APP-GW-GL-702, Revision 2
“Public Version of AP1000 Design Control Document”

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Design Control Document, Revision 20 (Public Version)

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The design, engineering, and other information contained in this document have been prepared by or on behalf of the Westinghouse Electric Company LLC in connection with its application to the United States Nuclear Regulatory Commission (NRC) for design certification of the AP1000 passive nuclear plant design pursuant to Title 10, Code of Federal Regulations Part 52. No use of or right to copy any of this information, other than by the NRC and its contractors in support of Westinghouse's application, is authorized.

The information provided in this document is a subset of a much larger set of know-how, technology and intellectual property rights pertaining to standardized, modularized, nuclear powered, electric generating facilities that utilize a minimum of active components and are characterized by a passive shutdown system, designed by Westinghouse and referred to as the AP1000® nuclear power plant design. Without access and a Westinghouse grant of rights to that larger set of know-how, technology and intellectual property rights, this document is not practically or rightfully usable by others, except by those with express authorization by Westinghouse, as set forth in the previous paragraph.

This document is a redacted version with sensitive unclassified non-safeguards information (SUNSI) related to the physical protection of an AP1000® nuclear plant removed. SUNSI information has been withheld pursuant to 10 CFR 2.390. The pages containing the SUNSI are marked with the following label at the top of the page. "Security-Related Information, Withheld Under 10 CFR 2.390."

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Design Control Document Introduction

1.0 Scope and Purpose of the Design Control Document

This Design Control Document (DCD) is a repository of information comprising the **AP1000®**(*) Standard Plant Design. The design control document also provides that design-related information to be incorporated by reference into Appendix D to 10 CFR Part 52 (the AP1000 design certification rule).

Applicants for a combined license pursuant to 10 CFR 52 must ensure that Appendix D to 10 CFR Part 52 and the associated Statements of Consideration are used when making licensing decisions relevant to the AP1000 Standard Plant Design.

Further sections of this introduction summarize the contents and use of the design control document. The design control document contains this introduction, the Tier 1 Information and the Tier 2 Information for the AP1000 Standard Plant Design.

Detailed information on the application and use of the AP1000 design control document may be found in Appendix D to 10 CFR Part 52.

If there is a conflict between this introduction and the AP1000 design certification rule, the AP1000 design certification rule controls.

1.1 Tier 1 Information

Tier 1 means the portion of the design-related information contained in the AP1000 design control document that is approved and certified by the NRC. Tier 1 information includes:

- Definitions and general provisions;
- Design descriptions;
- Inspections, tests, analyses, and acceptance criteria (ITAAC);
- Significant site parameters; and
- Significant interface requirements between the AP1000 Standard Plant Design and systems that are wholly or partially outside the scope of the AP1000 Standard Plant Design

The Tier 1 Information includes a table of contents, a figure legend, and an abbreviation list.

1.2 Tier 2 Information

Tier 2 means the portion of the design-related information contained in the AP1000 Design Control Document that is approved but not certified by the NRC. Tier 2 information includes:

- Information required by 10 CFR 52.47, with the exception of generic technical specifications and conceptual design information;
- Information required for a final safety analysis report under 10 CFR 50.34;
- Supporting information on the inspections, tests, and analyses that will be performed to demonstrate that the acceptance criteria in the ITAAC have been met; and

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- Combined license (COL) information items which identify certain matters that shall be addressed in the site-specific portion of the final safety analysis report (FSAR) by an applicant who references the AP1000 design certification rule.

Each volume of the Tier 2 Information includes a master table of contents and each chapter contains a chapter specific table of contents.

1.3 Relationship of the Tier 1 Information to the Tier 2 Information

The design descriptions, interface requirements, and site parameters in Tier 1 are derived from Tier 2 information.

Compliance with Tier 2 is required, but generic changes to and plant-specific departures from Tier 2 are governed by the AP1000 design certification rule. Compliance with Tier 2 provides a sufficient, but not the only acceptable, method for complying with Tier 1. Compliance methods differing from Tier 2 must satisfy the change process in Section VIII of the AP1000 design certification rule.

1.4 Uses of the Design Control Document

An applicant for a license that wishes to reference the AP1000 design certification rule shall, in addition to complying with the requirements of 10 CFR 52.77, 52.79, and 52.80, comply with the following requirements:

- Incorporate by reference, as part of its application, the AP1000 design certification rule;
- Include, as part of its application:
    - A plant-specific design control document containing the same information and utilizing the same organization and numbering as the generic design control document for the AP1000 Standard Plant Design, as modified and supplemented by the applicant's exemptions and departures;
    - The reports on departures from and updates to the plant-specific design control document required by Section X of the AP1000 design certification rule;
    - Plant-specific technical specifications, consisting of the generic and site-specific technical specifications, that are required by 10 CFR 50.36 and 50.36a;
    - Information demonstrating compliance with the site parameters and interface requirements;
    - Information that addresses the COL information items; and
    - Information required by 10 CFR 52.47(a) that is not within the scope of the AP1000 design certification rule.
    - Physically include, in the plant-specific design control document, the proprietary information referenced in the AP1000 design control document.

The Commission reserves the right to determine in what manner the AP1000 design certification rule may be referenced by an applicant for a construction permit or operating license under 10 CFR Part 50.

2.0 Effect of the Tier 1 Information

The following provisions describe the scope and effect of the Tier 1 Information.

2.1 Compliance with Tier 1 Information

All of the information in the Tier 1 Information is approved by the NRC and is applicable to a license application for a license that references the AP1000 design certification rule, and is

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among the "matters resolved" under 10 CFR52.63 (a)(4). The provisions and methods specified in the Tier 1 Information shall be complied with unless a plant specific exemption is granted by the NRC or a change is made to the Tier 1 Information in accordance with the change process specified in Section VIII of the AP1000 design certification rule.

2.2 Design Descriptions

The Design Descriptions pertain only to the design of structures, systems and components of an AP1000 Standard Plant Design and not to their operation, maintenance, and administration. In the event of an inconsistency between the Design Descriptions and the Tier 2 Information, the Design Descriptions shall govern.

2.3 Inspections, Tests, Analyses and Acceptance Criteria

An applicant or licensee who references the AP1000 design certification rule shall perform and demonstrate conformance with the ITAAC before fuel load. With respect to activities subject to an ITAAC, an applicant for a license may proceed at its own risk with design and procurement activities, and a licensee may proceed at its own risk with design, procurement, construction, and preoperational activities, even though the NRC may not have found that any particular ITAAC has been satisfied.

In the event that an activity is subject to an ITAAC, and the applicant or licensee who references the AP1000 design certification rule has not demonstrated that the ITAAC has been satisfied, the applicant or licensee may either take corrective actions to successfully complete that ITAAC, request an exemption from the ITAAC in accordance with Section VIII of the AP1000 design certification rule and 10 CFR 52.97(b), or petition for rulemaking to amend the AP1000 design certification rule by changing the requirements of the ITAAC, under 10 CFR 2.802 and 52.97(b).

In accordance with 10 CFR 52.99 and 52.103(g), the Commission shall find that the acceptance criteria in the ITAAC for the license are met before fuel load.

After the Commission has made the finding required by 10 CFR 52.103(g), the ITAAC do not, by virtue of their inclusion within the design control document, constitute regulatory requirements either for licensees or for renewal of the license; except for specific ITAAC, which are the subject of a Section 103(a) hearing, their expiration will occur upon final Commission action in such proceeding. However, subsequent modifications must comply with the Tier 1 and Tier 2 design descriptions in the plant-specific design control document unless the licensee has complied with the applicable requirements of 10 CFR 52.97 and Section VIII of the AP1000 design certification rule.

2.4 Tier 1 Site Parameters

Site parameters are specified in the Tier 1 Information to establish the bounding parameters to be used in the selection of a suitable site for the facility referencing the AP1000 certified design. Since the Tier 1 Information Site Parameters were used in the bounding evaluations of the certified design, they define the requirements for the design that must be met to ensure that a facility built on the site remains in conformance with the design certification. In the event that an inconsistency between the Tier 1 Information Site Parameters and the Tier 2 Information, the Tier 1 Information Site Parameters shall govern.

2.5 Tier 1 Interface Requirements

The Tier 1 Interface Requirements describe the significant design provisions for interfaces between the AP1000 Standard Plant Design and structures, systems and components that are

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wholly or partially outside the scope of the AP1000 Standard Plant Design. Tier 1 Interface Requirements also define the significant attributes and performance characteristics that the out-of-scope portion of the plant must have in order to support the in-scope portion of the design. The FSAR shall contain provisions which implement the Interface Requirements in accordance with 10 CFR 52.79(b). Any plant-specific application for a COL shall contain additional ITAAC corresponding to these implementing provisions. In the event of an inconsistency between the Tier 1 Interface Requirements and the Tier 2 Information, the Tier 1 Interface Requirements shall govern.

3.0 Effect of the Tier 2 Information

The following provisions describe the scope and effect of the Tier 2 Information.

3.1 Compliance with the Tier 2 Information

All of the information in the Tier 2 Information is approved by the NRC and, with the exceptions noted in Sections 3.2, 3.3, and 3.5 below, is applicable to a license that references the AP1000 design certification rule and is among the "matters resolved" under 10 CFR 52.63(a)(4). Compliance with the Tier 2 Information is a sufficient, but not necessarily the only, method of complying with the Tier 1 Information. The provisions and methods specified in the Tier 2 Information shall be followed unless a change is made in accordance with Section VIII of the AP1000 design certification rule.

3.2 Standard FSAR Text

Standard final safety analysis report (FSAR) text from the Reference Plant (Vogtle Electric Generating Plant 3&4) is embedded directly into this AP1000 generic DCD. The Standard FSAR text included in DCD Tier 2 information is colored blue. This information is included within the DCD to encourage the standardization of applications, enhance the safety of nuclear power plants, and facilitate a predictable and consistent method for application review.

This Standard FSAR text is outside the scope of the AP1000 Standard Plant Design and does not impose requirements applicable to a COL, nor an application for a COL, that references the AP1000 design certification rule. Standard FSAR text in the DCD is not subject to the change process in 10 CFR 52 Appendix D. An applicant referencing the AP1000 design certification rule must confirm the applicability of the Standard FSAR text as part of the COL application.

3.3 COL Information Items

Combined license (COL) information items identify certain matters that shall be addressed in the site-specific portion of the FSAR by an applicant who references the AP1000 design certification rule. These items constitute information requirements but are not the only acceptable set of information in the FSAR. An applicant may depart from or omit these items, provided that the departure or omission is identified and justified in the FSAR. After issuance of a construction permit or COL, these items are not requirements for the licensee unless such items are restated in the FSAR.

A summary of the AP1000 COL Information Items is provided in Table 1.8-2 of the Tier 2 Information.

The COL information sections contained in the DCD may contain blue standard FSAR text, which is provided for information, but must be confirmed by an applicant for inclusion in their FSAR.

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3.4 Tier 2 Interface Requirements

The Tier 2 Interface Requirements describe the design provisions for interfaces between the AP1000 Standard Plant Design and structures, systems and components that are wholly or partially outside the scope of the AP1000 Standard Plant Design. Tier 2 Interface Requirements, summarized in Table 1.8-1 of the Tier 2 Information, also define the attributes and performance characteristics that the out-of-scope portion of the plant must have in order to support the in-scope portion of the design. The FSAR shall contain provisions which implement the Tier 2 Interface Requirements in accordance with 10 CFR 52.79(b). In the event of an inconsistency between the Tier 1 Interface Requirements and the Tier 2 Interface Requirements, the Tier 1 Interface Requirements shall govern.

3.5 Conceptual Designs

Conceptual designs for those portions of the plant that are outside the scope of the AP1000 Standard Plant Design are described and designated as out-of-scope in various places in the Tier 2 Information. As provided by 10 CFR 52.47(a)(1)(ix), these conceptual designs are not a part of the design certification for the AP1000 Standard Plant Design and do not impose requirements applicable to a COL, nor an application for a COL, that references the AP1000 design certification rule. Those portions of the AP1000 Standard Plant Design for which conceptual designs are included in the Tier 2 Information are identified by double brackets and listed in Section 1.8 of the Tier 2 Information.

3.6 Plant-Specific Changes to Designated Information in the Tier 2 Information

Tier 2* means the portion of the Tier 2 information, designated as such in the AP1000 design control document, which is subject to the change process in Section VIII of the AP1000 design certification rule. This designation expires for some Tier 2* information under Section VIII of the AP1000 design certification rule.

An applicant who references the AP1000 design certification rule may depart from Tier 2* information, which is designated with italicized text or brackets and an asterisk in the AP1000 design control document, without NRC approval, except for departures that:

1. Involve a deviation from a code or standard credited in the plant-specific DCD for establishing the criteria for the design or construction of a structure, system, or component (SSC) important to safety,
2. Result in a change to a design process described in the plant-specific DCD that is material to implementation of an industry standard or endorsed regulatory guidance,
3.
   (i) Result in a change to the fuel criteria evaluation process, the fuel principal design requirements, or the nuclear design of the fuel or the reactivity control system that is material to a fuel or reactivity control system design function, or the evaluation process in WCAP-12488, “Westinghouse Fuel Criteria Evaluation Process,” or
   (ii) Result in any change to the maximum fuel rod average burn-up limits; or the small break LOCA analysis information in UFSAR Subsections 15.6.5.4B.2.2 or 15.6.5.4B.2.3,
4. Adversely affect the containment debris limits or debris screen design criteria,
5. Change the Reactor Coolant Pump (RCP) type from a canned motor to a different type of RCP,
6. Result in a change to the Passive Residual Heat Removal Heat Exchanger natural circulation test (first plant test), the Core Makeup Tank Heated Recirculation Tests (first

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three plants test), or the Automatic Depressurization System Blowdown Test (first three plants test) that is material to the test objectives or test performance criteria,

7. Involve structural materials or analytical or design methods, including design codes and analytical assumptions, that deviate from those credited in the plant-specific DCD for critical sections,
8. Result in a change to the design of the steel faceplates, internal trusses, tie bars, or headed studs of the steel-concrete (SC) module walls in the Nuclear Island or the Shield Building, including SC-to-reinforced concrete (RC) connections,
9. Result in an increase in the demand to capacity (D/C) ratio of a critical section of the structure. SNC shall determine the D/C ratio under this condition for each critical section structural member including, but not limited to, wall segments, wall sections, concrete panels, slabs, or basemat sections, affected by a departure by:
   (i) Using the Tier 2* information in the UFSAR Section 3.8 or Appendix 3H table that directly states the D/C ratio or states the area of steel provided and the area of steel required for the affected structural member, or
   (ii) Providing the same total area of steel across the entire critical section using any combination of rebar sizes and spacing allowed by the design basis codes used in the UFSAR as the total area of steel specified in UFSAR Section 3.8 and Appendix 3H tables marked Tier 2*.

The AP1000 Tier 2* information, summarized in Table 1-1 of this introduction, is designated with italicized text in the Tier 2 Information. Certain figures that are indicated to be Tier 2* may contain information beyond that that is considered to be Tier 2*. A review of the text referencing the figure may be necessary to determine what information on the figure is considered to be Tier 2*. The AP1000 Tier 2* information for which the Tier 2* designation expires when the COL holder first achieves 100% power operation is indicated in Table 1-1 of this introduction.

3.7 Treatment of Probabilistic Risk Assessment Information

The Probabilistic Risk Assessment (PRA) completed for the AP1000 Reference Plant is included in Chapter 19 of the Tier 2 information. Combined License holders referencing the DCD must update the Probabilistic Risk Assessment prior to initial fuel load.

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Table 1-1
Index of AP1000 Tier 2 Information Requiring NRC Approval for Change

Item	Expiration at First Full Power	Tier 2 Reference
Piping Modeling	Yes	3.6.2.1.1.1 3.6.2.1.1.2 3.6.2.1.1.3 3.9.1.2
Pipe Stress Analysis Criteria	Yes	3.6.2.2 3.6.3.3 3.9.3.1.2 3.9.3.1.5 Table 3.9-5 Table 3.9-6 Table 3.9-7 Table 3.9-8 Table 3.9-9 Table 3.9-10 Table 3.9-11
Dimensions for Nuclear Island Structures	Yes	3.7.1.4 Table 3.7.1-2 Figure 3.7.1-14
Nuclear Island Key Structural Dimensions	Yes	3.7.2 Figure 3.7.2-12
Polar Crane Parked Orientation	Yes	3.7.2.3.2
Nuclear Island Seismic Model	Yes	3.7.2.3.3
Composite Modal Damping Approach	Yes	3.7.2.14
ASME Class 1 Piping, Fatigue Evaluation	Yes	3.7.3.2
Equivalent Static Load Method	Yes	3.7.3.5
Distributed Mass with Dynamic Response Single Mode Dominant	Yes	3.7.3.5.1
Multiple Mode Dominant Response	Yes	3.7.3.5.2
Three Components of Earthquake Motion	Yes	3.7.3.6
High Frequency or Rigid Modes	Yes	3.7.3.7
PIPESTRESS Left-Out-Force Method	Yes	3.7.3.7.1.1
GAPPIPE, High Frequency Responses	Yes	3.7.3.7.1.1
SRP Section 3.7.2 Use for Combination of High-Frequency Modes	Yes	3.7.3.7.1.2

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Table 1-1
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Item	Expiration at First Full Power	Tier 2 Reference
Combination of Low-Frequency Modes	Yes	3.7.3.7.2
Analytical Procedure for Piping	Yes	3.7.3.8
Supporting Systems	Yes	3.7.3.8.1
Large-Diameter Auxiliary Piping	Yes	3.7.3.8.2.1
Small-Diameter Auxiliary Piping	Yes	3.7.3.8.2.2
Piping Systems on Modules	Yes	3.7.3.8.3
Piping Systems with Gapped Supports	Yes	3.7.3.8.4
Seismic Anchor Motions	Yes	3.7.3.9
Independent Support Response Spectrum Methods	Yes	3.7.3.9
Torsional Effects of Eccentric Masses	Yes	3.7.3.11
Buried Piping Systems and Tunnels	Yes	3.7.3.12
Interaction of Connected Systems with Seismic Category I Piping Interaction of Piping with Seismic Category I Piping Systems, Structures, and Components	Yes	3.7.3.13 3.7.3.13.4
Design Method of Piping to Prevent Adverse Spatial Interactions	Yes	3.7.3.13.4.1 3.7.3.13.4.2 3.7.3.13.4.3
Composite Modal Damping for Coupled Building and Piping Systems	Yes	3.7.3.15
Time History Analysis of Piping Systems	Yes	3.7.3.17
SSE Damping Valves	Yes	Table 3.7.1-1
Containment Vessel Design Characteristics and Spacing Between Each Pair of Ring Supports	Yes	3.8.2.1.1 3.8.2.1.3 3.8.2.1.4 3.8.2.1.5
2001 Edition of ASME Code, Section III, including 2002 Addenda for Containment Design	Yes	3.8.2.2
ASME Code Case N-284-1	Yes	3.8.2.2 3.8.2.5

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Table 1-1
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Item	Expiration at First Full Power	Tier 2 Reference
Containment Vessel Materials, Quality Control, and Special Construction Techniques	Yes	3.8.2.6
Specific Tier 2* Details of Figure 3.8.3-2, Figure 3.8.3-8, Sheets 1, 2, 3, and Figure 3.8.3-15	Yes	3.8.3.1 3.8.3.1.3
Structural Steel Modules Materials, Quality Control, and Special Construction Techniques	Yes	3.8.3.6
Use of ACI-349-01	Yes	3.8.3.2 3.8.3.5 3.8.4.2 3.8.4.4.1 3.8.4.5 3.8.4.5.1 3.8.5.5 Table 3.8.4-2
Use of AISC N690-1994	Yes	3.8.3.2 3.8.4.2 3.8.4.4.1 3.8.4.5 3.8.4.5.2 Table 3.8.4-1
Use of AISI	Yes	3.8.4.4.1 3.8.4.5

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Table 1-1
Index of AP1000 Tier 2 Information Requiring NRC Approval for Change

Item	Expiration at First Full Power	Tier 2 Reference
Design Summary of Critical Sections Inside Containment	Yes	3.8.3.5.8.1 3.8.3.5.8.2 3.8.3.5.8.3 3.8.3.8 Table 3.8.3-3 Table 3.8.3-4 Table 3.8.3-5 Table 3.8.3-6 Table 3.8.3-7 Figure 3.8.3-1 Figure 3.8.3-2 Figure 3.8.3-8 Figure 3.8.3-14 Figure 3.8.3-15 Figure 3.8.3-17 Figure 3.8.3-18 3.8.4.6.1.1 3.8.4.6.1.2
Steel Wall of IRWST Minimum Size of Angles and Channels Spacing of Faceplates, Trusses, Channels, and Headed Studs. Mechanical Connectors. Use of Anchors, Studs, and Bars. Spacing of Faceplates and Trusses.		

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Table 1-1
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Item	Expiration at First Full Power	Tier 2 Reference
Design Summary of Critical Sections Outside Containment	Yes	3.8.4.1.1 3.8.4.5.4 3.8.4.5.5 3.8.4.5.5.5 3.8.4.6.1.1 3.8.4.6.1.2 3.8.4.6.1.3
Design and Analysis Procedures		Figure 3.8.4-2
Seismic Loads		Figure 3.8.4-4
Bearing Pressure		3.8.5.1 3.8.5.4 App 3H.1 App 3H.2.1 App 3H.3 App 3H.3.1 App 3H.3.2 App 3H.3.3 App 3H.3.4 App 3H.4 App 3H.4.1 App 3H.5 App 3H.5.1 App 3H.5.1.1 App 3H.5.1.2 App 3H.5.1.3 App 3H.5.1.4 App 3H.5.2 App 3H.5.2.1 App 3H.5.2.2 App 3H.5.3 App 3H.5.3.1 App 3H.5.4 App 3H.5.5 App 3H.5.5.1 App 3H.5.6 App 3H.5.6.1 App 3H.5.6.2 App 3H.5.6.3 App 3H.5.7.1 App 3H.5.7.2

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Table 1-1
Index of AP1000 Tier 2 Information Requiring NRC Approval for Change

Item	Expiration at First Full Power	Tier 2 Reference
Design Summary of Critical Sections Outside Containment (Cont.)		Table 3H.5-1 Table 3H.5-2 Table 3H.5-3 Table 3H.5-4 Table 3H.5-5 Table 3H.5-6 Table 3H.5-7 Table 3H.5-8 Table 3H.5-9 Table 3H.5-10 Table 3H.5-11 Table 3H.5-12 Table 3H.5-13 Table 3H.5-14 Figure 3H.2-1 Figure 3H.5-1 Figure 3H.5-2 Figure 3H.5-3 Figure 3H.5-4 Figure 3H.5-5 Figure 3H.5-6 Figure 3H.5-7 Figure 3H.5-8 Figure 3H.5-9 Figure 3H.5-10 Figure 3H.5-11 Figure 3H.5-12 Figure 3H.5-15 Figure 3H.5-16
Design Summary of Critical Sections for Nuclear Island Basemat	Yes	3.8.5.4.4 Table 3.8.5-3
Reference to APP-GW-GLR-602, Rev 1	Yes	3.8.7 3H.5.8
Design Transients	Yes	3.9.1.1
Loads for Class I Components and Core/Component Supports	Yes	3.9.3.1.2
Use of Square-Root-Sum-of-the-Squares Method for SSE plus Pipe Rupture	Yes	3.9.3.1.3
Analysis of Reactor Coolant Loop Piping	Yes	3.9.3.1.4

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Table 1-1
Index of AP1000 Tier 2 Information Requiring NRC Approval for Change

Item	Expiration at First Full Power	Tier 2 Reference
Analysis Methods	Yes	3.9.1.3 3.9.3.1.5
ASME Classes 1, 2, and 3 Piping	Yes	3.9.3.1.5
Pipe Support Criteria	Yes	3.9.3.4 3.9.3.5
Design of Spring-Loaded Safety Valves	Yes	3.9.3.3.1
Design and Analysis Requirement for Open and Closed Discharge Systems	Yes	3.9.3.3 3.9.3.3.3
Component and Piping Supports for Dynamic Loading	Yes	3.9.3.4
Class 2 and Class 3 Component Supports	Yes	3.9.3.4.2
Piping System Seismic Stress Analysis	Yes	3.9.3.4.3
Seismic Qualification Standards	Yes	3.10.1.1
Seismic Response Spectra	Yes	App 3G.4.3.3 Figure 3G.4-5X Figure 3G.4-5Y, Figure 3G.4-5Z Figure 3G.4-6X Figure 3G.4-6Y Figure 3G.4-6Z Figure 3G.4-7X Figure 3G.4-7Y Figure 3G.4-7Z Figure 3G.4-8X Figure 3G.4-8Y Figure 3G.4-8Z Figure 3G.4-9X Figure 3G.4-9Y Figure 3G.4-9Z Figure 3G.4-10X Figure 3G.4-10Y Figure 3G.4-10Z
Fuel Principal Design Requirements	No	4.1.1

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Table 1-1
Index of AP1000 Tier 2 Information Requiring NRC Approval for Change

Item	Expiration at First Full Power	Tier 2 Reference
WCAP-12488-P-A, “Fuel Criteria Evaluation Process,” October 1994 WCAP-14204-A, “Fuel Criteria Evaluation Process,” October 1994	No	Table 1.6-1 4.1.3 4.2 4.2.1 4.2.1.1.2 4.2.1.1.3 4.2.1.5 4.2.1.6 4.2.3 4.2.6 4.3.1 4.3.1.1.1 4.3.5 4.4.8
Maximum Fuel Rod Average Burnup	No	4.3.1.1.1
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WCAP-17201-P, “AC160 High Speed Link Communication Compliance to DI&C-ISG-04 Staff Position 9, 12, 13, and 15,” Rev 0, February 2010	Yes	Table 1.6-1 7.1.7
WCAP-15927 (Non-Proprietary), “Design Process for AP1000 Common Q Safety Systems,” Rev 2	Yes	Table 1.6-1 7.1.2.14.1 7.1.7

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WCAP-16097-P-A, “Common Qualified Platform,” Rev 0	Yes	Table 1.6-1 7.1.2.14.2 7.1.7
WCAP-16096-NP-A, “Software Program Manual for Common Q Systems,” Rev 01A	Yes	Table 1.6-1 7.1.2.14 7.1.2.14.1 7.1.7
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Small-break loss-of-coolant accident (LOCA) analysis methodology	No	15.6.5.4B.2.2 15.6.5.4B.2.3
WCAP-14396, “Man-in-the-Loop Test Plan Description,” Rev 3	Yes	Table 1.6-1 18.8.1.4 18.8.6
WCAP-15860, “Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan,” Rev 2	Yes	Table 1.6-1 18.8 18.8.1.2 18.8.1.7 18.8.6 18.11.2
WCAP-14651, “Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan,” Rev 2	Yes	Table 1.6-1 18.5.1 18.5.5 18.7 18.7.2 18.8.1.9 18.8.2.1 18.8.6 18.12.2 18.12.5

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WCAP-15847, “AP1000 Quality Assurance Procedures Supporting NRC review of AP1000 SSAR Sections 18.2 and 18.8,” Rev 1	Yes	Table 1.6-1 18.2.3.5 18.2.7
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HFE Technical Program and Milestones	Yes	18.2.5
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Integration of Human Reliability Analysis with HFE	Yes	18.7
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Operator Decision-Making Model Used by Task Analysis Activities	Yes	18.8.1.8
Critical Human Actions and Risk-Important Tasks	Yes	18.8.1.9
Safety Parameter Display System 10 CFR 50.34(f)(2)(iv)	Yes	18.8.2 18.8.6
NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability”	Yes	18.8.2 18.8.3.5 18.8.6
Implementation Plan for Integrating Human Reliability Analysis with HFE	Yes	18.8.2.1
Display of Safety Parameters	Yes	18.8.2.2
Safety Parameter Display System HFE	Yes	18.8.2.5
Minimum Information, Safety Parameter Display System Design	Yes	18.8.2.6
NUREG-1342, “A Status Report Regarding Industry Implementation of Safety Parameter Display Systems”	Yes	18.8.2.6 18.8.6
Main Control Area Mission and Major Tasks Regulatory Guide 1.97	Yes	18.8.3.2
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Human Performance Issues to be Addressed by HSI Design	Yes	Table 18.8-1
APP-OCS-GEH-120, “AP1000 Human Factors Engineering Design Verification Plan,” Rev B	Yes	Table 1.6-1 18.11.2
APP-OCS-GEH-220, “AP1000 Human Factors Engineering Task Support Verification Plan,” Rev B	Yes	Table 1.6-1 18.11.2
APP-OCS-GEH-320, “AP1000 Human Factors Engineering Integrated System Validation Plan,” Rev D	Yes	Table 1.6-1 18.11.2
APP-OCS-GEH-420, “AP1000 Human Factors Engineering Discrepancy Resolution Process,” Rev B	Yes	Table 1.6-1 18.11.2
APP-OCS-GEH-520, “AP1000 Plant Startup Human Factors Engineering Verification Plan,” Rev B	Yes	Table 1.6-1 18.11.2
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Implementation Process for Identification of Critical PRA Operator Actions	Yes	18.12.2
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Table 1-2 Not Used.

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1.0 Introduction

1.1 Definitions

The following definitions apply to terms used in the design descriptions and associated inspections, tests, analyses, and acceptance criteria (ITAAC).

Acceptance Criteria means the performance, physical condition, or analysis result for a structure, system, or component that demonstrates that the design commitment is met.

Analysis means a calculation, mathematical computation, or engineering or technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar structures, systems, or components.

As-built means the physical properties of a structure, system, or component following the completion of its installation or construction activities at its final location at the plant site. In cases where it is technically justifiable, determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation, provided that subsequent fabrication, handling, installation, and testing does not alter the properties.

Column Line is the designation applied to a plant reference grid used to define the location of building walls and columns. Column lines may not represent the center line of walls and columns.

Design Commitment means that portion of the design description that is verified by ITAAC.

Design Description means that portion of the design that is certified.

Design Plant Grade means the elevation of the soil around the nuclear island assumed in the design of the AP1000, i.e., floor elevation 100'-0".

Division (for electrical systems or electrical equipment) is the designation applied to a given safety-related system or set of components that is physically, electrically, and functionally independent from other redundant sets of components.

Floor Elevation is the designation applied to name a floor. The actual elevation may vary due to floor slope and layout requirements.

Functional Arrangement (for a system) means the physical arrangement of systems and components to provide the service for which the system is intended, and which is described in the system design description.

Inspect or **Inspection** means visual observations, physical examinations, or reviews of records based on visual observation or physical examination that compare the structure, system, or component condition to one or more design commitments. Examples include walkdowns, configuration checks, measurements of dimensions, or nondestructive examinations.

Inspect for Retrievability of a display means to visually observe that the specified information appears on a monitor when summoned by the operator.

$\mathbf{L_a}$ is the maximum allowable containment leakage as defined in 10 CFR 50 Appendix J.

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Physical Arrangement (for a structure) means the arrangement of the building features (e.g., floors, ceilings, walls, and basemat) and of the structures, systems, and components within, which are described in the building design description.

Qualified for Harsh Environment means that equipment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of its safety function, for the time required to perform the safety function. These environmental conditions include applicable time-dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, submergence, and their synergistic effects which have a significant effect on the equipment performance. Equipment identified in the Design Description as being Qualified for Harsh Environment includes the:

a. equipment itself
b. sensors, switches and lubricants that are an integral part of the equipment
c. electrical components connected to the equipment (wiring, cabling and terminations)

Items b and c are Qualified for Harsh Environment only when they are necessary to support operation of the equipment to meet its safety-related function listed in the Design Description table and to the extent such equipment is located in a harsh environment during or following a design basis accident.

Sensor means a transmitter, resistance temperature detector, thermocouple or other transducer, plus associated cables, connectors, preamplifiers, reference junction boxes, or other signal processing equipment that is located in the immediate proximity of the sensor and subject to the same environmental conditions.

Site Grade means the as-built elevation of the soil to the west side of the nuclear island. Adjacent buildings are located on the other sides of the nuclear island.

Tag Number in the ITAACs represents the complete tag number or a portion of the tag number used to identify the actual hardware (or associated software). For instrumentation, the tag number identified in the ITAACs does not include the type of instrument (for example, the Containment Exhaust Fan A Flow Sensor, VFS-11A, does not include the designators FE [flow element] or FT [flow transmitter], which would appear on the actual hardware or in the associated software). This is because the designator VFS-11A and the equipment description are sufficient to uniquely identify the channel associated with the designated instrument function, and this method of identification eliminates the need to list every portion of the instrumentation channel required to perform the function. In most cases, the channel number includes physical hardware. There are, however, a few places where the channel number represents only a calculation in software. In those cases, the channel data can be displayed. In many instances, the word "sensor" is used in the equipment description to identify that the item is an instrument.

Test means the actuation, operation, or establishment of specified conditions to evaluate the performance or integrity of as-built structures, systems, or components, unless explicitly stated otherwise.

Transfer Open (Closed) means to move from a closed (open) position to an open (closed) position.

Type Test means a test on one or more sample components of the same type and manufacturer to qualify other components of the same type and manufacturer. A type test is not necessarily a test of the as-built structures, systems, or components.

UA of a heat exchanger means the product of the heat transfer coefficient and the surface area.

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1.2 General Provisions

The following general provisions are applicable to the design descriptions and associated ITAAC.

Treatment of Individual Items

The absence of any discussion or depiction of an item in the design description or accompanying figures shall not be construed as prohibiting a licensee from utilizing such an item, unless it would prevent an item from performing its safety functions as discussed or depicted in the design description or accompanying figures.

If an inspections, tests, or analyses (ITA) requirement does not specify the temperature or other conditions under which a test must be run, then the test conditions are not constrained.

When the term "operate," "operates," or "operation" is used with respect to an item discussed in the acceptance criteria, it refers to the actuation and running of the item. When the term "exist," "exists," or "existence" is used with respect to an item discussed in the acceptance criteria, it means that the item is present and meets the design commitment.

When the Design Commitment or ITAAC provides that an item or activity must comply with ASME Code Section III, this means compliance with the ASME Code Section III, as incorporated by reference in 10 CFR 50.55a with specific conditions, or in accordance with alternatives authorized by the NRC pursuant to 10 CFR 50.55a.

Implementation of ITAAC

The ITAACs are provided in tables with the following three-column format:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

Each design commitment in the left-hand column of the ITAAC tables has an associated ITA requirement specified in the middle column of the tables.

The identification of a separate ITA entry for each design commitment shall not be construed to require that separate inspections, tests, or analyses must be performed for each design commitment. Instead, the activities associated with more than one ITA entry may be combined, and a single inspection, test, or analysis may be sufficient to implement more than one ITA entry.

An ITA may be performed by the licensee of the plant or by its authorized vendors, contractors, or consultants. Furthermore, an ITA may be performed by more than a single individual or group, may be implemented through discrete activities separated by time, and may be performed at any time prior to fuel load (including before issuance of the combined license for those ITAACs that do not necessarily pertain to as-installed equipment). Additionally, an ITA may be performed as part of the activities that are required to be performed under 10 CFR Part 50 (including, for example, the quality assurance (QA) program required under Appendix B to Part 50); therefore, an ITA need not be performed as a separate or discrete activity.

Many of the acceptance criteria include the words “A report exists and concludes that…” When these words are used, it indicates that the ITAAC for that design commitment will be met when it is confirmed that appropriate documentation exists and the documentation shows that the design commitment is met. Appropriate documentation can be a single document or a collection of documents that show that the stated

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acceptance criteria are met. Examples of appropriate documentation include design reports, test reports, inspection reports, analysis reports, evaluation reports, design and manufacturing procedures, certified data sheets, commercial dedication procedures and records, quality assurance records, calculation notes, and equipment qualification data packages. For plants at sites which are qualified using the hard rock high frequency (HRHF) ground motion response spectra (GMRS), high frequency seismic screening and qualification testing required as a result of the evaluation of potential high frequency sensitive components is included in the equipment qualification data packages.

Many entries in the ITA column of the ITAAC tables include the words "Inspection will be performed for the existence of a report verifying…" When these words are used it indicates that the ITA is tests, type tests, analyses, or a combination of tests, type tests, and analyses and a report will be produced documenting the results. This report will be available to inspectors.

Many ITAAC are only a reference to another Tier 1 location, either a section, subsection, or ITAAC table entry (for example, "See Tier 1 Material…"). A reference to another ITAAC location is always in both the ITA and acceptance criteria columns for a design commitment. This reference is an indication that the ITA and acceptance criteria for that design commitment are satisfied when the referenced ITA are completed and the acceptance criteria for the referenced Tier 1 sections, subsections, or table entries are satisfied. If a complete Tier 1 section is referenced, this indicates that all the ITA and acceptance criteria in that section must be met before the referencing design commitment is satisfied.

Discussion of Matters Related to Operations

In some cases, the design descriptions in this document refer to matters that relate to operation, such as normal valve or breaker alignment during normal operation modes. Such discussions are provided solely to place the design description provisions in context (for example, to explain automatic features for opening or closing valves or breakers upon off-normal conditions). Such discussions shall not be construed as requiring operators during operation to take any particular action (for example, to maintain valves or breakers in a particular position during normal operation).

Interpretation of Figures

In many but not all cases, the design descriptions in Section 2 include one or more figures. The figures may represent a functional diagram, general structural representation, or another general illustration. For instrumentation and control (I&C) systems, figures may also represent aspects of the relevant logic of the system or part of the system. Unless specified explicitly, the figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, and components. In particular, the as-built attributes of structures, systems, and components may vary from the attributes depicted on the figures, provided that those safety functions discussed in the design description pertaining to the figure are not adversely affected.

Maximum Reactor Core Thermal Power

The initial rated reactor core thermal power for the AP1000 certified design is 3400 megawatts thermal (MWt).

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1.3 Figure Legend

The conventions used in this section are for figures described in the design description. The figure legend is provided for information and is not part of the Tier 1 Material.

VALVES

Valve

Check Valve

Relief Valve

VALVE OPERATORS

Operator Of Unspecified Type

Motor Operator

Solenoid Operator

Pneumatic/Hydraulic Operator

Pneumatic Operator

Squib Valve

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MECHANICAL EQUIPMENT

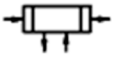
Centrifugal Pump

Pump Type Not Specified

Tank

Centrifugal Fan

Axial Fan

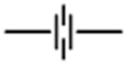
Heat Exchanger 

Vent

Drain

Pipe Cap

Blind Flange

Orifice 

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DAMPERS

Gravity Or Manually Operated Damper

Remotely Operated Damper

ELECTRICAL EQUIPMENT

Battery

Circuit Breaker

Disconnect Switch

Isolation

Transformer

Fuse

Heater

Generator

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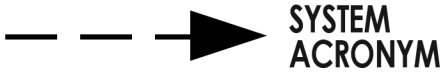
MISCELLANEOUS

A component that is part of the system functional arrangement shown on the figure and is included in the design commitments for the system.

A component that is part of the system functional arrangement shown on the figure.

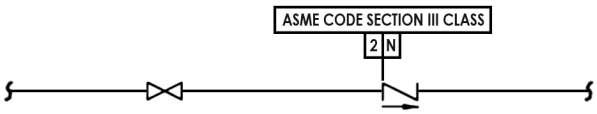
A system or component of another system that is not part of the system functional arrangement shown on the figure.

A functional connection to another system that is not part of the system functional arrangement shown on the figure.



ASME CODE CLASS BREAK

An ASME Code class break is identified by a single line to the designated location for the class break, as shown in the example below (see note 1).



NOTES:

1. The header, "ASME Code Section III Class," must appear at least once on each figure on which ASME class breaks are shown, but need not appear at every class break shown on a figure.
   N Indicates Non-ASME Code Section III

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1.4 List of Acronyms and Abbreviations

The acronyms presented in this section are used in the Tier 1 Material. The acronyms are provided for information and are not part of the Tier 1 Material.

ac	Alternating Current
AC	Acceptance Criteria
ACC	Accumulator
ADS	Automatic Depressurization System
AHU	Air Handling Units
ANS	Alert and Notification System
ASME	American Society of Mechanical Engineers
BTU	British Thermal Unit
CAS	Compressed and Instrument Air System
CAV	Cumulative Absolute Velocity
CCS	Component Cooling Water System
CDM	Certified Design Material
CDS	Condensate System
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CL	Cold Leg
CMT	Core Makeup Tank
CNS	Containment System
COL	Combined Operating License/Combined License
CRDM	Control Rod Drive Mechanism
CSA	Control Support Area
CST	Condensate Storage Tank
CVS	Chemical and Volume Control System
DAC	Design Acceptance Criteria
DAS	Diverse Actuation System
DBA	Design Basis Accident
dc	Direct Current
DC	Design Commitment
DDS	Data Display and Processing System
DOS	Standby Diesel Fuel Oil System
DPU	Distributed Processing Unit
D-RAP	Design Reliability Assurance Program
DTS	Demineralized Water Treatment System
DVI	Direct Vessel Injection

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List of Acronyms and Abbreviations (cont.)

DWS	Demineralized Water Transfer and Storage System
ECS	Main ac Power System
EDS	Non-Class 1E dc and Uninterruptible Power Supply System
EFS	Communication System
EGS	Grounding and Lightning Protection System
ELS	Plant Lighting System
EMI	Electromagnetic Interference
ERF	Emergency Response Facility
ESD	Electrostatic Discharge
ESF	Emergency Safety Features
ESFAS	Engineering Safety Feature Actuation System
F	Fahrenheit
FHM	Fuel Handling Machine
FHS	Fuel Handling and Refueling System
FID	Fixed Incore Detector
FPS	Fire Protection System
ft	Feet
FTS	Fuel Transfer System
FWS	Main and Startup Feedwater System
gpm	Gallons per Minute
HEPA	High Efficiency Particulate Air
HFE	Human Factors Engineering
HL	Hot Leg
hr	Hour
HSI	Human-System Interface
HVAC	Heating, Ventilation, and Air Conditioning
HX	Heat Exchanger
Hz	Hertz
I&C	Instrumentation and Control
IDS	Class 1E dc and Uninterruptible Power Supply System
IIS	In-core Instrumentation System
ILRT	Integrated Leak Rate Test
IHP	Integrated Head Package
in	Inches
I/O	Input/Output
I&C	Instrumentation and Control

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List of Acronyms and Abbreviations (cont.)

IRC	Inside Reactor Containment
IRWST	In-containment Refueling Water Storage Tank
ISI	Inservice Inspection
IST	Inservice Testing
ITA	Inspections, Tests, Analyses
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
JIC	Joint Information Center
LBB	Leak Before Break
LTOP	Low Temperature Overpressure Protection
MBtu	Million British Thermal Units
MCC	Motor Control Center
MCR	Main Control Room
MHS	Mechanical Handling System
MMIS	Man-machine Interface System
MOV	Motor-operated Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSS	Main Steam System
MTS	Main Turbine System
MW	Megawatt
MWe	Megawatt Electric
MWt	Megawatt Thermal
N/A	Not Applicable
NDE	Nondestructive Examination
NI	Nuclear Island
NSSS	Nuclear Steam Supply System
OCS	Operation and Control Centers System
ORC	Outside Reactor Containment
ORE	Occupational Radiation Exposure
OSA	Operational Sequence Analyses
OSC	Operations Support Center
PAR	Passive Autocatalytic Recombiner
PCCAWST	Passive Containment Cooling Ancillary Water Storage Tank
PCCWST	Passive Containment Cooling Water Storage Tank
PCS	Passive Containment Cooling System
P&ID	Piping and Instrument Diagram

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List of Acronyms and Abbreviations (cont.)

PGS	Plant Gas System
pH	Potential of Hydrogen
PLS	Plant Control System
PMS	Protection and Safety Monitoring System
PORV	Power-operated Relief Valve
PRA	Probabilistic Risk Assessment
PRHR	Passive Residual Heat Removal
psia	Pounds per Square Inch Absolute
PSS	Primary Sampling System
PXS	Passive Core Cooling System
PWR	Pressurized Water Reactor
RAP	Reliability Assurance Program
RAT	Reserve Auxiliary Transformer
RCA	Radiologically Controlled Area
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RFI	Radio Frequency Interference
RM	Refueling Machine
RMS	Radiation Monitoring System
RNS	Normal Residual Heat Removal System
RPV	Reactor Pressure Vessel
RSR	Remote Shutdown Room
RSW	Remote Shutdown Workstation
RTD	Resistance Temperature Detector
RXS	Reactor System
RV	Reactor Vessel
scf	Standard Cubic Feet
scfm	Standard Cubic Feet per Minute
SFP	Spent Fuel Pool
SFS	Spent Fuel Pool Cooling System
SG	Steam Generator
SGS	Steam Generator System
SJS	Seismic Monitoring System
SMS	Special Monitoring System

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List of Acronyms and Abbreviations (cont.)

SSAR	Standard Safety Analysis Report
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SWC	Surge Withstand Capability
SWS	Service Water System
TID	Total Integrated Dose
TSC	Technical Support Center
UAT	Unit Auxiliary Transformer
UBC	Uniform Building Code
UPS	Uninterruptible Power Supply
V	Volt
VAS	Radiologically Controlled Area Ventilation System
VBS	Nuclear Island Nonradioactive Ventilation System
VCS	Containment Recirculation Cooling System
VES	Main Control Room Emergency Habitability System
VFS	Containment Air Filtration System
VHS	Health Physics and Hot Machine Shop Areas
VLS	Containment Hydrogen Control System
VWS	Central Chilled Water System
VXS	Annex/Auxiliary Building Nonradioactive Ventilation System
VZS	Diesel Generator Building Ventilation System
WGS	Gaseous Radwaste System
WLS	Liquid Radwaste System
WSS	Solid Radwaste System
ZOS	Onsite Standby Power System

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2.1 Reactor

2.1.1 Fuel Handling and Refueling System

Design Description

The fuel handling and refueling system (FHS) transfers fuel assemblies and core components during fueling operations and stores new and spent fuel assemblies in the new and spent fuel storage racks. The refueling machine (RM) and the fuel transfer tube are operated during refueling mode. The fuel handling machine (FHM) is operated during normal modes of plant operation, including startup, power operation, cooldown, shutdown and refueling.

The component locations of the FHS are as shown in Table 2.1.1-2.

1. The functional arrangement of the FHS is as described in the Design Description of this Section 2.1.1.

2. The FHS has the RM, the FHM, and the new and spent fuel storage racks.

3. The FHS preserves containment integrity by isolation of the fuel transfer tube penetrating containment.

4. The RM and FHM/spent fuel handling tool (SFHT) gripper assemblies are designed to prevent opening while the weight of the fuel assembly is suspended from the grippers.

5. The lift height of the RM mast and FHM hoist(s) is limited such that the minimum required depth of water shielding is maintained.

6. The RM and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake.

7. The new and spent fuel storage racks maintain the effective neutron multiplication factor required by 10 CFR 50.68 limits during normal operation, design basis seismic events, and design basis dropped spent fuel assembly accidents over the spent fuel storage racks.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the FHS.

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Table 2.1.1-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
3. Not used.		
2. The FHS has the refueling machine (RM), the fuel handling machine (FHM), and the new and spent fuel storage racks.	Inspection of the system will be performed.	The FHS has the RM, the FHM, and the new and spent fuel storage racks.
4. The RM and FHM/spent fuel handling tool (SFHT) gripper assemblies are designed to prevent opening while the weight of the fuel assembly is suspended from the grippers.	The RM and FHM/SFHT gripper assemblies will be tested by operating the open controls of the gripper while suspending a dummy fuel assembly.	The RM and FHM/SFHT gripper assemblies will not open while suspending a dummy test assembly.
5. The lift height of the RM mast and FHM hoist(s) is limited such that the minimum required depth of water shielding is maintained.	The RM and FHM will be tested by attempting to raise a dummy fuel assembly.	The bottom of the dummy fuel assembly cannot be raised to within 24 ft, 6 in. of the operating deck floor.
6. The RM and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake.	i) Inspection will be performed to verify that the RM and FHM are located on the nuclear island.	i) The RM and FHM are located on the nuclear island.
7. The new and spent fuel storage racks maintain the effective neutron multiplication factor required by 10 CFR 50.68 limits during normal operation, design basis seismic events, and design basis dropped spent fuel assembly accidents over the spent fuel storage racks.	ii) Inspection will be performed to verify that the new and spent fuel storage racks are located on the nuclear island.	ii) The new and spent fuel storage racks are located on the nuclear island.
6. The RM and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake.	ii) Type test, analysis, or a combination of type tests and analyses of the RM and FHM will be performed.	ii) A report exists and concludes that the RM and FHM can withstand seismic design basis dynamic loads without loss of load carrying or structural integrity functions.

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Table 2.1.1-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The new and spent fuel storage racks maintain the effective neutron multiplication factor required by 10 CFR 50.68 limits during normal operation, design basis seismic events, and design basis dropped spent fuel assembly accidents over the spent fuel storage racks.	i) Analyses will be performed to calculate the effective neutron multiplication factor in the new and spent fuel storage racks during normal conditions. iii) Seismic analysis of the new and spent fuel storage racks will be performed. iv) Analysis of the spent fuel storage racks under design basis dropped spent fuel assembly loads will be performed.	i) The calculated effective neutron multiplication factor for the new and spent fuel storage racks meets the requirements of 10 CFR 50.68(1) limits under normal conditions. iii) A report exists and concludes that the new and spent fuel racks can withstand seismic design basis dynamic loads and maintain the calculated effective neutron multiplication factor required by 10 CFR 50.68(1) limits. iv) A report exists and concludes that the spent fuel racks can withstand design basis dropped spent fuel assembly loads and maintain the calculated effective neutron multiplication factor required by 10 CFR 50.68(1) limits.

Note:

1. The requirements of 10 CFR 50.68 are summarized as follows:
   - For new fuel storage racks:
     - The effective neutron multiplication factor (K-effective) must not exceed 0.95 when flooded with unborated water and
     - K-effective must not exceed 0.98 with optimum moderator conditions.
   - For spent fuel storage racks:
     - If methodology does not take credit for soluble boron:
       - K-effective must not exceed 0.95 when flooded with unborated water.
     - Or if methodology takes credit for soluble boron:
       - K-effective must not exceed 0.95 when flooded with borated water and
       - K-effective must remain below 1.0 when flooded with unborated water.

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Table 2.1.1-2

Component Name	Tag No.	Component Location
Refueling Machine	FHS-FH-01	Containment
Fuel Handling Machine	FHS-FH-02	Auxiliary Building
Spent Fuel Storage Racks	FHS-FS-02	Auxiliary Building
New Fuel Storage Racks	FHS-FS-01	Auxiliary Building
Fuel Transfer Tube	FHS-FT-01	Auxiliary Building/Containment

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2.1.2 Reactor Coolant System

Design Description

The reactor coolant system (RCS) removes heat from the reactor core and transfers it to the secondary side of the steam generators for power generation. The RCS contains two vertical U-tube steam generators, four sealless reactor coolant pumps (RCPs), and one pressurizer.

The RCS is as shown in Figure 2.1.2-1 and the component locations of the RCS are as shown in Table 2.1.2-5.

1. The functional arrangement of the RCS is as described in the Design Description of this Section 2.1.2.

2. a) The components identified in Table 2.1.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.1.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.1.2-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.1.2-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.1.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.1.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. a) The seismic Category I equipment identified in Table 2.1.2-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.1.2-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

6. Each of the as-built lines identified in Table 2.1.2-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

7. a) The Class 1E equipment identified in Table 2.1.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

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b) The Class 1E components identified in Table 2.1.2-1 are powered from their respective Class 1E division.

c) Separation is provided between RCS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

8. The RCS provides the following safety-related functions:

a) The pressurizer safety valves provide overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

b) The reactor coolant pumps (RCPs) have a rotating inertia to provide RCS flow coastdown on loss of power to the pumps.

c) Each RCP flywheel assembly can withstand a design overspeed condition.

d) The RCS provides automatic depressurization during design basis events.

e) The RCS provides emergency letdown during design basis events.

9. The RCS provides the following nonsafety-related functions:

a) The RCS provides circulation of coolant to remove heat from the core.

b) The RCS provides the means to control system pressure.

c) The pressurizer heaters trip after a signal is generated by the PMS.

10. Safety-related displays identified in Table 2.1.2-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.1.2-1 to perform active functions.

b) The valves identified in Table 2.1.2-1 as having protection and safety monitoring system (PMS) control perform an active safety function after receiving a signal from the PMS.

c) The valves identified in Table 2.1.2-1 as having diverse actuation system (DAS) control perform an active safety function after receiving a signal from DAS.

12. a) The valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.

b) After loss of motive power, the remotely operated valves identified in Table 2.1.2-1 assume the indicated loss of motive power position.

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13. a) Controls exist in the MCR to trip the RCPs.

    b) The RCPs trip after receiving a signal from the PMS.

    c) The RCPs trip after receiving a signal from the DAS.

14. Controls exist in the MCR to cause the components identified in Table 2.1.2-3 to perform the listed function.

15. Displays of the parameters identified in Table 2.1.2-3 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1.2-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the RCS.

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Table 2.1.2-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Steam Generator 1	RCS-MB-01	Yes	Yes	-	-/-	-	-	-	-
Steam Generator 2	RCS-MB-02	Yes	Yes	-	-/-	-	-	-	-
RCP 1A	RCS-MP-01A	Yes	Yes	-	No/No	No	Yes/Yes (pump trip)	No	-
RCP 1B	RCS-MP-01B	Yes	Yes	-	No/No	No	Yes/Yes (pump trip)	No	-
RCP 2A	RCS-MP-02A	Yes	Yes	-	No/No	No	Yes/Yes (pump trip)	No	-
RCP 2B	RCS-MP-02B	Yes	Yes	-	No/No	No	Yes/Yes (pump trip)	No	-
Pressurizer	RCS-MV-02	Yes	Yes	-	No/No (heaters)	-	Yes/No (heater trip)	No	-
Automatic Depressurization System (ADS) Sparger A	PXS-MW-01A	Yes	Yes	-	-/-	-	-/-	-	-
ADS Sparger B	PXS-MW-01B	Yes	Yes	-	-/-	-	-/-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Pressurizer Safety Valve	RCS-PL-V005A	Yes	Yes	No	-/-	No	-/-	Transfer Open/ Transfer Closed	-
Pressurizer Safety Valve	RCS-PL-V005B	Yes	Yes	No	-/-	No	-/-	Transfer Open/ Transfer Closed	-
First-stage ADS Motor-operated Valve (MOV)	RCS-PL-V001A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
First-stage ADS MOV	RCS-PL-V001B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Second-stage ADS MOV	RCS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Second-stage ADS MOV	RCS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Third-stage ADS MOV	RCS-PL-V003A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is

Note: Dash (-) indicates not applicable.

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Third-stage ADS MOV	RCS-PL-V003B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS Squib Valve	RCS-PL-V004A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS Squib Valve	RCS-PL-V004B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS Squib Valve	RCS-PL-V004C	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS Squib Valve	RCS-PL-V004D	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
ADS Discharge Header A Vacuum Relief Valve	RCS-PL-V010A	Yes	Yes	No	No/Yes	No	No/No	Transfer Open	-
ADS Discharge Header B Vacuum Relief Valve	RCS-PL-V010B	Yes	Yes	No	No/Yes	No	No/No	Transfer Open	-
First-stage ADS Isolation MOV	RCS-PL-V011A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
First-stage ADS Isolation MOV	RCS-PL-V011B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Second-stage ADS Isolation MOV	RCS-PL-V012A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is

Note: Dash (-) indicates not applicable.

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Second-stage ADS Isolation MOV	RCS-PL-V012B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Third-stage ADS Isolation MOV	RCS-PL-V013A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Third-stage ADS Isolation MOV	RCS-PL-V013B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS MOV	RCS-PL-V014A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	None	As Is
Fourth-stage ADS MOV	RCS-PL-V014B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	None	As Is
Fourth-stage ADS MOV	RCS-PL-V014C	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	None	As Is
Fourth-stage ADS MOV	RCS-PL-V014D	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	None	As Is
Reactor Vessel Head Vent Valve	RCS-PL-V150A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed/ Transfer Open	Closed
Reactor Vessel Head Vent Valve	RCS-PL-V150B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed/ Transfer Open	Closed

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Reactor Vessel Head Vent Valve	RCS-PL-V150C	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed/ Transfer Open	Closed
Reactor Vessel Head Vent Valve	RCS-PL-V150D	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed/ Transfer Open	Closed
RCS Hot Leg 1 Flow Sensor	RCS-101A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 1 Flow Sensor	RCS-101B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 1 Flow Sensor	RCS-101C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 1 Flow Sensor	RCS-101D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 2 Flow Sensor	RCS-102A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 2 Flow Sensor	RCS-102B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 2 Flow Sensor	RCS-102C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 2 Flow Sensor	RCS-102D	-	Yes	-	Yes/Yes	Yes	-/-	-	-

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
RCS Cold Leg 1A Narrow Range Temperature Sensor	RCS-121A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 1B Narrow Range Temperature Sensor	RCS-121B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 1B Narrow Range Temperature Sensor	RCS-121C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 1A Narrow Range Temperature Sensor	RCS-121D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 2B Narrow Range Temperature Sensor	RCS-122A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 2A Narrow Range Temperature Sensor	RCS-122B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 2A Narrow Range Temperature Sensor	RCS-122C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 2B Narrow Range Temperature Sensor	RCS-122D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 1A Dual Range Temperature Sensor	RCS-125A	-	Yes	-	Yes/Yes	Yes (Wide Range)	-/-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
RCS Cold Leg 2A Dual Range Temperature Sensor	RCS-125B	-	Yes	-	Yes/Yes	Yes (Wide Range)	-/-	-	-
RCS Cold Leg 1B Dual Range Temperature Sensor	RCS-125C	-	Yes	-	Yes/Yes	Yes (Wide Range	-/-	-	-
RCS Cold Leg 2B Dual Range Temperature Sensor	RCS-125D	-	Yes	-	Yes/Yes	Yes (Wide Range)	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-131A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-131B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-131C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-131D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-132A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-132B	-	Yes	-	Yes/Yes	No	-/-	-	-

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-132C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-132D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-133A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-133B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-133C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-133D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Wide Range Temperature Sensor	RCS-135A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 2 Wide Range Temperature Sensor	RCS-135B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Wide Range Pressure Sensor	RCS-140A	-	Yes	-	Yes/Yes	Yes	-/-	-	-

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
RCS Wide Range Pressure Sensor	RCS-140B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Wide Range Pressure Sensor	RCS-140C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Wide Range Pressure Sensor	RCS-140D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 1 Level Sensor	RCS-160A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 2 Level Sensor	RCS-160B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Passive Residual Heat Removal (PRHR) Return Line Temperature Sensor	RCS-161	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Pressure Sensor	RCS-191A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Pressure Sensor	RCS-191B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Pressure Sensor	RCS-191C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Pressure Sensor	RCS-191D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Reference Leg Temperature Sensor	RCS-193A	-	Yes	-	Yes/Yes	Yes	-/-	-	-

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Pressurizer Level Reference Leg Temperature Sensor	RCS-193B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Reference Leg Temperature Sensor	RCS-193C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Reference Leg Temperature Sensor	RCS-193D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Sensor	RCS-195A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Sensor	RCS-195B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Sensor	RCS-195C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Sensor	RCS-195D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1A Bearing Water Temperature Sensor	RCS-211A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1A Bearing Water Temperature Sensor	RCS-211B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1A Bearing Water Temperature Sensor	RCS-211C	-	Yes	-	Yes/Yes	Yes	-/-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
RCP 1A Bearing Water Temperature Sensor	RCS-211D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1B Bearing Water Temperature Sensor	RCS-212A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1B Bearing Water Temperature Sensor	RCS-212B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1B Bearing Water Temperature Sensor	RCS-212C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1B Bearing Water Temperature Sensor	RCS-212D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2A Bearing Water Temperature Sensor	RCS-213A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2A Bearing Water Temperature Sensor	RCS-213B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2A Bearing Water Temperature Sensor	RCS-213C	-	Yes	-	Yes/Yes	Yes	-/-	-	-

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Table 2.1.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
RCP 2A Bearing Water Temperature Sensor	RCS-213D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2B Bearing Water Temperature Sensor	RCS-214A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2B Bearing Water Temperature Sensor	RCS-214B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2B Bearing Water Temperature Sensor	RCS-214C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2B Bearing Water Temperature Sensor	RCS-214D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1A Pump Speed Sensor	RCS-281	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1B Pump Speed Sensor	RCS-282	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2A Pump Speed Sensor	RCS-283	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 2B Pump Speed Sensor	RCS-284	-	Yes	-	Yes/Yes	Yes	-/-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.1.2-2

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
Hot Legs	RCS-L001A RCS-L001B	Yes	Yes	Yes
Cold Legs	RCS-L002A RCS-L002B RCS-L002C RCS-L002D	Yes	Yes	Yes
Pressurizer Surge Line	RCS-L003	Yes	Yes	Yes
ADS Inlet Headers	RCS-L004A/B RCS-L006A/B RCS-L030A/B RCS-L020A/B	Yes	Yes	Yes
Safety Valve Inlet Piping	RCS-L005A RCS-L005B	Yes	Yes	Yes
Safety Valve Discharge Piping	RCS-L050A/B RCS-L051A/B	Yes	No	Yes
	RCS-L064A/B	Yes	No	No
ADS First-stage Valve Inlet Piping	RCS-L010A/B RCS-L011A/B	Yes	No	Yes
ADS Second-stage Valve Inlet Piping	RCS-L021A/B RCS-L022A/B	Yes	Yes No	Yes
ADS Third-stage Valve Inlet Piping	RCS-L131 RCS-L031A/B RCS-L032A/B	Yes	Yes Yes No	Yes
ADS Outlet Piping	RCS-L012A/B RCS-L023A/B RCS-L033A/B RCS-L061A/B RCS-L063A/B RCS-L200 RCS-L069A/B PXS-L130A/B	Yes	No	Yes
	RCS-L240A/B	Yes	No	No
ADS Fourth-stage Inlet Piping	RCS-L133A/B RCS-L135A/B RCS-L136A/B RCS-L137A/B	Yes	Yes	Yes

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Table 2.1.2-2 (cont.)

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
Pressurizer Spray Piping	RCS-L106 RCS-L110A/B RCS-L212A/B RCS-L213 RCS-L215	Yes	No	No
RNS Suction Piping	RCS-L139 RCS-L140	Yes	Yes	No
CVS Purification Piping	RCS-L111 RCS-L112	Yes	No	No

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Table 2.1.2-3

Equipment	Tag No.	Display	Control Function
RCP 1A Breaker (Status)	ECS-ES-31	Yes	-
RCP 1A Breaker (Status)	ECS-ES-32	Yes	-
RCP 1B Breaker (Status)	ECS-ES-41	Yes	-
RCP 1B Breaker (Status)	ECS-ES-42	Yes	-
RCP 2A Breaker (Status)	ECS-ES-51	Yes	-
RCP 2A Breaker (Status)	ECS-ES-52	Yes	-
RCP 2B Breaker (Status)	ECS-ES-61	Yes	-
RCP 2B Breaker (Status)	ECS-ES-62	Yes	-
Pressurizer Heaters	RCS-EH-03	Yes	On/Off
Pressurizer Heaters	RCS-EH-04A	Yes	On/Off
Pressurizer Heaters	RCS-EH-04B	Yes	On/Off
Pressurizer Heaters	RCS-EH-04C	Yes	On/Off
Pressurizer Heaters	RCS-EH-04D	Yes	On/Off
Fourth-stage ADS Squib Valve (Position Indication)	RCS-PL-V004A	Yes	-
Fourth-stage ADS Squib Valve (Position Indication)	RCS-PL-V004B	Yes	-
Fourth-stage ADS Squib Valve (Position Indication)	RCS-PL-V004C	Yes	-
Fourth-stage ADS Squib Valve (Position Indication)	RCS-PL-V004D	Yes	-
Pressurizer Safety Valve (Position Indication)	RCS-PL-V005A	Yes	-
Pressurizer Safety Valve (Position Indication)	RCS-PL-V005B	Yes	-
Pressurizer Spray Valve (Position Indication)	RCS-PL-V110A	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.1.2-3 (cont.)

Equipment	Tag No.	Display	Control Function
Pressurizer Spray Valve (Position Indication)	RCS-PL-V110B	Yes	-
Reactor Vessel Head Vent Valve (Position Indication)	RCS-PL-V150A	Yes	-
Reactor Vessel Head Vent Valve (Position Indication)	RCS-PL-V150B	Yes	-
Reactor Vessel Head Vent Valve (Position Indication)	RCS-PL-V150C	Yes	-
Reactor Vessel Head Vent Valve (Position Indication)	RCS-PL-V150D	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.1.2-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the RCS is as described in the Design Description of this Section 2.1.2.	Inspection of the as-built system will be performed.	The as-built RCS conforms with the functional arrangement described in the Design Description of this Section 2.1.2.
2.a) The components identified in Table 2.1.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.1.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.1.2-1 and 2.1.2-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.1.2-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.1.2-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.1.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The piping identified in Table 2.1.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the components and piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components and piping identified in Tables 2.1.2-1 and 2.1.2-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

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Table 2.1.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b) Each of the lines identified in Table 2.1.2-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.1.2-2 for which functional capability is required meets the requirements for functional capability.
6. Each of the as-built lines identified in Table 2.1.2-2 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	Inspection will be performed for the existence of an LBB evaluation report or an evaluation report on the protection from dynamic effects of a pipe break. Tier 1 Material, Section 3.3, Nuclear Island Buildings, contains the design descriptions and inspections, tests, analyses, and acceptance criteria for protection from the dynamic effects of pipe rupture.	An LBB evaluation report exists and concludes that the LBB acceptance criteria are met by the as-built RCS piping and piping materials, or a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided.
5.a) The seismic Category I equipment identified in Table 2.1.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.1.2-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.1.2-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.

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Table 2.1.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) The Class 1E equipment identified in Table 2.1.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.1.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.1.2-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
7.b) Not used.		
7.c) Not used.		
8.a) The pressurizer safety valves provide overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	i) Inspections will be conducted to confirm that the value of the vendor code plate rating is greater than or equal to system relief requirements.	i) The sum of the rated capacities recorded on the valve ASME Code plates of the safety valves exceeds 1,500,000 lb/hr.
	ii) Testing and analysis in accordance with ASME Code Section III will be performed to determine set pressure.	ii) A report exists and concludes that the safety valves set pressure is 2485 psig $\pm$ 25 psi.
8.b) The RCPs have a rotating inertia to provide RCS flow coastdown on loss of power to the pumps.	A test will be performed to determine the pump flow coastdown curve.	The pump flow coastdown will provide RCS flows greater than or equal to the flow shown in Figure 2.1.2-2, “Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down.”

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Table 2.1.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.c) Each RCP flywheel assembly can withstand a design overspeed condition.	Shop testing of each RCP flywheel assembly will be performed at the vendor facility at overspeed conditions.	Each RCP flywheel assembly has passed an overspeed condition of no less than 125% of operating speed.
8.d) The RCS provides automatic depressurization during design basis events.	i) A low pressure flow test and associated analysis will be conducted to determine the total piping flow resistance of each ADS valve group connected to the pressurizer (i.e., ADS Stages 1-3) from the pressurizer through the outlet of the downstream ADS control valves. The reactor coolant system will be at cold conditions with the pressurizer full of water. The normal residual heat removal pumps will be used to provide injection flow into the RCS discharging through the ADS valves. Inspections and associated analysis of the piping flow paths from the discharge of the ADS valve groups connected to the pressurizer (i.e., ADS Stages 1-3) to the spargers will be conducted to verify the line routings are consistent with the line routings used for design flow resistance calculations.	i) The calculated ADS piping flow resistance from the pressurizer through the sparger with all valves of each ADS group open is $\leq$ 2.91E-6 ft/ gpm^2.
	ii) Inspections and associated analysis of each fourth-stage ADS sub-loop (four valves and associated piping connected to each hot leg) will be conducted to verify the line routing is consistent with the line routing used for design flow resistance calculations.	ii) The calculated flow resistance for each fourth-stage ADS sub-loop valves and piping is: Loop 1: Sub-loop A: $\leq 5.91 \times 10^{-7}$ ft/gpm^2 Sub-loop C: $\leq 6.21 \times 10^{-7}$ ft/gpm^2 Loop 2: Sub-loop B: $\leq 4.65 \times 10^{-7}$ ft/gpm^2 Sub-loop D: $\leq 6.20 \times 10^{-7}$ ft/gpm^2
	iii) Inspections of each fourth-stage ADS valve will be conducted to determine the as-manufactured flow area through each valve.	iii) The as-manufactured flow area through each fourth-stage ADS valve is $\geq$ 67 in^2.

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Table 2.1.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	iv) Type tests and analysis will be performed to determine the effective flow area through each stage 1,2,3 ADS valve. vii) Inspection of each ADS sparger will be conducted to determine the flow area through the sparger holes.	iv) A report exists and concludes that the effective flow area through each stage 1 ADS valve $\geq$ 4.6 in^2 and each stage 2,3 ADS valve is $\geq$ 16 in^2. vii) The flow area through the holes in each ADS sparger is $\geq$ 274 in^2.
8.d) The RCS provides automatic depressurization during design basis events.	v) Inspections of the elevation of the ADS stage 4 valve discharge will be conducted. vi) Inspections of the ADS stage 4 valve discharge will be conducted. viii) Inspection of the elevation of each ADS sparger will be conducted.	v) The minimum elevation of the bottom inside surface of the outlet of these valves is greater than plant elevation 110 feet. vi) The discharge of the ADS stage 4 valves is directed into the steam generator compartments. viii) The centerline of the connection of the sparger arms to the sparger hub is $\leq$ 11.5 feet below the IRWST overflow level.
8.e) The RCS provides emergency letdown during design basis events.	Inspections of the reactor vessel head vent valves and inlet and outlet piping will be conducted.	A report exists and concludes that the capacity of the reactor vessel head vent is sufficient to pass not less than 9.0 lbm/sec at 2500 psia in the RCS.
9.a) The RCS provides circulation of coolant to remove heat from the core.	Testing and analysis to measure RCS flow with four reactor coolant pumps operating at no-load RCS pressure and temperature conditions will be performed. Analyses will be performed to convert the measured pre-fuel load flow to post-fuel load flow with 10-percent steam generator tube plugging.	The calculated post-fuel load RCS flow rate is $\geq$ 301,670 gpm.

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Table 2.1.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.b) The RCS provides the means to control system pressure.	i) Inspections will be performed to verify the rated capacity of pressurizer heater backup groups A and B. ii) Tests will be performed to verify that the pressurizer spray valves can open and close when operated from the MCR.	i) Pressurizer heater backup groups A and B each has a rated capacity of at least 168 kW. ii) Controls in the MCR operate to cause the pressurizer spray valves to open and close.
9.c) The pressurizer heaters trip after a signal is generated by the PMS.	Testing will be performed to confirm trip of the pressurizer heaters identified in Table 2.1.2-3.	The pressurizer heaters identified in Table 2.1.2-3 trip after a signal is generated by the PMS.
10. Safety-related displays identified in Table 2.1.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.1.2-1 can be retrieved in the MCR.
11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.1.2-1 to perform active functions.	ii) Stroke testing will be performed on the other remotely operated valves listed in Table 2.1.2-1 using controls in the MCR.	ii) Controls in the MCR operate to cause the remotely operated valves (other than squib valves) to perform active functions.
11.b) The valves identified in Table 2.1.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	ii) Testing will be performed on the other remotely operated valves identified in Table 2.1.2-1 using real or simulated signals into the PMS. iii) Testing will be performed to demonstrate that remotely operated RCS valves RCS-V001A/B, V002A/B, V003A/B, V011A/B, V012A/B, V013A/B open within the required response times.	ii) The other remotely operated valves identified in Table 2.1.2-1 as having PMS control perform the active function identified in the table after receiving a signal from PMS. iii) These valves open within the following times after receipt of an actuation signal: V001A/B ≤ 48 sec V002A/B, V003A/B ≤ 120 sec V011A/B ≤ 30 sec V012A/B, V013A/B ≤ 60 sec
12.b) After loss of motive power, the remotely operated valves identified in Table 2.1.2-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valve identified in Table 2.1.2-1 assumes the indicated loss of motive power position.

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Table 2.1.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.1.2-1 to perform active functions.	i) Testing will be performed on the squib valves identified in Table 2.1.2-1 using controls in the MCR without stroking the valve.	i) Controls in the MCR operate to cause a signal at the squib valve electrical leads which is capable of actuating the squib valve.
11.b) The valves identified in Table 2.1.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	i) Testing will be performed on the squib valves identified in Table 2.1.2-1 using real or simulated signals into the PMS without stroking the valve.	i) The squib valves receive a signal at the valve electrical leads that is capable of actuating the squib valve.
11.c) The valves identified in Table 2.1.2-1 as having DAS control perform an active safety function after receiving a signal from DAS.	i) Testing will be performed on the squib valves identified in Table 2.1.2-1 using real or simulated signals into the DAS without stroking the valve. ii) Testing will be performed on the other remotely operated valves identified in Table 2.1.2-1 using real or simulated signals into the DAS.	i) The squib valves receive a signal at the valve electrical leads that is capable of actuating the squib valve. ii) The other remotely operated valves identified in Table 2.1.2-1 as having DAS control perform the active function identified in the table after receiving a signal from DAS.

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Table 2.1.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.1.2-1 under design conditions.
	ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tests or type tests.	ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tests or type tests.
	iii) Tests of the motor-operated valves will be performed under pre-operational flow, differential pressure and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.1.2-1 under pre-operational test conditions.
	iv) Tests or type tests of squib valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	iv) A test report exists and concludes that each squib valve changes position as indicated in Table 2.1.2-1 under design conditions.
	v) Inspection will be performed for the existence of a report verifying that the as-built squib valves are bounded by the tests or type tests.	v) A report exists and concludes that the as-built squib valves are bounded by the tests or type tests.
	vi) Not used.	
	vii) Not used.	
	viii) See item 8.d.iii in this table.	viii) See item 8.d.iii in this table.
	ix) See item 8.d.iv in this table.	ix) See item 8.d.iv in this table.
13.a) Controls exist in the MCR to trip the RCPs.	Testing will be performed on the RCPs using controls in the MCR.	Controls in the MCR operate to trip the RCPs.
13.b) The RCPs trip after receiving a signal from the PMS.	Testing will be performed using real or simulated signals into the PMS.	The RCPs trip after receiving a signal from the PMS.
13.c) The RCPs trip after receiving a signal from the DAS.	Testing will be performed using real or simulated signals into the DAS.	The RCPs trip after receiving a signal from the DAS.

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Table 2.1.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
14. Controls exist in the MCR to cause the components identified in Table 2.1.2-3 to perform the listed function.	Testing will be performed on the components in Table 2.1.2-3 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.1.2-3 to perform the listed functions.
15. Displays of the parameters identified in Table 2.1.2-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the RCS parameters in the MCR.	The displays identified in Table 2.1.2-3 can be retrieved in the MCR.

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Table 2.1.2-5

Component Name	Tag No.	Component Location
Steam Generator 1	RCS-MB-01	Containment
Steam Generator 2	RCS-MB-02	Containment
Reactor Coolant Pump 1A	RCS-MP-01A	Containment
Reactor Coolant Pump 1B	RCS-MP-01B	Containment
Reactor Coolant Pump 2A	RCS-MP-02A	Containment
Reactor Coolant Pump 2B	RCS-MP-02B	Containment
Pressurizer	RCS-MV-02	Containment
ADS Sparger A	PXS-MW-01A	Containment
ADS Sparger B	PXS-MW-01B	Containment

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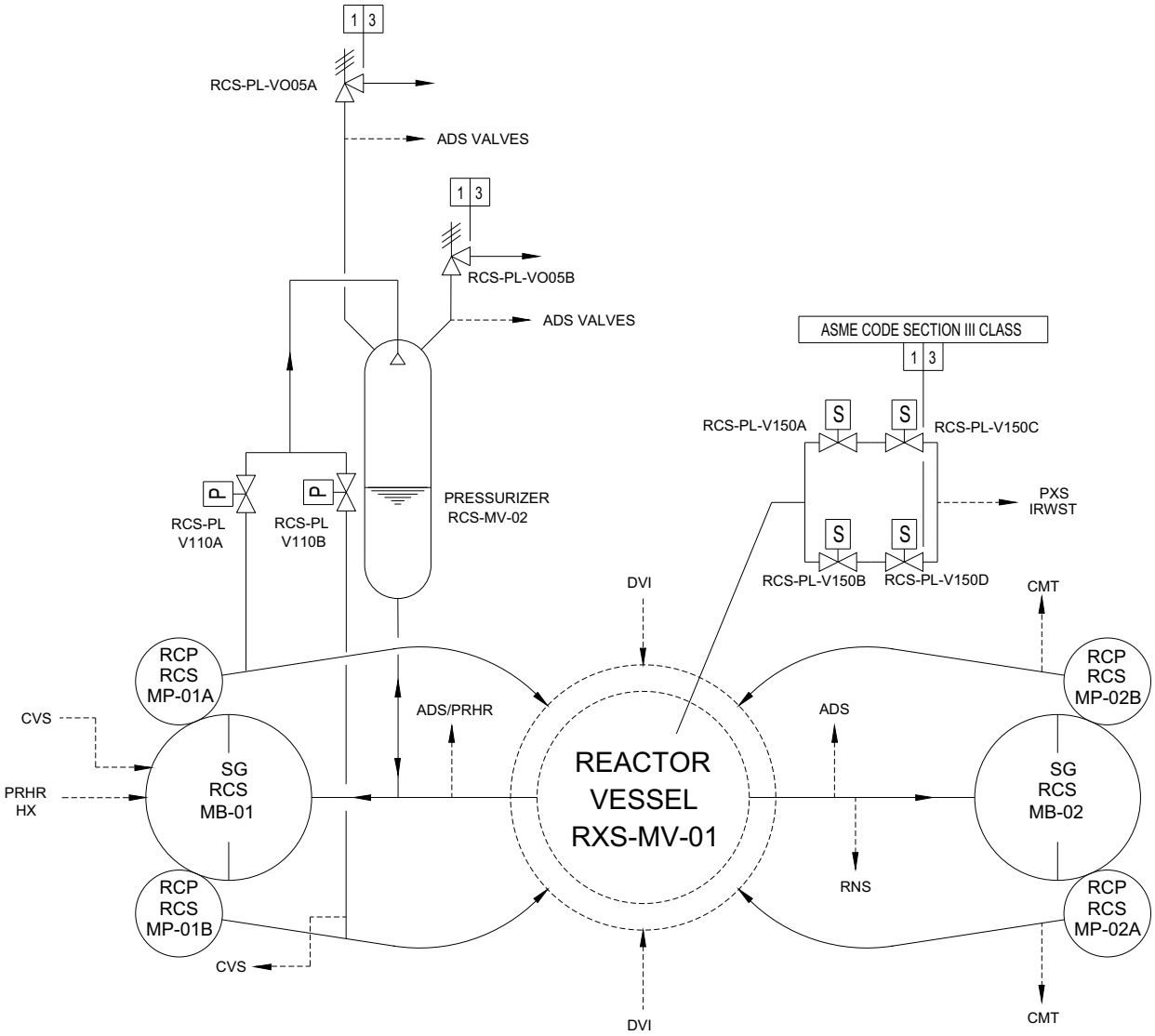

Figure 2.1.2-1 (Sheet 1 of 2)
Reactor Coolant System

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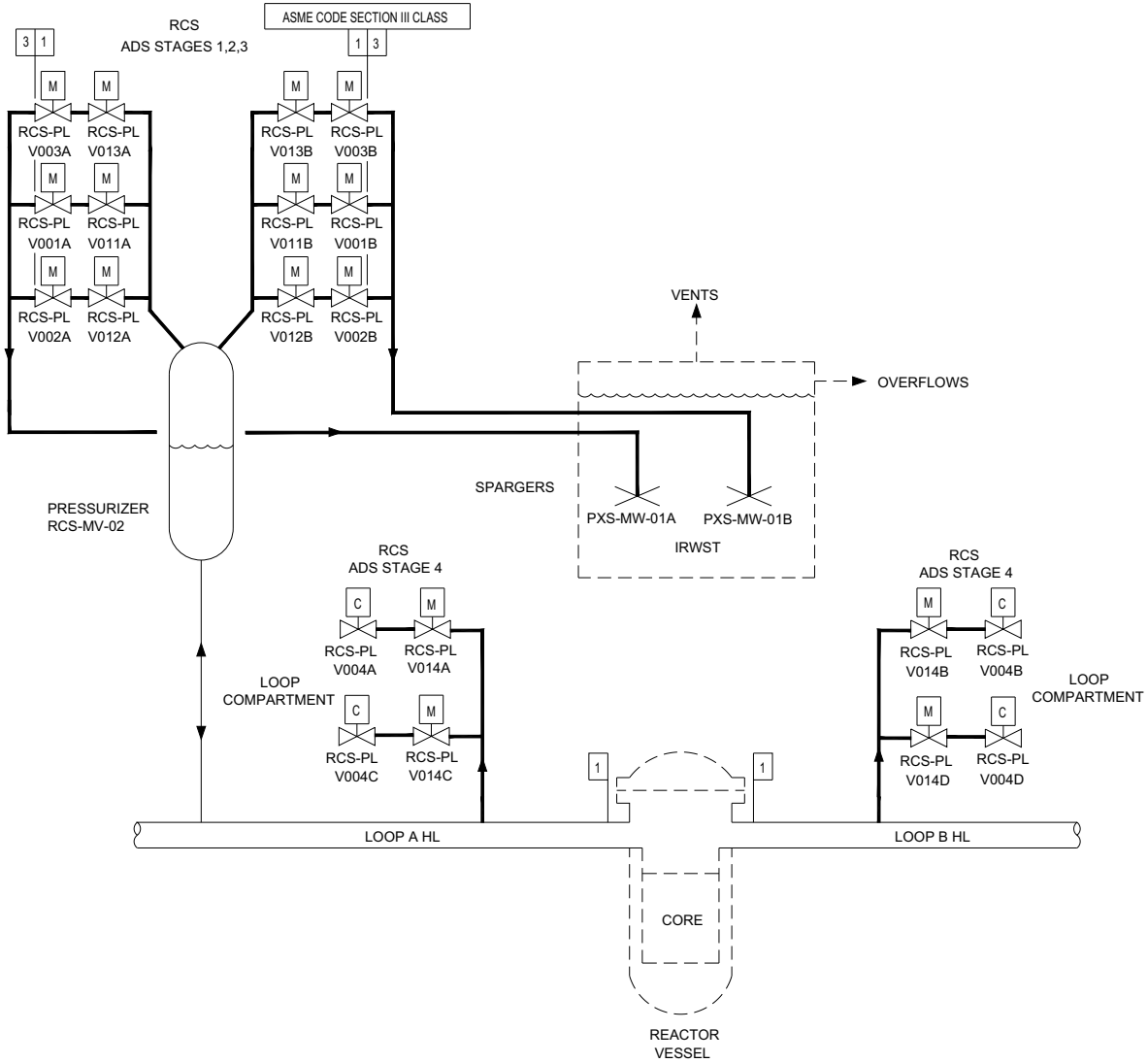

Figure 2.1.2-1 (Sheet 2 of 2)
Reactor Coolant System

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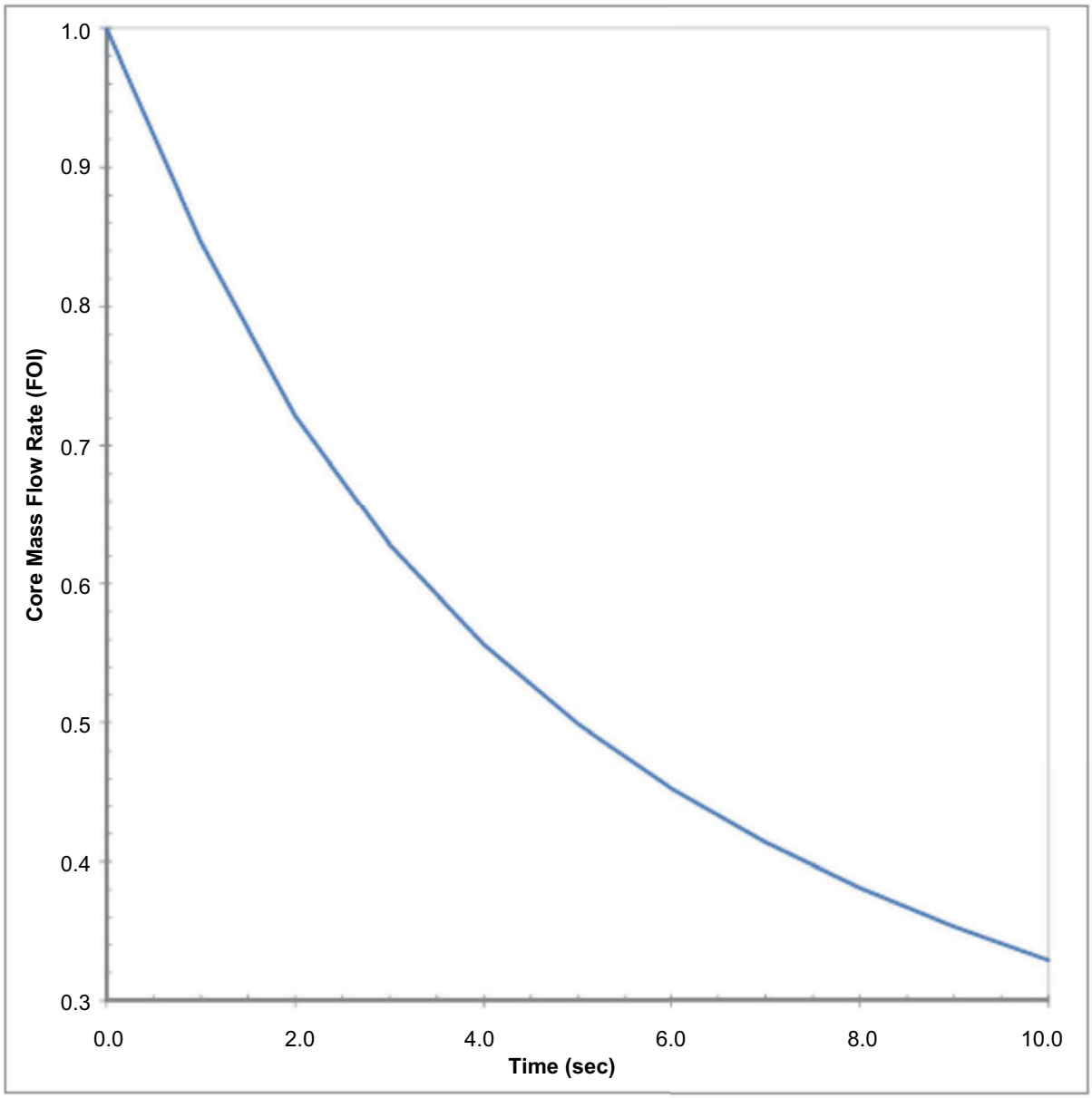

Note: FOI = Fraction of Initial

Figure 2.1.2-2
Flow Transient for Four Cold Legs
in Operation, Four Pumps Coasting Down

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2.1.3 Reactor System

Design Description

The reactor system (RXS) generates heat by a controlled nuclear reaction and transfers the heat generated to the reactor coolant, provides a barrier that prevents the release of fission products to the atmosphere and a means to insert negative reactivity into the reactor core and to shutdown the reactor core.

The reactor core contains a matrix of fuel rods assembled into fuel assemblies using structural elements. Rod cluster control assemblies (RCCAs) are positioned and held within the fuel assemblies by control rod drive mechanisms (CRDMs). The CRDMs unlatch upon termination of electrical power to the CRDM thereby releasing the RCCAs. The fuel assemblies and RCCAs are designed in accordance with the principal design requirements.

The RXS is operated during normal modes of plant operation, including startup, power operation, cooldown, shutdown and refueling.

The component locations of the RXS are as shown in Table 2.1.3-3.

1. The functional arrangement of the RXS is as described in the Design Description of this Section 2.1.3.

2. a) The reactor upper internals rod guide arrangement is as shown in Figure 2.1.3-1.

   b) The rod cluster control and drive rod arrangement is as shown in Figure 2.1.3-2.

   c) The reactor vessel arrangement is as shown in Figure 2.1.3-3.

3. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

4. Pressure boundary welds in components identified in Table 2.1.3-1 as ASME Code Section III meet ASME Code Section III requirements.

5. The pressure boundary components (reactor vessel [RV], control rod drive mechanisms [CRDMs], and incore instrument QuickLoc assemblies) identified in Table 2.1.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.

7. The reactor internals will withstand the effects of flow induced vibration.

8. The reactor vessel direct injection nozzle limits the blowdown of the reactor coolant system (RCS) following the break of a direct vessel injection line.

9. a) The Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

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b) The Class 1E components identified in Table 2.1.3-1 are powered from their respective Class 1E division.

c) Separation is provided between RXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

10. The reactor lower internals assembly is equipped with holders for at least eight capsules for storing material surveillance specimens.

11. The reactor pressure vessel (RPV) beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.

12. Safety-related displays of the parameters identified in Table 2.1.3-1 can be retrieved in the main control room (MCR).

13. The fuel assemblies and rod cluster control assemblies intended for initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.

14. A top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle, can be performed.

Inspections, Tests, Analysis, and Acceptance Criteria

Table 2.1.3-2 specifies the inspections, tests, analysis, and associated acceptance criteria for the RXS.

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Table 2.1.3-1

Equipment Name	Tag No.	ASME Code Section III Classification	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
RV	RXS-MV-01	Yes	Yes	-	-
Reactor Upper Internals Assembly	RXS-MI-01	Yes	Yes	-	-
Reactor Lower Internals Assembly	RXS-MI-02	Yes	Yes	-	-
Fuel Assemblies (157 locations)	RXS-FA-A07/A08/A09/B05/B06/B07/B08/ B09/B10/B11/C04/C05/C06/C07/C08/C09/C10/ C11/C12/D03/D04/D05/D06/D07/D08/D09/ D10/D11/D12/D13/E02/E03/E04/E05/E06/E07/ E08/E09/E10/E11/E12/E13/E14/F02/F03/F04/ F05/F06/F07/F08/F09/F10/F11/F12/F13/F14/ G01/G02/G03/G04/G05/G06/G07/G08/G09/ G10/G11/G12/G13/G14/G15/H01/H02/H03/ H04/H05/H06/H07/H08/H09/H10/H11/H12/ H13/H14/H15/J01/J02/J03/J04/J05/J06/J07/J08/ J09/J10/J11/J12/J13/J14/J15/K02/K03/K04/ K05/K06/K07/K08/K09/K10/K11/K12/K13/ K14/L02/L03/L04/L05/L06/L07/L08/L09/L10/ L11/L12/L13/L14/M03/M04/M05/M06/M07/ M08/M09/M10/M11/M12/M13/N04/N05/N06/ N07/N08/N09/N10/N11/N12/P05/P06/P07/P08/ P09/P10/P11/ R07/R08/R09	No(1)	Yes	-	-

Note: Dash (-) indicates not applicable.

1. Fuel assemblies are designed using ASME Section III as a general guide.

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Table 2.1.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III Classification	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Rod Cluster Control Assemblies (RCCAs) (minimum 53 locations)	RXS-FR-B06/B10/C05/C07/C09/C11/D06/ D08/D10/E03/E05/E07/E09/E11/E13/F02/F04/ F12/F14/G03/G05/G07/G09/G11/G13/H04/ H08/H12/J03/J05/J07/J09/J11/J13/K02/K04/ K12/K14/L03/L05/L07/L09/L11/L13/M06/ M08/M10/N05/N07/N09/N11/P06/P10	No(1)	Yes	-	-
Gray Rod Cluster Assemblies (GRCAs) (16 locations)	RXS-FG-B08/D04/D12/F06/F08/F10/H02/H06/ H10/H14/K06/K08/K10/M04/M12/P08	No(1)	Yes	-	-
Control Rod Drive Mechanisms (CRDMs) (69 Locations)	RXS-MV-11B06/11B08/11B10/11C05/11C07/ 11C09/11C11/11D04/11D06/11D08/11D10/ 11D12/11E03/11E05/11E07/11E09/11E11/ 11E13/11F02/11F04/11F06/11F08/11F10/ 11F12/11F14/11G03/11G05/11G07/11G09/ 11G11/11G13/11H02/11H04/11H06/11H08/ 11H10/11H12/11H14/11J03/11J05/11J07/ 11J09/11J11/11J13/11K02/11K04/11K06/ 11K08/11K10/11K12/11K14/11L03/11L05/ 11L07/11L09/11L11/11L13/11M04/11M06/ 11M08/11M10/11M12/11N05/11N07/11N09/ 11N11/11P06/11P08/11P10	Yes	Yes	No/No	No
Incore Instrument QuickLoc Assemblies (8 Locations)	RXS-MY-Y11 through Y18	Yes	Yes	-	-

Note: Dash (-) indicates not applicable.

1. Fuel assemblies are designed using ASME Section III as a general guide.

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Table 2.1.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III Classification	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Source Range Detectors (4)	RXS-JE-NE001A/NE001B/NE001C/NE001D	-	Yes	Yes/Yes	No
Intermediate Range Detectors (4)	RXS-JE-NE002A/NE002B/NE002C/NE002D	-	Yes	Yes/Yes	Yes
Power Range Detectors – Lower (4)	RXS-JE-NE003A/NE003B/NE003C/NE003D	-	Yes	Yes/Yes	No
Power Range Detectors – Upper (4)	RXS-JE-NE004A/NE004B/NE004C/NE004D	-	Yes	Yes/Yes	No

Note: Dash (-) indicates not applicable.

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Table 2.1.3-2
Inspections, Tests, Analysis, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analysis	Acceptance Criteria
1. Not used.	Not used.	Not used.
2.a) The reactor upper internals rod guide arrangement is as shown in Figure 2.1.3-1.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the fuel assembly and control rod drive mechanism pattern shown in Figure 2.1.3-1.
2.b) The control assemblies (rod cluster and gray rod) and drive rod arrangement is as shown in Figure 2.1.3-2.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the control assemblies (rod cluster and gray rod) and drive rod arrangement shown in Figure 2.1.3-2.
2.c) The reactor vessel arrangement is as shown in Figure 2.1.3-3.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the reactor vessel arrangement shown in Figure 2.1.3-3.
3. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.1.3-1 as ASME Code Section III.
4. Pressure boundary welds in components identified in Table 2.1.3-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
5. The pressure boundary components (RV, CRDMs, and incore instrument QuickLoc assemblies) identified in Table 2.1.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components of the RXS required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the pressure boundary components (RV, CRDMs, and incore instrument QuickLoc assemblies) conform with the requirements of the ASME Code Section III.

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Table 2.1.3-2 (cont.)
Inspections, Tests, Analysis, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analysis	Acceptance Criteria
6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.1.3-1 (except fuel assemblies, rod cluster control assemblies, gray rod cluster assemblies, and incore instrument QuickLoc assemblies) is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.1.3-1 (except fuel assemblies, rod cluster control assemblies, gray rod cluster assemblies, and incore instrument QuickLoc assemblies) is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the equipment including anchorage is seismically bounded by the tested or analyzed conditions. This inspection must be performed on the as-built equipment except for the fuel assemblies, rod cluster control assemblies, gray rod cluster assemblies, and incore instrument QuickLoc assemblies.	iii) A report exists and concludes that the equipment including anchorage is seismically bounded by the tested or analyzed conditions.
9.a) The Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analysis, or a combination of type tests and analysis will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.1.3-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.

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Table 2.1.3-2 (cont.)
Inspections, Tests, Analysis, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analysis	Acceptance Criteria
7. The reactor internals will withstand the effects of flow induced vibration.	i) Not used.	i) Not used.
	ii) A pre-test inspection, a flow test and a post-test inspection will be conducted on the as-built reactor internals.	ii) The as-built reactor internals have no observable damage or loose parts.
10. The reactor lower internals assembly is equipped with holders for at least eight capsules for storing material surveillance specimens.	Inspection of the reactor lower internals assembly for the presence of capsules will be performed.	At least eight capsules are in the reactor lower internals assembly.
8. The reactor vessel direct vessel injection nozzle limits the blowdown of the RCS following the break of a direct vessel injection line.	An inspection will be conducted to verify the flow area of the flow limiting venturi within each direct vessel injection nozzle.	The throat area of the direct vessel injection line nozzle flow limiting venturi is less than or equal to 12.57 in^2.
9.b) Not used.		
9.c) Not used.		
11. The RPV beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.	Manufacturing tests of the Charpy V-Notch specimen of the RPV beltline material will be performed.	A report exists and concludes that the initial RPV beltline Charpy upper-shelf energy is no less than 75 ft-lb.
12. Safety-related displays of the parameters identified in Table 2.1.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.1.3-1 can be retrieved in the MCR.
13. The fuel assemblies and rod cluster control assemblies intended for initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the established design requirements.	An analysis is performed of the reactor core design.	A report exists and concludes that the fuel assemblies and rod cluster control assemblies intended for the initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.
14. A top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle, can be performed.	A preservice visual examination of the reactor vessel head top surface and penetration nozzles will be performed.	A report exists that documents the results of the top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle.

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Table 2.1.3-3

Component Name	Tag No.	Component Location
RV	RXS-MV-01	Containment
Reactor Upper Internals Assembly	RXS-MI-01	Containment
Reactor Lower Internals Assembly	RXS-MI-02	Containment
Fuel Assemblies (157 locations)	RXS-FA-A07/A08/A09/B05/ B06/B07/B08/B09/B10/B11/ C04/C05/C06/C07/C08/C09/ C10/C11/C12/D03/D04/D05/ D06/D07/D08/D09/D10/D11/ D12/D13/E02/E03/E04/E05/ E06/E07/E08/E09/E10/E11/E12/ E13/E14/F02/F03/F04/F05/F06/ F07/F08/F09/F10/F11/F12/F13/ F14/G01/G02/G03/G04/G05/ G06/G07/G08/G09/G10/G11/ G12/G13/G14/G15/H01/H02/ H03/H04/H05/H06/H07/H08/ H09/H10/H11/H12/H13/H14/ H15/J01/J02/J03/J04/J05/J06/ J07/J08/J09/J10/J11/J12/J13/ J14/J15/K02/K03/K04/K05/ K06/K07/K08/K09/K10/K11/ K12/K13/K14/L02/L03/L04/ L05/L06/L07/L08/L09/L10/L11/ L12/L13/L14/M03/M04/M05/ M06/M07/M08/M09/M10/M11/ M12/M13/N04/N05/N06/N07/ N08/N09/N10/N11/N12/P05/ P06/P07/P08/P09/P10/P11/R07/ R08/R09	Containment (located in auxiliary building prior to fuel loading)
Rod Cluster Control Assemblies (RCCAs) (minimum 53 locations)	RXS-FR-B06/B10/C05/C07/ C09/C11/D06/D08/D10/E03/ E05/E07/E09/E11/E13/F02/F04/ F12/F14/G03/G05/G07/G09/ G11/G13/H04/H08/H12/J03/ J05/J07/J09/J11/J13/K02/K04/ K12/K14/L03/L05/L07/L09/ L11/L13/M06/M08/M10/N05/ N07/N09/N11/P06/P10	Containment (located in auxiliary building prior to fuel loading)
Gray Rod Cluster Assemblies (GRCAs) (16 locations)	RXS-FG-B08/D04/D12/F06/ F08/F10/H02/H06/H10/H14/ K06/K08/K10/M04/M12/P08	Containment (located in auxiliary building prior to fuel loading)

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Table 2.1.3-3 (cont.)

Component Name	Tag No.	Component Location
Control Rod Drive Mechanisms (CRDMs) (69 Locations)	RXS-MV-11B06/11B08/ 11B10/11C05/11C07/11C09/ 11C11/11D04/11D06/11D08/ 11D10/11D12/11E03/11E05/ 11E07/11E09/11E11/11E13/ 11F02/11F04/11F06/11F08/ 11F10/11F12/11F14/11G03/ 11G05/11G07/11G09/11G11/ 11G13/11H02/11H04/11H06/ 11H08/11H10/11H12/11H14/ 11J03/11J05/11J07/11J09/11J11/ 11J13/11K02/11K04/11K06/ 11K08/11K10/11K12/11K14/ 11L03/11L05/11L07/11L09/ 11L11/11L13/11M04/11M06/ 11M08/11M10/11M12/11N05/ 11N07/11N09/11N11/11P06/ 11P08/11P10	Containment
Incore Instrument QuickLoc Assemblies (8 Locations)	RXS-MY-Y11 through Y18	Containment
Source Range Detectors (4)	RXS-JE-NE001A/NE001B/ NE001C/NE001D	Containment
Intermediate Range Detectors (4)	RXS-JE-NE002A/NE002B/ NE002C/NE002D	Containment
Power Range Detectors – Lower (4)	RXS-JE-NE003A/NE003B/ NE003C/NE003D	Containment
Power Range Detectors – Upper (4)	RXS-JE-NE004A/NE004B/ NE004C/NE004D	Containment

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Table 2.1.3-4
Key Dimensions and Acceptable Variations of the Reactor Vessel and Internals
(Figure 2.1.3-2 and Figure 2.1.3-3)

Description	Dimension or Elevation (inches)	Nominal Value (inches)	Acceptable Variation (inches)
RV inside diameter at beltline (inside cladding)	A	159.0	+1.0/-1.0
RV wall thickness at beltline (without cladding)	B	8.4	+1.0/-0.12
RV wall thickness at bottom head (without cladding)	C	6.0	+1.0/-0.12
RV inlet nozzle inside diameter at safe end	D	22.0	+0.35/-0.10
RV outlet nozzle inside diameter at safe end	E	31.0	+0.35/-0.10
Elevation from RV mating surface to centerline of inlet nozzle	F	62.5	+0.25/-0.25
Elevation from RV mating surface to centerline of outlet nozzle	G	80.0	+0.25/-0.25
Elevation from RV mating surface to centerline of direct vessel injection nozzle	H	100.0	+0.25/-0.25
Elevation from RV mating surface to inside of RV bottom head (inside cladding)	I	397.59	+1.0/-0.50
Elevation from RV mating surface to top of lower core support plate	J	327.3	+0.50/-0.50
Separation distance between bottom of upper core plate and top of lower core support with RV head in place	K	189.8	+0.20/-0.20

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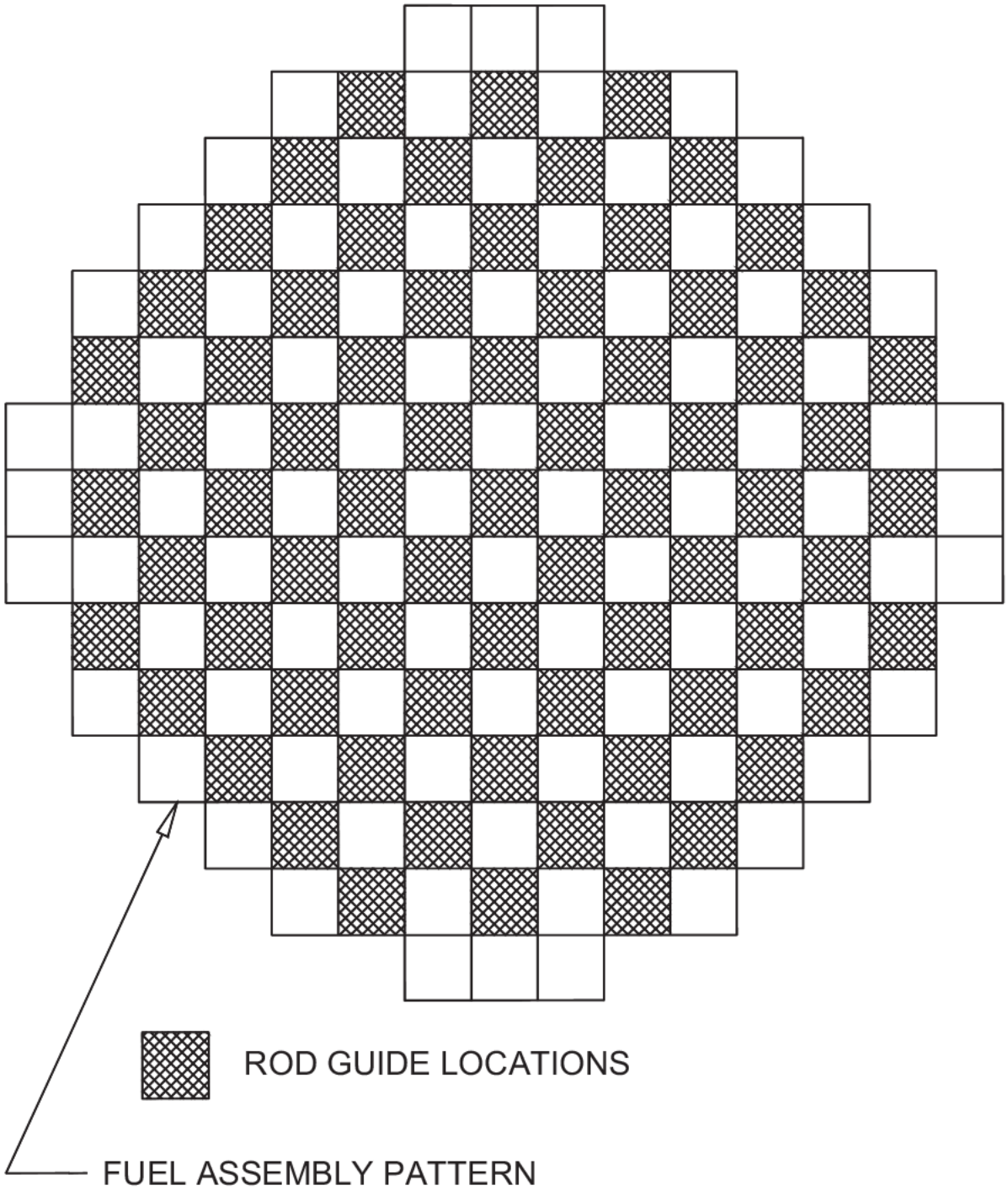

Figure 2.1.3-1
Reactor Upper Internals Rod Guide Arrangement

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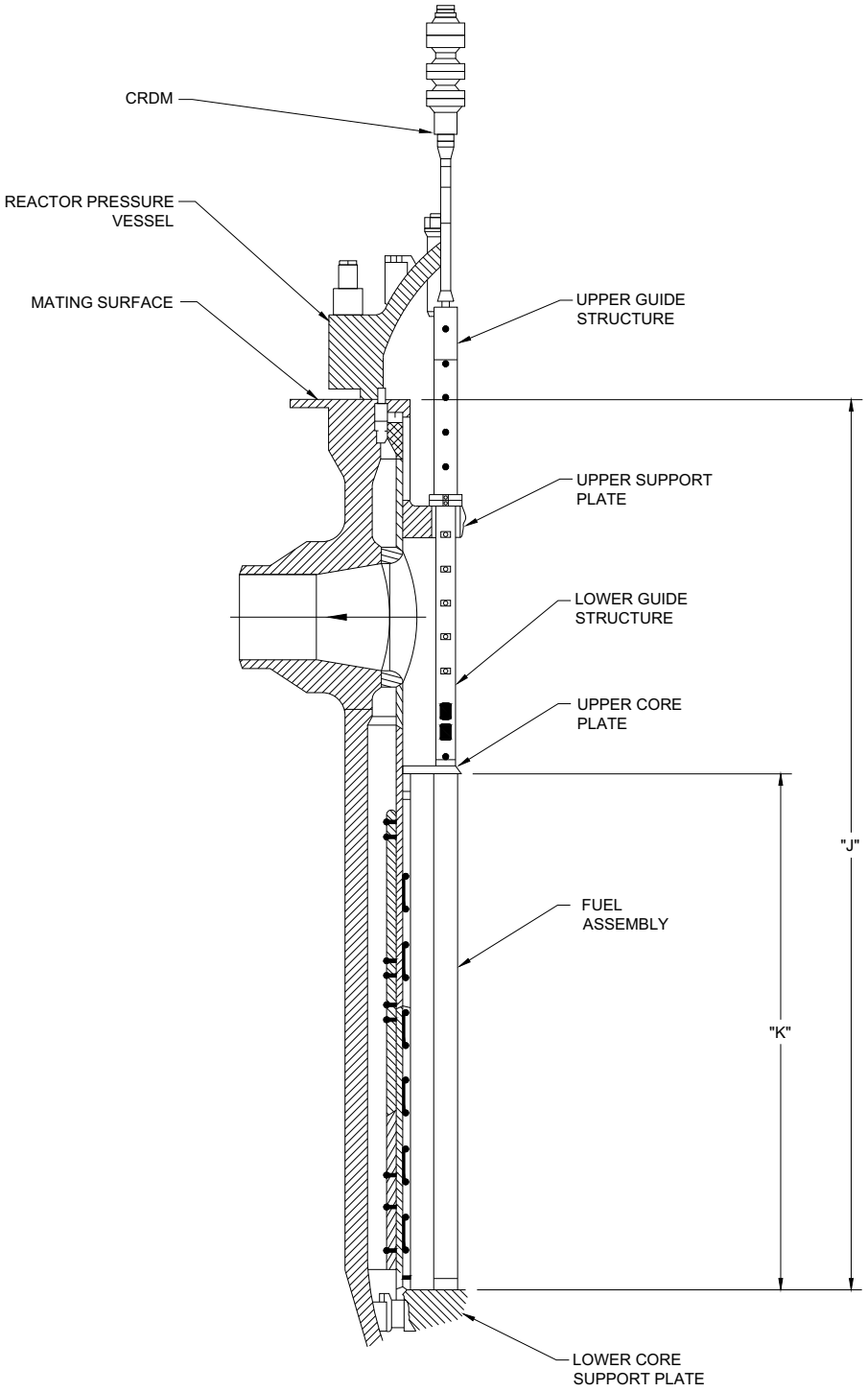

Figure 2.1.3-2
Rod Cluster Control and Drive Rod Arrangement

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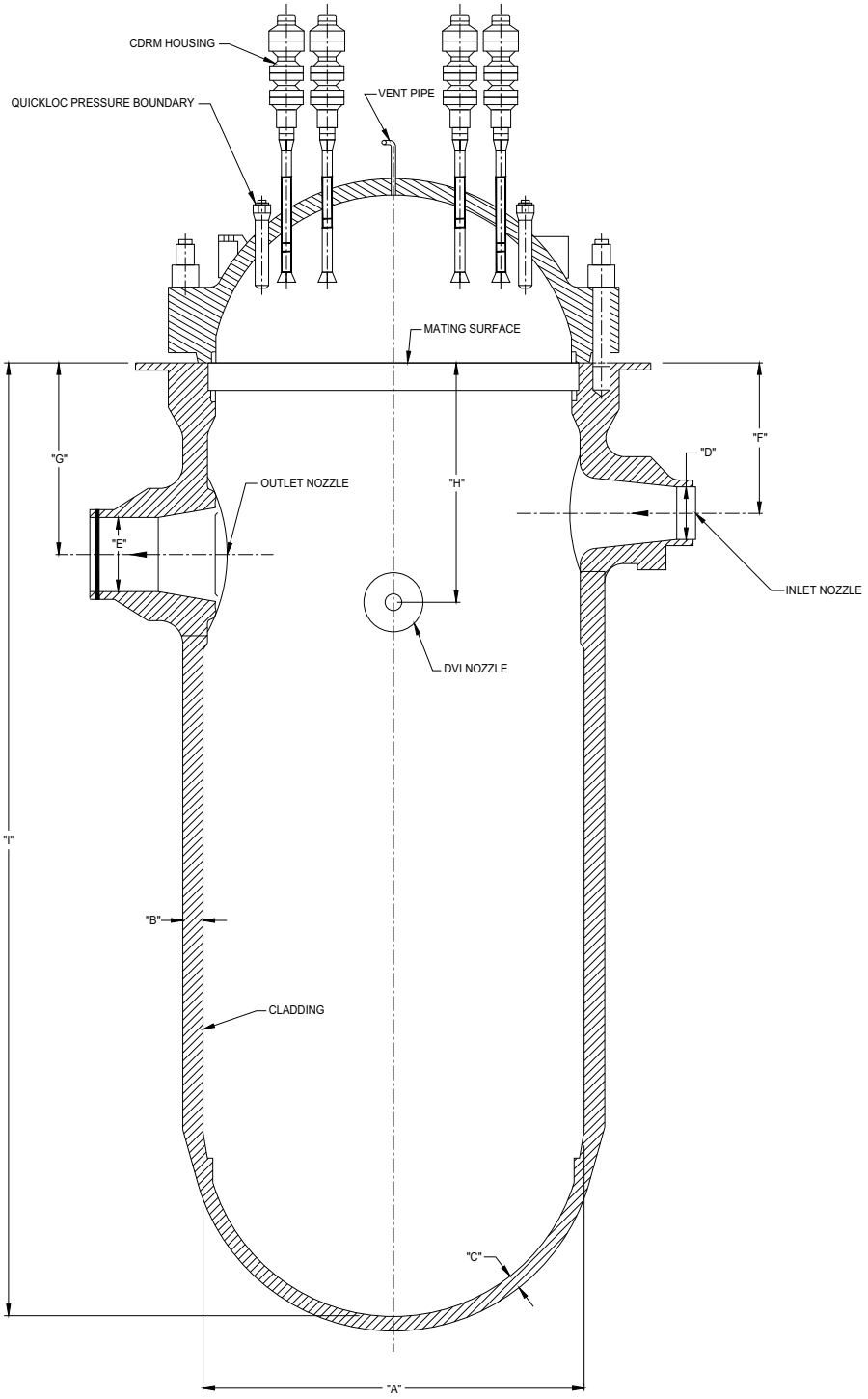

Figure 2.1.3-3
Reactor Vessel Arrangement

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2.2 Nuclear Safety Systems

2.2.1 Containment System

Design Description

The containment system (CNS) is the collection of boundaries that separates the containment atmosphere from the outside environment during design basis accidents.

The CNS is as shown in Figure 2.2.1-1 and the component locations of the CNS are as shown in Table 2.2.1-4.

1. The functional arrangement of the CNS and associated systems is as described in the Design Description of this Section 2.2.1.

2. a) The components identified in Table 2.2.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.2.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.2.1-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.2.1-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.2.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.2.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of structural integrity and safety function.

6. a) The Class 1E equipment identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.2.1-1 are powered from their respective Class 1E division.

   c) Separation is provided between CNS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

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d) The non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of containment pressure boundary integrity.

7. The CNS provides the safety-related function of containment isolation for containment boundary integrity and provides a barrier against the release of fission products to the atmosphere.

8. Containment electrical penetration assemblies are protected against currents that are greater than the continuous ratings.

9. Safety-related displays identified in Table 2.2.1-1 can be retrieved in the main control room (MCR).

10. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.2.1-1 to perform active functions.

    b) The valves identified in Table 2.2.1-1 as having protection and safety monitoring system (PMS) control perform an active function after receiving a signal from the PMS.

    c) The valves identified in Table 2.2.1-1 as having diverse actuation system (DAS) control perform an active function after receiving a signal from the DAS.

11. a) The motor-operated and check valves identified in Table 2.2.1-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.2.1-1 assume the indicated loss of motive power position.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.2.1-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the CNS.

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Table 2.2.1-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
Service Air Supply Outside Containment Isolation Valve	CAS-PL-V204	Yes	Yes	No	-/-	No	-/-	None	-
Service Air Supply Inside Containment Isolation Check Valve	CAS-PL-V205	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-
Instrument Air Supply Outside Containment Isolation Valve	CAS-PL-V014	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Instrument Air Supply Inside Containment Isolation Check Valve	CAS-PL-V015	Yes	Yes	No	-/-	-	-/-	Transfer Closed	-
Component Cooling Water System (CCS) Containment Isolation Motor-operated Valve (MOV) – Inlet Line Outside Reactor Containment (ORC)	CCS-PL-V200	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
CCS Containment Isolation Check Valve – Inlet Line Inside Reactor Containment (IRC)	CCS-PL-V201	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.2.1-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
CCS Containment Isolation MOV – Outlet Line IRC	CCS-PL-V207	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
CCS Containment Isolation MOV – Outlet Line ORC	CCS-PL-V208	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
CCS Containment Isolation Relief Valve – Outlet Line IRC	CCS-PL-V220	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Demineralized Water Supply Containment Isolation Valve ORC	DWS-PL-V244	Yes	Yes	No	-/-	No	-/-	None	-
Demineralized Water Supply Containment Isolation Check Valve IRC	DWS-PL-V245	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-
Fuel Transfer Tube	FHS-FT-01	Yes	Yes	-	-/-	-	-/-	-	-
Fuel Transfer Tube Isolation Valve	FHS-PL-V001	Yes	Yes	-	-/-	-	-/-	Transfer Closed	-
Fire Water Containment Supply Isolation Valve – Outside	FPS-PL-V050	Yes	Yes	No	-/-	No	-/-	None	-
Fire Water Containment Isolation Supply Check Valve – Inside	FPS-PL-V052	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.2.1-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
Spent Fuel Pool Cooling System (SFS) Discharge Line Containment Isolation Check Valve – IRC	SFS-PL-V037	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-
SFS Discharge Line Containment Isolation MOV – ORC	SFS-PL-V038	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
SFS Suction Line Containment Isolation MOV – IRC	SFS-PL-V034	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
SFS Suction Line Containment Isolation MOV – ORC	SFS-PL-V035	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
SFS Suction Line Containment Isolation Relief Valve – IRC	SFS-PL-V067	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Containment Purge Inlet Containment Isolation Valve – ORC	VFS-PL-V003	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Containment Purge Inlet Containment Isolation Valve – IRC	VFS-PL-V004	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

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Table 2.2.1-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
Integrated Leak Rate Testing Vent Discharge Containment Isolation Valve – ORC	VFS-PL-V008	Yes	Yes	No	-/-	No	-/-	None	-
Containment Purge Discharge Containment Isolation Valve – IRC	VFS-PL-V009	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Containment Purge Discharge Containment Isolation Valve – ORC	VFS-PL-V010	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Vacuum Relief Containment Isolation A MOV – ORC	VFS-PL-V800A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed/ Transfer Open	As Is
Vacuum Relief Containment Isolation B MOV – ORC	VFS-PL-V800B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed/ Transfer Open	As Is
Vacuum Relief Containment Isolation Check Valve A – IRC	VFS-PL-V803A	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Vacuum Relief Containment Isolation Check Valve B – IRC	VFS-PL-V803B	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-

Note: Dash (-) indicates not applicable.

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Table 2.2.1-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
Fan Coolers Return Containment Isolation Valve – IRC	VWS-PL-V082	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Fan Coolers Return Containment Isolation Valve – ORC	VWS-PL-V086	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Fan Coolers Return Containment Isolation Relief Valve – IRC	VWS-PL-V080	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Fan Coolers Supply Containment Isolation Valve – ORC	VWS-PL-V058	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Fan Coolers Supply Containment Isolation Check Valve – IRC	VWS-PL-V062	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-
Reactor Coolant Drain Tank (RCDT) Gas Outlet Containment Isolation Valve – IRC	WLS-PL-V067	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
RCDT Gas Outlet Containment Isolation Valve – ORC	WLS-PL-V068	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

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Table 2.2.1-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
Sump Discharge Containment Isolation Valve – IRC	WLS-PL-V055	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Sump Discharge Containment Isolation Valve – ORC	WLS-PL-V057	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Sump Discharge Containment Isolation Relief Valve – IRC	WLS-PL-V058	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Spare Penetration	CNS-PY-C01	Yes	Yes	-	-/-	-	-/-	-	-
Spare Penetration	CNS-PY-C02	Yes	Yes	-	-/-	-	-/-	-	-
Spare Penetration	CNS-PY-C03	Yes	Yes	-	-/-	-	-/-	-	-
Main Equipment Hatch	CNS-MY-Y01	Yes	Yes	-	-/-	-	-/-	-	-
Maintenance Hatch	CNS-MY-Y02	Yes	Yes	-	-/-	-	-/-	-	-
Upper Personnel Hatch	CNS-MY-Y03	Yes	Yes	-	-/-	-	-/-	-	-
Lower Personnel Hatch	CNS-MY-Y04	Yes	Yes	-	-/-	-	-/-	-	-
Containment Vessel	CNS-MV-01	Yes	Yes	-	-/-	-	-/-	-	-
Electrical Penetration P03	DAS-EY-P03Z	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P01	ECS-EY-P01X	Yes	Yes	-	No/Yes	-	-/-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.1-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
Electrical Penetration P02	ECS-EY-P02X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P06	ECS-EY-P06Y	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P07	ECS-EY-P07X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P09	ECS-EY-P09W	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P10	ECS-EY-P10W	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P11	IDSA-EY-P11Z	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P12	IDSA-EY-P12Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P13	IDSA-EY-P13Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P14	IDSD-EY-P14Z	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P15	IDSD-EY-P15Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P16	IDSD-EY-P16Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P17	ECS-EY-P17X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P18	ECS-EY-P18X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P19	ECS-EY-P19Z	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P20	ECS-EY-P20Z	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P21	EDS-EY-P21Z	Yes	Yes	-	No/Yes	-	-/-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.1-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
Electrical Penetration P22	ECS-EY-P22X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P23	ECS-EY-P23X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P24	ECS-EY-P24	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P25	ECS-EY-P25W	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P26	ECS-EY-P26W	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P27	IDSC-EY-P27Z	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P28	IDSC-EY-P28Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P29	IDSC-EY-P29Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P30	IDSB-EY-P30Z	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P31	IDSB-EY-P31Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P32	IDSB-EY-P32Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Instrument Penetration P46	PCS-PY-C01	Yes	Yes	-	-/-	-	-/-	-	-
Instrument Penetration P47	PCS-PY-C02	Yes	Yes	-	-/-	-	-/-	-	-
Instrument Penetration P48	PCS-PY-C03	Yes	Yes	-	-/-	-	-/-	-	-
Instrument Penetration P49	PCS-PY-C04	Yes	Yes	-	-/-	-	-/-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.1-2

Line Name	Line Number	ASME Code Section III
Instrument Air In	CAS-PL-L015	Yes
Service Air In	CAS-PL-L204	Yes
Component Cooling Water Supply to Containment	CCS-PL-L201	Yes
Component Cooling Water Outlet from Containment	CCS-PL-L207	Yes
Demineralized Water In	DWS-PL-L245, L230	Yes
Fire Protection Supply to Containment	FPS-PL-L107	Yes
Containment Atmosphere Return Line	PSS-PL-L038	Yes
Common Primary Sample Line A/B	PSS-PL-T005A/B	Yes
Containment Atmosphere Sample Line	PSS-PL-T031	Yes
Spent Fuel Pool Cooling Discharge	SFS-PL-L017	Yes
Spent Fuel Pool Cooling Suction from Containment	SFS-PL-L038	Yes
Containment Purge Inlet to Containment	VFS-PL-L104, L105, L106	Yes
Containment Purge Discharge from Containment	VFS-PL-L203, L204, L205, L800, L801A/B, L803, L804, L805A/B, L810A/B, L832	Yes
Fan Cooler Supply Line to Containment	VWS-PL-L032	Yes
Fan Cooler Return Line from Containment	VWS-PL-L055	Yes
RCDT Gas Out	WLS-PL-L022	Yes
Waste Sump Out	WLS-PL-L073	Yes

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Table 2.2.1-3
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the CNS and associated systems is as described in the Design Description of this Section 2.2.1.	Inspection of the as-built system will be performed.	The as-built CNS conforms with the functional arrangement as described in the Design Description of this Section 2.2.1.
2.a) The components identified in Table 2.2.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.2.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.2.1-1 and 2.2.1-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.2.1-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.2.1-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

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Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.a) The components identified in Table 2.2.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	i) A hydrostatic or pressure test will be performed on the components required by the ASME Code Section III to be tested.	i) A report exists and concludes that the results of the pressure test of the components identified in Table 2.2.1-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
4.b) The piping identified in Table 2.2.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic or pressure test will be performed on the piping required by the ASME Code Section III to be pressure tested.	A report exists and concludes that the results of the pressure test of the piping identified in Table 2.2.1-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
4.a) The components identified in Table 2.2.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	ii) Impact testing will be performed on the containment and pressure-retaining penetration materials in accordance with the ASME Code Section III, Subsection NE, to confirm the fracture toughness of the materials.	ii) A report exists and concludes that the containment and pressureretaining penetration materials conform with fracture toughness requirements of the ASME Code Section III.
5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of structural integrity and safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.2.1-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.2.1-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of structural integrity and safety function. iii) The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.

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Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.a) The Class 1E equipment identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.2.1-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
6.d) The non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of containment pressure boundary integrity.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on non-Class 1E electrical penetrations located in a harsh environment.	i) A report exists and concludes that the non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of containment pressure boundary integrity.
	ii) Inspection will be performed of the as-built non-Class 1E electrical penetrations located in a harsh environment.	ii) A report exists and concludes that the as-built non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
6.b) Not used.		
6.c) Not used.		

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Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The CNS provides the safety-related function of containment isolation for containment boundary integrity and provides a barrier against the release of fission products to the atmosphere.	i) A containment integrated leak rate test will be performed. ii) Testing will be performed to demonstrate that remotely operated containment isolation valves close within the required response times.	i) The leakage rate from containment for the integrated leak rate test is less than L_a. ii) The containment purge isolation valves (VFS-PL-V003, -V004, -V009, and -V010) close within 10 seconds, containment vacuum relief isolation valves (VFS-PL-V800A and -V800B) close within 30 seconds, SGS valves SGS-PL-V040A/B and SGS-PL-V057A/B are covered in Tier 1 Material, subsection 2.2.4, Table 2.2.4-4 (item 11.b.ii) and all other containment isolation valves close within 60 seconds upon receipt of an actuation signal.
8. Containment electrical penetration assemblies are protected against currents that are greater than the continuous ratings.	An analysis for the as-built containment electrical penetration assemblies will be performed to demonstrate (1) that the maximum current of the circuits does not exceed the continuous rating of the containment electrical penetration assembly, or (2) that the circuits have redundant protection devices in series and that the redundant current protection devices are coordinated with the containment electrical penetration assembly's rated short circuit thermal capacity data and prevent current from exceeding the continuous current rating of the containment electrical penetration assembly.	Analysis exists for the as-built containment electrical penetration assemblies and concludes that the penetrations are protected against currents which are greater than their continuous ratings.

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Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Safety-related displays identified in Table 2.2.1-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.1-1 can be retrieved in the MCR.
10.a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.2.1-1 to perform active functions.	Stroke testing will be performed on remotely operated valves identified in Table 2.2.1-1 using the controls in the MCR.	Controls in the MCR operate to cause remotely operated valves identified in Table 2.2.1-1 to perform active safety functions.
10.b) The valves identified in Table 2.2.1-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	Testing will be performed on remotely operated valves listed in Table 2.2.1-1 using real or simulated signals into the PMS.	The remotely operated valves identified in Table 2.2.1-1 as having PMS control perform the active function identified in the table after receiving a signal from PMS.
10.c) The valves identified in Table 2.2.1-1 as having DAS control perform an active safety function after receiving a signal from DAS.	Testing will be performed on remotely operated valves listed in Table 2.2.1-1 using real or simulated signals into the DAS.	The remotely operated valves identified in Table 2.2.1-1 as having DAS control perform the active function identified in the table after receiving a signal from DAS.
11.a) The motor-operated and check valves identified in Table 2.2.1-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed to demonstrate the capability of each valve to operate under design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.2.1-1 under design conditions.
	ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tests or type tests.	ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tests or type tests.
	iii) Tests of the motor-operated valves will be performed under preoperational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.2.1-1 under pre-operational test conditions.
	iv) Exercise testing of the check valves with active safety functions identified in Table 2.2.1-1 will be performed under preoperational test pressure, temperature and fluid flow conditions.	iv) Each check valve changes position as indicated in Table 2.2.1-1.

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Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11.b) After loss of motive power, the remotely operated valves identified in Table 2.2.1-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.1-1 assumes the indicated loss of motive power position.

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Table 2.2.1-4

Component Name	Tag. No.	Component Location
Containment Vessel	CNS-MV-01	Shield Building

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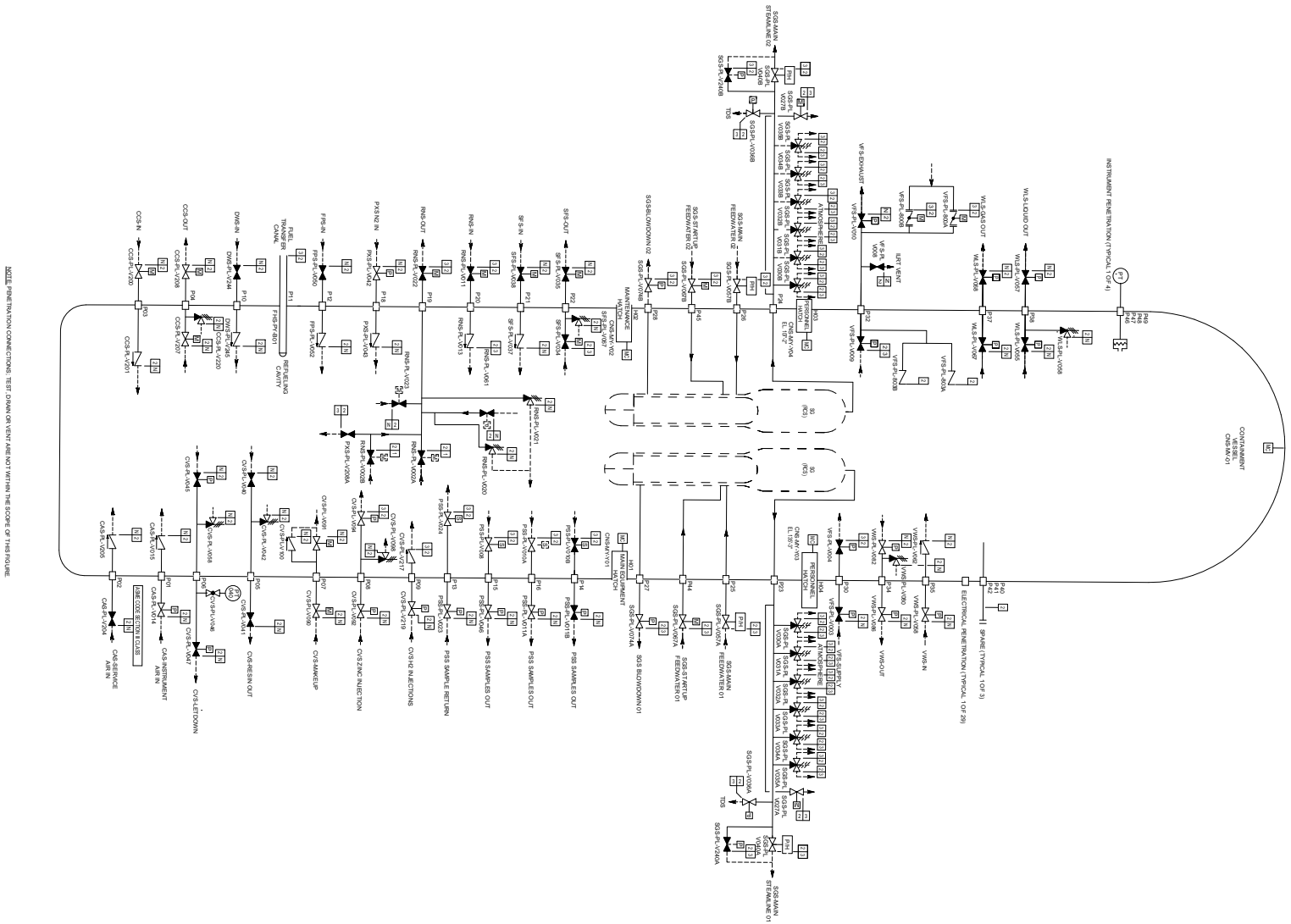

Figure 2.2.1-1
Containment System

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2.2.2 Passive Containment Cooling System

Design Description

The passive containment cooling system (PCS) removes heat from the containment during design basis events.

The PCS is as shown in Figure 2.2.2-1 and the component locations of the PCS are as shown in Table 2.2.2-4.

1. The functional arrangement of the PCS is as described in the Design Description of this Section 2.2.2.

2. a) The components identified in Table 2.2.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The pipelines identified in Table 2.2.2-2 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.2.2-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in the pipelines identified in Table 2.2.2-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.2.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The pipelines identified in Table 2.2.2-2 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

5. a) The seismic Category I components identified in Table 2.2.2-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the pipelines identified in Table 2.2.2-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

   c) The passive containment cooling ancillary water storage tank (PCCAWST) can withstand a seismic event.

6. a) The Class 1E components identified in Table 2.2.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.2.2-1 are powered from their respective Class 1E division.

   c) Separation is provided between PCS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

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7. The PCS performs the following safety-related functions:

   a) The PCS delivers water from the PCCWST to the outside, top of the containment vessel.

   b) The PCS wets the outside surface of the containment vessel. The inside and outside of the containment vessel above the operating deck are coated with an inorganic zinc coating.

   c) The PCS provides air flow over the outside of the containment vessel by a natural circulation air flow path from the air inlets to the air discharge structure.

   d) The PCS drains the excess water from the outside of the containment vessel through the two upper annulus drains.

   e) The PCS provides a flow path for long-term water makeup to the passive containment cooling water storage tank (PCCWST).

   f) The PCS provides a flow path for long-term water makeup from the PCCWST to the spent fuel pool.

8. The PCS performs the following nonsafety-related functions:

   a) The PCCAWST contains an inventory of cooling water sufficient for PCS containment cooling from hour 72 through day 7.

   b) The PCS delivers water from the PCCAWST to the PCCWST and spent fuel pool simultaneously.

   c) The PCCWST includes a water inventory for the fire protection system.

9. Safety-related displays identified in Table 2.2.2-1 can be retrieved in the main control room (MCR).

10. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.2-1 to perform active functions.

    b) The valves identified in Table 2.2.2-1 as having protection and safety monitoring system (PMS) control perform an active safety function after receiving a signal from the PMS.

    c) The valves identified in Table 2.2.2-1 as having diverse actuation system (DAS) control perform an active safety function after receiving a signal from the DAS.

11. a) The motor-operated valves identified in Table 2.2.2-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.2.2-1 assume the indicated loss of motive power position.

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Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.2.2-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the PCS.

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Table 2.2.2-1

Component Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
PCCWST	PCS-MT-01	No	Yes	-	-	-	-	-	-
Water Distribution Bucket	PCS-MT-03	No	Yes	-	-	-	-	-	-
Water Distribution Wiers	PCS-MT-04	No	Yes	-	-	-	-	-	-
PCCWST Isolation Valve	PCS-PL-V001A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Open	Open
PCCWST Isolation Valve	PCS-PL-V001B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Open	Open
PCCWST Isolation Valve MOV	PCS-PL-V001C	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
PCCWST Isolation Block MOV	PCS-PL-V002A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Open	As Is
PCCWST Isolation Block MOV	PCS-PL-V002B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Open	As Is
PCCWST Isolation Block MOV	PCS-PL-V002C	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Open	As Is
PCS Recirculation Return Isolation Valve	PCS-PL-V023	Yes	Yes	-	-/No	No	-	Transfer Close	-
PCCWST Supply to Fire Protection System Isolation Valve	PCS-PL-V005	Yes	Yes	-	-/No	No	-	Transfer Close	-

Note: Dash (-) indicates not applicable.

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Table 2.2.2-1 (cont.)

Component Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
PCS Makeup to SFS Isolation Valve	PCS-PL-V009	Yes	Yes	-	-/No	No	-	Transfer Open	-
Water Makeup Isolation Valve	PCS-PL-V044	Yes	Yes	-	-/No	No	-	Transfer Open	-
Water Bucket Makeup Line Drain Valve	PCS-PL-V015	Yes	Yes	-	-/No	No	-	Transfer Close	-
Water Bucket Makeup Line Isolation Valve	PCS-PL-V020	Yes	Yes	-	-/No	No	-	Transfer Open	-
PCCWST Long-Term Makeup Line Check Valve	PCS-PL-V039	Yes	Yes	-	-/No	No	-	Transfer Open	-
PCCWST Long-Term Makeup Drain Isolation	PCS-PL-V042	Yes	Yes	-	-/No	No	-	Transfer Close	-
PCS Discharge to SFS Pool Isolation Valve	PCS-PL-V045	Yes	Yes	-	-/No	No	-	Transfer Open	-
Recirc Header Discharge to PCCWST Isolation Valve	PCS-PL-V046	Yes	Yes	-	-/No	No	-	Transfer Close	-
PCCWST Drain Isolation Valve	PCS-PL-V049	Yes	Yes	-	-/No	No	-	Transfer Close	-
Recirc Header Discharge to SFS Pool Isolation Valve	PCS-PL-V050	Yes	Yes	-	-/No	No	-	Transfer Open/Close	-
PCCWST Discharge to SFS Pool Isolation Valve	PCS-PL-V051	Yes	Yes	-	-/No	No	-	Transfer Open/Close	-
PCS Water Delivery Flow Sensor	PCS-001	No	Yes	-	Yes/No	Yes	-	-	-

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Table 2.2.2-1 (cont.)

Component Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
PCS Water Delivery Flow Sensor	PCS-002	No	Yes	-	Yes/No	Yes	-	-	-
PCS Water Delivery Flow Sensor	PCS-003	No	Yes	-	Yes/No	Yes	-	-	-
PCS Water Delivery Flow Sensor	PCS-004	No	Yes	-	Yes/No	Yes	-	-	-
Containment Pressure Sensor	PCS-005	No	Yes	-	Yes/Yes	Yes	-	-	-
Containment Pressure Sensor	PCS-006	No	Yes	-	Yes/Yes	Yes	-	-	-
Containment Pressure Sensor	PCS-007	No	Yes	-	Yes/Yes	Yes	-	-	-
Containment Pressure Sensor	PCS-008	No	Yes	-	Yes/Yes	Yes	-	-	-
PCCWST Water Level Sensor	PCS-010	No	Yes	-	Yes/No	Yes	-	-	-
PCCWST Water Level Sensor	PCS-011	No	Yes	-	Yes/No	Yes	-	-	-
High-range Containment Pressure Sensor	PCS-012	No	Yes	-	Yes/Yes	Yes	-	-	-
High-range Containment Pressure Sensor	PCS-013	No	Yes	-	Yes/Yes	Yes	-	-	-
High-range Containment Pressure Sensor	PCS-014	No	Yes	-	Yes/Yes	Yes	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.2-2

Pipeline Name	Line Number	ASME Code Section III	Functional Capability Required
PCCWST Discharge Lines	PCS-PL-L001A/B/C/D	Yes	Yes
PCCWST Discharge Cross-connect Line	PCS-PL-L002	Yes	Yes
PCCWST Discharge Header Lines	PCS-PL-L003A/B PCS-PL-L005	Yes	Yes
Post-72-hour Supply Line Connection	PCS-PL-L051 PCS-PL-L054 PCS-PL-L065	Yes	Yes
Post-72-hour Containment Cooling Makeup From Supply Line Connections	PCS-PL-L004 PCS-PL-L007 PCS-PL-L008 PCS-PL-L023 PCS-PL-L050	Yes	Yes
Post-72-hour SFS Makeup From PCCWST	PCS-PL-L011 PCS-PL-L017 PCS-PL-L018 PCS-PL-L030* PCS-PL-L039* PCS-PL-L041 PCS-PL-L049* PCS-PL-L073	Yes	Yes
Post-72-hour SFS Makeup From Supply Line Connection	PCS-PL-L025 PCS-PL-L029 PCS-PL-L030* PCS-PL-L039* PCS-PL-L048 PCS-PL-L049* PCS-PL-L052	Yes	Yes

Note:
* Lines PCS-PL-L049, L039, and L030 comprise a common makeup line from both sources.

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Table 2.2.2-3
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the PCS is as described in the Design Description of this Section 2.2.2.	Inspection of the as-built system will be performed.	The as-built PCS conforms to the functional arrangement as described in the Design Description of this Section 2.2.2.
2.a) The components identified in Table 2.2.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The pipelines identified in Table 2.2.2-2 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.2.2-1 and 2.2.2-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.2.2-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in the pipelines identified in Table 2.2.2-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.2.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The pipelines identified in Table 2.2.2-2 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components and piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components and piping identified in Tables 2.2.2-1 and 2.2.2-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

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Table 2.2.2-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b) Each of the pipelines identified in Table 2.2.2-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report concluding that the as-built pipelines meet the requirements for functional capability.	A report exists and concludes that each of the as-built pipelines identified in Table 2.2.2-2 for which functional capability is required meets the requirements for functional capability.
5.a) The seismic Category I components identified in Table 2.2.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I components and valves identified in Table 2.2.2-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I components will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built components including anchorage are seismically bounded by the tested or analyzed conditions.	i) The seismic Category I components identified in Table 2.2.2-1 are located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I components can withstand seismic design basis loads without loss of safety function. iii) The report exists and concludes that the as-built components including anchorage are seismically bounded by the tested or analyzed conditions.
6.a) The Class 1E components identified in Table 2.2.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests or a combination of type tests and analyses will be performed on Class 1E components located in a harsh environment. ii) Inspection will be performed of the as-built Class 1E components and the associated wiring, cables, and terminations located in a harsh environment.	i) A report exists and concludes that the Class 1E components identified in Table 2.2.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function. ii) A report exists and concludes that the as-built Class 1E components and the associated wiring, cables, and terminations identified in Table 2.2.2-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.

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Table 2.2.2-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.c) The PCCAWST can withstand a seismic event.	Inspection will be performed for the existence of a report verifying that the as-built PCCAWST and its anchorage are designed using seismic Category II methods and criteria.	A report exists and concludes that the as-built PCCAWST and its anchorage are designed using seismic Category II methods and criteria.
6.b) The Class 1E components identified in Table 2.2.2-1 are powered from their respective Class 1E division.	Testing will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E components identified in Table 2.2.2-1 when the assigned Class 1E division is provided the test signal.
6.c) Not used.		
7.a) The PCS delivers water from the PCCWST to the outside, top of the containment vessel.	iii) Inspection will be performed to determine the PCCWST standpipes elevations.	iii) The elevations of the standpipes above the tank floor are: - 16.8 ft ± 0.2 ft - 20.3 ft ± 0.2 ft - 24.1 ft ± 0.2 ft
7.f) The PCS provides a flow path for long-term water makeup from the PCCWST to the spent fuel pool.	ii) Inspection of the PCCWST will be performed.	ii) The volume of the PCCWST is greater than 756,700 gallons.
8.a) The PCCAWST contains an inventory of cooling water sufficient for PCS containment cooling from hour 72 through day 7.	Inspection of the PCCAWST will be performed.	The volume of the PCCAWST is greater than 780,000 gallons.

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Table 2.2.2-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) The PCS delivers water from the PCCWST to the outside, top of the containment vessel.	i) Testing will be performed to measure the PCCWST delivery rate from each one of the three parallel flow paths.	i) When tested, each one of the three flow paths delivers water at greater than or equal to: - 469.1 gpm at a PCCWST water level of 27.4 ft + 0.2, - 0.0 ft above the tank floor - 226.6 gpm when the PCCWST water level uncovers the first (i.e. tallest) standpipe - 176.3 gpm when the PCCWST water level uncovers the second tallest standpipe - 144.2 gpm when the PCCWST water level uncovers the third tallest standpipe - or a report exists and concludes that the as-measured flow rates delivered by the PCCWST to the containment vessel provides sufficient heat removal capability such that the limiting containment pressure and temperature values are not affected and the PCS is able to perform its safety function to remove heat from containment to maintain plant safety.
	ii) Testing and or analysis will be performed to demonstrate the PCCWST inventory provides 72 hours of adequate water flow.	ii) When tested and/or analyzed with all flow paths delivering and an initial water level at 27.4 + 0.2, - 0.00 ft, the PCCWST water inventory provides greater than or equal to 72 hours of flow, and the flow rate at 72 hours is greater than or equal to 100.7 gpm or a report exists and concludes that the as-measured flow rates delivered by the PCCWST to the containment vessel provides sufficient heat removal capability such that the limiting containment pressure and temperature values are not affected and the PCS is able to perform its safety function to remove heat from containment to maintain plant safety.

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Table 2.2.2-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.b) The PCS wets the outside surface of the containment vessel. The inside and the outside of the containment vessel above the operating deck are coated with an inorganic zinc material.	i) Testing will be performed to measure the outside wetted surface of the containment vessel with one of the three parallel flow paths delivering water to the top of the containment vessel.	i) A report exists and concludes that when the water in the PCCWST uncovers the standpipes at the following levels, the water delivered by one of the three parallel flow paths to the containment shell provides coverage measured at any elevation between elevation 266 ft. and the spring line that is equal to or greater than the stated coverages. - 24.1 ± 0.2 ft above the tank floor; at least 90% of the perimeter is wetted. - 20.3 ± 0.2 ft above the tank floor; at least 72.9% of the perimeter is wetted. - 16.8 ± 0.2 ft above the tank floor; at least 59.6% of the perimeter is wetted.
	ii) Inspection of the containment vessel exterior coating will be conducted.	ii) A report exists and concludes that the containment vessel exterior surface is coated with an inorganic zinc coating above elevation 135'-3".
	iii) Inspection of the containment vessel interior coating will be conducted.	iii) A report exists and concludes that the containment vessel interior surface is coated with an inorganic zinc coating above the operating deck.
7.c) The PCS provides air flow over the outside of the containment vessel by a natural circulation air flow path from the air inlets to the air discharge structure.	Inspections of the air flow path segments will be performed.	Flow paths exist at each of the following locations: - Air inlets - Base of the outer annulus - Base of the inner annulus - Discharge structure
7.d) The PCS drains the excess water from the outside of the containment vessel through the two upper annulus drains.	Testing will be performed to verify the upper annulus drain flow performance.	With a water level within the upper annulus 10" $\pm$ 1" above the annulus drain inlet, the flow rate through each drain is greater than or equal to 525 gpm.

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Table 2.2.2-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.e) The PCS provides a flow path for long-term water makeup to the PCCWST.	ii) Testing will be performed to measure the delivery rate from the long-term makeup connection to the PCCWST.	ii) With a water supply connected to the PCS long-term makeup connection, each PCS recirculation pump delivers greater than or equal to 100 gpm when tested separately.
9. Safety-related displays identified in Table 2.2.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.2-1 can be retrieved in the MCR.
10.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.2-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.2.2-1 using the controls in the MCR.	Controls in the MCR operate to cause remotely operated valves identified in Table 2.2.2-1 to perform active functions.
10.b) The valves identified in Table 2.2.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	Testing will be performed on the remotely operated valves in Table 2.2.2-1 using real or simulated signals into the PMS.	The remotely operated valves identified in Table 2.2.2-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
11.a) The motor-operated valves identified in Table 2.2.2-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under preoperational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.2.2-1 under preoperational test conditions.
11.b) After loss of motive power, the remotely operated valves identified in Table 2.2.2-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.2-1 assumes the indicated loss of motive power position.
7.e) The PCS provides a flow path for long-term water makeup to the PCCWST.	i) Not used.	

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Table 2.2.2-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.f) The PCS provides a flow path for long-term water makeup from the PCCWST to the spent fuel pool.	i) Testing will be performed to measure the delivery rate from the PCCWST to the spent fuel pool.	i) With the PCCWST water level at 27.4 ft + 0.2, - 0.0 ft above the bottom of the tank, the flow path from the PCCWST to the spent fuel pool delivers greater than or equal to 118 gpm.
8.b) The PCS delivers water from the PCCAWST to the PCCWST and spent fuel pool simultaneously.	Testing will be performed to measure the delivery rate from the PCCAWST to the PCCWST and spent fuel pool simultaneously.	With PCCAWST aligned to the suction of the recirculation pumps, each pump delivers greater than or equal to 100 gpm to the PCCWST and 35 gpm to the spent fuel pool simultaneously when each pump is tested separately.
8.c) Not used.		
10.c) The valves identified in Table 2.2.2-1 as having DAS control perform an active safety function after receiving a signal from the DAS.	Testing will be performed on the remotely operated valves listed in Table 2.2.2-1 using real or simulated signals into the DAS.	The remotely operated valves identified in Table 2.2.2-1 as having DAS control perform the active function identified in the table after receiving a signal from the DAS.
11.a) The motor-operated valves identified in Table 2.2.2-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed to demonstrate the capability of the valve to operate under its design conditions. ii) Inspection will be performed for the existence of a report verifying that the capability of the as-built motor-operated valves bound the tested conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.2.2-1 under design conditions. ii) A report exists and concludes that the capability of the as-built motor-operated valves bound the tested conditions.

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Table 2.2.2-4

Component Name	Tag No.	Component Location
PCCWST	PCS-MT-01	Shield Building
PCCAWST	PCS-MT-05	Yard
Recirculation Pump A	PCS-MP-01A	Auxiliary Building
Recirculation Pump B	PCS-MP-01B	Auxiliary Building

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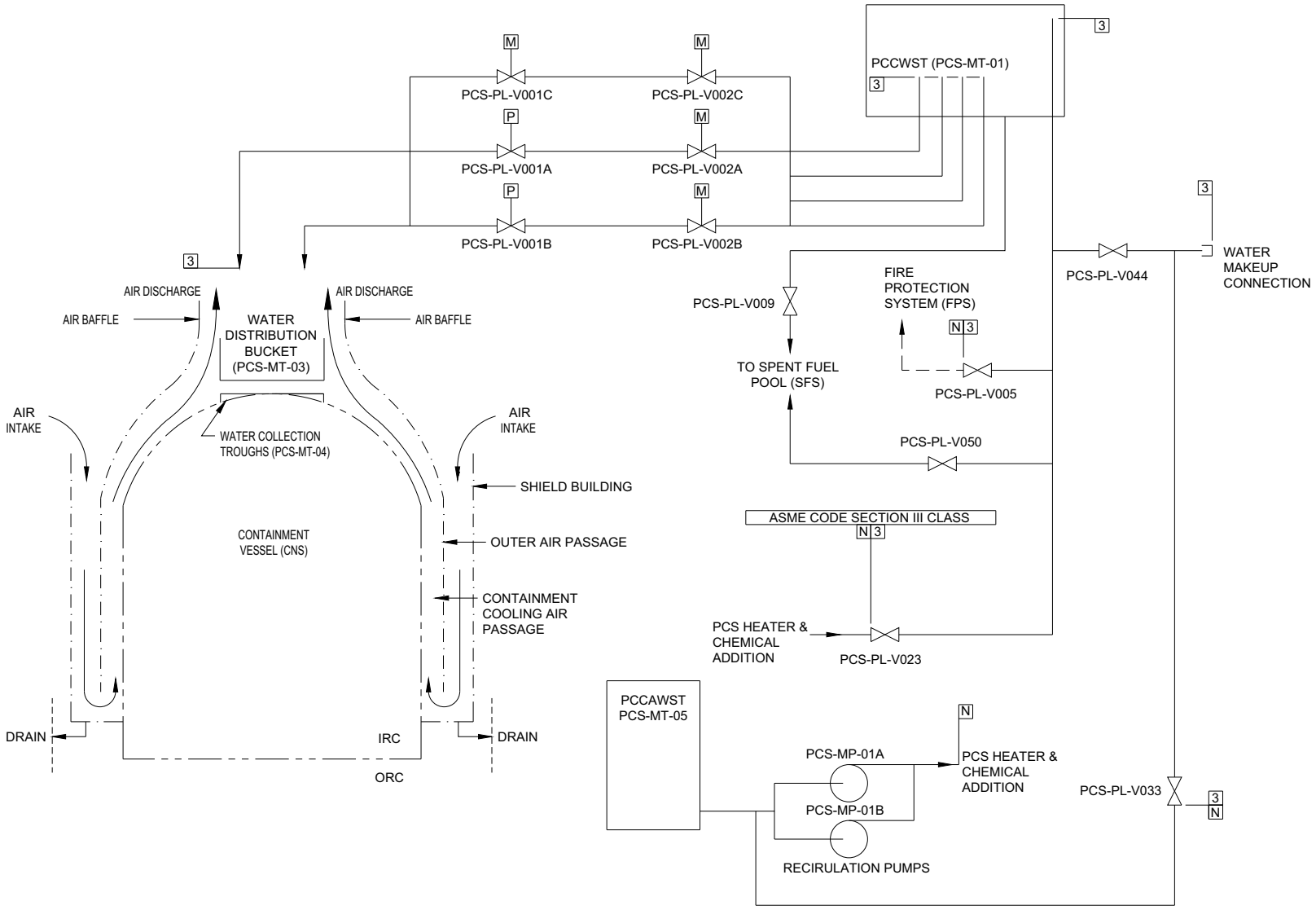

Figure 2.2.2-1
Passive Containment Cooling System

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2.2.3 Passive Core Cooling System

Design Description

The passive core cooling system (PXS) provides emergency core cooling during design basis events.

The PXS is as shown in Figure 2.2.3-1 and the component locations of the PXS are as shown in Table 2.2.3-5.

1. The functional arrangement of the PXS is as described in the Design Description of this Section 2.2.3.

2. a) The components identified in Table 2.2.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.2.3-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.2.3-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.2.3-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.2.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.2.3-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. a) The seismic Category I equipment identified in Table 2.2.3-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.2.3-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

6. Each of the as-built lines identified in Table 2.2.3-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

7. a) The Class 1E equipment identified in Table 2.2.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.2.3-1 are powered from their respective Class 1E division.

   c) Separation is provided between PXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

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8. The PXS provides the following safety-related functions:

   a) The PXS provides containment isolation of the PXS lines penetrating the containment.

   b) The PRHR HX provides core decay heat removal during design basis events.

   c) The CMTs, accumulators, in-containment refueling water storage tank (IRWST) and containment recirculation provide reactor coolant system (RCS) makeup, boration, and safety injection during design basis events.

   d) The PXS provides pH adjustment of water flooding the containment following design basis accidents.

9. The PXS has the following features:

   a) The PXS provides a function to cool the outside of the reactor vessel during a severe accident.

   b) The accumulator discharge check valves (PXS-PL-V028A/B and V029A/B) are of a different check valve type than the CMT discharge check valves (PXS-PL-V016A/B and V017A/B).

10. Safety-related displays of the parameters identified in Table 2.2.3-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.3-1 to perform their active function(s).

    b) The valves identified in Table 2.2.3-1 as having protection and safety monitoring system (PMS) control perform their active function after receiving a signal from the PMS.

    c) The valves identified in Table 2.2.3-1 as having diverse actuation system (DAS) control perform their active function after receiving a signal from the DAS.

12. a) The squib valves and check valves identified in Table 2.2.3-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.2.3-1 assume the indicated loss of motive power position.

13. Displays of the parameters identified in Table 2.2.3-3 can be retrieved in the MCR.

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Inspection, Tests, Analyses, and Acceptance Criteria

Table 2.2.3-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the PXS.

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Table 2.2.3-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Passive Residual Heat Removal Heat Exchanger (PRHR HX)	PXS-ME-01	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank A	PXS-MT-01A	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank B	PXS-MT-01B	Yes	Yes	-	- / -	-	- / -	-	-
Core Makeup Tank (CMT) A	PXS-MT-02A	Yes	Yes	-	- / -	-	- / -	-	-
CMT B	PXS-MT-02B	Yes	Yes	-	- / -	-	- / -	-	-
IRWST	PXS-MT-03	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen A	PXS-MY-Y01A	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen B	PXS-MY-Y01B	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen C	PXS-MY-Y01C	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen A	PXS-MY-Y02A	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen B	PXS-MY-Y02B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3A	PXS-MY-Y03A	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3B	PXS-MY-Y03B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 4A	PXS-MY-Y04A	No	Yes		- / -		- / -		
pH Adjustment Basket 4B	PXS-MY-Y04B	No	Yes		- / -		- / -		
Downspout Screen 1A	PXS-MY-Y81	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1B	PXS-MY-Y82	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1C	PXS-MY-Y83	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1D	PXS-MY-Y84	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2A	PXS-MY-Y85	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2B	PXS-MY-Y86	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2C	PXS-MY-Y87	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2D	PXS-MY-Y88	No	Yes	-	- / -	-	- / -	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
CMT A Inlet Isolation Motor-operated Valve	PXS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT B Inlet Isolation Motor-operated Valve	PXS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT A Discharge Isolation Valve	PXS-PL-V014A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V014B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Isolation Valve	PXS-PL-V015A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V015B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Check Valve	PXS-PL-V016A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT B Discharge Check Valve	PXS-PL-V016B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT A Discharge Check Valve	PXS-PL-V017A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT B Discharge Check Valve	PXS-PL-V017B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Accumulator A Pressure Relief Valve	PXS-PL-V022A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Accumulator B Pressure Relief Valve	PXS-PL-V022B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Accumulator A Discharge Isolation Valve	PXS-PL-V027A	Yes	Yes	Yes	- / -	No	- /No	None	As Is
Accumulator B Discharge Isolation Valve	PXS-PL-V027B	Yes	Yes	Yes	- / -	No	- /No	None	As Is
Accumulator A Discharge Check Valve	PXS-PL-V028A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator B Discharge Check Valve	PXS-PL-V028B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator A Discharge Check Valve	PXS-PL-V029A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator B Discharge Check Valve	PXS-PL-V029B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Nitrogen Supply Containment Isolation Valve	PXS-PL-V042	Yes	Yes	Yes	Yes/No	Yes (position)	Yes/No	Transfer Closed	Close
Nitrogen Supply Containment Isolation Check Valve	PXS-PL-V043	Yes	Yes	No	- / -	No	- / -	Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
PRHR HX Inlet Isolation Motor-operated Valve	PXS-PL-V101	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/No	None	As Is
PRHR HX Control Valve	PXS-PL-V108A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
PRHR HX Control Valve	PXS-PL-V108B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
Containment Recirculation A Isolation Motor-operated Valve	PXS-PL-V117A	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/No	None	As Is
Containment Recirculation B Isolation Motor-operated Valve	PXS-PL-V117B	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/No	None	As Is
Containment Recirculation A Squib Valve	PXS-PL-V118A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
Containment Recirculation B Squib Valve	PXS-PL-V118B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
Containment Recirculation A Check Valve	PXS-PL-V119A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Containment Recirculation B Check Valve	PXS-PL-V119B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Containment Recirculation A Squib Valve	PXS-PL-V120A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
Containment Recirculation B Squib Valve	PXS-PL-V120B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is

Note: Dash (-) indicates not applicable.

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Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
IRWST Injection A Check Valve	PXS-PL-V122A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection B Check Valve	PXS-PL-V122B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection A Squib Valve	PXS-PL-V123A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection B Squib Valve	PXS-PL-V123B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection A Check Valve	PXS-PL-V124A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection B Check Valve	PXS-PL-V124B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection A Squib Valve	PXS-PL-V125A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection B Squib Valve	PXS-PL-V125B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Gutter Isolation Valve	PXS-PL-V130A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Closed	Closed
IRWST Gutter Isolation Valve	PXS-PL-V130B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

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Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
CMT A Level Sensor	PXS-011A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011B	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011C	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011D	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-012A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-014A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014B	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014C	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014D	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Wide Range Level Sensor	PXS-046	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Wide Range Level Sensor	PXS-047	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Wide Range Level Sensor	PXS-048	-	Yes	-	Yes/Yes	Yes	- / -	-	-
PRHR HX Flow Sensor	PXS-049A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
PRHR HX Flow Sensor	PXS-049B	-	Yes	-	Yes/Yes	Yes	- / -	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.3-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Containment Flood-up Level Sensor	PXS-050	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Containment Flood-up Level Sensor	PXS-051	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Containment Flood-up Level Sensor	PXS-052	-	Yes	-	Yes/Yes	Yes	-/-	-	-
IRWST Lower Narrow Range Level Sensor	PXS-066	-	Yes	-	Yes/Yes	Yes	-/-	-	-
IRWST Lower Narrow Range Level Sensor	PXS-067	-	Yes	-	Yes/Yes	Yes	-/-	-	-
IRWST Lower Narrow Range Level Sensor	PXS-068	-	Yes	-	Yes/Yes	Yes	-/-	-	-
IRWST Lower Narrow Range Level Sensor	PXS-069	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RNS Suction Leak Test Valve	PXS-PL-V208A	Yes	Yes	No	- / -	No	-/-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.3-2

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
PRHR HX inlet line from hot leg and outlet line to steam generator channel head	RCS-L134, PXS-L102, PXS-L103, PXS-L104A, PXS-L104B, PXS-L105, RCS-L113	Yes	Yes	Yes
	PXS-L107	Yes	Yes	No
CMT A inlet line from cold leg C and outlet line to reactor vessel direct vessel injection (DVI) nozzle A	RCS-L118A, PXS-L007A, PXS-L015A, PXS-L016A, PXS-L017A, PXS-L018A, PXS-L020A, PXS-L021A	Yes	Yes	Yes
	PXS-L070A	Yes	Yes	No
CMT B inlet line from cold leg D and outlet line to reactor vessel DVI nozzle B	RCS-L118B, PXS-L007B, PXS-L015B, PXS-L016B, PXS-L017B, PXS-L018B, PXS-L020B, PXS-L021B	Yes	Yes	Yes
	PXS-L070B	Yes	Yes	No
RNS A discharge line to PXS from RNS check valve RNS-PL-V017A to DVI line A	PXS-L019A	Yes	Yes	Yes
RNS B discharge line to PXS from RNS check valve RNS-PL-V017B to DVI line B	PXS-L019B	Yes	Yes	Yes
Accumulator A discharge line to DVI line A	PXS-L025A, PXS-L027A, PXS-L029A	Yes	Yes	Yes
Accumulator B discharge line to DVI line B	PXS-L025B, PXS-L027B, PXS-L029B	Yes	Yes	Yes

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Table 2.2.3-2 (cont.)

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
IRWST injection line A to DVI line A	PXS-L123A, PXS-L125A, PXS-L127A	Yes	Yes	Yes
	PXS-L124A, PXS-L118A, PXS-L117A, PXS-L116A, PXS-L112A	Yes	No	Yes
	PXS-L133A, PXS-L134A	Yes	Yes	No
IRWST injection line B to DVI line B	PXS-L123B, PXS-L125B, PXS-L127B	Yes	Yes	Yes
	PXS-L124B, PXS-L118B, PXS-L117B, PXS-L116B, PXS-L114, PXS-L112B, PXS-L120	Yes	No	Yes
	PXS-L133B, PXS-L134B	Yes	Yes	No
IRWST screen cross-connect line	PXS-L180A, PXS-L180B	Yes	No	Yes
Containment recirculation line A	PXS-L113A, PXS-L131A, PXS-L132A	Yes	No	Yes
Containment recirculation line B	PXS-L100, PXS-L101, PXS-L106 PXS-L113B, PXS-L131B, PXS-L132B	Yes	No	Yes
IRWST gutter drain line	PXS-L142A, PXS-L142B, PXS-L183A, PXS-L183B, PXS-L184A, PXS-L184B, PXS-L185A, PXS-L185B	Yes	No	Yes
	PXS-L141A, PXS-L141B	Yes	No	No
Downspout drain lines from polar crane girder and internal stiffener to collection box A	PXS-L301A, PXS-L302A, PXS-L303A, PXS-L304A, PXS-L305A, PXS-L306A, PXS-L307A, PXS-L308A, PXS-L309A, PXS-L310A	Yes	No	Yes
Downspout drain lines from polar crane girder and internal stiffener to collection box B	PXS-L301B, PXS-L302B, PXS-L303B, PXS-L304B, PXS-L305B, PXS-L306B, PXS-L307B, PXS-L308B, PXS-L309B, PXS-L310B	Yes	No	Yes

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Table 2.2.3-3

Equipment	Tag No.	Display	Control Function
CMT A Discharge Isolation Valve (Position)	PXS-PL-V014A	Yes (Position)	-
CMT B Discharge Isolation Valve (Position)	PXS-PL-V014B	Yes (Position)	-
CMT A Discharge Isolation Valve (Position)	PXS-PL-V015A	Yes (Position)	-
CMT B Discharge Isolation Valve (Position)	PXS-PL-V015B	Yes (Position)	-
Accumulator A Nitrogen Vent Valve (Position)	PXS-PL-V021A	Yes (Position)	-
Accumulator B Nitrogen Vent Valve (Position)	PXS-PL-V021B	Yes (Position)	-
Accumulator A Discharge Isolation Valve (Position)	PXS-PL-V027A	Yes (Position)	-
Accumulator B Discharge Isolation Valve (Position)	PXS-PL-V027B	Yes (Position)	-
PRHR HX Control Valve (Position)	PXS-PL-V108A	Yes (Position)	-
PRHR HX Control Valve (Position)	PXS-PL-V108B	Yes (Position)	-
Containment Recirculation A Isolation Valve	PXS-PL-V117A	Yes (Position)	-
Containment Recirculation B Isolation Valve	PXS-PL-V117B	Yes (Position)	-
Containment Recirculation A Isolation Valve (Position)	PXS-PL-V118A	Yes (Position)	-
Containment Recirculation B Isolation Valve (Position)	PXS-PL-V118B	Yes (Position)	-
Containment Recirculation A Isolation Valve (Position)	PXS-PL-V120A	Yes (Position)	-
Containment Recirculation B Isolation Valve (Position)	PXS-PL-V120B	Yes (Position)	-
IRWST Line A Isolation Valve (Position)	PXS-PL-V121A	Yes (Position)	-
IRWST Line B Isolation Valve (Position)	PXS-PL-V121B	Yes (Position)	-
IRWST Injection A Isolation Squib (Position)	PXS-PL-V123A	Yes (Position)	-

Note: Dash (-) indicates not applicable.

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Table 2.2.3-3 (cont.)

Equipment	Tag No.	Display	Control Function
IRWST Injection B Isolation Squib (Position)	PXS-PL-V123B	Yes (Position)	-
IRWST Injection A Isolation Squib (Position)	PXS-PL-V125A	Yes (Position)	-
IRWST Injection B Isolation Squib (Position)	PXS-PL-V125B	Yes (Position)	-
IRWST Gutter Bypass Isolation Valve (Position)	PXS-PL-V130A	Yes (Position)	-
IRWST Gutter Bypass Isolation Valve (Position)	PXS-PL-V130B	Yes (Position)	-
Accumulator A Level Sensor	PXS-JE-L021	Yes	-
Accumulator B Level Sensor	PXS-JE-L022	Yes	-
Accumulator A Level Sensor	PXS-JE-L023	Yes	-
Accumulator B Level Sensor	PXS-JE-L024	Yes	-
PRHR HX Inlet Temperature Sensor	PXS-JE-T064	Yes	-
IRWST Surface Temperature Sensor	PXS-JE-T041	Yes	-
IRWST Surface Temperature Sensor	PXS-JE-T042	Yes	-
IRWST Bottom Temperature Sensor	PXS-JE-T043	Yes	-
IRWST Bottom Temperature Sensor	PXS-JE-T044	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.2.3-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the PXS is as described in the Design Description of this Section 2.2.3.	Inspection of the as-built system will be performed.	The as-built PXS conforms with the functional arrangement as described in the Design Description of this Section 2.2.3.
2.a) The components identified in Table 2.2.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.2.3-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.2.3-1 and 2.2.3-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.2.3-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.2.3-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.2.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The piping identified in Table 2.2.3-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the components and piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components and piping identified in Tables 2.2.3-1 and 2.2.3-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b) Each of the lines identified in Table 2.2.3-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.2.3-2 for which functional capability is required meets the requirements for functional capability.
6. Each of the as-built lines identified in Table 2.2.3-2 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	Inspection will be performed for the existence of an LBB evaluation report or an evaluation report on the protection from dynamic effects of a pipe break. Tier 1 Material, Section 3.3, Nuclear Island Buildings, contains the design descriptions and inspections, tests, analyses, and acceptance criteria for protection from the dynamic effects of pipe rupture.	An LBB evaluation report exists and concludes that the LBB acceptance criteria are met by the as-built PXS piping and piping materials, or a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.a) The seismic Category I equipment identified in Table 2.2.3-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.2.3-1 are located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.2.3-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of safety function. For the PXS containment recirculation and IRWST screens, a report exists and concludes that the screens can withstand seismic dynamic loads and also post-accident operating loads, including head loss and debris weights.
	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions. For the PXS containment recirculation and IRWST screens, a report exists and concludes that the as-built screens including their anchorage are bounded by the seismic loads and also post-accident operating loads, including head loss and debris weights.
7.a) The Class 1E equipment identified in Table 2.2.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.2.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.2.3-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.b) Not used.		
7.c) Not used.		
8.a) Not used.		
8.b) The PXS provides core decay heat removal during design basis events.	A heat removal performance test and analysis of the PRHR HX will be performed to determine the heat transfer from the HX. For the test, the reactor coolant hot leg temperature will be initially at ≥ 350°F with the reactor coolant pumps running. The IRWST water level for the test will be above the top of the HX. The test will continue until the hot leg temperature is ≤ 250°F. Inspection of the elevation of the PRHR HX will be conducted.	A report exists and concludes that the PRHR HX heat transfer rate with the design basis number of PRHR HX tubes plugged is: ≥ 8.46×10^7 Btu/hr with 250°F HL Temp and an initial IRWST temperature of 80°F. The heat transfer rate measured in the test should be adjusted to account for differences in the HL and IRWST temperatures and the number of tubes plugged. The elevation of the centerline of the HX's upper channel head is greater than the HL centerline by at least 26.3 ft.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	i) A low-pressure injection test and analysis for each CMT, each accumulator, each IRWST injection line, and each containment recirculation line will be conducted. Each test is initiated by opening isolation valve(s) in the line being tested. Test fixtures may be used to simulate squib valves.	i) The injection line flow resistance from each source is as follows:
	CMTs: Each CMT will be initially filled with water. All valves in these lines will be open during the test.	CMTs: The calculated flow resistance between each CMT and the reactor vessel is $\geq 1.81 \times 10^{-5}$ ft/gpm^2 and $\leq 2.25 \times 10^{-5}$ ft/gpm^2.
	Accumulators: Each accumulator will be partially filled with water and pressurized with nitrogen. All valves in these lines will be open during the test. Sufficient flow will be provided to fully open the check valves.	Accumulators: The calculated flow resistance between each accumulator and the reactor vessel is $\geq 1.47 \times 10^{-5}$ ft/gpm^2 and $\leq 1.83 \times 10^{-5}$ ft/gpm^2.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	IRWST Injection: The IRWST will be partially filled with water. All valves in these lines will be open during the test. Sufficient flow will be provided to open the check valves.	IRWST Injection: The calculated flow resistance for each IRWST injection line between the IRWST and the reactor vessel is: Line A: $\geq 5.35 \times 10^{-6}$ ft/gpm^2 and $\leq 9.09 \times 10^{-6}$ ft/gpm^2 and Line B: $\geq 6.15 \times 10^{-6}$ ft/gpm^2 and $\leq 1.05 \times 10^{-5}$ ft/gpm^2.
	Containment Recirculation: A temporary water supply will be connected to the recirculation lines. All valves in these lines will be open during the test. Sufficient flow will be provided to open the check valves.	Containment Recirculation: The calculated flow resistance for each containment recirculation line between the containment and the reactor vessel is: Line A: $\leq 1.33 \times 10^{-5}$ ft/gpm^2 and Line B: $\leq 1.21 \times 10^{-5}$ ft/gpm^2.
	ii) A low-pressure test and analysis will be conducted for each CMT to determine piping flow resistance from the cold leg to the CMT. The test will be performed by filling the CMT via the cold leg balance line by operating the normal residual heat removal pumps.	ii) The flow resistance from the cold leg to the CMT is $\leq 7.21 \times 10^{-6}$ ft/gpm^2.
	iii) Inspections of the routing of the following pipe lines will be conducted: - CMT inlet line, cold leg to high point - PRHR HX inlet line, hot leg to high point	iii) These lines have no downward sloping sections between the connection to the RCS and the high point of the line.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	iv) Inspections of the elevation of the following pipe lines will be conducted: 1. IRWST injection lines; IRWST connection to DVI nozzles	iv) The maximum elevation of the top inside surface of these lines is less than the elevation of: 1. IRWST bottom inside surface
	v) Inspections of the elevation of the following tanks will be conducted: 2. IRWST	v) The elevation of the bottom inside tank surface is higher than the direct vessel injection nozzle centerline by the following: 2. IRWST ≥ 3.4 ft
	iv) Inspections of the elevation of the following pipe lines will be conducted: - Containment recirculation lines; containment to IRWST lines - CMT discharge lines to DVI connection - PRHR HX outlet line to SG connection	iv) The maximum elevation of the top inside surface of these lines is less than the elevation of: - IRWST bottom inside surface - CMT bottom inside surface - PRHR HX lower channel head top inside surface
	v) Inspections of the elevation of the following tanks will be conducted: - CMTs	v) The elevation of the bottom inside tank surface is higher than the direct vessel injection nozzle centerline by the following: - CMTs ≥ 7.5 ft
	vi) Inspections of each of the following tanks will be conducted: - CMTs - Accumulators - IRWST	vi) The calculated volume of each of the following tanks is as follows: - CMTs ≥ 2487 ft^3 - Accumulators ≥ 2000 ft^3 - IRWST ≥ 73,100 ft^3 between the tank outlet connection and the tank overflow

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	vii) Inspection of the as-built components will be conducted for the plate located above the containment recirculation screens.	vii) The plate located above the containment recirculation screens is no more than 1 ft, 3 in above the top of the face of the screens and extends at least 8 ft, 3 in perpendicular to the front and at least 7 ft to the side of the face of the screens.
	viii) Inspections of the IRWST and containment recirculation screens will be conducted. The inspections will include measurements of the pockets and the number of pockets used in each screen. The pocket frontal face area is based on a width times a height. The width is the distance between pocket centerlines for pockets located beside each other. The height is the distance between pocket centerlines for pockets located above each other. The pocket screen area is the total area of perforated plate inside each pocket; this area will be determined by inspection of the screen manufacturing drawings.	viii) The screens utilize pockets with a frontal face area of ≥ 6.2 in^2 and a screen surface area ≥ 140 in^2 per pocket. IRWST Screens A and B each have a sufficient number of pockets to provide a frontal face area ≥ 25 ft^2, a screen surface area ≥ 575 ft^2, and a screen mesh size of ≤ 0.0625 inch. IRWST Screen C has a sufficient number of pockets to provide a frontal face area ≥ 50 ft^2, a screen surface area ≥ 1150 ft^2, and a screen mesh size ≤ 0.0625 inch. Each containment recirculation screen has a sufficient number of pockets to provide a frontal face area ≥ 105 ft^2, a screen surface area ≥ 2500 ft^2, and a screen mesh size ≤ 0.0625 inch. A debris curb exists in front of the containment recirculation screens which is ≥ 2 ft above the loop compartment floor. The bottoms of the IRWST screens are located ≥ 6 in above the bottom of the IRWST.
	ix) Inspections will be conducted of the insulation used inside the containment on the ASME Class 1 lines, reactor vessel, reactor coolant pumps, pressurizer and steam generators.	ix) The type of insulation used on these lines and equipment is a metal reflective type or a suitable equivalent. If an insulation other than metal reflective insulation is used, a report must exist and conclude that the insulation is a suitable equivalent.
	Inspections will be conducted of other insulation used inside the containment within the zone of influence (ZOI).	The type of insulation used on these lines and equipment is a metal reflective type or a suitable equivalent. If an insulation other than metal reflective insulation is used, a report must exist and conclude that the insulation is a suitable equivalent.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	Inspection will be conducted of other insulation below the maximum flood level of a design basis loss-of-coolant accident (LOCA).	The type of insulation used on these lines is metal reflective insulation, jacketed fiberglass, or a suitable equivalent. If an insulation other than metal reflective or jacketed fiberglass insulation is used, a report must exist and conclude that the insulation is a suitable equivalent.
	x) Inspections will be conducted of the as-built nonsafety-related coatings or of plant records of the nonsafety-related coatings used inside containment on walls, floors, ceilings, and structural steel except in the CVS room. Inspections will be conducted of the as-built non-safety-related coatings or of plant records of the non-safety-related coatings used on components below the maximum flood level of a design basis LOCA or located above the maximum flood level and not inside cabinets or enclosures.	x) A report exists and concludes that the coatings used on these surfaces have a dry film density of $\geq$ 100 lb/ft^3. If a coating is used that has a lower dry film density, a report must exist and conclude that the coating will not transport. A report exists and concludes that inorganic zinc coatings used on these surfaces are Safety – Service Level I or have been quantified and justified in a program for management of unqualified coatings to demonstrate the unqualified coatings are acceptable for use.
	Inspections will be conducted on caulking, tags, and signs used inside containment below the maximum flood level of a design basis LOCA or located above the maximum flood level and not inside cabinets or enclosures.	A report exists and concludes that tags and signs used in these locations are made of steel or another metal with a density $\geq$ 100 lb/ft^3. In addition, a report exists and concludes that caulking used in these locations or coatings used on these signs or tags have a dry film density of $\geq$ 100 lb/ft^3. If a material is used that has a lower density, a report must exist and conclude that there is insufficient water flow to transport lightweight caulking, signs, or tags.
	Inspections will be conducted of ventilation filters and fiber-producing fire barriers used inside containment within the ZOI or below the maximum flood level of a design basis LOCA.	A report exists and concludes that the ventilation filters and fire barriers in these locations have a density of $\geq$ 100 lb/ft^3.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	xi) Inspection of the as-built CMT inlet diffuser will be conducted.	xi) The CMT inlet diffuser has a flow area $\geq$ 165 in^2.
	xii) Inspections will be conducted of the CMT level sensors (PXS-11A/B/D/C, - 12A/B/C/D, - 13A/B/C/D, - 14A/B/C/D) upper level tap lines.	xii) Each upper level tap line has a downward slope of $\geq$ 2.4 degrees from the centerline of the connection to the CMT to the centerline of the connection to the standpipe.
	xiii) Inspections will be conducted of the surfaces in the vicinity of the containment recirculation screens. The surfaces in the vicinity of the containment recirculation screens are the surfaces located above the bottom of the recirculation screens up to and including the bottom surface of the plate discussed in Table 2.2.3-4, item 8.c.vii, out at least 8 ft, 3 in perpendicular to the front and at least 7 feet to the side of the face of the screens.	xiii) These surfaces are stainless steel.
	xiv) Inspection will be conducted of the excore (source range, intermediate range, and power range) detectors.	xiv) A report exists and concludes that the aluminum surfaces of the excore detectors are encased in a watertight stainless steel or titanium housing.
8.d) The PXS provides pH adjustment of water flooding the containment following design basis accidents.	Inspections of the pH adjustment baskets will be conducted.	pH adjustment baskets exist, with a total calculated volume $\geq$ 560 ft^3. The pH baskets are located below plant elevation 107 ft, 2 in.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.a) The PXS provides a function to cool the outside of the reactor vessel during a severe accident.	i) A flow test and analysis for each IRWST drain line to the containment will be conducted. The test is initiated by opening isolation valves in each line. Test fixtures may be used to simulate squib valves.	i) The calculated flow resistance for each IRWST drain line between the IRWST and the containment is $\leq 4.44 \times 10^{-6}$ ft/gpm^2.
	ii) Inspections of the as-built reactor vessel insulation will be performed.	ii) The combined total flow area of the water inlets is not less than 6 ft^2. The combined total flow area of the steam outlet(s) is not less than 12 ft^2. A report exists and concludes that the minimum flow area between the vessel insulation and reactor vessel for the flow path that vents steam is not less than 12 ft^2 considering the maximum deflection of the vessel insulation with a static pressure of 12.95 ft of water.
	iii) Inspections will be conducted of the flow path(s) from the loop compartments to the reactor vessel cavity.	iii) A flow path with a flow area not less than 6 ft^2 exists from the loop compartment to the reactor vessel cavity.
9.b) The accumulator discharge check valves (PXS-PL-V028A/B and V029A/B) are of a different check valve type than the CMT discharge check valves (PXS-PL-V016A/B and V017A/B).	An inspection of the accumulator and CMT discharge check valves is performed.	The accumulator discharge check valves are of a different check valve type than the CMT discharge check valves.
9.c) Not used.		
10. Safety-related displays of the parameters identified in Table 2.2.3-1 can be retrieved in the MCR.	Inspection will be performed for the retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.3-1 can be retrieved in the MCR.
11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.3-1 to perform their active function(s).	ii) Stroke testing will be performed on remotely operated valves other than squib valves identified in Table 2.2.3-1 using the controls in the MCR.	ii) Controls in the MCR operate to cause remotely operated valves other than squib valves to perform their active functions.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11.b) The valves identified in Table 2.2.3-1 as having PMS control perform their active function after receiving a signal from the PMS.	ii) Testing will be performed on the remotely operated valves other than squib valves identified in Table 2.2.3-1 using real or simulated signals into the PMS.	ii) Remotely operated valves other than squib valves perform the active function identified in the table after a signal is input to the PMS.
	iii) Testing will be performed to demonstrate that remotely operated PXS isolation valves PXS-V014A/B, V015A/B, V108A/B open within the required response times.	iii) These valves open within 20 seconds after receipt of an actuation signal.
12.b) After loss of motive power, the remotely operated valves identified in Table 2.2.3-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.3-1 assumes the indicated loss of motive power position.
13. Displays of the parameters identified in Table 2.2.3-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.2.3-3 in the MCR.	Displays identified in Table 2.2.3-3 can be retrieved in the MCR.
11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.3-1 to perform their active function(s).	i) Testing will be performed on the squib valves identified in Table 2.2.3-1 using controls in the MCR, without stroking the valve.	i) Controls in the MCR operate to cause a signal at the squib valve electrical leads that is capable of actuating the squib valve.
11.b) The valves identified in Table 2.2.3-1 as having PMS control perform their active function after receiving a signal from the PMS.	i) Testing will be performed on the squib valves identified in Table 2.2.3-1 using real or simulated signals into the PMS without stroking the valve.	i) Squib valves receive an electrical signal at the valve electrical leads that is capable of actuating the valve after a signal is input to the PMS.
11.c) The valves identified in Table 2.2.3-1 as having DAS control perform their active function after receiving a signal from the DAS.	i) Testing will be performed on the squib valves identified in Table 2.2.3-1 using real or simulated signals into the DAS without stroking the valve.	i) Squib valves receive an electrical signal at the valve electrical leads that is capable of actuating the valve after a signal is input to the DAS.
	ii) Testing will be performed on the remotely operated valves other than squib valves identified in Table 2.2.3-1 using real or simulated signals into the DAS.	ii) Remotely operated valves other than squib valves perform the active function identified in Table 2.2.3-1 after a signal is input to the DAS.

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Table 2.2.3-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12.a) The squib valves and check valves identified in Table 2.2.3-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of squib valves will be performed that demonstrate the capability of the valve to operate under its design condition.	i) A test report exists and concludes that each squib valve changes position as indicated in Table 2.2.3-1 under design conditions.
	ii) Inspection will be performed for the existence of a report verifying that the as-built squib valves are bounded by the tests or type tests.	ii) A report exists and concludes that the as-built squib valves are bounded by the tests or type tests.
	iii) Deleted	iii) Deleted
	iv) Exercise testing of the check valves with active safety functions identified in Table 2.2.3-1 will be performed under preoperational test pressure, temperature, and fluid flow conditions.	iv) Each check valve changes position as indicated in Table 2.2.3-1

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Table 2.2.3-5

Component Name	Tag No.	Component Location
Passive Residual Heat Removal Heat Exchanger (PRHR HX)	PXS-ME-01	Containment Building
Accumulator Tank A	PXS-MT-01A	Containment Building
Accumulator Tank B	PXS-MT-01B	Containment Building
Core Makeup Tank (CMT) A	PXS-MT-02A	Containment Building
CMT B	PXS-MT-02B	Containment Building
IRWST	PXS-MT-03	Containment Building
IRWST Screen A	PXS-MY-Y01A	Containment Building
IRWST Screen B	PXS-MY-Y01B	Containment Building
IRWST Screen C	PXS-MY-Y01C	Containment Building
Containment Recirculation Screen A	PXS-MY-Y02A	Containment Building
Containment Recirculation Screen B	PXS-MY-Y02B	Containment Building
pH Adjustment Basket 3A	PXS-MY-Y03A	Containment Building
pH Adjustment Basket 3B	PXS-MY-Y03B	Containment Building
pH Adjustment Basket 4A	PXS-MY-Y04A	Containment Building
pH Adjustment Basket 4B	PXS-MY-Y04B	Containment Building

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Table 2.2.3-6 is not used.

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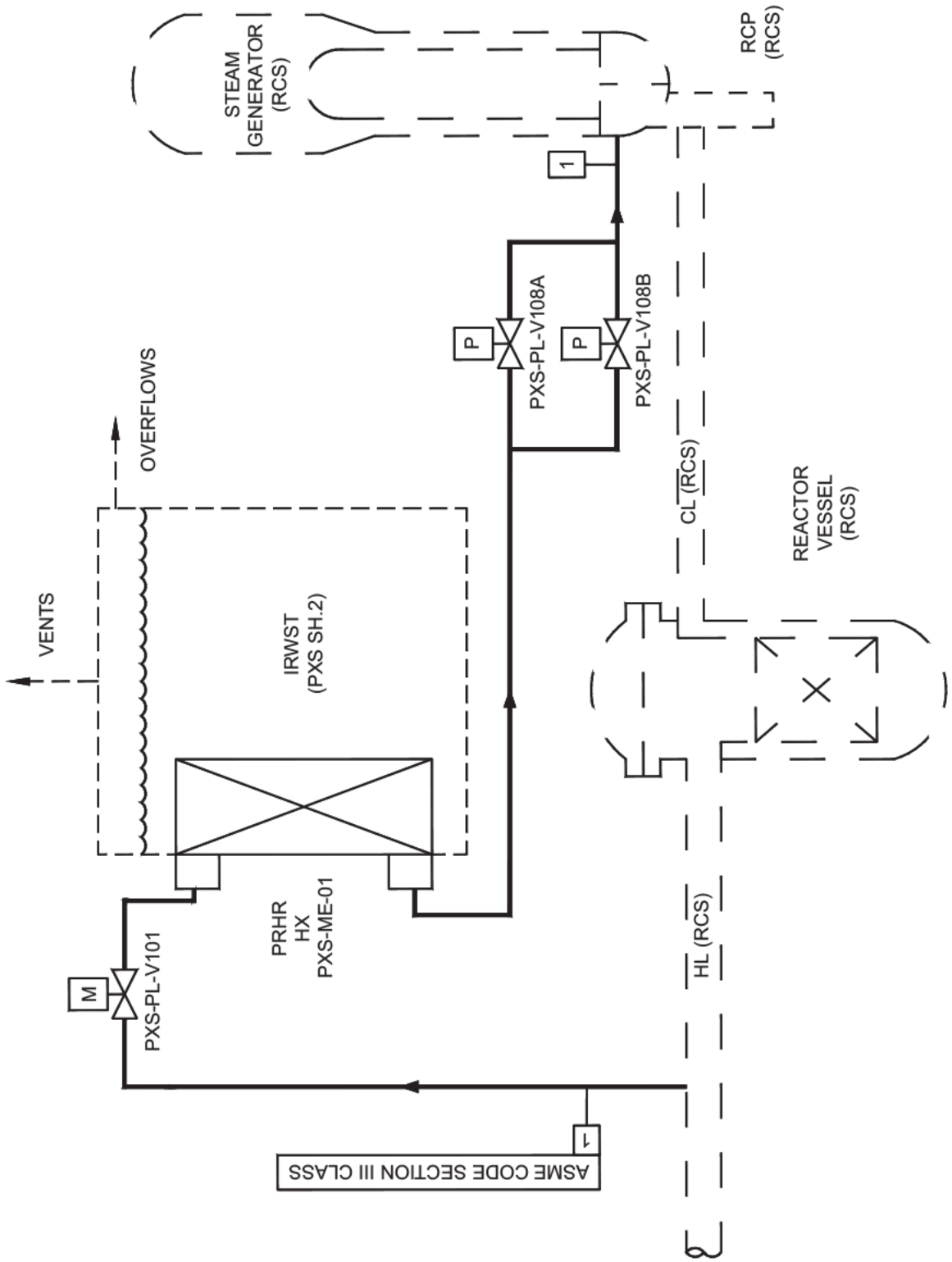

Figure 2.2.3-1 (Sheet 1 of 2)
Passive Core Cooling System

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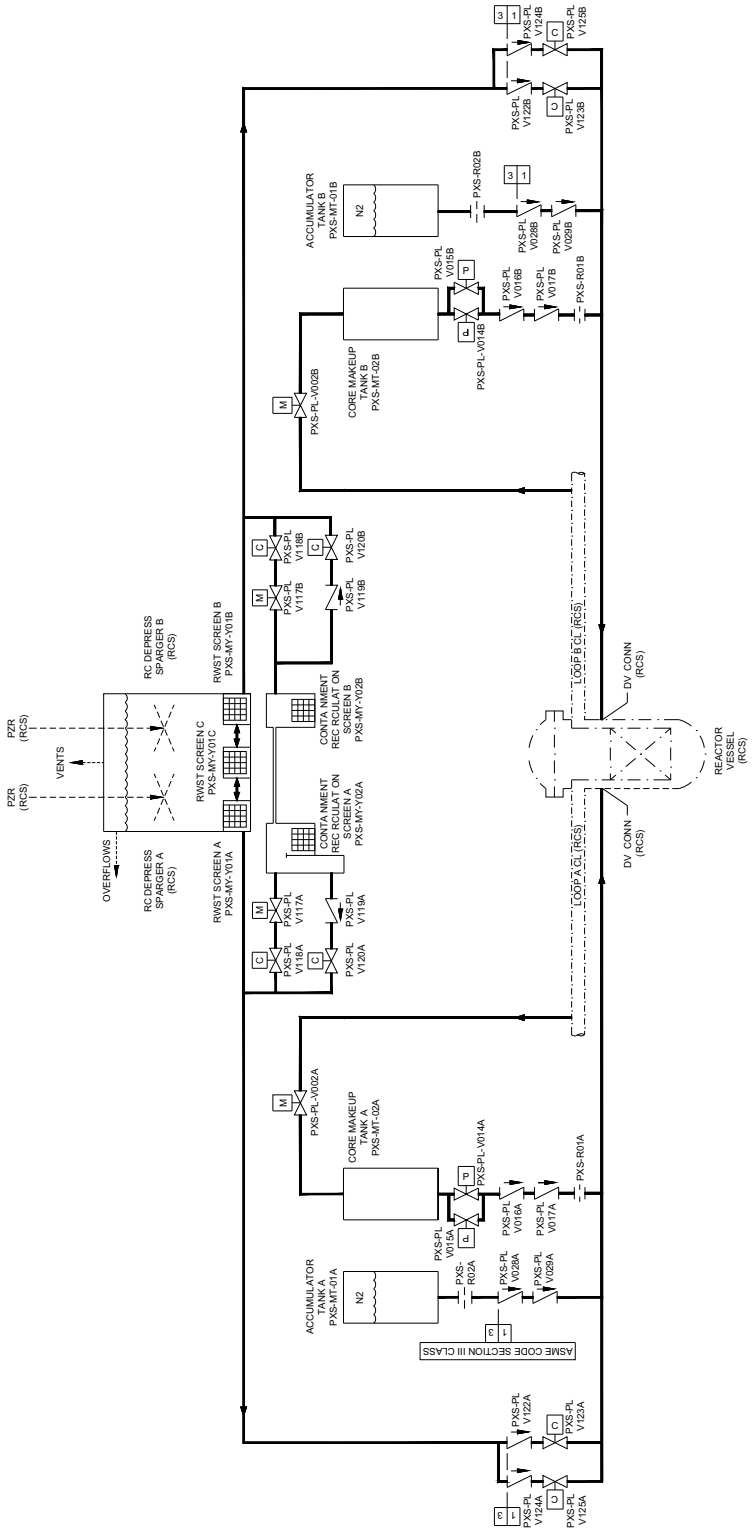

Figure 2.2.3-1 (Sheet 2 of 2)
Passive Core Cooling System

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2.2.4 Steam Generator System

Design Description

The steam generator system (SGS) and portions of the main and startup feedwater system (FWS) transport and control feedwater from the condensate system to the steam generators during normal operation. The SGS and portions of the main steam system (MSS) and turbine system (MTS) transport and control steam from the steam generators to the turbine generator during normal operations. These systems also isolate the steam generators from the turbine generator and the condensate system during design basis accidents.

The SGS is as shown in Figure 2.2.4-1, sheets 1 and 2, and portions of the FWS, MSS, and MTS are as shown in Figure 2.2.4-1, sheet 3, and the locations of the components in these systems is as shown in Table 2.2.4-5.

1. The functional arrangement of the SGS and portions of the FWS, MSS, and MTS are as described in the Design Description of this Section 2.2.4.

2. a) The components identified in Table 2.2.4-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.2.4-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.2.4-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.2.4-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.2.4-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.2.4-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. a) The seismic Category I equipment identified in Table 2.2.4-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.2.4-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

6. Each of the as-built lines identified in Table 2.2.4-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

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7. a) The Class 1E equipment identified in Table 2.2.4-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.2.4-1 are powered from their respective Class 1E division.

   c) Separation is provided between SGS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

8. The SGS provides the following safety-related functions:

   a) The SGS provides a heat sink for the reactor coolant system (RCS) and provides overpressure protection.

   b) During design basis events, the SGS limits steam generator blowdown and feedwater flow to the steam generator.

   c) The SGS preserves containment integrity by isolation of the SGS lines penetrating the containment. The inside containment isolation function (isolating the RCS and containment atmosphere from the environment) is provided by the steam generator, tubes, and SGS lines inside containment while isolation outside containment is provided by manual and automatic valves.

9. The SGS provides the following nonsafety-related functions:

   a) Components within the main steam system, main and startup feedwater system, and the main turbine system identified in Table 2.2.4-3 provide backup isolation of the SGS to limit steam generator blowdown and feedwater flow to the steam generator.

   b) During shutdown operations, the SGS removes decay heat by delivery of startup feedwater to the steam generator and venting of steam from the steam generators to the atmosphere.

10. Safety-related displays identified in Table 2.2.4-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.4-1 to perform active functions.

    b) The valves identified in Table 2.2.4-1 as having PMS control perform an active safety function after receiving a signal from PMS.

12. a) The motor-operated valves identified in Table 2.2.4-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.2.4-1 assume the indicated loss of motive power position.

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Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.2.4-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the SGS.

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Table 2.2.4-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Main Steam Safety Valve SG01	SGS-PL-V030A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL-V030B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG01	SGS-PL-V031A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL-V031B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG01	SGS-PL-V032A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL-V032B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.2.4-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Main Steam Safety Valve SG01	SGS-PL-V033A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL–V033B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG01	SGS-PL–V034A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL–V034B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG01	SGS-PL–V035A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL–V035B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.2.4-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Power-operated Relief Valve Block Motor-operated Valve Steam Generator 01	SGS-PL-V027A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Power-operated Relief Valve Block Motor-operated Valve Steam Generator 02	SGS-PL-V027B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Steam Line Condensate Drain Isolation Valve	SGS-PL-V036A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Line Condensate Drain Isolation Valve	SGS-PL-V036B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Steam Line Isolation Valve	SGS-PL-V040A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Main Steam Line Isolation Valve	SGS-PL-V040B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Steam Line Condensate Drain Control Valve	SGS-PL-V086A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Line Condensate Drain Control Valve	SGS-PL-V086B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed

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Table 2.2.4-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Main Feedwater Isolation Valve	SGS-PL-V057A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Main Feedwater Isolation Valve	SGS-PL-V057B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Startup Feedwater Isolation Motor-operated Valve	SGS-PL-V067A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Startup Feedwater Isolation Motor-operated Valve	SGS-PL-V067B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Steam Generator Blowdown Isolation Valve	SGS-PL-V074A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Generator Blowdown Isolation Valve	SGS-PL-V074B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Generator Blowdown Isolation Valve	SGS-PL-V075A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Generator Blowdown Isolation Valve	SGS-PL-V075B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Power-operated Relief Valve	SGS-PL-V233A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Power-operated Relief Valve	SGS-PL-V233B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed

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Table 2.2.4-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Main Steam Isolation Valve Bypass Isolation	SGS-PL-V240A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Steam Isolation Valve Bypass Isolation	SGS-PL-V240B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Feedwater Control Valve	SGS-PL-V250A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Feedwater Control Valve	SGS-PL-V250B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Startup Feedwater Control Valve	SGS-PL-V255A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Startup Feedwater Control Valve	SGS-PL-V255B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Feedwater Thermal Relief Valve	SGS-PL-V257A	Yes	Yes	No	-/-	-	-	Transfer Open/ Transfer Closed	-
Main Feedwater Thermal Relief Valve	SGS-PL-V257B	Yes	Yes	No	-/-	-	-	Transfer Open/ Transfer Closed	-
Startup Feedwater Thermal Relief Valve	SGS-PL-V258A	Yes	Yes	No	-/-	-	-	Transfer Open/ Transfer Closed	-

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Table 2.2.4-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Startup Feedwater Thermal Relief Valve	SGS-PL-V258B	Yes	Yes	No	-/-	-	-	Transfer Open/ Transfer Closed	-
Steam Generator 1 Narrow Range Level Sensor	SGS-001	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Narrow Range Level Sensor	SGS–002	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Narrow Range Level Sensor	SGS–003	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Narrow Range Level Sensor	SGS–004	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Narrow Range Level Sensor	SGS–005	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Narrow Range Level Sensor	SGS–006	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Narrow Range Level Sensor	SGS–007	No	Yes	-	Yes/Yes	Yes	-	-	-

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Table 2.2.4-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Steam Generator 2 Narrow Range Level Sensor	SGS-008	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Wide Range Level Sensor	SGS-011	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Wide Range Level Sensor	SGS-012	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Wide Range Level Sensor	SGS-013	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Wide Range Level Sensor	SGS-014	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Wide Range Level Sensor	SGS-015	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Wide Range Level Sensor	SGS-016	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Wide Range Level Sensor	SGS-017	No	Yes	-	Yes/Yes	Yes	-	-	-

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Table 2.2.4-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Steam Generator 2 Wide Range Level Sensor	SGS-018	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 1 Pressure Sensor	SGS-030	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 1 Pressure Sensor	SGS-031	No	Yes	-	Yes/No	Yes	-	-	-
Main Steam Line Steam Generator 1 Pressure Sensor	SGS-032	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 1 Pressure Sensor	SGS-033	No	Yes	-	Yes/No	Yes	-	-	-
Main Steam Line Steam Generator 2 Pressure Sensor	SGS-034	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 2 Pressure Sensor	SGS-035	No	Yes	-	Yes/No	Yes	-	-	-
Main Steam Line Steam Generator 2 Pressure Sensor	SGS-036	No	Yes	-	Yes/Yes	Yes	-	-	-

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Table 2.2.4-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Main Steam Line Steam Generator 2 Pressure Sensor	SGS-037	No	Yes	-	Yes/No	Yes	-	-	-
Steam Generator 1 Startup Feedwater Flow Sensor	SGS-55A	No	Yes	-	Yes/No	Yes	-	-	-
Steam Generator 1 Startup Feedwater Flow Sensor	SGS-55B	No	Yes	-	Yes/No	Yes	-	-	-
Steam Generator 2 Startup Feedwater Flow Sensor	SGS-56A	No	Yes	-	Yes/No	Yes	-	-	-
Steam Generator 2 Startup Feedwater Flow Sensor	SGS-56B	No	Yes	-	Yes/No	Yes	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.4-2

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
Main Feedwater Line	SGS-PL-L002A, L002B	Yes	No	No
Main Feedwater Line	SGS-PL-L003A, L003B	Yes	No	No
Startup Feedwater Line	SGS-PL-L004A, L004B	Yes	No	No
Startup Feedwater Line	SGS-PL-L005A, L005B	Yes	No	No
Main Steam Line (within containment)	SGS-PL-L006A, L006B	Yes	Yes	Yes
Main Steam Line (outside of containment)	SGS-PL-L006A, L006B	Yes	No	Yes
Main Steam Line	SGS-PL-L007A, L007B	Yes	No	No
Safety Valve Inlet Line	SGS-PL-L015A, L015B, L015C, L015D, L015E, L015F, L015G, L015H, L015J, L015K, L015L, L015M	Yes	No	Yes
Safety Valve Discharge Line	SGS-PL-L018A, L018B, L018C, L018D, L018E, L018F, L018G, L018H, L018J, L018K, L018L, L018M	Yes	No	Yes
Power-operated Relief Block Valve Inlet Line	SGS-PL-L024A, L024B	Yes	No	No
Power-operated Relief Valve Inlet Line	SGS-PL-L014A, L014B	Yes	No	No

Note: Dash (-) indicates not applicable.

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Table 2.2.4-2 (cont.)

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
Main Steam Isolation Valve Bypass Inlet Line	SGS-PL-L022A, L022B	Yes	No	No
Main Steam Isolation Valve Bypass Outlet Line	SGS-PL-L023A, L023B	Yes	No	No
Main Steam Condensate Drain Line	SGS-PL-L021A, L021B	Yes	No	No
Steam Generator Blowdown Line	SGS-PL-L009A, L009B	Yes	No	No
Steam Generator Blowdown Line	SGS-PL-L027A, L027B	Yes	No	No
Steam Generator Blowdown Line	SGS-PL-L010A, L010B	Yes	No	No

Note: Dash (-) indicates not applicable.

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Table 2.2.4-3

Equipment Name	Tag No.	Control Function
Turbine Stop Valve	MTS-PL-V001A	Close
Turbine Stop Valve	MTS-PL-V001B	Close
Turbine Control Valve	MTS-PL-V002A	Close
Turbine Control Valve	MTS-PL-V002B	Close
Turbine Stop Valve	MTS-PL-V003A	Close
Turbine Stop Valve	MTS-PL-V003B	Close
Turbine Control Valve	MTS-PL-V004A	Close
Turbine Control Valve	MTS-PL-V004B	Close
Turbine Bypass Control Valve	MSS-PL-V001	Close
Turbine Bypass Control Valve	MSS-PL-V002	Close
Turbine Bypass Control Valve	MSS-PL-V003	Close
Turbine Bypass Control Valve	MSS-PL-V004	Close
Turbine Bypass Control Valve	MSS-PL-V005	Close
Turbine Bypass Control Valve	MSS-PL-V006	Close
Moisture Separator Reheater 2nd Stage Steam Isolation Valve	MSS-PL-V015A	Close
Moisture Separator Reheater 2nd Stage Steam Isolation Valve	MSS-PL-V015B	Close
Main Feedwater Pump	FWS-MP-02A	Trip
Main Feedwater Pump	FWS-MP-02B	Trip
Main Feedwater Pump	FWS-MP-02C	Trip
Startup Feedwater Pump	FWS-MP-03A	Trip
Startup Feedwater Pump	FWS-MP-03B	Trip

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Table 2.2.4-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the SGS and portions of the FWS, MSS, and MTS are as described in the Design Description of this Section 2.2.4.	Inspection of the as-built system will be performed.	The as-built SGS and portions of the FWS, MSS, and MTS conform with the functional arrangement as defined in the Design Description of this Section 2.2.4.
2.a) The components identified in Table 2.2.4-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.2.4-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.2.4-1 and 2.2.4-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.2.4-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.2.4-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

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Table 2.2.4-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.a) The components identified in Table 2.2.4-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The piping identified in Table 2.2.4-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the components and piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components and piping identified in Tables 2.2.4-1 and 2.2.4-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
5.b) Each of the lines identified in Table 2.2.4-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report concluding that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.2.4-2 for which functional capability is required meets the requirements for functional capability.
6. Each of the as-built lines identified in Table 2.2.4-2 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	Inspection will be performed for the existence of an LBB evaluation report or an evaluation report on the protection from effects of a pipe break. Tier 1 Material, Section 3.3, Nuclear Island Buildings, contains the design descriptions and inspections, tests, analyses, and acceptance criteria for protection from the dynamic effects of pipe rupture.	An LBB evaluation report exists and concludes that the LBB acceptance criteria are met by the as-built SGS piping and piping materials, or a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided.
5.a) The seismic Category I equipment identified in Table 2.2.4-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.2.4-1 is located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.2.4-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.

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Table 2.2.4-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) The Class 1E equipment identified in Table 2.2.4-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.2.4-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.2.4-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
7.b) Not used.		
7.c) Not used.		

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Table 2.2.4-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.a) The SGS provides a heat sink for the RCS and provides overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	i) Inspections will be conducted to confirm that the value of the vendor code plate rating of the steam generator safety valves is greater than or equal to system relief requirements. ii) Testing and analyses in accordance with ASME Code Section III will be performed to determine set pressure.	i) The sum of the rated capacities recorded on the valve vendor code plates of the steam generator safety valves exceeds 8,240,000 lb/hr per steam generator. ii) A report exists to indicate the set pressure of the valves is less than 1305 psig.
8.b) During design basis events, the SGS limits steam generator blowdown and feedwater flow to the steam generator.	i) Not used. ii) Inspection will be performed for the existence of a report confirming that the area of the flow limiting orifice within the SG main steam outlet nozzle will limit releases to the containment.	ii) A report exists to indicate the installed flow limiting orifice within the SG main steam line discharge nozzle does not exceed 1.4 sq. ft.
8.c) Not used.		
9.a) Components within the main steam system, main and startup feedwater system, and the main turbine system identified in Table 2.2.4-3 provide backup isolation of the SGS to limit steam generator blowdown and feedwater flow to the steam generator.	ii) Testing will be performed to confirm the trip of the pumps identified in Table 2.2.4-3.	ii) The pumps identified in Table 2.2.4-3 trip after a signal is generated by the PMS.
9.b) During shutdown operations, the SGS removes decay heat by delivery of startup feedwater to the steam generator and venting of steam from the steam generators to the atmosphere.	i) Not used. ii) Type tests and/or analyses will be performed to demonstrate the ability of the power-operated relief valves to discharge steam from the steam generators to the atmosphere.	ii) A report exists and concludes that each power-operated relief valve will relieve greater than 300,000 lb/hr at 1106 psia ±10 psi.
9.a) Components within the main steam system, main and startup feedwater system, and the main turbine system identified in Table 2.2.4-3 provide backup isolation of the SGS to limit steam generator blowdown and feedwater flow to the steam generator.	i) Testing will be performed to confirm closure of the valves identified in Table 2.2.4-3.	i) The valves identified in Table 2.2.4-3 close after a signal is generated by the PMS.

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Table 2.2.4-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. Safety-related displays identified in Table 2.2.4-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.4-1 can be retrieved in the MCR.
11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.4-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves listed in Table 2.2.4-1 using controls in the MCR.	Controls in the MCR operate to cause the remotely operated valves to perform active safety functions.
11.b) The valves identified in Table 2.2.4-1 as having PMS control perform an active safety function after receiving a signal from PMS.	i) Testing will be performed on the remotely operated valves listed in Table 2.2.4-1 using real or simulated signals into the PMS.	i) The remotely-operated valves identified in Table 2.2.4-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
	ii) Testing will be performed to demonstrate that remotely operated SGS isolation valves SGS-V027A/B, V040A/B, V057A/B, V250A/B close within the required response times.	ii) These valves close within the following times after receipt of an actuation signal: V027A/B < 44 sec V040A/B, V057A/B < 5 sec V250A/B < 5 sec
12.a) The motor-operated valves identified in Table 2.2.4-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under pre-operational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.2.4-1 under pre-operational test conditions.
12.b) After loss of motive power, the remotely operated valves identified in Table 2.2.4-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.4-1 assumes the indicated loss of motive power position. Motive power to SGS-PL-V040A/B and SGS-PL-V057A/B is electric power to the actuator from plant services.
12.a) The motor-operated valves identified in Table 2.2.4-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed to demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.2.4-1 under design conditions.
	ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tests or type tests.	ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tests or type tests.

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Table 2.2.4-5

Component Name	Tag No.	Component Location
Main Steam Line Isolation Valve	SGS-PL-V040A	Auxiliary Building
Main Steam Line Isolation Valve	SGS-PL-V040B	Auxiliary Building
Main Feedwater Isolation Valve	SGS-PL-V057A	Auxiliary Building
Main Feedwater Isolation Valve	SGS-PL-V057B	Auxiliary Building
Main Feedwater Control Valve	SGS-PL-V250A	Auxiliary Building
Main Feedwater Control Valve	SGS-PL-V250B	Auxiliary Building
Turbine Stop Valves	MTS-PL-V001A MTS-PL-V001B MTS-PL-V003A MTS-PL-V003B	Turbine Building
Turbine Control Valves	MTS-PL-V002A MTS-PL-V002B MTS-PL-V004A MTS-PL-V004B	Turbine Building
Main Feedwater Pumps	FWS-MP-02A FWS-MP-02B FWS-MP-02C	Turbine Building
Feedwater Booster Pumps	FWS-MP-01A FWS-MP-01B FWS-MP-01C	Turbine Building

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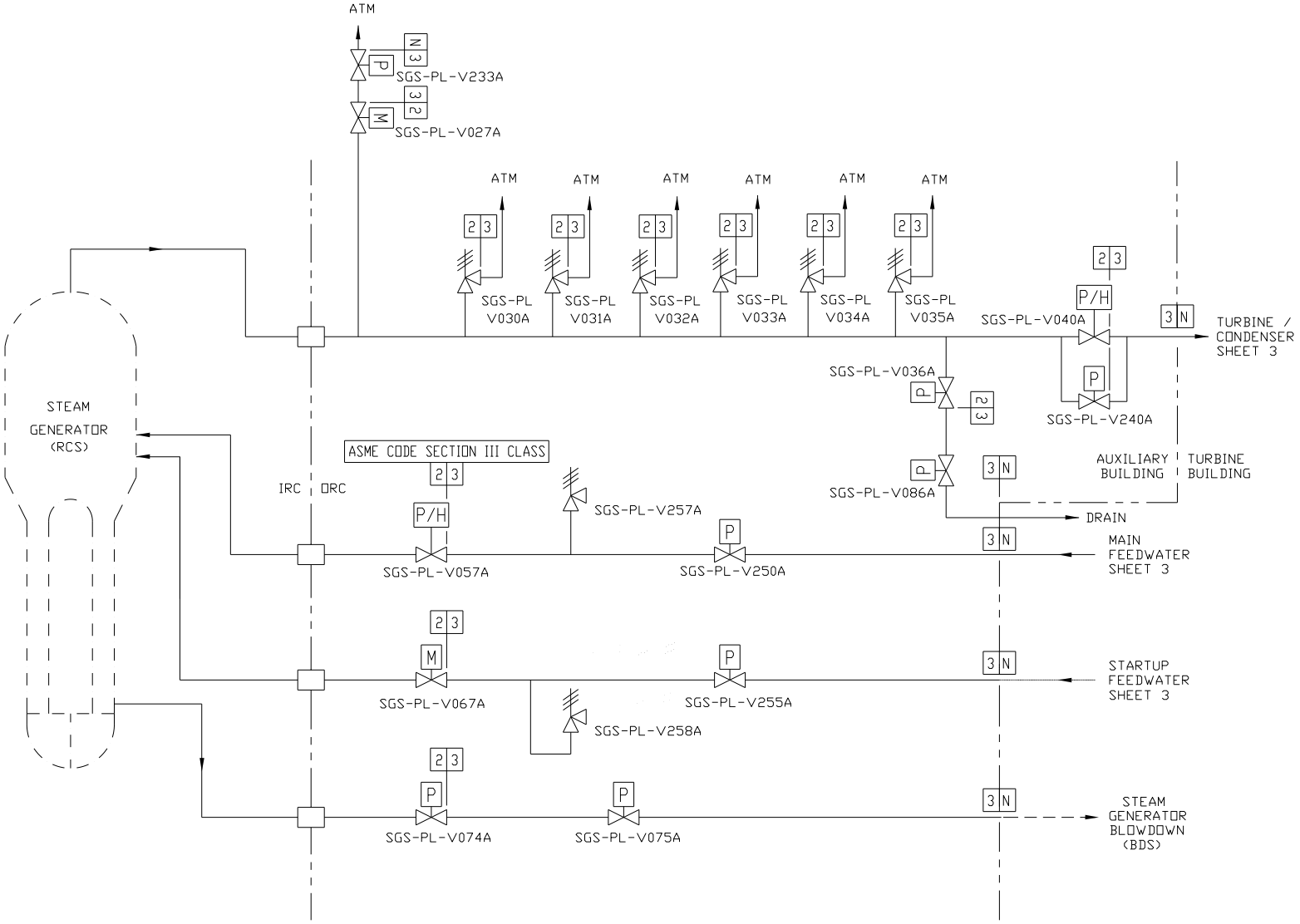

Figure 2.2.4-1 (Sheet 1 of 3)
Steam Generator System

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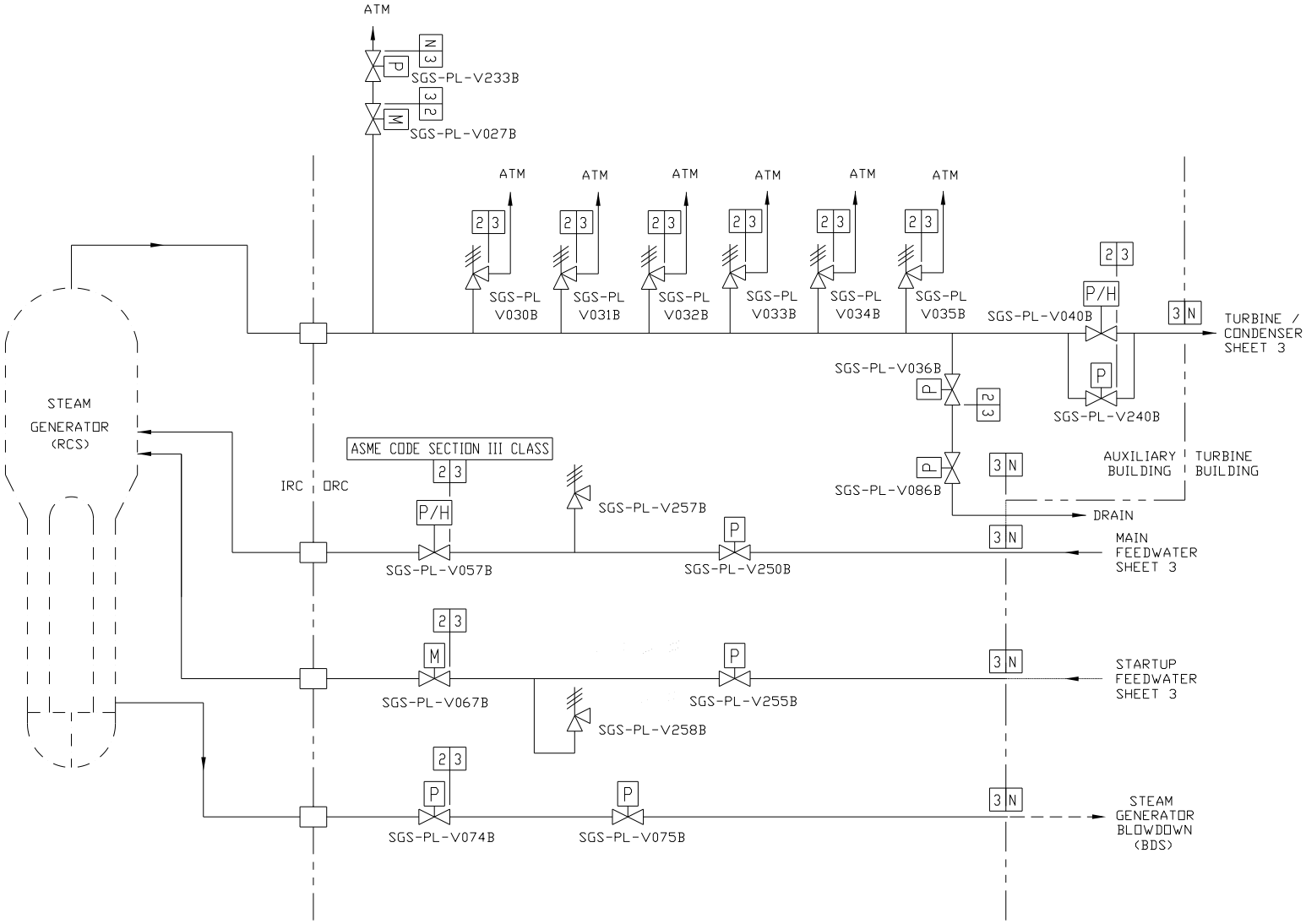

Figure 2.2.4-1 (Sheet 2 of 3)
Steam Generator System

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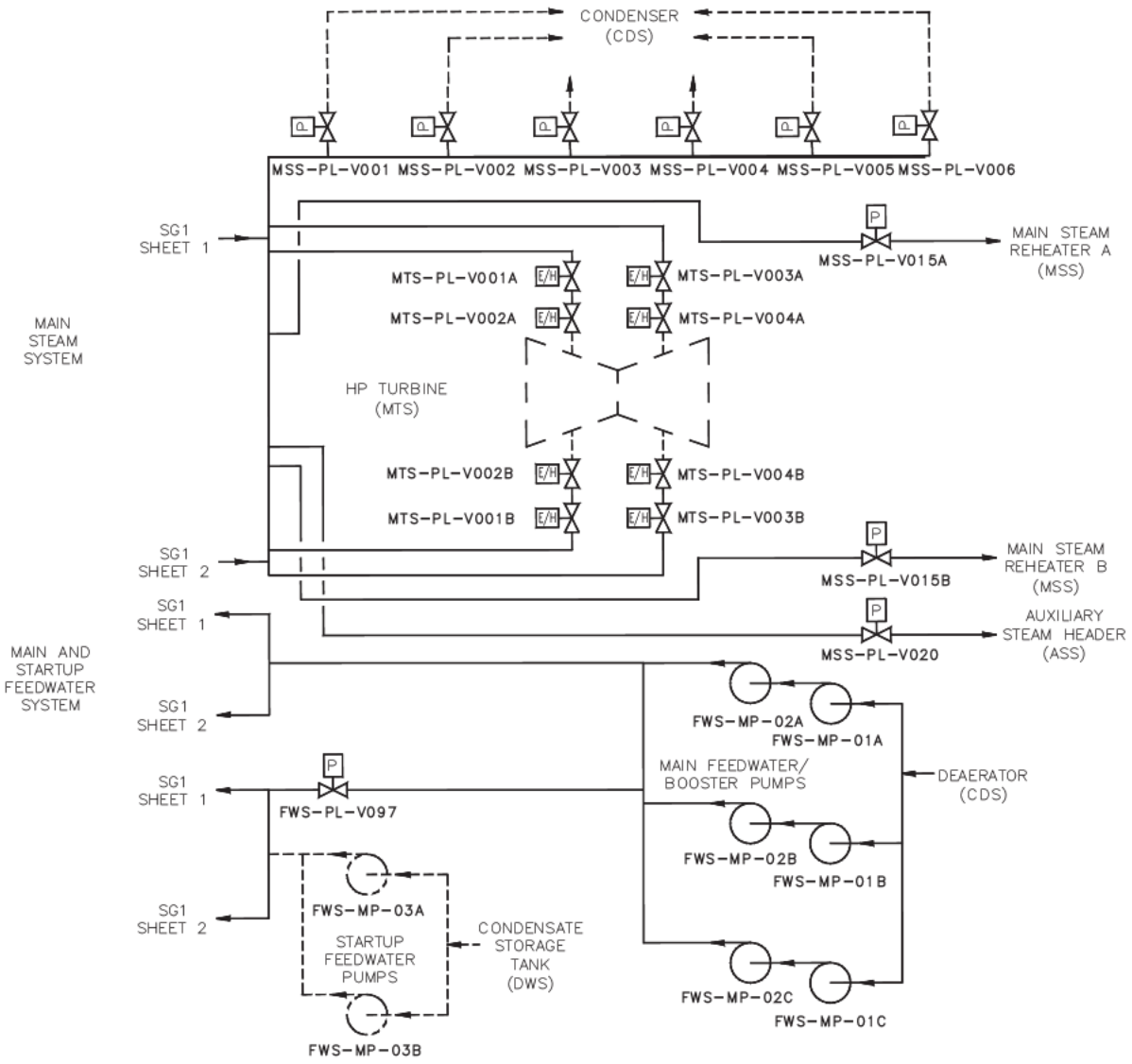

Figure 2.2.4-1 (Sheet 3 of 3)
Steam Generator System

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2.2.5 Main Control Room Emergency Habitability System

Design Description

The main control room emergency habitability system (VES) provides a supply of breathable air for the main control room (MCR) occupants and maintains the MCR at a positive pressure with respect to the surrounding areas whenever ac power is not available to operate the nuclear island nonradioactive ventilation system (VBS), MCR differential pressure is not maintained, or high radioactivity is detected in the MCR air supply. (See Tier 1 material, Section 3.5 for Radiation Monitoring). The VES also limits the heatup of the MCR, the 1E instrumentation and control (I&C) equipment rooms, and the Class 1E dc equipment rooms by using the heat capacity of surrounding structures.

The VES is as shown in Figure 2.2.5-1 and the component locations of the VES are as shown in Table 2.2.5-6.

1. The functional arrangement of the VES is as described in the Design Description of this Section 2.2.5.

2. a) The components identified in Table 2.2.5-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.2.5-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.2.5-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.2.5-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.2.5-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.2.5-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. a) The seismic Category I equipment identified in Table 2.2.5-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.2.5-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

6. a) The Class 1E components identified in Table 2.2.5-1 are powered from their respective Class 1E division.

   b) Separation is provided between VES Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

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7. The VES provides the following safety-related functions:

   a) The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.

   b) The VES maintains the MCR pressure boundary at a positive pressure with respect to the surrounding areas. There is a discharge of air through the MCR vestibule.

   c) The heat loads within the MCR, the I&C equipment rooms, and the Class 1E dc equipment rooms are within design basis assumptions to limit the heatup of the rooms identified in Table 2.2.5-4.

   d) The system provides a passive recirculation flow of MCR air to maintain main control room dose rates below an acceptable level during VES operation.

   e) The system provides shielding below the VES filter that is sufficient to ensure main control room doses are below an acceptable level during VES operation.

8. Safety-related displays identified in Table 2.2.5-1 can be retrieved in the MCR.

9. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.2.5-1 to perform their active functions.

   b) The valves identified in Table 2.2.5-1 as having protection and safety monitoring system (PMS) control perform their active safety function after receiving a signal from the PMS.

   c) The MCR Load Shed Panels identified in Table 2.2.5-1 perform their active safety function after receiving a signal from the PMS.

10. After loss of motive power, the remotely operated valves identified in Table 2.2.5-1 assume the indicated loss of motive power position.

11. Displays of the parameters identified in Table 2.2.5-3 can be retrieved in the MCR.

12. The background noise level in the MCR does not exceed 65 dB(A) at the operator workstations when the VES is operating.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.2.5-5 specifies the inspections, tests, analyses, and associated acceptance criteria for the VES.

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Table 2.2.5-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
MCR Load Shed Panel 1	VES-EP-01	No	Yes	-	Yes/No	Yes	Yes	De-energize MCR Loads	-
MCR Load Shed Panel 2	VES-EP-02	No	Yes	-	Yes/No	Yes	Yes	De-energize MCR Loads	-
Emergency Air Storage Tank 01	VES-MT-01	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 02	VES-MT-02	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 03	VES-MT-03	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 04	VES-MT-04	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 05	VES-MT-05	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 06	VES-MT-06	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 07	VES-MT-07	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 08	VES-MT-08	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 09	VES-MT-09	No	Yes	-	-/-	-	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.5-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Emergency Air Storage Tank 10	VES-MT-10	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 11	VES-MT-11	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 12	VES-MT-12	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 13	VES-MT-13	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 14	VES-MT-14	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 15	VES-MT-15	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 16	VES-MT-16	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 17	VES-MT-17	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 18	VES-MT-18	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 19	VES-MT-19	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 20	VES-MT-20	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 21	VES-MT-21	No	Yes	-	-/-	-	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.5-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Emergency Air Storage Tank 22	VES-MT-22	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 23	VES-MT-23	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 24	VES-MT-24	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 25	VES-MT-25	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 26	VES-MT-26	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 27	VES-MT-27	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 28	VES-MT-28	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 29	VES-MT-29	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 30	VES-MT-30	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 31	VES-MT-31	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 32	VES-MT-32	No	Yes	-	-/-	-	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.5-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Air Delivery Alternate Isolation Valve	VES-PL-V001	Yes	Yes	No	-/-	No	-	Transfer Open	-
Eductor Flow Path Isolation Valve	VES-PL-V045	Yes	Yes	No	-/-	No	-	Transfer Close	-
Eductor Bypass Isolation Valve	VES-PL-V046	Yes	Yes	No	-/-	No	-	Transfer Open	-
Pressure Regulating Valve A	VES-PL-V002A	Yes	Yes	No	-/-	No	-	Throttle Flow	-
Pressure Regulating Valve B	VES-PL-V002B	Yes	Yes	No	-/-	No	-	Throttle Flow	-
MCR Air Delivery Isolation Valve A	VES-PL-V005A	Yes	Yes	Yes	Yes/No	Yes	Yes	Transfer Open	Open
MCR Air Delivery Isolation Valve B	VES-PL-V005B	Yes	Yes	Yes	Yes/No	Yes	Yes	Transfer Open	Open
Temporary Instrument Isolation Valve A	VES-PL-V018	Yes	Yes	No	-/-	No	No	Transfer Open	-
Temporary Instrument Isolation Valve B	VES-PL-V019	Yes	Yes	No	-/-	No	No	Transfer Open	-
MCR Pressure Relief Isolation Valve A	VES-PL-V022A	Yes	Yes	Yes	Yes/No	Yes	Yes	Transfer Open	Open
MCR Pressure Relief Isolation Valve B	VES-PL-V022B	Yes	Yes	Yes	Yes/No	Yes	Yes	Transfer Open	Open

Note: Dash (-) indicates not applicable.

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Table 2.2.5-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Air Tank Safety Relief Valve A	VES-PL-V040A	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Close	-
Air Tank Safety Relief Valve B	VES-PL-V040B	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Close	-
Air Tank Safety Relief Valve C	VES-PL-V040C	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Close	-
Air Tank Safety Relief Valve D	VES-PL-V040D	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Close	-
Main Air Flow Path Isolation Valve	VES-PL-V044	Yes	Yes	No	-/-	No	-	Transfer Close	-
MCR Air Filtration Line Eductor	VES-PY-N01	Yes	Yes	-	-	-	-	-	-
MCR Air Filtration Line Charcoal Filter	VES-MY-F01	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line HEPA Filter	VES-MY-F02	No	Yes	-	-	-	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.5-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
MCR Air Filtration Line Postfilter	VES-MY-F03	No	Yes	-	-	-	-	-	-
MCR Filter Shielding	12401-NS-01	No	Yes	-	-	-	-	-	-
MCR Gravity Relief Dampers	VES-MD-D001A	No	Yes	-	-	-	-	-	-
MCR Gravity Relief Dampers	VES-MD-D001B	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Supply Damper	VES-MD-D002	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Supply Damper	VES-MD-D003	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Silencer	VES-MY-Y01	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Silencer	VES-MY-Y02	No	Yes	-	-	-	-	-	-
MCR Air Delivery Line Flow Sensor	VES-003A	No	Yes	-	Yes/No	Yes	-	-	-
MCR Air Delivery Line Flow Sensor	VES-003B	No	Yes	-	Yes/No	Yes	-	-	-
MCR Differential Pressure Sensor A	VES-004A	No	Yes	-	Yes/No	Yes	-	-	-
MCR Differential Pressure Sensor B	VES-004B	No	Yes	-	Yes/No	Yes	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.2.5-2

Line Name	Line Number	ASME Code Section III	Functional Capability Required
MCR Relief Line	VES-PL-022A	Yes	Yes
MCR Relief Line	VES-PL-022B	Yes	Yes

Table 2.2.5-3

Equipment	Tag No.	Display
Air Storage Tank Pressure	VES-001A	Yes
Air Storage Tank Pressure	VES-001B	Yes

Table 2.2.5-4

Room Name	Room Numbers	Heat Load 0 to 24 Hours (Btu/s)	Heat Load 24 to 72 Hours (Btu/s)
MCR Envelope	12401	23.5 (hour 0 to 0.5) 14.5 (hour 0.5 to 3.5) 4.75 (hour 3.5 through 24)	3.95
I&C Rooms	12301	3.50	1.10
	12302	3.20	2.50
	12304	4.30	3.20
	12305	4.20	1.50
dc Equipment Rooms	12201	2.00	2.00
	12203	3.30	3.30
	12205	2.40	2.40
	12207	3.45	3.45

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Table 2.2.5-5
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the VES is as described in the Design Description of this Section 2.2.5.	Inspection of the as-built system will be performed.	The as-built VES conforms with the functional arrangement described in the Design Description of this Section 2.2.5.
2.a) The components identified in Table 2.2.5-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.2.5-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.2.5-1 and 2.2.5-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.2.5-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.2.5-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.2.5-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The piping identified in Table 2.2.5-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A pressure test will be performed on the components and piping required by the ASME Code Section III to be pressure tested.	A report exists and concludes that the results of the pressure test of the components and piping identified in Tables 2.2.5-1 and 2.2.5-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

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Table 2.2.5-5 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b) Each of the lines identified in Table 2.2.5-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.2.5-2 for which functional capability is required meets the requirements for functional capability.
5.a) The seismic Category I equipment identified in Table 2.2.5-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.2.5-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.2.5-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
6.a) Not used.		
6.b) Not used.		

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Table 2.2.5-5 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.	i) Testing will be performed to confirm that the required amount of air flow is delivered to the MCR.	i) The air flow rate from the VES is at least 60 scfm and not more than 70 scfm.
	iii) MCR air samples will be taken during VES testing and analyzed for quality.	iii) The MCR air is of breathable quality.
7.b) The VES maintains the MCR pressure boundary at a positive pressure with respect to the surrounding areas.	i) Testing will be performed with VES flow rate between 60 and 70 scfm to confirm that the MCR is capable of maintaining the required pressurization of the pressure boundary.	i) The MCR pressure boundary is pressurized to greater than or equal to 1/8-in. water gauge with respect to the surrounding area.
	ii) Air leakage into the MCR will be measured during VES testing using a tracer gas.	ii) Air leakage into the MCR is less than or equal to 10 cfm.
7.d) The system provides a passive recirculation flow of MCR air to maintain main control room dose rates below an acceptable level during VES operation.	Testing will be performed to confirm that the required amount of air flow circulates through the MCR passive filtration system.	The air flow rate at the outlet of the MCR passive filtration system is at least 600 cfm greater than the flow measured by VES-003A/B.
8. Safety-related displays identified in Table 2.2.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.5-1 can be retrieved in the MCR.
9.a) Controls exist in the MCR to cause remotely operated valves identified in Table 2.2.5-1 to perform their active functions.	Stroke testing will be performed on remotely operated valves identified in Table 2.2.5-1 using the controls in the MCR.	Controls in the MCR operate to cause remotely operated valves identified in Table 2.2.5-1 to perform their active safety functions.
9.b) The valves identified in Table 2.2.5-1 as having PMS control perform their active safety function after receiving a signal from the PMS.	Testing will be performed on remotely operated valves listed in Table 2.2.5-1 using real or simulated signals into the PMS.	The remotely operated valves identified in Table 2.2.5-1 as having PMS control perform the active safety function identified in the table after receiving a signal from the PMS.
10. After loss of motive power, the remotely operated valves identified in Table 2.2.5-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.5-1 assumes the indicated loss of motive power position.

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Table 2.2.5-5 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. Displays of the parameters identified in Table 2.2.5-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.2.5-3 can be retrieved in the MCR.
12. The background noise level in the MCR does not exceed 65 dB(A) at the operator workstations when VES is operating.	The as-built VES will be operated, and background noise levels in the MCR will be measured at the operator work stations with the plant not operating.	The background noise level in the MCR does not exceed 65 dB(A) at the operator work stations when the VES is operating.
7.a) The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.	ii) Analysis of storage capacity will be performed based on manufacturers data.	ii) The calculated storage capacity is greater than or equal to 327,574 scf.
7.c) The heat loads within the MCR, the I&C equipment rooms, and the Class 1E dc equipment rooms are within design basis assumptions to limit the heatup of the rooms identified in Table 2.2.5-4.	An analysis will be performed to determine that the heat loads from as-built equipment within the rooms identified in Table 2.2.5-4 are less than or equal to the design basis assumptions.	A report exists and concludes that: the heat loads within rooms identified in Table 2.2.5-4 are less than or equal to the specified values or that an analysis report exists that concludes: - The temperature and humidity in the MCR remain within limits for reliable human performance for the 72-hour period. - The maximum temperature for the 72-hour period for the I&C rooms is less than or equal to 120°F. - The maximum temperature for the 72-hour period for the Class 1E dc equipment rooms is less than or equal to 120°F.
7.e) Shielding below the VES filter is capable of providing attenuation that is sufficient to ensure main control room doses are below an acceptable level during VES operation.	Inspection will be performed for the existence of a report verifying that the as-built shielding meets the requirements for functional capability.	A report exists and concludes that the as-built shielding identified in Table 2.2.5-1 meets the functional requirements and exists below the filtration unit, and within its vertical projection.
9.c) The MCR Load Shed Panels identified in Table 2.2.5-1 perform their active safety function after receiving a signal from the PMS.	Testing will be performed on the MCR Load Shed Panels listed in Table 2.2.5-1 using real or simulated signals into the PMS.	The MCR Load Shed Panels identified in Table 2.2.5-1 perform their active safety function identified in the table after receiving a signal from the PMS.

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Table 2.2.5-6

Component Name	Tag Number	Component Location
Emergency Air Storage Tank 01	VES-MT-01	Auxiliary Building
Emergency Air Storage Tank 02	VES-MT-02	Auxiliary Building
Emergency Air Storage Tank 03	VES-MT-03	Auxiliary Building
Emergency Air Storage Tank 04	VES-MT-04	Auxiliary Building
Emergency Air Storage Tank 05	VES-MT-05	Auxiliary Building
Emergency Air Storage Tank 06	VES-MT-06	Auxiliary Building
Emergency Air Storage Tank 07	VES-MT-07	Auxiliary Building
Emergency Air Storage Tank 08	VES-MT-08	Auxiliary Building
Emergency Air Storage Tank 09	VES-MT-09	Auxiliary Building
Emergency Air Storage Tank 10	VES-MT-10	Auxiliary Building
Emergency Air Storage Tank 11	VES-MT-11	Auxiliary Building
Emergency Air Storage Tank 12	VES-MT-12	Auxiliary Building
Emergency Air Storage Tank 13	VES-MT-13	Auxiliary Building
Emergency Air Storage Tank 14	VES-MT-14	Auxiliary Building
Emergency Air Storage Tank 15	VES-MT-15	Auxiliary Building
Emergency Air Storage Tank 16	VES-MT-16	Auxiliary Building
Emergency Air Storage Tank 17	VES-MT-17	Auxiliary Building
Emergency Air Storage Tank 18	VES-MT-18	Auxiliary Building
Emergency Air Storage Tank 19	VES-MT-19	Auxiliary Building
Emergency Air Storage Tank 20	VES-MT-20	Auxiliary Building
Emergency Air Storage Tank 21	VES-MT-21	Auxiliary Building
Emergency Air Storage Tank 22	VES-MT-22	Auxiliary Building
Emergency Air Storage Tank 23	VES-MT-23	Auxiliary Building
Emergency Air Storage Tank 24	VES-MT-24	Auxiliary Building
Emergency Air Storage Tank 25	VES-MT-25	Auxiliary Building
Emergency Air Storage Tank 26	VES-MT-26	Auxiliary Building
Emergency Air Storage Tank 27	VES-MT-27	Auxiliary Building
Emergency Air Storage Tank 28	VES-MT-28	Auxiliary Building

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Table 2.2.5-6 (cont.)

Component Name	Tag Number	Component Location
Emergency Air Storage Tank 29	VES-MT-29	Auxiliary Building
Emergency Air Storage Tank 30	VES-MT-30	Auxiliary Building
Emergency Air Storage Tank 31	VES-MT-31	Auxiliary Building
Emergency Air Storage Tank 32	VES-MT-32	Auxiliary Building

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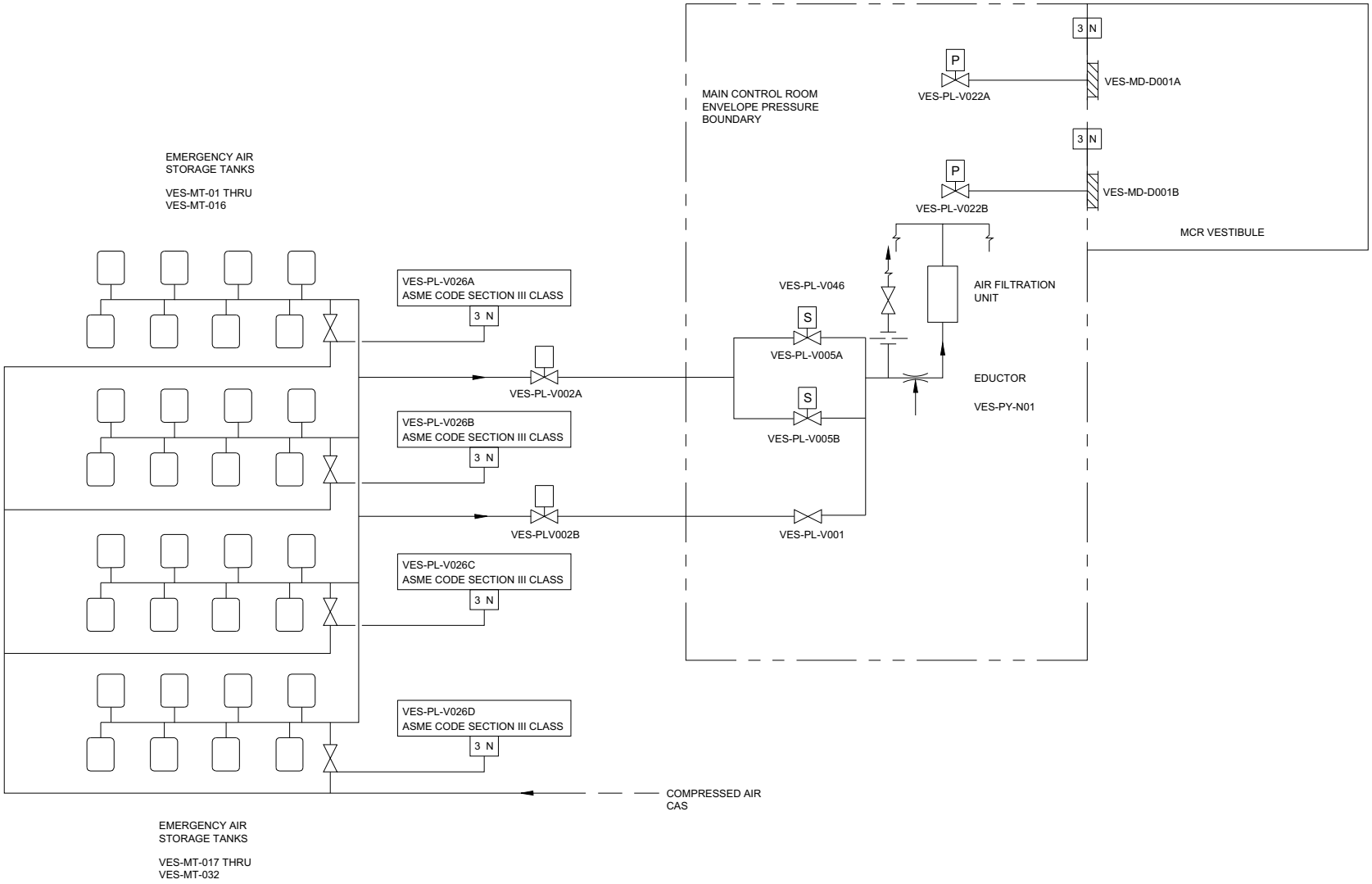

Figure 2.2.5-1
Main Control Room Emergency Habitability System

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2.3 Auxiliary Systems

2.3.1 Component Cooling Water System

Design Description

The component cooling water system (CCS) removes heat from various plant components and transfers this heat to the service water system (SWS) during normal modes of plant operation including power generation, shutdown and refueling. The CCS has two pumps and two heat exchangers.

The CCS is as shown in Figure 2.3.1-1 and the CCS component locations are as shown in Table 2.3.1-3.

1. The functional arrangement of the CCS is as described in the Design Description of this Section 2.3.1.

2. The CCS preserves containment integrity by isolation of the CCS lines penetrating the containment.

3. The CCS provides the nonsafety-related functions of transferring heat from the normal residual heat removal system (RNS) during shutdown and the spent fuel pool cooling system during all modes of operation to the SWS.

4. Controls exist in the main control room (MCR) to cause the pumps identified in Table 2.3.1-1 to perform the listed functions.

5. Displays of the parameters identified in Table 2.3.1-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.1-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the CCS.

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Table 2.3.1-1

Equipment Name	Tag No.	Display	Control Function
CCS Pump A	CCS-MP-01A	Yes (Run Status)	Start
CCS Pump B	CCS-MP-01B	Yes (Run Status)	Start
CCS Discharge Header Flow Sensor	CCS-101	Yes	-
CCS to Normal Residual Heat Removal System Heat Exchanger (RNS HX) A Flow Sensor	CCS-301	Yes	-
CCS to RNS HX B Flow Sensor	CCS-302	Yes	-
CCS to Spent Fuel Pool Cooling System (SFS) HX A Flow Sensor	CCS-341	Yes	-
CCS to SFS HX B Flow Sensor	CCS-342	Yes	-
CCS Surge Tank Level Sensor A	CCS-130	Yes	-
CCS Surge Tank Level Sensor B	CCS-131	Yes	-
CCS Heat Exchanger Inlet Temperature Sensor	CCS-121	Yes	-
CCS Heat Exchanger Outlet Temperature Sensor	CCS-122	Yes	-
CCS Flow to Reactor Coolant Pump (RCP) 1A Valve (Position Indicator)	CCS-PL-V256A	Yes	-
CCS Flow to RCP 1B Valve (Position Indicator)	CCS-PL-V256B	Yes	-
CCS Flow to RCP 2A Valve (Position Indicator)	CCS-PL-V256C	Yes	-
CCS Flow to RCP 2B Valve (Position Indicator)	CCS-PL-V256D	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.1-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. Not used.		
3. The CCS provides the nonsafety-related functions of transferring heat from the RNS during shutdown and the spent fuel pool cooling system during all modes of operation to the SWS.	i) Inspection will be performed for the existence of a report that determines the heat transfer capability of the CCS heat exchangers.	i) A report exists and concludes that the UA of each CCS heat exchanger is greater than or equal to 14.0 million Btu/hr-°F.
3. The CCS provides the nonsafety-related functions of transferring heat from the RNS during shutdown and the spent fuel pool cooling system during all modes of operation to the SWS.	ii) Testing will be performed to confirm that the CCS can provide cooling water to the RNS HXs while providing cooling water to the SFS HXs.	ii) Each pump of the CCS can provide at least 2685 gpm of cooling water to one RNS HX and at least 1200 gpm of cooling water to one SFS HX while providing at least 4415 gpm to other users of cooling water.
4. Controls exist in the MCR to cause the pumps identified in Table 2.3.1-1 to perform the listed functions.	Testing will be performed to actuate the pumps identified in Table 2.3.1-1 using controls in the MCR.	Controls in the MCR operate to cause pumps listed in Table 2.3.1-1 to perform the listed functions.
5. Displays of the parameters identified in Table 2.3.1-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	Displays identified in Table 2.3.1-1 can be retrieved in the MCR.

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Table 2.3.1-3

Component Name	Tag No.	Component Location
CCS Pump A	CCS-MP-01A	Turbine Building
CCS Pump B	CCS-MP-01B	Turbine Building
CCS Heat Exchanger A	CCS-ME-01A	Turbine Building
CCS Heat Exchanger B	CCS-ME-01B	Turbine Building

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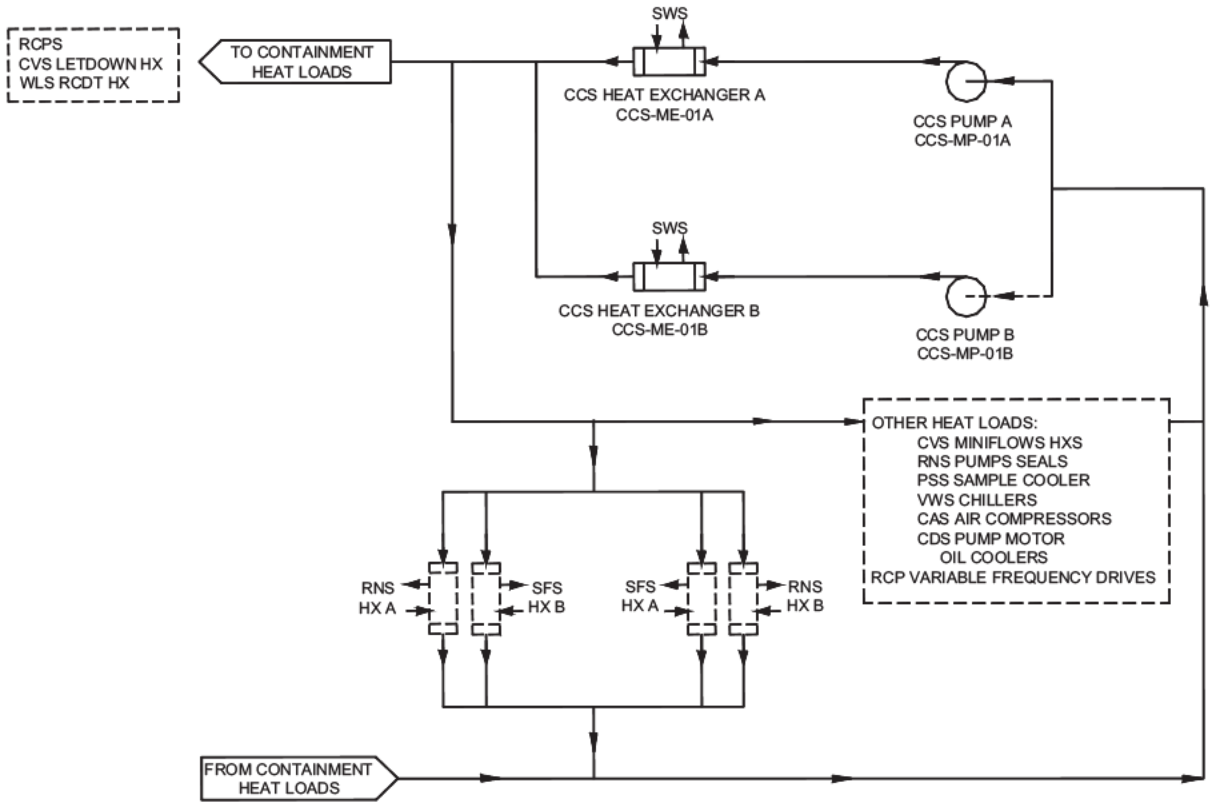

Figure 2.3.1-1
Component Cooling Water System

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2.3.2 Chemical and Volume Control System

Design Description

The chemical and volume control system (CVS) provides reactor coolant system (RCS) purification, RCS inventory control and makeup, chemical shim and chemical control, oxygen control, and auxiliary pressurizer spray. The CVS performs these functions during normal modes of operation including power generation and shutdown.

The CVS is as shown in Figure 2.3.2-1 and the component locations of the CVS are as shown in Table 2.3.2-5.

1. The functional arrangement of the CVS is as described in the Design Description of this Section 2.3.2.

2. a) The components identified in Table 2.3.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.3.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.3.2-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.3.2-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.3.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.3.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. The seismic Category I equipment identified in Table 2.3.2-1 can withstand seismic design basis loads without loss of safety function.

6. a) The Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.3.2-1 are powered from their respective Class 1E division.

   c) Separation is provided between CVS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

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7. The CVS provides the following safety-related functions:

   a) The CVS preserves containment integrity by isolation of the CVS lines penetrating the containment.

   b) The CVS provides termination of an inadvertent RCS boron dilution by isolating demineralized water from the RCS.

   c) The CVS provides isolation of makeup to the RCS.

8. The CVS provides the following nonsafety-related functions:

   a) The CVS provides makeup water to the RCS.

   b) The CVS provides the pressurizer auxiliary spray.

9. Safety-related displays in Table 2.3.2-1 can be retrieved in the main control room (MCR).

10. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.

    b) The valves identified in Table 2.3.2-1 as having protection and safety monitoring system (PMS) control perform an active safety function after receiving a signal from the PMS.

11. a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.

    b) After a loss of motive power, the remotely operated valves identified in Table 2.3.2-1 assume the indicated loss of motive power position.

12. a) Controls exist in the MCR to cause the pumps identified in Table 2.3.2-3 to perform the listed function.

    b) The pumps identified in Table 2.3.2-3 start after receiving a signal from the PLS.

13. Displays of the parameters identified in Table 2.3.2-3 can be retrieved in the MCR.

14. The nonsafety-related piping located inside containment and designated as reactor coolant pressure boundary, as identified in Table 2.3.2-2 (pipe lines with "No" in the ASME Code column), has been designed to withstand a seismic design basis event and maintain structural integrity.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.2-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the CVS.

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Table 2.3.2-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
RCS Purification Motor-operated Isolation Valve	CVS-PL-V001	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Purification Motor-operated Isolation Valve	CVS-PL-V002	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Purification Motor-operated Isolation Valve	CVS-PL-V003	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
CVS Resin Flush Line Containment Isolation Valve	CVS-PL-V040	Yes	Yes	No	- / -	-	-	-	-
CVS Resin Flush Line Containment Isolation Valve	CVS-PL-V041	Yes	Yes	No	- / -	-	-	-	-
CVS Demineralizer Resin Flush Line Containment Isolation Thermal Relief Valve	CVS-PL-V042	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Letdown Containment Isolation Valve	CVS-PL-V045	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Letdown Containment Isolation Valve	CVS-PL-V047	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

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Table 2.3.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
CVS Letdown Line Containment Isolation Thermal Relief Valve	CVS-PL-V058	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Makeup Return Line Bypass Check Valve	CVS-PL-V067	Yes	Yes	No	- / -	-	-	Transfer Open Transfer Closed	-
CVS Purification Return Line Pressure Boundary Check Valve	CVS-PL-V080	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Purification Return Line Pressure Boundary Isolation Check Valve	CVS-PL-V081	Yes	Yes	No	- / -	No	-	Transfer Closed	-
CVS Purification Return Line Pressure Boundary Check Valve	CVS-PL-V082	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Auxiliary Pressurizer Spray Line Pressure Boundary Valve	CVS-PL-V084	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Auxiliary Pressurizer Spray Line Pressure Boundary Check Valve	CVS-PL-V085	Yes	Yes	No	- / -	-	-	Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.3.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
CVS Makeup Line Containment Isolation Motor-operated Valve	CVS-PL-V090	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	As Is
CVS Makeup Line Containment Isolation Motor-operated Valve	CVS-PL-V091	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
CVS Zinc Injection Containment Isolation Valve ORC	CVS-PL-V092	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Zinc Injection Containment Isolation Valve IRC	CVS-PL-V094	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Zinc Addition Line Ctmt Isol Thermal Relief Valve	CVS-PL-V098	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Makeup Line Containment Isolation Thermal Relief Valve	CVS-PL-V100	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Demineralized Water Isolation Valve	CVS-PL-V136A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

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Table 2.3.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
CVS Demineralized Water Isolation Valve	CVS-PL-V136B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Hydrogen Injection Containment Isolation Check Valve IRC	CVS-PL-V217	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Hydrogen Injection Containment Isolation Valve ORC	CVS-PL-V219	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

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Table 2.3.2-2

Line Name	Line Number	ASME Code Section III
CVS Purification Line	L001 L040	Yes Yes
CVS Resin Flush Containment Penetration Line	L026	Yes
CVS Purification Line Return	L038	Yes
CVS Pressurizer Auxiliary Spray Connection	L070 L071	Yes Yes
CVS Letdown Containment Penetration Line	L051	Yes
CVS Makeup Containment Penetration Line	L053	Yes
CVS Hydrogen Injection Containment Penetration Line	L213 L214 L217	Yes
CVS Zinc Injection Containment Penetration Line	L061	Yes
CVS Supply Line to Regenerative Heat Exchanger	L002	No
CVS Return Line from Regenerative Heat Exchanger	L018 L036	No Yes
CVS Line from Regenerative Heat Exchanger to Letdown Heat Exchanger	L003	No
CVS Lines from Letdown Heat Exchanger to Demin. Tanks	L004 L005 L072	No No No
CVS Lines from Demin Tanks to RC Filters and Connected Lines	L006(1) L007(1) L010(1) L011(1) L012 L015(1) L016(1) L020 L021 L022 L023(1) L024(1) L029 L037	No No No No No No No No No No No No No No

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Table 2.3.2-2 (cont.)

Line Name	Line Number	ASME Code Section III
CVS Lines from RC Filters to Regenerative Heat Exchanger	L030 L031 L034 L050 L073	No No No No No
CVS Resin Fill Lines to Demin. Tanks	L008(1) L013(1) L025(1)	No No No

Note:

1. Special seismic requirements include only the portion of piping normally exposed to RCS pressure. Piping beyond the first normally closed isolation valve is evaluated as seismic Category II piping extending to either an interface anchor, a rigid support following a six-way anchor, or the last seismic support of a rigidly supported region of the piping system as necessary to satisfy analysis requirements for piping connected to seismic Category I piping systems.

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Table 2.3.2-3

Equipment	Tag No.	Display	Control Function
CVS Makeup Pump A	CVS-MP-01A	Yes (Run Status)	Start
CVS Makeup Pump B	CVS-MP-01B	Yes (Run Status)	Start
Purification Flow Sensor	CVS-001	Yes	-
Purification Return Flow Sensor	CVS-025	Yes	-
CVS Purification Return Line (Position Indicator)	CVS-PL-V081	Yes	-
Auxiliary Spray Line Isolation Valve (Position Indicator)	CVS-PL-V084	Yes	-
Boric Acid Storage Tank Level Sensor	CVS-109	Yes	-
Boric Acid Flow Sensor	CVS-115	Yes	-
Makeup Blend Valve (Position Indicator)	CVS-PL-V115	Yes	-
CVS Demineralized Water Isolation Valve (Position Indicator)	CVS-PL-V136A	Yes	-
CVS Demineralized Water Isolation Valve (Position Indicator)	CVS-PL-V136B	Yes	-
Makeup Pump Discharge Flow Sensor	CVS-157	Yes	-
Makeup Flow Control Valve (Position Indicator)	CVS-PL-V157	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.2-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) The components identified in Table 2.3.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.3.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.3.2-1 and 2.3.2-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.3.2-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.3.2-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.3.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The piping identified in Table 2.3.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the components and piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components and piping identified in Tables 2.3.2-1 and 2.3.2-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

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Table 2.3.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The seismic Category I equipment identified in Table 2.3.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.2-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.2-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
6.a) The Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.3.2-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
6.b) Not used.		
6.c) Not used.		
7.a) Not used.		
7.b) Not used.		
7.c) Not used.		

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Table 2.3.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.a) The CVS provides makeup water to the RCS.	i) Testing will be performed by aligning a flow path from each CVS makeup pump, actuating makeup flow to the RCS at pressure greater than or equal to 2000 psia, and measuring the flow rate in the makeup pump discharge line with each pump suction aligned to the boric acid storage tank.	i) Each CVS makeup pump provides a flow rate of greater than or equal to 100 gpm.
8.b) The CVS provides the pressurizer auxiliary spray.	Testing will be performed by aligning a flow path from each CVS makeup pump to the pressurizer auxiliary spray and measuring the flow rate in the makeup pump discharge line with each pump suction aligned to the boric acid storage tank and with RCS pressure greater than or equal to 2000 psia.	Each CVS makeup pump provides spray flow to the pressurizer.
9. Safety-related displays identified in Table 2.3.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.2-1 can be retrieved in the MCR.
10.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.3.2-1 using the controls in the MCR.	Controls in the MCR operate to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.
10.b) The valves identified in Table 2.3.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	i) Testing will be performed using real or simulated signals into the PMS.	i) The valves identified in Table 2.3.2-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
	ii) Testing will be performed to demonstrate that the remotely operated CVS isolation valves CVS-V090, V091, V136A/B close within the required response time.	ii) These valves close within the following times after receipt of an actuation signal: V090, V091 < 30 sec V136A/B < 20 sec

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Table 2.3.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11.a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under pre-operational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.3.2-1 under pre-operational test conditions.
	iv) Exercise testing of the check valves with active safety functions identified in Table 2.3.2-1 will be performed under pre-operational test pressure, temperature and fluid flow conditions.	iv) Each check valve changes position as indicated in Table 2.3.2-1.
11.b) After loss of motive power, the remotely operated valves identified in Table 2.3.2-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valve identified in Table 2.3.2-1 assumes the indicated loss of motive power position.
12.a) Controls exist in the MCR to cause the pumps identified in Table 2.3.2-3 to perform the listed function.	Testing will be performed to actuate the pumps identified in Table 2.3.2-3 using controls in the MCR.	Controls in the MCR cause pumps identified in Table 2.3.2-3 to perform the listed function.
12.b) The pumps identified in Table 2.3.2-3 start after receiving a signal from the PLS.	Testing will be performed to confirm starting of the pumps identified in Table 2.3.2-3.	The pumps identified in Table 2.3.2-3 start after a signal is generated by the PLS.
13. Displays of the parameters identified in Table 2.3.2-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.3.2-3 in the MCR.	Displays identified in Table 2.3.2-3 can be retrieved in the MCR.
8.a) The CVS provides makeup water to the RCS.	ii) Inspection of the boric acid storage tank volume will be performed.	ii) The volume in the boric acid storage tank is at least 70,000 gallons between the tank suction point and the tank overflow.
	iii) Testing will be performed to measure the delivery rate from the DWS to the RCS. Both CVS makeup pumps will be operating and the RCS pressure will be below 6 psig.	iii) The total CVS makeup flow to the RCS is less than or equal to 175 gpm.

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Table 2.3.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11.a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions. ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tested conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.3.2-1 under design conditions. ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tests or type tests.
14. The nonsafety-related piping located inside containment and designated as reactor coolant pressure boundary, as identified in Table 2.3.2-2, has been designed to withstand a seismic design basis event and maintain structural integrity.	Inspection will be conducted of the as-built piping as documented in the CVS Seismic Analysis Report.	The CVS Seismic Analysis Reports exist for the non-safety related piping located inside containment and designated as reactor coolant pressure boundary as identified in Table 2.3.2-2.

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Table 2.3.2-5

Component Name	Tag No.	Component Location
CVS Makeup Pump A	CVS-MP-01A	Auxiliary Building
CVS Makeup Pump B	CVS-MP-01B	Auxiliary Building
Boric Acid Storage Tank	CVS-MT-01	Yard
Regenerative Heat Exchanger	CVS-ME-01	Containment
Letdown Heat Exchanger	CVS-ME-02	Containment
Mixed Bed Demineralizer A	CVS-MV-01A	Containment
Mixed Bed Demineralizer B	CVS-MV-01B	Containment
Cation Bed Demineralizer	CVS-MV-02	Containment
Reactor Coolant Filter A	CVS-MV-03A	Containment
Reactor Coolant Filter B	CVS-MV-03B	Containment

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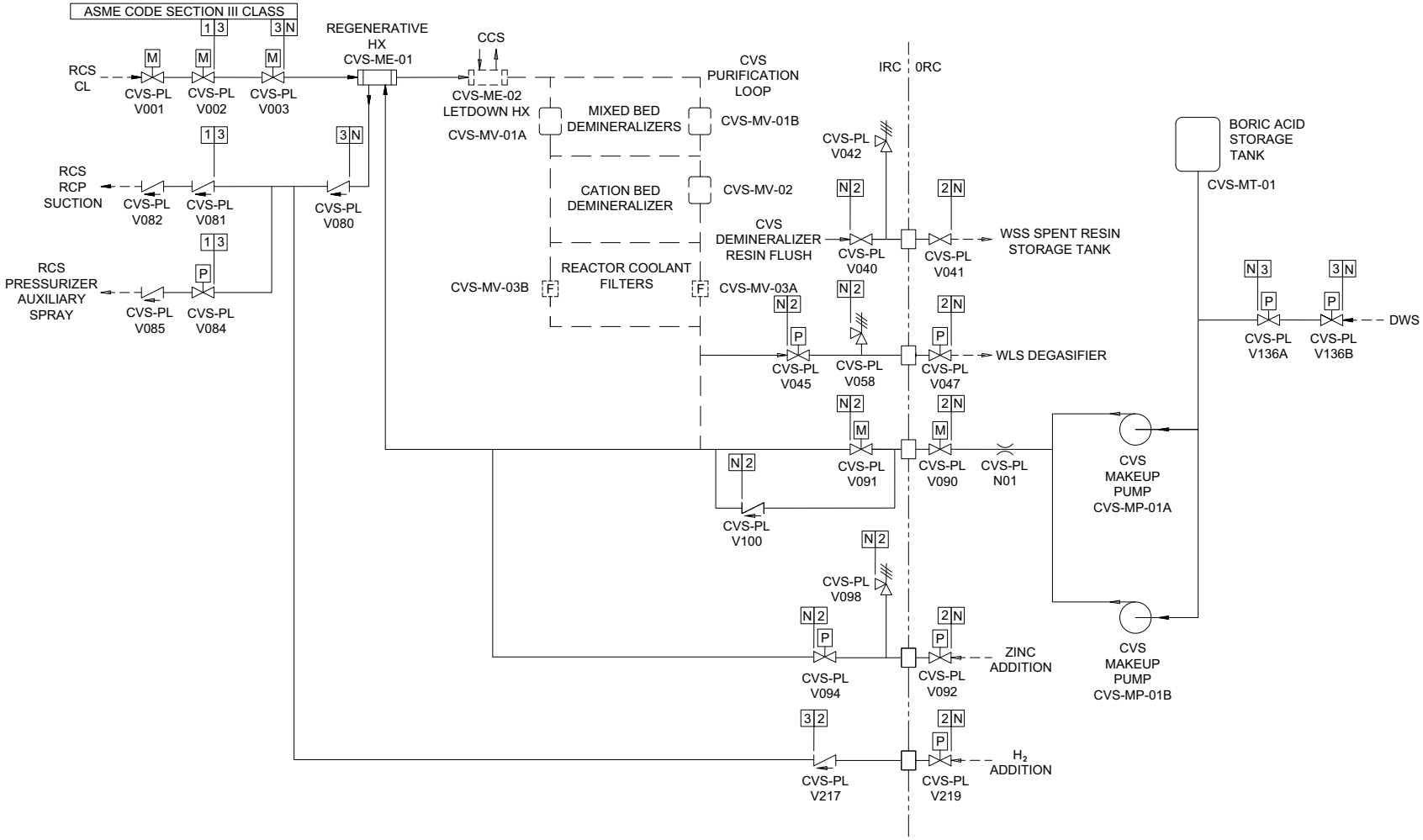

Figure 2.3.2-1
Chemical and Volume Control System

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2.3.3 Standby Diesel Fuel Oil System

Design Description

The standby diesel fuel oil system (DOS) supplies diesel fuel oil for the onsite standby power system. The diesel fuel oil is supplied by two above-ground fuel oil storage tanks. The DOS also provides fuel oil for the ancillary diesel generators. A single fuel oil storage tank services both ancillary diesel generators.

The DOS is as shown in Figure 2.3.3-1 and the component locations of the DOS are as shown in Table 2.3.3-3.

1. The functional arrangement of the DOS is as described in the Design Description of this Section 2.3.3.

2. The ancillary diesel generator fuel tank can withstand a seismic event.

3. The DOS provides the following nonsafety-related functions:

    a) Each fuel oil storage tank provides for at least 7 days of continuous operation of the associated standby diesel generator.

    b) Each fuel oil day tank provides for at least four hours of continuous operation of the associated standby diesel engine generator.

    c) The fuel oil flow rate to the day tank of each standby diesel generator provides for continuous operation of the associated diesel generator.

    d) The ancillary diesel generator fuel tank is sized to supply power to long-term safety-related post-accident monitoring loads and control room lighting through a regulating transformer and one PCS recirculation pump for a period of 4 days.

4. Controls exist in the main control room (MCR) to cause the components identified in Table 2.3.3-1 to perform the listed function.

5. Displays of the parameters identified in Table 2.3.3-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.3-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the DOS.

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Table 2.3.3-1

Equipment Name	Tag No.	Display	Control Function
Diesel Fuel Oil Pump A (Motor)	DOS-MP-01A	Yes (Run Status)	Start
Diesel Fuel Oil Pump B (Motor)	DOS-MP-01B	Yes (Run Status)	Start
Diesel Generator Fuel Oil Day Tank A Level	DOS-016A	Yes	-
Diesel Generator Fuel Oil Day Tank B Level	DOS-016B	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.3-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The ancillary diesel generator fuel tank can withstand a seismic event.	Inspection will be performed for the existence of a report verifying that the as-built ancillary diesel generator fuel tank and its anchorage are designed using seismic Category II methods and criteria.	A report exists and concludes that the as-built ancillary diesel generator fuel tank and its anchorage are designed using seismic Category II methods and criteria.
3.a) Each fuel oil storage tank provides for at least 7 days of continuous operation of the associated standby diesel generator.	Inspection of each fuel oil storage tank will be performed.	The volume of each fuel oil storage tank available to the standby diesel generator is greater than or equal to 55,000 gallons.
3.b) Each fuel oil storage day tank provides for at least 4 hours of operation of the associated standby diesel generator.	Inspection of the fuel oil day tank will be performed.	The volume of each fuel oil day tank is greater than or equal to 1300 gallons.
3.c) The fuel oil flow rate to the day tank of each standby diesel generator provides for continuous operation of the associated diesel generator.	Testing will be performed to determine the flow rate.	The flow rate delivered to each day tank is 8 gpm or greater.
3.d) The ancillary diesel generator fuel tank is sized to supply power to long-term safety-related post accident monitoring loads and control room lighting through a regulating transformer and one PCS recirculation pump for four days.	Inspection of the ancillary diesel generator fuel tank will be performed.	The volume of the ancillary diesel generator fuel tank is greater than or equal to 650 gallons.
4. Controls exist in the MCR to cause the components identified in Table 2.3.3-1 to perform the listed function.	Testing will be performed on the components in Table 2.3.3-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.3.3-1 to perform the listed functions.
5. Displays of the parameters identified in Table 2.3.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.3.3-1 can be retrieved in the MCR.

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Table 2.3.3-3

Component Name	Tag No.	Component Location
Diesel Oil Transfer Package A	DOS-MS-01A	Yard
Diesel Oil Transfer Package B	DOS-MS-01B	Yard
Fuel Oil Storage Tank A	DOS-MT-01A	Yard
Fuel Oil Storage Tank B	DOS-MT-01B	Yard
Diesel Generator A Fuel Oil Day Tank	DOS-MT-02A	Diesel Building
Diesel Generator B Fuel Oil Day Tank	DOS-MT-02B	Diesel Building
Ancillary Diesel Fuel Oil Storage Tank	DOS-MT-03	Annex Building

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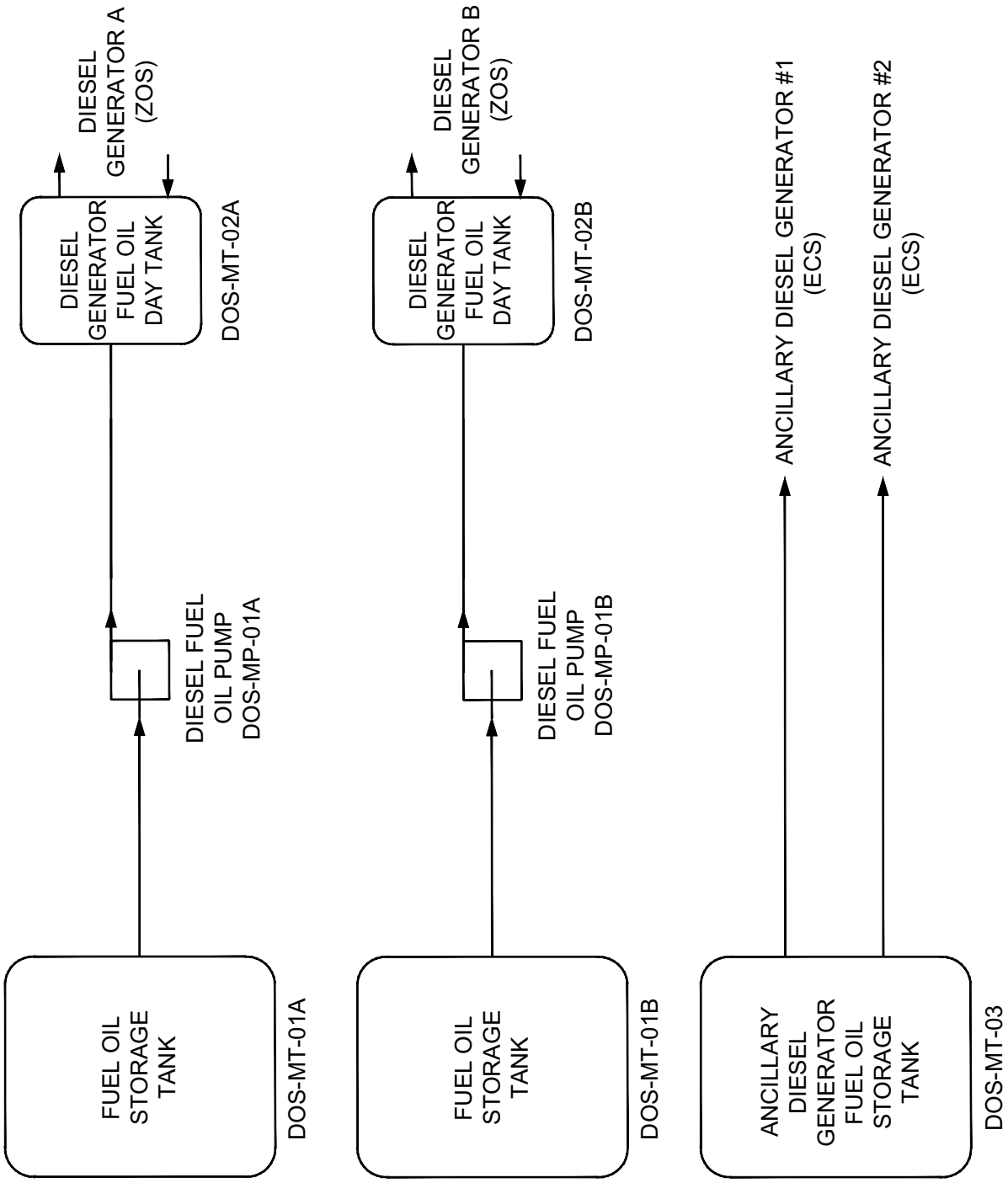

Figure 2.3.3-1
Standby Diesel Fuel Oil System

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2.3.4 Fire Protection System

Design Description

The fire protection system (FPS) detects and suppresses fires in the plant. The FPS consists of water distribution systems, automatic and manual suppression systems, a fire detection and alarm system, and portable fire extinguishers. The FPS provides fire protection for the nuclear island, the annex building, the turbine building, the radwaste building and the diesel generator building.

The FPS is as shown in Figure 2.3.4-1 and the component locations of the FPS are as shown in Table 2.3.4-3.

1. The functional arrangement of the FPS is as described in the Design Description of this Section 2.3.4.

2. The FPS piping shown on Figure 2.3.4-2 remains functional following a safe shutdown earthquake.

3. The FPS provides the safety-related function of preserving containment integrity by isolation of the FPS line penetrating the containment.

4. The FPS provides for manual fire fighting capability in plant areas containing safety-related equipment.

5. Displays of the parameters identified in Table 2.3.4-1 can be retrieved in the main control room (MCR).

6. The FPS provides nonsafety-related containment spray for severe accident management.

7. The FPS provides two fire water storage tanks, each capable of holding at least 100 percent of the water supply necessary for FPS use.

8. Two FPS fire pumps provide at least 2000 gpm each at a total head of at least 350 ft.

9. The fuel tank for the diesel-driven fire pump is capable of holding at least 385 gallons.

10. Individual fire detectors provide fire detection capability and can be used to initiate fire alarms in areas containing safety-related equipment.

11. The FPS seismic standpipe subsystem can be supplied from the FPS fire main by opening the normally closed cross-connect valve to the FPS plant fire main.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.4-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the FPS.

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Table 2.3.4-1

Equipment Name	Tag No.	Display	Control Function
Motor-driven Fire Pump	FPS-MP-01A	Yes (Run Status)	Start
Diesel-driven Fire Pump	FPS-MP-01B	Yes (Run Status)	Start
Jockey Pump	FPS-MP-02	Yes (Run Status)	Start

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Table 2.3.4-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The FPS piping shown on Figure 2.3.4-2 remains functional following a safe shutdown earthquake.	i) Inspection will be performed to verify that the piping shown on Figure 2.3.4-2 is located on the Nuclear Island. ii) A reconciliation analysis using the as-designed and as-built piping information will be performed, or an analysis of the as-built piping will be performed.	i) The piping shown on Figure 2.3.4-2 is located on the Nuclear Island. ii) The as-built piping stress report exists and concludes that the piping remains functional following a safe shutdown earthquake.
3. Not used.		
4. The FPS provides for manual fire fighting capability in plant areas containing safety-related equipment. 6. The FPS provides nonsafety-related containment spray for severe accident management. 7. The FPS provides two fire water storage tanks, each capable of holding at least 100 percent of the water supply necessary for FPS use.	i) Inspection of the passive containment cooling system (PCS) storage tank will be performed. Inspection of the containment spray headers will be performed. Inspection of each fire water storage tank will be performed.	i) The volume of the PCS tank above the standpipe feeding the FPS and below the overflow is at least 18,000 gal. The FPS has spray headers and nozzles as follows: At least 44 nozzles at plant elevation of at least 260 feet, and 24 nozzles at plant elevation of at least 275 feet. The volume of water dedicated to FPS use provided in each fire water storage tank is at least 396,000 gallons.
4. The FPS provides for manual fire fighting capability in plant areas containing safety-related equipment.	ii) Testing will be performed by measuring the water flow rate as it is simultaneously discharged from the two highest fire-hose stations and when the water for the fire is supplied from the PCS storage tank.	ii) Water is simultaneously discharged from each of the two highest fire-hose stations in plant areas containing safety-related equipment at not less than 75 gpm.

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Table 2.3.4-2 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Displays of the parameters identified in Table 2.3.4-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.3.4-1 can be retrieved in the MCR.
8. Two FPS fire pumps provide at least 2000 gpm each at a total head of at least 350 ft.	Testing and/or analysis of each fire pump will be performed.	The tests and/or analysis concludes that each fire pump provides a flow rate of at least 2000 gpm at a total head of at least 350 ft.
9. The fuel tank for the diesel-driven fire pump is capable of holding at least 385 gallons.	Inspection of the diesel-driven fire pump fuel tank will be performed.	The volume of the diesel driven fire pump fuel tank is at least 385 gallons.
10. Individual fire detectors provide fire detection capability and can be used to initiate fire alarms in areas containing safety-related equipment.	Testing will be performed on the as-built individual fire detectors in the fire areas identified in Tier 1 Material, subsection 3.3, Table 3.3-3. (Individual fire detectors will be tested using simulated fire conditions.)	The tested individual fire detectors respond to simulated fire conditions.
11. The FPS seismic standpipe subsystem can be supplied from the FPS fire main by opening the normally closed cross-connect valve to the FPS plant fire main.	Inspection for the existence of a cross-connect valve from the FPS seismic standpipe subsystem to FPS plant fire main will be performed.	Valve FPS-PL-V101 exists and can connect the FPS seismic standpipe subsystem to the FPS plant fire main.

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Table 2.3.4-3

Component Name	Tag No.	Location
Motor-driven Fire Pump	FPS-MP-01A	Yard
Diesel-driven Fire Pump	FPS-MP-01B	Yard
Jockey Pump	FPS-MP-02	Yard
Primary Fire Water Tank	FPS-MT-01A	Yard
Secondary Fire Water Tank	FPS-MT-01B	Yard
Fire Pump Diesel Fuel Day Tank	FPS-MT-02	Yard

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Table 2.3.4-4 is not used.

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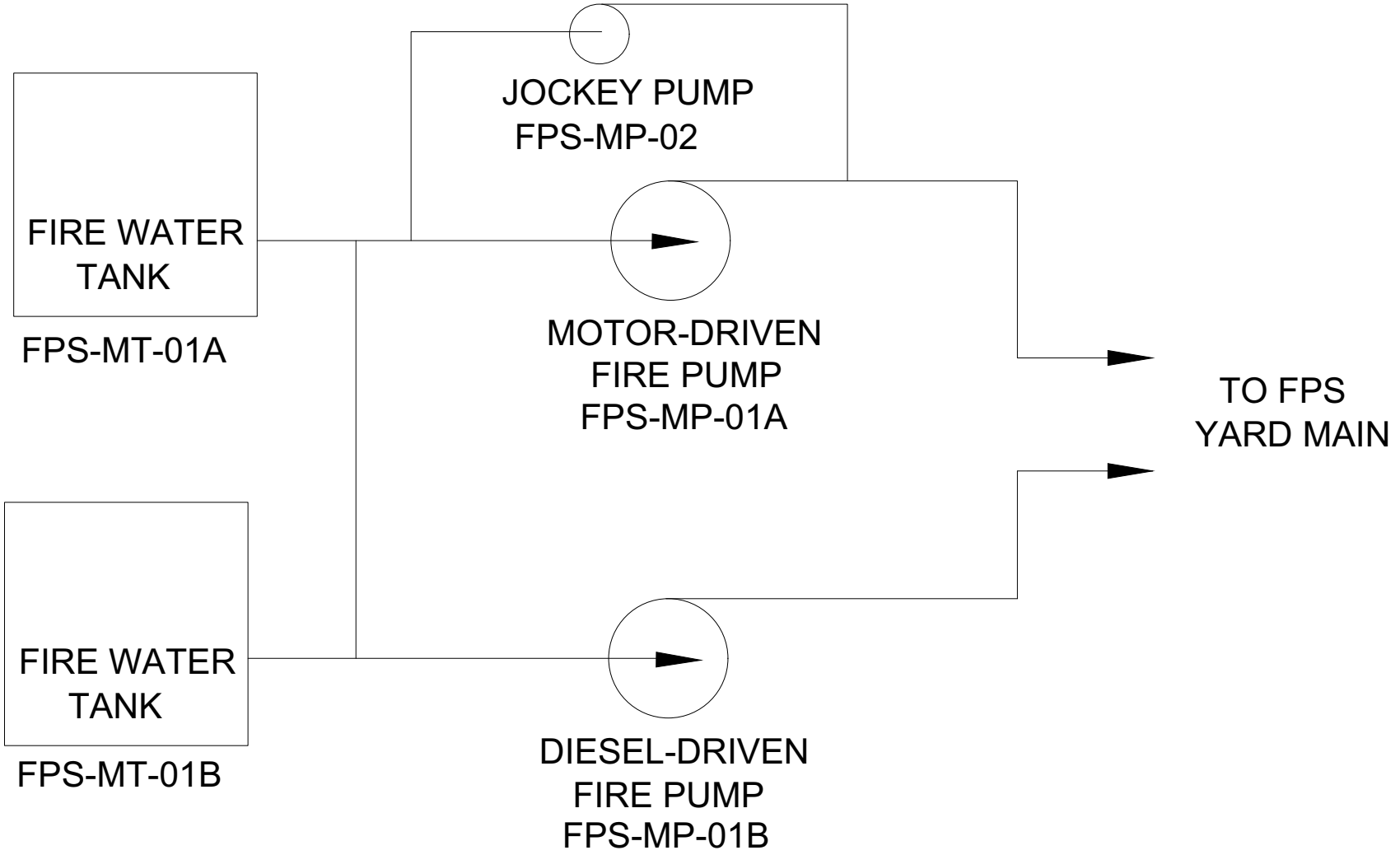

Figure 2.3.4-1 (Sheet 1 of 2)
Fire Protection System

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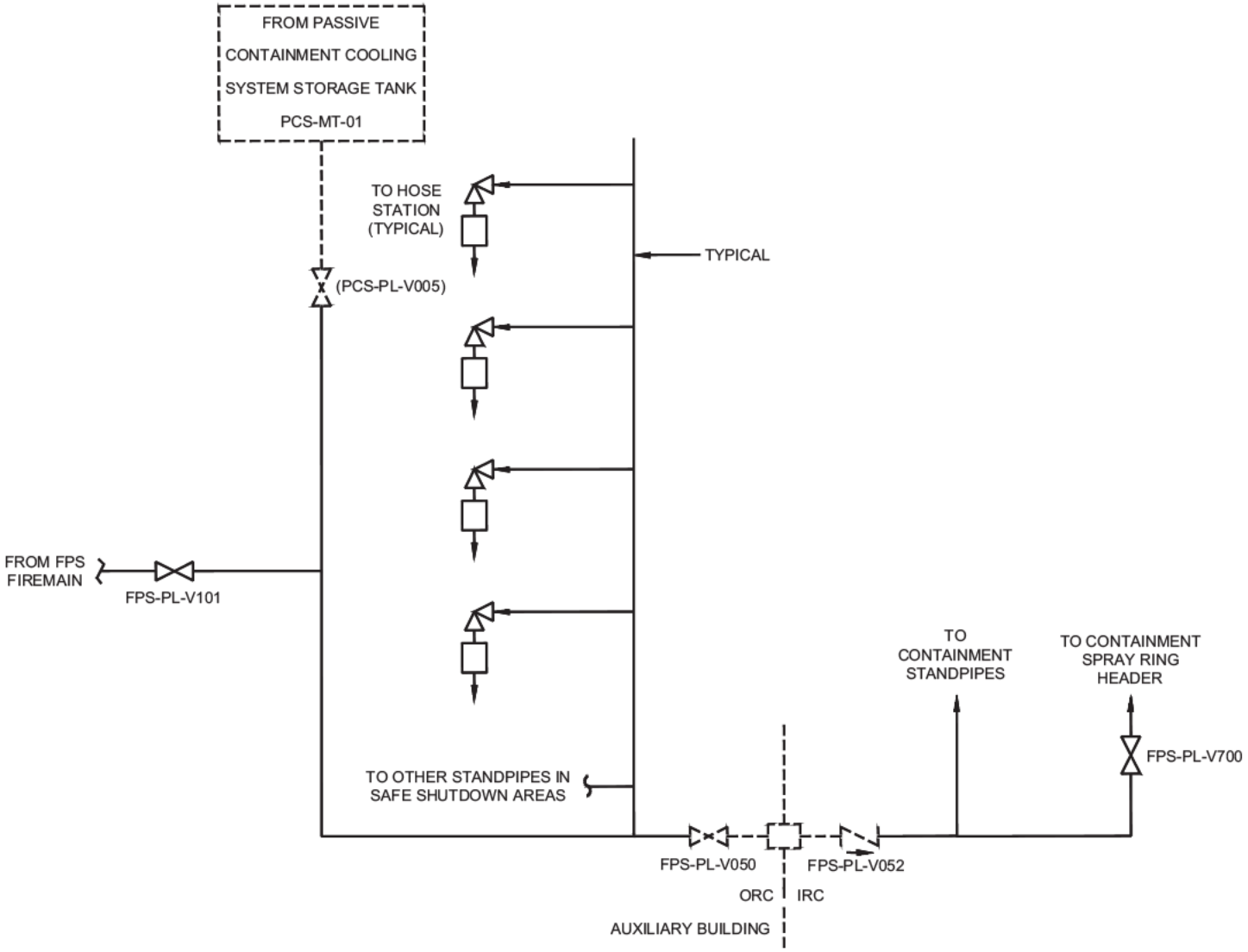

Figure 2.3.4-1 (Sheet 2 of 2)
Fire Protection System

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NOTE:
ALL VALVES ARE FPS UNLESS OTHERWISE IDENTIFIED.

PIPING LOCATED WITHIN THE BOUNDARY VALVES OF THIS FIGURE, INCLUDING THE PIPING DOWNSTREAM OF VALVE V301, ARE DESIGNED TO REMAIN FUNCTIONAL FOLLOWING A SAFE SHUTDOWN EARTHQUAKE. THIS INCLUDES ALL RCF AND JCB CLASS FPS PIPING IN CONTAINMENT AND THE NON RADIOLOGICAL AREA OF THE AUXILIARY BUILDING. DASHED LINES ARE RCF CLASS PCS PIPING.

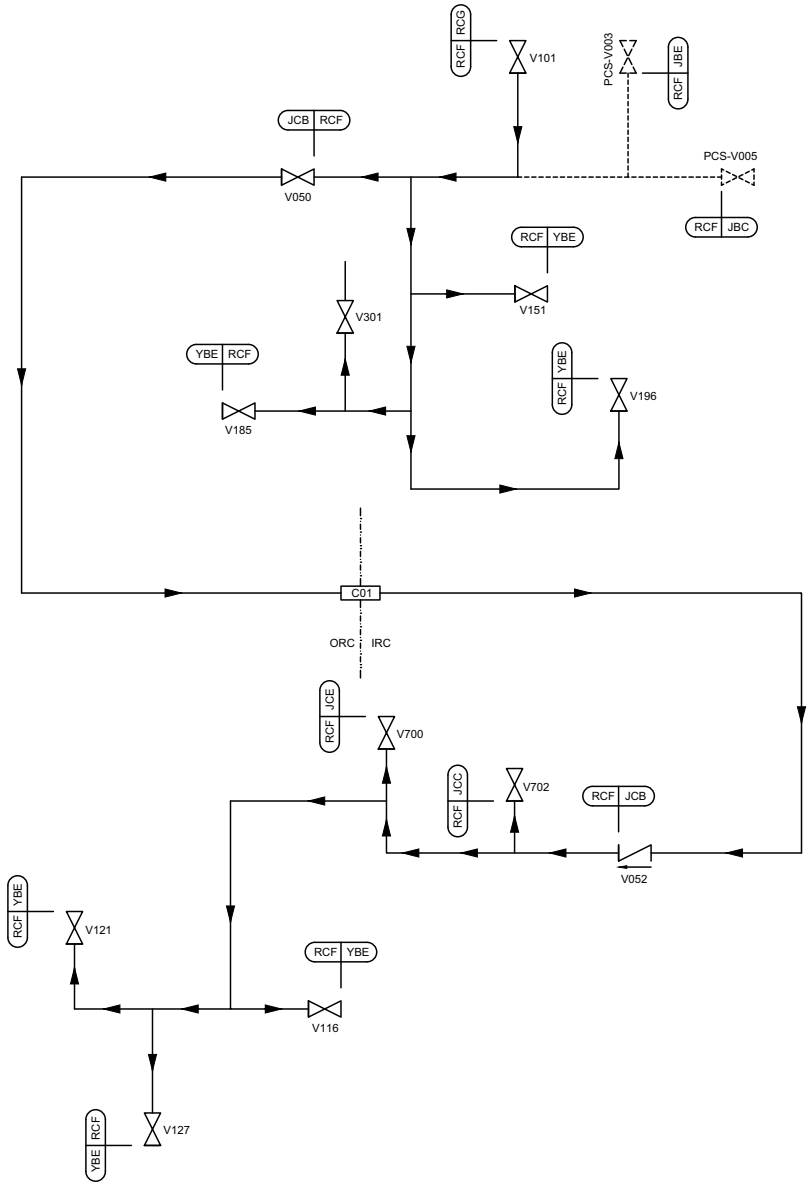

Figure 2.3.4-2
Boundaries of Fire Protection System That Must Remain Functional Following a Safe Shutdown Earthquake

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2.3.5 Mechanical Handling System

Design Description

The mechanical handling system (MHS) provides for lifting heavy loads. The MHS equipment can be operated during shutdown and refueling.

The component locations of the MHS are as shown in Table 2.3.5-3.

1. The functional arrangement of the MHS is as described in the Design Description of this Section 2.3.5.

2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.

3. The MHS components listed below are single failure proof:

    a) Polar crane
    b) Cask handling crane
    c) Equipment hatch hoist
    d) Maintenance hatch hoist

4. The cask handling crane cannot move over the spent fuel pool.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.5-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the MHS.

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Table 2.3.5-1

Equipment Name	Tag No.	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety Function
Containment Polar Crane	MHS-MH-01	Yes	No/No	Avoid uncontrolled lowering of heavy load.
Cask Handling Crane	MHS-MH-02	Yes	No/No	Avoid uncontrolled lowering of heavy load.
Equipment Hatch Hoist	MHS-MH-05	Yes	No/No	Avoid uncontrolled lowering of heavy load.
Maintenance Hatch Hoist	MHS-MH-06	Yes	No/No	Avoid uncontrolled lowering of heavy load.

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Table 2.3.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.5-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.5-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
3.a) The polar crane is single failure proof.	i) Validation of double design factors is provided for hooks where used as load bearing components. Validation of redundant factors is provided for load bearing components such as: • Hoisting ropes • Sheaves • Equalizer assembly • Holding brakes	i) A report exists and concludes that the polar crane is single failure proof. A certificate of conformance from the vendor exists and concludes that the polar crane is single failure proof.
	ii) Testing of the polar crane is performed.	ii) The polar crane shall be static-load tested to 125% of the rated load.
	iii) Testing of the polar crane is performed.	iii) The polar crane shall lift a test load that is 100% of the rated load. Then it shall lower, stop, and hold the test load.

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Table 2.3.5-2 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3.b) The cask handling crane is single failure proof.	i) Validation of double design factors is provided for hooks where used as load bearing components. Validation of redundant factors is provided for load bearing components such as: • Hoisting ropes • Sheaves • Equalizer assembly • Holding brakes	i) A report exists and concludes that the cask handling crane is single failure proof. A certificate of conformance from the vendor exists and concludes that the cask handling crane is single failure proof.
3.b) The cask handling crane is single failure proof.	ii) Testing of the cask handling crane is performed.	ii) The cask handling crane shall be static load tested to 125% of the rated load.
	iii) Testing of the cask handling crane is performed.	iii) The cask handling crane shall lift a test load that is 100% of the rated load. Then it shall lower, stop, and hold the test load.
4. The cask handling crane cannot move over the spent fuel pool.	Testing of the cask handling crane is performed.	The cask handling crane does not move over the spent fuel pool.
3.c) The equipment hatch hoist is single failure proof.	i) Validation of double design factors is provided for hooks where used as load bearing components. Validation of redundant factors is provided for load bearing components such as: • Hoisting ropes • Sheaves • Equalizer assembly • Holding brakes	i) A report exists and concludes that the equipment hatch hoist is single failure proof. A certificate of conformance from the vendor exists and concludes that the equipment hatch hoist is single failure proof.
	ii) Testing of the equipment hatch hoist is performed.	ii) The equipment hatch hoist holding mechanism shall stop and hold the hatch.

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Table 2.3.5-2 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3.d) The maintenance hatch hoist is single failure proof.	i) Validation of double design factors is provided for hooks where used as load bearing components. Validation of redundant factors is provided for load bearing components such as: • Hoisting ropes • Sheaves • Equalizer assembly • Holding brakes	i) A report exists and concludes that the maintenance hatch hoist is single failure proof. A certificate of conformance from the vendor exists and concludes that the maintenance hatch hoist is single failure proof.
	ii) Testing of the maintenance hatch hoist is performed.	ii) The maintenance hatch hoist holding mechanism shall stop and hold the hatch.

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Table 2.3.5-3

Component Name	Tag No.	Component Location
Containment Polar Crane	MHS-MH-01	Containment
Cask Handling Crane	MHS-MH-02	Auxiliary Building
Equipment Hatch Hoist	MHS-MH-05	Containment
Maintenance Hatch Hoist	MHS-MH-06	Containment

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2.3.6 Normal Residual Heat Removal System

Design Description

The normal residual heat removal system (RNS) removes heat from the core and reactor coolant system (RCS) and provides RCS low temperature over-pressure (LTOP) protection at reduced RCS pressure and temperature conditions after shutdown. The RNS also provides a means for cooling the in-containment refueling water storage tank (IRWST) during normal plant operation.

The RNS is as shown in Figure 2.3.6-1 and the RNS component locations are as shown in Table 2.3.6-5.

1. The functional arrangement of the RNS is as described in the Design Description of this Section 2.3.6.

2. a) The components identified in Table 2.3.6-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.3.6-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.3.6-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.3.6-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.3.6-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.3.6-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. a) The seismic Category I equipment identified in Table 2.3.6-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.3.6-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

6. Each of the as-built lines identified in Table 2.3.6-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

7. a) The Class 1E equipment identified in Table 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.3.6-1 are powered from their respective Class 1E division.

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c) Separation is provided between RNS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

8. The RNS provides the following safety-related functions:

a) The RNS preserves containment integrity by isolation of the RNS lines penetrating the containment.

b) The RNS provides a flow path for long-term, post-accident makeup to the RCS.

9. The RNS provides the following nonsafety-related functions:

a) The RNS provides low temperature overpressure protection (LTOP) for the RCS during shutdown operations.

b) The RNS provides heat removal from the reactor coolant during shutdown operations.

c) The RNS provides low pressure makeup flow from the SFS cask loading pit to the RCS for scenarios following actuation of the automatic depressurization system (ADS).

d) The RNS provides heat removal from the in-containment refueling water storage tank.

10. Safety-related displays identified in Table 2.3.6-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.3.6-1 to perform active functions.

b) The valves identified in Table 2.3.6-1 as having protection and safety monitoring system (PMS) control perform active safety functions after receiving a signal from the PMS.

12. a) The motor-operated and check valves identified in Table 2.3.6-1 perform an active safety-related function to change position as indicated in the table.

b) After loss of motive power, the remotely operated valves identified in Table 2.3.6-1 assume the indicated loss of motive power position.

13. Controls exist in the MCR to cause the pumps identified in Table 2.3.6-3 to perform the control function.

14. Displays of the RNS parameters identified in Table 2.3.6-3 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.6-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the RNS.

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Table 2.3.6-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
RNS Pump A (Pressure Boundary)	RNS-MP-01A	Yes	Yes	-	-/-	-	-	No	-
RNS Pump B (Pressure Boundary)	RNS-MP-01B	Yes	Yes	-	-/-	-	-	No	-
RNS Heat Exchanger A (Tube Side)	RNS-ME-01A	Yes	Yes	-	-/-	-	-	-	-
RNS Heat Exchanger B (Tube Side)	RNS-ME-01B	Yes	Yes	-	-/-	-	-	-	-
RCS Inner Hot Leg Suction Motor-operated Isolation Valve	RNS-PL-V001A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Inner Hot Leg Suction Motor-operated Isolation Valve	RNS-PL-V001B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Outer Hot Leg Suction Motor-operated Isolation Valve	RNS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Outer Hot Leg Suction Motor-operated Isolation Valve	RNS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is

Note: Dash (-) indicates not applicable.

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Table 2.3.6-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
RCS Pressure Boundary Thermal Relief Check Valve	RNS-PL-V003A	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RCS Pressure Boundary Thermal Relief Check Valve	RNS-PL-V003B	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Discharge Motor-operated Containment Isolation Valve	RNS-PL-V011	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	As Is
RNS Discharge Containment Isolation Test Connection	RNS-PL-V012	Yes	Yes	No	-/-	No	No	Transfer Open/ Transfer Closed	-
RNS Discharge Header Containment Isolation Check Valve	RNS-PL-V013	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Discharge RCS Pressure Boundary Check Valve	RNS-PL-V015A	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.3.6-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
RNS Discharge RCS Pressure Boundary Check Valve	RNS-PL-V015B	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Discharge RCS Pressure Boundary Check Valve	RNS-PL-V017A	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Discharge RCS Pressure Boundary Check Valve	RNS-PL-V017B	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Hot Leg Suction Pressure Relief Valve	RNS-PL-V020	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Hot Leg Suction Pressure Relief Valve	RNS-PL-V021	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Suction Header Motor-operated Containment Isolation Valve	RNS-PL-V022	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	As Is
RNS Suction from IRWST Motor-operated Isolation Valve	RNS-PL-V023	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RNS Discharge to IRWST Motor-operated Isolation Valve	RNS-PL-V024	Yes	Yes	Yes	-/-	No	No	No	As Is

Note: Dash (-) indicates not applicable.

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Table 2.3.6-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
RNS Pump Discharge Relief	RNS-PL-V045	Yes	Yes	No	-/-	No	-	No	-
RNS Suction from Cask Loading Pit Motor-operated Isolation Valve	RNS-PL-V055	Yes	Yes	Yes	No/No	No	No	No	As Is
RNS Suction from Cask Loading Pit Check Valve	RNS-PL-V056	Yes	Yes	No	-/-	No	-	No	-
RNS Pump Miniflow Air-Operated Isolation Valve	RNS-PL-V057A	Yes	Yes	Yes	No/No	No	No	No	Open
RNS Pump Miniflow Air-Operated Isolation Valve	RNS-PL-V057B	Yes	Yes	Yes	No/No	No	No	No	Open
RNS Return from Chemical and Volume Control System (CVS) Containment Isolation Valve	RNS-PL-V061	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

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Table 2.3.6-2

Line Name	Line No.	ASME Code Section III	Leak Before Break	Functional Capability Required
RNS Suction Lines, from the RCS Hot Leg Connection to the RCS Side of Valves RNS PL-V001A and RNS-PL-V001B	RNS-L001 RNS-L002A RNS-L002B	Yes	Yes	No
RNS Suction Lines, from the RCS Pressure Boundary Valves, RNS-PL-V001A and RNS-PL-V001B, to the RNS pumps	RNS-L004A RNS-L004B RNS-L005 RNS-L006 RNS-L007A RNS-L007B RNS-L009A RNS-L009B	Yes	No	Yes Yes Yes Yes Yes Yes Yes Yes
RNS Suction Line from CVS	RNS-L061	Yes	No	Yes
RNS Suction Line from IRWST	RNS-L029	Yes	No	Yes
RNS Suction Line LTOP Relief	RNS-L040 RNS-L090	Yes	No	Yes
RNS Discharge Lines, from the RNS Pumps to the RNS Heat Exchangers RNS-ME-01A and RNS-ME-01B	RNS-L011A RNS-L011B	Yes	No	Yes
RNS Discharge Lines, from RNS Heat Exchanger RNS-ME-01A to Containment Isolation Valve RNS-PL-V011	RNS-L012A RNS-L014	Yes	No	Yes
RNS Discharge Line, from RNS Heat Exchanger RNS-ME-01B to Common Discharge Header RNS-L014	RNS-L012B	Yes	No	Yes
RNS Discharge Lines, Containment Isolation Valve RNS-PL-V011 to Containment Isolation Valve RNS-PL-V013	RNS-L016	Yes	No	Yes

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Table 2.3.6-2 (cont.)

Line Name	Line No.	ASME Code Section III	Leak Before Break	Functional Capability Required
RNS Suction Line from Cask Loading Pit	RNS-L065	Yes	No	No
RNS Discharge Lines, from Containment Isolation Valve RNS-PL-V013 to RCS Pressure Boundary Isolation Valves RNS-PL-V015A and RNS-PL-V015B	RNS-L017 RNS-L018A RNS-L018B	Yes	No	Yes
RNS Discharge Lines, from Direct Vessel Injection (DVI) Line RNS-BBC-L018A to Passive Core Cooling System (PXS) IRWST Return Isolation Valve RNS-PL-V024	RNS-L020	Yes	No	No
RNS Discharge Lines, from RCS Pressure Boundary Isolation Valves RNS-PL-V015A and RNS-PL-V015B to RCS Pressure Boundary Isolation Valves RNS-PL-V017A and RNS-PL-V017B	RNS-L019A RNS-L019B	Yes	No	Yes
RNS Heat Exchanger Bypass	RNS-L008A RNS-L008B	Yes	No	No
RNS Suction from Spent Fuel Pool	RNS-L052	Yes	No	No
RNS Pump Miniflow Return	RNS-L030A RNS-L030B	Yes	No	No
RNS Discharge to Spent Fuel Pool	RNS-L051	Yes	No	No
RNS Discharge to CVS Purification	RNS-L021	Yes	No	No

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Table 2.3.6-3

Equipment Name	Tag No.	Display	Control Function
RNS Pump A (Motor)	RNS-MP-01A	Yes (Run Status)	Start
RNS Pump B (Motor)	RNS-MP-01B	Yes (Run Status)	Start
RNS Flow Sensor	RNS-01A	Yes	-
RNS Flow Sensor	RNS-01B	Yes	-
RNS Suction from Cask Loading Pit Isolation Valve (Position Indicator)	RNS-PL-V055	Yes	-
RNS Pump Miniflow Isolation Valve (Position Indicator)	RNS-PL-V057A	Yes	-
RNS Pump Miniflow Isolation Valve (Position Indicator)	RNS-PL-V057B	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.6-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) The components identified in Table 2.3.6-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.3.6-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.3.6-1 and 2.3.6-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.3.6-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.3.6-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.3.6-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The piping identified in Table 2.3.6-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the components and piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components and piping identified in Tables 2.3.6-1 and 2.3.6-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
5.b) Each of the lines identified in Table 2.3.6-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.3.6-2 for which functional capability is required meets the requirements for functional capability.

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Table 2.3.6-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Each of the as-built lines identified in Table 2.3.6-2 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	Inspection will be performed for the existence of an LBB evaluation report or an evaluation report on the protection from dynamic effects of a pipe break. Tier 1 Material, Section 3.3, Nuclear Island Buildings, contains the design descriptions and inspections, tests, analyses, and acceptance criteria for protection from the dynamic effects of pipe rupture.	An LBB evaluation report exists and concludes that the LBB acceptance criteria are met by the as-built RNS piping and piping materials, or a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided.
5.a) The seismic Category I equipment identified in Table 2.3.6-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.6-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.6-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
7.a) The Class 1E equipment identified in Tables 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.3.6-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
7.b) Not used.		

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Table 2.3.6-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.c) Not used.		
8.a) Not used.		
8.b) Not used.		
9.a) The RNS provides LTOP for the RCS during shutdown operations.	i) Inspections will be conducted on the low temperature overpressure protection relief valves to confirm that the capacities of the vendor code plate ratings are greater than or equal to system relief requirements. ii) Testing and analysis in accordance with the ASME Code Section III will be performed to determine set pressure.	i) The rated capacities recorded on the valves' vendor code plates are not less than the flow required to provide low-temperature overpressure protection for the RCS, as determined by the LTOPS evaluation based on the pressure-temperature curves developed for the as-procured reactor vessel material. ii) A report exists and concludes that the relief valves open at a pressure not greater than the set pressures required to provide low-temperature overpressure protection for the RCS, as determined by the LTOPS evaluation based on the pressure-temperature curves developed for the as-procured reactor vessel material.
9.b) The RNS provides heat removal from the reactor coolant during shutdown operations.	i) Inspection will be performed for the existence of a report that determines the heat removal capability of the RNS heat exchangers.	i) A report exists and concludes that the product of the overall heat transfer coefficient and the effective heat transfer area, UA, of each RNS heat exchanger is greater than or equal to 2.2 million Btu/hr-°F.

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Table 2.3.6-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.b) The RNS provides heat removal from the reactor coolant during shutdown operations.	ii) Testing will be performed to confirm that the RNS can provide flow through the RNS heat exchangers when the pump suction is aligned to the RCS hot leg and the discharge is aligned to both PXS DVI lines with the RCS at atmospheric pressure.	ii) When tested individually, each RNS pump provides at least 1400 gpm net flow to the RCS when the hot leg water level is at an elevation 15.5 inches ± 2 inches above the bottom of the hot leg.
	iii) Inspection will be performed of the reactor coolant loop piping.	iii) The RCS cold legs piping centerline is 17.5 inches ± 2 inches above the hot legs piping centerline.
	iv) Inspection will be performed of the RNS pump suction piping.	iv) The RNS pump suction piping from the hot leg to the pump suction piping low point does not form a local high point (defined as an upward slope with a vertical rise greater than 3 inches).
	v) Inspection will be performed of the RNS pump suction nozzle connection to the RCS hot leg.	v) The RNS suction line connection to the RCS is constructed from 20-inch Schedule 140 pipe.
9.c) The RNS provides low pressure makeup flow from the cask loading pit to the RCS for scenarios following actuation of the ADS.	Testing will be performed to confirm that the RNS can provide low pressure makeup flow from the cask loading pit to the RCS when the pump suction is aligned to the cask loading pit and the discharge is aligned to both PXS DVI lines with RCS at atmospheric pressure.	When tested individually, each RNS pump provides at least 1100 gpm net flow to the RCS with the water level maintained within the range of >10 ft and <15 ft from the bottom of the cask loading pit.
9.d) The RNS provides heat removal from the in-containment refueling water storage tank (IRWST).	Testing will be performed to confirm that the RNS can provide flow through the RNS heat exchangers when the pump suction is aligned to the IRWST and the discharge is aligned to the IRWST.	Two operating RNS pumps provide at least 2000 gpm to the IRWST.

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Table 2.3.6-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12.a) The motor-operated and check valves identified in Table 2.3.6-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under preoperational flow, differential pressure and temperature conditions. iv) Exercise testing of the check valves active safety functions identified in Table 2.3.6-1 will be performed under preoperational test pressure, temperature and fluid flow conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.3.6-1 under preoperational test conditions. iv) Each check valve changes position as indicated in Table 2.3.6-1.
10. Safety-related displays identified in Table 2.3.6-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.6-1 can be retrieved in the MCR.
11.a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.3.6-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.3.6-1 using the controls in the MCR.	Controls in the MCR operate to cause those remotely operated valves identified in Table 2.3.6-1 to perform active functions.
11.b) The valves identified in Table 2.3.6-1 as having PMS control perform active safety functions after receiving a signal from the PMS.	Testing will be performed using real or simulated signals into the PMS.	The valves identified in Table 2.3.6-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
12.b) After loss of motive power, the remotely operated valves identified in Table 2.3.6-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valve identified in Table 2.3.6-1 assumes the indicated loss of motive power position.
13. Controls exist in the MCR to cause the pumps identified in Table 2.3.6-3 to perform the control function.	Testing will be performed to actuate the pumps identified in Table 2.3.6-3 using controls in the MCR.	Controls in the MCR cause pumps identified in Table 2.3.6-3 to perform the control function.
14. Displays of the RNS parameters identified in Table 2.3.6-3 can be retrieved in the MCR.	Inspection will be performed for retrievability in the MCR of the displays identified in Table 2.3.6-3.	Displays of the RNS parameters identified in Table 2.3.6-3 are retrieved in the MCR.

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Table 2.3.6-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12.a) The motor-operated and check valves identified in Table 2.3.6-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions. ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tested conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.3.6-1 under design conditions. ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tested conditions.

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Table 2.3.6-5

Component Name	Tag No.	Component Location
RNS Pump A	RNS-MP-01A	Auxiliary Building
RNS Pump B	RNS-MP-01B	Auxiliary Building
RNS Heat Exchanger A	RNS-ME-01A	Auxiliary Building
RNS Heat Exchanger B	RNS-ME-01B	Auxiliary Building

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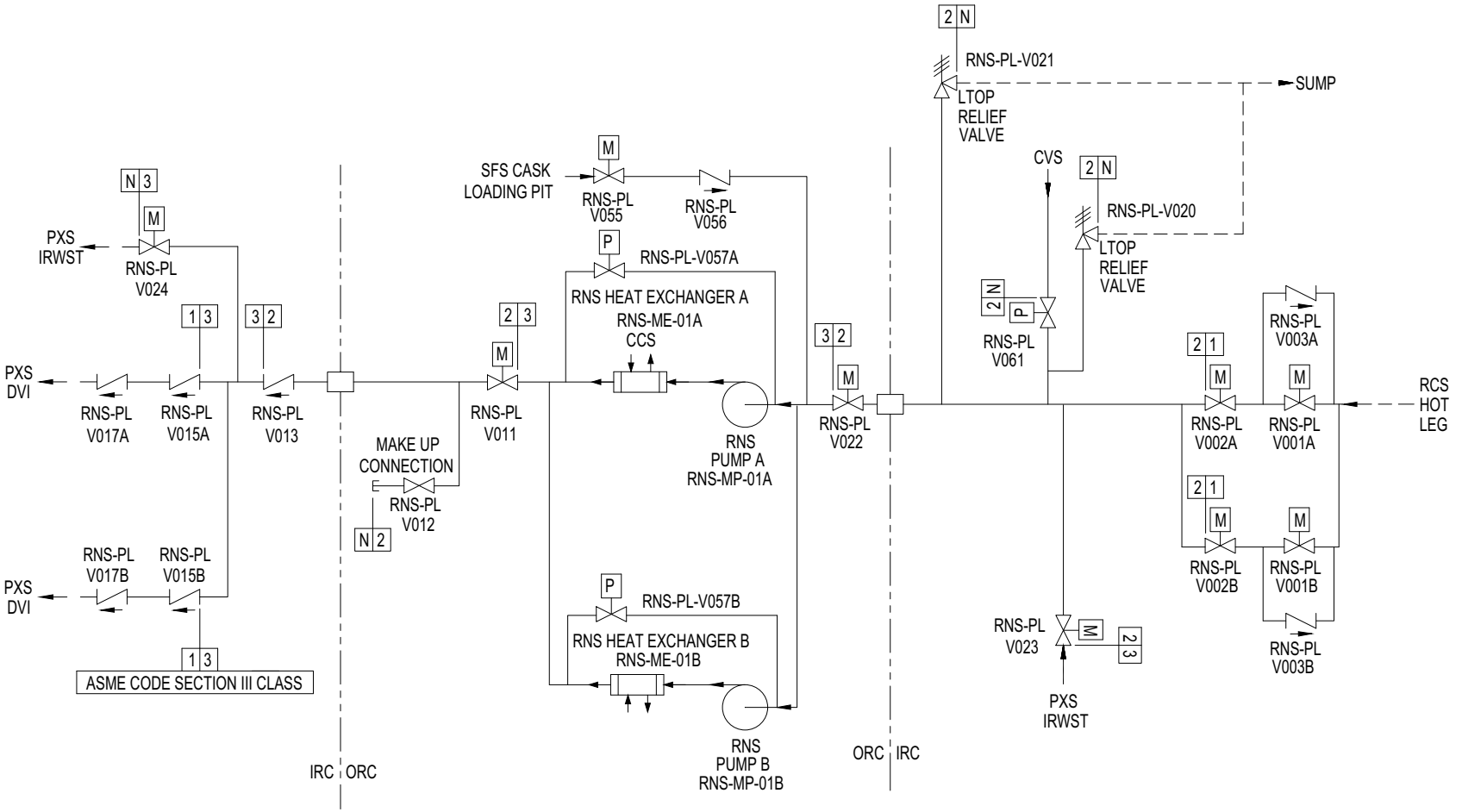

Figure 2.3.6-1
Normal Residual Heat Removal System

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2.3.7 Spent Fuel Pool Cooling System

Design Description

The spent fuel pool cooling system (SFS) removes decay heat from spent fuel by transferring heat from the water in the spent fuel pool to the component cooling water system during normal modes of operation. The SFS purifies the water in the spent fuel pool, fuel transfer canal, and in-containment refueling water storage tank during normal modes of operation. Following events such as earthquakes, or fires, if the normal heat removal method is not available, decay heat is removed from spent fuel by boiling water in the pool. In the event of long-term station blackout, makeup water is supplied to the spent fuel pool from onsite storage tanks.

The SFS is as shown in Figure 2.3.7-1 and the component locations of the SFS are as shown in Table 2.3.7-5.

1. The functional arrangement of the SFS is as described in the Design Description of this Section 2.3.7.

2. a) The components identified in Table 2.3.7-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping lines identified in Table 2.3.7-2 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

3. Pressure boundary welds in piping lines identified in Table 2.3.7-2 as ASME Code Section III meet ASME Code Section III requirements.

4. The piping lines identified in Table 2.3.7-2 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

5. The seismic Category I components identified in Table 2.3.7-1 can withstand seismic design basis loads without loss of safety function.

6. a) The Class 1E components identified in Table 2.3.7-1 are powered from their respective Class 1E division.

   b) Separation is provided between SFS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

7. The SFS performs the following safety-related functions:

   a) The SFS preserves containment integrity by isolating the SFS piping lines penetrating the containment.

   b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site water storage tanks.

   c) The SFS provides check valves in the drain line from the refueling cavity to prevent flooding of the refueling cavity during containment flooding.

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8. The SFS provides the nonsafety-related function of removing spent fuel decay heat using pumped flow through a heat exchanger.

9. Safety-related displays identified in Table 2.3.7-1 can be retrieved in the main control room (MCR).

10. Controls exist in the MCR to cause the pumps identified in Table 2.3.7-3 to perform their listed functions.

11. Displays of the SFS parameters identified in Table 2.3.7-3 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.7-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the SFS.

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Table 2.3.7-1

Component Name	Tag No.	ASME Code Section III	Seismic Cat 1	Remotely Operated Valve	Class 1E/ Qual for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Spent Fuel Pool Level Sensor	SFS-019A	No	Yes	-	Yes/No	Yes	-	-	-
Spent Fuel Pool Level Sensor	SFS-019B	No	Yes	-	Yes/No	Yes	-	-	-
Spent Fuel Pool Level Sensor	SFS-019C	No	Yes	-	Yes/No	Yes	-	-	-
Refueling Cavity Drain to SGS Compartment Isolation Valve	SFS-PL-V031	Yes	Yes	No	-/-	Yes	-	-	-
Refueling Cavity to SFS Pump Suction Isolation Valve	SFS-PL-V032	Yes	Yes	No	-/-	No	-	-	-
Refueling Cavity Drain to Containment Sump Isolation Valve	SFS-PL-V033	Yes	Yes	No	-/-	Yes	-	-	-
IRWST to SFS Pump Suction Line Isolation Valve	SFS-PL-V039	Yes	Yes	No	-/-	No	-	-	-
Fuel Transfer Canal to SFS Pump Suction Iso. Valve	SFS-PL-V040	Yes	Yes	No	-/-	No	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.3.7-1 (cont.)

Component Name	Tag No.	ASME Code Section III	Seismic Cat 1	Remotely Operated Valve	Class 1E/ Qual for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Cask Loading Pit to SFS Pump Suction Isolation Valve	SFS-PL-V041	Yes	Yes	No	-/-	No	-	-	-
Cask Loading Pit to SFS Pump Suction Isolation Valve	SFS-PL-V042	Yes	Yes	No	-/-	No	-	Transfer Closed	-
SFS Pump Discharge Line to Cask Loading Pit Isolation Valve	SFS-PL-V045	Yes	Yes	No	-/-	No	-	Transfer Closed	-
Cask Loading Pit to WLS Isolation Valve	SFS-PL-V049	Yes	Yes	No	-/-	No	-	Transfer Closed	-
Spent Fuel Pool to Cask Washdown Pit Isolation Valve	SFS-PL-V066	Yes	Yes	No	-/-	No	-	Transfer Open	-
Cask Washdown Pit Drain Isolation Valve	SFS-PL-V068	Yes	Yes	No	-/-	No	-	Transfer Open	-
Refueling Cavity Drain Line Check Valve	SFS-PL-V071	Yes	Yes	No	-/-	No	-	Transfer Open Transfer Closed	-

Note: Dash (-) indicates not applicable.

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Table 2.3.7-1 (cont.)

Component Name	Tag No.	ASME Code Section III	Seismic Cat 1	Remotely Operated Valve	Class 1E/ Qual for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
Refueling Cavity Drain Line Check Valve	SFS-PL-V072	Yes	Yes	No	-/-	No	-	Transfer Open Transfer Closed	-
SFS Containment Floodup Isolation Valve	SFS-PL-V075	Yes	Yes	No	-/-	Yes	-	-	-

Note: Dash (-) indicates not applicable.

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Table 2.3.7-2

Piping Line Name	Line Number	ASME Code Section III
Spent Fuel Pool to RNS Pump Suction	L014	Yes
Cask Loading Pit to RNS Pump Suction	L115	Yes
Refueling Cavity Drain	L033	Yes
PXS IRWST to SFS Pump Suction	L035	Yes
Refueling Cavity Skimmer to SFS Pump Suction	L036	Yes
Refueling Cavity Drain	L037	Yes
Refueling Cavity Drain	L044	Yes
Fuel Transfer Canal Drain	L047	Yes
Cask Washdown Pit Drain	L068	Yes
Cask Loading Pit Drain	L043	Yes
Cask Pit Transfer Branch Line	L045	Yes
Spent Fuel Pool Containment Isolation Thermal Relief Line	L052	Yes
Refueling Cavity Drain	L030	Yes
Upender Pit Drain/Fill Line	L121	Yes
Spent Fuel Pool Drain	L066	Yes
Cask Loading Pit to WLS	L067	Yes
RNS Return to Spent Fuel Pool	L100	Yes
SFS Containment Floodup Line	L120	Yes

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Table 2.3.7-3

Component Name	Tag No.	Display	Control Function
SFS Pump A	SFS-MP-01A	Yes (Run Status)	Start
SFS Pump B	SFS-MP-01B	Yes (Run Status)	Start
SFS Flow Sensor	SFS-13A	Yes	-
SFS Flow Sensor	SFS-13B	Yes	-
Spent Fuel Pool Temperature Sensor	SFS-018	Yes	-
Cask Loading Pit Level Sensor	SFS-022	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.7-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) The components identified in Table 2.3.7-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping lines identified in Table 2.3.7-2 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the ASME as-built components and piping lines as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping lines identified in Tables 2.3.7-1 and 2.3.7-2 as ASME Code Section III.
3. Pressure boundary welds in piping lines identified in Table 2.3.7-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4. The piping lines identified in Table 2.3.7-2 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the piping lines required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping lines identified in Table 2.3.7-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

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Table 2.3.7-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The seismic Category I components identified in Table 2.3.7-1 can withstand seismic design basis loads without loss of safety functions.	i) Inspection will be performed to verify that the seismic Category I components identified in Table 2.3.7-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I components identified in Table 2.3.7-1 are located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
6.a) Not used.		
6.b) Not used.		
7.a) Not used.		

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Table 2.3.7-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site storage tanks.	i) Inspection will be performed to verify that the spent fuel pool includes a sufficient volume of water.	i) The volume of the spent fuel pool, fuel transfer canal, and both gate areas above the fuel assemblies and below the spent fuel pool cooling suction piping is greater than or equal to 130,350 gallons.
	ii) Inspection will be performed to verify the cask washdown pit includes sufficient volume of water.	ii) The water volume of the cask washdown pit from the cask washdown pit floor to 13.75 feet above the cask washdown pit floor is greater than or equal to 34,100 gallons.
	iii) Not used.	
	iv) Not used	
	v) Not used.	
	vi) Not used.	
	vii) Inspection will be performed to verify the cask loading pit includes sufficient volume of water.	vii) The water volume of the cask loading pit above the bottom of the gate and below the spent fuel pool cooling suction piping is greater than or equal to 46,050 gallons.
7.c) The SFS provides check valves in the drain line from the refueling cavity to prevent flooding of the refueling cavity during containment flooding.	Exercise testing of the check valves with active safety-functions identified in Table 2.3.7-1 will be performed under pre-operational test pressure, temperature and flow conditions.	Each check valve changes position as indicated on Table 2.3.7-1.
8. The SFS provides the nonsafety-related function of removing spent fuel decay heat using pumped flow through a heat exchanger.	ii) Testing will be performed to confirm that each SFS pump provides flow through its heat exchanger when taking suction from the SFP and returning flow to the SFP.	ii) Each SFS pump produces at least 900 gpm through its heat exchanger.
9. Safety-related displays identified in Table 2.3.7-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.7-1 can be retrieved in the MCR.
10. Controls exist in the MCR to cause the pumps identified in Table 2.3.7-3 to perform their listed functions.	Testing will be performed to actuate the pumps identified in Table 2.3.7-3 using controls in the MCR.	Controls in the MCR cause pumps identified in Table 2.3.7-3 to perform the listed functions.
11. Displays of the SFS parameters identified in Table 2.3.7-3 can be retrieved in the MCR.	Inspection will be performed for retrievability in the MCR of the displays identified in Table 2.3.7-3.	Displays of the SFS parameters identified in Table 2.3.7-3 are retrieved in the MCR.

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Table 2.3.7-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The SFS provides the nonsafety-related function of removing spent fuel decay heat using pumped flow through a heat exchanger.	i) Inspection will be performed for the existence of a report that determines the heat removal capability of the SFS heat exchangers.	i) A report exists and concludes that the heat transfer characteristic, UA, of each SFS heat exchanger is greater than or equal to 2.2 million Btu/hr-°F.

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Table 2.3.7-5

Component Name	Tag No.	Component Location
SFS Pump A	SFS-MP-01A	Auxiliary Building
SFS Pump B	SFS-MP-01B	Auxiliary Building
SFS Heat Exchanger A	SFS-ME-01A	Auxiliary Building
SFS Heat Exchanger B	SFS-ME-01B	Auxiliary Building

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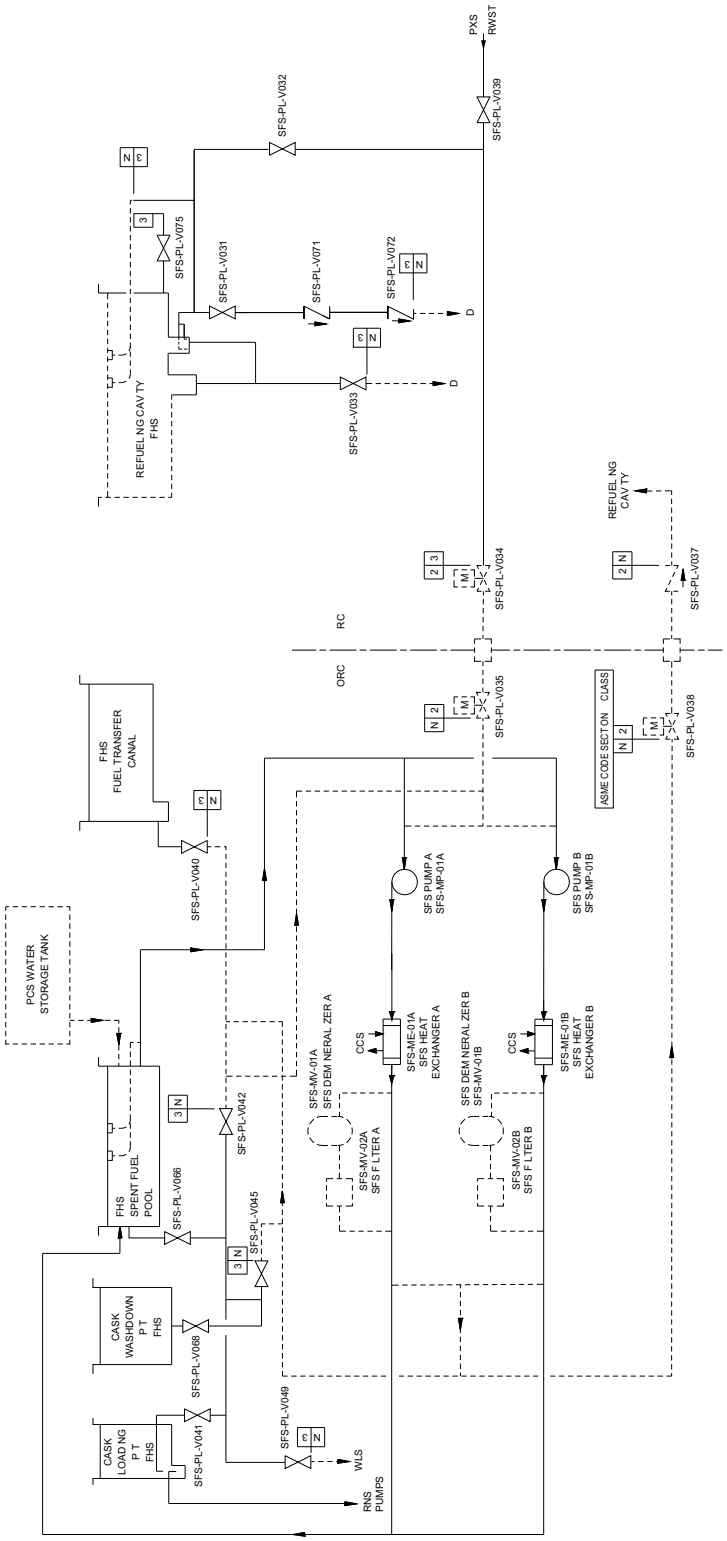

Figure 2.3.7-1
Spent Fuel Pool Cooling System

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2.3.8 Service Water System

Design Description

The service water system (SWS) transfers heat from the component cooling water heat exchangers to the atmosphere. The SWS operates during normal modes of plant operation, including startup, power operation (full and partial loads), cooldown, shutdown, and refueling.

The SWS is as shown in Figure 2.3.8-1 and the component locations of the SWS are as shown Table 2.3.8-3.

1. The functional arrangement of the SWS is as described in the Design Description of this Section 2.3.8.

2. The SWS provides the nonsafety-related function of transferring heat from the component cooling water system (CCS) to the surrounding atmosphere to support plant shutdown and spent fuel pool cooling.

3. Controls exist in the main control room (MCR) to cause the components identified in Table 2.3.8-1 to perform the listed function.

4. Displays of the parameters identified in Table 2.3.8-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.8-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the SWS.

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Table 2.3.8-1

Equipment Name	Tag No.	Display	Control Function
Service Water Pump A (Motor)	SWS-MP-01A	Yes (Run Status)	Start
Service Water Pump B (Motor)	SWS-MP-01B	Yes (Run Status)	Start
Service Water Cooling Tower Fan A (Motor)	SWS-MA-01A	Yes (Run Status)	Start
Service Water Cooling Tower Fan B (Motor)	SWS-MA-01B	Yes (Run Status)	Start
Service Water Pump A Flow Sensor	SWS-004A	Yes	-
Service Water Pump B Flow Sensor	SWS-004B	Yes	-
Service Water Pump A Discharge Valve	SWS-PL-V002A	Yes (Valve Position)	Open
Service Water Pump B Discharge Valve	SWS-PL-V002B	Yes (Valve Position)	Open
Service Water Pump A Discharge Temperature Sensor	SWS-005A	Yes	-
Service Water Pump B Discharge Temperature Sensor	SWS-005B	Yes	-
Service Water Cooling Tower Basin Level	SWS-009	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.8-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The SWS provides the nonsafety-related function of transferring heat from the component cooling water system to the surrounding atmosphere to support plant shutdown and spent fuel pool cooling.	i) Testing will be performed to confirm that the SWS can provide cooling water to the CCS heat exchangers.	i) Each SWS pump can provide at least 10,000 gpm of cooling water through its CCS heat exchanger.
3. Controls exist in the MCR to cause the components identified in Table 2.3.8-1 to perform the listed function.	Testing will be performed on the components in Table 2.3.8-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.3.8-1 to perform the listed functions.
4. Displays of the parameters identified in Table 2.3.8-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.3.8-1 can be retrieved in the MCR.
2. The SWS provides the nonsafety-related function of transferring heat from the component cooling water system to the surrounding atmosphere to support plant shutdown and spent fuel pool cooling.	ii) Inspection will be performed for the existence of a report that determines the heat transfer capability of each cooling tower cell.	ii) A report exists and concludes that the heat transfer rate of each cooling tower cell is greater than or equal to 170 million Btu/hr at a 80.1°F ambient wet bulb temperature and a cold water temperature of 90°F.
	iii) Testing will be performed to confirm that the SWS cooling tower basin has adequate reserve volume.	iii) The SWS tower basin contains a usable volume of at least 230,000 gallons at the basin low level alarm setpoint.

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Table 2.3.8-3

Component Name	Tag No.	Component Location
Service Water Pump A	SWS-MP-01A	Turbine Building or yard
Service Water Pump B	SWS-MP-01B	Turbine Building or yard
Service Water Cooling Tower	SWS-ME-01	Yard

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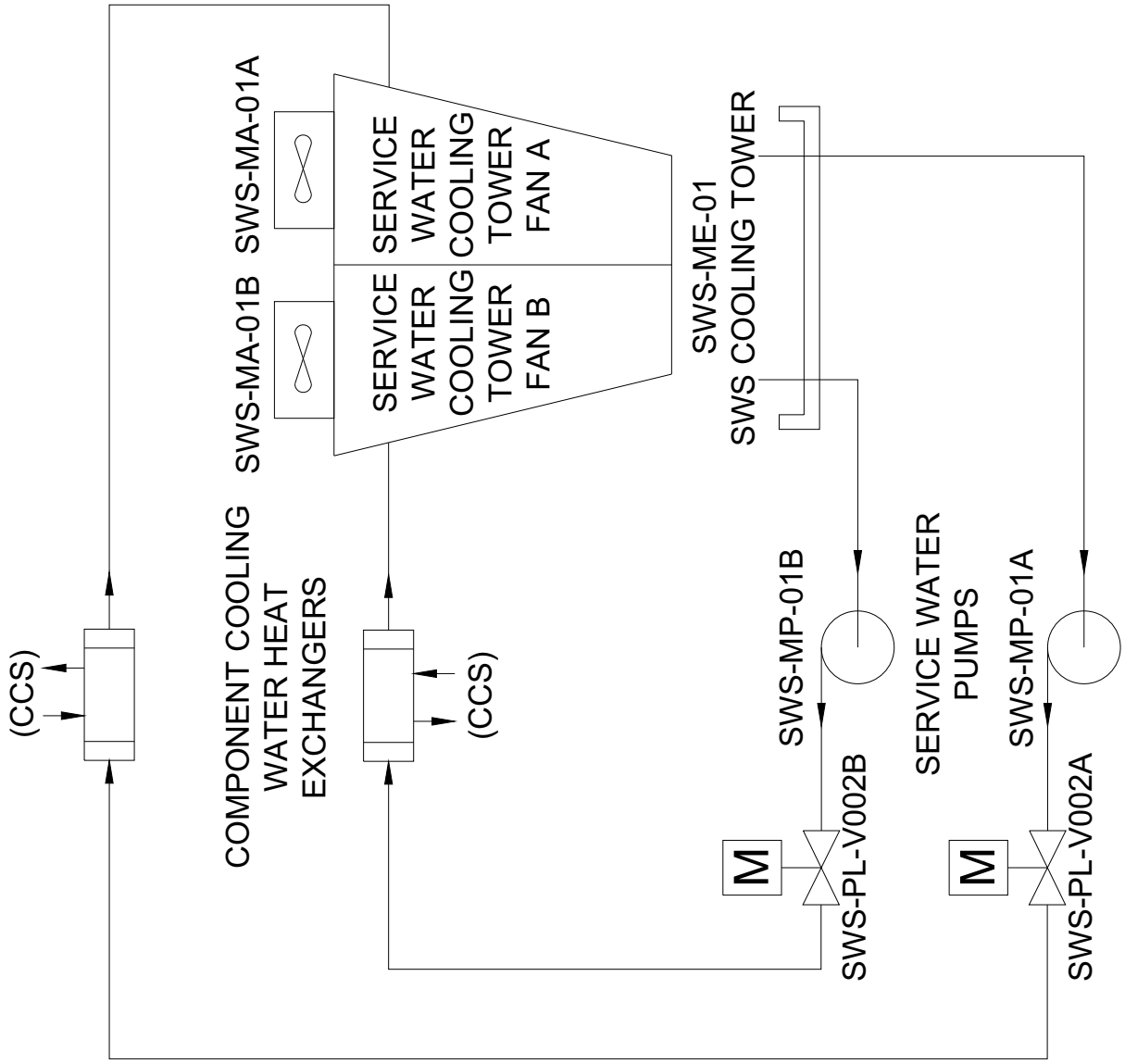

Figure 2.3.8-1
Service Water System

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2.3.9 Containment Hydrogen Control System

Design Description

The containment hydrogen control system (VLS) limits hydrogen gas concentration in containment during accidents.

The VLS has catalytic hydrogen recombiners (VLS-MY-E01A and VLS-MY-E01B) that are located inside containment. The VLS has hydrogen igniters located as shown on Table 2.3.9-2.

1. The functional arrangement of the VLS is as described in the Design Description of this Section 2.3.9.

2. a) The hydrogen monitors identified in Table 2.3.9-1 are powered by the non-Class 1E dc and UPS system.

   b) The components identified in Table 2.3.9-2 are powered from their respective non-Class 1E power group.

3. The VLS provides the non-safety related function to control the containment hydrogen concentration for beyond design basis accidents.

4. a) Controls exist in the MCR to cause the components identified in Table 2.3.9-2 to perform the listed function.

   b) The components identified in Table 2.3.9-2 perform the listed function after receiving a manual signal from the diverse actuation system (DAS).

5. Displays of the parameters identified in Table 2.3.9-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.9-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the VLS.

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Table 2.3.9-1

Equipment	Tag No.	Display
Containment Hydrogen Monitor	VLS-001	Yes
Containment Hydrogen Monitor	VLS-002	Yes
Containment Hydrogen Monitor	VLS-003	Yes

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Table 2.3.9-2

Equipment Name	Tag Number	Function	Power Group Number	Location	Room No.
Hydrogen Igniter 01	VLS-EH-01	Energize	1	Tunnel connection loop compartments	11204
Hydrogen Igniter 02	VLS-EH-02	Energize	2	Tunnel connection loop compartments	11204
Hydrogen Igniter 03	VLS-EH-03	Energize	1	Tunnel connection loop compartments	11204
Hydrogen Igniter 04	VLS-EH-04	Energize	2	Tunnel connection loop compartments	11204
Hydrogen Igniter 05	VLS-EH-05	Energize	1	Loop compartment 02	11402
Hydrogen Igniter 06	VLS-EH-06	Energize	2	Loop compartment 02	11502
Hydrogen Igniter 07	VLS-EH-07	Energize	2	Loop compartment 02	11402
Hydrogen Igniter 08	VLS-EH-08	Energize	1	Loop compartment 02	11502
Hydrogen Igniter 09	VLS-EH-09	Energize	1	In-containment refueling water storage tank (IRWST) sparger side	11305
Hydrogen Igniter 10	VLS-EH-10	Energize	2	IRWST sparger side	11305
Hydrogen Igniter 11	VLS-EH-11	Energize	2	Loop compartment 01	11401
Hydrogen Igniter 12	VLS-EH-12	Energize	1	Loop compartment 01	11501
Hydrogen Igniter 13	VLS-EH-13	Energize	1	Loop compartment 01	11401
Hydrogen Igniter 14	VLS-EH-14	Energize	2	Loop compartment 01	11501
Hydrogen Igniter 15	VLS-EH-15	Energize	2	IRWST vacuum breaker vents	11305
Hydrogen Igniter 16	VLS-EH-16	Energize	1	IRWST vacuum breaker vents	11305
Hydrogen Igniter 17	VLS-EH-17	Energize	2	Northeast valve room	11207
Hydrogen Igniter 18	VLS-EH-18	Energize	1	Northeast accumulator room	11207
Hydrogen Igniter 19	VLS-EH-19	Energize	2	East valve room	11208
Hydrogen Igniter 20	VLS-EH-20	Energize	2	Southeast accumulator room	11206
Hydrogen Igniter 21	VLS-EH-21	Energize	1	Southeast valve room	11206
Hydrogen Igniter 22	VLS-EH-22	Energize	1	Lower compartment area (core makeup tank [CMT] and valve area)	11400
Hydrogen Igniter 23	VLS-EH-23	Energize	2	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 24	VLS-EH-24	Energize	2	Lower compartment area (CMT and valve area)	11400

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Table 2.3.9-2 (cont.)

Equipment Name	Tag Number	Function	Power Group Number	Location	Room No.
Hydrogen Igniter 25	VLS-EH-25	Energize	2	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 26	VLS-EH-26	Energize	2	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 27	VLS-EH-27	Energize	1	Lower compartment area (CMT and valve area)	11300
Hydrogen Igniter 28	VLS-EH-28	Energize	1	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 29	VLS-EH-29	Energize	1	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 30	VLS-EH-30	Energize	2	Lower compartment area (CMT and valve area)	11403
Hydrogen Igniter 31	VLS-EH-31	Energize	1	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 32	VLS-EH-32	Energize	1	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 33	VLS-EH-33	Energize	2	North CVS equipment room	11209
Hydrogen Igniter 34	VLS-EH-34	Energize	1	North CVS equipment room	11209
Hydrogen Igniter 35	VLS-EH-35	Energize	1	IRWST hooded vents	11500
Hydrogen Igniter 36	VLS-EH-36	Energize	2	IRWST hooded vents	11500
Hydrogen Igniter 37	VLS-EH-37	Energize	1	IRWST hooded vents	11500
Hydrogen Igniter 38	VLS-EH-38	Energize	2	IRWST hooded vents	11500
Hydrogen Igniter 39	VLS-EH-39	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 40	VLS-EH-40	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 41	VLS-EH-41	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 42	VLS-EH-42	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 43	VLS-EH-43	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 44	VLS-EH-44	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 45	VLS-EH-45	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 46	VLS-EH-46	Energize	2	Upper compartment lower region	11500

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Table 2.3.9-2 (cont.)

Equipment Name	Tag Number	Function	Power Group Number	Location	Room No.
Hydrogen Igniter 47	VLS-EH-47	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 48	VLS-EH-48	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 49	VLS-EH-49	Energize	1	Pressurizer compartment	11503
Hydrogen Igniter 50	VLS-EH-50	Energize	2	Pressurizer compartment	11503
Hydrogen Igniter 51	VLS-EH-51	Energize	1	Upper compartment mid-region	11500
Hydrogen Igniter 52	VLS-EH-52	Energize	2	Upper compartment mid-region	11500
Hydrogen Igniter 53	VLS-EH-53	Energize	2	Upper compartment mid-region	11500
Hydrogen Igniter 54	VLS-EH-54	Energize	1	Upper compartment mid-region	11500
Hydrogen Igniter 55	VLS-EH-55	Energize	1	Refueling cavity	11504
Hydrogen Igniter 56	VLS-EH-56	Energize	2	Refueling cavity	11504
Hydrogen Igniter 57	VLS-EH-57	Energize	2	Refueling cavity	11504
Hydrogen Igniter 58	VLS-EH-58	Energize	1	Refueling cavity	11504
Hydrogen Igniter 59	VLS-EH-59	Energize	2	Pressurizer compartment	11503
Hydrogen Igniter 60	VLS-EH-60	Energize	1	Pressurizer compartment	11503
Hydrogen Igniter 61	VLS-EH-61	Energize	1	Upper compartment-upper region	11500
Hydrogen Igniter 62	VLS-EH-62	Energize	2	Upper compartment-upper region	11500
Hydrogen Igniter 63	VLS-EH-63	Energize	1	Upper compartment-upper region	11500
Hydrogen Igniter 64	VLS-EH-64	Energize	2	Upper compartment-upper region	11500
Hydrogen Igniter 65	VLS-EH-65	Energize	1	IRWST roof vents	11500
Hydrogen Igniter 66	VLS-EH-66	Energize	2	IRWST roof vents	11500

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Table 2.3.9-3
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) The hydrogen monitors identified in Table 2.3.9-1 are powered by the non-Class 1E dc and UPS system.	Testing will be performed by providing a simulated test signal in each power group of the non-Class 1E dc and UPS system.	A simulated test signal exists at the hydrogen monitors identified in Table 2.3.9-1 when the non-Class 1E dc and UPS system is provided the test signal.
2.b) The components identified in Table 2.3.9-2 are powered from their respective non-Class 1E power group.	Testing will be performed by providing a simulated test signal in each non-Class 1E power group.	A simulated test signal exists at the equipment identified in Table 2.3.9-2 when the assigned non-Class 1E power group is provided the test signal.
3. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	i) Inspection for the number of igniters will be performed. ii) Operability testing will be performed on the igniters.	i) At least 66 hydrogen igniters are provided inside containment at the locations specified in Table 2.3.9-2. ii) The surface temperature of the igniter meets or exceeds 1700°F.
4.a) Controls exist in the MCR to cause the components identified in Table 2.3.9-2 to perform the listed function.	Testing will be performed on the igniters using the controls in the MCR.	Controls in the MCR operate to energize the igniters.
5. Displays of the parameters identified in Table 2.3.9-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.3.9-1 in the MCR.	Displays identified in Table 2.3.9-1 can be retrieved in the MCR.

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Table 2.3.9-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	iii) An inspection of the as-built containment internal structures will be performed.	iii) The equipment access opening and CMT-A opening constitute at least 98% of the vent path area from Room 11206 to Room 11300. The minimum distance between the equipment access opening and the containment shell is at least 24.3 feet. The minimum distance between the CMT-A opening and the containment shell is at least 9.4 feet. The CMT-B opening constitutes at least 98% of the vent path area from Room 11207 to Room 11300 and is a minimum distance of 24.6 feet away from the containment shell. Other openings through the ceilings of these rooms must be at least 3 feet from the containment shell.
	iv) An inspection will be performed of the as-built IRWST vents that are located in the roof of the IRWST along the side of the IRWST next to the containment shell.	iv) The discharge from each of these IRWST vents is oriented generally away from the containment shell.
4.b) The components identified in Table 2.3.9-2 perform the listed function after receiving a manual signal from DAS.	Testing will be performed on the igniters using the DAS controls.	The igniters energize after receiving a signal from DAS.

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2.3.10 Liquid Radwaste System

Design Description

The liquid radwaste system (WLS) receives, stores, processes, samples and monitors the discharge of radioactive wastewater.

The WLS has components which receive and store radioactive or potentially radioactive liquid waste. These are the reactor coolant drain tank, the containment sump, the effluent holdup tanks and the waste holdup tanks. The WLS components store and process the waste during normal operation and during anticipated operational occurrences. Monitoring of the liquid waste is performed prior to discharge.

The WLS is as shown in Figure 2.3.10-1 and the component locations of the WLS are as shown in Table 2.3.10-5.

1. The functional arrangement of the WLS is as described in the Design Description of this Section 2.3.10.

2. a) The components identified in Table 2.3.10-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.3.10-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.3.10-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.3.10-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.3.10-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.3.10-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. a) The seismic Category I equipment identified in Table 2.3.10-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.3.10-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

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6. The WLS provides the following safety-related functions:

   a) The WLS preserves containment integrity by isolation of the WLS lines penetrating the containment.

   b) Check valves in drain lines to the containment sump limit cross flooding of compartments.

7. The WLS provides the nonsafety-related functions of:

   a) Detecting leaks within containment to the containment sump.

   b) Controlling releases of radioactive materials in liquid effluents.

8. Controls exist in the main control room (MCR) to cause the remotely operated valve identified in Table 2.3.10-3 to perform its active function.

9. The check valves identified in Table 2.3.10-1 perform an active safety-related function to change position as indicated in the table.

10. Displays of the parameters identified in Table 2.3.10-3 can be retrieved in the MCR.

11. a) The Class 1E components identified in Table 2.3.10-1 are powered from their respective Class 1E division.

    b) Separation is provided between WLS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

12. Safety-related displays identified in Table 2.3.10-1 can be retrieved in the main control room (MCR).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.10-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the WLS.

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Table 2.3.10-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Active Function
WLS Containment Sump Level Sensor	WLS-034	No	Yes	No	No/No	No	-
WLS Containment Sump Level Sensor	WLS-035	No	Yes	No	No/No	No	-
WLS Containment Sump Level Sensor	WLS-036	No	Yes	No	No/No	No	-
WLS Drain from Passive Core Cooling System (PXS) Compartment A (Room 11206) Check Valve	WLS-PL-V071B	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from PXS Compartment A (Room 11206) Check Valve	WLS-PL-V072B	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from PXS Compartment B (Room 11207) Check Valve	WLS-PL-V071C	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from PXS Compartment B (Room 11207) Check Valve	WLS-PL-V072C	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from Chemical and Volume Control System (CVS) Compartment (Room 11209) Check Valve	WLS-PL-V071A	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from CVS Compartment (Room 11209) Check Valve	WLS-PL-V072A	Yes	Yes	No	-/-	No	Transfer Closed
WLS Auxiliary Building Radiologically Controlled Area (RCA) Floodup Level Sensor	WLS-400A	No	Yes	No	Yes/No	Yes	-
WLS Auxiliary Building RCA Floodup Level Sensor	WLS-400B	No	Yes	No	Yes/No	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.10-2

Line Name	Line No.	ASME Section III	Functional Capability Required
WLS Drain from PXS Compartment A	WLS-PL-L062	Yes	Yes
WLS Drain from PXS Compartment B	WLS-PL-L063	Yes	Yes
WLS Drain from CVS Compartment	WLS-PL-L061	Yes	Yes

Table 2.3.10-3

Equipment Name	Tag No.	Display	Active Function
WLS Effluent Discharge Isolation Valve	WLS-PL-V223	-	Close
Reactor Coolant Drain Tank Level	WLS-JE-LT002	Yes	-
Letdown Flow from CVS to WLS	WLS-JE-FT020	Yes	-
WLS Auxiliary Building RCA Floodup Level Sensor	WLS-400A	Yes	-
WLS Auxiliary Building RCA Floodup Level Sensor	WLS-400B	Yes	-

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Table 2.3.10-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) The components identified in Table 2.3.10-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.3.10-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design report exists for the as built components and piping identified in Tables 2.3.10-1 and 2.3.10-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.3.10-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.3.10-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.3.10-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The piping identified in Table 2.3.10-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the components and piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components and piping identified in Tables 2.3.10-1 and 2.3.10-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

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Table 2.3.10-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b) Each of the lines identified in Table 2.3.10-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.3.10-2 for which functional capability is required meets the requirements for functional capability.
5.a) The seismic Category I equipment identified in Table 2.3.10-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.10-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.10-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
6.a) Not used.		
6.b) Not used.		
7.a) The WLS provides the nonsafety-related function of detecting leaks within containment to the containment sump.	i) Inspection will be performed for retrievability of the displays of containment sump level channels WLS-034, WLS-035, and WLS-036 in the MCR.	i) Nonsafety-related displays of WLS containment sump level channels WLS-034, WLS-035, and WLS-036 can be retrieved in the MCR.
	ii) Testing will be performed by adding water to the sump and observing display of sump level.	ii) A report exists and concludes that sump level channels WLS-034, WLS-035, and WLS-036 can detect a change of 1.34 ± 0.5 inches.
7.b) The WLS provides the nonsafety-related function of controlling releases of radioactive materials in liquid effluents.	Tests will be performed to confirm that a simulated high radiation signal from the discharge radiation monitor, WLS-RE-229, causes the discharge isolation valve WLS-PL-V223 to close.	A simulated high radiation signal causes the discharge control isolation valve WLS-PL-V223 to close.
8. Controls exist in the MCR to cause the remotely operated valve identified in Table 2.3.10-3 to perform its active function.	Stroke testing will be performed on the remotely operated valve listed in Table 2.3.10-3 using controls in the MCR.	Controls in the MCR operate to cause the remotely operated valve to perform its active function.

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Table 2.3.10-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. The check valves identified in Table 2.3.10-1 perform an active safety-related function to change position as indicated in the table.	Exercise testing of the check valves with active safety functions identified in Table 2.3.10-1 will be performed under pre-operational test pressure, temperature and flow conditions.	Each check valve changes position as indicated on Table 2.3.10-1.
10. Displays of the parameters identified in Table 2.3.10-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.3.10-3 in the MCR.	Displays identified in Table 2.3.10-3 can be retrieved in the MCR.
11.a) Not used.		
12. Safety-related displays identified in Table 2.3.10-1 can be retrieved in the main control room (MCR).	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.10-1 can be retrieved in the MCR.

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Table 2.3.10-5

Component Name	Tag No.	Component Location
WLS Reactor Coolant Drain Tank	WLS-MT-01	Containment
WLS Containment Sump	WLS-MT-02	Containment
WLS Degasifier Column	WLS-MV-01	Auxiliary Building
WLS Effluent Holdup Tanks	WLS-MT-05A WLS-MT-05B	Auxiliary Building
WLS Waste Holdup Tanks	WLS-MT-06A WLS-MT-06B	Auxiliary Building
WLS Waste Pre-Filter	WLS-MV-06	Auxiliary Building
WLS Ion Exchangers	WLS-MV-03 WLS-MV-04A WLS-MV-04B WLS-MV-04C	Auxiliary Building
WLS Waste After-Filter	WLS-MV-07	Auxiliary Building
WLS Monitor Tanks	WLS-MT-07A WLS-MT-07B WLS-MT-07C	Auxiliary Building
	WLS-MT-07D WLS-MT-07E WLS-MT-07F	Radwaste Building

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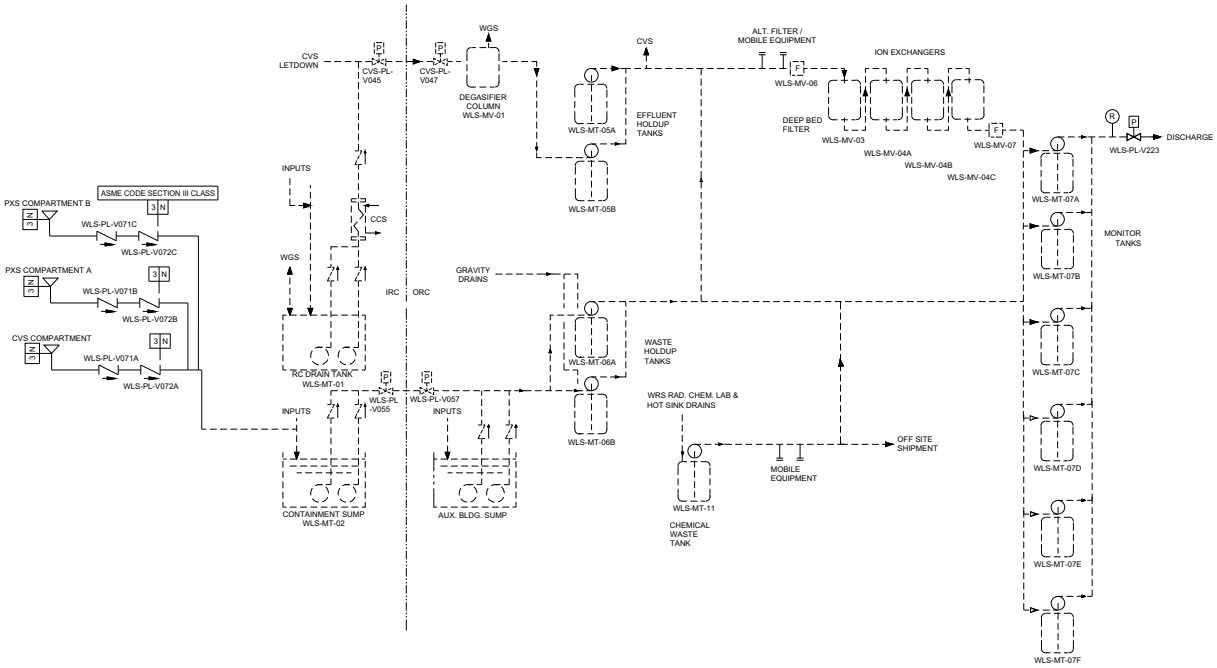

Figure 2.3.10-1
Liquid Radwaste System

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2.3.11 Gaseous Radwaste System

Design Description

The gaseous radwaste system (WGS) receives, processes, and discharges the radioactive waste gases received within acceptable off-site release limits during normal modes of plant operation including power generation, shutdown and refueling.

The WGS is as shown in Figure 2.3.11-1 and the component locations of the WGS are as shown in Table 2.3.11-3.

1. The functional arrangement of the WGS is as described in the Design Description of this Section 2.3.11.

2. The equipment identified in Table 2.3.11-1 can withstand the appropriate seismic design basis loads without loss of its structural integrity function.

3. The WGS provides the nonsafety-related functions of:

    a) Processing radioactive gases prior to discharge.

    b) Controlling the releases of radioactive materials in gaseous effluents.

    c) The WGS is purged with nitrogen on indication of high oxygen levels in the system.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.11-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the WGS.

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Table 2.3.11-1

Equipment Name	Tag No.	Seismic Category I
WGS Activated Carbon Delay Bed A	WGS-MV-02A	No(1)
WGS Activated Carbon Delay Bed B	WGS-MV-02B	No(1)
WGS Discharge Isolation Valve	WGS-PL-V051	No

Note:
1. The WGS activated carbon delay beds (WGS-MV-02A and B) are designed to one-half SSE.

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Table 2.3.11-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The equipment identified as having seismic design requirements in Table 2.3.11-1 can withstand seismic design basis loads without loss of its structural integrity function.	i) Inspection will be performed to verify that the equipment identified as having seismic design requirements in Table 2.3.11-1 is located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismically designed equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The equipment identified as having seismic design requirements in Table 2.3.11-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismically designed equipment can withstand appropriate seismic design basis loads without loss of its structural integrity function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
3.a) The WGS provides the nonsafety-related function of processing radioactive gases prior to discharge.	Inspection will be performed to verify the contained volume of each of the activated carbon delay beds, WGS-MV-02A and WGS-MV-02B.	A report exists and concludes that the contained volume in each of the activated carbon delay beds, WGS-MV-02A and WGS-MV-02B, is at least 80 ft^3.
3.b) The WGS provides the nonsafety-related function of controlling the releases of radioactive materials in gaseous effluents.	Tests will be performed to confirm that the presence of a simulated high radiation signal from the discharge radiation monitor, WGS-017, causes the discharge control isolation valve WGS-PL-V051 to close.	A simulated high radiation signal causes the discharge control isolation valve WGS-PL-V051 to close.
3.c) The WGS is purged with nitrogen on indication of high oxygen levels in the system.	Tests will be performed to confirm that the presence of a simulated high oxygen level signal from the oxygen monitors (WGS-025A, -025B) causes the nitrogen purge valve (WGS-PL-V002) to open and the WLS degasifier vacuum pumps (WLS-MP-03A, -03B) to stop.	A simulated high oxygen level signal causes the nitrogen purge valve (WGS-PL-V002) to open and the WLS degasifier vacuum pumps (WLS-MP-03A, -03B) to stop.

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Table 2.3.11-3

Equipment Name	Tag No.	Component Location
WGS Gas Cooler	WGS-ME-01	Auxiliary Building
WGS Moisture Separator	WGS-MV-03	Auxiliary Building
WGS Activated Carbon Delay Bed A	WGS-MV-02A	Auxiliary Building
WGS Activated Carbon Delay Bed B	WGS-MV-02B	Auxiliary Building

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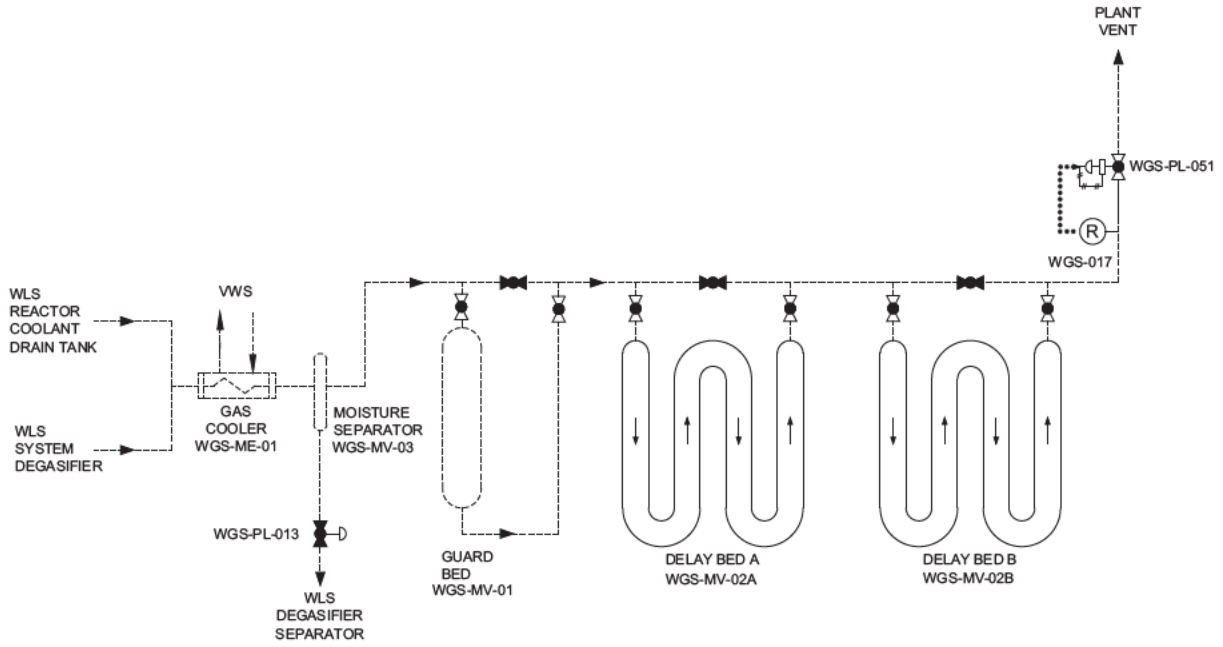

Figure 2.3.11-1
Gaseous Radwaste System

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2.3.12 Solid Radwaste System

Design Description

The solid radwaste system (WSS) receives, collects, and stores the solid radioactive wastes received prior to their processing and packaging by mobile equipment for shipment off-site.

The component locations of the WSS are as shown in Table 2.3.12-2.

1. The functional arrangement of the WSS is as described in the Design Description of this Section 2.3.12.

2. The WSS provides the nonsafety-related function of storing radioactive spent resins prior to processing or shipment.

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Table 2.3.12-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The WSS provides the nonsafety-related function of storing radioactive solids prior to processing or shipment.	Inspection will be performed to verify that the volume of each of the spent resin tanks, WSS-MV01A and WSS-MV01B, is at least 250 ft^3.	A report exists and concludes that the volume of each of the spent resin tanks, WSS-MV01A and WSS-MV01B, is at least 250 ft^3.

Table 2.3.12-2

Component Name	Tag No.	Component Location
WSS Spent Resin Tank A	WSS-MV-01A	Auxiliary Building
WSS Spent Resin Tank B	WSS-MV-01B	Auxiliary Building

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2.3.13 Primary Sampling System

The primary sampling system collects samples of fluids in the reactor coolant system (RCS) and the containment atmosphere during normal operations.

The PSS is as shown in Figure 2.3.13-1. The PSS Grab Sampling Unit (PSS-MS-01) is located in the Auxiliary Building.

1. The functional arrangement of the PSS is as described in the Design Description of this Section 2.3.13.

2. The components identified in Table 2.3.13-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

3. Pressure boundary welds in components identified in Table 2.3.13-1 as ASME Code Section III meet ASME Code Section III requirements.

4. The components identified in Table 2.3.13-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

5. The seismic Category I equipment identified in Table 2.3.13-1 can withstand seismic design basis loads without loss of safety function.

6. a) The Class 1E equipment identified in Table 2.3.13-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of their safety function, for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.3.13-1 are powered from their respective Class 1E division.

   c) Separation is provided between PSS Class 1E divisions, and between Class 1E divisions and non-Class 1E divisions.

7. The PSS provides the safety-related function of preserving containment integrity by isolation of the PSS lines penetrating the containment.

8. The PSS provides the nonsafety-related function of providing the capability of obtaining reactor coolant and containment atmosphere samples.

9. Safety-related displays identified in Table 2.3.13-1 can be retrieved in the MCR.

10. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.3.13-1 to perform active functions.

    b) The valves identified in Table 2.3.13-1 as having protection and safety monitoring system (PMS) control perform an active function after receiving a signal from the PMS.

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11. a) Deleted.

    b) After loss of motive power, the remotely operated valves identified in Table 2.3.13-1 assume the indicated loss of motive power position.

12. Controls exist in the MCR to cause the valves identified in Table 2.3.13-2 to perform the listed function.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.13-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the PSS.

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Table 2.3.13-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS	Active Function	Loss of Motive Power Position
Containment Air Sample Containment Isolation Valve Inside Reactor Containment (IRC)	PSS-PL-V008	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Liquid Sample Line Containment Isolation Valve IRC	PSS-PL-V010A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Liquid Sample Line Containment Isolation Valve IRC	PSS-PL-V010B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Liquid Sample Line Containment Isolation Valve Outside Reactor Containment (ORC)	PSS-PL-V011A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Liquid Sample Line Containment Isolation Valve ORC	PS-PL-V011B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Sample Return Line Containment Isolation Valve ORC	PSS-PL-V023	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Sample Return Containment Isolation Valve IRC	PSS-PL-V024	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Air Sample Line Containment Isolation Valve ORC	PSS-PL-V046	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed

Note: A dash (-) indicates not applicable.

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Table 2.3.13-2

Equipment Name	Tag No.	Control Function
Hot Leg 1 Sample Isolation Valve	PSS-PL-V001A	Transfer Open/Transfer Closed
Hot Leg 2 Sample Isolation Valve	PSS-PL-V001B	Transfer Open/Transfer Closed

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Table 2.3.13-3
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The components identified in Table 2.3.13-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.3.13-1 as ASME Code Section III.
3. Pressure boundary welds in components identified in Table 2.3.13-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4. The components identified in Table 2.3.13-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.3.13-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
5. The seismic Category I equipment identified in Table 2.3.13-1 can withstand seismic design basis loads without loss of its safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.3.13-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.3.13-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.

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Table 2.3.13-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.a) The Class 1E equipment identified in Tables 2.3.13-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of their safety function, for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.3.13-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of its safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.3.13-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
6.b) Not used.		
6.c) Not used.		
7. Not used.		

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Table 2.3.13-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The PSS provides the nonsafety-related function of providing the capability of obtaining reactor coolant and containment atmosphere samples.	Testing will be performed to obtain samples of the reactor coolant and containment atmosphere.	A sample is drawn from the reactor coolant and the containment atmosphere.
9. Safety-related displays identified in Table 2.3.13-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	The safety-related displays identified in Table 2.3.13-1 can be retrieved in the MCR.
10.a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.3.13-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.3.13-1 using the controls in the MCR.	Controls in the MCR operate to cause those remotely operated valves identified in Table 2.3.13-1 to perform active functions.
10.b) The valves identified in Table 2.3.13-1 as having PMS control perform an active function after receiving a signal from the PMS.	Testing will be performed on remotely operated valves listed in Table 2.3.13-1 using real or simulated signals into the PMS.	The remotely operated valves identified in Table 2.3.13-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
11.b) After loss of motive power, the remotely operated valves identified in Table 2.3.13-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.3.13-1 assumes the indicated loss of motive power position.
12. Controls exist in the MCR to cause the valves identified in Table 2.3.13-2 to perform the listed function.	Testing will be performed on the components in Table 2.3.13-2 using controls in the MCR.	Controls in the MCR cause valves identified in Table 2.3.13-2 to perform the listed functions.
11.a) Deleted.		

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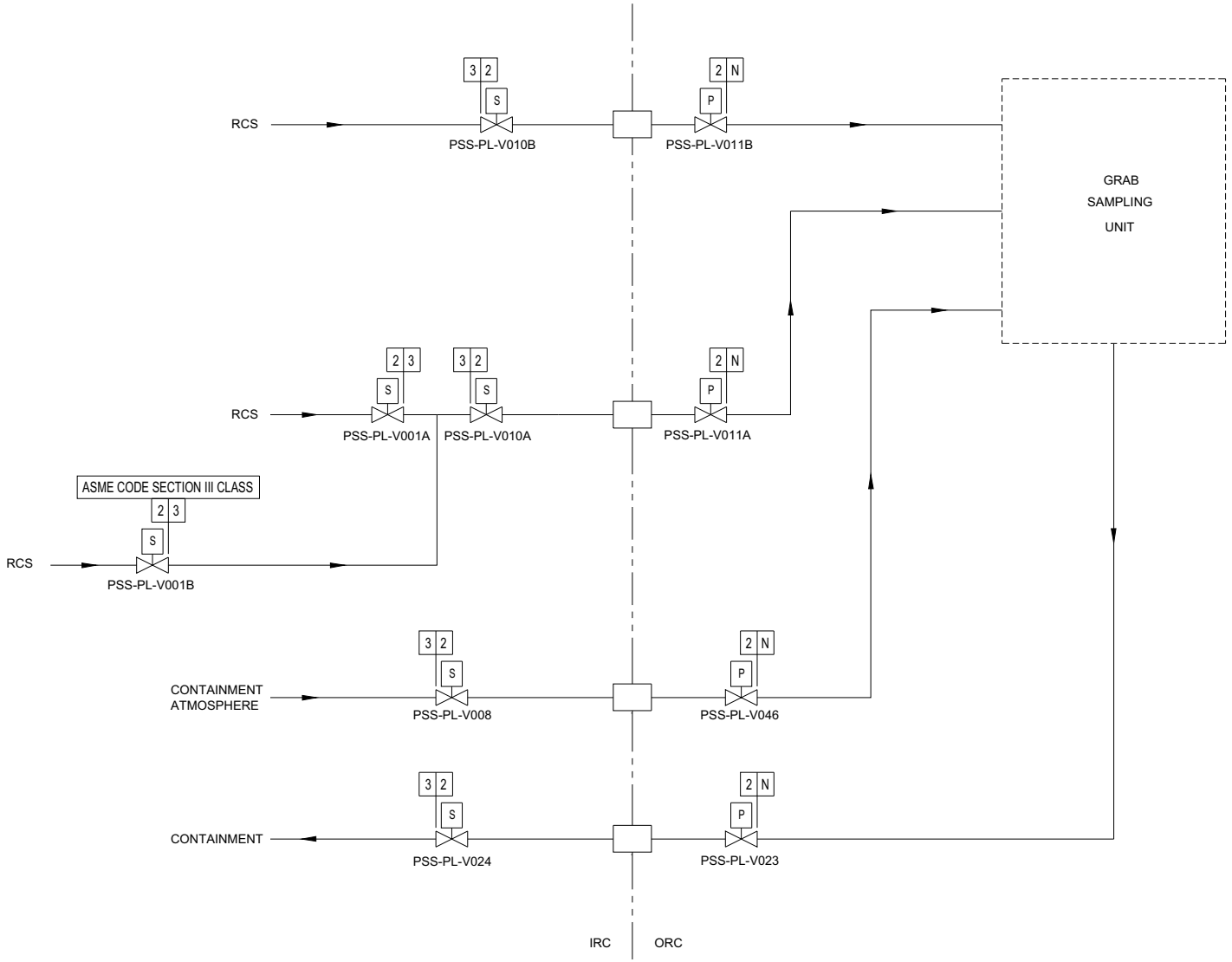

Figure 2.3.13-1
Primary Sampling System

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2.3.14 Demineralized Water Transfer and Storage System

Design Description

The demineralized water transfer and storage system (DWS) receives water from the demineralized water treatment system (DTS), and provides a reservoir of demineralized water to supply the condensate storage tank and for distribution throughout the plant. Demineralized water is processed in the DWS to remove dissolved oxygen. In addition to supplying water for makeup of systems which require pure water, the demineralized water is used to sluice spent radioactive resins from the ion exchange vessels in the chemical and volume control system (CVS), the spent fuel pool cooling system (SFS), and the liquid radwaste system (WLS) to the solid radwaste system (WSS).

The component locations of the DWS are as shown in Table 2.3.14-3.

1. The functional arrangement of the DWS is as described in the Design Description of this Section 2.3.14.

2. The DWS provides the safety-related function of preserving containment integrity by isolation of the DWS lines penetrating the containment.

3. The DWS condensate storage tank (CST) provides the nonsafety-related function of water supply to the FWS startup feedwater pumps.

4. Displays of the parameters identified in Table 2.3.14-1 can be retrieved in the main control room (MCR).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.14-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the DWS.

Table 2.3.14-1

Equipment Name	Tag No.	Display	Control Function
Condensate Storage Tank Water Level	DWS-006	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.14-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. Not used.		
3. The DWS CST provides the nonsafety-related function of water supply to the FWS startup feedwater pumps.	Inspection of the DWS CST will be performed.	The volume of the CST between the tank overflow and the startup feedwater pumps supply connection is greater than or equal to 325,000 gallons.
4. Displays of the parameters identified in Table 2.3.14-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.3.14-1 can be retrieved in the MCR.

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Table 2.3.14-3

Component Name	Tag No.	Component Location
Demineralizer Water Storage Tank Degasification System Package	DWS-MS-01	Annex Building
Condensate Storage Tank Degasification System Package	DWS-MS-02	Turbine Building
Demineralized Water Storage Tank	DWS-MT-01	Yard
Condensate Storage Tank	DWS-MT-02	Yard

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2.3.15 Compressed and Instrument Air System

Design Description

The compressed and instrument air system (CAS) consists of three subsystems: instrument air, service air, and high-pressure air. The instrument air subsystem supplies compressed air for air-operated valves and dampers. The service air subsystem supplies compressed air at outlets throughout the plant to power air-operated tools and is used as a motive force for air-powered pumps. The service air subsystem is also utilized as a supply source for breathing air. The high-pressure air subsystem supplies air to the main control room emergency habitability system (VES) and fire fighting apparatus recharge station.

The CAS is required for normal operation and startup of the plant.

The component locations of the CAS are as shown in Table 2.3.15-3.

1. The functional arrangement of the CAS is as described in the Design Description of this Section 2.3.15.

2. The CAS provides the safety-related function of preserving containment integrity by isolation of the CAS lines penetrating the containment.

3. Displays of the parameters identified in Table 2.3.15-1 can be retrieved in the main control room (MCR).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.15-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the CAS.

Table 2.3.15-1

Equipment Name	Tag No.	Display	Control Function
Instrument Air Pressure	CAS-011	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.3.15-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. Not used.		
3. Displays of the parameters identified in Table 2.3.15-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.3.15-1 can be retrieved in the MCR.

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Table 2.3.15-3

Component Name	Tag No.	Component Location
Instrument Air Compressor Package A	CAS-MS-01A	Turbine Building
Instrument Air Compressor Package B	CAS-MS-01B	Turbine Building
Instrument Air Dryer Package A	CAS-MS-02A	Turbine Building
Instrument Air Dryer Package B	CAS-MS-02B	Turbine Building
Service Air Compressor Package A	CAS-MS-03A	Turbine Building
Service Air Compressor Package B	CAS-MS-03B	Turbine Building
Service Air Dryer Package A	CAS-MS-04A	Turbine Building
Service Air Dryer Package B	CAS-MS-04B	Turbine Building
High Pressure Air Compressor and Filter Package	CAS-MS-05	Turbine Building
Instrument Air Receiver A	CAS-MT-01A	Turbine Building
Instrument Air Receiver B	CAS-MT-01B	Turbine Building
Service Air Receiver	CAS-MT-02	Turbine Building

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2.3.16 Potable Water System

No entry for this system.

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2.3.17 Waste Water System

No entry for this system.

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2.3.18 Plant Gas System

No entry. Covered in Section 3.3, Buildings.

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2.3.19 Communication System

Design Description

The communication system (EFS) provides intraplant communications during normal, maintenance, transient, fire, and accident conditions, including loss of offsite power.

1. a) The EFS has handsets, amplifiers, loudspeakers, and siren tone generators connected as a telephone/page system.

   b) The EFS has sound-powered equipment connected as a system.

2. The EFS provides the following nonsafety-related functions:

   a) The EFS telephone/page system provides intraplant, station-to-station communications and area broadcasting between the main control room (MCR) and the locations listed in Table 2.3.19-1.

   b) The EFS provides sound-powered communications between the MCR, the remote shutdown workstation (RSW), the Division A, B, C, D dc equipment rooms (Rooms 12201/12203/12205/12207), the Division A, B, C, D I&C rooms (Rooms 12301/12302/12304/12305), and the diesel generator building (Rooms 60310/60320) without external power.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.19-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the EFS.

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Table 2.3.19-1

Telephone/Page System Equipment	Location
Fuel Handling Area	12562
Division A, B, C, D dc Equipment Rooms	12201/12203/12205/12207
Division A, B, C, D I&C Rooms	12301/12302/12304/12305
Maintenance Floor Staging Area	12351
Containment Maintenance Floor	11300
Containment Operating Deck	11500

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Table 2.3.19-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.a) The EFS has handsets, amplifiers, loudspeakers, and siren tone generators connected as a telephone/page system.	Inspection of the as-built system will be performed.	The as-built EFS has handsets, amplifiers, loudspeakers, and siren tone generators connected as a telephone/page system.
1.b) The EFS has sound-powered equipment connected as a system.	Inspection of the as-built system will be performed.	The as-built EFS has sound-powered equipment connected as a system.
2.a) The EFS telephone/page system provides intraplant, station-to-station communications and area broadcasting between the MCR and the locations listed in Table 2.3.19-1.	An inspection and test will be performed on the telephone/page communication equipment.	Telephone/page equipment is installed and voice transmission and reception from the MCR are accomplished.
2.b) EFS provides sound-powered communications between the MCR, the RSW, the Division A, B, C, D dc equipment rooms (Rooms 12201/12203/12205/12207), the Division A, B, C, D I&C rooms (Rooms 12301/12302/12304/12305), and the diesel generator building (Rooms 60310/60320) without external power.	An inspection and test will be performed of the sound-powered communication equipment.	Sound-powered equipment is installed and voice transmission and reception are accomplished.

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2.3.20 Turbine Building Closed Cooling Water System

No entry for this system.

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2.3.21 Secondary Sampling System

No entry for this system.

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2.3.22 Containment Leak Rate Test System

No entry. Covered in Section 2.2.1, Containment System.

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2.3.24 Demineralized Water Treatment System

No entry for this system.

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2.3.25 Gravity and Roof Drain Collection System

No entry for this system.

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2.3.27 Sanitary Drainage System

No entry for this system.

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2.3.28 Turbine Island Vents, Drains, and Relief System

No entry for this system.

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2.3.29 Radioactive Waste Drain System

Design Description

The radioactive waste drain system (WRS) collects radioactive and potentially radioactive liquid wastes from equipment and floor drains during normal operation, startup, shutdown, and refueling. The liquid wastes are then transferred to appropriate processing and disposal systems.

Nonradioactive wastes are collected by the waste water system (WWS). The WRS is as shown in Figure 2.3.29-1.

1. The functional arrangement of the WRS is as described in the Design Description of this Section 2.3.29.

2. The WRS collects liquid wastes from the equipment and floor drainage of the radioactive portions of the auxiliary building, annex building, and radwaste building and directs these wastes to a WRS sump or WLS waste holdup tanks located in the auxiliary building.

3. The WRS collects chemical wastes from the auxiliary building chemical laboratory drains and the decontamination solution drains in the annex building and directs these wastes to the chemical waste tank of the liquid radwaste system.

4. The WWS stops the discharge from the turbine building sumps upon detection of high radiation in the discharge stream to the oil separator.

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Table 2.3.29-1
Inspection, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspection, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The WRS collects liquid wastes from the equipment and floor drainage of the radioactive portions of the auxiliary building, annex building, and radwaste building and directs these wastes to a WRS sump or WLS waste holdup tanks located in the auxiliary building.	A test is performed by pouring water into the equipment and floor drains in the radioactive portions of the auxiliary building, annex building, and radwaste building.	The water poured into these drains is collected either in the auxiliary building radioactive drains sump or the WLS waste holdup tanks.
3. The WRS collects chemical wastes from the auxiliary building chemical laboratory drains and the decontamination solution drains in the annex building and directs these wastes to the chemical waste tank of the liquid radwaste system.	A test is performed by pouring water into the auxiliary building chemical laboratory and the decontamination solution drains in the annex building.	The water poured into these drains is collected in the chemical waste tank of the liquid radwaste system.
4. The WWS stops the discharge from the turbine building sumps upon detection of high radiation in the discharge stream to the oil separator.	Tests will be performed to confirm that a simulated high radiation signal from the turbine building sump discharge radiation monitor, WWS-021 causes the sump pumps (WWS-MP-01A and B, and WWS-MP-07A and B) to stop operating, stopping the spread of radiation outside of the turbine building.	A simulated high radiation signal causes the turbine building sump pumps (WWS-MP-01A and B, and WWS-MP-07A and B) to stop operating, stopping the spread of radiation outside of the turbine building.

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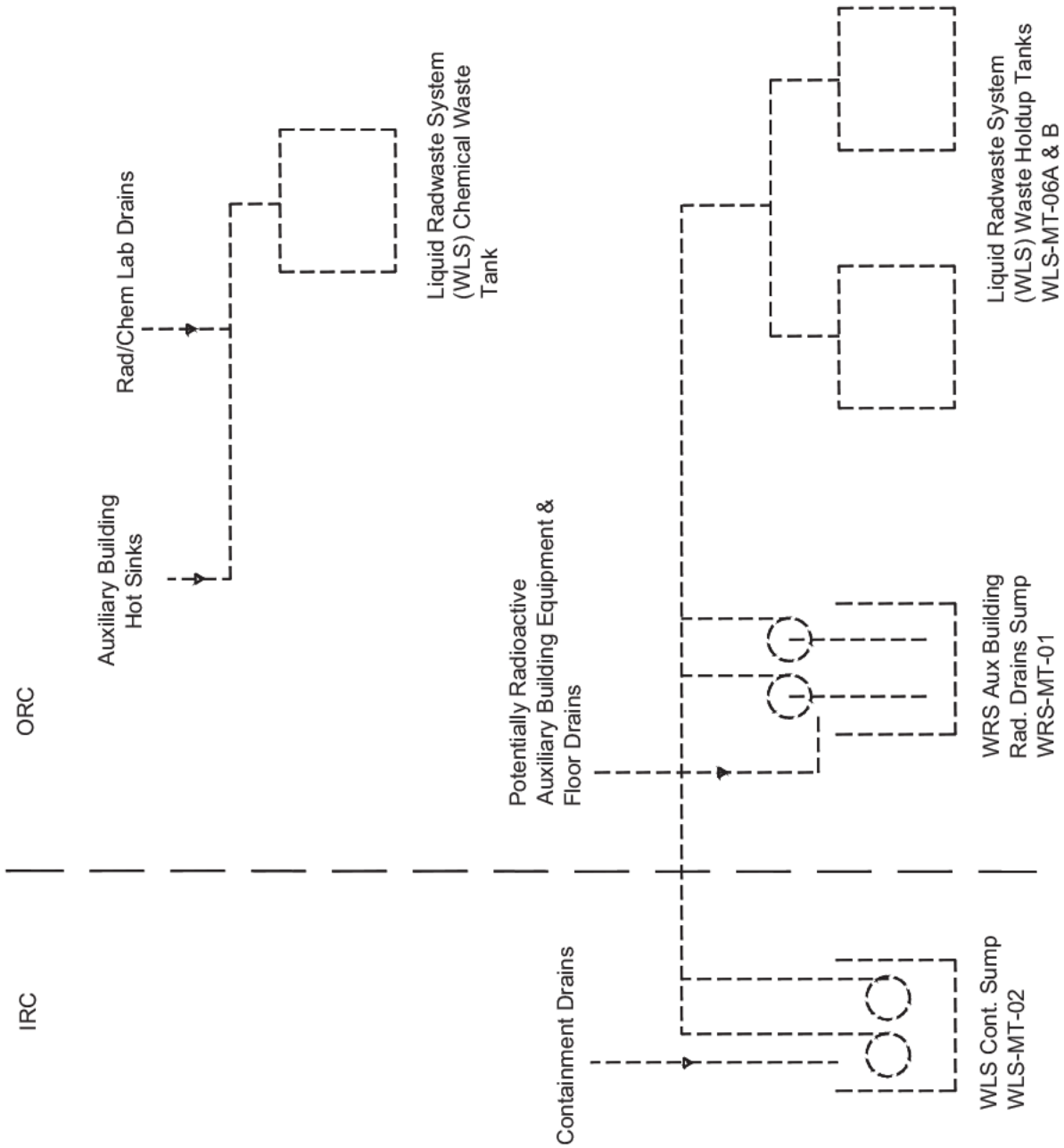

Figure 2.3.29-1
Radioactive Waste Drain System

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2.4 Steam and Power Conversion Systems

2.4.1 Main and Startup Feedwater System

See Section 2.2.4 for information on the main feedwater system.

Design Description

The startup feedwater system supplies feedwater to the steam generators during plant startup, hot standby and shutdown conditions, and during transients in the event of main feedwater system unavailability.

1. The functional arrangement of the startup feedwater system is as described in the Design Description of this Section 2.4.1.

2. The FWS provides the following nonsafety-related functions:

   The FWS provides startup feedwater flow from the condensate storage tank (CST) to the steam generator system (SGS) for heat removal from the RCS.

3. Controls exist in the main control room (MCR) to cause the components identified in Table 2.4.1-1 to perform the listed function.

4. Displays of the parameters identified in Table 2.4.1-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.4.1-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the FWS.

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Table 2.4.1-1

Equipment Name	Tag No.	Display	Control Function
Startup Feedwater Pump A (Motor)	FWS-MP-03A	Yes (Run Status)	Start
Startup Feedwater Pump B (Motor)	FWS-MP-03B	Yes (Run Status)	Start
Startup Feedwater Pump A Isolation Valve	FWS-PL-V013A	Yes (Valve Position)	Open
Startup Feedwater Pump B Isolation Valve	FWS-PL-V013B	Yes (Valve Position)	Open

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Table 2.4.1-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The FWS provides startup feedwater flow from the CST to the SGS for heat removal from the RCS.	Testing will be performed to confirm that each of the startup feedwater pumps can provide water from the CST to both steam generators.	Each FWS startup feedwater pump provides a flow rate greater than or equal to 260 gpm to each steam generator system at a steam generator secondary side pressure of at least 1106 psia.
3. Controls exist in the MCR to cause the components identified in Table 2.4.1-1 to perform the listed function.	Testing will be performed on the components in Table 2.4.1-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.4.1-1 to perform the listed functions.
4. Displays of the parameters identified in Table 2.4.1-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.4.1-1 can be retrieved in the MCR.

Table 2.4.1-3

Component Name	Tag No.	Component Location
Startup Feedwater Pump A	FWS-MP-03A	Turbine Building
Startup Feedwater Pump B	FWS-MP-03B	Turbine Building

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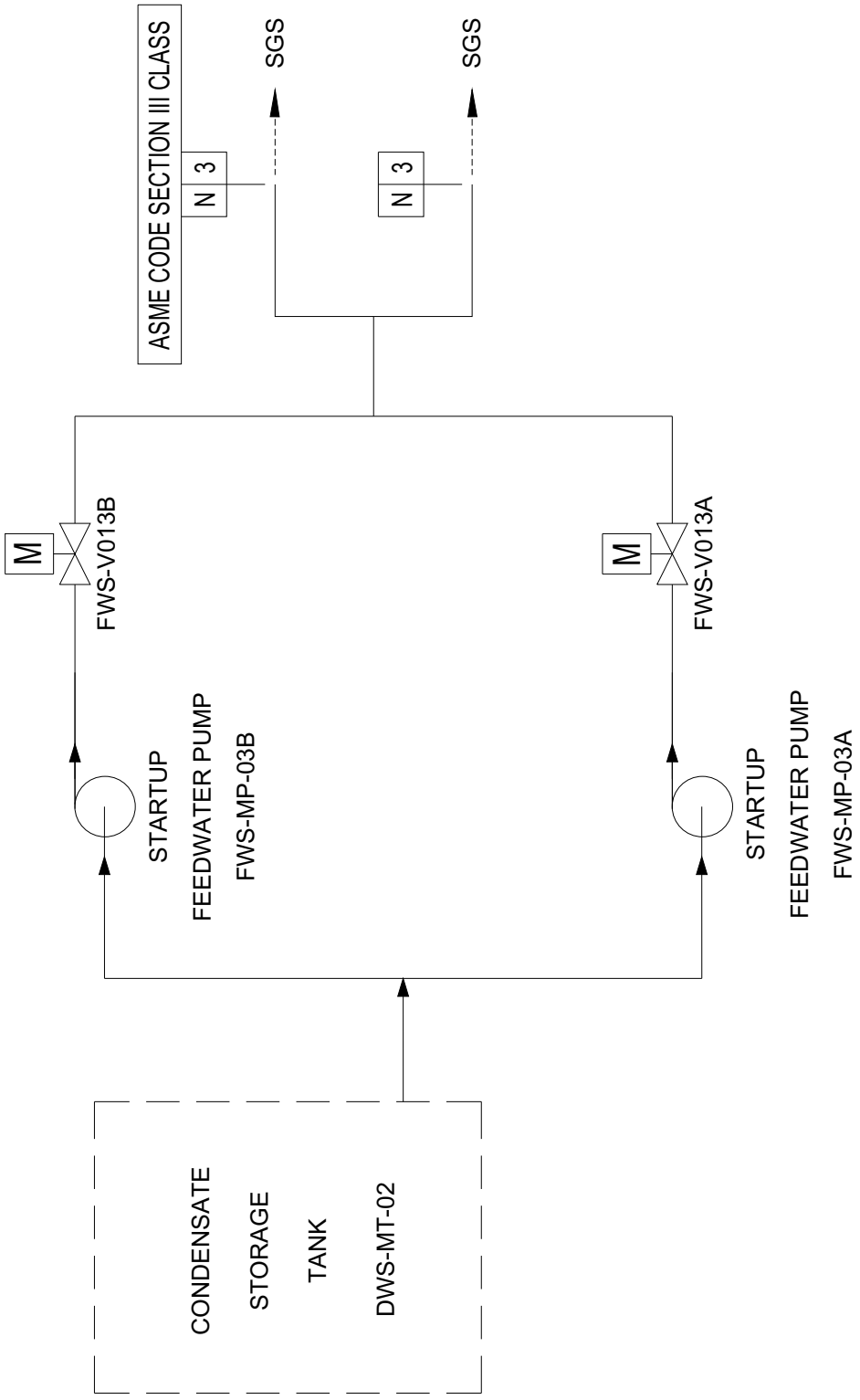

Figure 2.4.1-1
Main and Startup Feedwater System

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2.4.2 Main Turbine System

Design Description

The main turbine system (MTS) is designed for electric power production consistent with the capability of the reactor and the reactor coolant system.

The component locations of the MTS are as shown in Table 2.4.2-2.

1. The functional arrangement of the MTS is as described in the Design Description of this Section 2.4.2.

2. a) Controls exist in the MCR to trip the main turbine-generator.

   b) The main turbine-generator trips after receiving a signal from the PMS.

   c) The main turbine-generator trips after receiving a signal from the DAS.

3. The overspeed trips for the AP1000 turbine are set for 110% and 111% (±1% each). Each trip is initiated electrically in separate systems. The trip signals from the two turbine electrical overspeed protection trip systems are isolated from, and independent of, each other.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.4.2-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the MTS.

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Table 2.4.2-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
1. Not used.		
2.a) Controls exist in the MCR to trip the main turbine-generator.	Testing will be performed on the main turbine-generator using controls in the MCR.	Controls in the MCR operate to trip the main turbine-generator.
2.c) The main turbine-generator trips after receiving a signal from the DAS.	Testing will be performed using real or simulated signals into the DAS.	The main turbine-generator trips after receiving a signal from the DAS.
3) The trip signals from the two turbine electrical overspeed protection trip systems are isolated from, and independent of, each other.	ii) Testing of the as-built system will be performed using simulated signals from the turbine speed sensors.	ii) The main turbine-generator trips after overspeed signals are received from the speed sensors of the 110% emergency electrical overspeed trip system, and the main turbine-generator trips after overspeed signals are received from the speed sensors of the 111% backup electrical overspeed trip system.
2.b) The main turbine-generator trips after receiving a signal from the PMS.	Testing will be performed using real or simulated signals into the PMS.	The main turbine-generator trips after receiving a signal from the PMS.
3) The trip signals from the two turbine electrical overspeed protection trip systems are isolated from, and independent of, each other.	i) The system design will be reviewed.	i) The system design review shows that the trip signals of the two electrical overspeed protection trip systems are isolated from, and independent of, each other.
	iii) Inspection will be performed for the existence of a report verifying that the two turbine electrical overspeed protection systems have diverse hardware and software/firmware.	iii) A report exists and concludes that the two electrical overspeed protection systems have diverse hardware and software/firmware.

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Table 2.4.2-2

Component Name	Tag No.	Component Location
HP Turbine	MTS-MG-01	Turbine Building
LP Turbine A	MTS-MG-02A	Turbine Building
LP Turbine B	MTS-MG-02B	Turbine Building
LP Turbine C	MTS-MG-02C	Turbine Building
Gland Steam Condenser	GSS-ME-01	Turbine Building
Gland Condenser Vapor Exhauster 1A	GSS-MA-01A	Turbine Building
Gland Condenser Vapor Exhauster 1B	GSS-MA-01B	Turbine Building
Electrical Overspeed Trip Device	--	Turbine Building
Emergency Electrical Overspeed Trip Device	--	Turbine Building

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2.4.3 Main Steam System

No entry. Covered in Section 2.2.4, Steam Generator System.

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2.4.4 Steam Generator Blowdown System

No entry. Containment isolation function covered in Section 2.2.1, Containment System and 2.2.4, Steam Generator System.

No entry. Steam generator isolation function covered in Section 2.2.4, Steam Generator System.

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2.4.5 Condenser Air Removal System

No entry. Covered in Section 3.5, Radiation Monitoring.
(Note: Monitor is TDS-RE001.)

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2.4.6 Condensate System

Design Description

The condensate system (CDS) provides feedwater at the required temperature, pressure, and flow rate to the deaerator. Condensate is pumped from the main condenser hotwell by the condensate pumps and passes through the low-pressure feedwater heaters to the deaerator. The circulating water system (CWS) removes heat from the condenser and is site specific starting from the interface at the locations where the CWS piping enters and exits the turbine building.

The CDS operates during plant startup and power operations (full and part loads).

The component locations of the CDS are as shown in Table 2.4.6-3.

1. The functional arrangement of the CDS is as described in the Design Description of this Section 2.4.6.

2. Displays of the parameters identified in Table 2.4.6-1 can be retrieved in the main control room (MCR).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.4.6-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the CDS.

Table 2.4.6-1

Equipment Name	Tag No.	Display
Condenser Backpressure	CDS-056A	Yes
Condenser Backpressure	CDS-056B	Yes
Condenser Backpressure	CDS-056C	Yes

Table 2.4.6-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. Displays of the parameters identified in Table 2.4.6-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.4.6-1 can be retrieved in the MCR.

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Table 2.4.6-3

Component Name	Component Location
Low Pressure Feedwater Heaters	Turbine Building
Deaerator Feedwater Heater and Storage Tank	Turbine Building
Main Condenser Shell A	Turbine Building
Main Condenser Shell B	Turbine Building
Main Condenser Shell C	Turbine Building
Condensate Pump A	Turbine Building
Condensate Pump B	Turbine Building
Condensate Pump C	Turbine Building

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2.4.7 Circulating Water System

No entry for this system.

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2.4.8 Auxiliary Steam Supply System

No entry for this system.

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2.4.9 Condenser Tube Cleaning System

No entry for this system.

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2.4.10 Turbine Island Chemical Feed System

No entry for this system.

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2.4.11 Condensate Polishing System

No entry for this system.

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2.4.12 Gland Seal System

No entry. Covered in Section 2.4.2, Main Turbine System.

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2.4.13 Generator Hydrogen and CO_2 System

No entry for this system.

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2.4.14 Heater Drain System

No entry for this system.

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2.4.15 Hydrogen Seal Oil System

No entry for this system.

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2.4.16 Main Turbine and Generator Lube Oil System

No entry for this system.

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2.5 Instrumentation and Control Systems

2.5.1 Diverse Actuation System

Design Description

The diverse actuation system (DAS) initiates reactor trip, actuates selected functions, and provides plant information to the operator.

The component locations of the DAS are as shown in Table 2.5.1-5.

1. The functional arrangement of the DAS is as described in the Design Description of this Section 2.5.1.

2. The DAS provides the following nonsafety-related functions:

    a) The DAS provides an automatic reactor trip on low wide-range steam generator water level, or on low pressurizer water level, or on high hot leg temperature, separate from the PMS.

    b) The DAS provides automatic actuation of selected functions, as identified in Table 2.5.1-1, separate from the PMS.

    c) The DAS provides manual initiation of reactor trip and selected functions, as identified in Table 2.5.1-2, separate from the PMS. These manual initiation functions are implemented in a manner that bypasses the control room multiplexers, if any; the PMS cabinets; and the signal processing equipment of the DAS.

    d) The DAS provides main control room (MCR) displays of selected plant parameters, as identified in Table 2.5.1-3, separate from the PMS.

3. The DAS has the following features:

    a) The signal processing hardware of the DAS uses input modules, output modules, and microprocessor or special purpose logic processor boards that are different than those used in the PMS.

    b) The display hardware of the DAS uses a different display device than that used in the PMS.

    c) Software diversity between DAS and PMS will be achieved through the use of different algorithms, logic, program architecture, executable operating system, and executable software/logic.

    d) The DAS has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency (RFI), and electrostatic discharge (ESD) conditions that exist where the DAS equipment is located in the plant.

    e) The sensors identified on Table 2.5.1-3 are used for DAS input and are separate from those being used by the PMS and plant control system.

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f) The DAS is powered by non-Class 1E uninterruptible power supplies that are independent and separate from the power supplies which power the PMS.

g) The DAS signal processing cabinets are provided with the capability for channel testing without actuating the controlled components.

h) The DAS equipment can withstand the room ambient temperature and humidity conditions that will exist at the plant locations in which the DAS equipment is installed at the times for which the DAS is designed to be operational.

4. The DAS hardware and any software are developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages:

   a) Development phase for hardware and any software
   b) System test phase
   c) Installation phase

The planned design process also provides for the use of commercial off-the-shelf hardware and software.

5. The DAS manual actuation of ADS, IRWST injection, and containment recirculation can be executed correctly and reliably.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5.1-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the DAS.

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Table 2.5.1-1
Functions Automatically Actuated by the DAS

1. Reactor and Turbine Trip on Low Wide-range Steam Generator Water Level or Low Pressurizer Water Level or High Hot Leg Temperature
2. Passive Residual Heat Removal (PRHR) Actuation and In-containment Refueling Water Storage Tank (IRWST) Gutter Isolation on Low Wide-range Steam Generator Water Level or on High Hot Leg Temperature
3. Core Makeup Tank (CMT) Actuation and Trip All Reactor Coolant Pumps on Low Wide-Range Steam Generator Water Level or Low Pressurizer Water Level
4. Isolation of Selected Containment Penetrations and Initiation of Passive Containment Cooling System (PCS) on High Containment Temperature

Table 2.5.1-2
Functions Manually Actuated by the DAS

1. Reactor and Turbine Trip
2. PRHR Actuation and IRWST Gutter Isolation
3. CMT Actuation and Trip All Reactor Coolant Pumps
4. First-stage Automatic Depressurization System (ADS) Valve Actuation
5. Second-stage ADS Valve Actuation
6. Third-stage ADS Valve Actuation
7. Fourth-stage ADS Valve Actuation
8. PCS Actuation
9. Isolation of Selected Containment Penetrations
10. Containment Hydrogen Igniter Actuation
11. IRWST Injection Actuation
12. Containment Recirculation Actuation
13. Actuate IRWST Drain to Containment

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Table 2.5.1-3
DAS Sensors and Displays

Equipment Name	Tag Number
Reactor Coolant System (RCS) Hot Leg Temperature	RCS-300A
RCS Hot Leg Temperature	RCS-300B
Steam Generator 1 Wide-range Level	SGS-044
Steam Generator 1 Wide-range Level	SGS-045
Steam Generator 2 Wide-range Level	SGS-046
Steam Generator 2 Wide-range Level	SGS-047
Pressurizer Water Level	RCS-305A
Pressurizer Water Level	RCS-305B
Containment Temperature	VCS-053A
Containment Temperature	VCS-053B
Core Exit Temperature	IIS-009
Core Exit Temperature	IIS-013
Core Exit Temperature	IIS-030
Core Exit Temperature	IIS-034
Rod Control Motor Generator Voltage	PLS-001
Rod Control Motor Generator Voltage	PLS-002

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Table 2.5.1-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) The DAS provides an automatic reactor trip on low wide-range steam generator water level, or on low pressurizer water level, or on high hot leg temperature, separate from the PMS.	Electrical power to the PMS equipment will be disconnected and an operational test of the as-built DAS will be performed using real or simulated test signals.	The generator field control relays (contained in the control cabinets for the rod drive motor-generator sets) open after the test signal reaches the specified limit.
2.b) The DAS provides automatic actuation of selected functions, as identified in Table 2.5.1-1, separate from the PMS.	Electrical power to the PMS equipment will be disconnected and an operational test of the as-built DAS will be performed using real or simulated test signals.	Appropriate DAS output signals are generated after the test signal reaches the specified limit.
2.c) The DAS provides manual initiation of reactor trip, and selected functions, as identified in Table 2.5.1-2, separate from the PMS. These manual initiation functions are implemented in a manner that bypasses the control room multiplexers, if any; the PMS cabinets; and the signal processing equipment of the DAS.	Electrical power to the control room multiplexers, if any, and PMS equipment will be disconnected and the outputs from the DAS signal processing equipment will be disabled. While in this configuration, an operational test of the as-built system will be performed using the DAS manual actuation controls.	i) The generator field control relays (contained in the control cabinets for the rod drive motor-generator sets) open after reactor and turbine trip manual initiation controls are actuated. ii) DAS output signals are generated for the selected functions, as identified in Table 2.5.1-2, after manual initiation controls are actuated.
2.d) The DAS provides MCR displays of selected plant parameters, as identified in Table 2.5.1-3, separate from the PMS.	Electrical power to the PMS equipment will be disconnected and inspection will be performed for retrievability of the selected plant parameters in the MCR.	The selected plant parameters can be retrieved in the MCR.
3.f) The DAS is powered by non-Class 1E uninterruptible power supplies that are independent and separate from the power supplies which power the PMS.	Electrical power to the PMS equipment will be disconnected. While in this configuration, a test will be performed by providing simulated test signals in the non-Class 1E uninterruptible power supplies.	A simulated test signal exists at the DAS equipment when the assigned non-Class 1E uninterruptible power supply is provided the test signal.
3.g) The DAS signal processing cabinets are provided with the capability for channel testing without actuating the controlled components.	Channel tests will be performed on the as built system.	The capability exists for testing individual DAS channels without propagating an actuation signal to a DAS controlled component.

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Table 2.5.1-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3.a) The signal processing hardware of the DAS uses input modules, output modules, and microprocessor or special purpose logic processor boards that are different than those used in the PMS.	Inspection of the as-built DAS and PMS signal processing hardware will be performed.	The DAS signal processing equipment uses input modules, output modules, and micro-processor or special purpose logic processor boards that are different than those used in the PMS. The difference may be a different design, use of different component types, or different manufacturers.
3.b) The display hardware of the DAS uses a different display device than that used in the PMS.	Inspection of the as-built DAS and PMS display hardware will be performed.	The DAS display hardware is different than the display hardware used in the PMS. The difference may be a different design, use of different component types, or different manufacturers.
3.c) Software diversity between the DAS and PMS will be achieved through the use of different algorithms, logic, program architecture, executable operating system, and executable software/logic.	Inspection of the DAS and PMS design documentation will be performed.	Any DAS algorithms, logic, program architecture, executable operating systems, and executable software/logic are different than those used in the PMS.
3.d) The DAS has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency (RFI), and electrostatic discharge (ESD) conditions that exist where the DAS equipment is located in the plant.	Type tests, analyses, or a combination of type tests and analyses will be performed on the equipment.	A report exists and concludes that the DAS equipment can withstand the SWC, EMI, RFI and ESD conditions that exist where the DAS equipment is located in the plant.
3.e) The sensors identified on Table 2.5.1-3 are used for DAS input and are separate from those being used by the PMS and plant control system.	Inspection of the as-built system will be performed except for the core exit temperature sensor installation.	The sensors identified on Table 2.5.1-3 are used by DAS and are separate from those being used by the PMS and plant control system.
3.h) The DAS equipment can withstand the room ambient temperature and humidity conditions that will exist at the plant locations in which the DAS equipment is installed at the times for which the DAS is designed to be operational.	Type tests, analyses, or a combination of type tests and analyses will be performed on the equipment.	A report exists and concludes that the DAS equipment can withstand the room ambient temperature and humidity conditions that will exist at the plant locations in which the DAS equipment is installed at the times for which the DAS is designed to be operational.

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Table 2.5.1-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The DAS hardware and any software are developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages: a) Development phase for hardware and any software b) System test phase c) Installation phase The planned design process also provides for the use of commercial off-the-shelf hardware and software.	Inspection will be performed of the process used to design the hardware and any software.	A report exists and concludes that the process defines the organizational responsibilities, activities, and configuration management controls for the following: a) Documentation and review of hardware and any software. b) Performance of tests and the documentation of test results during the system test phase. c) Performance of tests and inspections during the installation phase. The process also defines requirements for the use of commercial off-the-shelf hardware and software.
5. Not used.		

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Table 2.5.1-5

Component Name	Tag No.	Component Location
DAS Processor Cabinet 1	DAS-JD-001	Auxiliary Building
DAS Processor Cabinet 2	DAS-JD-002	Auxiliary Building
DAS Squib Valve Control Cabinet	DAS-JD-003	Auxiliary Building

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2.5.2 Protection and Safety Monitoring System

Design Description

The protection and safety monitoring system (PMS) initiates reactor trip and actuation of engineered safety features in response to plant conditions monitored by process instrumentation and provides safety-related displays. The PMS has the equipment identified in Table 2.5.2-1. The PMS has four divisions of Reactor Trip and Engineered Safety Features Actuation, and four divisions of safety-related post-accident parameter displays. The functional arrangement of the PMS is depicted in Figure 2.5.2-1 and the component locations of the PMS are as shown in Table 2.5.2-9.

1. The functional arrangement of the PMS is as described in the Design Description of this Section 2.5.2.

2. The seismic Category I equipment, identified in Table 2.5.2-1, can withstand seismic design basis loads without loss of safety function.

3. The Class 1E equipment, identified in Table 2.5.2-1, has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency interference (RFI), and electrostatic discharge (ESD) conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

4. The Class 1E equipment, identified in Table 2.5.2-1, can withstand the room ambient temperature, humidity, pressure, and mechanical vibration conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

5. a) The Class 1E equipment, identified in Table 2.5.2-1, is powered from its respective Class 1E division.

   b) Separation is provided between PMS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

6. The PMS provides the following safety-related functions:

   a) The PMS initiates an automatic reactor trip, as identified in Table 2.5.2-2, when plant process signals reach specified limits.

   b) The PMS initiates automatic actuation of engineered safety features, as identified in Table 2.5.2-3, when plant process signals reach specified limits.

   c) The PMS provides manual initiation of reactor trip and selected engineered safety features as identified in Table 2.5.2-4.

7. The PMS provides the following nonsafety-related functions:

   a) The PMS provides process signals to the plant control system (PLS) through isolation devices.

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b) The PMS provides process signals to the data display and processing system (DDS) through isolation devices.

c) Data communication between safety and nonsafety systems does not inhibit the performance of the safety function.

d) The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.

e) The PMS receives signals from non-safety equipment that provides interlocks for PMS test functions through isolation devices.

8. The PMS, in conjunction with the operator workstations, provides the following functions:

a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved in the main control room (MCR). The fixed position controls listed with a "Yes" in the "Control" column are provided in the MCR.

b) The PMS provides for the transfer of control capability from the MCR to the remote shutdown workstation (RSW) using multiple transfer switches. Each individual transfer switch is associated with only a single safety-related group or with nonsafety-related control capability.

c) Displays of the open/closed status of the reactor trip breakers can be retrieved in the MCR.

9. a) The PMS automatically removes blocks of reactor trip and engineered safety features actuation when the plant approaches conditions for which the associated function is designed to provide protection. These blocks are identified in Table 2.5.2-6.

b) The PMS two-out-of-four initiation logic reverts to a two-out-of-three coincidence logic if one of the four channels is bypassed. All bypassed channels are alarmed in the MCR.

c) The PMS does not allow simultaneous bypass of two redundant channels.

d) The PMS provides the interlock functions identified in Table 2.5.2-7.

10. Setpoints are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation.

11. The PMS hardware and software is developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages:

a) Design requirements phase, may be referred to as conceptual or project definition phase (Complete)

b) System definition phase

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c) Hardware and software development phase, consisting of hardware and software design and implementation

d) System integration and test phase

e) Installation phase

12. The PMS software is designed, tested, installed, and maintained using a process which incorporates a graded approach according to the relative importance of the software to safety and specifies requirements for:

    a) Software management including documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action.

    b) Software configuration management including historical records of software and control of software changes.

    c) Verification and validation including requirements for reviewer independence.

13. The use of commercial grade hardware and software items in the PMS is accomplished through a process that specifies requirements for:

    a) Review of supplier design control, configuration management, problem reporting, and change control.

    b) Review of product performance.

    c) Receipt acceptance of the commercial grade item.

    d) Final acceptance based on equipment qualification and software validation in the integrated system.

14. The Component Interface Module (CIM) is developed using a planned design process which provides for specific design documentation and reviews.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5.2-8 specifies the inspections, tests, analyses, and associated acceptance criteria for the PMS.

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Table 2.5.2-1
PMS Equipment Name and Classification

Equipment Name	Seismic Cat. I	Class 1E	Qual. for Harsh Envir.
PMS Cabinets, Division A	Yes	Yes	No
PMS Cabinets, Division B	Yes	Yes	No
PMS Cabinets, Division C	Yes	Yes	No
PMS Cabinets, Division D	Yes	Yes	No
Reactor Trip Switchgear, Division A	Yes	Yes	No
Reactor Trip Switchgear, Division B	Yes	Yes	No
Reactor Trip Switchgear, Division C	Yes	Yes	No
Reactor Trip Switchgear, Division D	Yes	Yes	No
MCR/RSW Transfer Panels	Yes	Yes	No
MCR Safety-related Display, Division A	Yes	Yes	No
MCR Safety-related Display, Division B	Yes	Yes	No
MCR Safety-related Display, Division C	Yes	Yes	No
MCR Safety-related Display, Division D	Yes	Yes	No
MCR Safety-related Controls	Yes	Yes	No

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Table 2.5.2-2
PMS Automatic Reactor Trips

Source Range High Neutron Flux Reactor Trip
Intermediate Range High Neutron Flux Reactor Trip
Power Range High Neutron Flux (Low Setpoint) Trip
Power Range High Neutron Flux (High Setpoint) Trip
Power Range High Positive Flux Rate Trip
Reactor Coolant Pump High-2 Bearing Water Temperature Trip
Overtemperature Delta-T Trip
Overpower Delta-T Trip
Pressurizer Low-2 Pressure Trip
Pressurizer High-2 Pressure Trip
Pressurizer High-3 Water Level Trip
Low-2 Reactor Coolant Flow Trip
Low-2 Reactor Coolant Pump Speed Trip
Low-2 Steam Generator Narrow Range Water Level Trip
High-3 Steam Generator Water Level Trip
Automatic or Manual Safeguards Actuation Trip
Automatic or Manual Depressurization System Actuation Trip
Automatic or Manual Core Makeup Tank (CMT) Injection Trip
Passive Residual Heat Removal (PRHR) Actuation Reactor Trip

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Table 2.5.2-3
PMS Automatically Actuated Engineered Safety Features

Safeguards Actuation
Containment Isolation
Automatic Depressurization System (ADS) Actuation
Main Feedwater Isolation
Reactor Coolant Pump Trip
CMT Injection
Turbine Trip (Isolated signal to nonsafety equipment)
Steam Line Isolation
Steam Generator Relief Isolation
Steam Generator Blowdown Isolation
Passive Containment Cooling Actuation
Startup Feedwater Isolation
Passive Residual Heat Removal (PRHR) Heat Exchanger Alignment
Block of Boron Dilution
Chemical and Volume Control System (CVS) Makeup Line Isolation
Steam Dump Block (Isolated signal to nonsafety equipment)
Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization
Auxiliary Spray and Purification Line and Zinc/Hydrogen Addition Isolation
Containment Air Filtration System Isolation
Normal Residual Heat Removal Isolation
Refueling Cavity and Spent Fuel Pool Cooling System (SFS) Isolation
In-Containment Refueling Water Storage Tank (IRWST) Injection
IRWST Containment Recirculation
CVS Letdown Isolation
Pressurizer Heater Block (Isolated signal to nonsafety equipment)
Containment Vacuum Relief
Component Cooling System Containment Isolation Valve Closure

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Table 2.5.2-4
PMS Manually Actuated Functions

Reactor Trip
Safeguards Actuation
Containment Isolation
Depressurization System Stages 1, 2, and 3 Actuation
Depressurization System Stage 4 Actuation
Feedwater Isolation
Core Makeup Tank Injection Actuation
Steam Line Isolation
Passive Containment Cooling Actuation
Passive Residual Heat Removal Heat Exchanger Alignment
IRWST Injection
Containment Recirculation Actuation
Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization
Steam Generator Relief Isolation
Chemical and Volume Control System Isolation
Normal Residual Heat Removal System Isolation
Containment Vacuum Relief

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Table 2.5.2-5
Minimum Inventory of Displays, Alerts, and Fixed Position Controls in the MCR

Description	Control	Display	Alert[1]
Neutron Flux	-	Yes	Yes
Neutron Flux Doubling[2]	-	No	Yes
Startup Rate	-	Yes	Yes
Reactor Coolant System (RCS) Pressure	-	Yes	Yes
Wide-range Hot Leg Temperature	-	Yes	No
Wide-range Cold Leg Temperature	-	Yes	Yes
RCS Cooldown Rate Compared to the Limit Based on RCS Pressure	-	Yes	Yes
Wide-range Cold Leg Temperature Compared to the Limit Based on RCS Pressure	-	Yes	Yes
Change of RCS Temperature by more than 5°F in the last 10 minutes	-	No	Yes
Containment Water Level	-	Yes	Yes
Containment Pressure	-	Yes	Yes
Pressurizer Water Level	-	Yes	Yes
Pressurizer Water Level Trend	-	Yes	No
Pressurizer Reference Leg Temperature	-	Yes	No
Reactor Vessel-Hot Leg Water Level	-	Yes	Yes
Pressurizer Pressure	-	Yes	No
Core Exit Temperature	-	Yes	Yes
RCS Subcooling	-	Yes	Yes
RCS Cold Overpressure Limit	-	Yes	Yes
IRWST Water Level	-	Yes	Yes
PRHR Flow	-	Yes	Yes
PRHR HX Outlet Temperature	-	Yes	Yes
PRHR HX Inlet Isolation and Control Valve Status	-	Yes	Yes

Note: Dash (-) indicates not applicable.

1. These parameters are used to generate visual alerts that identify challenges to the critical safety functions. For the main control room, the visual alerts are embedded in the safety-related displays as visual signals.

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Table 2.5.2-5 (cont.)
Minimum Inventory of Displays, Alerts, and Fixed Position Controls in the MCR

Description	Control	Display	Alert[1]
Passive Containment Cooling System (PCS) Storage Tank Water Level	-	Yes	No
PCS Cooling Flow	-	Yes	No
IRWST to Normal Residual Heat Removal System (RNS) Suction Valve Status[2]	-	Yes	Yes
Remotely Operated Containment Isolation Valve Status[2]	-	Yes	No
Containment Area High-range Radiation Level	-	Yes	Yes
Containment Pressure (Extended Range)	-	Yes	No
CMT Level	-	Yes	No
Manual Reactor Trip (also initiates turbine trip)	Yes	-	-
Manual Safeguards Actuation	Yes	-	-
Manual CMT Actuation	Yes	-	-
Manual MCR Emergency Habitability System Actuation	Yes	-	-
Manual ADS Stages 1, 2, and 3 Actuation	Yes	-	-
Manual ADS Stage 4 Actuation	Yes	-	-
Manual PRHR Actuation	Yes	-	-
Manual Containment Cooling Actuation	Yes	-	-
Manual IRWST Injection Actuation	Yes	-	-
Manual Containment Recirculation Actuation	Yes	-	-
Manual Containment Isolation	Yes	-	-
Manual Main Steam Line Isolation	Yes	-	-
Manual Feedwater Isolation	Yes	-	-
Manual Containment Vacuum Relief	Yes		
Manual ADS and IRWST Injection Unblock	Yes	-	-

Note: Dash (-) indicates not applicable.

2. These instruments are not required after 24 hours.

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Table 2.5.2-6
PMS Blocks

Reactor Trip Functions:
Source Range High Neutron Flux Reactor Trip
Intermediate Range High Neutron Flux Reactor Trip
Power Range High Neutron Flux (Low Setpoint) Trip
Pressurizer Low-2 Pressure Trip
Pressurizer High-3 Water Level Trip
Low-2 Reactor Coolant Flow Trip
Low-2 Reactor Coolant Pump Speed Trip
High-3 Steam Generator Water Level Trip
Low-2 Steam Generator Narrow Range Water Level Trip
Engineered Safety Features:
ADS and IRWST Injection Actuation
Automatic Safeguards
Containment Isolation
Main Feedwater Isolation
Reactor Coolant Pump Trip
Core Makeup Tank Injection
Steam Line Isolation
Startup Feedwater Isolation
Block of Boron Dilution
Chemical and Volume Control System Isolation
Chemical and Volume Control System Letdown Isolation
Refueling Cavity and Spent Fuel Pool Cooling System (SFS) Isolation
Steam Dump Block
Auxiliary Spray and Letdown Purification Line Isolation
Passive Residual Heat Removal Heat Exchanger Alignment
Normal Residual Heat Removal System Isolation

Table 2.5.2-7
PMS Interlocks

RNS Suction Valves
PRHR Heat Exchanger Inlet Isolation Valve
CMT Cold Leg Balance Line Isolation Valves
Containment Vacuum Relief Isolation Valves

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Table 2.5.2-8
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The seismic Category I equipment, identified in Table 2.5.2-1, can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.5.2-1 is located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.5.2-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
3. The Class 1E equipment, identified in Table 2.5.2-1, has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency interference (RFI), and electrostatic discharge (ESD) conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on the equipment.	A report exists and concludes that the Class 1E equipment identified in Table 2.5.2-1 can withstand the SWC, EMI, RFI, and ESD conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
4. The Class 1E equipment, identified in Table 2.5.2-1, can withstand the room ambient temperature, humidity, pressure, and mechanical vibration conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment identified in Table 2.5.2-1.	A report exists and concludes that the Class 1E equipment identified in Table 2.5.2-1 can withstand the room ambient temperature, humidity, pressure, and mechanical vibration conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
5.a) Not used.		

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Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.b) Not used.		
6.a) The PMS initiates an automatic reactor trip, as identified in Table 2.5.2-2, when plant process signals reach specified limits.	An operational test of the as-built PMS will be performed using real or simulated test signals.	i) The reactor trip switchgear opens after the test signal reaches the specified limit. This only needs to be verified for one automatic reactor trip function.
6.a) The PMS initiates an automatic reactor trip, as identified in Table 2.5.2-2, when plant process signals reach specified limits.	An operational test of the as-built PMS will be performed using real or simulated test signals.	ii) PMS output signals to the reactor trip switchgear are generated after the test signal reaches the specified limit. This needs to be verified for each automatic reactor trip function.
6.b) The PMS initiates automatic actuation of engineered safety features, as identified in Table 2.5.2-3, when plant process signals reach specified limits.	An operational test of the as-built PMS will be performed using real or simulated test signals.	Appropriate PMS output signals are generated after the test signal reaches the specified limit. These output signals remain following removal of the test signal. Tests from the actuation signal to the actuated device(s) are performed as part of the system-related inspection, test, analysis, and acceptance criteria.
6.c) The PMS provides manual initiation of reactor trip and selected engineered safety features as identified in Table 2.5.2-4.	An operational test of the as-built PMS will be performed using the PMS manual actuation controls.	ii) PMS output signals are generated for reactor trip and selected engineered safety features as identified in Table 2.5.2-4 after the manual initiation controls are actuated.

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Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved in the MCR. The fixed position controls listed with a "Yes" in the "Control" column are provided in the MCR.	i) An inspection will be performed for retrievability of plant parameters in the MCR. iii) An operational test of the as-built system will be performed using each MCR fixed position control.	i) The plant parameters listed in Table 2.5.2-5 with a "Yes" in the "Display" column, can be retrieved in the MCR. iii) For each test of an as-built fixed position control listed in Table 2.5.2-5 with a "Yes" in the "Control" column, an actuation signal is generated. Tests from the actuation signal to the actuated device(s) are performed as part of the system-related inspection, test, analysis and acceptance criteria.
8.c) Displays of the open/closed status of the reactor trip breakers can be retrieved in the MCR.	Inspection will be performed for retrievability of displays of the open/closed status of the reactor trip breakers in the MCR.	Displays of the open/closed status of the reactor trip breakers can be retrieved in the MCR.
9.a) The PMS automatically removes blocks of reactor trip and engineered safety features actuation when the plant approaches conditions for which the associated function is designed to provide protection. These blocks are identified in Table 2.5.2-6.	An operational test of the as-built PMS will be performed using real or simulated test signals.	The PMS blocks are automatically removed when the test signal reaches the specified limit.
9.b) The PMS two-out-of-four initiation logic reverts to a two-out-of-three coincidence logic if one of the four channels is bypassed. All bypassed channels are alarmed in the MCR.	An operational test of the as-built PMS will be performed.	The PMS two-out-of-four initiation logic reverts to a two-out-of-three coincidence logic if one of the four channels is bypassed. All bypassed channels are alarmed in the MCR.
9.c) The PMS does not allow simultaneous bypass of two redundant channels.	An operational test of the as-built PMS will be performed. With one channel in bypass, an attempt will be made to place a redundant channel in bypass.	The redundant channel cannot be placed in bypass.
6.c) The PMS provides manual initiation of reactor trip and selected engineered safety features as identified in Table 2.5.2-4.	An operational test of the as-built PMS will be performed using the PMS manual actuation controls.	i) The reactor trip switchgear opens after manual reactor trip controls are actuated.

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Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) The PMS provides process signals to the PLS through isolation devices.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.
7.b) The PMS provides process signals to the DDS through isolation devices.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.
7.c) Data communication between safety and nonsafety systems does not inhibit the performance of the safety function.	Type tests, analyses, or a combination of type tests and analyses of the PMS gateways will be performed.	A report exists and concludes that data communication between safety and nonsafety systems does not inhibit the performance of the safety function.
7.d) The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.	Type tests, analyses, or a combination of type tests and analyses of the PMS manual control circuits and algorithms will be performed.	A report exists and concludes that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.
7.e) The PMS receives signals from non-safety equipment that provides interlocks for PMS test functions through isolation devices.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.
8.a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved in the MCR. The fixed position controls listed with a "Yes" in the "Control" column are provided in the MCR.	ii) An inspection and test will be performed to verify that the plant parameters are used to generate visual alerts that identify challenges to critical safety functions.	ii) The plant parameters listed in Table 2.5.2-5 with a "Yes" in the "Alert" column are used to generate visual alerts that identify challenges to critical safety functions. The visual alerts actuate in accordance with their correct logic and values.

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Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.b) The PMS provides for the transfer of control capability from the MCR to the RSW using multiple transfer switches. Each individual transfer switch is associated with only a single safety-related group or with nonsafety-related control capability.	i) An inspection will be performed to verify that a transfer switch exists for each safety-related division and the nonsafety-related control capability. ii) An operational test of the as-built system will be performed to demonstrate the transfer of control capability from the MCR to the RSW.	i) A transfer switch exists for each safety-related division and the nonsafety-related control capability. ii) Actuation of each transfer switch results in an alarm in the MCR and RSW, the activation of operator control capability from the RSW, and the deactivation of operator control capability from the MCR for the associated safety-related division and nonsafety-related control capability.
9.d) The PMS provides the interlock functions identified in Table 2.5.2-7.	An operational test of the as-built PMS will be performed using real or simulated test signals.	Appropriate PMS output signals are generated as the interlock conditions are changed.
10. Setpoints are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation.	Inspection will be performed for a document that describes the methodology and input parameters used to determine the PMS setpoints.	A report exists and concludes that the PMS setpoints are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation.

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Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. The PMS hardware and software is developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages: a) Not used b) System definition phase c) Hardware and software development phase, consisting of hardware and software design and implementation d) System integration and test phase e) Installation phase	Inspection will be performed of the process used to design the hardware and software.	A report exists and concludes that the process defines the organizational responsibilities, activities, and configuration management controls for the following: a) Not used. b) Specification of functional requirements. c) Documentation and review of hardware and software. d) Performance of system tests and the documentation of system test results, including a response time test performed under maximum CPU loading to demonstrate that the PMS can fulfill its response time criteria. e) Performance of installation tests and inspections.

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Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. The PMS software is designed, tested, installed, and maintained using a process which incorporates a graded approach according to the relative importance of the software to safety and specifies requirements for: a) Software management including documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action. b) Software configuration management including historical records of software and control of software changes. c) Verification and validation including requirements for reviewer independence.	Inspection will be performed of the process used to design, test, install, and maintain the PMS software.	A report exists and concludes that the process establishes a method for classifying the PMS software elements according to their relative importance to safety and specifies requirements for software assigned to each safety classification. The report also concludes that requirements are provided for the following software development functions: a) Software management including documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action. Software management requirements may be documented in the software quality assurance plan, software management plan, software development plan, software safety plan, and software operation and maintenance plan; or these requirements may be combined into a single software management plan. b) Software configuration management including historical records of software and control of software changes. Software configuration management requirements are provided in the software configuration management plan. c) Verification and validation including requirements for reviewer independence. Verification and validation requirements are provided in the verification and validation plan.

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Table 2.5.2-8 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. The use of commercial grade computer hardware and software items in the PMS is accomplished through a process that specifies requirements for: a) Review of supplier design control, configuration management, problem reporting, and change control. b) Review of product performance. c) Receipt acceptance of the commercial grade item. d) Acceptance based on equipment qualification and software validation in the integrated system.	Inspection will be performed of the process defined to use commercial grade components in the application.	A report exists and concludes that the process has requirements for: a) Review of supplier design control, configuration management, problem reporting, and change control. b) Review of product performance. c) Receipt acceptance of the commercial grade item. d) Acceptance based on equipment qualification and software validation in the integrated system.
14. The Component Interface Module (CIM) is developed using a planned design process which provides for specific design documentation and reviews. {Design Acceptance Criteria}	An inspection and or an audit will be performed of the processes used to design the hardware, development software, qualification and testing.	A report exists and concludes that CIM meets the below listed life cycle stages. Life cycle stages: a. Design requirements phase, may be referred to as conceptual or project definition phase b. System definition phase c. Hardware and software development phase, consisting of hardware and software design and implementation d. System integration and test phase e. Installation phase

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Table 2.5.2-9

Component Name	Component Location
PMS Cabinets, Division A	Auxiliary Building
PMS Cabinets, Division B	Auxiliary Building
PMS Cabinets, Division C	Auxiliary Building
PMS Cabinets, Division D	Auxiliary Building
Reactor Trip Switchgear, Division A	Auxiliary Building
Reactor Trip Switchgear, Division B	Auxiliary Building
Reactor Trip Switchgear, Division C	Auxiliary Building
Reactor Trip Switchgear, Division D	Auxiliary Building
MCR/RSW Transfer Panels	Auxiliary Building
MCR Safety-related Displays	Auxiliary Building
MCR Safety-related Controls	Auxiliary Building

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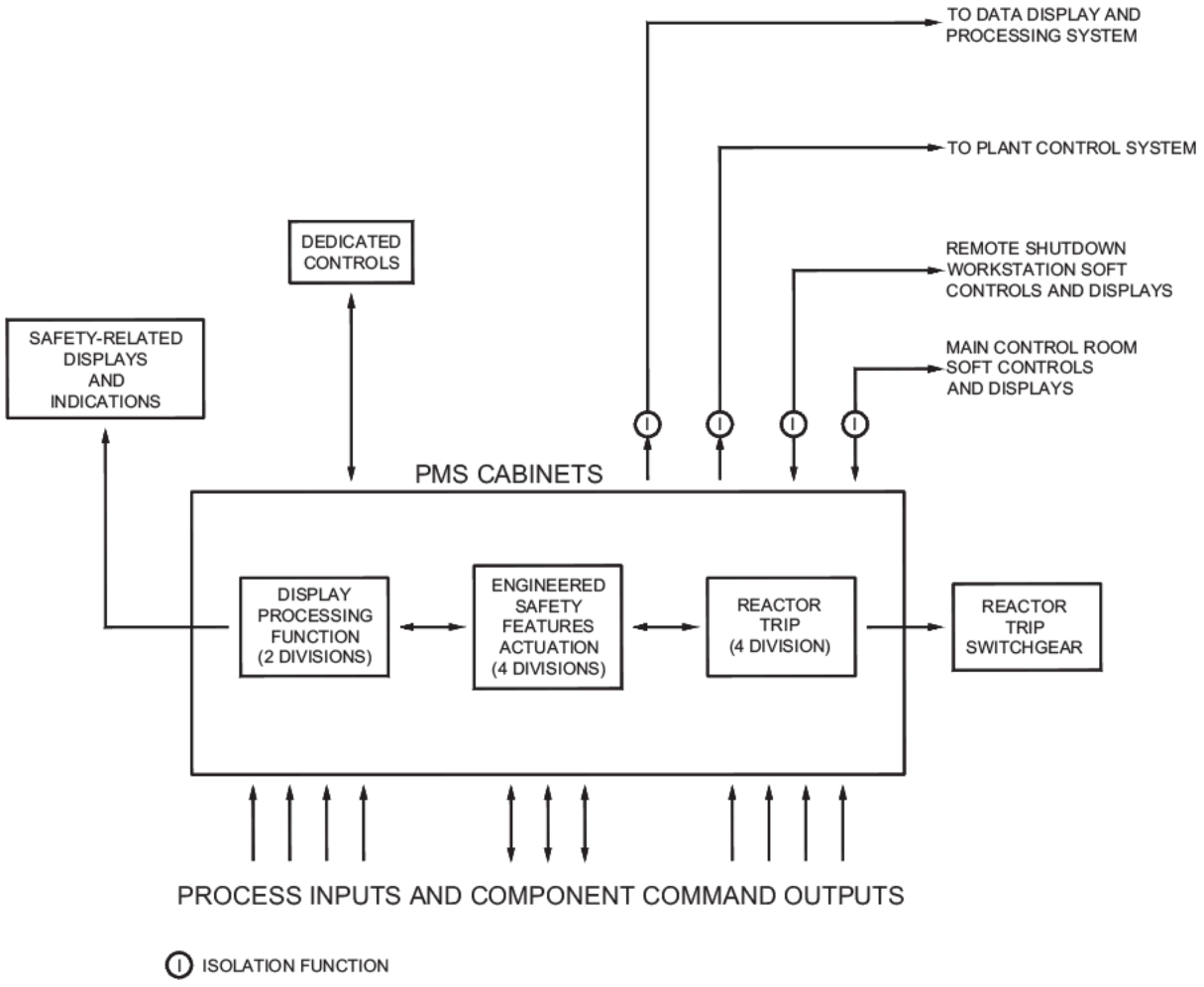

Figure 2.5.2-1
Protection and Safety Monitoring System

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2.5.3 Plant Control System

Design Description

The plant control system (PLS) provides for automatic and manual control of nonsafety-related plant components during normal and emergency plant operations. The PLS has distributed controllers and operator controls interconnected by computer data links or data highways.

1. The functional arrangement of the PLS is as described in the Design Description of this Section 2.5.3.

2. The PLS provides control interfaces for the control functions listed in Table 2.5.3-1.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5.3-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the PLS.

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Table 2.5.3-1
Control Functions Supported by the PLS

1. Reactor Power	5. Steam Generator Feedwater
2. Reactor Rod Position	6. Steam Dump
3. Pressurizer Pressure	7. Rapid Power Reduction
4. Pressurizer Water Level	

Table 2.5.3-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The PLS provides control interfaces for the control functions listed in Table 2.5.3-1.	An operational test of the system will be performed using simulated input signals. System outputs or component operations will be monitored to determine the operability of the control functions.	The PLS provides control interfaces for the control functions listed in Table 2.5.3-1.

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2.5.4 Data Display and Processing System

Design Description

The data display and processing system (DDS) provides nonsafety-related alarms and displays, analysis of plant data, plant data logging and historical storage and retrieval, and operational support for plant personnel. The DDS has distributed computer processors and video display units to support the data processing and display functions.

1. The functional arrangement of the DDS is as described in the Design Description of this Section 2.5.4.

2. The DDS, in conjunction with the operator workstations, provides the following function:

   The DDS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.4-1. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved at the remote shutdown workstation (RSW). The controls listed with a "Yes" in the "Control" column are provided at the RSW.

3. The DDS provides information pertinent to the status of the protection and safety monitoring system.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5.4-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the DDS.

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Table 2.5.4-1
Minimum Inventory of Controls, Displays, and Alerts at the RSW

Description	Control	Display	Alert(1)
Neutron Flux	-	Yes	Yes
Neutron Flux Doubling	-	No	Yes
Startup Rate	-	Yes	Yes
Reactor Coolant System (RCS) Pressure	-	Yes	Yes
Wide-range Hot Leg Temperature	-	Yes	No
Wide-range Cold Leg Temperature	-	Yes	Yes
RCS Cooldown Rate Compared to the Limit Based on RCS Pressure	-	Yes	Yes
Wide-range Cold Leg Temperature Compared to the Limit Based on RCS Pressure	-	Yes	Yes
Change of RCS Temperature by more than 5°F in the last 10 minutes	-	No	Yes
Containment Water Level	-	Yes	Yes
Containment Pressure	-	Yes	Yes
Pressurizer Water Level	-	Yes	Yes
Pressurizer Water Level Trend	-	Yes	No
Pressurizer Reference Leg Temperature	-	Yes	No
Reactor Vessel-Hot Leg Water Level	-	Yes	Yes
Pressurizer Pressure	-	Yes	No
Core Exit Temperature	-	Yes	Yes
RCS Subcooling	-	Yes	Yes
RCS Cold Overpressure Limit	-	Yes	Yes
In-containment Refueling Water Storage Tank (IRWST) Water Level	-	Yes	Yes
Passive Residual Heat Removal (PRHR) Flow	-	Yes	Yes
Manual Containment Vacuum Relief	Yes	-	-

Note: Dash (-) indicates not applicable.

1. These parameters are used to generate visual alerts that identify challenges to the critical safety functions. For the RSW, the visual alerts are embedded in the nonsafety-related displays as visual signals.

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Table 2.5.4-1 (cont.)
Minimum Inventory of Controls, Displays, and Alerts at the RSW

Description	Control	Display	Alert(1)
PRHR HX Outlet Temperature	-	Yes	Yes
PRHR HX Inlet Isolation and Control Valve Status	-	Yes	Yes
Passive Containment Cooling System (PCS) Storage Tank Water Level	-	Yes	No
PCS Cooling Flow	-	Yes	No
IRWST to Normal Residual Heat Removal System (RNS) Suction Valve Status	-	Yes	Yes
Remotely Operated Containment Isolation Valve Status	-	Yes	No
Containment Area High-range Radiation Level	-	Yes	Yes
Containment Pressure (Extended Range)	-	Yes	No
Core Makeup Tank (CMT) Level	-	Yes	No
Manual Reactor Trip (also initiates turbine trip)	Yes	-	-
Manual Safeguards Actuation	Yes	-	-
Manual CMT Actuation	Yes	-	-
Manual Automatic Depressurization System (ADS) Stages 1, 2, and 3 Actuation	Yes	-	-
Manual ADS Stage 4 Actuation	Yes	-	-
Manual PRHR Actuation	Yes	-	-
Manual Containment Cooling Actuation	Yes	-	-
Manual IRWST Injection Actuation	Yes	-	-
Manual Containment Recirculation Actuation	Yes	-	-
Manual Containment Isolation	Yes	-	-
Manual Main Steam Line Isolation	Yes	-	-
Manual Feedwater Isolation	Yes	-	-
Manual Containment Hydrogen Igniter (Nonsafety-related)(2)	Yes	-	-

Note: Dash (-) indicates not applicable.

1. These parameters are used to generate visual alerts that identify challenges to the critical safety functions. For the RSW, the visual alerts are embedded in the nonsafety-related displays as visual signals.

2. Containment hydrogen igniter control is provided as a “soft” control.

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Table 2.5.4-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the DDS is as described in the Design Description of this Section 2.5.4.	Inspection of the as-built system will be performed.	The as-built DDS conforms with the functional arrangement as described in the Design Description of this Section 2.5.4.
2. The DDS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.4-1. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved at the RSW. The controls listed with a "Yes" in the "Control" column are provided at the RSW.	i) An inspection will be performed for retrievability of plant parameters at the RSW. ii) An inspection and test will be performed to verify that the plant parameters are used to generate visual alerts that identify challenges to critical safety functions. iii) An operational test of the as-built system will be performed using each RSW control.	i) The plant parameters listed in Table 2.5.4-1 with a "Yes" in the "Display" column can be retrieved at the RSW. ii) The plant parameters listed in Table 2.5.4-1 with a "Yes" in the "Alert" column are used to generate visual alerts that identify challenges to critical safety functions. The visual alerts actuate in accordance with their logic and values. iii) For each test of a control listed in Table 2.5.4-1 with a "Yes" in the "Control" column, an actuation signal is generated. Tests from the actuation signal to the actuated device(s) are performed as part of the system-related inspection, test, analysis and acceptance criteria.
3. The DDS provides information pertinent to the status of the protection and safety monitoring system.	Tests of the as-built system will be performed.	The as-built system provides displays of the bypassed and operable status of the protection and safety monitoring system.

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2.5.5 In-Core Instrumentation System

Design Description

The in-core instrumentation system (IIS) provides safety-related core exit thermocouple signals to the protection and safety monitoring system (PMS). The IIS also provides nonsafety-related core exit thermocouple signals to the diverse actuation system (DAS). The core exit thermocouples are housed in the core instrument assemblies. Multiple core instrument assemblies are used to provide radial coverage of the core. At least three core instrument assemblies are provided in each core quadrant.

1. The functional arrangement of the IIS is as described in the Design Description of this Section 2.5.5.

2. The seismic Category I equipment identified in Table 2.5.5-1 can withstand seismic design basis loads without loss of safety function.

3. a) The Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment can withstand environmental conditions that would exist before, during, and following a design basis accident without loss of safety function, for the time required to perform the safety function.

   b) The Class 1E cables between the Core Exit Temperature sensors and the connector plates have sheaths.

   c) For cables other than those covered by 3.b, separation is provided between IIS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

4. Safety-related displays of the parameters identified in Table 2.5.5-1 can be retrieved in the main control room (MCR).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5.5-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the IIS.

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Table 2.5.5-1

Equipment Name	Seismic Cat. I	ASME Code Classification	Class 1E	Qual. for Harsh Envir.	Safety-Related Display
Incore Thimble Assemblies (at least three assemblies in each core quadrant)	Yes	–	Yes(1)	Yes(1)	Core Exit Temperature(1)

Note: Dash (-) indicates not applicable.

1. Only applies to the safety-related assemblies. There are at least two safety-related assemblies in each core quadrant.

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Table 2.5.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The seismic Category I equipment identified in Table 2.5.5-1 can withstand seismic design basis dynamic loads without loss of safety function.	i) Not used. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) Not used. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of safety function. iii) A report exists and concludes that the equipment including anchorage is seismically bounded by the tested or analyzed conditions.
3.a) The Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function, for the time required to perform the safety function.	i) Type tests, analysis, or a combination of type tests and analysis will be performed on Class 1E equipment located in a harsh environment. ii) Inspection will be performed of the Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment. This equipment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function. ii) A report exists and concludes that the Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.5.5-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
3.b) The Class 1E cables between the Core Exit Temperature sensors and the connector plates have sheaths.	Inspection of the Class 1E cables will be performed.	The Class 1E cables between the Core Exit Temperature sensors and the connector plates have sheaths.
3.c) Not used.		

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Table 2.5.5-2 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Safety-related displays of the parameters identified in Table 2.5.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.5.5-1 can be retrieved in the MCR.

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2.5.6 Special Monitoring System

Design Description

The special monitoring system (SMS) monitors the reactor coolant system (RCS) for the occurrence of impacts characteristic of metallic loose parts. Metal impact monitoring sensors are provided to monitor the RCS at the upper and lower head region of the reactor pressure vessel, and at the reactor coolant inlet region of each steam generator.

1. The functional arrangement of the SMS is as described in the Design Description of this Section 2.5.6.

2. Data obtained from the metal impact monitoring sensors can be retrieved in the main control room (MCR).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5.6-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the SMS.

Table 2.5.6-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. Data obtained from the metal impact monitoring sensors can be retrieved in the MCR.	Inspection will be performed for retrievability of data from the metal impact monitoring sensors in the MCR.	Data obtained from the metal impact monitoring sensors can be retrieved in the MCR.

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2.5.7 Operation and Control Centers System

Design Description

The operation and control centers system (OCS) is developed and implemented based upon a human factors engineering (HFE) program. The human system interface (HSI) scope includes the design of the OCS and each of the HSI resources. For the purposes of the HFE program, the OCS includes the main control room, remote shutdown workstation, the local control stations, and the associated workstations for each of these centers. Implementation of the HFE program involves the completion of the human factors engineering analyses and plans described in Tier 1 Material Section 3.2, Human Factors Engineering.

Inspections, Tests, Analyses, and Acceptance Criteria

The inspections, tests, analyses, and associated acceptance criteria for the OCS are provided in Table 3.2-1.

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2.5.8 Radiation Monitoring System

No entry. Radiation monitoring function covered in Section 3.5, Radiation Monitoring.

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2.5.9 Seismic Monitoring System

Design Description

The seismic monitoring system (SJS) provides for the collection of seismic data in digital format, analysis of seismic data, notification of the operator if the ground motion exceeds a threshold value, and notification of the operator (after analysis of data) that a predetermined cumulative absolute velocity (CAV) has been exceeded. The SJS has at least four triaxial acceleration sensor units and a time-history analyzer and recording system. The time-history analyzer and recording system are located in the auxiliary building.

1. The functional arrangement of the SJS is as described in the Design Description of this Section 2.5.9.

2. The SJS can compute CAV and the 5 percent of critical damping response spectrum for frequencies between 1 and 10 Hertz.

3. The SJS has a dynamic range of 0.001g to 1.0g and a frequency range of 0.2 to 50 Hertz.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5.9-1 specifies the inspections, tests, analyses, and associated acceptance criteria for SJS.

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Table 2.5.9-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The SJS can compute CAV and the 5 percent of critical damping response spectrum for frequencies between 1 and 10 Hz.	Type tests using simulated input signals, analyses, or a combination of type tests and analyses, of the SJS time-history analyzer and recording system will be performed.	A report exists and concludes that the SJS time-history analyzer and recording system can record data at a sampling rate of at least 200 samples per second, that the pre-event recording time is adjustable from less than or equal to 1.2 seconds to greater than or equal to 15.0 seconds, and that the initiation value is adjustable from less than or equal to 0.002g to greater than or equal to 0.02g.
3. The SJS has a dynamic range of 0.001g to 1.0g and a frequency range of 0.2 to 50 Hertz.	Type tests, analyses, or a combination of type tests and analyses, of the SJS triaxial acceleration sensors will be performed.	A report exists and concludes that the SJS triaxial acceleration sensors have a dynamic range of at least 0.001g to 1.0g and a frequency range of at least 0.2 to 50 Hertz.

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2.5.10 Main Turbine Control and Diagnostic System

No entry. Covered in Section 2.4.2, Main Turbine System.

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2.6 Electrical Power Systems

2.6.1 Main ac Power System

Design Description

The main ac power system (ECS) provides electrical ac power to nonsafety-related loads and non-Class 1E power to the Class 1E battery chargers and regulating transformers during normal and off-normal conditions.

The ECS is as shown in Figures 2.6.1-1 and the component locations of the ECS are as shown in Table 2.6.1-5.

1. The functional arrangement of the ECS is as described in the Design Description of this Section 2.6.1.

2. The seismic Category I equipment identified in Table 2.6.1-1 can withstand seismic design basis loads without loss of safety function.

3. a) The Class 1E breaker control power for the equipment identified in Table 2.6.1-1 are powered from their respective Class 1E division.

   b) Separation is provided between ECS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

4. The ECS provides the following nonsafety-related functions:

   a) The ECS provides the capability for distributing non-Class 1E ac power from onsite sources (ZOS) to nonsafety-related loads listed in Table 2.6.1-2.

   b) The 6900 Vac circuit breakers in switchgear ECS-ES-1 and ECS-ES-2 open after receiving a signal from the onsite standby power system.

   c) Each standby diesel generator 6900 Vac circuit breaker closes after receiving a signal from the onsite standby power system.

   d) Each ancillary diesel generator unit is sized to supply power to long-term safety-related post-accident monitoring loads and control room lighting and ventilation through a regulating transformer; and for one passive containment cooling system (PCS) recirculation pump.

   e) The ECS provides two loss-of-voltage signals to the onsite standby power system (ZOS), one for each diesel-backed 6900 Vac switchgear bus.

   f) The ECS provides a reverse-power trip of the generator circuit breaker which is blocked for at least 15 seconds following a turbine trip.

5. Controls exist in the main control room (MCR) to cause the circuit breakers identified in Table 2.6.1-3 to perform the listed functions.

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6. Displays of the parameters identified in Table 2.6.1-3 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.1-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the ECS.

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Table 2.6.1-1

Equipment Name	Tag No.	Seismic Category I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Reactor Coolant Pump (RCP) Circuit Breaker	ECS-ES-31	Yes	Yes/No (Trip open only)	Yes
RCP Circuit Breaker	ECS-ES-32	Yes	Yes/No (Trip open only)	Yes
RCP Circuit Breaker	ECS-ES-41	Yes	Yes/No (Trip open only)	Yes
RCP Circuit Breaker	ECS-ES-42	Yes	Yes/No (Trip open only)	Yes
RCP Circuit Breaker	ECS-ES-51	Yes	Yes/No (Trip open only)	Yes
RCP Circuit Breaker	ECS-ES-52	Yes	Yes/No (Trip open only)	Yes
RCP Circuit Breaker	ECS-ES-61	Yes	Yes/No (Trip open only)	Yes
RCP Circuit Breaker	ECS-ES-62	Yes	Yes/No (Trip open only)	Yes

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Table 2.6.1-2

Load Description	Power Source
Load Center Transformers EK-11, EK-12, EK-13, EK-14	ZOS-MG-02A
ECS Panel Transformers	ZOS-MG-02A
Diesel Oil Transfer Module Enclosure A Electric Unit Heater	ZOS-MG-02A
Diesel Oil Transfer Module Enclosure A Fan	ZOS-MG-02A
Class 1E Division A Regulating Transformer	ZOS-MG-02A
Class 1E Division C Regulating Transformer	ZOS-MG-02A
Diesel Generator Fuel Oil Transfer Pump A	ZOS-MG-02A
Diesel Generator Room A Building Standby Exhaust Fans 1A and 2A	ZOS-MG-02A
Diesel Generator Service Module A Air Handling Unit (AHU) 01A Fan	ZOS-MG-02A
Diesel Generator Fuel Oil Cooler Fan A	ZOS-MG-02A
Diesel Fuel Oil Transfer Module Unit Heater A	ZOS-MG-02A
Diesel Generator Jacket Water Radiator Fans A	ZOS-MG-02A
Diesel Generator AC/OC Radiator Fan A	ZOS-MG-02A
Diesel Generator Building Engine AHU MS 03A Fan	ZOS-MG-02A
Fuel Oil Day Tank Vault Exhaust Fan A	ZOS-MG-02A
Diesel Generator Lube Oil Cooling Motors A (Front/Rear)	ZOS-MG-02A
Diesel Generator Transformer A	ZOS-MG-02A
Day Tank Heater Pad A	ZOS-MG-02A
Startup Feedwater Pump A	ZOS-MG-02A
Service Water Pump A	ZOS-MG-02A
Service Water Cooling Tower Fan A	ZOS-MG-02A
MCR/Control Support Area (CSA) AHU A Supply and Return Fans	ZOS-MG-02A
Divisions A/C Class 1E Electrical Room AHU A Supply and Return Fans	ZOS-MG-02A
Divisions B/D Class 1E Electrical Room AHU D Supply and Return Fans	ZOS-MG-02A
Air-cooled Chiller Pump 2	ZOS-MG-02A
Component Cooling Water Pump A	ZOS-MG-02A

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Table 2.6.1-2 (cont.)

Load Description	Power Source
Air-cooled Chiller 2	ZOS-MG-02A
Air-cooled Chiller 2 Piping Heat Trace	ZOS-MG-02A
Air-cooled Chiller 2 Control and Heat Trace	ZOS-MG-02A
Chemical and Volume Control System (CVS) Makeup Pump A	ZOS-MG-02A
CVS Pump Room Unit Cooler Fan A	ZOS-MG-02A
Normal Residual Heat Removal System (RNS) Pump A	ZOS-MG-02A
RNS Pump Room Unit Cooler Fan A	ZOS-MG-02A
Equipment Room AHU Supply and Return Fans VXS-MA-01A/02A	ZOS-MG-02A
Switchgear Room A AHU Supply and Return Fans VXS-MA-05A/06A	ZOS-MG-02A
Non-1E Battery Charger EDS1-DC-1	ZOS-MG-02A
Non-1E Battery Room A Exhaust Fan	ZOS-MG-02A
Non-1E Battery Charger EDS3-DC-1	ZOS-MG-02A
Class 1E Division A Battery Charger 1 (24-hour)	ZOS-MG-02A
Class 1E Division C Battery Charger 1 (24-hour)	ZOS-MG-02A
Class 1E Division C Battery Charger 2 (72-hour)	ZOS-MG-02A
Divisions A/C Class 1E Battery Room Exhaust Fan A	ZOS-MG-02A
Divisions B/D Class 1E Battery Room Exhaust Fan D	ZOS-MG-02A
Supplemental Air Filtration Unit Fan A	ZOS-MG-02A
Backup Group 4A Pressurizer Heaters	ZOS-MG-02A
Spent Fuel Cooling Pump A	ZOS-MG-02A
Load Center Transformers EK-21, EK-22, EK-23, EK-24	ZOS-MG-02B
ECS Panel Transformers	ZOS-MG-02B
Diesel Oil Transfer Module Enclosure B Electric Unit Heater	ZOS-MG-02B
Diesel Oil Transfer Module Enclosure B Fan	ZOS-MG-02B
Class 1E Division B Regulating Transformer	ZOS-MG-02B
Class 1E Division D Regulating Transformer	ZOS-MG-02B
Diesel Generator Fuel Oil Transfer Pump B	ZOS-MG-02B
Diesel Generator Room B Building Standby Exhaust Fans 1B and 2B	ZOS-MG-02B
Diesel Generator Service Module B AHU 01B Fan	ZOS-MG-02B

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Table 2.6.1-2 (cont.)

Load Description	Power Source
Diesel Generator Fuel Oil Cooler Fan B	ZOS-MG-02B
Diesel Fuel Oil Transfer Module Unit Heater B	ZOS-MG-02B
Diesel Generator Jacket Water Radiator Fans B	ZOS-MG-02B
Diesel Generator AC/OC Radiator Fan B	ZOS-MG-02B
Diesel Generator Building Engine AHU MS 03B Fan	ZOS-MG-02B
Fuel Oil Day Tank Vault Exhaust Fan B	ZOS-MG-02B
Diesel Generator Lube Oil Cooling Motors B (Front/Rear)	ZOS-MG-02B
Diesel Generator Transformer B	ZOS-MG-02B
Day Tank Heater Pad B	ZOS-MG-02B
Startup Feedwater Pump B	ZOS-MG-02B
Service Water Pump B	ZOS-MG-02B
Service Water Cooling Tower Fan B	ZOS-MG-02B
MCR/CSA AHU B Supply and Return Fans	ZOS-MG-02B
Divisions B/D Class 1E Electrical Room AHU B Supply and Return Fans	ZOS-MG-02B
Divisions A/C Class 1E Electrical Room AHU C Supply and Return Fans	ZOS-MG-02B
Air-cooled Chiller Pump 3	ZOS-MG-02B
Component Cooling Water Pump B	ZOS-MG-02B
Air-cooled Chiller 3	ZOS-MG-02B
Air-cooled Chiller 3 Piping Heat Trace	ZOS-MG-02B
CVS Makeup Pump B	ZOS-MG-02B
CVS Pump Room Unit Cooler Fan B	ZOS-MG-02B
RNS Pump B	ZOS-MG-02B
RNS Pump Room Unit Cooler Fan B	ZOS-MG-02B
Equipment Room B AHU Supply and Return Fans VXS-MA-01B/02B	ZOS-MG-02B
Switchgear Room B AHU Supply and Return Fans VXS-MA-05B/06B	ZOS-MG-02B
Non-1E Battery Charger EDS2-DC-1	ZOS-MG-02B
Non-1E Battery Charger EDS4-DC-1	ZOS-MG-02B
Non-1E Battery Room B Exhaust Fan	ZOS-MG-02B

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Table 2.6.1-2 (cont.)

Load Description	Power Source
Class 1E Division B Battery Charger 1 (24-hour)	ZOS-MG-02B
Class 1E Division B Battery Charger 2 (72-hour)	ZOS-MG-02B
Class 1E Division D Battery Charger 1 (24-hour)	ZOS-MG-02B
Divisions B/D Class 1E Battery Room Exhaust Fan B	ZOS-MG-02B
Divisions A/C Class 1E Battery Room Exhaust Fan C	ZOS-MG-02B
Supplemental Air Filtration Unit Fan B	ZOS-MG-02B
Backup Group 4B Pressurizer Heaters	ZOS-MG-02B
Spent Fuel Cooling Pump B	ZOS-MG-02B

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Table 2.6.1-3

Equipment	Tag No.	Display	Control Function
6900 V Switchgear Bus 1	ECS-ES-1	Yes (Bus voltage, breaker position for all breakers on bus)	Yes (Breaker open/close)
6900 V Switchgear Bus 2	ECS-ES-2	Yes (Bus voltage, breaker position for all breakers on bus)	Yes (Breaker open/close)
Unit Auxiliary Transformer A	ZAS-ET-2A	Yes (Secondary Voltage)	No
Unit Auxiliary Transformer B	ZAS-ET-2B	Yes (Secondary Voltage)	No
Reserve Auxiliary Transformer A	ZAS-ET-4A	Yes (Secondary Voltage)	No
Reserve Auxiliary Transformer B	ZAS-ET-4B	Yes (Secondary Voltage)	No

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Table 2.6.1-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The seismic Category I equipment identified in Table 2.6.1-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.6.1-1 is located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.6.1-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
3.a) Not used.		
3.b) Not used.		
4.b) Not used.		
4.c) Not used.		
4.d) Each ancillary diesel generator unit is sized to supply power to long-term safety-related post-accident monitoring loads and control room lighting and ventilation through a regulating transformer; and for one PCS recirculation pump.	Each ancillary diesel generator will be operated with fuel supplied from the ancillary diesel generator fuel tank and with a load of 35 kW or greater and a power factor between 0.9 and 1.0 for a time period required to reach engine temperature equilibrium plus 2.5 hours.	Each diesel generator provides power to the load with a generator terminal voltage of 480 ± 10% volts and a frequency of 60 ± 5% Hz.
4.a) The ECS provides the capability for distributing non-Class 1E ac power from onsite sources (ZOS) to nonsafety-related loads listed in Table 2.6.1-2.	Tests will be performed using a test signal to confirm that an electrical path exists for each selected load listed in Table 2.6.1-2 from an ECS-ES-1 or ECS-ES-2 bus. Each test may be a single test or a series of over-lapping tests.	A test signal exists at the terminals of each selected load.

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Table 2.6.1-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.e) The ECS provides two loss-of-voltage signals to the onsite standby power system (ZOS), one for each diesel-backed 6900 Vac switchgear bus.	Tests on the as-built ECS system will be conducted by simulating a loss-of-voltage condition on each diesel-backed 6900 Vac switchgear bus.	A loss-of-voltage signal is generated when the loss-of-voltage condition is simulated.
4.f) The ECS provides a reverse-power trip of the generator circuit breaker which is blocked for at least 15 seconds following a turbine trip.	Tests on the as-built ECS system will be conducted by simulating a turbine trip signal followed by a simulated reverse-power condition. The generator circuit breaker trip signal will be monitored.	The generator circuit breaker trip signal does not occur until at least 15 seconds after the simulated turbine trip.
5. Controls exist in the MCR to cause the circuit breakers identified in Table 2.6.1-3 to perform the listed functions.	Tests will be performed to verify that controls in the MCR can operate the circuit breakers identified in Table 2.6.1-3.	Controls in the MCR cause the circuit breakers identified in Table 2.6.1-3 to operate.
6. Displays of the parameters identified in Table 2.6.1-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.6.1-3 in the MCR.	Displays identified in Table 2.6.1-3 can be retrieved in the MCR.

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Table 2.6.1-5

Component Name	Tag No.	Component Location
RCP Circuit Breaker	ECS-ES-31	Auxiliary Building
RCP Circuit Breaker	ECS-ES-32	Auxiliary Building
RCP Circuit Breaker	ECS-ES-41	Auxiliary Building
RCP Circuit Breaker	ECS-ES-42	Auxiliary Building
RCP Circuit Breaker	ECS-ES-51	Auxiliary Building
RCP Circuit Breaker	ECS-ES-52	Auxiliary Building
RCP Circuit Breaker	ECS-ES-61	Auxiliary Building
RCP Circuit Breaker	ECS-ES-62	Auxiliary Building
6900 V Switchgear Bus 1	ECS-ES-1	Annex Building
6900 V Switchgear Bus 2	ECS-ES-2	Annex Building
6900 V Switchgear Bus 3	ECS-ES-3	Turbine Building
6900 V Switchgear Bus 4	ECS-ES-4	Turbine Building
6900 V Switchgear Bus 5	ECS-ES-5	Turbine Building
6900 V Switchgear Bus 6	ECS-ES-6	Turbine Building
Main Generator	ZAS-MG-01	Turbine Building
Generator Circuit Breaker	ZAS-ES-01	Turbine Building
Main Step-up Transformer	ZAS-ET-1A	Yard
Main Step-up Transformer	ZAS-ET-1B	Yard
Main Step-up Transformer	ZAS-ET-1C	Yard
Unit Auxiliary Transformer A	ZAS-ET-2A	Yard
Unit Auxiliary Transformer B	ZAS-ET-2B	Yard
Reserve Auxiliary Transformer A	ZAS-ET-4A	Yard
Reserve Auxiliary Transformer B	ZAS-ET-4B	Yard
Ancillary Diesel Generator #1	ECS-MG-01	Annex Building
Ancillary Diesel Generator #2	ECS-MG-02	Annex Building
Ancillary Diesel Generator Distribution Panel 1	ECS-ED-01	Annex Building
Ancillary Diesel Generator Distribution Panel 1	ECS-ED-02	Annex Building

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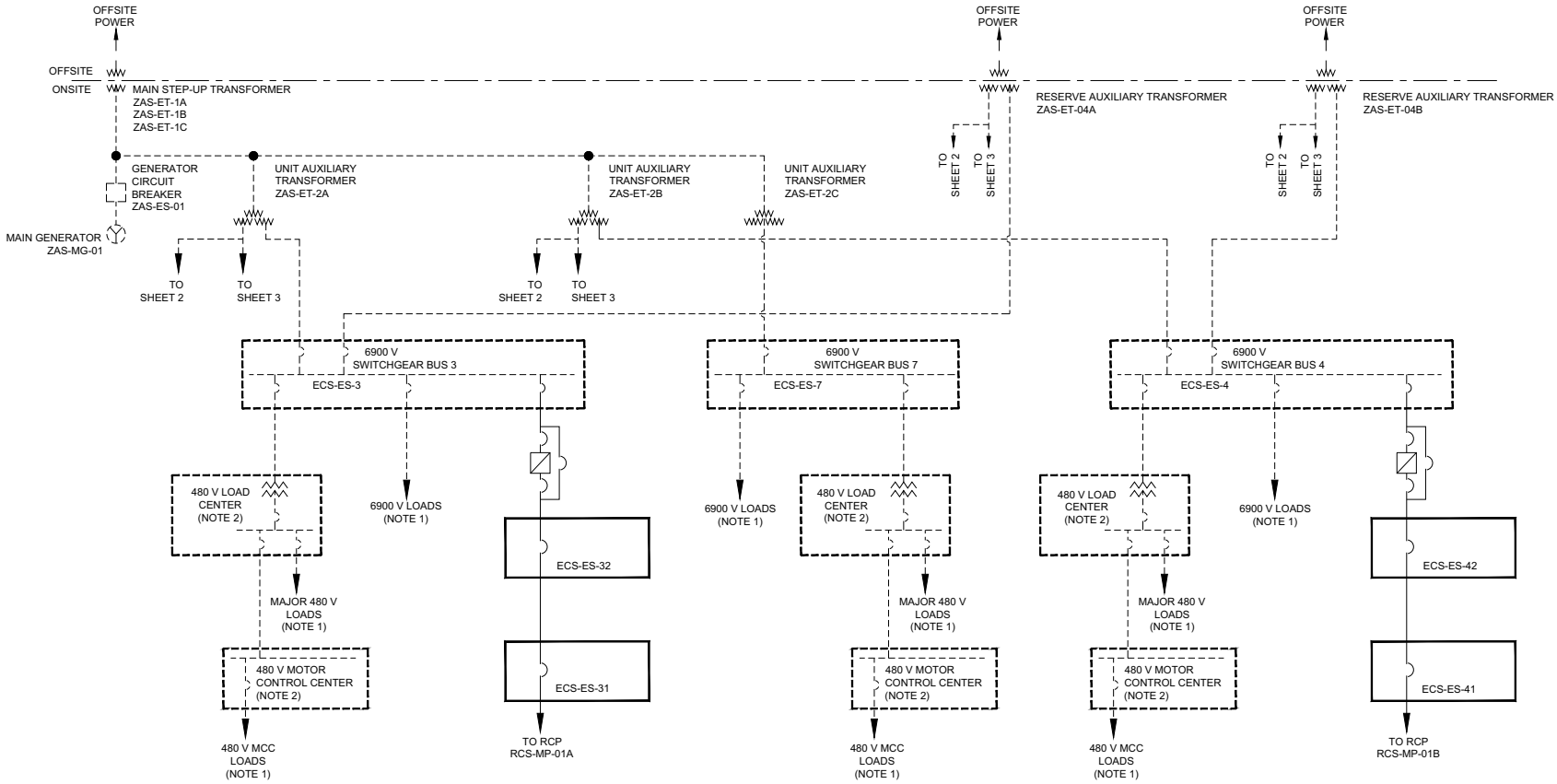

NOTES:
1. ALL LOADS ARE TYPICAL OF ONE OR MORE.
2. LOAD CENTERS AND MOTOR CONTROL CENTERS ARE TYPICAL OF ONE OR MORE.

Figure 2.6.1-1 (Sheet 1 of 4)
Main ac Power System

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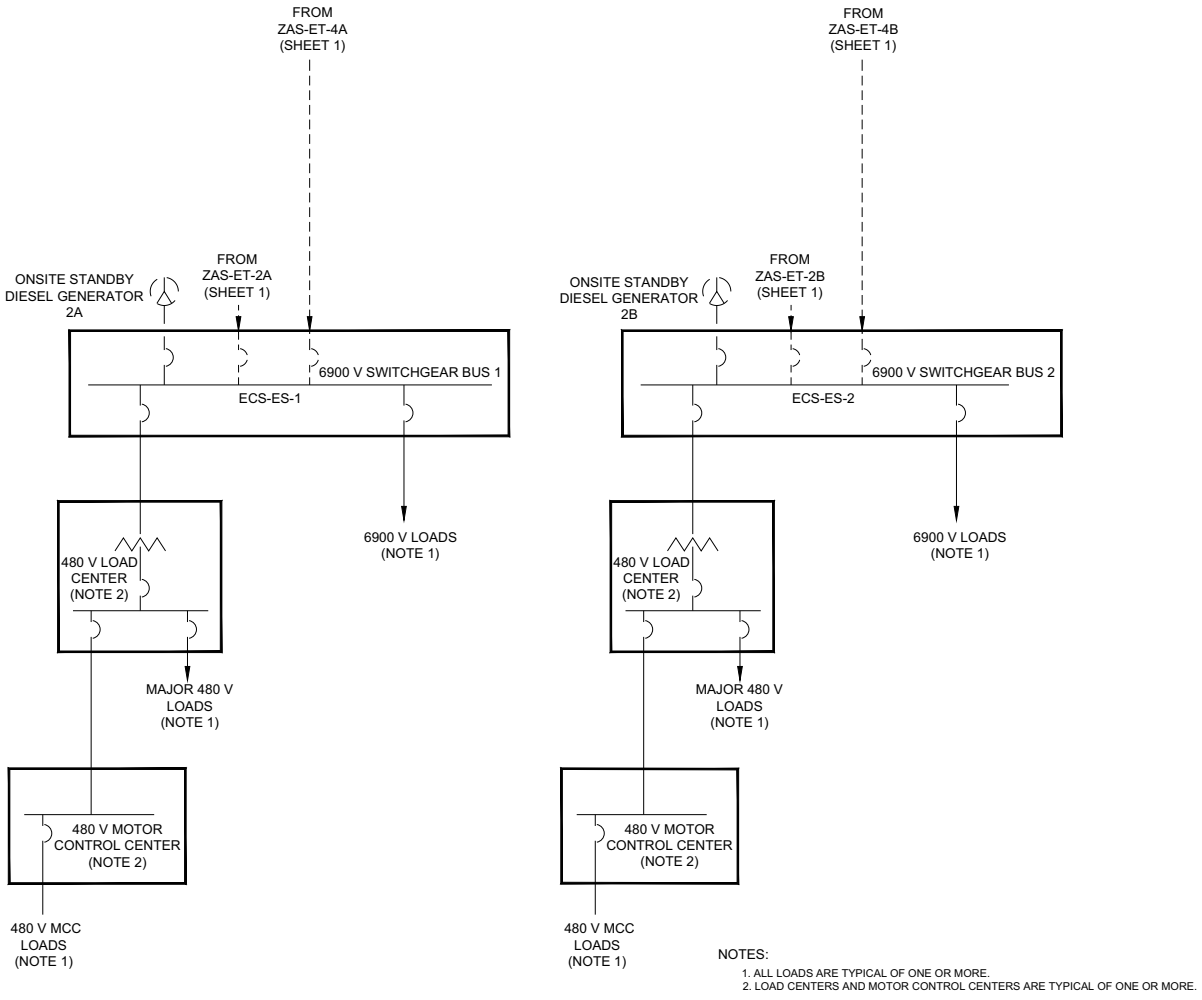

Figure 2.6.1-1 (Sheet 2 of 4)
Main ac Power System

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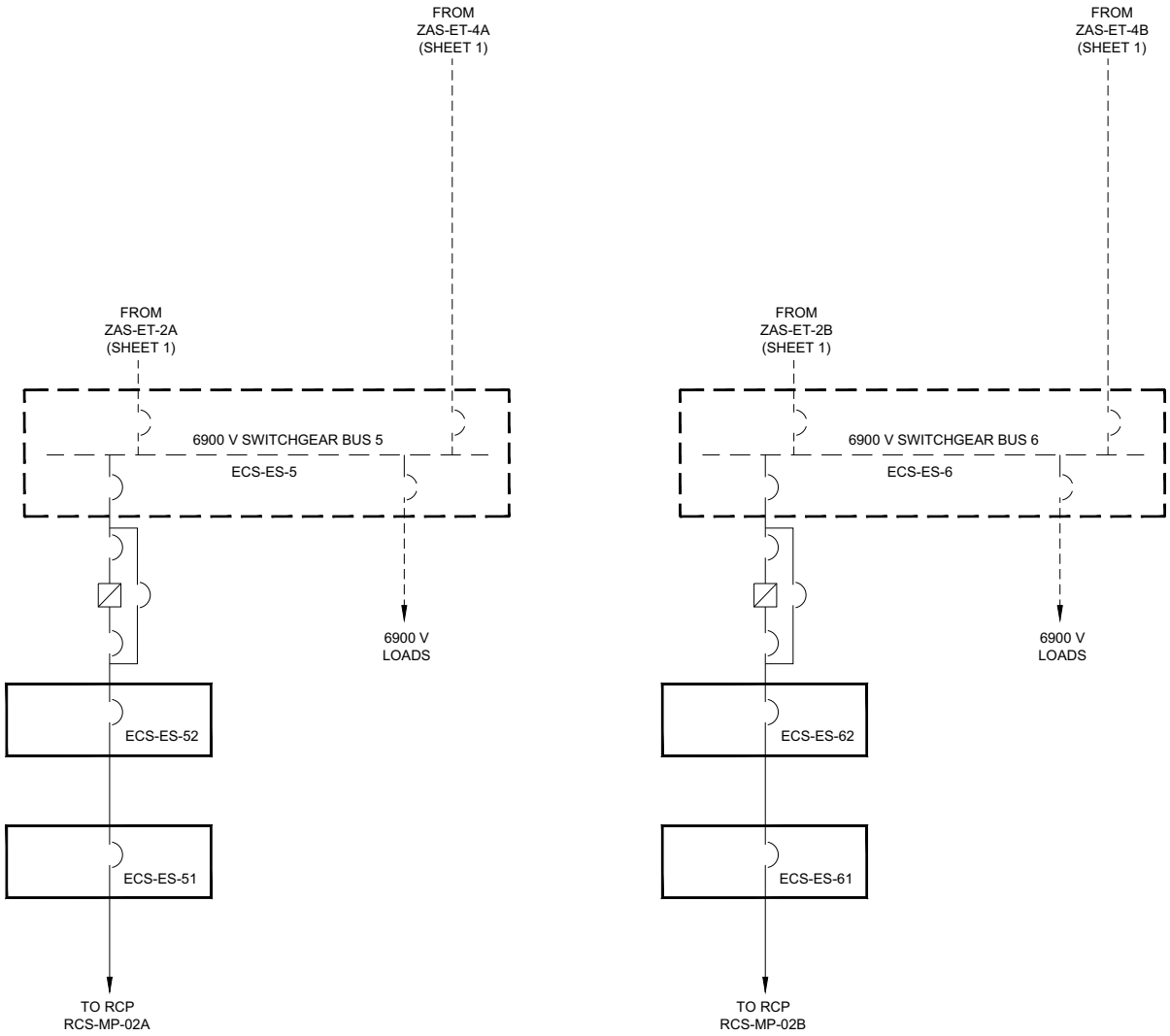

Figure 2.6.1-1 (Sheet 3 of 4)
Main ac Power System

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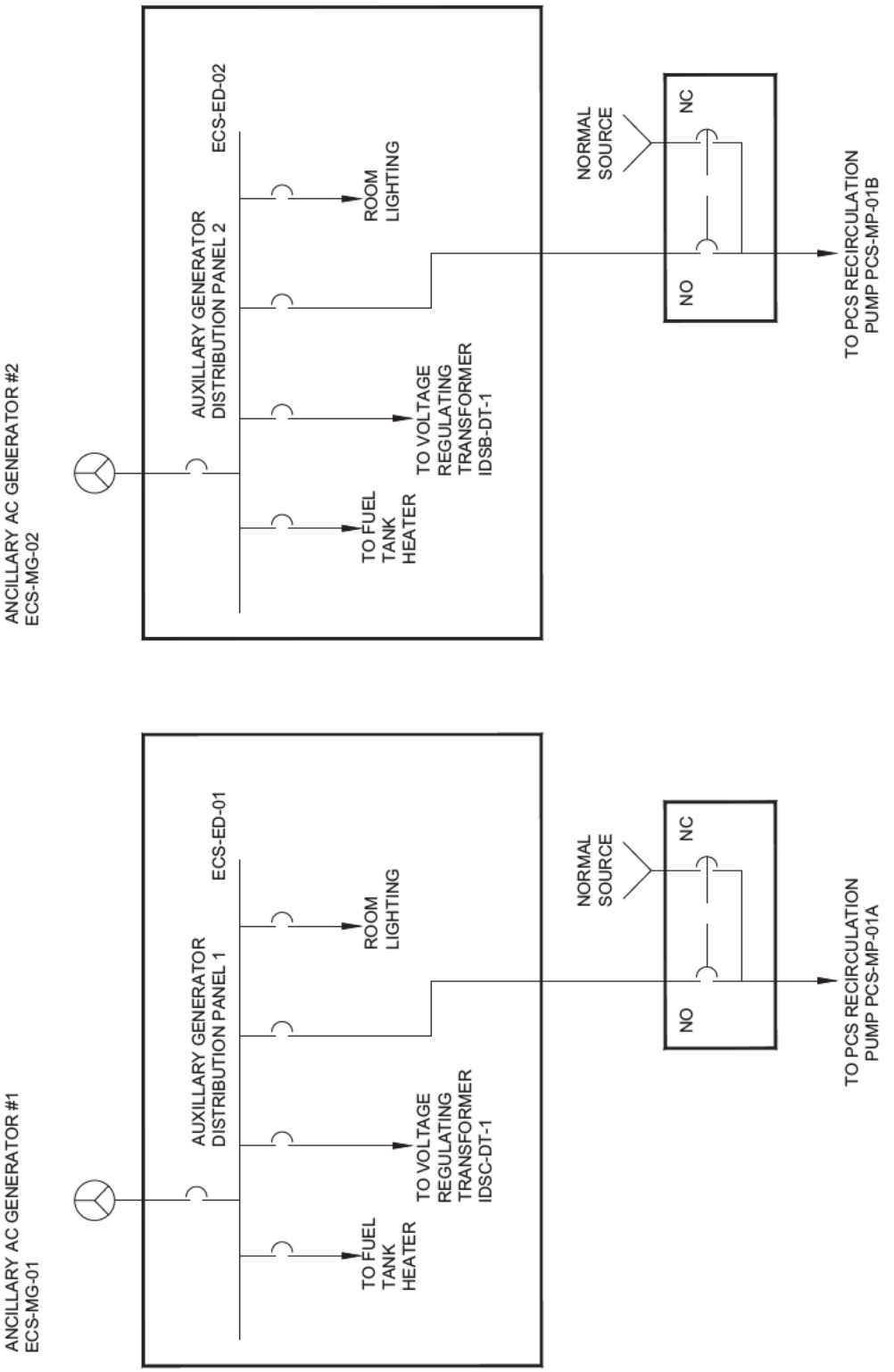

Figure 2.6.1-1 (Sheet 4 of 4)
Main ac Power System

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2.6.2 Non-Class 1E dc and Uninterruptible Power Supply System

Design Description

The non-Class 1E dc and uninterruptible power supply system (EDS) provides dc and uninterruptible ac electrical power to nonsafety-related loads during normal and off-normal conditions.

The EDS is as shown in Figure 2.6.2-1 and the component locations of the EDS are as shown in Table 2.6.2-2.

1. The functional arrangement of the EDS is as described in the Design Description of this Section 2.6.2.

2. The EDS provides the following nonsafety-related functions:

    a) Each EDS load group 1, 2, 3, and 4 battery charger supplies the corresponding dc switchboard bus load while maintaining the corresponding battery charged.

    b) Each EDS load group 1, 2, 3, and 4 battery supplies the corresponding dc switchboard bus load for a period of 2 hours without recharging.

    c) Each EDS load group 1, 2, 3, and 4 inverter supplies the corresponding ac load.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.2-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the EDS.

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Table 2.6.2-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) Each EDS load group 1, 2, 3, and 4 battery charger supplies the corresponding dc switchboard bus load while maintaining the corresponding battery charged.	Testing of each as-built battery charger will be performed by applying a simulated or real load, or a combination of simulated or real loads.	Each battery charger provides an output current of at least 900 amps with an output voltage in the range 105 to 140 V.
2.b) Each EDS load group 1, 2, 3, and 4 battery supplies the corresponding dc switchboard bus load for a period of 2 hours without recharging.	Testing of each as-built battery will be performed by applying a simulated or real load, or a combination of simulated or real loads. The test will be conducted on a battery that has been fully charged and has been connected to a battery charger maintained at 135 ± 1 V for a period of no less than 24 hours prior to the test.	The battery terminal voltage is greater than or equal to 105 V after a period of no less than 2 hours, with an equivalent load greater than 850 amps.
2.c) Each EDS load group 1, 2, 3, and 4 inverter supplies the corresponding ac load.	Testing of each as-built inverter will be performed by applying a simulated or real load, or a combination of simulated or real loads, equivalent to a resistive load greater than 55 kW.	Each inverter provides a line-to-line output voltage of 208 ± 2% V at a frequency of 60 ± 0.5% Hz.

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Table 2.6.2-2

Component Name	Tag No.	Component Location
Load Group 1 Battery	EDS1-DB-1	Annex Building
Load Group 2 Battery	EDS2-DB-1	Annex Building
Load Group 3 Battery	EDS3-DB-1	Annex Building
Load Group 4 Battery	EDS4-DB-1	Annex Building
Load Group 1 Battery Charger	EDS1-DC-1	Annex Building
Load Group 2 Battery Charger	EDS2-DC-1	Annex Building
Load Group 3 Battery Charger	EDS3-DC-1	Annex Building
Load Group 4 Battery Charger	EDS4-DC-1	Annex Building
Load Group 1 125 Vdc Switchboard	EDS1-DS-1	Annex Building
Load Group 2 125 Vdc Switchboard	EDS2-DS-1	Annex Building
Load Group 3 125 Vdc Switchboard	EDS3-DS-1	Annex Building
Load Group 4 125 Vdc Switchboard	EDS4-DS-1	Annex Building
Load Group 1 Inverter	EDS1-DU-1	Annex Building
Load Group 2 Inverter	EDS2-DU-1	Annex Building
Load Group 3 Inverter	EDS3-DU-1	Annex Building
Load Group 4 Inverter	EDS4-DU-1	Annex Building

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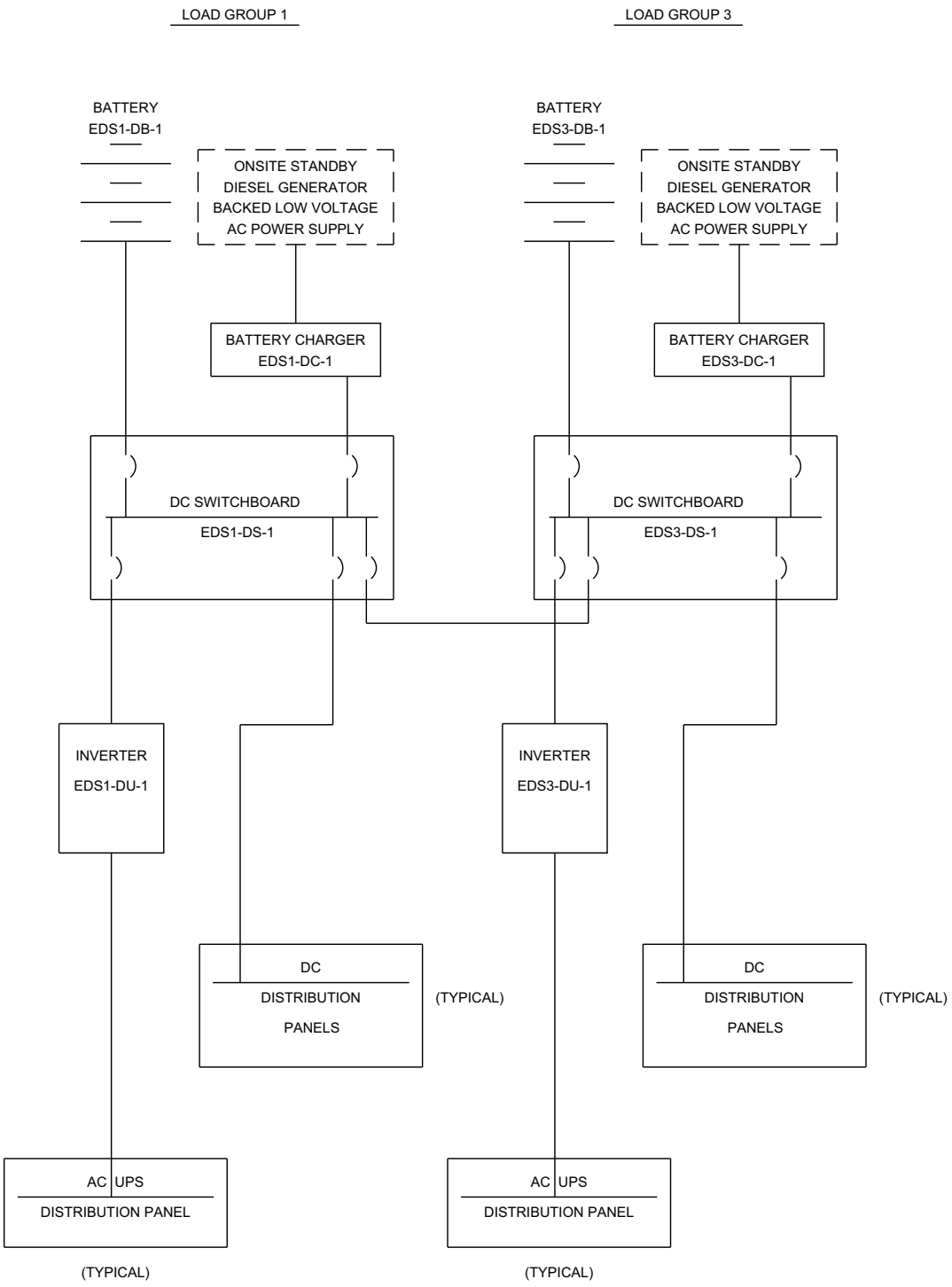

Figure 2.6.2-1 (Sheet 1 of 2)
Non-Class 1E dc and Uninterruptible Power Supply System

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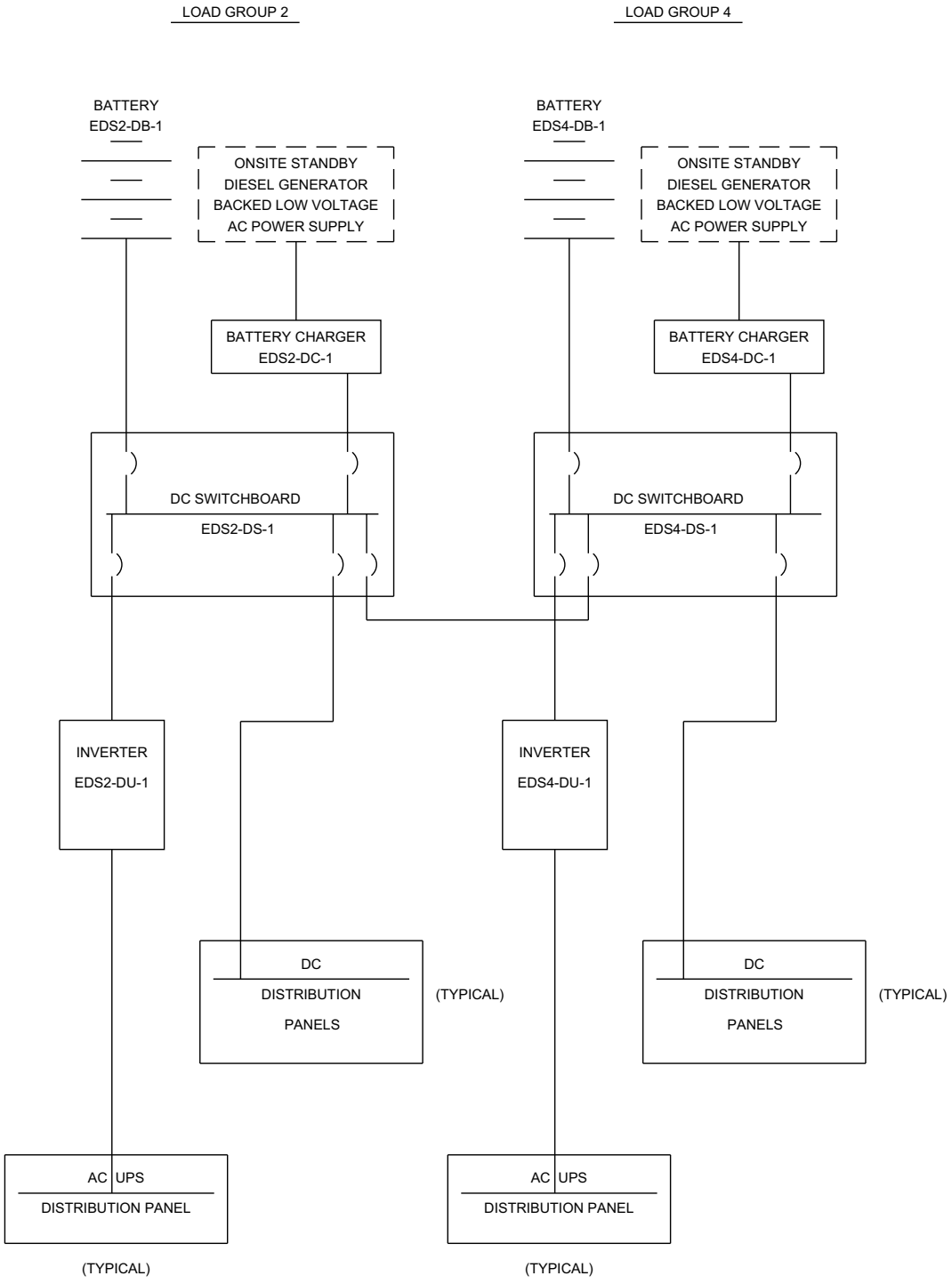

Figure 2.6.2-1 (Sheet 2 of 2)
Non-Class 1E dc and Uninterruptible Power Supply System

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2.6.3 Class 1E dc and Uninterruptible Power Supply System

Design Description

The Class 1E dc and uninterruptible power supply system (IDS) provides dc and uninterruptible ac electrical power for safety-related equipment during normal and off-normal conditions.

The IDS is as shown in Figure 2.6.3-1 and the component locations of the IDS are as shown in Table 2.6.3-4.

1. The functional arrangement of the IDS is as described in the Design Description of this Section 2.6.3.

2. The seismic Category I equipment identified in Table 2.6.3-1 can withstand seismic design basis loads without loss of safety function.

3. Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cables.

4. The IDS provides the following safety-related functions:

    a) The IDS provides electrical independence between the Class 1E divisions.

    b) The IDS provides electrical isolation between itself and the non-Class 1E ac power system and the non-Class 1E lighting in the MCR.

    c) Each IDS 24-hour battery bank supplies a dc switchboard bus load for a period of 24 hours without recharging.

    d) Each IDS 72-hour battery bank supplies a dc switchboard bus load for a period of 72 hours without recharging.

    e) The IDS spare battery bank supplies a dc load equal to or greater than the most severe switchboard bus load for the required period without recharging.

    f) Each IDS 24-hour inverter supplies its ac load.

    g) Each IDS 72-hour inverter supplies its ac load.

    h) Each IDS 24-hour battery charger provides the protection and safety monitoring system (PMS) with two loss-of-ac input voltage signals.

    i) The IDS supplies an operating voltage at the terminals of the Class 1E motor-operated valves identified in Tier 1 Material subsections 2.1.2, 2.2.1, 2.2.2, 2.2.3, 2.2.4, 2.3.2, 2.3.6, and 2.7.1 that is greater than or equal to the minimum design voltage.

    j) The IDS provides electrical isolation between the non-Class 1E battery monitors and the Class 1 E battery banks.

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5. The IDS provides the following nonsafety-related functions:

   a) Each IDS 24-hour battery charger supplies a dc switchboard bus load while maintaining the corresponding battery charged.

   b) Each IDS 72-hour battery charger supplies a dc switchboard bus load while maintaining the corresponding battery charged.

   c) Each IDS regulating transformer supplies an ac load when powered from the 480 V motor control center (MCC).

   d) The IDS Divisions B and C regulating transformers supply their post-72 hour ac loads when powered from an ancillary diesel generator.

6. Safety-related displays identified in Table 2.6.3-1 can be retrieved in the MCR.

7. The IDS dc battery fuses and battery charger circuit breakers, and dc distribution panels, MCCs, and their circuit breakers and fuses, are sized to supply their load requirements.

8. Circuit breakers and fuses in IDS battery, battery charger, dc distribution panel, and MCC circuits are rated to interrupt fault currents.

9. The IDS batteries, battery chargers, dc distribution panels, and MCCs are rated to withstand fault currents for the time required to clear the fault from its power source.

10. The IDS electrical distribution system cables are rated to withstand fault currents for the time required to clear the fault from its power source.

11. Displays of the parameters identified in Table 2.6.3-2 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.3-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the IDS.

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Table 2.6.3-1

Equipment Name	Tag No.	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Division A 250 Vdc 24-Hour Battery Bank	IDSA-DB-1	Yes	Yes/No	No
Division B 250 Vdc 24-Hour Battery Bank 1	IDSB-DB-1	Yes	Yes/No	No
Division B 250 Vdc 72-Hour Battery Bank 2	IDSB-DB-2	Yes	Yes/No	No
Division C 250 Vdc 24-Hour Battery Bank 1	IDSC-DB-1	Yes	Yes/No	No
Division C 250 Vdc 72-Hour Battery Bank 2	IDSC-DB-2	Yes	Yes/No	No
Division D 250 Vdc 24-Hour Battery Bank	IDSD-DB-1	Yes	Yes/No	No
Spare 250 Vdc Battery Bank	IDSS-DB-1	Yes	Yes/No	No
Division A 24-Hour Battery Charger 1	IDSA-DC-1	Yes	Yes/No	No
Division B 24-Hour Battery Charger 1	IDSB-DC-1	Yes	Yes/No	No
Division B 72-Hour Battery Charger 2	IDSB-DC-2	Yes	Yes/No	No
Division C 24-Hour Battery Charger 1	IDSC-DC-1	Yes	Yes/No	No
Division C 72-Hour Battery Charger 2	IDSC-DC-2	Yes	Yes/No	No
Division D 24-Hour Battery Charger 1	IDSD-DC-1	Yes	Yes/No	No
Spare Battery Charger 1	IDSS-DC-1	Yes	Yes/No	No
Division A 250 Vdc Distribution Panel	IDSA-DD-1	Yes	Yes/No	No
Division B 250 Vdc Distribution Panel	IDSB-DD-1	Yes	Yes/No	No
Division C 250 Vdc Distribution Panel	IDSC-DD-1	Yes	Yes/No	No
Division D 250 Vdc Distribution Panel	IDSD-DD-1	Yes	Yes/No	No
Division A 120 Vac Distribution Panel 1	IDSA-EA-1	Yes	Yes/No	No
Division A 120 Vac Distribution Panel 2	IDSA-EA-2	Yes	Yes/No	No
Division B 120 Vac Distribution Panel 1	IDSB-EA-1	Yes	Yes/No	No
Division B 120 Vac Distribution Panel 2	IDSB-EA-2	Yes	Yes/No	No
Division B 120 Vac Distribution Panel 3	IDSB-EA-3	Yes	Yes/No	No
Division C 120 Vac Distribution Panel 1	IDSC-EA-1	Yes	Yes/No	No
Division C 120 Vac Distribution Panel 2	IDSC-EA-2	Yes	Yes/No	No
Division C 120 Vac Distribution Panel 3	IDSC-EA-3	Yes	Yes/No	No
Division D 120 Vac Distribution Panel 1	IDSD-EA-1	Yes	Yes/No	No

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Table 2.6.3-1 (cont.)

Equipment Name	Tag No.	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Division D 120 Vac Distribution Panel 2	IDSD-EA-2	Yes	Yes/No	No
Division A Fuse Panel 4	IDSA-EA-4	Yes	Yes/No	No
IDSA Battery Monitor Fuse Panel	IDSA-EA-5	Yes	Yes/No	No
Division B Fuse Panel 4	IDSB-EA-4	Yes	Yes/No	No
Division B Fuse Panel 5	IDSB-EA-5	Yes	Yes/No	No
Division B Fuse Panel 6	IDSB-EA-6	Yes	Yes/No	No
IDSB Battery Monitor Fuse Panel	IDSB-EA-7	Yes	Yes/No	No
IDSB Battery Monitor Fuse Panel	IDSB-EA-8	Yes	Yes/No	No
Division C Fuse Panel 4	IDSC-EA-4	Yes	Yes/No	No
Division C Fuse Panel 5	IDSC-EA-5	Yes	Yes/No	No
Division C Fuse Panel 6	IDSC-EA-6	Yes	Yes/No	No
IDSC Battery Monitor Fuse Panel	IDSC-EA-7	Yes	Yes/No	No
IDSC Battery Monitor Fuse Panel	IDSC-EA-8	Yes	Yes/No	No
Division D Fuse Panel 4	IDSD-EA-4	Yes	Yes/No	No
IDSD Battery Monitor Fuse Panel	IDSD-EA-5	Yes	Yes/No	No
IDSS Battery Monitor Fuse Panel	IDSS-EA-1	Yes	Yes/No	No
Division A Fused Transfer Switch Box 1	IDSA-DF-1	Yes	Yes/No	No
Division B Fused Transfer Switch Box 1	IDSB-DF-1	Yes	Yes/No	No
Division B Fused Transfer Switch Box 2	IDSB-DF-2	Yes	Yes/No	No
Division C Fused Transfer Switch Box 1	IDSC-DF-1	Yes	Yes/No	No
Division C Fused Transfer Switch Box 2	IDSC-DF-2	Yes	Yes/No	No
Division D Fused Transfer Switch Box 1	IDSD-DF-1	Yes	Yes/No	No
Spare Fused Transfer Switch Box 1	IDSS-DF-1	Yes	Yes/No	No
Division A 250 Vdc MCC	IDSA-DK-1	Yes	Yes/No	No
Division B 250 Vdc MCC	IDSB-DK-1	Yes	Yes/No	No
Division C 250 Vdc MCC	IDSC-DK-1	Yes	Yes/No	No
Division D 250 Vdc MCC	IDSD-DK-1	Yes	Yes/No	No

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Table 2.6.3-1 (cont.)

Equipment Name	Tag No.	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Division A 250 Vdc Switchboard 1	IDSA-DS-1	Yes	Yes/No	Yes (Bus Voltage)
Division B 250 Vdc Switchboard 1	IDSB-DS-1	Yes	Yes/No	Yes (Bus Voltage)
Division B 250 Vdc Switchboard 2	IDSB-DS-2	Yes	Yes/No	Yes (Bus Voltage)
Division C 250 Vdc Switchboard 1	IDSC-DS-1	Yes	Yes/No	Yes (Bus Voltage)
Division C 250 Vdc Switchboard 2	IDSC-DS-2	Yes	Yes/No	Yes (Bus Voltage)
Division D 250 Vdc Switchboard 1	IDSD-DS-1	Yes	Yes/No	Yes (Bus Voltage)
Division A Regulating Transformer	IDSA-DT-1	Yes	Yes/No	No
Division B Regulating Transformer	IDSB-DT-1	Yes	Yes/No	No
Division C Regulating Transformer	IDSC-DT-1	Yes	Yes/No	No
Division D Regulating Transformer	IDSD-DT-1	Yes	Yes/No	No
Division A 24-Hour Inverter 1	IDSA-DU-1	Yes	Yes/No	No
Division B 24-Hour Inverter 1	IDSB-DU-1	Yes	Yes/No	No
Division B 72-Hour Inverter 2	IDSB-DU-2	Yes	Yes/No	No
Division C 24-Hour Inverter 1	IDSC-DU-1	Yes	Yes/No	No
Division C 72-Hour Inverter 2	IDSC-DU-2	Yes	Yes/No	No
Division D 24-Hour Inverter 1	IDSD-DU-1	Yes	Yes/No	No
Spare Battery Termination Box	IDSS-DF-3	Yes	Yes/No	No

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Table 2.6.3-2

Equipment	Tag No.	Display/Status Indication
Division A Battery Monitor	IDSA-DV-1	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division B 24-Hour Battery Monitor	IDSB-DV-1	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division B 72-Hour Battery Monitor	IDSB-DV-2	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division C 24-Hour Battery Monitor	IDSC-DV-1	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division C 72-Hour Battery Monitor	IDSC-DV-2	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division D Battery Monitor	IDSD-DV-1	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division A Fused Transfer Switch Box	IDSA-DF-1	Yes (Battery Current, Battery Disconnect Switch Position)
Division B 24-Hour Fused Transfer Switch Box	IDSB-DF-1	Yes (Battery Current, Battery Disconnect Switch Position)
Division B 72-Hour Fused Transfer Switch Box	IDSB-DF-2	Yes (Battery Current, Battery Disconnect Switch Position)
Division C 24-Hour Fused Transfer Switch Box	IDSC-DF-1	Yes (Battery Current, Battery Disconnect Switch Position)
Division C 72-Hour Fused Transfer Switch Box	IDSC-DF-2	Yes (Battery Current, Battery Disconnect Switch Position)
Division D Fused Transfer Switch Box	IDSD-DF-1	Yes (Battery Current, Battery Disconnect Switch Position)

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Table 2.6.3-2 (cont.)

Equipment	Tag No.	Display/Status Indication
Division A Battery Charger	IDSA-DC-1	Yes (Charger Output Current, Charger Trouble(1))
Division B 24-Hour Battery Charger	IDSB-DC-1	Yes (Charger Output Current, Charger Trouble(1))
Division B 72-Hour Battery Charger	IDSB-DC-2	Yes (Charger Output Current, Charger Trouble(1))
Division C 24-Hour Battery Charger	IDSC-DC-1	Yes (Charger Output Current, Charger Trouble(1))
Division C 72-Hour Battery Charger	IDSC-DC-2	Yes (Charger Output Current, Charger Trouble(1))
Division D Battery Charger	IDSD-DC-1	Yes (Charger Output Current, Charger Trouble(1))

Note: (1) Battery charger trouble includes charger dc output under/over voltage

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Table 2.6.3-3
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The seismic Category I equipment identified in Table 2.6.3-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.6.3-1 is located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.6.3-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
3. Not used.		
4.a) The IDS provides electrical independence between the Class 1E divisions.	Testing will be performed on the IDS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.6.3-1 when the assigned Class 1E division is provided the test signal.
4.b) The IDS provides electrical isolation between itself and the non-Class 1E ac power system and the non-Class 1E lighting in the MCR.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the battery chargers, regulating transformers, and isolation fuses prevent credible faults from propagating into the IDS.

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Table 2.6.3-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.c) Each IDS 24-hour battery bank supplies a dc switchboard bus load for a period of 24 hours without recharging.	Testing of each 24-hour as-built battery bank will be performed by applying a simulated or real load, or a combination of simulated or real loads which envelope the battery bank design duty cycle. The test will be conducted on a battery bank that has been fully charged and has been connected to a battery charger maintained at 270±2 V for a period of no less than 24 hours prior to the test.	The battery terminal voltage is greater than or equal to 210 V after a period of no less than 24 hours with an equivalent load that equals or exceeds the battery bank design duty cycle capacity.
4.d) Each IDS 72-hour battery bank supplies a dc switchboard bus load for a period of 72 hours without recharging.	Testing of each 72-hour as-built battery bank will be performed by applying a simulated or real load, or a combination of simulated or real loads which envelope the battery bank design duty cycle. The test will be conducted on a battery bank that has been fully charged and has been connected to a battery charger maintained at 270±2 V for a period of no less than 24 hours prior to the test.	The battery terminal voltage is greater than or equal to 210 V after a period of no less than 72 hours with an equivalent load that equals or exceeds the battery bank design duty cycle capacity.
4.e) The IDS spare battery bank supplies a dc load equal to or greater than the most severe switchboard bus load for the required period without recharging.	Testing of the as-built spare battery bank will be performed by applying a simulated or real load, or a combination of simulated or real loads which envelope the most severe of the division batteries design duty cycle. The test will be conducted on a battery bank that has been fully charged and has been connected to a battery charger maintained at 270±2 V for a period of no less than 24 hours prior to the test.	The battery terminal voltage is greater than or equal to 210 V after a period with a load and duration that equals or exceeds the most severe battery bank design duty cycle capacity.
4.f) Each IDS 24-hour inverter supplies its ac load.	Testing of each 24-hour as-built inverter will be performed by applying a simulated or real load, or a combination of simulated or real loads, equivalent to a resistive load greater than 12 kW. The inverter input voltage will be no more than 210 Vdc during the test.	Each 24-hour inverter supplies a line-to-line output voltage of 208 ± 2% V at a frequency of 60 ± 0.5% Hz.

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Table 2.6.3-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.g) Each IDS 72-hour inverter supplies its ac load.	Testing of each 72-hour as-built inverter will be performed by applying a simulated or real load, or a combination of simulated or real loads, equivalent to a resistive load greater than 7 kW. The inverter input voltage will be no more than 210 Vdc during the test.	Each 72-hour inverter supplies a line-to-line output voltage of 208 ± 2% V at a frequency of 60 ± 0.5% Hz.
4.h) Each IDS 24-hour battery charger provides the PMS with two loss-of-ac input voltage signals.	Testing will be performed by simulating a loss of input voltage to each 24-hour battery charger.	Two PMS input signals exist from each 24-hour battery charger indicating loss of ac input voltage when the loss-of-input voltage condition is simulated.
5.a) Each IDS 24-hour battery charger supplies a dc switchboard bus load while maintaining the corresponding battery charged.	Testing of each as-built 24-hour battery charger will be performed by applying a simulated or real load, or a combination of simulated or real loads.	Each 24-hour battery charger provides an output current of at least 150 A with an output voltage in the range 210 to 280 V.
5.b) Each IDS 72-hour battery charger supplies a dc switchboard bus load while maintaining the corresponding battery charged.	Testing of each 72-hour as-built battery charger will be performed by applying a simulated or real load, or a combination of simulated or real loads.	Each 72-hour battery charger provides an output current of at least 125 A with an output voltage in the range 210 to 280 V.
5.c) Each IDS regulating transformer supplies an ac load when powered from the 480 V MCC.	Testing of each as-built regulating transformer will be performed by applying a simulated or real load, or a combination of simulated or real loads, equivalent to a resistive load greater than 30 kW when powered from the 480 V MCC.	Each regulating transformer supplies a line-to-line output voltage of 208 ± 2% V.
6. Safety-related displays identified in Table 2.6.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.6.3-1 can be retrieved in the MCR.
11. Displays of the parameters identified in Table 2.6.3-2 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.6.3-2 in the MCR.	Displays identified in Table 2.6.3-2 can be retrieved in the MCR.

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Table 2.6.3-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.i) The IDS supplies an operating voltage at the terminals of the Class 1E motor operated valves identified in Tier 1 Material subsections 2.1.2, 2.2.1, 2.2.2, 2.2.3, 2.2.4, 2.3.2, 2.3.6, and 2.7.1 that is greater than or equal to the minimum design voltage.	Testing will be performed by measuring the voltage during motor starting at both the IDS battery and motor-operated valve motor terminals while each motor-operated valve is stroked. Analyses will be performed to verify that the voltage at the motor-operated valve motor terminals is greater than or equal to the minimum design voltage of each motor-operated valve with an IDS battery terminal voltage of 210 Vdc.	A report exists and concludes that IDS can provide a voltage greater than or equal to each valve's minimum design voltage to the motor terminals of each motor-operated valve when power is supplied under design conditions from IDS batteries with battery terminal voltage at 210 Vdc while each motor-operated valve is stroked.
4.j) The IDS provides electrical isolation between non-Class 1E battery monitors and the Class 1E battery banks.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the battery monitor fuse isolation panels prevent credible faults from propagating into the Class 1E portions of the IDS.
5.d) The IDS Divisions B and C regulating transformers supply their post-72-hour ac loads when powered from an ancillary diesel generator.	Inspection of the as-built system will be performed.	i) Ancillary diesel generator 1 is electrically connected to regulating transformer IDSC-DT-1 ii) Ancillary diesel generator 2 is electrically connected to regulating transformer IDSB-DT-1.

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Table 2.6.3-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The IDS dc battery fuses and battery charger circuit breakers, and dc distribution panels, MCCs, and their circuit breakers and fuses, are sized to supply their load requirements.	Analyses for the as-built IDS dc electrical distribution system to determine the capacities of the battery fuses and battery charger circuit breakers, and dc distribution panels, MCCs, and their circuit breakers and fuses, will be performed.	Analyses for the as-built IDS dc electrical distribution system exist and conclude that the capacities of as-built IDS battery fuses and battery charger circuit breakers, and dc distribution panels, MCCs, and their circuit breakers and fuses, as determined by their nameplate ratings, exceed their analyzed load requirements.
8. Circuit breakers and fuses in IDS battery, battery charger, dc distribution panel, and MCC circuits are rated to interrupt fault currents.	Analyses for the as-built IDS dc electrical distribution system to determine fault currents will be performed.	Analyses for the as-built IDS dc electrical distribution system exist and conclude that the analyzed fault currents do not exceed the interrupt capacity of circuit breakers and fuses in the battery, battery charger, dc distribution panel, and MCC circuits, as determined by their nameplate ratings.
9. The IDS batteries, battery chargers, dc distribution panels, and MCCs are rated to withstand fault currents for the time required to clear the fault from its power source.	Analyses for the as-built IDS dc electrical distribution system to determine fault currents will be performed.	Analyses for the as-built IDS dc electrical distribution system exist and conclude that the fault current capacities of as-built IDS batteries, battery chargers, dc distribution panels, and MCCs, as determined by manufacturer's ratings, exceed their analyzed fault currents for the time required to clear the fault from its power source as determined by the circuit interrupting device coordination analyses.
10. The IDS electrical distribution system cables are rated to withstand fault currents for the time required to clear the fault from its power source.	Analyses for the as-built IDS dc electrical distribution system to determine fault currents will be performed.	Analyses for the as-built IDS dc electrical distribution system exist and conclude that the IDS dc electrical distribution system cables will withstand the analyzed fault currents, as determined by manufacturer's ratings, for the time required to clear the fault from its power source as determined by the circuit interrupting device coordination analyses.

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Table 2.6.3-4

Component Name	Tag No.	Component Location
Division A 250 Vdc 24-Hour Battery Bank	IDSA-DB-1	Auxiliary Building
Division B 250 Vdc 24-Hour Battery Bank 1	IDSB-DB-1	Auxiliary Building
Division B 250 Vdc 72-Hour Battery Bank 2	IDSB-DB-2	Auxiliary Building
Division C 250 Vdc 24-Hour Battery Bank 1	IDSC-DB-1	Auxiliary Building
Division C 250 Vdc 72-Hour Battery Bank 2	IDSC-DB-2	Auxiliary Building
Division D 250 Vdc 24-Hour Battery Bank	IDSD-DB-1	Auxiliary Building
Spare 250 Vdc Battery Bank	IDSS-DB-1	Auxiliary Building
Division A 24-Hour Battery Charger 1	IDSA-DC-1	Auxiliary Building
Division B 24-Hour Battery Charger 1	IDSB-DC-1	Auxiliary Building
Division B 72-Hour Battery Charger 2	IDSB-DC-2	Auxiliary Building
Division C 24-Hour Battery Charger 1	IDSC-DC-1	Auxiliary Building
Division C 72-Hour Battery Charger 2	IDSC-DC-2	Auxiliary Building
Division D 24-Hour Battery Charger 1	IDSD-DC-1	Auxiliary Building
Spare Battery Charger 1	IDSS-DC-1	Auxiliary Building
Division A 250 Vdc Distribution Panel	IDSA-DD-1	Auxiliary Building
Division B 250 Vdc Distribution Panel	IDSB-DD-1	Auxiliary Building
Division C 250 Vdc Distribution Panel	IDSC-DD-1	Auxiliary Building
Division D 250 Vdc Distribution Panel	IDSD-DD-1	Auxiliary Building
Division A 120 Vac Distribution Panel 1	IDSA-EA-1	Auxiliary Building
Division A 120 Vac Distribution Panel 2	IDSA-EA-2	Auxiliary Building
Division B 120 Vac Distribution Panel 1	IDSB-EA-1	Auxiliary Building
Division B 120 Vac Distribution Panel 2	IDSB-EA-2	Auxiliary Building
Division B 120 Vac Distribution Panel 3	IDSB-EA-3	Auxiliary Building
Division C 120 Vac Distribution Panel 1	IDSC-EA-1	Auxiliary Building
Division C 120 Vac Distribution Panel 2	IDSC-EA-2	Auxiliary Building
Division C 120 Vac Distribution Panel 3	IDSC-EA-3	Auxiliary Building
Division D 120 Vac Distribution Panel 1	IDSD-EA-1	Auxiliary Building
Division D 120 Vac Distribution Panel 2	IDSD-EA-2	Auxiliary Building

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Table 2.6.3-4 (cont.)

Component Name	Tag No.	Component Location
Division A Fuse Panel 4	IDSA-EA-4	Auxiliary Building
IDSA Battery Monitor Fuse Panel	IDSA-EA-5	Auxiliary Building
Division B Fuse Panel 4	IDSB-EA-4	Auxiliary Building
Division B Fuse Panel 5	IDSB-EA-5	Auxiliary Building
Division B Fuse Panel 6	IDSB-EA-6	Auxiliary Building
IDSB Battery Monitor Fuse Panel	IDSB-EA-7	Auxiliary Building
IDSB Battery Monitor Fuse Panel	IDSB-EA-8	Auxiliary Building
Division C Fuse Panel 4	IDSC-EA-4	Auxiliary Building
Division C Fuse Panel 5	IDSC-EA-5	Auxiliary Building
Division C Fuse Panel 6	IDSC-EA-6	Auxiliary Building
IDSC Battery Monitor Fuse Panel	IDSC-EA-7	Auxiliary Building
IDSC Battery Monitor Fuse Panel	IDSC-EA-8	Auxiliary Building
Division D Fuse Panel 4	IDSD-EA-4	Auxiliary Building
IDSD Battery Monitor Fuse Panel	IDSD-EA-5	Auxiliary Building
IDSS Battery Monitor Fuse Panel	IDSS-EA-1	Auxiliary Building
Division A Fused Transfer Switch Box 1	IDSA-DF-1	Auxiliary Building
Division B Fused Transfer Switch Box 1	IDSB-DF-1	Auxiliary Building
Division B Fused Transfer Switch Box 2	IDSB-DF-2	Auxiliary Building
Division C Fused Transfer Switch Box 1	IDSC-DF-1	Auxiliary Building
Division C Fused Transfer Switch Box 2	IDSC-DF-2	Auxiliary Building
Division D Fused Transfer Switch Box 1	IDSD-DF-1	Auxiliary Building
Spare Fused Transfer Switch Box 1	IDSS-DF-1	Auxiliary Building
Division A 250 Vdc MCC	IDSA-DK-1	Auxiliary Building
Division B 250 Vdc MCC	IDSB-DK-1	Auxiliary Building
Division C 250 Vdc MCC	IDSC-DK-1	Auxiliary Building
Division D 250 Vdc MCC	IDSD-DK-1	Auxiliary Building
Division A 250 Vdc Switchboard 1	IDSA-DS-1	Auxiliary Building
Division B 250 Vdc Switchboard 1	IDSB-DS-1	Auxiliary Building
Division B 250 Vdc Switchboard 2	IDSB-DS-2	Auxiliary Building

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Table 2.6.3-4 (cont.)

Component Name	Tag No.	Component Location
Division C 250 Vdc Switchboard 1	IDSC-DS-1	Auxiliary Building
Division C 250 Vdc Switchboard 2	IDSC-DS-2	Auxiliary Building
Division D 250 Vdc Switchboard 1	IDSD-DS-1	Auxiliary Building
Division A Regulating Transformer	IDSA-DT-1	Auxiliary Building
Division B Regulating Transformer	IDSB-DT-1	Auxiliary Building
Division C Regulating Transformer	IDSC-DT-1	Auxiliary Building
Division D Regulating Transformer	IDSD-DT-1	Auxiliary Building
Division A 24-Hour Inverter 1	IDSA-DU-1	Auxiliary Building
Division B 24-Hour Inverter 1	IDSB-DU-1	Auxiliary Building
Division B 72-Hour Inverter 2	IDSB-DU-2	Auxiliary Building
Division C 24-Hour Inverter 1	IDSC-DU-1	Auxiliary Building
Division C 72-Hour Inverter 2	IDSC-DU-2	Auxiliary Building
Division D 24-Hour Inverter 1	IDSD-DU-1	Auxiliary Building
Spare Battery Termination Box	IDSS-DF-3	Auxiliary Building

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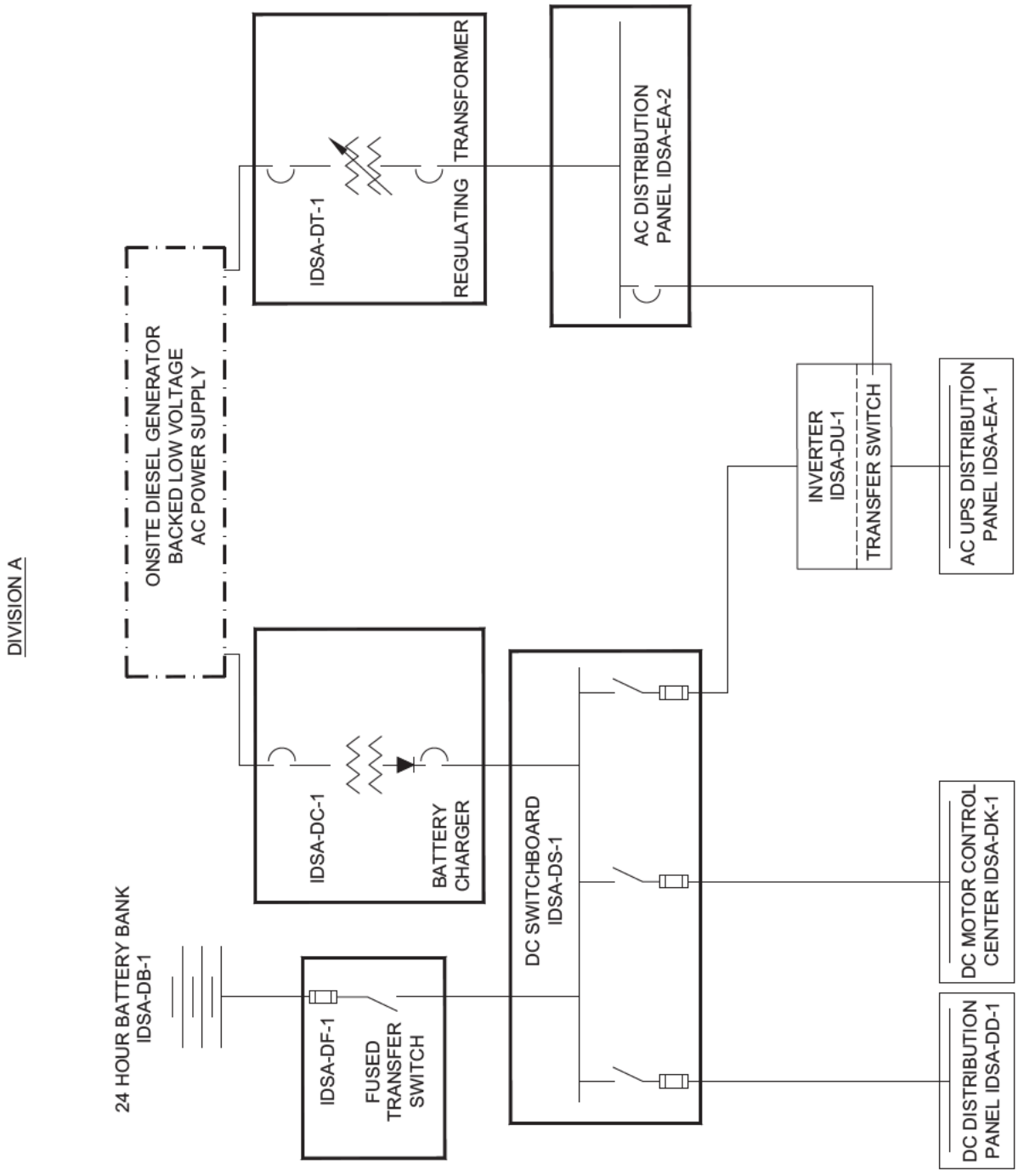

Figure 2.6.3-1 (Sheet 1 of 4)
Class 1E dc and Uninterruptible Power Supply System (Division A)

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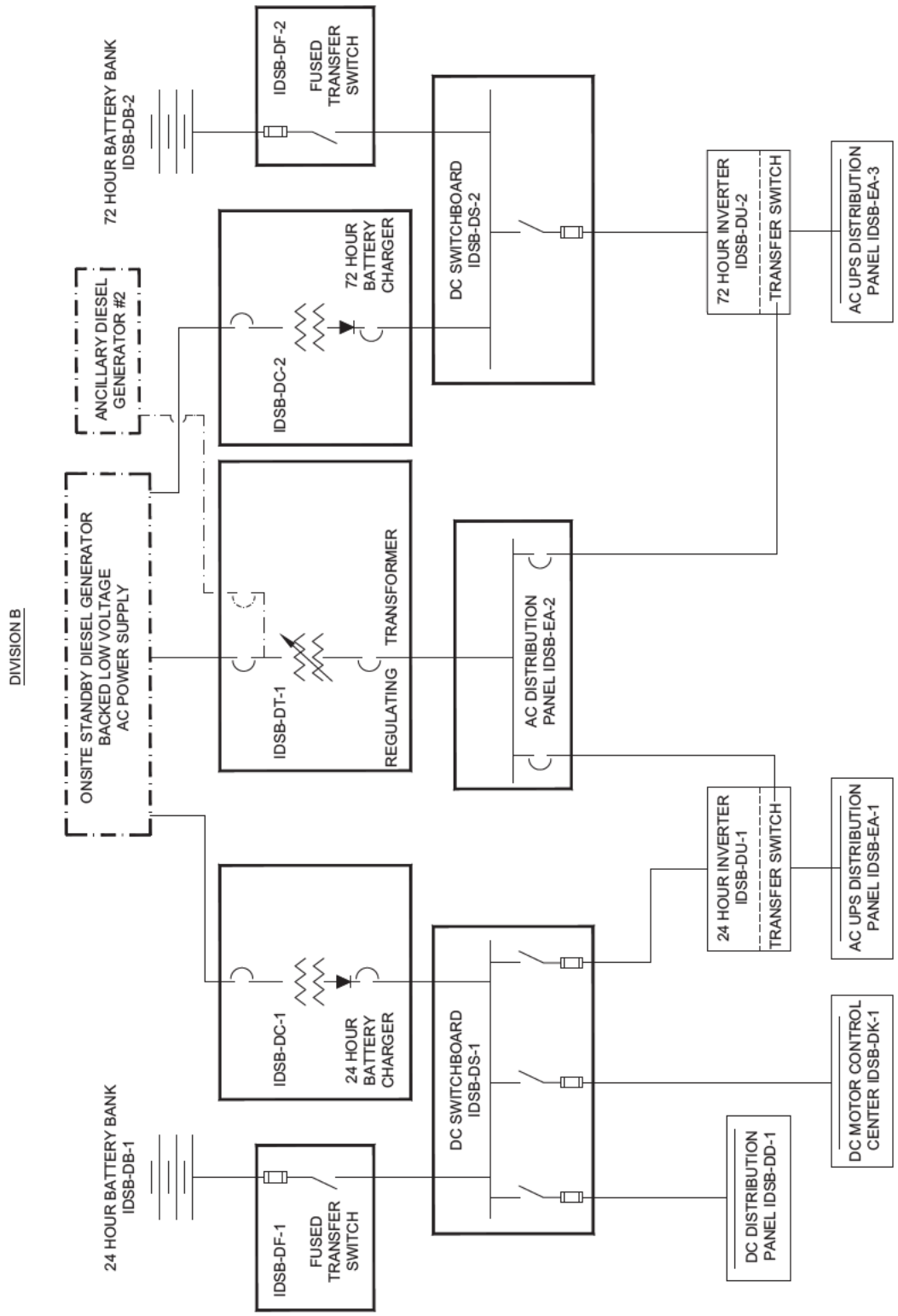

Figure 2.6.3-1 (Sheet 2 of 4)
Class 1E dc and Uninterruptible Power Supply System (Division B)

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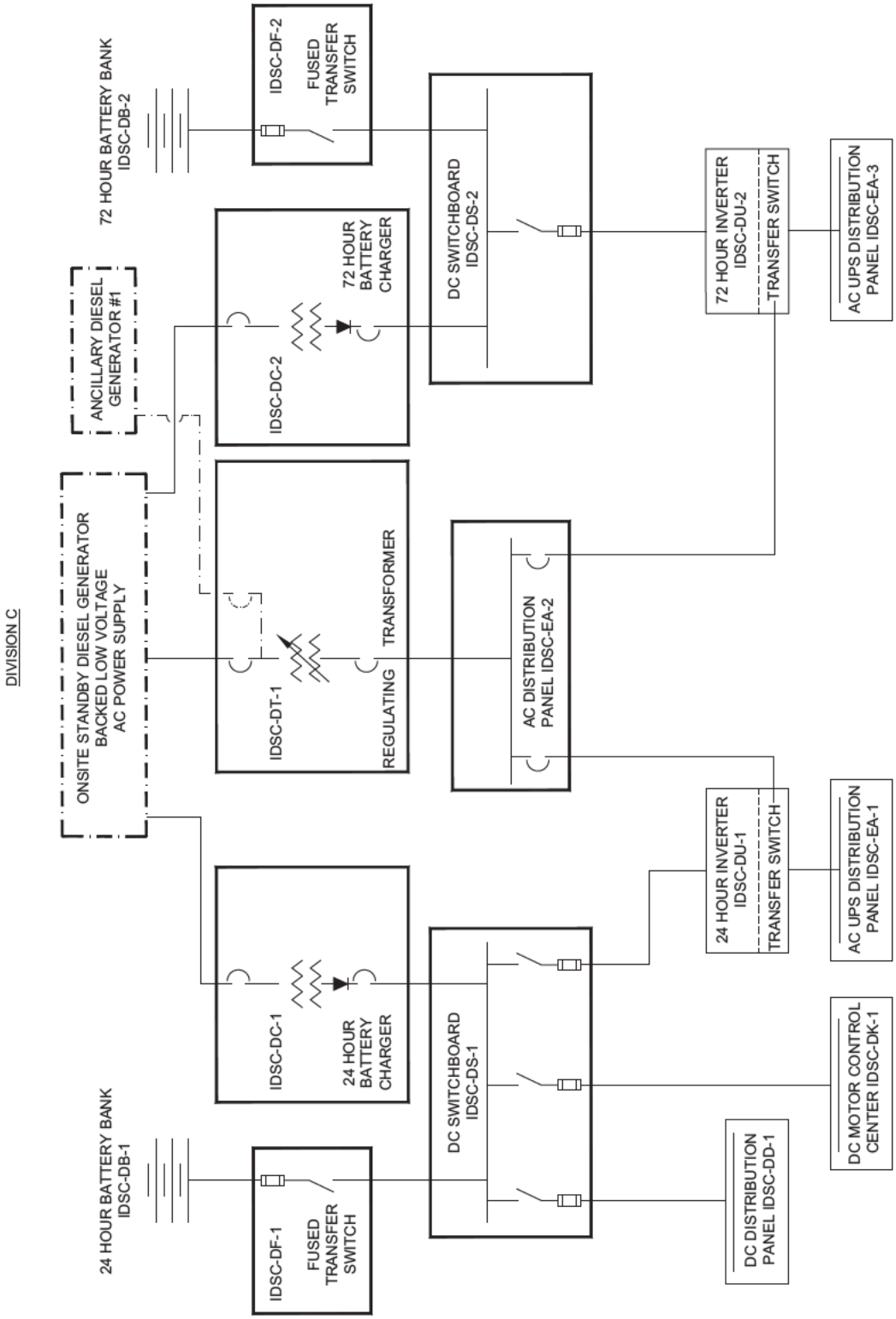

Figure 2.6.3-1 (Sheet 3 of 4)
Class 1E dc and Uninterruptible Power Supply System (Division C)

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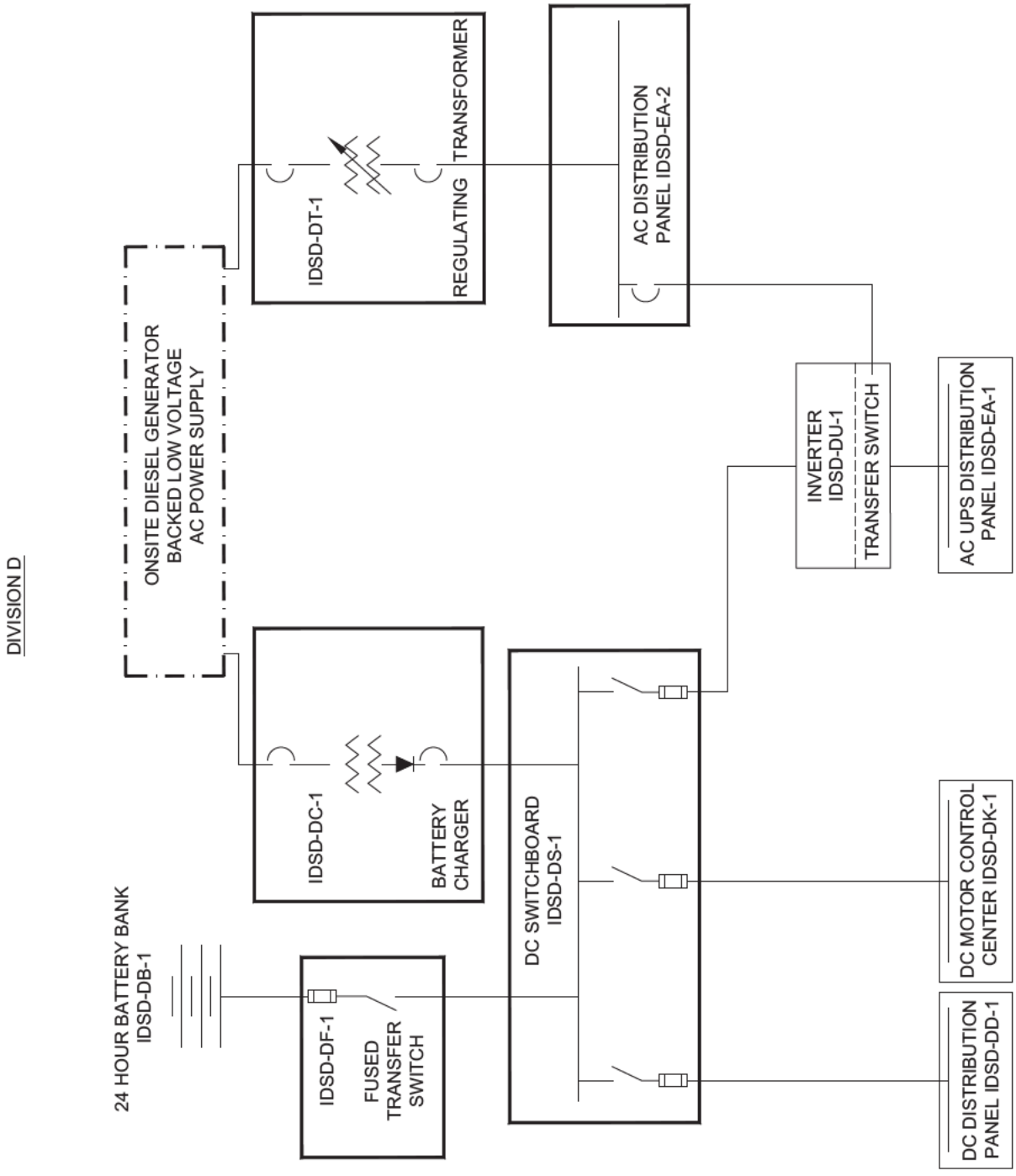

Figure 2.6.3-1 (Sheet 4 of 4)
Class 1E dc and Uninterruptible Power Supply System (Division D)

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2.6.4 Onsite Standby Power System

Design Description

The onsite standby power system (ZOS) provides backup ac electrical power for nonsafety-related loads during normal and off-normal conditions.

The ZOS has two standby diesel generator units and the component locations of the ZOS are as shown in Table 2.6.4-2. The centerline of the diesel engine exhaust gas discharge is located more than twenty (20) feet higher than that of the combustion air intake.

1. The functional arrangement of the ZOS is as described in the Design Description of this Section 2.6.4.

2. The ZOS provides the following nonsafety-related functions:

    a) On loss of power to a 6900 volt diesel-backed bus, the associated diesel generator automatically starts and produces ac power at rated voltage and frequency. The source circuit breakers and bus load circuit breakers are opened, and the generator is connected to the bus.

    b) Each diesel generator unit is sized to supply power to the selected nonsafety-related electrical components.

    c) Automatic-sequence loads are sequentially loaded on the associated buses.

3. Displays of diesel generator status (running/not running) and electrical output power (watts) can be retrieved in the main control room (MCR).

4. Controls exist in the MCR to start and stop each diesel generator.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.4-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the ZOS.

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Table 2.6.4-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) On loss of power to a 6900 volt diesel-backed bus, the associated diesel generator automatically starts and produces ac power at rated voltage and frequency. The source circuit breakers and bus load circuit breakers are opened, and the generator is connected to the bus.	Tests on the as-built ZOS system will be conducted by providing a simulated loss-of-voltage signal. The starting air supply receiver will not be replenished during the test.	Each as-built diesel generator automatically starts on receiving a simulated loss-of-voltage signal and attains a voltage of 6900 $\pm$ 10% V and frequency 60 $\pm$ 5% Hz after the start signal is initiated. The source circuit breakers and bus load circuit breakers are opened, and the generator circuit breaker is closed on the associated 6900 V bus.
2.b) Each diesel generator unit is sized to supply power to the selected nonsafety-related electrical components.	Each diesel generator will be operated with a load of 4000 kW or greater and a power factor between 0.9 and 1.0 for a time period required to reach engine temperature equilibrium plus 2.5 hours.	Each diesel generator provides power to the load with a generator terminal voltage of 6900 $\pm$ 10% V and a frequency of 60 $\pm$ 5% Hz.
3. Displays of diesel generator status (running/not running) and electrical output power (watts) can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays in the MCR.	Displays of diesel generator status and electrical output power can be retrieved in the MCR.
4. Controls exist in the MCR to start and stop each diesel generator.	A test will be performed to verify that controls in the MCR can start and stop each diesel generator.	Controls in the MCR operate to start and stop each diesel generator.
2.c) Automatic-sequence loads are sequentially loaded on the associated buses.	An actual or simulated signal is initiated to start the load sequencer operation. Output signals will be monitored to determine the operability of the load sequencer. Time measurements are taken to determine the load stepping intervals.	The load sequencer initiates a closure signal within ±5 seconds of the set intervals to connect the loads.

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Table 2.6.4-2

Component Name	Tag No.	Component Location
Onsite Diesel Generator A Package	ZOS-MS-05A	Diesel Generator Building
Onsite Diesel Generator B Package	ZOS-MS-05B	Diesel Generator Building

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2.6.5 Lighting System

Design Description

The lighting system (ELS) provides the normal and emergency lighting in the main control room (MCR) and at the remote shutdown workstation (RSW).

1. The functional arrangement of the ELS is as described in the Design Description of this Section 2.6.5.

2. The ELS has six groups of emergency lighting fixtures located in the MCR and at the RSW. Each group is powered by one of the Class 1E inverters. The ELS has four groups of panel lighting fixtures located on or near safety panels in the MCR. Each group is powered by one of the Class 1E inverters in Divisions B and C (one 24-hour and one 72-hour inverter in each Division).

3. The lighting fixtures located in the MCR utilize seismic supports.

4. The panel lighting circuits are classified as associated and treated as Class 1E. These lighting circuits are routed with the Divisions B and C Class 1E circuits. Separation is provided between ELS associated divisions and between associated divisions and non-Class 1E cable.

5. The normal lighting can provide 50 foot candles at the safety panel and at the workstations in the MCR and at the RSW.

6. The emergency lighting can provide 10 foot candles at the safety panel and at the workstations in the MCR and at the RSW.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.5-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the ELS.

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Table 2.6.5-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The ELS has six groups of emergency lighting fixtures located in the MCR and at the RSW. Each group is powered by one of the Class 1E inverters. The ELS has four groups of panel lighting fixtures located on or near safety panels in the MCR. Each group is powered by one of the Class 1E inverters in Divisions B and C (one 24-hour and one 72-hour inverter in each Division).	i) Inspection of the as-built system will be performed.	i) The as-built ELS has six groups of emergency lighting fixtures located in the MCR and at the RSW. The ELS has four groups of panel lighting fixtures located on or near safety panels in the MCR.
	ii) Testing of the as-built system will be performed using one Class 1E inverter at a time.	ii) Each of the six as-built emergency lighting groups is supplied power from its respective Class 1E inverter and each of the four as-built panel lighting groups is supplied power from its respective Class 1E inverter.
5. The normal lighting can provide 50 foot candles at the safety panel and at the workstations in the MCR and at the RSW.	i) Testing of the as-built normal lighting in the MCR will be performed.	i) When adjusted for maximum illumination and powered by the main ac power system, the normal lighting in the MCR provides at least 50 foot candles at the safety panel and at the workstations.
	ii) Testing of the as-built normal lighting at the RSW will be performed.	ii) When adjusted for maximum illumination and powered by the main ac power system, the normal lighting in the RSW provides at least 50 foot candles at the safety panel and at the workstations.
6. The emergency lighting can provide 10 foot candles at the safety panel and at the workstations in the MCR and at the RSW.	i) Testing of the as-built emergency lighting in the MCR will be performed.	i) When adjusted for maximum illumination and powered by the six Class 1E inverters, the emergency lighting in the MCR provides at least 10 foot candles at the safety panel and at the workstations.
	ii) Testing of the as-built emergency lighting at the RSW will be performed.	ii) When adjusted for maximum illumination and powered by the six Class 1E inverters, the emergency lighting provides at least 10 foot candles at the RSW.

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Table 2.6.5-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The lighting fixtures located in the MCR utilize seismic supports.	i) Inspection will be performed to verify that the lighting fixtures located in the MCR are located on the Nuclear Island. ii) Analysis of seismic supports will be performed.	i) The lighting fixtures located in the MCR are located on the Nuclear Island. ii) A report exists and concludes that the seismic supports can withstand seismic design basis loads.
4. Not used.		

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2.6.6 Grounding and Lightning Protection System

Design Description

The grounding and lightning protection system (EGS) provides electrical grounding for instrumentation grounding, equipment grounding, and lightning protection during normal and off-normal conditions.

1. The EGS provides an electrical grounding system for: (1) instrument/computer grounding; (2) electrical system grounding of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, and onsite standby diesel generators; and (3) equipment grounding of equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, motor control centers, and control cabinets. Lightning protection is provided for exposed structures and buildings housing safety-related and fire protection equipment. Each grounding system and lightning protection system is grounded to the station grounding grid.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.6-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the EGS.

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Table 2.6.6-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The EGS provides an electrical grounding system for: (1) instrument/computer grounding; (2) electrical system grounding of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, and onsite standby diesel generators; and (3) equipment grounding of equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, motor control centers, and control cabinets. Lightning protection is provided for exposed structures and buildings housing safety-related and fire protection equipment. Each grounding system and lightning protection system is grounded to the station grounding grid.	i) An inspection for the instrument/ computer grounding system connection to the station grounding grid will be performed. ii) An inspection for the electrical system grounding connection to the station grounding grid will be performed. iii) An inspection for the equipment grounding system connection to the station grounding grid will be performed. iv) An inspection for the lightning protection system connection to the station grounding grid will be performed.	i) A connection exists between the instrument/computer grounding system and the station grounding grid. ii) A connection exists between the electrical system grounding and the station grounding grid. iii) A connection exists between the equipment grounding system and the station grounding grid. iv) A connection exists between the lightning protection system and the station grounding grid.

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2.6.7 Special Process Heat Tracing System

No entry for this system.

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2.6.8 Cathodic Protection System

No entry.

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2.6.9 Plant Security System

Design Description

The physical security system provides physical features to detect, delay, assist response to, and defend against the design basis threat (DBT) for radiological sabotage. The physical security system consists of physical barriers and an intrusion detection system. The details of the physical security system are categorized as Safeguards Information. The physical security system provides protection for vital equipment and plant personnel.

1. The external walls, doors, ceiling, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.

2. Not used.

3. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

4. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.

5. a) Security alarm annunciation and video assessment information is displayed concurrently in the central alarm station and the secondary alarm station, and the video image recording with real time playback capability can provide assessment of activities before and after each alarm annunciation within the perimeter barrier.

   b) The central and secondary alarm stations are located inside the protected area, and the interior of each alarm station is not visible from the perimeter of the protected area.

   c) The central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to detect and assess alarms and communicate with onsite and offsite response personnel.

6. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.

7. a) Vital equipment is located only within a vital area.

   b) Access to vital equipment requires passage through the vital area barrier.

8. Isolation zones and exterior areas within the protected area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.

9. Emergency exits through the vital area boundaries are locked, alarmed, and equipped with a crash bar to allow for emergency egress.

10. Not used.

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11. Not used.

12. Not used.

13. a) The central and secondary alarm stations have conventional (landline) telephone service with the main control room and local law enforcement authorities.

    b) The central and secondary alarm stations are capable of continuous communications with security personnel.

    c) Non-portable communication equipment in the central and secondary alarm stations remains operable from an independent power source in the event of loss of normal power.

14. Not used.

15. a) Security alarm devices including transmission lines to annunciators are tamper indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms and emergency exit alarm) and location.

    b) Intrusion detection and assessment systems concurrently provide visual displays and audible annunciation of alarms in the central and secondary alarm station.

16. Equipment exists to record onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.9-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the physical security system.

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Table 2.6.9-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. Not used.		
3. Not used.		
4. Not used.		
5.a) Security alarm annunciation and video assessment information is displayed concurrently in the central alarm station and the secondary alarm station, and the video image recording with real time playback capability can provide assessment of activities before and after each alarm annunciation within the perimeter area barrier.	Test, inspection, or a combination of test and inspections of the installed systems will be performed.	Security alarm annunciation and video assessment information is displayed concurrently in the central alarm station and the secondary alarm station, and the video image recording with real time playback capability provides assessment of activities before and after alarm annunciation within the perimeter barrier.
15.b) Intrusion detection and assessment systems concurrently provide visual displays and audible annunciation of alarms in the central and secondary alarm stations.	Tests will be performed on intrusion detection and assessment equipment.	The intrusion detection system concurrently provides visual displays and audible annunciations of alarms in both the central and secondary alarm stations.
5.b) The central and secondary alarm stations are located inside the protected area, and the interior of each alarm station is not visible from the perimeter of the protected area.	Inspections of the central and secondary alarm stations will be performed.	The central and secondary alarm stations are located inside the protected area, and the interior of each alarm station is not visible from the perimeter of the protected area.
5.c) The central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to detect and assess alarms and communicate with onsite and offsite response personnel.	Inspections and/or analysis of the central and secondary alarm station will be performed.	The central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, equipment needed to maintain the functional capability of either alarm station to detect and assess alarms and communicate with onsite and offsite response personnel exists.

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Table 2.6.9-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.	Inspections and analysis will be performed for the vehicle barrier system .	The vehicle barrier system will protect against the DBT vehicle bombs based upon the stand-off distance of the system.
7.a) Not used.		
7.b) Not used.		
8. Isolation zones and exterior areas within the protected area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.	Inspection of the illumination in the isolation zones and external areas of the protected area will be performed.	The illumination in isolation zones and exterior areas within the protected area is 0.2 foot candles measured horizontally at ground level or, alternatively, sufficient to permit observation.
9. Not used.		
10. Not used.	.	
11. Not used.		
12. Not used.		
13.a) The central and secondary alarm stations have conventional (landline) telephone service with the main control room and local law enforcement authorities.	Tests, inspections, or a combination of tests and inspections of the central and secondary alarm stations' conventional telephone services will be performed.	The central and secondary alarm stations are equipped with conventional (landline) telephone service with the main control room and local law enforcement authorities.
13.b) The central and secondary alarm stations are capable of continuous communication with security personnel.	Tests, inspections, or a combination of tests and inspections of the central and secondary alarm stations' continuous communication capabilities will be performed.	The central and secondary alarm stations are equipped with the capability to continuously communicate with security officers, watchmen, armed response individuals, or any security personnel that have responsibilities during a contingency event.
13.c) Non-portable communication equipment in the central and secondary alarm stations remains operable from an independent power source in the event of loss of normal power.	Tests, inspections, or a combination of tests and inspections of the non-portable communications equipment will be performed.	Non-portable communication devices (including conventional telephone systems) in the central and secondary alarm stations are wired to an independent power supply that enables the system to remain operable in the event of loss of normal power.

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Table 2.6.9-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
14. Not used.		
15.a) Security alarm devices, including transmission lines to annunciators, are tamper indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms and emergency exit alarm) and location.	A test will be performed to verify that security alarms, including transmission lines to annunciators, are tamper indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power) and that alarm annunciation indicates the type of alarm (e.g., intrusion alarms and emergency exit alarms) and location.	A report exists and concludes that security alarm devices, including transmission lines to annunciators, are tamper indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when the system is on standby power) and that alarm annunciation indicates the type of alarm (e.g., intrusion alarms and emergency exit alarms) and location.
16. Equipment exists to record onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.	Test, analysis, or a combination of test and analysis will be performed to ensure that equipment is capable of recording each onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.	A report exists and concludes that equipment is capable of recording each onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.

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2.6.10 Main Generation System

No entry. Covered in Section 2.6.1, Main ac Power System.

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2.6.11 Excitation and Voltage Regulation System

No entry for this system.

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2.7 HVAC Systems

2.7.1 Nuclear Island Nonradioactive Ventilation System

Design Description

The nuclear island nonradioactive ventilation system (VBS) serves the main control room (MCR), control support area (CSA), Class 1E dc equipment rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, remote shutdown room (RSR), reactor coolant pump trip switchgear rooms, adjacent corridors, and passive containment cooling system (PCS) valve room during normal plant operation. The VBS consists of the following independent subsystems: the main control room/control support area HVAC subsystem, the class 1E electrical room HVAC subsystem, and the passive containment cooling system valve room heating and ventilation subsystem. The VBS provides heating, ventilation, and cooling to the areas served when ac power is available. The system provides breathable air to the control room and maintains the main control room and control support area areas at a slightly positive pressure with respect to the adjacent rooms and outside environment during normal operations. The VBS monitors the main control room supply air for radioactive particulate and iodine concentrations and provides filtration of main control room/control support area air during conditions of abnormal "High-1" airborne radioactivity. In addition, the VBS isolates the HVAC penetrations in the main control room boundary on "High-2" particulate or iodine radioactivity in the main control room supply air duct or on a loss of ac power for more than 10 minutes or if main control room differential pressure is below the "Low" setpoint for more than 10 minutes. The Sanitary Drainage System (SDS) also isolates a penetration in the main control room boundary on "High-2" particulate or iodine radioactivity in the main control room supply air duct or on a loss of ac power for more than 10 minutes or if main control room differential pressure is below the "Low" setpoint for more than 10 minutes. Additional penetrations from the SDS and Potable Water System (PWS) into the main control room boundary are maintained leak tight using a loop seal in the piping, and the Waste Water System (WWS) is isolated using a normally closed safety related manual isolation valve. These features support operation of the main control room emergency habitability system (VES), and have been included in Tables 2.7.1-1 and 2.7.1-2.

The VBS is as shown in Figure 2.7.1-1 and the component locations of the VBS are as shown in Table 2.7.1-5.

1. The functional arrangement of the VBS is as described in the Design Description of this subsection 2.7.1.

2. a) The components identified in Table 2.7.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.7.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.7.1-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.7.1-2 as ASME Code Section III meet ASME Code Section III requirements.

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4. a) The components identified in Table 2.7.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.7.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. The seismic Category I equipment identified in Table 2.7.1-1 can withstand seismic design basis loads without loss of safety function.

6. a) The Class 1E components identified in Table 2.7.1-1 are powered from their respective Class 1E division.

   b) Separation is provided between VBS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

7. The VBS and SDS provide the safety-related function to isolate the pipes that penetrate the MCR pressure boundary.

8. The VBS provides the following nonsafety-related functions:

   a) The VBS provides cooling to the MCR, CSA, RSR, and Class 1E electrical rooms.

   b) The VBS provides ventilation cooling to the Class 1E battery rooms.

   c) The VBS maintains MCR and CSA habitability when radioactivity is detected.

   d) The VBS provides ventilation cooling via the ancillary equipment in Table 2.7.1-3 to the MCR and the division B&C Class 1E I&C rooms.

9. Safety-related displays identified in Table 2.7.1-1 can be retrieved in the MCR.

10. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.7.1-1 to perform their active functions.

    b) The valves identified in Table 2.7.1-1 as having protection and safety monitoring system (PMS) control perform their active safety function after receiving a signal from the PMS.

11. After loss of motive power, the valves identified in Table 2.7.1-1 assume the indicated loss of motive power position.

12. Controls exist in the MCR to cause the components identified in Table 2.7.1-3 to perform the listed function.

13. Displays of the parameters identified in Table 2.7.1-3 can be retrieved in the MCR.

14. The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.

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Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.1-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the VBS.

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Table 2.7.1-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS(1)	Active Function	Loss of Motive Power Position
MCR Supply Air Isolation Valve	VBS-PL-V186	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Supply Air Isolation Valve	VBS-PL-V187	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Return Air Isolation Valve	VBS-PL-V188	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Return Air Isolation Valve	VBS-PL-V189	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Exhaust Air Isolation Valve	VBS-PL-V190	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Exhaust Air Isolation Valve	VBS-PL-V191	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
PWS MCR Isolation Valve	PWS-PL-V418	Yes	Yes	No	-/-	No	No	Transfer Closed	-
PWS MCR Isolation Valve	PWS-PL-V420	Yes	Yes	No	-/-	No	No	Transfer Closed	-
PWS MCR Vacuum Relief	PWS-PL-V498	Yes	Yes	No	-/-	No	No	Transfer Open	-
MCR SDS (Vent) Isolation Valve	SDS-PL-V001	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR SDS (Vent) Isolation Valve	SDS-PL-V002	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is

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Table 2.7.1-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/DAS(1)	Active Function	Loss of Motive Power Position
MCR WWS Isolation Valve	WWS-PL-V506	Yes	Yes	No	–	No	No	–	–

1. DAS = diverse actuation system

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Table 2.7.1-2

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
Main Control Room Supply	VBS-L311	Yes	No	No
Main Control Room Exhaust	VBS-L312	Yes	No	No
Main Control Room Toilet Exhaust	VBS-L313	Yes	No	No
Main Control Room Sanitary Vent Line	SDS-PL-L016	Yes	No	No
Main Control Room Sanitary Drain Line	SDS-PL-L179	Yes	No	No
Main Control Room Sanitary Drain Line	SDS-PL-L182	Yes	No	No
Main Control Room Water Line	PWS-PL-L319	Yes	No	No
Main Control Room Water Line	PWS-PL-L320	Yes	No	No
Main Control Room Waste Water Line	WWS-PL-L808	Yes	No	No
Main Control Room Waste Water Line	WWS-PL-L851	Yes	No	No

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Table 2.7.1-3

Equipment	Tag No.	Display	Control Function
Supplemental Air Filtration Unit Fan A	VBS-MA-03A	Yes (Run Status)	Start
Supplemental Air Filtration Unit Fan B	VBS-MA-03B	Yes (Run Status)	Start
MCR/CSA Supply Air Handling Units (AHU) A Fans	VBS-MA-01A VBS-MA-02A	Yes (Run Status)	Start
MCR/CSA Supply AHU B Fans	VBS-MA-01B VBS-MA-02B	Yes (Run Status)	Start
Division "A" and "C" Class 1E Electrical Room AHU A Fans	VBS-MA-05A VBS-MA-06A	Yes (Run Status)	Start
Division "A" and "C" Class 1E Electrical Room AHU C Fans	VBS-MA-05C VBS-MA-06C	Yes (Run Status)	Start
Division "B" and "D" Class 1E Electrical Room AHU B Fans	VBS-MA-05B VBS-MA-06B	Yes (Run Status)	Start
Division "B" and "D" Class 1E Electrical Room AHU D Fans	VBS-MA-05D VBS-MA-06D	Yes (Run Status)	Start
Division "A" and "C" Class 1E Battery Room Exhaust Fans	VBS-MA-07A VBS-MA-07C	Yes (Run Status)	Start
Division "B" and "D" Class 1E Battery Room Exhaust Fans	VBS-MA-07B VBS-MA-07D	Yes (Run Status)	Start
MCR Ancillary Fans	VBS-MA-10A VBS-MA-10B	No	-
Division B Room Ancillary Fan	VBS-MA-11	No	-
Division C Room Ancillary Fan	VBS-MA-12	No	-

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Table 2.7.1-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) The components identified in Table 2.7.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. 2.b) The piping identified in Table 2.7.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components and piping identified in Tables 2.7.1-1 and 2.7.1-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.7.1-1 as ASME Code Section III meet ASME Code Section III requirements. 3.b) Pressure boundary welds in piping identified in Table 2.7.1-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for nondestructive examination of pressure boundary welds.
4.a) The components identified in Table 2.7.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure. 4.b) The piping identified in Table 2.7.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A pressure test will be performed on the components and piping required by the ASME Code Section III to be pressure tested.	A report exists and concludes that the results of the pressure test of the components and piping identified in Tables 2.7.1-1 and 2.7.1-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

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Table 2.7.1-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The seismic Category I equipment identified in Table 2.7.1-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.7.1-1 is located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.7.1-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
6.a) Not used.		
6.b) Not used.		
7. Not used.		
8.a) Not used.		
8.b) Not used.		
8.c) Not used.		
8.d) The VBS provides ventilation cooling via the ancillary equipment in Table 2.7.1-3 to the MCR and the division B&C Class 1E I&C rooms.	Testing will be performed on the components in Table 2.7.1-3.	The fans start and run.
9. Safety-related displays identified in Table 2.7.1-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.7.1-1 can be retrieved in the MCR.
10.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.7.1-1 to perform their active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.7.1-1 using the controls in the MCR.	Controls in the MCR operate to cause the remotely operated valves identified in Table 2.7.1-1 to perform their active functions.
10.b) The valves identified in Table 2.7.1-1 as having PMS control perform their active safety function after receiving a signal from the PMS.	Testing will be performed using real or simulated signals into the PMS.	The valves identified in Table 2.7.1-1 as having PMS control perform their active safety function after receiving a signal from PMS.

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Table 2.7.1-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. After loss of motive power, the remotely operated valves identified in Table 2.7.1-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valves identified in Table 2.7.1-1 assumes the indicated loss of motive power position.
12. Controls exist in the MCR to cause the components identified in Table 2.7.1-3 to perform the listed function.	Testing will be performed on the components in Table 2.7.1-3 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.1-3 to perform the listed functions.
13. Displays of the parameters identified in Table 2.7.1-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.1-3 can be retrieved in the MCR.
14. The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.	The as-built VBS will be operated, and background noise levels in the MCR and RSR will be measured.	The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.

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Table 2.7.1-5

Component Name	Tag No.	Component Location
Supplemental Air Filtration Unit A	VBS-MS-01A	Auxiliary Building
Supplemental Air Filtration Unit B	VBS-MS-01B	Auxiliary Building
MCR/CSA Supply Air Handling Unit A	VBS-MS-02A	Auxiliary Building
MCR/CSA Supply Air Handling Unit B	VBS-MS-02B	Annex Building
Division "A" and "C" Class 1E Electrical Room AHU A	VBS-MS-03A	Auxiliary Building
Division "A" and "C" Class 1E Electrical Room AHU C	VBS-MS-03C	Auxiliary Building
Division "B" and "D" Class 1E Electrical Room AHU B	VBS-MS-03B	Auxiliary Building
Division "B" and "D" Class 1E Electrical Room AHU D	VBS-MS-03D	Auxiliary Building
MCR Toilet Exhaust Fan	VBS-MA-04	Auxiliary Building
Division "A&C" Class 1E Battery Room Exhaust Fan	VBS-MA-07A	Auxiliary Building
Division "A&C" Class 1E Battery Room Exhaust Fan	VBS-MA-07C	Auxiliary Building
Division "B&D" Class 1E Battery Room Exhaust Fan	VBS-MA-07B	Auxiliary Building
Division "B&D" Class 1E Battery Room Exhaust Fan	VBS-MA-07D	Auxiliary Building
PCS Valve Room Vent Fan	VBS-MA-08	Containment Shield Building
CSA Toilet Exhaust Fan	VBS-MA-09	Annex Building
MCR Ancillary Fan A	VBS-MA-10A	Auxiliary Building
MCR Ancillary Fan B	VBS-MA-10B	Auxiliary Building
Division B Ancillary Fan	VBS-MA-11	Auxiliary Building
Division C Ancillary Fan	VBS-MA-12	Auxiliary Building

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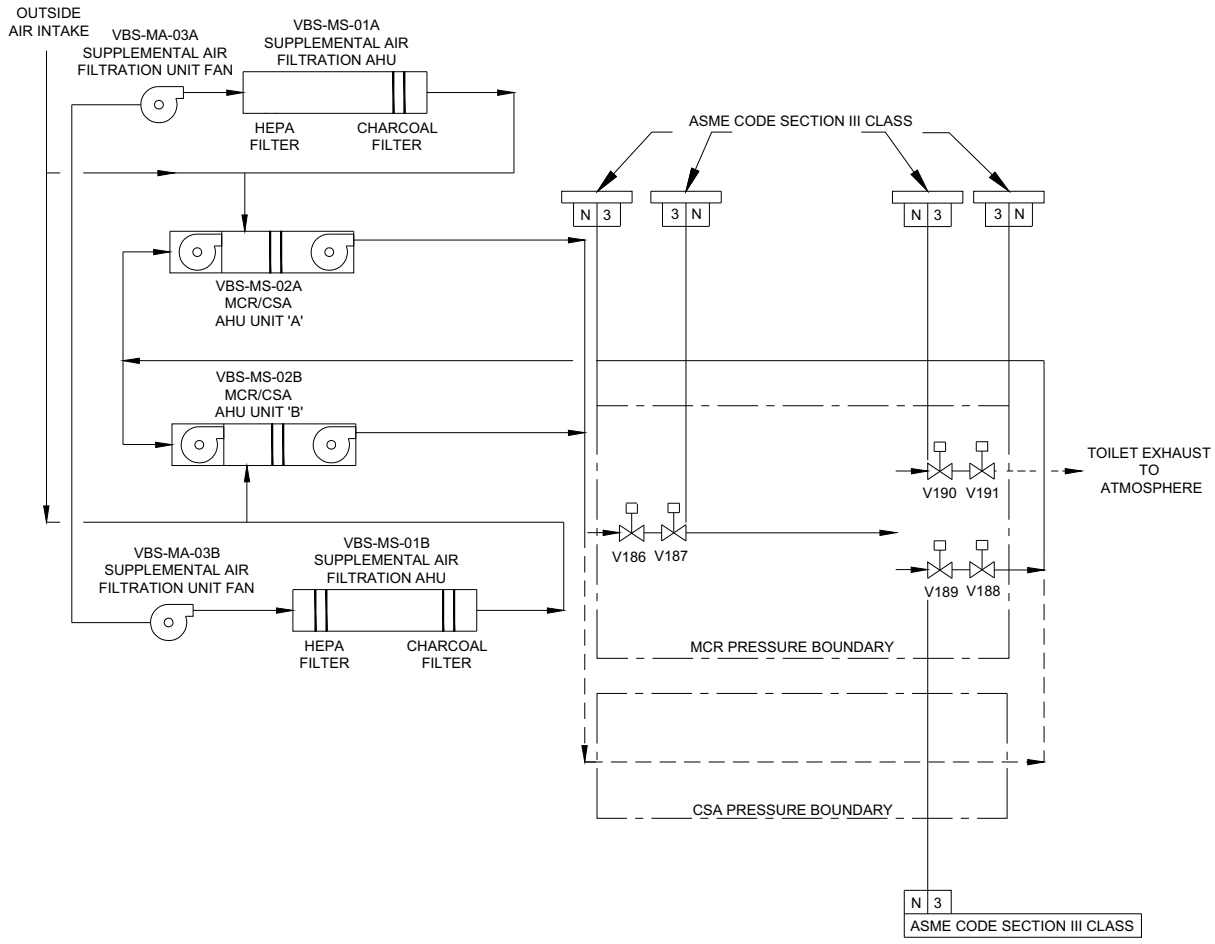

Figure 2.7.1-1 (Sheet 1 of 2)
Nuclear Island Nonradioactive Ventilation System

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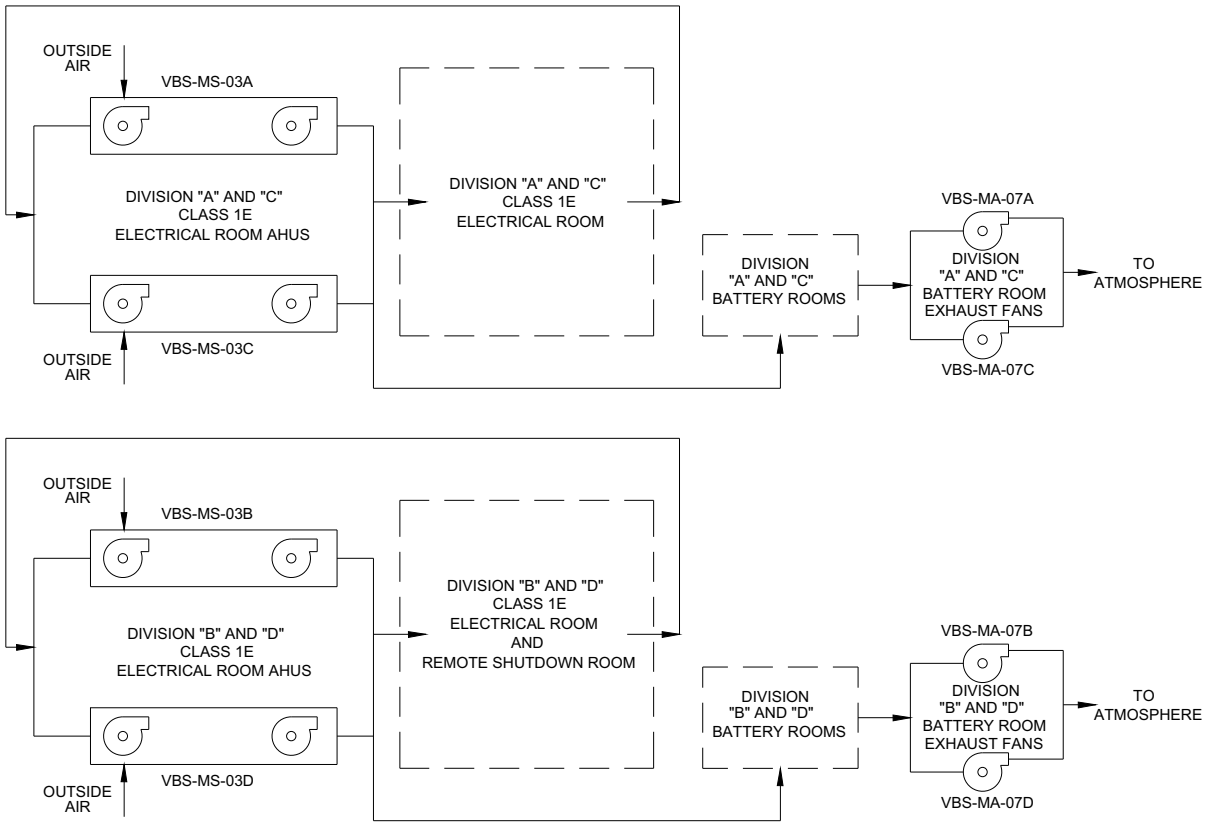

Figure 2.7.1-1 (Sheet 2 of 2)
Nuclear Island Nonradioactive Ventilation System

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2.7.2 Central Chilled Water System

Design Description

The plant heating, ventilation, and air conditioning (HVAC) systems require chilled water as a cooling medium to satisfy the ambient air temperature requirements for the plant. The central chilled water system (VWS) supplies chilled water to the HVAC systems and is functional during reactor full-power and shutdown operation. The VWS also provides chilled water to selected process systems.

The VWS is as shown in Figure 2.7.2-1 and the component locations of the VWS are as shown Table 2.7.2-3.

1. The functional arrangement of the VWS is as described in the Design Description of this Section 2.7.2.

2. The VWS provides the safety-related function of preserving containment integrity by isolation of the VWS lines penetrating the containment.

3. The VWS provides the following nonsafety-related functions:

    a) The VWS provides chilled water to the supply air handling units serving the MCR, the Class 1E electrical rooms, and the unit coolers serving the RNS and CVS pump rooms.

    b) The VWS air-cooled chillers transfer heat from the VWS to the surrounding atmosphere.

4. Controls exist in the MCR to cause the components identified in Table 2.7.2-1 to perform the listed function.

5. Displays of the parameters identified in Table 2.7.2-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.2-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the VWS.

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Table 2.7.2-1

Equipment Name	Tag No.	Display	Control Function
Air-cooled Chiller	VWS-MS-02	Yes (Run Status)	Start
Air-cooled Chiller	VWS-MS-03	Yes (Run Status)	Start
Air-cooled Chiller Pump	VWS-MP-02	Yes (Run Status)	Start
Air-cooled Chiller Pump	VWS-MP-03	Yes (Run Status)	Start
CVS Pump Room Unit Cooler Fan A	VAS-MA-07A	Yes (Run Status)	Start
CVS Pump Room Unit Cooler Fan B	VAS-MA-07B	Yes (Run Status)	Start
RNS Pump Room Unit Cooler Fan A	VAS-MA-08A	Yes (Run Status)	Start
RNS Pump Room Unit Cooler Fan B	VAS-MA-08B	Yes (Run Status)	Start
Air-cooled Chiller Water Valve	VWS-PL-V210	Yes (Position Status)	Open
Air-cooled Chiller Water Valve	VWS-PL-V253	Yes (Position Status)	Open

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Table 2.7.2-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. Not used.		
3.a) The VWS provides chilled water to the supply air handling units serving the MCR, the Class 1E electrical rooms, and the unit coolers serving the RNS and CVS pump rooms.	Testing will be performed by measuring the flow rates to the chilled water cooling coils.	The water flow to each cooling coil equals or exceeds the following: Coil — Flow (gpm) VBS MY C01A/B — 96 VBS MY C02A/C — 97 VBS MY C02B/D — 52 VAS MY C07A/B — 12.3 VAS MY C12A/B — 8.2 VAS MY C06A/B — 8.2
4. Controls exist in the MCR to cause the components identified in Table 2.7.2-1 to perform the listed function.	Testing will be performed on the components in Table 2.7.2-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.2-1 to perform the listed functions.
5. Displays of the parameters identified in Table 2.7.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.7.2-1 can be retrieved in the MCR.
3.b) The VWS air-cooled chillers transfer heat from the VWS to the surrounding atmosphere.	Inspection will be performed for the existence of a report that determines the heat transfer capability of each air-cooled chiller.	A report exists and concludes that the heat transfer rate of each air-cooled chiller is greater than or equal to 230 tons.

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Table 2.7.2-3

Component Name	Tag No.	Component Location
Water Chiller Pump A	VWS-MP-01A	Turbine Building
Water Chiller Pump B	VWS-MP-01B	Turbine Building
Air Cooled Chiller Pump 2	VWS-MP-02	Auxiliary Building
Air Cooled Chiller Pump 3	VWS-MP-03	Annex Building
Water Chiller A	VWS-MS-01A	Turbine Building
Water Chiller B	VWS-MS-01B	Turbine Building
Air Cooled Chiller 2	VWS-MS-02	Auxiliary Building
Air Cooled Chiller 3	VWS-MS-03	Auxiliary Building

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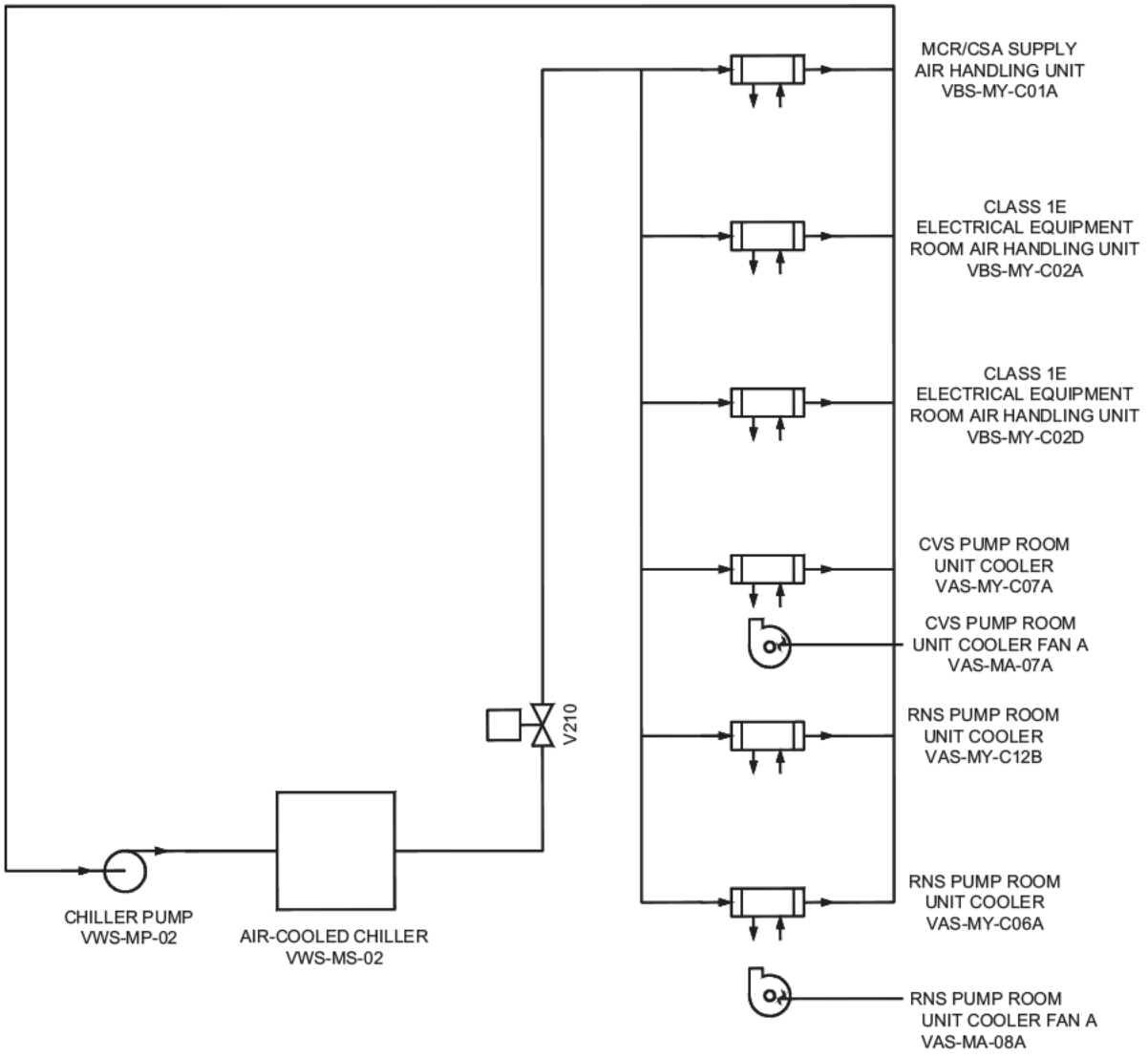

Figure 2.7.2-1 (Sheet 1 of 2)
Central Chilled Water System

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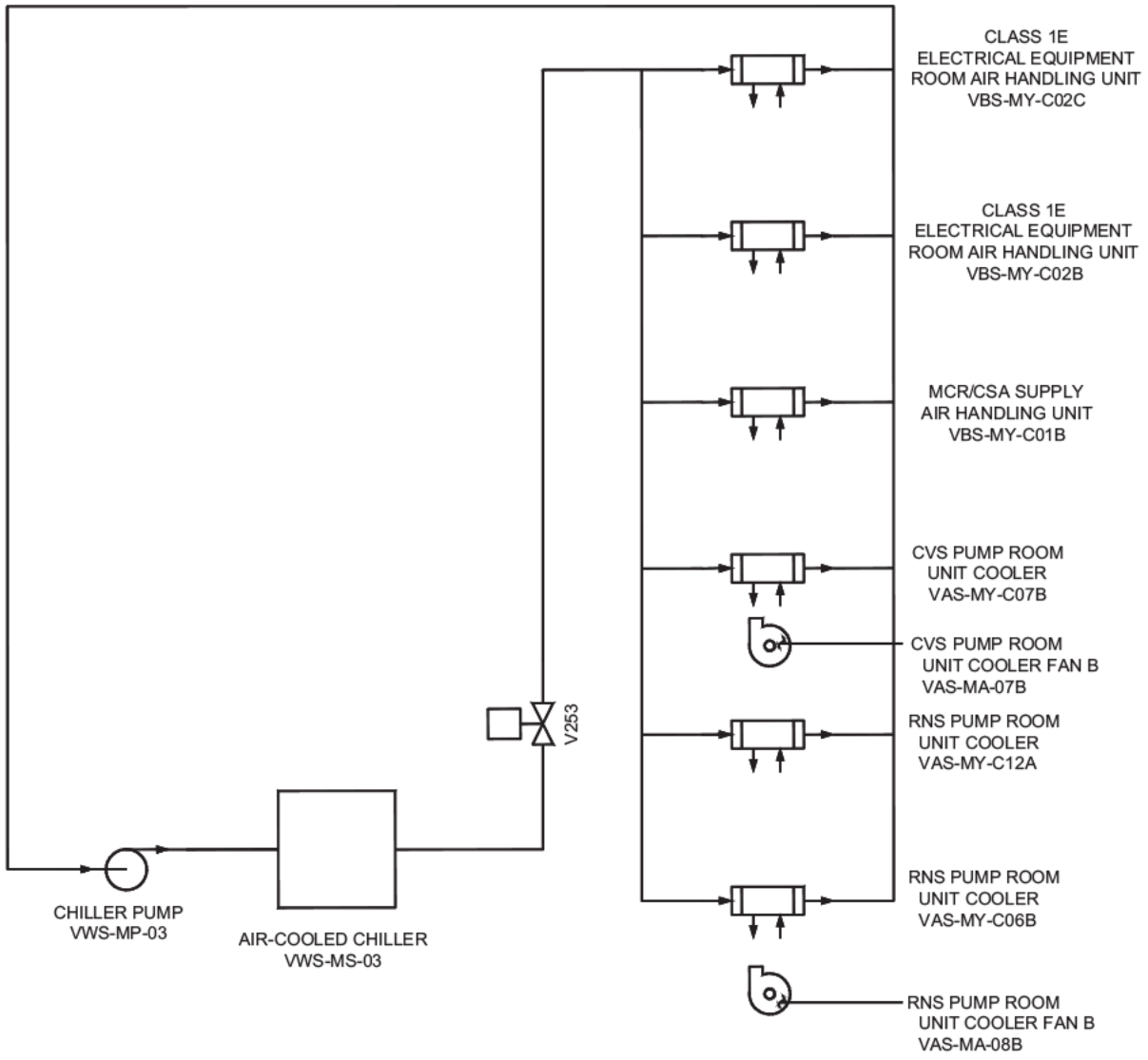

Figure 2.7.2-1 (Sheet 2 of 2)
Central Chilled Water System

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2.7.3 Annex/Auxiliary Building Nonradioactive Ventilation System

Design Description

The annex/auxiliary buildings nonradioactive HVAC system (VXS) serves the nonradioactive personnel and equipment areas, electrical equipment rooms, clean corridors, the ancillary diesel generator room and demineralized water deoxygenating room in the annex building, and the main steam isolation valve compartments, reactor trip switchgear rooms, and piping and electrical penetration areas in the auxiliary building. The VXS consists of the following independent subsystems: the general area HVAC subsystem, the switchgear room HVAC subsystem, the equipment room HVAC subsystem, the MSIV compartment HVAC subsystem, the mechanical equipment areas HVAC subsystem and the valve/piping penetration room HVAC subsystem.

The VXS is as shown in Figure 2.7.3-1 and the component locations of the VXS are as shown in Table 2.7.3-3.

1. The functional arrangement of the VXS is as described in the Design Description of this Section 2.7.3.

2. The VXS provides the following nonsafety-related functions:

    a) The VXS provides cooling to the electrical switchgear, the battery charger, and the annex building nonradioactive air handling equipment rooms.

    b) The VXS provides ventilation cooling to the electrical switchgear, the battery charger, and the annex building nonradioactive air handling equipment rooms when the ZOS operates during a loss of offsite power coincident with loss of chilled water.

3. Controls exist in the main control room (MCR) to cause the components identified in Table 2.7.3-1 to perform the listed function.

4. Displays of the parameters identified in Table 2.7.3-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.3-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the VXS.

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Table 2.7.3-1

Equipment Name	Tag No.	Display	Control Function
Switchgear Room Air Handling Units (AHU) A Fans	VXS-MA-05A VXS-MA-06A	Yes (Run Status)	Start
Switchgear Room AHU B Fans	VXS-MA-05B VXS-MA-06B	Yes (Run Status)	Start
Equipment Room AHU A Fans	VXS-MA-01A VXS-MA-02A	Yes (Run Status)	Start
Equipment Room AHU B Fans	VXS-MA-01B VXS-MA-02B	Yes (Run Status)	Start

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Table 2.7.3-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) Not used.		
2.b) Not used.		
3. Controls exist in the MCR to cause the components identified in Table 2.7.3-1 to perform the listed function.	Testing will be performed on the components in Table 2.7.3-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.3-1 to perform the listed functions.
4. Displays of the parameters identified in Table 2.7.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.3-1 can be retrieved in the MCR.

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Table 2.7.3-3

Component Name	Tag No.	Component Location
Annex Building General Area AHU A	VXS-MS-01A	Annex Building
Annex Building General Area AHU B	VXS-MS-01B	Annex Building
Annex Building Equipment Room AHU A	VXS-MS-02A	Annex Building
Annex Building Equipment Room AHU B	VXS-MS-02B	Annex Building
MSIV Compartment A AHU-A	VXS-MS-04A	Auxiliary Building
MSIV Compartment B AHU-B	VXS-MS-04B	Auxiliary Building
MSIV Compartment B AHU-C	VXS-MS-04C	Auxiliary Building
MSIV Compartment A AHU-D	VXS-MS-04D	Auxiliary Building
Switchgear Room AHU A	VXS-MS-05A	Annex Building
Switchgear Room AHU B	VXS-MS-05B	Annex Building
Mechanical Equipment Area AHU Unit A	VXS-MS-07A	Annex Building
Mechanical Equipment Area AHU Unit B	VXS-MS-07B	Annex Building
Valve/Piping Penetration Room AHU A	VXS-MS-08A	Auxiliary Building
Valve/Piping Penetration Room AHU B	VXS-MS-08B	Auxiliary Building
Battery Room #1 Exhaust Fan	VXS-MA-09A	Annex Building
Battery Room #2 Exhaust Fan	VXS-MA-09B	Annex Building
Toilet Exhaust Fan	VXS-MA-13	Annex Building
Annex Building Nonradioactive Air Handling Equipment Room Unit Heater A	VXS-MY-W01A	Annex Building
Annex Building Nonradioactive Air Handling Equipment Room Unit Heater B	VXS-MY-W01B	Annex Building
Annex Building Nonradioactive Air Handling Equipment Room Unit Heater C	VXS-MY-W01C	Annex Building

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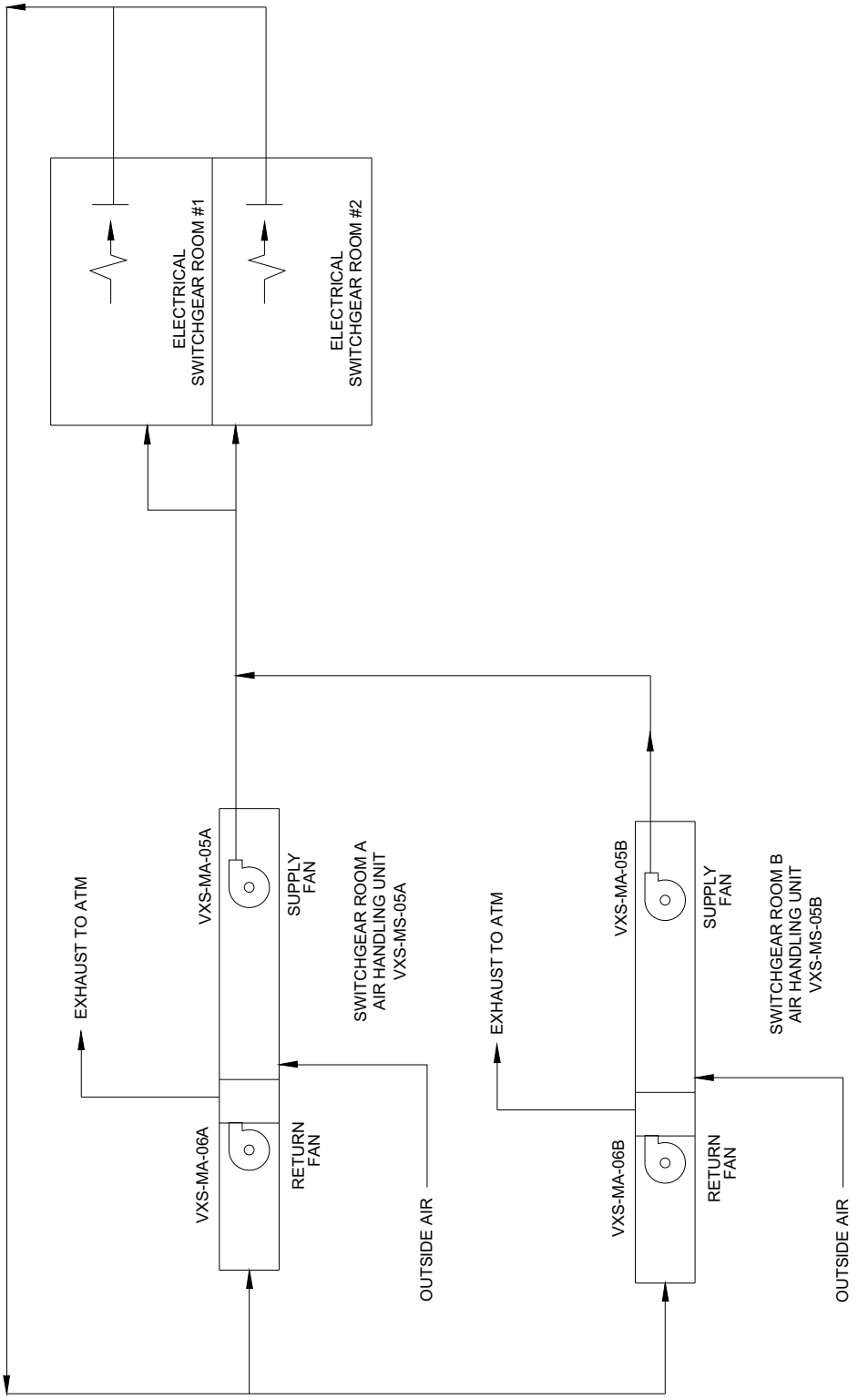

Figure 2.7.3-1 (Sheet 1 of 2)
Annex/Auxiliary Building Nonradioactive Ventilation System

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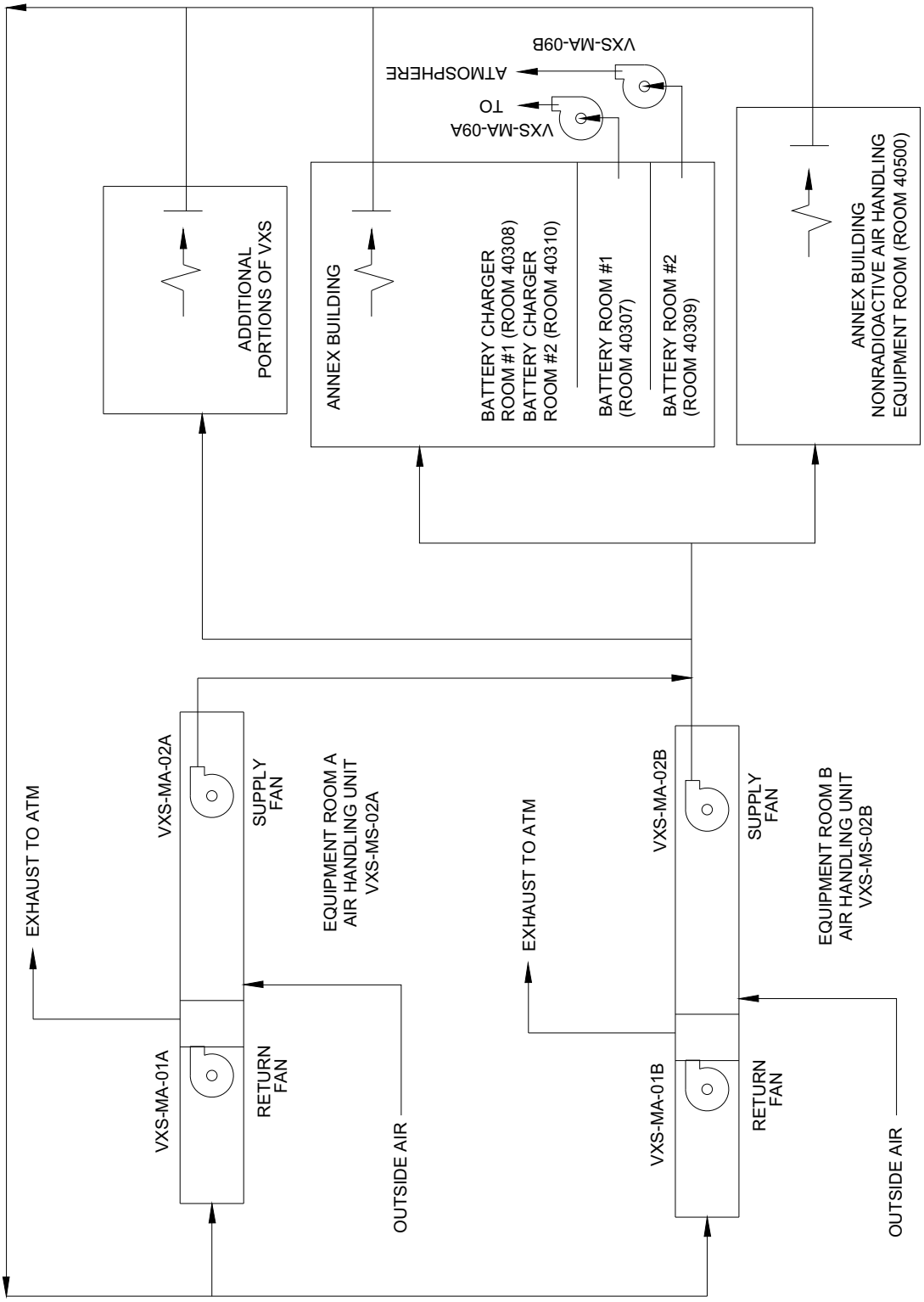

Figure 2.7.3-1 (Sheet 2 of 2)
Annex/Auxiliary Building Nonradioactive Ventilation System

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2.7.4 Diesel Generator Building Ventilation System

Design Description

The diesel generator building ventilation system (VZS) provides ventilation cooling of the diesel generator building for the onsite standby power system. The VZS also provides heating and ventilation within the diesel oil transfer module enclosure. The VZS consists of the following subsystems: the normal diesel building heating and ventilation subsystem, the standby diesel building exhaust ventilation subsystem, the fuel oil day tank vault exhaust subsystem and the diesel oil transfer module enclosures ventilation and heating subsystem.

The VZS is as shown in Figure 2.7.4-1 and the component locations of the VZS are as shown in Table 2.7.4-3.

1. The functional arrangement of the VZS is as described in the Design Description of this Section 2.7.4.

2. The VZS provides the following nonsafety-related functions:

    a) The VZS provides ventilation cooling to the diesel generator rooms when the diesel generators are operating.

    b) The VZS provides ventilation cooling to the electrical equipment service modules when the diesel generators are operating.

    c) The VZS provides normal heating and ventilation to the diesel oil transfer module enclosure.

3. Controls exist in the main control room (MCR) to cause the components identified in Table 2.7.4-1 to perform the listed functions.

4. Displays of the parameters identified in Table 2.7.4-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.4-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the VZS.

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Table 2.7.4-1

Equipment Name	Tag No.	Display	Control Function
Diesel Generator Room A Standby Exhaust Fans	VZS-MY-V01A VZS-MY-V02A	Yes (Run Status)	Start
Diesel Generator Room B Standby Exhaust Fans	VZS-MY-V01B VZS-MY-V02B	Yes (Run Status)	Start
Service Module A Air Handling Units (AHU) Supply Fan	VZS-MA-01A	Yes (Run Status)	Start
Service Module B AHU Supply Fan	VZS-MA-01B	Yes (Run Status)	Start
Diesel Oil Transfer Module Enclosure A Exhaust Fan	VZS-MY-V03A	Yes (Run Status)	Start
Diesel Oil Transfer Module Enclosure A Electric Unit Heater	VZS-MY-U03A	Yes (Run Status)	Energize
Diesel Oil Transfer Module Enclosure B Exhaust Fan	VZS-MY-V03B	Yes (Run Status)	Start
Diesel Oil Transfer Module Enclosure B Electric Unit Heater	VZS-MY-U03B	Yes (Run Status)	Energize

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Table 2.7.4-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2.a) Not used.		
2.b) Not used.		
2.c) Not used.		
3. Controls exist in the MCR to cause the components identified in Table 2.7.4-1 to perform the listed function.	Testing will be performed on the components in Table 2.7.4-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.4-1 to perform the listed functions.
4. Displays of the parameters identified in Table 2.7.4-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.4-1 can be retrieved in the MCR.

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Table 2.7.4-3

Component Name	Tag No.	Component Location
Service Module AHU A	VZS-MS-01A	Diesel-Generator Building
Service Module AHU B	VZS-MS-01B	Diesel-Generator Building
Diesel Oil Transfer Module Enclosure A Unit Heater	VZS-MY-U03A	Yard
Diesel Oil Transfer Module Enclosure B Unit Heater	VZS-MY-U03B	Yard
D/G Building Standby Exhaust Fan 1A	VZS-MY-V01A	Diesel-Generator Building
D/G Building Standby Exhaust Fan 1B	VZS-MY-V01B	Diesel-Generator Building
D/G Building Standby Exhaust Fan 2A	VZS-MY-V02A	Diesel-Generator Building
D/G Building Standby Exhaust Fan 2B	VZS-MY-V02B	Diesel-Generator Building
Diesel Oil Transfer Module Enclosure A Exhaust Fan	VZS-MY-V03A	Yard
Diesel Oil Transfer Module Enclosure B Exhaust Fan	VZS-MY-V03B	Yard
Fuel Oil Day Tank Vault Exhaust Fan	VZS-MA-02A	Diesel-Generator Building
Fuel Oil Day Tank Vault Exhaust Fan	VZS-MA-02B	Diesel-Generator Building

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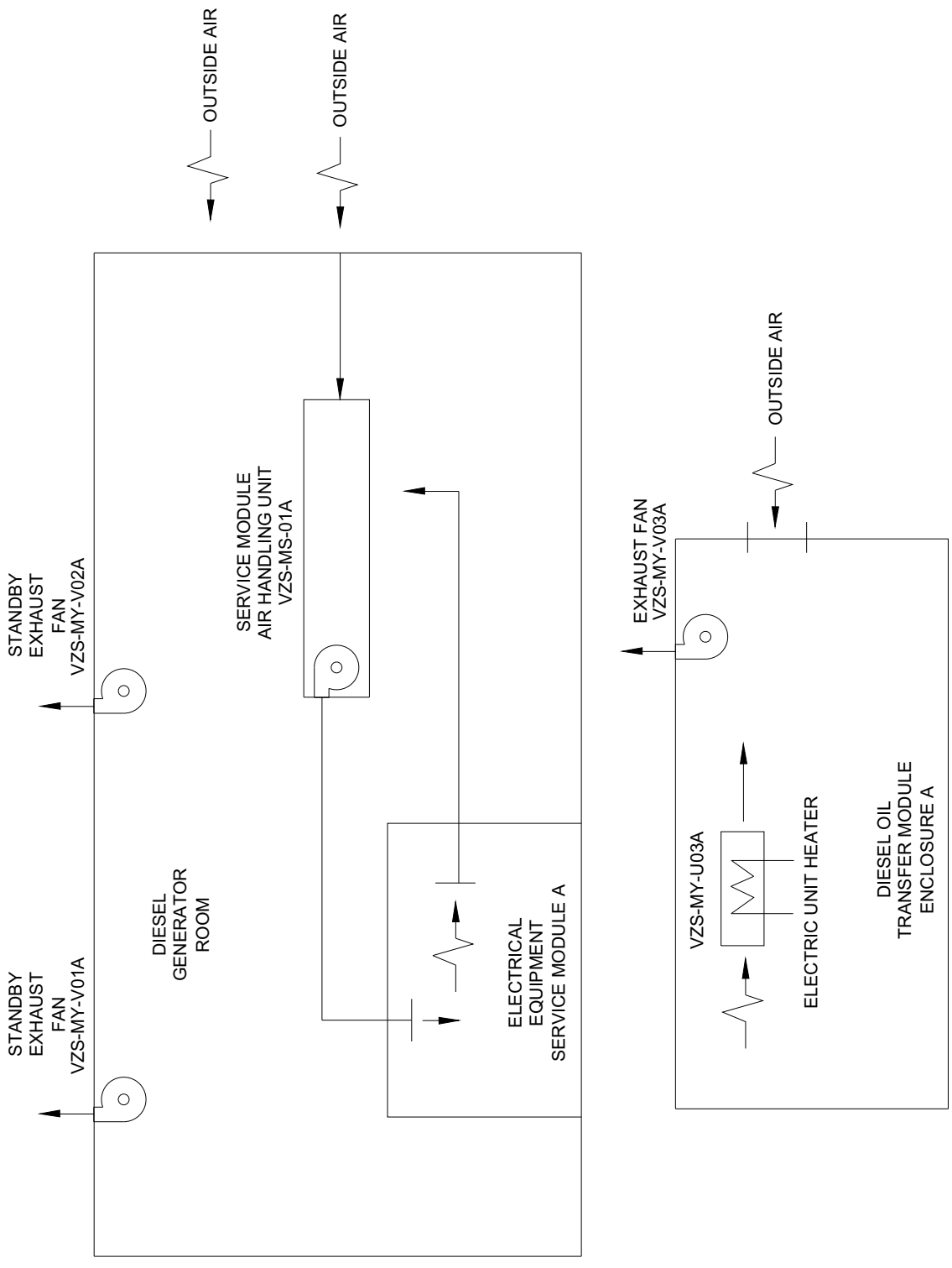

Figure 2.7.4-1 (Sheet 1 of 2)
Diesel Generator Building Ventilation System

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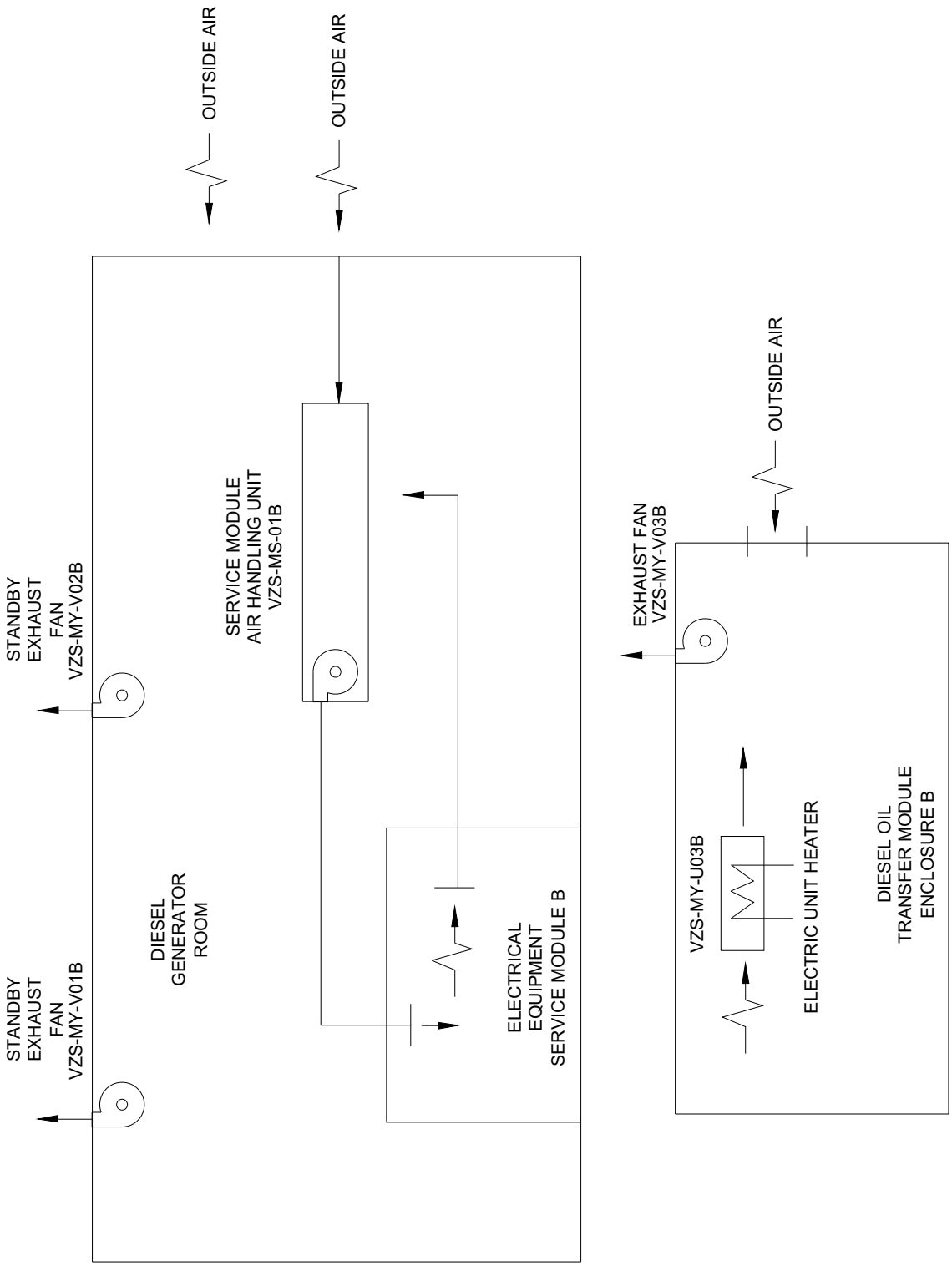

Figure 2.7.4-1 (Sheet 2 of 2)
Diesel Generator Building Ventilation System

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2.7.5 Radiologically Controlled Area Ventilation System

Design Description

The radiologically controlled area ventilation system (VAS) serves the fuel handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas, which are provided with a separate ventilation system (VHS). The VAS consists of two subsystems: the auxiliary/annex building ventilation subsystem and the fuel handling area ventilation subsystem. The subsystems provide ventilation to maintain occupied areas, and access and equipment areas within their design temperature range. They provide outside air for plant personnel and prevent the unmonitored release of airborne radioactivity to the atmosphere or adjacent plant areas. The VAS automatically isolates selected building areas by closing the supply and exhaust duct isolation dampers and starts the containment air filtration system (VFS) when high airborne radioactivity in the exhaust air duct or high ambient pressure differential is detected.

The component locations of the VAS are as shown in Table 2.7.5-3.

1. The functional arrangement of the VAS is as described in the Design Description of this Section 2.7.5.

2. The VAS maintains each building area at a slightly negative pressure relative to the atmosphere or adjacent clean plant areas.

3. Displays of the parameters identified in Table 2.7.5-1 can be retrieved in the main control room (MCR).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.5-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the VAS.

Table 2.7.5-1

Equipment	Tag No.	Display	Control Function
Fuel Handling Area Pressure Differential Indicator	VAS-030	Yes	-
Annex Building Pressure Differential Indicator	VAS-032	Yes	-
Auxiliary Building Pressure Differential Indicator	VAS-033	Yes	-
Auxiliary Building Pressure Differential Indicator	VAS-034	Yes	-

Note: Dash (-) indicates not applicable.

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Table 2.7.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The VAS maintains each building area at a slightly negative pressure relative to the atmosphere or adjacent clean plant areas.	i) Testing will be performed to confirm that the VAS maintains each building at a slightly negative pressure when operating all VAS supply AHUs and all VAS exhaust fans.	i) The time average pressure differential in the served areas of the annex, fuel handling and radiologically controlled auxiliary buildings as measured by each of the instruments identified in Table 2.7.5-1 is negative.
	ii) Testing will be performed to confirm the ventilation flow rate through the auxiliary building fuel handling and rail car bay/solid radwaste system areas when operating all VAS supply AHUs and all VAS exhaust fans.	ii) A report exists and concludes that the calculated exhaust flow rate based on the measured flow rates is greater than or equal to 10,710 cfm.
	iii) Testing will be performed to confirm the auxiliary building radiologically controlled area ventilation flow rate when operating all VAS supply AHUs and all VAS exhaust fans.	iii) A report exists and concludes that the calculated exhaust flow rate based on the measured flow rates is greater than or equal to 22,500 cfm.
3. Displays of the parameters identified in Table 2.7.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.5-1 can be retrieved in the MCR.

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Table 2.7.5-3

Component Name	Tag No.	Component Location
Auxiliary/Annex Building Supply AHU A	VAS-MS-01A	Annex Building
Auxiliary/Annex Building Supply AHU B	VAS-MS-01B	Annex Building
Fuel Handling Area Supply AHU A	VAS-MS-02A	Annex Building
Fuel Handling Area Supply AHU B	VAS-MS-02B	Annex Building
CVS Pump Room Unit Cooler A	VAS-MS-05A	Auxiliary Building
CVS Pump Room Unit Cooler B	VAS-MS-05B	Auxiliary Building
RNS Pump Room Unit Cooler A	VAS-MS-06A	Auxiliary Building
RNS Pump Room Unit Cooler B	VAS-MS-06B	Auxiliary Building
Auxiliary/Annex Building Exhaust Fan A	VAS-MA-02A	Auxiliary Building
Auxiliary/Annex Building Exhaust Fan B	VAS-MA-02B	Auxiliary Building
Fuel Handling Area Exhaust Fan A	VAS-MA-06A	Auxiliary Building
Fuel Handling Area Exhaust Fan B	VAS-MA-06B	Auxiliary Building

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2.7.6 Containment Air Filtration System

Design Description

The containment air filtration system (VFS) provides intermittent flow of outdoor air to purge and filter the containment atmosphere of airborne radioactivity during normal plant operation, and continuous flow during hot or cold plant shutdown conditions to reduce airborne radioactivity levels for personnel access. The VFS can also provide filtered exhaust for the radiologically controlled area ventilation system (VAS) during abnormal conditions.

The VFS is as shown in Figure 2.7.6-1 and the component locations of the VFS are as shown in Table 2.7.6-3.

1. The functional arrangement of the VFS is as described in the Design Description of this Section 2.7.6.

2. The VFS provides the safety-related functions of preserving containment integrity by isolation of the VFS lines penetrating containment and providing vacuum relief for the containment vessel.

3. The VFS provides the intermittent flow of outdoor air to purge the containment atmosphere during normal plant operation, and continuous flow during hot or cold plant shutdown conditions.

4. Controls exist in the main control room (MCR) to cause the components identified in Table 2.7.6-1 to perform the listed function.

5. Displays of the parameters in Table 2.7.6-1 can be retrieved in the MCR.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.6-2 specifies the inspections, tests, analyses, and associated criteria for the VFS.

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Table 2.7.6-1

Equipment	Tag No.	Display	Control Function
Containment Air Handling Units (AHU) Supply Fan A	VFS-MA-01A	Yes (Run Status)	Start
Containment AHU Supply Fan B	VFS-MA-01B	Yes (Run Status)	Start
Containment AHU Supply Fan A Flow Sensor	VFS-012A	Yes	-
Containment AHU Supply Fan B Flow Sensor	VFS-012B	Yes	-
Containment Exhaust Fan A	VFS-MA-02A	Yes (Run Status)	Start
Containment Exhaust Fan B	VFS-MA-02B	Yes (Run Status)	Start
Containment Exhaust Fan A Flow Sensor	VFS-011A	Yes	-
Containment Exhaust Fan B Flow Sensor	VFS-011B	Yes	-

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Table 2.7.6-2

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. The VFS provides the safety-related functions of preserving containment integrity by isolation of the VFS lines penetrating containment and providing vacuum relief for the containment vessel.	i) Not used. ii) Testing will be performed to demonstrate that remotely operated containment vacuum relief isolation valves open within the required response time.	ii) The containment vacuum relief isolation valves (VFS-PL-V800A and VFS-PL-V800B) open within 30 seconds.
3. The VFS provides the intermittent flow of outdoor air to purge the containment atmosphere during normal plant operation, and continuous flow during hot or cold plant shutdown conditions.	i) Testing will be performed to confirm that containment supply AHU fan A when operated with containment exhaust fan A provides a flow of outdoor air. ii) Testing will be performed to confirm that containment supply AHU fan B when operated with containment exhaust fan B provides a flow of outdoor air. iii) Inspection will be conducted of the containment purge discharge line (VFS-L204) penetrating the containment.	i) The flow rate measured at each fan is greater than or equal to 3,600 scfm. ii) The flow rate measured at each fan is greater than or equal to 3,600 scfm. iii) The nominal line size is ≥ 36 in.
4. Controls exist in the MCR to cause the components identified in Table 2.7.6-1 to perform the listed function.	Testing will be performed on the components in Table 2.7.6-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.6-1 to perform the listed functions.
5. Displays of the parameters identified in Table 2.7.6-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.6-1 can be retrieved in the MCR.

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Table 2.7.6-3

Component Name	Tag No.	Component Location
Containment Air Filtration Supply AHU A	VFS-MS-01A	Annex Building
Containment Air Filtration Supply AHU B	VFS-MS-01B	Annex Building
Containment Air Filtration Exhaust Unit A	VFS-MS-02A	Annex Building
Containment Air Filtration Exhaust Unit B	VFS-MS-02B	Annex Building

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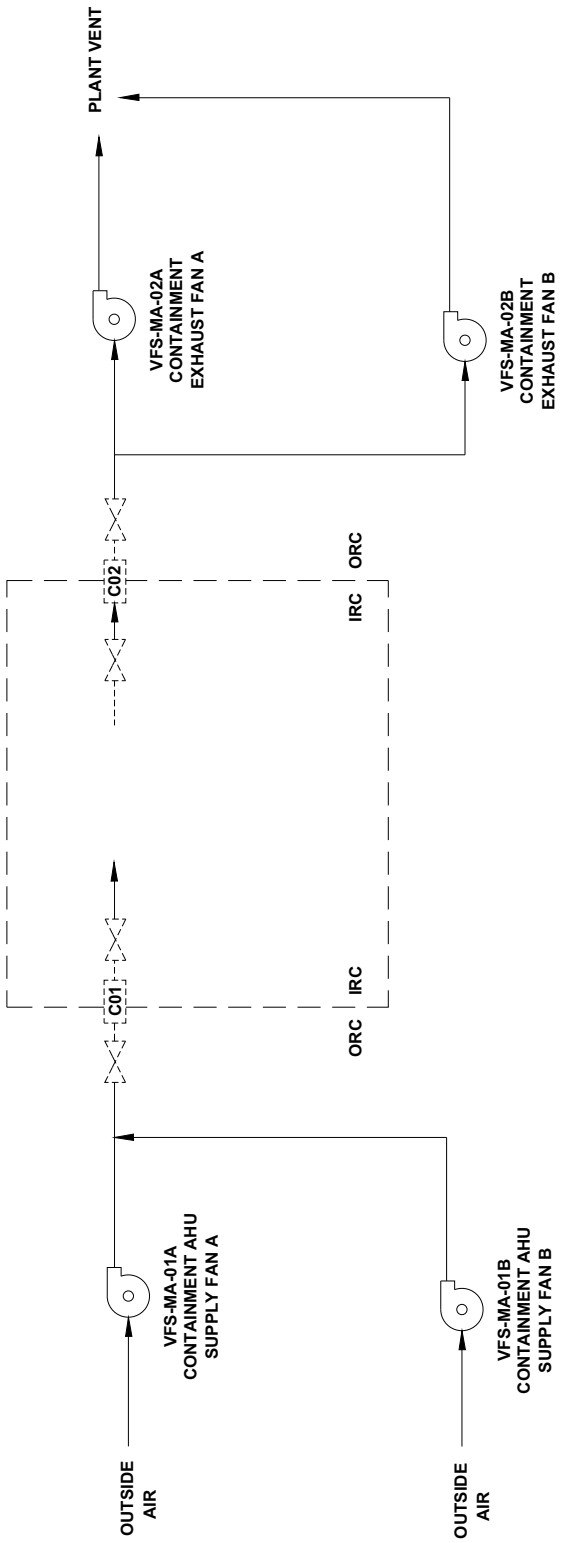

Figure 2.7.6-1
Containment Air Filtration System

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2.7.7 Containment Recirculation Cooling System

Design Description

The containment recirculation cooling system (VCS) controls the containment air temperature and humidity during normal operation, refueling and shutdown.

The locations of the VCS are as shown in Table 2.7.7-3.

1. The functional arrangement of the VCS is as described in the Design Description of this Section 2.7.7.

2. Displays of the parameters identified in Table 2.7.7-1 can be retrieved in the main control room (MCR).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.7-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the VCS.

Table 2.7.7-1

Equipment Name	Tag No.	Display
Containment Temperature Channel	VCS-061	Yes
Containment Fan Cooler Fan	VCS-MA-01A VCS-MA-01C VCS-MA-01B VCS-MA-01D	Yes (Run Status) Yes (Run Status) Yes (Run Status) Yes (Run Status)

Note: Dash (-) indicates not applicable.

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Table 2.7.7-2
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		
2. Displays of the parameters identified in Table 2.7.7-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.7-1 are retrieved in the MCR.

Table 2.7.7-3

Component Name	Tag No.	Component Location
Reactor Containment Recirculation Fan Coil Unit Assembly A	VCS-MS-01A	Containment
Reactor Containment Recirculation Fan Coil Unit Assembly B	VCS-MS-01B	Containment
Reactor Containment Recirculation Fan Coil Unit Assembly C	VCS-MS-01C	Containment
Reactor Containment Recirculation Fan Coil Unit Assembly D	VCS-MS-01D	Containment

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2.7.8 Radwaste Building HVAC System

The radwaste building HVAC system (VRS) serves the radwaste building and provides radiation monitoring of exhaust prior to release to the environment.

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2.7.9 Turbine Island Building Ventilation System

No entry for this system.

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2.7.10 Health Physics and Hot Machine Shop HVAC System

The health physics and hot machine shop system (VHS) serves the health physics and hot machine shop area of the annex building and provides radiation monitoring of exhaust prior to release to the environment.

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2.7.11 Hot Water Heating System

No entry for this system.

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3.1 Emergency Response Facilities

Design Description

The technical support center (TSC) is a facility from which management and technical support is provided to main control room (MCR) personnel during emergency conditions. The operations support center (OSC) provides an assembly area where operations support personnel report in an emergency. The control support area (CSA) is an area nearby the main control room from which support can be provided to the main control room.

1. The TSC has floor space of at least 75 ft^2 per person for a minimum of 25 persons.

2. The TSC has voice communication equipment for communication with the MCR, emergency operations facility, OSC, and the U.S. Nuclear Regulatory Commission (NRC).

3. The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, Data Display and Processing System (DDS), with a "Yes" in the "Display" column, can be retrieved in the TSC.

4. The OSC has voice communication equipment for communication with the MCR and TSC.

5. The TSC and OSC are in different locations.

6. The CSA provides a habitable workspace environment.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the emergency response facilities.

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Table 3.1-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Not used.		
Not used.		
3. The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, DDS, with a "Yes" in the "Display" column, can be retrieved in the TSC.	An inspection will be performed for retrievability of the plant parameters in the TSC.	The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, DDS, with a "Yes" in the "Display" column, can be retrieved in the TSC.
Not used.		
5. The TSC and OSC are in different locations.	An inspection will be performed of the location of the TSC and OSC.	The TSC and OSC are in different locations.
6. Not used.		

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3.2 Human Factors Engineering

Design Description

The AP1000 human-system interface (HSI) will be developed and implemented based upon a human factors engineering (HFE) program. Figure 3.2-1 illustrates the HFE program elements. The HSI scope includes the design of the operation and control centers system (OCS) and each of the HSI resources. For the purposes of the HFE program, the OCS includes the main control room (MCR), the remote shutdown workstation (RSW), the local control stations, and the associated workstations for each of these centers. The HSI resources include the wall panel information system, alarm system, plant information system (nonsafety-related displays), qualified safety-related displays, and soft and dedicated controls. Minimum inventories of controls, displays, and visual alerts are specified as part of the HSI for the MCR and the RSW.

The MCR provides a facility and resources for the safe control and operation of the plant. The MCR includes a minimum inventory of displays, visual alerts and fixed-position controls. Refer to item 8.a and Table 2.5.2-5 of subsection 2.5.2 for this minimum inventory.

The remote shutdown room (RSR) provides a facility and resources to establish and maintain safe shutdown conditions for the plant from a location outside of the MCR. The RSW includes a minimum inventory of displays, controls, and visual alerts. Refer to item 2 and Table 2.5.4-1 of subsection 2.5.4 for this minimum inventory. As stated in item 8.b of subsection 2.5.2, the protection and safety monitoring system (PMS) provides for the transfer of control capability from the MCR to the RSW.

The mission of local control stations is to provide the resources, outside of the MCR, for operations personnel to perform monitoring and control activities.

Implementation of the HFE program includes activity 1 below. The MCR includes design features specified by items 2 through 4 below. The RSW includes the design features specified by items 5 through 8 below. Local control stations include the design feature of item 9.

1. The HFE program verification and validation implementation plans are developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan. The implementation plans establish the methods for conducting evaluations of the integrated HSI design. The development of the HFE verification and validation plans are complete. The following documents were developed:

    a) HSI task support verification – APP-OCS-GEH-220, “AP1000 Human Factors Engineering Task Support Verification Plan,” Westinghouse Electric Company LLC

    b) HFE design verification – APP-OCS-GEH-120, “AP1000 Human Factors Engineering Design Verification Plan,” Westinghouse Electric Company LLC

    c) Integrated system validation – APP-OCS-GEH-320, “AP1000 Human Factors Engineering Integrated System Validation Plan,” Westinghouse Electric Company LLC

    d) Issue resolution verification – APP-OCS-GEH-420, “AP1000 Human Factors Engineering Discrepancy Resolution Process,” Westinghouse Electric Company LLC

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e) Plant HFE/HSI (as designed at the time of plant startup) verification – APP-OCS-GEH-520, "AP1000 Plant Startup Human Factors Engineering Design Verification Plan," Westinghouse Electric Company LLC

2. The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.

3. The MCR provides a suitable workspace environment for use by MCR operators.

4. The HSI resources available to the MCR operators include the alarm system, plant information system (nonsafety-related displays), wall panel information system, nonsafety-related controls (soft and dedicated), and computerized procedure system.

5. The RSW includes reactor operator workstation(s) from which licensed operators perform remote shutdown operations.

6. The RSR provides a suitable workspace environment, separate from the MCR, for use by the RSW operators.

7. The HSI resources available at the RSW include the alarm system displays, the plant information system, and the controls.

8. The RSW and the available HSI permit execution of tasks by licensed operators to establish and maintain safe shutdown.

9. The capability to access displays and controls is provided (controls as assigned by the MCR operators) for local control and monitoring from selected locations throughout the plant.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.2-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the HFE program, MCR, RSW, and local control stations.

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Table 3.2-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities:		
a) HSI Task support verification	a) An evaluation of the implementation of the HSI task support verification will be performed.	a) A report exists and concludes that: Task support verification was conducted in conformance with the implementation plan and includes verification that the information and controls provided by the HSI match the display and control requirements generated by the function-based task analyses and the operational sequence analyses.
b) HFE design verification	b) An evaluation of the implementation of the HFE design verification will be performed.	b) A report exists and concludes that: HFE design verification was conducted in conformance with the implementation plan and includes verification that the HSI design is consistent with the AP1000 specific design guidelines developed for each HSI resource.
c) Integrated system validation	c) (i) An evaluation of the implementation of the integrated system validation will be performed.	c) (i) A report exists and concludes that: The test scenarios listed in the implementation plan for integrated system validation were executed in conformance with the plan and noted human deficiencies were addressed.

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Table 3.2-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	c) (ii) Tests and analyses of the following plant evolutions and transients, using a facility that physically represents the MCR configuration and dynamically represents the MCR HSI and the operating characteristics and responses of the AP1000 design, will be performed: - Normal plant heatup and startup to 100% power - Normal plant shutdown and cooldown to cold shutdown - Transients: reactor trip and turbine trip - Accidents: - Small-break LOCA - Large-break LOCA - Steam line break - Feedwater line break - Steam generator tube rupture	c) (ii) A report exists and concludes that: The test and analysis results demonstrate that the MCR operators can perform the following: - Heat up and start up the plant to 100% power - Shut down and cool down the plant to cold shutdown - Bring the plant to safe shutdown following the specified transients - Bring the plant to a safe, stable state following the specified accidents
d) Issue resolution verification	d) An evaluation of the implementation of the HFE design issue resolution verification will be performed.	d) A report exists and concludes that: HFE design issue resolution verification was conducted in conformance with the implementation plan and includes verification that human factors issues documented in the design issues tracking system have been addressed in the final design.
e) Plant HFE/HSI (as designed at the time of plant startup) verification	e) An evaluation of the implementation of the plant HFE/HSI (as designed at the time of plant startup) verification will be performed.	e) A report exists and concludes that: The plant HFE/HSI, as designed at the time of plant startup, is consistent with the HFE/HSI verified in 1.a) through 1.d).

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Table 3.2-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.	An inspection of the MCR workstations and control panels will be performed.	The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.
3. Not used.		
4. The HSI resources available to the MCR operators include the alarm system, plant information system (nonsafety-related displays), wall panel information system, nonsafety-related controls (soft and dedicated), and computerized procedure system.	An inspection of the HSI resources available in the MCR for the MCR operators will be performed.	The HSI (at the time of plant startup) includes an alarm system, plant information system (nonsafety-related displays), wall panel information system, nonsafety-related controls (soft and dedicated), and computerized procedure system.
5. The RSW includes reactor operator workstation(s) from which licensed operators perform remote shutdown operations.	An inspection of the RSW will be performed.	The RSW includes reactor operator workstation(s).
6. Not used.		
7. The HSI resources available at the RSW include the alarm system displays, the plant information system, and the controls.	An inspection of the HSI resources available at the RSW will be performed.	The as-built HSI at the RSW includes the alarm system displays, the plant information system, and the controls.
8. The RSW and the available HSI permit execution of tasks by licensed operators to establish and maintain safe shutdown.	Test and analysis, using a workstation that physically represents the RSW and dynamically represents the RSW HSI and the operating characteristics and responses of the AP1000, will be performed.	A report exists and concludes that the test and analysis results demonstrate that licensed operators can achieve and maintain safe shutdown conditions from the RSW.
9. The capability to access displays and controls is provided (controls as assigned by the MCR operators) for local control and monitoring from selected locations throughout the plant.	An inspection of the local control and monitoring capability is provided.	The capability for local control and monitoring from selected locations throughout the plant exists.

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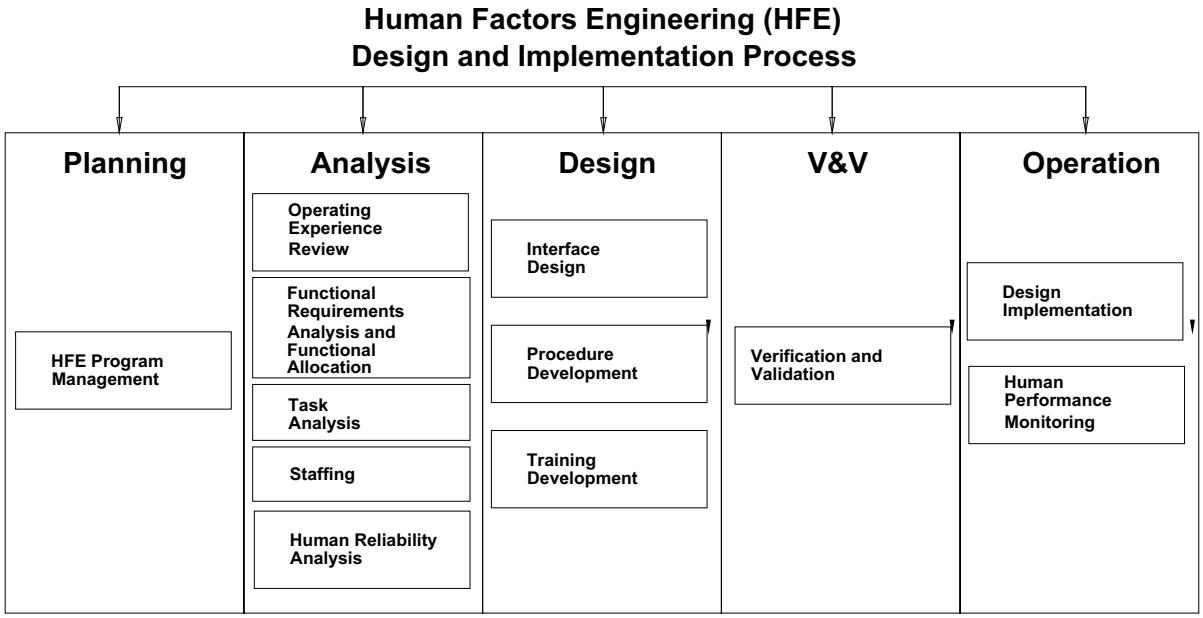

Figure 3.2-1
Human Factors Engineering (HFE)
Design and Implementation Process

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3.3 Buildings

Design Description

The nuclear island structures include the containment (the steel containment vessel and the containment internal structure) and the shield and auxiliary buildings. The containment, shield and auxiliary buildings are structurally integrated on a common basemat which is embedded below the finished plant grade level. The containment vessel is a cylindrical welded steel vessel with elliptical upper and lower heads, supported by embedding a lower segment between the containment internal structures concrete and the basemat concrete. The containment internal structure is reinforced concrete with structural modules used for some walls and floors. The shield building cylinder is a composite steel and concrete (SC) structure except for the portion surrounded by the auxiliary building, which is reinforced concrete (RC). The shield building, in conjunction with the internal structures of the containment building, provides shielding for the reactor coolant system and the other radioactive systems and components housed in the containment. The shield building roof is a reinforced concrete structure containing an integral, steel lined passive containment cooling water storage tank. The auxiliary building is reinforced concrete and houses the safety-related mechanical and electrical equipment located outside the containment and shield buildings.

The portion of the annex building adjacent to the nuclear island is a structural steel and reinforced concrete seismic Category II structure and houses the control support area, non-1E electrical equipment, and hot machine shop.

The radwaste building is a steel framed structure and houses the low level waste processing and storage.

The turbine building is a non-safety related structure that houses the main turbine generator and the power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the turbine building. The turbine building is located on a separate foundation. The turbine building structure is adjacent to the auxiliary building to the south and the annex building to the south and east. The turbine building consists of two separate superstructures, the first bay and the main area, both supported on a common reinforced concrete basemat. The first bay, next to the auxiliary building, consists of a combination of reinforced concrete walls and steel framing with reinforced concrete and steel grated floors. It is classified as a seismic Category II structure due to its immediate proximity to the auxiliary building. The main area of the turbine building, immediately to the north of the first bay, is a steel framed building with reinforced concrete and steel grated floors. It is classified as a non-seismic structure. The non-seismic portion of the turbine building is designed with a mix of concentrically and eccentrically braced framing.

The diesel generator building is a non-safety related structure that houses the two standby diesel engine powered generators and the power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the diesel generator building. The diesel generator building is located on a separate foundation at a distance from the nuclear island structures.

The plant gas system (PGS) provides hydrogen, carbon dioxide, and nitrogen gases to the plant systems as required. The component locations of the PGS are located in the yard areas.

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1. The physical arrangement of the nuclear island structures, the annex building, and the turbine building is as described in the Design Description of this Section 3.3, and as shown on Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building and the diesel generator building is as described in the Design Description of this Section 3.3.

2. a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads, as specified in the Design Description, without loss of structural integrity and the safety-related functions. The design bases loads are those loads associated with:

   - Normal plant operation (including dead loads, live loads, lateral earth pressure loads, and equipment loads, including hydrodynamic loads, temperature and equipment vibration);
   - External events (including rain, snow, flood, tornado, tornado generated missiles and earthquake); and
   - Internal events (including flood, pipe rupture, equipment failure, and equipment failure generated missiles).

   b) Site grade level is located relative to floor elevation 100'-0" per Table 3.3-5. Floor elevation 100'-0" is defined as the elevation of the floor at design plant grade.

   c) The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC.(1)

   d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.

   e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.

   f) The key dimensions of the nuclear island structures are as defined on Table 3.3-5.

   g) The containment vessel above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.

   h) The containment free volume below elevation 107.68' provides containment floodup during a postulated loss-of-coolant accident.

3. Walls and floors of the nuclear island structures as defined on Table 3.3-1, except for designed openings and penetrations, provide shielding during normal operations.

4. a) Walls and floors of the annex building as defined on Table 3.3-1, except for designed openings and penetrations, provide shielding during normal operations.

1. Containment isolation devices are addressed in subsection 2.2.1, Containment System.

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b) The walls on the outside of the waste accumulation room in the radwaste building provide shielding from accumulated waste.

5. a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.

b) The boundaries between mechanical equipment rooms and the electrical and instrumentation and control (I&C) equipment rooms of the auxiliary building as identified in Table 3.3-2 are designed to prevent flooding of rooms that contain safety-related equipment up to the maximum flood level for each room defined in Table 3.3-2.

c) The boundaries between the following rooms, which contain safety-related equipment – passive core cooling system (PXS) valve/accumulator room A (11206), PXS valve/accumulator room B (11207), and chemical and volume system (CVS) room (11209) – are designed to prevent flooding between these rooms.

6. a) The radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" contains adequate volume to contain the liquid volume of faulted liquid radwaste system (WLS) storage tanks. The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceeds the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).

b) The radwaste building waste accumulation room has a volume greater than or equal to 1417 cubic feet.

7. a) Class 1E electrical cables, communication cables associated with only one division, and raceways that route the Class 1E electrical cables and the communication cables are identified according to applicable color-coded Class 1E divisions.

b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.

c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.

d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

e) Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.

8. Systems, structures, and components identified as essential targets are protected from the dynamic and environmental effects of postulated pipe ruptures.

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9. The reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.

10. The shield building roof and the passive containment cooling system (PCS) storage tank support and retain the PCS water. The passive containment cooling system tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided over the tank boundary liner welds.

11. Deleted.

12. The extended turbine generator axis intersects the shield building.

13. Separation is provided between the structural elements of the turbine and annex buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in a safe shutdown earthquake without impact between structural elements of the buildings.

14. The external walls, doors, ceiling, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.

15. Deleted.

16. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

17. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.

18. Deleted.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.3-6 specifies the inspections, tests, analyses, and associated acceptance criteria for the buildings.

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Table 3.3-1
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)(18)	Applicable Radiation Shielding Wall (Yes/No)
Containment Building Internal Structure(15)				
Shield Wall between Reactor Vessel Cavity and RCDT Room	E-W wall parallel with column line 7 (Inside face is 3'-0" north of column line 7. Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity.)	From 71'-6" to 83'-0"	3'-0"(10)	Yes
West Reactor Vessel Cavity Wall	N-S wall parallel with column line N (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity).	From 83'-0" to 98'-0"	7'-6"(10)	Yes

1. The column lines and floor elevations are identified and included on Figures 3.3-1 through 3.3-13.
2. These wall (and floor) thicknesses have a construction tolerance of ±1 inch, except as noted and for exterior walls below grade where the tolerance is +12 inches, -1 inch. These tolerances are not applicable to the nuclear island basemat.
3. For walls that are part of structural modules, the concrete thickness also includes the steel face plates. Where faceplates with a nominal thickness of 0.5 inches are used in the construction of the wall modules, the wall thicknesses in this column apply. Where faceplates thicker than the nominal 0.5 inches are used in the construction of the structural wall modules, the wall thicknesses in the area of the thicker faceplates are greater than indicated in this column by the amount of faceplate thickness increase over the nominal 0.5 inches. Overlay plates are not considered part of the faceplates, and thus are not considered in the wall thicknesses identified in this column.
4. For floors with steel surface plates, the concrete thickness also includes the plate thickness.
5. Where a wall (or a floor) has openings, the concrete thickness does not apply at the opening.
6. The elevation ranges for the shield building items are rounded to the nearest inch.
7. The Wall or Section Description, Column Line information, and Floor Elevation or Elevation Ranges are provided as reference points to define the general location. The concrete thickness of an item intersecting other walls, roofs or floors at a designated location (e.g., column line) is not intended to be measured to the stated column line, but only to the point where the intersection occurs.
8. Where applicable, the upper wall portions extend to their associated roofs, which may vary in elevation, e.g., sloped roofs.
9. From one wall/floor section to another, the concrete thickness transitions from one thickness to another, consistent with the configurations in Figures 3.3-1 through 3.3-14.
10. This wall thickness has a tolerance of ±1-5/8 inch.
11. Not used.
12. The concrete thickness is the total floor thickness, including the metal decking, where applicable.
13. The concrete in the kitchen and restroom areas is 2 inches thinner.
14. The 6-foot concrete thickness includes the thickness of the containment vessel bottom head in a local area at the center of containment.
15. Reconciliation of construction deviations in the nuclear island structures from the thickness and tolerances specified in this table is included in the reconciliation reports, and demonstrate that the as-built structures will withstand design basis loads without loss of structural integrity or safety functions and without impacting compliance with the radiation protection licensing basis, such as GDC 19, established radiological zoning and equipment qualification in accordance with Table 3.3-6.
16. Construction deviations in the annex building from the thickness and tolerances specified in this table are evaluated in the thickness report to demonstrate that the as-built structures will withstand design basis loads without loss of structural integrity or safety functions and without impacting compliance with the radiation protection licensing basis, such as GDC 19, established radiological zoning and equipment qualification in accordance with Table 3.3-6.
17. Construction deviations in the turbine building from the thickness and tolerances specified in this table are evaluated in the thickness report to demonstrate that the as-built structures will withstand design basis loads without loss of structural integrity or safety functions in accordance with Table 3.3-6.
18. Nonconformances from the thicknesses and tolerances specified in Table 3.3-1 (i.e. out of tolerance conditions) are addressed under the 10 CFR Part 50, Appendix B process and subsequently are screened in accordance with the 10 CFR Part 52, Appendix D, Section VIII process, to ensure that the licensing basis is adequately maintained. Construction deviations will continue to be assessed against the licensing basis requirements and will be addressed in accordance with licensee procedures and regulatory requirements and, if applicable, a license amendment will be obtained prior to implementation of the change.

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Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)(18)	Applicable Radiation Shielding Wall (Yes/No)
North Reactor Vessel Cavity Wall	E-W wall parallel with column line 7 (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity).	From 83'-0" to 98'-0"	9'-0"(10)	Yes
East Reactor Vessel Cavity Wall	N-S wall parallel with column line N (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity.)	From 83'-0" to 98'-0"	7'-6"(10)	Yes
West Refueling Cavity Wall	N-S wall parallel with column line N	From 98'-0" to 135'-3"	4'-0"	Yes
North Refueling Cavity Wall	E-W wall parallel with column line 7	From 98'-0" to 135'-3"	4'-0"	Yes
East Refueling Cavity Wall	N-S wall parallel with column line N	From 98'-0" to 135'-3"	4'-0"	Yes
South Refueling Cavity Wall	E-W wall parallel with column line 7	From 98'-0" to 135'-3"	4'-0"	Yes
South wall of west steam generator compartment	Not Applicable	From 103'-0" to 153'-0"	2'-6"	Yes
West wall of west steam generator compartment	N-S wall parallel with column line N	From 103'-0" to 153'-0"	2'-6"	Yes
North wall of west steam generator compartment	Not Applicable	From 103'-0" to 153'-0"	2'-6"	Yes
South wall of pressurizer compartment	Not Applicable	From 103'-0" to 153'-6"	2'-6"	Yes
West wall of pressurizer compartment	N-S wall parallel with column line N	From 107'-2" to 160'-0"	2'-6"	Yes
North wall of pressurizer compartment	E-W wall parallel with column line 7	From 107'-2" to 160'-0"	2'-6"	Yes
East wall of pressurizer compartment	N-S wall parallel with column line N	From 118'-6" to 160'-0"	2'-6"	Yes
North-east wall of in-containment refueling water storage tank	Parallel to column line N	From 103'-0" to 135'-3"	2'-6"	Yes
West wall of in-containment refueling water storage tank	Not applicable	From 103'-0" to 135'-3"	5/8" steel plate with stiffeners	Yes
South wall of east steam generator compartment	Not Applicable	From 87'-6" to 153'-0"	2'-6"	Yes
East wall of east steam generator compartment	N-S wall parallel with column Line N	From 94'-0" to 153'-0"	2'-6"	Yes
North wall of east steam generator compartment	Not Applicable	From 87'-6" to 153'-0" with a 158'-0" portion	2'-6"	Yes

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Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)(18)	Applicable Radiation Shielding Wall (Yes/No)
Shield Building(6)(15)				
Shield Building Cylinder	Not Applicable	From 100'-0" to 248'-6"	3'-0" (including 3/4 inch thick min. steel plate liner on each face on portion not protected by auxiliary building)	Yes
Air Inlet	Not Applicable	From 248'-6" to 251'-6"	3'-0" (including 3/4 inch thick min. steel plate liner on each face)	Yes
		From 251'-6" to 254'-6"	3'-0" to 4'-6" (including 1 inch thick steel plate liner on each face)	Yes
		From 254'-6" to 266'-4"	4'-6" (including 1 inch thick min. steel plate liner on each face)	Yes
Tension Ring	Not Applicable	From 266'-4" to 271'-6" (at top of plate)	4'-6" (including 1-1/2 inch thick steel plate liner on each face)	Yes
Conical Roof	Not Applicable	From 271'-6" to 293'-9"	3'-0" (including 1/2 inch thick min. steel plate liner on bottom face, outside of PCS tank exterior wall)	Yes
PCS Tank External Cylindrical Wall	Not Applicable	From 293'-9" to 328'-9"	2'-0"	Yes
PCS Tank Internal Cylindrical Wall	Not Applicable	From 309'-4" to 329'-0"	1'-6"	Yes
PCS Tank Roof	Not Applicable	328'-9" (Lowest) 329'-0" (Highest)	1'-3"	No
Nuclear Island Basemat	Below shield building	From 60'-6" to containment vessel or 82'-6"	6'-0"(14) to 22'-0" (varies)	No

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Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)(18)	Applicable Radiation Shielding Wall (Yes/No)
Auxiliary Building Walls/Floors Radiologically Controlled(15)				
Column Line 1 wall	From I to N	From 66'-6" to 100'-0"	3'-0"	No
Column Line 1 wall	From I to 5'-7" west of K-2	From 100'-0" to 109'-3"	2'-3"	Yes
Column Line 1 wall	From 5'-7" west of K-2 to N	From 100'-0" to 109'-3"	3'-0"	Yes
Column Line 1 wall	From I to 5'-6" east of L-2	From 109'-3" to 180'-0"	2'-3"	Yes
Column Line 1 wall	From 5'-6" east of L-2 to N	From 109'-3" to 125'-0"	3'-0"	Yes
Column Line 1 wall	From 5'-6" east of L-2 to N	From 125'-0" to 180'-0"	2'-3"	Yes
Column Line 2 wall	From I to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 2 wall	From K-2 to L-2	From 66'-6" to 135'-3"	5'-0"	Yes
Column Line 2 wall	From L-2 to N	From 82'-6" to 135'-3"	2'-6"	Yes
Column Line 2 wall	From I to J-1	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line 3 wall	From J-1 to J-2	From 66'-6" to 82'-6"	2'-6"	Yes
Column Line 3 wall	From J-1 to J-2	From 100'-0" to 135'-3"	2'-6"	Yes
Column Line 3 wall	From J-2 to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 3 wall	From K-2 to L-2	From 66'-6" to 92'-8 1/2"	2'-6"	Yes
Column Line 4 wall	From I to J-1	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From J-1 to J-2	From 66'-6" to 92'-6"	2'-6"	Yes
Column Line 4 wall	From J-1 to J-2	From 107'-2" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From J-2 to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From I to intersection with shield building wall	From 135'-3" to 180'-0"	2'-0"	Yes
Column Line 5 wall	From I to shield building; with opening east of J-1 (below 107'-2" floor).	From 66'-6" to 160'-6"	2'-0"	Yes
Wall, 17'-3" north of Column Line 7	From I to 8' east of J-1	From 66'-6" to 82'-6"	2'-0"	Yes
Wall, 10'-6" south of Column Line 7.3	From I to 5'-6"east of J-1	From 66'-6" to 82'-6"	2'-0"	Yes
Wall, 10'-6" south of Column Line 7.3	From I to just east of J	From 82'-6" to 100'-0"	2'-0"	Yes
Column Line I wall	From 1 to 4	From 66'-6" to 100'-0"	3'-0"	No
Column Line I wall	From 4 to 16'-0" south of 5	From 66'-6" to 107'-2"	3'-0"	No
Column Line I wall	From 16'-0" south of 5 to 5	From 66'-6" to 105'-0"	3'-0"	No
Column Line I wall	From 5 to 7.3	From 66'-6" to 100'-0"	3'-0"	No

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Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building[1]

Wall or Section Description	Column Lines[7]	Floor Elevation or Elevation Range[7][8]	Concrete Thickness[2][3][4][5][9][18]	Applicable Radiation Shielding Wall (Yes/No)
Column Line I wall	From 1 to 3	From 100'-0" to roof	2'-0"	Yes
Column Line I wall	From 3 to 4	From 100'-0" to roof	2'-0"	Yes
Column Line I wall	From 4 to 16'-0" south of 5	From 107'-2" to roof	2'-0"	No
Column Line I wall	From 16'-0" south of 5 to 5	From 105'-0" to roof	2'-0"	No
Column Line J-1 wall	From 1 to 2	From 82'-6" to 100'-0"	2'-0"	Yes
Column Line J-1 wall	From 2 to 4	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line J-1 wall	From 2 to 4	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line J-1 wall	From 4 to shield building	From 66'-6" to 107'-2"	2'-0"	Yes
Column Line J-2 wall	From 2 to 4	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line J-2 wall	From 4 to intersection with shield building wall	From 66'-6" to 135'-3"	2'-0"	Yes
Column Line K-2 wall	From 2 to 4	From 66'-6" to 135'-3"	4'-9"	Yes
Column Line L-2 wall	From 2 to 4	From 66'-6" to 135'-3"	4'-0"	Yes
Column Line N wall	From 1 to 2	From 66'-6" to 100'-0"	3'-0"	No
Column Line N wall	From 1 to 12'-9" north of 1	From 100'-0" to 125'-0"	3'-9"	Yes
Column Line N wall	From 1 to 12'-9" north of 1	From 125'-0" to 135'-3"	2'-0"	Yes
Column Line N wall	From 12'-9" north of 1 to 2	From 100'-0" to 118'-2 1/2"	3'-0"	Yes
Column Line N wall	From 12'-9" north of 1 to 2	From 118'-2 1/2" to 135'-3"	2'-0"	Yes
Column Line N wall	From 2 to 4 (or to shield building)	From 66'-6" to 98'-1"	5'-6"	No
Column Line N wall	From 2 to 4 (or to shield building)	From 98'-1" to 135'-3"	5'-6"	Yes
Column Line N wall	From 1 to 4 (or to shield building)	From 135'-3" to 180'-0"	2'-0"	Yes
Labyrinth Wall between Col. Line 3 and 4 and J-1 to 5'-2" from J-2	Not Applicable	From 82'-6" to 92'-6"	2'-6"	Yes

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Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)(18)	Applicable Radiation Shielding Wall (Yes/No)
N-S Shield Wall (low wall)	5'-7" west of column line K-2 extending 16'-0" from column line 1 north	From 100'-0" to 110'-0"	2'-6"	Yes
N-S Shield Wall	2'-9" east of column line L-2 extending 12'-9" from column line 1 north	From 100'-0" to 109'-3"	3'-0"	Yes
N-S Shield Wall	2'-9" east of column line L-2 extending 12'-9" from column line 1 north	From 109'-3" to 125'-0"	2'-9"	Yes
E-W Shield Wall	10'-0" north of column line 1 extending 13'-3" from column line N east	From 100'-0" to 125'-0"	2'-9"	Yes
Auxiliary Area Basemat	From 1-7.3 and I-N, excluding shield building	From 60'-6" to 66'-6"	6'-0"	No
Floor	From 1 to 2 and I to N	82'-6"	2'-0"	Yes
Floor	From 2 to 4 and J-1 to J-2	82'-6"	2'-0"	Yes
Floor	From 4 north to the shield building ending 17'-4" south of column line 5 and J-1 to J-2	82'-6"	0'-9"	Yes
Pipe Chase Floor	From 2 north to the shield building ending 17'-4" south of column line 5 and J-1 to J-2	92'-6"	2'-0"	Yes
Floor	From 2 to 3 and J-2 to K-2	90'-3"	3'-0"	Yes
Floor	From 3 to 4 and J-2 to K-2	92'-6"	2'-0"	Yes
Floor	From 4 to 7.3 and I to J-1	82'-6"	2'-0"	Yes
Floor	From 1 to 2 and I to N	100'-0"	3'-0"	Yes
Floor	From 2 to 4 and K-2 to L-2	92'-8 1/2"	3'-2 1/2"	Yes
Floor	From I to J-2 and 4 to shield building and vertical wall 17'-0" south of column line 5	107'-2"	2'-0"	Yes
Floor	From I to shield building wall and from intersecting vertical wall before column line 5 to column line 5	105'-0"	0'-9"	Yes

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Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)(18)	Applicable Radiation Shielding Wall (Yes/No)
Floor	From column line 1 to 10'-0" north of column line 1 and from 2'-9" east of column line L-2 to N	125'-0"	3'-0"	Yes
Floor	From 12'-9" north of column line 1 to 2 and from 2'-9" east of column line L-2 to N	118'-2 1/2"	2'-0"	Yes
Floor	From 3 to 4 and J-2 to K-2	117'-6"	2'-0"	Yes
Floor	From 2 to 4 and I to J-1	153'-0"	1'-3"	Yes
Roof	From 1 to 4 and I to N	180'-0" to 180'-9"	1'-3"	Yes
Floor	From 4 to 16'-0" south of column line 5 and from I to intersection with shield building wall	135'-3"	0'-9"	Yes
Floor	From 16'-0" south of column line 5 to column line 5 and from I to intersection with shield building wall	133'-0"	0'-9"	Yes
Auxiliary Building Walls/Floors Non-Radiologically Controlled(15)				
Column Line 11 wall	From I to Q	From 66'-6" to 100'-0"	3'-0"	No
Column Line 11 wall	From I to Q	From 100'-0" to 117'-6"	2'-0"	Yes
Column Line 11 wall	From I to L	From 117'-6" to roof	2'-0"	Yes
Column Line 11 wall	From L to M	From 117'-6" to 135'-3"	4'-0"	Yes
Column Line 11 wall	From M to P	From 117'-6" to 135'-3"	2'-0"	Yes
Column Line 11 wall	From P to Q	From 117'-6" to 135'-3"	4'-0"	Yes
Column Line 11 wall	From L to Q	From 135'-3" to roof	2'-0"	Yes
Column Line 7.3 wall	From I to shield building	From 66'-6" to 100'-0"	3'-0"	Yes
Column Line 7.3 wall	From I to shield building	From 100'-0" to roof	2'-0"	No
Column Line I wall	From 7.3 to 11	From 66'-6" to 100'-0"	3'-0"	No
Column Line I wall	From 5 to 11	From 100'-0" to roof	2'-0"	No
Column Line J wall	From 7.3 to 11	From 66'-6" to 117'-6"	2'-0"	No
Column Line K wall	From shield building wall to 11	From 66'-6" to 135'-3"	2'-0"	Yes
Column Line L wall	From shield building wall to 11	From 66'-6" to roof	2'-0"	Yes
Column Line M wall	From shield building wall to 11	From 66'-6" to roof	2'-0"	Yes

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Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building[1]

Wall or Section Description	Column Lines[7]	Floor Elevation or Elevation Range[7][8]	Concrete Thickness[2][3][4][5][9][18]	Applicable Radiation Shielding Wall (Yes/No)
Column Line P wall	From shield building wall to 11	From 66'-6" to roof	2'-0"	Yes
Column Line Q wall	From shield building wall to 11	From 66'-6" to 100'-0"	3'-0"	No
Column Line Q wall	From shield building wall to 11	From 100'-0" to roof	2'-0"	Yes
Column Line 9.2 wall	From I to J and K to L	From 117'-6" to 135'-3"	2'-0"	Yes
Labyrinth Wall between Column Line 7.3 and 9.2 and J to K	J to K	From 117'-6" to 135'-3"	2'-0"	Yes
Auxiliary Area Basemat	From 7.3-11 and I-Q, excluding shield building	From 60'-6" to 66'-6"	6'-0"	No
Floor	From 5 to 10'-6" south of 7.3 and I to shield building wall	100'-0"	2'-0"	Yes
Floor	From 10'-6" south of 7.3 to 7.3 and I to shield building wall	100'-0"	3'-0"	Yes
Floor	From K to L and shield building wall to column line 10	100'-0"	0'-9"	Yes
Main Control Room Floor	From 9.2 to 11 and I to L	117'-6"	2'-0"	Yes
Floor	Bounded by shield bldg, 7.3, J, 9.2 and L	117'-6"	2'-0"	Yes
Floor	From shield building to 11 and L to Q	117'-6"	2'-0"	Yes
Floor	From 5 to 7.3 and from I to intersection with shield building wall	135'-3"	0'-9"	Yes
Annex Building[16]				
Column line 2 wall	From E to H	From 107'-2" to 135'-3"	19 3/4"	Yes
Column line 4 wall	From E to H	From 107'-2" to 162'-6" & 166'-0"	2'-0"	Yes
N-S Shield Wall between E and F	From 2 to 4	From 107'-2" to 135'-3"	1'-0"	Yes
Column line 4.1 wall	From E to H	From 107'-2" to 135'-3"	2'-0"	Yes
N-S Labyrinth Wall between column line 7.8 and 9 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	Yes
E-W Labyrinth Wall between column line 7.1 and 7.8 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	Yes
Column Line 9 wall	From E to connecting wall between G and H	From 107'-2" to 117'-6"	2'-0"	Yes

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Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)(18)	Applicable Radiation Shielding Wall (Yes/No)
Column Line E wall	From 9 to 13	From 100'-0" to 135'-3"	2'-0"	Yes
Column Line 13 wall	From E to I.1	From 100'-0" to 135'-3"	2'-0"	Yes
Column Line I.1 wall	From 11.09 to 13	From 100'-0" to 135'-3"	2'-0"	Yes
Corridor Wall between G and H	From 9 to near 13	From 100'-0" to 117'-6"	1'-6"	Yes
Corridor Wall between G and H	From 9 to 13	From 117'-6" to 135'-3"	1'-6"	Yes
Column Line 9 wall	From E to H	From 117'-6" to 158'-0"	2'-0"	Yes
Floor	From 2 to 4 and E to H	135'-3"	0'-8"	Yes
Floor	From 4 to 4.1 and E to H	135'-3"	1'-0"(12)	Yes
Floor	From 9 to 13 and E to I.1	117'-6"	0'-8"(12) (13)	Yes
Floor	From 9 to 13 and E to I.1	135'-3"	0'-8"(12)	Yes
Containment Filtration Rms A and B (North Wall)	Between column line E to H	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rms A and B (East wall)	Between column line E to F	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rm A (Floor)	Between column line E to H	135'-3"	1'-0"(12)	Yes
Containment Filtration Rm B (Floor)	Between column line E to H	150'-3"	0'-8"(12)	Yes
Turbine Building(17)				
Wall adjacent to Column Line I.2	From Col. Line 11.05 to 11.2	From 100'-0" to 169'-0"	3'-0"	No
Wall along Column Line 11.2	From near I.2 to near Col. Line R	From 100'-0" to 169'-0"	2'-0"	No
Wall adjacent to Column Line R	From Col. Line 11.2 to Col. Line 11.02	From 100'-0" to 169'-0"	3'-0"	No
Wall along Column Line 11.02	From near Col. Line R to near Col. Line Q	From 100'-0" to 169'-0"	2'-0"	No
Wall along Column Line 11.05	From Col. Line K.4 to near Col. Line I.2	From 100'-0" to 169'-0"	2'-0"	No

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Table 3.3-2
Nuclear Island Building Room Boundaries
Required to Have Flood Barrier Floors and Walls

Boundary/ Maximum Flood Level (inches)	Between Room Number to Room Number	
	Room with Postulated Flooding Source	Adjacent Room
Floor/36	12306	12211
Floor/3	12303	12203/12207
Floor/3	12313	12203/12207
Floor/1	12300	12201/12202/12207 12203/12204/12205
Floor/3	12312	12212
Wall/36	12306	12305
Floor/1	12401	12301/12302/12303 12312/12313
Wall/1	12401	12411/12412
Floor/36	12404	12304
Floor/4	12405	12305
Floor/36	12406	12306
Wall/36	12404	12401
Wall/30	12452	12421
Floor/3	12501	12401/12411/12412
Floor/3	12555	12421/12423/12422
Wall/168	12156	12111/12112
Wall/76.5	12158	12111
Wall/132	12258	12211/12212

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Table 3.3-3
Class 1E Divisions in Nuclear Island Fire Areas

Fire Area Number	Class 1E Divisions			
	A	C	B	D
Auxiliary Building Radiologically Controlled				
1200 AF 01	Yes	Yes	–	–
1220 AF 02	–	–	–	Yes
Auxiliary Building Non-Radiologically Controlled				
1200 AF 03	Yes	Yes	–	–
1201 AF 02	– Note 2	– Note 2	Yes	Yes
1201 AF 03	– Note 2	– Note 2	Yes	Yes
1201 AF 04	–	–	Yes	Yes
1201 AF 05	–	–	Yes	Yes
1201 AF 06	–	–	Yes	Yes
1202 AF 03	– Note 2	Yes	– Note 2	– Note 2
1202 AF 04	Yes	– Note 2	– Note 2	– Note 2
1220 AF 01	–	–	Yes	Yes
1230 AF 01	Yes	Yes	–	–
1230 AF 02	–	–	Yes	Yes
1240 AF 01	Yes	Yes	–	–
1242 AF 02	Yes	Yes	–	–
1250 AF 01	Yes	Yes	–	–

Note 1: Dash (–) indicates not applicable.
Note 2: Cables from this division in this fire area are limited to interdivisional cables terminating in this fire area.

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Table 3.3-4 is not used.

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Table 3.3-5
Key Dimensions of Nuclear Island Building Features

Key Dimension	Reference Dimension (Figure 3.3-14)	Nominal Dimension	Tolerance
Distance between Outside Surface of walls at Column Line I & N when Measured at Column Line 1	X1	91 ft-0 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line 1 to Column Line 7 when Measured at Column Line I	X2	138 ft-0 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line 11 to Column Line 7 when Measured at Column Line I	X3	118 ft-0 in	+3 ft -1 ft
Distance between Outside Surface of walls at Column Line I & Q when Measured at Column Line 11	X4	117 ft-6 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line Q to Column Line N when Measured at Column Line 11	X5	29 ft-0 in	+3 ft -1 ft
Distance between Outside Surface of shield building wall to shield building centerline when Measured on West Edge of Shield Building	X6	72 ft-6 in	+3 ft -1 ft
Distance between shield building centerline to Reactor Vessel centerline when Measured along Column Line N in North-South Direction	X7	7 ft-6 in	± 3 in
Distance from Bottom of Containment Sump to Top Surface of Embedded Containment Shell	–	2 ft-8 in	± 3 in
Distance from top of Basemat to Design Plant Grade	–	33 ft-6 in	± 1 ft
Distance of Design Plant Grade (Floor elevation 100'-0") relative to Site Grade	–	0 ft	± 3 ft-6 in
Distance from Design Plant Grade to Top Surface of Shield Building Roof	–	229 ft-0 in	± 1 ft

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Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The physical arrangement of the nuclear island structures and the annex building is as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building, the turbine building, and the diesel generator building is as described in the Design Description of this Section 3.3.	An inspection of the nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building will be performed.	The as-built nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building conform with the physical arrangement as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14.
2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	i) An inspection of the nuclear island structures will be performed. Deviations from the design due to as-built conditions will be analyzed for the design basis loads, and for radiation shielding.	i.a) A report exists which reconciles deviations during construction, including Table 3.3-1 wall and floor thicknesses, and concludes that the as-built containment internal structures, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions, and without impacting compliance with the radiation protection licensing basis.
3. Walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations, provide shielding during normal operations.		i.b) A report exists which reconciles deviations during construction, including Table 3.3-1 wall and floor thicknesses, and concludes that the as-built shield building structures, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions, and without impacting compliance with the radiation protection licensing basis.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		i.c) A report exists which reconciles deviations during construction, including Table 3.3-1 wall and floor thicknesses, and concludes that the as-built structures in the non-radiologically controlled area of the auxiliary building, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions, and without impacting compliance with the radiation protection licensing basis. i.d) A report exists which reconciles deviations during construction, including Table 3.3-1 wall and floor thicknesses, and concludes that the as-built structures in the radiologically controlled area of the auxiliary building, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions, and without impacting compliance with the radiation protection licensing basis.
	ii) An inspection of the as-built concrete thickness will be performed.	ii.a) Not used. ii.b) Not used. ii.c) Not used. ii.d) Not used.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions. 4.a) Walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	ii.) An inspection of the as-built concrete thickness will be performed.	ii.e) A report exists that concludes that the as-built concrete thicknesses of the annex building sections conform with the building sections defined in Table 3.3-1, except for designed openings or penetrations, or the report documents an evaluation of thickness deviations identified during construction and demonstrates that the as-built structures will withstand the design basis loads without loss of structural integrity and without impacting compliance with the radiation protection licensing basis.
2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	ii.) An inspection of the as-built concrete thickness will be performed.	ii.f) A report exists that concludes that the as-built concrete thicknesses of the turbine building sections conform to the building sections defined in Table 3.3-1, except for designed openings of penetrations, or the report documents an evaluation of thickness deviations identified during construction and demonstrates that the as-built structures will withstand the design basis loads without loss of structural integrity.
2.b) Site grade level is located relative to floor elevation 100'-0" per Table 3.3-5.	Inspection of the as-built site grade will be conducted.	Site grade is consistent with design plant grade within the dimension defined on Table 3.3-5.
2.c) Not used.		
2.d) Not used.		
2.e) Not used.		

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2.f) The key dimensions of nuclear island structures are defined on Table 3.3-5.	An inspection will be performed of the as-built configuration of the nuclear island structures.	A report exists and concludes that the key dimensions of the as-built nuclear island structures are consistent with the dimensions defined on Table 3.3-5.
2.g) The containment vessel above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.	The maximum containment vessel inside height from the operating deck is measured and the inner radius below the spring line is measured at two orthogonal radial directions at one elevation.	The containment vessel maximum inside height from the operating deck is 146'-7" (with tolerance of +12", -6"), and the inside diameter is 130 feet nominal (with tolerance of +12", -6").
2.h) The free volume in the containment allows for floodup to support long-term core cooling for postulated loss-of-coolant accidents.	An inspection will be performed of the as-built containment structures and equipment. The portions of the containment included in this inspection are the volumes that flood with a loss-of-coolant accident in passive core cooling system valve/equipment room B (11207). The in-containment refueling water storage tank volume is excluded from this inspection.	A report exists and concludes that the floodup volume of this portion of the containment is less than 71,960 ft^3 to an elevation of 107.68'.
3. Not used.		
4.a) Not used.		
4.b) Walls of the waste accumulation room in the radwaste building except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built radwaste building wall thicknesses will be performed.	A report exists and concludes that the shield walls of the waste accumulation room in the radwaste building except for designed openings or penetrations are consistent with the minimum concrete wall thicknesses of 1'-4", and a minimum concrete wall thickness of 1'-8" near the radwaste bunkers, or the report documents an evaluation of thickness deviations identified during construction and demonstrates there is no impact to compliance with the radiation protection licensing basis.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.	An inspection of the as-built water barrier will be performed during construction.	A report exists that confirms that a water barrier exists on the nuclear island exterior walls up to site grade.
5.b) The boundaries between rooms identified in Table 3.3-2 of the auxiliary building are designed to prevent flooding of rooms that contain safety-related equipment.	An inspection of the auxiliary building rooms will be performed.	A report exists that confirms floors and walls as identified on Table 3.3-2 have provisions to prevent flooding between rooms up to the maximum flood levels for each room defined in Table 3.3-2.
5.c) The boundaries between the following rooms, which contain safety-related equipment – PXS valve/accumulator room A (11206), PXS valve/accumulator room B (11207), and CVS room (11209) – are designed to prevent flooding between these rooms.	An inspection of the boundaries between the following rooms which contain safety-related equipment – PXS Valve/ Accumulator Room A (11206), PXS Valve/Accumulator Room B (11207), and CVS Room (11209) – will be performed.	A report exists that confirms that flooding of the PXS Valve/ Accumulator Room A (11206), and the PXS Valve/Accumulator Room B (11207) is prevented to a maximum flood level as follows: PXS A 110'-2", PXS B 110'-1"; and of the CVS room (11209) to a maximum flood level of 110'-0".
6.a) The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).	An inspection will be performed of the as-built radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" to define volume.	A report exists and concludes that the as-built available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).
6.b) The radwaste building waste accumulation room has a volume greater than or equal to 1417 cubic feet.	An inspection of the radwaste building waste accumulation room (50351) is performed.	The volume of the radwaste building waste accumulation room (50351) is greater than or equal to 1417 cubic feet.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) Class 1E electrical cables, communication cables associated with only one division, and raceways that route the Class 1E electrical cables and the communication cables are identified according to applicable color-coded Class 1E divisions.	Inspections of the as-built Class 1E cables and the as-built raceways that route the Class 1E cables will be conducted.	a) Class 1E electrical cables, and communication cables associated with only one division, and the raceways that route these cables inside containment are identified by the appropriate color code. b) Class 1E electrical cables, and communication cables associated with only one division, and the raceways that route these cables in the non-radiologically controlled area of the auxiliary building are identified by the appropriate color code. c) Class 1E electrical cables, and communication cables associated with only one division, and the raceways that route these cables in the radiologically controlled area of the auxiliary building are identified by the appropriate color code.
7.b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.	Inspections of the as-built Class 1E divisional cables and the as-built raceways that route the Class 1E cables will be conducted.	a) Class 1E electrical cables and communication cables inside containment associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division. b) Class 1E electrical cables and communication cables in the non-radiologically controlled area of the auxiliary building associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		c) Class 1E electrical cables and communication cables in the radiologically controlled area of the auxiliary building associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.
7.c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.	i) Inspections of the as-built Class 1E division electrical cables, as-built communication cables associated with only one division, and the as-built raceways that route the Class 1E divisional electrical cables and communication cables located in the fire areas identified in Table 3.3-3 will be conducted.	i.a) Results of the inspection will confirm that the separation between Class 1E divisions in the non-radiologically controlled area of the auxiliary building is consistent with Table 3.3-3. i.b) Results of the inspection will confirm that the separation between Class 1E divisions in the radiologically controlled area of the auxiliary building is consistent with Table 3.3-3.
	ii) Inspections of the as-built fire barriers between the fire areas identified in Table 3.3-3 will be conducted.	ii.a) Results of the inspection will confirm that fire barriers exist between fire areas identified in Table 3.3-3 inside the non-radiologically controlled area of the auxiliary building. ii.b) Results of the inspection will confirm that fire barriers exist between fire areas identified in Table 3.3-3 inside the radiologically controlled area of the auxiliary building.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built raceways that route Class 1E cables will be performed to confirm that the separation between raceways that route Class 1E cables of different divisions, and between raceways that route Class 1E cables and raceways that route non-Class 1E cables is consistent with the following:	Results of the inspection will confirm that the separation between raceways that route Class 1E cables of different divisions, and between raceways that route Class 1E cables and raceways that route non-Class 1E cables is consistent with the following:
	i) Within the main control room and remote shutdown room (non-hazard areas), the minimum separation for low-voltage power cables and below is defined by one of the following:	i) Within the main control room and remote shutdown room (non-hazard areas), the minimum separation for low-voltage power cables and below meets one of the following:
	1) For configurations involving open configurations to enclosed configurations with low-voltage power cables, the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.	1) For configurations involving open configurations to enclosed configurations with low-voltage power cables, the vertical separation is 3 inches or more and the horizontal separation is 1 inch or more.
	2) For configurations involving an enclosed raceway and an open raceway with low-voltage power cables, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway.	2) For configurations that involve an enclosed raceway and an open raceway with low-voltage power cables, the minimum vertical separation may be reduced to 1 inch if the enclosed raceway is below the open raceway.
	3) For configurations involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.	3) For configurations that involve enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.
	4) For configurations involving open configurations, and an enclosed raceway and an open raceway, with instrumentation and control cables, the minimum separation is 1 inch in both horizontal and vertical directions.	4) For configurations that involve open configurations, and an enclosed raceway and an open raceway, with instrumentation and control cables, the minimum separation is 1 inch in both horizontal and vertical directions.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	ii.a) Within other plant areas (hazard areas), the minimum separation is defined by one of the following: 1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables ≤2/0 AWG. This minimum vertical separation is 3 inches for the configuration with a conduit above and crossing the open tray at an angle equal to or greater than 45 degrees. 3) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches between a conduit and an open configuration for low-voltage power cables greater than 2/0 AWG but not greater than 750 kcmil. The vertical separation is 3 inches if a conduit is above and crossing an open tray at an angle equal to or greater than 45 degrees. 4) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch.	ii.a) Within other plant areas inside containment (hazard areas), the separation meets one of the following: 1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables ≤2/0 AWG. This minimum vertical separation may be reduced to 3 inches for the configuration with a conduit above and crossing the open tray at an angle equal to or greater than 45 degrees. 3) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches between a conduit and an open configuration for low-voltage power cables greater than 2/0 AWG but not greater than 750 kcmil. The vertical separation may be reduced to 3 inches if a conduit is above and crossing an open tray at an angle equal to or greater than 45 degrees. 4) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5) For configurations involving an enclosed raceway and an open raceway with low-voltage power cables, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway.	5) For configurations that involve an enclosed raceway and an open raceway with low-voltage power cables, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway.
	6) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.	6) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.
	7) The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch for configurations with a non-safety conduit and a free air safety cable with low-voltage power cables and below.	7) The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch for configurations with a non-safety conduit and a free air safety cable with low-voltage power cables and below.
	ii.b) Within other plant areas (limited hazard areas), the minimum separation is defined by one of the following:	ii.b) Within other plant areas inside the non-radiologically controlled area of the auxiliary building (limited hazard areas), the separation meets one of the following:
	1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet.	1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more.
	2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables ≤2/0 AWG. This minimum vertical separation is 3 inches for the configuration with a conduit above and crossing the open tray at an angle equal to or greater than 45 degrees.	2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables ≤2/0 AWG. This minimum vertical separation may be reduced to 3 inches for the configuration with a conduit above and crossing the open tray at an angle equal to or greater than 45 degrees.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	3) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches between a conduit and an open configuration for low-voltage power cables greater than 2/0 AWG but not greater than 750 kcmil. The vertical separation is 3 inches if a conduit is above and crossing an open tray at an angle equal to or greater than 45 degrees.	3) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches between a conduit and an open configuration for low-voltage power cables greater than 2/0 AWG but not greater than 750 kcmil. The vertical separation may be reduced to 3 inches if a conduit is above and crossing an open tray at an angle equal to or greater than 45 degrees.
	4) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch.	4) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch.
	5) For configurations involving an enclosed raceway and an open raceway with low-voltage power cables, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway.	5) For configurations that involve an enclosed raceway and an open raceway with low-voltage power cables, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway.
	6) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.	6) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.
	7) The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch for configurations with a non-safety conduit and a free air safety cable with low-voltage power cables and below.	7) The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch for configurations with a non-safety conduit and a free air safety cable with low-voltage power cables and below.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	ii.c) Within other plant areas (limited hazard areas), the minimum separation is defined by one of the following: 1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables ≤2/0 AWG. This minimum vertical separation is 3 inches for the configuration with a conduit above and crossing the open tray at an angle equal to or greater than 45 degrees. 3) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches between a conduit and an open configuration for low-voltage power cables greater than 2/0 AWG but not greater than 750 kcmil. The vertical separation is 3 inches if a conduit is above and crossing an open tray at an angle equal to or greater than 45 degrees. 4) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch.	ii.c) Within other plant areas inside the radiologically controlled area of the auxiliary building (limited hazard areas), the separation meets one of the following: 1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables ≤2/0 AWG. This minimum vertical separation may be reduced to 3 inches for the configuration with a conduit above and crossing the open tray at an angle equal to or greater than 45 degrees. 3) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches between a conduit and an open configuration for low-voltage power cables greater than 2/0 AWG but not greater than 750 kcmil. The vertical separation may be reduced to 3 inches if a conduit is above and crossing an open tray at an angle equal to or greater than 45 degrees. 4) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	5) For configurations involving an enclosed raceway and an open raceway with low-voltage power cables, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 6) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions. 7) The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch for configurations with a non-safety conduit and a free air safety cable with low-voltage power cables and below.	5) For configurations that involve an enclosed raceway and an open raceway with low-voltage power cables, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 6) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch. 7) The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch for configurations with a non-safety conduit and a free air safety cable with low-voltage power cables and below.
	iii) Where minimum separation distances are not maintained, the circuits are run in enclosed raceways or barriers are provided.	iii.a) Where minimum separation distances are not met inside containment, the circuits are run in enclosed raceways or barriers are provided. iii.b) Where minimum separation distances are not met inside the non-radiologically controlled area of the auxiliary building, the circuits are run in enclosed raceways or barriers are provided. iii.c) Where minimum separation distances are not met inside the radiologically controlled area of the auxiliary building, the circuits are run in enclosed raceways or barriers are provided.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	iv) Separation distances less than those specified above and not run in enclosed raceways or provided with barriers are based on analysis	iv.a) For areas inside containment, a report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed. iv.b) For areas inside the non-radiologically controlled area of the auxiliary building, a report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed. iv.c) For areas inside the radiologically controlled area of the auxiliary building, a report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed.
	v) Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is considered as associated circuits and subject to Class 1E requirements.	v.a) For areas inside containment, non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring. v.b) For areas inside the non-radiologically controlled area of the auxiliary building, non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring. v.c) For areas inside the radiologically controlled area of the auxiliary building, non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.e) Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.	Inspections of the as-built Class 1E communication cables will be conducted.	Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.
8. Systems, structures, and components identified as essential targets are protected from the dynamic and environmental effects of postulated pipe ruptures.	Following as-built reconciliation, an inspection will be performed of the as-built high and moderate energy pipe rupture mitigation features for systems, structures, and components identified as essential targets.	An as-built Pipe Rupture Hazard Analysis Report exists and concludes that systems, structures, and components identified as essential targets can withstand the effects of postulated pipe rupture without loss of required safety function.
9. The reactor cavity sump has a minimum concrete thickness as shown in Table 3.3-5 between the bottom of the sump and the steel containment.	An inspection of the as-built containment building internal structures will be performed.	A report exists and concludes that the reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The shield building roof and PCS storage tank support and retain the PCS water sources. The PCS storage tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided on the tank boundary liner welds.	i) A test will be performed to measure the leakage from the PCS storage tank based on measuring the water flow out of the leak chase collection system.	i) A report exists and concludes that total water flow from the leak chase collection system does not exceed 10 gal/hr.
	ii) An inspection of the PCS storage tank exterior tank boundary and shield building tension ring will be performed before and after filling of the PCS storage tank to the overflow level. The vertical elevation of the shield building roof will be measured at a location at the outer radius of the roof (tension ring) and at a location on the same azimuth at the outer radius of the PCS storage tank before and after filling the PCS storage tank.	ii) A report exists and concludes that inspection and measurement of the PCS storage tank and the tension ring structure, before and after filling of the tank, shows structural behavior under normal loads to be acceptable.
	iii) An inspection of the PCS storage tank exterior tank boundary and shield building tension ring will be performed before and after filling of the PCS storage tank to the overflow level. The boundaries of the PCS storage tank and the shield building roof above the tension ring will be inspected visually for excessive concrete cracking.	iii) A report exists and concludes that there is no visible water leakage from the PCS storage tank through the concrete and that there is no visible excessive cracking in the boundaries of the PCS storage tank and the shield building roof above the tension ring.
11. Deleted		
12. The extended turbine generator axis intersects the shield building.	An inspection of the as-built turbine generator will be performed.	The extended axis of the turbine generator intersects the shield building.
13. Separation is provided between the structural elements of the turbine and annex buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in the safe shutdown earthquake without impact between structural elements of the buildings.	An inspection of the separation of the nuclear island from the annex and turbine building structures will be performed. The inspection will verify the specified horizontal clearance between structural elements of the adjacent buildings, consisting of the reinforced concrete walls and slabs, structural steel columns and floor beams.	The minimum horizontal clearance above floor elevation 100′-0″ between the structural elements of the annex building and the nuclear island is 3 inches. The minimum horizontal clearance above floor elevation 100′-0″ between the structural elements of the turbine building and the nuclear island is 3 inches.

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Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
14. The external walls, doors, ceiling, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.	Type test, analysis, or a combination of type test and analysis will be performed for the external walls, doors, ceilings, and floors in the main control room, the central alarm station, and the secondary alarm station.	A report exists and concludes that the external walls, doors, ceilings, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.
15. Deleted.		
16. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.	An inspection will be performed to ensure that the location of the secondary security power supply equipment for alarm annunciator equipment and non-portable communications equipment is within a vital area.	Secondary security power supply equipment for alarm annunciator equipment and non-portable communication equipment is located within a vital area.
17. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.	An inspection of the as-built vital areas, and central and secondary alarm stations are performed.	Vital areas are locked and alarmed with active intrusion detection systems and intrusion is detected and annunciated in both the central and secondary alarm stations.
18. Deleted.		

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Table 3.3-7
Nuclear Island Critical Structural Sections

Containment Internal Structures
South west wall of the refueling cavity
South wall of the west steam generator compartment
North east wall of the in-containment refueling water storage tank
In-containment refueling water storage tank steel wall
Column supporting the operating floor
Auxiliary and Shield Building
South wall of auxiliary building (column line 1), elevation 66'-6" to elevation 180'-0"
Interior wall of auxiliary building (column line 7.3), elevation 66'-6" to elevation 160'-6"
West wall of main control room in auxiliary building (column line L), elevation 117'-6" to elevation 153'-0"
North wall of MSIV east compartment (column line 11 between lines L and M), elevation 117'-6" to elevation 153'-0"
Roof slab at elevation 180'-0" adjacent to shield building cylinder
Floor slab on metal decking at elevation 135'-3"
2'-0" slab in auxiliary building (tagging room ceiling) at elevation 135'-3"
Finned floor in the main control room at elevation 135'-3"
Shield building roof, exterior wall of the PCS water storage tank
Shield building roof, interior wall of the PCS water storage tank
Shield building roof, tension ring and air inlets
Divider wall between the spent fuel pool and the fuel transfer canal
Shield building SC cylinder
Shield building SC to RC connection
Nuclear Island Basemat Below Auxiliary Building
Bay between reference column lines 9.1 and 11, and K and L
Bay between reference column lines 1 and 2 and K-2 and N

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Figure 3.3-1
Nuclear Island Section A-A

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Figure 3.3-2
Nuclear Island Section B-B

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Figure 3.3-3
Nuclear Island Plan View at Elevation 66'-6"

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Figure 3.3-4
Nuclear Island Plan View at Elevation 82'-6"

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Figure 3.3-5
Nuclear Island Plan View at Elevation 96'-6"

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Figure 3.3-6
Nuclear Island Plan View at Elevation 100'-0"

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Figure 3.3-7
Nuclear Island Plan View at Elevation 117'-6"

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Figure 3.3-8
Nuclear Island Plan View at Elevation 135'-3"

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Figure 3.3-9
Nuclear Island Plan View at Elevation 153'-3" and 160'-6"

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Figure 3.3-10
Nuclear Island Plan View at Shield Building Roof

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Figure 3.3-11A
Annex Building Plan View at Elevation 100'-0"

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Figure 3.3-11B
Turbine Building General Arrangement Plan at Elevation 100'-0"

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Figure 3.3-12
Annex Building Plan View at Elevation 117'-6"

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Figure 3.3-13
Annex Building Plan View at Elevation 135'-3"

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Figure 3.3-14
Nuclear Island Dimensions at
Elevation 66'-6"

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3.4 Initial Test Program

Design Description

The Licensee will implement an initial test program.

An initial test program is performed for each AP1000 plant and consists of a series of tests categorized as preoperational (prior to fuel load) and startup (during and after fuel load). Prerequisites for preoperational testing, such as completion of associated construction activities, component and/or preliminary tests, and inspections, should be completed before beginning preoperational tests (to the degree that outstanding items could not be expected to affect the validity of test results). All ITAAC will be completed prior to fuel load; therefore, no ITAAC are performed during the startup test phase of the initial test program.

Preoperational tests are performed for each system after associated prerequisites are met, but prior to initial fuel loading to demonstrate that equipment and systems perform in accordance with design criteria so that initial fuel loading, initial criticality, and subsequent power operation can be safely undertaken. Preoperational tests include, as appropriate, logic and interlock tests, control and instrumentation functional tests, and system performance tests including expansion, vibration, and dynamic effects tests.

Startup tests begin with the initial fuel loading and are performed to demonstrate the capability of individual systems, as well as the integrated plant, to meet performance requirements. Startup testing is conducted in four categories: tests related to initial fuel loading, tests performed after initial fuel loading but prior to initial criticality, tests related to initial criticality and those performed at low power (less than 5 percent), and tests performed at power levels greater than 5 percent (ascension to power tests). Startup tests include a controlled fuel load, reactor core and component performance tests, initial criticality, control and protection system operational tests, and plant system performance tests.

Preoperational and startup tests are performed using test procedures, which delineate the test methods to be used in the conduct of preoperational and startup testing and the applicable acceptance criteria against which performance is evaluated. Test procedures are developed and reviewed by qualified personnel. Copies of the test procedures for preoperational tests are available to NRC personnel prior to the scheduled performance of these individual tests. Copies of the test procedures for startup tests are available to NRC inspection personnel prior to the scheduled fuel loading date. Administrative procedures are used to control the conduct of the initial test program; the review, evaluation and approval of test results; and test record retention.

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3.5 Radiation Monitoring

Design Description

Radiation monitoring is provided for those plant areas where there is a significant potential for airborne contamination, for those process and effluent streams where contamination is possible, and in accessible areas to provide indication of unusual radiological events as identified in Tables 3.5-1, 3.5-2, 3.5-3, 3.5-4, and 3.5-5. The radiation monitoring component locations are as shown in Table 3.5-7.

1. The seismic Category I equipment identified in Table 3.5-1 can withstand seismic design basis loads without loss of safety function.

2. The Class 1E equipment identified in Table 3.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

3. Separation is provided between system Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

4. Safety-related displays identified in Table 3.5-1 can be retrieved in the main control room (MCR).

5. The process radiation monitors listed in Table 3.5-2 are provided.

6. The effluent radiation monitors listed in Table 3.5-3 are provided.

7. The airborne radiation monitors listed in Table 3.5-4 are provided.

8. The area radiation monitors listed in Table 3.5-5 are provided.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.5-6 specifies the inspections, tests, analyses, and associated acceptance criteria for radiation monitoring.

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Table 3.5-1

Equipment Name	Tag No.	Seismic Cat. I	Class 1E	Qual. for Harsh Envir.	Safety-Related Display
Containment High Range Monitor	PXS-RE160	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE161	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE162	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE163	Yes	Yes	Yes	Yes
MCR Radiation Monitoring Package A(1)	VBS-JS01A	Yes	Yes	No	Yes
MCR Radiation Monitoring Package B(1)	VBS-JS01B	Yes	Yes	No	Yes
Containment Atmosphere Monitor (Gaseous)	PSS-RE026	Yes	No	No	No
Containment Atmosphere Monitor (particulate, for RCS pressure boundary leakage detection)	PSS-RE027	Yes	No	No	No

Notes: (1) Each MCR Radiation Monitoring Package includes particulate, iodine and gaseous radiation monitors.

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Table 3.5-2
Process Radiation Monitors

Equipment List	Equipment No.
Steam Generator Blowdown	BDS-RE010
Steam Generator Blowdown	BDS-RE011
Component Cooling Water	CCS-RE001
Main Steam Line(1)	SGS-RY026
Main Steam Line(1)	SGS-RY027
Service Water Blowdown	SWS-RE008
Primary Sampling System Liquid Sample	PSS-RE050
Primary Sampling System Gaseous Sample	PSS-RE052
Containment Air Filtration Exhaust	VFS-RE001
Gaseous Radwaste Discharge	WGS-RE017

Note:

1. Each main steam line monitor includes a noble gas detector and primary-to-secondary side leak detector.

Table 3.5-3
Effluent Radiation Monitors

Equipment List	Equipment No.
Plant Vent (Normal Range Particulate)	VFS-RE101
Plant Vent (Normal Range Iodine)	VFS-RE102
Plant Vent (Normal Range Radiogas)	VFS-RE103
Plant Vent (Mid Range Radiogas)	VFS-RE104A
Plant Vent (High Range Radiogas)	VFS-RE104B
Turbine Island Vent(1)	TDS-RE001
Liquid Radwaste Discharge	WLS-RE229
Wastewater Discharge	WWS-RE021

Note:

1. The turbine island vent includes a low and a high range detector.

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Table 3.5-4
Airborne Radiation Monitors

Equipment List	Equipment No.
Fuel Handling Area Exhaust Radiation Monitor	VAS-RE001
Auxiliary Building Exhaust Radiation Monitor	VAS-RE002
Auxiliary Building Exhaust Radiation Monitor	VAS-RE003
Annex Building Exhaust Radiation Monitor	VAS-RE008
Health Physics and Hot Machine Shop Exhaust Radiation Monitor	VHS-RE001
Radwaste Building Exhaust Radiation Monitor	VRS-RE023

Table 3.5-5
Area Radiation Monitors

Equipment List	Equipment No.
Primary Sampling Room	RMS-RE008
Containment Area – Upper Personnel Hatch Operating Deck	RMS-RE009
Main Control Room	RMS-RE010
Chemistry Laboratory	RMS-RE011
Fuel Handling Area 1	RMS-RE012
Rail Car Bay/Filter Storage Area (Auxiliary Building Loading Bay)	RMS-RE013
Liquid and Gaseous Radwaste Area	RMS-RY014
Control Support Area	RMS-RE016
Radwaste Building Mobile Systems Facility	RMS-RE017
Hot Machine Shop	RMS-RE018
Annex Staging and Storage Area	RMS-RE019
Fuel Handling Area 2	RMS-RE020
Containment Area – Lower Personnel Hatch Maintenance Level	RMS-RE021

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Table 3.5-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The seismic Category I equipment identified in Table 3.5-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 3.5-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 3.5-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
2. The Class 1E equipment identified in Table 3.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that Class 1E equipment identified in Table 3.5-1 as being located in a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 3.5-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
3. Not used.		

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Table 3.5-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Safety-related displays identified in Table 3.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays in the MCR.	Safety-related displays identified in Table 3.5-1 can be retrieved in the MCR.
5. The process radiation monitors listed in Table 3.5-2 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-2 exists.
6. The effluent radiation monitors listed in Table 3.5-3 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-3 exists.
7. The airborne radiation monitors listed in Table 3.5-4 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-4 exists.
8. The area radiation monitors listed in Table 3.5-5 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-5 exists.

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Table 3.5-7

Component Name	Tag No.	Component Location
Containment High Range Radiation Monitor	PXS-RE160	Containment
Containment High Range Radiation Monitor	PXS-RE161	Containment
Containment High Range Radiation Monitor	PXS-RE162	Containment
Containment High Range Radiation Monitor	PXS-RE163	Containment
MCR Radiation Monitoring Package A	VBS-JS01A	Auxiliary Building
MCR Radiation Monitoring Package B	VBS-JS01B	Auxiliary Building
Containment Atmosphere Radiation Monitor (Gaseous)	PSS-RE026	Auxiliary Building
Containment Atmosphere Radiation Monitor (particulate, for RCS pressure boundary leakage detection)	PSS-RE027	Auxiliary Building
Steam Generator Blowdown Radiation Monitor	BDS-RE010	Turbine Building
Steam Generator Blowdown Radiation Monitor	BDS-RE011	Turbine Building
Component Cooling Water Radiation Monitor	CCS-RE001	Turbine Building
Main Steam Line Radiation Monitor	SGS-RY026	Auxiliary Building
Main Steam Line Radiation Monitor	SGS-RY027	Auxiliary Building
Service Water Blowdown Radiation Monitor	SWS-RE008	Turbine Building
Primary Sampling System Liquid Sample Radiation Monitor	PSS-RE050	Auxiliary Building
Primary Sampling System Gaseous Sample Radiation Monitor	PSS-RE052	Auxiliary Building
Containment Air Filtration Exhaust Radiation Monitor	VFS-RE001	Annex Building
Gaseous Radwaste Discharge Radiation Monitor	WGS-RE017	Auxiliary Building
Plant Vent (Normal Range Particulate) Radiation Monitor	VFS-RE101	Auxiliary Building
Plant Vent (Normal Range Iodine) Radiation Monitor	VFS-RE102	Auxiliary Building
Plant Vent (Normal Range Radiogas) Radiation Monitor	VFS-RE103	Auxiliary Building
Plant Vent (Mid Range Radiogas) Radiation Monitor	VFS-RE104A	Auxiliary Building
Plant Vent (High Range Radiogas) Radiation Monitor	VFS-RE104B	Auxiliary Building
Turbine Island Vent Radiation Monitor	TDS-RE001	Turbine Building

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Table 3.5-7 (cont.)

Component Name	Tag No.	Component Location
Liquid Radwaste Discharge Monitor	WLS-RE229	Radwaste Building
Wastewater Discharge Radiation Monitor	WWS-RE021	Turbine Building
Fuel Handling Area Exhaust Radiation Monitor	VAS-RE001	Auxiliary Building
Auxiliary Building Exhaust Radiation Monitor	VAS-RE002	Auxiliary Building
Auxiliary Building Exhaust Radiation Monitor	VAS-RE003	Auxiliary Building
Annex Building Exhaust Radiation Monitor	VAS-RE008	Annex Building
Health Physics and Hot Machine Shop Exhaust Radiation Monitor	VHS-RE001	Annex Building
Radwaste Building Exhaust Radiation Monitor	VRS-RE023	Radwaste Building
Primary Sampling Room	RMS-RE008	Auxiliary Building
Containment Area – Personnel Hatch – Operating Deck	RMS-RE009	Auxiliary Building
Main Control Room	RMS-RE010	Auxiliary Building
Chemistry Laboratory	RMS-RE011	Auxiliary Building
Fuel Handling Area 1	RMS-RE012	Auxiliary Building
Rail Car Bay/Filter Storage Area (Auxiliary Building Loading Bay)	RMS-RE013	Auxiliary Building
Liquid and Gaseous Radwaste Area	RMS-RY014	Radwaste Building
Control Support Area	RMS-RE016	Annex Building
Radwaste Building Mobile Systems Facility	RMS-RE017	Radwaste Building
Hot Machine Shop	RMS-RE018	Annex Building
Annex Staging and Storage Area	RMS-RE019	Annex Building
Fuel Handling Area 2	RMS-RE020	Auxiliary Building
Containment Area – Personnel Hatch – Maintenance Level	RMS-RE021	Auxiliary Building

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3.6 Reactor Coolant Pressure Boundary Leak Detection

Design Description

The reactor coolant pressure boundary leakage detection monitoring provides a means of detecting and quantifying the reactor coolant leakage. To detect unidentified leakage inside containment, the following diverse methods are provided to quantify and assist in locating the leakage:

- Containment Sump Level
- Reactor Coolant System Inventory Balance
- Containment Atmosphere Radiation

Leakage detection monitoring is accomplished using instrumentation and other components of several systems.

1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.

Inspection, Tests, Analyses, and Acceptance Criteria

Table 3.6-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the leak detection equipment.

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Table 3.6-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Not used.		

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3.7 Design Reliability Assurance Program

The Design Reliability Assurance Program (D-RAP) is a program that will be performed during the detailed design and equipment specification phase prior to initial fuel load. The D-RAP evaluates and sets priorities for the structures, systems, and components (SSCs) in the design, based on their degree of risk significance. The risk-significant components are listed in Table 3.7-1.

The objective of the D-RAP program is to provide reasonable assurance that risk-significant SSCs (Table 3.7-1) are designed such that: (1) assumptions from the risk analysis are utilized, (2) SSCs (Table 3.7-1) when challenged, function in accordance with the assumed reliability, (3) SSCs (Table 3.7-1) whose failure results in a reactor trip, function in accordance with the assumed reliability, and (4) maintenance actions to achieve the assumed reliability are identified.

1. The D-RAP ensures that the design of SSCs within the scope of the reliability assurance program (Table 3.7-1) is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.7-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the D-RAP.

Table 3.7-1
Risk-Significant Components

Equipment Name	Tag No.
Component Cooling Water System (CCS)	
Component Cooling Water Pumps	CCS-MP-01A/B
Containment System (CNS)	
Containment Vessel	CNS-MV-01
Hydrogen Igniters	VLS-EH-1 through -66
Chemical and Volume Control System (CVS)	
Makeup Pumps	CVS-MP-01A/B
Makeup Pump Suction and Discharge Check Valves	CVS-PL-V113 CVS-PL-V160A/B
Letdown Discharge Isolation Valves	CVS-PL-V045 CVS-PL-V047
Diverse Actuation System (DAS)	
DAS Processor Cabinets and Control Panel (used to provide automatic and manual actuation)	DAS-JD-001 DAS-JD-002 DAS-JD-003 OCS-JC-20

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Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Auxiliary Building UPS Distribution Panels (provide power to DAS)	EDS2-EA-12, EDS3-EA-14A
Control Cabinets for the Rod Drive MG Sets (generator field control relays)	PLS-JD-RDM001 PLS-JD-RDM002
Containment Isolation Valves Controlled by DAS	CVS-PL-V045, -V047 VFS-PL-V003, -V004, -V009, -V010 WLS-PL-V055, -V057
Main ac Power System (ECS)	
Reactor Coolant Pump Switchgear	ECS-ES-31, -32, -41, -42, -51, -52, -61, -62
Ancillary Diesel Generators	ECS-MS-01, -02
6900 Vac Buses	ECS-ES-1, -2
Main and Startup Feedwater System (FWS)	
Startup Feedwater Pumps	FWS-MP-03A/B
General I&C	
IRWST Wide Range Level Sensors	PXS-046, -047, -048
IRWST Lower Narrow Range Level Sensors	PXS-066, -067, -068, -069
RCS Hot Leg Level Sensors	RCS-160A/B
Pressurizer Pressure Sensors	RCS-191A/B/C/D
Pressurizer Level Sensors	RCS-195A/B/C/D
Steam Generator Narrow-Range Level Sensors	SGS-001, -002, -003, -004, -005, -006, -007, -008
Steam Generator Wide-Range Level Sensors	SGS-011, -012, -013, -014, -015, -016, -017, -018
Main Steam Line Pressure Sensors	SGS-030, -031, -032, -033, -034, -035, -036, -037
Main Feedwater Low-Range Flow Sensors	FWS-050A/C/E, -051A/C/E
Startup Feedwater Flow Sensors	SGS-055A/B/C/D/E, -056A/B/C/D/E
CMT Level Sensors	PXS-011A/B/C/D, -012A/B/C/D, -013A/B/C/D, -014A/B/C/D

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Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Class 1E dc Power and Uninterruptible Power System (IDS)	
250 Vdc 24-Hour Batteries	IDSA-DB-1A/B, IDSB-DB-1A/B, IDSC-DB-1A/B, IDSD-DB-1A/B
250 Vdc 24-Hour Buses	IDSA-DS-1, IDSB-DS-1 IDSC-DS-1, IDSD-DS-1
250 Vdc 24-Hour Battery Chargers	IDSA-DC-1, IDSB-DC-1, IDSC-DC-1, IDSD-DC-1
250 Vdc and 120 Vac Distribution Panels	IDSA-DD-1, IDSA-EA-1/-2, IDSB-DD-1, IDSB-EA-1/-2/-3, IDSC-DD-1, IDSC-EA-1/-2/-3, IDSD-DD-1, IDSD-EA-1/-2
Fused Transfer Switch Boxes	IDSA-DF-1, IDSB-DF-1/-2, IDSC-DF-1/-2, IDSD-DF-1
250 Vdc Motor Control Centers	IDSA-DK-1, IDSB-DK-1, IDSC-DK-1, IDSD-DK-1
250 Vdc 24-Hour Inverters	IDSA-DU-1, IDSB-DU-1, IDSC-DU-1, IDSD-DU-1
Passive Containment Cooling System (PCS)	
Recirculation Pumps	PCS-MP-01A/B
PCCWST Drain Isolation Valves	PCS-PL-V001A/B/C
Plant Control System (PLS)	
PLS Actuation Software (used to provide control functions)	Refer to Table 3.7-2
PLS Actuation Hardware (used to provide control functions)	Refer to Table 3.7-2
Protection and Monitoring System (PMS)	
PMS Actuation Software (used to provide automatic control functions)	Refer to Tables 2.5.2-2 and 2.5.2-3
PMS Actuation Hardware (used to provide automatic control functions)	Refer to Tables 2.5.2-2 and 2.5.2-3
MCR 1E Displays and System Level Controls	OCS-JC-10, -11
Reactor Trip Switchgear	PMS-JD-RTS A01/02, B01/02, C01/02, D01/02

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Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Passive Core Cooling System (PXS)	
IRWST Hood Vent Covers	PXS-MY-Y21, -Y22, -Y23, -Y24, -Y25, -Y26, -Y27, -Y28, -Y29, -Y30, -Y31, -Y32, -Y33, -Y41, -Y47, -Y48
IRWST Steam Generator Wall Vent Covers	PXS-MY-Y61, -Y62, -Y63, -Y64
IRWST Overflow Weir Covers	PXS-MY-Y71, -Y72, -Y73, -Y74, -Y75, -Y76
IRWST Screens	PXS-MY-Y01A/B/C
Containment Recirculation Screens	PXS-MY-Y02A/B
CMT Discharge Isolation Valves	PXS-PL-V014A/B, -V015A/B
CMT Discharge Check Valves	PXS-PL-V016A/B, -V017A/B
Accumulator Discharge Check Valves	PXS-PL-V028A/B, -V029A/B
PRHR HX Control Valves	PXS-PL-V108A/B
Containment Recirculation Squib Valves	PXS-PL-V118A/B, -V120A/B
IRWST Injection Check Valves	PXS-PL-V122A/B, -V124A/B
IRWST Injection Squib Valves	PXS-PL-V123A/B, -V125A/B
IRWST Gutter Bypass Isolation Valves	PXS-PL-V130A/B
Reactor Coolant System (RCS)	
ADS Stage 1/2/3 Valves (MOVs)	RCS-PL-V001A/B, -V011A/B RCS-PL-V002A/B, -V012A/B RCS-PL-V003A/B, -V013A/B
ADS Stage 4 Valves (Squibs)	RCS-PL-V004A/B/C/D
Pressurizer Safety Valves	RCS-PL-V005A/B
Reactor Vessel Insulation Water Inlet and Steam Vent Devices	RXS-MN-01
Reactor Cavity Doorway Damper	–
Fuel Assemblies	157 assemblies with tag numbers beginning with RXS-FA

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Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Normal Residual Heat Removal System (RNS)	
Residual Heat Removal Pumps	RNS-MP-01A/B
RNS Motor-Operated Valves	RNS-PL-V011, -V022, -V023, -V055
RNS Stop Check Valves RNS Check Valves	RNS-PL-V015A/B RNS-PL-V017A/B
RNS Check Valves	RNS-PL-V007A/B, -V013, -V056
Spent Fuel Cooling System (SFS)	
Spent Fuel Cooling Pumps	SFS-MP-01A/B
Steam Generator System (SGS)	
Main Steam Safety Valves	SGS-PL-V030A/B, -V031A/B, -V032A/B, -V033A/B, -V034A/B, -V035A/B
Main Steam Line Isolation Valves	SGS-PL-V040A/B
Main Feedwater Isolation Valves	SGS-PL-V057A/B
Service Water System (SWS)	
Service Water Cooling Tower Fans	SWS-MA-01A/B
Service Water Pumps	SWS-MP-01A/B
Nuclear Island Nonradioactive Ventilation System (VBS)	
MCR Ancillary Fans	VBS-MA-10A/B
I&C Room B/C Ancillary Fans	VBS-MA-11, -12
Containment Air Filtration System (VFS)	
Containment Purge Isolation Valves	VFS-PL-V003 VFS-PL-V004 VFS-PL-V009 VFS-PL-V010
Chilled Water System (VWS)	
Air Cooled Chiller Pumps	VWS-MP-02, -03
Air Cooled Chillers	VWS-MS-02, -03
Liquid Radwaste System (WLS)	
Sump Containment Isolation Valves	WLS-PL-V055 WLS-PL-V057

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Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Onsite Standby Power System (ZOS)	
Engine Room Exhaust Fans	VZS-MY-V01A/B, -V02A/B
Onsite Diesel Generators	ZOS-MS-05A/B

Note: Dash (-) indicates not applicable.

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Table 3.7-2
PLS D-RAP Control Functions

CVS Reactor Makeup
RNS Reactor Injection from cask loading pit
Startup Feedwater from CST
Spent Fuel Cooling
Component Cooling of RNS and SFS Heat Exchangers
Service Water Cooling of CCS Heat Exchangers
Onsite Diesel Generators
Hydrogen Igniters

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Table 3.7-3
Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The D-RAP ensures that the design of SSCs within the scope of the reliability assurance program (Table 3.7-1) is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability).	An analysis will confirm that the design of RAP SSCs identified in Table 3.7-1 has been completed in accordance with applicable D-RAP activities.	An analysis report documents that safety-related SSCs identified in Table 3.7-1 have been designed in accordance with a 10 CFR 50 Appendix B quality program. An analysis report documents that non-safety-related SSCs identified in Table 3.7-1 have been designed in accordance with a program that satisfies quality assurance requirements for SSCs important to investment protection.

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4.0 Interface Requirements

The 10 CFR 52.47 (a)(25) requires identification of the interface requirements to be met by those portions of the plant for which the application does not seek certification. The 10 CFR 52.47 (a)(26) requires justification that these interfaces be verifiable through inspection, testing (either in the plant or elsewhere), or analysis. An applicant for a combined license (COL) that references the Certified Design must provide design features or characteristics that comply with the interface requirements for the plant design and inspections, tests, analyses, and acceptance criteria (ITAAC) for the site-specific portion of the facility design, in accordance with 10 CFR 52.79 (c).

No Tier 1 interfaces were identified for the AP1000 standard plant design.

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5.0 Site Parameters

Table 5.0-1 identifies the key site parameters that are specified for the design of safety-related aspects of structures, systems, and components for the AP1000. An actual site is acceptable if its site characteristics fall within the AP1000 plant site design parameters in Table 5.0-1.

Structures, systems, and components for the AP1000 are evaluated for generic envelope response spectra with high frequency seismic input. The spectra shown in Figure 5.0-3 and Figure 5.0-4 provide hard rock high frequency (HRHF) envelope response spectra at the foundation level for both the horizontal and vertical directions for 5% damping. An actual site is acceptable if its site-specific GMRS falls within the AP1000 HRHF parameters in Figures 5.0-3 and 5.0-4. No additional design or analyses are required for the structures, systems, and components for sites that fall within the AP1000 HRHF parameters.

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Table 5.0-1
Site Parameters

Maximum Ground Water Level	Plant elevation 98 ft
Maximum Flood Level	Plant elevation 100 ft (design grade elevation)
Precipitation	
Rain	20.7 in/hr [1-hr 1-mi^2 PMP]
Snow/Ice	Ground snow load of 75 lb/ft^2 with exposure factor of 1.0 and importance factor of 1.2
Air Temperature	Limits based on historical data excluding peaks of less than 2 hours duration Maximum temperature of 115° dry bulb/86.1°F coincident wet bulb Maximum wet bulb 86.1°F (noncoincident) Minimum temperature of -40°F
Tornado	
Wind Speed	Maximum wind speed of 300 mph
Maximum Pressure Differential	Maximum pressure differential of 2.0 lb/in^2
Tornado Missile Spectra	4000-lb automobile at 105 mph horizontal, 74 mph vertical 275-lb, 8-in. shell at 105 mph horizontal, 74 mph vertical 1-in.-diameter steel ball at 105 mph in the most damaging direction
Soil	
Average Allowable Static Soil Bearing Capacity	The allowable bearing capacity, including a factor of safety appropriate for the design load combination, shall be greater than or equal to the average bearing demand of 8,900 lb/ft^2 over the footprint of the nuclear island at its excavation depth.
Minimum Soil Angle of Internal Friction	The minimum soil angle of internal friction is greater than or equal to 35 degrees below the footprint of the nuclear island at its excavation depth. If the minimum soil angle of internal friction is below 35 degrees, a site-specific analysis shall be performed using the site-specific soil properties to demonstrate stability.
Dynamic Bearing Capacity for Normal Plus Safe Shutdown Earthquake (SSE)	The allowable bearing capacity, including a factor of safety appropriate for the design load combination, shall be greater than or equal to the maximum bearing demand of 35,000 lb/ft^2 at the edge of the nuclear island at its excavation depth, or site-specific analyses demonstrate factor of safety appropriate for normal plus safe shutdown earthquake loads.

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Table 5.0-1 (cont.)
Site Parameters

Soil (cont.)		
Limits of Acceptable Settlement Without Additional Evaluation(1)	Differential across nuclear island foundation mat	1/2 inch in 50 feet
	Total for nuclear island foundation mat	6 inches
	Differential between nuclear island and turbine building(2)	3 inches
	Differential between nuclear island and other buildings(2)	3 inches
Lateral Variability	Soils supporting the nuclear island should not have extreme variations in subgrade stiffness. This may be demonstrated by one of the following: 1. Soils supporting the nuclear island are uniform in accordance with Regulatory Guide 1.132 if the geologic and stratigraphic features at depths less than 120 feet below grade can be correlated from one boring or sounding location to the next with relatively smooth variations in thicknesses or properties of the geologic units, or 2. Site-specific assessment of subsurface conditions demonstrates that the bearing pressures below the footprint of the nuclear island do not exceed 120% of those from the generic analyses of the nuclear island at a uniform site, or 3. Site-specific analysis of the nuclear island basemat demonstrates that the site specific demand is within the capacity of the basemat.	

Notes:

1. Additional evaluation may include evaluation of the impact of the elevated estimated settlement values on the critical components of the AP1000, determining a construction sequence to control the predicted settlement behavior, or developing an active settlement monitoring system throughout the entire construction sequence as well as a long-term (plant operation) plan.

2. Differential settlement is measured at the center of the nuclear island and the center of adjacent structures.

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Table 5.0-1 (cont.)
Site Parameters

Soil (cont.)	
Lateral Variability (cont.)	As an example of sites that are considered uniform, the variation of shear wave velocity in the material below the foundation to a depth of 120 feet below finished grade within the nuclear island footprint and 40 feet beyond the boundaries of the nuclear island footprint meets the criteria in the case outlined below. Case 1: For a layer with a low strain shear wave velocity greater than or equal to 2500 feet per second, the layer should have approximately uniform thickness, should have a dip not greater than 20 degrees, and should have less than 20 percent variation in the shear wave velocity from the average velocity in any layer.
Shear Wave Velocity	Greater than or equal to 1000 ft/sec based on minimum low-strain soil properties over the footprint of the nuclear island at its excavation depth
Liquefaction Potential	No liquefaction considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures.
Seismic	
CSDRS	CSDRS free field peak ground acceleration of 0.30g with modified Regulatory Guide 1.60 response spectra (See Figures 5.0-1 and 5.0-2). The SSE is now referred to as CSDRS. Seismic input is defined at finished grade except for sites where the nuclear island is founded on hard rock. If the site-specific spectra exceed the response spectra in Figures 5.0-1 and 5.0-2 at any frequency, or if soil conditions are outside the range evaluated for AP1000 design certification, a site-specific evaluation can be performed. This evaluation will consist of a site-specific dynamic analysis and generation of in-structure response spectra at key locations to be compared with the floor response spectra of the certified design at 5-percent damping. The site is acceptable if the floor response spectra from the site-specific evaluation do not exceed the AP1000 spectra for each of the locations or the exceedances are justified.

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Table 5.0-1 (cont.)
Site Parameters

Seismic (cont.) CSDRS (cont.)	The hard rock high frequency (HRHF) envelope response spectra are shown in Figure 5.0-3 and Figure 5.0-4 defined at the foundation level for 5% damping. The HRHF envelope response spectra provide an alternative set of spectra for evaluation of site-specific GMRS. A site is acceptable if its site-specific GMRS falls within the AP1000 HRHF envelope response spectra. Evaluation of a site for application of the HRHF envelope response spectra includes consideration of the limitation on shear wave velocity identified for use of the HRHF envelope response spectra. This limitation is defined by a shear wave velocity at the bottom of the basemat equal to or higher than 7,500 fps, while maintaining a shear wave velocity equal to or above 8,000 fps at the lower depths.
Fault Displacement Potential	No potential fault displacement considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures.
Atmospheric Dispersion Factors (χ/Q) Site Boundary (0-2 hr) Site Boundary (annual average) Low Population Zone Boundary 0 - 8 hr 8 - 24 hr 24 - 96 hr 96 - 720 hr	 $\leq 5.1 \times 10^{-4}$ sec/m^3 $\leq 2.0 \times 10^{-5}$ sec/m^3 $\leq 2.2 \times 10^{-4}$ sec/m^3 $\leq 1.6 \times 10^{-4}$ sec/m^3 $\leq 1.0 \times 10^{-4}$ sec/m^3 $\leq 8.0 \times 10^{-5}$ sec/m^3

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Table 5.0-1 (cont.)
Site Parameters

Control Room Atmospheric Dispersion Factors (χ /Q) for Accident Dose Analysis						
χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[3]						
	Plant Vent or PCS Air Diffuser[5]	**Ground Level Containment Release Points[6]**	**PORV and Safety Valve Releases[7]**	**Steam Line Break Releases**	**Fuel Handling Area[8]**	**Condenser Air Removal Stack[9]**
0 - 2 hours	2.53E-03	4.00E-03	1.92E-02	2.13E-02	6.0E-3	6.0E-3
2 - 8 hours	1.98E-03	2.28E-03	1.60E-02	1.76E-02	4.0E-3	4.0E-3
8 - 24 hours	7.96E-04	1.03E-03	6.90E-03	7.50E-03	2.0E-3	2.0E-3
1 - 4 days	6.40E-04	9.03E-04	4.96E-03	5.43E-03	1.5E-3	1.5E-3
4 - 30 days	4.78E-04	7.13E-04	4.16E-03	4.55E-03	1.0E-3	1.0E-3
χ/Q (s/m^3) at Annex Building Door for the Identified Release Points[4]						
0 - 2 hours	1.0E-3	1.0E-3	4.0E-3	4.0E-3	6.0E-3	2.0E-2
2 - 8 hours	7.5E-4	7.5E-4	3.2E-3	3.2E-3	4.0E-3	1.8E-2
8 - 24 hours	3.5E-4	3.5E-4	1.2E-3	1.2E-3	2.0E-3	7.0E-3
1 - 4 days	2.8E-4	2.8E-4	1.0E-3	1.0E-3	1.5E-3	5.0E-3
4 - 30 days	2.5E-4	2.5E-4	8.0E-4	8.0E-4	1.0E-3	4.5E-3

Notes:

3. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the nonsafety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.

4. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.

5. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.

6. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.

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7. The listed values bound the dispersion factors for releases from the steam line safety and power-operated relief valves. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and the secondary side release from a rod ejection accident.

8. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.

9. This release point is included for information only as a potential activity release point. None of the design basis accident radiological consequences analyses model release from this point.

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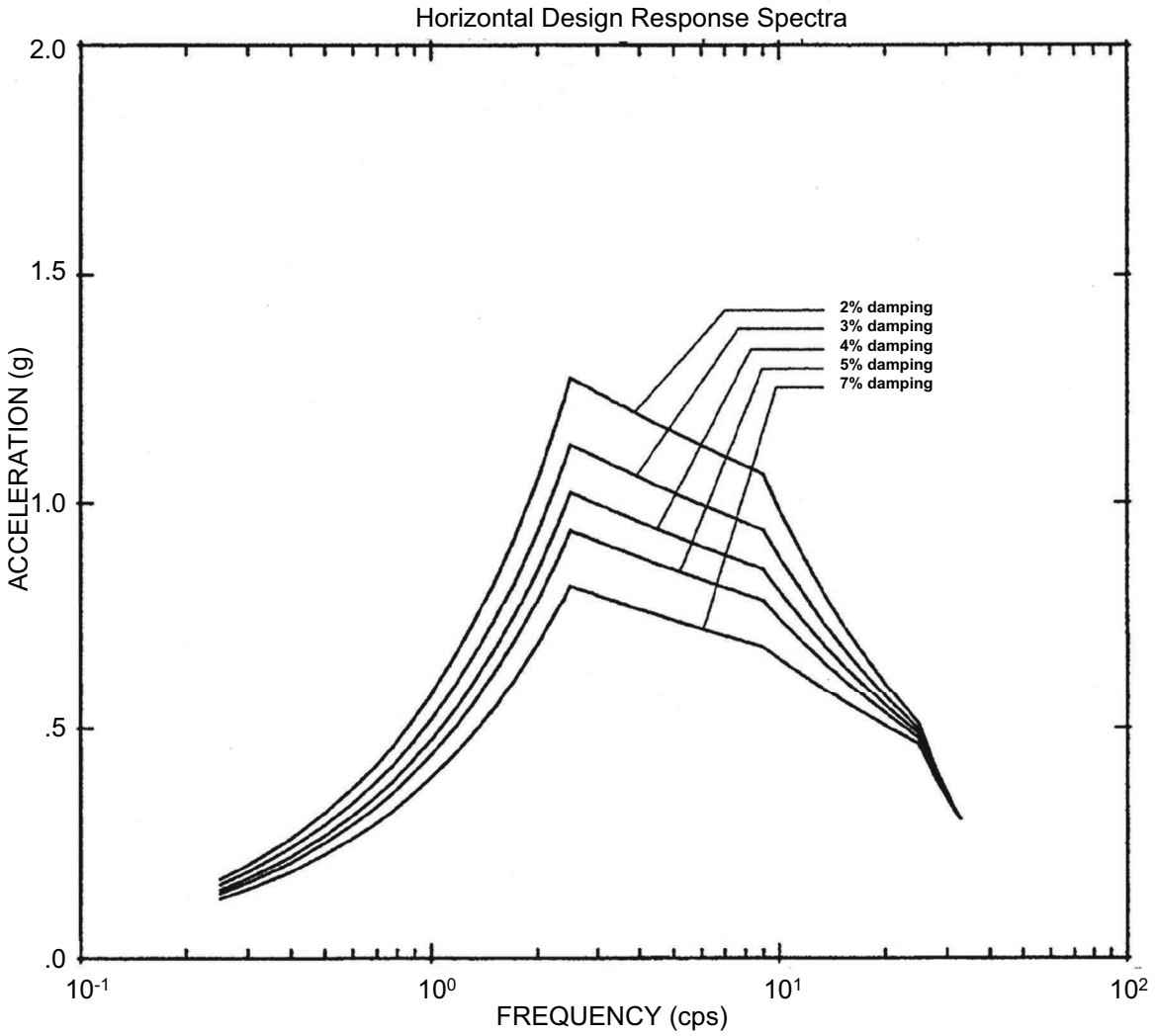

Figure 5.0-1
Horizontal Design Response Spectra
Safe Shutdown Earthquake

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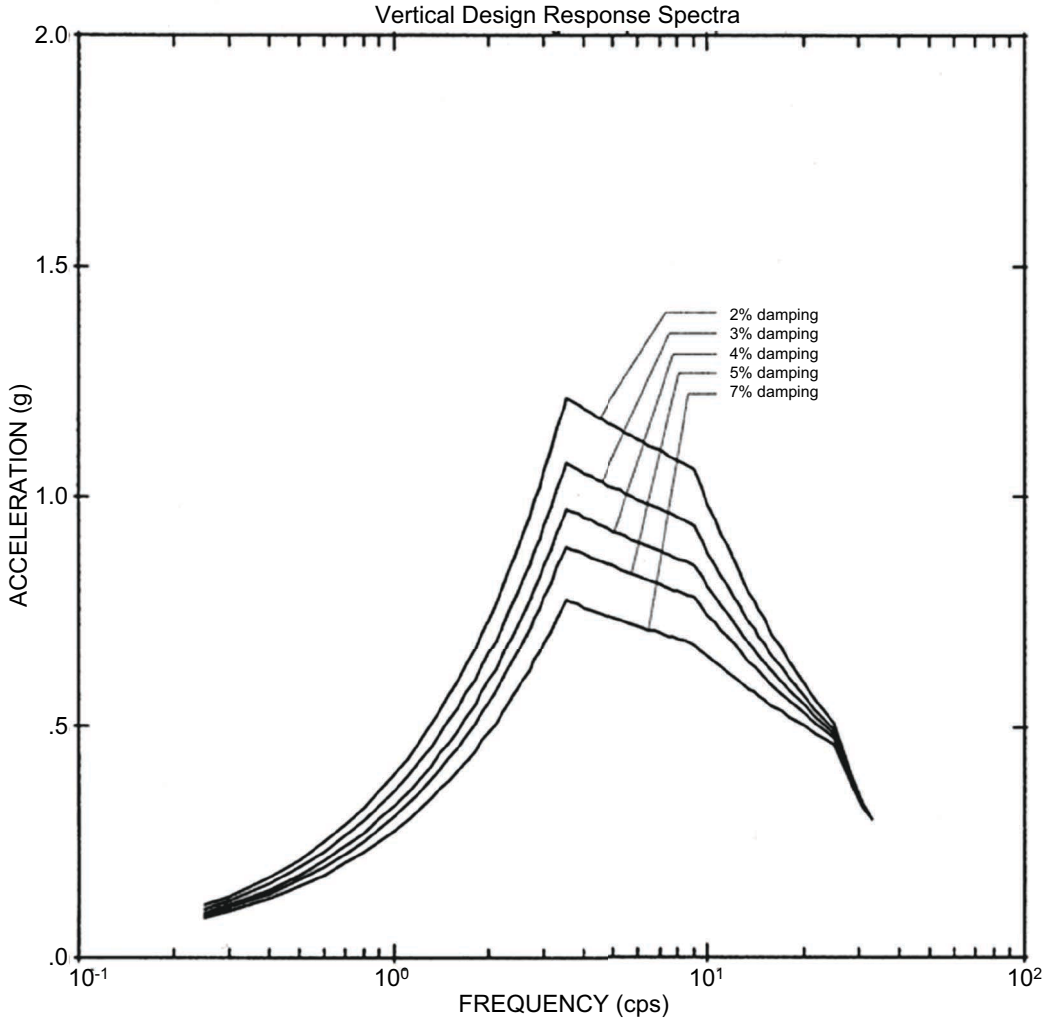

Figure 5.0-2
Vertical Design Response Spectra
Safe Shutdown Earthquake

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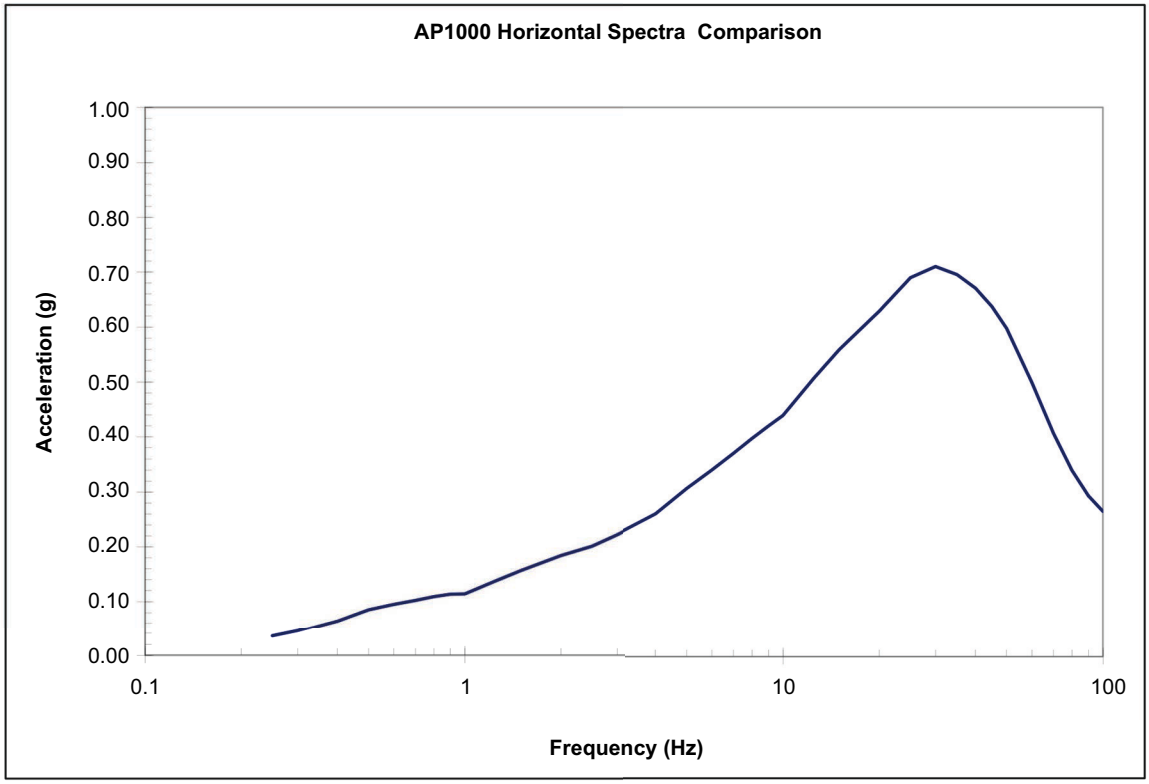

Figure 5.0-3
Horizontal HRHF Envelope Response Spectra
Safe Shutdown Earthquake

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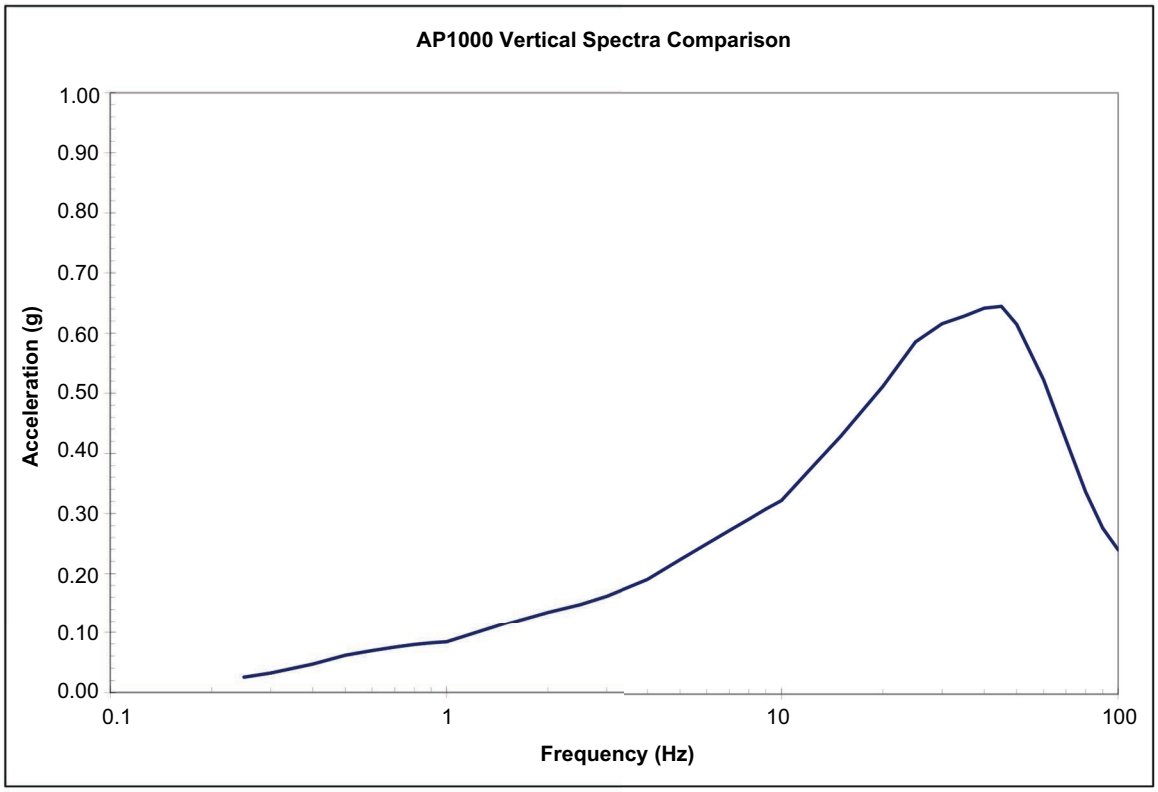

Figure 5.0-4
Vertical HRHF Envelope Response Spectra
Safe Shutdown Earthquake

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Chapter 1 Introduction and General Description of the Plant

1.1 Introduction

The Design Control Document (DCD) for the Westinghouse AP1000 simplified passive advanced light water reactor plant is incorporated by reference into the Design Certification Rule for the AP1000 design (Section II.A) of Appendix D to 10 CFR Part 52.

This Final Safety Analysis Report (FSAR) incorporates the Design Control Document (DCD) (as identified in Table 1.6-201) for a simplified passive advanced light water reactor plant provided by Westinghouse Electric Company, the entity originally sponsoring and obtaining the AP1000 design certification documented in 10 CFR Part 52, Appendix D. Throughout this FSAR, the “referenced DCD” is the AP1000 DCD submitted by Westinghouse as Revision 20 including any supplemental material as identified in Table 1.6-201. Unless otherwise specified, reference to the DCD refers to Tier 2 information, including references to the sensitive unclassified non-safeguards information (including proprietary information) and safeguards information, contained in the AP1000 DCD. Such DCD information was included in the combined license application in the same manner as it is included in the AP1000 DCD and the appropriate material has been included in the FSAR, i.e., references in the DCD are included as references in the FSAR, and material incorporated by reference into the DCD is incorporated by reference into the FSAR.

A limited set of DCD Tier 2 information was further designated as Tier 2* matter information. As provided in 10 CFR Part 52, Appendix D, much of the Tier 2* matter information reverts to Tier 2 information for the unit once the unit first achieves full power operation.

Appendix D to 10 CFR Part 52 is hereby incorporated by reference into the COL application.

1.1.1 Plant Location

The AP1000 is a standardized plant that is to be placed on a site with parameters described in Chapter 2, "Site Characteristics." The site parameters relate to the seismology, hydrology, meteorology, geology, heat sink, and other site-related aspects.

1.1.2 Containment Type

The containment building is a freestanding, cylindrical, steel containment vessel with elliptical upper and lower heads. It is surrounded by a seismic Category I shield building constructed of reinforced concrete and steel concrete composite modules. The containment vessel is an integral part of the passive containment cooling system. The vessel provides the safety-related interface with the ultimate heat sink, which is the surrounding atmosphere. Westinghouse is responsible, along with their contractor team members, for the design of the containment.

1.1.3 Reactor Type

The nuclear steam supply system (NSSS) for the AP1000 is a Westinghouse-designed pressurized water reactor.

1.1.4 Power Output

The plant's net producible electrical power to the grid is at least 1000 MWe, with a core power rating of 3400 MWt. In some safety evaluations a power level higher than the rated power level is employed.

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1.1.5 Schedule

The scheduled completion date and estimated commercial operation date of nuclear power plants referencing the AP1000 design certification are provided as discussed in Subsection 1.1.7.

1.1.6 Format and Content

1.1.6.1 Regulatory Guide 1.70

To the extent practical, the AP1000 DCD has used as a guide the format and content recommendations of Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," November 1978.

The DCD generally uses the same chapter, section, subsection, and paragraph headings used in the standard format. Where appropriate, the DCD is subdivided beyond the extent of the standard format to provide additional information specifically required for that area. Similarly, some of the passive features of the AP1000 require modification of the standard format and content either in terms of placement or type of material presented.

This FSAR generally follows the AP1000 DCD organization and numbering. Some organization and numbering differences are adopted where necessary to include additional material, such as additional content identified in Regulatory Guide 1.206.

1.1.6.2 Standard Review Plan

The technical guidance provided in NUREG-0800, is followed in the preparation of the AP1000 DCD. Standard Review Plan conformance is also determined in accordance with 10 CFR 50.34 to identify the deviations of the AP1000 DCD from the Standard Review Plan. See Subsection 1.9.2 for additional details on Standard Review Plan conformance.

1.1.6.3 Text, Tables, and Figures

AP1000 DCD tables of data are identified by the section or subsection number followed by a sequential number (for example, Table 3.2-3 is the third table of Section 3.2). Tables are located at the end of the section immediately following the text. Drawings, pictures, sketches, curves, graphs, plots, and engineering diagrams are identified as figures and are numbered sequentially by section or subsection similar to tables, and follow at the end of the applicable section or subsection.

FSAR tables, figures, and references are numbered in the same manner as the DCD, but the first new FSAR item is numbered as 201, the second 202, the third 203, and consecutively thereafter.

New appendices are included in the FSAR with double letter designations following the pertinent chapter (e.g., 12AA).

When it provides greater contextual clarity, an existing DCD table or figure is revised by adding new information to the table or figure and replacing the DCD table or figure with a new one in the FSAR. In this instance, the revised table or figure clearly identifies the information being added, and retains the same numbering as in the DCD, but the table or figure number is revised to end with the designation “R” to indicate that the table or figure has been revised and replaced. For example, revised “Table 4.2-1” would become “Table 4.2-1R.”

1.1.6.4 Numbering of Pages

Text pages are numbered sequentially within each section or subsection.

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1.1.6.5 Proprietary Information

The AP1000 DCD contains no proprietary information, with the exception of limited information located in Appendix 7B.

Some portions of the DCD may be considered as proprietary, personal, or sensitive and withheld from public disclosure pursuant to 10 CFR 2.390 and Regulatory Issue Summary (RIS) 2005-026. Such material is clearly marked and the withheld material is separately provided for NRC review.

1.1.6.6 Acronyms

Table 1.1-1 provides a list of acronyms used in the AP1000 DCD. Acronyms for systems are defined in the section in which they are used. Other acronyms may be defined in the section in which they are used. Table 1.7-2 provides a list of AP1000 system designators.

1.1.7 Combined License Information

Combined License applicants referencing the AP1000 certified design will provide the construction and startup schedule information.

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Table 1.1-1 (Sheet 1 of 12)
AP1000 Acronyms

ac	Alternating Current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AD	Aerodynamic Diameter
ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
AFD	Axial Flux Difference
AFT	As-Found Tolerance
AFW	Auxiliary Feedwater
AICC	Adiabatic, Isochoric, Complete Combustion
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	As-Low-As-Reasonably Achievable
ALHR	Average Linear Heart Rate
ALI	Annual Limits on Intake
ALT	As-Left Tolerance
ALWR	Advanced Light Water Reactor
AMC	Annual Maintenance Cost
AMCA	Air Movement and Control Association
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	Axial Offset
AOC	Annual Operating Cost
AOO	Anticipated Operational Occurrence
AOV	Air-Operated Valve
API	American Petroleum Institute
ARI	Air Conditioning and Refrigeration Institute
ARM	Area Radiation Monitor
ARO	All Rods Out
ASCE	American Society of Civil Engineers
ASD	Allowable Stress Design
ASHRAE	American Society of Heating, Refrigeration and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASS	Auxiliary Steam System
ASTM	American Society for Testing and Materials
ASTRUM	Automated Statistical Treatment of Uncertainty Method
ATWS	Anticipated Transient Without Scram
AVT	All Volatile Treatment
AWS	American Welding Society
AWWA	American Water Works Association
BDE	Boron Dilution Event

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Table 1.1-1 (Sheet 2 of 12)
AP1000 Acronyms

BDPS	Boron Dilution Protection System
BDS	Blowdown System
BEACON	Best Estimate Analyzer for Core Operations - Nuclear
BOC	Beginning of Cycle
BOL	Beginning of Life
BOP	Balance of Plant
BP	Blowout Panel
BP	Containment Bypass
B&PVC	Boiler and Pressure Vessel Code
BR	Breathing Rate
BRL	Ballistic Research Laboratory
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CAM	Continuous Air Monitor
CAOC	Constant Axial Offset Control
CASS	Cast Auestnitic Stainless Steel
CAV	Cumulative Absolute Velocity
CBI	Chicago Bridge & Iron Services
CCI	Core-Concrete Interaction
CCW	Component Cooling Water
CDE	Committed Dose Equivalent
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CEDE	Committed Effective Dose Equivalent
CET	Containment Event Tree
CFE	Containment Failure Early
CFI	Containment Failure Intermediate
CFL	Containment Failure Late
CFR	Code of Federal Regulations
CFS	Chemical Feed System
CHF	Critical Heat Flux
CIP	Clean In-Place
CIPS	Crud-Induced Power Shift
CIS	Containment Internal Structure
CLP	Cask Loading Pit
CMAA	Crane Manufacturers Association of America
CMT	Core Makeup Tank
COL	Combined Operating License/Combined License
COLA	Combined License Application
CP	Construction Phase
COLR	Core Operating Limits Report

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Table 1.1-1 (Sheet 3 of 12)
AP1000 Acronyms

COT	Channel Operational Test
CR	Control Room
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CRDS	Control Rod Drive System
CSDRS	Certified Seismic Design Response Spectra
CRF	Capital Recovery Factor
CS	Containment Shell
CSA	Control Support Area
CSP	Cyber Security Plan
CVAP	Comprehensive Vibration Assessment Program
CVS	Chemical and Volume Control System
CWO	Core Wide Oxidation
CWP	Cask Washdown Pit
DCH	Direct Containment Heating
DDT	Deflagration to Detonation Transition
DAC	Derived Air Concentration
DAC	Design Acceptance Criteria
dc	Direct Current
DBA	Design Basis Accident
DBE	Design Basis Event
DCD	Design Control Document
D-EHC	Digital Electrohydraulic Control
DEMA	Diesel Engine Manufacturers Association
DF	Decontamination Factor
DFBN	Debris Filter Bottom Nozzle
DID	Defense-In-Depth
DG	Diesel Generator
DLC	Direct Labor Cost
DLF	Dynamic Load Factor
DMIMS	Digital Mental Impact Monitoring System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy
DPU	Distributed Processing Unit
DRAP	Design Reliability Assurance Program
DRPI	Digital Rod Position Indication
DVI	Direct Vessel Injection
DWDI	Double Width Double Inlet
ECCS	Emergency Core Cooling System
EDE	Effective Dose Equivalent

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Table 1.1-1 (Sheet 4 of 12)
AP1000 Acronyms

EDI	Electrodeionization
EDL	Equivalent Dead Load
EDMG	Extensive Damage Mitigation Guidelines
EFPD	Effective Full Power Days
EMI	Electromagnetic Interference
ENS	Emergency Notification System
EOC	End of Cycle
EOF	Emergency Offsite Facility
EOL	End of Life
EOP	Emergency Operating Procedures
EPA	Environmental Protection Agency
EQSR	Equipment Qualification Summary Report
EP-ITAAC	Emergency Planning ITAAC
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
EQMEL	EQ Master Equipment List
ERF	Emergency Response Facility
ERG	Emergency Response Guidelines
ERO	Emergency Response Organization
FRP	Fiberglass Reinforced Plastic
ESDM	Excess Shutdown Margin
ESF	Engineered Safety Features
ESFAC	Engineered Safety Features Actuation Cabinet
ESFAS	Engineered Safety Features Actuation System
EW	East-West
FAC	Flow Accelerated Corrosion
FATT	Fracture Appearance Transitions Temperatures
FBTA	Function-Based Task Analysis
FCW	Flux-Core Welding
FFD	Fitness For Duty
FHA	Fire Hazards Analysis
FID	Fixed Incore Detector
FLB	Feedwater Line Break
FMEA	Failure Modes and Effects Analysis
FOAK	First of a Kind
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FTC	Fuel Transfer Canal
FWS	Feedwater System
GDC	General Design Criteria
GL	Generic Letter

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Table 1.1-1 (Sheet 5 of 12)
AP1000 Acronyms

GMAW	Gas Metal Arc Welding
GMRS	Ground Motion Response Spectrum
GRCA	Gray Rod Cluster Assembly
GRDM	Gray Rod Drive Mechanism
GSI	Generic Safety Issues
GTAW	Gas Tungsten Arc Welding
HCLPF	High Confidence, Low Probability of Failure
HELB	High Energy Line Break
HEM	Homogenous Equilibrium
HEPA	High Efficiency Particulate Air
HFE	Human Factors Engineering
HFP	Hot Full Power
HID	High-Intenisty Discharge
HP	Health Physics
HPCI	High Pressure Coolant Injection
HPME	High Pressure Melt Ejection
HPS	High Pressure Sodium
HRHF	Hard Rock High Frequency
HSI	Human System Interface
HSS	High Safety Significance
HV	High Voltage
HVAC	Heating, Ventilation and Air Conditioning
HX	Heat Exchanger
HZP	Hot Zero Power
I&C	Instrumentation and Control
IBC	International Building Code
IC	Intact Containment
ICEA	Insulated Cable Engineers Association
ICF	Indirect Cost Factor
IEEE	Institute of Electrical and Electronics Engineers
IES	Illumination Engineering Society
IFM	Intermediate Flow Mixing
IHS	Integrated Head Package
IIS	Incore Instrumentation System
ILRT	Integrated Leak Rate Test
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Plant Operations
I/O	Input/Output
IPC	Integrated Protection Cabinets
IRC	Inside Reactor Containment
IRWST	In-containment Refueling Water Storage Tank

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Table 1.1-1 (Sheet 6 of 12)
AP1000 Acronyms

ISA	Instrument Society of America
ISI	Inservice Inspection
ISLH	Inservice Leak and Hydrostatic
ISM	Independent Support Motion
ISRS	In-Structure Response Spectra
IST	Inservice Testing
ITA	Inspections, Tests, Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITC	Isothermal Temperature Coefficient
ITP	Initial Test Program
IVR	In-Vessel Retention
JOG	Joint Owners Group
JTWG	Joint Test Working Group
LBB	Leak-Before-Break
LCCF	Laor Cost Correction Factor
LCO	Limiting Condition for Operation
LERF	Large Early Release Frequency
LHR	Linear Heart Rate
LHX	Letdown Heat Exchanger
LHX	Letdown Heat Exchanger
LLC	Limited Liability Corporation
LLW	Low Level Waste
LMRS	Land Mobile Radio Service
LOCA	Loss of Coolant Accident
LOF	Loss-of-Flow with Failure to Scram
LOFT	Loss of Flow Test
LOOP	Loss of Offsite Power
LOSP	Loss of System Pressure with Degraded ECCS Operation
LPME	Low Pressure Melt Ejection
LPZ	Low Population Zone
LRF	Large Release Frequency
LSB	Last Stage Blade
LSS	Low Safety Significance
LSSS	Limiting Safety System Settings
LTOP	Low Temperature Overpressure Protection
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Programs
MCC	Motor Control Center
MCR	Main Control Room
MCRE	Main Control Room Envelope
MFCV	Main Feedwater Control Valve

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Table 1.1-1 (Sheet 7 of 12)
AP1000 Acronyms

MFIV	Main Feedwater Isolation Valve
MMI	Man-Machine Interface System
MOV	Motor-Operated Valve
MRI	Metallic Reflective Insulation
MRI	Maintenance Rule
MSE	Mechanically Stabilized Earth
MSHA	Mine Safety and Health Administration
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSR	Moisture Sepeartor/Reheater
MSSV	Main Steam Safety Valves
MTC	Moderator Temperature Coefficient
MW	Megawatt
MWe	Megawatt, electric
MWt	Megawatt, thermal
ND	Nuclear Development
NDCT	Natural Draft Cooling Tower
NDE	Non-destructive Examination
NDRC	National Defense Research Committee
NDTT	Nil-Ductility Transition Temperature
NE	North East
NEC	National Electrical Code
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
NI	Nuclear Island
NIOSH	National Institute for Occupational Safety and Health
NLO	Non-Licensed Operator
NNS	Non-Nuclear Safety
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTS	Nominal Trip Setpoint
NUMARC	Nuclear Management and Resources Council (Superseded by NEI)
NUREG	Report designator for NRC reports
NUS	Nuclear Utility Services
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
OE	Operating Experience
OJT	On-the-Job Training

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Table 1.1-1 (Sheet 8 of 12)
AP1000 Acronyms

OM	Operations and Maintenance
OPDMS	On-Line Power Distribution Monitoring System
ORC	Outside Reactor Containment
ORE	Occupation Radiation Exposure
OSC	Operations Support Center
OSP	Overspeed Protection System
P&ID	Piping and Instrumentation Diagram
PAM	Post Accident Monitoring
PAW	Plasma Arc Welding
PBX	PCCP
PC	Plant Condition
PCC	Power Coordination Center
PCCAWST	Passive Containment Cooling Ancillary Water Storage Tank
PCCP	Prestressed Concrete Cylinder Pipe
PCCWST	Passive Containment Cooling Water Storage Tank
PCMI	Pellet Clad Mechanical Interaction
PCP	Process Control Program
PCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PGA	Peak Ground Acceleration
PGS	Plant Gas System
PIRT	Phenomena Identification and Ranking Table
PIV	Pressure Isolation Valve
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PRHR	Passive Residual Heat Removal
PRHR HX	Passive Residual Heat Removal Heat Exchanger
PSARV	Pressure Safety and Relief Valve
PSI	Preservice Inspection
PS-ITAAC	Physical Security ITAAC
PVC	Polyvinyl Chloride
PWR	Pressurized Water Reactor
PWS	Potable Water System
PXS	Passive Core Cooling System
PZR	Pressurizer
QA	Quality Assurance
QAPD	Quality Assurance Program Description
QIN	Quickloc Instrument Nozzle
QMS	Quality Management System

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Table 1.1-1 (Sheet 9 of 12)
AP1000 Acronyms

QPTR	Quadrant Power Tilt Ratio
RAOC	Relaxed Axial Offset Control
RAP	Reliability Assurance Program
RAT	Reserve Auxiliary Transformer
RC	Reinforced Concrete
RCA	Radiological Controlled Area
RCCA	Rod Cluster Control Assmebly
RCDT	Reactor Coolant Drain Tank
RCIC	Reactor Core Isolation Cooling
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
REA	Rod Ejection Accident
RFI	Radio Frequency Interference
R.G.	Regulatory Guide
RHR	Residual Heat Removal
RHX	Regenerative Heat Exchanger
RIM	Required Input Motion
RIS	Regulatory Issue Summary
RLE	Review Level Earthquake
RNS	Normal Residual Heat Removal
RO	Reactor Operator
RO	Reverse Osmosis
RP	Radiation Protection
RPR	Rapid Power Reduction
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRS	Required Response Spectra
RS	Review Standard
RSW	Remote Shutdown Workstation
RT	Radiography Techniques
RTB	Reactor Trip Breaker
RTCB	Reactor Trip Circuit Breaker
RTD	Resistance Temperature Detector
RTDP	Revised Thermal Design Procedure
RTM	Reactor Trip Matrices
RTP	Rated Thermal Power
RTS	Reactor Trip System
RV	Reactor Vessel
RVHV	Reactor Vessel Head Vent

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Table 1.1-1 (Sheet 10 of 12)
AP1000 Acronyms

RWP	Radiation Work Permit
SAL	Safety Analysis Limit
SAMDA	Severe Accident Mitigation Design Alternatives
SAMG	Severe Accident Management Guidance
SAR	Safety Analysis Report
SAT	Systematic Approach to Training
SAW	Submerged Arc Welding
SBO	Station Blackout
SC	Steel Composite
SCBA	Self Contained Breathing Apparatus
SDM	Shutdown Margin
SDS	Sanitary Drains System
SECY	Secretary of the Commission Letter
SEI	Southern Electric International
SFAHT	Spent Fuel Assembly Handling Tool
SFDP	Safety Function Determination Program
SFP	Spent Fuel Pool
SFRSI	Spent Fuel Rack Storage Insert
SFW	Secondary Feedwater
SG	Steam Generator
SGMP	Steam Generator Management Program
SGS	Steam Generation System
SGTR	Steam Generator Tube Rupture
Shaw	Shaw Stone & Webster Nuclear Services
SL	Safety Limits
SLB	Steam Line Break
SMA	Seismic Margin Analysis
SMACNA	Sheet Metal and Air Conditioning Contractors National Association
SMAW	Shielded Metal Arc Welding
SNM	Special Nuclear Material
SOER	Significant Operating Event Reports
SOV	Solenoid Operated Valve
SP	Setpoint Program
SPRA	Seismic PRA
SR	Surveillance Requirement
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRSS	Square Root Sum of the Squares
SSAR	Standard Safety Analysis Report
SSC	Seismic Source Characterization
SSC	Stress Corrosion Cracking

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AP1000 Acronyms

SSC(s)	Structure(s), System(s), and Component(s)
SSD	System Specification Document
SSE	Safe Shutdown Earthquake
SSI	Soil Structure Interaction
SS-ITAAC	Site-Specific ITAAC
ST	Shunt Trip
STA	Shift Technical Advisor
SWSI	Single Width Single Inlet
TAC	Total Annual Cost
TADOT	Trip Actuating Device Operational Test
TF	Transfer Functions
TID	Total Integrated Dose
TLD	Thermoluminescent Dosimeter
TMD	Transient Mass Distribution
TMI	Three Mile Island
TOC	Total Organic Carbon
TRS	Test Response Spectrum
TS	Technical Specification(s)
TSC	Technical Support Center
TSP	Trisodium Phosphate
UAT	Unit Auxiliary Transformer
UBC	Uniform Building Code
UFSAR	Updated Final Safety Analysis Report
UL	Underwriters Laboratories
UP	Upper Plenum
UPS	Uninterruptible Power Supply
URD	Utility Requirements Document
USE	Upper Shelf Energy
USI	Unresolved Safety Issue
UT	Ultrasonic Techniques
UV	Undervoltage
V & V	Verification and Validation
VDU	Visual Display Unit
VFTP	Ventilation Filter Testing Program
VHRA	Very High Radiation Areas
WAC	Waste Acceptance Criteria
WBGT	Wet Bulb Globe Temperature
WABA	Wet Annular Burnable Absorber
WCAP	Westinghouse report designator, originally Westinghouse Commercial Atomic Power
WEC	Westinghouse Electric Company
WMA	Wildlife Management Area

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AP1000 Acronyms

WIN	Westinghouse Integral Nozzle
WPIS	Wall Panel Information System
ZPA	Zero Period Acceleration

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1.2 General Plant Description

This section includes a general discussion of the objectives, design criteria, operating characteristics and safety considerations for the AP1000 and provides a general description of the plant site, the site criteria, the general plant arrangement, the plant arrangement criteria and key features of each of the individual buildings that are collectively defined as the power generation complex.

Design Certification is sought for the power generation complex, excluding those elements and features considered site-specific. The AP1000 design extends beyond those structures, systems, and equipment which are safety-related. All safety-related structures, systems, and components are located on the nuclear island and are to be included in the design certification. To provide a better understanding of the safety-related features of the AP1000, nonsafety-related features are also described in this DCD. In addition, some plant design features which are outside the boundary of the AP1000, and considered to be site-specific, are described for completeness and to provide a basis for quantification of the required interfaces, as required by 10 CFR 52.47 (a)(25). The site-specific structures located off the nuclear island are neither safety-related nor seismic Category I. A more complete description of interfaces for the standard design is contained in Section 1.8.

1.2.1 Design Criteria, Operating Characteristics, and Safety Considerations

This section provides an overview of the AP1000 design objectives, design criteria, operating characteristics and safety considerations.

1.2.1.1 Overall Plant

The primary objective of the AP1000 design is to meet applicable safety requirements and goals defined for advanced light water pressurized water reactors with passive safety features. Since the AP600 has already received a Design Certification, it is also a design objective for AP1000 to be as similar as possible to the AP600.

Westinghouse was a principal participant in the development of the EPRI sponsored Utility Requirements Document (URD) and continues to be involved with EPRI on changes to that document. Therefore, an objective of the AP1000 design is to remain as consistent as possible with the EPRI URD. Additional design objectives for the AP1000 are to provide a greatly simplified plant with respect to design, licensing, construction, operation, inspection and maintenance. Specific design objectives follow.

1.2.1.1.1 Power Capability Objectives

- The plant's net electrical power to the grid is at least 1000 MWe with a nuclear steam supply system power rating (core plus reactor coolant pump heat) of about 3415 MWt.
- The plant is designed for rated performance with up to 10 percent of the steam generator tubes plugged and with a maximum hot leg temperature of 610°F.
- The plant is designed to accept a step load increase or decrease of 10 percent between 25 and 100 percent power without reactor trip or steam dump system actuation provided the rated power level is not exceeded.
- The plant is designed to accept a 100 percent load rejection from full power to house loads without reactor trip or operation of the pressurizer or steam generator safety valves. The design provides for a turbine capable of continued stable operation at house loads.

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- The plant is designed to accept ramp load changes of 5 percent per minute while operating in the range of 25 to 100 percent of full power without reactor trip or steam dump actuation subject to core power distribution limits and provided the rated power level is not exceeded.

- The plant is designed to permit a design basis daily load follow cycle for at least 90 percent of the fuel cycle length. The design basis daily load follow cycle is defined as the daily (24 hour period) cycle of operation at 100 percent power, followed by a 2-hour linear ramp to 50 percent power, operation at 50 percent power and a 2-hour linear ramp back to 100 percent power. The duration of time at 50 percent power can vary between 2 and 10 hours. This load follow capability is achievable during 90 percent of each fuel cycle.

- During load follow the plant is designed to routinely make load changes of ≤ 10 percent at ± 2 percent per minute between 50 and 100 percent power without exceeding the core power distribution limits for the purpose of responding to grid frequency changes. No change to the reactor coolant boron concentration is required during these load follow maneuvers.

1.2.1.1.2 Reliability and Availability Objectives

- The overall plant availability goal is greater than 90 percent considering all forced and planned outages.

- The rate of unplanned reactor trips goal is less than one per year.

- The plant is designed with significantly fewer components and significantly fewer safety-related components than a current pressurized water reactor of a comparable size.

- The plant design objective is 60 years without the planned replacement of the reactor vessel which itself has a 60 year design objective based on conservative assumptions. The design provides for the replaceability of other major components, including the steam generators.

- The design of the major components required for power generation such as the steam generators, reactor coolant pumps, fuel, internals, turbine and generator is based on equipment that has successfully operated in power plants. Modifications to these proven designs were based on similar equipment that had successful operating experience in similar or more severe conditions.

1.2.1.1.3 Safety Design Criteria

- The plant design conforms to applicable regulations as discussed in Sections 1.9 and 3.1.

- The plant is designed to be fabricated, erected, and operated in such a manner that the release of radioactive materials to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations and for design basis transients and accidents.

- Gaseous and liquid waste disposal facilities are designed so that the discharge of radioactive effluents can be made in accordance with applicable regulations.

- The design provides means by which plant operators are alerted when limits on the release of radioactive effluent are approached.

- The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient.

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- The reactor is designed so that there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the reactor with other appropriate plant systems.
- Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered by plant safety analysis.
- Essential safety actions are provided by equipment of sufficient redundancy and independence so that no single failure of active components can prevent the required actions.
- Provisions are made for control of active components of nuclear safety systems and engineered safety features from the control room.
- Those portions of the nuclear steam supply system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material containment barrier following design basis operational transients and accidents.
- Nuclear safety systems and engineered safety features functions are designed so that no damage to the reactor coolant pressure boundary results from internal pressures caused by design basis operational transients and accidents.
- Nuclear safety systems and engineered safety features are designed to permit demonstrations of their functional performance requirements.
- The design of nuclear safety systems and engineered safety features includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the station site.
- Standby electrical power sources have sufficient capacity to power the nuclear safety systems and engineered safety features requiring electrical power. Safety-related electrical power requirements needed during a loss of all ac power are supplied via Class 1E dc power.
- Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.
- A containment is provided which completely encloses the reactor system.
- The containment is designed to allow periodic integrity and leak tightness testing.
- The containment, in conjunction with other engineered safety features, limits the release of radioactivity from inside the containment, in the event of a design basis accident. This has the effect of limiting radiological consequences of a design basis accident to within an appropriate fraction of regulatory guidelines.
- Piping that penetrates the containment and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation is effected in time to limit radiological effects to less than the specified acceptable limits.
- Provisions are made for passively removing energy from the containment to maintain the integrity of the containment system following accidents that release energy to the containment.

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- Passive core cooling systems are provided to limit fuel cladding temperature to less than the limits established by 10 CFR 50.46 in the event of a loss-of-coolant accident.
- The passive core cooling system provides for core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary.
- Actuation of the passive core cooling system occurs automatically when required, regardless of the availability of offsite power supplies and the normal generating system.
- The control room is shielded against radiation so that continued occupancy under accident conditions is possible.
- In the event that the control room becomes uninhabitable, it is possible to bring the reactor from power range operation to safe shutdown conditions by utilizing the remote shutdown workstation located outside the control room.
- Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shutdown the reactor from any normal operating conditions and subsequently to maintain the shutdown condition.
- The fuel handling and storage facility is designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel.

1.2.1.1.4 Site Objectives

- The plant is designed for location at a site with the parameters set forth in Chapter 2, Site Characteristics.

1.2.1.1.5 Other Objectives

- The radiation exposure goal for plant personnel resulting from normal operation, inspection and maintenance is less than 100 man-Rem/year. Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operations.
- The total low level radioactive waste volume goal is less than 1,970 cubic feet per year after de-watering. This waste includes items such as spent resins, spent filter elements, tank sludge, chemical wastes, and clothing. Spent condensate polishing resins are not included. The total wet radioactive waste volume produced from spent resin and filter elements, tank sludge and chemical waste is designed not to exceed 550 cubic feet per year (de-watered).

1.2.1.2 Reactor Coolant System Design

The AP1000 reactor coolant system (Figure 5.1-2) is designed to remove or to enable removal of heat from the reactor during all modes of operation, including shutdown and accident conditions.

The system consists of two heat transfer circuits, each with a steam generator, two reactor coolant pumps, a single hot leg and two cold legs, for circulating reactor coolant. In addition the system includes a pressurizer, interconnecting piping, valves and instrumentation necessary for operational control and safeguards actuation. All system equipment is located in the reactor containment.

During operation, the reactor coolant pumps circulate pressurized water through the reactor vessel and the steam generators. The water, which serves as coolant, moderator and solvent for boric acid (chemical shim control), is heated as it passes through the core to the steam generators where the

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heat is transferred to the steam system. The water is then is returned to the reactor (core) by the pumps to repeat the process.

The reactor coolant system pressure is controlled by operation of the pressurizer, where water and steam are maintained in equilibrium by the activation of electrical heaters and/or a water spray. Steam is formed by the heaters or condensed by the water spray to control pressure variations due to expansion and contraction of the reactor coolant.

Overpressure protection for the reactor coolant system is provided by the spring loaded safety valves installed on the pressurizer. These valves discharge to the containment atmosphere. The valves for the first three stages of automatic depressurization are also mounted on the pressurizer. These valves discharge steam through spargers to the in-containment refueling water storage tank. The discharged steam is condensed and cooled by mixing with water in the tank.

The reactor coolant system is also served by a number of auxiliary systems, including the chemical and volume control system, the passive core cooling system, the spent fuel pool cooling system, the steam generator system, the primary sampling system, the liquid radwaste system and the component cooling water system.

1.2.1.2.1 Reactor Design

- The core is designed for an 18-month fuel cycle.
- There are no reactor vessel penetrations below the top of the core.
- The core is designed for a moderator temperature coefficient that is non-positive over the entire fuel cycle at any power level with the reactor coolant at the normal operating temperature.
- A core design is maintained for projected fuel cycles.
- The core design provides adequate margin so that departure from nucleate boiling will not occur with a 95 percent probability and 95 percent confidence basis for all Condition I and II events.
- The core is located low in the vessel to minimize core temperature during loss-of-coolant accidents.
- The vessel and internals are designed so coolant at approximately the average of T_{cold} and T_{hot} is maintained in the head and control rod drive mechanism regions.
- The lower internals are designed to prevent flow jetting into the core.
- Bottom mounted incore instrumentation is not used. No vessel penetrations exist below the top of the core.
- An integrated head package which contains a forced air cooling system, radial arm hoist system, control rod drive mechanism, seismic support system, integrated head package lifting rig, radiation shielding, integrated cable support system, integrated head package service structures, and shroud that provides housing for the control rod drive mechanisms is employed.
- A welded permanent cavity seal ring is used to provide the seal between the vessel flange and the refueling cavity floor.

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1.2.1.2.2 Steam Generator Design

- The Model Delta 125 steam generator of proven design is employed. The steam generator employs thermally treated nickel-chromium-iron Alloy 690 tubes and a steam separator area sludge trap with clean out provisions.
- The channel head is designed for the direct attachment of two reactor coolant pumps.
- The channel head is designed for both manual and robotic accessibility for inspection, plugging, sleeving and nozzle dam placement operations.

1.2.1.2.3 Reactor Coolant Pump Design

- Sealless pumps of proven design are used.
- Two reactor coolant pumps are attached directly to each steam generator channel head with the motor located below the channel head to simplify the loop piping and minimize fuel uncovery during small loss-of-coolant accidents.
- Each reactor coolant pump includes sufficient internal rotating inertia to provide a flow coastdown to avoid departure from nucleate boiling following a loss of reactor coolant flow accident.
- The reactor coolant pump motors are designed with appropriate lifting and handling attachments (lugs and trunnions) to facilitate maintenance.
- The reactor coolant pumps are designed such that they are not damaged due to a loss of all cooling water until a reactor coolant pump trip occurs on High-2 bearing water temperature. This automatic protection is provided to protect the reactor coolant pumps from an extended loss of coolant water.

1.2.1.2.4 Pressurizer and Loop Arrangement

- The piping layout is designed for adequate thermal expansion flexibility assuming a fixed vessel and a free floating steam generator/reactor coolant pump support system.
- The reactor coolant loop and surge line piping are designed to leak-before-break criteria.
- The pressurizer is designed such that, with design spray flow rates, the power-operated relief valve function is not required nor provided.

1.2.1.3 Steam and Power Conversion System Design

1.2.1.3.1 Turbine Design

- The turbine is a power conversion system designed to change the thermal energy of the steam flowing through the turbine into rotational mechanical work which rotates a generator to provide electrical power. It consists of a double flow high pressure cylinder (high pressure turbine) and three double flow low pressure cylinders (low pressure turbines) which exhaust to the condenser. It is a six flow tandem compound, 1800 rpm machine. The turbine system includes stop, control and intercept valves attached to the turbine and in the steam flow path, crossover and crossunder piping between the turbine cylinders and the moisture separator reheater.

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- The high pressure turbine has connections for two stages of feedwater heating. The high pressure turbine exhaust steam provides steam for one stage of feedwater heating in the deaerator. The low pressure turbines have extraction connections for four stages of feedwater heating.

- The moisture separator reheaters are an integral component of the turbine system which extracts moisture from the steam and reheats the steam to improve the turbine system performance. There are two moisture separator reheaters located between the high pressure turbine exhaust and the low pressure turbine inlet. The reheater has two stages of reheat.

- The turbine orientation minimizes potential interaction between turbine missiles and safety-related structures and components.

1.2.1.3.2 Main Steam System Design

- The main steam system is designed to supply steam from the steam generators to the high pressure turbine over a range of flows and pressures for the entire plant operating range, i.e., from system warmup to valves-wide-open turbine conditions.

- The main steam system is also designed to dissipate heat generated by the nuclear steam supply system to the condenser through steam dump valves or to the atmosphere through power-operated atmospheric relief valves or spring-loaded main steam safety valves when either the turbine-generator or the condenser is not available.

- Six steam generator safety valves are utilized per steam header. There are two steam headers.

1.2.1.3.3 Main Feedwater and Condensate System Design

- The main feedwater system is designed to supply the steam generators with adequate feedwater during all modes of plant operation including transient conditions. The condensate system is designed to condense and collect steam from the low-pressure turbines and turbine steam bypass systems and then, transfer this condensate from the main condenser to the deaerator.

- The main feedwater and condensate systems are designed for increased availability and improved dissolved oxygen control.

- A deaerating heater is employed.

1.2.1.4 Auxiliary Fluid Systems Design

1.2.1.4.1 Engineered Safeguards Systems Design

- The safety systems are designed to mitigate design basis accidents with a single failure, as defined in Chapter 15.

- The safety systems are designed to maximize the use of natural driving forces such as pressurized nitrogen, gravity flow and natural circulation flow. They do not use active components such as pumps, fans or diesel generators. A minimum number of valves are used for the purpose of initially aligning the safety systems.

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- The safety systems are designed to function without safety-related support systems such as alternating current, component cooling water, service water, heating, ventilation and air conditioning.

- The number and complexity of operator actions required to control the safety systems are minimized. In meeting this objective, the approach was to eliminate the required action and not to automate them.

- An automatic reactor coolant system depressurization feature is included in the design and meets the following criteria:

  – The reliability (redundancy and diversity) of the automatic depressurization system valves and controls satisfies single failure criteria as well as the failure tolerance required by the low core melt frequency goals.

  – The design provides for both real demands (such as reactor coolant system leaks and failure of the chemical and volume control system makeup pumps) and spurious instrumentation signals. The probability of significant flooding of the containment due to the use of the automatic depressurization system is less than once in 600 years.

- The passive safety-related systems can operate for at least 1 hour following anticipated transients without release of contaminants that require significant plant cleanup. The automatic depressurization system is designed not to activate for anticipated transients.

- The passive safety-related systems are designed to cool the reactor coolant system from normal operating temperatures to safe shutdown conditions.

- The passive containment cooling system maintains the containment pressure and temperature within the appropriate design limits for both design basis and severe accident scenarios.

1.2.1.4.2 Nonsafety-Related Systems Designs

- The nonsafety-related systems designs are simplified; the number of systems and components and the complexity of operation and maintenance are reduced from current operating plants.

- The nonsafety-related systems are not relied upon to provide safety functions required to mitigate design basis accidents.

- Nonsafety-related systems that are required for normal plant operation provide high plant availability. These systems have appropriate redundancy, are powered by onsite standby power supplies and have sufficient capacity to prevent automatic passive safety system actuation following anticipated Condition II events.

  – The reactor coolant system makeup capability design is sufficient for reactor coolant leaks up to 3/8 inch.

  – Steam generator feedwater capability from the startup feedwater system is designed to provide sufficient flow for a loss of main feedwater event.

  – The normal containment sump pumps (part of the liquid radwaste system) are designed to assist in recovery from leakage to the containment sump.

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- Boric acid solutions are designed to be stored at concentrations that do not require heat tracing or room temperatures above normal values.

1.2.1.5 Electrical and Control Systems Designs

1.2.1.5.1 Control and Protection Systems Designs

- The design provides that during normal operation, a single failure in the protection and safety monitoring system does not result in a reactor trip, with the exception of a limited number of PMS component failures, e.g., certain CIMs. This is true even with a channel under maintenance or test.

- The potential for reactor trip and for safeguards actuation due to failures in the plant control system is reduced relative to current operating plants.

- The number of measured plant variables used for reactor trip and for safeguards actuation is minimized relative to current operating plants.

- The margin between the normal operating conditions and the protection system setpoints is increased relative to current operating plants.

- The potential for interaction between the protection and safety monitoring system and the plant control system is reduced relative to current operating plants by incorporating a signal selector function that selects signals for control and for protection.

1.2.1.5.2 Alternating Current and Direct Current Power Design

- Safety-related direct current (dc) power is provided to support reactor trip and engineered safeguards actuation. Batteries are sized to provide the necessary dc power and uninterruptible ac power for items such as the protection and safety monitoring system actuation, the control room functions including habitability, dc-powered valves in the passive safety-related systems and containment isolation.

- All safety-related electrical power is provided from the Class 1E dc power system. No separate safety-related ac power system is required.

1.2.1.5.3 Control Room Design

- A main control room is provided that is able to control the plant during normal and anticipated transients and design basis accidents. The main control room includes indications and controls capable of monitoring and controlling the plant safety systems as well as the nonsafety-related control systems.

- A remote shutdown capability is provided. The remote shutdown workstation contains the indications and controls that allow an operator to achieve and maintain safe shutdown of the plant following an event when the main control room is unavailable. Additional nonsafety-related indications and controls are provided as described in Chapter 7.

- The remote shutdown workstation contains indications and controls consistent with its intended use; i.e., the remote shutdown workstation is to be used in the unlikely event that the main control room is not available.

- Access to the remote shutdown workstation transfer mechanism is under strict administrative control.

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- The main control room is serviced by reliable and redundant nonsafety-related power sources and heating, ventilation and air conditioning systems during normal operation.
- In the unlikely event that the normal power source or the heating, ventilating and air conditioning system becomes unavailable, there are passive systems (batteries, compressed air) to support the main control room for up to 3 days.
- The main control room contains the safety-related instrumentation and controls to allow the operator to achieve and maintain safe shutdown following any design basis accident.
- The safety-related power sources and passive cooling system are designed to provide a habitable environment for the operating staff assuming that no ac power is available. Installed equipment provides for at least 3 days of operation, as stated above. After 3 days, it is possible to continue operation with the control room cooled and ventilated with circulation of outside air.
- A mechanism is provided to allow the operating staff to transfer control from the main control room to the remote shutdown workstation.
- The system prevents spurious signals caused by fire damage from being issued to components once transfer to the remote shutdown workstation has been effected.
- The transfer of the control of components to the remote shutdown workstation is alarmed in the main control room.
- Both the main control room and the remote shutdown workstation are designed in accordance with human factors engineering principles and practices.
- Human factors considerations are utilized so that the indications and controls for the remote shutdown workstation are similar to those provided in the main control room.
- The safety-related instrumentation (equipment racks) is maintained at acceptable ambient conditions for 3 days following a loss of all ac power by using a passive cooling system. After 3 days, it is possible to continue operation with the instrumentation and control rooms cooled by circulation of outside air.
- A technical support center is provided.

1.2.1.6 Plant Arrangement and Construction

1.2.1.6.1 Plant Arrangement

- The plant arrangement is comprised of five principal building structures; the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building (see Figure 1.2-3).
- The nuclear island is structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29. The nuclear island consists of a free-standing steel containment building, a concrete shield building, and an auxiliary building. The foundation for the nuclear island is an integral basemat which supports these buildings.

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- The nuclear island structures are designed to withstand the effects of natural phenomena such as hurricanes, floods, tornados, tsunamis, and earthquakes without loss of capability to perform safety functions. Design for natural phenomena is based on the industry standards as described in Chapters 2 and 3.

- The nuclear island is designed to withstand the effects of postulated internal events such as fires and flooding without loss of capability to perform safety functions.

- The main area of the turbine building structure is designed to International Building Code requirements. The first bay of the turbine building is designed to seismic Category II requirements. The turbine building is supported on a single basemat foundation.

- The annex building area outlined by Column lines E – I.1 and 2 – 13 is designed to seismic Category II requirements. The rest of the annex building area is designed to International Building Code requirements. The annex building includes functions such as the health physics area, the control support area, access control, and personnel facilities (shower and locker rooms).

- The diesel generator building houses two diesel generators and their associated heating, ventilation and air conditioning equipment. The building is a nonseismic structure designed for wind and seismic loads in accordance with the Uniform Building Code.

- The radwaste building contains facilities for the handling and storage of plant wastes. It is a nonseismic structure designed for wind and seismic loads in accordance with the Uniform Building Code. The foundation for the building is a reinforced concrete mat on grade.

- Radioactive equipment and piping in all buildings are arranged and shielded to minimize radiation exposure.

- The overall plant arrangement utilizes building configurations and structural designs to minimize the building volumes and quantities of bulk materials (concrete, structural steel, rebar) consistent with safety, operational, maintenance, and structural needs.

- The plant arrangement provides separation between safety-related and nonsafety-related systems to preclude adverse interaction between safety-related and nonsafety-related equipment. Separation between redundant safety-related equipment and systems provides confidence that the safety design functions can be performed. In general this separation is provided by partitioning an area with concrete walls.

- The plant arrangement provides separation for radioactive and non-radioactive equipment and provides separate pathways to these areas for personnel access.

- Pathways through the plant are designed to accommodate equipment maintenance and equipment removal from within the plant. The size of the pathways is dictated by the largest appropriate piece of equipment that may have to be removed or installed after initial installation. Where required, laydown space is provided for disassembling large pieces of equipment to accommodate the removal or installation process.

- Adequate space is provided for equipment maintenance, laydown, removal and inspection. Hatches, monorails, hoists, and removable shield walls are provided to facilitate maintenance.

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1.2.2 Site Description

Site Characteristics

The AP1000 is a standard plant that is to be placed on a site with parameters bounded by those used as a basis for design certification as described in Chapter 2, Site Characteristics. The site parameters relate to the seismology, hydrology, meteorology, geology, heat sink and other site-related aspects. The allowable site interface parameters bound a large percentage of potential sites.

The AP1000 is designed on the basis that the equipment, modules, structures, and bulk material can be shipped to the site by commercial rail or truck. This does not preclude the shipment of large equipment or structures by barges should a specific site be accessible by water.

Site Plan

The site plan is defined in the site specific licensing process. A proposed plan has been provided for site interface purposes. Specific details of the site plan will be covered in the site application or in the combined license application.

A typical site plan for the single unit AP1000 reference is shown on Figure 1.2-2. Direction of north, south, east and west used in this description are nominal site description directions and have no relationship to directions on an actual site. With the exception of the parking area, the entire facility is contained within the perimeter fence. The area within the 1400 feet x 775 feet perimeter fence is approximately 25 acres.

As previously stated in Subsection 1.2.1.6.1, the power block complex consists of five principal building structures; the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building. Each of these building structures is constructed on an individual basemat. The nuclear island consists of the containment building, the shield building, and the auxiliary building, all of which are constructed on a common basemat.

As shown on the site plan, Figure 1.2-2, these building structures are oriented such that the turbine building is located to the north of the nuclear island, with the other principal buildings adjacent to the nuclear island to meet their functional purpose.

The reference plant main cooling tower-circulating water pump complex consists of a natural draft cooling tower, a pump basin, and circulating water pumps. The final configuration of the cooling tower is site-specific.

The circulating water pumps circulate the cooling water from the pump basin to the main condenser and back to the cooling tower through two precast concrete supply and return pipes that are below grade.

The transformer area is located immediately adjacent to and north of the turbine building. The unit auxiliary transformers, the reserve auxiliary transformer and the main step-up transformers are located in the transformer area. The main switchyard area is site-specific.

The rail and road accesses to the site are through the east perimeter fence.

During construction, a heavy lift crane is used to place major pieces of equipment such as the turbine-generator, the reactor vessel, the steam generators, containment ring sections, large structural modules and other large or heavy equipment modules.

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Figure 1.2-3 provides a functional representation of the principal systems and components that are located in each of the key AP1000 buildings. This figure identifies major systems and components that are contained in these structures.

1.2.3 Plant Arrangement Description

Building Definition

A set of the general arrangement drawings for the AP1000 is provided in Figures 1.2-4 through 1.2-29.

Figure 1.2-201 replaces Figure 1.2-18 to reflect the relocation of the Operations Support Center.

The AP1000 consists of the following five principal structures. Each of these buildings is constructed on an individual basemat:

- Nuclear island
- Turbine building
- Annex building
- Diesel generator building
- Radwaste building

The structures that make up the nuclear island are:

- Containment building
- Shield building
- Auxiliary building

These nuclear island buildings are depicted on the site plan. The safety-related equipment designed to perform accident mitigation functions is located in the nuclear island.

1.2.4 Nuclear Island

1.2.4.1 Containment Building

Building Function

The containment building is the containment vessel and the structures contained within the containment vessel. The containment building is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity following postulated design basis accidents and providing shielding for the reactor core and the reactor coolant system during normal operations.

The containment vessel is an integral part of the passive containment cooling system. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents.

The containment building is designed to house the reactor coolant system and other related systems and provides a high degree of leak tightness.

Civil/Structural Features

The containment building, a seismic Category I structure, is a freestanding cylindrical steel containment vessel with elliptical upper and lower heads. It is surrounded by a seismic Category I shield building.

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Figures 1.2-13 through 1.2-16 provide sectional views through the containment that show the configuration of the containment vessel and the internal structures of the containment.

There are two floor elevations (grade access maintenance floor and operating deck) and four lower equipment compartments within the containment building. Removable hatches are provided for access to equipment at other elevations.

Figures 1.2-7, 1.2-8, 1.2-9, 1.2-15, and 1.2-16 depict the configuration of the refueling water storage tank. This tank is located below the operating deck. The capacity of the refueling water storage tank exceeds the quantity of water required to accomplish safety functions or to fill the refueling cavity during refueling operations. The refueling cavity has two floor elevations. The upper and lower reactor internals storage area is at the lower elevation as is the fuel transfer tube.

Equipment Arrangement

The principal system located within the containment building is the reactor coolant system that consists of two main coolant loops, a reactor vessel, two steam generators, four sealless reactor coolant pumps, and a pressurizer. Figures 1.2-9, 1.2-14 and 1.2-16 depict the reactor coolant system component locations in the containment.

The main steam and feedwater lines are routed from the steam generators to a horizontal run below the operating deck. The steam and feedwater lines penetrate the north side of the containment vessel and are routed through the main steam isolation valve area in the auxiliary building to the turbine island.

The passive core cooling system is also located in the containment building. The primary components of the passive core cooling system are two core makeup tanks, two accumulators, the refueling water storage tank, the passive residual heat removal heat exchanger, and two spargers. The first three stages of the automatic depressurization valves are located above the pressurizer and consist of a two-tier valve module.

The passive residual heat removal heat exchanger and the spargers are located within the refueling water storage tank (Figures 1.2-7 and 1.2-9). The core makeup tanks are located on floor elevation 107′2″ level (Figures 1.2-7 and 1.2-9).

The chemical and volume control system equipment module is located in the containment below the maintenance floor level. This module represents the high pressure purification loop of the chemical and volume control system (Figure 1.2-14).

The reactor coolant drain tank, the reactor coolant drain tank heat exchanger and the containment sump pumps are located in the compartment adjacent to the reactor vessel cavity. Access to the reactor vessel cavity is via a stairwell that descends from the maintenance floor (Figure 1.2-14).

Four containment recirculation fan coil unit assemblies are located adjacent to the steam generator compartments. Each consists of a vaneaxial fan, cooling coils, and the associated exit ducts and inlet plenum. The four fan coil unit assemblies are connected to the common exit plenum (ring header). Several vertical ducts branch off from the ring header to provide cooling flow to the lower compartments in the containment while other vertical ducts are directed up to provide cooling flow to the upper regions of the containment vessel.

Equipment and Material Handling

A seismic Category I polar crane is provided in the containment and its bridge is sized for lifting the steam generator during a steam generator removal operation. A temporary construction trolley is

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required for this operation. The polar crane support is attached to the steel cylindrical shell of the containment as shown in Figures 1.2-14 and 1.2-16.

The layout of the containment is designed to permit the removal of either steam generator through a temporary opening cut through the top of containment, then through the center of the passive containment cooling air diffuser. During a steam generator removal operation, the steam generator is lifted from the steam generator compartment by a temporary construction trolley and then through containment by a large mobile crane.

The polar crane trolley is designed for normal refueling operations such as lifting the integrated head package, the lower internals package and the upper internals package.

An auxiliary hook is provided with the polar crane for easier movement of smaller equipment. The polar crane is used for lifting reactor coolant pump motor/impeller assemblies from the steam generator/loop compartments to the operating deck in the event that the reactor coolant pump motor/impeller assemblies have to be removed from the containment for major maintenance.

A reactor coolant pump maintenance cart is provided for use in either of the two steam generator/loop compartments for removing the reactor coolant pump motor/impeller assemblies from the bottom head of the steam generators. This maintenance cart transports the reactor coolant pump motor/impeller assemblies to a designated area in each of the steam generator/loop compartments where the assemblies are lifted from the compartment to the operating deck by the polar crane. Removable sections of grating at all platform levels in the steam generator/loop compartments permit direct access to the pumps. From the operating deck level, the reactor coolant pump motor/impeller assemblies are removed from the containment via the main equipment hatch into the annex building maintenance area.

A refueling machine is provided to move fuel between the fuel transfer system and the reactor core (Figure 1.2-14). The refueling machine consists of a rectilinear bridge and a trolley crane with a vertical mast extending down into the refueling cavity. The bridge spans the refueling cavity and runs on rails set into the edge of the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly. In addition, the refueling machine is equipped with an auxiliary hoist which provides additional capability for other refueling operations.

A fuel transfer system is provided to transfer nuclear fuel assemblies between the refueling cavity in the containment building and the fuel transfer canal/spent fuel pool located in the fuel handling area of the auxiliary building. The fuel transfer system also has the capability to transfer control rod clusters.

Building Access and Exit

Access to the containment is provided through a personnel airlock and the main equipment hatch located at the operating deck level and a personnel airlock and a maintenance hatch at the maintenance floor level. Access to the containment can be controlled by the health physics office in the annex building.

In the event that large numbers of temporary personnel require access to the containment during a major outage, temporary personnel facilities can be provided immediately adjacent to the health physics area in the annex building.

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1.2.4.2 Shield Building

Building Function

The shield building is the structure that surrounds the containment vessel. During normal operations, a primary function of the shield building is to provide shielding for the containment vessel and the radioactive systems and components located in the containment building. The shield building, in conjunction with the internal structures of the containment building, provides the required shielding for the reactor coolant system and the other radioactive systems and components housed in the containment.

Another function of the shield building is to protect the containment building from external events. The shield building protects the containment vessel and the reactor coolant system from the effects of tornadoes and tornado produced missiles.

During accident conditions, the shield building provides the required shielding for radioactive airborne materials that may be dispersed in the containment as well as radioactive particles in the water distributed throughout the containment.

The shield building is an integral part of the passive containment cooling system.

Civil/Structural Features

The shield building is a seismic Category I structure. It shares a common basemat with the containment building and the auxiliary building.

Figures 1.2-13 through 1.2-16 provide sectional views of the shield building which show the basic configuration of the shield building and the annulus area between the containment vessel and the shield building.

The following items represent the significant features of the shield building and the annulus area:

- Shield building cylindrical structure
- Shield building roof structure
- Connections between reinforced concrete and steel concrete composite modules
- Tension ring at the interface between the roof and shield building cylinder
- Knuckle Region at interface of roof and outer wall of passive containment cooling system water storage tank
- Compression ring below the inner wall of the passive containment cooling system water storage tank
- Lower annulus area
- Middle annulus area
- Upper annulus area
- Passive containment cooling system air inlets at the top of the shield building cylinder
- Passive containment cooling system air inlet plenum
- Passive containment cooling system water storage tank
- Passive containment cooling system air diffuser
- Passive containment cooling system air baffle

The cylindrical section of the shield building serves as shielding and a missile barrier and is a key component of the passive containment cooling system. It structurally supports the roof and is a major structural member for the entire nuclear island. Floor slabs and structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

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A watertight seal is provided between the upper and middle annulus areas to provide an environmental barrier. The middle annulus area contains the majority of containment penetrations and radioactive piping. This environmental barrier is provided to protect against the following:

- In the event of an accident or spurious actuation, the passive containment cooling system drains the system water storage tank. The water, which runs down the outside of the containment vessel, is prevented from draining into the middle annulus area by the watertight seal. Drains are provided to direct the passive containment cooling system runoff water out of the shield building.

- The passive containment cooling system is designed to perform with the upper annulus permanently open to the environment to permit sufficient air flow through the shield building in the event of an accident. The watertight seal protects the middle annulus area from ambient environmental conditions.

The shield building roof is a reinforced concrete conical shell supporting the passive containment cooling system water storage tank and air diffuser. Air intakes are located at the top of the cylindrical portion of the shield building. The conical roof supports the passive containment cooling system water storage tank which is constructed with a stainless steel liner attached to reinforced concrete walls. The air diffuser in the center of the roof discharges containment cooling air directly upwards.

The passive containment cooling system air baffle is located in the upper annulus area. It is attached to the cylindrical section of the containment vessel. The function of the passive containment cooling system air baffle is to provide a pathway for natural circulation of cooling air in the event that a design basis accident results in a large release of energy into the containment. In this event the outer surface of the containment vessel transfers heat to the air between the baffle and the containment shell. This heated and thus, lower density air flows up through the air baffle to the air diffuser and cooler and higher density air is drawn into the shield building through the air inlets at the top cylindrical portion of the shield building.

Equipment and Material Handling

A monorail is provided in the upper annulus area of the shield building to facilitate the initial installation of the passive containment cooling system air baffle panels and to permit the removal of these air baffle panels when an inspection or repainting of the containment vessel is required.

Two personnel workstation platforms are provided for transporting staff and equipment from the operating deck floor level of the upper annulus area to the top of the shield building. The work station platforms are powered from their respective monorail sections and are able to be positioned at any circumferential position or height beneath the monorail sections. Figures 1.2-14 and 1.2-16 depict the monorail system and the personnel work station platforms.

1.2.4.3 Auxiliary Building

Building Function

The primary function of the auxiliary building is to provide protection and separation for the seismic Category I mechanical and electrical equipment located outside the containment building.

The auxiliary building provides protection for the safety-related equipment against the consequences of either a postulated internal or external event. The auxiliary building also provides shielding for the radioactive equipment and piping that is housed within the building.

The most significant equipment, systems, and functions contained within the auxiliary building are the following:

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- Main control room
- Class 1E instrumentation and control systems
- Class 1E electrical system
- Fuel handling area
- Mechanical equipment areas
- Containment penetration areas
- Main steam and feedwater isolation valve compartment

Main control room: The main control room provides the human system interfaces required to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions. The main control room includes the main control area, the operations work area, the operations break area, and an office for the shift manager.

Instrumentation and control systems: The protection and safety monitoring system and the plant control system provide monitoring and control of the plant during startup, ascent to power, powered operation, and shutdown. The instrumentation and control systems include the protection and safety monitoring system, the plant control system, and the data display and processing system.

Class 1E electrical system: The Class 1E system provides 250 volts dc power for safety-related and vital control instrumentation loads including monitoring and control room emergency lighting. It is required for safe shutdown of the plant during a loss of ac power and during a design basis accident with or without concurrent loss of offsite power.

Fuel handling area: The primary function of the fuel handling area is to provide for the handling and storage of new and spent fuel. The fuel handling area in conjunction with the annex building provides the means for receiving, inspecting and storing the new fuel assemblies. It also provides for safe storage of spent fuel as described in Section 9.1, Fuel Storage and Handling.

The fuel handling area provides for transferring new fuel assemblies from the auxiliary building rail car bay to and from the new fuel storage area to the containment building and for transferring spent fuel assemblies from the containment building to the spent fuel storage pit within the auxiliary building.

The fuel handling area provides the means for removing the spent fuel assemblies from the spent fuel storage pit and loading the assemblies into a shipping cask for transfer from the facility.

The fuel handling area is protected from external events such as tornadoes and tornado produced missiles. Protection is provided for the spent fuel assemblies, the new fuel assemblies and the associated radioactive systems from external events.

The fuel handling area is constructed so that the release of airborne radiation following any postulated design basis accident that could result in damage to the fuel assemblies or associated radioactive systems does not result in unacceptable site boundary radiation levels.

Mechanical equipment areas: The mechanical equipment located in radiological control areas of the auxiliary building are the normal residual heat removal pumps and heat exchangers, the spent fuel cooling system pumps and heat exchangers, the solid, liquid, and gaseous radwaste pumps, tanks, demineralizers and filters, the chemical and volume control pumps, and the heating, ventilating and air conditioning exhaust fans.

The mechanical equipment located in the clean areas of the auxiliary building are the heating, ventilating and air conditioning air handling units, associated equipment that service the main control room, instrumentation and control cabinet rooms, the battery rooms, the passive containment cooling

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system recirculation pumps and heating unit and the equipment associated with the air cooled chillers that are an integral part of the chilled water system.

Containment penetration areas: The auxiliary building contains all of the containment penetration areas for mechanical, electrical, and instrumentation and control penetrations. The auxiliary building provides separation of the radioactive piping penetration areas from the non-radioactive penetration areas and separation of the electrical and instrumentation and control penetration areas from the mechanical penetration areas. Also provided is separation of redundant divisions of instrumentation and control and electrical equipment.

Main steam and feedwater isolation valve compartment: The main steam and feedwater isolation valve compartment is contained within the auxiliary building. The auxiliary building provides an adequate venting area for the main steam and feedwater isolation valve compartment in the event of a postulated leak in either a main steam line or feedwater line.

Civil/Structural Features

The auxiliary building is a seismic Category I reinforced concrete structure. It shares a common basemat with the containment building and the shield building.

The auxiliary building wraps around approximately 70 percent of the circumference of the shield building. Floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

Equipment and Material Handling

A cask handling crane is located in the fuel handling area of the auxiliary building. The cask handling crane is designed to transport the spent fuel cask between the rail car bay, the cask loading pit, and the cask washdown pit. The crane rail length and rail stop limits the crane travel and thus precludes the movement of this crane in the near vicinity of the spent fuel pool. A fuel handling machine is provided to transfer new fuel from the rail car bay to the new fuel rack to the new fuel elevator. A bridge crane is provided in the rail car bay for handling the spent resin waste container fill station cover, the spent resin waste container, and the high activity filter transfer casks.

The major components of the fuel transfer system are located in the fuel transfer canal. The fuel transfer system is designed to transfer fuel assemblies between the fuel transfer canal located in the fuel handling area and the refueling cavity located in the containment building. The fuel transfer system consists of a transfer car/fuel container, a drive car, a traverse drive mechanism, an upending mechanism, the transfer tube, a quick opening hatch/blind flange on the containment side of the transfer tube and a valve on the fuel handling area side of the transfer tube.

A fuel handling machine is provided to move the spent fuel assemblies between the fuel transfer canal, the spent fuel pool and the cask loading pit. The fuel handling machine is a gantry crane, which has an extension to the south wall to reach the new fuel racks.

The high bay area is designed for a trailer to enter the building through a slide-up door. When used to transport a spent fuel cask to the fuel handling area, the trailer is positioned in the high bay area and the cask lifting rig is attached to the cask handling crane. When the cask is in the vertical position, it is disconnected from the trunnion and lifted to the operating deck through the equipment hatch and placed in the cask loading pit.

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1.2.5 Annex Building

Building Function

The annex building (Figures 1.2-17 through 1.2-19) provides the main personnel entrance to the power generation complex. It includes accessways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area. The building includes the health physics facilities for the control of entry to and exit from the radiological control area as well as personnel support facilities such as locker rooms. The annex building also includes office areas, a fire brigade locker room, and a radiation protection calibration lab. The building also contains the non-1E ac and dc electric power systems, the ancillary diesel generators and their fuel supply, other electrical equipment, the control support area, and various heating, ventilating and air conditioning systems. No safety-related equipment is located in the annex building.

The annex building includes the health physics facilities and provides personnel and equipment accessways to and from the containment building and the rest of the radiological control area via the auxiliary building. Provided are large, direct accessways to the upper and lower equipment hatches of the containment building for personnel access during outages and for large equipment entry and exit. The building includes a hot machine shop for servicing radiological control area equipment. The hot machine shop includes decontamination facilities including a portable decontamination system that may be used for decontamination operations throughout the nuclear island.

Civil/Structural Features

The seismic classification of the annex building can be found in Table 3.2-2. No protection against missile penetration is required. However, certain areas of the building, such as the hot machine shop, radiation protection calibration lab, and the control support area, are provided with shielding for protection against low level radiation from either internal sources or external sources under accident conditions. This is accomplished by either reinforced concrete walls, reinforced masonry walls, or lead-lined fire rated plywood/gypsum wall boards on metal studs. The control support area is designed so that it may be used as a technical support center (TSC) if desired.

The annex building is a combination of reinforced concrete structure and steel framed structure with insulated metal siding. Floor slabs are reinforced concrete supported by metal decking. Roof slabs are either reinforced concrete or rigid insulation supported on metal decking. Floors are designed to act as diaphragms to transmit horizontal loads to side wall bracing and to concrete shear walls. The building foundation is a reinforced concrete mat.

1.2.6 Diesel Generator Building

Building Function

The diesel generator building (Figure 1.2-20) houses two identical slide along diesel generators separated by a three hour fire wall. These generators provide backup power for plant operation in the event of disruption of normal power sources. No safety-related equipment is located in the diesel generator building.

Civil/Structural Features

The diesel generator building houses the two diesel generators and their associated heating, ventilating and air conditioning equipment, none of which are required for the safe shutdown of the plant. The seismic classification of the diesel generator building can be found in Table 3.2-2. The building is designed as a structure subject to wind loads in accordance with the Uniform Building Code.

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The building is a single story steel framed structure with insulated metal siding. The roof is composed of a metal deck supporting a concrete slab and serves as a horizontal diaphragm to transmit lateral loads to sidewall bracing and thereby to the foundation.

The foundation consists of a reinforced concrete mat. The diesel generators are skid-mounted and rest on vibration isolators supported directly from the mat.

1.2.7 Radwaste Building

Building Function

The radwaste building includes facilities for segregated storage of various categories of waste prior to processing, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building. Dedicated floor areas and trailer parking space for mobile processing systems is provided for the following:

- Contaminated laundry shipping for offsite processing
- Dry waste processing and packaging
- Hazardous/mixed waste shipping for offsite processing
- Chemical waste treatment
- Empty waste container receiving and storage
- Storage and loading packaged wastes for shipment

The radwaste building also provides for temporary storage of other categories of plant wastes.

Three liquid waste monitor tanks are located within the radwaste building. These tanks contain processed effluents which are ready for release to the environment.

Civil/Structural Features

The radwaste building general arrangement is shown on Figure 1.2-21. The seismic design of the radwaste building can be found in Table 3.2-2. The liquid radwaste processing areas are designed to contain any liquid spills. These provisions include a raised perimeter and floor drains that lead to the liquid radwaste system waste holdup tanks. The foundation for the entire building is a reinforced concrete mat on grade.

1.2.8 Turbine Building

Building Function

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It provides weather protection for the laydown and maintenance of major turbine/generator components. The turbine building also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

Civil/Structure Features

The turbine building, shown in Figures 1.2-22 through 1.2-29, consists of two sections, the first bay and the main area which houses the turbine. The first bay is immediately adjacent to the auxiliary building, and it consists of reinforced concrete walls and steel framing with reinforced concrete and steel grated floors. The main area is a steel framed building with reinforced concrete and steel grated floors. The first bay and the main area are two independent structures. The turbine building ground floor (structural mat) is a reinforced concrete slab shared by the first bay and main area structure. The seismic design of the turbine building can be found in Table 3.2-2.

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The turbine-generator is low-tuned by means of spring supports. The design consists of a reinforced concrete deck mounted on springs. The springs are supported on a structural steel framework that forms an integral part of the turbine building structural system. Lateral bracing serves to provide lateral support for the building as well as the turbine-generator support. The spring-supported concept isolates dynamically the turbine-generator deck from the remainder of the structure for operating frequencies, thus allowing for an integrated structure below the deck. This includes an integrated reinforced concrete foundation mat that supports both the turbine generator and the building. The condenser is attached rigidly to the low pressure turbine exhaust and is supported on springs. The foundation for the entire building is a reinforced concrete mat.

1.2.9 Combined License Information

This section contained no requirement for additional information.

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Figure 1.2-1
Not used

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Figure 1.2-2
Site Plan

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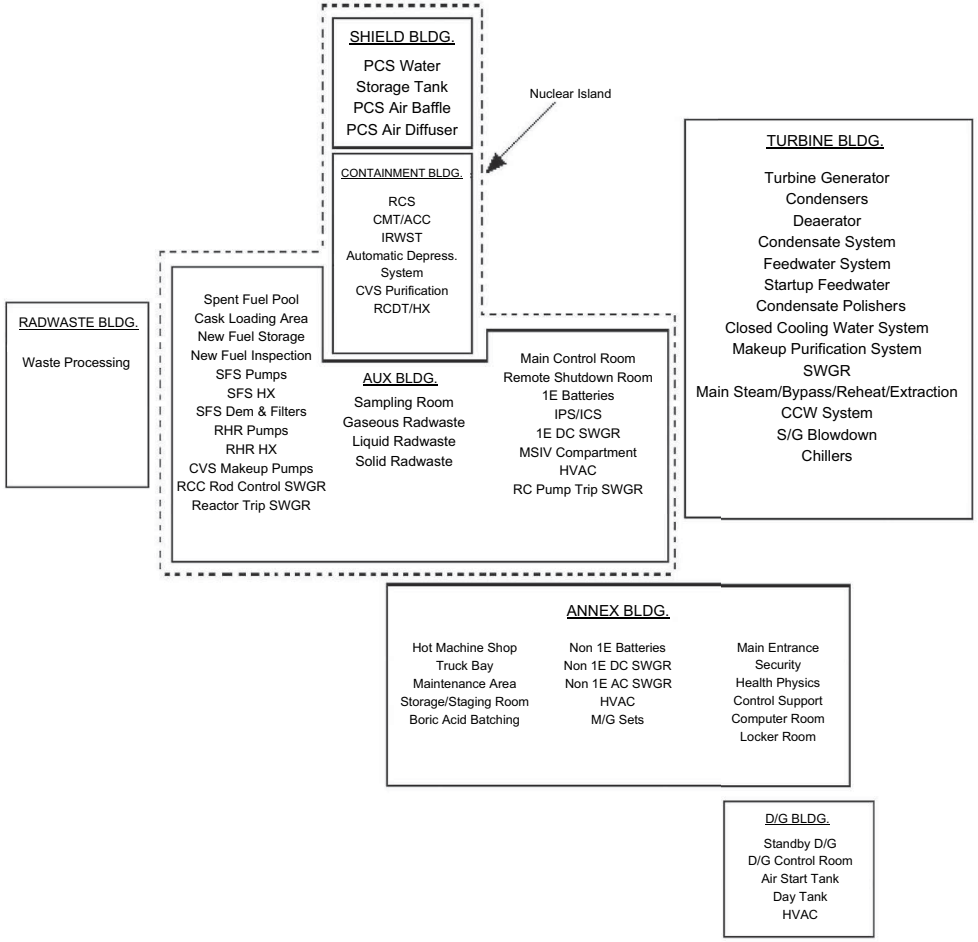

Figure 1.2-3
Functional Allocation of System Components of AP1000 Power Generation Complex

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Figure 1.2-4
Nuclear Island General Arrangement
Plan at Elevation 66'-6"

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Figure 1.2-5
Nuclear Island General Arrangement
Plan at Elevation 82'-6"

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Figure 1.2-6
Nuclear Island General Arrangement
Plan at Elevation 96'-6"

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Figure 1.2-7
Nuclear Island General Arrangement
Plan at Elevation 100'-0" & 107'-2"

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Figure 1.2-8
Nuclear Island General Arrangement
Plan at Elevation 117'-6" & 130'-0"

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Figure 1.2-9
Nuclear Island General Arrangement
Plan at Elevation 117'-6" with Equipment

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Figure 1.2-10
Nuclear Island General Arrangement
Plan at El. 135'-3"

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Figure 1.2-11
Nuclear Island General Arrangement
Plan at Elevation 153'-0" & 160'-6"

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Figure 1.2-12
Nuclear Island General Arrangement
Plan at Elevation 160'-6", 180'-0", & 329'-0"

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Figure 1.2-13
Nuclear Island General Arrangement
Section A-A

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Figure 1.2-14
Nuclear Island General Arrangement
Section A-A with Equipment

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Figure 1.2-15
Nuclear Island General Arrangement
Section B-B

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Figure 1.2-16
Nuclear Island General Arrangement
Section B-B with Equipment

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Figure 1.2-17
Annex Building General Arrangement
Section A-A

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(Note: This figure replaces Figure 1.2-18.)

Figure 1.2-201
Annex Building General Arrangement
Plan at Elevation 100'-0" & 107'-2"

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Figure 1.2-19
Annex Building General Arrangement
Plan at Elevation 117'-6" & 126'-3"

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Figure 1.2-20
Annex Building General Arrangement
Plan at Elevation 135'-3" through 183'-6"

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Figure 1.2-21
Diesel Generator Building General Arrangement
Plan at Elevation 100'-0" & Section A-A

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Figure 1.2-22
Radwaste Building General Arrangement
Plan at Elevation 100'-0"

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Figure 1.2-23
Turbine Building General Arrangement
Plan at Elevation 100'-0″

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Figure 1.2-24
Turbine Building General Arrangement
Plan at Elevation 120'-6"

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Figure 1.2-25
Turbine Building General Arrangement
Plan at Elevation 141'-3"

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Figure 1.2-26
Turbine Building General Arrangement
Plan at Elevation 170'-0"

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Figure 1.2-27
Turbine Building General Arrangement
Plan at Elevation 170'-0" with Equipment

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Figure 1.2-28
Turbine Building General Arrangement
Plan at Elevation 255'-3"

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Figure 1.2-29
Turbine Building General Arrangement
Section A-A

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Figure 1.2-30
Turbine Building General Arrangement
Section B-B

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1.3 Comparisons with Similar Facility Designs

A comparison of the major AP1000 design features and nominal parameters with the certified AP600 and a typical two-loop Westinghouse plant is provided in Table 1.3-1. The values provided for AP1000 are nominal and provided for comparison. Design parameter values for design certification are delineated in the sections referenced. The values provided in Table 1.3-1 for the reference AP600 and two-loop plants are typical. The two-loop plant parameters are represented by Waterford Unit 3.

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Table 1.3-1 (Sheet 1 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Plant design objective	1.2	60 yrs	60 yrs	40 yrs
NSSS power	4.0	3,415 MWt	1,940 MWt	3,410 MWt
Core power	4.0	3,400 MWt	1,933 MWt	3,390 MWt
Net electrical output	1.2	≥ 1,000 MWe	600 MWe	1,075 MWe
Reactor operating pressure	5.1	2,250 psia	2,250 psia	2,250 psia
Hot leg temp	5.1	610°F	600°F	611°F (Cycle 1) 603°F (current)
Steam generator design pressure	5.4	1200 psia	1200 psia	1100 psia
Main feedwater temp	10.3	440°F	435°F	445°F
Core	4.0			
Number fuel assem.		157	145	217
Active fuel length		168 in	144 in	150 in
Fuel assembly array		17 x 17	17 x 17	16 x 16
Fuel rod OD		0.374 in	0.374 in	0.382 in
Number control assem.		53	45	83
– Absorber material		Ag-In-Cd	Ag-In-Cd	B_4C/Ag-In-Cd
Number gray rod assem.		16	16	8 (part length)
– Absorber material		Tungsten	SS-304/Ag-In-Cd	Inconel 625/ B_4C
Avg linear power		5.72 kW/ft	4.11 kW/ft	5.34 kW/ft
Heat flux hot channel factor, FQ		2.60	2.60	2.35

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Table 1.3-1 (Sheet 2 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Reactor Vessel	5.3			
Vessel ID		159 in	157 in	172 in
Construction		forged rings	forged rings	welded plate
Number hot leg nozzles		2	2	2
– ID		31.0 in	31.0 in	42 in
Number cold leg nozzles		4	4	4
– ID		22.0 in	22.0 in	30 in
Number safety injection nozzles		2	2	0
Steam Generators	5.4.2			
Type		Vertical U-tube Recirc. design	Vertical U-tube Recirc. design	Vertical U-tube Recirc. design
Model		Delta-125	Delta-75	–
Number		2	2	2
Heat transfer area/SG		123,538 ft^2	75,180 ft^2	103,574 ft^2
Number tubes/SG		10,025	6,307	9,300
Tube material		I 690 TT	I 690 TT	I 600 TT
Separate startup feedwater nozzle		Yes	Yes	No
Reactor Coolant Pumps	5.4.1			
Type		canned	canned	shaft seal
Number		4	4	4
Rated HP		7,300 hp/pump	≤3,500 hp/pump	9,700 hp/pump
Estimated flow/loop		150,000 gpm	102,000 gpm	198,000 gpm

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Table 1.3-1 (Sheet 3 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Pressurizer	5.4.5			
Total volume		2,100 ft^3	1,600 ft^3	1,500 ft^3
Volume/MWt		0.618 ft^3/MWt	0.825 ft^3/MWt	0.440 ft^3/MWt
Safety valves #/size		2 – 6"x8"	2 – 6"x6"	3 – 6"
PORV #/size		no	no	no
PRT volume		no	no	2,400 ft^3
Auto depressurization		yes	yes	no
Turbine Island	10.2			
Turbine – # HP cylinder		1	1	1
# LP cylinders		3	2	3
Max blade length		52 in	47 in	40 in
Number reheat stages		2	1	1
Feedwater heating stages				
– # LP stages		4	4	5
– # HP stages		2	2	1
Deaerator		yes	yes	no
Main feedwater pumps		3 motor driven	2 motor driven	2 turbine driven
Condensate pumps		3	3	3
Condenser tube material		Ti	Ti	SS
Condensate polishing		0–33%	33%	0–100%

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Table 1.3-1 (Sheet 4 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Containment	6.2			
Type		Steel	Steel	Steel
Inside dia.		130 ft	130 ft	140 ft
Volume		2.06 E+06 ft^3	1.76 E+06 ft^3	2.677 E+06 ft^3
Volume/MWt		606 ft^3/MWt	910 ft^3/MWt	785 ft^3/MWt
Post accident cooling		Air and water on outside of steel containment vessel	Air and water on outside of steel containment vessel	Component cooling water cooled fan coolers
Safety Injection	6.3			
Accumulator – #/volume		2/2,020 ft^3	2/2,000 ft^3	4/2,250 ft^3
Core makeup tank – #/volume		2/2,500 ft^3	2/2,000 ft^3	no
High head pumps – #		none	none	3
– runout flow		–	–	380 gpm
– shutoff head		–	–	1,365 psi
Low head pumps – #		none	none	See RHR pumps
Refuel water storage tank – #		1	1	1
– location		in containment	in containment	ex-containment
– volume		590,000 gal	530,000 gal	475,000 gal
Boron inject tank #/vol		no	no	1/630 gal (batching) 2/11,800 gal (makeup)
Normal Residual Heat Removal (NRHR)	5.4.7			
Design pressure		900 psig	900 psig	650 psig
Normal RHR pumps – #/design flow		2/1,500 gpm per pump	2/1,000 gpm per pump	2/4,050 gpm per pump

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Table 1.3-1 (Sheet 5 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Cooling Water Systems	9.2			
Safety-related		no	no	yes
Component cooling water pumps		2	2	3
Service water pumps		2	2	none
Heat sink		Separate mechanical draft cooling towers	Separate mechanical draft cooling towers	Separate mechanical draft cooling towers
Startup/Auxiliary Feedwater	10.4			
Motor pumps – #/flow per pump/ safety-related		2/520 gpm/no	2/380 gpm/no	2/350 gpm/yes 1/900 gpm/no
Turbine pumps – #/flow		none/–	none/–	1/700 gpm
Passive RHR HX – #/heat removal/ safety-related		1/60 MW/Yes	1/42 MW/Yes	None/–/–
Chemical and Volume Control	9.3.6			
Purification/Letdown flow				
– normal		100 gpm	100 gpm	38 gpm
– max		100 gpm	100 gpm	126 gpm
Purification location		IRC	IRC	ORC
RCP seal injection/pump		none	none	5 – 8 gpm
Charging pumps		2 @ 100 gpm	2 @ 100 gpm	3 @ 44 gpm
– SI use		no	no	no
– safe shutdown use		no	no	yes
– continuous oper.		no	no	yes
Boron thermal regeneration		no	no	no
Boron recycle evaporator		no	no	no

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Table 1.3-1 (Sheet 6 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Instrumentation and Control	7.7			
Type I&C system		digital	digital	analog
Type control room		work station	work station	control boards
Electrical				
Diesels – #	8.3.1	2	2	2
– safety-related		no	no	yes
– capacity		4,000 kW	4,000 kW	4,400 kW
1E batteries – total capacity	8.3.2	14,400 amp-hr	28,800 amp-hr	3 x 2,320 amp-hr (@ 8 hour rate)

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1.4 Identification of Agents and Contractors

1.4.1 Applicant – Program Manager

Westinghouse Electric Company, LLC (Westinghouse), is responsible for the overall design and design certification of the AP1000 nuclear power plant. A significant portion of the AP1000 design is the same as the design of AP600. Westinghouse Electric Company was also responsible for the overall design and design certification of AP600.

Westinghouse has designed, developed, and manufactured nuclear facilities since the 1950s, beginning with the world's first large central station nuclear plant (Shippingport), which produced power from 1957.

Westinghouse has designed and delivered more than 100 commercial nuclear power plants with a combined electrical generating capacity in excess of 90,000 MW. The company's manufacturing facilities include the commercial nuclear fuel fabrication facility at Columbia, South Carolina; and nuclear component manufacturing facilities at Blairsville, Pennsylvania; and Newington, New Hampshire.

Westinghouse has been involved with advanced light water reactor plant design efforts for over fifteen years. Included is the development of the advanced, passive pressurized water reactors known as the AP600 and AP1000.

Westinghouse has substantial, proven experience, knowledge, and capability to design, manufacture and furnish technical assistance for the installation, startup and service of nuclear power plants.

1.4.2 Not Used

1.4.3 Combined License Information

This section contained no requirement for additional information.

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1.5 Requirements for Further Technical Information

Introduction

Tests were conducted during the AP600 Conceptual Design Program (1986 through 1989) to provide input for plant design and to demonstrate the feasibility of unique design features. Tests for the AP600 design certification and design program were devised to provide input for the final safety analyses, to verify the safety analysis models (computer codes), and to provide data for final design and verification of plant components. An AP1000 specific Phenomena Identification and Ranking Table (PIRT) and scaling analysis (Reference 25) and a review of safety analysis evaluations of AP1000 (Chapter 15 of this DCD) show that AP600 and AP1000 exhibit a similar range of conditions for the events analyzed. This provides justification that the database of test information generated during the AP600 Conceptual Design Program is sufficient to meet the requirements of 10 CFR Part 52 for AP1000. Table 1.5-1is a list of the AP600 tests and AP1000 evaluations with references to test and evaluation documentation. Note that Reference 25 reviews each of the AP600 tests described and assesses their applicability to AP1000. The evaluations of Reference 25 show that the AP600 tests are sufficient to support AP1000 safety analysis.

The AP600 tests related to the plant safety functions were selected based on the plant features that are different from current PWRs and where directly applicable experimental data are not available. The tests simulate plant features as required to demonstrate the phenomena being examined. To validate the computer models, these experiments are modeled using the same computer codes used for plant analyses.

Testing of some plant component designs is required to verify their reliability and manufacturability. Other component tests provide data for design optimization. The completed component design tests are described below.

1.5.1 AP600 Safety-Related Tests

The AP600 safety-related experiments are designed to meet several goals:

- Provide input for safety analysis
- Provide data on the passive safeguards systems to validate the safety analysis methods and computer codes
- Assess the design margin in the passive safety system performance

To accomplish these goals, the AP600 test program utilizes the available data from the NRC and industry light water reactor safety research programs as well as specific tests which address the uniqueness of the AP600. The AP600 safety-related test program utilizes component experiments and integral tests to determine the transient behavior of the AP600 safety system components such that computer models can be developed and verified.

The range of plant conditions for design basis accidents and transients, and the new features of the AP600 design were evaluated against current Westinghouse designs and safety-related data available in the literature (NUREG-1230). The results of this assessment were used to determine the data needs, and to define the experiments to support the AP600 safety analysis. Based upon the experiments performed for AP600 and the AP1000 range of plant conditions for design basis accidents and transients, the tests were shown to be sufficient to support AP1000 safety analysis as well.

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1.5.1.1 Large-Break LOCA

For large-break LOCA safety analysis, the relevant new features of the AP600 were the core makeup tanks (CMTs), which drain by gravity, and the use of hermetically sealed, high inertia centrifugal canned-motor reactor coolant pumps (RCPs). Two-phase pump flow data exists for Westinghouse designed pumps and others (NUREG-1230) that can be used to characterize the AP600 pumps. The core makeup tank is unique to the AP600 and AP1000 design. A specific AP600 test was conducted for this component. In addition, a test of passive safety injection system check valve flow vs. ΔP with low differential pressure has been completed. The evaluation of Reference 25 shows that these tests are applicable to AP1000.

Core Makeup Tank Performance Test

The purpose of this experiment is to verify the natural circulation and draining behavior of the core makeup tank over a full range of flowrates, pressures and temperatures, and to provide data to support the design and operation of the tank level indication which acts as a control for the automatic depressurization system (ADS). When actuated, the CMT adds water mass to the reactor coolant system (RCS) by natural circulation when the cold leg contains hot water. The water in the core makeup tank drains by gravity head into the RCS when steam is provided from the cold leg to the top of the core makeup tank. This steam replaces the water drained from the core makeup tank. Some of the steam condenses upon entry into the core makeup tank and can affect the tank draining performance. The objective of the test is to verify that the tank will drain as predicted.

A one-eighth diameter and one-half height scale core makeup tank was constructed and instrumented to obtain the condensation rates within the tank to verify the computer model. The core makeup tank water delivery was examined.

Passive Safety Injection System Check Valve Tests

The AP1000 uses check valves to isolate passive systems from the reactor coolant system. Tests have been performed in the AP600 test program on these check valves to demonstrate their operability.

Tests were conducted to measure check valve pressure drop from very low flow to full flow conditions. Detailed data on initial valve opening, valve disk behavior and flow versus differential pressure were obtained for individual check valves as well as for valves installed in series.

Initial check valve low differential opening tests have determined the characteristic valve flow under the expected gravity drain conditions. A review of existing utility information has been conducted to assess check valve performance under conditions similar to those which would be experienced by the gravity drain check valves.

1.5.1.2 Small-Break LOCA

For small-break LOCA safety analysis, the relevant new features of the AP600 and the AP1000 designs are the core makeup tank, and the automatic depressurization system which depressurizes the primary system to near containment pressure.

The core makeup tank provides injection flow to the reactor vessel at any reactor coolant system pressure. The core makeup tank tests described above duplicated small-break conditions as well as the large-break conditions. The automatic depressurization system provides controlled venting of the reactor coolant system to reduce pressure to allow transition to gravity driven injection from the IRWST. Full-scale tests were conducted in the AP600 test program to obtain data on the performance of the automatic depressurization system. As shown by the AP1000 evaluations, these tests also support AP1000 safety analysis.

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Automatic Depressurization System Hydraulic Tests

The purpose of these tests is to simulate the automatic depressurization system, to confirm the capacity of the automatic depressurization system valves and spargers, and to determine the dynamic effects on the IRWST structure.

A pressurized, heated water/steam source was used to simulate the water/steam flow rate from the AP600 reactor coolant system during various stages of the automatic depressurization system blowdown. Two test phases were conducted. Phase A consisted of steam only blowdowns at bounding volumetric flowrates. The flow is piped to a full sized sparger submerged in a quench tank simulating the IRWST. The Phase B1 portion of the test included steam/water blowdowns at bounding mass flowrates through a simulation of one of the two ADS stage 1,2,3 flowpaths. Instrumentation to measure water and steam flow rate, and IRWST dynamic loads was installed. Sparger behavior was obtained from ambient to fully saturated IRWST water temperatures.

1.5.1.3 Containment Cooling

Tests to characterize the heat removal capabilities of the AP600 containment design were performed to provide the database for the containment cooling models. These include the following:

- Study of water film behavior and wetting of a steel plate simulating the containment exterior surface
- Heated plate tests to examine the evaporating heat transfer of water from the steel surface of the containment and heat transfer with only air cooling
- Containment external cooling air flow path pressure drop tests to characterize the hydraulic losses
- Steam condensation heat transfer experiments on a flat cool surface at different angles of inclination to simulate the condensation on the inside of the containment in the presence of noncondensible gases

In addition, tests were performed to examine the integrated behavior of the steam condensation on the inside, and the evaporative film cooling and air cooling on the outside of a pressure vessel. The cylindrical vessel used for this integral test was 3 feet wide and 24 feet high. These experiments included transient and steady-state tests which have been used as the basis for the containment analyses. The limits of coolability and the effect of cold weather conditions were also examined.

As shown by the AP1000 evaluations, these tests also support AP1000 safety analysis.

Integral Containment Cooling Tests

This test examines the combined effect of natural convection and condensation on the interior of the containment while the exterior is cooled by film evaporation and air flow. This test demonstrated the operation of the passive containment cooling system over a range of operating conditions, including operation at low environmental temperatures. This test, in conjunction with completed conceptual design phase testing and the large scale containment test described below, characterize the passive containment cooling system design and performance.

Passive Containment Cooling System Heat Transfer Test

A one-eighth scale steel containment structure with external water film and natural circulation air cooling and modeled containment internal compartments was constructed.

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This test accurately models both the containment dome and side wall heat transfer areas. It complements the integral containment experiment which simulates the side wall condensation and evaporating film heat transfer. This test was used to verify the containment analysis analytical methods.

Instrumentation measured the condensation heat flux distribution, the resulting heat transfer coefficients, the air/steam mass ratios, and the resulting liquid film evaporation rates. Both the current integral containment cooling test and this larger scale containment test have been modeled to verify the Chapter 15 analysis computer code and to demonstrate the scalability of the results.

Passive Containment Cooling System Water Distribution Test

A passive containment cooling system water distribution experiment was performed to examine and finalize the AP600 containment water distribution. The results provide input into the containment safety analysis computer codes for water coverage of the containment shell.

The test was performed on a full-scale 1/8th sector of the containment dome. The AP600 water supply/distribution arrangement was modeled. Tests were conducted to demonstrate and measure the water spreading from the top center of the dome to the outer edges. Tests have been conducted to verify the performance of the water distribution system design. Tests were conducted with the surface coated with the prototypic AP600 containment coating. Measurements of water film velocities and film thickness variation as a function of flow rate and radial distance on the dome were obtained.

Passive Containment Cooling System Wind Tunnel Tests

Containment cooling relies on natural circulation of air to enhance evaporative cooling of the containment shell during a design basis event. Wind tunnel tests were performed to demonstrate that wind does not adversely affect natural circulation air cooling through the shield building and around the containment shell.

An approximately 1/100-scale model of the AP600 plant, including the adjacent buildings and cooling tower structure, was constructed and instrumented with pressure taps. The model was placed in a boundary layer wind tunnel and tested at different wind directions. The results were used to design the shield building air inlet and exhaust arrangement and to determine the loads on the air baffle. Variations in site layout and topography have been addressed using an approximately 1/800-scale model of the site buildings and local topography.

Tests were also conducted in a larger, higher speed wind tunnel on an approximately 1/30-scale model. These tests were conducted to confirm that the early test results conservatively represented those expected at full scale Reynolds numbers and to obtain better estimates of the baffle loads in the presence of a cooling tower.

1.5.1.4 Non-LOCA Transient Analysis

The non-LOCA accidents are evaluated using the same transient analysis methods used on existing Westinghouse PWR designs. Passive core cooling system computer models have been developed and added to the transient analysis codes. These models consist of a core makeup tank model and a passive residual heat removal (PRHR) heat exchanger model. As shown by the AP1000 evaluations, these tests also support AP1000 safety analysis.

Passive Residual Heat Removal Heat Exchanger Performance Test

The PRHR heat exchanger is located in the IRWST. This heat exchanger, which is connected directly to the reactor coolant system, transfers core decay heat and sensible heat energy to the IRWST water and depends only on natural circulation driving forces.

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The passive residual heat removal heat exchanger test determined the heat transfer characteristics of the PRHR heat exchanger and the mixing characteristics in the IRWST. These results confirm the heat exchanger size and configuration.

The test facility consisted of three full-length heat exchanger tubes placed vertically in a cylindrical tank filled with water and baffled to simulate the AP600 IRWST. Water at prototypic natural circulation and forced flow rates was run through the heat exchanger tubes at prototypic system pressure and temperatures. Data was taken with IRWST water cold to saturation temperature to define the PRHR heat transfer correlation. Tests were also conducted using a baffle to simulate the effect of other rows of tubes have on heat exchanger thermal performance and tank mixing.

Departure from Nucleate Boiling Test

Due to the shorter coastdown of the AP600 canned motor reactor coolant pumps, the flow rates at the time of minimum DNBR are somewhat below previously correlated flow rates. DNB testing was performed to extend the DNB correlation to these lower flows.

These critical heat flux tests were conducted using a 5x5, full length heated rod bundle with non-uniform radial and axial heating distributions.

1.5.1.5 Integral Systems Testing

In the AP600, the water injected into the reactor coolant system comes from the CMTs, accumulators, and the IRWST. Two integral systems tests were conducted in the AP600 test program, a low-pressure scaled test and a full-height, full-pressure test. In addition, the NRC conducted tests in the low-pressure scaled test facility (Reference 27). As shown in the AP1000 evaluations (Reference 25), these three test programs are sufficient to support AP1000 safety analysis.

Low-Pressure Integral Systems Test

The primary purpose of this experiment was to examine the operation of the long-term makeup path from the in-containment refueling water storage tank. In addition, analysis of this experiment demonstrates water flow through the core to limit the long-term concentration of boric acid. The facility is capable of simulating high-pressure system responses.

The test models the reactor vessel, steam generators, reactor coolant pumps, in-containment refueling water storage tank, the automatic depressurization system vent paths, the lower containment, and the connecting piping. The hot legs and cold legs are modeled as are the core makeup tanks, PRHR heat exchangers, accumulators, and pressurizer.

Water is the working fluid and the core is simulated with electric heater rods scaled to match the core power levels consistent with the test scaling approach. Tests were performed to simulate various small-break LOCAs with different break locations, break sizes, with and without nonsafety systems operating. The analysis methods in Chapter 15 were compared to the test.

Full-Height, Full-Pressure Integral Systems Test

A test was performed to provide data on system performance at high pressure. This test facility is configured as a full-height, full-pressure integral test with AP600 features including two loops with one hot leg and two cold legs per loop, two core makeup tanks, two accumulators, a PRHR heat exchanger and an automatic depressurization system. The facility includes a scaled reactor vessel, steam generators, pressurizer and reactor coolant pumps. Water is the working fluid and the core is simulated with electric heater rods.

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Tests were performed simulating small break LOCAs, steam generator tube ruptures and a steam line break transient. The analysis methods in Chapter 15 were compared to the test results.

1.5.2 AP600 Component Design Tests

The component design tests will provide a larger database for design optimization during the detailed design of the plant. Tests on selected plant components were performed to confirm their reliability or that materials and fabrication methods meet ASME requirements. These tests are also applicable to the AP1000 design and analysis.

Incore Instrumentation System Tests

Systems similar to the AP600 and AP1000 top mounted fixed incore detector (FID) instrumentation have been demonstrated in operating plants. A test was performed to demonstrate that the system will not be susceptible to electro-magnetic interference (EMI) from the nearby control rod drive mechanisms.

The electro-magnetic interference test was performed by mocking up instrument cables, bringing them into close proximity with an operating control rod drive mechanism, and measuring the resulting noise induced on simulated flux signals.

Reactor Coolant Pump/Steam Generator Airflow Test

The airflow test was performed to identify effects on the pump performance due to non-uniform channel head flow distribution, pressure losses of the channel head nozzle dams and pump suction nozzle, and possible vortices in the channel head induced by the pump impeller rotation.

The air test facility was constructed as an approximate one-half scale mockup of the outlet half of the channel head, the two pump suction nozzles, and two pump impellers and diffusers. The channel head tube sheet was constructed from clear plastic to allow smoke flow stream patterns to be seen.

The results of the test showed no flow anomalies or vortices in the channel head were induced by the dual impellers.

Reactor Coolant Pump High Inertia Rotor/Bearing Tests

In the AP600 a rotor, manufactured of depleted uranium clad with stainless steel, was incorporated into the hermetically sealed, high inertia centrifugal canned motor reactor coolant pump to provide the required flow coastdown performance for loss of flow transients.

Tests have been performed to verify manufacturability of the rotor, to determine friction and drag losses, to verify the operating performance of the pivoted-pad bearings, and to develop a detailed quantitative knowledge of the factors influencing bearing design and performance.

Tests were performed to verify the drag losses of the rotor with the journal bearing located on the pump shaft. Approximately 1000 cycles of starts and stops were also performed as a life test to demonstrate that the rotor will maintain its dimensional stability. These tests were performed on the specially-constructed, full-scale rotor/bearing test rig.

The results of these tests were used to check analytical methods used in the design of the AP1000 reactor coolant pump.

1.5.3 Combined License Information

This section contained no requirement for additional information.

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1.5.4 References

1. WCAP-14217, “Core Makeup Tank Final Data Report,” (Proprietary), WCAP-14218 (Nonproprietary), November 1994.

2. WCAP-13286, “AP600 Passive Core Cooling System Check Valve Test Final Report,” (Proprietary), WCAP-13287 (Nonproprietary), April 1992.

3. WCAP-13891, “AP600 Automatic Depressurization System Phase A Test Data Report,” (Proprietary), WCAP-14095 (Nonproprietary), May 1994.

4. WCAP-14324, “Final Data Report for ADS Phase B1 Tests,” (Proprietary), WCAP-14325 (Nonproprietary), April 1995.

5. WCAP-14134, “AP600 Passive Containment Cooling System Integral Small-Scale Tests Final Report,” (Proprietary), WCAP-14137 (Nonproprietary), August 1994.

6. WCAP-13566, “AP600 1/8th Large Scale Passive Containment Cooling System Heat Transfer Baseline Data Report,” (Proprietary), Revision 1, WCAP-13567 (Nonproprietary), Revision 0, December 1992.

7. WCAP-14135, “Final Test Report for PCS Large Scale Phase 2 and Phase 3 Tests,” (Proprietary), WCAP-14138 (Nonproprietary), Revision 3, September 1998.

8. WCAP-13353, “Passive Containment Cooling System Water Distribution Phase 1 Test Data Report,” (Proprietary), WCAP-13354 (Nonproprietary), April 1992.

9. WCAP-13296, “PCS Water Distribution Test Phase II Test Data Report,” (Proprietary), WCAP-13297 (Nonproprietary), March 1992.

10. WCAP-13960, “PCS Water Distribution Phase 3 Test Data Report,” (Proprietary), WCAP-13961 (Nonproprietary), December 1993.

11. WCAP-13294, “Phase I Wind Tunnel Testing for the Westinghouse AP600 Reactor,” (Proprietary), WCAP-13295 (Nonproprietary), April 1992.

12. WCAP-13323, “Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor,” (Proprietary), WCAP-13324 (Nonproprietary), August 1992.

13. WCAP-14068, “Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor,” (Proprietary), WCAP-14084 (Nonproprietary), May 1994.

14. WCAP-14169, “Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report,” (Proprietary), WCAP-14170 (Nonproprietary), September 1994.

15. WCAP-14091, “Phase IVb Wind Tunnel Testing for the Westinghouse AP600 Reactor,” (Proprietary), WCAP-14092 (Nonproprietary), July 1994.

16. WCAP-12980, “AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report,” (Proprietary), WCAP-13573 (Nonproprietary), Revision 3, April 1997.

17. WCAP-14371, “AP600 Low Flow Critical Heat Flux (CHF) Test Data Analysis,” (Proprietary), WCAP-14372 (Nonproprietary), May 1995.

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18. WCAP-14252, “AP600 Low Pressure 1/4 Height Integral Systems Tests - Final Data Report,” (Proprietary), WCAP-14253 (Nonproprietary), Revision 1, November 1998.

19. WCAP-14309, “AP600 Design Certification Program, SPES-2 Tests Final Data Report,” (Proprietary), WCAP-14310 (Nonproprietary), Revision 2, May 1997.

20. WCAP-12648, “AP600 Incore Instrumentation System Electromagnetic Interference Test Report,” (Proprietary), WCAP-13322 (Nonproprietary), Revision 1, April 1992.

21. WCAP-13298, “RCP Air Model Test Report,” (Proprietary), WCAP-13299 (Nonproprietary), August 1991.

22. WCAP-12668, “AP600 High Inertia Rotor Testing - Phase I, Test Report,” (Proprietary), WCAP-13321 (Nonproprietary), March 1990.

23. WCAP-13319, “AP600 High Inertia Rotor Testing - Phase 2 Report,” (Proprietary), WCAP-13320 (Nonproprietary), August 1991.

24. WCAP-13758, “High Inertia Rotor Test Phase 3 Report,” (Proprietary), WCAP-13759 (Nonproprietary), June 1993.

25. WCAP-15613, “AP1000 PIRT and Scaling Assessment,” (Proprietary), WCAP-15706 (Nonproprietary), March 2001.

26. Not used.

27. NUREG/CR-6641, “Final Report of NRC Research Conducted at Oregon State University,” August 1999.

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Table 1.5-1
AP600 Design Tests and AP1000 Evaluation

Test	Reference
LOCA Mitigation	
Core Makeup Tank Performance Test	(1)
Passive Safety Injection System Check Valve Test	(2)
Automatic Depressurization System Hydraulic Test	(3), (4)
Containment Cooling	
Integral Containment Cooling Test	(5)
Passive Containment Cooling System Heat Transfer Test	(6), (7)
Passive Containment Cooling System Water Distribution Test	(8), (9), (10)
Passive Containment Cooling System Wind Tunnel Test	(11), (12), (13), (14), (15)
Non-LOCA Transients	
Passive Residual Heat Removal Heat Exchanger Performance Test	(16)
Departure from Nucleate Boiling Test	(17)
Integral Systems Tests	
Low Pressure Integral Systems Test	(18)
Full Height Full Pressure Integral Systems Test	(19)
NRC Low Pressure Integral Systems Test	(27)
Component Design Tests	
Incore Instrumentation System Test	(20)
Reactor Coolant Pump/Steam Generator Airflow Test	(21)
Reactor Coolant Pump High Inertia Rotor/Bearing Test	(22), (23), (24)
AP1000 Evaluation	
AP1000 PIRT and Scaling Assessment	(25)

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1.6 Material Referenced

The AP1000 Design Control Document references various Westinghouse technical support documents; these documents are listed by DCD section in Table 1.6-1.

Table 1.6-201 provides a list of the various technical documents incorporated by reference in the FSAR in addition to those technical documents incorporated by reference in the AP1000 DCD. Note that the incorporation by reference may be all, or only a portion, of the identified document. The material incorporated by reference is specifically identified in the text of the section in which the document is referenced.

Table 1.6-203 provides a list of documents which include a methodology that is required by the Technical Specifications. Changes to these methodologies require a license amendment to change the Technical Specification reference.

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Table 1.6-1 (Sheet 1 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
1.9	WCAP-15993	Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria, Revision 2, June 2017
	WCAP-15799	AP1000 Compliance with SRP Acceptance Criteria, Revision 1, August 2003
	WCAP-15800	Operational Assessment for AP1000, Revision 3, July 2004
	WCAP-15992	AP1000 Adverse Systems Interactions Evaluation Report, Revision 1, February 2003
	WCAP-15776	Safety Criteria for the AP1000 Instrumentation and Control Systems, Revision 1, July 2019
1A	WCAP-8577	The Application of Pre-Heat Temperature After Welding of Pressure Vessel Steels, September 1975
	WCAP-16650-P (P) WCAP-16650-NP	Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines, Revision 0, February 2007
3.3	WCAP-13323 (P) WCAP-13324	Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor, August 1992
	WCAP-14068 (P) WCAP-14084	Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor, May 1994
	WCAP-14169 (P) WCAP-14170	Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report, September 1994
	WCAP-13294-P (P) WCAP-13295-NP	Phase I Wind Tunnel Testing for the Westinghouse AP600 Reactor, April 1992
3.6	WCAP-8077 (P) WCAP-8078	Ice Condenser Containment Pressure Transient Analysis Methods, March 1973
	WCAP-8708 (P) WCAP-8709-A	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-8252	Documentation of Selected Westinghouse Structural Analysis Computer Codes, Revision 1, May 1977
3.7	WCAP-7921-AR	Damping Values of Nuclear Power Plant Components, May 1974
	WCAP-9903 (P)	Justification of the Westinghouse Equivalent Static Analysis Method for Seismic Qualification of Nuclear Power Plant Auxiliary Mechanical Equipment, August 1980

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Table 1.6-1 (Sheet 2 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
3.8	[*APP-GW-GLR-602*	*AP1000 Shield Building Design Details for Select Wall and RC/SC Connections, Revision 5, October 2014, Westinghouse Electric Company LLC*]*
3.9	WCAP-8708-P-A (P) Volumes 1 and 2 WCAP-8709-A Volumes 1 and 2	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-17983-P (P) WCAP-17983-NP	Comprehensive Vibration Assessment Program (CVAP) Measurement and Inspection Programs for the AP1000 Plant, March 2015
	WCAP-17984-P (P) WCAP-17984-NP	Comprehensive Vibration Assessment Program (CVAP) Vibration Analysis Program for the AP1000 Plant, July 2015

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Table 1.6-1 (Sheet 3 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
3H	[*APP-GW-GLR-602*	*AP1000 Shield Building Design Details for Select Wall and RC/SC Connections, Revision 5, October 2014, Westinghouse Electric Company LLC*]*
4.1	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985, and VANTAGE 5H Fuel Assembly, Addendum 2A, February 1989
	WCAP-12610-P-A (P) WCAP-14342-A	VANTAGE+ Fuel Assembly Reference Core Report, April 1995
	[*WCAP-12488-A (P)* *WCAP-14204-A*]*	*Fuel Criteria Evaluation Process, October 1994*]*
4.2	[*WCAP-12488-A (P)* *WCAP-14204-A*]*	*Fuel Criteria Evaluation Process, October 1994*]*
	WCAP-10125-P-A (P) WCAP-10126-NP-A	Extended Burnup Evaluation of Westinghouse Fuel, December 1985
	WCAP-9179 (P) WCAP-9224	Properties of Fuel and Core Component Materials, July 1978
	WCAP-12610-P-A (P) WCAP-14342-A	VANTAGE+ Fuel Assembly Reference Core Report, June 1990/ April 1995
	WCAP-10851-P-A (P) WCAP-11873-A	Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations, August 1988
	WCAP-13589-A (P) WCAP-14297-A	Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel, March 1995
	WCAP-10021-P-A (P) WCAP-10377-NP-A	Westinghouse Wet Annular Burnable Absorber Evaluation Report, Revision 1, October 1983
	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985
	WCAP-8691 (P) WCAP-8692	Fuel Rod Bow Evaluation, Revision 1, July 1979
	WCAP-8236 (P) WCAP-8288	Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident, December 1973

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Table 1.6-1 (Sheet 4 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
4.2	WCAP-9401-P-A (P) WCAP-9402-A	Verification, Testing, and Analysis of the 17x17 Optimized Fuel Assembly, August 1981
	WCAP-9283	Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events, March 1978
	WCAP-15063-P-A (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), Revision 1, July 2000
	WCAP-8377 (P)	Revised Clad Flattening Model, July 1974
	WCAP-16652-NP	AP1000 Core & Fuel Design Technical Report, Revision 0, October 2006
	WCAP-16943-P-A (P) WCAP-16943-NP-A	Enhanced GRCA Rodlet Design, September 2012
4.3	WCAP-9272-P-A (P) WCAP-9273-NP-A	Westinghouse Reload Safety Evaluation Methodology, July 1985
	[*WCAP-12488-P-A (P)* [*WCAP-14204-A*]*	*Fuel Criteria Evaluation Process, October 1994*]*
	WCAP-12472-P-A (P) WCAP-12473-A	BEACON: Core Monitoring and Operations Support System, August 1994; Addendum 1, May 1996; Addendum 2, March 2001
	WCAP-12472-P-A (P) WCAP-12472-NP-A	BEACON: Core Monitoring and Operations Support System, Addendum 4, September 2012
	WCAP-8218-P-A (P) WCAP-8219-A	Fuel Densification Experimental Results and Model for Reactor Application, March 1975
	WCAP-8359	Effects of Fuel Densification Power Spikes on Clad Thermal Transients, July 1974
	WCAP-7811	Power Distribution Control of Westinghouse Pressurized Water Reactors, December 1971
	WCAP-8385 (P) WCAP-8403	Power Distribution Control and Load Following Procedures, September 1974
	WCAP-17661-P-A (P) WCAP-17661-NP-A	Improved RAOC and CAOC F_Q Surveillance Technical Specifications, Revision 1, February 2019
	WCAP-10216-P-A (P) WCAP-10217-A	Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification, Revision 1A, February 1994
	WCAP-7912-P-A (P) WCAP-7912-A	Power Peaking Factors, January 1975
	WCAP-7048-P-A (P) WCAP-7757-A	The PANDA Code, February 1975
	WCAP-7213-A (P) WCAP-7758-A	The TURTLE 24.0 Diffusion Depletion Code, February 1975

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Table 1.6-1 (Sheet 5 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
4.3	WCAP-6073 (P)	LASER – A Depletion Program for Lattice Calculations Based on MUFT and THERMOS, April 1966
	WCAP-2048 (P)	The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements, July 1962
	WCAP-11596-P-A (P) WCAP-11597-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, June 1988
	WCAP-10841 (P) WCAP-10842	Qualification of the PHOENIX/POLCA Nuclear Design and Analysis Program for Boiling Water Reactors, June 1985
	WCAP-10965-P-A (P) WCAP-10966-A	ANC: A Westinghouse Advanced Nodal Computer Code, September 1986
	WCAP-13589-A (P) WCAP-14297-A	Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel, March 1995
	WCAP-13524-P-A (P) WCAP-14952-NP-A	APOLLO - A One Dimensional Neutron Theory Program, Revision 1-A, September 1997
	WCAP-16652-NP	AP1000 Core & Fuel Design Technical Report, Revision 0, October 2006

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Table 1.6-1 (Sheet 6 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
4.3	WCAP-16045-P-A (P) WCAP-16045-NP-A	Qualification of the Two-Dimensional Transport Code PARAGON, August 2004
	WCAP-16943-P-A (P) WCAP-16943-NP-A	Enhanced GRCA Rodlet Design, September 2012
	WCAP-16045-P-A Addendum 1-A (P) WCAP-16045-NP-A Addendum 1-A	Qualification of the NEXUS Nuclear Data Methodology, August 2007
	WCAP-10965-P-A Addendum 2-A (P) WCAP-10966-A Addendum 2-A	Qualification of the New Pin Power Recovery Methodology, September 2010
4.4	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985
	[*WCAP-12488-A (P)*	*Westinghouse Fuel Criteria Evaluation Process, October 1994*]*
	WCAP-8691 (P) WCAP-8692	Fuel Rod Bow Evaluation, Revision 1, July 1979
	WCAP-2923	In-Pile Measurement of UO_2 Thermal Conductivity, March 1966
	WCAP-10851-P-A (P) WCAP-11873-A	Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations, August 1988
	WCAP-7912-P-A (P) WCAP-7912-A	Power Peaking Factors, January 1975
	WCAP-15025-P-A (P) WCAP-15026-NP-A	Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, April 1999
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, October 1999
	WCAP-15063-P-A (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), Revision 1, July 2000
	WCAP-16652-NP	AP1000 Core & Fuel Design Technical Report, Revision 0, October 2006
	WCAP-14565-P-A, Addendum 1-A (P)	Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code, August 2004
	WCAP-14565-P-A, Addendum 2-P-A (P)	Addendum 2 to WCAP-14565-P-A Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications, Revision 0, April 2008
5.2	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984

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Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
5.3	WCAP-15557	Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology, August 2000
	WCAP-14040-NP-A	Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, Revision 4, May 2004
6.2	WCAP-8077 (P) WCAP-8078	Ice Condenser Containment Pressure Transient Analysis Methods, March 1973
	WCAP-10325-P-A (P) WCAP-10326-A	Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version, May 1983
	WCAP-8822 (P) WCAP-8860 WCAP-8822-P-S1 (P) WCAP-8822-S2-P-A (P)	Mass and Energy Releases Following A Steam Line Rupture, September 1976 Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture, January 1985 Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Subatmospheric Containment Designs, September 1986
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-15846 (P) WCAP-15862	WGOTHIC Application to AP600 and AP1000, Revision 5, September 2016
	WCAP-15846-P Addendum 1 WCAP-15846-NP Addendum 1	Effective Thermal Conductivity Model of Inorganic Zinc Coating for Application to AP1000, Revision 0, October 2013
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997

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Table 1.6-1 (Sheet 8 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
6.2	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
6.3	WCAP-13594 (P) WCAP-13662 (NP)	FMEA of Advanced Passive Plant Protection System, Revision 1, June 1998
7.1	WCAP-15775	AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report, Revision 8, May 2017
	[*WCAP-16096-P-A* *WCAP-16096-NP-A*	*Software Program Manual for Common Q™ Systems, Revision 4, February 2013*(1) *(as modified by the SPM alternatives in WCAP-15927, Revision 8)*]*
	[*WCAP-16097-P-A* *WCAP-16097-NP-A*	*Common Qualified Platform Topical Report, Revision 3, February 2013 (as modified by the Topical Report alternatives in WCAP-15927, Revision 8)*]*
	WCAP-15776	Safety Criteria for the AP1000 Instrumentation and Control Systems, Revision 1, July 2019
	WCAP-16674-P WCAP-16674-NP	AP1000 I&C Data Communication and Manual Control of Safety Systems and Components, Revision 9, June 2019
	WCAP-16675-P WCAP-16675-NP	AP1000 Protection and Safety Monitoring System Architecture Technical Report, Revision 10, September 2020

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Table 1.6-1 (Sheet 9 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
7.1	[*WCAP-15927-P* *WCAP-15927-NP*	*Design Process for AP1000 Common Q Safety Systems, Revision 8, September 2020*]*
		Westinghouse Electric Company Quality Management System (QMS), (Non-Proprietary), Revision 7, October 2013
	APP-GW-J0R-012	AP1000 Protection and Safety Monitoring System Computer Security Plan, Revision 5, September 2020
7.2	WCAP-15776	Safety Criteria for the AP1000 Instrumentation and Control Systems, Revision 1, July 2019
7.3	WCAP-15776	Safety Criteria for the AP1000 Instrumentation and Control Systems, Revision 1, July 2019
7.7	WCAP-17184-P	AP1000™ Diverse Actuation System Planning and Functional Design Summary Technical Report, Revision 13, May 2019
	WCAP-15775	AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report, Revision 8, May 2017
	WCAP-17179-P WCAP-17179-NP	AP1000 Component Interface Module Technical Report, Revision 6, April 2016
10.2	WCAP-16650-P (P) WCAP-16650-NP	Analysis of the Probability of the Generation of Missiles for AP1000 Fully Integral Low Pressure Turbines, Revision 0, February 2007
	WCAP-16651-P (P) WCAP-16651-NP	Probabilistic Evaluation of Turbine Valve Test Frequency, Revision 1, May 2009

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Table 1.6-1 (Sheet 10 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
15.0	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-10054-P-A (P) WCAP-10081	Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985
	WCAP-12945-P-A (P) WCAP-14747	Code Qualification Document for Best Estimate LOCA Analysis, Revision 1, March 1998
	WCAP-7908-A	FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-16009-P-A (P) WCAP-16009-NP-A	Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005
15.1	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-9226-P-A (P) WCAP-9227-A	Reactor Core Response to Excessive Secondary Steam Releases, Revision 1, February 1998
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis, October 1999

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Table 1.6-1 (Sheet 11 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
15.2	WCAP-7769	Overpressure Protection for Westinghouse Pressurized Water Reactors, Revision 1, June 1972
	WCAP-16779-NP	Overpressure Protection Report for AP1000 Nuclear Power Plant, April 2007
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis, October 1999
15.3	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-8424	An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs, Revision 1, May 1975
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis, October 1999
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
15.4	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984

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Table 1.6-1 (Sheet 12 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
15.4	WCAP-15806-P-A (P) WCAP-15807-NP-A	Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics, November 2003
	WCAP-10965-P-A (P) WCAP-10966-A	ANC: A Westinghouse Advanced Nodal Computer Code, September 1986
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-11596-P-A (P) WCAP-11597-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, June 1988
	WCAP-16045-P-A (P) WCAP-16045-NP-A	Qualification of the Two-Dimensional Transport Code PARAGON, August 2004
	WCAP-10965-P-A, Addendum 1 (P) WCAP-10966-A Addendum 1	ANC – A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery, April 1989
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, October 1999
	WCAP-15063-P-A, Revision 1 with Errata (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), July 2000
	WCAP-16045-P-A Addendum 1-A (P) WCAP-16045-NP-A Addendum 1-A	Qualification of the NEXUS Nuclear Data Methodology, August, 2007
	WCAP-10965-P-A, Addendum 2-A (P)	Qualification of the New Pin Power Recovery Methodology, September, 2010
15.5	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
15.6	WCAP-12945-P-A (P) WCAP-14747	Code Qualification Document for Best Estimate Analysis, Revision 2, March 1998
	WCAP-10079-P-A (P) WCAP-10080-A	NOTRUMP – A Nodal Transient Small Break and General Network Code, August 1985

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Table 1.6-1 (Sheet 13 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
15.6	WCAP-10054-P-A (P) WCAP-10081-A	Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14601 (P) WCAP-15062	AP600 Accident Analyses – Evaluation Models, Revision 2, May 1998
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997
	WCAP-14171 (P) WCAP-14172	WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident, Revision 2, March 1998
	WCAP-14807 (P) WCAP-14808	NOTRUMP Final Validation Report for AP600, Revision 5, August 1998
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-16009-P-A (P) WCAP-16009-NP-A	Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005
	WCAP-14449-P-A (P) WCAP-14450	Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection, Revision 1, October 1999
16.1	WCAP-17661-P-A (P) WCAP-17661-NP-A	Improved RAOC and CAOC F_Q Surveillance Technical Specifications, Revision 1, February 2019

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Table 1.6-1 (Sheet 14 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
18.2	WCAP-14644	AP600/AP1000 Functional Requirements Analysis and Function Allocation, Revision 1, October 2008
18.3	WCAP-14645	Human Factors Engineering Operating Experience Review Report for the AP1000 Nuclear Power Plant, Revision 3, May 2006
18.4	WCAP-14644	AP600/AP1000 Functional Requirements Analysis and Function Allocation, Revision 1, October 2008

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Table 1.6-1 (Sheet 15 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
18.5	[*WCAP-14695*	*Description of the Westinghouse Operator Decision Making Model and Function Based Task Analysis Methodology, July 1996*]*
	[*WCAP-14651*	*Integration of Human Reliability Analysis and Human Factors Engineering Design Implementation Plan, Revision 2, May 1997*]*
18.7	[*WCAP-14651*	*Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan, Revision 2, May 1997*]*
18.7	WCAP-16555	AP1000 Identification of Critical Human Actions and Risk Important Tasks, Revision 0, March 2006
18.8	[*WCAP-14651*	*Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan, Revision 2, May 1997*]*
	[*WCAP-15860*	*Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan, Revision 2, October 2003*]*
	[*WCAP-14695*	*Description of the Westinghouse Operator Decision Making Model and Function Based Task Analysis Methodology, July 1996*]*
	WCAP-14655	Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel, Revision 1, August 1996
	WCAP-14694	Designer's Input to Determination of the AP600 Main Control Room Staffing Level, July 1996
	[*WCAP-14396*	*Man-in-the-Loop Test Plan Description, Revision 3, November 2002*]*
18.10	WCAP-14655	Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel, Revision 1, August 1996

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Table 1.6-1 (Sheet 16 of 16)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
18.11	[*WCAP-15860*	*Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan, Revision 2, October 2003*]*
	[*APP-OCS-GEH-120*	*AP1000 Human Factors Engineering Design Verification Plan,*]* Revision 3, December 2014, Westinghouse Electric Company LLC
	[*APP-OCS-GEH-220*	*AP1000 Human Factors Engineering Task Support Verification Plan,*]* Revision 6, February 2018, Westinghouse Electric Company LLC
	[*APP-OCS-GEH-320*	*AP1000 Human Factors Engineering Integrated System Validation Plan,*]* Revision 7, February 2018, Westinghouse Electric Company LLC [*(as modified by changes provided in Appendix 18A)*]*
	[*APP-OCS-GEH-420*	*AP1000 Human Factors Engineering Discrepancy Resolution Process,*]* Revision 2, December 2014, Westinghouse Electric Company LLC
	[*APP-OCS-GEH-520*	*AP1000 Plant Startup Human Factors Engineering Design Verification Plan,*]* Revision 4, December 2014, Westinghouse Electric Company LLC
18.12	WCAP-13793	The AP600 System/Event Matrix, June 1994
19E	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14171 (P) WCAP-14172	WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident, Revision 2, March 1998

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Notes: [*Section 12, "Secure Development and Operational Environment Plan," of WCAP-16096, "Software Program Manual for Common Q™ Systems," Revision 4, is not applicable to the AP1000.*]*

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Table 1.6-201
Additional Material Referenced

Author/ Report Number	Title	Revision	Section	Document Transmittal	ADAMS Accession Number
10 CFR Part 52 Appendix D	Design Certification Rule for the AP1000 Design	–	1.1	–	–
QAPD	Quality Assurance Manual	–	17.5	–	–
Westinghouse/ APP-GW-GL-700	AP1000 Design Control Document	20	All	–	–

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Table 1.6-203 Methodologies Required By Technical Specifications

Technical Specification	Methodology Document Number	Methodology Document Title
5.5.14	WCAP-16361-P	Westinghouse Setpoint Methodology for Protection Systems – AP1000, February 2011
5.6.3 – 1	WCAP-9272-P-A (P) WCAP-9273-NP-A	Westinghouse Reload Safety Evaluation Methodology, July 1985
5.6.3 – 2a	WCAP-8385 (P) WCAP-8403	Power Distribution Control and Load Following Procedures, September 1974
5.6.3 – 2b	January 31, 1980 Memo Attachment	T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package
5.6.3 – 2c	BTP CPB 4.3-1	NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981
5.6.3 – 3	WCAP-10216-P-A (P) WCAP-10217-A	Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification, Revision 1A, February 1994
5.6.3 – 4a	WCAP-16009-P-A (P) WCAP-16009-NP-A	Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), Revision 0, January 2005
5.6.3 – 4b	APP-GW-GLE-026 (P) APP-GW-GLE-026-NP	Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000, Revision 1, January 2009
5.6.3 – 5	WCAP-12472-P-A (P) WCAP-12473-A	BEACON Core Monitoring and Operations Support System, August 1994; Addendum 1, May 1996; and Addendum 2, March 2001
5.6.3 – 5	WCAP-12472-P-A (P) WCAP-12472-NP-A	BEACON Core Monitoring and Operations Support System, Addendum 4, September 2012
5.6.3 – 6	APP-GW-GLR-137	Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/ OTΔT) Reactor Trips, Revision 1
5.6.3 – 7a	WCAP-16045-P-A WCAP-16045-NP-A	Qualification of the Two-Dimensional Transport Code PARAGON, August 2004
5.6.3 – 7b	WCAP-16045-P-A WCAP-16045-NP-A	Qualification of the NEXUS Nuclear Data Methodology, Addendum 1-A, August 2007
5.6.3 – 7c	WCAP-13749-P-A	Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement, March 1997
5.6.4	WCAP-14040-NP-A	Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, Revision 4, May 2004

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1.7 Drawings and Other Detailed Information

The figures referenced in Subsections 1.7.1 and 1.7.2 may represent a functional diagram, general structural representation, or another general illustration. For instrumentation and control (I&C) systems, figures may also represent aspects of the relevant logic of the system or part of the system. Unless specified explicitly, the figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, and components. In particular, the as-built attributes of structures, systems, and components may vary from the attributes depicted on the figures, provided that those safety functions discussed in the design description pertaining to the figure are not adversely affected.

1.7.1 Electrical and Instrumentation and Control Drawings

Instrument and control functional diagrams, electrical one-line diagrams, and onsite standby diesel generator loading sequence and initiating circuit logic diagrams are listed in Table 1.7-1.

The legend for electrical power, control, lighting, and communication drawings is provided in Figure 1.7-1, sheets 1, 2, and 3. The index, notes, and symbols for instrument and control functional diagrams are provided in Figure 7.2-1.

1.7.2 Piping and Instrumentation Diagrams

Table 1.7-2 contains a list of piping and instrumentation diagrams (P&IDs) and the corresponding figure numbers. The three letter system names are provided in Table 1.7-2. Figures appear at the end of the respective text section. The P&ID legend, Figure 1.7-2, sheets 1, 2, and 3, provides an explanation of AP1000 symbols and characters used in these figures.

1.7.3 Combined License Information

This section contained no requirement for additional information.

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Table 1.7-1
I&C Functional and Electrical One-Line Diagrams

DCD Figure Number	Title
7.2-1 (Sheet 1)	Index and Symbols
7.2-1 (Sheet 2)	Reactor Trip Function
7.2-1 (Sheet 3)	Nuclear Startup Protection
7.2-1 (Sheet 4)	Nuclear Overpower Protection
7.2-1 (Sheet 5)	Core Heat Removal Protection and Reactor Coolant Pump Trip
7.2-1 (Sheet 6)	Primary Overpressure & Loss of Heat Sink Protection
7.2-1 (Sheet 7)	Loss of Heat Sink Protection
7.2-1 (Sheet 8)	Loss of Heat Sink Protection
7.2-1 (Sheet 9)	Steam Line Isolation
7.2-1 (Sheet 10)	Feedwater Isolation
7.2-1 (Sheet 11)	Safeguards Actuation
7.2-1 (Sheet 12)	Core Makeup Tank Actuation
7.2-1 (Sheet 13)	Containment and Other Protection
7.2-1 (Sheet 14)	Turbine Trip and Refueling Cavity and SFS Isolation
7.2-1 (Sheet 15)	Automatic RCS Overpressurization Valve Sequencing
7.2-1 (Sheet 16)	Incontainment Refueling Water Storage Tank Actuations
7.2-1 (Sheet 17)	Passive Residual Heat Removal and Core Makeup Tank Isolation Valve Interlocks
7.2-1 (Sheet 18)	Normal Residual Heat Removal System Isolation Valve Interlocks
7.2-1 (Sheet 19)	Containment Vacuum Relief Protection
7.2-1 (Sheet 20)	Diverse Actuation System Logic, Automatic Actuations
7.2-1 (Sheet 21)	Diverse Actuation System Logic, Manual Actuations
8.3.1-1	AC Power System - Station One-Line Diagram (Sheets 1 & 2)
8.3.1-2	On-site Standby Diesel Generator Initiation Circuit Logic Diagram
8.3.1-3	Ancillary Diesel Generator One Line Diagram
8.3.2-1	Class 1E DC System One-Line Diagrams (Sheets 1 & 2)
8.3.2-2	Class 1E 208Y/120V UPS One-Line Diagram
8.3.2-3	Non-Class 1E DC & UPS System One-Line Diagrams (Sheets 1, 2 & 3)

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Table 1.7-2 (Sheet 1 of 3)
AP1000 System Designators and System Diagrams

Designator	System (Note 1)	Section	Figure (Note 2)
ASS	Auxiliary Steam Supply System	10.4.10	None
BDS	Steam Generator Blowdown System	10.4.8	10.4.8-1
CAS	Compressed and Instrument Air Systems	9.3.1	9.3.1-1
CCS	Component Cooling Water System	9.2.2	9.2.2-2
CDS	Condensate System	10.4.7	10.4.7-1
CES	Condenser Tube Cleaning System	10.4.1.2.1, 10.4.5.2.3	None
CFS	Turbine Island Chemical Feed System	10.4.11	None
CMS	Condenser Air Removal System	10.4.2	None
CNS	Containment System	6.2.3	None
CPS	Condensate Polishing System	10.4.6	10.4.6-1
CVS	Chemical and Volume Control System	9.3.6	9.3.6-1
CWS	Circulating Water System	10.4.5	None
DAS	Diverse Actuation System	7.7	7.2-1 (Sh. 19 & 20)
DDS	Data Display and Processing System	7.1 & 7.7	7.1-1
DOS	Standby Diesel Fuel Oil System	9.5.4	9.5.4-1
DRS	Storm Drain System (Wholly out of scope)	None	None
DTS	Demineralized Water Treatment System	9.2.3	None
DWS	Demineralized Water Transfer and Storage System	9.2.4	9.2.4-1
ECS	Main ac Power System	8.3.1	8.3.1-1
EDS	Non Class 1E dc and UPS System	8.3.2	8.3.2-3
EFS	Communication Systems	9.5.2	None
EGS	Grounding and Lightning Protection System	8.3.1.1	None
EHS	Special Process Heat Tracing System	8.3.1.1	None
ELS	Plant Lighting System	9.5.3	None
EQS	Cathodic Protection System (Partially out of scope)	None	None
FHS	Fuel Handling and Refueling System	9.1.1, 9.1.2, 9.1.4	9.1 - various
FPS	Fire Protection System	9.5.1, 6.5.2	9.5.1-1
FWS	Main and Startup Feedwater System	10.4.7, 10.4.9	10.4.7-1
GSS	Gland Seal System	10.4.3	10.4.3-1
HCS	Generator Hydrogen and CO_2 Systems	10.2	None
HDS	Heater Drain System	10.4.7	None
HSS	Hydrogen Seal Oil System	10.2	None
IDS	Class 1E dc and UPS System	8.3.2	8.3.2-1
IIS	In-core Instrumentation System	4.4.6	None

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Table 1.7-2 (Sheet 2 of 3)
AP1000 System Designators and System Diagrams

Designator	System (Note 1)	Section	Figure (Note 2)
LOS	Main Turbine and Generator Lube Oil System	10.2	None
MES	Meteorological and Environmental Monitoring System (Wholly out of scope)	2.3.3	None
MHS	Mechanical Handling System	9.1	None
MSS	Main Steam System	10.3	10.3.2-2
MTS	Main Turbine System	10.2	10.2-1
OCS	Operation and Control Centers System	7.1, Ch. 18	7.1-1
PCS	Passive Containment Cooling System	6.2.2	6.2.2-1
PGS	Plant Gas Systems	9.3.2	None
PLS	Plant Control System	7.1 & 7.7	7.1-1
PMS	Protection and Safety Monitoring System	Ch. 7	7.2-1
PSS	Primary Sampling System	9.3.3	9.3.3-1
PWS	Potable Water System (Partially out of scope)	9.2.5	9.2.5-1 (Note 4)
PXS	Passive Core Cooling System	6.3	6.3-1
RCS	Reactor Coolant System	5.1	5.1-5
RDS	Gravity and Roof Drain Collection System (Partially out of scope)	None	None
RMS	Radiation Monitoring System	11.5	None
RNS	Normal Residual Heat Removal System	5.4.7	5.4-7
RWS	Raw Water System (Wholly out of scope)	9.2.1.2.2, 9.2.1.2.3.1, 9.2.3, 9.2.5	None
RXS	Reactor System	3.9.4, 3.9.5, 4.2.2.2, 4.2.2.3.1, 5.3	5.3-1
SDS	Sanitary Drainage System (Partially out of scope)	9.2.6	None
SES	Plant Security System (Partially out of scope)	13.6	None
SFS	Spent Fuel Pool Cooling System	9.1.3	9.1-6
SGS	Steam Generator System	10.3, 10.4.7, 10.4.9	10.3.2-1
SJS	Seismic Monitoring System	3.7.4	None
SMS	Special Monitoring System	4.4.6.4	None
SSS	Secondary Sampling System	9.3.4	None
SWS	Service Water System	9.2.1	9.2.1-1
TCS	Turbine Building Closed Cooling Water System	9.2.8	None

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Table 1.7-2 (Sheet 3 of 3)
AP1000 System Designators and System Diagrams

Designator	System (Note 1)	Section	Figure (Note 2)
TDS	Turbine Island Vents, Drains and Relief System	9.2.9.2.2, 10.4.2.2.1, 10.4.3.1.2, 10.4.3.2.2, 10.4.6.3	None
TOS	Main Turbine Control and Diagnostics System	10.2.2.4	None
TVS	Closed Circuit TV System (Wholly out of scope)	None	None
VAS	Radiologically Controlled Area Ventilation System	9.4.3	9.4.3-1
VBS	Nuclear Island Nonradioactive Ventilation System	9.4.1	9.4.1-1
VCS	Containment Recirculation Cooling System	9.4.6	9.4.6-1
VES	Main Control Room Emergency Habitability System	6.4	6.4-2
VFS	Containment Air Filtration System	9.4.7	9.4.7-1
VHS	Health Physics and Hot Machine Shop HVAC System	9.4.11	9.4.11-1
VLS	Containment Hydrogen Control System	6.2.4	6.2.4 - various
VRS	Radwaste Building HVAC System	9.4.8	9.4.8-1
VTS	Turbine Building Ventilation System	9.4.9	9.4.9-1
VUS	Containment Leak Rate Test System	6.2.5	6.2.5-1
VWS	Central Chilled Water System	9.2.7	9.2.7-1
VXS	Annex/Auxiliary Non-Radioactive Ventilation System	9.4.2	9.4.2-1
VYS	Hot Water Heating System	9.2.10	None
VZS	Diesel Generator Building Ventilation System	9.4.10	9.4.10-1
WGS	Gaseous Radwaste System	11.3	11.3-2
WLS	Liquid Radwaste System	11.2	11.2-2
WRS	Radioactive Waste Drain System	9.3.5, 11.2	9.3.5-1
WSS	Solid Radwaste System	11.4	11.4-1
WWS	Waste Water System (Partially out of scope)	9.2.9	None
YFS	Yard Fire Water System (Wholly out of scope)	None	None
ZAS	Main Generation System (Note 3)	8.1	None
ZBS	Transmission Switchyard and Offsite Power System (Wholly out of scope)	8.2	None
ZOS	Onsite Standby Power System	8.2.1, 8.3.1	8.3.1-4, 8.3.1-5
ZRS	Offsite Retail Power System (Wholly out of scope)	None	None
ZVS	Excitation and Voltage Regulation System	10.2.2.3	None

Notes:

1. For the System names:
   a) An entry with the system name only means the system is wholly in the scope of the AP1000 design certification.
   b) An entry with the system name followed by (Partially out of scope) means the system is partially in the scope of the AP1000 design certification.
   c) An entry with the system name followed by (Wholly out of scope) means the system is not in the scope of the AP1000 design certification.
2. For the DCD Figures:
   In the AP1000 design documentation system, Piping and Instrumentation Diagrams are numbered xxx-M6-yyy, where xxx is the system designator and yyy is the sheet number. Electrical One-Line Diagrams are numbered xxx-E3-yyy, where xxx is the system designator and yyy is the sheet number. I&C Functional Logic Diagrams are numbered xxx-J1-yyy, where xxx is the I&C system designator and yyy is the sheet number.
3. For the Main Generation System:
   The high side voltage of the main step-up transformer and the reserve auxiliary transformer is site specific.
4. Figure 9.2.5-1 is a partial Piping and Instrumentation Diagram entitled, “Main Control Room Potable Water System Isolation.”

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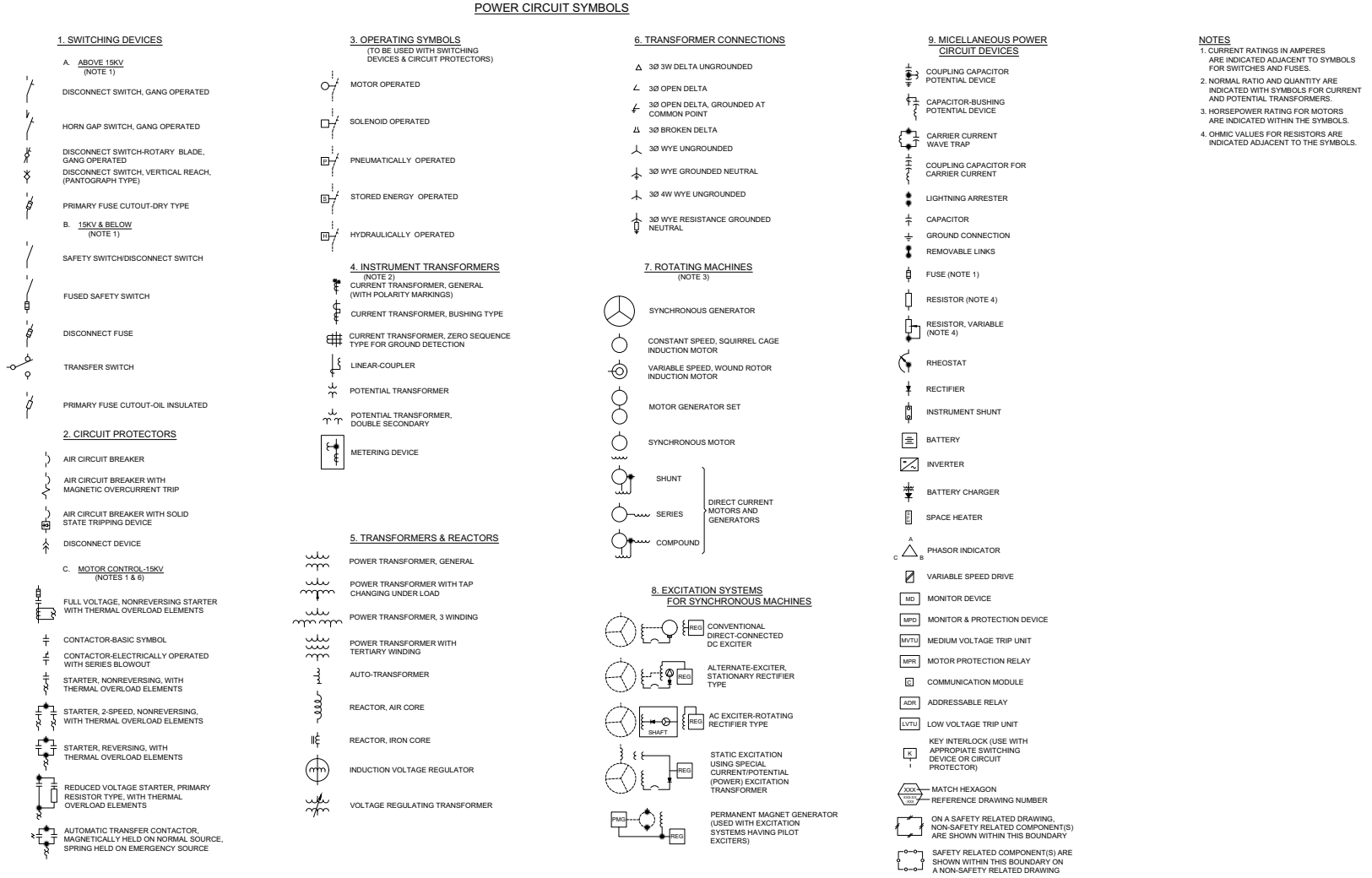

Figure 1.7-1 (Sheet 1 of 3)
Legend for Electrical Power, Lighting, and Communication Drawings

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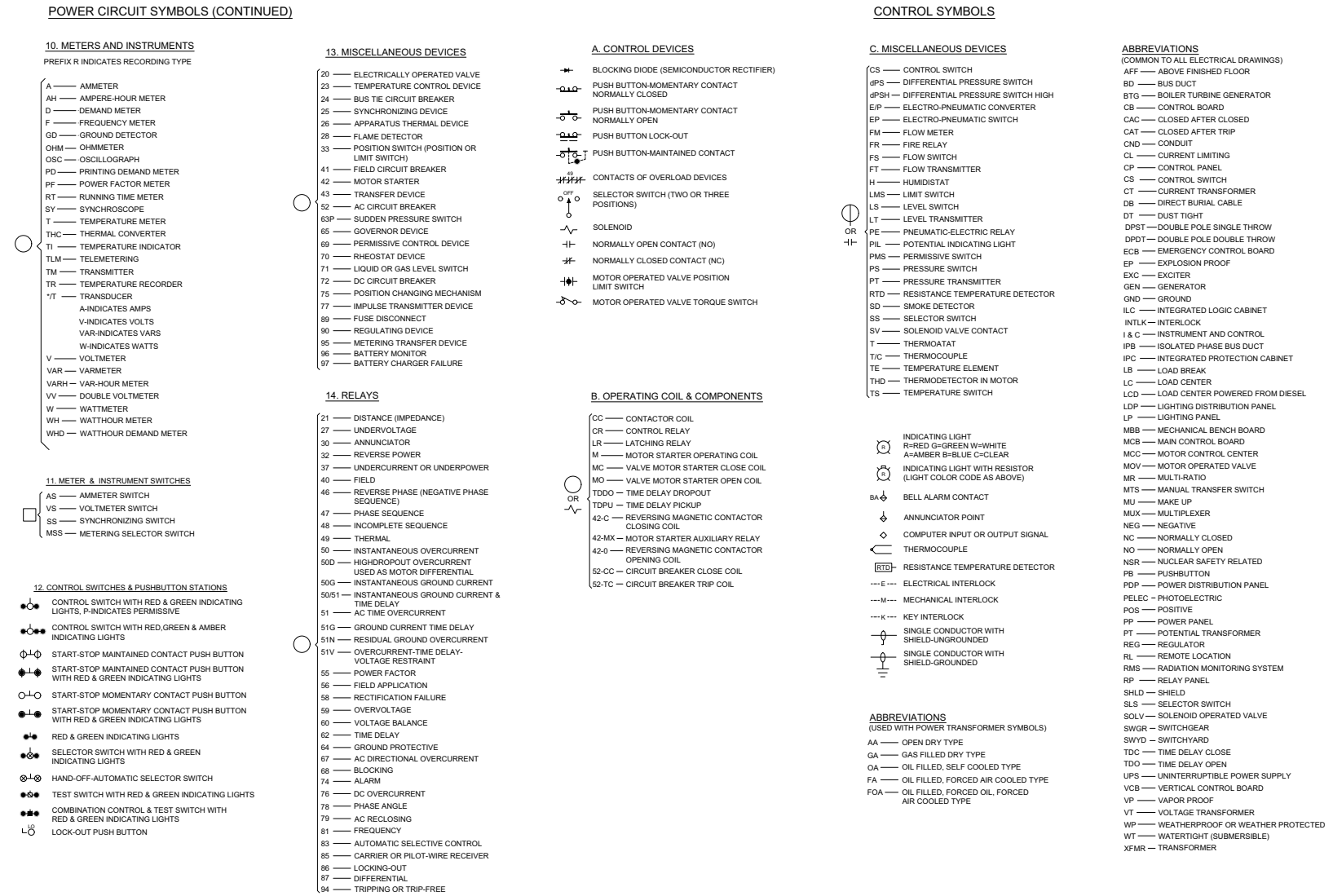

Figure 1.7-1 (Sheet 2 of 3)
Legend for Electrical Power, Lighting, and Communication Drawings

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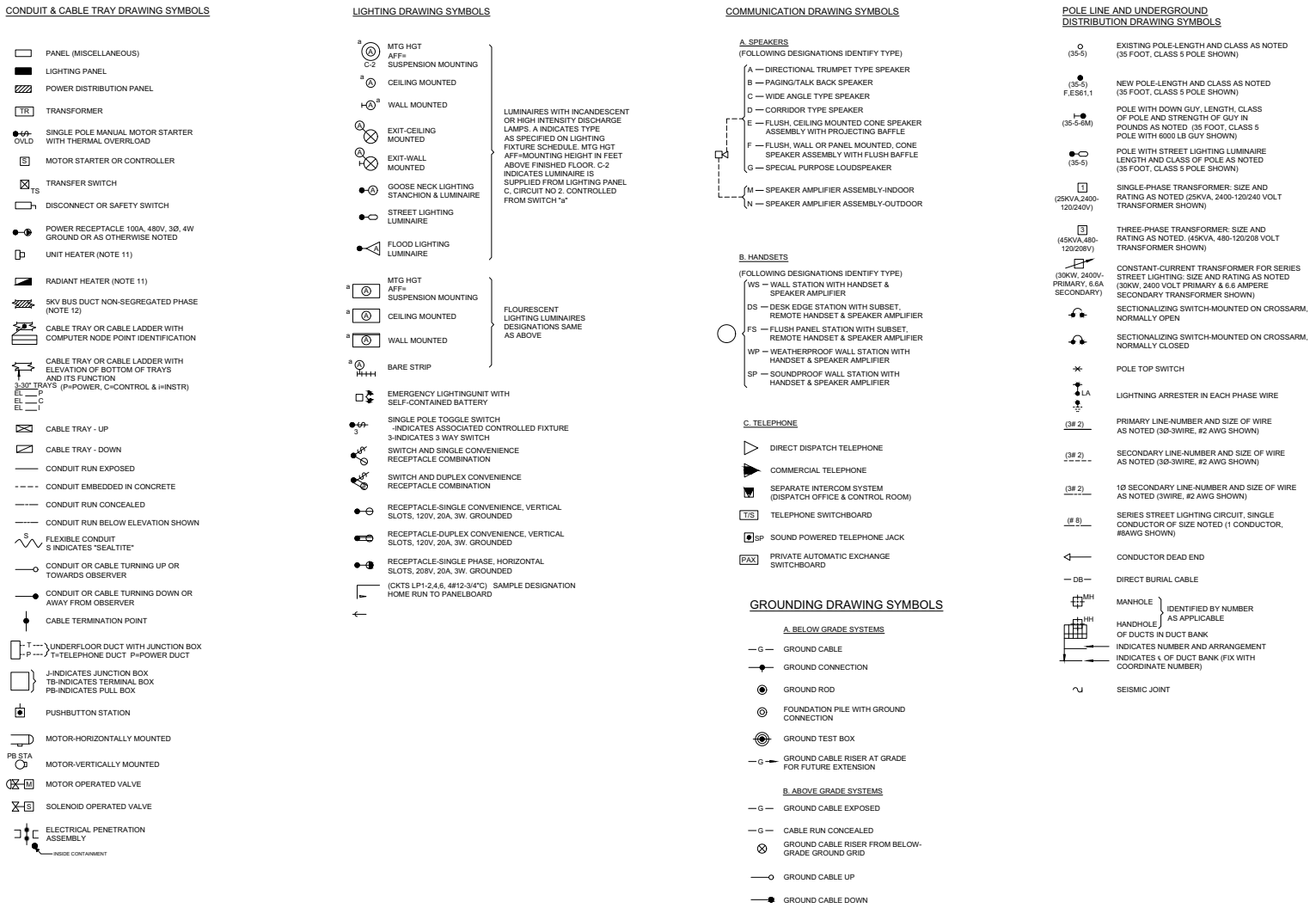

Figure 1.7-1 (Sheet 3 of 3)
Legend for Electrical Power, Lighting, and Communication Drawings

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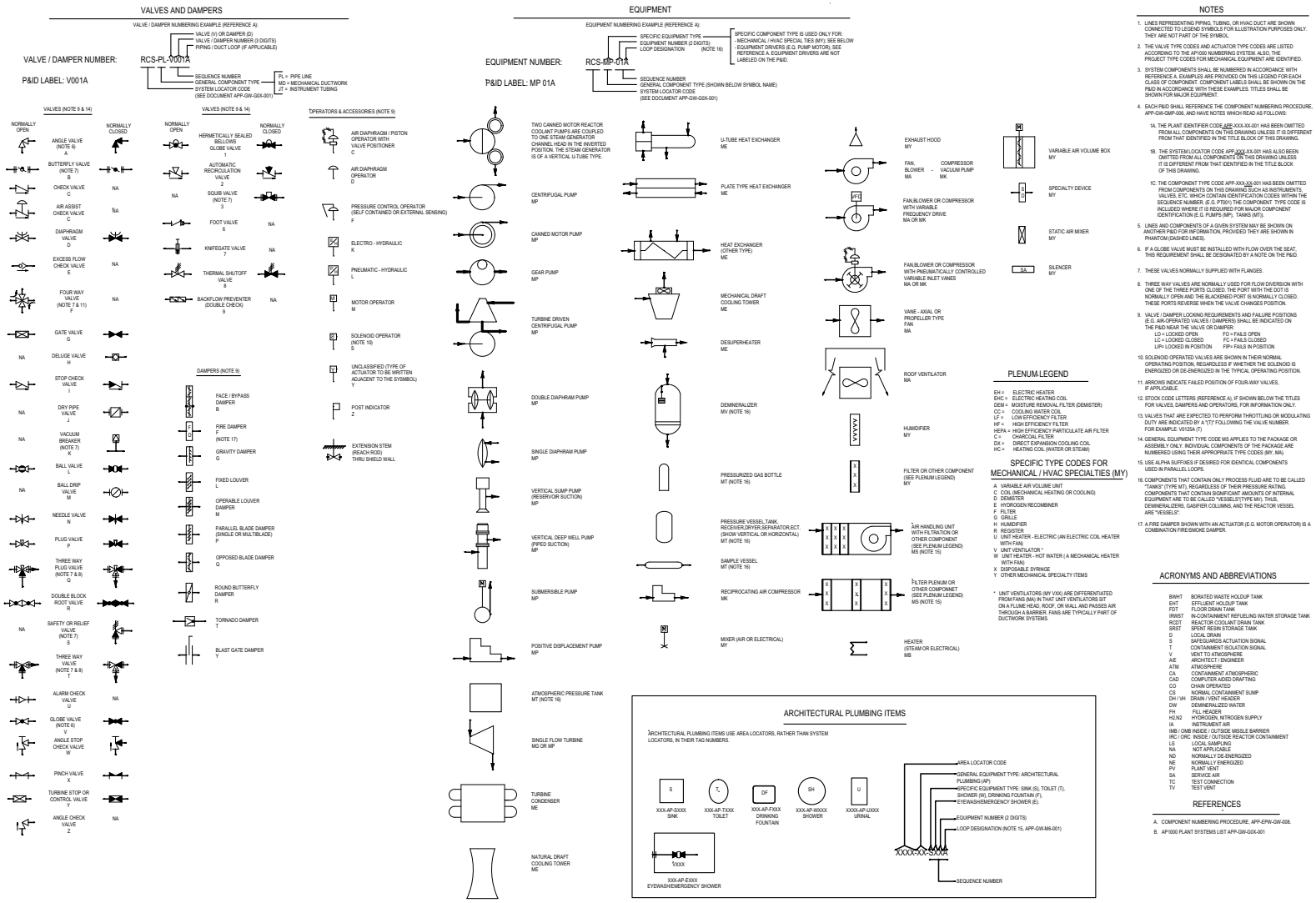

Figure 1.7-2 (Sheet 1 of 3)
Piping and Instrumentation Diagram Legend

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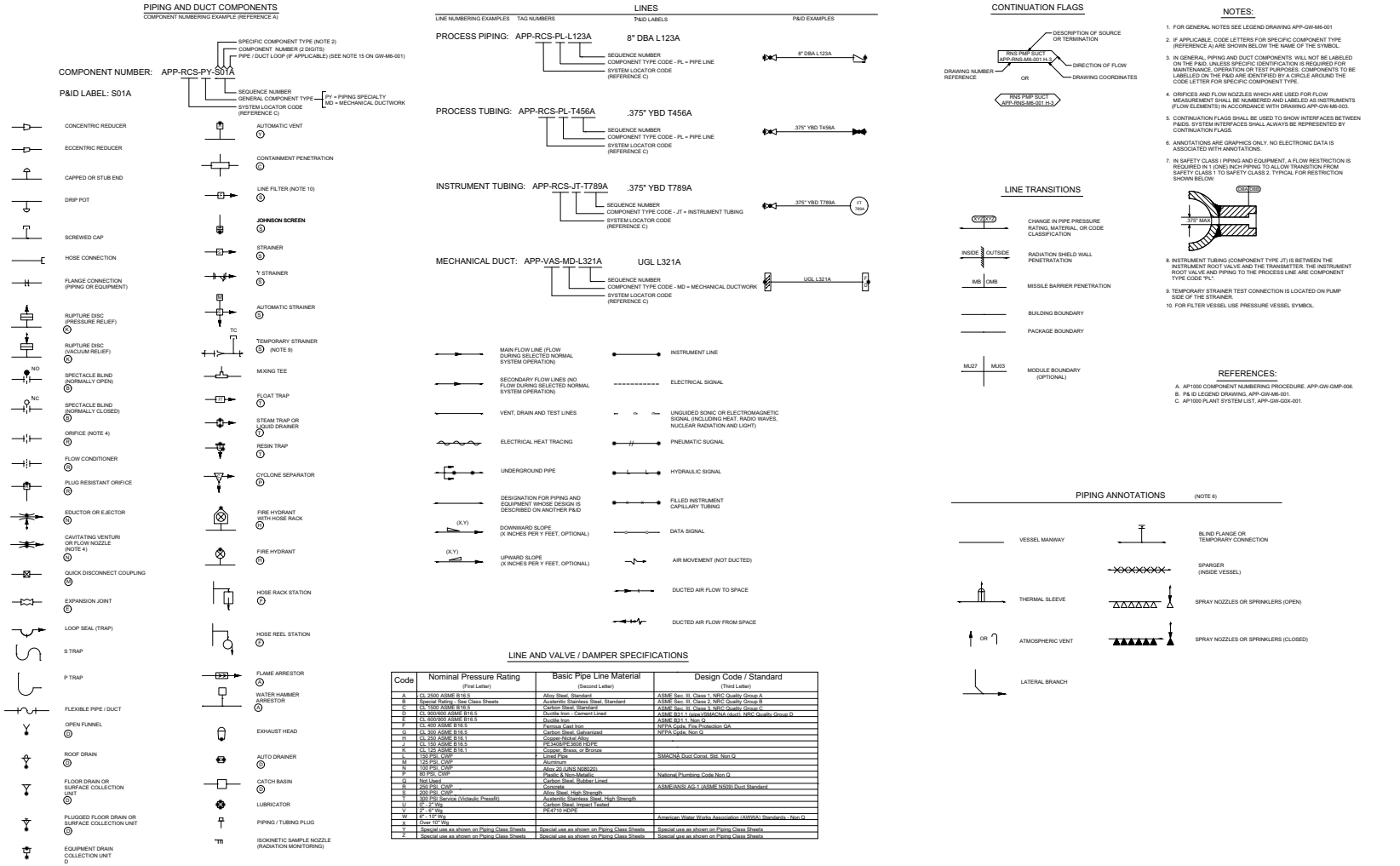

Code	Nominal Pressure Rating (First Letter)	Basic Pipe Line Material (Second Letter)	Design Code / Standard (Third Letter)
A	CL 2500 ASME B16.5	Alloy Steel, Standard	ASME Sec. III, Class 1, NRC Quality Group A
B	Special Rating - See Class Sheets	Austenitic Stainless Steel, Standard	ASME Sec. III, Class 2, NRC Quality Group B
C	CL 1500 ASME B16.5	Carbon Steel, Standard	ASME Sec. III, Class 3, NRC Quality Group C
D	CL 900/600 ASME B16.5	Ductile Iron - Cement Lined	ASME B31.1 (pipe)/SMACNA (duct), NRC Quality Group D
E	CL 600/300 ASME B16.5	Ductile Iron	ASME B31.1, Non Q
F	CL 400 ASME B16.5	Famous Cast Iron	NFPA Code, Fire Protection QA
G	CL 300 ASME B16.5	Carbon Steel, Galvanized	NFPA Code, Non Q
H	CL 250 ASME B16.1	Copper-Nickel Alloy	
J	CL 150 ASME B16.5	PE3408/PE3608 HDPE	
K	CL 125 ASME B16.1	Copper, Brass, or Bronze	
L	150 PSI, CWP	Lined Pipe	SMACNA Duct Const. Std. Non Q
M	125 PSI, CWP	Aluminum	
N	150 PSI, CWP	Alloy 20 (UNS N08020)	
P	80 PSI, CWP	Plastic & Non-Metallic	National Plumbing Code Non Q
Q	Not Used	Carbon Steel, Rubber Lined	
R	250 PSI, CWP	Concrete	ASME/ANSI AG-1 (ASME N509) Duct Standard
S	200 PSI, CWP	Alloy Steel, High Strength	
T	300 PSI Service (Victaulic Presafit)	Austenitic Stainless Steel, High Strength	
U	0" - 2" Wg	Carbon Steel, Impact Tested	
V	2" - 6" Wg	PE4710 HDPE	
W	6" - 10" Wg		American Water Works Association (AWWA) Standards - Non Q
X	Over 10" Wg		
Y	Special use as shown on Piping Class Sheets	Special use as shown on Piping Class Sheets	Special use as shown on Piping Class Sheets
Z	Special use as shown on Piping Class Sheets	Special use as shown on Piping Class Sheets	Special use as shown on Piping Class Sheets

Figure 1.7-2 (Sheet 2 of 3)
Piping and Instrumentation Diagram Legend

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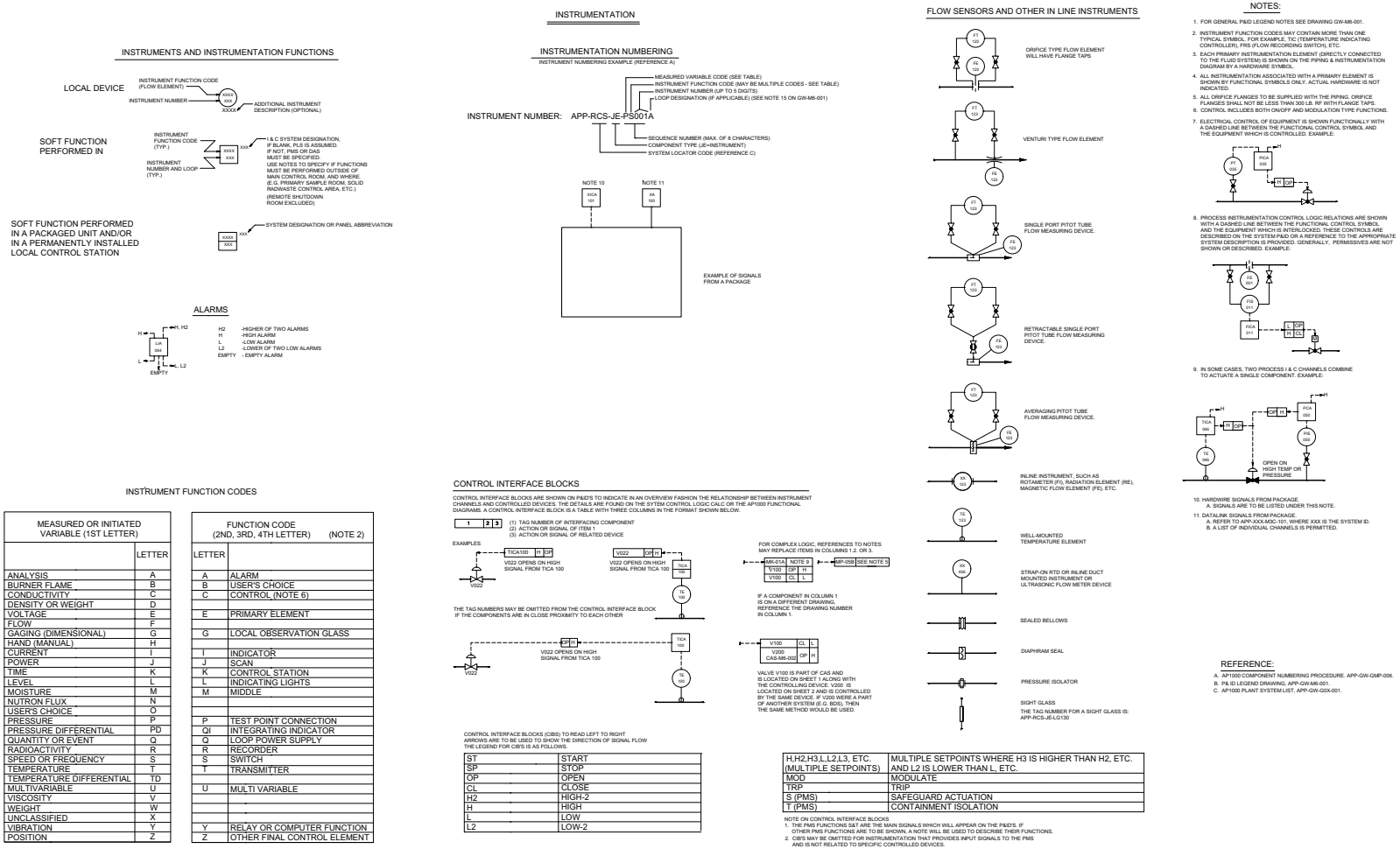

INSTRUMENT FUNCTION CODES

MEASURED OR INITIATED VARIABLE (1ST LETTER)	LETTER
ANALYSIS	A
BURNER FLAME	B
CONDUCTIVITY	C
DENSITY OR WEIGHT	D
VOLTAGE	E
FLOW	F
GAGING (DIMENSIONAL)	G
HAND (MANUAL)	H
CURRENT	I
POWER	J
TIME	K
LEVEL	L
MOISTURE	M
NUTRON FLUX	N
USER'S CHOICE	O
PRESSURE	P
PRESSURE DIFFERENTIAL	PD
QUANTITY OR EVENT	Q
RADIOACTIVITY	R
SPEED OR FREQUENCY	S
TEMPERATURE	T
TEMPERATURE DIFFERENTIAL	TD
MULTIVARIABLE	U
VISCOSITY	V
WEIGHT	W
UNCLASSIFIED	X
VIBRATION	Y
POSITION	Z

FUNCTION CODE (2ND, 3RD, 4TH LETTER) (NOTE 2)	
LETTER	
A	ALARM
B	USER'S CHOICE
C	CONTROL (NOTE 6)
E	PRIMARY ELEMENT
G	LOCAL OBSERVATION GLASS
I	INDICATOR
J	SCAN
K	CONTROL STATION
L	INDICATING LIGHTS
M	MIDDLE
P	TEST POINT CONNECTION
QI	INTEGRATING INDICATOR
Q	LOOP POWER SUPPLY
R	RECORDER
S	SWITCH
T	TRANSMITTER
U	MULTI VARIABLE
Y	RELAY OR COMPUTER FUNCTION
Z	OTHER FINAL CONTROL ELEMENT

ST	START
SP	STOP
OP	OPEN
CL	CLOSE
H2	HIGH-2
H	HIGH
L	LOW
L2	LOW-2

H,H2,H3,L,L2,L3, ETC. (MULTIPLE SETPOINTS)	MULTIPLE SETPOINTS WHERE H3 IS HIGHER THAN H2, ETC. AND L2 IS LOWER THAN L, ETC.
MOD	MODULATE
TRP	TRIP
S (PMS)	SAFEGUARD ACTUATION
T (PMS)	CONTAINMENT ISOLATION

Figure 1.7-2 (Sheet 3 of 3)
Piping and Instrumentation Diagram Legend

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1.8 Interfaces for Standard Design

This section identifies the AP1000 standard plant scope, interfaces related to design certification between the AP1000 plant design and the Combined License applicant, and the site-specific items to be included in an application for a Combined License. It is submitted to satisfy the requirements of 10 CFR 52.47(a)(25).

The AP1000 is a plant incorporating six buildings, the equipment in them, and the associated yard structures and tankage. This includes the nuclear island (consisting of the containment/shield building and the auxiliary building), the annex building and associated equipment, the diesel generator building and associated equipment, the turbine building, the turbine/generator equipment, and the radwaste facilities. The physical boundary of the portion of the AP1000 included in this application is shown on the site plan, Figure 1.2-2. It includes arrangement and placement of structures within the indicated boundary. Additionally, the red zone delay barrier necessary for security is included, but the boundary fence and vehicle barrier are not included since they are site-specific. As a result, no interfaces need to be identified between or among the portions of the plant within the boundary. They are addressed in their appropriate section of this DCD. There are no safety-related interfaces to site-specific elements of the plant outside the scope of this certification application. The following site-specific elements are outside the scope of the AP1000 standard plant:

(1) The portions of the circulating water system and its heat sink outside the AP1000 buildings, the portions of the service water blowdown lines outside the AP1000 buildings, which transfer fluid to the circulating water cooling tower basins, as well as the specific design details of the main condenser. A conceptual design is presented, delineated by Double Brackets ([[ ]]), in Subsection 10.4.5, based upon a cooling tower approach.

(2) The offsite power transmission system outside the low voltage terminals of the main and reserve transformers. Location and design of the main switchyard area and the equipment located therein, as well as design details such as voltage level for the main step-up transformers. A conceptual design of this system is included, delineated by Double Brackets ([[ ]]), in Section 8.2 for reference.

(3) Raw water source and treatment outside the turbine building. An interface specification of amount and water chemistry limits is provided.

(4) Sanitary and other drain systems outside the buildings identified above. This DCD is based upon the COL applicant providing adequate overall site drain collection and processing systems

(5) Communications systems and equipment outside the buildings identified above. This DCD is based upon the COL applicant providing adequate external communications.

(6) Location and design of administrative and training structures.

(7) Landscaping features.

(8) Size and location of the waste water retention basins and the associated plant outfall piping.

A more detailed listing of the systems included in the standard AP1000 plant is included in Section 3.2.

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There are a number of information interfaces between the AP1000 design and other portions of a completely licensed facility which must be addressed by parties that reference the AP1000 design. These interfaces are identified in Table 1.8-1 in the order they are presented in this DCD.

The safety-related interface requirements in Table 1.8-1 have been selected based on a review of interfaces between the AP1000 plant design and other Combined License applicant or site-specific items. Satisfying the referenced information for each of the interfaces listed will provide confidence that systems, structures and components within the AP1000 can perform their safety functions. The specific details of the interface parameters are identified in the DCD sections identified in Table 1.8-1. The interface specifications have been selected to suit a wide range of potential sites. Values identified by a Combined License applicant to be outside the range of acceptable parameters may be demonstrated to be acceptable. Such cases will be documented in the appropriate sections of the specific Combined License application.

The classification of interface types is based on the sources of interfaces listed in Appendix A of Regulatory Guide 1.70. The first four types below are directly related to the four sources of interfaces. They have been redefined slightly to reflect the fact that AP1000 is an essentially complete plant design. The classification of interface types is as follows:

- **Requirement of AP1000** – Requirements for operation of the AP1000 design that must be satisfied by the matching portion of the site, utility or Combined License applicant administration.

- **AP1000 Interface** – Interface condition used for AP1000 design which must be more precisely defined during the coordination effort between the AP1000 design team and the Combined License applicant.

- **Site Interface** – Site-related interface data upon which the AP1000 design is based.

- **Pertinent Criteria** – Criteria pertinent to the AP1000 design that may be useful for the design and staff review of the matching systems, components and structures.

- **Not an Interface** – Interface items identified in Appendix A of Regulatory Guide 1.70 which are wholly within the boundaries of the AP1000 plant. As a result, the "Matching Interface Item" in Table 1.8-1 is identified as N/A (not applicable).

- **Non-Nuclear Safety (NNS)** – Interface items identified in Appendix A of Regulatory Guide 1.70 which are non-nuclear safety-related because of the design features of AP1000.

Note that all plant interfaces listed in Appendix A of Regulatory Guide 1.70 have been listed in Table 1.8-1. As noted above and in Table 1.8-1, a number of these interfaces do not apply to the AP1000 plant as described in this DCD. In some cases, the interface listed in Appendix A of Regulatory Guide 1.70 is totally within the AP1000 plant and therefore not an interface. Other interfaces from Appendix A of Regulatory Guide 1.70 are identified as non-nuclear safety. The classification of systems, structures and components is described in Section 3.2. Only safety-related interfaces are detailed in Table 1.8-1. An example of an "NNS" (non-nuclear safety) type of interface is any of those associated with site service water. AP1000 does not rely on site service water as a safety grade ultimate heat sink. Neither the cooling tower nor the diesel-generator building is safety-related in AP1000. As such, there are no safety-related interfaces for these features.

Interfaces are listed in the order discussed in the DCD. General interfaces are listed as they relate to a particular section of this DCD. No specific system-by-system interface listings are required due to the complete nature of the AP1000 plant design. All safety-related systems are contained within the AP1000 plant design. The listing includes identification of the interface classification and the

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matching interface item to be specified by the Combined License applicant. In addition, the section of this DCD which addresses the listed interface is identified. To satisfy the requirements of 10 CFR 52.47(a)(25), representative conceptual designs are included in this DCD for those portions of the plant for which Westinghouse does not seek certification to aid the NRC staff in its review of the DCD and the probabilistic risk assessment to be submitted in support of the application, and to permit assessment of the adequacy of interface requirements.

Combined License Information

Combined License applicants referencing the AP1000 certified design will be required to provide site-specific information, verification that interface criteria are satisfied, information related to operating procedures, standard FSAR text, and other information required to support the AP1000 Design Certification. The description of information to be provided by the Combined License applicant is found in the DCD sections applicable to the specific information. Table 1.8-2 is a listing of the Combined License Information Items and the DCD location of the description of the information. In some cases, the activity required by a COL Information Item requires as-built information or other conditions that are not available when the COL application is submitted. These items are noted in the applicable DCD sections and Table 1.8-2. These activities are completed prior to fuel load.

1.8.1 Combined License Information

The Combined License applicant referencing the AP1000 certified design will address the applicability of the standard FSAR text in their application.

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Table 1.8-1 (Sheet 1 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
2.1	Envelope of AP1000 plant site related parameters	Site Interface	Site specific parameters	2.0
2.2	External missiles from man-made hazards and accidents	Site Interface	Site specific parameters	2.2
2.3	Maximum loads from man-made hazards and accidents	Site Interface	Site specific parameters	2.2
2.4	Limiting meteorological parameters (χ/Q) for design basis accidents and for routine releases and other extreme meteorological conditions for the design of systems and components exposed to the environment.	Site Interface	Site specific parameters	2.3
2.5	Tornado and operating basis wind loadings	Site Interface	Site specific parameters	2.3
2.6	External missiles generated by natural phenomena	Site Interface	Site specific parameters	2.3
2.7	Snow, ice and rain loads	Site Interface	Site specific parameters	2.3
2.8	Ambient air temperatures	Site Interface	Site specific parameters	2.3
2.9	Onsite meteorological measurement program	Requirement of AP1000	Combined License applicant program	2.3.3
2.10	Flood and ground water elevations	Site Interface	Site specific parameters	2.4
2.11	Hydrostatic loads on systems, components and structures	Site Interface	Site specific parameters	2.4
2.12	Seismic parameters peak ground acceleration response spectra shear wave velocity	Site Interface	Site specific parameters	 2.5 2.5 2.5

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Table 1.8-1 (Sheet 2 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
2.13	Required bearing capacity of foundation materials	Site Interface	Site specific parameters	2.5
3.1	Deleted			
3.2	Operating procedures to minimize water hammer	Requirement of AP1000	Combined License applicant procedure	3.6, 10
3.3	Site seismic sensor location and "trigger" value	Requirement of AP1000	Onsite implementation	3.7.4
3.4	Depth of overburden	Requirement of AP1000	Onsite implementation	3.8
3.5	Depth of embedment	Requirement of AP1000	Onsite implementation	3.8
3.6	Specific depth of waterproofing	Requirement of AP1000	Onsite implementation	3.8.5
3.7	Foundation Settlement Monitoring	Requirement of AP1000	Combined License applicant coordination	3.8.5
3.8	Lateral earth pressure loads	Not an Interface	N/A	3
3.9	Preoperational piping vibration test parameters	Not an Interface	N/A	3
3.10	Inservice Inspection requirements and locations	Requirement of AP1000	Combined License applicant program	3.9.6 5.2.4 6.6
3.11	Maintenance of preservice and reference test data for inservice testing of pumps and valves	Requirement of AP1000	Combined License applicant program	3.9.6 5.2.4 6.6
3.12	Earthquake response procedures	Requirement of AP1000	Combined License applicant program	3.7.4
5.1	Steam Generator Tube Surveillance Requirements	Requirement of AP1000	Combined License applicant program	5.4.2
6.1	Inservice Inspection requirements for the containment	Requirement of AP1000	Combined License applicant program	6.2.1
6.2	Offsite environmental conditions assumed for Main Control Room and control support area habitability design	AP1000 Interface	Site specific parameter	6.4

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Table 1.8-1 (Sheet 3 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
7.1	Listing of all design criteria applied to the design of the I&C systems	Not an Interface	N/A	7
7.2	Power required for site service water instrumentation	NNS and Not an Interface	N/A	7
7.3	Other provisions for site service water instrumentation	NNS and Not an Interface	N/A	7
7.4	Post-accident monitoring system	NNS	Combined License applicant coordination	7.5.5
8.1	Listing of design criteria applied to the design of the offsite power system	NNS	Combined License applicant coordination	8
8.2	Offsite ac requirements Steady-state load Inrush kVA for motors Nominal voltage Allowable voltage regulation Nominal frequency Allowable frequency fluctuation Maximum frequency decay rate Limiting under frequency value for RCP	NNS	Combined License applicant coordination	8

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Table 1.8-1 (Sheet 4 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
8.3	Offsite transmission system analysis: Loss of AP1000 or largest unit Voltage operating range Transient stability must be maintained and the RCP bus voltage must remain above the voltage required to maintain the flow assumed in Chapter 15 analyses for a minimum of three (3) seconds following a turbine trip. The protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip.	NNS	Combined License applicant analysis	8.2
8.4	Listing of design criteria applied to the design of onsite ac power systems	NNS and Not an Interface	N/A	8
8.5	Onsite ac requirements	NNS and Not an Interface	N/A	8
8.6	Diesel generator room coordination	NNS and Not an Interface	N/A	8
8.7	Listing of design criteria applied to the design of onsite dc power systems	Not an Interface	N/A	8
8.8	Provisions of dc power systems to accommodate the site service water system	NNS and Not an Interface	N/A	8
9.1	Listing of design criteria applied to the design of portions of the site service water within AP1000	NNS and Not an Interface	N/A	9
9.2	Integrated heat load to site service water system	NNS and Not an Interface	N/A	9
9.3	Plant cooling water systems parameters	NNS and Not an Interface	N/A	9
9.4	Plant makeup water quality limits	NNS	Site specific parameter	9

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Table 1.8-1 (Sheet 5 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
9.5	Requirements for location and arrangement of raw and sanitary water systems	NNS	Site implementation	9
9.6	Ventilation requirements for diesel-generator room	NNS and Not an Interface	N/A	9
9.7	Requirements to satisfy fire protection program	AP1000 Interface	Combined License applicant program	9.5.1
9.8	Requirements for location and size of waste water retention basins and associated plant outfall	NNS	Site Implementation	9
11.1	Expected release rates of radioactive material from the Liquid Waste System including: Location of release points Effluent temperature Effluent flow rate Size and shape of flow orifices	Site Interface	Site specific parameters	11.2
11.2	Expected release rates of radioactive materials from the Gaseous Waste System including: Location of release points Height above grade Height relative to adjacent buildings Effluent temperature Effluent flow rate Effluent velocity Size and shape of flow orifices	Site Interface	Site specific parameters	11.3
11.3	Expected release rates of radioactive material from the Solid Waste System including: Location of release points Material types Material qualities Size and shape of material containers	Site Interface	Site specific parameters	11.4

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Table 1.8-1 (Sheet 6 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
11.4	Requirements for offsite sampling and monitoring of effluent concentrations	AP1000 Interface	Combined License applicant program	11.5
12.1	Identification of miscellaneous radioactive sources	AP1000 Interface	Combined License applicant program	12.2
13.1	Features that may affect plans for coping with emergencies	AP1000 Interface	Combined License applicant program	13.3
13.2	Physical Security Plan consistent with AP1000 plant	AP1000 Interface	Combined License applicant program	13.6
14.1	Identification of special features to be considered in development of the initial test program	Requirement of AP1000	Combined License applicant program	14
14.2	Maintenance of preoperational test data and inservice inspection baseline data	AP1000 Interface	Combined License applicant program	14
16.1	Administrative requirements associated with reliability information maintenance	AP1000 Interface	Combined License applicant program	16
16.2	Administrative requirements associated with the Technical Specifications	Requirement of AP1000	Combined License applicant implementation	16
16.3	Site and operator related information associated with the Reliability Assurance Program (D-RAP)	Requirement of AP1000	Combined License applicant program	16.2
18.1	Operating staff consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.6
18.2	Operator training consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.8 18.10
18.3	Operating Procedures consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.8 18.10

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Summary of AP1000 Standard Plant
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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
1.1-1	Construction and Startup Schedule	1.1.7	N/A	Yes	–
1.8-1	Confirmation of Standard FSAR Text	1.8.1	N/A	Yes	–
1.9-1	Regulatory Guide Conformance	1.9.1.5	APP-GW-GLN-129	Yes	–
2.1-1	Geography and Demography	2.1.1	N/A	Yes	–
2.2-1	Identification of Site-specific Potential Hazards	2.2.1	N/A	Yes	–
2.3-1	Regional Climatology	2.3.6.1	N/A	Yes	–
2.3-2	Local Meteorology	2.3.6.2	N/A	Yes	–
2.3-3	Onsite Meteorological Measurements Program	2.3.6.3	N/A	Yes	–
2.3-4	Short-Term Diffusion Estimates	2.3.6.4	N/A	Yes	–
2.3-5	Long-Term Diffusion Estimates	2.3.6.5	N/A	Yes	–
2.4-1	Hydrological Description	2.4.1.1	N/A	Yes	–
2.4-2	Floods	2.4.1.2	N/A	Yes	–
2.4-3	Cooling Water Supply	2.4.1.3	N/A	Yes	–
2.4-4	Groundwater	2.4.1.4	N/A	Yes	–
2.4-5	Accidental Release of Liquid Effluents into Ground and Surface Water	2.4.1.5	N/A	Yes	–
2.4-6	Flood Protection Emergency Operation Procedures	2.4.1.6	N/A	Yes	–

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Summary of AP1000 Standard Plant
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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
2.5-1	Basic Geologic and Seismic Information	2.5.1	N/A	Yes	–
2.5-2	Site Seismic and Tectonic Characteristics Information	2.5.2.1	N/A	Yes	–
2.5-3	Geoscience Parameters	2.5.2.3	N/A	Yes	–
2.5-4	Surface Faulting	2.5.3	N/A	Yes	–
2.5-5	Site and Structures	2.5.4.6.1	N/A	Yes	–
2.5-6	Properties of Underlying Materials	2.5.4.6.2	N/A	Yes	–
2.5-7	Excavation and Backfill	2.5.4.6.3	N/A	Yes	–
2.5-8	Ground Water Conditions	2.5.4.6.4	N/A	Yes	–
2.5-9	Liquefaction Potential	2.5.4.6.5	N/A	Yes	–
2.5-10	Bearing Capacity	2.5.4.6.6	N/A	Yes	–
2.5-11	Earth Pressures	2.5.4.6.7	N/A	Yes	–
2.5-12	Static and Dynamic Stability of Facilities	2.5.4.6.9	N/A	Yes	–
2.5-13	Subsurface Instrumentation	2.5.4.6.10	N/A	Yes	–
2.5-14	Stability of Slopes	2.5.5	N/A	Yes	–
2.5-15	Embankments and Dams	2.5.6	N/A	Yes	–
2.5-16	Settlement of Nuclear Island	2.5.4.6.11	N/A	Yes	–
2.5-17	Waterproofing System	2.5.4.6.12	N/A	Yes	–

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Summary of AP1000 Standard Plant
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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
3.3-1	Wind and Tornado Site Interface Criteria	3.3.3	APP-GW-GLR-020	Yes	–
3.4-1	Site-Specific Flooding Hazards Protective Measures	3.4.3	N/A	Yes	–
3.5-1	External Missile Protection Requirements	3.5.4	APP-GW-GLR-020	Yes	–
3.6-1	Pipe Break Hazards Analysis	3.6.4.1	APP-GW-GLR-021 APP-GW-GLR-074	No	Yes
3.6-4	Primary System Inspection Program for Leak-Before-Break Piping	3.6.4.4	N/A	Yes	–
3.7-1	Seismic Analysis of Dams	3.7.5.1	N/A	Yes	–
3.7-2	Post-Earthquake Procedures	3.7.5.2	N/A	Yes	–
3.7-3	Seismic Interaction Review	3.7.5.3	APP-GW-GLR-021	No	Yes
3.7-4	Reconciliation of Seismic Analyses of Nuclear Island Structures	3.7.5.4	APP-GW-GLR-021 APP-GW-S2R-010	No	Yes
3.7-5	Location of Free-Field Acceleration Sensor	3.7.5.5	N/A	Yes	–
3.8-5	Structures Inspection Program	3.8.6.5	N/A	Yes	–
3.8-6	Construction Procedures Program	3.8.6.6	N/A	No	Yes
3.9-2	Design Specification and Reports	3.9.8.2	APP-GW-GLR-021	No	Yes
3.9-3	Snubber Operability Testing	3.9.8.3	N/A	Yes	–
3.9-4	Valve Inservice Testing	3.9.8.4	APP-GW-GLN-020	Yes	–
3.9-5	Surge Line Thermal Monitoring	3.9.8.5	N/A	Yes	–
3.11-1	Equipment Qualification File	3.11.5	APP-GW-GLN-110	No	Yes

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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
4.4-2	Confirm Assumptions for Safety Analyses DNBR Limits	4.4.7.2	N/A	–	Yes
5.2-1	ASME Code and Addenda	5.2.6.1	N/A	Yes	–
5.2-2	Plant Specific Inspection Program	5.2.6.2	N/A	Yes	–
5.2-3	Response to Unidentified Reactor Coolant System Leakage Inside Containment	5.2.6.3	N/A	Yes	–
5.3-1	Reactor Vessel Pressure – Temperature Limit Curves	5.3.6.1	APP-GW-GLR-021	No	Yes
5.3-2	Reactor Vessel Materials Surveillance Program	5.3.6.2	N/A	Yes	–
5.3-4	Reactor Vessel Materials Properties Verification	5.3.6.4.1	N/A	No	Yes
5.4-1	Steam Generator Tube Integrity	5.4.15	N/A	Yes	–
5.3-7	Quickloc Weld Buildup ISI	5.3.6.6	N/A	Yes	–
6.1-1	Procedure Review for Austenitic Stainless Steels	6.1.3.1	N/A	Yes	–

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Table 1.8-2 (Sheet 5 of 12)
Summary of AP1000 Standard Plant
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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
6.1-2	Coating Program	6.1.3.2	N/A	Yes	–
6.2-1	Containment Leak Rate Testing	6.2.6	N/A	Yes	–
6.3-1	Containment Cleanliness Program	6.3.8.1	N/A	Yes	–
6.4-1	Local Hazardous Gas Services and Monitoring	6.4.7	N/A	Yes	–
6.4-2	Procedures for Training for Control Room Habitability	6.4.7	N/A	Yes	–
6.6-1	Inspection Programs	6.6.9.1	N/A	Yes	–
6.6-2	Construction Activities	6.6.9.2	N/A	Yes	–
7.1-1	Setpoint Calculations for Protective Functions	7.1.6.1	WCAP-16361-P	Yes	Yes
7.5-1	Post-Accident Monitoring	7.5.5	N/A	Yes	No
8.2-1	Offsite Electrical Power	8.2.5	N/A	Yes	–
8.2-2	Technical Interfaces	8.2.5	N/A	Yes	–

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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
8.3-1	Grounding and Lightning Protection	8.3.3	N/A	Yes	–
8.3-2	Onsite Electrical Power Plant Procedures	8.3.3	N/A	Yes	–
9.1-5	Inservice Inspection Program of Cranes	9.1.6.5	N/A	Yes	–
9.1-6	Radiation Monitor	9.1.6.6	N/A	Yes	–
9.1-7	Metamic Monitoring Program	9.1.6.7	N/A	No	Yes
9.2-1	Potable Water	9.2.12.1	N/A	Yes	–
9.2-2	Waste Water Retention Basins	9.2.12.2	N/A	Yes	–
9.3-1	Air Systems (NUREG-0933 Issue 43)	9.3.7	N/A	Yes	–
9.4-1	Ventilation Systems Operations	9.4.12	N/A	Yes	–
9.5-1	Qualification Requirements for Fire Protection Program	9.5.1.9.1	N/A	Yes	–
9.5-2	Fire Protection Analysis Information	9.5.1.9.2	N/A	Yes	–
9.5-3	Regulatory Conformance	9.5.1.9.3	N/A	Yes	–

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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
9.5-4	NFPA Exceptions	9.5.1.9.4	N/A	Yes	–
9.5-6	Verification of Field Installed Fire Barriers	9.5.1.9.6	N/A	No	Yes
9.5-8	Establishment of Procedures to Minimize Risk for Fire Areas Breached During Maintenance	9.5.1.9.7	N/A	Yes	–
9.5-9	Offsite Interfaces	9.5.2.5.1	N/A	Yes	–
9.5-10	Emergency Offsite Communications	9.5.2.5.2	N/A	Yes	–
9.5-11	Security Communications	9.5.2.5.3	N/A	Yes	–
9.5-13	Fuel Degradation Protection	9.5.4.7.2	APP-GW-GLR-120	Yes	–
10.1-1	Erosion-Corrosion Monitoring	10.1.3	N/A	No	Yes
10.2-1	Turbine Maintenance and Inspection	10.2.6	APP-GW-GLN-018	No	Yes
10.4-1	Circulating Water Supply	10.4.12.1	N/A	Yes	–
10.4-2	Condensate, Feedwater and Auxiliary Steam System Chemistry Control	10.4.12.2	N/A	Yes	–
10.4-3	Potable Water	10.4.12.3	N/A	Yes	–

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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
11.2-1	Liquid Radwaste Processing by Mobile Equipment	11.2.5.1	N/A	Yes	–
11.2-2	Cost Benefit Analysis of Population Doses	11.2.5.2	N/A	Yes	–
11.3-1	Cost Benefit Analysis of Population Doses	11.3.5.1	N/A	Yes	–
11.4-1	Solid Waste Management System Process Control Program	11.4.6	N/A	Yes	–
11.5-1	Plant Offsite Dose Calculation Manual (ODCM)	11.5.8	N/A	Yes	–
11.5-2	Effluent Monitoring and Sampling	11.5.8	N/A	Yes	–
11.5-3	10 CFR 50, Appendix I	11.5.8	N/A	Yes	–
12.1-1	ALARA and Operational Policies	12.1.4	N/A	Yes	–
12.2-1	Additional Contained Radiation Sources	12.2.3	N/A	Yes	–
12.3-1	Administrative Controls for Radiological Protection	12.3.5.1	N/A	Yes	–
12.3-2	Criteria and Methods for Radiological Protection	12.3.5.2	N/A	Yes	–
12.3-3	Groundwater Monitoring Program	12.3.5.3	N/A	Yes	–

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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
12.3-4	Record of Operational Events of Interest for Decommissioning	12.3.5.4	N/A	Yes	–
12.5-1	Radiological Protection Organization and Procedures	12.5.5	N/A	Yes	–
13.1-1	Organizational Structure of Combined License Applicant	13.1.1	N/A	Yes	–
13.2-1	Training Program for Plant Personnel	13.2.6	N/A	Yes	–
13.3-1	Emergency Planning and Communications	13.3.1	N/A	Yes	–
13.3-2	Activation of Emergency Operations Facility	13.3.1	N/A	Yes	–
13.4-1	Operational Review	13.4.1	N/A	Yes	–
13.5-1	Plant Procedures	13.5.3	APP-GW-GLR-040	Yes	–
13.6-1	Security	13.6.1	APP-GW-GLR-062 APP-GW-GLR-066 APP-GW-GLR-068	Yes	–
13.6-5	Cyber Security Program	13.6.1	APP-GW-GLR-104	No	Yes
14.4-1	Organization and Staffing	14.4.1	N/A	Yes	–
14.4-2	Test Specifics and Procedures	14.4.2	APP-GW-GLR-037	No	Yes
14.4-3	Conduct of Test Program	14.4.3	APP-GW-GLR-038	No	Yes
14.4-4	Review and Evaluation of Test Results	14.4.4	N/A	No	Yes
14.4-5	Testing Interface Requirements	14.4.5	N/A	Yes	–
15.0-1	Documentation of Plant Calorimetric Uncertainty Methodology	15.0.15.1	NA	–	Yes

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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
15.7-1	Consequences of Tank Failure	15.7.6	N/A	Yes	–
16.1-1	Technical Specification Preliminary Information	16.1	APP-GW-GLR-064 APP-GW-GLN-075	Yes	–
16.3-1	Procedure to Control Operability of Investment Protection Systems, Structures and Components	16.3.2	N/A	Yes	–
17.5-1	Quality Assurance Design Phase	17.7.1	N/A	Yes	–

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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
17.5-2	Quality Assurance for Procurement, Fabrication, Installation, Construction and Testing	17.7.2	N/A	Yes	–
17.5-4	Quality Assurance Program for Operations	17.7.4	N/A	Yes	–
17.5-8	Operational Reliability Assurance Program Integration with Quality Assurance Program	17.7.8	N/A	Yes	–
18.2-2	Design of the Emergency Operations Facility	18.2.6.2	APP-GW-GLR-136	Yes	No
18.6-1	Plant Staffing	18.6.1	APP-GW-GLR-090	Yes	–
18.10-1	Training Program Development	18.10.1	APP-GW-GLR-090	Yes	–
18.14-1	Human Performance Monitoring	18.14	APP-GW-GLR-090	Yes	–
19.11-1	Site-Specific PRA	19.11	N/A	No	Yes
19.59.10-1	As-Built SSC HCLPF Comparison to Seismic Margin Evaluation	19.11	APP-GW-GLR-021	No	Yes
19.59.10-2	Evaluation of As-Built Plant Versus Design in AP1000 PRA and Site-Specific PRA External Events	19.11	APP-GW-GLR-101	Yes	Yes
19.59.10-3	Internal Fire and Internal Flood Analyses	19.11	APP-GW-GLR-021	No	Yes
19.59.10-4	Develop and Implement Severe Accident Management Guidance	19.11	APP-GW-GLR-070	Yes	–
19.59.10-5	Equipment Survivability	19.11	APP-GW-GLR-021	No	Yes

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Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
19.59.10-6	Confirm that the Seismic Margin Assessment analysis is applicable to the COL site.	19.11	n/a	Yes	No
	Bulletins and Generic Letters (WCAP-15800, Revision 3, July 2004)	1.9.5.5	n/a	Yes	No
	Unresolved Safety Issues and Generic Safety Issues	Table 1.9-2	APP-GW-GLN-129	Yes	No

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1.9 Compliance with Regulatory Criteria

1.9.1 Regulatory Guides

Regulatory guides are issued by the NRC in the following 10 broad divisions:

- Division 1 - Power Reactors
- Division 2 - Research and Test Reactors
- Division 3 - Fuels and Materials Facilities
- Division 4 - Environmental and Siting
- Division 5 - Materials and Plant Protection
- Division 6 - Products
- Division 7 - Transportation
- Division 8 - Occupational Health
- Division 9 - Antitrust and Financial Review
- Division 10 - General

Divisions 2, 3, 6, 7, 9, and 10 of the regulatory guides do not apply to the design and design certification phase of AP1000. The following sections provide a summary discussion of NRC Divisions 1, 4, 5, and 8 of the regulatory guides applicable to the design and design certification phase of AP1000.

Appendix 1A provides a discussion of AP1000 regulatory guide conformance.

Divisions 2, 3, 6, 7, 9, and 10 of the regulatory guides do not apply to the construction or operational safety considerations and are not addressed.

Division 4 of the regulatory guides applies to the Environmental Report and the topics are addressed in the Environmental Report. Three Division 4 Regulatory Guides are addressed in Appendix 1A.

Division 5 of the regulatory guides applies to materials and plant protection. As appropriate, the Division 5 regulatory guide topics are addressed in the DCD and plant-specific security plans (i.e., Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Cyber Security Plan).

Applicable Division 8 Regulatory Guides are addressed in Appendix 1A.

Appendix 1A provides a discussion of plant specific regulatory guide conformance, addressing new Regulatory Guides and new revisions not addressed by the referenced DCD. Regulatory Guides that are completely addressed by the DCD are not listed.

The following subsections provide a summary discussion of Divisions 1, 4, 5 and 8 of the regulatory guides as applicable to the content of this document, or to the construction and/or operations phases.

1.9.1.1 Division 1 Regulatory Guides - Power Reactors

Currently there are approximately 190 Division 1 regulatory guides that have been issued by the NRC for implementation or for comment.

Appendix 1A provides an evaluation of the degree of AP1000 compliance with NRC Division 1 regulatory guides. The revisions of the regulatory guides against which AP1000 is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. For those regulatory guides applicable to the AP1000 Table 1.9-1 identifies

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the appropriate cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

The superseded or canceled regulatory guides are not considered in Appendix 1A or Table 1.9-1.

Appendix 1A provides an evaluation of the degree of compliance with Division 1 regulatory guides as applicable to the content of this document, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the degree of compliance is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the certified design (see Notes at the end of Appendix 1A). Table 1.9-1 identifies the appropriate regulatory guide to cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

Superseded or canceled regulatory guides are not considered in Appendix 1A or Table 1.9-1.

1.9.1.2 Division 4 Regulatory Guides - Environmental and Siting

One Division 4 regulatory guide, Regulatory Guide 4.7, merits discussion.

Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," provides guidelines for identifying suitable candidate sites for nuclear power stations. The guidance of this regulatory guide is considered as appropriate in the establishment of the AP1000 site interface criteria, and is described in Sections 2.1 and 2.5.

Division 4 of the regulatory guides applies to the Environmental Report and the topics are addressed in the Environmental Report. Appendix 1A provides an evaluation of the degree of compliance with Division 4 regulatory guides as applicable to the content of this document, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the plant is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the certified design (see Notes at the end of Appendix 1A). For those regulatory guides applicable, Table 1.9-1 identifies the appropriate cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

1.9.1.3 Division 5 Regulatory Guides - Materials and Plant Protection

Division 5 of the regulatory guides applies to materials and plant protection. Appendix 1A provides an evaluation of the degree of conformance with Division 5 regulatory guides as applicable to the content of the AP1000 DCD and the plant-specific Cyber Security Plan. The plant-specific physical security plans (i.e., Physical Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan) were developed using the template in NEI 03-12, Revision 6, “Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program],” which was endorsed for use by NRC letter dated April 9, 2009. The plant-specific physical security plans include no substantive deviations from the NRC-endorsed template in NEI 03-12, Revision 6. Therefore, the degree of conformance with Division 5 regulatory guides for the plant-specific physical security plans is consistent with the degree of conformance of NEI 03-12, Revision 6.

Three Division 5 regulatory guides, Regulatory Guides 5.9, 5.12, and 5.65, merit discussion.

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Regulatory Guide 5.9, "Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material," provides guidelines for data acquisition systems associated with the use of a lithium-drifted germanium gamma ray spectroscopy system. This regulatory guide is not applicable to AP1000 design certification.

Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials," provides guidelines for the selection and use of commercially available locks in the protection of facilities and special nuclear material. The guidance of this regulatory guide is considered as appropriate in the AP1000 design.

Regulatory Guide 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," is not applicable to design certification.

1.9.1.4 Division 8 Regulatory Guides - Occupational Health

Two Division 8 regulatory guides, Regulatory Guides 8.8 and 8.19 merit discussion.

Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations will be As Low As is Reasonably Achievable (ALARA)," provides NRC guidance for meeting the requirements of 10 CFR Part 20. This regulatory guide includes guidance in the following areas for maintaining radiation exposures ALARA:

- Overall program (e.g., policy, organization, and training)
- Facility and equipment design features
- Radiation protection program
- Radiation protection facilities, instrumentation, and equipment

Regulatory Guide 8.8 is written primarily for utility applicants and licensees. However, Westinghouse has established policy, design, and operational considerations that will be applied in the AP1000 design in accordance with this regulatory guide. These considerations are discussed in Section 12.1.

Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants" describes a method acceptable to the NRC staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process involved in designing a light-water-cooled power reactor so that occupational radiation exposures will be ALARA. This regulatory guide includes guidance for estimating occupational radiation exposures (principally during the design stage) as a result of:

- Reactor operations and surveillance
- Routine maintenance
- Waste processing
- Refueling
- Inservice inspection
- Special maintenance

Occupational radiation exposure estimates that are in accordance with Regulatory Guide 8.19 are described in Section 12.4.

Appendix 1A provides an evaluation of the degree of compliance with Division 8 regulatory guides as applicable to the content of this document, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the plant is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the

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certified design (see Notes at the end of Appendix 1A). For those regulatory guides applicable,Table 1.9-1 identifies the appropriate cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

Superseded or canceled regulatory guides are not considered in Appendix 1A or Table 1.9-1.

1.9.1.5 Combined License Information

Division 1, 4, 5 and 8 Regulatory Guides applicable to the content of this document, or to the site-specific design, construction and/or operational aspects are listed in Table 1.9-1 and Appendix 1A.

1.9.2 Compliance with Standard Review Plan (NUREG-0800)

WCAP-15799, "AP1000 Compliance with SRP Acceptance Criteria," provides the results of a review of the AP1000 compliance with the acceptance criteria for each section of the Standard Review Plan, NUREG-0800.

1.9.3 Three Mile Island Issues

This section identifies the Three Mile Island issues of 10 CFR 50.34(f) that are addressed by AP1000 design features or program plans. The additional issues of NUREG-0660 and NUREG-0737 that apply to the AP1000 are resolved in accordance with the guidance of NUREG-0933, with specific details provided in the applicable sections of the DCD.

Some of the 10 CFR 50.34(f) issues initially identified as applicable only to Boiling Water Reactors (BWRs) or Babcock and Wilcox plants have also been addressed for the AP1000 design. For example, the AP1000 design incorporates an automatic depressurization system with some similarity to that utilized for BWRs.

10 CFR 50.34(f):

(1)(i) Plant/Site Specific TMI-Related Risk Assessment (NUREG-0660 Item II.B.8)

"Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."

AP1000 Response:

A plant-specific Probabilistic Risk Assessment (PRA) performed on the AP1000 design evaluates the plant in terms of core damage frequency and containment integrity. The PRA supports the design effort and establishes the capability of the design to meet established safety goals. Level 1 (Plant) and Level 2 (Containment) PRA evaluations, and offsite dose evaluation, including internal and external events:

- Demonstrate that the plant design meets the NRC safety goals
- Identify design vulnerabilities, evaluate alternate design features and operational strategies, and modify the design to reduce risk

The PRA process has been integrated into the design process to verify that the design effort meets the targeted goals and resolves the identified vulnerabilities. As a result, specific design changes were incorporated into the plant systems to improve the reliability of the core and containment heat removal systems.

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Close interaction between the plant designers and PRA analysts is maintained to consider severe accident vulnerabilities as part of the design process. The AP1000 PRA is provided to the NRC as a separate document.

(1)(ii) Auxiliary Feedwater System Evaluation (NUREG-0737 Item II.E.1)

"Perform an evaluation of the proposed auxiliary feedwater system, to include (applicable to pressurized water reactors only): (A) a simplified Auxiliary Feedwater System reliability analysis using event-tree and fault-tree logic techniques, (B) a design review of Auxiliary Feedwater System, and (C) an evaluation of Auxiliary Feedwater System flow design bases and criteria."

AP1000 Response:

The AP1000 design does not utilize an auxiliary feedwater system. A nonsafety-related startup feedwater system is provided to remove the core decay heat after the reactor trip during postulated non-LOCA event. Decay heat removal maintains core subcooling and prevents water relief from the pressurizer safety valves by preventing heatup of the reactor coolant system. The startup feedwater pumps automatically start following anticipated transients resulting in low steam generator level. However, operation of the nonsafety-related startup feedwater system is not credited to mitigate licensing design basis accidents described in Chapter 15.

The safety-related passive core cooling system provides emergency core decay heat removal during transients, accidents, or whenever the normal nonsafety-related heat removal paths are unavailable.

The safety-related passive core cooling system design basis and criteria are described in Section 6.3.

(1)(iii) Reactor Coolant Pump Seals (NUREG-0737 Items II.K.2.16 and II.K.3.25)

"Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break loss of coolant accident with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss of coolant accident with subsequent reactor coolant pump seal damage."

AP1000 Response:

The AP1000 design uses sealless motor pumps for circulating primary reactor coolant through the reactor core, piping, and steam generators. In the sealless design, all rotating components are inside a pressure vessel; therefore, no seal can fail and initiate reactor coolant system leakage.

(1)(iv) Automatic Power-Operated Relief Valve Isolation System (NUREG-0737 Item II.K.3.2)

"Perform an analysis of the probability of a small-break loss of coolant accident caused by a stuck-open power-operated relief valve. If this probability is a significant contributor to the probability of small-break loss of coolant accidents from all causes, provide a description and evaluation of the effect on small-break loss of coolant accident probability of an automatic power-operated relief valve isolation system that would operate when the reactor coolant system pressure falls after the power-operated relief valve has opened."

AP1000 Response:

The AP1000 design does not include power-operated relief valves. The pressurizer volume is about 40 percent larger than the pressurizer volume in current plants with a comparable power rating. The larger pressurizer increases transient operation margins and prevents safety valve actuation in most accident situations. The pressurizer surge line is also larger to permit a more rapid transfer of coolant

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between the reactor coolant system and the pressurizer, and also to accommodate the automatic depressurization system first- to third-stage flow rates. The surge line limits the pressure drop during maximum anticipated surge (Condition II loss of load transient) to prevent exceeding the maximum reactor coolant system pressure limit.

Overpressure protection is provided by two totally enclosed pop-type safety valves. These valves are spring-loaded and self-actuated and they are designed to meet the requirements of the ASME Code, Section III. If the pressurizer pressure exceeds the set pressure, the safety valves start lifting. A temperature indicator in the discharge piping for each safety valve alarms on high temperature to alert the operator to the presence of high temperature fluid from leakage or when the valves open.

The AP1000 design also includes an automatic depressurization system. The system consists of four stages of valves. Three stages are connected to the pressurizer. The fourth stage is connected to the hot legs. These valves are not actuated on a high pressure signal. Design features are included to reduce the chance of spurious automatic depressurization system actuation including appropriate interlocks, 2-out-of-4 instrument actuation, fail as is valves, redundant, closed first, second, and third stage valves in each line, and redundant series controllers for fourth stage valves. Probabilistic risk assessment is used to determine the probability of a loss of coolant accident caused by failure of the automatic depressurization system. Results of this evaluation are factored into the design process. See Chapter 5 and Section 6.3 for additional information.

(1)(v) Separation of HPCI and RCIC System Initiation Levels (NUREG-0737 Item II.K.3.13)

"Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words 'high pressure core spray' for 'high pressure coolant injection' and 'HPCS' for 'HPCI')."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(vi) Relief Valve Challenges (NUREG-0737 Item II.K.3.16)

"Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(vii) Automatic Depressurization System Activation (NUREG-0737 Item II.K.3.18)

"Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system design modifications that would eliminate the need for manual activation to ensure adequate core cooling."

AP1000 Response:

Although this issue is identified as applicable to BWRs only, the AP1000 design uses an automatic depressurization system with some similarity to that used on BWRs.

The automatic depressurization system actuates on Low-3 core makeup tank level, coincident with a core makeup tank actuation signal. Therefore manual actuation of the automatic depressurization

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system is not required to maintain core cooling. As discussed in Section (1)(i), PRA analysis confirms the reliability of the automatic actuation. Additional information is provided in Section 6.3.

(1)(viii) Core Spray and Low Pressure Coolant Injection Systems (NUREG-0737 Item II.K.3.21)

"Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(ix) RCIC and HPCI Additional Space Cooling (NUREG-0737 Item II.K.3.24)

"Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words 'high pressure core spray' for 'high pressure coolant injection' and 'HPCS' for 'HPCI')."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(x) Automatic Depressurization System Functionality During Accidents (NUREG-0737 Item II.K.3.28)

"Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves."

AP1000 Response:

Although this issue is identified as applicable to BWRs only, the AP1000 uses a safety-related automatic depressurization system that is different from that presently used on BWRs. The AP1000 automatic depressurization system uses safety-related dc motor-operated valves and squib valves to initiate depressurization. The motive power for these valves is safety-related dc power. There is no nonsafety-related equipment or instrumentation, including instrument air or nitrogen supply, relied on in the operation of these valves.

These valves are designed and qualified to function in the conditions of an accident. They will also be subject of pre-operational and in-service testing. They will be included in the reliability assurance program. Additional information is provided in Section 6.3 for the passive core cooling system, Subsection 3.9.3 for valve operability requirements, Chapter 14 for the initial test program, Subsection 3.9.6 for in-service testing, and Section 16.2 for the reliability assurance program.

(1)(xi) Depressurization Methods/Rapid Cooldown (NUREG-0737 Item II.K.3.45)

"Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown."

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AP1000 Response:

This issue is applicable to BWRs only.

(1)(xii) Hydrogen Control System Evaluation (NUREG-0660 Item II.B.8)

"Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section (50.34). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include: (A) a comparison of costs and benefits of the alternative systems considered, (B) for the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section (50.34), and (C) for the selected system, preliminary design descriptions of equipment, function, and layout."

AP1000 Response:

Continuous indication of hydrogen concentration in the containment atmosphere is provided. The containment hydrogen control system maintains hydrogen concentrations below 10 percent following the reaction of 100 percent of the active zircaloy cladding.

Hydrogen igniters control rapid releases of hydrogen during and after postulated degraded core and core melt accidents to maintain concentration below 10 percent.

Sufficient vent area is provided for each subcompartment in the containment to prevent high local concentrations of hydrogen.

See Subsection 6.2.4 for additional information.

(2)(i) Simulator Capability (NUREG-0933 Item I.A.4.2)

"Provide simulator capability that correctly models the control room and includes the capability to simulate small-break loss of coolant accidents."

AP1000 Response:

Simulator capability is not included within the scope of the AP1000 design certification. Functional requirements for simulator capability are derived from Human Factors Engineering Program described in Chapter 18.

(2)(ii) Plant Procedures (NUREG-0933 Item I.C.9)

"Establish a program to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts."

AP1000 Response:

See Chapter 13 for a discussion of plant procedures, training of operations personnel and emergency planning.

(2)(iii) Control Room Design (NUREG-0737 Item I.D.1)

"Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts."

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AP1000 Response:

The human factors engineering design process of the AP1000 has been developed to conform with NUREG-0711, "Human Factors Engineering Program Review Model." The elements of the design process provide a structured top-down system analysis using accepted human factors engineering principles. The design of the main control room and the other operation and control centers reflect state-of-the-art human factors principles. See Appendix 1A for information on conformance with applicable regulatory guides. See Chapter 18 for additional information on the AP1000 human factors engineering design process.

(2)(iv) Safety Parameter Display System (NUREG-0737 Item I.D.2)

"Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded."

AP1000 Response:

The purpose of the plant safety parameter display console (or safety parameter display system) is to display important plant variables in the main control room in order to assist in rapidly and reliably determining the safety status of the plant.

The requirements for the safety parameter display system are specified during the main control room design process, and are met by the main control room design, specifically as part of the alarms, displays, and controls. The requirements for a safety parameter display system (NUREG-0696, Reference 1) are met by grouping the alarms by plant process or purpose, as directly related to the critical safety functions.

The process data presented on the graphic displays is similarly grouped, facilitating an easy transition for the operators. The safety parameter display system requirement for presentation of plant data in an analog fashion prior to reactor trip is met by the design of the graphic CRT displays.

Displays are available at the operator workstations, the remote shutdown workstation, and at the technical support center. See Chapter 18 for additional information pertaining to the safety parameter display system design.

(2)(v) Safety System Status Indication (NUREG-0933 Item I.D.3)

"Provide for automatic indication of the bypassed and [in]operable status of safety systems."

AP1000 Response:

The AP1000 main control room meets the NRC Regulatory Guide 1.47 recommendations, including automatic indication of bypassed and inoperable status of plant safety systems, as described in Appendix 1A.

Plant safety parameters, protection system status, and plant component status signals are processed by the protection and safety monitoring system and made available to the entire instrumentation and control system via the redundant monitor bus.

Class 1E signals are provided to the qualified data processor, which is part of the protection and safety monitoring system, for accident monitoring displays. The display of this data is incorporated in the process data displays on the PMS safety displays in the AP1000 main control room.

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See Chapters 7 and 18 for additional information pertaining to bypass inoperable status indication. Appendix 1A describes conformance with Regulatory Guide 1.47.

(2)(vi) Reactor Coolant System High Point Vents (NUREG-0737 Item II.B.1)

"Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity."

AP1000 Response:

In the AP1000 design, the capability for remotely operated high point venting of the reactor coolant system is provided by the safety-related automatic depressurization system valves and the safety-related reactor vessel head vent system. Both of these vent paths discharge to the in-containment refueling water storage tank.

During loss of cooling accident events, the automatic depressurization system automatically depressurizes the reactor coolant system so that the passive core cooling system may effectively deliver core cooling flow. Depressurization via the automatic depressurization system results in creation of a gas-steam volume in the upper region of the vessel. This vapor volume expands down to the inside of the hot leg before it begins venting through the hot leg either via the automatic depressurization system paths connected to the pressurizer or directly from the hot legs via the fourth stage automatic depressurization system paths. This process provides an open injection and steam venting flow path through the reactor vessel, maintaining required core cooling flow.

The reactor vessel head vent system can also be operated from the main control room to directly vent from the top of the reactor vessel head. Subsection 5.4.12 provides additional information pertaining to the reactor coolant system venting capabilities.

(2)(vii) Plant Radiation Shielding (NUREG-0737 Item II.B.2)

"Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID-14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment."

AP1000 Response:

Post-accident radiation sources, used in the shield design and assessment of post-accident access to vital areas, are addressed in Subsection 12.2.1.3. The post-LOCA instantaneous and integrated source strengths as a function of time are also included as Tables 12.2-20 and 12.2-21, respectively. The sources are based on the core activity release model from Regulatory Guide 1.183, which supersedes the TID-14844 source term assumptions as reflected in Regulatory Guide 1.4.

Vital areas for post-accident personnel access are addressed in Section 12.3, including radiation zone maps that show projected dose rates in these areas and access routes for the various post-accident actions in vital areas. Time estimates have been made for ingress, egress, and performance of actions at the vital area locations and have been used in demonstrating that total individual radiation doses are limited to less than 5 rem and that Item II.B.2 of NUREG-0737 and GDC-19 requirements are met.

Environmental qualification of safety-related equipment is addressed in Section 3.11. The determination of the radiation environments during postulated accident situations considers the

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activity release model based on NUREG-1465, which supersedes the source term definition of Parts 1 and 4 of Item II.B.2 of NUREG-0737.

Subsection 12.2.3 defines the responsibility to address any additional contained radiation sources not identified in 12.2.1. Thus, appropriate source terms have been identified and used in establishing that the requirements of Item II.B.2 of NUREG-0737 and GDC 19 are met and the issues are resolved.

(2)(viii) Post-Accident Sampling (NUREG-0737 Item II.B.3)

"Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 50 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations."

AP1000 Response:

Recently the NRC published a model Safety Evaluation Report on eliminating post-accident sampling system requirements from technical specifications for operating plants (Federal Register Volume 65, Number 211, October 31, 2000). The AP1000 sampling design is consistent with the approach in the Model safety evaluation report and not the guidance outlined in NUREG-0737 and Regulatory Guide 1.97. The primary sampling system design is consistent with contingency plans to obtain and analyze highly radioactive post-accident samples from the reactor coolant system, the containment sump, and the containment atmosphere.

(2)(ix) Hydrogen Control (NUREG-0660 Item II.B.8)

"Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1)(xii) of this section (50.34) is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that:

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation."

AP1000 Response:

See the response provided for issue (1)(xii).

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(2)(x) Reactor Coolant System Valve Testing (NUREG-0737 Item II.D.1)

"Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for pressurized water reactors, power-operated relief valves, block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed."

AP1000 Response:

The AP1000 reactor coolant system design does not include power-operated relief valves and their associated block valves. However, the safety valve and discharge piping used in the AP1000 design will be either of design similar to those items tested by EPRI and documented in EPRI Report EPRI NP-2770-LD (Reference 2) or will be tested in accordance with the guidelines of Item [II.D.1] of NUREG-0737.

The AP1000 design includes automatic depressurization system valves which are used to depressurize the plant and establish conditions for injection from the accumulators and the in-containment refueling water storage tank. The operability of the automatic depressurization system valves and spargers is confirmed by a test program. See Section 1.5 for information pertaining to the testing program.

Accident analyses for the AP1000 determine fluid conditions expected under operating conditions, transients, and accidents, and the postulated system responses to these conditions, including the operation of reactor coolant system safety valves. Anticipated transients without scram events are analyzed. Appropriate valve qualification documentation is maintained.

(2)(xi) Valve Position Indication (NUREG-0737 Item II.D.3)

"Provide direct indication of relief and safety valve position (open or closed) in the control room."

AP1000 Response:

The AP1000 design does not include power-operated relief valves and their associated block valves from the reactor coolant system.

Direct indication of relief and safety valve position (open or closed) is provided in the main control room.

(2)(xii) Auxiliary Feedwater System Initiation and Indication (NUREG-0737 Item II.E.1.2)

"Provide automatic and manual auxiliary feedwater system initiation, and provide auxiliary feedwater system flow indication in the control room."

AP1000 Response:

As previously noted in the AP1000 response to Issue (1)(ii), the AP1000 design includes a nonsafety-related startup feedwater system, but not an auxiliary feedwater system. Flow indication of the startup feedwater system is provided in the main control room.

The startup feedwater pumps automatically start following anticipated transients resulting in low steam generator level. The startup feedwater control valves automatically control feedwater flow to the steam generators during operation. They can also be operated manually from the main control room.

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The safety-related passive core cooling system provides for emergency core decay heat removal during transients, accidents, or whenever the normal heat removal paths are unavailable. Automatic and manual actuation and flow rate indication are available in the main control room.

(2)(xiii) Pressurizer Heater Power Supplies (NUREG-0737 Item II.E.3.1)

"Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available."

AP1000 Response:

The AP1000 pressurizer heaters are powered from the nonsafety-related ac power system. During loss of offsite power events, a portion of the pressurizer heaters is capable of being powered from the nonsafety-related onsite standby power system. The pressurizer heaters are capable of establishing and maintaining natural circulation in hot standby condition, with only the diesel generators supplying electrical power.

With only safety-related dc (Class 1E dc) power available, the safety-related passive core cooling system can establish and maintain natural circulation cooling using the passive residual heat removal heat exchangers, transferring the decay heat to the in-containment refueling water storage tank water and to the passive containment cooling system.

Therefore, the nonsafety-related pressurizer heaters are not required for core decay heat removal following a loss of offsite power. See Section 8.3 for additional information.

(2)(xiv) Containment Isolation System (NUREG-0737 Item II.E.4.2)

"Provide containment isolation systems that: (A) ensure all nonessential systems are isolated automatically by the containment isolation system, (B) for each non-essential penetration (except instrument lines) have two isolation barriers in series, (C) do not result in reopening of the containment isolation valves on resetting of the isolation signal, (D) utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation, and (E) include automatic closing on a high radiation signal for all systems that provide a path to the environs."

AP1000 Response:

The AP1000 containment isolation design satisfies NRC requirements, including post-TMI requirements. In general, this means that two barriers are provided -- one inside containment and the other outside containment. Usually these barriers are valves, but in some cases they are closed, seismic Category I piping systems not connected to the reactor coolant system or to the containment atmosphere. Table 6.2.3-1 identifies containment isolation design provisions for mechanical penetrations. The isolation signal and maximum closure times are defined for each remotely operated valve. Containment penetrations, other than equipment hatches and flanges, incorporate two isolation barriers in series.

The AP1000 design incorporates a reduction in the number of required penetrations compared to the number in previous plant designs. The majority of these penetrations are normally closed. Those few that are normally open, use automatically closed isolation valves.

Containment isolation is automatically actuated by a safeguards actuation signal, using two-out-of-four coincident logic. The containment isolation actuation is set as low as reasonable without creating potential for spurious trips during normal operations. Containment isolation can also be initiated

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manually from the main control room. Containment penetrations do not automatically reopen on the resetting of the isolation signal. See Subsection 6.2.3 for additional information.

(2)(xv) Containment Purging/Venting (NUREG-0933 Item II.E.4.4)

"Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions."

AP1000 Response:

Containment purging for the AP1000 is provided by the nonsafety-related containment air filtration system. The function of the system is to clean up the containment atmosphere to acceptable radiation levels during plant operation and prior to personnel entry. It can also be used for containment pressure equalization.

The containment air filtration system is designed to reliably isolate under accident conditions. There are two penetrations and two containment filtration subsystems for AP1000.

See Subsection 9.4.7 for additional information.

(2)(xvi) ECCS Actuation Cycles (NUREG-0933 Item II.E.5.1)

"Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both the expected transients and accidents)."

AP1000 Response:

This issue is applicable to Babcock & Wilcox designs only.

The AP1000 design uses the passive core cooling system to provide emergency reactor coolant inventory control and emergency decay heat removal. Component design criteria have been established for the number of actuation cycles for the passive core cooling system. The identified actuation cycles include inadvertent actuation, as well as the system response to expected plant trip occurrences, including overcooling events.

Automatic depressurization system operation is not expected for either design basis or best estimate overcooling events. See Subsection 3.9.1 for additional information.

(2)(xvii) Specific Accident Monitoring Instrumentation (NUREG-0737 Item II.F.1)

"Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples."

AP1000 Response:

AP1000 post-accident monitoring is described in Chapter 7.

AP1000 post-accident monitoring provides for indication of the specified parameters as follows:

- Containment pressure
- Containment water level

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- Containment radiation intensity (high level)
- Noble gas effluents - to ascertain reactor coolant system integrity

Other noble gas effluents are designated Type E variables and include information to permit the operators to:

- Monitor the habitability of the main control room
- Monitor plant areas where access may be required to service equipment necessary to monitor or mitigate the consequences of an accident
- Estimate the magnitude of release of radioactive materials through identified pathways
- Monitor radiation levels and radioactivity in the environment surrounding the plant

Subsection 11.5.5 has additional information on measurement of radioactive effluents and conformance with Regulatory Guide 1.97.

The AP1000 primary sampling system is designed to provide post accident sampling functions. See Subsection 9.3.3.1 for additional information on the post accident sampling system.

The human factors aspects of the AP1000 are discussed in Chapter 18.

(2)(xviii) Inadequate Core Cooling Instrumentation (NUREG-0737 Item II.F.2)

"Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs."

AP1000 Response:

The AP1000 reactor system includes instrumentation for detecting voids in the reactor vessel head and other reactor vessel inventory deficits that could lead to inadequate core cooling.

The available instrumentation includes core subcooling margin monitors, core exit thermocouples, pressurizer level indicators, and reactor coolant system hot leg level.

The AP1000 design uses hot leg level instrumentation to provide those functions necessary to support monitoring of the passive plant design features and to provide indication that the reactor vessel water level remains above the elevation of the fuel.

The AP1000 features that provide margin to or indication of inadequate core cooling include the following:

- A larger pressurizer than most current PWRs, with a pressurizer that is located above the reactor pressure vessel head
- No automatic power-operated relief valves
- An improved reactor vessel head venting capability
- A passive core cooling system
- A passive containment cooling system

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- No dependence on ac power to maintain adequate core and containment cooling
- Reactor coolant system hot leg level instrumentation
- Improved reactor system instrumentation
- Core subcooling monitoring

See Sections 6.3 and 7.5 for additional information.

(2)(xix) Post-Accident Monitoring Instrumentation (NUREG-0933 Item II.F.3)

"Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage."

AP1000 Response:

The AP1000 post-accident monitoring system was developed by using Regulatory Guide 1.97 as a guidance document.

Data used for post-accident monitoring is displayed either by the normal control room display system or by the PMS safety display subsystem.

The normal control room display system is used for display of nonsafety-related signals which are not required to be displayed by a qualified system. The PMS safety display subsystem provides for the display of signals which must be displayed by a qualified system.

The qualified data processing system is a microprocessor-based, safety-related system that provides instrumentation to monitor the plant variables and systems during and following an accident. The system consists of two independent, electrically isolated, physically separated divisions.

Additional details pertaining to this system are provided in the AP1000 response to issue (2)(xvii) and in Chapter 7.

(2)(xx) Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators (NUREG-0737 Item II.G.1)

"Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) level indicators are powered from vital buses, (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety, and (C) electric power is provided from emergency power sources."

AP1000 Response:

The AP1000 design does not include power-operated relief valves and their associated block valves from the reactor coolant system.

Pressurizer level indication is provided by instrumentation powered from the Class 1E dc and UPS system. The system provides safety-related, uninterruptible power for the Class 1E plant instrumentation, control, monitoring, and other vital functions, including safety-related components that are essential for safe shutdown of the plant.

The Class 1E direct current system is designed such that these critical plant loads are powered during emergency plant conditions when both onsite and offsite ac power sources are unavailable.

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See Chapter 7 and Section 8.3 for additional information.

(2)(xxi) Auxiliary Heat Removal Systems (NUREG-0933 Item II.K.1.22)

"Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable."

AP1000 Response:

Although this issue is applicable to BWRs only, there are some considerations for AP1000.

Following a loss of main feedwater for the AP1000, there are a number of plant systems that automatically actuate to provide decay heat removal. The startup feedwater system is a nonsafety-related system, that can be powered by the nonsafety-related diesel generators, and is automatically actuated and controlled by steam generator level.

For design basis events, the safety-related passive core cooling system includes a passive residual heat removal heat exchanger which automatically actuates to provide emergency core decay heat removal if the nonsafety-related systems are not available.

The AP1000 main control room meets the NRC guidelines for manual actuation of protective functions including those that are used in the event of a loss of normal feedwater.

See Sections 6.3 and 10.4 for additional information.

(2)(xxii) Failure Mode and Effects Analysis for Control Systems (NUREG-0933 Item II.K.2.9)

"Provide a failure modes and effects analysis of the integrated control system to include consideration of failures and effects of input and output signals to the integrated control system."

AP1000 Response:

This issue is applicable to Babcock & Wilcox plants only.

(2)(xxiii) Safety-Grade Anticipatory Reactor Trip (NUREG-0737 Item II.K.2.10)

"Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip."

AP1000 Response:

This issue is applicable to Babcock & Wilcox plants only.

The AP1000 trip logic includes an anticipatory reactor trip for loss of main feedwater using low steam generator water level. See Section 7.2 for additional information.

Since the AP1000 design does not include power-operated relief valves and their associated block valves in the reactor coolant system, the anticipatory reactor trip on turbine trip is not required for AP1000.

(2)(xxiv) Central Water Level Recording (NUREG-0933 Item II.K.3.23)

"Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements."

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AP1000 Response:

This issue is applicable to BWRs only.

(2)(xxv) Emergency Response Facilities (NUREG-0737 Item III.A.1.2)

"Provide an onsite technical support center, an onsite operational support center, and, for construction permit applications only, a nearsite emergency operations facility."

AP1000 Response:

The AP1000 provides for an onsite technical support center and an operational support center. See the figures in Section 1.2 for additional information on the location. The detailed design of the workstations and the associated man-machine interface for the technical support center and the operational support center is guided by the human factors engineering design process described in Chapter 18. The offsite emergency response facility is discussed in Subsection 18.2.6.

(2)(xxvi) Leakage Control Outside Containment (NUREG-0737 Item III.D.1.1)

"Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency."

AP1000 Response:

As described in issue (2)(vii), the safety-related AP1000 passive systems do not recirculate radioactive fluids outside of containment following an accident. A nonsafety-related system can be used to recirculate coolant outside of containment following an accident, but this system is not operated when high containment radiation levels exist.

(2)(xxvii) In-Plant Monitoring (NUREG-0737 Item III.D.3.3)

"Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions."

AP1000 Response:

Area radiation monitors (ARMs) are provided to supplement the personnel and area radiation survey provisions of the AP1000 health physics program described in Section 12.5 and to comply with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, and Regulatory Guides 1.97, 8.2, and 8.8. In addition to the installed detectors, periodic plant environmental surveillance is established.

(2)(xxviii) Control Room Habitability (NUREG-0737 Item III.D.3.4)

"Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID-14844 source term release, and make necessary design provisions to preclude such problems."

AP1000 Response:

Normally, a nonsafety-related HVAC system keeps the AP1000 main control room slightly pressurized to prevent infiltration of air from other plant areas. During accident conditions, a safety-related isolation of the main control room is automatically actuated.

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Upon the loss of nonsafety-related ac power, the main control room environment is sufficient to protect the operators and support the man-machine interfaces necessary to establish and maintain safe shutdown conditions for the plant following postulated design basis accident conditions. The sources are based on the core activity release model from Regulatory Guide 1.183, which supersedes the TID-14844 source term assumptions as reflected in Regulatory Guide 1.4.

The main control room is sealed with safety-related connections to a safety-related compressed air breathing source. This compressed air system provides continued pressurization and a source of fresh air for operator habitability. The air supply is sized to last for 72 hours following an accident. It is expected that the onsite nonsafety-related normal HVAC system will be operational before the installed compressed air supply is exhausted.

The nonsafety-related HVAC system, equipped with a refrigeration-type air conditioning unit, normally provides main control room cooling. This equipment is powered from the onsite diesel generators. If the normal HVAC system is not available, outside air is not allowed into the main control room, and the safety-related compressed air storage system is actuated.

(3)(i) Industry Experience (NUREG-0737 Item I.C.5)

"Provide administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant."

AP1000 Response:

AP1000 design engineers are continually involved in reviewing industry experiences from sources such as NRC Bulletins, Licensee Event Reports, NRC request for information letters to holders of operating licenses for nuclear power reactors, Federal Register information, and generic letters. Lessons learned experience was incorporated in the AP600 through the Westinghouse participation in developing Volume III of the ALWR Utility Requirements Document and participation in the ALWR Utility Steering Committee activities. The AP1000 design is closely based on the AP600. See Subsection 1.9.5.5 for additional information.

(3)(ii) Quality Assurance List (NUREG-0933 Item I.F.1)

"Ensure that the quality assurance list required by Criterion II, Appendix B, 10 CFR Part 50 includes all structures, systems and components important to safety."

AP1000 Response:

The AP1000 Quality Assurance Plan is described in Chapter 17. Structures, systems, and components are classified as described in Section 3.2.

(3)(iii) Quality Assurance Program (NUREG-0737 Item I.F.2)

"Establish a quality assurance program based on consideration of: (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including Quality Assurance personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining Quality Assurance programmatic requirements; (E) establishing qualification requirements for Quality Assurance and Quality Control personnel; (F) sizing the Quality Assurance staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a Quality Assurance role in design and analysis activities."

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AP1000 Response:

The AP1000 Quality Assurance Plan described in Chapter 17 meets the requirements of issue 1.F.2.

(3)(iv) Dedicated Containment Penetrations (NUREG-0660 Item II.B.8)

"Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system."

AP1000 Response:

The containment analysis for the AP1000, including PRA and severe accident assessments, demonstrate that the containment, with its passive heat rejection capability, does not need a filtered vent to prevent overpressurization.

The 36-inch diameter containment air filtration system penetration provided for AP1000 meets the requirement of 10 CFR 50.34(f)(3)(iv). See Figure 9.4.7-1, sheet 1, note 6, for additional information.

(3)(v) Containment Design (NUREG-0660 Item II.B.8)

"Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2 Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100 percent fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above, appropriate for each type of containment, will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraph (f)(3)(v)(A)(1) and (f)(3) (v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. . . .

(B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, Service Load Category), (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting."

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AP1000 Response:

The AP1000 containment vessel is designed to meet the requirements of the ASME Code, Section III, Division I, Subsection NE. A severe accident containment analysis is conducted to support the design effort. The results of the analysis are fission product source terms and plant thermal-hydraulic response for each of the accident sequences chosen to be representative of the plant damage states determined in level 1 PRA analysis.

Results of the analysis indicate that containment failure is not predicted for cases in which the passive containment cooling system cooling water is available. The hydrogen igniter system controls hydrogen and mitigates threats to the containment due to hydrogen.

See Section 6.2 for additional information.

(3)(vi) Hydrogen Recombiners (NUREG-0737 Item II.E.4.1)

"For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere."

AP1000 Response:

Since external hydrogen recombiners are not provided for the AP1000, this requirement is not applicable. See Section 6.2 for additional information.

(3)(vii) Management Plan (NUREG-0933 Item II.J.3.1)

"Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort."

AP1000 Response:

The AP1000 design team has developed a management plan for the AP1000 project which consists of a properly structured organization with open lines of communication, clearly defined responsibilities, well-coordinated technical efforts, and appropriate control channels. The procedures to be used in the construction, startup, and operation phases of the plant are provided in accordance with the Master Plan and Procedure Development Process identified in APP-GW-GLR-040 (Reference 72).

1.9.4 Unresolved Safety Issues and Generic Safety Issues

Proposed technical resolutions of Unresolved Safety Issues and medium- and high-priority Generic Safety Issues, as identified in NUREG-0933, Reference 3 are required for new plants as part of the NRC policy on severe accidents and are required for design certification in accordance with 10 CFR 52.47(a)(21).

The current program for identifying and establishing the priority of open safety issues is summarized in NUREG-0933. This program provides for the prioritization and tracking of previously categorized

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Unresolved Safety Issues and Generic Safety Issues, New Generic Issues, TMI Action Plan Items Under Development, and Human Factors Program Plan Issues.

The following subsection reviews each of the NUREG-0933 safety issues and identifies the safety issues that are applicable to the AP1000. For each of these issues guidance is provided on how the issue is addressed for the AP1000.

1.9.4.1 Review of NRC List of Unresolved Safety Issues and Generic Safety Issues

Applicants for design certification are required by 10 CFR 52.47(a)(21) to identify:

"Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues which are identified in the version of NUREG-0933 current on the date six months prior to application and which are technically relevant to the design."

NUREG-0933, "A Prioritization of Generic Safety Issues," through Supplement 25 identifies hundreds of issues. The issues tabulated in Supplement 25 were reviewed to determine which issues are technically relevant to the AP1000 design. In this review process, the following screening criteria were applied:

a. Issue has been prioritized as **Low**, **Drop**, or has not been prioritized.

b. Issue is not an AP1000 design issue. Issue is applicable to GE, B&W, or CE designs only.

c. Issue resolved with no new requirements.

d. Issue is not a design issue (Environmental Issue, Licensing Issue, Regulatory Impact Issue, or covered in an existing NRC program).

e. Issue superseded by one or more issues.

f. Issue is not an AP1000 design certification issue. Issue is applicable to NTOL plants only, responsibility of combined license applicant, or issue is limited to current generation operating plants.

Issues meeting one or more of the preceding screening criteria were screened out of the review process as issues that are not applicable to the AP1000 design. The remaining issues fall into one of the following two categories:

g. Issue is resolved by establishment of new regulatory requirements and/or guidance.

h. Issue is unresolved pending generic resolution (e.g., prioritized as **High**, **Medium**, or possible resolution identified).

Table 1.9-2 identifies the results of the screening review. For those issues identified as relevant to the AP1000 design (i.e., issues screened as **g** or **h**), Table 1.9-2 identifies the subsection that addresses the issue.

Table 1.9-2 addresses the second un-numbered COL Information Item identified at the end of Table 1.8-2, "Unresolved Safety Issues and Generic Safety Issues." As such, Table 1.9-2 lists those issues identified by Note "d," which apply to other than design issues, Note "f," which apply either to resolution of Combined License (COL) Information Items or to nuclear power plant operations issues, Note "h," which apply to issues unresolved pending generic resolution at the time of submittal of the AP1000 DCD, and any new Unresolved Safety Issues and Generic Safety Issues that have been

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included in NUREG-0933 (through supplement 30) since the DCD was developed. Many of these have since been resolved and incorporated into the applicable licensing regulations or guidance (e.g., the standard review plans). These resolved items (as indicated by NUREG-0933) are identified only as "Resolved per NUREG-0933." Many others are not in the list of items in NUREG-0933 Appendix B identified as applicable to new plants. These items are identified only as "Not applicable to new plants." For the remaining items, the table provides the sections that address the topic.

1.9.4.2 AP1000 Resolution of Unresolved Safety Issues and Generic Safety Issues

1.9.4.2.1 TMI Action Plan Issues

TMI Action Plan issues that were not incorporated in 10CFR50.34(f) are addressed in the following. Those issues incorporated into 10CFR50.34(f) are addressed in Subsection 1.9.3.

I.D.5(2) Plant Status and Post-Accident Monitoring Discussion:

TMI action plant item I.D.5(2) addresses the need to improve the operators' ability to prevent, diagnose and properly respond to accidents. The emphasis is on the information needs (i.e., indication of plant status) of the operator. This issue was resolved with the issuance of Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident."

AP1000 Response:

The AP1000 conforms to and meets the intent of Regulatory Guide 1.97. Regulatory Guide 1.97 provides the requirements for post-accident monitoring of nuclear reactor safety parameters, including plant process parameters important to safety and the monitoring of effluent paths and plant environs for radioactivity. These guidelines include definition and categorization of plant variables that are available to the main control room operators for monitoring the plant safety status following a design basis event.

For the AP1000, an analysis is conducted to identify the appropriate variables and to establish the appropriate design basis and qualification criteria for instrumentation used by the operator for monitoring conditions in the reactor coolant system, the secondary heat removal system, the containment, and the systems used for attaining a safe shutdown condition, as discussed in Section 7.5.

This instrumentation is used by the operator to monitor and maintain the safety of the plant during operating conditions, including anticipated operational occurrences and accident and post-accident conditions. A set of plant parameters identified according to the Regulatory Guide 1.97 guidelines are processed by the qualified data processing system (QDPS) and displayed by the safety displays of the PMS safety display subsystem, which is discussed in Section 18.8. The verification and validation (V&V) of the safety displays complies with the V&V process described in Section 18.11.

I.D.5(3) On-Line Reactor Surveillance System

Discussion:

TMI action plan item I.D.5(3) addresses the benefit to plant safety and operations of continuous on-line automated surveillance systems. Continuous on-line surveillance systems that automatically monitor reactors can assist plant operations by providing diagnostic information which can predict anomalous behavior.

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Various methods of on-line reactor surveillance have been used, including neutron noise monitoring in boiling water reactors (BWRs) to detect internals vibration, and pressure noise surveillance at TMI-2 to monitor primary loop degasification.

AP1000 Response:

The AP1000 reactor coolant pressure boundary is monitored for leaks from the reactor coolant and associated systems by a variety of components located in multiple systems. The leak detection system provides information permitting the plant operators to take corrective action if any detected leakage exceeds technical specifications. The leak detection system is designed according to the requirements of 10 CFR 50, Appendix A, General Design Criterion 30. The system provides a means to detect and, to the extent practical, to identify the source of the reactor coolant pressure boundary leakage. Subsection 5.2.5 provides further discussion of leak detection.

A digital metal impact monitoring system (DMIMS) monitors the reactor coolant system for the presence of loose metallic parts. This system conforms with the guidance provided in Regulatory Guide 1.133, Rev. 1, May 1981. An advanced microprocessor-based system, employing digital technology, automatically actuates audible and visual alarms if a signal exceeds the preset alarm level.

I.F.1 Expand Quality Assurance List

Discussion:

Item I.F.1 addressed the issue of systems that are "important to safety" that are not on the Quality Assurance List. The suggestion was made that equipment important to safety be ranked and that ranking used to determine systems that should be added to the Quality Assurance List. This approach has not been implemented by the NRC on either a generic or cases-by case basis. In NUREG-0933 this item was classified as resolved with no additional requirements established.

AP1000 Response:

The requirements of 10 CFR Appendix B apply to safety-related systems and components. See Subsection 3.2.2 for a discussion of the AP1000 equipment classification system and the associated quality assurance requirements, including requirements for nonsafety-related systems.

I.G.1 Training Requirements

Discussion:

Item I.G.1 included the issue of natural circulation testing for use as input into operator training.

AP1000 Response:

For the AP1000, natural circulation heat removal using the steam generators is not safety-related, as in current plants. This safety-related function is performed by the passive residual heat removal system. Testing of the passive residual heat removal system meets the intent of the requirement to perform natural circulation testing and the results of this testing is factored into the operator training.

For the AP1000, the tests outlined below are contained in the AP1000 initial test plan and demonstrate the effectiveness of natural circulation cooling.

1. Not Used.

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2. After fuel loading, but prior to criticality, with the reactor system at no-load operating temperature and pressure and all reactor coolant pumps operating, the depressurization rate is determined by de-energizing the heaters and pressure is further reduced through use of sprays (See Subsection 14.2.10.1.19).

3. After criticality is achieved and the plant is at ~ 3% power, the plant is placed in a natural circulation mode by tripping all reactor coolant pumps and observing the plant response using the steam generators (See Subsection 14.2.10.3.6). This test is performed for the first plant only.

4. A PRHR natural circulation test (see Subsection 14.2.10.4.29) is performed using decay heat generated following reactor operation. The plant is operated at power for a sufficient duration to generate decay heat. The reactor is then tripped, followed by the tripping of all reactor coolant pumps and placing the PRHR heat exchanger into service. The plant response is observed using the PRHR and IRWST as the primary heat sink. This test is performed for the first plant only.

5. A loss-of-offsite power test is performed with the plant at minimum power level supplying normal house loads. The turbine is tripped and the plant is placed in a stable condition using batteries and the diesel generator (See Subsection 14.2.10.4.26).

6. Data obtained from the first plant only natural circulation tests using the steam generators and PRHR is provided for operator training on a plant simulator at the earliest opportunity.

This response as modified for the AP1000 design is consistent with the response to NUREG-0737, action item I.G.1 which provided a proposal for low power testing of existing and future Westinghouse pressurized water reactors in Attachment 4 to letter NS-EPR-2465 from Westinghouse (E. P. Rahe) to the NRC (H. R. Denton) dated July 8, 1981.

I.G.2 Scope of Test Program

Discussion:

TMI Action Plan Items I.G.2 recommended additional testing during preoperational and startup programs to search for anomalies in a plants response to transients. The Standard Review Plan, Section 14 was revised to provide additional guidance for preoperational and startup test programs.

AP1000 Response:

The program plan for preoperational and startup testing of the AP1000 is in Section 14.2. This section addresses the Standard Review Plan, Section 14. The conformance with Standard Review Plan, Section 14 is outlined in AP1000 Compliance with SRP Acceptance Criteria, WCAP-15799.

II.E.1.3 Update Standard Review Plan and Develop Regulatory Guide

Discussion:

This item was a requirement to update Section 10.4.9 of the Standard Review Plan to address the requirements of Items II.E.1.1 and II.F.1.2 for auxiliary feedwater systems. Standard Review Plan 10.4.9 was revised and this issue is classified as resolved.

AP1000 Response:

The AP1000 does not have a safety-related auxiliary feedwater system. For conformance of the AP1000 with Items II.E.1.1 and II.E.1.2 see the write-up for (1)(ii) and (2)(xii) in Subsection 1.9.3. For conformance with Standard Review Plan Section 10.4.9 see WCAP-15799.

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II.E.6.1 Test Adequacy Study

Discussion:

This item was intended to establish the adequacy of requirements for safety-related valve testing. Subsequent to this item, expanded requirements were written into the ASME OM Code for valve testing.

AP1000 Response:

The AP1000 is designed for an in-service test program in accordance with the ASME OM Code. See Subsection 3.9.6 for additional information on the in-service testing program plan.

II.K.1(10) Review and Modify Procedures for Removing Safety-related Systems from Service

Discussion:

This item required operating plants to review and modify (as required) their procedures for removing safety-related systems from service to assure operability status is known.

AP1000 Response:

Section 13.5 describes the AP1000 procedure development, preparation, and responsibility.

II.K.1(13) Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items.

Discussion:

This item required that operating plants propose technical specification changes to address Bulletin items.

AP1000 Response:

The AP1000 Technical Specifications are based on and were reviewed against the Westinghouse Standard Technical Specifications, which incorporated the requirements of the bulletins for the TMI Action Plan.

II.K.1(17) Trip PZR Level Bistable So That Low Pressure Will Initiate Safety Injection

Discussion:

This item required operating licensees and operating license applicants with Westinghouse designed nuclear steam supply systems to trip the pressurizer level bistable so that the pressurizer low pressure (rather than the pressurizer low pressure and pressurizer low level coincidence) would initiate safety injection.

AP1000 Response:

This issue does not apply to AP1000. The AP1000 does not rely on coincident low pressurizer pressure and low pressurizer level for actuation. See Section 6.3 for a discussion of actuation of the passive core cooling system.

II.K.1(24) Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and Reactor Coolant Pump Trip

Discussion:

This item requires analyses to provide the basis for the comparison of analytical methods.

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AP1000 Response:

The analyses documented in Chapter 15 cover a range of small break sizes. The AP1000 automatically trips the reactor coolant pump on an S signal. The need to look at time lapses between reactor trip and pump trip is not required.

II.K.3(5) Automatic Trip of Reactor Coolant Pumps

Discussion:

This item requires that operating plants and operating plant applicants study the need for automatic trip of reactor coolant pumps and to modify procedures of designs as appropriate.

AP1000 Response:

The AP1000 design provides for an automatic trip of the reactor coolant pumps on actuation of the passive core cooling system. This trip is provided to prevent reactor coolant pump interaction with the operation of the core makeup tank. See Section 6.3 for additional information.

II.K.3(9) Proportional Integral Derivative Controller Modification

Discussion:

TMI action plan item II.K.3(9) required all Westinghouse plants to raise the interlock bistable trip setting to preclude derivative action from opening the PORVs.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not include power-operated relief valves. See Subsections 5.1.2 and 5.2.2 for additional information.

1.9.4.2.2 Task Action Plan Items

A-1 Water Hammer

Discussion:

Generic Safety Issue A-1 was raised after the occurrence of various incidents of water hammer that involved steam generator feedrings and piping, emergency core cooling systems, residual heat removal systems, containment spray, service water, feedwater, and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage has been relatively minor and involved pipe hangers and restraints. However, several incidents have resulted in piping and valve damage. This item was originally identified in NUREG-0371, (Reference 4) and was later determined to be an Unresolved Safety Issue.

AP1000 Response:

Specific sections of the Standard Review Plan (NUREG-0800) address criteria for mitigation of water hammer concerns. The applicable Standard Review Plan sections as well as information provided in NUREG-0927 (Reference 5) were reviewed. The AP1000 meets the water hammer provisions as specified. The discussion that follows provides a brief description of selected systems identified as being subject to water hammer occurrences and special design features that mitigate or prevent water hammer damage.

Design features are incorporated as appropriate to prevent water hammer damage in applicable systems including steam generator feedrings and piping, passive core cooling system, passive residual heat removal system, service water system, feedwater system, and steam lines.

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Water hammer issues are considered in the design of the AP1000 passive core cooling system. The passive core cooling system design includes a number of design features specifically to prevent or mitigate water hammer.

The automatic depressurization system operation uses multiple, sequenced valve stages to provide a relatively slow, controlled depressurization of the reactor coolant system, which helps to reduce the potential for water hammer.

Once the depressurization is complete, gravity injection from the in-containment refueling water storage tank is initiated by opening squib valves and then check valves, which reposition slowly. Gravity injection flow actuates slowly, without water hammer, as the pressure differential across the gravity injection check valves equalizes, and the valves open and initiate flow.

The passive residual heat removal heat exchanger is normally aligned with an open inlet valve and closed discharge valves. This alignment keeps the system piping at reactor coolant system pressure, preventing water hammer upon initiation of flow through the heat exchanger. Instrumentation is provided at the system high point to detect a void in the system.

The core makeup tanks are normally aligned with an open inlet line from the reactor coolant cold leg to keep the tanks at reactor coolant system pressure. This alignment keeps the system piping at reactor coolant pressure, preventing water hammer upon initiation of flow through the tank. In addition, instrumentation is provided at each high point to detect voids within the system. Section 6.3 provides additional information on the passive core cooling system.

The potential for water hammer in the feedwater line is minimized by the improved design and operation of the feedwater delivery system. The steam generator features include introducing feedwater into the steam generator at an elevation above the top of the tube bundles and below the normal water level by a top discharge spray tube feedring. The feedring is welded to the feedwater nozzle to limit the potential for inadvertent draining. The layout of the feedwater line is consistent with industry standard recommendations to reduce the potential of a steam generator water hammer.

The startup feedwater system is a nonsafety-related system that provides feedwater during normal plant startup, shutdown, and hot standby. The startup feedwater line is separate from the main feedwater line and therefore does not contribute to the potential of water hammer in the feedwater piping or steam generator feedring.

The main steam line drains are designed to remove accumulated condensate from the main steam lines and to maintain the turbine bypass header at operating temperature during plant operation. The system is designed to accommodate drain flows during startup, shutdown, transient, and normal operation to protect the turbine and the turbine bypass valves from water slug damage.

A-2 Asymmetric Blowdown Loads on Reactor Primary Coolant Systems

Discussion:

Generic Safety Issue A-2 pertains to asymmetric loadings that could act on a pressurized water reactor's primary system as the result of a postulated double-ended rupture of the piping in the primary coolant system. The magnitude of these loads is potentially large enough to damage the supports of the reactor vessel, the reactor internals, and other primary components of the system. Therefore, the NRC initiated a generic study to develop criteria for an evaluation of the response of the primary systems in pressurized water reactors to these loads.

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AP1000 Response:

The use of mechanistic pipe break criteria permits elimination of the evaluation of dynamic effects of sudden circumferential and longitudinal pipe breaks in the structural analysis of structures, systems, and components. General Design Criterion 4 allows the use of analyses to eliminate from the design basis the dynamic effects of pipe ruptures postulated at locations defined in Subsection 3.6.2. Dynamic effects include jet impingement, pipe whip, jet reaction forces on other portions of the piping and components, subcompartment pressurization including reactor cavity asymmetric pressurization transients, and traveling pressure waves from the depressurization of the system.

The AP1000 reactor coolant loop and pressurizer surge line are designed in accordance with mechanistic pipe break criteria. In addition, other high energy ASME Code, Section III, Class 1 and 2 piping of 6 inches and greater nominal diameter is evaluated against leak-before-break criteria. The evaluation methodology is described in Subsection 3.6.3 and Appendix 3B.

A-3 Steam Generator Tube Integrity

Discussion:

Pressurized water reactor steam generator tube integrity is subject to various degradation mechanisms, including corrosion-induced wastage, cracking, reduction in tube diameter, denting, (which leads to primary side stress corrosion cracking), vibration-induced fatigue cracks, and wear or fretting due to loose parts in the secondary system. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Steam generator tube integrity concerns for the three steam generator suppliers, Westinghouse, Combustion Engineering, and Babcock and Wilcox, are addressed by an integrated NRC program for Generic Safety Issues A3, A4, and A5. This program addresses the areas of steam generator integrity, plant systems response, human factors, radiological consequences, and the response of various organizations to a steam generator tube rupture.

AP1000 Response:

The AP1000 steam generators are designed in accordance with the recommendations of Generic Letter 85-02 and NUREG-0844 (References 6 and 7). The AP1000 steam generator is equipped with a number of features to enhance steam generator tube performance and reliability. These features are described in Subsection 5.4.2.

A-9 Anticipated Transients Without Scram

Discussion:

Generic Safety Issue A-9 was resolved with the publication of 10 CFR 50.62. This regulation sets forth the requirements for reduction of risks from anticipated transients without scram.

AP1000 Response:

The AP1000 complies with the requirements of 10 CFR 50.62 except that the AP1000 does not have a safety-related auxiliary feedwater system. In lieu of the automatic initiation of the auxiliary feedwater system under conditions indicative of an ATWS as required by 10 CFR 50.62 (c)(1), the AP1000 automatically initiates the passive residual heat removal system as discussed in Section 6.3.

A discussion of the AP1000 design features used to address the probability of an ATWS is presented in Subsection 1.9.5 and Section 7.7.

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A-11 Reactor Vessel Materials Toughness

Discussion:

Generic Issue A-11 addresses a concern with the reduction of reactor vessel fracture toughness as plants accumulate more and more service time. 10 CFR 50, Appendix G provides requirements for reactor vessel material toughness.

AP1000 Response:

The AP1000 reactor vessel design complies with the requirements of 10 CFR 50, Appendix G and includes numerous features to reduce neutron fluence, enhance material toughness at low temperature and eliminate weld seams in critical areas. Material requirements are provided in Subsection 5.3.2. Pressure and temperature limits are provided in Subsection 5.3.3.

A-12 Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

Discussion:

Generic Safety Issue A-12 addresses a concern with the potential for lamellar tearing of steam generator and RCP support material. NUREG-0577 (Reference 8) categorizes operating plants relative to the adequacy of the plant's steam generator and reactor coolant pump supports with respect to fracture toughness.

AP1000 Response:

The steam generator and reactor coolant pump supports are described in Subsection 5.4.10. The supports are designed in accordance with subsection NF of Section III of the ASME Code. Design and fabrication of these supports in accordance with Subsection NF requirements provide acceptable fracture toughness of materials, and conform with NUREG-0577.

A-13 Snubber Operability Assurance

Discussion:

Generic Issue A-13 addresses snubber operability concerns. Snubbers are utilized primarily as seismic and pipe whip restraints at nuclear power plants. Their safety function is to operate as rigid supports for restraining the motion of attached systems or components under rapidly applied load conditions such as earthquakes, pipe breaks, and severe hydraulic transients.

Operating experience reports show that a substantial number of snubbers have leaked hydraulic fluid and that the rejection rate from functional testing and inspection is high. This has led to an NRC and ACRS concern regarding the effect of snubber malfunctions on plant safety.

AP1000 Response:

The use of snubbers is minimized in the AP1000. Gapped support devices, leak-before-break considerations, and state-of-the-art piping analysis methods are used to minimize the use of snubbers. Snubbers applied in safety-related applications are constructed to ASME Code, Section III, Subsection NF as discussed in Subsection 3.9.3.4.3.

A-17 Systems Interactions in Nuclear Power Plants

Discussion:

This item addresses the potential systems interactions among systems including safety-related and nonsafety-related structures, systems, and components. There can be unintended and unrecognized dependencies among structures, systems, and components. A number of specific types of

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interactions have been addressed in other generic safety issues and NRC staff activities. These include guidance for inclusion of internal flooding in the IPE program, requirements that address seismically-induced systems interactions, and evaluation of electric power supplies for electric power reliability. NUREG-0933 classifies this item as resolved with no new requirements.

AP1000 Response:

In addition to addressing the specific system interaction guidance mentioned above, the AP1000 was the subject of a systematic evaluation of potential adverse systems interactions documented in WCAP-15992, "AP1000 Adverse Systems Interactions Evaluation Report" (Reference 69).

A-24 Qualification of Class 1E Safety-Related Equipment

Discussion:

Generic Issue A-24 was resolved with the publication of 10 CFR 50.49, prescribing aging and testing for synergistic effects. The NRC has also issued Revision 1 to Regulatory Guide 1.89 for comment. The proposed revision describes a method acceptable to the NRC staff to demonstrate compliance with the requirements of 10 CFR 50.49.

AP1000 Response:

The AP1000 environmental qualification methodology described in Appendix 3D is based on the generic Westinghouse qualification program approved by the NRC. The Westinghouse methodology addresses the requirements of General Design Criteria 4 and 10 CFR 50.49, as well as the guidance of Regulatory Guide 1.89 and IEEE Standard 323-1974. See Appendix 1A and Reference 9.

A-25 Non-Safety Loads on Class 1E Power Sources

Discussion:

Generic Issue A-25 addresses whether nonsafety-related loads should be allowed to share Class 1E power sources with safety-related plant systems. Past regulatory practice has allowed the connection of nonsafety-related loads in addition to the required safety loads to Class 1E power sources by imposing some restrictions. The purpose of this issue is for the NRC to determine whether the reliability of the Class 1E power sources is significantly affected by the sharing of safety and nonsafety-related loads.

The NRC considers this issue as technically resolved with the issuance of Revision 2 to Regulatory Guide 1.75. This regulatory guide includes special requirements for connection of nonsafety-related loads to a Class 1E source.

AP1000 Response:

The AP1000 conforms with the criteria of Regulatory Guide 1.75 with minor exceptions (see Appendix 1A and IEEE 384-1974). The AP1000 safety-related power source is the Class 1E dc and UPS system, which supplies power to the ac inverters for the plant instrumentation and control systems. The system also provides power to dc loads associated with the four protection channels and the accident monitoring system. Non-Class 1E loads powered from Class 1E sources are limited to loads that need connection to a reliable power source. No Credible failure of non-Class 1E equipment or systems will degrade the Class 1E system below an acceptable level. Subsection 8.3.2.1.1 provides a discussion on the Class 1E power source.

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A-26 Reactor Vessel Pressure Transient Protection

Discussion:

Generic Issue A-26 addresses the need to provide reactor vessel overpressure protection whenever plants are in a cold shutdown condition. Branch Technical Position RSB 5-2 establishes the current NRC criteria for a low-temperature overpressurization protection system.

AP1000 Response:

The AP1000 conforms with the criteria established in Branch Technical Position RSB 5-2. The AP1000 pressurizer is sized to accommodate most pressure transients. Overpressure protection for the reactor coolant system is provided by either the pressurizer safety valves or the normal residual heat removal relief valves, as described in Subsection 5.2.2.

A-28 Increase in Spent Fuel Pool Storage Capacity

Discussion:

Generic Issue A-28 addresses the safety significance of damage to spent fuel, primarily from a lack of adequate cooling, that could result in the release of radioactivity.

AP1000 Response:

The AP1000 incorporates the NRC criteria. The heat load is evaluated for the spent fuel storage capacity.

A-29 Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Description

This item addresses potential methods to reduce vulnerability to sabotage. The NRC staff concluded that existing requirements dealing with plant physical security, controlled access to vital areas, screening for reliable personnel appear to be effective. This item was resolved with no new requirements.

AP1000 Response:

The passive systems in the AP1000 provided to mitigate the effects of potential accidents may have an inherent advantage when considering potential acts of sabotage compared to the active systems in operating plants. The AP1000 includes provisions for access control to the vital area. The provisions for security are discussed in the AP1000 Security Design Report and outlined in Section 13.6.

A-31 Residual Heat Removal Requirements

Discussion:

Generic Issue A-31 addresses the desire for plants to be able to go from hot-standby to cold-shutdown conditions (when this is determined to be the safest course of action) under an accident condition. The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident has been typically interpreted as achieving a hot standby condition (the reactor is shut down, but system temperature and pressure are at or near normal operating values). There are events that require eventual cooldown and long-term cooling to perform inspection and repairs.

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AP1000 Response:

The AP1000 employs safety-related core decay heat removal systems that establish and maintain the plant in a safe, stable condition following design basis events. It is not necessary that these passive systems achieve cold shutdown as defined by Regulatory Guide 1.139.

The AP1000 complies with General Design Criteria 34 by using a more reliable and simplified system design. The passive core cooling system is employed for both hot-standby and long-term cooling modes. Hot-standby conditions are achieved immediately and a temperature of 420°F is reached within 36 hours as discussed in Subsection 19E.4.10.2. Reactor pressure is controlled and can be reduced to about 250 psig. The passive residual heat removal system provides a closed cooling system to maintain long-term core cooling. Passive feed and bleed cooling, using the passive injection features for the feed and the automatic depressurization system for bleed, provides safety-related cooling capability. See Section 7.4 for a discussion of safe shutdown and Section 6.3 for a description of the passive core cooling system.

Since the passive core cooling system maintains safe conditions indefinitely, cold shutdown is necessary only to gain access to the reactor coolant system for inspection or repair. On the AP1000, cold shutdown is accomplished by using non-safety-related systems. These systems are highly reliable. They have similar redundancy as current generation safety-related systems and are supplied with ac power from either onsite or offsite sources. See Subsection 5.4.7 for a description of the normal residual heat removal system and Subsection 7.4.1.3 for a discussion of cold shutdown achieved by use of non-safety-related systems.

A-35 Adequacy of Offsite Power Systems

Discussion:

Generic Issue A-35 addresses the susceptibility of safety-related electric equipment to offsite power source degradation. The NRC considers this issue as technically resolved with the issuance of the Standard Review Plan, Section 8.3.1 criteria specified in Appendix A, Branch Technical Position BTP PSB 1, "Adequacy of Station Electric Distribution System Voltages."

AP1000 Response:

The AP1000 ac power system is discussed in Sections 8.1 through 8.3. The AP1000 does not require any ac power source to achieve and maintain safe shutdown.

A-36 Control of Heavy Loads Near Spent Fuel

Discussion:

Generic Issue A-36 addresses the need to review requirements, facility designs, and Technical Specifications regarding the movement of heavy loads near spent fuel. The NRC has documented its technical position on this issue in NUREG-0612 (Reference 10) and that issued Standard Review Plan, Section 9.1.5, which includes NUREG-0612 as a part of the review plan.

AP1000 Response:

The AP1000 design conforms to NUREG-0612 and Standard Review Plan, Section 9.1.5. Light load handling systems are described in Subsection 9.1.4, and overhead heavy-load handling systems are described in Subsection 9.1.5.

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A-39 Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments

Discussion:

Generic Issue A-39 addresses operation of BWR primary system pressure relief valves whose operation can result in hydrodynamic loads on the suppression pool retaining structures or those structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching due to high local pool temperatures. This USI was resolved with the issuance of SRP Section 6.2.1.1.C and a series of NUREG reports.

Generic Issue A-39 is not directly applicable to the AP1000. However, the AP1000 in-containment refueling water storage tank (IRWST) has some functional similarity to a suppression pool when the automatic depressurization system (ADS) is actuated.

AP1000 Response:

The AP1000 in-containment refueling water storage tank design includes consideration of loads due to automatic depressurization system operation. The effect of hydrodynamic loads is addressed in Subsection 3.8.3.4.2.

A-40 Seismic Design Criteria - Short Term Program

Discussion:

Generic Issue A-40 addresses a desire to identify and quantify conservatism in the seismic design process. The Standard Review Plan, Section 3.7 provides clarification of development of site-specific spectra, justification for use of single synthetic time-history by power spectral density function, location and reductions of input ground motion for soil-structure interaction, and design of above-ground vertical tanks. The revised provisions are used for margin studies and re-evaluations or individual plant examination for external events.

AP1000 Response:

The AP1000 conforms to the criteria outlined in the Standard Review Plan, Section 3.7. The seismic design criteria and seismic evaluation methodology are described in Section 3.7.

The AP1000 employs generic, enveloping seismic design criteria and applies established seismic evaluation methodology that complies with current regulations and regulatory guidance. For sites having specific characteristics outside the range of the selected parameters, the AP1000 is evaluated to demonstrate acceptability to the site-specific characteristics.

A-43 Containment Emergency Sump Performance

Discussion:

Generic Issue A-43 addresses technical concerns as follows:

- Pressurized water reactor sump (or boiling water reactor residual heat removal system suction intake) hydraulic performance under post-loss-of-coolant accident adverse conditions resulting from potential vortex formation, air ingestion, and subsequent pump failure

- The possible transport of large quantities of insulation debris generated by a loss-of-coolant accident resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling

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- The capability of residual heat removal and containment spray system pumps to continue pumping when subjected to possible air, debris, or other effects, such as particulate ingestion on pump seal and bearing systems

AP1000 Response:

Air ingestion, vortexing, and debris blockage are not significant concerns for the AP1000. Containment recirculation includes sump screens that conform to the criteria specified in Regulatory Guide 1.82. The recirculation screens have a large cross-sectional area to reduce the fluid flow velocity through the screen and to provide a large screening area to accommodate accumulated debris. A horizontal plate located above the recirculation screens precludes debris being deposited in the water directly adjacent to the screens. Pipes that are the subject of loss of coolant pipe breaks and in the vicinity of these breaks use reflective metallic insulation (or a suitable equivalent, as demonstrated per the criteria in Subsection 6.3.2.2.7.1) to preclude the generation of fibrous insulation debris. See Subsection 6.3.2.2.7 for additional information on the design of the screens and limits on use of fibrous insulation.

Since the AP1000 design does not use pumps to provide safety injection flow, the passive core cooling system injection flow rates are substantially lower than those for plants with pumped injection flow. This results in lower fluid flow velocities through the screens, reducing the potential to draw debris into the sump screens.

The containment recirculation sump piping inlet is located slightly above the compartment floor, which is substantially below the expected flood-up water level. This precludes air ingestion in the piping since recirculation does not initiate until the flood-up water level is well above the piping inlet.

The elimination of pumps also eliminates concerns about the effects on safety injection capability for vortexing, air ingestion, and blockage effects on pump net positive suction head.

The AP1000 includes the capability to use nonsafety-related normal residual heat removal pumps to take a suction from the containment recirculation sump to provide reactor coolant system injection. The sump screen design addresses concerns with screen debris, vortexing, and air ingestion.

Section 6.3 provides additional information on the operation of the passive core cooling system. Appendix 1A describes conformance with Regulatory Guide 1.82. Section 6.2 provides additional information on the containment recirculation sump.

A-44 Station Blackout

Discussion:

Generic Issue A-44 was resolved with the publication of 10 CFR 50.63, which provides requirements that light-water-cooled nuclear power plants be able to withstand for a specified duration and recover from a station blackout. It specifies that an alternate ac power source constitutes acceptable capability to withstand station blackout provided an analysis is performed that demonstrates that the plant has this capability from the onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate.

10 CFR 50.2 for the alternate ac source notes that the alternate ac power source must have sufficient capability and reliability for operation of all systems required for coping with station blackout for the time required to place and maintain the plant in safe shutdown.

AP1000 Response:

AC electrical power is not needed to establish or maintain a plant safe shutdown condition for the AP1000. The ac power system is discussed in Chapter 8. In addition, two nonsafety-related standby

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diesel generators are provided as alternate sources of electrical power to nonsafety-related active systems that provide a defense-in-depth function.

A-46 Seismic Qualification of Equipment in Operating Plants

Discussion:

Generic Issue A-46 addresses the variability among operating plants in the margins of safety provided in equipment to resist seismically induced loads and perform the intended safety functions. The NRC believes that the seismic qualification of equipment in operating plants must, therefore, be reassessed to confirm the ability to bring the plant to a safe shutdown condition when it is subject to a seismic event.

AP1000 Response:

This issue applies to operating plants and, as such, does not specifically apply to the AP1000, which is designed in accordance with current seismic requirements. The seismic Category I mechanical and electrical equipment utilized for the AP1000 is qualified in accordance with the AP1000 qualification methodology discussed in Section 3.10. The methodology is based on the generic Westinghouse qualification program previously approved by the NRC. This methodology addresses IEEE Standard 344-1987 (Reference 13) and Regulatory Guide 1.100. See Subsection 1.9.1 (Appendix 1A).

A-47 Safety Implications of Control Systems

Discussion:

Generic Issue A-47 addresses the safety impact of non-safety-related control systems on plant dynamics. Instrumentation and control systems used by nuclear plants comprise safety-related protection systems and nonsafety-related control systems. Safety-related systems are used to trip the reactor when specified parameters exceed allowable limits and to protect the core from overheating by initiating emergency core cooling systems. Nonsafety-related control systems are used to maintain the plant within prescribed parameters during shutdown, startup, normal load, and varying power operation. Nonsafety-related systems are not relied on to perform any safety functions during or following postulated accidents, but are used to control plant processes.

AP1000 Response:

For the AP1000, control system failures are considered as potential initiating events. The analyses of these transients demonstrate that the consequences of such failures are bounded by ANS Condition II criteria. No design basis failure of a control system violates Condition II criteria.

The integrated control system for the AP1000 obtains certain control input signals from signals used in the integrated protection system. With the integrated control and protection system, functional independence of the control and protection systems is maintained by providing a signal selection device in the control system for those signals used in the protection system. The purpose of the signal selection device is to prevent a failed signal, caused by the failure of a protection channel, from resulting in a control action that could lead to a plant condition requiring that protective action. The signal selection device provides this capability by comparing the redundant signals and automatically eliminating an aberrant signal from use in the control system. This capability exists for bypassed sensors or for sensors whose signals diverge from the expected error tolerance.

The plant control system incorporates design features such as redundancy, automatic testing, and self-diagnostics to prevent challenges to the protection and safety monitoring system. Chapter 7 provides a discussion of the AP1000 instrumentation and controls. The surveillance requirements for the main and startup feedwater control are found in Technical Specifications 3.7.3 and 3.7.7.

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A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Discussion:

Generic Issue A-48 addresses postulated light water reactor accidents resulting in a degraded or melted core that could result in the generation and release to the containment of large quantities of hydrogen. One source of hydrogen is from the reaction of the zirconium fuel cladding with the steam at high temperatures. The NRC requires design provisions for handling hydrogen releases associated with rapid reaction of a large portion of fuel cladding (10 CFR 50.44 and 10 CFR 50.34).

AP1000 Response:

The AP1000 design complies with the provisions of draft changes to 10 CFR 50.44 and 10 CFR 50.34 (f). The mechanisms used to monitor and control hydrogen inside containment are discussed in Subsection 6.2.4.

A-49 Pressurized Thermal Shock

Discussion:

Generic Issue A-49 addresses transients and accidents postulated to occur in pressurized water reactors that can result in severe overcooling (thermal shock) of the reactor vessel, concurrent with high pressure. In these pressurized thermal shock events, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall that produces a thermal stress with maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress varies with the rate of change of temperature and is compounded by coincident pressure stresses.

As long as the fracture resistance of the reactor vessel material is relatively high, these events are not expected to cause vessel failure. The fracture resistance of the reactor vessel material decreases with the integrated exposure to fast neutrons. The rate of decrease is dependent on the chemical composition of the vessel wall and weld materials.

AP1000 Response:

The AP1000 complies with the requirements of 10 CFR 50.61. Material requirements and pressure-temperature limits are discussed in Subsections 5.3.2 and 5.3.3.

B-5 Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

Discussion:

Part I - Ductility of Two-Way Slabs and Shells

Generic Issue B-5 involved a concern over the lack of information on the behavior of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension, flexure, and shear. The NRC Staff concluded that there is sufficient information pertaining to the design of two-way slabs subjected to dynamic loads and biaxial tension to enable a reasonably accurate analysis.

Part II - Buckling Behavior of Steel Containments

Generic Issue B-5 involves a concern over the lack of a uniform, well defined approach for design evaluation of steel containments. Of particular interest was potential instability of the shell during dynamic loadings. Based on the conclusion of the NRC Staff that existing steel containments had adequate margins against buckling and that the issue of steel containment buckling had very little safety impact, this item was classified as resolved with no new requirements.

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AP1000 Response:

The design requirements and analysis methods used for two-way reinforced concrete slabs and for the steel containment are outlined in Section 3.8.

B-17 Criteria for Safety-Related Operator Actions

Discussion:

Generic Issue B-17 addresses the development of a time criterion for safety-related operator actions including a determination of whether or not automatic actuation is required. The evaluation of this issue includes Issue 27, Manual versus Automated Actions.

AP1000 Response:

The AP1000 automatically initiates the safety-related actions required to protect the plant during design basis events. The plant systems are designed to provide the required information to the operator to monitor plant conditions and to evaluate the performance of the safety-related passive systems, as well as the nonsafety-related active systems. The active systems are designed to automatically actuate and provide defense-in-depth for various plant events, to preclude unnecessary actuation of the safety-related passive systems. The plant design also provides the capability for a backup manual initiation of both the safety-related systems and the nonsafety-related defense-in-depth systems.

As described in Chapter 15, the AP1000 safety systems maintain the plant in a safe condition following design basis events. For the design basis events described in Chapter 15, this is accomplished without the need for operator action for up to 72 hours. Operator action is planned and expected during plant events to achieve the most effective plant response consistent with event conditions and equipment availability. For events where operator action is taken, the plant design maximizes the time available to complete actions for events. For example, during a steam generator tube rupture, no operator action is required to establish safe shutdown conditions or prevent steam generator overfill. It is expected that the main control room operators take actions similar to those taken in current plants to identify and isolate the faulted steam generator and to stabilize plant conditions.

For events where operator actions are taken, the AP1000 design is based on previous experience and the guidance of ANSI 58.8-1984 (Reference 21). At least 30 minutes is available following design basis events for the operator to initiate planned actions.

B-22 LWR Fuel

Discussion:

Generic Issue B-22 addresses the reliability of fuel behavior predictions during normal operation and postulated accidents. Standard Review Plan, Section 4.2 provides detailed NRC criteria for the design of fuel and core components.

AP1000 Response:

The AP1000 reactor core design complies with the Standard Review Plan, Section 4.2. See Section 4.2 for a discussion of the fuel system design.

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B-29 Effectiveness of Ultimate Heat Sinks

Discussion:

Generic Issue B-29 addresses NRC confirmation of currently used mathematical models for prediction of ultimate heat sink performance by comparing model performance with field data and development of better guidance regarding the criteria for weather record selection to define ultimate heat sink design basis meteorology.

The NRC considers this issue to be technically resolved with the publication of three reports: NUREG-0693, NUREG-0733, and NUREG-0858 (References 23, 24 and 25).

AP1000 Response:

The AP1000 passive containment cooling system complies with Standard Review Plan, Section 9.2.5 by providing passive decay heat removal that transfers heat to the atmosphere, which is the ultimate heat sink for accident conditions. The passive containment cooling system is described in Subsection 6.2.2.

B-32 Ice Effects on Safety-Related Water Supplies

Discussion:

Generic Issue B-32 addresses the potential effects of extreme cold weather and ice buildup on the reliability of various plant water supplies. Current NRC criteria are provided in Standard Review Plan, Section 2.4.7, "Ice Effects."

AP1000 Response:

Subsection 6.2.2 describes the ultimate heat sink design and discusses the features that prevent freezing in the passive containment cooling system.

B-36 Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems

Discussion:

Generic Issue B-36 addresses the development of revisions to current guidance and technical positions regarding engineered safety features and normal ventilation system air filtration and adsorption units. The NRC considers this issue technically resolved with the issuance of Revision 2 to Regulatory Guide 1.52 and Revision 1 to Regulatory Guide 1.140.

AP1000 Response:

The AP1000 main control room emergency habitability system (VES) includes a passive filtration system that is contained entirely within the main control room envelope. Regulatory Guide 1.52 was written for active safety-related filtration systems. To the extent applicable, system design criteria are established in accordance with Regulatory Guide 1.52 Revision 3. The passive filtration portion of the AP1000 VES contains no active equipment.

B-53 Load Break Switch

Discussion:

Generic Issue B-53 addresses the use of the generator load break switch for isolating the generator from the step-up transformer following turbine trip. Plant designs that utilize generator load circuit breakers to satisfy the requirement for an immediate access circuit stated in General Design Criterion

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17, "Electric Power Systems," must prototype-test the generator load circuit breaker to demonstrate functional capability.

AP1000 Response:

The AP1000 design incorporates a generator load circuit breaker to provide a reliable source of ac power to the electrical systems. Exceptions to General Design Criteria 17, as discussed in Section 3.1, are due to the AP1000 design not requiring ac power sources for a design basis accident. Subsection 8.2.2.5 provides further discussion.

B-56 Diesel Reliability

Discussion:

Generic Safety Issue B-56 addresses the reliability of emergency onsite diesel-generators. Diesel reliability is a factor in the criteria associated with the resolution of Unresolved Safety Issue A-44. The resolution of issue B-56 is the development of guidelines for an acceptable emergency diesel-generator reliability program to ensure conformance with the emergency diesel-generator target reliability (0.95 to 0.975) identified in the proposed resolution of Unresolved Safety Issue A-44.

AP1000 Response:

The AP1000 diesel-generators are not safety related. The AP1000 diesel-generator reliability is based on diesel-generator industry standards and practices. The diesel generator is discussed in Subsection 8.3.1. The diesel generator reliability is modeled in the PRA. The reliability assurance program is discussed in Section 16.2.

B-61 Allowable ECCS Equipment Outage Periods

Discussion:

Generic Safety Issue B-61 addresses surveillance test intervals and allowable equipment outage periods in the technical specifications for safety-related systems. This task involves the NRC development of analytically based criteria for use in confirming or modifying these surveillance intervals and allowable equipment outage periods.

AP1000 Response:

The AP1000 surveillance test intervals and allowable outage times help to meet plant safety goals while maximizing plant availability and operability. In determining these limits for the AP1000 technical specifications, a combination of NUREG-1431 precedent, system design, and safety-related function is considered.

B-63 Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary

Discussion:

Generic Issue B-63 addresses the adequacy of the isolation of low-pressure systems that are connected to the reactor coolant pressure boundary. The NRC staff requires that valves forming the interface between high- and low-pressure systems associated with the reactor coolant boundary have sufficient redundancy to prevent the low-pressure systems from being subjected to pressures that exceed their design limits.

AP1000 Response:

The AP1000 includes interconnections between high- and low-pressure systems. Each of these systems interfaces contains appropriate isolation provisions. Valves at the interface between high-

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and low-pressure systems have redundancy to prevent low-pressure systems from being subjected to pressures that exceed their design limits. The AP1000 design meets the provisions of the Standard Review Plan, Section 3.9.6.

The normal residual heat removal system interface is addressed in Subsection 5.4.7. WCAP-15993 (Reference 56) provides an evaluation of the AP1000 conformance to intersystem loss-of-coolant accident regulatory criteria.

B-66 Control Room Infiltration Measurements

Discussion:

Generic Safety Issue B-66 addresses the adequacy of control room area ventilation systems and control building layout to ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases. The NRC considers this issue as being technically resolved, and criteria have been incorporated in Standard Review Plan, Section 6.4.

AP1000 Response:

The AP1000 main control room is essentially leak-tight. A description of the control room habitability systems is contained in Section 6.4.

Verification of design infiltration rates is as specified in Standard Review Plan, Section 6.4. The AP1000 minimizes unfiltered in-leakage by maintaining the main control room at a slightly positive pressure.

C-1 Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Discussion:

Generic Issue C-1 addresses the long-term capability of hermetically sealed instruments and equipment that must function in post-accident environments. The NRC considers this issue as being technically resolved with the issuance of current criteria for qualification of safety-related electrical equipment.

AP1000 Response:

The AP1000 environmental qualification program described in response to Unresolved Safety Issue A-24 addresses qualification of safety-related instrumentation and electrical equipment that must function under accident conditions. This program confirms the integrity of seals employed in the design of Class 1E equipment. See item A-24 of this subsection and Section 3.11 for AP1000 qualification methodology.

C-4 Statistical Methods for ECCS Analysis

Discussion:

Generic Issue C-4 addresses NRC development of a statistical assessment of the certainty level of the peak clad temperature limit. Appendix K, "ECCS Evaluation Models," to 10 CFR 50 specifies the requirements for ECCS analysis. These requirements call for conservatisms to be applied to certain models and assumptions used in the analysis to account for data uncertainties at the time Appendix K was written. The resulting conservatism in the calculated peak clad temperature (PCT) has not been thoroughly compared against the uncertainty in peak clad temperature obtained from a realistically calculated (best-estimate) LOCA. The staff allows voluntary use of statistical uncertainty analysis to justify relaxation of all but the required conservatisms contained in current ECCS evaluation models.

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AP1000 Response:

Chapter 15 discusses the LOCA analysis for the AP1000.

C-5 Decay Heat Update

Discussion:

Generic Issue C-5 involves following the work of research groups in determining best-estimate decay heat data and associated uncertainties for use in LOCA calculations.

The staff has determined that the 1979 ANSI 5.1 is technically acceptable and has allowed the use of this data to justify relaxation of non-required conservatisms in current ECCS evaluation models. The ECCS rule change allows the use of this new data. This issue was determined to be resolved.

AP1000 Response:

The large-break LOCA analyses for the AP1000, which employ the best-estimate W COBRA/TRAC analysis methodology (Subsection 15.6.5), use the decay heat model identified in the 1979 ANSI 5.1 (Reference 26).

C-6 LOCA Heat Sources

Discussion:

Generic Issue C-6 addresses the impact on LOCA calculations of LOCA heat sources, their associated uncertainties, and the manner in which they are combined. An evaluation was made of the combined effect of power density, decay heat, stored energy, fission power decay, and their associated uncertainties with regard to calculations of LOCA heat sources.

AP1000 Response:

See Subsection 15.6.5 for a discussion of LOCA heat sources.

C-10 Effective Operation of Containment Sprays in a LOCA

Discussion:

Generic Issue C-10 addresses the effectiveness of containment sprays to remove airborne radioactive materials that could be present within the containment following a LOCA. The NRC considers this issue as being technically resolved with the issuance of ANSI 56.5-1979 (Reference 28), which is referenced in Standard Review Plan, Section 6.5.2.

AP1000 Response:

The AP1000 design does not employ a safety-related containment spray system for removal of airborne radioactive materials in containment. Subsection 15.6.5.3 provides details of source term and mitigation techniques.

C-17 Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Discussion:

Generic Issue C-17 addresses the development of criteria for acceptability of radwaste solidification agents. The NRC considers this issue as technically resolved with the issuance of 10 CFR 61.56.

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AP1000 Response:

The AP1000 solid radwaste system transfers, stores, and prepares spent ion exchange resins for disposal. It also provides for disposal of filter elements; sorting, shredding, and compaction of compressible dry active wastes. The solid radwaste system does not provide for liquid waste concentration or solidification. These functions, if used, are provided using mobile systems. Solidification of wastes is not performed by permanently installed systems.

1.9.4.2.3 New Generic Issues

These items were identified in NUREG-0933 as New Generic Issues and surfaced after the publication of the NUREGs that included the Task Action Plan items other unresolved safety issues.

Issue 14 PWR Pipe Cracks

Discussion:

This issue addresses the occurrences of main feedwater line cracking found in operating plants. This issue was classified as resolved with no new requirements.

AP1000 Response:

The design and inspection requirements for the feedwater lines are discussed in Subsection 10.4.7.

Issue 15 Radiation Effects on Reactor Vessel Supports

Discussion:

Generic Safety Issue 15 addresses the potential problem of radiation embrittlement of reactor vessel support structures. There is a potential for radiation embrittlement of the reactor vessel support structure from long-term exposure to neutrons with an energy of 1 MeV or greater. Embrittlement due to neutron damage may increase the potential for propagation of existing flaws.

AP1000 Response:

The supports for the AP1000 reactor pressure vessel are designed for loading conditions and environmental factors including consideration of neutron fluence levels. The material requirements include fracture toughness requirements and impact testing requirements in compliance with ASME Code, Section III, Subsection NF requirements. The reactor pressure vessel supports are not in the region of high neutron fluence where neutron embrittlement of the supports would be a significant concern.

Issue 22 Inadvertent Boron Dilution Event

Discussion:

Some operating plants do not have provisions to detect boron dilution during cold shutdown. This could result in inadvertent criticality. The NRC staff concluded that existing review criteria are adequate. This issue was classified as resolved with no new requirements.

AP1000 Response:

The provisions in the design to preclude inadvertent boron dilution events are outlined in Subsection 9.3.6.

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Issue 23 Reactor Coolant Pump Seal Failures

Discussion:

Generic Safety Issue 23 addresses reactor coolant pump seal failures that challenge the makeup capacity in PWRs. Such seal failures represent small-break loss-of-coolant accidents.

AP1000 Response:

The AP1000 reactor coolant pumps are sealless pumps. A sealless pump contains the motor and all rotating components inside a pressure vessel designed for full reactor coolant system pressure. The shaft for the impeller and rotor is contained within the pressure boundary; therefore, seals are not required in order to restrict leakage out of the pump into containment. Subsection 5.4.1 provides additional information on the sealless pump design for the AP1000 reactor coolant pumps. Since the reactor coolant pumps do not rely on seals as a reactor coolant pressure boundary, this issue is not applicable to the AP1000.

Issue 24 Automatic ECCS Switchover to Recirculation

Discussion:

This issue addresses the issue of switchover from safety injection to recirculation using manual valve alignment or automatic valve alignment.

AP1000 Response:

The AP1000 does not switch from injection to recirculation in the sense that injection is not isolated when recirculation is opened. The AP1000 does provide for automatic opening of the recirculation line on a low level signal from the in-containment refueling water storage tank. See Section 6.3 for additional details.

Issue 29 Bolting Degradation or Failure in Nuclear Power Plants

Discussion:

Generic Safety Issue 29 addresses a concern about pressure boundary integrity and component support reliability associated with bolt failures.

As documented in Generic Letter 91-17, the NRC has provided resolution of this issue. The resolution is documented in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," and NUREG-1445, "Regulatory Analysis for the Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." The resolution was based on a number of industry initiatives and NRC staff actions. NRC staff actions include issuing a number of bolting-related bulletins, generic letters and information notices. Industry initiatives include the publishing of EPRI Reports NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," and NP-5067, "Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel."

EPRI Report NP-5769 establishes the characteristic that bolted connections exhibit leakage prior to failure resulting from bolt degradation. The NRC has endorsed the recommendation in NP-5769 that plant-specific bolting integrity programs be established that encompass safety-related bolting. NUREG-1339 includes recommendations and guidelines for the content of a comprehensive bolting integrity program.

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AP1000 Response:

The elements of resolution pertain to the design, material selection, fabrication, and in-service inspection of the bolted connections found in the AP1000. To address this, resolutions found in NUREG-1339 are incorporated into the design, material selection, fabrication, and maintenance of the bolted connections. The maintenance practices are addressed by the maintenance program of the combined license holder. Conformance to ASME Code, Section III requirements for pressure boundary components and related supports provides safe operation in the event of bolting degradation. Because of the emphasis in the AP1000 design on access for maintenance and inspection, the recommended maintenance practices can be implemented.

Issue 43 Reliability of Air Systems

Discussion:

This issue addresses the concern that compressed air system degradation or malfunction may cause malfunction of safety-related systems and components. Of particular interest are air operated valves because of problems with the quality of the air supply or the manner in which the compressed air system fails. Generic Letter 88-14 and NUREG-1275 were issued in response to this issue.

AP1000 Response:

The compressed air systems are described in Subsection 9.3.1. Provisions are included to maintain the quality of the air supply. The AP1000 safety-related, air-operated valves do not rely on the air supply to perform their safety-related function.

Issue 45 Inoperability of Instrumentation Due to Extreme Cold Weather

Discussion:

Generic Safety Issue 45 addresses the inoperability of instrumentation due to extreme cold weather. This issue was resolved with the issuance of changes to Standard Review Plan, Section 7.1, Appendix A to Section 7.1, Section 7.5, and Section 7.7.

AP1000 Response:

The AP1000 complies with Standard Review Plan Section 7.1, Appendix A to Section 7.1, Section 7.5, and Section 7.7.

Issue 51 Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

Discussion:

Generic Safety Issue 51 addresses the susceptibility of open cycle service water systems to fouling including the buildup of aquatic bivalves and corrosion products that can significantly degrade the performance of the system. In operating plants, the service water system is typically used to cool safety-related equipment and to transfer decay heat to the ultimate heat sink.

AP1000 Response:

The service water system in the AP1000 provides cooling water to the component cooling water system and has no safety-related functions. None of the safety-related equipment requires cooling water to effect a safe shutdown or mitigate the effects of design basis events. Heat transfer to the ultimate heat sink is accomplished by heat transfer through the containment shell to air and water flowing on the outside of the shell.

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The design of the service water system and the provisions for minimizing long-tern corrosion and organic fouling are described in Subsection 9.2.1.

Issue 57 Effects of Fire Protection System Actuation

Discussion:

Generic Safety Issue 57 addresses the potential for adverse interactions from actuation of the fire protection system with safety-related equipment. Operating experience has shown that safety-related equipment subject to fire protection system water spray and other suppressant chemicals can be rendered inoperable.

AP1000 Response:

The fire protection system and fire protection program in the AP1000 minimize the potential for adverse interactions of safety-related equipment with the fire protection system. The means used to achieve this result include: isolating combustible material and limiting the spread of fire by subdividing the plant into fire areas separated by fire barriers, providing separate and redundant safe shut down components and associated electrical divisions to preserve the ability to safely shutdown the plant following a fire, and providing floor drains sized to remove expected firefighting water without flooding safety-related equipment. The design of the fire protection system is described in Subsection 9.5.1.

Issue 67.3.3 Improved Accident Monitoring

Discussion:

This issue addresses weaknesses in accident monitoring. The recommended solution is to implement Regulatory Guide 1.97.

AP1000 Response:

The guidance of Regulatory Guide 1.97 is followed to determine the appropriate parameters to monitor in the AP1000.

Issue 73 Detached Thermal Sleeves

Discussion:

This issue addresses problems with "generation 3" thermal sleeves.

AP1000 Response:

The AP1000 does not use generation 3 thermal sleeves and includes design provisions to preclude failures of thermal sleeves.

Issue 75 Generic Implications of ATWS Events at the Salem Nuclear Plant

Discussion:

This issue considers the failure of reactor trip breakers to open and issues related to design and testing of the reactor protection system. Issues to be considered include the capability to record and display reactor trip system parameters, equipment classification information, post-maintenance testing, and reliability improvements in operating plants. Generic letter 83-28 and IE Bulletins 83-01 and 83-04 were issued by the staff with specific requirements.

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AP1000 Response

The design of the reactor trip breakers and the reactor protection system is outlined in Section 7.1. Information on the functional requirements for reactor trip and conformance with industry and regulatory guidance is outlined in Section 7.2.

The provisions provided to display and record parameters used by the reactor trip system are outlined in Subsections 7.1.2.6 and 7.1.2.13. Section 7.5 also provides information on requirements for safety-related display information.

Subsection 7.1.1 identifies the safety-related functions provided by the protection and safety monitoring system and the items that are included in the system including the reactor trip switchgear. Conformance of safety-related systems and components to industry and regulatory criteria is identified in Subsection 7.1.4.

The reliability and fault tolerance of the protection and safety monitoring system for test maintenance and bypass conditions are outlined in Subsection 7.1.2.10.

The changes in the design of the reactor trip breakers and associated logic to enhance reliability in operating nuclear power plants have been incorporated in the AP1000 design as appropriate. The reactor trip system includes built-in test capability.

WCAP-15800 addresses conformance with generic letters and bulletins.

Issue 79 Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

Discussion:

Generic Safety Issue 79 addresses the thermal stresses that occur in the reactor vessel head flange during a natural circulation cooldown. High stresses in the flange or studs during a natural circulation cooldown in PWRs could violate ASME code allowables. Cycling of the stresses could reduce the fatigue margin. Generic Letter 92-02 repeated the reporting requirements of 10CFR 50.73 (a)(2)(ii)(B), "Licensee event report system."

AP1000 Response:

The natural circulation cooldown transient is evaluated as part of ASME Code vessel evaluations and is discussed in Subsection 3.9.1.1.2.11. The reporting requirements to address the requirements of 10CFR 50.73 (a)(2)(ii)(B) referenced in Generic Letter 92-02 are the responsibility of the Combined License holder.

Issue 82 Beyond Design Basis Accidents in Spent Fuel Pools

Discussion:

This issue addresses the concern of a beyond design basis accident in which the spent fuel pool is drained and spent fuel stored there subsequently catches on fire releasing very large amounts of radioactive contamination. This issue is classified as resolved with no new requirements.

AP1000 Response:

The AP1000 includes design provisions that preclude draining of the spent fuel pool. Also, provisions are available to supply water to the pool in the event the water covering the spent fuel begins to boil off.

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Issue 83 Control Room Habitability

Discussion:

Loss of control room habitability following an accidental release of external toxic or radioactive material or smoke can impair or cause loss of the control room operators' capability to safely control the reactor. Use of the remote shutdown workstation outside the control room following such events is unreliable since this station has no emergency habitability or radiation protection provisions.

AP1000 Response:

Habitability of the main control room is provided by the main control room/control support area HVAC subsystem of the nonsafety-related nuclear island nonradioactive ventilation system (VBS). If ac power is unavailable for more than 10 minutes or if main control room differential pressure is below the “Low” setpoint for more than 10 minutes or if “High-2” particulate or iodine radioactivity is detected in the main control room supply air duct, which may lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The safety-related main control room emergency habitability system supplies breathable quality air for the main control room operators while the main control room is isolated.

In the event of external smoke or radiation release, the nonsafety-related nuclear island nonradioactive ventilation system provides for a supplemental filtration mode of operation, as discussed in Section 9.4. In the unlikely event of a toxic chemical release, the safety-related main control room emergency habitability system has the capability to be manually actuated by the operators. Further, a 6-hour supply of self-contained portable breathing equipment is stored inside the main control room pressure boundary.

Issue 87 Failure of HPCI Steam Line Without Isolation

Discussion:

Generic Safety Issue 87 addresses the uncertainty regarding the operability of the motor-operated isolation valves for the steam supply lines of the high-pressure coolant injection (HPCI) system in boiling water reactors following a postulated break in the supply line. A break in the line could lead to high flow or high differential pressure that may inhibit closure of the isolation valve. These valves typically cannot be tested in-situ for the design flow rates and pressures. Although the AP1000 does not have a high-pressure coolant injection system, it does have isolation valves designed to close against high flow or high pressure differential in the event of a postulated pipe break.

The issue of the operability of motor-operated valves has received considerable attention since Generic Safety Issue 87 was initiated. The NRC provided guidance for inservice testing of motor-operated, safety-related valves in Generic Letter 89-10. SECY-93-087 identifies the proposed position on inservice testing of safety-related valves for advance light water reactors. The guidance in these documents recommends that safety-related valves be tested under full flow under actual plant conditions where practical. EPRI has a program to demonstrate operation of motor-operated valves.

AP1000 Response:

Safety-related valves must meet the requirements of ASME Code, Section III to provide pressure boundary integrity. Valves and valve operators are sized to provide operation under a full range of design basis flow and pressure drop conditions. For the AP1000, safety-related motor-operated valve designs are subject to qualification testing to demonstrate the capability of the valve to open, close, and seat against maximum pressure differential and flow. The requirements for this testing are based

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on ASME QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.” See Subsection 5.4.8 for an outline of AP1000 valve requirements.

The in-service testing program for safety-related valves is discussed in Subsection 3.9.6. Subsection 3.9.6 includes a discussion of the factors to be considered to determine which valves and test conditions are to be used for operability testing of power-operated valves. Sufficient flow is provided to fully open check valves during testing unless the maximum accident flows are not sufficient to fully open the check valve. The valves built to ASME Code, Section III are tested in compliance with the requirements found in the ASME code, "Code for Operation and Maintenance of Nuclear Power Plants." For additional information on inservice testing of safety-related valves, see Subsection 3.9.6.

Issue 93 Steam Binding of Auxiliary Feedwater Pumps

Discussion:

Generic Safety Issue 93 addresses the potential for a common mode failure of the pumps in an auxiliary or emergency feedwater system. Hot water leaking through one or more isolation valves can flash to steam at the auxiliary feedwater pump potentially resulting in the failure of the pump to operate if required because of steam binding. The NRC addressed this issue in Bulletin 85-01, and reinforced it in Generic Letter 88-03, by requesting that the fluid conditions in the auxiliary feedwater system be monitored and procedures be developed to recognize steam binding and restore the auxiliary feedwater system to operable status if steam binding should occur.

AP1000 Response:

The AP1000 does not have a safety-related auxiliary feedwater system. The passive core cooling system provides the safety-related function of cooling the reactor coolant system in the event of loss of feedwater. The startup feedwater system provides the steam generators with feedwater during plant conditions of startup, hot standby, and cooldown and when the main feedwater pumps are unavailable. The startup feedwater system has no safety-related function.

The startup feedwater system includes temperature instrumentation in the pump discharge for monitoring of the temperature of the startup feedwater system. The system also includes a normally closed isolation valve and a normally closed check valve for each pump limiting potential back leakage.

Issue 94 Additional Low-Temperature Overpressure Protection for Light Water Reactors

Discussion:

Generic Safety Issue 94 addresses the establishment of additional guidance for reactor coolant system low-temperature overpressure protection to ensure reactor vessel and reactor coolant system integrity beyond that identified in the resolution to Generic Safety Issue (GSI) A-26. Low-pressure overpressurization events that occurred subsequent to the implementation of the guidelines for resolution of GSI A-26 indicated a need for additional low-temperature overpressure protection. To resolve this issue, the NRC issued Generic Letter 90-06 which required a revision to plant technical specifications for operability of the low-temperature overpressure protection system. Other possible solutions identified in GL 90-06 included hardware modifications including use of residual heat removal system relief valves and requiring the low temperature overpressure protection system to be fully safety related.

AP1000 Response:

The reactor vessel for the AP1000 is designed to be less susceptible to brittle fracture during low temperature overpressure events. The material requirements and welding processes are developed

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to enhance resistance to embrittlement. See Subsection 5.3.2 for additional information on the requirements to address fracture toughness of the reactor vessel.

The normal residual heat removal system is designed to provide the safety-related function of low temperature overpressure protection for the reactor coolant system during refueling, startup, and shutdown operations. The system is designed to limit the reactor coolant system pressure within the limits specified in 10 CFR 50, Appendix G. The relief valves in the normal residual heat removal system are used to provide the overpressure protection. See Subsection 5.4.7 for additional information on the design of the normal residual heat removal system and the overpressure protection function.

Issue 103 Design for Probable Maximum Precipitation

Discussion:

Generic Safety Issue 103 addresses the methodology used for determining the design flood level for a particular reactor site. This issue was resolved by incorporating the methodology into the Standard Review Plan.

AP1000 Response:

This is a site-related parameter. The AP1000 is designed for air temperatures, humidity, precipitation, snow, wind, and tornado conditions as specified in Table 2-1. The site is acceptable if the site characteristics fall within the AP1000 plant site design parameters in Table 2-1. For cases where a site characteristic exceeds the envelope parameter, see Chapter 2.

Issue 105 Interfacing System LOCA at BWRs

Discussion:

Generic Safety Issue 105 addresses concerns over the adequacy of isolation valves between the reactor coolant system and low-pressure interfacing systems in BWRs. This issue, which is limited to pressure isolation valves in BWRs, is related to Generic Safety Issue 96, which considers the failure of the pressure isolation valves between the reactor coolant system and the RHR system in PWRs. Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. This may result in a core melt accident with an energetic release outside the containment building that could cause a significant offsite radiation release. Designing interfacing systems to withstand full reactor pressure is an acceptable means of resolving this issue.

AP1000 Response:

For information on this issue, see Subsection 1.9.5.1 SECY-90-016 Issues. See Subsection 5.4.7 for additional information on the normal residual heat removal system design.

Issue 106 Piping and Use of Highly Combustible Gases in Vital Areas

Discussion:

Generic Safety Issue 106 addresses the normal process system use of relatively small amounts of combustible gases on site and also addresses leaks or breaks in the hydrogen piping and supply system that could result in the accumulation of a combustible or an explosive mixture of air and hydrogen within the auxiliary systems building. The accumulation of combustible or explosive mixtures of gas in the auxiliary systems building could represent a threat to safety-related equipment if the combustible gases are inadvertently ignited.

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AP1000 Response:

The AP1000 uses small amounts of combustible gases for normal plant operation. Most of these gases are used in limited quantities and are associated with plant functions or activities that do not jeopardize any safety-related equipment. These gases are found in areas of the plant that are removed from the Nuclear Island (see Subsection 9.3.2 for a description of the plant gas system). The exception to this is the hydrogen supply line to the chemical and volume control system (CVS).

The chemical and volume control system is the only system on the nuclear island that uses hydrogen gas. Hydrogen is supplied to the AP1000 CVS inside containment from a hydrogen injection package supplied from high pressure hydrogen bottles. The release of the contents of high pressure hydrogen bottles in the limiting building volumes (both inside containment and in the auxiliary building) would not result in a volume percent of hydrogen large enough to reach a detonable level.

The chemical and volume control system hydrogen supply piping is routed through the turbine building and into the auxiliary building and then into containment. The H_2 supply line is routed through the piping/valve room on elevation 100′-0″ of the auxiliary building. The piping/valve penetration room in the auxiliary building on elevation 100′-0″ is designed as a 3-hour fire zone. A fire in this area would not inhibit the safe shutdown of the plant. More information is contained in Appendix 9A.

The turbine building does not house any safety-related systems or equipment. The release of hydrogen into an area of the turbine building does not represent a threat to the safety of the plant.

The AP1000 containment has hydrogen sensors that would detect hydrogen leaks. The containment hydrogen concentration monitoring subsystem is described in Subsection 6.2.4.1.

Issue 113 Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers

Discussion:

Generic Safety Issue 113 addresses the requirements for qualification and periodic operability testing of large bore hydraulic snubber for operating plants. Large-bore hydraulic snubbers are used to a limited extent on the AP1000 to provide support, particularly for seismic events, of piping systems and components while allowing for movement due to thermal expansion. The NRC, in a draft regulatory guide (SC-708-4, "Qualification and Acceptance Test for Snubbers Used in Systems Important to Safety"), has established recommendations for testing of hydraulic snubbers on a forward-fit basis; that is, units without a license at the time the recommendations were established.

AP1000 Response:

The AP1000 plant uses significantly fewer hydraulic snubbers than do currently operating plants. In addition to the recommendations in the draft regulatory guide, testing requirements have been established in ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants," as implemented in the Inservice Testing Program Plan. Subsection 3.9.3.4.3 discusses requirements for production and qualification testing. The design of the hydraulic snubbers permits required preoperational and inservice testing.

Subsection 3.9.8.3 defines the responsibility to provide information on snubber operability testing.

Issue 120 On-Line Testability of Protection System

Discussion:

This issue is related to the protection system of some older plants that do not provide for as complete a degree of on-line protection system testing surveillance capability as is now required. Testing

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requirements and guidance are found in GDC 21, Regulatory Guides 1.22 and 1.118 and IEEE Standard 338. This item is classified as resolved with no additional requirements.

AP1000 Response:

This item does not apply to the AP1000. The provision for testing of the protection system in conformance with the regulatory guidance is found in Section 7.1.

Issue 121 Hydrogen Control for Large, Dry PWR Containments

Discussion:

Generic Safety Issue 121 concerns ongoing NRC experimental and analytical programs addressing the likelihood of safe shutdown equipment surviving a hydrogen burn. The staff also intends to explore the possibility and probable consequences of the formation of local detonable concentrations in large, dry PWRs. The concerns are prediction of conditions in realistic configurations, and containment and equipment survivability.

AP1000 Response:

The AP1000 includes provisions for hydrogen control for the unlikely severe accident cases in which large amounts of hydrogen could be generated because of degraded core events. Analyses were performed to examine the consequences of hydrogen burn and to evaluate the likelihood of deflagration to detonable transitions (see Appendix 19D).

For severe accident cases, the containment hydrogen control system prevents hydrogen burn initiation at high hydrogen concentration levels. Hydrogen igniters promote burning when the lower flammability limit is reached and limits the containment hydrogen concentration to less than 10 volume percent during and following a degraded core or core melt.

Thus, for severe accident cases, the AP1000 is designed to prevent the occurrence of hydrogen detonation, thereby preventing the possibility of the resultant large pressure spikes in containment, which is the source of concern for containment integrity and equipment survival. Details of the hydrogen ignition subsystem are provided in Subsection 6.2.4.2.3. Placement of the hydrogen igniters is discussed in Subsection 6.2.4.

A hydrogen burn analysis shows that the AP1000 hydrogen igniter system is effective in maintaining the hydrogen concentration throughout the containment close to the lower flammability limit, and that the peak pressure in the containment during and following hydrogen burn remains well below ASME service level C stress intensity limits. The hydrogen concentration is similar in all compartments analyzed, indicating that the hydrogen released mixes well in the AP1000 containment. The analyses predict conditions in realistic configurations. Peak gas temperatures and pressures in each compartment for each case analyzed are provided, thus providing the hydrogen burn thermal environment that containment equipment will experience. Details are provided in Chapter 14 of the AP1000 PRA report.

The challenge to the AP1000 containment integrity from hydrogen deflagrations and detonations during core damage events is examined in the hydrogen deflagration and detonation analyses. This bounding evaluation assumes that an amount of hydrogen equivalent to 100-percent active cladding oxidation burns all at once in the AP1000 containment, with no credit taken for the hydrogen igniters. The evaluation concludes that a hydrogen deflagration is unlikely to cause containment failure. Other analyses show that a deflagration to detonation transition in any part of the AP1000 containment is unlikely. Containment failure from a detonation is not considered a credible event for the AP1000 because of the lack of conditions supporting a deflagration to detonation transition, the provision and placement of hydrogen igniters, and the containment design features resulting in a well-mixed atmosphere. Details are provided in subsection 10.2.5 of the AP1000 PRA evaluation report.

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The hydrogen igniters and the containment electrical and mechanical penetrations are designed to operate in the most limiting severe accident environment, including a hydrogen burn. (See subsection 10.2.5 of the AP1000 PRA evaluation report.) The approach of using controlled burning to prevent accidental hydrogen burn initiation provides confidence that safety-related equipment will continue to operate during and after hydrogen burns. (See Subsection 6.2.4.)

Issue 124 Auxiliary Feedwater System Reliability

Discussion:

Generic Safety Issue 124 addresses the use of probabilistic risk assessment to evaluate the reliability of the auxiliary feedwater system. The issue was resolved by the NRC's issuing plant-specific requirements for a few plants that did not initially have a reliability higher than a minimum criteria.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not have a safety-related auxiliary feedwater system. The passive core cooling system provides the safety-related function of cooling of the reactor coolant system in the event of loss of feedwater. The startup feedwater system provides the steam generators with feedwater during plant conditions of startup, hot standby, and cooldown and when the main feedwater pumps are unavailable. The startup feedwater system has no safety-related function beyond containment isolation.

Issue 128 Electrical Power Reliability

Discussion:

Generic Safety Issue 128 addresses the reliability of onsite electrical systems and encompasses GSI 48, GSI 49, and GSI A-30.

AP1000 Response:

The design basis and design criteria for the Class 1E dc and UPS system is provided in Subsections 8.1.4.2.1 and 8.1.4.3. The class 1E dc and UPS system design is described in Subsection 8.3.2.1.1. Specifically, this design addresses IEEE Standards 603 and 308. This includes the following generic issues:

- Generic Safety Issue 48, LCO for Class 1E vital instrument buses in operating reactors. The AP1000 Technical Specifications provide the limiting conditions for operation in the event of a loss of one or more Class 1E 120-vac vital instrument buses and the associated inverters. The AP1000 Class 1E buses have no tie breakers

- Generic Safety Issue 49, interlocks and LCOs for Class 1E tie breakers. Based on the historical background, this issue is not applicable to the AP1000 design. There are no tie breakers between the four class 1E divisions.

- Generic Safety Issue A-30, adequacy of safety-related dc power supplies. The AP1000 incorporates the following recommended enhancements:

  – The Class 1E dc distribution system design is in accordance with the guidelines of IEEE Standard 384 and Regulatory Guide 1.75.

  – Four separate divisions of Class 1E dc power are provided.

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The AP1000 design provides additional testing capability through the installed spare battery bank with one installed battery charger. The spare battery bank permits frequent full-component testing, as described in Subsection 8.3.2.1.1, without compromising plant availability. Battery equalization can be performed off-line. The battery and battery charger can be tested and maintained separately.

Issue 130 Essential Service Water Pump Failure at Multiple Plant Sites

Discussion:

Generic Safety Issue 130 addresses the use of shared or cross-connected essential service water systems at sites with two or more reactor plants. During some situations the crosstied pumps may not be available for accident mitigation operations.

AP1000 Response:

The AP1000 is a single, independent plant that does not share or cross-tie systems or components with another plant. See Section 1.2 for a general description of the plant. This issue is not applicable to the AP1000.

Issue 135 Integrated Steam Generator Issues

Discussion:

Generic Safety Issue 135 was initiated to provide an integrated work plan for the resolution of steam generator issues including steam generator overfill consequences, water hammer, and eddy current testing. The issue was divided into the following four tasks:

1. Assessing current capabilities of eddy current testing and developing recommendations.

2. Reviewing SGTR results and conclusions to develop regulatory analysis supporting Standard Review Plan changes.

3. Reassessing SGTR associated issues including radiological, design basis, tube integrity, procedures, and RCS pressure control.

4. Reviewing the effects of water hammer, overfill and water carryover.

The results of the tasks will provide the staff with a basis to develop a position on offsite dose, operator action, tube integrity, water hammer, and valve operability.

AP1000 Response:

The AP1000 design features are discussed below.

TASK 1: Appendix 1A identifies the level of conformance with Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." As detailed in Appendix 1A, the AP1000 conforms with the regulatory guidance except where state-of-the-art advances have enhanced inservice inspection techniques. Further, as specified in Subsection 5.4.2.5, the steam generators permit access to tubes for inspection and/or repair or plugging, if necessary, per the guidelines described in Regulatory Guide 1.83. The AP1000 steam generator includes features to enhance robotics inspection of steam generator tubes.

TASK 2: Subsection 15.6.3.1.4 discusses anticipated operator recovery actions and the effects of those actions in the mitigation of a steam generator tube rupture (SGTR). As discussed in Subsection 15.6.3.2, the AP1000 incorporates automatic steam generator overfill protection. The

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details of the design are provided in Subsection 15.6.3.2, with the control logic provided in Section 7.2.

TASK 3: The following sections provide pertinent details on SGTR issues.

- Reassessment of radiological consequences: Subsection 15.6.3 provides details of the scenario, analysis assumptions, and results.
- Re-evaluation of design basis SGTR: The design basis SGTR evaluated on the AP1000 is discussed in Subsection 15.6.3, providing details of the scenario, analysis assumptions and results.
- Supplemental Tube Inspections: See Subsection 5.4.2.5, Appendix 1A, Regulatory Guide 1.83.
- Denting criteria: Subsection 5.4.2.4.3 provides a discussion of steam generator design and tubing compatibility with secondary coolants.
- Improved accident monitoring and reactor vessel inventory measurement: Section 7.5 discusses the safety related display information.
- Reactor coolant pump trip: Subsection 7.3.1.2.5 discusses reactor coolant pump trip.
- Control room design: Sections 7.5 and 18.8 discuss the control room design and design process.
- Emergency operating procedures: Section 18.9 addresses the development of emergency operating procedures.
- Organizational responses: Chapter 13 identifies the requirements for organizational responses.
- Reactor coolant pressure control: Subsection 7.7.1.6 addresses primary system pressure control.

TASK 4: Steam generator overfill, water carryover and water hammer are addressed as discussed in Subsection 15.6.3.2, with the control logic provided in Section 7.2.

Issue 142 Leakage Through Electrical Isolators in Instrumentation Circuits

Discussion:

Generic Issue 142 addresses the susceptibility to leakage of isolation devices between safety- and nonsafety-related electrical systems. The NRC requires that licensees identify isolation devices in instrumentation circuits that are potentially susceptible to electrical leakage, define and perform an inspection and test program, replace failed or unacceptable isolators, and implement an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

AP1000 Response:

The use of isolation devices in the AP1000 Instrumentation and Control Architecture is described in Subsections 7.1.2.10, "Isolation Devices," 7.7.1.11, "Diverse Actuation System," and WCAP-15776 (Reference 70), Section 3.9, “Conformance to the Requirements to Maintain Independence Between Safety Systems and Other Interconnected Equipment (Paragraph 5.6.3.1 of IEEE 603-1991).” As stated in WCAP-15776, Section 3.9, the isolation devices are tested to conform to requirements. This

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testing meets the requirement for an inspection and test program and identifies those devices that are potentially susceptible to electrical leakage. Implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems is the responsibility of the Combined License holder. The use of fiber-optic data links eliminates electrically conductive paths between receiving and transmitting terminals, and eliminates the potential for electrically generated noise caused by leakage through an isolator. These communication links also use extensive testing and error checking to minimize erroneous transmissions. These data links are described in Subsection 7.1.2.8, "Communication Functions." In addition, electromagnetic design, testing, and qualification is performed as described in WCAP-15776, Section 2.6, "Design Basis: Range of Conditions for Safety System Performance (Paragraph 4.7 of IEEE 603-1991.)"

Issue 143 Availability of Chilled Water System and Room Cooling

Discussion:

This issue relates to the need to maintain air cooling systems in some rooms containing safety-related system components.

AP1000 Response:

This issue does not apply to the AP1000. The AP1000 does not rely on active safety systems to provide safe shutdown of the plant. A total loss of HVAC systems will not prevent a safe shutdown.

Issue 153 Loss of Essential Service Water in LWRs

Discussion:

This issue is related to the reliability of essential service water and the failure of such systems due to fouling mechanisms, ice effects, design deficiencies, flooding, multiple equipment failures, and personnel errors. This issue has been the subject of a number of generic communications from the NRC staff.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not rely on the service water and component cooling water systems to provide safety-related safe shutdown.

Issue 163 Multiple Steam Generator Tube Leakage

Discussion:

This issue identifies a safety concern associated with potential multiple steam generator tube leaks triggered by a main steam line break outside containment that cannot be isolated. This sequence of events could lead to core damage due to the loss of all primary system coolant and safety injection fluid in the refueling water storage tank.

AP1000 Response:

The AP1000 plant response to a main steam line break (MSLB) scrams the reactor automatically and removes decay heat via the intact generator or the PRHR heat exchanger. If the MSLB is not isolated, the RCS will continue to lose coolant after shutdown through leaking steam generator tubes; the plant responds to the scenario as a small LOCA. The core makeup tanks drain and produce a low level signal. The plant protection and monitoring system depressurizes the RCS via the automatic depressurization system (ADS). The core remains covered throughout the scenario. Once the RCS is depressurized, the much lower reactor coolant system pressure stops the water loss through the leaking steam generator tubes. Therefore, no long-term core uncovery is expected.

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Issue 168 Environmental Qualification of Electrical Equipment

Discussion:

This issue is related to the effects of cable aging and whether the licensing basis for older plants should be reassessed or enhanced in connection with license renewal, or whether they should be reassessed for the current license term.

AP1000 Position:

This issue applies to operating plants and does not apply to the AP1000.

Issue 185 Control of Recriticality Following Small-Break LOCAs in PWRs

Discussion:

This issue is related to the potential for large reactivity transients, including prompt criticality, and significant heat generation resulting from natural circulation flow of unborated water formed in steam generators following small-break LOCAs.

AP1000 Position:

This scenario is not a safety concern for the AP1000 because of the passive safety systems designed to mitigate the consequences of a LOCA. Specifically, the automatic depressurization system operates to reduce primary system pressure and, thus, prevents significant heat transfer in the steam generators. Consequently, the steam generators should not generate any significant amount of boron-free condensate via reflux condensation over an extended period during a LOCA event. In the AP1000 design, the steam generator functions as a "heat source" as the RCS depressurizes, rather than a "heat sink" as it does in conventional PWR designs. Therefore, the differential temperature across the primary and secondary side of the generators is such that steam from the reactor will not condense on the tubes.

Another important design feature of the AP1000 that reduces the significance of this event is the elimination of the loop seal in the inlet to the reactor coolant pump. By elimination of the crossover leg piping, a large volume of boron-free condensate cannot collect in the loop piping. Thus, restart of the reactor coolant pumps following a LOCA will not result in a large slug of unborated water entering the core.

Post-LOCA, the PRHR heat exchanger can act as a heat sink and potentially could be a source of unborated water post-LOCA. However, condensate from the PRHR heat exchanger outlet mixes with the borated injection from the core makeup tanks and accumulators, and adequately mixes in the reactor vessel downcomer to prevent post-LOCA boron dilution. Long-term boration of the core is provided by the injection from the borated IRWST.

Issue 186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants

Discussion:

This issue concerns licensees operating within the regulatory guidelines of Generic Letter 85-11 that may not have taken adequate measures to assess and mitigate the consequences of dropped heavy loads.

FSAR Position:

There are no planned heavy load lifts outside those already described in the DCD. However, over the plant life there may be occasions when heavy loads not presently addressed need to be lifted (i.e. in

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support of special maintenance/ repairs). For these occasions, special procedures are generated that address the activity. Further discussion is provided in Subsection 9.1.5.3.

Issue 189 Susceptibility of Ice Condenser and Mark III Containments to Early Failure From Hydrogen Combustion During a Severe Accident Description

Discussion:

This issue concerns the early containment failure probability for ice condenser and BWR MARK III containments given the relatively low containment free volume and low containment strength in these designs.

FSAR Position:

The AP1000 design does not have an ice condenser containment or a Mark III containment. Therefore, this issue is not addressed here.

Issue 191 Assessment of Debris Accumulation on PWR Sump Performance

Discussion:

This issue addresses new contributors to debris and possible blockage of PWR sumps. Generic Letter (GL) 2004-02 (Reference 2), issued in September 2004, identified actions that utilities must take to address the sump blockage issue. The NRC position is that plants must be able to demonstrate that debris transported to the sump screen after a LOCA will not lead to unacceptable head loss for the recirculating flow. For the AP1000, this requirement is interpreted as demonstrating that debris transported to recirculating screens will not significantly impede flow through the PXS and will not adversely affect the long-term operation of the PXS.

AP1000 Position:

The AP1000 Nuclear Power Plant uses natural recirculation for cooling the core following a loss of coolant accident (LOCA).

Screens are provided in strategic areas of the plant to remove debris that might migrate with the water in containment and adversely affect core cooling. Accordingly, it must be assured that the screens themselves are not susceptible to plugging.

Technical report APP-GW-GLR-079 (Reference 71) evaluates the potential for debris to plug the AP1000 screens consistent with Regulatory Guide 1.82 Revision 3 and subsequently issued Nuclear Regulatory Commission guidance. The evaluation considers the various potential contributors to screen plugging. It considers debris that could be produced by a LOCA as well as resident fibers and particles that could be present in containment prior to the LOCA. It considers the AP1000 containment design, equipment locations, and containment cleanliness program. The evaluation uses debris characteristics based on sample measurements from operating plants and evaluates the generation of chemical precipitants considering materials used inside the AP1000 containment, the post-accident water chemistry, and applicable research and testing. The AP1000 screen designs are acceptable.

Issue 191 Assessment of Debris Accumulation on PWR Sump Performance (REV. 1)

Discussion:

Results of research on BWR ECCS suction strainer blockage identified new phenomena and failure modes that were not considered in the resolution of Issue A-43. In addition, operating experience identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings.

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FSAR Position:

The design aspects of this issue are addressed by the DCD. The protective coatings program controls the procurement, application, inspection, and monitoring of Service Level I and Service Level III coatings with the quality assurance features discussed above. The protective coatings program complies with Regulatory Guide 1.54, and is controlled and implemented by administrative procedures. The program is discussed in Subsection 6.1.2.1.6.

Administrative procedures implement the containment cleanliness program. Implementation of the program minimizes the amount of debris that might be left in containment following refueling and maintenance outages. The program is consistent with the containment cleanliness program used in the evaluation discussed in Subsection 6.3.8.2. The program is discussed in Subsection 6.3.8.1.

Issue 196 Boral Degradation

Discussion:

The issue specifically addresses the use of Boral in long-term dry storage casks for spent reactor fuel.

FSAR Position:

Long-term dry storage casks for spent reactor fuel are not used and therefore this issue is not addressed here.

1.9.4.2.4 Human Factors Issues

These issues were outlined in the Human Factors Program Plan and are documented in NUREG-0985, Revision 1. The Human Factors Program Plan includes the human factors tasks required to address NUREG-0660.

HF4.4 Guidelines for Upgrading Other Procedures

Discussion:

The need was evaluated to develop technical guidance for use in upgrading normal operating procedures and abnormal operating procedures, similar to what the NRC staff completed for emergency operating procedures. NUREG-0933 classified this item as resolved with no new requirements.

AP1000 Response:

The process to manage the development, review and approval of AP1000 Normal Operating, Abnormal Operating, Emergency Operating, Refueling and outage planning, Alarm response, Administrative, Maintenance, Inspection, Test and Surveillance Procedures as well as the procedures which address the operation of post-72 hour equipment is delineated in APP-GW-GLR-040 (Reference 72).

Writer's Guidelines have been developed which control the preparation of Normal Operating Procedures and Two-Column Format Procedures. The Writer's Guidelines establish programmatic guidelines. The criteria and methodology for procedure development is described in this technical report and in Westinghouse Writer's Guidelines, and Human Factors-related procedures have been developed in accordance with these criteria/guidelines.

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HF5.1 Local Control Stations

Discussion:

Human Factors Issue 5.1 addresses the need to develop additional guidance for the design of local control stations.

AP1000 Response:

The AP1000 local control stations are designed using the same human factors engineering (HFE) design process as is used for the main control room (MCR). The human factors engineering design process is described in Chapter 18. Section 18.8 provides a description of the human system interface (HSI) design element of the overall design process. As part of the human system interface design process, design guidelines for each interface, such as workstation displays, are generated. These guidelines are used when designing the respective interface and control stations. This provides consistency of human system interface design, including local control stations, with the main control room.

HF5.2 Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation

Discussion:

Human Factors Issue 5.2 addresses review criteria for human factors aspects of advanced controls and instrumentation.

AP1000 Response:

Chapter 18 describes the human factors engineering (HFE) program for the AP1000. Section 18.4 includes a description of the Functional Requirements Analysis and Allocation (element 3) for the AP1000. The objective of this allocation process is to define the AP1000 safety function requirements and allocate functions between the human and the machine appropriately. Section 18.8 also presents the implementation plan for the human system interface (HSI) design. This description of the human system interface design process includes the development of design guidelines, the execution of man-in-the-loop concept testing, review of human system interface design, and the use of a full-scale mockup.

The AP1000 human system interface (HSI)/man-machine interface (MMI) includes the following resources:

- Alarm system (also identified as alarm presentation system)
- Computerized Procedure System
- Plant Information System
- PMS Safety Display Subsystem
- Controls (dedicated and soft)
- Wall Panel Information System (WPIS)

The implementation plan for the design of each of these human system interfaces (HSI design) is described in Section 18.8. The mission statements and high-level information for each of these resources is also provided in Section 18.8. The plant information system provides display at the operators workstation. The protection and safety monitoring system provides qualified (Class 1E) displays to the operator, located at the dedicated safety panel. The alarm system provides alarm overviews which are integrated into the wall panel information system and it provides alarm system displays at the operator's workstation. Alarms are integrated into the workstation displays. There is a navigational link from an alarm system display for a specific alarm to its associated alarm response procedure as presented to the operator by the alarm presentation system. Design guidelines for each

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human system interface is developed as part of the human system interface design (as described in Section 18.8). These design guidelines are developed from existing industry guidelines and considerations specific to the technology planned for the human system interface. Human factors engineering specialists are part of the human factors engineering/ man-machine interphase design team (Section 18.2) and were involved in the development of the design guidelines.

1.9.5 Advanced Light Water Reactor Certification Issues

This subsection addresses the advanced light water reactor issues identified by the NRC in SECY-90-016 (Reference 29), in the February 27, 1992 NRC letter from D. M. Crutchfield to E. E. Kintner (Reference 30).

1.9.5.1 SECY-90-016 Issues

The following issues were outlined in SECY-90-016 (Reference 29).

1.9.5.1.1 Advanced Light Water Reactor Public Safety Goal

NRC Position:

Based on current regulatory guidance, including the NRC Severe Accident Policy Statement, Standardization Policy Statement, and Safety Goal Policy Statement, it is expected that any new standard plant design will result in a higher level of severe accident safety than current plant designs. This is achieved by improving safety and by striking a balance between accident prevention and mitigation.

The overall objective of the public safety goal is to significantly reduce or eliminate the likelihood of known major safety issues.

The safety goals approved by the NRC in the Staff Requirements Memorandum to SECY-90-016 (Reference 31) are as follows:

- The mean core damage frequency target for each design should be less than 1.0×10^{-4} per reactor year.

- The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation, where a large release is defined as one that has a potential for causing an early offsite fatality.

AP1000 Response:

The AP1000 level 1, 2, and 3 PRA evaluations for both internal and external events (excluding seismic events) demonstrate conformance with the NRC safety goals. The AP1000 PRA evaluates shutdown events and provides additional information and specific results.

1.9.5.1.2 Use of Physically-Based Source Term

NRC Position:

As noted in SECY-95-172 (Reference 57), the NRC plans to use the accident source term model from NUREG-1465 (Reference 58). This source term model provides a physically based approach to modelling of activity releases from the reactor core to the containment in the event of a core degradation accident. As discussed in SECY-94-302 (Reference 59), for the design basis accident, release of activity from the core will not be assumed to extend beyond the in-vessel release phase.

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In calculating the radiological consequences of accidents, as stated in Reference 57, the NRC intends to use the model presented in SECY-94-194 (Reference 60) which identifies the proposed changes to 10 CFR Parts 50 and 100. The pertinent features that will be applied to the determination of accident radiological consequences are:

- In place of thyroid and whole body dose limits, dose limits are specified as total effective dose equivalent (TEDE). The offsite dose limits of 25 rem whole body and 300 rem thyroid are replaced by a limit of 25 rem TEDE. The dose limit for the control room operators (currently identified in SRP Section 6.4 as 5 rem whole body, 30 rem thyroid, and 30 rem beta skin) is replaced by 5 rem TEDE which is consistent with GDC 19.

- Instead of calculating the site boundary dose over the first two hours of the accident, the dose is to be calculated for the two hour interval over which the highest dose would be calculated.

AP1000 Response:

The AP1000 radiological consequence analysis utilizes the accident source term provided in Regulatory Guide 1.183.

1.9.5.1.3 Anticipated Transients Without Scram (ATWS)

NRC Position:

This former unresolved safety issue was resolved with the issuance of Rule 10 CFR 50.62. Requirements for currently operating pressurized water reactors include diverse reactor trip (except for Westinghouse plants) and diverse actuation of auxiliary feedwater and turbine trips.

The Staff Requirements Memorandum to SECY-90-016 (Reference 31) approved the requirement for diverse reactor trip systems for evolutionary advanced light water reactors. However, it added that if the applicant can demonstrate that the consequences of an anticipated transient without reactor trip are acceptable, the NRC should accept the demonstration as an alternative to the diverse reactor trip system.

AP1000 Response:

The AP1000 complies with the current rules on an anticipated transient without reactor trip as specified in 10 CFR 50.62.

The AP1000 design includes the following design features aimed at minimizing the probability of occurrence of an anticipated transient without reactor trip and at mitigating the consequences if it occurs.

- The design of the protection and safety monitoring system is highly reliable, using a two out of four coincidence logic and featuring continuous diagnostic testing. The system incorporates fail-safe features to the extent practical. It is designed to generate a reactor trip signal and to generate an actuation signal for most engineered safety features components when protection system failures occur.

- For a reactor trip, the switchgear consists of eight circuit breakers arranged in a two out of four matrix located in two separate cabinets. The trip is implemented by undervoltage trip attachments and diverse shunt trip devices on the circuit breakers. To initiate a reactor trip, power is interrupted to the undervoltage trip attachment, while the shunt trip attachment is energized. Either device trips the breaker. The eight-breaker configuration permits testing of the reactor trip breakers without the use of auxiliary bypass breakers.

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- The reactor trip switchgear can be actuated manually from the main control room by reactor trip switches. The reactor trip switches are connected to the undervoltage trip attachment via the protection and safety monitoring system reactor trip matrix termination units and interposing relays. The reactor trip switches are also connected to the shunt trip attachment via the reactor trip matrix termination units. This manual trip signal does not pass through any software-based portion of the protection and safety monitoring system. In addition, it is possible to manually initiate a reactor trip from the main control room by turning off the motor-generators that provide power for control rod operation.

- A nonsafety-related diverse actuation system is included in the AP1000 design. The diverse actuation system inserts control rods by de-energizing the field windings of the control rod motor-generators.

- The diverse actuation system trips the turbine and diversely actuates selected other engineered safeguards functions. Additional details of the diverse actuation system are included in Section 7.7.

Section 15.8 describes the evaluation of an anticipated transient without reactor trip.

1.9.5.1.4 Midloop Operation

NRC Position:

Loss of decay heat removal function has occurred on a number of occasions in operating plants. In response to these events, the NRC issued Generic Letter 87-12 requesting that operating plants provide information regarding mid-loop operation. Generic Letter 88-17 requested additional information and provided guidance to operating utilities. Subsequent NRC evaluations have indicated that loss of decay heat removal during midloop operation may contribute significantly to public risk.

It is the NRC position that for future plants, conformance with Generic Letter 88-17 is insufficient, and additional hardware features should be incorporated into the design.

The Staff Requirements Memorandum to SECY-90-016 (Reference 31) approved the proposed NRC position, with the following four additional recommendations made by the ACRS:

- Design provisions to help ensure continuity of flow through the core and residual heat removal system with low liquid levels at the junction of the decay heat removal system suction lines and the reactor coolant system

- Provisions to ensure availability of reliable systems for decay heat removal

- Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the decay heat removal system suction lines and the reactor coolant system

- Provisions for maintaining containment closure or for rapid closure of containment openings.

AP1000 Response:

The following features are incorporated into the design of the reactor coolant system and the normal residual heat removal system for continued performance of the residual heat removal function during midloop operation:

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- The layout of the reactor coolant system hot leg piping and the steam generator channel head is such that installation of the nozzle dams can be performed with an 80 percent level in the hot leg piping. This is about 9 inches above the actual hot leg piping midplane elevation. (The hot leg piping has a 31-inch inside diameter.)

- A specially designed vortex breaker is used for the normal residual heat removal system suction nozzle. This vortex breaker connects vertically to the bottom of the hot leg piping. The normal residual heat removal system suction piping is connected to the bottom of this vortex breaker. With the vortex breaker, the amount of air entrainment remains below 10 percent unless the hot leg is essentially drained. Therefore, the potential for a loss of normal residual heat removal system flow and damage to the normal residual heat removal pump is substantially reduced.

- The normal residual heat removal pump suction piping is designed to be self-venting to a condition acceptable for RNS pump operation such that no manual venting is required to start the pumps if stopped on cavitation due to excessive air entrainment. This is achieved by sloping most of the RNS pump suction piping upward from the RNS pump suction to the reactor coolant system hot leg. In the level portions of piping, there are no local high points. If the pump should stop during midloop operation, any air bubbles present in the pump or suction piping are vented back up through the suction line to the water surface in the hot leg. This feature allows the operator to rapidly restart the pump with no additional action required.

- The normal residual heat removal pumps are designed to minimize susceptibility to cavitation and other adverse conditions. The plant piping layout configuration (such as piping elevations and routing) and the available and required net positive suction head characteristics are designed such that the normal residual heat removal pumps normally operate without the need for throttling a residual heat removal control valve when the level in the reactor coolant system is reduced to a midloop level. If the reactor coolant system is at saturated conditions and midloop level, some throttling of a residual heat removal flow control valve is necessary to maintain adequate net positive suction head. The normal residual heat removal system is readily restored after a temporary loss of decay heat removal.

- The core makeup tanks, accumulators, and the in-containment refueling water storage tank are isolated, but can be manually actuated during midloop operations. In addition, the in-containment refueling water storage tank is automatically actuated on a sustained loss of shutdown decay heat removal. This arrangement provides a reliable water source for maintaining the reactor coolant system inventory that is either automatically or manually actuated.

- Redundant narrow-range level instrumentation indicates the reactor coolant system water level between the bottom of the hot leg and the top of the steam generator inlet elbow. Indication and low level alarms are provided in the main control room. In addition, this instrumentation actuates the in-containment refueling water storage tank makeup.

- Wide-range pressurizer level instrumentation used during cold plant operations is expanded to the bottom of the hot legs. This provides a continuous level indication in the main control room, from the normal level in the pressurizer to the range of the two narrow-range hot leg level instrumentation.

- Normal residual heat removal system heat exchanger discharge flow instrumentation provides main control room indication of return flow to the reactor vessel. A low-flow alarm alerts the operator to a decrease in normal residual heat removal system return flow from either heat exchanger.

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- The drain-down of the reactor coolant system to the midloop operating level and the subsequent reactor coolant system inventory control during midloop operation are performed by the operator from the main control room.

The plant design precludes the need to locally coordinate actions in the containment with the main control room operators to control the reactor coolant system drain-down rate and level.

- The normal residual heat removal temperature instruments upstream of the heat exchangers indicate reactor coolant system hot leg temperature when in reduced inventory conditions. In addition, at least two incore thermocouple channels are available to directly measure the core exit temperature during midloop residual heat removal operation. These two thermocouple channels are associated with separate electrical divisions.

- The automatic depressurization system first-, second-, and third-stage valves, connected to the top of the pressurizer, are open whenever the core makeup tanks are blocked during shutdown conditions while the reactor vessel upper internals are in place. This provides a vent flow path that, when combined with actuation of the automatic depressurization system fourth-stage valves, sufficient steam venting area is provided to reduce reactor coolant system pressure. Actuation of ADS stage 4, either automatically or manually (as described in Subsection 19E.4.8.3) allows the in-containment water storage tank to provide injection flow following a loss of decay heat removal.

Administrative controls require containment closure capability in modes 5 and 6, during reduced inventory operations, and when the upper internals are in place. Containment closure capability is defined as the capability to close the containment prior to core uncovery following a loss of the normal decay heat removal system (that is, normal residual heat removal system). The containment design also includes penetrations for temporary cables and hoses needed for shutdown operations. These penetrations are isolated in an emergency.

In addition to these design features, appropriate procedures are defined to guide and direct the operator in the proper conduct of midloop operation and to aid in identifying and correcting abnormal conditions that might occur during shutdown operations.

1.9.5.1.5 Station Blackout

NRC Position:

The NRC has issued NUREG-0649 (Reference 34), NUREG-1032 (Reference 35), and NUREG-1109 (Reference 36) to address the unresolved safety issue of station blackout (USI-44). See Subsection 1.9.4 for a discussion of USI-44.

To resolve this issue, the NRC published 10 CFR 50.63 and Regulatory Guide 1.155, which establish new requirements so that an operating plant can safely shut down following a loss of all ac power. SECY-94-084 (Reference 67), discusses station blackout for passive plants.

AP1000 Response:

The AP1000 is in conformance with the NRC guidelines for station blackout.

The AP1000 design minimizes the potential risk contribution of station blackout by not requiring ac power sources for design basis events. Safety-related systems do not need nonsafety-related ac power sources to perform safety-related functions.

The AP1000 safety-related passive systems automatically establish and maintain safe, stable conditions for the plant following design basis events, including an extended loss of ac power

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sources. The passive systems can maintain these safe, stable conditions after design basis events for at least 72 hours, without operator action, following a loss of both onsite and offsite ac power sources. Subsection 1.9.5.4 provides additional information on long-term actions following an extended station blackout beyond 72 hours.

The AP1000 also includes redundant nonsafety-related onsite ac power sources (diesel-generators) to provide electrical power for the nonsafety-related active systems which provide defense in depth.

AP1000 design features that mitigate the consequences of a station blackout are as follows:

- A full-load rejection capability to reduce the probability of loss of onsite power
- Safety-related passive residual heat removal heat exchanger
- Safety-related passive containment cooling
- Bleed and feed capability, using the safety-related automatic depressurization system in conjunction with the water available from the core makeup tanks, the accumulators, and the in-containment refueling water storage tank
- Class 1E batteries sized for 72 hours of operation under station blackout conditions
- Nonsafety-related reserve auxiliary transformers to provide power to selected ac power systems
- A nonsafety-related ac power system that includes two diesel-generators that automatically start on loss of normal ac power and offsite power
- An automatic nonsafety-related load-sequencing circuit that starts the following redundant nonsafety-related equipment after a loss of normal ac power and offsite power, once the associated diesel-generator is started:
  - Startup feedwater pump
  - Component cooling water pump
  - Service water pump
- Reactor coolant pumps without shaft seals
- Passive cooling for the rooms containing equipment assumed to operate during station blackout conditions (the protection and safety monitoring system cabinet rooms and the main control room) so that this equipment continues to operate. (Section 6.4 provides additional information.)

Training and procedures to mitigate a 10 CFR 50.63 “loss of all alternating current power” (or station blackout (SBO)) event are implemented in accordance with Sections 13.2 and 13.5, respectively. As recommended by NUMARC 87-00 (Reference 201), the SBO event mitigation procedures address response (e.g., restoration of onsite power sources), ac power restoration (e.g., coordination with transmission system load dispatcher), and severe weather guidance (e.g., identification of actions to prepare for the onset of severe weather such as an impending tornado), as applicable. The AP1000 is a passive design and does not rely on offsite or onsite ac sources of power for at least 72 hours after an SBO event, as described above.

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Restoration from an SBO event will be contingent upon ac power being made available from any one of the transmission lines described in Section 8.2 or any one of the standby diesel generators.

1.9.5.1.6 Fire Protection

NRC Position:

Current fire protection criteria are contained in GDC 3 and 10 CFR 50.48, guidelines for compliance with these criteria are provided in the Standard Review Plan, Section 9.5.1, including Branch Technical Position CMEB 9.5-1. Reference 37 identifies the following enhancements:

- Alternative, dedicated shutdown capability for main control room fires.
- Safe shutdown capability required for a fire in any other fire area, without reliance on any equipment in that area or re-entry into that area for repairs or for performance of operator actions.
- Fire protection for redundant shutdown systems in the reactor containment building must be provided to ensure, to the extent practicable, that on shutdown the division will be free of fire damage.
- Migration of smoke, hot gases, or fire-suppressant chemicals into other applicable fire areas must be minimized by design to prevent any adverse impact on safe shutdown capability, including operator actions.

SECY-98-161 (Reference 66) presents the results of the NRC review of the AP1000 Fire Protection System.

AP1000 Response:

Enhanced fire protection has been one of the goals of the AP1000 design. The following physical separation philosophy is used:

Outside Containment:

- Within the nuclear island, redundant divisions of safety-related equipment outside containment are located in safety-related areas separated from each other and from other areas in the plant by fire barriers with a minimum fire resistance rating of 3-hours to provide that safe shutdown can be achieved. Since most safety-related mechanical equipment is located inside containment, this applies primarily to the protection and safety monitoring system and the Class 1E dc and UPS system.
- Each safety-related area is provided with ventilation isolation provisions at the fire barrier boundaries to minimize the migration of smoke, hot gasses, or fire suppressant chemicals into other safety-related areas. Fiber-optic cables are used to provide communication between redundant protection and safety monitoring divisions.
- Exceptions to the use of three-hour fire barriers outside containment are made only in cases where physical separation conflicts with other requirements or where the equipment is not clearly division oriented, such as the main control room, the remote shutdown room, the main steam tunnel, and the passive containment cooling system valve room.

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Inside Containment:

- The containment is a single fire area. Separation by three-hour fire barriers inside containment is not practical due to issues of hydrogen venting, compartment pressure equalization, and during high-energy line breaks and for system functionality. To the extent practical, separation is provided between redundant safety-related equipment.

- Separation between redundant safety-related equipment is accomplished by using existing structural walls. Where this is not possible, other methods are used, such as physical separation with no intervening combustibles.

- To the extent practical, the containment is split into two different fire zones for the purpose of routing of protection and safety monitoring system cabling and electrical power cabling. Divisions A and C cabling is routed below the operating deck, while Divisions B and D cabling is routed above the operating deck. Additional separation is provided by existing floors and walls and by the physical separation of cabling runs. Protection for the primary input sensors and the final actuation devices is accomplished by the physical separation of the various sensors and components using existing containment walls as barriers.

- The in-containment fire area contains reduced combustible material due to the use of sealless reactor coolant pump motors that do not use oil lubrication and due to strict combustible material limitations.

Main Control Room:

- Functionality requirements dictate that the main control room be a single fire zone. Features are included in the main control room to:

  – Reduce the probability of fire initiation
  – Reduce the likelihood of fire spreading
  – Increase the probability of fire detection
  – Effectively mitigate the effects of a fire

- In the event of main control room evacuation, safe shutdown conditions are established and maintained using the remote shutdown workstation.

See Appendix 9A.3 for information on the main steam tunnel and the passive containment cooling system valve room. See Subsection 9.5.1 and Appendix 9A for additional information.

1.9.5.1.7 Intersystem LOCA

NRC Position:

Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. This may result in a core melt accident with an energetic release outside the containment building that could cause a significant offsite radiation release.

It is the NRC position that designing interfacing systems to withstand full reactor pressure is an acceptable means of resolving this issue. The Staff Requirements Memorandum to SECY-90-016 (Reference 31) added that consideration should be given to all elements of the low-pressure system (such as instrument lines, pump seals, heat exchanger tubes, and valve bonnets). For interfacing systems not designed to withstand full reactor coolant system pressure, it is necessary to provide leak testing capability for the pressure isolation valves, main control room position indication for de-energized reactor coolant system isolation valves, and high pressure alarms to alert control room

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operators when increasing reactor coolant system pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

AP1000 Response:

The AP1000 has incorporated various design features to address intersystem loss-of-coolant accident challenges. These design features result in very low AP1000 core damage frequency for intersystem loss-of-coolant accidents compared with operating nuclear power plants. The design features are primarily associated with the normal residual heat removal system and are discussed in Section 3 of WCAP-15993 (Reference 56) as well as Subsection 5.4.7. WCAP-15993 was prepared to document the evaluation of the AP1000 for conformance to the intersystem loss-of-coolant accident regulatory criteria identified in various NRC documents. See that document for additional information on conformance to intersystem loss-of-coolant accident regulatory criteria.

1.9.5.1.8 Hydrogen Generation and Control

NRC Position:

It is the NRC position that the likelihood of early containment failure from hydrogen combustion should be reduced. Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, advanced light water reactors should be designed to:

- Accommodate hydrogen equivalent to 100 percent metal-water reaction of the fuel cladding
- Limit containment hydrogen concentration to no greater than 10 percent

Further, because hydrogen control is necessary to preclude local concentrations of hydrogen below detonable limits, and given uncertainties in present analytical capabilities, advanced light water reactors should provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) includes the lessening of pressure and temperature loadings on the containment and essential equipment.

AP1000 Response:

The AP1000 design includes mechanisms for monitoring and controlling hydrogen inside the containment. The containment hydrogen control system maintains hydrogen concentrations below 10 percent following the reaction of 100 percent of the zircaloy cladding.

Passive autocatalytic hydrogen recombiners control hydrogen concentration following design basis events. Nonsafety-related hydrogen igniters control rapid releases of hydrogen during and after postulated events with degraded core conditions or with core melt.

Sufficient vent area is provided for each subcompartment in the containment to prevent high local concentrations of hydrogen.

The containment air filtration system provides a capability to purge the containment atmosphere.

See Subsection 6.2.4 for additional information.

1.9.5.1.9 Core-Concrete Interaction - Ability to Cool Core Debris

NRC Position:

Containment integrity could be breached in the event of a severe accident in which the core melts through the reactor vessel, resulting in interaction between core debris and concrete, which can generate large quantities of hydrogen and other gases. It is the NRC position that sufficient reactor

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cavity floor space be provided to enhance debris spreading, and that a method for quenching debris in the reactor cavity be incorporated. The NRC staff has not formulated specific criteria for debris bed coolability and reviews each vendor's design to determine how they address the general criteria for debris spreading and quenching.

AP1000 Response:

The AP1000 design provides superior protection against core-concrete interaction by reliably depressurizing the reactor vessel and flooding the reactor cavity to cool the vessel and prevent debris from relocating from the vessel into the containment. Based on the DOE/ARSAP analysis of the thermal-hydraulics of in-vessel debris retention (see Section 19.16 and Appendix 19B as supported by Theofanous, T. G., et al., Reference 62) performed using the Risk Oriented Accident Analysis Methodology, the AP1000 has a large margin to reactor vessel failure in the depressurized, flooded cavity condition. This strategy eliminates the large uncertainties associated with ex-vessel debris relocation that could result in containment failure even while meeting the NRC criteria for debris coolability in the cavity.

In the event that cavity flooding fails, the floor area under the vessel provides debris spreading area to enhance the coolability of the debris. The AP1000 containment design drains the water from the reactor coolant system, core makeup tanks and accumulators to the reactor cavity to provide enough water to quench ex-vessel debris. The heat is ultimately removed from the containment via the passive containment cooling system, and the condensate is returned to the cavity to continue to provide cooling water to the debris bed.

1.9.5.1.10 High Pressure Core Melt Ejection

NRC Position:

Direct containment heating associated with the ejection of molten core debris, under high pressure, from the reactor vessel can result in a rapid addition of energy to the containment atmosphere. It is the NRC position that, pending completion of ongoing research, it is prudent to provide protection against this potential failure mode. This protection should include the following two aspects:

- Providing a rate of reactor coolant system depressurization to preclude molten core ejection and creep rupture of steam generator tubes

- Arranging the reactor cavity so that high-pressure core debris ejection resulting from reactor vessel failure does not impinge on the containment boundary

AP1000 Response:

The AP1000 design includes an automatic depressurization system that is redundant, diverse, independent of ac power sources, and automatically actuated. The automatic depressurization system can also be manually actuated. Any of the automatic depressurization system lines can sufficiently reduce the reactor coolant system pressure to help preclude direct containment heating. Subsection 5.4.6 and Section 6.3 provide additional information on the automatic depressurization system.

In addition, the reactor cavity region and lower containment of the AP1000 are designed to preclude transport of significant core debris to the upper containment in the unlikely event of a high pressure melt ejection scenario from the reactor vessel. This is a passive feature involving the geometric configuration of the reactor cavity lower containment. There is no direct pathway from the cavity to the upper compartment.

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1.9.5.1.11 Containment Performance

NRC Position:

The NRC opinion is that because there are substantial uncertainties in core damage predictions, and because it is very important to maintain defense in depth, it is necessary that the containment boundary serve as a reliable barrier against fission product release for credible severe accident challenges. Hence, a containment performance criterion has been proposed by the NRC.

The objective of the containment performance criterion is to provide a leaktight barrier against radioactive releases for two distinct categories of severe accident challenges:

- Rapid energy release, hydrogen combustion, and initial release of stored reactor coolant system energy

- Slow energy release, including decay heat and noncondensible gas generation, due to core-concrete interaction

The NRC position is that the reactor containment boundary should serve as a reliable barrier against fission product release for credible severe accident challenges. A conditional containment failure probability of 0.1 should be used unless a deterministic containment performance goal can offer comparable protection.

An alternate deterministic criterion proposed in SECY-90-016 (Reference 29) states that "...The containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage..."

This capability should, to the extent practical, be provided by the passive capability of the containment and any related passive design features. The NRC further believes that following this 24-hour period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

AP1000 Response:

The AP1000 design includes several features to minimize the potential for large fission product releases in the event of a severe accident. These features are aimed at both the prevention and the mitigation of severe accident phenomena that can threaten containment integrity. An adequate margin to containment performance is maintained.

The AP1000 containment is continuously cooled by natural air circulation outside the steel shell. During accident conditions, water drains on the outside of the containment vessel to increase heat transfer. The containment design best-estimate performance analysis alone shows that the maximum containment pressure reached maintains the containment shell stresses below the ASME Code Service Level C stress intensity limits, using a factor of safety of 1.5 for buckling of the top head.

Additionally, the probability of containment bypass scenarios is reduced by improved containment isolation, by designing to protect against interfacing system LOCAs, thereby reducing the associated core melt frequency, and by reducing the steam generator tube rupture core melt frequency.

The interfacing system LOCA core melt frequency is reduced by the use of several features, including effective leak testing of the normal residual heat removal system motor-operated isolation valves. A third valve is provided to the normal residual heat removal system suction line. It is a motor-operated valve located outside containment. This prevents inadvertently aligning the reactor coolant system to the normal residual heat removal system. The normal residual heat removal system design

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pressure is 900 psig. Therefore the ultimate rupture strength of the system prevents it from failing when exposed to the normal reactor coolant system operating pressure (2250 psia). See the position on intersystem LOCA for additional information on the normal residual heat removal system design against overpressurization.

Steam generator tube rupture core melt frequency is reduced by incorporating multiple levels of defense that are both redundant and diverse. The first level of defense relies on the use of nonsafety-related active systems and operator action. The second level of defense uses safety-related passive systems and equipment, such as the core makeup tanks and passive residual heat removal heat exchangers, without the safety-related automatic depressurization of the reactor coolant system. The third level of defense uses the redundant and diverse safety-related automatic depressurization system valves to depressurize the reactor coolant system and initiate low-pressure passive injection. Any of these levels of defense can prevent core damage during a steam generator tube rupture event.

Finally, containment isolation capabilities are substantially improved by reducing the number of penetrations and the number of open paths. Most of the open containment penetration lines use fail-closed valves for automatic isolation.

1.9.5.1.12 ABWR Containment Vent Design

This issue is specific to BWRs and PWRs with ice condenser containments. Therefore this issue does not apply to the AP1000 design.

1.9.5.1.13 Equipment Survivability

NRC Position:

Safety-related equipment used to mitigate design basis events is subject to a comprehensive set of criteria such as redundancy, diversity, environmental qualification, and quality assurance to provide reasonable assurance that they perform their intended functions, if needed. However, equipment used to mitigate the effects of severe accidents should not be treated in the same manner because of large differences in the likelihood of occurrence. There should be reasonable assurance that the equipment will operate in the severe accident environment for which they are intended and over the time span for which they are needed. However, equipment provided only for severe accident protection need not be subject to the 10 CFR 50.49, environmental qualification requirements, 10 CFR 50, Appendix B quality assurance requirements, and 10 CFR 50 Appendix A, redundancy and diversity requirements.

AP1000 Response:

The equipment used to mitigate severe accidents is identified in the AP1000 PRA evaluation report. Because of the nature of the passive safety features of the AP1000, there is very little equipment in this category. Equipment used to mitigate severe accidents is designed to survive the environmental conditions identified in the AP1000 PRA evaluation (see Appendix 19D).

1.9.5.1.14 Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE)

NRC Position:

Currently, 10 CFR 100 requires that the magnitude of the operating basis earthquake be at least one-half that of the safe shutdown earthquake. This forces the safety-related system design at some plants to be controlled by the operating basis earthquake, but the NRC agrees that the operating basis earthquake should not control the safety-related system design. Therefore, the NRC recommends eliminating the operating basis earthquake from the design of systems, structures, and components. Until final rulemaking is approved for 10 CFR 100, Appendix A, the elimination of the

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operating basis earthquake from the design of passive plants will require an exemption from current regulations, with acceptable supporting justification from the designer. The details of this process will be resolved with the NRC through the appropriate code-related activities or supplemental regulatory guidance.

AP1000 Response:

The operating basis earthquake is not used as a design basis for AP1000 safety-related structures, systems, and components. For safety-related equipment, the safe shutdown earthquake is used as the design basis. In specifying design criteria for this earthquake, consideration is given to lower magnitude earthquakes having a greater probability of occurrence, as well as to larger magnitude earthquakes having a lower probability.

Cyclic stresses due to earthquakes are included in the design of those components sensitive to fatigue. Analysis methods and allowable stresses provide margin for the design requirements for the safe shutdown earthquake. Sections 3.7 and 3.10 provide additional information.

1.9.5.1.15 In-Service Testing of Pumps and Valves

NRC Position:

Periodic testing according to ASME Code, Section XI is required to confirm operability of safety-related pumps and valves. The NRC believes that these testing requirements do not necessarily verify the capability of the components to perform their intended safety function. To address this concern, the NRC has issued Generic Letters 89-04 (Reference 38) and 89-10 (Reference 39), and has proposed rulemaking to extend in-service testing beyond code components and to demonstrate capability to perform safety functions. Reference 29 includes the following provisions to be applied to safety-related pumps and valves (not limited to only ASME Code Class 1, 2, or 3):

- Piping design should incorporate provisions for full-flow testing (maximum design flow) of pumps and check valves.
- Designs should incorporate provisions to test motor-operated valves under design basis differential pressure.
- Check valve testing should incorporate the use of advanced, nonintrusive techniques to address degradation and performance characteristics.
- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, nonintrusive techniques.

In June 1990, the NRC position was approved, additionally noting that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

The NRC concluded that this was an issue for passive plant designs in SECY-94-084 (Reference 67), because the safety-related passive systems rely on the proper operation of equipment such as check valves and depressurization valves to mitigate the effects of transients.

AP1000 Response:

The AP1000 safety-related passive systems include the following design features:

- The AP1000 does not include any active ASME Section III safety-related pumps.

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- The motor-operated valve design is simplified by extending opening and closing times and by using simplified, conservative valve designs.
- Safety-related motor-operated valves are designed to be cycled with the plant at power.
- Features are included in the design to provide proper operational testing of the appropriate check valves, motor-operated valves, and air-operated valves, including flow and differential pressure testing during shutdown conditions.

Subsection 3.9.8.4 defines the responsibility for the in-service testing program for ASME Code Class 1, 2, and 3 valves.

Subsection 3.9.6 summarizes the requirements for the in-service testing program, including industry standards and NRC recommendations. The AP1000 system and valve designs generally allow implementation of the NRC recommendations in Generic Letters 89-04 and 89-10. Requirements for nonsafety-related pumps and valves that support the operation of systems that preclude unnecessary operation of the safety-related passive systems are outlined in Subsection 3.9.6.

The AP1000 in-service testing program provides for periodic testing of the safety-related passive system components. The safety-related passive system components and systems are designed to meet the intent of the ASME Code, Section XI, for in-service inspection.

The AP1000 is designed for the following basic types of in-service testing of safety-related components:

- Periodic functional testing of active components during power operation (such as cycling of specific valves)
- Periodic flow/differential pressure operability testing of active components
- Periodic leak testing of the containment isolation valves.
- Periodic system flow or heat transfer rate testing of passive safety-related injection or cooling features during plant shutdown

The passive system design includes specific features to support in-service test performance:

- Remotely operated valves can be exercised during plant operation.
- Level, pressure, flow, and valve position instrumentation is provided for monitoring passive system equipment during plant operation and testing.
- Permanently installed test lines and connections are provided for performance of the containment isolation valve leakage testing.

1.9.5.2 Other Evolutionary and Passive Design Issues

Other evolutionary and passive design issues were identified in Reference 30.

1.9.5.2.1 Industry Codes and Standards

NRC Position:

SECY-91-273 (Reference 40) discusses NRC concerns with the use of recently developed or modified design codes and industry standards that the ALWR vendors are using in applications, but

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that have not yet been reviewed by the NRC for acceptability. The NRC recommends using the newest codes and standards endorsed by the NRC in the review of passive design applications. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis.

AP1000 Response:

When the AP1000 design is based on revisions of industry codes and standards later than those required by NRC regulation, such use is explicitly discussed in the appropriate section. Use of codes and standards later than those recommended in NRC guidance documents is also discussed in the appropriate section.

Appendix 1A discusses regulatory guide conformance. For those standards endorsed by regulatory guides and subsequently superseded by a more recent revision, when the later revision is used its use is discussed or indicated in Appendix 1A.

1.9.5.2.2 Electrical Distribution

NRC Position:

The Commission approved the recommendations in SECY-91-078 (Reference 41) for evolutionary plant designs to include the following:

1. An alternate power source for nonsafety-related loads unless design margins for loss of nonsafety-related loads are no more severe than turbine-trip-only events in current plants

2. At least one offsite circuit to each redundant safety division supplied directly from offsite power sources with no intervening nonsafety-related buses

The applicability of this issue to passive designs is discussed in SECY-94-084 (Reference 67).

AP1000 Response:

See the response to station blackout in Subsection 1.9.5.1.

1.9.5.2.3 Seismic Hazard Curves and Design Parameters

NRC Position:

To assess the seismic risk associated with an ALWR design, EPRI proposed the use of generic bounding seismic hazard curves for sites in the central and eastern United States. EPRI proposes that these curves be used in the seismic PRA. NRC regulations do not require performance of a seismic PRA to determine site acceptability.

The NRC has compared the proposed EPRI ALWR seismic hazard bounding curve for rock sites to hazard curves derived by Lawrence Livermore National Laboratories (LLNL) using historical earthquake methodology in NUREG/CR-4885 and to hazard curves generated by EPRI for the Seabrook site. The LLNL hazard curves are generally higher than the EPRI results for the same sites.

The proposed EPRI bounding curve is exceeded for accelerations below 0.1g and the NRC questions the adequacy of the proposed EPRI bounding curve at higher peak accelerations. The NRC concludes that the EPRI bounding hazards curve is nonconservative and also that its use in a seismic PRA assessment would underpredict the core damage frequency. Therefore, the EPRI curves are not sufficiently conservative for ALWR designer use.

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The Combined License applicant must demonstrate that site-specific seismic parameters meet the certified design parameters, or a site-specific analysis will be required to confirm site acceptability.

AP1000 Response:

The AP1000 includes a seismic margin assessment performed in lieu of a seismic PRA. The seismic margin assessment follows the guidelines established in NUREG-1407 (Reference 42). This assessment demonstrates that the AP1000, located at a site having the most severe seismic inputs meeting the AP1000 site interface requirements, has a seismic risk comparable to that at existing nuclear power plants.

1.9.5.2.4 Leak-Before-Break

NRC Position:

GDC 4 provides the basis for the leak-before-break (LBB) analysis that has been approved for PWR primary piping, and the pressurizer surge, accumulator, and residual heat removal piping. In addition, it has been used for primary piping inside containment and for piping at least 6 inches nominal diameter and for both austenitic and carbon steel (clad with stainless) materials.

The NRC will evaluate the acceptability in ALWR designs, based on the justification provided by a deterministic fracture mechanics analysis submitted as part of the design. The NRC concluded that the analyses should be based on specific data, such as piping geometry, materials, and piping loads. However, the analyses may incorporate an initial set of bounding values and preliminary stress analysis results during the design certification phase. Subsequent verification of the preliminary analysis will be required.

The LBB approach has established certain limitations for excluding piping susceptible to failure from degradation mechanisms. In addition, the LBB introduced acknowledged inconsistency in the design basis, but the NRC published clarifications for the intended treatment of the containment, emergency core cooling systems, and environmental qualification in the LBB application.

The NRC position on LBB for the AP1000 is presented in SECY-95-172 (Reference 68).

AP1000 Response:

The AP1000 incorporates the leak-before-break approach for most high-energy lines inside containment that are 6 inches in diameter or larger. Detailed methodology and criteria are defined in Subsection 3.6.3 and are consistent with those accepted by the NRC on existing nuclear power plants.

1.9.5.2.5 Classification of Main Steam Line of Boiling Water Reactors (BWRs)

This issue is specific to BWRs and therefore does not apply to the AP1000 design.

1.9.5.2.6 Tornado Design Basis

NRC Position:

WASH-1300 (Reference 43) and Regulatory Guide 1.76 contain the current NRC regulatory position for design basis tornados. Based on a contractor review of Regulatory Guide 1.76, the NRC recommends a maximum tornado speed of 300 mph be used for design basis tornado for passive ALWR designs.

The tornado design basis requirements have been used in establishing structural requirements against effects not covered explicitly in review guidance such as Regulatory Guides or the SRP. The

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Combined License applicant will have to demonstrate that the design will also be sufficient to withstand other site hazards such as aviation crashes, nearby explosions, and explosion debris and missiles.

AP1000 Response:

The AP1000 is designed in accordance with the NRC recommendations for a maximum tornado wind speed of 300 mph, as described in Section 3.3. The AP1000 site interface defined in Chapter 2 provides information to evaluate other site hazards if appropriate.

1.9.5.2.7 Containment Bypass

NRC Position:

Reasonable efforts should be made to minimize the possibility of containment bypass leakage, and ALWR designs should allow for a certain amount of leakage in the containment design. The NRC is evaluating the need for containment spray for all ALWRs. The containment spray provides containment temperature and pressure suppression effects and scrubs the containment atmosphere of fission products, mitigating the effects on the fission product bypass distribution.

AP1000 Response:

Although the phenomenon described for this item is primarily applicable to BWRs, the AP1000 has a variety of design features that help to reduce the potential for containment bypass leakage.

The response to the containment performance issue in Subsection 1.9.5 provides additional information pertaining to various improvements that help to reduce containment bypass.

The safety-related passive containment cooling system design also contributes to the containment performance. The system includes multiple flow paths to provide cooling water for containment during severe accident conditions. The containment is also capable of successfully removing core decay heat with air-cooling alone.

The containment has a significantly reduced number of penetrations. The number of normally open containment penetrations is also reduced. The result is a low containment leak rate and a low probability of bypass.

The response to intersystem LOCA in Subsection 1.9.5.1 provides additional information pertaining to applicable AP1000 design features that reduce the potential for intersystem LOCA and the potential for containment bypass.

Improvements are made to the steam generator design, such as the use of improved tube materials and tube supports. These improvements reduce the potential for tube leakage, which contributes to a reduction in containment bypass. Subsection 5.4.2 provides additional information on the steam generator design.

During a steam generator tube rupture event, the safety-related passive core cooling system automatically mitigates the effects of the event, including automatic safety-related protection against steam generator overfill.

The safety-related passive core cooling system provides long-term pH control for the containment sump, which helps to reduce the levels of airborne radioactivity, thereby reducing the consequences of leakage from the containment. Section 6.3 includes additional information on the passive core cooling system.

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The diverse actuation system includes containment isolation features to provide isolation for the most risk-significant containment penetrations. AP1000 PRA Chapter 24 discusses the provisions for isolating risk significant containment penetrations.

The performance of the passive fission product removal process and minimal potential for containment bypass precludes the need for a safety-related containment spray system on AP1000.

1.9.5.2.8 Containment Leak Rate Testing

NRC Position:

SECY-91-348 (Reference 44) proposes changes to 10 CFR 50, Appendix J to allow an increased interval from 24 months to 30 months for Type C containment leakage rate tests, until rule change proceedings are completed.

AP1000 Response:

10 CFR 50 Appendix J has been revised since SECY-91-348 was issued. AP1000 type C testing and compliance with 10 CFR 50 Appendix J is discussed in Subsection 6.2.5.

1.9.5.2.9 Post-Accident Sampling System

NRC Position:

Regulatory Guide 1.97 and NUREG-0737 (Reference 45) provide guidance regarding the design of the post-accident sampling system. 10 CFR 50.34 required the capability to obtain and analyze samples from containment and the reactor coolant system that may contain TID-14844 source term radioactive materials, without exceeding specified radiation exposures. The analysis and quantification are required for certain specified radionuclides that are indicators of the degree of core damage, containment hydrogen, dissolved gases, chloride, and boron concentrations.

The NRC concluded that adequate capability for monitoring post-accident hydrogen is provided by the safety-grade containment hydrogen monitoring instrumentation.

The NRC requires sampling the reactor coolant system for dissolved hydrogen, chloride, and oxygen. The time for taking these samples can be extended to 24 hours after the accident.

The NRC requires sampling the reactor coolant system for boron and for activity measurements. The time for taking these samples can be extended to 8 hours after power operation for boron and 24 hours after power operation for activity measurements.

AP1000 Response:

The post-accident sampling system is a subsystem of the primary sampling system, described in Subsection 9.3.3.

The primary sampling system is designed to conform to the guidelines of the model Safety Evaluation Report on eliminating post-accident sampling system requirements from technical specifications for operating plants (Federal Register Volume 65, Number 211, October 31, 2000). The primary sampling system conforms with the most recent NRC position.

1.9.5.2.10 Level of Detail

NRC Position:

The Staff Requirements Memorandum for SECY-90-377 (Reference 47) provided guidance on the level of detail to be provided for a design certification application under 10 CFR 52. The guidance

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was that the application should include the information traditionally provided in a final safety analysis report, less the site-specific and as-procured information. This information should be supplemented by design inspections, tests, analysis, and acceptance criteria for those areas where the NRC is unable to make a final safety decision because of not having the site-specific information or the as-procured information, or because the technology is evolving so rapidly that it would be inappropriate to lock in the design.

AP1000 Response:

The AP1000 submittals are consistent with the requirements of 10 CFR 52 and the position in Reference 47.

1.9.5.2.11 Prototyping

NRC Position:

10 CFR 52.47 requires that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of normal operating conditions, transient conditions, and specified accident conditions. Further, the interdependent effects among the safety features of the design must be found acceptable by analysis, appropriate test programs, experience, or a combination thereof. SECY-91-057 (Reference 48) informed the Commission of the steps the NRC was taking to identify the research needs for the AP600. SECY-91-074 (Reference 49) outlined the process the NRC would use to determine the need for a prototype or other demonstration facility for advanced reactor designs. SECY-91-273 (Reference 40) presented to the Commission the staff's recommendations for reviewing, monitoring and approving the Westinghouse test program to support the AP600 design certification application. SECY-92-030 (Reference 50) presented the Commission with the NRC opinion that there was a need for a full-height, full-pressure integral systems test to support the issuance of a final design approval.

AP1000 Response:

The Westinghouse testing program to assess the analytical methodologies used for the AP1000 safety analysis is described in Section 1.5 and is in conformance with the NRC position.

1.9.5.2.12 Inspections, Test, Analyses, and Acceptance Criteria (ITAAC)

NRC Position:

10 CFR 52 requires that the design certification application include the proposed tests, inspections, analyses, and the associated acceptance criteria. For certified standard designs, these tests, inspections, and analyses must apply to those portions of the facility covered by the design certification.

The Staff Requirements Memorandum for SECY-91-178 (Reference 51) provided guidance regarding development of ITAAC for final design approval and design certification applications.

AP1000 Response:

The AP1000 design certification application includes ITAACs.

1.9.5.2.13 Reliability Assurance Program

NRC Position:

SECY-89-013 (Reference 52) requires a reliability assurance program for design certification. The program would ensure that the design reliability of safety significant systems, structures, and components is maintained over the life of a plant.

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The NRC is working on the development of a detailed guidance document consisting of two levels. The vendor submittal is the first level, consisting of a top-level program that identifies the scope, conceptual framework, and essential elements of an effective program. The Combined License applicant fully develops and implements the program based on the plant-specific design information.

AP1000 Response:

Section 16.2 includes a description of the reliability assurance program. The program description identifies the scope, conceptual framework, and essential elements of the program. The reliability assurance program confirms that the performance of the safety-related systems, structures, and components is consistent with the assumptions made for the design basis analysis.

In addition, the reliability assurance program monitors the long-term performance of important nonsafety-related structures, systems, and components that provide defense-in-depth against unnecessary actuation of the passive safety-related systems.

1.9.5.2.14 Site-Specific Probabilistic Risk Assessments (PRAs)

NRC Position:

10 CFR 52.47 requires all applicants for standard design certification to provide a PRA with enveloping analyses for seismic events and tornadoes. The Combined License applicant is responsible for the site-specific PRA information that addresses site-specific events such as river flooding, storm surge, tsunami, volcanism, and hurricanes.

AP1000 Response:

The AP1000 PRA submitted as a part of the design certification application is based on a site that bounds a large percentage of plant sites in the United States and is described in Chapter 2. APP-GW-GLR-101 (Reference 73) identifies the potential external events that may impact the AP1000 risk on a site-specific basis. This technical report considers a wide range of site-specific external events as long as a site can show that the external events listed in this report bound those applicable to the site. The report also discusses impact of site selection on PRA Offsite Dose Evaluation requirements.

1.9.5.2.15 Severe Accident Mitigation Design Alternatives

NRC Position:

The National Environmental Policy Act (NEPA) requires that alternatives be investigated for actions that may significantly affect the quality of the human environment. The timing of the NEPA hearing is at the Early Site Permit or Combined License stage. One objective of the 10 CFR 52 design certification rulemaking is to preclude changes to a certified standard plant design. The U.S. Court of Appeals has required the NRC to include consideration of severe accident mitigation design alternatives (SAMDAs) as a part of their environmental impact review for operating license applications. If this same process is followed for a plant design that had been certified, it may be necessary to reopen issues that had been resolved in the design certification rulemaking. To avoid this situation, the NRC issued SECY-91-229 (Reference 53) which recommended that SAMDAs be specifically addressed during the design certification rulemaking.

AP1000 Response:

The severe accident mitigation design alternatives (SAMDA) evaluation for AP1000 is contained in Appendix 1B.

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1.9.5.2.16 Generic Rulemaking Related to Design Certification

NRC Position:

SECY-91-262 (Reference 54) provides the NRC recommendations to proceed with design-specific rulemaking where appropriate for passive designs, as information becomes available from ongoing efforts on those issues, independent of the design review and certification processes. In SECY-93-087 the NRC staff concludes that the design of passive plants is not sufficiently developed to determine whether generic rulemaking should be initiated for passive plant designs.

Generic rulemaking activities for source terms during severe accidents are ongoing, and the results may be used during design certification of the passive plants, focusing on updating 10 CFR 100 siting criteria, and planning to incorporate the revised source criteria in 10 CFR 50.

AP1000 Response:

No response necessary. See Subsection 1.9.5.1.2 for a discussion of the use of a physically based source term.

1.9.5.3 Passive Design Issues

Issues related to the passive design were outlined in Reference 30.

1.9.5.3.1 Regulatory Treatment of Non-Safety Systems

NRC Position:

The NRC believes that its review of passive designs requires not only a review of the passive safety-related systems, but also a review of the functional capability and availability of the active nonsafety-related systems to provide significant defense-in-depth and accident and core damage prevention capability. The NRC issued a commission policy paper SECY-94-084 (Reference 67), on the regulatory treatment of non-safety systems (RTNSS), that outlines the process for resolving the RTNSS issue. This process includes a combination of probabilistic and deterministic criteria to identify risk-significant nonsafety-related systems.

AP1000 Response:

The AP1000 nonsafety-related active systems are designed to provide reliable support for normal plant operations and to provide defense-in-depth to minimize unnecessary challenges to the safety-related passive systems. These active systems are designed for more probable component and system failures. The systems include reliable, proven equipment and component designs. These active systems are capable of being powered by the nonsafety-related diesel-generators. The systems have nonsafety-related automatic actuation and controls that are separate from those of the safety-related systems.

These systems are designed to provide highly reliable performance. The design standards and operability provisions for these systems are discussed in Subsection 3.2.2.6. Availability controls were developed for nonsafety related structures, systems, and components found to be important via the RTNSS process. The availability controls for the AP1000 are documented in Section 16.3 and are the same as those for the AP600.

1.9.5.3.2 Definition of Passive Failure

NRC Position:

The NRC considered redefining failure of check valves in passive safety systems, where the valve fails to provide the mechanical movement to complete its intended safety function, to that of an active

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failure, as defined in Appendix A to 10 CFR 50. The NRC was concerned, since safety-related check valves in passive designs operate under different conditions (low flow and pressure without pump pressure to open valves) than current generation reactors and evolutionary designs. The check valves have increased safety significance to the operation of the passive safety-related systems, and operating experience has shown that they have a lower reliability than originally anticipated. The Staff position is described in SECY-94-084 (Reference 67).

AP1000 Response:

AP1000 is designed to tolerate the single failure of a check valve to change position to perform a safety-related function. Valve redundancy is provided for the core makeup tank discharge check valves (to close), the in-containment refueling water storage tank gravity injection check valves (to open), the containment recirculation gravity injection check valves (to open), and containment isolation line check valves (to close). The redundancy in the design for each of these safety-related flow paths is sufficient to accommodate the single failure of a check valve to reposition as required to perform its safeguards function.

Section 6.3 provides additional information on the failures assumed for the passive core cooling system including exceptions to the single failure criteria.

1.9.5.3.3 SBWR Stability

This issue is applicable to BWRs only.

1.9.5.3.4 Safe Shutdown Requirements

NRC Position:

GDC 34 requires that a residual heat removal system be provided to remove residual heat from the reactor core so that specified, acceptable fuel design limits are not exceeded. Regulatory Guide 1.139 and Branch Technical Position 5-1 implement this requirement and set forth conditions to cold shutdown (200°F for a PWR) using only safety-related systems within 36 hours.

The NRC evaluated the alternate means of addressing GDC 34 using passive safety-related systems to achieve a safe shutdown condition of 420°F. Additionally, the NRC reviewed the acceptability of using active, nonsafety-related systems to take a plant to cold shutdown conditions. The results of this review are presented in SECY-94-084 (Reference 67).

AP1000 Response:

The AP1000 includes safety-related passive systems and equipment that are designed to automatically establish and indefinitely maintain safe shutdown conditions for the plant following design basis events.

Sections 6.3 and 7.4 provide additional information pertaining to safe shutdown, using the safety-related passive systems.

1.9.5.3.5 Control Room Habitability

NRC Position:

10 CFR 50, Appendix A, GDC 19 requires adequate radiation protection to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of five rem whole body, or its equivalent, to any part of the body, for the duration of the accident. Section 6.4 of the Standard Review Plan defines this dose criterion in terms of specific whole-body and organ doses (5 rem to whole body, and 30 rem each to thyroid and skin).

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The NRC requires that the analyses of main control room habitability be based on the dose criterion defined in GDC 19 of Appendix A to 10 CFR 50 and Section 6.4 of the Standard Review Plan (5 rem to whole body, and 30 rem each to thyroid and skin). In addition, the analyses of control room habitability should be based on the duration of the accident according to GDC 19 of Appendix A to 10 CFR 50.

AP1000 Response:

The AP1000 design includes a passive, safety-related main control room habitability system to meet the requirements of GDC 19. Section 6.4 provides additional information.

As described in Subsection 15.6.5.3, the main control room operator doses following a design basis loss of coolant accident are within the dose criterion of GDC 19 (5 rem TEDE as applied to the AP1000 design).

1.9.5.3.6 Radionuclide Attenuation

NRC Position:

The NRC is concerned that use of the auxiliary building for holdup may require additional restrictions to be placed on the auxiliary building during normal operation. In addition, the NRC is continuing its evaluation of the need for a containment spray system for passive plant designs.

AP1000 Response:

The AP1000 design does not have a safety-related containment spray or take credit for auxiliary building holdup for mitigation of the design basis loss of coolant accident. The design includes a low-leakage-rate containment (0.10 percent per day) together with credit for aerosol removal by naturally occurring processes and pool scrubbing in containment. The low-leakage containment and natural aerosol removal are adequate to meet 10 CFR 50.34 dose limits, consistent with the physically-based source term.

1.9.5.3.7 Simplification of Off-Site Emergency Planning

NRC Position:

The NRC states that changes to emergency planning regulatory requirements may be appropriate, but that an NRC determination on this issue will require detailed design evaluation. Summaries of specific NRC conclusions are as follows:

- Unique characteristics of the designs should be considered in determining the extent of emergency planning, including the ability to prevent significant release of radioactive material or to provide delay times for all but the most unlikely events.

- A very low likelihood of all containment bypass sequences will be required before relaxing emergency planning requirements.

- Lack of information on source term and risk precludes further NRC evaluation of emergency preparedness for the passive plants at this time.

- Emergency planning requirements following the TMI-2 accident were not premised on specific assumptions regarding severe accident probability. So, as a policy matter, even very low calculated probabilities may not be a sufficient basis for changes to emergency planning requirements.

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The industry and the NRC are working to determine a process, including developing technical criteria and methods, that would justify simplification of offsite emergency planning. The results of this process would be used as input to a generic rulemaking proposal to be initiated by nuclear industry organizations.

AP1000 Response:

The AP1000 PRA evaluation risk assessment includes calculations of the AP1000 response to severe accidents. This response includes the release of radionuclides following a severe accident. This analysis supports the technical basis for simplification of offsite emergency planning. The offsite emergency planning is discussed in Section 13.3.

1.9.5.4 Additional Licensing Issue

Post-72 Hour Support Actions

The AP1000 includes safety-related passive systems and equipment that are sufficient to automatically establish and maintain safe shutdown conditions for the plant following design basis events, assuming that the most limiting single failure occurs. The safety-related passive systems maintain safe shutdown conditions after an event -- without operator action, without onsite and offsite ac power sources.

The AP1000 includes nonsafety-related active systems and equipment designed to provide multiple levels of defense for a wide range of events. For the more probable events, these nonsafety-related systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related passive systems. These nonsafety-related systems establish and maintain safe shutdown conditions for the plant following design basis events, provided that at least one of the standby nonsafety-related ac power sources is available.

Although event scenarios that result in an extended loss of the nonsafety-related systems or both offsite and onsite ac power sources for more than 72 hours are very unlikely, this potential is considered in the AP1000 design.

The actions described below are required following an extended loss of these nonsafety-related systems.

The safety functions required include the following:

- Core cooling, inventory, and reactivity control
- Containment cooling and ultimate heat sink
- Main control room habitability and post-accident monitoring
- Spent fuel pool cooling

The AP1000 design includes the capability to:

- Provide electrical power to supply the post-accident and spent fuel pool monitoring instrumentation using the ancillary diesel generators to feed the safety-related voltage regulating transformer. A connection is also available for use with a portable generator. See Section 8.3 for additional information.

- Provide makeup water to the passive containment cooling water storage tank to maintain external containment cooling water flow using one of the two PCS recirculation pumps powered by an ancillary diesel generator. A safety-related connection is also available for use with a portable pump. See Subsection 6.2.2 for additional information.

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- Ventilation and cooling of the main control room, the instrumentation and control rooms, and the dc equipment rooms is provided by open doors and ancillary fans or portable fans which receive power via the safety-related voltage regulating transformer. See Subsections 6.4.2.2 and 9.4.1 for additional information.

- Provide makeup water to the spent fuel pool from the passive containment cooling water storage tank, passive containment cooling ancillary water storage tank, and from the long term makeup connection. See Subsection 6.2.2.2.4 for a discussion of the operation of the passive containment cooling system and Subsection 9.1.3.4.3 and 9.1.3.5 for discussion of makeup to the spent fuel pool.

- Provide a vent path between the fuel handling area and outside environment to vent water vapor generated by elevated spent fuel pool water temperature. See Subsection 9.1.3.4.3.4 for additional information.

These actions are accomplished by the site support personnel, in coordination with the main control room operators. These actions are performed separate from, but in parallel with, other actions taken by the plant operators to directly mitigate the consequences of an event.

1.9.5.5 Operational Experience

Operational experience highlighted in NRC bulletins, generic letters, and information notices has been incorporated into the AP1000 design. Generic letters and bulletins are identified in WCAP-15800 (Reference 65). The applicability of each generic letter and bulletin to the AP1000 is assessed in WCAP-15800. If required, additional information for applicable issues is provided in the referenced sections of the DCD.

Table 1.9-204 lists the Bulletins and Generic Letters addressed by topical discussion in this FSAR. Table 1.9-204 also lists Bulletins and Generic Letters categorized as part of the first un-numbered COL Information Item identified at the end of Table 1.8-2. Table 1.9-204 provides the appropriate cross-references for the discussion of the topics addressed by those Bulletins and Generic Letters. Bulletins or Generic Letters issued after those listed in the DCD are also included in Table 1.9-204. Issues identified as “procurement” or “maintenance” or “surveillance” in WCAP-15800 are addressed as part of the scope of the certified design and are not specifically identified in Table 1.9-204. Issues identified as “procedural” in WCAP-15800 are addressed by the procedures discussed in Section 13.5 and are not specifically identified in Table 1.9-204. Other items in WCAP-15800, including the Circulars and Information Notices, are considered to have been adequately addressed based on the guidance identified in Regulatory Guide 1.206 and the NRC Standard Review Plans.

1.9.6 References

1. NUREG-0696, "Functional Criteria for Emergency Response Facilities," 1981.

2. Report NP-2770-LD, "EPRI PWR Safety Valve Test Report," December 1982.

3. NUREG-0933, "A Prioritization of Generic Safety Issues," June 2001.

4. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

5. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1984.

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6. NRC letter to all PWR Licensees of Operating Reactors, Applicants for Operating Licensees and Holders of Construction Permits, and Ft. St. Vrain, "Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity," (Generic Letter 85-02) April 17, 1985.

7. NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, A-5 Regarding Steam Generator Tube Integrity," U.S. Nuclear Regulatory Commission, September 1988.

8. NUREG-0577, Revision 1, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," U.S. Nuclear Regulatory Commission, October 1983.

9. IEEE 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

10. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1980.

11. NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1979.

12. NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plant Stations," U.S. Nuclear Regulatory Commission, February 1981.

13. IEEE 344-1987, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

14. NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants."

15. NUREG-0660, "NRC Action Plan Developed as a result of the TMI-2 accident," May 1980.

16. NUREG-0985, "Nuclear Regulatory Commission Human Factors Program Plan Revision 2," April 1986.

17. IEEE 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," Institute of Electrical and Electronics Engineers.

18. NUREG-0471, "Generic Task Problem Descriptions (Category B, C, and D Tasks)" and NUREG-0933, "A Prioritization of Generic Safety Issues," June 1978.

19. NUREG-0484, Revision 1, "Methodology for Combining Dynamic Responses," May 1980.

20. IEEE 317-1983, "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

21. ANSI/ANS-58.8-1984. "Time Response Design Criteria for Nuclear Safety Related Operator Actions."

22. NUREG/CR-2425, "Sediment and Radionuclide Transport in Rivers," December 1982.

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23. NUREG-0693, "Analysis of Ultimate-Heat-Sink Cooling Ponds," November 1980.

24. NUREG-0733, "Analysis of Ultimate Heat-Sink Spray Ponds," August 1981.

25. NUREG-0858, "Comparison Between Field Data and Ultimate Heat Sink Cooling-Pond and Spray-Pond Models," September 1980.

26. ANSI 5.1, "Decay Heat Power in Light Water Reactors," American National Standards Institute, 1979.

27. NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," September 1980.

28. ANSI 56.5-1979, "PWR and BWR Containment Spray System Design Criteria."

29. USNRC, SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues And Their Relationship to Current Regulatory Requirements," January 12, 1990.

30. NRC letter, Subject: Identification of Issues Concerning the Evolutionary and Passive Plant Designs, Dennis M. Crutchfield, USNRC Director, Division of Advanced Reactors and Special Projects, to E. E. Kintner, Chairman ALWR Steering Committee, February 27, 1992.

31. Staff Requirements Memorandum, Subject: SECY-90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements, Samuel J. Chilk, USNRC Secretary, to James M. Taylor, USNRC Executive Director for Operations, June 26, 1990.

32. NUREG-1150, "Severe Accident Risk: An Assessment for Five U.S. Nuclear Power Plants," June 1989.

33. "Passive ALWR Source Term," D. E. Leaver, et al., DOE/ID-10321, February 1991.

34. NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," Revision 1, September 1984.

35. NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44," June 1988.

36. NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," June 1988.

37. NUREG-0800, U.S. Nuclear Regulatory Commission Standard Review Plan, Section 9.5.1, “Fire Protection Program,” Revision 3, July 1981, including Branch Technical Position (BTP) CMEB 9.5-1, “Guidelines for Fire Protection for Nuclear Power Plants,” Revision 2, July 1981.

38. Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," April 3, 1989.

39. Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," June 28, 1989.

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40. SECY-91-273, "Review of Vendors' Test Programs to Support the Design Certification of Passive Light Water Reactors," August 27, 1991.

41. SECY-91-078, "Chapter 11 of EPRI's Requirements Document and Additional Evolutionary Light Water Reactor Certification Issues," March 25, 1991.

42. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

43. WASH-1300, “Technical Basis for Interim Regional Tornado Criteria,” May 1974.

44. SECY-91-348, preliminary untitled SECY related to containment leakrate testing, issued to the Commission for review, and not yet released by the NRC.

45. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

46. NUREG-4330, "Review of Light Water Reactor Regulatory Requirements, Volume 1, Identification of Regulatory Requirements That May Have Marginal Importance to Risk," April 1986.

47. SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," November 8, 1990.

48. SECY-91-057, "Early Review of AP600 and SBWR Research Needs," March 1, 1991.

49. SECY-91-074, "Prototype Decisions for Advanced Reactor Designs," March 19, 1991.

50. SECY-92-030, "Integral System Testing Requirements for Westinghouse's AP600 Plant," January 27, 1992.

51. SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," June 12, 1991.

52. SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)," January 19, 1989.

53. SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," July 31, 1991.

54. SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," August 16, 1991.

55. Not used.

56. WCAP-15993, “Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria,” Revision 2, June 2017.

57. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," June 30, 1995.

58. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," L. Soffer, et al., February 1995.

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59. SECY-94-302, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs," December 19, 1994.

60. SECY-94-194, "Proposed Revisions to 10 CFR Part 100 and 10 CFR Part 50, and New Appendix S to 10 CFR Part 50," July 27, 1994.

61. Not used.

62. Theofanous, T. G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

63. WCAP-15799, "AP1000 Compliance with SRP Acceptance Criteria," Revision 1, August 2003.

64. NCRP Report No. 116, Limitation of Exposure to Ionizing Radiation, March 31, 1993.

65. WCAP-15800, "Operational Assessment for AP1000," Revision 3, July 2004.

66. SECY-98-161, "The Westinghouse AP1000 Standard Design as it Relates to the Fire Protection and the Spent Fuel Pool Cooling Systems," July 1, 1998.

67. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994.

68. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP1000 Standardized Passive Reactor Design," June 30, 1995.

69. WCAP-15992, "AP1000 Adverse Systems Interactions Evaluation Report," Revision 1, February 2003.

70. WCAP-15776, “Safety Criteria for the AP1000 Instrumentation and Control Systems,” Revision 1, July 2019.

71. APP-GW-GLR-079, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," Westinghouse Electric Company LLC.

72. APP-GW-GLR-040, "Plant Operations, Surveillance, and Maintenance Procedures," Westinghouse Electric Company LLC.

73. APP-GW-GLR-101, "AP1000 Probabilistic Risk Assessment Site-Specific Considerations," Westinghouse Electric Company LLC.

201. NUMARC 87-00, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, Revision 1, August 1991.

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Table 1.9-1 (Sheet 1 of 17)
Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Rev. 0, November 2, 1970)	This regulatory guide is not applicable to AP1000.
1.2	Withdrawn	
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-coolant Accident for Boiling Water Reactors (Rev. 2, June 1974)	This regulatory guide is not applicable to AP1000.
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors (Rev. 2, June 1974)	The guidance of Reg. Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” will be followed instead of Reg. Guide 1.4.
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Rev. 0, March 10, 1971)	This regulatory guide is not applicable to AP1000.
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Rev. 0, March 10, 1971)	8.1 8.3.1 8.3.2 16.1 Bases 16 (TS Bases 3.8.1)
1.7	Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident (Rev. 2, November 1978)	6.1.1 6.2.4 15.6.3 Appendix 15A
1.7	Control of Combustible Gas Concentrations in Containment (Rev. 3, March 2007)	DCD discussion only
1.8	Qualification and Training of Personnel for Nuclear Power Plants (Rev. 3, 1 May 2000)	This regulatory guide is not applicable to AP1000 design certification. 12.1 (NEI 07-08A) 12.1.1.3 Appendix 12AA Appendix 12AA (NEI 07-03A) 13.2 (NEI 06-13A) 16 (TS 5.3.1) 17.5 (QAPD, IV)
1.9	Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (Proposed Rev. 3, November 1988)	This regulatory guide is not applicable to AP1000.
1.10	Withdrawn	
1.11	Instrument Lines Penetrating Primary Reactor Containment (Rev. 0, March 10, 1971)	3.6.2 6.2.3
1.11	Instrument Lines Penetrating Primary Reactor Containment (Rev. 1, March 2010)	DCD discussion only
1.12	Instrumentation for Earthquakes (Rev. 2, March 1997)	3.7.4 3.7.4.1
1.13	Spent Fuel Storage Facility Design Basis (Proposed Rev. 2, December 1981)	9.1.2 9.1.3 9.1.4 16.1 Bases

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Division 1 Regulatory Guide		Chapter, Section or Subsection
1.13	Spent Fuel Storage Facility Design Basis (Rev. 2, March 2007)	16 (TS Bases 3.7.11) 16 (TS Bases 3.7.12)
1.14	Reactor Coolant Pump Flywheel Integrity (Rev. 1, August 1975)	5.4.1
1.15	Withdrawn	
1.16	Reporting of Operating Information - Appendix A Technical Specifications (Rev. 4, August 1975).	This regulatory guide is not applicable to AP1000 design certification.
1.17	Withdrawn	
1.18	Withdrawn	
1.19	Withdrawn	
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 2, May 1976)	3.9.2 14
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 3, March 2007)	DCD discussion only
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants (Rev. 1, June 1974)	11.5 11.5.1.2 11.5.4.1 12.3.4
1.22	Periodic Testing of Protection System Actuation Functions (Rev. 0, February 17, 1972)	7.1 7.2 7.4
1.23	Onsite Meteorological Program (Second Proposed Rev. 1, April 1986)	2.3
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Rev. 0, March 23, 1972)	This regulatory guide is not applicable to AP1000.
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Rev. 0, March 23, 1972)	The guidance of Reg. Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” will be followed instead of Reg. Guide 1.25.
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 3, February 1976)	3.2.2
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 4, March 2007)	5.2.4.1 17.5 (QAPD IV)
1.27	Ultimate Heat Sink for Nuclear Power Plants (Rev. 2, January 1976)	6.2.2
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985)	2.5 17 14.2.2.2 17.5 (QAPD, II, 17.1) 17.5 (QAPD, IV)
1.29	Seismic Design Classification (Rev. 3, September 1978)	3.2.1
1.29	Seismic Design Classification (Rev. 4, March 2007)	17.5 (QAPD IV)

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Division 1 Regulatory Guide		Chapter, Section or Subsection
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Rev. 0, August 11, 1972)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978)	4.5.1 4.5.2 5.2.3 5.3.2 6.1.1 6.1.1.2
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (Rev. 2, February 1977)	8.1 8.2 8.3.1 8.3.2 16.1 Bases
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants and Rev. 3, March 2004	16 (TS Bases 3.8.1)
1.33	Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978)	3.11.2.1 3D.4.1.2 3D.6.4 16 (TS 5.4.1) 17.5 (QAPD, IV)
1.34	Control of Electroslag Weld Properties (Rev. 0, December 28, 1972)	4.5.2 5.2.3 5.3.2
1.35	Inservice Inspection of Ungrouted Tendons in Pre-stressed Concrete Containments (Rev. 3, July 1990)	This regulatory guide is not applicable to AP1000.
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments (Rev. 0, July 1990)	This regulatory guide is not applicable to AP1000.
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (Rev. 0, February 23, 1973)	5.2.3 6.1.1
1.37	Quality Assurance Requirements for cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 0, March 1973)	17
1.37	Quality Assurance Requirements for cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 1, March 2007)	17.5 (QAPD, II, 13.2) 17.5 (QAPD IV)
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, May 1977)	17
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, September 1977)	17
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (Rev. 0, March 16, 1973)	This regulatory guide is not applicable to AP1000.
1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments (Rev. 0, March 16, 1973)	14
1.42	Withdrawn	

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1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	5.2.3 5.3.2
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	4.5.1 4.5.2 5.2.3 5.3.2 6.1.1 10.3 6.1.1.2
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 0, May 1973)	5.2.5 16.1 Bases 16 (TS Bases 3.4.7) 16 (TS Bases 3.4.9)
1.46	Withdrawn	
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)	6.3 7.2 7.3 7.4 7.5 8.3.2
1.48	Withdrawn	
1.49	Withdrawn	
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	5.2.3 5.3.2 6.1.1
1.51	Withdrawn	
1.52	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 3, June 2001)	6.4 16 (TS 3.7.6) 16 (TS Bases B 3.7.6) 1.9.4.2.2
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 0, June 1973)	7.1 7.2 7.4 15.2 15.3 15.4 15.5 15.6
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 2, November 2003)	DCD discussion only
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000 and Rev. 2, October 2010)	6.1.2 1.9.4.2.3 6.1.2.1.6
1.55	Withdrawn	
1.56	Maintenance of Water Purity in Boiling Water Reactors (Rev. 1, July 1978)	This regulatory guide is not applicable to AP1000.
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 0, June 1973)	3.8.2 3.8.3
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 1, March 2007)	DCD discussion only

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1.58	Withdrawn	
1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, August 1977)	3.4 3.4.1.1.1
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	2.5 3.7.1
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 0, October 1973)	3.7.1 3.9.23.10 Appendix 3D
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 1, March 2007)	DCD discussion only
1.62	Manual Initiation of Protective Actions (Rev. 0, October 1973)	7.1 7.2
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants (Task EE 405-4) (Rev. 3, February 1987)	8.3.1 Appendix 3D
1.64	Withdrawn	
1.65	Materials and Inspections for Reactor Vessel Closure Studs (Rev. 0, October 1973)	5.3.2
1.66	Withdrawn	
1.67	Withdrawn	
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants (Rev. 2, August 1978)	14 16.1 Bases
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants (Rev. 3, March 2007)	14.2.1 14.2.3 14.2.8 14.2.5.2 16 (TS Bases 3.1.8)
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants (Rev. 1, January 1977)	This regulatory guide is not applicable to AP1000.
1.68.2	Initial Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (Rev. 1, July 1978)	14
1.68.3	Preoperational Testing of Instrument and Air Control Systems (Task RS 709-4) (Rev. 0, April 1982)	9.3.1 14
1.69	Concrete Radiation Shields for Nuclear Power Plants (Rev. 0, December 1973)	3.8.4 12.3
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Rev. 3, November 1978)	1.1 1.1.6.1
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 0, December 1973)	5.2.3.4.6
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 1, March 2007)	DCD discussion only
1.72	Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin (Rev. 2, November 1978)	This regulatory guide is not applicable to AP1000.
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)	3.11 Appendix 3D
1.74	Withdrawn	

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1.75	Physical Independence of Electric Systems (Rev. 2, September 1978)	7.1 7.2 7.3 7.4 7.5 8.1 8.3.1 8.3.2 9.5.1
1.75	Criteria for Independence of Electrical Safety Systems (Rev 3, February 2005)	DCD discussion only
1.76	Design Basis Tornado for Nuclear Power Plants (Rev. 0, April 1974)	2.3 3.3
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Rev. 1, March 2007)	Not referenced; see Appendix 1A
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	The interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, is followed for the analysis of the RCCA ejection accident instead of Regulatory Guide 1.77. The guidance of Regulatory Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” is also followed instead of Regulatory Guide 1.77. See Subsection 15.4.8.1.2.
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Rev. 1, December 2001)	2.2 6.4 9.4.1 9.5.1 16.1 Bases 6.4.3 16 (TS Bases 3.7.6)
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, September 1975)	14
1.80	Withdrawn	
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plant (Rev. 1, January 1975)	This regulatory guide is not applicable to AP1000.
1.82	Water Sources for Long Term Recirculation Cooling Following a Loss-of-Coolant Accident (Task 203-4) (Rev. 3, November 2003)	6.3
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)	5.4.2

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1.84	Design and Fabrication Code Case Acceptability ASME Section III Division 1 (Rev. 32, June 2003)	4.5.1 4.5.2 5.2.1 5.2.3 10.3
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III (Rev. 33, August 2005)	DCD discussion only
1.85	Withdrawn	
1.86	Termination of Operating Licenses for Nuclear Reactors (Rev. 0, June 1974)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Rev. 1, June 1975)	This regulatory guide is not applicable to AP1000.
1.88	Withdrawn	
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (Task EE 042-2) (Rev. 1, June 1984)	3.11 Appendix 3D 1.9.4 8.1.4 8.3.2 11.5.2 11.5.6
1.90	Inservice Inspection of Prestressed Concrete Containment Structures With Grouted Tendons (Rev. 1, August 1977)	This regulatory guide is not applicable to AP1000.
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites (Rev. 1, February 1978)	9.3.2.3
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 1, February 1976; Rev. 2, July 2006)	3.7 Appendix 3D
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 2, July 2006)	DCD discussion only
1.93	Availability of Electric Power Sources (Rev. 0, December 1974)	8.1 8.3 16.1 Bases 16 (TS Bases 3.8.1) 16 (TS Bases 3.8.5)
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1, April 1976)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.95	Withdrawn	
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants (Rev. 1, June 1976)	This regulatory guide is not applicable to AP1000.

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1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident (Rev. 3, May 1983)	7.5 18.8 16.1 Bases Appendix 3D Table 7.5-201 Appendix 12AA 16 (TS Bases 3.3.17)
1.97	Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants (Rev. 4, June 2006)	Not referenced; See Appendix 1A
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (Rev. 0, March 1976)	This regulatory guide is not applicable to AP1000.
1.99	Radiation Embrittlement of Reactor Vessel Materials (Task ME 305-4) (Rev. 2, May 1988)	5.3.2 5.3.3 16.1 Bases
1.99	Radiation Embrittlement of Reactor Vessel Materials (Rev. 2, May 1988)	16 (TS Bases 3.4.3)
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants (Task EE 108-5) (Rev. 2, June 1988)	3.10 Appendix 3D 1.9.4 3.11.2 8.1.4 8.3.2 11.5.2 11.5.6
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Rev. 3, August 1992)	This regulatory guide is not applicable to AP1000 design certification.
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Rev. 4, July 2003)	13.3 (EP) 9.5.1.8.2.2
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Rev. 5, June 2005)	Not referenced; see Appendix 1A
1.102	Flood Protection for Nuclear Power Plants (Rev. 1, September 1976)	3.4
1.103	Withdrawn	
1.104	Withdrawn	
1.105	Instrument Setpoints for Safety-Related Systems (Task 1C 010-5) (Rev. 3, December 1999)	7.1 16
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Rev. 1, March 1977)	8.1
1.107	Qualifications for Cement Grouting Tendons for Prestressing Tendons in Containment Structures (Rev. 1, February 1977)	This regulatory guide is not applicable to AP1000.
1.108	Withdrawn	
1.109	Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50 Appendix I (Rev. 1, October 1977)	11.3.3
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Rev. 0, March 1976)	11.2 11.3
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors (Rev. 1, July 1977)	2.3

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1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors (Rev. 0-R, May 1977)	11.2.3 11.3.3
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors (Rev. 1, March 2007)	DCD discussion only
1.113	Estimating Aquatic Dispersion of Effluents From Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Rev. 1, April 1977)	This regulatory guide is not applicable to AP1000 design certification.
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit (Rev. 2, May 1989)	This regulatory guide is not applicable to AP1000 design certification.
1.115	Protection Against Low-Trajectory Turbine Missiles (Rev 1, July 1977)	3.5 3.8.4 3.5.1.3
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. 0-R, May 1977)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.117	Tornado Design Classification (Rev. 1, April 1978)	3.5 9.1.2
1.118	Periodic Testing of Electric Power and Protection Systems (Rev. 3, April 1995)	7.1 8.1 8.3
1.119	Withdrawn	
1.120	Fire Protection Guidelines for Nuclear Power Plants (Rev. 1, November 1977)	9.5.1
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (Rev. 0, August 1976)	5.4.2 16.1 Bases 16 (TS Bases 3.4.17)
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, February 1978)	3.7 Appendix 3D
1.123	Withdrawn	
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (Rev. 1, January 1978)	3.9.3, 9.1.2.1, 9.1.1.1
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports (Rev. 2, February 2007)	DCD discussion only
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, October 1978)	2.4
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, March 1978)	4.2
1.127	Inspection of Water-Control Structures Associated With Nuclear Power Plants (Rev. 1, March 1978)	This regulatory guide is not applicable to AP1000.
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, October 1978)	8.3.2
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	DCD discussion only
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Table 8.1-1

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1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 1, October 1978)	3.9.3
1.130	Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Supports (Rev. 2, March 2007)	DCD discussion only
1.131	Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants (Rev. 0, August 1977)	3.11 Appendix 3D
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 1, March 1979)	This regulatory guide is not applicable to AP1000 design certification.
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 2, October 2003)	This regulatory guide is not applicable to AP1000 design certification.
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, May 1981)	4.4.6.4 Not referenced; see Appendix 1A
1.134	Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses (Rev. 3, March 1998)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.135	Normal Water Level and Discharge at Nuclear Power Plants (Rev. 0, September 1977)	2.4
1.136	Material for Concrete Containments (Rev. 2, June 1981)	This regulatory guide is not applicable to AP1000.
1.137	Fuel-Oil Systems for Standby Diesel Generators (Rev. 1, October 1979)	9.5.4
1.138	Laboratory Investigation of Soils for Engineering Analysis and Design of Nuclear Power Plants (Rev. 0, April 1978)	This regulatory guide is not applicable to AP1000 design certification.
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants (Rev. 2, December 2003)	This regulatory guide is not applicable to AP1000 design certification.
1.139	Guidance for Residual Heat Removal (Rev. 0, May 1978)	6.3 7.4
1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 2, June 2001)	9.4.1 9.4.4 9.4.5 9.4.7 9.4.9 9.4.1.4 9.4.7.4 16.3 (TRM Bases 3.9.2)
1.141	Containment Isolation Provisions for Fluid Systems (Rev. 0, April 1978)	6.2.4
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) (Rev. 2, November 2001)	3.8.3 3.8.4 3.8.5

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1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Rev. 2, November 2001)	3.8.4 10.4.8 11.2 11.3 11.4 11.5 11.2.1.2.5.2 11.2.3.6 11.3.3.6 11.4.5 11.4.6.2
1.144	Withdrawn	
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Rev. 1, November 1982)	This regulatory guide is not applicable to AP1000 design certification.
1.146	Withdrawn	
1.147	Inservice Inspection Code Case Acceptability ASME Section XI Division 1 (See PSI and ISI Programs)	5.2.4 6.6
1.148	Withdrawn	
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations (Rev. 2, April 1996)	This regulatory guide is not applicable to AP1000 design certification.
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements (Rev. 4, April 2011)	13.2 (NEI 06-13A)
1.150	Ultrasonic Testing of Reactor Pressure Vessel Welds During Preservice and Inservice Examinations (Rev. 1, February 1983)	5.2.4 5.3.2 5.3.4 DCD discussion only
1.151	Instrument Sensing Lines (Task 1C 126-5) (Rev. 0, July 1983)	7.1 7.5 7.6 7.7
1.152	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants (Rev. 1, January 1996)	7
1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants (Rev. 2, January 2006)	Not referenced; see Appendix 1A
1.153	Criteria for Power, Instrumentation, and Control Portions of Safety Systems (Task 1C 609-5) (Rev. 1, June 1996)	7.1 7.2 7.3 7.4 7.5 7.6
1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors (Rev. 0, January 1987)	5.3
1.155	Station Blackout (Task SI 501-4) (Rev. 0, August 1988)	8.2 8.3.1

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1.155	Station Blackout (Rev. 0, August 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants (Task EE 404-4) (Rev. 0, November 1987)	Appendix 3D
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance (Task RS 701-4) (Rev. 0, May 1989)	6.3
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants (Task EE 006-5) (Rev. 0, February 1989)	Appendix 3D 8.3.2
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors (Rev. 0, August 1990)	This regulatory guide is not applicable to AP1000 design certification.
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors (Rev. 1, October 2003)	Not referenced; see Appendix 1A
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Rev. 4, August 2018)	3.8.3.7 3.8.4.7 3.8.5.7
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb (Rev. 0, June 1995)	This regulatory guide is not applicable to AP1000 design certification.
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels (Rev. 0, February 1996)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.163	Performance Based Containment Leak-Test Program (Rev. 0, September 1995)	6.2 16.1 Bases 6.2.5.1 6.2.5.2.2 16 (TS 5.5.8)
1.165	Identification and Characterization of Seismic Sources and Determination Safe Shutdown Earthquake Ground Motion (Rev. 0, March 1997)	Not referenced; see Appendix 1A
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions (Rev. 0, March 1997)	This regulatory guide is not applicable to AP1000 design certification. 3.7.4.4
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event (Rev. 0, March 1997)	This regulatory guide is not applicable to AP1000 design certification. 3.7.4.4
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7 (Common Q discussions)
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 1, February 2004)	7 (Component Interface Module discussions)
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 1, February 2004)	DCD discussion only
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7

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Division 1 Regulatory Guide		Chapter, Section or Subsection
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.174	An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis (Rev. 0, July 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.174	An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis (Rev. 1, November 2002)	Not referenced; see Appendix 1A
1.175	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing (Rev. 0, July 1998)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.176	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance (Rev. 0, August 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.177	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications (Rev. 0, August 1998)	Not referenced; see Appendix 1A
1.178	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Piping (Rev. 0, September 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.178	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Piping (Rev. 1, September 2003)	Not referenced; see Appendix 1A
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors (Rev. 0, January 1999)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (Rev. 1, October 2003)	Appendix 3D 7.1.4.2
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e) (Rev. 0, September 1999)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.182	Withdrawn	

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Division 1 Regulatory Guide		Chapter, Section or Subsection
1.183	Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors (Rev. 0, July 2000)	2.3 Appendix 3D 4.2 6.5.1 15.4 15.6.3 15.7 16.1 Bases 16 (TS Bases 3.7.5) 16 (TS Bases 3.9.4) 16 (TS Bases 3.9.5)
1.184	Decommissioning of Nuclear Power Reactors (Rev. 0, August 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.184	Decommissioning of Nuclear Power Reactors (Rev. 0, July 2000)	Not referenced; see Appendix 1A
1.185	Standard Format and Content for Post-shutdown Decommissioning Activities Report (Rev. 0, August 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.185	Standard Format and Content for Post- shutdown Decommissioning Activities Report (Rev. 0, July 2000)	Not referenced; see Appendix 1A
1.186	Guidance and Examples of Identifying 10 CFR 50.2 Design Bases (Rev. 0, December 2000)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments (Rev. 0, November 2000)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.188	Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses (Rev. 1, September 2005)	Not referenced; see Appendix 1A
1.189	Fire Protection for Operating Nuclear Power Plants (Rev. 0, April 2001)	This regulatory guide is not applicable to AP1000 design certification.
1.189	Fire Protection for Nuclear Power Plants (Rev. 1, March 2007)	9.5.1.8.1.1 9.5.1.8.2.2 Table 9.5.1-1 17.5 (QAPD III.2)
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Rev. 0, March 2001)	5.3.2.6 5.3.3.1
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown (Rev. 0, May 2001)	Not referenced; see Appendix 1A
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code (Rev. 3, October 2019)	Not referenced; See Appendix 1A
1.193	ASME Code Cases Not Approved for Use (Rev. 1, August 2005)	Not referenced; see Appendix 1A
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (Rev. 0, June 2003)	15A.3.3

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Division 1 Regulatory Guide		Chapter, Section or Subsection
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (Rev. 0, May 2003)	Not referenced; see Appendix 1A
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors (Rev. 1, January 2007)	6.4.3 16 (TS Bases 3.7.6)
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	9.4.1 6.4.5
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites (Rev. 0, November 2003)	Not referenced; see Appendix 1A
1.199	Anchoring Components and Structural Supports in Concrete (Rev. 0, November 2003)	3.8.3.5 3.8.4.5.1 3.8.5.5
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Rev. 2, March 2009)	19
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance (Rev. 1, May 2006)	Not referenced; see Appendix 1A
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors (Rev. 0, February 2005)	Not referenced; see Appendix 1A
1.203	Transient and Accident Analysis Methods (Rev. 0, December 2005)	Not referenced; see Appendix 1A
1.204	Guidelines for Lightning Protection of Nuclear Power Plants (Rev. 0, November 2005)	Not referenced; see Appendix 1A
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants (Rev. 0, May 2006)	Not referenced; see Appendix 1A
1.206	Applications for Nuclear Power Plants (Rev. 1, October 2018)	1.1.6.1 1.9.5.5 See Appendix 1A 12.1.1.3 12AA.5 14.2.1 Appendix 14A
1.207	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors (Rev. 0, March 2007)	Not referenced; see Appendix 1A
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion (Rev. 0, March 2007)	Not referenced; see Appendix 1A
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants (Rev. 0, March 2007)	Not referenced; see Appendix 1A
Division 4 Regulatory Guides		
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) — Effluent Streams and the Environment (Rev. 1, February 1979)	11.5.1.2 11.5.3 11.5.4 11.5.6.5
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) — Effluent Streams and the Environment (Rev. 2, July 2007)	11.5.3

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Division 1 Regulatory Guide		Chapter, Section or Subsection
Division 5 Regulatory Guides		Note c
5.71	Cyber Security Programs for Nuclear Facilities (Rev. 0, January 2010)	Not referenced; see Appendix 1A
5.84	Fitness-For-Duty Programs at New Reactor Construction Sites (Rev. 0, July 2015)	13.7
Division 8 Regulatory Guides		
8.2	Guide for Administrative Practices in Radiation Monitoring (Rev. 0, February 1973)	12.1 (NEI 07-08A) 12.3.4 Appendix 12AA (NEI 07-03A)
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters (Rev. 0, February 1973)	Appendix 12AA (NEI 07-03A)
8.5	Withdrawn	
8.6	Standard Test Procedure for Geiger-Muller Counters (Rev. 0, May 1973)	Appendix 12AA (NEI 07-03A)
8.7	Instructions for Recording and Reporting Occupational Radiation Dose Data (Rev. 2, November 2005)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.8	Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable (Rev. 3, June 1978)	12.1 (NEI 07-08A) 12.3.4 Appendix 12AA Appendix 12AA (NEI 07-03A)
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (Rev. 1, July 1993)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable (Rev. 1-R, May 1977)	12.1 (NEI 07-08A) 12.3.4 Appendix 12AA Appendix 12AA (NEI 07-03A)
8.13	Instruction Concerning Prenatal Radiation Exposure (Rev. 3, June 1999)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.15	Acceptable Programs for Respiratory Protection (Rev. 1, October 1999)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants (Rev. 0, March 1981)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.28	Audible-Alarm Dosimeters (Rev. 0, August 1981)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.29	Instruction Concerning Risks from Occupational Radiation Exposure (Rev. 1, February 1996)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.34	Monitoring Criteria and Methods To Calculate Occupational Radiation Doses (Rev. 0, July 1992)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.35	Planned Special Exposures (Rev. 0, June 1992)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.36	Radiation Dose to the Embryo/Fetus (Rev. 0, July 1992)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)

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Division 1 Regulatory Guide		Chapter, Section or Subsection
8.38	Control of Access to High and Very High Radiation Areas of Nuclear Plants (Rev. 1, May 2006)	12.1 (NEI 07-08A) Appendix 12AA Table 12AA-201 Appendix 12AA (NEI 07-03A)

a. Not used.
b. Not used.
c. Division 5 of the regulatory guides applies to materials and plant protection. As appropriate, the Division 5 regulatory guide topics are addressed in the DCD and plant-specific security plans (i.e., Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Cyber Security Plan)

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Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
TMI Action Plan Items			
I.A.1.1	Shift Technical Advisor	f	Resolved per NUREG-0933
I.A.1.2	Shift Supervisor Administrative Duties	f	Resolved per NUREG-0933
I.A.1.3	Shift Manning	f	Resolved per NUREG-0933
I.A.1.4	Long-Term Upgrading	f	See Subsection 13.1.1 Resolved per NUREG-0933
I.A.2.1(1)	Qualifications - Experience	f	Resolved per NUREG-0933
I.A.2.1(2)	Training	f	
I.A.2.1(2)	Immediate Upgrading of RO & SRO Training and Qualifications, Training	f	Resolved per NUREG-0933
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	f	Resolved per NUREG-0933
I.A.2.2	Training and Qualifications of Operations Personnel	c	
I.A.2.3	Administration of Training Programs	f	Resolved per NUREG-0933
I.A.2.4	NRR Participation in Inspector Training	d	Not applicable to new plants
I.A.2.5	Plant Drills	c	
I.A.2.6(1)	Revise Regulatory Guide 1.8	f	Resolved per NUREG-0933
I.A.2.6(2)	Staff Review of NRR 80-117	c	
I.A.2.6(3)	Revise 10 CFR 55	e	
I.A.2.6(4)	Operator Workshops	c	
I.A.2.6(5)	Develop Inspection Procedures for Training Programs	c	
I.A.2.6(6)	Nuclear Power Fundamentals	a	
I.A.2.7	Accreditation of Training Institutions	c	
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	f	Resolved per NUREG-0933
I.A.3.2	Operator Licensing Program Changes	c	
I.A.3.3	Requirements for Operator Fitness	c	
I.A.3.4	Licensing of Additional Operations Personnel	c	
I.A.3.5	Establish Statement of Understanding with INPO and DOE	d	Not applicable to new plants

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.A.4.1(1)	Short-Term Study of Training Simulators	c	
I.A.4.1(2)	Interim Changes in Training Simulators	f	Resolved per NUREG-0933
I.A.4.2(1)	Research on Training Simulators	f	Resolved per NUREG-0933
I.A.4.2(2)	Upgrade Training Simulator Standards	f	Resolved per NUREG-0933
I.A.4.2(3)	Regulatory Guide on Training Simulators	f	See Subsection 1.9.3, item (2)(i) Resolved per NUREG-0933
I.A.4.2(4)	Review Simulators for Conformance to Criteria	f	Resolved per NUREG-0933
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	d	Not applicable to new plants
I.A.4.4	Feasibility Study of NRC Engineering Computer	d	Not applicable to new plants
I.B.1.1(1)	Prepare Draft Criteria	c	
I.B.1.1(2)	Prepare Commission Paper	c	
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	c	
I.B.1.1(4)	Review Responses to Determine Acceptability	c	
I.B.1.1(5)	Review Implementation of the Upgrading Activities	c	
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	e	
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	e	
I.B.1.2(1)	Prepare Draft Criteria	c	
I.B.1.2(2)	Review Near-Term Operating License Facilities	c	
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	c	
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	d	Not applicable to new plants
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	d	Not applicable to new plants
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	d	Not applicable to new plants
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	d	Not applicable to new plants

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	d	Not applicable to new plants
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Ensure Proper Testing and Return to Service	d	Not applicable to new plants
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	d	Not applicable to new plants
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	d	Not applicable to new plants
I.B.2.1(6)	Observe Routine Maintenance	d	Not applicable to new plants
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	d	Not applicable to new plants
I.B.2.2	Resident Inspector at Operating Reactors	d	Not applicable to new plants
I.B.2.3	Regional Evaluations	d	Not applicable to new plants
I.B.2.4	Overview of Licensee Performance	d	Not applicable to new plants
I.C.1(1)	Small Break LOCAs	f	Resolved per NUREG-0933
I.C.1(2)	Inadequate Core Cooling	f	Resolved per NUREG-0933
I.C.1(3)	Transients and Accidents	f	Resolved per NUREG-0933
I.C.1(4)	Confirmatory Analyses of Selected Transients	c	
I.C.2	Shift and Relief Turnover Procedures	f	Resolved per NUREG-0933
I.C.3	Shift Supervisor Responsibilities	f	Resolved per NUREG-0933
I.C.4	Control Room Access	f	Resolved per NUREG-0933
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	g	See Subsection 1.9.3, item (3)(i)
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	f	Resolved per NUREG-0933
I.C.7	NSSS Vendor Review of Procedures	f	Resolved per NUREG-0933
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	f	Resolved per NUREG-0933

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.C.9	Long-Term Program Plan for Upgrading of Procedures	c	See Subsections 13.5.1 and 1.9.3, item (2)(ii)
I.D.1	Control Room Design Reviews	g	See Subsection 1.9.3, item (2)(iii)
I.D.2	Plant Safety Parameter Display Console	g	See Subsection 1.9.3, item (2)(iv)
I.D.3	Safety System Status Monitoring	c	See Subsection 1.9.3, item (2)(v)
I.D.4	Control Room Design Standard	c	
I.D.5(1)	Operator-Process Communication	c	
I.D.5(2)	Plant Status and Post-Accident Monitoring	g	See Subsection 1.9.4, item I.D.5(2)
I.D.5(3)	On-Line Reactor Surveillance System	c	See Subsection 1.9.4, item I.D.5(3)
I.D.5(4)	Process Monitoring Instrumentation	c	
I.D.5(5)	Disturbance Analysis Systems	d	Not applicable to new plants
I.D.6	Technology Transfer Conference	d	Not applicable to new plants
I.E.1	Office for Analysis and Evaluation of Operational Data	d	Not applicable to new plants
I.E.2	Program Office Operational Data Evaluation	d	Not applicable to new plants
I.E.3	Operational Safety Data Analysis	d	Not applicable to new plants
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	d	Not applicable to new plants
I.E.5	Nuclear Plant Reliability Data Systems	d	Not applicable to new plants
I.E.6	Reporting Requirements	d	Not applicable to new plants
I.E.7	Foreign Sources	d	Not applicable to new plants
I.E.8	Human Error Rate Analysis	d	Not applicable to new plants
I.F.1	Expand QA List	c, j	See Subsections 1.9.4.2.1, item I.F.1 and 1.9.3, item (3)(ii)
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	a	See Section 17.5
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	g	See Subsection 1.9.3, item (3)(iii)
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	g	See Subsection 1.9.3, item (3)(iii)

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Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	a	See Section 17.5
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	a	See Section 17.5
I.F.2(6)	Increase the Size of Licensees' QA Staff	f	Resolved per NUREG-0933
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	a	See Section 17.5
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	a	See Section 17.5
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	f	Resolved per NUREG-0933
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	a	See Section 17.5
I.F.2(11)	Define Role of QA in Design and Analysis Activities	a	See Section 17.5
I.G.1	Training Requirements	f, j	See Subsection 1.9.4.2.1, item I.G.1 Resolved per NUREG-0933
I.G.2	Scope of Test Program	f, j	See Subsection 1.9.4.2.1, item I.G.2 Resolved per NUREG-0933
II.A.1	Siting Policy Reformulation	c	
II.A.2	Site Evaluation of Existing Facilities	e	
II.B.1	Reactor Coolant System Vents	g	See Subsection 1.9.3, item (2)(vi)
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	g	See Subsection 1.9.3, item (2)(vii)
II.B.3	Post-Accident Sampling	g	See Subsection 1.9.3, item (2)(viii)
II.B.4	Training for Mitigating Core Damage	f	Resolved per NUREG-0933
II.B.5(1)	Behavior of Severely Damaged Fuel	d	Not applicable to new plants
II.B.5(2)	Behavior of Core Melt	d	Not applicable to new plants
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structures	d	Not applicable to new plants
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	f	Resolved per NUREG-0933
II.B.7	Analysis of Hydrogen Control	e	
II.B.8	Rulemaking Proceedings on Degraded Core Accidents	g	See Subsection 1.9.3, items (1)(i), (1)(xii), (2)(ix), (3)(iv), and (3)(v)
II.C.1	Interim Reliability Evaluation Program	c	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.C.2	Continuation of Interim Reliability Evaluation Program	c	
II.C.3	Systems Interaction	e	
II.C.4	Reliability Engineering	c	
II.D.1	Testing Requirements	g	See Subsection 1.9.3, item (2)(x)
II.D.2	Research on Relief and Safety Valve Test Requirements	a	
II.D.3	Relief and Safety Valve Position Indication	g	See Subsection 1.9.3, item (2)(xi)
II.E.1.1	Auxiliary Feedwater System Evaluation	g	See Subsection 1.9.3, item (1)(ii)
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	g	See Subsection 1.9.3, items (1)(ii) and (2)(xii)
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	d, j	See Subsection 1.9.4.2.1, item II.E.1.3 Resolved per NUREG-0933
II.E.2.1	Reliance on ECCS	e	
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	c	
II.E.2.3	Uncertainties in Performance Predictions	a	
II.E.3.1	Reliability of Power Supplies for Natural Circulation	g	See Subsection 1.9.3, item (2)(xiii)
II.E.3.2	Systems Reliability	e	
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	e	
II.E.3.4	Alternate Concepts Research	c	
II.E.3.5	Regulatory Guide	e	
II.E.4.1	Dedicated Penetrations	g	See Subsection 1.9.3, item (3)(vi)
II.E.4.2	Isolation Dependability	g	See Subsection 1.9.3, item (2)(xiv)
II.E.4.3	Integrity Check	c	
II.E.4.4	Purging	g	See Subsection 1.9.3, item (2)(xv)
II.E.5.1	Design Evaluation	b	
II.E.5.2	B&W Reactor Transient Response Task Force	b	
II.E.6.1	Test Adequacy Study	d, j	See Subsection 1.9.4.2.1, item II.E.6.1 Resolved per NUREG-0933
II.F.1	Additional Accident Monitoring Instrumentation	g	See Subsection 1.9.3, item (2)(xvii)
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	g	See Subsection 1.9.3, item (2)(xviii)

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.F.3	Instruments for Monitoring Accident Conditions	g	See Subsection 1.9.3, item (2)(xix)
II.F.4	Study of Control and Protective Action Design Requirements	a	
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	d	Not applicable to new plants
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	g	See Subsection 1.9.3, item (2)(xx)
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	c	
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	b	
II.H.3	Evaluate and Feed Back Information Obtained from TMI	e	
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	d	Not applicable to new plants
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	d	Not applicable to new plants
II.J.1.2	Modify Existing Vendor Inspection Program	d	Not applicable to new plants
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	d	Not applicable to new plants
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	d	Not applicable to new plants
II.J.2.1	Reorient Construction Inspection Program	d	Not applicable to new plants
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	d	Not applicable to new plants
II.J.2.3	Assign Resident Inspectors to All Construction Sites	d	Not applicable to new plants
II.J.3.1	Organization and Staffing to Oversee Design and Construction	f	See Subsection 1.9.3, item (3)(vii) Not applicable to new plants
II.J.3.2	Issue Regulatory Guide	e	
II.J.4.1	Revise Deficiency Reporting Requirements	f	Resolved per NUREG-0933
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	f	Resolved per NUREG-0933
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	b	
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	f	Resolved per NUREG-0933
II.K.1(4)	Review Operating Procedures and Training Instructions	f	Resolved per NUREG-0933
II.K.1(5)	Safety-Related Valve Position Description	f	Resolved per NUREG-0933

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	f	Resolved per NUREG-0933
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	b	
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	b	
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	f	Resolved per NUREG-0933
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	f, j	See Subsection 1.9.4.2.1, item II.K.1(10) Resolved per NUREG-0933
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	f	Resolved per NUREG-0933
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	f	Resolved per NUREG-0933
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	f, j	See Subsection 1.9.4.2.1, item II.K.1(13) Resolved per NUREG-0933
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	f	Resolved per NUREG-0933
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	f	Resolved per NUREG-0933
II.K.1(16)	Implement Procedures That Identify PZR PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	f	Resolved per NUREG-0933
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	f, j	See Subsection 1.9.4.2.1, item II.K.1(17) Resolved per NUREG-0933
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	b	
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	b	
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	b	
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	b	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	g	See Subsection 1.9.3, item (2)(xxi)
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	b	
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	e, j	See Subsection 1.9.4.2.1, item II.K.1(24)
II.K.1(25)	Develop Operator Action Guidelines	e	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	f	Resolved per NUREG-0933
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	e	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	e	
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	b	
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	b	
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	b	
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	b	
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	b	
II.K.2(6)	Reevaluate Analysis of Dual-Level Setpoint Control	b	
II.K.2(7)	Reevaluate Transient of September 24, 1977	b	
II.K.2(8)	Continued Upgrading of AFW System	e	
II.K.2(9)	Analysis and Upgrading of Integrated Control System	e	
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	b	See Subsection 1.9.3, item (2)(xxiii)
II.K.2(11)	Operator Training and Drilling	b	
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	e	
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	b	
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	b	
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	b	
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	g	See Subsection 1.9.3, item (1)(iii)
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	b	
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	e	
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	b	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA	b	
II.K.2(21)	LOFT L3-1 Predictions	b	
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	g	See Subsection 1.9.3, item (1)(iv)
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	g	See Subsection 1.9.3, item (1)(iv)
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	f	Resolved per NUREG-0933
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	e	
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	f, j	See Subsection 1.9.4.2.1, item II.K.3(5) Resolved per NUREG-0933
II.K.3(6)	Instrumentation to Verify Natural Circulation	e	
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	b	
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	e	
II.K.3(9)	Proportional Integral Derivative Controller Modification	g	See Subsection 1.9.4, item II.K.3(9)
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	f	Resolved per NUREG-0933
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	f	Resolved per NUREG-0933
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	f	Resolved per NUREG-0933
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	b	
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	b	
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	b	
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	b	
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	b	
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	g	See Subsection 1.9.3, item (1)(vii)
II.K.3(19)	Interlock on Recirculation Pump Loops	b	
II.K.3(20)	Loss of Service Water for Big Rock Point	b	
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	b	
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	b	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.3(23)	Central Water Level Recording	e	
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	b	
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	g	See Subsection 1.9.3, item (1)(iii)
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	e	
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	b	
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	g	See Subsection 1.9.3, item (1)(x)
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	b	
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	f	Resolved per NUREG-0933
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	f	Resolved per NUREG-0933
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	e	
II.K.3(33)	Evaluate Elimination of PORV Function	e	
II.K.3(34)	Relap-4 Model Development	e	
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	e	
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	e	
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	e	
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	e	
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	e	
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	e	
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	e	
II.K.3(42)	Submit Requested Information on the Effects of Non Condensible Gases	e	
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	e	
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	b	
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	b	
II.K.3(46)	Response to List of Concerns from ACRS Consultant	b	
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	e	
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	e	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.3(49)	Review of Procedures (NRC)	e	
II.K.3(50)	Review of Procedures (NSSS Vendors)	e	
II.K.3(51)	Symptom-Based Emergency Procedures	e	
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	e	
II.K.3(53)	Two Operators in Control Room	e	
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	e	
II.K.3(55)	Operator Monitoring of Control Board	e	
II.K.3(56)	Simulator Training Requirements	e	
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	b	
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	f	Resolved per NUREG-0933
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	f	Not applicable to new plants
III.A.1.2	Upgrade Licensee Emergency Support Facilities	g	See Subsection 1.9.3, item (2)(xxv)
III.A.1.3(1)	Maintain Supplies of Thyroid-Blocking Agent - Workers	c	
III.A.1.3(2)	Maintain Supplies of Thyroid-Blocking Agent - Public	c	
III.A.2.1(1)	Publish Proposed Amendments to the Rules	d	Resolved per NUREG-0933
III.A.2.1(2)	Conduct Public Regional Meetings	d	Not applicable to new plants
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	d	Not applicable to new plants
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	d	Resolved per NUREG-0933
III.A.2.2	Development of Guidance and Criteria	d	Resolved per NUREG-0933
III.A.3.1(1)	Define NRC Role in Emergency Situations	c	
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	c	
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	c	
III.A.3.1(4)	Prepare Commission Paper	c	
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	c	
III.A.3.2	Improve Operations Centers	c	
III.A.3.3	Communications	d	See Subsection 9.5.2.5.2 Resolved per NUREG-0933
III.A.3.4	Nuclear Data Link	c	
III.A.3.5	Training, Drills, and Tests	c	
III.A.3.6(1)	Interaction of NRC and Other Agencies - International	c	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.A.3.6(2)	Federal	c	
III.A.3.6(3)	State and Local	c	
III.B.1	Transfer of Responsibilities to FEMA	c	
III.B.2(1)	The Licensing Process	c	
III.B.2(2)	Federal Guidance	c	
III.C.1(1)	Review Publicly Available Documents	d	Not applicable to new plants
III.C.1(2)	Recommend Publication of Additional Information	d	Not applicable to new plants
III.C.1(3)	Program of Seminars for News Media Personnel	d	Not applicable to new plants
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	d	Not applicable to new plants
III.C.2(2)	Provide Training for Members of the Technical Staff	d	Not applicable to new plants
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	g	See Subsection 1.9.3, item (2)(xxvi)
III.D.1.1(2)	Review Information on Provisions for Leak Detection	a	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	a	
III.D.1.2	Radioactive Gas Management	a	
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	a	
III.D.1.3(2)	Review and Revise SRP	a	
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	a	
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	c	
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	a	
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	a	
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	a	
III.D.2.1(3)	Revise Regulatory Guides	a	
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	c	
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	e	
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	e	
III.D.2.2(4)	Revise SRP and Regulatory Guides	e	
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	c	
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	c	
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	c	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.D.2.3(4)	Prepare a Summary Assessment	c	
III.D.2.4(1)	Study Feasibility of Environmental Monitors	c	
III.D.2.4(2)	Place 50 TLDs Around Each Site	d	Not applicable to new plants
III.D.2.5	Offsite Dose Calculation Manual	c	
III.D.2.6	Independent Radiological Measurements	d	Not applicable to new plants
III.D.3.1	Radiation Protection Plans	c	
III.D.3.2(1)	Amend 10 CFR 20	d	Not applicable to new plants
III.D.3.2(2)	Issue a Regulatory Guide	d	Not applicable to new plants
III.D.3.2(3)	Develop Standard Performance Criteria	d	Not applicable to new plants
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	d	Not applicable to new plants
III.D.3.3	In-plant Radiation Monitoring	g COL Item 12.3-2	See Subsection 1.9.3, item (2)(xxvii) 12.3.4, Appendix 12AA
III.D.3.4	Control Room Habitability	g	See Subsection 1.9.3, item (2)(xxviii)
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	d	Not applicable to new plants
III.D.3.5(2)	Investigate Methods of Obtaining Employee Health Data by Nonlegislative Means	d	Not applicable to new plants
III.D.3.5(3)	Revise 10 CFR 20	d	Not applicable to new plants
IV.A.1	Seek Legislative Authority	d	Not applicable to new plants
IV.A.2	Revise Enforcement Policy	d	Not applicable to new plants
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	d	Not applicable to new plants
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	c	
IV.D.1	NRC Staff Training	d	Not applicable to new plants
IV.E.1	Expand Research on Quantification of Safety Decision-Making	d	Not applicable to new plants
IV.E.2	Plan for Early Resolution of Safety Issues	d	Not applicable to new plants
IV.E.3	Plan for Resolving Issues at the CP Stage	d	Not applicable to new plants

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
IV. E.4	Resolve Generic Issues by Rulemaking	d	Not applicable to new plants
IV.E.5	Assess Currently Operating Reactors	c	
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	c	
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	c	
IV.G.1	Develop a Public Agenda for Rulemaking	d	Not applicable to new plants
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	d	Not applicable to new plants
IV.G.3	Improve Rulemaking Procedures	d	Not applicable to new plants
IV.G.4	Study Alternatives for Improved Rulemaking Process	d	Not applicable to new plants
IV.H.1	NRC Participation in the Radiation Policy Council	d	Not applicable to new plants
V.A.1	Develop NRC Policy Statement on Safety	d	Not applicable to new plants
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	d	Not applicable to new plants
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	d	Not applicable to new plants
V.C.2	Study Need for Additional Advisory Committees	d	Not applicable to new plants
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	d	Not applicable to new plants
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	d	Not applicable to new plants
V.D.2	Study Construction-During-Adjudication Rules	d	Not applicable to new plants
V.D.3	Reexamine Commission Role in Adjudication	d	Not applicable to new plants
V.D.4	Study the Reform of the Licensing Process	d	Not applicable to new plants
V.E.1	Study the Need for TMI-Related Legislation	d	Not applicable to new plants
V.F.1	Study NRC Top Management Structure and Process	d	Not applicable to new plants
V.F.2	Reexamine Organization and Functions of the NRC Offices	d	Not applicable to new plants
V.F.3	Revise Delegations of Authority to Staff	d	Not applicable to new plants
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	d	Not applicable to new plants

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	d	Not applicable to new plants
V.G.1	Achieve Single Location, Long-Term	d	Not applicable to new plants
V.G.2	Achieve Single Location, Interim	d	Not applicable to new plants
Task Action Plan Items			
A-1	Water Hammer (former USI)	g	See Subsection 1.9.4, item A-1
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	g	See Subsection 1.9.4, item A-2
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	g COL Item 5.4-1	See Subsection 1.9.4, item A-3 5.4.2.5
A-4	CE Steam Generator Tube Integrity (former USI)	b	
A-5	B&W Steam Generator Tube Integrity (former USI)	b	
A-6	Mark I Short-Term Program (former USI)	b	
A-7	Mark I Long-Term Program (former USI)	b	
A-8	Mark II Containment Pool Dynamic Loads Long-Term Program (former USI)	b	
A-9	ATWS (former USI)	g	See Subsection 1.9.4, item A-9
A-10	BWR Feedwater Nozzle Cracking (former USI)	b	
A-11	Reactor Vessel Materials Toughness (former USI)	g	See Subsection 1.9.4, item A-11
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	g	See Subsection 1.9.4, item A-12
A-13	Snubber Operability Assurance	g	See Subsection 1.9.4, item A-13
A-14	Flaw Detection	a	
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	c	
A-16	Steam Effects on BWR Core Spray Distribution	b	
A-17	Systems Interactions in Nuclear Power Plants (former USI)	c, j	See Subsection 1.9.4.2.2, item A-17
A-18	Pipe Rupture Design Criteria	a	
A-19	Digital Computer Protection System	d	Not applicable to new plants
A-20	Impacts of the Coal Fuel Cycle	d	Not applicable to new plants
A-21	Main Steam Line Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	a	
A-22	PWR Main Steam Line Break - Core, Reactor Vessel and Containment Building Response	a	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
A-23	Containment Leak Testing	d COL Item 6.2-1	6.2.5.1
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	g	See Subsection 1.9.4, item A-24
A-25	Non-Safety Loads on Class 1E Power Sources	g	See Subsection 1.9.4, item A-25
A-26	Reactor Vessel Pressure Transient Protection (former USI)	g	See Subsection 1.9.4, item A-26
A-27	Reload Applications	d	Not applicable to new plants
A-28	Increase in Spent Fuel Pool Storage Capacity	g	See Subsection 1.9.4, item A-28
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	c, j	See Subsection 1.9.4.2.2, item A-29
A-30	Adequacy of Safety-Related DC Power Supplies	e	
A-31	RHR Shutdown Requirements (former USI)	g	See Subsection 1.9.4, item A-31
A-32	Missile Effects	e	
A-33	NEPA Review of Accident Risks	i	
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	e	
A-35	Adequacy of Offsite Power Systems	g	See Subsection 1.9.4, item A-35
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	g	See Subsection 1.9.4, item A-36
A-37	Turbine Missiles	a	
A-38	Tornado Missiles	a	
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	b	See Subsection 1.9.4, item A-39
A-40	Seismic Design Criteria - Short Term Program (former USI)	g	See Subsection 1.9.4, item A-40
A-41	Long Term Seismic Program	c	
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	b	
A-43	Containment Emergency Sump Performance (former USI)	g	See Subsection 1.9.4, item A-43
A-44	Station Blackout (former USI)	g	See Subsection 1.9.4, item A-44
A-45	Shutdown Decay Heat Removal Requirements (former USI)	c	
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	g	See Subsection 1.9.4, item A-46
A-47	Safety Implications of Control Systems (former USI)	g	See Subsection 1.9.4, item A-47

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	g	See Subsection 1.9.4, item A-48
A-49	Pressurized Thermal Shock (former USI)	g	See Subsection 1.9.4, item A-49
B-1	Environmental Technical Specifications	d	Not applicable to new plants
B-2	Forecasting Electricity Demand	d	Not applicable to new plants
B-3	Event Categorization	a	
B-4	ECCS Reliability	e	
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	c, j	See Subsection 1.9.4.2.2, item B-5
B-6	Loads, Load Combinations, Stress Limits	e	
B-7	Secondary Accident Consequence Modeling	a	
B-8	Locking Out of ECCS Power Operated Valves	a	
B-9	Electrical Cable Penetrations of Containment	c	
B-10	Behavior of BWR Mark III Containments	b	
B-11	Subcompartment Standard Problems	d	Not applicable to new plants
B-12	Containment Cooling Requirements (Non-LOCA)	c	
B-13	Marviken Test Data Evaluation	d	Not applicable to new plants
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	e	
B-15	CONTEMPT Computer Code Maintenance	a	
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	e	
B-17	Criteria for Safety-Related Operator Actions	c	See Subsection 1.9.4, item B-17
B-18	Vortex Suppression Requirements for Containment Sumps	e	
B-19	Thermal-Hydraulic Stability	c	
B-20	Standard Problem Analysis	d	Not applicable to new plants
B-21	Core Physics	a	
B-22	LWR Fuel	a	See Subsection 1.9.4, item B-22
B-23	LMFBR Fuel	a	
B-24	Seismic Qualification of Electrical and Mechanical Components	e	
B-25	Piping Benchmark Problems	d	Not applicable to new plants
B-26	Structural Integrity of Containment Penetrations	c	
B-27	Implementation and Use of Subsection NF	d	Not applicable to new plants

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
B-28	Radionuclide/Sediment Transport Program	d	Not applicable to new plants
B-29	Effectiveness of Ultimate Heat Sinks	d	See Subsection 1.9.4, item B-29 Not applicable to new plants
B-30	Design Basis Floods and Probability	d	Not applicable to new plants
B-31	Dam Failure Model	a	
B-32	Ice Effects on Safety-Related Water Supplies	e	See Subsection 1.9.4, item B-32
B-33	Dose Assessment Methodology	d	Not applicable to new plants
B-34	Occupational Radiation Exposure Reduction	e	
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	d	Not applicable to new plants
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	g	See Subsection 1.9.4, item B-36
B-37	Chemical Discharges to Receiving Waters	d	Not applicable to new plants
B-38	Reconnaissance Level Investigations	a	
B-39	Transmission Lines	a	
B-40	Effects of Power Plant Entrainment on Plankton	a	
B-41	Impacts on Fisheries	a	
B-42	Socioeconomic Environmental Impacts	d	Not applicable to new plants
B-43	Value of Aerial Photographs for Site Evaluation	d	Not applicable to new plants
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	d	Not applicable to new plants
B-45	Need for Power - Energy Conservation	e	
B-46	Cost of Alternatives in Environmental Design	a	
B-47	Inservice Inspection of Supports - Classes 1, 2, 3, and MC Components	a	
B-48	BWR CRD Mechanical Failure (Collet Housing)	b	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	d	Not applicable to new plants
B-50	Post-Operating Basis Earthquake Inspections	a	
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	e	
B-52	Fuel Assembly Seismic and LOCA Responses	e	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
B-53	Load Break Switch	g	See Subsection 1.9.4, item B-53
B-54	Ice Condenser Containments	c	
B-55	Improved Reliability of Target Rock Safety Relief Valves	b	
B-56	Diesel Reliability	g	See Subsection 1.9.4, item B-56
B-57	Station Blackout	e	
B-58	Passive Mechanical Failures	c	
B-59	(N-1) Loop Operation in BWRs and PWRs	d	Not applicable to new plants
B-60	Loose Parts Monitoring System	c	
B-61	Allowable ECCS Equipment Outage Periods	g	See Subsection 1.9.4, item B-61
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	a	
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	g	See Subsection 1.9.4, item B-63
B-64	Decommissioning of Reactors	f	Resolved per NUREG-0933
B-65	Iodine Spiking	a	
B-66	Control Room Infiltration Measurements	g	See Subsection 1.9.4, item B-66
B-67	Effluent and Process Monitoring Instrumentation	e	
B-68	Pump Overspeed During LOCA	a	
B-69	ECCS Leakage Ex-Containment	e	
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	c	
B-71	Incident Response	e	
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	d	Not applicable to new plants
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	e	
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	g	See Subsection 1.9.4, item C-1
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	c	
C-3	Insulation Usage Within Containment	e	
C-4	Statistical Methods for ECCS Analysis	d	See Subsection 1.9.4, item C-4 Not applicable to new plants

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
C-5	Decay Heat Update	d	See Subsection 1.9.4, item C-5 Not applicable to new plants
C-6	LOCA Heat Sources	d	See Subsection 1.9.4, item C-6 Not applicable to new plants
C-7	PWR System Piping	c	
C-8	Main Steam Line Leakage Control Systems	b	
C-9	RHR Heat Exchanger Tube Failures	a	
C-10	Effective Operation of Containment Sprays in a LOCA	g	See Subsection 1.9.4, item C-10
C-11	Assessment of Failure and Reliability of Pumps and Valves	c	
C-12	Primary System Vibration Assessment	c	
C-13	Non-Random Failures	e	
C-14	Storm Surge Model for Coastal Sites	a	
C-15	NUREG Report for Liquids Tank Failure Analysis	a	
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	a	
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	g	See Subsection 1.9.4, item C-17
D-1	Advisability of a Seismic Scram	a	
D-2	Emergency Core Cooling System Capability for Future Plants	a	
D-3	Control Rod Drop Accident	c	
New Generic Issues			
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	a	
2.	Failure of Protective Devices on Essential Equipment	a	
3.	Set Point Drift in Instrumentation	c	
4.	End-of-Life and Maintenance Criteria	c	
5.	Design Check and Audit of Balance-of-Plant Equipment	e	
6.	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	c	
7.	Failures Due to Flow-Induced Vibrations	a	
8.	Inadvertent Actuation of Safety Injection in PWRs	e	
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	e	
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	a	
11.	Turbine Disc Cracking	e	
12.	BWR Jet Pump Integrity	b	
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	a	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
14.	PWR Pipe Cracks	c, j	See Subsection 1.9.4.2.3, item 14
15.	Radiation Effects on Reactor Vessel Supports	c	See Subsection 1.9.4, item 15
16.	BWR Main Steam Isolation Valve Leakage Control Systems	e	
17.	Loss of Offsite Power Subsequent to LOCA	a	
18.	Steam Line Break with Consequential Small LOCA	e	
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	e	
20.	Effects of Electromagnetic Pulse on Nuclear Power Plants	c	
21.	Vibration Qualification of Equipment	a	
22.	Inadvertent Boron Dilution Events	c, j	See Subsection 1.9.4.2.3, item 22
23.	Reactor Coolant Pump Seal Failures	c	See Subsection 1.9.4, item 23
24.	Automatic Emergency Core Cooling System Switch to Recirculation	a, j	See Subsection 1.9.4.2.3, item 24
25.	Automatic Air Header Dump on BWR Scram System	b	
26.	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	e	
27.	Manual vs. Automated Actions	e	
28.	Pressurized Thermal Shock	e	
29.	Bolting Degradation or Failure in Nuclear Power Plants	c	See Subsection 1.9.4, item 29
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	a	
31.	Natural Circulation Cooldown	e	
32.	Flow Blockage in Essential Equipment Caused by Corbicula	e	
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	e	
34.	RCS Leak	a	
35.	Degradation of Internal Appurtenances in LWRs	a	
36.	Loss of Service Water	c	
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	e	
38.	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	a	
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	e	
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	b	
41.	BWR Scram Discharge Volume Systems	b	
42.	Combination Primary/Secondary System LOCA	e	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
43.	Reliability of Air Systems	f, j	See Subsection 1.9.4.2.3, item 43 Resolved per NUREG-0933
44.	Failure of Saltwater Cooling System	e	
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	g	See Subsection 1.9.4, item 45
46.	Loss of 125 Volt DC Bus	e	
47.	Loss of Off-Site Power	c	
48.	LCO for Class e Vital Instrument Buses in Operating Reactors	e	
49.	Interlocks and LCOs for Redundant Class e Tie Breakers	e	
50.	Reactor Vessel Level Instrumentation in BWRs	c	
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	g	See Subsection 1.9.4, item 51
52.	SSW Flow Blockage by Blue Mussels	e	
53.	Consequences of a Postulated Flow Blockage Incident in a BWR	a	
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	e	
55.	Failure of Class e Safety-Related Switchgear Circuit Breakers to Close on Demand	a	
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	e	
57.	Effects of Fire Protection System Actuation	c	See Subsection 1.9.4, item 57
58.	Inadvertent Containment Flooding	a	
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	d	Not applicable to new plants
60.	Lamellar Tearing of Reactor Systems Structural Supports	e	
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	c	
62.	Reactor Systems Bolting Applications	e	
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	a	
64.	Identification of Protection System Instrument Sensing Lines	c	
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	e	
66.	Steam Generator Requirements	c	
67.2.1	Integrity of Steam Generator Tube Sleeves	d	Not applicable to new plants
67.3.1	Steam Generator Overfill	e	
67.3.2	Pressurized Thermal Shock	e	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
67.3.3	Improved Accident Monitoring	e, j	See Subsection 1.9.4.2.3, item 67.3.3
67.3.4	Reactor Vessel Inventory Measurements	e	
67.4.1	RCP Trip	e	
67.4.2	Control Room Design Review	e	
67.4.3	Emergency Operating Procedures	e	
67.5.1	Reassessment of SGTR Design Basis	d	
67.5.1	Reassessment of Radiological Consequences	d	Not applicable to new plants
67.5.2	Reevaluation of SGTR Design Basis	d	Not applicable to new plants
67.5.3	Secondary System Isolation	a	
67.6.0	Organizational Responses	e	
67.7.0	Improved Eddy Current Tests	e	
67.8.0	Denting Criteria	e	
67.9.0	Reactor Coolant System Pressure Control	e	
67.10.0	Supplement Tube Inspections	d	Not applicable to new plants
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pumps Steam Supply Line Rupture	e	
69.	Make-up Nozzle Cracking in B&W Plants	c	
70.	PORV and Block Valve Reliability	g	See Subsection 1.9.3, item (1)(iv)
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	a	
72.	Control Rod Drive Guide Tube Support Pin Failures	a	
73.	Detached Thermal Sleeves	a, j	See Subsection 1.9.4.2.3, item 73
74.	Reactor Coolant Activity Limits for Operating Reactors	a	
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	g, j	See Subsection 1.9.4.2.3, item 75
76.	Instrumentation and Control Power Interactions	a	
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	e	
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	c	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Circulation Cooldown	c	See Subsection 1.9.4.2.3, item 79
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	a	
81.	Impact of Locked Doors and Barriers on Plant and Personnel Safety	a	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
82.	Beyond Design Basis Accidents in Spent Fuel Pools	c, j	See Subsection 1.9.4.2.3, item 82
83.	Control Room Habitability	c	See Subsection 1.9.4.2.3, item 83
84.	CE PORVs	c	
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	a	
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	b	
87.	Failure of HPCI Steam Line Without Isolation	g	See Subsection 1.9.4, item 87
88.	Earthquakes and Emergency Planning	c	
89.	Stiff Pipe Clamps	h (Medium)	
90.	Technical Specifications for Anticipatory Trips	a	
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	c	
92.	Fuel Crumbling During LOCA	a	
93.	Steam Binding of Auxiliary Feedwater Pumps	g	See Subsection 1.9.4, item 93
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	g	See Subsection 1.9.4, item 94
95.	Loss of Effective Volume for Containment Recirculation Spray	c	
96.	RHR Suction Valve Testing	e	
97.	PWR Reactor Cavity Uncontrolled Exposures	e	
98.	CRD Accumulator Check Valve Leakage	a	
99.	RCS/RHR Suction Line Valve Interlock on PWRs	f	Resolved per NUREG-0933
100.	OTSG Level	b	
101.	BWR Water Level Redundancy	c	
102.	Human Error in Events Involving Wrong Unit or Wrong Train	c	
103.	Design for Probable Maximum Precipitation	g	See Subsection 1.9.4, item 103
104.	Reduction of Boron Dilution Requirements	a	
105.	Interfacing Systems LOCA at BWRs	c	See Subsection 1.9.4, item 105
106.	Piping and Use of Highly Combustible Gases in Vital Areas	c	See Subsection 1.9.4, item 106
107.	Main Transformer Failures	a	
108.	BWR Suppression Pool Temperature Limits	a	
109.	Reactor Vessel Closure Failure	a	
110.	Equipment Protective Devices on Engineered Safety Features	a	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	d	Not applicable to new plants
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	d	Not applicable to new plants
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	c	See Subsection 1.9.4, item 113
114.	Seismic-Induced Relay Chatter	e	
115.	Enhancement of the Reliability of Westinghouse Solid State Protection System	c	
116.	Accident Management	a	
117.	Allowable Time for Diverse Simultaneous Equipment Outages	a	
118.	Tendon Anchorage Failure	f	Resolved per NUREG-0933.
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	d	Not applicable to new plants
119.2	Piping Damping Values	a	
119.3	Decoupling the OBE from the SSE	d	Not applicable to new plants
119.4	BWR Piping Materials	d	Not applicable to new plants
119.5	Leak Detection Requirements	d	Not applicable to new plants
120.	On-Line Testability of Protection Systems	c, j	See Subsection 1.9.4.2.3, item 120
121.	Hydrogen Control for Large, Dry PWR Containments	c	See Subsection 1.9.4, item 121
122.1.a	Failure of Isolation Valves in Closed Position	e	
122.1.b	Recovery of Auxiliary Feedwater	e	
122.1.c	Interruption of Auxiliary Feedwater Flow	e	
122.2	Initiating Feed-and-Bleed	c	
122.3	Physical Security System Constraints	a	
123.	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	a	
124.	Auxiliary Feedwater System Reliability	g	See Subsection 1.9.4, item 124
125.I.1	Availability of the STA	a	
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	e	
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	e	
125.I.2.c	Need for Additional Protection Against PORV Failure	a	

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Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	e	
125.I.3	SPDS Availability	c	
125.I.4	Plant-Specific Simulator	a	
125.I.5	Safety Systems Tested in All Conditions Required by Design Basis Analysis	a	
125.I.6	Valve Torque Limit and Bypass Switch Settings	a	
125.I.7.a	Recover Failed Equipment	a	
125.I.7.b	Realistic Hands-On Training	a	
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	a	
125.II.1.a	Two-Train AFW unavailability	a	
125.II.1.b	Review Existing AFW Systems for Single Failure	e	
125.II.1.c	NUREG-0737 Reliability Improvements	a	
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	a	
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	a	
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	a	
125.II.4	Thermal Stress of OTSG Components	a	
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	a	
125.II.6	Reexamine PRA-Based Estimates of the Likelihood of a Severe Core Damage Accident Based on Loss of All Feedwater	a	
125.II.7	Reevaluate Provisions to Automatically Isolate Feedwater from Steam Generator During a Line Break	c	
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	a	
125.II.9	Enhanced Feed-and-Bleed Capability	a	
125.II.10	Hierarchy of Impromptu Operator Actions	a	
125.II.11	Recovery of Main Feedwater as Alternative to AFW	a	
125.II.12	Adequacy of Training Regarding PORV Operation	a	
125.II.13	Operator Job Aids	a	
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	a	
126.	Reliability of PWR Main Steam Safety Valves	d	
127.	Testing and Maintenance of Manual Valves in Safety-Related Systems	a	
128.	Electrical Power Reliability	h (High)	See Subsection 1.9.4, item 128 Resolved per NUREG-0933.

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Table 1.9-2 (Sheet 28 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	a	
130.	Essential Service Water Pump Failures at Multiplant Sites	f	See Subsection 1.9.4, item 130 See Subsection 1.9.4, item 130
131.	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System in Westinghouse Plants	e	
132.	RHR Pumps Inside Containment	a	
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	d	Not applicable to new plants
134.	Rule on Degree and Experience Requirements	c	
135.	Steam Generator and Steam Line Overfill	c	See Subsection 1.9.4, item 135
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	d	Not applicable to new plants
137.	Refueling Cavity Seal Failure	a	
138.	Deinerting Upon Discovery of RCS Leakage	a	
139.	Thinning of Carbon Steel Piping in LWRs	d	Not applicable to new plants
140.	Fission Product Removal Systems	a	
141.	LBLOCA With Consequential SGTR	a	
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	c	See Subsection 1.9.4, item 142
143.	Availability of Chilled Water Systems	c, j	See Subsection 1.9.4.2.3, item 143
144.	Scram Without a Turbine/Generator Trip	a	
145.	Actions to Reduce Common Cause Failures	c	
146.	Support Flexibility of Equipment and Components	d	Not applicable to new plants
147.	Fire-Induced Alternate Shutdown Control Room Panel Interactions	d	Not applicable to new plants
148.	Smoke Control and Manual Fire-Fighting Effectiveness	d	Not applicable to new plants
149.	Adequacy of Fire Barriers	a	
150.	Overpressurization of Containment Penetrations	a	
151.	Reliability of Recirculation Pump Trip During an ATWS	c	

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Table 1.9-2 (Sheet 29 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
152.	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	a	
153.	Loss of Essential Service Water in LWRs	c, j	See Subsection 1.9.4.2.3, item 153
154.	Adequacy of Emergency and Essential Lighting	a	
155.1	More Realistic Source Term Assumptions	g	
155.2	Establish Licensing Requirements For Non-Operating Facilities	d	Not applicable to new plants
155.3	Improve Design Requirements For Nuclear Facilities	a	
155.4	Improve Criticality Calculations	a	
155.5	More Realistic Severe Reactor Accident Scenario	a	
155.6	Improve Decontamination Regulations	a	
155.7	Improve Decommissioning Regulations	a	
156	Systematic Evaluation Program	f	Not applicable to new plants
156.6.1	Pipe Break Effects on Systems and Components	High	The AP1000 is a new plant that takes the effects of a pipe break into account and therefore issue 156.6.1 is not applicable.
157	Containment Performance	c	
158	Performance Of Safety-Related Power-Operated Valves Under Design Basis Conditions	c	
159	Qualification Of Safety-Related Pumps While Running On Minimum Flow	a	
160	Spurious Actuations Of Instrumentation Upon Restoration Of Power	a	
161	Use Of Non-Safety-Related Power Supplies In Safety-Related Circuits	a	
162	Inadequate Technical Specifications For Shared Systems At Multiplant Sites When One Unit Is Shut Down	a	
163	Multiple Steam Generator Tube Leakage	h (Medium) h (High)	See Subsection 1.9.4.2.3, item 163 See Subsection 1.9.4.2.3, item 163
164	Neutron Fluence In Reactor Vessel	a	
165	Spring-Actuated Safety And Relief Valve Reliability	c	
166	Adequacy Of Fatigue Life Of Metal Components	c	
167	Hydrogen Storage Facility Separation	a	

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Table 1.9-2 (Sheet 30 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
168	Environmental Qualification Of Electrical Equipment	f	See Subsection 1.9.4.2.3, item 168 Not applicable to new plants
169	BWR MSIV Common Mode Failure Due To Loss Of Accumulator Pressure	a	
170	Fuel Damage Criteria For High Burnup Fuel	c	
171	ESF Failure From LOOP Subsequent To A LOCA	c	
172	Multiple System Responses Program	e	
173.A	Spent Fuel Storage Pool Operating Facilities	c	
173.B	Spent Fuel Storage Pool Permanently Shutdown Facilities	c	
174	Fastener Gaging Practices	c	
175	Nuclear Power Plant Shift Staffing	c	
176	Loss Of Fill-Oil In Rosemount Transmitters	c	
177	Vehicle Intrusion At TMI	g	
178	Effect Of Hurricane Andrew On Turkey Point	d	Not applicable to new plants
179	Core Performance	c	
180	Notice Of Enforcement Discretion	d	Not applicable to new plants
181	Fire Protection	d	Not applicable to new plants
182	General Electric Extended Power Uprate	b	
183	Cycle-Specific Parameter Limits In Technical Specifications	d	Not applicable to new plants
184	Endangered Species	d	Not applicable to new plants
185	Control of Recriticality following Small-Break LOCA in PWRs	h (High)	See Subsection 1.9.4.2.3, item 185 Not applicable to new plants
186	Potential Risk and Consequences of Heavy Load Drops	a	
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	Continue	1.9.4.2.3 9.1.5.3
187	The Potential impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants.	a	
188	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	a	
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Concentration during a Severe Accident	a	
189	Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident Description	Continue	Not applicable to the AP1000.

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Table 1.9-2 (Sheet 31 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
190	Fatigue Evaluation Of Metal Components For 60-Year Plant Life	c	
191	Assessment Of Debris Accumulation On PWR Sump Performance	h (High)	See Subsections 6.3.2.2.7 and 1.9.4.2.3, item 191 See Subsections 6.3.2.2.7 and 1.9.4.2.3, item 191
199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	Issue to be Prioritized by NRC in the Future	2.5
Human Factors Issues			
HF1.1	Shift Staffing	f	18.6
HF1.2	Engineering Expertise on Shift	c	
HF1.3	Guidance on Limits and Conditions of Shift Work	c	
HF2.1	Evaluate Industry Training	d	Not applicable to new plants
HF2.2	Evaluate INPO Accreditation	d	Not applicable to new plants
HF2.3	Revise SRP Section 13.2	d	Not applicable to new plants
HF3.1	Develop Job Knowledge Catalog	d	Not applicable to new plants
HF3.2	Develop License Examination Handbook	d	Not applicable to new plants
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	e	
HF3.4	Examination Requirements	e	
HF3.5	Develop Computerized Exam System	d	Not applicable to new plants
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	c, i	
HF4.2	Procedures Generation Package Effectiveness Evaluation	d	Not applicable to new plants
HF4.3	Criteria for Safety-Related Operator Actions	e	
HF4.4	Guidelines for Upgrading Other Procedures	c, j	See Subsection 1.9.4.2.4, item HF4.4
HF4.5	Application of Automation and Artificial Intelligence	e	
HF5.1	Local Control Stations	c	See Subsection 1.9.4, item HF5.1

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Table 1.9-2 (Sheet 32 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	g	See Subsection 1.9.4, item HF5.2
HF5.3	Evaluation of Operational Aid Systems	e	
HF5.4	Computers and Computer Displays	e	
HF6.1	Develop Regulatory Position on Management and Organization	e	
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	e	
HF7.1	Human Error Data Acquisition	d	Not applicable to new plants
HF7.2	Human Error Data Storage and Retrieval	d	Not applicable to new plants
HF7.3	Reliability Evaluation Specialist Aids	d	Not applicable to new plants
HF7.4	Safety Event Analysis Results Applications	d	Not applicable to new plants
HF8	Maintenance and Surveillance Program	c	
Chernobyl Issues			
CH1.1A	Symptom-Based EOPs	d	Not applicable to new plants
CH1.1B	Procedure Violations	d	Not applicable to new plants
CH1.2A	Test, Change, and Experiment Review Guidelines	d	Not applicable to new plants
CH1.2B	NRC Testing Requirements	d	Not applicable to new plants
CH1.3A	Revise Regulatory Guide 1.47	d	Not applicable to new plants
CH1.4A	Engineered Safety Feature Availability	d	Not applicable to new plants
CH1.4B	Technical Specification Bases	d	Not applicable to new plants
CH1.4C	Low Power and Shutdown	d	Not applicable to new plants
CH1.5	Operating Staff Attitudes Toward Safety	d	Not applicable to new plants
CH1.6A	Assessment of NRC Requirements on Management	d	Not applicable to new plants
CH1.7A	Accident Management	d	Not applicable to new plants
CH2.1A	Reactivity Transients	d	Not applicable to new plants
CH2.2	Accidents at Low Power and at Zero Power	e	
CH2.3A	Control Room Habitability	e	

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Table 1.9-2 (Sheet 33 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
CH2.3B	Contamination Outside Control Room	d	Not applicable to new plants
CH2.3C	Smoke Control	d	Not applicable to new plants
CH2.3D	Shared Shutdown Systems	d	Not applicable to new plants
CH2.4A	Firefighting With Radiation Present	d	Not applicable to new plants
CH3.1A	Containment Performance	d	Not applicable to new plants
CH3.2A	Filtered Venting	d	Not applicable to new plants
CH4.1	Size of the Emergency Planning Zones	a	
CH4.2	Medical Services	a	
CH4.3A	Ingestion Pathway Protective Measures	d	Not applicable to new plants
CH4.4A	Decontamination	d	Not applicable to new plants
CH4.4B	Relocation	d	Not applicable to new plants
CH5.1A	Mechanical Dispersal in Fission Product Release	d	Not applicable to new plants
CH5.1B	Stripping in Fission Product Release	d	Not applicable to new plants
CH5.2A	Steam Explosions	d	Not applicable to new plants
CH5.3	Combustible Gas	a	
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	a	
CH6.1B	Structural Graphite Experiments	d	Not applicable to new plants
CH6.2	Assessment	d	Not applicable to new plants

Notes:
a. Issue has been prioritized as **Low, Drop** or has not been prioritized.
b. Issue is not an AP1000 design issue. Issue is applicable to GE, B&W, or CE designs only.
c. Issue resolved with no new requirements.
d. Issue is not a design issue (Environmental, Licensing, or Regulatory Impact Issue; or covered in an existing NRC program).
e. Issue superseded by one or more issues.
f. Issue is not an AP1000 design certification issue. Issue is applicable to current operating plants or is programmatic in nature.
g. Issue is resolved by establishment of new regulatory requirements and/or guidance.
h. Issue is unresolved pending generic resolution (for example, prioritized as **High, Medium**, or possible resolution identified).
i. The AP600 DSER (Draft NUREG-01512) identified this item as not being required to be addressed by 10 CFR 52.47.
j. The AP600 DSER (Draft NUREG-01512) identified this item as required to be discussed.

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Table 1.9-201
Not Used

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Table 1.9-202
Not Used

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Table 1.9-203
Not Used

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Table 1.9-204 (Sheet 1 of 6)
Generic Communications Assessment

Number	Title	Comment
BULLETIN		
80-06	Engineered Safety Feature (ESF) Reset Controls (3/80)	See Note a.
80-10	Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment (5/80)	Appendix 12AA
80-15	Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power (6/80)	9.5.2.5.1
88-11	Pressurizer Surge Line Thermal Stratification	3.9.3.1.2
02-01	Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity	5.2.4 See Note a.
02-02	Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs	5.2.4 See Note a.
03-01	Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors	6.3 See Note a.
03-02	Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity	5.2.4.3 See Note a.
03-03	Potentially Defective 1-inch Valves for Uranium Hexafluoride Cylinders	N/A
03-04	Rebaselining of Data in the Nuclear Materials Management and Safeguards System	N/A One time report.

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Table 1.9-204 (Sheet 2 of 6)
Generic Communications Assessment

Number	Title	Comment
04-01	Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors	See Note a.
05-01	Material Control and Accounting at Reactors and Wet Spent Fuel Storage Facilities	13.5.2.2.9
05-02	Emergency Preparedness and Response Actions for Security-Based Events	13.3
GENERIC LETTERS		
80-22	Transmittal of NUREG-0654 “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans” (3/80)	13.3
80-26	Qualifications of Reactor Operators (3/80)	13.2 18.10
80-51	On-Site Storage Of Low-Level Waste (6/90)	11.4.6
80-55	Possible Loss of Hotline With Loss Of Off-Site Power	See Bulletin 80-15
80-77	Refueling Water Level (8/80)	16.1 See Note a.
80-094	Emergency Plan (11/80)	13.3
80-099	Technical Specification Revisions for Snubber Surveillance (11/80)	Snubbers no longer in generic Tech Specs See Note a.
80-108	Emergency Planning (12/80)	13.3
81-02	Analysis, Conclusions and Recommendations Concerning Operator Licensing (1/81)	13.2

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Table 1.9-204 (Sheet 3 of 6)
Generic Communications Assessment

Number	Title	Comment
81-10	Post-TMI Requirements for the Emergency Operations Facility (2/81)	13.3
81-38	Storage of Low-Level Radioactive Waste at Power Reactor Sites (11/81)	11.4.6
81-40	Qualifications of Reactor Operators (12/81)	13.1 13.2
82-02	Commission Policy on Overtime (2/82)	16.1
82-04	Use of INPO See-in Program (3/82)	13.1 13.5
82-13	Reactor Operator and Senior Reactor Operator Examinations (6/82)	For information only.
82-18	Reactor Operator and Senior Reactor Operator Requalification Examinations (10/82)	13.2
83-06	Certificates and Revised Format For Reactor Operator and Senior Reactor Operator Licenses (1/83)	13.2
83-11	Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (2/83)	13.1 See Note a.
83-12	Issuance of NRC FORM 398 - Personal Qualifications Statement - Licensee (2/83)	13.2
83-17	Integrity of the Requalification Examinations for Renewal of Reactor Operator and Senior Reactor Operator Licenses (4/83)	13.1
83-22	Safety Evaluation of "Emergency Response Guidelines" (6/83)	18.9

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Generic Communications Assessment

Number	Title	Comment
83-40	Operator Licensing Examination (12/83)	13.2
84-10	Administration of Operating Tests Prior to Initial Criticality (10 CFR 55.25) (4/84)	13.2
84-14	Replacement and Requalification Training Program (5/84)	13.2
84-17	Annual Meeting to Discuss Recent Developments Regarding Operator Training, Qualifications, and Examinations (7/84)	Administrative
84-20	Scheduling Guidance for Licensee Submittals of Reloads That Involve Unreviewed Safety Questions (8/84)	13.5
85-04	Operating Licensing Examinations (1/85)	Administrative
85-05	Inadvertent Boron Dilution Events (1/85)	13.5
85-14	Commercial Storage At Power Reactor Sites Of Low Level Radioactive Waste Not Generated By The Utility (8/85)	Administrative
85-18	Operator Licensing Examinations (9/85)	Administrative
85-19	Reporting Requirements On Primary Coolant Iodine Spikes (9/85)	16.1
86-14	Operator Licensing Examinations (8/86)	Administrative
87-14	Operator Licensing Examinations (8/87)	Administrative
88-05	Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (3/88)	5.2.4 See Note a.
88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment (8/88)	9.3.7
88-18	Plant Record Storage on Optical Disk (10/88)	17

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Table 1.9-204 (Sheet 5 of 6)
Generic Communications Assessment

Number	Title	Comment
89-07	Power Reactors Safeguards Contingency Planning for Surface Vehicle Bombs (4/89)	13.6
89-07 S1	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	13.6
89-08	Erosion/Corrosion-Induced Pipe Wall Thinning	10.1.3.1
89-12	Operator Licensing Examination (7/89)	13.2
89-15	Emergency Response Data System (8/89)	9.5.2 13.3
89-17	Planned Administrative Changes to the NRC Operator Licensing Written Examination Process (9/89)	N/A
91-14	Emergency Telecommunications (9/91)	9.5.2 13.3
91-16	Licensed Operators and Other Nuclear Facility Personnel Fitness for Duty (10/91)	13.7
92-01	Reactor Vessel Structural Integrity (1/92)	5.3.2.6.3
93-01	Emergency Response Data System Test Program	13.3
93-03	Verification of Plant Records	17
96-02	Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat (2/96)	13.6
03-01	Control Room Habitability	6.4 See Note a.
04-01	Requirements for Steam Generator Tube Inspections	5.4.2.5 16.1 See Note a.

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Table 1.9-204 (Sheet 6 of 6)
Generic Communications Assessment

Number	Title	Comment
04-02	Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors	6.3.8.1 See Note a.
06-01	Steam Generator Tube Integrity and Associated Technical Specifications	5.4.2.5 16.1 See Note a.
06-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	8.2.1.1 8.2.2 See Note a.
06-03	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations	9.5.1.8 See Note a.
07-01	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients.	17.6 See Note a.

(a) The design aspects of this topic are as stated in the AP1000 DCD.

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1.10 Nuclear Power Plants to be Operated on Multi-Unit Sites

The certification for the AP1000 is for a single unit. Dual siting of AP1000 is achievable, provided that the centerlines of the units are sufficiently separated. The primary consideration in setting this separation distance is the space needed to support plant construction via the use of a heavy-lift crane.

Security controls during construction and operation are addressed in the Physical Security Plan.

Management and administrative controls are established to identify potential hazards to structures, systems, and components (SSCs) of an operating unit as a result of construction activities at a unit under construction. Controls within this section are not required unless there is an operating unit on the site, i.e., a unit with fuel loaded into the reactor vessel. Advance notification, scheduling and planning allow site management to implement interim controls to reduce the potential for impact to SSCs.

This section presents an assessment of the potential impacts of construction of one unit on SSCs important to safety for an operating unit, in accordance with 10 CFR 52.79(a)(31). This assessment includes:

- Identification of potential construction activity hazards
- Identification of SSCs important to safety and limiting conditions for operation (LCOs) for the operating unit
- Identification of potentially impacted SSCs and LCOs
- Identification of applicable managerial and administrative controls

1.10.1 Potential Construction Activity Hazards

Construction activities may include site exploration, grading, clearing, and installation of drainage and erosion-control measures; boring, drilling, dredging, pile driving and excavating; transportation, storage and warehousing of equipment; and construction, erection, and fabrication of new facilities.

Construction activities and their representative hazards to an operating unit are shown in Table 1.10-201.

1.10.2 Potentially Impacted SSCs and Limiting Conditions for Operation

The construction activities described above were reviewed for possible impact to operating unit SSCs important to safety.

The initial assessment consisted of a review of individual SSCs and LCOs to determine whether an item is applicable, or may be eliminated due to either examination or being internal and specific to an operating unit. The assessment identified the SSCs that could reasonably be expected to be impacted by construction activities unless administrative and managerial controls are established. The results of the assessment are presented in Table 1.10-202.

1.10.3 Managerial and Administrative Controls

To eliminate or mitigate construction hazards that could potentially impact operating unit SSCs important to safety, specific managerial and administrative controls have been identified as shown in Table 1.10-203.

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Although not all of the managerial and administrative construction controls are necessary to protect the operating unit, the identified controls are applied to any operating unit as a conservative measure. This conservative approach provides reasonable assurance of protecting the identified SSCs from potential construction hazards and preventing the associated LCOs specified in the operating unit Technical Specifications from being exceeded as a result of construction activities, as discussed below.

The majority of the operating unit SSCs important to safety are contained and protected within safety-related structures. The managerial controls protect these internal SSCs from postulated construction hazards by maintaining the integrity and design basis of the safety-related structures and foundations. Heavy load drop controls, crane boom failure standoff requirements, ground vibration controls and construction generated missile(s) control are examples of managerial controls that provide this protection.

Other managerial controls support maintaining offsite power, control of hazardous materials and gases, and protection of cooling water supplies and safety system instrumentation. These managerial controls prevent or mitigate external construction impacts that could affect SSCs important to safety. These controls also prevent or mitigate unnecessary challenges to safety systems caused by plant construction hazards, such as disruption of offsite transmission lines or impact to plant cooling water supplies.

The above discussed controls to eliminate or mitigate construction hazards that could potentially impact operating unit SSCs important to safety are in place when there is an operating nuclear unit on the site.

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Table 1.10-201 (Sheet 1 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Site Exploration, Grading, Clearing, Installation of Drainage and Erosion Control Measures	• Overhead Power Lines • Transmission Towers • Underground Conduits, Piping, Tunnels, etc. • Site Access and Egress • Drainage Facilities and Structures • Onsite Transportation Routes • Slope Stability • Soil Erosion and Local Flooding • Construction-Generated Dust and Equipment Exhausts • Encroachment on Plant Control Boundaries • Encroachment on Structures and Facilities
Boring, Drilling, Pile Driving, Dredging, Demolition, Excavation	• Underground Conduits, Piping, Tunnels, etc. • Foundation Integrity • Structural Integrity • Slope Stability • Erosion and Turbidity Control • Groundwater and Groundwater Monitoring Facilities • Dewatering Structures, Systems and Components • Nearby Structures, Systems and Components • Vibratory Ground Motion

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Table 1.10-201 (Sheet 2 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Equipment Movement, Material Delivery, Vehicle Traffic	• Overhead Power Lines • Transmission Towers • Underground Conduits, Piping, Tunnels • Crane Load Drops • Crane or Crane Boom Failures • Vehicle Accidents • Rail Car Derailments
Equipment and Material Laydown, Storage, Warehousing	• Releases of Flammable, Hazardous or Toxic Materials • Wind-Generated, Construction-Related Debris and Missiles
General Construction, Erection, Fabrication	• Physical Integrity of Structures, Systems and Components • Adjacent or Nearby Structures, Systems and Components • Instrumentation and Control Systems and Components • Electrical Systems and Components·Cooling Water Systems and Components • Waste Heat Environmental Controls and Parameters • Radioactive Waste Release Points and Parameters • Abandonment of Structures, Systems or Components • Relocation of Structures, Systems or Components • Removal of Structures, Systems or Components

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Table 1.10-201 (Sheet 3 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Connection, Integration, Testing	• Instrumentation and Control Systems and Components • Electrical and Power Systems and Components • Cooling Water Systems and Components

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Table 1.10-202 (Sheet 1 of 2)
Hazards During Construction Activities

CONSTRUCTION HAZARD	IMPACTED SSCs
Impact on Overhead Power Lines	• Offsite Power System
Impact on Transmission Towers	• Offsite Power Systems
Impact on Utilities, Underground Conduits, Piping, Tunnels, Tanks	• Fire Protection System • Service Water System[1]
Impact of Construction-Generated Dust and Equipment Exhausts	• Control Room Emergency HVAC Systems[1] • Diesel Generators
Impact of Vibratory Ground Motion	• Offsite Power System • Onsite Power Systems • Instrumentation and Seismic Monitors
Impact of Crane or Crane Boom Failures	• Safety-Related Structures
Impact of Releases of Flammable, Hazardous or Toxic Materials	• Control Room Emergency HVAC Systems[1]
Impact of Wind-Generated, Construction-Related Debris and Missiles	• Safety-Related Structures • Control Room Emergency HVAC Systems[1]
Impact on Electrical Systems and Components	• Offsite Power System • Onsite Power Systems
Impact on Cooling Water Systems and Components	• Service Water System[1] • Ultimate Heat Sink[1]
Impact on Radioactive Waste Release Points and Parameters	• Gaseous and Liquid Radioactive Waste Management Systems

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Table 1.10-202 (Sheet 2 of 2)
Hazards During Construction Activities

CONSTRUCTION HAZARD	IMPACTED SSCs
Impact of Relocation of Structures, Systems or Components	• Fire Protection System • Service Water System[1]
Impact of Site Groundwater Depression and Dewatering	• Safety-Related Structures and Foundations
Impact of Equipment Delivery and Heavy Equipment Delivery	• Safety-Related Structures and Foundations
Impact of Local Flooding	• Safety-related structures, systems, and components (SSCs)

1 Not applicable to AP1000 operating units.

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Table 1.10-203 (Sheet 1 of 3)
Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact on Transmission Power Lines and Offsite Power Lines	• Safe standoff clearance distances are established for transmission power lines, including verification of standoff distance for modules, the reactor vessel and other equipment to be transported beneath energized electric lines to meet minimum standoff clearance requirements. • Physical warning or caution barriers and signage are erected along transport routes.
Impact on Transmission Towers	• Establish controls or physical barriers to avoid equipment collisions with electric transmission support towers.
Impact on Utilities, Underground Conduits, Piping, Tunnels, Tanks	• Grading, excavation, and pile driving require location and identification of equipment or underground structures that must be relocated, removed, or left in place and protected prior to the work activity.
Impact of Construction-Generated Dust and Equipment Exhausts	• Fugitive dust and dust generation is controlled. Potentially affected system air intakes and filters are periodically monitored.
Impact of Vibratory Ground Motion	• Construction administrative procedures, methods, and controls are implemented to prevent exceeding ground vibration and instrumentation limit settings.
Impact of Crane or Crane Boom Failures	• Construction standoff distance controls prevent heavy load impacts from crane boom failures and crane load drops. Drop analyses may be substituted if minimum standoff distances are not practical.

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Table 1.10-203 (Sheet 2 of 3)
Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact of Releases of Flammable, Hazardous or Toxic Materials and Missile Generation	• Environmental, safety and health controls limit transport, storage, quantities, type and use of flammable, hazardous, toxic materials and compressed gasses. Construction safety and storage controls maintain potential missile generation events from compressed gasses within the operating unit design basis.
Impact of Wind-Generated, Construction-Related Debris and Missiles	• Administrative controls address equipment, material storage and transport during high winds or high wind warnings. • Plant procedures are followed during severe weather conditions which may call for power reduction or shut down.
Impact on Electrical Systems and Components	• Affected operating unit electrical systems and components within the construction area are identified and isolated or relocated or otherwise protected.
Impact on Cooling Water Systems and Components	• Transport of heavy load equipment over buried cooling water piping is prohibited without evaluation.
Impact on Radioactive Waste Release Points and Parameters	• Engineering evaluation and managerial controls are implemented, as necessary, to prevent radioactive releases beyond the established limits due to construction activity.
Impact of Relocation of Structures, Systems or Components	• Administrative controls identify SSCs that require relocation. Temporary or permanent design changes are implemented if necessary.
Impact of Equipment Delivery and Heavy Equipment Delivery	• Rail transport speed limits and maximum rail loading weights onsite are established. • General equipment and heavy equipment movement controls and limitations are established.

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Table 1.10-203 (Sheet 3 of 3)
Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact of Local Flooding	• Site grading and drainage provisions consider potential flooding impacts from local intense precipitation
Impact of Site Groundwater Dewatering	• Administrative controls address groundwater level monitoring

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Appendix 1A Conformance With Regulatory Guides

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
DIVISION 1 – Power Reactors			
Reg. Guide 1.1, Rev. 0, 11/70 – Net Positive Suction Head For Emergency Core Cooling and Containment Heat Removal System Pumps			
General		N/A	The AP1000 passive safety systems make maximum use of natural phenomena (gravity, natural circulation, and gas driven injection) and fail-safe position valves, and thus require no active pumps, diesel-generators, or fans. The AP1000 normal residual heat removal system is designed to take suction from the cask loading pit, the in-containment refueling water storage tank, and from containment, however it is not a safety-related system, and does not control or mitigate the consequences of an accident in the licensing basis accident analyses.
Reg. Guide 1.2 – Withdrawn			
Reg. Guide 1.3, Rev. 2, 6/74 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.4, Rev. 2, 6/74 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors			
General		Exception	The guidance of Reg. Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors" will be followed instead of Reg. Guide 1.4.
Reg. Guide 1.5, Rev. 0, 3/71 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.6, Rev. 0, 3/71 – Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems			
General		Exception	The AP1000 main ac power system is a nonsafety-related system. This regulatory guide is applicable only to the Class 1E dc and UPS system.
D.1		Conforms	Guidance applies only to the Class 1E dc and UPS system, since the AP1000 ac power system is a nonsafety-related system.
D.2		N/A	The main ac power system is a nonsafety-related system. Therefore, this regulatory position is not applicable. However, the AP1000 design includes connections to a preferred (offsite) power source and two nonsafety-related onsite standby diesel generators.
D.3		Conforms	
D.4		N/A	See comment on Criteria Section D.2.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
D.5		N/A	See comment on Criteria Section D.2.
Reg. Guide 1.7, Rev. 2, 11/78 and Rev. 3, 03/07 – Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident			
Conformance of the design aspects with Revision 2 of the Regulatory Guide is as stated below in the DCD.			
C.1		Conforms	Mixing of the containment atmosphere is accomplished through natural passive processes (natural circulation), not with an active system.
C.2		Conforms	
C.3	Regulatory Guide 1.29 Regulatory Guide 1.26	Conforms	The hydrogen recombiners are passive autocatalytic recombiners and nonsafety-related. They do not require and are not supplied with power.
C.4		Conforms	
C.5		Conforms	
C.6		Conforms	
Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.2		Conforms	
C.4		Conforms	
Reg. Guide 1.8, Rev. 3, 5/00 – Qualification and Training of Personnel for Nuclear Power Plants			
Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 13.2.1 defines the responsibility for the training program for plant personnel.
Conformance for programmatic and/or operational aspects is documented below.			
C.1		Conforms	
C.2	Section 4 of ANSI/ ANS-3.1-1993	Exception	Not able to meet Regulatory Guide 1.8, Rev. 3 qualification requirements for licensed personnel prior to operations. In lieu of the minimum experience requirements specified in ANSI/ANS-3.1-1993, paragraphs 4.5.3.1 (chemistry), 4.5.3.2 (radiation protection) and 4.5.3.3 (instrumentation and control) technicians, and in paragraphs 4.5.7.1 (electrical) and 4.5.7.2 (mechanical) maintenance personnel, the minimum experience requirements specified in ANSI/ANS-3.1-2014 paragraphs 4.5.3.1, 4.5.3.2, 4.5.3.3, 4.5.7.1 and 4.5.7.2, as endorsed by and clarified in RG 1.8, Rev. 4 apply. When fuel is initially loaded in the reactor, and thereafter, whenever fuel is in the reactor, at least one (1) radiation protection technician meeting the minimum qualifications of RG 1.8, Rev. 3, will be on-site and on duty, for compliance with TS 5.3.1. Additional radiation protection technicians on-site when fuel is in the reactor may be qualified to RG 1.8, Rev. 4.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.9, Rev. 2, 12/79 – Selection, Design, and Qualification of Diesel Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants			
General		N/A	Guidelines apply to Class 1E diesel-generators. They are not applicable to the AP1000.
C.1-14		N/A	Guidelines apply to Class 1E diesel-generators. They are not applicable to the AP1000.
Reg. Guide 1.10 – Withdrawn			
Reg. Guide 1.11, Rev.1, 3/10 – Instrument Lines Penetrating Primary Reactor Containment			
General		Conforms	
C.1		Conforms	
C.2.a-c		Conforms	
C.2.d	10 CFR 100	Conforms	
C.3.a-c		N/A	Design features described are not applicable to AP1000 Design.
C.4		Conforms	Regulatory position describes the AP1000 implementation.
C.5		Conforms	
C.6		Conforms	
C.7.a-b		N/A	There are no instrument lines penetrating containment that are not associated with protection or safety system instrumentation.
Reg. Guide 1.12, Rev. 2, 3/97 – Instrumentation for Earthquakes			
Conformance of the design aspects is as stated below in the DCD.			
C.1		Exception	Two elevations (excluding the foundation) on a structure internal to the containment are specified in the draft regulatory guide. A second sensor internal to the containment is not provided because access to a sensor at a lower elevation is inconsistent with maintaining occupational radiation exposures as low as reasonably achievable (ALARA) and the containment seismic analyses show such a location to be unnecessary. The response of the containment internal structures is well represented by the response obtained at elevation 138′-0″. Two independent Category I structure foundations where the response is different from that of the containment structure are also specified. Since all seismic Category I structures are part of the nuclear island, which has a common basemat, no additional foundation sensors are required.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.2		Conforms	Should the seismic response at multiple units at the same site be evaluated as not essentially the same, multiple seismic monitoring systems will be installed at the units. If the seismic response is essentially the same at the other units, the system will be installed at only one unit; however annunciation will be provided in the main control room of each unit.
C.3		Conforms	
C.4		Conforms	The system power panel provides timing signals to components of the entire system. The triaxial acceleration sensor input signals exceeding a preset value are used as the actuation signal for system recording and analysis.
C.5		Conforms	
C.6		Conforms	The triaxial acceleration sensor input signals exceeding a preset value are used as the actuation signal for system recording and analysis.
C.7		Conforms	See Criteria Section C.2.
C.8		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for development of procedures.
Conformance for programmatic and/or operational aspects is documented below.			
C.3		Conforms	
C.8		Conforms	
Reg. Guide 1.13, Rev. 1, 12/75 and Rev. 2, 03/07 – Spent Fuel Storage Facility Design Basis			
Conformance of the design aspects with Revision 1 of the Regulatory Guide is as stated below in the DCD.			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
C.4	Regulatory Guide 1.25	Exception	The ventilation system is not designed to mitigate the consequences of a fuel handling accident.
C.5		Conforms	
C.6		Conforms	
C.7		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.8		Exception	Normal makeup supply (demineralized water) is not seismic Category I. Long-term post-accident supply piping is seismic Category I.
Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.7		Conforms	
Reg. Guide 1.14, Rev. 1, 8/75 – Reactor Coolant Pump Flywheel Integrity			
1.a	ASTM A.20	Exception	The flywheel is made of a bi-metallic design. Heavy alloy segments are fitted to a stainless steel hub and held in place by a retaining ring. Therefore, the specific guidelines in this section are not directly applicable to the AP1000.
1.b		Exception	Fracture toughness and tensile properties are checked for components that are required for structural integrity of the bi-metallic flywheel.
1.c		N/A	This guideline is not applicable to the flywheel assembly. Therefore, the guideline is not applicable to the AP1000 reactor coolant pump.
1.d		Conforms	The components of the flywheel that are relied upon for structural integrity require no welding.
2.a-b		Conforms	
2.c-e	ASME Code, Section III	Exception	The limits and methods of ASME Code, Section III, Paragraph F-1331.1(b), (replacement for Paragraph F-1323.1) are not directly applicable to the flywheel assembly. The calculated stress levels in the flywheel are evaluated against the ASME Code, Section III, Subsection NG stress limits used as guidelines and the recommended stress limits in Positions 4.a and 4.c of the Standard Review Plan 5.4.1.1. The stress evaluated is the maximum primary stress.
2.f		Exception	The calculated stress levels in the flywheel satisfy the ASME Code, Section III, Subsection NG stress limits used as guidelines and the recommended stress limits in Position 4.a of the Standard Review Plan 5.4.1.1. The stress evaluated is the maximum primary stress.
2.g		Conforms	
3		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
4.a	ASME Code, Section III, NB-2545 or NB-2546,NB-2540, NB-2530	Exception	The inspections and guidelines referenced in the regulatory guide were developed for steel flywheels in shaft seal pumps. The paragraphs of Subsection NB referenced in the regulatory guide apply only to forged and plate steel components. The bi-metallic flywheel design will be manufactured using multiple processes and materials. In accordance with the regulatory guide, each structural component of the bi-metallic flywheel will be inspected prior to final assembly according to its fabrication and the procedures outlined in Section III, NB-2500 of the ASME Code. Inspection of the flywheel assembly inside the sealed enclosure following a spin test is not practical.
4.b	ASME Code, Section XI	Exception	Inservice inspection of the flywheel assembly is not required to support safe operation of the reactor coolant pump. Planned, routine inspections of the flywheel assembly require considerable occupational radiation exposure and are not recommended. Inservice inspection of the flywheel assemblies requires extensive disassembly. Postulated missiles from the failure of the flywheel are contained within the stator shell and the pressure boundary is not breached. Vibration of the shaft due to a small flywheel fracture or leak in the enclosure does not result in stresses in the pressure boundary of sufficient magnitude to result in a break in the primary pressure boundary.
Reg. Guide 1.15 – Withdrawn			
Reg. Guide 1.16, Rev. 4, 8/75 – Reporting of Operating Information – Appendix A Technical Specifications			
General		N/A	Not applicable to AP1000 design certification. Reporting requirements associated with the technical specifications are identified in Tech Spec. Section 5.6. Subsection 1.1.1 defines the responsibility for finalizing the technical specification.
Reg. Guide 1.17 – Withdrawn			
Reg. Guide 1.18 – Withdrawn			
Reg. Guide 1.19 – Withdrawn			
Reg. Guide 1.20, Rev. 2, 5/76 and Rev. 3, 3/07 – Comprehensive Vibration Assessment Program For Reactor Internals During Preoperational and Initial Startup Testing			
Conformance with Revision 2 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.			
General		Conforms	The AP1000 internals are similar to those for a three-loop XL Westinghouse 17 x 17 robust fuel assembly core internals, a core shroud and the new incore instrumentation system. The upper internals are not significantly changed from standard designs. An internals vibration measurement program is conducted during hot functional testing. The results are evaluated based on pre-established allowable levels.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			During hot functional testing the AP1000 internals are subjected to operational system flow conditions that are similar to those imposed on previous 3XL three-loop designs. The duration of the hot functional flow testing is sufficient to subject each critical reactor internals component to at least 10^6 cycles of vibration. Pre- and post-test inspections are conducted to confirm that the AP1000 internals experience no excessive motion or wear.
C.1		Conforms	Although the AP1000 internals do not represent a first of a kind or unique design on the basis of the arrangement, design, size, or operating conditions, for the purposes of the reactor internals preoperational test program, the first operational AP1000 reactor vessel internals are classified as a prototype. Subsequent plants will be classified as the appropriate Non-prototype Category based on the designation of the first AP1000 as a Valid Prototype. See Subsections 3.9.2.3 and 3.9.2.4 for additional information on the vibration assessment of the reactor vessel internals.
C.2		Conforms	A comprehensive vibration assessment program will be developed for the first AP1000 reactor vessel internals. With regard to transients, data are acquired only during the hot functional test. Additionally, data are calculated over the ranges of hot functional test temperatures and during startup, shutdown, and steady-state operation of various combinations of reactor coolant pumps. Subsection 3.9.8 addresses information provided about the vibration assessment program.
C.3		Conforms	Subsequent to completion of the vibration assessment program for the first AP1000 reactor vessel internals, the vibration analysis program will address the criteria for the appropriate Non-prototype Category internals.
Reg. Guide 1.21, Rev. 1, 6/74 – Measuring Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	The design guidance of this regulatory guide for the selection of locations and type of effluent measurements to cover major or potentially significant pathways of release of radioactive materials during normal reactor operation, including anticipated operational occurrences, are incorporated in the plant design and in the requirements of the radiological effluent technical specifications. The calibration of effluent monitoring systems is performed according to written plant procedures. Subsection 11.5.8 defines the responsibility for the radiation monitoring program. Subsection 13.5.1 defines the responsibility for the plant procedure preparation.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1		N/A	Not applicable to AP1000 design certification. Subsection 11.5.8 defines the responsibility for the radiation monitoring program.
C.2		Conforms	
C.3-5		N/A	Not applicable to AP1000 design certification. Subsection 11.5.8 defines the responsibility for the radiation monitoring program.
C.6	ANSI N13.1-1969	Exception	ANSI N13.1-1969 prescribes the use of isokinetic sampling systems along with the use of multipoint sample nozzles for sampling from large ducts or vents. The use of alternate non-isokinetic sampling systems with a single point shrouded nozzle is also an acceptable method for obtaining representative samples from ducts or vents provided the flow is well mixed at the sample extraction point in the flow stream.
C.7-14		N/A	Not applicable to AP1000 design certification. Subsection 11.5.8 defines the responsibility for the radiation monitoring program.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.1		Conforms	
C.3-C.5		Conforms	
C.6	ANSI N13.1-1969	Exception	ANS N13.1-1969 prescribes the use of isokinetic sampling systems along with the use of multipoint sample nozzles for sampling from large ducts and vents. The use of alternate sampling systems such as a single point shrouded nozzle is also an acceptable method for obtaining representative samples from ducts and vents provided the flow is well mixed as the extraction point in the flow stream.
C.7-C.14		Conforms	
Reg. Guide 1.22, Rev. 0, 2/72 – Periodic Testing of Protection System Actuation Functions			
General		Conforms	Safety actuation circuitry is provided with a capability for testing with the reactor at power. The protection system, including the engineered safety features test cabinet design, conforms to this regulatory guide. The protection functions are tested at power to the greatest extent practical. Only the device function and/or system level function is not universally tested. The logic associated with the devices has the capability for testing at power, at the subsystem and/or component level.
D.1		Conforms	The AP1000 protection system is designed to permit periodic testing.
D.2-4		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.23, Second Proposed Rev. 1, 4/86			
General		Conforms	The onsite meteorological measurement program is site-specific and will be defined as indicated in Subsection 2.3.6. The number and location of meteorological instrument towers are determined by actual site parameters. Subsection 2.3.6 defines the responsibility for the onsite meteorological program. The data display and processing system has the capability to record the data from the meteorological instruments and display the information in the main control room.
Reg. Guide 1.24, Rev. 0, 3/72 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure			
General		N/A	This regulatory guide applies to the evaluation of a waste gas storage tank failure. The AP1000 design does not include waste gas storage tanks. Therefore, it is not applicable to the AP1000.
Reg. Guide 1.25, Rev. 0, 3/72 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors			
General		Exception	The guidance of Reg. Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors" will be followed instead of Reg. Guide 1.25.
Reg. Guide 1.26, Rev. 3, 2/76 and Rev. 4, 3/07 – Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containment Components of Nuclear Power Plants			
Conformance with Revision 3 of the Regulatory Guide for DCD scope of work is as stated below in the DCD.			
C.1		Exception	A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(z). The alternate classification is discussed in Section 5.2.
C.1.a		Exception	For the AP1000 plant design, Quality Group B is reserved for the containment boundary including any extensions such as containment isolation valves and associated piping. Quality Group C is essentially equivalent quality except that it has less stringent ISI. For equipment such as passive safety system accumulators, minor leakage is not a problem for the following reasons: a. It is located inside containment so activity releases are contained. b. Minor leakage does not affect its functional performance, especially considering the limited duration of post-accident operation. c. There is continuous water level and gas pressure monitoring of the passive safety system accumulators that detects leaks.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			This approach results in the change of quality group (from Quality Group B to Quality Group C) for various components such as the IRWST. Portions of systems that provide emergency core cooling system functions and are constructed using ASME Code, Section III, Class 3 criteria have the additional requirement that radiography will be conducted on a random sample of welds during construction, see Subsection 3.2.2.5.
C.1.b		Exception	The AP1000 residual heat removal system is a nonsafety-related system, but it is classified as Quality Group C. The passive core cooling system provides the safety-related function that the residual heat removal system provides in current plants with active safety-related systems.
C.1.c		N/A	Applies to boiling water reactors only.
C.1.d		Conforms	Portions of the feedwater and steam systems are Quality Group B, up to the isolation valves.
C.1.e		Conforms	
C.2.a		Conforms	The component cooling water and the service water systems are Quality Group D since they perform no safety-related functions.
C.2.b		Conforms	Component cooling water is not required for safe shutdown of the AP1000. The reactor cooling pumps do not have seals and do not require seal water supply.
C.2.c		Conforms	
C.2.d		N/A	Regulatory Guide 1.143 supersedes this guideline.
C.2.e		N/A	Regulatory Guide 1.143 supersedes this guideline.
C.3		Exception	Systems that are normally radioactive are classified as Quality Group D. AP1000 also classifies as Quality Group D, nonsafety-related systems and components which have functions that have been identified as important as part of the implementation of the regulatory treatment of nonsafety-related systems or as defense-in-depth systems. Some structures, systems, and components that have the potential to be contaminated with radioactive fluids but normally do not contain radioactive fluids are not classified as Quality Group D.
Conformance with Revision 4 of this Regulatory Guide for remaining scope is documented below.			
General		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.27, Rev. 2, 1/76 – Ultimate Heat Sink for Nuclear Power Plants			
C.1		Conforms	The passive containment cooling system water storage tank is sized to provide water cooling to the containment vessel and provide heat removal to meet the requirements of General Design Criterion 38 to reduce and maintain the containment temperature and pressure following a postulated loss-of-coolant accident for 3 days following passive containment cooling system actuation. This water delivery is done in conjunction with the flow of air over the containment shell to provide the containment cooling. After 3 days of water delivery from the PCCWST, the PCS cooling water supply is continued through either: • simple operator action via installed safety-related piping and connection utilizing offsite or available onsite supplies of water and an offsite pump to resupply water to the tank; or, • simple operator action utilizing onsite seismically analyzed pumps, piping and 4 days of water inventory within the passive containment cooling ancillary water storage tank to resupply the PCCWST. Supplemental supplies would then be available from either onsite storage facilities or an offsite source. Since the passive containment cooling system can function with replenished water supplies from either onsite or offsite, the system meets the guideline of providing an ultimate heat sink for more than 30 days.
C.2		Conforms	The AP1000 design conforms to this regulatory position, provided that the definition of a single failure of a man-made structure does not include the safety-related, seismically-designed containment structure assembly. The AP1000 uses the atmosphere as the ultimate heat sink. A baffle located between the containment shell and the shield building sustains the natural circulation that provides for air flow over the containment shell to carry heat away. The baffle is composed of a large number of panels and will continue to function if damaged by an external missile.
C.3		Conforms	The seismically-designed passive containment cooling system water storage, integral to the containment structure meets this regulatory position.
C.4		Conforms	Technical Specifications exist for the passive containment cooling system and air inlet and air baffle.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.28, Rev. 3, 8/85 – Quality Assurance Program Requirements (Design and Construction)			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI/ASME N45.2-1977 ANSI/ASME NQA-1-1983 through NQA-1a-1983 Addenda	Exception	The Westinghouse quality assurance program is described in Chapter 17. Refer to "Westinghouse Electric Company Quality Management System" (QMS) referenced therein for Westinghouse positions on regulatory guides within the scope of the quality assurance program. In some cases current industry consensus standards have replaced the standards specifically referenced by certain regulatory guides. In particular, the N45.2 series standards have been replaced by ASME NQA-1. Therefore, the "Quality Management System" may reference ASME NQA-1 rather than the N45.2 series standards when describing the Westinghouse position. Work performed under QMS, Rev. 5 and Rev. 6, is based on application of Regulatory Guide 1.28, Revision 3, and ASME NQA-1-1994. Work performed under QMS, Rev. 7, is based on application of Regulatory Guide 1.28, Revision 4, and ASME NQA-1-2008 and NQA-1a-2009. Application of Regulatory Guide 1.28, Revision 4, is addressed in Appendix A of the Westinghouse QMS, Revision 6.
2.	Criteria 17 10 CFR 50 Appendix B	Conforms	
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.29, Rev. 3, 9/78 and Rev. 4 3/07 – Seismic Design Classification			
Conformance with Revision 3 of the Regulatory Guide for DCD scope of work is as stated below in the DCD.			
C.1.a		Conforms	
C.1.b		Conforms	
C.1.c		Conforms	
C.1.d		Exception	The AP1000 normal residual heat removal system is nonsafety-related. The safety-related function of decay heat removal is provided by the safety-related passive residual heat removal heat exchanger of the passive core cooling system that is seismic Category I. The spent fuel pool cooling system does not have active components that are required for the safety-related decay heat removal function. This function is provided passively through a large heat sink of water in the pool. The spent fuel pool is sized to keep the fuel covered for at least 72 hours without active cooling or makeup following a loss of ac power sources.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			The 72-hour sizing calculation accounts for the maximum loss of water due to the rupture of non-seismic piping. seismic Category I components within the spent fuel pool cooling system include the containment penetration, the connections for makeup, and the spent fuel pool.
C.1.e		N/A	Applies to boiling water reactors only.
C.1.f		Conforms	
C.1.g		Exception	The AP1000 does not have a safety-related auxiliary feedwater system. The safety-related decay heat removal function is provided by the passive residual heat removal heat exchanger. The safety-related functions of the essential service water system are provided by the passive residual heat removal heat exchangers and the passive containment cooling system. The component cooling system is a nonsafety-related system, since it performs no safety-related functions.
C.1.h		Conforms	
C.1.i		N/A	The diesel-generators are nonsafety-related. Therefore, this section is not applicable to the AP1000.
C.1.j		Conforms	
C.1.k		Conforms	
C.1.l		Conforms	
C.1.m		Conforms	
C.1.n		Exception	Structures or equipment whose failure results in incapacitating injury to the occupants of the main control room are classified as seismic Category II and covered under Position 2 of this regulatory guide.
C.1.o		Conforms	
C.1.p		Conforms	
C.1.q		Conforms	
C.2		Conforms	
C.3		Conforms	
C.4		Conforms	
Conformance with Revision 3 of this Regulatory Guide for the site-specific scope is documented below.			
General		Conforms	
Conformance with Revision 4 of this Regulatory Guide for remaining scope is documented below.			
C.4		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.30, Rev. 0, 8/72 – Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI/ASME N45.2.4-1972	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the Quality Assurance program.
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.31, Rev. 3, 4/78 – Control of Ferrite Content in Stainless Steel Weld Metal			
General		Conforms	
C.1-5		Conforms	
Reg. Guide 1.32, Rev. 2, 2/77 – Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants			
Regulatory Guide 1.32, Rev. 3, 03/04 – Criteria for Power Systems for Nuclear Power Plants			
Conformance of the design aspects with Revision 2 of the Regulatory Guide is as stated below in the DCD.			
1.	IEEE Std. 308-1974	Exception	Regulatory Guide 1.32 endorses IEEE Std. 308-1974 (Reference 5) which has been superseded by IEEE Std. 308-1991(Reference 6). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.32. The guidelines are applicable to the Class 1E dc and UPS system only. There are no safety-related ac power systems in the AP1000.
1.a	Regulatory Guide 1.93	N/A	The AP1000 has no safety-related ac power system. Therefore, the guidelines specified in this criterion section recommending the availability of offsite power "within a few seconds" are not applicable.
1.b	IEEE Std. 308-1974, Section 5.3.4	Exception	See comment on Criterion Section 1.
1.c	IEEE Std. 450-1975	Exception	Battery performance discharge test is per IEEE-Std 450-1995 as described in Bases for Technical Specification, Surveillance Requirement 3.8.7.6. The AP1000 has adopted a testing interval of 24 months that aligns with industry practice.
1.d	Regulatory Guide 1.6 Regulatory Guide 1.75	Exception	The guidelines are applicable to the Class 1E dc and UPS system only. There are no safety-related ac power systems in the AP1000.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
1.e	Regulatory Guide 1.75	Exception	The guidelines are applicable to the Class 1E dc and UPS system only. There are no safety-related ac power systems in the AP1000.
1.f	Regulatory Guide 1.9	N/A	Guidelines apply to Class 1E diesel generators. Therefore, they are not applicable to the AP1000.
2.a	IEEE Std. 308-1974, Section 8.2, 8.3.1; Regulatory Guide 1.81	N/A	The AP1000 is a single-unit plant. This criterion is not applicable to the AP1000. When two or more AP1000s are located adjacent, electrical systems are not shared.
2.b.	Regulatory Guide 1.93	Exception	The guidelines are applicable to the Class 1E dc and UPS system only. There are no safety-related ac power systems in the AP1000. See comments on Regulatory Guide 1.93.
Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.33, Rev. 2, 2/78 – Quality Assurance Program Requirements (Operation)			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N18.7-1976 ANS-3.2	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the Quality Assurance program. Regulatory Guide 1.33 is used in a specific manner for determining documentation adequacy in regard to the ongoing qualification method based on the assumption the utility programs are in conformance with Regulatory Guide 1.33.
Conformance for remaining scope is documented below.			
General	ANSI N18.7-1976 ANS-3.2	N/A	QA program requirements based on NEI 06-14A, Rev 7 as noted in Section 17.5.
Reg. Guide 1.34, Rev. 0, 12/72 – Control of Electroslag Weld Properties			
General	ASME Code, Sections III and IX	Conforms	The AP1000 prohibits the use of electroslag welding on reactor coolant pressure boundary components. AP1000 safety-related components that use electroslag welding conform to the provisions of the ASME Code and this regulatory guide.
C.1-5		Conforms	
Reg. Guide 1.35, Rev. 3, 7/90 – Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments			
General		N/A	The AP1000 does not have a concrete containment and does not use a prestressing tendon in the containment structure. Therefore, this regulatory guide is not applicable to the AP1000.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.35.1, Rev. 0, 7/90 – Determining Prestressing Forces for Inspection of Prestressed Concrete Containments			
General		N/A	The AP1000 does not have a concrete containment and does not use a prestressing tendon in the containment structure. Therefore, this regulatory guide is not applicable to the AP1000.
Reg. Guide 1.36, Rev. 0, 2/73 – Nonmetallic Thermal Insulation for Austenitic Stainless Steel			
General		Conforms	
C.1		Conforms	
C.2.a	ASTM C692-71 RDT M12-1T	Conforms	
C.2.b		Conforms	
C.3-4		Conforms	
Reg. Guide 1.37, Rev. 0, 3/73 and Rev. 1, 3/07 – Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants			
Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated below in the DCD.			
General	ANSI N45.2.1-1973	Exception	The ANSI N45.2 series of standards that are referenced by the current revisions of the Quality Assurance regulatory guides have been replaced by ASME NQA-1. ANSI N45.2.1, which is referenced in Regulatory Guide 1.37, has been incorporated into NQA-1 Subpart 2.1. The technical requirements specified in ANSI N45.2.1 and NQA-1 Part II, Subpart 2.1 are compatible. Therefore, compliance with NQA-1 Part II, Subpart 2.1 satisfies Regulatory Guide 1.37. Section 17.5 defines the responsibility for the Quality Assurance program.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.38, Rev. 2, 5/77 – Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N45.2.2-1972	Exception	The ANSI N45.2 series of standards that are referenced by the current revisions of the Quality Assurance regulatory guides have been replaced by ASME NQA-1. Refer to the Regulatory Guide 1.28 position. Section 17.5 defines the responsibility for the Quality Assurance program.
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.39, Rev. 2, 9/77 – Housekeeping Requirements for Water-Cooled Nuclear Power Plants			
Conformance for DCD scope of work is as stated below in the DCD.			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	ANSI N45.2.3-1973	Exception	The ANSI N45.2 series of standards that are referenced by the current revisions of the Quality Assurance regulatory guides have been replaced by ASME NQA-1. Refer to the Regulatory Guide 1.28 position. Section 17.5 defines the responsibility for the Quality Assurance program.
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.40, Rev. 0, 3/73 – Qualification Tests of Continuous-Duty, Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants			
General	IEEE Std. 334-1971	N/A	The AP1000 does not have continuous-duty safety-related motors installed inside the containment. Therefore, the regulatory guide is not applicable to the AP1000.
Reg. Guide 1.41, Rev. 0, 3/73 – Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments			
General		Exception	The guidelines are followed for Class 1E dc power systems during the preoperational testing of AP1000 redundant onsite electric power systems to verify proper load group assignments, except as follows. Complete preoperational testing of the startup, sequence loading, and functional performance of the load groups is performed where practical. In those cases where it is not practical to perform complete functional performance testing, an evaluation is used to supplement the testing.
Reg. Guide 1.42 – Withdrawn			
Reg. Guide 1.43, Rev. 0, 5/73 – Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components			
General		Conforms	The guidelines of this regulatory guide are followed during the welding process used for cladding ferritic steel components of the AP1000 with austenitic stainless steel. No qualifications are provided for by this regulatory guide for ASME SA-533 material and equivalent chemistry for forging grade ASME SA-508, Class 3, material. The reactor vessel, steam generator channel heads, accumulators, and core makeup tanks design specification restricts the low alloy steel forging material to ASME SA-508, Class 3, which is made to a fine grain practice only. Cladding of ASME SA-508, Class 2 is not applicable to the AP1000 design. The fabricator monitors and records the weld parameters to verify agreement with the parameters established by the procedure qualification as stated in Regulatory Position C.3.
C.1-3		N/A	The AP1000 material, specifically ASME SA-533 and SA-508 Class 3 made to a fine grain practice, is not subjected to the controls in this regulatory guide.
Reg. Guide 1.44, Rev. 0, 5/73 – Control of the Use of Sensitized Stainless Steel			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with Revision 0 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
C.4		Exception	Material exposed to reactor coolant has a controlled concentration of less than 0.10 ppm dissolved oxygen at all temperatures above 250°F during normal operation.
C.5		Conforms	
C.6		Conforms	
Reg. Guide 1.45, Rev. 0, 5/73 – Reactor Coolant Pressure Boundary Leakage Detection Systems			
Conformance of the design aspects is as stated below in the DCD.			
C.1		Conforms	
C.2		Conforms	
C.3		Exception	The AP1000 uses two methods for leak detection, including sump level and flow monitoring, and airborne particulate radioactivity monitoring.
C.4		Conforms	
C.5		Conforms	
C.6		Exception	Seismically qualified tubing from containment to the containment atmosphere particulate monitor is not provided. Leak detection is provided by the seismic Category I containment sump level monitor leakage detection system.
C.7		Conforms	
C.8		Conforms	
C.9		Conforms	
Conformance with programmatic and/or operational aspects is documented below.			
C.7		Conforms	
Reg. Guide 1.46 – Withdrawn			
Reg. Guide 1.47, Rev. 0, 5/73 – Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems			
General	IEEE Std. 279-1971, Section 4.13; and 10 CFR 50 App. B, Criterion XIV	Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1-4		Conforms	
Reg. Guide 1.48 – Withdrawn			
Reg. Guide 1.49 – Withdrawn			
Reg. Guide 1.50, Rev. 0, 5/73 – Control of Preheat Temperature for Welding of Low-Alloy Steel			
General	ASME Code, Sections III and IX	N/A	The guidelines of this regulatory guide are followed during the initial fabrication of low-alloy steel components of the AP1000. This regulatory guide is considered as applicable to ASME Code, Section III, Class 1 components. The AP1000 practice for Class 1 components is in agreement with the guidance of this regulatory guide except for Regulatory Positions C.1(b) and 2. For AP1000 Class 2 and 3 components, the guidelines provided by this regulatory guide are not applied, however all requirements of the ASME Boiler and Pressure Vessel Code are imposed.
C.1(b)		Exception	The welding procedures are qualified within the preheat temperature ranges required by ASME Code, Section IX. Experience has shown excellent quality of welds using the ASME qualification procedures.
C.2		Exception	The AP1000 position is that the guidance specified in this regulatory guide is both unnecessary and impractical. Code acceptable low-alloy steel welds have been and are being made under present procedures. It is not necessary to maintain the preheat temperature until a post-weld heat treatment has been performed in accordance with the guidance provided by this regulatory guide, in the case of large components. In some cases of reactor vessel main structural welds, the practice of maintaining preheat until the intermediate or final post-weld heat treatment has been followed. In other cases, an extended preheat practice has been utilized in accordance with the reactor vessel design specification. In this practice, the weld temperature is maintained at 400°F to 750°F for 4 hours after welding. The weld temperature may then be lowered to ambient without performing an intermediate or final post-weld heat treatment at 1100°F. The welds have shown high integrity. Westinghouse practices are documented in WCAP-8577 (Reference 9) which has been accepted by the Nuclear Regulatory Commission.
Reg. Guide 1.51 – Withdrawn			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Regulatory Guide 1.52, Rev. 3, 6/01 – Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants			
Conformance with the design and operational aspects is as stated below in the DCD.			
General		Conforms	The AP1000 main control room emergency habitability system (VES) includes a passive filtration system that is contained entirely within the main control room envelope. The passive filtration portion of the AP1000 VES contains no active equipment. This regulatory guide is followed where appropriate for the AP1000 MCR passive filtration system.
C.1		Conforms	The performance, design, construction, acceptance testing, and QA of the AP1000 MCR passive filtration system will be in accordance with ASME AG-1-1997 where appropriate. Also, it will be designed to ASME N509-1989 and tested to ASME N510-1989.
C.2		Conforms	The AP1000 MCR passive filtration is designed to operate under the environmental conditions specified by the guidelines in this regulatory position.
C.3.1 – 3.2		Exception	Redundant filtration trains are not needed in the AP1000 passive filtration system. The passive filtration system contains no active components or moving parts. Adequate margin is provided in the design to prevent the likelihood of a passive failure.
C.3.3		Conforms	The maximum pressure surge is experienced during the highly unlikely failure of the upstream pressure regulator valve in the VES. The passive filtration system is designed to operate under this pressure surge.
C.3.4		Conforms	All components in the MCR passive filtration system are designed to be seismic Category I.
C.3.5		Conforms	The mechanical design of the MCR passive filtration system is such that it can withstand the buildup of postulated radioactivity levels and maintain functionality.
C.3.6		Conforms	The AP1000 passive filtration system volumetric air-flow rate is below 30,000 CFM.
C.3.7		Exception	Temporary differential pressure gauges are provided to perform the initial and the in-service testing in compliance with the Ventilation Filter Testing Program of Technical Specification 5.5.13. During system operation, air flow through the passive filtration line, in lieu of pressure drop across the filters, provides indication of satisfactory flow through the filtration unit.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			The flow instrumentation is not required to be safety related. The instrumentation does not perform a safety function. The instrumentation monitoring flow from the VES emergency air storage tanks indicates whether there is sufficient flow coming from the compressed air tanks to induce the passive filtration.
C.3.8		Exception	The AP1000 passive filtration system does not rely upon, use, or need a power supply and/or electrical distribution.
C.3.9		Conforms	The AP1000 passive filtration system is designed to operate for 72 hours following a DBA.
C.3.10		Conforms	
C.3.11		Exception	There are no outdoor air intakes for the AP1000 passive filtration system. The system uses breathable compressed air that is stored in compressed air tanks during the post-72-hour operation time.
C.3.12		Exception	The AP1000 passive filtration system is located completely within the CRE. Leakages as explained in this regulatory position are not applicable to this system.
C.4.1 – 4.2		Exception	There are no moisture separators and/or heaters in the AP1000 passive filtration line.
C.4.3		Exception	ASME AG-1, 1997 Section FB requires filters be certified to UL 900, Class 1, 5th Edition 1987. As of May 2012 UL no longer classifies UL 900 filters with a Class 1 or Class 2 rating. The Class 1 requirements per UL 900, “Air Filter Units” 8th Edition, are obsolete and deleted, as UL 900 no longer distinguishes between Class 1 and Class 2 filters.
C.4.4		Exception	ASME AG-1, 1997 Section FC require HEPA Filters be listed to UL 586, 7th Edition. Manufacturers only list components per the latest Edition of the UL Standard. Previous UL Standard Editions/Revisions cannot be invoked.
C.4.5 - 4.7		Conforms	
C.4.8		Exception	There are no water drains in the AP1000 passive filtration line.
C.4.9		Conforms	The credited adsorber efficiencies are 90% for elemental iodine and 90% for organic iodine. These efficiencies assume no humidity control.
C.4.10		Conforms	Type II adsorbers are used in this application
C.4.11		Conforms	The AP1000 passive filtration line uses impregnated activated carbon as the absorbent. The absorber is designed for a minimum average atmosphere residence time of 0.25 seconds per 2 inches of absorbent bed.
C.4.12		Conforms	
C.4.13		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.4.14		Exception	The passive filtration line requires no fans.
C.5.1		N/A	Only one bank of filters is used.
C5.2		Exception	This system is not used for normal HVAC, and the filters should not build up unusual levels of particulate once installed.
C.6.1		Conforms	
C.6.2 – 6.5		Conforms	
C.7		Conforms	
Table 1		Conforms	The Technical Specification methyl iodide penetration acceptance limit for the AP1000 activated carbon adsorber is 5%, which correlates to 90% removal efficiency of both organic and elemental iodine. The calculated design basis for the AP1000 passive filtration adsorbers assumes a 90% organic iodine removal efficiency and a 90% elemental iodine efficiency. A 1% bypass leakage is accounted for by testing to increased organic iodine removal efficiency.

Reg. Guide 1.53, Rev. 0, 6/73 – Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems

Regulatory Guide 1.53, Rev. 2, 11/03 – Application of the Single-Failure Criterion to Safety Systems

Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated below in the DCD.This guidance is completely within the scope of the DCD.

General	IEEE Std. 379-1972	Exception	Regulatory Guide 1.53 endorses IEEE Std. 379-72 (Reference 10), which has been superseded by IEEE Std. 379-2000 (Reference 11). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.53. IEEE Std. 379-2000 is endorsed by DG-1118 (Proposed Revision of Regulatory Guide 1.53). The guidelines are applicable to safety-related dc power systems. There are no safety-related ac power sources in the AP1000.

Reg. Guide 1.54, Rev. 1, 7/00 and Rev. 2, 10/10 – Service Level I, II and III Protective Coatings Applied to Nuclear Power Plants

Conformance with Revision 1 of this regulatory guide is as stated below.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	ASTM D 3843-00, ASTM D 3911-95, ASTM D 5144-00	Exception	Some coatings inside containment are nonsafety-related and satisfy appropriate ASTM Standards. See Subsection 6.1.2 for additional information, e.g., delamination of Service Level II coatings during design basis accident testing is acceptable if the Service Level II coating meets the criteria identified in Subsection 6.1.2.1.1. Application is controlled by procedures using qualified personnel to provide a high quality product. The paint materials for coatings inside the containment are subject to 10 CFR Part 50 Appendix B Quality Assurance requirements. The quality assurance features of the AP1000 coatings systems are outlined in Subsection 6.1.2.1.6. Subsection 6.1.3 defines the responsibility for the coating program.
General		Exception	Revision 2 of the regulatory guide is applied to use for protective coatings on the containment vessel shell and attachments to the containment vessel shell.
C.4	ASTM D 5962-96	Exception	Unqualified coatings inside containment are managed consistent with the guidance of ASTM D 7491-08, “Standard Guide for Management of Non-Conforming Coatings in Coating Service Level I Areas of Nuclear Power Plants.” Regulatory Guide 1.54 Revision 2 endorses this standard.
Conformance with Revision 2 of this regulatory guide is as stated below.			
General	ASTM D 3843-00, ASTM D 3911-08, ASTM D 5144-08	Exception	Some coatings inside containment are nonsafety-related and satisfy appropriate ASTM Standards. See Subsection 6.1.2 for additional information. Application is controlled by procedures using qualified personnel to provide a high quality product. The paint materials for coatings inside the containment are subject to 10 CFR Part 50 Appendix B Quality Assurance requirements. The quality assurance features of the AP1000 coatings systems are outlined in Subsection 6.1.2.1.6. Subsection 6.1.3 defines the responsibility for the coating program.
General		Exception	Revision 2 of this regulatory guide is applied to use only for management of nonconforming coatings within Service Level I areas and protective coatings on the containment vessel shell and attachments to the containment vessel shell. Other protective coatings use Revision 1 of the regulatory guide.
General	ASTM D 3911-08	Exception	Regulatory Guide 1.54 provides additional acceptance criteria for Service Level I coatings when tested per ASTM D 3911-08. Of the additional acceptance criteria, a) Peeling and delamination shall not be permitted and b) Cracking is not considered a failure unless it is accompanied by delamination or loss of adhesion. These are not applicable to a Service Level I epoxy top coat applied over inorganic zinc. This is because delamination and/or detachment of epoxy coatings does not impact plant safety and does not adversely affect underlying inorganic zinc coating or the ability of the inorganic zinc coating to perform its safety functions.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with programmatic and/or operational aspects of Revision 1 is documented below. Revision 2 is not utilized for the programmatic and/or operational aspects except as noted below for management of unqualified coatings.			
C.4	ASTM D 5962-96	Exception	Programmatic and/or operational aspects regarding management of nonconforming coatings in Service Level I areas comply with ASTM D 7491-08 endorsed in Revision 2 of this regulatory guide.
Reg. Guide 1.55 – Withdrawn			
Reg. Guide 1.56, Rev. 1, 7/78 – Maintenance of Water Purity in Boiling Water Reactors			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.57, Rev. 0, 6/73 and Rev. 1, 03/07 – Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components			
Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.			
General	ASME Code, Section III	Exception	The regulatory guide was issued in 1973. It refers to the ASME Code through the Summer 1973 Addenda. The acceptance criteria have been defined in greater detail in SRP 3.8.2. The AP1000 complies with the SRP acceptance criteria with the exception that the operating basis earthquake is excluded.
Reg. Guide 1.58 – Withdrawn			
Reg. Guide 1.59, Rev. 2, 8/77 – Design Basis Floods for Nuclear Power Plants			
Conformance with the design aspects is as stated below in the DCD.			
C.1-4	Regulatory Guide 1.29	N/A	The maximum water level due to the probable maximum flood is established as a site interface in Chapter 2 and is used in the design of the AP1000.
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	Section 2.4 and Section 3.4 provide the site-specific information on historical flooding and potential flooding factors.
Reg. Guide 1.60, Rev. 1, 12/73 – Design Response Spectra for Seismic Design of Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
C.1		Conforms	
C.2		Conforms	
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms with Clarification	Site-specific response spectra is derived in accordance with 10 CFR Part 100 Subpart B 100.23. The standard spectral shape of Regulatory Guide is not used.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.61, Rev. 0, 10/73 and Rev. 1, 3/07 – Damping Values for Seismic Design of Nuclear Power Plants			
Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.			
General		Conforms	Damping values used in the AP1000 safe shutdown earthquake analyses are shown in Table 3.7.1-1. These values are based on Regulatory Guide 1.61, on the recommendations of ASCE 4-86 (Reference 12), and on values used and accepted on past projects (Reference 13). The values are conservative relative to realistic damping values reported in the literature (Reference 14) A site interface is established to verify that the site is within the range considered in the design.
Reg. Guide 1.62, Rev. 0, 10/73 – Manual Initiation of Protective Actions			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
C.4		Conforms	
C.5		Conforms	
C.6	IEEE Std. 279-1971, Section 4.16	Conforms	
Reg. Guide 1.63, Rev. 3, 2/87 – Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants			
General	IEEE Std. 317-1983 IEEE Std. 741-1986, Section 5.4	Exception	Regulatory Guide 1.63 endorses IEEE Std. 741-1986 (Reference 15), which has been superseded by IEEE Std. 741-1997 (Reference 16). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.63. Electric penetration assemblies are in conformance with IEEE Std. 317-1983 (Reference 17), Reference 16, and this regulatory guide with the clarification discussed below. The majority of low voltage control circuits are self-limiting in that circuit resistance limits the fault current to a level that does not damage the penetration. Where, on a case-by-case basis, a circuit is found not to be self-limiting, primary and backup breaker or fuse coordination or the addition of a subfeed over current protection as in the case of motor control centers, provide for safe operation. The energy levels in the instrument systems are such that damage cannot occur to the containment penetration.
Reg. Guide 1.64 – Withdrawn			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.65, Rev. 0, 10/73 – Materials and Inspections for Reactor Vessel Closure Studs			
Conformance of the design aspects is as stated below in the DCD.			
C.1.a	ASME Code, Section III, Subsection NB	Conforms	The reactor vessel closure stud bolting material is pro-cured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. The material meets the criteria of Appendix G to 10 CFR 50. The reactor vessel design specification requires the maximum tensile strength of 170,000 psi for the closure stud material.
C.1.b		Conforms	
C.2	ASME Code, Section III, NB-2580	Conforms	The guidelines of this regulatory guide are followed during the fabrication of the stud bolts and nuts.
C.3		Conforms	The guidance of this regulatory guide is followed during the venting and filling of the AP1000 pressure vessel.
C.4	ASME Code, Section XI; ASME Code, Section III, NB-2545 or NB-2546	Conforms	The guidelines of this regulatory guide are followed during the inservice examination of the AP1000 pressure vessel stud bolting.
Conformance with programmatic and/or operational aspects is documented below.			
C.3		Conforms	
C.4		Exception	ASME XI ISI criteria for reactor vessel closure stud examinations are applied in lieu of the ASME III NB 2545 and NB 2546 surface examinations. The volumetric examinations currently required by ASME XI provide improved (since 1973) detection of bolting degradation.
Reg. Guide 1.66 – Withdrawn			
Reg. Guide 1.67 – Withdrawn			
Reg. Guide 1.68, Rev. 2, 8/78 and Rev. 3, 3/07 – Initial Test Program for Water-Cooled Nuclear Power Plants			
Conformance with Revision 2 of the Regulatory Guide is documented below in the DCD. Conformance of the design aspects is as stated below in the DCD.			
C.1	App. A.1.a	Conforms	Applies to AP1000 reactor coolant system components. (Pressurizer power-operated relief valves and reactor vessel internal vent valves are not design features of the AP1000. Jet pumps are applicable to boiling water reactors only.)
	App. A.1.b	Conforms	Applies to the AP1000 reactivity control system, except the systems for boiling water reactors such as rod worth minimizers. Standby liquid control system is not a design feature of the AP1000.
	App. A.1.c	Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
	App. A.1.d	Conforms	The functions of these systems are replaced by the passive residual heat removal heat exchanger of the passive core cooling system. Reactor core isolation cooling system is not a design feature of the AP1000.
	App. A.1.e	Conforms	
	App. A.1.f	Conforms	
	App. A.1.g	Conforms	
	App. A.1.h	Conforms	The characteristics of the AP1000 passive safety systems allow the support systems such as the cooling water systems, the heating, ventilating, and air conditioning and the ac power sources to be nonsafety-related and simplified. The capability of these systems is established by testing. Air flow resistance of the passive containment cooling system is verified by scale testing and analysis. Cold water interlocks are not a design feature of the AP1000.
	App. A.1.i	Conforms	The AP1000 has no secondary containment. Therefore, this guideline applies only to primary containment. The following systems or functions are not design features of the AP1000 and are therefore not tested: • Containment supplementary leak collection • Standby gas treatment • Secondary containment system • Containment annulus and cleanup • Bypass leakage tests on pressure suppression • Ice condenser systems • Containment penetration cooling
	App. A.1.j	Conforms	Recirculation flow control, traversing incore probes, automatic dispatching control systems and hotwell level control are not design features of the AP1000.
	App. A.1.k	Conforms	
	App. A.1.l	Conforms	Condenser off gas systems are not a design feature of the AP1000.
	App. A.1.m	Conforms	
	App. A.1.n	Conforms	Seal water, boron recovery, shield cooling, refueling water storage tank heating, and equipment for establishing and maintaining subatmospheric pressures are not design features of the AP1000.
	App. A.1.o	Conforms	
	App. A.2	Conforms	As applicable for pressurized water reactor.
	App. A.3	Conforms	As applicable for pressurized water reactor.
	App. A.4	Conforms	As applicable for pressurized water reactor.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			Compliance with A.4.t is met for the AP1000 with the provisions to perform the pre-operational tests of the passive RHR heat exchanger, as well as the tests described in DCD test abstracts 14.2.10.3.6, “Natural Circulation (First Plant Only)” and 14.2.10.4.29, “Passive Residual Heat Removal Heat Exchanger (First Plant Only).” Natural circulation testing of the reactor coolant system will be performed using the steam generators and the PRHR for the first plant only, in conformance with the AP1000 position on TMI item I.G.1 as outlined in Subsection 1.9.4.2.1.
	App. A.5	Conforms	
C.2 through C.9		N/A	Section 14.2 describes the AP1000 plant initial test program. Section 14.4 describes the responsibilities required to perform the AP1000 plant initial test program.
General	Appendix B	N/A	Section 14.2 describes the AP1000 plant initial test program. Section 14.4 describes the responsibilities required to perform the AP1000 plant initial test program.
General	Appendix C	N/A	Section 14.2 describes the AP1000 plant initial test program. Section 14.4 describes the responsibilities required to perform the AP1000 plant initial test program.
Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.2-C.9 Appendix B Appendix C		Conforms	
		Exception	Exception to the statement in C.9 regarding "...the startup report, as discussed in RG 1.16." This statement is interpreted to mean the report is submitted to NRC on the docket as discussed in (Withdrawn) RG 1.16. The startup report will be completed on the schedule and include the content information as described in RG 1.68 Item C.9 (a) - (e), however the report will be retained onsite rather than submitted to NRC on the docket as discussed in (Withdrawn) RG 1.16.
Reg. Guide 1.68.1, Rev. 1, 1/77 – Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.68.2, Rev. 1, 7/78 – Initial Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants			
General		Conforms	
Reg. Guide 1.68.3, (Task RS 709-4), 4/82 – Preoperational Testing of Instrument and Control Air Systems			
General	Regulatory Guide 1.68	Exception	
C.6		Exception	Regulatory Guide 1.68.3 endorses ANSI/ISA Standard S7.3-1975. The AP1000 references ANSI/ISA S7.3-1981.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.7		N/A	The instrument and control air systems (CAS instrument air and service air subsystems) does not include redundant components or air supplies to meet the single failure criterion for any safety functions except for the containment isolation function which is tested outside the scope of this regulatory guide.
C.8		Exception	Testing for complete loss of system pressure, both sudden and gradual, and to partial reductions in system pressure is only performed on safety related air operated valves. Testing is performed on a component level, not by branches.
C.9		N/A	Service air is specifically designed as a backup for Instrument Air. The compressors, dryers, filters, receivers and interconnecting piping are designed to meet the same air quality standards as the instrument air system.
C.11		N/A	There is no credible failure affecting the performance of safety-related valves to perform their safety-function due to over pressurization of the CAS air supply as by design, the air operated valves required to perform a safety-related function either • have a pressure regulator (to protect the end-user valve) designed to withstand an input pressure greater than the CAS safety valve set point or • when a pressure regulator is not used to regulate the CAS air pressure to the air operated valve, the air operated valve itself is designed to withstand an input pressure greater than the CAS safety valve set point.
C.1-C.5, C.10		Conforms	
Reg. Guide 1.69, Rev. 0, 12/73 – Concrete Radiation Shields for Nuclear Power Plants			
General	ANSI N101.6-1972	Exception	Regulatory Guide 1.69 endorses ANSI N101.6-1972 (Reference 18), which has been superseded by ANSI/ANS 6.4 1997 (Reference 19) and ACI 349-R01 (Reference 44). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.69.
Reg. Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition), Rev. 3, 11/78			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	
Conformance with programmatic and/or operational aspects is documented below.			
General		Exception	The format and content of the FSAR follow Regulatory Guide 1.206 and AP 1000 Design Control Document as required by Appendix D of 10 CFR Part 52.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			Regulatory Guide 1.70 requires the security plan to be submitted as a separate document. The security plan will be submitted with the COL.
Reg. Guide 1.71, Rev. 0, 12/73 and Rev. 1, 3/07 – Welder Qualification for Areas of Limited Accessibility			
Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated below in the DCD.			
General		Exception	Current practice does not require qualification or requalification of welders for areas of limited accessibility as described by this regulatory guide. The performance of required nondestructive evaluations helps to confirm weld quality. Limited accessibility qualification or requalification in excess of ASME Code, Section III or IX requirements is considered an unduly restrictive requirement for component fabrication, where the welders' physical position relative to the welds is controlled and does not present significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding is closely supervised. For field application, the type of qualification is considered on a case-by-case basis due to the great variety of circumstances encountered.
Conformance with Revision 1 of the Regulatory Guide during the operational phase (i.e., after the construction phase is completed per the DCD) is documented below.			
General		Exception	Current practice does not require qualification or requalification of welders for areas of limited accessibility as described by this regulatory guide. The performance of required nondestructive evaluations helps to confirm weld quality. Limited accessibility qualification or requalification in excess of ASME Code, Section III or IX requirements is considered an unduly restrictive requirement for component fabrication, where the welders' physical position relative to the welds is controlled and does not present significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding is closely supervised. For field application, the type of qualification is considered on a case-by-case basis due to the great variety of circumstances encountered.
Reg. Guide 1.72, Rev. 2, 11/78 – Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin			
General	ASME Code CCN-155-1 (1792-1)	N/A	The AP1000 does not have safety-related spray pond piping components. Therefore, this regulatory guide is not applicable to the AP1000.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.73, Rev. 0, 1/74 – Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants			
General	IEEE Std. 382-1972	Exception	Qualification of valve appurtenances, such as motor operators, solenoid valves, and limit switches, is in accordance with this regulatory guide. For safety-related motor-operated valves located inside containment, environmental qualification is performed in accordance with IEEE Standards 382-1996 (Reference 21) and 323-1974 (Reference 22).
C.1-6		Conforms	
Reg. Guide 1.74 – Withdrawn			
Reg. Guide 1.75, Rev. 2, 9/78 – Physical Independence of Electric Systems			
Conformance with Revision 2 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.			
General	IEEE Std. 384-1974	Exception	Regulatory Guide 1.75 endorses IEEE Std. 384-74 (Reference 23) which has been superseded by a later revision, IEEE Std. 384-81 (Reference 24). It is the later version that is used for the referenced purposes. This version has not yet been endorsed by a regulatory guide. The differences between the two revisions are not expected to contribute to conflicting design configurations because the jurisdiction of Regulatory Guide 1.75 with regard to the onsite ac power sources is limited. Specifically, since the AP1000 does not use safety-related ac power sources, the guidelines of Regulatory Guide 1.75 are applicable on a very limited basis to provide guidance on the Class 1E/non-Class 1E electrical separation and isolation for the following ac components that employ safety-related and nonsafety-related circuits: a) Class 1E dc battery chargers b) Reactor coolant pump switchgear c) Class 1E dc and UPS system regulating transformers. See Subsection 8.3.2.4.2 for exceptions related to spacial separation between separation groups. Two fuses in series may be used as an isolation device for Class 1E and non-Class 1E isolation. Fiber optic cable may be used as an isolation device for Class 1E to non-Class 1E isolation as well as Class 1E inter-divisional isolation. IEEE 384-2008 exempts fiber optic cables from the need to undergo qualification testing to demonstrate their isolation capability due to their inherent electrical isolation characteristics. This provision has been adopted for the AP1000. Fiber optic cables are not capable of generating a fire due to an internally generated fault. Therefore, separation is not required for fiber optic cables when the cable normally considered as a potential fire source is a fiber optic cable.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.76, Rev. 0, 4/74 and Rev. 1, 3/07 – Design Basis Tornado for Nuclear Power Plants			
Regulatory Guide 1.76, Rev. 1, 3/07 – Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants			
Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guide is completely within the scope of the DCD.			
C.1		Exception	The design basis tornado for the AP1000 is defined by the following parameters: Maximum wind speed: 300 mph Maximum rotational speed: 240 mph Translational speed: 60 mph (maximum) 5 mph (minimum) Radius to maximum wind from center of tornado: 150 feet Atmospheric pressure drop: 2.0 psi Rate of pressure drop: 1.2 psi/sec. Chapter 2 provides design basis tornado interface parameters.
C.2		Conforms	
Reg. Guide 1.77, Rev. 0, 5/74 – Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors			
General		N/A	The interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, is followed for the analysis of the RCCA ejection accident instead of Regulatory Guide 1.77. The guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors" is also followed instead of Regulatory Guide 1.77.
Reg. Guide 1.78, Rev. 1, 12/01 – Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release			
Conformance with the design aspects is as stated below in the DCD.			
C.1		N/A	This criterion is site-specific. It is not applicable to AP1000 design certification. Subsection 2.2.1defines the responsibility for addressing site-specific information on identification of site-specific potential hazards.
C.2		N/A	This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. Subsection 2.1.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards. Subsection 6.4.7 defines the responsibility for addressing site-specific information amount and location of possible sources of toxic chemicals in or near the plant relative to control room habitability.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3.1		N/A	This criterion is site-specific. It is not applicable to AP1000 design certification. Subsection 2.2.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards. Subsection 6.4.7 defines the responsibility for addressing site-specific information amount and location of possible sources of toxic chemicals in or near the plant relative to control room habitability.
C.3.2		Conforms	
C.3.3		Exception	For AP1000 design certification, the atmospheric dispersion factors are not calculated (since there are no specific site data), but are selected so as to bound the majority of existing sites. Section 2.3 provides additional information.
C.3.4		Conforms	
C.4.1		N/A	This criterion is site-specific. It is not applicable to AP1000 design certification. Subsection 2.2.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards. Subsection 6.4.7 defines the responsibility for addressing site-specific information amount and location of possible sources of toxic chemicals in or near the plant relative to control room habitability.
C.4.2		Conforms	
C.4.3		Conforms	
C.5		N/A	Not applicable to AP1000 design certification. Subsection 2.1.1 defines the responsibility for addressing site-specific information on identification of site location and description, exclusion area authority and control, and population distribution. Subsection 2.2.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards. Section 13.3 defines the responsibility for addressing emergency planning.
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.79, Rev. 1, 9/75 – Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors			
General		Conforms	Preoperational testing is performed to test the functioning of the accumulators, core makeup tanks, passive residual heat removal heat exchanger, and automatic depressurization system, in a manner consistent with this regulatory guide. However, the AP1000 does not have high-head or low-head active safety-injection pumps. Therefore, many of the specific guidelines of this regulatory guide do not apply.
Reg. Guide 1.80 – Withdrawn			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.81, Rev. 1, 1/75 – Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plant			
General		N/A	The AP1000 is a single unit plant. When two or more AP1000s are located adjacent, electrical systems are not shared.
Reg. Guide 1.82, Rev. 3, 11/03 – Water Sources for Long Term Recirculation Cooling Following a Loss-of-Coolant Accident			
Conformance with the design aspects is as stated below in the DCD.			
General		Conforms	The AP1000 does not have high-head or low-head safety-injection pumps that need to take suction from the containment. The AP1000 does have a gravity-driven recirculation path that employs a containment recirculation arrangement. This containment recirculation can also be used to feed the normal residual heat removal pumps if they are available. The containment recirculation design conforms with the guidelines of this regulatory guide.
Conformance with programmatic and/or operational aspects is documented below.			
C.1.1.2		Conforms	
C.1.1.5		Conforms	
Reg. Guide 1.83, Rev. 1, 7/75 – Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes			
Conformance of the design aspects is as stated below in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (74 FR 58324, 11/12/2009).			
General		Conforms	A program for in-service inspection of AP1000 steam generator tubing is established and performed in accordance with the guidelines of this regulatory guide. The baseline inspection will be performed in accordance with Regulatory Position C.3.a. Should the Combined License applicant request a baseline examination at the manufacturing facility, it will be performed in accordance with Regulatory Position C.3.a.
C.1		Conforms	
C.2		Exception	The specification of equipment in Regulatory Position C.2.c does not represent state-of-the-art equipment for gathering and storing eddy current information. When an eddy current inspection of an AP1000 steam generator is done in the manufacturing facility, more capable equipment than that specified in the regulatory guide is used. The steam generator design is compatible with robotic eddy current inspection equipment.
C.3		Exception	As noted in the comment on Criteria Section C.2, any eddy current inspection done in the manufacturing facility uses equipment of more current technology than that specified in Criteria Section C.2.
C.4-7		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.8		Exception	The only corrective action recognized by the regulatory guide is plugging of the tube to remove it from service. Sleeving of tubes is in many cases an acceptable repair method. The AP1000 steam generator design provides increased access to tubes to implement the sleeving repair method or other repair methods which may be developed.
Reg. Guide 1.84, Rev. 32, 6/03 and Rev. 33, 8/05 – Design and Fabrication Code Case Acceptability ASME Section III Division 1			
Regulatory Guide 1.84, Rev. 33, 8/05 – Design, Fabrication, and Materials Code Case Acceptability, ASME Section III			
Conformance with Revision 32 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.			
General		Conforms	The ASME Code Cases required for design certification are listed in Table 5.2-3. These cases are included in Regulatory Guide 1.84 or have been accepted by the US Nuclear Regulatory Commission staff as part of the review of AP1000.
C.1	ASME Code, Section III, Code Cases	Conforms	As applicable for pressurized water reactor.
C.2-5		Conforms	
Reg. Guide 1.85, Rev. 31, 5/99 – Materials Code Case Acceptability - ASME Code, Section III, Division 1			
General		Conforms	Refer to the discussion on Regulatory Guide 1.84. Subsequent to Revision 31 Reg. Guide 1.85 was combined with Reg. Guide 1.84. The guidance and conditions included in the previous revisions of Reg. Guide 1.85 remains valid.
C.1	ASME Code, Section III, Code Cases	Conforms	As applicable for pressurized water reactor.
C.2-5		Conforms	
Reg. Guide 1.86, Rev. 0, 6/74 – Termination of Operating Licenses for Nuclear Reactors			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Reg. Guide 1.87, Rev. 1, 6/75 – Guidance for Construction of Class 1 Components in Elevated Temperature Reactors			
General		N/A	The AP1000 is not an elevated temperature reactor. See Section 1.2 for a general description of the plant and plant parameters. This regulatory guide is not applicable to the AP1000.
Reg. Guide 1.88 – Withdrawn			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.89, (Task EE 042-2), Rev. 1, 6/84 – Environmental Qualification of Certain Electric Equipment Important to Safety For Nuclear Power Plants			
General	IEEE Std. 323-1974	Conforms	Conformance of AP1000 Class 1E equipment with 10 CFR 50.49, Reference 26 and this regulatory guide is demonstrated by an appropriate combination of the following: type testing, operating experience, qualification by analysis and ongoing qualification.
C.1	App. A	Conforms	As applicable for pressurized water reactor.
	App. B	Conforms	As applicable for pressurized water reactor.
	Regulatory Guide 1.97	Conforms	As applicable for pressurized water reactor.
C.2		Conforms	
C.2.a	App. C	Conforms	As applicable for pressurized water reactor.
C.2.b		Conforms	
C.2.c	App. D	Conforms	
C.2.c.1		Exception	Source term definition is discussed in the exceptions to Regulatory Guide 1.4, Positions C.1.a and C.1.b.
C.2.c.2		Exception	The fission product inventories in the fuel are discussed in the exception to Regulatory Guide 1.25, Position C.1.d.
C.2.c.3-8		Conforms	
C.2.d		Conforms	
C.3-6		Conforms	
C.7	App. E	Conforms	
Reg. Guide 1.90, Rev. 1, 8/77 – Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons			
General		N/A	The AP1000 does not have a concrete containment and does not use a prestressing tendon in the containment structures. Therefore, this regulatory guide is not applicable to the AP1000.
Reg. Guide 1.91, Rev. 1, 2/78 – Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites			
Conformance of the design aspects is as stated below.			
General		Conforms	Offsite explosive materials are site-specific. See Subsection 2.2.1 for information for identification of site-specific potential hazards.
General		Exception	Onsite explosive materials conform to the guidelines of Reg. Guide 1.91, Rev. 2. See Subsection 2.2.1 for information on potential hazards.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.92, Rev. 1, 2/76 and Rev. 2, 07/06 – Combining Modal Responses and Spatial Components in Seismic Response Analysis			
Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
Reg. Guide 1.93, Rev. 0, 12/74 – Availability of Electric Power Sources			
C.1-2		N/A	The ac power sources are nonsafety-related. Therefore, these guidelines do not apply to the AP1000.
C.3		N/A	The function of the nonsafety-related diesel-generators for the AP1000 is to provide ac power for equipment and lighting during loss of offsite power but is not required for safe shutdown. Therefore, these guidelines do not apply to the AP1000.
C.4		N/A	See discussion on Criteria Section C.3.
C.5		Exception	AP1000 does not follow the guidance of C.5 for a 2-hour completion time for the limiting conditions of operation associated with the loss of one dc power subsystem. If one of the Class 1E dc electrical power subsystems is inoperable, the remaining Class 1E dc electrical power subsystems have the capacity to support a safe shutdown and to mitigate all design basis accidents, based on conservative analysis. Because of the passive system design and the use of fail-safe components, the remaining Class 1E dc electrical power subsystems have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst-case single failure. Also, with passive/fail-safe design, the risk associated with the loss of one Class 1E dc subsystem is similar to the loss of one ac supply for a conventional unit. AP1000 uses a 6-hour completion time for the limiting conditions of operation associated with the loss of one dc power subsystem to be consistent with the guidance in C.1 for a conventional plant with the loss of one ac source. The 6-hour completion time is reasonable based on engineering judgement balancing the risks of operation without one dc subsystem against the risks of a forced shutdown. Additionally, the completion time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown, and if necessary, prepare and effect an orderly and safe shutdown.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.94, Rev. 1, 4/76 – Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N45.2.5-1974	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the quality assurance program.
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.95 – Withdrawn			
Reg. Guide 1.96, Rev. 1, 6/76 – Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.97, Rev. 3, 5/83 and Rev. 4, 6/06 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident			
Regulatory Guide 1.97, Rev. 4, 6/06 – Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants			
Conformance with Revision 3 of the Regulatory Guide is as stated below in the DCD.			
General	ANS-4.5-1980	Conforms	The variables to be monitored are selected according to usage and need in the plant Emergency Response Guidelines. They are assigned design and qualification Category 1, 2, or 3 and classified as Type A, B, C, D, or E. Due to AP1000 specific design features, the selection of some plant-specific variables and their classifications and categories are different from those of this regulatory guide. For example, the use of the passive residual heat removal system as the safety grade heat sink allows steam generator wide range level to be category 2, not category 1 as specified in Regulatory Guide 1.97. The AP1000 has no Type A variables. See Section 7.5 for additional information. Since Category 3 instrumentation is not part of a safety-related system, it is not qualified to provide information when exposed to a post-accident adverse environment. Category 3 instrumentation is subject to servicing, testing, and calibration programs that are specified to maintain their capability. However, these programs are not in accordance with Regulatory Guide 1.118, which applies to safety-related systems.
C.1-2		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Exception	Portable equipment outside the DCD scope conforms to Revision 3 of this Regulatory Guide for consistency with DCD scope since Revision 4 indicates that partial implementation is not advised.
Reg. Guide 1.98, Rev. 0, 3/76 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.99, (Task ME 305-4), Rev. 2, 5/88 – Radiation Embrittlement of Reactor Vessel Materials			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
Reg. Guide 1.100, (Task EE 108-5), Rev. 2, 6/88 – Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants			
General	IEEE Std. 344-1987	Conforms	
Reg. Guide 1.101, Rev. 3, 8/92 and Rev. 5, 6/05 – Emergency Planning and Preparedness for Nuclear Power Reactors			
Conformance of the design aspects of Rev. 3 is as stated below in the DCD.			
General	NUREG-0654, FEMA-REP-1 NUMARC/NESP-007	Conforms	Section 13.3 defines the responsibility for addressing emergency planning. RG 1.101 (Revision 2) references NUREG-0654/ FEMA-REP-1 and item II.H, "Emergency Facilities and Equipment" of NUREG-0654/FEMA-REP-1, is applicable to the technical support center (TSC), operations support center (OSC), and the emergency operations facility (EOF) in the AP1000 design. Subsection 18.2.6 defines the responsibility for designing the EOF in accordance with the AP1000 human factors engineering program, including specification of its location. The AP1000 design conforms with the design criteria of item II.H that pertain to the TSC and the OSC.
Conformance with Rev. 5 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.102, Rev. 1, 9/76 – Flood Protection for Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
C.1		Conforms	
C.2	Regulatory Guide 1.59, C.2	Conforms	
C.3		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.103 – Withdrawn			
Reg. Guide 1.104 – Withdrawn			
Reg. Guide 1.105, Rev. 3, 12/99 – Instrument Setpoints for Safety-Related Systems			
General	ISA-S67.04-1994	Conforms	The technical specifications setpoints provide the margin from the nominal setpoint to the safety-analysis limit to account for drift when measured at the rack during periodic testing. The allowances between the technical specification limit and the safety limit include the following items: a) the inaccuracy of the instrument; b) process measurement accuracy; c) uncertainties in the calibration; and d) environmental effects on equipment accuracy caused by postulated or limiting postulated events (only for those systems required to mitigate consequences of an accident). The setpoints are chosen such that the accuracy of the instrument is adequate to meet the assumptions of the safety analysis. The instrumentation range is based on the span necessary for the associated function. Narrow range instruments are used where necessary. Instruments are selected based on expected environmental and accident conditions. The need for qualification testing is evaluated and justified on a channel-by-channel basis. Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism such that no integral setpoint securing device is required. Integral setpoint locking devices are not supplied. A plant-specific setpoint analysis must be performed to provide technical specification setpoints prior to plant startup. AP1000 conforms to the documentation requirements of the 1994 criteria.
Reg. Guide 1.106, Rev. 1, 3/77 – Thermal Overload Protection for Electric Motors on Motor-Operated Valves			
C.1	IEEE 279-1971, Sections 4.1, 4.2, 4.3, 4.4, 4.5, 4.10, and 4.13	Exception	Regulatory Guide 1.106 endorses IEEE Std. 279-1971 Reference 27, which has been replaced by IEEE STD 603-1991, (Reference 51). The AP1000 uses IEEE Std. 603-1991. This standard is endorsed by Regulatory Guide 1.153. The only safety-related electric motor-operated valves are dc.
C.2		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.107, Rev. 1, 2/77 – Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures			
General		N/A	The AP1000 does not have a concrete containment and does not use a prestressing tendon in the containment structure. Therefore, these guidelines are not applicable to the AP1000.
Reg. Guide 1.108 – Withdrawn			
Reg. Guide 1.109, Rev. 1, 10/77 – Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	This is applicable to the evaluation of specific sites. AP1000 design certification application evaluates how the AP1000 design is expected to compare with existing plants. This comparison is made based on the calculation of anticipated annual releases for the AP1000.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.110, Rev. 0, 3/76 – Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General	10 CFR 50, App. I, Section II.D	Exception	The disposal of effluents for the AP1000 is within the limits of Appendix I of 10 CFR 50, and the radwaste treatment systems have sufficient capacity to control effluents. The AP1000 approach to the design of radwaste systems is the result of a nuclear industry-sponsored program to optimize the radwaste systems design. A site-specific cost-benefit analysis is not required for sites that meet the site interface criteria.
Conformance with Revision 0 this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.111, Rev. 1, 7/77 – Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors			
Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. This is applicable to the evaluation of specific sites. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to design certification.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.112, Rev. 0-R, 5/77 and Rev. 1, 3/07 – Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors			
Conformance of the design aspects with Revision 0-R of the Regulatory Guide is as stated below in the DCD.			
C.1	10 CFR 20.1(c); 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, App. I	Exception	The reference to 10 CFR 20.1(c) is no longer valid in the current version of 10 CFR Part 20.
C.2	NUREG-0016; NUREG-0017	Exception	Revision 1 of NUREG-0017 is used. Plant-specific values for charcoal adsorption coefficients are used to calculate delay times in-place of the generic values given in NUREG-0017.
C.3		Conforms	
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General	ANSI 18.1-1999	Conforms	
Reg. Guide 1.113, Rev. 1, 4/77 – Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I			
Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. This is applicable to the evaluation of specific sites. Interface data are provided. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to design certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.114, Rev. 2, 5/89 – Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Section 13.2 defines the responsibility for training and Section 13.5 defines the responsibility for procedures.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.115, Rev. 1, 1/77 – Protection Against Low-Trajectory Turbine Missiles			
Conformance of the design aspects is as stated below in the DCD.			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	The SRP 3.5.1.3 issued in 1981 and Regulatory Guide 1.115, issued in 1977, provide criteria for protection against the effects of potential turbine missiles. Reference 28 issued in 1984 states that "the Nuclear Regulatory Commission staff has shifted emphasis in the reviews of the turbine missile issue from the strike and damage probability (P_2xP_3) to the missile generation probability (P_1) and, in the process, has attempted to integrate the various aspects of the issue into a single coherent evaluation." The AP1000 turbine is arranged in a radial orientation. The three low pressure turbines incorporate fully integral rotors. The turbine conforms with the criteria given in Reference 28 and WCAP-16650 (Reference 52).
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.116, Rev. O-R, 5/77 – Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N45.2.8-1975	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the Quality Assurance program.
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.117, Rev. 1, 4/78 – Tornado Design Classification			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
APPENDIX			
General		Conforms	For the AP1000, the leaktight integrity of the primary containment is maintained.
Reg. Guide 1.118, Rev. 3, 4/95 – Periodic Testing of Electric Power and Protection Systems			
General	IEEE Std. 338-1987	Conforms	Guidelines apply to safety-related dc power systems. Since the AP1000 has no safety-related ac power sources, the guidelines do not apply to the AP1000 ac power sources. The types of tests described in IEEE 338 Section 6.3 are not all applicable to the protection and safety monitoring system. In certain instances, the self-diagnostics included within the protection and safety monitoring system are used to verify that the safety system is capable of meeting its designed safety function in lieu of manual testing as part of the surveillance program.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			Specifically, logic system function tests, response time tests, and in certain instances, channel checks are not manually performed on the protection and safety monitoring system equipment as part of the AP1000 surveillance program. In these cases, self-diagnostic test features continuously monitor the system. Functional tests are only performed on the PMS equipment that do not have complete self-diagnostic coverage. The Technical Specifications provide the necessary manual functional testing requirements in these instances (e.g., ALT, TADOT, channel checks). Channel calibration verification tests are included in the AP1000 surveillance program.
Reg. Guide 1.119 – Withdrawn			
Reg. Guide 1.120, Rev. 1, 11/77 – Fire Protection Guidelines for Nuclear Power Plants			
General		Exception	The AP1000 design conforms with the Branch Technical Position CMEB 9.5-1 (Reference 32), which is attached to Section 9.5.1 of the Standard Review Plan, NUREG-0800 (Reference 33), as described in Section 9.5.1. Therefore, these guidelines are not applicable to the AP1000.
Reg. Guide 1.121, Rev. 0, 8/76 – Bases for Plugging Degraded Pressurizer Water Reactor Steam Generator Tubes			
General		Conforms	The only corrective action recognized by this regulatory guide is plugging of the tube to remove it from service. Sleeving of tubes is in many cases an acceptable repair method. The AP1000 steam generator design provides increased access to tubes to implement the sleeving repair method or other repair methods which may be developed.
C.1		Exception	Westinghouse interprets the term "unacceptable defects" to apply to those imperfections resulting from service induced mechanical or corrosion degradation of the tube walls which have penetrated to a depth or a length or a combination of both in excess of the plugging limit.
C.2.a.(1)		Exception	Westinghouse interprets this criterion to exclude the local region of the crack tip for Inconel tubing.
C.2.a.(2)		Exception	Tube minimum wall requirements are calculated in accordance with the following criteria. For normal plant operation, allowable membrane stress, Pm, is limited to a margin of three against exceeding the ultimate tensile strength of the material. As this regulatory guide constitutes an operating criterion, the allowable stress limits are based on expected lower bound material properties rather than ASME Code minimum values. Expected strength properties are obtained from statistical analysis of tensile test data of actual production tubing.
C.2.a(3)		Conforms	
C.2.a(4)		Exception	Refer to the discussion on Criteria Section C.2.a(2).

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.2.a.(5)-(6)		Conforms	
C.2..b.		Exception	In cases where sufficient inspection data exist to establish degradation allowance, the rate used is an average time-rate determined from the mean of the test data. Where requirements for minimum wall are markedly different for various areas of the tube bundle, such as the U-bend area versus straight length in Westinghouse designs, separate plugging limits are established to address the varying requirements in a manner which does not require unnecessary plugging of tubes.
C.3.a - c		Conforms	
C.3.d.(1)-(3)		Conforms	
C.3.e.(1)-(5)		Conforms	
C.3.e.(6)		Exception	Computer code names and references are supplied rather than actual codes.
C.3.e.(7)-(10)		Conforms	
C.3.f.(1)-(3)		Conforms	
Reg. Guide 1.122, Rev. 1, 2/78 – Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components			
C.1-3		Conforms	
Reg. Guide 1.123 – Withdrawn			
Reg. Guide 1.124, Rev. 1, 1/78 and Rev. 2, 02/07 – Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports			
Regulatory Guide 1.124, Rev. 2, 02/07 – Service Limits and Loading Combinations for Class 1 Linear-Type Supports			
Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD.			
General	ASME Code, Section III Subsection NF	Conforms	Many of the items addressed in this regulatory guide have since been incorporated into later ASME Code, Section III Editions and Addenda. The design conforms to this regulatory guide with the following interpretations to maintain consistency with the ASME Code: 1. References to ASME Code, Section III, Subsection NF and Appendix XVII paragraphs are interpreted to be references to the corresponding paragraph in Subsection NF of the ASME Code. 2. References to ASME Code Case 1644 are interpreted to be references to the accepted versions of ASME Code Cases N-249 and N-71.
C.1		Conforms	
C.2	Code Case 1644	Conforms	Values of Su at these elevated temperatures are determined by test rather than via the method 2 as given by this regulatory position.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3	ASME Code, Section III, Appendix XVII	Conforms	
C.4	ASME Code, Section III, Appendix XVII-2110(b)	Exception	Paragraph B.1(b) of this regulatory guide states that "Allowable service limits for bolted connections are derived from tensile and shear stress limits and their nonlinear interaction. They also change with the size of the bolt. For this reason, the increases permitted by ASME Code, Section III, Subsections NF-3231.1, XVII-2110(a), and F-1370(a) are not directly applicable to allowable shear stresses and allowable stresses for bolts and bolted connections." This regulatory position also states that "This increase of level A or B service limits does not apply to limits for bolted connections and shear stresses." As stated above, the increase in bolt allowable stress under emergency and faulted conditions is not permitted because the interaction between the allowable tension and shear stress in bolts is nonlinear, and the allowable tension and shear stress vary with the bolt size. The ASME Code, NF-3225, allows small increases in allowable stresses for Level B, Level C (previously termed "emergency"), Level D (previously termed "faulted"), and test conditions. The ASME Code rules are adequate since they satisfy the two objectives raised in the above quoted paragraph and will be used without further restrictions or justifications. This position is based on the following. 1. The interaction curve between the shear and tension stress in bolts is more closely represented by an ellipse and not a line. 2. The ASME Code specifies stress limits for bolts and represents this tension/shear relationship as a non-linear interaction equation (ellipse). This interaction equation has a built-in safety factor that ranges between two and three (depending on whether the bolt load is predominately tension or shear) based on the actual strength of the bolt as determined by test. See Reference 34.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			3. This regulatory position states that "Any increases of limits for shear stresses above 1.5 times the ASME Code, Level A service limits should be justified." Concerning allowable shear stresses, the AP1000 uses the ASME Code, Subsection NF requirements. The ASME Code shear stress limits (NF-3300 and Tables NF-3523.2 and NF-3623.2-1) generally meet the guidance provided by this regulatory position that shear stresses be maintained within 1.5 times Level A service limits. This limit may be exceeded slightly in some limited cases such as Level D limits for SA-36 material, in which case the NF shear stress limit of .42 Su is 13 percent greater than this regulatory guide limit of 1.5 x .4 Fy. Su and Fy are the material tensile and yield strengths, respectively.
C.5.a	ASME Code, Section III,	Exception	The AP1000 evaluates supports to current Level B stress limits for the upset load combination. Effects of constraint of free-end displacements are included in the upset loading condition while no further increase in allowable stresses over and above the Level B limits is permitted. The operating basis earthquake has been eliminated from the AP1000 design basis.
C.5.b-c	ASME Code, Section III, Subsection NF-3262.3, Appendix XVII-4200, Appendix XVII-4110(a)	Conforms	The operating basis earthquake has been eliminated from the AP1000 design basis.
C.6	ASME Code, Section III, Appendix XVII-2000, 2110(a) Subsection NF 2362.3, Appendix XVII-4200, 4110(a), II-1400	Conforms	
C.7.a	ASME Code, Section III, Appendix XVII-2000, and F-1370(a)	Conforms	
C.7.b		Exception	The AP1000 uses the provisions of the ASME Code, Section III, Appendix F to determine faulted condition allowable loads for supports designed by the load rating method. The method described in this regulatory position is conservative and inconsistent with the remainder of the faulted stress limits.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.7.c	ASME Code, Section III Appendix XVII-4200, and F-1370(b)	Conforms	
C.7.d	ASME Code, Section III, II-1400, and F-1370(b)	Conforms	
C.8		Exception	The reduction of allowable stresses to no greater than Level B limits for support structures in those systems with normal safety-related functions occurring during emergency or faulted plant conditions is overly conservative for components which are not required to mechanically function (inactive components). For service Level C and D loading conditions, Level C limits are used for the support of active components.
Conformance with Revision 2 for programmatic and/or operational aspects of the Regulatory Guide is as stated below. This guidance is completely within the scope of the DCD.			
Reg. Guide 1.125, Rev. 1, 10/78 – Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants			
Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.			
General		Conforms	
C.1-6		Conforms	
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.126, Rev. 1, 3/78 – An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification			
C.1-2		Exception	Westinghouse uses the densification model described in the Nuclear Regulatory Commission-approved topical reports WCAP-10851-A and WCAP-13589-A. Westinghouse conforms to the methodology of this regulatory guide when implementation of the methodology is required.
C.3-4		Conforms	
C.5		Conforms	
Reg. Guide 1.127, Rev. 1, 3/78 – Inspection of Water-Control Structures Associated With Nuclear Power Plants			
General		N/A	The AP1000 does not have water-control structures. Therefore, this guideline is not applicable to the AP1000. Subsection 2.5.6 defines the responsibility for embankments and dams.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.128, Rev. 1, 10/78 and Rev. 2, 2/07– Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants			
Regulatory Guide 1.128, Rev. 2, 2/07 – Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants			
Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD.			
General	IEEE Std. 484-1975	Exception	Regulatory Guide 1.128 endorses IEEE Std. 484-75 (Reference 36) which has been superseded by IEEE Std. 484-1996 (Reference 37). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.128.
Regulatory Guide 1.129, Rev. 2, 2/07 – Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants			
General	IEEE Std. 450-2002	Exception	Battery maintenance, testing, and replacement is in accordance with approved Technical Specifications which are based on IEEE Std. 450-1995.
Reg. Guide 1.130, Rev. 1, 10/78 – Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Component Supports			
Regulatory Guide 1.130, Rev. 2, 3/07 - Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Supports			
Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD.This guidance is completely within the scope of the DCD.			
General	ASME Code, Section III Subsection NF	Exception	Many of the items addressed in this regulatory guide have since been incorporated into later ASME Code, Section III, Editions and Addenda. The plant design conforms to this regulatory guide with the following interpretations to maintain consistency with the ASME Code: 1. Regulatory guide references to ASME Code, Section III, Subsection NF and Appendix XVII paragraphs are interpreted to be references to the corresponding paragraph in the ASME Code, Subsection NF. 2. Regulatory guide references to ASME Code Case 1644 are interpreted to be references to the latest acceptable versions of the ASME Code Case N-249 and N-71. Paragraph B.1 of this regulatory guide states that "Allowable stress limits for bolted connections are derived on a different basis that varies with the size of the bolt. For this reason, the increases permitted by NF-3222.3 and F-1323.1(a) of ASME Code, Section III are not directly applicable to bolts and bolted connections." The ASME Code rules are adequate for bolt design without further restriction or justification.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			The maximum stress increase factor allowed is 25 percent for the Service Level D condition, and the stress allowables do not vary with bolt size. The AP1000 takes exception to the guideline stated in Paragraph B.5 of this regulatory guide, that systems whose safety-related function occurs during emergency or faulted plant conditions should meet Level A limits. The reduction of allowable stresses to no greater than Level A limits for support structures in those systems with normal safety-related functions occurring during emergency or faulted plant conditions is overly conservative for components which are not required to mechanically function (inactive components). For service, Level C and D loading conditions, Level C limits are used for the support of active components.
C.1		Conforms	
C.2	Code Case 1644	Conforms	
C.3		Exception	Design margins of two for flat plates and three for shells are unnecessarily restrictive for normal, upset, and emergency conditions, as well as inconsistent with ASME Code requirements. For these loading conditions, the AP1000 limits the allowable buckling strength to 1/2 of the critical buckling strength.
C.4	ASME Code, Section III, NF-3221.1, NF-3221.2, NF-3222, NF-3262.2, II-1400	Exception	This regulatory position recommends that design stress limits be used in conjunction with a loading combination that includes operating basis earthquake. The ASME Code rules (in which Level B stress limits are typically used for the upset load combination) provide a conservative design basis. The AP1000 uses the latest rules (as of 4/2001) without further restriction or justification. The operating basis earthquake has been eliminated from the AP1000 design basis. Refer also to the discussion on Criteria Section C.3.
C.5.a	ASME Code, Section III, NF-3224	Exception	Refer to the discussion on Criteria Section C.3.
C.5.b-c	ASME Code, Section III, NF-3262.2, II-1400	Conforms	
C.6.a	ASME Code, Section III, F-1323.1(a), F-1370(c)	Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.6.b	ASME Code, Section III, NF-3262.1	Exception	The limit based on the test load given in this regulatory position is overly conservative and is inconsistent with ASME Code requirements. The AP1000 uses the provisions of the ASME Code, Section III, Appendix F to determine faulted condition allowable loads for supports designed by the load rating method.
C.6.c		Conforms	
C.6.d	ASME Code Section III, F-1324, F-1370(c)	Conforms	
C.7		Conforms	
Reg. Guide 1.131, Rev. 0, 8/77 – Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants			
General	IEEE Std. 383-1974	Exception	The insulating and jacketing material for electrical cables are selected to meet the fire and flame test requirements of IEEE Standard 1202, IEEE Standard 383, or the standards for the corresponding test types as listed in Tables K-1 and K-2 in NEI 04-02, Rev 3 excluding the option to use the alternate flame source, oil or burlap.
C.1-14		Conforms	
Reg. Guide 1.132, Rev. 1, 3/79 and Rev. 2, 10/03 – Site Investigations for Foundations of Nuclear Power Plants			
Conformance of the design aspects for Revision 1 of this Regulatory Guide is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Section 2.5 defines the responsibility for site investigations and the site specific information related to basic geological, seismological, and geotechnical engineering of the site.
Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.133, Rev. 1, 5/81 – Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	A digital metal impact monitoring system (DMIMS) monitors the reactor coolant system for the presence of loose metallic parts. The system actuates audible and visual alarms if a signal exceeds the preset alarm level. The digital metal impact monitoring system is not a Class 1E system. It serves as a diagnostic aid to detect loose parts in the reactor coolant system before damage occurs. Database calibration is made prior to plant startup and the capability for periodic online channel checks and channel functional tests are incorporated in the digital metal impact monitoring system design.
C.1.a-i		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.2		Conforms	
C.3.a		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for development of procedures.
C.3.b		Conforms	
C.4		Conforms	
C.5		Exception	Not applicable to AP1000 design certification. A loose parts detection system is excluded from technical specification requirements per the NRC screening selection criteria in 10 CFR 50.36(c)(2)(ii).
C.6		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for development of procedures.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.2b		Conforms	Procedures are addressed in Section 13.5
C.3a		Conforms	Procedures are addressed in Section 13.5
C.4g		Conforms	Procedures are addressed in Section 13.5
C.4h		Conforms	Procedures are addressed in Section 13.5
C.4i		Conforms	ALARA is addressed in Chapter 12 and Section 13.5
C.4j		Conforms	Training is addressed in Section 13.2
C.6		Exception	Regulatory Guide 1.16 has been withdrawn. Event reporting is performed in accordance with 10 CFR 50.72 and 50.73 utilizing the guidance of NUREG-1022
Reg. Guide 1.134, Rev. 3, 3/98 – Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses			
Regulatory Guide 1.134, Rev. 3, 3/98 – Medical Evaluation of Licensed Personnel at Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 plant design certification. Section 13.5 defines the responsibility for administrative procedures.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.135, Rev. 0, 9/77 – Normal Water Level and Discharge at Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (74 FR 39349, 08/06/2009).			
General		Conforms	The normal ground and surface water levels and surface water discharges for the AP1000 are determined using the postulated site parameters. Chapter 2 provides additional information.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.136, Rev. 2, 6/81 – Materials, Construction, and Testing of Concrete Containments			
General		N/A	The AP1000 does not have a concrete containment. Therefore, this guideline is not applicable to the AP1000.
Reg. Guide 1.137, Rev. 1, 10/79 – Fuel-Oil Systems for Standby Diesel Generators			
General		N/A	The AP1000 diesel-generators and the associated fuel-oil systems are nonsafety-related. Therefore, this guideline is not applicable to the AP1000.
Reg. Guide 1.138, Rev. 0, 4/78 and Rev. 2, 12/03 – Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants			
Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 2.5.4.6.2 defines the responsibility to establish the properties of the foundation soils including laboratory investigations of underlying materials.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.139, Rev. 0, 5/78 – Guidance for Residual Heat Removal			
Conformance with the design aspects is as stated below in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (73 FR 32750, 06/10/2008).			
C.1.a		Exception	The AP1000 employs a full pressure/temperature passive residual heat removal heat exchanger that is automatically actuated. The heat exchanger does not rely on ac or dc power. Fail-safe valves are used to manually isolate the heat exchanger. When these valves are open, the reactor coolant pumps (if available) or natural circulation produces flow through the heat exchangers. The heat exchanger is safety-related, seismically designed, and can tolerate single active failure. Continued operation of the heat exchanger brings the reactor coolant system pressure and temperature down to the point where the stress in the reactor coolant system pressure boundary is low. This temperature is about 400°F which allows an reactor coolant system pressure of 1/10 of design (250 psia).
C.1.b		Conforms	
C.1.c		Exception	See the comment on Criteria Section C.1.a. The passive residual heat removal heat exchanger does not rely on pumps, ac power sources, air systems, or water cooling systems.
C.2		Conforms	
C.3		Conforms	
C.4		N/A	The passive residual heat removal heat exchanger does not have pumps. Therefore, this guideline is not applicable to the AP1000.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.5	IEEE Std. 338; Regulatory Guide 1.22; Regulatory Guide 1.68	Conforms	IEEE Std. 338-1987 (Reference 31) is current standard.
C.6		N/A	The passive residual heat removal heat exchanger provides this function. As a result, the auxiliary feedwater system has been replaced by a nonsafety-related startup feedwater system. Therefore, this guideline is not applicable to the AP1000.
C.7	Regulatory Guide 1.33	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the quality assurance program.
Reg. Guide 1.140, Rev. 2, 06/01 – Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup System in Light-Water-Cooled Nuclear Power Plants			
C.1		Conforms	Regulatory Guide 1.140 endorses ASME Standard N509-1989 (Reference 39), ASME Standard N510-1989 (Reference 40), and ASME AG-1-1997 (Reference 38). The AP1000 uses the latest version of the industry standards (as of 3/2002).
C.2.1-2.4		Conforms	
C.3.1-3.2		Conforms	
C.3.3	ERDA 76-21, Section 5.6; ASME N509-1989 Section 4.9	Conforms	
C.3.4	Regulatory Guide 8.8	Conforms	
C.3.5		Conforms	
C.3.6	ASME AG-1-1997Article SA-4500	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure is designed to meet at least SMACNA design standards.
	ASME AG-1-1997,Section TA	Conforms	
C.4.1	ASME AG-1-1997,Section FB	Exception	ASME AG-1, 1997 Section FB requires filters be certified to UL 900, Class 1, 5th Edition 1987. As of May 2012 UL no longer classifies UL 900 filters with a Class 1 or Class 2 rating. The Class 1 requirements per UL 900, “Air Filter Units” 8th Edition, are obsolete and deleted, as UL 900 no longer distinguishes between Class 1 and Class 2 filters.
C.4.2	ASME AG-1-1997, Section CA	Exception	ASME AG-1, 1997 Section CA require heaters be listed to UL 1996, 2nd Edition. Manufacturers only list components per the latest Edition of the UL Standard. Previous UL Standard Editions/Revisions cannot be invoked.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.4.3	ASME AG-1-1997, Section FC, and Section TA	Exception	ASME AG-1, 1997 Section FC require HEPA Filters be listed to UL 586, 7th Edition. Manufacturers only list components per the latest Edition of the UL Standard. Previous UL Standard Editions/Revisions cannot be invoked.
C.4.4	ASME AG-1-1997, Section FG	Conforms	
C.4.5	ERDA 76-21, Section 4.4; ASME AG-1a-2000, Section HA	Conforms	
C.4.6	ASME N509-1989, Section 5.6; ASME AG-1a-2000, Section HA	Conforms	
C.4.7	ASME AG-1-1997, Section CA	Conforms	
C.4.8	ASME AG-1-1997, Section FD or FE	Conforms	
C.4.9	ASME AG-1-1997, Section FD and FE or, Section FF	Conforms	
C.4.10	ASME AG-1-1997 Section SA	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure is designed to meet at least SMACNA design standards.
C.4.11		Conforms	
C.4.12	ASME AG-1-1997 Section DA	Conforms	
C.4.13	ASME AG-1-1997, Section BA and SA	Conforms	
C.5.1	ERDA 76-21, Section 2.3.8; ASME AG-1a-2000, Section HA	Conforms	
C.5.2		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.6	ASME N510-1989	Conforms	ASME N510-1989 Section 13.1 states that the system bypass test requirements can be considered satisfied if it can be shown that all potential external bypass pathways (bypass dampers, ducts, conduits, floor drains, pipe penetrations, etc.) are challenged by the HEPA and adsorber filter in-place bank tests. Both the VBS and VFS filtration units employ a blow through design which pressurizes the entire filtration units limiting potential bypass pathways internal to the filtration units which are assessed by the HEPA and adsorber filter in-place bank tests.
C.6	ASME AG-1-1997	Exception	The structural capability and leak tightness of outside air intake isolation dampers VBS-MD-D201 and VBS-MD-D202 is affirmed by satisfactory performance of seat leakage testing performed by the damper manufacturer, per Paragraph DA-5740 of ASME AG-1-1997, supplemented by inleakage testing of the main control room pressure boundary during system startup testing, per ASTM E741.
C.7	ANSI N509-1989	Conforms	
Reg. Guide 1.141, Rev. 0, 4/78 – Containment Isolation Provisions for Fluid Systems			
General	ANSI N271-1976	Exception	Regulatory Guide 1.141 endorses ANSI N271-1976 (Reference 41) that has been superseded by ANS 56.2-1984 (Reference 42). The AP1000 uses the latest version of industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.141. Containment isolation for AP1000 fluid systems conforms to Reference 42 with the following exceptions and/or clarifications. ANS 56.2-1984, Section 3.6.3 states that "remote manual closure of isolation valves on engineered safeguards features or engineered safeguards features-related systems is acceptable when provisions are made to detect possible failure of the fluid lines inside and outside containment." The AP1000 engineered safeguards features are designed to avoid the need for transport of post-accident fluids outside of containment and thus avoid the concern associated with remote manual isolation of engineered safety feature lines. Non-engineered safety feature lines capable of providing engineered safety feature functions are provided with the capability for remote manual isolation. The nonsafety-related normal residual heat removal system has provisions to isolate on High-2 containment radiation. Radiation monitors are provided inside containment to assess continuation of the functions.
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.4	ANSI N271-1976, Section 4.4.8, Section 3.5 or 3.6.7	Conforms	
C.5	Regulatory Guide 1.7 & 1.89	Conforms	
C.6	ANSI N271-1976, Section 3.5 or 3.7	Conforms	
Reg. Guide 1.142, Rev. 2, 11/01 – Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)			
General	ACI 349-97	Exception	Regulatory Guide 1.142 endorses ACI 349-97 (Reference 43) that has been superseded by ACI 349-01 (Reference 44). The AP1000 uses the latest version of industry standards (as of October 2001). This version is not endorsed by a regulatory guide but its use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.142. In the following evaluation of conformance, the design is shown as conforming since the requirements of ACI 349-01 are similar to those of ACI 349-97.
		Clarification	ACI 349-01 (Reference 44) Section 1.2.1 requires that "Copies of structural drawings, typical details, and specifications for all reinforced concrete construction shall be signed by a licensed engineer." The signature of the licensed engineer shall be confirmable by documentation to be that of a licensed professional engineer.
C.1		N/A	The compartments within the containment are not designed to be leaktight since they must communicate with one another to preclude subcompartment pressurization. Therefore, this guideline is not applicable to the AP1000.
C.2	ANS 6.4-1997	Conforms	
C.3	ANSI/ACI 349-97	Conforms	
C.4		Conforms	
C.5		Conforms	
C.6	ACI 349-97, Section 9.2.1	Conforms	
C.7		Conforms	
C.8		Conforms	
C.9		N/A	The AP1000 does not include a pressure-suppression containment. Therefore, this guideline is not applicable to the AP1000.
C.10	ACI 349-97, App. C	Conforms	
C.11		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.12	ACI 349-97, App. A	Conforms	
C.13		Conforms	
C.14		N/A	The AP1000 containment vessel is steel.
C.15		Conforms	The provisions in Section 11.6 of ACI 349-01 are the same as those in ACI 318-99 (Reference 46).
Reg. Guide 1.143, Rev. 2, 11/01 - Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water Cooled Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
General			The AP1000 Radwaste Building provides space to store dry active waste and space for mobile waste processing systems and equipment and three liquid waste monitor tanks which contain liquid effluents which have been processed and are acceptable for release to the environment (within the requirements of classification RW-IIC of Regulatory Guide 1.143). It does not contain installed systems and components used to process, store, or handle gaseous or liquid waste.
C.1.1.1	Regulatory Guide 1.143, Table 1	Exception	Components in the liquid radwaste systems are designed and tested to the requirements set forth in the codes and standards listed in Table 1 of Regulatory Guide 1.143 with the exception of the use of the 2013 Edition of ASME, Section VIII, Div. 1, instead of the 1998 Edition (with 1999 Addenda) that is referenced in Appendix A of Regulatory Guide 1.143. This version is not endorsed by a regulatory guide, but its use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.143. Equipment classifications and design codes are listed in Table 3.2-3. Pressure vessels are designed and built according to ASME, Section VIII, Div. 1. Atmospheric tanks are per API 650 or ASME, Section III and heat exchangers to ASME Section VIII, Div. 1 and TEMA (for shell and tube). Piping and valves are per ASME B31.1 except the containment penetrations and isolation valves are per ASME, Section III, Class 2. Pumps are according to manufacturer's standards. Steam generator blowdown system (BDS) electrodeionization (EDI) modules within the EDI packages are designed and constructed in accordance with the manufacturer's standards and may be fabricated from non-metallic materials that differ from those identified in Table 1. Flexible hose is used in the BDS—flexible metal hoses connect the EDI reject radiation monitor package to external piping and flexible plastic hoses within the EDI packages connect EDI modules to adjacent piping.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1.1.2	ASME Code, Section II	Conforms	Materials, except elastomers for gaskets, seals, seats, diaphragms, and packing, are provided in accordance with the ASME Code, Section II when the ASME Code is the design and fabrication standard. Piping and valves materials are per ASTM specifications consistent with ASME B31.1. Pump materials are provided according to manufacturer's standards.
C.1.1.3		Conforms	The auxiliary building that contains the liquid radwaste system with the exception of three monitor tanks is designed to seismic Category I criteria. The seismic Category I structure will retain the maximum liquid inventory of the system. The lowest level of the auxiliary building, elevation 66'-6", contains the liquid radwaste system effluent holdup tanks, a monitor tank, and chemical waste tank within a common flood zone. This flood zone has watertight floors and walls. The enclosed volume within this flood zone is sufficient to contain the contents of the system. The effluent holdup tank rooms, the monitor tank room, and chemical waste tank room each have one or two floor drains that lead to the sump. Tank overflows or spills will be collected in the auxiliary building sump. The sump is automatically pumped to a waste holdup tank. The waste holdup tank rooms each have two normally isolated floor drains that are connected to a sump pump. The waste holdup tank overflows or spills will be manually pumped to a waste holdup tank. Two liquid radwaste system monitor tanks are above at elevations 92′-6" and 107′-2". Overflows or spills from these monitor tanks drain by gravity down through the drain system to a waste holdup tank. The seismic Category I criteria exceed the operating basis earthquake required by regulatory position C.6 of Regulatory Guide 1.143. The radwaste building that contains three liquid radwaste monitor tanks is designed to the Uniform Building Code. The basemat and curbed structure will retain the maximum liquid inventory of any of the three monitor tanks. Tank overflows or spills will be collected by the radioactive waste drain system and routed to the auxiliary building sump. The sump is automatically pumped to a waste holdup tank. The Uniform Building Code design of the radwaste building meets the requirements of regulatory position C.6 of Regulatory Guide 1.143 for structures classified as RW-IIC.
		Exception	The turbine building first bay, which houses the BDS, is seismic Category II. The BDS components are not designed for protection against natural phenomena or internal/ external hazards.
C.1.2.1		Conforms	Atmospheric tanks in the liquid radwaste system have level sensors, transmitters, and alarms. Local alarm is not provided because the tanks are located in shielded areas that are not normally occupied by people.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1.2.2		Conforms	Tank overflows, drains and sample lines that may contain radioactive water are routed to the liquid radwaste system for processing.
C.1.2.3		Conforms	Please refer to the discussion of conformance to C.1.1.3, which addresses the provisions in the buildings that contain radioactive waste to contain any spills.
C.1.2.4		Conforms	Please refer to the discussion of conformance to C.1.1.3, which addresses the provisions in the building that contain radioactive waste to contain any spills. The design includes the following provisions to prevent contaminated leakage from entering unmonitored and nonradioactive systems and ductwork in the area: the HVAC subsystems serving radiologically controlled areas (RCAs) normally maintain the served building areas at a slightly negative pressure to prevent the uncontrolled release of airborne radioactivity to the atmosphere or adjacent clean plant areas. Additionally, ductwork passing through RCAs is Class R and is designed, constructed, and tested in accordance with ASME AG-1 Section SA to prevent the infiltration of airborne radiation to unmonitored and nonradioactive systems. Interfaces with the adjacent buildings are limited to doorways, airlocks, and ductwork. Ductwork connecting the annex building and adjacent areas consists entirely of supply air ductwork handling outside air for the fuel handling area, health physics area, containment purge supply, and main control room. The main control room supplemental air filtration unit is in the HVAC equipment room; however, this unit has no radioactive material during normal plant operation.
		Exception	There are no provisions to prevent BDS leakage from entering the nonradioactive waste water system (WWS). WWS water is monitored for radioactivity prior to discharge and can be manually diverted to the liquid radwaste system (WLS) if necessary. There is no means to prevent BDS leakage from entering nonradioactive ductwork.
C.1.2.5		Conforms	This guideline does not apply because the liquid radwaste treatment system has no outdoor tanks. No other outside tanks store radioactive fluids.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.2.1	Regulatory Guide 1.143, Table 1	Exception	Components in the gaseous radwaste systems are designed and tested to the requirements set forth in the codes and standards listed in Table 1 of Regulatory Guide 1.143 with the exception of the use of the 2013 Edition of ASME, Section VIII, Div. 1, instead of the 1998 Edition (with 1999 Addenda) that is referenced in Appendix A of Regulatory Guide 1.143. This version is not endorsed by a regulatory guide, but its use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.143. Heat exchangers are designed and built according to ASME, Section VIII, Div. 1 and TEMA (for shell and tube). Piping and valves are per ASME B31.1. Pumps are according to manufacturer's standards.
C.2.2	ASME Code, Section II	Conforms	Materials, except elastomers for gaskets, seals, seats, diaphragms, and packing, are provided in accordance with the ASME Code Section II when the ASME Code is the design and fabrication standard. Piping and valves materials are per ASTM specifications consistent with ASME B31.1. Pump materials are provided according to manufacturer's standards.
C.2.3		Conforms	The guard bed and the delay beds, including supports, in the gaseous radwaste system are designed for seismic loads according to the requirements of Regulatory Guide 1.143. These are the only AP1000 components used to store or delay the release of gaseous radioactive waste. The beds are located in the seismic Category I auxiliary building at elevation 66′-6″. Seismic loads for this equipment will be established using one-half of the safe shutdown earthquake (SSE) floor response spectra. The loads due to this seismic response spectra are equivalent or greater than those due to an operating basis earthquake (OBE). Other equipment and supports will be designed in accordance with the codes indicated in Table 3.2-3.
C.3		Conforms	The regulatory guidance applies to the AP1000 solid waste processing system except for components and subsystems used to solidify or concentrate liquid waste. The AP1000 solid waste processing system does not have these components/subsystems. These functions are provided by contractors who process these wastes using mobile systems.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3.1	Regulatory Guide 1.143, Table 1, Reg. Pos. 3.2 and 3.3	Exception	The solid radwaste system is designed and tested to the requirements set forth in the codes and standards listed in Table 1 of Regulatory Guide 1.143 with the exception of the use of the 2013 Edition of ASME, Section VIII, Div. 1, instead of the 1998 Edition (with 1999 Addenda) that is referenced in Appendix A of Regulatory Guide 1.143. This version is not endorsed by a regulatory guide, but its use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.143. The spent resin tanks are designed and tested in accordance with ASME Code, Section VIII, Div. 1. Piping and valves are designed and tested according to ASME B31.1. The pumps are designed to manufacturers' standards and tested in accordance with the Hydraulic Institute standards.
C.3.2	ASME Code, Section II	Conforms	Materials, except elastomers for gaskets, seals, seats, diaphragms, and packing, are provided in accordance with the ASME Code, Section II when the ASME Code is the design and fabrication standard. Piping and valves materials are per ASTM specifications consistent with ASME B31.1. Pump materials are provided according to manufacturer's standards.
C.3.3		Conforms	The seismic Category I auxiliary building will retain the maximum liquid and spent resin inventory of the spent resin tanks. The seismic Category I criteria exceed the operating basis earthquake required by regulatory position C.6 of Regulatory Guide 1.143.
C.3.4		Conforms	The equipment and components used to collect, process, and store solid radwaste are nonseismic as permitted by this paragraph.
C.4.1	Regulatory Guide 8.8	Conforms	Chapter 12, "Radiation Protection," discusses the measures taken to maintain the radiation exposure to personnel as low as reasonably achievable.
C.4.2		Conforms	The quality assurance program for design, fabrication, procurement, and installation of radwaste systems is in accordance with the overall quality assurance program described in Chapter 17, which meets the requirements of Regulatory Guide 1.143, position C.7.
C.4.3	ASME Code, Section IX	Conforms	Pressure-containing components in the radwaste systems are of welded construction to the maximum practical extent. Flanged joints and quick connect fittings are used only where maintenance or operational requirements indicate that they are preferable. Screwed connections are not used except for some instrumentation and vents and drains where welded construction is not suitable. Process lines are 1 in. or larger. Butt welds are used in process lines, which contain radioactive fluids. Nonconsumable backing rings are not used in process piping welds. Process pipe welding is performed as required by ASME B31.1. Component welding is performed as required by the applicable construction code.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
		Exception	The EDI modules within the EDI packages have threaded inlet and outlet connections. There is also process piping less than 3/4" nominal diameter inside the EDI packages.
C.4.4		Conforms	Hydrostatic testing is performed as required by the applicable construction codes.
C.4.5		Conforms	In-service testing of the containment penetrations and isolation valves is performed as described in Subsection 3.9.6. Other tests, on nonsafety equipment, are performed on an item-by-item basis as judged necessary to confirm proper operation of the systems.
C.5	10 CFR Part 20		
C.5.1		Conforms	Systems containing enough activity to be possibly classified as RW-IIa are located in the Auxiliary Building. The Auxiliary Building is seismic Category I.
C.5.2		Conforms	
C.5.3	10 CFR Part 71 Appendix A	Conforms	AP1000 systems and components that store or process radioactive waste are located in the Auxiliary Building.
C.5.4	10 CFR Part 71 Appendix A	Conforms	AP1000 systems and components that store or process radioactive waste are located in the Auxiliary Building and radwaste building.
C.6.1.1	Table 2	Exception	Refer to the discussion on Criteria Section C.1.1.3 for the exception on the BDS piping and components.
C.6.1.2	Table 3	Exception	Refer to the discussion on Criteria Section C.1.1.3 for the exception on the BDS piping and components.
C.6.1.3	Table 4	Exception	Refer to the discussion on Criteria Section C.1.1.3 for the exception on the BDS piping and components.
C.6.1.4	Table 1 & 4	Exception	Refer to the discussion on Criteria Section C.1.1.3 for the exception on the BDS piping and components.
C.6.2.1	UBC 1997, ASCE 7-95	Conforms	The Radwaste Building is designed to UBC-1997 and ASCE 7-98.
C.6.2.2		Conforms	Shield structures, if used, will comply with Regulatory Guide 1.143, position C.6.2.
C.7	ANSI/ANS55.6-1993	Conforms	The quality assurance program for design, fabrication, procurement, and installation of radwaste systems is in accordance with the overall quality assurance program described in Chapter 17, which meets the requirements of Regulatory Guide 1.143, position C.6.

Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

General Conforms

Reg. Guide 1.144 – Withdrawn

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.145, Rev. 1, 11/82 (Revised 2/83 to correct page 1.145-7) – Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants			
Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. The atmospheric dispersion factors for use in determining potential accident consequences are selected to be representative of existing nuclear power plant sites and to bound the majority of them. Chapter 2 provides the interface criteria.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.146 – Withdrawn			
Reg. Guide 1.147, Rev. 12, 5/99 and Applicable Revision – Inservice Inspection Code Case Acceptability ASME Section XI Division 1			
Conformance with Revision 12 of the Regulatory Guide is documented below in the DCD. Conformance of the design aspects is as stated in the DCD.			
General	ASME Code, Section XI	Conforms	
Conformance with the applicable revision of this Regulatory Guide for programmatic and/or operational aspects is documented in the Preservice and Inservice Inspection (PSI and ISI) Programs in accordance with 10 CFR 50.55a.			
Reg. Guide 1.148 – Withdrawn			
Reg. Guide 1.149, Rev. 2, 4/96 – Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations			
Regulatory Guide 1.149, Rev. 4, 4/11 – Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements			
Conformance with Revision 2 of the Regulatory Guide is documented below in the DCD. Conformance of the design aspects is as stated in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 13.2.1 defines the responsibility to develop and implement training programs for plant personnel. These training programs will address the scope of licensing examinations.
Conformance with Revision 4 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.150, Rev. 1, 2/83 – Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations			
Conformance with the design aspects is as stated below in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (73 FR 7766, 02/11/2008).			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	The reactor vessel design includes features that permit conformance to the pre-service and in-service inspection of this regulatory guide. Guidelines for such features as positioning of welds, vessel contour, and weld surface preparation are included.
Reg. Guide 1.151, (Task 1C 126-5), Rev. 0, 7/83 – Instrument Sensing Lines			
General	ISA-S67.02	Conforms	This regulatory guide addresses the use of ISA-S67.02, “Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants,” 1980, to provide a basis for the design and installation of instrument sensing lines. It establishes the applicable ASME code requirements and boundaries for the design and installation of instrument sensing line in accordance with the appropriate parts of ASME Code, Section III. Where there is a conflict between the ISA standard and the ASME Code, Section III, the ASME Code requirements are followed. Because this instrument configuration (permanently sealed, fluid-filled tubing) is excluded from the scope of Section III, the tubing is designed and fabricated to ASME B31.1. Industry standard ISA-S67.02 reiterates and clarifies the practice of controlling documents such as interface requirements and regulations. The AP1000 uses the Piping and Instrumentation Diagram as the approved document to designate the safety classification system boundaries.
C.1		Conforms	
C.2	ASME Code, Class 2 SC I	Conforms	Safety-related instrumentation has safety class pressure boundaries, including the sensing line, valves, and instrumentation sensors. The pressure boundary is the same safety class as the equipment to which it is connected. The AP1000 credits design features such as flow restrictors and diaphragms as class separation. For that portion of a sensing line from the ASME Code, Class 1 piping or vessel out to the class separation device, ISA-S67.02 includes the ASME Code, Class 1 requirement. For that portion of the sensing line from the class separation device to the sensor is designated as ASME Code, Class 2 requirement. The AP1000 has no sensing lines penetrating the containment barrier.
C.3	ASME Code, Class 3 SC I	Exception	The guidelines apply to the AP1000 sensing lines, except the sensing lines that are connected to some ASME Code, Class 3 components that do not have a seismic design requirement. Sensing lines from these components are not ASME Code, seismic Category I.
C.4-6		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.152, Rev. 1, 1/96 – Criteria for Digital Computers in Safety Systems of Nuclear Power Plants			
Regulatory Guide 1.152, Rev. 2, 1/06 – Criteria for Use of Computers in Safety Systems of Nuclear Power Plants			
Conformance of the design aspects with Revision 1 of the Regulatory Guide is as stated below in the DCD.			
General	ANSI/ IEEE-ANS-7-4.3.2 -1993	Exception	The Common Q portion of the protection and safety monitoring system is developed using the Common Q Software Program Manual (SPM) (as modified by the SPM alternatives in WCAP-15927) and Common Q Topical Report (as modified by the Topical Report alternatives in WCAP-15927). The Common Q SPM and Topical Report were reviewed and approved by the NRC. The Common Q SPM and Topical Report meet IEEE Std. 7-4.3.2-2003, as endorsed by Regulatory Guide 1.152, Revision 3.
Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Exception	The Cyber Security Program is based on March 2009 revisions of the 10 CFR 73.54 regulations in lieu of Revision 2 of this Regulatory Guide.
Reg. Guide 1.153, Rev. 1, 6/96 – Criteria for Power, Instrumentation, and Control Portions of Safety Systems			
General	IEEE Std. 603-1991 including January 30, 1995 Correction sheet	Conforms	
Reg. Guide 1.154, Rev. 0, 1/87 – Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors			
Conformance of the design aspects with the Regulatory Guide is as stated below in the DCD.			
General		N/A	See Section 5.3 for additional information on pressurized thermal shock. Subsection 5.3.6 defines the responsibility to document reactor vessel materials and material evaluation.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.155, (Task SI 501-4), Rev. 0, 8/88 – Station Blackout			
General	10 CFR 50.63	N/A	There are no safety-related ac power sources. Therefore, this regulatory guide is not applicable to the AP1000.
Reg. Guide 1.156, (Task EE 404-4), Rev. 0, 11/87 – Environmental Qualification of Connection Assemblies for Nuclear Power Plants			
General	IEEE Std. 572-1985	Conforms	
Reg. Guide 1.157, (Task RS 701-4), Rev. 0, 5/89 – Best-Estimate Calculations of Emergency Core Cooling System Performance			
C.1		Conforms	
C.2		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3.1	10 CFR 50, App. A	Conforms	
C.3.2-12		Conforms	
C.3.13-14		N/A	Applies to boiling water reactors only.
C.3.15-16		Conforms	
C.4.1	10 CFR 50.46(a)(1)(i)	Conforms	
C.4.2-4		Conforms	
C.4.5		Conforms	
Reg. Guide 1.158, (Task EE 006-5), Rev. 0, 2/89 – Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants			
General	IEEE Std. 535-1986	Conforms	
Reg. Guide 1.159, Rev. 0, 8/90 and Rev. 1, 10/03 – Assuring the Availability of Funds for Decommissioning Nuclear Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.160, Rev. 4, 8/18 – Monitoring the Effectiveness of Maintenance at Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.161, Rev. 0, 6/95 – Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb			
General		N/A	The design and material specification for the reactor vessel do not permit a Charpy value less than 50 ft.-lb. Subsection 5.3.6.4 defines the responsibility for reactor vessel materials properties verification.
Reg. Guide 1.162, Rev. 0, 2/96 – Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.163, Rev. 0, 9/95 – Performance Based Containment Leak-Test Program			
Conformance of the design aspects is as stated below in the DCD.			
1	NEI94-01 ANSI/ ANS 56.8-1994	Conforms	
2	NEI Section 11.3.2	Conforms	
3	NEI 94-01 Section 9.2.1 NEI 94-01 Section 10.2.3.3	Conforms	
Conformance with Revision 0 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.165, Rev. 0, 3/97 – Identification and Characterization of Seismic Sources and Determination Safe Shutdown Earthquake Ground Motion			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Subsection 2.5.2.1 defines the responsibility to address site-specific information related to the vibratory ground motion aspects.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		Conforms with Clarification	Regulatory Guide 1.165 is used to (1) conduct geological, seismological, and geophysical investigations of the site and region around the site, (2) identify and characterize seismic sources, and (3) perform PSHA. The procedure to determine the SSE for the site departs from the Regulatory Guide 1.165 procedure. Site-specific SSE spectra following the procedures of ASCE 43-05 for defining the Design Response Spectra (DRS) using a Target Performance Goal (Pf) of a mean annual probability of exceedance of 1E- 05 is used to define the SSE design ground motion.
Reg. Guide 1.166, Rev. 0, 3/97 – Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 13.5.1 defines the responsibility for the plant procedure preparation.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.167, Rev. 0, 3/97 – Restart of a Nuclear Power Plant Shut Down by a Seismic Event			
Conformance of the design aspects is as stated below in the DCD.			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		N/A	Not applicable to AP1000 design certification. Subsection 13.5.1 defines the responsibility for the plant procedure preparation.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.168, Rev. 0, 9/97 and Rev. 1, 2/04 – Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated below in the DCD.			
General		Exception	See Chapter 7 for a discussion of the instrumentation and control software program related to Common Qualified Platform (Common Q). The Common Q portion of the protection and safety monitoring system is developed using the Common Q SPM (as modified by the SPM alternatives in WCAP-15927). The Common Q SPM was reviewed and approved by the NRC using the criteria of IEEE Std. 1012-1998 and IEEE Std. 1028-1997 as endorsed by Regulatory Guide 1.168, Revision 1.
Conformance of the design aspects with Revision 1 of the Regulatory Guide is as stated below.			
General		Exception	See Chapter 7 for a discussion of the instrumentation and control software program related to the Component Interface Module (CIM) subsystem. Exceptions to the standards endorsed by Regulatory Guide 1.168, Revision 1 (i.e., IEEE 1012-1998 and IEEE 1028-1997) related to the CIM subsystem are noted below.
General	IEEE 1012-1998	Exception	The CIM subsystem takes exception to the requirement in Section 7.6 that states each Independent Verification and Validation (IV&V) Anomaly Report shall contain “recommendations.” The IV&V Anomaly Reports do not require recommendations so that the independence of IV&V is maintained.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	IEEE 1028-1997	Exception	The CIM subsystem takes exception to the requirement in Section 4.5.4 that requires team members to review the relevant materials prior to the management review. The project manager is required to distribute review materials before the meeting, and the material is reviewed during the meeting. The CIM subsystem takes exception to Section 4.6, which requires a management review to be considered complete when specific criteria are met. There is no requirement to officially declare the management review complete. The management review is considered complete at the discretion of the program manager in conjunction with the management review team. However, the CIM subsystem management reviews adhere to the meeting goals and outputs as defined by IEEE 1028, Sections 4.6 and 4.7 (with the exceptions noted in this appendix). The CIM subsystem takes exception to the guidance in Section 4.7, which requires specific criteria for documenting and tracking action item statuses, anomalies, and review team member participation as management review outputs. Contrary to IEEE 1028: • Only significant action items are formally documented and tracked in management meeting minutes. • Anomalies are formally tracked using issue tracking databases. • Members of the review teams are not formally captured as a management review output. The CIM subsystem takes exception to the guidance in Section 8.5.5.1, which requires auditors to perform activities outside the scope of the audit plan, if the activities are required to define the full extent of an issue. Contrary to IEEE 1028, auditors are not directed to perform or document reviews outside of the assigned audit scope or to take actions beyond that required by the audit plan. Issues found during an audit are entered into a corrective action tracking system. The corrective action program formally evaluates the issues and determines if an “extent of condition” investigation is required.

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

General Conforms

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.169, Rev. 0, 9/97 – Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Exception	Westinghouse uses the Common Q SPM (as modified by the SPM alternatives in WCAP-15927) to develop and maintain the Common Q portion of the protection and safety monitoring system. The Common Q SPM was reviewed and approved by the NRC using the criteria of Regulatory Guide 1.169, Revision 0 and IEEE 828-2005. The CIM subsystem complies with Regulatory Guide 1.169, Revision 0 with the exception(s) identified below:
General	IEEE 828-1990	Exception	The CIM subsystem takes exception to the requirement in Section 2.3.4 to perform a configuration audit on a configuration item prior to its release. Configuration audits occur after the release of a configuration item document instead of prior to its release. Configuration audits are complete prior to installation of the Configuration Item in the plant. See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.170, Rev. 0, 9/97 – Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Exception	The Common Q portion of the protection and safety monitoring system is developed using the Common Q SPM (as modified by the SPM alternatives in WCAP-15927). The Common Q SPM was reviewed and approved by the NRC using the criteria of Regulatory Guide 1.170, Revision 0 and IEEE 829-1998. The CIM subsystem complies with Regulatory Guide 1.170, Revision 0 with the exception(s) identified below:
General	IEEE 829-1983	Exception	The CIM subsystem takes exception to the requirement in Section 3 to specify the version/revision level of the test items within the test plan. The CIM Test Plan is authored to be independent of the specific versions/revisions of the test items to be tested and requires the latest approved test plan version to be used.
General	IEEE 829-1983	Exception	The CIM subsystem takes exception to the required structure/format specified for Test-Design Specification (Section 4), Test-Case Specification (Section 5), Test-Procedure Specification (Section 6), Test-Item Transmittal Report (Section 7), Test Log (Section 8), Test-Incident Report (Section 9), and Test-Summary Report (Section 10). The CIM subsystem documents corresponding to the documents referred to in Sections 4 through 10 contain the content required by IEEE 829.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	IEEE 829-1983	Exception	The CIM subsystem takes exception to the requirements in Sections 7.2.2 and 7.2.4 to include a reference to the test plan and deviations from the test plan when transmitting test items. The version/revision of the test item is recorded in the test record. There will be no deviations from the identified software documentation at the time of the software release.
General	IEEE 829-1983	Exception	The CIM subsystem takes exception to the requirement in Section 8 to record the start and end times of a test event. The date of the event is recorded, not the specific start and end times.
General	IEEE 829-1983	Exception	The CIM subsystem takes exception to the requirement in Section 10 to summarize resource consumption data in the test report. Resource consumption data is addressed in project schedules and budget data. See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.171, Rev. 0, 9/97 – Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Conforms	See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.172, Rev. 0, 9/97 – Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Exception	The Common Q portion of the protection and safety monitoring system is developed using the Common Q SPM (as modified by the SPM alternatives in WCAP-15927). The Common Q SPM was reviewed and approved by the NRC using the criteria of Regulatory Guide 1.172, Revision 0 and IEEE 830-1998. See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.173, Rev. 0, 9/97 – Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Exception	Westinghouse uses the Common Q SPM (as modified by the SPM alternatives in WCAP-15927) to develop and maintain the Common Q portion of the protection and safety monitoring system. The Common Q SPM was reviewed and approved by the NRC using the criteria of IEEE 1074-1995 as endorsed by Regulatory Guide 1.173, Revision 0. The CIM subsystem complies with Regulatory Guide 1.173, Revision 0 with the exception(s) identified below.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	IEEE 1074-1995	Exception	The CIM subsystem takes exception to the requirements in Sections 4.1.4, 4.1.5, and 4.1.7, respectively, to perform the "Formulate Potential Approaches" and "Constraints and Benefits" activity; the "Conduct Feasibility Study" activity; and the "Refine and Finalize Idea or Need" activity. Using information from previous projects, the basic CIM subsystem design approach was already formulated at the time the project was initiated.
General	IEEE 1074-1995	Exception	The CIM subsystem takes exception to the requirement in Section 7.4.3 to consider the software requirements as an input to the development of the training program. The training plan was written to be independent of the software requirements developed later in the project. See Chapter 7 for a discussion of the instrumentation and control software program.

Reg. Guide 1.174, Rev. 0, 7/98 and Rev. 1, 11/02 – An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis

Conformance of the design aspects for Revision 0 of this Regulatory Guide is as stated below in the DCD.

General	N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

This Regulatory Guide is outside the scope of the FSAR.

Reg. Guide 1.175, Rev. 0, 7/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing

Regulatory Guide 1.175, Rev. 0, 8/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing

Conformance of the design aspects is as stated below in the DCD.

General	N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Inservice testing of ASME Section III components is discussed in Subsection 3.9.6.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

Risk-informed inservice testing is not being utilized for this plant.

Reg. Guide 1.176, Rev. 0, 8/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance

General	N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Quality assurance is discussed in Chapter 17.

Reg. Guide 1.177, Rev. 0, 8/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications

Conformance of the design aspects is as stated below in the DCD.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. The standard AP1000 Technical Specifications are provided in DCD Chapter 16.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

This Regulatory Guide is outside the scope of the FSAR.

Reg. Guide 1.178, Rev. 0, 9/98 and Rev. 1, 9/03 – An Approach for Plant-Specific Risk-informed Decisionmaking Inservice Inspection of Piping

Conformance of the design aspects is as stated below in the DCD.

General	N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Inservice inspection is discussed in Subsection 5.2.4 and Section 6.6.

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

Risk-informed inservice inspection is not being utilized for this plant.

Reg. Guide 1.179, Rev. 0, 1/99 – Standard Format and Content of License Termination Plans for Nuclear Power Reactors

Conformance of the design aspects is as stated below in the DCD.

General	N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

	N/A	This Regulatory Guide is outside the scope of the FSAR.

Reg. Guide 1.180, Rev. 1, 10/03 – Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems

Conformance of the design aspects is as stated below in the DCD.

General	Conforms	See Appendix 3D for a discussion of the EMI/RFI qualification.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General	Conforms	Exclusion zones are established through administrative controls to prohibit the activation of portable EMI/RFI emitters (e.g., welders and transceivers) in areas where safety-related I&C systems are installed.

Reg. Guide 1.181, Rev. 0, 9/99 – Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)

Conformance of the design aspects is as stated below in the DCD.

General	N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General	Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.182 – Withdrawn			
Reg. Guide 1.183, Rev. 0, 7/00 – Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.184, Rev. 0, 7/00 – Decommissioning of Nuclear Power Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.185, Rev. 0, 8/00 – Standard Format and Content for Post-shutdown Decommissioning Activities Report			
Regulatory Guide 1.185, Rev. 0, 7/00 – Standard Format and Content for Post-shutdown Decommissioning Activities Report			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.186, Rev. 0, 12/00 – Guidance and Examples of Identifying 10 CFR 50.2 Design Bases			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.187, Rev. 0, 11/00 – Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	
Regulatory Guide 1.188, Rev. 1, 9/05 – Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 1.189, Rev. 1, 3/07 – Fire Protection for Operating Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Subsection 9.5.1 describes the AP1000 Fire Protection System. Subsection 9.5.1.8 defines the responsibility for completing a fire protection program.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	Subsection 9.5.1.8 defines the protection program.
Reg. Guide 1.190, Rev. 0, 3/01 – Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence			
General		Exception	Subsection 5.3.2.6 describes the calculational and dosimetry methods for determining pressure vessel neutron fluence for the AP1000. Subsections 5.3.2.6.3 and 5.3.3.1 describe the reporting of the test results. Subsection 5.3.6.4 describes the reactor vessel materials properties verification.
Regulatory Guide 1.191, Rev. 0, 5/01 – Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 1.192, Rev. 3, 10/19 – Operation and Maintenance Code Case Acceptability, ASME OM Code			
General		Conforms	
Regulatory Guide 1.193, Rev. 1, 8/05– ASME Code Cases Not Approved for Use			
This Regulatory Guide is not applicable to the AP1000 certified design. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.194, Rev. 0, 6/03 – Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants			
Conformance with Revision 0 of the Regulatory Guide is documented in the DCD. Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.195, Rev. 0, 5/03 – Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors			
This Regulatory Guide is not applicable to the AP1000 certified design.			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Regulatory Guide 1.196, Rev. 1, 1/07 – Control Room Habitability at Light-Water Nuclear Power Reactors			
This Regulatory Guide is not applicable to the AP1000 certified design. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.197, Rev. 0, 5/03 – Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors			
Conformance with the design aspects is as stated in the DCD.			
General		Conforms	The design of the AP1000 control room and associated HVAC systems facilitates testing to demonstrate control room envelope integrity.
C.1.1-2.6		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for procedures. Section 14.4 describes the responsibilities required to perform the AP1000 plant initial test program.
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.198, Rev. 0, 11/03 – Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites			
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.199, Rev. 0, 11/03 – Anchoring Components and Structural Supports in Concrete			
Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.			
C.1 – C.7		Conforms	
Regulatory Guide 1.200, Rev. 2, 3/09 – An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities			
This Regulatory Guide is not applicable to the AP1000 certified design. Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.201, Rev. 1, 5/06 – Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance			
This Regulatory Guide is not applicable to the AP1000 certified design.			
Regulatory Guide 1.202, Rev. 0, 2/05 – Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors			
This Regulatory Guide is outside the scope of the FSAR.			
Regulatory Guide 1.203, Rev. 0, 12/05 – Transient and Accident Analysis Methods			
This Regulatory Guide is not applicable to the AP1000 certified design.			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Regulatory Guide 1.204, Rev. 0, 11/05 – Guidelines for Lightning Protection of Nuclear Power Plants			
This Regulatory Guide is not applicable to the AP1000 certified design. Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.205, Rev. 0, 5/06 – Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants			
This Regulatory Guide is not applicable to the AP1000 certified design.			
Regulatory Guide 1.206, Rev. 1, 10/18 – Applications for Nuclear Power Plants			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General	Format	Conforms	
General	Content	Exception	Exceptions to content are identified in the application and were reviewed and approved by the NRC staff. Applicable for Combined License Application phase. This position statement is not being updated after the COL is issued.
Regulatory Guide 1.207, Rev. 0, 3/07 – Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors			
This Regulatory Guide is not applicable to the AP1000 certified design.			
Regulatory Guide 1.208, Rev. 0, 3/07 – A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion			
This Regulatory Guide is not applicable to the AP1000 certified design.			
		N/A	Performance-based analysis per RG 1.165 and ASCE 43-05.
Regulatory Guide 1.209, Rev. 0, 3/07 – Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants			
This Regulatory Guide is not applicable to the AP1000 certified design.			
DIVISION 4 – Environmental and Siting			
Regulatory Guide 4.2 & Supp. 1, Rev. 2, 7/76, S-1, 9/00 – Preparation of Environmental Reports for Nuclear Power Stations			
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 4.7 Rev. 2, 4/98 – General Site Suitability Criteria for Nuclear Power Stations			
Conformance with the design aspects is as stated below in the DCD.			

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		N/A	Chapter 2 defines the site-related parameters for which the AP1000 plant is designed. These interface parameters envelop most potential sites in the United States. The guidelines in this regulatory guide are site-specific. Chapter 2 defines the responsibility for determining general site suitability.
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 4.15 Rev. 2, 7/07 – Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) – Effluent Streams and the Environment			
		Exception	The Guidance of Rev. 1, February 1979 will be followed as per the justification provided in Subsection 11.5.3.
DIVISION 5 – Materials and Plant Protection The plant-specific physical security plans include no substantive deviations from the NRC-endorsed template in NEI 03-12, Rev. 6. Therefore, the degree of conformance with Division 5 regulatory guides for the Physical Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan is consistent with the degree of conformance of NEI 03-12, Rev. 6.			
Reg. Guide 5.9 Rev. 2, 12/83 – Specifications for Ge (Li) Spectroscopy Systems for Material Protection Measurements Part 1: Data Acquisition Systems			
Regulatory Guide 5.9, Rev. 2, 12/83 – Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Laboratory Equipment is not included in the AP1000 design. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 5.12, Rev. 0, 11/73 – General Use of Locks in the Protection and Controls of Facilities and Special Nuclear Materials			
Conformance of the design aspects is as stated below in the DCD.			
C.1	UL-768	Conforms	
C.2	FF-P-110F	Conforms	
C.3	UL-437	Conforms	
C.4	FF-P-001480 (GSA FSS)	Conforms	
C.5-8		Conforms	
Conformance with programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 5.65, Rev. 0, 9/86 – Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	The AP1000 provides for physical protection of the vital area. Identification of the protected and vital areas and an outline of the physical protection system are presented in the AP1000 Security Design Report.
Conformance with programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 5.71, Rev. 0, 1/10 – Cyber Security Programs for Nuclear Facilities			
Conformance with regulatory positions C.1 through C.5 of Regulatory Guide 5.71, Rev. 0 is as stated in the Cyber Security Plan (CSP), with exceptions to the guidance as noted in Attachment A of the CSP.			
Regulatory Guide 5.84, Rev. 0, 7/15 - Fitness-For-Duty Programs at New Reactor Construction Sites			
Conformance with programmatic and/or operational aspects is documented below:			
General	10 CFR Part 26 NEI 06-06, Rev. 6 (April 2013)	Conforms	
DIVISION 8 – Occupational Health			
Reg. Guide 8.2, Rev. 0, 2/73 – Guide for Administrative Practices in Radiation Monitoring			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for administrative procedures. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
Conformance with programmatic and/or operational aspects is documented below.			
General	10 CFR Part 20; ANSI 13.2-1969	Exception	The reference to 10 CFR 20.401 is no longer valid in the current version of 10 CFR Part 20. ANSI N13.2-1969 was reaffirmed in 1988.
Regulatory Guide 8.4, Rev. 0, 2/73 - Direct-Reading and Indirect-Reading Pocket Dosimeters			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General	10 CFR Part 20 ANSI N13.5-1972	Exception	The reference to 10 CFR 20.202 (a) and 20.401 is no longer valid in the current version of 10 CFR Part 20. ANSI N13.5-1972 was reaffirmed in 1989. The two performance criteria specified in Regulatory Guide 8.4 (accuracy and leakage) for these devices are met using acceptance standards in ANSI N322-1997 “American National Standard Inspection, Test, Construction, and Performance Requirements for Direct Reading Electrostatic/Electroscope Type Dosimeters”.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Regulatory Guide 8.5, Rev. 1, 3/81 - Withdrawn			
Regulatory Guide 8.6, Rev. 0, 5/73 - Standard Test Procedure for Geiger-Muller Counters			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General		Exception	Instrument calibration program is based upon criteria in ANSI N323A-1997 (with 2004 Correction Sheet) “Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments.” The ANSI 42.3-1969 Standard is no longer recognized as sufficient for calibration of modern instruments.
Regulatory Guide 8.7, Rev. 2, 11/05 - Instructions for Recording and Reporting Occupational Radiation Dose Data			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.8, Rev. 3, 6/78 – Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable			
Conformance of the design aspects is as stated below in the DCD.			
1		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
1.a-c	Regulatory Guide 1.8	N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
1.d		Conforms	
2	ANSI N237-1976	Exception	Regulatory Guide 8.8 endorses ANSI-N237-1976 (Reference 49), which has been superseded by ANSI 18.1-1999 (Reference 50). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 8.8.
2.a	10 CFR 20-203	Conforms	
2.b-d		Conforms	
2.e		Conforms	See Subsection 12.3.1.1.1, “Materials,” and corresponding Table 12.3-1.
2.f-g		Conforms	
2.h	ANS N197 ANS 55.1 ANS N19	Conforms	ANS-55.1-1992-R2000 is Current Version
2.i		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
3		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
4.a		Conforms	
4.b-d		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
4.e		Conforms	
Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.1		Conforms	
C.3.a		Conforms	
C.3.b		Exception	Regulatory Guide 1.16 C.1.b.(3) data is no longer reported. Reporting per C.1.b(2) is also no longer required.
C.3.c		Conforms	
C.4.b-C.4.d	ANSI Z-88.2, Regulatory Guide 8.15, NUREG-0041	Conforms	Conformance is with the latest revision of NUREG-0041.
Regulatory Guide 8.9, Rev. 1, 7/93 – Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.10, Rev. 1-R, 5/77 – Operating Philosophy For Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.12 – Withdrawn			
Reg. Guide 8.13, Rev. 3, 6/99 – Instruction Concerning Prenatal Radiation Exposure			
Conformance of the design aspects is as stated below in the DCD.			
General	10 CFR 19.12	N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA. Section 13.5 defines the responsibility for administrative procedures.

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.14 – Withdrawn			
Reg. Guide 8.15, Rev. 1, 10/99 – Acceptable Programs for Respiratory Protection			
Conformance of the design aspects is as stated below in the DCD.			
General	10 CFR 20.103	N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA. See Section 12.3 for information on radiation protection design features. See Section 12.5 for information on health physics facilities. Section 13.5 defines the responsibility for administrative procedures.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.19, Rev. 1, 6/79 – Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates			
General		Conforms	
Regulatory Guide 8.27, Rev. 0, 3/81 – Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 8.28, Rev. 0, 8/81 – Audible-Alarm Dosimeters			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General	ANSI N13.27-1981	Conforms	
Regulatory Guide 8.29, Rev. 1, 2/96 – Instruction Concerning Risks from Occupational Radiation Exposure			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 8.34, Rev. 0, 7/92 – Monitoring Criteria and Methods To Calculate Occupational Radiation Doses			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 8.35, Rev. 0, 6/92 – Planned Special Exposures			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	

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Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Regulatory Guide 8.36, Rev. 0, 7/92 – Radiation Dose to the Embryo/Fetus			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
Reg. Guide 8.38, Rev. 0, 6/93 and Rev. 1, 5/06 – Control of Access to High and Very High Radiation Areas of Nuclear Plants			
Conformance of the design aspects for Revision 0 of this Regulatory Guide is as stated below in the DCD.			
General		Conforms	
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	

Note a. Not used.

Note b. Above stated general alternatives regarding the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD is provided to preserve the finality of the certified design. Further, each stated conformance with the programmatic and/or operational aspects is only to the extent that a design change or departure from the approved DCD is not required to implement those programmatic and/or operational aspects. As the operational and programmatic aspects become more fully defined (for example, during the preparation, approval, or initial implementation of plant procedures), there exists a potential that a conflict could be identified between the design as certified in the DCD and the programmatic and/or operational aspects of the guidance. In such cases, the design certification (rule) becomes the controlling factor, and the design conformance to the Regulatory Guide is per the revision stated in the DCD.

Note c. A “Criteria Section” entry of “General” indicates a scope for the conformance statement of “all regulatory guide positions related to programmatic and/or operational aspects.” Thus, an associated conformance statement of “Conforms” indicates that the applicant “complies with all regulatory guide positions related to programmatic and/or operational aspects.”

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1A.1 References

1. Sofer, et al., "Accident Source Terms for Light Water Nuclear Power Plants." NUREG-1465, February 1995.

2. Not used.

3. Not used.

4. Not used.

5. IEEE Std. 308-1974, IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations, 1974.

6. IEEE Std. 308-1991, IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations, 1991.

7. Not used.

8. Not used.

9. WCAP-8577, "The Application of Pre-Heat Temperature After Welding of Pressure Vessel Steels," September 1975.

10. IEEE 379-72, IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems, 1972.

11. IEEE 379-2000, IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems, 2000.

12. ASCE 4-86, Seismic Analysis of Safety-Related Nuclear Structures, September 1986.

13. RESAR/SP90, Westinghouse Advanced Pressurized Water Reactor, Paragraph 3.7.1.3, Damping Values, January 1989.

14. Design & Evaluation Guideline for Department of Energy Facilities Subjected to Natural Phenomena Hazards, Lawrence Livermore National Laboratory, UCRL-15910, R. P. Kennedy et al., May 1989.

15. IEEE 741-1986, IEEE Standard Criteria for Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations, 1986.

16. IEEE Std. 741-1997, IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations, 1997.

17. IEEE 317-1983, IEEE Standard for Electric Penetrations Assemblies in Containment Structures for Nuclear Power Generating Stations, 1983.

18. ANSI N101.6-1972, Atomic Industry Facility Design, Construction, and Operation Criteria, December 22, 1972.

19. ANSI/ANS 6.4 1997, Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants.

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20. Not used.

21. IEEE 382-1996, IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants, 1996.

22. IEEE 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, 1974.

23. IEEE 384-74, IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits, 1974.

24. IEEE 384-81, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits, 1981.

25. ANS 5.4-1982, American National Standard Method for Calculating the Fractional Release of Volatile Fission Products From Oxide Fuel, 1982.

26. Title 10 Code of Federal Regulations Part 50, 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants.

27. IEEE Std. 279-1971, IEEE Standard Criteria for Protection Systems for Nuclear Power Generating Stations, 1971.

28. NRC Safety Evaluation Report, letter from B. D. Liaw to J. A. Martin, December 27, 1984.

29. IEEE 384-2008, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits, 2008.

30. Not used.

31. IEEE Std. 338-1987, IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Stations Safety Systems, 1987.

32. NUREG-0800, U.S. Nuclear Regulatory Commission Standard Review Plan, Section 9.5.1, “Fire Protection Program,” Revision 3, July 1981, including Branch Technical Position (BTP) CMEB 9.5-1, “Guidelines for Fire Protection for Nuclear Power Plants,” Revision 2, July 1981.

33. SRP (NUREG-0800) Section 9.5.1 "Fire Protection Program," Revision 3, July 1986.

34. Guide to Design Criteria for Bolted and Riveted Joints, Fisher and Struik, copyright 1974, John Wiley and Sons.

35. Not used.

36. IEEE 484-75, IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations, 1975.

37. IEEE 484-1996, IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations, 1996.

38. ASME AG-1-1997, "Code on Nuclear Air and Gas Treatment," 1997.

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39. ASME Code N509-1989, "Nuclear Power Plant Air Cleaning Units and Components," 1989.

40. ASME Code N510-1989, "Testing of Nuclear Air Cleaning Systems," 1989.

41. ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems," 1976.

42. ANS 56.2-1984, Containment Isolation Provisions for Fluid Systems, 1984.

43. ACI 349-97, "Code Requirements for Nuclear Safety Related Concrete Structures," 1997.

44. ACI 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures," 2001.

45. Not used.

46. ACI 318-99, "Building Code Requirements for Reinforced Concrete," 1999.

47. Not used.

48. Not used.

49. ANSI N237-1976, Source Term Specification, 1976.

50. ANSI/ANS 18.1-1999, Radioactive Source Term for Normal Operation of Light Water Reactors, 1999.

51. IEEE Std. 603-1991, IEEE Standard Criteria for Power Generating Stations, 1991.

52. WCAP-16650-P (Proprietary) and WCAP-16650-NP (Non-Proprietary), "Analysis of the Probability of Generation of Missiles for Fully Integral Low Pressure Turbines," Revision 0, February 2007.

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Appendix 1B Severe Accident Mitigation Design Alternatives

1B.1 AP1000 SAMDA Evaluation

1B.1.1 Introduction

This response provides an evaluation of Severe Accident Mitigation Design Alternatives (SAMDA) for the Westinghouse AP1000 design. This evaluation is performed to evaluate whether or not the safety benefit of the SAMDA outweighs the costs of incorporating the SAMDA in the plant, and is conducted in accordance with applicable regulatory requirements as identified below.

The National Environmental Policy Act (NEPA), Section 102.(C)(iii) requires, in part, that:

> ... all agencies of the Federal Government shall ... (C) include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on ... (iii) alternatives to the proposed action.

The 10 CFR 52.47(a)(ii) requires an applicant for design certification to demonstrate:

> ... compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f) ...

A relevant requirement of 10 CFR 50.34(f) contained in subparagraph (1)(i) requires the performance of:

> ... a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant …

In SECY-91-229, the U.S. Nuclear Regulatory Commission (NRC) staff recommends that SAMDAs be addressed for certified designs in a single rulemaking process that would address both the 10 CFR 50.34 (f) and NEPA considerations in the 10 CFR Part 52 design certification rulemaking. SECY-91-229 further recommends that applicants for design certification assess SAMDAs and the applicable decision rationale as to why they will or will not benefit the safety of their designs. The Commission approved the staff recommendations in a memorandum dated October 25, 1991 (Reference 1).

1B.1.2 Summary

Note that the AP1000 is similar to the AP600, which has received Design Certification. The evaluation for AP1000 uses the conclusions of the AP600 SAMDA investigation as described below. An evaluation of candidate modifications to the AP600 design was conducted to evaluate the potential for such modifications to provide significant and practical improvements in the radiological risk profile of the AP600 design. Since the AP1000 is so similar to the AP600, the list of candidate modifications is the same.

The process used for identifying and selecting candidate design alternatives included a review of SAMDAs evaluated for other plant designs. Several SAMDA designs evaluated previously for other plants were excluded from the present evaluation because they have already been incorporated or otherwise addressed in the AP600 and AP1000 designs. These include the following:

- Hydrogen ignition system
- Reactor cavity flooding system

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- Reactor coolant pump seal cooling
- Reactor coolant system depressurization
- Reactor vessel exterior cooling

Additional design alternatives were identified based upon the results of the AP600 probabilistic risk assessment (Reference 3). The AP1000 probabilistic risk results are similar to those developed for the AP600. Fifteen candidate design alternatives were selected for further evaluation.

An evaluation of these alternatives was performed using a bounding methodology such that the potential benefit of each alternative is conservatively maximized. As part of this process, it was assumed that each SAMDA performs beyond expectations and completely eliminates the severe accident sequences that the design alternative addresses. In addition, the capital cost estimates for each alternative were intentionally biased on the low side to maximize the risk reduction benefit. This approach maximizes the potential benefits associated with each alternative.

The results show, for the AP600 and AP1000, that despite the significant conservatism used in the evaluation, none of the SAMDAs evaluated provide risk reductions that are cost beneficial. The results also show that even a conceptual "ideal SAMDA," one which reduces the total plant radiological risk to zero, would not be cost effective. This is due primarily to the already low-risk profile of the AP600 and AP1000 designs.

1B.1.3 Selection and Description SAMDAs

Candidate design alternatives were selected based upon design alternatives evaluated for other plant designs (References 4, 5, and 6) as well as suggestions from AP600 and AP1000 design personnel. Additional candidate design alternatives were selected based upon an assessment of the AP600 and AP1000 probabilistic risk assessment results. SAMDA design alternatives were finally selected for further evaluation. These SAMDAs are as follows:

- Chemical, volume, and control system (CVS) upgraded to mitigate small loss-of-coolant accidents (LOCAs)
- Filtered containment vent
- Normal residual heat removal system (RNS) located inside containment
- Self-actuating containment isolation valves
- Passive containment spray
- Active high-pressure safety injection system
- Steam generator shell-side passive heat removal system
- Steam generator safety valve flow directed to in-containment refueling water storage tank (IRWST)
- Increase of steam generator secondary side pressure capacity
- Secondary containment filtered ventilation
- Diverse IRWST injection valves

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- Diverse containment recirculation valves
- Ex-vessel core catcher
- High-pressure containment design
- Diverse actuation system improved reliability.

Each SAMDA and the benefit expected due to the modification is described below. In the evaluation of the risk reduction benefit, each SAMDA is assumed to operate perfectly with 100 percent efficiency, without failure of supporting systems. A perfect SAMDA reduces the frequency of accident sequences, which it addresses to zero. This is conservative as it maximizes the benefit of each design alternative. The SAMDA will reduce the risk by lowering the frequency, attenuating the release, or both. The benefit will be described in terms of the accident sequences and dose, which are affected by the SAMDAs, as well as the overall risk reduction. For these evaluations, increases to release category IC are not factored into the risk benefit calculations. The IC dose is sufficiently small that changes to the IC total frequency do not result in an appreciable change to overall results. This is also a conservative representation since this maximizes the risk reduction.

The cost benefit methodology of NUREG/BR-0184 (1997) is used to calculate the maximum attainable benefit. This includes replacement power costs. For expected benefit, the change in the CDF frequency (delta-F) is assumed to be equal to the sum of CDF frequencies from internal, external, and shutdown events that are already evaluated. This is bounding, used to calculate the maximum attainable benefit. In practice, there is no design alternative, or SAMDA strategy, whose implementation would reduce the plant CDF to zero (or to an infinitesimally small frequency).

Upgrade Chemical, Volume, and Control System for Small LOCAs

The chemical, volume, and control system is currently capable of maintaining the reactor coolant system inventory to a level in which the core remains covered in the event of a very small (< 3/8 inch diameter break) LOCA. This SAMDA involves providing IRWST containment recirculation connections to the chemical, volume, and control system and adding a second line from the chemical, volume, and control system makeup pumps to the reactor coolant system to be able to use the system to keep the core covered during small and intermediate LOCAs.

A perfect, upgraded chemical, volume, and control system is assumed to prevent core damage in the reactor coolant system leak, passive residual heat removal heat exchanger tube ruptures, small LOCA, and intermediate LOCA release categories. The chemical, volume, and control system is assumed to have perfect support systems (power supply and component cooling) and to work in all situations regardless of the common cause failures of other systems.

Filtered Vent

This SAMDA consists of placing a filtered containment vent and all associated piping and penetrations into the AP1000 containment design. The filtered vent could be used to vent the containment to prevent catastrophic overpressure failure, and it also provides filtering capability for source term release. With respect to the AP1000 Probabilistic Risk Assessment, the possible scenario in which the filtered vent could result in risk reduction would be late containment overpressure failures (release category CFL). Other containment overpressure failures occur due to dynamic severe accident phenomena, such as hydrogen burn and steam explosion. The late containment failures for AP1000 are failures of the passive containment cooling system. Analyses have indicated that for scenarios with passive containment cooling system failure, air cooling may limit the containment pressure to less than the ultimate pressure. However, for the Level 2 probabilistic risk assessment, failure of the passive containment cooling system is assumed to result

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in containment failure based on an adiabatic heatup. To conservatively consider the risk reduction of a filtered vent, the use of a filtered vent to preclude a late containment failure will be evaluated. A decontamination factor (DF) of 1000 will conservatively be assumed for each probabilistic risk assessment Level 1 accident classification, even though it is realized that the dose due to noble gases will not be impacted by the filtered vent since 100 percent of the noble gas fission products will still be released. Therefore, the risk reduction is equal to the decontamination factor assumed since the probabilistic risk assessment Level 1 accident classification frequencies do not change.

Self-Actuating Containment Isolation Valves

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as normal residual heat removal system and component cooling. The design alternative would be to add a self-actuating valve or enhance the existing inside containment isolation valve to provide for self-actuation in the event that containment conditions are indicative of a severe accident. Conceptually, the design would be either an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve. This provides the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures that occur at the time of the accident, such as in cases of vessel rupture or anticipated transients without scram.

Passive Containment Sprays

This SAMDA involves adding a passive safety-related spray system and all associated piping and support systems to the AP1000 containment. A passive containment spray system could result in risk benefits in the following ways:

- Scrubbing of fission products could be done primarily for CI failures.
- Assuming appropriate timing, containment spray could be used as an alternate means for flooding the reactor vessel (in-vessel retention) and for debris quenching should vessel failure occur.
- Containment spray could also be used to control containment pressure for cases in which passive containment cooling system has failed.

In order to envelop these potential risk benefits, the risk reduction evaluation will assume that containment sprays are perfectly effective for each of these benefits, with the exception of fission product scrubbing for containment bypass. Thus, the risk reduction can be conservatively estimated by assuming all release categories except BP are eliminated.

Active High-Pressure Safety Injection System

This SAMDA consists of adding a safety-related active high-pressure safety injection pump and all associated piping and support systems to the AP1000 design. A perfect high-pressure safety injection system is assumed to prevent core melt for all events but excessive LOCA and anticipated transients without scram. Therefore, to estimate the risk reduction, only the contributions to each release category of Level 1 accident classes 3C (vessel rupture) and 3A (anticipated transients without scram) need to be considered. This SAMDA would completely change the design approach from a plant with passive safety systems to a plant with passive plus active safety-related systems, and it is not consistent with design objectives.

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Steam Generator Shell-Side Heat Removal System

This SAMDA consists of providing a passive safety-related heat removal system to the secondary side of the steam generators. The system would provide closed loop cooling of the secondary using natural circulation and stored water cooling. This prevents a loss of primary heat sink in the event of a loss of startup feedwater and passive residual heat removal heat exchanger. A perfect secondary heat removal system would eliminate transients from each of the release categories. In order to evaluate the benefit of this SAMDA, the frequencies of all the transient sequences are subtracted from the overall frequency of each of the release categories and the risk is recalculated.

Direct Steam Generator Relief Flow to the In containment Refueling Water Storage Tank

This SAMDA consists of providing all the piping and valves required for redirecting the flow from the steam generator safety and relief valves to the IRWST. An alternate, lower cost option of this SAMDA consists of redirecting only the first-stage safety valve to the IRWST. This system would prevent or reduce fission product release from bypassing the containment in the event of a steam generator tube rupture event. In order to evaluate the benefit from this SAMDA (both options), this design change is assumed to eliminate the BP release category.

Increased Steam Generator Pressure Capability

This SAMDA consists of increasing the design pressure of the steam generator secondary side and safety valve set point to the degree that a steam generator tube rupture will not cause the secondary system safety valve to open. The design pressure would have to be increased sufficiently such that the combined heat capacity of the secondary system inventory and the passive residual heat removal system could reduce the reactor coolant system temperature below T_{sat} for the secondary design pressure. Although specific analysis would have to be performed, it is estimated that the design pressure would have to be increased several hundred psi. This design would also prevent the release of fission products that bypass the containment via the steam generator tube rupture.

Secondary Containment Filtered Ventilation

This SAMDA consists of providing the middle and lower annulus (below the 135’-3" elevation) of the secondary concrete containment with a passive annulus filter system to for filtration of elevated releases. The passive filter system is operated by drawing a partial vacuum on the middle annulus through charcoal and HEPA filters. The partial vacuum is drawn by an eductor with motive flow from compressed gas tanks. The secondary containment would then reduce particulate fission product release from any failed containment penetrations (containment isolation failure). In order to evaluate the benefit from such a system, this design change is assumed to eliminate the CI release category.

Diverse In containment Refueling Water Storage Tank Injection Valves

This SAMDA consists of changing the IRWST injection valve designs so that two of the four lines use diverse valves. Each of the four lines is currently isolated by a squib valve in series with a check valve. In order to provide diversity, the valves in two of the lines will be provided by a different vendor. For the check valves, alternate vendors are available. However, it is questionable if check valves of different vendors would be sufficiently different to be considered diverse unless the type of check valve was changed from the current swing disk check to another type. The swing disk type is the preferred type for this application and other types are considered to be less reliable. Squib valves are specialized valve designs for which there are few vendors. A vendor may not be willing to design, qualify, and build a reasonable squib valve design for this AP1000 application considering that they would only supply two valves per plant. As a result, this SAMDA is not really practicable because of the uncertainty in availability of a second squib valve design/vendor and because of the uncertainty in the reliability of another check valve type. However, the cost estimate for this SAMDA assumes that a

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second squib valve vendor exists and that the vendor provides only the two diverse IRWST squib valves. The cost impact does not include the additional first time engineering and qualification testing that will be incurred by the second vendor. Those costs are expected to be more than a million dollars.

This change will reduce the frequency of core melt by eliminating the common cause failure of the IRWST injection. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of IRWST injection are assumed to be averted. Core damage sequences resulting from a failure of IRWST injection correspond to probabilistic risk assessment Level 1 accident classification 3BE; thus, release category 3BE is eliminated.

Diverse Containment Recirculation Valves

This SAMDA consists of changing the containment recirculation valve designs so that two out of the four lines use diverse valves. Each of the four lines currently contains a squib valve; two of the lines contain check valves, and the other two contain motor-operated valves. In order to provide diversity, the squib valves in two lines will be made diverse. This change will reduce the frequency of core melt by eliminating the common cause failure of the containment recirculation. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of containment recirculation are assumed to be averted. Core damage sequences resulting from failure of containment recirculation correspond to probabilistic risk assessment Level 1 accident classification 3BL; thus, release category 3BL is eliminated.

In the AP1000 design for recirculation, valve diversity has been introduced to reduce some of the dominant failure modes that were discovered for the AP600.

The four AP600 recirculation squib valves were of the "low pressure" type and were a part of a single common cause group. In the AP1000, two of these valves that are in series with check valves are designated to be of "high pressure" type, which are in a common cause group with the same design of valves on the IRWST injection lines. Thus, the common cause failure mode that fails all four recirculation lines in the AP600 is eliminated, and it is replaced with the product of two common cause failure modes, one applicable to the group of six high pressure squib valves and the other to the two low pressure squib valves. This design change helps in reduction of recirculation failures.

Ex-Vessel Core Catcher

This SAMDA consists of designing a structure in the containment cavity or using a special concrete or coating that will inhibit core-concrete interaction (CCI), even if the debris bed dries out. A perfect core catcher would prevent CCI for all cases. However, the AP1000 incorporates a wet cavity design in which ex-vessel cooling is used to maintain the core debris in the vessel to prevent ex-vessel phenomena, such as CCI. Consequently, containment failure due to CCI is not considered in detail for the AP1000 Level 2 probabilistic risk assessment. For cases in which reactor vessel flooding is failed, it is assumed that containment failure occurs due to ex-vessel steam explosion or CCI. This containment failure is assumed to be an early containment failure, CFE (due to ex-vessel steam explosion) even though CCI and basemat melt-through would be a late containment failure. To conservatively estimate the risk reduction of an ex-vessel core catcher, this design change is assumed to eliminate the CFE release category.

High-Pressure Containment Design

This SAMDA design consists of using the massive high-pressure containment design in which the design pressure of the containment is approximately 300 psi (20 bar) for the AP1000 containment. The massive containment design has a passive containment cooling feature much like the AP1000 containment. The high design pressure is considered only for prevention of containment failures due

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to severe accident phenomena, such as steam explosions and hydrogen detonation. A perfect high-pressure containment design would reduce the probability of containment failures, but would have no reduction of the frequency or magnitude of the release from an unisolated containment (containment isolation failure or containment bypass). To estimate the risk reduction of a high pressure containment design, this design is assumed to eliminate the CFE, CFI, and CFL release categories.

Increase Reliability of Diverse Actuation System

This SAMDA design consists of improving the reliability of the diverse actuation system, which actuates engineered safety features and allows the operator to monitor the plant status. The design change would add a third instrumentation and control cabinet and a third set of diverse actuation system instruments to allow the use of two-out-of-three logic instead of two out-of-two logic. Other changes, such as adding another set of batteries, have not been included in the cost estimates. A perfectly reliable diverse actuation system would reduce the frequency of the release categories by the cumulative frequencies of all sequences in which diverse actuation system failure leads to core damage. In order to evaluate the benefit from the diverse actuation system upgrade, a Level 1 sensitivity analysis assuming perfect reliability of diverse actuation system was completed.

Locate Normal Residual Heat Removal Inside Containment

This SAMDA consists of placing the entire normal residual heat removal system and piping inside the containment pressure boundary. Locating the normal residual heat removal system inside the containment would prevent containment bypass due to interfacing system LOCAs (ISLOCA) of the residual heat removal system. In past probabilistic risk assessments of current generation nuclear power plants, the ISLOCA is the leading contributor of plant risk because of large offsite consequences. A failure of the valves which isolate the low-pressure residual heat removal system from the high pressure reactor coolant system causes the residual heat removal system to overpressurize and fail, releasing reactor coolant system coolant outside the containment where it cannot be recovered for recirculation cooling of the core. The result is core damage and the direct release of fission products outside the containment.

In the AP1000, the normal residual heat removal system is designed with a higher design pressure than the systems in current pressurized water reactors, and an additional isolation valve is provided in the design. In the probabilistic risk assessment, no ISLOCAs contribute significantly to the core damage frequency (CDF) of the AP1000 (Reference 2, Chapter 33). Therefore, relocating the normal residual heat removal system of the AP1000 inside containment will provide virtually no risk reduction benefit and will not be investigated further in terms of cost.

1B.1.4 Methodology

The severe accident mitigation design alternatives analysis uses a bounding methodology such that the benefit is conservatively maximized and the capital cost is conservatively minimized for each SAMDA.

1B.1.4.1 Total Population Dose

To assess the potential benefits associated with a design alternative, estimates are made of the offsite population doses resulting from each of the release categories (that is, source terms). MACCS2 version 1.12 (Reference 9) is used for the analysis. The NRC sponsored the development of this code. The code performs probabilistic estimates of offsite consequences from potential accidental releases in conformance with Chapter 9 of the probabilistic risk assessment guidelines described in NUREG/CR-2300 (Reference 10).

Doses are determined for the early exposure effects resulting from the initial 24 hours following the core damage initiation. The dose evaluation provides the conditional probability distributions for the

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consequence measures, which includes the whole-body dose for this analysis. These consequence probability distributions are based on the assumption that the accident that produced the source term has occurred. Therefore, the consequence probability distributions presented result from the variation in dose levels due to the various meteorological conditions. Hence, the actual probability of the identified dose levels would be the probability of the release category that produced the source term occurring multiplied by the probability of the dose level.

The dose risks are quantified by multiplying the calculated fission product release category frequency vector by the release category mean dose vectors. The frequencies for each of the six release categories are quantified in Chapter 45 of the AP1000 Probabilistic Risk Assessment (Reference 2), while the mean doses for each release category are identified in Chapter 49. Table 1B-1 presents the results of the dose risk calculations at the site boundary at 24 hours. The table presents the release category identifier, the release frequency (per reactor year), the mean dose (in rem), and the resulting risk (in rem per reactor-year). In addition, each table presents the total dose risk and the percent that each release category contributes to the total risk. The information from Table 1B-1 was extracted from Chapter 49 of the AP1000 Probabilistic Risk Assessment.

It is shown that release category CFE presents the largest risk to the site safety.

The release categories for the AP1000 are defined as follows:

- IC - intact containment. Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.
- CFE - containment failure early. Fission-product release through a containment failure caused by severe accident phenomenon occurring after the onset of core damage but prior to core relocation.
- CFI - containment failure intermediate. Fission-product release through a containment failure caused by severe accident phenomenon occurring after core relocation but before 24 hours.
- CFL - containment failure late. Fission-product release through a containment failure caused by severe accident phenomenon occurring after 24 hours.
- CI - containment isolation failure. Fission-product release through a failure of the system or valves that close the penetrations between the containment and the environment. Containment failure occurs prior to onset of core damage.
- BP - containment bypass. Fission products are released directly from the Reactor Coolant System to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.

The following subsections present a brief description of the AP1000 release categories.

Release Category IC - Intact Containment

If the containment integrity is maintained throughout the accident, then the release of radiation from the containment is due to nominal leakage and is expected to be within the design basis of the containment. This is the "no failure" containment failure mode and is termed intact containment. The main location for fission-product leakage from the containment is penetration leakage into the auxiliary building where significant deposition of aerosol fission products may occur.

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Release Category CFE - Early Containment Failure

Early containment failure is defined as failure that occurs in the time frame between the onset of core damage and the end of core relocation. During the core melt and relocation process, several dynamic phenomena can be postulated to result in rapid pressurization of the containment to the point of failure. The combustion of hydrogen generated in vessel, steam explosions, and reactor vessel failure from high pressure are major phenomena postulated to have the potential to fail the containment. If the containment fails during or soon after the time when the fuel is overheating and starting to melt, the potential for attenuation of the fission-product release diminishes because of short fission-product residence time in the containment. The fission products released to the containment prior to the containment failure are discharged at high pressure to the environment as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Containment failures postulated within the time of core relocation are binned into release category CFE.

Release Category CFI - Intermediate Containment Failure

Intermediate containment failure is defined as failure that occurs in the time frame between the end of core relocation and 24 hours after core damage. After the end of the in-vessel fission-product release, the airborne aerosol fission products in the containment have several hours for deposition to attenuate the source term. The global combustion of hydrogen generated in-vessel from a random ignition prior to 24 hours can be postulated to fail the containment. The fission products in the containment atmosphere are discharged at high pressure to the environment as the containment blows down. Containment failures postulated within 24 hours of the onset of core damage are binned into release category CFI.

Release Category CFL - Late Containment Failure

Late containment failure is defined as containment failure postulated to occur later than 24 hours after the onset of core damage. Since the probabilistic risk assessment assumes the dynamic phenomena, such as hydrogen combustion, to occur before 24 hours, this failure mode occurs only from the loss of containment heat removal via failure of the passive containment cooling system. The fission products that are airborne at the time of containment failure will be discharged at high pressure to the environment, as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Accident sequences with failure of containment heat removal are binned in release category CFL.

Release Category CI - Containment Isolation Failure

A containment isolation failure occurs because of the postulated failure of the system or valves that close the penetrations between the containment and the environment. Containment isolation failure occurs before the onset of core damage. For such a failure, fission-product releases from the reactor coolant system can leak directly from the containment to the environment with diminished potential for attenuation. Most isolation failures occur at a penetration that connects the containment with the auxiliary building. The auxiliary building may provide additional attenuation of aerosol fission-product releases. However, this decontamination is not credited in the containment isolation failure cases. Accident sequences in which the containment does not isolate prior to core damage are binned into release category CI.

Release Category BP - Containment Bypass

Accident sequences in which fission products are released directly from the reactor coolant system to the environment via the secondary system or other interfacing system bypass the containment. The containment failure occurs before the onset of core damage and is a result of the initiating event or

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adverse conditions occurring at core uncovery. The fission-product release to the environment begins approximately at the onset of fuel damage, and there is no attenuation of the magnitude of the source term from natural deposition processes beyond that which occurs in the reactor coolant system, in the secondary system, or in the interfacing system. Accident sequences that bypass the containment are binned into release category BP.

1B.1.4.2 AP1000 Risk (CDF, LRF, and POPULATION Dose)

Table 1B-2 presents a summary of the CDF and large release frequency (LRF) risks for the AP1000.

Level 3 analysis is performed only for internal events at power. The ensuing population dose was very low, and it was not pursued for other events. The population dose for internal events is given in Table 1B-3. The information from Table 1B-3 was extracted from Chapter 49 of the AP1000 Probabilistic Risk Assessment.

1B.1.5 Summary of Risk Significant Enhancements

This section summarizes the design enhancements already incorporated into the AP1000 plant due to probabilistic risk assessment insights and results.

- Changed normal position of the two containment motor-operated recirculation valves (in series with squib valves) from closed to open

  The normal position of the two motor-operated valve lines in the two sump recirculation lines has been changed from NORMALLY CLOSED to NORMALLY OPEN to improve the reliability of opening these paths. These two paths support containment recirculation for core cooling and IRWST draining for IVR. This change reduced the CDF and LRF contribution from the failure modes to open the motor-operated valves.

- Changed IRWST drain procedure so it occurs earlier for IVR support

  Credit is taken for operator action to drain the IRWST into the sump to preserve reactor vessel integrity following core melt. The procedure for this severe accident response has been modified so that the operator action associated with IRWST draining is moved to the beginning of the procedure to allow more time for operator success and also to fill the cavity as soon as possible. This improves the probability of success of the operator action.

- Improved IVR heat transfer

  In going from the AP600 to the AP1000, the heat loads during IVR are increased due to the larger core power level, which reduced the margins in the heat removal capability through the reactor vessel head during IVR. To compensate for the increase in core power, the critical heat flux limit on the outside of the reactor vessel has been increased by changes made to the flow path between the outside of the reactor vessel and the reactor vessel insulation. Testing has confirmed the robustness of the IVR heat transfer.

- Improved IRWST vents

  The larger core in the AP1000 can generate more hydrogen in a severe accident. In the AP1000 hydrogen analysis for Level II, it was observed that the standing hydrogen diffusion flames at the IRWST vents resulted in a larger thermal loads to the containment steel shell, potentially leading to containment wall failure. The design of the vents was changed so that the IRWST vents located well away from the containment would open and the IRWST vents located next to the containment would not open during a severe accident to eliminate or minimize this potential concern.

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- Incorporated low boron core (anticipated transients without scram)

  In the AP600, anticipated transients without scram (ATWS) contribution to LRF was noticed to be high relative to other initiating events. A low boron core was incorporated into the design to reduce the potential contribution of ATWS to plant risk.

- Added 3rd passive containment cooling drain valve (motor-operator valve diverse to air operated valve)

  Due to reduced containment surface area per MW of core power, natural air circulation without passive containment cooling system water drain may not always be sufficient for long-term (greater than 1 day) containment heat removal in the AP1000. For the AP600, it was always sufficient for an indefinite time. To reduce the uncertainty in whether air cooling is sufficient to provide adequate long-term containment heat removal, a third path was added to the passive containment cooling system drain lines to increase passive containment cooling system reliability. The isolation valve used in the third path is a motor-operated valve, which is diverse from the air-operated valves used in the other two lines. This provides considerable improvement in the passive containment cooling system water drain reliability.

- Reduced potential recirculation-line squib valve failures

  An examination of AP1000 plant CDF cutsets revealed that the common cause failure of 4/4 recirculation line squib valves is a dominant contributor to CDF and LRF. This failure mode can be reduced by re-aligning the diverse squib valves already used in the AP1000 (and AP600) IRWST injection paths (high-pressure valves) and the containment recirculation paths (low-pressure valves). By making the recirculation squib valves two sets of two low-pressure and high-pressure squib valves, which are different and belong to different common cause failure groups. This design change reduces the common cause failure contribution of the recirculation squib valves. The increase in the group size of the high-pressure squib valves from four to six (including the four from the IRWST injection lines) does not add an appreciable contribution to the plant CDF.

1B.1.6 Specific Site Characteristics

AP1000 Probabilistic Risk Assessment Chapter 49, "Offsite Dose Risk Quantification," is based on an Electric Power Research Institute (EPRI) report (Reference 11) to establish the specific site characteristics for AP1000. Reference 11 Annex B, "ALWR Reference Site," establishes a conservative reference site to represent the consequences of most potential sites with respect to exposure at the site boundary. This reference site was based on the characteristics of 91 U.S. reactor sites that are tabulated in the NRC document, "Technical Guidance for Siting Criteria Development," (NUREG CR-2239) (Reference 12). Annex B provides a summary of the meteorological data to be used in calculating offsite dose.

1B.1.7 Value of Eliminating Risk

The cost benefit methodology of NUREG/BR-0184 (1997) is used to calculate the maximum attainable benefit. This includes replacement power costs. The maximum improvement change in the CDF frequency (delta-F) is assumed to be equal to the sum of CDF frequencies from internal, external, and shutdown events that are already evaluated:

delta F = 5 E-07/year

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This is bounding and is used to calculate the maximum attainable benefit. In practice, there is no design alternative, or SAMDA strategy, whose implementation would reduce the plant CDF to zero (or to an infinitesimally small frequency).

PRA Table 49-10, Revision 4, is used to calculate the expected value of the person-rem exposure:

Dose = 179,000 person-rem (0.0432 / 2.41E-07, from Table 49-10)

It is assumed that this dose is applicable to all events (internal, external, at-power, and shutdown). Thus, the consequences (dose and other) from all events are included in the calculations. Uncertainty in this dose is analyzed in sensitivity case 2 given below.

The following cost categories are investigated (NUREG/BR-0184 notation is used):

C1	Public Health (Accident)		5.7.1	5.7.1.3	W(pha)
C2	Public Health (Routine)		5.7.2	5,7,2	V(phr)
C3	Occupational Health (Accident)	Sum of C4 and C5	5.7.3	5.7.3	V(oha)
C4		Accident Related Exposure - ID		5.7.3.3	W(io)
C5		LT Doses		5.7.3.3	W(lto)
C6	Occupational Health (Routine)		5.7.4	5.7.4	V(ohr)
C7	Offsite Property		5.7.5	5.7.5	V(fp)
C8	Onsite Property	Sum of C9, C10, and C11	5.7.6	5.7.6	V(op)
C9		Cleanup and Decon		5.7.6.1	U(cd)
C10		LT Replacement Power		5.7.6.2	U(rp)
C11		Repair and Refurbishment		5.7.6.3	

The present-dollar value equivalent for severe accidents at one unit of the AP1000 is the sum of the offsite exposure costs, offsite economic costs, onsite exposure costs, and onsite economic costs. The present-day value (at 7 percent discount rate) of eliminating all plant CDF (maximum attainable benefit) is calculated to be $21,000, which is a very small dollar value. Thus, any mitigating system or a SAMDA strategy/alternative that reduces the plant risk by a fraction of the total plant CDF must cost less than $21,000 to be cost-effective.

Another calculation of the maximum attainable benefit is made with the discount rate of 3 percent (Table 7-2). The resulting value is $43,000, which is still very small to justify any appreciable investment.

Even if a very conservative multiplicative error factor of 10 were used, the maximum attainable benefit would be limited to a cost below $207,000.

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Table 1B-4 summarizes the results of the base case and the sensitivity cases.

In all cases, the values are strongly affected (increased) because of the replacement power cost. This is an inappropriate bias for public decision making, since it does not relate to public safety and it is not a direct cost to the public since the costs are to the utility, and their impact on the electricity rates for the public is unpredictable.

The first sensitivity case is already discussed above. In the second sensitivity case, the dose values are increased (10 times for external, NUREG high-estimates for occupational health). The third sensitivity analysis acknowledges that the delta-F realistically cannot be equal to the total plant CDF; a factor of 0.5 is introduced.

Sensitivity case 4 examines the case where the CDF value (thus the delta-F) is increased by a factor of 2. Finally, sensitivity case 5 looks at what happens if a multiplicative error factor of 10 is applied to the base case. In all cases, the benefits range from very small to modest.

1B.1.8 Evaluation of Potential Improvements

The value of eliminating AP1000 total risk is $21,000, as discussed in Section 1B.1.7. This value is an upper bound for any single engineered design alternative, which would actually reduce CDF and/or LRF a fraction of the values assumed in the base case for calculating the $21,000 value.

For the AP1000, SAMDA design alternatives discussed in this section are found to be not cost effective. One of these alternatives is actually implemented in the AP1000 design (diverse containment recirculation squib valves) to help improve the success likelihood of cavity reflooding operator action in severe accidents. The costs associated with the remaining SAMDA design alternatives are provided in Table 1B-5. Only one design alternative, 3 - namely, self actuating containment isolation valves - has a cost near $30,000; the remaining alternatives are at least an order of magnitude more costly than $30,000. Thus, only design alternative 3 needs to be further discussed.

1B.1.8.1 Self-Actuating Containment Isolation Valves

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as normal residual heat removal system and component cooling. The design alternative would be to add a self-actuating valve or enhance the existing inside containment isolation valve to provide for self actuation in the event that containment conditions are indicative of a severe accident. Conceptually, the design would either be an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve. This provides the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures, which occur at the time of the accident such as in cases of vessel rupture or ATWS. This design alternative provides almost no benefit in reducing plant CDF.

Generously assuming that this design alternative will eliminate CI release totally and that Delta CDF is zero, the benefit of this design alternative is calculated to be at the order of a few thousand dollars. Thus, even the cheapest design alternative does not meet the benefit/cost ratio of 1.

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1B.1.8.2 Other New Design Changes

Other design changes, as discussed in Section 1B.1.5, are already incorporated into the AP1000. There is no cost/benefit analysis available for those changes already incorporated.

Two additional design changes not incorporated in the AP1000 were assessed as follows:

Larger Accumulators

Increasing the size of the accumulators would result in a significant increase in cost that would be greater than the cost threshold established by the perfect SAMDA evaluation. In order to have any benefit in the probabilistic risk assessment, the accumulators would have to be increased in size sufficiently to change the large LOCA success criteria from two of two accumulators to one of two accumulators. Westinghouse estimates that the accumulator tanks would have to be increased in size from 2000 ft^3 to 4000 ft^3, and the hardware costs associated with this change would be significant. Such a size increase would also likely result in a change to the design of the DVI piping subsystem. The design of this piping system was established in the AP600 design certification, and the design does not change significantly for AP1000. Recently, Westinghouse completed the leak-before break analysis of the DVI piping, and any change in the DVI piping would result in significant piping reanalysis of the DVI piping. Westinghouse estimates the redesign costs associated with the changes in hardware and piping re-design to be significantly greater than the cost threshold established for the perfect SAMDA discussed above. Therefore this design change was not incorporated.

Larger Fourth-Stage ADS Valves

Increasing the fourth-stage ADS valves in size would result in a significant increase in cost associated with redesigning the AP1000 loop piping and fourth-stage piping configuration. The AP1000 ADS valves were already increased in size compared to the AP600 valves more than the ratio of the power uprate of the AP1000. In order to have any benefit in the probabilistic risk assessment, the 4th stage ADS valves would have to be increased in size sufficiently to change the LOCA success criteria from three of four valves to two of four valves. To accommodate such a change, Westinghouse estimates that the fourth-stage ADS valves would have to increase in size from 14-inch to 18-inch valves and associated piping. In addition, the common fourth-stage inlet piping that connects to the hot leg would have to increase in size from 18-inch to at least 20-inch. This would require a significant redesign of the squib valve and would also result in redesign of the ADS-4 piping which in turn would impact the design of the reactor coolant loop piping. Finally, such a redesign would require Westinghouse to perform additional confirmatory testing of the passive core cooling system to verify that the behavior of the passive safety systems was not adversely impacted. Westinghouse estimates the cost of this change to be significantly larger than the cost threshold of the perfect SAMDA discussed above. Therefore, this design change was not incorporated.

1B.1.9 Results

Due to the existing low risk of the AP1000 plant, none of the design alternatives described in Section 1B.1.3 meets an acceptable benefit to cost ratio of 1 or greater.

Several of the design alternatives evaluated in other SAMDA analyses are included in the current AP1000 design. These design features include the following:

- Reactor coolant system depressurization system
- Passive residual heat removal system located inside containment
- Cavity flooding system

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- Passive containment cooling system
- Hydrogen igniters in a large-dry containment
- Diverse actuation system
- Sealless motor reactor coolant pumps
- Interfacing system with high design pressure

As the AP1000 plant CDF is lower than for existing plants, the benefits of additional design alternatives are small. The SAMDAs analyzed provided little or no benefit to the AP1000 design.

Assuming a hypothetical design alternative was developed which provides a 100 percent reduction in overall plant risk, representing an average averted risk of 4.32×10^{-2} man-rem per year, the capital benefit amounts to only $21,000.

1B.2 References

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4. "Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2," Docket Nos. 50-352/353, August 1989.

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Table 1B-1
Population Whole Body EDE Dose Risk - 24 Hours

Release Category	Release Frequency (per reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM per reactor year)	Percentage Contribution to Total Risk
CFI	1.89E-10	7.03E+03	7.03E+05	1.33E-04	0.3
CFE	7.47E-09	8.51E+03	8.51E+05	6.36E-03	14.7
IC	2.21E-07	7.19E+00	7.19E+02	1.59E-04	0.4
BP	1.05E-08	3.23E+04	3.23E+06	3.39E-02	78.4
CI	1.33E-09	2.01E+04	2.01E+06	2.67E-03	6.2
CFL	3.45E-13	7.37E+01	7.37E+03	2.54E-09	0.0
			Total Risk =	4.32E-02	100.0

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Table 1B-2
Summary of AP1000 PRA Results (CDF and LRF)

Events	Core Damage Frequency (per year)		Large Release Frequency (per year)	
	At-Power	Shutdown	At-Power	Shutdown
Internal Events	2.41E-07	1.23E-07	1.95E-08	2.05E-08
Internal Flood	8.82E-10	3.22E-09	7.14E-11	5.37E-10
Internal Fire	5.61E-08	8.5E-08	4.54E-09	1.43E-08
Sum =	2.97E-07	2.11E-07	2.41E-08	3.53E-08

Note:
For seismic risk, the seismic margins method is used. CDF and LRF are not quantified.

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Table 1B-3
Population Whole Body Dose (Effective Dose Equivalent [EDE]),
0-80.5 km person-sieverts

24-Hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	7.03E+03	5.33E+03	1.31E+04	1.82E+04	3.11E+04	3.59E+04	5.07E+04
CFE	8.51E+03	6.25E+03	1.62E+04	2.31E+04	4.13E+04	5.06E+04	6.40E+04
DIRECT	2.16E+01	1.20E+01	4.78E+01	8.13E+01	1.14E+02	1.23E+02	1.68E+02
IC	7.19E+00	4.21E+00	1.71E+01	2.95E+01	3.56E+01	3.84E+01	5.60E+01
BP	3.23E+04	2.10E+04	6.40E+04	1.03E+05	1.54E+05	1.82E+05	2.64E+05
CI	2.01E+04	1.13E+04	4.71E+04	6.60E+04	1.23E+05	1.48E+05	1.61E+05
CFL	7.37E+01	1.00E+01	1.62E+02	5.91E+02	9.76E+02	1.11E+03	2.56E+03
72-Hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	1.13E+04	9.02E+03	2.12E+04	2.63E+04	4.09E+04	4.89E+04	6.18E+04
CFE	9.36E+03	6.89E+03	1.898E+04	2.54E+04	4.25E+04	5.12E+04	6.77E+04
DIRECT	2.36E+01	1.35E+01	5.28E+01	8.32E+01	1.15E+02	1.25E+02	1.75E+02
IC	7.87E+00	4.75E+00	1.85E+01	3.00E+01	3.79E+01	4.20E+01	5.83E+01
BP	4.17E+04	2.94E+04	7.99E+04	1.16E+05	2.20E+05	2.61E+05	2.87E+05
CI	2.14E+04	1.25E+04	4.90E+04	7.40E+04	1.27E+05	1.53E+05	1.67E+05
CFL	4.79E+04	3.11E+04	9.57E+04	1.57E+05	2.62E+05	3.01E+05	4.14E+05

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Table 1B-4
Cost Benefit Calculation Results for Different Assumptions

	Case Studied	Benefit of Case
Base Case	7% Discount rate	21,000
SC-1	3% Discount rate	43,000
SC-2	High dose (10 times the base case)	36,000
SC-3	Realistic delta-F (SAMDA reduces CDF by 50% of total)	10,000
SC-4	Twice the base CDF	41,000
SC-5	10 times the benefit of base case	207,000

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Table 1B-5
Design Alternatives for SAMDA

No.	Design Alternative	Cost
1	Upgrade chemical, volume, and control system for small LOCA	1,500,000
2	Containment filtered vent	5,000,000
3	Self-actuating containment isolation valves	33,000
4	Safety grade passive containment spray	3,900,000
6	Steam generator shell-side heat removal	1,300,000
7	Steam generator relief flow to IRWST	620,000
8	Increased steam generator pressure capacity	8,200,000
9	Secondary containment ventilation with filtration	2,200,000
10	Diverse IRWST injection valves	570,000
11	Diverse containment recirculation valves	Already Implemented
12	Ex-vessel core catcher	1,660,000
13	High-pressure containment design	50,000,000
14	More reliable diverse actuation system	470,000

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Chapter 2 Site Characteristics

This chapter defines the site-related parameters for which the AP1000 plant is designed. The site parameters are in Table 2-1. These parameters envelope most potential sites in the United States. The sections of this chapter follow the standard format and discuss how the specific parameters are used in the AP1000 design and how the Combined License applicant is to demonstrate that the site meets the design parameters.

The site is acceptable if the site characteristics fall within the AP1000 plant site design parameters in Table 2-1. Should specific site parameters or characteristics be outside the envelope of assumptions established by Table 2-1, the Combined License applicant referencing the AP1000 will demonstrate that the design satisfies the requirements imposed by the specific site parameters and conforms to the design commitments and acceptance criteria described in the AP1000 Design Control Document.

2.1 Geography and Demography

The geography and demography are site specific and will be defined by the Combined License applicant.

2.1.1 Combined License Information for Geography and Demography

Combined License applicants referencing the AP1000 certified design will provide site-specific information related to site location and description, exclusion area authority and control, and population distribution.

Site Information – Site-specific information on the site and its location will include political subdivisions, natural and man-made features, population, highways, railways, waterways, and other significant features of the area.

Exclusion Area – Site-specific information on the exclusion area will include the size of the area and the exclusion area authority and control. Activity that may be permitted within the exclusion area will be included in the discussion.

Population Distribution – Site-specific information will be included on population distribution.

2.2 Nearby Industrial, Transportation, and Military Facilities

The plant has inherent capability to withstand certain types of external accidents due to the specified design conditions associated with earthquakes, wind loading, and radiation shielding. Acceptability for external accidents associated with a given site will be covered in the Combined License application.

Each Combined License applicant referencing the AP1000 will provide analyses of accidents external to the nuclear plant. The determination of the probability of occurrence of potential accidents which could have severe consequences will be based on analyses of available statistical data on the occurrence of the accident together with analyses of the effects of the accident on the plant's safety-related structures and components. If an accident is identified for which the probability of severe consequences is unacceptable, specific changes to the AP1000 will be identified in the Combined License safety analysis report. The criteria for not requiring

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changes to the AP1000 design is that the total annual frequency of occurrence is less than 10^{-6} per year for an external accident leading to severe consequences. The following accident categories will be considered in determining the frequency of occurrence, as appropriate:

Explosions – Accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels will be considered for facilities and activities in the vicinity of the plant where such materials are processed, stored, used, or transported in quantity.

The AP1000 includes onsite storage facilities for compressed and liquid hydrogen. Accidents involving accidental detonations of hydrogen from these storage facilities are evaluated as part of the AP1000 certified design. Combined License applicants referencing the AP1000 Design Certification are not required to provide analyses of accidents involving these storage facilities provided that the locations and size of the storage facilities are consistent with the safe distances defined by the AP1000 certified design. The bulk gas storage area for the plant gas system (PGS) is located sufficiently far from the nuclear island that an explosion would not result in damage to safety-related structures, systems, and components.

Evaluation of potential explosions due to exposure of chemical storage tanks to exterior fires has determined that all of these postulated accidents are safe distances away from safety-related items.

The AP1000 certified design does not include liquid oxygen or propane storage facilities.

Flammable Vapor Clouds (Delayed Ignition) – Accidental releases of flammable liquids or vapors that result in the formation of unconfined vapor clouds in the vicinity of the plant.

A flammable vapor cloud (delayed ignition) due to the accidental release of hydrogen from the PGS bulk gas storage area is evaluated as part of the AP1000 certified design. A detonation of such a hydrogen vapor cloud would not result in damage to safety-related structures, systems, and components. No other chemical has the possibility of developing unconfined flammable vapor clouds.

Toxic Chemicals – Accidents involving the release of toxic chemicals from nearby mobile and stationary sources.

Fires – Accidents leading to high heat fluxes or smoke, and to nonflammable gas or chemical-bearing clouds from the release of materials as the consequence of fires in the vicinity of the plant.

Airplane Crashes – Accidents involving aircraft crashes leading to missile impact or fire in the vicinity of the plant.

The AP1000 safe distance for material in onsite storage facilities for explosions, flammable vapor clouds, and fires is tabulated in Table 2.2-1.

2.2.1 Combined License Information for Identification of Site-Specific Potential Hazards

Combined License applicants referencing the AP1000 certified design will provide site-specific information related to the identification of potential hazards within the site vicinity, including an evaluation of potential accidents and verify that the frequency of site-specific potential hazards

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is consistent with the criteria outlined in Section 2.2. The site-specific information will provide a review of aircraft hazards, information on nearby transportation routes, and information on potential industrial and military hazards.

2.3 Meteorology

The AP1000 is designed for air temperatures, humidity, precipitation, snow, wind, and tornado conditions as specified in Table 2-1. The Combined License applicant must provide information to demonstrate that the site parameters are within the limits specified for the standard design.

The design wind is specified as a basic wind speed of 145 mph with an annual probability of occurrence of 0.02. Wind loads are calculated for exposure C, which is applicable to shorelines in hurricane prone areas. The site parameters for the design wind may be demonstrated to be acceptable for other exposures or topographic factors by comparison of the wind loads on the structures. For example, for a site at a location with exposure Category D, the wind speed should be equal to or less than 130 mph.

2.3.1 Regional Climatology

The regional climatology is site specific and will be defined by the Combined License applicant.

2.3.2 Local Meteorology

The local meteorology is site specific and will be defined by the Combined License applicant.

2.3.3 Onsite Meteorological Measurement Programs

The onsite meteorological measurement program is site specific and will be defined by the Combined License applicant. The number and location of meteorological instrument towers are determined by actual site parameters.

2.3.4 Short-Term Diffusion Estimates

In the absence of a specific site for use in determining values for short-term diffusion, a study was performed to determine the atmospheric dispersion factors (χ/Q values) that would envelope most current plant sites and that could be used to calculate the radiological consequences of design basis accidents. The χ/Q values thus derived for offsite are provided in Table 2-1.

This set of offsite χ/Q values is representative of potential sites for construction of the AP1000. The values are appropriate for analyses to determine the radiological consequences of accidents. These values were selected to bound 70 to 80 percent of U.S. sites.

The χ/Q values for the control room air intake or the door leading to the control room are dependent not only on the site meteorology but also on the plant design and layout. These χ/Q values are addressed in Appendix 15A. Separate sets of χ/Q values are identified for each combination of activity release location and receptor location.

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2.3.5 Long-Term Diffusion Estimates

The long-term diffusion estimates are site specific and will be provided by the Combined License applicant. The site boundary annual average χ/Q shown in Table 2-1 is used to calculate release concentrations at the site boundary for comparison with the activity release limits defined in 10 CFR 20. The value specified is expected to bound atmospheric conditions at most U.S. sites. If a selected site has a χ/Q value that exceeds this reference site value, the release concentrations reported in Section 11.3 would be adjusted proportionate to the change in χ/Q.

2.3.6 Combined License Information

2.3.6.1 Regional Climatology

Combined License applicants referencing the AP1000 certified design will address site-specific information related to regional climatology.

2.3.6.2 Local Meteorology

Combined License applicants referencing the AP1000 certified design will address site-specific local meteorology information.

2.3.6.3 Onsite Meteorological Measurements Program

Combined License applicants referencing the AP1000 certified design will address the site-specific onsite meteorological measurements program.

2.3.6.4 Short-Term Diffusion Estimates

Combined License applicants referencing the AP1000 certified design will address the site-specific χ/Q values specified in subsection 2.3.4. For a site selected that exceeds the bounding χ/Q values, the Combined License applicant will address how the radiological consequences associated with the controlling design basis accident continue to meet the dose reference values given in 10 CFR Part 50.34 and control room operator dose limits given in General Design Criteria 19 using site-specific χ/Q values. The Combined License applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and receptors. No further action is required for sites within the bounds of the site parameters for atmospheric dispersion.

With regard to assessment of the postulated impact of an accident on the environment, the COL applicant will provide χ/Q values for each cumulative frequency distribution which exceeds the median value (50 percent of the time).

2.3.6.5 Long-Term Diffusion Estimates

Combined License applicants referencing the AP1000 certified design will address long-term diffusion estimates and χ/Q values specified in subsection 2.3.5. The Combined License applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and

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other unusual conditions affecting atmospheric transport and diffusion between the source and receptors. No further action is required for sites within the bounds of the site parameter for atmospheric dispersion.

With regard to environmental assessment, the COL applicant will also provide estimates of annual average χ/Q values for 16 radial sectors to a distance of 50 miles from the plant.

2.4 Hydrologic Engineering

The AP1000 is designed for a normal groundwater elevation up to plant elevation 98′ and for a flood level up to plant elevation 100′. For structural analysis purposes, grade elevation is also established as plant elevation 100′. Actual grade will be a few inches lower to prevent surface water from entering doorways.

For a portion of the annex building the site grade will be 107 feet to permit truck access at the elevation of the floor in the annex building and inside containment. Subsection 3.4.1 describes design provisions for groundwater and flooding.

The Combined License applicant will evaluate events leading to potential flooding to demonstrate that the site meets the site parameter for flood level. As necessary, the Combined License applicant may propose measures to protect the plant according to the Standard Review Plan, Section 2.4.10. Events to be considered are those identified in Standard Review Plan, Section 2.4.2.

Adverse effects of flooding due to high water or ice effects do not have to be considered for site-specific nonsafety-related structures and water sources outside the scope of the certified design. Flooding of water intake structures, cooling canals, or reservoirs or channel diversions would not prevent safe operation of the plant.

2.4.1 Combined License Information

2.4.1.1 Hydrological Description

Combined License applicants referencing the AP1000 certified design will describe major hydrologic features on or in the vicinity of the site including critical elevations of the nuclear island and access routes to the plant.

2.4.1.2 Floods

Combined License applicants referencing the AP1000 certified design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.

- Probable Maximum Flood on Stream and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the probable maximum flood on streams and rivers.

- Dam Failures – Site-specific information on potential dam failures.

- Probable Maximum Surge and Seiche Flooding – Site-specific information on probable maximum surge and seiche flooding.

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- Probable Maximum Tsunami Loading – Site-specific information on probable maximum tsunami loading.

- Flood Protection Requirements – Site-specific information on flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

No further action is required for sites within the bounds of the site parameter for flood level.

2.4.1.3 Cooling Water Supply

Combined License applicants will address the water supply sources to provide makeup water to the service water system cooling tower.

2.4.1.4 Groundwater

Combined License applicants referencing the AP1000 certified design will address site-specific information on groundwater. No further action is required for sites within the bounds of the site parameter for ground water.

2.4.1.5 Accidental Release of Liquid Effluents in Ground and Surface Water

Combined License applicants referencing the AP1000 certified design will address site-specific information on the ability of the ground and surface water to disperse, dilute, or concentrate accidental releases of liquid effluents. Effects of these releases on existing and known future use of surface water resources will also be addressed.

2.4.1.6 Emergency Operation Requirement

Combined License applicants referencing the AP1000 certified design will address any flood protection emergency procedures required to meet the site parameter for flood level.

2.5 Geology, Seismology, and Geotechnical Engineering

Combined License applicants referencing the AP1000 certified design will address site specific information related to basic geological, seismological, and geotechnical engineering of the site and the region, as discussed in the following subsections.

2.5.1 Basic Geological and Seismic Combined License Information

Combined License applicants referencing the AP1000 certified design will address the following regional and site-specific geological, seismological, and geophysical information as well as conditions caused by human activities:

- Structural geology of the site
- Seismicity of the site
- Geological history
- Evidence of paleoseismicity
- Site stratigraphy and lithology
- Engineering significance of geological features
- Site groundwater conditions
- Dynamic behavior during prior earthquakes

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- Zones of alteration, irregular weathering, or structural weakness
- Unrelieved residual stresses in bedrock
- Materials that could be unstable because of mineralogy or unstable physical properties
- Effect of human activities in the area

2.5.2 Vibratory Ground Motion

The AP1000 is designed for an earthquake defined by a peak ground acceleration (PGA) of 0.30g and the design response spectra specified in subsection 3.7.1.1, and Figures 3.7.1-1 and 3.7.1-2. The AP1000 design earthquake is referred to as the AP1000 Certified Seismic Design Response Spectra (CSDRS). The AP1000 CSDRS was developed using the Regulatory Guide 1.60 response spectra as the base and modified to include additional high frequency amplification at a control point at 25 Hz. The peak ground accelerations in the two horizontal and the vertical directions are equal. The CSDRS also represents the AP1000 Foundation Input Response Spectra (FIRS) at a hard rock site.

The AP1000 is evaluated for high frequency input using the response spectra specified in Appendix 3I, Figures 3I.1-1 and 3I.1-2. The seismic response spectra given in Figures 3I.1-1 and 3I.1-2 are envelope response spectra with high frequency content.

2.5.2.1 Combined License Seismic and Tectonic Characteristics Information

Combined License applicants referencing the AP1000 certified design will address the following site-specific information related to the vibratory ground motion aspects of the site and region:

- Seismicity
- Geologic and tectonic characteristics of site and region
- Correlation of earthquake activity with seismic sources
- Probabilistic seismic hazard analysis and controlling earthquakes
- Seismic wave transmission characteristics of the site
- SSE ground motion

The site-specific ground motion response spectra (GMRS) for comparison against the CSDRS are determined in the free-field on the ground surface. For sites with soil layers that will be completely excavated to expose competent material, the GMRS is specified on an outcrop or a hypothetical outcrop that will exist after excavation. Motions at this hypothetical outcrop are developed as a free-surface motion, not as an in-column motion with no soil or backfill soil layers above the outcrop. Competent material may be defined as in-situ material having a low strain shear wave velocity equal to or greater than 1000 fps. The Combined License applicant must demonstrate that the proposed site meets the following requirements:

1. The free field peak ground acceleration at the finished grade level is less than or equal to a 0.30g SSE.

2. The site-specific ground motion response spectra (GMRS) at the finished grade level in the free-field are less than or equal to the AP1000 certified seismic design spectra (CSDRS) given in Figures 3.7.1-1 and 3.7.1-2.

3. In lieu of (1) and (2) above, for a site where the nuclear island is founded on hard rock with shear wave velocity greater than 8,000 feet per second, the site-specific ground motion

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response spectra (GMRS) may be defined at the foundation level and shown to be less than or equal to the CSDRS given in Figures 3.7.1-1 and 3.7.1-2.

4. In lieu of (1) and (2) above, for a site where the nuclear island is founded on hard rock defined by a shear wave velocity at the bottom of the basemat equal to or higher than 7,500 fps, while maintaining a shear wave velocity equal to or above 8,000 fps at the lower depths, site-specific spectra may be developed at the top of the competent rock and shown at the foundation level to be less than or equal to those given in Figures 3I.1-1 and 3I.1-2 over the entire frequency range. If a COL applicant has site GMRS exhibiting hard rock high frequency (HRHF) characteristics, but is not enveloped by the AP1000 HRHF envelope response spectra or the AP1000 CSDRS, or has shear wave velocities that are not associated with hard rock, the COL applicant may perform site-specific studies to demonstrate that the high frequency is not damaging. This may be accomplished by the following:

   a. Demonstrating that the site floor response spectra, developed at the locations of the spectra given in APP-GW-GLR-115, Sections 5.2 and 6.3 (Reference 3) using the seismic input defined by the site GMRS, are enveloped by the AP1000 HRHF envelope response spectra or CSDRS spectra.

   b. If it is shown in step one that the spectra are not enveloped, evaluations similar to those described in Appendix 3I (documented in Reference 3) would be made to demonstrate that the high frequency input is non-damaging.

5. Foundation material layers are approximately horizontal (dip less than 20 degrees), and the minimum estimate of the low strain shear wave velocity of the soil below the foundation of the nuclear island is greater than or equal to 1000 feet per second.

6. For sites where the nuclear island is founded on soil, the minimum estimate of the strain-compatible soil shear modulus (G), the shear wave velocity (Vs), and hysteretic damping is compared to the values used in the AP1000 generic analyses shown in Table 3.7.1-4 and Figure 3.7.1-17. Properties of soil layers within a depth of 120 feet below finished grade are compared to those in the generic soil site analyses (soft soil [SS], soft-to-medium soil [SM], and upper bound soft-to-medium soil [UBSM]). The shear wave velocity should generally increase with depth. The average low strain shear wave velocity in any layer should not be less than 80 percent of the average shear wave velocity in any layer at higher elevation. For the SS, SM, and UBSM soil profile, the shear wave velocity of each layer is to be within the lower bound and upper bound of the soil profiles. The lower bound and upper bound of shear wave velocity correspond to Gmax/1.5 and 1.5*Gmax. The lower bound shear wave velocity for the SS and SM soil profile is to be greater than or equal to 1000 fps.

7. In lieu of (1) to (6) above, a site-specific evaluation can be performed as described in subsection 2.5.2.3.

2.5.2.2 Site-Specific Seismic Structures

The AP1000 includes all seismic Category I structures, systems and components in the scope of the design certification.

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2.5.2.3 Site-Specific Seismic Evaluation

The Combined License applicant may identify site-specific features and parameters that are not clearly within the guidance provided in subsection 2.5.2.1. These features and parameters may be demonstrated to be acceptable by performing site-specific seismic analyses. These analyses may be either 2D or 3D. Where 2D or 3D analyses apply are as follows:

- The 3D SASSI analyses will be used to quantify the effects of exceedances of site-specific GMRS compared to the CSDRS, or the HRHF GMRS at a hard rock site (DCD Figures 3I.1-1 and 3I.1-2), or in cases where the site-specific velocity soil profiles do not fall within the range evaluated for the standard design.

- For site-specific cases outside of the certified design, such as loads not evenly applied on the foundation that can be caused by soil conditions not evenly applied throughout the AP1000 foundation, the site-specific analysis should consider 3D effects.

- The 2D analyses are performed for parameter studies.

Results will be compared to the corresponding 2D or 3D generic analyses.

2.5.2.3.1 2D Analyses

Where features of the site are not within the parameters specified for the AP1000, site-specific soil structure interaction analyses may be performed using the 2D SASSI models described in Appendix 3G for variations in site conditions that can be represented in these models. Results should be compared to the results of the 2D SASSI analyses described in Appendix 3G. Such analyses may be used to demonstrate that local features, such as soil degradation properties or backfill, are well within the bounds established by the design cases. If the results are not clearly enveloped at the significant frequencies of response at the six key locations compared with the floor response spectra of the certified design at 5-percent damping, then a 3D SASSI analysis may be required. These evaluations and comparisons will be provided and reviewed as part of the Combined License application.

2.5.2.3.2 3D Analyses

If required, a 3D evaluation will consist of a site-specific dynamic analysis and generation of in-structure response spectra at six key locations to be compared with the floor response spectra of the certified design at 5-percent damping. The certified seismic site design response spectra at the foundation level in the free-field given in Figures 3.7.1-1 and 3.7.1-2 were used to develop the floor response spectra. They were applied at foundation level for the hard rock site and at finished grade level for the soil sites. The site is acceptable if the floor response spectra from the site-specific evaluation do not exceed the AP1000 spectra for each of the locations identified below or the exceedances are justified:

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Containment internal structures at elevation of reactor vessel support	Figure 3G.4-5X to 3G.4-5Z
Containment operating floor	Figure 3G.4-6X to 3G.4-6Z
Auxiliary building NE corner at elevation 116'-6"	Figure 3G.4-7X to 3G.4-7Z
Shield building at fuel building roof	Figure 3G.4-8X to 3G.4-8Z
Shield building roof	Figure 3G.4-9X to 3G.4-9Z
Steel containment vessel at polar crane support	Figure 3G.4-10X to 3G.4-10Z

Site-specific soil structure interaction analyses are performed using the 3D SASSI models described in Appendix 3G. The site-specific soil structure interaction analyses use the site-specific soil conditions (including variation in soil properties in accordance with Standard Review Plan 3.7.2 and site-specific soil degradation models). The three components of the site-specific ground motion time history must satisfy the regulatory requirements for statistical independence and enveloping of the site design spectra at 5% damping. Floor response spectra determined from the site-specific analyses should be compared against the design basis of the AP1000 described above. These evaluations and comparisons will be provided and reviewed as part of the Combined License application.

If the site-specific spectra at foundation level at a rock site exceed the response spectra in Figures 3I.1-1 and 3I.1-2 at any frequency, a site-specific evaluation can be performed similar to that described in Appendix 3I.

2.5.3 Surface Faulting Combined License Information

Combined License applicants referencing the AP1000 certified design will address the following surface and subsurface geological, seismological, and geophysical information related to the potential for surface or near-surface faulting affecting the site:

- Geological, seismological, and geophysical investigations
- Geological evidence, or absence of evidence, for surface deformation
- Correlation of earthquakes with capable tectonic sources
- Ages of most recent deformation
- Relationship of tectonic structures in the site area to regional tectonic structures
- Characterization of capable tectonic sources
- Designation of zones of quaternary deformation in the site region
- Potential for surface tectonic deformation at the site

The AP1000 design has not been evaluated for a site where there is a fault displacement potential. A COL applicant will satisfy the requirement for no surface or near surface tectonic structure capable of displacements beneath the nuclear island and adjacent seismic Category II structures by completing geological, seismological, and geophysical investigations that are consistent with the guidance of Regulatory Guide 1.206.

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2.5.4 Stability and Uniformity of Subsurface Materials and Foundations

Combined License applicants referencing the AP1000 certified design will address the following site-specific information related to the stability and uniformity of subsurface materials and foundations.

- Excavation
- Bearing capacity
- Settlement
- Liquefaction

2.5.4.1 Excavation

Excavation for the nuclear island structures below grade may use either a sloping excavation or a vertical face as described in subsequent paragraphs. If sloping excavations are to be used on a soil site, the Combined License applicants must evaluate the 3D effects on the site response and perform site-specific SSI analyses using either or a combination of 2D or 3D SASSI models that reflect the sloping excavations. If backfill is to be placed adjacent to the exterior walls of the nuclear island, the Combined License applicant will provide information on the properties of backfill and its compaction requirements as described in subsection 2.5.4.6.3 and will evaluate its properties against those used in the seismic analyses described in subsection 3.7.2.

For the vertical face alternative, excavation in soil for the nuclear island structures below grade will establish a vertical face with lateral support of the adjoining undisturbed soil or rock. This vertical face will be covered by a waterproof membrane as described in subsection 3.4.1.1.1.1 and is used as the outside form for the exterior walls below grade of the nuclear island. Alternative methods include a soil nailing and mechanically stabilized earth (MSE) walls.

2.5.4.1.1 Vertical Face Using Soil Nails

Soil nailing is a method of retaining earth in-situ. As the nuclear island excavation progresses vertically downward, holes are drilled horizontally into the adjoining undisturbed soil, a metal rod is inserted into the hole, and grout is pumped into each hole to fill the hole and to anchor the “nail” rod.

As each increment of the nuclear island excavation is completed, nominal eight to ten inch diameter holes are drilled horizontally through the vertical face of the excavation into adjacent undisturbed soil. These “nail” holes, spaced horizontally and vertically on five to six feet centers, are drilled slightly downward to the horizontal. A “nail”, normally a metal bar/rod, is center located for the full length of the hole. The nominal length of soil nails is 60 percent to 70 percent of the wall height, depending upon soil conditions. The hole is filled with grout to anchor the rod to the soil. A metal face plate is installed on the exposed end of the rod at the excavated wall vertical surface. Welded wire mesh is hung on the wall surface for wall reinforcement and secured to the soil nail face plates for anchorage. A 4,000 psi to 5,000 psi non-expansive pea gravel shotcrete mix is blown onto the wire mesh to form a nominal four to six inch thick soil retaining wall. Installation of the soil retaining wall closely follows the progress of the excavation and is from the top down, with each wire mesh-reinforced, shotcreted wall section being supported by the soil “nails” and the preceding elevations of soil nailed wall placements.

Soil nailing as a method of soil retention has been successfully used on excavations up to 55 feet deep on projects in the U.S. Soils have been retained for up to 90 feet in Europe. The

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state of California CALTRANS uses soil nailing extensively for excavations and soil retention installations. Soil nailing design and installation has a successful history of application which is evidenced by its excellent safety record.

The soil nailing method produces a vertical surface down to the bottom of the excavation and is used as the outside forms for the exterior walls below grade of the nuclear island. Concrete is placed directly against the vertical concrete surface of the excavation.

For methods of soil retention other than soil nailing, such as for excavation in rock, four to six inches of shotcrete are blown on to the vertical surface. The concrete for the exterior walls is placed against the shotcrete.

2.5.4.1.2 Vertical Face Using Mechanically Stabilized Earth Walls

Mechanically stabilized earth walls (MSE) are flexible retaining wall systems that use strip, grid, or sheet type of tensile reinforcements in the soil mass, and a discrete modular pre-cast concrete, which is vertical. MSE walls function like, and are generally more economical than, conventional retaining walls. The tensile strength of the reinforcements and the slip at the interface of the reinforcement and the soil provide great internal stability to MSE walls. These walls may be used where the side soils have to be removed or the grade elevation needs to be raised. The walls and backfill are placed prior to construction of the nuclear island.

2.5.4.1.3 Mudmat

The mudmat provides a working surface prior to initiating the placement of reinforcement for the foundation mat structural concrete. The lower and upper mudmats are as follows:

- Lower mudmat – (6-inch layer) of un-reinforced concrete, with a minimum compressive strength of 2,500 psi. The lower mudmat will be used as the final dental concrete layer on the underlying foundation media.

- Upper mudmat – (6-inch layer) of un-reinforced concrete with a minimum compressive strength of 2,500 psi. This upper mudmat will support the chairs that, in turn, support the reinforcing steel.

The waterproofing system is described in subsection 2.5.4.6.12.

2.5.4.2 Bearing Capacity

The maximum bearing reaction determined from the 3D SASSI analyses described in Appendix 3G is less than 35,000 lb/ft^2 under all combined loads, including the safe shutdown earthquake. The maximum dynamic bearing demand of 35 ksf occurs under the west edge of the shield building and is primarily due to the response to the east-west component of the earthquake. The east edge of the nuclear island lifts off the soil. The Combined License applicant will verify that the site-specific allowable soil bearing capacities for static and dynamic loads at the site will exceed the static and dynamic bearing demand given in Table 2-1.

The evaluation of the allowable capacity of the soil is based on the properties of the underlying materials (see subsection 2.5.4.5.2), including appropriate laboratory test data to evaluate strength, and considering local site effects, such as fracture spacing, variability in properties,

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and evidence of shear zones. The allowable bearing capacity should provide a factor of safety appropriate for the design load combination, including safe shutdown earthquake loads.

If the shear wave velocity or the allowable bearing capacity is outside the range evaluated for AP1000 design certification, a site-specific evaluation can be performed using the AP1000 basemat model and methodology described in subsection 3.8.5. The safe shutdown earthquake loads are those from the AP1000 analyses described therein. Alternatively, bearing pressures may be determined from a site-specific analysis using site-specific inputs as described in subsection 2.5.2.3. For the site to be acceptable, the bearing pressures from the site-specific analyses, including static and dynamic loads, need to be less than the capacity of each portion of the basemat.

2.5.4.3 Settlement

The Combined License applicant will address short-term (elastic) and long-term (heave and consolidation) settlement for soil sites for the history of loads imposed on the nuclear island foundation and adjacent buildings consistent with the construction sequence. The resulting time-history of settlements includes construction activities such as dewatering, excavation, bearing surface preparation, placement of the basemat, and construction of the superstructure.

The AP1000 does not rely on structures, systems, or components located outside the nuclear island to provide safety-related functions. Differential settlement between the nuclear island foundation and the foundations of adjacent buildings does not have an adverse effect on the safety-related functions of structures, systems, and components. Differential settlement under the nuclear island foundation could cause the basemat and buildings to tilt. Much of this settlement occurs during civil construction prior to final installation of the equipment. Differential settlement of a few inches across the width of the nuclear island would not have an adverse effect on the safety-related functions of structures, systems, and components. Table 2.5-1 provides guidance to the Combined License applicant on predictions of absolute and differential settlement that are acceptable without further evaluation. The predicted settlements will cover the periods before construction begins through the construction phase, and for the subsequent plant operating period or otherwise justified. The predicted settlements will be based on conservative assumptions of soil properties. If the predicted settlements exceed the limits of Table 2.5-1, a detailed evaluation and construction plan will be described by the Combined License applicant. During construction and plant operation at a soil site, settlements would be measured and compared to the predicted settlement values and any exceedances would require additional investigation.

Alternatives for the additional evaluations are provided as follows:

1. Evaluate the impact of the elevated estimated settlement values on the critical components of the AP1000 including, but not limited to, piping spanning between the nuclear island and the adjacent structures, the equipment support pads, the construction gap between the nuclear island and adjacent buildings, and the stresses on the basemat (along with influences to the underlying soil).

2. Submit a construction sequence to control the predicted settlement behavior. A revised sequence should follow the specific schedule to distribute construction loads as necessary in order to obtain acceptable values. Depending on soil conditions, a significant amount of the settlement could occur during construction and can be controlled through the construction sequence.

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3. Provide a uniform excavation and engineered backfill to manage static building rotation and differential settlement between the nuclear island and adjacent structures.

4. Implement an active settlement monitoring system throughout the entire construction sequence as well as a long-term (plant operation) plan. By monitoring the settlement throughout construction, the Combined License applicant will be able to modify the construction sequences of adjacent buildings to conform to the site's settlement characteristics and minimize differential settlement. For soil sites, the potential heave or rebound of the excavation bottom, the effect of dewatering, and the effect of foundation loading during construction should be monitored by the Combined License applicant. The monitoring system shall consist of three primary elements as follows:

    - Piezometers to measure pore pressures in a soil layer prone to consolidation type settlement. Vibrating wire piezometers are preferred for this purpose because they are adequately sensitive and responsive and easily record positive and negative changes on a real time basis.

    - Settlement monuments placed directly on concrete, preferably on the mudmat for early construction monitoring and on the corners of the structures at grade once the mudmat monuments have been covered by backfill to be used for long-term monitoring. Monuments at grade are to be accessible with conventional surveying equipment.

    - Settlement telltales if monuments are not practical or if fills are used over consolidation type soils and it is necessary to monitor settlement of the consolidation type in-situ soils independent of the consolidation of the engineered fill soil. Most soil sites will not need this particular form of monitoring.

Develop graphs and plots of the field measurements to:

- Show movement (settlement or heave) versus time.
- Estimate construction loads versus time.
- Measure ground water levels from the dewatering activities versus time.

This data should be maintained during construction and post-construction as needed depending on the field measurement results.

2.5.4.4 Liquefaction

The Combined License applicant will demonstrate for soil sites that the potential for liquefaction is negligible for both the soil underneath the nuclear island foundation and the soil of the side embedment engaged in passive resistance adjacent to the nuclear island.

The AP1000 design has not been evaluated for a site where there is a liquefaction potential of the soil below the nuclear island. A COL applicant will satisfy the requirement that liquefaction beneath the nuclear island and adjacent seismic Category II structures needs not be considered by providing information concerning the properties and stability of supporting soils and rock consistent with the guidance of Regulatory Guide 1.206.

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2.5.4.5 Subsurface Uniformity

Soil structure interaction and foundation design are a function of the uniformity of the soil or rock below foundation. Although the design and analysis of the AP1000 is based on soil or rock conditions with uniform properties within horizontal layers, it includes provisions and design margins to accommodate many non-uniform sites. This subsection identifies the requirements for site investigation that may be used to demonstrate that:

- A site is “uniform” based on the criteria outlined in Site Foundation Material Evaluation Criteria below. If the site can be demonstrated to be “uniform,” no further site specific analysis is required to qualify the site for the AP1000.

- A “non-uniform” site is acceptable to locate the AP1000 based on the criteria for acceptability outlined in Site Foundation Material Evaluation Criteria below. Non-uniform sites may be shown to be acceptable as described in subsection 2.5.4.5.3.1 using site-specific evaluation as part of the Combined License application.

Considerations with respect to the materials underlying the nuclear island are the type of site, such as rock or soil, and whether the site can be considered uniform. If the site is non-uniform, the non-uniform soil characteristics, such as the location and profiles of soft and hard spots, should be considered. These considerations can be assessed with the information developed in response to Regulatory Guides 1.132 and 1.138. The geological investigations of subsections 2.5.1 and 2.5.4.6.1 provide information on the uniformity of the site, whether it may be geologically impacted, and whether the bedrock may be sloping or undulatory.

2.5.4.5.1 Not Used

2.5.4.5.2 Not Used

2.5.4.5.3 Site Foundation Material Evaluation Criteria

The AP1000 is designed for application at a site where the foundation conditions do not have extreme variation within the nuclear island footprint. This subsection provides criteria for evaluation of soil variability. The subsurface may consist of layers and these layers may dip with respect to the horizontal. If the dip is less than 20 degrees, the generic analysis using horizontal layers is applicable as described in NUREG/CR-0693 (Reference 2). The physical properties of the foundation medium may or may not vary systematically across a horizontal plane. The recommended methodology for checking uniformity is to calculate from the boring logs a series of “best-estimate” planes beneath the nuclear island footprint that define the top (and bottom) of each layer. The planes could represent stratigraphic boundaries, lithologic changes, and unconformities, but most important, they should represent boundaries between layers having different shear wave velocities. Shear wave velocity is the primary property used for defining uniformity of a site.

The distribution of bearing reactions under the basemat is a function of the subgrade modulus, which in turn is a function of the soil properties. The Combined License applicant shall demonstrate that the variation of subgrade modulus across the footprint is within the range considered for design of the nuclear island basemat. The farther that the non-uniform layer is located below the foundation, the less influence it has on the bearing pressures at the basemat. Lateral variability of the shear wave velocity at depths greater than 120 feet below grade (80 feet below the foundation) do not significantly affect the subgrade modulus.

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Subsurface conditions should be evaluated by the Combined License applicant based on the geologic investigation in accordance with Regulatory Guide 1.132. Subsurface conditions should be evaluated within the nuclear island footprint and 40 feet beyond the boundaries of the nuclear island footprint at depths less than 120 feet below grade. Subsurface conditions may be considered uniform if the geologic and stratigraphic features can be correlated from one boring or sounding location to the next with relatively smooth variations in thicknesses or properties of the geologic units. An occasional anomaly or a limited number of unexpected lateral variations may occur. If a site can be classified as uniform, it qualifies for the AP1000 based on analyses and evaluations performed to support design certification without additional site-specific analyses.

As an example of sites that are considered uniform, the variation of soil properties in the material below the foundation to a depth of 120 feet below finished grade within the nuclear island footprint and 40 feet beyond the boundaries of the nuclear island footprint meets the criteria outlined below:

- The depth to a given layer indicated on each boring log may not fall precisely on the postulated “best-estimate” plane. The deviation of the observed layers from the “best-estimate” planes should not exceed 5 percent of the observed depths from the ground surface to the plane. If the deviation is greater than 5 percent, additional planes may be appropriate or additional borings may be required. This thereby diminishes the spacing.

- For a layer with a low strain shear wave velocity greater than or equal to 2500 feet per second, the layer should have approximately uniform thickness and should have a dip no greater than 20 degrees, and the shear wave velocity at any location within any layer should not vary from the average velocity within the layer by more than 20 percent.

2.5.4.5.3.1 Site-Specific Subsurface Uniformity Design Basis

Many sites that do not meet the above criteria for a uniform site are acceptable for the AP1000. The key attribute for acceptability of the site for an AP1000 is the bearing pressure on the underside of the basemat. A site having local soft or hard spots within a layer or layers does not meet the criteria for a uniform site. Non-uniform soil conditions may also require evaluation of the AP1000 seismic response as described in subsection 2.5.2.3.

As described in subsection 3.8.5, the nuclear island foundation is designed specifically for bearing pressures of 120 percent of those of the uniform soil properties case. Evaluation criteria are defined to evaluate sites that do not satisfy the site parameters directly. The design basis provided below is included to provide a clear specification of the design commitment and evaluation criteria required to demonstrate that a site-specific application satisfies AP1000 requirements. Application of the AP1000 to sites using this site-specific evaluation is not approved as part of the AP1000 design certification and the evaluation should be provided and reviewed as part of the Combined License application.

Rigid Basemat Evaluation

A site with nonuniform soil properties may be demonstrated to be acceptable by evaluation of the bearing pressures on the underside of a rigid rectangular basemat equivalent to the nuclear island. The soils identified in the site investigation may be included in a finite element model of the soil to analyze the effect of the lateral variability. When the variability identified at the site can be modeled in two dimensions (there is not significant variability in one horizontal direction), 2D

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analyses may be used. Where the variability occurs in both horizontal directions, a 3D analysis should be performed. Bearing pressures are calculated in a linear analysis for unit vertical load and overturning moments. For the site to be acceptable, the bearing pressures from this analysis need to be less than or equal to 120 percent of the bearing pressures calculated in similar analyses for a site having uniform soil properties.

Alternatively, the safe shutdown earthquake loads may be determined from a site-specific seismic analysis of the nuclear island using site-specific inputs as described in subsections 2.5.2.1 or 2.5.2.3. For the site to be acceptable, the bearing pressures from the site-specific analyses (with site-specific response and site-specific soil properties) need to be less than or equal to 120 percent of the bearing pressures calculated in similar rigid basemat analyses using the AP1000 design ground motion at a site having uniform soil properties.

Flexible Basemat Evaluation

For sites having bedrock close to the foundation level, the assumption of a rigid basemat may be overly conservative because local deformation of the basemat will reduce the effect of local soil variability. For such sites, a site-specific analysis may be performed using the AP1000 basemat model and methodology described in subsection 3.8.5. The soils may be represented by soil springs or by a finite element model of the soil depending on the type of variability identified at the site. The safe shutdown earthquake loads are those from the AP1000 design soil case representative of the site-specific soil. Alternatively, bearing pressures may be determined from a site-specific soil structure interaction analysis using site-specific inputs as described in subsection 2.5.2.3. For the site to be acceptable, the bearing pressures from the site-specific analyses, including static and dynamic loads, need to be less than the design bearing strength of each portion of the basemat.

2.5.4.6 Combined License Information

Combined License applicants referencing the AP1000 design will address the following site specific information related to the geotechnical engineering aspects of the site. No further action is required for sites within the bounds of the site parameters.

2.5.4.6.1 Site and Structures – Site-specific information regarding the underlying site conditions and geologic features will be addressed. This information will include site topographical features, as well as the locations of seismic Category I structures.

2.5.4.6.2 The Combined License applicant will establish the properties of the foundation soils to be within the range considered for design of the nuclear island basemat.

Properties of Underlying Materials – A determination of the static and dynamic engineering properties of foundation soils and rocks in the site area will be addressed. This information will include a discussion of the type, quantity, extent, and purpose of field explorations, as well as logs of borings and test pits. Results of field plate load tests, field permeability tests, and other special field tests (e.g., bore-hole extensometer or pressuremeter tests) will also be provided. Results of geophysical surveys will be presented in tables and profiles. Data will be provided pertaining to site-specific soil layers (including their thicknesses, densities, moduli, and Poisson's ratios) between the basemat and the underlying rock stratum. Plot plans and profiles of site explorations will be provided. The Combined License applicant is to demonstrate that the minimum soil angle of internal friction is greater

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than or equal to 35 degrees below the footprint of the nuclear island at its excavation depth as specified in Table 2-1. If the minimum soil angle of internal friction is below 35 degrees, then the Combined License applicant will evaluate stability as described in subsections 3.8.5.5.3 and 3.8.5.5.5 using the site-specific soil properties.

Laboratory Investigations of Underlying Materials – Information about the number and type of laboratory tests and the location of samples used to investigate underlying materials will be provided. Discussion of the results of laboratory tests on disturbed and undisturbed soil and rock samples obtained from field investigations will be provided.

2.5.4.6.3 Excavation and Backfill – Information concerning the extent (horizontal and vertical) of seismic Category I excavations, fills, and slopes, if any will be addressed. The sources, quantities, and static and dynamic engineering properties of borrow materials will be described in the site-specific application. The compaction requirements, results of field compaction tests, and fill material properties (such as moisture content, density, permeability, compressibility, and gradation) will also be provided. Information will be provided concerning the specific soil retention system, for example, the soil nailing system or mechanically stabilized earth walls, including the length and size of the soil nails or tension reinforcement, which is based on actual soil conditions and applied construction surcharge loads. If backfill is to be placed adjacent to the exterior walls of the nuclear island, information will be provided concerning compaction of the backfill and any additional loads on the exterior walls of the nuclear island.

2.5.4.6.4 Ground Water Conditions – Groundwater conditions will be described relative to the foundation stability of the safety-related structures at the site. The soil properties of the various layers under possible groundwater conditions during the life of the plant will be compared to the range of values assumed in the standard design in Table 2-1.

2.5.4.6.5 Liquefaction Potential – Soils under and around seismic Category I structures will be evaluated for liquefaction potential for the site specific SSE ground motion. This should include justification of the selection of the soil properties, as well as the magnitude, duration, and number of excitation cycles of the earthquake used in the liquefaction potential evaluation (e.g., laboratory tests, field tests, and published data). Liquefaction potential will also be evaluated to address seismic margin.

2.5.4.6.6 Bearing Capacity – The Combined License applicant referencing the AP1000 certified design will verify that the site-specific allowable soil bearing capacities for static and dynamic loads are equal to or greater than the values documented in Table 2-1, or will provide a site-specific evaluation as described in subsection 2.5.4.2. The acceptance criteria for this evaluation are those of Standard Review Plan 2.5.4 as follows:

- The static and dynamic loads, and the stresses and strains induced in the soil surrounding and underlying the nuclear island, are conservatively and realistically evaluated.

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- The consequences of the induced soil stresses and strains, as they influence the soil surrounding and underlying the nuclear island, have been conservatively assessed.

2.5.4.6.7 Earth Pressures – The Combined License applicant will describe the design for static and dynamic lateral earth pressures and hydrostatic groundwater pressures acting on plant safety-related facilities using soil parameters as evaluated in previous subsections.

2.5.4.6.8 Soil Properties for Seismic Analysis of Buried Pipes – The AP1000 does not utilize safety-related buried piping. No additional information is required on soil properties.

2.5.4.6.9 Static and Dynamic Stability of Facilities – Soil characteristics affecting the stability of the nuclear island will be addressed including foundation rebound, settlement, and differential settlement.

2.5.4.6.10 Subsurface Instrumentation – Data will be provided on instrumentation, if any, proposed for monitoring the performance of the foundations of the nuclear island. This will specify the type, location, and purpose of each instrument, as well as significant details of installation methods. The location and installation procedures for permanent benchmarks and markers for monitoring the settlement will be addressed.

2.5.4.6.11 Settlement of Nuclear Island **–** Data will be provided on short-term (elastic) and long-term (heave and consolidation) settlement for soil sites for the history of loads imposed on the nuclear island foundation and adjacent buildings consistent with the construction sequence. The resulting time-history of settlements includes construction activities such as dewatering, excavation, bearing surface preparation, placement of the basemat, and construction of the superstructure. Subsection 3.8.5.4.2 includes analyses of settlement during construction completed to support the design certification and the required limitations on construction sequence for some sites. The limitations on construction sequence impose limits on the placement of concrete for the shield building and the auxiliary building prior to completion of both buildings at elevation 82′-6″. Special construction requirements will be described, if required, to accommodate site-specific settlement predicted to exceed the values shown in Table 2.5-1.

2.5.4.6.12 Waterproofing System – The Combined License applicant will provide a waterproofing system used for the foundation mat (mudmat) and below grade exterior walls exposed to flood and groundwater under seismic Category I structures. The waterproofing membrane should be placed immediately beneath the upper mudmat, and on top of the lower mudmat. The performance requirements to be met by the COL applicant for the waterproofing system are described in subsection 3.4.1.1.1.1.

2.5.5 Combined License Information for Stability of Slopes

Combined License applicants referencing the AP1000 design will address site-specific information about the static and dynamic stability of soil and rock slopes, the failure of which could adversely affect the nuclear island.

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2.5.6 Combined License Information for Embankments and Dams

Combined License applicants referencing the AP1000 design will address site-specific information about the static and dynamic stability of embankments and dams, the failure of which could adversely affect the nuclear island.

2.6 References

1. Not Used.

2. NUREG/CR-0693, “Seismic Input and Soil Structure Interaction,” February 1979.

3. APP-GW-GLR-115, “Effect of High Frequency Seismic Content on SSCs,” Westinghouse Electric Company LLC.

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Table 2-1 (Sheet 1 of 5)
Site Parameters

Air Temperature	
Maximum Safety [a]	115°F dry bulb/86.1°F coincident wet bulb[g] 86.1°F wet bulb (noncoincident)
Minimum Safety [a]	-40°F
Maximum Normal [b]	101°F dry bulb/80.1°F coincident wet bulb 80.1°F wet bulb (noncoincident) [d]
Minimum Normal [b]	-10°F
Wind Speed	
Operating Basis	145 mph (3 second gust); importance factor 1.15 (safety), 1.0 (nonsafety); exposure C; topographic factor 1.0
Tornado	300 mph Maximum pressure differential of 2.0 lb/in^2
Seismic	
CSDRS	CSDRS free field peak ground acceleration of 0.30g with modified Regulatory Guide 1.60 response spectra (See Tier 1 Figures 5.0-1 and 5.0-2). The SSE is now referred to as CSDRS. Seismic input is defined at finished grade except for sites where the nuclear island is founded on hard rock. If the site-specific spectra exceed the response spectra in Tier 1 Figures 5.0-1 and 5.0-2 at any frequency, or if soil conditions are outside the range evaluated for AP1000 design certification, a site-specific evaluation can be performed. This evaluation will consist of a site-specific dynamic analysis and generation of in-structure response spectra at key locations to be compared with the floor response spectra of the certified design at 5-percent damping. The site is acceptable if the floor response spectra from the site-specific evaluation do not exceed the AP1000 spectra for each of the locations or the exceedances are justified. The hard rock high frequency (HRHF) envelope response spectra are shown in Tier 1 Figure 5.0-3 and Tier 1 Figure 5.0-4 defined at the foundation level for 5% damping. The HRHF envelope response spectra provide an alternative set of spectra for evaluation of site-specific GMRS. A site is acceptable if its site-specific GMRS falls within the AP1000 HRHF envelope response spectra. Evaluation of a site for application of the HRHF envelope response spectra includes consideration of the limitation on shear wave velocity identified for use of the HRHF envelope response spectra. This limitation is defined by a shear wave velocity at the bottom of the basemat equal to or higher than 7,500 fps, while maintaining a shear wave velocity equal to or above 8,000 fps at the lower depths.

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Table 2-1 (Sheet 2 of 5)
Site Parameters

Fault Displacement Potential	No potential fault displacement considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures.
Soil	
Average Allowable Static Bearing Capacity	The allowable bearing capacity, including a factor of safety appropriate for the design load combination, shall be greater than or equal to the average bearing demand of 8,900 lb/ft^2 over the footprint of the nuclear island at its excavation depth.
Dynamic Bearing Capacity for Normal Plus SSE	The allowable bearing capacity, including a factor of safety appropriate for the design load combination, shall be greater than or equal to the maximum bearing demand of 35,000 lb/ft^2 at the edge of the nuclear island at its excavation depth, or Site-specific analyses demonstrate factor of safety appropriate for normal plus safe shutdown earthquake loads.
Shear Wave Velocity	Greater than or equal to 1,000 ft/sec based on minimum low-strain soil properties over the footprint of the nuclear island at its excavation depth

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Table 2-1 (Sheet 3 of 5)
Site Parameters

Lateral Variability	Soils supporting the nuclear island should not have extreme variations in subgrade stiffness. This may be demonstrated by one of the following: 1. Soils supporting the nuclear island are uniform in accordance with Regulatory Guide 1.132 if the geologic and stratigraphic features at depths less than 120 feet below grade can be correlated from one boring or sounding location to the next with relatively smooth variations in thicknesses or properties of the geologic units, or 2. Site-specific assessment of subsurface conditions demonstrates that the bearing pressures below the footprint of the nuclear island do not exceed 120% of those from the generic analyses of the nuclear island at a uniform site, or 3. Site-specific analysis of the nuclear island basemat demonstrates that the site-specific demand is within the capacity of the basemat. As an example of sites that are considered uniform, the variation of shear wave velocity in the material below the foundation to a depth of 120 feet below finished grade within the nuclear island footprint and 40 feet beyond the boundaries of the nuclear island footprint meets the criteria in the case outlined below: Case 1: For a layer with a low strain shear wave velocity greater than or equal to 2500 feet per second, the layer should have approximately uniform thickness, should have a dip not greater than 20 degrees, and should have less than 20 percent variation in the shear wave velocity from the average velocity in any layer.
Liquefaction Potential	No liquefaction considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures.
Minimum Soil Angle of Internal Friction	Greater than or equal to 35 degrees below footprint of nuclear island at its excavation depth. If the minimum soil angle of internal friction is below 35 degrees, a site-specific analysis shall be performed using the site-specific soil properties to demonstrate stability.
Missiles	
Tornado	4000 - lb automobile at 105 mph horizontal, 74 mph vertical 275 - lb, 8 in. shell at 105 mph horizontal, 74 mph vertical 1 inch diameter steel ball at 105 mph in the most damaging direction

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Table 2-1 (Sheet 4 of 5)
Site Parameters

Flood Level	Less than plant elevation 100′
Ground Water Level	Less than plant elevation 98′
Plant Grade Elevation	Less than plant elevation 100′ except for portion at a higher elevation adjacent to the annex building
Precipitation	
Rain	20.7 in./hr [1-hr 1-mi^2 PMP]
Snow/Ice	75 pounds per square foot on ground with exposure factor of 1.0 and importance factors of 1.2 (safety) and 1.0 (non-safety)
Atmospheric Dispersion Values - χ/Q[(e)]	
Site boundary (0-2 hr)	≤ 5.1 x 10^{-4} sec/m^3
Site boundary (annual average)	≤ 2.0 x 10^{-5} sec/m^3
Low population zone boundary	
0 - 8 hr	≤ 2.2 x 10^{-4} sec/m^3
8 - 24 hr	≤ 1.6 x 10^{-4} sec/m^3
24 - 96 hr	≤ 1.0 x 10^{-4} sec/m^3
96 - 720 hr	≤ 8.0 x 10^{-5} sec/m^3
Population Distribution	
Exclusion area (site)	0.5 mi

Notes:

(a) Maximum and minimum safety values are based on historical data and exclude peaks of less than 2 hours duration.

(b) The maximum normal value is the 1-percent seasonal exceedance temperature. The minimum normal value is the 99-percent seasonal exceedance temperature. The minimum temperature is for the months of December, January, and February in the northern hemisphere. The maximum temperature is for the months of June through September in the northern hemisphere. The 1-percent seasonal exceedance is approximately equivalent to the annual 0.4-percent exceedance. The 99-percent seasonal exceedance is approximately equivalent to the annual 99.6-percent exceedance.

(c) With ground response spectra as given in Figures 3.7.1-1 and 3.7.1-2. Seismic input is defined at finished grade except for sites where the nuclear island is founded on hard rock.

(d) The noncoincident wet bulb temperature is applicable to the cooling tower only.

(e) For AP1000, the terms “site boundary” and “exclusion area boundary” are used interchangeably. Thus, the χ/Q specified for the site boundary applies whenever a discussion refers to the exclusion area boundary.

(f) Sites that fall within the hard rock high frequency envelope response spectra given in Figure 3I.1-1 and Figure 3I.1-2, and satisfy the limitation on shear wave velocity in subsection 2.5.2.1, are acceptable.

(g) The containment pressure response analysis is based on a conservative set of dry-bulb and wet-bulb temperatures. These results envelop any conditions where the dry-bulb temperature is 115°F or less and wet-bulb temperature of less than or equal to 86.1°F.

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Table 2-1 (Sheet 5 of 5)
Site Parameters

Control Room Atmospheric Dispersion Factors (χ /Q) for Accident Dose Analysis						
χ/Q (s/m3) at HVAC Intake for the Identified Release Points(1)						
	Plant Vent or PCS Air Diffuser(3)	**Ground Level Containment Release Points(4)**	**PORV and Safety Valve Releases(5)**	**Steam Line Break Releases**	**Fuel Handling Area(6)**	**Condenser Air Removal Stack(7)**
0 - 2 hours	2.53E-03	4.00E-03	1.92E-02	2.13E-02	6.0E-3	6.0E-3
2 - 8 hours	1.98E-03	2.28E-03	1.60E-02	1.76E-02	4.0E-3	4.0E-3
8 - 24 hours	7.96E-04	1.03E-03	6.90E-03	7.50E-03	2.0E-3	2.0E-3
1 - 4 days	6.40E-04	9.03E-04	4.96E-03	5.43E-03	1.5E-3	1.5E-3
4 - 30 days	4.78E-04	7.13E-04	4.16E-03	4.55E-03	1.0E-3	1.0E-3
χ/Q (s/m^3) at Annex Building Door for the Identified Release Points(2)						
	Plant Vent or PCS Air Diffuser(3)	**Ground Level Containment Release Points(4)**	**PORV and Safety Valve Releases(5)**	**Steam Line Break Releases**	**Fuel Handling Area(6)**	**Condenser Air Removal Stack(7)**
0 - 2 hours	1.0E-3	1.0E-3	4.0E-3	4.0E-3	6.0E-3	2.0E-2
2 - 8 hours	7.5E-4	7.5E-4	3.2E-3	3.2E-3	4.0E-3	1.8E-2
8 - 24 hours	3.5E-4	3.5E-4	1.2E-3	1.2E-3	2.0E-3	7.0E-3
1 - 4 days	2.8E-4	2.8E-4	1.0E-3	1.0E-3	1.5E-3	5.0E-3
4 - 30 days	2.5E-4	2.5E-4	8.0E-4	8.0E-4	1.0E-3	4.5E-3

Notes:

1. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the non-safety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.
2. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.
3. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.

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4. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.
5. The listed values bound the dispersion factors for releases from the steam line safety & power-operated relief valves. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and for the secondary side release from a rod ejection accident.
6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.
7. This release point is included for information only as a potential activity release point. None of the design basis accident radiological consequences analyses model release from this point.

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Table 2.2-1
AP1000 Onsite Explosion Safe Distances

Material	Explosion Minimum Safe Distance[1] (feet)	Flammable Vapor Cloud Safe Distance[1] (feet)	AP1000 Distance to SSC (feet)
Liquid Hydrogen, H_2	577	175	635
Pressurized Gaseous Hydrogen, H_2 (PGS Hydrogen Switchover Station)	10.5	Not Applicable	24
Pressurized Gaseous Hydrogen, H_2 (Spare Cylinders)	6	Not Applicable	10
Hydrazine, N_2H_4	45	Not Applicable	123
Ethanolamine, C_2H_7NO Methoxypropylamine, (MOPA), $C_4H_{11}NO$ Morpholine, $O(CH_2CH_2)_2NH$	75	Not Applicable	123
No. 2 Diesel Fuel Oil (DOS-MT-01A/B)	119	Not Applicable	313
No. 2 Diesel Fuel Oil (DOS-MT-02A/B)	34	Not Applicable	182
No. 2 Diesel Fuel Oil (DOS-MT-03)	23	Not Applicable	53
No. 2 Diesel Fuel Oil (FPS-MT-02)	20	Not Applicable	368
No. 2 Diesel Fuel Oil (ASS-EP-006 Temporary Auxiliary Boiler Diesel Tank)	34	Not Applicable	250
Waste Oil	102	Not Applicable	201

Note:

1. Safe distance is to nearest point of nuclear island SSC.

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Table 2.5-1
Limits of Acceptable Settlement Without Additional Evaluation

Differential Across Nuclear Island Foundation Mat	Total for Nuclear Island Foundation Mat	Differential Between Nuclear Island and Turbine Building(1)	Differential Between Nuclear Island and Other Buildings(1)
1/2 inch 50 ft	6 inches	3 inches	3 inches

Note:

1. Differential settlement is measured at the center of the nuclear island and the center of the adjacent structures.

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Chapter 3 Design of Structures, Components, Equipment and Systems

3.1 Conformance with Nuclear Regulatory Commission General Design Criteria

This section discusses the extent to which the AP1000 design criteria for safety-related structures, systems, and components comply with 10 CFR 50, Appendix A. As presented in this section, each criterion is first quoted and then discussed. For some criteria, the AP1000 advanced passive design features are deemed to be significantly different in certain specific areas from those design features considered when the General Design Criteria were formulated. In those instances, the means by which the AP1000 design complies with the intent of the General Design Criterion is indicated. Where additional information is required for a complete discussion, the appropriate Design Control Document (DCD) sections are referenced.

3.1.1 Overall Requirements

Criterion 1 – Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified, as necessary, to assure a quality product, in keeping with the required safety function.

A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

AP1000 Compliance

The Quality Assurance Program for the AP1000 provides confidence that safety-related items and services are designed, procured, fabricated, inspected, and tested to quality standards commensurate with the safety-related functions to be performed. This program also applies to design services subcontracted to external organizations. The quality assurance program for erection of structures, systems, and components will be identified before the construction phase of the AP1000 project. The AP1000 quality assurance program is described in Chapter 17, including its compliance with ASME NQA-1.

Design, procurement, fabrication, inspection, and testing are performed according to recognized codes, standards, and design criteria that comply with the requirements of 10 CFR 50.55a. As necessary, supplemental standards, design criteria, and requirements are developed by the AP1000 designers. A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(z). The alternate classification is discussed in Subsection 5.2.1.3.

Appropriate records documenting that design, procurement, fabrication, inspection, and testing comply with the applicable codes, standards, and design criteria are maintained according to appropriate, applicable laws and regulations, either by or under the control of the Combined License applicant.

In the passive AP1000 design, systems necessary to provide the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures

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comparable to the guideline exposures of 10 CFR 50.34 are classified as safety-related. Therefore, the AP1000 complies with the intent of Criterion 1.

The principal design criteria, design bases, codes, and standards applied to the facility are identified in Section 3.2. Additional details may be found in the pertinent sections dealing with safety-related structures, systems, and components.

Criterion 2 – Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of the capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

AP1000 Compliance

The safety-related structures, systems, and components are designed to withstand the effects of natural phenomena without loss of the capability to perform their safety-related functions, or are designed such that their response or failure will be in a safe condition. Those structures, systems, and components vital to the shutdown capability of the reactor are designed to withstand the maximum probable natural phenomena at the intended site.

Accident analyses consider conservative site conditions that envelope expected sites. Appropriate combinations of structural loadings from normal, accident, and natural phenomena are considered in the plant design. The design of the plant in relationship to those natural phenomena is addressed.

Seismic and quality group classifications and other pertinent standards and information are given in the sections discussing individual structures, systems, and components as well as in Chapter 3. The nature and magnitude of the natural phenomena considered in the design of this plant are discussed in Chapter 2.

Criterion 3 – Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

AP1000 Compliance

The safety-related structures, systems, and components are designed to minimize the probability and effect of fires and explosions. Noncombustible and fire-resistant materials are used in the containment and main control room. Additionally noncombustible and fire-resistant materials are used on components of safety-related systems, and elsewhere in the plant where fire is a potential risk to safety-related systems.

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For example, electrical cables have a fire-retardant jacketing, and fire barriers are used at fire area boundaries. The AP1000 design approach includes designing the safety-related systems with redundant divisions, and locating these redundant divisions in separate safety-related areas.

Equipment and facilities for fire protection, including detecting, alarming, and extinguishing functions, are provided to help protect both plant equipment and personnel from fire, explosion, and the resultant release of toxic vapors. Fire protection is provided by deluge systems (water spray), sprinklers, and portable extinguishers. Fire fighting systems are designed so that their rupture or inadvertent operation will not prevent safety-related systems from performing their design functions.

The following codes, guides, and standards are used as guidelines in the design of the fire protection system and equipment. The system and equipment conform to the applicable portions of the following documents:

- National Fire Protection Association Codes and Standards
- BTP-CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," July 1981, as addressed in Table 9.5.1-1

Subsection 9.5.1 describes the AP1000 fire protection system and equipment, including conformance with the applicable portions of these codes and standards and reference to specific fire protection codes and standards.

Criterion 4 – Environmental and Dynamic Effects Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

AP1000 Compliance

Safety-related structures, systems, and components are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents.

The AP1000 design has emphasized the minimization of missiles, pipe whip, and fluid discharge by a combination of separation of safe shutdown components and design to prevent the dynamic effects of postulated pipe ruptures based on the application of the leak-before-break approach. This analysis is discussed in Subsection 3.4.3.5 and Section 3.6.

The AP1000 structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. Details of the design, environmental testing, and construction of these structures, systems, and components are given in the sections that discuss individual structures, systems, and components, as well as in Sections 3.5 and 3.6.

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Criterion 5 – Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit.

AP1000 Compliance

The AP1000 is a single-unit plant. If more than one unit were built on the same site, none of the safety-related systems would be shared.

3.1.2 Protection by Multiple Fission Product Barriers

Criterion 10 – Reactor Design

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

AP1000 Compliance

The reactor core and associated coolant, control, and protection systems are designed to the following criteria:

- No fuel damage occurs during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II). For normal operation, the plant is designed to accommodate a fuel defect level of up to 0.25 percent. Fuel damage, as used here, is defined as penetration of the fission product barrier, that is, the fuel rod cladding. The small number of clad defects that may occur are within the capability of the plant cleanup system and are consistent with the plant design bases. For additional information see Section 11.1.

- The reactor can be returned to a safe shutdown state following a Condition III event, with only a small fraction of the fuel rods damaged, although sufficient fuel damage might occur to preclude the immediate resumption of operation.

- The core remains intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

The reactor protection system is designed to actuate a reactor trip whenever necessary to prevent exceeding the fuel design limits. The core design, together with the process and decay heat removal systems, provide this capability under expected conditions of normal operation, with appropriate margins for uncertainties and anticipated transient situations. This includes the effects of the loss of reactor coolant flow, trip of the turbine generator, loss of normal feedwater, and loss of both normal and preferred power sources.

Chapter 4, Reactor, describes the mechanical components of the reactor and reactor core, including the fuel rods and fuel assemblies, the mechanical design, nuclear design, and the thermal hydraulic design. Chapter 7 provides details of the control and protection systems instrumentation design and logic. This information supports the accident analyses documented in Chapter 15. The acceptable fuel design limits are not exceeded for Condition I and II events. Acceptable core cooling is provided for Condition III and IV events.

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Criterion 11 – Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power-operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

AP1000 Compliance

When the reactor is critical, the negative fuel temperature reactivity effects (Doppler feedback) provide prompt reactivity feedback to compensate for a rapid, uncontrolled reactivity excursion. The negative Doppler coefficient of reactivity is provided by the use of a low-enrichment fuel design. This Doppler feedback is the primary reactivity feedback mechanism to provide the inherent core reactivity protection during rapid core reactivity excursions.

For slower reactivity transients that result in moderator temperature increases, the nonpositive moderator temperature coefficient of reactivity provides compensatory reactivity feedback to help control these slower transients. The overall core design establishes a nonpositive moderator temperature coefficient of reactivity.

Chapter 4 provides information pertaining to the core design.

Criterion 12 – Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

AP1000 Compliance

Power oscillations of the fundamental mode are inherently eliminated by negative Doppler and nonpositive moderator temperature coefficients of reactivity.

Oscillations, due to xenon spatial effects, in the radial and azimuthal overtone modes are heavily damped because of the inherent design and due to the negative Doppler and nonpositive moderator temperature coefficients of reactivity.

Oscillations due to xenon spatial effects may occur in the axial first overtone mode. Reactor trip functions are provided, using the measured axial power imbalance as an input, so that the fuel design limits are not exceeded during axial xenon oscillations.

If it is necessary to maintain axial imbalance within the limits (that is, imbalances that are alarmed to the operator and are within the imbalance trip setpoints), the operator can suppress axial xenon oscillations by control rod motions or temporary power reductions or both.

Oscillations due to spatial xenon effects, in axial modes higher than the first overtone, are heavily damped because of the inherent design and the negative Doppler coefficient of reactivity.

The stability of the core against xenon-induced power oscillations and the functional requirements of instrumentation for monitoring and measuring core power distribution are discussed in Chapter 4. Details of the instrumentation design and logic are discussed in Chapter 7.

Criterion 13 – Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its

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associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

AP1000 Compliance

Instrumentation and controls are provided to monitor and control neutron flux, control rod position, fluid temperatures, pressures, flows, and levels, as necessary, to maintain plant safety. Instrumentation is provided in the reactor coolant system, steam and power conversion system, containment, engineered safety systems, radioactive waste management systems, and other auxiliary systems.

See Section 7.5 for a discussion of indications that are required for operator use under normal operating and accident conditions. Criteria regarding layout of the controls and displays are provided in Chapter 18.

The quantity and types of process instrumentation used provide safe and orderly operation of systems over the design range of plant operations, including accident conditions.

Criterion 14 – Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

AP1000 Compliance

The reactor coolant pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, while maintaining stresses within applicable limits. Consideration is given to loadings under normal operating conditions and to abnormal loadings, such as seismic loadings. The piping is protected from overpressure by means of pressure-relieving devices, as required by ASME Code, Section III. See Subsection 5.2.2 for additional information.

Reactor coolant pressure boundary materials and fabrication techniques are such that there is a low probability of gross rupture or significant leakage. The AP1000 reactor coolant system design incorporates revised pipe-break criteria (leak-before-break) to reduce or eliminate the need to consider the dynamic effects of pipe breaks. The configuration and materials of the reactor coolant system have been selected such that the pipe stresses meet the leak-before-break criteria. See Subsection 3.6.3 for additional information.

The AP1000 reactor core and reactor internals are designed to limit neutron fluence on the reactor vessel. See Section 5.4 and Chapter 4 for additional information.

The reactor vessel is manufactured from low-alloy carbon steel clad with 308L stainless steel weld overlay on wetted surfaces. The vessel shell is constructed of ring-rolled forgings that eliminate vertical weld seams. Chemical composition of the forging material is controlled to improve radiation resistance of the vessel. (See Criterion 31 for further discussion of the reactor coolant pressure boundary.)

Coolant chemistry is controlled to protect the materials of construction of the reactor coolant pressure boundary from corrosion. See Subsection 5.2.3 for additional information.

The reactor coolant pressure boundary welds are accessible for in-service inspections to assess structural and leaktight integrity. For the reactor vessel, a material surveillance program is provided. Instrumentation is provided to detect significant leakage from the reactor coolant pressure boundary, with indication in the main control room. See Subsection 5.2.4 for additional information.

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A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary is nonsafety-related. This portion of the system is capable of being automatically isolated by safety-related valves that are designed and qualified for the design requirements.

Criterion 15 – Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation, including anticipated operational occurrences.

AP1000 Compliance

Steady-state and transient analyses are performed to demonstrate that reactor coolant system design conditions are not exceeded during normal operation. Protection and control setpoints are based on these analyses. See Chapter 15 for additional information.

The reactor coolant system stress analysis and the leak-before-break analyses are described in Appendices 3B and 3C. See Section 5.3 for additional information.

Two safety valves are provided for the reactor coolant system. These valves and their setpoints meet the ASME Code, Section III criteria for overpressure protection. See Subsection 5.2.2 for additional information.

Criterion 16 – Containment Design

The reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

AP1000 Compliance

The containment is an integral part of the overall containment system, whose function is to contain the release of airborne radioactivity following postulated design basis accidents and to provide shielding for the reactor core and the reactor coolant system during normal operations. The containment consists of a steel containment vessel and is surrounded by a concrete shield building.

The containment vessel, which is a free-standing steel shell, is an integral part of the passive containment cooling system, whose function is to provide the safety-related ultimate heat sink for the removal of the reactor coolant system sensible heat, core decay heat, and stored energy. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents.

The containment is designed to house the reactor coolant system and other related systems. The containment vessel functions as an essentially leaktight barrier. It is protected against postulated missiles from external sources as well as missiles produced by internal equipment failures.

Containment penetrations are isolated according to the provisions of GDCs 54, 55, 56, and 57.

Criterion 17 – Electrical Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming that the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational

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occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system shall have sufficient independence, redundancy, and testability to perform their safety functions, assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights-of-way) designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time, following a loss of all onsite alternating current power supplies and other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss of coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

AP1000 Compliance

The AP1000 plant design supports an exemption to the requirement of GDC 17 for two physically independent offsite circuits by providing safety-related passive systems for core cooling and containment integrity, and multiple nonsafety-related onsite and offsite electric power sources for other functions. See Section 6.3 for additional information on the systems for core cooling.

A reliable dc power source supplied by batteries provides power for the safety-related valves and instrumentation during transient and accident conditions.

The Class 1E dc and UPS system is the only safety-related power source required to monitor and actuate the safety-related passive systems. Otherwise, the plant is designed to maintain core cooling and containment integrity, independent of nonsafety-related ac power sources indefinitely. The only electric power source necessary to accomplish these safety-related functions is the Class 1E dc and UPS power system which includes the associated safety-related 120V ac distribution switchgear.

Although the AP1000 is designed with reliable nonsafety-related offsite and onsite ac power that are normally expected to be available for important plant functions, nonsafety-related ac power is not relied upon to maintain the core cooling or containment integrity.

The nonsafety-related ac power system is designed such that plant auxiliaries can be powered from the grid under all modes of operation. During loss of normal ac power and offsite power, the ac power is supplied by the onsite standby diesel-generators. Preassigned loads and equipment are automatically loaded on the diesel-generators in a predetermined sequence. Additional loads can be manually added as required. The onsite standby power system is not required for safe shutdown of the plant.

Criterion 18 – Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The

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systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

AP1000 Compliance

The AP1000 is designed so that only the Class 1E dc and UPS system is required in order to initiate and actuate the systems necessary for maintaining core cooling and containment integrity. The safety-related dc power system design complies with GDC 18. Compliance with GDC 18 is achieved by designing testability and inspection capability into the system. The associated testing requirements are contained in the Technical Specifications.

Criterion 19 – Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

AP1000 Compliance

The AP1000 main control room provides the man-machine interfaces required to operate the plant safely and efficiently under normal conditions and to maintain it in a safe manner under accident conditions, including LOCAs. Simplified passive safety-related system designs are provided that do not rely upon operator action to maintain core cooling for design basis accidents. Operator action outside the main control room to mitigate the consequences of an accident is permitted.

The main control room is shielded by the containment and auxiliary building from direct gamma radiation and inhalation doses resulting from the postulated release of fission products inside containment. Refer to Chapter 15 for additional information on accident conditions. The main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS) allows access to and occupancy of the main control room under accident conditions as described in Subsection 9.4.1. Sufficient shielding and the main control room/control support area HVAC subsystem provide adequate protection so that personnel will not receive radiation exposure in excess of 5 rem whole-body or its equivalent to any part of the body for the duration of the accident.

If ac power is unavailable for more than 10 minutes or if main control room differential pressure is below the “Low” setpoint for more than 10 minutes or if “High-2” particulate or “High-2” iodine radioactivity is detected in the main control room supply air duct, which may lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The main control room emergency habitability system also allows access to and occupancy of the main control room under accident conditions. The emergency main control room habitability system is designed to satisfy seismic Category I requirements as described in Section 3.2; the system design is described in Section 6.4.

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In the event that the operators are forced to abandon the main control room, a workstation is provided with remote shutdown capability. A main control room evacuation is not assumed to occur simultaneously with design basis events. The remote shutdown workstation is described in Section 7.4.

3.1.3 Protection and Reactivity Control Systems

Criterion 20 – Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

AP1000 Compliance

The protection system is a microprocessor-based system that trips the reactor and actuates engineered safety features when predetermined limits are exceeded or when manually initiated.

The reactor trip portion of the protection system includes four independent, redundant, physically separated, electrically-isolated divisions. The coincidence circuits guard against the loss of protection or the generation of false protection signals due to equipment failures through the use of a two-out-of-four logic and built-in operational bypasses.

Independent, redundant, physically separated, electrically-isolated engineered safety features trains are provided. Signal conditioning for the plant sensors is provided.

See Chapter 7 for additional information concerning the design of the protection system.

Criterion 21 – Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (l) no single failure results in the loss of the protection function and (2) removal from service of any component or channel does not result in the loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

AP1000 Compliance

The protection system is designed for functional reliability and in-service testability. The design employs redundant logic trains and measurement and equipment diversity.

The protection system equipment includes integral testing circuits. System equipment, from input to output, in the protection cabinets and the engineered safety features cabinets, is tested. Simulated inputs replace the field signals. Outputs are monitored for validity. Manual and automatic testing is used to test the final stages of the reactor trip circuits and the reactor trip switchgear. Testing of cabinets and communications links verifies the functional operation of the equipment and the hardware. See Chapter 7 for further information concerning the test capabilities of the protection system.

Criterion 22 – Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels

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do not result in the loss of the protection function or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

AP1000 Compliance

Design of the protection systems includes consideration of natural phenomena, normal maintenance, testing, and accident conditions so that the protection functions are available.

Protection system components are designed, arranged, and qualified for operation in the environment accompanying any emergency situation in which the components are required to function.

Functional diversity has been designed into the system. The extent of this functional diversity is demonstrated for a variety of postulated accidents. Diverse protection functions automatically serve to mitigate the consequences of an event. Chapter 15 identifies the primary and diverse protective functions for each of the analyzed events.

Sufficient redundancy and independence are designed into the protection systems so that no single failure or removal from service of any component or channel of a system results in loss of the protection function. Functional diversity and location diversity are designed into the system.

Automatic reactor trip is initiated by neutron flux measurements, reactor coolant system overtemperature delta-T, reactor coolant system overpower delta-T, pressurizer pressure and level measurements, reactor coolant flow, reactor coolant pump speed, reactor coolant pump bearing water temperature, either of the PRHR heat exchanger discharge valves is not closed, ADS actuation, CMT actuation and steam generator water level measurements. Trips may also be initiated manually or by a safeguards actuation signal.

For additional information pertaining to the reactor trip logic, see Section 7.2.

High-quality components, conservative design and quality control, inspection, calibration, and tests are used to guard against common-mode failure. Qualification testing and analysis are performed on the safety-related systems to demonstrate functional operation at normal and post-accident conditions of temperature, humidity, pressure, and radiation for specified periods, as required. Typical protection system equipment is subjected to type tests under simulated seismic conditions, using conservatively large accelerations and applicable frequencies.

See Section 7.1 for additional information concerning the equipment design of the protection and safety monitoring system.

See Sections 3.10 and 3.11 for information pertaining to environmental and seismic qualification of the protection system equipment.

The AP1000 includes a nonsafety-related diverse actuation system. The diverse actuation system provides specific automatic functions including control rod insertion, turbine trip, passive residual heat removal heat exchanger actuation, core makeup tank actuation, isolation of critical containment lines, and passive containment cooling system actuation. This system is diverse and independent from the reactor protection system from sensors up to the actuation devices.

See Section 7.7 for additional information concerning the diverse actuation system.

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Criterion 23 – Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air) or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

AP1000 Compliance

The protection system is designed considering the most probable failure modes of the components under various perturbations of the environment and energy sources. Reactor trip channels are designed on the deenergize-to-trip principle so that a single event (that is, loss of power) that could affect many functions at the same time causes the channels to actuate to their tripped conditions.

Criterion 24 – Separation of Protection and Control Systems

The protection system shall be separated from the control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

AP1000 Compliance

The protection system is separate and distinct from the control systems. Control systems are, in some cases, dependent on the protection system for control signals that are derived from protection system measurements, where applicable. These signals are transferred to the control system by isolation devices classified as protection components.

The adequacy of the system isolation is verified by testing under conditions of postulated credible faults. Due to their inherent electrical isolation characteristics, fiber optic cables are exempt from electrical isolation qualification testing. The failure of a single control system component or channel, or the failure or removal from service of a single protection system component or channel common to the control and protection system, leaves intact a system that satisfies the requirements of the protection system. The removal of a protection division from service is allowed during testing of the division.

Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of the control rods.

AP1000 Compliance

The protection system is designed to limit reactivity transients so that the fuel design limits are not exceeded. Reactor shutdown by control rod insertion is independent of the normal control functions since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Thus, in the postulated accidental withdrawal of a control rod or control rod bank (assumed to be initiated by a control malfunction), neutron flux, temperature, pressure, level, and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

The AP1000 is designed to automatically terminate a boron dilution during manual or automatic operation at power, and also during startup and shutdown conditions. See Chapter 7 for a discussion of the signals used in the logic to terminate a boron dilution. Subsection 9.3.6.4.5 discusses the

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chemical and volume control system design features for addressing boron dilution. The Chapter 15 safety analyses demonstrate that fuel design limits are not exceeded.

Criterion 26 – Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that the acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

AP1000 Compliance

Two reactivity control systems are provided. These are rod cluster control assemblies and gray rod assemblies, and chemical shim (boric acid). The rod cluster control and gray rod assemblies are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. See Section 4.3 for additional information.

The shutdown and control rod banks are designed to provide reactivity margin to shut down the reactor during normal operating conditions and during anticipated operational occurrences, without exceeding specified fuel design limits. The safety analyses assume the most restrictive time in the core operating cycle and that the most reactive control rod cluster assembly is in the fully withdrawn position. See Chapter 15 for summaries of the analyses, assumptions, and results.

The safety-related passive systems provide the required boration to establish and maintain safe shutdown condition for the reactor core. See Section 6.3 for additional information.

Criterion 27 – Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

AP1000 Compliance

The plant is provided with the means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. Combined use of the control rod and the chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth control rod assembly is assumed to be stuck in the fully withdrawn position for this determination.

Criterion 28 – Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (l) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include

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consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

AP1000 Compliance

The maximum reactivity worth of the control rods and the maximum rates of reactivity increase employing control rods and boron removal are limited by design and operating procedures.

The appropriate reactivity addition rate for the withdrawal of control rods and the dilution rate of the boric acid in the reactor coolant system are specified in the precautions, limitations, and setpoint document and the control system setpoint study. Technical specifications explicitly specify control rod bank alignment and insertion limits in addition to shutdown margin reactivity requirements.

The control rod reactivity addition rate is determined by the allowable rod control system withdrawal speed, in conjunction with the control rod worth, which varies throughout the operating cycle. The capability to change boron concentration is determined by the various plant systems that provide makeup to the reactor coolant system. The reactivity insertion rates, rod withdrawal limits, and boron dilution limits are discussed in Chapter 4.

Core cooling capability following events such as rod ejection and steam line breaks is provided by keeping the reactor coolant pressure boundary stresses within faulted condition limits, as specified by applicable ASME codes. Structural deformations are also checked and limited to values that do not jeopardize the operation of needed safety-related features.

Criterion 29 – Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

AP1000 Compliance

The protection and reactivity control systems have an extremely high probability of performing their required safety-related functions in the event of anticipated operational occurrences. High-quality equipment, diversity, and redundancy, support this probability. Loss of power to the protection system results in a reactor trip. Defense in depth is designed into AP1000 to reduce challenges to the protection and reactivity control systems.

3.1.4 Fluid Systems

Criterion 30 – Quality of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

AP1000 Compliance

Reactor coolant pressure boundary components are designed, fabricated, inspected, and tested in conformance with the ASME Code, Section III. A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(z). The alternate classification is discussed in Section 5.2.

Leakage detection monitoring is accomplished using instrumentation and other components of several systems. See Subsection 5.2.5 for additional information. Reactor coolant pressure boundary leakage is classified as either identified or unidentified leakage.

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Auxiliary systems connected to the reactor coolant pressure boundary incorporate design and administrative provisions that limit leakage. Leakage is detected by increasing auxiliary system level, temperature, flow, or pressure, by lifting of relief valves, or by increasing values of monitored radiation in the auxiliary system.

Leakage from the reactor coolant pressure boundary and other components not otherwise identified inside the containment will condense and flow by gravity via the floor drains and other drains to the containment sump. Leakage is indicated by an increase in the sump level.

Reactor coolant system inventory monitoring provides an indication of system leakage. The reactor coolant system inventory balance is a quantitative inventory or mass balance calculation.

Leakage from the reactor coolant pressure boundary will result in an increase in the radioactivity levels inside containment. The containment atmosphere is monitored for airborne gaseous radioactivity and F18 particulate. From the concentration of F18 particulate and the power level, reactor coolant pressure boundary leakage can be estimated.

Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (l) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (l) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.

AP1000 Compliance

Control is maintained over material selection and fabrication for the reactor coolant pressure boundary components so that the boundary behaves in a nonbrittle manner. The portion of the chemical and volume control system that uses an alternate classification is not required to meet the requirements to prevent brittle failure. The reactor coolant pressure boundary materials exposed to the coolant are corrosion-resistant stainless steel or nickel-chromium-iron alloy. The nil-ductility transition reference temperature of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR 50, Appendix G (Reference 1). See Section 5.3 for additional information.

The following requirements are imposed in addition to those specified by the ASME Code, Section III.

- A 100 percent volumetric ultrasonic shear wave test of reactor vessel plate and a post-hydrotest ultrasonic map of welds in the pressure vessel are required. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the ASME Code, Section III is also required in order to preclude interpretation problems during in-service inspection.

- In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and l/2T compact tension specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM, E-185 (Reference 2).

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- Reactor vessel core region material chemistry (copper, phosphorous, and vanadium) is controlled to reduce sensitivity to embrittlement due to irradiation over the life of the plant.

The fabrication and quality control techniques used in the fabrication of the reactor coolant system are governed by ASME Code, Section III requirements.

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the ASME Code, Section III, Appendix G. The approach specifies that the allowable stress intensity factors for vessel-operating conditions do not exceed the reference stress intensity factor for the metal temperature. Operating specifications include conservative margins for predicted changes in the material reference temperatures due to irradiation.

Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (l) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel.

AP1000 Compliance

The design of the reactor coolant pressure boundary provides accessibility to the internal surfaces of the reactor vessel and most external zones of the vessel, including the nozzle-to-reactor coolant piping welds, the top and bottom heads, and external surfaces of the reactor coolant piping, except for the area of pipe within the primary shield concrete. The inspection capability complements the leakage detection systems in assessing the integrity of the pressure boundary components. The reactor coolant pressure boundary will be periodically inspected under the provisions of the ASME Code, Section XI. Section 5.1 provides the reactor coolant system primary loop drawings. The portion of the chemical and volume control system that uses an alternate classification is constructed to requirements that do not require in-service inspection.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates, forgings, weldments, and associated heat-treated zones is performed according to 10 CFR 50, Appendix H. Additionally, samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes conventional tensile and impact tests and fracture mechanics specimens. The observed shifts in the nil-ductility transition reference temperature of the core region materials with irradiation is used to confirm the allowable limits calculated for operational transients.

The design of the reactor coolant pressure boundary piping provides for accessibility of welds requiring in-service inspection under the provisions of the ASME Code, Section XI. Removable insulation is provided at welds requiring in-service inspection. See Section 5.3 and Subsection 5.2.4 for additional information.

Criterion 33 – Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished

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using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

AP1000 Compliance

Changes in the reactor coolant volume will be accommodated by the pressurizer level program for normal power changes, including the transition from hot standby to full-power operation and returning to hot standby. In addition, the pressurizer has sufficient volume to accommodate minor reactor coolant system leakage.

Safety-related passive reactor coolant system makeup is provided to accommodate small leaks when the normal makeup system is unavailable and to accommodate larger leaks resulting from loss of coolant accidents. Safety-related reactor coolant makeup and safety injection are provided by two core makeup tanks, two accumulators, and an in-containment refueling water storage tank. Long-term cooling is provided by containment gravity recirculation of reactor coolant within containment. See Section 6.3 for additional information. The safety-related reactor coolant makeup relies on the Class 1E and UPS system. Neither onsite or offsite ac power is required.

In addition, the nonsafety-related chemical and volume control system automatically provides inventory control to accommodate minor leakage from the reactor coolant system, expansion during heatup from cold shutdown, and contraction during cooldown. This inventory control is provided by letdown and makeup connections to the chemical and volume control system purification loop. Redundant pumps with connections to redundant nonsafety-related onsite ac power are provided when offsite power is not available and these pumps can be supplied from offsite power when onsite power is not available. See Section 5.2 for additional information.

Criterion 34 – Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

"Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

AP1000 Compliance

The AP1000 design satisfies the intent of GDC 34 by reducing the risk associated with loss of the decay heat removal function through a combination of safety-related passive systems, together with nonsafety-related active systems. Specific decay heat removal systems include the following:

- A safety-related passive residual heat removal heat exchanger that uses natural circulation flow and that does not require electrical power for operation
- Automatic, safety-related feed and bleed using the core makeup tanks, accumulators, and the in-containment refueling water storage tank for injection and the automatic depressurization system valves for reactor coolant system venting
- The nonsafety-related main feedwater system with motor-driven pumps supplied by the main generator or by offsite power

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- The nonsafety-related startup feedwater system with motor-driven pumps supplied by offsite or onsite power, including automatic sequencing on the nonsafety-related diesel generators

- The nonsafety-related normal residual heat removal system with motor-driven pumps supplied by offsite or onsite power, including nonsafety-related diesel generators, for use at low reactor coolant system pressures

A safety-related emergency feedwater system is not required for the AP1000 design. An active safety-related residual heat removal system is not required for the AP1000.

The AP1000 passive core cooling system, in conjunction with the passive containment cooling system, provides a reliable capability for removing decay heat from the reactor core and maintains sufficient water inventory to provide adequate core cooling for an extended period of time. The system does not depend upon pumped injection or recirculation, and actuates automatically, requiring no operator actions.

The containment arrangement addresses the Regulatory Guide 1.82 issues. Functional performance of the system addresses the guidelines of Regulatory Guide 1.139, except that cooldown rate is somewhat more limited when using the passive residual heat removal equipment. See Subsection 1.9.1 for additional information.

The passive core cooling system provides both gravity injection and gravity recirculation, automatically shifting injection modes when the proper containment flood-up conditions are achieved.

The AP1000 design provides a passive decay heat removal system that functions independent of nonsafety-related ac power supplies and can accommodate single active failures. (The Class 1E dc and UPS system supplies power to the safety-related monitoring and control instrumentation.) The passive core cooling system complies with General Design Criterion 34 by providing the capability to remove decay heat without relying on nonsafety-related ac power.

Criterion 35 – Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (l) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

AP1000 Compliance

The AP1000 design provides for safety-related passive reactor coolant makeup. Core makeup tanks accommodate small leaks when the normal makeup system is unavailable and provide safety injection for small-break loss of coolant accidents. Accumulators provide the high makeup flow required for a large loss of coolant accident and initiate injection when the reactor coolant system pressure is below the static accumulator pressure during a small-break loss of coolant accident.

The in-containment refueling water storage tank, and after containment flood-up, containment recirculation capability provide the long-term source of gravity injection to the core after the reactor coolant system is depressurized. The automatic depressurization system valves provide the vent path to transfer the core decay heat to the containment and then to the ultimate heat sink.

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The AP1000 design provides a passive core cooling system that functions independent of ac power supplies, assuming single active failures. The passive core cooling system does not need the nonsafety-related diesel-generators for electrical power to either actuate or operate the various system components. Therefore, the passive core cooling system complies with the intent of GDC 35 by providing the capability for core cooling without relying on nonsafety-related ac power sources.

Criterion 36 – Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

AP1000 Compliance

The AP1000 design includes a passive core cooling system that provides emergency core decay heat removal, emergency reactor coolant system makeup and boration, safety injection, and containment sump pH control. The system piping and components are designed to permit access for periodic inspection and testing of equipment, according to the ASME Code and technical specification requirements, to provide confidence in the integrity and capability of the system.

The core makeup tanks, accumulators, and passive residual heat removal heat exchanger have manways which permit access for inspection and required maintenance. The in-containment refueling water storage tank design provides access for both the tank itself and for the passive residual heat removal heat exchanger, spargers, and other components located inside the tank.

In addition, the system piping provides accessibility for inspection and maintenance to the extent practical. See Section 6.3 for additional information.

Criterion 37 – Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (l) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

AP1000 Compliance

The AP1000 passive core cooling system is designed to permit the periodic inspection and testing of the appropriate system components. The testing capabilities of the system including in-service testing and inspection to confirm the structural and leaktight integrity of various components, technical specification operability and performance of the active system components, and additional in-service testing to confirm the overall operability of the system.

The stage 1, 2, and 3 automatic depressurization system valves have provisions for shutdown in-service testing and at-power operability testing.

Planned shutdown testing includes operability testing of the component and system performance, including operation of applicable portions of the protection and safety monitoring system and the use of the appropriate power sources for the system.

The AP1000 design has significantly reduced the support systems required for system operation. In-service testing of the required support systems is also planned.

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Criterion 38 – Containment Heat Removal System

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss of coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

AP1000 Compliance

The AP1000 design uses passive systems for post-loss of coolant accident core and containment heat removal and for the prevention of overpressurization failure of the containment building. Heat is transferred from the containment atmosphere to the steel containment shell by natural convection and condensation. Heat removal from the exterior of the containment shell is enhanced by a directed-flow natural convection design and a passive, external cooling water distribution system.

The AP1000 passive containment cooling system is designed with sufficient capacity to prevent the containment from exceeding its design pressure with no operator action or outside assistance for a minimum of 3 days. After 3 days, limited operator action is required.

The AP1000 passive containment cooling system consists of a steel containment shell and associated water supplies, piping, valves, and air baffle. The passive containment cooling system is a passive system that uses gravity and natural circulation as driving forces. The design of the AP1000 passive containment cooling system does not require the use of any pumps, and it functions independent of nonsafety-related ac power sources for 3 days. Therefore, the passive containment cooling system can function during loss of offsite or onsite power. GDC 38 is satisfied by using appropriate redundancy and by the design of the passive containment cooling system and its reliance on natural forces.

Criterion 39 – Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles and piping, to assure the integrity and capability of the system.

AP1000 Compliance

The AP1000 design uses safety-related passive means for containment heat removal. The design of the system allows for inspection of piping, valves, the containment shell and air baffle, and other components to provide confidence in the integrity and capability of the system.

The periodic inspections specified in the ASME Code and technical specifications provide confidence that the capability of these heat removal systems is retained through plant life.

Criterion 40 – Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (l) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable

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portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

AP1000 Compliance

The AP1000 design includes a passive containment cooling system that provides containment heat removal to limit the peak containment pressure following design basis events. The system piping and components are designed to permit access for periodic inspection and testing of equipment, according to the ASME Code and technical specification requirements, to provide confidence in the integrity and capability of the system.

The passive containment cooling water storage tank design allows access for both the tank and for the various components located inside the tank.

In addition, the system piping provides accessibility for inspection and maintenance to the extent practical. See Section 6.2 for additional information.

Criterion 41 – Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided, as necessary, to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

AP1000 Compliance

Fission product control for the AP1000 plant is provided via natural removal processes within containment and by limiting containment leakage. The passive removal processes such as deposition and sedimentation are evaluated based on a physically-based source term with large scale core damage. See Section 6.5 for additional details. The containment and penetration design includes features specifically designed to minimize overall containment leakage. See Subsection 6.2.3 for additional details.

The generation of hydrogen in the containment under post-accident conditions has been evaluated, and the containment hydrogen control system has been designed such that the following criteria are satisfied:

- In compliance with Section 50.44 of 10 CFR 50, means are provided to measure and control post-loss of coolant accident hydrogen concentrations.
- The combustible concentrations of hydrogen do not accumulate in the areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.
- Internal passive autocatalytic recombiners are provided for hydrogen control following a design basis loss of coolant accident.

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- Hydrogen igniters are provided to limit local and global hydrogen concentrations to below 10 percent following a degraded core event with the reaction of 100 percent of the zircaloy cladding.

- The concentration of uniformly distributed hydrogen produced by the equivalent of a 75 percent active fuel-clad metal water reaction does not exceed 13 percent by volume during and following a degraded core event. (The AP1000 containment volume is large enough to provide passive protection for the hydrogen produced by 75 percent zircaloy cladding reaction following a severe accident.)

- The nonsafety-related ventilation system, normally used during refueling, is designed with the capability for a controlled purge of the containment atmosphere to assist in post-accident cleanup, but is not required for hydrogen control.

Criterion 42 – Inspection of Containment Atmosphere Cleanup System

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

AP1000 Compliance

The containment atmosphere cleanup systems are designed and located so that they can be inspected periodically, as appropriate.

Criterion 43 – Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

AP1000 Compliance

The appropriate portions of the containment atmosphere cleanup system are designed to permit periodic pressure and functionality testing.

As described in GDC 41, the containment atmosphere cleanup system has no safety-related post-accident cleanup functions. Dose mitigation is passively provided by the containment isolation and integrity, natural removal processes, and limited containment leakage. Periodic containment integrity is verified in accordance with 10 CFR 50 Appendix J testing as described in Subsection 6.2.3.

Criterion 44 – Cooling Water

A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

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AP1000 Compliance

The passive containment cooling system is the ultimate heat sink for the AP1000 and does not rely upon offsite or onsite ac power sources. Heat transfer by convection from the containment shell to the atmosphere meets the intent of GDC 44. Additional information is provided in the responses for GDC 34 and GDC 38.

Criterion 45 – Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

AP1000 Compliance

Refer to the discussion provided for GDC 39.

Criterion 46 – Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss of coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

AP1000 Compliance

Refer to the discussion provided for GDC 40.

3.1.5 Reactor Containment

Criterion 50 – Containment Design Basis

The reactor containment structure, including access opening, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

AP1000 Compliance

The design of the containment structure is based on the containment design basis accidents, which include the rupture of a reactor coolant pipe or the rupture of a main steam or feedwater line. The maximum pressure and temperature reached, a description of the calculational model, and input parameters for a containment design basis accident are presented in Section 6.2. The containment design provides margin to the design basis limits.

Criterion 51 – Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in

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a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

AP1000 Compliance

Principal load-carrying components of ferritic materials of the reactor containment boundary exposed to the external environment are selected so that they behave in a nonbrittle manner and so that the probability of fracture propagation is minimized. See Subsection 3.8.2 for additional information.

Criterion 52 – Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

AP1000 Compliance

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leakage rate tests according to the requirements of 10 CFR 50, Appendix J.

Criterion 53 – Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows.

AP1000 Compliance

Provisions exist for conducting individual leakage rate tests on containment penetrations. Penetrations are visually inspected and pressure-tested for leak tightness at periodic intervals. Other inspections are performed as required by 10 CFR 50, Appendix J.

Criterion 54 – Piping Systems Penetrating Containment

Piping systems penetrating the primary reactor containment shall be provided with leak detection, isolation and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

AP1000 Compliance

Piping systems penetrating the primary reactor containment are provided with containment isolation valves. Penetrations that close for containment isolation have redundant valving. Automatic isolation valves with air-, solenoid-, or motor-operators, which do not restrict normal plant operation, are periodically tested to verify operability.

The AP1000 containment isolation design satisfies the current NRC requirements including the post-TMI requirements, as discussed in Subsection 1.9.3. In general, this means that two barriers are provided, one inside containment and the other outside containment. Usually these barriers are valves, but in some cases they are closed piping systems not connected to the reactor coolant system or to the containment atmosphere.

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The AP1000 design incorporates a reduction in the number of existing penetrations. Most penetrations are normally closed. Those few that are normally open and are required to close use remotely operated valves for isolation that close automatically. See Subsection 6.2.3 for additional information.

Nonessential systems that may be normally open, such as the mini-purge system, are provided with automatic containment isolation valves that close automatically on a containment isolation signal. The containment isolation signal is actuated by the protection and safety monitoring system. See Section 7.3 for additional information.

Piping and electrical containment penetrations are equipped with test connections and test vents or have other provisions to allow periodic leak rate testing so that leakage is within the acceptable limits established in technical specifications consistent with 10 CFR 50, Appendix J.

Criterion 55 – Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided, as necessary, to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, and use characteristics, and physical characteristics of the site environs.

AP1000 Compliance

Lines that penetrate containment that are connected to the reactor coolant pressure boundary are provided with containment isolation valves in accordance with one of the acceptable arrangements as described in GDC 55. Additional information is found in Subsection 6.2.3.

Criterion 56 – Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates the primary reactor containment shall be provided with containment isolation valves as follows, unless it can be

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demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside the containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside the containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside the containment.

Isolation valves outside the containment shall be located as close to the containment as practical and, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

AP1000 Compliance

Lines connecting directly with the containment atmosphere and penetrating the reactor containment are normally provided with two isolation valves in series, one inside and one outside the containment, in accordance with one of the acceptable arrangements as described in GDC 56. Isolation of instrument lines for containment pressure measurement is demonstrated on a different basis and does not require isolation valves. Additional information is found in Subsection 6.2.3.

Criterion 57 – Closed System Isolation Valves

Each line that penetrates the primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, locked closed, or capable of remote manual operation. This valve shall be outside the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

AP1000 Compliance

Lines that penetrate the containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere are considered closed systems within the containment and are equipped with at least one containment isolation valve of one of the following types:

- An automatic isolation valve (a simple check valve is not used as this automatic valve)

- A locked-closed valve

This valve is located outside the containment and as close to the containment wall as practical.

3.1.6 Fuel and Reactivity Control

Criterion 60 – Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during

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normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for the retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

AP1000 Compliance

Means are provided to control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

The radioactive waste management systems are designed to minimize the potential for an inadvertent release of radioactivity from the facility and to provide confidence that the discharge of radioactive wastes is maintained below regulatory limits of 10 CFR 50, Appendix I, during normal operation. The gaseous radwaste and liquid radwaste processing systems include continuous radiation monitoring of their discharge paths. High radiation automatically closes a discharge isolation valve. The liquid radwaste system also has provisions to prevent inadvertent siphoning of its monitor tank contents which could cause an uncontrolled discharge. The radioactive waste management systems, the design bases, and the estimated amounts of radioactive effluent releases to the environment are described in Chapter 11.

Criterion 61 – Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

AP1000 Compliance

The spent fuel pool cooling system, and the fuel handling and refueling system are designed to provide cooling and shielding for the fuel assemblies stored in the spent fuel pool and to provide purification of the water in the pit. The system design provides adequate safety under normal and postulated accident conditions.

The spent fuel pool cooling system normal system operation is described in Subsection 9.1.3. Sampling of the spent fuel pool water for gross activity, tritium, and particulate matter is conducted periodically. The concentration of tritium in the spent fuel pool water is maintained at less than 0.5 microcuries per gram to provide confidence that the airborne concentration of tritium in the fuel handling area is within the limits specified in 10 CFR 20, Appendix B. See Subsection 12.2.2 for additional information.

The spent fuel pool is designed so that a water level is maintained above the spent fuel assemblies for at least 72 hours following a loss of the spent fuel pool cooling system, without ac power. See Subsection 9.1.2 for additional information.

The spent fuel pool cooling system maintains the water in the in-containment refueling water storage tank consistent with activity requirements of the water in the refueling cavity during a refueling. Two spent fuel pool cooling filters are provided, one downstream of each demineralizer in the purification branch line of each mechanical train. The filters are sized to collect particulates and suspended solids passed by the demineralizer.

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The AP1000 spent fuel pool cooling system is not required to operate to mitigate design basis events. In the event the spent fuel pool cooling system is unavailable, the spent fuel pool cooling is provided by the heat capacity of the water in the pool and in the passive sources of makeup water.

Normal HVAC to the spent fuel pool area is provided by a subsystem of the radiologically controlled area ventilation system described in Subsection 9.4.3. No credit is taken for this system in evaluation of fuel handling accidents discussed in Subsection 15.7.4.

Connections to the spent fuel pool are provided at an elevation that prevents inadvertent draining of the water in the pool to an unacceptable level.

The design of spent fuel storage pool and the spent fuel pool cooling system satisfies GDC 61. See Subsection 9.1.3 for additional information.

Criterion 62 – Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

AP1000 Compliance

The restraints, interlocks, and physical arrangement provided for the safe handling and storage of new and spent fuel are discussed in Section 9.1. The spent fuel assemblies are stored in the spent fuel pool until fission product activity is low enough to permit shipment.

Criterion 63 – Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in the fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in the loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

AP1000 Compliance

Instrumentation is provided to monitor spent fuel storage pool temperature and water level. Indication and alarms are provided in the main control room. Area radiation monitoring is provided in the fuel storage area for personnel protection and general surveillance. The area monitor alarms locally and in the main control room.

If radiation levels in the ventilation effluent reach a predetermined point, an alarm is actuated in the main control room, and the ventilation discharge path is automatically transferred through filter absorber units that provide filtration before discharge from the plant vent.

Criterion 64 – Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss of coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

AP1000 Compliance

The containment atmosphere is monitored during normal and transient operations by the containment gaseous radiation monitors. Under accident conditions, samples of the containment atmosphere taken via the sampling system provide data on airborne radioactive concentrations within the containment.

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No reactor coolant fluids are required to be recirculated outside of containment following an accident. Radioactivity levels contained in the facility effluent and discharge paths and in the plant environs are monitored during normal and accident conditions by the plant radiation monitoring systems. High radiation in a discharge path causes automatic closure of the discharge isolation valve.

Area radiation monitors (ARMs) are provided to supplement the personnel and area radiation survey provisions of the AP1000 health physics program described in Section 12.5 and to comply with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, and Regulatory Guides 1.97, 8.2, 8.8, and 8.12. In addition to the installed detectors, periodic plant environmental surveillance is established.

Measurement capability and reporting of effluents are based on the guidelines of Regulatory Guides 1.4 and 1.21, as discussed in Subsection 1.9.1. Additional information is contained in Chapters 11 and 12.

3.1.7 Combined License Information

This section contained no requirement for additional information.

3.1.8 References

1. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

2. American Society of Testing Materials E-185, Standard Recommended Practice for Surveillance Test for Nuclear Reactor Vessels, and the requirements for 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

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3.2 Classification of Structures, Components, and Systems

Structures, systems, and components in the AP1000 are classified according to nuclear safety classification, quality groups, seismic category, and codes and standards. This section provides the methodology used for safety-related and seismic classification of AP1000 structures, systems, and components. The seismic classification is described in Subsection 3.2.1. Subsection 3.2.2 describes the classification including nuclear safety-related classification and the corresponding codes and standards. Additionally, Subsection 3.2.2 describes nonsafety-related equipment classifications.

3.2.1 Seismic Classification

General Design Criterion 2 requires that nuclear power plant “Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.” 10 CFR 100, Appendix A sets forth the criteria to which the plant design bases demonstrate the capability to function during and after vibratory ground motion associated with the safe shutdown earthquake conditions.

The seismic classification methodology used in AP1000 complies with the preceding criteria, as well as with recommendations stated within Regulatory Guide 1.29. Conformance with the recommendations of Regulatory Guide 1.29 is discussed in Subsection 1.9.1. The methodology classifies structures, systems, and components into three categories: seismic Category I (C-I), seismic Category II (C-II) and non-seismic (NS).

Seismic Category I applies to both functionality and integrity, and seismic Category II applies only to integrity. Non-seismic items located in the proximity of safety-related items, the failure of which during a safe shutdown earthquake could result in loss of function of safety-related items, are designated as seismic Category II.

There are no safety-related structures, systems, or components outside the scope of the DCD. The nonsafety-related structures, systems, and components outside the scope of the DCD are classified as non-seismic (NS).

3.2.1.1 Definitions

3.2.1.1.1 Seismic Category I (C-I)

Seismic Category I applies to, in general, safety-related structures, systems, and components. Seismic Category I also applies to those structures, systems, and components required to support or protect safety-related structures, systems, and components. The exceptions to this general rule are a limited number of structures, such as those required for tornado missile protection, which do not have a safety-related function to perform during or following a seismic event. (See Subsection 3.2.2.5.)

Safety-related items are those necessary to provide for the following:

- The integrity of the reactor coolant pressure boundary
- The capability to shut down the reactor and maintain it in a safe shutdown condition
- Capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34.

Seismic Category I structures, systems, and components are designed to withstand the appropriate seismic loads, as discussed in Section 3.7, and other applicable loads without loss of function.

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Seismic Category I structures are protected from interaction with adjacent non-seismic structures as described in Subsection 3.7.2.8.

Systems and components identified as safety-related systems and components in Table 3.2-3, and electrical and instrumentation components identified in Table 3.11-1, are the systems and components necessary for continued operation that must remain functional without undue risk of the health and safety of the public during and following an operating basis earthquake. Systems and components identified as Equipment Class A, B, and C in Table 3.2-3, and electrical and instrumentation components identified in Table 3.11-1, are the systems and components that per the criteria of 10 CFR Part 50, Appendix S, must be demonstrated, prior to resuming operations, to have no functional damage following a seismic ground motion exceeding the operating basis earthquake ground motion. See Section 3.7 for information on the operating basis earthquake.

Seismic Category I structures, systems, and components meet the quality assurance requirements of 10 CFR 50, Appendix B. The criteria used for the design of seismic Category I structures, systems, and components are discussed in Section 3.7.

3.2.1.1.2 Seismic Category II (C-II)

Seismic Category II applies to plant structures, systems, and components which perform no safety-related function, and the continued function of which is not required. Seismic Category II applies to structures, systems, and components designed to prevent their collapse under the safe shutdown earthquake. Structures, systems and components are classified as seismic Category II to preclude their structural failure during a safe shutdown earthquake or interaction with seismic Category I items which could degrade the functioning of a safety-related structure, system, or component to an unacceptable level, or could result in incapacitating injury to occupants of the main control room. The turbine building first bay building structure, including Wall 11.2, is a seismic Category II structure as identified in Table 3.2-2. The seismic Category II systems, structures, and components which provide tornado missile protection for openings and penetrations are described in Subsection 3.5.2. Wall 11.2 protects Wall 11 from the dynamic effects of pipe failure events in the nonseismic portion of the turbine building.

Seismic Category II structures, systems, and components are designed so that the safe shutdown earthquake does not cause unacceptable structural failure of or interaction with seismic Category I items. Seismic Category II fluid systems require an appropriate level of pressure boundary integrity if located near sensitive equipment.

The criteria used for the design of seismic Category II structures, systems, and components are discussed in Section 3.7. As identified in Subsection 3.7.2, seismic Category II building structures are designed for the safe shutdown earthquake using the same methods and design stress limits as are used for seismic Category I structures. Seismic Category II building structures are also designed to withstand the design basis tornado loads, including missiles, in accordance with the loading combinations identified in Table 3.8.4-2.

Pertinent portions of 10 CFR 50, Appendix B apply to the analysis and design of seismic Category II structures, systems, and components. The quality assurance requirements for the analysis and design of seismic Category II structures, systems, and components are performed in accordance with the Westinghouse AP1000 quality plan as described in Section 17.3 and are sufficient to provide that these components will meet the requirement to not cause unacceptable structural failure of or interaction with seismic Category I items. See Section 17.7 for the Combined License applicant quality assurance program requirement. These quality requirements are applicable to the seismic Category II turbine building first bay building structure.

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3.2.1.1.3 Non-Seismic

Non-seismic (NS) structures, systems, and components are those that are not classified seismic Category I or Category II.

The criteria used for the design of non-seismic structures, components and systems are discussed in Section 3.7.

The non-seismic lines and associated equipment are routed, to the extent practicable, outside of safety-related buildings and rooms to avoid adverse system interactions. In cases where these lines are routed in safety-related areas, the non-seismic item is evaluated for the safe shutdown earthquake and is upgraded to seismic Category II if a credible failure could cause an unacceptable interaction.

Although the seismic category for an item located in the proximity of safety-related structures, systems, and components may be upgraded to seismic Category II, its pre-assigned equipment class remains unchanged.

3.2.1.2 Classifications

Table 3.2-1 illustrates the general relationship between safety-related equipment classes and seismic categories. In most cases, except as noted in Subsection 3.2.2.5, safety-related items are also seismic Category I items. When portions of systems are identified as seismic Category I, the boundaries of seismic Category I portions of the system are shown on the piping and instrumentation diagram (P&ID) of that system. See Subsection 1.7.2 for a list of the piping and instrumentation diagrams.

3.2.1.3 Classification of Building Structures

Building structures are assigned a seismic category as indicated in Table 3.2-2. Codes and standards used in the design and construction of building structures are given in Section 3.8. The building structures are not assigned a safety classification in Subsection 3.2.2 with the exception of the containment vessel.

3.2.2 AP1000 Classification System

The assignment of safety-related classification and use of codes and standards conforms to the requirements of 10 CFR 50.55a for the development of a Quality Group classification and the use of codes and standards. The description of the equipment classification which follows identifies the classifications requiring the full 10 CFR 50, Appendix B quality assurance program as described in Chapter 17 and the Quality Group associated with each classification.

The classification system provides a means of identifying the extent to which structures, systems, and components are related to safety-related and seismic requirements. The classification system provides an easily recognizable means of identifying the extent to which structures, systems, and components are related to ANS nuclear safety classification, NRC quality groups, ASME Code, Section III classification, seismic category and other applicable industry standards, as shown in Table 3.2-3.

There are no safety-related structures, systems, or components outside the scope of the DCD.

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3.2.2.1 Classification Definitions

The definitions used in the classification of structures, systems and components are provided in the following. Unless otherwise noted these definitions apply throughout the Design Control Document. These definitions are consistent with the ANS Definitions for Light Water Reactor Standards (ANS-58.14-1993).

Safety-related is a classification applied to items relied upon to remain functional during or following a design basis event to provide a safety-related function. Safety-related also applies to documentation and services affecting a safety-related item.

Safety-related function is a function that is relied upon during or following a design basis event to provide for the following:

- The integrity of the reactor coolant pressure boundary
- The capability to shut down the reactor and maintain it in a safe shutdown condition
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34.

Design basis event is an event that is a condition of normal operation (including anticipated operational occurrences), a design basis accident, an external event, or natural phenomena for which the plant must be designed so that the safety-related functions are achievable.

Design basis accidents and transients are those design basis events that are accidents and transients and are postulated in the safety analyses. The design basis accidents and transients are used in the design of the plant to establish acceptable performance requirements for structures, systems, and components.

3.2.2.2 Application of Classification

The AP1000 requires adaptation of safety classification documents and standards because of the way that the AP1000 accomplishes safety-related functions.

In addition to 10 CFR 50.55a, the AP1000 classification has been developed considering requirements and guidelines in the following:

- ANSI N18.2a (Reference 1) – safety classification
- ANS 51.1 (Reference 2) – safety classification
- Regulatory Guide 1.26 – Quality Groups
- Regulatory Guide 1.97 – instrumentation requirements
- 10 CFR 21.

Conformance with the guidelines of Regulatory Guides 1.26 and 1.97 is discussed in Subsection 1.9.1.

The general guidelines for safety classification in the ANSI and ANS standards are useful in the development of the AP1000 classification. The specific classifications for various structures, systems, and components included in Regulatory Guide 1.26 and ANSI N18.2a and ANS 51.1 are based on a nuclear power plant with active safety systems and are not necessarily appropriate for the passive safety systems of the AP1000.

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For the purposes of equipment classification, structures, systems, and components are classified as Class A, B, C, D, E, F, G, L, P, R, or W. For mechanical equipment Classes A, B, and C are equivalent to ANS Safety Class 1, 2, and 3. For electrical equipment Class C is equivalent to Class 1E. Structures, systems, and components classified equipment class A, B, or C or seismic Category I are basic components as defined in 10 CFR Part 21.

Equipment Class D is a nonsafety-related class. Classes E, F, G, L, P, R, and W are nonsafety-related classes associated with different industry codes and standards.

Components are classified down to the replacement part level according to the definitions and criteria of the classification system. A single item or portion thereof, which provides two or more functions of different classes, is classified according to the most stringent function. Different portions of the same structure, system, or component may perform different functions and be assigned to different equipment classes if the structure, system, or component contains a suitable interface boundary.

The definitions and criteria for the AP1000 equipment classes follow.

3.2.2.3 Equipment Class A

Class A is a safety-related class equivalent to ANS Safety Class 1. It applies to the reactor coolant system pressure boundary, including the required isolation valves and mechanical supports. This class has the highest integrity, and the lowest probability of leakage.

10 CFR 21 applies to Class A structures, systems, and components. Class A structures, systems, and components are seismic Category I and use codes and standards consistent with the guidelines for NRC Quality Group A. 10 CFR 50, Appendix B applies. ASME Code, Section III, Class 1 applies to pressure retaining components.

3.2.2.4 Equipment Class B

Class B is a safety-related class equivalent to ANS Safety Class 2. It limits the leakage of radioactive material from the containment following a design basis accident. This class is designed to accomplish the following:

- It provides fission product barrier or primary containment radioactive material holdup or isolation.

- It provides the containment boundary including penetrations and isolation valves. This also includes piping that functions as the containment boundary. For example, the steam and feedwater system inside containment and the secondary shell of the steam generator are Class B by this criterion.

- It circulates a non-containment/non-reactor coolant fluid to provide a post-accident safety-related function into and out of the containment. These lines have a Class B pressure boundary inside the containment. The outside containment lines in this circulation loop can be Class C or a nonsafety-related class if suitable containment isolation valves are provided.

- It introduces emergency negative reactivity to make the reactor subcritical (for example, control rods).

- This class also applies to structures, systems, and components where leakage could cause a loss of adequate core cooling. In isolating leaks, credit can be taken for automatic safety-related isolation and for appropriate operator action. As a minimum, operator action needs redundant safety-related indication and alarm followed by 30 minutes for operator action.

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10 CFR 21 applies to Class B structures, systems, and components. Class B structures, systems, and components are seismic Category I and use codes and standards consistent with the guidelines for NRC Quality Group B. 10 CFR 50, Appendix B applies. ASME Code, Section III, Class 2 or Class MC applies to pressure retaining components. ASME Code, Section III, Subsection NE applies to the containment vessel and guard pipes.

3.2.2.5 Equipment Class C

Class C is a safety-related class equivalent to ANS Safety Class 3. It applies to other safety-related functions required to mitigate design basis accidents and other design basis events. Minor leakage will not prevent Class C structures, systems, and components from meeting the safety-related function, either from the regard of radiation dose or system functioning.

This class also applies to equipment that, upon rupturing, would cause dose limits for unrestricted areas, as specified in 10 CFR 20, to be exceeded or would cause a loss of core cooling.

10 CFR 21 applies to Class C structures, systems, and components. Class C structures, systems, and components use codes and standards consistent with the guidelines for NRC Quality Group C. Class C structures, systems, and components are seismic Category I except those noted below which are not required to provide a safety-related function following a seismic event. 10 CFR 50, Appendix B applies. ASME Code, Section III, Class 3 applies to pressure retaining components with some exceptions as identified below. In addition to these requirements, for systems that provide emergency core cooling functions, full radiography in accordance with the requirements of ASME Code, Section III, ND-5222 of a random sample of welds will be conducted on the piping butt welds during construction. For Class C air and gas storage tanks fabricated without welding, ASME Code, Section VIII, Appendix 22 may be used in lieu of Section III, Class 3. For Class C fuel handling system weir gate pressure boundary components containing only compressed air, Manufacturer's Standard may be used in lieu of Section III, Class 3. 10 CFR 50, Appendix B requirements and 10 CFR 21 apply to the manufacture of safety-related air and gas storage tanks. For core support structures ASME Code, Section III, Subsection NG applies. For electrical systems, appropriate IEEE standards, including IEEE standard 323-74 (Reference 3) and IEEE standard 344-87 (Reference 4), apply.

Class C applies to structures, systems, and components not included in Class A or Class B that are designed and relied upon to accomplish one or more of the following safety-related functions:

- Provide safety injection or maintain sufficient reactor coolant inventory to allow for core cooling
- Provide core cooling
- Provide containment cooling
- Provide for removal of radiation from the containment atmosphere as necessary to meet the offsite dose limits
- Limit the buildup of radioactive material in the atmosphere of rooms and areas outside containment as necessary to meet the offsite dose limits
- Introduce negative reactivity control measures to achieve or maintain safe shutdown conditions (for example, boron addition)
- Maintain geometry of structures inside the reactor vessel so that the control rods can be inserted (when required) and the fuel remains in a coolable geometry

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- Provide load-bearing structures and supports for Class A, B, and C structures, systems, and components. This applies to structures and supports that are not part of the pressure boundary.

- Provide structures to protect Class A, B, and C structures, systems, and components from events such as internal/external missiles, seismic, flooding, and the dynamic and environmental effects of pipe failures. Structures protecting equipment from nonseismic events are not required to be seismic Category I. Exceptions to this equipment Class C criterion are the equipment Class D barriers and seismic Category II structures described in Subsection 3.5.2 that provide tornado missile protection. Turbine building first bay building structure Wall 11.2 provides protection of Wall 11 from HELB loads resulting from postulated ruptures in main steam and main feedwater piping north of the turbine building first bay.

- Provide permanent radiation shielding to allow operator access to the main control room and to limit the exposure to Class A, B and C structures, systems, and components

- Provide safety support functions to Class A, B and C structures, systems, and components, such as, heat removal, room cooling, and electrical power

- Provide instrumentation and controls for automatic or manual actuation of Class A, B, and C structures, systems, and components necessary to perform the safety-related functions of the Class A, B, or C structure, system or component. This includes the processing of signals and interlock functions required for proper safety performance of these structures, systems, and components.

- Maintain spent fuel integrity, the failure of which could result in fuel damage such that significant quantities of radioactive material could be released from the fuel and results in offsite doses greater than normal limits (for example, spent fuel pool, fuel transfer tube isolation valve)

- Maintain spent fuel sub-critical

- Monitor variables to indicate status of Class A, B or C structures, systems, and components required for post-accident mitigation

- Provide for functions defined in Class B where structures, systems, and components, or portions thereof are not within the scope of the ASME Code, Section III, Class 2.

- Provide provisions for connecting temporary equipment to extend the use of safety related systems. See Subsection 1.9.5 for a discussion of actions required for an extended loss of onsite and offsite ac power sources.

The components and portions of systems that provide emergency core cooling functions and are required to have radiography of a random sample of welds during construction include the following:

- Accumulators

- Injection piping from the accumulators to the reactor coolant system isolation check valves in the direct vessel injection line

- Piping from the in-containment refueling water storage tank (IRWST) and recirculation screens to the reactor coolant system isolation check valves in the direct vessel injection line

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- Piping from the Stage 1, 2, and 3 automatic depressurization system valves to the IRWST including the spargers.

The IRWST is formed from portions of structural modules that are elements of the containment internal structures. The inspection requirements for the welds in these structural modules are provided in Subsection 3.8.3.6.2.

3.2.2.6 Equipment Class D

Class D is nonsafety-related with some additional requirements on procurement, inspection or monitoring.

For Class D structures, systems, and components containing radioactivity, it is demonstrated by conservative analysis that the potential for failure due to a design basis event does not result in exceeding the normal offsite doses per 10 CFR 20. This criterion is in conformance with the definition of Class D in Regulatory Guide 1.26.

A structure, system or component is classified as Class D when it directly acts to prevent unnecessary actuation of the passive safety systems. Structures, systems and components which support those which directly act to prevent the actuation of passive safety systems are also Class D. The inclusion of these nonsafety-related structures, systems, and components in Class D recognizes that these systems provide an important first level of defense that helps to reduce the calculated probabilistic risk assessment core melt frequency. These structures, systems, and components are normally used to support plant cooldown and depressurization and to maintain shutdown conditions during maintenance and refueling outages.

For Class D structures, systems, and components considered to be risk significant as defined in the reliability assurance program (see Section 16.2). Provisions are made to check for operability, including appropriate testing and inspection, and to repair out-of-service structures, systems, and components. These provisions are documented and administered in the plant reliability assurance plan and operating and maintenance procedures.

A portion of chemical and volume control system is defined as the reactor coolant pressure boundary and is Class D. This portion of the chemical and volume control system is seismically analyzed. See Subsection 5.2.1.1 for the seismic analysis requirements.

Some Class D structures, systems, and components are assumed to function in a severe containment environment. The design requirements for these components include operation in such an environment. An evaluation is done to confirm that the structure, system, or component can be expected to function in such an environment.

Standard industrial quality assurance standards are applied to Class D structures, systems, and components to provide appropriate integrity and function although 10 CFR 50, Appendix B and 10 CFR 21 do not apply. 10 CFR 50, Appendix B and 10 CFR 21 do apply to Class D structures, systems, and components that are seismic Category I. Pertinent portions of 10 CFR 50, Appendix B are applied to seismic Category II applications as described in Subsection 3.2.1.1.2. These industrial quality assurance standards are consistent with the guidelines for NRC Quality Group D. The industry standards used for Class D structures, systems and components are widely used industry standards. Typical industrial standards used for Class D systems and components are provided as follows:

- Pressure vessels – ASME Code, Section VIII

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- Piping – ASME B31.1. Power Piping, (Reference 5). NDE of welds during fabrication and installation may use methods and acceptance criteria of ASME B31.1, 2007 Edition. Post Weld Heat Treatment of welds may use methods and acceptance criteria of ASME B31.1, 2014 Edition.
- Pumps – API 610 (Reference 6), or Hydraulic Institute Standards
- Valves – ANSI B16.34 (Reference 8)
- Atmospheric storage tanks – API-650 (Reference 9), AWWA D 100 (Reference 10), or ANSI B96.1 (Reference 11)
- 0 - 15 psig Storage Tanks – API-620 (Reference 12)
- AC motor and generators – NEMA MG1 (Reference 13)
- Circuit breakers, switchgear, relays, substations and fuses – IEEE C37 (Reference 14).

The NS buildings (except for the NS portions of the turbine and annex buildings outlined in Table 3.2-2) containing Class D structures, systems, and components, as well as the anchorage of the structures, systems, and components to the building, are designed to the seismic requirements of the Uniform Building Code (Reference 15). The NS portions of the turbine building and annex building are designed to the requirements of the International Building Code, IBC-06 (Reference 19). The systems and components are not designed for seismic loads. However, when Class D structures, systems, and components are located near a Class A, B, or C structure, system, or component, the requirements for seismic Category II may apply. The tornado missile barriers in the turbine building first bay building structure identified in Table 3.5-1 are seismic Category II, equipment Class D.

For Class D structures, systems, and components required to be monitored for maintenance effectiveness by 10 CFR 50.65, the availability parameters and criteria are included in the maintenance monitoring plan for evaluating the effectiveness of the maintenance program.

As examples, Class D applies to structures, systems, and components not included in Class A, B or C that provide the following functions:

- Provide core or containment cooling which prevents challenges to the passive core cooling system and the passive containment cooling system
- Process, extract, encase, store or reuse radioactive fluid or waste
- Verify that plant operating conditions are within technical specification limits
- Provide permanent shielding for post accident access to Class A, B or C structures, systems, and components or of offsite personnel
- Handle spent fuel, the failure of which could result in fuel damage such that limited quantities of radioactive material could be released from the fuel (for example, fuel handling machine, spent fuel handling tool, new and spent fuel racks)
- Protect Class B or C structures, systems, and components necessary to attain or maintain safe shutdown following fire

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- Indicate the status of protection system bypasses that are not automatically removed as a part of the protection system operation
- Aid in determining the cause or consequences of an event for post-accident investigation
- Prevent interaction that could result in preventing Class A, B or C structures, systems, and components from performing required safety-related functions
- Limit the buildup of hydrogen in the containment atmosphere to acceptable values

3.2.2.7 Other Equipment Classes

Equipment classes E, F, G, L, P, R, and W are nonsafety-related. They apply to structures, systems, and components not covered in the above classes. They have no safety-related function to perform. They do not contain sufficient radioactive material that a release could exceed applicable limits.

Structures, systems, and components that do not normally contain radioactive fluids, gases, or solids but have the potential to become radioactively contaminated are classified as one of these nonsafety-related classes if all of the following criteria are satisfied:

- The system is only potentially radioactive and does not normally contain radioactive material, and
- The system has shown in plant operations that the operation with the system containing radioactive material meets or can meet unrestricted area release limits, and
- An evaluation of the system confirms that the system contains features and components that keep the consequences of a system failure as low as reasonably achievable, and
- The system has no other regulatory guidance requiring its inclusion in Classes A, B, C or D.

This review of the system functions, features, and components includes the following as a minimum:

- Features and components that control and limit the radioactive contamination in the system
- Features that facilitate an expeditious cleanup should the system become contaminated
- Features and components that limit and control the radiological consequences of a potential system failure
- Monitor radioactive effluent to confirm that release rates or total releases are within limits established for normal operations and transient operation
- The means by which the system prevents propagation to an event of greater consequence.

There are no special quality assurance requirements for Class E, F, G, L, P, R, and W structures, systems, and components. Unless specifically specified, 10 CFR Part 21 and Part 50, Appendix B do not apply. The systems and components are normally not designed for seismic loading. However, there may be special cases where some seismic design is required. See Subsection 3.2.1 for more details.

Procurement, design, fabrication, and testing of structures, systems, and components is performed using normal industrial quality standards at the discretion of the designer. The following provides examples of industry standards which may be used for these classes:

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Class E – This class is used for nonsafety-related structures, systems, and components constructed to normal industrial quality standards which are not noted in the following classes. Class Y is a subset of Class E and is not utilized in the licensing basis.

Class F and G – These classes are used for Fire Protection Systems. They comply with National Fire Protection Association Codes which invoke ANSI B31.1 (Reference 5), AWWA (American Water Works Association), API (American Petroleum Institute), Underwriters Laboratories (UL), and other codes, depending on service. See Subsection 9.5.1 for quality assurance requirements for fire protection structures, systems, and components. Portions of fire protection systems that protect safety-related SSCs are designated as AP1000 equipment **Class F**, which meets the requirements of ASME B31.1 and requires seismic analysis. NDE of welds during fabrication and installation may use methods and acceptance criteria of ASME B31.1, 2007 Edition. Post Weld Heat Treatment of welds may use methods and acceptance criteria of ASME B31.1, 2014 Edition.

Class L – This class is used in heating, ventilation and air-conditioning systems. It complies with SMACNA - 1995 (Reference 16). Components may also be procured to AMCA, ASHRAE, and other SMACNA standards, as described in Section 9.4.

Class P – This class is used for plumbing equipment. It complies with the National Plumbing Code (Reference 17).

Class R – This class is for air cleaning units and components that may be required to contain, clean, or exclude radioactively contaminated air. It complies with ASME N509 (Reference 18) and ASME AG-1 (Reference 20) as noted in Table 3.2-3. When used with 10 CFR Part 50 Appendix B quality assurance and 10 CFR Part 21, it is equivalent to Class C. If Class R components are nonsafety-related, the quality assurance requirements invoked in ASME N509 and ASME AG-1 do not apply.

Class W – This class complies with American Water Works Association guidelines with no specific quality assurance requirements.

3.2.2.8 Instrumentation and Control Line Interface Criteria

Class C instrumentation, as defined in Subsection 3.2.2.5 have a safety-related equipment class pressure boundary including the sensing line, valves and instrument sensor. The pressure boundary is the same safety-related equipment class as the systems or components it is connected to. Sensing lines connected to the reactor coolant system pressure boundary are Class B if a suitable flow restrictor is provided.

Nonsafety-related instrumentation that monitors safety-related fluid systems is Class D, as defined in Subsection 3.2.2.6. The instrument sensing line is safety-related, seismic Category I from the connected fluid system to the instrument manifold. The instrument, manifold, and impulse line (interconnecting tubing between the manifold and instrument) are Class D, seismic Category II, as defined in Subsection 3.2.1.1.2. The Class D quality requirements include a pressure test at 1.5 times design pressure and a requirement for certified material test reports.

The parts of the sensor, outside the pressure boundary, are designated Class C (1E) if they provide a safety-related function per Subsection 3.2.2.1. They are Class D if the instrument supports Class D functions per Subsection 3.2.2.6. Otherwise the parts are Class E.

3.2.2.9 Electrical Classifications

Safety-related electrical equipment is equipment Class C, as outlined in Subsection 3.2.2.5, and is constructed to IEEE standards for Class 1E. The nonsafety-related electrical equipment and

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instrumentation is constructed to standards including non-Class 1E IEEE standards and National Electrical Manufacturers Association (NEMA) standards. Safety-related electrical equipment and instrumentation is identified in Section 3.11.

3.2.3 Inspection Requirements

Safety-related structures, systems, and components built to the requirements of the ASME Code, Section III, are required by 10 CFR 50.55a to have in-service inspections. The requirements of the in-service inspection program for ASME Code, Section III structures, systems, and components are found in Section XI of the ASME Code.

The following ASME standards apply to safety-related structures, systems, and components:

- Pumps (Class A, B, C) – ASME OM Code, Subsection ISTB
- Valves (Class A, B, C) – ASME OM Code, Subsection ISTC
- Equipment supports (Class A, B, C) – ASME Code, Section XI, Subsection IWF
- Metal containments and vessels – ASME Code, Section XI, Subsection IWE
- Other Class A components such as pipes and tanks – ASME Code, Section XI, Subsection IWB
- Other Class B components such as pipes and tanks – ASME Code, Section XI, Subsection IWC
- Other Class C components such as pipes and tanks – ASME Code, Section XI, Subsection IWD.

The inspection requirements, if applicable, for Class D structures, systems, and components are established by the designer for each structure, system, and component. These inspection requirements are developed so that the reliability of the structures, systems, and components is not degraded. The inspection requirements are included in the administratively controlled inspection or maintenance plans.

3.2.4 Application of AP1000 Safety-Related Equipment and Seismic Classification System

The application of the AP1000 equipment and seismic classification system to AP1000 systems and components is shown in Table 3.2-3. Table 3.2-3 lists safety-related and seismic Category I mechanical and fluid system component and associated equipment class and seismic category as well as other related information. The table also provides information on the systems that contain Class D components. Additional information on the Class D functions of the various systems can be found in the description in the Design Certification Document (DCD) for the systems. Mechanical and fluid systems that contain no safety-related or Class D systems are included in the table and general information provided on the system. Supports for piping and components have the same classification as the component or piping supported. Supports for AP1000 equipment Class A, B, and C mechanical components and piping are constructed to ASME Code, Section III, Subsection NF requirements. The principal construction code for supports for nonsafety-related components and piping is the same as that for the supported component or piping.

Following the name of each system is the building location of the system components. Some of the systems supply all or most of the buildings. This is indicated by identifying the location as various.

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Where a system includes piping or ducts that only passed through a building without including any components that building is generally not included in the list.

The following list includes the systems in Table 3.2-3. The three letters in the beginning of each line is the acronym for the system. The systems included in Table 3.2-3 are listed alphabetically by three letter acronym. Those systems marked with an asterisk * are electrical or instrumentation systems and are not included in Table 3.2-3. The components in the incore instrumentation system that have a pressure boundary function are included in the table. See Section 3.11 for identification of safety-related electrical and instrumentation equipment.

NSSS/Steam Generator Controls and Auxiliaries

BDS	Steam Generator Blowdown System
CNS	Containment System
CVS	Chemical and Volume Control System
PCS	Passive Containment Cooling System
PXS	Passive Core Cooling System
RCS	Reactor Coolant System
RNS	Normal Residual Heat Removal System
RXS	Reactor System
SGS	Steam Generator System

Nuclear Control and Monitoring

*DAS	Diverse Actuation System
IIS	Incore Instrumentation System
*OCS	Operation and Control Centers
*PMS	Protection and Safety Monitoring System
PSS	Primary Sampling System
*RMS	Radiation Monitoring System
*SJS	Seismic Monitoring System
*SMS	Special Monitoring System

Main Power Cycle and Auxiliaries

CDS	Condensate System

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CFS	Turbine Island Chemical Feed System
CPS	Condensate Polishing System
DTS	Demineralized Water Treatment System
DWS	Demineralized Water Transfer and Storage System
FWS	Main and Startup Feedwater System
GSS	Gland Seal System
HDS	Heater Drain System
MSS	Main Steam System
MTS	Main Turbine System
RWS	Raw Water System
TDS	Turbine Island Vents, Drains and Relief System

Class 1E and Emergency Power Systems

*IDS	Class 1E dc and UPS System

Cooling and Circulating Water

CCS	Component Cooling Water System
CES	Condenser Tube Cleaning System
CWS	Circulating Water System
SFS	Spent Fuel Pool Cooling System
SWS	Service Water System
TCS	Turbine Building Closed Cooling Water System

Auxiliary Steam

ASS	Auxiliary Steam Supply System

Generation and Transmission

*ZAS	Main Generation System

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*ZBS	Transmission Switchyard and Offsite Power System
*ZVS	Excitation and Voltage Regulation System

Radwaste

WGS	Gaseous Radwaste System
WLS	Liquid Radwaste System
WRS	Radioactive Waste Drain System
WSS	Solid Radwaste System

HVAC

VAS	Radiologically Controlled Area Ventilation System
VBS	Nuclear Island Nonradioactive Ventilation System
VCS	Containment Recirculation Cooling System
VES	Main Control Room Emergency Habitability System
VFS	Containment Air Filtration System
VHS	Health Physics and Hot Machine Shop HVAC System
VLS	Containment Hydrogen Control System
VRS	Radwaste Building HVAC System
VTS	Turbine Building Ventilation System
VUS	Containment Leak Rate Test System
VWS	Central Chilled Water System
VXS	Annex/Auxiliary Nonradioactive Ventilation System
VYS	Hot Water Heating System
VZS	Diesel Generator Building Ventilation System

Turbine-Generator Controls and Auxiliary

CMS	Condenser Air Removal System
HCS	Generator Hydrogen and CO_2 Systems

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HSS	Hydrogen Seal Oil System
LOS	Main Turbine and Generator Lube Oil System
*TOS	Main Turbine Control and Diagnostics System

Material Handling

FHS	Fuel Handling and Refueling System
MHS	Mechanical Handling System

Piping Services

CAS	Compressed and Instrument Air Systems
DOS	Standby Diesel Fuel Oil System
FPS	Fire Protection System
PGS	Plant Gas Systems
PWS	Potable Water System

Non-Class 1E Power Systems

*ECS	Main AC Power System
*EDS	Non-Class 1E dc and UPS System
ZOS	Onsite Standby Power System
*ZRS	Offsite Retail Power System

Miscellaneous Electrical Systems

*EFS	Communication Systems
*EGS	Grounding and Lightning Protection System
*EHS	Special Process Heat Tracing System
*ELS	Plant Lighting System
*EQS	Cathodic Protection System

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Non-Nuclear Controls and Monitoring

*DDS	Data Display and Processing System
*MES	Meteorological and Environmental Monitoring System
*PLS	Plant Control System
*SES	Plant Security System
SSS	Secondary Sampling System
*TVS	Closed Circuit TV System

Non-Radioactive Drains

DRS	Storm Drain System
RDS	Gravity and Roof Drain Collection System
SDS	Sanitary Drainage System
WWS	Waste Water System

Those systems marked with an asterisk (*) are electrical or instrumentation systems and are not included in Table 3.2-3.

3.2.5 Combined License Information

This section contained no requirement for additional information.

3.2.6 References

1. ANSI N18.2a - 1975, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.”

2. ANS/ANSI 51.1 - 1983, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.”

3. IEEE 323-74, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.”

4. IEEE 344-1987, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.”

5. ASME B31.1-1989 Edition, “Power Piping,” including 1989 Addenda.

6. API 610, “Centrifugal Pumps for General Refinery Services.”

7. Not used.

8. ANSI B 16.34 - 1981, “Valves - Flanged and Buttwelding End.”

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9. API-650-80, “Welded Steel Tanks for Oil Storage,” Revision 1, February 1984.

10. AWWA D100-84, “Welded Steel Tanks for Water Storage.”

11. ANSI B96.1-81, “Welded Aluminum-Alloy Storage Tanks.”

12. API-620-82, “Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks,” Revision 1, April 1985.

13. NEMA MG-1-98, “Motors and Generators,” Revision 1, January 1998, National Electric Manufacturers Association.

14. IEEE C37, IEEE standards on circuit breakers, switch gear, relays, substations, fuses, etc.

15. “Uniform Building Code (1997),” International Conference of Building Officials.

16. SMACNA - 1995, HVAC Duct Construction Standards - Metal and Flexible, Sheet Metal and Air-Conditioning Contractors National Association.

17. The BOCA Basic/National Plumbing Code 1984: Model Plumbing Regulations for the Protection of Public Health, Safety and Welfare: Sixth Edition, Building Officials and Code Administrators International.

18. ASME N509-1989, “Nuclear Power Plant Air Cleaning Units and Components.”

19. International Building Code, 2006.

20. ASME AG-1-1997, “Code on Nuclear Air and Gas Treatment.”

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Table 3.2-1
Comparison of Safety Classification Requirements

AP1000 Code Letter (1)	ANS Equipment Safety Class (2)	RG 1.29 Seismic Design Reqmnts (3)	ASME Code, Sec. III Class (4)	IEEE Requirements	RG 1.26 NRC Quality Group (5)	10 CFR 50 Appendix B (6)	Inspection & Testing Requirements	Required Test & Maint.
A	SC-1	I	1	NA	GROUP A	YES	YES(7)	(8)
B	SC-2	I	2	NA	GROUP B	YES	YES(7)	(8)
C	SC-3	I(14)	3	1E	GROUP C	YES	YES(7)	(8)
D	NNS(2)	NA(9)(14)	NA(10)	(10)	GROUP D	NO(10)	YES(11)	(11)
OTHER	NNS(2)	NA(13)	NA	NA	NA	NA(12)	NA	NA

NA - Not Applicable OTHER includes Classes E, F, L, P, R, and W.

Notes:

1. A single letter equipment classification identifies the safety class, quality group, and other classifications for AP1000. See the Subsection 3.2.2 for definition.
2. AP1000 safety classification is an adaptation of that defined in ANSI 51.1. The NNS defined in the ANSI 51.1 standard is divided into several AP1000 equipment classifications namely, Classes D E, F, L, P, R, and W.
3. See Subsection 3.2.1 for definition of seismic categories.
4. ASME Boiler and Pressure Vessel Code, Section III defines various classes of structures, systems, and components for nuclear power plants. It defines criteria and requirements based on the classification. It is not applicable for nonsafety-related components.
5. The quality group classification corresponds to those provided in Regulatory Guide 1.26.
6. “Yes” means quality assurance program is required according to 10 CFR 50 Appendix B.
   “No” means quality assurance program is not required according to 10 CFR 50 Appendix B.
7. Class A, B, and C, structures, systems, and components built to ASME Code, Section III are inspected to ASME Code, Section XI requirements. See the text for additional specification of requirements.
8. Class A, B, and C structures, systems, and components that are required to function to mitigate design base accidents have some testing requirements included in the plant technical specifications. In addition to the requirements in the technical specifications, testing and maintenance requirements are included in an administratively controlled reliability assurance plan.
9. See Subsection 3.2.1 for cases when seismic Category II requirements are applicable for Class D structures, systems, and components.
10. See the text for a discussion of the industry standards used in the construction of Class D structures, systems and components.
11. Class D structures, systems, and components have selected reliability assurance programs and procedures to provide availability when needed. These programs are administratively controlled programs and are not included in the technical specifications.
12. Normal industrial quality standards are followed in procuring, designing, fabricating, and testing these nonsafety-related structures, systems, and components, at the discretion of the designer. When documentation of compliance with normal industrial quality standards is unavailable, the SSC's adequacy may be affirmed with an evaluation by the designer of the confirmed attributes of the SSC design or construction.
13. Some Class E, F, G, L, P, R, and W structures, systems, and components may be classified as seismic Category II. See Subsection 3.7.3.
14. See Subsection 3.5.2 for seismic Category II tornado missile barriers that provide tornado missile protection for the openings and penetrations in auxiliary building walls.

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Table 3.2-2
Seismic Classification of Building Structures

Structure	Category[1]
Nuclear Island Basemat Containment Interior Shield Building Auxiliary Building Containment Air Baffle	C-I
Containment Vessel	C-I
Plant Vent and Stair Structure	C-II
Turbine Building – First bay adjacent to Nuclear Island outlined by Columns I.1 to R, 11.05 to 11.2, and 11.02 to 11.2	C-II
Turbine Building – All portions of Turbine Building except first bay adjacent to Nuclear Island as outlined by Columns H.05 to R and 12.1 to 20	NS[2]
Turbine Building	NS[2]
Annex Building Area Outlined by Columns A - D and 8 - 13 Area Outlined by Columns A - G and 13 - 16	NS[2]
Annex Building Area Outlined by Columns E - I.1 and 2 - 13	C-II
Radwaste Building	NS[2]
Diesel-Generator Building	NS[3]
Circulating Water Pumphouse and Towers	NS

C-I – Seismic Category I
C-II – Seismic Category II
NS – Non-seismic

Note:

1. Within the broad definition of seismic Category I and II structures, these buildings contain members and structural subsystems the failure of which would not impair the capability for safe shutdown. Examples of such systems would be elevators, stairwells not required for access in the event of a postulated earthquake, and nonstructural partitions in nonsafety-related areas. These substructures are classified as non-seismic.
2. The NS designation for the turbine building, the radwaste building, and a portion of the annex building indicates that the buildings are not seismic Category I or seismic Category II. The seismic requirements for these buildings are outlined in Subsection 3.7.2.8.
3. The NS designation for the diesel-generator building indicates that the building is not seismic Category I or seismic Category II. The seismic requirements for buildings containing Class D equipment, including the diesel generator building, are outlined in Subsection 3.2.2.6.

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Table 3.2-3 (Sheet 1 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Auxiliary Steam Supply System (ASS)				Location: Turbine Building	
System components are Class E					
Steam Generator Blowdown System (BDS)				Location: Turbine Building	
BDS-PL-V009	Discharge to WLS Vent	D	NS	ANSI B16.34	
BDS-PL-V037	Discharge to WLS Valve	D	NS	ANSI B16.34	
BDS-PL-V046	WWS Contaminated Flow Connection	D	NS	ANSI B16.34	
Balance of system components are Class E					
Compressed and Instrument Air System (CAS)				Location: Various	
CAS-PL-V014	Instrument Air Supply Outside Containment Isolation	B	I	ASME III-2	
CAS-PL-V015	Instrument Air Supply Inside Containment Isolation	B	I	ASME III-2	
CAS-PL-V027	Containment Penetration Test Connection Isolation	B	I	ASME III-2	
CAS-PL-V204	Service Air Supply Outside Containment Isolation	B	I	ASME III-2	
CAS-PL-V205	Service Air Supply Inside Containment Isolation	B	I	ASME III-2	
CAS-PL-V219	Containment Penetration Test Connection Isolation	B	I	ASME III-2	
CAS-PY-C02	Containment Instrument Air Inlet Penetration	B	I	ASME III, MC	
CAS-PY-C03	Containment Service Air Inlet Penetration	B	I	ASME III, MC	
Balance of system components are Class E					
Component Cooling Water System (CCS)			Location: Auxiliary Building and Turbine Building		
n/a	Heat Exchangers, CCS and SWS Side	D	NS	ASME VIII	
n/a	Pumps	D	NS	Hydraulic Institute Stds.	
n/a	Tanks	D	NS	ASME VIII	
n/a	Valves Providing CCS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
CCS-PL-V200	CCS Containment Isolation Valve - Inlet Line ORC	B	I	ASME III-2	
CCS-PL-V201	CCS Containment Isolation Valve - Inlet Line IRC	B	I	ASME III-2	
CCS-PL-V207	CCS Containment Isolation Valve - Outlet Line IRC	B	I	ASME III-2	
CCS-PL-V208	CCS Containment Isolation Valve - Outlet Line ORC	B	I	ASME III-2	
CCS-PL-V209	Containment Isolation Valve Test Connection - Outlet Line	B	I	ASME III-2	
CCS-PL-V214	CCS Supply Containment Isolation - IRC	C	I	ASME III-3	

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Table 3.2-3 (Sheet 2 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Component Cooling Water System (Continued)					
CCS-PL-V215	CCS Supply Containment Isolation Valve Test Connection - IRC	C	I	ASME III-3	
CCS-PL-V216	Containment Leak Test Outlet Line - IRC	C	I	ASME III-3	
CCS-PL-V217	Containment Isolation Valve V207 Body Test Connection Valve	C	I	ASME III-3	
CCS-PL-V270	CCS IRC Relief Valve	C	I	ASME III-3	
CCS-PL-V271	CCS IRC Relief Valve	C	I	ASME III-3	
CCS-PL-V220	CCS Containment Isolation Relief Valve	B	I	ASME III-2	
CCS-PL-V257	Containment Isolation Valve Test Connection - Inlet Line	B	I	ASME III-2	
CCS-PY-C01	Containment Supply Header Penetration	B	I	ASME III, MC	
CCS-PY-C02	Containment Return Header Penetration	B	I	ASME III, MC	
Balance of system components are Class E					
Condensate System (CDS)				Location: Turbine Building	
System components are Class E					
Condenser Tube Cleaning System (CES)				Location: Turbine Building	
System components are Class E					
Turbine Island Chemical Feed System (CFS)				Location: Turbine Building and SWS Chemical Treatment Building/Area	
System components are Class E					
Condenser Air Removal System (CMS)				Location: Turbine Building	
n/a	Condenser Vacuum Breakers	E	NS	ANSI B16.34	
Balance of system components are Class D					
Containment System (CNS)				Location: Containment	
CNS-MV-01	Containment Vessel	B	I	ASME III, MC	
CNS-MY-Y01	Equipment Hatch	B	I	ASME III, MC	
CNS-MY-Y02	Maintenance Hatch	B	I	ASME III, MC	
CNS-MY-Y03	Upper Personnel Hatch - 135′-3″	B	I	ASME III, MC	
CNS-MY-Y04	Lower Personnel Hatch - 107′-2″	B	I	ASME III, MC	
n/a	Spare Containment Penetrations	B	I	ASME III, MC	
Condensate Polishing System (CPS)				Location: Turbine Building	
System components are Class E					
Chemical and Volume Control System (CVS)				Location: Containment, Auxiliary Building, and Annex Building	
n/a	Heat Exchangers, CVS and CCS Side	D	NS	ASME VIII/ TEMA	

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Table 3.2-3 (Sheet 3 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Chemical and Volume Control System (Continued)					
n/a	Pumps	D	NS	Hydraulic Institute Stds.	
n/a	Tanks (Except CVS-MT-03 and CVS-MT-05)	D	NS	API 650	
CVS-MT-03	CVS Chemical Mixing Tank	E	NS	ASME VIII	
CVS-MT-05	CVS Air Intrusion Prevention Tank	D	NS	ASME VIII	
n/a	Demineralizers	D	NS	ASME VIII	
n/a	Filters	D	NS	ASME VIII	
n/a	Valves Providing CVS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
CVS-PL-V001	RCS Purification Stop	A	I	ASME III-1	
CVS-PL-V002	RCS Purification Stop	A	I	ASME III-1	
CVS-PL-V003	RCS Purification Stop	C	I	ASME III-3	
CVS-PL-V040	Resin Flush IRC Isolation	B	I	ASME III-2	
CVS-PL-V041	Resin Flush ORC Isolation	B	I	ASME III-2	
CVS-PL-V042	Flush Line Containment Isolation Relief	B	I	ASME III-2	
CVS-PL-V045	Letdown Containment Isolation IRC	B	I	ASME III-2	
CVS-PL-V046	Letdown Pressure Instrument Root	B	I	ASME III-2	
CVS-PL-V047	Letdown Containment Isolation ORC	B	I	ASME III-2	
CVS-PL-V058	Letdown Line Containment Isolation Thermal Relief	B	I	ASME III-2	
CVS-PL-V065	Zinc Addition – IRC Shutoff	C	I	ASME III-3	
CVS-PL-V067	Makeup Return Line Bypass Check Valve	A	I	ASME III-1	
CVS-PL-V080	RCS Purification Return Line Check Valve	C	I	ASME III-3	
CVS-PL-V081	RCS Purification Return Line Stop Valve	A	I	ASME III-1	
CVS-PL-V082	RCS Purification Return Line Check Valve	A	I	ASME III-1	
CVS-PL-V084	Auxiliary Pressurizer Spray Line Isolation	A	I	ASME III-1	
CVS-PL-V085	Auxiliary Pressurizer Spray Line	A	I	ASME III-1	
CVS-PL-V090	Makeup Line Containment Isolation	B	I	ASME III-2	
CVS-PL-V091	Makeup Line Containment Isolation	B	I	ASME III-2	
CVS-PL-V092	Zinc Injection Containment Isolation ORC	B	I	ASME III-2	

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Table 3.2-3 (Sheet 4 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Chemical and Volume Control System (Continued)					
CVS-PL-V094	Zinc Injection Containment Isolation IRC	B	I	ASME III-2	
CVS-PL-V095	Zinc Add Containment Isolation Test Connection	C	I	ASME III-3	
CVS-PL-V096	Zinc Injection Containment Isolation Test Connection	B	I	ASME III-2	
CVS-PL-V098	Zinc Addition Line Containment Isolation Thermal Relief Valve	B	I	ASME III-2	
CVS-PL-V100	Makeup Line Containment Isolation Relief	B	I	ASME III-2	
CVS-PL-V136A	Demineralized Water System Isolation	C	I	ASME III-3	
CVS-PL-V136B	Demineralized Water System Isolation	C	I	ASME III-3	
CVS-PL-V215	Hydrogen Injection – IRC Shutoff	C	I	ASME III-3	
CVS-PL-V216	Hydrogen Injection Containment Isolation Test Connection	C	I	ASME III-3	
CVS-PL-V217	Hydrogen Injection Containment Isolation Check IRC	B	I	ASME III-2	
CVS-PL-V218	Hydrogen Injection Containment Isolation Test Connection	B	I	ASME III-2	
CVS-PL-V219	Hydrogen Injection Containment Isolation ORC	B	I	ASME III-2	
CVS-PY-C01	Demineralizer Resin Flush Line Containment Penetration	B	I	ASME III, MC	
CVS-PY-C02	Letdown Line Containment Penetration	B	I	ASME III, MC	
CVS-PY-C03	Makeup Line Containment Penetration	B	I	ASME III, MC	
CVS-PY-C04	Zinc Add Line Containment Penetration	B	I	ASME III, MC	
CVS-PY-C05	Hydrogen Add Line Containment Penetration	B	I	ASME III, MC	
Balance of system components are Class D or E					
Circulating Water System (CWS)			Location: Turbine Building and pump intake structure		
System components are Class E					

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Table 3.2-3 (Sheet 5 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Standby Diesel Fuel Oil System (DOS)				Location: Diesel Generator Building and yard	
n/a	Fuel Oil Transfer Package	D	NS	Manufacturer Std.	
n/a	Fuel Oil Storage Tanks	D	NS	API 650	
n/a	Fuel Oil Day Tanks	D	NS	ASME VIII	
n/a	Valves Providing DOS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
n/a	Ancillary Diesel Generator Fuel Tank	D	II	Non-Stamped ASME VIII	Located in Annex Building
Balance of system components are Class E					
Storm Drain System (DRS)				Location: Various	
System components are Class E					
Demineralized Water Treatment System (DTS)				Location: Turbine Building	
System components are Class E					
Demineralized Water Transfer and Storage System (DWS)				Location: Various	
n/a	Condensate Storage Tanks	D	NS	API 650	
n/a	Valves Providing DWS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
DWS-PL-V241	DWS Containment Penetration Thermal Relief Valve	C	I	ASME III-3	
DWS-PL-V244	Demineralized Water Supply Containment Isolation - Outside	B	I	ASME III-2	
DWS-PL-V245	Demineralized Water Supply Containment Isolation - Inside	B	I	ASME III-2	
DWS-PL-V248	Containment Penetration Test Connection Isolation	B	I	ASME III-2	
DWS-PY-C01	Containment Demineralized Water Supply Penetration	B	I	ASME III, MC	
Balance of system components are Class E					
Electrical Distribution System (ECS)				Location: Annex Building	
n/a	Ancillary Diesel Generator Engines	D	NS	Manufacturer Standard	Anchorage is SCII
n/a	Ancillary Diesel Generator Radiators	D	NS	Manufacturer Std.	
n/a	Ancillary Diesel Generator Silencers	D	NS	Manufacturer Std.	
n/a	Valve providing fuel to ECS Ancillary Diesel Generators	D	NS	ANSI B16.34	
Balance of system components are Class E					

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Table 3.2-3 (Sheet 6 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Fuel Handling and Refueling System (FHS)			Location: Containment and Auxiliary Building		
FHS-FH-01	Refueling Machine	D	II	AISC	
FHS-FH-02	Fuel Handling Machine	D	II	AISC	
FHS-FH-04	New Fuel Elevator	D	II	AISC	
FHS-FH-05	Fuel Transfer System	D	II	AISC	
FHS-FH-52	Spent Fuel Assembly Handling Tool	D	II	AISC	
FHS-FS-01	New Fuel Storage Rack	D	I	Manufacturer Std.	
FHS-FS-06	In-containment Fuel Storage Rack	D	I	Manufacturer Std.	
FHS-FS-21	Spent Fuel Storage Rack A1	D	I	Manufacturer Std.	
FHS-FS-22	Spent Fuel Storage Rack A2	D	I	Manufacturer Std.	
FHS-FS-23	Spent Fuel Storage Rack A3	D	I	Manufacturer Std.	
FHS-FS-24	Spent Fuel Storage Rack B1	D	I	Manufacturer Std.	
FHS-FS-25	Spent Fuel Storage Rack B2	D	I	Manufacturer Std.	
FHS-FS-26	Spent Fuel Storage Rack B3	D	I	Manufacturer Std.	
FHS-FS-27	Spent Fuel Storage Rack B4	D	I	Manufacturer Std.	
FHS-FS-28	Spent Fuel Storage Rack C1	D	I	Manufacturer Std.	
FHS-FT-01	Fuel Transfer Tube	B	I	ASME III Class MC	Jurisdictional Boundary CV Only, Remaining Portion Optional
FHS-MT-01	Spent Fuel Pool	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MT-02	Fuel Transfer Canal	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MT-03	Refueling Cavity	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MT-05	Spent Fuel Cask Loading Pit	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MT-06	Spent Fuel Cask Washdown Pit	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MY-Y01	Spent Fuel Transfer Gate	C	I	ANSI/AISC N690 for structural steel	Manufacturer's Std for non-structural components
FHS-MY-Y02	Spent Fuel Cask Loading Pit Gate	C	I	ANSI/AISC N690 for structural steel	Manufacturer's Std for non-structural components
FHS-MY-Y03	Permanent Cavity Seal Ring	C	I	ANSI/AISC N690	
FHS-PL-V001	Fuel transfer tube Isolation Valve	C	I	ASME-III-3	
FHS-PY-B01	Fuel Transfer Tube Hatch/Blind Flange	B	I	ASME III Class MC	
Balance of system components are Class E					

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Table 3.2-3 (Sheet 7 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Fire Protection System (FPS)				Location: Various	
FPS-PL-V050	Fire Water Containment Supply Isolation	B	I	ASME III-2	
FPS-PL-V051	Fire Water Containment Test Connection Isolation	B	I	ASME III-2	
FPS-PL-V052	Fire Water Containment Supply Isolation - Inside	B	I	ASME III-2	
FPS-PY-C01	Fire Protection Containment Penetration	B	I	ASME III, MC	
FPS-PL-V441	Auxiliary Connection to CCS Isolation	D	NS	ASME B31.1	
FPS-PL-V702	FPS Containment Penetration Thermal Relief Valve	C	I	ASME III-3	
Containment standpipe and fire hose connections	Includes all FPS components Inside Reactor Containment with the exception of those used for containment isolation, cable tray suppression, and containment spray	F	NS	ASME B31.1	Seismic Analysis Consistent with ASME Section III Class 3 Systems
Various	Auxiliary Building (Non-Radiologically Controlled) Standpipe and Non-1E Equipment Penetration Room Preaction Sprinkler System piping	F	NS	ASME B31.1	Seismic Analysis Consistent with ASME Section III Class 3 Systems
Balance of system components are Class E, F & G					
Main and Startup Feedwater System (FWS)				Location: Turbine Building	
n/a	Startup Feedwater Pumps	D	NS	Hydraulic Institute Standards	
n/a	Valves Providing SFW AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
Balance of system components are Class E					
Gland Seal System (GSS)				Location: Turbine Building	
System components are Class D					
Generator Hydrogen and CO_2 Systems (HCS)				Location: Turbine Building	
System components are Class E					
Heater Drain System (HDS)				Location: Turbine Building	
System components are Class E					
Hydrogen Seal Oil System (HSS)				Location: Turbine Building	
System components are Class E					
Incore Instrumentation System (IIS)				Location: Containment	
n/a	In-core Instrument Thimble Assembly Tubes	D	II	Manufacturer Std.	
n/a	Thimble assemblies	B	I	Manufacturer Std.	

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Table 3.2-3 (Sheet 8 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Main Turbine and Generator Lube Oil System (LOS)				Location: Turbine Building	
System components are Class E					
Mechanical Handling System (MHS)				Location: Various	
MHS-MH-01	Containment Polar Crane	D	I	NUREG-0554 supplemented by ASME NOG-1	
MHS-MH-02	Cask Handling Crane	D	I	NUREG-0554 supplemented by ASME NOG-1	
MHS-MH-05	Equipment Hatch Hoist	C	I	NUREG-0554 supplemented by ASME NOG-1	
MHS-MH-06	Maintenance Hatch Hoist	C	I	NUREG-0554 supplemented by ASME NOG-1	
Balance of system components are Class E					
Main Steam System (MSS)				Location: Turbine Building	
System components are Class E					
Main Turbine System (MTS)				Location: Turbine Building	
System components are Class E					
Passive Containment Cooling System (PCS)			Location: Containment Shield Building and Auxiliary Building		
PCS-JE-FE001	PCCWST Flow Through L001A Element	C	I	ASME III-3	
PCS-JE-FE002	PCCWST Flow Through L001B Element	C	I	ASME III-3	
PCS-JE-FE003	PCCWST Flow Through L001C Element	C	I	ASME III-3	
PCS-JE-FE004	PCCWST Flow Through L001D Element	C	I	ASME III-3	
PCS-MT-01	Passive Containment Cooling Water Storage Tank	C	I	ACI 349	See subsection 6.2.2.2.3 for additional design requirements
PCS-MT-02	PCS Chemical Addition Tank	D	II	ASME VIII	
PCS-MT-03	Water Distribution Bucket	C	I	Manufacturer Std.	See subsection 6.2.2.2.3 for additional design requirements
PCS-MT-04	Water Collection Troughs	C	I	Manufacturer Std.	See subsection 6.2.2.2.3 for additional design requirements
PCS-MT-05	Passive Containment Cooling Ancillary Water Storage Tank	D	II	API 650	
PCS-MT-06	PCCWST Leak Chase Collection Pot	D	II	ASME VIII	
PCS-MY-Y01	PCCWST Screen 1	C	I	Manufacturer Std.	
PCS-MY-Y02	PCCWST Screen 2	C	I	Manufacturer Std.	
PCS-MY-Y03	PCCWST Screen 3	C	I	Manufacturer Std.	

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Table 3.2-3 (Sheet 9 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-MY-Y04	PCCWST Screen 4	C	I	Manufacturer Std.	
PCS-MY-Y05	PCCWST Screen 5	C	I	Manufacturer Std.	
PCS-PL-V001A	PCCWST Isolation	C	I	ASME III-3	
PCS-PL-V001B	PCCWST Isolation	C	I	ASME III-3	
PCS-PL-V001C	PCCWST Isolation	C	I	ASME III-3	
PCS-MP-01A	PCS Recirculation Pump	D	NS	Hydraulic Institute Standards	Equipment Anchorage is Seismic Category II
PCS-MP-01B	PCS Recirculation Pump	D	NS	Hydraulic Institute Standards	Equipment Anchorage is Seismic Category II
PCS-PL-V002A	PCCWST Series Isolation	C	I	ASME III-3	
PCS-PL-V002B	PCCWST Series Isolation	C	I	ASME III-3	
PCS-PL-V002C	PCCWST Series Isolation	C	I	ASME III-3	
PCS-PL-V004	Recirculation Bypass Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V005	PCCWST Supply to FPS Isolation	C	I	ASME III-3	
PCS-PL-V009	Spent Fuel Pool Emergency Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V010A	Flow Transmitter FT001 Root Valve	C	I	ASME III-3	
PCS-PL-V010B	Flow Transmitter FT001 Root Valve	C	I	ASME III-3	
PCS-PL-V011A	Flow Transmitter FT002 Root Valve	C	I	ASME III-3	
PCS-PL-V011B	Flow Transmitter FT002 Root Valve	C	I	ASME III-3	
PCS-PL-V012A	Flow Transmitter FT003 Root Valve	C	I	ASME III-3	
PCS-PL-V012B	Flow Transmitter FT003 Root Valve	C	I	ASME III-3	
PCS-PL-V013A	Flow Transmitter FT004 Root Valve	C	I	ASME III-3	
PCS-PL-V013B	Flow Transmitter FT004 Root Valve	C	I	ASME III-3	
PCS-PL-V014	Chemical Addition Tank Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V015	Water Bucket Makeup Line Drain Valve	C	I	ASME III-3	
PCS-PL-V016	PCCWST Drain Isolation Valve	C	I	ASME III-3	
PCS-PL-V017	Chemical Addition Tank Vent Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V018	Recirculation Pump Throttle Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V019	Chemical Addition Tank Fill Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-PL-V020	Water Bucket Makeup Line Isolation Valve	C	I	ASME III-3	
PCS-PL-V021	PCCWST TO Recirculation Pump Suction Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V022	Chemical Addition Tank Drain Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V023	PCS Recirculation Return Isolation	C	I	ASME III-3	
PCS-PL-V025	Pressure Transmitter PT 031 Root Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V026	Makeup to Distribution Bucket Isolation Valve	C	I	ASME III-3	
PCS-PL-V029	PCCWST Isolation Valve Leakage Detection Drain	C	I	ASME III-3	
PCS-PL-V030	PCCWST Isolation Valve Leakage Detection Crossconnect Valve	C	I	ASME III-3	
PCS-PL-V031A	Level Transmitter LT 016 & 010 Root Isolation Valve	C	I	ASME III-3	
PCS-PL-V031B	Level Transmitter LT 015 & 011 Root Isolation Valve	C	I	ASME III-3	
PCS-PL-V033	Recirculation Pump Suction from Long Term Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V035A	Recirculation Pump Suction Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V035B	Recirculation Pump Suction Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V036A/B	Recirculation Pump Discharge Check Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V037	PCCAWST Discharge Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V038	PCCAWST Drain Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V039	PCCWST Long-Term Makeup Check Valve	C	I	ASME III-3	
PCS-PL-V040	Recirculation Pump Suction from PCCAWST Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V041	PCCAWST Recirculation Return Line Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V042	PCCWST Long-Term Makeup Isolation Drain Valve	C	I	ASME III-3	
PCS-PL-V043	PCCAWST Recirculation Return Line Drain Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-PL-V044	PCCWST Long-Term Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V045	Emergency Makeup to the Spent Fuel Pool Isolation Valve	C	I	ASME III-3	
PCS-PL-V046	PCCWST Recirculation Return Isolation Valve	C	I	ASME III-3	
PCS-PL-V047A/B	PCS Recirculation Pump Discharge Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V048	Recirculation Pump Fire Suction Isolation Valve	D	NS	ANSI B16.34	Seismically Analyzed for Operability
PCS-PL-V049	Emergency Makeup to the Spent Fuel Pool Drain Isolation Valve	C	I	ASME III-3	
PCS-PL-V050	Spent Fuel Pool Long Term Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V051	Spent Fuel Pool Emergency Makeup Lower Isolation	C	I	ASME III-3	
PCS-PL-V052	Spent Fuel Pool Emergency Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V053	PCS Recirculation Heater Pressure Relief Valve	D	NS	ASME VIII	
PCS-PL-V060A	Shutoff Valve for Leakage Sensor	C	I	ASME III-3	
PCS-PL-V060B	Shutoff Valve for Leakage Sensor	C	I	ASME III-3	
PCS-PL-V100	Temporary Containment Washdown Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V301	PCCWST to Recirculation Pump Suction Drain Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V303	Recirculation Header Discharge to SFS Pool Vent Isolation Valve	C	I	ASME III-3	
PCS-PL-V304	Recirculation Header Discharge to SFS Pool Drain Isolation Valve	C	I	ASME III-3	
PCS-PL-V305	PCCWST Recirculation Return Drain Isolation Valve	C	I	ASME III-3	
PCS-PY-B01	Spent Fuel Pool Emergency Makeup Isolation	C	I	ASME III-3	
PCS-PY-C01	Containment Pressure Instrument Line Penetration	B	I	ASME, MC	
PCS-PY-C02	Containment Pressure Instrument Line Penetration	B	I	ASME, MC	
PCS-PY-C03	Containment Pressure Instrument Line Penetration	B	I	ASME, MC	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-PY-C04	Containment Pressure Instrument Line Penetration	B	I	ASME, MC	
Balance of system components are Class E or F					
Plant Gas Systems (PGS)				Location: Various	
System components are Class E					
Primary Sampling System (PSS)			Location: Containment and Auxiliary Building		
n/a	Grab Sample Unit	D	NS	Manufacturer Std.	
n/a	Sample Cooler, PSS and CCS Side	D	NS	ASME VIII/ TEMA	
n/a	Valves Providing PSS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
PSS-PL-V001A	Hot Leg Sample Isolation	B	I	ASME III-2	
PSS-PL-V001B	Hot Leg Sample Isolation	B	I	ASME III-2	
PSS-PL-V003	Pressurizer Sample Isolation	B	I	ASME III-2	
PSS-PL-V004A	PXS Accumulator Sample Isolation	C	I	ASME III-3	
PSS-PL-V004B	PXS Accumulator Sample Isolation	C	I	ASME III-3	
PSS-PL-V005A	PXS CMT A Sample Isolation	B	I	ASME III-2	
PSS-PL-V005B	PXS CMT B Sample Isolation	B	I	ASME III-2	
PSS-PL-V005C	PXS CMT A Sample Isolation	B	I	ASME III-2	
PSS-PL-V005D	PXS CMT B Sample Isolation	B	I	ASME III-2	
PSS-PL-V008	Containment Air Sample Containment Isolation IRC	B	I	ASME III-2	
PSS-PL-V010A	Liquid Sample Line Containment Isolation IRC	B	I	ASME III-2	
PSS-PL-V010B	Liquid Sample Line Containment Isolation IRC	B	I	ASME III-2	
PSS-PL-V011A	Liquid Sample Line Containment Isolation ORC	B	I	ASME III-2	
PSS-PL-V011B	Liquid Sample Line Containment Isolation ORC	B	I	ASME III-2	
PSS-PL-V012A	Liquid Sample Isolation Valve	C	I	ASME III-3	
PSS-PL-V012B	Liquid Sample Check Valve	C	I	ASME III-3	
PSS-PL-V013	RCS Pressurizer Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V014A	RCS Hot Leg 1 Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V014B	RCS Hot Leg 2 Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V015A	PXS Accumulator Sample Isolation Valve	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Primary Sampling System (Continued)					
PSS-PL-V015B	PXS Accumulator Sample Isolation Valve	C	I	ASME III-3	
PSS-PL-V016A	PXS CMT A Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V016B	PXS CMT B Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V016C	PXS CMT A Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V016D	PXS CMT B Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V023	Sample Return Line Containment Isolation ORC	B	I	ASME III-2	
PSS-PL-V024	Sample Return Containment Isolation IRC	B	I	ASME III-2	
PSS-PL-V046	Air Sample Line Containment Isolation ORC	B	I	ASME III-2	
PSS-PL-V076A	Containment Testing Boundary Isolation Valve	C	I	ASME III-3	
PSS-PL-V076B	Containment Testing Boundary Isolation Valve	C	I	ASME III-3	
PSS-PL-V082	Containment Isolation Test Connection Isolation Valve	C	I	ASME III-3	
PSS-PL-V083	Containment Isolation Test Connection Isolation Valve	C	I	ASME III-3	
PSS-PL-V085	Containment Isolation Test Connection Isolation Valve	B	I	ASME III-2	
PSS-PL-V086	Containment Isolation Test Connection Isolation Valve	C	I	ASME III-3	
PSS-PL-V902A	PXS CMT A Sample Isolation Valve	B	I	ASMEIII-2	
PSS-PL-V902B	PXS CMT B Sample Isolation Valve	B	I	ASMEIII-2	
PSS-PL-V903	PXS CMT B Sample Isolation Valve	B	I	ASMEIII-2	
PSS-PY-C01	Common Primary Sample Line Penetration	B	I	ASME III, MC	
PSS-PY-C02	Containment Atmosphere Sample Line Penetration	B	I	ASME III, MC	
PSS-PY-C03	Containment Atmosphere Sample Line Penetration	B	I	ASME III, MC	
PSS-PY-C04	RCS Hot Leg Sample Line Penetration	B	I	ASME III, MC	
PSS-PY-Y01	Delay Coil 1 for RCS Hot Leg 1	C	I	ASME III-3	
PSS-PY-Y02	Delay Coil 2 for RCS Hot Leg 2	C	I	ASME III-3	
PSS-MY-Y05	Delay Coil Assembly	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Primary Sampling System (Continued)					
Balance of system components are Class E					
Potable Water System (PWS)				Location: Various	
PWS-PL-V418	PWS MCR Isolation Valve	C	I	ASME III-3	
PWS-PL-V420	PWS MCR Isolation Valve	C	I	ASME III-3	
PWS-PL-V498	PWS MCR Vacuum Relief	C	I	ASME III-3	
Balance of system components are Class E, P, and W					
Passive Core Cooling System (PXS)				Location: Containment	
PXS-ME-01	Passive Residual Heat Removal Heat Exchanger	A	I	ASME III-1	
PXS-MT-01A	Accumulator Tank A	C	I	ASME III-3	
PXS-MT-01B	Accumulator Tank B	C	I	ASME III-3	
PXS-MT-02A	Core Makeup Tank A	A	I	ASME III-1	
PXS-MT-02B	Core Makeup Tank B	A	I	ASME III-1	
PXS-MT-03	In-Containment Refueling Water Storage Tank	C	I	ACI 349/ANSI/AISC N690	ACI 349 is used for Evaluation of Structural Boundary
PXS-MT-04	IRWST Gutter	C	I	Manufacturer Std.	
PXS-MW-01A	Reactor Coolant Depressurization Sparger A	C	I	ASME III-3	
PXS-MW-01B	Reactor Coolant Depressurization Sparger B	C	I	ASME III-3	
PXS-MY-Y01A	IRWST Screen A	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y01B	IRWST Screen B	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y01C	IRWST Screen C	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y02A	Containment Recirculation Screen A	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-MY-Y02B	Containment Recirculation Screen B	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y03A	pH Adjustment Basket 3A	C	I	Manufacturer Std.	
PXS-MY-Y03B	pH Adjustment Basket 3B	C	I	Manufacturer Std.	
PXS-MY-Y04A	pH Adjustment Basket 4A	C	I	Manufacturer Std.	
PXS-MY-Y04B	pH Adjustment Basket 4B	C	I	Manufacturer Std.	
PXS-MY-Y11A	CMT A Upper Level Standpipe	A	I	ASME III-1	
PXS-MY-Y11B	CMT B Upper Level Standpipe	A	I	ASME III-1	
PXS-MY-Y12A	CMT A Upper Level Standpipe	A	I	ASME III-1	
PXS-MY-Y12B	CMT B Upper Level Standpipe	A	I	ASME III-1	
PXS-MY-Y13A	CMT A Lower Level Standpipe	A	I	ASME III-1	
PXS-MY-Y13B	CMT B Lower Level Standpipe	A	I	ASME III-1	
PXS-MY-Y14A	CMT A Lower Level Standpipe	A	I	ASME III-1	
PXS-MY-Y14B	CMT B Lower Level Standpipe	A	I	ASME III-1	
PXS-MY-Y21	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y22	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y23	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y24	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y25	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y26	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y27	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y28	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y29	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y30	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y31	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y32	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y33	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y41	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y47	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y48	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y61	IRWST SG Wall Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y62	IRWST SG Wall Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y63	IRWST SG Wall Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y64	IRWST SG Wall Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y71	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y72	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y73	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y74	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y75	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y76	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-MY-Y81	Downspout Screen 1A	C	I	Manufacturer Std.	
PXS-MY-Y82	Downspout Screen 1B	C	I	Manufacturer Std.	
PXS-MY-Y83	Downspout Screen 1C	C	I	Manufacturer Std.	
PXS-MY-Y84	Downspout Screen 1D	C	I	Manufacturer Std.	
PXS-MY-Y85	Downspout Screen 2A	C	I	Manufacturer Std.	
PXS-MY-Y86	Downspout Screen 2B	C	I	Manufacturer Std.	
PXS-MY-Y87	Downspout Screen 2C	C	I	Manufacturer Std.	
PXS-MY-Y88	Downspout Screen 2D	C	I	Manufacturer Std.	
PXS-PL-V002A	CMT A CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V002B	CMT B CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V010A	CMT A Upper Sample	B	I	ASME III-2	
PXS-PL-V010B	CMT B Upper Sample	B	I	ASME III-2	
PXS-PL-V011A	CMT A Lower Sample	B	I	ASME III-2	
PXS-PL-V011B	CMT B Lower Sample	B	I	ASME III-2	
PXS-PL-V012A	CMT A Drain	A	I	ASME III-1	
PXS-PL-V012B	CMT B Drain	A	I	ASME III-1	
PXS-PL-V013A	CMT A Discharge Manual Isolation	A	I	ASME III-1	
PXS-PL-V013B	CMT B Discharge Manual Isolation	A	I	ASME III-1	
PXS-PL-V014A	CMT A Discharge Isolation	A	I	ASME III-1	
PXS-PL-V014B	CMT B Discharge Isolation	A	I	ASME III-1	
PXS-PL-V015A	CMT A Discharge Isolation	A	I	ASME III-1	
PXS-PL-V015B	CMT B Discharge Isolation	A	I	ASME III-1	
PXS-PL-V016A	CMT A Discharge Check	A	I	ASME III-1	
PXS-PL-V016B	CMT B Discharge Check	A	I	ASME III-1	
PXS-PL-V017A	CMT A Discharge Check	A	I	ASME III-1	
PXS-PL-V017B	CMT B Discharge Check	A	I	ASME III-1	
PXS-PL-V019A	RNS to CMT Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V019B	RNS to CMT Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V020A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V020B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V021A	Accumulator A Nitrogen Vent	C	I	ASME III-3	
PXS-PL-V021B	Accumulator B Nitrogen Vent	C	I	ASME III-3	
PXS-PL-V022A	Accumulator A Pressure Relief	C	I	ASME III-3	
PXS-PL-V022B	Accumulator B Pressure Relief	C	I	ASME III-3	
PXS-PL-V023A	Accumulator A Pressure Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V023B	Accumulator B Pressure Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V024A	Accumulator A Pressure Transmitter A Isolation	C	I	ASME III-3	

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Table 3.2-3 (Sheet 17 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V024B	Accumulator B Pressure Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V025A	Accumulator A Sample	C	I	ASME III-3	
PXS-PL-V025B	Accumulator B Sample	C	I	ASME III-3	
PXS-PL-V026A	Accumulator A Drain	C	I	ASME III-3	
PXS-PL-V026B	Accumulator B Drain	C	I	ASME III-3	
PXS-PL-V027A	Accumulator A Discharge Isolation	C	I	ASME III-3	
PXS-PL-V027B	Accumulator B Discharge Isolation	C	I	ASME III-3	
PXS-PL-V028A	Accumulator A Discharge Check	A	I	ASME III-1	
PXS-PL-V028B	Accumulator B Discharge Check	A	I	ASME III-1	
PXS-PL-V029A	Accumulator A Discharge Check	A	I	ASME III-1	
PXS-PL-V029B	Accumulator B Discharge Check	A	I	ASME III-1	
PXS-PL-V030A	CMT A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V030B	CMT B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V031A	CMT A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V031B	CMT B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V033A	Accumulator A Check Valve Drain	B	I	ASME III-2	
PXS-PL-V033B	Accumulator B Check Valve Drain	B	I	ASME III-2	
PXS-PL-V042	Nitrogen Supply Containment Isolation ORC	B	I	ASME III-2	
PXS-PL-V043	Nitrogen Supply Containment Isolation IRC Check Valve	B	I	ASME III-2	
PXS-PL-V052	Accumulator Nitrogen Containment Penetration TC	B	I	ASME III-2	
PXS-PL-V080A	CMT A WR Level Isolation	B	I	ASME III-2	
PXS-PL-V080B	CMT B WR Level Isolation	B	I	ASME III-2	
PXS-PL-V081A	CMT A WR Level Isolation	B	I	ASME III-2	
PXS-PL-V081B	CMT B WR Level Isolation	B	I	ASME III-2	
PXS-PL-V082A	CMT A Upper Level A Isolation 1	A	I	ASME III-1	
PXS-PL-V082B	CMT B Upper Level A Isolation 1	A	I	ASME III-1	
PXS-PL-V083A	CMT A Upper Level A Isolation 2	A	I	ASME III-1	
PXS-PL-V083B	CMT B Upper Level A Isolation 2	A	I	ASME III-1	
PXS-PL-V084A	CMT A Upper Level A Vent	B	I	ASME III-2	
PXS-PL-V084B	CMT B Upper Level A Vent	B	I	ASME III-2	

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Table 3.2-3 (Sheet 18 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V085A	LT-011 A/C Drain Valve	B	I	ASME III-2	
PXS-PL-V085B	LT-012 A/C Drain Valve	B	I	ASME III-2	
PXS-PL-V086A	CMT A Upper Level B Isolation 1	A	I	ASME III-1	
PXS-PL-V086B	CMT B Upper Level B Isolation 1	A	I	ASME III-1	
PXS-PL-V087A	CMT A Upper Level B Isolation 2	A	I	ASME III-1	
PXS-PL-V087B	CMT B Upper Level B Isolation 2	A	I	ASME III-1	
PXS-PL-V088A	CMT A Upper Level B Vent	B	I	ASME III-2	
PXS-PL-V088B	CMT B Upper Level B Vent	B	I	ASME III-2	
PXS-PL-V089A	CMT A Upper Level B Drain	B	I	ASME III-2	
PXS-PL-V089B	CMT B Upper Level B Drain	B	I	ASME III-2	
PXS-PL-V092A	CMT A Lower Level A Isolation 1	A	I	ASME III-1	
PXS-PL-V092B	CMT B Lower Level A Isolation 1	A	I	ASME III-1	
PXS-PL-V093A	CMT A Lower Level A Isolation 2	A	I	ASME III-1	
PXS-PL-V093B	CMT B Lower Level A Isolation 2	A	I	ASME III-1	
PXS-PL-V094A	CMT A Lower Level A Vent	B	I	ASME III-2	
PXS-PL-V094B	CMT B Lower Level A Vent	B	I	ASME III-2	
PXS-PL-V095A	CMT A Lower Level A Drain	B	I	ASME III-2	
PXS-PL-V095B	CMT B Lower Level A Drain	B	I	ASME III-2	
PXS-PL-V096A	CMT A Lower Level B Isolation 1	A	I	ASME III-1	
PXS-PL-V096B	CMT B Lower Level B Isolation 1	A	I	ASME III-1	
PXS-PL-V097A	CMT A Lower Level B Isolation 2	A	I	ASME III-1	
PXS-PL-V097B	CMT B Lower Level B Isolation 2	A	I	ASME III-1	
PXS-PL-V098A	CMT A Lower Level B Vent	B	I	ASME III-2	
PXS-PL-V098B	CMT B Lower Level B Vent	B	I	ASME III-2	
PXS-PL-V099A	CMT A Lower Level B Drain	B	I	ASME III-2	
PXS-PL-V099B	CMT B Lower Level B Drain	B	I	ASME III-2	
PXS-PL-V101	PRHR HX Inlet Isolation	A	I	ASME III-1	
PXS-PL-V102A	PRHR HX Inlet Head Vent	B	I	ASME III-2	
PXS-PL-V102B	PRHR HX Inlet Head Drain	B	I	ASME III-2	
PXS-PL-V103A	PRHR HX Outlet Head Vent	B	I	ASME III-2	
PXS-PL-V103B	PRHR HX Outlet Head Drain	B	I	ASME III-2	
PXS-PL-V104A	PRHR HX Flow Transmitter A Isolation	B	I	ASME III-2	

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Table 3.2-3 (Sheet 19 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V104B	PRHR HX Flow Transmitter B Isolation	B	I	ASME III-2	
PXS-PL-V105A	PRHR HX Flow Transmitter A Isolation	B	I	ASME III-2	
PXS-PL-V105B	PRHR HX Flow Transmitter B Isolation	B	I	ASME III-2	
PXS-PL-V106	Containment Recirculation A Highpoint Vent	C	I	ASME III-3	
PXS-PL-V107	Containment Recirculation A Highpoint Vent	C	I	ASME III-3	
PXS-PL-V108A	PRHR HX Control	A	I	ASME III-1	
PXS-PL-V108B	PRHR HX Control	A	I	ASME III-1	
PXS-PL-V109	PRHR HX/RCS Return Isolation	A	I	ASME III-1	
PXS-PL-V111A	PRHR HX Highpoint Vent	B	I	ASME III-2	
PXS-PL-V111B	PRHR HX Highpoint Vent	B	I	ASME III-2	
PXS-PL-V113	PRHR HX Pressure Transmitter Isolation	B	I	ASME III-2	
PXS-PL-V115A	Containment Recirculation A Drain	C	I	ASME III-3	
PXS-PL-V115B	Containment Recirculation B Drain	C	I	ASME III-3	
PXS-PL-V116A	Containment Recirculation A Drain	C	I	ASME III-3	
PXS-PL-V116B	Containment Recirculation B Drain	C	I	ASME III-3	
PXS-PL-V117A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V117B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V118A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V118B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V119A	Containment Recirculation A Check	C	I	ASME III-3	
PXS-PL-V119B	Containment Recirculation B Check	C	I	ASME III-3	
PXS-PL-V120A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V120B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V121A	IRWST Line A Isolation	C	I	ASME III-3	
PXS-PL-V121B	IRWST Line B Isolation	C	I	ASME III-3	
PXS-PL-V122A	IRWST Injection A Check	A	I	ASME III-1	
PXS-PL-V122B	IRWST Injection B Check	A	I	ASME III-1	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V123A	IRWST Injection A Isolation	A	I	ASME III-1	
PXS-PL-V123B	IRWST Injection B Isolation	A	I	ASME III-1	
PXS-PL-V124A	IRWST Injection A Check	A	I	ASME III-1	
PXS-PL-V124B	IRWST Injection B Check	A	I	ASME III-1	
PXS-PL-V125A	IRWST Injection A Isolation	A	I	ASME III-1	
PXS-PL-V125B	IRWST Injection B Isolation	A	I	ASME III-1	
PXS-PL-V126A	IRWST Injection Check Test	C	I	ASME III-3	
PXS-PL-V126B	IRWST Injection Check Test	C	I	ASME III-3	
PXS-PL-V127	IRWST Injection Line A Drain	C	I	ASME III-3	
PXS-PL-V128A	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V128B	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V129A	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V129B	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V130A	IRWST Gutter Bypass A Isolation	C	I	ASME III-3	
PXS-PL-V130B	IRWST Gutter Bypass B Isolation	C	I	ASME III-3	
PXS-PL-V131A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V131B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V132A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V132B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V133A	IRWST Injection Line A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V133B	IRWST Injection Line B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V134A	IRWST Injection Line A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V134B	IRWST Injection Line B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V135A	IRWST Injection Line A Highpoint Vent Isolation	B	I	ASME III-2	
PXS-PL-V135B	IRWST Injection Line B Highpoint Vent Isolation	B	I	ASME III-2	
PXS-PL-V149	RNS Suction Pump Line Drain	C	I	ASME III-3	
PXS-PL-V151B	IRWST Wide Range Level Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V151C	IRWST Wide Range Level Transmitter C Isolation	C	I	ASME III-3	
PXS-PL-V151D	IRWST Wide Range Level Transmitter D Isolation	C	I	ASME III-3	
PXS-PL-V161A	IRWST Lower Narrow Range Level Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V161B	IRWST Lower Narrow Range Level Transmitter B Isolation	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V161C	IRWST Lower Narrow Range Level Transmitter C Isolation	C	I	ASME III-3	
PXS-PL-V161D	IRWST Lower Narrow Range Level Transmitter D Isolation	C	I	ASME III-3	
PXS-PL-V162A	IRWST Lower Narrow Range Level Transmitter A Reference Leg Isolation	C	I	ASME III-3	
PXS-PL-V162B	IRWST Lower Narrow Range Level Transmitter B Reference Leg Isolation	C	I	ASME III-3	
PXS-PL-V162C	IRWST Lower Narrow Range Level Transmitter C Reference Leg Isolation	C	I	ASME III-3	
PXS-PL-V162D	IRWST Lower Narrow Range Level Transmitter D Reference Leg Isolation	C	I	ASME III-3	
PXS-PL-V170A	PRHR Flow Transmitter A Vent	B	I	ASME III-2	
PXS-PL-V170B	PRHR Flow Transmitter B Vent	B	I	ASME III-2	
PXS-PL-V171A	PRHR Flow Transmitter A Vent	B	I	ASME III-2	
PXS-PL-V171B	PRHR Flow Transmitter B Vent	B	I	ASME III-2	
PXS-PL-V201A	Accumulator A Leak Test	B	I	ASME III-2	
PXS-PL-V201B	Accumulator B Leak Test	B	I	ASME III-2	
PXS-PL-V202A	Accumulator A Leak Test	C	I	ASME III-3	
PXS-PL-V202B	Accumulator B Leak Test	C	I	ASME III-3	
PXS-PL-V205A	RNS Discharge Leak Test	B	I	ASME III-2	
PXS-PL-V205B	RNS Discharge Leak Test	B	I	ASME III-2	
PXS-PL-V206	RNS Discharge Leak Test	C	I	ASME III-3	
PXS-PL-V207A	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V207B	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V208A	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V217	PXS Leak Test Line Isolation	D	NS	ASME B31.1	
PXS-PL-V221	Test Header to IRWST	D	NS	ASME B31.1	
PXS-PL-V230A	CMT A Fill Isolation	B	I	ASME III-2	
PXS-PL-V230B	CMT B Fill Isolation	B	I	ASME III-2	
PXS-PL-V231A	CMT A Fill Check	B	I	ASME III-2	
PXS-PL-V231B	CMT B Fill Check	B	I	ASME III-2	
PXS-PL-V232A	Accumulator A Fill/Drain Isolation	C	I	ASME III-3	
PXS-PL-V232B	Accumulator B Fill/Drain Isolation	C	I	ASME III-3	
PXS-PL-V233A	Accumulator A to PXS Test Header Iso Valve	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V233B	Accumulator B to PXS Test Header Iso Valve	C	I	ASME III-3	
PXS-PL-V250A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V250B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V251A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V251B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V252A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V252B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PY-C01	Nitrogen Makeup Containment Penetration	B	I	ASME III, MC	
PXS-PY-E01	DVI-B Inline Expansion Joint	C	I	ASME III-3	
PXS-PY-R01A	Core Makeup Tank A Orifice	A	I	ASME III-1	
PXS-PY-R01B	Core Makeup Tank B Orifice	A	I	ASME III-1	
PXS-PY-R02A	Accumulator Tank A Orifice	C	I	ASME III-3	
PXS-PY-R02B	Accumulator Tank B Orifice	C	I	ASME III-3	
Balance of system components are Class E					
Reactor Coolant System (RCS)				Location: Containment	
RCS-MB-01	Steam Generator 1	A	I	ASME III-1	
n/a	SG 1 Shell	B	I	ASME III-1	
n/a	SG 1 Channel Head Divider Plate	B	I	ASME III-1	
n/a	SG 1 Tube Bundle Support Assembly	C	I	Manufacturer Std	
n/a	SG 1 Steam Flow Limiting Venturi	B	I	ASME III-1	See Subsections 5.4.4.2 and 5.4.4.3 for additional requirements.
n/a	SG 1 Feedwater Distribution Ring Supports	C	I	Manufacturer Std	
RCS-MB-02	Steam Generator 2	A	I	ASME III-1	
n/a	SG 2 Shell	B	I	ASME III-1	
n/a	SG 2 Channel Head Divider Plate	B	I	ASME III-1	
n/a	SG 2 Tube Bundle Support Assembly	C	I	Manufacturer Std	
n/a	SG 2 Steam Flow Limiting Venturi	B	I	ASME III-1	See Subsections 5.4.4.2 and 5.4.4.3 for additional requirements.

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
n/a	SG 2 Feedwater Distribution Ring Supports	C	I	Manufacturer Std	
RCS-MP-01A/B	SG 1A(B) Reactor Coolant Pump	A	I	ASME III-1	Pump Motor – Class D
n/a	Rotor Shaft	C	I	Manufacturer Std	
n/a	Impeller	C	I	Manufacturer Std	
n/a	Flywheel	C	I	Manufacturer Std	
n/a	RCP Heat Exchanger (Tube Side)	A	I	ASME III-1	Shellside – Class D, ASME VIII, Div. 1
n/a	Pump Motor Cooling Water to HX Inlet Connector	A	I	ASME III-1	
n/a	Pump Motor Cooling Water from HX Outlet Connector	A	I	ASME III-1	
RCS-MP-02A/B	SG 2A(B) Reactor Coolant Pump	A	I	ASME III-1	Pump Motor – Class D
n/a	Rotor Shaft	C	I	Manufacturer Std	
n/a	Impeller	C	I	Manufacturer Std	
n/a	Flywheel	C	I	Manufacturer Std	
n/a	RCP Heat Exchanger (Tube Side)	A	I	ASME III-1	Shellside – Class D, ASME VIII, Div. 1
n/a	Pump Motor Cooling Water to HX Inlet Connector	A	I	ASME III-1	
n/a	Pump Motor Cooling Water from HX Outlet Connector	A	I	ASME III-1	
RCS-MV-01	Reactor Vessel	A	I	ASME III-1	The reactor vessel has two tag numbers; RCS-MV-01 for the reactor coolant system (RCS) and RXS-MV-01 for the reactor system (RXS)
RCS-MV-02	Pressurizer	A	I	ASME III-1	
RCS-PL-V001A	First Stage ADS	A	I	ASME III-1	
RCS-PL-V001B	First Stage ADS	A	I	ASME III-1	
RCS-PL-V002A	Second Stage ADS	A	I	ASME III-1	
RCS-PL-V002B	Second Stage ADS	A	I	ASME III-1	
RCS-PL-V003A	Third Stage ADS	A	I	ASME III-1	
RCS-PL-V003B	Third Stage ADS	A	I	ASME III-1	
RCS-PL-V004A	Fourth Stage ADS	A	I	ASME III-1	
RCS-PL-V004B	Fourth Stage ADS	A	I	ASME III-1	
RCS-PL-V004C	Fourth Stage ADS	A	I	ASME III-1	
RCS-PL-V004D	Fourth Stage ADS	A	I	ASME III-1	
RCS-PL-V005A	Pressurizer Safety Valve	A	I	ASME III-1	
RCS-PL-V005B	Pressurizer Safety Valve	A	I	ASME III-1	
RCS-PL-V007A	ADS Test Valve	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
RCS-PL-V007B	ADS Test Valve	B	I	ASME III-2	
RCS-PL-V007C	ADS Test Valve	B	I	ASME III-2	
RCS-PL-V008	ADS Valve Leakage Check Valve	C	I	ASME III-3	
RCS-PL-V010A	ADS Discharge Header A Vacuum Relief	C	I	ASME III-3	
RCS-PL-V010B	ADS Discharge Header B Vacuum Relief	C	I	ASME III-3	
RCS-PL-V011A	First Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V011B	First Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V012A	Second Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V012B	Second Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V013A	Third Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V013B	Third Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V014A	Fourth Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V014B	Fourth Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V014C	Fourth Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V014D	Fourth Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V095	Hot Leg 2 Level Instrument Root	B	I	ASME III-2	
RCS-PL-V096	Hot Leg 2 Level Instrument Root	B	I	ASME III-2	
RCS-PL-V097	Hot Leg 1 Level Instrument Root	B	I	ASME III-2	
RCS-PL-V098	Hot Leg 1 Level Instrument Root	B	I	ASME III-2	
RCS-PL-V101A	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101B	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101C	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101D	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101E	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101F	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102A	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102B	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102C	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102D	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
RCS-PL-V102E	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102F	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V103	PRHR HX Outlet Line Drain	B	I	ASME III-2	
RCS-PL-V108A	Hot Leg 1 Sample Isolation	B	I	ASME III-2	
RCS-PL-V108B	Hot Leg 2 Sample Isolation	B	I	ASME III-2	
RCS-PL-V110A	Pressurizer Spray Valve	A	I	ASME III-1	
RCS-PL-V110B	Pressurizer Spray Valve	A	I	ASME III-1	
RCS-PL-V111A	Pressurizer Spray Block Valve	A	I	ASME III-1	
RCS-PL-V111B	Pressurizer Spray Block Valve	A	I	ASME III-1	
RCS-PL-V120	Reactor Vessel Flange Leakoff	D	NS	ASME B31.1	
RCS-PL-V121	Reactor Vessel Flange Leakoff	D	NS	ASME B31.1	
RCS-PL-V122A	Reactor Vessel Flange Leakoff	D	NS	ASME B31.1	
RCS-PL-V122B	Reactor Vessel Flange Leakoff	D	NS	ASME B31.1	
RCS-PL-V150A	Reactor Vessel Head Vent	A	I	ASME III-1	
RCS-PL-V150B	Reactor Vessel Head Vent	A	I	ASME III-1	
RCS-PL-V150C	Reactor Vessel Head Vent	A	I	ASME III-1	
RCS-PL-V150D	Reactor Vessel Head Vent	A	I	ASME III-1	
RCS-PL-V171A	Cold Leg 1A Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V171B	Cold Leg 1A Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V172A	Cold Leg 1B Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V172B	Cold Leg 1B Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V173A	Cold Leg 2A Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V173B	Cold Leg 2A Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V174A	Cold Leg 2B Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V174B	Cold Leg 2B Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V204	Pressurizer Manual Vent	A	I	ASME III-1	
RCS-PL-V205	Pressurizer Manual Vent	A	I	ASME III-1	
RCS-PL-V210A	Pressurizer Spray Bypass	B	I	ASME III-2	
RCS-PL-V210B	Pressurizer Spray Bypass	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
RCS-PL-V015	Pressurizer Vent to RCDT Test Valve	D	NS	ANSI B16.34	
RCS-PL-V225A	Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V225B	Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V225C	Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V225D	Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V226A	Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V226B	Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V226C	Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V226D	Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V227A	Pressurizer Level Reference Leg Tubing Instrumentation Root	B	I	ASME III-2	
RCS-PL-V227B	Pressurizer Level Reference Leg Tubing Instrumentation Root	B	I	ASME III-2	
RCS-PL-V227C	Pressurizer Level Reference Leg Tubing Instrumentation Root	B	I	ASME III-2	
RCS-PL-V227D	Pressurizer Level Reference Leg Tubing Instrumentation Root	B	I	ASME III-2	
RCS-PL-V228	Pressurizer Liquid Sample Isolation	B	I	ASME III-2	
RCS-PL-V229	Wide Range Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V232	Manual Head Vent	C	I	ASME III-3	
RCS-PL-V233	Head Vent Isolation	C	I	ASME III-3	
RCS-PL-V241	ADS Valve Discharge Header Drain Isolation	C	I	ASME III-3	
RCS-PL-V242	ADS Valve Discharge Header Drain Check	D	NS	ANSI B16.34	
RCS-PL-V250	ADS Discharge Line Isolation	C	I	ASME III-3	
RCS-PL-V260A	RCP 1A Vent	A	I	ASME III-1	
RCS-PL-V260B	RCP 1B Vent	A	I	ASME III-1	
RCS-PL-V260C	RCP 2A Vent	A	I	ASME III-1	
RCS-PL-V260D	RCP 2B Vent	A	I	ASME III-1	
RCS-PL-V261A	RCP 1A Drain	A	I	ASME III-1	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
RCS-PL-V261B	RCP 1B Drain	A	I	ASME III-1	
RCS-PL-V261C	RCP 2A Drain	A	I	ASME III-1	
RCS-PL-V261D	RCP 2B Drain	A	I	ASME III-1	
RCS-PY-B01	ADS Hydrostatic Test Spectacle Flange	C	I	ASME III-3	
RCS-PY-B02	ADS Hydrostatic Test Spectacle Flange	C	I	ASME III-3	
RCS-PY-B03	RCS Vacuum Pump Suction Spectacle Blind	C	I	ASME III-3	
RCS-PY-B04	RCS Vacuum Ejector Package Suction Spectacle Blind	C	I	ASME III-3	
RCS-PY-K03	Safety Valve Discharge Chamber Rupture Disk	C	I	ASME III-3	
RCS-PY-K04	Safety Valve Discharge Chamber Rupture Disk	C	I	ASME III-3	
RCS-PY-R01A	Reactor Vessel Head Vent Flow Orifice A	C	I	ASME III-3	
RCS-PY-R01B	Reactor Vessel Head Vent Flow Orifice B	C	I	ASME III-3	
RCS-PY-Y01A	Pressurizer Level Reference Leg L225A Flex Hose	B	I	ASME III-2	
RCS-PY-Y01B	Pressurizer Level Reference Leg L225B Flex Hose	B	I	ASME III-2	
RCS-PY-Y01C	Pressurizer Level Reference Leg L225C Flex Hose	B	I	ASME III-2	
RCS-PY-Y01D	Pressurizer Level Reference Leg L225D Flex Hose	B	I	ASME III-2	
Gravity and Roof Drain Collection System (RDS)				Location: Various	
System components are Class E					
Normal Residual Heat Removal System (RNS)			Location: Containment and Auxiliary Building		
RNS-JE-FE001A	RNS Train A Discharge Flow Element	C	I	ASME III-3	
RNS-JE-FE001B	RNS Train B Discharge Flow Element	C	I	ASME III-3	
RNS-ME-01A	Normal Residual Heat Removal Heat Exchanger A (Tube Side)	C	I	ASME III-3	Shellside – Class D ASME VIII, Div. 1
RNS-ME-01B	Normal Residual Heat Removal Heat Exchanger B (Tube Side)	C	I	ASME III-3	Shellside – Class D ASME VIII, Div. 1
RNS-MP-01A	Residual Heat Removal Pump A	C	I	ASME III-3	Pump Motor - Class D
RNS-MP-01B	Residual Heat Removal Pump B	C	I	ASME III-3	Pump Motor - Class D
RNS-PL-V001A	RNS HL Suction Isolation - Inner	A	I	ASME III-1	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Normal Residual Heat Removal System (Continued)					
RNS-PL-V001B	RNS HL Suction Isolation - Inner	A	I	ASME III-1	
RNS-PL-V002A	RNS HL Suction and Containment Isolation - Outer	A	I	ASME III-1	
RNS-PL-V002B	RNS HL Suction and Containment Isolation - Outer	A	I	ASME III-1	
RNS-PL-V003A	RCS Pressure Boundary Valve Thermal Relief	B	I	ASME III-2	
RNS-PL-V003B	RCS Pressure Boundary Valve Thermal Relief	B	I	ASME III-2	
RNS-PL-V004A	RCS Pressure Boundary Valve Thermal Relief Isolation	B	I	ASME III-2	
RNS-PL-V004B	RCS Pressure Boundary Valve Thermal Relief Isolation	B	I	ASME III-2	
RNS-PL-V005A	RNS Pump A Suction Isolation	C	I	ASME III-3	
RNS-PL-V005B	RNS Pump B Suction Isolation	C	I	ASME III-3	
RNS-PL-V006A	RNS HX A Outlet Flow Control	C	I	ASME III-3	
RNS-PL-V006B	RNS HX B Outlet Flow Control	C	I	ASME III-3	
RNS-PL-V007A	RNS Pump A Discharge Isolation	C	I	ASME III-3	
RNS-PL-V007B	RNS Pump B Discharge Isolation	C	I	ASME III-3	
RNS-PL-V008A	RNS HX A Bypass Flow Control	C	I	ASME III-3	
RNS-PL-V008B	RNS HX B Bypass Flow Control	C	I	ASME III-3	
RNS-PL-V010	RNS Discharge Containment Isolation Valve Test	C	I	ASME III-3	
RNS-PL-V011	RNS Discharge Containment Isolation Valve - ORC	B	I	ASME III-2	
RNS-PL-V012	RNS Discharge Containment Isolation Valve Test Connection ORC	B	I	ASME III-2	
RNS-PL-V013	RNS Discharge Containment Isolation - IRC	B	I	ASME III-2	
RNS-PL-V014	RNS Discharge Containment Isolation Valve Test Connection	C	I	ASME III-3	
RNS-PL-V015A	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V015B	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V016	RNS Discharge Containment Penetration Isolation Valves Test	C	I	ASME III-3	
RNS-PL-V017A	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Normal Residual Heat Removal System (Continued)					
RNS-PL-V017B	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V020	RNS HL Suction Pressure Relief	B	I	ASME III-2	
RNS-PL-V021	RNS HL Suction Pressure Relief	B	I	ASME III-2	
RNS-PL-V022	RNS Suction Header Containment Isolation - ORC	B	I	ASME III-2	
RNS-PL-V023	RNS Suction from IRWST - Containment Isolation	B	I	ASME III-2	
RNS-PL-V024	RNS Discharge to IRWST Isolation	C	I	ASME III-3	
RNS-PL-V025	RNS Suction from IRWST - Bonnet Relief Isolation	C	I	ASME III-3	
RNS-PL-V026	RNS Suction from IRWST - Containment Isolation Test	C	I	ASME III-3	
RNS-PL-V029	RNS Discharge to CVS	C	I	ASME III-3	
RNS-PL-V030A	RNS HX A Shell Drain	D	NS	ASME B31.1	
RNS-PL-V030B	RNS HX B Shell Drain	D	NS	ASME B31.1	
RNS-PL-V031A	RNS Train A Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V031B	RNS Train B Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V032A	RNS Train A Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V032B	RNS Train B Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V033A	RNS Pump A Suction Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V033B	RNS Pump B Suction Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V034A	RNS Pump A Discharge Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V034B	RNS Pump B Discharge Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V035A	RNS HX A Shell Vent	D	NS	ANSI B16.34	
RNS-PL-V035B	RNS HX B Shell Vent	D	NS	ANSI B16.34	
RNS-PL-V036A	RNS Pump A Suction Piping Drain. Isolation	C	I	ASME III-3	
RNS-PL-V036B	RNS Pump B Suction Piping Drain. Isolation	C	I	ASME III-3	
RNS-PL-V045	RNS Pump Discharge Relief	C	I	ASME III-3	
RNS-PL-V048A	RNS Pump Seal Cooler A Vent Isolation	C	I	ASME III-3	
RNS-PL-V048B	RNS Pump Seal Cooler B Vent Isolation	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Normal Residual Heat Removal System (Continued)					
RNS-PL-V049A	RNS Pump Seal Cooler A Drain Isolation	C	I	ASME III-3	
RNS-PL-V049B	RNS Pump Seal Cooler B Drain Isolation	C	I	ASME III-3	
RNS-PL-V050	RNS Pump A Casing Drain. Isolation	C	I	ASME III-3	
RNS-PL-V051	RNS Pump B Casing Drain. Isolation	C	I	ASME III-3	
RNS-PL-V052	RNS Pump Suction From Spent Fuel Pool Isolation	C	I	ASME III-3	
RNS-PL-V053	RNS Pump Discharge to Spent Fuel Pool Isolation	C	I	ASME III-3	
RNS-PL-V055	RNS Pump Suction to Cask Loading Pit Isolation	C	I	ASME III-3	
RNS-PL-V056	RNS Pump Suction to Cask Loading Pit Isolation	C	I	ASME III-3	
RNS-PL-V057A	RNS Pump A Miniflow Isolation	C	I	ASME III-3	
RNS-PL-V057B	RNS Pump B Miniflow Isolation	C	I	ASME III-3	
RNS-PL-V059	RNS Pump Suction Containment Isolation Test Connection	C	I	ASME III-3	
RNS-PL-V061	RNS Return from CVS - Containment Isolation	B	I	ASME III-2	
RNS-PL-V065	RNS Discharge Drain Valve	C	I	ASME III-3	
RNS-PL-V066A	RNS Discharge to DVI Line A Drain	C	I	ASME III-3	
RNS-PL-V066B	RNS Discharge to DVI Line B Drain	C	I	ASME III-3	
RNS-PL-V067A	RNS Discharge to DVI Line A Drain	B	I	ASME III-2	
RNS-PL-V067B	RNS Discharge to DVI Line B Drain	B	I	ASME III-2	
RNS-PL-V068	RNS Discharge to IRWST Drain	C	I	ASME III-3	
RNS-PL-V069A	RNS Pump A Miniflow Vent	C	I	ASME III-3	
RNS-PL-V069B	RNS Pump B Miniflow Vent	C	I	ASME III-3	
RNS-PL-V071A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V071B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V072A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V072B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Normal Residual Heat Removal System (Continued)					
RNS-PL-V073A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V073B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V074A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V074B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V075A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V075B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V080	IRWST Suction Line to RNS Pump Vent	B	I	ASME III-2	
RNS-PL-V081	RNS Cask Loading Pit Suction Line Vent	C	I	ASME III-3	
RNS-PL-V082	RNS Discharge Drain	C	I	ASME III-3	
RNS-PY-C01	Normal Residual Heat Removal Suction Line Penetration	B	I	ASME III, MC	
RNS-PY-C02	Normal Residual Heat Removal Discharge Line Penetration	B	I	ASME III, MC	
RNS-PY-R01A	Residual Heat Removal Discharge to DVI Line Orifice A	C	I	ASME III-3	
RNS-PY-R01B	Residual Heat Removal Discharge to DVI Line Orifice B	C	I	ASME III-3	
RNS-PY-R02A	Residual Heat Removal Pump Miniflow Orifice A	C	I	ASME III-3	
RNS-PY-R02B	Residual Heat Removal Pump Miniflow Orifice B	C	I	ASME III-3	
RNS-PY-R03	Residual Heat Removal to IRWST Line Orifice	C	I	ASME III-3	
RNS-PY-R04	Residual Heat Removal to Spent Fuel Pool Line Orifice	C	I	ASME III-3	
Balance of system components are Class E					
Raw Water System (RWS)				Location: Yard, Turbine Building	
System components are Class E					
Reactor System (RXS)				Location: Containment	
n/a	Fuel Assemblies	C	I	Manufacturer Std.	
RXS-FR-B06	Control Rod Cluster B6	B	I	Manufacturer Std.	
RXS-FR-B10	Control Rod Cluster B10	B	I	Manufacturer Std.	
RXS-FR-C05	Control Rod Cluster C5	B	I	Manufacturer Std.	
RXS-FR-C07	Control Rod Cluster C7	B	I	Manufacturer Std.	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-FR-C09	Control Rod Cluster C9	B	I	Manufacturer Std.	
RXS-FR-C11	Control Rod Cluster C11	B	I	Manufacturer Std.	
RXS-FR-D06	Control Rod Cluster D6	B	I	Manufacturer Std.	
RXS-FR-D08	Control Rod Cluster D8	B	I	Manufacturer Std.	
RXS-FR-D10	Control Rod Cluster D10	B	I	Manufacturer Std.	
RXS-FR-E03	Control Rod Cluster E3	B	I	Manufacturer Std.	
RXS-FR-E05	Control Rod Cluster E5	B	I	Manufacturer Std.	
RXS-FR-E07	Control Rod Cluster E7	B	I	Manufacturer Std.	
RXS-FR-E09	Control Rod Cluster E9	B	I	Manufacturer Std.	
RXS-FR-E11	Control Rod Cluster E11	B	I	Manufacturer Std.	
RXS-FR-E13	Control Rod Cluster E13	B	I	Manufacturer Std.	
RXS-FR-F02	Control Rod Cluster F2	B	I	Manufacturer Std.	
RXS-FR-F04	Control Rod Cluster F4	B	I	Manufacturer Std.	
RXS-FR-F12	Control Rod Cluster F12	B	I	Manufacturer Std.	
RXS-FR-F14	Control Rod Cluster F14	B	I	Manufacturer Std	
RXS-FR-G03	Control Rod Cluster G3	B	I	Manufacturer Std.	
RXS-FR-G05	Control Rod Cluster G5	B	I	Manufacturer Std.	
RXS-FR-G07	Control Rod Cluster G7	B	I	Manufacturer Std.	
RXS-FR-G09	Control Rod Cluster G9	B	I	Manufacturer Std.	
RXS-FR-G11	Control Rod Cluster G11	B	I	Manufacturer Std.	
RXS-FR-G13	Control Rod Cluster G13	B	I	Manufacturer Std.	
RXS-FR-H04	Control Rod Cluster H4	B	I	Manufacturer Std.	
RXS-FR-H08	Control Rod Cluster H8	B	I	Manufacturer Std.	
RXS-FR-H12	Control Rod Cluster H12	B	I	Manufacturer Std.	
RXS-FR-J03	Control Rod Cluster J3	B	I	Manufacturer Std.	
RXS-FR-J05	Control Rod Cluster J5	B	I	Manufacturer Std.	
RXS-FR-J07	Control Rod Cluster J7	B	I	Manufacturer Std.	
RXS-FR-J09	Control Rod Cluster J9	B	I	Manufacturer Std.	
RXS-FR-J11	Control Rod Cluster J11	B	I	Manufacturer Std.	
RXS-FR-J13	Control Rod Cluster J13	B	I	Manufacturer Std.	
RXS-FR-K02	Control Rod Cluster K2	B	I	Manufacturer Std.	
RXS-FR-K04	Control Rod Cluster K4	B	I	Manufacturer Std.	
RXS-FR-K12	Control Rod Cluster K12	B	I	Manufacturer Std.	
RXS-FR-K14	Control Rod Cluster K14	B	I	Manufacturer Std.	
RXS-FR-L03	Control Rod Cluster L3	B	I	Manufacturer Std.	
RXS-FR-L05	Control Rod Cluster L5	B	I	Manufacturer Std.	
RXS-FR-L07	Control Rod Cluster L7	B	I	Manufacturer Std.	
RXS-FR-L09	Control Rod Cluster L9	B	I	Manufacturer Std.	
RXS-FR-L11	Control Rod Cluster L11	B	I	Manufacturer Std.	
RXS-FR-L13	Control Rod Cluster L13	B	I	Manufacturer Std.	
RXS-FR-M06	Control Rod Cluster M6	B	I	Manufacturer Std.	
RXS-FR-M08	Control Rod Cluster M8	B	I	Manufacturer Std.	
RXS-FR-M10	Control Rod Cluster M10	B	I	Manufacturer Std.	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-FR-N05	Control Rod Cluster N05	B	I	Manufacturer Std.	
RXS-FR-N07	Control Rod Cluster N07	B	I	Manufacturer Std.	
RXS-FR-N09	Control Rod Cluster N09	B	I	Manufacturer Std.	
RXS-FR-N11	Control Rod Cluster N11	B	I	Manufacturer Std.	
RXS-FR-P06	Control Rod Cluster P06	B	I	Manufacturer Std.	
RXS-FR-P10	Control Rod Cluster P10	B	I	Manufacturer Std.	
RXS-MI-01	Reactor Upper Internals	C	I	ASME III, CS	
RXS-MI-02	Reactor Lower Internals	C	I	ASME III, CS	
RXS-MI-10	Non-Threaded Fasteners	D	II	Manufacturer Std.	
RXS-MI-11	Threaded Structural Fasteners	C	I	ASME III, CS	
RXS-MI-20	Lower Core Support Plate	C	I	ASME III, CS	
RXS-MI-22	Vortex Suppression Plate	D	II	Manufacturer Std.	
RXS-MI-23	Core Shroud Assembly	D	II	Manufacturer Std.	
RXS-MI-24	Radial Supports [4]	C	I	ASME III, CS	
RXS-MI-25	Core Barrel	C	I	ASME III, CS	
RXS-MI-26	Core Barrel Nozzle	C	I	Manufacturer Std.	
RXS-MI-27	Head and Vessel Pins	D	II	Manufacturer Std.	
RXS-MI-28	Lower Support Plate Fuel Alignment Pins	C	I	ASME III, CS	
RXS-MI-29	Core Barrel Hold Down Spring	C	I	Manufacturer Std.	
RXS-MI-50	Upper Support	C	I	ASME III, CS	
RXS-MI-51	Upper Core Plate	C	I	ASME III, CS	
RXS-MI-52	Support Columns [42]	C	I	ASME III, CS	
RXS-MI-53	Guide Tube Assemblies [69]	C	I	Manufacturer Std.	
RXS-MI-54	Upper Core Plate Fuel Alignment Pins	C	I	ASME III, CS	
RXS-MI-55	Upper Core Plate Inserts	C	I	ASME III, CS	
RXS-MI-56	Safety Injection Deflector	D	II	Manufacturer Std.	
RXS-MI-57	Irradiation Specimen Guide Tubes	D	II	Manufacturer Std.	
RXS-MI-58	Head Cooling Nozzles	D	II	Manufacturer Std.	
n/a	Neutron Pad	D	II	Manufacturer Std.	
n/a	Instrument Grid Assembly	D	II	Manufacturer Std.	
n/a	DVI Flow Diverter	D	II	Manufacturer Std.	
RXS-MI-80	Reactor Vessel Flow Skirt	D	II	Manufacturer Std.	
RXS-MN-01	Reactor Vessel Cavity Reflective Insulation	D	II	Manufacturer Std.	
RXS-MV-01	Reactor Vessel	A	I	ASME III-1	The reactor vessel has two tag numbers; RCS-MV-01 for the reactor coolant system (RCS) and RXS-MV-01 for the reactor system (RXS)

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Table 3.2-3 (Sheet 34 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-10	Reactor Integrated Head Package	C	I	ANSI/AISC N690	
RXS-MV-10A	Integrated Head Package Shroud	C	I	ASME-NF	
RXS-MV-10B	Integrated Head Package Seismic Support System	C	I	ASME-NF	
RXS-MV-11B06/ B08/B10/C05/C07/ C09/C11/D04/ D06/D08/D10/ D12/E03/E05/E07/ E09/E11/E13/F02/ F04/F06/F08/F10/ F12/F14/G03/G05/ G07/G09/G11/G13/ H02/H04/H06/H08/ H10/H12/H14/J03/ J05/J07/J09/J11/ J13/K02/K04/K06/ K08/K10/K12/K14/ L03/L05/L07/L09/ L11/L13/M04/M06/ M08/M10/M12/ N05/N07/N09/N11/ P06/P08/P10	CRDM Latch Assemblies	C	I	Manufacturer Std.	
RXS-MV-11B06/ B08/B10/C05/C07/ C09/C11/D04/ D06/D08/D10/ D12/E03/E05/E07/ E09/E11/E13/F02/ F04/F06/F08/F10/ F12/F14/G03/G05/ G07/G09/G11/G13/ H02/H04/H06/H08/ H10/H12/H14/J03/ J05/J07/J09/J11/ J13/K02/K04/K06/ K08/K10/K12/K14/ L03/L05/L07/L09/ L11/L13/M04/M06/ M08/M10/M12/ N05/N07/N09/N11/ P06/P08/P10	CRDM Drive Rod Assemblies	D	NS	Manufacturer Std.	

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Table 3.2-3 (Sheet 35 of 59)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11B06/ B08/B10/C05/C07/ C09/C11/D04/D06/ D08/D10/D12/E03/ E05/E07/E09/E11/ E13/F02/F04/F06/ F08/F10/F12/F14/ G03/G05/G07/G09/ G11/G13/H02/H04/ H06/H08/H10/H12/ H14/J03/J05/J07/ J09/J11/J13/K02/ K04/K06/K08/K10/ K12/K14/L03/L05/ L07/L09/L11/L13/ M04/M06/M08/ M10/M12/N05/ N07/N09/N11/P06/ P08/P10	CRDM Coil Stack Assemblies	D	NS	Manufacturer Std.	
RXS-MV-11B06L/ B08L/B10L/C05L/ C07L/C09L/C11L/ D04L/D06L/D08L/ D10L/D12L/E03L/ E05L/E07L/E09L/ E11L/E13L/F02L/ F04L/F06L/F08L/ F10L/F12L/F14L/ G03L/G05L/G07L/ G09L/G11L/G13L/ H02L/H04L/H06L/ H08L/H10L/H12L/ H14L/J03L/J05L/ J07L/J09L/J11L/ J13L/K02L/K04L/ K06L/K08L/K10L/ K12L/K14L/L03L/ L05L/L07L/L09L/ L11L/L13L/M04L/ M06L/M08L/ M10L/M12L/N05L/ N07L/N09/N11L/ P06L/P08L/P10L	CRDM Latch Housings	A	I	ASME III-1	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11B06R/ B08R/B10R/C05R/ C07R/C09R/C11R/ D04R/D06R/D08R/ D10R/D12R/E03R/ E05R/E07R/E09R/ E11R/E13R/F02R/ F04R/F06R/F08R/ F10R/F12R/F14R/ G03R/G05R/G07R/ G09R/G11R/G13R/ H02R/H04R/H06R/ H08R/H10R/H12R/ H14R/J03R/J05R/ J07R/J09R/J11R/ J13R/K02R/K04R/ K06R/K08R/K10R/ K12R/K14R/L03R/ L05R/L07R/L09R/ L11R/L13R/M04R/ M06R/M08R/ M10R/M12R/ N05R/N07R/N11R/ P06R/P08R/P10R	CRDM Rod Travel Housings	A	I	ASME III-1	
RXS-MY-Y01	IHP Lower Shroud Assembly	C	I	ASME-NF	
RXS-MY-Y51	Integrated Head Package Lifting Rig	C	II	NUREG-0612, ANSI N14.6	
RXS-MY-Y11	Incore Instrumentation QuickLoc Assembly 1	A	I	ASME III-1	
RXS-MY-Y12	Incore Instrumentation QuickLoc Assembly 2	A	I	ASME III-1	
RXS-MY-Y13	Incore Instrumentation QuickLoc Assembly 3	A	I	ASME III-1	
RXS-MY-Y14	Incore Instrumentation QuickLoc Assembly 4	A	I	ASME III-1	
RXS-MY-Y15	Incore Instrumentation QuickLoc Assembly 5	A	I	ASME III-1	
RXS-MY-Y16	Incore Instrumentation QuickLoc Assembly 6	A	I	ASME III-1	
RXS-MY-Y17	Incore Instrumentation QuickLoc Assembly 7	A	I	ASME III-1	
RXS-MY-Y18	Incore Instrumentation QuickLoc Assembly 8	A	I	ASME III-1	
Balance of system components are Class E					

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Sanitary Drainage System (SDS)				Location: Various	
SDS-PL-V001	SDS MCR Vent Isolation Valve	C	I	ASME III-3	
SDS-PL-V002	SDS MCR Vent Isolation Valve	C	I	ASME III-3	
Balance of system components are Class D, P, and W					
Spent Fuel Pool Cooling System (SFS)			Location: Auxiliary Building, Containment		
n/a	Heat Exchangers, SFS and CCS Side	D	NS	ASME VIII	
n/a	Pumps	D	NS	Hydraulic Institute Std.	
n/a	Demineralizers	D	NS	ASME VIII	
n/a	Filters	D	NS	ASME VIII	
n/a	Valves Providing SFS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
SFS-PL-V024A	Spent Fuel Pool Level Instrument Isolation	C	I	ASME III-3	
SFS-PL-V024B	Spent Fuel Pool Level Instrument Isolation	C	I	ASME III-3	
SFS-PL-V024C	Spent Fuel Pool Level Instrument Isolation	C	I	ASME III-3	
SFS-PL-V028	Cask Washdown Pit Level Instrument Isolation	C	I	ASME III-3	
SFS-PL-V031	SFS Refueling Cavity Drain to SGS Compartment Isolation	C	I	ASME III-3	
SFS-PL-V032	SFS Refueling Cavity Suction Isolation	C	I	ASME III-3	
SFS-PL-V033	SFS Refueling Cavity Drain to Containment Sump Isolation	C	I	ASME III-3	
SFS-PL-V034	SFS Suction Line Containment Isolation	B	I	ASME III-2	
SFS-PL-V035	SFS Suction Line Containment Isolation	B	I	ASME III-2	
SFS-PL-V037	SFS Discharge Line Containment Isolation	B	I	ASME III-2	
SFS-PL-V038	SFS Discharge Line Containment Isolation	B	I	ASME III-2	
SFS-PL-V039	SFS Suction Line from IRWST Isolation	C	I	ASME III-3	
SFS-PL-V040	SFS Fuel Transfer Canal Drain Isolation	C	I	ASME III-3	
SFS-PL-V041	SFS Cask Loading Pit Drain Isolation	C	I	ASME III-3	
SFS-PL-V042	Cask Loading Pit to Pump Suction Isolation	C	I	ASME III-3	
SFS-PL-V043	Cask Loading Pit Level Transmitter Root Isolation Valve	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Spent Fuel Pool Cooling System (Continued)					
SFS-PL-V045	SFS Discharge to Cask Loading Pit Isolation	C	I	ASME III-3	
SFS-PL-V047	SFS Demineralized Water Makeup to SFP Reverse Flow Prevent	D	NS	ANSI B16.34	
SFS-PL-V048	SFS Containment Penetration Test Connection	B	I	ASME III-2	
SFS-PL-V049	SFS Cask Loading Pit Drain to WLS Isolation	C	I	ASME III-3	
SFS-PL-V056	SFS Containment Penetration Test Connection Isolation	B	I	ASME III-2	
SFS-PL-V058	SFS Containment Isolation Valve V034 Test	C	I	ASME III-3	
SFS-PL-V066	Spent Fuel Pool to Cask Washdown Pit Isolation	C	I	ASME III-3	
SFS-PL-V067	SFS Containment Isolation Relief Valve	B	I	ASME III-2	
SFS-PL-V068	Cask Washdown Pit Drain Isolation	C	I	ASME III-3	
SFS-PL-V071	Refueling Cavity Overflow to SG Compartment	C	I	ASME III-3	
SFS-PL-V072	Refueling Cavity Overflow to SG Compartment	C	I	ASME III-3	
SFS-PL-V075	SFS Containment Floodup Isolation Valve	C	I	ASME III-3	
SFS-PL-V117	Refueling Cavity Drain Line Test Connection	C	I	ASME III-3	
SFS-PY-C01	Spent Fuel Cooling Pump Discharge to IRWST	B	I	ASME III, MC	
SFS-PY-C02	Spent Fuel Cooling Pump Suction from IRWST	B	I	ASME III, MC	
SFS-PY-R04	SFS Cask Loading and Washdown Orifice	C	I	ASME III-3	
Balance of system components are Class D					
Steam Generator System (SGS)			Location: Containment and Auxiliary Building		
SGS-MY-Y01A	Steam Generator A PORV Silencer	D	NS	Manufacturer Std.	
SGS-MY-Y01B	Steam Generator B PORV Silencer	D	NS	Manufacturer Std.	
SGS-PL-V001A	LT001, LT011 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V001B	LT005, LT013 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V002A	LT001 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V002B	LT005 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V003A	LT002, LT012 Root Isolation Valve	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V003B	LT006, LT014 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V004A	LT002 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V004B	LT006 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V005A	LT003, LT015, LT044 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V005B	LT007, LT017, LT046 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V006A	LT003 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V006B	LT007 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V007A	LT004, LT016, LT045 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V007B	LT008, LT018, LT047 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V008A	LT004 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V008B	LT008 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V010A	LT011 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V010B	LT013 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V011A	LT011 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V011B	LT013 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V012A	LT012 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V012B	LT014 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V013A	LT012 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V013B	LT014 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V014A	PORV Discharge Condensate Drain Isolation	D	NS	ASME B31.1	
SGS-PL-V014B	PORV Discharge Condensate Drain Isolation	D	NS	ASME B31.1	
SGS-PL-V015A	FT021 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V015B	FT023 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V016A	FT020 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V016B	FT022 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V017A	FT021 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V017B	FT023 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V018A	FT020 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V018B	FT022 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V019A	Main Steam Line Vent Isolation	B	I	ASME III-2	
SGS-PL-V019B	Main Steam Line Vent Isolation	B	I	ASME III-2	
SGS-PL-V020A	FT024 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V020B	FT025 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V021A	FT024 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V021B	FT025 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V022A	PT030 Root Isolation Valve	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V022B	PT034 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V023A	PT031 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V023B	PT035 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V024A	PT032 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V024B	PT036 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V025A	PT033 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V025B	PT037 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V027A	PORV Block Valve SG 01	B	I	ASME III-2	
SGS-PL-V027B	PORV Block Valve SG 02	B	I	ASME III-2	
SGS-PL-V030A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V030B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V031A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V031B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V032A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V032B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V033A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V033B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V034A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V034B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V035A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V035B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V036A	Steam Line Condensate Drain Isolation	B	I	ASME III-2	
SGS-PL-V036B	Steam Line Condensate Drain Isolation	B	I	ASME III-2	
SGS-PL-V038A	Steam Line #1 Nitrogen Supply Isolation	B	I	ASME III-2	
SGS-PL-V038B	Steam Line #2 Nitrogen Supply Isolation	B	I	ASME III-2	
SGS-PL-V040A	Main Steam Line Isolation	B	I	ASME III-2	
SGS-PL-V040B	Main Steam Line Isolation	B	I	ASME III-2	
SGS-PL-V042A	MSIV Bypass Control Isolation	B	I	ASME III-2	
SGS-PL-V042B	MSIV Bypass Control Isolation	B	I	ASME III-2	
SGS-PL-V043A	MSIV Bypass Control Isolation	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V043B	MSIV Bypass Control Isolation	C	I	ASME III-3	
SGS-PL-V045A	SG 1 Condensate Pipe Drain Valve	B	I	ASME III-2	
SGS-PL-V045B	SG 2 Condensate Pipe Drain Valve	B	I	ASME III-2	
SGS-PL-V046A	LT015 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V046B	LT017 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V047A	LT015, LT044 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V047B	LT017, LT046 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V048A	LT016 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V048B	LT018 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V049A	LT016, LT045 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V049B	LT018, LT047 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V050A	LT044 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V050B	LT046 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V051A	LT044 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V051B	LT046 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V052A	LT045 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V052B	LT047 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V053A	LT045 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V053B	LT047 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V056A	PT062 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V056B	PT063 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V057A	Main Feedwater Isolation	B	I	ASME III-2	
SGS-PL-V057B	Main Feedwater Isolation	B	I	ASME III-2	
SGS-PL-V058A	Main Feedwater Check	B	I	ASME III-2	
SGS-PL-V058B	Main Feedwater Check	B	I	ASME III-2	
SGS-PL-V062A	FT055A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V062B	FT056A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V063A	FT055A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V063B	FT056A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V064A	FT055B Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V064B	FT056B Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V065A	FT055B Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V065B	FT056B Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V066A	FT055 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V066B	FT056 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V067A	Startup Feedwater Isolation	B	I	ASME III-2	
SGS-PL-V067B	Startup Feedwater Isolation	B	I	ASME III-2	
SGS-PL-V068A	FT055 Root Isolation Valve	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V068B	FT056 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V074A	SG Blowdown Isolation	B	I	ASME III-2	
SGS-PL-V074B	SG Blowdown Isolation	B	I	ASME III-2	
SGS-PL-V075A	SG Series Blowdown Isolation	C	I	ASME III-3	
SGS-PL-V075B	SG Series Blowdown Isolation	C	I	ASME III-3	
SGS-PL-V076A	Blowdown Vent Line Isolation	C	I	ASME III-3	
SGS-PL-V076B	Blowdown Vent Line Isolation	C	I	ASME III-3	
SGS-PL-V084A	SG 1 Nitrogen Sparging Isolation	B	I	ASME III-2	
SGS-PL-V084B	SG 2 Nitrogen Sparging Isolation	B	I	ASME III-2	
SGS-PL-V086A	Steam Line Condensate Drain Control	C	I	ASME III-3	
SGS-PL-V086B	Steam Line Condensate Drain Control	C	I	ASME III-3	
SGS-PL-V093A	Orifice Isolation Valve	C	I	ASME III-3	
SGS-PL-V093B	Orifice Isolation Valve	C	I	ASME III-3	
SGS-PL-V094A	Orifice Cleanout Line Isolation Valve	C	I	ASME III-3	
SGS-PL-V094B	Orifice Cleanout Line Isolation Valve	C	I	ASME III-3	
SGS-PL-V095A	Orifice Isolation Valve	C	I	ASME III-3	
SGS-PL-V095B	Orifice Isolation Valve	C	I	ASME III-3	
SGS-PL-V096A	Steam Line Condensate Drain Level Isolation Valve	B	I	ASME III-2	
SGS-PL-V096B	Steam Line Condensate Drain Level Isolation Valve	B	I	ASME III-2	
SGS-PL-V097A	Steam Line Condensate Drain Level Isolation Valve	B	I	ASME III-2	
SGS-PL-V097B	Steam Line Condensate Drain Level Isolation Valve	B	I	ASME III-2	
SGS-PL-V100A	Startup Feedwater Drain Isolation Valve	C	I	ASME III-3	
SGS-PL-V100B	Startup Feedwater Drain Isolation Valve	C	I	ASME III-3	
SGS-PL-V101A	Main Feedwater Drain Isolation Valve	B	I	ASME III-2	
SGS-PL-V101B	Main Feedwater Drain Isolation Valve	B	I	ASME III-2	
SGS-PL-V102A	Startup Feedwater Vent Isolation Valve	C	I	ASME III-3	
SGS-PL-V102B	Startup Feedwater Vent Isolation Valve	C	I	ASME III-3	
SGS-PL-V103A	Main Feedwater Vent Isolation Valve	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V103B	Main Feedwater Vent Isolation Valve	B	I	ASME III-2	
SGS-PL-V104A	Main Feedwater Drain Isolation Valve	C	I	ASME III-3	
SGS-PL-V104B	Main Feedwater Drain Isolation Valve	C	I	ASME III-3	
SGS-PL-V233A	Power Operated Relief Valve	C	I	ASME III-3	
SGS-PL-V233B	Power Operated Relief Valve	C	I	ASME III-3	
SGS-PL-V240A	MSIV Bypass Isolation	B	I	ASME III-2	
SGS-PL-V240B	MSIV Bypass Isolation	B	I	ASME III-2	
SGS-PL-V250A	Main Feedwater Control	C	I	ASME III-3	
SGS-PL-V250B	Main Feedwater Control	C	I	ASME III-3	
SGS-PL-V255A	Startup Feedwater Control	C	I	ASME III-3	
SGS-PL-V255B	Startup Feedwater Control	C	I	ASME III-3	
SGS-PL-V256A	Startup Feedwater Check Valve	C	I	ASME III-3	
SGS-PL-V256B	Startup Feedwater Check Valve	C	I	ASME III-3	
SGS-PL-V257A	Main Feedwater Thermal Relief Valve	C	I	ASME III-3	
SGS-PL-V257B	Main Feedwater Thermal Relief Valve	C	I	ASME III-3	
SGS-PL-V258A	Startup Feedwater Thermal Relief Valve	C	I	ASME III-3	
SGS-PL-V258B	Startup Feedwater Thermal Relief Valve	C	I	ASME III-3	
SGS-PL-V300	SG1 NR LT001 Upper Isol	B	I	ASME III-2	
SGS-PL-V301	SG1 NR LT001 Lower Isol	B	I	ASME III-2	
SGS-PL-V302	SG1 NR LT002 Upper Isol	B	I	ASME III-2	
SGS-PL-V303	SG1 NR LT002 Lower Isol	B	I	ASME III-2	
SGS-PL-V304	SG1 NR LT003 Upper Isol	B	I	ASME III-2	
SGS-PL-V305	SG1 NR LT003 Lower Isol	B	I	ASME III-2	
SGS-PL-V306	SG1 NR LT004 Upper Isol	B	I	ASME III-2	
SGS-PL-V307	SG1 NR LT004 Lower Isol	B	I	ASME III-2	
SGS-PL-V308	SG1 NR LT011 Lower Isol	B	I	ASME III-2	
SGS-PL-V309	SG1 NR LT012 Lower Isol	B	I	ASME III-2	
SGS-PL-V310	SG1 NR LT015 Lower Isol	B	I	ASME III-2	
SGS-PL-V311	SG1 NR LT016 Lower Isol	B	I	ASME III-2	
SGS-PL-V312	SG1 Main Steam FT020 Isol	B	I	ASME III-2	
SGS-PL-V313	SG1 Main Steam FT021 Isol	B	I	ASME III-2	
SGS-PL-V314	SG1 Main Steam FT024 Isol	B	I	ASME III-2	
SGS-PL-V320	SG2 NR LT005 Upper Isol	B	I	ASME III-2	
SGS-PL-V321	SG2 NR LT005 Lower Isol	B	I	ASME III-2	
SGS-PL-V322	SG2 NR LT006 Upper Isol	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V323	SG2 NR LT006 Lower Isol	B	I	ASME III-2	
SGS-PL-V324	SG2 NR LT007 Upper Isol	B	I	ASME III-2	
SGS-PL-V325	SG2 NR LT007 Lower Isol	B	I	ASME III-2	
SGS-PL-V326	SG2 NR LT008 Upper Isol	B	I	ASME III-2	
SGS-PL-V327	SG2 NR LT008 Lower Isol	B	I	ASME III-2	
SGS-PL-V328	SG2 NR LT013 Lower Isol	B	I	ASME III-2	
SGS-PL-V329	SG2 NR LT014 Lower Isol	B	I	ASME III-2	
SGS-PL-V330	SG2 NR LT017 Lower Isol	B	I	ASME III-2	
SGS-PL-V331	SG2 NR LT018 Lower Isol	B	I	ASME III-2	
SGS-PL-V332	SG2 Main Steam FT022 Isol	B	I	ASME III-2	
SGS-PL-V333	SG2 Main Steam FT023 Isol	B	I	ASME III-2	
SGS-PL-V334	SG2 Main Steam FT025 Isol	B	I	ASME III-2	
SGS-PY-C01A	Main Steam Line A Penetration	B	I	ASME III, MC	
SGS-PY-C01B	Main Steam Line B Penetration	B	I	ASME III, MC	
SGS-PY-C02A	Main Feedwater Line A Penetration	B	I	ASME III, MC	
SGS-PY-C02B	Main Feedwater Line B Penetration	B	I	ASME III, MC	
SGS-PY-C03A	Steam Generator A Blowdown Line Penetration	B	I	ASME III, MC	
SGS-PY-C03B	Steam Generator B Blowdown Line Penetration	B	I	ASME III, MC	
SGS-PY-C05A	Startup Feedwater Line A Penetration	B	I	ASME III, MC	
SGS-PY-C05B	Startup Feedwater Line B Penetration	B	I	ASME III, MC	
SGS-PY-R02A	SG 1 Main Steam Line Drain Plug-Resistant Orifice	C	I	ASME III-3	
SGS-PY-R02B	SG 2 Main Steam Line Drain Plug-Resistant Orifice	C	I	ASME III-3	
SGS-PY-Y01A	Flexible Hose from SG 1 to Upper Tap LT001/LT011	B	I	ASME III-2	
SGS-PY-Y01B	Flexible Hose from SG 2 to Upper Tap LT005/LT013	B	I	ASME III-2	
SGS-PY-Y02A	Flexible Hose from SG 1 to Upper Tap LT001	B	I	ASME III-2	
SGS-PY-Y02B	Flexible Hose from SG 2 to Upper Tap LT005	B	I	ASME III-2	
SGS-PY-Y03A	Flexible Hose from SG 1 to Upper Tap LT002/LT015	B	I	ASME III-2	
SGS-PY-Y03B	Flexible Hose from SG 2 to Upper Tap LT006/LT014	B	I	ASME III-2	
SGS-PY-Y04A	Flexible Hose from SG 1 to Upper Tap LT002	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PY-Y04B	Flexible Hose from SG 2 to Upper Tap LT006	B	I	ASME III-2	
SGS-PY-Y05A	Flexible Hose from SG 1 to Upper Tap LT003/LT012/ LT044	B	I	ASME III-2	
SGS-PY-Y05B	Flexible Hose from SG 2 to Upper Tap LT007/LT017/ LT046	B	I	ASME III-2	
SGS-PY-Y06A	Flexible Hose from SG 1 to Upper Tap LT003	B	I	ASME III-2	
SGS-PY-Y06B	Flexible Hose from SG 2 to Upper Tap LT007	B	I	ASME III-2	
SGS-PY-Y07A	Flexible Hose from SG 1 to Upper Tap LT004/LT016/ LT045	B	I	ASME III-2	
SGS-PY-Y07B	Flexible Hose from SG 2 to Upper Tap LT008/LT018/ LT047	B	I	ASME III-2	
SGS-PY-Y08A	Flexible Hose from SG 1 to Upper Tap LT004	B	I	ASME III-2	
SGS-PY-Y08B	Flexible Hose from SG 2 to Upper Tap LT008	B	I	ASME III-2	
SGS-PY-Y09A	Flexible Hose from SG 1 to Upper Tap LT011	B	I	ASME III-2	
SGS-PY-Y09B	Flexible Hose from SG 2 to Upper Tap LT013	B	I	ASME III-2	
SGS-PY-Y10A	Flexible Hose from SG 1 to Upper Tap LT012	B	I	ASME III-2	
SGS-PY-Y10B	Flexible Hose from SG 2 to Upper Tap LT014	B	I	ASME III-2	
SGS-PY-Y11A	Flexible Hose from SG 1 to Upper Tap LT015/LT044	B	I	ASME III-2	
SGS-PY-Y11B	Flexible Hose from SG 2 to Upper Tap LT017/LT046	B	I	ASME III-2	
SGS-PY-Y12A	Flexible Hose from SG 1 to Upper Tap LT016/LT045	B	I	ASME III-2	
SGS-PY-Y12B	Flexible Hose from SG 2 to Upper Tap LT018/LT047	B	I	ASME III-2	
SGS-PY-Y13A	Flexible Hose from SG 1 to Lower Tap FT024	B	I	ASME III-2	
SGS-PY-Y13B	Flexible Hose from SG 2 to Lower Tap FT025	B	I	ASME III-2	
SGS-PY-Y14A	Flexible Hose from SG 1 to Lower Tap FT021	B	I	ASME III-2	
SGS-PY-Y14B	Flexible Hose from SG 2 to Lower Tap FT023	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PY-Y15A	Flexible Hose from SG 1 to Lower Tap FT020	B	I	ASME III-2	
SGS-PY-Y15B	Flexible Hose from SG 2 to Lower Tap FT022	B	I	ASME III-2	
SGS-PY-Y16A	Flexible Hose from Main Steam Line to Upper Tap FT020	B	I	ASME III-2	
SGS-PY-Y16B	Flexible Hose from Main Steam Line to Upper Tap FT022	B	I	ASME III-2	
SGS-PY-Y17A	Flexible Hose from Main Steam Line to Upper Tap FT021	B	I	ASME III-2	
SGS-PY-Y17B	Flexible Hose from Main Steam Line to Upper Tap FT023	B	I	ASME III-2	
SGS-PY-Y18A	Flexible Hose from Main Steam Line to Upper Tap FT024	B	I	ASME III-2	
SGS-PY-Y18B	Flexible Hose from Main Steam Line to Upper Tap FT025	B	I	ASME III-2	
SGS-PY-Y19A	Flexible Hose from Main Steam Line to PT030	B	I	ASME III-2	
SGS-PY-Y19B	Flexible Hose from Main Steam Line to PT034	B	I	ASME III-2	
SGS-PY-Y20A	Flexible Hose from Main Steam Line to PT032	B	I	ASME III-2	
SGS-PY-Y20B	Flexible Hose from Main Steam Line to PT036	B	I	ASME III-2	
Secondary Sampling System (SSS)				Location: Turbine Building	
System components are Class E					
Service Water System (SWS)				Location: Turbine Building and Yard	
n/a	Service Water Cooling Tower Fans	D	NS	Manufacturer Std.	
n/a	Service Water Cooling Tower	D	NS	Manufacturer Std.	
n/a	Service Water Pumps	D	NS	Hydraulic Institute Std.	
n/a	Valves Providing SWS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
Turbine Building Closed Cooling Water System (TCS)				Location: Turbine Building	
System components are Class E					

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Turbine Island Vents, Drains and Relief System (TDS)				Location: Turbine Building	
n/a	Piping and components that provide the path from the GSS and CMS to atmosphere and rad monitor	D	NS	ASME B31.1	
Balance of system components are Class E					
Main Turbine Control and Diagnostic System (TOS)				Location: Turbine Building	
System components are Class E					
Radiologically Controlled Area Ventilation System (VAS)			Location: Auxiliary Building and Annex Building		
n/a	CVS Pump Room Coolers	Note 2	NS	Manufacturer Std.	
n/a	RNS Pump Room Coolers	Note 2	II	Manufacturer Std.	
n/a	Valves Providing VAS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500-D	
n/a	Shutoff and Isolation Dampers	R	NS	ASME AG-1	
n/a	Fire Dampers	L	NS	UL-555	
n/a	Air Handling Units	L	NS	Manufacturer Std.	
n/a	Filters	L	NS	UL 900	
n/a	Ductwork	L	NS	SMACNA, Note 7	
n/a	Ductwork in Auxiliary Building except ductwork attached to mechanical modules	L	II	SMACNA, Note 7	Ductwork in radiation chemistry laboratory is nonseismic.
n/a	Ductwork	R	II	ASME AG-1	
n/a	Fans	L	NS	AMCA, Note 7	
Balance of system components are Class E or Class L					
Nuclear Island Nonradioactive Ventilation System (VBS)			Location: Shield Building, Auxiliary Building, and Annex Building		
n/a	Battery Rooms Exhaust Fans	Note 2	II	AMCA, Note 7	
n/a	PCS Room Heaters	Note 3	II	Manufacturer Std.	
n/a	Fire Dampers	Note 2, L	NS	UL-555	
n/a	Fire Dampers	Note 2, R	NS	ASME AG-1 and UL-555	
n/a	Combination Fire/Smoke Dampers	Note 2, L	NS	UL-555 and UL-555S	
n/a	Combination Fire/Smoke Dampers	Note 2, R	NS	ASME AG-1, UL-555, and UL-555S	
n/a	Dampers Providing AP1000 Equipment Class D Function	Note 2	NS	ANSI/AMCA-500-D	
n/a	Shutoff, Balancing, and Isolation Dampers	R	NS	ASME AG-1	
n/a	Shutoff and Balancing Dampers	L	NS	ANSI/AMCA-500-D	
VBS-MP-01A	Sample Pump A	C	I	Manufacturer Std.	
VBS-MP-01B	Sample Pump B	C	I	Manufacturer Std.	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Nuclear Island Nonradioactive Ventilation System (Continued)					
n/a	MCR/CSA Supplemental Air Filtration Units	Note 2	II	ASME AG-1, ASME N509, Note 4	
VBS-PL-V186	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V187	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V188	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V189	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V190	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V191	MCR Isolation Valve	C	I	ASME III-3	
n/a	Valves Providing VBS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
n/a	Other Air Handling Units	Note 2	NS	Manufacturer Std.	
n/a	Filters	Note 2	NS	UL 900	
n/a	Other Fans	Note 2, L	NS	AMCA, Note 7	
VBS-MA-10A	Ancillary Fan	D	II	ANSI/AMCA 210, 300	Equipment Anchorage is Seismic Category II
VBS-MA-10B	Ancillary Fan	D	II	ANSI/AMCA 210, 300	Equipment Anchorage is Seismic Category II
VBS-MA-11	Ancillary Fan	D	II	ANSI/AMCA 210, 300	Equipment Anchorage is Seismic Category II
VBS-MA-12	Ancillary Fan	D	II	ANSI/AMCA 210, 300	Equipment Anchorage is Seismic Category II
n/a	MCR Pressure Boundary Penetration Cast Sleeve	C	I	ANSI/AISC N690	CA52
n/a	Ductwork in Auxiliary Building	Note 2, L	II	SMACNA, Note 7	
n/a	Ductwork in Annex Building	Note 2, L	NS	SMACNA, Note 7	
n/a	Ductwork in Auxiliary Building	Note 2, R	II	ASME AG-1	
n/a	Ductwork in Annex Building	Note 2, R	NS	ASME AG-1	
VBS-MY-Y100	Silencer	D	II	Manufacturer Std.	Equipment Anchorage is Seismic Category II
VBS-MY-Y101	Silencer	D	II	Manufacturer Std.	Equipment Anchorage is Seismic Category II
VBS-MY-Y102	Static Air Mixer	Note 1	I	Manufacturer Std.	Inside Seismic Category II Supported Duct; Commercially dedicated to ASME AG-1
VBS-PL-V184A	VBS-PDT-032A Isolation Valve	C	I	ASME III-3	
VBS-PL-V184B	VBS-PDT-032B Isolation Valve	C	I	ASME III-3	
VBS-PL-V184C	VBS-PDT-032C Isolation Valve	C	I	ASME III-3	
Balance of system components are Class E or Class L					

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Containment Recirculation Cooling System (VCS)				Location: Containment	
n/a	Dampers	L	II	ANSI/AMCA-500-D	
n/a	Fan Coil Units	L	II	Manufacturer Std.	
n/a	Ductwork	L	II	SMACNA, Note 7	
n/a	Fans	L	II	AMCA, Note 7	
n/a	Fire Damper	L	II	UL-555	
Balance of system components are Class L					
Main Control Room Emergency Habitability System (VES)				Location: Auxiliary Building	
VES-MT-01	Emergency Air Storage Tank 01	C	I	ASME VIII, Appendix 22	
VES-MT-02	Emergency Air Storage Tank 02	C	I	ASME VIII, Appendix 22	
VES-MT-03	Emergency Air Storage Tank 03	C	I	ASME VIII, Appendix 22	
VES-MT-04	Emergency Air Storage Tank 04	C	I	ASME VIII, Appendix 22	
VES-MT-05	Emergency Air Storage Tank 05	C	I	ASME VIII, Appendix 22	
VES-MT-06	Emergency Air Storage Tank 06	C	I	ASME VIII, Appendix 22	
VES-MT-07	Emergency Air Storage Tank 07	C	I	ASME VIII, Appendix 22	
VES-MT-08	Emergency Air Storage Tank 08	C	I	ASME VIII, Appendix 22	
VES-MT-09	Emergency Air Storage Tank 09	C	I	ASME VIII, Appendix 22	
VES-MT-10	Emergency Air Storage Tank 10	C	I	ASME VIII, Appendix 22	
VES-MT-11	Emergency Air Storage Tank 11	C	I	ASME VIII, Appendix 22	
VES-MT-12	Emergency Air Storage Tank 12	C	I	ASME VIII, Appendix 22	
VES-MT-13	Emergency Air Storage Tank 13	C	I	ASME VIII, Appendix 22	
VES-MT-14	Emergency Air Storage Tank 14	C	I	ASME VIII, Appendix 22	
VES-MT-15	Emergency Air Storage Tank 15	C	I	ASME VIII, Appendix 22	
VES-MT-16	Emergency Air Storage Tank 16	C	I	ASME VIII, Appendix 22	
VES-MT-17	Emergency Air Storage Tank 17	C	I	ASME VIII, Appendix 22	
VES-MT-18	Emergency Air Storage Tank 18	C	I	ASME VIII, Appendix 22	
VES-MT-19	Emergency Air Storage Tank 19	C	I	ASME VIII, Appendix 22	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Main Control Room Emergency Habitability System (Continued)					
VES-MT-20	Emergency Air Storage Tank 20	C	I	ASME VIII, Appendix 22	
VES-MT-21	Emergency Air Storage Tank 21	C	I	ASME VIII, Appendix 22	
VES-MT-22	Emergency Air Storage Tank 22	C	I	ASME VIII, Appendix 22	
VES-MT-23	Emergency Air Storage Tank 23	C	I	ASME VIII, Appendix 22	
VES-MT-24	Emergency Air Storage Tank 24	C	I	ASME VIII, Appendix 22	
VES-MT-25	Emergency Air Storage Tank 25	C	I	ASME VIII, Appendix 22	
VES-MT-26	Emergency Air Storage Tank 26	C	I	ASME VIII, Appendix 22	
VES-MT-27	Emergency Air Storage Tank 27	C	I	ASME VIII, Appendix 22	
VES-MT-28	Emergency Air Storage Tank 28	C	I	ASME VIII, Appendix 22	
VES-MT-29	Emergency Air Storage Tank 29	C	I	ASME VIII, Appendix 22	
VES-MT-30	Emergency Air Storage Tank 30	C	I	ASME VIII, Appendix 22	
VES-MT-31	Emergency Air Storage Tank 31	C	I	ASME VIII, Appendix 22	
VES-MT-32	Emergency Air Storage Tank 32	C	I	ASME VIII, Appendix 22	
VES-PL-V001	Air Delivery Alternate Isolation Valve	C	I	ASME III-3	
VES-PL-V002A	Pressure Regulating Valve A	C	I	ASME III-3	
VES-PL-V002B	Pressure Regulating Valve B	C	I	ASME III-3	
VES-PL-V005A	Air Delivery Main Isolation Valve A	C	I	ASME III-3	
VES-PL-V005B	Air Delivery Main Isolation Valve B	C	I	ASME III-3	
VES-PL-V006A	Air Delivery Line Pressure Instrument Isolation Valve A	C	I	ASME III-3	
VES-PL-V006B	Air Delivery Line Pressure Instrument Isolation Valve B	C	I	ASME III-3	
VES-PL-V010A	Air Delivery Line Maintenance Isolation Valve A	C	I	ASME III-3	
VES-PL-V010B	Air Delivery Line Maintenance Isolation Valve B	C	I	ASME III-3	
VES-PL-V011A	Air Delivery Line Maintenance Isolation Valve A	C	I	ASME III-3	
VES-PL-V011B	Air Delivery Line Maintenance Isolation Valve B	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Main Control Room Emergency Habitability System (Continued)					
VES-PL-V016	Temporary Instrument Isolation Valve A	C	I	ASME III-3	
VES-PL-V018	Temporary Instrument Isolation Valve A	C	I	ASME III-3	
VES-PL-V019	Temporary Instrument Isolation Valve B	C	I	ASME III-3	
VES-PL-V020	Temporary Instrument Isolation Valve B	C	I	ASME III-3	
VES-PL-V022A	Pressure Relief Isolation Valve A	C	I	ASME III-3	
VES-PL-V022B	Pressure Relief Isolation Valve B	C	I	ASME III-3	
VES-PL-V024A	Air Bank 1 Isolation Valve A	C	I	ASME III-3	
VES-PL-V024B	Air Bank 2 Isolation Valve B	C	I	ASME III-3	
VES-PL-V024C	Air Bank 3 Isolation Valve C	C	I	ASME III-3	
VES-PL-V024D	Air Bank 4 Isolation Valve D	C	I	ASME III-3	
VES-PL-V025A	Air Bank 1 Isolation Valve A	C	I	ASME III-3	
VES-PL-V025B	Air Bank 2 Isolation Valve B	C	I	ASME III-3	
VES-PL-V025C	Air Bank 3 Isolation Valve C	C	I	ASME III-3	
VES-PL-V025D	Air Bank 4 Isolation Valve D	C	I	ASME III-3	
VES-PL-V026A	Air Bank 1 Fill/Vent Isolation Valve A	C	I	ASME III-3	
VES-PL-V026B	Air Bank 2 Fill/Vent Isolation Valve B	C	I	ASME III-3	
VES-PL-V026C	Air Bank 3 Fill/Vent Isolation Valve C	C	I	ASME III-3	
VES-PL-V026D	Air Bank 4 Fill/Vent Isolation Valve D	C	I	ASME III-3	
VES-PL-V040A	Air Tank Safety Relief Valve A	C	I	ASME III-3	
VES-PL-V040B	Air Tank Safety Relief Valve B	C	I	ASME III-3	
VES-PL-V040C	Air Tank Safety Relief Valve C	C	I	ASME III-3	
VES-PL-V040D	Air Tank Safety Relief Valve D	C	I	ASME III-3	
VES-PL-V043A	Differential Pressure Instrument Line Isolation Valve A	C	I	ASME III-3	
VES-PL-V043B	Differential Pressure Instrument Line Isolation Valve B	C	I	ASME III-3	
VES-PL-V044	Main Air Flowpath Isolation Valve	C	I	ASME III-3	
VES-PL-V045	Eductor Flow Path Isolation Valve	C	I	ASME III-3	
VES-PL-V046	Eductor Bypass Isolation Valve	C	I	ASME III-3	
VES-PY-N01	MCR Air Filtration Line Eductor	C	I	ASME III-3	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Main Control Room Emergency Habitability System (Continued)					
VES-PY-N02	Eductor Bypass Isolation Discharge Silencer	C	I	ASME III-3	
VES-PY-R02	Eductor Bypass Flow Orifice	C	I	ASME III-3	
VES-MY-F01	MCR Air Filtration Line Charcoal Filter	Note 1	I	ASME AG-1 Section FD	
VES-MY-F02	MCR Air Filtration Line HEPA Filter	Note 1	I	ASME AG-1 Section FC	
VES-MY-F03	MCR Air Filtration Line Postfilter	Note 1	1	ASME AG-1	
VES-MD-D001A	Relief Damper A	Note 1	I	ASME N509/N510	
VES-MD-D001B	Relief Damper B	Note 1	I	ASME N509/N510	
VES-MD-D002	MCR Air Filtration Line Supply Damper	Note 1	I	ASME AG-1 Section DA	
VES-MD-D003	MCR Air Filtration Line Supply Damper	Note 1	I	ASME AG-1 Section DA	
VES-MY-G001	Ducting Grill	Note 1	I	ASME N509/N510	
VES-MY-R001	Ducting Register	Note 1	I	ASME N509/N510	
VES-MY-R002	Ducting Register	Note 1	I	ASME N509/N510	
VES-MY-R003	Ducting Register	Note 1	I	ASME N509/N510	
VES-MY-Y01	MCR Air Filtration Line Silencer	Note 1	I	ASME AG-1 Section SA	
VES-MY-Y02	MCR Air Filtration Line Silencer	Note 1	I	ASME AG-1 Section SA	
Containment Air Filtration System (VFS)			Location: Containment, Auxiliary Building, and Annex Building		
VFS-PY-C01	Containment Supply Duct Penetration	B	I	ASME III, MC	
VFS-PY-C02	Containment Exhaust Duct Penetration	B	I	ASME III, MC	
VFS-MY-Y01	Containment Air Supply Debris Screen	C	I	ASME Sec. III Class 3	
VFS-MY-Y02	Containment Air Exhaust Debris Screen	C	I	ASME Sec. III Class 3	
VFS-PL-V003	Containment Purge Supply Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V004	Containment Purge Supply Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V006	Containment Isolation Test Connection	C	I	ASME III-3	
VFS-PL-V007	RCS Ejector Discharge Isolation	C	I	ASME III-3	
VFS-PL-V008	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V009	Containment Purge Discharge Containment Isolation Valve	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Containment Air Filtration System (Continued)					
VFS-PL-V010	Containment Purge Discharge Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V012	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V015	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V101	Containment Air Supply Line Test Connection	C	I	ASME III-3	
VFS-PL-V202	Containment Atmosphere to Filtration Units Isolation	C	I	ASME III-3	
VFS-PL-V587	Filtration Units to Containment Atmosphere Manual Isolation	C	I	ASME III-3	
VFS-PL-V800A	Vacuum Relief Containment Isolation A – ORC	B	I	ASME III-2	
VFS-PL-V800B	Vacuum Relief Containment Isolation B – ORC	B	I	ASME III-2	
VFS-PL-V803A	Vacuum Relief Containment Isolation Check Valve A – IRC	B	I	ASME III-2	
VFS-PL-V803B	Vacuum Relief Containment Isolation Check Valve B – IRC	B	I	ASME III-2	
n/a	Valves Providing VFS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
n/a	Shutoff, Balancing, Isolation, and Pressure Differential Control Dampers	R	NS	ASME AG-1	
n/a	Shutoff and Balancing Dampers	L	NS	ANSI/AMCA-500-D	
n/a	Fire Dampers	L	NS	UL-555	
n/a	Fire Damper	R	NS	ASME AG-1 and UL-555	
n/a	Supply Air Handling Units	L	NS	Manufacturer Std.	
n/a	Air Exhaust Filtration Units	R	NS	ASME AG-1, ASME N509, Note 4	
n/a	Other Fans	L	NS	AMCA, Note 7	
n/a	Ductwork in Auxiliary Building	L	II	SMACNA, Note 7	
n/a	Ductwork in Annex Building	L	NS	SMACNA, Note 7	
n/a	Ductwork in Containment	R	II	ASME AG-1	
n/a	Ductwork in Annex Building	R	NS	ASME AG-1, Note 4	
Balance of system components are Class E, L and R					
Health Physics and Hot Machine Shop HVAC System (VHS)				Location: Annex Building	
n/a	Shutoff, Isolation, Balancing, and Backdraft Dampers	L	NS	ANSI/AMCA-500-D	
n/a	Shutoff Dampers	R	NS	ASME AG-1	
n/a	Fire Dampers	L	NS	UL-555	
n/a	Air Handling Units w/ Filters	L	NS	Manufacturer Std.	
n/a	Ductwork	L	NS	SMACNA, Note 7	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Health Physics and Hot Machine Shop HVAC System (Continued)					
n/a	Ductwork	R	NS	ASME AG-1	
n/a	Fans	L	NS	AMCA, Note 7	
Balance of system components are Class E or Class L					
Containment Hydrogen Control System (VLS)				Location: Containment	
n/a	Hydrogen Igniters	D	NS	Manufacturer Std.	Provides Hydrogen Control Following Severe Accidents
VLS-MY-E01A	Catalytic Hydrogen Recombiner A	D	NS	Manufacturer Std.	
VLS-MY-E01B	Catalytic Hydrogen Recombiner B	D	NS	Manufacturer Std.	
Balance of system components are Class E or Class L					
Radwaste Building Ventilation System (VRS)				Location: Annex Building and Radwaste Building	
n/a	Shutoff, Isolation, Balancing, and Backdraft Dampers	L	NS	ANSI/AMCA-500-D	
n/a	Shutoff Dampers	R	NS	ASME AG-1	
n/a	Fire Damper	R	NS	ASME AG-1 & UL-555	
n/a	Air Handling Units	L	NS	Manufacturer Std.	
n/a	Filters	L	NS	UL 900	
n/a	Ductwork	L	NS	SMACNA, Note 7	
n/a	Ductwork	R	NS	ASME AG-1	
n/a	Fans	L	NS	AMCA, Note 7	
Balance of system components are Class E or Class L					
Turbine Building Ventilation System (VTS)				Location: Turbine Building	
n/a	Shutoff, Isolation, Balancing, and Backdraft Dampers	L	NS	ANSI/AMCA-500-D	
n/a	Fire Dampers	L	NS	UL-555	
n/a	Air Handling Units w/ Filters	L	NS	Manufacturer Std., UL-900	
n/a	Ductwork	L	NS	SMACNA, Note 7	
n/a	Fans	L	II	AMCA, Note 7	
n/a	Fire Damper	L	II	UL-555	
Balance of system components are Class E or Class L					
Containment Leak Rate Test System (VUS)			Location: Containment and Auxiliary Building		
VUS-PL-V013	Main Equipment Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V014	Maintenance Equipment Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V015	Main Equipment Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V016	Maintenance Equipment Hatch Test Connection	B	I	ASME III-2	

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Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Containment Leak Rate Test System (Continued)					
VUS-PL-V017	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V018	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V019	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V020	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V021	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V022	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V023	Fuel Transfer Tube Test Connection	B	I	ASME III-2	
VUS-PL-V140	Spare Penetration Test Connection	B	I	ASME III-2	
VUS-PL-V141	Spare Penetration Test Connection	B	I	ASME III-2	
VUS-PL-V142	Spare Penetration Test Connection	B	I	ASME III-2	
Balance of system components are Class E					
Central Chilled Water System (VWS)				Location: Various	
VWS-MS-02	Air-Cooled Chiller 2	D	NS	ARI/ASME VIII	
VWS-MS-03	Air-Cooled Chiller 3	D	NS	ARI/ASME VIII	
n/a	Pumps	D	NS	Manufacturer Std.	
n/a	Tanks	D	NS	ASME VIII	
n/a	Valves Providing VWS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
VWS-PY-C01	Containment Chilled Water Return Penetration	B	I	ASME III, MC	
VWS-PY-C02	Containment Chilled Water Supply Penetration	B	I	ASME III, MC	
VWS-PL-V053	VWS Containment Penetration Thermal Relief Valve	C	I	ASME III-3	
VWS-PL-V057	VWS Containment Penetration Thermal Relief Valve	C	I	ASME III-3	
VWS-PL-V058	Fan Coolers Supply Containment Isolation	B	I	ASME III-2	
VWS-PL-V062	Fan Coolers Supply Containment Isolation Check Valve	B	I	ASME III-2	
VWS-PL-V080	VWS Containment Isolation Relief Valve	B	I	ASME III-2	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Central Chilled Water System (Continued)					
VWS-PL-V082	Fan Coolers Return Containment Isolation	B	I	ASME III-2	
VWS-PL-V086	Fan Coolers Return Containment Isolation	B	I	ASME III-2	
VWS-PL-V424	Containment Penetration Test Connection	B	I	ASME III-2	
VWS-PL-V425	Containment Penetration Test Connection	B	I	ASME III-2	
Balance of system components are Class E					
Annex/Auxiliary Nonradioactive Ventilation System (VXS)		Location: Auxiliary Building and Annex Building			
n/a	Air Handling Unit Fans Providing AP1000 Equipment Class D Function	Note 2	NS	AMCA, Note 7	
n/a	Dampers Providing VXS AP1000 Equipment Class D Function	Note 2	NS	ANSI/AMCA-500-D	
n/a	Exhaust Fan Providing Ancillary Diesel Room Ventilation	Note 2	NS	AMCA, Note 7	
n/a	Fire Dampers	L	NS	UL-555	
n/a	Fire Dampers	R	NS	ASME AG-1 and UL-555	
n/a	Combination Fire/Smoke Dampers	L	NS	UL-555 and UL-555S	
n/a	Combination Fire/Smoke Dampers	R	NS	ASME AG-1, UL-555, and UL-555S	
n/a	Air Handling Units	L	NS	Manufacturer Std.	
n/a	Filters	L	NS	UL 900	
n/a	Ductwork	L	NS	SMACNA, Note 7	
n/a	Ductwork in Auxiliary Building except ductwork attached to mechanical modules	L	II	SMACNA, Note 7	
n/a	Ductwork	R	NS	ASME AG-1	
n/a	Fans	L	NS	AMCA, Note 7	
Balance of system components are Class E or Class L					
Hot Water Heating System (VYS)				Location: Various	
System components are Class E					

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Diesel Generator Building Ventilation System (VZS)				Location: Diesel Generator Building	
n/a	Unit Heaters Providing AP1000 Equipment Class D Function	Note 3	NS	UL-2021; NFPA 70	
n/a	Fans Providing AP1000 Equipment Class D Function	Note 2	NS	AMCA, Note 7	
n/a	Dampers Providing VZS AP1000 Equipment Class D Function	Note 2	NS	ANSI/AMCA 500-D	
n/a	Fire Dampers	L	NS	UL-555	
n/a	Air Handling Units	L	NS	Manufacturer Std.	
n/a	Filters	L	NS	UL 900	
n/a	Ductwork	L	NS	SMACNA, Note 7	
n/a	Fans	L	II	AMCA, Note 7	
n/a	Fire Damper	L	II	UL-555	
Balance of system components are Class E or Class L					
Gaseous Radwaste System (WGS)				Location: Auxiliary Building	
n/a	Gas Cooler	D	NS	Manufacturer Std.	
n/a	Sample Pumps	D	NS	Manufacturer Std.	
n/a	Guard and Delay Beds	D	NS	ASME VIII	Design for 1/2 SSE
n/a	Moisture Separator	D	NS	ASME VIII	
n/a	Valves Providing WGS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
Liquid Radwaste System (WLS)			Location: Containment, Auxiliary, and Radwaste Buildings		
n/a	Heat Exchangers, WLS and CCS Side	D	NS	ASME VIII/ TEMA	
n/a	Pumps	D	NS	Manufacturer Std.	
n/a	Tanks (except WLS-MT-13A, WLS-MT-13B, WLS-MT-17, WLS-MT-23A, WLS-MT-23B	D	NS	ASME III without Code Stamp	
WLS-MT-13A	Waste Holdup Tank A Chemical Addition Pot	D	NS	ASME VIII-1	
WLS-MT-13B	Waste Holdup Tank B Chemical Addition Pot	D	NS	ASME-VIII-1	
WLS-MT-17	Chemical Waste Tank Chemical Addition Pot	D	NS	ASME VIII-1	
WLS-MT-23A	WLS Leak Chase Collection Pot A	D	NS	ASME VIII-1	
WLS-MT-23B	WLS Leak Chase Collection Pot B	D	NS	ASME VIII-1	
n/a	Degasifier	D	NS	ASME VIII	
n/a	Ion Exchangers	D	NS	ASME VIII	
n/a	Filters	D	NS	ASME VIII	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Liquid Radwaste System (Continued)					
n/a	Valves Providing WLS AP1000 Equipment Class D Function (local drain valves in Radwaste Building)	D	NS	ANSI B16.34	
n/a	Floor Drain Hubs	D	NS	Manufacturer Std.	
WLS-MT-02	Containment Sump	D	NS	ACI 349	ACI 349 Evaluation of Structural Boundary Only
WLS-PL-V055	Sump Discharge Containment Isolation IRC	B	I	ASME III-2	
WLS-PL-V057	Sump Discharge Containment Isolation ORC	B	I	ASME III-2	
WLS-PL-V058	WLS Containment Isolation Relief Valve	B	I	ASME III-2	
WLS-PL-V067	RCDT Gas Outlet Containment Isolation IRC	B	I	ASME III-2	
WLS-PL-V068	RCDT Gas Outlet Containment Isolation ORC	B	I	ASME III-2	
WLS-PL-V071A	CVS Compartment to Sump	C	I	ASME III-3	
WLS-PL-V071B	PXS A Compartment to Sump	C	I	ASME III-3	
WLS-PL-V071C	PXS B Compartment to Sump	C	I	ASME III-3	
WLS-PL-V072A	CVS Compartment to Sump	C	I	ASME III-3	
WLS-PL-V072B	PXS A Compartment to Sump	C	I	ASME III-3	
WLS-PL-V072C	PXS B Compartment to Sump	C	I	ASME III-3	
WLS-PY-C02	Reactor Coolant Drain Tank WLS Connection Penetration	B	I	ASME III, MC	
WLS-PY-C03	Containment Sump Pumps Combined Discharge Penetration	B	I	ASME III, MC	
WLS-MY-Y34	Containment Sump Level Instrument Stilling Well	D	I	Manufacturer Std.	
WLS-MY-Y35	Containment Sump Level Instrument Stilling Well	D	I	Manufacturer Std.	
WLS-MY-Y36	Containment Sump Level Instrument Stilling Well	D	I	Manufacturer Std.	
Balance of system components are Class E					
Radioactive Waste Drain System (WRS)				Location: Auxiliary Building	
n/a	Pumps	D	NS	Manufacturer Std.	
n/a	Valves Providing WRS AP1000 Equipment Class D Function	D	NS	ANSI B16.34	
n/a	Floor Drain Hubs	D	NS	Manufacturer Std.	
WRS-MT-02A	WRS Leak Chase Collection Pot A	D	NS	ASME VIII-1	
WRS-MT-02B	WRS Leak Chase Collection Pot B	D	NS	ASME VIII-1	

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AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category, Note 6	Principal Construction Code	Comments
Solid Radwaste System (WSS)				Location: Auxiliary Building	
n/a	Pumps	D	NS	Manufacturer Std.	
n/a	Tanks	D	NS	ASME VIII	
n/a	Filters	D	NS	ASME VIII	
n/a	Valves Providing WSS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
Balance of system components are Class E					
Waste Water System (WWS)				Location: Various	
WWS-PL-V506	MCR WWS Isolation Valve	C	I	ASME III-3	
Balance of system components are Class E					
Onsite Standby Power System (ZOS)				Location: Diesel Generator Building	
n/a	Diesel Generator Engines	D	NS	Manufacturer Std.	
n/a	Diesel Generator Starting Units	D	NS	CAGI	
n/a	Diesel Generator Radiators	D	NS	API 661	
n/a	Diesel Generator Silencers	D	NS	Manufacturer Std.	
n/a	Valves Providing ZOS Diesel Generator Engines AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
Balance of system components are Class E					

Notes:
1. Component performs a safety-related function equivalent to AP1000 equipment Class C. The component is constructed using the standards for Class R and a quality assurance program in conformance with 10 CFR Part 50 Appendix B.
2. Component performs an AP1000 equipment Class D function and is constructed using the standards for Class L or Class R.
3. Component performs an AP1000 Equipment Class D function and is constructed using the standards for Class E.
4. Construction is non-seismic and meets applicable portions of ASME AG-1 consistent with RG 1.140.
5. Not used.
6. The classification of NS for structures, systems and components is upgraded to seismic Category II if the criterion in Subsection 3.2.1.1.3 is met.
7. For specific SMACNA and AMCA standards, see the descriptions in Section 9.4.

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3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

The wind loadings for seismic Category I structures are in accordance with American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7-98 (Reference 1).

3.3.1.1 Design Wind Velocity

The design wind is specified as a basic wind speed of 145 mph with an annual probability of occurrence of 0.02 based on the most severe location identified in Reference 1 This wind speed is the 3 second gust speed at 33 feet above the ground in open terrain (Reference 1, exposure C). The basic wind speed of 145 mph is the 3 second gust speed that has become the basis of wind design codes since 1995. It corresponds to the 110 mph fastest mile wind used as the basis for the AP600 design in accordance with the 1988 edition of Reference 1.

Higher winds with a probability of occurrence of 0.01 are used in the design of seismic Category I structures by using an importance factor of 1.15. This is obtained by classifying the AP1000 seismic Category I structures as essential facilities and using the design provisions for Category IV of Reference 1.

Velocity pressure exposure coefficients and gust response factors are calculated according to Reference 1 for exposure C, which is applicable to shorelines in hurricane prone areas in the 1998 edition of Reference 1. The topographic factor is taken as unity.

The design wind loads calculated as described above exceed those required at other locations in the United States, where the more severe Exposure Category D is specified in Reference 1. Exposure Category D is applicable for sites near the open inland waterways, the Great Lakes, and the coastal areas of California, Oregon, Washington, and Alaska. For such locations, the basic wind speed is less than 130 mph.

3.3.1.2 Determination of Applied Forces

The procedures used in transforming the wind velocity into an effective pressure to be applied to structures and parts and portions of structures follow the guidelines of Reference 1.

Effective pressures applied to interior and exterior surfaces of the buildings and corresponding shape coefficients are calculated according to Reference 1 for exposure C. Shape coefficients, defining the variation around the circumference of the shield building, are calculated using ASCE Paper No. 3269 (Reference 2). These shape coefficients are consistent with those observed in the model tests described in Reference 6.

3.3.2 Tornado Loadings

Seismic Category I structures are designed to resist tornado wind loads without exceeding the allowable stresses defined in Subsection 3.8.4. These tornado loads exceed the loads for hurricanes with a probability of occurrence comparable to that of the tornado. In addition, seismic Category I structures are designed to remain functional when subjected to tornado-generated missiles as discussed in Subsection 3.5.1.4. Seismic Category I structures are permitted to sustain local missile damage such as partial penetration and local cracking or permanent deformation or both, provided that structural integrity is maintained and seismic Category I systems, components, and equipment required to function during or after passage of a tornado are not subject to damage by secondary missiles, such as from concrete spalling. See Subsection 3.5.2.

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3.3.2.1 Applicable Design Parameters

The design parameters applicable to the design basis tornado are as follows:

- Maximum wind speed – 300 mph
- Maximum rotational speed – 240 mph
- Maximum translational speed – 60 mph
- Radius of maximum rotational wind from center of tornado – 150 ft
- Atmospheric pressure drop – 2.0 psi
- Rate of pressure change – 1.2 psi/sec

It is estimated that the probability of wind speeds greater than the design basis tornado is between 10^{-6} and 10^{-7} per year for an AP1000 at a "worst location" anywhere within the contiguous United States.

3.3.2.2 Determination of Forces on Structures

The procedures described in Subsection 3.3.1.2 are used to transform the tornado wind loading and differential pressure loading into effective loads on structures, with a wind velocity of 300 mph (translational plus rotational velocities). The dynamic wind pressure is applied to the structures in the same manner as the wind loads described in Subsection 3.3.1.2, except that the importance factor, gust factor, and the variation of wind speed with height do not apply. Loading combinations and load factors used are as follows:

$W_t = W_w$

$W_t = W_p$

$W_t = W_m$

$W_t = W_w + 0.5\ W_p$

$W_t = W_w + W_m$

$W_t = W_w + 0.5\ W_p + W_m$

where:

W_t = total tornado load

W_w = total wind load

W_p = total differential pressure load

W_m = total missile load

The maximum pressure drop of 2.0 psi, applicable to a nonvented structure, is used for W_p for all structures except the upper portion of the shield building. The portion of the shield building surrounding the upper annulus is designed as fully vented (zero differential pressure) due to the large area of the air inlets and discharge stack. Figure 3.3-1 shows the velocity pressure variation with the radius from the center of the tornado. When the tornado loading includes the missile load, the structure locally may go into the plastic range because of missile impact. Subsection 3.5.3 discusses the procedure for analyzing local missile effects.

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3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

The failure of structures not designed for tornado loadings does not affect the capability of seismic Category I structures or safety-related systems performance. This is accomplished by one of the following:

- Designing the adjacent structure to seismic Category I structure tornado loading
- Investigating the effect of adjacent structure failure on seismic Category I structures to determine that no impairment of function results
- Designing a structural barrier to protect seismic Category I structures from adjacent structural failure.

The structures adjacent to the nuclear island are the annex building, the radwaste building, and the turbine building.

The portions of the annex and turbine buildings adjacent to the nuclear island are classified as seismic Category II and are designed to seismic Category I structure tornado loading. The acceptance criteria are based on ACI 349 for concrete structures, on ANSI/AISC N690 for steel structures, including the supplemental requirements described in Subsections 3.8.4.4.1 and 3.8.4.5, and on AWS D1.1-2000 for weld design, qualification, fabrication, and inspection as described in Subsections 3.8.3.2 and 3.8.4.2. The structures are constructed to the same requirements as nonseismic structures, ACI 318 for concrete structures, and AISC Specification for Structural Steel Buildings, Allowable Stress Design and Plastic Design — June 1989 for steel structures. Siding is permitted to blow off during the tornado.

The radwaste building is a small steel-frame building. If it were to collapse in the tornado, it would not impair the integrity of the reinforced concrete nuclear island.

The main area of the turbine building is classified as nonseismic and is designed to seismic Category I structure tornado loading. The acceptance criteria for tornado loading are based on ACI 318 for concrete structures using a load factor of 1.0 and on 1.7 times the AISC 360 allowables for steel structures. Siding is permitted to blow off during the tornado.

3.3.2.4 Tornado Loads on the Passive Containment Cooling System Air Baffle

The containment air baffle is located within the annulus between the containment vessel and the shield building. It interfaces with the passive containment cooling system and separates downward flowing air entering at the air intake openings at the top of the cylindrical portion of the shield building from upward flowing air that cools the containment vessel and flows out of the discharge diffuser.

Loads due to the atmospheric pressure drop (W_p) are calculated assuming the tornado is centered over the containment. Differential pressure between the air intakes and the discharge is calculated based on the radius of the shield building and the parameters of the tornado defined in Subsection 3.3.2.1. The differential pressure is used with the pressure loss coefficients in the air flow path to determine pressures throughout the flow path.

The development of loads on the air baffle due to the design wind and tornado (W_w) are described in the test reports (References 3, 4, and 5). Models of the AP600 were tested in a wind tunnel and subjected to representative wind profiles. Pressures were measured on each side of the baffle, and the differential pressures were normalized to the input wind velocity. The pressure coefficients are applied to the effective dynamic pressure for the design wind and the tornado to obtain the wind

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loads across the baffle. The tornado wind is specified to be constant with height. The tornado loads calculated for the AP600 are applicable to the AP1000. The AP1000 configuration is similar to the AP600. The height of the shield building roof increases by 20′ 6"; the exterior diameter of the passive containment cooling storage tank increases from 80′ 0" to 89′ 0". The pressure coefficients measured in the AP600 tests are not significantly affected by these changes in geometry.

Wind conditions result in a pressure reduction in the annulus between the shield building and the containment vessel as well as above the containment dome. This reduced pressure is equivalent to an increase in containment internal pressure and is within the normal operating range for containment pressure (-0.2 to 1.0 psig).

Wind conditions result in a small wind load across the containment vessel. This is maximum opposite the air intakes where positive pressures occur on the windward side and negative pressures occur on the leeward side. Lateral loads on the containment vessel are developed in Reference 5.

3.3.3 Combined License Information

The Combined License information requested in this subsection has been partially addressed in APP-GW-GLR-020 (References 7). The information item completion activities required of the Combined License applicant are defined in APP-GW-GLR-020. These activities include development of site-specific parameters and verification of bounding conditions, site arrangement, and building construction. Specifically, the information that the applicant must evaluate is the following, demonstrating that any exceedances or differences do not compromise the safety of the plant:

- Show site parameters for wind and tornado showing that the site satisfies the AP1000 site interface criteria for the wind and tornado conditions. If there are exceedances, they must be discussed and shown acceptable.

- Provide plant-specific site plan and discuss any differences between the plant-specific site plan and the typical site plan shown in Figure 1.2-2 of Section 1.2.

- Verify that there are no other structures adjacent to the nuclear island other than: turbine building, annex building, radwaste building, and passive containment cooling ancillary water storage tank. If there are, then the effect of these structures on the AP1000 nuclear island must be discussed.

- Show that missiles caused by external events separate from the tornado have energies less than the tornado missile spectrum energies that the AP1000 is designed to withstand. If missile energy is greater, evaluate and show that it will not compromise the safety of AP1000 safety-related structures and components.

3.3.4 References

1. American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7-98.

2. ASCE Paper No. 3269, "Wind Forces on Structures," Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961).

3. WCAP-13323-P and WCAP-13324-NP, "Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor," August 1992.

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4. WCAP-14068-P, "Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor," May, 1994.

5. WCAP-14169-P, "Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report," September, 1994.

6. WCAP-13294-P and WCAP-13295-NP, "Phase I Wind Tunnel Testing for the Westinghouse AP600 Reactor," April 1992.

7. APP-GW-GLR-020, "Wind and Tornado Site Interface Criteria," Westinghouse Electric Company LLC.

8. American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7-05.

9. AISC 360, "Specification for Structural Steel Buildings," March 9, 2005.

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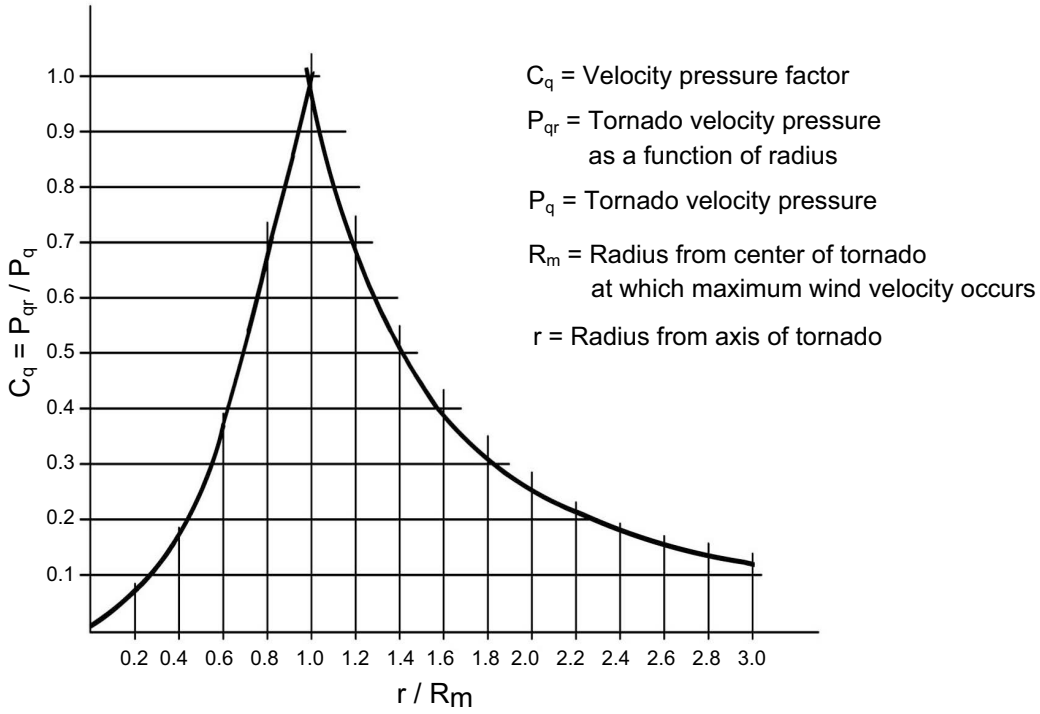

Figure 3.3-1
Velocity Pressure Variation with Radius from Center of Tornado

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3.4 Water Level (Flood) Design

External flooding of a nuclear power plant from natural causes can be attributed to probable maximum flood, site and adjacent area probable maximum precipitation runoff, seiche, and ground water. Criteria for the design basis flood are in accordance with the provisions of Regulatory Guide 1.59, Design Basis Floods for Nuclear Power Plants, and Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants. Conformance with the Regulatory Guides is described in Section 1.9. External events are described in Section 2.4. Chapter 2 provides interface data for AP1000 which has an interface flood level at plant grade.

Internal plant flooding can be attributed to piping ruptures, tank failures, or the actuation of fire suppression systems.

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Seismic Category I Structures, Systems, and Components

The seismic Category I structures, systems, and components identified in Section 3.2 are designed to withstand the effects of flooding due to natural phenomena or postulated component failures. A description of the structures is provided in Subsections 3.8.2. An alternate waterproofing system for the seismic Category I structures below grade is as presented in Subsection 3.8.5.1.1. and 3.8.4. None of the nonsafety-related structures, systems and components were found to be important based on flooding considerations. As a result, nonsafety-related structures, systems and components are not important in mitigation of flood events and are not required to be protected from either internal or external flooding.

3.4.1.1.1 Protection from External Flooding

The probable maximum flood for the AP1000 has been established at less than plant elevation 100′ as discussed previously in Section 2.4. The probable maximum flood results from site specific events, such as river flooding, upstream dam failure, or other natural causes.

Flooding does not occur from the probable maximum precipitation. The roofs do not have internal roof drains. The annex, radwaste, and diesel generator buildings have parapets with large openings to drain to scuppers/drains to preclude accumulation of water on the roofs. The roofs are sloped such that rainfall is directed towards gutters located along the edges of the roofs. Therefore, ponding of water on the roofs is precluded. Water from roof drains and/or scuppers, as well as runoff from the plant site and adjacent areas, is conveyed to catch basins, underground pipes, or directly to open ditches by sloping the tributary surface area. The site is graded to offer protection to the seismic Category I structures.

The high ground water table interface is at two feet below the grade elevation, as discussed in Section 2.4.

The components that may be potential sources for external flooding are nonsafety-related, nonseismic tanks as shown in Figure 1.2-2:

- Fire water tanks as described in Subsection 9.5.1. These two tanks have maximum usable volumes of approximately 504,000 gallons each, and are located at the north end of the turbine building. Water will drain from the tanks away from the nuclear island and adjacent buildings due to the required site grading.

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- Condensate storage tank as described in Subsection 9.2.4. This tank has a volume of approximately 656,000 gallons, and is located at the west side of the turbine building. Water will drain from the tank away from the turbine and auxiliary buildings due to site grading.

- Demineralized water tank as described in Subsection 9.2.4. This tank has a volume of 126,000 gallons and is located adjacent to the annex building at elevation 107′-2″. Water will drain from the tank away from the annex building to elevation 100′-0″. Nearby doors lead to areas in the annex building which do not contain safety-related components or systems.

- Boric acid storage tank as described in Subsection 9.3.6. This tank has a volume of 79,315 gallons and is located adjacent to the demineralized water storage tank.

- Diesel fuel oil tanks as described in Subsection 9.5.4. These two tanks have volumes of 94,000 gallons each. They are located remote from safety-related structures and are provided with dikes to retain leaks and spills.

- Passive containment cooling ancillary water storage tank, which is seismic category II, as described in Subsection 6.2.2.2.3. This tank has a volume of 1,080,027 gallons and is located at the west side of the auxiliary building. Water will drain from the tank away from the auxiliary building due to site grading.

In addition, failure of the cooling tower or the service water or circulating water piping under the yard could result in a potential flood source. However, these potential sources are located far from safety-related structures and the consequences of a failure in the yard would be enveloped by the analysis described in Subsection 10.4.5.

For the AP1000, the 100′-0″ building floor elevations are slightly above the grade elevation. In addition, the slope of the yard grade directs water away from the buildings. Because the probable maximum flood for AP1000 is less than grade elevation, the exterior doors are not required to be watertight for protection from external flooding.

Process piping penetrations through the exterior walls of the nuclear island below grade are embedded in the wall or are welded to a steel sleeve embedded in the wall. There are no access openings or tunnels penetrating the exterior walls of the nuclear island below grade.

The reinforced concrete seismic Category I structures, incorporating the waterproofing and sealing features described above and in Subsection 3.4.1.1.1.1, provide hardened protection for safety-related structures, systems, and components as defined in Regulatory Guide 1.59.

3.4.1.1.1.1 Waterproofing

A waterproof membrane or waterproofing system for the seismic Category I structures below grade will be installed as an architectural aide to limit the infiltration of subsurface water. The COL applicant will use a waterproofing system for foundation mat (mudmat) and the below grade exterior walls exposed to flood and groundwater that will demonstrate a friction coefficient ≥ 0.55 with all horizontal concrete surfaces. This friction coefficient is maintained for the life expectancy of the plant and will not introduce a horizontal slip plane increasing the potential for movement during an earthquake (see Subsection 3.8.5.5.3). Typical waterproofing approaches are described as follows:

- HDPE Double-Sided Textured Waterproof Membrane

  Figures 3.4-1and 3.4-2 show the typical application of this waterproofing approach for a mechanically stabilized earth (MSE) wall and for a step-back configuration.

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- HDPE Single-Sided Self-Adhering Sheet Waterproofing Membrane

  The HDPE single-sided adhesive sheet membrane is interchangeable with the HDPE double-sided textured membrane for use in the mudmat, and it may be used in certain circumstances to waterproof the walls.

- Self-adhesive, Rubberized Asphalt/Polyethylene Waterproofing Membrane

  The self-adhesive rubberized membrane is for application to waterproof the walls only.

- Sprayed-on Waterproofing Membrane

  This method may be used either for soil sites, in conjunction with an MSE wall, or for rock sites, where an open excavation may be used. The membrane consists of 100-percent solids materials based on polymer-modified asphalt or polyurea. This system may include a polyester reinforcement fabric having properties necessary to meet the performance requirements for the system. Figure 3.4-4 shows the typical installation using MSE walls with the sprayed-on, liquid-applied waterproofing membrane placed on the MSE wall panels and between the two layers of the mudmat. Where the vertical face of excavation is used as a form for the exterior walls, the waterproof membrane is installed on the vertical face of the excavation prior to placement of concrete in the exterior walls. An alternate waterproofing system for the seismic Category I structures below grade is as presented in Subsection 3.8.5.1.1.

- HDPE Waterproof Membrane System

  The HDPE waterproof membrane system uses an embedded HDPE Liner, with additional rubberized waterproof sealant materials forming a transition to the waterproof membrane on the MSE walls. It may be used for vertical waterproofing applications on the NI wall.

The waterproof function of the membrane is not safety-related; however, the membrane between the mudmats must provide adequate shear strength to transfer horizontal shear forces due to seismic (SSE) loading. This function is seismic Category I. The waterproof membrane will have physical properties, including surface and texture, to achieve the required coefficient of friction. Primer or geotextile may be added as required.

3.4.1.1.2 Protection from Internal Flooding

The nuclear island general arrangement drawings provided in Section 1.2 are a useful reference for the internal flooding discussion.

The AP1000 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components. As a result, component failures resulting from internal flooding do not prevent safe shutdown of the plant or prevent mitigation of the flooding event. Protection mechanisms are generally described in Subsection 3.6.1.3.2. The protection mechanisms related to minimizing the consequences of internal flooding include the following:

- Structural enclosures
- Structural barriers
- Curbs, weirs and elevated thresholds
- Watertight hatches
- Leak detection systems
- Drain systems
- Flood relief louvers

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- Pressure relief panels
- Flappers

The AP1000 minimizes the number of penetrations through enclosure or barrier walls below the flood level. Those few penetrations through flood protection walls that are below the maximum flood level are watertight. Any process piping penetrating below the maximum flood level either is embedded in the wall or floor or is welded to a steel sleeve embedded in the wall or floor. There are no watertight doors in the AP1000 used for internal flood protection because, as described in Subsection 3.4.1.2.2, they are not needed to protect safe shutdown components from the effects of internal flooding. Watertight hatches are provided to close openings not required for hydrogen venting. The walls, floors, hatches, and penetrations are designed to withstand the maximum anticipated hydrostatic loads associated with a pipe failure as described in Section 3.6. The two watertight doors on the waste holdup tank compartments limit the consequence of a failure on spent fuel pool water level.

3.4.1.2 Evaluation of Flooding Events

3.4.1.2.1 External Flooding

Base mat and exterior walls of seismic Category I structures are designed to resist upward and lateral pressures caused by the probable maximum flood and high ground water level. The vertical hydrostatic pressure acting uniformly at the bottom of the base mat is the product of the height to the high water level and the unit weight of water assumed as 62.4 lb/ft^3. The horizontal hydrostatic pressure acting on the exterior walls varies with height, with the maximum at the bottom of the wall and zero at the maximum water level. Minimum factors of safety for overturning, sliding, and flotation are described in Subsection 3.8.5. There are no dynamic water forces associated with the probable maximum flood or high ground water level because they are below the finished grade. Dynamic forces associated with the probable maximum precipitation are not factors in the analysis or design since the finished grade is adequately sloped.

There are no safety-related hydraulic structures for AP1000.

3.4.1.2.2 Internal Flooding

This section describes the consequences of compartment flooding for various postulated component failures. The equipment required to achieve and maintain safe shutdown depends on the initiating event. The safety-related systems and components available for safe shutdown are described in Section 7.4. This equipment is located in the auxiliary building and inside containment. Except for floor drains, no credit is taken in this evaluation for the availability of nonsafety-related systems or components.

Each area of the plant containing safety-related systems or equipment is reviewed to determine the postulated fluid system failures which would result in the most adverse internal flooding conditions. For the internal flooding analysis, the failure of safety-related systems, structures or components is acceptable provided they have no safe shutdown function or the safe shutdown function is otherwise accomplished. The internal flooding analysis shows that systems, structures, and components are not prevented from performing their required safe shutdown functions due to the effects of the postulated failure. In addition, the analysis identifies the protection features that mitigate the consequences of flooding in an area that contains safety-related equipment.

The flooding sources considered in the analysis consist of the following:

- High-energy piping (breaks and cracks)
- Through-wall cracks in seismically-supported moderate energy piping
- Breaks and through-wall cracks in non-seismically-supported moderate energy piping

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- Pump mechanical seal failures
- Storage tank ruptures
- Actuation of fire suppression systems
- Flow from upper elevations, lower elevations and adjacent areas

The analysis is performed based on the criteria and assumptions provided in Section 3.6 and ANS-56.11 (Reference 1). Section 3.6 provides the criteria used to define break and crack locations and configurations for high and moderate-energy piping failures. Additional design criteria pertaining to the internal flooding analysis are provided in this section.

The analysis consists of the following steps:

- Identification of the flood sources
- Identification of essential equipment in area
- Determination of flowrates and flood levels
- Evaluation of effects on essential equipment

As stated in Section 3.6, high-energy ASME Code Class 1, 2, and 3 piping of 6 inch nominal diameter or larger inside the containment is evaluated for mechanistic pipe break (leak-before-break) for AP1000. Those high-energy piping systems that do not satisfy the mechanistic pipe break requirements inside containment and high-energy lines outside containment are evaluated for non-mechanistic breaks and cracks, as above.

Fluid flowrates from high- and moderate-energy piping ruptures are determined based on the criteria provided in Section 3.6 and ANSI 56.11 (Reference 1). Fluid flowrates through stairwells, floor openings, and floor sleeves are determined in accordance with the formulas given in Reference 1.

No breaks or cracks are assumed for piping with nominal diameters of 1 inch or less. For each storage tank rupture, it is assumed that the entire tank inventory is drained.

The analysis of potential flooding events is performed on a floor-by-floor and room-by-room basis depending upon the relative location of safety-related equipment. No credit is taken for operation of sump pumps to mitigate the consequences of flooding.

3.4.1.2.2.1 Containment Flooding Events

General

The safe shutdown systems and components located inside the containment are associated with the passive core cooling system (PXS), the automatic depressurization system (ADS), and containment isolation.

The evaluation of containment flooding events addresses the impact of flooding on the safe shutdown systems and components. The AP1000 passive core cooling system, the internal containment compartments, and the equipment locations are designed for internal flooding to maintain post accident long-term cooling flow to the reactor core from the flooded volumes.

In the unlikely event of a loss-of-coolant accident (LOCA), the combined water inventory from available sources within the containment is sufficient to flood the reactor and steam generator compartments to a level above the reactor coolant system piping to provide water flow back into the reactor coolant system via the break location or via the passive core cooling system containment recirculation subsystem (see Section 6.3) flow path.

The potential for flooding safe shutdown components inside containment that would be required to perform safe shutdown functions is limited to two equipment compartments. These compartments

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are located in the southeast and northeast quadrants of the containment below the floor at elevation 107′-2″. For flood evaluation, these compartments extend up to the top of the curbs through the openings in the floor. These two compartments contain passive core cooling system components that provide two redundant means for delivering borated water to the reactor coolant system when required for safe shutdown.

The two passive core cooling system compartments primarily contain passive core cooling system components. The southeast compartment is referred to as the PXS-A compartment. The northeast compartment is interconnected with the RNS valve room by a passageway. These two compartments are treated, in this discussion, as one floodable volume referred to as the PXS-B compartment. The principal passive core cooling system component in each passive core cooling system compartment is an accumulator. A passive core cooling system core makeup tank is located above each passive core cooling system compartment. Each passive core cooling system compartment also contains isolation valves for the accumulator, the core makeup tank, the in-containment refueling water storage tank, and the passive core cooling system containment recirculation subsystem line.

There are eight automatically actuated containment isolation valves inside containment subject to flooding. These normally closed containment isolation valves are not required to operate during a safe shutdown operation and they would not fail open as a result of the compartment flooding. Also, there is a redundant, normally closed, containment isolation valve located outside containment in series with each of these valves.

The PXS-A compartment contains one normally closed spent fuel pool cooling system containment isolation valve and one normal residual heat removal system containment isolation valve. The PXS-B compartment contains four normally closed normal residual heat removal system containment isolation valves. The maintenance floor contains two normally closed liquid radwaste system containment isolation valves located partially below the maximum flood level.

Except for the valves mentioned above, the rest of the automatically actuated containment isolation valves are located above the maximum flood level; therefore, these components would not be adversely affected by postulated flooding.

Flooding can be postulated from a failure of several systems located inside the containment. The worst case flooding scenario is a LOCA. The maximum flood level for a LOCA is based on the combined inventory of the reactor coolant system, the two accumulators, the two core makeup tanks, and the in-containment refueling water storage tank flooding the containment. The maximum inventory also considers makeup from the cask loading pit and boric acid tank.

Curbs are provided around openings through the maintenance floor at elevation 107′-2″ or watertight hatches are provided to close openings not required for hydrogen venting to control flooding. Overflow into the refueling canal occurs through a pipe centered at elevation 110′-0″. Curbs around openings into the chemical and volume control system compartment extend up to elevation 110′-0″. Curbs around openings into the PXS-A compartment extend up to elevation 110′-2″. Curbs around openings into the PXS-B compartment extend up to elevation 110′-1″. With these curb elevations, water flooding the maintenance floor is directed first into the refueling canal, then into the CVS compartment, then into the PXS-B compartment, and finally into the PXS-A compartment.

The evaluation of containment flooding from postulated component failures includes the compartments that are located below the maximum flood level. There are seven subcompartments that contain components below the floor at elevation 107′-2″. The active safe shutdown components inside containment which are located below the maximum flood level are located in only two of the seven floodable compartments.

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The seven compartments partially or completely below the maximum flood level include the reactor vessel cavity, the two steam generator compartments, the vertical access tunnel, the two passive core cooling system compartments, and the chemical and volume control system compartment. The safe shutdown components are located in the two passive core cooling system compartments.

The reactor vessel cavity and the two steam generator compartments are interconnected by a large vertical access tunnel. These four compartments are treated, in this discussion, as one large floodable volume and they are referred to as the reactor coolant system compartment. Flooding of this compartment above elevation 107′-2″ also includes the maintenance floor outside the curbs around the other three compartments.

The PXS-A compartment (Room 11206), PXS-B compartment (Room 11207 and its connected RNS Valve Room 11208), and the chemical and volume control system compartment (Room 11209) are physically separated and isolated from each other by structural walls and curbs such that flooding in any one of these compartments or in the reactor coolant system compartment cannot cause flooding in any of the other compartments. Located in the 107'-2" floor are hatches which allow personnel access into PXS-A, PXS-B, and the CVS compartment for maintenance. These hatches are leak-tight and designed to prevent flooding through the 107'-2" floor into the compartments below. The access hatch to the PXS-B compartment is located near the containment wall and is normally closed to address severe accident considerations. The access hatch to the PXS-B compartment is accessible from Room 11300 on elevation 107′-2″.

The access hatches and curb on the 107'-2" maintenance floor are important design features which support containment flood-up during a LOCA. The access hatches are designed to be opened or removed during plant shutdown to support maintenance work in the PXS-A, PXS-B, and CVS compartments. The design of the curbs may also allow them to be removed to support component inspections or other plant shutdown maintenance activities. Removal of these components to conduct shutdown maintenance activities include limitations and compensatory measures to preserve an adequate flood-up recirculation level to be able to support core cooling in the unlikely event of a loss of shutdown cooling or loss of RCS inventory.

The fire protection system and the demineralized water transfer and storage system are open-cycle systems that enter the containment. During plant operation, the containment piping for these systems is isolated by containment isolation valves and is not a potential flooding source. These systems are not open systems as defined in Bulletin 80-24 (one that has an essentially unlimited source).

Reactor Coolant System Compartment

The reactor coolant system compartment, represented by the reactor vessel cavity, the two steam generator compartments, and the large vertical access tunnel, is the largest of the separate floodable compartments. With the exception of the pressurizer which is at a higher elevation, the principal components of the reactor coolant system are contained in this compartment.

The reactor vessel cavity and the adjoining equipment room are at the lowest level in the containment. The floor level of these rooms is at elevation 71′-6″. The floor level of the two steam generator compartments is at elevation 83′-0″. A portion of each compartment has low point areas at elevation 80′-0″.

The containment sump pumps are located in the equipment room at elevation 71′-6″. The arrangement for the floor drains from the two passive core cooling system compartments and the chemical and volume control system compartment provide a drain path for each compartment to the lowest level of containment (elevation 71′-6″) where the containment sump is located. Therefore, the source of the flooding in the reactor coolant system compartment is not limited to the components or systems contained within this compartment.

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Any leakage that occurs within the containment drains by gravity to the elevation 71′-6″ equipment room. Reverse flow into the two passive core cooling system compartments and the chemical and volume control system compartment is prevented by redundant backflow preventers in each of the three compartment drain lines.

Flooding in any compartment of the containment is detected by the containment sump level monitoring system and the containment flood-up level instrumentation.

The containment sump level monitoring system consists of three seismically qualified level sensors for the containment sump. These sensors transmit the sump level indication through a qualified isolation device to the main control room and the plant instrumentation system.

The plant instrumentation system monitors the rate of the sump level rise, calculates the leakage collection rate, and initiates the appropriate alarms in the main control room. A description of this leak detection system is provided in Subsection 5.2.5.3.1.

Another indication of flooding in this compartment is provided by the containment flood-up level instrumentation consisting of three redundant Class 1E level sensor racks. Multiple discrete level signals are provided from the bottom of the reactor vessel cavity to the top of the vertical access tunnel. These level sensors transmit the containment sump water level indication to the main control room.

In the event that the source of the containment flooding can not be terminated, the water level in the reactor vessel cavity and the steam generator compartment continues to increase until the water source has been depleted or the leak has been isolated. The maximum level that could occur in the compartment from all of the water which is available in containment is elevation 110′-2″.

Essential, safety-related components in the RCS compartment subjected to submergence are either protected or qualified for submergence. The flooding of this compartment has no impact on safe shutdown capability.

Passive Core Cooling System Compartments

The PXS-A and PXS-B compartments, located in the southeast and northeast quadrants of the containment, primarily contain components associated with the passive core cooling system. The safe shutdown related components of the passive core cooling system located in these two compartments are redundant and essentially identical. One set of the redundant equipment is located in each of the two separate compartments.

The redundant passive core cooling system components located in these two compartments provide coolant to the reactor vessel from the two core makeup tanks, the two accumulators, and the in-containment refueling water storage tank via two independent and redundant direct vessel injection lines.

Each passive core cooling system compartment contains a parallel set of normally closed, air operated, core makeup tank isolation valves that receive actuation signals to open during a safe shutdown operation. These valves are approximately 10 feet above the floor level of the passive core cooling system compartments and 26 feet above the floor of the reactor vessel cavity.

Each passive core cooling system compartment also contains one normally open accumulator isolation valve and one normally open in-containment refueling water storage tank isolation valve. These valves do not have to be repositioned during a safe shutdown operation and a coincident flooding event.

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In addition, each passive core cooling system compartment contains four passive core cooling system containment recirculation subsystem isolation valves. A normally closed, squib valve is located in each of two parallel flow paths. One of the lines includes a check valve in series with the squib valve. The other line includes a normally open, motor-operated valve in series with the squib valve. The squib valve and motor-operated valves are opened on a Low-3 in-containment refueling water storage tank level signal to provide a redundant flow path from the flooded reactor/steam generator compartments to the reactor vessel. One set of these redundant containment recirculation subsystem isolation valves is required to open to provide a redundant recirculation flow path to the reactor vessel. In the unlikely event that one of the two passive core cooling system compartments were to be flooded, the set of recirculation valves in the other, unflooded, compartment could be opened. Note that these squib valves are qualified to operate after being flooded.

The design bases for this system are described in Section 6.3. The passive core cooling system is designed to perform its safety functions in the unlikely event of the most limiting single failure occurring coincident with any design basis event. For example, a direct vessel injection line could break in one of the two passive core cooling system compartments, thus preventing the core makeup tank and the accumulator located in the compartment from delivering borated water to the reactor vessel. A coincident single failure in the other passive core cooling system compartment would prevent only one of the two parallel injection paths from opening. This series of events would not prevent the passive core cooling system from performing its safety function.

The maximum flooding rate to either of these passive core cooling system compartments would occur on a postulated LOCA of one of the eight inch direct vessel injection lines at a location inside one of the two compartments. This postulated rupture would result in direct blowdown from the reactor coolant system to the compartment as well as blowdown of the associated core makeup tank and accumulator. The resulting flooding in one of two passive core cooling system compartments would not prevent the passive core cooling system from performing its safe shutdown function.

Another postulated LOCA, that would cause rapid flooding in the PXS-B compartment, is a rupture of the 12 inch normal residual heat removal system line. This line is routed from one of two reactor coolant system hot legs to a containment penetration in the PXS-B compartment.

The evaluation of containment flooding events is also concerned with non-LOCA flooding events. The maximum flooding rate to either of the passive core cooling system compartments, for a non-LOCA event, would be based on a postulated rupture of one of the two in-containment refueling water storage tank lines or a postulated rupture of one of the two accumulator injection lines.

A 10-inch line is routed from the in-containment refueling water storage tank to the PXS-A compartment and a 10-inch line is routed to the PXS-B compartment. The driving head from a full in-containment refueling water storage tank to either of these two compartments is approximately 35 feet. A rupture in one of these lines would result in flooding of the associated passive core cooling system compartment and the reactor coolant system compartment via the normal drain path or by overflowing the passive core cooling system compartment.

The 8-inch accumulator injection lines are routed from the accumulators to the 8-inch direct vessel injection lines. A rupture of either of these two injection lines at a point upstream of the two series reactor coolant system pressure boundary check valves would result in the blowdown of the accumulator to the associated compartment. The water level attained in this case would be limited to the water volume of the accumulator. The water level would not reach the level of the core makeup tank isolation valves.

The total flood-up of either the PXS-A or PXS-B compartments from any source of water is acceptable and does not prevent the passive core cooling system from performing its required safe shutdown function.

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The PXS-A and the PXS-B compartments are physically separated and isolated from each other by a structural wall so that flooding in one compartment can not cause flooding in the other compartment. They are located below the maintenance floor level which is at elevation 107′-2″. A curb or watertight hatch is provided at openings that penetrate through the maintenance floor into these compartments from the elevation 107′-2″ floor level.

There are several HVAC ducts, cable trays, and pipes that penetrate the maintenance floor into the passive core cooling system compartments. These penetrations are properly protected to prevent leakage into the passive core cooling system compartments.

The floor drains for these two compartments are located at elevation 84′-6″. Reverse flow through the floor drains is blocked by redundant, safety-related backflow preventers in the drain lines.

When the flooding rate exceeds the ability of the floor drain lines to drain the water from the compartment, or in the event that the floor drain line is blocked, the water level in that compartment increases to the entrance curb elevation.

Should the flooding continue, the water overflows from that compartment to the maintenance floor at elevation 107′-2″. The water overflowing to this level would immediately drain to the reactor coolant system compartment via the vertical access tunnel. There is no curb at the entrance to the vertical access tunnel; therefore, water on the maintenance floor (elevation 107′-2″) flows freely into the reactor coolant system compartment. For LOCA events, flooding via this path continues to a level above the reactor coolant system cold legs.

If the leakage rate into PXS-A or PXS-B were not excessive, the compartment drain lines would prevent significant flood-up in that compartment. Consequently, the flooding of the components could be prevented for postulated flooding events of limited duration and flowrates less than the drain line capacity.

The flowrate from the compartments is a function of the water height in the PXS compartments and the water height in the reactor coolant system compartment. The differential head between the two water levels establishes the flowrate from the compartment.

The draining of these compartments initiates flooding of the reactor vessel cavity and the adjoining cavity equipment room. If the operator does not terminate the leak, action is taken to shut down the reactor.

If the flooding rate is not greater than the compartment drain line capacity, the large volume of the reactor vessel cavity and the adjoining equipment room provides additional time for the operator to identify the source of leakage before any significant flooding occurs in the compartments containing the passive core cooling system equipment.

Should the flooding continue, the water level eventually reaches the steam generator compartment floor at elevation 83′-0″. The large floor area of the two steam generator compartments and the vertical access tunnel provides additional volume for flood-up and reduces the rate of level increase.

The containment isolation valves in these two passive core cooling system compartments are located above elevation 95′-0″, but below the maximum flood-up level. The PXS-A compartment contains one normally closed, motor operated, spent fuel pool cooling system containment isolation valve. The PXS-B compartment contains four normally closed, motor operated, normal residual heat removal system containment isolation valves. These containment isolation valves are not required to operate for safe shutdown and they do not fail open as a result of compartment flooding. Also, there are redundant outside containment isolation valves for each line that penetrates the containment boundary.

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Chemical and Volume Control System Compartment

The majority of the components associated with the chemical and volume control system are located inside the containment in a separate compartment in the north quadrant of the containment below elevation 107′-2″.

There are several HVAC ducts, cable trays, and pipes that penetrate the maintenance floor into the chemical and volume control system compartment. These penetrations are properly protected to prevent leakage around the ducts into the chemical and volume control system compartment. The entrance curb elevation for the chemical and volume control system compartment is lower than the PXS-A and B compartment curbs to preferentially flood the chemical and volume control system compartment.

A single floor drain line is routed from this compartment to the containment sump at elevation 71′-6″. Reverse flow from the containment sump to this compartment is prevented by redundant, safety-related backflow preventers in the drain lines.

In the event that the single drain line were to be blocked, the water level in the chemical and volume control system compartment would flood to the level of the entrance curb elevation and would over flow to the maintenance floor at elevation 107′-2″. The water overflowing to this level would drain to the reactor coolant system compartment via the vertical access tunnel. There is no adverse effect on safe shutdown of the plant from flooding of the chemical and volume control system compartment.

The fire protection system and the demineralized water transfer and storage system are open-cycle systems that enter the containment. During plant operation, the containment piping for these systems is isolated by containment isolation valves and is not a potential flooding source. These systems are not open systems as defined in Bulletin 80-24 (one that has an essentially unlimited source).

3.4.1.2.2.2 Auxiliary Building Flooding Events

General

The AP1000 auxiliary building contains radiologically controlled areas and nonradiologically controlled areas which are physically separated by 2 and 3 foot structural walls and floor slabs. These structural barriers are designed to prevent flooding across the boundary between these areas by locating penetrations for piping and HVAC duct above maximum flood levels, or by sealing these penetrations. Process piping penetrations between the radiologically controlled areas and nonradiologically controlled areas are embedded in the wall or are welded to a steel sleeve in the wall. Electrical penetrations between the radiologically controlled areas and nonradiologically controlled areas are located above the maximum flood level. Electrical penetrations subject to the effects of the local build up of water on floors above the maximum flood level are also sealed.

For example, flooding in the auxiliary building at elevation 66′-6″ of the radiologically controlled area would not cause flooding in the nonradiologically controlled areas since the two areas are completely separated by a three foot thick structural wall. In the non-radiologically controlled area (non-RCA) of the auxiliary building, the four Class 1E electrical divisions are separated by 3-hour fire barriers. Portions of these fire barriers also serve as flood barriers.

- **Nonradiologically Controlled Areas**

  The safe shutdown systems and components that are located in the nonradiologically controlled area are associated with the protection and safety monitoring and Class 1E dc system, and containment isolation. The safe shutdown components associated with the protection and safety monitoring system are the instrumentation and control (I&C) cabinets located in the nonradioactive controlled area on level 3 (elevation 100′-0″). The safe shutdown components

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associated with the Class 1E dc system are the Class 1E batteries on level 1 (elevation 66′-6″) and level 2 (elevation 82′-6″) and dc electrical equipment also on level 2.

The nonradiologically controlled areas of the auxiliary building are designed to provide maximum separation between the mechanical and electrical equipment areas. This separation prevents the propagation of leaks from the piping areas and the mechanical equipment areas to the Class 1E electrical and Class 1E I&C equipment rooms.

The major piping compartments in the nonradiologically controlled area are the main steam isolation valve compartments on levels 4 and 5 (elevations 117′-6″ and 135′-3″, respectively) and the valve/piping penetration compartment on level 3 (elevation 100′-0″). The mechanical equipment rooms in the nonradiologically controlled area are the HVAC compartments on levels 4 and 5.

Drain lines are provided in each of the piping and mechanical equipment compartments which drain to the turbine building drain tank. Leakage from postulated pipe ruptures in these compartments will drain to the turbine building.

- **Radiologically Controlled Areas**

  The safe shutdown components located in radiologically controlled areas (RCA) are primarily containment isolation valves which are located near the containment vessel and above elevation 82′-6″. The containment isolation valves below the maximum flood level are either air-operated and fail closed, or remain closed during a safe shutdown operation.

The evaluation of potential flooding within the radiologically and nonradiologically controlled areas of the auxiliary building is performed on a floor-by-floor basis as described below.

Auxiliary Building Level 1 (Elevation 66′-6″)

- **Nonradiologically Controlled Area**

  Level 1 of the nonradiologically controlled area has five individual rooms that contain Class 1E batteries: four divisional (A, B, C, and D) Class 1E battery rooms and one Class 1E spare battery room. The doors are not water tight.

  The primary line of defense for level 1 is to exclude fluid systems and their associated piping from this area. The only fluid systems in level 1 are the potable water and fire protection systems. Potable or demineralized water brought into the area in portable containers is used for battery washdown. Potable water is used for the emergency eye wash/shower facilities. The maximum nominal diameter of potable water piping in this area is 1 inch; therefore, it is excluded from consideration as a source of flooding.

  The seismically qualified fire protection system piping routed through levels 1, 2, 3, and 4 is the only piping in level 1 that is greater than 1 inch in diameter.

  Fire fighting activities or fire protection system piping through-wall cracks in levels 1, 2, 3, or 4 would contribute to flooding in level 1. The drain lines, stairwells, and the elevator shaft direct the water from fire fighting activities down to the auxiliary building nonradiologically controlled area sump located on level 1.

  Fire fighting in these five battery rooms is accomplished by manual means from two fire hose stations located adjacent to the two stairwells. The maximum flowrate to this area from the two hose stations is assumed to be 250 gpm.

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A limited supply of water is initially provided to the fire protection system standpipe fire hose stations (see Subsection 9.5.1) from the passive containment cooling water storage tank. A nominal volume of 18,000 gallons is provided for the fire protection system. A volume of approximately 26,200 gallons is conservatively assumed; this is the volume in the tank between the elevations of the fire protection system inlet and the tank overflow, and associated system piping. In the event of a through-wall crack in the fire protection system piping, the maximum water depth in the battery rooms would be less than 7.5 inches, assuming that the water could propagate into all rooms on this level. This maximum water depth is substantially below the terminal height on the first row of battery jars, which are located 7.5 inches above the floor. The terminal height is approximately 30 inches above the floor.

Since a limited supply of fire water is provided, inadvertent initiation of the fire protection system can not exceed the flooding levels described above. Operator action to stop inadvertent water flow from the fire protection system is expected to limit flooding to only a small fraction of this water supply.

Structural walls, drain line routing, and raised platforms prevent leakage that may occur in piping or mechanical areas on levels 4 and 5 from propagating to the electrical areas on levels 1, 2, 3, or 4.

Dual sump pumps and water level sensors are also provided in the sump on level 1. The level sensors transmit water level indication to the main control room and the plant control system. Level alarms alert the operator to take corrective action.

The sump pumps are sized to remove approximately 250 gpm (with two pumps operating) based on a maximum flow from two fire hose stations of 250 gpm. The discharge of these pumps is directed to the turbine building drain tank of the waste water system (WWS) located on elevation 89′-0″ of the turbine building as described in Subsection 9.2.9. The discharge line into the tank is provided with a standpipe to prevent siphoning back to the auxiliary building nonradiologically controlled area sump. These sump pumps and level sensors are not required to maintain safe shutdown capability.

- **Radiologically Controlled Area**

  There are no safe shutdown components located on level 1 of the radiologically controlled area. The radiologically controlled area of the auxiliary building is subject to flooding from a variety of potential sources including, but not limited to, the component cooling water, central chilled water, hot water, spent fuel pool cooling, normal residual heat removal system, fire protection system, and chemical and volume control system, as well as various tanks. Most of the piping associated with these systems is above level 1; however, the flow from any postulated rupture in the radiologically controlled area will eventually flood level 1. The principal flow paths to level 1 are the vertical pipe chase and the floor gratings provided in the elevator lobbies on levels 2 and 3. Other flow paths include the floor drain system, the stairwell, and the elevator shaft. The western wall of the stairwell contains a safety-related, seismically qualified, and fire-rated flood relief louver that provides a flow path into the adjacent corridor in the event of a fire protection system line rupture.

  The auxiliary building radiologically controlled area sump is located on level 1 with dual sump pumps and water level sensor provided in the sump. The level sensor transmits water level indication to the main control room and the plant control system. High level alarms alert the operator to take corrective action. Two seismic Category I flood level sensors, which alert in the main control room, are also located in this area.

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The sump pumps are sized to remove approximately 250 gpm (with two pumps operating) based on a maximum flow from two fire hose stations of 250 gpm. The discharge of these pumps is directed to the waste holdup tanks of the liquid radioactive waste system as described in Section 11.2. These sump pumps and sump level sensors are not required to maintain safe shutdown capability.

For the component cooling water and central chilled water systems, the maximum flooding volume is bounded by the system volume plus a reasonable period of makeup. This includes any discharges from the pressure relief valves on the cooling water lines from the RNS heat exchangers. Flow from these two valves is also directed to the auxiliary building sump through the radioactive waste drain system. For the spent fuel pool cooling system, the maximum flooding volume is limited to the volume of water above the non-seismic spent fuel pool cooling suction line, plus a reasonable period of makeup. This flooding volume is approximately equal to that of the component cooling water and chilled water systems above.

The normal residual heat removal system is primarily operated when the plant is being shutdown. Since it is not normally operating, it is evaluated as a moderate-energy system. Flooding is determined based on the maximum flowrate from a through-wall crack in a 10 inch normal residual heat removal system suction line.

Flooding due to a break in the high-energy chemical and volume control system makeup pump discharge line bounds the normal residual heat removal system through-wall crack.

For a flooding event due to a moderate-energy line break of the fire protection system main header loop running through the auxiliary building, the maximum connected fire protection system inventory is assumed to be drained.

A flooding event due to a moderate-energy line break of the fire protection system main header loop running through the auxiliary building bounds the above conditions. Due to a circumferential break in this non-seismic moderate-energy piping, flooding flow output from both break ends originating from higher elevations eventually flows down to elevation 66'-6". Based on notification of flooding conditions received in the main control room, credited operator action is taken after pipe ruptures occur to limit the flood to the design basis level. Due to limited volume of the fire water storage tanks, maximum height is limited to 19 feet in non-tank rooms. To provide indication that supports initiation of credited operator action, two redundant, seismic Category I flood level sensors are used to alert the operators in the main control room of a floodup within the radiologically controlled area of the auxiliary building.

Flow from the postulated bounding break spreads throughout the level 1 rooms and corridor via flow under doors and interconnecting floor drains if the auxiliary building radiologically controlled area sump pumps are inoperable. The maximum flood level in the area, for any of the cases above, is 19 ft. This flooding will extend up to level 2 to elevation 85'-6" after credited operator action is taken to isolate fire protection system makeup at 7 hours past rupture initiation. This flooding has no impact on safe shutdown since there are no flooded components on level 1 or 2 that cannot perform the functions required for safe shutdown. The maximum flood height in waste holdup tank rooms (12166 and 12167) is 23 feet due to the water-tight doors and the locked-closed drain.

Flow from a tank rupture in one of the tank rooms will initially flood the tank room, and begin to flow to the auxiliary building radiologically controlled area sump via floor drains. If the sump pumps are inoperable, the tank volume floods the balance of level 1 via the interconnecting floor drains. The maximum flood level for this event is less than for the piping failures discussed above.

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Auxiliary Building Level 2 (Elevation 82′-6″)

- **Nonradiologically Controlled Area**

  Level 2 of the nonradiologically controlled area has two Class 1E battery rooms, four divisional Class 1E dc electrical equipment rooms, and one Class 1E reactor coolant pump trip switchgear room. The doors to these rooms are not water tight.

  Level 2 contains an arrangement of fire protection and potable water piping similar to level 1.

  The potential for flooding on this level is limited to fire fighting activities and through-wall cracks in fire protection system piping. Fire fighting in these rooms is accomplished by manual means from two fire hose stations located adjacent to the two stairwells. The maximum flowrate to this area from the two hose stations is assumed to be 250 gpm.

  The drains, elevator shafts, and stairwells drain water spilled on this level to level 1. Therefore, no significant accumulation of water occurs on level 2.

- **Radiologically Controlled Area**

  The radiologically controlled area on level 2 contains a few containment isolation valves and an effluent monitor tank at elevation 92’-6". The horizontal pipe chase at elevation 92′-6″ contains two normally closed normal residual heat removal system isolation valves. One spent fuel pool cooling system containment isolation valve is located, above 92′-6″, in the adjacent vertical pipe chase. The area on the north side of the lower annulus contains two chemical and volume control system and two liquid radwaste automatically operated containment isolation valves above elevation 82′-6″. These valves are required to close or remain closed during a safe shutdown operation.

  Two chemical and volume control system valves used to isolate the chemical and volume control system makeup pump suction from the demineralized water storage tank are located in the demineralizer/filter access area at 82′-6″. These safety-related valves close or remain closed to prevent boron dilution events. They are not required for safe shutdown.

  Potential sources of flooding for this area include, but are not limited to, the chemical and volume control system, a liquid radioactive waste system effluent monitor tank, and the fire protection system, including an automatic suppression system in the CVS makeup pump room. Flow from a component rupture or from fire fighting activities on level 2 drains to level 1 as described below.

  To protect the above valves from flooding, the makeup pump compartment at elevation 82′-6″ drains, via the floor grating located in the corridor adjacent to the stairwell, directly to elevation 66′-6″. Flooding in the lower annulus drains directly to elevation 66′-6″ via the floor grating, drains, and various openings to the tank rooms. The stairwell and elevator shaft on the east wall are additional flow paths to level 1. The horizontal pipe chase at elevation 92′-6″ drains under the door to elevation 66′-6″ via the vertical pipe chase in the spent fuel pool cooling system penetration room. As a result of these drain paths, there is less than a 12-inch accumulation of water in the horizontal pipe chase from any postulated pipe ruptures. The containment isolation valves in these areas are capable of moving to their safety position during the event.

  Due to the moderate-energy line break of the fire protection system and the subsequent drainage of the entire connected fire protection system inventory to level 1, the bounding flood height originating from elevation 66'-6" is 19 feet. That flood height extends up to level 2 and terminates at elevation 85'-6". Therefore, the bounding flood height for the rooms on auxiliary building level 2 is 3 feet for a room with a floor elevation of 82'-6". This flooding has no impact on safe shutdown

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since there are no components on level 2 that cannot perform the functions required for safe shutdown if flooded.

Auxiliary Building Level 3 (Elevation 100′-0″)

- **Nonradiologically Controlled Area**

  Level 3 of the nonradiologically controlled area includes the remote shutdown room, one reactor coolant pump trip switchgear room, four divisional Class 1E I&C rooms, one equipment room, and the valve/piping penetration room. The division A, B, C and D I&C rooms and the electrical room also include containment electrical penetrations. The doors are not water tight.

  The level 3 Class 1E and non-Class 1E electrical areas contain only fire protection system piping. Fire hose stations are provided near each of the two stairwells and normally dry fire protection piping, supplied from the passive containment system tank, serves the preaction sprinkler system in the non-1E equipment/penetration room. The potential for flooding in the electrical areas on this level is limited to fire fighting activities. The maximum flowrate to this area from either automatic or manual fire fighting activities is assumed to be 250 gpm. The floor drains, stairwells, and elevator shaft drain water spilled on this level down to level 1. Therefore, no significant accumulation of water occurs in this area.

  The valve/piping penetration room on level 3 is physically separated from the electrical rooms. The valve/piping penetration room contains automatically actuated containment isolation valves for the steam generator blowdown system and the hydrogen line in the chemical and volume control system. Access to this room is from the turbine building. The access door and drain lines provided in this room drain from the auxiliary building to the turbine building. Maximum postulated flood level for this room is less than 36 inches. The containment isolation valves in the area are located above this maximum flood level.

- **Radiologically Controlled Area**

  Level 3 of the radiologically controlled area includes two containment pressure transmitters, three spent fuel pool level transmitters, four spent fuel pool cooling system isolation valves, and two passive containment cooling system safety-related valves, which are required for safe shutdown.

  Potential sources of flooding for this area include, but are not limited to, the normal residual heat removal system, the component cooling water system, an effluent monitor tank and the fire protection system, including an automatic suppression system in the rail car bay. The worst case flooding scenario on this level is the moderate-energy line break in the fire protection system located in the rail car bay area. This scenario results in significant flooding in the rail car bay, residual heat removal system heat exchanger room, and resin transfer pump room due to the high flow rate in the fire protection system and limited flow paths out of the room. The maximum flood level in the rail car bay, resin transfer pump room, and residual heat removal system heat exchanger room is less than 4 feet. Fluid from this break will flow from the rail car bay through a safety-related flood relief flapper into the normal residual heat removal heat exchanger room, and through floor penetrations to the lower levels.

  The containment pressure transmitters and one of the spent fuel pool level transmitters are located in the middle annulus. The safety-related passive containment cooling valves are located in the maintenance floor staging area. A moderate-energy line break in the fire protection system located in the maintenance floor staging area causes significant flooding in the middle annulus. The maximum flood level in the middle annulus and the staging area is below elevation 107'-9". The spent fuel pool level transmitter will be submerged; however, the transmitter is qualified for submergence.

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The other safe shutdown components are located above the maximum flood level. The other two spent fuel pool level transmitters and the spent fuel pool cooling system isolation valves are located in the waste monitor tank room B. The maximum flood level in this room is due to a waste monitor tank rupture and will be less than 109 inches. The spent fuel pool level transmitters in this area are located below the maximum flood level; however, they are qualified for submergence.

For each scenario, the plant is maintained in a safe condition and the safety functions maintain their integrity without the need for operator action to mitigate the flooding. Flow from a component rupture or from fire fighting activities in other rooms on level 3 drains directly to level 1.

Auxiliary Building Level 4 (Elevation 117′-6″)

- **Nonradiologically Controlled Area**

Level 4 of the nonradiologically controlled area includes the main control room, one divisional Class 1E penetration room, one non-Class 1E electrical penetration room, two main steam isolation valve compartments, and one mechanical equipment room.

The doors to these rooms are not water tight. There are no doors from the main steam isolation valve compartments to the Class 1E electrical areas. The main steam isolation valve compartments are only accessible from the turbine building at elevation 135′-3″. The mechanical equipment room is only accessible from the turbine building at elevation 117′-6″.

The potential for flooding Class 1E electrical areas on this level is limited to fire fighting activities. The Class 1E electrical penetration room and main control room are accessible from a hose station near the east stairwell. While the main control room kitchen and restroom are provided with potable water, the lines are 1 inch and smaller, and are not evaluated for pipe ruptures. The main control room contains a water heater tank and expansion tank Section in the men’s toilet room to provide hot water for hand washing and dishwashing. To mitigate the effects of a postulated tank rupture, the water heater tank and expansion tank are enclosed in a seismic Category I structure and the room design is configured to contain the water and direct it to the floor drain, where it is gravity-drained to the sanitary drainage system (SDS).

Fire fighting in the control room is done manually using portable extinguishers or a fire hose from a hose station in the east corridor. In the event that a hose is brought into the main control room through the east corridor access doors, water accumulation is limited by flow through the access doors which are open. The threshold of the east corridor access door is at the elevation of the floor slab. Once in the corridor this flow drains, via floor drains, the stairwell and elevator shaft, to level 1. An emergency egress door and stairwell is located on the west end of the main control room, which leads down to the remote shutdown room. The threshold of the emergency egress door is flush with the raised portion of flooring in the main control room, which is approximately 14 inches above the east corridor entrance. Water being discharged in this area will flow through the porous raised flooring and flow back out the east access doors. The main control room has a normally closed floor drain which can be manually opened to drain water to the auxiliary building non-RCA sump at level 1. The drain paths prevent significant flooding of the adjacent rooms.

In the event of fire fighting activity in the non-Class 1E electrical penetration rooms, the accumulation of water is prevented by floor drains and flows through the stairwell and elevator shaft to level 1.

The mechanical equipment room contains containment isolation valves for the chilled water, compressed air, component cooling water, and passive core cooling (nitrogen) systems. Flooding in the mechanical equipment room due to fire fighting or piping ruptures is directed to the turbine building through the access door at elevation 117′-6″ or through floor drains to the turbine

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building. The maximum flood level for this room is 4 inches. The containment isolation valves in this area are located above this maximum flood level.

The main steam isolation valve compartments contain the main steam and main feedwater piping and their isolation valves. In the event of a pipe break or leak in the area, floor drains to the turbine building are provided. Structural walls and floors are designed to prevent flow of water to levels 1, 2, or 3. For larger flows, wall openings and pressure relief panels, located at floor elevation, open to drain the rooms to the turbine building. The maximum flood level for these rooms is less than 36 inches. The isolation valves in this area are located above this maximum flood level.

- **Radiologically Controlled Area**

  In the radiologically controlled area, there are six containment isolation valves on level 4. Five of these are located in the vertical pipe chase. These are for the primary sampling system, spent fuel pool cooling system and containment air filtration system. The primary source of flooding in the vertical pipe chase is from a moderate-energy line break in the fire protection system located in the personnel access area on level 5. Flow from this break will be directed through the open pipe penetration on level 5, into the vertical pipe chase, through the grating down to level 3 where water will flow under the door to the waste monitor tank room and through floor drains and an open sleeve penetration to the auxiliary building RCA sump which limits the flood level to less than 12 inches in the vertical pipe chase and 109 inches in the waste monitor tank room. The containment isolation valves are located above maximum flood level. The other containment isolation valve for the containment air filtration system is located in a separate compartment adjacent to column line 5. The principal source of flooding for this area is from a moderate-energy line break in the fire protection system located in the personnel access on level 5. Flow from this source will be directed through the open pipe penetration on level 5, into the containment air filtration system penetration room, under the door and through floor drains to the auxiliary building radiologically controlled area sump which limits the flood level to less than 30 inches in the containment air filtration system penetration room. The containment isolation valve body will be partially submerged; however, the limit switch and solenoid are above the maximum flood level. No other safe shutdown equipment is located in this area.

Auxiliary Building Level 5 (Elevation 135′-3″)

- **Nonradiologically Controlled Area**

  Level 5 of the nonradiologically controlled area contains two mechanical HVAC equipment rooms and the upper portion of the two main steam isolation valve compartments. The only safety-related equipment on level 5 that could be affected by flooding is the potable water system main control room isolation valve located in the main mechanical HVAC equipment room, but it is positioned above the maximum flood height for this room.

  The evaluation of the main steam isolation valve compartments is addressed in the discussion of level 4.

  Water from fire fighting, postulated pipe, or potable water storage tank ruptures in the main mechanical HVAC equipment rooms drains to the turbine building via floor drains. Therefore, no significant accumulation of water occurs in this room. Floor penetrations are sealed and a 6 inch platform is provided at the elevator and stairwell such that flooding in these rooms does not propagate to levels below.

  The mechanical room between the main steam isolation valve compartments at level 5 is accessed from the turbine building on the same level. This room is drained to the turbine building. In the event of fire fighting or postulated pipe ruptures, the accumulation of water is prevented by

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directing the flow to the drains into the turbine building. Floor penetrations are sealed such that flooding in this area does not propagate to other areas of the auxiliary building.

- **Radiologically Controlled Area**

  Level 5 of the radiologically controlled area contains the fuel handling area operating deck, HVAC equipment and access rooms, the main equipment hatch staging area, and the component cooling water system valve room. Safety-related equipment on level 5 includes the compressed air tanks for the main control room emergency habitability system and associated valves located in the main equipment hatch staging area, containment leak rate test system valves in the personnel access area, and spent fuel equipment located in the fuel transfer canal and spent fuel pool.

  Over-filling of the spent fuel pool would flood the fuel handling area operating deck. The flooding flowrate is limited by the makeup capacity from the demineralized water or chemical and volume control systems. Accumulation of water in this area is prevented by floor drains. Spent fuel pool cooling is not adversely affected by this event. There is no safe shutdown equipment in this area. The component cooling water system valves with the regulatory treatment of nonsafety-related system important missions located in the component cooling water system valve room which support the normal residual heat removal system, are located well above the maximum flood level for this area and are expected to remain functional in a flooding event.

  The shield building stairwell serves as a pipe chase for passive containment cooling system supply and return lines, drains for the passive containment cooling system valve room and passive containment cooling system air outlet shield plug, and a fire water line. Leakage from a crack in one of these lines flows down the stairwell to level 5, under the stairwell door to the HVAC equipment room, and then to the auxiliary building radiologically controlled area sump via floor drains or to the annex building. There is no significant accumulation in the stairwell or the equipment and access rooms. There is no safe shutdown equipment in this area. The passive containment cooling system supply and return line connections to the passive containment cooling system storage tank are above the maximum water level, thus a leak in these lines would not adversely affect the safe shutdown capability of the passive containment cooling system.

  Water from fire fighting in the main equipment hatch staging area drains to the auxiliary building radiologically controlled area sump via floor drains. Therefore, no significant accumulation of water occurs in this area.

Auxiliary Building Upper Annulus (Elevation 132′-3″)

This area serves as the air flow path for the passive containment cooling system. It is bounded by the seismic Category I shield building on the outside and the seismic Category I containment vessel on the inside. The floor has a curb on the outside with a flexible seal connected to the shield building. The curb and seal block communication with the middle annulus area, below. The outside wall of the annulus is provided with redundant safety-related drains to the yard drainage system.

The worst case flooding scenario is postulated as blockage of the nonsafety-related floor drains concurrent with inadvertent opening of a passive containment cooling system cooling water isolation valve. The maximum flood level is determined by the flow gradient to the operating drains. Maximum level will be approximately 2 feet. This level does not affect the capability of passive containment cooling system air cooling. No other safe shutdown equipment is affected. Passive containment cooling system operation or leakage is detected by sensors on the passive containment cooling system discharge line. During non-accident conditions the annulus is accessible to manually clear any drain blockage.

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PCS Valve Room (Elevation 284′-10″)

This room contains three redundant safety-related valve trains for the passive containment cooling system water cooling subsystem. One train must open to provide the required containment cooling. The only source of flooding for this room is a through-wall crack in the passive containment cooling system piping. The bounding through-wall crack is postulated in the 4 inch recirculation piping. This leak is not isolable from the 756,700-gallon passive containment cooling system water storage tank above the valve room.

The PCS valve room contains floor drains with sufficient capacity to prevent water accumulation in the room following a postulated through-wall crack. Water from the through-wall crack flows into the floor drains out to the auxiliary building roof, then to the storm drain system. The passive containment cooling system isolation valves are located above the maximum flood level in the valve room, so they remain operable.

The passive containment cooling water storage tank (PCCWST) level instruments are relied upon to monitor PCCWST water level for operation of the PCS and to identify leakage from the PCS. The leakage does not adversely affect containment or any other essential system.

3.4.1.2.2.3 Adjacent Structures Flooding Events

Turbine Building

The turbine building is subject to flooding from a variety of potential sources including the circulating water, service water, condensate/feedwater, component cooling water, turbine building cooling water, demineralized water and fire protection systems as well as the deaerator storage tank. Flow from any postulated ruptures above elevation 100′-0″ flows down to elevation 100′-0″ via floor grating and stairwells. Thus, there will be a negligible contribution from these sources to flooding of the auxiliary building compartments at elevations 135′-3″ and 117′-6″ via flow under doors. Auxiliary building flooding is bounded by the effects of postulated breaks in the compartments.

The bounding flooding source for the turbine building is a break in the circulating water piping which would result in flooding of the elevation 100′-0″ floor. Flow from this break runs out of the building to the yard through a relief panel in the turbine building west wall and limits the maximum flood level to less than 6 inches. The only area of the auxiliary building which interfaces with the turbine building at elevation 100′-0″ is the valve/piping penetration room. This room could be flooded via flow under the door or backflow through the drains, however the flood level would be less than postulated for a break in the valve/piping penetration room itself.

The waste water system (WWS) sump pumps located in the nonradiologically controlled area of the auxiliary building discharge to the turbine building drain tank. Backflow from the drain tank is prevented as described in Subsection 3.4.1.2.2.2

There is no safety-related equipment in the turbine building. The component cooling water and service water components on elevation 100′-0″, which provide the regulatory treatment of nonsafety-related systems important to support the normal residual heat removal system, are expected to remain functional following a flooding event in the turbine building since the pump motors and valve operators are above the expected flood level.

Annex Building

- **Nonradiologically Controlled Areas**

  The primary sources of flooding in the nonradiologically controlled areas of the annex building are the component cooling water, chilled water and fire protection systems. Water from postulated breaks above elevation 100′-0″ flows primarily through floor drains to the annex

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building sump that discharges to the turbine building drain tank. Alternate paths include flows to the turbine building via flow under access doors at elevations 135′-3″ and 117′-6″ and flows down to elevation 100′-0″ via stairwells and elevator shaft. Water accumulation at elevation 100′-0″ is minimized by floor drains to the annex building sump and by flow under the access doors leading directly to the yard area.

There is no safety-related equipment in the nonradiologically controlled area portion of the annex building. The main ac power system components with regulatory treatment of nonsafety-related systems important missions are located on elevation 117′-6″ in the electrical switchgear rooms, which are separated from potential flood sources. Water from manual fire fighting operations is collected by floor drains discharging to the annex building sump or down a hatch or stairwell to elevation 100′-0″. The non-Class 1E dc and UPS system (EDS) equipment with regulatory treatment of nonsafety-related systems important missions is located on elevation 100′-0″ in separate battery rooms. Water in one of these rooms due to manual fire fighting in the room is collected by floor drains to the annex building sump and by flow under the access doors leading directly to the yard area. This is not expected to affect functionality of equipment in the adjacent rooms.

- **Radiologically Controlled Areas**

  There is no safety-related equipment in the radiologically controlled area portion of the annex building. The primary sources of flooding in the radiologically controlled areas of the annex building are the component cooling water, chilled water and fire protection systems, including an automatic suppression system that protects the containment access corridor. Water from postulated breaks above elevation 100′-0″ drains through floor drains to the radioactive waste drain system sump in the radiologically controlled area of the auxiliary building or drains to elevation 100′-0″ via stairwells and equipment handling hatches or under access doors to the radiologically controlled area portion of the auxiliary building. Accumulated water at elevation 100′-0″ is minimized by floor drains discharging to the radioactive waste drain system sump or chemical waste tank in the auxiliary building. The contribution of water to the flooding of the radiologically controlled area portion of the auxiliary building is bounded by flooding events which could occur in the auxiliary building.

Radwaste Building

The potential sources of flooding in the radwaste building are the chilled water, hot water, and fire protection systems or from failure of one of the three waste monitor tanks. Flow from postulated breaks is directed to floor drains via a curb/sloped floor around the perimeter to drain to the radioactive waste drain system sump in the radiologically controlled area of auxiliary building. The contribution of water to flooding of the auxiliary building is bounded by flooding events which could occur in the auxiliary building. There are no safety-related systems or components or equipment with regulatory treatment of nonsafety-related systems important missions in the radwaste building.

Diesel Generator Building

The potential source of flooding in the diesel generator building is the fire protection system. There is no safety-related equipment in the diesel generator building. The diesel generator system which has regulatory treatment of nonsafety-related systems important mission has each diesel and associated auxiliaries in a separate compartment. Flooding due to a break in a fire water header is directed to the respective diesel generator building sump and subsequently pumped to the yard oil separator or is drained by gravity to the yard area under the access doors. The equipment in the adjacent diesel generator compartment should remain functional following the event.

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3.4.1.3 Permanent Dewatering System

The need for a permanent dewatering system is site specific and is defined as discussed in Subsection 3.4.3.

3.4.2 Analytical and Test Procedures

The AP1000 is designed so that the maximum water levels considered due to natural phenomena or internal flooding do not jeopardize the safety of the plant or the ability to achieve and maintain safe shutdown conditions. The analytical approach in the consideration of external and internal flooding events is described in Subsection 3.4.1.2.

3.4.3 Combined License Information

The Combined License applicant referencing the AP1000 certified design will demonstrate that the site satisfies the interface requirements as described in Section 2.4. If these criteria cannot be satisfied because of site-specific flooding hazards, the Combined License applicant may propose protective measures as discussed in Section 2.4.

3.4.4 References

1. ANSI/ANS-56.11-1988, "Design Criteria for Protection against the Effects of Compartment Flooding in Light Water Reactor Plants."

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NOTES:

1. HDPE DOUBLE-SIDED TEXTURED WATERPROOF MEMBRANE ON TOP OF FIRST LAYER OF MUDMAT AND ON OUTSIDE VERTICAL FACE OF AUXILIARY BUILDING WALL UP TO EL. 100' -0" GRADE LEVEL (WITH PROTECTIVE SHIELD ON VERTICAL FACE)

2. MSE WALL TO BE DESIGNED WITH GEOREINFORCED MATERIALS AND 18" THICK LAYERS OF COMPACTED FREE DRAINING GRANULAR SOIL

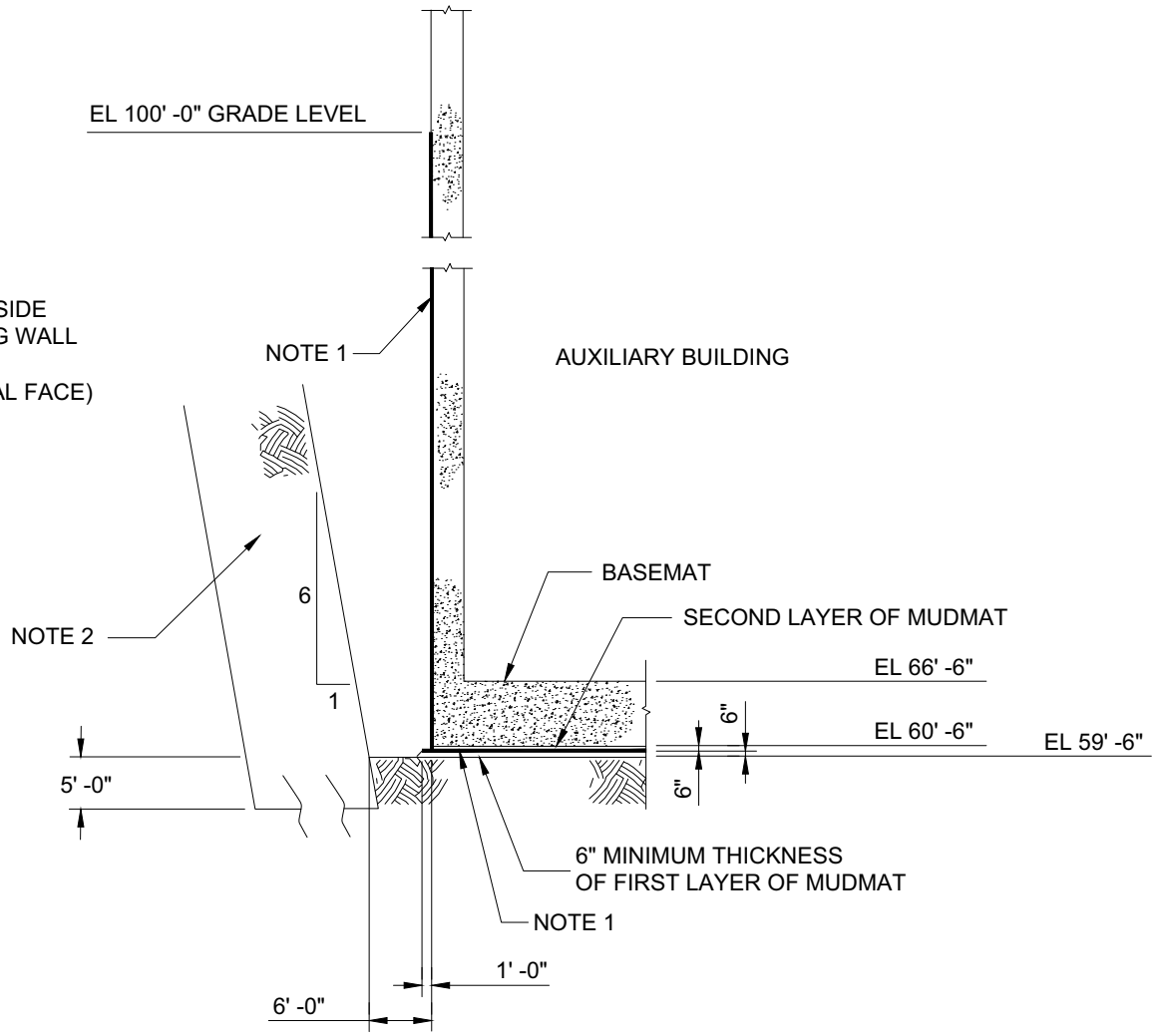

Figure 3.4-1
Typical Details of
Nuclear Island Waterproofing Below Grade

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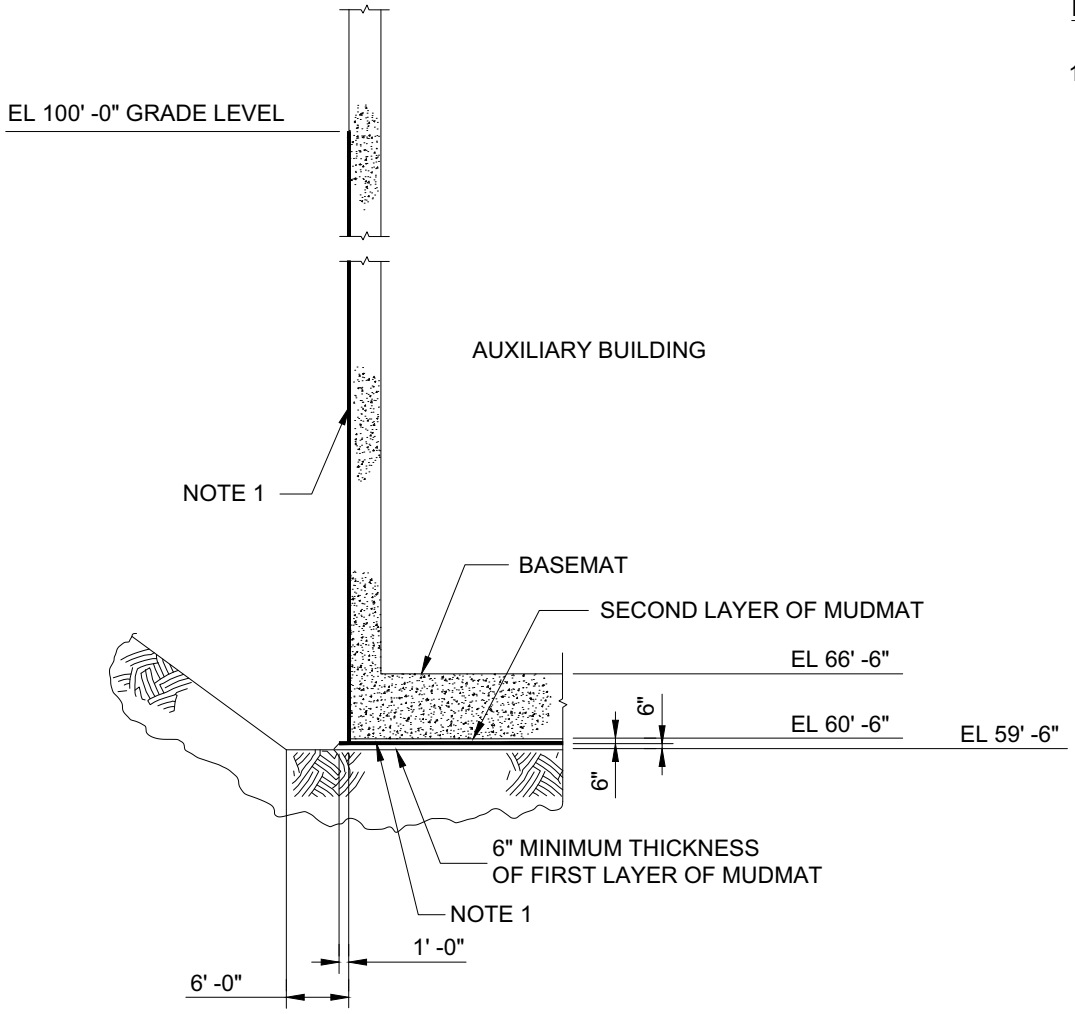

NOTES:

1. HDPE DOUBLE-SIDED TEXTURED WATERPROOF MEMBRANE ON TOP OF FIRST LAYER OF MUDMAT AND ON OUTSIDE VERTICAL FACE OF AUXILIARY BUILDING WALL UP TO EL 100' -0" GRADE LEVEL (WITH PROTECTIVE SHIELD ON VERTICAL FACE)

Figure 3.4-2
Typical Details of Nuclear Island
Waterproofing Below Grade with Step Back

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Figure 3.4-3 Not Used

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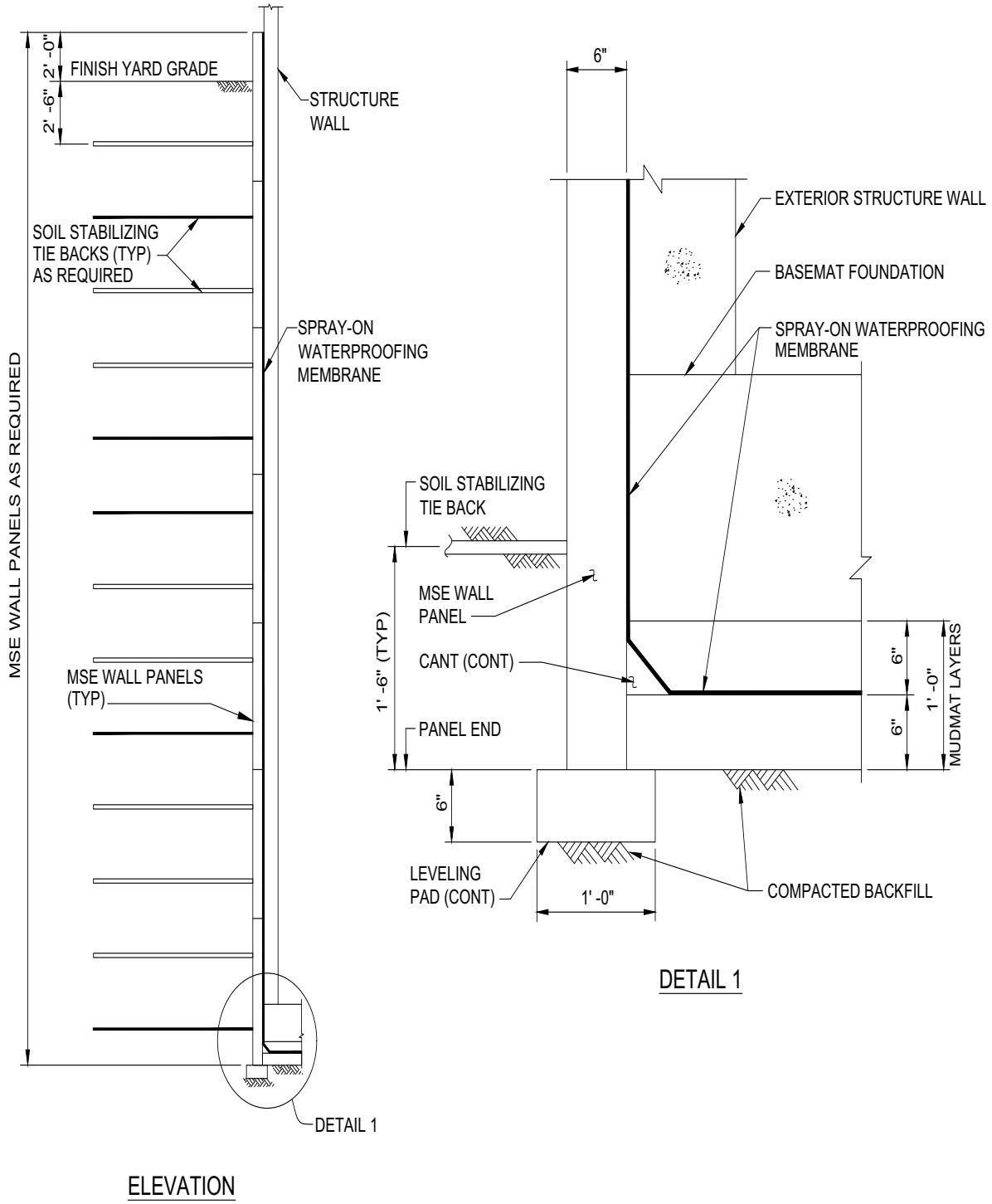

Figure 3.4-4
Typical Details of Membrane Corner Detail at Basemat and Exterior Wall

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3.5 Missile Protection

General Design Criterion 4 of Appendix A to 10 CFR 50 requires that structures systems and components important to safety be protected from the effects of missiles. The AP1000 criteria for protection from postulated missiles provide the capability to safely shut down the reactor and maintain it in a safe shutdown condition. The AP1000 criteria also protect the integrity of the reactor coolant system pressure boundary and maintain offsite radiological dose/concentration levels within the limits defined in 10 CFR 50.34.

Missiles may be generated by pressurized components, rotating machinery, and explosions within the plant and by tornadoes or transportation accidents external to the plant. Potential missile hazards are eliminated to the extent practical by minimizing the potential sources of missiles through proper selection of equipment, and by arrangement of structures and equipment in a manner to minimize the potential for damage from missiles. Potential missiles due to failures of nonseismic items are addressed in Subsection 3.7.3.13. Heavy load-drop evaluations are described in Subsection 9.1.5.

The following are definitions for missile protection terminology:

Internally Generated Missile – A mass that may be accelerated by energy sources continuously present on site.

Single Active Failure – Malfunction or loss of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic, or electrical malfunction, but not the loss of component structural integrity.

High-Energy System – Fluid systems that, during normal plant conditions, are operated or maintained pressurized with a maximum operating temperature greater than 200°F and/or a maximum operating pressure greater than 275 psig, as discussed in Subsection 3.6.1.

The following criteria are applied in the identification of missiles and the protection requirements that must be satisfied:

- A missile must not damage structures, systems, or components to the extent that could prevent achieving or maintaining safe shutdown of the plant or result in a significant release of radioactivity.

- A single active component failure is assumed in systems used to mitigate the consequences of the postulated missile and achieve a safe shutdown condition. The single active component failure is assumed to occur in addition to the postulated missile and any direct consequences of the missile. When the postulated missile is generated in one of two or more redundant trains of a dual-purpose safety-related fluid system, which is designed to seismic Category I standards and is capable of being powered from both onsite and offsite sources, a single active component failure need not be assumed in the remaining train(s), or associated supporting trains.

- Walls, partitions, and other items that enclose safety-related systems, or separate redundant trains of safety related equipment, must be constructed so that a postulated missile cannot damage components required to achieve safe shutdown nor damage components required to prevent a release of radioactivity producing offsite doses in excess of 10 CFR 50.34 limits.

- A postulated missile from the reactor coolant system must not cause loss of integrity of the primary containment, main steam, feedwater, or other loop of the reactor coolant system.

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- A postulated missile from any system other than the reactor coolant system must not cause loss of integrity of the containment or the reactor coolant system pressure boundary.
- Other plant accidents or severe natural phenomena are not assumed to occur in conjunction with a postulated missile (except for tornado).
- Offsite power is assumed to be unavailable if a trip of the turbine-generator or reactor protection system is a direct consequence of the postulated missile.
- Safe shutdown is accomplished using only safety-related systems with a coincident single active failure, although nonsafety-related systems not affected by the missile are available to support safe shutdown.
- Missiles are postulated to occur where the single failure of a retention mechanism can result in a missile, unless the missile is not considered credible as discussed later. Missiles created by the independent failures of two retention mechanisms are not postulated.
- The energy of postulated missiles produced by rotating components is based on a 120 percent overspeed condition, unless such an overspeed condition is not possible (such as a synchronous motor).
- Equipment required for safe shutdown is located in plant areas separate from potential missile sources wherever practical.
- Spatial separation may be used to demonstrate protection from missile hazards when it is shown that the range and trajectory of the generated missile is less than the distance to or is directed away from the potential target.

The AP1000 passive design minimizes the number of safety-related structures, systems, and components required for safe shutdown. Systems required for safe shutdown are identified in Chapter 7. Safety class structures, systems and components, their location, seismic category, and quality group classifications are given in Section 3.2. General arrangement drawings showing locations of the structures, systems, and components are given in Section 1.2 The areas required for safe shutdown, and the major systems and components housed therein that are required to be protected from internally and externally generated missiles for safe shutdown, are summarized below:

- The containment vessel, including the reactor coolant loop, and passive core cooling system inside containment
- The shield building, including the passive containment cooling system
- Containment penetration areas, including containment isolation valves and Class IE cables
- The control complex including the main control room, reactor protection system, batteries, and dc switchgear
- The spent fuel pool

The AP1000 relies on safety-related systems and equipment to establish and maintain safe shutdown conditions. There are no nonsafety-related systems or components that require protection from missiles.

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Evaluations are performed to demonstrate that the criteria are satisfied in the event a credible missile is produced coincident with a single active component failure. These evaluations include the following:

- For those potential missiles considered to be credible, a realistic assessment is made of the postulated missile size and energy, and its potential trajectories.
- Potentially impacted components associated with systems required to achieve and maintain safe shutdown are identified.
- Loss of these potentially impacted components coincident with an assumed single active component failure is evaluated to determine if sufficient redundancy remains to achieve and maintain a safe shutdown condition. If these criteria are satisfied, no further protection is required for the identified missile. If these conditions are not satisfied, additional protective features are incorporated (for example, plant layout is modified, or barriers are added).

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

3.5.1.1.1 Criteria for Missile Prevention

Equipment for the AP1000 is selected to minimize the potential for missiles to be generated. Missiles are postulated as described in Subsection 3.5.1.1.2. The following items are the major equipment selection considerations with regards to missile prevention:

- Safety-related rotating equipment is designed so that the surrounding housings would contain fragments in the event of failure of the rotating parts.
- Valves that have only a threaded connection between the body and the bonnet are not used in high-energy systems. ASME Code, Section III valves with removable bonnets should be of the pressure-seal type or have bolted bonnets.
- Valve stems of valves located in high-energy systems have at least two retention features. In addition to the stem threads, acceptable features include back seats on the stem or a power actuator, such as an air or motor operator.
- Thermowells and other instrument wells, vents, drains, test connections, and other fittings located in high-energy systems are attached to the piping or pressurized equipment by welding. The completed joint should have a greater design strength than the parent metal. Threaded connections in high-energy systems are avoided.
- High-pressure gas cylinders permanently installed in safety-related areas are constructed to the criteria of ASME Code, Section III or Section VIII. Portable and temporary cylinders and cylinders periodically replaced in safety-related areas are constructed and handled in accordance with applicable Department of Transportation requirements for seamless steel cylinders.

3.5.1.1.2 Missile Selection

3.5.1.1.2.1 Missiles not Considered Credible

This subsection describes internally generated missiles (outside of containment) not considered credible. Missiles not considered credible include the following:

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- Catastrophic failure of safety-related rotating equipment (such as pumps, fans, and compressors) leading to the generation of missiles is not considered credible. These components are designed to preclude having sufficient energy to move the masses of their rotating parts through the housings in which they are contained. In addition, material characteristics, inspections, quality control during fabrication and erection, and prudent operation as applied to the particular component reduce the likelihood of missile generation.

- Catastrophic failure of nonsafety-related rotating equipment is not considered credible in situations where measures similar to those just described for safety-related rotating equipment are applied to them. Protection from nonsafety-related equipment will normally be provided by separation. In special situations, equipment features may be used to prevent missile formation.

- Provisions to preclude generation of missiles due to failure of the turbine generator are discussed in Subsection 3.5.1.3.

- Missiles originating in non-high-energy fluid systems are not considered credible because these systems have insufficient stored energy.

- The valve bonnets of pressure-seal, bonnet-type valves, constructed in accordance with ASME Code, Section III, are not considered credible missiles. The valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke capturing the bonnet or reducing bonnet energy. Because of the conservative design of the retaining ring of these valves, bonnet ejection is unlikely.

- The valves of the bolted bonnet design, constructed in accordance with ASME Code, Section III, are not considered credible missiles. These bolted bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material according to ASME Code, Section III requirements, and by designing flanges in accordance with applicable code requirements. Even if bolt failure would occur, the likelihood of all bolts experiencing simultaneous complete severance failure is not credible. The widespread use of valves with bolted bonnets, and the low historical incidence of complete severance failure of the bonnet, confirm that bolted valve bonnets are not credible missiles. Safety-relief valves in high energy systems use the bolted bonnet design.

- Valve stems are not considered as credible missiles if at least one feature (in addition to the stem threads) is included in their design to prevent ejection. Valve stems with back seats are prevented from becoming missiles by this feature. In addition, the valve stems of valves with power actuators, such as air- or motor-operated valves, are effectively restrained by the valve actuator. Valve stems of rotary motion valves, such as plug valves, ball valves (except single-seat ball valves) and butterfly valves, as well as diaphragm and bellows type valves are not considered as credible missiles. Because these valves do not have a large reservoir of pressurized fluid acting on the valve stem, there is little stored energy available to produce a missile.

- Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are not considered as credible missiles.

- Thermowells and similar fittings attached to piping or pressurized equipment by welding are not considered as credible missiles where the completed joint has a greater design strength than the parent metal. Such a design makes missile formation not credible. Threaded connections are not used to connect instrumentation to high-energy systems or components.

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- Instrumentation such as pressure, level, and flow transmitters and associated piping and tubing are not considered as credible missiles. The quantity of high energy fluid in these instruments is limited and will not result in the generation of missiles. The connecting piping and tubing is made up using welded joints or compression fittings for the tubing. Tubing is small diameter and has only a small amount of stored energy.

- ASME Code, Section III vessel ruptures and ruptures of gas storage vessels constructed without welding using ASME Code, Section VIII criteria are not considered credible due to the conservative design, material characteristics, inspections, quality control during fabrication and erection, and prudent operation.

- Rotating components that operate less than 2 percent of the time are not considered credible sources of missiles. Components that are excluded by this criterion include motors on valve operators and pumps in systems that operate infrequently, such as the chemical and volume control makeup pumps. This exclusion is similar to the exclusion mentioned in Subsection 3.6.1.1, that is, of lines from the high-energy category of lines that have limited operating time in high energy conditions.

- Valves, rotating equipment, vessels, and small fittings not otherwise considered to be credible missiles due to design features or other considerations are not considered to be a potential source of missiles when struck by a falling object.

3.5.1.1.2.2 Explosions

Missiles can potentially be generated by a hydrogen explosion. Missiles that could prevent achieving or maintaining a safe shutdown or result in significant release of radioactivity are precluded by design of the plant systems that use or generate hydrogen.

- The battery compartments are ventilated by a system that is designed to preclude the possibility of hydrogen accumulation. Therefore, a hydrogen explosion in a battery compartment is not postulated.

- The hydrogen injection package (CVS-MS-02) in the turbine building is supplied from the Plant Gas System (PGS) high pressure hydrogen switchover station bottles located in the yard, adjacent to the turbine building. The hydrogen is not located in a compartment that contains safety-related systems or components. The hydrogen is supplied to the chemical and volume control system inside the containment. There are two parallel flow paths: one for “continuous” addition of hydrogen during power operation, and one for “batch” additions of hydrogen during startup and restart. Each line has a control valve and flow transmitter to provide operators with system parameters and control capabilities. The hydrogen injection line break analysis has determined that a worst case failure of the Chemical and Volume Control System (CVS) hydrogen injection line outside the containment building would not lead to an explosive atmosphere in the compartment where the failure occurred.

- Mixing within a compartment is achieved by normal convection caused by thermal forces from hot surfaces and air movement due to operation of HVAC systems. The hydrogen supply line is not routed through compartments that do not have air movement due to HVAC systems.

- The bulk gas plant storage area for the plant gas system (PGS) stores liquid hydrogen for use in generator cooling. This storage area is located sufficiently far from the nuclear island that an explosion would not result in missiles more energetic than the tornado missiles for

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which the nuclear island is designed. The liquid hydrogen is converted to gas in the storage area and then piped to the generator in the turbine building. The turbine building includes sufficient ventilation to prevent an explosive concentration of hydrogen in the event of a leak.

- A detonation of a flammable vapor cloud (delayed ignition) due to the accidental release of hydrogen from the PGS bulk gas storage area or from the high-pressure hydrogen bottles area would not result in missiles more energetic than the tornado missiles for which the nuclear island is designed.

3.5.1.1.2.3 Missiles to be Considered

The following missiles are considered:

- Nonsafety-related rotating equipment, not excluded above,
- Pressurized components, not excluded above, located in high-energy systems
- High pressure gas storage cylinders that may experience a failure of the outlet pipe or valve if accidentally impacted.

3.5.1.1.2.4 Credible Sources of Internally Generated Missiles (Outside Containment)

The consideration of missile sources outside containment that can adversely affect safety-related structures, systems or components is limited to a few rotating components inside the auxiliary building and a few pressurized components in the chemical and volume control system. The safety-related systems and components needed as described in Section 7.4 to bring the plant to a safe shutdown are located inside the containment shield building and auxiliary building, both of which have thick structural concrete exterior walls that provide protection from missiles generated in other portions of the plant. Safety-related systems and components located in the auxiliary building, including the main control room, are protected from missiles generated in other portions of the auxiliary building by the structural concrete interior walls and floors. Protection against potential missiles from the turbine-generator is discussed in Subsection 3.5.1.3.

Rotating components located inside the auxiliary building that are either safety-related or are constructed as canned motor pumps would contain fragments from a postulated fracture of the rotating elements. These are excluded from evaluation as missile sources. Rotating components used less than 2 percent of the time are also excluded from evaluation as missile sources. This exclusion of equipment that is used for a limited time is similar to the approach used for the definition of high-energy systems. Nonsafety-related rotating equipment in compartments surrounded by structural concrete walls with no safety-related systems or components inside the compartment is not considered a missile source. Rotating equipment with a housing or an enclosure that contains the fragments of a postulated impeller failure is not considered a credible source of missiles. For one or more of these reasons the nonsafety-related rotating equipment inside the auxiliary building is not considered to be a credible missile source. Nonsafety-related rotating equipment in compartments with safety-related systems or components that do not provide other separation features have design requirements for a housing or an enclosure to retain fragments from postulated failures of rotating elements.

The high-energy system inside the auxiliary building that includes pressurized components in the high-energy portions that are constructed to standards other than the ASME Code criteria outlined in Subsection 3.5.1.1.1 is the chemical and volume control system. The high-energy portion of this system inside the auxiliary building that is not constructed to ASME Code criteria outlined in Subsection 3.5.1.1.1 is from the makeup pumps to the containment and system isolation valves. The nonsafety-related, high-energy portion of this system is not required to be protected from missiles.

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The nonsafety-related, high-energy portion of the chemical and volume control system is not to be considered a missile source. It includes the design features that are outlined above to exclude components from consideration as missile sources. These considerations include features such as a pump housing or enclosure that contains fragments of a postulated impeller fracture, valve design requirements, vessel design requirements, or enclosure requirements. See Table 3.6-1 for a list of the high-energy systems.

Falling objects (i.e. gravitational missiles) heavy enough to generate a secondary missile are postulated as a result of movement of a heavy load or from a nonseismically designed structure, system, or component during a seismic event. Movements of heavy loads are controlled to protect safety-related structures, systems, and components, see Subsection 9.1.5. Safety-related structures, systems, or components are protected from nonseismically designed structures, systems, or components or the interaction is evaluated. See Subsection 3.7.3.13 for additional discussion on the interaction of other systems with Seismic Category I systems. Valves, rotating equipment, vessels, and small fittings not otherwise considered to be credible missiles due to design features or other considerations are not considered to be a potential source of missiles when struck by a falling object. The outlet pipes and valves for the air storage bottles for the main control room are constructed to the ASME Code, Section III, requirements and are designed for seismic loads. The attached pipes and valves are not credible missile sources due to an accidental impact. The air storage bottles are located within a structural steel frame and are in an area with no activity directly above. For the reasons noted above, secondary missiles are not considered credible missiles.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Selection of equipment for the AP1000 considers provisions to minimize the potential for missiles to be generated. The considerations previously discussed in Subsection 3.5.1.1 are also applicable to equipment inside the containment.

3.5.1.2.1 Missile Selection

3.5.1.2.1.1 Missiles not Considered Credible

Potential missiles are not considered credible when sufficient energy is not available to produce the missile, or by design the probability of creating a missile is negligible. The following are not considered credible sources of internally generated missiles:

- Reactor coolant pump design requirements are established so that any failure of the rotating parts would be retained within the casing at specified overspeed conditions. This is discussed in Subsection 5.4.1.3.6.

- Catastrophic failure of rotating equipment such as pumps, fans, and compressors leading to the generation of missiles is not considered credible as described previously in Subsection 3.5.1.1.2.

- Failure of the reactor vessel, steam generators, pressurizer, core makeup tanks, accumulators, reactor coolant pump castings, passive residual heat exchangers, and piping leading to the generation of missiles is not considered credible. This is due to the material characteristics, preservice and inservice inspections, quality control during fabrication, erection and operation, conservative design, and prudent operation as applied to the particular component.

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- Gross failure of a control rod drive mechanism housing, sufficient to create a missile from a piece of the housing or to allow a control rod to be ejected rapidly from the core, is not considered credible. This is because of the same reasons listed above for the reactor vessel and other components and is based on the following:

  – The control rod drive mechanisms are shop hydrotested to 125 percent of system design pressure.

  – The housings are hydrotested to 125 percent of system design pressure after they are installed on the reactor vessel head. They are included as part of the hydrotest of the completed reactor coolant system.

  – The housings are made of Type 304 or 316 stainless steel, which exhibits excellent notch toughness.

  – The allowable stress levels in the mechanism are not exceeded due to system thermal transients at power or by thermal movement of the coolant loops.

  – The welds in the pressure boundary of the control rod drive mechanism meet the same design, procedure, examination, and inspection requirements as the welds on other ASME Code, Section III, Class 1 components.

  – A nonmechanistic control rod ejection is considered in the safety analyses in Chapter 15 and the design transients in Subsection 3.9.1.1. The integrated head package and control rod drive mechanisms are not designed for the dynamic effects of a missile generated by a rupture of the control rod housing.

- Valves, valve stems, nuts and bolts, and thermowells in high-energy fluid systems and missiles originating in non-high-energy fluid systems are not considered credible missiles as discussed previously in Subsection 3.5.1.1.2.1.

3.5.1.2.1.2 Explosions

Missiles can potentially be generated by a hydrogen explosion. Missiles that could prevent achieving or maintaining a safe shutdown or result in significant release of radioactivity are precluded by design of the plant systems that use or generate hydrogen.

- The hydrogen injection package (CVS-MS-02) in the turbine building is supplied from the Plant Gas System (PGS) high pressure hydrogen switchover station bottles located in the yard, adjacent to the turbine building. The hydrogen is supplied to the chemical and volume control system inside the containment. Hydrogen is injected into the Reactor Coolant System (RCS) at a steady “continuous” flow rate for normal operation and at higher flows for “batch” operation when it is desirable to change the hydrogen concentration in the coolant quickly. In case of a Chemical and Volume Control System (CVS) hydrogen injection line failure inside containment, hydrogen would rapidly dilute due to the mixing with air. The hydrogen injection line break analysis conclusion has determined that a failure of the CVS hydrogen injection line inside the containment would not lead to an explosive atmosphere in the compartment where the failure occurred. Mixing within a compartment is achieved by normal convection caused by thermal forces from hot surfaces and air movement due to operation of HVAC systems.

3.5.1.2.1.3 Missiles to be Considered

The following missiles are considered:

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- Nonsafety related rotating equipment, not excluded above,
- Pressurized components, not excluded above, located in high-energy systems

3.5.1.2.1.4 Evaluation of Internally Generated Missiles (Inside Containment)

The consideration of credible missile sources inside containment that can adversely affect safety-related structures, systems, or components is limited to a few rotating components. The safety-related systems and components needed to bring the plant to a safe shutdown are inside the containment shield building and auxiliary building both of which have thick structural concrete exterior walls that provide protection from missiles generated in other portions of the plant.

Rotating components inside containment that are either safety-related or are constructed as sealless pumps would contain fragments from a postulated fracture of the rotating elements and are excluded from evaluation as missile sources. Rotating components in use less than 2 percent of the time are also excluded from evaluation as missile sources. This exclusion of equipment that is used for a limited time is similar to the approach used for the definition of high-energy systems. This includes the reactor coolant drain pumps, the containment sump pumps and motors for valve operators, and mechanical handling equipment. Non-safety-related rotating equipment in compartments surrounded by structural concrete walls with no safety-related systems or components inside the compartment is not considered a missile source. Rotating equipment with a housing or an enclosure that contains the fragments of a postulated impeller failure is not considered a credible source of missiles. For one or more of these reasons the nonsafety-related rotating equipment inside containment is considered not to be a credible missile source. Non-safety-related rotating equipment in compartments with safety-related systems or components that do not provide other separation features has design requirements for a housing or an enclosure to retain fragments from postulated failures of rotating elements.

The high-energy portions of high-energy systems inside the containment shield building except for a portion of the chemical and volume control system are constructed to the requirements of the ASME Code, Section III. The nonsafety-related, high-energy portion of the chemical and volume control system between the inside containment isolation valves and the outermost reactor coolant system isolation valves is not required to be protected from missiles and is not to be considered a missile source. It includes design features outlined above to exclude components from consideration as missile sources. In addition most of the nonsafety-related portion of the chemical and volume control system is contained in a compartment located away from safety-related equipment. See Table 3.6-1 for a list of the high-energy systems.

Falling objects heavy enough to generate a secondary missile are postulated as a result of movement of a heavy load or from a nonseismically designed structure, system, or component during a seismic event. Movements of heavy loads are controlled to protect safety-related structures, systems, and components (see Subsection 9.1.5). Design and operational procedures of the polar crane inside containment precludes dropping a heavy load. Additionally, movements of heavy loads inside containment occur during shutdown periods when most of the high-energy systems are depressurized. Valves, rotating equipment, vessels, and small fittings not otherwise considered to be credible missiles due to design features or other considerations are not considered to be a potential source of missiles when struck by a falling object. Secondary missiles are not considered credible. Striking a component with a falling object will not generate a secondary missile if design of the component precludes generation of missiles due to pressurization of the component. Safety-related structures, systems, or components are protected from nonseismically designed structures, systems, or components or the interaction is evaluated. Nonsafety-related equipment that could fall and damage safety-related equipment during an earthquake is classified as seismic Category II and is designed and supported to preclude such failure. See Subsection 3.7.3.13 for additional discussion on the interaction of other systems with Seismic Category I systems. There are no high-pressure gas

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storage cylinders inside the containment shield building. For the reasons noted above, secondary missiles are not considered credible missiles.

3.5.1.3 Turbine Missiles

The turbine generator is located north of the nuclear island with its shaft oriented north-south. In this orientation, the potential for damage from turbine missiles is negligible. Safety-related structures, systems and components are located outside the high-velocity, low-trajectory missile strike zone, as defined by Regulatory Guide 1.115. Thus, postulated low-trajectory missiles cannot directly strike safety-related areas.

The turbine and rotor design is described in Section 10.2. Protection is provided by the orientation of the turbine-generator and by the use of robust turbine rotors as described in Section 10.2. The rotor design, manufacturing, and material specification and the inspections recommended for the AP1000 provide an acceptably very low probability (see Subsection 10.2.2) of missile generation. Turbine rotor integrity is discussed in Subsection 10.2.3. This discussion includes fatigue and fracture analysis, material selection, and the maintenance program requirements.

The potential for a high-trajectory missile to impact safety-related areas of the AP1000 is less than 10^{-7}. Based on this very low probability, the potential damage from a high-trajectory missile is not evaluated. The probability of an impact in the safety-related areas is the product of the probability of missile generation from the turbine; the probability, assuming a turbine failure, that a high-trajectory missile would land within a few hundred feet from the turbine (10^{-7} per square foot); and the area of the safety-related area. In the AP1000, the safety-related area is contained within the containment shield building and the auxiliary building.

The potential for a turbine missile from another AP1000 plant in close proximity has been considered. As noted in Subsection 10.2.2, the probability of generation of a turbine missile (or P1 as identified in SRP 3.5.1.3) is less than 1×10^{-5} per year. This missile generation probability (P1) combined with an unfavorable orientation P2 x P3 conservative product value of 10^{-2} (from SRP 3.5.1.3) results in a probability of unacceptable damage from turbine missiles (or P4 value) of less than 10^{-7} per year per plant which meets the SRP 3.5.1.3 acceptance criterion and the guidance of Regulatory Guide 1.115. Thus, neither the orientation of the side-by-side AP1000 turbines nor the separation distance is pertinent to meeting the turbine missile generation acceptance criterion. In addition, the shield building and auxiliary building walls, roofs, and floors, provide further conservative, inherent protection of the safety-related SSCs from a turbine missile.

The turbine system maintenance and inspection program is discussed in Subsection 10.2.3.6.

3.5.1.4 Missiles Generated by Natural Phenomena

Tornado missiles are defined in accordance with Standard Review Plan, Subsection 3.5.1.4. The velocities are adjusted to the maximum wind velocity defined in Section 3.3. The following missiles are postulated:

- A massive high-kinetic-energy missile, which deforms on impact. It is assumed to be a 4000-pound automobile impacting the structure at normal incidence with a horizontal velocity of 105 mph or a vertical velocity of 74 mph. This missile is considered at all plant elevations up to 30 feet above grade. In addition, to consider automobiles parked within half a mile of the plant at higher elevations than the plant grade elevation, the evaluation of the automobile missile is considered at all plant elevations up to the junction of the outer wall of the passive containment cooling water storage tank with the roof of the shield building. This elevation is

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approximately 193 feet above grade. This evaluation bounds sites with automobiles parked within half a mile of the shield building and auxiliary building at elevations up to the equivalent of 163 feet above grade.

- A rigid missile of a size sufficient to test penetration resistance. It is assumed to be a 275 pound, eight inch armor-piercing artillery shell impacting the structure at normal incidence with a horizontal velocity of 105 mph or a vertical velocity of 74 mph.

- A small rigid missile of a size sufficient to just pass through any openings in protective barriers. It is assumed to be a one inch diameter solid steel sphere assumed to impinge upon barrier openings in the most damaging direction at a velocity of 105 mph.

In addition to the missile spectrum specified above, the impact of tornado-driven sheet metal siding on the shield building is evaluated. The evaluation considers siding representative of the siding used on the turbine building, radwaste building, diesel generator building, and portions of the annex building. The evaluation considers a flat steel sheet, which bounds the corrugated siding design used on the buildings adjacent to the nuclear island.

3.5.1.5 Missiles Generated by Events Near the Site

As described previously in Section 2.2, the site interface is established to address site specific missiles as discussed in Subsection 3.5.4. The AP1000 missile interface criteria are based on the tornado missiles described in Subsection 3.5.1.4. Additional analyses are required to evaluate other site specific missiles.

3.5.1.6 Aircraft Hazards

As described previously in Section 2.2, the site interface is established to address aircraft hazards as discussed in Subsection 3.5.4. The AP1000 missile interface criteria are based on the tornado missiles described in Subsection 3.5.1.4. Additional analyses are required to evaluate other site specific missiles. Aircraft crash probability, and the effects of this hazard on the plant, is determined as described in Section 2.2.

3.5.2 Protection from Externally Generated Missiles

Systems required for safe shutdown are protected from the effects of missiles. These systems are identified in Section 7.4. Protection from external missiles, including those generated by natural phenomena, is provided by the external walls and roof of the Seismic Category I nuclear island structures. The external walls and roofs are reinforced concrete. The structural design requirements for the shield building and auxiliary building are outlined in Subsection 3.8.4. Openings through these walls are evaluated on a case-by-case basis to provide confidence that a missile passing through the opening would not prevent safe shutdown and would not result in an offsite release exceeding the limits defined in 10 CFR 50.34.

Where necessary, adjacent structures and/or missile barriers are used to protect openings in the nuclear island building structures. Building structures credited in this evaluation to protect openings in seismic Category I nuclear island building structures are seismic Category I or seismic Category II (see Subsection 3.2.1.1.2). As identified in Subsection 3.7.2, seismic Category II building structures are designed for the safe shutdown earthquake using the same methods and design stress limits as are used for seismic Category I structures. Seismic Category II building structures are also designed to withstand the design basis tornado loads, including missiles, in accordance with the loading combinations identified in Table 3.8.4-2. The evaluation of site-specific hazards for external events that may produce missiles more energetic than tornado missiles is discussed in Subsection 2.2.1. Evaluation of turbine missiles is provided in Subsection 3.5.1.3. Evaluation of tornado missiles is

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provided in Subsection 3.5.1.4. Conformance with regulatory guide recommendations is provided in Appendix 1A.

3.5.2.1 Protection from Externally Generated Missile through Wall 11

Evaluation of the openings in the north wall of the auxiliary building (Wall 11) considered the protection provided by the seismic Category II turbine building first bay building structure and associated missile barriers, consistent with the provisions in Subsections 3.2.1.1.1, 3.2.1.1.2, 3.2.2.5, and 3.2.2.6. Since the turbine building first bay is surrounded by a large amount of structures, equipment and components in the turbine building and adjacent annex building, large deformable missiles (represented by automobile) passing through turbine building and annex building into turbine building first bay is a highly tortuous path. The structures, equipment and components in the turbine building and part of the annex building are credited to stop or break apart an automobile missile and protect Wall 11 from an automobile missile.

In accordance with the missile identification and protection criteria provided in Section 3.5, a realistic assessment of potential missile paths was conducted. Where the line of sight of a missile passing through a turbine building first bay building structure opening could potentially result in missile impact upon a Wall 11 opening, the missile was considered credible, and additional protection was provided or additional evaluation performed. As a result of this analysis, two missile barriers designed to stop the 8-inch artillery shell and 1-inch sphere missiles are provided within the interior of the turbine building first bay building structure, at elevation 117'-6", northeast of MSIV Compartment B, and at elevation 100', north of Valve/Piping Penetration Room. Where the 1-inch diameter solid steel sphere or 8-inch artillery shell missiles were determined to have a line of sight to a Wall 11 penetration, analysis was performed to confirm that there was no line of sight through the penetration, the contents of the penetration would prevent the missile from passing through Wall 11, or equipment required to achieve safe shutdown could not be adversely affected.

The steel tornado missile barriers located within the turbine building first bay are designed and analyzed in accordance with the barrier design procedures contained in Subsection 3.5.3 and applicable ANSI/AISC N690 requirements.

The steel tornado missile barriers located within the turbine building first bay, identified in Table 3.5-1, are designed and analyzed in accordance with the barrier design procedures and ductility requirements contained in Subsection 3.5.3 and applicable ANSI/AISC N690 requirements. The missile barriers are permanently anchored to the turbine building first bay or auxiliary building as applicable. Where anchored to concrete, the anchors conform with the anchorage requirements contained in Subsection 3.8.4.5.1. The configuration of the barriers is based on the lines of sight to be eliminated and the missile sizes pertinent to those lines of sight. The configuration also provides for functional requirements such as airflow, venting, and personnel access. Steel barrier designs are provided in both solid and grating-type configurations to address the required missile protection and functional requirements on a case-by-case basis. The missile protection provided for Wall 11 openings and penetrations is tabulated in Table 3.5-1. Protection of Wall 11 openings from external missiles is provided by the seismic Category II turbine building first bay building structure, the seismic Category II missile barriers located within the turbine building first bay building structure, the seismic Category I Wall 11 doors, and the seismic Category I Wall 11 penetrations and spare penetration covers within missile lines of sight.

3.5.2.2 Protection of MSSV Functions from Externally Generated Missile

The main steam safety valve (MSSV) vent stacks are evaluated for tornado missiles in the horizontal and vertical directions. Based on elevation of the vent stacks, site topography, and arrangement of surrounding buildings, large debris (represented by the automobile) does not have a line of sight to the MSSV vent stacks. For horizontal missile impact, it is analyzed and confirmed that the steam can

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be relieved through Wall 11 doors and vents into turbine building first bay without impacting the pressure limit in the MSIV rooms and Valve/Piping Penetration Room if a vent stack is crimped. For the vertical missile impact, it is evaluated and confirmed that 1) ASME limits are not exceeded under impact of small debris; 2) The main steam pressure boundary remains intact under the impact of the 8-inch artillery shell and 1-inch sphere missiles; 3) the passive safety features are not impacted. Therefore, the tornado missile impact on the MSSV vent stacks does not affect safe shutdown.

3.5.2.3 Protection from Externally Generated Missile through Wall I

Evaluation of the openings in the east wall of the auxiliary building (Wall I) considered the protection provided by the seismic Category II annex building structures and associated missile barriers. Since the auxiliary building is surrounded by a large amount of structures, equipment, and components in the annex building, large deformable missiles (represented by automobile) passing through the annex building into auxiliary building is a highly tortuous path. The structures, equipment and components in the annex building are credited to stop or break apart an automobile missile and protect Wall I from an automobile missile.

In accordance with the missile identification and protection criteria provided in Section 3.5, a realistic assessment of potential missile paths was conducted. Where the line of sight of a missile passing through the annex could potentially result in missile impact upon a Wall I opening or penetration, the missile was considered credible, and additional protection was provided or additional evaluation performed. The rooms with line of sight are 12351, 12421, 12501, and 12541. The missile protection provided for Wall I openings and penetrations is tabulated in Table 3.5-2.

3.5.2.4 Protection from Externally Generated Missile through Shield Building Air Inlets

Evaluation of the openings in the shield building air inlets considered the protection provided by the seismic Category I missile barriers and air baffles, and seismic Category II missile barriers, consistent with the provisions in Subsections 3.2.1.1.1, 3.2.1.1.2, 3.2.2.5, and 3.2.2.6.

In accordance with the missile identification and protection criteria provided in Section 3.5, a realistic assessment of potential missile paths was conducted. Where the line of sight of a missile passing through the shield building air inlets could potentially result in missile impact upon a safe shutdown required SSC, the missile was considered credible, and additional protection was provided or additional evaluation performed. As a result of this analysis, potential missile paths through some of the upper and lower air inlet openings to some PCS piping and Class 1E cabling/conduit are identified. Four missile barriers are provided on the outside of the exterior circular walkway of the shield building to prevent the 1-inch diameter solid steel sphere or 8-inch artillery shell missiles from entering the upper air inlet openings. The external shield building walkway prevents the 1-inch diameter solid steel sphere and 8-inch artillery shell missiles from entering the lower air inlet openings. The automobile missile is too large to enter the air inlets and therefore can be stopped by the shield building wall. The missile protection provided for the shield building air inlet openings is tabulated in Table 3.5-3.

3.5.3 Barrier Design Procedures

Missile barriers and protective structures are designed to withstand and absorb missile impact loads to prevent damage to safety-related components.

Formulae used for missile penetration calculations into steel or concrete barriers are the Modified National Defense Research Committee (NDRC) formula for concrete and either the Ballistic Research Laboratory (BRL) or Stanford formulae for steel.

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Concrete (Modified NDRC Formula)

$$x = \left[4\,KNWd\left(\frac{V}{1000\,d}\right)^{1.8} \right]^{0.5} \qquad \text{for } \frac{x}{d} \le 2.0$$

$$x = KNW\left(\frac{V}{1000\,d}\right)^{1.8} + d \qquad \text{for } \frac{x}{d} > 2.0$$

where

x = penetration depth, inches

W = missile weight, lbs

d = missile diameter, inches

N = missile shape factor = 1.0

V = impact velocity, feet/sec

K = experimentally obtained material coefficient for penetration = $\frac{180}{\sqrt{f_c'}}$

fc′ = concrete compressive strength

Scabbing thickness, t_s, and perforation thickness, t_p is given by:

$$\frac{t_s}{d} = 2.12 + 1.36\frac{x}{d} \qquad \text{for } 0.65 \le \frac{x}{d} \le 11.75$$

$$\frac{t_s}{d} = 7.91\left(\frac{x}{d}\right) - 5.06\left(\frac{x}{d}\right)^2 \qquad \text{for } \frac{x}{d} \le 0.65$$

$$\frac{t_p}{d} = 1.32 + 1.24\frac{x}{d} \qquad \text{for } 1.35 \le \frac{x}{d} \le 13.5$$

$$\frac{t_p}{d} = 3.19\left(\frac{x}{d}\right) - 0.718\left(\frac{x}{d}\right)^2 \qquad \text{for } \frac{x}{d} \le 1.35$$

Steel (Stanford Formula)

$$\frac{E}{D} = \frac{S}{46{,}500}\left(16{,}000\,T^2 + 1{,}500\frac{W}{W_s}T\right)$$

Where:

E = critical kinetic energy required for perforation, foot pounds

D = effective missile diameter, inches

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S = ultimate tensile strength of the target (steel plate), pounds per square inch

T = target plate thickness, inches

W = length of a square side between rigid supports, inches

Ws = length of a standard window, 4 inches

The ultimate tensile strength is directly reduced by the amount of bilateral tension stress already in the target. The equation is good within the following ranges:

$0.1 < T/D < 0.8$,

$0.002 < T/L < 0.05$,

$10 < L/D < 50$,

$5 < W/D < 8$,

$8 < W/T < 100$,

$70 < V < 400$

Where:

L = missile length, inches

V = impact velocity, feet/second

Steel (BRL Formula)

$$t_p = \frac{(E_k)^{2/3}}{672D}$$

Where:

t_p = steel plate thickness for threshold of perforation, inches

D = equivalent missile diameter, inches

E_k = missile kinetic energy, foot pounds

= $M V^2/2$

M = mass of the missile, lb-sec^2/ft.

In using the Modified NDRC, BRL and Stanford formulae for missile penetration, it is assumed that the missile impacts normal to the plane of the wall on a minimum impact area and, in the case of reinforced concrete, does not strike the reinforcing. Due to the conservative nature of these assumptions, the minimum thickness required for missile shields is taken as the thickness just perforated.

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Structural members designed to resist missile impact are designed for flexural, shear, and buckling effects using the equivalent static load obtained from the evaluation of structural response. Stress and strain limits for the equivalent static load comply with applicable codes and Regulatory Guide 1.142, and the limits on ductility of steel structures as given in Subsection 3.5.3.1. The consequences of scabbing are evaluated if the thickness is less than the minimum thickness to preclude scabbing.

The thicknesses of the exterior walls above grade and of the roof of the nuclear island are 24 inches and 15 inches, respectively. The roof is constructed using left-in-place metal deck. These thicknesses exceed the minimum thicknesses for Region II tornado missiles specified in Standard Review Plan 3.5.3.

3.5.3.1 Ductility Factors for Steel Structures

Ductility factors for the design of steel structures are as follows:

- For tension due to flexure, $\mu \leq 10.0$
- For columns with slenderness ratio (L/r) equal to or less than 20, $\mu \leq 1.3$
- For columns with slenderness ratio greater than 20, $\mu \leq 1.0$
  Where: L = effective length of the member
  r = the least radius of gyration
- For members subjected to tension, $\mu \leq .5*(e_u/e_y)$
  Where: e_u = ultimate strain
  e_y = yield strain

3.5.4 Combined License Information

The evaluation for those external events that produce missiles that are more energetic than the tornado missiles postulated for design of the AP1000 is addressed in APP-GW-GLR-020 (Reference 1).

The combined license applicant must evaluate the following information and demonstrate that any exceedances or differences do not compromise the safety of the plant:

- Show site parameters for wind and tornado showing that the site satisfies the AP1000 site interface criteria for the wind and tornado conditions. If there are exceedances, they must be discussed and shown acceptable.
- Provide plant-specific site plan and discuss any differences between the plant-specific site plan and the typical site plan shown in Figure 1.2-2 of Section 1.2.

- Verify that there are no other structures adjacent to the nuclear island other than: turbine building, annex building, radwaste building, and passive containment cooling ancillary water storage tank. If there are, then the effect of the structures on the AP1000 nuclear island must be discussed.

- Show that missiles caused by external events separate from the tornado have energies less than the tornado missile spectrum energies that the AP1000 is designed to withstand. If missile energy is greater, evaluate and show that it will not compromise the safety of AP1000 safety-related structures and components.

3.5.5 References

1. APP-GW-GLR-020, “Wind and Tornado Site Interface Criteria,” Westinghouse Electric Company LLC.

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Table 3.5-1 (Sheet 1 of 2)
External Missile Protection Provided for Auxiliary Building Wall 11 Openings

Wall 11 Opening	Protected Room	Elevation	Missile[1]	Protection[2][3]
Room 12306 doorway	Valve/Piping Penetration Room	100'-0"	Sphere	First bay building structure First bay interior missile barrier
			Artillery Shell	First bay building structure first bay interior missile barrier
			Automobile	First bay building structure Structure, equipment, and components in turbine building and annex building
Room 12404 vent	Lower MSIV Compartment B	117'-6"	Sphere	First bay building structure First bay interior missile barrier
			Artillery Shell	First bay building structure First bay interior missile barrier
			Automobile	First bay building structure Structure, equipment, and components in turbine building and annex building
Room 12405 doorway	Lower VBS B&D Equipment Room	117'-6"	Sphere	Room 12405 Wall 11 door
			Artillery Shell	Room 12405 Wall 11 door
			Automobile	First bay building structure Structure, equipment, and components in turbine building and annex building
Room 12406 vent	Lower MSIV Compartment A	117'-6"	Sphere	First bay building structure
			Artillery Shell	First bay building structure
			Automobile	First bay building structure Structure, equipment, and components in turbine building and annex building
Room 12504 doorway	Upper MSIV Compartment B	135'-3"	Sphere	First bay building structure
			Artillery Shell	First bay building structure
			Automobile	First bay building structure Structure, equipment, and components in turbine building and annex building
Room 12505 doorway	Upper VBS B&D Equipment Room	135'-3"	Sphere	Room 12505 Wall 11 door
			Artillery Shell	Room 12505 Wall 11 door
			Automobile	First bay building structure Structure, equipment, and components in turbine building and annex building
Room 12506 doorway	Upper MSIV Compartment A	135'-3"	Sphere	First bay building structure
			Artillery Shell	First bay building structure
			Automobile	First bay building structure Structure, equipment, and components in turbine building and annex building

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Table 3.5-1 (Sheet 2 of 2)
External Missile Protection Provided for Auxiliary Building Wall 11 Openings

Wall 11 Opening	Protected Room	Elevation	Missile(1)	Protection(2)(3)
Wall 11 penetrations	Various	Various	Sphere	First bay building structure Pipe sleeves Penetration contents Spare penetration covers
			Artillery Shell	First bay building structure Pipe sleeves Penetration contents Spare penetration covers
			Automobile	First bay building structure Structure, equipment, and components in turbine building and annex building

Notes:

1. Tornado missiles are defined in Subsection 3.5.1.4.
2. Turbine building first bay is a seismic Category II building structure. Wall 11 doors, Wall 11 penetration sleeves, and Wall 11 spare penetration covers that provide tornado missile protection are equipment Class C. Barriers in the turbine building first bay building structure that provide tornado missile protection are equipment Class D.
3. The materials of construction for the missile barriers installed within the turbine building first bay and on Wall 11 shall be steel as specified below, or steel with equal or better material properties:
   - First bay interior missile barrier at elevation 117'-6"
     - barrier plate - ASTM A572, Gr. 50
     - barrier support frame - ASTM A500, Gr. B
     - anchors - ASTM A325 or ASTM A490, and ASTM F1554, Gr. 105 as required
   - First bay interior missile barrier at elevation 100'-0"
     - barrier plate - ASTM A572, Gr. 50
     - barrier support frame - ASTM A500, Gr. B
     - anchors - manufacturer standard
   - Wall 11 missile doors
     - door plate - ASTM A572, Gr. 50
     - embed plates - ASTM A572, Gr. 50
     - anchors - ASTM A 1064
   - Wall 11 spare penetration covers
     - ASTM A240
   - Pipe sleeves
     - material shall be as specified for their penetration function

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Table 3.5-2
External Missile Protection Provided for Annex Building Wall I Openings(3)

Wall I Opening	Protected Room	Elevation	Missile(1)	Protection(2)
Room 12351 doorway	Maintenance Floor Staging Area	107'-2"	Sphere	Room 12351 Wall I door
			Artillery Shell	Room 12351 Wall I door
			Automobile	Room 12351 Wall I door
Room 12421 electrical penetrations	Non 1E Equipment/ Penetration Room	117'-6"	Sphere	Pipe sleeves Penetration contents Spare penetration covers
			Artillery Shell	Pipe sleeves Penetration contents Spare penetration covers
			Automobile	Annex Building Wall along column line E
Room 12501 doorway	VBS MCR/A&C Equipment Room	135'-3"	Sphere	Room 12501 Wall I door
			Artillery Shell	Security grating Electrical control panels Room 12501 Wall I door
			Automobile	Structures, equipment, and components in annex building
Room 12541 doorway	Operating Deck Staging Area	135'-3"	Sphere	Room 12556 Wall I door
			Artillery Shell	Room 12556 Wall I door
			Automobile	Room 12556 Wall I door

Notes:
1. Tornado missiles are defined in Subsection 3.5.1.4.
2. Annex Building Areas 1, 2 and 3 are a Seismic Category II building structure. Wall I doors, Wall I penetration sleeves and Wall I spare penetration covers that provide tornado missile protection are equipment Class C. Barriers in the Annex Building Areas 1, 2 and 3 building structures that provide tornado missile protection are equipment Class D.
3. Other Wall I door and penetrations are protected from the tornado missiles by the Seismic Category II Annex Building Structure.

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Table 3.5-3
External Missile Protection Provided for Shield Building Air Inlet Openings

Opening	Protected Room	Elevation	Missile	Protection
Shield Building Lower Air Inlet Openings	Upper Annulus	251'-8"	Sphere	External Shield Building Walkway
			Artillery Shell	External Shield Building Walkway
			Automobile	Shield Building Wall around Air Inlets
Shield Building Upper Air Inlet Openings	Upper Annulus	257'-9"	Sphere	Upper Air Baffle (SB02) External Shield Building Walkway Missile Barriers
			Artillery Shell	Upper Air Baffle (SB02) External Shield Building Walkway Missile Barriers
			Automobile	Shield Building Wall around Air Inlets

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3.6 Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping

The effects of a postulated pipe rupture in the AP1000 are of several types. This section considers the effects that are localized to the area of the break and are a result of the dynamic effects of the pipe rupture including jet impingement, pipe whip, subcompartment pressurization, and fluid system decompression. This section describes the evaluation of the potential for and effects of these dynamic effects. It describes measures taken to protect systems and equipment from dynamic effects of pipe rupture when necessary. This section also considers the effects of spray wetting and flooding from pipe ruptures and cracks.

Chapters 6 and 15 discuss the response of the system to changes in flow and pressure and loss of coolant and the response of the containment to the pressure and temperature changes. Pressure due to a break in a high energy line in the auxiliary building is vented into an adjacent building or to the atmosphere. The design transients listed in Subsection 3.9.1 are used in evaluating the components of the reactor coolant system for effects due to internal pressure and temperature changes from postulated accidents. Section 3.11 discusses the qualification of the equipment required to function in the adverse environmental conditions including temperature, humidity, pressure, radiation, spray wetting, and chemical consequences.

Pipe failure protection is provided according to the requirements of 10 CFR 50, Appendix A, General Design Criterion 4. In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided so that essential structures, systems, or components are not impacted by the adverse effects of postulated piping failure. Essential systems and components are those required to shut down the reactor and mitigate the consequences of the postulated piping failure. Nonsafety-related systems are not required to be protected from the dynamic and environmental effects associated with the postulated rupture of piping except as described in Subsection 3.6.1.1, item Q.

The criteria used to evaluate pipe failure protection are generally consistent with NRC guidelines including those in the Standard Review Plan Sections 3.6.1 and 3.6.2, NUREG-1061, Volume 3 (Reference 11) and applicable Branch Technical Positions.

Subsection 3.6.1 provides the design bases and criteria for the analysis required to demonstrate that essential systems are protected. The high- and moderate-energy systems representing the potential source of adverse effects are listed. Additionally, the criteria for separation and the effects of adverse consequences are defined.

Subsection 3.6.2 defines the criteria for postulated break location and configuration. High-energy pipes are evaluated for the effects of circumferential and longitudinal pipe breaks and through-wall cracks. Moderate-energy pipes are evaluated for the effects of through-wall cracks. Analysis methods and criteria for evaluating pipe whip and evaluating the consequences of jet impingement, motions of the pipe, and system depressurization on integrity and operability are provided. The evaluation of containment penetrations, pipe whip restraints, guard pipes, and other protective devices is also described. The criteria for excluding breaks in high-energy piping adjacent to containment penetrations are also provided.

Evaluation of the dynamic effects of postulated breaks in the reactor coolant loop, main steam lines inside containment, and other primary piping inside containment equal to or greater than the 6-inch nominal pipe size (NPS) is eliminated for AP1000 based on mechanistic pipe break (leak-before-break) considerations. Those sections of high-energy piping that qualify for mechanistic pipe break are evaluated for only the effects of leakage cracks.

Subsection 3.6.3 describes the application of leak-before-break criteria to permit the elimination of pipe rupture dynamic effects considerations. Design guidelines aid in the design of piping systems

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that satisfy the requirements for mechanistic pipe break. Dynamic effects of postulated breaks are evaluated for those analyzable sections of high-energy piping systems that do not use the mechanistic pipe break methods.

The safety analyses in Chapter 15 and the requirements for emergency core cooling discussed in Section 6.3 and the environmental qualification of equipment discussed in Section 3.11 of this report are not changed by the use of mechanistic pipe break considerations for pipe rupture dynamic effects evaluations. Chapter 6 describes the containment subcompartment pressurization analyses including mechanistic pipe break considerations.

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

A number of systems and components are necessary to shut the plant down in the event of a pipe rupture. These systems, termed essential systems, are protected from the postulated pipe ruptures. The essential systems for various pipe ruptures are the reactor coolant system, the steam generator system, the passive core cooling system, and the passive containment cooling system. In addition to these fluid systems, the protection and safety monitoring system and the Class 1E dc and UPS system are essential. The main control room and main control room habitability system are also protected as essential systems. In addition, containment penetrations and isolation valves (including those for nonessential systems) are essential.

Most of the equipment required for plant safety or safety-related shutdown is located inside containment. The piping inside containment also represents the most significant piping relative to plant safety and, therefore, is subject to the most stringent design and analysis requirements.

Essential equipment in the vicinity of piping that does not satisfy leak-before-break criteria is protected as required by the use of protective structures, pipe restraints, and separation. The need for protection of essential structures, systems and components is determined by evaluation of the dynamic effects. The design bases and criteria for the evaluation follow.

Evaluations are made based upon circumferential or longitudinal pipe breaks, through-wall cracks, or leakage cracks as determined by the appropriate criteria. At locations determined to be subject to a circumferential or longitudinal pipe break, dynamic effects such as jet impingement and pipe whip are evaluated.

At locations subject to through-wall cracks, only environmental effects (temperature, humidity, pressure, radiation, spray wetting and chemical consequences) and flooding are evaluated.

The pressurization loads on structures and components are evaluated for postulated circumferential breaks and longitudinal breaks in piping that does not meet leak-before-break requirements and for postulated leakage cracks in piping that meets the leak-before-break requirements. See Subsection 3.8.3.4 and Subsection 3.8.4.3.1.4 for a discussion of pressurization loads on structures.

The in-containment refueling water storage tank is evaluated for pressurization as described in Subsection 3.6.1.2.1.

Pressurization loads for pipe failures in the main steam and feedwater break exclusion zones for high-energy lines in the vicinity of containment penetrations are evaluated for a 1.0 square foot break. Structures in the steam generator blowdown break exclusion zone are evaluated for subcompartment pressurization effects due to worst case circumferential pipe rupture in the 4-inch steam generator blowdown piping. Pipe whip and jet impingement are not evaluated for structures in the break exclusion zones per NRC Branch Technical Position MEB 3-1, section B.1.b, except that the east wall and the floor at elevation 117′-6″ of the east main steam subcompartment is designed

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for pipe whip and jet impingement loads for worst case breaks in either the main steam line or the main feedwater line. See Subsection 3.6.1.2.2.

3.6.1.1 Design Basis

The following design bases relate to the evaluation of the effects of the pipe failures at locations determined in Subsection 3.6.2.

A. The selection of the failure type is based on whether the system is high or moderate-energy during normal operating conditions of the system. High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig. Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high-energy are considered moderate-energy. Piping systems that exceed 200°F or 275 psig for two percent or less of the time during which the system is in operation or that experience high-energy pressures or temperatures for less than one percent of the plant operation time are considered moderate-energy.

B. The following assumptions are used to determine the thermodynamic state in the piping system for the calculation of fluid reaction forces:

   1. For those portions of piping systems normally pressurized during operation at power, the thermodynamic state in the pipe and associated reservoirs is that of normal full-power operation.

   2. For those portions of piping systems pressurized only during other normal plant conditions (for example, startup, hot standby, reactor cooldown), the thermodynamic state and associated operating condition are determined as the mode giving the most severe fluid reaction forces. Moderate-energy systems that are occasionally at higher temperature or pressure (see design basis A.) are not evaluated for pipe failures at the high-energy conditions.

   3. High-stress pipe rupture locations are based on calculated stresses due to Level A and Level B loading. Seismic loads are not included.

C. Circumferential and longitudinal breaks in high-energy pipes, except in pipes satisfying leak-before-break requirements, are evaluated for effects including subcompartment pressurization, pipe whip, jet impingement, jet reaction thrust, internal fluid decompression loads, spray wetting, and flooding.

D. High-energy and moderate-energy pipe through-wall cracks are evaluated for environmental and flooding effects. Moderate-energy systems that are occasionally at higher temperature or pressure (see design basis A.) are evaluated for environmental effects at only the moderate-energy conditions. Dynamic effects are not evaluated for these cracks.

E. Through-wall cracks are not postulated in the break exclusion zones. The effects of flooding, spray wetting, and subcompartment pressurization are evaluated for a postulated 1.0 square foot break for the main steam and feedwater lines.

F. Where postulated, each longitudinal or circumferential break in high-energy fluid system piping, leakage crack in high-energy piping with mechanistic pipe break, or through-wall crack in high-energy or moderate-energy fluid system piping is considered separately as a single initial event occurring during normal plant conditions.

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For systems not seismically analyzed for a safe shutdown earthquake, the safe shutdown earthquake is assumed to cause a pressure boundary failure, as described in Subsections 3.6.2.1.1.3 and 3.6.2.1.2.2.

G. AC power is not required for the actuation of the passive safety systems. The only electrical system required to function is the Class 1E dc and UPS system.

H. A single active component failure is assumed in systems used to mitigate the consequences of the postulated piping failure or to safely shut down the reactor. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power.

I. The function of the containment to act as the ultimate heat sink is maintained for any postulated pipe rupture.

J. Safety-related systems and components are used to mitigate the effects of postulated pipe ruptures. However, the seismic Category II wall separating the first bay of the turbine building from the main area of the turbine building (Wall 11.2) is credited with protecting the north wall of the auxiliary building (Wall 11) from high energy line break (HELB) loads resulting from postulated ruptures in the main steam and main feedwater piping north of the turbine building first bay. In addition, the turbine control and stop, moisture separator reheater 2nd stage steam isolation, and turbine bypass (steam dump) valves (which are not safety-related) are credited in single failure analyses to mitigate postulated steam line ruptures.

K. A whipping pipe is considered capable of rupturing impacted pipes of smaller nominal pipe diameter, irrespective of pipe-wall thickness. This is based on the assumption that only piping is determined to do the impacting. A whipping pipe is considered capable of developing a through-wall crack in a pipe of equal or larger nominal pipe size with equal or thinner wall thickness, assuming that only piping is determined to do the impacting. The preceding criterion is not used where the potential exists for valves or other components in the whipping pipe to impact the targets, since these are treated on a case-by-case basis.

L. Pipe whip is assumed to occur in the plane defined by the piping geometry and to cause movement in the direction of the jet reaction.

If unrestrained, a whipping pipe having a constant energy source sufficient to form a plastic hinge is considered to form a plastic hinge and rotate about the nearest rigid pipe whip restraint, anchor, or wall penetration capable of resisting the pipe whip loads or the calculated dynamic plastic hinge location.

If the direction of the initial pipe movement caused by the thrust force is such that the whipping pipe impacts a flat surface normal to its direction of travel, it is assumed that the pipe comes to rest against that surface, with no pipe whip in other directions.

Pipe whip restraints are provided wherever postulated pipe breaks could impair the capability of any essential system or component to perform its intended safety functions.

M. The calculation of thrust and jet impingement forces considers any line restrictions (that is, flow limiter) between the pressure source and break location and the absence of energy reservoirs, as applicable.

N. Breaks are not postulated to occur in pump and valve bodies since the wall thickness exceeds that of connecting pipe.

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O. Components impacted by jets from breaks in piping containing high-pressure (870 to 2466 psia) steam or subcooled liquid (subcooled no more than 126°F) that would flash at the break, such as piping connected to the steam generators or reactor coolant loops, are evaluated as follows:

1. Impacted components within 10 piping inside diameters of the broken pipe are assumed to fail. Specific jet loads are calculated and evaluated only when failure of the component, when combined with a single active failure, could adversely affect safe shutdown or accident mitigation capability. These jet loads are calculated according to Subsection 3.6.2.3.1.

2. Components beyond 10 inside diameters of the broken pipe are considered to be undamaged by the jet and are not analyzed. The basis for these criteria is contained in NUREG/CR-2913 (Reference 1).

P. Pipe breaks are not postulated to occur in systems for which postulated leakage cracks have been shown to be stable for worst case loadings. (See Subsection 3.6.3.) Leak detection systems are provided that are capable of detecting the leakage from a postulated leakage crack.

For these systems, leakage cracks are postulated and evaluated for subcompartment pressure loads on structures and components. When the mechanistic pipe break approach is used, subcompartment pressure loads on structures and essential components are based on the small leakage crack determined from the mechanistic pipe break approach. Where the subcompartment includes lines not qualified for mechanistic pipe break, subcompartment pressurization is evaluated for a break in the line with the largest effect.

The leakage crack effects of jet impingement, pipe whip, and internal fluid system loads are considered negligible and are not evaluated. The leakage crack effects of flooding and environmental effects are less limiting than the corresponding effects for postulated high-energy through-wall cracks. These through-wall cracks are not eliminated by mechanistic pipe break.

Q. Nonessential systems, structures, and components are not required to meet the criteria outlined in this section. However, while none of the nonessential systems are needed during or following a pipe break event, pipe whip protection is evaluated in cases where a high-energy nonessential system failure could initiate a failure in an essential system or component or where a high-energy nonessential system failure could initiate a failure in another nonessential system whose failure could affect an essential system.

R. The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture will not preclude:

- Subsequent access to any areas, as required, to recover from the postulated pipe rupture
- Habitability of the control room
- Capability of essential instrumentation, electric power supplies, components, and controls to perform safety functions to the extent necessary to meet the criteria outlined in this section

3.6.1.2 Description

Essential systems are evaluated to demonstrate conformance with the design bases and to determine their susceptibility to the failure effects. Table 3.6-1 identifies systems which contain high

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and moderate-energy lines. The systems listed include all high- and moderate-energy systems inside containment plus the high- and moderate-energy systems in the auxiliary building near containment penetrations (including access hatches), the main control room, the Class 1E dc and UPS system or the portions of the passive containment cooling system located in the auxiliary building. The table does not list systems that operate at or close to atmospheric pressure including air handling and gravity drains. High energy system piping in the turbine building adjacent to the auxiliary building is evaluated for potential effects on the main control room. These systems are included on Table 3.6-1.

The definition of high and moderate-energy systems is provided in paragraph A of Subsection 3.6.1.1.

The postulated break, through-wall crack, and leakage crack locations are determined according to Subsections 3.6.2 and 3.6.3.

Equipment is considered to be separated from the dynamic effects of pipe rupture when the equipment is located in a different subcompartment. For the case of pipe whip, equipment may be considered separated for dynamic effects based on the distance from the pipe and the length of pipe that is moving. For the case of jet impingement in a line with saturated or subcooled fluid, equipment more than ten inside pipe diameters from the break location and the tip of the pipe whip trajectory, including the resting location of the broken pipe, is considered separated for dynamic effects.

Equipment located in the same subcompartment as a break, through-wall crack, or leakage crack is subject to potential environmental and flooding effects. Equipment may also be subject to environmental and flooding effects of steam and water vented into a subcompartment from an adjoining subcompartment.

3.6.1.2.1 Pressurization Response

Pressurization response analyses are performed for subcompartments containing high-energy piping for which break locations are defined by Subsections 3.6.2.1.1.1, 3.6.2.1.1.2, and 3.6.2.1.1.3 or postulated leakage flaws are defined based on Subsection 3.6.3.3. Table 3.6-2 identifies those pipe breaks considered for the evaluation of the effects of pressurization loads on subcompartments. The pipe breaks inside containment that are postulated in piping that is not evaluated to the leak-before-break requirements of Subsection 3.6.3 are summarized in Table 3.6-2. The subcompartments are identified using the room numbers and room names given on Figures 1.2-4 through 1.2-10 as supplemented by Table 3.6-2. The subcompartments inside containment are designed to accommodate the pressurization loads from these breaks. In order to account for high stress break locations and the additional pressure boundary leakages from manways and flanges, pressurization loads on compartments inside containment enclosing high-energy piping are designed as described in Subsection 3.8.3.4.

There is no high-energy piping that can pressurize the annulus between the containment vessel and the shield building. Guard pipes are provided for the main steam, feedwater, startup feedwater and steam generator blowdown containment penetrations passing through the annulus as shown on Figure 3.8.2-4. The chemical and volume control system makeup piping is classified as high energy due to its design pressure, but does not cause pressurization because it is at ambient temperature.

The pressurization loads for the in-containment refueling water storage tank are based on the pressure and hydrodynamic loads due to the maximum discharge through the first, second, and third stages of the automatic depressurization system valves.

The pressurization loads for the reactor vessel annulus for the evaluation of asymmetric compartment pressurization are negligible based on a 5-gallon per minute leakage crack in the primary loop piping. The internal reactor pressure vessel asymmetric pressurization loads are based

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on a break in the largest pipe connected to the reactor coolant system that does not qualify for the application of mechanistic pipe break.

There are limited areas in the auxiliary building where the potential for pressurization loads from high-energy lines are considered. The pressurization loads for the steam tunnels are addressed in the discussion of loads due to a break in the break exclusion zone of the main steam and feedwater lines. The pressurization loads for the Elevation 100′ containment penetration room containing the steam generator blowdown break exclusion zone are based on a circumferential rupture of the 4-inch steam generator blowdown piping. The areas through which the chemical and volume control system make-up line run, including the annulus between the containment and the containment shield building, are not subject to pressurization since the temperature of these lines is less than 212°F.

For a discussion of the criteria and analysis methods for subcompartment pressurization analysis, see Subsection 6.2.1.2. The analytical methods for transient mass distribution, used for pressure response analysis, are the same as described in WCAP-8077 (Reference 2).

3.6.1.2.2 Main Control Room Habitability

The high-energy lines in closest proximity to the main control room are the main steam line and main feedwater line. The portions of these lines near the main control room are in the main steam line isolation valve compartment and are part of the break exclusion areas.

The main control room is separated from the isolation valve compartment by two structural walls. The areas between the two walls is used for nonessential office and administrative space associated with the control room. The walls separating the main control room from the main steam isolation valve compartment are thick, reinforced-concrete walls.

Consistent with the criteria for evaluation of leaks in the break exclusion area, the subcompartment, including the walls, is evaluated for the effects of flooding, spray wetting and subcompartment pressurization from a 1-square-foot break from either main steam or feedwater line within the respective break exclusion areas. The wall between the main steam line isolation valve compartment and the main control room, and the floor slab between the main steam line isolation valve compartment and the safety related electrical equipment room are also evaluated for pipe whip and jet impingement loads for worse case breaks in either the main steam line or the main feedwater line. The subcompartment pressure loads from the 1-square-foot break are not combined with the pipe whip and jet impingement loads for the worse case breaks.

The effects upon the habitability of the main control room resulting from postulated pipe breaks and cracks in the auxiliary building are evaluated. In addition to pipe ruptures and cracks in lines in the auxiliary building, the main control room is evaluated for the dynamic effects and environmental effects of a postulated circumferential or longitudinal break of either the main steam line or main feedwater line in the turbine building.

Further description of the control room habitability systems, including options for remote shutdown, is provided in Section 6.4. The remote shutdown workstation is not subject to adverse effects of high-energy pipe rupture.

Main Steam Isolation Valve Compartment De-Pressurization Flow Paths

The MSIV Compartment B (Figures 1.2-7, 1.2-8, 1.2-10, and 1.2-11), which is adjacent to the main control room, houses the main feedwater, startup feedwater, and main steam lines, as they pass to and from primary containment through the shield building, and to and from the turbine building, the MSIVs, the main steam safety valves (MSSVs), the power-operated relief valves (PORVs), the PORV block valves, and other essential components. As a result of a main steam or main feedwater

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high-energy line pipe break in MSIV Compartment B adjacent to the Main Control Room or MSIV Compartment A, the MSIV compartments de-pressurize by venting through the following pathways:

- MSIV Compartment A
  - drains/vents to Turbine Building first bay
  - floor penetrations to Room 12306, which is the Valve/Piping Penetration Room located below, through Room 12306 door to Turbine Building first bay
  - door to Turbine Building first bay
  - roof vent to atmosphere
- MSIV Compartment B
  - drains/vents to Turbine Building first bay
  - door to Turbine Building first bay
  - roof vent to atmosphere

These pressure relief flow paths limit the design pressure loading while maintaining the environmental and structural integrity of the MSIV compartments, adjacent compartments, the Main Control Room, and equipment housed therein.

3.6.1.3 Safety Evaluation

3.6.1.3.1 General

An analysis of postulated pipe failures is performed to determine the impact of such failures on those safety-related systems or components that provide protective actions and are required to mitigate the consequences of the failure. Through such protective measures, as separation, barriers, and pipe whip restraints, the effects of breaks, through-wall cracks, and leakage cracks are prevented from damaging essential items to an extent that would impair their essential function or necessary component operability.

Typical measures used for protecting the essential systems, components, and equipment are outlined in the next subsection and are discussed in Subsection 3.6.2. The capability of specific safety-related systems to withstand a single active failure concurrent with the postulated event is discussed, as applicable. When the results of the pipe failure effects analysis show that the effects of a postulated pipe failure are isolated, physically remote, or restrained by protective measures from essential systems or components, no further dynamic analysis is performed.

3.6.1.3.2 Protection Mechanisms

The plant arrangement is based on maximizing the physical separation of redundant or diverse safety-related components and systems from each other and from nonsafety-related items. Therefore, in the event a pipe failure occurs, there is a minimal effect on other essential systems or components required for safe shutdown of the plant or to mitigate the consequences of the failure.

The effects associated with a particular pipe failure are mechanistically consistent with the failure. Thus, pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the specific measures for protection against the consequences of postulated failures.

Protection against the dynamic effects of pipe failures is provided by physical separation of systems and components, barriers, equipment shields, and pipe whip restraints. The precise method chosen depends largely upon considerations such as accessibility and maintenance. The preferred method of providing protection is by separation. When separation is not practical pipe whip restraints are used. Barriers or shields are used when neither separation nor pipe whip restraints are practical. This protection is not required when piping satisfies leak-before-break criteria.

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Separation

The plant arrangement provides separation, to the extent practicable, between redundant safety systems (including their appurtenances) to prevent loss of safety function as a result of events for which the system is required to be functional. Separation between redundant safety systems, with their related appurtenances, therefore, is the basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

In general, separation is achieved by:

- Safety-related systems located remotely from high-energy piping, where practicable
- Redundant safety systems located in separate compartments, where practicable
- Specific components enclosed to retain the redundancy required for those systems that must function to mitigate specific piping failures
- Drainage systems provided for flooding control
- Pressure relief panels, safety-related, seismically qualified, and fire-rated barrier louvers and flappers to prevent flooding of essential areas

Where physical separation is not possible, the pipe rupture hazard analysis includes an evaluation to determine the systems and components that require a structure for separation from the effects of a break in a high energy line. For these structures specifically included to separate breaks from essential systems or components, the evaluation considers that the break may be at the closest point in the line to the separating structure; not only at the break locations identified in Subsection 3.6.2.1.1. High energy lines qualified as leak-before-break lines and the lines in containment penetration break exclusion areas are not included as possible break locations in this evaluation. For a discussion of the information included in the pipe rupture hazard analysis see Subsection 3.6.2.5.

Barriers and Shields

Protection requirements are met through the protection afforded by walls, floors, columns, abutments, and foundations. Where adequate protection does not already exist as a result of separation, a separating structure such as additional barriers, deflectors, or shields is provided to meet the functional protection requirements.

Inside the containment, the secondary shield wall serves as a barrier between the reactor coolant loops and the containment. In addition, the refueling cavity walls, operating floor, and secondary shield walls minimize the possibility of an accident that may occur in any one reactor coolant loop affecting the other loop or the containment. Those portions of the steam and feedwater lines located within the containment are routed in such a manner that possible interaction between these lines and the reactor coolant piping is minimized. The direct vessel injection valves for train A and train B are separated by the secondary shield wall.

Outside the containment, the wall separating the first bay of the turbine building from the main area of the turbine building (Wall 11.2) is credited with protecting the north wall of the auxiliary building (Wall 11) from HELB loads resulting from postulated ruptures in main steam and main feedwater piping north of the turbine building first bay. The wall is designed and analyzed for the loadings associated with the effects of postulated pipe ruptures in high energy lines north of the turbine building first bay in conjunction with the required load combinations identified in Subsection 3.8.4, and determined to meet the acceptance criteria contained in ACI 349-01 and ANSI/AISC N690-1994.

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Barriers and shields that are identified as required by the pipe rupture hazard analysis are designed for loads from a break in the line at the closest location to the structure. This criterion is in conformance with the guidance of Branch Technical Position MEB 3-1. Rev. 2. Subsection 3.6.2.4 further discusses barriers and shields.

Piping Restraint Protection

Measures for protection against pipe whip are provided where the unrestrained pipe movement of either end of the ruptured pipe could cause damage at an unacceptable level to any structure, system, or components required to meet the criteria outlined in this subsection.

Subsection 3.6.2.3 gives the design criteria for and description of pipe whip restraints.

3.6.1.3.3 Specific Protection Considerations

The analysis of the consequences of pipe breaks, through-wall cracks, and leakage cracks uses the following criteria.

- High-energy containment penetrations are subject to special protection mechanisms. Restraints are provided to maintain the operability of the isolation valves and the integrity of the penetration due to a break in the safety-related and nonsafety piping beyond the restraint if required. These restraints are located as close as practicable to the containment isolation valves associated with these penetrations.

- Instrumentation required to function following a pipe rupture is protected.

- High-energy fluid system pipe whip restraints and protective measures are designed so that a postulated break in one pipe cannot lead to a rupture of other nearby essential pipes or components, if the secondary rupture results in consequences that are unacceptable for the initial postulated break.

For those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event, the turbine control, turbine stop, moisture separator reheater 2nd stage steam isolation, and turbine bypass valves, and to a limited extent, the control systems for the turbine stop and feedwater control valves, are credited in single failure analysis to mitigate the event. This equipment is not protected from pipe ruptures in the turbine building because the postulated pipe rupture for which it provides protection is inside containment. The assumed single active failure for this analysis is the function of the safety-related valve that would normally isolate the piping. This isolation function is addressed in more detail in Chapter 10.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This subsection describes the design bases for locating postulated breaks and cracks in high- and moderate-energy piping systems inside and outside the containment; the procedures used to define the jet thrust reaction at the break location; the procedures used to define the jet impingement loading on adjacent essential structures, systems, or components; pipe whip restraint design; and the protective assembly design. Pipe breaks in several high-energy systems, including the reactor coolant loop and surge line, are replaced by small leakage cracks when the leak-before-break criteria are applied. (See Subsection 3.6.3.) Jet impingement and pipe whip effects are not evaluated for these small leakage cracks.

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3.6.2.1 Criteria Used to Define High- and Moderate-Energy Break and Crack Locations and Configurations

The NRC Branch Technical Position MEB 3-1 is used as the basis of the criteria for the postulation of high-energy pipe breaks and through-wall cracks, except for piping that satisfies the requirements for mechanistic pipe break, as described in Subsection 3.6.3.

A postulated high-energy pipe break is defined as a sudden, gross failure of the pressure boundary of a pipe either in the form of a complete circumferential severance (that is, a guillotine break) or as a sudden longitudinal, uncontrolled crack. For high-energy and moderate-energy fluid systems, pipe failures are also defined by postulation of controlled through-wall cracks in piping. For those piping lines that satisfy leak-before-break requirements, the guillotine breaks and sudden longitudinal cracks are replaced by postulated controlled leakage cracks.

Subsection 3.6.1 describes the evaluation and criteria for the effects of these breaks and cracks on the safety-related equipment.

3.6.2.1.1 High-Energy Break Locations

The locations for postulated breaks in high-energy piping are dependent on the classification, quality group, and design standards used for the piping system. The break locations for high-energy piping are described in the following subsections. These locations are based on the design configuration and include changes due to the as-built piping configuration. As a result of piping reanalysis due to differences between the design configuration and the as-built configuration, the high stress and usage factor location may be shifted. The intermediate break (if any) locations need not be changed unless one of the following conditions exists:

A. The dynamic effects from new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

B. There is a significant change in pipe design parameters such as pipe size, wall thickness or pressure rating.

Breaks are not postulated in piping in the vicinity of containment penetrations. The portion of the piping that does not have postulated breaks is the break exclusion area. Subsection 3.6.2.1.1.4 identifies the requirements for the piping in the containment penetration break exclusion area.

Breaks are not postulated for those sections of pipe, including the reactor coolant loop and pressurizer surge line, that meet the requirements for leak-before-break as described in Subsection 3.6.3.

The leak-before-break methodology is applied to the candidate high-energy lines in the nuclear island identified in Appendix 3E. This appendix also identifies other high-energy lines in the nuclear island with diameters larger than 1 inch and the break exclusion areas inside and outside containment. The evaluation criteria for lines that do not satisfy the leak-before-break criteria are described in Subsection 3.6.2.

3.6.2.1.1.1 ASME Code, Section III, Division 1 – Class 1 Piping

[*Pipe breaks are postulated to occur at the following locations in piping designed and constructed to the requirements for Class 1 piping in the ASME Code, Section III, Division 1.*

- *At terminal ends of the piping, including:*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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- *The extremity of piping connected to structures, components, or anchors that act as essentially rigid restraints to piping translation and rotational motion due to static or dynamic loading.*

- *Branch intersection points are considered a terminal end for the branch line unless the following are met: The branch and the main piping systems are modeled in the same static, dynamic and thermal analyses, and the branch and main run are of comparable size and fixity (that is, the nominal size of the branch is at least one-half of that of the main run).*

- *In piping runs that are maintained pressurized during normal plant conditions for only a portion of the run, the terminal end, for purposes of defining break locations, is the piping connection to the first normally closed valve.*

- *At intermediate locations where the following conditions are satisfied:*

  - *Intermediate locations where the maximum stress range as calculated by Equation (10) of Paragraph NB-3653 of the ASME Code, Section III exceeds 2.4 Sm (where Sm is the design stress intensity), and either Equation (12) or Equation (13) of Paragraph NB-3653.6, exceed 2.4 Sm.*

  - *Intermediate locations where the cumulative usage factor as determined by the ASME Code exceeds 0.1.*

  - *Efforts will be made to avoid intermediate break locations through appropriate piping layout and pipe support design.*

The loading conditions considered for the stress range and usage factors calculated to determine break locations are those defined for Level A and B Service conditions for the piping system with the exception that seismic loads do not need to be considered for the postulation of intermediate break locations.

For those sections of pipe that satisfy the requirements for leak-before-break, leakage cracks are postulated for evaluation of subcompartment pressurization.]*

3.6.2.1.1.2 ASME Code, Section III – Class 2 and Class 3 Piping Systems

[*For those piping system lines designed and analyzed to the requirements of the ASME Code, Section III, Class 2 and 3, except for those sections that satisfy the mechanistic pipe break criteria (subsection 3.6.3), the following criteria apply.*

- *Pipe breaks are postulated to occur at terminal ends, using the same definition for terminal ends as for Class 1 pipe.*

- *Pipe breaks are postulated at intermediate locations between terminal ends where the maximum stress value, as calculated by the sum of Equations (9) and (10) in Subarticle NC-3600 (Class 2) and ND-3600 (Class 3) of the ASME Code, Section III, considering Level A and B Service conditions. That is, breaks are postulated at locations for sustained loads, occasional loads, and thermal expansion exceeding 0.8 (1.8 Sh + SA) or 0.8 (1.5 Sy + SA), where Sh, SA, and Sy are the allowable stress at maximum hot temperature stress, allowable stress range for thermal expansion, and yield strength, respectively, for Class 2 and 3 piping, as defined in Subarticle NC-3600 and Subarticle ND-3600 of the ASME Code, Section III. Efforts will be made to avoid intermediate break locations through appropriate piping layout and pipe support design.*

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For those ASME Code, Section III, Class 2 and 3 systems that satisfy the leak-before-break criteria, postulated leakage crack locations are defined in the same way as for the Class 1 systems.]*

3.6.2.1.1.3 Piping Not Designed to ASME Code

[*Breaks in piping systems designed to requirements other than the ASME Code, such as ASME B31.1 (Reference 3), are postulated at the following locations:*

- *If the piping is analyzed and supported to withstand safe shutdown earthquake loadings, pipe ruptures are postulated to occur at the following locations:*
  - *At terminal ends, using the same definition for terminal ends as for Class 1 pipe*
  - *At intermediate locations where the stresses, as calculated by the sum of Equations (9) and (10) in Subarticle NC3600 of the ASME Code, Section III, considering normal and upset plant conditions, exceeds 0.8 (1.8 Sh + SA) or 0.8 (1.5 Sy + SA)*
  - *Efforts will be made to avoid intermediate break locations through appropriate piping layout and pipe support design.*]*
- In the absence of stress analysis, breaks in non-nuclear piping are postulated at the following locations in each run or branch run:
  - Terminal ends
  - Intermediate fittings; (short- and long-radius elbows, crosses, flanges, nonstandard fittings, tees, reducers, welded attachments, and valves)

3.6.2.1.1.4 High-Energy Piping in Containment Penetration Areas

The AP1000 does not have any ASME Code, Section III Class 1 pipe in containment penetration areas. Breaks are not postulated in the portions of ASME Code, Section III, Class 2 or Class 3 piping, defined below as break exclusion piping, provided subject piping meets the following provisions:

- Stresses do not exceed those specified in Subsection 3.6.2.1.1.2.
- The maximum stress in this piping as calculated by Equation (9), of paragraph NC-3653 of ASME Code Section III, when subjected to the combined loadings of internal pressure, deadweight, and postulated pipe rupture outside the break exclusion zone, does not exceed 2.25 Sh or 1.8 Sy.
- The number of circumferential piping welds is minimized by using pipe bends in place of welding elbows when practicable. There are no longitudinal piping welds in the break exclusion zone. Where guard pipes are used, there are no circumferential or longitudinal welds in the piping enclosed within the guard pipe. Details of the arrangement are shown in Figure 3.8.2-4.
- When required for isolation valve operability, structural integrity, or containment integrity, anchors or five-way restraints capable of resisting torsional and bending moments produced by a postulated pipe break, either upstream or downstream of the piping and valves which form the containment isolation boundary, are located reasonably close to the isolation valves or penetration.

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The anchors or five-way restraints do not prevent the access required to conduct in-service inspection examinations specified in Section XI of the ASME Code. In-service examinations completed during each inspection interval provide 100-percent volumetric examination (according to IWA-2400, ASME Code, Section XI) of circumferential pipe welds within the boundary of these portions of piping during each inspection interval. This volumetric inspection applies to circumferential piping welds that are equal to or greater than a 3-inch nominal diameter and piping branch connection welds that are equal to or greater than 4-inch nominal diameter.

- Welded attachments to these portions of piping for pipe supports or other purposes are avoided. Where welded attachments are necessary, detailed stress analyses are performed to demonstrate compliance with the limits of Subsection 3.6.2.1.1 and applicable requirements of Section XI of the ASME Code.

- The requirements of ASME Code, Section III, Subarticle NE-1120, are satisfied for the containment penetration.

- Class 3 pipe satisfies the fabrication and inspection requirements for Section III, Class 2 pipe.

- For evaluation of spray wetting, flooding, and subcompartment pressurization effects, longitudinal cracks (with crack flow areas of 1 square foot) are postulated in the main steam and main feedwater piping. The dynamic effects of pipe whip and jet impingement are not evaluated for these cracks. Locations having the greatest effect on essential equipment are chosen.

- Guard pipe assemblies for high-energy piping in the containment annulus region between the containment shell and shield building that are part of the containment boundary are designed according to the rules of Class MC, subsection NE, of the ASME Code. The following requirements also apply. The design pressure and temperature are equal to or greater than the maximum operating pressure and temperature of the enclosed process pipe under normal plant conditions. Level C service limits of the ASME Code, Section III, Paragraph NE-3221, are not exceeded by the loadings associated with containment design pressure and temperature in combination with a safe shutdown earthquake. The guard pipe assemblies are subjected to a pressure test performed at the maximum operating pressure of the enclosed process pipe.

Areas of system piping where no breaks, except as noted in Subsections 3.6.1.2.1 and 3.6.1.2.2, are postulated are as follows:

- The main steam piping from the containment penetration flued head inboard weld to the auxiliary building anchor downstream of the main steam isolation valves, including the main steam safety valves and the connecting branch piping

- The main feedwater piping from the auxiliary building side of the containment penetration flued head to the auxiliary building anchor upstream of the isolation valve

- The startup feedwater piping from the auxiliary building side of the containment penetration flued head to the auxiliary building anchor upstream of the isolation valve

- The steam generator blowdown piping from the auxiliary building side of the containment penetration flued head to the auxiliary building anchor downstream of the isolation valve

- The chemical and volume control system makeup piping from the containment penetration flued head to the outboard isolation valve

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- The chemical and volume control system makeup piping from the containment penetration flued head to the inboard isolation valve

For main steam line breaks in the Turbine Building, Wall 11 is capable of resisting torsional and bending moments produced by the postulated pipe breaks. When Wall 11.2 protects Wall 11 against the whipping effects of main steam line breaks, its piping sleeve helps in minimizing axial movement of Wall 11. The axial movement is due to pipe deflection at Wall 11.2, resulting from a main steam piping break in the Turbine Building north of Wall 11.2, and deflection of the main steam piping at Wall 11.2. Wall 11.2, along with its piping sleeve, acts as a barrier which helps in minimizing the lateral displacement of the main steam piping at Wall 11.2. This, in turn, reduces the axial piping displacements at Wall 11 due to pipe deflection at Wall 11.2, resulting in stresses in the break exclusion zone portion of the main steam piping that meet the criteria of Subsection 3.6.2.1.1.4.

Those portions of the containment penetration flued heads identified above that have the same nominal dimensions as the connected pipe are also considered as part of the break exclusion zone piping. The auxiliary building anchors also have flued head designs and the same requirement applies to these.

The main steam and main feedwater containment penetration flued heads are attached to expansion bellows, which are attached to the containment vessel via insert plates (Subsection 3.8.2.1.5, Figure 3.8.2-4, Sheet 1 and Sheet 8, respectively). The function of the expansion bellows is to minimize any piping loads applied to the containment vessel. The containment is not a terminal end for these piping analyses; the terminal ends are the main steam and main feedwater piping anchors in the auxiliary building exterior wall and their respective steam generator nozzles inside containment. The portion of the main steam piping that is inside containment is evaluated to meet the leak-before-break mechanistic pipe break criteria in accordance with Subsection 3.6.3; the portion of the main feedwater piping that is inside containment is analyzed to meet the high-energy pipe break criteria in accordance with Subsection 3.6.2.

All other fluid system containment penetrations are for moderate-energy systems or for pipe of 1-inch nominal diameter or smaller. See Subsection 6.2.3 for a discussion of containment penetrations.

3.6.2.1.2 Types of Breaks/Cracks Postulated

3.6.2.1.2.1 Break in Piping – High-Energy

The following types of breaks are postulated to occur in ASME Code Class 1, 2, and 3 and non-ASME Code, Section III high-energy piping at the locations determined according to Subsection 3.6.2.1.1, except when the leak-before-break criteria are satisfied.

- In piping with a nominal diameter of greater than or equal to 4 inches, both circumferential and longitudinal breaks are postulated at each selected break location unless eliminated by comparison of longitudinal and axial stresses with the maximum stress as follows:

  – If the maximum stress range exceeds the limits specified in Subsections 3.6.2.1.1.1, 3.6.2.1.1.2, and 3.6.2.1.1.3, but the circumferential stress range is at least 1.5 times the axial stress range, only a longitudinal break is postulated.

  – If the maximum stress range exceeds the limits specified in Subsections 3.6.2.1.1.1, 3.6.2.1.1.2, and 3.6.2.1.1.3, but the axial stress is at least 1.5 times the circumferential stress range, only a circumferential break is postulated.

  – Longitudinal breaks, however, are not postulated at terminal ends.

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- In piping with a nominal diameter of greater than 1 inch but less than 4 inches, only circumferential breaks are postulated at each selected break location.

- No breaks are postulated for piping with a nominal diameter of 1 inch or less.

3.6.2.1.2.2 Through-Wall Cracks in High- or Moderate-Energy Piping

Through-wall cracks are postulated in high-energy or moderate-energy piping, including branch runs larger than 1-inch nominal diameter as defined in the following paragraphs:

A. Through-wall cracks are not postulated in the break exclusion areas of high-energy pipe defined in Subsection 3.6.2.1.1.4 and in those portions of moderate-energy piping between containment isolation valves, provided the containment penetration meets the requirements of ASME Code, Section III, Sub-article NE-1120, and the piping is designed so that the maximum stress range based on the sum of equations (9) and (10) in Subarticle NC3600 of the ASME Code, Section III, considering Level A and B Service conditions, does not exceed either 0.4 (1.8 Sh + SA) or 0.4 (1.5 Sy + SA).

B. Through-wall cracks are not postulated in high- or moderate-energy fluid system piping located in an area where a break in the high-energy fluid system is postulated, provided that such cracks do not result in environmental conditions more limiting than the high-energy pipe break.

C. Subject to Paragraphs A and D, through-wall cracks are postulated in:

   - ASME Code, Section III, Division 1 – Class 1 piping where the maximum stress range as calculated by Equation (10) of Paragraph NB-3653 of the ASME Code, Section III exceeds 1.2 Sm. Cracks are also postulated at locations where the cumulative usage factor exceeds 0.1.

   - ASME Code, Section III, Division 1 – Class 2 or 3 piping at locations where the maximum stress range, as calculated by the sum of Equations (9) and (10) in Subarticle NC-3600 (Class 2) and ND-3600 (Class 3) of the ASME Code, Section III, considering Level A and B Service conditions, is greater than 0.4 (1.8 Sh + SA) or 0.4 (1.5 Sy + SA).

   - Seismically analyzed ASME B31.1 piping at locations defined in the same way as ASME Code, Section III, Class 3 piping.

   - Nonseismically analyzed ASME B31.1 piping at the following locations:

     – Terminal ends

     – Intermediate fittings; (short- and long-radius elbows, crosses, flanges, nonstandard fittings, tees, reducers, welded attachments, and valves)

D. Individual through-wall cracks are not postulated at specific locations determined by stress analyses when a review of the piping layout and plant arrangement drawings shows that the effects of through-wall leakage cracks at any location in the piping designed to seismic or nonseismic standards are isolated or are physically remote from structures, systems, and components required for safe shutdown.

E. Through-wall cracks are postulated to be in those circumferential locations that result in the most severe environmental consequences.

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3.6.2.1.2.3 Leakage Cracks in High-Energy Piping with Leak-before-Break

In those sections of piping that satisfy the requirements for leak-before-break, leakage cracks are postulated for evaluation of subcompartment pressurization. The size of the crack is such that the expected leakage is 10 times the minimum leak detection capability for that location. See Subsection 3.6.3 for a discussion of crack size and leakage detection.

3.6.2.1.3 Break and Crack Configuration

3.6.2.1.3.1 High-Energy Break Configuration

Following a circumferential break, the two ends of the broken pipe are assumed to move clear of each other unless physically limited by piping restraints, structural members, or piping stiffness. The effective cross-sectional (inside diameter) flow area of the pipe is used in the jet discharge evaluation. Movement is assumed to be in the direction of the jet reaction initially with the total path controlled by the piping geometry.

The orientation of a longitudinal break, except when otherwise justified by a detailed stress analysis, is assumed to be at opposing points on a line perpendicular to the plane of a fitting for a non-axisymmetric fitting. The flow area of such a break is equal to the cross-sectional flow area of the pipe. The geometry of the longitudinal break may be assumed elliptical (2D along pipe axis and D/2 along pipe transverse) or circular. Both circumferential and longitudinal breaks are postulated to occur, but not concurrently, in high-energy piping systems at the locations specified in Subsection 3.6.2.1.2.1, except as follows:

- Where the postulated break location is at a tee or elbow, the locations and types of breaks are determined as follows:

  – Without the benefit of a detailed stress analysis, such as a finite element analysis, circumferential breaks are postulated to occur individually at each pipe-to-fitting weld. Longitudinal breaks are postulated to occur individually (except in piping with a nominal diameter less than 4-inches) on each side of the fitting at its center and oriented perpendicular to the plane of the fitting, or

  – Alternatively, if a detailed stress analysis or test is performed, the results may be used to predict the most probable rupture location(s) and type of break.

- Where the postulated break location is at a branch/run connection, a circumferential break is postulated at the branch pipe-to-branch fitting weld unless otherwise justified by detailed analysis.

- Where the postulated break location is at a welded attachment (lugs, stanchions), a circumferentially oriented break is postulated at the centerline of the welded attachment unless otherwise justified by a detailed analysis. The break area is equal to the pipe surface area that is bounded by the welded attachment.

- Where the postulated break location is at a reducer, circumferential breaks are postulated at each pipe-to-fitting weld. Longitudinal breaks are oriented to produce out-of-plane bending of the piping configuration on both sides of the reducer at each pipe-to-fitting weld.

3.6.2.1.3.2 High-Energy and Moderate-Energy Through-Wall Crack Configuration

High-and moderate-energy through-wall crack openings are assumed to be a circular orifice with cross-sectional flow area equal to that of a rectangle one-half the pipe inside diameter in length and

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one-half pipe wall thickness in width. The flow from a through-wall crack is assumed to result in an environment that wets unprotected components within the compartment with consequent flooding in the compartment and communicating compartments, unless analysis shows otherwise. Flooding effects are determined on the basis of a conservatively estimated time period required to take corrective actions.

3.6.2.2 Analytical Methods to Define Jet Thrust Forcing Functions and Response Models

To determine the forcing function, the fluid conditions at the upstream source and at the break exit dictate the analytical approach and approximations that are used.

Analytical methods for calculation of jet thrust for the preceding situations are discussed in ANS-58.2-1988 (Reference 4) and Moody, F. J. (Reference 5). The discussion of the jet thrust forcing functions on the reactor coolant loop follows.

Since a rupture of the large-diameter reactor coolant loop piping does not have to be considered, based on satisfying mechanistic pipe break criteria, the jet thrust and reactive loads considered in the analysis are those associated with breaks in branch line sections that do not satisfy the mechanistic pipe break criteria.

To determine the thrust and reactive force loads to be applied to the reactor coolant loop during the postulated pipe rupture, it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the reactor coolant loops as a result of a postulated loss of coolant accident. These forces result from the transient flow and pressure histories in the reactor coolant system (RCS).

The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flowrates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates, and calculates the time-history of forces at appropriate locations in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the entire reactor coolant system. Key parameters calculated by the hydraulic model are pressure, mass flowrate, and density. These are supplied to the thrust calculation, together with plant layout information, to determine the time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated loss of coolant accident, the pressure and momentum flux terms are dominant. The inertia and gravitational terms are taken into account in the evaluation of the local fluid conditions in the hydraulic model.

The blowdown hydraulic analysis provides the basic information concerning the dynamic behavior of the reactor core environment for the loop forces. This requires the ability to predict the flow, quality, and pressure of the fluid throughout the reactor system. [*MULTIFLEX (Reference 6) or an equivalent computer code is used to provide this information.*]*

MULTIFLEX calculates the hydraulic transients within the entire primary coolant system. This hydraulic program considers a coupled, fluid-structure interaction by accounting for the deflection of the core support barrel. The depressurization of the system is calculated using the method of characteristics applicable to transient flow of a homogenous fluid in thermal equilibrium.

The ability to treat multiple flow branches and a large number of mesh points gives MULTIFLEX the flexibility to represent the various flow passages within the primary reactor coolant system. The system geometry is represented by a network of one-dimensional flow passages.

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[*The THRUST computer program or equivalent is used to compute the transient (blowdown) hydraulic loads resulting from a loss of coolant accident.*]*

The blowdown hydraulic loads on primary loop components are computed from the equation:

$$F = 144\,A\left[(P - 14.7) + \left(\frac{\dot{m}^2}{144\,\rho\,g\,(A_m)^2}\right)\right]$$

where:

F = Force (lbf)

A = Aperture area (ft^2)

P = System pressure (psia)

$\dot{m}$ = Mass flowrate (lbm/s)

ρ = Density (lbm/ft^3)

g = Gravitational constant = 32.174 ft-lbm/lbf - s2

A_m = Mass flow area (ft^2)

In the model to compute forcing functions, the reactor coolant loop system is represented by a model similar to that employed in the blowdown analysis. The entire loop layout is represented in a global coordinate system. Each node is described by blowdown hydraulic information and the orientation of the streamline of the force nodes in the system, which includes flow areas and projection coefficients along the three axes of the global coordinate system.

Each node is modeled as a separate control volume with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area.

Each force is divided into its x, y and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, a total y force, and a total z force. These thrust forces serve as input to the piping/ restraint dynamic analysis.

[*The THRUST code calculates forces the same way as the STHRUST code described in WCAP-8252 (Reference 7).*]*

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

This subsection describes the pipe rupture design criteria for auxiliary piping systems. Subsection 3.6.2.2 describes the analysis methods for thrust loadings. To mitigate each postulated pipe rupture, auxiliary piping systems required to maintain pressure boundary integrity or to provide for fluid flow are identified. The loadings on these systems may consist of jet impingement loads, transient motions at terminal end connections, or internal system depressurization loadings.

The application of leak-before-break analysis eliminates evaluation of postulated pipe ruptures in the primary coolant loop piping and selected piping systems of 6-inch nominal size or larger. The piping

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system mechanical components and supports are designed for the effects of the remaining postulated pipe ruptures and leaks.

To confirm the continued integrity of the essential components and the engineered safety systems, consideration is given to the consequential effects of the pipe break to the extent that:

- The minimum performance capabilities of the engineered safety systems are not reduced below that required to protect against the postulated break.
- The containment leaktightness is not decreased below the design value if the break leads to a loss of coolant accident.
- Propagation of damage is limited in type or degree or both to the extent that:
  - A pipe break that is not a loss of coolant accident, steam line break, or main feedwater break will not cause a loss of coolant accident or steam line or feedwater line break.
  - A break in the nonsafety portion of the chemical and volume control system purification loop will not cause a break in the safety-related portion of the system. In addition, the ability to isolate the reactor coolant system flow will not be adversely affected.
  - A reactor coolant system pipe break will not cause a steam or feedwater system pipe break, and vice versa.

3.6.2.3.1 Jet Impingement

Analytical methods for the calculation of jet impingement forces are based on Moody, F. J. (Reference 5), NUREG/CR-2913 (Reference 1), and Section 7.3 of ANS-58.2-1988 (Reference 4). For piping systems this loading is a suddenly applied load that can have significant energy content. These loads are generally treated as statically applied constant loads.

Two separate structural evaluations are performed. For the short-term response, snubber supports are considered to be active and a dynamic load factor of 2 is used. For the longer-term response, snubber supports are considered inactive, and no dynamic load factor is used.

If simplified static analysis is performed instead of a dynamic analysis, the preceding jet load (FT) is multiplied by a dynamic load factor. For an equivalent static analysis of the target structure, the jet impingement force is multiplied by a dynamic load factor of 1.2 to 2.0, depending upon the time variance of the jet load and the elastic/plastic behavior of the target. This factor assumes that the target can be represented as essentially a one-degree-of-freedom system.

3.6.2.3.2 Transient Motions at Terminal Ends

This loading is displacement limited and has a short duration of about 0.5 seconds. An example is the motions of the primary loop piping at the terminal end connection of the Class 1 pressurizer surge line piping due to a postulated pipe rupture in a Class 2 pipe connected to the steam generator.

When there are active in-line components in the piping system that must function to mitigate the postulated pipe rupture, dynamic structural analyses are performed for the terminal end motions. The calculated accelerations are evaluated to confirm the operability of the active in-line components. For piping systems with no active in-line components, static structural analyses with no dynamic amplification are performed for the terminal end motions.

These analyses may consider nonlinear geometric and material characteristics of the piping system.

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3.6.2.3.3 Internal System Depressurization

This loading has a short duration of approximately 0.5 seconds and arises from rapidly traveling pressure waves in piping systems connected to the broken piping system. Two types of configurations are possible: systems without check valves and systems with check valves. In systems with check valves, the valve closure can increase the duration and magnitude of these loads.

An example of the former is the pressure waves in the Class 1 letdown line of the chemical and volume control system piping due to a postulated pipe rupture in a Class 1 pipe connected to the primary loop piping. An example of the latter is the closure of the feedwater check valve due to a postulated pipe rupture upstream of the valve.

For piping systems without closing check valves, there is little energy in the high-frequency depressurization loadings. These loadings are therefore not considered in the piping and support analysis.

For piping system with closing check valves, the magnitude of the loadings depends on the valve closure time, with shorter closing times generally causing higher loadings. For this loading the potential system failure mechanisms evaluated are: 1) excessive pipe and valve hoop stress; 2) tensile loads on the valve pressure boundary bolting; and 3) excessive distortion of the valve disc or seat.

The maximum internal pressure and the kinetic energy of the valve disc at the time of closure are used to verify the pressure boundary integrity of the piping and valve based on the preceding failure mechanisms. RELAP5 is used to calculate the pressure and kinetic energy. The supports on these systems are designed in such a way that support failure will occur prior to local pipe pressure boundary failure at the support connection.

3.6.2.3.4 Pipe Whip Restraints

To satisfy varying requirements of available space, permissible pipe deflection, and equipment operability, the restraints are generally designed as a combination of an energy-absorbing element and a restraint structure suitable for the geometry required to pass the restraint load from the whipping pipe to the main building structure. The restraint structure is typically a structural steel frame or truss, and the energy-absorbing element is usually either stainless steel U-bars or energy-absorbing material.

Another option is for the restraints to be designed as entirely elastic structures and simply transfer the pipe break loads to the main building structure. In addition, pipe supports (including snubbers, struts, or rigid frames) can be used as pipe whip restraints when qualified for the pipe break loads.

3.6.2.3.4.1 Location of Pipe Whip Restraints

For purposes of determining pipe hinge length and thus locating the pipe whip restraints, the plastic moment of the pipe is determined in the following manner:

$$M_p = 1.1\, z_p S_y$$

where:

z_p = Plastic section modulus of pipe

S_y = Yield stress at pipe operating temperature

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1.1 = 10-percent factor to account for strain hardening.

Pipe whip restraints are located as close to the axis of the reaction thrust force as practicable. Pipe whip restraints are generally located so that a plastic hinge does not form in the pipe. If, because of physical limitations, pipe whip restraints are located so that a plastic hinge can form, the consequences of the whipping pipe and the jet impingement effect are further investigated. Lateral guides are provided where necessary to predict and control pipe motion.

Generally, pipe whip restraints are designed and located with sufficient clearances between the pipe and the restraint in such a way that they do not interact and cause additional piping stresses. A design hot position gap is provided that allows maximum predicted thermal, seismic, and seismic anchor movement displacements to occur without interaction.

Exception to this general criterion may occur when a pipe support and restraint are incorporated into the same structural steel frame, or when a zero design gap is required. In these cases, the pipe whip restraint is included in the piping analysis and designed to the requirements of pipe support structures for all loads except pipe break and designed to the requirements of pipe whip restraints when pipe break loads are included.

In general, the pipe whip restraints do not prevent the access required to conduct in-service inspection examination of piping welds. When the location of the restraint makes the piping welds inaccessible for in-service inspection, a portion of the restraint is designed to be removable to provide accessibility.

3.6.2.3.4.2 Analysis and Design of Pipe Whip Restraints

The criteria for analysis and design of pipe whip restraints for postulated pipe break effects are provided in the following. These criteria are consistent with the guidelines in ANS-58.2-1988 (Reference 4).

- Pipe whip restraints are designed based on energy absorption principles by considering the elastic-plastic, strain-hardening behavior of the materials used.

- Non-energy absorbing portions of the pipe whip restraints are designed to the requirements of ANSI/AISC N690 Code supplemented by the requirements given in Subsection 3.8.4.5. American Welding Society (AWS), Structural Welding Code-Steel, AWS D1.1-2000 provides an acceptable alternative for ANSI/AISC N690 weld requirements as described in Subsections 3.8.3.2 and 3.8.4.2.

- Standard pipe support components, such as struts and snubbers, used as pipe whip restraints are designed to ASME Section III, Subsection NF, Service Level D allowables.

- Loads on pipe whip restraints are developed either from an energy balance method (as described in ANS-58.2) with a 1.1 rebound factor or from dynamic time-history analyses.

- Except in cases where calculations are performed to verify that a plastic hinge is formed, the energy absorbed by the ruptured pipe is conservatively assumed to be zero. That is, the thrust force developed goes directly into moving the broken pipe and is not reduced by the force required to bend the pipe.

- Other structural members of the pipe whip restraints are designed for elastic response. A dynamic increase factor is used for those members that are designed to remain elastic.

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- The criteria for allowable strain in a pipe whip restraint are dependent on the type of restraint. The following discussions address the types of restraints used and the allowable strain for each. Note -ε = allowable strain used in design.

Energy Absorbing Materials

Stainless Steel U-Bar – This type of restraint consists of one or more U-shaped, upset-threaded rods of stainless steel looped around the pipe but not in contact with the pipe. This allows unimpeded pipe motion during seismic and thermal movement of the pipe. At rupture, the pipe moves against the U-bars, which absorb the kinetic energy of pipe motion by yielding plastically. Figure 3.6-1 shows a typical example of a U-bar restraint.

$$\varepsilon = 0.5\varepsilon u$$

where:

εu = ultimate uniform strain of stainless steel (strain at ultimate stress)

Crushable Pipe – The crushable pipe type of restraint consists of a short section of standard pipe (length at least three times diameter of crushable pipe) that is situated approximately perpendicular to the process pipe and whose longitudinal axis is approximately perpendicular to the plane of pipe movement. The crushable pipe is attached to a supporting structure. The crushable pipe is the part of the assembly that absorbs the kinetic energy that the broken pipe has accumulated after rupture, and the crushable pipe typically undergoes considerable plastic deformation during the pipe whip event. A design hot position gap is provided between the process pipe and the energy-absorbing crushable pipe to allow unimpeded pipe motion during seismic and thermal pipe movements. Figure 3.6-2 shows typical examples of a crushable pipe type whip restraint. The allowable capacity of the crushable pipe is limited to 80 percent of its rated energy dissipating capacity as determined by dynamic testing, at loading rates within ± 50 percent of the specified design loading rate. An additional conservative measure is used to maintain ductile yield of the energy absorbing pipe by limiting flattening of the crushed pipe to the limits provided in ASTM A530. The rated energy dissipating capacity is not greater than the area under the load-deflection curve as illustrated in Figure 3.6.2-1 of NUREG-0800, Standard Review Plan, Section 3.6.2, Revision 2.

3.6.2.4 Protective Assembly Design Criteria

In addition to pipe whip restraints, other protective devices are designed to protect against the effects of postulated pipe ruptures. Barriers and shields are designed to protect against jet impingement. Guard pipes in the break exclusion zones provide additional confidence that pipes will not leak into the annulus between the containment vessel and the shield building.

3.6.2.4.1 Jet Impingement Barriers and Shields

Barriers and shields, constructed of either steel or concrete, are provided to protect essential equipment, including instrumentation, from the effects of jet impingement resulting from postulated pipe breaks. Barriers differ from shields in that they may also accept the impact of whipping pipes. Barriers and shields include walls, floors, and structures specifically designed to provide protection from postulated pipe breaks. Barrier and shield design is based on elastic methods and the elastic-plastic methods for dynamic analysis included in Biggs, J. M. (Reference 9). Design criteria and loading combinations are according to Subsections 3.8.3 and 3.8.4.

3.6.2.4.2 Auxiliary Guardpipes

The use of guard pipes has been minimized by plant arrangement and routing of high-energy piping. Guard pipes in the containment annulus areas of the break exclusion zones are designed as

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described in Subsection 3.6.2.1.1.4. Other guard pipes are designed and constructed to the same ASME rules as the enclosed process pipe.

3.6.2.5 Evaluation of Dynamic Effects of Pipe Ruptures

The preceding information provides the criteria and methods for the evaluation of the dynamic effects of pipe ruptures. The pipe rupture hazard analysis report (also referred to as the pipe break evaluation report) includes the following:

- Prepare a stress summary
- Identify pipe break locations in high energy piping
- Identify through-wall crack locations in high and moderate energy piping
- Identify and locate essential structures, systems, and components
- Evaluate consequences of pipe whip and jet impingement

  For rooms with both high energy breaks and essential items, confirm that there is no adverse interaction between the essential items and the whipping pipe or jet.

  The plant layout is modified as required to provide separation to protect essential systems.
- Evaluate consequences of flooding, environment, and compartment pressurization

  Evaluate compartment pressurization in the break exclusion zones in the vicinity of containment penetrations due to 1.0 square foot breaks in the main steam and feedwater lines.
- Design and locate protective hardware
- Prepare isometric piping sketches that identify the break locations, the basis for these locations and the protective hardware which mitigates the consequences of these breaks.
- Reconciliation of as-built condition

Pipe breaks that are larger than 1-inch nominal diameter are evaluated for pipe whip and jet impingement. Lines that are located in a break exclusion zone or are qualified to leak-before-break are not evaluated for pipe whip and jet impingement effects on systems and components, except for the portions of the lines in the MSIV compartment adjacent to the main control room as noted in Subsection 3.6.1.2.2.

Where these systems are qualified for mechanistic pipe break and pipe rupture loads prior to fabrication, the qualification is based on design information, not on as-built information. As-built information and the final configuration of valves and other equipment is used to verify the design analysis.

High Energy Break Locations

High energy break locations evaluated are on the nuclear island and in the turbine building for evaluation of the wall loadings in the south end of the turbine building adjacent to the main control room.

For ASME Class 1 piping terminal end locations are determined from the piping isometric drawings. Intermediate break locations depend on the ASME Code stress report fatigue analysis results. These

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results were not available at design certification. For the design of the AP1000, breaks are postulated at locations typically associated with a high cumulative fatigue usage factor. These locations are at valves, tees, and branch connections which have significant structural discontinuities. These locations are part of the as-built reconciliation as discussed in Subsection 3.6.4.1. The following ASME Class 1 lines are evaluated to terminal end and intermediate high energy break locations if applicable.

Line	Diameter (inches)
Pressurizer Spray	4
Automatic Depressurization Stage 1	4
Chemical and Volume Control Letdown	3
Chemical and Volume Control Makeup	3
Reactor Coolant Pump Piping to/from Heat Exchangers (2/pump)	3
Pressurizer Auxiliary Spray	2

For ASME Class 2 and 3 piping, terminal end break locations are determined from the piping isometric drawings. The intermediate break locations depend on the stress level. The AP1000 ASME Class 2 and 3 lines do not have intermediate breaks based on the low stress. The following ASME Class 2 and 3 lines have terminal end high energy break locations.

Line	Diameter (inches)
Main Feedwater	16, 20
Startup Feedwater	6
Steam Generator Blowdown	4

For ASME B31.1 piping, terminal end break locations are determined from the piping isometric drawings. The intermediate break locations in seismically analyzed pipe depend on the stress level. The AP1000 ASME seismically analyzed ASME B31.1 piping does not have intermediate breaks based on the low stress. For nonseismically analyzed high-energy ASME B31.1, intermediate breaks locations are postulated at each fitting.

Rooms subject to pressurization due to high energy pipe break are listed in Table 3.6-2 with the break location.

Essential Systems and Components

In rooms that contain high energy pipe breaks, the systems and components that are needed to mitigate the postulated break and achieve a safe plant shutdown are identified. Rooms that contain both high energy pipe break locations and essential systems or components that must be protected are listed in Table 3.6-3. No high energy pipe break protection is required in other areas of the plant.

Essential Target Evaluation

To complete the essential target evaluation jet parameters, volumetric area of affected compartments, plant layout, and separating structures are considered. Parameters that determine the shape of the jet and the magnitude of the jet and thrust loads include pressure, temperature, and friction losses between the break and the reservoir. The volumetric area affected is determined by considering jet shape and loads at the postulated location of the breaks. Where an initial evaluation of essential targets indicated adverse effects, layout may be changed to relocate the target or

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postulated break. If necessary, the location of whip restraints and jet shields is established to protect essential systems and components. Essential equipment protected by pipe whip restraints or jet shields is listed in Table 3.6-3. The criteria for the break location postulated for evaluation of separating structures is outlined in Subsection 3.6.1.3.2.

Verification of the Pipe Break Hazard Analysis

A pipe rupture hazard analysis is prepared based on the as-designed piping stress analyses and pipe whip restraint design information. The as-designed piping analysis is based on piping routings, layouts, and isometrics. Intermediate break locations are identified using the as-designed piping stress analysis, including the fatigue analysis required for ASME Code Class 1 piping. As-designed piping stress analysis information is used to confirm the location and configuration of pipe whip restraints and jet impingement shields. The information included in Tables 3.6-2 and 3.6-3 is updated and validated as part of the as-designed pipe rupture hazard analysis. Large leakage cracks in moderate energy pipes are evaluated for adverse effects as part of the pipe break hazard evaluation.

The ASME Code, Section III, requires that each plant have a Design Report for the piping system that includes as-built information. Included in the Design Reports are the loads and loading combinations used in the analysis. Where mechanistic pipe break requirements are used to eliminate the evaluation of dynamic effects of pipe rupture in ASME Code, Section III, Class 1, 2, and 3 piping system, the basis for the exclusion is documented in the Design Report.

The final piping stress analyses, pipe whip restraint design, and as-built reconciliation of the pipe break hazard analysis is discussed in Subsection 3.6.4.1. The final piping stress analysis includes design properties and characteristics of procured components selected to be included in the piping system that are not available for the as-designed evaluation. The as-built reconciliation is required prior to fuel loading and includes evaluation of the ASME Code fatigue analysis, pipe break dynamic loads, reconciliation to the certified design floor response spectra, confirmation of the reactor coolant loop time history seismic analyses, changes in support locations, preoperational testing, and construction deviations.

3.6.2.6 Evaluation of Flooding Effects from Pipe Failures

The effect of flooding due to high and moderate energy pipe failures on essential systems and components is described in Section 3.4.

3.6.2.7 Evaluation of Spray Effects from High- and Moderate-Energy Through-Wall Cracks

Essential systems and components are evaluated for the potential effects of spray from high- and moderate-energy through-wall cracks. Spray effects are assumed to be limited to the compartment where the pipe failure occurs. The spray is assumed to wet unprotected components in the compartment. It is further assumed the spray does not damage non-electrical passive components, including piping, ducts, valve bodies, or mechanical components of valve operators. Spray may cause failure of electrical components not designed to withstand wetting. Components protected by NEMA 4 or NEMA 12 enclosures are not affected by spray effects.

The safe shutdown components inside containment are subject to wetting from design basis events inside containment. These conditions bound the effects of spray from moderate energy cracks. Sensitive components are qualified for this environment as described in Section 3.11.

The doors to the auxiliary Class 1E battery rooms are normally closed, so spray cannot affect the batteries if fire fighting activities or a pipe crack were to occur in the corridor. If fire fighting activities were to occur in a particular room, all of the equipment is assumed inoperable due to the fire, therefore, no further spray effects need be considered. The containment isolation valves subject to

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spray and the safe shutdown components in the main steam tunnels are evaluated for the consequence of spray; and, if essential functions are compromised, are provided with spray protection. The sensitive components of the main control room emergency habitability system are protected from spray effects.

3.6.3 Leak-before-Break Evaluation Procedures

This subsection describes the design basis for mechanistic pipe break (leak-before-break) evaluation of high-energy piping systems.

Mechanistic pipe break evaluations demonstrate that for piping lines meeting the criteria, sudden catastrophic failure of the pipe is not credible. It is demonstrated that piping that satisfies the criteria leaks at a detectable rate from postulated flaws prior to growth of the flaw to a size that would fail because of applied loads resulting from normal conditions, anticipated transients, and a postulated safe shutdown earthquake.

The use of mechanistic pipe break criteria represents a higher level of confidence of the integrity of piping systems based on additional criteria compared to the existing high level of integrity provided by the requirements of the ASME Code. Evaluations of the mechanistic pipe break criteria are commonly called leak-before-break evaluations.

The use of mechanistic pipe break criteria permits the elimination of the evaluation of dynamic effects of sudden circumferential and longitudinal pipe breaks in the design basis analysis of structures, systems, and components. General Design Criterion 4 of Appendix A, 10 CFR Part 50 allows the use of analyses to eliminate from the design basis the dynamic effects of pipe ruptures.

Without the application of mechanistic pipe break criteria, the dynamic effects are evaluated for pipe ruptures postulated at locations defined in Subsection 3.6.2. Dynamic effects include jet impingement, pipe whip, jet reaction forces on other portions of the piping and components, subcompartment pressurization including reactor cavity asymmetric pressurization transients, pump overspeed and traveling pressure waves from the depressurization of the system.

Incorporating leak-before-break criteria and guidelines into the design process maximizes the benefits of applying mechanistic pipe break. Eliminating the dynamic effects permits minimizing the size and number of protective structures and eliminates the use of pipe whip restraints. This permits design optimization and avoids obstruction of pipe welds for in-service inspection by protective structures and restraints.

High-energy ASME Code Section III piping that is evaluated to the leak-before-break criteria is identified in Appendix 3E. This applies to the main steam piping as follows. The main steam piping from the steam generator outlet nozzle to the anchor downstream of the isolation valve is analyzed for applicable loadings including the safe shutdown earthquake. This anchor is at the exterior wall of the auxiliary building. The portion of this piping from the containment penetration flued head inboard weld to the above anchor satisfies the break exclusion zone requirements described in Subsection 3.6.2. The portion of this piping from the steam generator outlet nozzle to flued head inboard weld, including the welds, is evaluated to the leak-before-break criteria. The portion of the auxiliary building flued head (anchor in the wall) that has the same nominal dimensions as the main steam pipe is also classified as a break exclusion zone. High-energy piping that does not satisfy the leak-before-break criteria is designed to the requirements discussed in Subsections 3.6.1 and 3.6.2.

The piping to which mechanistic pipe break is applied is analyzed to demonstrate that the piping has leak-before-break characteristics. The leak-before-break analysis is either a fracture-mechanics based stability analysis or a plastic-instability limit load analysis as appropriate. The analysis combines normal and abnormal (including seismic) loads to determine a critical crack size for a

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postulated through-wall crack. The critical crack size is compared to the size of a leakage crack for which, with appropriate margin, detection is certain. When the critical crack size is sufficiently larger than the leakage crack size the leak-before-break requirements are satisfied.

Mechanistic pipe break is not used for purposes of specifying non-structural design criteria for emergency core cooling, containment systems, or other non-structural engineered safety features, or for the evaluation of environmental effects including spray wetting, humidity, temperature, pressure, radiation, and adverse reactions with chemicals in the coolant. This includes piping for which leak-before-break is demonstrated.

A bounding analysis is performed for each piping system. The bounding analysis is applied as discussed in Subsection 3.6.4.2 to verify that the as-built piping satisfies the requirements for leak-before-break.

3.6.3.1 Application of Mechanistic Pipe Break Criteria

Piping systems to which mechanistic pipe break are applied are high integrity systems with well understood loading combinations and conditions. The piping systems to which it is applied satisfy the requirements of the ASME Code, Section III. ASME Code requirements also apply to the pre-service and in-service inspection which confirm continued integrity.

The mechanistic pipe break approach is applicable to high-energy piping provided plant design, operating experience, tests, or analyses have indicated low probability of failure from effects of intergranular stress corrosion cracking, water hammer, steam hammer, fatigue (thermal or mechanical), or erosion.

The plant design and operating features permit the application of the mechanistic pipe break approach. The piping to which the leak-before-break criteria is applied is evaluated for fatigue due to cyclic loads as required by the appropriate requirements of the ASME Code.

The piping in the AP1000 does not operate at temperatures for which creep or creep fatigue must be considered.

The reactor coolant loop piping, branch lines, and other lines in contact with reactor coolant are fabricated of austenitic stainless steel, which is very resistant to erosion and corrosion in typical reactor coolant chemistries and flowrates. Intergranular stress corrosion cracking has not been associated with reactor coolant piping in pressurized water reactors.

The design of the reactor coolant loop is not conducive to the generation of water hammer loads. The reactor coolant loop does not have any valves that could result in a water hammer due to rapid valve closure. The steam bubble in the pressurizer is not subject to the introduction of a large volume of cold water sufficient to result in a bubble collapse water hammer.

The design and component selection of reactor coolant branch lines and other lines evaluated for mechanistic pipe break follow design guidelines intended to minimize the potential for water hammer. Comparison of the AP1000 piping to the screening criteria in Subsection 5.29 of NUREG/CR-6519 (Reference 13) demonstrates that there is not a significant potential for water hammer in the leak-before-break piping.

Thermal stratification of water in stagnant or slowly flowing lines can result in thermal fatigue in a pipe. The piping and system design requirements for AP1000 address the potential for thermal stratification. For additional information of thermal stratification, see Subsections 3.9.3, 5.4.3, and 5.4.5.

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The composition of the main steam lines has been selected to minimize the potential for erosion and corrosion. The main steam lines are fabricated from SA335 Grade P11 Alloy steel, which is composed of sufficient levels of chromium to preclude erosion and corrosion mechanisms. The main steam lines are not subject to water hammer or thermal stratification by the nature of the fluid transported.

The steam line is protected from being filled with water due to steam generator overfill by implementation of operating instructions or isolation requirements included in the protection system logic or both. See Section 7.3 for information on the protection system design to prevent overfill.

In addition to requirements on the design, fabrication, and inspection of the piping systems, the application of mechanistic pipe break requires a qualified leak detection capability. Leak detection systems inside containment meet the guidelines of Regulatory Guide 1.45. See Subsection 5.2.5 for a discussion of the leak detection system for the reactor coolant system and connected piping.

3.6.3.2 Design Criteria for Leak-before-Break

The methods and criteria to evaluate leak-before-break in the AP1000 are consistent with the guidance in NUREG-1061 (Reference 11) and Draft Standard Review Plan 3.6.3 (Reference 12) The application of the mechanistic pipe break in AP1000 requires that the following design requirements are met.

- Pre-service inspection of welds is required.
- For ASME Code Class 1, Class 2, and Class 3 systems for which leak-before break is demonstrated, the ASME Code, Section III and Section XI preservice and inservice inspection requirements will provide for the integrity of each system. The weld and welder qualification, and weld inspection requirements for ASME Code, Section III, Class 3 leak-before-break lines are equivalent to the requirements for Class 2. The inservice inspection requirement for each Class 3 leak-before-break line includes a volumetric inspection equivalent to the requirements for Class 2 for the weld at or closest to the high stress location.
- Inservice inspection and testing of snubbers (if used) are performed to provide for a low snubber failure rate.
- For the maximum stress due to steady-state vibration refer to Subsection 3.9.2.
- The leak-before-break bounding analysis curves are developed for each applicable piping system. The bounding analysis methods are described in Appendix 3B. These curves give the design guidance to satisfy the stress limits and leak-before-break acceptance criteria. The highest stressed point (critical location) determined from the piping stress analysis is compared to the bounding analysis curve and has to fall on or under the curve. The points on or under the bounding analysis curve satisfy the requirements for leak-before-break.

  The analyzed normal stress and maximum stress are not required to construct the bounding analysis curve. The analyzed stresses are calculated by the equation;

$$\sigma = \frac{F_x}{A} + \frac{M}{Z}$$

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where:

σ is the stress

F_x is the axial force

M is the applied moment

A is the piping cross-sectional area

Z is the piping section modulus.

The normal stress is calculated by the algebraic summation of load combination method and the maximum stress is calculated by the absolute summation of load combination method.

- The corrosion-resistant piping materials, including base metal and welds, have an appropriate toughness. The piping materials containing primary coolant are wrought stainless steel. The welds in stainless steel pipe are made using the gas tungsten arc (GTAW) process. These materials are very resistant to crack extension. The tensile properties for the leak-before-break evaluation are those found in the Section II Appendices of the ASME Code. During the design stage, the material properties used are based on the ASME Code minimum values. During the as-built reconciliation stage, certified material test report values are reviewed to verify that ASME Code requirements are satisfied.

- For those lines fabricated using non-stainless ferritic materials, the materials used and the associated welds have adequate toughness to demonstrate that leak-before-break criteria are satisfied. The welds are made using the gas tungsten arc (GTAW) process. The tensile properties for the leak-before-break evaluation are obtained from actual material tests. During the design stage, the material properties are based on test results. During the as-built reconciliation stage, certified material test report values are reviewed to verify that the toughness and strength requirements of the ASME Code, Section III are satisfied.

- Potential degradation by erosion, erosion/corrosion and erosion cavitation is examined to provide low probability of pipe failure.

- Wall thicknesses in elbows and other fittings are evaluated to confirm that ASME Code, Section III piping requirements are met as a minimum.

- The as-built condition of the piping and support system is evaluated based on the guidelines in EPRI NP-5630 (Reference 10) and reconciled to the analysis of the leak-before-break criteria based on the design information. The locations and characteristics of the supports, including any gaps between the supports and piping, or other configurations that result in a nonlinear response are included in the as-built evaluation.

- Adjacent structures and components are designed for the safe shutdown earthquake event to provide low probability of indirect pipe failure.

- The piping supports are anchored to reinforced concrete structures, to concrete-filled steel plate structures, or to steel structures anchored to these types of structures. Piping is not supported by masonry block walls.

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3.6.3.3 Analysis Methods and Criteria

The methods used to develop the bounding analysis curves are described in Appendix 3B. Development of the bounding analysis curves provides an evaluation method that is consistent with NRC requirements and guidance. The calculation method and computer codes used for AP1000 are benchmarked to test data and have been previously accepted by the NRC for leak-before-break evaluations in operating nuclear power plants.

Analyzable sections run from one terminal end or anchor to another terminal end or anchor. A terminal end is typically a connection to a larger pipe or a component. For the structural analysis, a normally closed valve between pressurized and unpressurized portions of a line is not considered a terminal end. Figure 3.6-3 is a schematic of a portion of a piping system that illustrates the meaning of analyzable segments. In the figure the analyzable portion of the pipe runs from point A to point D.

The leak-before-break evaluation is based on a fracture mechanics stability analysis comparing the selected leakage crack to the critical crack size. The following discussion outlines the analysis method.

The development of leak-before-break bounding analysis curves assume that circumferentially oriented postulated cracks are limiting. Stability is established by analyzing through-wall flaws.

Leakage Flaw

Through-wall flaws in candidate leak-before break piping systems are postulated. [*The size of the postulated flaws are large enough so that the leakage is detectable with adequate margin, using 10 times the minimum installed leak detection capability when the pipes are subjected to normal operational loads combining by algebraic sum method.*]* That is, the size of the leakage flaw postulated would be expected to have a leak rate 10 times the size of the rated leak rate detection capability.

As noted in Subsection 5.2.5, the rated capability of the leak detection systems for the primary coolant inside containment is 0.5 gpm. The methods used to detect leakage are described in Subsection 5.2.5.3. The methods used for primary coolant are the containment sump level, inventory balance, and containment atmosphere radiation. The method used to detect leakage from the main steam line inside containment is the containment sump level. Containment water level sensors provide a diverse backup for main steam line leakage by detecting a 0.5 gpm leak within 17 days. Containment air cooler condensate flow, and containment atmosphere pressure, temperature, and humidity also provide an indication of possible leakage.

Stability and Critical Flaw Sizes

The local and global failure mechanisms are evaluated, as appropriate, to provide margin on flaw size and load. The local mode of failure addresses crack tip behavior: blunting, initiation, extension, and instability. The local failure mechanism is evaluated for ferritic steel piping systems using the J-integral method. The global mode of failure addresses the behavior of the net section: initial yielding, strain hardening, and plastic hinge formation. The global failure mechanism (limit load method) is evaluated for stainless steel piping with no cast material and GTAW welding. From these evaluations a critical crack size is determined. That is, a crack larger than the critical crack size would have unstable growth characteristics.

Acceptance Standards

[*The results of the preceding evaluations are compared to show that the critical flaw size, which is shown to be stable when the maximum loads are combined based on individual absolute values, is at least twice the size (to satisfy margin of 2 on flaw size) of the leakage flaw size. To satisfy a margin*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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on load of 1.0, the maximum loads are combined using absolute summation of individual values.]* The maximum loads are described in Appendix 3B Subsection 3B.3.3.

Bounding Analyses

Evaluations are provided for each different combination of material type, pipe size, pressure, and temperature. These evaluations are used to develop a set of curves of maximum faulted stress versus the corresponding normal stress that satisfy the criteria for leak-before-break. These curves are used in the design of the piping systems and will be used to verify that the as-built piping satisfies the requirements for leak-before-break as discussed in Subsection 3.6.4.2.

3.6.3.4 Documentation of Leak-before-Break Evaluations

The leak-before-break evaluation is used to support the elimination of dynamic effects of pipe breaks from the loading conditions for the piping analysis. An evaluation of leak-before-break using the as-built configuration of the piping system and supports is required as part of the Design Report (also referred to as LBB evaluation report where applicable) of the as-built configuration required to meet ASME Code requirements and LBB criteria. Appendix 3B contains a discussion of the bounding analysis methods for the leak-before-break evaluation.

The analysis methods, criteria, and loads used for evaluation of stress in piping systems are outlined in Subsections 3.7.3 and 3.9.3.

3.6.4 Combined License Information

3.6.4.1 Pipe Break Hazard Analysis

The as-built reconciliation of the pipe break hazards analysis and as-built pipe rupture hazard analysis are addressed in APP-GW-GLR-021 (Reference 14), and the applicable changes are incorporated.

The as-designed pipe rupture hazards evaluation is made available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets protected by associated mitigation features (Reference is Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate.

A pipe rupture hazard analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The report will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design.

The pipe whip restraint and jet shield design includes the properties and characteristics of procured components connected to the piping, components, and walls at identified break and target locations. The design will be completed prior to installation of the piping and connected components.

The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in Subsections 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load (in accordance with DCD Tier 1 Table 3.3-6, item 8).

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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This COL item is also addressed in Subsection 14.3.3.

3.6.4.2 Leak-before-Break Evaluation of As-Designed Piping

The leak-before-break evaluation of the as-designed piping is addressed in APP-GW-GLR-022 (Reference 15), and the applicable changes are incorporated.

3.6.4.3 Leak-before-Break Evaluation of As-Built Piping

Not used.

3.6.4.4 Primary System Inspection Program for Leak-before-Break Piping

Alloy 690 is not used in leak-before-break piping. No additional or augmented inspections are required beyond the inservice inspection program for leak-before-break piping. An as-built verification of the leak-before-break piping is required to verify that no change was introduced that would invalidate the conclusion reached in this subsection.

3.6.5 References

1. NUREG/CR-2913, “Two-Phase Jet Loads,” January 1983.

2. WCAP-8077 (Proprietary) and WCAP-8078 (Nonproprietary), “Ice Condenser Containment Pressure Transient Analysis Methods,” March 1973.

3. ASME B31.1 1989 Edition, “Power Piping,” including 1989 Addendum.

4. ANSI/ANS-58.2-1988, “Design Bases for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture.”

5. Moody, F. J., Fluid Reaction and Impingement Loads, paper presented at the ASCE Specialty Conference, Chicago, December 1973.

6. “MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics,” WCAP-8708 (Proprietary) and WCAP-8709 (Nonproprietary), February 1976.

7. WCAP-8252, “Documentation of Selected Westinghouse Structural Analysis Computer Codes,” Revision 1, May 1977.

8. Not used.

9. Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill Book Company, New York, 1964.

10. EPRI NP-5630, “Guidelines for Piping System Reconciliation” (NCIG-05, Revision 1), May 1988.

11. NUREG-1061, Volume 3, Report of the U. S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks, November 1984.

12. Standard Review Plan 3.6.3, “Leak Before Break Evaluation Procedures,” Federal Register, Volume 52, Number 167, Friday, August 28, 1987; Notice (Public Comment Solicited), pp. 32626-32633.

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13. NUREG/CR-6519, Screening Reactor Steam/Water Systems for Water Hammer, November 1996.

14. APP-GW-GLR-021, “AP1000 As-Built COL Information Items,” Westinghouse Electric Company LLC.

15. APP-GW-GLR-022, “AP1000 Leak-Before-Break Evaluation of As-Designed Piping,” Westinghouse Electric Company LLC.

16. Information Systems Laboratories (2006), “RELAP5/MOD3.3 (Patch03) Code Manual Volume I to VIII, NUREG/CR-5535/Rev P3-Vol I to VIII,” prepared for U.S. Nuclear Regulatory Commission.

17. ASTM A530/A530M - 99, “Standard Specification for General Requirements for Specialized Carbon and Alloy Steel Pipe.”

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Table 3.6-1
High-Energy and Moderate-Energy Fluid Systems
Considered for Protection of Essential Systems[a]

System	High-Energy	Moderate-Energy
Reactor coolant (RCS)	•	
Steam generator (SGS)[b]	•	
Passive core cooling (PXS)	•	
Passive containment cooling (PCS)[c]		•
Main control room habitability (VES)	•	
Chemical and volume control (CVS)	•	
Primary sampling (PSS)	•	
Compressed and instrument air (CAS)	•	
Normal residual heat removal (RNS)[a]		•
Component cooling water (CCS)		•
Spent fuel pool cooling (SFS)		•
Demineralized water (DWS)		•
Liquid radwaste (WLS)		•
Radioactive drain (WRS)		•
Central chilled water (VWS)		•
Fire protection (FPS)		•
Steam generator blowdown (BDS)[d]	•	
Main and startup feedwater (FWS)[d]	•	
Main steam (MSS)[d]	•	
Hot water heating (VYS)		•

Notes:

a. Systems included on this list are high-energy or moderate-energy fluid systems located in the containment or the auxiliary building. Systems that operate at or close to atmospheric pressure such as ventilation and gravity drains are not included. The normal residual heat removal system lines are classified as moderate-energy based on the 1 percent rule. These lines experience high-energy conditions for less than 1 percent of the plant operating time. The portions of the normal residual heat removal system from the connections to the reactor coolant system and passive core cooling system to the first closed valve in each line are high energy. The spent fuel pool cooling system is classified as moderate energy based on the 2 percent rule. These systems experience high-energy conditions for less than 2 percent of the system operating time. See Subsection 3.6.1.1 Item A and Subsection 3.6.1.2 for additional information.

b. Main and startup feedwater, main steam, and steam generator blowdown lines located in the containment and auxiliary building are part of the steam generator system.

c. The essential portion of the system is at atmospheric pressure.

d. The portion of these systems in the turbine building adjacent to the auxiliary building are evaluated for the effect of a circumferential or longitudinal break on the main control room.

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Table 3.6-2 (Sheet 1 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Break Location
Steam Generator Compart-ment 1	11201	22 in. Cold Leg (RCS)	RC Pump Nozzles (2)	4 in. Pressurizer Spray (RCS)	Cold Leg Nozzles (2)
		18 in. Fourth Stage ADS (RCS)	Hot Leg Nozzle	3 in. RC Pump External Piping to/from Motor Bearing HX	RC Pump Cooling Nozzles: A/B (4)
				3 in. RC Pump External Piping to/from Motor Bearing HX	Motor Bearing Water Cooling HX Nozzles: A/B (4)
				3 in. RC Pump External Piping to/ from Motor Bearing HX	Intermediate breaks along elbow fittings: pumps A/B (8)
	11301	31 in. Hot Leg (RCS)	SG Nozzle	3 in. Purification (CVS)	3 in. SG Channel Head Nozzle
		18 in. Surge Line (RCS)	Hot Leg Nozzle		
		18 in. & 14 in. Fourth Stage ADS (RCS)	Valves: V004A/C		
		14 in. PRHR Return (RCS)	SG Channel Head Nozzle		
	11401	None		4 in. SG Blowdown (SGS)	4 in. SG Nozzle
	11501	None		None	
	11601			16 in and 20 in. Feedwater (SGS) 6 in. Startup Feedwater (SGS)	SG Nozzle SG Nozzle
	11701	38 in. Main Steam (SGS)	SG Nozzle	None	

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Table 3.6-2 (Sheet 2 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Break Location
Steam Generator Compart-ment 2	11202	22 in. Cold Leg (RCS)	RC Pump Nozzles (2)	3 in. RC Pump External Piping to/from Motor Bearing HX	RC Pump Cooling Nozzles: A/B (4)
		18 in. Fourth Stage ADS (RCS)	Hot Leg Nozzle	3 in. RC Pump External Piping to/from Motor Bearing HX	Motor Bearing Water Cooling HX Nozzles: A/B (4)
		20 in. Normal RHR (RCS)	Hot Leg Nozzle	3 in. RC Pump External Piping to/from Motor Bearing HX	Intermediate breaks along elbow fittings: pumps A/B (8)
		12 in. Normal RHR (RCS)	20 in. x 12 in. Reducer (This is not a terminal end)		
	11302	31 in. Hot Leg (RCS)	SG Nozzle	None	
		18 in. & 14 in. Fourth Stage ADS (RCS)	Valves: V004B/D		
		8 in. Cold Leg to CMT (RCS)	Cold Leg Nozzles (2)		
	11402	None		4 in. SG Blowdown (SGS)	4 in. SG Nozzle
	11502	None		None	
	11602			16 in. and 20 in. Feedwater (SGS) 6 in. Startup Feedwater (SGS)	SG Nozzle SG Nozzle

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Table 3.6-2 (Sheet 3 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Break Location
	11702	38 in. Main Steam (SGS)	SG Nozzle	None	
Reactor Vessel Nozzle Area	11205	31 in. Hot Leg (RCS)	Reactor Vessel Nozzles (2)	None	
		22 in. Cold Leg (RCS)	Reactor Vessel Nozzles (4)		
		8 in. Direct Vessel Injection (RCS)	Reactor Vessel Nozzles (2)		
PXS Valve and Accumulator Room A	11206	8 in. Accumulator Injection (PXS)	Accumulator Nozzle	None	
		8 in. CMT Injection (PXS)	CMT Nozzle		
		6 in. Line from Normal RHR (RNS)	Valve: V017A		
		8 in. Line from IRWST (PXS)	Valves: V125A & V123A		
PXS Valve Room B	11207 PXS	6 in. Line from Normal RHR (RNS)	Valve: V017B	None	
		8 in. Line from IRWST (PXS)	Valves: V125B & V123B		
Accumulator Room B	11207 ACCUM	8 in. Accumulator Injection (PXS)	Accumulator Nozzle	None	
		8 in. CMT Injection (PXS)	CMT Nozzle		
RNS Valve Room	11208	10 in. Normal RHR (RNS)	Valves: V001A/B	None	

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Table 3.6-2 (Sheet 4 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	**Room Number**	**Description**	**Terminal End Break Location Excluded by LBB**	**Description**	**Break Location**
Vertical Access	11204	None		3 in. Line from Regen HX to SG 01 (CVS)	Anchor to Wall
				3 in. Purification from Cold Leg to Regen HX (CVS)	Anchor to Wall
Lower Pressurizer Compartment	11303	18 in. Surge Line (RCS)	Pressurizer Nozzle	None	
Upper Pressurizer Compartment	11503	14 in. ADS (RCS)	Pressurizer Nozzle (2)	4 in. Pressurizer Spray (RCS)	Pressurizer Nozzle
Lower ADS Valve Area	11603	14 in. & 8 in. ADS (RCS) 6 in. Pressurizer Safety (RCS)	Valves: V012B & V013B Valves: V005A and V005B 14 in. x 6 in. Tees (2)	4 in. ADS (RCS)	Valve V0011B & 14 in. x 4 in. Branch
Upper ADS Valve Area	11703	14 in. & 8 in., ADS (RCS)	Valves: V012A & V013A	4 in. ADS (RCS)	Valve V0011A & 14 in. x 4 in. Branch
Maintenance Floor/ Mezzanine	11400	38 in. Main Steam (SGS)	Non-terminal End Location (2) at Boundary of Break Exclusion Zone	6 in. Startup Feedwater (SGS)	Anchors (2) at Containment Penetration
		14 in. Passive RHR (PXS)	PRHR HX Inlet Nozzle		
		8 in. CMT Balance Line Piping	CMT Nozzles (2)		

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Table 3.6-2 (Sheet 5 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Break Location
SG01 Access Room	11304	None		3 in. CVS Return from Regen HX	Intermediate breaks along pipe fittings: valve transition (3), weldolet (1)
Pressurizer Spray Valve Room	11403	None		4 in. Pressurizer Spray to 2 in. Auxiliary Spray	Intermediate breaks along pipe fitting: 4"x2" reducing tee (9)
Maintenance Floor	11300	14 in. Passive RHR (PXS)	PRHR HX Outlet Nozzle	1.5 in. PRHR Return flow instrument branch line (PXS)	Sockolet for flow element
Operating Deck	11500	None		None	
CVS Room	11209	None		3 in. Purification from Pressurizer Spray to Regen HX (CVS)	Regen HX Nozzle
				3 in. Return, Auxiliary Spray (CVS)	Regen HX Nozzle
				3 in. Return to RNS from Regen HX (CVS)	Valve: V079
				3 in. Supply from RNS to Letdown HX (CVS)	Valve: V072
				3 in. Supply from Regen HX to Letdown HX (CVS)	Nozzles: Regen HX, Letdown HX
CVS Room	11209 Pipe Chase	None		3 in. Purification from Anchor to Regen HX	Anchor
				3 in. Return from Regen HX to Anchor (CVS)	Anchor
				4 in. SG Blowdown (SGS)	Anchors (2) at Containment Penetration

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Table 3.6-2 (Sheet 6 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Break Location
Reactor Coolant Drain Tank Room	11104	None		None	
Reactor Vessel Cavity	11105	None		None	
MSIV Compartment B (Upper/Lower)	12504/ 12404	None		Main Steam Main Feedwater Startup Feedwater Lines[a]	Longitudinal Cracks with Crack Flow Areas of 1 Square Foot are Postulated
MSIV Compartment A (Upper/Lower)	12506/ 12406	None		Main Steam Main Feedwater Startup Feedwater Lines[a]	Longitudinal Cracks with Crack Flow Areas of 1 Square Foot are Postulated
Valve/Piping Penetration Room	12306	None		4 in. Steam Generator Blowdown[a]	Anchors (2) at Containment Penetrations Anchors (2) at Wall to Turbine Building

Note:

a. The piping in these areas is included in break exclusion zones. For additional information on the evaluation of these lines, see Subsection 3.6.1.2.1 for the steam generator blowdown line; Subsection 3.6.1.2.2 for information on the evaluation of lines in MSIV compartment B because of the proximity to the main control room; and Subsection 3.6.2.1.1.4 for general break exclusion zone requirements.

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Table 3.6-2 (Sheet 7 of 7)
Subcompartments and Postulated Pipe Ruptures

Room #	Description	Bottom Elevation	Top Elevation
11104	RCDT Room	66′-6″	81′-0″
11105	Reactor Vessel Cavity	66′-6″	98′
11205	Reactor Vessel Nozzle Area	98′	107′-2″
11201	SG Compartment 1	83′	104′-7″
11202	SG Compartment 2	83′	104′-7″
11204	Vertical Access	83′	107′-2″
11206	PXS Valve and Accumulator Room A	87′-6″	105′-2″
11300	Maintenance Floor	107′-2″	118′-6″
11301	SG Compartment 1	104′-7″	116′-6″
11302	SG Compartment 2	104′-7″	116′-6″
11400	Maintenance Floor/Mezzanine	118′-6″	133′-3″
11401	SG Compartment 1	116′-6″	135′-3″
11402	SG Compartment 2	116′-6″	135′-3″
11501	SG Compartment 1	135′-3″	153′-0″
11502	SG Compartment 2	135′-3″	153′-0″
11601	SG Compartment 1	153′-0″	166′-4″
11602	SG Compartment 2	153′-0″	166′-6″
11701	SG Compartment 1	166′-4″	----
11702	SG Compartment 2	166′-4″	----
11500	Operating Deck	135′-3″	281′-8 3/8″
11303	Lower Pressurizer Compartment	107′-2″	135′-3″
11304	SG01 Access Room	107′-2″	118′-6″
11403	Pressurizer Spray Valve Room	118′-6″	133′-3″
11503	Upper Pressurizer Compartment	135′-3″	166′-1.5″
11603	Lower ADS Valve Area	166′-1.5″	176′-10.5″
11703	Upper ADS Valve Area	176′-10.5″	----
11207 ACCUM	Accumulator Room B	87′-6″	105′-2″
11207 PXS	PXS Valve Room B	87′-6″	105′-2″
11208	RNS Valve Room	94′	105′-2″
11209	CVS Room	80′-6″	105′-2″
11209 PIPE CHASE	CVS Room Pipe Chase	100′-0″	105′-2″
12306	Valve/Piping Penetration Room	100′-0″	117′-6″
12504/12404	MSIV Compartment B (Upper/Lower)	117′-6″	153′-0″
12506/12406	MSIV Compartment A (Upper/Lower)	117′-6″	153′-0″

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Table 3.6-3 (Sheet 1 of 10)
NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11201	Steam Generator Compartment-01	Pipe Whip Restraint RCS-SH-W002 (RCS-002)	Reactor Coolant System (RCS) Pressurizer Spray Line, 4" L110A: Terminal End Break at RCS Cold Leg RCS-PL-L002A/N11A	RCS spray control valve (RCS-V110A) (Room 11403). RCS pressurizer spray bypass valve (RCS-V210A) (Room 11403). RCS spray control valve (RCS-V110B) (Room 11403). RCS pressurizer spray bypass valve (RCS-V210B) (Room 11403).
		Pipe Whip Restraint RCS-SH-W003 (RCS-003)	RCS Pressurizer Spray Line, 4" L106: Terminal End Break at RCS Cold Leg RCS-PL-L002B/N11B	RCS spray control valve (RCS-V110A) (Room 11403). RCS pressurizer spray bypass valve (RCS-V210A) (Room 11403). RCS spray control valve (RCS-V110B) (Room 11403). RCS pressurizer spray bypass valve (RCS-V210B) (Room 11403).
11204	Vertical Access Room	Dual Acting Pipe Support CVS-PH-11R2270 (CVS-004U)	Chemical and Volume Control System (CVS) Letdown Line, 3" L002: Terminal End Break at In-Line Anchor	SGS blowdown piping (L009B). CCS isolation valve (CCS-V201) (Room 11400). CCS isolation valve (CCS-V207) (Room 11400). CCS isolation valve (CCS-V220) (Room 11400). CVS purification stop valve (CVS-V001) (Room 11403). CVS purification stop valve (CVS-V002) (Room 11403). CVS purification stop valve (CVS-V003) (Room 11403).

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Table 3.6-3 (Sheet 2 of 10)
NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11204	Vertical Access Room	Dual Acting Pipe Support CVS-PH-11R2271 (CVS-007D)	CVS Purification Return Line, 3" L018: Terminal End Break at In-Line Anchor	CCS isolation valve (CCS-V201) (Room 11400). CCS isolation valve (CCS-V207) (Room 11400). CCS isolation valve (CCS-V220) (Room 11400). CVS regenerative HX shell side outlet check valve (CVS-V080) (Room 11303). CVS purification return line stop check valve (CVS-V081) (Room 11304). CVS purification return line check valve (CVS-V082) (Room 11303). CVS auxiliary pressurizer spray line isolation valve (CVS-V084) (Room 11304). CVS auxiliary pressurizer spray line isolation valve (CVS-V085) (Room 11304).
11209	CVS Room (Pipe Chase)	Pipe Whip Restraint SGS-SH-W004 (SGS-004)	Steam Generator System (SGS) Blowdown Line, 4" L009A: Terminal End Break at Containment Penetration P27	SGS blowdown piping (L009B). CVS isolation valve (CVS-V042) (Room 11209). CVS isolation valve (CVS-V045) (Room 11300). CVS isolation valve (CVS-V058) (Room 11300). CVS isolation valve (CVS-V091) (Room 11300). CVS isolation valve (CVS-V100) (Room 11300). CVS isolation valve (CVS-V217) (Room 11300). WLS isolation valve (WLS-V055) (Room 11300). WLS isolation valve (WLS-V058) (Room 11300). WLS isolation valve (WLS-V067) (Room 11300).

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NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11209 (cont.)	CVS Room (Pipe Chase)	Pipe Whip Restraint SGS-SH-W008 (SGS-008)	SGS Blowdown Line, 4" L009B: Terminal End Break at Containment Penetration P28	CVS isolation valve (CVS-V042) (Room 11209). CVS isolation valve (CVS-V045) (Room 11300). CVS isolation valve (CVS-V058) (Room 11300). CVS isolation valve (CVS-V091) (Room 11300). CVS isolation valve (CVS-V100) (Room 11300). CVS isolation valve (CVS-V217) (Room 11300). WLS isolation valve (WLS-V067) (Room 11300).
		Dual Acting Pipe Supports CVS-PH-11Y2265 CVS-PH-11Y2266 (CVS-056U)	CVS Makeup Line, 3" L056: Terminal End Break at In-Line Anchor	SGS blowdown piping (L009A). SGS blowdown piping (L009B). CVS isolation valve (CVS-V045) (Room 11300). CVS isolation valve (CVS-V058) (Room 11300). CVS isolation valve (CVS-V091) (Room 11300). CVS isolation valve (CVS-V100) (Room 11300). WLS isolation valve (WLS-V055) (Room 11300). WLS isolation valve (WLS-V058) (Room 11300). WLS isolation valve (WLS-V067) (Room 11300).

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Table 3.6-3 (Sheet 4 of 10)
NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11209 (cont.)	CVS Room (Pipe Chase)	Pipe Whip Restraint CVS-SH-W004D (CVS-004D)	CVS Letdown Line, 3" L002: Terminal End Break at In-Line Anchor	SGS blowdown piping (L009A). SGS blowdown piping (L009B). CVS isolation valve (CVS-V042) (Room 11209). CVS isolation valve (CVS-V045) (Room 11300). CVS isolation valve (CVS-V058) (Room 11300). CVS isolation valve (CVS-V091) (Room 11300). CVS isolation valve (CVS-V100) (Room 11300). CVS isolation valve (CVS-V217) (Room 11300). WLS isolation valve (WLS-V055) (Room 11300). WLS isolation valve (WLS-V058) (Room 11300). WLS isolation valve (WLS-V067) (Room 11300).
		Pipe Whip Restraint CVS-SH-W007U (CVS-007U)	CVS Purification Return Line, 3" L018: Terminal End Break at In-Line Anchor	SGS blowdown piping (L009A). SGS blowdown piping (L009B). CVS isolation valve (CVS-V042) (Room 11209). CVS isolation valve (CVS-V045) (Room 11300). CVS isolation valve (CVS-V058) (Room 11300). CVS isolation valve (CVS-V091) (Room 11300). CVS isolation valve (CVS-V100) (Room 11300). CVS isolation valve (CVS-V217) (Room 11300). WLS isolation valve (WLS-V055) (Room 11300). WLS isolation valve (WLS-V058) (Room 11300). WLS isolation valve (WLS-V067) (Room 11300).

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NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11301	Steam Generator 1 Lower Manway Area	Pipe Whip Restraint RCS-SH-W001 A/B (RCS-001)	CVS Purification Return Nozzle, 3" RCS L112: Terminal End Break at RCS-MB-01/N05	CVS isolation valve (CVS-V084) (Room 11304). SGS blowdown piping (L009A) (Room 11301). RCS Passive Residual Heat Removal (PRHR) Heat Exchanger (HX) return piping (L113) (Room 11301).
11303	Lower Pressurizer Compartment	Pipe Whip Restraint CVS-SH-W210 (CVS-210-C)	CVS Purification Return Line 3" RCS L112: Intermediate Break at Outlet CVS-V082	Column support for pressurizer (RCS-MV-02) (Room 11303). SG nozzle RCS-MB-01/nozzle N05 (Room 11301). RCS surge line – line RCS-L003 (Room 11303).
		Dual Acting Pipe Support RCS-PH-11Y1132 (RCS-201-C)	RCS Pressurizer Spray Line, 4" L215: Intermediate Break at 4" x 2" Reducing Tee	Lateral support struts for pressurizer (RCS-MV-02) (Room 11303). RCS instrument piping (L225A) (Room 11303). RCS level instrument (LT195A) (Room 11300). RCS pressure transmitter (PT191A) (Room 11300). Pressurizer nozzle – RCS-MV-02/nozzle N02 (Room 11503).
11400	Maintenance Floor Mezzanine	Pipe Whip Restraint SGS-SH-W006 (SGS-006)	SGS Startup Feedwater Line, 6" L005B: Terminal End Break at SGS-PY-C05B, Containment Penetration P45	Steam generator 2 (RCS-MB-02) (Room 11602).
		Pipe Whip Restraint SGS-SH-W002 (SGS-002)	SGS Startup Feedwater Line, 6" L005A: Terminal End Break at SGS-PY-C05A, Containment Penetration P44	Steam generator 1 (RCS-MB-01) (Room 11601). Main steam 1 pressure transmitters – PT030/PT032 (Room 11400). Steam generator 1 (RCS-MB-01) level instrumentation piping – SGS-PL-L039A/SGS-PL-L041A, (Room 11400).

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NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11401	Steam Generator 1 Tube Sheet Area	Pipe Whip Restraint SGS-SH-W003 (SGS-003)	SGS Blowdown Line, 4" L009A: Terminal End Break at RCS-MB-01/N06A	SGS blowdown penetration (SGS-PY-C03A).
11402	Steam Generator 2 Tube Sheet Area	Pipe Whip Restraint SGS-SH-W007 (SGS-007)	SGS Blowdown Line, 4" L009B: Terminal End Break at RCS-MB-02/N06A	SGS blowdown penetration (SGS-PY-C03B). Main steam 2 pressure transmitters – PT034/PT036 (Room 11402). Steam generator 2 (RCS-MB-02) level instrumentation piping – SGS-PL-L039B/SGS-PL-L041B (Room 11402).
11403	Pressurizer Spray Valve Room	Dual Acting Pipe Support RCS-PH-11R0084 (RCS-203-C)	RCS Pressurizer Spray Valve Line, 4" L213: Intermediate Break at 4" x 2" Reducing Tee	RCS spray control valve (RCS-V110A) (Room 11403). RCS pressurizer spray bypass valve (RCS-V210A) (Room 11403). RCS spray control valve (RCS-V110B) (Room 11403). RCS pressurizer spray bypass valve (RCS-V210B) (Room 11403).

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NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11503	Upper Pressurizer Compartment	Pipe Whip Restraint RCS-SH-W006 (RCS-006)	RCS Pressurizer Spray Line, 4" L215: Terminal End Break at RCS-MV-02/N02	ADS Stage 1, 2, and 3 valves (RCS-V001B, RCS-V002B, RCS-V003B, RCS-V011B, RCS-V012B, and RCS-V013B) (Room 11603). ADS Stage 1, 2, and 3 valves (RCS-V001A, RCS-V002A, RCS-V003A, RCS-V011A, RCS-V012A, and RCS-V013A) (Room 11703). Pressurizer safety valves (RCS-V005A, RCS-V005B) (Room 11603). ADS discharge header B vacuum relief valve (RCS-V010B) (Room 11603). ADS discharge header A vacuum relief valve (RCS-V010A) (Room 11703). ADS Stage 1, 2, and 3 support steel (Rooms 11503, 11603, and 11703). RCS spray control valve (RCS-V110A) (Room 11403). RCS pressurizer spray bypass valve (RCS-V210A) (Room 11403). RCS spray control valve (RCS-V110B) (Room 11403). RCS pressurizer spray bypass valve (RCS-V210B) (Room 11403).

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Table 3.6-3 (Sheet 8 of 10)
NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11601	Steam Generator 1 Feedwater Nozzle Area (153'-0")	Pipe Whip Restraint SGS-SH-W001 (SGS-001)	SGS Startup Feedwater Line, 6" L005A: Terminal End Break at RCS-MB-01/N20	RCS head vent valve (RCS-V150A) (Room 11601). RCS head vent valve (RCS-V150B) (Room 11601). RCS head vent valve (RCS-V150C) (Room 11601). RCS head vent valve (RCS-V150D) (Room 11601). SGS startup feedwater penetration (SGS-PY-C05A).
		Pipe Whip Restraint SGS-SH-W021 (SGS-021)	SGS Main Feedwater Line, 16" L008A/L003A: Terminal End Break at RCS-MB-01/N04.	SGS main feedwater penetration (SGS-PY-C02A). Break exclusion zone: SGS main feedwater piping (L002A).
11602	Steam Generator 2 Feedwater Nozzle Area (153'-0")	Pipe Whip Restraint SGS-SH-W005 (SGS-005)	SGS Startup Feedwater Line, 6" L005B: Terminal End Break at RCS-MB-02/N20	SGS startup feedwater penetration (SGS-PY-C05B).
		Pipe Whip Restraint SGS-SH-W022 (SGS-022)	SGS Main Feedwater Line, 16" L008B/L003B: Terminal End Break at RCS-MB-02/N04	SGS main feedwater penetration (SGS-PY-C02B). Break exclusion zone: SGS main feedwater piping (L002B).
		Jet Shield SGS-SH-P022 (SGS-022)	SGS Main Feedwater Line, 16" L008B/L003B: Terminal End Break at RCS-MB-02/N04	Main steam line snubber for support APP-SGS-PH-11Y7057 (Room 11602).

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Table 3.6-3 (Sheet 9 of 10)
NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
11603	ADS Valve Area – Lower Tier	Pipe Whip Restraint RCS-SH-W108 (RCS-108)	RCS ADS Stage 1 Train B Inlet Line, 4" L010B: Terminal End Break at First Stage ADS Isolation Valve RCS-PL-V011B	ADS stage 1, 2, and 3 valves (RCS-V002B, RCS-V003B, RCS-V012B, and RCS-V013B) (Room 11603). ADS stage 1, 2, and 3 valves (RCS-V001A, RCS-V002A, RCS-V003A, RCS-V011A, RCS-V012A, and RCS-V013A) (Room 11703). ADS discharge header B vacuum relief valve (RCS-V010B) (Room 11603). ADS discharge header A vacuum relief valve (RCS-V010A) (Room 11703). ADS stage 1, 2, and 3 support steel 1102-Q6-01 (Rooms 11503, 11603, and 11703).
11703	Upper ADS Valve Area – 176'-10 1/2"	Pipe Whip Restraint RCS-SH-W106 (RCS-106)	RCS ADS Stage 1 Train A Inlet Line, 4" L010A: Terminal End Break at First Stage ADS Isolation Valve RCS-PL-V011A	ADS stage 1, 2, and 3 valves (RCS-V001B, RCS-V002B, RCS-V003B, RCS-V011B, RCS-V012B, and RCS-V013B) (Room 11603). ADS stage 1, 2, and 3 valves (RCS-V002A, RCS-V003A, RCS-V012A, and RCS-V013A) (Room 11703). ADS discharge header B vacuum relief valve (RCS-V010B) (Room 11603). ADS discharge header A vacuum relief valve (RCS-V010A) (Room 11703). ADS stage 1, 2, and 3 support steel 1102-Q6-01 (Rooms 11503, 11603, and 11703).

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Table 3.6-3 (Sheet 10 of 10)
NI Rooms With Pipe Whip Restraints, Dual Acting Pipe Supports, Jet Shields and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Protection Mechanism (Break ID)	Hazard Source	Essential Target Description/Room
12258	Degasifier Column Room	Dual Acting Pipe Supports CVS-PH-12R2272 CVS-PH-12R2273 (CVS-061D)	CVS Makeup Pump Discharge Header, 3" L131: Terminal End Break at Anchor CVS-PH-12A0122	Break exclusion zone: CVS makeup piping (L053).
12404	East MSIV Compartment	Pipe Whip Restraint SGS-SH-W033 SGS-SH-W034 SGS-SH-W035	Main Feedwater Line, 20" L002B and L003B: Longitudinal Splits	Floor of Room 12404 (protection of electrical/I&C components in Room 12304 – room below east MSIV compartment).
		Pipe Whip Restraint SGS-SH-W031 SGS-SH-W032	Main Feedwater Line, 20" L002B and L003B: Longitudinal Splits and Guillotine Breaks	Wall 11 (Room 12404) and Wall L (Room 12404 – protection of main control room).

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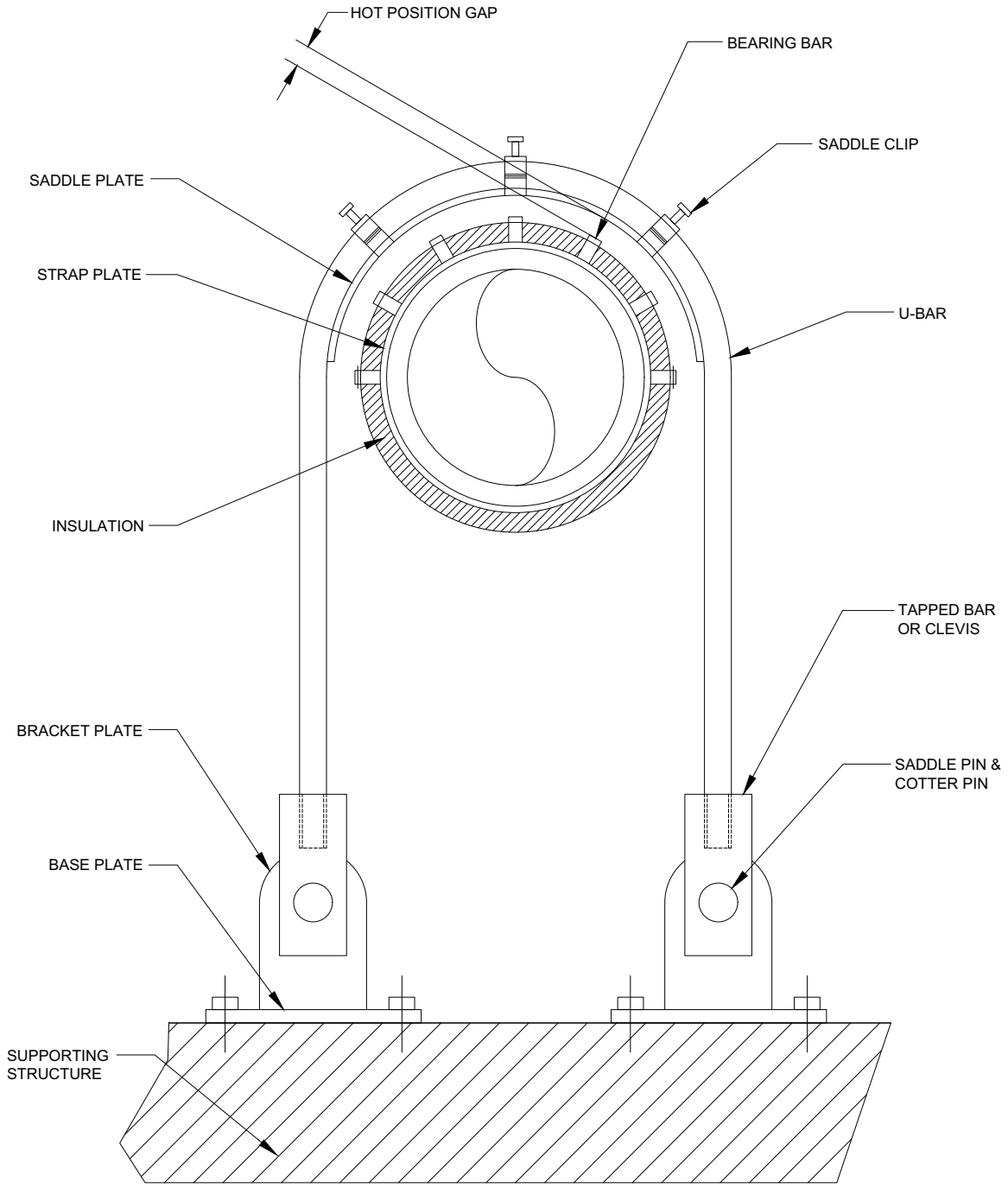

Figure 3.6-1
Typical U-Bar Restraint

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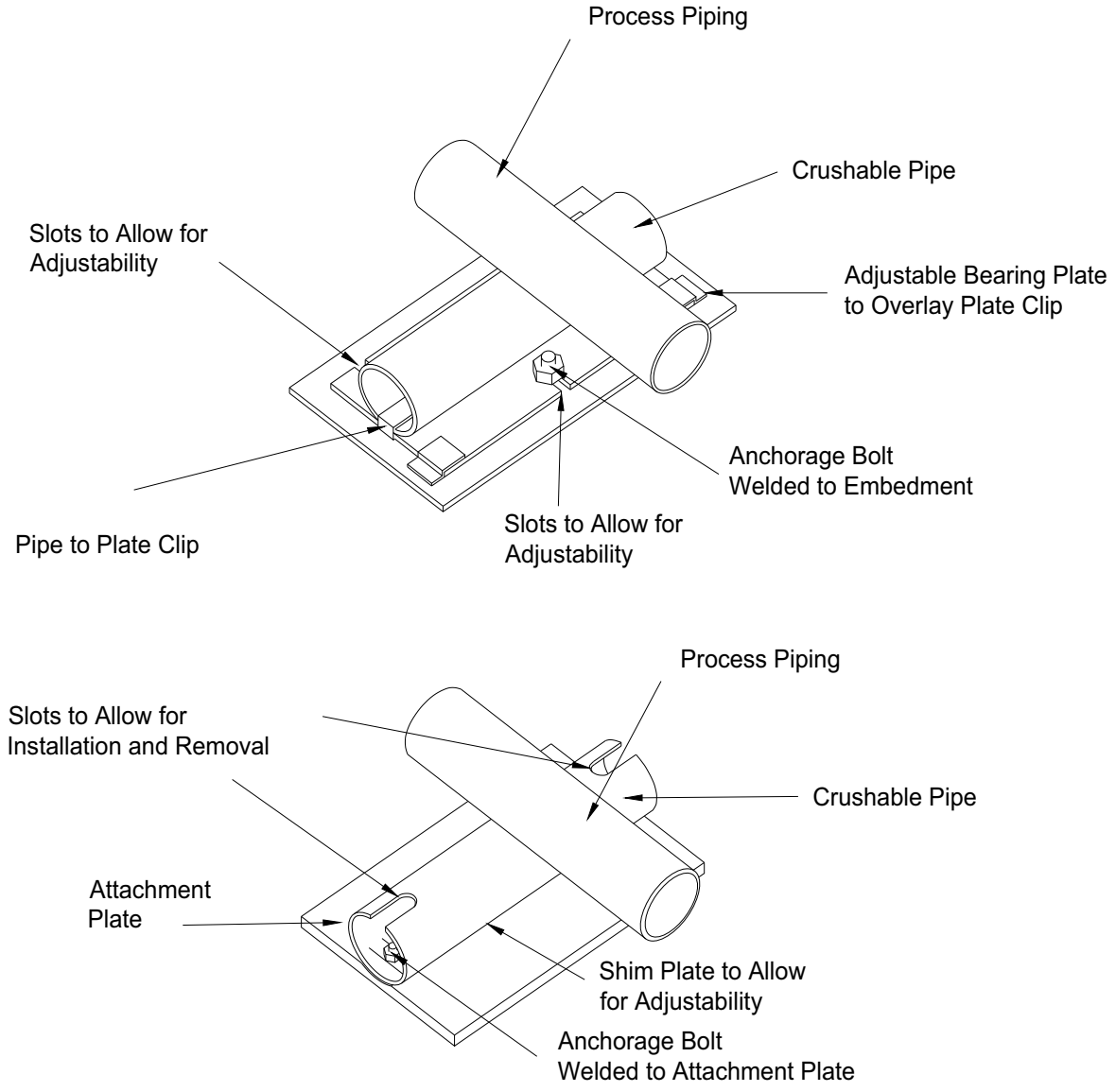

Figure 3.6-2
Typical Structural Components of Crushable Pipe Type Whip Restraints

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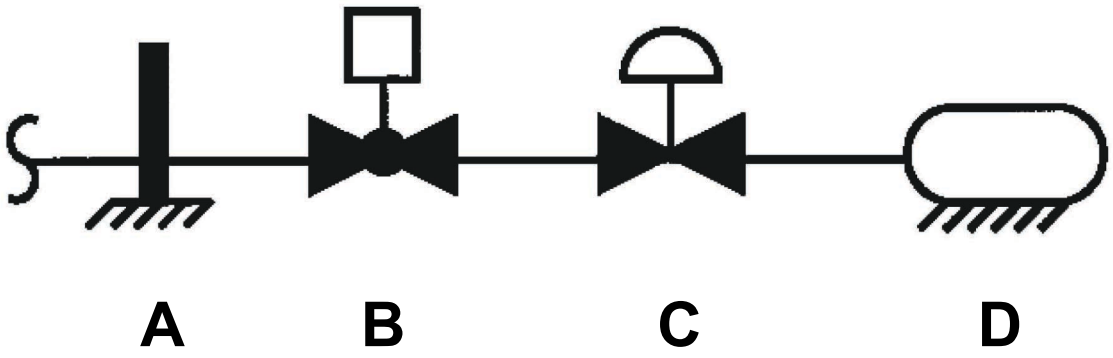

A – Anchor
B – Closed Valve
C – Closed Valve
D – Terminal End

A to B – High Energy
B to D – Moderate Energy

Figure 3.6-3
Terminal Ends Definitions

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3.7 Seismic Design

Plant structures, systems, and components important to safety are required by General Design Criterion (GDC) 2 of Appendix A of 10 CFR 50 to be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

Each plant structure, system, equipment, and component is classified in an applicable seismic category depending on its function. A three-level seismic classification system is used for the AP1000: seismic Category I, seismic Category II, and nonseismic. The definitions of the seismic classifications and a seismic classifications listing of structures, systems, equipment, and components are presented in Section 3.2.

Seismic design of the AP1000 seismic Categories I and II structures, systems, equipment, and components is based on the safe shutdown earthquake (SSE). The safe shutdown earthquake is defined as the maximum potential vibratory ground motion at the generic plant site as identified in Section 2.5.

The operating basis earthquake (OBE) has been eliminated as a design requirement for the AP1000. Low-level seismic effects are included in the design of certain equipment potentially sensitive to a number of such events based on a percentage of the responses calculated for the safe shutdown earthquake. Criteria for evaluating the need to shut down the plant following an earthquake are established using the cumulative absolute velocity approach according to EPRI Report NP-5930 (Reference 1) and EPRI Report TR-100082 (Reference 17). For the purposes of the shutdown criteria in Reference 1 the operating basis earthquake for shutdown is considered to be one-third of the safe shutdown earthquake.

Seismic Category I structures, systems, and components are designed to withstand the effects of the safe shutdown earthquake event and to maintain the specified design functions. Seismic Category II and nonseismic structures are designed or physically arranged (or both) so that the safe shutdown earthquake could not cause unacceptable structural interaction with or failure of seismic Category I structures, systems, and components.

3.7.1 Seismic Input

The geologic and seismologic considerations of the plant site are discussed in Section 2.5.

The peak ground acceleration of the safe shutdown earthquake, now referred to as the Certified Seismic Design Response Spectra (CSDRS), has been established as 0.30g for the AP1000 design. The vertical peak ground acceleration is conservatively assumed to equal the horizontal value of 0.30g as discussed in Section 2.5.

3.7.1.1 Design Response Spectra

The AP1000 design response spectra of the safe shutdown earthquake, now referred to as the Certified Seismic Design Response Spectra (CSDRS), are provided in Figures 3.7.1-1 and 3.7.1-2 for the horizontal and the vertical components, respectively.

The horizontal design response spectra for the AP1000 plant are developed, using the Regulatory Guide 1.60 spectra as the base and several evaluations to investigate the high frequency amplification effects. These evaluations included:

- Comparison of Regulatory Guide 1.60 spectra with the spectra predicted by recent eastern U.S. spectral velocity attenuation relations (References 23, 24, 25, and 26) using a suite of magnitudes and distances giving a 0.3 g peak acceleration

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- Comparison of Regulatory Guide 1.60 spectra with the 10^{-4} annual probability uniform hazard spectra developed for eastern U.S. nuclear power plants by both Lawrence Livermore National Laboratory (Reference 27) and Electric Power Research Institute (Reference 28)

- Comparison of Regulatory Guide 1.60 spectra with the spectra of 79 additional old and newer components of strong earthquake time histories not considered in the original derivation of Regulatory Guide 1.60

Based on the above described evaluations, it is concluded that the eastern U.S. seismic data exceed Regulatory Guide 1.60 spectra by a modest amount in the 15 to 33 hertz frequency range when derived either from published attenuation relations or from the 10^{-4} annual probability of exceedance uniform hazard spectra at eastern U.S. sites. This conclusion is consistent with findings of other investigators that eastern North American earthquakes have more energy at high frequencies than western earthquakes. Exceedance of Regulatory Guide 1.60 spectra at the high frequency range, therefore, would be expected since Regulatory Guide 1.60 spectra are based primarily on western U.S. earthquakes. The evaluation shows that, at 25 hertz (approximately in the middle of the range of high frequencies being considered, and a frequency for which spectral amplitudes are explicitly evaluated) the mean-plus-one-standard-deviation spectral amplitudes for 5 percent damping range from about 2.1 to 4 cm/sec and average 2.7 cm/sec. Whereas, the Regulatory Guide 1.60 spectral amplitude at the same frequency and damping value equal just over 2 cm/sec.

It is concluded, therefore, that an appropriate augmented 5 percent damping horizontal design velocity response spectrum for the AP1000 project is one with spectral amplitudes equal to the Regulatory Guide 1.60 spectrum at control frequencies 0.25, 2.5, 9 and 33 hertz augmented by an additional control frequency at 25 hertz with an amplitude equal to 3 cm/sec. This spectral amplitude equals 1.3 times the Regulatory Guide 1.60 amplitude at the same frequency. The additional control point's spectral amplitude of other damping values were determined by increasing the Regulatory Guide 1.60 spectral amplitude by 30 percent.

The AP1000 design vertical response spectrum is, similarly, based on the Regulatory Guide 1.60 vertical spectra at lower frequencies but is augmented at the higher frequencies equal to the horizontal response spectrum.

The AP1000 design response spectra's relative values of spectrum amplification factors for control points are presented in Table 3.7.1-3.

The design response spectra are applied at the foundation level in the free field at hard rock sites and at the finished grade in the free field at firm rock and soil sites. The resulting peak horizontal ground acceleration values are above 0.1g. This satisfies 10 CFR Part 50, Appendix S, which requires that the horizontal component of the SSE ground motion in the free-field at the foundation elevation (that is, bottom of foundation) has a peak ground acceleration of at least 0.1g together with an appropriate response spectrum. The definitions (characteristics) of hard rock, firm rock, and soil sites are provided in Subsection 3.7.1.4.

3.7.1.2 Design Time History

A "single" set of three mutually orthogonal, statistically independent, synthetic acceleration time histories is used as the input in the dynamic analysis of seismic Category I structures. The synthetic time histories were generated by modifying a set of actual recorded "TAFT" earthquake time histories. The design time histories include a total time duration equal to 20 seconds and a corresponding stationary phase, strong motion duration greater than 6 seconds. The acceleration, velocity, and displacement time-history plots for the three orthogonal earthquake components, "H1," "H2," and "V," are presented in Figures 3.7.1-3, 3.7.1-4, and 3.7.1-5. Design horizontal time history, H1, is applied in the north-south (Global X or 1) direction; design horizontal time history, H2, is

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applied in the east-west (global Y or 2) direction; and design vertical time history is applied in the vertical (global Z or 3) direction. The cross-correlation coefficients between the three components of the design time histories are as follows:

$$\rho_{12} = 0.05, \rho_{23} = 0.043, \text{ and } \rho_{31} = 0.140$$

where 1, 2, 3 are the three global directions.

Since the three coefficients are less than 0.16 as recommended in Reference 30, which was referenced by NRC Regulatory Guide 1.92, Revision 1, it is concluded that these three components are statistically independent. The design time histories are applied at the foundation level in the free field.

The ground motion time histories (H1, H2, and V) are generated with time step size of 0.010 second for applications in soil structure interaction analyses. For applications in the fixed-base mode superposition time-history analyses, the time step size is reduced to 0.005 second by linear interpolation. The maximum frequency of interest in the horizontal and vertical seismic analysis of the nuclear island is 33 hertz. Modes with higher frequencies are included in the analysis so that the mass in these higher modes is included in the member forces. The cutoff frequencies used in the soil structure interaction analyses are 33 hertz. The maximum "cut-off" frequency for the soil structure interaction analyses and the fixed-base analyses is well within the Nyquist frequency limit.

The comparison plots of the acceleration response spectra of the time histories versus the design response spectra for 2, 3, 4, 5, and 7 percent critical damping are shown in Figures 3.7.1-6, 3.7.1-7, and 3.7.1-8. The SRP 3.7.1, Table 3.7.1-1, provision of frequency intervals is used in the computation of these response spectra.

In SRP 3.7.1 the NRC introduced the requirement of minimum power spectral density to prevent the design ground acceleration time histories from having a deficiency of power over any frequency range. SRP 3.7.1, Revision 2, specifies that the use of a single time history is justified by satisfying a target power spectral density (PSD) requirement in addition to the design response spectra enveloping requirements. Furthermore, it specifies that when spectra other than Regulatory Guide 1.60 spectra are used, a compatible power spectral density shall be developed using procedures outlined in NUREG/CR-5347 (Reference 29).

The NUREG/CR-5347 procedures involve ad hoc hybridization of two earlier power spectral density envelopes. Since the modification to the RG 1.60 design spectra adopted for AP1000 (see Subsection 3.7.1.1) is relatively small (compared to the uncertainty in the fit to RG 1.60 of power spectral density-compatible time histories referenced in NUREG/CR-5347) and occurs only in the frequency range between 9 to 33 hertz, a project-specific power spectral density is developed using a slightly different hybridization for the higher frequencies.

Since the original RG 1.60 spectrum and the project-specific modified RG 1.60 spectrum are identical for frequencies less than 9 hertz, no modification to the power spectral density is done in this frequency range. At frequencies above 9 hertz, the third and the fourth legs of the power spectral density are slightly modified as follows:

- The frequency at which the design response spectrum inflected towards a 1.0 amplification factor at 33 hertz takes place at 25 hertz in the AP1000 spectrum rather than at 9 hertz as in the RG 1.60 spectrum. The third leg of the power spectral density, therefore, is extended to about 25 hertz rather than 16 hertz.

- The lead coefficient to the fourth leg of the power spectral density is changed to connect with the extended third leg.

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The AP1000 augmented power spectral density, anchored to 0.3 g, is as follows:

$S_0(f) = 58.5\ (f/2.5)^{0.2}\ in^2/sec^3$, $f \leq 2.5$ hertz

$S_0(f) = 58.5\ (2.5/f)^{1.8}\ in^2/sec^3$, 2.5 hertz $\leq f \leq$ 9 hertz

$S_0(f) = 5.832\ (9/f)^3\ in^2/sec^3$, 9 hertz $\leq f \leq$ 25 hertz

$S_0(f) = 0.27\ (25/f)^8\ in^2/sec^3$, 25 hertz $\leq f$

The AP1000 Minimum Power Spectral Density is presented in Figure 3.7.1-9. This AP1000 target power spectral density is compatible with the AP1000 horizontal design response spectra and envelops a target power spectral density compatible with the AP1000 vertical design response spectra. This AP1000 target power spectral density, therefore, is conservatively applied to the vertical response spectra.

The comparison plots of the power spectral density curve of the AP1000 acceleration time histories versus the target power spectral density curve are presented in Figures 3.7.1-10, 3.7.1-11, and 3.7.1-12. The power spectral density functions of the design time histories are calculated at uniform frequency steps of 0.0489 hertz. The power spectral densities presented in Figures 3.7.1-10 through 3.7.1-12 are the averaged power spectral density obtained over a moving frequency band of ±20 percent centered at each frequency. The power spectral density amplitude at frequency (f) has the averaged power spectral density amplitude between the frequency range of 0.8 f and 1.2 f as stated in appendix A of Revision 2 of SRP 3.7.1.

3.7.1.3 Critical Damping Values

Energy dissipation within a structural system is represented by equivalent viscous dampers in the mathematical model. The damping coefficients used are based on the material, load conditions, and type of construction used in the structural system. The safe shutdown earthquake damping values used in the dynamic analysis of various structures, supports, and equipment are presented in Table 3.7.1-1. The damping values are based on Regulatory Guide 1.61, Revision 0, ASCE Standard 4-98 (Reference 3), except for the damping value of the primary coolant loop piping, which is based on Reference 22, and conduits, cable trays and their related supports.

The damping values for conduits, cable trays and their related supports are shown in Table 3.7.1-1. The damping value of conduit, empty cable trays, and their related supports is similar to that of a bolted structure, namely 7 percent of critical. The damping value of filled cable trays and supports increases with increased cable fill and level of seismic excitation. Full cable trays use a 10-percent damping value consistent with RG 1.61, Revision 1. The limiting condition for design of the AP1000 standard cable tray supports is for full cable tray weight.

For structures or components composed of different material types, the composite modal damping is calculated using the stiffness-weighted method based on Reference 3. The modal damping values equal:

$$\beta_n = \sum_{i=1}^{nc} \frac{\{\phi_n\}^T \beta_i [K_t]_i \{\phi_n\}}{\{\phi_n\}^T [K_t] \{\phi_n\}}$$

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where:

β_n = ratio of critical damping for mode n

nc = number of elements

$\{\phi_n\}$ = mode n (eigenvector)

$[K_t]_i$ = stiffness matrix of element i

β_i = ratio of critical damping associated with element i

$[K_t]$ = total system stiffness matrix

The linear structural damping values were defined in the modeling codes as a parameter of material property defined for each element. This form of structural damping is used for seismic time history analyses. The structural models analyzed follow the damping criteria stated in Table 3.7.1-1 using 5-percent SSE damping for steel composite (SC) structures, including the shield building wall and modules, and 7-percent SSE damping for the remaining reinforced concrete (RC) structures throughout the nuclear island. A time history non-linear analysis confirms only minor cracking in the nuclear island structure.

3.7.1.4 Supporting Media for Seismic Category I Structures

The supporting media will be described consistent with the information items in Subsection 2.5.4. Seismic analyses for both rock and soil sites are described in Subsection 3.7.2 and Appendix 3G.

The AP1000 nuclear island consists of three seismic Category I structures founded on a common basemat. The three structures that make up the nuclear island are the coupled auxiliary and shield buildings, the steel containment vessel, and the containment internal structures. [*The nuclear island is shown in Figure 3.7.1-14.*]* The foundation embedment depth, foundation size, and total height of the seismic Category I structures are presented in Table 3.7.1-2.

For the design of seismic Category I structures, a set of six design soil profiles (that include hard rock) of various shear wave velocities is established from parametric studies as described in Appendix 3G.3. The soil cases selected for the AP1000 use parameters from the AP600 design, and the AP600 conclusions are applicable to the AP1000 due to the identical footprint to the AP600 and the similarity in overall mass.

For the AP1000 2D and 3D soil-structure interaction analyses, although some of the parabolic soil profiles are defined using a depth of 240 feet, the actual soil profile defined in SASSI (System for Analysis of Soil-Structure Interaction) (base rock) goes to only elevation 120′.

Soil-structure interaction analyses on soil sites for the AP1000 used the latest soil degradation curves recommended by EPRI TR-102293, although these represent more recent soils data and differ slightly for those used for the AP600.

These six profiles are sufficient to envelope sites where the shear wave velocity of the supporting medium at the foundation level exceeds 1000 feet per second (see Subsection 2.5.2). The design soil profiles include a hard rock site, a soft rock site, a firm rock site, an upper bound soft-to-medium soil site, a soft-to-medium soil site, and a soft soil site. The shear wave velocity profiles and related governing parameters of the six sites considered are as follows:

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- For the hard rock site, an upper bound case for rock sites using a shear wave velocity of 8000 feet per second.
- For the firm rock site, a shear wave velocity of 3500 feet per second to a depth of 120 feet and base rock at the depth of 120 feet.
- For the soft rock site, a shear wave velocity of 2400 feet per second at the ground surface, increasing linearly to 3200 feet per second at a depth of 240 feet, and base rock at the depth of 120 feet.
- For the upper bound soft-to-medium soil site, a shear wave velocity of 1414 feet per second at ground surface, increasing parabolically to 3394 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water at grade level. The initial soil shear modulus profile is twice that of the soft-to-medium soil site.
- For the soft-to-medium soil site, a shear wave velocity of 1000 feet per second at ground surface, increasing parabolically to 2400 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water is assumed at grade level.
- For the soft soil site, a shear wave velocity of 1000 feet per second at ground surface, increasing linearly to 1200 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water is assumed at grade level

The strain-dependent shear modulus curves for the foundation materials, together with the corresponding damping curves are taken from References 37 and 38 and are shown in Figures 3.7.1-15 and 3.7.1-16 for rock material and soil material respectively. The different curves for soil in Figure 3.7.1-16 apply to the range of depth within a soil column below grade. The strain-dependent soil material damping is limited to 15 percent of critical damping. The strain-dependent properties used in the SSI analyses for the safe shutdown earthquake are shown in Table 3.7.1-4 and Figure 3.7.1-17 for the firm rock, soft rock, upper bound soft-to-medium soil, soft-to-medium soil, and soft soil properties.

Some variation of soil modeling (water table, soil layering, soil degradation model, and the like) and combinations of these have been demonstrated to have no significant effect on the seismic response of the nuclear island structures. The governing parameters obtained for the AP600 soil studies are also applicable to the AP1000. Each of the parameters deemed not significant has been analyzed.

For instance, the combination of effects of the different strain dependent soil parameters that affect the strain-iterated shear wave velocity profiles was evaluated and shown not to result in exceedances of the envelope of the generic seismic design in-structure response spectra (ISRS).

3.7.2 Seismic System Analysis

Seismic Category I structures, systems, and components are classified according to Regulatory Guide 1.29. Seismic Category I building structures of AP1000 consist of the containment building (the steel containment vessel and the containment internal structures), the shield building, and the auxiliary building. These structures are founded on a common basemat and are collectively known as the nuclear island or nuclear island structures. [*Key dimensions, such as thickness of the basemat, floor slabs, roofs and walls, of the seismic Category I building structures are shown in Figure 3.7.2-12.*]*

Seismic systems are defined, according to SRP 3.7.2, Section II.3.a, as the seismic Category I structures that are considered in conjunction with their foundation and supporting media to form a soil-structure interaction model. The following subsections describe the seismic analyses performed

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for the nuclear island. Other seismic Category I structures, systems, equipment, and components not designated as seismic systems (that is, heating, ventilation, and air-conditioning systems; electrical cable tray systems; piping systems) are designated as seismic subsystems. The analysis of seismic subsystems is presented in Subsection 3.7.3.

Seismic Category I building structures are on the nuclear island. Other building structures are classified nonseismic or seismic Category II. Nonseismic structures are analyzed and designed for seismic loads according to the Uniform Building Code (Reference 2) requirements for Zone 2A. The main area of the turbine building structure is analyzed and designed for seismic loads in accordance with International Building Code requirements for an earthquake magnitude equivalent to the Uniform Building Code, Zone 3. Seismic Category II building structures are designed for the safe shutdown earthquake using the same methods and design allowables as are used for seismic Category I structures. The acceptance criteria are based on ACI 349 for concrete structures, on ANSI/AISC N690 for steel structures, including the supplemental requirements described in Subsections 3.8.4.4.1 and 3.8.4.5, and on AWS D1.1-2000 for weld design, qualification, fabrication, and inspection as described in Subsections 3.8.3.2 and 3.8.4.2.

Separate seismic analyses are performed for the nuclear island for each of the six design soil profiles defined in Subsection 3.7.1.4. The analyses generate one set of in-structure responses for each of the design soil profiles. The six sets of in-structure responses are enveloped to obtain the seismic design envelope (design member forces, nodal accelerations, nodal displacements, and floor response spectra), which are used in the design and analysis of seismic Category I structures, components, and seismic subsystems.

Appendix 3G summarizes the types of models and analysis methods that are used in the seismic analyses of the nuclear island, as well as the type of results that are obtained and where they are used in the design. The seismic analyses of the nuclear island are summarized in a seismic analysis summary report. This report describes the development of the finite element models, the soil structure interaction and fixed base analyses, and the results thereof. Seismic response spectra are given in Appendix 3G for the six key locations:

- Containment internal structures at reactor vessel support elevation 100.00′.
- Containment internal structures at operating deck elevation 134.25′.
- Auxiliary shield building north east corner at control room floor elevation 116.50′.
- Auxiliary shield building corner of fuel building roof at shield building elevation 179.19′.
- Auxiliary shield building roof area elevation 327.41′.
- Steel containment vessel near polar crane elevation 224.000′.

3.7.2.1 Seismic Analysis Methods

Seismic analyses of the nuclear island are performed in conformance with the criteria within SRP 3.7.2.

Seismic analyses – using response spectra analysis, the equivalent static acceleration method, the mode superposition time-history method, and the complex frequency response analysis method – are performed for the safe shutdown earthquake to determine the seismic force distribution for use in the design of the nuclear island structures, and to develop in-structure seismic responses (accelerations, displacements, and floor response spectra) for use in the analysis and design of seismic subsystems.

3.7.2.1.1 Equivalent Static Acceleration Analysis

Equivalent static analyses, using computer program ANSYS (Reference 36), are performed to obtain the seismic forces and moments required for the structural design of the steel containment vessel

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and the nuclear island basemat (see Subsection 3.8.2.4.1.1). Equivalent static loads are applied to the finite element models using the maximum acceleration results from the time history analyses for the six design soil profiles. Accidental torsional moments are applied as described in Subsection 3.7.2.11.

Equivalent static analyses are also performed for design of the shield building roof and radial roof beams, passive containment cooling water storage tank (PCWST), tension ring, and air inlet structure (see Subsection 3.8.4.4.1). The equivalent static loads are based on the maximum acceleration results from time history dynamic analysis of the nuclear island in Subsection 3.7.2.1.2.

3.7.2.1.2 Time-History Analysis and Complex Frequency Response Analysis

Mode superposition time-history analyses using computer program ANSYS and complex frequency response analysis using computer program SASSI are performed to obtain the in-structure seismic response needed in the analysis and design of seismic subsystems. Three-dimensional finite element shell models of the nuclear island structures are used in conjunction with the design soil profiles presented in Subsection 3.7.1.4 to obtain the in-structure responses. Stick models are coupled to the shell models of the concrete structures for the containment vessel, polar crane, reactor coolant loop, pressurizer, and core makeup tanks. Three models are used. The fine (NI10) model, as described in Subsection 3G.2.2.1, is used to define the seismic response for the hard rock site. The coarse (NI20) model, as described in Subsection 3G.2.2.2, is used for the soil structure interaction (SSI) analyses and is set up in both ANSYS and SASSI. The NI05 model, as described in Appendix 3G.2.2.4, is used to develop amplified seismic response for the envelope of soil profiles presented in Subsection 3.7.1.4 for flexible regions not captured by the coarser NI20 model. The models and analyses are described in Appendix 3G.

For the hard rock site, the soil-structure interaction effect is negligible. Therefore, for the hard rock site, the nuclear island is analyzed as a fixed-base structure, using computer program ANSYS without the foundation media. The three components of earthquake (two horizontal and one vertical time histories) are applied simultaneously in the analysis. Since the NI10 finite element model of the auxiliary and shield building uses shell elements to represent the 6-foot-thick basemat, the nodes of the basemat element are at the center of the basemat (elevation 63′-6″). The finite element model of the containment internal structures uses solid elements, which extend down to elevation 60′-6″. When the finite element models are combined and used in the time history analyses, the auxiliary building finite element model is fixed at the shell element basemat nodes (elevation 63′-6″) and the base of the containment internal structures is fixed at the bottom of the solid element base nodes (elevation 60′-6″). This difference in elevation of the base fixity is not significant since the concrete between elevations 60′-6″ and 63′-6″, below the auxiliary building, is nearly rigid. There is no lateral support due to soil or hard rock below grade. This case results in higher response than a case analyzed with full lateral support below grade.

For additional information on the method used to calculate displacement, see Appendix 3G.4.1 and Appendix 3G.4.2.

3.7.2.1.3 Response Spectrum Analysis

Response spectral analysis is used for the evaluation of the nuclear island structures. Response spectrum analyses are used to perform an analysis of a particular structure or portion of structure using the procedures described in Appendix 3G.4.3.1 and Subsections 3.7.2.6, 3.7.2.7, and 3.7.3.

Seismic response spectrum analysis of the auxiliary building, shield building, and containment internal structure is performed to develop the seismic design loads for these buildings, and the loads generated include the amplified load due to flexibility and the distribution of this load to the surrounding structures.

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3.7.2.2 Natural Frequencies and Response Loads

Modal analyses are performed for the shell and lumped-mass stick models of the seismic Category I structures on the nuclear island, as described in Appendix 3G. Seismic response spectra at the six key locations (Subsection 3.7.2) are given in Appendix 3G.

3.7.2.3 Procedure Used for Modeling

Based on the general plant arrangement, three-dimensional, finite element models are developed for the nuclear island structures: a finite element model of the coupled shield and auxiliary buildings, a finite element model of the containment internal structures, a finite element model of the shield building roof, and an axisymmetric shell model of the steel containment vessel. These three-dimensional, finite element models provide the basis for the development of the dynamic model of the nuclear island structures.

The finite element models of the coupled shield and auxiliary buildings, and the containment internal structures are based on the gross concrete section with the modulus based on the specified compressive strength of concrete reduced by a factor of 0.8 to consider the effect of cracking as recommended in Table 6-5 of FEMA 356 (Reference 5). This 80-percent value is supported by non-linear ABAQUS analyses performed on the nuclear island finite element model. The comparison between linear and non-linear models shows that the 80-percent stiffness model response spectra enveloped the non-linear model, providing a conservative approach in terms of response spectra and maximum stresses obtained in the shield building wall.

Seismic subsystems coupled to the overall dynamic model of the nuclear island include the coupling of the reactor coolant loop model to the model of the containment internal structures, and the coupling of the polar crane model to the model of the steel containment vessel. The criteria used for decoupling seismic subsystems from the nuclear island model are according to Section II.3.b of SRP 3.7.2, Revision 2. The total mass of other major subsystems and equipment is less than one percent of the respective supporting nuclear island structures; therefore, the mass of other major subsystems and equipment is included as concentrated lumped-mass only.

Several minor (basic building configuration not modified) design changes and model improvements include the following:

- Provision for heavier fuel racks in the spent fuel pool area. Fuel and rack masses are updated, and pool water volumes are modeled as lumped masses.
- Changes in the annulus configuration are incorporated into the dish model, and lower shield building and upper containment internal structures basemat nodes and elements are modified for compatibility.
- The core makeup tanks were added as stick models.
- The polar crane model has an updated weight and updated steel containment vessel local stiffness, and now includes polar crane truck stiffness.

The seismic analysis of the water inside the PCCWST was performed for the AP600. It was concluded that the low-frequency sloshing mode is not significant to the response of the nuclear island away from the shield building roof and that this conclusion could be extended to the AP1000 design. Further analysis indicated that the sloshing mass ratio remained essentially unchanged between the AP600 and AP1000.

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3.7.2.3.1 Coupled Shield and Auxiliary Buildings and Containment Internal Structures

The finite element models of the coupled shield and auxiliary buildings and the reinforced concrete portions of the containment internal structures are based on the gross concrete section with the modulus based on the specified compressive strength of concrete of contributing structural walls and slabs. The properties of the concrete-filled structural modules are computed using the combined gross concrete section and the transformed steel face plates of the structural modules. The modulus is reduced by a factor of 0.8 to consider the effect of cracking. Furthermore, the weight density of concrete plus the uniformly distributed miscellaneous dead weights are considered by adding surface mass or by adjusting the material mass density of the structural elements. An equivalent tributary slab area load of 50 pounds per square foot is considered to represent miscellaneous deadweight such as minor equipment, piping and raceways. 25 percent of the floor live load or 75 percent of the roof snow load, whichever is applicable, is considered as mass in the global seismic models.

Major equipment weights are distributed over the floor area or are included as concentrated lumped masses at the equipment locations. The major equipment supported by the containment internal structures is represented by stick models connected to the containment internal structures, and includes the reactor coolant loop, the pressurizer, and the core makeup tank. The core makeup tank model is used only in the nuclear island fine (NI10) model; the core makeup tank is represented by mass in the nuclear island coarse model (NI20). The finite element models of the coupled shield and auxiliary buildings and the containment internal structures are described in Appendix 3G. The auxiliary and shield building is modeled with shell elements and the base of the finite element model is at the middle of the basemat at elevation 63′-6″. The bottom of the containment and internal structures are modeled with solid elements and the base of the finite element model is at the underside of the basemat at elevation 60′-6″. The interface between the models is at a radius of 71′-0″ at the mid-surface of the shield building.

3.7.2.3.2 Steel Containment Vessel

The steel containment vessel is a freestanding, cylindrical, steel shell structure with ellipsoidal upper and lower steel domes. The three-dimensional, lumped-mass stick model of the steel containment vessel is developed based on the axisymmetric shell model. Figure 3G.2-4 presents the steel containment vessel stick model. In the stick model, the properties are calculated as follows:

- Members representing the cylindrical portion are based on the properties of the actual circular cross section of the containment vessel.

- Members representing the bottom head are based on equivalent stiffnesses calculated from the shell of revolution analyses for static 1.0g in vertical and horizontal directions.

- Shear, bending and torsional properties for members representing the top head are based on the average of the properties at the successive nodes, using the actual circular cross section. These are the properties that affect the horizontal modes. Axial properties, which affect the vertical modes, are based on equivalent stiffnesses calculated from the shell of revolution analyses for static 1.0g in the vertical direction.

The equivalent static acceleration analyses of the containment vessel use a finite element shell model with a refined mesh in the area adjacent to the large penetrations. Comparison of this with a time history analysis for the regions immediately surrounding the large penetrations verifies that the loads from equivalent static analysis are conservative to time history using a representative study.

This method used to construct a stick model from the axisymmetric shell model of the containment vessel is verified by comparison of the natural frequencies determined from the stick model and the shell of revolution model as shown in Table 3G.2-2. The shell of revolution vertical model (n = 0

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harmonic) has a series of local shell modes of the top head above elevation 265′ between 23 and 30 hertz. These modes are predominantly in a direction normal to the shell surface and cannot be represented by a stick model. These local modes have small contribution to the total response to a vertical earthquake as they are at a high frequency where seismic excitation is small. The only seismic Category I components attached to this portion of the top head are the water distribution weirs of the passive containment cooling system. These weirs are designed such that their fundamental frequencies are outside the 23 to 30 hertz range of the local shell modes.

Additional details of the steel containment vessel stick model are included in Appendix 3G.2.1.3.

The containment air baffle, presented in Subsection 3.8.4.1.3, is supported from the steel containment vessel at regular intervals so that a gap is maintained for airflow. It is constructed with individual panels which do not contribute to the stiffness of the containment vessel. The fundamental frequency of the baffle panels and supports is about twice the fundamental frequency of the containment vessel. The mass of the air baffle is small, equal to approximately 10 percent of the vessel plates to which it is attached. The air baffle, therefore, is assumed to have negligible interaction with the steel containment vessel. Only the mass of the air baffle is considered and added at the appropriate elevations of the steel containment vessel stick model.

The interaction between the polar crane and the containment vessel is significant and is included in the model. This polar crane model reflects the polar crane wheel assemblies. The polar crane is supported on a ring girder which is an integral part of the steel containment vessel at elevation 228′-0″ as shown in Figure 3.8.2-1. It is modeled as a multi-degree of freedom system attached to the steel containment shell at elevation 224′ (midpoint of ring girder) as shown in Figure 3G.2-4. The polar crane is modeled as shown in Figure 3G.2-5A with five masses at the mid-height of the bridge at elevation 236′-6″ and one mass for the trolley. The polar crane model includes the flexibility of the crane bridge girders and truck assembly. The containment shell's local flexibility is considered when combined with the global model. When fixed at the center of containment, the model shows fundamental frequencies of 4.6 hertz transverse to the bridge, 6.9 hertz vertically, and 9.4 hertz along the bridge.

[*During plant operating conditions, the polar crane is parked in the plant north-south direction with the trolley located at one end near the containment shell.*]* In the seismic model, the crane bridge spans in the north-south direction and the mass eccentricity of the trolley is considered by locating the mass of the trolley at the northern limit of travel of the main hook. Furthermore, the mass eccentricity of the two equipment hatches and the two personnel airlocks are considered by placing their mass at their respective center of mass as shown in Figure 3G.2-4. Any modeling change due to the as-procured crane data is resolved with the COL holder item in Subsection 3.7.5.4, "Reconciliation of Seismic Analyses of Nuclear Island Structures."

3.7.2.3.3 Nuclear Island Seismic Model

The nuclear island seismic models are described in Appendix 3G. The various building models are interconnected to form the overall dynamic model of the nuclear island. The mass properties of the models include all tributary mass expected to be present during plant operating conditions. This includes the dead weight of walls and slabs, weight of major equipment, and equivalent tributary slab area loads representing miscellaneous equipment, piping and raceways.

The hydrodynamic mass effect of the water within the passive containment cooling system water tank on the shield building roof, the in-containment refueling water storage tank within the containment internal structures, and the spent fuel pool in the auxiliary building is evaluated. Since the water in the PCCWST responds at a very low frequency (sloshing) and does not affect building response, the PCCWST water horizontal mass is reduced to exclude the low frequency water sloshing mass. The total mass of the water in the in-containment refueling water storage tank within the containment

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internal structures, and the spent fuel pool in the auxiliary building is included in the nuclear island seismic model.

Seismic response spectra are developed at the locations of the nodes. These response spectra are grouped and enveloped to define the seismic design response spectra. The nodes associated with a specific elevation and building structure (i.e., auxiliary and shield building and containment internal structures) are grouped. For the auxiliary and shield building where the floor at the elevation of interest is rigid (i.e. frequency > 33 hertz), it is only necessary to envelop the response spectra at edge points and interior nodes at the shield wall to obtain the largest seismic response spectra because of rigid motion. The edge nodes reflect the largest rocking and translational response of the auxiliary building, and the response spectra associated with the nodes on the shield wall will reflect the shield wall dynamic response. It is not necessary to include any nodes between the shield wall and auxiliary building edge since the floor is rigid, and the response cannot be worse than those enveloped.

A refined finite element shell model of the nuclear island concrete structures is reviewed for flexible regions, which may produce amplified response spectra. This model, called the NI05 model, has a tetrahedral mesh size of approximately 5 feet by 5 feet. Each of the principal walls and floors in the auxiliary and shield building as well as the containment internal structures are reviewed. A modal analysis of the NI05 model for both auxiliary and shield building and containment internal structures is reviewed for each of these regions for the existence of out of plane modes, which are considered flexible (less than 33 hertz) with significant participating mass. The survey reveals that some regions, typically in the middle of a floor or wall, exhibit amplified behavior compared to the critical nodes at the corner and edge building locations. [*These regions, which have flexible areas, are evaluated in one of two ways:*

- *Flexible areas, which have been previously identified, have amplified response spectra developed directly from the time history analyses for the envelope of soil sites.*

- *Flexible regions, which require a detailed analysis to obtain the amplified response spectra, use input directly from time history analysis. The NI05 finite element model is used to capture out-of-plane flexibilities that, because of mesh refinement, a more course model could not capture.*

If equipment or a structure is supported at more than one elevation, then the seismic input as an envelope of multiple groups based on the support locations will be defined. Therefore, if the equipment or structure is supported on rigid and flexible floor areas the response spectra (horizontal and vertical directions) used by the analysts will be the envelope of the rigid and flexible areas that include inside and outside nodes.

If an equipment or structure is supported exclusively by a floor or wall, only that spectra will be used for design.]*

3.7.2.4 Soil-Structure Interaction

Soil-structure interaction is not significant for the nuclear island founded on rock with a shear wave velocity greater than 8000 feet per second. The soil-structure interaction analyses for the firm rock and soil sites are described in Appendix 3G.

The computer program SASSI is used to perform the soil-structure interaction analysis. The SASSI model of the nuclear island is based on the NI20 Coarse Finite Element model. Soil-structure interaction analyses are performed based on the nuclear island 3D SASSI model for the three soil conditions established from the AP1000 2D SASSI analyses, in addition to soft rock and soft soil.

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SASSI uses key frequencies to perform its transfer function calculations. For a large model, resting on a very stiff soil (hard rock), SASSI gives conservative results at high frequencies. The significant responses for AP1000 soil cases occur at less than 10 hertz so the SASSI model is adequate for use.

Analyses are performed with large solid-shell finite element models at two levels. The fine (NI10) model is used to define the seismic response for the hard rock site. The coarse (NI20) model is used for the soil-structure interaction analyses. The NI20 coarse model has fewer nodes and elements than the NI10 model. It captures the essential features of the nuclear island configuration. The nominal shell and solid element dimension is about 20 feet.

3.7.2.5 Development of Floor Response Spectra

The design floor response spectra are generated according to Regulatory Guide 1.122.

Seismic floor response spectra are computed using time-history responses determined from the nuclear island seismic analyses. The time-history responses for the hard rock condition are determined from a mode superposition time history analysis using computer program ANSYS.

The time-history responses for the firm rock and soil conditions are determined from a complex frequency response analysis using computer program SASSI. Floor response spectra for damping values equal to 2, 3, 4, 5, 7, 10, and 20 percent of critical damping are computed at the required locations.

The floor response spectra for the design of subsystems and components are generated by broadening the enveloped nodal response spectra determined for the hard rock site and soil sites.

The spectral peaks are broadened by ±15 percent to account for the variation in the structural frequencies, due to the uncertainties in parameters such as material and mass properties of the structure and soil, damping values, seismic analysis technique, and the seismic modeling technique. Figure 3.7.2-14 shows the broadening procedure used to generate the design floor response spectra. Spectral peaks at frequencies associated with fundamental soil structure interaction frequencies are reviewed. If there is a “valley” between peaks due to different soil profiles and not the building modal response, then this valley is filled by extending the broadening of the lower peak horizontally until it meets the broadened upper peak.

Floor response spectra for the auxiliary building are obtained from the three-dimensional model as described in Appendix 3G. These spectra are developed for the specific location in the auxiliary building. Where spectra at a number of nodes have similar characteristics, a single set of spectra may be developed by enveloping the broadened spectra at each of the nodes.

The safe shutdown earthquake floor response spectra for 5 percent damping, at representative locations of the coupled auxiliary and shield buildings, the steel containment vessel, and the containment internal structures are presented in Appendix 3G.

3.7.2.6 Three Components of Earthquake Motion

Seismic system analyses are performed considering the simultaneous occurrences of the two horizontal and the vertical components of earthquake.

In mode superposition time-history analyses using computer program ANSYS, the three components of earthquake are applied either simultaneously or separately. In the ANSYS analyses with the three earthquake components applied simultaneously, the effect of the three components of earthquake motion is included within the analytical procedure so that further combination is not necessary.

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In analyses with the earthquake components applied separately and in the response spectrum and equivalent static analyses, the effect of the three components of earthquake motion are combined using one of the following methods:

- For seismic analyses with the statistically independent earthquake components applied separately, the time-history responses from the three earthquake components are combined algebraically at each time step to obtain the combined response time-history. This method is used in the SASSI analyses.

- The peak responses due to the three earthquake components from the response spectrum and equivalent static analyses are combined using the square root of the sum of squares (SRSS) method.

- The peak responses due to the three earthquake components from the equivalent static analyses are combined directly, using the assumption that when the peak response from one component occurs, the responses from the other two components are 40 percent of the peak (100 percent-40 percent-40 percent method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus), are considered. This method is used in the nuclear island basemat analyses, the containment vessel analyses, and the shield building roof analyses.

The containment vessel is analyzed using axisymmetric finite element models. These axisymmetric building structures are analyzed for one horizontal seismic input from any horizontal direction and one vertical earthquake component. Responses are combined by either the square root of the sum of squares method or by the 100 percent-40 percent-40 percent method in which one component is taken at 100 percent of its maximum value and the other components are taken at 40 percent of their maximum value.

For the seismic responses presented in Appendix 3G, the effect of three components of earthquake are considered as follows:

- Mode Superposition Time History Analysis (program ANSYS) and the Complex Frequency Response Analysis (program SASSI) – the time history responses from the three components of earthquake motion are combined algebraically at each time step.

A summary of the dynamic analyses performed and the combination techniques used are presented in Appendix 3G.

3.7.2.7 Combination of Modal Responses

The modal responses of the response spectrum system structural analysis are combined using the procedures described in Appendix 3G.4.3. In the fixed base mode superposition time history analysis of the hard rock site, the total seismic response is obtained by superposing the modal responses within the analytical procedure so that further combination is not necessary.

A summary of the dynamic analyses performed and the combination techniques used are presented in Appendix 3G.

3.7.2.8 Interaction of Seismic Category II and Nonseismic Structures with Seismic Category I Structures, Systems, or Components

Nonseismic structures are evaluated to determine that their seismic response does not preclude the safety functions of seismic Category I structures, systems or components. This is accomplished by satisfying one of the following:

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- The collapse of the nonseismic structure will not cause the nonseismic structure to strike a seismic Category I structure, system or component.

- The collapse of the nonseismic structure will not impair the integrity of seismic Category I structures, systems or components.

- The structure is classified as seismic Category II and is analyzed and designed to prevent its collapse under the safe shutdown earthquake.

The structures adjacent to the nuclear island are the annex building, the radwaste building, and the turbine building.

3.7.2.8.1 Annex Building

The portion of the annex building adjacent to the nuclear island is classified as seismic Category II. The structural configuration is shown in Figure 3.7.2-19. The annex building is analyzed for the safe shutdown earthquake for the six soil profiles described in Subsection 3.7.1.4. For the hard rock site, a range of soil properties is assumed for the layer above rock at the level of the nuclear island foundation. Seismic input is defined by response spectra applied at the base of a dynamic model of the annex building. The seismic response spectra input at the base of the annex building are the envelopes of the range of soil sites and also envelope the AP1000 design free field ground spectra shown in Figures 3.7.1-1 and 3.7.1-2. The envelope of the maximum building response acceleration values is applied as equivalent static loads to a more detailed static model. See Subsection 3.7.2.8.4 for more discussion of modeling and seismic analysis.

The minimum space required between the annex building and the nuclear island to avoid contact is obtained by absolute summation of the deflections of each structure obtained from either a time history or a response spectrum analysis for each structure. The maximum displacement of the roof of the annex building is 1.6 inches in the east-west direction. The minimum clearance between the structural elements of the annex building above grade and the nuclear island is 3 inches.

3.7.2.8.2 Radwaste Building

The radwaste building is classified as nonseismic and is designed to the seismic requirements of the Uniform Building Code, Zone 2A with an Importance Factor of 1.25. As shown in the radwaste building general arrangement in Figure 1.2-22, it is a small steel framed building. If it were to impact the nuclear island or collapse in the safe shutdown earthquake, it would not impair the integrity of the reinforced concrete nuclear island. The gap between the structural elements of the radwaste building above grade and the nuclear island is 4 inches.

Three methods are used to demonstrate that a potential radwaste building impact on the nuclear island during a seismic event will not impair its structural integrity:

- The maximum kinetic energy of the impact during a seismic event considers the maximum radwaste building and nuclear island velocities. The total kinetic energy is considered to be absorbed by the nuclear island and converted to strain energy. The deflection of the nuclear island is less than 0.2″. The shear forces in the nuclear island walls are less than the ultimate shear strength based on a minus one standard deviation of test data.

- Stress wave evaluation shows that the stress wave resulting from the impact of the radwaste building on the nuclear island has a maximum compressive stress less than the concrete compressive strength.

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- An energy comparison shows that the kinetic energy of the radwaste building is less than the kinetic energy of tornado missiles for which the exterior walls of the nuclear island are designed.

3.7.2.8.3 Turbine Building

The south end of the turbine building is separated from the rest of the turbine building by a 2'-0" thick reinforced concrete wall that provides a robust structure around the first bay. This wall isolates the first bay of the turbine building from the general area of the turbine building and from the adjacent yard area. The main segment of this wall is located on column line 11.2. This wall extends from El.100'-0" basemat to El.169'-0". The first bay of the turbine building is classified as seismic Category II. The other bays are classified as non-seismic. The modeling used to represent the structural configuration in the soil structure interaction analysis is shown in Figure 3.7.2-21.

The first bay of the turbine building is analyzed for the safe shutdown earthquake for the six soil profiles described in Subsection 3.7.1.4. For the hard rock site, a range of soil properties is assumed for the layer above rock at the level of the nuclear island foundation. Seismic input is defined by response spectra applied at the base of a dynamic model of the first bay of the turbine building. The seismic response spectra input at the base of the first bay of the turbine building are the envelopes of the range of soil sites and also envelope the AP1000 design free field ground spectra shown in Figures 3.7.1-1 and 3.7.1-2. See Subsection 3.7.2.8.4 for more discussion of modeling and seismic analysis.

The first bay is designed in accordance with ACI-349 for concrete features, ANSI/AISC N690 for steel features, including the supplemental requirements described in Subsections 3.8.4.4.1 and 3.8.4.5, and AWS D1.1-2000 for weld design, qualification, fabrication, and inspection as described in Subsections 3.8.3.2 and 3.8.4.2.

For the non-seismic portion of the Turbine Building, seismic design is upgraded in order to provide margin against collapse during the safe shutdown earthquake. The turbine building is a braced steel frame structure designed to meet the following criteria:

- The turbine building is designed in accordance with the 2006 International Building Code (Reference 40). This references ACI-318 for concrete structures and AISC for steel structures. Seismic loads are defined in accordance with the International Building Code with the maximum considered earthquake spectral parameters S_{DS} = 0.9, S_{D1} = 0.54 for Site Class D. This is consistent with the 1997 Uniform Building Code provisions for Zone 3 with an Importance Factor of 1.0. For a braced structure that has a mix of eccentric and special concentric bracing, the response modification factor is 6 (ASCE 7-05, Reference 42) using strength design.

- The design complies with the seismic requirements for eccentrically braced frames and special concentrically braced frames given in the 2005 AISC Seismic Provisions for Structural Steel Buildings (Reference 41). Quality assurance is in accordance with ASCE 7-05 (Reference 42).

3.7.2.8.4 Seismic Modeling and Analysis of Seismic Category II Building Structures

Seismic Category II structures, systems, and components are designed so that the safe shutdown earthquake does not cause unacceptable structural failure or interaction with seismic Category I items. Therefore, the seismic response of seismic Category II buildings must be obtained so that they can be designed to meet the seismic Category II requirements as given in Subsection 3.2.1.1.2. Seismic Category II structures are analyzed and evaluated in the same manner as seismic Category

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l structures. The foundation of the non-seismic portion is modeled with the associated mass distributed on it so that the soil structure interaction during a seismic event is reflected in the analysis.

The seismic analyses performed for the adjacent seismic Category II structures are simulated 3D analyses. The seismic analyses are performed primarily using 2D SASSI models. To properly account for the 3D effect, the response from 2D and 3D SASSI analyses of the seismic Category II buildings on rigid foundations are compared and a 3D effects factor is developed from this comparison. Three soil cases (upper bound soft to medium UBSM, soft to medium SM, and soft soil SS) are used to determine the 3D factor. Shown in Figures 3.7.2-20 and 3.7.2-21 are the 2D SASSI models with adjacent building structures. The seismic Category II buildings are modeled as stick models. The 3D model with adjacent structures is shown in Figure 3.7.2-22.

Seismic Category II buildings are designed using envelope foundation input response spectra (FIRS). The development of these FIRS shall be based on a number of analyses results from the SASSI analyses. The seismic Category II FIRS shall be the envelope of the SASSI seismic Category II foundation response spectra resulting from the following seismic inputs/soil profiles:

- AP1000 CSDRS – Hard rock at El. 60.5′.
- AP1000 CSDRS – Firm rock, soft rock, upper bound soft to medium, soft to medium, and soft soil, soil profiles with AP1000 CSDRS spectra input at plant grade; and
- AP1000 hard rock high frequency (HRHF) – For rock sites, HRHF at plant grade shall be developed using AP1000 HRHF spectra at El. 60.5′ and a range of backfill soil profiles. The backfill soil under the annex and turbine buildings has a parabolic soil profile as a function of depth (El. 100′ to El. 60.5′) and uses EPRI (1993) strain dependent curves. The HRHF at plant grade spectrum shall be generated using soil profiles corresponding to a shear wave velocity of 500 fps, 750 fps, and 1000 fps at El. 100′. The HRHF at plant grade shall be used as input to SASSI analyses to determine the FIRS at the base of the seismic Category II structures.

For each soil case, 2D SASSI analyses shall be performed and the results at three locations at the base of the seismic Category II structures are enveloped. The maximum bearing demand and maximum relative displacement shall be established from the 2D SASSI analyses. The 3D effect factor is applied to the envelope foundation spectra and used for the design of the annex building and turbine building first bay.

Response spectrum analyses (using detailed finite element building models) shall be used to obtain seismic design loads for the seismic Category II building design. The seismic input to the response spectrum analyses is the envelope foundation response spectra obtained from the SASSI analyses. The COL applicant will perform the following screening criteria to determine if it has to perform further analysis for its site. If the requirements given below are not met, then the site applicant can perform site-specific analyses to demonstrate that its site-specific seismic Category II foundation seismic response spectra are less than the AP1000 annex building and turbine building first bay generic design envelope foundation spectra.

- The site meets Subsection 2.5.4.2 DCD soil uniformity requirements.
- For soil sites, the site GMRS is enveloped by the AP1000 CSDRS with soil profiles SS, SM, UBSM, SR, FR, and HR.

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- For HRHF sites, the site GMRS is enveloped by the AP1000 HRHF response spectra with a minimum backfill surface shear wave velocity of 500 fps, and a minimum lateral extent of the backfill corresponding to a line extending down from the surface at a one horizontal to one vertical (1H:1V) slope from the outside footprint limit of the seismic Category II structure.

- The bearing capacity with appropriate factor of safety is greater than or equal to the bearing demand.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Seismic model uncertainties due to, among other things, uncertainties in material properties, mass properties, damping values, the effect of concrete cracking, and the modeling techniques are accounted for in the widening of floor response spectra, as described in Subsection 3.7.2.5. The effect of cracking of the concrete-filled structural modules inside containment due to thermal loads is discussed in Subsection 3.8.3.4.2.

3.7.2.10 Use of Constant Vertical Static Factors

The vertical component of the safe shutdown earthquake is considered to occur simultaneously with the two horizontal components in the seismic analyses. Therefore, constant vertical static factors are not used for the design of seismic Category I structures.

3.7.2.11 Method Used to Account for Torsional Effects

The seismic analysis models of the nuclear island incorporate the mass and stiffness eccentricities of the seismic Category I structures and the torsional degrees of freedom.

For the response spectrum analysis of the nuclear island, the seismic loads are combined by means of the square root of the sum of the squares (SRSS). The equation for SRSS is shown below.

$$\sqrt{(\alpha \cdot A_{NS})^2 + (\alpha \cdot A_{EW})^2 + (A_{VT})^2}$$

where,

A_{NS} maximum element forces due to SSE response analysis in X (NS)

A_{EW} maximum element forces due to SSE response analysis in Y (EW)

A_{VT} maximum element forces due to SSE response analysis in Z (VT)

α factor to account for accidental torsion effect in NS or EW (1.05)

Alternatively, for equivalent static analysis, the 100-40-40 rule is applicable in order to cover both negative and positive member forces. The equation for the 100-40-40 rule is shown below.

$$k_1 \cdot \text{sign}(A_{NS}) \cdot (\alpha \cdot A_{NS}) + k_2 \cdot \text{sign}(A_{EW}) \cdot (\alpha \cdot A_{EW}) + k_3 \cdot A_{VT}$$

where,

k_i combination factors (±1.0, ±0.4, ±0.4)

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sign(X) sign of variable X: X < 0 results -1; X ≥ 0 results +1

α factor to account for accidental torsion effect in NS or EW (1.05)

3.7.2.12 Methods for Seismic Analysis of Dams

Seismic analysis of dams is site specific design.

3.7.2.13 Determination of Seismic Category I Structure Overturning Moments

Subsection 3.8.5.5.4 describes the effects of seismic overturning moments.

3.7.2.14 Analysis Procedure for Damping

Subsection 3.7.1.3 presents the damping values used in the seismic analyses. [*For structures comprised of different material types, the composite modal damping approach utilizing the strain energy method is used to determine the composite modal damping values.*]* Subsection 3.7.2.4 presents the damping values used in the soil-structure interaction analysis.

3.7.3 Seismic Subsystem Analysis

This subsection describes the seismic analysis methodology for subsystems, which are those structures and components that do not have an interface with the soil-structure interaction analyses. Structures and components considered as subsystems include the following:

- Structures, such as floor slabs, walls, miscellaneous steel platforms and framing
- Equipment modules consisting of components, piping, supports, and structural frames
- Equipment including vessels, tanks, heat exchangers, valves, and instrumentation
- Distributive systems including piping and supports, electrical cable trays and supports, HVAC ductwork and supports, instrumentation tubing and supports, and conduits and supports

Subsection 3.9.2 describes dynamic analysis methods for the reactor internals. Subsection 3.9.3 describes dynamic analysis methods for the primary coolant loop support system. Subsection 3.7.2 describes the analysis methods for seismic systems, which are those structures and components that are considered with the foundation and supporting media. Section 3.2 includes the seismic classification of building structures, systems, and components.

3.7.3.1 Seismic Analysis Methods

The methods used for seismic analysis of subsystems include, modal response spectrum analysis, time-history analysis, and equivalent static analysis. The methods described in this subsection are acceptable for any subsystem. The particular method used is selected by the designer based on its appropriateness for the specific item. Items analyzed by each method are identified in the descriptions of each method in the following paragraphs.

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3.7.3.2 Determination of Number of Earthquake Cycles

Seismic Category I structures, systems, and components are evaluated for one occurrence of the safe shutdown earthquake (SSE). In addition, subsystems sensitive to fatigue are evaluated for cyclic motion due to earthquakes smaller than the safe shutdown earthquake. Using analysis methods, these effects are considered by inclusion of seismic events with an amplitude not less than one-third of the safe shutdown earthquake amplitude. The number of cycles is calculated based on IEEE-344-1987 (Reference 16) to provide the equivalent fatigue damage of two full safe shutdown earthquake events with 10 high-stress cycles per event. Typically, there are five seismic events with an amplitude equal to one-third of the safe shutdown earthquake response. Each of the one-third safe shutdown earthquake events has 63 high-stress cycles. [*For ASME Class 1 piping, the fatigue evaluation is performed based on five seismic events with an amplitude equal to one-third of the safe shutdown earthquake response. Each event has 63 high-stress cycles.*]*

When seismic qualification is based on dynamic testing for structures, systems, or components containing mechanisms that must change position in order to function, operability testing is performed for the safe shutdown earthquake preceded by one or more earthquakes. The number of preceding earthquakes is calculated based on IEEE-344-1987 (Reference 16) to provide the equivalent fatigue damage of one safe shutdown earthquake event. Typically, the preceding earthquake is one safe shutdown earthquake event or five one-half safe shutdown earthquake events.

3.7.3.3 Procedure Used for Modeling

The dynamic analysis of any complex system requires the discretization of its mass and elastic properties. This is accomplished by concentrating the mass of the system at distinct characteristic points or nodes, and interconnecting them by a network of elastic springs representing the stiffness properties of the systems. The stiffness properties are computed either by hand calculations for simple systems or by finite element methods for more complex systems.

Nodes are located at mass concentrations and at additional points within the system. They are selected in such a way as to provide an adequate representation of the mass distribution and high-stress concentration points of the system.

At each node, degrees of freedom corresponding to translations along three orthogonal axes, and rotations about these axes are assigned. The number of degrees of freedom is reduced by the number of constraints, where applicable. For equipment qualification, reduced degrees of freedom are acceptable provided that the analysis adequately and conservatively predicts the response of the equipment.

The size of the model is reviewed so that a sufficient number of masses or degrees of freedom are used to compute the response of the system. A model is considered adequate provided that additional degrees of freedom do not result in more than a 10 percent increase in response, or the number of degrees of freedom equals or exceeds twice the number of modes with frequencies less than 33 hertz.

Dynamic models of floor and roof slabs and miscellaneous steel platforms and framing include masses equal to 25 percent of the floor live load or 75 percent on the roof snow load, whichever is applicable.

Dynamic models are prepared for the following seismic Category I steel structures. Response spectrum or time history analyses are performed for structural design.

- Passive containment cooling valve room (room number 12701)

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- Steel framing around steam generators
- Containment air baffle

Seismic input for the subsystem and component design are the enveloped floor response spectra described in Subsection 3.7.2.5 or the response time histories as described in Subsection 3.7.2.1. Where amplified response spectra are required on the subsystem for design of components, such as for use in the decoupled analyses of piping or components described in Subsection 3.7.3.8.3, the amplified response spectra are generated and enveloped as described in Subsection 3.7.2.5.

3.7.3.4 Basis for Selection of Frequencies

The effect of the building amplification on equipment and components is addressed by the floor response spectra method or by a coupled analysis of the building and equipment. Certain components are designed for a natural frequency greater than 33 hertz. In those cases where it is practical to avoid resonance, the fundamental frequencies of components and equipment are selected to be less than one-half or more than twice the dominant frequencies of the support structure.

3.7.3.5 Equivalent Static Load Method of Analysis

[*The equivalent static load method involves equivalent horizontal and vertical static forces applied at the center of gravity of various masses. The equivalent force at a mass location is computed as the product of the mass and the seismic acceleration value applicable to that mass location. Loads, stresses, or deflections, obtained using the equivalent static load method, are adjusted to account for the relative motion between points of support when significant.*]*

3.7.3.5.1 Single Mode Dominant or Rigid Structures or Components

For rigid structures and components, or for cases where the response can be classified as single mode dominant, the following procedures are used. Examples of these systems, structures, and components are equipment, and piping lines, instrumentation tubing, cable trays, HVAC, and floor beams modeled on a span by span basis.

- For rigid systems, structures, and components (fundamental frequency ≥ 33 hertz), an equivalent seismic load is defined for the direction of excitation as the product of the component mass and the zero period acceleration value obtained from the applicable floor response spectra.

- A rigid component (fundamental frequency ≥ 33 hertz), whose support can be represented by a flexible spring, can be modelled as a single degree of freedom model in the direction of excitation (horizontal or vertical directions). The equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value at the natural frequency from the applicable floor response spectra. If the frequency is not determined, the peak acceleration from the applicable floor response spectrum is used.

[- *If the component has a distributed mass whose dynamic response will be single mode dominant, the equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value at the component natural frequency from the applicable floor response spectra times a factor of 1.5. A factor of less than 1.5 may be used if justified. Static factors smaller than 1.5 are not used for piping systems.*]* A factor of 1.0 is used for structures or equipment that can be represented as uniformly loaded cantilever, simply supported, fixed-simply supported, or fixed-fixed beams

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(References 10 and 11) when the fundamental frequency is higher than the peak acceleration frequency associated with the applicable floor response spectrum. If the frequency is not determined, the peak acceleration from the applicable floor response spectrum is used.

3.7.3.5.2 Multiple Mode Dominant Response

This procedure applies to piping, instrumentation tubing, cable trays, and HVAC that are multiple span models. The equivalent static load method of analysis can be used for design of piping systems, instrumentation and supports that have significant responses at several vibrational frequencies. In this case, [*a static load factor of 1.5 is applied to the peak accelerations of the applicable floor response spectra. For runs with axial supports which are rigid in the axial direction (fundamental frequency greater than or equal to 33 hertz), the acceleration value of the mass of piping in its axial direction may be limited to 1.0 times its calculated spectral acceleration value. The spectral acceleration value is based on the frequency of the piping system along the axial direction. The relative motion between support points is also considered.*]*

3.7.3.6 Three Components of Earthquake Motion

[*Two horizontal components and one vertical component of seismic response spectra are employed as input to a modal response spectrum analysis.*]* The spectra are associated with the safe shutdown earthquake. In the response spectrum and equivalent static analyses, the effects of the three components of earthquake motion are combined using one of the following methods:

[● *The peak responses due to the three earthquake components from the response spectrum analyses are combined using the square root of the sum of squares (SRSS) method.*

● *The peak responses due to the three earthquake components are combined directly, using the assumption that when the peak response from one component occurs, the responses from the other two components are 40 percent of the peak (100 percent-40 percent-40 percent method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus), are considered. This method is not used for piping systems.*

One set of three mutually orthogonal artificial time histories is used when time-history analyses are performed. The components of earthquake motion specified in the three directions are statistically independent and applied simultaneously. When this method is used, the responses from each of the three components of motion are combined algebraically at each time step.]*

In addition, an optional method for combining the response of the three components of earthquake motion is presented in the following paragraphs.

[*The time-history safe shutdown earthquake analysis of a subsystem can be performed by simultaneously applying the displacements and rotations at the interface point(s) between the subsystem and the system. These displacements and rotations are the results obtained from a model of a larger subsystem or a system that includes a simplified representation of the subsystem. The time-history safe shutdown earthquake analysis of the system is performed by applying three mutually orthogonal and statistically independent, artificial time histories.*]* Possible examples of the use of this method of seismic analysis include the following:

● The subsystem analysis is a flexible floor or miscellaneous structural steel frame. The corresponding system analysis is the soil-structure interaction analysis of the nuclear island structures.

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- The subsystem analysis is the primary loop piping system and interior concrete building structure. The interface point is the top of the basemat. The corresponding system analysis is the soil-structure interaction analysis of the nuclear island structures.

- The subsystem analysis is the reactor coolant pump and internal components. The interface points are the welds on the pump suction and discharge nozzles. The corresponding system analysis is the primary loop piping system and interior concrete building structure.

3.7.3.7 Combination of Modal Responses

[*For the seismic response spectra analyses, the zero period acceleration cut-off frequency is 33 hertz. High frequency or rigid modes are considered using the left-out-force method or the missing mass method*]* described in Subsection 3.7.3.7.1. The method to combine the low frequency modes is described in Subsection 3.7.3.7.2. [*The rigid mode results in the three perpendicular directions of the seismic input are combined by the SRSS method. The resultant response of the rigid modes is combined by SRSS with the flexible mode results.*]* The combination of modal responses in time history analyses of piping systems is described in Subsection 3.7.3.17. Modal responses in time history analyses of other subsystems are combined as described in Subsection 3.7.2.6.

3.7.3.7.1 Combination of High-Frequency Modes

This subsection describes alternative methods of accounting for high-frequency modes (generally greater than 33 hertz) in seismic response spectrum analysis. Higher-frequency modes can be excluded from the response calculation if the change in response is less than or equal to 10 percent.

3.7.3.7.1.1 Left-Out-Force Method or Missing Mass Correction for High Frequency Modes

The left-out-force method is based on the Left-Out-Force Theorem. This theorem states that for every time history load there is a frequency, f_r, called the "rigid mode cutoff frequency" above which the response in modes with natural frequencies above f_r will very closely resemble the applied load at each instant of time. These modes are called "rigid modes." [*The left-out-force method is used in program PIPESTRESS.*]*

The left-out-force vector, $\{Fr\}$, is calculated based on lower modes:

$$\{Fr\} = \left[1 - \Sigma M e_j e_j^T\right] f(t)$$

where:

f (t) = the applied load vector

M = the mass matrix

e_j = the eigenvector

Note that Σ is only for all the flexible modes, not including the rigid modes.

In the response spectra analysis, the total inertia force contribution of higher modes can be interpreted as:

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$$\{Fr\} = Am[M]\left[\{r\} - \Sigma P_j e_j\right]$$

where:

Am = the maximum spectral acceleration beyond the flexible modes

[M] = the mass matrix

$\{r\}$ = the influence vector or displacement vector due to unit displacement

P_j = participation factor

Since,

$$P_j = e_j{}^T[M]\{r\}, \{Fr\} = Am[M]\{r\}\left[1 - \Sigma M e_j e_j{}^T\right]$$

[*In PIPESTRESS, the low frequency modes are combined by one of the Regulatory Guide 1.92 methods in the response spectrum analysis.*]* For each support level, there is a pseudo-load vector or left-out-force vector in the X, Y and Z directions. These left-out-force vectors are used to generate left-out-force solutions which are multiplied by a scalar amplitude equal to a magnification factor specified by the user. This factor is usually the ZPA (zero period acceleration) of the response spectrum for the corresponding direction. The resultant low frequency responses are combined by square root of the sum of the squares with the high frequency responses (rigid modes results).

[*In GAPPIPE, the results from the high frequency responses are also combined by the square root of the sum of the squares with those from the resultant loads contributed by lower modes.*]* The missing mass correction for an independent support motion or multiple response spectra analysis is exactly the same as that for the single enveloped response spectrum analysis except that Am used is the envelope of all the zero period accelerations of all the independent support inputs.

3.7.3.7.1.2 SRP 3.7.2 Method

[*The method described in SRP Section 3.7.2 may also be used for combination of high-frequency modes.*]*

The following is the procedure for incorporating responses associated with high-frequency modes.

Step 1 Determine the modal responses only for those modes having natural frequencies less than that at which the spectral acceleration approximately returns to the zero period acceleration (33 hertz for the Regulatory Guide 1.60 response spectra). Combine such modes according to the methods discussed in Subsection 3.7.3.7.2.

Step 2 For each degree of freedom included in the dynamic analysis, determine the fraction of degree of freedom mass included in the summation of all modes included in Step 1. This fraction d_i for each degree of freedom is given by:

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$$d_i = \sum_{n=1}^{N} C_n \, x \, \phi_{n,i}$$

where:

n = order of mode under consideration

N = number of modes included in Step 1

$\phi_{n,i}$ = nth natural mode of the system

C_n is the participation factor given by:

$$C_n = \frac{(\phi_n)^T [m] (1)}{(\phi_n)^T [m] (\phi_n)}$$

Next, determine the fraction of degree of freedom mass not included in the summation of these modes:

$$e_i = d_i - \delta_{ij}$$

where δ_{ij} is the Kronecker delta, which is 1 if degree of freedom i is in the direction of the earthquake motion and 0 if degree of freedom i is a rotation or not in the direction of the earthquake input motion.

If, for any degree of freedom i, the absolute value of this fraction e_i exceeds 0.1, the response from higher modes is included with those included in Step 1.

Step 3 Higher modes can be assumed to respond in phase with the zero period acceleration and, thus, with each other. Hence, these modes are combined algebraically, which is equivalent to pseudostatic response to the inertial forces from these higher modes excited at the zero period acceleration. The pseudostatic inertial forces associated with the summation of all higher modes for each degree of freedom i are given by:

$$P_i = ZPA \, x \, M_i \, x \, e_i$$

where:

P_i = force or moment to be applied by degree of freedom i

M_i = mass or mass moment of inertia associated with degree of freedom i.

The subsystem is then statically analyzed for this set of pseudo static inertial forces applied to all degrees of freedom to determine the maximum responses associated with high-frequency modes not included in Step 1.

Step 4 The total combined response to high-frequency modes (Step 3) is combined by the square root of sum of the squares method with the total combined response from lower-frequency modes (Step 1) to determine the overall structural peak responses.

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3.7.3.7.2 Combination of Low-Frequency Modes

This subsection describes the method for combining modal responses in the seismic response spectra analysis. [*The total unidirectional seismic response for subsystems is obtained by combining the individual modal responses using the square root sum of the squares method. For subsystems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. For piping systems, the methods in Regulatory Guide 1.92 are used for modal combinations.*]* For other subsystems, the methods in Regulatory Guide 1.92 or the following alternative methods may be used. [*The groups of closely spaced modes are chosen so that the differences between the frequencies of the first mode and the last mode in the group do not exceed 10 percent of the lower frequency.*

Combined total response for systems having such closely spaced modal frequencies is obtained by adding to the square root sum of squares of all modes the product of the responses of the modes in each group of closely spaced modes and coupling factor.]* This can be represented mathematically as:

$$R_T{}^2 = \sum_{i=1}^{N} R_i{}^2 + 2 \sum_{j=1}^{S} \sum_{k=Mj}^{Nj-1} \sum_{\ell=k+1}^{Nj} R_k \, R_\ell \, \varepsilon_{k\ell}$$

where:

R_T = total unidirectional response

R_i = absolute value of response of mode i

N = total number of modes considered

S = number of groups of closely spaced modes

= lowest modal number associated with group j of closely spaced modes

N_j = highest modal number associated with group j of closely spaced modes

$\varepsilon_{k\ell}$ = coupling factor, defined as follows:

$$\varepsilon_{k\ell} = \left(1 + \frac{\left(w_k^{,} - w_\ell^{,} \right)^2}{\left(\beta_k^{,} w_k + \beta_\ell^{,} w_\ell \right)^2} \right)^{-1}$$

and,

$$w_k^{,} = w_k \left[1 - \left(\beta_k \right)^2 \right]^{1/2}$$

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$$\beta_k' = \beta_k + \frac{2}{w_k t_d}$$

where:

w_k = frequency of closely spaced mode k

β_k = fraction of critical damping in closely spaced mode k

t_d = duration of the earthquake (= 30 seconds)

[*Alternatively, a more conservative grouping method can be used in the seismic response spectra analyses. The groups of closely spaced modes are chosen so that the difference between two frequencies is no greater than 10 percent.*]* Therefore,

$$R_T{}^2 = \sum_{i=1}^{N} R_i{}^2 + 2\Sigma\, \varepsilon_{k\ell}\, R_k\, R_\ell$$

where:

$$\frac{|\, w_k - w_\ell \,|}{w_\ell} \leq 0.1$$

All other terms for the modal combination remain the same. The 10 percent grouping method is more conservative than the grouping method because the same mode can appear in more than one group.

In addition to the above methods, any of the other methods in Regulatory Guide 1.92 may be used for modal combination.

[*3.7.3.8 Analytical Procedure for Piping*

This subsection describes the modeling methods and analytical procedures for piping systems.

The piping system is modeled as beam elements with lump masses connected by a network of elastic springs representing the stiffness properties of the piping system. Concentrated weights such as valves or flanges are also modeled as lump masses. The effects of torsion (including eccentric masses), bending, shear, and axial deformations, and effects due to the changes in stiffness values of curved members are accounted for in the piping dynamic model.

The lump masses are selected so that the maximum spacing is not greater than the length that would produce a natural frequency equal to the zero period acceleration (ZPA) frequency of the seismic input when calculated based on a simply supported beam. As a minimum, the number of degrees of freedom is equal to twice the number of modes with frequencies less than the zero period acceleration frequency.

The piping system analysis model includes the effect of piping support mass when the contributory mass of the support is greater than 10 percent of the total mass of the effected piping spans. The contributory mass of the support is the portion of the support mass that is attached to the piping; such as clamps, bolts, trunnions, struts, and snubbers. Supports that are not directly attached to the

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piping, such as box frames, need not be considered for mass effects. The mass of the applicable support will not affect the response of the system in the supported direction, herefore only the unsupported direction needs to be considered. Based on this reasoning, the mass of full anchors can be neglected. The total mass of each effected piping span includes the mass of the piping, fluid contents, insulation, and any concentrated masses (for example, valves or flanges) between the adjacent supports in each unrestrained direction on both sides of the applicable support. For example; the contributory mass of an X direction support must be compared to the mass of the piping spans in the unrestrained Y and Z directions. A contributory support mass that is less than 10 percent of the masses of the effected spans will have insignificant effect on the response of the piping system and can be neglected.

The stiffness matrix of the piping system is calculated based on the stiffness values of the pipe elements and support elements. Default rigid or calculated support stiffness values are used (see subsections 3.9.3.1.5 and 3.9.3.4). When the support deflections are limited to 1/8 inches for the dynamic combined faulted loads, default rigid support stiffness values are used. If the dynamic combined faulted load deflection for any support exceeds 1/8 inches, calculated support stiffness values are used for the affected support.

Valves, equipment and piping modules are considered as rigid if the natural frequencies are greater than 33 hertz. Valves with lower frequencies are included in the piping system model. See subsection 3.7.3.8.2.1 for flexible equipment and subsection 3.7.3.8.3 for flexible modules.

See subsection 3.9.3.1.4 for the primary loop piping and support system.]*

3.7.3.8.1 Supporting Systems

This subsection deals with the analysis of piping systems that provide support to other piping systems. The methods used for the analysis of the primary loop piping are described in Appendix 3C. [*The supported piping system may be excluded from the analysis of the supporting piping system when the ratio of the supported pipe to supporting pipe moment of inertia is less than or equal to 0.04.*

If the ratio of the run piping outside diameter to the branch piping outside diameter (nominal pipe size) exceeds or equals 3.0, the branch piping can be excluded from the analysis of the run piping. The mass and stiffness effects of the branch piping are considered as described below.

Stiffness Effect

The stiffness effect of the decoupled branch pipe is considered significant when the distance from the run pipe outside diameter to the first rigid or seismic support on the decoupled branch pipe is less than or equal to one half the deadweight span of the branch pipe (given in ASME III Code Subsection NF).

Mass Effect

Considering one direction at a time, the mass effect is significant when the weight of half the span (from the decoupling point) of the branch pipe in one direction is more than 20 percent the weight of the main run pipe span in the same direction. Concentrated weights in the branch pipe are considered. A branch pipe span in x direction is the span between the decoupled branch point and the first seismic or rigid support in the x direction. A main run pipe span in the x direction is the piping bounded by the first seismic or rigid support in the x direction on both sides of the decoupled branch point. Similarly, the same definition applies to the spans in other directions (y and z).

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If the calculated branch pipe weight is less than 20 percent but more than 10 percent of the main run pipe weight, this weight is lumped at the decoupling point of the run pipe for the run pipe analysis. This weight can be neglected if it is less than 10 percent of the main run pipe weight.

Required Coupled Branch Piping

If the stiffness and/or mass effects are considered significant, the branch piping is included in the piping analysis for the run pipe analysis. The portion of branch piping considered in the analysis adequately represents the behavior of the run pipe and branch pipe. The branch line model ends in one of the following ways:

- *First six-way anchor*
- *Four rigid/seismic supports in each of the three perpendicular directions*
- *Rigidly supported zone as described* in *subsection 3.7.3.13.4.2*]*

3.7.3.8.2 Supported Systems

This subsection deals with the analysis of piping systems that are supported by other piping systems or by equipment.

3.7.3.8.2.1 Large Diameter Auxiliary Piping

[*This subsection deals with ASME Class 1 piping larger than 1-inch nominal pipe size and ASME Class 2 and 3 piping with nominal pipe size larger than 2 inches. The response spectra methodology is used.*

When the supporting system is a piping system, the supported pipe (branch) can be decoupled from the supporting pipe (run) when the ratio of the run piping nominal pipe size to branch pipe nominal pipe size is greater than or equal to three to one. Decoupling can also be done when the moment of inertia of the branch pipe is less than or equal to 4 percent of the moment of inertia of the run pipe.

During the analysis of the branch piping, resulting values of tee anchor reactions are checked against the capabilities of the tee.

The seismic inertia effects of equipment and piping that provide support to supported (branch) piping systems are considered when significant. When the frequency of the supporting equipment is less than 33 hertz, then either a coupled dynamic model of the piping and equipment is used, or the amplified response spectra at the equipment connection point is used with a decoupled model of the supported piping. When supported piping is supported by larger piping, one of the following methods is used:

- *A coupled dynamic model of the supported piping and the supporting piping*
- *Amplified response spectra at the connection point to the supporting piping with a decoupled model of the supported piping*]*

3.7.3.8.2.2 Small-Diameter Auxiliary Piping

[*This subsection deals with ASME Code Class 1 piping equal to or less than 1-inch nominal pipe size and ASME Class 2 and 3 piping with nominal pipe sizes less than or equal to 2 inches. This includes instrumentation tubing. These piping systems may be supported by equipment or primary loop piping or other auxiliary piping or both. The response spectra or equivalent static load methodology is used. One of the following methods may be used for these systems:*

- *Same method as described in subsection 3.7.3.8.2.1*

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- *Equivalent static analysis based on appropriate load factors applied to the response spectra acceleration values*]*

Subsections 3.9.3 and 3.9.8.2 discuss the final design and as built reconciliation of small bore piping.

3.7.3.8.3 Piping Systems on Modules

Many portions of the systems for the AP1000 are assembled as modules offsite and shipped to the plant as completed units. This method of construction does not result in any unique requirements for the analysis of these structures, systems, or components. Existing industry standards and regulatory requirements and guidelines are appropriate for the evaluation of structures, systems, and components included in modules.

The modules are constructed using a structural steel framework to support the equipment, pipe, and pipe supports in the module. The structural steel framework is designed as part of the building structure according to the criteria given in Subsection 3.8.4.

One exception is the pressurizer and safety relief valve module, which is attached to the top of the pressurizer. For this module the structures and piping arrangements support valves off the pressurizer and not the building structure. The structural steel frame is designed as a component support according to ASME Code, Section III, Subsection NF. [*Piping in modules is routed and analyzed in the same manner as in a plant not employing modules. Piping is analyzed from anchor point to anchor point, which are not necessarily at the boundaries of the module.*]* This is consistent with the manner in which room walls are treated in a nonmodule plant.

[*The supported piping or component may be decoupled from the seismic analysis of the structural frame based on the following criteria. The mass ratio, Rm, and the frequency ratio, Rf, are defined as follows:*

- *Rm = mass of supported component or piping/mass of supporting structural frame*
- *Rf = frequency of the component or piping/frequency of the structural frame*

Decoupling may be done when:

- *Rm < 0.01, for any Rf, or*
- *Rm ≥ 0.01 and ≤ 0.10, if Rf ≤ 0.8 or if Rf is ≥ 1.25.*

In addition, supported piping may be decoupled if analysis shows that the effect on the structural frame is small, that is, when the change in response is less than 10 percent. When piping or components are decoupled from the analysis of the frame, the contributory mass of the piping and components is included as a rigid mass in the model of the structural frame.]*

When piping or components are decoupled from the analysis of the frame using the preceding criteria, the effect of the frame is accounted for in the analysis of the decoupled components or piping. Either an amplified response spectra or a coupled model is used. The amplified response spectra are obtained from the time history safe shutdown earthquake analysis of the frame. The coupled model consists of a simplified mass and stiffness model of the frame connected to the seismic model of the components or piping.

Alternative criteria may be applied to simple frames that behave as pipe support miscellaneous steel. Decoupling may be done when the deflection of the frame due to dynamic combined faulted loading is less than or equal to 1/8 inch. These deflections are defined with respect to the structure to which the structural frame is attached. The stiffness of the intervening elements between the frame and the supported piping or component is considered as follows: Default rigid stiffness values are used for

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supports except that vendor stiffness values are used for snubbers and rigid gapped supports. The mass of the structural frame is evaluated as a self-weight excitation loading on the frame and the structures supporting the frame. The same approach is used for pipe support miscellaneous steel, as described in Subsection 3.9.3.4.

When the supported components or piping cannot be decoupled, they are included in the analysis model of the structural frame. The interaction between the piping and the frame is incorporated by including the appropriate stiffness and mass properties of the components, piping, and frame in the coupled model.

[***3.7.3.8.4 Piping Systems with Gapped Supports***

This subsection describes the analysis methods for piping systems with rigid gapped supports. These supports may be used to minimize the number of pipe support snubbers and the corresponding inservice testing and maintenance activities.

The analysis consists of an iterative response spectra analysis of the piping and support system. Iterations are performed to establish calculated piping displacements that are compatible with the stiffness and gap of the rigid gapped supports. The results of the computer program GAPPIPE, which uses this methodology, are supported with test data (Reference 13).

The method implemented in GAPPIPE to analyze piping systems supported by rigid gapped supports is based on the equivalent linearization technique. GAPPIPE analysis is performed whenever snubber supports are replaced by rigid gapped supports.

The basis of the concept is to find an equivalent linear spring with a response-dependent stiffness for each nonlinear rigid gapped support, or limit stop, in the mathematical model of the piping system. The equivalent linearized stiffness minimizes the mean difference in force in the support between the equivalent spring and the corresponding original gapped support. The mean difference is estimated by an averaging process in the time domain, that is, across the response duration, using the concept of random vibration. Details of the design and analysis methods and modeling assumptions are described in Reference 12.]*

3.7.3.9 Combination of Support Responses

This subsection describes alternative methods for combining the responses from the individual support or attachment points that connect the supported system or subsystem to the supporting system or subsystem. There are two aspects to the responses from the support or attachment points: seismic anchor motions and envelope or multiple-input response spectra methodology.

Seismic Anchor Motions – The response due to differential seismic anchor motions is calculated using static analysis (without including a dynamic load factor). In this analysis, the static model is identical to the static portion of the dynamic model used to compute the seismic response due to inertial loading. In particular, the structural system supports in the static model are identical to those in the dynamic model.

[*The effect of relative seismic anchor displacements is obtained either by using the worst combination of the peak displacements or by proper representation of the relative phasing characteristics associated with different support inputs. For components supported by a single concrete building (coupled shield and auxiliary buildings, or containment internal structures), the seismic motions at all elevations above the basemat are taken to be in phase. When the component supports are in the same structure, the relative seismic anchor motions are small and the effects are neglected. This is applicable to building structures and to those supplemental steel frames that are rigid in comparison to the components. Supplemental steel frames that are flexible can have*

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significant seismic anchor motions which are considered. When the components supports are in different structures, the relative seismic anchor motion between the structures is taken to be out-of-phase and the effects are considered. The results of the modal spectra analysis (multiple input or envelope) are combined with the results from seismic anchor motion by the absolute sum method or the SRSS method, as described in Tables 3.9-5 and 3.9-6.]*

Response Spectra Methods – The envelope broadened uniform-input response spectra can lead to excessive conservatism and unnecessary pipe supports. The peak shifting method and independent support motion spectra method are used to avoid unnecessary conservatism.

Seismic Response Spectra Peak Shifting

The peak shifting method may be used in place of the broadened spectra method, as described below.

Determine the natural frequencies $(f_e)_n$ of the system to be qualified in the broadened range of the maximum spectrum acceleration peak.

If no equipment or piping system natural frequencies exist in the ±15 percent interval associated with the maximum spectrum acceleration peak, then the interval associated with the next highest spectrum acceleration peak is selected and used in the following procedure.

Consider all N natural frequencies in the interval

$f_j - 0.15f_j \le (f_e)_n \le f_j + 0.15f_j$

where:

f_j = the frequency of maximum acceleration in the envelope spectra

n = 1 to N

The system is then evaluated by performing N + 3 separate analyses using the envelope unbroadened floor design response spectrum and the envelope unbroadened spectrum modified by shifting the frequencies associated with each of the spectral values by a factor of +0.15; -0.15; and

$$\frac{(f_e)_n - f_j}{f_j}$$

where:

n = 1 to N

The results of these separate seismic analyses are then enveloped to obtain the final result desired (e.g., stress, support loads, acceleration, etc.) at any given point in the system. If three different floor response spectrum curves are used to define the response in the two horizontal and the vertical directions, then the shifting of the spectral values as defined above may be applied independently to these three response spectrum curves.

Independent Support Response Spectrum Methods

The use of multiple-input response spectra accounts for the phasing and interdependence characteristics of the various support points. The following alternative methods are used for the

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AP1000 plant. These are based on the guidelines provided by the "Pressure Vessel Research Committee Technical Committee on Piping Systems" (Reference 14).

[*Envelope Uniform Response Spectra - Method A - The seismic response spectrum that envelopes the supports is used in place of the spectra at each support in the envelope uniform response spectra. Also, the contribution from the seismic anchor motion of the support points is assumed to be in phase and is added algebraically as follows:*

$$q_i = d_i \sum_{j=1}^{N} P_{ij}$$

where:

q_i = *combined displacement response in the normal coordinate for mode i*

d_i = *maximum value of* d_{ij}

d_{ij} = *displacement spectral value for mode i associated with support "j"*

P_{ij} = *participation factor for mode i associated with support j*

N = *number of support points*

Enveloped response spectra are developed as the seismic input in three perpendicular directions of the piping coordinate system to include the spectra at the floor elevations of the attachment points and the piping module or equipment if applicable. The mode shapes and frequencies below the cut-off frequency are calculated in the response spectrum analysis. The modal participation factors in each direction of the earthquake motion and the spectral accelerations for each significant mode are calculated. Based on the calculated mode shapes, participation factors, and spectral accelerations of individual modes, the modal inertia response forces, moments, displacements, and accelerations are calculated. For a given direction, these modal inertia responses are combined based on consideration of closely spaced modes and high frequency modes to obtain the resultant forces, moments, displacements, accelerations, and support loads. The total seismic responses are combined by square-root-sum-of-the-squares method for all three earthquake directions.

Independent Support Motion - Method B - When there are more than one supporting structure, the independent support motion (ISM) method for seismic response spectra may be used.

Each support group is considered to be in a random-phase relationship to the other support groups. The responses caused by each support group are combined by the square-root-sum-of-the-square method. The displacement response in the modal coordinate becomes:

$$q_i = \left[\sum_{j=1}^{N} \left(P_{ij}\, d_{ij} \right)^2 \right]^{1/2}$$

A support group is defined by supports that have the same time-history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.]*

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3.7.3.10 Vertical Static Factors

Constant static factors can be used in some cases for the design of seismic Category I subsystems and equipment. The criteria for using this method are presented in Subsection 3.7.3.5.

3.7.3.11 Torsional Effects of Eccentric Masses

[*The methods used to account for the torsional effects of valves and other eccentric masses (for example, valve operators) in the seismic subsystem analyses are as follows:*

- *When valves and other eccentric masses are considered rigid, the mass of the operator and valve body or other eccentric mass are located at their respective center of gravity. The eccentric components (that is, yoke, valve body) are modeled as rigid members.*

- *When valves and other eccentric masses are not considered rigid, the dynamic models are simulated by the lumped masses in discrete locations (that is, center of gravity of valve body and valve operator), coupled by elastic members with properties of the eccentric components.*]*

3.7.3.12 Seismic Category I Buried Piping Systems and Tunnels

[*There are no seismic Category I buried piping systems and tunnels in the AP1000 design.*]*

3.7.3.13 Interaction of Other Systems with Seismic Category I Systems

The safety functions of seismic Category I structures, systems, and components are protected from interaction with nonseismic structures, systems, and components; or their interaction is evaluated. The safety-related systems and components required for safe shutdown are described in Section 7.4. This equipment is located in selected areas of the auxiliary building and inside containment. The primary means of protecting safety-related structures, systems, and components from adverse seismic interactions are discussed in the following paragraphs in the order of preference.

- Separation – separation with the use of physical barriers

- Segregation – routing away from location of seismic Category I systems, structures, and components

- Impact Evaluation – contact with seismic Category I systems, structures, and components may occur, and there is insufficient energy in the impact to cause loss of safety function

- Support as seismic Category II

[*Interaction of connected systems with seismic Category I piping is considered by including the other piping in the analysis of the seismic Category I system.*]* Interaction of piping systems that are adjacent to Category I structures, systems, and components is also considered. This is discussed in Subsection 3.7.3.13.4.

The containment and each room outside containment containing safety-related systems or equipment, as identified in Table 3.7.3-1, are reviewed for potential adverse seismic interactions to demonstrate that systems, structures, and components are not prevented from performing their required safe shutdown functions. In addition, the review identifies the protection features required to mitigate the consequences of seismic interaction in an area that contains safety-related equipment.

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The evaluation steps to address seismic interaction taken for each room or building area containing seismic Category I systems, structures, and components are:

1. Define targets susceptible to damage (sensitive targets);
   Sensitive targets are those seismic Category I components for which adverse spatial interaction can result in loss of safety function.

2. Define sources which can potentially interact in an adverse manner with the target.

3. If possible, assure adequate free space to eliminate the possibility of seismically-induced damaging impacts for the sensitive targets.

4. Assess impact effects (interaction) when adequate free space is not present.

5. Correct adverse seismic interaction conditions.

The three-dimensional computer model and composites developed for the nuclear island are used during the design process of the systems and components in the nuclear island, to aid in evaluating and documenting the review for seismic interactions. This review is performed using the design criteria and guidelines described in Subsections 3.7.3.13.1 through 3.7.3.13.4.

The seismic interaction review is discussed in Subsection 3.7.5.3. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition.

3.7.3.13.1 Separation and Segregation

Separation – The general plant arrangement provides physical separation between the seismic Category I and nonseismic structures, systems, and components to the maximum extent practicable in the nuclear island. The objective is to assist in the preclusion of a potential adverse interaction if the nonseismic structures, systems and components were to fail during a seismic event. Whenever possible, nonseismic pipe, electrical raceway, or ductwork is not routed above or adjacent to safety-related equipment, pipe, electrical raceway systems, or ductwork, thereby eliminating the possibility of seismic interaction.

Workstations and other equipment in the Main Control Room are separated from piping. Further, as stated in Subsection 3.2.1.1.2, structures, systems, and components that are located overhead in the Main Control Room are supported as seismic Category II.

Segregation – Where separation by physical means cannot be accomplished and it becomes necessary to locate or route nonseismic structures, systems, and components in or through safety-related areas, the nonseismic structures, systems and components are segregated from the seismic Category I items to the extent practicable.

Nonseismic cabinets are separated or segregated from seismic Category I cabinets. Also, if a cabinet is a source or a target, the cabinet doors must be secured by latches or fasteners to assure they do not open during a seismic event.

3.7.3.13.2 Impact Analysis

Adverse spatial interaction (i.e., loss of structural integrity or function effecting safety) can potentially occur when two items are in close proximity. Adverse spatial interaction can result from contact or impact from overturning. Seismic Category I systems, structures, and components that are sensitive to seismic interaction are identified as potential targets. Sources are structures or components that

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can have adverse spatial interaction with the seismic Category I systems, structures, and components. Identification and evaluation of spatial interactions includes the following considerations:

- Proximity of the source to the target. That is, the location of the source within the impact evaluation zone (shown in Figure 3.7.3-1)

  If a source is outside the impact evaluation zone, and does not enter this zone if overturning occurs, no adverse spatial interaction can occur with the identified target. If the source is within the impact evaluation zone and the supports of the source fail, the source could free fall, potentially impacting the target.

- Robustness of target

  If a target has significant structural integrity, and its function is not an issue, adverse spatial interaction could not occur with the identified source.

- Energy of impact

  The energy of the source impacting the target may be so low as not to cause adverse spatial interaction with the target.

A specific nonseismic structure, system, or component identified as a source to a specific safety-related component can be acceptable without being supported as seismic Category II, if an analysis demonstrates that the weight and configuration of the source, relative to the target, and the trajectory of the source are such that the interaction would not cause unacceptable damage to the target. For example, a nonseismic instrument tube routed above a seismic Category I electrical cable tray system would not pose a hazard and would be acceptable.

Nonseismic equipment can overturn as a result of a safe shutdown earthquake. The trajectory of its fall is evaluated to determine if it poses a potential impact hazard to a safety-related structure, system, or component. If it poses a hazard, the equipment is relocated, or it is supported as described in Subsection 3.7.3.13.3.

Nonseismic walls, platforms, stairs, ladders, grating, handrail installations, or other structures next to safety-related structures, systems, and components are evaluated to determine if their failure is credible.

Should a nonseismic structure, system, or component be capable of being dislodged from its supports, the trajectory of its fall is evaluated for potential adverse impacts. If these present a hazard, the structure, system or component is relocated or supported as described in Subsections 3.7.3.13.3 and 3.7.3.13.4. Impact is assumed for sources within an impact evaluation zone around the safety-related equipment. The impact evaluation zone is defined as the envelope around the target for which a source, if located outside of the envelope, would not impact the target during a safe shutdown earthquake in the event the supports of the source were to fail and allow the source to fall. The impact evaluation zone is defined by the volume extending 6 feet horizontally from the perimeter of the seismic Category I object up to a height of 35 feet. The impact evaluation zone above 35 feet is defined by a 10-degree cone radiating vertically from the foot of the object, projected from its perimeter. This definition of the impact evaluation zone is illustrated in Figure 3.7.3-1. The impact evaluation zone need not extend beyond seismic Category I structures such as walls or floor slabs.

Using seismic experience data, the following seismic Category I equipment (potential targets) are not sensitive to piping, HVAC ducts, and cable tray interaction because they are robust to these types of impact:

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- Tanks, "heavy" equipment (for example, heat exchangers)
- Mechanical or electrical penetrations
- Heating, ventilation, and air conditioning (HVAC)
- Adjacent piping
- Raceway Systems
- Structures

3.7.3.13.3 Seismic Category II Supports

Where the preceding approaches of separation, segregation, or impact analysis cannot prevent unacceptable interaction, the source is classified and supported as seismic Category II. The seismic Category II designation provides confidence that these nonseismic structures, systems, and components can withstand the forces of a safe shutdown earthquake in addition to the loading imparted on the seismic Category II supports due to failure of the remaining nonseismically supported portions. This includes nozzle loads from the nonseismic piping. Design methods and stress criteria for systems, structures, and components classified as seismic Category II are the same as for seismic Category I systems, structures, and components, except for piping which is described in Subsection 3.7.3.13.4.2. However, the functionality of these seismic Category II sources does not have to be maintained following a safe shutdown earthquake.

HVAC duct and/or cable trays within the impact evaluation zone are seismically supported using the criteria given in Appendices 3F and 3A for seismic Category I assuring that the HVAC and cable tray segments identified as a source will not fall or adversely impact the sensitive target. Adequate free space between the source and target is assured using the load combination that includes the safe shutdown earthquake. The seismic displacement of the HVAC duct and/or cable tray is 6 inches or the calculated displacement.

Nonseismic equipment identified as a source within the impact evaluation zone is supported as seismic Category II. Support seismic loads include seismic inertia loads of the equipment determined as described in Subsection 3.7.3.5 and nozzle loads from attached piping determined as described in Subsection 3.7.3.13.4.2. Adequate free space is assessed considering a 6-inch deflection envelope for equipment identified as a source, or calculated deflections obtained using the safe shutdown earthquake load combination and elastic analysis.

[*3.7.3.13.4 Interaction of Piping with Seismic Category I Piping Systems, Structures, and Components*

This subsection describes the design methods for piping to prevent adverse spatial interactions.

3.7.3.13.4.1 Seismic Category I Piping

The safe shutdown earthquake piping displacements obtained for the seismic Category I piping are used for the evaluation of seismic interaction with sensitive equipment. Adequate free space between a source and a target is checked adding absolutely the piping safe shutdown earthquake deflection and the safe shutdown earthquake target deflection along with the other loads (e.g., dead weight, thermal) that are in the appropriate design criteria load combinations. Sensitive equipment for piping as the source is seismic Category I equipment shown in Table 3.7.3-2 along with the portion that must be protected ("zone of protection"). Supports may be added to limit seismic movement to eliminate potential adverse interaction.

3.7.3.13.4.2 Seismic Category II Piping

This subsection describes the methods and criteria for piping that is connected to seismic Category I piping. Interaction of seismic Category I piping and nonseismic Category I piping connected to it is

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achieved by incorporating into the analysis of the seismic Category I system a length of pipe that represents the actual dynamic behavior of the complete run of the nonseismic Category I system. The length considered is classified as seismic Category II and extends to the interface anchor or rigid support as described below.

The seismic Category II portion of the line, up to the interface anchor or interface rigid support (last seismic support), is analyzed according to Equation 9 of ASME Code, Section III, Class 3, with a stress limit equal to the smaller of 4.5 S_h and 3.0 S_y. In either case, the nonseismic piping is isolated from the seismic Category I piping by anchors or seismic supports. The anchor or seismic Category II supports are designed for loads from the nonseismic piping. This includes three plastic moment components (M_{p1}, M_{p2}, or M_{p3}) in each of three local coordinate directions. The responses to the three moments are evaluated independently. The seismic Category II portion of the line is analyzed by the response spectrum or equivalent static load method for safe shutdown earthquake.

Single Interface Anchor

The seismic Category II piping may be terminated at a single interface anchor (six-way). This anchor and the supports on the seismic Category II piping are evaluated for safe shutdown earthquake loadings using the rules of ASME III Subsection NF. If the anchor is an equipment nozzle, then the equipment load path through the equipment supports are evaluated to the same acceptance criteria as seismic Category I equipment.

Anchor Followed by a Series of Seismic Supports

The seismic Category II piping may be terminated at the last seismic support which follows a six-way anchor on the seismic Category II piping. This last seismic support and the supports on the seismic Category II piping are evaluated for safe shutdown earthquake loadings using the rules of ASME III Subsection NF. From the anchor to the last seismic support, the response to the plastic moments (M_{p1}, M_{p2}, or M_{p3}) is combined with the responses to seismic anchor motions and equivalent static seismic inertia of the piping system by the absolute sum method. The responses to these moments are evaluated independently. The support and anchor loads due to the plastic moments (M_{p1}, M_{p2}, or M_{p3}) of the seismically analyzed and supported section can be reduced if the elbow/bend resultant moments have exceeded the plastic limit moments of the elbow/bend. The value of the reduction factor RF is as follows:

RF = Multiplier used to reduce the interface anchor and support loads

RF = < l, (if RF > l, no reduction is applicable)

RF = M_L/M_a

M_a = Resultant moment at elbow/bend. Use maximum value if several elbows/bends are within seismically supported region.

M_L = $0.8h^{0.6} D^2t$ Sy for h < 1.45

M_L = D^2t Sy for h > 1.45

h = Flexibility characteristic of elbow/bend

D = Outside diameter of elbow/bend

t = Thickness of elbow/bend

R = Bend radius of elbow/bend

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Rigid Region

The seismic Category II piping may be terminated at the last seismic support of a rigidly supported region of the piping system. The rigid region is typically defined as either four bi-lateral supports around an elbow or six bilateral supports around a tee. The structural behavior of the rigid region is similar to that of a six-way anchor. The frequency of the piping system in the rigid region is greater than or equal to 33 hertz. This last seismic support in the rigid region and the supports on the seismic Category II piping are evaluated for safe shutdown earthquake loadings using the rules of ASME III Subsection NF.

3.7.3.13.4.3 Nonseismic Piping

Nonseismic piping within the impact evaluation zone is seismically supported, thereby ensuring that the pipe segment identified as a source will not fall or adversely impact the sensitive target (Table 3.7.3-2). This situation is shown in Figure 3.7.3-2, and the seismic supported piping criteria described below:

- *Supports within the impact evaluation zone, plus one transverse support in each transverse direction beyond the impact evaluation zone, are classified as seismic Category II and are evaluated for the safe shutdown earthquake loading using the rules of ASME III, Subsection NF.*

- *Piping within the impact evaluation zone plus one transverse support in each transverse direction are evaluated to Equation 9 of ASME Code, Section III, Class 3, with a stress limit equal to the smaller of 4.5 S_h and 3.0 S_y. Outside the impact evaluation zone, the nonseismic piping meets ASME B31.1 requirements.*

- *The nonseismic piping and seismic Category II supports are designed for loads from the nonseismic piping beyond the impact evaluation zone. This includes three plastic moment components (M_{p1}, M_{p2}, or M_{p3}) in each of three local coordinate directions applied at the first and last seismic Category II support. The responses to the three moments are evaluated independently. The response from the moments applied at the first seismic Category II support is combined with the response from the moments applied at the last seismic Category II support and with the responses to seismic anchor motions and equivalent static seismic inertia of the piping system by the absolute sum method. The support and anchor loads due to the plastic moments (M_{p1}, M_{p2}, or M_{p3}) of the seismically analyzed and supported section can be reduced if the elbow/bend resultant moments have exceeded the plastic limit moments of the elbow/bend. The value of the reduction factor RF is the same as the value for connected seismic Category II piping described above.*

- *The piping segment identified as the source has at least one effective axial support.*

- *Adequate free space between a source and a target is checked adding absolutely the piping safe shutdown earthquake deflections (defined following seismic Category II piping analysis methodology) and the safe shutdown earthquake target deflection. Also included are the displacements associated with the appropriate load cases.*

- *When the anchor is an equipment nozzle, the equipment is supported as seismic Category II as described in subsection 3.7.3.13.3.*]*

3.7.3.14 Seismic Analyses for Reactor Internals

See Subsection 3.9.2 for the dynamic analyses of reactor internals.

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3.7.3.15 Analysis Procedure for Damping

Damping values used in the seismic analyses of subsystems are presented in Subsection 3.7.1.3. Safe shutdown earthquake damping values used for different types of analysis are provided in Table 3.7.1-1. For subsystems that are composed of different material types, the composite modal damping approach with the weighted stiffness method is used to determine the composite modal damping value. Alternately, the minimum damping value may be used for these systems. [*Composite modal damping for coupled building and piping systems is used for piping systems that are coupled to the primary coolant loop system and the interior concrete building. Composite modal damping is used for piping systems that are coupled to flexible equipment or flexible valves. Piping systems analyzed by the uniform envelope response spectra method with rigid valves can be evaluated with 5 percent damping. Five percent damping is not used in piping systems that are susceptible to stress corrosion cracking.*]*

For the time history dynamic analysis and independent support motion response spectra analysis of piping systems, 4 percent, 3 percent, and 2 percent damping values are used as described in Table 3.7.1-1.

When piping systems and nonsimple module steel frames (simple frames are described in Subsection 3.7.3.8.3) are in a single coupled model, composite damping, as described in Subsection 3.7.1.3 is used.

3.7.3.16 Analysis of Seismic Category I Tanks

This subsection describes the seismic analyses for the large, atmospheric seismic Category I pools and tanks. These are reinforced concrete structures with stainless steel liners or with structural modules, as discussed in Subsections 3.8.3 and 3.8.4. They include the spent fuel pool in the auxiliary building, the in-containment refueling water storage tank, and the passive containment cooling water tank incorporated into the shield building roof. There are no other seismic Category I tanks.

The seismic analyses of the tank consider the impulsive and convective forces of the water as well as the flexibility of the walls. For the spent fuel pool, cask loading pit, cask washdown pit and fuel transfer canal, the impulsive loads are calculated by considering a portion of the water mass responding with the concrete walls. The impulsive forces are calculated by conventional methods for rigid tanks. The passive containment cooling water tank is analyzed using methods described in Reference 15 for toroidal tanks. It is also analyzed by finite element methods. The in-containment refueling water storage tank is irregular in plan and is analyzed by finite element methods.

3.7.3.17 Time History Analysis of Piping Systems

[*The time history dynamic analysis is an alternate seismic analysis method for response spectrum analysis when time history seismic input is used. This method is also used for dynamic analyses of piping systems subjected to time history hydraulic transient loadings or forcing functions induced by postulated pipe breaks. The modal superposition method is used to solve the equations of motion. The computer programs used are GAPPIPE, PIPESTRESS, ANSYS, and WECAN.*

The modal superposition method is based on the equations of motion which can be decoupled as long as the piping system is within its elastic limit. The modal responses are obtained from integrating the decoupled equations. The total responses are obtained by the algebraic sum of the individual responses of the individual modes at each time step. The cutoff frequency is selected based on the frequency content of the input forcing function and the highest significant frequency of the piping system. The integration time step is no larger than 10 percent of the period of the cutoff frequency.

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For dynamic analysis, including seismic analysis at a hard rock site, three separate analyses are performed for each loading case to account for uncertainties. The three analyses correspond to three different time scales: normal time, time expanded by 15 percent, and time compressed by 15 percent. For time history analysis of piping system models that include a dynamic model of the supporting concrete building either the building stiffness is varied by + and - 30 percent, or the time scale is shifted by + and - 15 percent. Alternately, when uniform enveloping time history analysis is performed, modeling uncertainties are accounted for by the spreading that is included in the broadened response spectra.

For time history analysis using the PIPESTRESS program, the response from the high frequency modes above the cutoff frequency is calculated based on the static response to the left-out-forces. This response is combined with the response from the low frequency modes by algebraic sum at each time step. Composite modal damping is used with PIPESTRESS program. The damping of the individual components is as listed in Table 3.7.1-1.

Alternately, for time history analysis using the PIPESTRESS, GAPPIPE, ANSYS, or WECAN programs, the number of modes used in the modal analysis is chosen so that the results of the dynamic analysis based on the chosen number of modes are within 10 percent of the results of the dynamic analysis based on the next higher number of modes used. The number of modes analyzed is selected to account for the principal vibration modes of the piping system. The modes are combined by algebraic sum. Composite modal damping is used with the ANSYS or WECAN programs. The damping of the individual components is as listed in Table 3.7.1-1.]*

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with Regulatory Guide 1.12

Compliance with Regulatory Guide 1.12 is discussed in this section and in Subsection 1.9.1.

Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operation and shutdown in accordance with Regulatory Guide 1.12.

3.7.4.1.1 Safety Design Basis

The seismic instrumentation serves no safety-related function and therefore has no nuclear safety design basis.

3.7.4.1.2 Power Generation Design Basis

The seismic instrumentation is designed to provide the following:

- Collection of seismic data in digital format
- Analysis of seismic data after a seismic event
- Operator notification that a seismic event meeting one-third SSE exceedance criteria has occurred
- Operator notification (after analysis of data) that a predetermined cumulative absolute velocity value has been exceeded

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3.7.4.2 Location and Description of Instrumentation

The following instrumentation and associated equipment are used to measure plant response to earthquake motion.Four triaxial acceleration sensor units, located as stated in Subsection 3.7.4.2.1, are connected to a time-history analyzer. The time-history analyzer recording and playback system is located in a panel in the nuclear island in a room near the main control room. Seismic event data from these sensors are recorded on a solid-state digital recording system at 200 samples per second per data channel.

This solid-state recording and analysis system has internal batteries and a charger to prevent the loss of data during a power outage, and to allow data collection and analysis in a seismic event during which the power fails. Normally 120 volt alternating current power is supplied from the non-Class 1E dc and uninterruptible power supply system. The system uses triaxial acceleration sensor input signals to initiate the time-history analyzer recording and main control room alarms. The system initiation value is adjustable from 0.002g to 0.02g.

The time-history analyzer starts recording triaxial acceleration data from each of the triaxial acceleration sensors after the initiation value has been exceeded. Pre-event recording time is adjustable from 1.2 to 15.0 seconds, and will be set to record at least 3 seconds of pre-event signal. Post-event run time is adjustable from 10 to 90 seconds. A minimum of 25 minutes of continuous recording is provided. Each recording channel has an associated timing mark record with 2 marks per second, with an accuracy of about 0.02 percent.

The instrumentation components are qualified to IEEE 344-1987 (Reference 16).

The sensor installation anchors are rigid so that the vibratory transmissibility over the design spectra frequency range is essentially unity.

3.7.4.2.1 Triaxial Acceleration Sensors

Each sensor unit contains three accelerometers mounted in a mutually orthogonal array mounted with one horizontal axis parallel to the major axis assumed in the seismic analysis. The triaxial acceleration sensors have a dynamic range of 1000 to 1 (0.001 to 1.0g) and a frequency range of 0.2 to 50 hertz.

One sensor unit will be located in the free field, as discussed in below. The AP1000 seismic monitoring system will provide for signal input from the free field sensor.

A second sensor unit is located on the nuclear island basemat in the spare battery charger room at elevation 66′-6″ near column lines 9 and L.

A third sensor unit is located on the shield building structure at elevation 266′ near column lines 4-1 and K.

The fourth sensor unit is located on the containment internal structure on the east wall of the east steam generator compartment just above the operating floor at elevation 138′ close to column lines 6 and K.

Seismic instrumentation is not located on equipment, piping, or supports since experience has shown that data obtained at these locations are obscured by vibratory motion associated with normal plant operation.

A free-field sensor will be located and installed to record the ground surface motion representative of the site. It will be located such that the effects associated with surface features, buildings, and

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components on the recorded ground motion will be insignificant. The trigger value is initially set at 0.01g.

3.7.4.2.2 Time-History Analyzer

The time-history analyzer receives input from the triaxial acceleration sensors and, when activated as described in Subsection 3.7.4.3, begins recording the triaxial data from each triaxial acceleration sensor and initiates audio and visual alarms in the main control room.

This recorded data will be used to evaluate the seismic acceleration of the structure on which the triaxial acceleration sensors are mounted.

The time-history analyzer is a multichannel, digital recording system with the capability to automatically download the recorded acceleration data to a dedicated computer for data storage, playback, and analysis after a seismic event.

The time-history analyzer can compute cumulative absolute velocity (CAV) and the 5 percent of critical damping response spectrum for frequencies between 1 and 10 Hz. The operator may select the analysis of either CAV or the response spectrum. Analysis results are printed out on a dedicated graphics printer that is part of the system and is located in the same panel as the time-history analyzer.

3.7.4.3 Control Room Operator Notification

The time-history analyzer provides for initiation of audible and visual alarms in the main control room when the one-third SSE levels sensed by any of the triaxial acceleration sensors are exceeded and when the system is activated to record a seismic event. In addition to alarming when the system is activated, the analyzer portion of the system will provide a second alarm if the predetermined cumulative absolute velocity value has been exceeded by any of the sensors. Alarms are annunciated in the main control room.

3.7.4.4 Comparison of Measured and Predicted Responses

The recorded seismic data is used by the combined license holder operations and engineering departments to evaluate the effects of the earthquake on the plant structures and equipment.

The criterion for initiating a plant shutdown following a seismic event will be exceedance of a specified response spectrum limit and a cumulative absolute velocity limit. The seismic instrumentation system is capable of computing the cumulative absolute velocity as described in EPRI Report NP-5930 (Reference 1) and EPRI Report TR-100082 (Reference 17).

Post-earthquake operating procedures utilize the guidance of EPRI Reports NP-5930, TR-100082, and NP-6695, as modified and endorsed by the NRC in Regulatory Guides 1.166 and 1.167. A response spectrum check up to 10Hz and the cumulative absolute velocity will be calculated based on the recorded motions at the free field instrument. If the operating basis earthquake ground motion is exceeded or significant plant damage occurs, the plant must be shutdown in an orderly manner.

In addition, the procedures address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit, between the individual spent fuel racks, and from the spent fuel racks to the spent fuel pool walls, and provide for appropriate corrective actions to be taken if needed (such as repositioning the racks or analysis of the as-found condition).

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3.7.4.5 Tests and Inspections

Periodic testing of the seismic instrumentation system is accomplished by the functional test feature included in the software of the time-history recording accelerograph. The system is modular and is capable of single-channel testing or single channel maintenance without disabling the remainder of the system.

Installation and acceptance testing of the triaxial acceleration sensors described in Subsection 3.7.4.2.1 is completed prior to initial startup. Installation and acceptance testing of the time-history analyzer described in Subsection 3.7.4.2.2 is completed prior to initial startup.

3.7.5 Combined License Information

3.7.5.1 Seismic Analysis of Dams

Combined License applicants referencing the AP1000 certified design will evaluate dams whose failure could affect the site interface flood level specified in subsection 2.4.1.2. The evaluation of the safety of existing and new dams will use the site-specific safe shutdown earthquake.

3.7.5.2 Post-Earthquake Procedures

Site-specific procedures for activities following an earthquake are addressed in Subsection 3.7.4.4.

3.7.5.3 Seismic Interaction Review

The seismic interaction review will be updated for as-built information. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition. The as-built seismic interaction review is completed prior to fuel load.

3.7.5.4 Reconciliation of Seismic Analyses of Nuclear Island Structures

The seismic analyses described in Subsection 3.7.2 will be reconciled for detailed design changes, such as those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information. Deviations are acceptable based on an evaluation consistent with the methods and procedure of Section 3.7 provided the amplitude of the seismic floor response spectra, including the effect due to these deviations, does not exceed the design basis floor response spectra by more than 10 percent. This reconciliation will be completed prior to fuel load.

3.7.5.5 Free Field Acceleration Sensor

The location for the free-field acceleration sensor is addressed in Subsection 3.7.4.2.1.

3.7.6 References

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Table 3.7.1-1
Safe Shutdown Earthquake Damping Values

	Percent
Welded and friction-bolted steel structures and equipment	4
Bearing bolted structures and equipment	7
Prestressed concrete structures	5
Reinforced concrete structures	7
Concrete filled steel plate structures	5
[*Piping (for uniform envelope response spectra analysis)*	*5*
Piping (alternative for time history analysis and independent support motion response spectra analysis)	
Less than or equal to 12-inch diameter	*2*
Greater than 12-inch diameter	*3*
Primary coolant loop	*4*]*
Fuel assemblies	20
Control rod drive mechanisms	5
Full cable trays and related supports	10
Empty cable trays and related supports	7
Conduits and related supports	7
HVAC ductwork	7
HVAC welded ductwork	4
Cabinets and panels for electrical equipment	5
Equipment such as welded instrument racks and tanks	3

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.7.1-2
Embedment Depth and Related
Dimensions of Category I Structures

Structure	Foundation Embedment Depth (ft)	Least Foundation Width (ft)	Structure Height (ft)
Shield Building	See Note	See Note	268.25
Steel Containment Vessel	See Note	See Note	215.33
Auxiliary Building	See Note	See Note	119.50

Note:

1. The seismic Category I structures are founded on a common basemat embedded 39.5 feet, [*with dimensions shown in Figure 3.7.1-14.*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.7.1-3
AP1000 Design Response Spectra
Amplification Factors for Control Points

HORIZONTAL					
Percent of Critical Damping	Acceleration[1]				Displacement[1]
	A (33 cps)	B' (25 cps)[2]	B (9 cps)	C (2.5 cps)	D (0.25 cps)
2.0	1.0	1.70	3.54	4.25	2.50
3.0	1.0	1.66	3.13	3.76	2.34
4.0	1.0	1.63	2.84	3.41	2.19
5.0	1.0	1.60	2.61	3.13	2.05
7.0	1.0	1.55	2.27	2.72	1.88

VERTICAL					
Percent of Critical Damping	Acceleration[1]				Displacement[1]
	A (33 cps)	B' (25 cps)[2]	B (9 cps)	C (3.5 cps)	D (0.25 cps)
2.0	1.0	1.70	3.54	4.05	1.67
3.0	1.0	1.66	3.13	3.58	1.56
4.0	1.0	1.63	2.84	3.25	1.46
5.0	1.0	1.60	2.61	2.98	1.37
7.0	1.0	1.55	2.27	2.59	1.25

Notes:

1. Maximum ground displacement is taken proportional to maximum ground acceleration, and is 36 inches for ground acceleration of 1.0 gravity.
2. The 5 percent damping amplification factor for control point B' is derived per discussion in Subsection 3.7.1.1. This 5 percent damping amplification factor equals 1.3 times the RG 1.60 response spectra at 25 hertz. The amplification factors at control point B' for other damping values are determined by increasing the RG 1.60 response spectra at 25 hertz by 30 percent.

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Table 3.7.1-4 (Sheet 1 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Firm Rock								
0.0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.15	57422	3500	57032	3499	0.015
10.0	5.0	2	0.15	57422	3500	56600	3486	0.016
15.0	5.0	3	0.15	57422	3500	55943	3465	0.017
20.0	5.0	4	0.15	57422	3500	55511	3452	0.018
25.0	5.0	5	0.15	56442	3500	55933	3465	0.016
30.0	5.0	6	0.15	56442	3500	55436	3450	0.017
33.5	3.5	7	0.15	57422	3500	56076	3470	0.015
39.5	6.0	8	0.15	57422	3500	55898	3464	0.015
45.0	5.5	9	0.15	57422	3500	55716	3458	0.016
50.0	5.0	10	0.15	57422	3500	55575	3454	0.016
60.0	10.0	11	0.15	56442	3500	55400	3449	0.017
80.0	20.0	12	0.15	56442	3500	54695	3427	0.019
100.0	20.0	13	0.15	56442	3500	53358	3384	0.021
120.0	20.0	14	0.15	56442	3500	52295	3351	0.023
Bedrock	–	–	0.15	300000	8000	298137	8000	0.02

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Table 3.7.1-4 (Sheet 2 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Soft Rock								
0.0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.15	27660	2429	27425	2426	0.016
10.0	5.0	2	0.15	29180	2495	28318	2466	0.018
15.0	5.0	3	0.15	30262	2541	28819	2487	0.020
20.0	5.0	4	0.15	30620	2556	28589	2477	0.023
25.0	5.0	5	0.15	30920	2568	29290	2508	0.019
30.0	5.0	6	0.15	31384	2588	29481	2516	0.021
33.5	3.5	7	0.15	31932	2610	30768	2570	0.017
39.5	6.0	8	0.15	32464	2632	31144	2586	0.018
45.0	5.5	9	0.15	33042	2655	31314	2593	0.019
50.0	5.0	10	0.15	33668	2680	31598	2604	0.020
60.0	10.0	11	0.15	34341	2707	31826	2614	0.021
80.0	20.0	12	0.15	35021	2733	31738	2610	0.024
100.0	20.0	13	0.15	35708	2760	31585	2604	0.026
120.0	20.0	14	0.15	36401	2787	31585	2604	0.028
Bedrock	–	–	0.15	300000	8000	298137	8000	0.020

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Table 3.7.1-4 (Sheet 3 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Upper Bound Soft to Medium Soil								
0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.11	6873	1414	6664	1397	0.018
10.0	5.0	2	0.11	9844	1692	9202	1641	0.023
15.0	5.0	3	0.11	13917	2012	12880	1942	0.024
20.0	5.0	4	0.11	14971	2087	13629	1997	0.027
25.0	5.0	5	0.11	15645	2133	14574	2065	0.022
30.0	5.0	6	0.11	16419	2186	15045	2099	0.024
33.5	3.5	7	0.11	17873	2280	16908	2225	0.019
39.5	6.0	8	0.11	19036	2353	17873	2287	0.020
45.0	5.5	9	0.11	20387	2435	18996	2358	0.021
50.0	5.0	10	0.11	21726	2514	20136	2428	0.021
60.0	10.0	11	0.11	23234	2600	21366	2501	0.022
80.0	20.0	12	0.11	24712	2681	22314	2556	0.024
100.0	20.0	13	0.11	26151	2758	23137	2602	0.026
120.0	20.0	14	0.11	27546	2831	24009	2651	0.027
Bedrock	–	–	0.15	300000	8000	298137	8000	0.020

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Table 3.7.1-4 (Sheet 4 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Soft-to-Medium Soil								
0.0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.11	3438	1000	3222	971	0.023
10.0	5.0	2	0.11	4923	1197	4355	1129	0.031
15.0	5.0	3	0.11	6960	1423	5987	1324	0.035
20.0	5.0	4	0.11	7487	1476	6161	1343	0.040
25.0	5.0	5	0.11	7824	1509	6699	1400	0.031
30.0	5.0	6	0.11	8211	1546	6891	1420	0.033
33.5	3.5	7	0.11	8938	1613	7872	1518	0.026
39.5	6.0	8	0.11	9520	1664	8317	1560	0.027
45.0	5.5	9	0.11	10195	1722	8834	1608	0.028
50.0	5.0	10	0.11	10864	1778	9347	1654	0.029
60.0	10.0	11	0.11	11618	1838	9818	1695	0.031
80.0	20.0	12	0.11	12357	1896	10031	1714	0.036
100.0	20.0	13	0.11	13077	1950	10201	1728	0.040
120.0	20.0	14	0.11	13774	2002	10512	1754	0.043
Bedrock	–		0.15	300000	8000	298137	8000	0.020

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Table 3.7.1-4 (Sheet 5 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Soft Soil								
0.0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.11	3438	1000	3222	971	0.023
10.0	5.0	2	0.11	3633	1028	3042	944	0.038
15.0	5.0	3	0.11	3865	1060	2974	933	0.047
20.0	5.0	4	0.11	3921	1068	2752	898	0.059
25.0	5.0	5	0.11	3955	1073	2922	925	0.049
30.0	5.0	6	0.11	3994	1078	2762	899	0.056
33.5	3.5	7	0.11	4065	1088	3022	941	0.046
39.5	6.0	8	0.11	4121	1095	2958	931	0.049
45.0	5.5	9	0.11	4183	1103	2896	921	0.053
50.0	5.0	10	0.11	4244	1111	2851	914	0.056
60.0	10.0	11	0.11	4310	1120	2774	901	0.062
80.0	20.0	12	0.11	4374	1128	2668	884	0.068
100.0	20.0	13	0.11	4434	1136	2691	888	0.069
120.0	20.0	14	0.11	4492	1143	2718	892	0.069
Bedrock	–	–	0.15	300000	8000	298137	8000	0.020

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Tables 3.7.2-1–3.7.2-16 Not Used

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Table 3.7.3-1 (Sheet 1 of 3)
Seismic Category I Equipment Outside Containment by Room Number

Room No.	Room Name	Equipment Description
12101	Division A battery room	Batteries
12102	Division C battery room 1	Batteries
12103	Spare battery room	Spare batteries
12104	Division B battery room 1	Batteries
12105	Division D battery room	Batteries
12113	Spare battery charger room	
12154	Auxiliary building sump room	RCA floodup level sensors
12162	RNS pump room A	RNS pressure boundary
12163	RNS pump room B	RNS pressure boundary
12201	Division A dc equipment room	dc equipment
12202	Division C battery room 2	Batteries
12203	Division C dc equipment room	dc equipment
12204	Division B battery room 2	Batteries
12205	Division D dc equipment room	dc equipment
12207	Division B dc equipment room	dc equipment
12211	Corridor	Divisional cables SGS instrumentation tubing
12212	Division B RCP trip switchgear room	RCP trip switchgear
12244	Lower annulus valve area	CVS/WLS containment isolation valves
12251	Demineralizer/filter access area	CVS/DWS isolation valves
12254	SFS penetration room	SFS containment isolation valve
12256	Containment isolation valve room	RNS containment isolation valves
12259	Pipe chase	RNS piping
12262	Piping/Valve room	RNS pressure boundary, SFS piping
12265	Waste monitor tank room C	SFS piping
12269	Pipe chase	RNS pressure boundary
12300	Corridor	Divisional cable, MCR load shed panel SGS instrumentation tubing
12301	Division A I&C room	Divisional I&C
12302	Division C I&C room	Divisional I&C

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Table 3.7.3-1 (Sheet 2 of 3)
Seismic Category I Equipment Outside Containment by Room Number

Room No.	Room Name	Equipment Description
12303	Remote shutdown room	Divisional cabling
12304	Division B I&C/penetration room	Divisional I&C/electrical penetrations
12305	Division D I&C/penetration room	Divisional I&C/electrical penetrations
12306	Valve/piping penetration room	CCS/CVS/DWS/FPS/SGS containment isolation valves SGS instrumentation tubing
12311	Corridor	Divisional cabling
12312	Division C RCP trip switchgear room	RCP trip switchgear
12313	Division C I&C/penetration room	Divisional I&C/electrical penetrations
12321	Non-1E equipment/penetration room	Divisional cabling/electrical penetrations
12341	Middle annulus	Class 1E electrical penetrations Various mechanical piping penetrations
12343	Division C middle annulus penetration room	Class 1E electrical penetrations
12344	Division B middle annulus penetration room	Class 1E electrical penetrations
12345	Division D middle annulus penetration room	Class 1E electrical penetrations
12351	Maintenance floor staging area	Divisional cabling (ceiling)
12352	Personnel hatch	Personnel airlock (interlocks)
12354	Middle annulus access room	PSS/SFS containment isolation valves
12362	RNS HX room	RNS pressure boundary
12365	Waste monitor tank room B	SFS piping
12400	Control room vestibule	Control room access
12401	Main control room	Dedicated safety panel VBS HVAC dampers VES isolation valves Lighting circuits Mounting for lighting fixtures
12404	Lower MSIV compartment B	SGS containment isolation valves, instrumentation
12405	Lower VBS B and D equipment room	VWS/PXS/CAS containment isolation valves
12406	Lower MSIV compartment A	SGS containment isolation valves, instrumentation
12411	Corridor	Divisional cabling
12412	Electrical penetration room Division A	Divisional electrical penetrations, MCR load shed panel

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Table 3.7.3-1 (Sheet 3 of 3)
Seismic Category I Equipment Outside Containment by Room Number

Room No.	Room Name	Equipment Description
12421	Non 1E equipment/penetration room	Divisional cabling/electrical penetrations
12422	Reactor trip switchgear II	Reactor trip switchgear
12423	Reactor trip switchgear I	Reactor trip switchgear
12452	VFS penetration room	VFS containment isolation valves, divisional cabling
12454	VFS/SFS/PSS penetration room	SFS/PSS/VFS containment isolation valves, RNS pressure boundary
12462	Cask washdown pit	SFS piping
12463	Cask loading pit/Spent fuel storage pit	Spent fuel transfer gate
12501	VBS MCR/A&C equipment room	Divisional cabling
12504	Upper MSIV compartment B	SGS CIVs, instrumentation
12506	Upper MSIV compartment A	SGS CIVs, instrumentation
12541	Upper annulus	PCS piping and cabling PCS air baffle
12553	Personnel access area	Personnel airlock (interlocks)
12555	VES air storage	VES high pressure air bottles
12651	VAS Equipment Room	VFS containment isolation valves
12563	Spent fuel storage pit	Spent fuel storage racks Spent fuel transfer gate Spent fuel cask loading pit gate
12701	PCS valve room	PCS isolation valves/instrumentation
12703	PCS water storage tank	PCS piping, level and temperature instrumentation

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Table 3.7.3-2
Equipment Classified as Sensitive Targets for
Seismically Analyzed Piping, HVAC Ducting, Cable Tray Systems

Component	Discussion	Zone of Protection
Seismic Category I Valve No Class 1E Electrical Equipment Not pressure sensitive	These are manual valves. The actuator must be protected from impact.	Valve body and actuator area
Seismic Category I Valve Class 1E Electrical Equipment Pressure sensitive	These valves have sensitive Class 1E equipment (e.g., Position indicators, limit switches, motor operator) or solenoid valves.	One support (acting in direction of impact) on each side of valve
Seismic Category I Dampers	The actuator must be protected along with any Class 1E equipment.	Within one support (acting in direction of impact) on each side of HVAC
Monitors	This includes: neutron detectors, radiation monitors, resistance temperature detectors, speed sensors, thermocouples, and transmitters.	Monitors and associated wiring
Sensitive Electrical Equipment Housed in Cabinets, Panels or Boards	This includes: relays, contractors, breakers, and switchgear.	Cabinets, panels, and boards housing sensitive devices
Class 1E exposed cables and wiring	Cables and wiring which are not housed in cable trays or conduits must be protected.	Exposed cables and wiring
Device or Instrument Tubing	Any device or tubing that could be damaged resulting in the loss of the pressure boundary of a safety class line.	Device or tubing
Penetrations	Rigid penetrations are considered robust. Floating penetrations with bellows are considered sensitive.	Floating penetration and associated bellows

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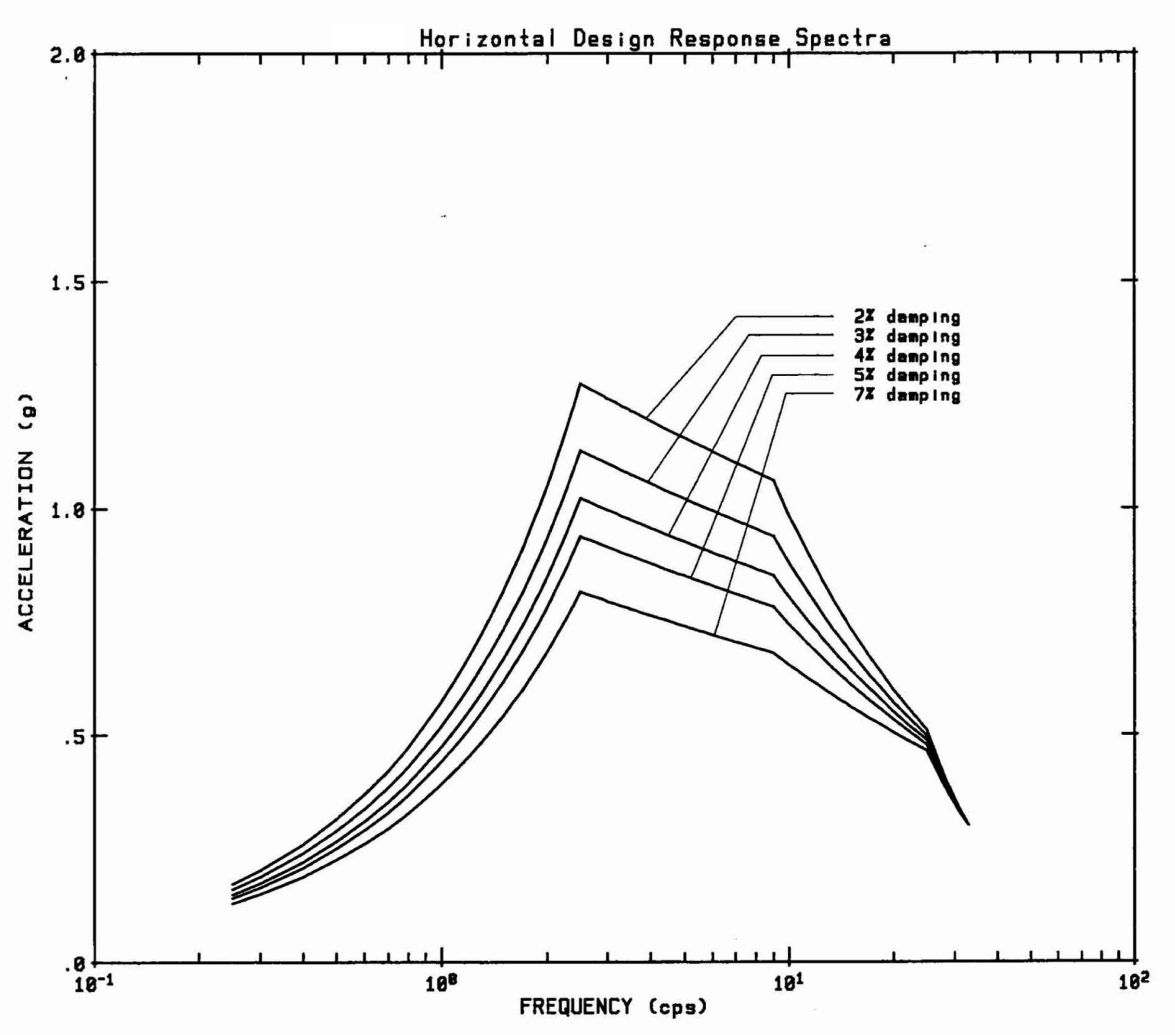

Figure 3.7.1-1
Horizontal Design Response Spectra
Safe Shutdown Earthquake

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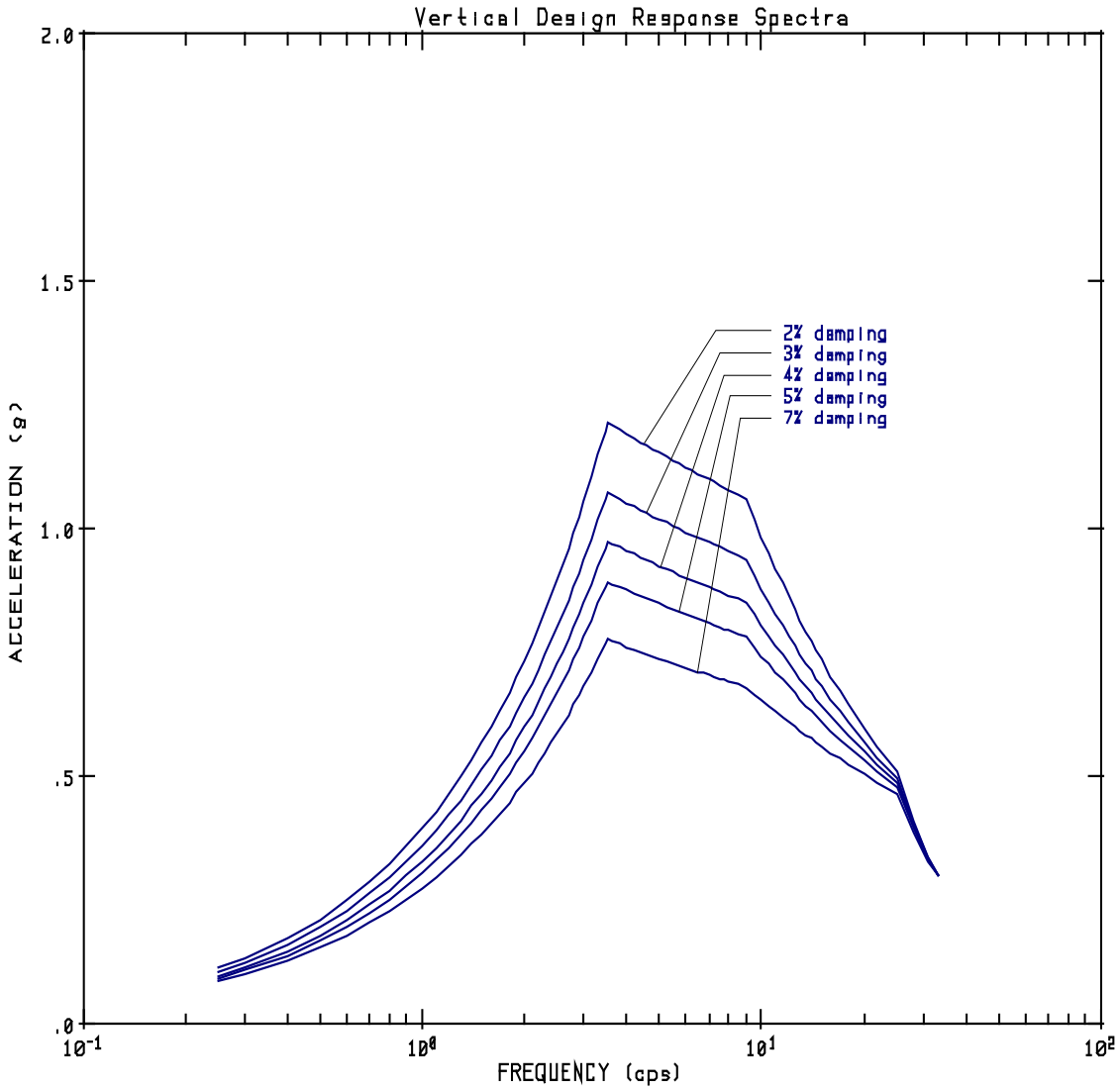

Figure 3.7.1-2
Vertical Design Response Spectra
Safe Shutdown Earthquake

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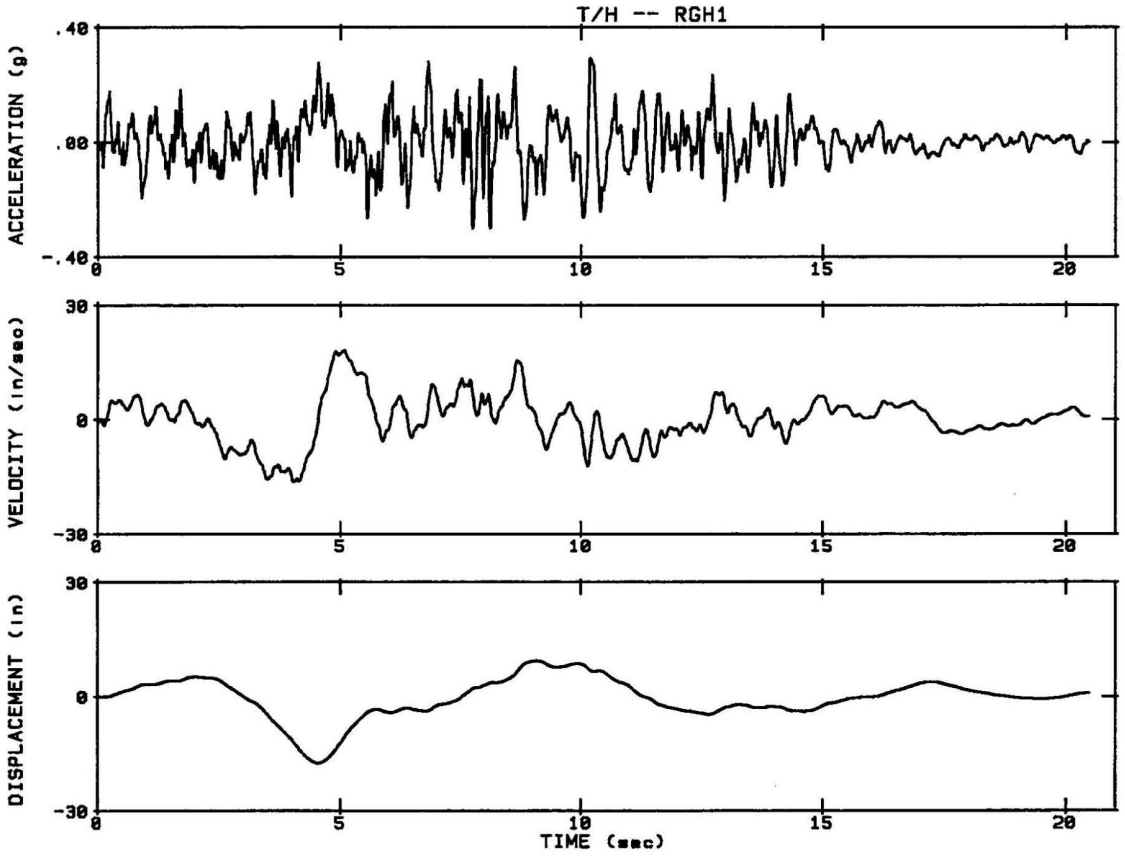

Figure 3.7.1-3
Design Horizontal Time History, "H1"
Acceleration, Velocity & Displacement Plots

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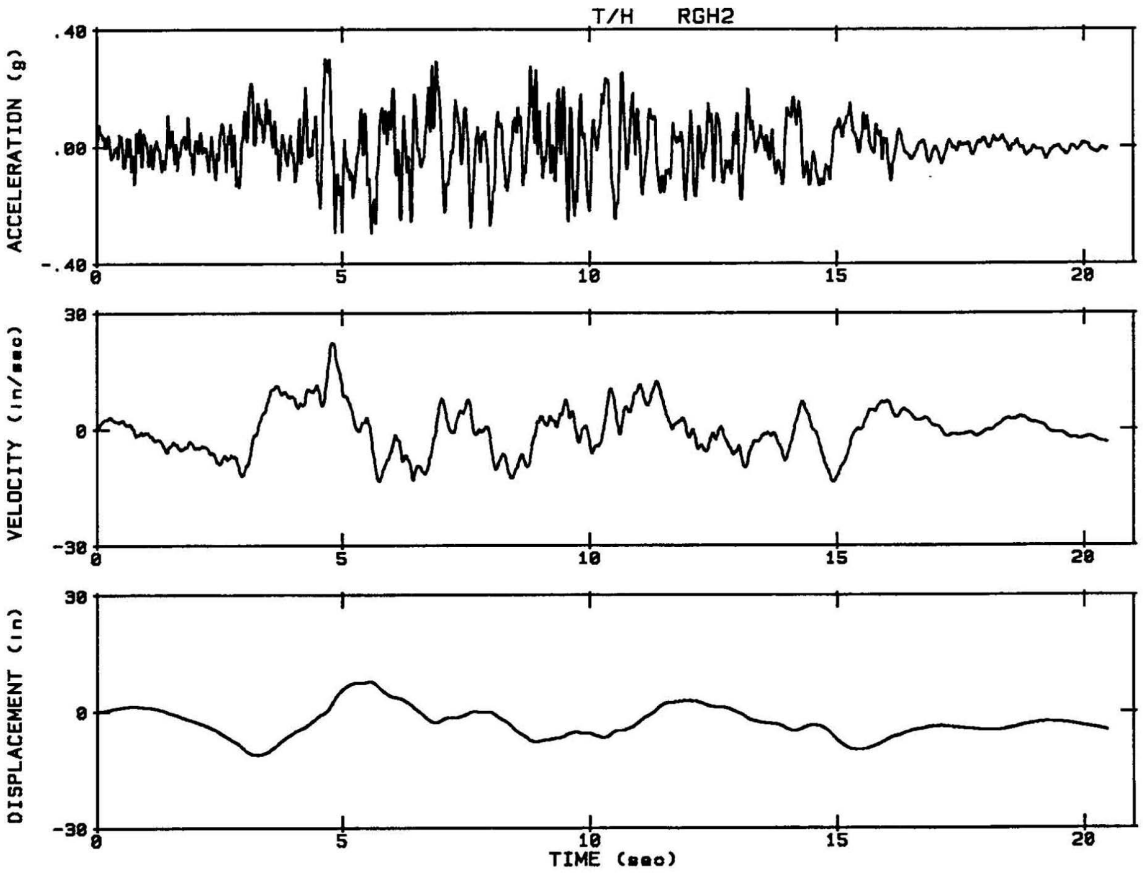

Figure 3.7.1-4
Design Horizontal Time History, "H2"
Acceleration, Velocity & Displacement Plots

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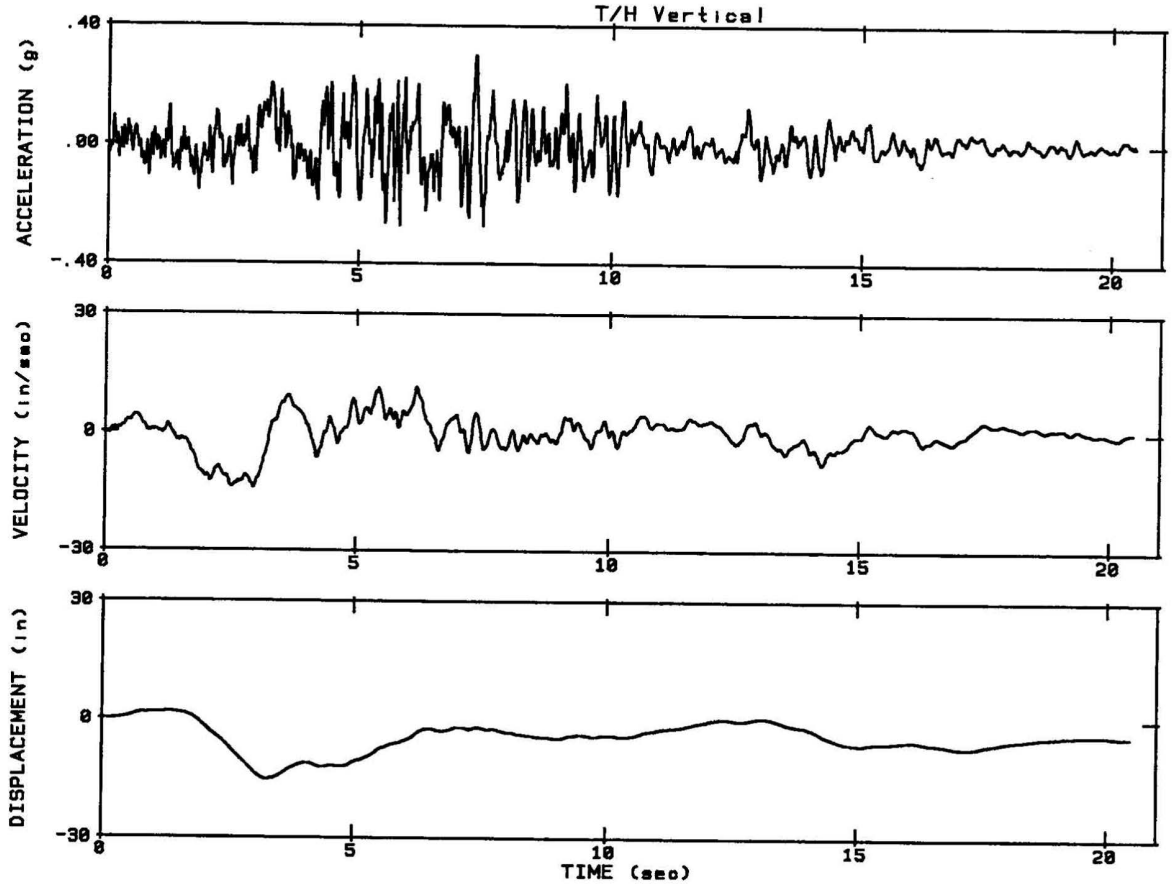

Figure 3.7.1-5
Design Vertical Time History
Acceleration, Velocity & Displacement Plots

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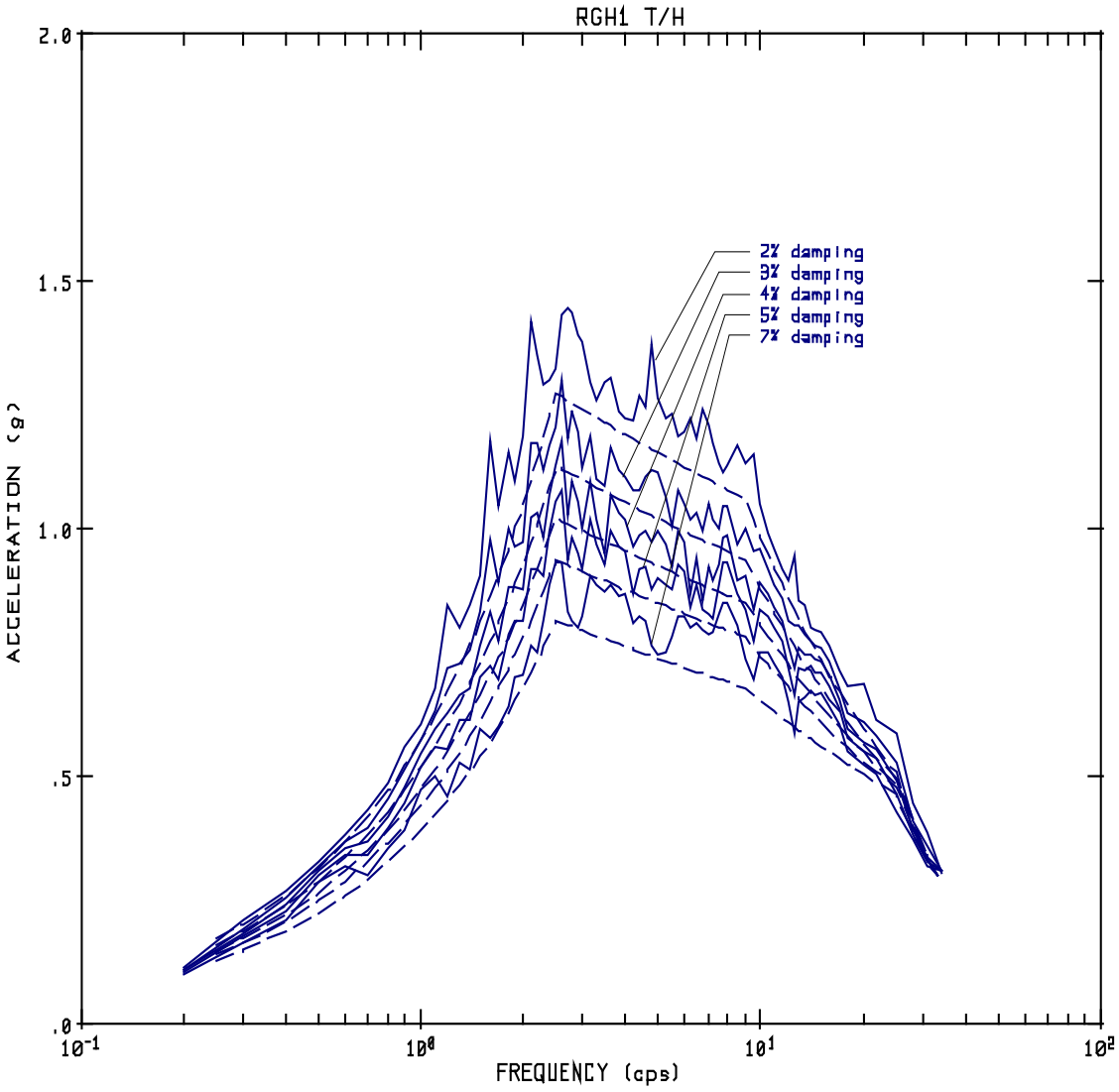

Figure 3.7.1-6
Acceleration Response Spectra of
Design Horizontal Time History, "H1"

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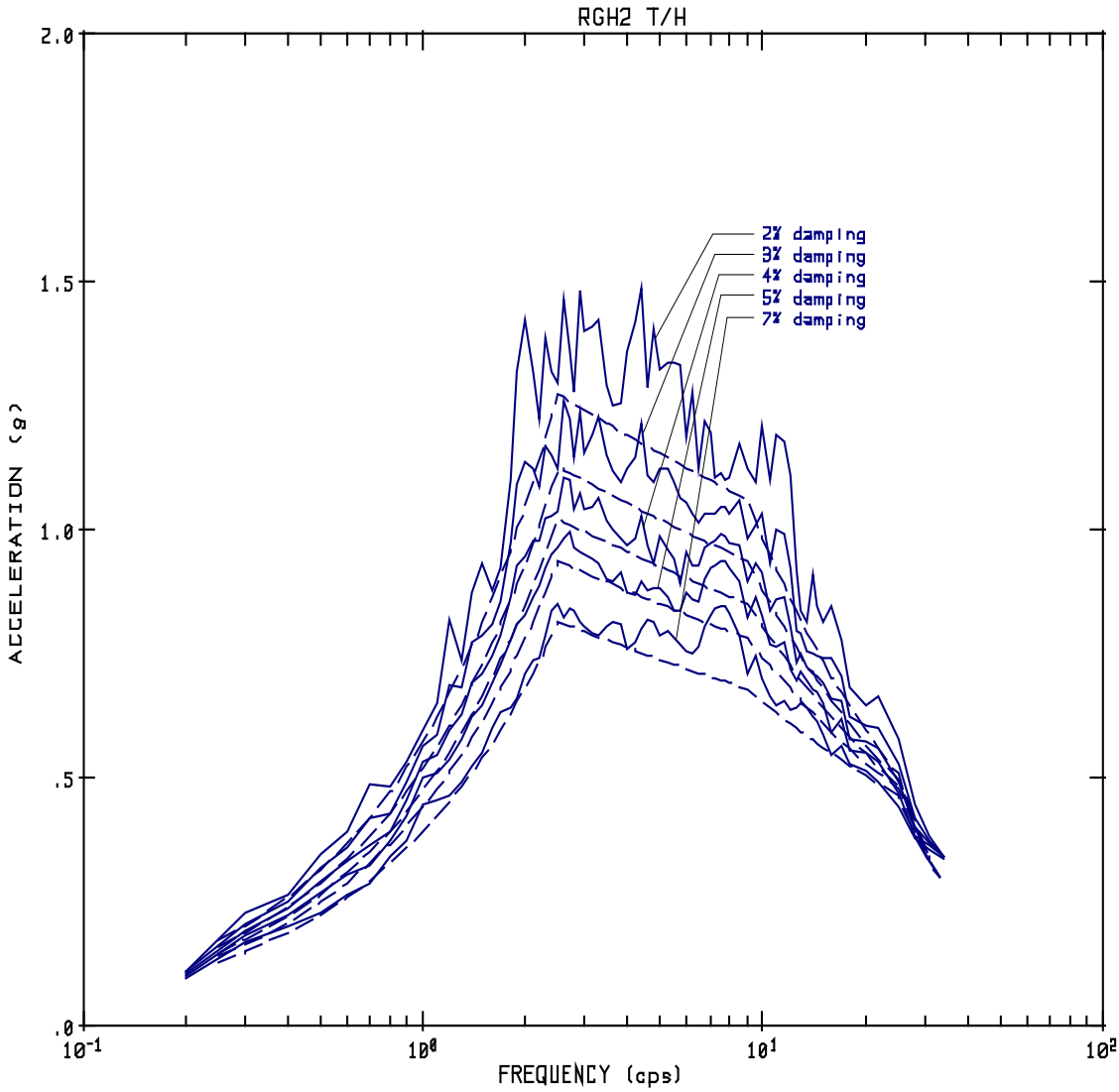

Figure 3.7.1-7
Acceleration Response Spectra of
Design Horizontal Time History, "H2"

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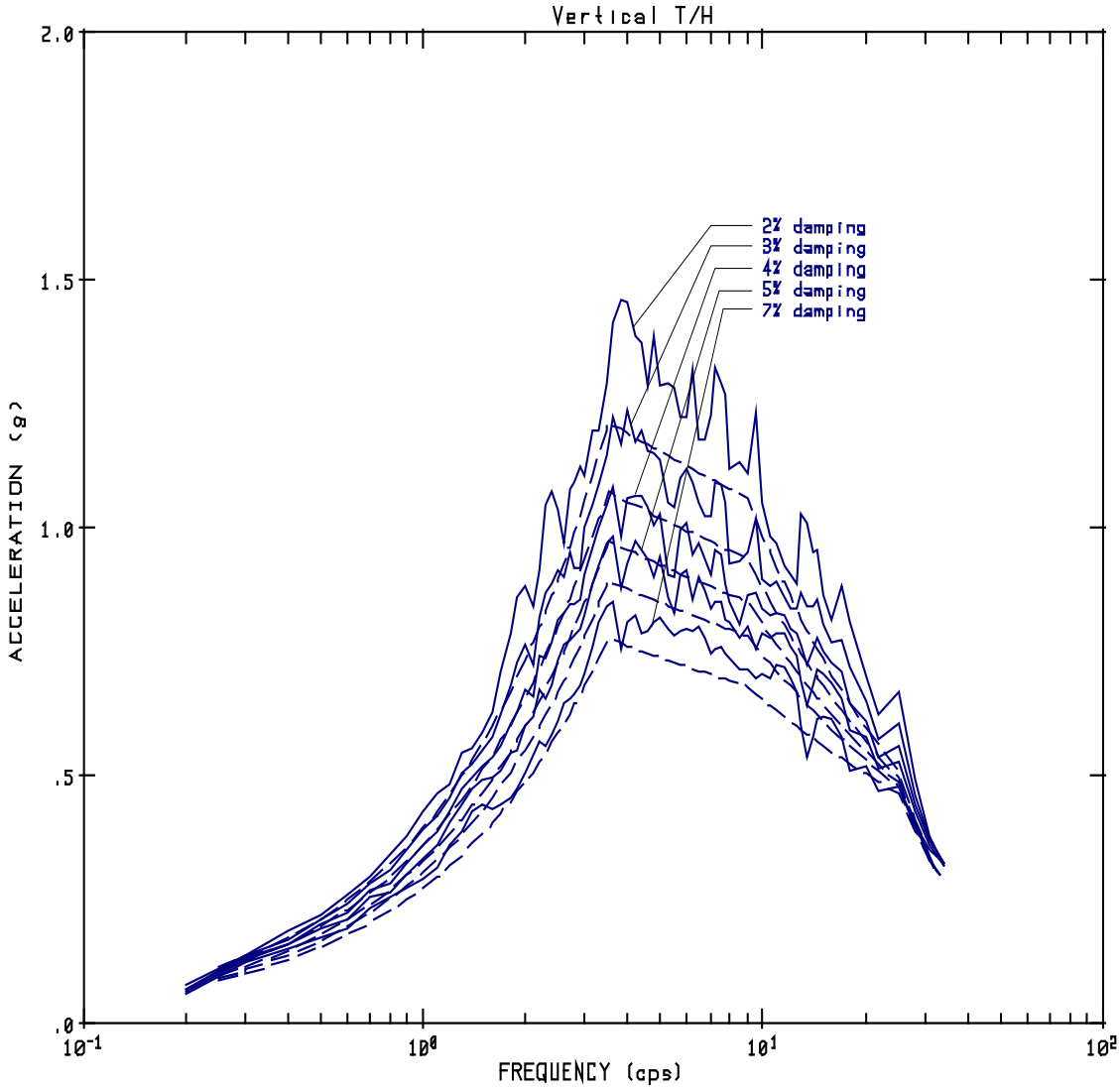

Figure 3.7.1-8
Acceleration Response Spectra of Design Vertical Time History

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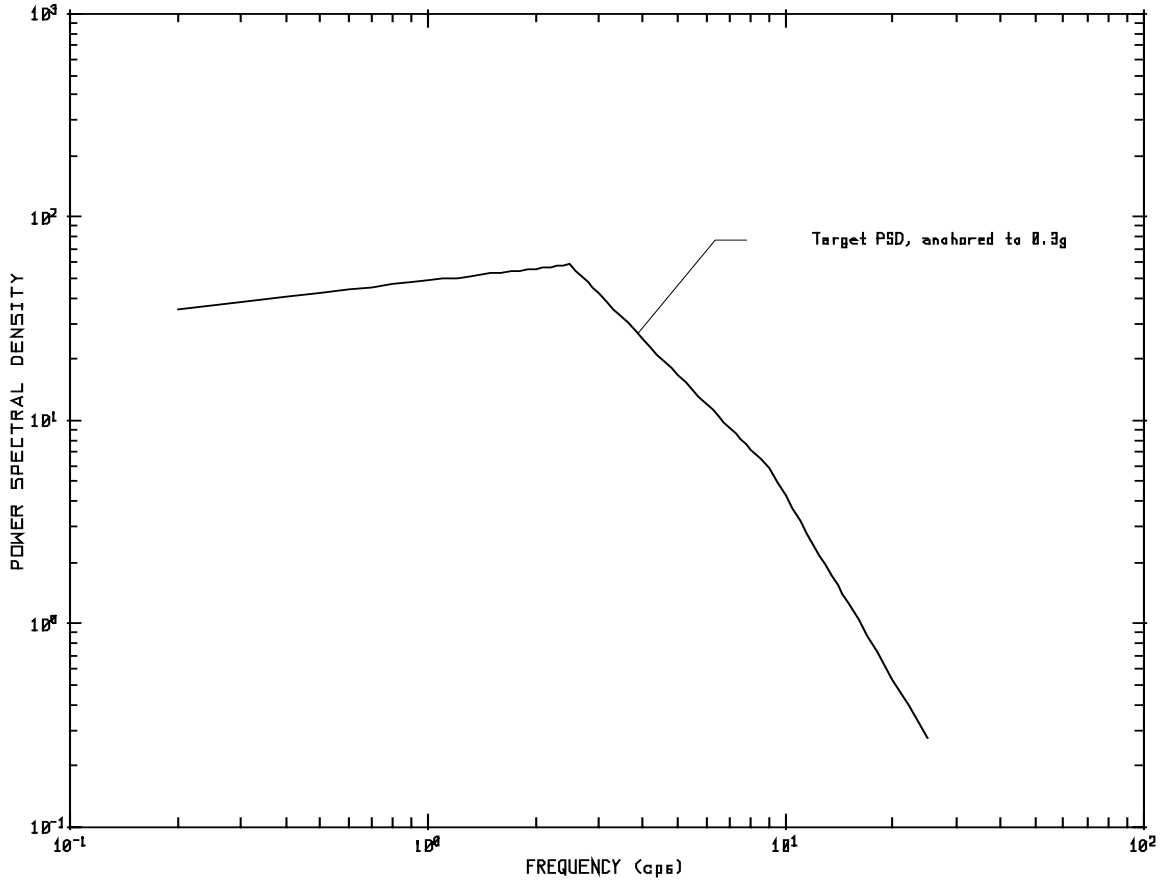

Figure 3.7.1-9
Minimum Power Spectral Density Curve
(Normalized to 0.3g)

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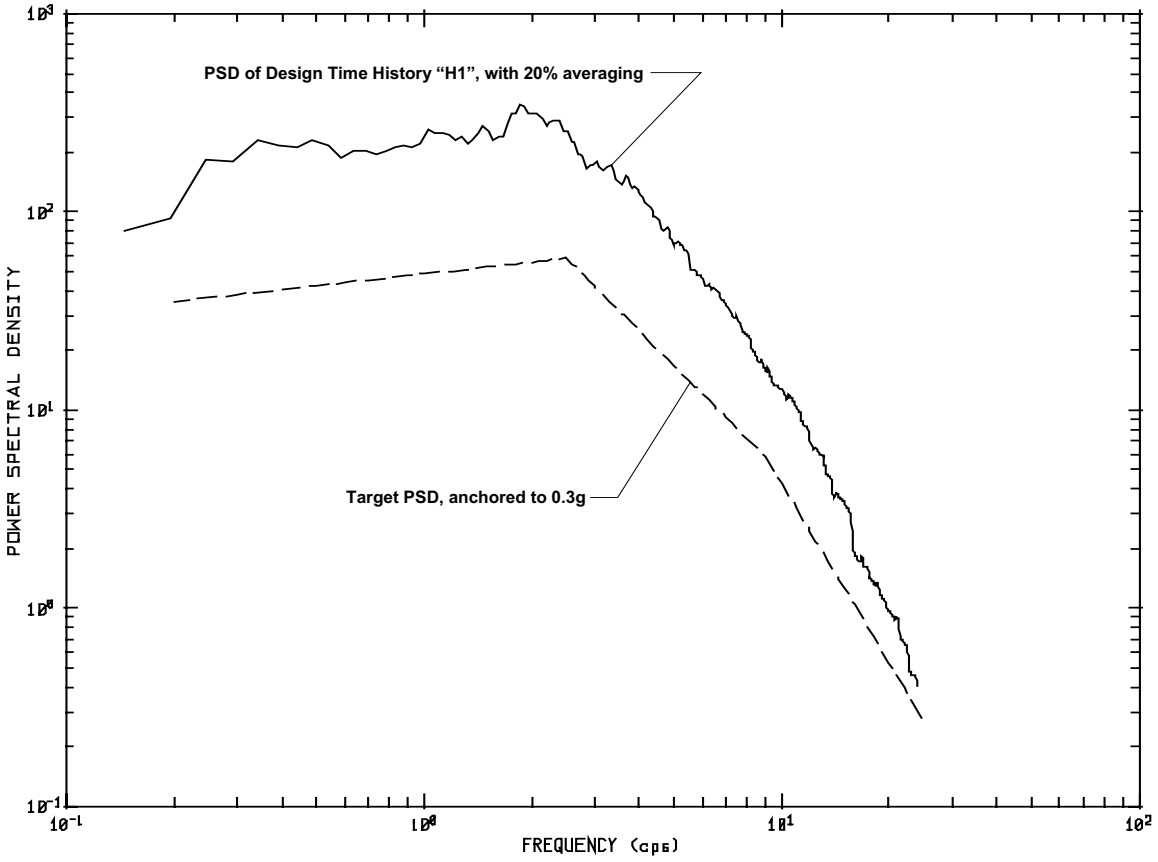

Figure 3.7.1-10
Power Spectral Density of
Design Horizontal Time History, "H1"

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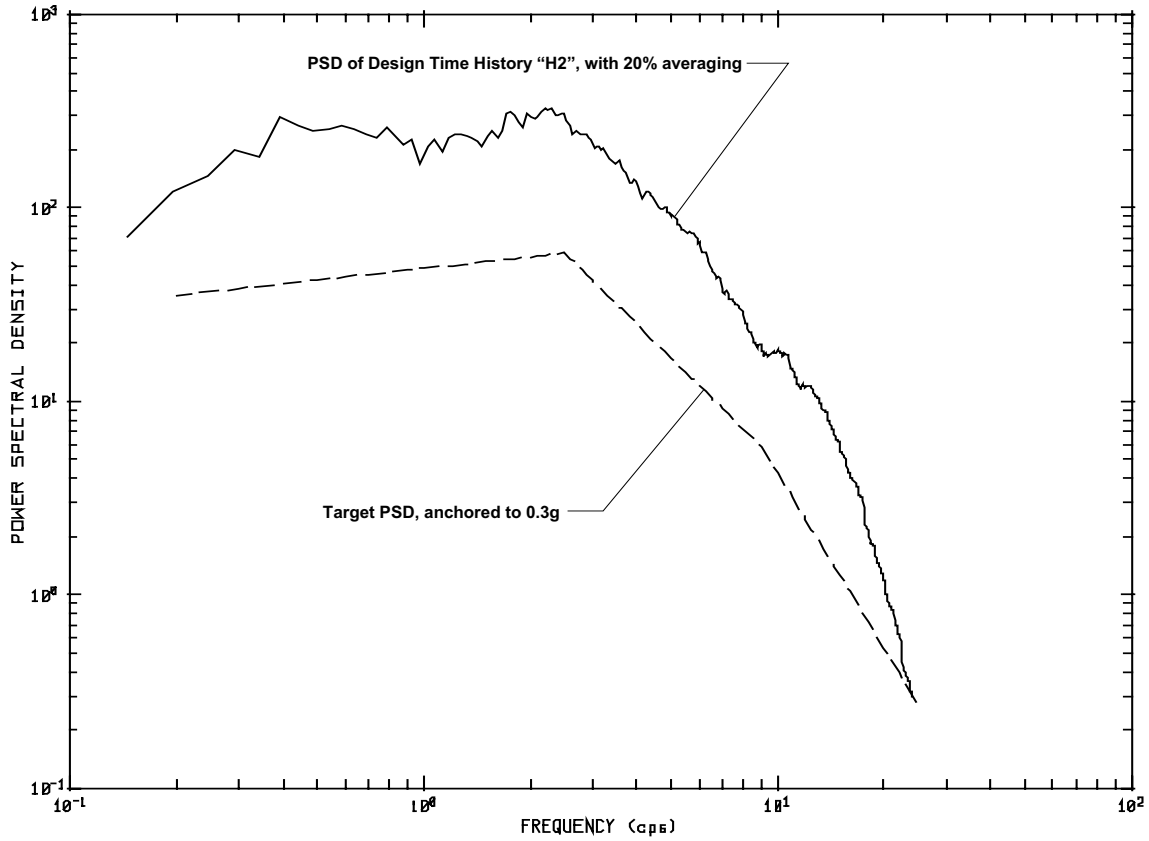

Figure 3.7.1-11
Power Spectral Density of
Design Horizontal Time History, "H2"

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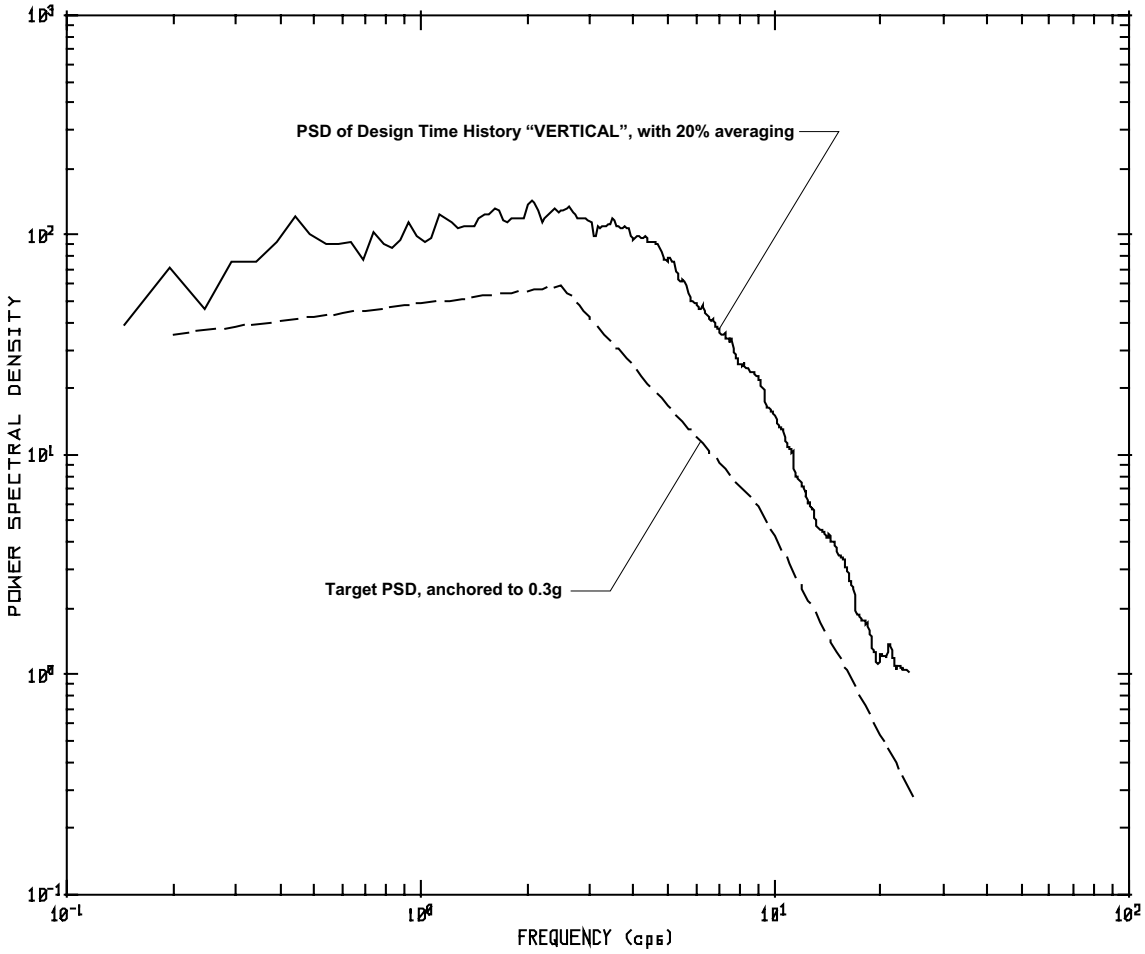

Figure 3.7.1-12
Power Spectral Density of
Design Vertical Time History

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Figure 3.7.1-13 Not Used

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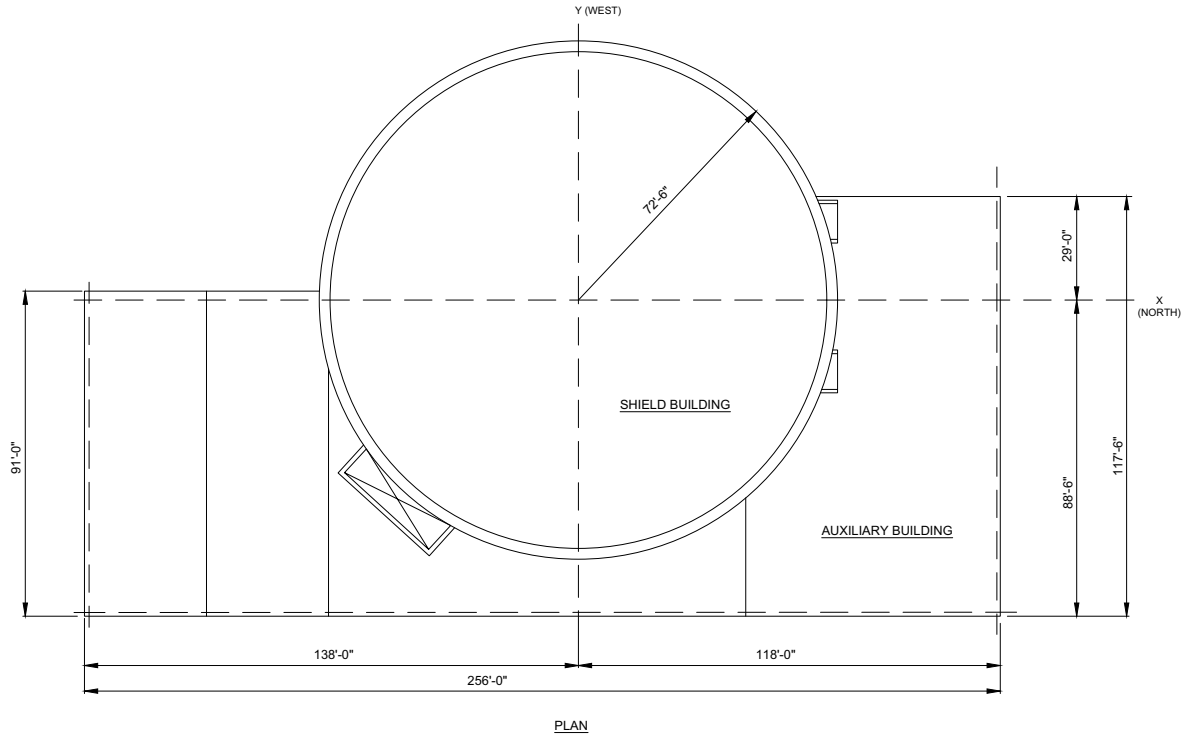

Figure 3.7.1-14
[*Nuclear Island Structures Dimensions*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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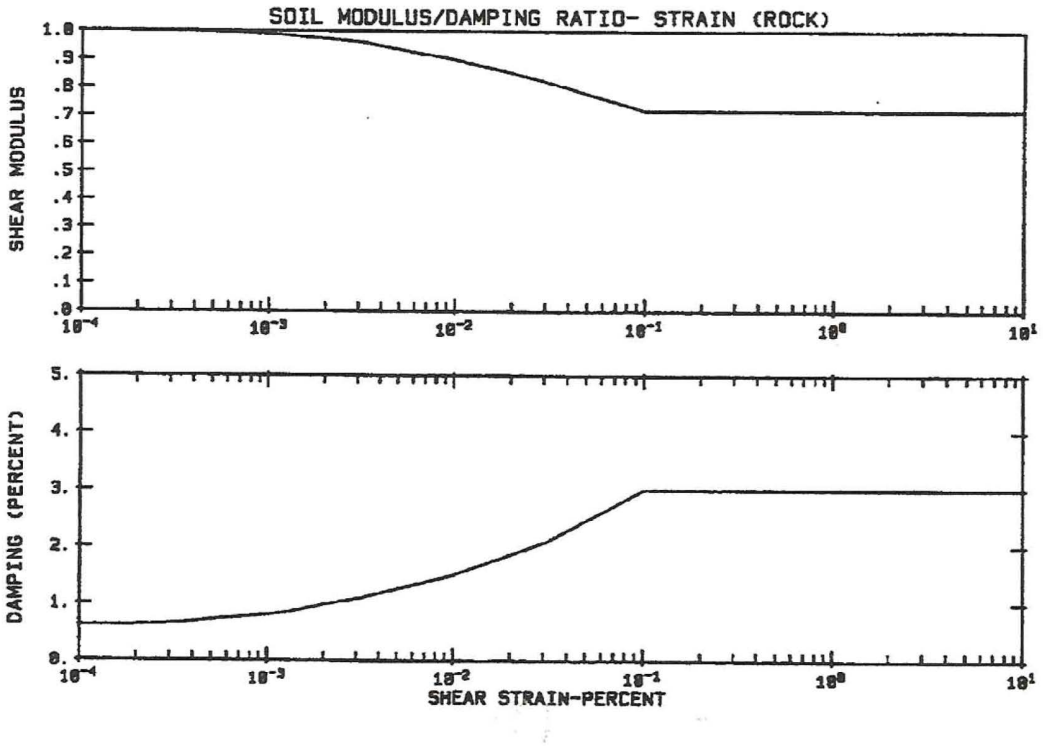

Modulus Reduction Curves for Generic Soil Sites

Figure 3.7.1-15
Strain Dependent Properties of Rock Material (Ref. 37)

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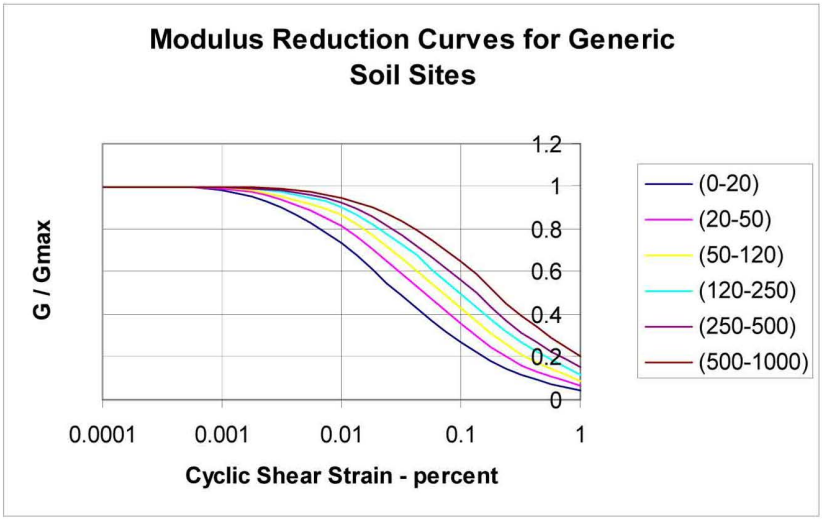

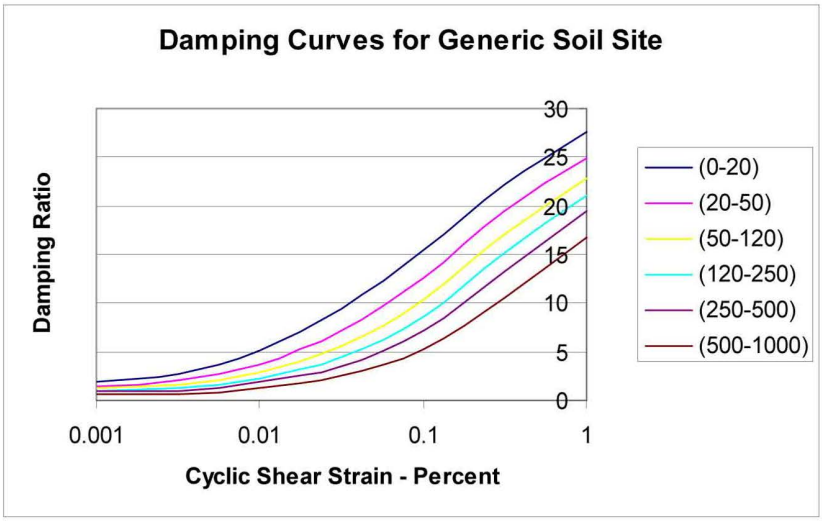

Figure 3.7.1-16
Strain Dependent Properties of Soil Material (Ref. 38)

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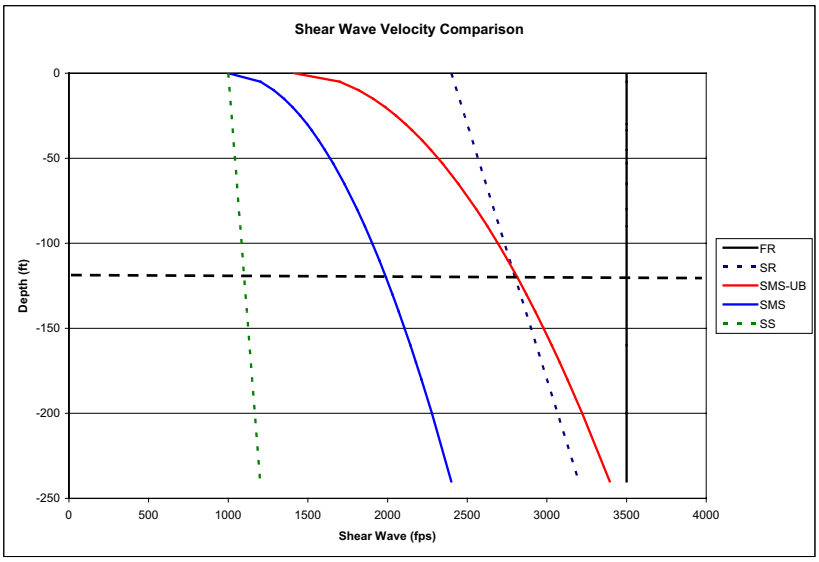
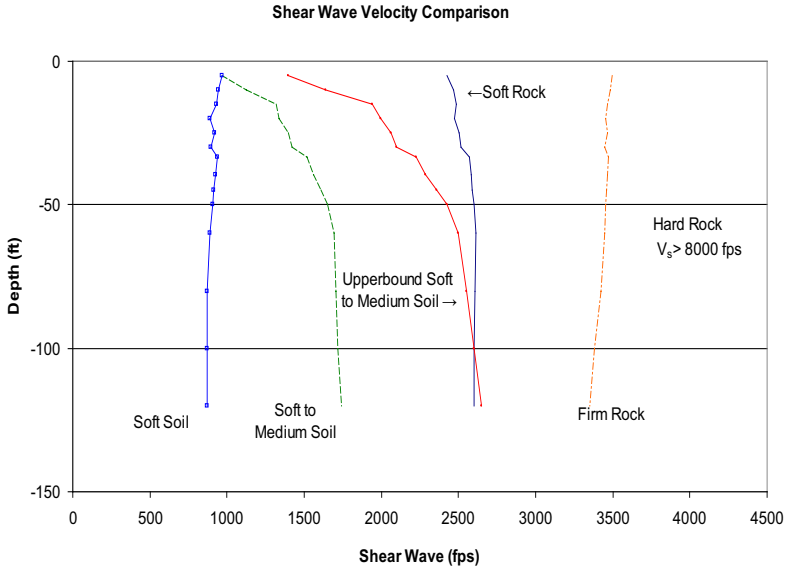


Initial Properties



Strain-Iterated Shear Wave Velocity Profiles

Note: Fixed base analyses were performed for hard rock sites. These analyses are applicable for shear wave velocity greater than 8000 feet per second.

Figure 3.7.1-17
Generic Soil Profiles

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Figures 3.7.2-1–3.7.2-11 Not Used

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Figure 3.7.2-12 (Sheet 1 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 66'-6"*]*

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Figure 3.7.2-12 (Sheet 2 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 82'-6"*]*

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Figure 3.7.2-12 (Sheet 3 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 100'-0" & 107'-2"*]*

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Figure 3.7.2-12 (Sheet 4 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 117'-6"*]*

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Figure 3.7.2-12 (Sheet 5 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 135'-3"*]*

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Figure 3.7.2-12 (Sheet 6 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 153'-0" & 160'-6"*]*

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Figure 3.7.2-12 (Sheet 7 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 160'-6", 180'-0", & 329'-0"*]*

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Figure 3.7.2-12 (Sheet 8 of 12)
**[*Nuclear Island Key Structural Dimensions*
Section A - A*]

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Figure 3.7.2-12 (Sheet 9 of 12)
**[*Nuclear Island Key Structural Dimensions*
Section B - B*]

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Figure 3.7.2-12 (Sheet 10 of 12)
[*Nuclear Island Key Structural Dimensions Sections C - C and H - H*]*

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Figure 3.7.2-12 (Sheet 11 of 12)
**[*Nuclear Island Key Structural Dimensions*
Section G - G*]

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Figure 3.7.2-12 (Sheet 12 of 12)
**[*Nuclear Island Key Structural Dimensions*
Section J - J*]

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Figure 3.7.2-13 Not Used

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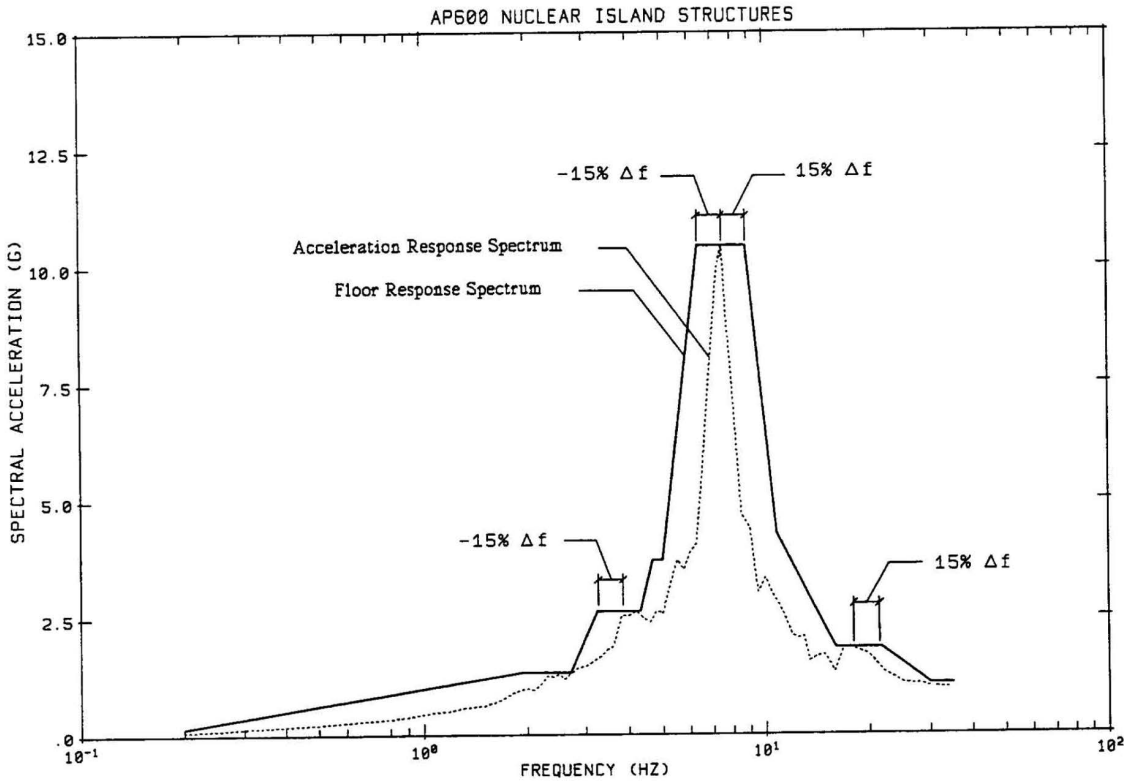

Figure 3.7.2-14
Typical Design Floor Response Spectrum

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Figures 3.7.2-15–3.7.2-18 Not Used

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Figure 3.7.2-19 (Sheet 1 of 10)
Annex Building Key Structural Dimensions
Plan at Elevation 100′-0″

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Figure 3.7.2-19 (Sheet 2 of 10)
Annex Building Key Structural Dimensions
Plan at Elevation 107′-2″ and 117′-6″

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Figure 3.7.2-19 (Sheet 3 of 10)
Annex Building Key Structural Dimensions
Plan at Elevation 135′-3″

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Figure 3.7.2-19 (Sheet 4 of 10)
Annex Building Key Structural Dimensions
Plan at Elevation 158′-0″ and 150′-3″

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Figure 3.7.2-19 (Sheet 5 of 10)
Annex Building Key Structural Dimensions
Roof Plan at Elevation 154′-0″ and 181′-11 3/4″

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Figure 3.7.2-19 (Sheet 6 of 10)
Annex Building Key Structural Dimensions
Section A - A

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Figure 3.7.2-19 (Sheet 7 of 10)
Annex Building Key Structural Dimensions
Section B - B

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Figure 3.7.2-19 (Sheet 8 of 10)
Annex Building Key Structural Dimensions
Section C - C

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Figure 3.7.2-19 (Sheet 9 of 10)
Annex Building Key Structural Dimensions
Sections D - D, E - E, & F - F

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Figure 3.7.2-19 (Sheet 10 of 10)
Annex Building Key Structural Dimensions
Sections G - G, H - H, & J - J

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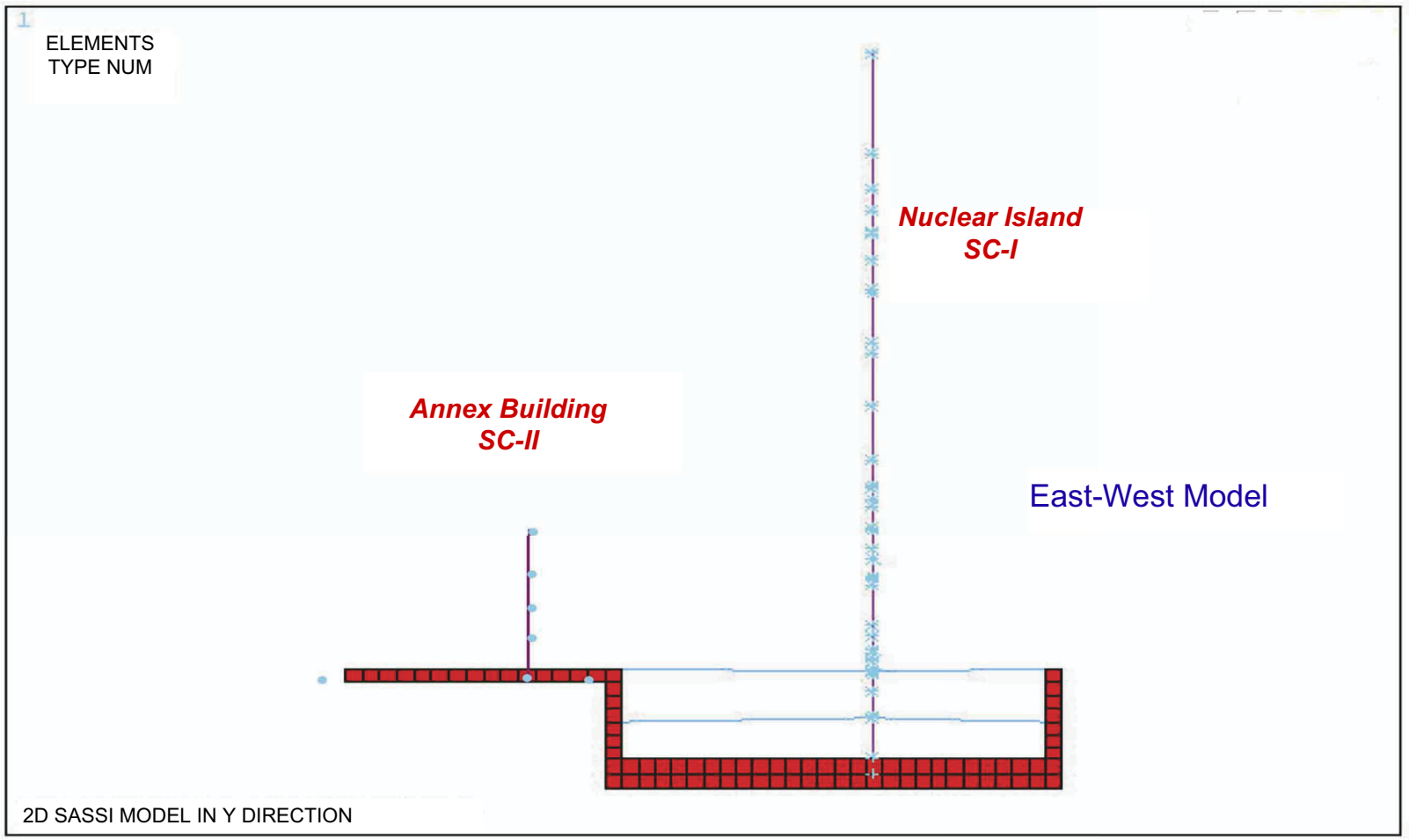

Figure 3.7.2-20
East-West 2D SASSI Model with Adjacent Buildings

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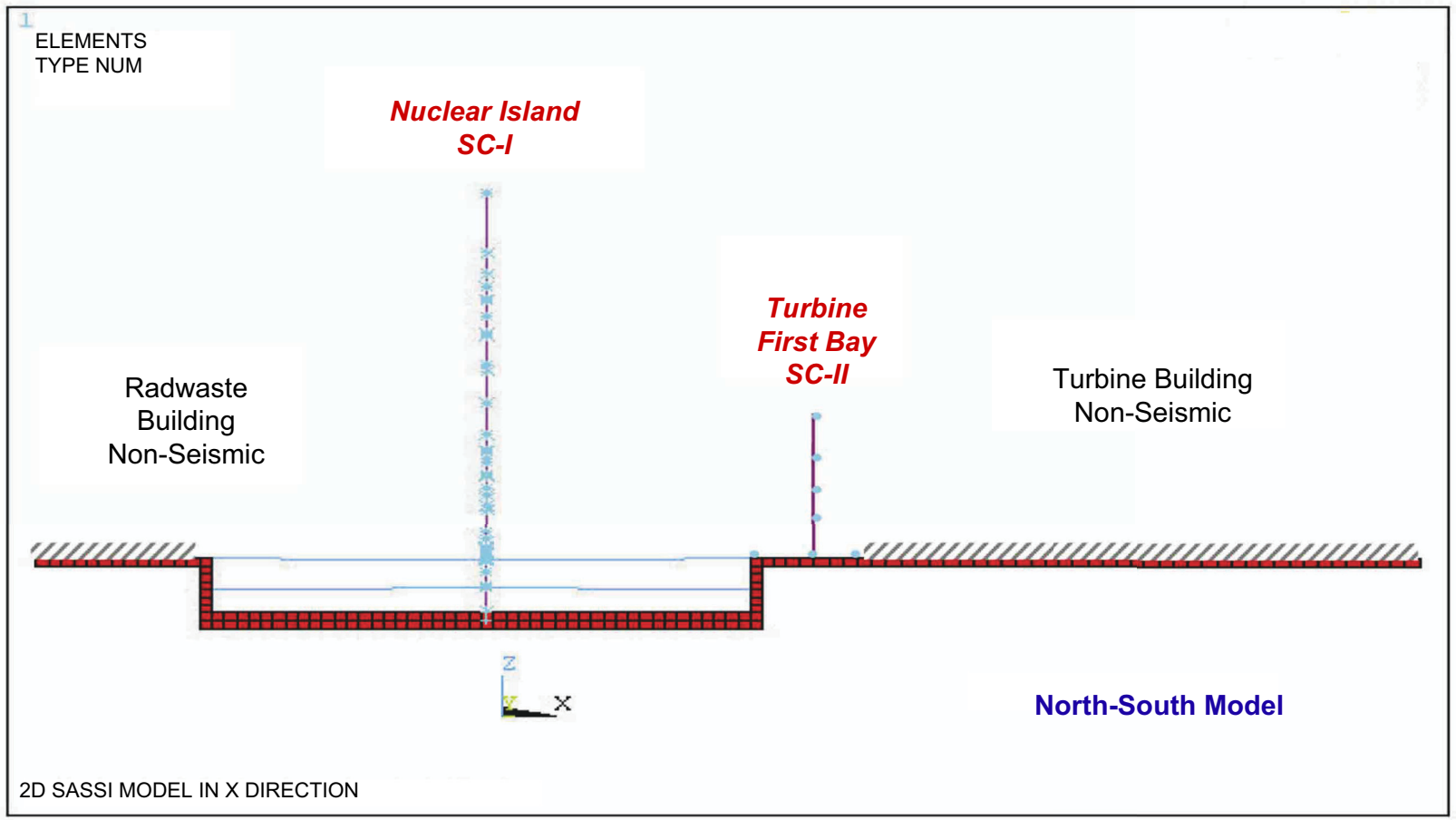

Figure 3.7.2-21
2D North-South SASSI Model with Adjacent Buildings

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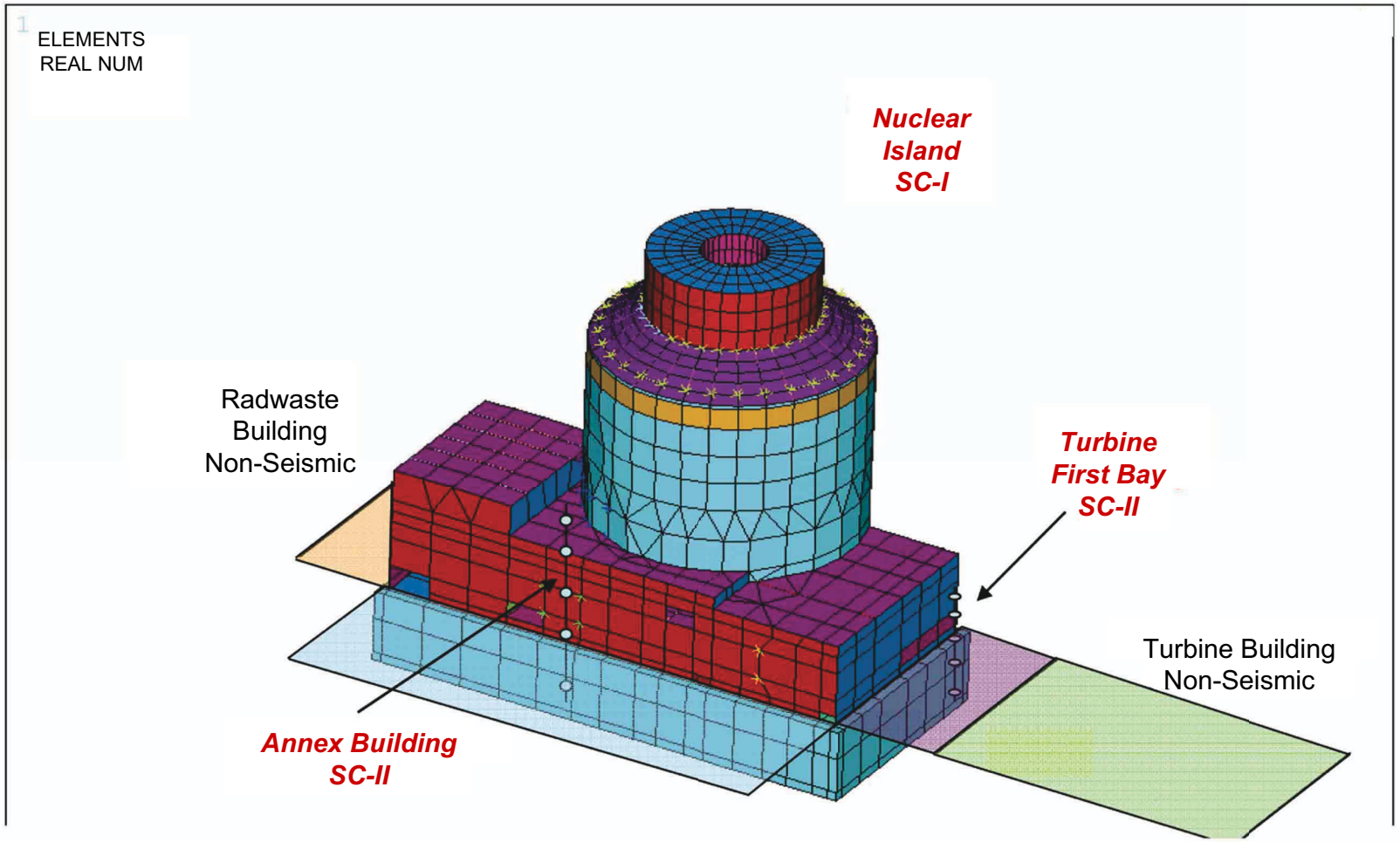

Figure 3.7.2-22
3D SASSI Model with Adjacent Buildings

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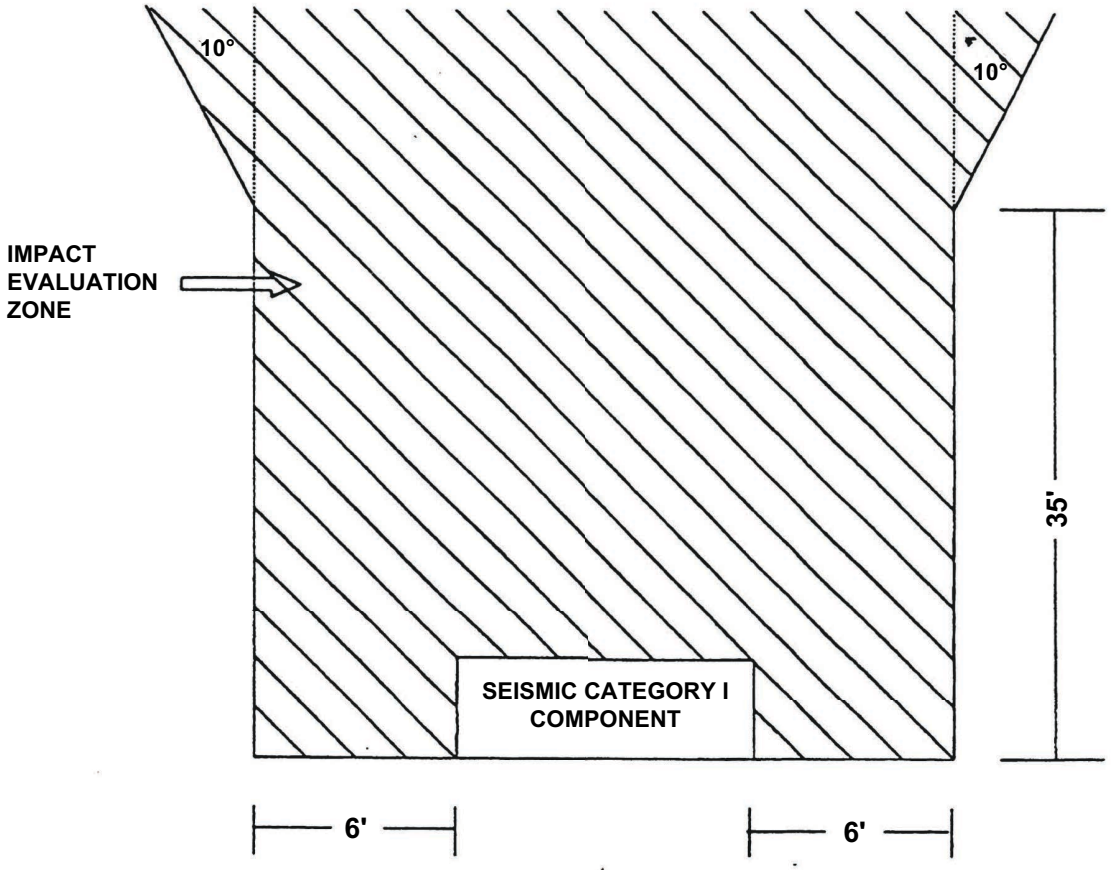

Figure 3.7.3-1
Impact Evaluation Zone

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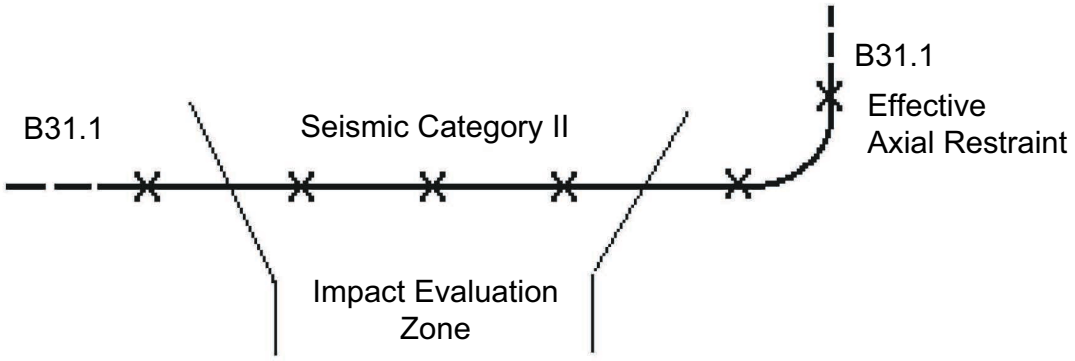

Figure 3.7.3-2
Impact Evaluation Zone and Seismic Supported Piping

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3.8 Design of Category I Structures

3.8.1 Concrete Containment

This subsection is not applicable to the AP1000.

3.8.2 Steel Containment

3.8.2.1 Description of the Containment

3.8.2.1.1 General

This subsection describes the structural design of the steel containment vessel and its parts and appurtenances. The steel containment vessel is an integral part of the containment system whose function is described in Section 6.2. It serves both to limit releases in the event of an accident and to provide the safety-related ultimate heat sink.

The containment vessel is an ASME metal containment. The information contained in this subsection is based on the design specification and preliminary design and analyses of the vessel. Final detailed analyses will be documented in the ASME Design Report.

The containment arrangement is indicated in the general arrangement figures in Section 1.2. The portion of the vessel above elevation 132′-3″ is surrounded by the shield building but is exposed to ambient conditions as part of the passive cooling flow path. A flexible watertight and airtight seal is provided at elevation 132′-3″ between the containment vessel and the shield building. The portion of the vessel below elevation 132′-3″ is fully enclosed within the shield building.

Figure 3.8.2-1 shows the containment vessel outline. It is a free-standing, cylindrical steel vessel with ellipsoidal upper and lower heads. [*The containment vessel has the following design characteristics:*

Diameter: 130 feet

Height: 215 feet 4 inches

Design Code: ASME III, Div. 1

Material: SA738, Grade B

Design Pressure: 59 psig

Design Temperature: 300°F

Design External Pressure: 1.7 psid

Lower Personnel Airlock: Elevation 110′-6″ and 107 degrees azimuth

Lower Equipment Hatch: Elevation 112′-6″ and 126 degrees azimuth

Upper Personnel Airlock Elevation 138′-7″ and 107 degrees azimuth

Upper Equipment Hatch Elevation 141′-6″ and 67 degrees azimuth

External Stiffener: Elevation 131′-9″

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Internal Stiffener: Elevation 170′-0″

Bottom Head Tangent Line Elevation 104′-1 1/2″

Upper Head Tangent Line Elevation 244′-2 1/2″

The tangent line is the elevation at which the vessel transitions from the cylinder to the head.

The wall thickness in most of the cylinder is 1.75 inches. The wall thickness of the lowest course of the cylindrical shell is increased to 1.875 inches to provide margin in the event of corrosion in the embedment transition region. The thickness of the heads is 1.625 inches.]* The heads are [*ellipsoidal*]* with a major diameter of 130 feet and a height of 37 feet, 7.5 inches.

The containment vessel includes the shell, hoop stiffeners and crane girder, equipment hatches, personnel airlocks, penetration assemblies, and miscellaneous appurtenances and attachments. The design for external pressure is dependent on the spacing of the hoop stiffeners and crane girder, which are shown on Figure 3.8.2-1. [*The spacing between each pair of ring supports (the bottom flange of the crane girder, the hoop stiffeners, and the concrete floor at elevation 100′-0″) is less than 50 feet, 6 inches. The design of the stiffeners and polar crane girder provides equal or greater radial and rotational stiffness than the design evaluated for the design certification.*]*

The polar crane is designed for handling the reactor vessel head during normal refueling. The crane girder and wheel assemblies are designed to support a special trolley to be installed in the event of steam generator replacement.

The containment vessel supports most of the containment air baffle as described in Subsection 3.8.4. The air baffle is arranged to permit inspection of the exterior surface of the containment vessel. Steel plates are welded to the dome as part of the water distribution system, described in Subsection 6.2.2. The polar crane system is described in Subsection 9.1.5.

3.8.2.1.2 Containment Vessel Support

The bottom head is embedded in concrete, with concrete up to elevation 100′ on the outside and to the maintenance floor at elevation 107′-2″ on the inside. The containment vessel is assumed as an independent, free-standing structure above elevation 100′. The thickness of the lower head is the same as that of the upper head. There is no reduction in shell thickness even though credit could be taken for the concrete encasement of the lower head.

Vertical and lateral loads on the containment vessel and internal structures are transferred to the basemat below the vessel by shear studs, friction, and bearing. The shear studs are not required for design basis loads. They provide additional margin for earthquakes beyond the safe shutdown earthquake.

Seals are provided at the top of the concrete on the inside and outside of the vessel to prevent moisture between the vessel and concrete. A typical cross section design of the seal is presented in Figure 3.8.2-8, sheets 1 and 2.

3.8.2.1.3 Equipment Hatches

Two equipment hatches are provided. One is at the operating floor (elevation 135′-3″) with an inside diameter of 16 feet. The other is at floor elevation 107′-2″ to permit grade-level access into the containment, with an inside diameter of 16 feet. The hatches, shown in Figure 3.8.2-2, consist of a cylindrical sleeve with a pressure seated dished head bolted on the inside of the vessel. The containment internal pressure acts on the convex face of the dished head and the head is in

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compression. The flanged joint has double O-ring or gum-drop seals with an annular space that may be pressurized for leak testing the seals. Each of the two equipment hatches is provided with an electrically powered hoist and with a set of hardware, tools, equipment and a self-contained power source for moving the hatch from its storage location and installing it in the opening. [*The information in Figure 3.8.2-2 that is considered to be Tier 2* information is the minimum thickness of the hatch cover, the inside diameter of the sleeve, the diameter of the insert plate, the minimum thickness of the insert plate, and the nominal spherical radius of the hatch cover.*]*

3.8.2.1.4 Personnel Airlocks

Two personnel airlocks are provided, one located adjacent to each of the equipment hatches. Figure 3.8.2-3 shows the typical arrangement. Each personnel airlock has about a 10-foot external diameter to accommodate a door opening of width 3 feet 6 inches and height 6 feet 8 inches. The airlocks are long enough to provide a clear distance of 8 feet, which is not impaired by the swing of the doors within the lock. The airlocks extend radially out from the containment vessel through the shield building. They are supported by the containment vessel. [*Area reinforcement for the personnel airlocks is provided by a minimum of 3-3/4-inch-thick insert plates. The surface area of the personnel airlock insert plate, not including the sleeve, is a minimum of 51.9 ft^2.*]*

Each airlock has two double-gasketed, pressure-seated doors in series. The doors are mechanically interlocked to prevent simultaneous opening of both doors and to allow one door to be completely closed before the second door can be opened. The interlock can be bypassed by using special tools and procedures.

3.8.2.1.5 Mechanical Penetrations

The mechanical penetrations consist of the fuel transfer penetration and mechanical piping penetrations and are listed in Table 6.2.3-1. Area is added to the shell by the addition of an insert plate that is thicker than the shell or by increasing the thickness of the nozzle neck or a combination of both. This piping penetration design is then evaluated for external loads on the penetration imposed by the piping system.

Design requirements for the mechanical penetrations are as follows:

- Design and construction of the process piping follow ASME, Section III, Subsection NC. Design and construction of the remaining portions follow ASME Code, Section III, Subsection NE. The boundary of jurisdiction is according to ASME Code, Section III, Subsection NE.

- Penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation.

- Guard pipes are designed for pipe ruptures as described in Subsection 3.6.2.1.1.4.

- Bellows are stainless steel or nickel alloy and are designed to accommodate axial and lateral displacements between the piping and the containment vessel. These displacements include thermal growth of the main steam and feedwater piping during plant operation, relative seismic movements, and containment accident and testing conditions. Cover plates are provided to protect the bellows from foreign objects during construction and operation. The cover plates for the main steam bellows are removable to permit in-service inspection. Access to permit in-service inspection of the pressure retaining portions of the main feedwater bellows is provided via holes in the weld ends of the bellows assembly.

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Figure 3.8.2-4, sheet 1, shows typical details for the main steam penetration. This includes bellows to minimize piping loads applied to the containment vessel and a guardpipe to protect the bellows and to prevent overpressurization of the containment annulus in a postulated pipe rupture event. Details for the main feedwater penetration are shown in Figure 3.8.2-4, Sheet 8. [*The main steam and feedwater penetrations are combined into a common 3-3/4-inch-thick insert plate. This thickness is a minimum value. The main steam penetration has an inside sleeve diameter of 57 inches. The feedwater penetration has an inside sleeve diameter of 38 inches.*]* The insert plates for the main steam and feedwater penetrations are shown in Figure 3.8.2-4, Sheet 7. The insert plate also includes the penetration for the 6-inch-diameter startup feedwater pipe. The insert plate is designed in accordance with NE-3330, “Openings and Reinforcement,” of the ASME Code.

Figure 3.8.2-4, sheet 2, shows typical details for the startup feedwater penetration. This includes a guardpipe to prevent overpressurization of the containment annulus in a postulated pipe rupture event. Similar details are used for the steam generator blowdown penetration.

Figure 3.8.2-4, sheet 3, shows typical details for the normal residual heat removal penetration. Similar details are used for other penetrations below elevation 107′-2″ where there is concrete inside the containment vessel. The flued head is integral with the process piping and is welded to the containment sleeve. The welds are accessible for in-service inspection. The containment sleeve is separated from the concrete by compressible material.

Figure 3.8.2-4, sheet 4 shows representative details for the other mechanical penetrations. These consist of:

- A sleeve welded to containment with flued head welded to the sleeve (detail A).
- A sleeve welded to containment with the process pipe welded directly to the sleeve and in direct contact with the process fluid (detail B).
- A sleeve welded to containment with process pipe passing through the sleeve (detail C).
- A sleeve welded to a thicker insert place welded to containment with sleeve welded to pipe and in contact with process fluid (detail D).

Flued heads are used for stainless piping greater than 2 inches in nominal diameter and for piping with high operating temperatures.

Note: The type of weld connecting the sleeve to pipe may vary.

The fuel transfer penetration, shown in Figure 3.8.2-4, sheet 5, is provided to transfer fuel between the containment and the fuel handling area of the auxiliary building. The fuel transfer tube is welded to the penetration sleeve. The containment boundary is a double-gasketed blind flange at the refueling canal end. The expansion bellows are not a part of the containment boundary. Rather, they are water seals during refueling operations and accommodate differential movement between the containment vessel, containment internal structures, and the auxiliary building.

3.8.2.1.6 Electrical Penetrations

Figure 3.8.2-4, sheet 6, shows a typical 18-inch-diameter electrical penetration. The penetration assemblies consist of conductor modules (or medium voltage cable modules in a similar 18-inch-diameter penetration) passing through a bulkhead attached to the containment nozzle. Electrical design of these penetrations is described in Subsection 8.3.1.1.6.

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Electrical penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation. Double barriers permit testing of each assembly to verify that containment integrity is maintained. Design and testing is according to IEEE Standard 317-83 and IEEE Standard 323-74.

3.8.2.1.7 Instrument Line Penetrations

Instrument line penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation.

Figure 3.8.2-4, sheet 4, detail B, shows typical details for the containment pressure instrumentation penetrations. The penetrations consist of sleeves welded to the containment vessel. Pressure transmitters outside containment are connected to pressure sensors inside containment by sealed, fluid-filled tubing (capillary), which passes through the sleeves. The capillary tubing is welded directly to the sleeve at a tubing coupling, which has a thicker wall and larger diameter than the capillary tubing.

Design and construction of the penetrations are in accordance with ASME Section III. The penetration sleeves, including the welds to the tubing couplings, follow ASME Section III, Subsection NE. Because ASME Section III, Subsection NCA excludes the sealed-tubing instrument configuration from the scope of Section III, the capillary tubing is designed and fabricated in accordance with ASME B31.1.

3.8.2.2 Applicable Codes, Standards, and Specifications

[*The containment vessel is designed*]* and constructed [*according to the 2001 edition of the ASME Code, Section III, Subsection NE, Metal Containment, including the 2002 Addenda. Stability of the containment vessel and appurtenances is evaluated using ASME Code, Case N-284-1, Metal Containment Shell Buckling Design Methods, Class MC, Section III, Division 1, as published in the 2001 Code Cases, 2001 Edition, July 1, 2001.*]*

Structural steel nonpressure parts, such as ladders, walkways, and handrails are designed to the requirements for steel structures defined in Subsection 3.8.4.

Section 1.9 discusses compliance with the Regulatory Guides and the Standard Review Plans.

3.8.2.3 Loads and Load Combinations

Table 3.8.2-1 summarizes the design loads, load combinations and ASME Service Levels. They meet the requirements of the ASME Code, Section III, Subsection NE. The loads and load combinations used in the analysis are considered to be part of the method of evaluation. The containment vessel is designed for the following loads specified during construction, test, normal plant operation and shutdown, and during accident conditions:

D Dead loads or their related internal moments and forces, including any permanent piping and equipment loads

L Live loads or their related internal moments and forces, including crane loads

P_o Operating pressure loads during normal operating conditions resulting from pressure variations either inside or outside containment

T_o Thermal effects and loads during normal operating conditions, based on the most critical transient or steady-state condition

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R_o Piping and equipment reactions during normal operating conditions, based on the most critical transient or steady-state condition

W Loads generated by the design wind on the portion of the containment vessel above elevation 132′, as described in Subsection 3.3.1.1

E_s Loads generated by the safe shutdown earthquake (SSE) as described in Section 3.7

W_t Loads generated by the design tornado on the portion of the containment vessel above elevation 132′, as described in Subsection 3.3.2

P_t Test pressure

P_d Containment vessel design pressure that exceeds the pressure load generated by the postulated pipebreak accidents and passive cooling function

P_e Containment vessel external pressure

T_a Thermal loads under thermal conditions generated by the postulated break or passive cooling function and including T_o. This includes variations around the shell due to the surrounding buildings and maldistribution of the passive containment cooling system water.

R_a Piping and equipment reactions under thermal conditions generated by the postulated break, as described in Section 3.6, and including R_o

Y_r Loads generated by the reaction on the broken high-energy pipe during the postulated break, as described in Section 3.6

Y_j Jet impingement load on a structure generated by the postulated break, as described in Section 3.6

Y_m Missile impact load on a structure generated by or during the postulated break, as from pipe whipping, as described in Section 3.6

Post-accident flooding load combination is not applicable in the design of the AP1000 containment vessel. The post-loss-of-coolant accident (LOCA) flooding event is enveloped by the other design cases.

The AP1000 addresses the production of large quantities of hydrogen from the oxidation of zirconium and other metals as a result of a postulated severe accident. The AP1000 includes hydrogen igniters inside containment to ensure that hydrogen generated in a severe accident is burned prior to reaching an explosive mixture. The discussion of the generation and burning of hydrogen as a result of a severe accident is included in Section 19.18.

The containment vessel is protected from the direct effects of wind/tornado loads (and associated potential missiles) by virtue of its location inside the shield building. The differential pressure effects of a tornado are also reduced because of the location and are bounded by other pressure loadings for which the containment vessel is designed.

The containment is evaluated for the deterministic severe accident pressure capacity. This evaluation is discussed in Subsection 3.8.2.4.2, “Evaluation of Ultimate Capacity.” According to 10 CFR 50.44, the hydrogen generated pressure loads from 100 percent fuel clad-coolant reaction plus the peak pressure from a hydrogen burn must be less than ASME Service Level C (not including buckling). The Service Level C maximum capacity is 117 psig at 300°F as presented in Subsection 3.8.2.4.2.8

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The peak pressure from the 100 percent fuel clad-coolant reaction plus the hydrogen burn (Pg1 + Pg2) is 90.3 psig as reported in Section 41.11 and Table 41-4 of the AP1000 Probabilistic Risk Assessment report. The severe accident conditions are beyond design basis accidents, and the load combinations for these severe accident evaluations are not included in the load combinations and service limits for the containment vessel.

The AP1000 does not have a post-accident inerting system. Therefore, there is no load combination that includes inerting of the containment.

Note that loads associated with flooding of the containment below elevation 107′ are resisted by the concrete structures and not by the containment vessel.

3.8.2.4 Design and Analysis Procedures

The design and analysis procedures for the containment vessel are according to the requirements of the ASME Code, Section III, Subsection NE.

The design certification analyses are summarized in Table 3.8.2-4. The detailed analyses used a series of general-purpose finite element, axisymmetric shell and special purpose computer codes to conduct such analyses. Code development, verification, validation, configuration control, and error reporting and resolution are according to the Quality Assurance requirements of Chapter 17.

Subsections 3.8.2.4.1 through 3.8.2.4.2.8 describe containment vessel (CV) design and analysis procedures to support design certification of the AP1000.

3.8.2.4.1 Analyses for Design Conditions

3.8.2.4.1.1 Axisymmetric Shell Analyses

The containment vessel is modelled as an axisymmetric shell and analyzed using the ANSYS computer program. A model used for static analyses is shown in Figure 3.8.2-6.

Dynamic analyses of the axisymmetric model, which is similar to that shown in Figure 3.8.2-6, are performed to obtain frequencies and mode shapes. These are used to confirm the adequacy of the containment vessel stick model as described in Subsection 3.7.2.3.2. Stress analyses are performed for each of the following loads:

- Dead load
- Internal pressure
- Seismic
- Polar crane wheel loads
- Wind loads
- Thermal loads

The seismic analysis performed envelopes all soil conditions. The global seismic loads are applied as equivalent static accelerations using the maximum accelerations shown in Table 3.8.2-5. These accelerations are the maximum accelerations from the nuclear island stick model on hard rock. The global member forces from the equivalent static case exceed those from the soil cases for soil conditions described in Appendix 3G. Based on these comparisons, the design acceleration values used for the global analyses are appropriate for both the hard rock and the soil sites. The seismic analysis of the nuclear island is discussed in Section 3.7 and Appendix 3G. The torsional moments, which include the effects of the eccentric masses, are increased to account for accidental torsion and are evaluated in a separate calculation.

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The results of these load cases are factored and combined in accordance with the load combinations identified in Table 3.8.2-1. These results are used to evaluate the general shell away from local penetrations and attachments, that is, for areas of the shell represented by the axisymmetric geometry. The results for the polar crane wheel loads are also used to establish local shell stiffnesses for inclusion in the containment vessel stick model described in Subsection 3.7.2.3. The results of the analyses and evaluations are included in the containment vessel design report.

Design of the containment shell is primarily controlled by the internal pressure of 59 psig. The meridional and circumferential stresses for the internal pressure case are shown in Figure 3.8.2-5. The most highly stressed regions for this load case are the portions of the shell away from the hoop stiffeners and the knuckle region of the top head. In these regions the stress intensity is close to the allowable for the design condition.

Table 3.8.2-1 includes a design load combination to address external pressure. For the design external pressure, a conservatively large magnitude of 1.7 psi differential pressure is used. Design external pressure is defined as a value greater than the external pressure at which the vacuum relief system will open and mitigate the external pressure. This is a part of the containment air filtration system (see Subsection 9.4.7). Upon actuation, the external pressure transient is immediately controlled and the external pressure is relieved. This design external pressure is combined with a coincident -40°F outside air temperature, which corresponds to a -18.5°F metal temperature for the portions of the containment vessel shell not insulated from ambient conditions. The portions of the containment vessel shell that are below the external stiffener are insulated from the cold outside air conditions and result in a metal temperature of 70°F.

A bounding case was analyzed to provide an indication of the margin to acceptance criteria associated with the minimum allowable service metal temperature for the AP1000 containment vessel. Various types of transients were considered to evaluate the minimum service metal temperature, including inadvertent fan cooler cases, inadvertent passive containment cooling system (PCS) actuation, and loss of ac power. The evaluation considered variations in initial conditions for parameters, including humidity, internal temperature, external temperature, and wind speed. These evaluations demonstrate that the -18.5°F service metal temperature is adequate.

Design external pressure is used in load combinations that include thermal loads and are used to evaluate Service Level A and D stress limits. These external pressure conditions are included in the loading combinations in Table 3.8.2-1.

Operating pressures range from -0.2 psig to 1.0 psig, which are then combined with an ambient temperature for the containment vessel. Design internal pressure is 59 psig combined with a containment vessel metal temperature of 300°F to be evaluated in the ASME service limits as well as the design conditions.

A load combination that combines design wind plus internal design pressure is not included in Table 3.8.2-1 because the wind loads are small (within the normal operating range for containment pressure) and because the combination of the design wind and accident pressure is a lower probability than either the design wind or the accident pressure acting alone.

Major loads that induce compressive stresses in the containment vessel are internal and external pressure and crane and seismic loads. Each of these loads and the evaluation of the compressive stresses are discussed below.

- Internal pressure causes compressive stresses in the knuckle region of the top head and in the equipment hatch covers. The evaluation methods are similar to those discussed in Subsection 3.8.2.4.2 for the ultimate capacity.

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- Evaluation of external pressure loads is performed in accordance with ASME Code, Section III, Subsection NE, Paragraph NE-3133.

- Crane wheel loads due to crane dead load, live load, and seismic loads result in local compressive stresses in the vicinity of the crane girder. These are evaluated in accordance with ASME Code, Case N-284.

- Overall seismic loads result in axial compression and tangential shear stresses at the base of the cylindrical portion. These are evaluated in accordance with ASME Code, Case N-284.

The bottom head is embedded in the concrete base at elevation 100 feet. This leads to circumferential compressive stresses at the discontinuity under thermal loading associated with the design basis accident. The containment vessel design includes a Service Level A combination in which the vessel above elevation 107′-2″ is specified at the design temperature of 300°F and the portion of the embedded vessel (and concrete) below elevation 100 feet is specified at a temperature of 70°F. The temperature profile for the vessel is linear between these elevations. Containment shell buckling close to the base is evaluated against the criteria of ASME Code, Case N-284.

Revision 1 of Code Case N-284 is used for the evaluation of the containment shell and equipment hatches.

3.8.2.4.1.2 Local Analyses

The penetrations and penetration reinforcements are designed in accordance with the rules of ASME III, Subsection NE. The design of the large penetrations for the two equipment hatches and the two airlocks use the results of finite element analyses which consider the effect of the penetration and its dynamic response (Reference 53).

The personnel airlocks and equipment hatches are modeled in a 3-D shell finite element model of the containment. A 3D shell, finite element model of the containment vessel was developed in ANSYS to consider the effect of the penetrations and their quasi-static response due to a seismic event. The large masses and local stiffness of the personnel locks and equipment hatches are discretely modeled. The polar crane wheel loads are incorporated by appropriate loadings (dead load and seismic loadings). The bottom of the model is fixed at elevation 100′ where the containment vessel is embedded in concrete. This means that rotations and displacements are conservatively fixed at this location.

Static analyses are performed using the finite element model shown in Figure 3.8.2-7 for internal pressure, dead load (including the polar crane in the parked position), thermal loads and seismic loads. The global seismic loads are applied as equivalent static accelerations using the maximum accelerations shown in Table 3.8.2-5. The amplified local responses are included separately for each of the four penetrations. Local seismic axial and rotational accelerations about both horizontal and vertical axes are applied based on the maximum amplified response determined from a time history analysis on a less refined dynamic model with seismic time histories at elevation 100′.

Stresses are evaluated against the stress intensity criteria of ASME Section III, Subsection NE for the load combinations described in Table 3.8.2-1. Stability is evaluated against ASME Code Case N-284. Local stresses in the regions adjacent to the major penetrations are evaluated in accordance with paragraph 1700 of the code case. Stability is not evaluated in the reinforced penetration neck and insert plate which are substantially stiffer than the adjacent shell.

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3.8.2.4.2 Evaluation of Ultimate Capacity

The capacity of the containment vessel has been calculated for internal pressure loads for use in the probabilistic risk assessment analyses and severe accident evaluations. These analyses include the evaluation of the peak pressure from the hydrogen-generated pressure loads from 100-percent fuel cladding metal-water reaction plus the hydrogen burn. Each element of the containment vessel boundary was evaluated to estimate the maximum pressure at an ambient temperature of 100°F corresponding to the following stress and buckling criteria:

- Deterministic severe accident pressure capacity corresponding to ASME Service Level C limits on stress intensity, ASME paragraph NE-3222, and ASME Code Case N-284 for buckling of the equipment hatch covers, and 60 percent of critical buckling for the top head. The deterministic severe accident pressure capacity corresponds to the approach in SECY 93-087, to maintain a reliable leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges. This approach was approved by the Nuclear Regulatory Commission as outlined in the Staff Requirements Memorandum on SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs, Dated July 21, 1993.

- Best estimate capacity corresponding to gross membrane yield at the ASME-specified minimum yield stress (SA738, Grade B, yield stress = 60 ksi, ultimate stress = 85 ksi), and critical buckling for the equipment hatch covers and top head.

The results are shown in Table 3.8.2-2. The analyses at a temperature of 100°F are described in the following paragraphs for each element. The critical regions identified in this table are then examined further for their response at higher temperatures. This results in the best-estimate capacity based on the ASME-specified minimum yield properties. The evaluation considered the containment boundary elements including:

- Cylindrical shell
- Top and bottom heads
- Equipment hatches and covers
- Personnel airlocks
- Mechanical and electrical penetrations

The evaluation identified the most likely failure mode to be that associated with gross yield of the cylindrical shell. Loss of containment function would be expected to occur because the large post-yield deflections would lead to local failures at penetrations, bellows, or other local discontinuities.

3.8.2.4.2.1 Tensile Stress Evaluation of Shell

Results of the axisymmetric analyses of the cylinder and top head described in Subsection 3.8.2.4.1 for dead load and internal pressure were evaluated to determine the pressure at which stresses reach yield at an ambient temperature of 100°F. The analyses assume the shell is fixed at elevation 100′, where the bottom head is embedded in concrete. The steel bottom head is identical to the top head and has a pressure capability greater than the top head due to the additional strength of the embedment concrete.

The allowable stress intensity under Service Level C loads is equal to yield. This corresponds to an internal pressure of 135 psig. The critical section is the cylinder, where the general primary membrane stress intensity is greatest.

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The best-estimate yield analysis uses the von Mises criterion to establish yield rather than the more conservative ASME stress intensity approach. This increases the yield stress by about 15 percent for the cylinder, where the longitudinal stress is equal to one-half of the hoop stress resulting in first yield at an internal pressure of 155 psig. At this pressure, hoop stresses in the cylinder reach yield. The radial deflection is about 1.6 inches. As pressure increases further, large deflections occur. For a material such as SA738, where the yield plateau extends from a strain of 0.2 percent to 0.6 percent, deflections would increase to 4.8 inches at yield without a substantial increase in pressure. Strain hardening would then permit a further increase in pressure with large radial deflections, as described in Subsection 3.8.2.4.2.6.

3.8.2.4.2.2 Buckling Evaluation of Top Head

The top head has a radius-to-height ratio of 1.728. This is not as shallow as most ellipsoidal or torispherical heads, which typically have a radius-to-height ratio of 2. The ratio was specifically selected to minimize the local stresses and buckling in the knuckle region due to internal pressure. As the ratio decreases, the magnitude of compressive stresses in the knuckle region decreases; for a radius-to-height ratio of 1.4 or smaller, there are no compressive stresses and therefore there is no potential for buckling.

Theoretical Buckling Capacity

The top head was analyzed using the BOSOR-5 computer code (Reference 1). This code permits consideration of both large displacements and nonlinear material properties. It calculates shell stresses and checks stability at each load step. The analysis included a portion of the cylinder with a thickness of 1.625 inches. In this analysis, yield of the cylinder started at a pressure of 144 psig using elastic – perfectly plastic material properties, a yield stress of 60 ksi, and the von Mises yield criterion. Yield of the top of the crown started at an internal pressure of 146 psig. Yield of the knuckle region started at 152 psig. A theoretical plastic buckling pressure of 174 psig was determined. At this pressure, the maximum effective prebuckling strain was 0.23 percent in the knuckle region where buckling occurred and 2.5 percent at the crown. The maximum deflection at the crown was 15.9 inches. A similar analysis was performed using nonlinear material properties considering the effects of residual stresses; buckling did not occur in this analysis, and failure would occur once strains at the crown reach ultimate. The failure mode was found to be an axisymmetric plastic collapse resulting from excessive vertical displacements at the crown. The maximum displacement was 43 inches at 195 psig.

Predicted Pressure Capacity

The actual buckling capacity may be lower than the theoretical buckling capacity because of effects not included in the analysis such as imperfections and residual stresses. This is considered by the use of capacity reduction factors that are based upon a correlation of theory and experiment. The capacity reduction factor for the top head was evaluated based on comparisons of BOSOR-5 analyses against test results of ellipsoidal and torispherical heads. This evaluation is described below and concludes that no reduction in capacity need be considered; that is, a capacity reduction factor of 1.0 is appropriate.

The knuckle region of ellipsoidal and torispherical heads is subjected to meridional tension and circumferential compression. The meridional tension tends to stabilize the knuckle region and reduces its sensitivity to imperfection. The radius-to-height ratio of 1.728 of the AP1000 head results in a larger ratio of meridional tension to circumferential compression than on shallower heads, further reducing the sensitivity to imperfection.

Welding Research Council Bulletin 267 (Reference 22) shows a comparison of BOSOR-5 predictions of buckling against the results of 20 tests of small head models. These results are summarized in Table 4 of the reference and show ratios (capacity reduction factors) of actual buckling to the

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BOSOR-5 prediction with an average of 1.2. Only one of the 20 cases shows a capacity reduction factor less than 1.0.

Table 3.8.2-3 shows the key parameters, test results, and BOSOR-5 predictions for two large, fabricated 2:1 torispherical heads tested and reported in NUREG/CR-4926 (Reference 23). The theoretical plastic buckling pressure predicted by BOSOR-5 represents initial buckling based on actual material properties. The initial buckling did not cause failure for either of the tests, and test pressure continued to increase until rupture occurred in the spherical cap. The collapse pressures were three to four times the initial buckling pressures.

- **Test Head 1** – The test result of 58 psig is 79 percent of the predicted theoretical plastic buckling pressure of 74 psig. Many of the buckles occurred directly on the meridional weld seams of the knuckle. The knuckle welds were noticeably flatter than the corresponding welds of the Test 2 head. The as-built configuration extended inside the theoretical shape at some of the meridional weld seams and was most pronounced at the location of the first observed buckle. Model 1 exceeded the tolerances for formed heads specified for containment vessels in NE-4222.2 of ASME, Section III, Subsection NE.

- **Test Head 2** – The test result of 106 psi is 100 percent of the BOSOR-5 predicted theoretical plastic buckling pressure. For test head 2, the welds had no noticeable flat spots and there was a smooth transition between the sphere and knuckle sections. Test head 2 was well within the Code allowable deviations.

The low-capacity reduction factor of 0.79 for test head 1 is attributed to excessive imperfections associated with the fabrication of relatively thin plate (0.196 inch). These imperfections were visible and were outside the tolerances permitted by the ASME Code. The results of test head 1 are therefore not considered applicable to the AP1000. The results of test head 2 and of the small-scale models described in the Welding Research Council Bulletin support the application of a capacity reduction factor of 1.0.

The capacity of the AP1000 head was also investigated using an approach similar to that permitted in ASME Code, Case N284. This code case provides alternate rules for certain containment vessel geometries such as cylindrical shells. The theoretical elastic buckling pressure was calculated to be 536 psi using the linear elastic computer code, BOSOR-4 (Reference 24). A reduction factor (defined as the product of the capacity reduction factor and the plastic reduction factor) was established as 0.385 based on the lower bound curve of test results of 20 ellipsoidal and 28 torispherical test specimens, which also include the two large fabricated heads previously discussed. This resulted in a predicted buckling capacity of 206 psig.

The preceding paragraphs addressed incipient buckling. It is concluded that buckling would not occur prior to reaching the pressure of 174 psig predicted in the BOSOR-5 analyses. Tests indicate that pressure can be significantly increased prior to rupture after the formation of the initial buckles. Failure would occur when local strains reach ultimate either close to a local buckle in the knuckle or at the center of the crown. The best estimate capacity of the head is taken as the theoretical plastic buckling pressure of 174 psig predicted in the BOSOR-5 analyses.

The deterministic severe accident pressure capacity is taken as 60 percent of critical buckling. This is consistent with the safety factor for Service Level C in ASME Code, Case N-284 and results in a containment head capacity of 104 psig.

3.8.2.4.2.3 Equipment Hatches

SECY 93-087 permits evaluation of certain severe accident scenarios against ASME Service Level C limits. The equipment hatch covers were evaluated for buckling against ASME paragraph NE-3222

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and according to ASME Code, Case N-284. Use of ASME Code, Case N-284 for this application was confirmed to be appropriate by ASME. The containment internal pressure acts on the convex face of the dished head and the hatch covers are in compression under containment internal pressure loads. The critical buckling capacity is based on classical buckling capacities reduced by capacity reduction factors to account for the effects of imperfections and plasticity. These capacity reduction factors are based on test data and are generally lower-bound values for the tolerances specified in the ASME Code.

The critical buckling pressure is 211 psig for the 16-foot-diameter hatch at an ambient temperature of 100°F. For the Service Level C limits in accordance with paragraph NE 3222, a safety factor of 2.50 is specified, resulting in capabilities of 84 psig (16-foot-diameter). For the Service Level C limits in accordance with Code Case N284, a safety factor of 1.67 is specified, resulting in capabilities of 126 psig (16-foot-diameter).

Typical gaskets have been tested for severe accident conditions as described in NUREG/CR-5096 (Reference 25). The gaskets for the AP1000 will be similar to those tested with material such as Presray EPDM E 603. For such gaskets the onset of leakage occurred at a temperature of about 600°F.

3.8.2.4.2.4 Personnel Airlocks

The capacity of the personnel airlocks was determined by comparing the airlock design to that tested and reported in NUREG/CR-5118 (Reference 3). Critical parameters are the same, so the results of the test apply directly. In the tests the inner door and end bulkhead of the airlock withstood a maximum pressure of 300 psig at 400°F. The capacity of the airlock is therefore at least 300 psig at ambient temperature. The maximum pressure corresponding to Service Level C is conservatively estimated by reducing this capacity in the ratio of the minimum specified material yield to ultimate.

3.8.2.4.2.5 Mechanical and Electrical Penetrations

Subsections 3.8.2.1.3 through 3.8.2.1.7 describe the containment penetrations. Penetration reinforcement is designed following the area replacement method of the ASME Code. The insert plates and sleeves permit development of the hoop tensile yield stresses predicted as the limiting capacity in Subsection 3.8.2.4.1. Capacities of the equipment hatch covers are discussed in Subsection 3.8.2.4.2.3 and of the personnel airlocks in Subsection 3.8.2.4.2.4.

Mechanical penetrations welded directly to the containment vessel are generally piping systems with design pressures greater than that of the containment vessel. Thicknesses of the flued head or end plate are established based on piping support loads or stiffness requirements. The capacities of these penetrations are greater than the capacity of the containment vessel cylinder.

Mechanical penetrations for the large-diameter high-energy lines, such as the main steam and feedwater piping, include expansion bellows. The piping and flued head have large pressure capability. The response of expansion bellows to severe pressure and deformations is described in NUREG/CR-5561 (Reference 4). The bellows can withstand large pressure loading but may tear once the containment vessel deflection becomes large. Testing reported in NUREG/CR-6154 (Reference 26) has shown that the bellows remain leaktight even when subjected to large deflections sufficient to fully compress the bellows. Such large deflections do not occur as long as the containment vessel remains elastic. As described in Subsection 3.8.2.4.2.6, the radial deflection of the shell increases substantially once the containment cylinder yields. The resulting deflections are assumed to cause loss of containment function. The containment penetration bellows are designed for a pressure of 90 psig at design temperature within Service Level C limits, concurrent with the relative displacements imposed on the bellows when the containment vessel is pressurized to these magnitudes.

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Electrical penetrations have a pressure boundary consisting of the sleeve and an end plate containing a series of modules. The electrical pressure boundary is designed and built to the requirements of the ASME Code, Section III, Class MC, Subsection NE. The pressure capacity of these elements is large and is greater than the capacity of the containment vessel cylinder at temperatures up to the containment design temperature. Electrical penetration assemblies are also designed to satisfy ASME Service Level C stress limits under a pressure of 90 psig at design temperature. Tests at pressures and temperatures representative of severe accident conditions are described in NUREG/CR-5334 (Reference 5), where typical nuclear industry penetrations were irradiated, aged, then tested. One design was tested to 135 psia at 700°F. Other electrical penetration assemblies were tested to 75 psia at 400°F and 155 psia at 361°F. These tests showed that the electrical penetration assemblies withstand severe accident conditions. The electrical penetration assemblies are qualified for the containment design basis event conditions as described in Appendix 3D. The assemblies are similar to one of those tested by Sandia as reported in NUREG/CR-5334 (Reference 5). The ultimate pressure capacity of the electrical penetration assemblies is primarily determined by the temperature. The maximum temperature of the containment vessel below the operating deck during a severe accident is below the temperature at which the assemblies from the three suppliers in the Sandia tests were tested.

3.8.2.4.2.6 Material Properties

The containment vessel is designed using SA738, Grade B material. This has a specified minimum yield of 60 ksi and ultimate of 85 ksi. Test data for materials having similar chemical properties were reviewed. In a sample of 122 tests for thicknesses equaling or exceeding 1.50 inches and less than 1.75 inches, the actual yield had a mean value of 69.1 ksi with a standard deviation of 3.3 ksi. Thus, the actual yield is expected to be about 15 percent higher than the minimum yield. Membrane yield of the cylinder is predicted to occur at an internal pressure of 178 psig.

A stress-strain curve for material with chemistry similar to SA738, Grade B, indicated constant yield stress of 81.3 ksi from a strain of 0.002 to 0.006 followed by strain-hardening up to a maximum stress of 94.5 ksi at a strain of 0.079. The first portion of the strain-hardening is nearly linear, with a stress of 90 ksi at a strain of 4 percent. This strain occurs at a stress 10 percent above yield. Thus, a pressure load 10 percent higher than that corresponding to yield of the shell would result in 4 percent strain and a 31-inch radial deflection of the containment cylinder. Such a deflection is expected to cause major distress for penetrations, the air flow path, and local areas where other structures are close to the containment vessel. Loss of function is therefore assumed for the containment once gross yield of the containment cylinder occurs.

3.8.2.4.2.7 Effect of Temperature

The evaluations described in the preceding subsections are based on an ambient temperature of 100°F. Nonmetallic items, such as gaskets, are qualified to function at the design temperature. The capacity of steel elements is reduced in proportion to the reduction due to temperature in yield stress, ultimate stress, or elastic modulus. The cylinder is governed by yield stress, and elastic buckling of the hatch covers is governed by the elastic modulus. The reduction in capacity is estimated using the tables given for material properties in the ASME Code. At 400°F, the yield stress is reduced by 17 percent and the pressure capacity corresponding to gross yield is reduced from 155 to 129 psig.

3.8.2.4.2.8 Summary of Containment Pressure Capacity

The ultimate pressure capacity for containment function is expected to be associated with leakage caused by excessive radial deflection of the containment cylindrical shell. This radial deflection causes distress to the mechanical penetrations, and leakage would be expected at the expansion bellows for the main steam and feedwater piping. There is high confidence that this failure would not occur before stresses in the shell reach the minimum specified material yield. This is calculated to

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occur at a pressure of 155 psig at ambient temperature and 129 psig at 400°F. Failure would be more likely to occur at a pressure about 15 percent higher based on expected actual material properties.

The deterministic severe accident pressure that can be accommodated according to the ASME Service Level C stress intensity limits and using a factor of safety of 1.67 for buckling of the top head is determined by the capacity of the 16-foot-diameter equipment hatch cover and the ellipsoidal head. The maximum capacity of the hatch cover, calculated according to ASME paragraph NE-3222, Service Level C, is 84 psig at an ambient temperature of 100°F and 81 psig at 300°F. When calculated in accordance with ASME Code, Case N-284, Service Level C, the maximum capacity is 126 psig at an ambient temperature of 100°F and 121 psig at 300°F. The maximum capacity of the ellipsoidal head is 104 psig at 100°F and 91 psig at 300°F.

The maximum pressure that can be accommodated according to the ASME Service Level C stress intensity limits, excluding evaluation of instability, is determined by yield of the cylinder and is 135 psig at an ambient temperature of 100°F and 117 psig at 300°F. This limit is used in the evaluations required by 10 CFR 50.44.

3.8.2.5 Structural Criteria

The containment vessel is designed, fabricated, installed, and tested according to the ASME Code, Section III, Subsection NE, and will receive a code stamp.

Stress intensity limits are according to ASME Code, Section III, Paragraph NE-3221 and Table NE-3221-1. [*Critical buckling stresses are checked according to the provisions of ASME Code, Section III, Paragraph NE-3222, or ASME Code Case N-284.*]*

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

Materials for the containment vessel, including the equipment hatches, personnel locks, penetrations, attachments, and appurtenances meet the requirements of NE-2000 of the ASME Code. [*The basic containment material is SA-738, Grade B, plate. The procurement specification for the SA-738, Grade B, plate includes SA-20 supplemental requirements S1, Vacuum Treatment and S20, Maximum Carbon Equivalent for Weldability.*]* This material has been selected to satisfy the lowest service metal temperature requirement of -18.5°F. This temperature is established by analysis for the portion of the vessel exposed to the environment when the minimum ambient air temperature is -40°F. Impact test requirements are as specified in NE-2000.

[*The material of construction for the insert plates and fabricated nozzle necks of penetrations is SA-738 Grade B. The material of construction for forged nozzle neck forgings is SA-350, LF2, Class 1 for penetrations greater than 2 inches nominal diameter and less than 24 inches inside diameter, and for the containment air filtration system penetration nozzle necks. The maximum carbon equivalent for the SA-350, LF2, Class 1 used for penetrations is 0.52 percent. A vacuum refining process is required for SA-350, LF2, Class 1. The SA-350, LF2, Class 1 material is used in portions of the containment vessel that are below the external stiffener.*]* These portions of the containment vessel are insulated from the cold outside air temperature. Insulation is provided around the upper equipment hatch and personnel airlock, including the insert plates, in order to insulate the penetrations from the ambient air temperature in the upper annulus.

The containment vessel is coated with an inorganic zinc coating, except for those portions fully embedded in concrete. The inside of the vessel below the operating floor also has an epoxy top coat. Below elevation 100′ the vessel is fully embedded in concrete with the exception of the few penetrations at low elevations (see Figure 3.8.2-4, sheet 3, for typical details). Embedding the steel vessel in concrete protects the steel from corrosion.

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The AP1000 configuration is shown in the general arrangement figures in Section 1.2 and in Figure 3.8.2-1. The exterior of the vessel is embedded at elevation 100′ and concrete is placed against the inside of the vessel up to the maintenance floor at elevation 107′-2″. Above this elevation the inside and outside of the containment vessel are accessible for inspection of the coating. The vessel is coated with an inorganic zinc primer to a level just below the concrete.

Seals are provided at the surface of the concrete inside and outside the vessel so that moisture is not trapped next to the steel vessel just below the top of the concrete. The seal on the inside accommodates radial growth of the vessel due to pressurization and heatup.

The plate thickness for the first course (elevation 104′-1.5″ to minimum 109′-0″) of the cylinder is 1.875 inches, which is 1/8-inch thicker than the rest of the vessel. This provides margin in the event there would be any corrosion in the transition region despite the coatings and seals described previously. Equivalent margin is available for the 1.625-inch-thick bottom head in the transition region (elevation 100′ to 104′-1.5″). The plate thickness for the head is a constant thickness and is established by the stresses in the knuckle. As a result, the pressure stresses in the transition zone are well below the allowable stress, providing margin in the event of corrosion in this region.

The quality control program involving welding procedures, erection tolerances, and nondestructive examination of shop- and field-fabricated welds conforms with Subsections NE-4000 and NE-5000 of the ASME Code. The containment vessel is designed to permit its construction using large subassemblies. These subassemblies consist of the two heads and three or more ring sections. These are assembled in an area near the final location, using plates fabricated in a shop facility.

3.8.2.7 Testing and In-Service Inspection Requirements

Testing of the containment vessel and the pipe assemblies forming the pressure boundary within the containment vessel will be according to the provisions of NE-6000 and NC-6000, respectively.

Subsection 6.2.5 describes leak-rate testing of the containment system including the containment vessel.

In-service inspection of the containment vessel will be performed. See Section 6.6 for information on inservice inspection for the containment vessel and penetrations.

3.8.3 Concrete and Steel Internal Structures of Steel Containment

3.8.3.1 Description of the Containment Internal Structures

The containment internal structures are those concrete and steel structures inside (not part of) the containment pressure boundary that support the reactor coolant system components and related piping systems and equipment. The concrete and steel structures also provide radiation shielding. The containment internal structures are shown on the general arrangement drawings in Section 1.2. The containment internal structures consist of the primary shield wall, reactor cavity, secondary shield walls, in-containment refueling water storage tank (IRWST), refueling cavity walls, operating floor, intermediate floors, and various platforms. The polar crane girders are considered part of the containment vessel. They are described in Subsection 3.8.2.

Component supports are those steel members designed to transmit loads from the reactor coolant system to the load-carrying building structures. The component configuration is described in this subsection including the local building structure backing up the component support. The design and construction of the component supports are described in Subsection 5.4.10.

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The containment internal structures are designed using reinforced concrete and structural steel. At the lower elevations conventional concrete and reinforcing steel are used, except that permanent steel forms are used in some areas in lieu of removable forms based on constructibility considerations. These steel form modules (liners) consist of plate reinforced with angle stiffeners and tee sections, as shown in Figure 3.8.3-16. The angles and the tee sections are on the concrete side of the plate. Welded studs, or similar embedded steel elements, are attached on the concrete face of the permanent steel form where surface attachments transfer loads into the concrete. Where these surface attachments are seismic Category I, the portion of the steel form module transferring the load into the concrete is classified as seismic Category I.

Walls and floors are concrete filled steel plate structural modules. The walls are supported on the mass concrete containment internal structures basemat with the steel surface plate extending down to the concrete floor on each side of the wall. The steel surface plates of the structural modules provide reinforcement in the concrete. The structural modules are anchored to the base concrete by mechanical connections welded to the steel plate as shown in Figure 3.8.3-8, Sheet 2. Figure 3.8.3-1 shows the location of the structural modules. Figures 3.8.3-2 and 3.8.3-15 show the typical structural configuration of the wall modules. Key structural elements of the module design are identified as Tier 2* information in the text and figures of this section. See DCD Introduction, Section 3.5 for a discussion of Tier 2* information. [*The information in Figure 3.8.3-2 that is Tier 2* is the minimum size of the angles and channels used to fabricate the modules. The information in Figure 3.8.3-15 that is Tier 2* is the design spacing between the face plates for the 4-foot-thick refueling canal wall in the containment internal structures and the design spacing between the trusses used to fabricate the modules in locations away from openings or penetrations in the wall. Local variation in the design of the trusses and spacing of the trusses and shear studs may be required to address internal obstructions and accessibility for fabrication and inspection. The obstructions include features such as leak chases; internal structures such as reinforcements, embedments, and backup structures; and internal conduit and piping.*]* See Subsection 3.8.3.5.3.6 for criteria for design of the shear stud size and spacing.

A representative floor module is shown in Figure 3.8.3-3 and also in Figure 3.8.3-16 combined with the liner module. These structural modules are structural elements built up with welded steel structural shapes and plates. The details shown in Figure 3.8.3-3 are a representative design for a specific portion of the operating deck floor in containment. The size, material, and configuration of the structural shapes and plates, the amount and arrangement of the reinforcement, and the type and size of the welds may vary in the final design of floor modules in this and other locations. Concrete is used where required for shielding, but reinforcing steel is not normally used.

Walls and floors exposed to water during normal operation or refueling are constructed using stainless steel plates.

3.8.3.1.1 Reactor Coolant Loop Supports

3.8.3.1.1.1 Reactor Vessel Support System

The reactor vessel is supported by four supports located under the cold legs, which are spaced 90 degrees apart in the primary shield wall. The supports are designed to provide for radial thermal growth of the reactor coolant system, including the reactor vessel, but they prevent the vessel from lateral and torsional movement. The ends of the supports are bolted to steel embedments in the concrete wall surrounding the reactor vessel. Each support connects to the embedments with eight 2-3/8-inch-diameter dowel studs. In addition, the center portion of the support is attached to the concrete with eight threaded anchor bolts embedded in the primary shield wall concrete. The reactor vessel supports loads are carried by the dowel studs to embedded steel structures and the anchor bolts to the concrete. Figure 3.8.3-4 shows the reactor vessel supports.

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3.8.3.1.1.2 Steam Generator Support System

The steam generator vertical support consists of a single vertical column extending from the steam generator compartment floor to the bottom of the steam generator channel head. The column is constructed of heavy plate sections and is pinned at both ends to permit unrestricted radial displacement of the steam generator during plant heatup and cooldown. The location of this column is such that it will allow full access to the steam generator for routine maintenance activities. It is located a sufficient distance away from the reactor coolant pump motors to permit pump maintenance and in-service inspection.

The lower steam generator horizontal support is located at the top of the vertical column. It consists of a tension/compression link bar and extension bar oriented approximately perpendicular to the hot leg. The link bar and extension bar is pinned at both the wall bracket and the vertical column to permit movement of the generator during plant heatup and cooldown.

The upper steam generator horizontal support in the direction of the hot leg is located on the upper shell just above the transition cone. It consists of two large hydraulic snubbers oriented parallel with the hot leg centerline. One snubber is mounted on each side of the generator on top of the steam generator compartment wall. The hydraulic snubbers are valved to permit steam generator movement for thermal transition conditions, and to "lock-up" and act as rigid struts under dynamic loads.

The upper steam generator horizontal support in the direction normal to the hot leg is located on the lower shell just below the transition cone. It consists of two rigid tension/compression struts oriented perpendicular to the hot leg. The two rigid struts are mounted on the steam generator compartment wall at the elevation of the operating deck. The steam generator loads are transferred to the struts and snubbers through trunnions on the generator shell. Figure 3.8.3-5 shows the steam generator supports.

The steam generator supports are anchored using anchor bolts or steel weldments embedded in the concrete, designed in accordance with Appendix B of ACI 349. The lower portion of the column pedestal, embedded in the concrete, as shown on sheet 1 of Figure 3.8.3-5, transfers the vertical load into the reinforced concrete basemat. The lower and intermediate horizontal supports are located so that the loads are transferred into the plane of the adjacent floor. The upper supports are located so that the loads are transferred into the plane of the steam generator compartment walls.

3.8.3.1.1.3 Reactor Coolant Pump Support System

Because the reactor coolant pumps are integrated into the steam generator channel head, they do not have individual supports. They are supported by the steam generators.

3.8.3.1.1.4 Pressurizer Support System

The pressurizer is supported by four columns mounted from the pressurizer compartment floor. A lateral support is provided at the top of the columns. This lateral support consists of eight struts connecting it to the pressurizer compartment walls. A lateral support is also provided on the upper portion of the pressurizer. This lateral support consists of a ring girder around the pressurizer and eight struts connecting it to the pressurizer compartment walls. Figure 3.8.3-6 shows the pressurizer supports.

3.8.3.1.2 Containment Internal Structures Basemat

The containment internal structures basemat is the reinforced concrete structure filling the bottom head of the containment vessel. It extends from the bottom of the containment vessel head at

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elevation 66′-6″ up to the bottom of the structural modules that start between elevations 71′-6″ and 103′-0″. The basemat includes rooms as shown on Figure 1.2-5. The primary shield wall and reactor cavity extend from elevation 71′-6″ to elevation 107′-2″. They provide support for the reactor vessel and portions of the secondary shield walls and refueling cavity walls. The general arrangement drawings in Section 1.2 show the location and configuration of the primary shield wall and reactor cavity. The walls of the primary shield, the steam generator compartment and the CVS room are structural modules as shown in Figure 3.8.3-1. The rest of the basemat is reinforced concrete.

3.8.3.1.3 Structural Wall Modules

Structural wall modules are used for the primary shield wall around the reactor vessel, the wall between the vertical access and the CVS room, secondary shield walls around the steam generators and pressurizer, for the east side of the in-containment refueling water storage tank, and for the refueling cavity. The general arrangement drawings in Section 1.2 show the location and configuration. Locations of the structural modules are shown in Figure 3.8.3-1. Isometric views of the structural modules are shown in Figure 3.8.3-14. The figure shows the use of modules, overall configuration, and general design approach for incorporation of structural modules into the structural design. Alternatives to the design details shown in Figure 3.8.3-14, including the size, location, and type of structural shapes attached to the faceplates and embedded in the concrete, may be used. The embedded lower portions of the modules in the concrete, attachments, and penetrations may also be added or differ from that shown in the figure. The secondary shield walls are a series of walls that, together with the refueling cavity wall, enclose the steam generators. Each of the two secondary shield wall compartments provides support and houses a steam generator and reactor coolant loop piping. The in-containment refueling water storage tank is approximately 30 feet high. The floor elevation of this tank is 103′-0″. The tank extends up to about elevation 133′-3″, directly below the operating deck. On the west side, along the containment vessel wall, the tank wall consists of a stainless steel plate stiffened with structural steel sections in the vertical direction and angles in the horizontal direction. Structural steel modules, filled with concrete and forming, in part, the refueling cavity, steam generator compartment, and pressurizer compartment walls, compose the east wall. The refueling cavity has two floor elevations. The area around the reactor vessel flange is at elevation 107′-2″. The lower level is at elevation 98′-1″. The upper and lower reactor internals storage is at the lower elevation, as is the fuel transfer tube. The center line of the fuel transfer tube is at elevation 100′-5.75″.

Structural wall modules consist of steel faceplates connected by steel trusses. The primary purpose of the trusses is to stiffen and hold together the faceplates during handling, erection, and concrete placement. The nominal spacing of the trusses is 30 inches. The nominal thickness of the steel faceplates is 0.5 inch. [*In the CA01 and CA05 structural wall modules, the faceplate thickness is greater than the nominal thickness where required to support local demand. For areas subject to these high out-of-plane loads, the faceplate thickness may be increased up to a thickness of 1.5 inches. To support loading associated with the steam generator lateral supports, the faceplate thickness may be increased up to a thickness of 3.0 inches.*]* At corner locations, the trusses are replaced with diaphragms. These diaphragms have similar purpose as the trusses and are used to fabricate the modules in the corners. The detail design of the trusses and the design of the diaphragms shown in Figure 3.8.3-8, Sheet 1 are representative. Details such as the location, type, and size of welds and length and end configuration of channels used to assemble the internal trusses may vary in the design implemented from that shown in the figure. The size, shape, and location of welds, structural shapes, and other elements used to fabricate and stiffen the diaphragms and connect them to the faceplates and the location, size, and shape of the access opening may vary from that shown in the figure. Steel plates, structural shapes, reinforcement bars, or tie bars are installed between the structural wall module faceplates and embedded in the concrete as additional structural elements that support out-of-plane loads and structural integrity. This reinforcement is designed to the applicable requirements of ACI 349 and ANSI/AISC N690. Each module has different characteristics that require the detailed design to be specific for that module and loading and

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interferences within that module.Shear studs are welded to the inside faces of the steel faceplates. Face plates are welded to adjacent plates with full penetration welds so that the weld is at least as strong as the plate. The full penetration welds are identified in Figure 3.8.3-8, Sheet 1 with the weld symbol that includes the notation CJP for complete joint penetration. Plates on each face of the wall module extend down to the elevation of the adjacent floor. Where the floors in the rooms on each side of the wall module are at different elevations, one of the plates at the bottom of the module extends further down than the other. This single-sided configuration is designated on Figure 3.8.3-1 as “CA Structure Module with Single Surface Plate” for lower elevations in containment. A typical example with the single-sided configuration at the bottom of the module is shown in Figure 3.8.3-8. Some modules or portions of modules do not have the single-sided configuration at the bottom of the module. The module functions as a wall above the upper floor level (elevation 103′-0″ in Figure 3.8.3-8). The single plate below this elevation is designed to transfer the reactions at the base of the wall into the basemat. This plate also acts as face reinforcement for the basemat. Basemat reinforcement dowels are provided at the bottom of the single plate as shown in Figure 3.8.3-8.

[*Shear studs, structural shapes, or horizontally oriented reinforcement bars mechanically connected or welded to the module faceplates may be used to anchor the CA01 and CA05 structural modules to the base concrete. Shear lugs designed in accordance with ACI 349 may also be used to transfer loads at the connection. Where horizontal bars are attached perpendicular to the faceplate, they function like shear studs to provide for faceplate development. Where additional reinforcement is required for detailing base concrete at module walls with offsets in faceplate elevation, the reinforcement is developed in accordance with ACI 349. Where horizontal bars are attached to the edge of a faceplate at an elevation offset, the bars are designed for a combination of applicable demand and developed in accordance with ACI 349.*]*

[*The information in Figure 3.8.3-8, Sheet 1 that is considered to be Tier 2* information is the design spacing of the faceplates, trusses, channels in the trusses and the minimum size and design spacing of the headed studs for the modular wall in the containment internal structure in locations away from openings or penetrations in the walls. Local variation in the design of the trusses and spacing of the trusses and shear studs may be required to address internal obstructions and accessibility for fabrication and inspection. As shown in Figure 3.8.3-1, Sheets 2 and 4, some CA01 and CA05 wall thicknesses are different than the wall thicknesses listed in Figure 3.8.3-8. Wall thickness increase due to local faceplate thickness increase is below the level of detail depicted in Figure 3.8.3-1. The module wall thicknesses are determined based on radiation protection considerations and are designed to the applicable requirements of ANSI/AISC N690 and ACI 349. The use of full penetration welds to connect the faceplates of the modules is also considered to be Tier 2* information. The information in Figure 3.8.3-8, Sheet 2 that is considered to be Tier 2* information is the use of mechanical connectors and the development length requirement for the mechanical connectors.*]* The detail design of the mechanical connectors is governed by ANSI/AISC N690 and ACI 349, and the representative design shown is not considered to be Tier 2* information. The material, size, and configuration of plates, size and type of welds, method of attaching the connectors to the modules, and size of deformed bars anchoring the embedments into the concrete may be different for each module and may vary from that shown in the figure provided that the embedment design satisfies the ANSI/AISC N690 and ACI 349 codes. Sheet 3 of Figure 3.8.3-8 shows a wall of the IRWST that is a steel structure anchored in concrete and is not a concrete filled module. [*The information in Figure 3.8.3-8, Sheet 3 that is considered to be Tier 2* information is the plate thickness for the IRWST wall, the use, spacing, and size of angles and tees to stiffen the wall, the number, size, and use of studs provided to anchor the module, and the number, size, vertical spacing, and development length of the deformed bars provided to connect the module to the mass concrete. The design implemented in fabrication and construction drawings and instructions will have the design shown, an equal design, or a better design for the key structural elements.*]*

The structural wall modules are anchored to the concrete base by reinforcing steel dowels or other types of connections mechanically connected or welded to the modules and embedded in the

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reinforced concrete below or adjacent to the module. [*The reinforcing steel used to anchor the modules to the concrete may be oriented horizontally or vertically and has a development that satisfies the requirements of ACI 349.*]* After erection, concrete is placed between the faceplates. Typical design details of the structural modules are shown in Figures 3.8.3-2 and 3.8.3-8. Shear-friction reinforcement may be included in construction joints at the transition between the base concrete and concrete in the modules and is not part of the mechanical connection between the module and base concrete described above. The shear-friction reinforcement at the construction joints is spaced and sized in accordance with ACI 349 requirements and is not connected to the module faceplates.

Representative design details of the connections of floor modules to structural wall modules are shown in Figure 3.8.3-17. These connections connect the floor modules that make up the maintenance floor at elevation 107'-2" and the operating deck at elevation 135'-3". Similar connection designs are used within structural modules to connect floors to walls. The design details for the floor modules, including the size and spacing of the reinforcement, the use of shear studs, and the sizes of the structural shapes, are provided for background information and vary based on loading conditions and geometry. The seat angles, beam seats, shear plates, and other design elements supporting the floor modules and connecting the floors to the wall modules are sized to satisfy ANSI/AISC N690 criteria and requirements. The clip angles and shear plates shown in Figure 3.8.3-17 and other design elements are used as required at beam locations for the connection. The module faceplates in the area of the connection may be thicker than 0.5 inch, up to 1.5-inch thick, in some locations. The backup structure shown in the module wall is a representative design, and the details for the design implemented include variation of the size and thickness of the plates used in the backup structure. The designs of the backup structures satisfy ANSI/AISC N690 criteria and requirements. As an alternative, locally thicker faceplates or other design elements embedded in the wall and connected to the faceplate may be used to provide sufficient strength in the connection. Where the reinforcement is included as part of the connection design, the size, spacing, and length of the reinforcement satisfies ACI 349 requirements. The standard hooks in the wall module and the reinforcement bar connectors on the faceplates satisfy ACI 349 requirements.

3.8.3.1.4 Structural Floor Modules

Structural floor modules are used for the operating floor at elevation 135′-3″ over the in-containment refueling water storage tank, for the southeast quadrant of the operating floor and for the 107′-2″ floor over the rooms in the containment internal structures basemat. The floors are shown on the general arrangement drawings in Section 1.2. The floor modules consist of shear studs, steel tee, wide flange, and channel sections, welded to horizontal steel bottom plates stiffened by transverse stiffeners. Shear studs are welded on the top of the floor bottom plate. After erection, concrete is placed on top of the horizontal plate and around the structural steel section. The remaining region of the operating floor consists of a concrete slab, placed on Q decking supported by structural steel beams. The operating floor is supported by the in-containment refueling water storage tank walls, refueling cavity walls, the secondary shield walls, and steel columns originating at elevation 107′-2″. Structural details for a representative example of the containment operating deck floor structural module are shown in Figure 3.8.3-3. The design details for the floor module are provided for background information and vary based on loading conditions and geometry.

3.8.3.1.5 Internal Steel Framing

The region of the operating floor away from the in-containment refueling water storage tank consists of a concrete slab, placed on Q decking supported by structural steel beams. The floor at elevation 118′-6″ consists of steel grating supported by structural steel framing. In addition, a number of steel platforms are located above and below the operating floor. These platforms support either grating floors or equipment, such as piping and valves.

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3.8.3.2 Applicable Codes, Standards, and Specifications

The following documents are applicable to the design, materials, fabrication, construction, inspection, or testing of the containment internal structures:

[● *American Concrete Institute (ACI), Code Requirements for Nuclear Safety Related Concrete Structures, ACI 349-01*]* (refer to Subsection 3.8.4.5 for supplemental requirements)

● American Concrete Institute (ACI), ACI Detailing Manual

● American Concrete Institute (ACI), Standard Specifications for Tolerances for Concrete Construction and Materials, ACI-117

● American Concrete Institute (ACI), Guide to Formwork for Concrete, ACI-347

[● *American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, ANSI/AISC N690-1994*]* (refer to Subsection 3.8.4.5 for supplemental requirements)

[● *American Welding Society (AWS), Structural Welding Code - Steel, AWS D1.1-2000*

AWS D1.1-2000 is an acceptable code for use in lieu of AWS D1.1-1992 for the design, qualification, fabrication, and inspection for ANSI/AISC N690 applications.

The use of AWS D1.1-2000 to replace and supplement provisions in ANSI/AISC N690-1994 includes the provisions noted below:

– *The provisions in Section 2.3.4.1 of AWS D1.1-2000 are used to size complete joint penetration groove welds for tubular members.*

– *ASTM A992 is an alternative to ASTM A36 steel for structural steel shapes. Where ASTM A992 steel is used, the provisions for matching weld metal identified in AWS D1.1-2000, Section 3.3 and Table 3.1 are used to supplement the information in ANSI/AISC N690 Section Q1.4.4.*

– *Where directionality is applied to increase allowable weld stress for fillet welds identified in ANSI/AISC N690, Section Q1.5.3 and Table Q1.5.3, the provisions in Sections 2.14.4 and 2.14.5 of AWS D1.1-2000 shall be used.*

For fillet weld groups concentrically loaded in the plane of the axis of the weld and consisting of elements oriented both longitudinally and transversely to the direction of applied load, the deformation capability of the elements shall be considered. As a supplement to the provisions in AWS D1.1-2000, the strength of the weld group is limited to the larger of the two following evaluations:

1. *The combined strength of the individual weld elements using 1.0 times the strength of the longitudinally oriented element and 1.0 times the strength of the transversely oriented element*

2. *The combined strength of the individual weld elements using 0.825 times the strength of the longitudinally oriented element and 1.5 times the strength of the transversely oriented element*

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The provisions for directionality are not applied to fillet welds of tubular member T, Y, and K connections described in Table 2.5 of AWS D1.1-2000.

The provisions in AWS D1.1-2000 for the application of directionality to linear fillet welds are supplemented by extending the provisions to weld configurations not specified in AWS D1.1-2000 as noted below:

– *Where directionality is considered for concentric out-of-plane loading for linear fillet weld groups, including lines and rectangular configurations, the directionality provisions in AWS D1.1-2000 are applied.*

– *Where directionality is considered for concentric out-of-plane loading for circular-shaped fillet weld configurations, the directionality provisions in AWS D1.1-2000 are applied. The use of directionality provisions in AWS D1.1-2000 for circular-shaped fillet weld configurations is limited to tension loading concentrically applied out-of-plane to the axis of the weld.*]*

Nationally recognized industry standards, such as American Society for Testing and Materials, American Concrete Institute, and American Iron and Steel Institute, are used to specify material properties, testing procedures, fabrication, and construction methods. Section 1.9 describes conformance with the Regulatory Guides and the Standard Review Plans.

Welding and inspection activities for seismic Category I structural steel, including building structures, structural modules, cable tray supports, and heating, ventilating and air-conditioning (HVAC) duct supports are accomplished in accordance with written procedures and meet the requirements of the American Institute of Steel Construction (ANSI/AISC N690). The welded seams of the plates forming part of the leaktight boundary of the in-containment refueling water storage tank are examined by liquid penetrant after fabrication to confirm that the boundary does not leak.

3.8.3.3 Loads and Load Combinations

The loads and load combinations used in the analysis are considered to be part of the method of evaluation. The loads and load combinations for the containment internal structures are the same as for other Category I structures described in Subsection 3.8.4.3 and the associated tables, except for the following modifications:

Wind loads (W), tornado loads (W_t), and precipitation loads (N) are not applicable to the design of the containment internal structures because of the protection provided by the steel containment. Therefore, these loading terms have been excluded in the load combinations for the containment internal structures.

3.8.3.3.1 Passive Core Cooling System Loads

Structures are evaluated for pressure and thermal transients associated with operation of the passive core cooling system. The effects of temperatures higher than 100°F on the modulus of elasticity and yield strength of steel are considered.

The passive core cooling system and the automatic depressurization system (ADS) are described in Section 6.3. The automatic depressurization system is in part composed of two spargers that are submerged in the in-containment refueling water storage tank. The spargers provide a controlled distribution of steam flow to prevent imposing excessive dynamic loads on the tank structures. Capped vent pipes are installed in the roof of the tank on the side near the containment wall.

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These caps prevent debris from entering the tank from the containment operating deck, but they open under slight pressurization of the in-containment refueling water storage tank. This provides a path to vent steam released by the spargers. An overflow is provided from the in-containment refueling water storage tank to the refueling cavity to accommodate volume and mass increases during automatic depressurization system operation. Two sets of loads representing bounding operational or inadvertent transients are considered in the design of the in-containment refueling water storage tank.

- ADS_1 – This automatic depressurization system load is associated with blowdown of the primary system through the spargers when the water in the in-containment refueling water storage tank is cold and the tank is at ambient pressure. Dynamic loads on the in-containment refueling water storage tank due to automatic depressurization system operation are determined using the results from the automatic depressurization system hydraulic test as described in Subsection 3.8.3.4.2. The hydrodynamic analyses described in Subsection 3.8.3.4.2 show that member forces in the walls of the in-containment refueling water storage tank are bounded by a case with a uniform pressure of 5 psi applied to the walls. The in-containment refueling water storage tank is designed for a uniform pressure of 5 psi applied to the walls. This pressure is taken as both positive and negative due to the oscillatory nature of the hydrodynamic loads. This automatic depressurization system transient is of short duration such that the concrete walls do not heat up significantly. It is combined with ambient thermal conditions. Long-term heating of the tank is bounded by the design for the ADS_2 load.

- ADS_2 – This automatic depressurization system transient considers heatup of the water in the in-containment refueling water storage tank. This may be due to prolonged operation of the passive residual heat removal heat exchanger or due to an automatic depressurization system discharge. For structural design, an extreme transient is defined starting at 50°F since this maximizes the temperature gradient across the concrete-filled structural module walls. Prolonged operation of the passive residual heat removal heat exchanger raises the water temperature from an ambient temperature of 50°F to saturation in about 4 hours, increasing to about 260°F within about 10 hours. Steaming to the containment atmosphere initiates once the water reaches its saturation temperature. The temperature transient is shown in Figure 3.8.3-7. Blowdown of the primary system through the spargers may occur during this transient and occurs prior to 24 hours after the initiation of the event. Since the flow through the sparger cannot fully condense in the saturated conditions, the pressure increases in the in-containment refueling water storage tank and steam is vented through the in-containment refueling water storage tank roof. The in-containment refueling water storage tank is designed for an equivalent static internal pressure of 5 psi in addition to the hydrostatic pressure occurring at any time up to 24 hours after the initiation of the event.

The ADS_1 and ADS_2 loads are considered as live loads. The dynamic ADS_1 load is combined with the safe shutdown earthquake by the square root sum of the squares (SRSS). ADS_2 is an equivalent static pressure which is included algebraically with other normal loads and then combined with plus/minus SSE loads.

3.8.3.3.2 Concrete Placement Loads

The steel faceplates of the structural wall modules, designed for the hydrostatic pressure of the concrete, act as concrete forms. The concrete placement loads are 1050 pounds per square foot determined in accordance with ACI-347. The bending stress in the faceplate due to this hydrostatic pressure of the concrete is approximately 13 ksi, based on the assumption of a continuous faceplate, or 20 ksi based on the assumption of simple spans. The minimum yield strength of material for the faceplates is 36 ksi. The stress is well below the allowable, since the plate is designed to limit the out-of-plane deflection. After the concrete has gained strength, these stresses remain in the steel;

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however, since the average residual stress is zero and since the concrete no longer requires hydrostatic support, the ultimate strength of the composite section is not affected, and the full steel plate is available to carry other loads as described below.

The steel plates and the concrete act as a composite section after the concrete has reached sufficient strength. The composite section resists bending moment by one face resisting tension and the other face resisting compression. The steel plate resists the tension and behaves as reinforcing steel in reinforced concrete. The composite section is underreinforced so that the steel would yield before the concrete reaches its strain limit of 0.003 in/in. As the steel faceplates are strained beyond yield to allow the composite section to attain its ultimate capacity, the modest residual bending stress from concrete placement is relieved, since the stress across the entire faceplate in tension is at yield. The small residual strain induced by the concrete placement loads is secondary and has negligible effect on the ultimate bending capacity of the composite section. The stresses in the faceplates resulting from concrete placement are therefore not combined with the stresses in the post-construction load combinations.

3.8.3.4 Analysis Procedures

This subsection describes the modelling and overall analyses of the containment internal structures, including the concrete-filled structural modules. Concrete and steel composite structures are used extensively in conventional construction. Applications include concrete slabs on steel beams and concrete-filled steel columns. Testing of concrete-filled structural modules is described in References 27 through 29 for in-plane loading and in References 30 through 33 for out-of-plane loading. The tests indicate that these composite structures behave in a manner similar to reinforced concrete structures. The initial load deflection behavior is well predicted using the gross properties of the steel and concrete. This is similar to the behavior of reinforced concrete elements where the initial stiffness is predicted by the gross properties. As the load is increased on reinforced concrete members, cracking of the concrete occurs and the stiffness decreases. The behavior of concrete and steel composite structures is similar in its trends to reinforced concrete but has a superior performance. The results of the test program by Akiyama et al. (Reference 27)

indicate that concrete and steel composites similar to the structural modules have significant advantages over reinforced concrete elements of equivalent thickness and reinforcement ratios:

- Over 50 percent higher ultimate load carrying capacity
- Three times higher ductility
- Less stiffness degradation under peak cyclic loads, 30 percent for concrete and steel composites versus 65 percent for reinforced concrete

Methods of analysis for the structural modules are similar to the methods used for reinforced concrete. Table 3.8.3-2 summarizes the finite element analyses of the containment internal structures and identifies the purpose of each analysis and the stiffness assumptions for the concrete filled steel modules. For static loads the analyses use the monolithic (uncracked) stiffness of each concrete element. The elastic modulus is taken as 0.80 times the value calculated based on the ACI Code. This reduced elastic modulus considers a small degree of cracking as described in the seismic analyses in Subsection 3.7.2.3. For thermal and dynamic loads the analyses consider the extent of concrete cracking as described in later subsections.

Stiffnesses are established based on analyses of the behavior and review of the test data related to concrete-filled structural modules. The stiffnesses directly affect the member forces resulting from restraint of thermal growth. The in-plane shear stiffness of the module influences the fundamental horizontal natural frequencies of the containment internal structures in the nuclear island seismic

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analyses described in Subsection 3.7.2. The out-of-plane flexural stiffness of the module influences the local wall frequencies in the seismic and hydrodynamic analyses of the in-containment refueling water storage tank. Member forces are evaluated against the strength of the section calculated as a reinforced concrete section with zero strength assigned to the concrete in tension.

ACI 349, Section 9.5.2.3 specifies an effective moment of inertia for calculating the deflection of reinforced concrete beams. For loads less than the cracking moment, the moment of inertia is the gross (uncracked) inertia of the section. The cracking moment is specified as the moment corresponding to a maximum flexural tensile stress of $7.5\sqrt{f_c'}$. For large loads, the moment of inertia is that of the cracked section transformed to concrete. The effective moment of inertia provides a transition between these two dependent on the ratio of the cracking moment to the maximum moment in the beam at the stage the deflection is to be computed.

Table 3.8.3-1 summarizes in-plane shear and out-of-plane flexural stiffness properties of the 48-inch and 30-inch walls based on a series of different assumptions. The stiffnesses are expressed for unit length and height of each wall. The ratio of the stiffness to the stiffness of the monolithic case is also shown.

- Case 1 assumes monolithic behavior of the steel plate and uncracked concrete. This stiffness is supported by the test data described in References 27 through 33 for loading that does not cause significant cracking. This stiffness is the basis for the stiffness of the concrete-filled steel module walls in the nuclear island seismic analyses and in the uncracked case for the hydrodynamic analyses.

- Case 2 considers the full thickness of the wall as uncracked concrete. This stiffness value is shown for comparison purposes. It is applicable for loads that do not result in significant cracking of the concrete and is the basis for the stiffness of the reinforced concrete walls in the nuclear island seismic analyses (prior to the reduction in concrete stiffness by a factor of 0.8). This stiffness was used in the harmonic analyses of the internal structures described in Subsection 3.8.3.4.2.2.

- Case 3 assumes that the concrete in tension has no stiffness. For the flexural stiffness this is the conventional stiffness value used in working stress design of reinforced concrete sections. For in-plane shear stiffness, a 45-degree diagonal concrete compression strut is assumed with tensile loads carried only by the steel plate. The in-plane stiffnesses calculated by these assumptions are lower than the stiffnesses measured in the tests described in References 27 through 29 for loading that causes cracking.

3.8.3.4.1 Seismic Analyses

3.8.3.4.1.1 Finite Element Model

The structural modules are simulated within the finite element model using 3D shell elements. Equivalent shell element thickness and modulus of elasticity of the structural modules are computed as shown below. The shell element properties are computed using the combined gross concrete section and the transformed steel faceplates of the structural modules. This representation models the composite behavior of the steel and concrete. The equivalent modulus of concrete, E_m, is reduced by a factor of 0.8 to consider the effect of cracking as recommended in Table 6-5 of FEMA 356 (Reference 5 in Subsection 3.7.6). See Section 3.7 and Appendix 3G for further discussion of the containment internal structures finite element model.

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- Axial and Shear Stiffnesses of module:

$$\sum E A = E_c (L t + 2 (n - 1) L t_s)$$

- Bending Stiffness of module:

$$\sum E I = E_c \left[\frac{L}{12} t^3 + 2 \frac{L}{12} (n-1)\ t_s^3 + 2 (n-1) L t_s \left(\frac{t}{2} \right)^2 \right]$$

where:

E_c = concrete modulus of elasticity

n = modular ratio of steel to concrete

L = length of wall module

t = thickness of wall module

t_s = thickness of plate on each face of wall module

These equations lead to an equivalent thickness, t_m, and modulus of elasticity of the plate elements, E_m, as shown below:

$$t_m = \left[\frac{1 + 3 \alpha (n-1)}{1 + \alpha (n-1)} \right]^{1/2} t$$

$$E_m = [1 + \alpha (n-1)] \left[\frac{1 + 3 \alpha (n-1)}{1 + \alpha (n-1)} \right]^{-1/2} E_c$$

where $\alpha = 2t_s / t$ and terms of order α^3 are neglected (for a typical 30-inch thick wall with 1/2-inch steel plates, $\alpha = 0.033$).

3.8.3.4.1.2 Stiffness Assumptions for Local Seismic Analyses of In-Containment Refueling Water Storage Tank

The seismic analyses of the in-containment refueling water storage tank address the local response of the walls and water and are performed to verify the structural design of the tank. The local analyses performed uses the cracked section stiffness values based on composite behavior with zero stiffness for the concrete in tension (Case 3 of Table 3.8.3-1). The local analyses use the finite element model described in Subsection 3.8.3.4.2.2. Response spectrum analyses are performed using the floor response spectra at the base of the tank.

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3.8.3.4.1.3 Damping of Structural Modules

Damping of the structural modules is reported in Reference 27 based on the cyclic load tests of a containment internal structure model. The equivalent viscous damping at the design load level was 5 percent for the concrete-filled steel model. This was almost constant up to the load level at which the steel plate started yielding. Dynamic analyses are performed using 7 percent damping for the reinforced concrete and 5 percent for the structural modules as shown in Subsection 3.7.1.

3.8.3.4.2 Hydrodynamic Analyses

This subsection describes the hydrodynamic analyses performed for the AP600 which demonstrated that design of the walls of the in-containment refueling water storage tank for 5 psi as described in Subsection 3.8.3.3.1 would bound the loads from the time history transient analysis. The analyses were performed using the AP600 test results. The peak values from these tests are also applicable to the AP1000 since they occur at the beginning of the transient, and the automatic depressurization system and the initial conditions are the same for the two plant designs (Reference 52). The structural configuration of the tank is identical. The minor differences in the height of the steam generator and pressurizer compartment walls and in the mass of the steam generators and pressurizer will have only a minor effect on the significant structural frequencies. Since the time histories applied in the AP600 analyses cover a broad range of frequencies the response of the AP1000 tank boundary will be similar to that of the AP600. The 5 psi pressure design basis for the tank boundary is therefore also applicable to the AP1000.

Hydrodynamic analyses were performed for the AP600 for automatic depressurization system discharge into the in-containment refueling water storage tank. This discharge is designated as ADS_1 in the load description of Subsection 3.8.3.3.1 and results in higher hydrodynamic loading than the ADS discharge into a hot tank in ADS_2. The first three stages of the automatic depressurization system valves discharge into the tank through spargers under water, producing hydrodynamic loads on the tank walls and equipment. Hydrodynamic loads, measured in hydraulic tests of the automatic depressurization system sparger in a test tank, are evaluated using the source load approach (Reference 34). Analyses of the tests define source pressure loads that are then used in analyses of the in-containment refueling water storage tank to give the dynamic responses of the containment internal structures. The basic analysis approach consists of the following steps:

1. A pressure source, an impulsive forcing function at the sparger discharge, is selected from the tests using a coupled fluid structure finite element model of the test tank, taking into account fluid compressibility effects. This source development procedure is based on a comparison between analysis and test results, both near the sparger exit and at the boundaries of the test tank.

2. The pressure source is applied at each sparger location in a coupled fluid structure finite element model of the in-containment refueling water storage tank structure and of the contained water. The mesh characteristics of the model at the sparger locations and the applied forcing functions correspond to those of the test tank analysis.

3.8.3.4.2.1 Sparger Source Term Evaluation

A series of tests was conducted with discharge conditions representative of one sparger for the AP600 (References 35 and 36). Pressure traces measured during the test discharges were investigated, at both sparger exit and tank boundaries to (1) bound the expected discharge from the automatic depressurization system; (2) characterize the pressure wave transmission through the pool water; (3) determine the maximum pressure amplitudes and the frequency content; and (4) produce reference data for qualification of the analytical procedure. Pressure time histories and

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power spectrum densities were examined at reference sensors, both for the total duration of the discharge transient (about 50 seconds) and for critical time intervals.

Fluid-structure interaction analyses were performed with the ANSYS computer code (Reference 37). The mathematical model consists of a 3D sector finite element model, 15 degrees wide, as shown in Figure 3.8.3-9. It uses STIF30 fluid and STIF63 structural ANSYS finite elements, which take into account fluid compressibility and fluid-structure interaction. Rayleigh damping of 4 percent is used for the concrete structure, and fluid damping is neglected. Direct step-by-step time integration is used. The measured discharge pressures for single time intervals are imposed as uniform forcing functions on the idealized spherical surface of the steam/water interface, and pressures transmitted through the water to the tank boundary are calculated and compared with test measurements. The analyses of the test tank showed satisfactory agreement for the pressures at the tank boundary.

The examination of test results related to the structural design of the in-containment refueling water storage tank under automatic depressurization system hydrodynamic excitation and the comparison with the analytical procedure previously described, lead to the following conclusions regarding the sparger source term definition:

- The automatic depressurization system discharge into cold water produces the highest hydrodynamic pressures. The tests at higher water temperatures produce significantly lower pressures.

- Two pressure time histories, characterized by different shapes and frequency content, can be selected as representative of the sparger discharge pressures; they are assumed as acting on a spherical bubble centered on the sparger centerline and enveloping the ends of the sparger arms.

- The application of such time histories as forcing functions to an analytical model, simulating the fluid structure interaction effects in the test tank, has been found to predict the measured tank wall pressures, for the two selected reference time intervals.

- The two defined sparger source term pressure time histories can be used as forcing functions for global hydrodynamic analyses of the in-containment refueling water storage tank by developing a comprehensive fluid-structure finite element model and reproducing the test tank mesh pattern in the sparger region.

- The hydrodynamic loads on the vessel head support columns and ADS sparger piping located in the IRWST are developed from the forcing functions using the methodology documented in Reference 51.

3.8.3.4.2.2 In-Containment Refueling Water Storage Tank Analyses

The in-containment refueling water storage tank is constructed as an integral part of the containment internal structures as described in Subsection 3.8.3.1.3. It contains two depressurization spargers that are submerged approximately 9 feet below the normal water level. Transmission of the hydrodynamic pressures from the sparger discharge to the wetted in-containment refueling water storage tank is evaluated using the coupled fluid-structure interaction method similar to that described for the test tank analysis in the previous subsection.

The 3D ANSYS finite element model includes the in-containment refueling water storage tank boundary, the water within the in-containment refueling water storage tank, the adjacent structural walls of the containment internal structures, and the operating floor. The model of the in-containment refueling water storage tank, shown in Figures 3.8.3-10 (sheet 2), 3.8.3-11, and 3.8.3-12, represents the outer steel structures, the inner concrete walls, and the water. The model of the adjacent

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structural walls and floors is shown in Figure 3.8.3-10 (sheet 1). The flexible steel outer wall is represented using beam and shell elements; isotropic plate elements are used to represent the inner structural module walls. The water is modelled as a compressible fluid to provide an acoustic medium to transmit the source pressure. The model has two bubble boundaries representing the spargers. Pressure loads are applied to the solid element faces adjacent to the air bubbles. The forcing functions at the sparger locations are conservatively assumed to be in phase. Rayleigh damping of 5 percent is used for the concrete-filled structural modules and fluid damping is neglected. All degrees of freedom were retained in the step-by-step direct integration solution procedure for the in-containment refueling water storage tank boundary and the water. Degrees of freedom in the adjacent walls and floor were condensed by Guyon reduction.

Significant structural frequencies of the AP600 containment internal structures were analyzed using the harmonic response option with the ANSYS model of the in-containment refueling water storage tank and containment internal structures. A harmonic unit pressure is applied at the surface of the spherical bubble representing the automatic depressurization system spargers. Material properties for the concrete elements are based on the uncracked gross concrete section (Case 2 of Table 3.8.3-1). The results of these harmonic response analyses show the response deflection as a function of input frequency at nodes in the containment internal structures. The harmonic response analyses show that the largest responses are close to the wetted boundary of the in-containment refueling water storage tank and that the significant frequencies are from 18 to 50 hertz.

Two time histories are identified for the structural hydrodynamic analyses; one has significant frequencies below 40 hertz while the other has significant frequencies in the range of 40 to 60 hertz. Both time history inputs are used in the hydrodynamic analyses with the monolithic uncracked section properties for all walls. The lower frequency input is also applied in lower bound analyses using the cracked section stiffness values (Case 3 of Table 3.8.3-1) for the concrete walls that are boundaries of the in-containment refueling water storage tank. Monolithic properties are used for the other walls. Results from these cases are enveloped, thereby accounting for variabilities in the structural analyses.

The analyses of the AP600 in-containment refueling water storage tank give wall pressures, displacements, accelerations, hydrodynamic floor response spectra, and member forces due to the automatic depressurization system discharge pressure forcing functions. Consideration of pressure wave transmission and fluid-structure interaction shows a significant wall pressure attenuation with distance from the sparger region and with increasing wall flexibilities, relative to the measured sparger pressure forcing function. The member stresses are evaluated against the allowable stresses specified in Subsection 3.8.3.5 for seismic Category I structures, considering the hydrodynamic loads as live loads. The analyses show that the member forces in the walls of the in-containment refueling water storage tank are bounded by a case with a uniform pressure of 5 psi applied to the walls.

3.8.3.4.3 Thermal Analyses

The in-containment refueling water storage tank water and containment atmosphere are subject to temperature transients as described in Subsection 3.8.3.3.1. The temperature transients result in a nonlinear temperature distribution within the wall modules. Temperatures within the concrete wall are calculated in a unidimensional heat flow analysis. The average and equivalent linear gradients are applied to a finite element model of the containment internal structures at selected times during the transient. The effect of concrete cracking is considered in the stiffness properties for the concrete elements subjected to the thermal transient. The finite element model is that described in Subsection 3.8.3.4.2.2 except that the model of the water in the IRWST is not needed.

The structural modules are subject to a rapid temperature transient in the event of a loss-of-coolant accident (LOCA) or a main steam line break (MSLB). The structural modules were evaluated for

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these rapid temperature transients. The evaluation considered both carbon and stainless steel faceplates. The steel plate heats up most rapidly in the LOCA event with temperatures up to 270°F in the first few minutes for an ambient initial temperature of 50°F. The faceplate of the structural module will see differential temperatures of 220°F relative to the concrete. The concrete heats up more slowly and does not see a significant temperature increase during the early part of the transient. There is relative thermal growth of the faceplate, causing shear loads in the shear studs, and embedded angles of the structural steel trusses that are welded to the faceplate. The heatup of the surface plates during the initial portion of the LOCA transient results in cracking of the concrete walls except in regions where there is significant external restraint. The structural module maintains its integrity throughout the rapid thermal transient.

Thermal transients for the design basis accidents are described in Section 6.3. The analyses for these transients are similar to those described above.

3.8.3.5 Design Procedures and Acceptance Criteria

The containment internal structures that contain reinforcing steel including most of the areas below elevation 98′, are designed by the strength method, as specified in the ACI Code Requirements for Nuclear Safety Related Structures, ACI-349. This code includes ductility criteria for use in detailing, placing, anchoring, and splicing of the reinforcing steel.

The internal steel framing is designed according to the AISC Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, ANSI/AISC N690, supplemented by the requirements given in Subsection 3.8.4.5

[*The design and construction of anchors and embedments conform to the procedures and standards of Appendix B to ACI 349-01*]* and are in conformance with the regulatory positions of NRC Regulatory Guide 1.199, Revision 0. [*Alternative requirements to ACI 349, Appendix B apply to the anchoring of headed shear reinforcement for the basemat (see Subsection 3.8.4.4.1). Alternative requirements to ACI 349, Appendix B apply to the anchoring of headed reinforcement above the basemat (see Subsection 3.8.4.4.1).*]*

The secondary shield walls, in-containment refueling water storage tank, refueling cavity, and operating floor above the in-containment refueling water storage tank are designed using structural modules. Concrete-filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349, as supplemented in the following paragraphs. Structural floor modules are designed as composite structures in accordance with ANSI/AISC N690. Methods of analysis used are based on accepted principles of structural mechanics and are consistent with the geometry and boundary conditions of the structures.

The methods described in Subsection 3.7.2 are employed to obtain the safe shutdown earthquake loads at various locations in the containment internal structures. The safe shutdown earthquake loads are derived from the equivalent static analysis of a three-dimensional, finite element model representing the entire containment internal structures.

The determination of pressure and temperature loads due to pipe breaks is described in Subsections 3.6.1 and 6.2.1.2. Subcompartments inside containment containing high energy piping are designed for a pressurization load of 5 psi. The pipe tunnel in the CVS room (room 11209, Figure 1.2-6) is designed for a pressurization load of 7.5 psi. These subcompartment design pressures bound the pressurization effects due to postulated breaks in high energy pipe. The design for the effects of postulated pipe breaks is performed as described in Subsection 3.6.2. Determination of pressure loads resulting from actuation of the automatic depressurization system is described in Subsection 3.8.3.3.1.

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Determination of reactor coolant loop support loads is described in Subsection 3.9.3. Design of the reactor coolant loop supports within the jurisdiction of ASME Code, Section III, Division 1, Subsection NF is described in Subsections 3.9.3 and 5.4.10.

Computer codes used are general purpose codes. The code development, verification, validation, configuration control, and error reporting and resolution are according to the Quality Assurance requirements of Chapter 17.

3.8.3.5.1 Reactor Coolant Loop Supports

3.8.3.5.1.1 Reactor Vessel Support System

The reactor vessel supports are described in Subsection 3.8.3.1.1.1. Figure 3.8.3-4 shows the embedments and anchor bolts for the reactor vessel supports. The embedded portions of the reactor vessel supports, which are outside the ASME jurisdictional boundary, are designed by elastic methods of analysis. Note the dowel studs connecting the support to the steel embedment are not embedded in the concrete and are within the ASME jurisdictional boundary. The embedded portions, including the anchor bolts, are analyzed and designed to resist the applicable loads and load combinations given in Subsection 3.8.4.3. The design is according to ANSI/AISC N690 and ACI-349. The design of the steel to resist the loads satisfies the AISC-N690 requirements. The design and evaluation to assure that the embedment transfers the loads to the concrete and is retained in the concrete satisfies the requirements of ACI 349.

3.8.3.5.1.2 Steam Generator Support System

The embedded portions of the steam generator supports, which are outside the ASME jurisdictional boundary, are designed by elastic methods of analysis. They are analyzed and designed to resist the applicable loads and load combinations given in Subsection 3.8.4.3. The design is according to ANSI/AISC N690 and ACI-349. Figure 3.8.3-5 shows the jurisdictional boundaries.

3.8.3.5.1.3 Reactor Coolant Pump Support System

The reactor coolant pumps are integrated into the steam generator channel head and consequently do not have a separate support system.

3.8.3.5.1.4 Pressurizer Support System

The embedded portions of the pressurizer supports, which are outside the ASME jurisdictional boundary, are designed by elastic methods of analysis. They are analyzed and designed to resist the applicable loads and load combinations given in Subsection 3.8.4.3. The design is according to ANSI/AISC N690 and ACI-349. Figure 3.8.3-6 shows the jurisdictional boundaries.

3.8.3.5.2 Containment Internal Structures Basemat

The containment internal structures basemat including the primary shield wall and reactor cavity are designed for dead, live, thermal, pressure, and safe shutdown earthquake loads. The structural modules are designed as described in Subsection 3.8.3.5.3.

The reinforced concrete forming the base of the containment internal structures is designed according to ACI 349.

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3.8.3.5.3 Structural Wall Modules

Structural modules without concrete fill, such as the west wall of the in-containment refueling water storage tank, are designed as steel structures, according to the requirements of ANSI/AISC N690. This code is applicable since the module is constructed entirely out of structural steel plates and shapes. In local areas stresses due to restraint of thermal growth may exceed yield and the allowable stress intensity is 3 S_{m1}. This allowable is based on the allowable stress intensity for Service Level A loads given in ASME Code, Section III, Subsection NE, Paragraph NE-3221.4.

The concrete-filled steel module walls are designed for dead, live, thermal, pressure, safe shutdown earthquake, and loads due to postulated pipe breaks. The in-containment refueling water storage tank walls are also designed for the hydrostatic head due to the water in the tank and the hydrodynamic pressure effects of the water due to the safe shutdown earthquake, and automatic depressurization system pressure loads due to sparger operation. The walls of the refueling cavity are also designed for the hydrostatic head due to the water in the refueling cavity and the hydrodynamic pressure effects of the water due to the safe shutdown earthquake.

Figure 3.8.3-8 shows the typical design details of the structural modules, typical configuration of the wall modules, typical anchorages of the wall modules to the reinforced base concrete, and connections between adjacent modules. Variations in the module design details and design elements used are described in the Figure 3.8.3-8 notes and in Subsections 3.8.3.1 and 3.8.3.1.3. Concrete-filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349, as supplemented in the following paragraphs. The faceplates are considered as the reinforcing steel, bonded to the concrete by headed studs. The application of ACI-349 and the supplemental requirements are supported by the behavior studies described in Subsection 3.8.3.4.1. The steel plate modules are anchored to the reinforced concrete basemat by mechanical connections welded to the steel plate. Loads are transferred directly from the faceplates to the base concrete using reinforcing bars, structural shapes, and shear studs connected directly to the modules with mechanical connectors or welds. The reinforcing steel used to anchor the modules to the concrete has a development that satisfies the requirements of ACI 349. The design of the surface plate, base plate, and vertical stiffeners is checked by finite element analysis. The design of critical sections is described in Subsection 3.8.3.5.8.

3.8.3.5.3.1 Design for Axial Loads and Bending

Design for axial load (tension and compression), in-plane bending, and out-of-plane bending is in accordance with the requirements of ACI-349, Chapters 10 and 14.

3.8.3.5.3.2 Design for In-Plane Shear

Design for in-plane shear is in accordance with the requirements of ACI-349, Chapters 11 and 14. The steel faceplates are treated as reinforcing steel, contributing as provided in Section 11.10 of ACI-349.

3.8.3.5.3.3 Design for Out-of-Plane Shear

Design for out-of-plane shear is in accordance with the requirements of ACI-349, Chapter 11.

3.8.3.5.3.4 Evaluation for Thermal Loads

The effect of thermal loads on the structural wall modules, with and without concrete fill, is evaluated by using the working stress design method for load combination 3 of Tables 3.8.4-1 and 3.8.4-2. This evaluation is in addition to the evaluation using the working stress design method of ANSI/AISC N690 or the strength design method of ACI-349 for the load combinations without the

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thermal load. Acceptance for the load combination with normal thermal loads, which includes the thermal transients described in Subsection 3.8.3.3.1, is that the stress in general areas of the steel plate be less than yield. In local areas where the stress may exceed yield the total stress intensity range is less than twice yield. This evaluation of thermal loads is based on the ASME Code philosophy for Service Level A loads given in ASME Code, Section III, Subsection NE, Paragraphs NE-3213.13 and 3221.4.

3.8.3.5.3.5 Design of Trusses

The trusses provide a structural framework for the modules, maintain the separation between the faceplates, support the modules during transportation and erection, and act as "form ties" between the faceplates when concrete is being placed. After the concrete has cured, the trusses act with shear studs to provide interfacial shear transfer between the faceplates and concrete. The trusses also provide structural integrity and permit the development of the nominal out-of-plane shear strength of the concrete without shear reinforcement. In areas of a module wall where the hooked dowels from a floor slab do not overlap the hooked dowels of an opposing floor slab on the opposite face of the wall, the horizontal channel member of the trusses may be credited as a tension tie, by confinement of the faceplates, preventing separation of the concrete. The welded connection of the truss to the faceplate is sized to fully develop the specified minimum yield strength of the channel member. As described in Subsection 3.8.3.1.3, additional structural capacity is provided between the module faceplates in some areas to support localized loads. The trusses, connecting steel plates, structural shapes, reinforcement bars, and tie bars are designed according to the applicable requirements of ANSI/AISC N690 and ACI 349.

3.8.3.5.3.6 Design of Shear Studs

The wall structural modules are designed as reinforced concrete elements, with the faceplates serving as reinforcing steel. Since the faceplates do not have deformation patterns typical of reinforcing steel, shear studs are provided to transfer the forces between the concrete and the steel faceplates. The shear studs make the concrete and steel faceplates behave compositely. In addition, the shear studs permit anchorage for piping and other items attached to the walls.

The design of the shear stud size and spacing includes consideration of shear transfer and faceplate buckling. Shear transfer is evaluated in accordance with ANSI/AISC N690 Sections Q1.11.1 and Q1.11.4. Faceplate buckling is evaluated in accordance with ANSI/AISC N690 Section Q1.5.1.3. Conformance with these requirements precludes the possible failure modes for structural module construction for design basis loads.

3.8.3.5.4 Structural Floor Modules

Figure 3.8.3-3 shows the design details for a representative example of the floor modules. The details shown in Figure 3.8.3-3 are the representative design for a specific portion of the operating deck floor in containment. The size, material, and configuration of the structural shapes and plates, the amount and arrangement of the reinforcement, the location, size, and spacing of the shear studs welded on the top of the floor bottom plate, and the type and size of the welds may vary in the final design of floor modules in this and other locations. The operating floor is designed for dead, live, thermal, safe shutdown earthquake, and pressure due to automatic depressurization system operation or due to postulated pipe break loads. The operating floor region above the in-containment refueling water storage tank is a series of structural modules. The remaining floor is designed as a composite structure of concrete slab and steel beams in accordance with ANSI/AISC N690.

For vertical downward loads, the floor modules are designed as a composite section, according to the requirements of Section Q1.11 of ANSI/AISC N690. Composite action of the steel section and concrete fill is assumed based on meeting the intent of Section Q1.11.1 for beams totally encased in

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concrete. Although the bottom plate of the floor, which is welded to the web and acts as the bottom flange, is not encased within concrete, the design configuration of the floor module provides complete concrete confinement to prevent spalling. It also provides a better natural bonding than the code-required configuration. When used, the shear studs welded on the top of the floor bottom plate supplement the natural bottom plate-to-concrete bond capability. The shear stud size and spacing is not directly credited in the composite section design. The capacity of the studs, whenever required for localized effects, is determined in accordance with ACI 349.

For vertical upward loads, no credit is taken for composite action. The steel members are relied upon to provide load-carrying capacity. Concrete, together with the embedded angle stiffeners, is assumed to provide stability to the plates.

Floor modules are designed using the following basic assumptions and related requirements:

- Concrete provides restraint against buckling of steel plates. The buckling unbraced length of the steel plate, therefore, is assumed to equal the span length between the fully embedded steel plates and shapes.

- Although the floor modules forming the top (ceiling) of the in-containment refueling water storage tank are not in contact with water, stainless steel plates are used for the tank boundary.

- The floor modules are designed as simply supported beams.

3.8.3.5.4.1 Design for Vertical Downward Loads

The floor modules are designed as a one-way composite concrete slab and steel beam system in supporting the vertical downward loads. The effective width of the concrete slab is determined according to Section Q1.11.1 of ANSI/AISC N690. The effective concrete compression area is extended to the neutral axis of the composite section. The concrete compression area is treated as an equivalent steel area based on the modular ratio between steel and concrete material. Figure 3.8.3-13 shows the effective composite sections. The steel section is proportioned to support the dead load and construction loads existing prior to hardening of the concrete. The allowable stresses are provided in Table 3.8.4-1.

3.8.3.5.4.2 Design for Vertical Upward Loads

For vertical upward loads, the floor modules are designed as noncomposite steel structures. The effective width, b_e, of the faceplate in compression is based on post-buckling strength of steel plates and is determined from Equation (4.16) of Reference 44. The faceplates of the structural floor modules are stiffened and supported by embedded horizontal angles. Hence, the buckling unbraced length of the faceplates is equal to the span length between the horizontal angles. The buckling stress, f_{cr}, of the faceplates is determined from Sections 9.2 and 9.3 of Reference 45. The effective width of the faceplates of the structural floor modules in compression is shown in Figure 3.8.3-13. The allowable stresses are provided in Table 3.8.4-1.

3.8.3.5.4.3 Design for In-Plane Loads

In-plane shear loads acting on the floor modules are assumed to be resisted only by the steel faceplate without reliance on the concrete for strength. The stresses in the faceplate due to the in-plane loads are combined with those due to vertical loads. The critical stress locations of the floor faceplate are evaluated for the combined normal and shear stress, based on the von Mises yield criterion:

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For the particular case of a two-dimension stress condition the equation is:

$$(\sigma_1)^2 - \sigma_1\sigma_2 + (\sigma_2)^2 = (f_y)^2$$

where σ_1 and σ_2 are the principal stresses and f_y is the uniaxial yield stress.

For the faceplate where normal, σ, and shear, τ, stresses are calculated, the principal stresses can be expressed as follows:

$$\sigma_1 = \left(\frac{\sigma}{2}\right) + \sqrt{\frac{\sigma^2}{4} + \tau^2}$$

$$\sigma_2 = \left(\frac{\sigma}{2}\right) - \sqrt{\frac{\sigma^2}{4} + \tau^2}$$

Therefore, the condition at yield becomes:

$$\sigma^2 + 3\tau^2 = (f_y)^2$$

For the design of the structural floor module faceplate, the allowable stresses for the various loading conditions are as follows:

Normal condition:

$$\sigma^2 + 3\tau^2 \leq (0.6\ f_y)^2$$

Severe condition:

$$\sigma^2 + 3\tau^2 \leq (0.6\ f_y)^2$$

Extreme/abnormal condition:

$$\sigma^2 + 3\tau^2 \leq (0.96\ f_y)^2$$

Thermal stresses in the faceplates result from restraint of growth during the thermal transients described in Subsection 3.8.3.3.1. Evaluation for thermal stresses is the same as discussed in Subsection 3.8.3.5.3.4 for the wall modules.

3.8.3.5.5 Internal Steel Framing

Internal steel framing is analyzed and designed according to ANSI/AISC N690. Seismic analysis methods are described in Subsection 3.7.3.

3.8.3.5.6 Steel Form Modules

The steel form modules consist of plate reinforced with angle stiffeners and tee sections as shown in Figure 3.8.3-16. The steel form modules are designed for concrete placement loads defined in Subsection 3.8.3.3.2.

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The steel form modules are designed as steel structures according to the requirements of ANSI/AISC N690. This code is applicable since the form modules are constructed entirely out of structural steel plates and shapes and the applied loads are resisted by the steel elements.

3.8.3.5.7 Design Summary Report

A design summary report is prepared for containment internal structures documenting that the structures meet the acceptance criteria specified in Subsection 3.8.3.5.

Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Section 3.7 and 3.8 provided the following acceptance criteria are met.

- The structural design meets the acceptance criteria specified in Section 3.8
- The seismic floor response spectra meet the acceptance criteria specified in Subsection 3.7.5.4

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

3.8.3.5.8 Design Summary of Critical Sections

3.8.3.5.8.1 Structural Wall Modules

[*This subsection summarizes the design of the following critical sections:*

- *Southwest wall of the refueling cavity (4′ 0″ thick)*
- *South wall of west steam generator compartment (2′ 6″ thick)*
- *North east wall of in-containment refueling water storage tank (2′ 6″ thick)*

The thicknesses and locations of these walls which are part of the boundary of the in-containment refueling water storage tank are shown in Figure 3.8.3-18. The structural configuration and typical details are shown in Figures 3.8.3-1, 3.8.3-2, 3.8.3-8, 3.8.3-14 and 3.8.3-15. Plate thickness, structural shape size, and reinforcement provided may be increased locally. The design implemented in fabrication and construction drawings and instructions may have alternative structural shapes or reinforcement arrangements if they provide equal or better load capacity. As described in Subsection 3.8.3.1.3, steel plates, structural shapes, reinforcement bars, or tie bars are used in other locations between the structural module faceplates to address local out-of-plane loads.]* The structural analyses are described in Subsection 3.8.3.4 and summarized in Table 3.8.3-2. The design procedures are described in Subsection 3.8.3.5.3.

[*The details shown in Figure 3.8.3-17 are representative of connections between floors in containment and walls constructed using steel plate concrete composite construction.*]* The design details for the floor module are provided for background information and vary based on loading conditions and geometry. The seat angles, beam seats, clip angles, shear plates, shear studs, and reinforcement bars shown in Figure 3.8.3-17 and other design elements are used as required for the connection. A key feature of the connection design is that the design elements provide a direct load path from the floor into the wall. Additional information on the connection design is provided in Subsection 3.8.3.1.3. Details of the connection design, including plate thickness, structural shape type and size, use of specific design elements, and reinforcement arrangement, vary based on local

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loads. The design implemented in fabrication and construction drawings and instructions may have alternative structural shapes or reinforcement arrangements if they provide sufficient load capacity to satisfy ANSI/AISC N690 and ACI 349 criteria and requirements.

[*The three walls extend from the floor of the in-containment refueling water storage tank at elevation 103′ 0″ to the operating floor at elevation 135′ 3″. The south west wall is also a boundary of the refueling cavity and has stainless steel plate on both faces. The other walls have stainless steel on one face and carbon steel on the other.*]* For each wall design information is summarized in Tables 3.8.3-3, 3.8.3-4, 3.8.3-5, and 3.8.3-6. [*Results are shown at the middle of the wall (mid span at mid height), at the base of the wall at its mid point (mid span at base) and at the base of the wall at the end experiencing greater demand (corner at base). The first part of each table shows the member forces due to individual loading. The lower part of the table shows governing load combinations. The steel plate thickness required to resist mechanical loads is shown at the bottom of the table as well as the thickness provided. The maximum principal stress for the load combination including thermal is also tabulated. If this value exceeds the yield stress at temperature a supplemental evaluation is performed*]* as described in Subsection 3.8.3.5.3.4; [*for these cases the maximum stress intensity range is shown together with the allowable stress intensity range which is twice the yield stress at temperature.*]*

3.8.3.5.8.2 In-Containment Refueling Water Storage Tank Steel Wall

[*The in-containment refueling water storage tank steel wall is the circular boundary of the in-containment refueling water storage tank. The structural configuration and typical details are shown in sheet 3 of Figure 3.8.3-8.*]* The structural analyses are described in Subsection 3.8.3.4 and summarized in Table 3.8.3-2. The design procedures are described in Subsection 3.8.3.5.3. [*The steel wall extends from the floor of the in-containment refueling water storage tank at elevation 103′ 0″ to the operating floor at elevation 135′ 3″. The wall is a 5/8″ thick stainless steel plate. It has internal vertical stainless steel T-section columns spaced 4′-8″ apart and external hoop carbon steel (L-section) angles spaced 18″ to 24″ apart. The wall is fixed to the adjacent modules and floor except for the top of columns which are free to slide radially and to rotate around the hoop direction.*

The wall is evaluated as vertical and horizontal beams. The vertical beams comprise the T-section columns plus the effective width of the plate. The horizontal beams comprise the L-section angles plus the effective width of the plate. Table 3.8.3-7 shows the ratio of the design stresses to the allowable stresses. When thermal effects result in stresses above yield, the evaluation is in accordance with the supplemental criteria]* as described in Subsection 3.8.3.5.3.4.

3.8.3.5.8.3 Column Supporting Operating Floor

[*This subsection summarizes the design of the most heavily loaded column in the containment internal structures. The column extends from elevation 107′-2″ to the underside of the operating floor at elevation 135′-3″. In addition to supporting the operating floor, it also supports a steel grating floor at elevation 118′-0″.*

The load combinations in Table 3.8.4-1 were used to assess the adequacy of the column. For mechanical load combinations, the maximum interaction factor due to biaxial bending and axial load is 0.59. For load combinations with thermal loads, the maximum interaction factor is 0.94. Since the interaction factors are less than 1, the column is adequate for all the applied loads.]*

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

Subsection 3.8.4.6 describes the materials and quality control program used in the construction of the containment internal structures.

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[*The structural steel modules are constructed using carbon steel plates and shapes (ASTM A36, ASTM A992, or steel with equal or better material properties*). *Duplex 2101 (American Society for Testing and Materials A240, designation S32101) stainless steel plates or steel with equal or better material properties*]* are used on the surfaces of the modules in contact with water during normal operation or refueling. [*The material of construction for studs attached to the module plates used to transfer loads into the concrete is A-108 or steel with equal or better material properties. Bars used to anchor the modules in the concrete are deformed bars according to Reference 19, Grade 60, and Reference 20.*]*

The structural wall and floor modules are fabricated and erected in accordance with ANSI/AISC N690. Loads during fabrication and erection due to handling and shipping are considered as normal loads as described in Subsection 3.8.4.3.1.1. Packaging, shipping, receiving, storage and handling of structural modules are in accordance with NQA-1, Part II, Subpart 2.2 (formerly ANSI/ ASME N45.2.2 as specified in ANSI/AISC N690).

3.8.3.6.1 Fabrication, Erection, and Construction of Structural Modules

Modular construction techniques are used extensively in the containment internal structures (Figure 3.8.3-1). Subassemblies, sized for commercial rail shipment, are assembled offsite and transported to the site. Onsite fabrication consists of combining the subassemblies in structural modules, which are then installed in the plant. A typical modular construction technique is described in the following paragraphs for Module CA01, which is the main structural module in the containment internal structures.

The CA01 module is a multicompartmented structure which, in its final form, comprises the central walls of the containment internal structures. The vertical walls of the module house the refueling cavity, the reactor vessel compartment, and the two steam generator compartments. The module (Figure 3.8.3-14) is in the form of a "T" and is approximately 88 feet long, 95 feet wide and 77 feet high. The module is assembled from about 40 prefabricated wall sections called structural submodules (Figure 3.8.3-15). The submodules are designed for railroad transport from the fabricator's shop to the plant site with sizes up to 12 feet by 12 feet by 80 feet long. The submodule weights range from approximately 1 to 49 tons. The submodules are assembled outside the nuclear island with full penetration welds between the faceplates of adjacent subunits.

The completed CA01 module is lifted to its final location within the containment vessel by the heavy lift construction crane. Following placement of the CA01 module within the containment building, the hollow wall structures are filled with concrete, forming a portion of the structural walls of the containment internal structures.

Tolerances for fabrication, assembly and erection of the structural modules conform to the requirements of section 4 of ACI-117, applicable sections of AWS D1.1, and sections Q1.23 and Q1.25 of ANSI/AISC N690. In walls around the reactor vessel cavity, where the concrete is placed between portions of unconnected modules or between a module and a left-in-place form, the tolerance for the wall thickness may be increased over those in ACI 117. These walls have been evaluated against ACI 349-01 reinforcement design requirements and demonstrated sufficient margin to accommodate the increased tolerance. Tolerances for shear stud spacing requirements are identified on Figure 3.8.3-8, Sheet 1 and conform to AWS D1.1, Paragraph 7.4.5.

3.8.3.6.2 Nondestructive Examination

Nondestructive examination of the submodules and module is performed according to ANSI/AISC N690 and AWS D1.1. Welds are visually examined for 100 percent of their length. Full penetration welds are inspected by ultrasonic or radiographic examination for 10 percent of their

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length, or for 100 percent of the length of 1 weld in 10. Partial penetration welds are inspected by magnetic particle or liquid penetrant examination for 10 percent of their length, or for 100 percent of the length of 1 weld in 10.

3.8.3.6.3 Concrete Placement

After installation of the CA01 module in the containment, the hollow walls are filled with concrete. The concrete is placed through multiple delivery trunks located along the top of the wall or through windows in the module walls or pumping ports built into the module wall. It is placed in incremental layers with the placement rate based on the pressure of the wet concrete and its setting time. During concrete placement, workers and inspectors have access to the inside of the modules. The arrangement of the module internal trusses provides communication to aid in the free flow of concrete and movement of personnel.

3.8.3.7 In-Service Testing and Inspection Requirements

The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.3.8 Construction Inspection

Construction inspection is conducted to verify the concrete wall thickness and the surface plate thickness. The location for measurement of the structural wall modules is at the locations of the trusses used to provide the structural framework for the modules. Inspections will be measured at applicable sections excluding designed openings or penetrations. Inspections will confirm that each section provides the minimum required steel and concrete thicknesses as shown in Table 3.8.3-3. The minimum required steel and concrete thicknesses represent the minimum values to meet the design basis loads. Table 3.8.3-3 also indicates the steel plate thickness provided which may exceed the minimum required value for the following reasons:

- Structural margin
- Ease of construction
- Construction loads
- Use of standard thicknesses

[*As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.*]*

3.8.4 Other Category I Structures

The other seismic Category I structures are the shield building and the auxiliary building. New fuel and spent fuel racks are described in Section 9.1.

General criteria in this section describing the loads, load combinations, materials, and quality control are also applicable to the containment internal structures described in Subsection 3.8.3.

3.8.4.1 Description of the Structures

3.8.4.1.1 Shield Building

The shield building is the structure and annulus area that surrounds the containment building. It shares a common basemat with the containment building and the auxiliary building. The shield building uses concrete-filled steel plate construction (SC) as well as reinforced concrete (RC)

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structure. The figures in Section 1.2 show the layout of the shield building and its interface with the other buildings of the nuclear island.

A non-linear analyses performed on the nuclear island finite element model is used to validate the stiffness used in the evaluation of concrete-filled steel modules. This analysis used benchmarked layered shell elements for the reinforced concrete/steel concrete composite (RC/SC) connection, which transferred all stresses to the steel in the connections once a maximum yield stress in concrete was reached. These elements were included in the comparison made between linear and non-linear models. Results show the 80-percent stiffness model response spectra enveloped the non-linear model and provide a conservative approach in terms of response spectra and maximum stresses obtained in the shield building wall.

Figure 3.8.4-5 shows the following significant features and the principal systems and components of the shield building:

- Shield building cylindrical structure
- Shield building roof structure
- RC/SC connections
- Air inlets and tension ring
- Knuckle region (connection to exterior wall of PCS tank)
- Compression ring (connection to interior wall of PCS tank)
- Passive containment cooling system water storage (PCS) tank

The overall configuration of the shield building is established from functional requirements related to radiation shielding, missile barrier, passive containment cooling, tornado, and seismic event protection. These functional requirements led to establishing the design based on two primary design codes used for nuclear plant structures: 1) ACI 349 for reinforced concrete design, and 2) ANSI/AISC N690 for structural steel design.

The shield building SC walls are anchored to the RC basemat and shield building RC wall by mechanical connections. [*These RC-to-SC connections are shown in Figures 1, 2, 3, and 4 of APP-GW-GLR-602 (Reference 57).*]* These RC-to-SC connections are also used in the other regions of the shield building, including:

- Auxiliary building RC roof connection to the shield building SC wall
- Auxiliary building RC wall connection to shield building SC wall
- Tension ring connection to the shield building RC roof

The connections provide for the direct transfer of forces from the RC reinforcing steel to the SC liner plates. A typical auxiliary building RC roof connection to the shield building SC wall is shown in [*Figure 7 of APP-GW-GLR-602 (Reference 57).*]*

The cylindrical shield wall has an outside radius of 72.5 feet and a thickness of 36 inches. The cylindrical wall section that is a few feet below the auxiliary building roof line is a reinforced concrete structure. The section that is not protected by the auxiliary building is a steel concrete composite structure; see Figure 3H.5-16. The overall thickness of 36 inches is the same as the RC wall below. The concrete for the SC portion is standard concrete with compressive strength of 6000 psi. The SC portion is constructed with steel surface plates, which act as concrete reinforcement. The 0.75-inch tie bars are welded to the steel faceplates to develop composite behavior of the steel faceplates and concrete. The shear studs are welded to the inside surface of the steel plate. The tie bar spacing is reduced in the higher stress regions. A typical SC wall panel is shown in Figure 3H.5-13 and [*Figure 5 of APP-GW-GLR-602 (Reference 57).*]*

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The tension ring is located at the interface of the shield building steel concrete composite air inlet structures and the shield building reinforced concrete roof. The top of the tension ring interfaces with the RC roof slab. The tension ring supports the roof girders that are located under the RC roof slab. The bottom of the tension ring is attached to the air inlets structure. The bottom of the air inlets structure is attached to the top of the cylindrical SC wall of the shield building. The connection of the tension ring to the roof is a mechanical connection design and is described above. The tension ring and air inlets are shown in Figure 3H.5-14.

The primary function of the tension ring is to resist the thrust from the shield building roof. The air inlets structure is located directly below the tension ring and includes the air openings that provide for natural circulation of cooling air. Though its steel plates are connected to the concrete infill by studs and tie bars, the tension ring is conservatively designed as a hollow steel box girder. The concrete infill is credited only for out-of-plane shear transfer and for stability of the steel plates. The tension ring is designed to have high stiffness and to remain elastic under required load combinations.

The air inlets structure is a 4.5-foot-thick SC structure with through-wall openings for air flow. The air inlet openings consist of circular pipes at a downward inclination of 38 degrees from the vertical. Steel plates on each face, aligned with the inner and outer flanges of the tension ring, serve as primary reinforcement. The concrete infill is connected to the steel plates with tie bars and studs. The top of the air inlets structure is welded to the underside of the tension ring. The bottom of the air inlets structure is welded to the SC wall.

The shield building conical roof steel structure consists of 32 radial beams. Between each pair of radial beams there are circumferential beams. A steel plate is welded to the top flanges of each beam and forms a surface on which the concrete is placed. The steel structure forms a conical shell that spans the area from the compression ring to the tension ring.

The outside diameter of the passive containment cooling water storage tank (PCCWST) intersects with the shield building roof at the knuckle region. Outside of the PCCWST, the concrete roof slab thickness is 3 feet and at the bottom of the PCCWST the concrete thickness is 2 feet. The wall from the PCCWST applies a load to the roof slab, and also provides stiffness and increases the strength of the roof in that region.

The inside diameter of the PCCWST intersects with the roof slab at the compression ring. The compression ring provides the compression support for the conical roof dome. It consists of a composite structure having a curved steel beam section, which supports the concrete roof directly above it. The inside wall of the PCCWST is located above the concrete roof. Studs are placed on the top flange of the steel girder to allow the steel and concrete sections to act as a composite unit. The curved girder is designed to provide support for the steel structure during construction and during the initial placement of the concrete roof before the concrete has hardened sufficiently.

The PCCWST sits on top of the shield building roof. It is supported by and acts integrally with the conical roof. The inside surface has a liner that functions to provide leak protection, but is not required to provide structural strength to the structure. Leak chase channels are provided over the liner welds. The top elevation of the water inside the PCCWST has sufficient freeboard to preclude impact on the roof during the SSE.

Appendix 3H provides additional information regarding the shield building design, testing, and analysis that demonstrates the robust behavior of the design with respect to ductility, effects of creep, reserve strength of seismic margins, buckling, and shrinkage.

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3.8.4.1.2 Auxiliary Building

The auxiliary building is a reinforced concrete and structural steel structure. Three floors are above grade and two are located below grade. It is one of the three buildings that make up the nuclear island and shares a common basemat with the containment building and the shield building.

The auxiliary building is a C-shaped section of the nuclear island that wraps around approximately 50 percent of the circumference of the shield building. The floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

The figures in Section 1.2 show the layout of the auxiliary building and its interface with the other buildings of the nuclear island. The following are the significant features and the principal systems and components of the auxiliary building:

- Main control room
- Remote shutdown room
- Class 1E dc switchgear
- Class 1E batteries
- Reactor trip switchgear
- Reactor coolant pump trip switchgear
- Main steam and feedwater piping
- Main control room heating, ventilating, and air conditioning (HVAC)
- Class 1E switchgear rooms heating, ventilating, and air conditioning
- Spent fuel pool
- Fuel transfer canal
- Cask loading and washdown pits
- New fuel storage area
- Cask handling crane
- Fuel handling machine
- Chemical and volume control system (CVS) makeup pumps
- Normal residual heat removal system (RNS) pumps and heat exchangers
- Liquid radwaste tanks and components
- Spent fuel cooling system
- Gaseous radwaste processing system
- Mechanical and electrical containment penetrations

Structural modules are used for part of the south side of the auxiliary building. These structural modules are structural elements built up with welded steel structural shapes and plates. Concrete is used where required for shielding, but reinforcing steel is not normally used. These modules include the spent fuel pool, fuel transfer canal, and cask loading and cask washdown pits.

The configuration of the auxiliary building structural modules is similar to the structural modules described in Subsection 3.8.3.1 for the containment internal structures. As in the containment internal structures, the structural wall modules consist of steel faceplates connected by steel trusses with shear studs welded to the inside faces of the steel faceplates. Figure 3.8.4-4 shows the location of the structural modules in the auxiliary building. The structural modules extend from elevation 66′-6″ to elevation 135′-3″.

The auxiliary building modules are different in configuration and design details compared to the containment internal structures. The following examples are provided. The thickness of the structural wall modules differs from the figures for the containment internal structures modules and ranges from 2′-6″ to 5′-0″ for the key structural walls. The auxiliary building modules also includes smaller structural walls that act as labyrinth walls or barrier access walls with thicknesses less than 2′-6″. Local variation in the design of the trusses and spacing of the trusses and shear studs may be

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required to address internal obstructions and accessibility for fabrication and inspection. The minimum design thickness of the faceplates is 0.5 inch. Portions of the faceplates are thicker than the nominal design thickness to provide strength for localized loads due to attachments and connections. The auxiliary building modules are placed on the nuclear island basemat and do not have an offset of elevation of the bottom of the module faceplates through the thickness of the wall that is shown in Figure 3.8.3-8 for a portion of a module inside containment. The detail design of the internal connections within the modules including those at the corners, and for design elements and structures within the modules such as the diaphragms associated with the corner design, is different than depicted for the containment internal structures. The design of the mechanical connection of the module to the basemat varies for details such as size of deformed bars, thickness and configuration of plates, design of welds, and use of bolted connections in lieu of welds.

The ceiling of the main control room (floor at elevation 135′-3″), and the instrumentation and control rooms (floor at elevation 117′-6″) are designed as finned floor modules (Figure 3H.5-9). A finned floor consists of a 24-inch-thick concrete slab poured over a stiffened steel plate ceiling. The fins are rectangular plates welded perpendicular to the plate. Shear studs are welded on the other side of the steel plate, and the steel and concrete act as a composite section. Figure 3H.5-9 shows the finned floor above the main control room and the connection of the floor to the wall on column line 11. The finned floors above the instrumentation and control rooms are similar, but differ with some design details, such as the size and spacing of reinforcement and the connecting dowels used in the connection of the floor to the wall. Penetrations and other interferences in the floors and adjacent walls may cause localized variances with the design details shown in the figure. The fins are exposed to the environment of the room, and enhance the heat-absorbing capacity of the ceiling (see Design Control Document (DCD) Subsection 6.4.2.2). Several shop-fabricated steel panels, placed side by side, are used to construct the stiffened plate ceiling in a modularized fashion. The number of panels used is determined by the size of the room and fabricator capabilities. The stiffened plate is designed to withstand construction loads prior to concrete hardening. Additional bottom layer reinforcing steel is provided in the finned floors at elevation 117'-6" where needed to maintain the structural integrity of the fire barrier during a fire event due to potential deterioration of mechanical properties of exposed steel fin plates during the fire.

The new fuel storage area is a separate reinforced concrete pit providing temporary dry storage for the new fuel assemblies.

A cask handling crane travels in the east-west direction. The location and travel of this crane prevents the crane from carrying loads over the spent fuel pool, thus precluding them from falling into the spent fuel pool.

3.8.4.1.3 Containment Air Baffle

The containment air baffle is located within the upper annulus of the shield building, providing an air flow path for the passive containment cooling system. The air baffle separates the downward air flow entering at the air inlets from the upward air flow that cools the containment vessel and flows out of the discharge stack. The upper portion is supported from the shield building roof and the remainder is supported from the containment vessel. The air baffle is a seismic Category I structure designed to withstand the wind and tornado loads defined in Section 3.3. The air baffle structural configuration is depicted in Figures 1.2-14 and 3.8.4-1. The baffle includes the following sections:

- A wall supported off the shield building roof (see Figure 1.2-14)
- A series of panels attached to the containment vessel cylindrical wall and the knuckle region of the dome

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- A flexible seal closing the gap between the wall and the panels fixed to the containment vessel, designed to accommodate the differential movements between the containment vessel and shield building

The air baffle is designed to meet the following functional requirements:

- The baffle and its supports are configured to minimize pressure losses as air flows through the system
- The baffle and its supports have a design objective of 60 years
- The baffle and its supports are configured to permit visual inspection and maintenance of the air baffle as well as the containment vessel. Periodic visual inspections are primarily to inspect the condition of the coatings
- The baffle is designed to maintain its function during postulated design basis accidents
- The baffle is designed to maintain its function under specified external events including earthquakes, hurricanes and tornadoes

The general arrangement of the containment air baffle is shown in sheet 1 of Figure 3.8.4-1. The course layout, plate/insert plate geometry, weld seams for the containment, hatches, airlocks, and the number and configuration of the containment air baffle panels are provided for illustrative purposes only. The lowest elevation of the air baffle is approximately 146′-3″. The upper air baffle panels are on the head. The air baffle is supported by U-shaped support attachments welded to the containment vessel. The attachments can be installed in the subassembly area and, therefore, should not interfere with the containment vessel erection welds. The only penetrations through the containment vessel above the operating deck at elevation 135′-3″ are the main equipment hatch and personnel airlock. Air baffle panels around the equipment hatch and personnel airlock are designed to accomodate the hatch and airlock insulated enclosures.

Panels are attached to the containment vessel above the cylindrical portion. The panels are angled to follow the curvature of the knuckle region of the containment vessel head, forming a conical baffle that provides a transitional flow region into the upper shield building. A flexible seal is provided between this upper row of panels and the air baffle that is attached directly to the shield building roof as shown in sheet 4 of Figure 3.8.4-1. This flexible seal is attached to the top of the upper row of panels. The flexible seal is set at ambient conditions to permit relative movements from minus 2 inches to plus 3 inches radially and minus 1 inch to plus 4 inches vertically. This accommodates the differential movement between the containment vessel and the shield building, based on the absolute sum of the containment pressure and temperature deflections and of the seismic deflections, such that the integrity of the air baffle is maintained.

The panels accommodate displacements between each panel due to containment pressure and thermal growth. Radial and circumferential growth of the containment vessel are accommodated by slip at the bolts between the horizontal beams and the U shaped attachment resulting in small gaps between adjacent panels. Vertical growth is accommodated by slip between the panel and the horizontal beam supporting the top of the panel. Seal plates between the panels limit leakage during and after occurrence of these differential displacements.

3.8.4.1.4 Seismic Category I Cable Tray Supports

Electric cables are routed in horizontal and vertical steel trays supported by channel type struts made out of cold rolled channel type sections. Spacing of the supports is determined by allowable loads in the trays and stresses in the supports. The supports are attached to the walls, floors, and ceiling of

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the structures as required by the arrangement of the cable trays. Longitudinal and transverse bracing is provided where required.

3.8.4.1.5 Seismic Category I Heating, Ventilating, and Air Conditioning Duct Supports

Heating, ventilating, and air conditioning duct supports consist of structural steel members or cold rolled channel type sections attached to the walls, floors, and ceiling of the structures as required by the arrangement of the duct. Spacing of the supports is determined by allowable stresses in the duct work and supports. Longitudinal and transverse bracing is provided where required.

3.8.4.2 Applicable Codes, Standards, and Specifications

The following standards are applicable to the design, materials, fabrication, construction, inspection, or testing:

[● *American Concrete Institute (ACI), Code Requirements for Nuclear Safety Related Concrete Structures, ACI 349-01*]* (refer to Subsection 3.8.4.5 for supplemental requirements)

● [*American Concrete Institute (ACI), Building Code Requirements for Structural Concrete, ACI 318-11, Section 12.6*]* (refer to subsection 3.8.4.4.1 for application of ACI 318-11 Section 12.6)

● American Concrete Institute (ACI), ACI Detailing Manual

● American Concrete Institute (ACI), Self-Consolidating Concrete, ACI-237R

[● *American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, ANSI/AISC N690-1994*]* (refer to Subsection 3.8.4.5 for supplemental requirements)

[● *American Welding Society (AWS), Structural Welding Code - Steel, AWS D1.1-2000*

AWS D1.1-2000 is an acceptable code for use in lieu of AWS D1.1-1992 for the design, qualification, fabrication, and inspection for ANSI/AISC N690 applications.

The use of AWS D1.1-2000 to replace and supplement provisions in ANSI/AISC N690-1994 includes the provisions noted below:

– *The provisions in Section 2.3.4.1 of AWS D1.1-2000 are used to size complete joint penetration groove welds for tubular members.*

– *ASTM A992 is an alternative to ASTM A36 steel for structural steel shapes. Where ASTM A992 steel is used, the provisions for matching weld metal identified in AWS D1.1-2000, Section 3.3 and Table 3.1 are used to supplement the information in ANSI/AISC N690 Section Q1.4.4.*

– *Where directionality is applied to increase allowable weld stress for fillet welds identified in ANSI/AISC N690, Section Q1.5.3 and Table Q1.5.3, the provisions in Sections 2.14.4 and 2.14.5 of AWS D1.1-2000 shall be used.*

For fillet weld groups concentrically loaded in the plane of the axis of the weld and consisting of elements oriented both longitudinally and transversely to the direction of applied load, the deformation capability of the elements shall be considered. As a supplement to the provisions

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in AWS D1.1-2000, the strength of the weld group is limited to the larger of the two following evaluations:

1. *The combined strength of the individual weld elements using 1.0 times the strength of the longitudinally oriented element and 1.0 times the strength of the transversely oriented element*

2. *The combined strength of the individual weld elements using 0.825 times the strength of the longitudinally oriented element and 1.5 times the strength of the transversely oriented element*

The provisions for directionality are not applied to fillet welds of tubular member T, Y, and K connections described in Table 2.5 of AWS D1.1-2000.

The provisions in AWS D1.1-2000 for the application of directionality to linear fillet welds are supplemented by extending the provisions to weld configurations not specified in AWS D1.1-2000 as noted below:

– *Where directionality is considered for concentric out-of-plane loading for linear fillet weld groups, including lines and rectangular configurations, the directionality provisions in AWS D1.1-2000 are applied.*

– *Where directionality is considered for concentric out-of-plane loading for circular-shaped fillet weld configurations, the directionality provisions in AWS D1.1-2000 are applied. The use of directionality provisions in AWS D1.1-2000 for circular-shaped fillet weld configurations is limited to tension loading concentrically applied out-of-plane to the axis of the weld.*]*

- American Iron and Steel Institute (AISI), Specification for the Design of Cold Formed Steel Structural Members, Parts 1 and 2, 1996 Edition and 2000 Supplement

Section 1.9 describes conformance with the Regulatory Guides.

Welding and inspection activities for seismic Category I structural steel, including building structures, structural modules, cable tray supports and heating, ventilating, and air conditioning duct supports are accomplished in accordance with written procedures and meet the requirements of the American Institute of Steel Construction (ANSI/AISC N690). The welded seam of the plates forming part of the leaktight boundary of the spent fuel pool, fuel transfer canal, cask loading pit, and cask washdown pit is examined by liquid penetrant and vacuum box after fabrication to confirm that the boundary does not leak. Vacuum box examination is not required where geometry prevents examination from being performed and where an alternative leak detection method can be performed.

3.8.4.3 Loads and Load Combinations

The loads and load combinations used in the analysis are considered to be part of the method of evaluation.

3.8.4.3.1 Loads

The loads considered are normal loads, severe environmental loads, extreme environmental loads, and abnormal loads.

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3.8.4.3.1.1 Normal Loads

Normal loads are those loads to be encountered, as specified, during initial construction stages, during test conditions, and later, during normal plant operation and shutdown. They include the following:

D = Dead loads or their related internal moments and forces, including any permanent piping and equipment loads

F = Lateral and vertical pressure of liquids or their related internal moments and forces

L = Live loads or their related internal moments and forces, including any movable equipment loads and other loads that vary with intensity and occurrence

H = Static earth pressure or its related internal moments and forces

T_o = Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition

R_o = Piping and equipment reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

3.8.4.3.1.2 Severe Environmental Loads

The severe environmental load is the following:

W = Loads generated by the design wind specified for the plant in Subsection 3.3.1.1

3.8.4.3.1.3 Extreme Environmental Loads

Extreme environmental loads are the following:

E_s = Loads generated by the safe shutdown earthquake specified for the plant, including the associated hydrodynamic and dynamic incremental soil pressure. Loads generated by the safe shutdown earthquake are specified in Section 3.7.

Wt = Loads generated by the design tornado specified for the plant in Subsection 3.3.2, including loads due to tornado wind pressure, differential pressure, and tornado-generated missiles.

N = Loads generated by the probable maximum precipitation (provided previously in Table 2-1).

3.8.4.3.1.4 Abnormal Loads

Abnormal loads are those loads generated by a postulated high-energy pipe break accident for pipes not qualified for leak-before-break. Abnormal loads include the following:

P_a = Pressure load within or across a compartment generated by the postulated break. The main steam isolation valve (MSIV) compartments are designed for a 6.5 psi pressurization load. The steam generator blowdown valve compartment is designed for a 6 psi pressurization load. The subcompartment design pressure bounds the pressurization effects due to postulated breaks in high energy pipe. Determination of subcompartment pressure loads is discussed in Subsection 6.2.1.2.

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Ta = Thermal loads under thermal conditions generated by the postulated break and including To. Determination of subcompartment temperatures is discussed in Subsection 6.2.1.3.

R_a = Piping and equipment reactions under thermal conditions generated by the postulated break and including R_o. Determination of pipe reactions generated by postulated breaks is discussed in Section 3.6.

Y_r = Load on the structure generated by the reaction on the broken high-energy pipe during the postulated break. Determination of the loads is discussed in Section 3.6.

Y_j = Jet impingement load on the structure generated by the postulated break. Determination of the loads is discussed in Section 3.6.

Y_m = Missile impact load on the structure generated by or during the postulated break, as from pipe whipping. Determination of the loads is discussed in Section 3.6.

3.8.4.3.1.5 Dynamic Effects of Abnormal Loads

The dynamic effects from the impulsive and impactive loads caused by P_a, R_a, Y_r, Y_j, Y_m, and tornado missiles are considered by one of the following methods:

- Applying an appropriate dynamic load factor to the peak value of the transient load
- Using impulse, momentum, and energy balance techniques
- Performing a time-history dynamic analysis

Elastoplastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Dynamic increase factors appropriate for the strain rates involved may be applied to static material strengths of steel and concrete for purposes of determining section strength.

3.8.4.3.2 Load Combinations

3.8.4.3.2.1 Steel Structures

The steel structures and components are designed according to the elastic working stress design methods of the ANSI/AISC N690 specification using the load combinations specified in Table 3.8.4-1.

3.8.4.3.2.2 Concrete Structures

The concrete structures and components are designed according to the strength design methods of ACI-349 Code, using the load combinations specified in Table 3.8.4-2.

3.8.4.3.2.3 Live Load for Seismic Design

Floor live loads, based on requirements during plant construction and maintenance activities, are specified varying from 50 to 250 pounds per square foot (with the exception of the containment operating deck which is designed for 800 pounds per square foot specified for plant maintenance condition).

For the local design of members, such as the floors and beams, seismic loads include the response due to masses equal to 25 percent of the specified floor live loads or 75 percent of the roof snow load, whichever is applicable. These seismic loads are combined with 100 percent of these specified

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live loads, or 75 percent of the roof snow load, whichever is applicable, except in the case of the containment operating deck. For the seismic load combination, the containment operating deck is designed for a live load of 200 pounds per square foot which is appropriate for plant operating condition. The mass of equipment and distributed systems is included in both the dead and seismic loads.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Seismic Category I Structures

[*The design and analysis procedures for the seismic Category I structures (other than the containment vessel, containment internal structures, and other structures constructed using concrete-filled steel plate construction), including assumptions on boundary conditions and expected behavior under loads, are in accordance with ACI-349 with clarification as provided below for concrete structures, with ANSI/AISC N690 for steel structures, and AISI for cold formed steel structures.*]* The structural wall modules in the auxiliary building are designed using similar design concepts as the structural modules in the containment internal structures described in Subsection 3.8.3.5.3. The shield building is designed using the procedures and requirements described in Subsection 3.8.4.5.5.

[*The criteria of ACI-349, Chapter 12, are applied in development and splicing of the reinforcing steel. The ductility criteria of ACI-349, Chapter 21, are applied in detailing and anchoring of the reinforcing steel. Provisions in ACI 318-11 Section 12.6*]* (Reference 58) [*for headed and mechanically anchored deformed bars apply to development of headed reinforcement as an alternative to the provisions in ACI 349 Appendix B.*]* The specific application of ACI 318-11 Section 12.6 is used for development of the headed reinforcement to provide ductile behavior. Alternatively, a rebar hook with requirements according to ACI 349 may also be used.

[*The application of Chapter 21 detailing is demonstrated in the reinforcement details of critical sections*]* in Subsection 3.8.5 and Appendix 3H.

[*Sections 21.2 through 21.5 of Chapter 21 of ACI 349 are applicable to frame members resisting earthquake effects. These requirements are considered in detailing structural elements subjected to significant flexure and out-of-plane shear. These elements include the following examples described in Appendix 3H*:]*

- Reinforcement details for the basemat are described in Subsection 3.8.5. [*Shear stirrups have T headed anchors at each end. Provisions in ACI 318-11 Section 12.6*]* (Reference 58) [*for headed and mechanically anchored deformed bars apply to headed shear reinforcement as an alternative to the provisions in ACI 349 Appendix B.*]* The specific application of ACI 318-11 Section 12.6 is used for development of the headed shear reinforcement to provide ductile behavior.

- Reinforcement details for the exterior walls below grade are described in Subsection 3H.5.1.1. [*Shear ties have T headed anchors at each end. Provisions in ACI 318-11 Section 12.6*]* (Reference 58) [*for headed and mechanically anchored deformed bars apply to headed reinforcement as an alternative to the provisions in ACI 349 Appendix B.*]* The specific application of ACI 318-11 Section 12.6 is used for development of the headed reinforcement to provide ductile behavior. Alternatively, a conventional tie with alternating 90 and 135 degree hooks around the longitudinal reinforcement may be used.

[*Sections 21.2 and 21.6 of Chapter 21 of ACI 349 are applicable to walls, diaphragms, and trusses serving as parts of the earthquake force-resisting systems as well as to diaphragms, struts, ties,*

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chords and collector elements. These requirements are considered in the detailing of reinforcement in the walls and floors of the auxiliary building and in the shield building cylindrical wall and roof.]*

- Reinforcement for the shear walls and floors are shown in Subsections 3H.5.1 to 3H.5.4. [*Transverse reinforcement terminating at the edges of structural walls or at openings is detailed in accordance with 21.6.6.5 of ACI 349.*]*

The bases of design for the tornado, pipe breaks, and seismic effects are discussed in Sections 3.3, 3.6, and 3.7, respectively. The foundation design is described in Subsection 3.8.5.

The seismic Category I structures are reinforced concrete, concrete-filled steel plate, and structural module shear wall structures consisting of vertical shear/bearing walls and horizontal slabs supported by structural steel framing. Seismic forces are obtained from the response spectrum analysis of the three dimensional finite element models described in Table 3G.1-1. These results are modified to account for accidental torsion as described in Subsection 3.7.2.11. Where the refinement of these finite element models is insufficient for design of the reinforcement, for example in walls with a large number of openings, detailed finite element models are used. Also evaluated and considered in the shear wall and floor slab design are out-of-plane bending and shear loads, such as live load, dead load, seismic, lateral earth pressure, hydrostatic, hydrodynamic, and wind pressure. These out-of-plane bending and shear loads are obtained from the response spectrum analyses supplemented by hand calculations.

The exterior walls of the seismic Category I structures below the grade are designed to resist the worst case lateral earth pressure loads (static and dynamic), soil surcharge loads, and loads due to external flooding as described in Section 3.4. The lateral earth pressure loads are evaluated for two cases:

- Lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with ASCE 4-98 (Reference 56), Subsection 3.5.3, Figure 3.5-1, “Variation of Normal Dynamic Soil Pressures for the Elastic Solution”

- Lateral earth pressure equal to the passive earth pressure

The shield building roof and the passive containment cooling water storage tank are analyzed using the three-dimensional finite element quadrant model described in Subsection 3G.2.3.1 with the ANSYS computer code. Loads and load combinations are given in Subsection 3.8.4.3 and include construction, dead, live, thermal, wind and seismic loads. The seismic response of the water in the tank is applied as static pressures corresponding to the impulsive and convective response. The results are used in the design of the tension ring, air inlet structure, PCS tank, shield building roof, and radial roof beams. The PCS tank is designed using the maximum accelerations at the applicable elevation resulting from time history dynamic analyses of the nuclear island. The tension ring and air inlet use maximum accelerations that are increased such that the member forces in these regions envelope those from a response spectrum analysis using the refined NI05 model, as described in Appendix 3G.2.2.4.

The liner for the passive containment cooling water storage system tank is analyzed by hand calculation. The design considers construction loads during concrete placement, loads due to handling and shipping, normal loads including thermal, and the safe shutdown earthquake. Buckling of the liner is prevented by anchoring the liner using the embedded stiffeners and welded studs. The liner is designed as a seismic Category I steel structure in accordance with ANSI/AISC N690 with the supplemental requirements given in Subsection 3.8.4.

The structural steel framing is used primarily to support the concrete slabs and roofs. Metal decking, supported by the steel framing, is used as form work for the concrete slabs and roofs. The structural

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steel framing is designed for vertical loads. Appendix 3H shows typical structural steel framing in the auxiliary building.

Computer codes used are general purpose computer codes. The code development, verification, validation, configuration control, and error reporting and resolution are according to the quality assurance requirements of Chapter 17.

[*The finned floors for the main control room and the instrumentation and control room ceilings are designed as reinforced concrete slabs in accordance with ACI-349. The steel panels are designed and constructed in accordance with ANSI/AISC N690. For positive bending, the steel plate is in tension and the steel plate with fin stiffeners serves as the bottom reinforcement. For negative bending, compression is resisted by the stiffened plate and tension by top reinforcement in the concrete.*]*

The concrete floors on steel plates, including the floors in the CA20 module, are designed as reinforced concrete slabs in accordance with ACI-349. The steel panels are designed and constructed in accordance with ANSI/AISC N690. For positive bending, the steel plate is in tension and the steel plate serves as the bottom reinforcement. For negative bending, compression is resisted by the concrete and steel plate and the tension by top reinforcement in the concrete. This methodology is similar to that described for the control room ceiling in Subsection 3H.5.4.

The design of the connection meets ACI 349 and ANSI/AISC N690 requirements for shear stud capacity. The anchorage of the reinforcing dowels may vary from that shown in Figure 3H.5-9, Sheet 2. Differences in dowel anchorage details meet ACI 349 requirements for standard hooks and ACI 318-11 requirements for headed reinforcement.

The construction sequence for CA20 floors is as follows:

- Each panel (steel plates, structural tees, and shear connectors) is fabricated in the shop, brought to the floor location, and placed in position. In some cases, panels are preassembled and placed as a module. The bottom plate is supported by brackets connected to the walls.

- The steel plate is used as the formwork for concrete placement. The wet concrete load is carried by the steel plates and the structural tees that act to stiffen the bottom plate.

- During concrete placement, shoring may be provided for floors with longer spans. Local shoring of the steel plates at penetrations and other openings in the floor and supporting wall, or to act as temporary support, may also be provided.

3.8.4.4.2 Seismic Category I Cable Tray Supports

The design and analysis procedures for seismic Category I cable trays and their supports are described in Appendix 3F.

3.8.4.4.3 Seismic Category I Heating, Ventilating, and Air Conditioning Duct Supports

The design and analysis procedures for seismic Category I heating, ventilating, and air conditioning ducts and their supports are described in Appendix 3A.

3.8.4.4.4 Below Grade Exterior Walls

The design and analysis procedures for seismic Category I exterior walls below grade are described below.

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The nuclear island exterior walls below grade are subjected to various loads, including the lateral earth pressure loads. Lateral loads used in design of the nuclear island are based on conservative assumptions (soil profiles with highest lateral loads) for the properties of the soil adjacent to the exterior walls. Lateral loads are calculated for a range of possible soil properties and a conservative set of loads is specified for design.

The plant grade elevation is 100′-0″, the high groundwater level is 98′-0″, and the maximum flood level is at plant elevation 100′-0″.

Load Conditions

Hydrostatic (groundwater) (Live, L)

The design high groundwater level is at elevation 98′-0″, and the probable maximum flood level is at elevation 100′-0″. Both of these loads are treated as live loads. The hydrostatic unit water pressure (P_w) at a depth h (units: feet) below ground level is calculated as:

$$P_w = \gamma_w h \quad (1)$$

Where, γ_w = unit weight of water = 62.4 pcf

At-Rest Earth Pressure (Earth, H)

Static earth pressure is based on “at -rest” conditions, and the coefficient of earth pressure for the at-rest condition (K_o) is determined from the following relationship:

$$P_o = K_o \gamma h \quad (2)$$

Where,

K_o = 1-SIN(φ)

φ = angle of internal friction

h = depth below grade (El. 100′-0″)

$\gamma = \gamma_s$ = Saturated unit weight of granular back fill above water table, or

$\gamma = \gamma_s - \gamma_w$ below water table

Static and Dynamic Surcharge Pressures

The static surcharge pressure (Dead, D) is considered a static pressure load, and therefore, the at-rest coefficient (K_o) is used. The static lateral surcharge pressure is defined for the same soil case as the at-rest pressure as follows:

$$P_{surch} = K_o q \quad (3)$$

Where,

q = static surcharge pressure based on footprint loads of adjacent structures

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The dynamic lateral surcharge pressure (Seismic, Es_1) is based on 0.3 times the static surcharge pressure. The static and dynamic lateral surcharge pressures act uniformly along the height of the exterior walls.

Dynamic Earth Pressure (Seismic, Es_2)

The dynamic earth pressure is calculated in accordance with ASCE 4-98 (Reference 56), Subsection 3.5.3, Figure 3.5-1, "Variation of Normal Dynamic Soil Pressures for the Elastic Solution." The Poisson's ratio (v) for the soil varies between 0.35 and 0.4. The most conservative dynamic soil pressure distribution, obtained using 0.4 for v, is used. The seismic acceleration levels are the maximum accelerations associated with the seismic response of the nuclear island from elevation 60.5′ to elevation 100′. The north-south seismic excitation acceleration values are associated with Walls 1 and 11. The east-west seismic acceleration values apply to Walls N, Q, and I.

Passive Earth Pressure (Seismic, Es_3)

The nuclear island includes passive pressure on the exterior walls to resist sliding during an SSE. Therefore, the exterior walls below grade shall also be designed for passive earth pressure in the load combinations that include Es. The passive earth pressure is calculated from:

$$P_P = K_P \gamma h \quad (4)$$

Where,

$K_P = \text{Tan}^2(45° + \varphi / 2)$

φ = angle of internal friction

h = depth below grade (El. 100′-0″)

$\gamma = \gamma_s$ = Saturated unit weight of granular back fill above water table or

$\gamma = \gamma_s - \gamma_w$ below water table

Load Combinations

Lateral loading conditions, identified above, are used for the following load combinations:

- Load Combination 3: (L) + (H) + (D) + (Es_1) + (Es_2)
- Load Combination 7: (L) + (D) + (Es_1) + (Es_3)
- Load Combinations 1 & 2: 1.4 x (D) + 1.7 x ((L) + (H))
- Load Combinations 4, 5 & 6: (L) + (H) + (D)
- Load Combinations 8 & 9: 1.05 x (D) + 1.3 x ((L) + (H))

The design of the exterior below grade walls is designed from the load combinations listed above, which incorporate the effects of full lateral passive earth pressure.

Based on the results for the load combinations, the critical lateral pressure distribution for the design loads along the exterior, below grade walls would result when combining the effects of Load Combinations (LC) 3 and 7 along the nuclear island's east-west axis. The pressure distribution for LC 3 would control from elevation 100′ (ground surface) to the intersection with the LC 7 pressure distribution. Graphically, this occurs at approximately elevation 91′. The LC 7 plot would control from this intersection to elevation 60.5′ (bottom of the basemat). LC 7 incorporates the effects of full lateral passive earth pressure.

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Lateral Soil Pressure with Vertical Seismic Effects

Soil Pressure – Vertical seismic is the at-rest earth pressure for the generic soil model multiplied by 0.3 (seismic acceleration factor). This same factor was used to determine the dynamic lateral surcharge pressure (Seismic, Es_2) noted above. These tabular values where combined with the values from the dynamic soil pressure (along the east-west axis) by the SRSS Method (square root, sum of squares) and plotted against the dynamic soil pressure (along the east-west axis) alone. The contribution of vertical seismic effect on the at-rest earth pressure is negligible to the dynamic soil pressure. Therefore, vertical seismic effect is neglected and the dynamic soil pressure alone was used in determining the load combination.

3D SASSI Analyses

3D SASSI soil-structure interaction (SSI) analyses for soil conditions soft-to-medium (SM) and upper-bound-soft-to-medium (UBSM) soil were used to estimate the dynamic soil pressure on the exterior nuclear island walls. Five locations were used to calculate lateral pressure on the structure, including spring elements at elevation 100′ and 82.5′, and structure elements at elevation 91′, 71′, and 57′. Axial spring forces normal to and along each wall are tabulated and converted to a unit load and corresponding stress at each elevation. Similarly, element stresses normal to the plane of the wall are tabulated for each elevation, and the maximum normal stress was determined.

The 3D SASSI UBSM and SM dynamic soil pressure for the north-south and east-west vertical profiles are compared to the ASCE 4-98 lateral soil pressure distribution curve.

3D ANSYS Analyses

The analyses of the nuclear island for the lateral earth pressure loads are performed using 3D linear elastic ANSYS analysis. The lateral earth pressure loads are applied to the finite element nodes associated with embedded nuclear island structural walls. The analysis performed is linear elastic. Loads and moments at the base of the exterior walls are extracted from the analyses and are added in the design analyses of the basemat.

3.8.4.5 Structural Criteria

[*The analysis and design of concrete conform to ACI-349 as supplemented below and with clarifications provided in Subsection 3.8.4.4.1. The analysis and design of structural steel conform to ANSI/AISC N690 as supplemented below with clarifications provided in Subsection 3.8.4.5.2. The analysis and design of cold-formed steel structures conform to AISI. The margins of structural safety are as specified by those codes.*]*

3.8.4.5.1 Supplemental Requirements for Concrete Structures

[*Supplemental requirements for ACI-349-01 are given in the position on Regulatory Guide 1.142 in Appendix 1A. The structural design meets the supplemental requirements identified in Regulatory Positions 2 through 8, 10 through 13, and 15.*]*

Paragraph 21.6.1 of ACI 349-01 should reference 21.6.6 instead of 21.6.5. Paragraph 21.6.5 in ACI 349-97 was renumbered to 21.6.6 in ACI 349-01, and the reference in 21.6.1 was not updated. The errata for ACI 349-01 are being updated to include this correction. This makes the paragraph consistent with ACI 349-97, which was endorsed by Regulatory Guide 1.142.

[*Design and construction of fastening to concrete is in accordance with ACI 349-01, Appendix B*]* and are in conformance with the regulatory positions of NRC Regulatory Guide 1.199, Revision 0. [*Alternative requirements to ACI 349, Appendix B apply to the anchoring of headed shear reinforcement for the basemat (see Subsection 3.8.4.4.1). Alternative requirements to ACI 349,*

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Appendix B apply to the anchoring of headed reinforcement above the basement (see Subsection 3.8.4.4.1).

Weldable coupler connections of reinforcing bar to structural steel shall develop 125% of the specified yield strength of the bar in accordance with ACI 349-01 Section 12.14.3.4. Qualification of the C2/C3J coupler welds is demonstrated as follows:

For reinforcing bar sizes #4, #5, and #6, the coupler connection weld adequacy is demonstrated by calculations in accordance with ANSI/AISC N690-1994 requirements using a stress limit coefficient (SLC) of 1.6.

For reinforcing bar sizes #7, #8, #9, #10, and #11, coupler connection weld adequacy is demonstrated through testing, as permitted by ANSI/AISC N690-1994 Section Q1.0.1, and more specifically, Section Q1.22.2. Two sets of testing are performed to demonstrate that the strength of the reinforcing bar is the limiting feature of the coupler reinforcing bar splice and weld system:

(1) Six static tension tests are performed for each reinforcing bar size on samples of the coupler reinforcing bar splice and weld system, retaining the 90%/95% upper confidence limit of the coupler system static tension test results. Three cyclic tests are also performed as described in ACI 349-01 12.14.3.4.1(b) to confirm that there is not significant coupler system degradation under cyclic demand.

(2) Static tension testing of the coupler weld to failure using ten representative sample weld configurations from each of the five reinforcing bar coupler sizes is performed to determine the 90%/95% lower confidence limit weld capacity.

The tests demonstrate that the factor of safety between the weld 90%/95% lower confidence limit and 125% of the specified yield strength of the reinforcing bar exceeds 1.4. The tests also demonstrate that the factor of safety between the weld 90%/95% lower confidence limit and the 90%/95% upper confidence limit of the coupler system static tension test results exceeds 1.0. The analyses and tests establish a minimum fillet reinforcement size for the C2/C3J couplers:

C2/C3J Coupler Size	*#4*	*#5*	*#6*	*#7*	*#8*	*#9*	*#10*	*#11*
Min. Fillet Weld Size (inches)	*1/4*	*1/4*	*1/4*	*1/4*	*1/4*	*5/16*	*1/4*	*3/8*]*

3.8.4.5.2 Supplemental Requirements for Steel Structures

[*Supplemental requirements for use of ANSI/AISC N690 are as follows:*

- *In Section Q1.0.2, the definition of secondary stress applies to stresses developed by temperature loading only.*
- *In Section Q1.3, where the structural effects of differential settlement are present, they are included with the dead load, D.*
- *In Table Q1.5.7.1, the stress limit coefficients for compression are as follows:*

  1.3 instead of 1.5 in load combinations 2, 5, and 6.

  1.4 instead of 1.6 in load combinations 7, 8, and 9.

  1.6 instead of 1.7 in load combination 11.

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- *In Section Q1.5.8, for constrained members (rotation and/or displacement constraint such that a thermal load causes significant stresses), supporting safety-related structures, systems, or components, the stresses under load combinations 9, 10, and 11 are limited to those allowed in Table Q1.5.7.1 as modified above.*

- *Sections Q1.24 and Q1.25.10 are supplemented as follows:*

  Shop painting is in accordance with Section M of the Manual of Steel Construction, Load and Resistance Factor Design, First Edition. Exposed areas after installation are field painted in accordance with the applicable portion of Chapter M of the Manual of Steel Construction, Load and Resistance Factor Design, First Edition.]* See Subsection 6.1.2.1 for additional description of the protective coatings.

3.8.4.5.3 Design Summary Report

A design summary report is prepared for seismic Category I structures documenting that the structures meet the acceptance criteria specified in Subsection 3.8.4.5.

Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Sections 3.7 and 3.8 provided the following acceptance criteria are met:

- the structural design meets the acceptance criteria specified in Section 3.8
- the seismic floor response spectra meet the acceptance criteria specified in Subsection 3.7.5.4

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

3.8.4.5.4 Design Summary of Critical Sections

[*The design of representative critical elements of the following structures is described in Appendix 3H.*

- *South wall of auxiliary building (column line 1), elevation 66'-6" to elevation 180'-0" – see subsection 3H.5.1.1 and Figures 3H.5-2 and 3H.5-3*

- *Interior wall of auxiliary building (column line 7.3), elevation 66'-6" to elevation 160'-6" – see subsection 3H.5.1.2 and Figure 3H.5-4*

- *West wall of main control room in auxiliary building (column line L), elevation 117'-6" to elevation 153'-0" – see subsection 3H.5.1.3 and Figure 3H.5-12*

- *North wall of MSIV east compartment (column line 11 between lines L and M), elevation 117'-6" to elevation 153'-0" – see subsection 3H.5.1.4 and Figure 3H.5-5*

- *Roof slab at elevation 180'-0" adjacent to shield building cylinder – see subsection 3H.5.2.1 and Figure 3H.5-7*

- *Floor slab on metal decking at elevation 135'-3" – see subsection 3H.5.2.2 and Figure 3H.5-6*

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- *2'-0" slab in auxiliary building (operations work area (tagging room) ceiling) at elevation 135'-3" – see subsection 3H.5.3.1 and Figure 3H.5-8 (Note: The 'Tagging Room' has been renamed as "Operations Work Area." However, to avoid changing the associated design and analysis documents, this room is referred to as the 'Tagging Room.')*

- *Finned floor in the main control room at elevation 135'-3" – see subsection 3H.5.4 and Figure 3H.5-9*

- *Shield building roof, exterior wall of the PCS water storage tank – see subsection 3H.5.6.3 and Figure 3H.5-11*

- *Shield building roof, interior wall of the PCS water storage tank – see subsection 3H.5.6.2 and Figure 3H.5-11*

- *Shield building roof, tension ring, and air inlet structure – see subsections 3H.5.6 and 3H.5.6.1*

- *Divider wall between the spent fuel pool and the fuel transfer canal – see subsection 3H.5.5.1 and Figure 3H.5-10*

- *Shield building SC cylinder – see subsection 3H.5.7.1, Figure 3H.5-16, and Figures 5 and 6 of APP-GW-GLR-602 (Reference 57)*

- *Shield building SC to RC connection – see Subsection 3H.5.7.2, Figure 3H.5-16, and Figures 1, 2, 3, and 4 of APP-GW-GLR-602 (Reference 57)*]*

3.8.4.5.5 Shield Building Structural Wall Modules

[*The shield building concrete-filled steel module walls are designed for loads such as dead, live, thermal, wind, and safe shutdown earthquake loads as identified in Table 3.8.4-2. Concrete-filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI 349, and supplemented with additional requirements discussed in subsection 3.8.3.5.3 and below. The faceplates are considered as the reinforcing steel, bonded to the concrete by headed studs. The steel plate modules are anchored to the reinforced concrete by mechanical connections welded to the steel plate.*]*

Figure 5 in Reference 57 shows the typical design details of the shield building concrete-filled steel module walls. The design of the shield building critical sections is described in Appendix 3H.

3.8.4.5.5.1 Design for Axial Loads and Bending

Design for axial load (tension and compression), in-plane bending, and out-of-plane bending is in accordance with the requirements of ACI 349, Chapters 10 and 14. The reinforcement for in-plane tension and compression combined with out-of-plane moments is calculated using a rectangular concrete stress distribution, in accordance with Sections 10.2 and 10.3 of the ACI 349 code.

3.8.4.5.5.2 Design for In-Plane Shear

Design for in-plane shear is in accordance with the requirements of ACI 349, Chapters 11 and 14. The steel faceplates are treated as reinforcing steel, contributing as provided in Section 11.10 of ACI 349.

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3.8.4.5.5.3 Design for Out-of-Plane Shear

Design for out-of-plane shear is in accordance with the requirements of ACI 349, Chapter 11.

3.8.4.5.5.4 Evaluation for Thermal Loads

Thermal loads are combined with other loads according to the load combinations defined in Table 3.8.4-2. The effect of concrete cracking is considered in the stiffness properties for the concrete elements subjected to the thermal loads. Thermal forces and moments are calculated by multiplying linear elastic thermal stress analysis results by a stiffness reduction ratio α. The stiffness reduction ratio α is calculated as the ratio of the cracked stiffness to the elastic stiffness of the section to be evaluated.

3.8.4.5.5.5 Design of Shear Studs and Tie Bars

[*The shear stud connectors and tie bars are sized to carry the horizontal shear at the junction of the steel faceplates and the concrete infill. The shear stud and tie bar design is such that the shear stud and tie bar strength is sufficient to develop the full yield strength of the steel plate in approximately 3 x the thickness of the wall, or about 9 feet.*

The tie bars provide a structural framework for the modules, maintain the separation between the faceplates, support the modules during transportation and erection, and act as "form ties" between the faceplates when concrete is being placed. The tie bars provide additional shear capacity between the steel plates and concrete as well as additional strength similar to that provided by stirrups in reinforced concrete. The area and spacing of the tie bars satisfy the requirement for minimum shear reinforcement for beams given in the ACI codes. The connection between the tie bars and the steel faceplates is designed to develop 125% of the specified yield strength of the tie bar.]*

3.8.4.5.5.6 Design of Connections

The shield building steel plate modules are connected to standard reinforced concrete using a connection design satisfying the requirements outlined below. The locations of these connections include the following:

1. At the base of the shield building wall into the concrete below

2. Between SC and RC portions of the shield building wall

3. Where RC floors, roofs, and walls connect to the SC modules.

The steel plate modules are connected to the reinforced concrete by reinforcement or deformed bars directly connected to the modules. These connections are sized using a strength design approach, and the mechanical connection portion of the connection satisfies the requirements of ANSI/AISC N690. In the area of the connections, the faceplates are thicker (up to 1.0-inch nominal thickness), as necessary, to address loads from the connections. The connection design satisfies the following requirements:

- The connections provide for the direct transfer of forces from the RC reinforcing steel to the SC faceplates. Lap splice type connections between dowels extending out of the RC into the SC adjacent to the headed shear studs are not used to connect the modules to the base concrete.

- Loads are transferred directly from the faceplates to the reinforced concrete using reinforcing bars, mechanical connection, and welds.

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- The mechanical connections develop 125 percent of specified yield of the connected reinforcement bar.

- Loads that are out of plane of the wall, such as from roofs or walls, are carried through the full thickness of the shield building walls.

Figures 1, 2, 3, and 4 in APP-GW-GLR-602 (Reference 57) show the typical design details for the connection of the wall modules to the reinforced concrete. Figure 5 in Reference 57 shows the typical design details of the shield building concrete-filled steel module walls.

Figure 3H.5-7 and Figure 7 in Reference 57 show typical auxiliary building reinforced concrete roof connections to the shield building SC wall. The details of the connections between the auxiliary building roof and the shield building wall vary because of loads on the connection and the orientation of the wall to the roof reinforcement arrangement as outlined in the notes on Figure 3H.5-7 and in Section 4 of Reference 57. Figure 3H.5-7 also shows typical connections between the auxiliary building walls and the shield building SC wall modules. The connection elements used and the design details vary because of the orientation of the auxiliary building wall to the shield building wall. These connection design details satisfy the requirements of ANSI/AISC N690 and the requirements identified above.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

This subsection contains information relating to the materials, quality control program, and special construction techniques used in the construction of the other seismic Category I structures, as well as the containment internal structures. An edition of the referenced specifications applicable after the start of construction or procurement activities will be used.

3.8.4.6.1 Materials

3.8.4.6.1.1 Concrete

[*The compressive strength of concrete used in the seismic Category I structures and containment internal structures is f_c' = 4000 psi except as noted in the following. For the nuclear island basemat (the nominal 6 ft. thick foundation described in Subsection 3.8.5.1) the compressive strength of concrete is f_c' = 5000 psi. For the SC composite portion of the shield building structure including the connection region below the SC/RC interface and the shield building roof, the compressive strength of concrete is f_c' = 6000 psi.*]* The test age of concrete containing pozzolan is up to 56 days. The test age of concrete without pozzolan is up to 28 days. Concrete is mixed, batched, and placed according to Reference 6, Reference 7, and ACI-349.

Portland cement conforms to Reference 8, Type II. Certified copies of mill test reports showing that the chemical composition and physical properties conform to the specification are obtained for each cement delivery.

Aggregates conform to Reference 9. The fineness modulus of fine aggregate (sand) is not less than 2.3, nor more than 3.1, per Reference 9. In at least four of five successive test samples, such modulus is not allowed to vary more than 0.20 from the moving average established by the last five tests. Coarse aggregates may be rejected if the loss from the Los Angeles abrasion test, Reference 10, using Grading A or Reference 11, exceeds 40 percent by weight at 500 revolutions. Acceptance of source and aggregates is based on the tests specified in Table 3.8.4-3 as applicable based on type of aggregate and ACI-349 requirements.

Water and ice used in mixing concrete do not contain more than 250 parts per million of chlorides (as Cl) as determined in accordance with Reference 12. They do not contain more than 2000 parts per

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million of total solids as determined in accordance with Reference 13. Water meets the criteria in Table 3.8.4-4 in regard to the effects of the proposed mixing water on hardened cement pastes and mortars compared with distilled water.

The concrete contains an air entraining admixture when required by ACI-349 and may contain pozzolans and a water-reducing admixture. A viscosity modifying admixture (VMA) may be used in self-consolidating concrete (SCC) for adjusting the viscosity and improving its stability as outlined in ACI 237R, “Self-Consolidating Concrete.” Admixtures, except pozzolan, are stored in liquid solution.

Admixture types A, B, C, and F are used as noted herein to produce quality concrete. The admixtures shall be proven compatible with all constituents and other admixtures in mix prior to use in production of concrete. Furthermore, all admixtures shall be measured, introduced into the mix, and used in strict accordance to the manufacturer’s recommendations.

Admixtures do not contain added chlorides except as contained in potable drinking water used for manufacture of the admixtures. The chloride content is stated in the manufacturer's material certification.

Pozzolan conforms to Reference 14, except that the ignition loss does not exceed 6 percent.

Pozzolan is sampled and tested in accordance with Reference 15 for source approval.

Air entraining admixture conforms to Reference 16.

Water-reducing admixtures conform to Reference 17 and are of types A or F. A type A admixture may be used to increase the slump for conventional mixes (may use type F, optional). A type F admixture shall be used for SCC admixtures.

Retarding admixtures conform to Reference 17 and are of type B. Type B admixtures may be used to retard set times for conventional and SCC mixes (i.e., for mass concrete).

Accelerating admixtures conform to Reference 17 and are type C. Type C admixtures may be used to accelerate the set time for conventional and SCC mixes (i.e., for thin members placed in cold weather).

Manufacturer's certification for the air entraining admixture is required demonstrating compliance with Reference 16, Section 4 requirements.

Manufacturer's certification for the water-reducing admixture is required demonstrating compliance with Reference 17, Section 5 requirements.

Manufacturer's test reports are required for each delivery of pozzolan showing the chemical composition and physical properties and certifying that the pozzolan complies with the specification.

Proportioning of the concrete mix is in accordance with Reference 18 and Option B of Reference 6.

A testing laboratory designs and tests the concrete mixes. Only mixes meeting the design requirements specified for concrete are used.

Forms for concrete are designed as recommended in ACI 347.

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3.8.4.6.1.2 Reinforcing Steel

[*Reinforcing bars for concrete are deformed bars according to Reference 19, Grade 60, and Reference 20.*]* Certified material test reports are provided by the supplier for each heat of reinforcing steel delivered showing physical (both tensile and bend test results) and chemical analysis. In addition, a minimum of one tensile test is performed for each 50 tons of each bar size produced from each heat of steel.

In areas where reinforcing steel splices are necessary and lap splices are not practical, welded splices and other mechanical connections are acceptable.

[*Headed reinforcement meeting the requirements of ASTM A970 (Reference 49) is used where mechanical anchorage is required,*]* such as for shear reinforcement in the nuclear island basemat and in the exterior walls below grade.

As stated in Subsection 3.4.1.1.1, seismic Category I structures that are located below grade elevation are protected against flooding by a waterproofing system. This, in conjunction with the ACI 349 Code requirements for concrete cover for exposure to earth or weather for the reinforcing steel, provides sufficient protection for the reinforcing steel. Therefore, the use of coated reinforcing steel is not planned.

3.8.4.6.1.3 Structural Steel

Basic materials used in the structural and miscellaneous steel construction conform to the ASTM standards listed in Table 3.8.4-6. The table includes the major structural steel materials of construction. Materials used in limited applications and minor amounts may not be included. [*Additional information on structural and reinforcing steel for the shield building is provided in Table 1 of APP-GW-GLR-602 (Reference 57).*]*

3.8.4.6.1.4 Masonry Walls

There are no safety-related masonry walls used in the nuclear island.

3.8.4.6.2 Quality Control

The quality assurance program is described in Chapter 17. Conformance to Regulatory Guide 1.94 is as described in Section 1.9.

3.8.4.6.3 Special Construction Techniques

Construction techniques for the structural modules are the same as special construction techniques for the containment internal structures discussed previously in Subsection 3.8.3.6.1. Design-specific tolerances for alignment of the faceplates at welds connecting the shield building structural wall modules permit up to 1/4" offset in the alignment of the faceplates. The separation between the faying surfaces of the shield building wall modules butt joints landing on a backing bar is allowed up to 1/4". These design-specific tolerances modify plate alignment tolerances and faying surface separation requirements specified in AWS D1.1 (subsections 5.22.1.1 and 5.22.3) and are used in conjunction with a weld configuration that is designed for the larger alignment offset and the larger backing bar gap. The shield building wall design using the modified weld joint is in conformance with applicable requirements of ACI 349 and ANSI/AISC N690.

Where out-of-plane shear reinforcement is provided within the radial/circumferential reinforcement pattern for the main steam line anchor in auxiliary building Wall 11 (Figure 3H.5-5), tolerance on

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shear reinforcement is specified to maintain the maximum spacing requirements of ACI 349-01, and the minimum spacing requirements of ACI 318-11 for headed reinforcement.

3.8.4.7 Testing and In-Service Inspection Requirements

Structures supporting the passive containment cooling water storage tank on the shield building roof will be examined before and after first filling of the tank.

- The boundaries of the passive containment cooling water storage tank and the shield building roof above the tension ring at the intersection of the shield building roof and the shield building cylinder will be inspected visually for excessive concrete cracking before and after first filling of the tank. Any significant concrete cracking will be documented and evaluated in accordance with ACI 349.3R-96 (Reference 50). The structure around the passive containment cooling water storage tank will be inspected for water leaking out of the tank through the concrete.

- The vertical elevation of the passive containment cooling water storage tank relative to the top of the shield building cylindrical wall at the tension ring will be measured before and after first filling. The change in relative elevation will be compared against the predicted deflection.

- A test will be performed to measure the leakage from the passive containment cooling water storage tank based on measuring the water flow out of the leak chase collection system.

- A report will be prepared summarizing the test and evaluating the results.

During the operation of the plant, the condition of these structures should be monitored by the Combined License holder to provide reasonable confidence that the structures are capable of fulfilling their intended functions. The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.4.8 Construction Inspection

Construction inspection is conducted to verify the concrete wall thickness and quantity of concrete reinforcement. The construction inspection includes concrete wall thickness and reinforcement expressed in units of in^2/ft (linear length) equivalent when compared to standard reinforcement bar sections. Inspections will be measured at applicable sections excluding designed openings or penetrations. Inspections will confirm that each applicable section provides the minimum required reinforcement, steel plate thickness, and concrete thickness. The minimum required reinforcement, steel plate thickness, and concrete thickness represent the minimum values to meet the design basis loads. Appendix 3H also indicates the reinforcement provided which may exceed the minimum required reinforcement for the following reasons:

- Structural margin
- Ease of construction
- Use of standardized reinforcement sizes and spacing

A shield building construction mockup program will be used to build full-scale replicas of areas of the shield building that present critical construction areas identified as challenging areas of construction. These mockups provide an opportunity to apply and evaluate alternate and innovative construction and inspection methods and procedures. The fabrication, assembly, and erection of a full-scale mockup provides the accurate physical representation needed to evaluate true working conditions, physical configuration, accessibility, and quality control issues that may be encountered in construction. Construction practices review and examination of the mockups will be used to confirm

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the adequacy of construction means, methods, and procedures. If defects are found, the procedures will be revised and the mockup repeated until the required result is obtained. The major tasks that will be performed on each mockup include the following: field performance testing of the quality of concrete mixes, methods of concrete placement, inspections and surveillance, and post-placement activities.

In addition to these important process control tasks performed on the mockups, an inspection program will be undertaken on the AP1000 construction site mockups that uses the enhanced shield building design. Both visual inspection and non-destructive examination (NDE) will be performed for assessing defects that may impact structural integrity, such as crack distress, deterioriation caused by honeycomb, voids, and delaminations. The NDE inspection of the mockups will be performed at the first construction site of an AP1000 plant with an enhanced shield building design, and mockups applicability will be evaluated for the subsequent AP1000 site. This is to demonstrate the construction quality process control for concrete placement, and develop and document insights and requirements for corrective action, if required, to be used in the construction inspection program for all AP1000 plants.

3.8.5 Foundations

3.8.5.1 Description of the Foundations

The nuclear island structures, consisting of the containment building, shield building, and auxiliary building are founded on a common 6-foot-thick, cast-in-place, reinforced concrete basemat foundation. The top of the foundation is at elevation 66′-6″.

[*The turbine building and annex building are structurally separated from the nuclear island structures by a 2-inch gap at and below the grade. A 3-inch minimum gap is provided above grade.*]* This provides space to prevent interaction between the nuclear island structures and the adjacent seismic Category II structures during a seismic event. The maximum relative seismic displacement between the roof of the nuclear island and the turbine and annex buildings is less than 2 inches. This results in a clearance (gap) between buildings greater than 1 inch during a seismic event. Therefore, there are no interactions between the nuclear island and adjacent seismic Category II buildings during a seismic event. The radwaste building(1) is separated from the nuclear island by a 2-inch gap at and below grade and a 4-inch gap above grade. Figure 3.8.5-1 shows the foundations for the nuclear island structures and the adjoining structures.

Resistance to sliding of the concrete basemat foundation is provided by passive soil pressure and soil friction. This provides the required factor of safety against lateral movement under the most stringent loading conditions.

For ease of construction, the foundation is built on a mud mat. The mud mat is lean, nonstructural concrete and rests upon the load-bearing soil. Waterproofing requirements are described in Subsection 3.4.1.1.1.

3.8.5.2 Applicable Codes, Standards, and Specifications

The applicable codes, standards, and specifications are described in Subsection 3.8.4.2.

1. It should be noted that an evaluation of the radwaste building was made to consider its impact on the nuclear island or collapse in the safe shutdown earthquake; it was concluded that it would not impair the integrity of the nuclear island (see Subsection 3.7.2.8.2).

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3.8.5.3 Loads and Load Combinations

Loads and load combinations are described in Subsection 3.8.4.3. The loads and load combinations used in the analysis are considered to be part of the method of evaluation. As described in Subsection 3.8.2.1.2, the bottom head of the steel containment vessel is the same as the upper head and is capable of resisting the containment internal pressure without benefit of the nuclear island basemat. However, containment pressure loads affect the nuclear island basemat since the concrete is stiffer than the steel head. The containment design pressure is included in the design of the nuclear island basemat as an accident pressure in load combinations 5, 6, and 7 of Table 3.8.4-2. In addition to the load combinations described in Subsection 3.8.4.3, the nuclear island is checked for resistance against sliding and overturning due to the safe shutdown earthquake, winds and tornados, and against flotation due to floods and groundwater according to the load combinations presented in Table 3.8.5-1.

The effect of the groundwater level has been studied extensively. For the AP1000, Westinghouse has performed a time history analysis using a saturated and unsaturated soft-medium soil profile (Poisson's ratio = 0.35) and compared the floor response spectra of the two analyses. Generic SSI analyses for the AP1000 assume the water table to be at grade level with saturated soil properties supporting the nuclear island. The unsaturated soil profile was produced where the water table was assumed to be well below the nuclear island.

The results of this analysis concluded that the depth of the water table used for SSI analyses has a negligible effect on the floor response spectra at the key nodes. Since the floor response spectra differences between the two models are negligible, no additional analyses are required to compare member forces or deformations.

3.8.5.4 Design and Analysis Procedures

The seismic Category I structures are concrete, shear-wall structures consisting of vertical shear/bearing walls and horizontal floor slabs. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs. The walls then transmit the loads to the basemat. The walls also provide stiffness to the basemat and distribute the foundation loads between them.

The design of the basemat consists primarily of applying the design loads to the structures, calculating shears and moments in the basemat, and determining the required reinforcement. For a site with hard rock below the underside of the basemat vertical loads are transmitted directly through the basemat into the rock. Horizontal loads due to seismic are distributed on the underside of the basemat resulting primarily in small membrane forces in the mat. The 6-foot-thick basemat is designed for the upward hydrostatic pressure due to groundwater reduced by the downward deadweight of the mat.

[*Seismic loads for the evaluation of the basemat of the Nuclear Island are developed from the results of the global seismic analyses on hard rock. They are specified as equivalent static accelerations.*

The equivalent static accelerations used in the non-linear design analyses of the nuclear island basemat are evaluated for the revised design with the enhanced shield building by comparing total base reactions and bearing pressures in a linear analysis using these equivalent static accelerations to those from a dynamic analysis of the nuclear island 3D finite element model. A time history fixed base analysis of the model is performed using time history inputs that envelope the basemat response given by the 3D SASSI analyses at the corners and centers of the basemat for all the specified generic soil cases.]*

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The basemat reactions for the equivalent static analyses compare well against those of the “all soils” time history. [*Bearing pressures are calculated from the basemat reactions assuming a rigid basemat for dead, live, and seismic loads. Seismic loads are considered using the 1.0, 0.4, 0.4 combination method.*]* The bearing pressures resulting from the equivalent static accelerations are similar to those due to the “all soils” time history analysis demonstrating the adequacy of the equivalent static accelerations applied in the basemat analyses.

3.8.5.4.1 Analyses for Loads during Operation

The analyses of the basemat use the three-dimensional ANSYS finite element models of the auxiliary building and containment internal structures, which are described in Subsections 3G.2.1.1 and 3G.2.1.2 and shown in Figures 3G.2-1 and 3G.2-2. The model considers the interaction of the basemat with the overlying structures and with the soil. Provisions are made in the model for two possible uplifts. One is the uplift of the containment internal structures from the lower basemat. The other is the uplift of the basemat from the soil.

The three-dimensional finite element model of the basemat includes the structures above the basemat and their effect on the distribution of loads on the basemat. The finite element model of the basemat is shown on sheet 1 of Figure 3.8.5-2.

The subgrade is modeled with one vertical spring and two horizontal springs at each node of the basemat. The vertical springs act in compression only. The horizontal springs are active when the vertical spring is closed and inactive when the vertical spring lifts off. The analyses of the basemat accounted for the range of soil sites described in Section 2.5. Horizontal bearing reactions on the side walls below grade are conservatively neglected.

The nuclear island basemat below the containment vessel, and the containment internal structures basemat above the containment vessel, are simulated with solid tetrahedral elements. Nodes on the two basemats are connected with spring elements normal to the theoretical surface of the containment vessel.

Normal and extreme environmental loads and containment pressure loads are considered in the analysis. The normal loads include dead loads and live loads. Extreme environmental loads include the safe shutdown earthquake.

Dead loads are applied as inertia loads. Live loads and the safe shutdown earthquake loads are applied as concentrated loads on the nodes. The safe shutdown earthquake loads are applied as equivalent static loads using the assumption that while maximum response from one direction occurs, the responses from the other two directions are 40 percent of the maximum. Combinations of the three directions of the safe shutdown earthquake are considered.

Linear analyses are performed for all specified load combinations assuming that the soil springs can take tension. Critical load cases are then selected for non-linear analyses with basemat liftoff based on the results of the linear cases. The results from the analysis include the forces, shears, and moments in the basemat; the bearing pressures under the basemat; and the area of the basemat that is uplifted. Reinforcing steel areas are calculated from the member forces for each load combination case.

The required reinforcing steel for the portion of the basemat under the auxiliary building and under the shield building is determined by considering the reinforcement envelope for the full non-linear iteration of the most critical load combination cases. Additional reinforcement is provided in the design of the 6′ mat for soil sites such that the basemat can resist loads 20 percent greater than the demand calculated by the equivalent static acceleration analyses on uniform soil springs. This

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increase accommodates potential site specific lateral variability of the soil investigated separately in a series of parametric studies. Figure 3.8.5-3 shows the basemat reinforcement.

3.8.5.4.2 Analyses of Settlement During Construction

Construction loads are evaluated in the design of the nuclear island basemat. This evaluation is performed for soil sites meeting the site interface requirements of Subsection 2.5.4 at which settlement is predicted to be maximum. In the expected basemat construction sequence, concrete for the mat is placed in a single placement. Concrete may be placed below the sumps and elevator pits, if determined to enhance constructability, prior to the remaining single pour of the nuclear island basemat. The placement includes the first 6 feet of the thicker basemat below the containment vessel and shield building, but excludes the central zone directly below the bottom of the containment vessel. Construction continues with a portion of the shield building foundation and containment internal structure and the walls of the auxiliary building. The critical location for shear and moment in the basemat is around the perimeter of the shield building. Once the shield building and auxiliary building walls are completed to elevation 82′-6″, the load path changes and loads are resisted by the basemat stiffened by the shear walls.

The analyses account for the construction sequence, the associated time varying load and stiffness of the nuclear island structures, and the resulting settlement time history. To maximize the potential settlement, the analyses consider a 360 feet deep soft soil site with soil properties consistent with the soft soil case described in Subsection 2A.2. Two soil profiles are analyzed to represent limiting foundation conditions, and address both cohesive and cohesionless soils and combinations thereof:

- A soft soil site with alternating layers of sand and clay. The assumptions in this profile maximize the settlement in the early stages of construction and maximize the impact of dewatering.

- A soft soil site with clay. The assumptions maximize the settlement during the later stages of construction and during plant operation.

The analyses focus on the response of the basemat in the early stages of construction when it could be susceptible to differential loading and deformations. As subsequent construction incorporates concrete shear walls associated with the auxiliary building and the shield building, the structural system significantly strengthens, minimizing the impact of differential settlement. The displacements, and the moments and shear forces induced in the basemat are calculated at various stages in the construction sequence. These member forces are evaluated in accordance with ACI 349 using the load factors given in Table 3.8.4-2. Three construction sequences are examined to demonstrate construction flexibility within broad limits.

- A base construction sequence which assumes no unscheduled delays. The site is dewatered and excavated. Concrete for the basemat is placed. Concrete for the exterior walls below grade is placed after the basemat is in place. Exterior and interior walls of the auxiliary building are placed in 16 to 18-foot lifts.

- A delayed shield building case which assumes a delay in the placement of concrete in the shield building while construction continues in the auxiliary building. This bounding case maximizes tension stresses on the top of the basemat. The delayed shield building case assumes that no additional concrete is placed in the shield building after the pedestal for the containment vessel head is constructed. The analysis incorporates construction in the auxiliary building to elevation 117′-6″ and filling the CA20 module with concrete to elevation 135′-3″, and thereafter assumes that construction is suspended.

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- A delayed auxiliary building case which assumes a delay in the construction of the auxiliary building while concrete placement for the shield building continues. This bounding case maximizes tension stresses in the bottom of the basemat. There are two construction sequence limitations from this case:

  – The first construction sequence limitation assumes a delayed auxiliary building case in which construction can continue in the shield building to elevation 139′-6″. Shield building construction may proceed up to 139′-6″ with the following requirements and exceptions:

    - Auxiliary building walls must be completed up to an elevation of 82′-6″. Completion of auxiliary building 82′-6″ floors is not required. Completion of module CA20 is not required provided that the steel portions are constructed.

    - Wall N is an exception to the completion of the auxiliary building walls to elevation 82′-6″. This wall does not have to be constructed in order for the shield building to proceed up to 139′-6″.

    - Construction of module CA01 is limited to steel only.

    - No concrete is placed in the SC shield building walls above elevation 131′-6″.

    - Construction in the RC portion of the shield building between AZ 181.98° and AZ 340.00° may proceed to an elevation of 141′-6″ provided that no concrete is placed above elevation 100′-0″.

    - Shield building construction may proceed to an elevation of 149′-6″ between AZ 181.98° and AZ 183.01° to permit installation of the RC/SC connection panel. No concrete is placed in this panel.

    - Containment vessel rings 1 and 2 may be placed.

    - Modules CA02, CA03, CA04, and CA05 may be fully constructed.

  – The second construction sequence limitation assumes the auxiliary building is complete to 82′-6″ excluding floors, and that the concrete has been placed in module CA20 walls to elevation 128′-0″ except for the north end of the east and west walls of the spent fuel pool, which has been placed to elevation 87′-3″. Construction in the shield building may proceed up to elevation 159′-6″, with the following requirements and exceptions:

    - Shield building concrete elevation must remain at or below elevation 149′-6″.

    - Wall N is an exception to the completion of the auxiliary building walls to elevation 82′-6″. This wall does not have to be constructed in order for the shield building concrete to continue to 149′-6″.

    - The containment vessel ring 3 and top head are not placed.

    - Module CA01 may be fully constructed.

  – Resumption in construction of the shield building above elevation 159′-6″ can proceed once the auxiliary building is completed to elevation 100′-0″, including all walls and 100′-0″ elevation floors.

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For the base construction sequence, the largest basemat moments and shears occur at the interface with the shield building before the connections between the auxiliary building and the shield building are credited. Once the shield building and auxiliary building walls are completed to elevation 82′-6″, the load path for successive loads changes and the loads are resisted by the basemat stiffened by the shear walls. Dewatering is discontinued once construction reaches grade, resulting in the rebound of the subsurface.

Of the three construction scenarios analyzed, the delayed auxiliary building case results in the largest demand for the bottom reinforcement in the basemat. The delayed shield building results in the largest demand for the top reinforcement in the basemat. The analyses of the three construction sequences demonstrate the following:

- The design of the basemat and superstructure accommodates the construction-induced stresses considering the construction sequence and the effects of the settlement time history.

- The design of the basemat can accommodate delays in the shield building so long as the auxiliary building construction is suspended at elevation 117′-0″. Resumption in construction of the auxiliary building can proceed once the shield building is advanced to elevation 100′-0″.

- The design of the basemat can accommodate delays in the auxiliary building so long as the shield building construction is suspended at the elevation limits as previously discussed in the delayed auxiliary building case.

- After the structure is in place and cured to elevation 100′-0″, the basemat and structure act as an integral 40 foot deep structure and the loading due to construction above this elevation is not expected to cause significant additional flexural demand with respect to the basemat and the shield building concrete below the containment vessel. Accordingly, there is no need for placing constraints on the construction sequence above elevation 100′-0″.

The site conditions considered in the evaluation provide reasonable bounds on construction induced stresses in the basemat. Accordingly, the basemat design is adequate for practically all soil sites and it can tolerate major variations in the construction sequence without causing excessive deformations, moments and shears due to settlement over the plant life.

The analyses of alternate construction scenarios considered the softest material properties satisfying the shear wave velocity limit of 1000 feet per second. These analyses show that member forces in the basemat are acceptable subject to the limits shown below on the relative level of construction of the buildings. Construction of the AP1000 will satisfy the limits shown below, or a site-specific analysis of settlement and member forces will be completed. These limits do not apply to AP1000 units with a soil profile where the shear wave velocity exceeds 7500 feet per second.

Prior to completion of the shield building at elevation 82′-6″:

- Concrete may not be placed above elevation 117′-6″ in the auxiliary building, except in the CA20 structural module, where it may be placed to elevation 135′-3″.

Member forces in the basemat considering settlement during construction differ from those obtained from the analyses on uniform elastic soil springs described in Subsection 3.8.5.4.1.

Although the bearing pressures at the end of construction are similar in the two analyses, the resulting member forces differ due to the progressive changes in structural configuration during construction. The design using the results of the analyses of Subsection 3.8.5.4.1 provides sufficient structural strength to resist the specified loads including bearing reactions on the underside of the

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basemat. The member forces in these analyses are those due to primary externally applied loads and do not consider secondary stresses and strains locked in during early stages of construction. A confirmatory evaluation was performed to demonstrate that the member forces due to design basis loads, including locked-in forces due to construction settlement, remain within the capacity of the section. The evaluation was performed for critical locations which were selected as locations where the effect of locked in member forces were judged to be most significant.

The governing scenario is the case with a delay in the auxiliary building construction for the soft soil site with alternating layers of sand and clay. The delay is postulated to occur just prior to the stage where the auxiliary building walls are constructed. Member forces at the end of construction are calculated considering the effects of settlement during construction. The difference in these member forces from those calculated for dead load in the analyses on soil springs are added as additional dead loads in the critical safe shutdown earthquake load combination.

The member forces for the load combination of dead load plus safe shutdown earthquake, including the member forces locked-in during various stages of plant construction, are within the design capacity for the five critical locations. The evaluation demonstrates that the member forces including locked-in forces calculated by elastic analyses remain within the capacity of the section.

3.8.5.4.3 Design Summary Report

A design summary report is prepared for the basemat documenting that the structures meet the acceptance criteria specified in Subsection 3.8.5.5.

Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Sections 3.7 and 3.8 provided the following acceptance criteria are met.

- The structural design meets the acceptance criteria specified in Section 3.8
- The seismic floor response spectra meet the acceptance criteria specified in Subsection 3.7.5.4

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

3.8.5.4.4 Design Summary of Critical Sections

The basemat is designed to meet the acceptance criteria specified in Subsection 3.8.4.5. Two critical portions of the basemat are identified below together with a summary of their design. The boundaries are defined by the walls and column lines which are shown in Figure 3.7.2-12 (sheet 1 of 12). Table 3.8.5-3 shows the reinforcement required and the reinforcement provided for the critical sections.

[*Basemat between column lines 9.1 and 11 and column lines K and L*

This portion of the basemat is designed as a two way slab with the shorter directions spanning a distance of 23′ 6″ between the walls on column lines K and L. The slab is continuous with the adjacent slabs to the east and west. The critical loading is the bearing pressure on the underside of the slab due to dead and seismic loads. This establishes the demand for the top flexural reinforcement at mid span and for the bottom flexural and shear reinforcement at the walls. The basemat is designed for the member forces from the analyses]* described in Subsection 3.8.5.4.1. [*The top and bottom reinforcement in the east west direction of span are equal. The reinforcement*

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provided is shown in sheets 1, 2 and 5 of Figure 3.8.5-3. Typical reinforcement details showing use of headed reinforcement for shear reinforcement are shown in Figure 3H.5-3.

Basemat between column lines 1 and 2 and column lines K-2 and N

This portion of the basemat is designed as a two way slab with the shorter direction spanning a distance of 22′ 0″ between the walls on column lines 1 and 2. The slab is continuous with the adjacent slabs to the north and with the exterior wall to the south. The critical loading is the bearing pressure on the underside of the slab due to dead and seismic loads. This establishes the demand for the top flexural reinforcement at mid span and for the bottom flexural and shear reinforcement at wall 2. The basemat is designed for the member forces from the analyses on uniform soil springs]* described in Subsection 3.8.5.4.1. [*The reinforcement provided is shown in sheets 1, 2 and 5 of Figure 3.8.5-3. Typical reinforcement details showing use of headed reinforcement for shear reinforcement are shown in Figure 3H.5-3.*]*

Figure 3.8.5-3 shows minimum design reinforcement for the concrete in the areas defined. This location information does not define the exact length or position of the reinforcement bars. The detailed reinforcement design, including requirements for detailing and concrete cover, is in conformance with ACI-349 requirements for foundations. The reinforcement arrangement implemented may have a greater amount of reinforcement than that shown in the figure. The size and spacing of the reinforcement bars may vary from that shown as long as the reinforcement area provided is equal to or greater than the area for the size and spacing identified in the figure. Incorporation of embedments, drains and other piping, support structures, and other obstructions may result in local variations from the reinforcement arrangement shown. Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Sections 3.7 and 3.8 provided the following acceptance criteria are met.

- The structural design meets the acceptance criteria specified in Section 3.8.
- The amplitude of the seismic floor response spectra does not exceed the design basis floor response spectra by more than 10 percent.

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design.

3.8.5.5 Structural Criteria

The analysis and design of the foundation for the nuclear island structures are according to ACI-349 with margins of structural safety as specified within it. The limiting conditions for the foundation medium, together with a comparison of actual capacity and estimated structure loads, are described in Section 2.5. The minimum required factors of safety against sliding, overturning, and flotation for the nuclear island structures are given in Table 3.8.5-1.

[*The design and construction of anchors and embedments conform to the procedures and standards of Appendix B to ACI 349-01*]* and are in conformance with the regulatory positions of NRC Regulatory Guide 1.199, Revision 0. [*Alternative requirements to ACI 349, Appendix B apply to the anchoring of headed shear reinforcement for the basemat (see Subsection 3.8.4.4.1).*]*

[*The basemat below the auxiliary building is designed for shear in accordance with the following supplemental provisions which are based on requirements for continuous deep flexural members in ACI 349-01. As permitted by paragraph 11.5.5.1 of ACI 349-01, shear reinforcement is not provided when the factored shear force, V_u, is less than one half of the shear strength provided by the concrete, φV_c.*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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- *The design for shear is based on 11.1 through 11.5 of ACI 349-01 except that the critical section measured from the face of the support is taken at a distance of 0.15 l_n.*

- *Shear strength, V_n, is not taken greater than $8\sqrt{f_c'}\, b_w d$ when l_n/d is less than 2.*
  – *When l_n/d is between 2 and 5, $V_n = 2/3\ (10 + l_n/d)\ \sqrt{f_c'}\, b_w d$*

- *Minimum vertical shear reinforcement is provided in each bay. The area of vertical shear reinforcement, A_v, is not less than $0.0015\ b_w s$*
  – *Spacing of shear reinforcement, s, based on provisions in Paragraph 11.5.4.1 does not exceed d/2, nor 24 in.*

- *Shear reinforcement required at the critical section is used throughout the span.*

The terms φ, d, l_n, V_c, A_v, b_w, s, and f_c' are defined in ACI 349.]*

3.8.5.5.1 Nuclear Island Maximum Bearing Pressures

The foundation will be demonstrated to be capable of withstanding the bearing demand from the nuclear island as described in Subsection 2.5.4.2.

3.8.5.5.2 Flotation

The factor of safety against flotation of the nuclear island is shown in Table 3.8.5-2 and is calculated as follows:

$$\text{F.S.} = \frac{\text{D}}{(\text{F or B})}$$

where:

F.S. = factor of safety against flotation

D = total weight of structures and foundation

F = buoyant force due to the design basis flood

B = buoyant force due to high ground water table

3.8.5.5.3 Sliding

The factor of safety against sliding of the nuclear island during a tornado or a design wind is shown in Table 3.8.5-2 and is calculated as follows:

$$\text{F.S.} = \frac{\text{F}_\text{S}}{\text{F}_\text{H}}$$

where:

F.S. = factor of safety against sliding from tornado or design wind

F_S = shearing or sliding resistance at bottom of basemat

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F_H = maximum lateral force due to active soil pressure, including surcharge, and tornado or design wind load

The factor of safety against sliding of the nuclear island during a safe shutdown earthquake is shown in Table 3.8.5-2 and is calculated as follows:

$$F.S. = \frac{F_S}{F_D}$$

where:

F.S. = factor of safety against sliding from a safe shutdown earthquake

F_S = shearing or sliding resistance at bottom of basemat

F_D = seismic force from safe shutdown earthquake

The sliding resistance is based on the friction force developed between the basemat and the foundation with a static coefficient of friction of 0.55. The governing friction value in the soil below the mudmat has an angle of internal friction of 35°. The effect of buoyancy due to the water table is included in calculating the sliding resistance. Passive soil pressure resistance is not included in the equations above because passive pressure is not considered for sliding stability. Since there is no passive pressure considered, active and overburden soil pressures are also not considered.

3.8.5.5.4 Overturning

The factor of safety against overturning of the nuclear island during a tornado or a design wind is shown in Table 3.8.5-2 and is calculated as follows:

$$F.S. = \frac{M_R}{M_O}$$

where:

F.S. = factor of safety against overturning from tornado or design wind

M_R = resisting moment

M_O = overturning moment of tornado or design wind

The factor of safety against overturning of the nuclear island during a safe shutdown earthquake is shown in Table 3.8.5-2 and is evaluated using the time history analysis assuming overturning about the edge of the nuclear island at the bottom of the basemat. The factor of safety is defined as follows:

$F.S. = (M_R)/(M_O + M_{AO})$

where:

F.S. = factor of safety against overturning from a safe shutdown earthquake

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M_R = nuclear island's resisting moment against overturning

M_O = maximum safe shutdown earthquake induced overturning moment acting on the nuclear island, applied as a static moment

M_{AO} = Moment due to lateral forces caused by active and overburden pressures

The resisting moment is equal to the nuclear island dead weight, minus buoyant force from ground water table, multiplied by the distance from the edge of the nuclear island to its center of gravity. The overturning moment is the maximum moment about the same edge from the time history analyses of the nuclear island NI20 model described in Subsection 3.7.2 and Appendix 3G.2. Resistance moment due to passive pressure is not included in the equation above because passive pressure is not considered for overturning stability.

3.8.5.5.5 Seismic Stability Analysis

The factors of safety for sliding and overturning for the SSE are calculated for each soil case for the base reactions in terms of shear and bending moments about column lines 1, 11, I, and the west side of the shield building at each time step of the seismic time history. The 2D SASSI reactions (Fx, Fy, and Fz) are used to obtain seismic response factors between the hard rock (HR) case to the UBSM soil case, and the SM soil case. These factors are used to adjust the hard rock time history to reflect the seismic response for the other two potential governing soil cases UBSM and SM. The firm rock, soft rock, and soft soil cases have higher factors of safety against sliding and, therefore, are not considered.

A non-linear analysis with sliding friction elements using a 2D ANSYS model was performed. The 2D ANSYS model that was used to study the basemat uplift (see Subsection 3.8.5.5.6 and Appendix 3G.2.2.5). This 2D non-linear model is for the east-west direction. There is no need to consider the north-south direction since the nuclear island deflections calculated to maintain a factor of safety of 1.1 are largest in the east-west direction. This model was modified introducing friction elements along the bottom of the basemat and soil media interface. Direct time integration analysis was performed with vertical uplift and sliding allowed. The three cases that have the lowest factor of safety related to sliding were evaluated. These three cases are HR, UBSM, and SM. The seismic input was increased by 10 percent to maintain the factor of safety against sliding of 1.1. No passive soil resistance is considered. The resulting maximum displacement at the base of the nuclear island basemat (elevation 60.5′) using a coefficient of friction of 0.55 is 0.12 inch without buoyant force consideration, and 0.19 inch with buoyant force considered. This is negligible sliding during the seismic event, and no passive soil resistance is necessary from the backfill (side soil). Therefore, it can be concluded that the nuclear island is stable against sliding.

The minimum seismic stability factors of safety values are reported in Table 3.8.5-2.

3.8.5.5.6 Effect of Nuclear Island Basemat Uplift on Seismic Response

The effects of basemat uplift were evaluated using an east-west lumped-mass stick model of the nuclear island structures supported on a rigid basemat with nonlinear springs (see Appendix 3G.2.2.5). Floor response spectra from safe shutdown earthquake time history analyses, which included basemat uplift, were compared to those from analyses that did not include uplift. The comparisons showed that the effect of basemat uplift on the floor response spectra is not significant.

The design analyses of the nuclear island basemat include consideration of sliding and liftoff between the containment internal structures and the containment vessel and of sliding between the containment vessel and the nuclear island basemat. Analyses of stability demonstrate that there was no uplift or sliding at the interface of the containment internal structures and the containment vessel.

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3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The materials and quality control program used in the construction of the nuclear island structures foundation are described in Subsection 3.8.4.6.

There are no special construction techniques used in the construction of the nuclear island structures foundation. Subsection 2.5.4.2 describes information related to the excavation, backfill, and mudmat.

3.8.5.7 In-Service Testing and Inspection Requirements

The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

The need for foundation settlement monitoring is site-specific as discussed in Subsection 2.5.4.3.

3.8.5.8 Construction Inspection

Construction inspection is conducted to verify the concrete wall thickness and quantity of concrete reinforcement. The construction inspection includes concrete wall thickness and reinforcement expressed in units of in^2/ft (linear length) equivalent when compared to standard reinforcement bar sections. Inspections will be measured at applicable sections excluding designed openings or penetrations. Inspections will confirm that each section provides the minimum required reinforcement and concrete thickness as shown in Table 3.8.5-3. The minimum required reinforcement and concrete thickness represent the required minimum values to meet the design basis loads. Table 3.8.5-3 also indicates the reinforcement provided which may exceed the required minimum reinforcement for the following reasons:

- Structural margin
- Ease of construction
- Use of standardized reinforcement sizes and spacing

3.8.6 Combined License Information

3.8.6.1 Containment Vessel Design Adjacent to Large Penetrations

The final design of containment vessel elements (reinforcement) adjacent to concentrated masses (penetrations) is addressed in APP-GW-GLR-005 (Reference 53), and the applicable changes are incorporated.

3.8.6.2 Passive Containment Cooling System Water Storage Tank Examination

Not used.

3.8.6.3 As-Built Summary Report

Not used.

3.8.6.4 In-Service Inspection of Containment Vessel

Not used.

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3.8.6.5 Structures Inspection Program

The structures inspection program to address maintenance requirements for the seismic Category I and seismic Category II structures is addressed in Subsections 3.8.3.7, 3.8.4.7, 3.8.5.7, and 17.6.

3.8.6.6 Construction Procedures Program

Construction and inspection procedures for concrete filled steel plate modules address activities before and after concrete placement, use of construction mock-ups, and inspection of modules before and after concrete placement as discussed in Subsection 3.8.4.8. The procedures will be made available to NRC inspectors prior to use.

3.8.7 References

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.2-1
Load Combinations and Service Limits for Containment Vessel

Load Description		Load Combination and Service Limit										
		Con	**Test**	**Des.**	**Des.**	**A**	**A**	**A**	**C**	**D**	**C**	**D**
Dead	D	x	x	x	x	x	x	x	x	x	x	x
Live	L	x	x	x	x	x	x	x	x	x	x	x
Wind(5)	W	x						x				
Safe shutdown earthquake	E_s								x	x		x
Tornado	W_t										x	
Test pressure	P_t		x									
Test temperature	T_t		x									
Operating pressure	P_o							x			x	
Design pressure	P_d			x			x		x			x
Design external pressure	P_e				x	x				x		
Normal reaction	R_o				x	x		x		x	x	
Normal thermal(4)	T_o				x	x		(3)		x	(3)	
Accident thermal reactions	R_a			x			x		x			x
Accident thermal	T_a			x			x		x			x
Accident pipe reactions	Y_r											x
Jet impingement	Y_j											x
Pipe impact	Y_m											x

Notes:
1. Service limit levels are per ASME-NE.
2. Where any load reduces the effects of other loads, that load is to be taken as zero, unless it can be demonstrated that the load is always present or occurs simultaneously with the other loads.
3. Temperature of vessel is 70°F.
4. Temperature distribution for normal operation in cold weather.
5. Wind load for the construction load combination is based on a 70 mph wind. Wind load for the Service Level A load combination is analyzed as a reduction in external pressure.

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Table 3.8.2-2
Containment Vessel Pressure Capabilities

Containment Element		Pressure Capability				
		Deterministic Severe Accident Capacity[1]			Maximum Pressure Capability[2]	
Temperature		100°F	300°F	400°F	100°F	400°F
Cylinder		135 psig	117 psig	112 psig	155 psig	129 psig
Ellipsoidal Head		104 psig	91 psig	87 psig	174 psig	144 psig
16-foot equipment hatch	F.S. = 1.67	126 psig	121 psig	118 psig	210 psig	198 psig
	F.S. = 2.50	84 psig	81 psig	79 psig		
Personnel airlocks[3]		>163 psig	>163 psig	>163 psig	>300 psig	>300 psig

Notes:
1. The buckling capacity of the ellipsoidal head is taken as 60 percent of the critical buckling pressure calculated by the BOSOR-5 nonlinear analyses; the buckling capacity at higher temperatures is calculated by reducing the capacity at 100°F by the ratio of yield at 100°F to yield at the higher temperature. Evaluations of the equipment hatch covers are shown both for ASME paragraph NE-3222 (F.S. = 2.50) and Code Case N-284 (F.S. = 1.67). Evaluations of the other elements are according to ASME Service Level C.
2. The estimated maximum pressure capability is based on minimum specified material properties.
3. The capacities of the personnel airlocks are estimated from test results.

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Table 3.8.2-3
Analysis and Test Results of Fabricated Heads
(Reference 23)

	Test Model #1	Test Model #2
Cylinder radius	96.0 inches	96.0 inches
Knuckle radius	32.64 inches	32.64 inches
Spherical radius	172.8 inches	172.8 inches
Thickness	0.196 inches	0.27 inches
Head height/radius	0.5	0.5
Radius/thickness	490	356
Test initial buckling pressure	58 psig	106 psig
Test collapse pressure	229 psig	332 psig
Collapse pressure/initial buckling pressure	3.95	3.13
BOSOR-5 predicted buckling pressure	73.6 psig	106.6 psig

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Table 3.8.2-4
Summary of Containment Vessel Models and Analysis Methods

Model	Analysis Method	Program	Purpose
Axisymmetric shell	Modal analysis	ANSYS	To calculate frequencies and mode shapes for comparison against stick model
Lumped mass stick model	Modal analysis	ANSYS	To create equivalent stick model for use in nuclear island seismic analyses
Axisymmetric shell	Static analyses using Fourier harmonic loads	ANSYS	To calculate containment vessel shell stresses
Axisymmetric shell	Nonlinear bifurcation	BOSOR5	To calculate buckling capacity close to base under thermal loads To calculate pressure capacity of top head
Finite element shell	Linear bifurcation	ANSYS	To study local effect of large penetrations and embedment on buckling capacity for axial and external pressure loads
Finite element shell	Modal analysis	ANSYS	To calculate frequencies and mode shapes for local effects of equipment hatches and personnel airlocks
Finite element shell	Static analyses	ANSYS	To calculate local shell stress in vicinity of the equipment hatches and personnel airlocks

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Table 3.8.2-5
Maximum Absolute Nodal Acceleration (ZPA)
Steel Containment Vessel

Hard Rock Site Condition

Elevation (ft)	Maximum Absolute Nodal Acceleration, ZPA (g)					
	N-S Direction		E-W Direction		Vertical Direction	
	Mass Center	Edge	Mass Center	Edge	Mass Center	Edge
281.90	1.48		1.56		1.25	
273.83	1.43		1.50		1.02	
265.83	1.38		1.43		0.85	
255.02	1.31		1.34		0.73	
244.21	1.23	1.28	1.26	1.30	0.68	0.71
224.00	1.09	1.13	1.11	1.17	0.66	0.68
200.00	0.90	0.94	0.94	0.98	0.61	0.63
169.93	0.69	0.71	0.72	0.75	0.53	0.55
162.00	0.63	0.65	0.67	0.68	0.51	0.53
141.50	0.49	0.50	0.54	0.54	0.45	0.47
131.68	0.43	0.44	0.47	0.48	0.41	0.44
112.50	0.40	0.41	0.37	0.38	0.35	0.40
104.12	0.38	0.40	0.38	0.40	0.32	0.38
100.00	0.38	0.40	0.39	0.41	0.31	0.34

Notes:
1. Enveloped response results at the north, south, east, and west edge nodes of the structure are shown. This is the maximum value of the response at any of these edge nodes.
2. Elevation 236'-6" is the mid-height of the polar crane bridge.

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Table 3.8.3-1
Shear and Flexural Stiffnesses of Structural Module Walls

Case	Analysis Assumption	Shear Stiffness[1],[2]				Flexural Stiffness[1],[2]			
		48" Wall		30" Wall		48" Wall		30" Wall	
		GA x 10^6 lbs	Ratio	GA x 10^6 lbs	Ratio	EI x 10^9 lbs. in^2	Ratio	EI x 10^9 lbs. in^2	Ratio
1	Monolithic section considering steel plates and uncracked concrete. For shear stiffness this is (A_c G_c + A_s G_s).	83.5	1.0	55.8	1.0	47.5	1.0	13.6	1.0
2	Uncracked gross concrete section (full wall thickness considering steel plate as concrete)	73.9	0.89	46.2	0.83	33.2	0.70	8.1	0.60
3	Transformed cracked section considering steel plates and concrete (no concrete tension stiffness)	25.0	0.30	22.6	0.41	22.1	0.47	8.0	0.59

Notes:
1. The shear stiffness, GA, is calculated for the full thickness of wall. The flexural stiffness is calculated per unit length of the wall.
2. Stiffness calculations are based on the following material properties: E_c = 3,605,000 psi,n = 8, v_c = 0.17, v_s = 0.30

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Table 3.8.3-2
Summary of Containment Internal Structures
Models and Analysis Methods

Computer Program and Model	Analysis Method	Purpose	Concrete Stiffness (1)
3D ANSYS finite element of containment internal structures	Static	To obtain the in-plane and out-of-plane mechanical forces for the design of floors and walls (dead, live, hydrostatic, pressure)	Monolithic Case 1 with E_c reduced by factor of 0.8
3D ANSYS finite element of containment internal structures	Response spectra analyses(2)	To obtain the in-plane and out-of-plane seismic forces for the design of floors and walls	Monolithic Case 1 with E_c reduced by factor of 0.8.
3D ANSYS finite element of containment internal structures	Static analyses	To obtain the in-plane and out-of-plane member forces for thermal loads	Cracked Case 3
The following AP600 analyses are used as background to develop the AP1000 design loads.			
3D ANSYS finite element of containment internal structures fixed at elevation 103′-0″	Harmonic analyses	To evaluate natural frequencies potentially excited by hydrodynamic loads	Uncracked Case 2
	Time history analyses	To obtain dynamic response of IRWST boundary for hydrodynamic loads	Monolithic and cracked Cases 1 & 3

Note:
1. See Table 3.8.3-1 for stiffness case description.
2. See Section 3.7 for discussion of the containment internal structures seismic analyses.

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Table 3.8.3-3
Definition of Critical Locations and Thicknesses for Containment Internal Structures(1)(4)

Wall Description (see detail in subsection 3.8.3.5.8.1)	Applicable Column Lines	Applicable Elevation Range	[*Concrete Thickness*(2)]*	Required Thickness of Surface Plates (inches)(3)	[*Thickness of Surface Plates Provided (inches)*]*
Containment Structures					
Module Wall 1	Southwest wall of refueling cavity	Wall separating IRWST and refueling cavity from elevation 103′ to 135′-3″	[*4′-0″ concrete-filled structural wall module with 0.5-in.-thick steel plate on inside and outside of wall*]*	0.12	[*0.5 -0.01 +0.1*]*
Module Wall 2	South wall of west steam generator compartment	Wall separating IRWST and west steam generator compartment from elevation 103′ to 135′-3″	[*2′-6″ concrete-filled structural wall module with 0.5-in.-thick steel plate on inside and outside of wall*]*	0.43	[*0.5 -0.01 +0.1*]*
CA02 Module Wall	North east boundary wall of IRWST	Wall separating IRWST and maintenance floor from elevation 103′ to 135′-3″	[*2′-6″ concrete-filled structural wall module with 0.5-in.-thick steel plate on inside and outside of wall*]*	0.36	[*0.5 -0.01 +0.1*]*

Notes:

1. The applicable column lines and elevation levels are identified and included in Figures 1.2-9, 3.7.2-12 (sheets 1 through 12), 3.7.2-19 (sheets 1 through 3) and on Table 1.2-1.

[2. *The concrete thickness includes the steel face plates. Thickness greater than 3′-0″ have a construction tolerance of +1″, -3/4″.Thickness less than or equal to 3′-0″ have a construction tolerance of +1/2″, -3/8″.*]*

3. These plate thicknesses represent the thickness required for operating and design basis loads except for designed openings or penetrations. These values apply for each face of the applicable wall unless specifically indicated on the table. For load combinations with thermal loads, the evaluation is performed as described in Subsection 3.8.3.5.3.4.

[4. *As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thicknesses.*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-4 (Sheet 1 of 3)
Design Summary of Southwest Wall of Refueling Cavity Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Mid-Height

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	1	-20	1	1	1	0	0	0	–
Hydro (F)	2	0	1	24	30	-2	0	0	–
Live (L)	0	-7	0	3	2	0	0	1	During refueling
Live (L_o)	0	-2	0	1	1	0	0	0	During operation
ADS	2	7	6	19	19	-4	0	1	–
E_s	13	49	56	41	40	15	1	8	During operation
Thermal (T_0)	-74	-34	-47	141	113	0	4	-1	During operation
LC (1)	8	-18	14	70	76	-8	0	3	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	5	-40	3	41	46	-2	0	2	[*1.4D+1.4F+1.7L_r*]*
LC (3)	8	-15	14	69	76	-8	0	3	[*1.4D+1.4F+1.7ADS*]*
LC (4)	18	35	65	87	90	17	1	10	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-12	-78	-60	-34	-28	-20	-1	-8	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-56	0	18	228	203	17	5	9	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-86	-112	-108	107	85	-19	3	-9	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	18	39	65	86	90	17	1	10	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101870
Plate thickness required for load combinations excluding thermal: 0.12 inches (Maximum)
[*Plate thickness provided:* *0.50 inches -0.01 +0.10*
As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*
Maximum principal stress for load combinations including thermal: 12.9 ksi
[*Yield stress at temperature:* *65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal: 19.6 ksi
Allowable stress intensity range for load combinations including thermal: 130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-4 (Sheet 2 of 3)
Design Summary of Southwest Wall of Refueling Cavity Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Base

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-1	-31	1	-1	-4	0	0	1	–
Hydro (F)	3	1	2	-2	-41	-1	-1	17	–
Live (L)	-1	-6	0	0	-3	0	0	1	During refueling
Live (L_o)	0	-2	0	0	-1	0	0	0	During operation
ADS	2	6	7	-2	-32	-1	-1	10	–
E_s	11	55	65	12	87	3	3	18	During operation
Thermal (T_0)	-189	-31	-76	165	86	-2	-5	0	During operation
LC (1)	7	-35	16	-8	118	-3	-3	41	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	2	-53	5	-5	-68	-2	-1	25	[*1.4D+1.4F+1.7L_r*]*
LC (3)	7	-32	16	-8	117	-3	-3	41	[*1.4D+1.4F+1.7ADS*]*
LC (4)	15	29	75	11	73	3	4	45	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-11	-93	-68	-17	165	-6	-5	-10	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-174	-2	-1	175	160	1	-1	44	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-200	124	144	147	78	-8	-10	-11	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	16	34	74	11	74	3	4	45	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101788
Plate thickness required for load combinations excluding thermal: 0.09 inches (Maximum)
[*Plate thickness provided:* *0.50 inches -0.01 +0.10*
As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*
Maximum principal stress for load combinations including thermal: 15.5 ksi
[*Yield stress at temperature:* *65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal: 24.8 ksi
Allowable stress intensity range for load combinations including thermal: 130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-4 (Sheet 3 of 3)
Design Summary of Southwest Wall of Refueling Cavity Design Loads, Load Combinations, and Comparison to Acceptance Criteria North End Bottom Corner

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-2	-29	-7	0	-3	0	0	0	–
Hydro (F)	3	1	4	-10	-16	4	1	3	–
Live (L)	-1	-4	0	0	-1	0	0	0	During refueling
Live (L_o)	0	-1	0	0	0	0	0	0	During operation
ADS	3	-2	7	-7	-18	2	1	2	–
E_s	23	43	64	14	68	6	6	7	During operation
Thermal (T_0)	-173	-88	-42	138	94	-15	-30	13	During operation
LC (1)	6	-44	8	-27	-57	8	4	7	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	0	-46	-4	-16	-28	5	3	4	[*1.4D+1.4F+1.7L_r*]*
LC (3)	6	-42	9	-27	-57	8	4	7	[*1.4D+1.4F+1.7ADS*]*
LC (4)	26	16	68	10	66	11	9	11	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-25	-74	-74	-32	-104	-4	-6	-5	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-147	-73	26	148	160	-4	-21	25	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-198	-162	-116	106	-11	-20	-36	8	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	26	19	69	11	67	11	9	11	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101794

Plate thickness required for load combinations excluding thermal: 0.09 inches (Maximum)

[*Plate thickness provided:* *0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal: 11.1 ksi

[*Yield stress at temperature:* *65.0 ksi (Minimum)*]*

Maximum stress intensity range for load combinations including thermal: 21.3 ksi

Allowable stress intensity range for load combinations including thermal: 130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-5 (Sheet 1 of 3)
Design Summary of South Wall of West Steam Generator Compartment Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Mid-Height

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-1	-23	0	1	1	0	0	0	–
Hydro (F_o)	-2	3	-5	20	22	0	0	-1	–
Live (L)	0	-7	-1	1	0	0	0	0	During refueling
Live (L_o)	0	-2	0	0	0	0	0	0	During operation
ADS	-1	9	-10	17	16	-1	0	1	–
E_s	7	65	58	30	26	5	1	4	During operation
Thermal (T_0)	-42	-53	19	69	59	2	-1	-1	During operation
LC (1)	-5	-17	-24	59	59	-2	1	0	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	-4	-39	-7	32	32	-1	1	-1	[*1.4D+1.4F+1.7L_r*]*
LC (3)	-5	-13	-24	59	59	-2	1	0	[*1.4D+1.4F+1.7ADS*]*
LC (4)	6	52	64	69	65	5	2	4	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-11	-96	-73	-26	-19	-6	-1	-6	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-36	-1	83	138	124	8	1	3	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-53	-149	-54	43	41	-4	-2	-7	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	6	56	64	68	64	5	2	4	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 104228

Plate thickness required for load combinations excluding thermal: 0.15 inches (Maximum)

[*Plate thickness provided:* *0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal: 16.5 ksi

[*Yield stress at temperature:* *36.0 ksi (Minimum)*]*

Maximum stress intensity range for load combinations including thermal: 16.5 ksi

Allowable stress intensity range for load combinations including thermal: 72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-5 (Sheet 2 of 3)
Design Summary of South Wall of West Steam Generator Compartment Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Base

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-3	-28	-1	0	2	0	0	0	–
Hydro (F)	3	4	-9	-4	-38	0	0	15	–
Live (L)	-1	-5	-1	0	-1	0	0	0	During refueling
Live (L_o)	0	-2	0	0	0	0	0	0	During operation
ADS	2	10	-10	-3	-30	0	0	9	–
E_s	14	74	52	9	62	1	1	14	During operation
Thermal (T_0)	-136	-13	43	79	65	5	4	-1	During operation
LC (1)	3	-20	-32	-10	-102	-1	-1	36	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	-2	-43	-16	-5	-52	-1	0	21	[*1.4D+1.4F+1.7L_r*]*
LC (3)	4	-17	-32	-10	-102	-1	-1	36	[*1.4D+1.4F+1.7ADS*]*
LC (4)	16	58	52	9	55	1	1	38	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-17	-110	-73	-16	-128	-2	-2	-8	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-120	45	95	88	120	6	5	37	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-152	-122	-30	63	-63	3	3	-9	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	16	62	52	9	55	1	1	38	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101943

Plate thickness required for load combinations excluding thermal: 0.14 inches (Maximum)

[*Plate thickness provided:* *0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal: 17.6 ksi

[*Yield stress at temperature:* *36.0 ksi (Minimum)*]*

Maximum stress intensity range for load combinations including thermal: 18.7 ksi

Allowable stress intensity range for load combinations including thermal: 72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-5 (Sheet 3 of 3)
Design Summary of South Wall of West Steam Generator Compartment Design Loads, Load Combinations, and Comparison to Acceptance Criteria West End Bottom Corner

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-6	-37	3	-1	6	1	-1	-3	–
Hydro (F)	5	14	-10	-6	-14	3	2	4	–
Live (L)	-2	-11	1	0	1	0	0	-1	During refueling
Live (L_o)	-1	-4	0	0	1	0	0	-1	During operation
ADS	9	36	-10	-3	-15	2	2	4	–
E_s	49	248	53	10	72	4	10	31	During operation
Thermal (T_0)	-125	-61	91	26	64	-4	-27	-23	During operation
LC (1)	14	21	-28	-15	-34	9	5	7	[$1.4D+1.4F+1.7L_o+1.7ADS$]*
LC (2)	-4	-52	-8	-10	-8	6	0	-1	[$1.4D+1.4F+1.7L_l$]*
LC (3)	15	27	-28	-15	-35	9	5	8	[$1.4D+1.4F+1.7ADS$]*
LC (4)	57	256	57	6	80	10	13	35	[$D+F+L_o+\|ADS\|+E_s$]*
LC (5)	-59	-311	-71	-20	-93	-2	-12	-36	[$D+F+L_o-\|ADS\|-E_s$]*
LC (6)	-67	195	147	32	144	6	-14	13	[$D+F+L_o+\|ADS\|+T_0+E_s$]*
LC (7)	-184	-372	20	7	-29	-6	-39	-58	[$D+F+L_o-\|ADS\|+T_0-E_s$]*
LC (8)	59	264	56	6	79	10	14	36	[$0.9D+1.0F+1.0\|ADS\|+1.0E_s$]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101933

Plate thickness required for load combinations excluding thermal: 0.43 inches (Maximum)

[*Plate thickness provided:* *0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal: 31.9 ksi

[*Yield stress at temperature:* *36.0 ksi (Minimum)*]*

Maximum stress intensity range for load combinations including thermal: 36.0 ksi

Allowable stress intensity range for load combinations including thermal: 72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-6 (Sheet 1 of 3)
Design Summary of North-East Wall of IRWST Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Mid-Height

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-1	-18	-1	0	1	-1	-2	-2	–
Hydro (F)	0	1	-1	16	17	3	2	0	–
Live (L)	0	-13	-1	5	5	-1	-2	-1	During refueling
Live (L_o)	0	-4	-1	4	5	-1	-2	-1	During operation
ADS	-2	3	0	16	14	5	2	1	–
E_s	16	42	47	35	39	12	11	16	During operation
Thermal (T_0)	-32	-13	48	49	51	-2	-5	-3	During operation
LC (1)	-5	-26	-3	58	58	8	0	-3	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	-2	-46	-4	30	33	0	-3	-5	[*1.4D+1.4F+1.7L_r*]*
LC (3)	-5	-19	-1	51	48	10	3	-1	[*1.4D+1.4F+1.7ADS*]*
LC (4)	16	24	45	72	76	17	11	14	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-19	-67	-49	-31	-31	-16	-15	-21	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-16	11	93	121	127	15	6	11	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-51	-80	-1	18	20	-18	-20	-24	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	16	29	46	68	71	18	13	16	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 140027

Plate thickness required for load combinations excluding thermal:	0.13 inches (Maximum)
[*Plate thickness provided:*	*0.50 inches -0.01 +0.10*]*
Maximum principal stress for load combinations including thermal:	18.1 ksi
[*Yield stress at temperature:*	*36.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal:	18.1 ksi
Allowable stress intensity range for load combinations including thermal:	72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-6 (Sheet 2 of 3)
Design Summary of North-East Wall of IRWST Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Bottom-Elevation 107'-2"

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	0	-19	1	0	0	0	0	0	–
Hydro (F)	1	2	-1	2	-8	2	0	11	–
Live (L)	-1	-7	0	0	-3	0	0	1	During refueling
Live (L_o)	-1	-1	0	0	-2	0	0	1	During operation
ADS	0	4	1	2	-7	3	0	8	–
E_s	23	54	47	13	48	8	6	17	During operation
Thermal (T_0)	-92	-58	71	64	63	-1	0	-1	During operation
LC (1)	0	-20	3	7	-27	8	-1	32	[$1.4D+1.4F+1.7L_o+1.7ADS$]*
LC (2)	1	-37	1	3	-16	3	-1	18	[$1.4D+1.4F+1.7L_r$]*
LC (3)	1	-18	3	6	-24	8	0	30	[$1.4D+1.4F+1.7ADS$]*
LC (4)	24	39	49	17	45	13	5	38	[$D+F+L_o+\|ADS\|+E_s$]*
LC (5)	-23	-77	-47	-12	-65	-8	-7	-13	[$D+F+L_o-\|ADS\|-E_s$]*
LC (6)	-68	-18	120	81	108	12	5	37	[$D+F+L_o+\|ADS\|+T_0+E_s$]*
LC (7)	-114	-134	24	52	-2	-9	-7	-14	[$D+F+L_o-\|ADS\|+T_0-E_s$]*
LC (8)	25	42	49	17	47	13	6	37	[$0.9D+1.0F+1.0\|ADS\|+1.0E_s$]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 140005

Plate thickness required for load combinations excluding thermal:	0.12 inches (Maximum)
[*Plate thickness provided:*	*0.50 inches -0.01 +0.10*]*
Maximum principal stress for load combinations including thermal:	17.0 ksi
[*Yield stress at temperature:*	*36.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal:	21.9 ksi
Allowable stress intensity range for load combinations including thermal:	72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.3-6 (Sheet 3 of 3)
Design Summary of North-East Wall of IRWST Design Loads, Load Combinations, and Comparison to Acceptance Criteria
North End Bottom Corner-Elevation 107'-2"

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-1	-19	1	0	-4	0	0	0	–
Hydro (F)	2	12	6	5	10	12	-7	-11	–
Live (L)	0	-10	1	-1	-4	1	1	1	During refueling
Live (L_o)	0	-3	0	0	-2	1	0	0	During operation
ADS	0	19	7	7	16	11	-5	-11	–
E_s	8	136	38	26	98	17	14	31	During operation
Thermal (T_0)	38	28	60	-39	24	-11	17	34	During operation
LC (1)	2	17	21	16	31	35	-17	-33	[$1.4D+1.4F+1.7L_o+1.7ADS$]*
LC (2)	2	-28	10	4	2	17	-8	-13	[$1.4D+1.4F+1.7L_r$]*
LC (3)	2	21	21	17	35	34	-18	-33	[$1.4D+1.4F+1.7ADS$]*
LC (4)	10	144	52	37	118	40	12	32	[$D+F+L_o+\|ADS\|+E_s$]*
LC (5)	-7	-165	-38	-29	-110	-16	-25	-52	[$D+F+L_o-\|ADS\|-E_s$]*
LC (6)	48	172	111	-2	142	29	29	66	[$D+F+L_o+\|ADS\|+T_0+E_s$]*
LC (7)	31	-137	22	-68	-86	-27	-8	-19	[$D+F+L_o-\|ADS\|+T_0-E_s$]*
LC (8)	10	149	51	37	120	39	12	32	[$0.9D+1.0F+1.0\|ADS\|+1.0E_s$]*

Notes:
x-direction is horizontal; y-direction is vertical.

Element number 140001

Plate thickness required for load combinations excluding thermal:	0.36 inches (Maximum)
[*Plate thickness provided:*	*0.50 inches -0.01 +0.10*]*
Maximum principal stress for load combinations including thermal:	29.5 ksi
[*Yield stress at temperature:*	*36.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal:	30.3 ksi
Allowable stress intensity range for load combinations including thermal:	72.0 ksi (Minimum)

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Table 3.8.3-7
Design Summary of Steel Wall of IRWST

Mechanical Loads Only AISC Interaction Ratio				
Section Location and Element Number	**T Section**	**L Section**		**Load Combination**
TYPICAL COLUMN AT MIDDLE OF WALL				
Top	0.15	0.15		*[D+F+L_o -|ADS|- E_s (LC#5)]**
Mid-height	0.30	0.12		*[D+F+L_o -|ADS|- E_s (LC#5)]**
	0.17	0.26		*[D+F+L_o +|ADS|+E_s (LC#4)]**
Bottom	0.22	0.01		*[D+F+L_o -|ADS|- E_s (LC#5)]**
ENVELOPE OF ALL LOCATIONS AND LOAD COMBINATIONS				
	0.47	0.64		*[D+F+ADS (LC#3)]**
Mechanical Plus Thermal Loads Ratio of Stress to AISC or ASME (2 * Sy = 105.6 ksi at T_0 = 260 F)[2]				
Section Location and Element Number	**Flange of T Section**	**Flange of L Section**	**Plate**	**Load Combination**
TYPICAL COLUMN AT MIDDLE OF WALL				
Top	0.17 AISC	0.16 AISC	0.28 AISC	*[D+F+L_o -|ADS|+T_0-E_s (LC#7)]**
	0.10 AISC	0.34 AISC	0.17 AISC	*[D+F+L_o +|ADS|+T_0+E_s (LC#6)]**
Mid-height	0.52 AISC	0.24 AISC	0.34 AISC	*[D+F+L_o -|ADS|+T_0-E_s (LC#7)]**
Bottom	0.27 ASME	0.32 AISC	0.44 AISC	*[D+F+L_o +|ADS|+T_0+E_s (LC#6)]**
	0.15 AISC	0.52 AISC	0.64 AISC	*[D+F+L_o -|ADS|+T_0-E_s (LC#7)]**
ENVELOPE OF ALL LOCATIONS AND LOAD COMBINATIONS				
–	0.67 ASME	0.50 ASME	0.84[2] ASME	*[D+F+L_o +|ADS|+T_0+E_s (LC#6)]**

Notes:

1. Results of the evaluation of mechanical and thermal loads are shown against the AISC allowables when the stresses are less than yield. Portions of the steel wall at the end of the wall exceed yield due to the restraint provided by the adjacent concrete. These areas are evaluated against the ASME allowables as described in Subsection 3.8.3.5.3.4.
2. T_0 = 260 F ADS2 event per Subsection 3.8.3.3.1.

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Table 3.8.4-1
[*LOAD COMBINATIONS AND LOAD FACTORS FOR SEISMIC CATEGORY I STEEL STRUCTURES*]*

Combination No.		*Load Combination and Factors*								
		1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*
Load Description										
Dead	D	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*
Liquid	F	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*
Live	L	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*
Earth pressure	H	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*
Normal reaction	R_o	*1.0*	*1.0*	*1.0*	*1.0*				*1.0*	*1.0*
Normal thermal	T_o			*1.0*	*1.0*				*1.0*	*1.0*
Wind	W		*1.0*							*1.0*
Safe shutdown earthquake	E_S			*1.0*				*1.0*		
Tornado	W_t				*1.0*					
Accident pressure	P_a					*1.0*	*1.0*	*1.0*		
Accident thermal	T_a					*1.0*	*1.0*	*1.0*		
Accident thermal reactions	R_a					*1.0*	*1.0*	*1.0*		
Accident pipe reactions	Y_r						*1.0*	*1.0*		
Jet impingement	Y_j						*1.0*	*1.0*		
Pipe impact	Y_m						*1.0*	*1.0*		
Stress Limit Coefficient [1],[3] *(except for compression)*		*1.0*	*1.0*	*1.6*	*1.6*	*1.6*	*1.6*	*1.7*	*1.5*	*1.5*
(for compression)		*1.0*	*1.0*	*1.4*	*1.4*	*1.4*	*1.4*	*1.6*	*1.3*	*1.3*

Notes:

1. *Allowable stress limits coefficients are applied to the basic stress allowables of AISI or AISC. The coefficients for ANSI/AISC N690 are supplemented by the requirements identified in subsection 3.8.4.5.*
2. *Where any load reduces the effects of other loads, the coefficient for that load is taken as zero unless it can be demonstrated that the load is always present or occurs simultaneously with the other loads.*
3. *In no instance does the allowable stress exceed $0.7F_u$ in axial tension nor $0.7F_u$ times the ratio of the plastic to elastic section modulus for tension plus bending.*
4. *Loads due to maximum precipitation are evaluated using load combination 4 with the maximum precipitation in place of the tornado load.*

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Table 3.8.4-2
[*LOAD COMBINATIONS AND LOAD FACTORS FOR SEISMIC CATEGORY I CONCRETE STRUCTURES*]*

Load Combination and Factors												
Combination No.		***1***	***2***	***3***	***4***	***5***	***6***	***7***	***8***	***9***	***10***	***11***
Dead	*D*	*1.4*	*1.4*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.05*	*1.05*	*1.4*	*1.0*
Liquid	*F*	*1.4*	*1.4*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.05*	*1.05*	*1.4*	*1.0*
Live	*L*	*1.7*	*1.7*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.3*	*1.3*	*1.7*	*1.0*
Earth	*H*	*1.7*	*1.7*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.3*	*1.3*	*1.7*	*1.0*
Design Pressure	P_d										*1.5*	*1.0*
Normal reaction	R_o	*1.7*	*1.7*	*1.0*	*1.0*				*1.3*	*1.3*	*1.7*	*1.0*
Normal thermal	T_o			*1.0*	*1.0*				*1.2*	*1.2*		
Wind	*W*		*1.7*							*1.3*		
Safe shutdown earthquake	E_s			*1.0*				*1.0*				*1.0*
Tornado	W_t				*1.0*							
Accident pressure	P_a					*1.4*	*1.25*	*1.0*				
Accident thermal	T_a					*1.0*	*1.0*	*1.0*				
Accident thermal reactions	R_a					*1.0*	*1.0*	*1.0*				
Accident pipe reactions	Y_r						*1.0*	*1.0*				
Jet impingement	Y_j						*1.0*	*1.0*				
Pipe impact	Y_m						*1.0*	*1.0*				

Notes:

1. *Design for mechanical loads is in accordance with ACI-349 Strength Design Method for all load combinations. Design for combinations including thermal loads is described in subsection 3.8.3.5.3.4. Shield building design for combinations including thermal loads is described in subsection 3.8.4.5.5.4.*
2. *Where any load reduces the effects of other loads, the corresponding coefficient for that load is taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise the coefficient for the load is taken as zero.*
3. *Loads due to maximum precipitation are evaluated using load combination 4 with the maximum precipitation in place of the tornado load.*
4. *Load combinations 10 and 11 are applicable to nuclear island basemat design only. P_d, the containment design pressure, is 59 psig.*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3.8.4-3
Acceptance Tests for Concrete Aggregates

Method of Test	Designation
Organic impurities in sand	ASTM C 40
Effect of organic impurities on strength of mortar	ASTM C 87
Soundness of aggregates	ASTM C 88
Material finer than No. 200 sieve	ASTM C 117
Specific gravity and absorption - coarse aggregates	ASTM C 127
Specific gravity and absorption - fine aggregates	ASTM C 128
Los Angeles abrasion of small-size coarse aggregates	ASTM C 131
Sieve analysis	ASTM C 136
Friable particles	ASTM C 142
Potential reactivity of aggregates (chemical)	ASTM C 289
Petrographic examination of aggregates	ASTM C 295
Resistance to degradation of large-size coarse aggregates by abrasion and impact in the Los Angeles machine	ASTM C 535
Potential alkali reactivity of carbonate rocks for concrete aggregates	ASTM C 586
Resistance of concrete to rapid freezing and thawing	ASTM C 666
Flat and elongated particles	ASTM D4791

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Table 3.8.4-4
Criteria for Water Used in Production of Concrete

Requirements and Test Method	Criteria
Compressive strength ASTM C 109	Reduction in strength not in excess of 10 percent
Soundness ASTM C 151	Increase in length limited to 0.10 percent
Time of setting ASTM C 191	± 10 min for initial set, ± 1 hour for final set

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Table 3.8.4-5
Not Used

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Table 3.8.4-6 (Sheet 1 of 2)
Materials Used in Structural and Miscellaneous Steel

Standard	Construction Material
ASTM A1	Carbon steel rails
ASTM A36/A36M	Rolled shapes, plates, and bars
ASTM A53	Welded and Seamless Steel Pipe, Grade B
ASTM A106	Seamless Carbon Steel Pipe For High Temperature Service
ASTM A108	Weld studs
ASTM A123	Zinc coatings (hot galvanized)
ASTM A167	Stainless and Heat-Resisting Chromium Nickel Steel Plate, Sheet and Strip
ASTM A193	Alloy Steel and Stainless Steel Bolting Materials for High-Temperature Service
ASTM A194	Carbon and Alloy Steel Nuts for Bolts for High-Pressure or High-Temperature Service, or Both
ASTM A240	Duplex 2101 stainless steel (designation S32101)
ASTM A242	High-strength low alloy structural steel
ASTM A276	Stainless and Heat-Resisting Steel Bars and Shapes
ASTM A307	Low carbon steel bolts
ASTM A312	Seamless and Welded Austenitic Stainless Steel Pipe
ASTM A325	High strength bolts
ASTM A354	Quenched and tempered alloy steel bolts (Grade BC)
ASTM A441	High-strength low alloy structural manganese vanadium steel
ASTM A490	Structural Bolts, Alloy Steel, Heat Treated, 150 ksi Minimum Tensile Strength
ASTM A496	High strength deformed wire
ASTM A500	Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes
ASTM A501	Hot-Formed Welded and Seamless Carbon Steel Structural Tubing
ASTM A505	Standard Specification for Steel, Sheet and Strip, Alloy, Hot-Rolled and Cold-Rolled
ASTM A514	High-Yield Strength Quenched and Tempered Alloy Steel Plate, Suitable for Welding
ASTM A517	Standard Specification for Pressure Vessel Plates, Alloy Steel, High-Strength, Quenched and Tempered
ASTM A540	Alloy Steel Bolting for Special Applications

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Table 3.8.4-6 (Sheet 2 of 2)
Materials Used in Structural and Miscellaneous Steel

Standard	Construction Material
ASTM A564	Hot-Rolled and Cold-Finished, Age-Hardening Stainless and Heat-Resisting Steel Bars and Shapes
ASTM A570	Hot-Rolled Carbon Steel Sheets and Strip, Structural Quality, Grades C, D, and E
ASTM A572	High-strength low alloy structural steel
ASTM A588	High-strength low alloy structural steel
ASTM A607	Steel Sheet and Strip, Hot-Rolled and Cold-Rolled, High-Strength, Low Alloy, Columbium, and/or Vanadium
ASTM A615	Deformed and Plain Billet Steel Bars for Concrete Reinforcement
ASTM A618	Hot-Formed Welded and Seamless High-Strength Low Alloy Structural Tubing
ASTM A706	Low Alloy Steel Deformed Bars for Concrete Reinforcement
ASTM A970	Headed Steel Bars for Concrete Reinforcement
ASTM A992	Rolled shapes, plates, and bars
ASTM-F1554	Steel anchor bolts, 36, 55, and 105-ksi yield strength

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Table 3.8.5-1
Minimum Required Factor of Safety
for Overturning and Sliding of Structures

Load Combination	Overturning	Sliding	Flotation
D + H + B + W	1.5	1.5	-
D + H + B + E_s	1.1	1.1	-
D + H + B + W_t	1.1	1.1	-
D + F	-	-	1.1
D + B	-	-	1.5

where:

D	=	dead load excluding the fluid loads
H	=	lateral earth pressure
W	=	wind load
E_s	=	safe shutdown earthquake load
W_t	=	tornado load
F	=	buoyant force due to the design basis flood
B	=	buoyant force on submerged structure due to high ground water table

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Table 3.8.5-2
Factors of Safety for Flotation, Overturning and Sliding of Nuclear Island Structures

Environmental Effect	Factor of Safety[1]
Flotation	
High Ground Water Table	3.7
Design Basis Flood	3.5
Sliding	
Design Wind, North-South	14.0
Design Wind, East-West	10.1
Design Basis Tornado, North-South	7.7
Design Basis Tornado, East-West	5.9
Safe Shutdown Earthquake, North-South	1.1[2]
Safe Shutdown Earthquake, East-West	1.1[2]
Overturning	
Design Wind, North-South	51.5
Design Wind, East-West	27.9
Design Basis Tornado, North-South	17.7
Design Basis Tornado, East-West	9.6
Safe Shutdown Earthquake, North-South	1.77
Safe Shutdown Earthquake, East-West	1.17

Notes:
1. Factor of safety is calculated for the envelope of the soil and rock sites described in Subsection 3.7.1.4.
2. From non-linear sliding analysis using friction elements, the horizontal movement is negligible (0.12 inch without buoyant force consideration, and 0.19 inch with buoyant force considered).

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Table 3.8.5-3
Definition of Critical Locations, Thicknesses and Reinforcement for Nuclear Island Basemat(1) (in^2/ft)

Basemat Segment (see detail in subsection 3.8.5.4.4)	Location	Required(3)			[*Provided (Minimum)*(4)]*		
		North-South	East-West	Shear	*North-South*	*East-West*	*Shear*
Auxiliary Building Basemat Elevation 66′ 6″							
Column line K to L and from Shield Building to Col. Line 11(2)(6)							
	Top Face	Note 5	Note 5		[2.25	*2.25*	
	Bottom Face	Note 5	1.85		*2.25*	*2.25*	
				0.24			*0.25*]*
Column line 1 to 2 and from Column Line K-2 to N wall(2)							
General area	Top Face	Note 5	Note 5		[*2.25*	*2.25*	
Central zone	Top Face	3.11	Note 5		*3.25*	*2.25*	
	Bottom Face	Note 5	1.85		*2.25*	*2.25*	
				0.47			*0.50*]*

Notes:

1. The applicable column lines and elevation levels are identified and included in Figures 1.2-9 and 3.7.2-12 (sheets 1 through 12).
[2. *The thickness of these sections is 6′ 0″ with a construction tolerance of +4 inches, -3/4 inch.*]*
3. These concrete reinforcement values represent the minimum reinforcement required for structural requirements except for designed openings, penetrations, sumps or elevator pits.
[4. *These concrete reinforcement values represent the provided reinforcement for structural requirements except for designed openings, penetrations, sumps or elevator pits.*]*
5. Reinforcement demand is low. Basemat reinforcement is fairly uniform across all of the basemat and the governing location is in other segments (see Figure 3.8.5-3, sheets 3 to 6).
6. The 5'-0" portion of the Basemat between column line L to 5' east of column line and shield building to column line 9.2, as shown in Figure 3.8.5-3, Sheet 6, are not included in this area.

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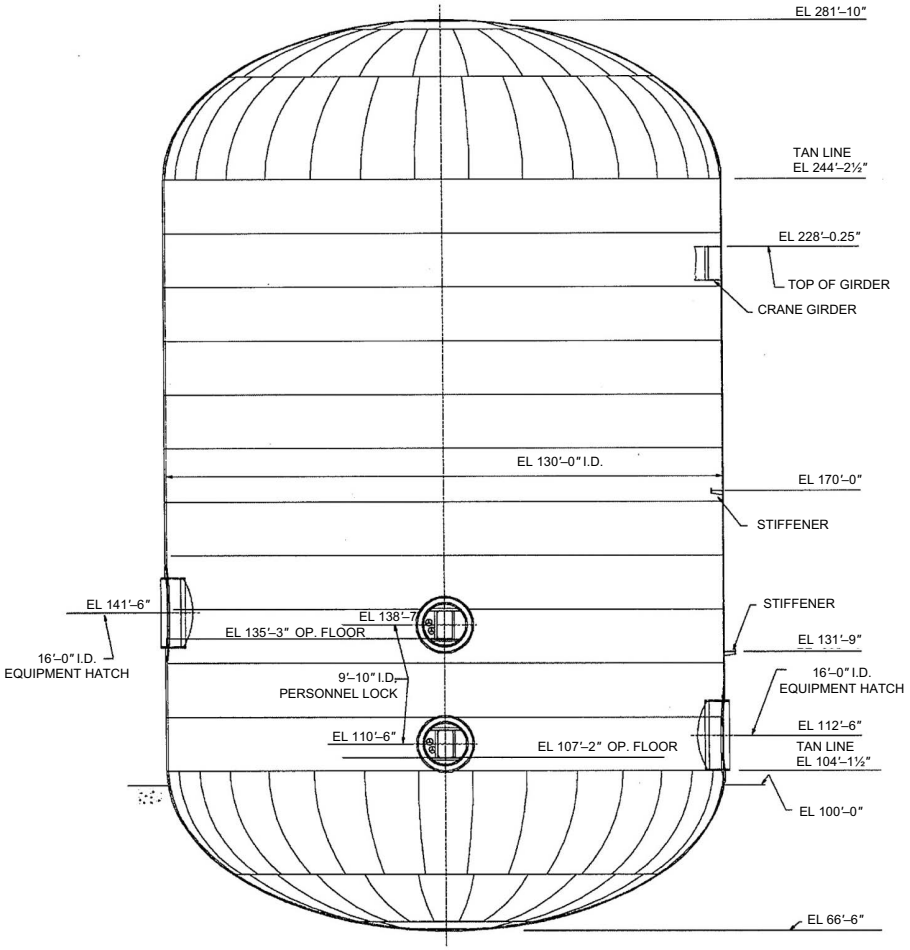

Equipment hatches rotated out of position to show configuration: see Sheet 2 for azimuth orientation.

Note: Course layout, plate/insert plate geometry, and weld seams are shown for illustrative purposes only.

Figure 3.8.2-1 (Sheet 1 of 3)
Containment Vessel General Outline

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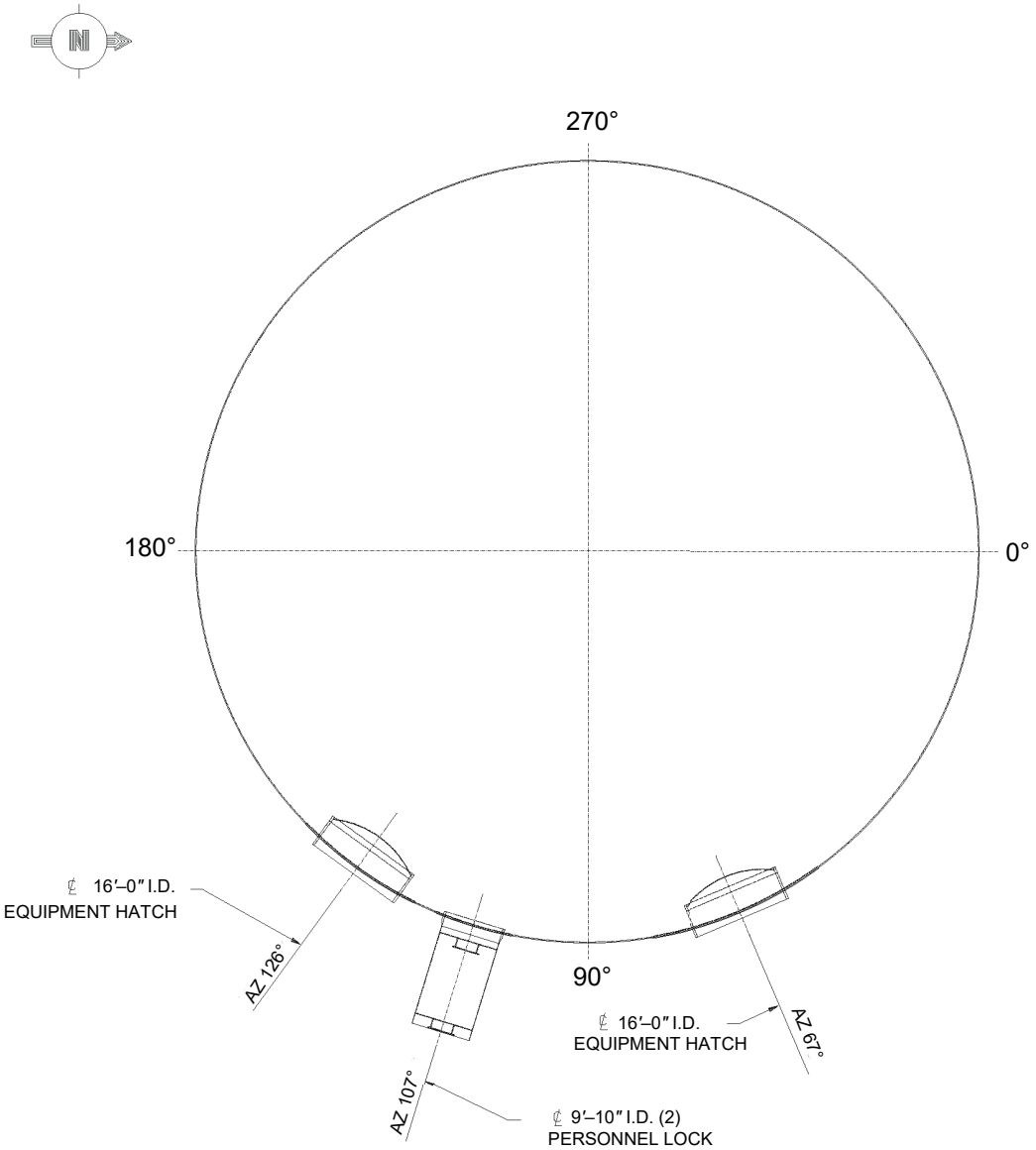

ORIENTATION OF MAJOR PENETRATIONS

Equipment hatch at azimuth of 126° is the lower hatch.
Equipment hatch at azimuth of 67° is the upper hatch.

Figure 3.8.2-1 (Sheet 2 of 3)
Containment Vessel General Outline

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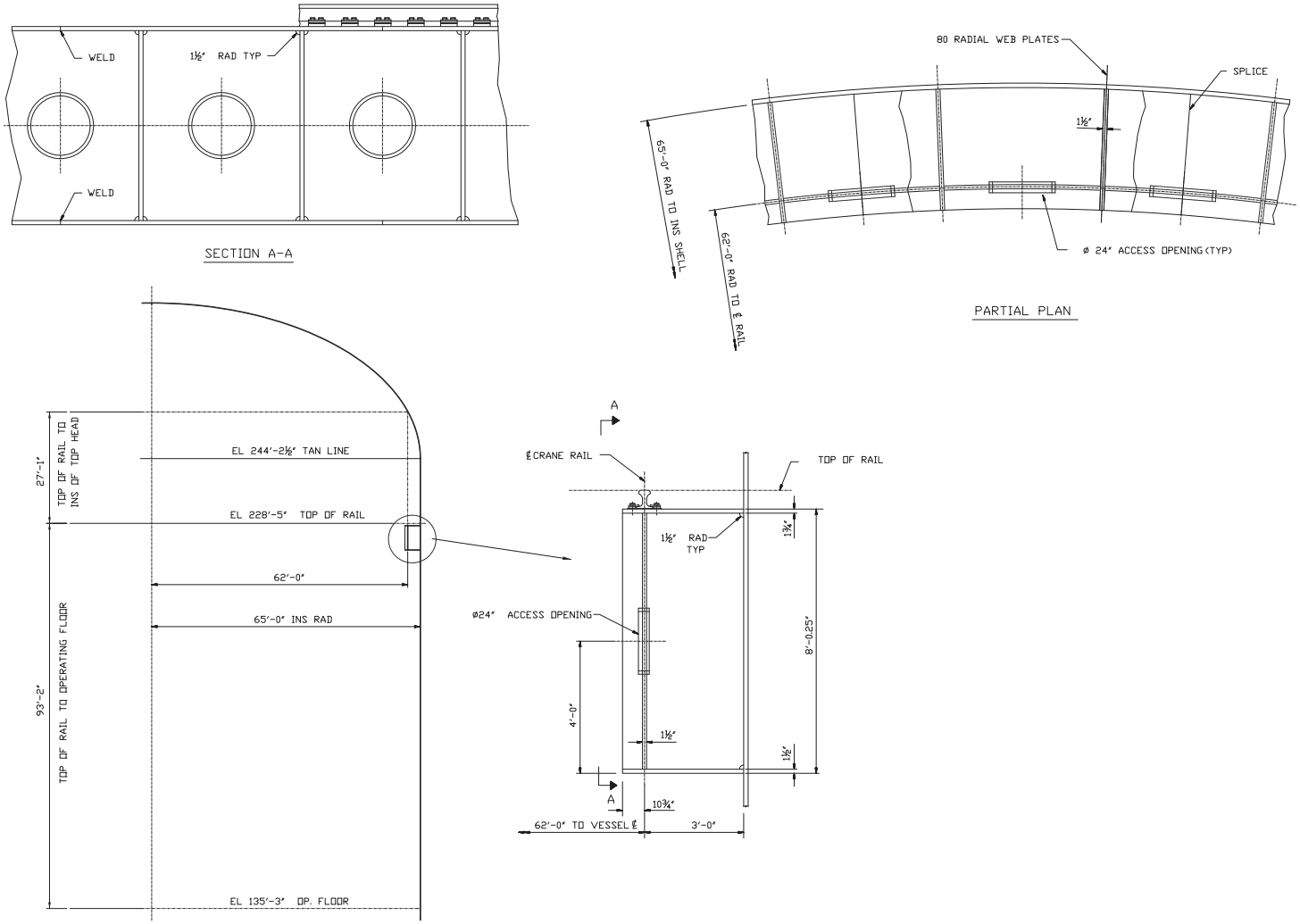

Figure 3.8.2-1 (Sheet 3 of 3)
Containment Vessel General Outline

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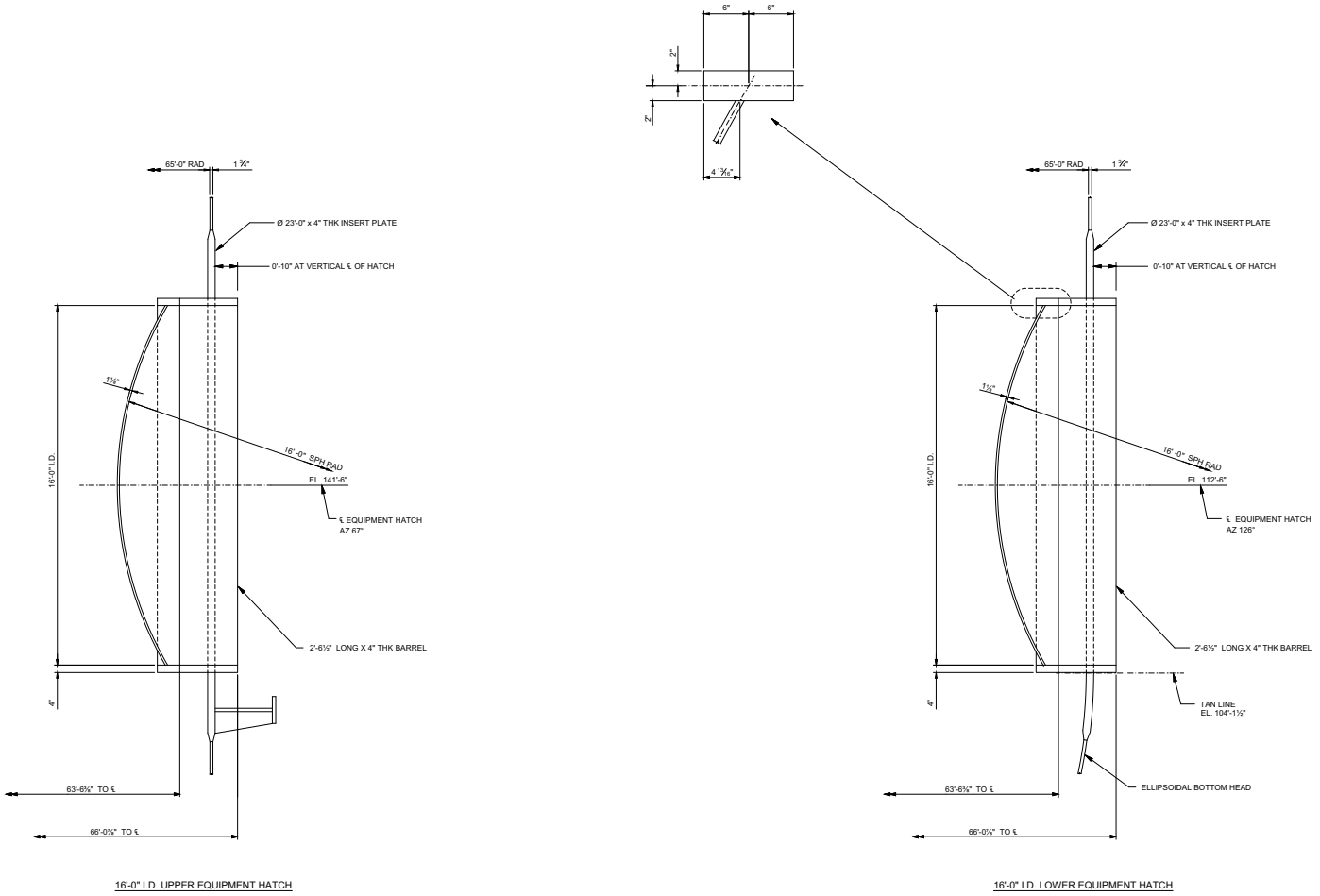

MATERIAL: SA738 GRADE B

See subsection 3.8.2.1.3 for information that is designated as Tier 2 *.

Figure 3.8.2-2
Equipment Hatches

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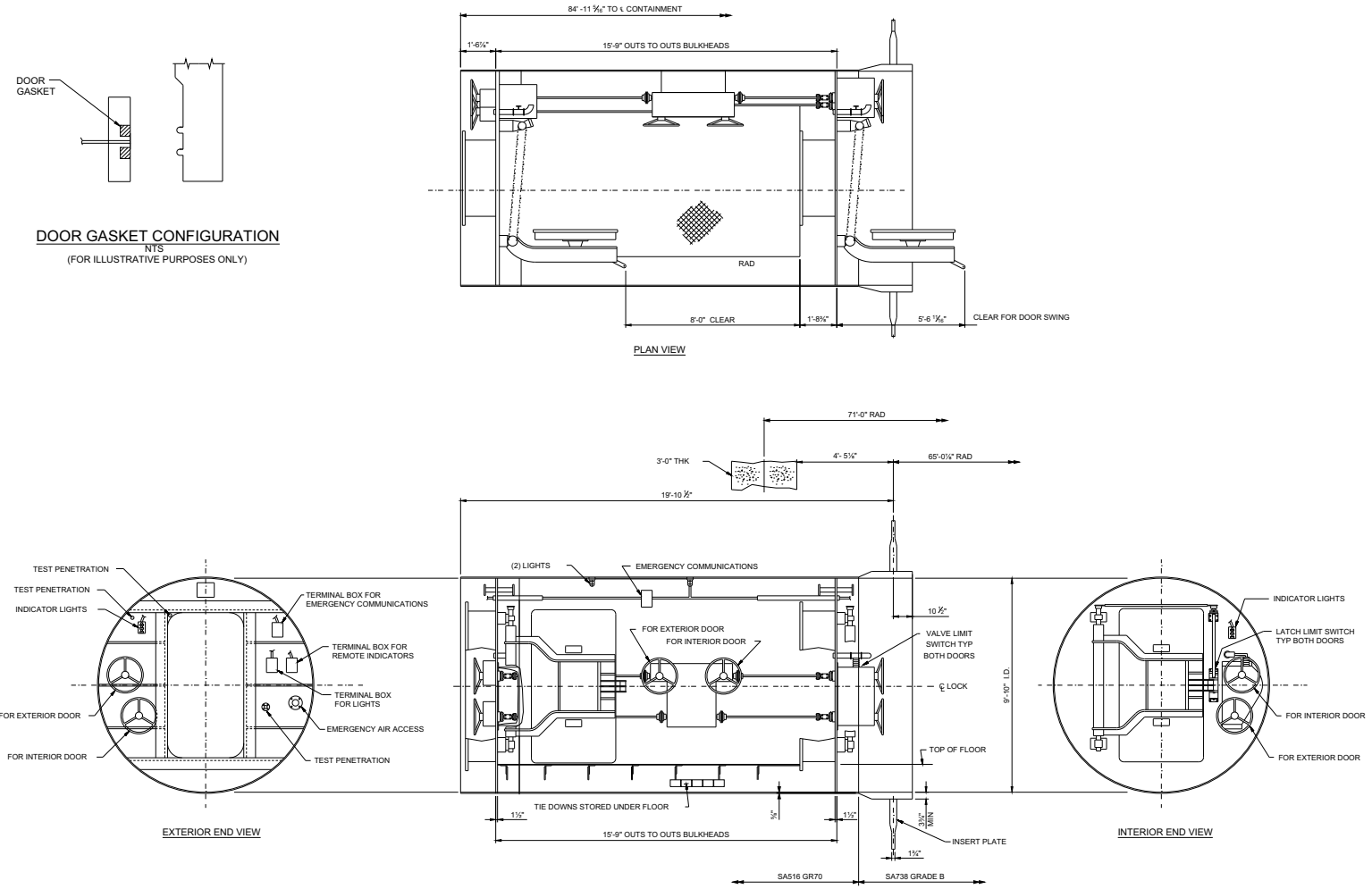

Figure 3.8.2-3
Personnel Airlock

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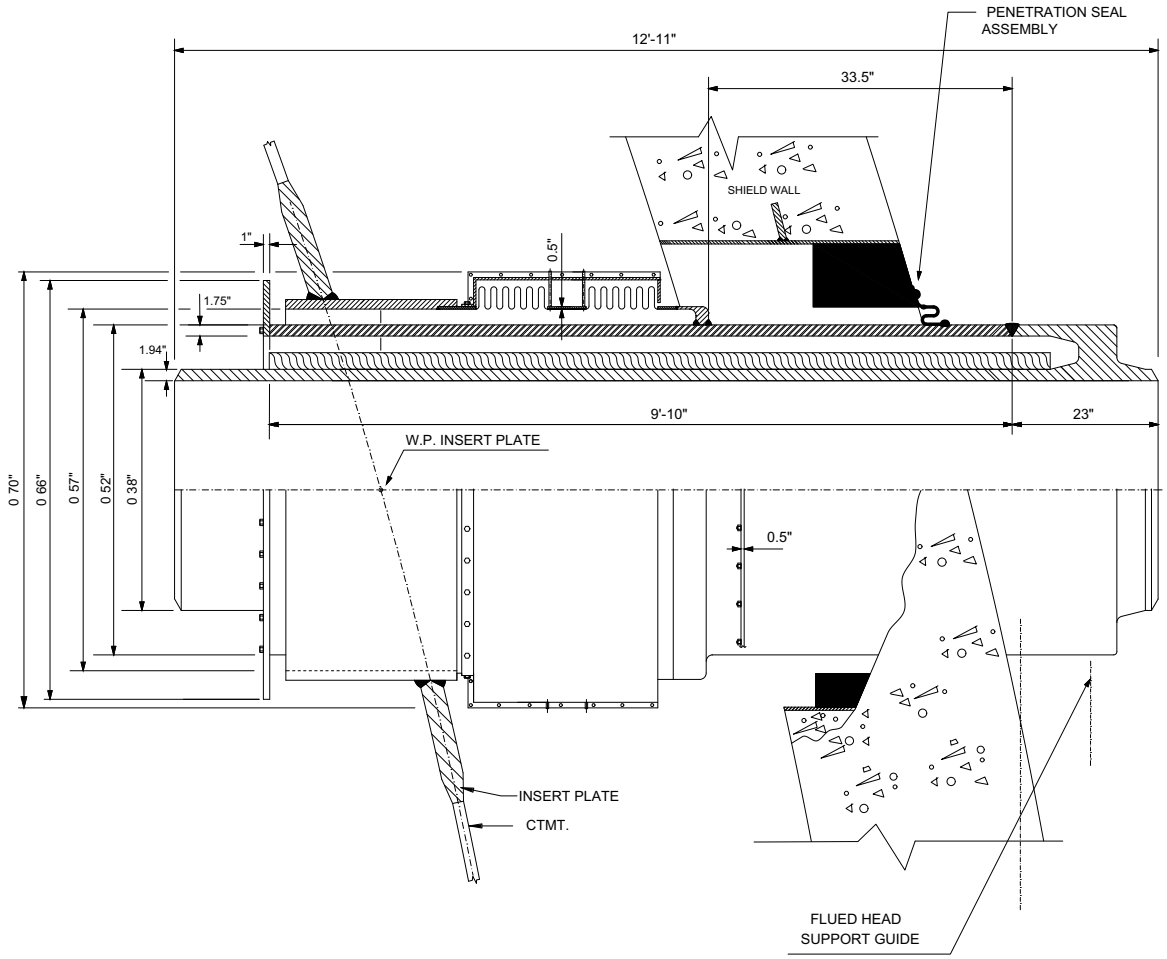

Figure 3.8.2-4 (Sheet 1 of 8)
Containment Penetrations Main Steam

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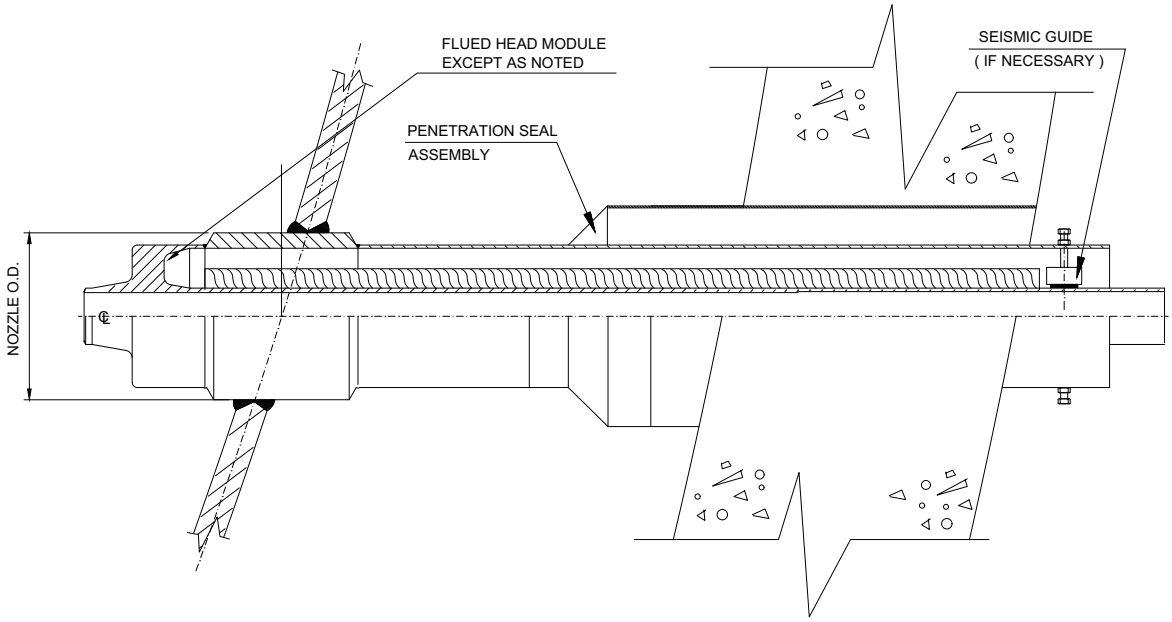

Figure 3.8.2-4 (Sheet 2 of 8)
Containment Penetrations Startup Feedwater

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(P20) RNS-PY-C02 PENETRATION

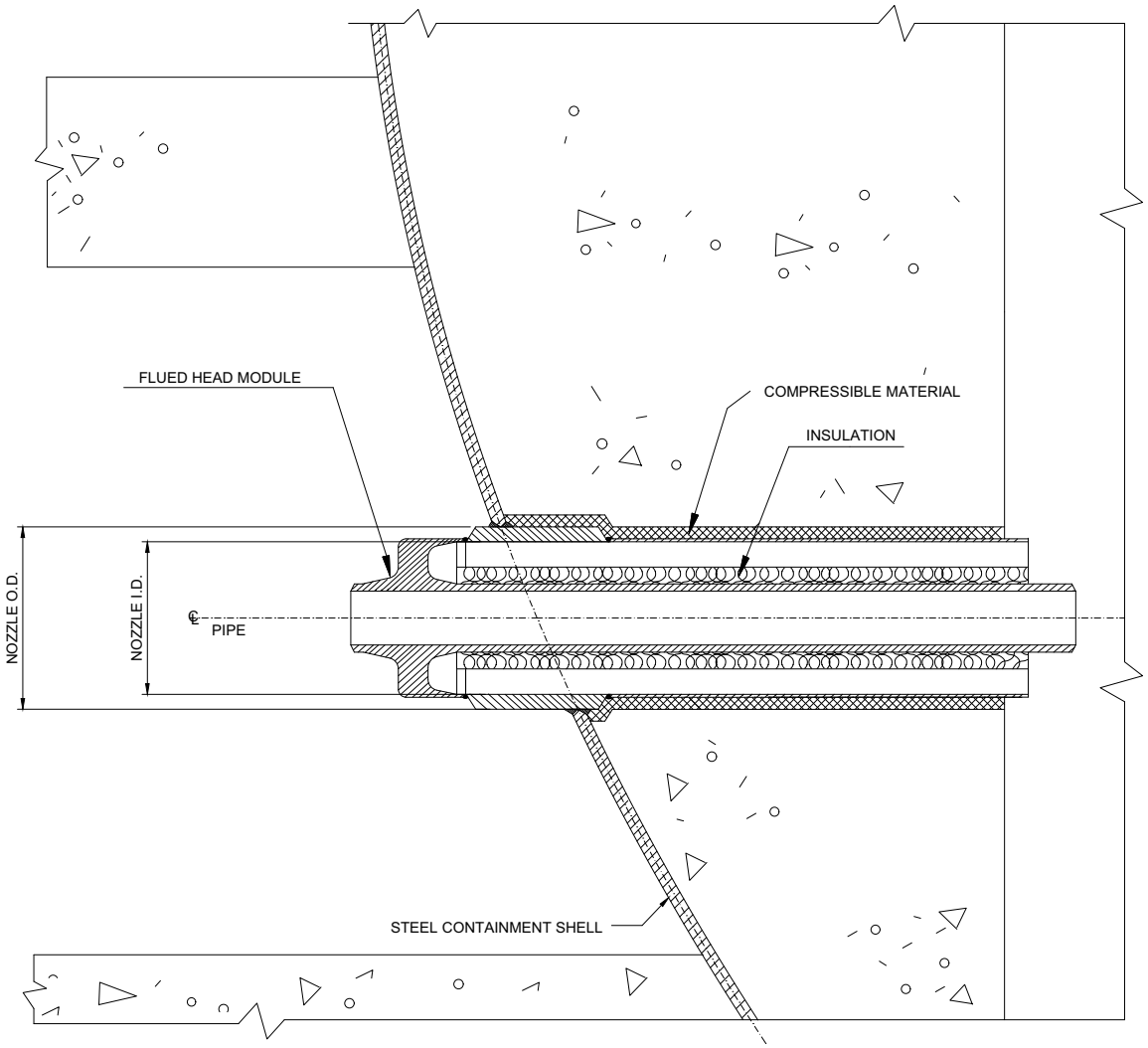

Figure 3.8.2-4 (Sheet 3 of 8)
Containment Penetrations Normal RHR Piping

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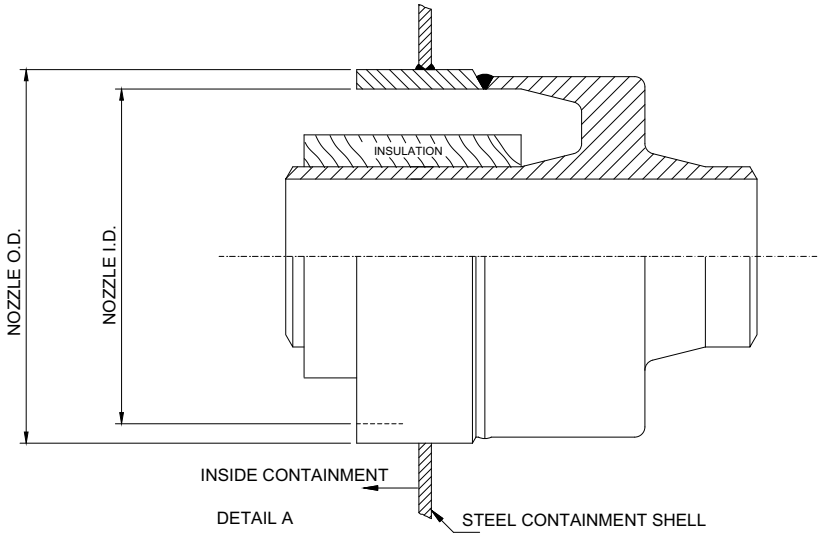

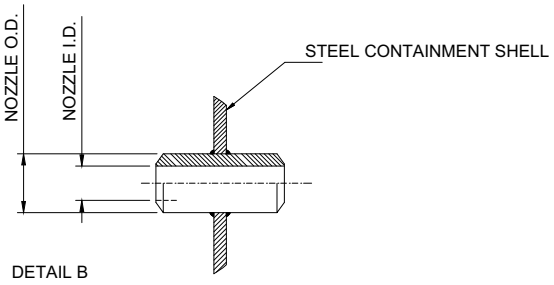

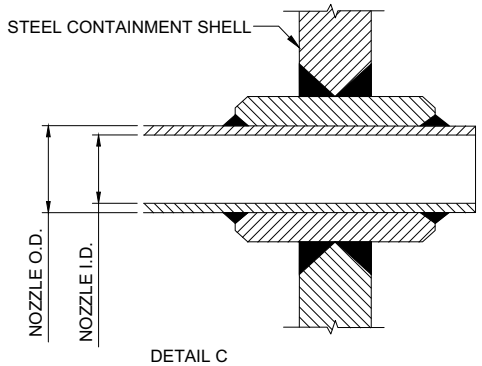

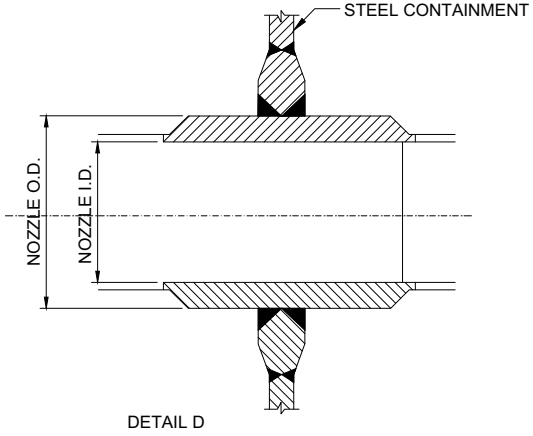

Figure 3.8.2-4 (Sheet 4 of 8)
Containment Penetrations

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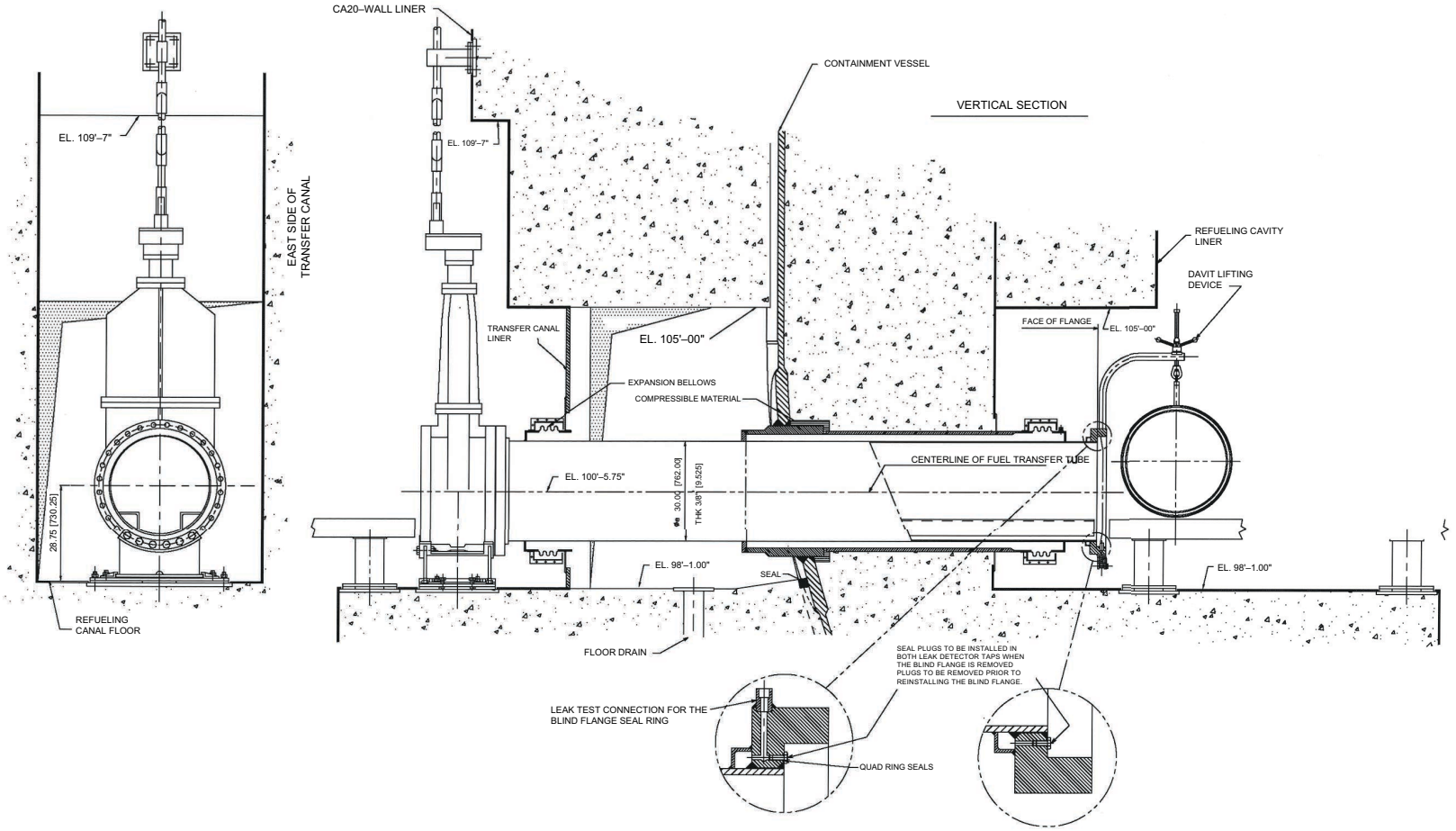

Figure 3.8.2-4 (Sheet 5 of 8)
Containment Penetrations
Fuel Transfer Penetration

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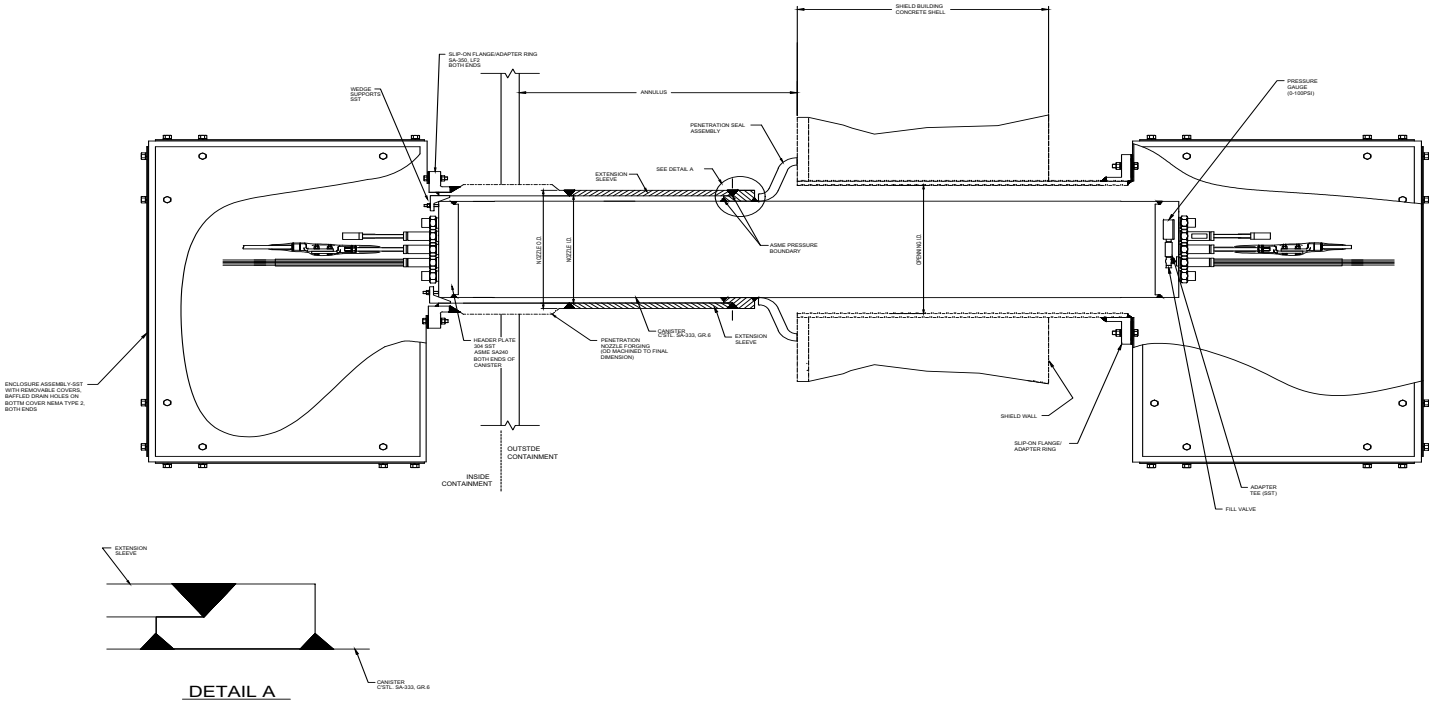

Figure 3.8.2-4 (Sheet 6 of 8)
Containment Penetrations
Typical Electrical Penetration

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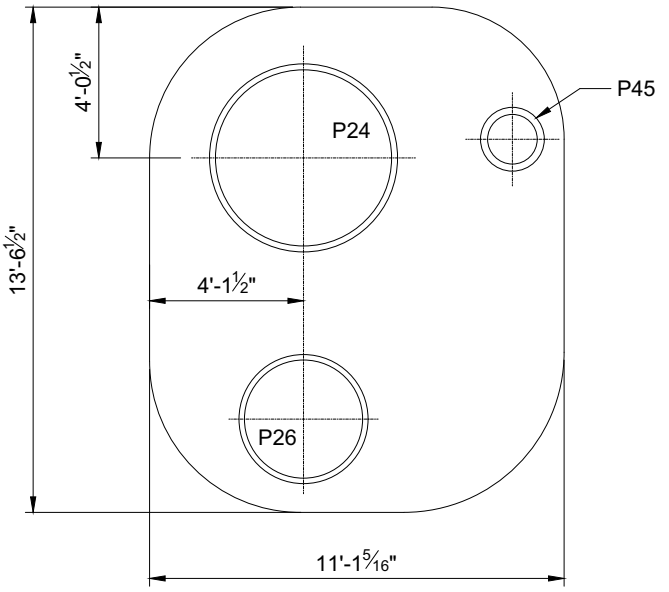

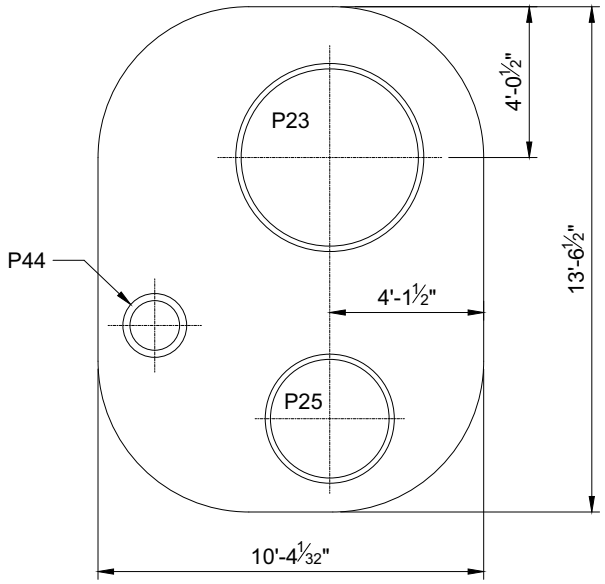

Figure 3.8.2-4 (Sheet 7 of 8)
Containment Penetrations
Steam Line and Feedwater Line Insert Plates

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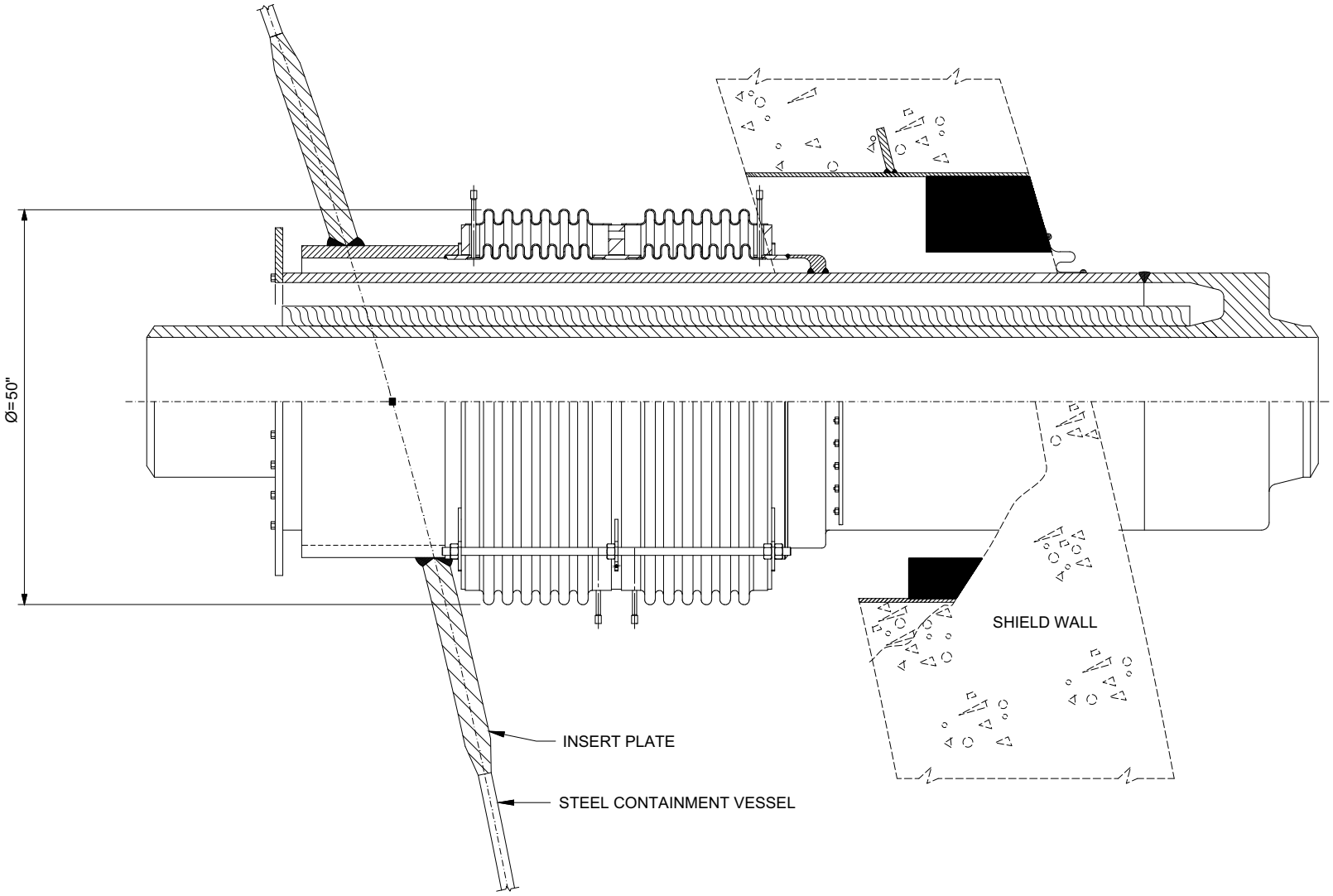

Figure 3.8.2-4 (Sheet 8 of 8)
Containment Penetrations Main Feedwater

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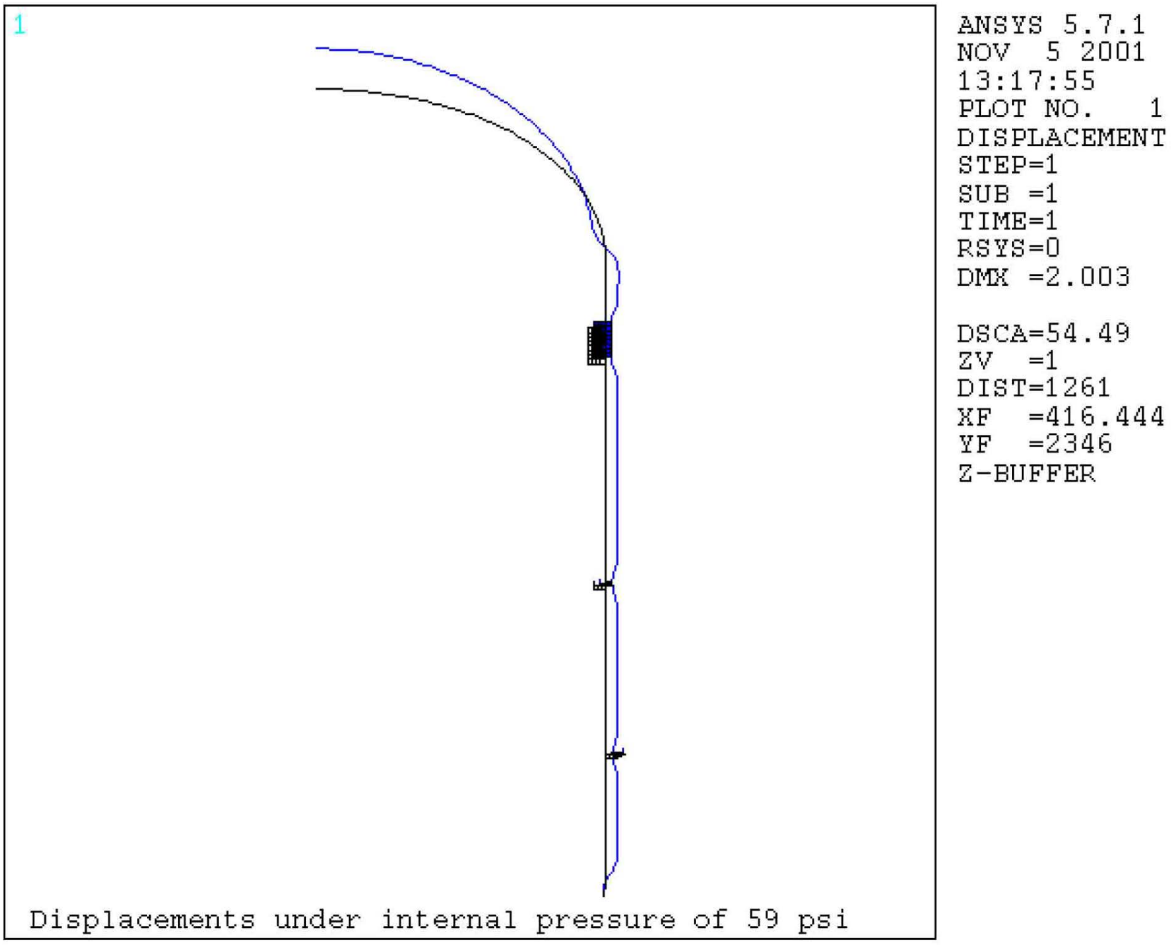

Figure 3.8.2-5 (Sheet 1 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Displaced Shape Plot

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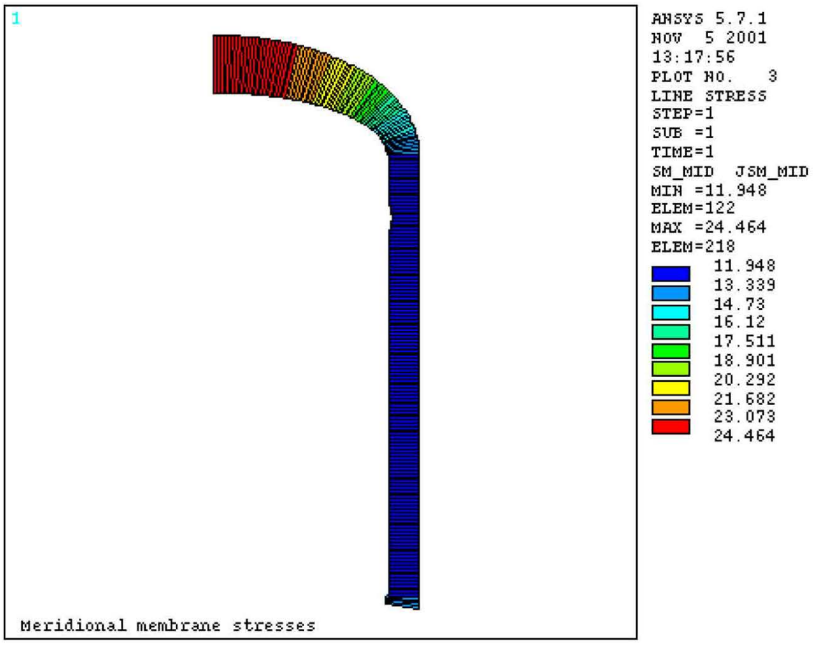
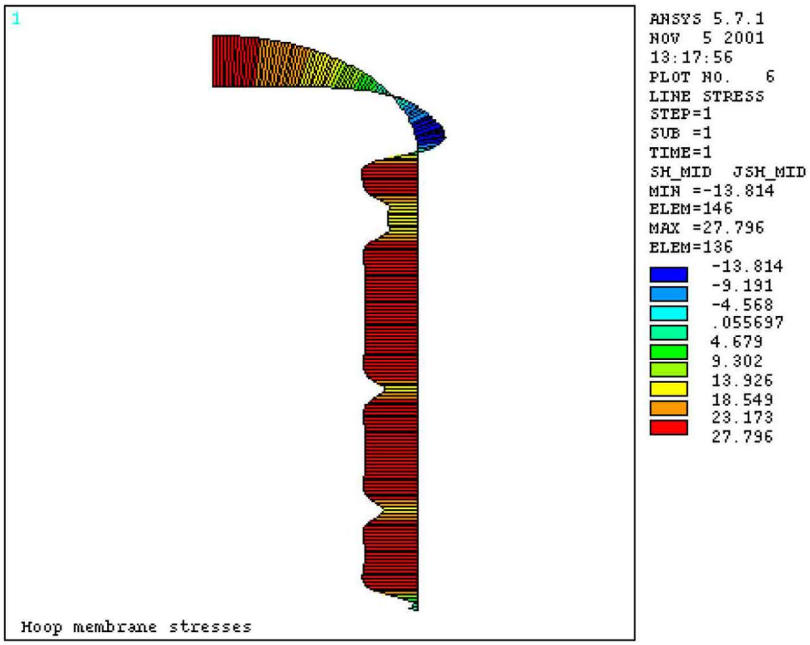


Meridional Membrane Stress (ksi)



Circumferential Membrane Stress (ksi)

Figure 3.8.2-5 (Sheet 2 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Membrane Stresses (ksi)

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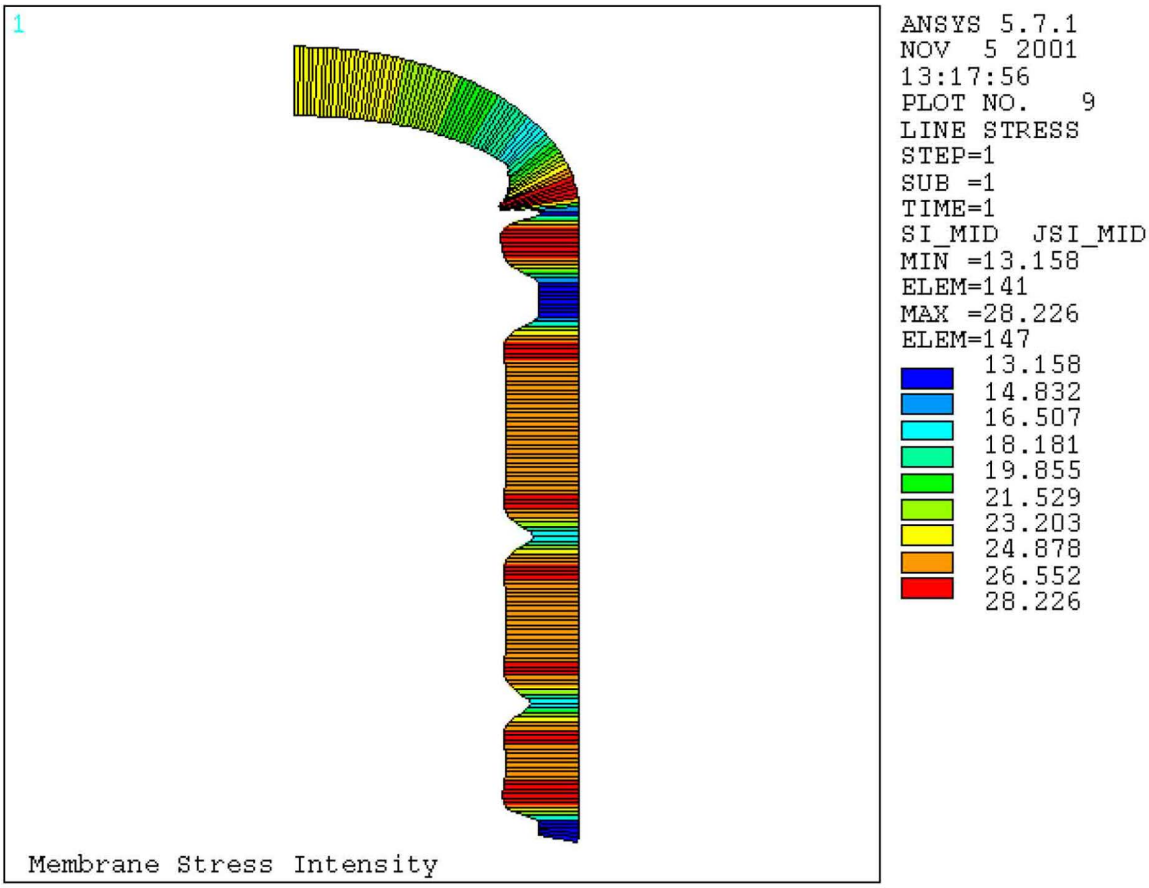

Figure 3.8.2-5 (Sheet 3 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Surface Meridional Stress (ksi)

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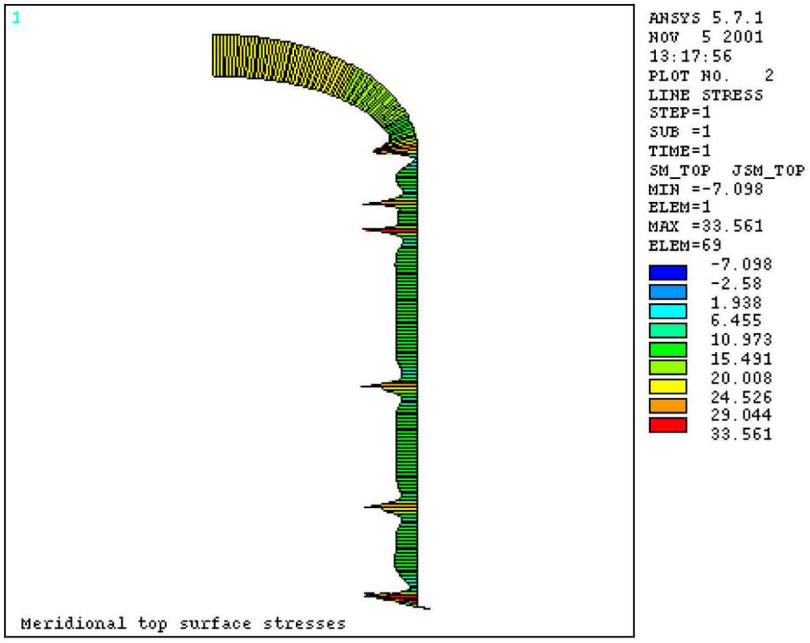
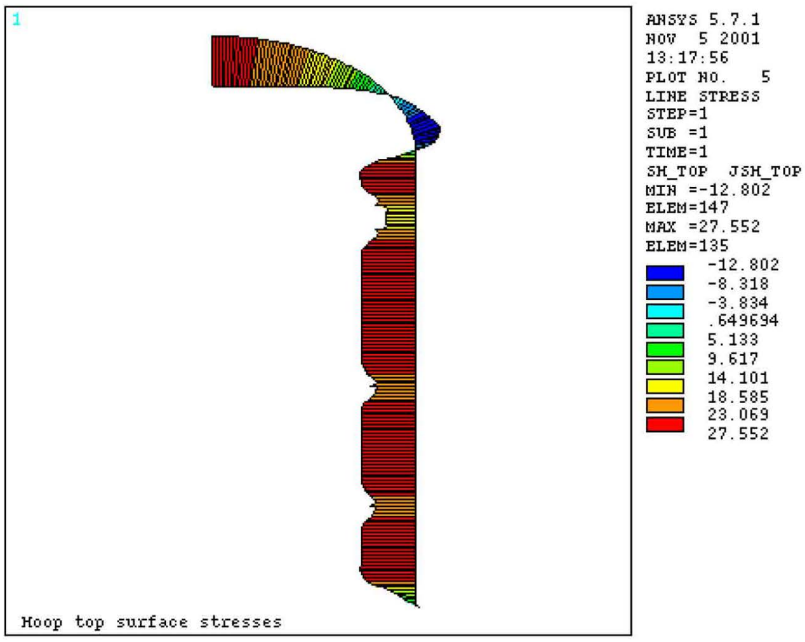


Meridional Top Surface Stress (ksi)



Circumferential Top Surface Stress (ksi)

Figure 3.8.2-5 (Sheet 4 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Outside Surface Stresses (ksi)

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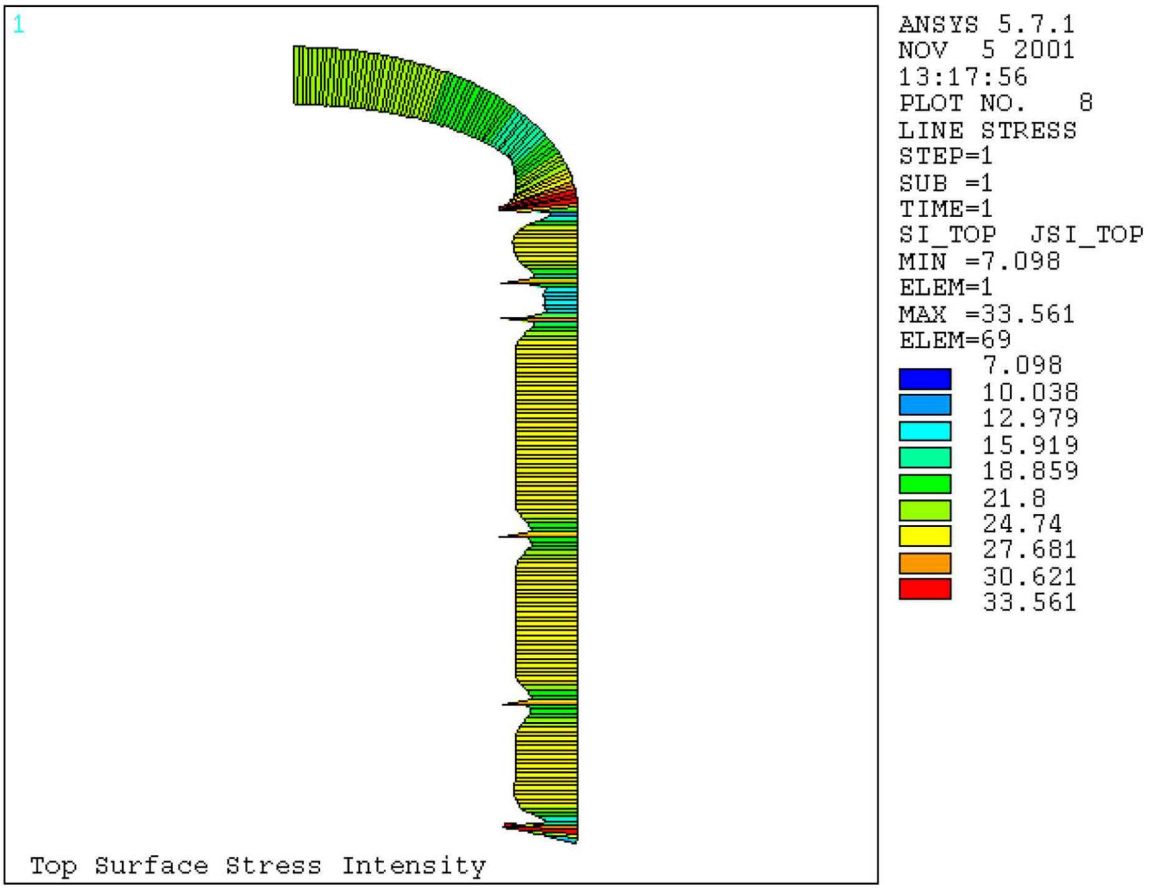

Figure 3.8.2-5 (Sheet 5 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Outer Stress Intensity (ksi)

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Figure 3.8.2-6 (Sheet 1 of 2)
Containment Vessel Axisymmetric Model

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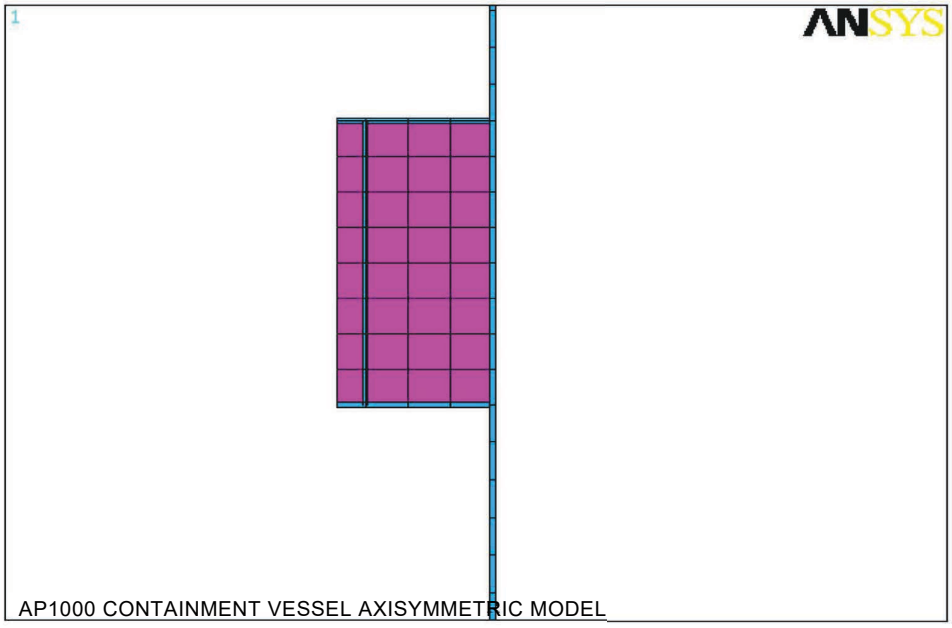
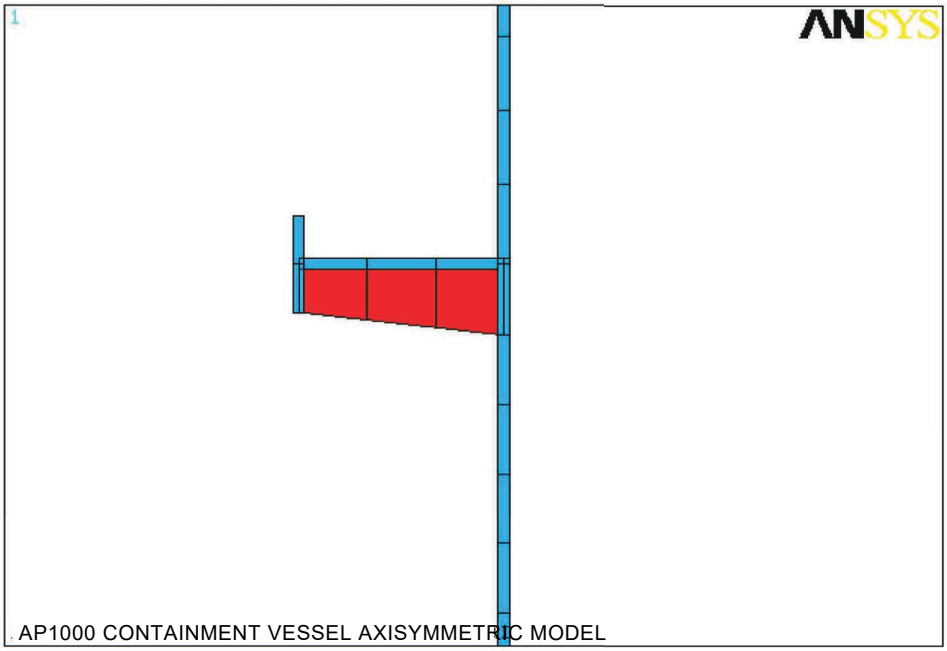


Crane Girder



Internal Stiffener at Elev. 170′–0″

Figure 3.8.2-6 (Sheet 2 of 2)
Containment Vessel Axisymmetric Model

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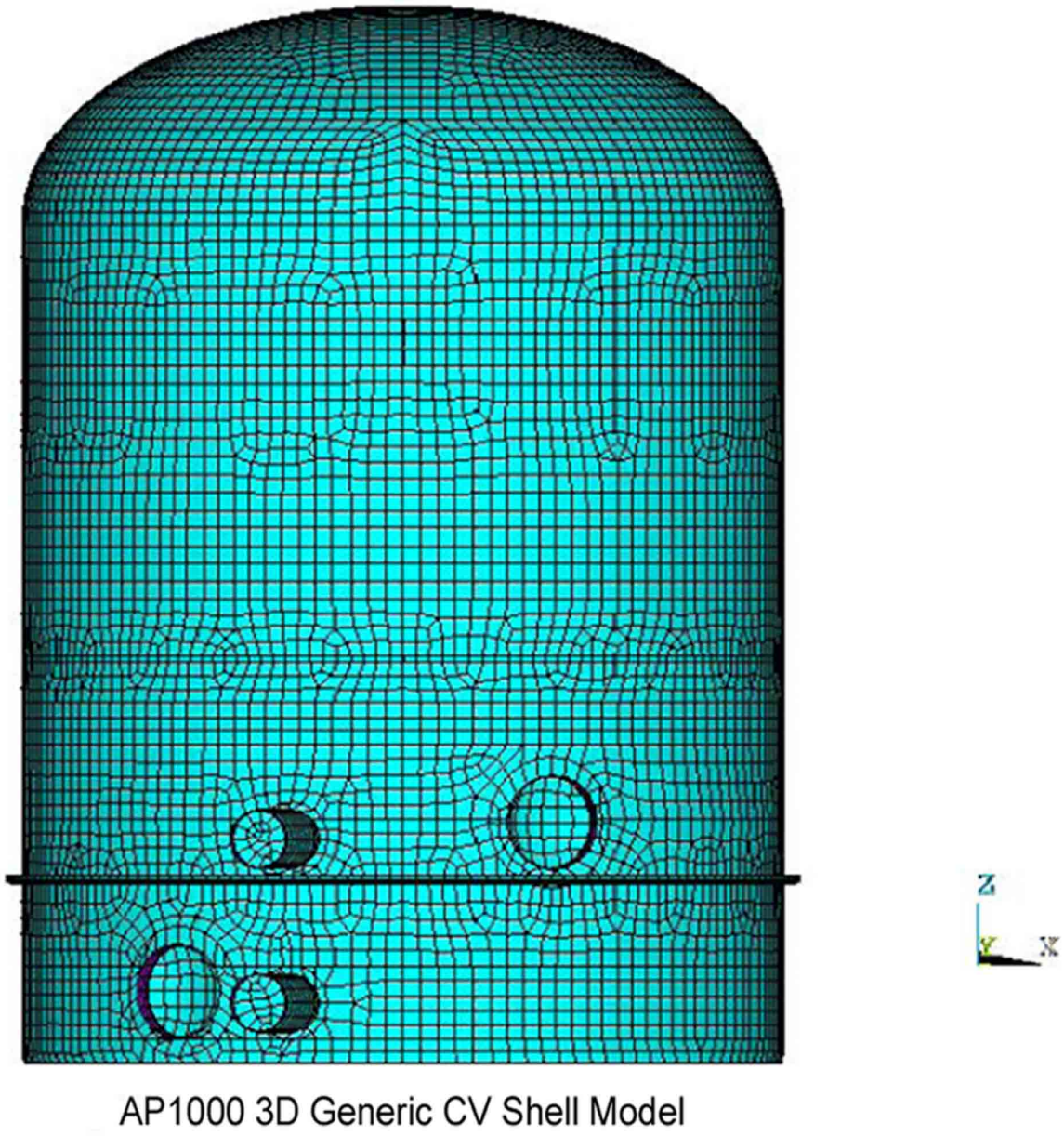

Figure 3.8.2-7
Finite Element Model for Local Buckling Analyses

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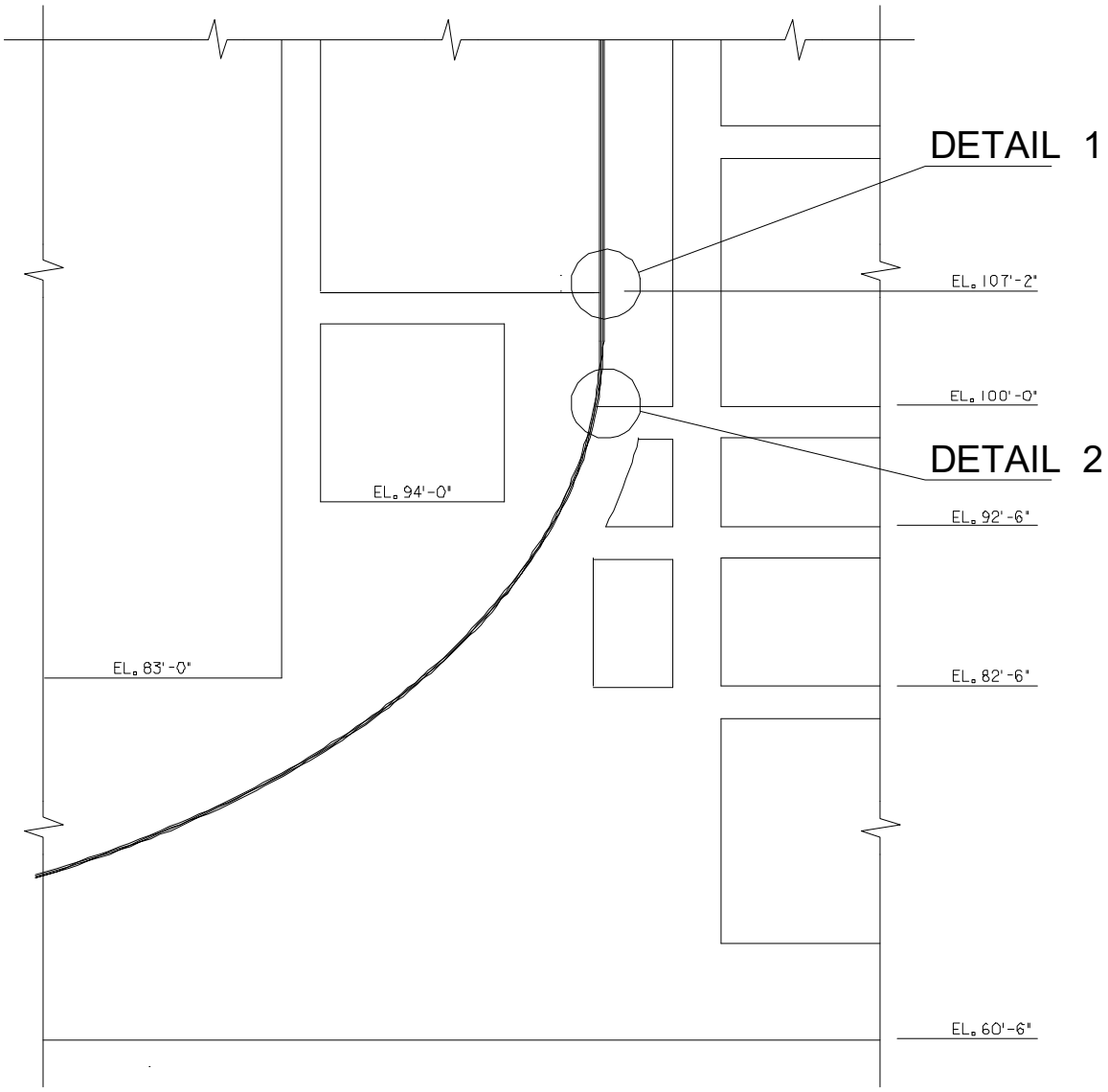

Figure 3.8.2-8 (Sheet 1 of 2)
Location of Containment Seal

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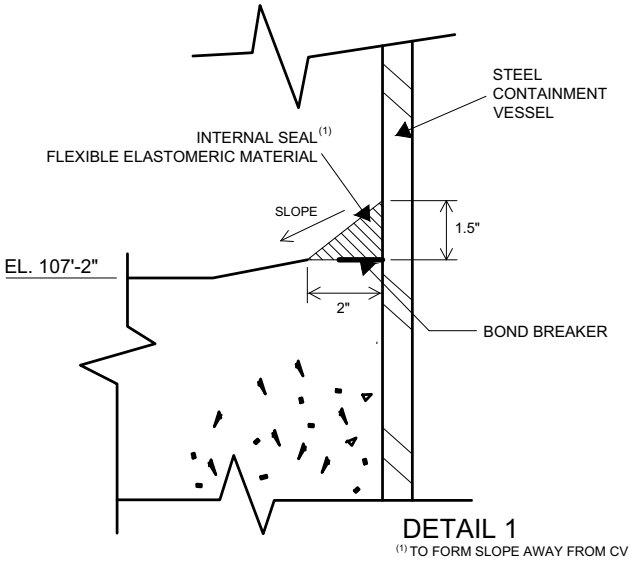

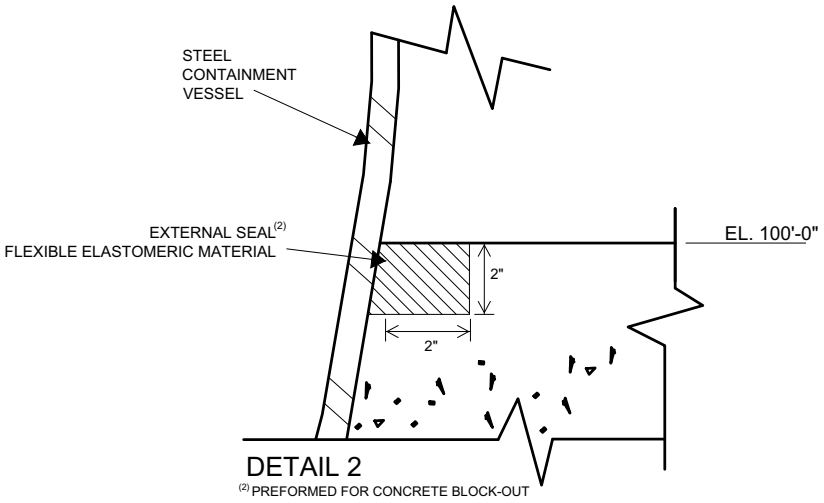

GENERAL NOTE: DIMENSIONS, ELEVATIONS AND CONFIGURATIONS SHOWN ARE APPROXIMATE AND CAN VARY WITH DIFFERENT SEAL MATERIALS. ALTERNATE SEAL CONFIGURATIONS AT ELEVATIONS OTHER THAN TYPICAL CAN BE UTILIZED WHERE REQUIRED TO ACCOMMODATE INTERFERENCES PROVIDED THE SEAL PREVENTS MOISTURE ACCUMULATION BETWEEN THE VESSEL AND CONCRETE.

Figure 3.8.2-8 (Sheet 2 of 2)
Seal Sections and Details

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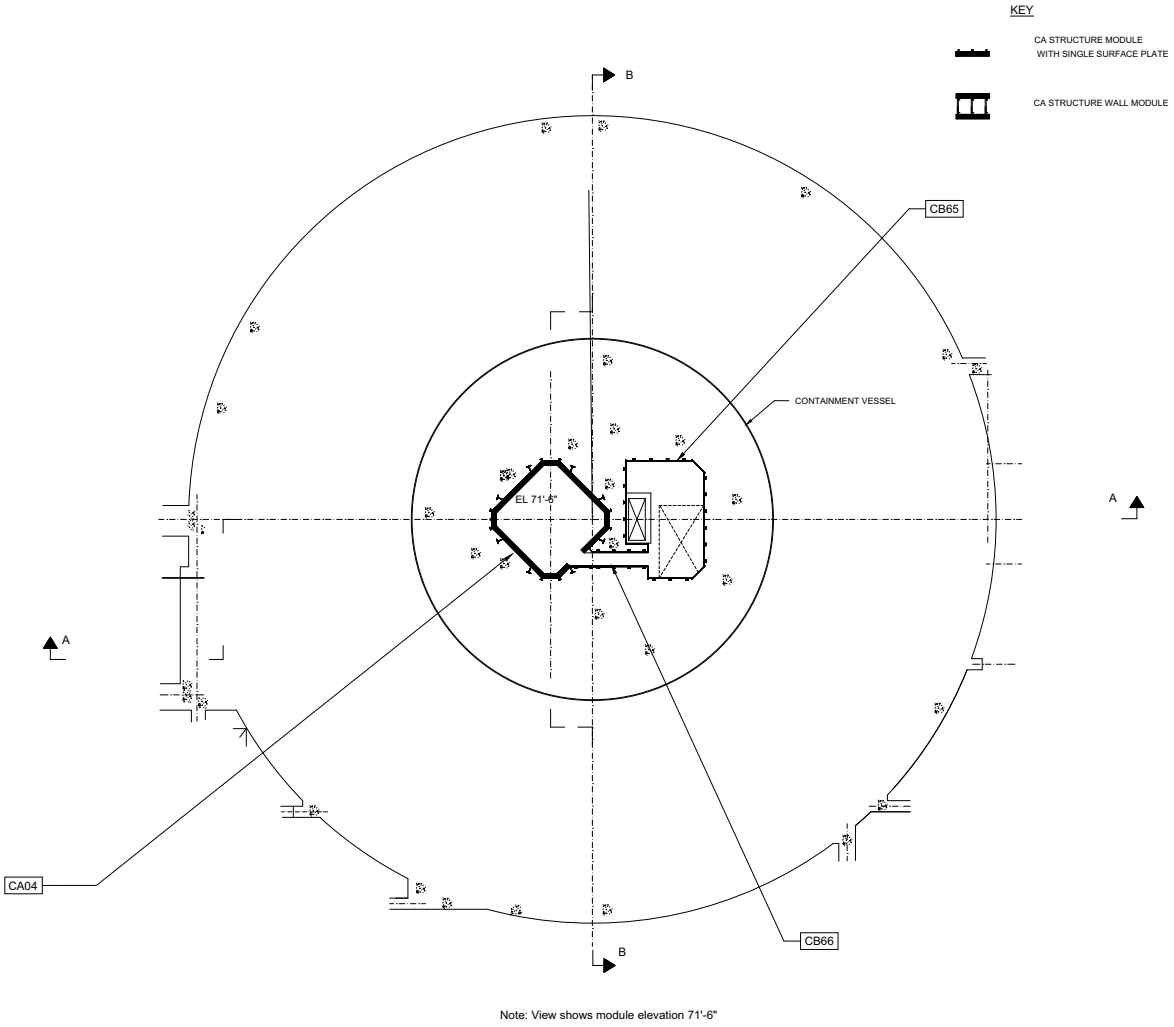

Figure 3.8.3-1 (Sheet 1 of 7)
[*Structural Modules in Containment Internal Structures*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Figure 3.8.3-1 (Sheet 2 of 7)
[*Structural Modules in Containment Internal Structures*]*

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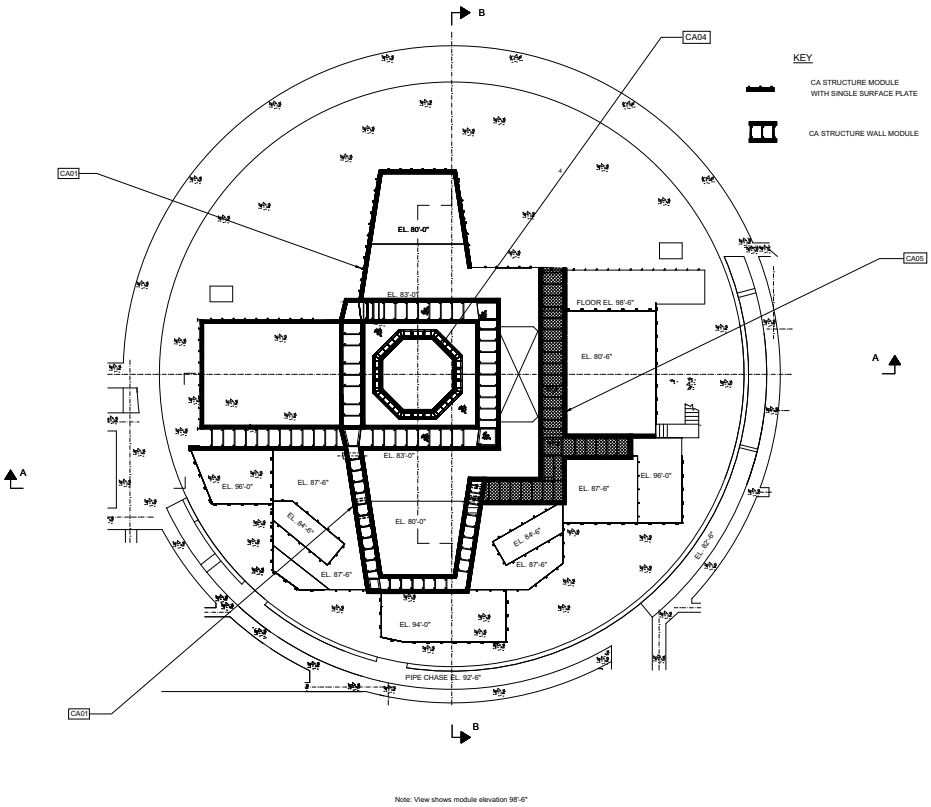

Figure 3.8.3-1 (Sheet 3 of 7)
[*Structural Modules in Containment Internal Structures*]*

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Figure 3.8.3-1 (Sheet 4 of 7)
[*Structural Modules in Containment Internal Structures*]*

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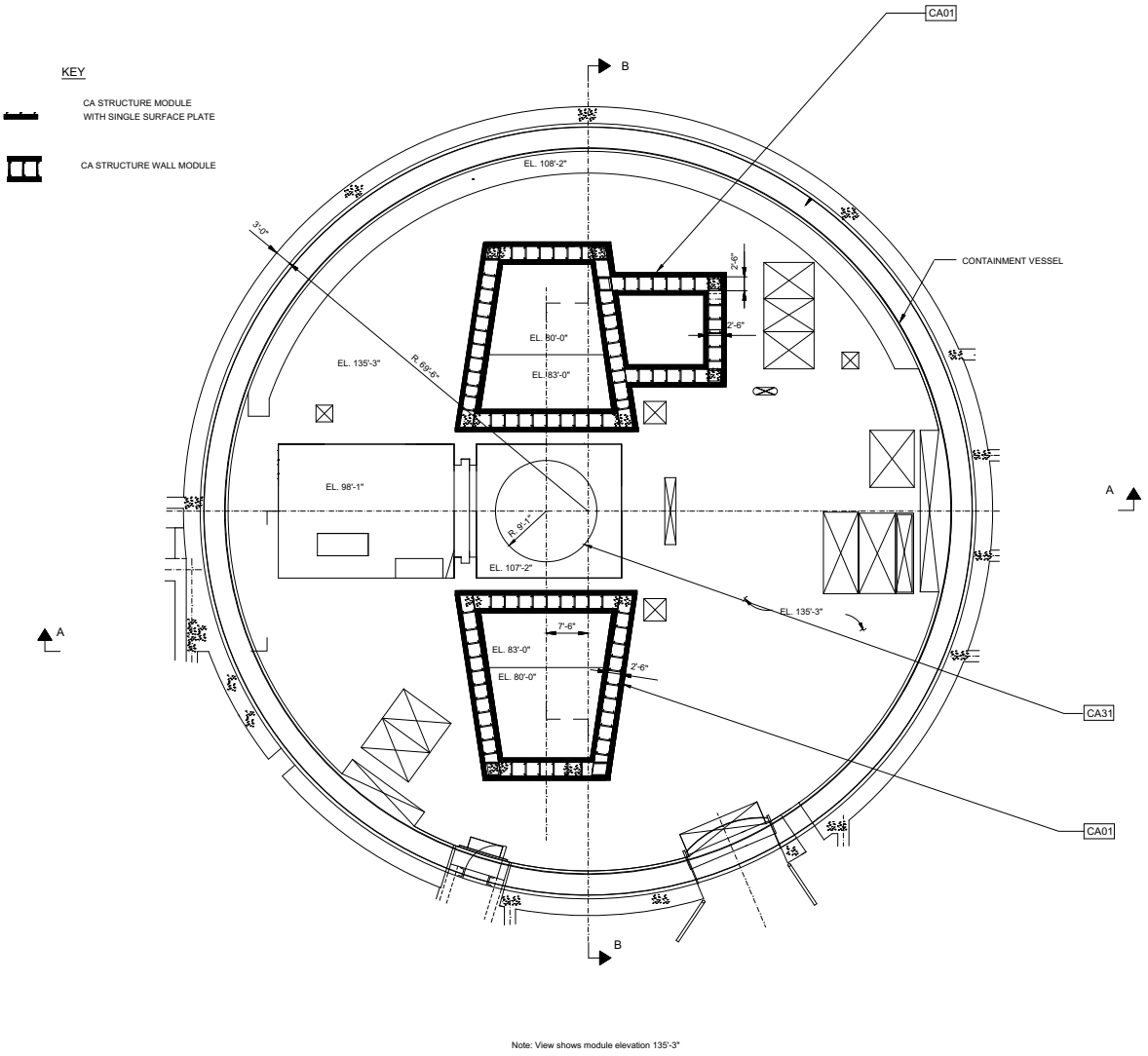

Figure 3.8.3-1 (Sheet 5 of 7)
[*Structural Modules in Containment Internal Structures*]*

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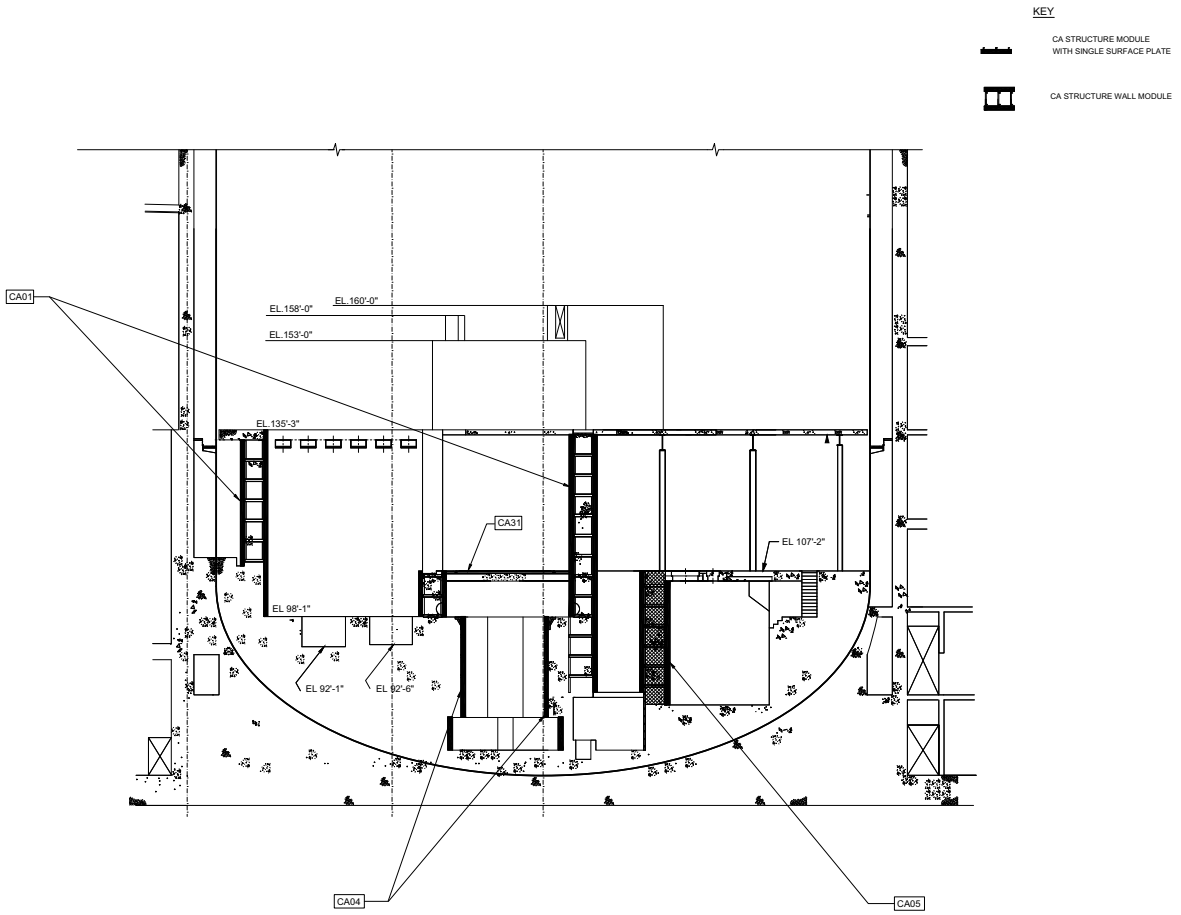

Figure 3.8.3-1 (Sheet 6 of 7)
[*Structural Modules in Containment Internal Structures*]*

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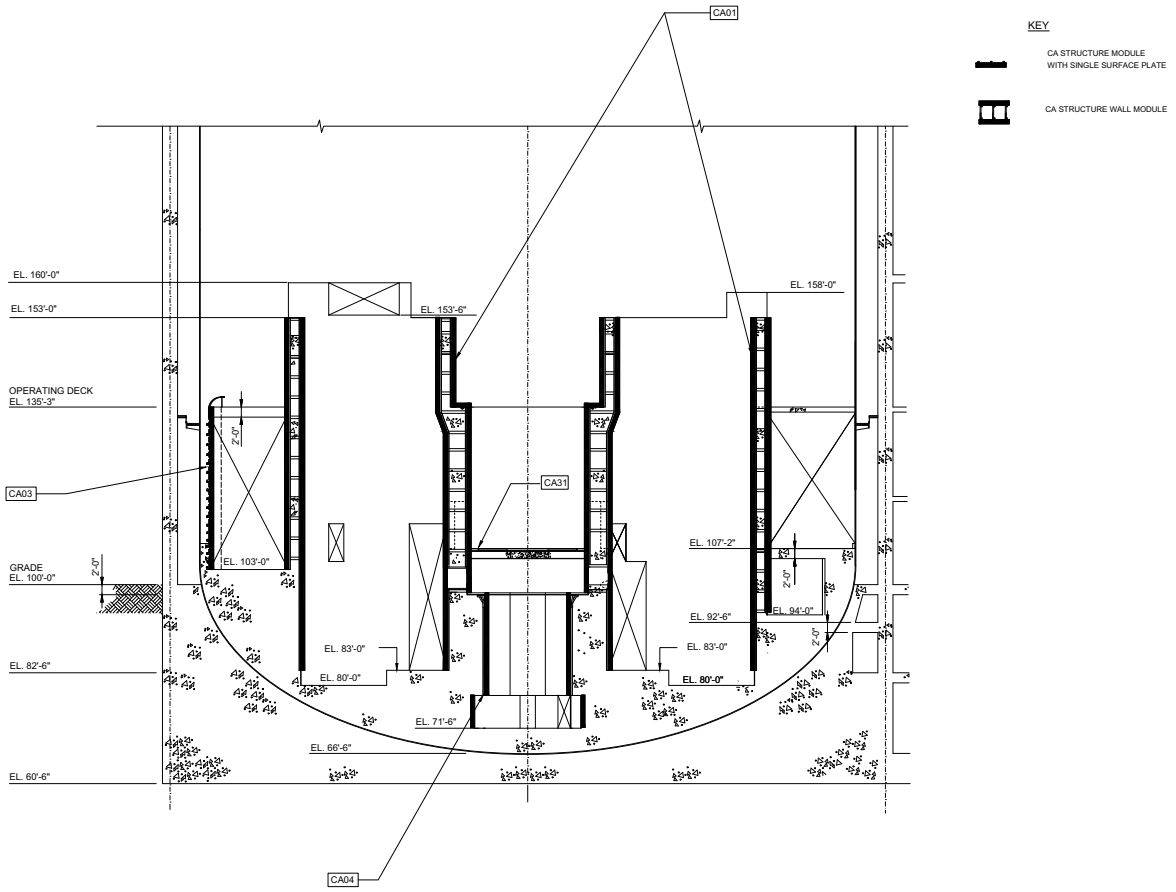

Figure 3.8.3-1 (Sheet 7 of 7)
[*Structural Modules in Containment Internal Structures*]*

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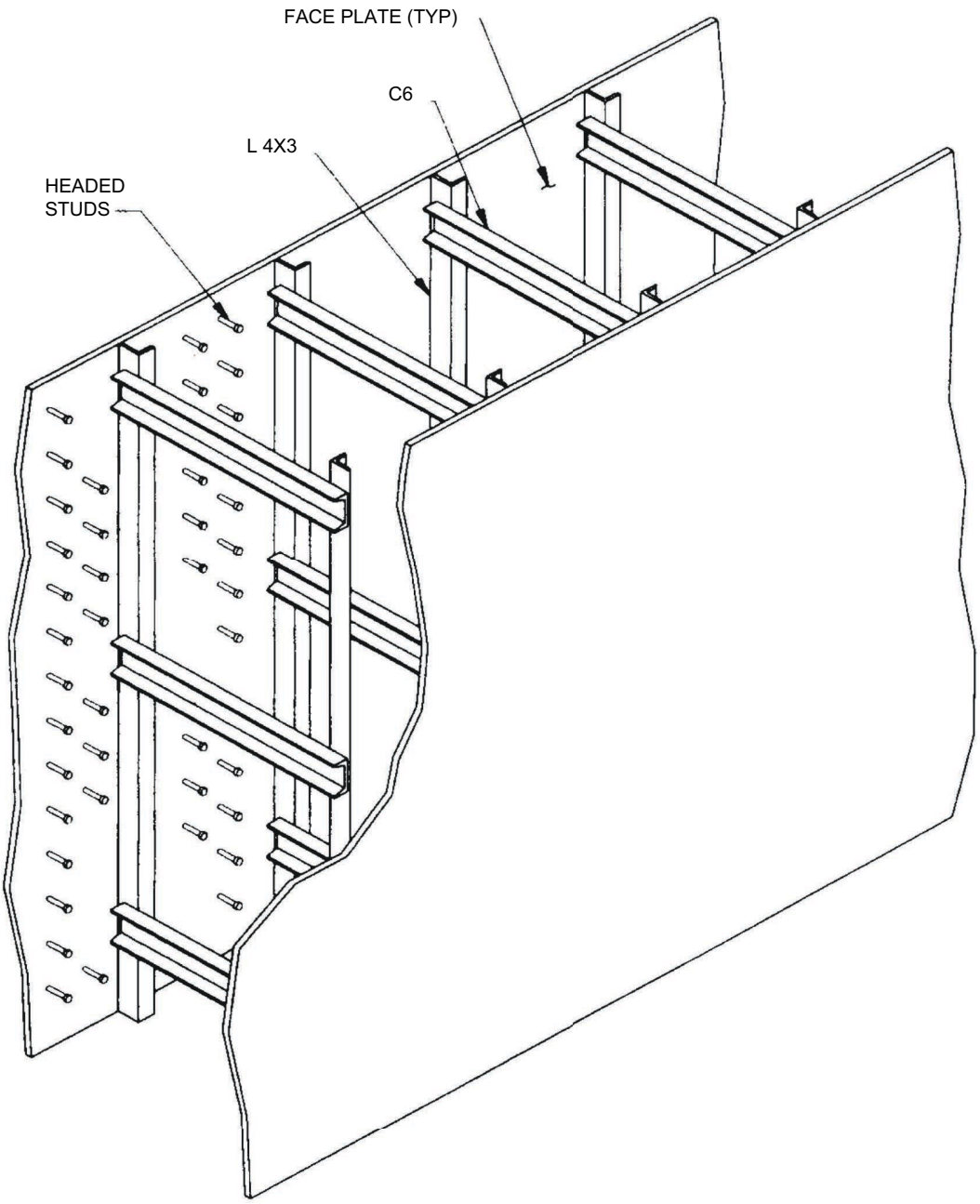

Note: See Figure 3.8.3-8 for fabrication detail.

Figure 3.8.3-2
[*Typical Structural Wall Module*]*

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NOTES:

1. DETAILS SHOWN ARE REPRESENTATIVE OF FLOOR MODULES INSIDE CONTAINMENT. CHANNELS AND WIDE FLANGE BEAMS USED IN SOME LOCATIONS ARE NOT SHOWN. REFER TO SUBSECTION 3.8.3.1.4 AND OTHER NOTES FOR ADDITIONAL INFORMATION ABOUT DESIGN DETAILS AND VARIATIONS.
2. REINFORCEMENT USE, SIZE, AND SPACING IN THE FLOOR MODULE CONCRETE SATISFY THE REQUIREMENTS OF ACI 349. WHERE USED, THE REINFORCEMENT SIZE RANGE IS #7 TO #11.
3. THE REINFORCEMENT AND FLOOR DESIGN ELEMENTS SHOWN ARE FOR LOCATIONS AWAY FROM OPENINGS, PENETRATIONS AND OTHER OBSTRUCTIONS.
4. DESIGN OF THE PLATES, BEAMS, AND STIFFENERS IN THE FLOOR, INCLUDING PLATE SIZE AND SPACING, AND TYPE, SIZE, AND SPACING OF STRUCTURAL SHAPES VARIES AND SATISFIES THE REQUIREMENTS OF AISC N690.
5. REFER TO SUBSECTION 3.8.3.5.4 FOR INFORMATION ABOUT THE SHEAR STUDS IN THE FLOOR MODULE.

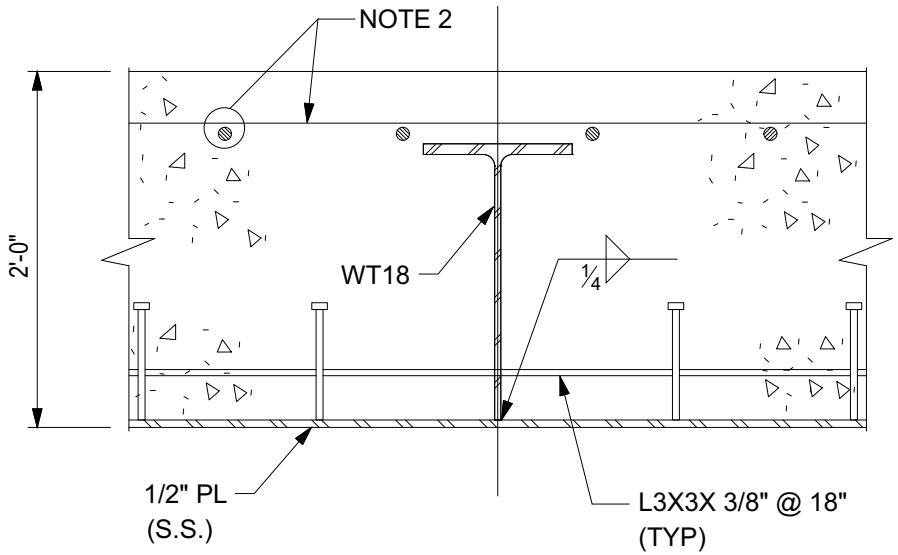

Figure 3.8.3-3
Structural Floor Module

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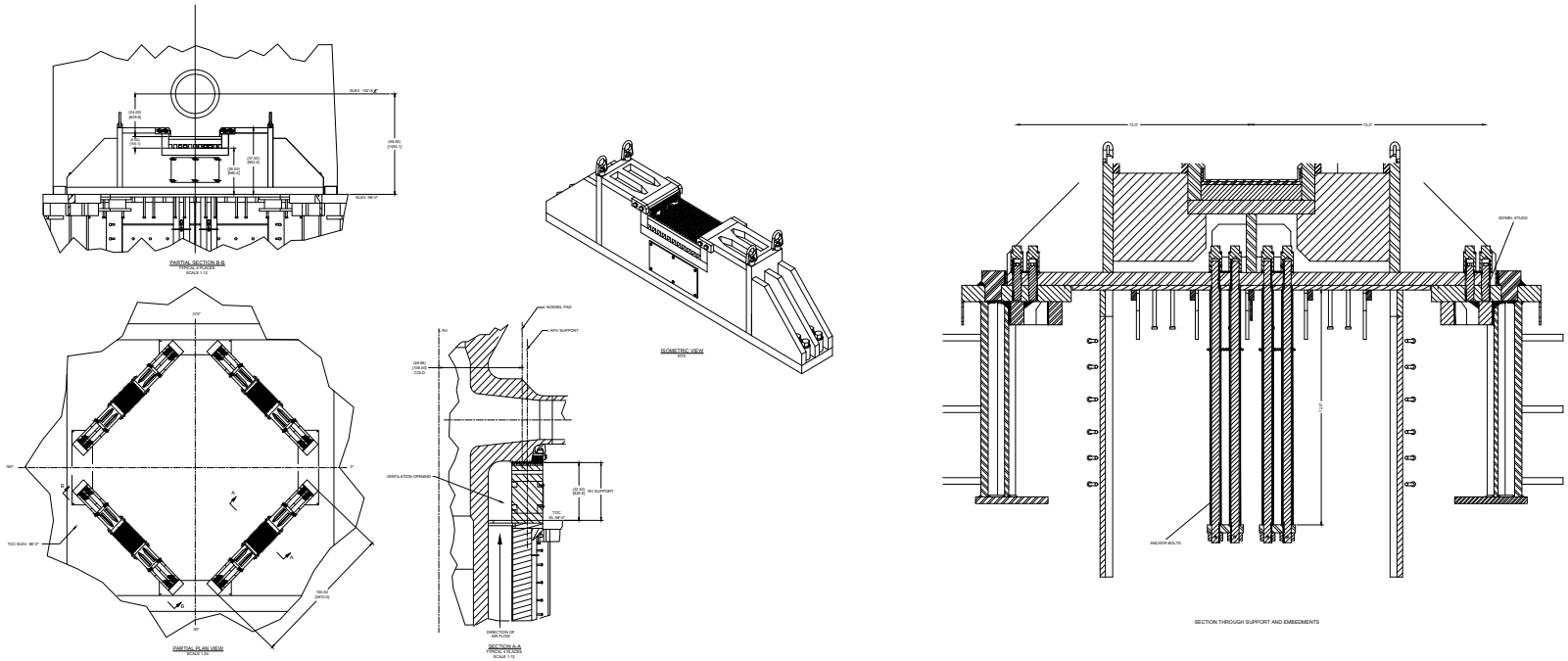

Figure 3.8.3-4
Reactor Vessel Supports

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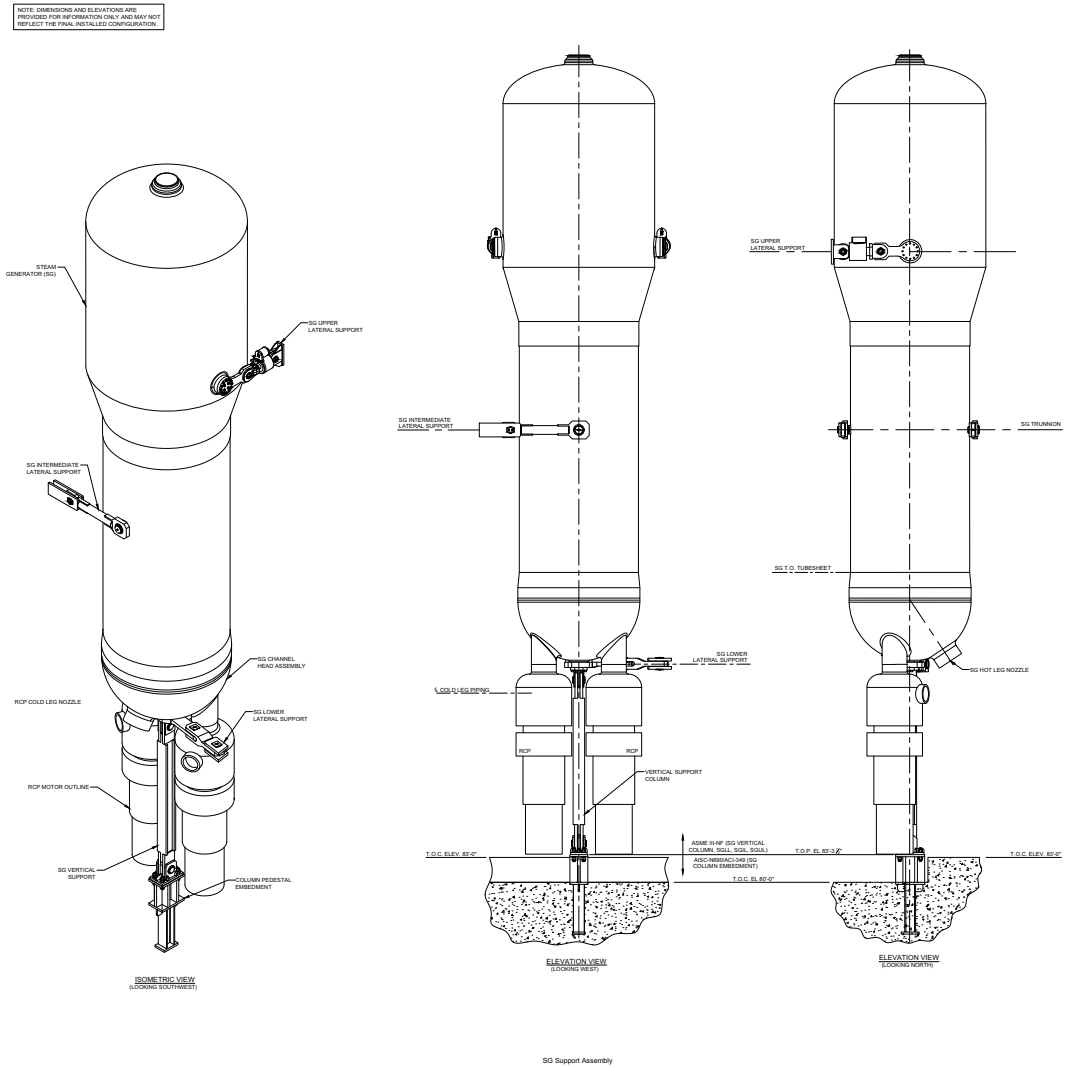

Figure 3.8.3-5 (Sheet 1 of 5)
Steam Generator Supports

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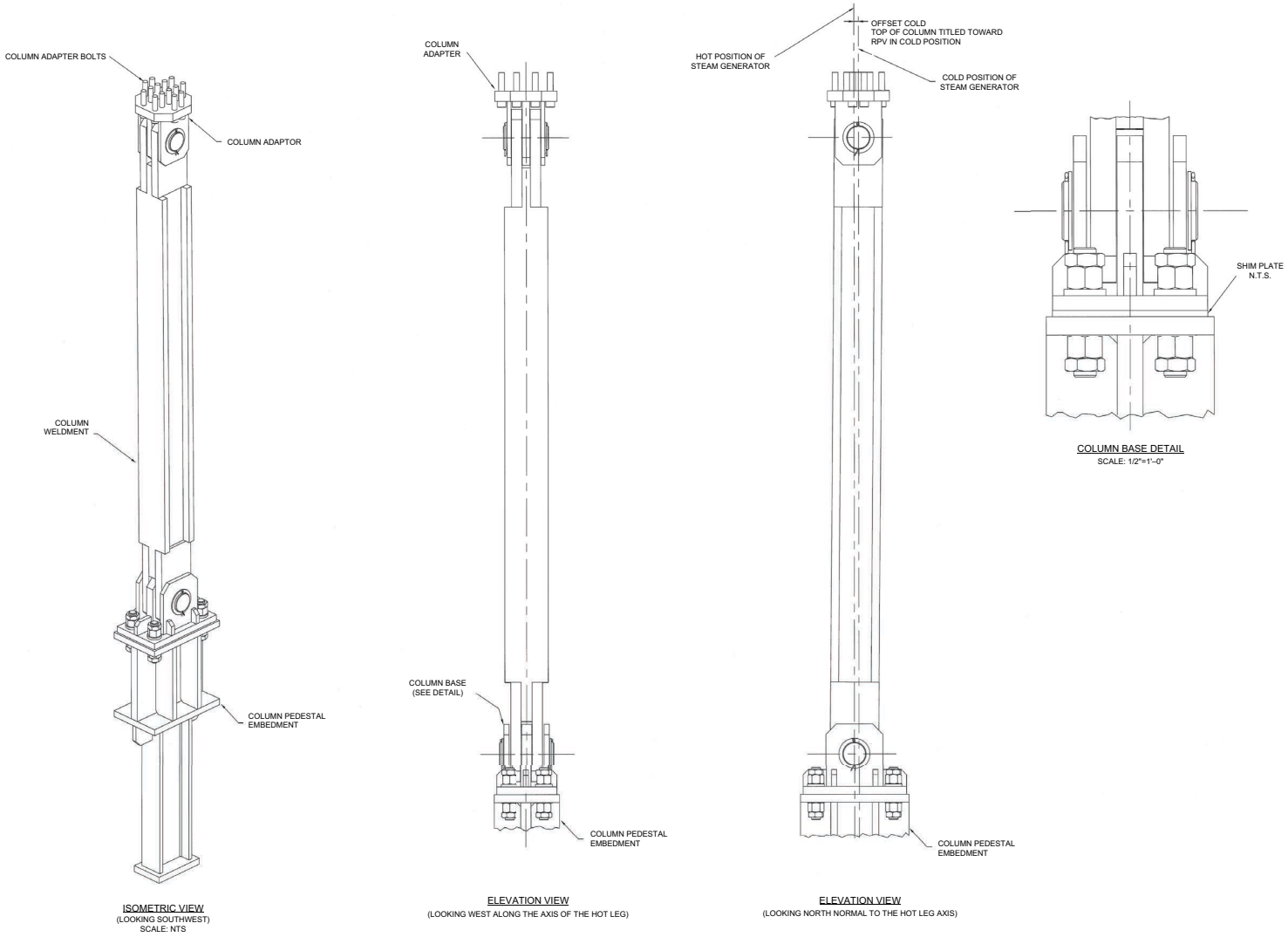

Figure 3.8.3-5 (Sheet 2 of 5)
Steam Generator Supports

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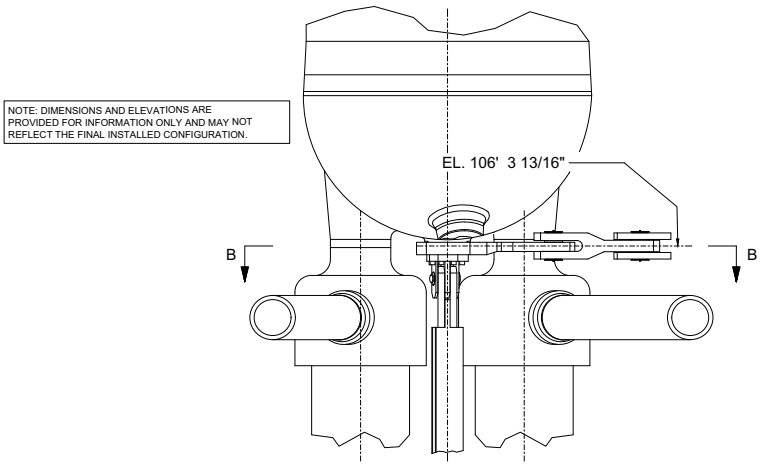
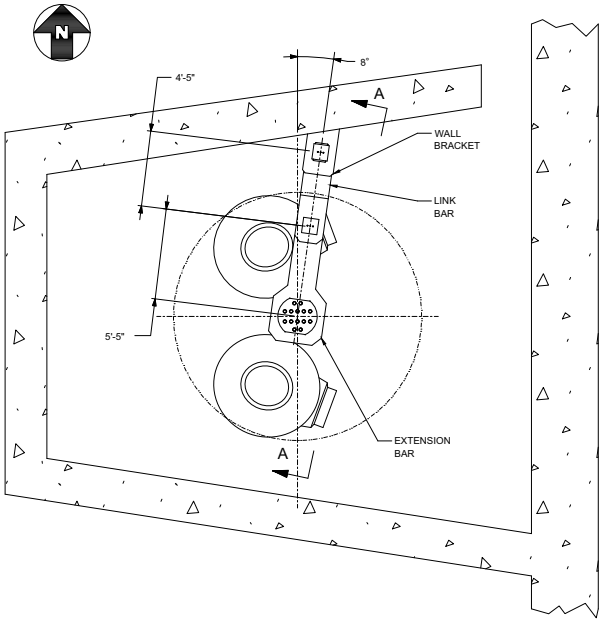


ELEVATION VIEW
(ROTATED VIEW A-A)

Lower Lateral Support Elevation View



PLAN VIEW
SECTION B-B

Lower Lateral Support Plan View

Figure 3.8.3-5 (Sheet 3 of 5)
Steam Generator Supports

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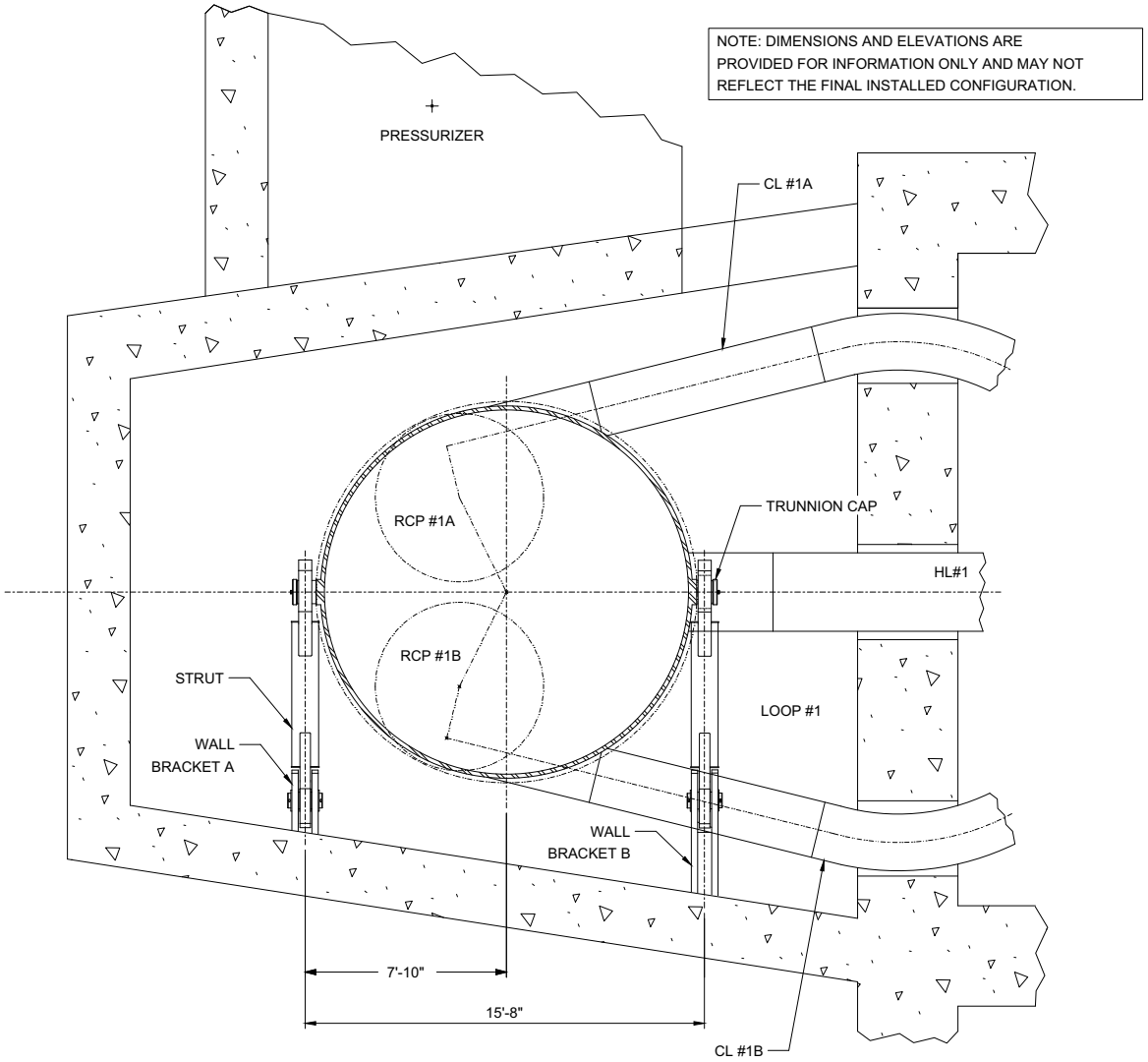

INTERMEDIATE LATERAL SUPPORT

Figure 3.8.3-5 (Sheet 4 of 5)
Steam Generator Supports

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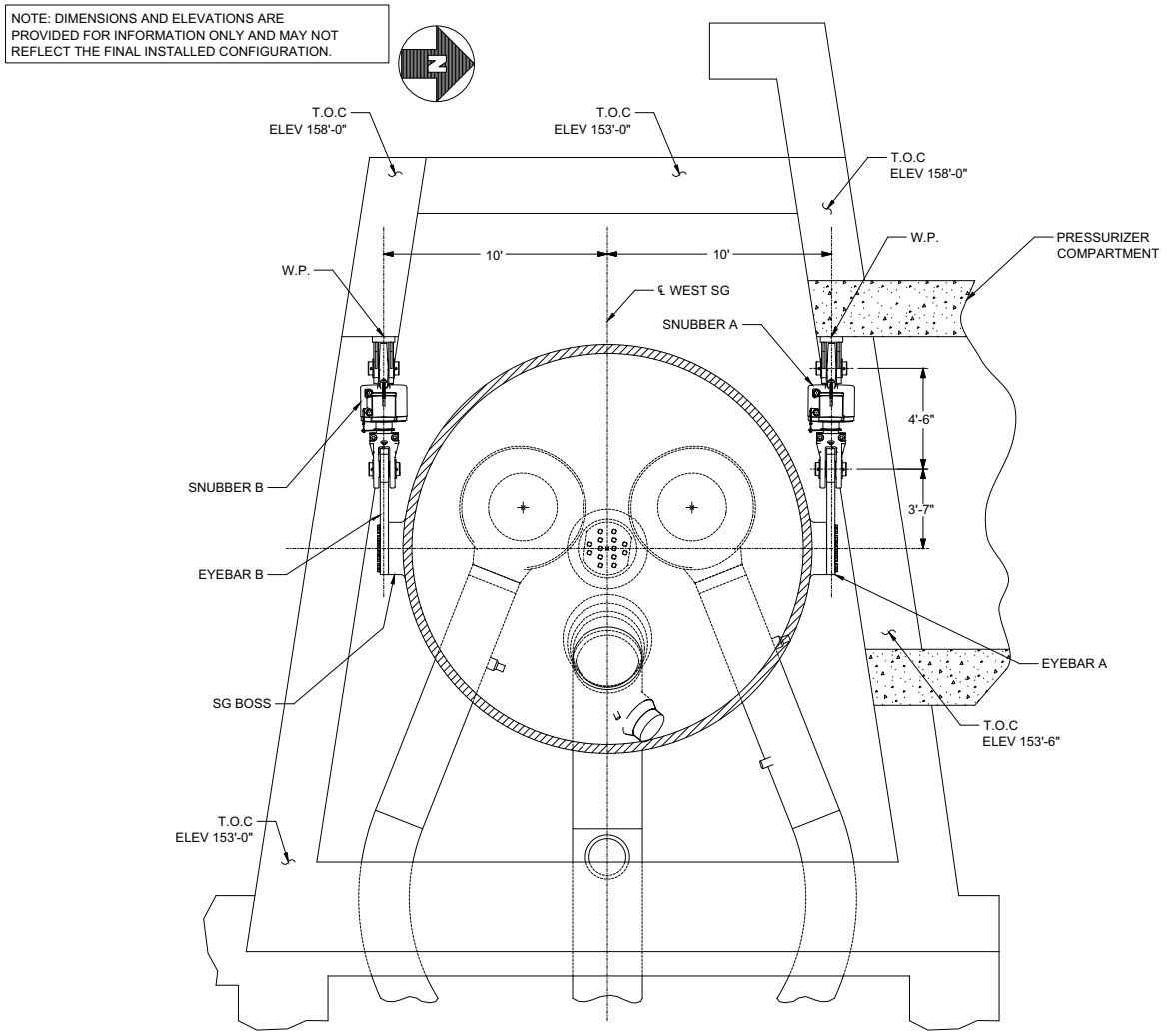

PLAN VIEW

Upper Lateral Support

Figure 3.8.3-5 (Sheet 5 of 5)
Steam Generator Supports

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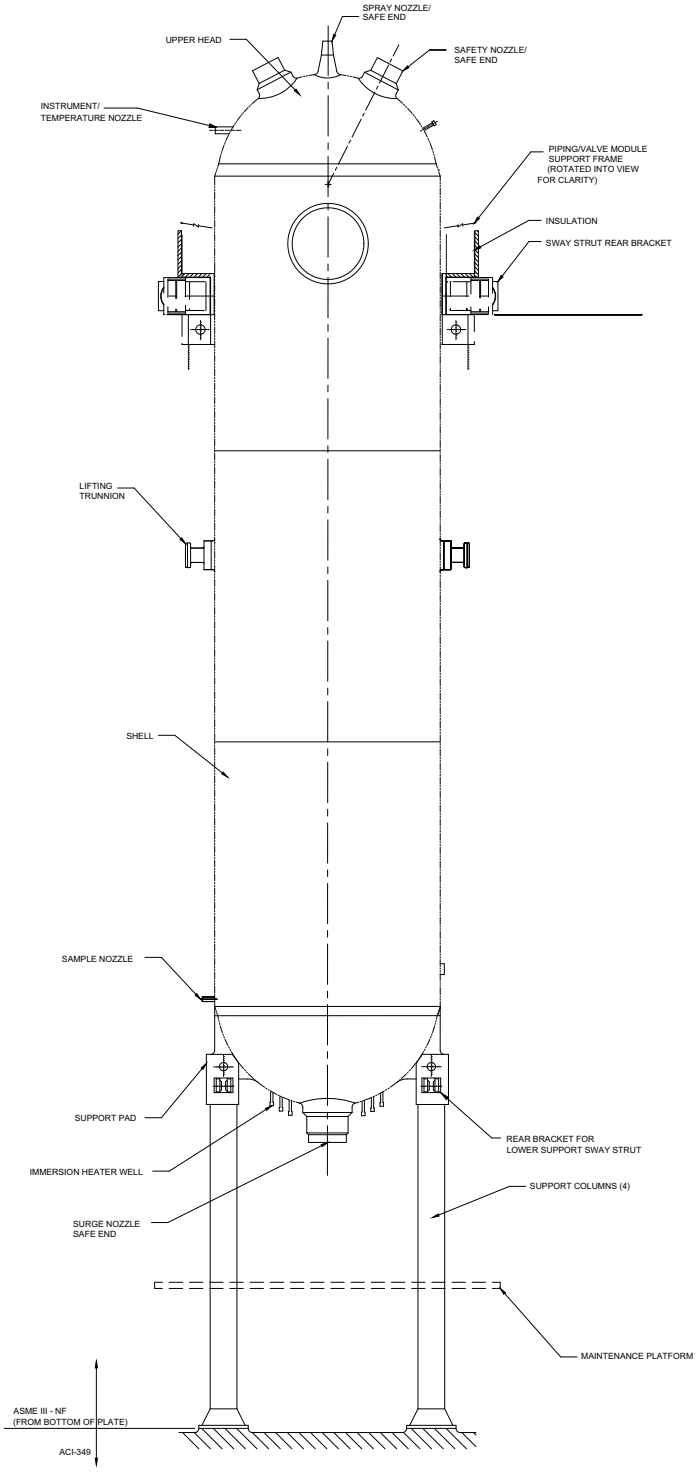

Figure 3.8.3-6 (Sheet 1 of 4)
Pressurizer Support Columns

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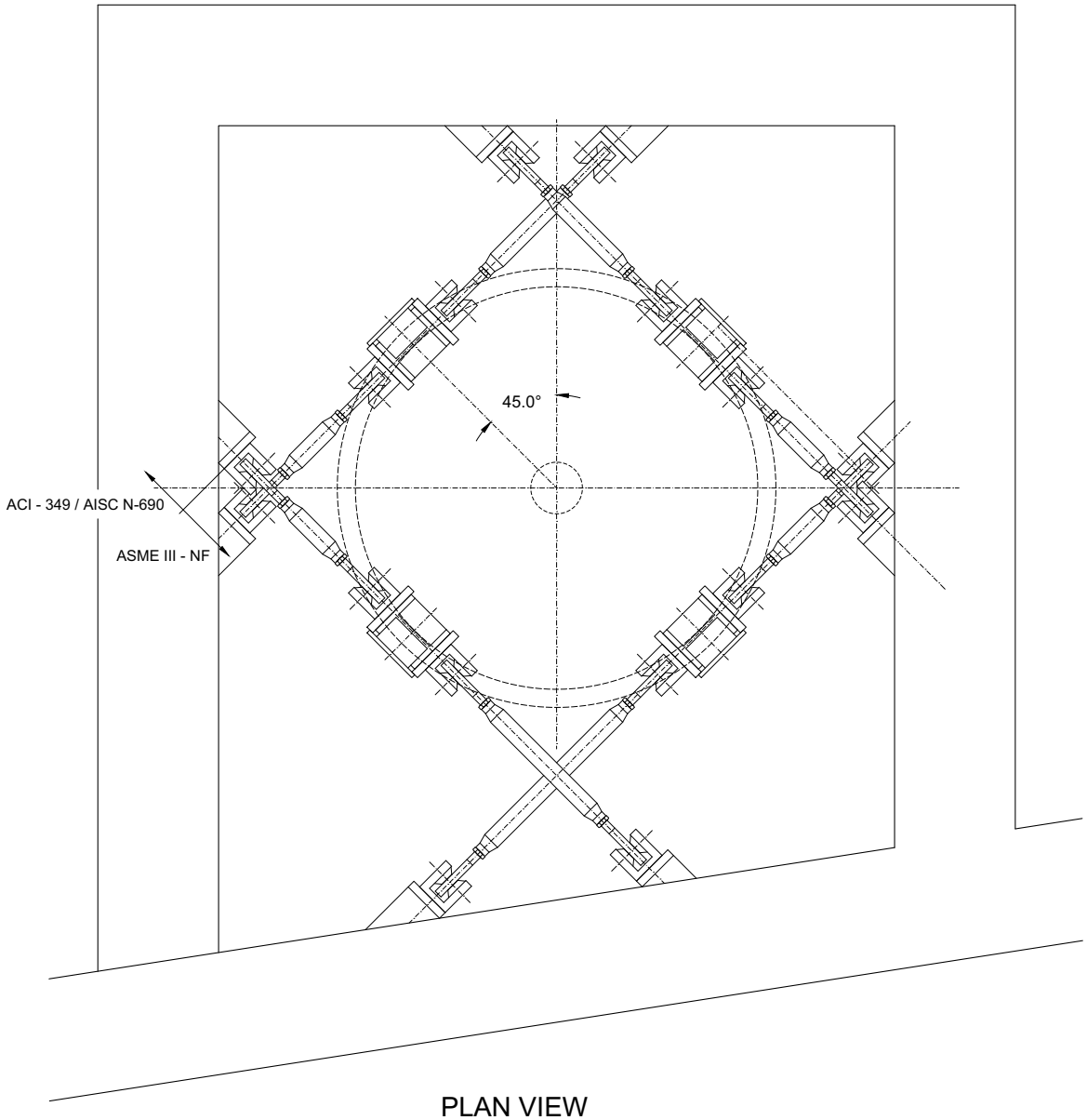

Figure 3.8.3-6 (Sheet 2 of 4)
Pressurizer Lower Lateral Supports

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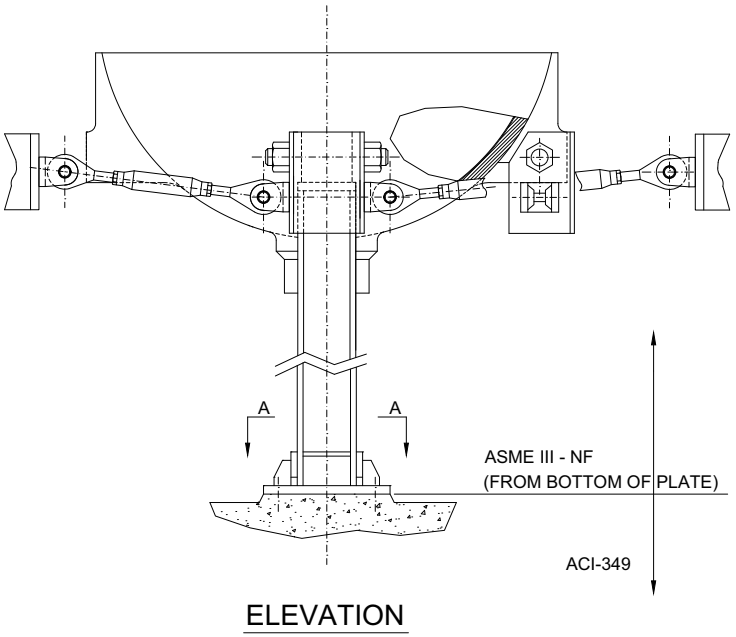
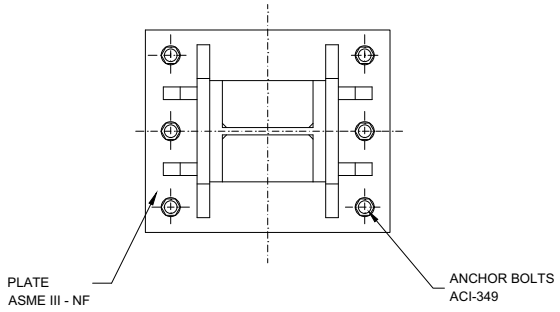


ELEVATION



VIEW A-A

Figure 3.8.3-6 (Sheet 3 of 4)
Pressurizer Lower Supports

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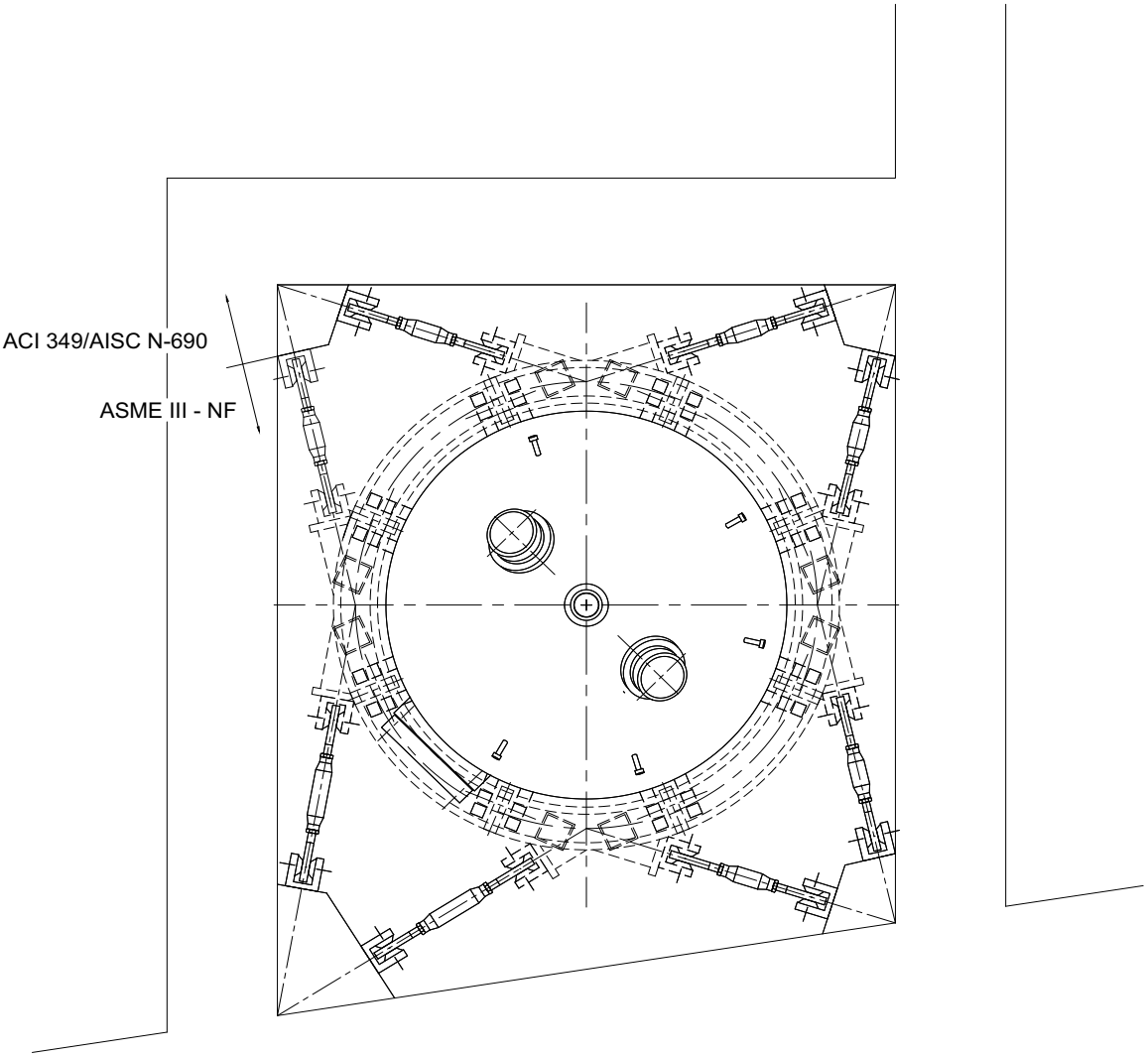

PLAN VIEW

Figure 3.8.3-6 (Sheet 4 of 4)
Pressurizer Upper Supports

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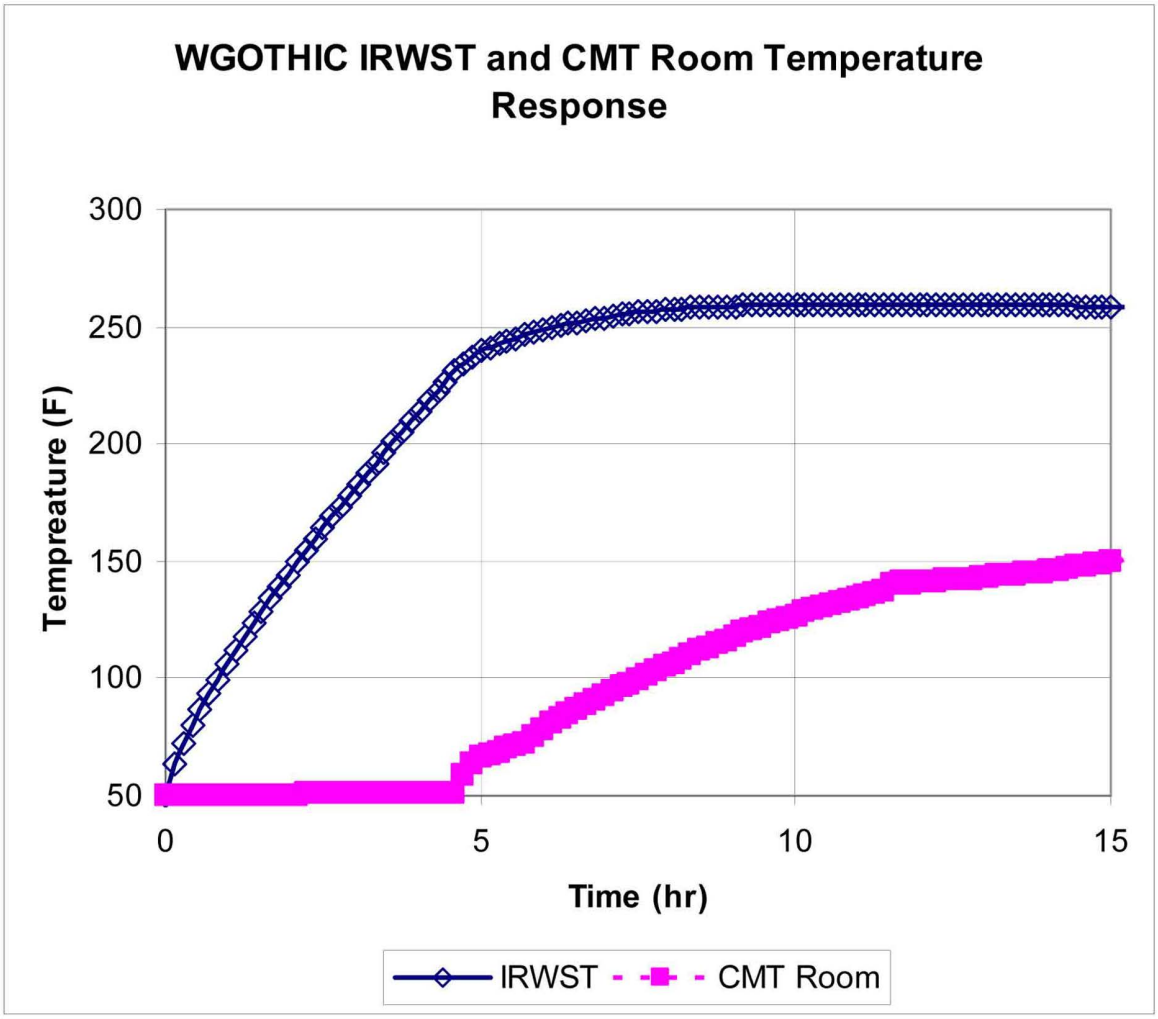

Figure 3.8.3-7
IRWST Temperature Transient

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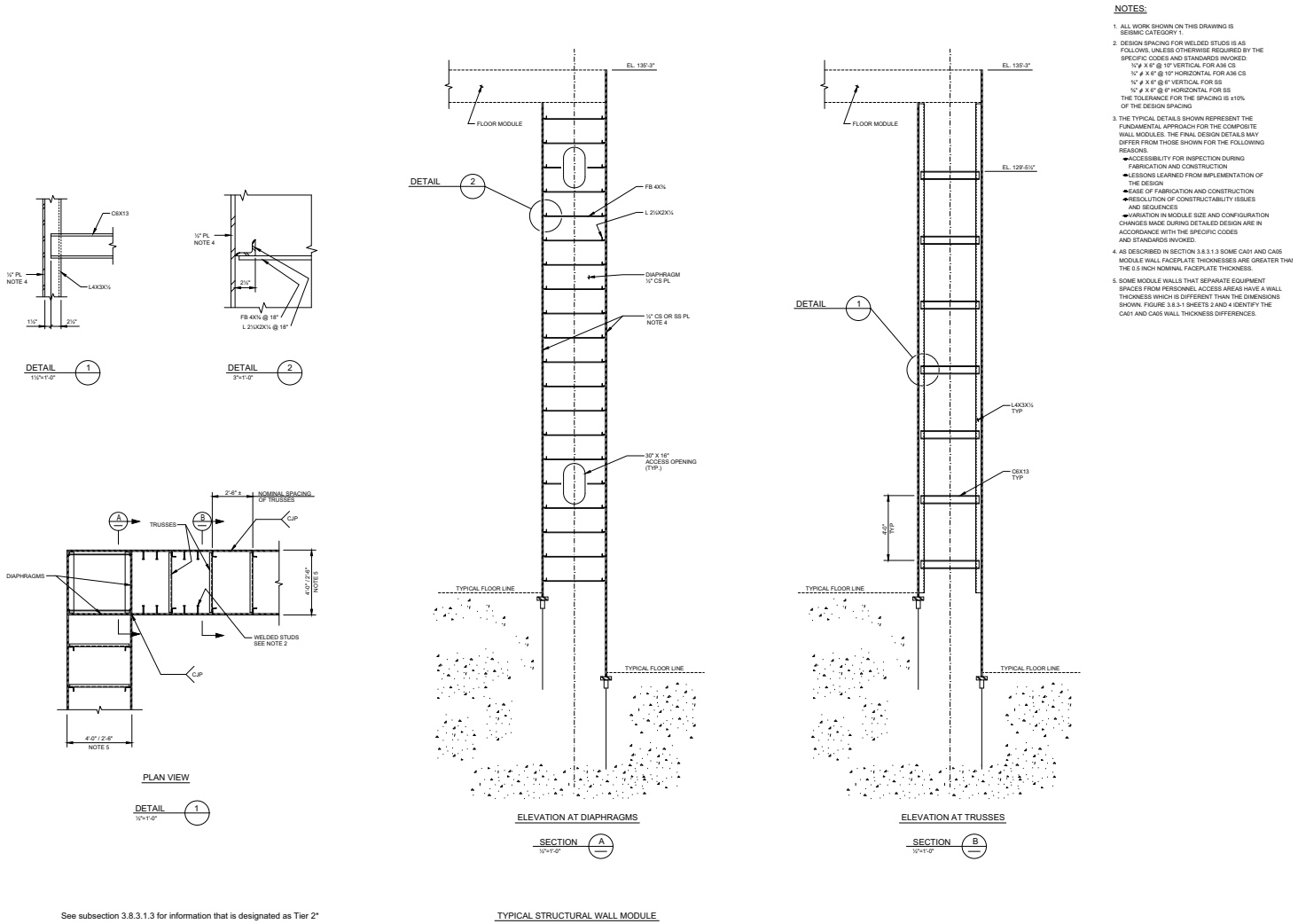

Figure 3.8.3-8 (Sheet 1 of 3)
[*Structural Modules – Typical Design Details*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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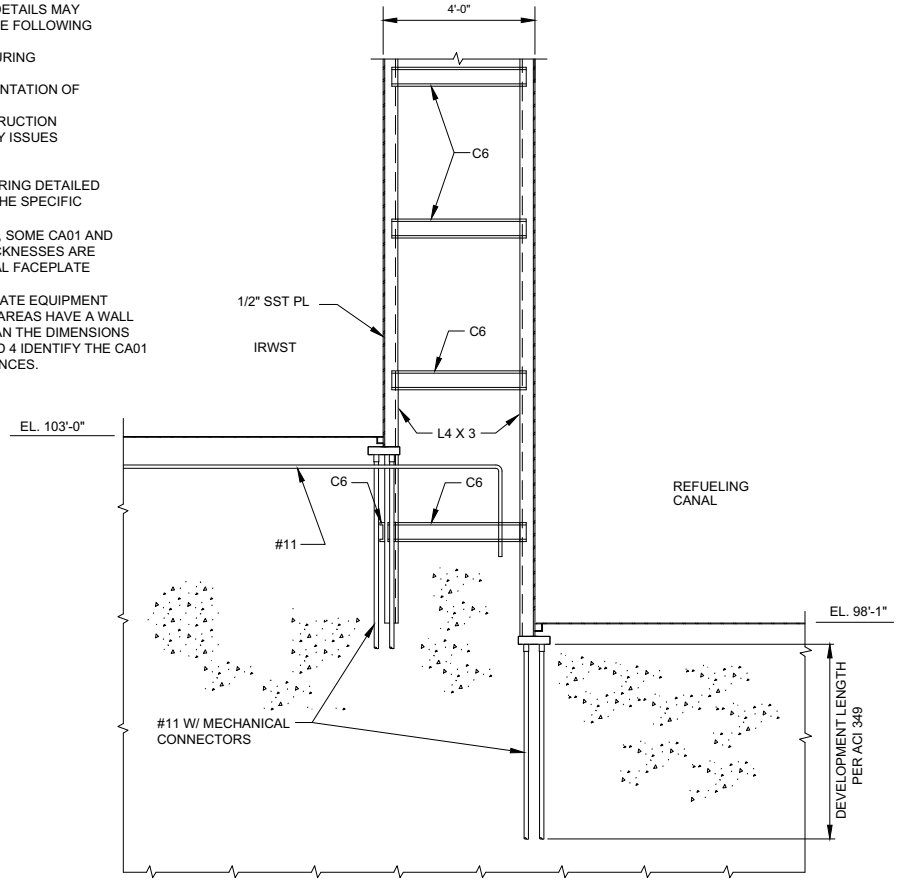
NOTES:

THE TYPICAL DETAILS SHOWN REPRESENT THE FUNDAMENTAL APPROACH FOR THE COMPOSITE WALL MODULES. THE FINAL DESIGN DETAILS MAY DIFFER FROM THOSE SHOWN FOR THE FOLLOWING REASONS.

- ACCESSIBILITY FOR INSPECTION DURING FABRICATION AND CONSTRUCTION
- LESSONS LEARNED FROM IMPLEMENTATION OF THE DESIGN
- EASE OF FABRICATION AND CONSTRUCTION
- RESOLUTION OF CONSTRUCTABILTY ISSUES AND SEQUENCES
- VARIATION IN MODULE SIZE AND CONFIGURATION CHANGES MADE DURING DETAILED DESIGN ARE IN ACCORDANCE WITH THE SPECIFIC CODES AND STANDARDS INVOKED.
- AS DESCRIBED IN SECTION 3.8.3.1.3, SOME CA01 AND CA05 MODULE WALL FACEPLATE THICKNESSES ARE GREATER THAN THE 0.5 INCH NOMINAL FACEPLATE THICKNESS.
- SOME MODULE WALLS THAT SEPARATE EQUIPMENT SPACES FROM PERSONNEL ACCESS AREAS HAVE A WALL THICKNESS WHICH IS DIFFERENT THAN THE DIMENSIONS SHOWN. FIGURE 3.8.3-1 SHEETS 2 AND 4 IDENTIFY THE CA01 AND CA05 WALL THICKNESS DIFFERENCES.



CHANGES TO THE MECHANICAL CONNECTOR DETAIL SHOWN WILL MAINTAIN A DIRECT LOAD PATH TO TRANSFER LOADS FROM THE MODULE FACE PLATES TO THE REINFORCEMENT BARS IN THE BASE CONCRETE THROUGH THE USE OF INTERVENING PLATES, MECHANICAL CONNECTORS, AND WELDS.

SEE SUBSECTION 3.8.3.1.3 FOR INFORMATION THAT IS DESIGNATED AS TIER 2*.

Figure 3.8.3-8 (Sheet 2 of 3)
[*Structural Modules – Typical Design Details*]*

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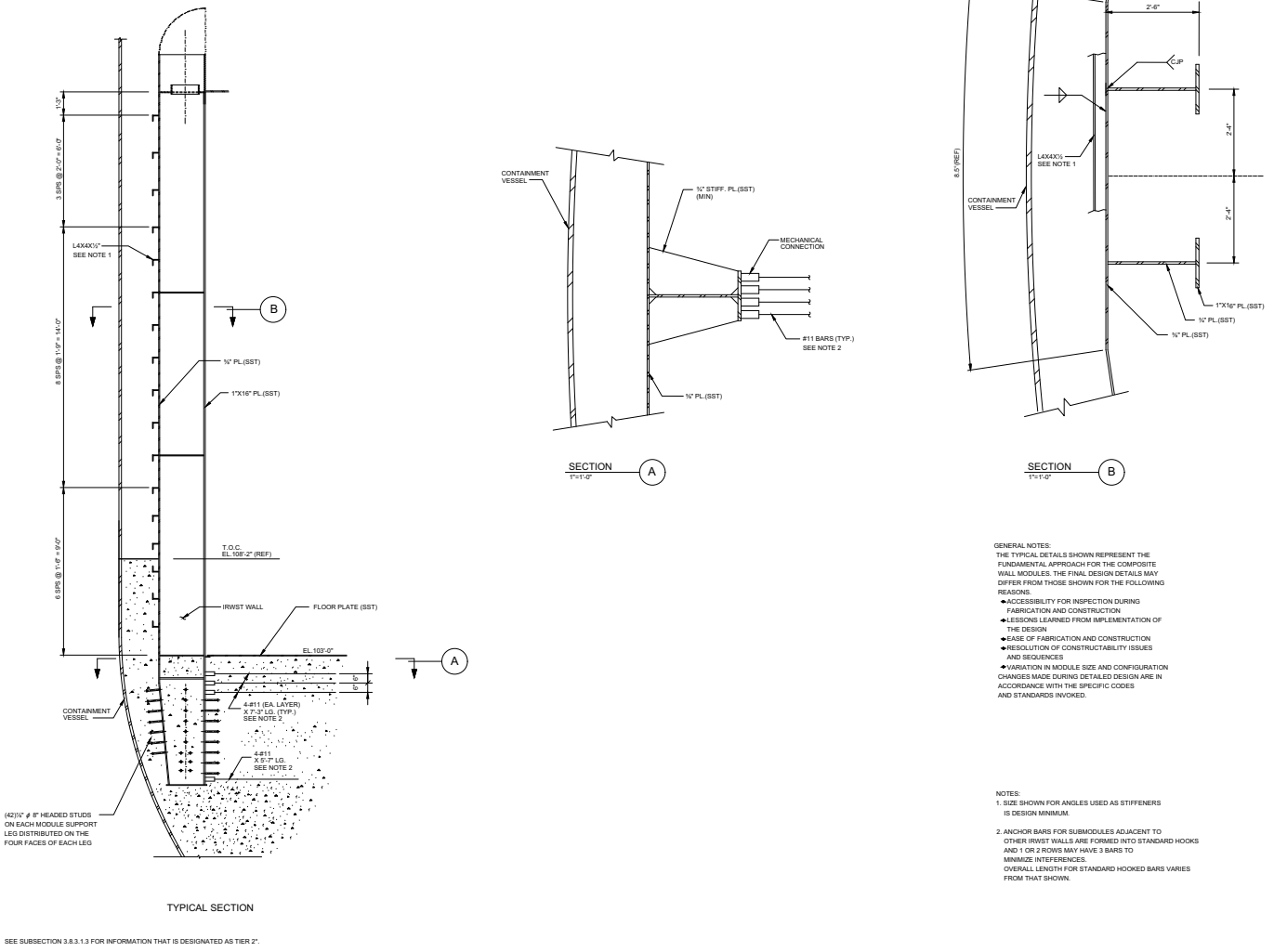

Figure 3.8.3-8 (Sheet 3 of 3)
[*Structural Modules – Typical Design Details*]*

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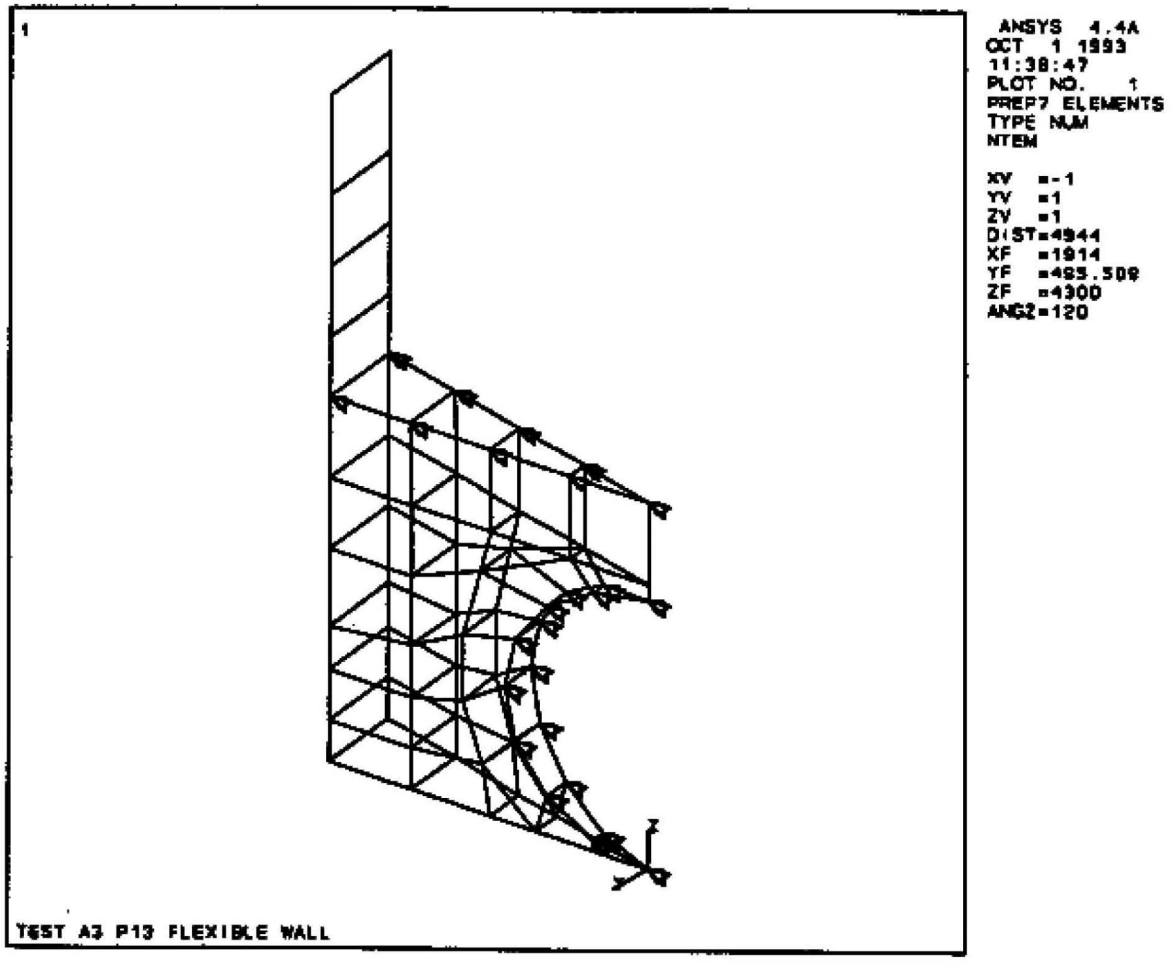

Figure 3.8.3-9
Test Tank Finite Element Model

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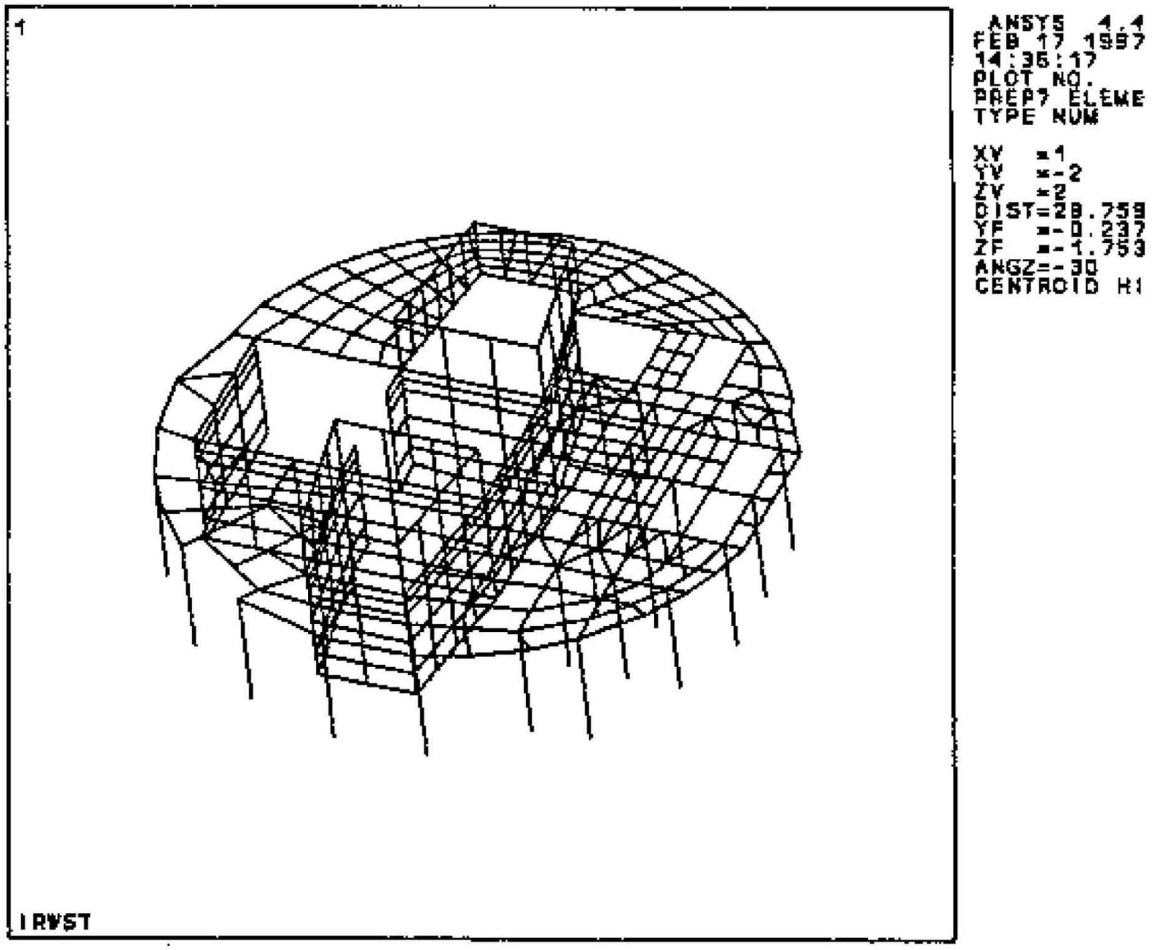

Figure 3.8.3-10 (Sheet 1 of 2)
IRWST Fluid Structure Finite Element Model
CIS Structural Model

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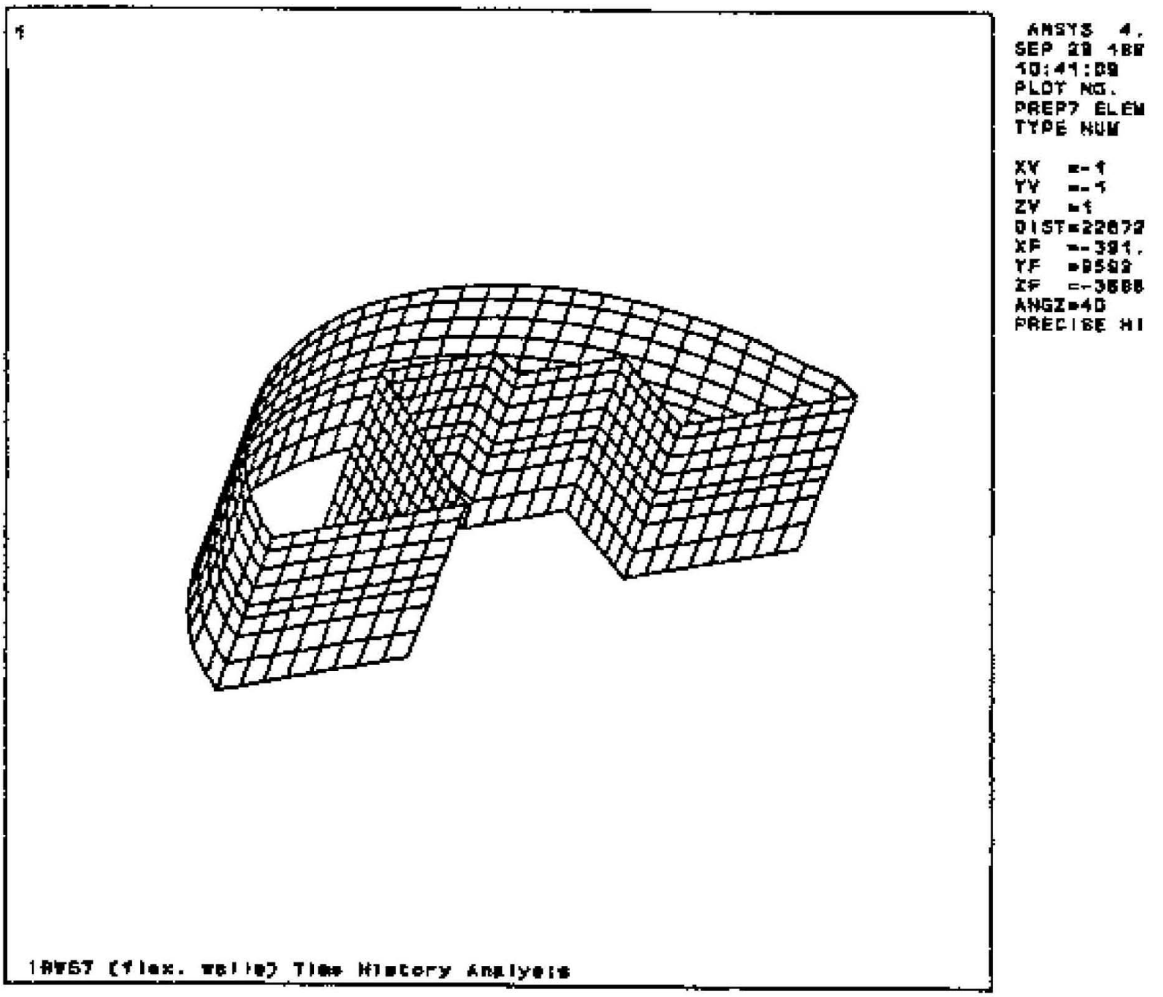

Figure 3.8.3-10 (Sheet 2 of 2)
IRWST Fluid Structure Finite Element Model
IRWST Structural Model

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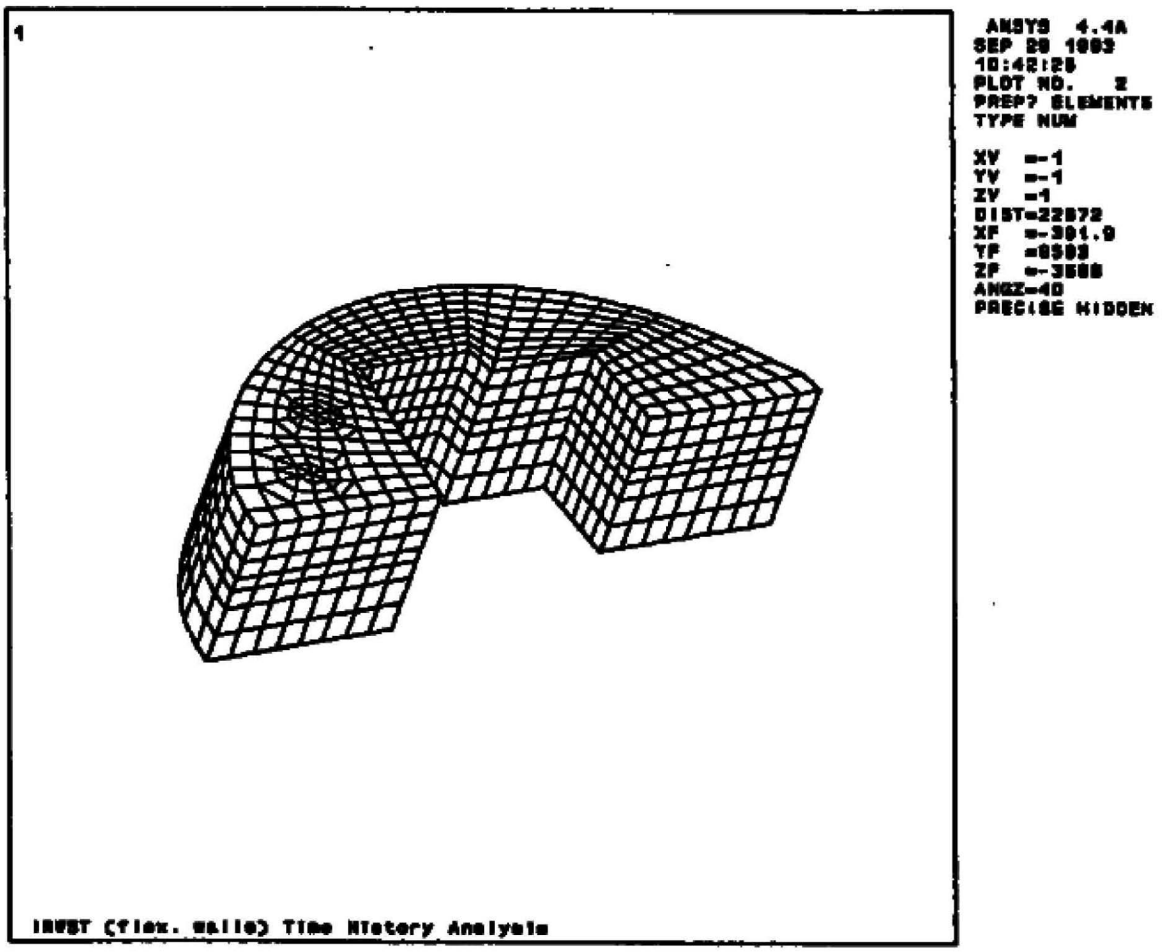

Figure 3.8.3-11
IRWST Fluid Structure Finite Element Model
Fluid Model

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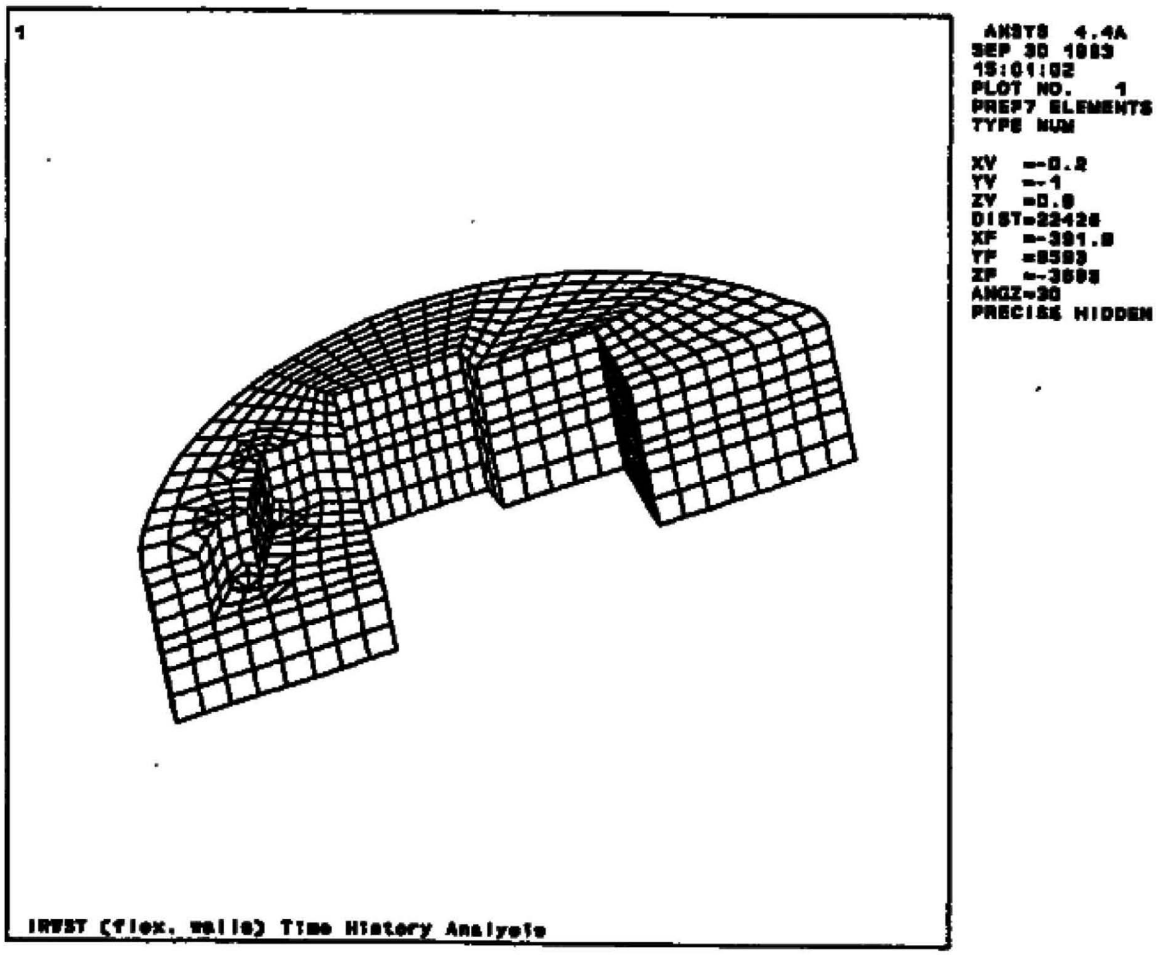

Figure 3.8.3-12
IRWST Fluid Structure Finite Element Model
Sparger Region Detail

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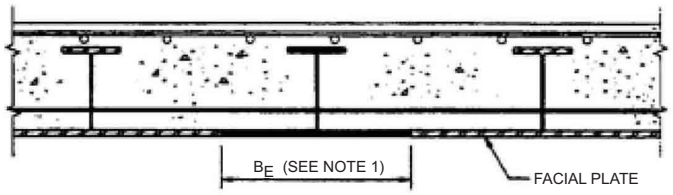
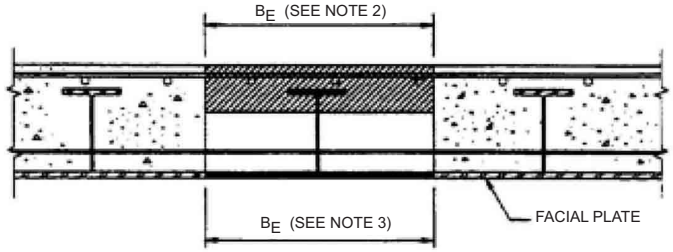


FACIAL PLATE IN COMPRESSION



FACIAL PLATE IN TENSION

EFFECTIVE SOLUTIONS FOR FLOOR MODULES

NOTES:

1. FOR FACIAL PLATE IN COMPRESSION, b_e, IS DETERMINED PER SECTION 3.8.3.5.4.2
2. EFFECTIVE WIDTH OF CONCRETE b_e, IS DETERMINED PER SECTION Q 1.11.1 OF AISC N690.
3. FOR FACIAL PLATE IN TENSION, b_e, IS TAKEN TO BE ONE HALF OF THE DISTANCE TO THE ADJACENT BEAMS.

Figure 3.8.3-13
Effective Sections for Floor Modules

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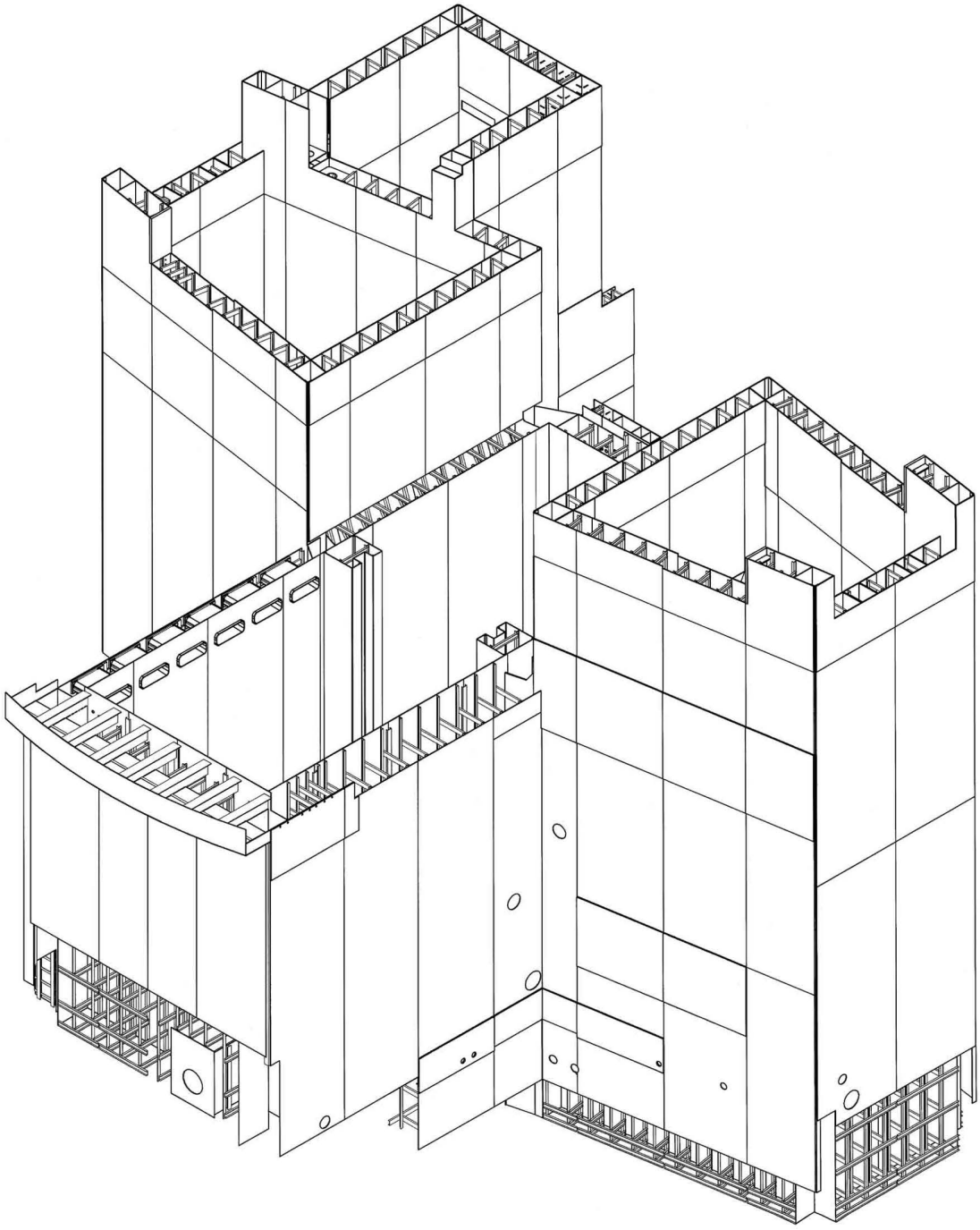

Figure 3.8.3-14 (Sheet 1 of 5)
[*CA-01 Module*]*

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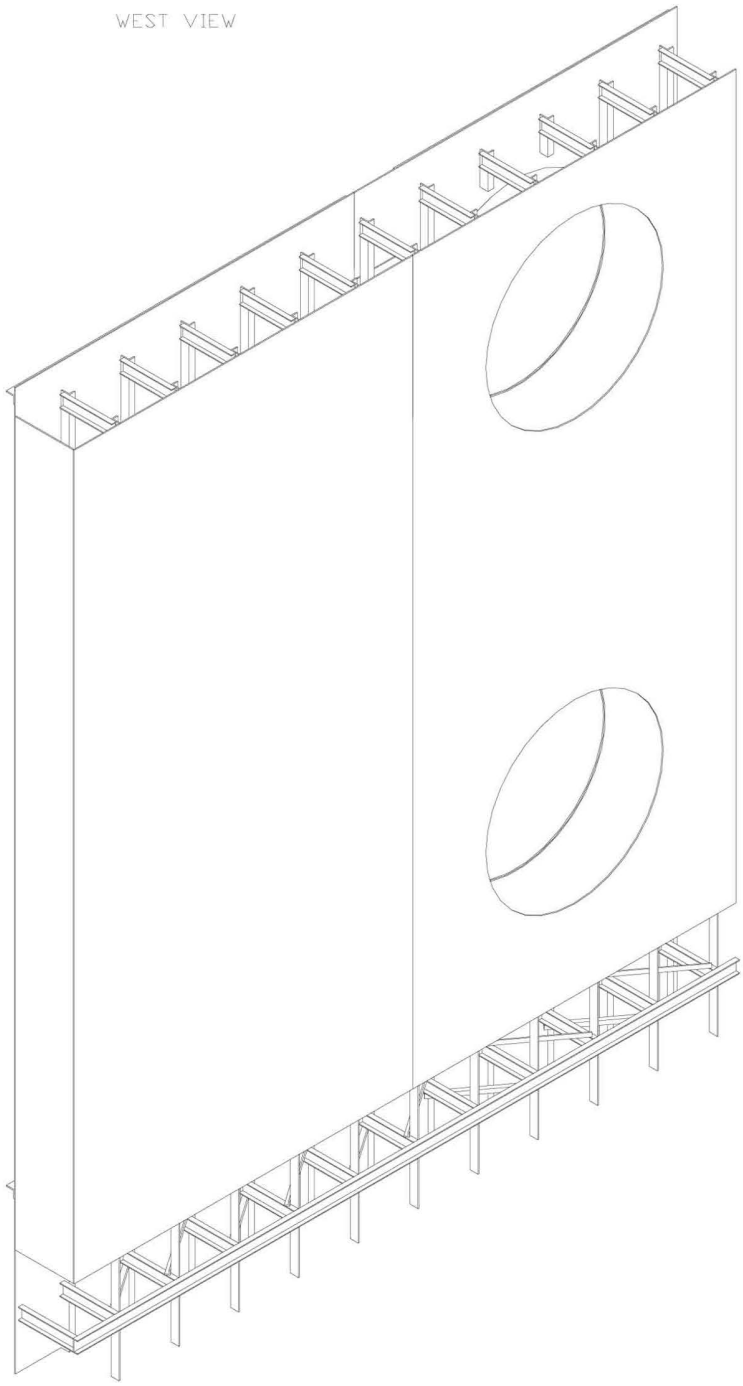

Figure 3.8.3-14 (Sheet 2 of 5)
[*CA-02 Module*]*

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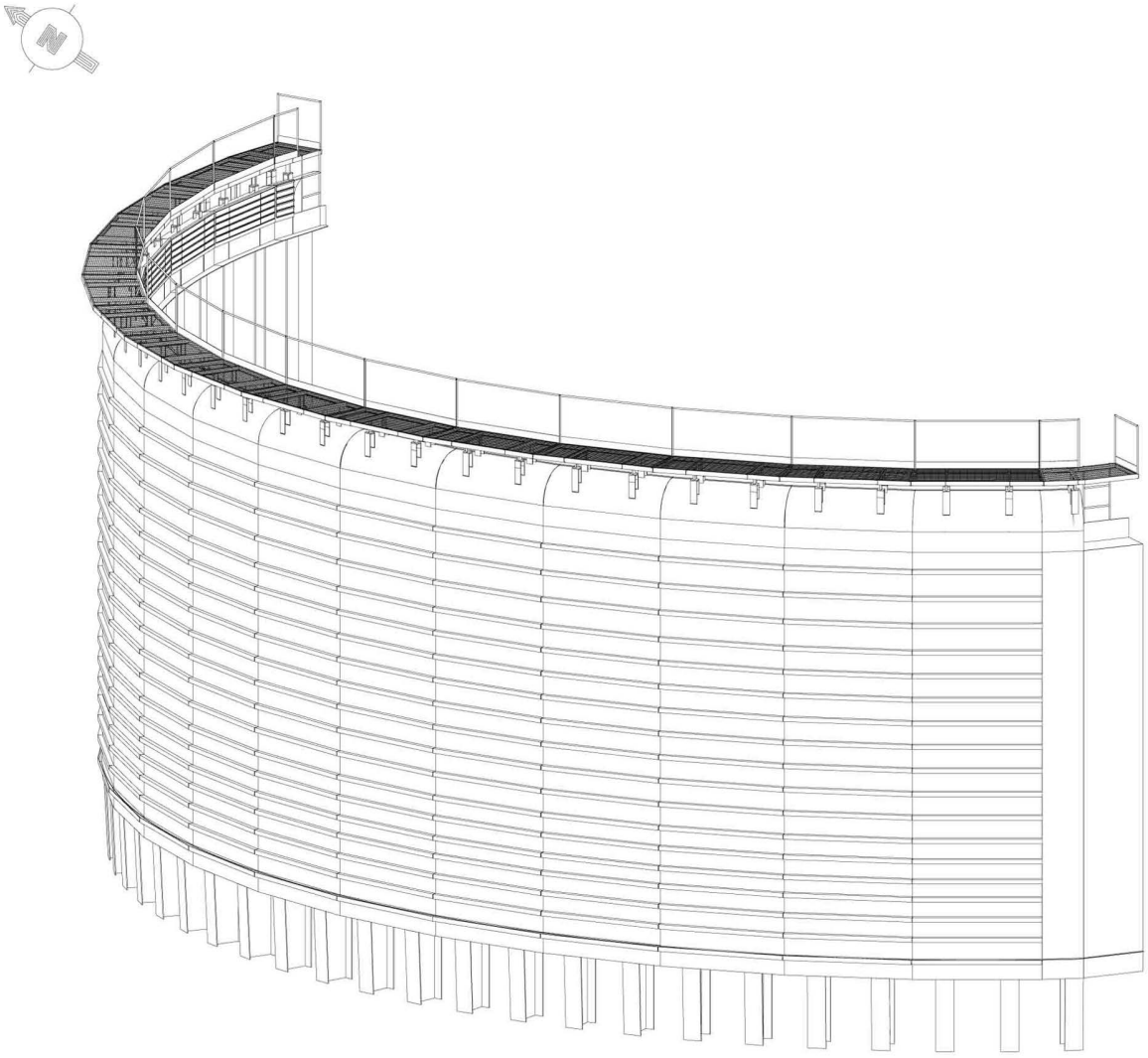

Figure 3.8.3-14 (Sheet 3 of 5)
[*CA-03 Module*]*

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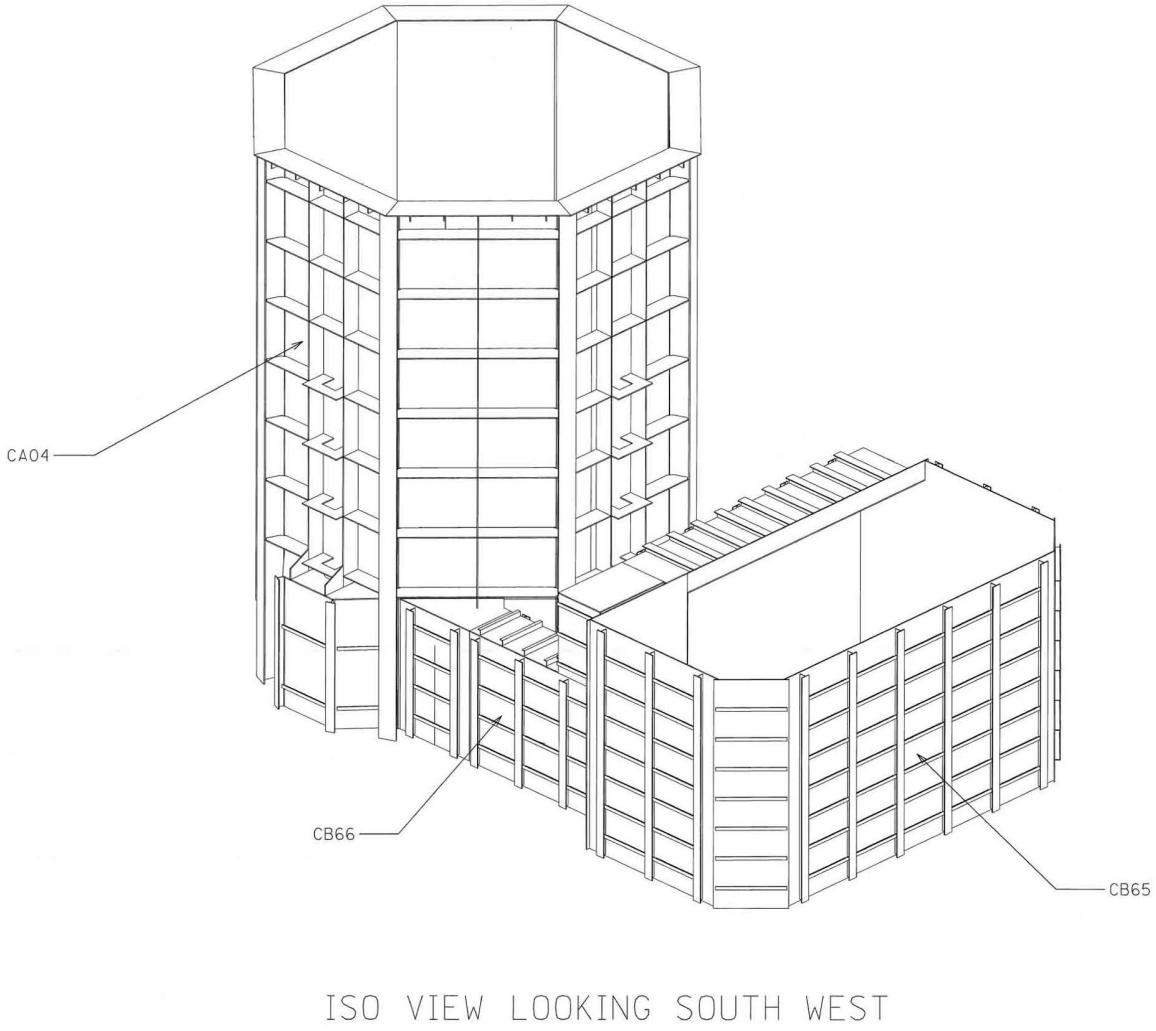

Figure 3.8.3-14 (Sheet 4 of 5)
[*CA-04 Structural Module*]*

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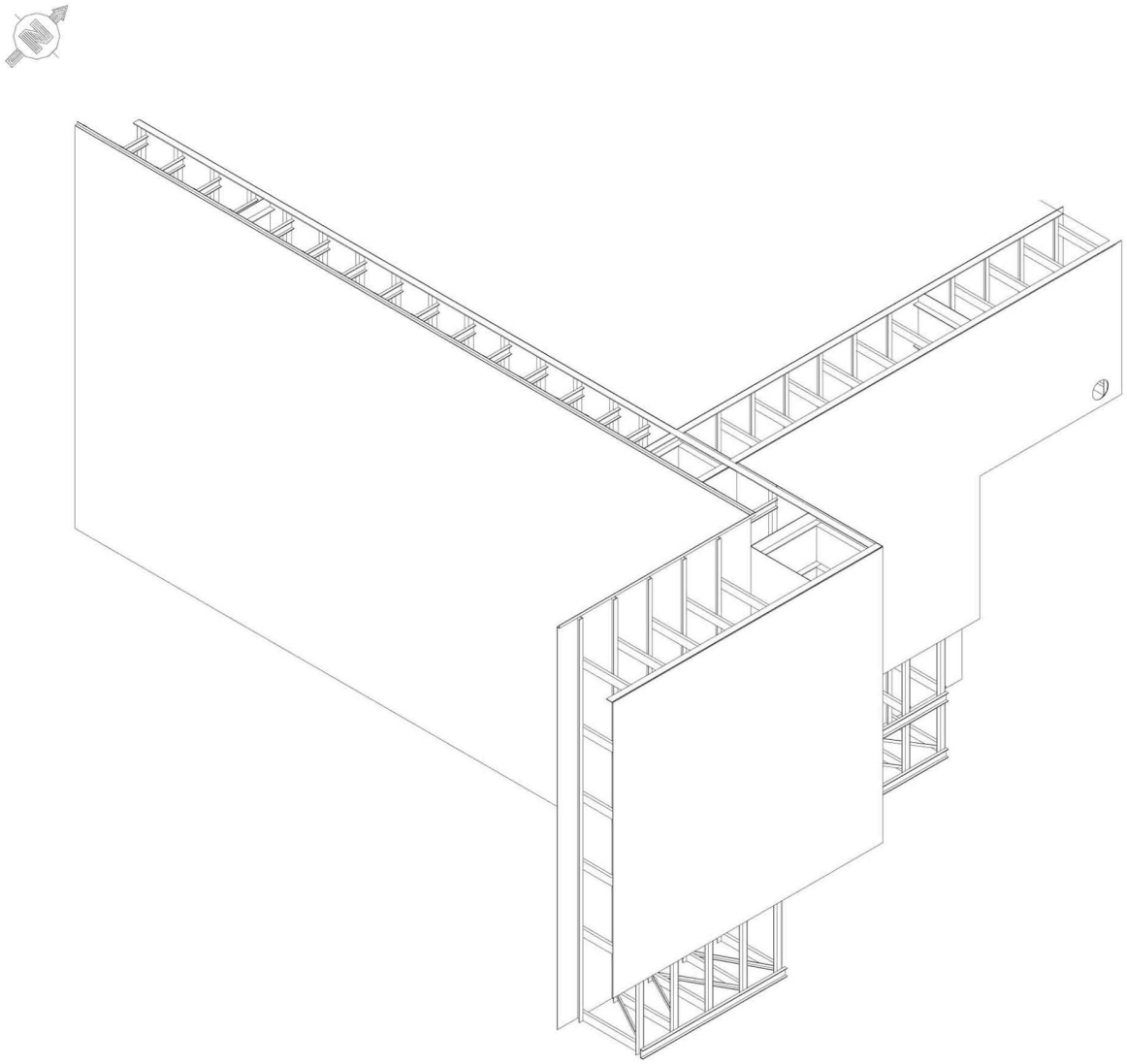

Figure 3.8.3-14 (Sheet 5 of 5)
[*CA-05 Module*]*

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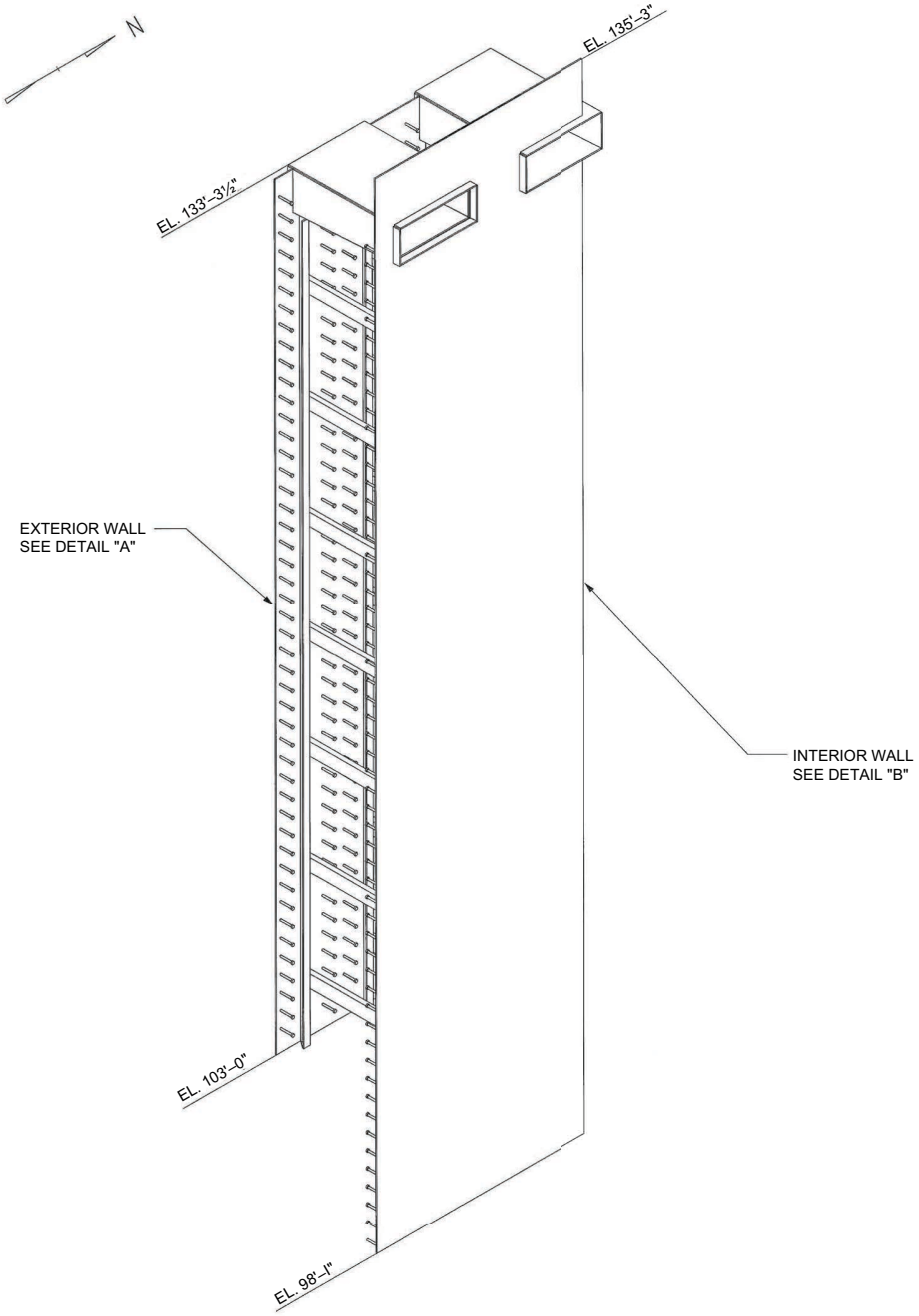

Note: See Figure 3.8.3-8 for fabrication detail.

Figure 3.8.3-15 (Sheet 1 of 2)
[*Typical Submodule*]*

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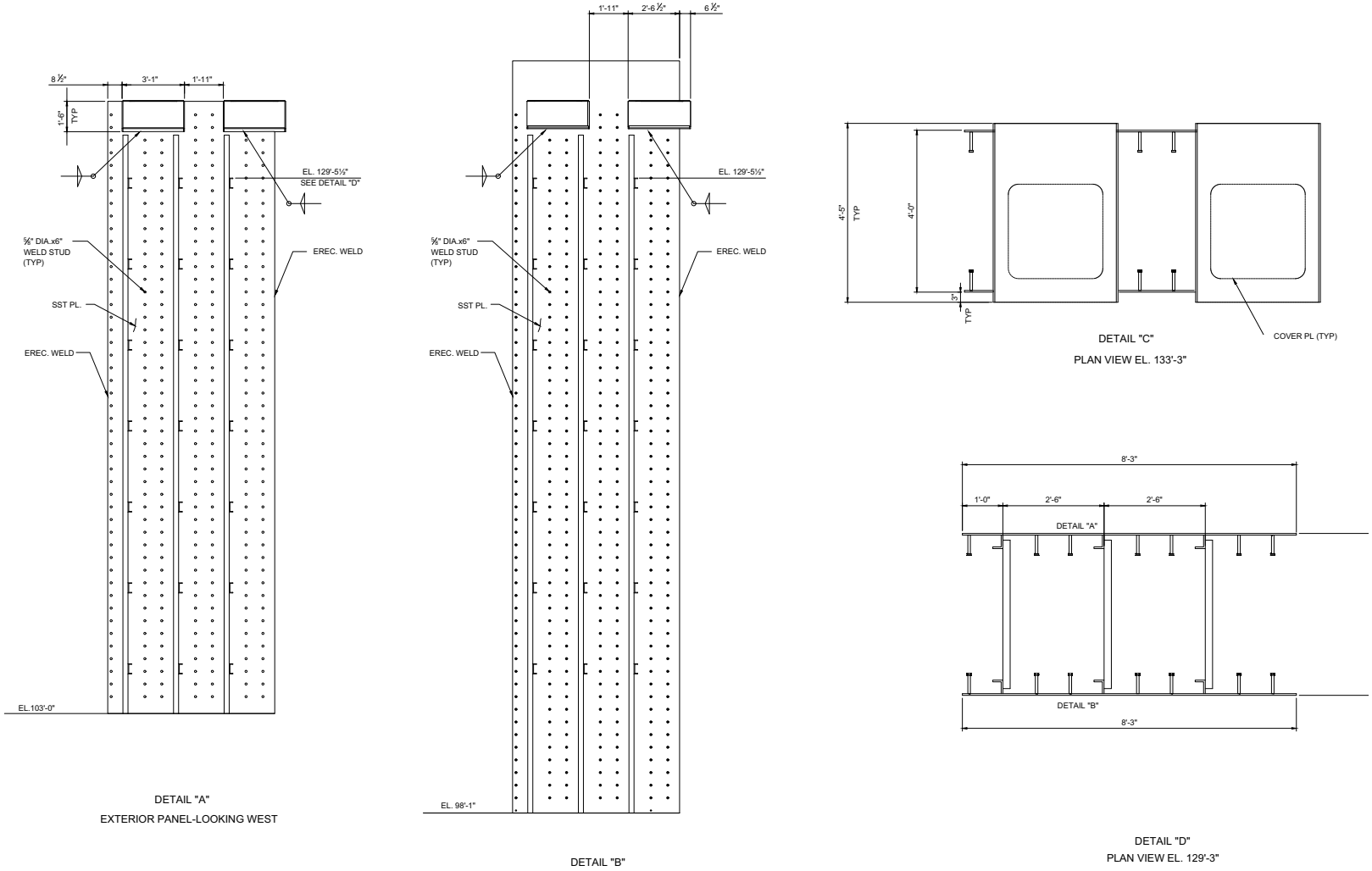

Figure 3.8.3-15 (Sheet 2 of 2)
[*Typical Submodule*]*

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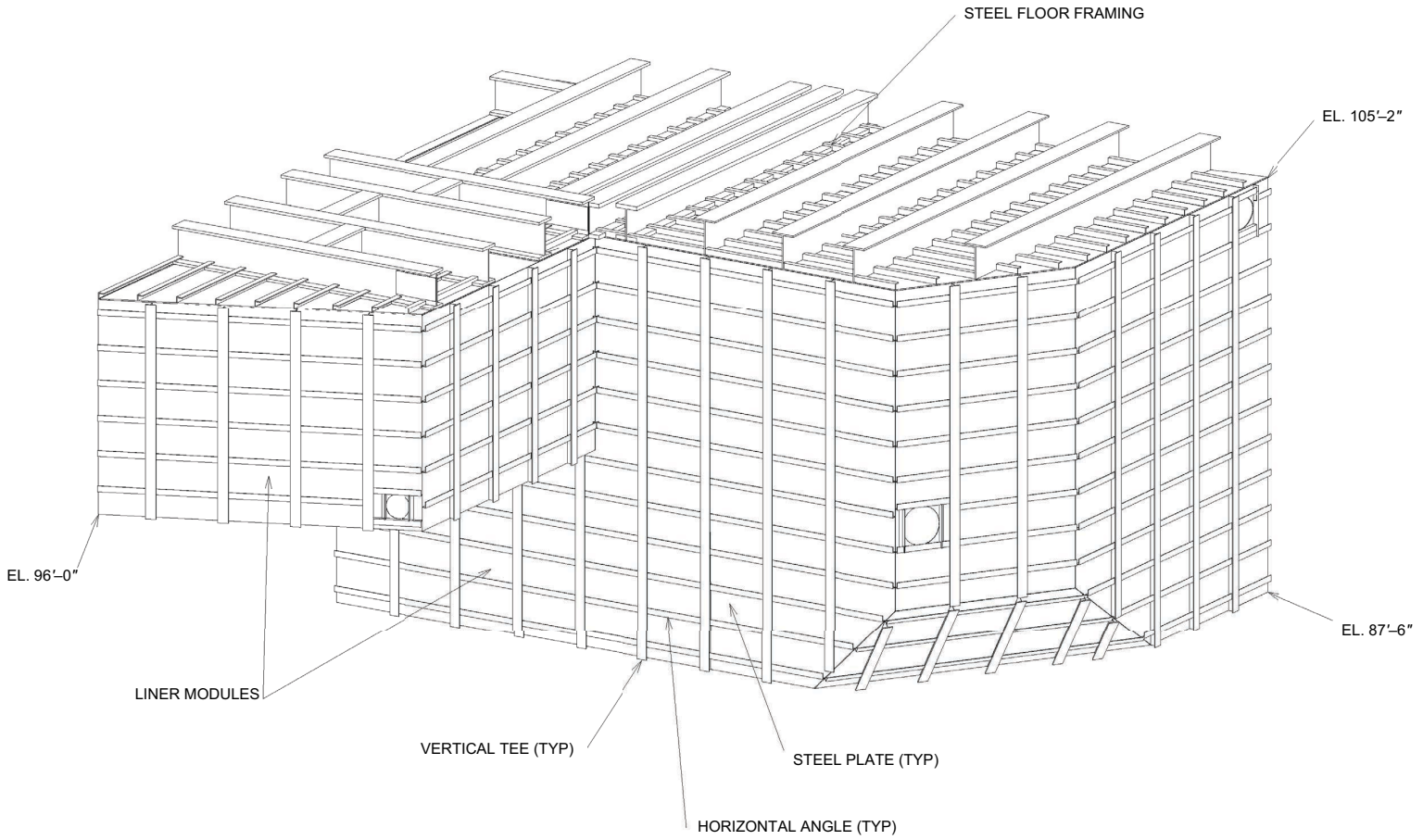

Figure 3.8.3-16
Liner Modules

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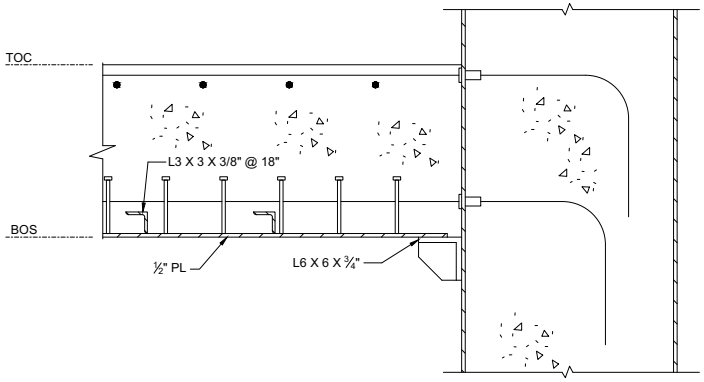
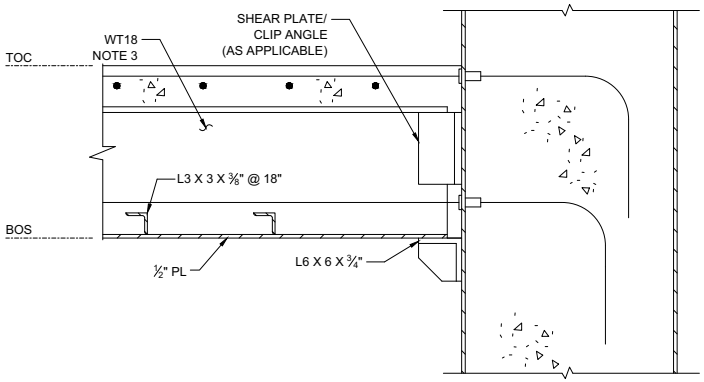


SECTION BETWEEN BEAMS IN FLOOR
NOTES: 1, 2, 3, 4, 5, 6, 8 & 9



SECTION AT BEAMS IN FLOOR
NOTES: 1, 2, 3, 4, 5, 6, 7 & 8

SEE SUBSECTION 3.8.3.5.8.1 FOR INFORMATION ON TIER 2* DESIGNATION.

NOTES:
1. DETAILS SHOWN ARE REPRESENTATIVE OF FLOOR MODULES AT EL 107'-2" AND 135'-3" AND THE CONNECTIONS OF FLOOR MODULES TO WALL MODULES INSIDE CONTAINMENT. REFER TO SUBSECTION 3.8.3.1.3 AND OTHER NOTES FOR ADDITIONAL INFORMATION ABOUT DESIGN DETAILS AND VARIATIONS FOR FLOOR MODULE TO WALL MODULE CONNECTIONS.
2. REINFORCEMENT USE, SIZE, AND SPACING IN THE FLOOR MODULE CONCRETE SATISFY THE REQUIREMENTS IN ACI 349. WHERE USED, THE REINFORCEMENT SIZE RANGE IS #7 TO #11. THE DESIGN OF THE STANDARD HOOKS IN THE WALL MODULES AND THE COUPLERS CONNECTING THE FLOOR REINFORCEMENT TO THE HOOKS IN THE WALL MODULES SATISFIES THE REQUIREMENTS OF ACI 349.
3. DESIGN OF THE PLATES, BEAMS, AND STIFFENERS IN THE FLOOR, INCLUDING PLATE SIZE AND SPACING, AND TYPE, SIZE, AND SPACING OF STRUCTURAL SHAPES VARIES AND SATISFIES THE REQUIREMENTS OF AISC N690.
4. THE REINFORCEMENT AND FLOOR DESIGN ELEMENTS SHOWN ARE FOR LOCATIONS AWAY FROM OPENINGS, PENETRATIONS, AND OTHER OBSTRUCTIONS.
5. IN SOME LOCATIONS, REINFORCEMENT AT THE CONNECTION IS NOT REQUIRED BECAUSE THE LOADS ARE TRANSFERRED THROUGH THE STEEL SECTION DIRECTLY TO THE BACKUP STRUCTURES OR THE SEAT ANGLE. THE DESIGN OF THE FLOORS SATISFIES THE APPLICABLE REQUIREMENTS IN ACI 349 AND AISC N690.
6. THE DETAIL DESIGN, LOCATION, AND ATTACHMENT OF THE FLOOR AND BEAM SUPPORTS ARE DESIGNED TO THE REQUIREMENTS OF AISC N690. SUPPORT CONFIGURATIONS, INCLUDING THE USE OF PLATES, STRUCTURAL SHAPES, AND STIFFENERS, ARE BASED ON LOADING AND LOCAL GEOMETRY CONSIDERATIONS.
7. THE DESIGNS OF THE CONNECTIONS AND BACKUP STRUCTURES WITHIN THE MODULES WALL, INCLUDING PLATE SIZE, AND SPACING, AND TYPE, SIZE, AND SPACING OF STRUCTURAL SHAPES VARY AND SATISFY THE REQUIREMENTS OF AISC N690.
8. THE THICKNESS OF THE ADJACENT WALL IS BASED ON THE WALL DESIGN REQUIREMENTS AND LOCATION.
9. REFER TO SUBSECTION 3.8.3.5.4 FOR INFORMATION ABOUT THE SHEAR STUDS IN THE FLOOR MODULE.

Figure 3.8.3-17 (Sheet 1 of 2)
[*Structural Modules – Design Details Standard Floor Connection*]*

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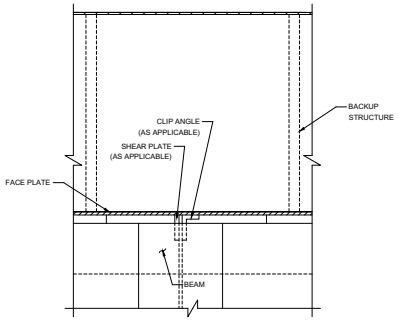
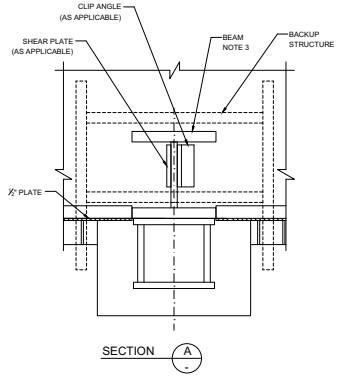
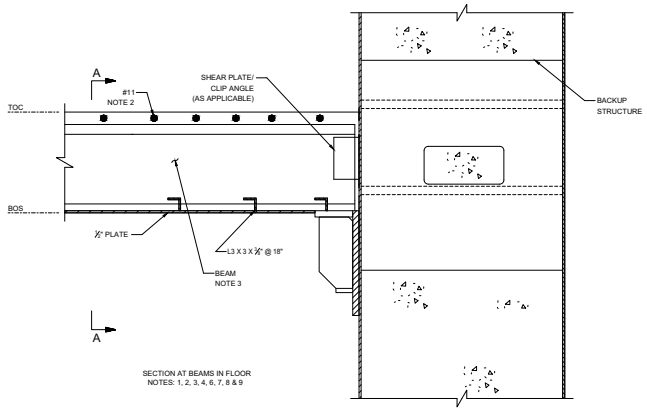


TOP VIEW AT BEAMS IN FLOOR
NOTE: 7

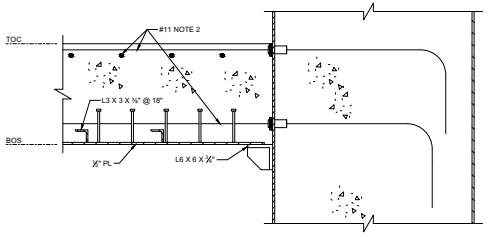


END VIEW AT BEAMS IN FLOOR
NOTES: 3, 6, 7 & 9

NOTES:
1. DETAILS SHOWN ARE REPRESENTATIVE OF FLOOR MODULES AT EL 107'-2" AND 135'-3" AND THE CONNECTIONS OF FLOOR MODULES TO WALL MODULES INSIDE CONTAINMENT. REFER TO SUBSECTION 3.8.3.1.3 AND OTHER NOTES FOR ADDITIONAL INFORMATION ABOUT DESIGN DETAILS AND VARIATIONS FOR FLOOR MODULE TO WALL MODULE CONNECTIONS.
2. REINFORCEMENT USE, SIZE, AND SPACING IN THE FLOOR MODULE CONCRETE SATISFY THE REQUIREMENTS IN ACI 349. WHERE USED, THE REINFORCEMENT SIZE RANGE IS #7 TO #11. THE DESIGN OF THE STANDARD HOOKS IN THE WALL MODULES AND THE COUPLERS CONNECTING THE FLOOR REINFORCEMENT TO THE HOOKS IN THE WALL MODULES SATISFIES THE REQUIREMENTS OF ACI 349.
3. DESIGN OF THE PLATES, BEAMS, AND STIFFENERS IN THE FLOOR, INCLUDING PLATE SIZE AND SPACING, AND TYPE, SIZE, AND SPACING OF STRUCTURAL SHAPES VARIES AND SATISFIES THE REQUIREMENTS OF AISC N690.
4. THE REINFORCEMENT AND FLOOR DESIGN ELEMENTS SHOWN ARE FOR LOCATIONS AWAY FROM OPENINGS, PENETRATIONS, AND OTHER OBSTRUCTIONS.
5. IN SOME LOCATIONS, REINFORCEMENT AT THE CONNECTION IS NOT REQUIRED BECAUSE THE LOADS ARE TRANSFERRED THROUGH THE STEEL SECTION IRECTLY TO THE BACKUP STRUCTURES OR THE SEAT ANGLE. THE DESIGN OF THE FLOORS SATISFIES THE APPLICABLE REQUIREMENTS IN ACI 349 AND AISC N690.
6. THE DETAIL DESIGN, LOCATION, AND ATTACHMENT OF THE FLOOR AND BEAM SUPPORTS ARE DESIGNED TO THE REQUIREMENTS OF AISC N690. SUPPORT CONFIGURATIONS, INCLUDING THE USE OF PLATES, STRUCTURAL SHAPES, AND STIFFENERS, ARE BASED ON LOADING AND LOCAL GEOMETRY CONSIDERATIONS.
7. THE DESIGNS OF THE CONNECTIONS AND BACKUP STRUCTURES WITHIN THE MODULES WALL, INCLUDING PLATE SIZE AND SPACING, AND TYPE, SIZE, AND SPACING OF STRUCTURAL SHAPES VARY AND SATISFY THE REQUIREMENTS OF AISC N690.
8. THE THICKNESS OF THE ADJACENT WALL IS BASED ON THE WALL DESIGN REQUIREMENTS AND LOCATION.
9. THE BEAM SEAT SHOWN IS NOT USED AT ALL BEAM LOCATIONS. THE SEAT ANGLE SHOWN ON SHEET 1 IS USED AT OTHER LOCATIONS.
10. REFER TO SUBSECTION 3.8.3.5.4 FOR INFORMATION ABOUT THE SHEAR STUDS IN THE FLOOR MODULE.



SECTION AT BEAMS IN FLOOR
NOTES: 1, 2, 3, 4, 6, 7, 8 & 9



SECTION BETWEEN BEAMS IN FLOOR
NOTES: 1, 2, 3, 4, 5, 6, 8 & 10

SEE SUBSECTION 3.8.3.5.8.1 FOR INFORMATION ON TIER 2* DESIGNATION.

Figure 3.8.3-17 (Sheet 2 of 2)
[*Structural Modules – Design Details Heavily Loaded Floor Connection*]*

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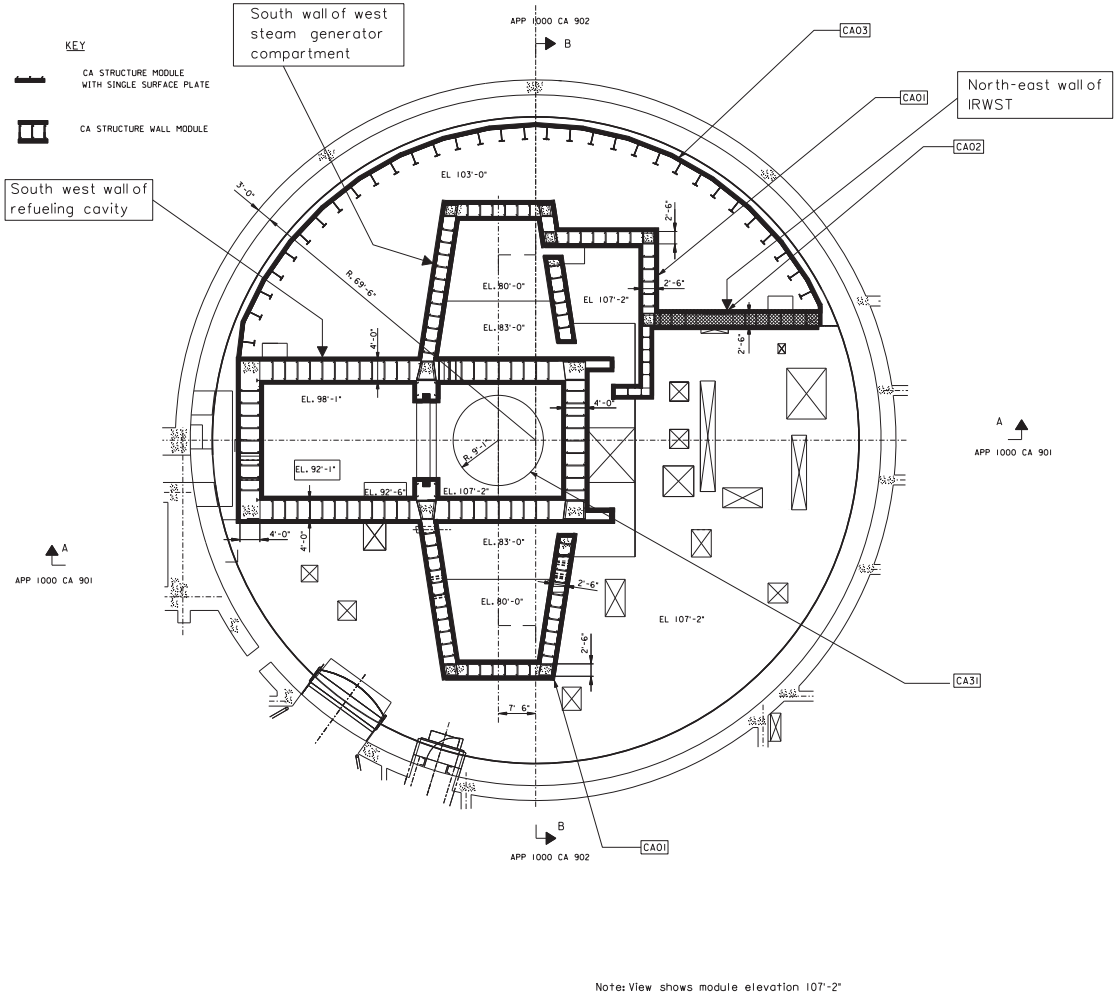

Figure 3.8.3-18
[*Location of Structural Wall Modules*]*

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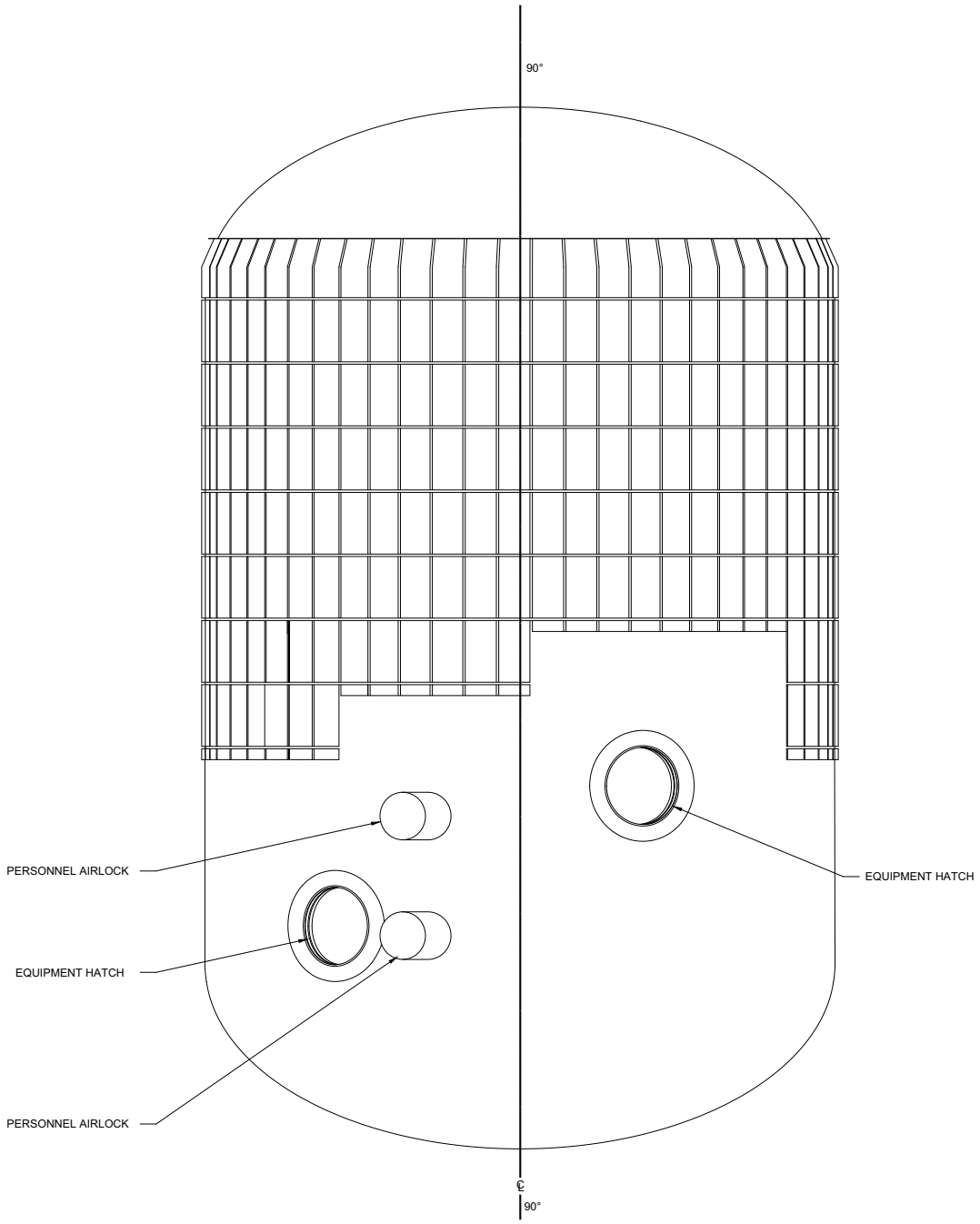

NOTE: COURSE LAYOUT, PLATE/INSERT PLATE GEOMETRY, WELD SEAMS FOR THE CONTAINMENT, HATCHES, AIRLOCKS, AND THE NUMBER AND CONFIGURATION OF THE CONTAINMENT AIR BAFFLE PANELS ARE PROVIDED FOR ILLUSTRATIVE PURPOSES ONLY.

Figure 3.8.4-1 (Sheet 1 of 4)
Containment Air Baffle
General Arrangement

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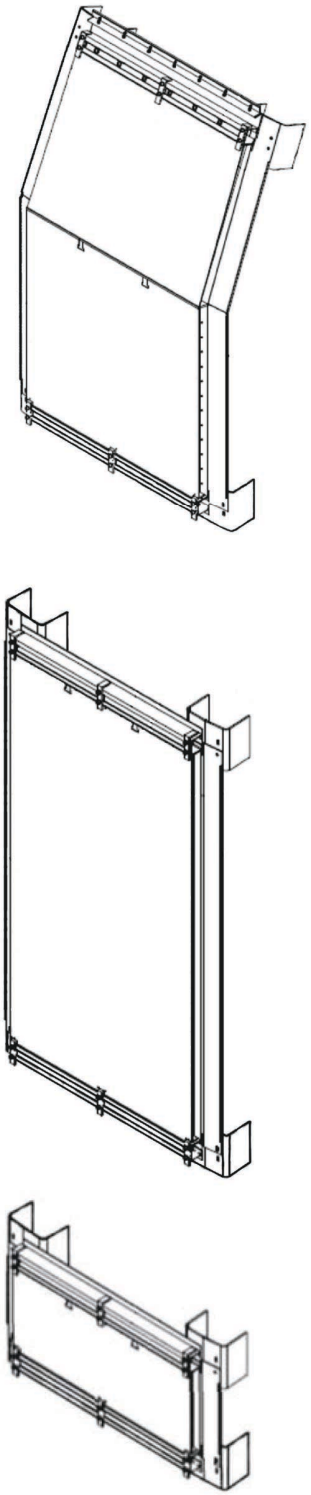

Figure 3.8.4-1 (Sheet 2 of 4)
Containment Air Baffle
Panel Types

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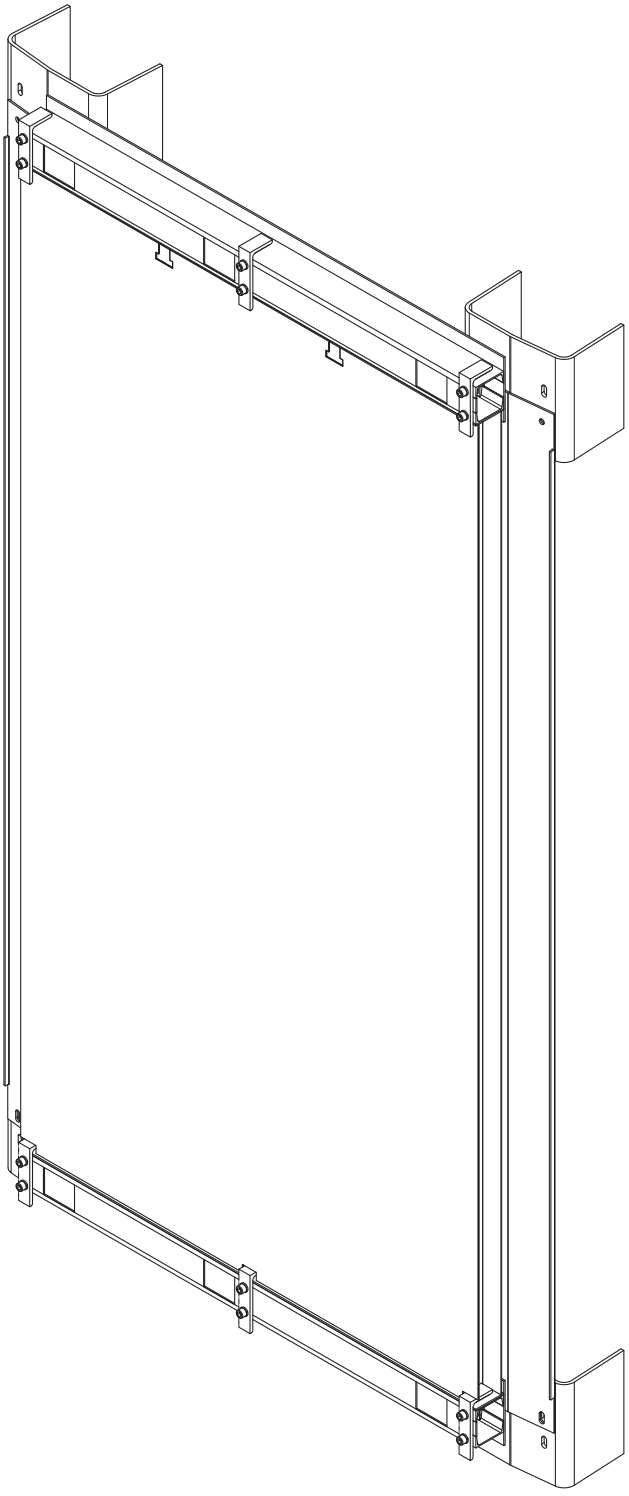

Figure 3.8.4-1 (Sheet 3 of 4)
Containment Air Baffle
Typical Panel on Cylinder

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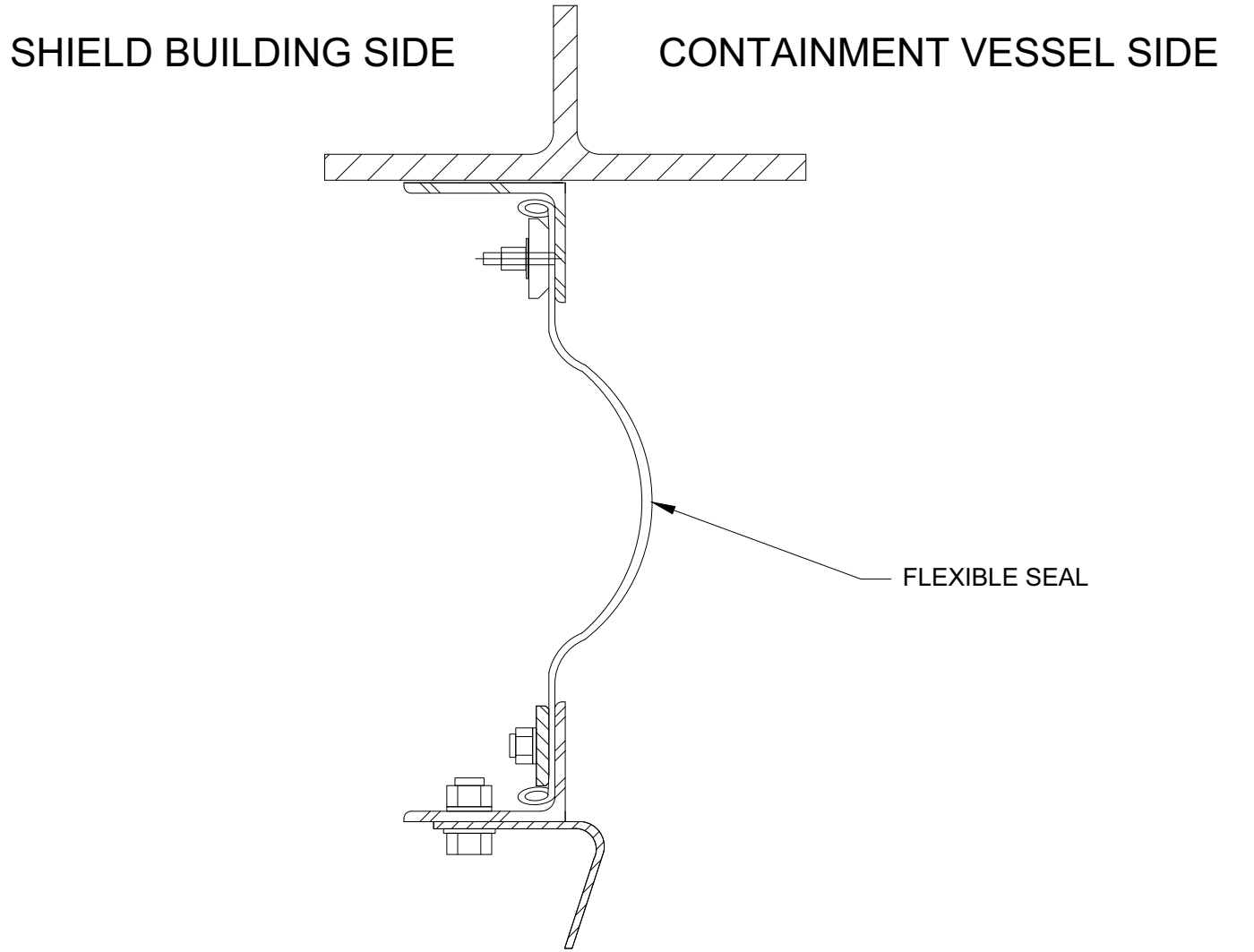

Figure 3.8.4-1 (Sheet 4 of 4)
Containment Air Baffle
Flexible Seal

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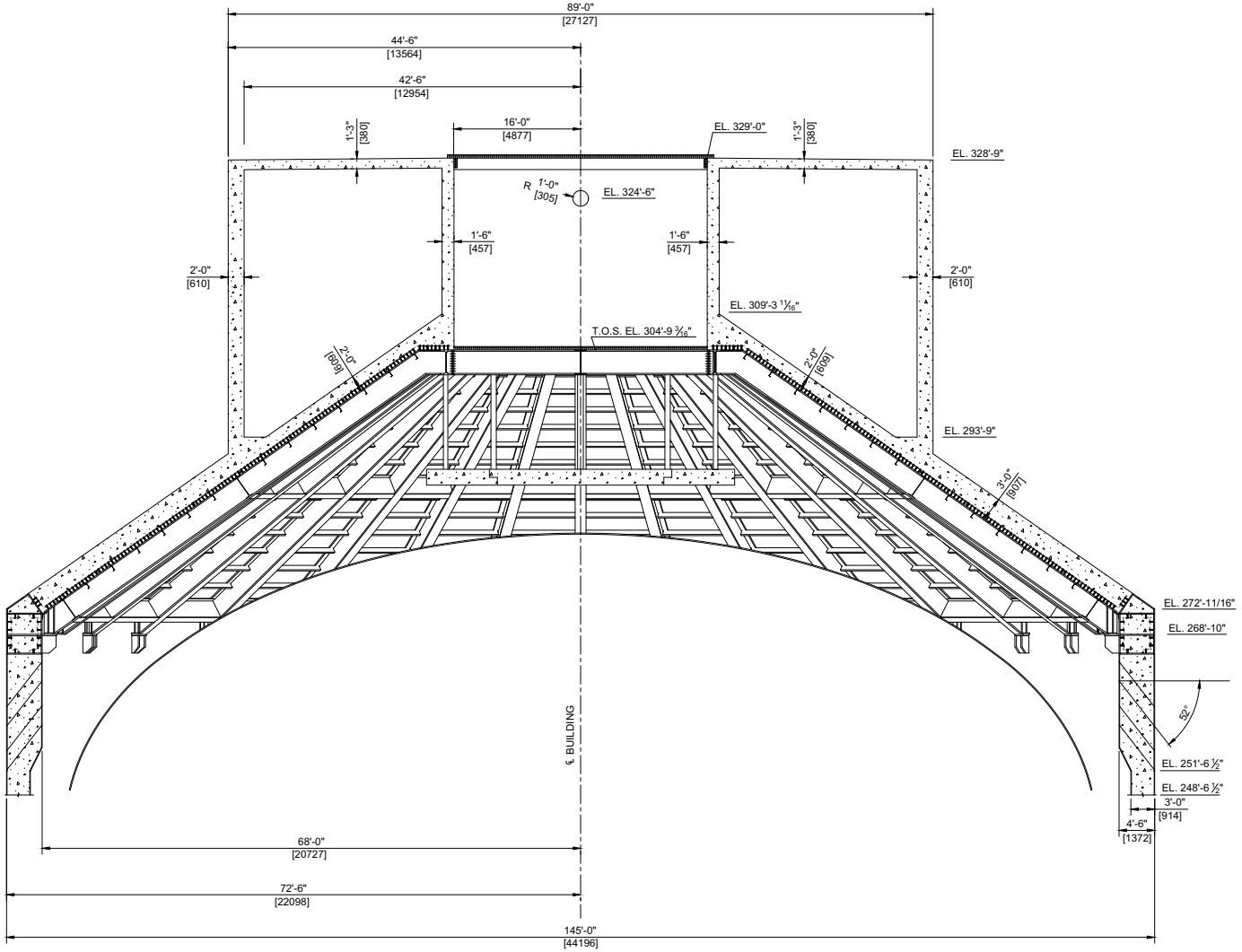

Figure 3.8.4-2
[*Passive Containment Cooling Tank*]*

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Figure 3.8.4-3 Not Used

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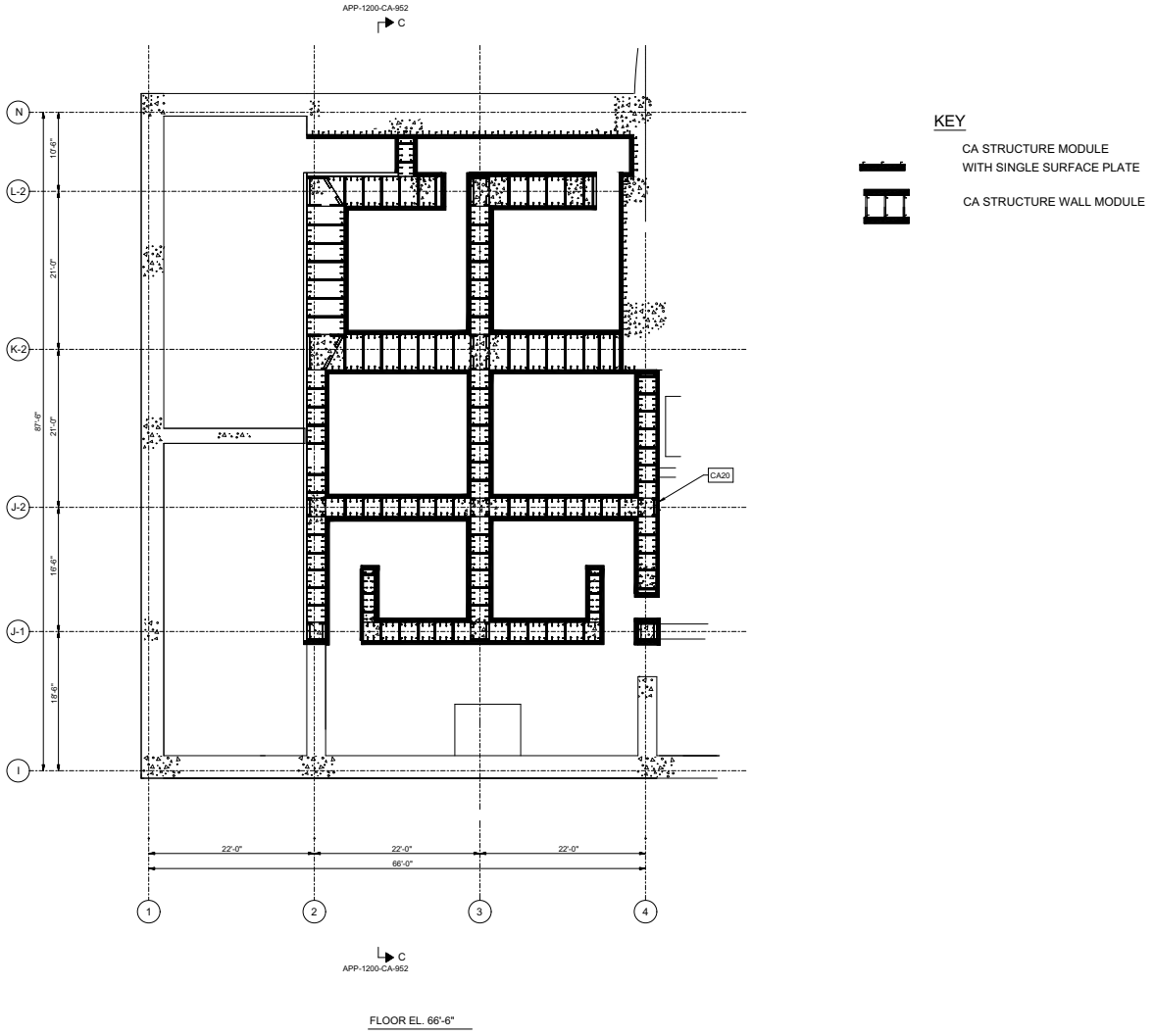

Figure 3.8.4-4 (Sheet 1 of 5)
[*Structural Modules in Auxiliary Building*]*

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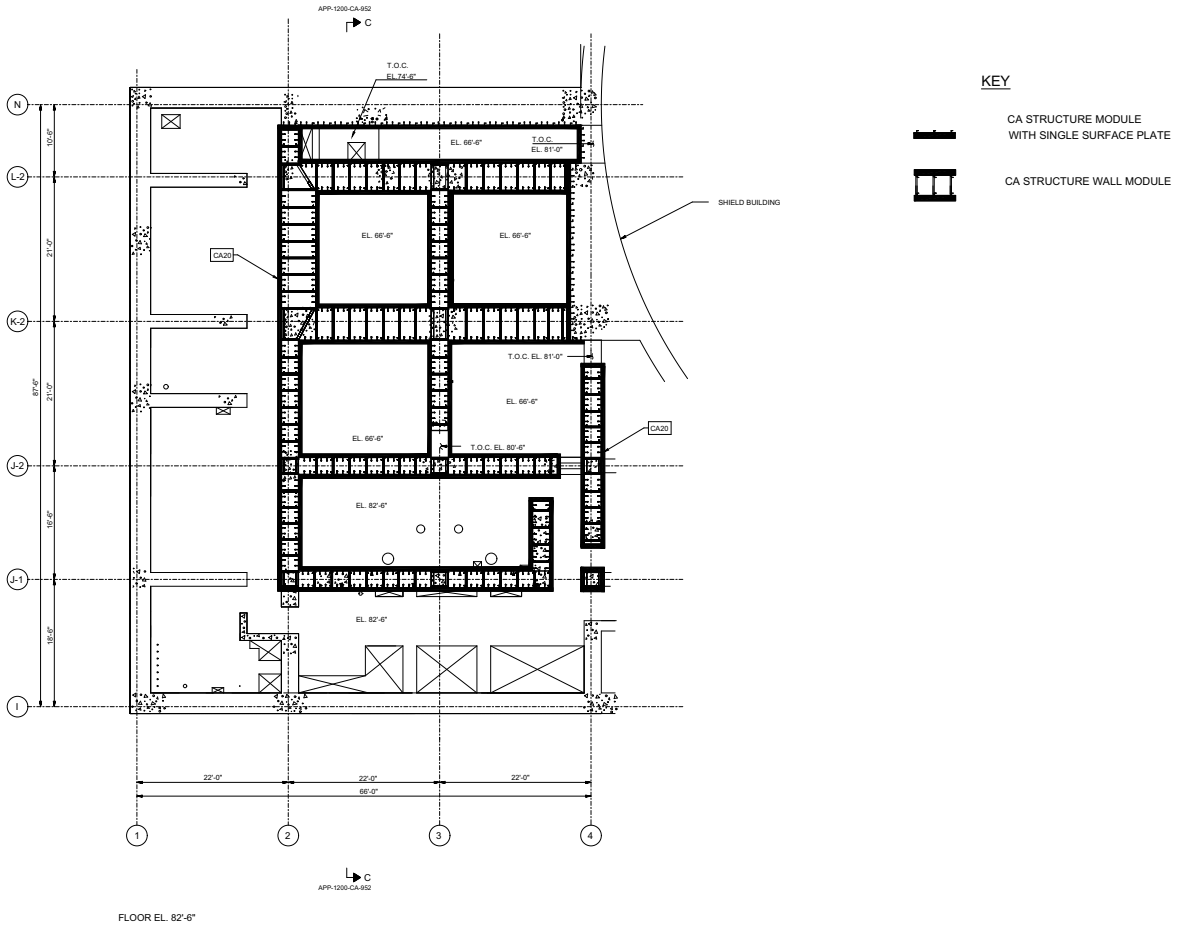

Figure 3.8.4-4 (Sheet 2 of 5)
[*Structural Modules in Auxiliary Building*]*

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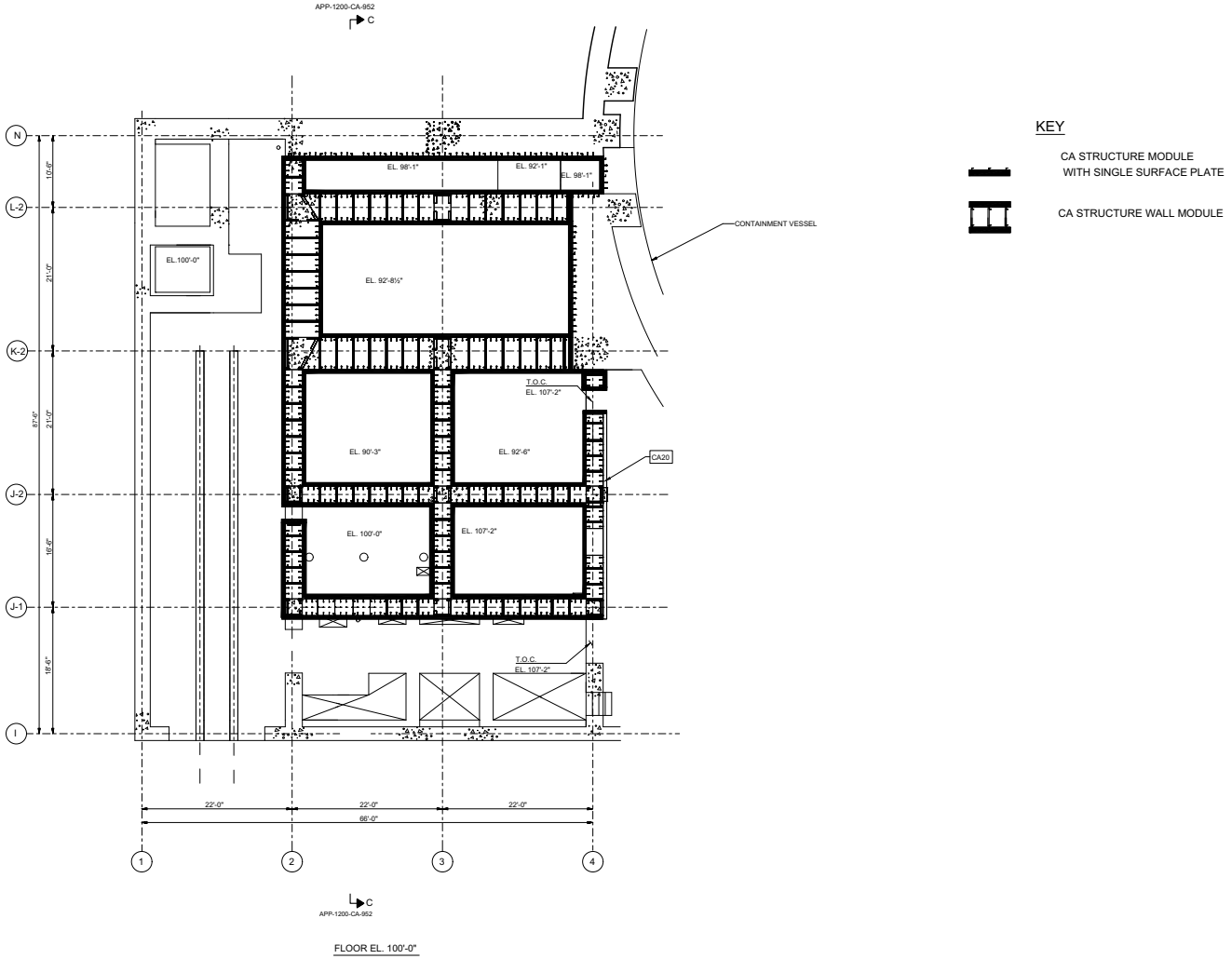

Figure 3.8.4-4 (Sheet 3 of 5)
[*Structural Modules in Auxiliary Building*]*

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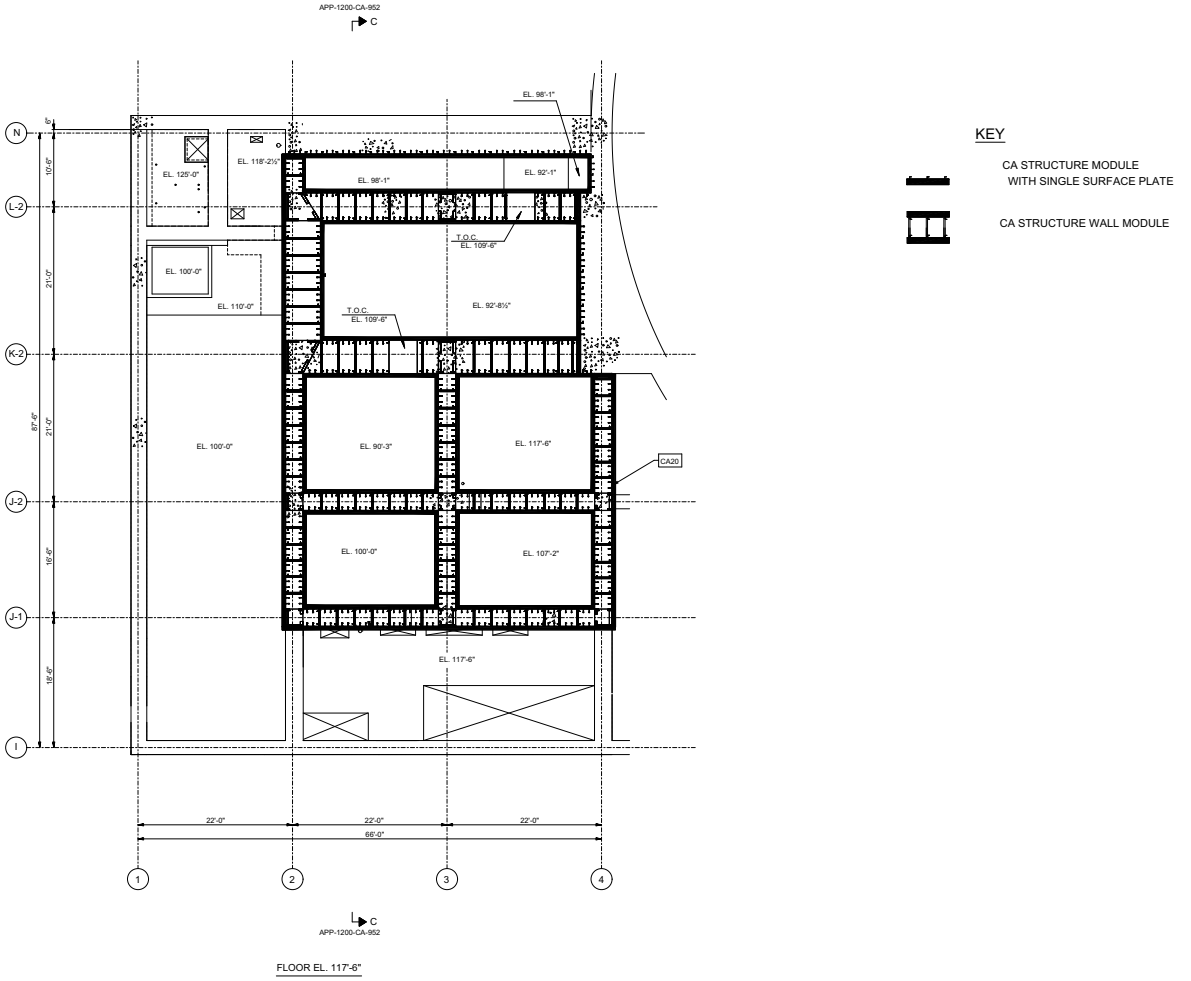

Figure 3.8.4-4 (Sheet 4 of 5)
[*Structural Modules in Auxiliary Building*]*

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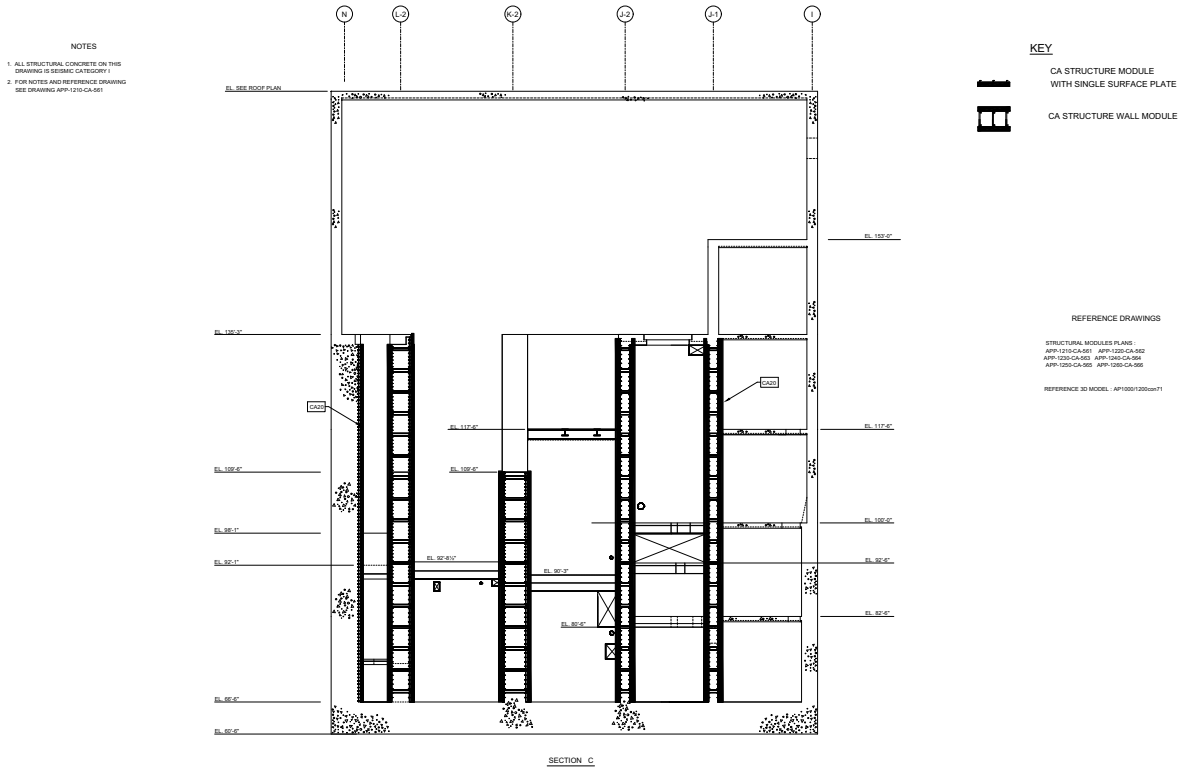

Figure 3.8.4-4 (Sheet 5 of 5)
[*Structural Modules in Auxiliary Building*]*

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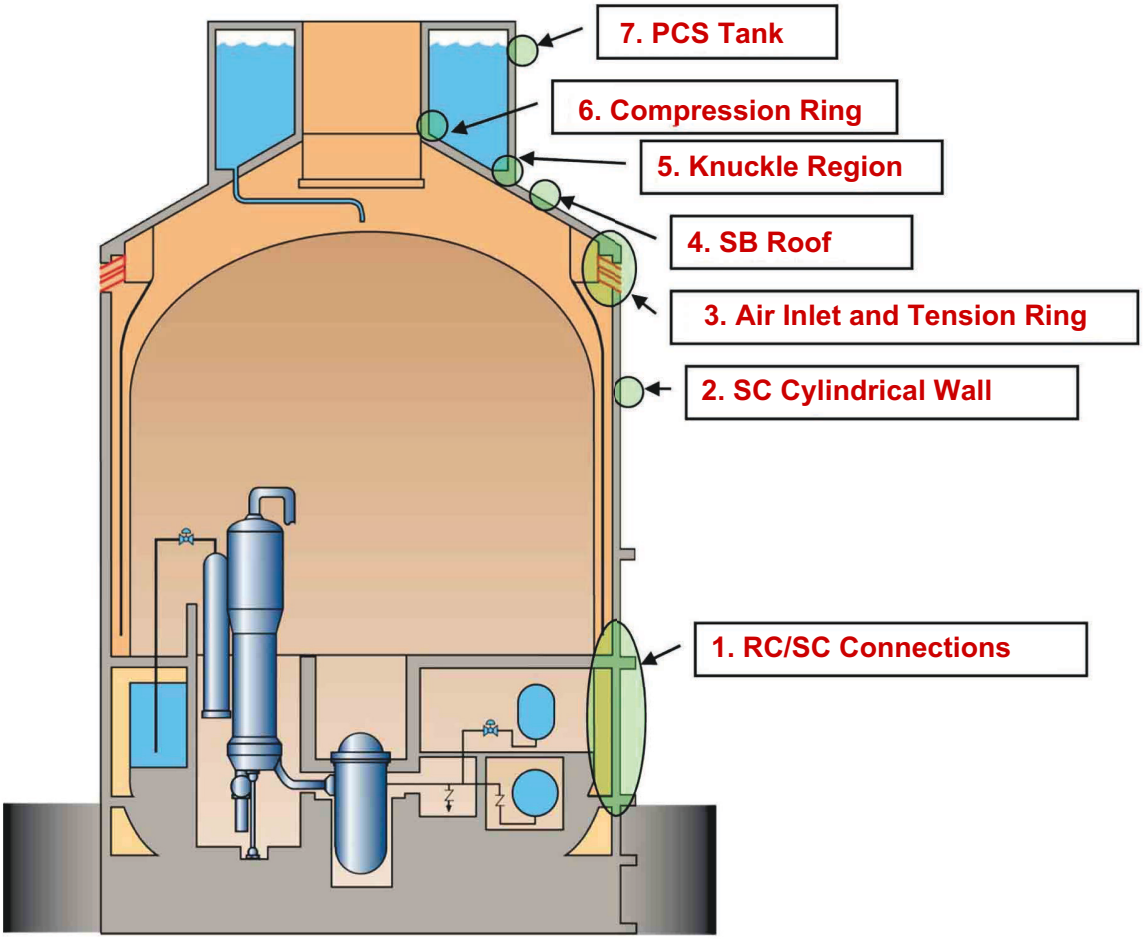

Figure 3.8.4-5
Shield Building Structure Key Areas

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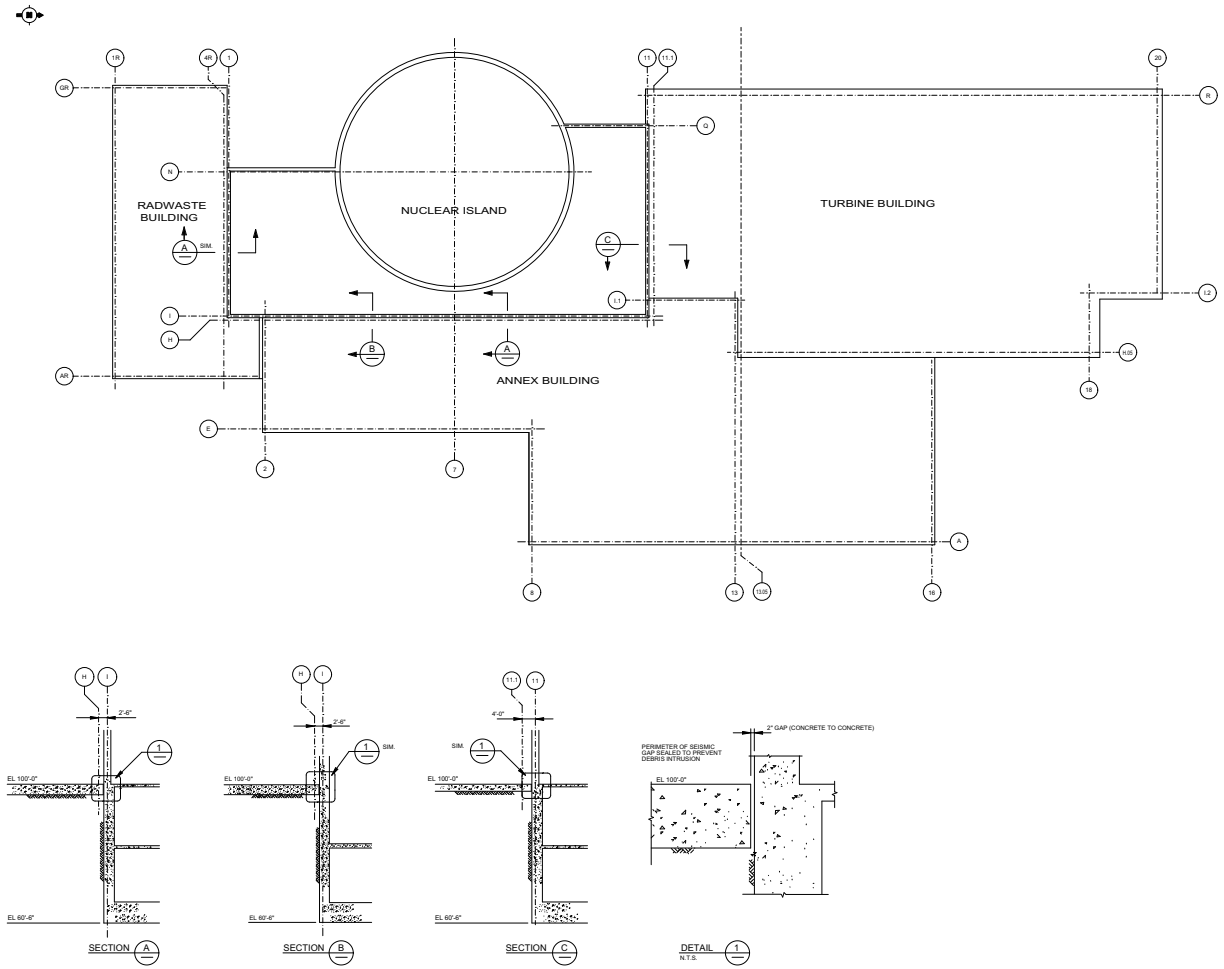

NUCLEAR ISLAND AND ADJACENT STRUCTURES

FOUNDATION PLANS AND SECTIONS

Figure 3.8.5-1
Foundation Plan

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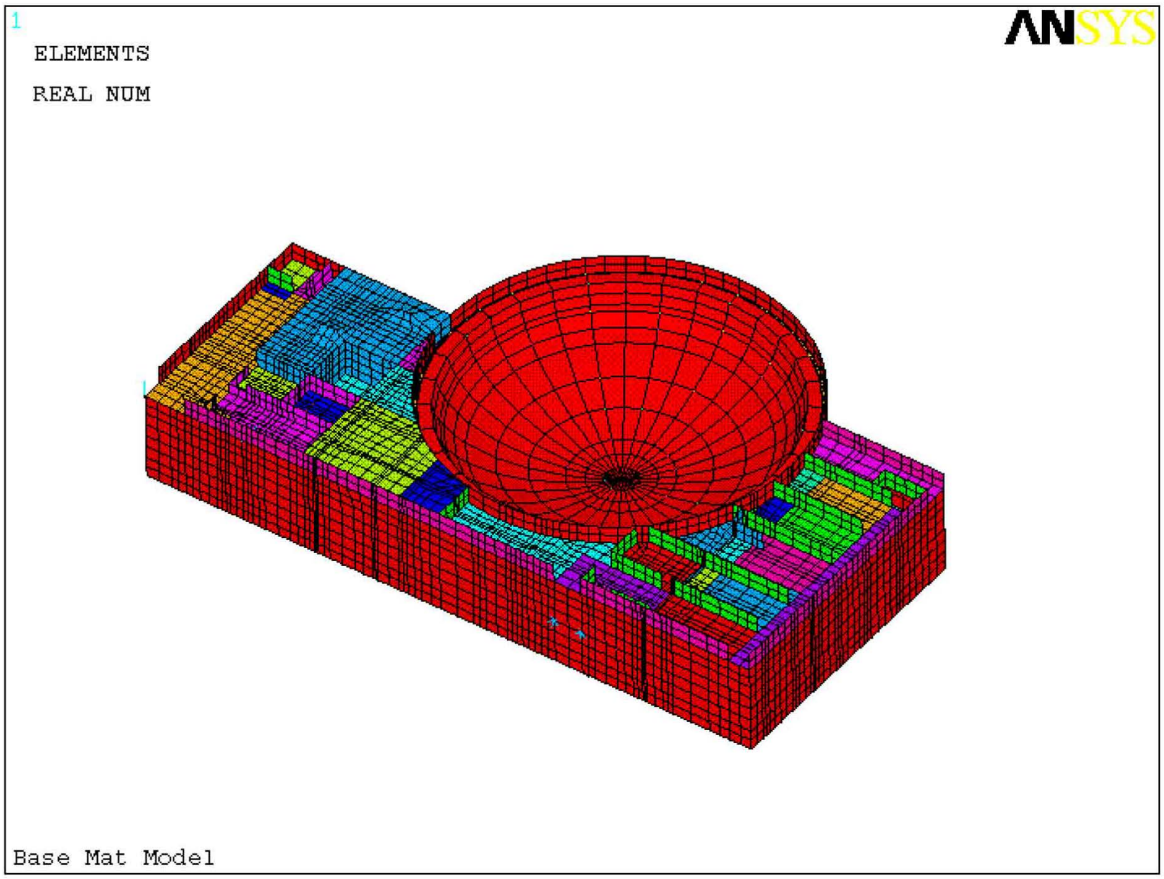

Figure 3.8.5-2
Isometric View of Finite Element Model

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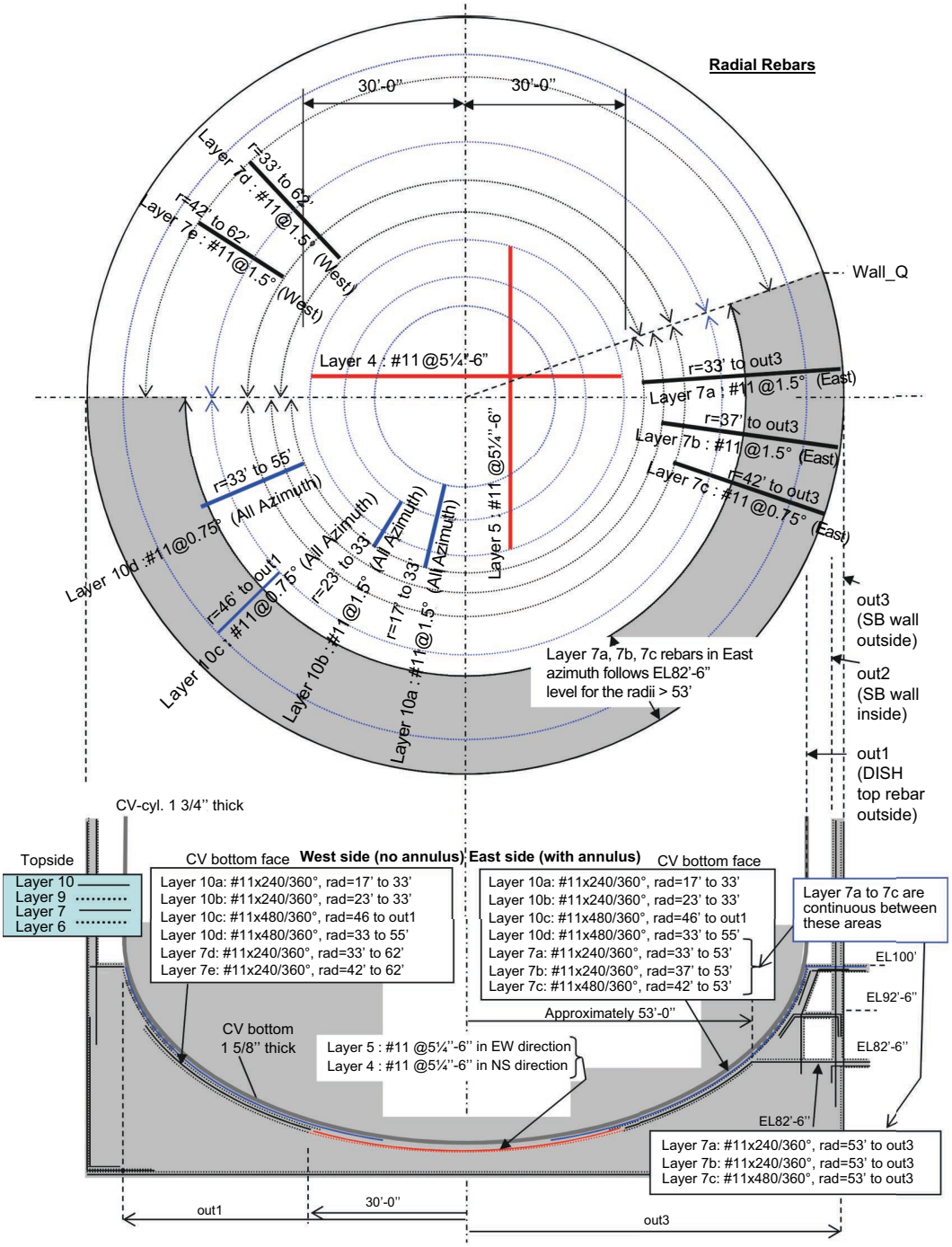

Figure 3.8.5-3 (Sheet 1 of 7)
Radial Reinforcement, Top Side of DISH

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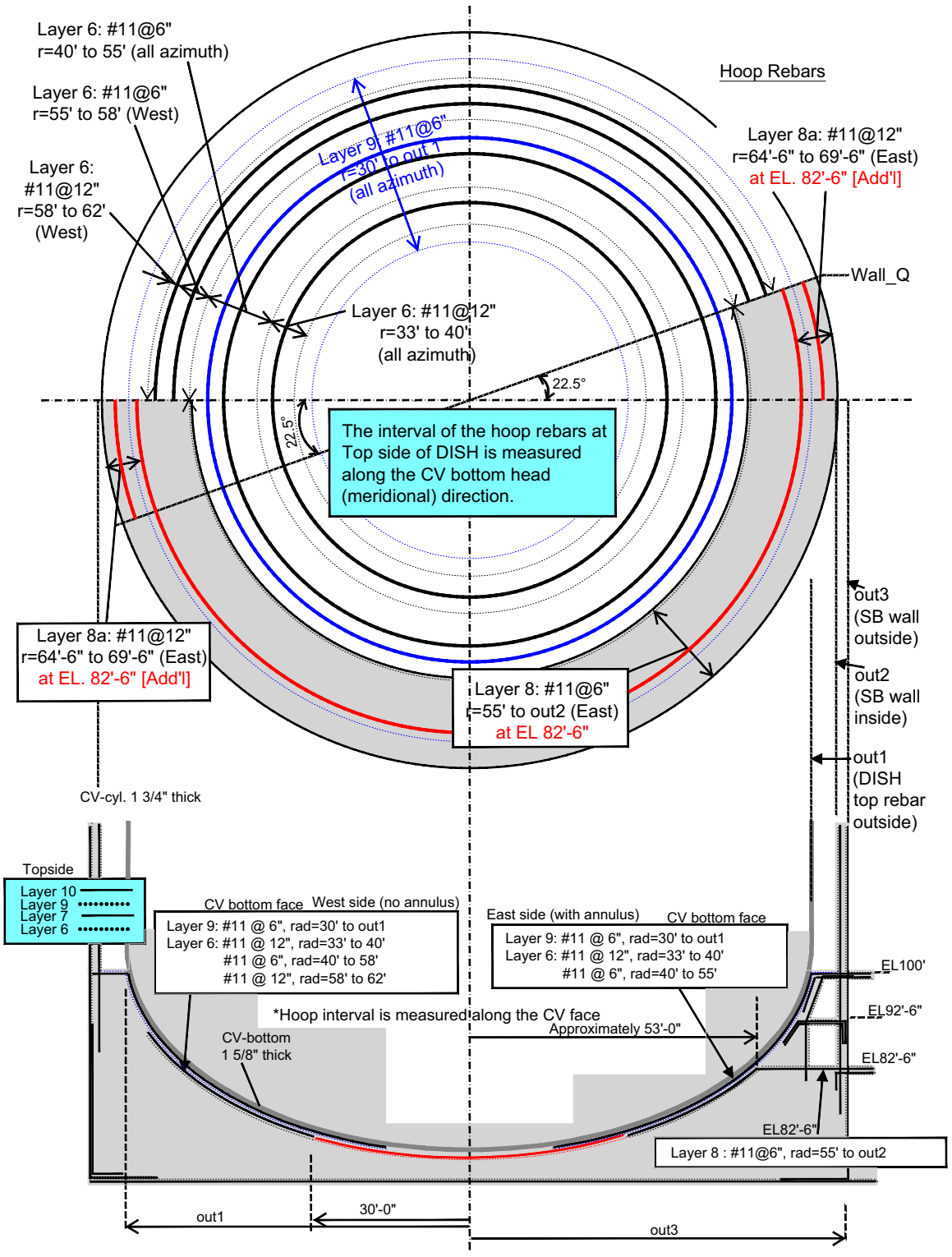

Figure 3.8.5-3 (Sheet 2 of 7)
Circumferential Reinforcement, Top Side of DISH

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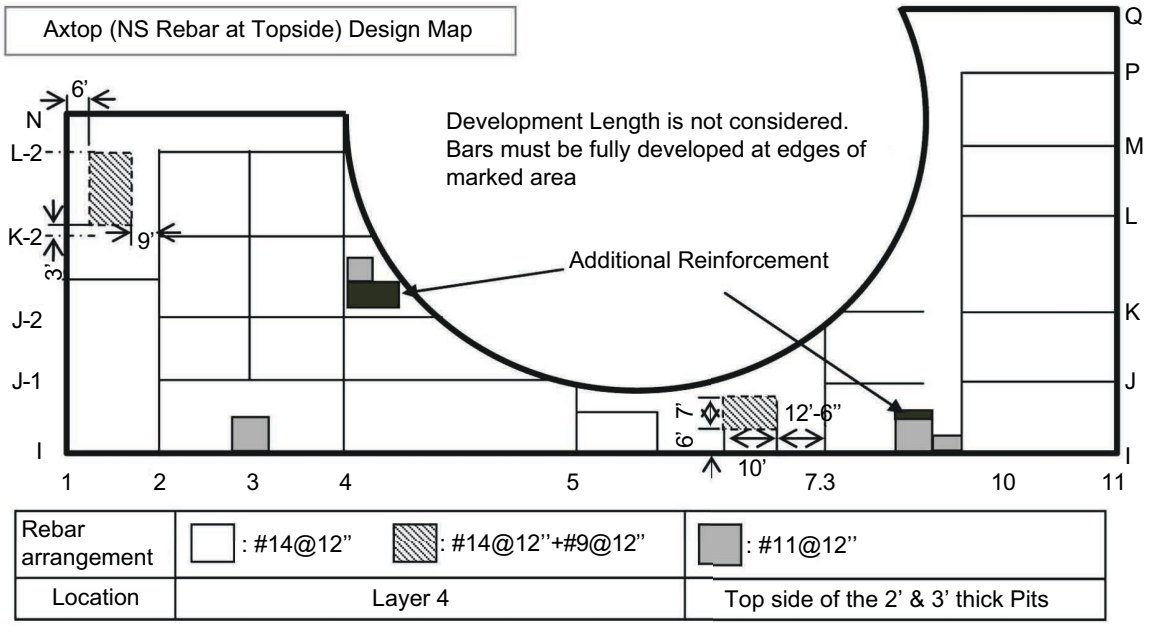

Figure 3.8.5-3 (Sheet 3 of 7)
Longitudinal Reinforcement Map,
Top Side in NS Direction

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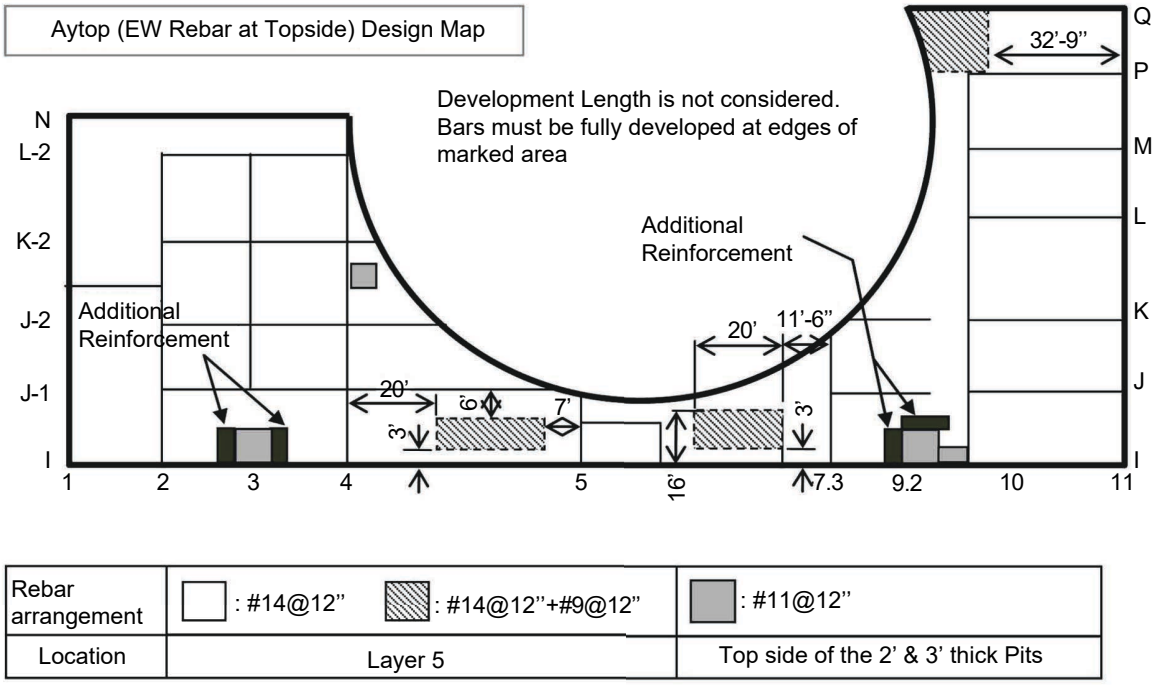

Figure 3.8.5-3 (Sheet 4 of 7)
Longitudinal Reinforcement Map,
Top Side in EW Direction

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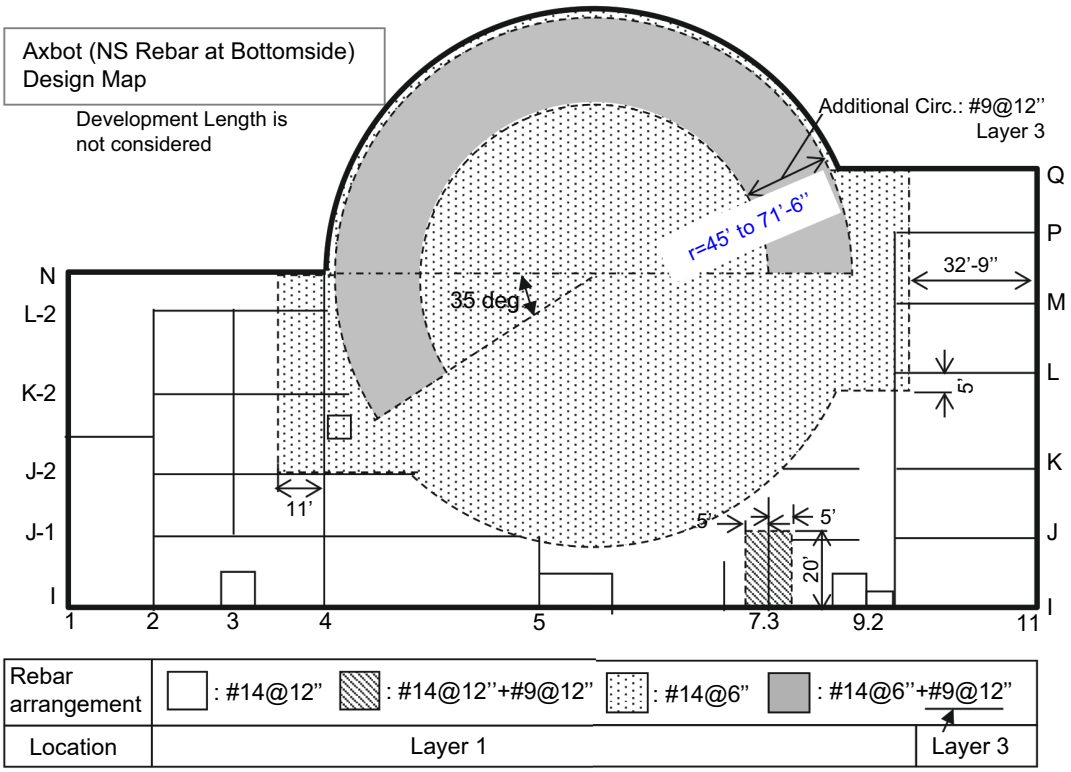

Figure 3.8.5-3 (Sheet 5 of 7)
Longitudinal Reinforcement,
Bottom Side of DISH and 6′ Basemat (NS)

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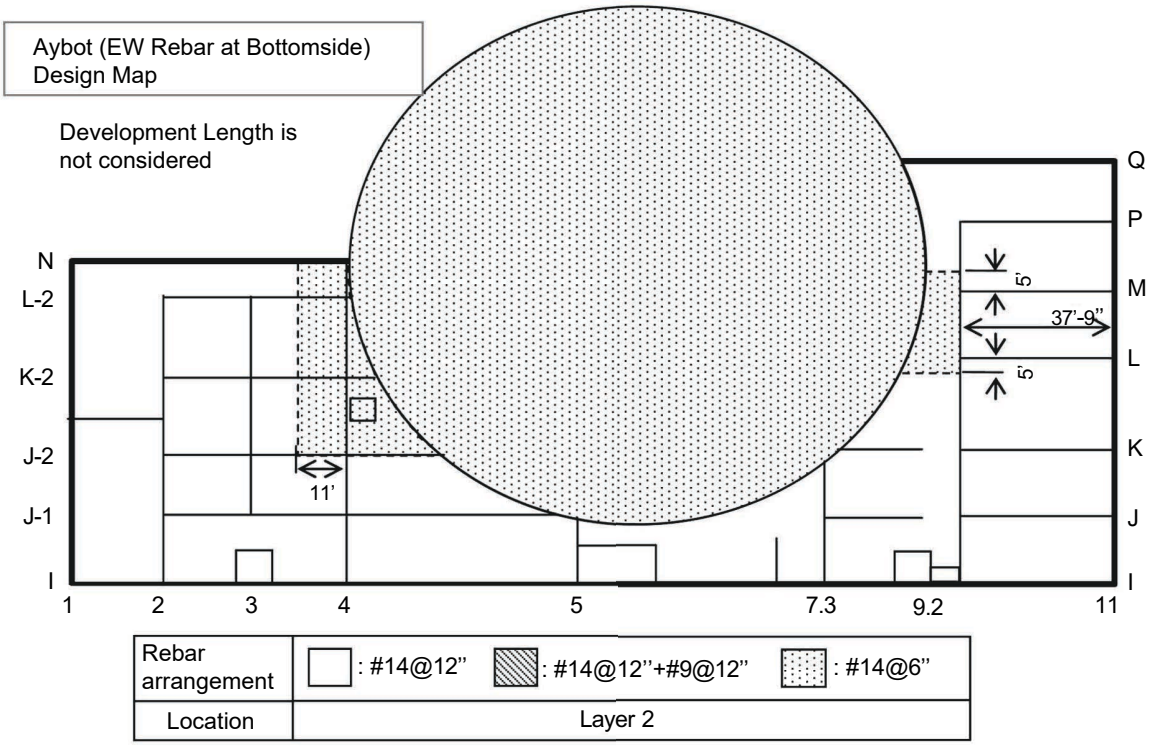

Figure 3.8.5-3 (Sheet 6 of 7)
Longitudinal Reinforcement,
Bottom Side of DISH and 6′ Basemat (EW)

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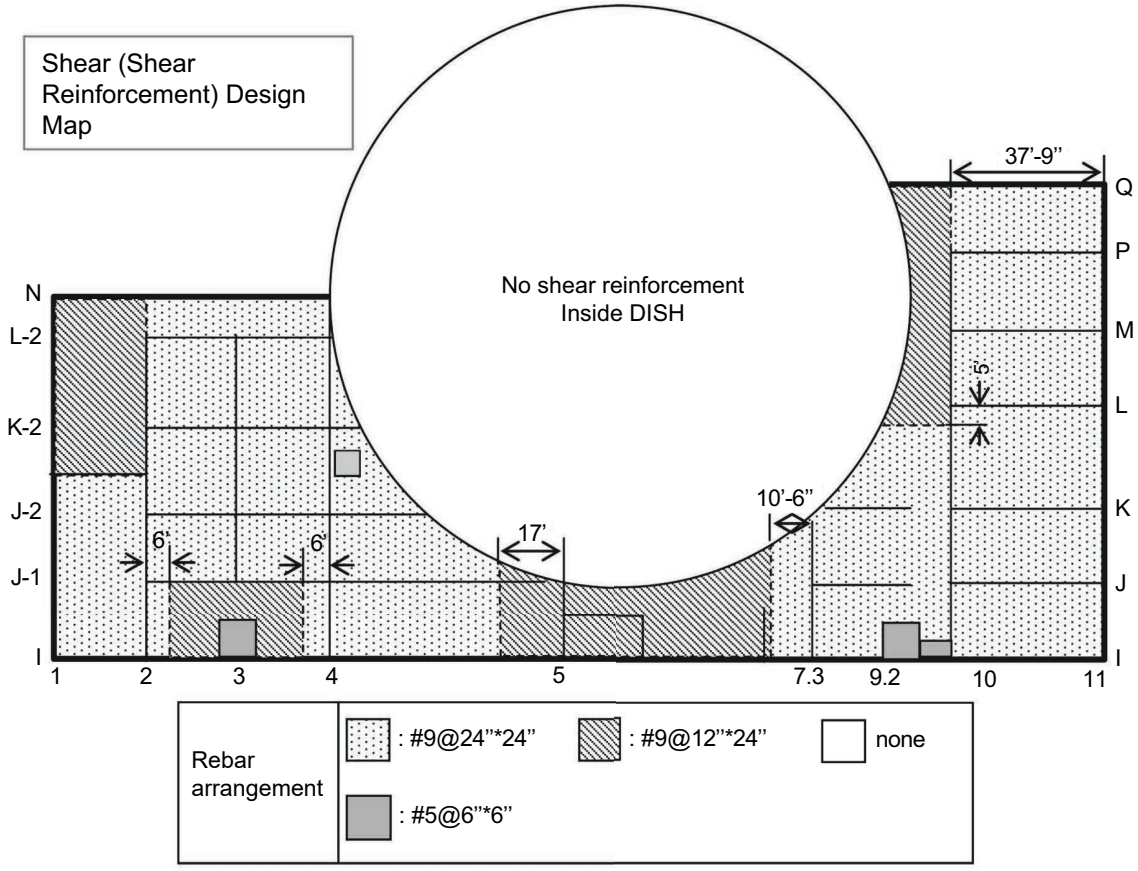

Figure 3.8.5-3 (Sheet 7 of 7)
Shear Reinforcement Map

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3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

To provide a high degree of integrity for the equipment in the reactor coolant system (RCS), components designed and constructed to the requirements for Class 1 in ASME Code, Section III are evaluated for design, service, and test conditions.

The design conditions include those pressure, temperature, and mechanical loadings selected as the basis for the design. Service conditions cover those normal operating conditions, anticipated transients, and accident conditions expected or postulated to occur during operation. The evaluation of the service and testing conditions includes an evaluation of fatigue due to cyclic stresses.

The following five operating conditions, as defined in ASME Code, Section III, are considered in the design of the reactor coolant system Class 1 components, auxiliary Class 1 components, reactor coolant system component supports, and reactor internals.

Level A Service Conditions - (Normal Conditions) These conditions include any condition in the course of system startup, operation in the design power range, hot standby, and system shutdown other than Level B, Level C, or Level D service conditions or testing conditions. Tests not requiring a pressure greater than the component design pressure are considered to be normal condition design transients.

Level B Service Conditions - (Upset Conditions, Incidents of Moderate Frequency) These conditions include any deviations from Level A service conditions anticipated to occur often enough that the design includes a capability to withstand the conditions without operational impairment. The Level B service conditions include those transients resulting from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Level B service conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of Level B service condition is included in the design specifications.

Level C Service Conditions - (Emergency Conditions, Infrequent Incidents) These conditions include those deviations from Level A service conditions that require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to establish that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The postulated occurrences for such events which result in more than 25 strong stress cycles are evaluated for cyclic fatigue using Level B service limits. Strong stress cycles are those having an alternating stress intensity value greater than that for 10^6 cycles from the applicable fatigue design curves.

Level D Service Conditions - (Faulted Conditions, Limiting Faults) These conditions include those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety is involved. Such considerations require compliance with safety criteria as may be specified by regulatory authorities.

Testing Conditions - Testing conditions are those pressure overload tests that include primary and secondary hydrostatic tests and steam generator tube leak tests specified. Other types of tests are classified under one of the other service condition categories.

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In addition to the design transients defined for evaluation of the ASME Code, Section III, Class 1 components, other transients are defined to address the same normal operating conditions, anticipated transients, and postulated accident conditions. These alternate transients are developed for evaluations of other effects. For example, a set of transients is developed for equipment qualification (see Section 3.11) and a set for accident analysis (see Chapter 15). These transients have somewhat different assumptions for the number of transients and sequence of events than do the design transients.

To provide a high degree of integrity for the equipment in the reactor coolant system, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients that may occur during plant operation.

To a large extent, the specific transients to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The plant condition (PC) categorization defined in ANS/ANSI 51.1 (Reference 1), which categorizes transients on the basis of expected frequency, are also part of the process to define transients and which service condition applies for a given transient.

The transients selected are severe enough or frequent enough to be of possible significance to component cyclic behavior. The transients selected are a conservative representation of transients that, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application for the 60-year design objective. These transients are described by pertinent variations in pressure, fluid temperature, and fluid flow. Because of the large number of possible operating transients, design transients are chosen to provide a conservative representation for component cyclic analysis. The frequency in some cases is greater than the maximum frequency that defines the plant condition in ANS N51.1 (Reference 1).

The design transients and the number of events of each that are normally used for fatigue evaluations of components are presented in Table 3.9-1. A limited number of events affecting only the core makeup tank or passive residual heat removal heat exchanger are not included in the design transients. Subsections 5.4.13 and 5.4.14 describe these events.

The effects of each transient vary in consequence for each of the analyzed components. For example, the reactor vessel is subject to the pressure and temperature variations in the reactor coolant loop flow, but, the core makeup tank and passive residual heat removal heat exchangers are subject only to the pressure changes for many of the reactor coolant system transients. Additionally, the steam generator is subject to changes in the feedwater and steam system parameters that may have little or no effect on the other Class 1 components.

The individual component fatigue evaluations are based on component specific analyses of the stress levels and cycles of applied stress of the design transients. In many cases, the fatigue evaluations for the individual components combine two or more of the design transients into one bounding condition for that component.

In some cases the use of the total number of the design transients in a component fatigue analysis may indicate the requirement for a significant redesign of a portion of a component. In such cases, the number of one or more of the transients evaluated in the analysis may be reduced. In each case, the number of transients to be included in the analysis is specified in the component design specification.

In accordance with ASME Code, Section III, Level D service conditions and up to 25 stress cycles for Level C service conditions may be excluded from cyclic fatigue analysis. Any Level C service condition in excess of the 25-cycle limit is evaluated for the effect on cyclic fatigue, using Level B

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criteria. The determination of which transients and seismic events are included in the 25-cycle exclusion is made separately for each component and piping line.

Levels C and D events are included in the design transients to provide the basis for pressure and temperatures used in the component stress analyses of these events. The number of events is given in the description of the transients and in Table 3.9-1 to support the determination that the fatigue evaluations do not have to consider these events.

[*The stress analysis, including analysis of fatigue, of the piping, applicable component nozzles, and piping and component supports includes the effect of thermal stratification and thermal cycling.*]* Thermal stratification may occur in piping when fluid rates are low and do not result in adequate mixing. Thermal cycling due to stratification may occur because of leaking valves or operational practices.

The design of piping and component nozzles in the AP1000 includes provisions to minimize the potential for and the effects of thermal stratification and cycling. [*Piping and component supports are designed and evaluated for the thermal expansion of the piping resulting from potential stratification modes. The evaluation includes consideration of the information on thermal cycling and thermal stratification included in NRC Bulletins 88-08 and 88-11 and other applicable design standards.*]*

The effects of earthquakes are not considered directly in the analyses leading to the fluid systems design transients. The presence or absence of seismic activity has no effect on the input data used for the analyses nor on the resulting pressure, temperature, and flow transients. Therefore, where applicable, in addition to the effects produced by the transients, earthquake loadings must be considered. See Subsection 3.9.3 for a description of the seismic loads and other mechanical loads and loading combinations evaluated.

3.9.1.1.1 Level A Service Conditions (Normal Conditions)

The following reactor coolant system transients are considered normal operating transients (plant condition PC-1 per ANS/ANSI 51.1) and are analyzed using Level A service limits:

- Reactor coolant pump startup and shutdown
- Plant heatup and cooldown
- Unit loading and unloading between 0 and 15 percent of full power
- Unit loading and unloading at 5 percent of full power per minute
- Step load increase and decrease of 10 percent of full power
- Large step load decrease with steam dump
- Steady-state fluctuations and load regulation
- Boron concentration equalization
- Feedwater cycling
- Core lifetime extension
- Feedwater heaters out of service
- Refueling
- Turbine roll test
- Primary leakage test
- Secondary leakage test
- Passive residual heat removal test
- Reactor coolant system makeup
- Daily load follow operations

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3.9.1.1.1.1 Reactor Coolant Pump Startup and Shutdown

The reactor coolant pumps are started and stopped during such routine operations as plant heatup and cooldown and in connection with recovery from certain transients, such as loss of power. Other (undefined) circumstances may also require pump starting and stopping.

Of the spectrum of reactor coolant system pressure and temperature conditions under which these operations may occur, three conditions have been selected for defining transients:

- Cold condition: 70°F and 400 psig - The minimum pressure required for reactor coolant pump operation may be as low as 100 psig. A pressure of 400 psig is considered a conservative value for design purposes.

- Pump restart condition: 100°F and 400 psig - These conditions are included to cover situations requiring stopping and restarting the pumps after plant heatup has commenced.

- Hot condition: 557°F and 2235 psig

These pressure and temperature values are defined for use in the design and fatigue evaluation processes. Actual pump starting and stopping conditions may be controlled by other factors such as reactor vessel material ductility considerations.

For reactor coolant pump starting and stopping operations, it is assumed that variations in reactor coolant system primary-side temperature and in-pressurizer pressure and temperature are negligible. Temperature and pressure changes in the steam generator secondary side are also assumed negligible. The only significant variables are the primary system flow and the pressure changes resulting from the pump operations.

The following cases are considered.

Case 1 - Pump Startup

Two startup cases are defined as described below.

One case represents the variations in reactor coolant loop flow at cold conditions, which accompany the simultaneous ramp up in speed of both reactor coolant pumps in one loop being driven from low speed to high speed. This case involves both the highest dynamic pressure loss and the largest change in dynamic pressure loss for cold conditions and, thus, conservatively represents the variations in reactor coolant flow accompanying operation of any single pump during cold conditions.

A case is also defined that represents the variations in reactor coolant loop flow at hot conditions, which accompany the startup of a second pump in the loop with a pump currently operating and both pumps in the other loop idle. This case involves both the highest dynamic pressure loss and the largest change in dynamic pressure loss and, thus, conservatively represents the variations in reactor coolant loop flow accompanying startup of any given single pump during hot conditions.

Case 2 - Pump Shutdown

This case represents the variations in reactor coolant loop flow, which accompany the shutdown of a pump when the only other operating pump is in the same loop. This transient is the reverse of the hot condition pump startup transient and involves an equally limiting change in dynamic pressure loss. It thus conservatively represents the variations in reactor coolant flow accompanying shutdown of any given single pump.

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Design values for the pump starting/stopping conditions are given in Table 3.9-2 along with the assumed number of occurrences.

An example of the consequence of a pump startup is that the loop flow change associated with pump startup develops a pressure differential in the normal (forward) direction across the divider plate of the steam generator in that loop. In the loop undergoing reverse flow, the direction of the divider plate differential is reversed. The magnitude of the dynamic pressure drop depends on the volumetric flowrate through the loop and on the density and viscosity of the reactor coolant.

3.9.1.1.1.2 Plant Heatup and Cooldown

For the purpose of designing the major reactor coolant system components, the plant heatup and cooldown operations are conservatively represented by uniform ramp temperature changes of 100°F per hour when the system temperature is above 350°F. (For the pressurizer vessel, the design cooldown rate is 200°F per hour.) This rate bounds both potential nuclear heatup operations and cooldown using the steam dump system when system temperatures are greater than 350°F. Below 350°F, only reactor coolant pump heat and small amounts of decay heat are available to heat the reactor coolant system. Cooldown between 350°F and the shutdown temperature of 125°F is accomplished via the normal residual heat removal system. In this range, a uniform ramp rate of 50°F per hour is considered to bound the temperature changes resulting from these operations.

The number of plant heatup and cooldown operations is defined as 200 each, which corresponds to approximately three occurrences per year for design purposes.

3.9.1.1.1.3 Unit Loading and Unloading Between 0 and 15 Percent of Full Power

The unit loading and unloading cases between the 0 and 15 percent load are represented by continuous and uniform ramp steam load changes, which require 30 minutes for loading and five minutes for unloading. During loading, reactor coolant temperatures are changed from their no-load values to their normal load programmed temperature values at 15 percent load. The reverse temperature change occurs during unloading.

Before loading, it is assumed that the plant is at hot standby, with feedwater cycling from the main or startup feedwater system. Loading commences, and during the first two hours, the feedwater temperature increases to the 15 percent load value, because of steam dump and turbine startup heat input to the feedwater.

After unloading, feedwater heating is reduced, steam dump is reduced to residual heat removal requirements, and feedwater temperature decreases from the 15 percent load value. Reactor coolant system pressure and pressurizer pressure are assumed to remain constant at the normal operating values during these operations.

The number of these loading and unloading transients is assumed to be 500 each for design purposes.

3.9.1.1.1.4 Unit Loading and Unloading at Five Percent of Full Power per Minute

The unit loading and unloading operations are conservatively represented by continuous and uniform ramp power changes of 5 percent per minute between the 15 percent and 100 percent power levels. This load swing is the maximum possible that is consistent with operation under automatic reactor control. The reactor temperature will vary with load prescribed by the reactor control system.

The number of loading and unloading operations is defined as 2000 each for the 60-year plant design objective. The 2000 occurrences includes the plant loading and unloading for the normal plant

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startup/shutdown, and loading resulting from all service levels B, C, and D transients that result in a reactor trip.

3.9.1.1.1.5 Step Load Increase and Decrease of 10 Percent of Full Power

The 10 percent step change in load demand results from disturbances in the electrical network to which the unit is tied. The reactor control system is designed to restore plant equilibrium without reactor trip following a 10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions between 15 and 100 percent of full load (the power range for automatic reactor control).

Following a step decrease in turbine load, the secondary-side steam pressure and temperature initially increase. The reactor coolant system average temperature and pressurizer pressure also increase, but this change lags slightly behind the secondary-side change. Because of the coolant temperature increase and the power mismatch between turbine and reactor, the control system automatically inserts the control rods to reduce core power. The reactor coolant temperature then decreases from its peak value to a value below its initial equilibrium value.

Pressurizer pressure also decreases from its peak value and follows the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash. This reduces the rate of pressure decrease. Subsequently, the pressurizer heaters turn on and restore the pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs. The secondary-side steam pressure and temperature initially decreases and the reactor coolant average temperature pressure initially decreases. The control system automatically withdraws the control rods to increase core power.

The decreasing pressure transient is reversed by actuation of the pressurizer heaters, and eventually the system pressure is restored to its normal value. The reactor coolant average temperature rises to a value above its initial equilibrium value.

The number of operations is specified as 3000 times each, or 50 times per year, for design purposes.

3.9.1.1.1.6 Large Step Load Decrease With Steam Dump

This transient applies to a step decrease in turbine load from full power. This step decrease in turbine load results in a rapid mismatch between nuclear and turbine power that automatically initiates a secondary-side steam dump and actuates the rapid power reduction system, which prevents both reactor trip and lifting of steam generator safety valves.

After the large step load decrease, reactor power is reduced at a controlled rate, which results in lower flow through the steam dump system.

The AP1000 plant is designed to accept a step load change to house load, without a reactor trip, with up to 40 percent of the load reduction provided by the steam dump capability. The balance of the load reduction is provided by the rapid power reduction system. The number of occurrences of this transient is specified at 200 times for design purposes.

3.9.1.1.1.7 Steady-State Fluctuations and Load Regulation

Reactor coolant pressure and temperature can vary around the nominal (steady-state) values during power operation. These variations can occur at many frequencies but for design purposes two cases are considered.

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Initial Fluctuations - Initial fluctuations are due to rod cycling during the first 20 full power months of reactor operation. Reactor coolant temperature is assumed to vary by ± 3°F, and pressure by ± 25 psi once during each 2-minute period. The total number of occurrences is specified as 1.5×10^5. The fluctuations are assumed to occur consecutively, but not simultaneously, with random fluctuations.

Random Fluctuations - Reactor coolant temperature is assumed to vary by ± 0.5°F, and pressure by ± six psi, once during each six-minute period. The total number of occurrences for design purposes does not exceed 4.6×10^6.

These small, primary-side fluctuations have no effect on the steam generator secondary side.

The above described steady-state fluctuations and the following load regulation transients are considered to be mutually exclusive. Component evaluations are based on the more limiting of either the steady-state fluctuations or the load regulation transients.

Load Regulation - Load regulation refers to the relatively small, rapid fluctuations in load, regarding some nominal operating condition that is the result of the participation of the plant in some form of grid frequency control. The nominal operating condition is either a constant power level or a very slowly changing power level such as that which occurs because of a daily load follow maneuver.

For design purposes it is assumed that the plant experiences load changes of 10 percent of rated load peak-to-peak at a rate of 2 percent of rated load per minute. It is assumed that up to 35 of these load swings may occur during any given day of plant operation. Frequency control capability is to be provided while performing ramp power changes required for load follow maneuvers. This capability is to be provided within 15 to 100 percent of full power. Load regulation is performed with backup heaters continuously energized and spray flow modulated to control pressure.

The load regulation transient can result in considerable cycling of components affected by pressurizer spray, the pressurizer backup heaters, and the control rod drive mechanisms (CRDMs). The 37,000 design occurrences of load regulation correspond to the following component cycles:

- Control rod drive mechanism stepping ≤ 1.3×10^6 steps
- Pressurizer spray on-off cycling ≤ 94,000
- Pressurizer backup heater on-off cycling ≤ 20,000

This cycling of the components should be considered to occur in addition to the duty cycles imposed on these components due to other modes of plant operation.

3.9.1.1.1.8 Boron Concentration Equalization

Following any large change in boron concentration in the reactor coolant system, the pressurizer spray is operated to equalize the concentration between the loops and the pressurizer. This can be done by manually operating the pressurizer backup heaters, which causes a pressure increase and spray initiation at a pressurizer pressure of approximately 2275 psia. The pressure increases to approximately 2280 psia before being returned to 2250 psia by the proportional spray. The pressure is maintained at 2250 psia by spray operation, matching the heat input from the backup heaters, until the concentration is equalized.

Since reactor coolant system boron concentration changes are not required for daily load follow, it is assumed that this operation is performed about once each week. For design purpose the total number of occurrences is placed at 2900.

The operations cause no significant effects on the steam generator secondary side. The only effects of these operations on the primary system are as follows:

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- The reactor coolant pressure varies in step with the pressurizer pressure.

- The pressurizer surge line nozzle at the hot leg will experience the temperature transient associated with outflow from the pressurizer.

3.9.1.1.1.9 Feedwater Cycling

The feedwater cycling transient occurs when the plant is being maintained at hot standby or no-load condition. With the plant in the steam pressure mode of steam dump control, a low steam generation rate occurs because of dissipation of decay and/or pump heat. This low steam generation rate decreases steam generator water level.

To compensate for the decreasing level, the steam generators are fed using the startup feedwater control. Either the main or startup feedwater pumps can continuously provide flow to the steam generators and maintain the desired steam generator level. For this case the reactor coolant system transient is relatively moderate.

If the startup feedwater control system is unavailable, the feedwater is provided intermittently in a slug-feeding mode.

Two modes of slug-feeding the steam generators are considered. In the first mode, it is assumed that the steam generators are slug-fed through the startup feedwater nozzle once every two hours. In the second mode, it is assumed that tighter control of steam generator water level is maintained by slug-feeding once every 24 minutes.

For design purposes, the following numbers of feedwater cycling transients are considered:

- Mode 1: Slug feed every 2 hours – 3000 cycles
- Mode 2: Slug feed every 24 minutes – 15,000 cycles

The component designers consider both modes of slug-feeding. Each component evaluation is based on the more limiting of the two modes.

3.9.1.1.1.10 Core Lifetime Extension

These transients occur at the end of core life when the critical boron concentration required to maintain full thermal power conditions becomes less than achievable (approximately 10 ppm). To extend core lifetime beyond this point, the operator does the following:

- Allows the reactor coolant system average temperature to decrease below the normal programmed temperature, thereby adding reactivity to the core through the negative moderator temperature coefficient.

- Manually controls the turbine to maintain full electrical load until the turbine control valves have fully opened.

- Reduces steam flow by an amount that will maintain the plant at full rated electrical load after a feedwater heater has been taken out of service and allow plant conditions to reach a new steady state.

- Takes a feedwater heater out of service.

This process is repeated until the maximum allowable feedwater heaters have been taken out of service and the turbine control valves have fully opened.

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For design purposes, the number of occurrences of these transients is a total of 40 transients. During this mode of operation, the plant is not capable of daily load follow operation. Thus, this transient is considered separately from unit loading and unloading transients.

3.9.1.1.1.11 Feedwater Heaters Out of Service

These transients occur when one or more feedwater heaters are taken out of service. During the time that the heaters are out of service, the operator maintains the plant at full rated thermal load. To accomplish this, the operator performs the following:

- Calculates the appropriate steam flow reduction which will maintain the plant at full rated thermal load after the heater has been taken out of service.
- Reduces steam flow by the appropriate amount and allow plant conditions to reach a new steady state (approximately 10 minutes).
- Takes heater (or heaters) out of service.

The transient is based on the maximum allowable number of heaters out of service. For design purposes, the number of occurrences of this transient is a total of 180.

3.9.1.1.1.12 Refueling

At the beginning of the refueling operation, the reactor coolant system is assumed to have been cooled down to 140°F. The vessel head is removed and the refueling canal is filled. This is done by transferring water from the in-containment refueling water storage tank, which is conservatively assumed to be at 70°F, to the refueling cavity by the spent fuel pool cooling system (SFS).

This operation is assumed to occur 40 times over the expected plant design. This transient is experienced only by the primary system.

3.9.1.1.1.13 Turbine Roll Test

This transient is imposed upon the plant during the hot functional test for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature (no-load conditions), and the steam generator is used to perform a turbine roll test. However, the plant cooldown during this test exceeds the 100°F per hour design rate.

The number of such test cycles is specified at 20 times, to be performed at the beginning of plant operation before reactor operation. This transient occurs before plant startup, so the number of cycles is independent of other operating transients.

The transient curves and the number of cycles are based on a conservatively high steam flowrate for turning the turbine.

3.9.1.1.1.14 Primary-Side Leakage Test

A leakage test is performed after each opening of the primary system. During this test the primary system pressure is raised (for design purposes) to 2500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks.

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The secondary side of the steam generator is pressurized so that the pressure differential across the tubesheet does not exceed 1650 psi. This is accomplished with the steam, feedwater, and blowdown lines closed.

For design purposes the number of occurrences is a total of 200 cycles.

3.9.1.1.1.15 Secondary-Side Leakage Test

A secondary side leakage test is performed after each opening of the secondary system to check closures for leakage. For design purposes, it is assumed that the steam generator secondary side is pressurized to just below its design pressure to prevent the safety valves from lifting. So that a secondary-side to primary-side pressure differential of 670 psi is not exceeded, the primary side must also be pressurized. The 670 psi differential is the steam generator design differential pressure for secondary-to-primary pressure. The primary system must be above the minimum temperature imposed by reactor vessel material ductility requirements (that is, between 120°F and 250°F). It is assumed that this test is performed 80 times for design purposes.

3.9.1.1.1.16 Not Used

3.9.1.1.1.17 Passive Residual Heat Removal Test

During the initial test program, the passive residual heat removal flow and heat transfer rates are tested. Passive residual heat removal flow is initiated by opening the passive residual heat removal isolation valves. For component design purposes, the temperature and pressure responses to this testing are based on a conservative definition of the test conditions with a total of 5 occurrences.

3.9.1.1.1.18 Reactor Coolant System Makeup

The chemical and volume control system makeup subsystem is used to accommodate normal minor leakage from the reactor coolant system. On a low programmed pressurizer level signal one of the chemical and volume control system makeup pumps starts automatically in order to provide makeup. The pump automatically stops when the pressurizer level increases to the high programmed setpoint. The addition of the makeup water to the reactor coolant system via the chemical and volume control system purification loop and attendant changes in reactor coolant system parameters constitute the reactor coolant system makeup design transient. The total number of occurrences of the makeup transient is 5640, which corresponds to twice per week during the plant design objective of 60 years assuming a 90 percent availability factor for the plant.

3.9.1.1.1.19 Daily Load Follow Operations

During the load follow operations, the plant power is reduced from the 100-percent power to 50-percent at a prescribed rate and remains there for a specified time, and then the power ramps up to 100-percent power at a prescribed rate. Power remains at 100-percent power for the remainder of the 24-hour cycle. The reactor coolant temperature will vary with load as prescribed by the reactor control systems.

The AP1000 features a rod control system that provides a load follow capability without requiring a change in the boron concentration in the coolant. Thus, the reactivity gain available from temperature reduction is not required for load follow, and reduced temperature return to power is not applicable to the AP1000.

The number of daily load follow operations is specified as 17,800 times during the plant design objective of 60 years. One swing of load follow operation consists of one power ramp down from

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steady-state 100-percent power to 50-percent power and one power ramp up from steady-state 50-percent power to 100-percent power.

3.9.1.1.2 Level B Service Conditions (Upset Conditions)

The following paragraphs describe the reactor coolant system upset condition transients, which are considered to be plant condition PC-2 and PC-3 per ANS/ANSI 51.1. From the standpoint of the use of design transient in the evaluation of cyclic fatigue, there is no difference between PC-2 and PC-3. These transients are analyzed using Level B service limits and are as follows:

- Loss of load
- Loss of power
- Reactor trip from reduced power
- Reactor trip from full power
  Case A - with no inadvertent cooldown
  Case B - with cooldown and no safeguards actuation
  Case C - with cooldown and safeguards actuation
- Control rod drop - three cases
- Cold overpressure
- Inadvertent safeguards actuation - three cases
- Partial loss of reactor coolant flow
- Inadvertent reactor coolant system depressurization
- Excessive feedwater flow
- Loss of power with natural circulation cooldown
  Case A - loss of power with natural circulation cooldown with onsite ac power
  Case B - loss of power with natural circulation cooldown without onsite ac power
- Loss of feedwater with non-safety PRHR HX modulation control

Under the upset condition transients listed, 505 reactor trip cases are assumed. A total of 505 reactor trips for design purposes exceeds the design goal. The design goal for the AP1000 is less than one unplanned trip per year. The number of reactor trips in the design transients represents a conservative number for the analysis of cyclic stresses in the components and is not to be considered an estimate of expected plant performance.

For some component or portions of components, the number of reactor trips analyzed for the effects of cyclic loads may be reduced from 505. In each case the number of reactor trips analyzed is greater than the design goal of one per year.

3.9.1.1.2.1 Loss of Load

This transient involves a step decrease in turbine load from full power (turbine trip) without immediate automatic reactor trip or rapid power reduction. These conditions produce the most severe pressure

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transient on the reactor coolant system under upset conditions. The reactor is assumed to trip as a consequence of a trip initiated by the reactor protection system. Since redundant means for tripping the reactor are provided by the reactor protection system, a transient of this nature is not expected, but is included to confirm conservative component design.

The number of occurrences of this transient is specified at 30 times for design purposes.

3.9.1.1.2.2 Loss of Power

This transient applies to a blackout situation involving the loss of outside electrical power to the station, assumed to be operating initially at 102 percent power, followed by reactor and turbine trips. The reactor coolant pumps are de-energized, as are electrical loads connected to the turbine-generator bus, including the main feedwater and condensate pumps.

As the reactor coolant pumps coast down, reactor coolant system flow reaches an equilibrium value through natural circulation. This condition permits removal of core residual heat through the steam generators, which receive feedwater from the startup feedwater system. For this event reactor coolant temperature stabilizes at hot standby conditions.

The number of occurrences of this transient is specified at 30 times for design purposes.

For one occurrence, a worst case is postulated in which the shell side of a single steam generator is assumed to be emptied after the blackout. The startup feedwater flow is then delivered into the hot, dry shell side. The steam generator tube and secondary shell integrity are evaluated for this condition.

3.9.1.1.2.3 Reactor Trip from Reduced Power

A significant percentage of reactor trips occur at low power as the plant is being brought up from hot standby to power. The low power reactor trip is provided to bound these occurrences without the excessive conservatism associated with the reactor trip from full power. The transient is assumed to start at 25 percent load, which bounds the conditions associated with achieving criticality, turbine roll, and turbine synchronization; establishing automatic rod control; and making the transitions in feedwater control from the startup feedwater nozzle to the main feedwater nozzle.

Reactor coolant system temperature and pressure variations are similar to those of reactor trip from full power, but are smaller. The transients continue until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. Controlled steam dump and startup feedwater remove any core residual heat and prevent steam generator safety valve actuation. For design purposes, 180 reactor trips from reduced power are postulated.

3.9.1.1.2.4 Reactor Trip from Full Power

Reactor trips from full power may occur for a variety of reasons. The reactor coolant temperature and pressure undergo rapid decreases from full power values as the reactor protection system causes the control rods to move into the core. Transients also occur in the secondary side of the steam generator because of continued heat transfer from the reactor coolant through the steam generators.

These transients continue until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. Continuation of feedwater flow and controlled steam dump remove the core residual heat and prevent steam generator safety valve actuation. For design purposes, reactor trip from full power is assumed to occur 120 times.

Three reactor trip cooldown transients are considered.

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Case A - Reactor Trip With No Inadvertent Cooldown

Case A of the reactor trip transient represents the likely case in which the systems function as designed and as such no inadvertent cooldown occurs. This case maintains conservatism over the best estimate reactor trip via assumptions that result in the RCS temperatures initially cooling below the no load conditions before steam and feedwater flow are both controlled to bring the plant back to no-load conditions and maintain it at no load. For design purposes, 50 occurrences of this transient are specified.

It is assumed that for most reactor trip Case A transients, the turbine control system operates as designed. For five of the reactor trip Case A transients, it is conservatively assumed that the control system fails, which results in an emergency turbine overspeed. This situation could be initiated with malfunction of the turbine control system, which results in a turbine speed increase past the overspeed trip setpoint. It is assumed that the reactor then trips and that the turbine speed increases to 120 percent of nominal, with accompanying proportional increases in generator bus frequency, reactor coolant pump speed and reactor coolant flowrate. None of the other reactor coolant system primary side, pressurizer, or steam generator secondary side variables is affected.

For design purposes it is assumed that the emergency turbine overspeed constitutes a special case of the reactor trip with no inadvertent cooldown transient. Thus, for five of the 50 occurrences, the effects of the reactor coolant flow variation are considered in addition to the basic pressure and temperature variations.

Case B - Reactor Trip With Cooldown and No Safeguards Actuation

This transient is similar to Case A, except that a failure in the main feedwater control valves is assumed to delay main feedwater isolation. Additionally, for this case, it is assumed that the startup feedwater pumps are started within five seconds of the reactor trip. Both main and startup feedwater flow continue until the feedwater flow is assumed to be isolated just prior to the actuation of safeguards system. This maintains a high heat transfer rate through the steam generator, which continues to drive the primary side pressure and temperature down. Flow through the startup feedwater nozzle is continued until isolated on Low-2 T_{cold}. No other safety functions are assumed to be actuated following the Low-2 T_{cold} signal. The plant is then brought back to the no-load condition.

For design purposes, 50 occurrences of this transient are specified.

Case C - Reactor Trip with Cooldown and Safeguards Actuation

This transient is similar to Case B, but it is assumed that the steam generator secondary side shrinkage is sufficient to actuate the passive residual heat removal heat exchanger in addition to the PLS starting of the startup feedwater pumps. The core makeup tanks are then actuated on Low-2 pressurizer water level signal. Main and startup feedwater are assumed to continue until isolated on Low-2 T_{cold}. After 30 minutes, the plant is assumed to be manually controlled to either cold shutdown or returned to the no-load conditions.

For design purposes, 20 occurrences of this transient are specified. The components analyses should be based on 20 occurrences of either the reactor trip - Case C transients or 20 occurrences of the control rod drop - Case B transients, whichever transient is more severe for the component being evaluated.

3.9.1.1.2.5 Control Rod Drop

This transient occurs when one or more rod cluster control assemblies inadvertently drop into the core because of equipment failure or operator error. If this rod drop occurs while the plant is at power, pressure and temperature transients will occur in the reactor coolant system and on the secondary side of the steam generators. The severity of the rod drop accident depends on a number of factors,

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such as the number and worth of rod cluster control assemblies that drop, and the value of the moderator temperature coefficient of reactivity.

The following three types of control rod drop transients are postulated for design purposes.

Control Rod Drop - Case A

This transient occurs when the worth of the dropped control rods is high. When the rods drop, reactor power quickly decreases, but plant load is maintained at its initial value.

The steam load-reactor power mismatch causes the plant to cool down, eventually leading to a reactor trip, main feedwater isolation, turbine trip, and steamline isolation following a Low-2 steam line pressure signal. No other safety functions are assumed to be actuated during this transient. Following the reactor trip, the steam generator water level falls because of shrinkage in the secondary side. This is assumed to cause startup feedwater flow to actuate on low steam generator level, thus continuing to drive the primary system temperature and pressure down. The transient is terminated 10 minutes after rod drop.

For design purposes, 30 occurrences of this transient are specified.

Control Rod Drop - Case B

This transient is similar to Case A, except the PRHR HX and CMT are assumed to be actuated. Therefore, the response of the various plant parameters during this transient are very similar to those of reactor trip from full power - Case C.

For design purposes, a combined total of 20 occurrences of the control rod drop - Case B transient and the reactor trip from full power - Case C transient are defined. The components analyses should be based on 20 occurrences of either the control rod drop - Case B transients or 20 occurrences of the reactor trip - Case C transients, whichever transient is more severe for the component being evaluated.

Control Rod Drop - Case C

This transient occurs when the worth of the dropped rod is relatively low and the moderator temperature coefficient of reactivity is negative. When the rod drops, reactor power is reduced but no trip occurs.

However, plant steam load is maintained at the initial value through the transient.

The steam load-reactor power mismatch causes the plant to cool down. With a negative moderator temperature coefficient of reactivity, reactor power returns to its initial value. The plant eventually stabilizes, with reactor power, plant steam flow, reactor coolant system pressure, and pressurizer pressure equal to their initial values, but the reactor coolant system temperature and steam generator secondary-side temperature and pressure are lower.

The magnitude of the reactor coolant system temperature reduction is proportional to the relative worth of the dropped control rod and the negative moderator temperature coefficient of reactivity. For design purposes, 30 occurrences of this transient are specified.

At the end of the control rod drop - Case C transient, plant parameters stabilize at their final values. After plant parameters achieve their final values, the plant remains at these conditions indefinitely. Subsequently, plant parameters are returned to their initial values.

Following initiation of recovery, hot and cold leg temperatures and steam generator steam temperature and pressure return to their initial values over a 30 minute period. Pressurizer water

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volume returns to its initial value in about the same amount of time as the return of hot and cold leg temperatures to their initial values. Pressurizer surge rate variation is consistent with the increase in pressurizer water level.

3.9.1.1.2.6 Cold Overpressure

The relief valves located in the residual heat removal pump suction piping provide the capability for additional reactor coolant system inventory letdown in order to maintain the reactor coolant system pressure consistent with the reactor vessel pressure temperature limits, as required by Appendix G of 10 CFR Part 50. Reactor coolant system cold overpressurization occurs at low temperature (below 350°F) during plant heatup or cooldown, and can occur with or without a steam bubble in the pressurizer. A cold overpressurization is especially severe when the reactor coolant system is water solid. The event is inadvertent, and can be generated by an equipment malfunction or an operator error.

Cold overpressure events are initiated by either a mass addition that exceeds normal letdown capabilities, or a heat addition that attempts to expand the reactor coolant system water volume.

Under water-solid conditions, a worst-case scenario, the mass addition causes an increase in system pressure until the relief valve set pressure, plus accumulation, is reached. The relief valves remain open, with the system pressure stabilizing at the set pressure plus accumulation, until the mass injection is terminated by the operator. Heat addition, also under water-solid conditions, results in a system pressure increase that eventually is terminated by the relief valves.

Once thermal equilibrium is established between the heat source and the reactor coolant system, and the volume expansion has been let down through the relief valves, system pressure stabilizes at the relief valve set pressure.

Fifteen reactor coolant system cold overpressure events, as described above, are specified for design purposes.

3.9.1.1.2.7 Inadvertent Safeguards Actuation

A spurious system-level actuation of the passive core cooling system results in an immediate reactor trip followed by actuation of the various components of the passive core cooling system. The resulting transient is bounded by the reactor trip Case C. The number of reactor trip transients is sufficient to cover a system-level inadvertent safeguards actuation.

A spurious actuation of the passive residual heat removal heat exchanger isolation valves or the core makeup tank valves causes cold reactor coolant to flow into the reactor coolant system. Rapid changes in the temperature of the core makeup tank or passive residual heat removal heat exchanger and associated piping occur. Ten events of this limited transient are postulated.

3.9.1.1.2.8 Partial Loss of Reactor Coolant Flow

This transient applies to a partial loss of flow from full power in which a reactor coolant pump is tripped out of service as a result of loss of power to that pump. The consequences of such an accident are a reactor trip on Low-2 reactor coolant flow, followed by a turbine trip; actuation of startup feed control; and automatic opening of the steam dump system. Flow reversal occurs in the associated cold leg. The normal flow direction is maintained in the hot leg of the affected loop but at a reduced rate. Flow through the operating pump in this loop increases.

Operation of the steam dump system tends to bring the plant toward no-load conditions. Cold feedwater from the startup feedwater system then enters the steam generators, causing the plant to

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cool down. This cooldown continues until termination of startup feed water. The plant is then returned to no-load conditions.

The number of occurrences of this transient is specified as 60 times for design purposes.

3.9.1.1.2.9 Inadvertent Reactor Coolant System Depressurization - Umbrella Case

Several events can be postulated as occurring during normal plant operation that cause rapid depressurization of the reactor coolant system. These include the following:

- Actuation of a single pressurizer safety valve with failure of the valve to reclose
- Malfunction of a single pressurizer pressure controller causing two pressurizer spray valves to open
- Inadvertent opening of one pressurizer spray valve
- Inadvertent opening of the auxiliary spray valve

Umbrella Case - Of these events, the pressurizer safety valve actuation causes the most severe reactor coolant system pressure and temperature transients. It can be used as an umbrella case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

Although inadvertent actuation of the pressurizer spray is included among the transient events covered by the umbrella case, the pressurizer safety valve actuation case selected to represent the depressurization transients does not involve spray operation. Therefore, for the umbrella case, it is assumed that pressurizer spray is not actuated and that no temperature transients due to flow occur at the spray nozzle.

Inadvertent Pressurizer Spray - The inadvertent pressurizer spray transient represents the depressurization transient, with the most significant temperature variations on portions of the pressurizer, spray nozzle, and spray piping. Should auxiliary spray flow be inadvertently initiated, it could cause a rapid temperature change at the pressurizer spray nozzle and on the pressurizer vessel. Therefore, to provide a conservative design for these components, an inadvertent pressurizer spray transient is defined.

An inadvertent pressurizer spray occurs if the normal spray valve is opened during normal plant operation because of either failure of a control component or operator error. This introduces water at reactor coolant system cold leg temperature into the pressurizer. The flowrate is assumed to be the maximum design spray flowrate. This transient results in a pressure decrease and, eventually, in a low-pressure reactor trip.

An inadvertent auxiliary spray occurs if the auxiliary spray valve is opened during normal plant operation because of either failure of a control component or operator error. The opening of the auxiliary spray valve causes an inadvertent spray transient only during the limited time that the makeup pump in the chemical and volume control system is operating. The inadvertent auxiliary spray introduces cold water into the pressurizer, which results in a sharp pressure decrease and, eventually, in a Low-2 pressure reactor trip.

The temperature of the auxiliary spray flow is dependent upon the performance of the regenerative heat exchanger. The most conservative case assumes that the letdown stream is shut off and that unheated charging fluid enters the 653°F pressurizer. It is assumed that the temperature of the spray water is 70°F and that the spray flowrate is equal to the normal charging rate.

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For inadvertent auxiliary spray, it is also assumed that the spray flow continues for five minutes before it is shut off and that the temperature changes at the pressurizer and spray nozzle occur as steps. For design purposes, it is assumed that no reactor coolant temperature changes occur as the result of inadvertent auxiliary spray.

For design purposes, 20 occurrences of the inadvertent reactor coolant system depressurization transient are specified. Component evaluations are based on the more limiting of either the umbrella case or the inadvertent spray case, or, alternatively, are based on a combination of design transients (5 stuck open PSV and 15 inadvertent spray) that total up to the 20 transients. For those components for which the limiting transient is caused by the inadvertent pressurizer spray transient, 10 occurrences of inadvertent normal spray and five occurrences of inadvertent auxiliary spray are postulated.

3.9.1.1.2.10 Excessive Feedwater Flow

An excessive feedwater flow transient is conservatively defined as an umbrella case to cover the occurrence of several events of the same general nature. The postulated transient results from inadvertent opening of a feedwater control valve while the plant is at the hot standby or no-load condition, with the feedwater, condensate, and heater drain systems in operation.

It is assumed that the stem of a feedwater control valve fails and the valve immediately reaches the full open position. In the steam generator directly affected by the malfunctioning valve (failed loop), the feedwater flow step increases from essentially zero flow to the value determined by the system resistance and the developed head of the operating feedwater pumps. Steam flow is assumed to remain at zero.

The CMTs and PRHR HX are actuated on the Low-2 pressurizer water level signal. The startup feedwater and main feedwater flow in the faulted loop are effectively isolated on Low-2 T_{cold}.

This transient is assumed to occur 30 times for design purposes.

3.9.1.1.2.11 Loss of Power with Natural Circulation Cooldown

This event is the same as a loss of power transient, except that the reactor coolant system temperature is reduced by natural circulation through the operation of either the startup feedwater pumps and steam dump through the power-operated relief valves if onsite power is available or the passive residual heat removal system transferring heat to the in-containment refueling water storage tank if onsite power is not available. For design purposes 30 natural circulation cooldown occurrences, are assumed. These two cases are discussed below.

Case A - Loss of Power with Natural Circulation Cooldown with Onsite ac Power

For this case, natural circulation cooldown is performed with onsite ac power available. This permits operation of the startup feedwater pumps which enables steam dump through the steam generator power-operated relief valves. This transient is analyzed assuming at least one onsite diesel is operable. For this case the startup feedwater pumps operate and the control rod drive mechanism fan coolers operate to maintain the temperature of the reactor vessel head close to the temperature of the remainder of the reactor vessel. For design purposes, 20 occurrences of this transient are assumed.

Case B - Loss of Power with Natural Circulation Cooldown without Onsite ac Power

For this case, the reactor coolant is cooled by natural circulation with the passive residual heat removal heat exchangers. For this case, no credit is taken for nonsafety-related equipment including the diesel generators. For design purposes, 10 occurrences of this transient are assumed.

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3.9.1.1.2.12 Loss of Feedwater with Non-Safety PRHR HX Modulation Control

A loss of feedwater transient with non-safety PRHR HX throttling occurs if Low-2 narrow range steam generator level and Low-2 SFW flow are present in at least one steam generator, and at least one RCP is running in loop 1. For this transient, PRHR is designed to throttle the PRHR HX outlet valves to control Tcold to a fixed setpoint such that the PRHR HX acts as a heat sink when one or both steam generators are unavailable as heat sinks.

This transient results in unique thermal-hydraulic conditions that impact the steam generator PRHR nozzle which is not bounded by existing transients. For the RCS, pressurizer, and steam generator secondary side, this transient is bounded by reactor trip Case A.

For the PRHR steam generator nozzle and PRHR system, which includes the PRHR heat exchanger and the piping and valves between the PRHR heat exchanger outlet to the steam generator nozzle, the component stress fatigue analyses are based upon 20 occurrences of the Reactor Trip Case C transient or 20 occurrences of the Loss of Feedwater with Non-Safety PRHR HX throttling, whichever is more limiting.

3.9.1.1.3 Level C Service Conditions (Emergency Conditions)

The following paragraphs describe the reactor coolant system emergency condition transients considered to be plant condition PC-4 per ANS/ANSI 51.1. A list of these transients follows. The effect of these events are analyzed using Service Level C limits. As noted previously, up to 25 strong stress cycles due to these transients are not analyzed for cyclic fatigue. Any cycles exceeding the 25 excluded are analyzed for cyclic fatigue using Service Level B limits. The mechanical loads due to pipe rupture are analyzed using Service Level D limits. See Subsection 3.6.2 for a discussion of the analysis of mechanical loads due to pipe break.

- Small loss of coolant accident
- Small steam line break
- Complete loss of flow
- Small feedwater line break
- Steam generator tube rupture
- Inadvertent opening of automatic depressurization system valves

3.9.1.1.3.1 Small Loss-of-Coolant Accident

For design transient purposes, the small loss-of-coolant accident is a pipe break equivalent to the severance of a 1-inch ID branch connection to the reactor coolant system. It is assumed that the passive core cooling system is actuated and that it delivers water at a minimum temperature of 70°F to the reactor vessel.

It is assumed that this transient occurs five times for design purposes.

3.9.1.1.3.2 Small Steam Line Break

For design purposes, a small steam line break is a break equivalent to a steam generator safety valve opening and remaining open.

For design purposes, it is assumed that this transient occurs five times.

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3.9.1.1.3.3 Complete Loss of Flow

This accident involves a complete loss of flow from full power resulting from the simultaneous loss of power to all reactor coolant pumps. The consequences are a reactor trip on Low-2 pump speed, followed by an automatic turbine trip.

This event is considered to be bounded by the loss of power transient. The frequency of occurrence of loss of power transients incorporates the frequency of occurrence of complete loss of flow accidents.

3.9.1.1.3.4 Small Feedwater Line Break

This transient is postulated as a small break in the piping between the steam generator and the main feedwater isolation valve. The main feedwater control system is assumed to malfunction. The malfunction of the main feedwater flow in the affected loop is equivalent to the fluid spilling through the break. No main feedwater is supplied to either steam generator.

After reactor trip, the main feedwater control system is assumed to be lost and reverse flow is assumed to be initiated from the pipe with the break. During the course of the transient, reactor trip, turbine trip, the passive core cooling system and the startup feedwater system are actuated because of low level in the steam generator.

For design purposes, this transient is assumed to occur five times.

3.9.1.1.3.5 Steam Generator Tube Rupture

This transient is postulated as the double-ended rupture of a single steam generator tube, which results in decreases in pressurizer level and reactor coolant pressure. Assuming no operator action, the reactor eventually trips on overtemperature ΔT or Low-2 pressurizer pressure. Reactor trip initiates a turbine trip. Reactor coolant system pressure continues to decrease after the trip because of energy transfer from the primary system to the secondary side and continued primary to secondary leakage through the ruptured steam generator tube. Continued reactor coolant system leakage results in an actuation of the passive core cooling system because of Low-2 pressurizer level or pressure.

For design purposes this transient is assumed to occur five times.

3.9.1.1.3.6 Inadvertent Opening of Automatic Depressurization System Valves

Rapid depressurization of the reactor coolant system results from the inadvertent opening of the automatic depressurization system (ADS) valves. This event is postulated to occur by:

- Inadvertent opening of an ADS flowpath through two 4-inch (ADS Stage 1) motor-operated valves connected to the pressurizer. Inadvertent opening of an ADS flowpath through two 8-inch valves (ADS Stage 2 or ADS Stage 3), prior to ADS Stage 1 actuation, is not a credible design transient. Inadvertent opening of the 14-inch (ADS Stage 4) valves connected to the reactor coolant system hot legs is not possible at normal operating pressure. The inadvertent opening of the ADS Stage 1 - 4 valves out of sequence is prevented by the Protection and Safety Monitoring System (PMS) logic for a controlled, staged opening (Figure 7.2-1, Sheet 15) and inadvertent opening of ADS valves individually or in any sequence is prevented by the PMS ADS blocking device described in Subsection 7.3.1.2.4.1 and Figure 7.2-1, Sheet 19.

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- Inadvertent automatic depressurization system actuation due to a spurious system level signal.

For design purposes, 15 occurrences of the system level inadvertent opening of automatic depressurization system valves transient are assumed, which bounds the inadvertent opening of a single 4-inch flowpath.

3.9.1.1.4 Level D Service Condition (Faulted Conditions)

The following paragraphs discuss the reactor coolant system faulted condition transients considered to be plant condition PC-5 per ANS/ANSI 51.1. A list of these transients follows. These transients are analyzed using Level D service limits and are not analyzed for fatigue due to cyclic loads. See Subsection 3.6.2 for a discussion of the analysis of mechanical loads due to pipe break.

The components are not evaluated for the dynamic effects of pipe rupture for the pipe break events when the requirements for mechanistic pipe break have been satisfied for the connecting piping. See Subsection 3.6.3 for a discussion of the leak-before-break requirements for mechanistic pipe break. The maximum fluid pressure on components is evaluated for these events when leak-before-break requirements are satisfied.

- Reactor coolant pipe break (large loss-of-coolant accident)
- Large steam line break
- Large feedwater line break
- Reactor coolant pump locked rotor
- Control rod ejection

Each of these accidents is evaluated for one occurrence only.

3.9.1.1.4.1 Reactor Coolant Pipe Break (Large Loss-of-Coolant Accident)

Following a rupture of a reactor coolant pipe or connecting branch line that results in a large loss of coolant, the primary system pressure decreases rapidly. This rapid decrease causes the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will remain at or near operating temperature during blowdown. The passive safety injection system is actuated to introduce water into the reactor coolant system (reactor vessel). The RCS thermal hydraulic response to this transient is developed assuming injection occurs at a maximum initial temperature of 120°F, while the component analyses account for stress associated with injection at a minimum temperature of 50°F. The safeguards actuation signal also trips the reactor and the turbine.

3.9.1.1.4.2 Large Steam Line Break

This transient is based on a double-ended rupture of a main steam line. The analyses performed are based on the following conservative assumptions:

- The plant is initially at no-load condition.
- The steam line break results in an immediate reactor trip.
- Main steam line isolation valves are initially open.
- The passive core cooling system operates as designed, and no single failures are assumed. This maximizes the extent and rate of plant cooldown.

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- Reactor coolant pumps continue to operate until tripped on core makeup tank actuation coincident with Low-2 pressurizer water level.

An alternate definition of large steam break if postulated for evaluation of steam generator pressure boundary components, with respect to stress levels in the steam generator tubes and tubesheet, may represent a more severe transient. The alternate definition is as follows: If the break should occur while the plant is operating at full power instead of no load, and the break is located outside of containment, the affected steam generator will quickly blow down to atmospheric pressure. Flow through the startup feedwater nozzle is then delivered to the hot, dry shell side of the affected steam generator. The primary side pressurizes to 2600 psia (set pressure of pressurizer safety valves plus one percent set pressure error plus 3 percent accumulation). This results in a large differential pressure across the tubes and tubesheet. The combination of parameters giving the most conservative results is used.

The simultaneous, complete severance of both a main steam line and a feedwater line is not a credible event in the AP1000. In addition to the application of criteria to demonstrate leak-before-break on these lines, layout and support requirements are imposed to prevent extensive steam line or feedwater line displacement following rupture.

3.9.1.1.4.3 Large Feedwater Line Break

This postulated accident involves the double-ended rupture of a main feedwater line, which results in rapid blowdown of the affected steam generator and termination of feedwater flow to the other. The plant is assumed to be operating at an initial power level of 102 percent of design rating, with temperatures conservatively higher than the nominal full power values when the break occurs. The feedwater line break results in immediate reactor and turbine trips. The passive core cooling system is actuated, the passive residual heat removal heat exchanger operates, and the reactor coolant pumps are tripped.

In the analysis, no credit is taken for operation of pressure control systems, steam dump, or steam generator power-operated relief valves. The intact steam generator feeds the break through the main steam header after the faulted steam generator discharges its liquid inventory. Steam flow continues until the main steam lines are isolated on Low-2 steam line pressure.

3.9.1.1.4.4 Reactor Coolant Pump Locked Rotor

This accident is based on the seizure of the rotating assembly of a reactor coolant pump rotor, with the plant operating at full power. Reactor trip occurs rapidly, as the result of Low-2 coolant flow in the affected cold leg. Assumptions made in the analysis include the following:

- Initially the plant is operating at 102 percent of design rating.
- Initially temperature is conservatively higher than the program value.
- No return to criticality occurs in the core.
- No credit is taken for reactor coolant system pressure control.

For the determination of the increase in pressure and response of the reactor core to the reduction in flow, the seizure is assumed to occur instantaneously. For the evaluation of dynamic effects imposed on the pump casing, steam generator, and connecting piping, the rotating assembly is assumed to come to a stop rapidly but not instantaneously. See Subsection 5.4.1 for a discussion of the time for a locked rotor to occur.

Level D pressure limits are applied to the affected reactor coolant pump, steam generator channel head and piping, and Level B pressure limits are applied to the rest of the reactor coolant system.

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The system effects and the maximum fluid pressure are evaluated for this condition on components not affected by the dynamic effects.

3.9.1.1.4.5 Control Rod Ejection

This accident is based on the single most reactive control rod being instantaneously ejected from the core. This reactivity insertion in a particular region of the core causes a severe pressure increase in the reactor coolant system in such a way that the pressurizer safety valves will lift. It also causes a more severe temperature transient in the loop associated with the affected region (the hot loop) than in the other loop.

Since the pressure boundary of the control rod drive mechanism is constructed using the requirements of the ASME Code, Section III, the ejection of the control rod is postulated as a nonmechanistic event and not as the result of a rupture of the control rod drive housing. The analysis of the system response is based on the reactivity insertion without any mitigating effects (on the pressure transient) of coolant blowdown through the hole in the vessel head above the rod. The maximum fluid pressure on the components is evaluated for this condition.

3.9.1.1.5 Test Conditions Transients

The following paragraphs describe the following reactor coolant system test conditions transients:

- Primary-side hydrostatic test
- Secondary-side hydrostatic test
- Tube leakage test

3.9.1.1.5.1 Primary-Side Hydrostatic Test

The pressure tests covered by this subsection include both shop and field hydrostatic tests that occur as a result of component or system testing. This hydrostatic test is performed at a water temperature compatible with reactor material ductility requirements and a test pressure of 3107 psig (1.25 times design pressure). In this test, the reactor coolant system is pressurized to 3107 psig coincident with steam generator secondary-side pressure of zero psig. The reactor coolant system is designed for 10 cycles of these hydrostatic tests. The number of cycles is independent of other operating transients.

Additional, lower-pressure hydrostatic tests may be performed to meet the inservice inspection requirements of ASME Code, Section XI, Subarticle IWB-5200. Four such tests are expected. The increase in the fatigue usage factor caused by these tests is covered by the primary-side leakage tests that are considered for design. No additional specification is required.

3.9.1.1.5.2 Secondary-Side Hydrostatic Test

The secondary side of the steam generator is pressurized to 1.25 design pressure, with a minimum water temperature of 70°F. For design purposes it is assumed that the steam generator will experience 10 cycles of this test. These hydrostatic test cycles are considered in the stress and fatigue analyses.

These tests may be performed either before plant startup or after major repairs or both. The number of cycles is independent of other operating transients.

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3.9.1.1.5.3 Tube Leakage Test

It may be necessary to check the steam generator for tube leakage and tube-to-tubesheet leakage. This is done by inspecting the underside (channel-head side) of the tubesheet for water leakage, with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdown.

For these tests, the secondary side of the steam generator is pressurized with water, initially at a relatively low pressure, and the primary system remains depressurized. The underside of the tubesheet is examined for leaks. If any are observed, the secondary side is depressurized and the leaking tube is plugged. The secondary side is then repressurized (to a higher pressure), and the underside of the tubesheet is again checked for leaks. This process is repeated until the leaks are repaired. The maximum (final) secondary-side test pressure reached is 840 psig.

The total number of tube leakage test cycles is defined as 800 for design purposes. The following is a breakdown of the anticipated number of occurrences at each secondary side test pressure:

Test Pressure (psig)	Number of Occurrences
200	400
400	200
600	120
840	80

During these tests, both the primary and the secondary sides of the steam generators are at ambient temperatures. Neither the primary-side nor secondary-side design pressure is exceeded. The expected secondary-to-primary pressure differential exceeds the design value of 670 psi for some of the test cycles.

3.9.1.2 Computer Programs Used in Analyses

A number of computer programs that are used in the dynamic and static analyses of mechanical loads, stresses, and deformations, and in the hydraulic transient load analyses, of seismic Category I components and supports are listed in Table 3.9-15. [*The Combined License applicant will implement the NRC benchmark program using AP1000 specific problems (Reference 20) if a piping analysis computer program other than those used for design certification (PIPESTRESS, GAPPIPE, WECAN, and ANSYS) is used.*]*

The development process, verification, validation, configuration control and error reporting and resolution for computer programs used in these analyses for the AP1000 are completed in compliance with an established quality assurance program. The quality assurance program is described in Chapter 17. The verification conforms to at least one of the following methods:

- Hand calculations
- Alternate verified calculational methods
- Results of other verified programs
- Results obtained from experiments and tests

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- Known solutions for similar or standard problems
- Measured and documented plant data
- Confirmed published data and correlations
- Results of standard programs and benchmarks
- Parametric sensitivity analysis
- Reference to a verification and validation that has been reviewed and accepted by an independent third party

3.9.1.3 Experimental Stress Analysis

For the reactor internals, measured results from prototype plants and various scale model tests are used to validate the analysis of vibrations of reactor vessel internals as discussed in Subsection 3.9.2. [*No other experimental stress analysis is used for the AP1000.*]*

3.9.1.4 Considerations for the Evaluation of the Faulted Conditions

Subsection 3.9.3 describes the analytical methods used to evaluate ASME Code, Section III, Class 1 components for Service Level D Conditions (faulted conditions).

3.9.1.5 Module Interaction, Coupling, and Other Issues

Many portions of the systems for the AP1000 are assembled as modules and shipped to the plant as completed or partially completed units. The following provides a discussion of influence of modularization on the structural analysis, inservice inspection, and maintenance in the AP1000.

The modules are constructed using a structural steel framework to support the equipment, pipe, and pipe supports in the module. Piping in the modules is routed and analyzed in the same manner as in a plant built by traditional methods. See Subsection 3.7.3 for additional discussion of the structural analysis of modules.

The modules are designed and engineered to provide access for inservice inspection and maintenance activities. Field run pipes and equipment supports do not hinder access for maintenance and inspection.

The quality assurance requirements for the installation and welding of components, piping, supports, and structural elements are the same as in a plant built by traditional methods. The improved access to the parts of the module during fabrication enhances inspection.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

A pre-operational test program as described in Section 14.2 is implemented as required by NB-3622.3, NC-3622, and ND-3622 of the ASME Code, Section III to verify that the piping and piping restraints will withstand dynamic effects due to transients, such as pump trips and valve trips, and that piping vibrations are within acceptable levels. The piping systems to be tested include ASME Code, Section III, Class 1, 2, and 3 systems, high energy systems inside seismic Category I structures, high energy portions of systems whose failure could reduce the functioning of seismic Category I features to an unacceptable level, and the seismic Category I portions of moderate-

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energy piping systems located outside containment. This includes ASME instrumentation lines up to the first support in each of three orthogonal directions from the process pipe or equipment connection point.

The pre-operational test program for the ASME Code, Section III, Class 1, 2, and 3, and other high-energy or seismic Category I piping systems simulates actual operating modes to demonstrate that the components comprising these systems meet functional design requirements and that piping vibrations are within acceptable levels. The pre-operational testing programs are outlined in Subsection 14.2.9. Piping systems are checked in three sequential steps or series of tests and inspections.

Construction acceptance, the first step, entails inspection of components for correct installation. During this phase, pipe and equipment supports are checked for correct assembly and setting. The cold locations of reactor coolant system components, such as steam generators and reactor coolant pumps, are recorded.

During the second step of testing, plant heatup, the plant is heated to normal operating temperatures. During the heatup, systems are observed periodically to verify proper expansion. Expansion data is recorded at the end of heatup.

During the third step of testing, performance testing, systems are operated to check the performance of critical pumps, valves, controls, and auxiliary equipment. This phase of testing includes transient tests performed as outlined in Chapter 14. During this phase of testing, the piping and piping restraints are observed for vibration and expansion response. Automatic safety devices, control devices, and other major equipment are observed for indications of overstress, excess vibration, overheating, and noise. Each system test includes critical valve operation during transient system modes.

The locations in the piping system selected for observation during the testing, and the respective acceptance standards, are provided in the preoperational vibration, thermal expansion, and dynamic effects test program plan.

Provisions are made to verify the operability of essential snubbers by recording hot and cold positions. If vibration during testing exceeds the acceptance standard, corrective measures are taken and the test is performed again to demonstrate adequacy.

3.9.2.1.1 Piping Vibration Details

Piping vibration loadings can be placed in two categories: transient-induced vibrations and steady-state vibrations. The first is a dynamic system response to a transient, time-dependent forcing function, such as fast valve closure. The second is a constant vibration, usually flow-induced. Piping vibration testing and assessment is performed in accordance with ASME OM-S/G (Reference 39), Part 3.

Transient Response

Dynamic events falling in this category are anticipated operational occurrences. The systems and the transients included in the preoperational test program are outlined in Section 14.2.

For those types of transients where a time-dependent dynamic analysis is performed on the system, the stresses obtained are combined with system stresses resulting from other operating conditions in according to the criteria identified in Subsection 3.9.3.

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Details of the program and the pipe monitoring displacement transducers or scratch plates and strain gage or load cells locations, including the criteria for evaluation of data gained, are provided in the test procedures.

Steady-State Vibration

System vibrations resulting from flow disturbances are considered steady-state vibration. Since the exact nature of the flow disturbance is not known prior to pump operation, no analysis is performed. If system vibration is evident during initial operation, the maximum amplitudes are measured and related to alternating stress intensity levels based on the guidance of ASME OM-S/G (Reference 39), Part 3.

In some locations, vibration monitors are installed to provide additional indication to the operator of the vibration levels on the piping system and supports. The information provided by the monitoring prevents unnecessary fatigue accumulation in the piping system.

The AP1000 preoperational vibration monitoring program includes appropriate safety-related instrument lines up to the first support in each of three orthogonal directions from the process pipe or equipment connection point. The acceptance standard is that the maximum alternating stress intensity, S_{alt}, calculated from the measured amplitudes, shall be limited as defined in the following:

A. For ASME Class 1 piping systems:

$$S_{alt} = \frac{C_2 K_2}{Z} M \leq \frac{S_{el}}{\propto}$$

where:

C_2 = secondary stress index as defined in the ASME Code

$\propto$ = allowable stress reduction factor: 1.3 for materials covered by Figure I-9.1; or 1.0 for materials covered by Figure I-9.2.1 or I-9.2.2 of the ASME Code, Section III, Appendices

K_2 = local stress index as defined in the ASME Code

M = maximum zero to peak dynamic moment loading due to vibration only, or in combination with other loads as required by the system design specification

S_{el} = $0.8\ S_A$, where S_A is the alternating stress at 10^6 cycles from Figure I-9.1; or S_A at 10^{11} cycles from Figure I-9.2.2 of the ASME Code. The influence of temperature on the modulus of elasticity is considered.

Z = section modulus of the pipe

B. For ASME Class 2 and 3 or ASME B31.1 piping:

$$S_{alt} = \frac{C_2 K_2}{Z} M \leq \frac{S_{el}}{\propto}$$

where:

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C_2K_2 = 2i

i = stress intensification factor, as defined in subsection NC and ND of the ASME Code or in ASME B31.1.

If significant vibration levels are detected during the test program that have not been previously considered in the piping system analysis, consideration is given to modifying the design specification to re-verify applicable code conformance using the measured vibration as input.

If required, additional restraints are provided to reduce stresses to below the acceptance levels.

3.9.2.1.2 Piping Thermal Expansion Program

The piping thermal expansion testing program verifies that the piping systems expand within acceptable limits during heatup and cooldown. Also, this program verifies that the standard component supports (including spring hangers, snubbers, and struts) can accommodate the expansion of the piping within an acceptable range for required modes of operation. Test specifications for thermal expansion testing of piping systems during preoperational and start-up testing will be in accordance with ASME OM-S/G (Reference 39), Part 7.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

Safety-related mechanical equipment and supports are tested or analyzed to demonstrate functional integrity during and following a postulated safe shutdown earthquake. Equipment that must be active to shut down the plant or mitigate the effects of postulated accidents is analyzed or tested to verify operability. The operability requirements for active valves are discussed more fully in Subsection 3.9.3.2.

Section 3.2 lists the equipment classification and seismic category for components and equipment in the AP1000. Table 3.9-12 lists the active valves. The AP1000 has no safety-related active pumps or turbines.

Seismic Category I safety-related equipment is shown to have structural integrity by analysis satisfying the stress criteria applicable to the particular piece of equipment or by tests showing that the equipment retains its structural integrity under the simulated test environment.

Analyses used to verify functional integrity demonstrate that stresses do not exceed the allowables specified for the appropriate loading combinations listed in Subsection 3.9.3. Deformations do not exceed those that permit the component to perform its required safety function.

Subsection 3.7.3 describes the methods for seismic subsystem analysis.

Tests used to verify operability demonstrate that the component is not prevented from performing its required safety function during and after the test.

The testing procedures used in the seismic qualification of instrumentation and electrical equipment are discussed in Section 3.10. The operability of active valves includes the operability of the valve operator. Valves and operators are tested for operability as an assembly. Section 3.10 includes a description of operability testing for ASME Code Classes 1, 2, and 3 valves and respective drives, operators, and vital auxiliary equipment. That section includes a description of the seismic operability criteria.

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Dynamic testing, analysis, or a combination of the two may be used to qualify safety-related seismic Category I mechanical equipment for a postulated safe shutdown earthquake. The criteria used to decide whether dynamic testing or analysis is used are as follows:

Analysis without Testing

Structural analysis without testing is used if structural integrity alone can verify the intended design function. Equipment which falls into this category includes:

- Piping
- Ductwork
- Tanks and vessels
- Heat exchangers
- Filters
- Passive valves

Dynamic analysis without testing is used to qualify heavy machinery too large to be tested. For active equipment, it is verified that deformations due to seismic loadings do not cause binding of moving parts to the extent that the component cannot perform its required safety function.

Dynamic Testing

Dynamic testing is used for components with mechanisms that must change position in order to perform the required safety function. Section 3.10 discusses the seismic qualification of electrical equipment and combinations of valves and valve operators. Such components include the following:

- Electric motor valve operators
- Valve position sensors
- Similar appurtenances for other active valves

Combinations of Analysis with Testing

Combinations of analysis, static testing, and dynamic testing are used for seismic qualification of complex valves. Section 3.10 discusses the requirements for these combinations for equipment, which includes the following:

- Main steam and main feedwater isolation valves
- Other active valves

3.9.2.3 Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow-induced excitation are complex and not readily ascertained by analytical means alone. Assessment of vibrational response is done using a combination of analysis and testing. Comparisons of results obtained from reference plant vibration measurement programs have been used to confirm the validity of scale model tests and other prediction methods as well to confirm the adequacy of reference plant internals regarding flow-induced vibration (FIV). The flow-induced vibration assessment was originally documented in WCAP-15949 (Reference 18) during development of the AP1000 Design Control Document. The FIV assessment program for the final reactor internals design is provided in WCAP-17983 (Reference 40) and WCAP-17984 (Reference 41). WCAP-17983 and WCAP-17984 supersede WCAP-15949, and the reference to WCAP-15949 is retained for historical purposes.

Reactor components are excited by flowing coolant, which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area provides the forcing functions to be used in the dynamic analysis of the structures. In view of the complexities of the geometries and

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the random character of the pressure oscillations, a closed form solution of the vibration problem by the integration of the differential equations is not always practical and realistic.

The determination of forcing functions as a direct correlation of pressure oscillations cannot be practically performed independently of the dynamic characteristics of the reactor vessel internals structure. The main objective is to establish the characteristics of the forcing functions that determine the response of the structures.

By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function are deduced. These studies indicate that the most important forcing functions are flow turbulence and pump-related excitation. The relevance of such excitation depends on factors that include the type and location of components and flow conditions.

The effects of these forcing functions have been studied in tests performed on models and reference plants. These effects will be factored into the analysis models used to evaluate flow-induced vibrations in the AP1000 reactor internals.

The structural analysis models include modeling of the gaps between the core barrel and reactor vessel, the core barrel and upper core plate, and the core barrel and core shroud. To ensure that the range of possible gaps between the components is evaluated, time-history analyses are performed with various sets of gaps. The resulting loads are evaluated, and the resulting highest load for each component is used in the structural analysis.

Loads on the reactor vessel internals components result from the reactor coolant pump pulsations. Using a reactor coolant system analysis, the resulting pressure differences across the components are calculated for inclusion in the structural evaluation models. The uncertainties associated with these component loads are considered by using a general design basis in which the reactor coolant pump related pressure pulsations are taken to be coincident with the component natural frequency if the natural frequency is within ±10% of the reactor coolant pump excitation frequency. The calculated maximum forces from this resonance condition are used in the component structural evaluation.

The vibration assessment program for the AP1000 reactor internals determines, prior to testing of the first AP1000, that the internals are not expected to be subject to unacceptable flow-induced vibrations. The assessment is consistent with the guidelines of Regulatory Guide 1.20. Conformance with Regulatory Guide 1.20 is summarized in Subsection 1.9.1.

The reactor vessel internals in the AP1000 are similar in size and overall configurations to the reactor vessel internals in previous Westinghouse-designed three-loop nuclear power plants.

The original reference plant for Westinghouse three-loop plant reactor internals flow-induced vibration is H. B. Robinson. The results of vibrations testing at H. B. Robinson are reported in WCAP-7765-AR (Reference 3).

Successive design changes that have been incorporated into the AP1000 design since the reference plant tests have also been included in previous preoperational plant vibration testing including the following:

- Inverted hat upper internals and 17x17 guide tubes at DOEL 3 and Sequoyah 1
- XL lower core support structure at DOEL 4
- Core shroud at Yonggwang 4
- Neutron panels at Trojan 1

These tests confirmed that the internals behaved as expected and that the vibration levels were within allowable values. The vibration testing for 17x17 fuel internals and inverted hat upper internals

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is reported in WCAP-8766 (Reference 4) and WCAP-8516-P (Reference 5). The vibration testing of three-loop XL type lower core support structure in DOEL 4 is reported in WCAP-10846 (Reference 6). The vibration evaluations of upper and lower internals assemblies for a four-loop XL plant are reported in WCAP-10865 (Reference 7). The vibration testing of the core shroud lower internals design is reported in Reference 13.

The results of the Doel 3 and Doel 4 reactor internals vibration test programs are utilized to support the vibration assessment of the AP1000 reactor internals. The measured responses from Doel 3 and Doel 4 are adjusted to the higher AP1000 flowrate to support the determination of the expected upper internals and lower internals vibration levels respectively. The velocity through the core is approximately the same as that of Doel 4.

The results of the Trojan 1 tests showed that the lower internals vibrations are lower with neutron panels than with a circular thermal shield as reported in WCAP-8766 (Reference 4).

Subsequent operation of numerous plants has further demonstrated the adequacy of the reactor vessel internals regarding flow-induced vibration.

AP1000 includes design features that differ from the design in plants in which the reactor internals have been tested as outlined previously. These design differences include the following:

- The design has four inlet nozzles and two outlet nozzles in a three-loop size reactor vessel with a three-loop size core barrel diameter.
- The AP1000 core barrel overall length is 11 inches longer than that of the standard 3XL design.
- The skirt of the internals support structure is 11-inches longer than the skirt of previous three-loop internals designs.
- The upper support plate has sixty-nine 9.78 inch diameter holes as compared to sixty-one 9.50 inch diameter holes in the previous three-loop design. The plate thickness is identical at 12 inches in both designs.
- The design has a new in-core instrumentation system.
- The structures below the lower core support plate and the height of the lower plenum have been changed. The core barrel restraint elevation is within the radius of the lower head.
- The reactor coolant is moved using a sealless pump instead of a shaft seal pump.
- A flow skirt is included in the reactor vessel lower head.

The vibrations of the upper internals components are well characterized by previous plant testing based on the following: The control assembly guide tubes and support column designs are similar to those in a previously tested plant. Vibratory loads on these components are expected to be comparable to previously tested three-loop plants, because the AP1000 outlet nozzle velocity is approximately the same as in the three-loop plants.

The AP1000 upper internals design is substantially the same as that measured in the Doel 3 plant and 3XL scale model tests. The AP1000 support columns, guide tubes and upper support assembly are nearly identical to the components in the 3XL scale model test, although the AP1000 arrangement includes more guide tubes and support columns. Preliminary consideration indicates

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that the corresponding AP1000 responses will be calculated to be similar to the previous plant responses.

The vibration assessment evaluation will demonstrate that the vibration levels of the AP1000 lower internals are acceptable. Comparison of lower internals design features between the AP1000 and standard 3XL are discussed below.

The core barrel outside diameter, inside diameter, and the reactor vessel inside diameter are approximately the same as the tested three-loop plants. The core barrel length is 11 inches longer (~6%). Although the inlet nozzle and upper downcomer configuration of the AP1000 design differs from that of the 3XL design, the AP1000 reactor coolant inlet velocity is approximately the same as in the three-loop plants. The reactor coolant velocity in the lower plenum is comparable to the lower plenum velocity in the three-loop plants. The coolant velocity in the downcomer annulus between the core barrel and the reactor vessel wall is lower in the AP1000 design than in previous three-loop plants.

The vibrational response of the core barrel was measured during the Doel 4 reactor internals vibration measurement program. The diameter, length, and thickness are nearly identical to the AP1000 core barrel, and both use the single combined lower core support plate and neutron panels.

The core shroud is shorter than the core barrel, has a smaller outer diameter than the core barrel inside diameter, and is more rigidly clamped at its axially supported end, so that it is not expected to have a significant effect on core barrel vibration.

The replacement of the baffle-former structure with the core shroud reduces the stiffness of the lower internals assembly. The AP1000 shell mode amplitudes are estimated to be higher than three-loop core barrel responses based on scaling the measured responses to the AP1000 reduced core barrel stiffness. The AP1000 shell mode amplitudes are expected to be acceptable.

The AP1000 core barrel and core shroud will be instrumented during the pre-operational testing of the first plant to determine the shell mode and beam mode frequencies and amplitudes.

The in-core instrumentation cables are inside the upper internals support columns and are thus shielded from core plenum coolant flows. The instrumentation cables are subjected to fuel assembly outlet nozzle turbulence. This is judged to be not greater than the inlet nozzle turbulence to which in-core instrumentation thimbles in previous plants were subjected.

One of the changes below the lower core support is the addition of a vortex suppressor. The vortex suppressor design is subject to flow-induced vibrations from coolant flows in the core inlet plenum and by motions of the core barrel. The other significant changes below the lower core support plate are the removal of bottom mounted instrumentation and associated guide tubes and the reduction of the plenum height.

The sealless reactor coolant pumps of the AP1000, have a higher rotational speed and the same number of impeller blades as in previous plants. An evaluation of pump-induced loads is included in the vibration assessment. Calculations of the pump induced pulsations acting on the AP1000 reactor internals are based on full-scale AP1000 pump test data.

3.9.2.4 Pre-operational Flow-Induced Vibration Testing of Reactor Internals

The reactor internals are classified as non-prototype, Category I as defined in Regulatory Guide 1.20. Sanmen Unit 1, as the first AP1000 plant, is the valid prototype for the AP1000 reactor internal design. Based on this classification, an instrumented test is not required. The recommendations of Regulatory Guide 1.20 are satisfied by conducting the confirmatory

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preand post-hot functional examination for integrity. Conformance with Regulatory Guide 1.20 is summarized in Subsection 1.9.1.

The reactor internals flow-induced vibration assessment program is documented in WCAP-17983 (Reference 40) and WCAP-17984 (Reference 41), including comparison of the AP1000 plant and operating plants that have undergone vibration measurement programs.

The acceptance criteria for the vibration predictions are established and related to the ASME Code allowables for long term steady-state conditions.

During the hot functional test, the internals are subjected to flow conditions representative of normal operation for a sufficient length of time to generate a cyclic loading of greater than 10^6 cycles on the main structural elements of the internals. In addition, there is some operating time with one, two, or three pumps operating.

Inspection before and after the hot functional test serves to confirm that the structural integrity of the internals with regard to flow-induced vibrations. When no indications of harmful vibrations or signs of abnormal wear are detected and no apparent structural changes take place, the core support structures are considered to be structurally adequate and sound for operation. If such indications are detected, further evaluation is required.

The testing and inspection plan includes features with emphasis on the areas susceptible to FIV, including flow-induced wear. Consistent with Regulatory Guide 1.20, these areas include as a minimum:

- The major load-bearing elements of the reactor internals relied upon to retain the core support structures in position,
- The lateral, vertical, and torsional restraints provided within the vessel,
- The locking and bolting components whose failure could adversely affect the structural integrity of the reactor internals,
- Those surfaces that are known to be or may become contact surfaces during operation,
- Those critical locations on the reactor vessel internal components as identified by the vibration analysis, and
- The interior of the reactor vessel for evidence of loose parts or foreign material.

The expected and acceptable vibration levels and expected natural frequencies will be determined as part of the vibration assessment program. Appropriate acceptance criteria for the inspection of the reactor internals before and after the hot functional testing are defined as part of the vibration assessment program (WCAP-17983, Reference 40).

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The reactor internals analysis for Level D Service condition events considers safe shutdown earthquake seismic events and pipe rupture conditions. Subsection 3.9.3 defines the loads and loading combinations considered.

The standard for acceptability, in regard to mechanical integrity analyses, are that adequate core cooling and core shutdown must be provided. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently,

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the limitations established for the internals are concerned with the deflections and stability of the parts in addition to stress criteria to confirm integrity of the components.

The AP1000 design loads for LOCA conditions are based on the use of mechanistic pipe break criteria (see Subsection 3.6.3).

3.9.2.5.1 Reactor Internals Analysis Methodology

The evaluation of the reactor internals consists of two major steps. The first step is the three-dimensional response of the reactor internals resulting from the seismic and pipe rupture conditions caused by breaks in the pipe that are not qualified by leak-before-break criteria. The breaks evaluated are those which have the greatest dynamic effect on the reactor internals.

The second step of the evaluation is the component stress evaluations. Maximum stresses and displacements under seismic plus pipe rupture conditions are obtained for the reactor internal components and are combined by the square root of the sum of the squares rule. These maximum stresses and displacements are compared to the allowable values for Level D service conditions.

3.9.2.5.1.1 Dynamic Response of Reactor Pressure Vessel System for Postulated Pipe Rupture

The structural analysis of the reactor vessel system for a postulated pipe rupture considers simultaneous application of the time-history loads that could result from the rupture. The mechanical loads are limited to those due to the movement of the fluid through the reactor internals and a small depressurization effect. Because of the application of mechanistic pipe rupture criteria, evaluation of dynamic effects such as cavity pressurization loads, jet impingement loads, and internal hydraulic pressure transients is limited to those pipe breaks which are not excluded by mechanistic pipe break criteria.

The vessel is restrained by reactor vessel supports beneath four of the reactor vessel nozzles and the reactor coolant loop piping. The reactor coolant loop piping is also supported by the steam generator and steam generator supports.

Analysis of the reactor internals for the loads resulting from a postulated pipe rupture is based on the time-history response of the internals to simultaneously applied forcing functions. The forcing functions are defined at points in the system where changes in cross section or direction of flow occur in such a way that differential loads are generated during the transient. The dynamic mechanical analysis can employ the displacement method, lumped parameters, and stiffness matrix formulations. Because of the complexity of the system and the components, finite element stress analysis codes are used to provide information at various points.

A digital computer program modeling the blowdown of coolant out of the break (see WCAP-8708-P-A, Reference 8), has been developed to calculate local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a loss of coolant accident. This program is applied to the subcooled, transition, and saturated two-phase blowdown regimes. The program is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy are solved numerically, using a fixed mesh in both space and time.

Although spatially, one-dimensional conservation laws are used, the code can be applied to describe three-dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surfaces

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(such as in the pressurizer). System losses such as friction, contraction, and expansion, are considered.

The program evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to a program that uses a detailed geometric description in evaluating the loadings on the reactor internals.

Each reactor component for which calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing up the effects of the following:

- Pressure differential across the element
- Flow stagnation on and unrecovered orifice losses across the element
- Friction losses along the element

Input to the code, in addition to the pressure and velocity transients, includes the effective area of each element on which the force acts because of the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe and the larger the pipe, the more severe the imposed loading is on the components. With the application of mechanistic pipe rupture and the determination of leak-before-break characteristics in large diameter pipe, the pressure waves are of small consequence compared with the seismic loads.

3.9.2.5.1.2 Reactor Vessel, Core Support, and Internal Components Modeling

The mathematical model of the reactor pressure vessel (RV) is a three-dimensional, nonlinear, finite element model that represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. This model is known as the Reactor Equipment System Model (RESM). The model is developed using a general purpose finite element computer code. The model consists of four major concentric, structural submodels connected by nonlinear impact elements and hydrodynamic mass. The four major concentric submodels represent the following components:

- Reactor vessel shell, hemispherical domed head, the lower dome, flow skirt, inlet and outlet nozzles, and integrated head package

- Lower internals assembly including: the core barrel (CB), core shroud (CS), lower core support plate (LCSP), secondary core support structure (SCSS) and vortex suppression plate, and CB neutron panels

- Upper internals (UI) assembly consisting of the upper support plate (USP), guide tubes (GT), upper support columns (USC), upper core plate (UCP), and instrumentation grid assembly (IGA)

- “Simplified” fuel assemblies (FA)

Several structural interfaces, for both linear and nonlinear impact elements, and hydrodynamic mass matrices include the following:

- Lower radial keys between the RV and CB

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- RV flange and CB flange
- RV and UI flange
- UI flange and hold-down spring
- Hydrodynamic mass matrices between the RV and CB, and between the CB and CS
- Restraint provided by the support stiffness at the RV four inlet nozzle support pads of the primary loop and ground
- Hot and cold leg loops

Nonlinear impact elements are included for both radial and tangential gaps and stiffnesses at several interface locations. The RV-to-USP flange, USP-flange-to-CB flange (hold-down spring), and CB-flange-to-RV interfaces are modeled as a series of elements that simulate preload, friction, rotational stiffness, and impact stiffness in the joints. Stiffnesses are developed for each interface, while gaps are established by calculating the steady-state hot gap values. Likewise, the CS-FA annulus is simulated using gap elements at various elevations. The locations of gap elements in the AP1000 RESM are as follows:

- RV and CB at the lower radial keys
- CB nozzles and RV outlet nozzles
- RV and CB flanges and the RV and UI flanges.
- UCP-CB alignment plate
- CS-CB alignment plate
- CB-CS annulus and
- CS-FA annulus

The dynamic time history of the RESM for seismic and loss-of-coolant-accident (LOCA) events provide a number of outputs for all of the components that are included in the model. Thus, for example, the impact forces at all the gap elements are one of the key output results provided to the component analysts. Other results include the internal forces and moments, acceleration and displacement time histories for the detailed stress analyses of the core support, and internals components. Head and lower dome motions, core plate, and CS displacement time histories are also provided for evaluation of the CRDMs, SCCS, flow skirt, and fuel assembly (core) components.

3.9.2.5.2 Analytical Methods

The time-history effects of the internals hydraulic loads and loop mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response.

The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports that are combined with other applicable Level D Service condition loads and used to calculate the stresses in the supports.

Also, the reactor vessel displacements are applied as input to the pipe rupture blowdown analysis of the primary loop piping. The resulting loads and stresses in the piping components and supports include both pipe rupture blowdown loads and reactor vessel displacements. Thus, the effect of vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

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The system seismic analysis of the reactor vessel and its internals for the safe shutdown earthquake (SSE) event is performed using the time-history method. The seismic response is combined with the pipe rupture response, as outlined in Subsection 3.9.3, in order to obtain the maximum stresses and deflections.

Reactor internals components are within acceptable stress and deflection limits for the postulated pipe rupture combined with the safe shutdown earthquake condition.

3.9.2.5.3 Control Rod Insertion

During full power plant operation, rod cluster control assemblies and the corresponding drive rod assemblies are held at a fully withdrawn position by their respective control rod drive mechanisms. During certain accident conditions, such as small break loss of coolant accident or a safe shutdown earthquake condition or both, control assemblies are assumed to drop to their fully inserted position. The guide tubes are evaluated to demonstrate the function of the control rods for a break size consistent with use of the leak-before-break criteria.

No credit for the function of the control rods is assumed for large breaks in the safety analyses outlined in Chapter 15. However, for break sizes consistent with use of the leak-before-break criteria, the design of the guide tubes permits control rod insertion at each control rod position.

3.9.2.6 Correlation of Reactor Internals Vibration Tests with the Analytical Results

The results of dynamic analysis of reactor internals have been compared to the results of preoperational testing in reference plants. This comparison verifies that the analytical model used provides appropriate results.

The preoperational vibration test program for the reactor vessel internals of the AP1000 conducted on the first plant, conforms to the intent of the guidelines in Regulatory Guide 1.20 for a comprehensive vibration assessment program. This program includes a correlation of the analysis and test results. This comparison provides additional verification for the analytical model.

3.9.3 ASME Code Classes 1, 2, and 3 Components, Component Supports, and Core Support Structures

Pressure-retaining components, core support structures, and component supports that are safety-related are classified as Class A, B, or C (see Subsection 3.2.2) and are constructed according to the rules of the ASME Code, Section III, Division 1. As noted in Subsection 3.2.2, Classes A, B, and C mechanical components meet the requirements of Code Classes 1, 2, and 3 respectively.

This subsection discusses the application of the ASME Code to safety-related components and core support structures, the operability of pumps and valves, the design and installation criteria for overpressure protection devices, automatic depressurization devices and the requirements for component supports.

Section 3.8 addresses the loads, loading combinations, and stress limits for structures, including containment.

The ASME Code, Section III requires that a design specification be prepared for ASME Class 1, 2, and 3 components. The specification conforms to and is certified to the requirements of ASME Code, Section III. The Code also requires a design report for safety-related components, to demonstrate that the as-built component meets the requirements of the relevant ASME Design Specification and the applicable ASME Code. The design specifications and design reports will be completed as

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discussed in Subsection 3.9.8.2. Design specifications for ASME Class 1, 2, and 3 components and piping are prepared utilizing procedures that meet the ASME Code. The design report includes as-built reconciliation.

The as-built reconciliation includes the evaluation of pipe break dynamic loads, changes in support locations, and construction deviations.

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

The integrity of the pressure boundary of safety-related components is provided by the use of the ASME Code. Using the methods and equations in the ASME Code, stress levels in the components and supports are calculated for various load combinations. These load combinations may include the effects of internal pressure, dead weight of the component and insulation, and fluid, thermal expansion, dynamic loads due to seismic motion, and other loads.

To determine if a design is acceptable for the loading combination, the calculated stress levels are compared to acceptance standards in the ASME Code. The acceptance standards in the ASME Code differ depending on the plant operating modes and loads considered. The ASME Code includes a design limit and four service limits (A, B, C, and D) against which to evaluate design conditions and plant and system operating conditions.

The design transients for the AP1000 are defined in Subsection 3.9.1. The transients are classified into Level A, B, C, and D Service conditions and test conditions, depending on the expected frequency of occurrence and severity. The description of the transients in Subsection 3.9.1 provides the initial plant operating condition and identifies the different component operating conditions. The design transients for Levels A and B are used in the evaluation of cyclic fatigue for the Class 1 components and piping. The effects of seismic events are also included in the evaluation of cyclic fatigue (See Subsection 3.9.3.1.2). Level D and up to 25 strong stress cycles of Level C service conditions are not required by the rules of the ASME Code to be included in the fatigue evaluation.

3.9.3.1.1 Seismic Loads and Combinations Including Seismic Loads

Seismic Category I systems and components, including core support structures, are designed for one occurrence of the safe shutdown earthquake which is evaluated as a Service Level D condition for pressure boundary integrity. In addition, systems and components sensitive to fatigue are evaluated for cyclic motion due to earthquakes smaller than the safe shutdown earthquake. Using analysis methods, these effects are considered by inclusion of seismic events with an amplitude not less than one-third of the safe shutdown earthquake amplitude. The number of cycles is calculated based on IEEE-344-1987 (Reference 21) to provide the equivalent fatigue damage of two full safe shutdown earthquake events with 10 high-stress cycles per event. There are five seismic events with an amplitude equal to one-third of the safe shutdown earthquake response. Each of the one-third safe shutdown earthquake events has 63 high-stress cycles.

ASME Class 1, 2, 3 and CS systems, components and supports are analyzed for the safe shutdown earthquake with other dynamic events. See Tables 3.9-5 and 3.9-8 for load combinations.

The safe shutdown earthquake is analyzed in combination with those operating modes that occur more than 10 percent of the time. Plant conditions combined with safe shutdown earthquake include the following:

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- Normal 100-percent power operation. Material properties are based on those at operating temperatures. Water inventories are based on normal operating levels. The in-containment refueling water storage tank is full, the refueling canal is empty, the spent fuel pool, fuel transfer canal, cask loading pit and cask washdown pit are full, and the passive containment cooling system tank is full.

- The safe shutdown earthquake, which is postulated to occur with the plant at normal 100-percent power operation, is assumed to cause nonsafety-related systems, including ac power sources, to be unavailable. A single active failure in the safety-related systems is also postulated.

- The timing and causal relationships that exist between the safe shutdown earthquake and transients such as valve discharge are considered and the events combined when the safe shutdown earthquake is the cause of the transient condition. For analysis of piping systems, the timing and causal relationships are not used to exclude load combinations. The safe shutdown earthquake duration is assumed to be 30 seconds. Nonseismically analyzed structures and components are assumed to be unavailable at the beginning of the safe shutdown earthquake. A single active component failure is assumed to occur at the time the component would be expected to function after the failure of the nonseismic components and structures.

- Nonsafety-related systems are evaluated to confirm that their failure in an earthquake does not jeopardize plant safety.

- A water source is provided for limited fire protection after occurrence of the safe shutdown earthquake. See Section 9.5 for additional information on fire protection.

The AP1000 is also designed for special combinations of events that are not based on the probability of occurrence but are based on past precedents and regulatory guidelines. These special combinations are treated as load combinations, not event sequences. That is, even though the safe shutdown earthquake event does not occur coincident with another event, the loads were combined to provide additional design margin.

- ASME Code components, supports, and support miscellaneous steel for these components are designed for the safe shutdown earthquake combined by the square root of the sum of the squares method, with short-term dynamic loads due to postulated pipe ruptures. The pipe ruptures included in this combination are those postulated in accordance with Subsections 3.6.1 and 3.6.2, but do not include those postulated in the break exclusion zone for evaluation of spray wetting, flooding, and subcompartment pressurization effects, nor those excluded by application of mechanistic pipe rupture criteria. (See Subsection 3.6.3.) This combination is used for components and supports that are required to mitigate the effects of the postulated pipe rupture.

- The containment boundary is designed for the safe shutdown earthquake in combination with containment design pressure at containment design temperature.

- The polar crane is designed assuming occurrence of the safe shutdown earthquake during handling of a critical load, such as the reactor vessel head.

3.9.3.1.2 Loads for Class 1 Components, Core Support, and Component Supports

The loads used in the analysis of the Class 1 components, core supports, and component supports are described in the following paragraphs. The loads are listed in Table 3.9-3. Additional information on the loads, stress limits and analysis methods for piping is described in Subsection 3.9.3.1.5.

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Pressure loading is identified as either design pressure or operating pressure. [*The design pressure is used in minimum wall thickness calculations in accordance with the ASME Code.*]* The term "operating pressure" is associated with Service Levels A, B, C, and D conditions.

[*A dead-weight analysis is performed to meet ASME Code requirements by applying a load equal to the acceleration due to gravity (1.0g) downward on the piping system and components. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe, insulation, and contained fluid during normal operating conditions.*]*

The analysis of the safe shutdown earthquake loads demonstrates pressure boundary integrity of the Class 1 systems and components. Seismic loads are identified as either seismic inertia loads or seismic anchor motion loads. The seismic inertia loads represent the dynamic portion of the response, and the seismic anchor motion loads represent the static portion. Subsection 3.7.3 describes seismic analysis methods.

Transient dynamic flow and pressure loads resulting from a postulated pipe break are analyzed. Structural consideration of dynamic effects of postulated pipe breaks requires postulation of a finite number of break locations. Section 3.6 defines postulated pipe break locations.

Safety-related piping including the reactor coolant loops, the main steam piping, and reactor coolant system branch lines equal to or larger than six inches nominal pipe size is evaluated with a leak-before-break analysis to verify that there are no locations subject to a sudden, unanticipated rupture of one of these lines. As a result, the piping and components in these systems do not have to be analyzed for the dynamic effects of a break in the pipe when the leak-before-break criteria are satisfied.

The pipe rupture event considered as a loading is the largest pipe that does not satisfy leak-before-break criteria. The leak-before-break analyses use the acceptance standard of the broad scope rule change to General Design Criterion 4 and NUREG 1061, Volume 3. Subsection 3.6.3 outlines the acceptance standard and approach, including application, methodology, and limits.

Transient dynamic loads are also associated with valve opening and closing. The categories associated with valve operation include automatic depressurization system actuation, fast valve closure, relief valve closed system, relief valve open system, and safety valve discharge. Transient dynamic loads include those due to actuation of the explosive device in squib valves. Components and piping are evaluated for the dynamic response to these transient loads. The relief valve open system (sustained) is evaluated as a static load.

In addition to the loads that apply to the ASME Code Class 1 components, additional miscellaneous loads apply to selected components. These loads are evaluated on a case-by-case basis and are not combined with any other Level C or D Service condition. These miscellaneous loads include the following circumstances.

The reactor coolant pump, steam generator channel head, and connected piping are evaluated for a postulated seized rotor event. For this condition, the rotating mass of the reactor coolant pump is assumed to come to a rapid (but not instantaneous) stop and to transfer the angular momentum through the motor enclosure and pump casing to the steam generator nozzle and reactor coolant piping. The stresses calculated for this event are evaluated using Level D limits for the immediately affected components and supports and using Level B limits for components in the other loop.

For additional information on the specification and analysis of locked rotor loads, see the description in the reactor coolant pump information in Subsection 5.4.1.

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The passive residual heat removal heat exchanger is evaluated for hydraulic loads from the discharge of steam from the automatic depressurization system valves through the spargers in the in-containment refueling water storage tank. These loads include pressure pulses from the introduction of steam into the tank and collapse of the steam bubbles and the gross movement of water in the tank. The stresses in the passive residual heat removal heat exchanger calculated for this event are evaluated using Level B stress limits.

For additional information on the specification and analysis of hydraulic loads see the description in the passive residual heat removal heat exchanger information in Subsection 5.4.14.

Portions of the integrated head package that provide seismic restraint for the control rod drive mechanisms also act as part of the load path for the lifting rig function of the integrated head package. These components are designed and evaluated for heavy load lifting.

For additional information on the design and evaluation of the components of the integrated head package in the load path of the lifting rig, see Subsection 3.9.7.

The ASME Code, Section III requires satisfaction of certain requirements relative to design transient conditions for Class 1 components. Subsection 3.9.1.1 summarizes the design transients.

To provide integrity for the reactor coolant system, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operation conditions. Generally, only Level A and B service condition design transients are evaluated in the analysis of cyclic fatigue. Up to 25 stress cycles for Level C service conditions may be excluded from cyclic fatigue analysis in conformance with ASME Code, Section III criteria. Any Level C service conditions which are in excess of the 25-cycle limit are evaluated for the effect on cyclic fatigue using Level B criteria. For the evaluation of cyclic fatigue, the cycles included for seismic events are evaluated using Level B criteria and are not excluded from the fatigue evaluation regardless of the size of the stress range considered. The determination of which transient events are included in the 25-cycle exclusion is made separately for each component and line of piping.

The effects of seismic events on the design of components other than piping are considered in one of the following ways. The effects of seismic events are considered by including 20 full cycles of the maximum safe shutdown earthquake stress range in the fatigue analysis. The seismic contribution to the fatigue evaluation is based on five seismic events with an amplitude of one-third the safe shutdown earthquake and 63 cycles per event. The seismic evaluation of piping components is discussed in Subsection 3.9.3.1.5.

Thermal Stratification, Cycling, and Striping

Thermal stratification, cycling and striping (TASCS) are phenomena that have resulted in pipe cracking at nuclear power plants. As a result of these incidents, the United States Nuclear Regulatory Commission has issued several bulletins, which are discussed below.

Thermal stratification may occur in piping when flowrates are low and adequate mixing of hot and cold fluid layers does not occur. Thermal cycling due to stratification may occur because of leaking valves or plant operation. Thermal striping is a cyclic mechanism caused by instabilities in the hot-cold fluid interface in stratified fluid during relatively steady flow conditions. The design of piping and component nozzles in the AP1000 includes provisions to minimize the potential for and the effects of thermal stratification and cycling. [*Piping and component supports are designed and evaluated for the thermal expansion of the piping resulting from potential stratification modes. The evaluation includes consideration of the information on thermal cycling and thermal stratification included in NRC Bulletins 79-13, 88-08, and 88-11, and other applicable design standards.*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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NRC Bulletin 79-13

Bulletin 79-13 (Reference 16) was issued as a result of a feedwater line cracking incident at Donald C. Cook Unit 2. This bulletin required that inspections of operating plant feedwater lines be performed. This resulted in the discovery of cracks in the feedwater lines of several plants. To provide a uniform approach to address this issue, a Feedwater Line Cracking Owners Group was established. The specific tasks of the Owners Group Program were to evaluate the thermal, hydraulic, structural and environmental conditions which could individually or collectively contribute to feedwater line crack initiation and growth. The Feedwater Line Cracking Owners Group was disbanded in 1981, after the original investigations were completed. The results of this program indicated that the primary cause of the cracking was thermal fatigue loading induced by thermal stratification and high-cycle thermal striping during low flow auxiliary feedwater injection. The mode of failure was concluded to be corrosion fatigue. This information is documented in WCAP-9693 (Reference 17).

The AP1000 steam generators are equipped with separate nozzles for the main feedwater and startup feedwater lines. Analyses of the AP1000 main feedwater nozzles are performed to demonstrate that the applicable requirements of the ASME Section III Code are met. Thermal stratification is prevented in the main feedwater line based on the flowrate limitations within the main feedwater line and the flow control stability for feedwater control. Low feedwater flow duty is provided by the startup feedwater line while higher feedwater flowrates are provided and controlled via the main feedwater line. The switchover from the startup to the main feedwater line occurs above a minimum flowrate to prevent thermal stratification for limiting temperature deviations. Main feedwater control valve positioning during normal operation is the function of the plant control system. The control scheme enhances steam generator level stability and thus reduces potential feedwater thermal stratification resulting from temporary low flow transients

NRC Bulletin 88-08

Bulletin 88-08, Supplement 1, Supplement 2, and Supplement 3 (Reference 12) were issued following the discovery of cracks in unisolable piping at several nuclear power plants. These cracks were attributed to unanalyzed thermal stresses resulting from isolation valve leakage. This bulletin required that utilities: 1) review systems connected to the reactor coolant system to determine whether unisolable sections of piping connected to the reactor coolant system can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves and that were not evaluated in the design analysis of the piping, 2) nondestructively examine the welds, heat-affected zones and high stress locations, including geometric discontinuities and base metal, as appropriate, to provide assurance that there are no existing flaws, and 3) plan and implement a program to provide continuing assurance of piping integrity. This assurance may be provided by designing the system to withstand the stresses from valve leakage, instrumenting the piping to detect adverse temperature distributions and establishing appropriate limits on these temperature distributions, or providing a means that pressure upstream from isolation valves that might leak into the reactor coolant system is monitored and does not exceed reactor coolant system pressure. In addition to leakage into the reactor coolant system, leakage out of the reactor coolant system may also result in adverse thermal stresses as discussed in Supplement 3 of the bulletin.

For adverse stresses from leakage to occur in unisolable piping, three conditions are necessary:

1. A component with the potential for leakage must exist. In most cases, this will be a valve.

2. A pressure differential capable of forcing leakage through the pressure-retaining component must exist. Leakage in unisolable piping sections may be directed toward the reactor coolant system (“inleakage”), or from the reactor coolant system (“outleakage”).

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3. A temperature differential between the unisolable piping section and the leakage source sufficient to produce significant stresses in the event of leakage must exist. For cases involving inleakage, this could result from a cold leakage entering hot sections of unisolable piping. For cases involving outleakage, this could result from hot leakage from the reactor coolant system entering cold sections of unisolable piping.

The criteria used in the evaluation of the AP1000 systems design for susceptibility to adverse stresses from valve leakage are summarized below:

- Single isolation valves can leak, regardless of design except for explosively actuated valves.
- It is generally assumed that two or more closed valves in series are sufficient to limit the amount of leakage to a magnitude which would have a negligible effect on piping integrity.
- Valves which have external operators may leak through the valve seat and packing. In the case of leaking through the packing, additional in-series closed valves may not be beneficial.
- A positive pressure difference should be considered as a possible leak source.
- Cross-leakage is possible between interconnected lines that are attached to different reactor coolant loop pipes and are isolated by single check valves.

[● *Sections of piping systems which have a slope of greater than 45 degrees from the horizontal plane are not subject to thermal stratification, cycling and striping thermal loadings.*

● *Pipe lines, or sections of lines less than or equal to 1-inch nominal size do not require a thermal stratification, cycling and striping evaluation.*]*

The unisolable portions of the following lines connected to the reactor coolant system have been reviewed and are not susceptible to thermal stratification, cycling or striping:

- Core make up lines from the cold legs to the core make up tanks.
- Auxiliary pressurizer spray from the pressurizer spray line to the auxiliary spray check valve.
- Chemical and volume control purification line from the passive residual heat removal line to the charging valve.
- Pressurizer safety valve lines from the pressurizer to the safety valve.
- Pressurizer spray lines from the cold legs to the pressurizer.
- Automatic depressurization Stage 1, 2, and 3 lines from the pressurizer to the depressurization valves.

The unisolable portions of the following lines connected to the reactor coolant system have been reviewed and are determined to be susceptible to thermal stratification, cycling or striping:

- Passive residual heat removal line from the passive residual heat removal heat exchanger to the steam generator channel head.
- Automatic depressurization Stage 4 lines from the hot legs to the Stage 4 depressurization valves.

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- Passive residual heat removal lines from the hot leg to the passive residual heat removal heat exchanger.
- Chemical and volume control purification line from the pressurizer spray line to the letdown valve.
- Normal residual heat removal suction lines from the hot legs to the isolation valves.

Direct vessel injection lines from the reactor vessel nozzle up to the accumulator injection valves, core make up injection valves, in-containment refueling water storage tank injection valves, and normal residual heat removal injection valves.

Analyses of these lines are performed to demonstrate that the applicable requirements of the ASME Section III Code are met. This analysis includes consideration of plant operation and thermal stratification using temperature distributions which are developed from finite element fluid flow and heat transfer analysis.

The final design reports for ASME components, including reconciliation of the as-built piping is discussed in Subsection 3.9.8. This reconciliation includes verification of the thermal cycling and stratification loadings considered in the stress analysis.

NRC Bulletin 88-11

Bulletin 88-11 (Reference 14) was issued after Portland General Electric Company experienced difficulties in setting whip restraint gap sizes on the pressurizer surge line at Trojan plant. The cold gaps were adjusted to design settings several times and were found to be out of specification after each operating cycle. The gap changes were caused by plastic deformation in the surge line piping resulting from excessive thermal loadings. The thermal loadings were determined to be caused by thermal stratification based on monitoring and analysis. Several similar incidents were subsequently discovered in other surge lines, and an industry-wide program to evaluate this phenomena was undertaken by the various PWR owners groups.

The purpose of Bulletin 88-11 was a request to addresses, establish, and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification, and to require addressees to inform the NRC staff of the actions taken to resolve this issue.

The actions requested in the bulletin are discussed below, and the manner in which AP1000 addresses the actions, if required, for surge line stratification:

For all licensees of operating PWRs:

Request 1.

The actions included under this heading are not applicable to the AP1000.

For all applicants for PWR Operating Licenses:

Request 2. a)

Before issuance of the low power license, applicants are requested to demonstrate that the pressurizer surge line meets the applicable design codes and other FSAR and regulatory commitments for the licensed life of the plant. This may be accomplished by performing a plant specific or generic bounding analysis. The analysis should include consideration of thermal stratification and thermal striping to ensure that fatigue and stresses are in compliance with applicable code limits. The analysis and hot functional testing should verify that piping thermal

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deflections result in no adverse consequences, such as contacting the pipe whip restraints. If analysis or test results show Code noncompliance, conduct of all actions specified below is requested.

AP1000 Conformance

Analysis of the AP1000 surge line considers thermal stratification and thermal striping, and demonstrates that the surge line meets applicable code requirements for the licensed life of the plant. Hot functional testing requirements for the AP1000 ensure that piping thermal deflections result in no adverse consequences.

Request 2. b)

Applicants are requested to evaluate operational alternatives or piping modifications needed to reduce fatigue and stresses to acceptable levels.

AP1000 Conformance

Analysis of the AP1000 surge line ensures that stress and fatigue requirements are satisfied, therefore the evaluation of operational alternatives or piping modifications is not required.

Request 2. c)

Applicants are requested to either monitor the surge line for the effects of thermal stratification, beginning with hot functional testing, or obtain data through collective efforts to assess the extent of thermal stratification, thermal striping and piping displacements.

AP1000 Conformance

As part of the Westinghouse Owners Group program on surge line thermal stratification, Westinghouse collected surge line physical design and plant operational data for all domestic Westinghouse PWRs. In addition, Westinghouse collected surge line monitoring data from approximately 30 plants. This experience was used in the development of the AP1000 thermal stratification loadings. As described in the AP1000 Conformance to Request 3 of Bulletin 88-11, monitoring will be performed during hot functional testing and during the first cycle of the first AP1000 plant. This Combined License item is identified in Subsection 3.9.8.5. Subsequent monitoring of the AP1000 surge line is not required.

Request 2. d)

Applicants are requested to update stress and fatigue analyses, as necessary, to ensure Code compliance. The analyses should be completed no later than one year after issuance of the low power license.

AP1000 Conformance

Revision of the stress and fatigue analyses is not required for the AP1000 surge line, since the design analysis considers thermal stratification and thermal striping.

Request 3)

Addressees are requested to generate records to document the development and implementation of the program requested by Items 1 or 2, as well as any subsequent corrective

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actions, and maintain these records in accordance with 10 CFR Part 50, Appendix B and plant procedures.

AP1000 Conformance

AP1000 procedures require documentation and maintenance of records in accordance with 10 CFR Part 50, Appendix B.

A monitoring program will be implemented as discussed in Subsection 3.9.8.5 at the first AP1000 to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters such as pressurizer temperature and level, hot leg temperature, and reactor coolant pump status. Monitoring will be performed during hot functional testing and during the first fuel cycle. The resulting monitoring data will be evaluated to show that it is within the bounds of the analytical temperature distributions and displacements.

PRESSURIZER SURGE LINE MONITORING

General

The pressurizer surge line is monitored at the first AP1000 plant to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters. This monitoring occurs during the hot functional testing and first fuel cycle. The resulting monitoring data is evaluated to verify that the pressurizer surge line is within the bounds of the analytical temperature distributions and displacements.

Subsequent AP1000 plants (after the first AP1000 plant) confirm that the heatup and cooldown procedures are consistent with the pertinent attributes of the first AP1000 plant surge line monitoring. In addition, changes to the heatup and cooldown procedures consider the potential impact on stress and fatigue analyses consistent with the concerns of NRC Bulletin 88-11.

The pressurizer surge line monitoring activities include the following methodology and requirements:

Monitoring Method

The pressurizer surge line pipe wall is instrumented with outside mounted temperature and displacement sensors. The data from this instrumentation is supplemented by data display and processing system data from related process and control parameters.

Locations to be Monitored

In addition to the existing permanent plant temperature instrumentation, temperature and displacement monitoring will be included at critical locations on the surge line. The additional locations utilized for monitoring during the hot functional testing and the first fuel cycle are selected based on the capability to provide effective monitoring.

Data Evaluation

Data evaluation is performed at the completion of the monitoring period (one fuel cycle). The evaluation includes a comparison of the data evaluation results with the thermal profiles and transient loadings defined for the pressurizer surge line, accounting for expected pipe outside wall temperatures. Interim evaluations of the data are performed during the hot functional testing period, up to the start of normal power operation, and again once three months worth of normal operating data has been collected, to identify any unexpected conditions in the pressurizer surge line.

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Other Applications

Thermal stratification in the reactor coolant loops resulting from actuation of passive safety features is evaluated as a design transient. Stratification effects due to both Level B and Level D service conditions are considered. The criteria used in the evaluation of the stress in the loop piping due to stratification is the same as that applicable for other Level B and Level D service conditions.

3.9.3.1.3 ASME Code Class 1 Components and Supports and Class CS Core Support Loading Combinations and Stress Limits

Tables 3.9-5 and 3.9-8 list loading combinations for ASME Class 1 components and component supports and Class CS core support structures. Table 3.9-9 lists the stress limits for these components. Table 3.9-3 lists the loads included in the loading combinations.

The stress limits for Service Level D that allow inelastic deformation are supplemented with the requirements of “Rules for Evaluation of Service Loadings with Level D Service Limits,” Appendix F of ASME Code, Section III. The limits and rules of Appendix F confirm that pressure boundary integrity and core support structural integrity are maintained but do not confirm operability. The limits and rules of Appendix F do not apply to the portion of the component or support in which the failure has been postulated. Subsection 3.9.1 provides a discussion of design transients used in the analysis of cyclic fatigue.

The structural stress analyses performed on the ASME Code Class 1 components and supports and Class CS core support structures consider the loadings specified, as shown in Table 3.9-3. These loads result from thermal expansion, pressure, weight, earthquake, pipe rupture, and plant operational thermal and pressure transients. Dynamic effects of pipe rupture, including the loss of coolant accident, are not included in loading combinations when the leak-before-break criteria are satisfied. The methods and acceptance standard for leak-before-break analyses are described in Subsection 3.6.3.

[*The combination of safe shutdown earthquake plus pipe rupture*]* (those breaks not excluded by mechanistic pipe break criteria) [*loads by square-root-sum-of the squares is considered.*]* This loading combination is evaluated for ASME Code components and piping that are required to mitigate the effects of the postulated pipe rupture and the supports for those components and piping.

The dynamic effects of pipe rupture that are combined with safe shutdown earthquake in loading combinations are those for lines for which the leak-before-break criteria are not satisfied. [*When the safe shutdown earthquake event is determined mechanistically to result in concurrent transient loads due to relief valve or safety valve discharge in ASME Code Class 1, 2, or 3 systems, the maximum response due to the safe shutdown earthquake is combined with the maximum response due to the valve opening discharge transient. The responses are combined using the square-root-sum-of-the-squares method.*]* Concurrent sustained loads due to open system relief valve discharge are combined with safe shutdown earthquake by absolute sum.

3.9.3.1.4 Analysis of Reactor Coolant Loop Piping

The reactor coolant loop and support system model consists of the primary loop piping (hot and cold legs), the connecting components (reactor vessel, steam generator, and reactor coolant pump) and the components supports (steam generator and reactor vessel).

The integrated reactor coolant loop and supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, and the stiffnesses of auxiliary line piping affecting the system. This model is used to

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determine the static and dynamic loads on the primary loop piping and the component supports and the interfacing loads on the connecting components.

The analysis of the connecting components is based on more detailed models of the steam generator, reactor vessel, and reactor coolant pump. Appendix 3C describes the analytical methods used in evaluating the piping of the reactor coolant loops.

[*The primary loop analysis for the safe shutdown earthquake uses the time-history integration or response spectra methods for seismic dynamic analysis.*]* Appendix 3C provides a description of the model.

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. [*In the time-history seismic analysis, the containment internals structure is included in the system coupled model unless appropriate time-history seismic input from the building is provided at multiple support locations.*]* The effect of the equipment motion on the reactor coolant loop and supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The main loop piping and the surge line satisfy the leak-before-break requirements for the elimination of nonmechanistic pipe breaks. See Subsection 3.6.3 for a description of the evaluation of piping for leak-before-break requirements. Reactor coolant system piping of 6-inch nominal pipe size or larger is evaluated for leak-before-break characteristics. The reactor coolant loop piping is evaluated for loads due to a break in the largest connected pipe that does not meet leak-before-break requirements. [*The primary loop analysis for pipe breaks uses time-history integration or equivalent static analysis to determine the structural response due to jet impingement loads, thrust loads, and subcompartment pressure loads.*]*

Operating transients in a nuclear power plant cause thermal or pressure fluctuations or both in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. The transients as summarized in Subsection 3.9.1.1 are used to define the fluctuations in plant parameters.

A one-dimensional finite difference heat transfer program is generally used to solve the thermal transient problem. The pipe is represented by many elements through the thickness of the pipe. The convective heat-transfer coefficient used in this program represents the time-varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic, while the inner surface boundary experiences the temperature of the coolant fluid.

Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness that varies with time. The average through-wall temperature, T_A, is calculated by integrating the temperature distribution across the wall. This integration is performed over each time step so that T_A is determined as a function of time.

A load-set is defined as a set of pressure loads, moment loads, and through-wall thermal effects at a given location and time in each transient. The through-wall thermal effects are functions of time and can be subdivided into four parts:

- Average temperature (T_A), which is the average temperature through-wall of the pipe that contributes to general expansion loads
- Radial linear thermal gradient, which contributes to the through-wall bending moment (ΔT_1)
- Radial nonlinear thermal gradient (ΔT_2), which contributes to a peak stress associated with shearing of the surface

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- Discontinuity temperature (T_A - T_B) which represents the difference in average temperature at the cross sections on each side of a discontinuity

Each transient is described by at least two load-sets representing the maximum and minimum stress states during each transient. The construction of the load-sets is accomplished by combining the following to yield the maximum (minimum) stress state during each transient.

- ΔT_1
- ΔT_2
- $\alpha_A T_A$ - $\alpha_B T_B$
- Moment loads due to T_A
- Pressure loads

This procedure produces at least twice as many load-sets as transients for each point.

For the possible load-set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors (K_e), and cumulative usage factors (U) are calculated.

The combination of load-sets yielding the highest alternating stress intensity range is first used to calculate the incremental usage factor. The next most severe combination is then determined, and the incremental usage factor is calculated. This procedure is repeated until the combinations having an allowable number of cycles less than 10^{11} are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.3.1.5 ASME Classes 1, 2, and 3 Piping

The loads for ASME Code Classes 1, 2, and 3 piping are included in the loads listed in Table 3.9-3. [*Tables 3.9-5, 3.9-6, and 3.9-9 list the loading combinations and stress limits for Class 1 piping. Tables 3.9-5, 3.9-7, and 3.9-10 list the loading combinations and stress limits for Class 2 and 3 piping.*

Piping systems are designed and analyzed for Levels A, B, and C service conditions, and corresponding service level requirements to the rules of the ASME Code, Section III. The analysis or test methods and associated stress or load allowable limits that are used in evaluation of Level D service conditions are those that are defined in Appendix F of the ASME Code, Section III. Inelastic analysis methods are not used.]*

Subsection 3.7.3 summarizes seismic analysis methods and criteria. Subsection 3.6.2 summarizes pipe break analysis methods.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Alternate methods for support stiffnesses representation is provided in Subsection 3.9.3.4. Shock suppressors that resist rapid motions and limit stop supports with gaps are also included in the analysis. The solution for the seismic disturbance uses the response spectra method. This method uses the lumped mass technique, linear elastic properties, and the principle of modal superposition. Alternatively, the time-history method may be used for the solution of the seismic disturbance.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system and the response from differential anchor motions (see Subsection 3.7.3). The stresses resulting from the anchor motions are considered to be secondary and are evaluated to the limits in Tables 3.9-6 and 3.9-7.

The mathematical models used in the seismic analyses of the Class 1, 2, and 3 piping systems lines are also used for pipe rupture effect analysis. To obtain the dynamic solution for auxiliary lines with

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active valves, the time-history deflections from the analysis of the reactor coolant loop are applied at nozzle connections. For other lines that must maintain structural integrity or that have no active valves, the motion of the reactor coolant loop is applied statically.

[*The functional capability requirements for ASME piping systems that must maintain an adequate fluid flow path to mitigate a Level C or Level D plant event are shown in Table 3.9-11.*]* These requirements are based on Reference 19.

Thermal analysis is required to obtain the stresses and loadings above the stress free state of the system. The stress free state of a piping system is defined as a temperature of 70°F. [*If the piping system operating temperature is 150°F or less, no thermal expansion analysis is required. If the piping system does not contain at least one 90-degree bend, then thermal expansion analysis is required.*]* This type of layout is avoided when practical. [*The thermal anchor displacements are also considered as negligible if they are 1/16 inches or less.*]* This is consistent with the practice that 1/16-inch of gap is allowed at a pipe support.

[*A thermal transient heat transfer analysis is performed for each different piping component on the Class 1 branch lines larger than 1-inch nominal diameter.*]* The following discussion on the evaluation of cyclic fatigue is not applicable to Class 2 and 3 pipe.

[*The Level A and B service condition and test condition transients identified in subsection 3.9.1.1 are included in the fatigue evaluation. For each thermal transient, two load-sets are defined representing the maximum and minimum stress states for that transient. The effects of seismic events on the design of piping are considered in one of the following ways. The effects of seismic events are considered by including 20 full cycles of the maximum safe shutdown earthquake stress range in the fatigue analysis. Alternatively, the seismic contribution to the fatigue evaluation is based on five seismic events with an amplitude of one-third the safe shutdown earthquake and 63 cycles per event.*

The primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors, and cumulative usage factors are calculated for the possible load-set combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.]*

The combination of load-sets yielding the highest alternating stress intensity range is determined, and the incremental usage factor is calculated. Likewise, the next most severe combination is then determined, and the incremental usage factor is calculated. This procedure is repeated until the combinations having an allowable cycle of less than 10^{11} are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

The methods and criteria used in the design and analysis of the ASME Code Classes 1, 2, and 3 piping are itemized in Table 3.9-19. These methods are included in the design specifications and design reports required for piping in Subsection 3.9.8.2.

3.9.3.1.6 Analysis of Primary Components and Class 1 Valves and Auxiliary Components

Primary components that serve as part of the pressure boundary in the reactor coolant loop include the steam generators, reactor coolant pumps, pressurizer, and reactor vessel. This equipment is AP1000 Equipment Class A. The pressure boundary meets the requirements of ASME Code, Section III. This equipment is evaluated for the loading combinations outlined in Table 3.9-5. The equipment is analyzed for the normal loads of weight, pressure, and temperature; mechanical transients of safe shutdown earthquake and auxiliary line pipe ruptures; and pressure and temperature transients are outlined in Subsection 3.9.1.1.

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The results of the reactor coolant loop analysis and other ASME Code, Section III, Class 1, 2, and 3 piping analyses are used to determine the seismic loads acting on the equipment nozzles and the support and component interface locations. Subsection 3.7.3 summarizes seismic analysis methods and criteria used for analysis of primary components. The results of the reactor coolant loop analysis, other ASME Code, Section III, Class 1, 2, and 3 piping analyses, and the reactor vessel system analysis are used to determine pipe break loads on the equipment nozzles and the support component interface locations for those lines that do not meet the leak-before-break requirements.

Section 3.6 summarizes the pipe break analysis methods used to determine pipe rupture loads for the ASME Code Class 1 components.

Seismic analyses are performed individually for the reactor coolant pump, pressurizer, and steam generator. Detailed and complex dynamic models are used for the dynamic analyses. Seismic analyses for the steam generator, reactor coolant pump, and pressurizer are performed using 4 percent damping for the safe shutdown earthquake.

The reactor pressure vessel is seismically qualified in accordance with ASME Code, Section III. The loadings used in the analysis are based on loads generated by a dynamic system analysis.

Auxiliary equipment that serves as part of the reactor coolant system pressure boundary includes ASME Code, Section III, Class 1 valves, core makeup tanks, and passive residual heat removal heat exchanger. Components and valves which form part of the reactor coolant system pressure boundary are designed and analyzed according to the appropriate portions of the ASME Code, Section III. This equipment is evaluated for the loading combinations and stress limits in Tables 3.9-5 and 3.9-9. The operability criteria for these valves are described in Subsection 3.9.3.2.

Valves in sample and instrument lines connected to the reactor coolant system are not considered to be AP1000 Equipment Class A nor ASME Class 1. This is because the nozzles where the lines connect to the primary system piping include an orifice with a 3/8-inch hole. This hole restricts the flow so that loss through a severance of one of these lines can be made up by normal charging flow. These small lines are seismically analyzed as described in Subsection 3.7.3.

3.9.3.1.7 ASME Code Class 2 and 3 Components

Table 3.9-3 lists the loads for ASME Code Class 2 and 3 components. Table 3.9-5 provides the loading combinations. The loading conditions for ASME Class 2 and 3 piping are presented in Table 3.9-3. Table 3.9-10 presents the stress limits for the various service levels. Functional capability requirements are presented in Table 3.9-11. Subsection 3.7.3 summarizes the seismic analysis methods and criteria for these components. The pipe break analysis methods are summarized in Subsection 3.6.2. Analysis methods for Class 2 and 3 piping are summarized in Subsection 3.9.3.1.5.

The allowable stress limits established for the components are low enough so that breach of the pressure-retaining boundary does not occur. Active valves requirements are further described in Subsection 3.9.3.2.

3.9.3.2 Pump and Valve Operability Assurance

The design and service limits specified by the ASME Code, Section III are established to confirm the pressure-retaining or support function of the ASME Code-class component. To assess the functional capability of required components to operate, additional criteria and considerations, including collapse and deflection limits, are developed.

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3.9.3.2.1 Pump Operability

There are no active pumps relied upon to perform a safety-related function in the AP1000.

3.9.3.2.2 Valve Operability

Active valves are those whose operability is relied upon to perform a safety-related function during transients or events considered in the respective operating condition categories. Inactive components are those whose operability is not relied upon to perform a safety-related function for the various transients and plant conditions. Table 3.9-12 lists the active valves.

Table 3.9-9 provides the stress limits used for active Class 1 valves. Table 3.9-10 provides the stress limits used for active Class 2 and Class 3 valves.

Active valves are subjected to a series of tests and inspections prior to service and during the plant life. These tests and inspections along with controls on maintenance and operation provide appropriate reliability of the valve for the design life objective of the plant.

Prior to service, qualification of functional capability of active valve assemblies is performed in accordance with the requirements of ASME QME-1-2007 (Reference 36). Prior to installation, the following tests, as appropriate to the function and mission of the valve, are performed: shell hydrostatic test, backseat and main seat leakage tests, disc hydrostatic tests, and operational tests to verify that the valve opens and closes.

Cold hydro tests, hot functional tests, periodic inservice inspections, and periodic inservice operations are performed in situ to verify the functional capability of the valve.

Refer to Section 3.11 for the operability qualification of motor operators for the environmental conditions.

For active valves with extended structures, an analysis of the extended structure is performed for equivalent static seismic safe shutdown earthquake loads applied at the center of gravity of the extended structure.

In addition to these tests and analyses, a representative number of valves of each design type are tested for verification of operability during a simulated Service Level D (safe shutdown earthquake) condition event by demonstrating operational capabilities within the specified limits. Valve sizes that cover the range of sizes in service are tested.

When seismic qualification is based on dynamic or equivalent static load testing for structures, systems or subsystems that contain mechanisms that must change position in order to function, operability testing is performed for the safe shutdown earthquake preceded by one or more earthquakes. The number of preceding earthquakes is calculated based on IEEE-344-1987 to provide the equivalent fatigue damage of one safe shutdown earthquake event.

The seismic qualification testing procedures for valve operability testing are as follows: The valve is mounted in a manner that will conservatively represent typical valve installations. The valve includes the operator, accessory solenoid valves, and position sensors when attached to the valve in service.

The operability of the valve during a Service Level D condition is demonstrated by satisfying the following criteria:

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- A static load or loads equivalent to those resulting from the accelerations due to Service Level D conditions is applied to the extended structure center of gravity so that the resulting deflection is in the nearest direction of the extended structure. The design pressure of the valve is applied to the valve during the static deflection tests.

- The valve is cycled while in the deflected position. The valve must function within the specified operating time limits while subject to design pressure.

- Electrical motor operators, position sensors, and pilot solenoid valves necessary for operation are qualified in accordance with IEEE seismic qualification standards. Section 3.10 describes the methods and criteria used to qualify electrical equipment.

Active valves that do not have an extended structure, such as check valves and safety valves, are considered separately.

Check valves are characteristically simple in design, and their operation is not affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact, and there are no extended structures or masses whose motion could cause distortions that could restrict operation of the valve. These valves are designed such that if structural integrity is maintained, the valve operability is maintained. In addition to these design considerations, the check valves also undergo the following: in-shop hydrostatic test, in-shop seat leakage test, and periodic in situ valve testing and inspection.

Pressurizer and main steam safety valves are qualified for operability in the same manner as valves with extended structures. The qualification methods include analysis of the bonnet for equivalent static safe shutdown earthquake loads, in shop hydrostatic and seat leakage tests, and periodic in situ valve inspection.

To verify analysis methods, representative safety valves are tested. This test is described as follows:

- The safety valve is mounted to represent the specified installation.

- The valve body is pressurized to its normal system pressure.

- A static load representing the Service Level D condition load is applied to the top of the valve bonnet in the weakest direction of the extended structure.

- The pressure is increased until the valve actuates.

- Actuation of the valve at its setpoint provides for operability during the Service Level D condition load.

Using these methods, the active valves in the system are qualified for operability during a Service Level D condition event. These methods conservatively simulate the seismic event, and confirm that the active valves perform their safety-related function when necessary.

3.9.3.3 Design and Installation Criteria of Class 1, 2, and 3 Pressure Relieving Devices

[*The design of pressure relieving valves comply with the requirements of ASME Code, Section III, Appendix O, "Rules for the Design of Safety Valve Installations."*]* When there is more than one valve on the same run of pipe, the sequence of valve openings is based on the anticipated sequence of valve opening. This sequence is determined by the set point pressures or control system logic. The applicable stress limits are satisfied for the components in the piping run and connecting systems

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including supports. The reaction forces and moments are based on a dynamic load factor of 2.0 unless a dynamic structural analysis is performed to calculate these forces and moments.

3.9.3.3.1 Pressure Relief Devices and Automatic Depressurization Valves Connected to the Pressurizer

The pressurizer safety valves provide overpressure protection for the reactor coolant system. The safety valves connected to the pressurizer are the only ASME Code, Section III, Class 1 pressure relief valves in the AP1000. The automatic depressurization system valves that provide a means to reduce reactor coolant system pressure to allow the passive core cooling system to fully function are not designed to provide overpressure protection and are not classified as pressure relief devices.

The safety valves and the first three stages of the automatic depressurization valves are mounted in and supported by the pressurizer safety and relief valve (PSARV) module located above the pressurizer. The valves are connected to two piping manifolds that are connected to two nozzles located in the pressurizer upper head. [*The spring loaded safety valves are designed to prevent system pressure from exceeding design pressure by more than ten percent.*]*

If the pressure exceeds the setpoint of either safety valve, the valve opens and steam is discharged through a rupture disk to the containment atmosphere. Normal pressurizer liquid and vapor volumes combined with operation of the plant control systems limit pressure increases so that opening of the safety valves is not required for any Level A or B service condition transient. The connecting pipe between the pressurizer and safety valves does not include a loop seal. The safety valves seal against the steam and any noncondensible gas in the upper portion of the pressurizer.

The valves for the automatic depressurization system open when required for the passive safety injection system. The motor-operated automatic depressurization valves open in sequence to reduce reactor coolant system pressure when required to allow stored water sources to cool the core. The valves open in stages as required by the controls for the automatic depressurization system. The automatic depressurization system valves open more slowly than do the safety valves. The operation of the automatic depressurization system is outlined in Subsection 5.4.6 and Section 6.3. For the three stages that are connected to the pressurizer, the valves discharge into the in-containment refueling water storage tank through a sparger. The piping connection of the automatic depressurization system valves to the pressurizer contain loop seals.

The valve opening generates transient thrust forces at each change in flow direction or area. The analysis of the piping system and support considers the transient forces associated with valve opening.

For each pressurizer safety and automatic depressurization system piping system, an analytical hydraulic model is developed. The piping from the pressurizer nozzle to the rupture disk and in-containment refueling water storage tank sparger is modeled as a series of single pipes. The pressurizer is modeled as a reservoir that contains steam at constant pressure and at constant temperature. Fluid acceleration inside the pipe generates reaction forces on the segments of the line that are bounded at either end by an elbow or bend. Reaction forces resulting from fluid pressure and momentum variations are calculated. These forces are defined in terms of the fluid properties for the transient hydraulic analysis.

3.9.3.3.2 Pressure Relief Devices for Class 2 Systems and Components

Pressure relieving devices for ASME Code, Section III, Class 2 systems include the safety valves and power operated relief valves on the steam line and the relief valves on the containment isolation portion of the normal residual heat removal system.

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The design and analysis requirements for the safety and relief valves and discharge piping for the steam line are described in Subsection 10.3.2.

In addition to providing overpressure protection for the normal residual heat removal system, the relief valves also provide low temperature overpressure protection for the reactor coolant system. The location and connection for the valves on the residual heat removal system are discussed in Subsection 5.4.7.

3.9.3.3.3 Design and Analysis Requirements for Pressure Relieving Devices

The design of pressure-relieving devices can be generally grouped in two categories: open discharge and closed discharge.

Open Discharge

An open discharge is characterized by a relief or safety valve discharging to the atmosphere or to a vent stack open to the atmosphere. [*The design and analysis of open discharge valve stations includes the following considerations:*

- *Stresses in the valve header, the valve inlet piping, and local stresses in the header-to-valve inlet piping junction due to thermal effects, internal pressure, seismic loads, and thrust loads are considered.*
- *Thrust forces include both pressure and momentum effects.*
- *Where more than one safety or relief valve is installed on the same pipe run, valve spacing requirements are as specified in the ASME Code. The minimum moments to be used in stress calculations are those specified in the ASME Code.*
- *The effects of the valve discharge on piping connected to the valve header are considered.*
- *The reaction forces and moments used in stress calculations include the effects of a dynamic load factor (DLF), or are the maximum instantaneous values obtained from a time-history structural analysis.*]*

Closed Discharge

The closed discharge system is characterized by piping between the valve and a tank or some other terminal end. Under steady-state conditions, there are no net unbalanced forces. [*The initial transient response and resulting stresses are determined using either a time-history computer solution*]* or a conservative equivalent static solution. [*In calculating initial transient forces, pressure and momentum terms as well as water slug effects are included.*]*

3.9.3.4 Component and Piping Supports

[*The supports for ASME Code, Section III, Class 1, 2, and 3 components including pipe supports satisfy the requirements of the ASME Code, Section III, Subsection NF.*]* In addition to meeting the requirements of ASME Code, Section III, Subsection NF, the welded connections of ASTM A500 Grade B tube steel members (identified as hollow structural sections [HSSs]) satisfy the requirements of the AISC “Load and Resistance Factor Design Specification for Steel Hollow Structural Sections,” November 10, 2000. The nominal strength of the connections, as provided in Sections 8 and 9 of the AISC “Load and Resistance Factor Design Specification for Steel Hollow Structural Sections,” November 10, 2000, will be divided by a safety factor to generate the allowable stress design (ASD) connection capacity. [*The boundary between the supports and the building structure is based on the rules found in Subsection NF.*]* Table 3.9-3 presents the loading conditions.

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[*Table 3.9-8 summarizes the load combinations. The stress limits are presented in Tables 3.9-9 and 3.9-10 for the various service levels.*]*

The criteria of Appendix F of the ASME Code Section III is used for the evaluation of Level D service conditions. When supports for components not built to ASME Code, Section III criteria are evaluated for the effect of Level D service conditions, the allowable stress levels are based on tests or accepted industry standards comparable to those in Appendix F of ASME Code, Section III.

In order to provide for operability of active equipment, including valves, ASME limits for Service Level C loadings are met for the supports of these items.

Dynamic loads for components loaded in the elastic range are calculated using dynamic load factors, time-history analysis, or any other method that accounts for elastic behavior of the component. A component is assumed to be in the elastic range if yielding across a section does not occur. Local yielding due to stress concentration is assumed not to affect the validity of the assumptions of elastic behavior. The stress allowables of Appendix F for elastically analyzed components are used for Code components. Inelastic stress analysis is not used.

The stiffness of the pipe support miscellaneous steel is controlled by one of the following methods so that component nozzle loads are not adversely affected by support deformation:

[*Pipe support miscellaneous steel deflections are limited for dynamic combined faulted loading to 1/8 inch in each restrained direction. The dynamic faulted loading combinations considered are those in Table 3.9-8 associated with Level D service limits.*]* These deflections are defined with respect to the structure to which the miscellaneous steel is attached. These deflection limits, provide adequate stiffness for seismic analysis and are small enough so that nozzle loads are not affected by pipe support deformation. In this case, the pipe support and miscellaneous steel are represented by the following stiffness values in the piping system analysis. Default rigid stiffness values are used for supports except that vendor stiffness values are used for snubbers and rigid gapped supports. [*The mass of the pipe support miscellaneous steel is evaluated as a self-weight excitation loading on the steel and the structures supporting the steel.*]*

[*Alternatively, if the deflections for dynamic loading exceeds 1/8 inches, the pipe support and miscellaneous steel are represented by calculated stiffness values in the piping system analysis.*]*

Use of baseplates with concrete expansion anchors is minimized in the AP1000. Concrete expansion anchors may be used for pipe supports. For these pipe support baseplate designs, the baseplate flexibility requirements of IE Bulletin 79-02, Revision 2, dated November 8, 1979 are met by accounting for the baseplate flexibility in the calculation of anchor bolt loads. Supplemental requirements for fastening anchor bolts to concrete are outlined in Subsection 3.8.4.5.1.

[*Friction forces induced by the pipe on the support must be considered in the analysis of sliding type supports, such as guides or box supports, when the resultant unrestrained thermal motion is greater than 1/16 inch. The friction force is equal to the coefficient of friction times the pipe load, and acts in the direction of pipe movement. A coefficient of friction of 0.35 for steel-on-steel sliding surfaces shall be used. If a self-lubricated bearing plate is used, a 0.15 coefficient of friction shall be used. The pipe load from which the friction force is developed includes only deadweight and thermal loads. The friction force can not be greater than the product of the pipe movement and the stiffness of the pipe support in the direction of movement.*]*

Small gaps are provided for frame type supports built around the pipe. These gaps allow for radial thermal expansion of the pipe as well as allowing for pipe rotation. The minimum gap (total of opposing sides) between the pipe and the support is equal to the diametral expansion of the pipe due

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to temperature and pressure. [*The maximum gap is equal to the diametral expansion of the pipe due to temperature and pressure plus 1/8 inch.*]*

For standard component pipe supports, the manufacturer's functional limitations for example, travel limits and sway angles, should be followed. This criterion is applicable to limit stops, snubbers, rods, hangers and sway struts. Snubber settings should be chosen such that pipe movement occurs over the mid range of the snubber travel. Some margin should be provided between the expected pipe movement and the maximum or minimum snubber-stroke to accommodate construction tolerance.

3.9.3.4.1 ASME Code Class 1 Component Supports

The load combinations and allowable stresses for ASME Code Class 1 component supports are given in Tables 3.9-8 and 3.9-9.

3.9.3.4.1.1 Class 1 Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped-parameter, multimass structural models. The equipment support structure models are dual-purpose, since they represent quantitatively the elastic restraints that the supports impose upon the component, and represent the individual support member stresses due to the forces imposed upon the supports by the component.

A description of the supports for the reactor pressure vessel, steam generator, and pressurizer is found in Subsection 5.4.10. The supports are modeled using elements such as beams, plates, and springs where applicable.

The reactor vessel supports are located at each of the four inlet nozzles and are modeled using a finite element computer program.

Steam generator supports include a column support below the steam generator, a lateral support attached to the top of the column support, a lateral support transverse to the hot leg attached to the secondary shell at the operating floor, and a lateral support (snubbers) parallel to the hot leg attached to the secondary shell at the top of the steam generator compartment, and are normally modeled as linear or nonlinear springs. The reactor coolant pump is supported by the connection to the steam generator and does not have separate supports.

The pressurizer is supported by four columns. Each core makeup tank is supported by eight columns.

The passive residual heat removal heat exchanger is supported by the in-containment refueling water storage tank. The channel heads are outside of the tank and the tubesheets are connected to the tank wall. The tubes are inside the tank, exposed to fluid motion and supported by a structure resting on the floor of the tank and attached to the tank wall.

For each operating condition, the loads (obtained from the reactor coolant loop analysis or the analysis of the component) acting on the reactor pressure vessel, steam generator, and pressurizer supports are appropriately combined. The adequacy of each member of the supports, is verified by solving the stress and interaction equations of ASME Code, Section III, Subsection NF and Appendix F. The adequacy of the reactor pressure vessel support structure is verified using a finite element computer program and comparing the resultant stresses to the criteria given in ASME Code, Section III, Subsection NF and Appendix F.

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The test load method given in Appendix F is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis. The test load method is not used in the AP1000 to qualify supports of components built to ASME Code, Section III requirements.

3.9.3.4.2 ASME Code Class 2 and 3 Supports

[*Class 2 and 3 component supports are designed and analyzed for design condition, and Level A, B, C, and D service conditions to the rules and requirements of ASME Section III, Subsection NF, and Appendix F.*]* The analyses or test methods and associated stress or load allowable limits that are used in the evaluation of linear supports for Level D service conditions are those defined in the ASME Code. Plate and shell type supports satisfy the Level D service condition limits provided in Appendix F of the ASME Code, Section III. Tables 3.9-8 and 3.9-10 outline the allowable stresses and loading combinations for ASME Code, Section III, Class 2 and 3 component supports.

3.9.3.4.3 Snubbers Used as Component and Piping Supports

The location and size of the snubbers are determined by stress analysis. Access for the testing, inspection, and maintenance of snubbers is considered in the AP1000 layout. The location and line of action of a snubber are selected based on the necessity of limiting seismic stresses in the piping and nozzle loads on equipment. Snubbers are chosen in lieu of rigid supports where restricting thermal growth would induce excessive thermal stresses in the piping or nozzle loads or equipment. Snubbers that are designed to lock up at a given velocity are specified with lock-up velocities sufficiently large to envelope the highest thermal growth rates of the pipe or equipment for design thermal transients. The snubbers are constructed to ASME Code, Section III, Subsection NF standards.

[*In the piping system seismic stress analysis, the snubbers are modeled as stiffness elements. The stiffness value is based on vendor stiffness data for the snubber, snubber extension, and pipe clamp assembly.*]* Supports for active valves are included in the overall design and qualification of the valve.

The elimination of the analysis of dynamic effects of pipe breaks due to leak-before-break considerations, as outlined in Subsection 3.6.3, permits the use of fewer snubbers than in plants that were designed without considering leak before break. Also, the AP1000 uses gapped support devices to minimize the use of snubbers. The evaluation of those snubbers used as supports is outlined below.

Design specifications for snubbers include:

- Seismic requirements
- Normal environmental parameters
- Accident/post-accident environmental parameters
- Full-scale performance test to measure pertinent performance requirements
- Instructions for periodic maintenance (in technical manuals)

Two types of tests will be performed on the snubbers to verify proper operation:

- Production tests, including dynamic testing, on every unit to verify proper operability

- Qualification tests on randomly selected production models to demonstrate the required load performance (load rating)

The tests for large hydraulic snubbers (that is, those with capacities of 50 kips or greater) include the following qualification tests:

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1) a full Level D load test to verify sufficient load capacity,

2) testing at full load to verify proper bleed with the control valve closed,

and the following production tests:

1) testing to verify the control valve closes within the specified velocity range, and

2) testing to demonstrate that breakaway and drag loads are within the design limits.

The operability of essential snubbers is verified as described in Subsection 3.9.3.4.4 and in the inservice testing program required by Subsection 3.9.8.3 by verifying the proper installation of the snubbers, and performing visual inspections and measurements of the cold and hot positions of the snubbers as required during plant heatup to verify the snubbers are performing as intended.

3.9.3.4.4 Inspection, Testing, Repair, and/or Replacement of Snubbers

A program for inservice examination and testing of dynamic supports (snubbers) is prepared in accordance with the requirements of ASME OM Code, Subsection ISTD.

The inservice examination and testing includes a thermal motion monitoring program that is established for verification of snubber movement, adequate clearance and gaps, including motion measurements, and acceptance criteria to assure compliance with ASME Section III, Subsection NF.

The inservice examination and testing plan for applicable snubbers is prepared in accordance with the requirements of the ASME OM Code, Subsection ISTD. Snubber maintenance, repairs, replacements, and modifications are performed in accordance with the requirements of the ASME OM Code, Subsection ISTD. Details of the inservice examination and testing program are reported in the inservice inspection and testing plan included in the inservice testing program required by Subsection 3.9.8.3.

a. Snubber Design and Testing

1. A list of safety-related snubbers is included in Table 3.9-201.

2. The snubbers are tested to verify they can perform as required during the seismic events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. Production and qualification test programs for both hydraulic and mechanical snubbers are carried out by the snubber vendors. Acceptance criteria for compliance with ASME Section III Subsection NF, and other applicable codes, standards, and requirements, are as follows:

   - Snubber production and qualification test programs are carried out by strict adherence to the manufacturer's testing and technical manuals. These manuals are prepared by the snubber manufacturer and subjected to review for compliance with the applicable provisions of the ASME Pressure Vessel and Piping Code of record. The test program is periodically audited during implementation for compliance.

   - Snubbers are inspected and tested for compliance with the design drawings and functional requirements of the procurement specifications.

   - Snubbers are load rated by testing in accordance with the snubber manufacturer's testing program.

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- Design compliance of the snubbers per ASME Section III Paragraph NF-3128, and Subparagraphs NF-3411.3 and NF-3412.4.
- The snubbers are tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test. The functional parameters cited in Subparagraph NF-3412.4 are included in the snubber qualification and testing program. Other parameters in accordance with the ASME OM Code will be incorporated.
- The codes and standards used for snubber qualification and production testing are as follows:
  - ASME B&PV Code Section III (Code of Record date) and Subsection NF.
  - ASME OM Code, Subsection ISTD.
- Large bore hydraulic snubbers are full Service Level D load tested on a random sampling basis, including verifying proper bleed rates with the control valve closed. Snubbers receive testing that ensure control valve closure within the specified velocity range and demonstration that drag forces/breakaway forces are within design limits.

3. Safety-related snubbers are identified in Table 3.9-201, including the snubber identification and the associated system or component, e.g., line number. The snubbers on the list are hydraulic and constructed to ASME Section III, Subsection NF. The snubbers are used for shock loading only. None of the snubbers are dual purpose or vibration arrestor type snubbers. Dual purpose snubbers are those snubbers used for shock arrestor (low cycle) and vibration arrestor (high cycle), i.e. snubbers used for a) impact loads from seismic, pipe break, etc., and b) control of high cycle operational vibration. Snubbers used for low cycle shock loadings for any purpose are considered a single purpose.

b. Snubber Installation Requirements

Installation instructions contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing that contains the installation location of the snubber on the pipe and structure, the cold setting, and additional information needed to install the particular snubber.

c. Snubber Preservice Examination and Testing

Snubbers are examined and tested during the preservice period in accordance with the Preservice Test Plan, which was prepared in accordance with the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) (Reference 2), Subsection ISTD. The required ISTD examinations are supplemented with an additional examination requirement to verify that snubbers are not seized, frozen or jammed.

d. Snubber Inservice Examination and Testing

Inservice examination and testing of snubbers is conducted in accordance with the requirements of the ASME OM Code, Subsection ISTD as outlined in the Inservice Test program plan.

Visual examinations are performed on the snubbers to identify conditions that could adversely affect a snubber's operational readiness. Snubbers evaluated as unacceptable during visual

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examination may be accepted for continued service by successful completion of an operational readiness test. The schedule of snubber visual examinations is based on snubber performance.

Snubbers are tested according to the established sampling plan each fuel cycle. When an in-place test or bench test cannot be performed, snubber subcomponents that control the parameters to be verified are examined and tested. Snubbers that do not meet test requirements are evaluated and additional testing is conducted until the appropriate sample plan completion criteria are satisfied.

There is also an assessment performed to ensure that the service life of a snubber is not exceeded prior to removing the snubber from service.

3.9.3.5 Instrumentation Line Supports

[*The design loads, load combinations, and acceptance criteria for safety-related instrumentation supports are similar to those of pipe supports. Design loads include deadweight, thermal, and seismic (as appropriate). The acceptance criteria is ASME Subsection NF.*]*

3.9.4 Control Rod Drive System (CRDS)

3.9.4.1 Descriptive Information of CRDS

3.9.4.1.1 Control Rod Drive Mechanism (CRDM)

The AP1000 control rod drive mechanism is based on a proven Westinghouse design that has been used in many operating nuclear power plants. Figure 3.9-4 shows the control rod drive mechanism. Figure 4.2-8 shows the configuration of the driveline, including the control rod drive mechanism. Subsection 4.2.2 describes the design of the rod cluster control assemblies and gray rod cluster assemblies. The material requirements for the control rod drive mechanisms and the control assemblies are discussed in Section 4.5.

Control rod drive mechanisms are located on the head of the reactor vessel. They are coupled to rod cluster control assemblies (RCCAs) that have neutron absorber material over the active length of the control rods. The control rod drive mechanisms are also attached to gray rod cluster assemblies (GRCAs) that are used for load follow. The gray rod cluster assemblies are geometrically identical to the rod cluster control assemblies except that most of the rodlets are fabricated from a material specified in Subsection 4.2.2.

The control rod drive mechanisms for both the rod cluster control assemblies and the gray rod cluster assemblies are identical. Although the gray rod cluster assemblies are expected to drop during a trip insertion, the insertion of these assemblies is not required in order to shut down the reactor.

The primary functions of the control rod drive mechanism is to insert or withdraw, at a designated speed, 53 rod cluster control assemblies and 16 gray rod cluster assemblies from the core to control average core temperature. During startup and shutdown the control assemblies control changes in reactivity.

Operation of the control rod drive mechanisms is integrated to move groups of assemblies together. Each cluster assembly is in a bank of assemblies which is used for reactivity control, axial power distribution control, or shutdown control. The assemblies of each bank of several rod cluster control assemblies or gray rod cluster assemblies move at the same time.

The design of the control rod drive mechanisms and the control assemblies permits load follow without the use of chemical shim over most of the life of the core. The design of the control rod drive

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mechanisms also permits holding the rod cluster control assemblies and the gray rod cluster assemblies at any step elevation within the range of rod travel during normal operation. The rod cluster control assemblies and gray rod cluster assemblies have the same mechanical coupling with the control rod drive mechanism.

The control rod drive mechanism is a magnetically operated jack (magjack). A magnetic jack is an arrangement of three electromagnets energized in a controlled sequence by a power cycle to insert or withdraw rod cluster control assemblies and gray rod cluster assemblies in the reactor core in discrete steps. The control rod drive mechanism is designed to release the drive rod and rod cluster control assembly during any part of the power cycle sequencing if electrical power to the coils is interrupted. When released from the control rod drive mechanism, the drive rod and rod cluster control assembly or gray rod cluster assembly falls by gravity into a fully inserted position.

The control rod drive mechanism consists of four separate subassemblies. These are the pressure vessel, coil stack assembly, latch assembly, and drive rod assembly.

The pressure vessel includes a latch housing and a rod travel housing that are connected by a threaded, seal-welded, maintenance joint that facilitates removal of the latch assembly. The top of the rod travel housing provides seismic support by an interface with the integrated head package. The latch housing is the lower portion of the vessel and contains the latch assembly. The latch housing is welded to the mechanism nozzle by a bi-metallic weld. The nozzle of the control rod drive mechanism is attached to the vessel head by a shrink-fit and a partial penetration weld. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the core.

The coil stack assembly includes the coil housings, electrical conduit and connector, and three operating coils: the stationary gripper coil, the movable gripper coil, and the lift coil. The coil stack assembly is a separate unit. It is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment. Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches: the movable gripper latches and the stationary gripper latches. The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8-inch steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next 5/8-inch step.

The drive rod assembly includes a coupling, drive rod, disconnect button, disconnect rod, and locking button. The drive rod has a 5/8-inch pitch from groove to groove that engage the latches during holding or moving of the drive rod. The coupling is attached to the drive rod and provides the means for coupling to the rod cluster control assembly directly below the control rod drive mechanism. The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the rod cluster control assembly and permit remote disconnection of the drive rod.

The control rod drive mechanism withdraws and inserts a rod cluster control assembly or gray rod cluster assembly as shaped electrical pulses are received by the operating coils. An on or off sequence, repeated by silicon-controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Withdrawal of the drive rod and rod cluster control assembly or gray rod cluster assembly is accomplished by magnetic forces. Insertion is by gravity. The mechanism is capable of raising or lowering a maximum 400-pound load (which includes the drive rod weight) at a rate of 45 inches per minute.

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During plant operation, the stationary gripper coil and movable gripper coil of the drive mechanism holds the rod cluster control assembly in a static position until a stepping sequence is initiated, at which time the stationary gripper coil, movable gripper coil, and lift coil are energized sequentially.

The control rod position is measured by 48 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2500 psia. The three operating coils are designed to operate at 392°F, with forced-air cooling required to maintain the coil internal temperature at or below 392°F. The air for cooling is provided by fans and shrouds included as part of the integrated head package. A loss of the air cooling would be expected to result in the release of the drive rod in the worst case. For this reason, the cooling air is not required to be a safety-related system and does not require an emergency power supply.

The design and construction of the control rod drive mechanism includes provisions to establish that gross failure of the housing sufficient to allow a control rod to be ejected from the core is not credible. These provisions include the following:

- Construction of the housing of Type 304 or 316 stainless steel, which exhibits excellent notch toughness at the temperatures that will be encountered.

- The allowable stress levels in the mechanism are not exceeded due to system thermal transients at power or by thermal movement of the reactor coolant loops.

- The control rod drive mechanisms are hydrotested after manufacture at a minimum of 125 percent of system design pressure.

- The housings are hydrotested at a minimum of 125 percent of system design pressure after installation to the reactor vessel head and included as part of the hydro test of the completed reactor coolant system.

The analyses of postulated accidents discussed in Chapter 15 include the evaluation of a nonmechanistic control rod ejection. Section 3.5 does not consider ejected rods to be a credible missile.

3.9.4.1.2 Control Rod Withdrawal

The rod cluster control assembly is withdrawn by repeating the following sequence of events. The sequence, starting with the stationary and movable grippers energized in the hold position, is as follows:

1. Moveable Gripper Coil B - OFF

   The latch-locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove. Note: the movable gripper is turned off at the start of rod motion even though it will be turned on in step 2 to provide identical CRDM cycling for all steps since a step ends with the movable gripper off.

2. Movable Gripper Coil B - ON

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The latch-locking plunger rises and swings the movable gripper latches into the drive rod assembly groove. A small axial clearance exists between the latch teeth and the drive rod.

3. Stationary Gripper Coil A - OFF

   The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 1/16 inch, transferring the load of the drive rod assembly and attached control rod to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

4. Lift Coil C - ON

   The 5/8-inch gap between the movable gripper pole and the lift pole closes, and the drive rod assembly rises one step length (5/8 inch).

5. Stationary Gripper Coil A - ON

   The plunger rises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) a small fraction of an inch. The small vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

6. Movable Gripper Coil B - OFF

   The latch-locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

7. Lift Coil C - OFF

   The gap between the movable gripper pole and the lift pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to a drive rod assembly groove.

8. Repeat Step

   The sequence just described (items 1 through 6) is termed one step or one cycle. The rod cluster control assembly moves 5/8 inch for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute, and the drive rod assembly (which has a 5/8-inch groove pitch) is raised 72 grooves per minute. The rod cluster control assembly is thus withdrawn at a rate of up to 45 inches per minute. The gray rod cluster assemblies are withdrawn in an identical manner.

   After rod motion has ended, movable gripper coil B is turned on to re-establish double-hold (that is, holding the rod in place with both the stationary and movable grippers).

3.9.4.1.3 Control Rod Insertion

The sequence for rod cluster control assembly insertion is similar to that for control rod withdrawal, except that the timing of lift coil C ON and OFF is changed to permit lowering of the control assembly. The sequence, starting with the stationary gripper energized in the hold position, is as follows:

1. Movable Gripper Coil B - OFF

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The latch-locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

2. Lift Coil C - ON

   The 5/8-inch gap between the movable gripper and lift the pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

3. Movable Gripper Coil B - ON

   The latch-locking plunger rises and swings the movable gripper latches into a drive rod assembly groove. A small axial clearance exists between the latch teeth and the drive rod assembly.

4. Stationary Gripper Coil A - OFF

   The force of gravity, acting upon the drive rod assembly and attached rod cluster control assembly, causes the stationary gripper latches and plunger to move downward 1/16 inch transferring the load of the drive rod assembly and attached rod cluster control assembly to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

5. Lift Coil C - OFF

   The force of gravity and spring force separate the movable gripper pole from the lift pole. The drive rod assembly and attached rod cluster control assembly drop down 5/8 inch.

6. Stationary Gripper Coil A - ON

   The plunger rises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) a small fraction of an inch. The small, vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

7. Movable Gripper Coil B - OFF

   The latch-locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

8. Repeat Step

   The sequence is repeated, as for rod cluster control assembly withdrawal, up to 72 times per minute, which gives an insertion rate of 45 inches per minute. The gray rod cluster assemblies are inserted in an identical manner.

   After rod motion has ended, movable gripper coil B is turned on to re-establish double-hold (that is, holding the rod in place with both the stationary and movable grippers).

3.9.4.1.4 Holding and Tripping of the Control Rods

During most of the plant operating time, the control rod drive mechanisms hold the rod cluster control assemblies withdrawn from the core in a static position. During most plant operation the gray rod

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cluster assemblies are held by the control rod drive mechanisms withdrawn or inserted in the core in a static position as directed by flux shape considerations. In the holding mode, two coils – stationary gripper coil A and movable gripper coil B – are energized on each mechanism. The drive rod assembly and attached rod cluster control assemblies or gray rod cluster assemblies hang suspended from the three latches of the stationary gripper. Additionally, the three latches of the movable gripper are engaged to hold the drive rod assembly in place in the event of a failure that would cause the release of the stationary gripper.

When the drive line is positioned in the last few steps, the rod cluster control assemblies and gray rod cluster assemblies are out of the last portion of the core, although not fully withdrawn from the fuel assemblies. This covers the range of steps from 263-266. The control rod drive mechanism may be located at any one or more of these step locations during operation of the plant and be considered fully out without any adverse impact on the control rod drive mechanism or plant operation.

The rod clusters cannot be physically withdrawn from the guide tubes by the control rod drive mechanisms since no additional grooves are machined in the drive rod past the last position.

If power to the stationary and movable gripper coils is cut off, the combined weights of the drive rod assembly and the rod cluster control assembly or gray rod cluster assembly (plus the stationary gripper and movable gripper return springs) move the latches out of the drive rod assembly groove. The trip occurs as follows:

- The magnetic field, holding the stationary gripper plunger against the stationary gripper pole, collapses, and the stationary gripper plunger is forced down by the stationary gripper return spring and the weight acting upon the latches.

- The magnetic field, holding the moveable gripper plunger against the movable gripper pole, collapses, and the movable gripper plunger is forced down by the moveable gripper return spring and the weight acting upon the latches.

The control rod falls by gravity into the core. After the driveline is released by the mechanism, it falls freely until the control rods enter the dashpot section of the fuel assembly where the coolant in the guide tubes slows the rate of descent until the rods are fully inserted.

3.9.4.1.5 Testing Program

As noted earlier, the AP1000 control rod drive mechanism is based on a proven Westinghouse design that has been used in many operating nuclear power plants. The control rod cluster and fuel assembly thimble tube mechanical design is also based on a proven design. The production tests that each control rod drive mechanism undergoes are outlined in Subsection 3.9.4.4.

3.9.4.2 Applicable CRDS Design Specifications

The specifications for the design, fabrication, construction, and operation of the control rod drive system (CRDS) include provisions related to the functional requirements, pressure boundary integrity, strength and durability of the internal components, and electrical requirements for the operating mechanism. The specifications and design requirements are consistent with the safety classification of the various parts of the control rod drive system as defined in Section 3.2.

The materials used in the control rod drive mechanisms are discussed in Subsection 4.5.1. The rod position instrumentation is described in Section 7.7.

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Since the AP1000 control rod drive mechanism is a design previously provided for other nuclear power plants the specifications are well established. The specifications are outlined in the following discussions.

3.9.4.2.1 Control Rod Drive Mechanism Functional Requirements

The suitability of the functional requirements for the step size and rate of withdrawal and insertion during normal operation and the time to drop into the core have been demonstrated during many years of successful operation of similar Westinghouse-designed control rod drive mechanisms. The time required for the control rod drive system to release the rod cluster control assemblies into the core is evaluated to determine that it is sufficient in analyses of postulated accident conditions. For a discussion of the evaluation of the performance of the reactivity control function of the AP1000 control rod drive system and specific AP1000 accident analyses see Section 4.6 and Chapter 15.

The basic operational requirements for the control rod drive mechanisms follow:

- 5/8-inch step
- 166.755 inch travel, maximum (cold conditions)
- 400-pound maximum load
- Step in or out at 45 inches per minute (72 steps per minute) maximum, 5 inches per minute (8 steps per minute) minimum
- Electrical power interruption initiating release of drive rod assembly
- Trip delay time of less than or equal to 150 millisecond. Free fall of drive rod assembly is to begin less than 150 millisecond after power interruption, no matter what holding or stepping action is being executed, with any load and coolant temperature of 100°F to 650°F.
- 60-year design objective with normal refurbishment

  Testing and operating experience has validated these requirements and the capability of the AP1000 control rod drive mechanism design to meet them.

3.9.4.2.2 Pressure Housing Requirements

The pressure housing portion of the control rod drive mechanism, the latch housing and rod travel housing, comprises a portion of the reactor coolant pressure boundary. The design pressure and temperature for the control rod drive mechanism pressure housing are the same as for the reactor vessel.

As part of the reactor coolant pressure boundary, the pressure housing is constructed in conformance with requirements in 10 CFR 50.55a. The conformance of the reactor coolant pressure boundary with applicable code and standards is discussed in Section 5.2. The pressure housing meets design, material, fabrication, analysis, quality assurance, and other requirements for Class 1 components in ASME Code, Section III. The pressure housing is required to meet stress requirements for design and transient conditions. The AP1000 control rod drive mechanism pressure housing uses only one canopy seal weld, which is between the latch housing and the rod travel housing. The rod travel housing includes a vent hole to allow water to drain from the canopy seal weld area. This eliminates the presence of an oxygenated environment near the weld. The single AP1000 canopy seal weld has reduced stresses compared to the lower seal weld in past designs.

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These design improvements over past control rod drive mechanisms aid to prevent the occurrence of stress corrosion cracking.

3.9.4.2.3 Internal Component Requirements

The internal components of the control rod drive mechanism include the latch assembly, drive rod and the coupling that attaches the drive rod to the rod cluster control assemblies and gray rod cluster assemblies.

The design, fabrication, inspection, and testing of these non-pressure boundary components typically do not come under the jurisdiction of the ASME Code. For those materials which do not have established stress limits the limits are based on the material specification mechanical property requirements.

In addition to dead-weight and operational loads, the design of the driveline is evaluated for loads due to safe shutdown earthquake and flow induced vibration as described in Subsection 3.9.5.3, and the resulting deflections are limited such that the CRDM will function properly.

Postulated failures of drive rod assemblies and latch mechanisms by fracture or decoupling lead to a decrease in reactivity. A postulated failure leading to the release of a drive rod or portion of a drive rod results in an insertion of control rods guided by the control rod assembly. A control rod drop is indicated by instrumentation that monitors the nuclear reaction and detect a decrease in reactivity.

A postulated failure of a control rod drive mechanism to insert a control assembly due to sticking or galling of the drive rod or latch assembly is accounted for in the safety analyses, which assume that the control assembly at the most reactive core location is inoperable.

In addition to the requirements related to the strength of the internal components, criteria have been developed for clearances in the latch assembly and between the latch arms and drive rod. The latch assembly has parts of austenitic and ferritic stainless steels and other alloys. Differential thermal expansion could eliminate clearances and result in binding or otherwise restrict movement of the latch assembly if not allowed for in the design.

The design requirement is that sufficient clearance exist between the moving parts in the latch assembly at expected operating and design condition temperatures. An evaluation of the thermal expansion, room temperature clearances, and geometry demonstrates that an appropriate clearance is available at design and normal operating conditions.

For the magjack mechanism to work properly to insert or withdraw the control rods, the latch arms contacting the drive rod, (that is the movable and stationary gripper latches) must not be under load at the same time. The effect of the differential thermal expansion on the latch arms, pressure housing, is evaluated to provide that the appropriate clearance between the drive rod and the unloaded latch arm is maintained.

3.9.4.2.4 Coil Stack Assembly Requirements

The coil stack assembly provides the electromotive force to move the latches in the latch assembly. The safety function of rapid insertion of the control rods can be accomplished by removing power from the coils. The separation and redundancy required of the control system and power supplied to the control rod drive system is discussed in Section 4.6.

Postulated electrical or structural failures of the coil assembly do not result in a condition which would prevent control rod insertion. As a result, the electrical coils are built using standard industrial quality assurance and are not required to be built to IEEE Class 1E standards.

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The coil stack assembly is located outside the pressure housing. The assembly does not come in contact with the reactor coolant and does not have any pressure-retaining function. The operating temperature of the coils is maintained below 392°F.

The coil stack assembly slides over the pressure housing and remains in place without a permanent mechanical or welded attachment. The assembly clearances permit removal of an assembly even when the control rod drive mechanism is at normal operating temperature. Thus, a malfunctioning coil assembly could be replaced without a complete cooldown of the plant. The clearances between the coil and coil housing are selected to minimize the gap at normal operating temperature to facilitate coil cooling.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The pressure housing portion of the control rod drive mechanism is a Class 1 component required to meet the requirements of ASME Code, Section III. Subsection 3.9.3 defines the loading combinations considered in the evaluation of ASME Code, Section III, pressure boundary components.

For each loading combination, the appropriate stresses due to pressure, component weight, external loads, hydraulic forces, thermal gradients, and seismic dynamic forces are evaluated and demonstrated to be less than the applicable stress limits. The cyclic stresses are combined with constant stresses to evaluate the fatigue usage due to cyclic loads. The transients used in the evaluation of cyclic loads are described in Subsection 3.9.1. The effect of seismic events is addressed by considering a seismic event with an amplitude equal to one-third of the safe shutdown earthquake evaluated as a Level B event. The seismic contribution to the fatigue evaluation is based on five seismic events with an amplitude of one-third the safe shutdown earthquake and with 63 cycles per event. The results of the stress evaluation are documented in a component stress report, as required by the ASME Code.

The control rod drive mechanism is supported by the attachment of the bottom of the assembly to the reactor vessel head and a connection to the integrated head package at the top of the rod travel housing. The integrated head package also provides the support to the cooling air shrouds and control rod drive mechanism electrical supply cables to prevent excessive loading on the control rod drive mechanisms during seismic events.

Hydrostatic tests according to the requirements of the ASME Code verify the pressure boundary integrity of the pressure housing prior to operation. The nozzle is attached to the reactor vessel head by the head supplier and is hydro tested as required. The rod travel housing seal weld is performed prior to final assembly following the assembly of the travel housing to the latch assembly housing. The hydrostatic test of the connection between the rod travel housing and the latch housing is done on the reactor vessel closure head after the housings are threaded together and seal welded.

To assure functional capability of the control rod dive mechanism following a seismic event or a pipe break, the bending moments on the control rod drive mechanisms are limited to those that produce stress levels in the pressure boundary of the control rod drive mechanism less than ASME Code limits during anticipated transient conditions. This limit provides that the rod travel housing does not bend to the extent that the drive rod binds during insertion of the control rods. The analysis evaluates the load combinations that include safe shutdown earthquake and pipe break. The pipe break considered is at least as large as the largest pipe in or connected to the reactor coolant system that is not qualified as leak before break line. See Subsection 3.9.7 for information on the control rod drive mechanism deflection limit requirements for the integrated head package.

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3.9.4.4 Control Rod Drive Mechanism Performance Assurance Program

The capability of the pressure housing components to perform throughout the 60 year design objective is confirmed by the stress analysis report required by the ASME Code, Section III.

To confirm the operational adequacy of the combination of fuel assembly, control rod drive mechanism, and rod cluster control assembly, functional test programs have been conducted. These tests verify that the trip time achieved by the control rod drive mechanisms meets the design requirements. These tests have been reported in WCAP-8446 (Reference 9).

The units are production tested prior to shipment to confirm the capability of the control rod drive mechanism to meet design specification operation requirements. Each production control rod drive mechanism undergoes a production test as listed in Table 3.9-13.

The trip time requirement is confirmed for each control rod drive mechanism prior to initial reactor operation and at periodic intervals after initial reactor operation, as required by the technical specifications. See Section 14.2 for preoperational and startup testing.

To demonstrate proper operation of the control rod drive mechanism and to provide acceptable core power distributions, rod cluster control assembly partial movement checks are performed as required by the Technical Specifications. In addition, periodic drop tests of the rod cluster control assembly are performed at each refueling shutdown to demonstrate continued capability to meet trip time requirements, consistent with safety analyses in Chapter 15.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Design Arrangements

The AP1000 reactor internals consist of two major assemblies - the lower internals and the upper internals. The reactor internals provide the protection, alignment and support for the core, control rods, and gray rods to provide safe and reliable reactor operation. In addition, the reactor internals help to accomplish the following: direct the main coolant flow to and from the fuel assemblies; absorb control rod dynamic loads, fuel assembly loads, and other loads and transmit these loads to the reactor vessel; support instrumentation within the reactor vessel; provide protection for the reactor vessel against excessive radiation exposure from the core; and position and support reactor vessel radiation surveillance specimens.

During reactor operation, the core barrel directs the coolant flow from the reactor vessel inlet nozzles, through the downcomer annulus, and into the lower plenum below the lower core support plate. The flow then turns and passes through the lower support plate and into the core region. After leaving the core, it passes through the upper core plate; then bypasses through and around the control rod guide tubes and the support columns to reach the outlet nozzles. During operation, a small amount of inlet coolant is diverted from the core to cool the core shroud and the vessel head area.

3.9.5.1.1 Lower Core Support Assembly

The major containment and support member of the reactor internals is the lower core support assembly, shown in Figure 3.9-5. This assembly consists of the core barrel, lower core support plate, secondary core support, vortex suppression plate, core shroud, neutron panels, radial supports, and related attachment hardware. The major material for this structure is 300 series austenitic stainless steel. The lower core support assembly is supported at its upper flange from a ledge in the reactor vessel flange. Its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. The radial support system consists of keys attached to the lower end of the core barrel subassembly. These keys engage clevis inserts in the reactor vessel. This system

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restricts the lower end of the core barrel from rotational and/or translational movement, but allows for radial thermal growth and axial displacement.

The core shroud is located inside the core barrel and above the lower core support. This shroud forms the radial periphery of the core. Through the dimensional control of the cavity (the gap between the fuel assemblies and the shroud) and the shroud cooling flow inlets, the core shroud provides directional and metered control of the reactor coolant through the core. The core shroud serves to provide a transition from the round core barrel to the square fuel assemblies.

Loads acting vertically downward from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried by the lower core support plate into the core supports. The loads are then carried through the core barrel shell to the core barrel flange, which is supported by the vessel flange. Transverse loads from earthquake acceleration, coolant cross-flow, and vibration are carried by the core barrel shell and distributed through the lower radial support to the vessel wall and to the vessel flange. Transverse loads from the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core support plate to the barrel wall, and by the upper core plate inserts to the core barrel alignment plates described in Subsection 3.9.5.1.2.

The main radial support system of the lower end of the core barrel is accomplished by key and keyway joints to the reactor vessel wall. Clevis blocks are welded to the vessel inner diameter at equally spaced points around the inner circumference of the vessel. Another insert block is bolted to each of these blocks and has a keyway geometry. Opposite each of these is a key attached to the internals. During assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. Correct positioning of the internals is provided by the installation equipment (lifting rig) guide studs and bushings. In this design, the internals have a support at the furthest extremity, such that the core barrel is laterally supported at the top and bottom.

Radial and axial expansion of the core barrel is accommodated, but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal structures are within ASME Code, Section III, Subsection NG limits.

In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy-absorbing devices limit the dynamic force imposed on the reactor vessel. The energy absorbing device is the secondary core support. In addition, the secondary core support also transmits the vertical load of the core uniformly to the reactor vessel, limits the displacement to prevent withdrawal of the control rods from the core, and limits the displacement to prevent loss of alignment of the core with the upper core support to allow the control rods to be inserted into the reactor.

The lower plenum vortex suppressor plate is positioned in the vessel lower plenum to suppress flow vortices formed by the reactor coolant flow reversal in this region. The suppressor plate is supported by columns from the lower core support plate.

3.9.5.1.2 Upper Core Support Assembly

The AP1000 upper core support assembly consists of the upper support, the upper core plate, the support columns, and the guide tube assemblies. Figure 3.9-6 shows the upper core support assembly.

The support grid assembly establishes the spacing between the upper support and the upper core plate. The support columns are fastened at the top and bottom to these plates. The support columns transmit the mechanical loadings between the two plates and some serve the supplementary function of supporting the tubes that house the fixed in-core detectors.

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The IGA feeds instrumentation from eight reactor vessel head penetrations to 42 locations within the fuel assemblies. The IGA is composed of a number of plate and shell members welded or bolted together, and rests on the top surface of the upper support plate. The top of the IGA is constrained where Quickloc stalks interface with the vessel head penetrations.

The IGA is designed to prevent excessive vibration of the incore instrumentation thimble assemblies (IITAs), which contain the fixed in-core flux detectors and core-exit thermocouples. The IGA provides a pathway from the Quickloc instrument nozzles through the upper support assemblies and upper support columns to the top of the fuel assemblies. The pathway protects and guides the IITAs during initial installation, reactor operation, and withdrawal/insertion during outages.

The guide tube assemblies sheath and guide the control rod drive shafts and control rods. The guide tubes are fastened to the upper support and are restrained by pins in the upper core plate for proper orientation and support.

The upper core support assembly is positioned in its proper orientation, with respect to the lower core support assembly, by alignment plates in the core barrel. Four equally spaced alignment plates are located at an elevation in the core barrel where the upper core plate is positioned. Four mating sets of inserts are located in the upper core plate at the same positions. As the upper support assembly is lowered into the lower support assembly, the inserts engage the alignment plates in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design.

Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. This system of locating pins and guidance arrangement provides proper alignment of the lower core support assembly, the upper core support assembly, the fuel assemblies, and control rods.

The upper and lower core support assemblies are preloaded by a large circumferential spring, which rests between the upper barrel flange and the upper core support assembly. This spring is compressed by installation of the reactor vessel head.

Vertical loads from weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns, to the upper support, and then into the reactor vessel head. Transverse loads from coolant cross-flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support and upper core plate. The upper support plate is particularly stiff to minimize deflection.

3.9.5.1.3 Core Shroud

The core shroud is between the lower core barrel and core, surrounding the core and forming the core cavity. The core shroud consists of formed vertical plates with fully welded vertical seams to prevent lateral flow from the fuel assemblies. This core shroud is a proven design that is currently utilized in operating plants.

3.9.5.1.4 Flow Skirt

The flow skirt is a perforated cylindrical ring, shown schematically in Figure 3.9-9, that is an attachment to the reactor vessel bottom head. However, since this structure is located entirely within the pressure boundary, it will be described in this reactor internals section. The flow skirt is welded to support lugs on the inside surface of the reactor vessel bottom head. A vertical clearance is provided between the top of the flow skirt and the bottom surface of the lower core support plate to prevent contact during operation and postulated core drop accident conditions. The flow skirt provides a more uniform core inlet flow distribution.

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3.9.5.1.5 Reactor Internals Interface Arrangement

Figure 3.9-8 shows the arrangement of reactor internals components shown in Figures 3.9-5 and 3.9-6 and their relative position in the reactor vessel. As shown in the figure, the lower reactor internal (Figure 3.9-5) rests on the vessel ledge. The upper core support structure (Figure 3.9-6) also rests at the same location on the top of a large compression spring (hold down spring). The hold down spring is between the upper support plate flange and the core barrel flange as shown in the figure. Both the assemblies are held together by reactor vessel closure studs, which clamp the upper head to upper shell of reactor vessel. The lower reactor internals are also guided laterally by four support lugs welded to the bottom head of reactor vessel.

3.9.5.2 Design Loading Conditions

3.9.5.2.1 Level A and B Service Conditions

The level A and B service conditions that provide the basis for the design of the reactor internals are:

- Fuel assembly and reactor internals weight
- Fuel assembly and core component spring forces, including spring preloading forces
- Differential pressure and coolant flow forces
- Temperature gradients
- Operational thermal transients listed in Table 3.9-1
- Differences in thermal expansion, due to temperature differences and differential expansion of materials
- Loss of load/pump overspeed
- Earthquake (included only in fatigue evaluation; amplitude equal to one-third of the safe shutdown earthquake response)

3.9.5.2.2 Level C Service Conditions

The Level C service conditions that are the basis for the design of the reactor internals are small break loss of coolant accident, and small steam line break.

3.9.5.2.3 Level D Service Conditions

The Level D service conditions that are the basis for the design of the reactor internals are safe shutdown earthquake (SSE), and pipe rupture. The pipe ruptures are evaluated for lines for which dynamic effect can not be excluded based on mechanistic pipe break criteria. See Subsection 3.6.3 for a description of mechanistic pipe break criteria. The breaks considered are those inside containment, in systems that carry reactor coolant, steam and feedwater. These breaks have the greatest effect on the reactor internals response.

3.9.5.2.4 Design Loading Categories

The combination of design loadings fit into either the service level A, B, C, or D conditions shown on Figures NG-3221-1 and NG-3224-1, NG-3232-1, and by Appendix F of the ASME Code, Section III.

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3.9.5.3 Design Bases

The reactor vessel internals components designated as ASME III Class CS core support structures are designed, fabricated, and examined in accordance with the requirements of ASME III, Subsection NG for Core Support Structures. The design documentation for these Class CS core support structures include a certified Design Specification and a certified Design Report conforming to the requirements of ASME III, Subsection NCA.

The basis used for design, construction, and examination, for those reactor vessel internals components not designated ASME III Class CS core support structures, is defined by Westinghouse as provided in the ASME Code, Subsection NG.

The scope of the stress analysis requires many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation.

3.9.5.3.1 Mechanical Design Basis

The design bases for the mechanical design of the AP1000 reactor vessel internals components are as follows:

- The reactor internals, in conjunction with the fuel assemblies, direct reactor coolant through the core to achieve an acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for the varying modes of operation. In addition, required cooling for the reactor pressure vessel head is provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

- The core shroud forms the core cavity and directs the reactor coolant flow through the fuel assemblies.

- Provisions are made for installing in-core instrumentation useful for plant operation, and vessel material test specimens required for a pressure vessel irradiation surveillance program.

- The core internals are designed to withstand mechanical loads arising from the safe shutdown earthquake and to meet the requirements of the following item.

  The reactor has mechanical provisions which are sufficient to adequately support the core and internals and to maintain the core intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

- Following a design basis accident, the plant is capable of being shut down and cooled in an orderly fashion, so that the fuel cladding temperature is kept within specified limits. Therefore, the deformation of certain critical reactor internals is kept sufficiently small to allow continued core cooling.

The functional limitations for the core structures and internal structures during the design basis accident are shown in Table 3.9-14.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Subsection 3.9.2.

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3.9.5.3.2 Allowable Deflections

Loads and deflections imposed on components, as a result of shock and vibration, are determined analytically and/or experimentally in both scaled models and operating reactors. The cyclic stresses resulting from these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions, as summarized in Table 3.9-1.

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For normal operating and accident conditions, the deflection criteria of internal structures are the limiting values given in Table 3.9-14. The upper barrel radial inward deflection limit is based on preventing contact between the barrel and the peripheral upper guide tubes during a LOCA event. The rod cluster control assembly can be dropped during the LOCA event if the guide tubes are not contacted by the barrel. The radial outward (uniform) deflection is based on maintaining flow in the downcomer annulus between the core barrel and pressure vessel wall. A peak deflection greater than the uniform allowable is acceptable provided that the annulus blockage from the deflected core barrel is less than the non-uniform radial outward deflection limit. The upper package allowable deflection is based on the clearance between the upper core plate and guide tube support pin shoulder. Exceeding this value could result in potential buckling of the guide tube and potential loss of function during operating or accident conditions. The rod cluster guide tube allowable lateral deflection is based on test data that indicates the rod cluster control assembly drop time will not be impaired.

The criteria for the postulated core drop accident are based on analyses that determine the total downward displacement of the internal structures, following a hypothetical core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structure and the reactor vessel lower head in the hot condition is approximately 0.6 inches. An additional displacement of approximately 0.7 inches would occur from the strain of the energy-absorbing devices of the secondary core support. Therefore, the total drop distance is about 1.3 inches. That distance is less than the distance that permits the tips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

The secondary core support is only required to function during an accident involving the hypothetical catastrophic failure of core support (such as core barrel or barrel flange). There are four supports in each reactor. This structure limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel.

The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support. The energy is absorbed during the plastic deformation of the controlled volume of stainless steel loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 0.7 inches in the hot condition, after which a positive stop is provided. The maximum deformation of the secondary core support allows for the maintenance of flow paths through the lower portion of the vessel and lower core support to provide cooling of the fuel under forced and natural circulation conditions.

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3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of pumps and valves is performed in accordance with the ASME Operations and Maintenance (OM) Code (Reference 2) as required by 10 CFR 50.55a(f), except where a deviation from Code requirements has been granted by the NRC as outlined in Subsection 3.9.6.3.

The requirements for system pressure tests are defined in the ASME Code, Section XI, IWA-5000. These tests verify the pressure boundary integrity and are part of the inservice inspection program, not part of the inservice test program.

A preservice test plan, which identifies the required functional testing, is to be submitted to the NRC prior to performing the tests and following the start of construction. The inservice test plan, which identifies requirements for functional testing, is to be submitted to the NRC prior to the anticipated date of commercial operation. These plans will document compliance with the OM Code except where a deviation from Code requirements has been granted by the NRC as outlined in Subsection 3.9.6.3. The execution and updating of these plans, as required, will ensure that all applicable pumps and valves are in a state of operational readiness to perform their safety function throughout the life of the plant.

Table 13.4-201 provides milestones for preservice and inservice test program implementation.

3.9.6.1 Inservice Testing of Pumps

ASME OM Code ISTA-1100 and ISTF-1000 provide the scoping criteria for pumps to be included in the preservice testing and inservice testing (IST) program plans. ISTF-1000 limits pumps to be included in the IST program to those that have an emergency power source. The emergency power source (Class 1E) for the AP1000 is the Class 1E DC distribution system, and there are no pumps powered from this source. Therefore, no pumps will be included in the preservice test plan or IST program plan.

3.9.6.2 Inservice Testing of Valves

The Preservice Test Plan and IST Program Plan will include lists of safety-related valves (i.e., those valves that perform a specific function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident) and other selected valves are subject to operational readiness testing. The valve testing portion of the preservice and IST plans shall conform to the requirements of the ASME OM Code and comply with all applicable portions of 10 CFR 50.55a(f), except where a deviation from Code requirements has been authorized by the NRC as outlined in Subsection 3.9.6.3.

Leakage testing requirements for pressure isolation valves, which provide isolation between high and low pressure systems, are contained in both the surveillance requirements for Technical Specification 3.4.15 and the OM Code. The pressure isolation valves that require leakage testing are identified in Table 3.9-18.

Pre-conditioning of valves or their associated actuators or controls prior to IST testing undermines the purpose of IST testing and is not allowed. Pre-conditioning includes manipulation, pre-testing, maintenance, lubrication, cleaning, exercising, stroking, operating, or disturbing the valve to be tested in any way, except as may occur in an unscheduled, unplanned, and unanticipated manner during normal operation. The IST program will include practices to prevent, where practical, unacceptable preconditioning.

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3.9.6.3 Deviations from Code Requirements

When it is desirable to deviate from the requirements of the OM Code for Inservice or Preservice testing an NRC approved alternative can be used. The alternative can be an ASME OM Code Case approved by the NRC in 10 CFR 50.55a. Also, an Alternative to the OM Code requirements, or requirements of 10 CFR 50.55a, may be submitted in accordance with 10 CFR 50.55a(z), however this Alternative must be authorized by the NRC prior to use.

3.9.6.4 Additional Testing

This section contains the testing commitments that are outside of the OM Code/IST.

3.9.6.4.1 System Level Tests

The test program includes periodic systems level tests and inspections that demonstrate the capability of safety-related features to perform their safety-related functions such as passing flow or transferring heat. For this system level testing, the test and inspection frequency is once every 10 years. Staggering of the tests of redundant components is not required. These tests may be performed in conjunction with inservice tests conducted to exercise check valves or to perform power-operated valve operability tests. Alternate means of performing these tests and inspections that provide equivalent demonstration may be developed. Table 3.9-17 identifies the system level tests.

3.9.6.4.2 Check Valve Low Differential Pressure Tests

Safety-related check valves that perform a safety related mission to transfer open under low differential pressure conditions have periodic testing to verify the capability of the valve to initiate flow.

The intent of this supplemental test is to determine the pressure required to initiate flow. This differential pressure will verify that the valve will initiate flow at low differential pressure. This low pressure differential test is performed in addition to exercise inservice tests.

The specified frequency for this testing is once each refueling cycle.

3.9.6.4.3 CVS/RCS Pressure Boundary Valve Leakage Testing

To support the nonsafety-related classification of the CVS purification subsystem inside containment the CVS RCS pressure boundary valves are subject to leak testing. These valves are not included as pressure isolation valves in Table 3.9-18, as the non-safety CVS inside containment is designed for full RCS pressure. The leakage through valves CVS-V001, CVS-V002, and CVS-V080 will be tested separately with a leakage limit of 1.5 gpm for each valve. The leakage through valves CVS-V067, V081, V082, V084, and V085 will be tested at the same time as a group with a leakage limit of 1 gpm for the group. The leak test(s) will be performed at reduced RCS pressure. The observed leakage at lower pressure will be confirmed to be below the limit when adjusted to normal operating conditions using an NRC-accepted methodology. The test requirements including the test pressure and method for verifying test acceptability are defined in the testing procedures. The specified frequency for this testing is once each refueling cycle.

3.9.6.4.4 Control Room Emergency Ventilation Pressure Regulating Valves (VES-V002A/B)

Pressure regulating valves that cannot be externally controlled are exempt from testing per the OM Code, and the Code does not provide testing methodology for these valves. These valves are

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functionally tested quarterly, and by the Technical Specification surveillance which requires monthly operation of VES for ≥ 15 minutes. Additionally, these valves will be tested through the range of inlet pressures, on a frequency established by the Equipment Reliability program.

3.9.6.4.5 Air-Operated Valves Additional Testing

Additional testing is performed as part of the air-operated valve (AOV) program, which includes the key elements for an AOV Program as identified in the JOG AOV program document, Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000 (References 42 and 43). The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air- and other power-operated valves included in the IST program. For example, key lessons learned addressed in the AOV program include:

- Valves are categorized according to their safety significance and risk ranking.
- Setpoints for AOVs are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).
- Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, to identify potential degradation, unless those valves are periodically cycled during normal plant operation, under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If required based on valve qualification or operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform its required functions.
- Sufficient diagnostics are used to collect relevant data (e.g., valve stem thrust and torque, fluid pressure and temperature, stroke time, operating and/or control air pressure, etc.) to verify the valve meets the functional requirements of the qualification specification.
- Uncertainties associated with performance of these tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.
- Test frequency is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with References 42 and 43, with a minimum of 5 years (or 3 refueling cycles) of data collected and evaluated before extending test intervals.
- Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when maintenance on the valve, repair or replacement, have the potential to affect valve functional performance.
- Guidance is included to address lessons learned from other valve programs specific to the AOV program.

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- Documentation from AOV testing, including maintenance records and records from the corrective action program are retained and periodically evaluated as a part of the AOV program.

3.9.6.4.6 Regulatory Treatment of Non-Safety Systems (RTNSS) Component Testing

10 CFR 50.55a(b)(3)(iii)(D) requires that the operational readiness of pumps, valves and dynamic restraints within the scope of Regulatory Treatment of Non-Safety Systems be assessed. Since the components are outside the scope of the OM Code the method used to meet this requirement is not included in the IST program. The method for assessment of operational readiness for these components is documented in the equipment reliability assessment for the associated components and includes testing of the components performed per Section 16.3.

3.9.6.4.7 Additional Testing of POVs Other Than MOVs or AOVs

The attributes of the AOV testing program described in Subsection 3.9.6.4.5 above, to the extent that they apply to and can be implemented on other safety-related power-operated valves, such as electro-hydraulic valves, are applied to those other power-operated valves.

3.9.6.4.8 Valve Disassembly to Assess Valve Degradation

Disassembly and inspection of valves will be performed based on information from inservice testing, or other program requirements, as noted below:

- PRA importance measures.
- Design reliability assurance program contained in Section 16.2.
- Historical performance of power-operated valves (identify valve types which experience unacceptable degradation in service.)
- Basic design of valves including the use of components subject to aging and requiring periodic replacement.
- Analysis of trends of valve test parameters during valve inservice tests.
- Results of nonintrusive techniques. Disassembly and inspection may not be needed if nonintrusive techniques are sufficient to detect unacceptable valve degradation.

3.9.7 Integrated Head Package

The integrated head package (IHP) combines several components in one assembly to simplify refueling the reactor. Figure 3.9-7 illustrates the integrated head package. The integrated head package includes a lifting rig, seismic restraints for control rod drive mechanisms, support for reactor head vent piping, cable bridge, power cables, cables for in-core instrumentation, cable supports, and shroud assembly.

The integrated head package provides the ability to rapidly disconnect cables, including the CRDM power cables, digital rod position indication cables, and in-core instrument cables from the components. The integrated head package also provides the ability to rapidly disconnect the reactor head vent system.

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The integrated head package provides the ability to move these components as an assembly to permit their lifting and removal with the reactor vessel head. In addition, the integrated head package provides support for the vessel head stud tensioner/detensioner during refueling.

The lifting rig function is discussed in Subsection 9.1.5. The control rod drive mechanisms are discussed in Subsection 3.9.4. The control rod drive mechanism support and cooling function is discussed in Section 4.6. The reactor vessel head vent function is discussed in Subsection 5.4.12. The function and requirements of the in-core instrumentation are discussed in Chapter 7.

3.9.7.1 Design Bases

Components required to provide seismic restraint for the control rod drive mechanisms and the valves and piping of the reactor head vent are AP1000 equipment Class C, seismic Category I. These are designed in accordance with the ASME Code, Section III, Subsection NF requirements.

The loads and loading combinations due to seismic loads for these components are developed using the appropriate seismic spectra.

The structural design of the integrated head package is based on a design temperature consistent with the heat loads from the vessel head, the control rod drive mechanisms, and electrical power cables. The design also considers changes in temperature resulting from plant design transients and loss of power to the cooling fans.

Components required to provide cooling to the control rod drive mechanisms are nonnuclear safety-related AP1000 equipment Class E. Section 4.6 offers a discussion of the effect of failure of cooling of the control rod drive mechanisms.

Those components that function as part of the lifting rig are required to be capable of lifting and carrying the total assembled load of the package. This includes the vessel head, control rod drive mechanisms, control rod drive mechanism seismic supports, shroud, cooling ducts, and insulation. The lifting rig components are required to meet the guidance for special lifting rigs, in NUREG-0612 (Reference 10). The lifting rig components are safety-related, AP1000 equipment Class C, seismic Category II.

The components of the incore instrumentation system (IIS) that interface with the IHP are the QuickLoc stalk assembly and the IIS cables and connectors. Access to each QuickLoc assembly – including connection/disconnection of the QuickLoc, connection/disconnection of the IIS cables, and attaching/removing the bullet nose assembly – is a maintenance requirement for design of the IHP shroud. The IHP must also provide support of the IIS cables.

The shroud assembly is required to provide radiation shielding of the control rod drive mechanism.

The shroud also minimizes the effects of external events such as jets from through-wall cracks in high- and moderate-energy pipes. The control rod drive mechanisms and small diameter piping, tubing and conduit within the shroud do not represent credible sources of missiles or jets due to breaks or cracks. Therefore, the shroud is not required to act as a missile shield to contain missiles generated within the integrated head package. It is also not required to deflect any jets originating within the integrated head package.

The cables and connectors, within the integrated head package, for the in-core instrumentation system are AP1000 equipment Class C, Class 1E. These cables are required to be physically and electrically independent of other cables including control rod drive mechanism power cables. Section 7.1 describes separation requirements. The cables and connector must be environmentally qualified, as discussed in Section 3.11. The cables are required to terminate at a connector plate

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located so that the cables can be readily connected or disconnected. The other cables within the integrated head package, including power cables and cables for the digital rod position indicator system, are not Class 1E.

The cable support provides seismic support and maintains separation for instrumentation and power cables.

3.9.7.2 Design Description

The integrated head package combines several separate components in one assembly to simplify refueling of the reactor. The purpose of the integrated head package is to reduce the outage time and personnel radiation exposure by combining operations associated with movement of the reactor vessel head during the refueling outage. In addition, the integrated head concept reduces the laydown space required in the containment. With the integrated head package, disconnections from and connections to the control rod drive mechanisms and rod position indicators (RPI) and other components within the cooling shroud assembly are not made at the individual component.

The integrated head package consists of the following main elements:

- Shroud assembly
- Lifting system
- Mechanism seismic support structure
- Cable support structure
- Cables
- In-core instrumentation
- Cable bridge

Brief descriptions of the principal elements of the integrated head package are provided in the following paragraphs.

Shroud assembly - The shroud assembly is a carbon steel structure that includes a shielding shroud and an air baffle. During normal operation, it directs the flow of cooling air to the control rod drive mechanism coil stacks. The rod position indicators are also cooled by this air flow. The duct work and air baffle are integral with, and supported by, the shroud assembly. The air cooling fans are attached to the IHP. Structurally, the shroud is integrated with the mechanism seismic support structure. The shroud also provides shielding at the vessel flange region.

The shroud structure is bolted to attachment lugs on the reactor vessel head.

Cabling, conduit and their supports and attachment hardware for the nonsafety-related control rod drive mechanism cables; digital rod position indication cables; miscellaneous power, control, and instrumentation cables; and safety-related in-core instrumentation cables are routed around the cable support attached to the shroud. The cable bridge attaches to the shroud.

Lifting system - This apparatus lifts the reactor vessel head and integrated head package as a unit. The lift legs transfer the head load during a head lift from the head attachment lugs to the lift rig. The lifting system consists of lift legs, sling block, clevises, and sling rods required to interface with the polar crane hook.

Mechanism seismic support structure - This structure provides seismic restraint for the control rod drive mechanisms. It is located near the top of the control rod drive mechanism rod travel housings. The digital rod position indication connector plate attached to the rod travel housing interfaces with spacer plates, where required, to form a system of bumpers that interface with the mechanism

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seismic support structure. This support interfaces with the shroud assembly to transfer seismic loads from the mechanisms to the reactor vessel head.

Cable support structure - The cable support is located at an elevation above the top of the rod travel housings. It provides permanent support and routing for the control rod drive mechanism power cables and rod position indication cables, which remain with the integrated head package and are normally not disturbed. These cables terminate at the connector plates, which constitute the interface with the mating cables. Cable disconnects are made at the connector plates.

Cables - The integrated head package cables include those portions of the control rod drive mechanism power cables, in-core instrumentation, and rod position indication instrumentation cables extending from the connector plates to the user devices. These cables remain with the integrated head package and are normally not disturbed. The individual cable length is sized to provide an orderly arrangement. A cable bridge spanning between the integrated head package to the operating deck is used to support the routing of integrated head package cables. A connector rack mounted to the end of the cable bridge is used to manage, support, and provide a means to quickly connect and disconnect the cables during the refueling outage.

In-core instrumentation – The in-core instrumentation system consists of thermocouples to measure fuel assembly coolant outlet temperature, and in-core flux thimbles containing fixed detectors for measurement of the neutron flux distribution within the reactor core. The incore thimble tubes have enhanced resistance to fluid-induced vibration and wear. The thimble is stiffer than the design in previous operating plants and the gap between the thimble tube and the tubes used to guide and protect the thimble inside the reactor vessel is smaller to minimize vibration. The design of the thimble tube assembly also precludes an non-isolable leak of reactor coolant. The thermocouples and neutron detectors are routed through the integrated head package. These are inserted into the core through the reactor vessel head and upper internals assembly.

Cable bridge – The cables from the integrated head package are routed in circumferential trays mounted to the shroud. Cables routed to those cable trays are routed via a cable bridge that attaches to the shroud. The cable bridge is raised to a vertical position during the outage and sits on the head stand during refueling as part of the integrated head package.

3.9.7.3 Design Evaluation

The components of the integrated head package, which provide seismic support including the control rod drive mechanism seismic support and the shroud, are designed using the ASME Code, Section III, Subsection NF. Because of the application of mechanistic pipe break evaluations, the supporting elements do not have to be designed for loads due to a postulated break in a reactor coolant loop pipe. Pipes down to 6-inch nominal diameter are evaluated using mechanistic pipe break criteria and the integrated head package is analyzed for movement of the reactor vessel due to a break of any pipe not qualified for leak-before-break. See Subsection 3.6.3 for a discussion of the mechanistic pipe break requirements.

The integrated head package satisfies the limit on deflection of the top of the control rod drive mechanism rod travel housing. This limit restricts the bending moments on the control rod drive mechanisms to less than those that produce stress levels in the pressure boundary of the control rod drive mechanism greater than ASME Code limits during anticipated transient or postulated accident conditions. This deflection limit provides that the rod travel housing does not bend to the extent that the drive rod binds during insertion of the control rods. This limit is based on the results of drive line drop testing with control rod drives travel housings in deflected positions.

The components of the integrated head package included in the load path of the lifting rig are designed to satisfy the requirements for lifting of heavy loads in NUREG-0612 (Reference 10). The

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criteria of ANSI N14.6, (Reference 11) is used to evaluate the loads and stresses during a lift. See Subsection 9.1.5 for discussion of special lifting rigs for heavy loads. Components which are part of the lifting load path are evaluated for the load due to the proof test required per ANSI N14.6.

Those cables and connectors for the in-core instrumentation system that are required to meet Class 1E requirements are evaluated for environmental conditions including normal operation and postulated accident conditions.

3.9.7.4 Inspection and Testing Requirements

The components in the lifting load path are proof tested to 300 percent of the rated load per the requirements of ANSI N14.6. The components load tested are surface examined by appropriate examination methods before and after the proof test.

3.9.8 Combined License Information

3.9.8.1 Reactor Internals Vibration Assessment and Predicted Response

The information consistent with the guidance of Regulatory Guide 1.20 is addressed in WCAP-16687-P (Reference 34), and the applicable changes are incorporated.

3.9.8.2 Design Specifications and Reports

As-designed design specifications and design reports for the major ASME Code, Section III components are available for NRC review. Design Specifications and selected design analysis information for ASME Code, Section III valves and auxiliary components are available for NRC review.

Design specifications and design reports for ASME Section III piping are made available for NRC review. Reconciliation of the as-built piping (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in Subsection 3.9.3.1.2) is completed by the COL holder after the construction of the piping systems and prior to fuel load (in accordance with DCD Tier 1 Section 2 ITAAC line item for the applicable systems).

The consistency of the reactor vessel core support materials relative to known issues of irradiation-assisted stress corrosion cracking or void swelling has been evaluated and addressed in APP-GW-GLR-035 (Reference 21).

3.9.8.3 Snubber Operability Testing

A program to verify operability of essential snubbers is addressed in Subsection 3.9.3.4.4.

3.9.8.4 Valve Inservice Testing

The inservice test program is addressed in Subsection 3.9.6.

3.9.8.5 Surge Line Thermal Monitoring

A monitoring program to record temperature distributions and thermal displacements of the surge line piping is addressed in Subsection 3.9.3.1.2.

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3.9.8.6 Piping Benchmark Program

The benchmark program as described in Subsection 3.9.1.2 (if a piping analysis computer program other than one of those used for design certification is used) is addressed in APP-GW-GLR-006 (Reference 35).

The piping benchmark problems identified in (Reference 20) for the Westinghouse AP600 are also representative for the AP1000 and can be used for the AP1000 piping benchmark program if required.

3.9.8.7 As-Designed Piping Analysis

The piping packages chosen to demonstrate the AP1000 piping design are identified to be any piping package containing Class 1 piping larger than 1 inch in diameter plus the additional packages identified in Table 3.9-20.

Note: Piping packages included in Table 3.9-20 are generally Class 2/3 lines, with the exception of APP-RCS-PLR-230, which contains the 1-inch Class 1 reactor head vent piping.

The as-designed piping analysis is provided for the piping lines chosen to demonstrate all aspects of the piping design. A design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class 1 piping using the methods and criteria outlined in Table 3.9-19 is made available for NRC review.

This topic is also addressed in Subsection 14.3.3, including the application of the completed as-designed piping analysis for the Reference Plant to subsequent standard AP1000 plants.

3.9.9 References

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24. Not used.

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26. Not used.

27. Not used.

28. Not used.

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32. Not used.

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Table 3.9-1 (Sheet 1 of 2)
Reactor Coolant System Design Transients

Event	Cycles
Level A Service Conditions	
Reactor coolant pump startup and shutdown (cycles of start and stop)	3000
Heatup at 100°F per hour	200
Cooldown at 100°F per hour	200
Unit loading between 0 and 15 percent of full power	500
Unit unloading between 0 and 15 percent of full power	500
Unit loading at 5 percent of full power per minute	2000
Unit unloading at 5 percent of full power per minute	2000
Step load increase of 10 percent of full power	3000
Step load decrease of 10 percent of full power	3000
Large step load decrease with steam dump	200
Steady-state fluctuation and load regulation	
Initial	1.5×10^5
Random	4.6×10^6
Load regulation	37,000(3)
Boron concentration equalization	2900
Feedwater cycling at hot shutdown	
Mode 1	3000
Mode 2	15,000
Core lifetime extension	40
Feedwater heaters out of service	180
Refueling	40
Turbine roll test	20
Primary-side leakage test	200
Secondary-side leakage test	80
Passive residual heat removal tests	5
Reactor coolant system makeup	5640
Daily load follow operation	17,800
Level B Service Conditions	
Loss of load (without reactor trip)	30
Loss of offsite power	30
Reactor trip from reduced power	180
Reactor trip from full power	
Case A - With no inadvertent cooldown	50
Case B - With cooldown and no safeguards actuation	50
Case C - With cooldown and safeguards actuation	20(1)

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Table 3.9-1 (Sheet 2 of 2)
Reactor Coolant System Design Transients

Event	Cycles
Level B Service Conditions	
Control rod drop	
Case A	30
Case B	20(2)
Case C	30(2)
Cold overpressure	15
Inadvertent safeguards actuation	10
Partial loss of reactor coolant flow	60
Inadvertent RCS depressurization	20
Excessive feedwater flow	30
Loss of offsite power - with natural circulation cooldown	
Case A - loss of power with natural circulation cooldown with onsite ac power	20
Case B - loss of power with natural circulation cooldown without onsite ac power	10
Loss of feedwater with non-safety PRHR HX modulation control	20
Level C Service Conditions	
Small loss of coolant accident	5
Small steam line break	5
Small feedwater line break	5
Steam generator tube rupture	5
Inadvertent opening of automatic depressurization system valves	15
Level D Service Conditions	
Reactor coolant pipe break (large loss-of-coolant accident)	1
Large steam line break	1
Large feedwater line break	1
Reactor coolant pump locked rotor	1
Control rod ejection	1
Test Conditions	
Primary side hydrotest	10
Secondary side hydrotest	10
Steam generator tube leakage test	
Secondary-side pressure, psig	
200	400
400	200
600	120
840	80

Notes:
1. Reactor Trip - Case C shares its number of occurrences (20) with Control Rod Drop Case B
2. Control Rod Drop - Case B shares its number of occurrences (20) with Reactor Trip Case C
3. The Load Regulation transient occurrences are limited to 37,000 to maintain main spray flow cycles within PRHA fatigue design limits of the pressurizer spray piping. Other RCS components continue to be designed for 750,000 Load Regulation transient occurrences.

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Table 3.9-2 Pump Starting/Stopping Conditions

Plant Condition	RCS (°F)/(psig)	SG Secondary (°F)/(psig)	Number of Starts/Stops	Operation
Cold	70/400	70/0	200	Cold Startup Transients
Cold	70/400	70/0	200	RCS heatup, cooldown
Restart	100/400	100/0	400	Hot functional RCP stops, starts
Hot	557/2235	557/1091	2200	Transients and miscellaneous

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Table 3.9-3 (Sheet 1 of 2)
Loadings for ASME Class 1, 2, 3, CS and Supports

Load	Description
P	Internal design pressure
PMAX	Peak pressure
DW	Dead weight
DML	Design Mechanical Loads (other than DW). This includes Service Level A loads and RVOS loads that are Service Level B.
XL	External mechanical loads, such as the nozzle reactions associated with piping systems, shall be combined with other loads in the loading combination expressions.
SSE	Safe shutdown earthquake (inertia portion)
E	Earthquake smaller than SSE (inertia portion)
FV	Fast valve closure
RVC	Relief/safety valve - closed system (transient)
RVOS	Relief/safety valve - open system (sustained)
RVOT	Relief/safety valve - open system (transient)
DY	Dynamic load associated with various service conditions including FV, RVC, and RVOT as applicable (transient)
DN	Dynamic load associated with Level A (Normal) service conditions including FV, RVC, and RVOT as applicable (transient)
DU	Dynamic load associated with Level B (Upset) service conditions including FV, RVC, and RVOT as applicable (transient)
DE	Dynamic load associated with Level C (Emergency) service conditions including FV, RVC, and RVOT as applicable (transient)
DF	Dynamic load associated with Level D (Faulted) service conditions during which, or following which, the piping system being evaluated must remain intact including FV, RVC, and RVOT as applicable. This includes postulated pipe rupture events (transient)
DYS	Dynamic load associated with various service conditions (sustained)
SSES	Seismic anchor motion portion of SSE
ES	Seismic anchor motion of earthquake smaller than SSE
TH	Thermal loads for the various service conditions

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Table 3.9-3 (Sheet 2 of 2)
Loadings for ASME Class 1, 2, 3, CS and Supports

Load	Description
TNU	Service Level A and B (normal and upset) plant condition thermal loads; including thermal stratification and thermal cycling
TN	Service Level A (normal) plant condition thermal loads
TU	Service Level B (upset) plant condition thermal loads
TE	Service Level C (emergency) plant condition thermal loads
TF	Service Level D (faulted) plant condition thermal loads
SCVNU	Static displacement of steel containment vessel - normal and upset conditions
SCVE	Static displacement of steel containment vessel - emergency condition
SCVF	Static displacement of steel containment vessel - faulted condition
HTDW	Hydrostatic test dead weight
DBPB	Design basis pipe break, includes LOCA and non-LOCA (transient)
LOCA	Loss-of-coolant accident
HYDSP	Building structure motions due to automatic depressurization system sparger discharge
DBPBS	Design basis pipe break, includes LOCA and non-LOCA (sustained)

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Table 3.9-4
Not Used

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Table 3.9-5
Minimum Design Loading Combinations For
ASME Class 1, 2, 3 And CS Systems And Components

Condition	Design Loading Combinations[3][6]
[*Design*	*P + DW + DML + XL*
Level A Service	*PMAX[1] + DW + XL[4]*
	PMAX + DW + DN + XL[8]
Level B Service	*PMAX + DW + DU + XL[8]*
Level C Service	*PMAX + DW + DE[5] + XL[8]*
	PMAX + DW + DY + HYDSP + XL[9]
Level D Service	*PMAX + DW + DF + XL[8]*
	PMAX + DW + SRSS[2] ((SSE + SSES) + DBPB)[7] + XL[4]
	PMAX + DW + RVOS + SRSS (SSE + SSES)[7] + XL[11]
	PMAX + DW + DYS + DBPBS + SRSS ((SSE + SSES)[7] + DY + HYDSP) + XL[9][10]]*

Notes:

1. The values of PMAX in the load combinations may be different for different levels of service conditions as provided in the design transients. For earthquake loadings PMAX is equal to normal operating pressure at 100% power.
2. SRSS equals the square root of the sum of the squares.
3. Appropriate loads due to static displacements of the steel containment vessel and building settlement should be added to the loading combinations expressions for ASME Code, Section III, Class 2 and 3 systems.
4. In combining loads, the timing and causal relationships that exist between PMAX, and XL, are considered for determination of the appropriate load combinations.
5. The pressurizer safety valve discharge is a Level C service condition.
6. See Table 3.9-3 for description of loads.
7. For components that behave as anchors to the piping system, such as equipment nozzles, SSE and SSES are combined by absolute sum. For other components, such as straight pipe, tees, and valves, SSE and SSES are combined by SRSS method.
8. In combining loads, the timing and causal relationships that exist between PMAX, DN, DU, DE, DF, and XL, are considered for determination of the appropriate load combinations.
9. In combining loads, the timing and causal relationships that exist between PMAX, DY, HYDSP, and XL, are considered for determination of the appropriate load combinations.
10. In combining loads, the timing and causal relationships that exist between PMAX, DY, and XL, are considered for determination of the appropriate load combinations.
11. In combining loads, the timing and causal relationships that exist between PMAX, RVOS, and XL, are considered for determination of the appropriate load combinations.

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Table 3.9-6
Additional Load Combinations and Stress Limits for ASME Class 1 Piping(9)

Condition	Loads(7)	Equation (NB3650)	Stress Limit
[*Level A/B*	*PMAX(1), TNU, E, ES, RVC, DN, DU,* *SCVNU(2)(4)(5)* *RVOS(2)*	*10* *11, 14*	*3.0 S_m* *CUF = 1.0*
	TNU	*12*	*3.0 S_m*
	PMAX + DW + DU	*13*	*3.0 S_m*
	PMAX + DW + RVOS(2)	*13*	*3.0 S_m*
Level C	*TE + SCVE*	*Note 3*	*Note 3*
Level D(8)	*SSES*	*F_{AM}/A_M(6)*	*1.0 S_m*
	TF + SCVF	*Note 3*	*Note 3*
	TNU + SSES	*C_2D_o (M1 + M2)/2I(8)*	*6.0 S_m]**

Notes:

1. The values of PMAX in the load combinations may be different for different levels of service conditions. For earthquake loading, PMAX is equal to normal operating pressure at 100% power.
2. Pressurizer safety valve discharge is classified as a Level C event.
3. See Table 3.9-11 for functional capability requirements.
4. The earthquake loads are assumed to occur at normal 100 percent power operation for the purposes of determining the total moment ranges.
5. Square root sum of the squares (SRSS) combination is used for ES, E, and other transient loads.
6. F_{AM} is amplitude of axial force for SSES; A_M is nominal pipe metal area.
7. See Table 3.9-3 for description of loads.
8. Where: M1 is range of moments for TNU, M2 is one half the range of SSES moments,
   M1 + M2 is larger of M1 plus one half the range of SSES, or full range of SSES
   C_2, D_o, I based on ASME III
9. When the simplified requirements of ASME Section III, NB-3640 and NB-3650 are not satisfied, the more detailed methods of NB-3200 may be used as stated in NB-3630.

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Table 3.9-7
Additional Load Combinations and Stress Limits for ASME Class 2, 3 Piping

Condition	Loads[3]	Equation (NC/ND3650)	Stress Limit
[*Level A/B*	*PMAX[1] + DW + TNU + SCVNU[4]*	*11*	$S_h + S_A$
	Building Settlement	*10a*	$3.0\ S_C$
Level C	*TE + SCVE[4]*	*Note 6*	*Note 6*
Level D	*TNU + SSES*	*i (M1 + M2)/Z[2]*	$3.0\ S_h$
	SSES	F_{AM}/A_M[5]	$1.0\ S_h$
	TF + SCVF[4]	*Note 6*	*Note 6*]*

Notes:

1. The values of PMAX in the load combinations may be different for different levels of service conditions. For earthquake loading PMAX is equal to normal operating pressure at 100% power.
2. Where: M1 is range of moments for TNU, M2 is one half the range of SSES moments, M1 + M2 is larger of M1 plus one half the range of SSES, or full range of SSES
3. See Table 3.9-3 for description of loads.
4. The timing and causal relationships among TNU, TE, TF, SCVNU, SCVE, and SCVF are considered to determine appropriate load combinations.
5. F_{AM} is amplitude of axial force for SSES; A_M is nominal pipe metal area.
6. See Table 3.9-11 for functional capability requirements.

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Table 3.9-8
Minimum Design Loading Combinations for Supports for ASME Class 1, 2, 3 Piping and Components(2)

Condition	Design Loading Combinations(3)
[*Design*	*DW + DML*
Level A Service	*DW + TH + DN*(4)
Level B Service	*DW + TH + DU*(4)
Level C Service	*DW + TH + DE*(5)(4)
	DW + TH + DY + HYDSP(7)
Level D Service	*DW + TH + RVOS + SSE + SSES + SWE*(6)(8)
	DW + TH + DF(4)
	DW + TH + SRSS (DBPB + (SSE + SSES + SWE))(6)
	DW + TH + DYS + DBPBS + SRSS ((SSE + SSES + SWE)(6) *+ DY + HYDSP)*(7)(9)
Hydrostatic Test	*HTDW*]*

Notes:

1. SRSS - square root of the sum of the squares
2. Appropriate loads due to static displacement of the steel containment vessel and building settlement should be added to the loading combinations expressions for Class 2 and 3 systems.
3. See Table 3.9-3 for description of loads.
4. The timing and causal relationships between TH and DY are considered to determine appropriate load combinations.
5. The pressurizer safety valve discharge is a Level C Service condition.
6. Combine SSE, SSES, and SWE by absolute sum method. SWE is self weight excitation, the effect of the acceleration of the support mass caused by building filtered loads such as SSE.
7. In combining loads, the timing and causal relationships that exist among TH, DY, and HYDSP are considered for determination of the appropriate load combinations.
8. In combining loads, the timing and causal relationships that exist among TH and RVOS are considered for determination of the appropriate load combinations.
9. In combining loads, the timing and causal relationships that exist among TH and DY are considered for determination of the appropriate load combinations.

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Table 3.9-9
Stress Criteria for ASME Code Section III
Class 1 Components[(a)] and Supports and Class CS Core Supports

Design/Service Level	Vessels/Tanks Pumps	Piping (h, i)	Core Supports	Valves, Disks & Seats	Components Supports [(c,d)]
Design and Service Level A	ASME Code, Section III NB-3221, 3222	[*ASME Code, Section III NB-3652, Equation 9*]*	ASME Code, Section III NG-3221, 3222, 3231, 3232	ASME Code, Section III NB-3520, 3525	[*ASME Code, Section III Subsection NF (e)*]*
Service Level B (Upset)	ASME Code, Section III NB-3223	[*ASME Code, Section III NB-3654, Equation 9*]*	ASME Code, Section III NG-3223, 3233	ASME Code, Section III NB-3525	[*ASME Code, Section III Subsection NF (e)*]*
Service Level C (Emergency)	ASME Code, Section III NB-3224	[*ASME Code, Section III NB-3655, Equation 9*]*	ASME Code, Section III NG-3224, 3234	ASME Code, Section III NB-3526	[*ASME Code, Section III Subsection NF (e)*]*
Service Level D (Faulted)	ASME Code, Section III (see Chapter 3.9.1.4) NB-3225 (no active Class 1 pumps used)	[*ASME Code, Section III NB-3656, Equation 9*]*	ASME Code, Section III (see chapter 3.9.1) NG-3225, 3235	(b) (g)	[*ASME Code, Section III Subsection NF, (e) (see Chapter 3.9.1) (f)*]*

Notes:

a. A test of the components may be performed in lieu of analysis.
b. Class 1 valve service Level D criteria for inactive valves is based on the criteria in ASME III, Appendix F, F-1420 for verification of pressure boundary integrity.
c. Including pipe supports.
d. In instances where the determination of allowable stress values utilizes S_u (ultimate tensile stress) at temperatures not included in ASME Code Section III, S_u shall be calculated using one of the methods provided in Regulatory Guide 1.124, Revision 1.
e. ASME Table 3131(a)-1.
f. See Subsection 3.9.3.4 for supports for active equipment, valves, and piping with active valves.
g. For active valves, pressure integrity verification will be based on using the ASME Code allowables one level less than the service loading condition. For example, for the evaluation of Level D loading, Level C allowables will be used. Valve operability is demonstrated by testing or analysis. Check valve operability may be shown by analysis. See Subsection 3.9.3.2.2 for an outline of test requirements.
h. Table 3.9-6 includes additional stress limits for Class 1 piping.
i. When the simplified requirements of ASME Section III, NB-3640 and NB-3650 are not satisfied, the more detailed methods of NB-3200 may be used as stated in NB-3630.

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Table 3.9-10
Stress Criteria for ASME Code Section III
Class 2 and 3 Components and Supports

Design/ Service Level	Vessels/Tanks	Piping (f)	Pumps	Valves, Disks, Seats	Component Supports (a) (b)
Design and Service Level A	ASME Code Section III NC-3217 NC/ ND-3310, 3320	[*ASME Code, Section III NC/ND-3652, Equation 8*]*	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3510	[*ASME Code Section III (c)*]*
Service Level B (Upset)	ASME Code Section III NC/ND-3310, 3320	[*ASME Code, Section III NC/ND-3653, Equation 9*]*	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520	[*ASME Code Section III (c)*]*
Service Level C (Emergency)	ASME Code Section III NC/ND-3310, 3320	[*ASME Code, Section III NC/ND-3654, Equation 9*]*	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520	[*ASME Code Section III (c)*]*
Service Level D (Faulted)	ASME Code Section III NC/ND-3310, 3320	[*ASME Code, Section III NC/ND-3655, Equation 9*]*	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520 (e)	[*ASME Code Section III (c) (d)*]*

See following page for notes.

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Notes:

a. Including pipe supports.
b. In instances where the determination of allowable stress values utilizes S_u (ultimate tensile stress) at temperatures not included in ASME Code Section III, S_u shall be calculated using one of the methods provided in Regulatory Guide 1.124, Revision 1.
c. ASME Table 3131(a)-1.
d. See Subsection 3.9.3.4 for supports for active equipment, valves, and piping with active valves.
e. For active valves, pressure integrity verification will be based on using the ASME Code allowables one level less than the service loading condition. For example, for the evaluation of Level D loading, Level C allowables will be used. Valve operability is demonstrated by testing or analysis. Check valve operability may be shown by analysis. See Subsection 3.9.3.2.2 for an outline of test requirements.
f. Table 3.9-7 includes additional stress limits for Class 2 and 3 piping.

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Table 3.9-11
Piping Functional Capability – ASME Class 1, 2, and 3(1)

[*Wall Thickness:*	$D_o/t \leq 50$, where D_o, t are per ASME III
Service Level D Conditions	Equation 9 ≤ smaller of 2.0 S_y and 3.0 S_m(2, 4, 5) Equation 9 ≤ smaller of 2.0 S_y and 3.0 S_h(3, 4, 6)
External Pressure:	$P_{external} \leq P_{internal}$
TE + SCVE	$C2*M*D_o/2I \leq 6.0\ S_m$(2) (NB-3650) Equation 10a (NC3653.2) ≤ 3.0 S_c(3)
TF + SCVF	$C2*M*D_o/2I \leq 6.0\ S_m$(2) (NB-3650) Equation 10a (NC 3653.2) ≤ 3.0 S_c(3)]*

Notes:

1. Applicable to Level C or Level D plant events for which the piping system must maintain an adequate fluid flow path
2. Applicable to ASME Code Class 1 piping. When the simplified requirements of ASME Section III, NB-3640 and NB-3650 are not satisfied, the more detailed methods of NB-3200 may be used to calculate stress. The allowables listed above shall be met.
3. Applicable to ASME Code Class 2 and 3 piping
4. Applicable to ASME Code Class 1, 2 and 3 piping when the following limitations are met:
   - 4.1 Dynamic loads are reversing (slug-flow water hammer loads are non-reversing)
   - 4.2 Slug-flow water-hammer loads are combined with other design basis loads (for example: SSE; pipe break loads)
   - 4.3 Steady-state bending stress from deadweight loads does not exceed:

$$\frac{B2*M}{Z} \leq 0.25\,S_Y$$

   - 4.4 When elastic response spectrum analysis is used, dynamic moments are calculated using 15% peak broadening and not more than 5% damping
5. For Class 1 piping, when slug-flow water hammer loads are only combined with pressure, weight and other sustained mechanical loads, the Equation 9 stress does not exceed the smaller of 1.8 S_y and 2.25 S_m.
6. For Class 2 and 3 piping, when slug-flow water hammer loads are only combined with pressure, weight and other sustained mechanical loads, the Equation 9 stress does not exceed the smaller of 1.8 S_y and 2.25 S_h.

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Table 3.9-12 (Sheet 1 of 7)
List of ASME Class 1, 2, and 3 Active Valves

Valve No.	Description	Function[a]
Compressed and Instrument Air System		
CAS-PL-V014	Instrument Air Supply Outside Containment Isolation	2
CAS-PL-V015	Instrument Air Supply Inside Containment Isolation Check Valve	2
CAS-PL-V205	Service Air Supply Inside Containment Isolation Check Valve	2
Component Cooling Water System		
CCS-PL-V200	Containment Isolation Valve – Inlet Line Isolation	2, 3
CCS-PL-V201	Containment Isolation Valve – Inlet Line Check Valve	2, 3
CCS-PL-V207	Containment Isolation Valve – Outlet Line Isolation	2, 3
CCS-PL-V208	Containment Isolation Valve – Outlet Line Isolation	2, 3
CCS-PL-V220	Containment Isolation Thermal Relief Valve	2
CCS-PL-V270	CCS IRC Relief Valve	3
CCS-PL-V271	CCS IRC Relief Valve	3
Chemical and Volume Control System		
CVS-PL-V001	Reactor Coolant System Purification Stop	1
CVS-PL-V002	Reactor Coolant System Purification Stop	1
CVS-PL-V003	Reactor Coolant System Purification Stop	1
CVS-PL-V042	Flush Line Containment Isolation Relief	2
CVS-PL-V045	Letdown Containment Isolation IRC	2
CVS-PL-V047	Letdown Containment Isolation ORC	2
CVS-PL-V058	Letdown Line Containment Isolation Thermal Relief	2
CVS-PL-V067	Makeup Return Line Bypass Check Valve	1
CVS-PL-V080	Reactor Coolant System Purification Return Line Check Valve	1
CVS-PL-V081	Reactor Coolant System Purification Return Line Stop Valve	1
CVS-PL-V082	Reactor Coolant System Purification Return Line Check Valve	1
CVS-PL-V084	Auxiliary Pressurizer Spray Line Isolation	1
CVS-PL-V085	Auxiliary Pressurizer Spray Line Check Valve	1
CVS-PL-V090	Makeup Line Containment Isolation	2
CVS-PL-V091	Makeup Line Containment Isolation	2
CVS-PL-V092	Zinc Injection Containment Isolation Valve ORC	2
CVS-PL-V094	Zinc Injection Containment Isolation Valve IRC	2
CVS-PL-V098	Zinc Addition Line Containment Isolation Thermal Relief Valve	2
CVS-PL-V100	Makeup Line Containment Isolation Thermal Relief Check Valve	2
CVS-PL-V136A	Demineralized Water System Isolation	3
CVS-PL-V136B	Demineralized Water System Isolation	3
CVS-PL-V217	Hydrogen Injection Containment Isolation Check Valve IRC	2
CVS-PL-V219	Hydrogen Injection Containment Isolation Valve ORC	2
Demineralized Water System		
DWS-PL-V241	DWS Containment Penetration Thermal Relief Valve	3
DWS-PL-V245	Demineralized Water Supply Inside Containment Isolation Check Valve	2
Fuel Handing System		
FHS-PL-V001	Fuel Transfer Tube Isolation Valve	3

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Table 3.9-12 (Sheet 2 of 7)
List of ASME Class 1, 2, and 3 Active Valves

Valve No.	Description	Function[a]
Fire Protection System		
FPS-PL-V052	Fire Water Supply Inside Containment Isolation Check Valve	2
FPS-PL-V702	FPS Containment Penetration Thermal Relief Valve	3
Passive Containment Cooling System		
PCS-PL-V001A	Passive Containment Cooling Water Storage Tank Isolation	3,4
PCS-PL-V001B	Passive Containment Cooling Water Storage Tank Isolation	3,4
PCS-PL-V001C	Passive Containment Cooling Water Storage Tank Isolation	3,4
PCS-PL-V002A	Passive Containment Cooling Water Storage Tank Series Isolation	3,4
PCS-PL-V002B	Passive Containment Cooling Water Storage Tank Series Isolation	3,4
PCS-PL-V002C	Passive Containment Cooling Water Storage Tank Series Isolation	3,4
PCS-PL-V005	Passive Containment Cooling Water Storage Tank Supply to Fire Protection System Isolation Valve	3,4
PCS-PL-V009	Spent Fuel Pool Emergency Makeup Isolation	3
PCS-PL-V015	Water Bucket Makeup Line Drain Valve	3,4
PCS-PL-V020	Water Bucket Makeup Line Isolation Valve	3,4
PCS-PL-V023	PCS Recirculation Return Isolation Valve	3,4
PCS-PL-V039	PCCWST Long-Term Makeup Check Valve	3,4
PCS-PL-V042	PCCWST Long Term Makeup Isolation Drain Valve	3,4
PCS-PL-V044	PCCWST Long Term Makeup Isolation Valve	3,4
PCS-PL-V045	Emergency Makeup to the Spent Fuel Pool Isolation Valve	3
PCS-PL-V046	PCCWST Recirculation Return Isolation Valve	3,4
PCS-PL-V049	Emergency Makeup to the Spent Fuel Pool Drain Isolation Valve	3
PCS-PL-V050	Spent Fuel Pool Long Term Makeup Isolation Valve	3
PCS-PL-V051	Spent Fuel Pool Emergency Makeup Lower Isolation Valve	3
Primary Sampling System		
PSS-PL-V008	Containment Isolation – Containment Air Sample Isolation	2
PSS-PL-V010A	Containment Isolation – Liquid Sample Line	2
PSS-PL-V010B	Containment Isolation – Liquid Sample Line	2
PSS-PL-V011A	Containment Isolation – Liquid Sample Line	2
PSS-PL-V011B	Containment Isolation – Liquid Sample Line	2
PSS-PL-V023	Containment Isolation – Sample Return Line	2
PSS-PL-V024	Containment Isolation – Sample Return Line	2
PSS-PL-V046	Containment Isolation – Air Sample Line	2
Potable Water System		
PWS-PL-V418	PWS MCR Isolation Valve	3
PWS-PL-V420	PWS MCR Isolation Valve	3
PWS-PL-V498	PWS MCR Vacuum Relief	3
Passive Core Cooling System		
PXS-PL-V014A	Core Makeup Tank A Discharge Isolation	3,4
PXS-PL-V014B	Core Makeup Tank B Discharge Isolation	3,4

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Table 3.9-12 (Sheet 3 of 7)
List of ASME Class 1, 2, and 3 Active Valves

Valve No.	Description	Function[a]
Passive Core Cooling System (Cont.)		
PXS-PL-V015A	Core Makeup Tank A Discharge Isolation	3,4
PXS-PL-V015B	Core Makeup Tank B Discharge Isolation	3,4
PXS-PL-V016A	Core Makeup Tank A Discharge Check	3,4
PXS-PL-V016B	Core Makeup Tank Discharge Check	3,4
PXS-PL-V017A	Core Makeup Tank A Discharge Check	3,4
PXS-PL-V017B	Core Makeup Tank B Discharge Check	3,4
PXS-PL-V022A	Accumulator A Pressure Relief	3
PXS-PL-V022B	Accumulator B Pressure Relief	3
PXS-PL-V028A	Accumulator A Discharge Check	1,3,4
PXS-PL-V028B	Accumulator B Discharge Check	1,3,4
PXS-PL-V029A	Accumulator A Discharge Check	1,3,4
PXS-PL-V029B	Accumulator B Discharge Check	1,3,4
PXS-PL-V042	Nitrogen Supply Containment Isolation ORC Isolation Valve	2
PXS-PL-V043	Nitrogen Supply Containment Isolation IRC Check Valve	2
PXS-PL-V108A	Passive Residual Heat Removal Heat Exchanger Control	3,4
PXS-PL-V108B	Passive Residual Heat Removal Heat Exchanger Control	3,4
PXS-PL-V118A	Recirculation Sump A Isolation	3,4
PXS-PL-V118B	Recirculation Sump B Isolation	3,4
PXS-PL-V119A	Recirculation Sump A Check	3,4
PXS-PL-V119B	Recirculation Sump B Check	3,4
PXS-PL-V120A	Recirculation Sump A Isolation	3,4
PXS-PL-V120B	Recirculation Sump B Isolation	3,4
PXS-PL-V122A	In-Containment Refueling Water Storage Tank Injection A Check	1,3,4
PXS-PL-V122B	In-Containment Refueling Water Storage Tank Injection B Check	1,3,4
PXS-PL-V123A	In-Containment Refueling Water Storage Tank Injection A Isolation	1,3,4
PXS-PL-V123B	In-Containment Refueling Water Storage Tank Injection B Isolation	1,3,4
PXS-PL-V124A	In-Containment Refueling Water Storage Tank Injection A Check	1,3,4
PXS-PL-V124B	In-Containment Refueling Water Storage Tank Injection B Check	1,3,4
PXS-PL-V125A	In-Containment Refueling Water Storage Tank Injection A Isolation	1,3,4
PXS-PL-V125B	In-Containment Refueling Water Storage Tank Injection B Isolation	1,3,4
PXS-PL-V130A	In-Containment Refueling Water Storage Tank Gutter Isolation	3,4
PXS-PL-V130B	In-Containment Refueling Water Storage Tank Gutter Isolation	3,4
Reactor Coolant System		
RCS-PL-V001A	First Stage Automatic Depressurization System	1,3,4
RCS-PL-V001B	First Stage Automatic Depressurization System	1,3,4
RCS-PL-V002A	Second Stage Automatic Depressurization System	1,3,4
RCS-PL-V002B	Second Stage Automatic Depressurization System	1,3,4

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Table 3.9-12 (Sheet 4 of 7)
List of ASME Class 1, 2, and 3 Active Valves

Valve No.	Description	Function[a]
Reactor Coolant System (Cont.)		
RCS-PL-V003A	Third Stage Automatic Depressurization System	1,3,4
RCS-PL-V003B	Third Stage Automatic Depressurization System	1,3,4
RCS-PL-V004A	Fourth Stage Automatic Depressurization System	1,3,4
RCS-PL-V004B	Fourth Stage Automatic Depressurization System	1,3,4
RCS-PL-V004C	Fourth Stage Automatic Depressurization System	1,3,4
RCS-PL-V004D	Fourth Stage Automatic Depressurization System	1,3,4
RCS-PL-V005A	Pressurizer Safety Valve	1,3
RCS-PL-V005B	Pressurizer Safety Valve	1,3
RCS-PL-V010A	Automatic Depressurization System Discharge Header A Vacuum Relief	3
RCS-PL-V010B	Automatic Depressurization System Discharge Header B Vacuum Relief	3
RCS-PL-V011A	First Stage Automatic Depressurization System Isolation	1,3,4
RCS-PL-V011B	First Stage Automatic Depressurization System Isolation	1,3,4
RCS-PL-V012A	Second Stage Automatic Depressurization System Isolation	1,3,4
RCS-PL-V012B	Second Stage Automatic Depressurization System Isolation	1,3,4
RCS-PL-V013A	Third Stage Automatic Depressurization System Isolation	1,3,4
RCS-PL-V013B	Third Stage Automatic Depressurization System Isolation	1,3,4
RCS-PL-V150A	Reactor Vessel Head Vent	1,3
RCS-PL-V150B	Reactor Vessel Head Vent	1,3
RCS-PL-V150C	Reactor Vessel Head Vent	1,3
RCS-PL-V150D	Reactor Vessel Head Vent	1,3
Normal Residual Heat Removal System		
RNS-PL-V001A	Reactor Coolant System Inner HL Suction Isolation	1
RNS-PL-V001B	Reactor Coolant System Inner HL Suction Isolation	1
RNS-PL-V002A	Reactor Coolant System Outer HL Suction Isolation	1,2
RNS-PL-V002B	Reactor Coolant System Outer HL Suction Isolation	1,2
RNS-PL-V003A	Reactor Coolant System Pressure Boundary Valve Thermal Relief Check Valve	1
RNS-PL-V003B	Reactor Coolant System Pressure Boundary Valve Thermal Relief Check Valve	1
RNS-PL-V011	RNS Discharge Containment Isolation Valve	2, 4
RNS-PL-V012	RNS Discharge Containment Isolation Test Connection Valve	2, 4
RNS-PL-V013	RNS Discharge Containment Isolation Check Valve	2, 4
RNS-PL-V015A	RNS Discharge Reactor Coolant System Pressure Boundary	1, 4
RNS-PL-V015B	RNS Discharge Reactor Coolant System Pressure Boundary	1, 4
RNS-PL-V017A	RNS Discharge Reactor Coolant System Pressure Boundary	1, 4
RNS-PL-V017B	RNS Discharge Reactor Coolant System Pressure Boundary	1, 4
RNS-PL-V020	RNS HL Suction Pressure Relief	2
RNS-PL-V021	RNS HL Suction Pressure Relief	2

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Table 3.9-12 (Sheet 5 of 7)
List of ASME Class 1, 2, and 3 Active Valves

Valve No.	Description	Function[a]
Normal Residual Heat Removal System (Cont.)		
RNS-PL-V022	RNS Suction Header Containment Isolation	2
RNS-PL-V023	RNS Suction from In-Containment Refueling Water Storage Tank Isolation	2
RNS-PL-V061	RNS – Chemical and Volume Control System Containment Isolation	2
Sanitary Drainage System		
SDS-PL-V001	SDS MCR Vent Isolation Valve	3
SDS-PL-V002	SDS MCR Vent Isolation Valve	3
Spent Fuel Pool Cooling System		
SFS-PL-V034	Spent Fuel Pool Cooling System Suction Line Containment Isolation	2
SFS-PL-V035	Spent Fuel Pool Cooling System Suction Line Containment Isolation	2
SFS-PL-V037	Spent Fuel Pool Cooling System Discharge Line Containment Isolation	2
SFS-PL-V038	Spent Fuel Pool Cooling System Discharge Line Containment Isolation	2
SFS-PL-V042	Cask Loading Pit to SFS Pump Suction Isolation Valve	3
SFS-PL-V045	SFS Pump Discharge Line to Cask Loading Pit Isolation Valve	3
SFS-PL-V049	Cask Loading Pit to WLS Isolation Valve	3
SFS-PL-V066	Spent Fuel Pool to Cask Washdown Pit Isolation	3
SFS-PL-V067	Containment Isolation Thermal Relief Valve	2
SFS-PL-V068	Cask Washdown Pit Drain Isolation	3
SFS-PL-V071	Refueling Cavity to Steam Generator Compartment	3
SFS-PL-V072	Refueling Cavity to Steam Generator Compartment	3
Steam Generator System		
SGS-PL-V027A	Power Operated Relief Valve Block Valve Steam Generator 01	2,3,4
SGS-PL-V027B	Power Operated Relief Valve Block Valve Steam Generator 02	2,3,4
SGS-PL-V030A	Main Steam Safety Valve Steam Generator 01	2,3,4
SGS-PL-V030B	Main Steam Safety Valve Steam Generator 02	2,3,4
SGS-PL-V031A	Main Steam Safety Valve Steam Generator 01	2,3,4
SGS-PL-V031B	Main Steam Safety Valve Steam Generator 02	2,3,4
SGS-PL-V032A	Main Steam Safety Valve Steam Generator 01	2,3,4
SGS-PL-V032B	Main Steam Safety Valve Steam Generator 02	2,3,4
SGS-PL-V033A	Main Steam Safety Valve Steam Generator 01	2,3,4
SGS-PL-V033B	Main Steam Safety Valve Steam Generator 02	2,3,4
SGS-PL-V034A	Main Steam Safety Valve Steam Generator 01	2,3,4
SGS-PL-V034B	Main Steam Safety Valve Steam Generator 02	2,3,4
SGS-PL-V035A	Main Steam Safety Valve Steam Generator 01	2,3,4
SGS-PL-V035B	Main Steam Safety Valve Steam Generator 02	2,3,4
SGS-PL-V036A	Steam Line Condensate Drain Isolation	2,3,4
SGS-PL-V036B	Steam Line Condensate Drain Isolation	2,3,4

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Table 3.9-12 (Sheet 6 of 7)
List of ASME Class 1, 2, and 3 Active Valves

Valve No.	Description	Function[a]
Steam Generator System (Cont.)		
SGS-PL-V040A	Main Steam Line Isolation	2,3,4
SGS-PL-V040B	Main Steam Line Isolation	2,3,4
SGS-PL-V057A	Main Feedwater Isolation	2,3,4
SGS-PL-V057B	Main Feedwater Isolation	2,3,4
SGS-PL-V067A	Startup Feedwater Isolation	2,3,4
SGS-PL-V067B	Startup Feedwater Isolation	2,3,4
SGS-PL-V074A	Steam Generator Blowdown Isolation	2,3,4
SGS-PL-V074B	Steam Generator Blowdown Isolation	2,3,4
SGS-PL-V075A	Steam Generator Series Blowdown Isolation	3,4
SGS-PL-V075B	Steam Generator Series Blowdown Isolation	3,4
SGS-PL-V086A	Steam Line Condensate Drain Control	3,4
SGS-PL-V086B	Steam Line Condensate Drain Control	3,4
SGS-PL-V233A	Power Operated Relief Valve	3,4
SGS-PL-V233B	Power Operated Relief Valve	3,4
SGS-PL-V240A	Main Steam Isolation Valve Bypass Isolation	2,3,4
SGS-PL-V240B	Main Steam Isolation Valve Bypass Isolation	2,3,4
SGS-PL-V250A	Main Feedwater Control	3,4
SGS-PL-V250B	Main Feedwater Control	3,4
SGS-PL-V255A	Startup Feedwater Control	3,4
SGS-PL-V255B	Startup Feedwater Control	3,4
SGS-PL-V257A	Main Feedwater Thermal Relief Valve	3,4
SGS-PL-V257B	Main Feedwater Thermal Relief Valve	3,4
SGS-PL-V258A	Startup Feedwater Thermal Relief Valve	3,4
SGS-PL-V258B	Startup Feedwater Thermal Relief Valve	3,4
Nuclear Island Nonradioactive Ventilation System		
VBS-PL-V186	MCR Supply Air Isolation Valve	3
VBS-PL-V187	MCR Supply Air Isolation Valve	3
VBS-PL-V188	MCR Return Air Isolation Valve	3
VBS-PL-V189	MCR Return Air Isolation Valve	3
VBS-PL-V190	MCR Exhaust Air Isolation Valve	3
VBS-PL-V191	MCR Exhaust Air Isolation Valve	3
Main Control Room Habitability System		
VES-PL-V001	Air Delivery Alternate Isolation Valve	3
VES-PL-V002A	Pressure Regulating Valve A	3
VES-PL-V002B	Pressure Regulating Valve B	3
VES-PL-V005A	Air Delivery Isolation Valve A	3
VES-PL-V005B	Air Delivery Isolation Valve B	3
VES-PL-V018	Temporary Instrument Isolation Valve A	3
VES-PL-V019	Temporary Instrument Isolation Valve B	3
VES-PL-V022A	Pressure Relief Isolation Valve A	3
VES-PL-V022B	Pressure Relief Isolation Valve B	3
VES-PL-V040A	Air Tank Safety Relief Valve A	3
VES-PL-V040B	Air Tank Safety Relief Valve B	3

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Table 3.9-12 (Sheet 7 of 7)
List of ASME Class 1, 2, and 3 Active Valves

Valve No.	Description	Function[a]
Main Control Room Habitability System (Cont.)		
VES-PL-V040C	Air Tank Safety Relief Valve C	3
VES-PL-V040D	Air Tank Safety Relief Valve D	3
VES-PL-V044	Main Air Flow Path Isolation Valve	3
VES-PL-V045	Eductor Flow Path Isolation Valve	3
VES-PL-V046	Eductor Bypass Isolation Valve	3
Containment Air Filtration System		
VFS-PL-V003	Containment Purge Inlet Containment Isolation Valve	2
VFS-PL-V004	Containment Purge Inlet Containment Isolation Valve	2
VFS-PL-V009	Containment Purge Discharge Containment Isolation Valve	2
VFS-PL-V010	Containment Purge Discharge Containment Isolation Valve	2
VFS-PL-V800A	Vacuum Relief Containment Isolation A ORC	2
VFS-PL-V800B	Vacuum Relief Containment Isolation B ORC	2
VFS-PL-V803A	Vacuum Relief Containment Isolation Check Valve A IRC	2
VFS-PL-V803B	Vacuum Relief Containment Isolation Check Valve B IRC	2
Central Chilled Water System		
VWS-PL-V053	VWS Containment Penetration Thermal Relief Valve	3
VWS-PL-V057	VWS Containment Penetration Thermal Relief Valve	3
VWS-PL-V058	Fan Coolers Supply Containment Isolation	2
VWS-PL-V062	Fan Coolers Supply Containment Isolation Check Valve	2
VWS-PL-V080	Containment Isolation Thermal Relief Valve	2
VWS-PL-V082	Fan Coolers Return Containment Isolation	2
VWS-PL-V086	Fan Coolers Return Containment Isolation	2
Liquid Radwaste System		
WLS-PL-V055	Sump Containment Isolation IRC	2
WLS-PL-V057	Sump Containment Isolation ORC	2
WLS-PL-V058	Containment Isolation Relief Valve	2
WLS-PL-V067	Reactor Coolant Drain Tank Gas Containment Isolation IRC	2
WLS-PL-V068	Reactor Coolant Drain Tank Gas Containment Isolation ORC	2
WLS-PL-V071A	Chemical and Volume Control System Compartment to Sump	3
WLS-PL-V071B	Passive Core Cooling System A Compartment to Sump	3
WLS-PL-V071C	Passive Core Cooling System B Compartment to Sump	3
WLS-PL-V072A	Chemical and Volume Control System Compartment to Sump	3
WLS-PL-V072B	Passive Core Cooling System A Compartment to Sump	3
WLS-PL-V072C	Passive Core Cooling System B Compartment to Sump	3

Note:

a. Function: 1 – Reactor coolant system pressure boundary
2 – Containment isolation
3 – Accident mitigation
4 – Safe shutdown

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Table 3.9-13
Control Rod Drive Mechanism Production Tests

Test	Acceptance Standard
Cold (ambient) hydrostatic	ASME Code, Section III
Confirm step length and load transfer (stationary gripper to movable gripper or movable gripper to stationary gripper)	Step length: 0.625 ± 0.015 inch axial movement Load transfer: 0.055 inch nominal axial movement
Performance test at design load - Cycled under ambient water temperature conditions for approximately one full drive rod travel excursion. Cycled under hot test conditions for approximately 5,000 steps.	Operating speed: 45 inches/minute Trip delay: Free fall of drive rod to begin within 150 milliseconds

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Table 3.9-14
Maximum Deflections Allowed for
Reactor Internal Support Structures

Component	Allowable Deflections (in.)
Upper Core Barrel	
Radial inward (uniform)	4.1
Radial outward (uniform)(1)	1.0
Upper package – relative vertical motion between upper core plate and upper support plate	0.20
Rod cluster guide tubes – radial toward the reactor vessel outlet	1.00

Note:
1. Non-uniform radial outward deflections are limited such that > 90-percent of the annulus area is maintained.

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Table 3.9-15
Computer Programs for Seismic Category 1 Components

Program	Application
ABAQUS	Finite element structural analysis
ANSYS	Finite element structural analysis
FATCON	ASME fatigue analysis of piping components
GAPPIPE	Static and dynamic analysis of piping systems
MAXTRAN	Transient stress evaluation of piping components
WESTEMS	ASME Class 1 fatigue analysis of piping and components
PIPESTRESS	Static and dynamic analysis of piping systems
PIPSAN	Structural and ASME stress analysis of piping and component supports
GTSTRUDL	Structural and ASME stress analysis of piping and component supports and structural frames
STAAD-III	Static and dynamic analysis of structural frames
THERST	Transient heat transfer analysis of piping components
WECAN	Finite element structural analysis
WEGAP	Dynamic structural response of the reactor core
WECEVAL	ASME stress evaluation of mechanical components
ITCH	Transient hydraulic analysis
FORFUN	Computes unbalanced hydraulic forces between piping elbows
RELAP5	Transient dynamic analysis
THRUST	Computes time-history hydraulic forcing functions
MULTIFLEX	Thermal-hydraulic-structural system analysis
GEC2	Computes time-history hydraulic forcing functions
FATSTR	ASME stress evaluation of piping components
HSTA	Hydraulic system transient analysis
E0781	Axisymmetric containment shell analysis
FLOW 3D	Finite element fluid flow and heat transfer

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Table 3.9-16
Not Used

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Table 3.9-17
System Level Operability Test Requirements

System/Feature	Test Purpose	Test Method	Tech Spec[a]
PCS			
PCCWST drain lines	Flow capability and water coverage	Note 1	SR 3.6.6.6
PXS			
Accumulator injection lines	Flow capability	Note 2	SR 3.5.1.6
CMT injection lines	Flow capability	Note 3	SR 3.5.2.7
PRHR HX	Heat transfer capability	Note 4	SR 3.5.4.9
IRWST injection lines	Flow capability	Note 5	SR 3.5.6.11
Containment recirculation lines	Flow capability	Note 6	SR 3.5.6.11

Alpha Note:

a. Refer to the Technical Specification surveillance identified in this column for the test frequency.

Notes:

1. The flow capability of each PCS water drain line is demonstrated by conducting a test where water is drained from the PCS water storage tank onto the containment shell by opening two of the three parallel isolation valves. During this flow test the water coverage is also demonstrated. The test is terminated when the flow measurement is obtained and the water coverage is observed. The test demonstrates that the PCS delivers water from the PCCWST to the outside, top of the containment vessel, such that a) the minimum allowable flowrate is 469.1 gpm with the passive containment cooling water storage tank level 27.5 feet (nominal) above the tank floor, or b) a report exists and concludes that the as-measured flow rates delivered by the PCCWST to the containment vessel provide sufficient heat removal capability such that the limiting containment pressure and temperature values are not affected and the PCS is able to perform its safety function to remove heat from containment to maintain plant safety (see item 7.a)(i) of ITAAC Table 2.2.2-3). The test may be run with a higher water level and the test results adjusted for the increased level. Water coverage is demonstrated by visual inspection that there is unobstructed flow from the lower weirs. In addition, air baffle panels may be removed at the containment vessel spring line, if needed, to permit visual inspection of the water coverage and the vessel coating. The water coverage observed at four locations, approximately 90 degrees apart, will be compared against the coverage measured at the same locations during pre-operational testing (see item 7.b)(i) of ITAAC Table 2.2.2-3).
2. The flow capability of each accumulator is demonstrated by conducting a test during cold shutdown conditions. The initial conditions of the test include reduced accumulator pressure. Flow from the accumulator to the RCS is initiated by opening the accumulator isolation valve. Sufficient flow is provided to fully open the check valves. The test is terminated when the flow measurement is obtained. The allowable calculated flow resistance between each accumulator and the reactor vessel is $\geq 1.47 \times 10^{-5}$ ft/gpm^2 and $\leq 1.83 \times 10^{-5}$ ft/gpm^2.
3. The flow capability of each CMT is demonstrated by conducting a test during cold shutdown conditions. The initial conditions of the test include the RCS loops drained to a level below the top of the RCS hot leg. Flow from the CMT to the RCS is initiated by opening both CMT discharge isolation valves. The test is terminated when the flow measurement is obtained. The allowable calculated flow resistance between each CMT and the reactor vessel is $\geq 1.81 \times 10^{-5}$ ft/gpm^2 and $\leq 2.25 \times 10\text{-}^{5}$ ft/gpm^2.
4. The heat transfer capability of the passive residual heat exchanger is demonstrated by conducting a test during shutdown conditions. The test is conducted with the RCPs in operation and the RCS at a reduced temperature. Due to the low RCS temperature, RCP limitations require the test to be run at 50-percent RCP speed. Flow through the heat exchanger is initiated by opening one outlet isolation valve. The test is terminated when the flow and temperature measurements are obtained. The allowable calculated heat transfer is ≥ 8.46E7-Btu/hr (50-percent pump speed) with an inlet temperature of 250°F and an initial IRWST temperature of 80°F and the design basis number of tubes plugged. The heat transfer rate measured in the test should be adjusted to account for differences in the hot leg and IRWST temperatures and the number of tubes plugged.
5. The flow capability of each IRWST injection line is demonstrated by conducting flow tests and inspections. A flow test is conducted to demonstrate the flow capability of the injection line from the IRWST through the IRWST injection check valves. Water flow from the IRWST through the IRWST injection check valve demonstrates the flow capability of this portion of the line. Sufficient flow is provided to open the check valves. The test is terminated when the flow measurement is obtained. The allowable calculated flow resistance from the IRWST through each direct vessel injection flow path to the reactor vessel is: Line A: $\geq 5.35 \times 10^{-6}$ ft/gpm^2 and $\leq 9.09 \times 10^{-6}$ ft/gpm^2 and Line B: $\geq 6.15 \times 10^{-6}$ ft/gpm^2 and $\leq 1.05 \times 10^{-5}$ ft/gpm^2. The flow capability of the portion of the line from the IRWST check valves to the DVI line is demonstrated by conducting an inspection of the inside of the line. The inspection shows that the lines are not obstructed. It is not necessary to operate the IRWST injection squib valves for this inspection.
6. The flow capability of each containment recirculation line is demonstrated by conducting an inspection. The line from the containment to the containment recirculation squib valve is inspected from the containment side. The line from the squib valve to the IRWST injection line is inspected from the IRWST side. The inspection shows that the lines are not obstructed. It is not necessary to operate the containment recirculation squib valves for this inspection.

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Table 3.9-18
AP1000 Pressure Isolation Valves

Valve Number	Description
PXS-V028A PXS-V028B PXS-V029A PXS-V029B	Accumulator Discharge Check Valves
RNS-V001A RNS-V001B RNS-V002A RNS-V002B	RNS Hot Leg Suction Isolation Valves
RNS-V015A RNS-V015B RNS-V017A RNS-V017B	RNS Discharge RCS Pressure Boundary

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Table 3.9-19 (Sheet 1 of 2)
Critical Piping Design Methods and Criteria (Piping Design Criteria)

Commitment	Tier 2 Reference (Section or Table)
ASME Code and Code Cases for AP1000 Piping and Pipe Support Design	Table 3.9-9, Table 3.9-10, 5.2.1.1, 5.2.1.2, Table 5.2-3
Analysis Methods: experimental stress analysis, independent support motion, inelastic analysis, non-seismic/seismic interaction, buried piping	3.7.3.9, 3.7.3.12, 3.7.3.13, 3.9.1.3, 3.9.3.1.5
Piping Modeling: piping benchmark program, decoupling criteria	3.6.2.1.1.1, 3.6.2.1.1.2, 3.6.2.1.1.3, 3.7.3.8.2.1, 3.9.1.2
Pipe Stress Analysis Criteria: loading and load combinations, damping values, combination of modal responses, high frequency modes, thermal oscillations in piping connected to the reactor coolant system, thermal stratification, safety-related valve design, installation and testing, functional capability, combination of inertial and seismic motion effects, welded attachments, modal damping for composite structures, minimum temperature for thermal analysis	3.6.2.2, 3.6.3.3, 3.7.2.14, 3.7.3.2, 3.7.3.7, 3.7.3.8.2.1, 3.7.3.9, Table 3.7.1-1, 3.9.3.1.2, 3.9.3.1.5, 3.9.3.3, Table 3.9-5, Table 3.9-6, Table 3.9-7, Table 3.9-8, Table 3.9-9, Table 3.9-10, Table 3.9-11
Pipe Support Criteria: applicable codes, jurisdictional boundaries, pipe support baseplate and anchor bolt design, use of energy absorbers and limit stops, pipe support stiffnesses, seismic self-weight excitation, design of supplementary steel, considerations of friction forces, pipe support gaps and clearances, instrument line support criteria	3.9.1.2, 3.9.3.4, 3.9.3.5
Equivalent Static Load Method of Analysis	3.7.3.5, 3.7.3.5.1, 3.7.3.5.2
Three Components of Earthquake Motion	3.7.3.6
Left-Out-Force Method Used in PIPESTRESS Program	3.7.3.7.1.1
SRP 3.7.2 Method for High-Frequency Modes	3.7.3.7.1.2
Combination of Low-Frequency Modes	3.7.3.7.2
Modeling Methods and Analytical Procedures for Piping Systems	3.7.3.8, 3.7.3.8.1, 3.7.3.8.2.2, 3.7.3.8.3, 3.7.3.8.4
Seismic Anchor Motions	3.7.3.9
Methods Used to Account for Torsional Effects of Eccentric Masses	3.7.3.11
Design Methods of Piping to Prevent Adverse Spatial Interactions	3.7.3.13.4, 3.7.3.13.4.1, 3.7.3.13.4.2, 3.7.3.13.4.3
Analysis Procedure for Damping	3.7.3.15
Time History Analysis of Piping Systems	3.7.3.17

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Table 3.9-19 (Sheet 2 of 2)
Critical Piping Design Methods and Criteria (Piping Design Criteria)

Commitment	Tier 2 Reference (Section or Table)
Design Transients Use of NRC Bulletins 88-08 and 88-11	3.9.1.1
Loads for Class 1 Components and Core/Component Supports	3.9.3.1.2
Use of Square-Root-Sum-of-the-Squares Method for SSE plus Pipe Rupture	3.9.3.1.3
Analysis of Reactor Coolant Loop Piping	3.9.3.1.4
ASME Classes 1, 2, and 3 Piping Use of ASME Code, Section III	3.9.3.1.5
Design of Spring-Loaded Safety Valves	3.9.3.3.1
Design and Analysis Requirement for Open and Closed Discharge Systems	3.9.3.3.3
Component and Piping Supports for Dynamic Loading	3.9.3.4
Class 2 and Class 3 Component Supports Use of ASME Section III	3.9.3.4.2
Piping System Seismic Stress Analysis	3.9.3.4.3
Design Report for ASME Classes 1, 2, and 3 Piping	3.9.8.2
Hard Rock High Frequency (HRHF) Seismic Response Effect on Piping Systems	Appendix 3I.6.3
Integrity of Nonsafety-Related CVS Piping Inside Containment Compliance with 10 CFR 50.55a and ASME B31.1 Code	5.2.1.1

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Table 3.9-20 (Sheet 1 of 2)
Piping Packages Chosen to Demonstrate Piping Design for Piping DAC Closure

(in addition to Class 1 lines larger than 1 inch in diameter)

ASME Class	Inside/Outside Containment (IC/OC)	Line(s) Description	Piping Package Number
2	IC	Hot Leg 2 I&C	APP-RCS-PLR-210
1,3	IC	Reactor Head Vent	APP-RCS-PLR-230
2	IC	Hot Leg 1 I&C	APP-RCS-PLR-260
2	IC	Hot Leg 1 Sampling	APP-RCS-PLR-460
2	IC	Hot Leg 2 Sampling	APP-RCS-PLR-470
2	IC	Pressurizer Sampling	APP-RCS-PLR-480
2	IC	Pressurizer I&C	APP-RCS-PLR-510
2/3	IC	Component Cooling from Penetration C01 IRC	APP-CCS-PLR-040
2/3	IC	Component Cooling from Penetration C02 IRC	APP-CCS-PLR-050
2/3	IC	CVS Makeup from Penetration C03 IRC	APP-CVS-PLR-090
2/3	IC	CVS Letdown from Penetration 002 IRC 2	APP-CVS-PLR-100
2/3	IC	ACC-A/B3 Makeup	APP-PXS-PLR-620
2/3	IC	SFS from Penetration C01	APP-SFS-PLR-600
2/3	IC/OC	Feedwater to SG 01	APP-SGS-PLR-010
2/3	IC/OC	Feedwater to SG 02	APP-SGS-PLR-020
2/3	IC/OC	Main Steam to SG 01	APP-SGS-PLR-030
2/3	IC/OC	Main Steam to SG 02	APP-SGS-PLR-040
2/3	IC	SG01 Blowdown to Penetration C03A	APP-SGS-PLR-070
2/3	IC	SG02 Blowdown to Penetration C03B	APP-SGS-PLR-080
2/3	IC	SG01 Startup Feed Water from Penetration C05A	APP-SGS-PLR-310
2/3	IC	SG02 Startup Feed Water from Penetration C05B	APP-SGS-PLR-320
2/3	IC	VWS Supply from Containment Penetration C02	APP-VWS-PLR-500
2/3	IC	VWS Return to Containment Penetration C01	APP-VWS-PLR-530
2/3	OC	Component Cooling from Penetration C01 ORC	APP-CCS-PLR-810

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Table 3.9-20 (Sheet 2 of 2)
Piping Packages Chosen to Demonstrate Piping Design for Piping DAC Closure

(in addition to Class 1 lines larger than 1 inch in diameter)

ASME Class	Inside/Outside Containment (IC/OC)	Line(s) Description	Piping Package Number
2/3	OC	Component Cooling from Penetration C02 ORC	APP-CCS-PLR-820
2/3	OC	CVS Makeup from Penetration C03 ORC	APP-CVS-PLR-530
2/3	OC	PCS Recirculation, DWS Supply, and FPS Supply	APP-PCS-PLR-100
2/3	OC	From Spent Fuel to RNS and PCCWST Drain	APP-RNS-PLR-100
2/3	OC	Normal RHR to Heat Exchangers and Pumps from Containment Penetrations C01 and C02	APP-RNS-PLR-170
2/3	OC	Spent Fuel Cooling Module R3-65	APP-SFS-PLR-350
2/3	OC	SG01 Startup Feedwater to Penetration C05A	APP-SGS-PLR-110
2/3	OC	SG02 Startup Feedwater to Penetration C05B	APP-SGS-PLR-120
2/3	OC	Nonradioactive Vent Return from Main Control Room	APP-VBS-PLR-010
2/3	OC	Nonradioactive Vent Supply to Main Control Room	APP-VBS-PLR-030
2/3	OC	VWS Supply to Containment Penetration C02	APP-VWS-PLR-910
2/3	OC	VWS Return from Containment Penetration C01	APP-VWS-PLR-920

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Table 3.9-201
Safety Related Snubbers

System	Snubber (Hanger) No.	Line #
CVS	APP-CVS-PH-11Y0262	L040
CVS	APP-CVS-PH-11Y0268	L038
CVS	APP-CVS-PH-11Y2040	L001
CVS	APP-CVS-PH-11Y2223	L001
CVS	APP-CVS-PH-11Y2224	L001
CVS	APP-CVS-PH-11Y2227	L040
CVS	APP-CVS-PH-11Y2229	L071
CVS	APP-CVS-PH-11Y2265	L056
CVS	APP-CVS-PH-11Y2266	L056
CVS	APP-CVS-PH-11Y7074	L068
PXS	APP-PXS-PH-11Y0020	L127A
PXS	APP-PXS-PH-11Y0578	L113A
PXS	APP-PXS-PH-11Y2052	L114
PXS	APP-PXS-PH-11Y2057	L117A
PXS	APP-PXS-PH-11Y2059	L132A
RCS	APP-RCS-PH-11Y0039	L215
RCS	APP-RCS-PH-11Y0060	L215
RCS	APP-RCS-PH-11Y0067	L005B
RCS	APP-RCS-PH-11Y0081	L112
RCS	APP-RCS-PH-11Y0082	L112
RCS	APP-RCS-PH-11Y0090	L118A
RCS	APP-RCS-PH-11Y0103	L003
RCS	APP-RCS-PH-11Y0112	L032A
RCS	APP-RCS-PH-11Y0388	L133B
RCS	APP-RCS-PH-11Y0391	L137B
RCS	APP-RCS-PH-11Y0528	L005A
RCS	APP-RCS-PH-11Y0810	L011A
RCS	APP-RCS-PH-11Y0811	L010A
RCS	APP-RCS-PH-11Y0813	L010B
RCS	APP-RCS-PH-11Y1127	L110A
RCS	APP-RCS-PH-11Y1130	L212B
RCS	APP-RCS-PH-11Y1132	L215
RCS	APP-RCS-PH-11Y1134	L215
RCS	APP-RCS-PH-11Y1140	L214
RCS	APP-RCS-PH-11Y1141	L215
RCS	APP-RCS-PH-11Y1144	L214
RCS	APP-RCS-PH-11Y2005	L022A
RCS	APP-RCS-PH-11Y2101	L032B
RCS	APP-RCS-PH-11Y2106	L137A
RCS	APP-RCS-PH-11Y2107	L133A
RCS	APP-RCS-PH-11Y2264	L112
RNS	APP-RNS-PH-12Y2060	L006
SGS	APP-SGS-PH-11Y0001	L003B
SGS	APP-SGS-PH-11Y0002	L003B
SGS	APP-SGS-PH-11Y0004	L003B
SGS	APP-SGS-PH-11Y0057	L003A
SGS	APP-SGS-PH-11Y0058	L003A
SGS	APP-SGS-PH-11Y0063	L003A
SGS	APP-SGS-PH-11Y0463	L006A
SGS	APP-SGS-PH-11Y0464	L006A
SGS	APP-SGS-PH-11Y0470	L006B
SGS	APP-SGS-PH-11Y2002	L006A
SGS	APP-SGS-PH-11Y2021	L006A
SGS	APP-SGS-PH-11Y3101	L006B
SGS	APP-SGS-PH-11Y3102	L006B
SGS	APP-SGS-PH-11Y3121	L006B
SGS	APP-SGS-PH-11Y7057	L006B
SGS	APP-SGS-PH-12Y0108	L023B
SGS	APP-SGS-PH-12Y0136	L023A
SGS	APP-SGS-PH-12Y0507	L022B
SGS	APP-SGS-PH-12Y0510	L022A
SGS	APP-SGS-PH-12Y7056	L022A
SGS	APP-SGS-PH-12Y7058	L022B
SGS	SG 1 Snubber A (1A)	(1)
SGS	SG 1 Snubber B (1B)	(1)
SGS	SG 2 Snubber A (2A)	(1)
SGS	SG 2 Snubber B (2B)	(1)

(1) These snubbers are on the upper lateral support assembly of the steam generators.

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Figures 3.9-1–3.9-3 Not Used

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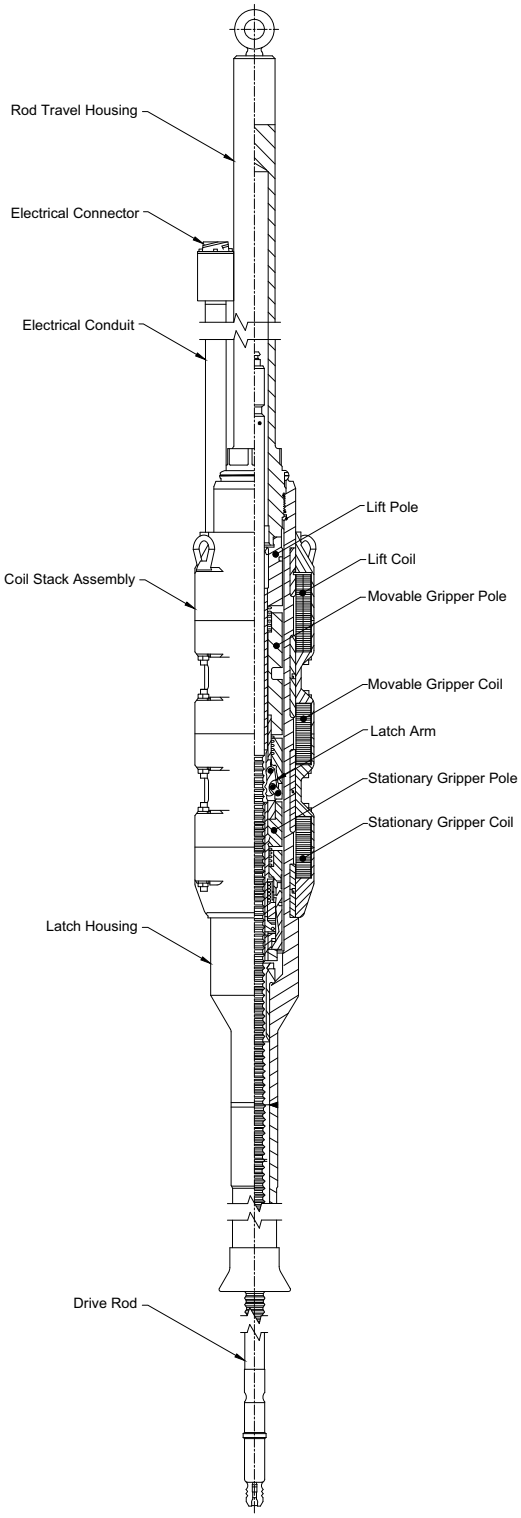

Figure 3.9-4
Control Rod Drive Mechanism

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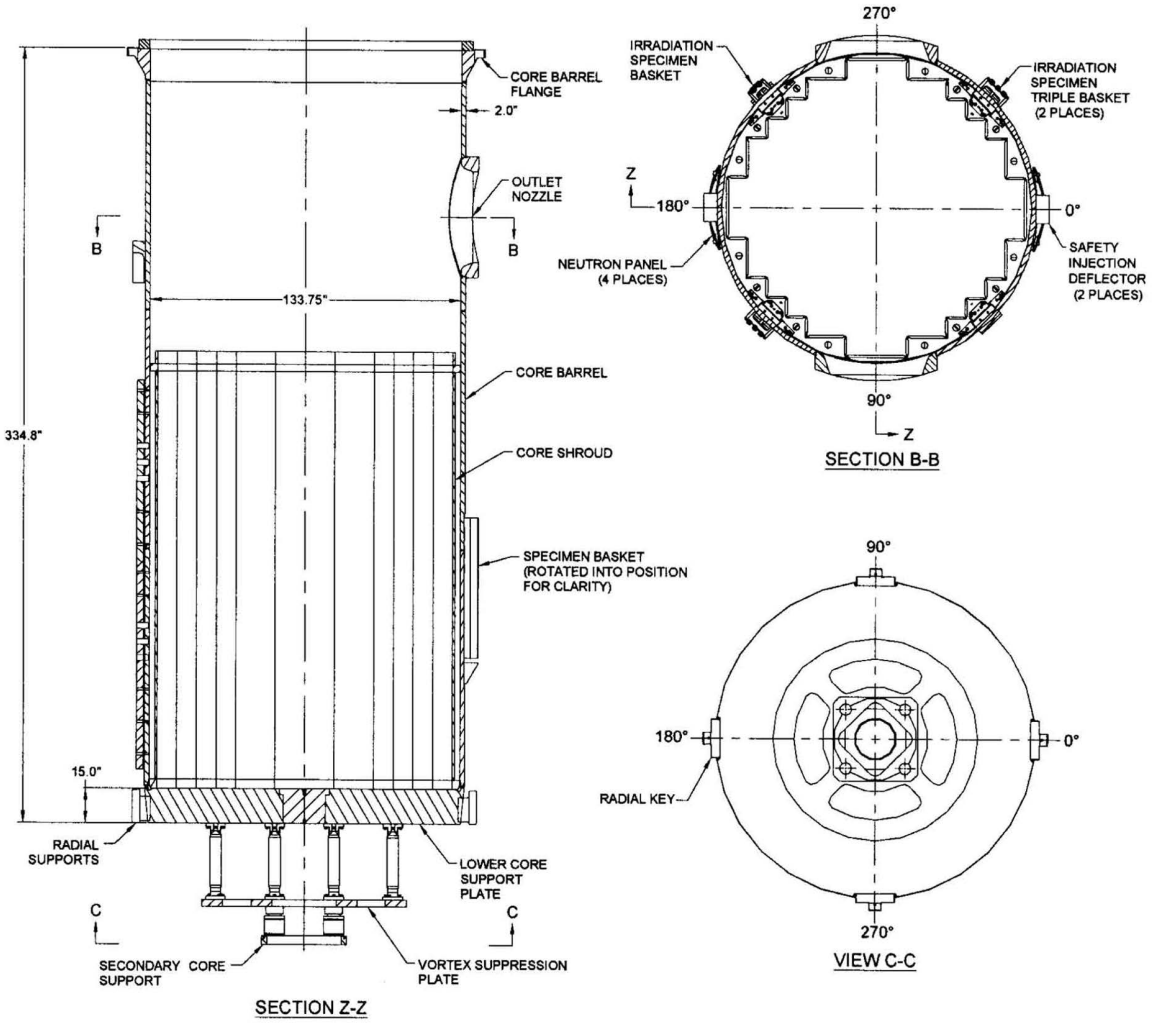

Figure 3.9-5
Lower Reactor Internals

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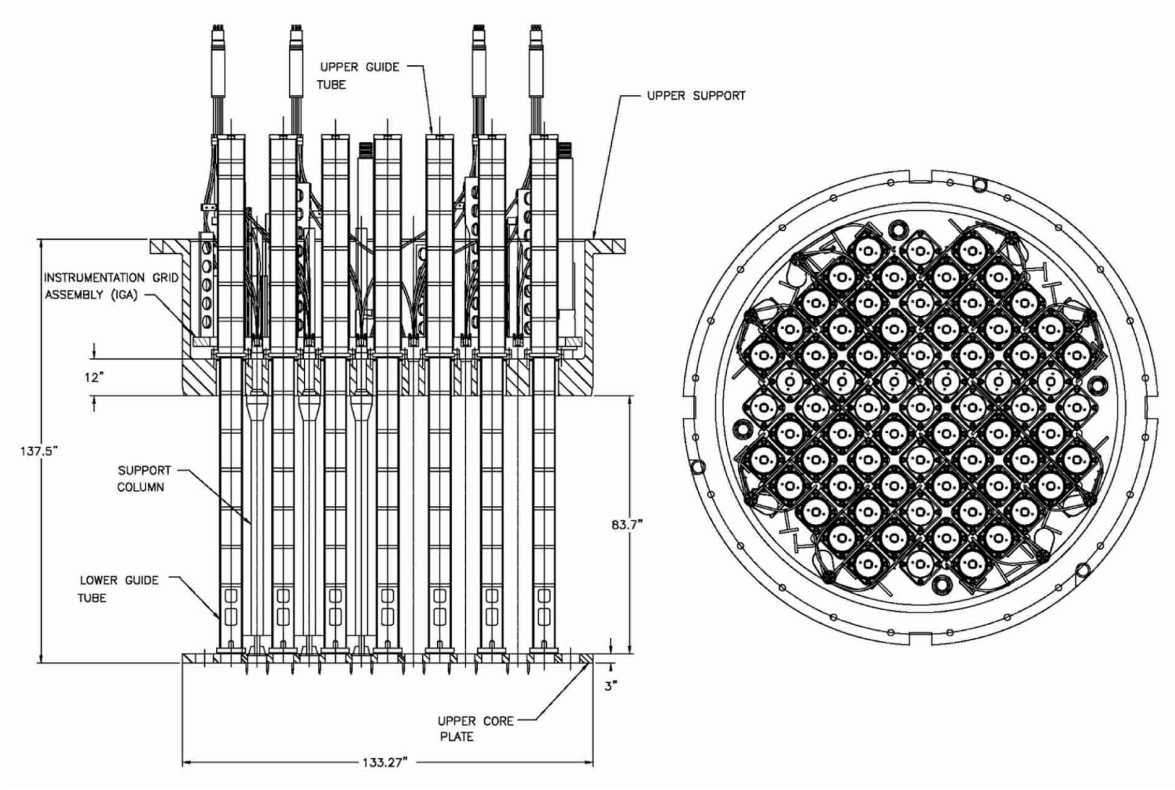

Figure 3.9-6
Upper Core Support Structure

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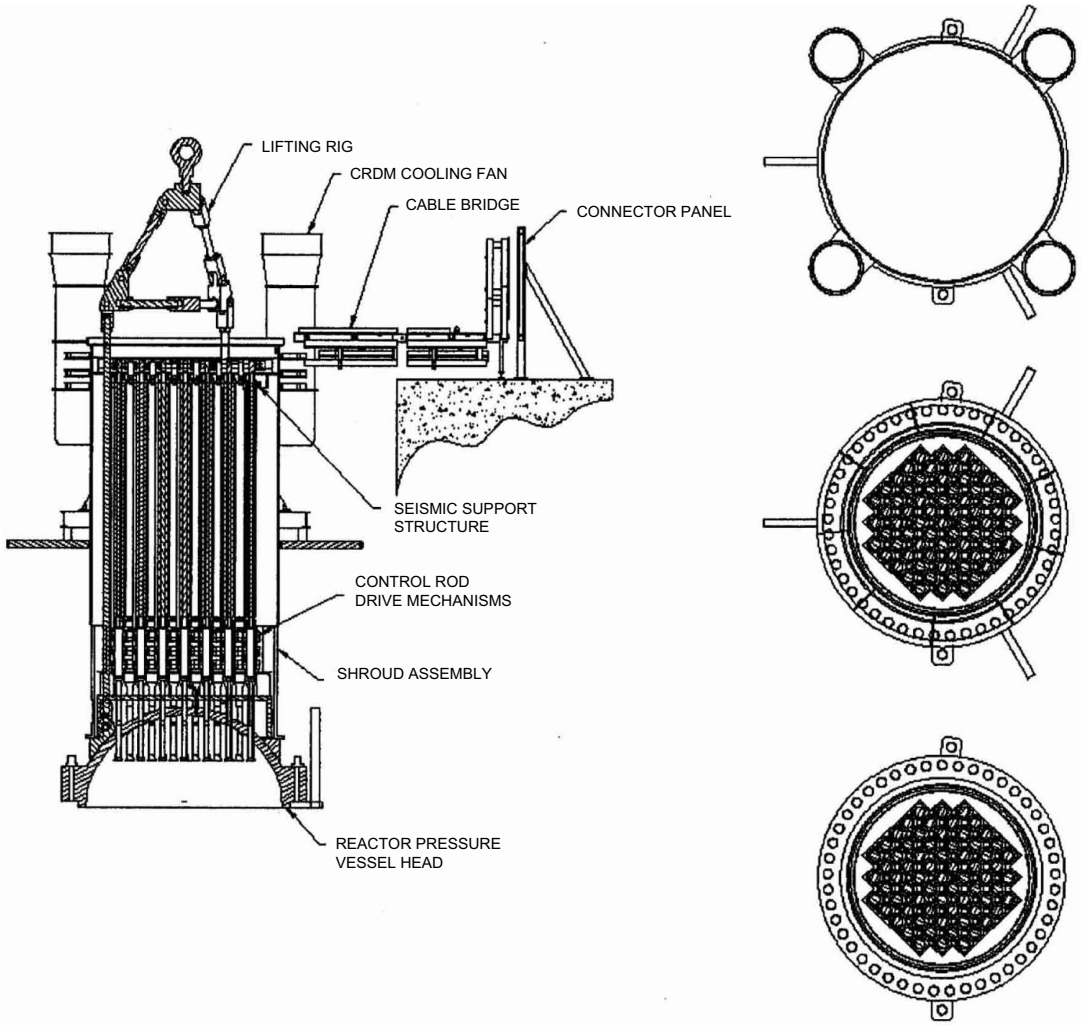

Figure 3.9-7
Integrated Head Package

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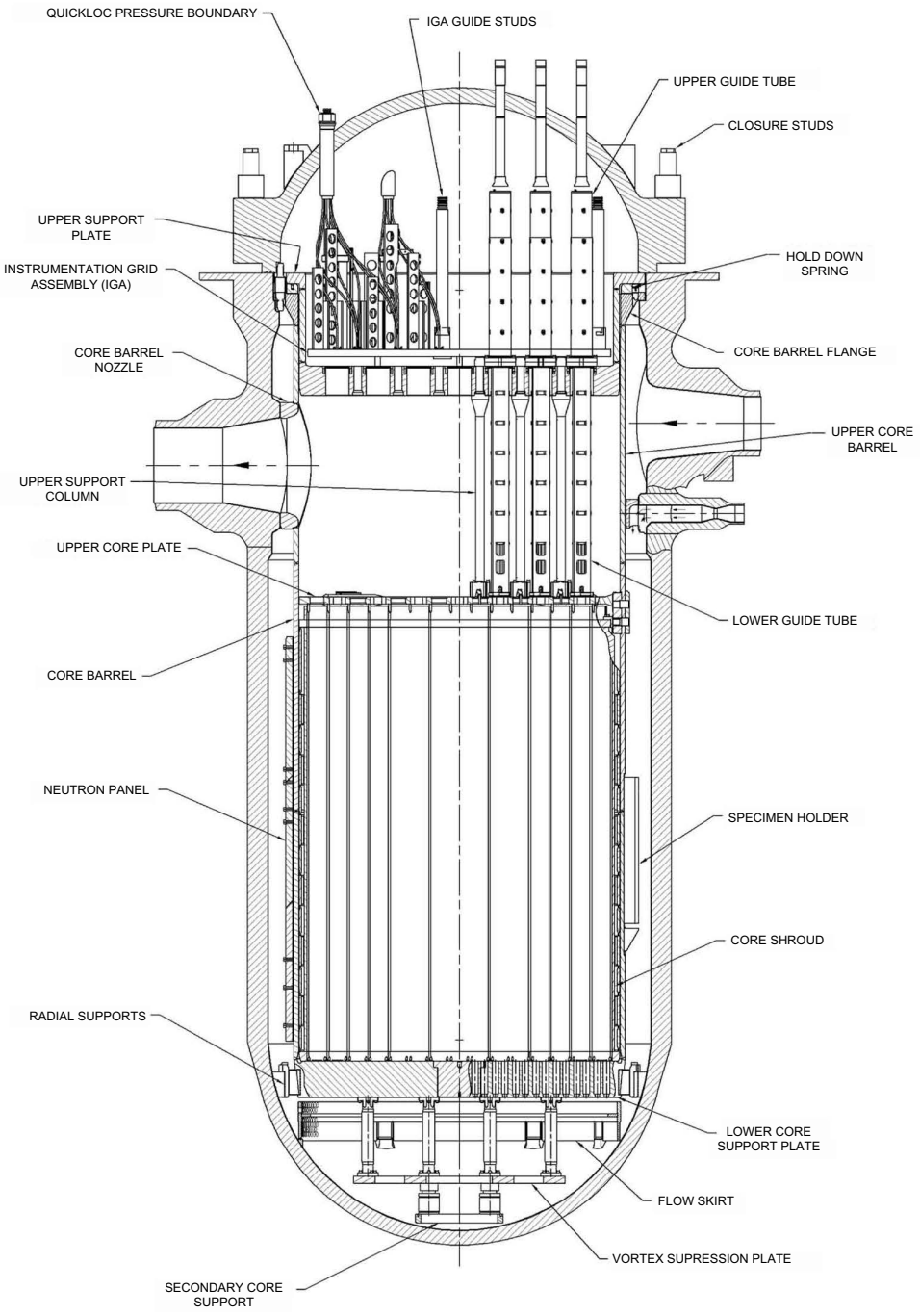

Figure 3.9-8
Reactor Internals Interface Arrangement

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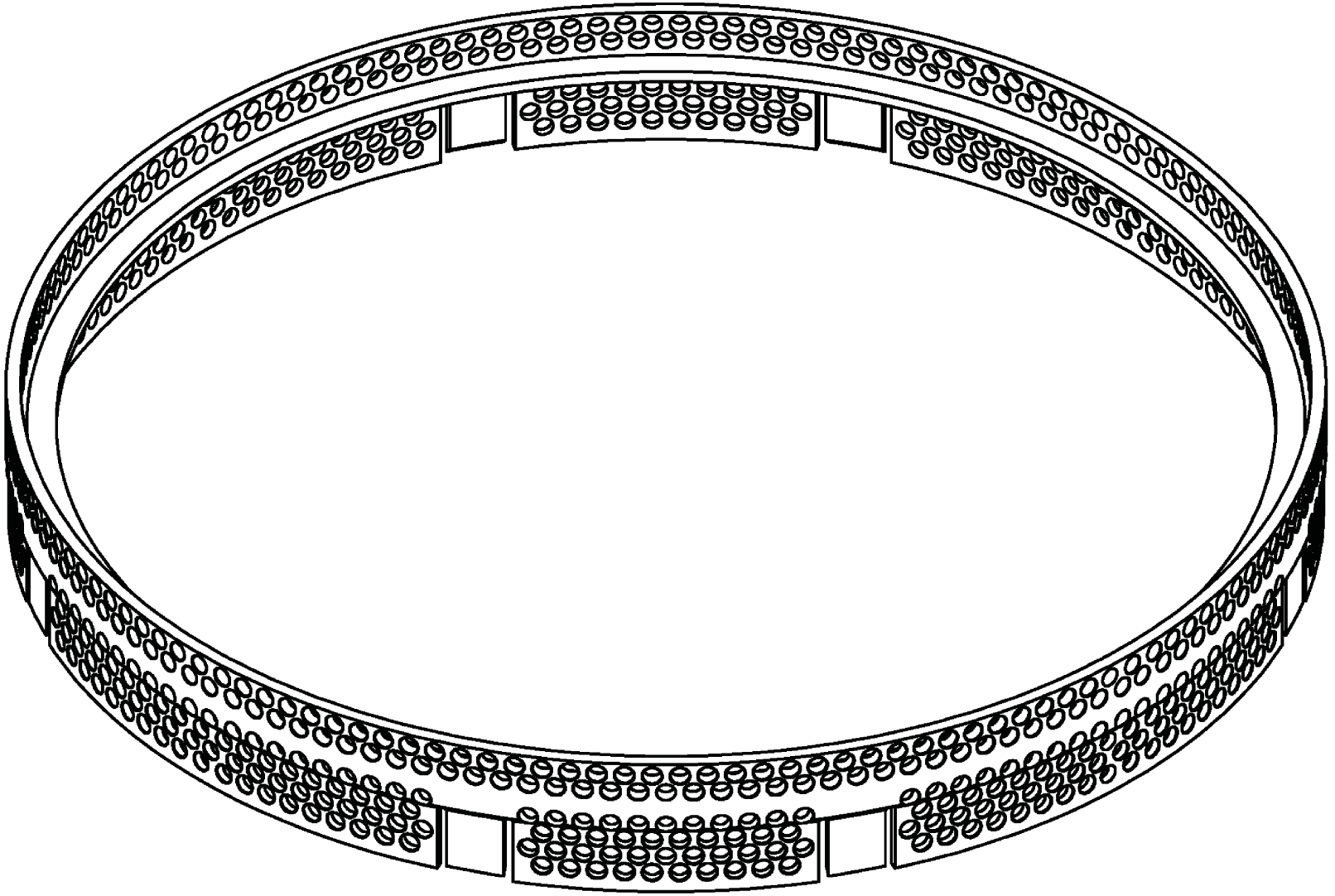

Figure 3.9-9
Flow Skirt Schematic

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3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

Safety-related equipment and selected portions of post-accident monitoring equipment are classified as seismic Category I, as discussed in Subsection 3.2.1.1. This section addresses the seismic and dynamic qualification of this equipment other than piping and includes the following types:

- Safety-related instrumentation and electrical equipment and certain monitoring equipment.
- Safety-related active mechanical equipment that performs a mechanical motion while accomplishing a system safety-related function. These devices include the control rod drive mechanisms; HVAC and fluid system valves.
- Safety-related, nonactive mechanical equipment whose mechanical motion is not required while accomplishing a system safety-related function, but whose structural integrity must be maintained in order to fulfill its design safety-related function.

This section presents or references information to demonstrate that mechanical equipment, electrical equipment, instrumentation, and, where applicable, their supports classified as seismic Category I are capable of performing their designated safety-related functions under the full range of normal and accident (including seismic) loadings. This equipment includes devices associated with systems essential to safe shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive material to the environment or in mitigating the consequences of accidents. The information presented or referenced includes:

- Identification of the seismic Category I instrumentation, electrical equipment, and appropriate mechanical equipment
- Qualification criteria employed for each type of equipment
- Designated safety-related functional requirements
- Definition of the applicable seismic environment
- Definition of other normal and accident loadings
- Documentation of the qualification process employed to demonstrate the required structural integrity and operability of mechanical and electrical equipment and instrumentation in the event of a safe shutdown earthquake (SSE) after a number of postulated occurrences of an earthquake smaller than a safe shutdown earthquake in combination with other relevant dynamic and static loads.

The AP1000 plant is based on the Certified Seismic Design Response Spectra (CSDRS) defined in Subsection 3.7.1.1. The CSDRS are based on Regulatory Guide 1.60 design response spectra with an increase in the 25 hertz region. The Ground Motion Response Spectra (GMRS) for some

Central and Eastern United States rock sites show higher amplitude at high frequency than the CSDRS. Evaluations for high frequency exceedance at AP1000 plant rock sites have been performed as described in Appendix 3I. It is the conclusion of these evaluations that AP1000 plant systems, structures, and components are qualified for the high frequency seismic response based on the CSDRS with the exception of potential high frequency sensitive components (APP-GW-GLN-144, Reference 3). Specific models of components are not identified as part of the AP1000 certified design and are evaluated for high frequency sensitivity as part of the equipment qualification. Appendix 3I

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provides the criteria for addressing potential high frequency sensitive components for plant locations where there is CSDRS exceedance in the high frequency region.

3.10.1 Seismic and Dynamic Qualification Criteria

3.10.1.1 Qualification Standards

The methods of meeting the general requirements for the seismic and dynamic qualification of seismic Category I mechanical and electrical equipment and instrumentation as described by General Design Criteria (GDC) 1, 2, 4, 14, 23, and 30 are described in Section 3.1. The general methods of implementing the requirements of Appendix B to 10CFR50 are described in Chapter 17.

The Nuclear Regulatory Commission (NRC) recommendations concerning the methods employed for seismic qualification of mechanical and electrical equipment are contained in Regulatory Guide 1.100, which endorses IEEE 344-1987 (Reference 1).

[*AP1000 meets IEEE 344-1987, as modified by Regulatory Guide 1.100, by either type testing or analysis or by an appropriate combination of these methods*]* employing the methodology described in Appendix 3D.

The guidance provided in the ASME Code, Section III, is followed in the design of seismic Category I mechanical equipment to achieve the structural integrity of pressure boundary components. In addition, the AP1000 implements an operability program for active valves based on ASME QME-1-2007, as addressed in Section 3.9.

3.10.1.2 Performance Requirements for Seismic Qualification

An equipment qualification data package (EQDP) is developed for the instrumentation and electrical equipment classified as seismic Category I. Table 3.11-1 of Section 3.11 identifies the seismic Category I electrical equipment and instrumentation supplied for the AP1000. Each equipment qualification data package contains a section entitled “Performance Requirements.” This section establishes the safety-related functional requirements of the equipment to be demonstrated during and after a seismic event. The required response spectra employed by the AP1000 for generic seismic qualification are also identified in the section.

For active seismic Category I mechanical components, the performance requirements are defined in the appropriate design and equipment specifications. Requirements for active valves are discussed in Subsection 3.10.2.2.

For other seismic Category I mechanical components, the performance requirement is to maintain structural integrity under appropriate loading conditions. Seismic qualification documentation requirements for mechanical equipment are discussed in Subsection 3.10.4.

A master list and summary of seismic qualification of safety-related seismic Category I electrical and mechanical equipment are maintained as part of the equipment qualification file.

3.10.1.3 Performance Criteria

Seismic and dynamic loading qualification demonstrates that seismic Category I instrumentation and electrical equipment and active valves are capable of performing their designated safety-related functions under applicable plant loading conditions, including the safe shutdown earthquake. The qualification also demonstrates the structural integrity of seismic Category I nonactive valves, mechanical supports, and structures. Some permanent deformation of supports and structures is

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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acceptable at the safe shutdown earthquake level, provided that the capability to perform the designated safety-related functions is not impaired.

3.10.2 Methods and Procedures for Qualifying Electrical Equipment, Instrumentation, and Mechanical Components

Testing is the preferred method to qualify equipment. Both dynamic and static test approaches are used to demonstrate structural integrity and operability of mechanical and electrical equipment in the event of a safe shutdown earthquake preceded by five earthquakes of a magnitude equal to 50 percent of the calculated safe shutdown earthquake. Test samples are selected according to type, load level, and size, as well as other pertinent factors on a prototype basis.

Analysis using mathematical modeling techniques correlated to tests performed on similar equipment or structures and verified analytical approaches are used to qualify equipment. Combined analysis and testing is also used to qualify equipment.

The analytical approach to seismic qualification without testing is used under the following conditions:

- If operability can be demonstrated by analysis alone.
- If only maintaining structural integrity is required for the safety-related function.
- If the equipment is too large or heavy to obtain a representative test input at existing test facilities. (The essential control devices and electrical parts of large equipment are tested separately if required.)
- If the interfaces (for example, interconnecting cables to the cabinet or other complex inputs) cannot be conservatively considered during testing.
- If the response of the equipment is essentially linear or has a simple nonlinear behavior that can be predicted by conservative analytical methods.

A combination of testing and analysis is used when complete testing is not practical.

Equipment that has been previously qualified by means of test and analysis equivalent to those described herein is acceptable provided that proper documentation is submitted.

Seismic qualification of seismic Category I instrumentation and electrical equipment is demonstrated by either type testing or a combination of test and analysis. The qualification method employed by the AP1000 for a particular item of equipment is based upon many factors including practicability, complexity of equipment, economics, and availability of previous seismic qualification. The qualification method employed for a particular item of instrumentation or electrical equipment is identified in the individual equipment qualification data package.

For active valves, the AP1000 uses a combination of tests and analyses to demonstrate the structural integrity and operability of such components. Other seismic Category I mechanical equipment is qualified by analysis to demonstrate structural integrity.

The methods of load combination and methods of combining dynamic responses for mechanical equipment are discussed in Section 3.9. For instrumentation and electrical equipment, the only dynamic loads considered in testing are seismic loads and hydrodynamic and vibratory loads where applicable. Other dynamic loads to which instrumentation and electrical equipment may be subjected are enveloped by this testing or are addressed by analysis.

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3.10.2.1 Seismic Qualification of Instrumentation and Electrical Equipment

3.10.2.1.1 Type Testing

For seismic Category I instrumentation and electrical equipment, seismic qualification by test is performed according to IEEE 344-1987. Where testing is used, multifrequency, multiaxis inputs are developed by the general procedures outlined in Appendix 3D. The test results contained in the individual equipment qualification data packages demonstrate that the measured test response spectrum envelops the required response spectrum defined in the equipment qualification data package.

Alternative test methods, such as single-frequency, single-axis inputs for line-mounted equipment, are used in selected cases as permitted by IEEE 344-1987 and Regulatory Guide 1.100. These methods are further described in Appendix 3D.

3.10.2.1.2 Analysis

Seismic qualification by the analysis method can be used to demonstrate qualification for equipment where structural integrity or limitations of deformation provide the safety-related function. Seismic analysis is widely used to demonstrate qualification of equipment where testing is impractical, equipment is easily modeled (no secondary structures), and no complex equipment functions are required.

Analysis may complement tests when needed to extrapolate or interpolate experimental data. Analysis may be used to investigate established failure modes related to structural integrity, fatigue, and stress-strain behavior. Two methods of computation are: (1) static equivalent load, which yields conservative results, and (2) dynamic analysis, which takes into account the dynamic response properties of the structure and which can be suitably represented by linear models.

The analysis method is not recommended for complex equipment that cannot be modeled to adequately predict its response.

3.10.2.1.3 Combination of Test and Analysis

The AP1000 uses a combination of test and analysis to qualify seismic Category I instrumentation and electrical equipment. The test methods are similar to those described for type testing. Available test results are employed in combination with the analysis methods described in IEEE 344-1987 to demonstrate seismic qualification. The analytical methods include both static and dynamic techniques, which are described in detail in Appendix 3D.

3.10.2.2 Seismic and Operability Qualification of Active Mechanical Equipment

Active mechanical equipment is qualified for both structural integrity and operability for its intended service conditions by a combination of test and analysis. These methods address the applicable loading conditions, such as thermal transients, significant flow loads, and/or degraded flow conditions. The test and analysis methods utilized in qualification of these components provide adequate confidence of operability under required plant conditions.

Qualification methods used for active valves are described in this subsection. The qualification methods used for control rod drive mechanisms and snubbers are described in Section 3.9. The qualification program for valves that are part of the reactor coolant pressure boundary shall include testing or analysis that demonstrate that these valves will not experience leakage beyond the limits defined in the design specification for each valve when subjected to design loading.

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Safety-related active valves, listed in Table 3.11-1, are required to function at the time of an accident. Tests and analyses are conducted to qualify active valves providing confidence that these valves operate during a seismic event.

The safety-related valves are subjected to a series of type tests or actual tests before service and during the plant life. Before installation, the following tests are performed: body hydrostatic test to ASME Code, Section III, requirements, back-seat and main seat leakage tests, disc hydrostatic tests, and operational tests to verify that the valve opens and closes within stroke time requirements. For the qualification of motor operators for environmental conditions, see Section 3.11. After installation, the valves undergo system level pressure tests, construction acceptance tests, and preoperational tests. Where applicable, periodic in-service inspections and operations are performed in-situ to verify the functional capability of the valve. On active valves, an analysis of the extended structure is performed for static equivalent seismic safe shutdown earthquake loads applied at the center of gravity of the extended structure. The maximum stress limits used for active Class 1, 2, and 3 valves are compared to acceptable standards in the ASME Code. Valve discs are evaluated for maximum design line pressure and maximum differential pressure resulting from plant operating, transient, and accident conditions. Feedwater line valve discs are evaluated, using appropriate ASME Code, Section III limits, for the effect of dynamic loads by considering the effect of an equivalent differential pressure. The equivalent differential pressure is developed from a transient analysis based on wave mechanics that includes consideration of system arrangement and valve closing dynamics. Valve operating conditions are included as part of the valve design specification and are used to evaluate the valve disc. Additional information is provided on the controlled-closure, feedwater check valve in Subsection 10.4.7.2.2.

In addition to these tests and analyses, representative valves of each design type having extended structures are subjected to static pull tests and nozzle load tests as appropriate. These tests verify operability of a rigid valve (natural frequency equaling or exceeding 33 hertz) during a simulated plant faulted-condition event by demonstrating operational capabilities within the specified limits. A representative valve of a specific design type is identified for this testing by the specification (for example, globe valve, motor-operated valve) for that particular type of valve. A further subdivision of design is based upon the valve size, pressure rating, type of operator, and previous operability testing to evaluate the need for additional testing of a particular design type. The testing procedures are described in Appendix 3D.

The accelerations used for the static valve qualification are equivalent, as justified by analysis, to 6.0g in two orthogonal horizontal directions and 6.0g vertical. For testing, the required input motion (RIM) curve shall be consistent with the profile of Figure 6 of IEEE 382-1996 (Reference 2), with the acceleration magnitude increased to 6.0g. The piping design maintains the operator accelerations to these levels. If the natural frequency of the valve is less than 33 hertz, a dynamic analysis of the valve is performed to determine the equivalent loads to be applied during the static test.

Valves that are safety related but are classified as not having an extended structure, such as check valves and safety valves, are considered separately.

Check valves are characteristically simple in design. Their operation is not affected by seismic accelerations or the maximum applied nozzle loads. These valves are designed so that once the structural integrity of the valve is verified using standard methods, the capability of the valve to operate is demonstrated by its design features. The valve also undergoes in-shop hydrostatic and seat leakage tests, and periodic in situ valve exercising and inspection to verify the functional capability of the valve.

The pressurizer safety valves are qualified by the following procedures (these valves are also subjected to tests and analysis similar to check valves): stress and deformation analyses of critical items that affect operability for faulted condition loads, in-shop hydrostatic and seat leakage tests,

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and periodic in situ valve inspection. In addition to these tests, a static load equivalent to that applied by the faulted condition is applied at the top of the bonnet, and the fluid pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve demonstrates its over-pressurization safety capabilities during a seismic event.

Safety-related active valves mounted in HVAC ductwork used to isolate main control room areas during design events are listed in Table 3.11-1. These valves are qualified to operate on demand using air operators.

Using these methods, the safety-related valves are qualified for operability during a faulted event. These methods conservatively simulate the seismic event and demonstrate that the active valves perform their safety-related function when necessary.

3.10.2.3 Valve Operator Qualification

Active valve motor operators, position sensors, and solenoid valves are seismically qualified according to IEEE 382-1996, as discussed in the appropriate equipment qualification data packages.

3.10.2.4 Seismic Qualification of Other Seismic Category I Mechanical Equipment

For seismic Category I mechanical equipment not defined as active, the AP1000 uses analysis to demonstrate structural integrity. The analysis methods are described in Sections 3.7 and 3.9 and in Appendix 3D.

3.10.3 Method and Procedures for Qualifying Supports of Electrical Equipment, Instrumentation, and Mechanical Components

The equipment qualification data packages and seismic analysis report for non-active mechanical equipment (with no environmental qualification requirements) identify the equipment mounting employed for qualification and establish interface requirements for the equipment to provide confidence that subsequent in-plant installation does not degrade the established qualification. Interface requirements are defined based on the test configuration and other design requirements. Dynamic coupling effects resulting from mounting the component according to these interface criteria are considered in the qualification program.

Information concerning the structural integrity of pressure-retaining components, their supports, and core supports is presented in Section 3.9.

The following bases are used in the design and analysis of cable tray supports and instrument tubing supports:

- The methods used in the seismic analysis of cable tray supports are described in Appendix 3F.

- The seismic Category I instrument tubing systems are supported so that the allowable stresses permitted by ASME Code, Section III, are not exceeded when the tubing is subjected to the loads specified in Section 3.9.

3.10.4 Documentation

The results of tests and analyses verifying that the criteria established in Subsection 3.10.1 are satisfied, employing the qualification methods described in Subsections 3.10.2 and 3.10.3, are included in the individual equipment qualification data packages, seismic analysis report, and test

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reports. The upkeep of the equipment qualification file is maintained during the equipment selection and procurement phase is discussed in Subsection 3.11.5.

Seismic qualification of equipment is documented in equipment qualification data packages, seismic analysis report, and test reports. Appendix 3D provides guidance in this area.

3.10.5 Standard Review Plan Evaluation

A summary describing the Standard Review Plan differences in regard to seismic and dynamic qualification of mechanical and electrical equipment is provided Subsection 1.9.2.

3.10.6 Combined License Information Item on Experienced-Based Qualification

Not used.

3.10.7 References

1. IEEE 344-1987, “Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.”

2. IEEE 382-1996, “IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants.”

3. APP-GW-GLN-144, “AP1000 Design Control Document High Frequency Seismic Tier 1 Changes,” Westinghouse Electric Company LLC.

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3.11 Environmental Qualification of Mechanical and Electrical Equipment

This section presents information to demonstrate that the mechanical and electrical portions of the engineered safety features, the reactor protection systems, and selected portions of the post-accident monitoring system are capable of performing their designated functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. The information presented includes identification of the equipment required to be environmentally qualified and, for each item of equipment, the designated functional requirements, definition of the applicable environmental parameters, and documentation of the qualification process employed to demonstrate the required environmental capability. The seismic qualification of mechanical and electrical equipment is presented in Section 3.10. The portions of post-accident monitoring equipment required to be environmentally qualified are identified in Table 7.5-1.

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Equipment Identification

A complete list of environmentally qualified electrical and mechanical equipment that is essential to emergency reactor shutdown, containment isolation, reactor core cooling, or containment and reactor heat removal, or that is otherwise essential in preventing significant release of radioactive material to the environment, is provided in Table 3.11-1. A list of environmentally qualified electrical and mechanical equipment and a summary of electrical and mechanical equipment qualification results are maintained as part of the equipment qualification file, as outlined in Appendix 3D. The equipment qualification file is maintained during the equipment selection and procurement phase as defined in Subsection 3.11.5.

3.11.1.2 Definition of Environmental Conditions

Appendix 3D identifies applicable normal, abnormal, and design basis accident environmental conditions conforming to General Design Criterion 4. These environmental conditions are associated with various plant areas by an environmental zone, as noted in Table 3D.5-1 and Table 3.11-1.

For mild environments, the area conditions do not change as the result of an accident. There are no degrading environmental effects that lead to common mode failure of the equipment. The qualification of mechanical and electrical equipment located in a mild environment is demonstrated by conducting the plant surveillance activities carried out during the operational phase of the plant.

The environmental conditions identified in Appendix 3D are defined as follows.

Normal operating environmental conditions are defined as those conditions existing during routine plant operations for which the equipment is expected to be available on a continuous basis to perform required functions.

Abnormal environmental conditions are those plant conditions for which the equipment is designed to operate for a period of time without accelerating normal periodic tests, inspections, and maintenance schedules for that equipment. The maximum and minimum conditions identified as the abnormal condition are based on the design limits for the affected areas.

Design basis accident (DBA) and post-design basis accident conditions are those plant conditions resulting from various postulated equipment and piping failures during which the equipment identified in Table 3.11-1 must operate without impairment of the function. The design basis accident and post-design basis accident conditions are discussed in Appendix 3D.

Compatibility of equipment with the specified environmental conditions is achieved by the following.

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Systems and components required to mitigate the consequences of a design basis accident or to perform safe shutdown operation are qualified to remain functional after exposure to the environmental conditions in Table 3D.5-5.

Environmentally qualified equipment exposed to a harsh environment has a qualified life goal of 60 years. Demonstration of qualified life by test or test and analysis is provided to address applicable aging effects. For critical components susceptible to aging, a qualified life is established that includes the effects of the total integrated radiation dose experienced at their respective locations within the plant. When a 60-year qualified life is not achievable, a shorter qualified life is established, and a replacement program is implemented.

For equipment located in a mild environment, a design life goal is established by using known significant aging mechanisms and reliability data.

Equipment qualification takes into account the most severe environmental conditions resulting from the design basis high-energy line break. Included in these conditions are the short-term peak transient temperature following a main steam line break (MSLB) and a radiation exposure and temperature due to a loss of coolant accident (LOCA) within the reactor containment.

Postulated high-energy line failures as defined in Subsection 3.6.2.1.2 are assumed in areas where high-energy lines greater than 1 inch are routed. Essential equipment is protected against the effects of jet impingement (Subsection 3.6.2.4.1) and evaluated for spray effects if required (Subsection 3.6.2.7).

Active mechanical equipment is qualified for operability as discussed in Subsection 3.9.3 and Section 3.10. This operability program, combined with the qualification of the electrical appurtenances (valve operators, solenoids, limit switches), demonstrates qualification under required environmental conditions. Active mechanical equipment is defined as equipment that performs a mechanical motion as part of its safety-related function.

Nonactive mechanical equipment whose only safety function is structural integrity is designed according to ASME Code guidelines. The accident and post-accident environmental effects are considered in the design of such structural components as pump casings and valve bodies.

The environmental qualification program is restricted to evaluating the design of critical nonmetallic subcomponents of active devices in a harsh environment, where failure results in loss of the active component.

In the event of potential flooding/wetting, one of the following criteria is applied for protection of equipment for service in such an environment:

- Equipment will be qualified for submergence due to flooding/wetting.
- Equipment will be protected from wetting due to spray.
- Equipment will be evaluated to show that failure of the equipment due to flooding/wetting is acceptable since its safety-related function is not required or has otherwise been accomplished.

3.11.1.3 Equipment Operability Times

For the AP1000 Class 1E electrical and active mechanical equipment, post-accident operability times are shown in Table 3D.4-2 in Appendix 3D.

Specific information for each device qualified as part of the IEEE 323-1974 qualification program is contained in the appropriate equipment qualification data package.

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The active mechanical component is qualified for operability as discussed in Section 3.10, using test, analysis, or a combination of tests and analyses. This operability program, combined with the qualification of the electrical appurtenances (for example, valve operators) discussed in the appropriate equipment qualification data packages, demonstrates qualification.

3.11.1.4 Standard Review Plan Evaluation

A discussion of the Standard Review Plan requirements in regard to environmental qualification of mechanical equipment is provided in Subsection 1.9.2.

3.11.2 Qualification Tests and Analysis

3.11.2.1 Environmental Qualification of Electrical Equipment

The AP1000 approach for environmental qualification of Class 1E equipment is outlined in Appendix 3D. The Appendix 3D methodology is developed based on the guidelines provided in IEEE 323-1974 (Reference 1) and IEEE 344-1987 (Reference 2).

Qualification for equipment in a harsh environment is based on type testing or testing and analysis. Analysis may be used to determine significant aging mechanisms in mild environment applications. Type testing includes thermal and mechanical aging, radiation, and exposure to extremes of environmental, seismic, and vibration effects. Type testing is done with representative samples of the production line equipment according to the sequence indicated in IEEE 323-1974 to the specified service conditions, including margin. The testing takes into account normal and abnormal plant operation and design basis accident and post-design basis accident operations, as required.

When reliable data and proven analytical methods are available, environmental qualification may be based on analysis supported by partial type test data. This method includes justification of the methods, theories, and assumptions used (that is, mathematical or logical proof based on actual test data) that the equipment meets or exceeds its specified performance requirements when subjected to normal, abnormal, and design basis accident environmental conditions.

Regulatory guides providing guidance for meeting the requirements of 10CFR50, Appendix A, General Design Criteria 1, 2, 4, 23, and 50; Appendix B, Criteria III, XI, and XVII to 10CFR50 and 10CFR50.49, include Regulatory Guide 1.89, Regulatory Guide 1.30, Regulatory Guide 1.63, Regulatory Guide 1.73, Regulatory Guide 1.100, and Regulatory Guide 1.131. The maintenance surveillance program follows the guidance of Regulatory Guide 1.33.

Additional information regarding conformance with each of these regulatory guides is given in Section 1.9.

3.11.2.2 Environmental Qualification of Mechanical Equipment

AP1000 mechanical components identified in Table 3.11-1 are qualified by design to perform their required functions under the appropriate environmental effects of normal, abnormal, accident, and post-accident conditions as required by General Design Criterion 4 and discussed in Appendix 3D. For mild environments, the area conditions do not change as a result of an accident. There are no degrading environmental effects that lead to common mode failure of equipment in mild environments. Mechanical equipment located in harsh environmental zones is designed to perform under the appropriate environmental conditions.

For mechanical equipment, there are two categories of components:

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- Active equipment – equipment that performs a mechanical motion as part of its safety-related function.

  The program for environmental qualification of active mechanical components is based on a combination of design, test, and analysis of critical sub-components, which is supported by maintenance and surveillance programs.

- Nonactive equipment – equipment whose only safety-related function is structural integrity. Nonactive components are designed for structural integrity according to ASME Code, Section III, as discussed in Section 3.9.

3.11.3 Loss of Ventilation

The abnormal environmental conditions shown on Tables 3D.5-3 and 3D.5-4 reflect anticipated maximum conditions based on loss of normal ventilation systems.

Normal containment heat removal is provided by the nonsafety-related containment air recirculation cooling system. If this system is out of service for an extended period of time, the passive containment cooling system may be initiated to maintain the temperature and pressure below the limits noted. Environmentally qualified equipment located in containment performs its functions under these conditions until the normal containment cooling system is restored.

Equipment areas outside containment and outside the main control room are maintained at normal environmental conditions by nonsafety-related HVAC systems. If these systems are disabled, the heat generated by this equipment is absorbed by the surrounding passive heat sinks with an ambient temperature rise that does not exceed the abnormal condition. Normal HVAC is restored within 8 hours or ventilation is provided as discussed in Subsection 9.4.1.2.3.2.

If the normal nonsafety-related main control room HVAC is lost, the heat generated by equipment and people is absorbed by the surrounding passive heat sinks. Normal heating, ventilation, and air-conditioning is restored within 72 hours or ventilation is provided as discussed in Section 6.4.

3.11.4 Estimated Radiation and Chemical Environment

The plant-specific estimates of the radiation dose incurred by equipment during normal operation is shown in Table 3D.5-2 and the estimated doses following a loss-of-coolant accident are defined in Table 3D.5-5.

The identified equipment is qualified to perform functions in the radiation environments present during normal and design basis accident conditions. The normal operational exposure is based upon design source terms presented in Chapter 11 and Subsection 12.2.1. The equipment and shielding configurations are presented in Section 12.3. Post-accident monitoring, reactor trip and engineered safety features system and component radiation exposures are dependent on the location of the equipment in the plant. Source terms and other accident parameters are presented in Subsection 12.2.1 and Chapter 15.

The maximum combined integrated radiation dose inside containment is based on the effects of the normally expected radiation environment (gamma) over the equipment's installed life plus that associated with the most severe design basis event (gamma and beta) during or following which the equipment is required to remain functional.

The chemical environment following a loss of coolant accident is primarily based on the chemistry of the reactor coolant system fluid since there is no caustic containment spray. Sump pH adjustments are considered for certain qualification tests. This is discussed further in Appendix 3D.

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3.11.5 Combined License Information Item for Equipment Qualification File

Westinghouse Electric Company LLC will act as the contractor for the COL holder during the equipment design phase, equipment selection and procurement phase, equipment qualification phase, plant construction phase, and ITAAC inspection phases.

The COL holder is responsible for the maintenance of the equipment qualification file upon receipt from the reactor vendor. The documentation necessary to support the continued qualification of the equipment installed in the plant that is within the Environmental Qualification (EQ) Program scope is available in accordance with 10 CFR Part 50 Appendix A, General Design Criterion 1.

EQ files developed by the reactor vendor are maintained as applicable for equipment and certain post-accident monitoring devices that are subject to a harsh environment. The contents of the qualification files are discussed in Section 3D.7. The files are maintained for the operational life of the plant.

For equipment not located in a harsh environment, design specifications received from the reactor vendor are retained. Any plant modifications that impact the equipment use the original specifications for modification or procurement. This process is governed by applicable plant design control or configuration control procedures.

Central to the EQ Program is the EQ Master Equipment List (EQMEL). This EQMEL identifies the electrical and mechanical equipment or components that must be environmentally qualified for use in a harsh environment. The EQMEL consists of equipment that is essential to emergency reactor shutdown, containment isolation, reactor core cooling, or containment and reactor heat removal, or that is otherwise essential in preventing significant release of radioactive material to the environment. This list is developed from the equipment list provided in Table 3.11-1. The EQMEL and a summary of equipment qualification results are maintained as part of the equipment qualification file for the operational life of the plant.

Administrative programs are in place to control revision to the EQ files and the EQMEL. When adding or modifying components in the EQ Program, EQ files are generated or revised to support qualification. The EQMEL is revised to reflect these new components. To delete a component from the EQ Program, a deletion justification is prepared that demonstrates why the component can be deleted. This justification consists of an analysis of the component, an associated circuit review if appropriate, and a safety evaluation. The justification is released and/or referenced on an appropriate change document. For changes to the EQMEL, supporting documentation is completed and approved prior to issuing the changes. This documentation includes safety reviews and new or revised EQ files. Plant modifications and design basis changes are subject to change process reviews, e.g. reviews in accordance with 10 CFR 50.59 or Section VIII of Appendix D to 10 CFR Part 52, in accordance with appropriate plant procedures. These reviews address EQ issues associated with the activity. Any changes to the EQMEL that are not the result of a modification or design basis change are subject to a separate review that is accomplished and documented in accordance with plant procedures.

Engineering change documents or maintenance documents generated to document work performed on an EQ component, which may not have an impact on the EQ file, are reviewed against the current revision of the EQ files for potential impact. Changes to EQ documentation may be due to, but not limited to, plant modifications, calculations, corrective maintenance, or other EQ concerns.

Table 13.4-201 provides milestones for EQ implementation.

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3.11.6 References

1. IEEE 323-1974, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.”

2. IEEE 344-1987, “IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.”

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Table 3.11-1 (Sheet 1 of 52)
Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
BATTERIES					
IDSA 125V 60 Cell Battery 1A	IDSA-DB-1A	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSA 125V 60 Cell Battery 1B	IDSA-DB-1B	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB 125V 60 Cell Battery 1A	IDSB-DB-1A	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB 125V 60 Cell Battery 1B	IDSB-DB-1B	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB 125V 60 Cell Battery 2A	IDSB-DB-2A	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSB 125V 60 Cell Battery 2B	IDSB-DB-2B	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSC 125V 60 Cell Battery 1A	IDSC-DB-1A	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSC 125V 60 Cell Battery 1B	IDSC-DB-1B	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSC 125V 60 Cell Battery 2A	IDSC-DB-2A	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSC 125V 60 Cell Battery 2B	IDSC-DB-2B	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSD 125V 60 Cell Battery 1A	IDSD-DB-1A	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSD 125V 60 Cell Battery 1B	IDSD-DB-1B	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
Spare 125V 60 Cell Battery 1A	IDSS-DB-1A	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
Spare 125V 60 Cell Battery 1B	IDSS-DB-1B	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E

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Table 3.11-1 (Sheet 2 of 52)
Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
BATTERY CHARGERS					
IDSA Battery Charger	IDSA-DC-1	2	ISOL RT ESF	24 hrs 5 min 24 hrs	E
IDSB Battery Charger 1	IDSB-DC-1	2	ISOL RT ESF	24 hrs 5 min 24 hrs	E
IDSB Battery Charger 2	IDSB-DC-2	2	ISOL	72 hrs	E
IDSC Battery Charger 1	IDSC-DC-1	2	ISOL RT ESF	24 hrs 5 min 24 hrs	E
IDSC Battery Charger 2	IDSC-DC-2	2	ISOL	72 hrs	E
IDSD Battery Charger	IDSD-DC-1	2	ISOL RT ESF	24 hrs 5 min 24 hrs	E
Spare Battery Charger	IDSS-DC-1	2	ISOL	72 hrs	E
DISTRIBUTION PANELS					
IDSA 250 Vdc Dist Panel	IDSA-DD-1	2	ESF	24 hrs	E
IDSB 250 Vdc Dist Panel	IDSB-DD-1	2	ESF	24 hrs	E
IDSC 250 Vdc Dist Panel	IDSC-DD-1	2	ESF	24 hrs	E
IDSD 250 Vdc Dist Panel	IDSD-DD-1	2	ESF	24 hrs	E
IDSA 120 Vac Dist Panel 1	IDSA-EA-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSA 120 Vac Dist Panel 2	IDSA-EA-2	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB 120 Vac Dist Panel 1	IDSB-EA-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB 120 Vac Dist Panel 2	IDSB-EA-2	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
IDSB 120 Vac Dist Panel 3	IDSB-EA-3	2	PAMS	2 wks	E
IDSC 120 Vac Dist Panel 1	IDSC-EA-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSC 120 Vac Dist Panel 2	IDSC-EA-2	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
IDSC 120 Vac Dist Panel 3	IDSC-EA-3	2	PAMS	2 wks	E
IDSD 120 Vac Dist Panel 1	IDSD-EA-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E

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Table 3.11-1 (Sheet 3 of 52)
Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
IDSD 120 Vac Dist Panel 2	IDSD-EA-2	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
FUSE PANELS					
IDSA Fuse Panel	IDSA-EA-4	2	ISOL	24 hrs	E
IDSA Battery Monitor Fuse Panel	IDSA-EA-5	2	ISOL	24 hrs	E
IDSB Fuse Panel	IDSB-EA-4	2	ISOL	24 hrs	E
IDSB Fuse Panel	IDSB-EA-5	2	ISOL	2 wks	E
IDSB Fuse Panel	IDSB-EA-6	2	ISOL	2 wks	E
IDSB Battery Monitor Fuse Panel	IDSB-EA-7	2	ISOL	24 hrs	E
IDSB Battery Monitor Fuse Panel	IDSB-EA-8	2	ISOL	2 wks	E
IDSC Fuse Panel	IDSC-EA-4	2	ISOL	24 hrs	E
IDSC Fuse Panel	IDSC-EA-5	2	ISOL	2 wks	E
IDSC Fuse Panel	IDSC-EA-6	2	ISOL	2 wks	E
IDSC Battery Monitor Fuse Panel	IDSC-EA-7	2	ISOL	24 hrs	E
IDSC Battery Monitor Fuse Panel	IDSC-EA-8	2	ISOL	2 wks	E
IDSD Fuse Panel	IDSD-EA-4	2	ISOL	24 hrs	E
IDSD Battery Monitor Fuse Panel	IDSD-EA-5	2	ISOL	24 hrs	E
IDSS Battery Monitor Fuse Panel	IDSS-EA-1	2	ISOL	2 wks	E
STAND-ALONE ELECTRICAL ISOLATION DEVICES					
Current-to-Current Transmitter 1	WLS-JD-MISX01	2	ISOL	24 hrs	E
Current-to-Current Transmitter 1	WLS-JD-MISX02	2	ISOL	24 hrs	E
Current-to-Current Transmitter 1	WLS-JD-MISX03	2	ISOL	24 hrs	E
TRANSFER SWITCHES					
IDSA Fused Transfer Switch Box 1	IDSA-DF-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB Fused Transfer Switch Box 1	IDSB-DF-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB Fused Transfer Switch Box 2	IDSB-DF-2	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSC Fused Transfer Switch Box 1	IDSC-DF-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSC Fused Transfer Switch Box 2	IDSC-DF-2	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSD Fused Transfer Switch Box 1	IDSD-DF-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E

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Table 3.11-1 (Sheet 4 of 52)
Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
IDSS Fused Transfer Switch Box 1 (Spare)	IDSS-DF-1	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSS Spare Battery Termination Box	IDSS-DF-3	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
MOTOR CONTROL CENTERS					
IDSA 250 Vdc MCC	IDSA-DK-1	2	ESF	24 hrs	E
IDSB 250 Vdc MCC	IDSB-DK-1	2	ESF	24 hrs	E
IDSC 250 Vdc MCC	IDSC-DK-1	2	ESF	24 hrs	E
IDSD 250 Vdc MCC	IDSD-DK-1	2	ESF	24 hrs	E
SWITCHBOARDS					
IDSA 250 Vdc Switchboard 1	IDSA-DS-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB 250 Vdc Switchboard 1	IDSB-DS-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB 250 Vdc Switchboard 2	IDSB-DS-2	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSC 250 Vdc Switchboard 1	IDSC-DS-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSC 250 Vdc Switchboard 2	IDSC-DS-2	2	RT ESF PAMS	5 min 24 hrs 72 hrs	E
IDSD 250 Vdc Switchboard 1	IDSD-DS-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
TRANSFORMERS					
IDSA Regulating Transformer 1	IDSA-DT-1	2	ISOL	24 hrs	E
IDSB Regulating Transformer 1	IDSB-DT-1	2	ISOL PAMS	72 hrs 2 wks	E
IDSC Regulating Transformer 1	IDSC-DT-1	2	ISOL PAMS	72 hrs 2 wks	E
IDSD Regulating Transformer 1	IDSD-DT-1	2	ISOL	24 hrs	E
INVERTERS					
IDSA Inverter	IDSA-DU-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSB Inverter 1	IDSB-DU-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E

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Table 3.11-1 (Sheet 5 of 52)
Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
IDSB Inverter 2	IDSB-DU-2	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
IDSC Inverter 1	IDSC-DU-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
IDSC Inverter 2	IDSC-DU-2	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
IDSD Inverter	IDSD-DU-1	2	RT ESF PAMS	5 min 24 hrs 24 hrs	E
SWITCHGEAR					
RCP 1A 6900V Switchgear 31	ECS-ES-31	2	ESF PAMS	5 min 2 wks	E
RCP 1A 6900V Switchgear 32	ECS-ES-32	2	ESF PAMS	5 min 2 wks	E
RCP 2A 6900V Switchgear 51	ECS-ES-51	2	ESF PAMS	5 min 2 wks	E
RCP 2A 6900V Switchgear 52	ECS-ES-52	2	ESF PAMS	5 min 2 wks	E
RCP 1B 6900V Switchgear 41	ECS-ES-41	2	ESF PAMS	5 min 2 wks	E
RCP 1B 6900V Switchgear 42	ECS-ES-42	2	ESF PAMS	5 min 2 wks	E
RCP 2B 6900V Switchgear 61	ECS-ES-61	2	ESF PAMS	5 min 2 wks	E
RCP 2B 6900V Switchgear 62	ECS-ES-62	2	ESF PAMS	5 min 2 wks	E
Junction Box Division A	PMS-EJ-001A	1	RT ESF PAMS	5 min 24 hrs 4 mos	E*S
Junction Box Division B	PMS-EJ-001B	1	RT ESF PAMS	5 min 24 hrs 4 mos	E*S
Junction Box Division C	PMS-EJ-001C	1	RT ESF PAMS	5 min 24 hrs 4 mos	E*S
Junction Box Division D	PMS-EJ-001D	1	RT ESF PAMS	5 min 24 hrs 4 mos	E*S
Junction Box Division A	PMS-EJ-002A	1	RT ESF PAMS	5 min 24 hrs 4 mos	E * S

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Table 3.11-1 (Sheet 6 of 52)
Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Junction Box Division B	PMS-EJ-002B	1	RT ESF PAMS	5 min 24 hrs 4 mos	E * S
Junction Box Division C	PMS-EJ-002C	1	RT ESF PAMS	5 min 24 hrs 4 mos	E * S
Junction Box Division D	PMS-EJ-002D	1	RT ESF PAMS	5 min 24 hrs 4 mos	E * S
Reactor Trip Switchgear	PMS-JD-RTSA01	4	RT PAMS	5 min 2 wks	E
Reactor Trip Switchgear	PMS-JD-RTSA02	4	RT PAMS	5 min 2 wks	E
Reactor Trip Switchgear	PMS-JD-RTSB01	4	RT PAMS	5 min 2 wks	E
Reactor Trip Switchgear	PMS-JD-RTSB02	4	RT PAMS	5 min 2 wks	E
Reactor Trip Switchgear	PMS-JD-RTSC01	4	RT PAMS	5 min 2 wks	E
Reactor Trip Switchgear	PMS-JD-RTSC02	4	RT PAMS	5 min 2 wks	E
Reactor Trip Switchgear	PMS-JD-RTSD01	4	RT PAMS	5 min 2 wks	E
Reactor Trip Switchgear	PMS-JD-RTSD02	4	RT PAMS	5 min 2 wks	E
LEVEL TRANSMITTERS					
Core Makeup Tank A Narrow Range Upper Level	PXS-JE-LE011A	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT011A	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank A Narrow Range Upper Level	PXS-JE-LE011B	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT011B	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank A Narrow Range Upper Level	PXS-JE-LE011C	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT011C	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank A Narrow Range Upper Level	PXS-JE-LE011D	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT011D	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank B Narrow Range Upper Level	PXS-JE-LE012A	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT012A	2	ESF PAMS	24 hrs 4 mos	E

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Table 3.11-1 (Sheet 7 of 52)
Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Core Makeup Tank B Narrow Range Upper Level	PXS-JE-LE012B	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT012B	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank B Narrow Range Upper Level	PXS-JE-LE012C	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT012C	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank B Narrow Range Upper Level	PXS-JE-LE012D	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT012D	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank A Narrow Range Lower Level	PXS-JE-LE013A	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT013A	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank A Narrow Range Lower Level	PXS-JE-LE013B	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT013B	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank A Narrow Range Lower Level	PXS-JE-LE013C	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT013C	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank A Narrow Range Lower Level	PXS-JE-LE013D	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT013D	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank B Narrow Range Lower Level	PXS-JE-LE014A	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT014A	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank B Narrow Range Lower Level	PXS-JE-LE014B	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT014B	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank B Narrow Range Lower Level	PXS-JE-LE014C	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT014C	2	ESF PAMS	24 hrs 4 mos	E
Core Makeup Tank B Narrow Range Lower Level	PXS-JE-LE014D	1	ESF PAMS	24 hrs 4 mos	E * S
	PXS-JE-LT014D	2	ESF PAMS	24 hrs 4 mos	E
Containment Floodup Level	PXS-JE-LE050	1	PAMS	4 mos	E * S
	PXS-JE-LT050	2	PAMS	4 mos	E
Containment Floodup Level	PXS-JE-LE051	1	PAMS	4 mos	E * S
	PXS-JE-LT051	2	PAMS	4 mos	E

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Containment Floodup Level	PXS-JE-LE052	1	PAMS	4 mos	E * S
	PXS-JE-LT052	2	PAMS	4 mos	E
NEUTRON DETECTORS					
Source Range Neutron Detector	RXS-JE-NE001A	1	RT ESF	Note 3 Note 3	E *
Source Range Neutron Detector	RXS-JE-NE001B	1	RT ESF	Note 3 Note 3	E *
Source Range Neutron Detector	RXS-JE-NE001C	1	RT ESF	Note 3 Note 3	E *
Source Range Neutron Detector	RXS-JE-NE001D	1	RT ESF	Note 3 Note 3	E *
Intermediate Range Neutron Detector	RXS-JE-NE002A	1	RT PAMS	Note 3 4 mos	E * S
Intermediate Range Neutron Detector	RXS-JE-NE002B	1	RT PAMS	Note 3 4 mos	E * S
Intermediate Range Neutron Detector	RXS-JE-NE002C	1	RT PAMS	Note 3 4 mos	E * S
Intermediate Range Neutron Detector	RXS-JE-NE002D	1	RT PAMS	Note 3 4 mos	E * S
Power Range Neutron Detector (Lower)	RXS-JE-NE003A	1	RT	5 min	E * S
Power Range Neutron Detector (Lower)	RXS-JE-NE003B	1	RT	5 min	E * S
Power Range Neutron Detector (Lower)	RXS-JE-NE003C	1	RT	5 min	E * S
Power Range Neutron Detector (Lower)	RXS-JE-NE003D	1	RT	5 min	E * S
Power Range Neutron Detector (Upper)	RXS-JE-NE004A	1	RT	5 min	E * S
Power Range Neutron Detector (Upper)	RXS–JE-NE004B	1	RT	5 min	E * S
Power Range Neutron Detector (Upper)	RXS-JE-NE004C	1	RT	5 min	E * S
Power Range Neutron Detector (Upper)	RXS-JE-NE004D	1	RT	5 min	E * S
RADIATION MONITORS					
Containment High Range Area Monitor	PXS-JE-RE160	1	ESF PAMS	24 hrs 4 mos	E * S
Containment High Range Area Monitor	PXS-JE-RE161	1	ESF PAMS	24 hrs 4 mos	E * S
Containment High Range Area Monitor	PXS-JE-RE162	1	ESF PAMS	24 hrs 4 mos	E * S
Containment High Range Area Monitor	PXS-JE-RE163	1	ESF PAMS	24 hrs 4 mos	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Control Room Supply Air Area Radiation Monitor - Particulate Detector	VBS-JE-RE001A	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Particulate Detector	VBS-JE-RE001B	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Iodine Detector	VBS-JE-RE002A	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Iodine Detector	VBS-JE-RE002B	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Noble Gas Detector	VBS-JE-RE003A	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Noble Gas Detector	VBS-JE-RE003B	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Air Flow Transmitter	VBS-JE-FT011A	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Air Flow Transmitter	VBS-JE-FT011B	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Sample Pump	VBS-MP-01A	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Sample Pump	VBS-MP-01B	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Local Radiation Processor	VBS-JE-RY001A	2	ESF	24 hrs	E
Control Room Supply Air Area Radiation Monitor - Local Radiation Processor	VBS-JE-RY001B	2	ESF	24 hrs	E
RESISTANCE TEMPERATURE DETECTORS					
PRHR HX Outlet Temperature	RCS-JE-TE161	1	PAMS	4 mos	E * S
RCS Cold Leg 1A Narrow Range Temperature	RCS-JE-TE121A	1	RT ESF	5 min 5 min	E * S
RCS Cold Leg 1A Narrow Range Temperature	RCS-JE-TE121D	1	RT ESF	5 min 5 min	E * S
RCS Cold Leg 1B Narrow Range Temperature	RCS-JE-TE121B	1	RT ESF	5 min 5 min	E * S
RCS Cold Leg 1B Narrow Range Temperature	RCS-JE-TE121C	1	RT ESF	5 min 5 min	E * S
RCS Cold Leg 2A Narrow Range Temperature	RCS-JE-TE122B	1	RT ESF	5 min 5 min	E * S
RCS Cold Leg 2A Narrow Range Temperature	RCS-JE-TE122C	1	RT ESF	5 min 5 min	E * S
RCS Cold Leg 2B Narrow Range Temperature	RCS-JE-TE122A	1	RT ESF	5 min 5 min	E * S
RCS Cold Leg 2B Narrow Range Temperature	RCS-JE-TE122D	1	RT ESF	5 min 5 min	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE131A	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE131C	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE132A	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE132C	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE133C	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE133A	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE131B	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE131D	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE132B	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE132D	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE133B	1	RT ESF	5 min 5 min	E * S
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE133D	1	RT ESF	5 min 5 min	E * S
RCS Cold Leg 1A Dual Range Temperature	RCS-JE-TE125A	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
RCS Cold Leg 1B Dual Range Temperature	RCS-JE-TE125C	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
RCS Cold Leg 2A Dual Range Temperature	RCS-JE-TE125B	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
RCS Cold Leg 2B Dual Range Temperature	RCS-JE-TE125D	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
RCS Hot Leg 1 Wide Range Temperature	RCS-JE-TE135A	1	PAMS	4 mos	E * S
RCS Hot Leg 2 Wide Range Temperature	RCS-JE-TE135B	1	PAMS	4 mos	E * S
PZR Reference Leg Level Temperature	RCS-JE-TE193A	1	RT ESF PAMS	5 min 5 min 4 mos	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
PZR Reference Leg Level Temperature	RCS-JE-TE193B	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
PZR Reference Leg Level Temperature	RCS-JE-TE193C	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
PZR Reference Leg Level Temperature	RCS-JE-TE193D	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
RCP 1A Bearing Water Temperature	RCS-JE-TE211A	1	RT ESF	Note 3 5 min	E *
RCP 1A Bearing Water Temperature	RCS-JE-TE211B	1	RT ESF	Note 3 5 min	E *
RCP 1A Bearing Water Temperature	RCS-JE-TE211C	1	RT ESF	Note 3 5 min	E *
RCP 1A Bearing Water Temperature	RCS-JE-TE211D	1	RT ESF	Note 3 5 min	E *
RCP 1B Bearing Water Temperature	RCS-JE-TE212A	1	RT ESF	Note 3 5 min	E *
RCP 1B Bearing Water Temperature	RCS-JE-TE212B	1	RT ESF	Note 3 5 min	E *
RCP 1B Bearing Water Temperature	RCS-JE-TE212C	1	RT ESF	Note 3 5 min	E *
RCP 1B Bearing Water Temperature	RCS-JE-TE212D	1	RT ESF	Note 3 5 min	E *
RCP 2A Bearing Water Temperature	RCS-JE-TE213A	1	RT ESF	Note 3 5 min	E *
RCP 2A Bearing Water Temperature	RCS-JE-TE213B	1	RT ESF	Note 3 5 min	E *
RCP 2A Bearing Water Temperature	RCS-JE-TE213C	1	RT ESF	Note 3 5 min	E *
RCP 2A Bearing Water Temperature	RCS-JE-TE213D	1	RT ESF	Note 3 5 min	E *
RCP 2B Bearing Water Temperature	RCS-JE-TE214A	1	RT ESF	Note 3 5 min	E *
RCP 2B Bearing Water Temperature	RCS-JE-TE214B	1	RT ESF	Note 3 5 min	E *
RCP 2B Bearing Water Temperature	RCS-JE-TE214C	1	RT ESF	Note 3 5 min	E *
RCP 2B Bearing Water Temperature	RCS-JE-TE214D	1	RT ESF	Note 3 5 min	E *
Containment Operating Floor Temperature	VCS-JE-TE052A	1	PAMS	4 mos	E *
Containment Operating Floor Temperature	VCS-JE-TE052B	1	PAMS	4 mos	E *

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Steam Generator 1 Area Containment Temperature	VCS-JE-TE054	1	PAMS	4 mos	E *
Pressurizer Area Containment Temperature	VCS-JE-TE055	1	PAMS	4 mos	E *
CVS Module Containment Temperature	VCS-JE-TE056	1	PAMS	4 mos	E *
Accumulator Room B Containment Temperature	VCS-JE-TE057	1	PAMS	4 mos	E *
Steam Generator 2 Area Containment Temperature	VCS-JE-TE058	1	PAMS	4 mos	E *
Accumulator Room A Containment Temperature	VCS-JE-TE-059	1	PAMS	4 mos	E *
SPEED SENSORS					
RCP 1A Pump Speed	RCS-JE-ST281	1	RT	Note 3	E *
RCP 1B Pump Speed	RCS-JE-ST282	1	RT	Note 3	E *
RCP 2A Pump Speed	RCS-JE-ST283	1	RT	Note 3	E *
RCP 2B Pump Speed	RCS-JE-ST284	1	RT	Note 3	E *
RCP 1A Speed Sensor Preamplifier Enclosure	PMS-JD-MIS01A	2	RT	Note 3	E
RCP 1B Speed Sensor Preamplifier Enclosure	PMS-JD-MIS01B	2	RT	Note 3	E
RCP 2A Speed Sensor Preamplifier Enclosure	PMS-JD-MIS02A	2	RT	Note 3	E
RCP 2B Speed Sensor Preamplifier Enclosure	PMS-JD-MIS02B	2	RT	Note 3	E
THERMOCOUPLES					
Incore Thermocouples (Note 4)	IIS-JE-TE001 through IIS-JE-TE042	1	PAMS	1 yr	E * S
TRANSMITTERS					
PCS Water Delivery Flow	PCS-JE-FT001	9	PAMS	2 wks	E **
PCS Water Delivery Flow	PCS-JE-FT002	9	PAMS	2 wks	E **
PCS Water Delivery Flow	PCS-JE-FT003	9	PAMS	2 wks	E **
PCS Water Delivery Flow	PCS-JE-FT004	9	PAMS	2 wks	E **
PCS Storage Tank Water Level	PCS-JE-LT010	9	PAMS	2 wks	E **
PCS Storage Tank Water Level	PCS-JE-LT011	9	PAMS	2 wks	E **
PRHR HX Flow	PXS-JE-FT049A	1	PAMS	4 mos	E * S
PRHR HX Flow	PXS-JE-FT049B	1	PAMS	4 mos	E * S
RCS Hot Leg 1 Flow	RCS-JE-FT101A	1	RT	Note 3	E *
RCS Hot Leg 1 Flow	RCS-JE-FT101B	1	RT	Note 3	E *
RCS Hot Leg 1 Flow	RCS-JE-FT101C	1	RT	Note 3	E *
RCS Hot Leg 1 Flow	RCS-JE-FT101D	1	RT	Note 3	E *
RCS Hot Leg 2 Flow	RCS-JE-FT102A	1	RT	Note 3	E *
RCS Hot Leg 2 Flow	RCS-JE-FT102B	1	RT	Note 3	E *
RCS Hot Leg 2 Flow	RCS-JE-FT102C	1	RT	Note 3	E *
RCS Hot Leg 2 Flow	RCS-JE-FT102D	1	RT	Note 3	E *

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
SG1 Startup Feedwater Flow	SGS-JE-FT055A	2	ESF PAMS	5 min 2 wks	E
SG1 Startup Feedwater Flow	SGS-JE-FT055B	2	ESF PAMS	5 min 2 wks	E
SG2 Startup Feedwater Flow	SGS-JE-FT056A	2	ESF PAMS	5 min 2 wks	E
SG2 Startup Feedwater Flow	SGS-JE-FT056B	2	ESF PAMS	5 min 2 wks	E
Flexible Hose from SG1 to Upper Tap LT001/LT011	SGS-PY-Y01A	1	PB	1 yr	M *
Flexible Hose from SG2 to Upper Tap LT005/LT013	SGS-PY-Y01B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap LT001	SGS-PY-Y02A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap LT005	SGS-PY-Y02B	1	PB	1 yr	M *
Flexible Hose from SG1 to Upper Tap LT002/LT015	SGS-PY-Y03A	1	PB	1 yr	M *
Flexible Hose from SG2 to Upper Tap LT006/LT014	SGS-PY-Y03B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap LT002	SGS-PY-Y04A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap LT006	SGS-PY-Y04B	1	PB	1 yr	M *
Flexible Hose from SG1 to Upper Tap LT003/LT012/LT044	SGS-PY-Y05A	1	PB	1 yr	M *
Flexible Hose from SG2 to Upper Tap LT007/LT017/LT046	SGS-PY-Y05B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap LT003	SGS-PY-Y06A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap LT007	SGS-PY-Y06B	1	PB	1 yr	M *
Flexible Hose from SG1 to Upper Tap LT004/LT016/LT045	SGS-PY-Y07A	1	PB	1 yr	M *
Flexible Hose from SG2 to Upper Tap LT008/LT018/LT047	SGS-PY-Y07B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap LT004	SGS-PY-Y08A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap LT008	SGS-PY-Y08B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap LT011	SGS-PY-Y09A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap LT013	SGS-PY-Y09B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap LT012	SGS-PY-Y10A	1	PB	1 yr	M *

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Flexible Hose from SG2 to Lower Tap LT014	SGS-PY-Y10B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap LT015/LT044	SGS-PY-Y11A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap LT017/LT046	SGS-PY-Y11B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap LT016/LT045	SGS-PY-Y12A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap LT018/LT047	SGS-PY-Y12B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap FT024	SGS-PY-Y13A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap FT025	SGS-PY-Y13B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap FT021	SGS-PY-Y14A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap FT023	SGS-PY-Y14B	1	PB	1 yr	M *
Flexible Hose from SG1 to Lower Tap FT020	SGS-PY-Y15A	1	PB	1 yr	M *
Flexible Hose from SG2 to Lower Tap FT022	SGS-PY-Y15B	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to Upper Tap FT020	SGS-PY-Y16A	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to Upper Tap FT022	SGS-PY-Y16B	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to Upper Tap FT021	SGS-PY-Y17A	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to Upper Tap FT023	SGS-PY-Y17B	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to Upper Tap FT024	SGS-PY-Y18A	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to Upper Tap FT025	SGS-PY-Y18B	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to PT030	SGS-PY-Y19A	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to PT034	SGS-PY-Y19B	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to PT032	SGS-PY-Y20A	1	PB	1 yr	M *
Flexible Hose from Main Steam Line to PT036	SGS-PY-Y20B	1	PB	1 yr	M *
MCR Air Delivery Line Flowrate – A	VES-JE-FT003A	3	PAMS	72 hrs	E
MCR Air Delivery Line Flowrate – B	VES-JE-FT003B	3	PAMS	72 hrs	E
IRWST Wide Range Level	PXS-JE-LT046	1	PAMS	4 mos	E * S
IRWST Wide Range Level	PXS-JE-LT047	1	PAMS ESF	4 mos 24 hrs	E * S

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IRWST Wide Range Level	PXS-JE-LT048	1	PAMS ESF	4 mos 24 hrs	E * S
IRWST Lower Narrow Range Level	PXS-JE-LT066	1	ESF PAMS	24 hrs 24 hrs	E * S
IRWST Lower Narrow Range Level	PXS-JE-LT067	1	ESF PAMS	24 hrs 24 hrs	E * S
IRWST Lower Narrow Range Level	PXS-JE-LT068	1	ESF PAMS	24 hrs 24 hrs	E * S
IRWST Lower Narrow Range Level	PXS-JE-LT069	1	ESF PAMS	24 hrs 24 hrs	E * S
RCS Hot Leg Water Level	RCS-JE-LT160A	1	PAMS	2 wks	E * S
RCS Hot Leg Water Level	RCS-JE-LT160B	1	PAMS	2 wks	E * S
PZR Level	RCS-JE-LT195A	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
PZR Level	RCS-JE-LT195B	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
PZR Level	RCS-JE-LT195C	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
PZR Level	RCS-JE-LT195D	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
SG1 Narrow Range Level	SGS-JE-LT001	1	RT ESF PAMS	5 min 5 min 2 wks	E * S
SG1 Narrow Range Level	SGS-JE-LT002	1	RT ESF PAMS	5 min 5 min 2 wks	E * S
SG1 Narrow Range Level	SGS-JE-LT003	1	RT ESF PAMS	5 min 5 min 2 wks	E * S
SG1 Narrow Range Level	SGS-JE-LT004	1	RT ESF PAMS	5 min 5 min 2 wks	E * S
SG2 Narrow Range Level	SGS-JE-LT005	1	RT ESF PAMS	5 min 5 min 2 wks	E * S
SG2 Narrow Range Level	SGS-JE-LT006	1	RT ESF PAMS	5 min 5 min 2 wks	E * S
SG2 Narrow Range Level	SGS-JE-LT007	1	RT ESF PAMS	5 min 5 min 2 wks	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
SG2 Narrow Range Level	SGS-JE-LT008	1	RT ESF PAMS	5 min 5 min 2 wks	E * S
SG1 Wide Range Level	SGS-JE-LT011	1	ESF PAMS	5 min 2 wks	E * S
SG1 Wide Range Level	SGS-JE-LT012	1	ESF PAMS	5 min 2 wks	E * S
SG1 Wide Range Level	SGS-JE-LT015	1	ESF PAMS	5 min 2 wks	E * S
SG1 Wide Range Level	SGS-JE-LT016	1	ESF PAMS	5 min 2 wks	E * S
SG2 Wide Range Level	SGS-JE-LT013	1	ESF PAMS	5 min 2 wks	E * S
SG2 Wide Range Level	SGS-JE-LT014	1	ESF PAMS	5 min 2 wks	E * S
SG2 Wide Range Level	SGS-JE-LT017	1	ESF PAMS	5 min 2 wks	E * S
SG2 Wide Range Level	SGS-JE-LT018	1	ESF PAMS	5 min 2 wks	E * S
Spent Fuel Pool Level	SFS-JE-LT019A	6	ESF PAMS	24 hrs 2 wks	E ** S
Spent Fuel Pool Level	SFS-JE-LT019B	7	ESF PAMS	24 hrs 2 wks	E ** S
Spent Fuel Pool Level	SFS-JE-LT019C	6	ESF PAMS	24 hrs 2 wks	E ** S
WLS Auxiliary Building RCA Floodup Level	WLS-JE-LT400A	6	(Note 8)	24 hrs	E ** S
WLS Auxiliary Building RCA Floodup Level	WLS-JE-LT400B	6	(Note 8)	24 hrs	E ** S
Containment Pressure Normal Range	PCS-JE-PT005	7	ESF PAMS	5 min 4 mos	E
Containment Pressure Normal Range	PCS-JE-PT006	7	ESF PAMS	5 min 4 mos	E *
Containment Pressure Normal Range	PCS-JE-PT007	7	ESF PAMS	5 min 4 mos	E
Containment Pressure Normal Range	PCS-JE-PT008	7	ESF PAMS	5 min 4 mos	E *
Containment Pressure Extended Range	PCS-JE-PT012	7	PAMS	4 mos	E
Containment Pressure Extended Range	PCS-JE-PT013	7	PAMS	4 mos	E *
Containment Pressure Extended Range	PCS-JE-PT014	7	PAMS	4 mos	E
Containment Pressure Sensor (Shared Diaphragm)	PCS-JE-PT005 PCS-JE-PT012	1	ESF PAMS	5 min 4 mos	E *

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Containment Pressure Sensor (Shared Diaphragm)	PCS-JE-PT006 PCS-JE-PT013	1	ESF PAMS	5 min 4 mos	E *
Containment Pressure Sensor (Shared Diaphragm)	PCS-JE-PT007 PCS-JE-PT014	1	ESF PAMS	5 min 4 mos	E *
Containment Pressure Sensor (Diaphragm)	PCS-JE-PT008	1	ESF PAMS	5 min 4 mos	E *
Accumulator A Pressure Transmitter	PXS-JE-PT027	1	PAMS	4 mos	E *
Accumulator B Pressure Transmitter	PXS-JE-PT028	1	PAMS	4 mos	E *
Accumulator A Pressure Transmitter	PXS-JE-PT029	1	PAMS	4 mos	E *
Accumulator B Pressure Transmitter	PXS-JE-PT030	1	PAMS	4 mos	E *
RCS Wide Range Pressure	RCS-JE-PT140A	1	PAMS ESF	4 mos 5 min	E * S
RCS Wide Range Pressure	RCS-JE-PT140B	1	PAMS ESF	4 mos 5 min	E * S
RCS Wide Range Pressure	RCS-JE-PT140C	1	PAMS ESF	4 mos 5 min	E * S
RCS Wide Range Pressure	RCS-JE-PT140D	1	PAMS ESF	4 mos 5 min	E * S
PZR Pressure	RCS-JE-PT191A	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
PZR Pressure	RCS-JE-PT191B	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
PZR Pressure	RCS-JE-PT191C	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
PZR Pressure	RCS-JE-PT191D	1	RT ESF PAMS	5 min 5 min 4 mos	E * S
Main Steam Line SG1 Pressure	SGS-JE-PT030	1	ESF PAMS	5 min 2 wks	E * S
Main Steam Line SG1 Pressure	SGS-JE-PT031	2	ESF PAMS	5 min 2 wks	E
Main Steam Line SG1 Pressure	SGS-JE-PT032	1	ESF PAMS	5 min 2 wks	E * S
Main Steam Line SG1 Pressure	SGS-JE-PT033	2	ESF PAMS	5 min 2 wks	E
Main Steam Line SG2 Pressure	SGS-JE-PT034	1	ESF PAMS	5 min 2 wks	E * S
Main Steam Line SG2 Pressure	SGS-JE-PT035	2	ESF PAMS	5 min 2 wks	E
Main Steam Line SG2 Pressure	SGS-JE-PT036	1	ESF PAMS	5 min 2 wks	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Main Steam Line SG2 Pressure	SGS-JE-PT037	2	ESF PAMS	5 min 2 wks	E
Main Control Room Differential Pressure	VES-JE-PDT004A	3	ESF PAMS	2 wks 2 wks	E
Main Control Room Differential Pressure	VES-JE-PDT004B	3	ESF PAMS	2 wks 2 wks	E
PROTECTION AND SAFETY MONITORING SYSTEMS					
Bistable Coincident Logic Cabinet 01 Division A	PMS-JD-BCCA01	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Bistable Coincident Logic Cabinet 02 Division A	PMS-JD-BCCA02	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Bistable Coincident Logic Cabinet 01 Division B	PMS-JD-BCCB01	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Bistable Coincident Logic Cabinet 02 Division B	PMS-JD-BCCB02	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Bistable Coincident Logic Cabinet 01 Division C	PMS-JD-BCCC01	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Bistable Coincident Logic Cabinet 02 Division C	PMS-JD-BCCC02	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Bistable Coincident Logic Cabinet 01 Division D	PMS-JD-BCCD01	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Bistable Coincident Logic Cabinet 02 Division D	PMS-JD-BCCD02	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Integrated Logic Cabinet 01 Division A	PMS-JD-ILCA01	2	ESF	24 hrs	E
Integrated Logic Cabinet 02 Division A	PMS-JD-ILCA02	2	ESF	24 hrs	E
Integrated Logic Cabinet 03 Division A	PMS-JD-ILCA03	2	ESF	24 hrs	E
Integrated Logic Cabinet 01 Division B	PMS-JD-ILCB01	2	ESF	24 hrs	E
Integrated Logic Cabinet 02 Division B	PMS-JD-ILCB02	2	ESF	24 hrs	E
Integrated Logic Cabinet 03 Division B	PMS-JD-ILCB03	2	ESF	24 hrs	E
Integrated Logic Cabinet 04 Division B	PMS-JD-ILCB04	2	ESF	24 hrs	E
Integrated Logic Cabinet 01 Division C	PMS-JD-ILCC01	2	ESF	24 hrs	E
Integrated Logic Cabinet 02 Division C	PMS-JD-ILCC02	2	ESF	24 hrs	E
Integrated Logic Cabinet 01 Division D	PMS-JD-ILCD01	2	ESF	24 hrs	E
Integrated Logic Cabinet 02 Division D	PMS-JD-ILCD02	2	ESF	24 hrs	E
Integrated Logic Cabinet 03 Division D	PMS-JD-ILCD03	2	ESF	24 hrs	E
Integrated Logic Cabinet 04 Division D	PMS-JD-ILCD04	2	ESF	24 hrs	E
Maintenance and Test Cabinet Division A	PMS-JD-MTCA01	2	PAMS	2 wks	E
Maintenance and Test Cabinet Division B	PMS-JD-MTCB01	2	PAMS	2 wks	E
Maintenance and Test Cabinet Division C	PMS-JD-MTCC01	2	PAMS	2 wks	E

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Maintenance and Test Cabinet Division D	PMS-JD-MTCD01	2	PAMS	2 wks	E
Nuclear Instrumentation Cabinet Division A	PMS-JD-NICA01	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Nuclear Instrumentation Cabinet Division B	PMS-JD-NICB01	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Nuclear Instrumentation Cabinet Division C	PMS-JD-NICC01	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Nuclear Instrumentation Cabinet Division D	PMS-JD-NICD01	2	RT ESF PAMS	5 min 24 hrs 2 wks	E
Qualified Data Processing System Cabinet Division B	PMS-JD-QDPB01	2	ESF PAMS	24 hrs 2 wks	E
Qualified Data Processing System Cabinet Division C	PMS-JD-QDPC01	2	ESF PAMS	24 hrs 2 wks	E
Division A Squib Valve Controller Cabinet	PMS-JD-SVCA01	2	ESF	24 hrs	E
Division B Squib Valve Controller Cabinet	PMS-JD-SVCB01	2	ESF	24 hrs	E
Division C Squib Valve Controller Cabinet	PMS-JD-SVCC01	2	ESF	24 hrs	E
Division D Squib Valve Controller Cabinet	PMS-JD-SVCD01	2	ESF	24 hrs	E
Source Range Neutron Flux Preamplifier Panel A	PMS-JW-005A	2	RT, ESF	Note 3	E
Source Range Neutron Flux Preamplifier Panel B	PMS-JW-005B	2	RT, ESF	Note 3	E
Source Range Neutron Flux Preamplifier Panel C	PMS-JW-005C	2	RT, ESF	Note 3	E
Source Range Neutron Flux Preamplifier Panel D	PMS-JW-005D	2	RT, ESF	Note 3	E
Intermediate Range Neutron Flux Preamplifier Panel A	PMS-JW-006A	2	RT PAMS	Note 3 2 wks	E
Intermediate Range Neutron Flux Preamplifier Panel B	PMS-JW-006B	2	RT PAMS	Note 3 2 wks	E
Intermediate Range Neutron Flux Preamplifier Panel C	PMS-JW-006C	2	RT PAMS	Note 3 2 wks	E
Intermediate Range Neutron Flux Preamplifier Panel D	PMS-JW-006D	2	RT PAMS	Note 3 2 wks	E
MAIN CONTROL ROOM					
Operator Workstation A	N/A	3	RT ESF PAMS	5 min 24 hrs 2 wks	E
Operator Workstation B	N/A	3	RT ESF PAMS	5 min 24 hrs 2 wks	E

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Supervisor Workstation	N/A	3	RT ESF PAMS	5 min 24 hrs 2 wks	E
Primary Dedicated Safety Panel	OCS-JC-10	3	RT ESF	5 min 24 hrs	E
Secondary Dedicated Safety Panel	OCS-JC-11	3	RT ESF	5 min 24 hrs	E
MCR/RSR Transfer Switch Panel	OCS-JW-001	2	RT ESF	5 min 24 hrs	E
Qualified Safety Display Division A	PMS-JD-QSPA01	3	ESF PAMS	24 hrs 2 wks	E
Qualified Safety Display Division B	PMS-JD-QSPB01	3	ESF PAMS	24 hrs 2 wks	E
Qualified Safety Display Division C	PMS-JD-QSPC01	3	ESF PAMS	24 hrs 2 wks	E
Qualified Safety Display Division D	PMS-JD-QSPD01	3	ESF PAMS	24 hrs 2 wks	E
MCR Load Shed Panel 1	VES-EP-01	2	ESF PAMS	24 hrs 2 wks	E S
MCR Load Shed Panel 2	VES-EP-02	2	ESF PAMS	24 hrs 2 wks	E
PENETRATIONS					
Penetrations (Mechanical)	See Table 6.2.3-1				M * S
Equipment Hatch	CNS-MY-Y01	1, 7, 9	PB	4 mos	M * S
Maintenance Hatch	CNS-MY-Y02	1, 7	PB	4 mos	M * S
Upper Personnel Hatch – 135'-3"	CNS-MY-Y03	1, 7, 9	PB	4 mos	M * S
Lower Personnel Hatch – 107'-2"	CNS-MY-Y04	1, 7	PB	4 mos	M * S
Penetrations (Electrical)	See Figure 3.8.2-4				E * S
ACTIVE VALVES					
Containment Isolation – Air Out	CAS-PL-V014	2	ESF	5 min	M **
Solenoid Valve	CAS-PL-V014-S	2	ESF	5 min	E
Limit Switch	CAS-PL-V014-L	2	PAMS	2 wks	E
Containment Isolation – Air In	CAS-PL-V015	1	ESF	24 hrs	M * S
Containment Isolation – Inlet	CCS-PL-V200	2	ESF	5 min	M **
Limit Switch	CCS-PL-V200-L	2	PAMS	2 wks	E
Motor Operator	CCS-PL-V200-M	2	ESF	5 min	E
Service Air Supply Inside Containment Isolation	CAS-PL-V205	1	ESF	24 hrs	M * S
Containment Isolation – Inlet	CCS-PL-V201	1	ESF	24 hrs	M * S
Containment Isolation – Outlet	CCS-PL-V207	1	ESF	5 min	M * S
Limit Switch	CCS-PL-V207-L	1	PAMS	1 yr	E * S
Motor Operator	CCS-PL-V207-M	1	ESF	5 min	E * S
Containment Isolation – Outlet	CCS-PL-V208	2	ESF	5 min	M **
Limit Switch	CCS-PL-V208-L	2	PAMS	2 wks	E
Motor Operator	CCS-PL-V208-M	2	ESF	5 min	E

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
CCS Containment Isolation Relief	CCS-PL-V220	1	ESF	24 hrs	M * S
CCS IRC Relief Valve	CCS-PL-V270	1	ESF	24 hrs	M* S
CCS IRC Relief Valve	CCS-PL-V271	1	ESF	24 hrs	M* S
RCS Purification Stop Valve	CVS-PL-V001	1	ESF	5 min	M * S
Limit Switch	CVS-PL-V001-L	1	PAMS	1 yr	E * S
Motor Operator	CVS-PL-V001-M	1	ESF	5 min	E * S
RCS Purification Stop Valve	CVS-PL-V002	1	ESF	5 min	M * S
Limit Switch	CVS-PL-V002-L	1	PAMS	1 yr	E * S
Motor Operator	CVS-PL-V002-M	1	ESF	5 min	E * S
RCS Letdown Stop Valve	CVS-PL-V003	1	ESF	5 min	M * S
Limit Switch	CVS-PL-V003-L	1	PAMS	1 yr	E * S
Motor Operator	CVS-PL-V003-M	1	ESF	5 min	E * S
Demineralizer Flush Line Relief Valve	CVS-PL-V042	1	ESF	24 hrs	M * S
WLS Letdown IRC Isolation	CVS-PL-V045	1	ESF	5 min	M * S
Limit Switch	CVS-PL-V045-L	1	PAMS	1 yr	E * S
Solenoid Valve	CVS-PL-V045-S1	1	ESF	5 min	E * S
Letdown Flow ORC Isolation	CVS-PL-V047	7	ESF	5 min	M ** S
Limit Switch	CVS-PL-V047-L	7	PAMS	2 wks	E ** S
Solenoid Valve	CVS-PL-V047-S1	7	ESF	5 min	E ** S
Letdown Line Containment Isolation Thermal Relief	CVS-PL-V058	1	ESF	24 hrs	M * S
CVS Makeup Line Bypass Check Valve	CVS-PL-V067	1	ESF	5 min	M* S
RCS Purification Check Valve	CVS-PL-V080	1	ESF	5 min	M * S
RCS Purification Stop Valve	CVS-PL-V081	1	ESF	5 min	M * S
RCS Purification Check Valve	CVS-PL-V082	1	ESF	5 min	M * S
Auxiliary PZR Spray Isolation	CVS-PL-V084	1	ESF	5 min	M * S
Limit Switch	CVS-PL-V084-L	1	PAMS	1 yr	E * S
Solenoid Valve	CVS-PL-V084-S	1	ESF	5 min	E * S
Auxiliary PZR Spray Isolation	CVS-PL-V085	1	ESF	5 min	M * S
Makeup Line Containment Isolation	CVS-PL-V090	7	ESF	5 min	M ** S
Limit Switch	CVS-PL-V090-L	7	PAMS	2 wks	E ** S
Motor Operator	CVS-PL-V090-M	7	ESF	5 min	E ** S
Makeup Line Containment Isolation	CVS-PL-V091	1	ESF	5 min	M * S
Limit Switch	CVS-PL-V091-L	1	PAMS	1 yr	E * S
Motor Operator	CVS-PL-V091-M	1	ESF	5 min	E * S
Zinc Injection Containment Isolation	CVS-PL-V092	10	ESF	5 min	M * S
Limit Switch	CVS-PL-V092-L	10	PAMS	2 wks	E * S
Solenoid Valve	CVS-PL-V092-S	10	ESF	5 min	E * S
Zinc Injection Containment Isolation IRC	CVS-PL-V094	1	ESF	5 min	M * S
Limit Switch	CVS-PL-V094-L	1	PAMS	1 yr	E * S
Solenoid Valve	CVS-PL-V094-S	1	ESF	5 min	E * S
Zinc Injection Containment Isolation Thermal Overpressurization Relief Valve	CVS-PL-V098	1	ESF	24 hrs	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Makeup Containment Isolation	CVS-PL-V100	1	ESF	24 hrs	M * S
Demineralizer Water System Isolation	CVS-PL-V136A	7	ESF	5 min	M **
Limit Switch	CVS-PL-V136A-L	7	PAMS	2 wks	E **
Solenoid Valve	CVS-PL-V136A-S	7	ESF	5 min	E **
Demineralized Water System Isolation	CVS-PL-V136B	7	ESF	5 min	M **
Limit Switch	CVS-PL-V136B-L	7	PAMS	2 wks	E **
Solenoid Valve	CVS-PL-V136B-S	7	ESF	5 min	E **
Hydrogen Injection Containment Isolation Check IRC	CVS-PL-V217	1	ESF	24 hrs	M * S
Hydrogen Injection Containment Isolation	CVS-PL-V219	10	ESF	5 min	M * S
Limit Switch	CVS-PL-V219-L	10	PAMS	2 wks	E * S
Solenoid Valve	CVS-PL-V219-S	10	ESF	5 min	E * S
DWS Containment Penetration Thermal Relief Valve	DWS-PL-V241	1	ESF	24 hrs	M * S
Demin Water Supply Containment Isolation – Inside	DWS-PL-V245	1	ESF	24 hrs	M * S
Fuel Transfer Tube Gate Valve	FHS-PL-V001	11	ESF	2 wks	M **
Fire Water Containment Supply Isolation – Inside	FPS-PL-V052	1	ESF	24 hrs	M * S
FPS Containment Penetration Thermal Relief Valve	FPS-PL-V702	1	ESF	24 hrs	M * S
PCCWST Isolation Valve	PCS-PL-V001A	9	ESF	5 min	M **
Limit Switch	PCS-PL-V001A-L	9	PAMS	2 wks	E **
Solenoid Valve	PCS-PL-V001A-S1	9	ESF	5 min	E **
PCCWST Isolation Valve	PCS-PL-V001B	9	ESF	5 min	M **
Limit Switch	PCS-PL-V001B-L	9	PAMS	2 wks	E **
Solenoid Valve	PCS-PL-V001B-S1	9	ESF	5 min	E **
PCCWST Isolation Valve	PCS-PL-V001C	9	ESF	5 min	M **
Limit Switch	PCS-PL-V001C-L	9	PAMS	2 wks	E **
Motor Operator	PCS-PL-V001C-M	9	ESF	5 min	E **
PCCWST Isolation Valve	PCS-PL-V002A	9	ESF	5 min	M **
Limit Switch	PCS-PL-V002A-L	9	PAMS	2 wks	E **
Motor Operator	PCS-PL-V002A-M	9	ESF	5 min	E **
PCCWST Isolation Valve	PCS-PL-V002B	9	ESF	5 min	M **
Limit Switch	PCS-PL-V002B-L	9	PAMS	2 wks	E **
Motor Operator	PCS-PL-V002B-M	9	ESF	5 min	E **
PCCWST Isolation Valve	PCS-PL-V002C	9	ESF	5 min	M **
Limit Switch	PCS-PL-V002C-L	9	PAMS	2 wks	E **
Motor Operator	PCS-PL-V002C-M	9	ESF	5 min	E **
PCCWST Fire Protection Isolation	PCS-PL-V005	10	ESF	72 hrs	M * S
PCCWST Emergency Spent Fuel Pool Makeup Isolation	PCS-PL-V009	9	ESF	2 wks	M **
Water Bucket Makeup Line Drain Valve	PCS-PL-V015	10	ESF	2 wks	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Water Bucket Makeup Line Isolation Valve	PCS-PL-V020	10	ESF	2 wks	M * S
PCS Recirculation Isolation	PCS-PL-V023	10	ESF	72 hrs	M * S
PCCWST Long-Term Makeup Check Valve	PCS-PL-V039	10	ESF	2 wks	M * S
PCCWST Long Term Makeup Isolation Drain Valve	PCS-PL-V042	10	ESF	2 wks	M * S
PCCWST Long Term Makeup Isolation Valve	PCS-PL-V044	10	ESF	2 wks	M * S
Emergency Makeup to the Spent Fuel Pool Isolation Valve	PCS-PL-V045	7	ESF	2 wks	M
PCCWST Recirculation Return Isolation Valve	PCS-PL-V046	10	ESF	2 wks	M * S
Emergency Makeup to the Spent Fuel Pool Drain Isolation Valve	PCS-PL-V049	7	ESF	2 wks	M
Spent Fuel Pool Long Term Makeup Isolation Valve	PCS-PL-V050	10	ESF	2 wks	M * S
Spent Fuel Pool Emergency Makeup Lower Isolation Valve	PCS-PL-V051	7	ESF	2 wks	M
Containment Isolation – Air Sample Line	PSS-PL-V008	1	ESF	4 mos	M * S
Limit Switch	PSS-PL-V008-L	1	PAMS	1 yr	E * S
Solenoid Operator	PSS-PL-V008-S	1	ESF	5 min	E * S
Voltage Control Box	PSS-DJ-05	2	ESF	5 min	E
Containment Isolation – Liquid Sample Line	PSS-PL-V010A	1	ESF	4 mos	M * S
Limit Switch	PSS-PL-V010A-L	1	PAMS	1 yr	E * S
Solenoid Operator	PSS-PL-V010A-S	1	ESF	5 min	E * S
Voltage Control Box	PSS-DJ-03	2	ESF	5 min	E
Containment Isolation – Liquid Sample Line	PSS-PL-V010B	1	ESF	4 mos	M * S
Limit Switch	PSS-PL-V010B-L	1	PAMS	1 yr	E * S
Solenoid Operator	PSS-PL-V010B-S	1	ESF	5 min	E * S
Voltage Control Box	PSS-DJ-04	2	ESF	5 min	E
Containment Isolation – Liquid Sample Line	PSS-PL-V011A	6	ESF	2 wks	M **
Limit Switch	PSS-PL-V011A-L	6	PAMS	2 wks	E **
Solenoid Valve	PSS-PL-V011A-S	6	ESF	5 min	E **
Containment Isolation – Liquid Sample Line	PSS-PL-V011B	6	ESF	2 wks	M **
Limit Switch	PSS-PL-V011B-L	6	PAMS	2 wks	E **
Solenoid Valve	PSS-PL-V011B-S	6	ESF	5 min	E **
Containment Isolation - Sample Return Line	PSS-PL-V023	6	ESF	2 wks	M **
Limit Switch	PSS-PL-V023-L	6	PAMS	2 wks	E **
Solenoid Valve	PSS-PL-V023-S	6	ESF	5 min	E **

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Containment Isolation – Sample Return Line	PSS-PL-V024	1	ESF	4 mos	M * S
Limit Switch	PSS-PL-V024-L	1	PAMS	1 yr	E * S
Solenoid Operator	PSS-PL-V024-S	1	ESF	5 min	E * S
Voltage Control Box	PSS-DJ-06	2	ESF	5 min	E
Containment Isolation - Air Sample Line	PSS-PL-V046	6	ESF	2 wks	M **
Limit Switch	PSS-PL-V046-L	6	PAMS	2 wks	E **
Solenoid Valve	PSS-PL-V046-S	6	ESF	2 wks	E **
PWS MCR Isolation Valve	PWS-PL-V418	2	ESF	5 min	M
PWS MCR Isolation Valve	PWS-PL-V420	3	ESF	5 min	M
PWS MCR Vacuum Relief	PWS-PL-V498	3	ESF	5 min	M
Core Makeup Tank A Discharge Isolation	PXS-PL-V014A	1	ESF	5 min	M * S
Limit Switch	PXS-PL-V014A-L	1	PAMS	1 yr (Note 9)	E * S
Solenoid Valve	PXS-PL-V014A-S1	1	ESF	5 min	E * S
Solenoid Valve	PXS-PL-V014A-S3	1	ESF	5 min	E * S
Core Makeup Tank B Discharge Isolation	PXS-PL-V014B	1	ESF	5 min	M * S
Limit Switch	PXS-PL-V014B-L	1	PAMS	1 yr (Note 9)	E * S
Solenoid Valve	PXS-PL-V014B-S1	1	ESF	5 min	E * S
Solenoid Valve	PXS-PL-V014B-S3	1	ESF	5 min	E * S
Core Makeup Tank A Discharge Isolation	PXS-PL-V015A	1	ESF	5 min	M * S
Limit Switch	PXS-PL-V015A-L	1	PAMS	1 yr (Note 9)	E * S
Solenoid Valve	PXS-PL-V015A-S1	1	ESF	5 min	E * S
Solenoid Valve	PXS-PL-V015A-S3	1	ESF	5 min	E * S
Core Makeup Tank B Discharge Isolation	PXS-PL-V015B	1	ESF	5 min	M * S
Limit Switch	PXS-PL-V015B-L	1	PAMS	1 yr (Note 9)	E * S
Solenoid Valve	PXS-PL-V015B-S1	1	ESF	5 min	E * S
Solenoid Valve	PXS-PL-V015B-S3	1	ESF	5 min	E * S
Core Makeup Tank A Discharge	PXS-PL-V016A	1	ESF	5 min	M * S
Core Makeup Tank B Discharge	PXS-PL-V016B	1	ESF	5 min	M * S
Core Makeup Tank A Discharge	PXS-PL-V017A	1	ESF	5 min	M * S
Core Makeup Tank B Discharge	PXS-PL-V017B	1	ESF	5 min	M * S
Accumulator A Pressure Relief	PXS-PL-V022A	1	ESF	5 min	M * S
Accumulator B Pressure Relief	PXS-PL-V022B	1	ESF	5 min	M * S
Accumulator A Discharge	PXS-PL-V028A	1	ESF	5 min	M * S
Accumulator B Discharge	PXS-PL-V028B	1	ESF	5 min	M * S
Accumulator A Discharge	PXS-PL-V029A	1	ESF	5 min	M * S
Accumulator B Discharge	PXS-PL-V029B	1	ESF	5 min	M * S

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Table 3.11-1 (Sheet 25 of 52)
Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Nitrogen Supply Outside Containment Isolation	PXS-PL-V042	2	ESF	5 min	M **
Limit Switch	PXS-PL-V042-L	2	PAMS	2 wks	E
Solenoid Valve	PXS-PL-V042-S	2	ESF	5 min	E
Nitrogen Supply Containment Isolation IRC Check Valve	PXS-PL-V043	1	ESF	24 hrs	M * S
PRHR HX Discharge Isolation	PXS-PL-V108A	1	ESF	5 min	M * S
Limit Switch	PXS-PL-V108A-L	1	PAMS	1 yr (Note 10)	E * S
Solenoid Valve	PXS-PL-V108A-S1	1	ESF	5 min	E * S
Solenoid Valve	PXS-PL-V108A-S3	1	ESF	5 min	E * S
PRHR HX Discharge Isolation	PXS-PL-V108B	1	ESF	5 min	M * S
Limit Switch	PXS-PL-V108B-L	1	PAMS	1 yr (Note 10)	E * S
Solenoid Valve	PXS-PL-V108B-S1	1	ESF	5 min	E * S
Solenoid Valve	PXS-PL-V108B-S3	1	ESF	5 min	E * S
Recirc Sump A Isolation	PXS-PL-V118A	1	ESF	72 hrs	M * S
Limit Switch	PXS-PL-V118A-L	1	PAMS	1 yr (Note 9)	E * S
Squib Operator	PXS-PL-V118A-T	1	ESF	72 hrs	E * S
Recirc Sump B Isolation	PXS-PL-V118B	1	ESF	72 hrs	M * S
Limit Switch	PXS-PL-V118B-L	1	PAMS	1 yr (Note 9)	E * S
Squib Operator	PXS-PL-V118B-T	1	ESF	72 hrs	E * S
Recirc Sump A	PXS-PL-V119A	1	ESF	1 yr	M * S
Recirc Sump B	PXS-PL-V119B	1	ESF	1 yr	M * S
Recirc Sump A	PXS-PL-V120A	1	ESF	72 hrs	M * S
Limit Switch	PXS-PL-V120A-L	1	PAMS	1 yr (Note 9)	E * S
Squib Operator	PXS-PL-V120A-T	1	ESF	72 hrs	E * S
Recirc Sump B	PXS- PL-V120B	1	ESF	72 hrs	M * S
Limit Switch	PXS-PL-V120B-L	1	PAMS	1 yr (Note 9)	E * S
Squib Operator	PXS-PL-V120B-T	1	ESF	72 hrs	E * S
IRWST Injection A	PXS-PL-V122A	1	ESF	1 yr	M * S
IRWST Injection B	PXS-PL-V122B	1	ESF	1 yr	M * S
IRWST Injection A	PXS-PL-V123A	1	ESF	72 hrs	M * S
Limit Switch	PXS-PL-V123A-L	1	PAMS	1 yr (Note 9)	E * S
Squib Operator	PXS-PL-V123A-T	1	ESF	72 hrs	E * S
IRWST Injection B	PXS-PL-V123B	1	ESF	72 hrs	M * S
Limit Switch	PXS-PL-V123B-L	1	PAMS	1 yr (Note 9)	E * S
Squib Operator	PXS-PL-V123B-T	1	ESF	72 hrs	E * S
IRWST Injection A	PXS-PL-V124A	1	ESF	1 yr	M * S
IRWST Injection B	PXS-PL-V124B	1	ESF	1 yr	M * S
IRWST Injection A	PXS-PL-V125A	1	ESF	72 hrs	M * S
Limit Switch	PXS-PL-V125A-L	1	PAMS	1 yr (Note 9)	E * S
Squib Operator	PXS-PL-V125A-T	1	ESF	72 hrs	E * S
IRWST Injection B	PXS-PL-V125B	1	ESF	72 hrs	M * S
Limit Switch	PXS-PL-V125B-L	1	PAMS	1 yr (Note 9)	E * S
Squib Operator	PXS-PL-V125B-T	1	ESF	72 hrs	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
IRWST Gutter Drain Isolation A	PXS-PL-V130A	1	ESF	5 min	M * S
Limit Switch	PXS-PL-V130A-L	1	PAMS	1 yr	E * S
Solenoid Valve	PXS-PL-V130A-S1	1	ESF	5 min	E * S
IRWST Gutter Drain Isolation B	PXS-PL-V130B	1	ESF	5 min	M * S
Limit Switch	PXS-PL-V130B-L	1	PAMS	1 yr	E * S
Solenoid Valve	PXS-PL-V130B-S1	1	ESF	5 min	E * S
First Stage ADS	RCS-PL-V001A	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V001A-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V001A-M	1	ESF	24 hrs	E * S
First Stage ADS	RCS-PL-V001B	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V001B-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V001B-M	1	ESF	24 hrs	E * S
Second Stage ADS	RCS-PL-V002A	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V002A-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V002A-M	1	ESF	24 hrs	E * S
Second Stage ADS	RCS-PL-V002B	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V002B-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V002B-M	1	ESF	24 hrs	E * S
Third Stage ADS	RCS-PL-V003A	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V003A-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V003A-M	1	ESF	24 hrs	E * S
Third Stage ADS	RCS-PL-V003B	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V003B-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V003B-M	1	ESF	24 hrs	E * S
Fourth Stage ADS	RCS-PL-V004A	1	ESF	72 hrs	M * S
Limit Switch	RCS-PL-V004A-L	1	PAMS	1 yr	E * S
Squib Operator	RCS-PL-V004A-T	1	ESF	72 hrs	E * S
Fourth Stage ADS	RCS-PL-V004B	1	ESF	72 hrs	M * S
Limit Switch	RCS-PL-V004B-L	1	PAMS	1 yr	E * S
Squib Operator	RCS-PL-V004B-T	1	ESF	72 hrs	E * S
Fourth Stage ADS	RCS-PL-V004C	1	ESF	72 hrs	M * S
Limit Switch	RCS-PL-V004C-L	1	PAMS	1 yr	E * S
Squib Operator	RCS-PL-V004C-T	1	ESF	72 hrs	E * S
Fourth Stage ADS	RCS-PL-V004D	1	ESF	72 hrs	M * S
Limit Switch	RCS-PL-V004D-L	1	PAMS	1 yr	E * S
Squib Operator	RCS-PL-V004D-T	1	ESF	72 hrs	E * S
PZR Safety Valve	RCS-PL-V005A	1	ESF	5 min	M * S
Limit Switch	RCS-PL-V005A-L	1	PAMS	1 yr	E * S
PZR Safety Valve	RCS-PL-V005B	1	ESF	5 min	M * S
Limit Switch	RCS-PL-V005B-L	1	PAMS	1 yr	E * S
ADS Discharge Header A Relief	RCS-PL-V010A	1	ESF	24 hrs	M * S
ADS Discharge Header B Relief	RCS-PL-V010B	1	ESF	24 hrs	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
First Stage ADS Isolation	RCS-PL-V011A	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V011A-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V011A-M	1	ESF	24 hrs	E * S
First Stage ADS Isolation	RCS-PL-V011B	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V011B-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V011B-M	1	ESF	24 hrs	E * S
Second Stage ADS Isolation	RCS-PL-V012A	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V012A-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V012A-M	1	ESF	24 hrs	E * S
Second Stage ADS Isolation	RCS-PL-V012B	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V012B-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V012B-M	1	ESF	24 hrs	E * S
Third Stage ADS Isolation	RCS-PL-V013A	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V013A-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V013A-M	1	ESF	24 hrs	E * S
Third Stage ADS Isolation	RCS-PL-V013B	1	ESF	24 hrs	M * S
Limit Switch	RCS-PL-V013B-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V013B-M	1	ESF	24 hrs	E * S
Reactor Vessel Head Vent	RCS-PL-V150A	1	ESF	1 yr	M * S
Limit Switch	RCS-PL-V150A-L	1	PAMS	1 yr	E * S
Solenoid Operator	RCS-PL-V150A-S	1	ESF	1 yr	E * S
Voltage Control Box	RCS-DJ-01	2	ESF	1 yr	E
Reactor Vessel Head Vent	RCS-PL-V150B	1	ESF	1 yr	M * S
Limit Switch	RCS-PL-V150B-L	1	PAMS	1 yr	E * S
Solenoid Operator	RCS-PL-V150B-S	1	ESF	1 yr	E * S
Voltage Control Box	RCS-DJ-02	2	ESF	1 yr	E
Reactor Vessel Head Vent	RCS-PL-V150C	1	ESF	1 yr	M * S
Limit Switch	RCS-PL-V150C-L	1	PAMS	1 yr	E * S
Solenoid Operator	RCS-PL-V150C-S	1	ESF	1 yr	E * S
Voltage Control Box	RCS-DJ-03	2	ESF	1 yr	E
Reactor Vessel Head Vent	RCS-PL-V150D	1	ESF	1 yr	M * S
Limit Switch	RCS-PL-V150D-L	1	PAMS	1 yr	E * S
Solenoid Operator	RCS-PL-V150D-S	1	ESF	1 yr	E * S
Voltage Control Box	RCS-DJ-04	2	ESF	1 yr	E
RCS Inner Suction Isolation	RNS-PL-V001A	1	ESF	5 min	M * S
Limit Switch	RNS-PL-V001A-L	1	PAMS	1 yr (Note 9)	E * S
Motor Operator	RNS-PL-V001A-M	1	ESF	5 min	E * S
RCS Inner Suction Isolation	RNS-PL-V001B	1	ESF	5 min	M * S
Limit Switch	RNS-PL-V001B-L	1	PAMS	1 yr (Note 9)	E * S
Motor Operator	RNS-PL-V001B-M	1	ESF	5 min	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
RCS Outer Suction Isolation	RNS-PL-V002A	1	ESF	5 min	M * S
Limit Switch	RNS-PL-V002A-L	1	PAMS	1 yr (Note 9)	E * S
Motor Operator	RNS-PL-V002A-M	1	ESF	5 min	E * S
RCS Outer Suction Isolation	RNS-PL-V002B	1	ESF	5 min	M * S
Limit Switch	RNS-PL-V002B-L	1	PAMS	1 yr (Note 9)	E * S
Motor Operator	RNS-PL-V002B-M	1	ESF	5 min	E * S
RCS Thermal Relief	RNS-PL-V003A	1	ESF	24 hrs	M * S
RCS Thermal Relief	RNS-PL-V003B	1	ESF	24 hrs	M * S
RHR Control/Isolation Valve	RNS-PL-V011	6	ESF	5 min	M **
Limit Switch	RNS-PL-V011-L	6	PAMS	2 wks	E **
Motor Operator	RNS-PL-V011-M	6	ESF	5 min	E **
RNS Discharge Containment Isolation Valve Test Connection	RNS-PL-V012	6	ESF	1 yr	M **
RNS Discharge Containment Isolation	RNS-PL-V013	1	ESF	24 hrs	M * S
RNS Discharge RCS Pressure Boundary	RNS-PL-V015A	1	ESF	5 min	M * S
RNS Discharge RCS Pressure Boundary	RNS-PL-V015B	1	ESF	5 min	M * S
RNS Discharge RCS Pressure Boundary	RNS-PL-V017A	1	ESF	5 min	M * S
RNS Discharge RCS Pressure Boundary	RNS-PL-V017B	1	ESF	5 min	M * S
RNS Hot Leg Suction Relief	RNS-PL-V020	1	ESF	24 hrs	M * S
RNS Hot Leg Suction Relief	RNS-PL-V021	1	ESF	24 hrs	M * S
RNS Suction Header Containment Isolation	RNS-PL-V022	6	ESF	5 min	M **
Limit Switch	RNS-PL-V022-L	6	PAMS	2 wks	E **
Motor Operator	RNS-PL-V022-M	6	ESF	5 min	E **
IRWST Suction Line Isolation	RNS-PL-V023	1	ESF	5 min	M * S
Limit Switch	RNS-PL-V023-L	1	PAMS	1 yr (Note 9)	E * S
Motor Operator	RNS-PL-V023-M	1	ESF	5 min	E * S
RNS – CVS Containment Isolation	RNS-PL-V061	1	ESF	5 min	M * S
Limit Switch	RNS-PL-V061-L	1	PAMS	1 yr (Note 9)	E * S
Air Operator	RNS-PL-V061-S	1	ESF	5 min	E * S
SDS MCR Vent Isolation Valve	SDS-PL-V001	3	ESF	24 hrs	M
Limit Switch	SDS-PL-V001-L	3	PAMS	2 wks	E
Motor Operator	SDS-PL-V001-M	3	ESF	5 min	E
SDS MCR Vent Isolation Valve	SDS-PL-V002	3	ESF	24 hrs	M
Limit Switch	SDS-PL-V002-L	3	PAMS	2 wks	E
Motor Operator	SDS-PL-V002-M	3	ESF	5 min	E
SFS Suction Line Containment Isolation	SFS-PL-V034	1	ESF	5 min	M * S
Limit Switch	SFS-PL-V034-L	1	PAMS	1 yr (Note 9)	E * S
Motor Operator	SFS-PL-V034-M	1	ESF	5 min	E * S
SFS Suction Line Containment Isolation	SFS-PL-V035	6	ESF	5 min	M **
Limit Switch	SFS-PL-V035-L	6	PAMS	2 wks	E **
Motor Operator	SFS- PL-V035-M	6	ESF	5 min	E **
SFS Discharge Containment Isolation	SFS-PL-V037	1	ESF	24 hrs	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
SFS Discharge Line Containment Isolation	SFS-PL-V038	6	ESF	5 min	M **
Limit Switch	SFS-PL-V038-L	6	PAMS	2 wks	E **
Motor Operator	SFS-PL-V038-M	6	ESF	5 min	E **
SFS Cask Loading Pit to SFS Pump	SFS-PL-V042	6	ESF	2 wks	M **
SFS Pump to Cask Loading Pit	SFS-PL-V045	6	ESF	2 wks	M **
Cask Loading Pit to WLS	SFS-PL-V049	6	ESF	2 wks	M **
Spent Fuel Pool to Cask Washdown Pit Isolation	SFS-PL-V066	6	ESF	2 wks	M **
SFS Containment Isolation Relief	SFS-PL-V067	1	ESF	24 hrs	M * S
Cask Washdown Pit Drain Isolation	SFS-PL-V068	6	ESF	2 wks	M **
Refueling Cavity to SG Compartment	SFS-PL-V071	1	ESF	2 wks	M * S
Refueling Cavity to SG Compartment	SFS-PL-V072	1	ESF	2 wks	M * S
PORV Block Valve	SGS-PL-V027A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V027A-L	5	PAMS	2 wks	E * S
Motor Operator	SGS-PL-V027A-M	5	ESF	5 min	E * S
PORV Block Valve	SGS-PL-V027B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V027B-L	5	PAMS	2 wks	E * S
Motor Operator	SGS-PL-V027B-M	5	ESF	5 min	E * S
Main Steam Safety Valve SG01	SGS-PL-V030A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V030A-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG02	SGS-PL-V030B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V030B-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG01	SGS-PL-V031A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V031A-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG02	SGS-PL-V031B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V031B-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG01	SGS-PL-V032A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V32A-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG02	SGS-PL-V032B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V032B-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG01	SGS-PL-V033A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V033A-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG02	SGS-PL-V033B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V033B-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG01	SGS-PL-V034A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V034A-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG02	SGS-PL-V034B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V034B-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG01	SGS-PL-V035A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V035A-L	5	PAMS	2 wks	E * S
Main Steam Safety Valve SG02	SGS-PL-V035B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V035B-L	5	PAMS	2 wks	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Steam Line Condensate					
Drain Isolation	SGS-PL-V036A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V036A-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V036A-S	5	ESF	5 min	E * S
Steam Line Condensate Drain Isolation	SGS-PL-V036B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V036B-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V036B-S	5	ESF	5 min	E * S
Main Steam Line Isolation	SGS-PL-V040A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V040A-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V040A-S1	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V040A-S2	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V040A-S3	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V040A-S4	5	ESF	5 min	E * S
Main Steam Line Isolation	SGS-PL-V040B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V040B-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V040B-S1	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V040B-S2	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V040B-S3	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V040B-S4	5	ESF	5 min	E * S
Main Feedwater Isolation	SGS-PL-V057A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V057A-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V057A-S1	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V057A-S2	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V057A-S3	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V057A-S4	5	ESF	5 min	E * S
Main Feedwater Isolation	SGS-PL-V057B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V057B-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V057B-S1	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V057B-S2	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V057B-S3	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V057B-S4	5	ESF	5 min	E * S
Startup Feedwater Isolation	SGS-PL-V067A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V067A-L	5	PAMS	2 wks	E * S
Motor Operator	SGS-PL-V067A-M	5	ESF	5 min	E * S
Startup Feedwater Isolation	SGS-PL-V067B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V067B-L	5	PAMS	2 wks	E * S
Motor Operator	SGS-PL-V067B-M	5	ESF	5 min	E * S
SG Blowdown Isolation	SGS-PL-V074A	10	ESF	5 min	M * S
Limit Switch	SGS-PL-V074A-L	10	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V074A-S	10	ESF	5 min	E * S
SG Blowdown Isolation	SGS-PL-V074B	10	ESF	5 min	M * S
Limit Switch	SGS-PL-V074B-L	10	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V074B-S	10	ESF	5 min	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
SG Series Blowdown Isolation	SGS-PL-V075A	10	ESF	5 min	M * S
Limit Switch	SGS-PL-V075A-L	10	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V075A-S	10	ESF	5 min	E * S
SG Series Blowdown Isolation	SGS-PL-V075B	10	ESF	5 min	M * S
Limit Switch	SGS-PL-V075B-L	10	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V075B-S	10	ESF	5 min	E * S
Steam Line Condensate Drain	SGS-PL-V086A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V086A-L	5	PAMS	2 wks	E * S
Isolation Solenoid Valve	SGS-PL-V086A-S	5	ESF	5 min	E * S
Steam Line Condensate Drain	SGS-PL-V086B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V086B-L	5	PAMS	2 wks	E * S
Isolation Solenoid Valve	SGS-PL-V086B-S	5	ESF	5 min	E * S
Power Operated Relief Valve	SGS-PL-V233A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V233A-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V233A-S	5	ESF	5 min	E * S
Power Operated Relief Valve	SGS-PL-V233B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V233B-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V233B-S	5	ESF	5 min	E * S
MSIV Bypass Isolation Valve	SGS-PL-V240A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V240A-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V240A-S1	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V240A-S2	5	ESF	5	E * S
MSIV Bypass Isolation Valve	SGS-PL-V240B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V240B-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V240B-S1	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V240B-S2	5	ESF	5 min	E * S
Main Feedwater Control Valve	SGS-PL-V250A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V250A-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V250A-S1	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V250A-S2	5	ESF	5 min	E * S
Main Feedwater Control Valve	SGS-PL-V250B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V250B-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V250B-S1	5	ESF	5 min	E * S
Solenoid Valve	SGS-PL-V250B-S2	5	ESF	5 min	E * S
Startup Feedwater Control Valve	SGS-PL-V255A	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V255A-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V255A-S	5	ESF	5 min	E * S
Startup Feedwater Control Valve	SGS-PL-V255B	5	ESF	5 min	M * S
Limit Switch	SGS-PL-V255B-L	5	PAMS	2 wks	E * S
Solenoid Valve	SGS-PL-V255B-S	5	ESF	5 min	E * S
Main Feedwater Thermal Relief Valve	SGS-PL-V257A	5	ESF	24 hrs	M * S
Main Feedwater Thermal Relief Valve	SGS-PL-V257B	5	ESF	24 hrs	M * S
Startup Feedwater Thermal Relief Valve	SGS-PL-V258A	5	ESF	24 hrs	M * S
Startup Feedwater Thermal Relief Valve	SGS-PL-V258B	5	ESF	24 hrs	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
MCR Isolation Valve	VBS-PL-V186	3	ESF	24 hrs	M
Limit Switch	VBS-PL-V186-L	3	PAMS	2 wks	E
Motor Operator	VBS-PL-V186-M	3	ESF	24 hrs	E
MCR Isolation Valve	VBS-PL-V187	3	ESF	24 hrs	M
Limit Switch	VBS-PL-V187-L	3	PAMS	2 wks	E
Motor Operator	VBS-PL-V187-M	3	ESF	24 hrs	E
MCR Isolation Valve	VBS-PL-V188	3	ESF	24 hrs	M
Limit Switch	VBS-PL-V188-L	3	PAMS	2 wks	E
Motor Operator	VBS-PL-V188-M	3	ESF	24 hrs	E
MCR Isolation Valve	VBS-PL-V189	3	ESF	24 hrs	M
Limit Switch	VBS-PL-V189-L	3	PAMS	2 wks	E
Motor Operator	VBS-PL-V189-M	3	ESF	24 hrs	E
MCR Isolation Valve	VBS-PL-V190	3	ESF	24 hrs	M
Limit Switch	VBS-PL-V190-L	3	PAMS	2 wks	E
Motor Operator	VBS-PL-V190-M	3	ESF	24 hrs	E
MCR Isolation Valve	VBS-PL-V191	3	ESF	24 hrs	M
Limit Switch	VBS-PL-V191-L	3	PAMS	2 wks	E
Motor Operator	VBS-PL-V191-M	3	ESF	24 hrs	E
Air Delivery Isolation Valve	VES-PL-V001	3	ESF	2 wks	M
Pressure Regulator Valve A	VES-PL-V002A	7	ESF	2 wks	M **
Pressure Regulator Valve B	VES-PL-V002B	7	ESF	2 wks	M **
Air Delivery Isolation Valve A	VES-PL-V005A	3	ESF	2 wks	M
Limit Switch	VES-PL-V005A-L	3	PAMS	2 wks	E
Solenoid Operator	VES-PL-V005A-S	3	ESF	2 wks	E
Voltage Control Box	VES-DJ-01	2	ESF	2 wks	E
Air Delivery Isolation Valve B	VES-PL-V005B	3	ESF	2 wks	M
Limit Switch	VES-PL-V005B-L	3	PAMS	2 wks	E
Solenoid Operator	VES-PL-V005B-S	3	ESF	2 wks	E
Voltage Control Box	VES-DJ-02	2	ESF	2 wks	E
Temporary Instrument Isolation Valve A	VES-PL-V018	7	ESF	2 wks	M **
Temporary Instrument Isolation Valve B	VES-PL-V019	7	ESF	2 wks	M **
Relief Isolation Valve A	VES-PL-V022A	3	ESF	2 wks	M
Limit Switch	VES-PL-V022A-L	3	PAMS	2 wks	E
Solenoid Valve	VES-PL-V022A-S	3	ESF	2 wks	E
Relief Isolation Valve B	VES-PL-V022B	3	ESF	2 wks	M
Limit Switch	VES-PL-V022B-L	3	PAMS	2 wks	E
Solenoid Valve	VES-PL-V022B-S	3	ESF	2 wks	E
Air Tank Safety Relief Valve A	VES-PL-V040A	7	ESF	2 wks	M **
Air Tank Safety Relief Valve B	VES-PL-V040B	7	ESF	2 wks	M **
Air Tank Safety Relief Valve C	VES-PL-V040C	7	ESF	2 wks	M **
Air Tank Safety Relief Valve D	VES-PL-V040D	7	ESF	2 wks	M **
Main Air Flow Path Isolation Valve	VES-PL-V044	3	ESF	2 wks	M
Eductor Flow Path Isolation Valve	VES-PL-V045	3	ESF	2 wks	M
Eductor Bypass Isolation Valve	VES-PL-V046	3	ESF	2 wks	M

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Containment Purge Inlet Isolation	VFS-PL-V003	7	ESF	5 min	M
Limit Switch	VFS-PL-V003-L	7	PAMS	2 wks	E
Solenoid Valve	VFS-PL-V003-S1	7	ESF	5 min	E
Containment Purge Inlet Isolation	VFS-PL-V004	1	ESF	5 min	M * S
Limit Switch	VFS-PL-V004-L	1	PAMS	1 yr	E * S
Solenoid Valve	VFS-PL-V004-S1	1	ESF	5 min	E * S
Containment Purge Discharge Isolation	VFS-PL-V009	1	ESF	5 min	M * S
Limit Switch	VFS-PL-V009-L	1	PAMS	1 yr	E * S
Solenoid Valve	VFS-PL-V009-S1	1	ESF	5 min	E * S
Containment Purge Discharge Isolation	VFS-PL-V010	6	ESF	5 min	M **
Limit Switch	VFS-PL-V010-L	6	PAMS	2 wks	E **
Solenoid Valve	VFS-PL-V010-S1	6	ESF	5 min	E **
Vacuum Relief Containment Isolation Valve A - ORC	VFS-PL-V800A	7	ESF	24 hrs	M **
Limit Switch	VFS-PL-V800A-L	7	PAMS	1 yr	E **
Motor Operator	VFS-PL-V800A-M	7	ESF	24 hrs	E
Vacuum Relief Containment Isolation Valve B - ORC	VFS-PL-V800B	7	ESF	24 hrs	M **
Limit Switch	VFS-PL-V800B-L	7	PAMS	1 yr	E **
Motor Operator	VFS-PL-V800B-M	7	ESF	24 hrs	E
Vacuum Relief Containment Isolation Check Valve A - IRC	VFS-PL-V803A	1	ESF	24 hrs	M* S
Vacuum Relief Containment Isolation Check Valve B - IRC	VFS-PL-V803B	1	ESF	24 hrs	M* S
VWS Containment Penetration Thermal Relief Valve	VWS-PL-V053	1	ESF	24 hrs	M* S
VWS Containment Penetration Thermal Relief Valve	VWS-PL-V057	1	ESF	24 hrs	M* S
Fan Cooler Supply Isolation	VWS-PL-V058	2	ESF	5 min	M **
Limit Switch	VWS-PL-V058-L	2	PAMS	2 wks	E
Solenoid Valve	VWS-PL-V058-S	2	ESF	5 min	E
Fan Coolers Supply Containment Isolation Check Valve	VWS-PL-V062	1	ESF	24 hrs	M * S
VWS Containment Isolation Relief	VWS-PL-V080	1	ESF	24 hrs	M * S
Fan Cooler Return Isolation	VWS-PL-V082	1	ESF	5 min	M * S
Limit Switch	VWS-PL-V082-L	1	PAMS	1 yr	E * S
Solenoid Valve	VWS-PL-V082-S	1	ESF	5 min	E * S
Fan Cooler Return Isolation	VWS-PL-V086	2	ESF	5 min	M **
Limit Switch	VWS-PL-V086-L	2	PAMS	2 wks	E
Solenoid Valve	VWS-PL-V086-S	2	ESF	5 min	E
Sump Containment Isolation IRC	WLS-PL-V055	1	ESF	5 min	M * S
Limit Switch	WLS-PL-V055-L	1	PAMS	1 yr	E * S
Solenoid Valve	WLS-PL-V055-S1	1	ESF	5 min	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Sump Containment Isolation ORC	WLS-PL-V057	7	ESF	5 min	M ** S
Limit Switch	WLS-PL-V057-L	7	PAMS	2 wks	E **
Solenoid Valve	WLS-PL-V057-S1	7	ESF	5 min	E ** S
WLS Containment Isolation Relief	WLS-PL-V058	1	ESF	24 hrs	M * S
RCDT Gas Containment Isolation	WLS-PL-V067	1	ESF	5 min	M * S
Limit Switch	WLS-PL-V067-L	1	PAMS	1 yr (Note 9)	E * S
Solenoid Valve	WLS-PL-V067-S	1	ESF	5 min	E * S
RCDT Gas Containment Isolation	WLS-PL-V068	7	ESF	5 min	M ** S
Limit Switch	WLS-PL-V068-L	7	PAMS	2 wks	E **
Solenoid Valve	WLS-PL-V068-S	7	ESF	5 min	E ** S
CVS To Sump	WLS-PL-V071 A	1	ESF	2 wks	M * S
PXS A To Sump	WLS-PL-V071 B	1	ESF	2 wks	M * S
PXS B To Sump	WLS-PL-V071 C	1	ESF	2 wks	M * S
CVS To Sump	WLS-PL-V072 A	1	ESF	2 wks	M * S
PXS A To Sump	WLS-PL-V072 B	1	ESF	2 wks	M * S
PXS B To Sump	WLS-PL-V072 C	1	ESF	2 wks	M * S
MISCELLANEOUS					
Non-Active Valves					
Containment Penetration Test Connection Isolation	CAS-PL-V027	1	PB	1 yr	M* S
Service Air Supply Outside Containment Isolation	CAS-PL-V204	2	PB	1 yr	M **
Containment Penetration Test Connection Isolation	CAS-PL-V219	1	PB	1 yr	M * S
Containment Isolation Valve Test Connection – Outlet Line	CCS-PL-V209	1	PB	1 yr	M * S
CCS Supply Containment Isolation – IRC	CCS-PL-V214	1	PB	1 yr	M* S
CCS Supply Containment Isolation Valve Test Connection – IRC	CCS-PL-V215	1	PB	1 yr	M* S
Containment Leak Test Outlet Line – IRC	CCS-PL-V216	1	PB	1 yr	M* S
Containment Isolation Valve V207 Body Test Connection Valve	CCS-PL-V217	1	PB	1 yr	M* S
Containment Isolation Valve Test Connection – Inlet Line	CCS-PL-V257	2	PB	1 yr	M **
Resin Flush IRC Isolation	CVS-PL-V040	1	PB	1 yr	M * S
Resin Flush ORC Isolation	CVS-PL-V041	7	PB	1 yr	M ** S
Letdown PZR Instrument Root	CVS-PL-V046	7	PB	1 yr	M ** S
Zinc Addition – IRC Shutoff	CVS-PL-V065	1	PB	1 yr	M * S
Zinc Add Cont Isolation Test Connection	CVS-PL-V095	1	PB	1 yr	M * S
Zinc Addition Containment Isolation Test Connection	CVS-PL-V096	10	PB	1 yr	M * S
Hydrogen Injection – IRC Shutoff	CVS-PL-V215	1	PB	1 yr	M * S
Hydrogen Addition Containment Isolation Test Connection	CVS-PL-V216	1	PB	1 yr	M * S
Hydrogen Addition Containment Isolation Test Connection	CVS-PL-V218	1	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Demin Water Supply Containment Isolation – Outside	DWS-PL-V244	10	PB	1 yr	M * S
Containment Penetration Test Connection Isolation	DWS-PL-V248	10	PB	1 yr	M * S
Fire Water Containment Supply Isolation	FPS-PL-V050	10	PB	1 yr	M * S
Fire Water Containment Test Connection Isolation	FPS-PL-V051	10	PB	1 yr	M * S
Flow Transmitter FT001 Root Valve	PCS-PL-V010A	9	PB	1 yr	M **
Flow Transmitter FT001 Root Valve	PCS-PL-V010B	9	PB	1 yr	M **
Flow Transmitter FT002 Root Valve	PCS-PL-V011A	9	PB	1 yr	M **
Flow Transmitter FT002 Root Valve	PCS-PL-V011B	9	PB	1 yr	M **
Flow Transmitter FT003 Root Valve	PCS-PL-V012A	9	PB	1 yr	M **
Flow Transmitter FT003 Root Valve	PCS-PL-V012B	9	PB	1 yr	M **
Flow Transmitter FT004 Root Valve	PCS-PL-V013A	9	PB	1 yr	M **
Flow Transmitter FT004 Root Valve	PCS-PL-V013B	9	PB	1 yr	M **
PCCWST Drain Isolation Valve	PCS-PL-V016	9	PB	1 yr	M **
Makeup to Distr. Bucket Isolation Valve	PCS-PL-V026	6	ESF	2 wks	M **
PCCWST Isolation Valve Leakage Detection Drain	PCS-PL-V029	9	PB	1 yr	M **
PCCWST Isolation Valve Leakage Detection Crossconn	PCS-PL-V030	9	PB	1 yr	M **
PCCWST Level Instrument Root Valve	PCS-PL-V031A	9	PB	1 yr	M **
PCCWST Level Instrument Root Valve	PCS-PL-V031B	9	PB	1 yr	M **
Recirculation Pump Suction from Long Term Makeup Isolation Valve	PCS-PL-V033	10	ESF	2 wks	M * S
Spent Fuel Pool Emergency Makeup Isolation	PCS-PL-V052	7	PB	1 yr	M
Shutoff Valve for Leakage Sensor	PCS-PL-V060A	9	PB	1 yr	M **
Shutoff Valve for Leakage Sensor	PCS-PL-V060B	9	PB	1 yr	M **
Recirculation Header Discharge to SFS Pool Vent Isolation Valve	PCS-PL-V303	10	PB	1 yr	M * S
Recirculation Header Discharge to SFS Pool Drain Isolation Valve	PCS-PL-V304	7	PB	1 yr	M *
PCCWST Recirculation Return Drain Isolation Valve	PCS-PL-V305	10	PB	1 yr	M * S
Hot Leg 1 Sample Isolation	PSS-PL-V001A	1	PB	1 yr	M * S
Limit Switch	PSS-PL-V001A-L	1	PAMS	1 yr	E * S
Solenoid Valve	PSS-PL-V001A-S	1	ESF	24 hrs	E * S
Voltage Control Box	PSS-DJ-01	2	ESF	24 hrs	E

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Hot Leg 2 Sample Isolation	PSS-PL-V001B	1	PB	1 yr	M * S
Limit Switch	PSS-PL-V001B-L	1	PAMS	1 yr	E * S
Solenoid Valve	PSS-PL-V001B-S	1	ESF	24 hrs	E * S
Voltage Control Box	PSS-DJ-02	2	ESF	24 hrs	E
Pressurizer Sample Isolation	PSS-PL-V003	1	PB	1 yr	M * S
PXS Accumulator Sample Isolation	PSS-PL-V004A	1	PB	1 yr	M * S
PXS Accumulator Sample Isolation	PSS-PL-V004B	1	PB	1 yr	M * S
PXS CMT A Sample Isolation	PSS-PL-V005A	1	PB	1 yr	M * S
PXS CMT B Sample Isolation	PSS-PL-V005B	1	PB	1 yr	M * S
PXS CMT A Sample Isolation	PSS-PL-V005C	1	PB	1 yr	M * S
PXS CMT B Sample Isolation	PSS-PL-V005D	1	PB	1 yr	M * S
Liquid Sample Isolation Valve	PSS-PL-V012A	1	PB	1 yr	M * S
Liquid Sample Check Valve	PSS-PL-V012B	1	PB	1 yr	M * S
RCS Pressurizer Sample Isolation Valve	PSS-PL-V013	1	PB	1 yr	M * S
RCS Hot Leg 1 Sample Isolation Valve	PSS-PL-V014A	1	PB	1 yr	M * S
RCS Hot Leg 2 Sample Isolation Valve	PSS-PL-V014B	1	PB	1 yr	M * S
PXS Accumulator Sample Isolation Valve	PSS-PL-V015A	1	PB	1 yr	M * S
PXS Accumulator Sample Isolation Valve	PSS-PL-V015B	1	PB	1 yr	M * S
PXS CMT A Sample Isolation Valve	PSS-PL-V016A	1	PB	1 yr	M * S
PXS CMT B Sample Isolation Valve	PSS-PL-V016B	1	PB	1 yr	M * S
PXS CMT A Sample Isolation Valve	PSS-PL-V016C	1	PB	1 yr	M * S
PXS CMT B Sample Isolation Valve	PSS-PL-V016D	1	PB	1 yr	M * S
Containment Testing Boundary Isolation Valve	PSS-PL-V076A	1	PB	1 yr	M * S
Containment Testing Boundary Isolation Valve	PSS-PL-V076B	1	PB	1 yr	M * S
Containment Isolation Test Connection Isolation Valve	PSS-PL-V082	1	PB	1 yr	M * S
Containment Isolation Test Connection Isolation Valve	PSS-PL-V083	1	PB	1 yr	M * S
Containment Isolation Test Connection Isolation Valve	PSS-PL-V085	1	PB	1 yr	M * S
Containment Isolation Test Connection Isolation Valve	PSS-PL-V086	1	PB	1 yr	M * S
PXS CMT A Sample Isolation Valve	PSS-PL-V902A	1	PB	1 yr	M * S
PXS CMT B Sample Isolation Valve	PSS-PL-V902B	1	PB	1 yr	M * S
PXS CMT B Sample Isolation Valve	PSS-PL-V903	1	PB	1 yr	M * S
Core Makeup Tank A CL Inlet Isolation	PXS-PL-V002A	1	PB	1 yr	M * S
Limit Switch	PXS-PL-V002A-L	1	PAMS	1 yr	E * S
Motor Operator	PXS-PL-V002A-M	1	ESF	5 min	E * S
Core Makeup Tank B CL Inlet Isolation	PXS-PL-V002B	1	PB	1 yr	M * S
Limit Switch	PXS-PL-V002B-L	1	PAMS	1 yr	E * S
Motor Operator	PXS-PL-V002B-M	1	ESF	5 min	E * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Core Makeup Tank A Upper Sample	PXS-PL-V010A	1	PB	1 yr	M * S
Core Makeup Tank B Upper Sample	PXS-PL-V010B	1	PB	1 yr	M * S
Core Makeup Tank A Lower Sample	PXS-PL-V011A	1	PB	1 yr	M * S
Core Makeup Tank B Lower Sample	PXS-PL-V011B	1	PB	1 yr	M * S
Core Makeup Tank A Drain	PXS-PL-V012A	1	PB	1 yr	M * S
Core Makeup Tank B Drain	PXS-PL-V012B	1	PB	1 yr	M * S
Core Makeup Tank A Discharge Manual Isolation	PXS-PL-V013A	1	PB	1 yr	M * S
Core Makeup Tank B Discharge Manual Isolation	PXS-PL-V013B	1	PB	1 yr	M * S
RNS to CMT Injection Line A Drain	PXS-PL-V019A	1	PB	1 yr	M * S
RNS to CMT Injection Line B Drain	PXS-PL-V019B	1	PB	1 yr	M * S
IRWST Injection Line A Drain	PXS-PL-V020A	1	PB	1 yr	M * S
IRWST Injection Line B Drain	PXS-PL-V020B	1	PB	1 yr	M * S
Accumulator A N_2 Vent	PXS-PL-V021A	1	PB	1 yr	M * S
Accumulator B N_2 Vent	PXS-PL-V021B	1	PB	1 yr	M * S
Accumulator A PZR Transmitter Isolation	PXS-PL-V023A	1	PB	1 yr	M * S
Accumulator B PZR Transmitter Isolation	PXS-PL-V023B	1	PB	1 yr	M * S
Accumulator A PZR Transmitter Isolation	PXS-PL-V024A	1	PB	1 yr	M * S
Accumulator B PZR Transmitter Isolation	PXS-PL-V024B	1	PB	1 yr	M * S
Accumulator A Sample	PXS-PL-V025A	1	PB	1 yr	M * S
Accumulator B Sample	PXS-PL-V025B	1	PB	1 yr	M * S
Accumulator A Drain	PXS-PL-V026A	1	PB	1 yr	M * S
Accumulator B Drain	PXS-PL-V026B	1	PB	1 yr	M * S
Accumulator A Discharge Isolation	PXS-PL-V027A	1	PB	1 yr	M * S
Accumulator B Discharge Isolation	PXS-PL-V027B	1	PB	1 yr	M * S
Core Makeup Tank A Highpoint Vent	PXS-PL-V030A	1	PB	1 yr	M * S
Core Makeup Tank B Highpoint Vent	PXS-PL-V030B	1	PB	1 yr	M * S
Core Makeup Tank A Highpoint Vent	PXS-PL-V031A	1	PB	1 yr	M * S
Core Makeup Tank B Highpoint Vent	PXS-PL-V031B	1	PB	1 yr	M * S
Accumulator A Check Valve Drain	PXS-PL-V033A	1	PB	1 yr	M * S
Accumulator B Check Valve Drain	PXS-PL-V033B	1	PB	1 yr	M * S
Accumulator N_2 Containment Penetration Test Connection	PXS-PL-V052	1	PB	1 yr	M * S
CMT A Wide Level Upper Root	PXS-PL-V080A	1	PB	1 yr	M * S
CMT B Wide Level Upper Root	PXS-PL-V080B	1	PB	1 yr	M * S
CMT A Wide Level Lower Root	PXS-PL-V081A	1	PB	1 yr	M * S
CMT B Wide Level Lower Root	PXS-PL-V081B	1	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
CMT A Upper Level A Isolation 1	PXS-PL-V082A	1	PB	1 yr	M * S
CMT B Upper Level A Isolation 1	PXS-PL-V082B	1	PB	1 yr	M * S
CMT A Upper Level A Isolation 2	PXS-PL-V083A	1	PB	1 yr	M * S
CMT B Upper Level A Isolation 2	PXS-PL-V083B	1	PB	1 yr	M * S
CMT A Upper Level A Vent	PXS-PL-V084A	1	PB	1 yr	M * S
CMT B Upper Level A Vent	PXS-PL-V084B	1	PB	1 yr	M * S
LT-011 A/C Drain Valve	PXS-PL-V085A	1	PB	1 yr	M * S
LT-012 A/C Drain Valve	PXS-PL-V085B	1	PB	1 yr	M * S
CMT A Upper Level B Isolation 1	PXS-PL-V086A	1	PB	1 yr	M * S
CMT B Upper Level B Isolation 1	PXS-PL-V086B	1	PB	1 yr	M * S
CMT A Upper Level B Isolation 2	PXS-PL-V087A	1	PB	1 yr	M * S
CMT B Upper Level B Isolation 2	PXS-PL-V087B	1	PB	1 yr	M * S
CMT A Upper Level B Vent	PXS-PL-V088A	1	PB	1 yr	M * S
CMT B Upper Level B Vent	PXS-PL-V088B	1	PB	1 yr	M * S
CMT A Upper Level B Drain	PXS-PL-V089A	1	PB	1 yr	M * S
CMT B Upper Level B Drain	PXS-PL-V089B	1	PB	1 yr	M * S
CMT A Lower Level A Isolation 1	PXS-PL-V092A	1	PB	1 yr	M * S
CMT B Lower Level A Isolation 1	PXS-PL-V092B	1	PB	1 yr	M * S
CMT A Lower Level A Isolation 2	PXS-PL-V093A	1	PB	1 yr	M * S
CMT B Lower Level A Isolation 2	PXS-PL-V093B	1	PB	1 yr	M * S
CMT A Lower Level A Vent	PXS-PL-V094A	1	PB	1 yr	M * S
CMT B Lower Level A Vent	PXS-PL-V094B	1	PB	1 yr	M * S
CMT A Lower Level A Drain	PXS-PL-V095A	1	PB	1 yr	M * S
CMT B Lower Level A Drain	PXS-PL-V095B	1	PB	1 yr	M * S
CMT A Lower Level B Isolation 1	PXS-PL-V096A	1	PB	1 yr	M * S
CMT B Lower Level B Isolation 1	PXS-PL-V096B	1	PB	1 yr	M * S
CMT A Lower Level B Isolation 2	PXS-PL-V097A	1	PB	1 yr	M * S
CMT B Lower Level B Isolation 2	PXS-PL-V097B	1	PB	1 yr	M * S
CMT A Lower Level B Vent	PXS-PL-V098A	1	PB	1 yr	M * S
CMT B Lower Level B Vent	PXS-PL-V098B	1	PB	1 yr	M * S
CMT A Lower Level B Drain	PXS-PL-V099A	1	PB	1 yr	M * S
CMT B Lower Level B Drain	PXS-PL-V099B	1	PB	1 yr	M * S
PRHR HX Inlet Isolation	PXS-PL-V101	1	PB	1 yr	M * S
Limit Switch	PXS-PL-V101-L	1	PAMS	1 yr	E * S
Motor Operator	PXS-PL-V101-M	1	ESF	5 min	E * S
PRHR HX Inlet Head Vent	PXS-PL-V102A	1	PB	1 yr	M * S
PRHR HX Inlet Head Drain	PXS-PL-V102B	1	PB	1 yr	M * S
PRHR HX Outlet Head Vent	PXS-PL-V103A	1	PB	1 yr	M * S
PRHR HX Outlet Head Drain	PXS-PL-V103B	1	PB	1 yr	M * S
PRHR HX Flow Transmitter A Isolation	PXS-PL-V104A	1	PB	1 yr	M * S
PRHR HX Flow Transmitter B Isolation	PXS-PL-V104B	1	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
PRHR HX Flow Transmitter A Isolation	PXS-PL-V105A	1	PB	1 yr	M * S
PRHR HX Flow Transmitter B Isolation	PXS-PL-V105B	1	PB	1 yr	M * S
Containment Recirculation A Highpoint Vent	PXS-PL-V106	1	PB	1 yr	M * S
Containment Recirculation A Highpoint Vent	PXS-PL-V107	1	PB	1 yr	M * S
PRHR HX/RCS Return Isolation	PXS-PL-V109	1	PB	1 yr	M * S
PRHR HX Highpoint Vent	PXS-PL-V111A	1	PB	1 yr	M * S
PRHR HX Highpoint Vent	PXS-PL-V111B	1	PB	1 yr	M * S
PRHR HX Pressure Transmitter Isolation	PXS-PL-V113	1	PB	1 yr	M * S
Containment Recirculation A Drain	PXS-PL-V115A	1	PB	1 yr	M * S
Containment Recirculation B Drain	PXS-PL-V115B	1	PB	1 yr	M * S
Containment Recirculation A Drain	PXS-PL-V116A	1	PB	1 yr	M * S
Containment Recirculation B Drain	PXS-PL-V116B	1	PB	1 yr	M * S
Recirc Sump A Isolation	PXS-PL-V117A	1	ESF	24 hrs	M * S
Limit Switch	PXS-PL-V117A-L	1	PAMS	1 yr (Note 9)	E * S
Motor Operator	PXS-PL-V117A-M	1	ESF	24 hrs	E * S
Recirc Sump B Isolation	PXS-PL-V117B	1	ESF	24 hrs	M * S
Limit Switch	PXS-PL-V117B-L	1	PAMS	1 yr (Note 9)	E * S
Motor Operator	PXS-PL-V117B-M	1	ESF	24 hrs	E * S
IRWST Line A Isolation	PXS-PL-V121A	1	PB	1 yr	M * S
IRWST Line B Isolation	PXS-PL-V121B	1	PB	1 yr	M * S
IRWST Injection Check Test	PXS-PL-V126A	1	PB	1 yr	M * S
IRWST Injection Check Test	PXS-PL-V126B	1	PB	1 yr	M * S
IRWST Injection Line A Drain	PXS-PL-V127	1	PB	1 yr	M * S
IRWST Injection Check Test	PXS-PL-V128A	1	PB	1 yr	M * S
IRWST Injection Check Test	PXS-PL-V128B	1	PB	1 yr	M * S
IRWST Injection Check Test	PXS-PL-V129A	1	PB	1 yr	M * S
IRWST Injection Check Test	PXS-PL-V129B	1	PB	1 yr	M * S
IRWST Injection Line A Drain	PXS-PL-V131A	1	PB	1 yr	M * S
IRWST Injection Line B Drain	PXS-PL-V131B	1	PB	1 yr	M * S
IRWST Injection Line A Drain	PXS-PL-V132A	1	PB	1 yr	M * S
IRWST Injection Line B Drain	PXS-PL-V132B	1	PB	1 yr	M * S
IRWST Injection Line A Highpoint Vent	PXS-PL-V133A	1	PB	1 yr	M * S
IRWST Injection Line B Highpoint Vent	PXS-PL-V133B	1	PB	1 yr	M * S
IRWST Injection Line A Highpoint Vent	PXS-PL-V134A	1	PB	1 yr	M * S
IRWST Injection Line B Highpoint Vent	PXS-PL-V134B	1	PB	1 yr	M * S
IRWST Injection Line A Highpoint Vent Isolation	PXS-PL-V135A	1	PB	1 yr	M * S
IRWST Injection Line B Highpoint Vent Isolation	PXS-PL-V135B	1	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
RNS Suction Pump Line Drain	PXS-PL-V149	1	PB	1 yr	M * S
IRWST Wide Range Level Transmitter B Isolation	PXS-PL-V151B	1	PB	1 yr	M * S
IRWST Wide Range Level Transmitter C Isolation	PXS-PL-V151C	1	PB	1 yr	M * S
IRWST Wide Range Level Transmitter D Isolation	PXS-PL-V151D	1	PB	1 yr	M * S
IRWST Lower Narrow Range Level Transmitter A Isolation	PXS-PL-V161A	1	PB	1 yr	M * S
IRWST Lower Narrow Range Level Transmitter B Isolation	PXS-PL-V161B	1	PB	1 yr	M * S
IRWST Lower Narrow Range Level Transmitter C Isolation	PXS-PL-V161C	1	PB	1 yr	M * S
IRWST Lower Narrow Range Level Transmitter D Isolation	PXS-PL-V161D	1	PB	1 yr	M * S
IRWST Lower Narrow Range Level Transmitter A Reference Leg Isolation	PXS-PL-V162A	1	PB	1 yr	M * S
IRWST Lower Narrow Range Level Transmitter B Reference Leg Isolation	PXS-PL-V162B	1	PB	1 yr	M * S
IRWST Lower Narrow Range Level Transmitter C Reference Leg Isolation	PXS-PL-V162C	1	PB	1 yr	M * S
IRWST Lower Narrow Range Level Transmitter D Reference Leg Isolation	PXS-PL-V162D	1	PB	1 yr	M * S
PRHR Flow Transmitter A Vent	PXS-PL-V170A	1	PB	1 yr	M * S
PRHR Flow Transmitter B Vent	PXS-PL-V170B	1	PB	1 yr	M * S
PRHR Flow Transmitter A Vent	PXS-PL-V171A	1	PB	1 yr	M * S
PRHR Flow Transmitter B Vent	PXS-PL-V171B	1	PB	1 yr	M * S
Accumulator A Leak Test	PXS-PL-V201A	1	PB	1 yr	M * S
Accumulator B Leak Test	PXS-PL-V201B	1	PB	1 yr	M * S
Accumulator A Leak Test	PXS-PL-V202A	1	PB	1 yr	M * S
Accumulator B Leak Test	PXS-PL-V202B	1	PB	1 yr	M * S
RNS Discharge Leak Test	PXS-PL-V205A	1	PB	1 yr	M * S
RNS Discharge Leak Test	PXS-PL-V205B	1	PB	1 yr	M * S
RNS Discharge Leak Test	PXS-PL-V206	1	PB	1 yr	M * S
RNS Suction Leak Test	PXS-PL-V207A	1	PB	1 yr	M * S
RNS Suction Leak Test	PXS-PL-V207B	1	PB	1 yr	M * S
RNS Suction Leak Test	PXS-PL-V208A	1	PB	1 yr	M * S
Core Makeup Tank A Fill Isolation	PXS-PL-V230A	1	PB	1 yr	M * S
Core Makeup Tank B Fill Isolation	PXS-PL-V230B	1	PB	1 yr	M * S
Core Makeup Tank A Fill Check	PXS-PL-V231A	1	PB	1 yr	M * S
Core Makeup Tank B Fill Check	PXS-PL-V231B	1	PB	1 yr	M * S
Accumulator A Fill/Drain Isolation	PXS-PL-V232A	1	PB	1 yr	M * S
Accumulator B Fill/Drain Isolation	PXS-PL-V232B	1	PB	1 yr	M * S
Accumulator A to PXS Test Header Iso Valve	PXS-PL-V233A	1	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Accumulator B to PXS Test Header Iso Valve	PXS-PL-V233B	1	PB	1 yr	M * S
CMT A Check Valve Test Valve	PXS-PL-V250A	1	PB	1 yr	M * S
CMT B Check Valve Test Valve	PXS-PL-V250B	1	PB	1 yr	M * S
CMT A Check Valve Test Valve	PXS-PL-V251A	1	PB	1 yr	M * S
CMT B Check Valve Test Valve	PXS-PL-V251B	1	PB	1 yr	M * S
CMT A Check Valve Test Valve	PXS-PL-V252A	1	PB	1 yr	M * S
CMT B Check Valve Test Valve	PXS-PL-V252B	1	PB	1 yr	M * S
ADS Test Valve	RCS-PL-V007A	1	PB	1 yr	M * S
ADS Test Valve	RCS-PL-V007B	1	PB	1 yr	M * S
ADS Test Valve	RCS-PL-V007C	1	PB	1 yr	M * S
ADS Valve Leakage Check Valve	RCS-PL-V008	1	PB	1 yr	M * S
Fourth Stage ADS Isolation	RCS-PL-V014A	1	PB	1 yr	M * S
Limit Switch	RCS-PL-V014A-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V014A-M	1	ESF	24 hrs	E * S
Fourth Stage ADS Isolation	RCS-PL-V014B	1	PB	1 yr	M * S
Limit Switch	RCS-PL-V014B-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V014B-M	1	ESF	24 hrs	E * S
Fourth Stage ADS Isolation	RCS-PL-V014C	1	PB	1 yr	M * S
Limit Switch	RCS-PL-V014C-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V014C-M	1	ESF	24 hrs	E * S
Fourth Stage ADS Isolation	RCS-PL-V014D	1	PB	1 yr	M * S
Limit Switch	RCS-PL-V014D-L	1	PAMS	1 yr	E * S
Motor Operator	RCS-PL-V014D-M	1	ESF	24 hrs	E * S
Hot Leg 2 Level Instrument Root	RCS-PL-V095	1	PB	1 yr	M * S
Hot Leg 2 Level Instrument Root	RCS-PL-V096	1	PB	1 yr	M * S
Hot Leg 1 Level Instrument Root	RCS-PL-V097	1	PB	1 yr	M * S
Hot Leg 1 Level Instrument Root	RCS-PL-V098	1	PB	1 yr	M * S
Hot Leg 1 Flow Instrument Root	RCS-PL-V101A	1	PB	1 yr	M * S
Hot Leg 1 Flow Instrument Root	RCS-PL-V101B	1	PB	1 yr	M * S
Hot Leg 1 Flow Instrument Root	RCS-PL-V101C	1	PB	1 yr	M * S
Hot Leg 1 Flow Instrument Root	RCS-PL-V101D	1	PB	1 yr	M * S
Hot Leg 1 Flow Instrument Root	RCS-PL-V101E	1	PB	1 yr	M * S
Hot Leg 1 Flow Instrument Root	RCS-PL-V101F	1	PB	1 yr	M * S
Hot Leg 2 Flow Instrument Root	RCS-PL-V102A	1	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Hot Leg 2 Flow Instrument Root	RCS-PL-V102B	1	PB	1 yr	M * S
Hot Leg 2 Flow Instrument Root	RCS-PL-V102C	1	PB	1 yr	M * S
Hot Leg 2 Flow Instrument Root	RCS-PL-V102D	1	PB	1 yr	M * S
Hot Leg 2 Flow Instrument Root	RCS-PL-V102E	1	PB	1 yr	M * S
Hot Leg 2 Flow Instrument Root	RCS-PL-V102F	1	PB	1 yr	M * S
PRHR HX Outlet Line Drain	RCS-PL-V103	1	PB	1 yr	M * S
Hot Leg 1 Sample Isolation	RCS-PL-V108A	1	PB	1 yr	M * S
Hot Leg 2 Sample Isolation	RCS-PL-V108B	1	PB	1 yr	M * S
PZR Spray Valve	RCS-PL-V110A	1	PB	1 yr	M * S
PZR Spray Valve	RCS-PL-V110B	1	PB	1 yr	M * S
PZR Spray Block Valve	RCS-PL-V111A	1	PB	1 yr	M * S
PZR Spray Block Valve	RCS-PL-V111B	1	PB	1 yr	M * S
Cold Leg 1A Bend Instrument Root	RCS-PL-V171A	1	PB	1 yr	M * S
Cold Leg 1A Bend Instrument Root	RCS-PL-V171B	1	PB	1 yr	M * S
Cold Leg 1B Bend Instrument Root	RCS-PL-V172A	1	PB	1 yr	M * S
Cold Leg 1B Bend Instrument Root	RCS-PL-V172B	1	PB	1 yr	M * S
Cold Leg 2A Bend Instrument Root	RCS-PL-V173A	1	PB	1 yr	M * S
Cold Leg 2A Bend Instrument Root	RCS-PL-V173B	1	PB	1 yr	M * S
Cold Leg 2B Bend Instrument Root	RCS-PL-V174A	1	PB	1 yr	M * S
Cold Leg 2B Bend Instrument Root	RCS-PL-V174B	1	PB	1 yr	M * S
PZR Manual Vent	RCS-PL-V204	1	PB	1 yr	M * S
PZR Manual Vent	RCS-PL-V205	1	PB	1 yr	M * S
PZR Spray Bypass	RCS-PL-V210A	1	PB	1 yr	M * S
PZR Spray Bypass	RCS-PL-V210B	1	PB	1 yr	M * S
PZR Level Steam Space Instrument Root	RCS-PL-V225A	1	PB	1 yr	M * S
PZR Level Steam Space Instrument Root	RCS-PL-V225B	1	PB	1 yr	M * S
PZR Level Steam Space Instrument Root	RCS-PL-V225C	1	PB	1 yr	M * S
PZR Level Steam Space Instrument Root	RCS-PL-V225D	1	PB	1 yr	M * S
PZR Level Liquid Space Instrument Root	RCS-PL-V226A	1	PB	1 yr	M * S
PZR Level Liquid Space Instrument Root	RCS-PL-V226B	1	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
PZR Level Liquid Space Instrument Root	RCS-PL-V226C	1	PB	1 yr	M * S
PZR Level Liquid Space Instrument Root	RCS-PL-V226D	1	PB	1 yr	M * S
Pressurizer Level Reference Leg Tubing Instrument Root	RCS-PL-V227A	1	PB	1 yr	M *
Pressurizer Level Reference Leg Tubing Instrument Root	RCS-PL-V227B	1	PB	1 yr	M *
Pressurizer Level Reference Leg Tubing Instrument Root	RCS-PL-V227C	1	PB	1 yr	M *
Pressurizer Level Reference Leg Tubing Instrument Root	RCS-PL-V227D	1	PB	1 yr	M *
Pressurizer Liquid Sample Isolation	RCS-PL-V228	1	PB	1 yr	M * S
Wide Range PZR Level Liquid Space Instrument Root	RCS-PL-V229	1	PB	1 yr	M * S
Manual Head Vent	RCS-PL-V232	1	PB	1 yr	M * S
Head Vent Isolation	RCS-PL-V233	1	PB	1 yr	M * S
ADS Valve Discharge Header Drain Isolation	RCS-PL-V241	1	PB	1 yr	M * S
ADS Discharge Line Isolation	RCS-PL-V250	1	PB	1 yr	M * S
RCP 1A Vent	RCS-PL-V260A	1	PB	1 yr	M * S
RCP 1B Vent	RCS-PL-V260B	1	PB	1 yr	M * S
RCP 2A Vent	RCS-PL-V260C	1	PB	1 yr	M * S
RCP 2B Vent	RCS-PL-V260D	1	PB	1 yr	M * S
RCP 1A Drain	RCS-PL-V261A	1	PB	1 yr	M * S
RCP 1B Drain	RCS-PL-V261B	1	PB	1 yr	M * S
RCP 2A Drain	RCS-PL-V261C	1	PB	1 yr	M * S
RCP 2B Drain	RCS-PL-V261D	1	PB	1 yr	M * S
RCS Pressure Boundary Valve Thermal Relief Isolation	RNS-PL-V004A	1	PB	1 yr	M * S
RCS Pressure Boundary Valve Thermal Relief Isolation	RNS-PL-V004B	1	PB	1 yr	M * S
RNS Pump A Suction Isolation	RNS-PL-V005A	6	PB	1 yr	M ** S
RNS Pump B Suction Isolation	RNS-PL-V005B	6	PB	1 yr	M ** S
RNS HX A Outlet Flow Control	RNS-PL-V006A	6	PB	1 yr	M ** S
RNS HX B Outlet Flow Control	RNS-PL-V006B	6	PB	1 yr	M ** S
RNS Pump A Discharge Isolation	RNS-PL-V007A	6	PB	1 yr	M ** S
RNS Pump B Discharge Isolation	RNS-PL-V007B	6	PB	1 yr	M ** S
RNS HX A Bypass Flow Control	RNS-PL-V008A	6	PB	1 yr	M ** S
RNS HX B Bypass Flow Control	RNS-PL-V008B	6	PB	1 yr	M ** S
RNS Discharge Containment Isolation Valve Test	RNS-PL-V010	6	PB	1 yr	M ** S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
RNS Discharge Containment Isolation Valve Test Connection	RNS-PL-V014	1	PB	1 yr	M * S
RNS Discharge Containment Penetration Isolation Valves Test	RNS-PL-V016	1	PB	1 yr	M * S
RNS Discharge to IRWST Isolation	RNS-PL-V024	1	PB	1 yr	M * S
RNS Suction from IRWST – Bonnet Relief Isolation	RNS-PL-V025	1	PB	1 yr	M * S
RNS Suction from IRWST – Containment Isolation Test	RNS-PL-V026	1	PB	1 yr	M * S
RNS Discharge to CVS	RNS-PL-V029	1	PB	1 yr	M * S
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V031A	6	PB	1 yr	M ** S
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V031B	6	PB	1 yr	M ** S
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V032A	6	PB	1 yr	M ** S
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V032B	6	PB	1 yr	M ** S
RNS Pump A Suction Pressure Instrument Isolation	RNS-PL-V033A	6	PB	1 yr	M **
RNS Pump B Suction Pressure Instrument Isolation	RNS-PL-V033B	6	PB	1 yr	M **
RNS Pump A Discharge Pressure Instrument Isolation	RNS-PL-V034A	6	PB	1 yr	M ** S
RNS Pump B Discharge Pressure Instrument Isolation	RNS-PL-V034B	6	PB	1 yr	M ** S
RNS Pump A Suction Piping Drain Isolation	RNS-PL-V036A	6	PB	1 yr	M ** S
RNS Pump B Suction Piping Drain Isolation	RNS-PL-V036B	6	PB	1 yr	M ** S
RNS Pump Discharge Relief	RNS-PL-V045	6	PB	1 yr	M **
RNS Pump Seal Cooler A Vent Isolation	RNS-PL-V048A	6	PB	1 yr	M** S
RNS Pump Seal Cooler B Vent Isolation	RNS-PL-V048B	6	PB	1 yr	M** S
RNS Pump Seal Cooler A Drain Isolation	RNS-PL-V049A	6	PB	1 yr	M** S
RNS Pump Seal Cooler B Drain Isolation	RNS-PL-V049B	6	PB	1 yr	M** S
RNS Pump A Casing Drain Isolation	RNS-PL-V050	6	PB	1 yr	M ** S
RNS Pump B Casing Drain Isolation	RNS-PL-V051	6	PB	1 yr	M ** S
RNS Suction from SFP Isolation	RNS-PL-V052	6	PB	1 yr	M **
RNS Discharge to SFP Isolation	RNS-PL-V053	6	PB	1 yr	M **
RNS Suction from Cask Loading Pit Isolation Valve	RNS-PL-V055	6	PB	1 yr	M **
RNS Pump Suction to Cask Loading Pit Isolation	RNS-PL-V056	6	PB	1 yr	M **

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
RNS Train A Miniflow Isolation Valve	RNS-PL-V057A	6	PB	1 yr	M **
RNS Train B Miniflow Isolation Valve	RNS-PL-V057B	6	PB	1 yr	M ** S
RNS Pump Suction Containment Isolation Test Connection	RNS-PL-V059	6	PB	1 yr	M ** S
RNS Discharge Drain Valve	RNS-PL-V065	6	PB	1 yr	M** S
RNS Discharge to DVI Line A Drain	RNS-PL-V066A	1	PB	1 yr	M * S
RNS Discharge to DVI Line B Drain	RNS-PL-V066B	1	PB	1 yr	M * S
RNS Discharge to DVI Line A Drain	RNS-PL-V067A	1	PB	1 yr	M * S
RNS Discharge to DVI Line B Drain	RNS-PL-V067B	1	PB	1 yr	M * S
RNS Discharge to IRWST Drain	RNS-PL-V068	1	PB	1 yr	M * S
RNS HX A Channel Head Drain Isolation	RNS-PL-V071A	6	PB	1 yr	M**
RNS HX B Channel Head Drain Isolation	RNS-PL-V071B	6	PB	1 yr	M**
RNS HX A Channel Head Drain Isolation	RNS-PL-V072A	6	PB	1 yr	M**
RNS HX B Channel Head Drain Isolation	RNS-PL-V072B	6	PB	1 yr	M**
RNS HX A Channel Head Drain Isolation	RNS-PL-V073A	6	PB	1 yr	M**
RNS HX B Channel Head Drain Isolation	RNS-PL-V073B	6	PB	1 yr	M**
RNS HX A Channel Head Drain Isolation	RNS-PL-V074A	6	PB	1 yr	M**
RNS HX B Channel Head Drain Isolation	RNS-PL-V074B	6	PB	1 yr	M**
RNS HX A Channel Head Drain Isolation	RNS-PL-V075A	6	PB	1 yr	M**
RNS HX B Channel Head Drain Isolation	RNS-PL-V075B	6	PB	1 yr	M**
RNS Cask Loading Pit Suction Line Vent	RNS-PL-V081	6	PB	1 yr	M **
RNS Discharge Drain	RNS-PL-V082	1	PB	1 yr	M* S
LT019A Root Isolation Valve	SFS-PL-V024A	6	PB	1 yr	M **
LT019B Root Isolation Valve	SFS-PL-V024B	6	PB	1 yr	M **
LT019C Root Isolation Valve	SFS-PL-V024C	6	PB	1 yr	M **
LT020 Root Isolation Valve	SFS-PL-V028	6	PB	1 yr	M **
SFS Refueling Cavity Drain to SGS Compartment Isolation Limit Switch	SFS-PL-V031 SFS-PL-V031-L	1 1	PB PAMS	1 yr 1 yr (Note 10)	M * S E * S
SFS Refueling Cavity Suction Isolation	SFS-PL-V032	1	PB	1 yr	M * S
SFS Refueling Cavity Drain to Containment Sump Isolation Limit Switch	SFS-PL-V033 SFS-PL-V033-L	1 1	PB PAMS	1 yr 1 yr (Note 10)	M * S E * S
SFS Suction Line from IRWST Isolation	SFS-PL-V039	1	PB	1 yr	M * S
SFS Fuel Transfer Canal Suction Isolation	SFS-PL-V040	6	PB	1 yr	M **
SFS Cask Loading Pit Suction Isolation	SFS-PL-V041	6	PB	1 yr	M **
Cask Loading Pit Level Transmitter Root Isolation Valve	SFS-PL-V043	6	PB	1 yr	M **
SFS Containment Penetration Test Connection	SFS-PL-V048	6	PB	1 yr	M **

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Environmentally Qualified Electrical and Mechanical Equipment

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
SFS Containment Penetration Test Connection Isolation	SFS-PL-V056	1	PB	1 yr	M * S
SFS Containment Isolation Valve V034 Test	SFS-PL-V058	1	PB	1 yr	M * S
SFS Containment Floodup Isolation Valve	SFS-PL-V075	1	PB	1 yr	M * S
Limit Switch	SFS-PL-V075-L	1	PAMS	1 yr (Note 10)	E * S
Refueling Cavity Drain Line Test Connection	SFS-PL-V117	1	PB	1 yr	M * S
LT001, LT011 Root Isolation Valve	SGS-PL-V001A	1	PB	1 yr	M * S
LT005, LT013 Root Isolation Valve	SGS-PL-V001B	1	PB	1 yr	M * S
LT001 Root Isolation Valve	SGS-PL-V002A	1	PB	1 yr	M * S
LT005 Root Isolation Valve	SGS-PL-V002B	1	PB	1 yr	M * S
LT002, LT012 Root Isolation Valve	SGS-PL-V003A	1	PB	1 yr	M * S
LT006, LT014 Root Isolation Valve	SGS-PL-V003B	1	PB	1 yr	M * S
LT002 Root Isolation Valve	SGS-PL-V004A	1	PB	1 yr	M * S
LT006 Root Isolation Valve	SGS-PL-V004B	1	PB	1 yr	M * S
LT003, LT015, LT044 Root Isolation Valve	SGS-PL-V005A	1	PB	1 yr	M * S
LT007, LT017, LT046 Root Isolation Valve	SGS-PL-V005B	1	PB	1 yr	M * S
LT003 Root Isolation Valve	SGS-PL-V006A	1	PB	1 yr	M * S
LT007 Root Isolation Valve	SGS-PL-V006B	1	PB	1 yr	M * S
LT004, LT016, LT045 Root Isolation Valve	SGS-PL-V007A	1	PB	1 yr	M * S
LT008, LT018, LT047 Root Isolation Valve	SGS-PL-V007B	1	PB	1 yr	M * S
LT004 Root Isolation Valve	SGS-PL-V008A	1	PB	1 yr	M * S
LT008 Root Isolation Valve	SGS-PL-V008B	1	PB	1 yr	M * S
LT011 Root Isolation Valve	SGS-PL-V010A	1	PB	1 yr	M * S
LT013 Root Isolation Valve	SGS-PL-V010B	1	PB	1 yr	M * S
LT011 Root Isolation Valve	SGS-PL-V011A	1	PB	1 yr	M * S
LT013 Root Isolation Valve	SGS-PL-V011B	1	PB	1 yr	M * S
LT012 Root Isolation Valve	SGS-PL-V012A	1	PB	1 yr	M * S
LT014 Root Isolation Valve	SGS-PL-V012B	1	PB	1 yr	M * S
LT012 Root Isolation Valve	SGS-PL-V013A	1	PB	1 yr	M * S
LT014 Root Isolation Valve	SGS-PL-V013B	1	PB	1 yr	M * S
FT021 Root Isolation Valve	SGS-PL-V015A	1	PB	1 yr	M * S
FT023 Root Isolation Valve	SGS-PL-V015B	1	PB	1 yr	M * S
FT020 Root Isolation Valve	SGS-PL-V016A	1	PB	1 yr	M * S
FT022 Root Isolation Valve	SGS-PL-V016B	1	PB	1 yr	M * S
FT021 Root Isolation Valve	SGS-PL-V017A	1	PB	1 yr	M * S
FT023 Root Isolation Valve	SGS-PL-V017B	1	PB	1 yr	M * S
FT020 Root Isolation Valve	SGS-PL-V018A	1	PB	1 yr	M * S
FT022 Root Isolation Valve	SGS-PL-V018B	1	PB	1 yr	M * S
Main Steam Line Vent Isolation	SGS-PL-V019A	1	PB	1 yr	M * S
Main Steam Line Vent Isolation	SGS-PL-V019B	1	PB	1 yr	M * S
FT024 Root Isolation Valve	SGS-PL-V020A	1	PB	1 yr	M * S
FT025 Root Isolation Valve	SGS-PL-V020B	1	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
FT024 Root Isolation Valve	SGS-PL-V021A	1	PB	1 yr	M * S
FT025 Root Isolation Valve	SGS-PL-V021B	1	PB	1 yr	M * S
PT030 Root Isolation Valve	SGS-PL-V022A	1	PB	1 yr	M * S
PT034 Root Isolation Valve	SGS-PL-V022B	1	PB	1 yr	M * S
PT031 Root Isolation Valve	SGS-PL-V023A	2	PB	1 yr	M **
PT035 Root Isolation Valve	SGS-PL-V023B	2	PB	1 yr	M **
PT032 Root Isolation Valve	SGS-PL-V024A	1	PB	1 yr	M * S
PT036 Root Isolation Valve	SGS-PL-V024B	1	PB	1 yr	M * S
PT033 Root Isolation Valve	SGS-PL-V025A	2	PB	1 yr	M **
PT037 Root Isolation Valve	SGS-PL-V025B	2	PB	1 yr	M **
Steam Line 1 Nitrogen Supply Isolation	SGS-PL-V038A	5	PB	1 yr	M * S
Steam Line 2 Nitrogen Supply Isolation	SGS-PL-V038B	5	PB	1 yr	M * S
MSIV Bypass Control Isolation	SGS-PL-V042A	5	PB	1 yr	M * S
MSIV Bypass Control Isolation	SGS-PL-V042B	5	PB	1 yr	M * S
MSIV Bypass Control Isolation	SGS-PL-V043A	5	PB	1 yr	M * S
MSIV Bypass Control Isolation	SGS-PL-V043B	5	PB	1 yr	M * S
SG1 Condensate Pipe Drain Valve	SGS-PL-V045A	5	PB	1 yr	M * S
SG2 Condensate Pipe Drain Valve	SGS-PL-V045B	5	PB	1 yr	M * S
LT015 Root Isolation Valve	SGS-PL-V046A	1	PB	1 yr	M * S
LT017 Root Isolation Valve	SGS-PL-V046B	1	PB	1 yr	M * S
LT015, LT044 Root Isolation Valve	SGS-PL-V047A	1	PB	1 yr	M * S
LT017, LT046 Root Isolation Valve	SGS-PL-V047B	1	PB	1 yr	M * S
LT016 Root Isolation Valve	SGS-PL-V048A	1	PB	1 yr	M * S
LT018 Root Isolation Valve	SGS-PL-V048B	1	PB	1 yr	M * S
LT016, LT045 Root Isolation Valve	SGS-PL-V049A	1	PB	1 yr	M * S
LT018, LT047 Root Isolation Valve	SGS-PL-V049B	1	PB	1 yr	M * S
LT044 Root Isolation Valve	SGS-PL-V050A	1	PB	1 yr	M * S
LT046 Root Isolation Valve	SGS-PL-V050B	1	PB	1 yr	M * S
LT044 Root Isolation Valve	SGS-PL-V051A	1	PB	1 yr	M * S
LT046 Root Isolation Valve	SGS-PL-V051B	1	PB	1 yr	M * S
LT045 Root Isolation Valve	SGS-PL-V052A	1	PB	1 yr	M * S
LT047 Root Isolation Valve	SGS-PL-V052B	1	PB	1 yr	M * S
LT045 Root Isolation Valve	SGS-PL-V053A	1	PB	1 yr	M * S
LT047 Root Isolation Valve	SGS-PL-V053B	1	PB	1 yr	M * S
PT062 Root Isolation Valve	SGS-PL-V056A	5	PB	1 yr	M * S
PT063 Root Isolation Valve	SGS-PL-V056B	5	PB	1 yr	M * S
Main Feedwater Check	SGS-PL-V058A	5	PB	1 yr	M * S
Main Feedwater Check	SGS-PL-V058B	5	PB	1 yr	M * S
FT055A Root Isolation Valve	SGS-PL-V062A	5	PB	1 yr	M * S
FT056A Root Isolation Valve	SGS-PL-V062B	5	PB	1 yr	M * S
FT055A Root Isolation Valve	SGS-PL-V063A	5	PB	1 yr	M * S
FT056A Root Isolation Valve	SGS-PL-V063B	5	PB	1 yr	M * S

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
FT055B Root Isolation Valve	SGS-PL-V064A	5	PB	1 yr	M * S
FT056B Root Isolation Valve	SGS-PL-V064B	5	PB	1 yr	M * S
FT055B Root Isolation Valve	SGS-PL-V065A	5	PB	1 yr	M * S
FT056B Root Isolation Valve	SGS-PL-V065B	5	PB	1 yr	M * S
FT055 Root Isolation Valve	SGS-PL-V066A	5	PB	1 yr	M * S
FT056 Root Isolation Valve	SGS-PL-V066B	5	PB	1 yr	M * S
FT055 Root Isolation Valve	SGS-PL-V068A	5	PB	1 yr	M * S
FT056 Root Isolation Valve	SGS-PL-V068B	5	PB	1 yr	M * S
SG 1 Blowdown Vent Line Isolation	SGS-PL-V076A	10	PB	1 yr	M *
SG 2 Blowdown Vent Line Isolation	SGS-PL-V076B	10	PB	1 yr	M *
SG1 Nitrogen Sparging Isolation	SGS-PL-V084A	1	PB	1 yr	M * S
SG2 Nitrogen Sparging Isolation	SGS-PL-V084B	1	PB	1 yr	M * S
Orifice Isolation Valve	SGS-PL-V093A	5	PB	1 yr	M * S
Orifice Isolation Valve	SGS-PL-V093B	5	PB	1 yr	M * S
Orifice Cleanout Line Isolation Valve	SGS-PL-V094A	5	PB	1 yr	M * S
Orifice Cleanout Line Isolation Valve	SGS-PL-V094B	5	PB	1 yr	M * S
Orifice Isolation Valve	SGS-PL-V095A	5	PB	1 yr	M * S
Orifice Isolation Valve	SGS-PL-V095B	5	PB	1 yr	M * S
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V096A	5	PB	1 yr	M * S
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V096B	5	PB	1 yr	M * S
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V097A	5	PB	1 yr	M * S
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V097B	5	PB	1 yr	M * S
Startup Feedwater Drain Isolation Valve	SGS-PL-V100A	5	PB	1 yr	M * S
Startup Feedwater Drain Isolation Valve	SGS-PL-V100B	5	PB	1 yr	M * S
Main Feedwater Drain Isolation Valve	SGS-PL-V101A	5	PB	1 yr	M * S
Main Feedwater Drain Isolation Valve	SGS-PL-V101B	5	PB	1 yr	M * S
Startup Feedwater Vent Isolation Valve	SGS-PL-V102A	5	PB	1 yr	M * S
Startup Feedwater Vent Isolation Valve	SGS-PL-V102B	5	PB	1 yr	M * S
Main Feedwater Vent Isolation Valve	SGS-PL-V103A	5	PB	1 yr	M * S
Main Feedwater Vent Isolation Valve	SGS-PL-V103B	5	PB	1 yr	M * S
Main Feedwater Drain Isolation Valve	SGS-PL-V104A	5	PB	1 yr	M * S
Main Feedwater Drain Isolation Valve	SGS-PL-V104B	5	PB	1 yr	M * S
Startup Feedwater Check Valve	SGS-PL-V256A	5	PB	1 yr	M * S
Startup Feedwater Check Valve	SGS-PL-V256B	5	PB	1 yr	M * S
SG 1 NR LT001 Upper Isolation Valve	SGS-PL-V300	1	PB	1 yr	M *
SG 1 NR LT001 Lower Isolation Valve	SGS-PL-V301	1	PB	1 yr	M *
SG 1 NR LT002 Upper Isolation Valve	SGS-PL-V302	1	PB	1 yr	M *
SG 1 NR LT002 Lower Isolation Valve	SGS-PL-V303	1	PB	1 yr	M *
SG 1 NR LT003 Upper Isolation Valve	SGS-PL-V304	1	PB	1 yr	M *
SG 1 NR LT003 Lower Isolation Valve	SGS-PL-V305	1	PB	1 yr	M *
SG 1 NR LT004 Upper Isolation Valve	SGS-PL-V306	1	PB	1 yr	M *

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
SG 1 NR LT004 Lower Isolation Valve	SGS-PL-V307	1	PB	1 yr	M *
SG 1 WR LT011 Lower Isolation Valve	SGS-PL-V308	1	PB	1 yr	M *
SG 1 WR LT012 Lower Isolation Valve	SGS-PL-V309	1	PB	1 yr	M *
SG 1 WR LT015 Lower Isolation Valve	SGS-PL-V310	1	PB	1 yr	M *
SG 1 WR LT016 Lower Isolation Valve	SGS-PL-V311	1	PB	1 yr	M *
SG 1 Main Steam FT020 Isolation Valve	SGS-PL-V312	1	PB	1 yr	M *
SG 1 Main Steam FT021 Isolation Valve	SGS-PL-V313	1	PB	1 yr	M *
SG 1 Main Steam FT024 Isolation Valve	SGS-PL-V314	1	PB	1 yr	M *
SG 2 NR LT005 Upper Isolation Valve	SGS-PL-V320	1	PB	1 yr	M *
SG 2 NR LT005 Lower Isolation Valve	SGS-PL-V321	1	PB	1 yr	M *
SG 2 NR LT006 Upper Isolation Valve	SGS-PL-V322	1	PB	1 yr	M *
SG 2 NR LT006 Lower Isolation Valve	SGS-PL-V323	1	PB	1 yr	M *
SG 2 NR LT007 Upper Isolation Valve	SGS-PL-V324	1	PB	1 yr	M *
SG 2 NR LT007 Lower Isolation Valve	SGS-PL-V325	1	PB	1 yr	M *
SG 2 NR LT008 Upper Isolation Valve	SGS-PL-V326	1	PB	1 yr	M *
SG 2 NR LT008 Lower Isolation Valve	SGS-PL-V327	1	PB	1 yr	M *
SG 2 WR LT013 Lower Isolation Valve	SGS-PL-V328	1	PB	1 yr	M *
SG 2 WR LT014 Lower Isolation Valve	SGS-PL-V329	1	PB	1 yr	M *
SG 2 WR LT017 Lower Isolation Valve	SGS-PL-V330	1	PB	1 yr	M *
SG 2 WR LT018 Lower Isolation Valve	SGS-PL-V331	1	PB	1 yr	M *
SG 2 Main Steam FT022 Isolation Valve	SGS-PL-V332	1	PB	1 yr	M *
SG 2 Main Steam FT023 Isolation Valve	SGS-PL-V333	1	PB	1 yr	M *
SG 2 Main Steam FT025 Isolation Valve	SGS-PL-V334	1	PB	1 yr	M *
VBS-PDT-032A Isolation Valve	VBS-PL-V184A	3	PB	1 yr	M
VBS-PDT-032B Isolation Valve	VBS-PL-V184B	3	PB	1 yr	M
VBS-PDT-032C Isolation Valve	VBS-PL-V184C	3	PB	1 yr	M
Air Delivery Line Pressure Instrument Isolation Valve A	VES-PL-V006A	7	PB	1 yr	M **
Air Delivery Line Pressure Instrument Isolation Valve B	VES-PL-V006B	7	PB	1 yr	M **
Air Delivery Line Maintenance Isolation Valve A	VES-PL-V010A	7	PB	1 yr	M **
Air Delivery Line Maintenance Isolation Valve B	VES-PL-V010B	7	PB	1 yr	M **
Air Delivery Line Maintenance Isolation Valve A	VES-PL-V011A	7	PB	1 yr	M **
Air Delivery Line Maintenance Isolation Valve B	VES-PL-V011B	7	PB	1 yr	M **
Temporary Instrument Isolation Valve A	VES-PL-V016	7	PB	1 yr	M **
Temporary Instrument Isolation Valve B	VES-PL-V020	7	PB	1 yr	M **
Air Bank 1 Isolation Valve A	VES-PL-V024A	7	PB	1 yr	M **
Air Bank 2 Isolation Valve B	VES-PL-V024B	7	PB	1 yr	M **
Air Bank 3 Isolation Valve C	VES-PL-V024C	7	PB	1 yr	M **
Air Bank 4 Isolation Valve D	VES-PL-V024D	7	PB	1 yr	M **

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Air Bank 1 Isolation Valve A	VES-PL-V025A	7	PB	1 yr	M **
Air Bank 2 Isolation Valve B	VES-PL-V025B	7	PB	1 yr	M **
Air Bank 3 Isolation Valve C	VES-PL-V025C	7	PB	1 yr	M **
Air Bank 4 Isolation Valve D	VES-PL-V025D	7	PB	1 yr	M **
Air Bank 1 Fill/Vent Isolation Valve A	VES-PL-V026A	7	PB	1 yr	M **
Air Bank 2 Fill/Vent Isolation Valve B	VES-PL-V026B	7	PB	1 yr	M **
Air Bank 3 Fill/Vent Isolation Valve C	VES-PL-V026C	7	PB	1 yr	M **
Air Bank 4 Fill/Vent Isolation Valve D	VES-PL-V026D	7	PB	1 yr	M **
DP Instrument Line Isolation Valve A	VES-PL-V043A	3	PB	1 yr	M
DP Instrument Line Isolation Valve B	VES-PL-V043B	3	PB	1 yr	M
Containment Isolation Test Connection	VFS-PL-V006	1	PB	1 yr	M *
RCS Ejector Discharge Isolation	VFS-PL-V007	1	PB	1 yr	M *
Containment Isolation Test Connection	VFS-PL-V008	6	PB	1 yr	M **
Containment Isolation Test Connection	VFS-PL-V012	1	PB	1 yr	M * S
Containment Isolation Test Connection	VFS-PL-V015	1	PB	1 yr	M * S
Containment Air Supply Line Test Connection	VFS-PL-V101	1	PB	1 yr	M *
Containment Atmosphere to Filtration Units Isolation	VFS-PL-V202	1	PB	1 yr	M *
Filtration Units to Containment Atmosphere Manual Isolation	VFS-PL-V587	1	PB	1 yr	M *
Main Equipment Hatch Test Connection	VUS-PL-V013	1	PB	1 yr	M *
Maintenance Equipment Hatch Test Connection	VUS-PL-V014	1	PB	1 yr	M *
Main Equipment Hatch Test Connection	VUS-PL-V015	1	PB	1 yr	M *
Maintenance Equipment Hatch Test Connection	VUS-PL-V016	1	PB	1 yr	M *
Personnel Hatch Test Connection	VUS-PL-V017	7	PB	1 yr	M **
Personnel Hatch Test Connection	VUS-PL-V018	7	PB	1 yr	M **
Personnel Hatch Test Connection	VUS-PL-V019	7	PB	1 yr	M **
Personnel Hatch Test Connection	VUS-PL-V020	7	PB	1 yr	M **
Personnel Hatch Test Connection	VUS-PL-V021	7	PB	1 yr	M **
Personnel Hatch Test Connection	VUS-PL-V022	7	PB	1 yr	M **
Fuel Transfer Tube Test Connection	VUS-PL-V023	1	PB	1 yr	M * S
Spare Penetration Test Connection	VUS-PL-V140	1	PB	1 yr	M *
Spare Penetration Test Connection	VUS-PL-V141	1	PB	1 yr	M *
Spare Penetration Test Connection	VUS-PL-V142	1	PB	1 yr	M *
VWS Supply Containment Penetration IRC Test Connection/Vent	VWS-PL-V424	1	PB	1 yr	M * S
VWS Return Containment Penetration ORC Test Connection/Vent	VWS-PL-V425	2	PB	1 yr	M **
WWS Main Control Room Isolation Valve	WWS-PL-V506	3	PB	1 yr	M

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Heat Exchangers					
Normal Residual Heat Removal Heat Exchanger A	RNS-ME-01A	6	PB	1 yr	M **
Normal Residual Heat Removal Heat Exchanger B	RNS-ME-01B	6	PB	1 yr	M **
Tanks					
Spent Fuel Pool	FHS-MT-01	11	ESF	1 yr	M **
Fuel Transfer Canal	FHS-MT-02	11	ESF	1 yr	M **
Refueling Cavity	FHS-MT-03	11	ESF	1 yr	M **
Spent Fuel Cask Loading Pit	FHS-MT-05	6	ESF	1 yr	M **
Spent Fuel Cask Washdown Pit	FHS-MT-06	6	ESF	1 yr	M **
Spent Fuel Transfer Gate	FHS-MY-Y01	6, 11	ESF (Note 11)	1 wk	M ** S
Spent Fuel Cask Loading Pit Gate	FHS-MY-Y02	11	ESF (Note 11)	1 wk	M ** S
Passive Containment Cooling Water Storage Tank	PCS-MT-01	9 (Note 12)	ESF	1 yr	M **
Water Distribution Bucket	PCS-MT-03	9	ESF	1 yr	M **
Water Collection Troughs	PCS-MT-04	9	ESF	1 yr	M **
Passive RHR Heat Exchanger	PXS-ME-01	1	ESF	1 yr	M * S
Accumulator Tank A	PXS-MT-01A	1	ESF	1 yr	M * S
Accumulator Tank B	PXS-MT-01B	1	ESF	1 yr	M * S
Core Makeup Tank A	PXS-MT-02A	1	ESF	1 yr	M * S
Core Makeup Tank B	PXS-MT-02B	1	ESF	1 yr	M * S
In-Containment Refueling Water Storage Tank	PXS-MT-03	1	ESF	1 yr	M * S
Emergency Air Storage Tank 01	VES-MT-01	7	ESF	1 yr	M **
Emergency Air Storage Tank 02	VES-MT-02	7	ESF	1 yr	M **
Emergency Air Storage Tank 03	VES-MT-03	7	ESF	1 yr	M **
Emergency Air Storage Tank 04	VES-MT-04	7	ESF	1 yr	M **
Emergency Air Storage Tank 05	VES-MT-05	7	ESF	1 yr	M **
Emergency Air Storage Tank 06	VES-MT-06	7	ESF	1 yr	M **
Emergency Air Storage Tank 07	VES-MT-07	7	ESF	1 yr	M **
Emergency Air Storage Tank 08	VES-MT-08	7	ESF	1 yr	M **
Emergency Air Storage Tank 09	VES-MT-09	7	ESF	1 yr	M **
Emergency Air Storage Tank 10	VES-MT-10	7	ESF	1 yr	M **
Emergency Air Storage Tank 11	VES-MT-11	7	ESF	1 yr	M **
Emergency Air Storage Tank 12	VES-MT-12	7	ESF	1 yr	M **
Emergency Air Storage Tank 13	VES-MT-13	7	ESF	1 yr	M **
Emergency Air Storage Tank 14	VES-MT-14	7	ESF	1 yr	M **
Emergency Air Storage Tank 15	VES-MT-15	7	ESF	1 yr	M **
Emergency Air Storage Tank 16	VES-MT-16	7	ESF	1 yr	M **
Emergency Air Storage Tank 17	VES-MT-17	7	ESF	1 yr	M **
Emergency Air Storage Tank 18	VES-MT-18	7	ESF	1 yr	M **
Emergency Air Storage Tank 19	VES-MT-19	7	ESF	1 yr	M **
Emergency Air Storage Tank 20	VES-MT-20	7	ESF	1 yr	M **
Emergency Air Storage Tank 21	VES-MT-21	7	ESF	1 yr	M **

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Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
Emergency Air Storage Tank 22	VES-MT-22	7	ESF	1 yr	M **
Emergency Air Storage Tank 23	VES-MT-23	7	ESF	1 yr	M **
Emergency Air Storage Tank 24	VES-MT-24	7	ESF	1 yr	M **
Emergency Air Storage Tank 25	VES-MT-25	7	ESF	1 yr	M **
Emergency Air Storage Tank 26	VES-MT-26	7	ESF	1 yr	M **
Emergency Air Storage Tank 27	VES-MT-27	7	ESF	1 yr	M **
Emergency Air Storage Tank 28	VES-MT-28	7	ESF	1 yr	M **
Emergency Air Storage Tank 29	VES-MT-29	7	ESF	1 yr	M **
Emergency Air Storage Tank 30	VES-MT-30	7	ESF	1 yr	M **
Emergency Air Storage Tank 31	VES-MT-31	7	ESF	1 yr	M **
Emergency Air Storage Tank 32	VES-MT-32	7	ESF	1 yr	M **
Eductor Bypass Isol. Discharge Silencer	VES-PY-N02	3	ESF	2 wks	M
DVI-B Inline Expansion Joint	PXS-PY-E01	1	PB	1 yr	M *

Notes:

1. RT (Reactor Trip), ESF (Engineered Safeguards Feature), PAMS (Post-Accident Monitoring), ISOL (Isolation), PB (Pressure Boundary); all active valves in this table have “PB-1 yr” in addition to any other requirements.
2. Zones identified in Table 3D.5-1
3. Not required post-accident
4. IIS-JE-TE009, IIS-JE-TE013, IIS-JE-TE030, and IIS-JE-TE-034 are not safety-related and therefore excluded from this list.
5. Reference Table 3D.4-2. The operating time presented is a conservatively established qualification time enveloping the actual operating time for component function.
6. E = Electrical Equipment Program
   M = Mechanical Equipment Program
   * = Harsh Environment
   ** = Radiation-Harsh Environment (See Appendix 3D, Subsection 3D.4.3.)
   + = Seismic Qualification not required
   S = Qualified for submergence or operation with spray
7. The Protection and Safety Monitoring Cabinets will be qualified to meet the function operating times identified in this table.
8. Provides the function of floodup level indication to the operator in the main control room (MCR), including during a seismic event.
9. Operating time required while submerged is 4 months.
10. Operation while submerged is not required.
11. Not required unless irradiated fuel is stored in the spent fuel pool. ESF leakage prevention from the spent fuel pool may also be met by maintaining equivalent water pressure (level) on the opposite side of the gate seal.
12. Conservatively identified as Zone 9 due to adjacent rooms.

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Appendix 3A HVAC Ducts and Duct Supports

This appendix provides the design criteria for seismic Category I and II HVAC ducts and their supports. These design criteria maintain structural integrity for seismic Category I and II ducts and functional capability for seismic Category I duct.

The structural components of a typical HVAC duct system include the sheet metal ducts, stiffeners for the ducts, duct supports, and other inline components such as duct heaters, dampers, etc.

3A.1 Codes and Standards

The design of the HVAC ducts and their supports conform to the following codes and standards:

- ASME N509-1989(R1996), Nuclear Power Plants Air Cleaning Units and Components
- ASME/ANSI AG-1-1997, Code on Nuclear Air and Gas Treatment
- American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, ANSI/AISC N690-1994. American Welding Society (AWS), Structural Welding Code - Steel, AWS D1.1-2000 provides an acceptable alternative for ANSI/AISC N690 for weld requirements as described in Subsections 3.8.3.2 and 3.8.4.2.
- American Iron and Steel Institute (AISI), Specification for the Design of Cold Formed Steel Structural Members, 1996 Edition and Supplement No. 1, July 30, 1999
- SMACNA, HVAC Duct Construction Standards, Metal and Flexible, Second Edition 1995.

3A.2 Loads and Load Combinations

3A.2.1 Loads

3A.2.1.1 Dead Load (D)

Dead load includes the weight of the duct sheet, stiffeners and inline components such as duct heaters and dampers. It also includes permanently attached items such as insulation and fireproofing, where applicable, and the weight of the duct supports. Temporary items used during construction or maintenance are removed prior to operation.

3A.2.1.2 Construction Live Load (L)

Live load consists of a load of 250 pounds to be applied only during construction or maintenance on an area of 10 square inches on the duct at a critical location to maximize flexural and shear stresses. This load is not combined with seismic loads.

3A.2.1.3 Pressure (P)

The duct metal thickness and stiffener requirements are based on maximum system design pressures. SMACNA or ASME guidelines, as applicable, are used in the design of duct metal thickness and stiffener requirements.

The pressure loads occur during normal plant operation, including plant start up testing, damper closure and normal airflow. Occasionally, overpressure transient loads such as rapid damper closure may also produce short duration pressure differential.

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3A.2.1.4 Safe Shutdown Earthquake (E_s)

Seismic response of the HVAC ductwork and its support system are produced due to seismic excitation of the supports.

3A.2.1.5 Wind Loads (W)

Ductwork within partially or fully vented buildings is subject to wind effects. Design wind loads are discussed in Section 3.3.

3A.2.1.6 Tornado Loads (W_t)

Ductwork within partially or fully vented buildings is subject to tornado differential pressure effects. Tornado loads are discussed in Section 3.3. Seismic Category I HVAC ductwork is protected from impact by tornado missiles.

3A.2.1.7 External Pressure Differential Loads (P_A)

Seismic Category I HVAC ductwork and its supports are designed to withstand dynamic external pressure differential loads resulting from postulated accident conditions. Usually HVAC ducts are routed outside the areas of potential pipe break.

3A.2.1.8 Thermal (T_O/T_A)

Stresses on the supports resulting from the ductwork expansion due to temperature changes are avoided by designing the system to take care of the expansion or by utilizing expansion joints. For ducts of gasketed companion angle construction, thermal loads are negligible. For ducts exposed to higher temperatures during a postulated accident condition, an evaluation is performed on a case by case basis for its effect.

3A.2.2 Load Combinations

The load combinations for various service levels are as follows:

Service Level		Load Combination
A	(Construction / maintenance)	$D + L + P + T_O$
A	(Normal Operating Condition)	$D + P + T_O$
B	(Severe Condition)	$D + W + P + T_O$
C	(Extreme Condition)	$D + E_s + P + T_O$
C	(Extreme Condition)	$D + W_t + P + T_O$
D	(Abnormal Condition)	$D + P + P_A + E_s + T_A$

3A.3 Analysis and Design

The HVAC duct support system is designed to maintain structural integrity of the duct. Function is not required for the seismic Category II ductwork. The stresses are maintained within the allowable limits specified in Subsection 3A.3.4. Section properties and masses are calculated in accordance with SMACNA standard.

The damping values for seismic analysis are as follows:

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- Welded HVAC Ductwork 4 percent
- Bolted HVAC Ductwork 7 percent

The duct design due to pressure loads is based on ASME/ANSI AG-1 for seismic Category I ducts and SMACNA for seismic Category II ducts.

The global behavior of the duct is determined from the overall bending of the duct between the supports. It is similar to the beam type bending. The dead load is combined with the seismic inertial load to determine the maximum bending moment. For determining the section modulus, the corners of the duct are considered effective. The corner length in each direction equals 32 times the thickness of the duct (t) for this purpose.

3A.3.1 Response Due to Seismic Loads

The methodology for seismic analysis is provided in Subsection 3.7.3. Seismic loads are determined by either using the equivalent static load method of analysis or by performing dynamic analysis.

Stresses are determined for the seismic excitation in two horizontal and one vertical direction. The stresses in the three directions are combined using the square root of sum of the squares (SRSS) method or the 100-40-40 method as described in Subsection 3.7.3.6.

3A.3.2 Deflection Criteria

Deflections for panels and stiffeners conform to the limits stated in the “Code on Nuclear Air and Gas Treatment.”

3A.3.3 Relative Movement

Clearances are provided for allowing relative movement between equipment, other commodities, and HVAC system.

3A.3.4 Allowable Stresses

The basic stress allowables for the HVAC ducts are in accordance with paragraph SA-4220 of ASME/ANSI AG-1.

The basic stress allowables for duct supports utilizing rolled structural shapes are in accordance with ANSI/AISC N690 and the supplemental requirements described in Subsection 3.8.4.5.2. The basic stress allowables for supports utilizing light gage cold rolled channel type sections are based on the manufacturer's published catalog values.

Service Level A and B	Basic Allowable
Service Level C and D	1.6 times basic allowable for tension and 1.4 times basic allowable for compression

3A.3.5 Connections

Connections are designed in accordance with the applicable codes and standards listed in Section 3A.1. For connections used with light gage cold rolled channel type sections, design is based on the manufacturer's published catalog values. Supports are attached to the building structure by bolted or welded connections. Fastening of the supports to concrete structures meets the supplemental requirements given in Subsection 3.8.4.5.1.

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Appendix 3B Leak-Before-Break Evaluation of the AP1000 Piping

General Design Criterion 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of conditions associated with normal operation, anticipated transients, and postulated accident conditions. However, the dynamic effects associated with pipe rupture may be excluded when analysis demonstrates that the probability of fluid system pipe rupture is extremely low. Dynamic effects are not considered for those segments of piping that are shown mechanistically, with a large margin, not to be susceptible to a pipe rupture.

The dynamic effects associated with pipe rupture include effects such as pipe break reaction loads, jets and jet impingement, subcompartment pressurization loads, and transient pipe rupture depressurization loads on other components.

The use of mechanistic pipe break to eliminate evaluation of dynamic effects of pipe rupture includes material selection, inspection, leak detection, and analysis. Subsection 3.6.3 outlines considerations relative to material selection, inspections, and leak detection. Subsection 5.2.5 describes the leak detection system inside containment. This appendix describes the analysis methods used to support the application of mechanistic pipe break to high-energy piping in the AP1000.

The analysis and criteria to eliminate dynamic effects of pipe breaks are encompassed in a methodology called leak-before-break (LBB). This methodology has been validated by theoretical investigations and test demonstrations sponsored by the industry and the NRC.

The primary regulatory documents for leak-before-break analyses are General Design Criterion No. 4 (GDC-4), Draft Standard Review Plan 3.6.3 (SRP 3.6.3) (Reference 1), and NUREG-1061, Volume 3 (Reference 2). Although SRP 3.6.3 has been issued only as a draft, its provisions are followed as guidelines to leak-before-break analyses.

Leak-before-break methodology has been applied to the reactor coolant loop and high-energy auxiliary line piping in operating nuclear power plants. The leak-before-break analysis used to support the piping design of the AP1000 is an application of the same methodology used in leak-before-beak evaluations previously accepted by the NRC.

In the AP1000, leak-before-break evaluations are performed for the reactor coolant loop, the surge line, selected other branch lines containing reactor coolant down to and including 6-inch diameter nominal pipe size, and portions of the main steam line. Those lines not qualified to the leak-before-break criteria are evaluated using the pipe rupture protection criteria outlined in Subsections 3.6.1 and 3.6.2.

This appendix provides a leak-before-break analysis for the applicable piping systems. Table 3B-1 provides a list of AP1000 leak-before-break piping systems.

3B.1 Leak-before-Break Criteria for AP1000 Piping

The methodology used for leak-before-break analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2). The steps are:

- Evaluate potential failure mechanisms
- Perform bounding analysis

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3B.2 Potential Failure Mechanisms for AP1000 Piping

In high-energy piping, there are material degradation mechanisms that could adversely affect the integrity of the system as well as its suitability for leak-before-break analysis. The following lists potential degradation (or "failure") mechanisms:

- Erosion-corrosion induced wall thinning
- Stress corrosion cracking (SCC)
- Water hammer
- Fatigue
- Thermal aging
- Thermal stratification
- Other mechanisms

The stainless steel piping is fabricated of SA312TP316LN or SA312TP304L material. The type 304L material is used in the accumulator discharge lines. The main steam piping is fabricated of SA335 Grade P11. The welds are made by the gas tungsten arc welding (GTAW) method.

The various degradation mechanisms are discussed in the following subsections.

3B.2.1 Erosion-Corrosion Induced Wall Thinning

Primary Loop Piping

Wall thinning by erosion and erosion-corrosion effects does not occur in the primary loop piping because Series 300 austenitic stainless steel material is highly resistant to these effects. The coolant velocity in the AP1000 primary loop is about 76 feet per second. This flow velocity is not expected to create erosion-corrosion effects since stainless steels are considered to be virtually immune (Reference 3). A review of erosion-corrosion in nuclear power systems (Reference 4) reported that "stainless steels are increasingly being used due to their excellent resistance to erosion-corrosion, even at high water velocities, 40 m/s (131 ft/sec)." The bend radii in the AP1000 hot and cold legs are greater than the bend radii used in the crossover legs of operating plants. There is no record of erosion-corrosion induced wall thinning in the primary loops of operating plants.

Auxiliary Stainless Steel Piping

Wall thinning by erosion-corrosion effects does not occur in the auxiliary stainless steel piping because Series 300 austenitic stainless materials are highly resistant to these effects. The coolant velocity in these systems is lower than in comparable systems in operating Westinghouse-designed pressurized water reactors. There is no record of erosion-corrosion induced wall thinning in the stainless steel piping of operating plants.

Main Steam Line

Main steam lines in the AP1000 are fabricated from SA335 Grade P11 Alloy steel. Erosion-corrosion induced wall thinning is not expected in the main steam line. Extensive work has been done investigating erosion-corrosion in carbon steel pipes. The main steam line has low susceptibility to erosion due to the pipe material composition, which has sufficient levels of chromium to preclude erosion-corrosion material loss. Susceptibility is also low due to the relatively high operating temperature and the high quality steam in the main steam line.

Based on the above discussion, erosion-corrosion induced wall thinning does not have an adverse effect on the integrity of the AP1000 leak-before-break piping systems.

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3B.2.2 Stress Corrosion Cracking

Stress corrosion cracking is not expected to occur in the AP1000 piping systems because the three conditions necessary for stress corrosion cracking to take place are not present. If any of these three conditions is not present, stress corrosion cracking will not take place. The three conditions are:

- There must be a corrosive environment.
- The material itself must be susceptible.
- Tensile stresses must be present in the material.

Primary Loop Piping

During plant operation, the reactor coolant water chemistry is monitored and maintained within specific limits (see Subsection 5.2.3 for a discussion of reactor coolant chemistry). Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking. The major water chemistry control standards are included in the plant operating procedures as a condition for plant operation.

The key to avoidance of a corrosive environment is control of oxygen. During normal power operation, oxygen concentration in the reactor coolant system is controlled to extremely low levels by controlling charging flow chemistry and maintaining a hydrogen overpressure in the reactor coolant at specified concentrations. Halogen concentration is controlled by maintaining concentrations of chlorides and fluorides within the specified limits. During plant operations, the likelihood of stress corrosion cracking in the primary loop piping systems is very low.

The elements of a water environment known to increase the susceptibility of austenitic stainless steel to stress corrosion are oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (for example, sulfides, sulfites, and thionates). Pipe cleaning standards prior to operation and careful water chemistry control during plant operation are applied to prevent the occurrence of a corrosive environment. Before being placed in service the piping is cleaned. During flushes and preoperational testing, water chemistry is controlled according to written specifications. Standards on chlorides, fluorides, conductivity, and pH are included in the guidelines for water for cleaning the piping.

Series 300 stainless steel materials have been chosen for the AP1000 due to their proven operating experience. These materials have operated in low-oxygen or no-oxygen environments with no incidents for a number of years. The requirements of Regulatory Guide 1.44, with the exceptions as stated in Subsection 1.9.1, will be used to maintain the experiences of the PWR applications for the use of Series 300 stainless steel materials.

Design tensile stresses in the reactor coolant loop are within the ASME Code, Section III allowables. Residual tensile stresses are expected in the welds and such stresses are not considered when designing by the ASME Code, Section III because these stresses are self-equilibrating and do not affect the failure loads. The residual stresses should not be more severe than for the operating Westinghouse pressurized water reactor plants (which have not experienced stress corrosion cracking in the primary loop).

The material used for buttering nozzles at the stainless-to-carbon steel safe ends is a high nickel alloy. The nickel-chromium-iron alloy selected and qualified for this application is not susceptible to primary water stress corrosion cracking.

Auxiliary Stainless Steel Piping

The discussion above regarding the necessary conditions for primary loop piping stress corrosion cracking is also applicable to the other stainless steel piping of the primary system.

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Series 300 stainless steel materials have been chosen for the AP1000 due to their proven operating experience. These materials have operated in low-oxygen or no-oxygen environments with no incidents for a number of years. The requirements of Regulatory Guide 1.44, with the exceptions as stated in Subsection 1.9.1, will be used to maintain the experiences of the PWR applications for the use of Series 300 stainless steel materials.

Design tensile stresses in the other stainless steel piping are within the ASME Code, Section III allowables. Residual tensile stresses are expected in the welds; however, the residual stresses should not be more severe than for the operating Westinghouse pressurized water reactor plants (which have not experienced stress corrosion cracking in the auxiliary stainless steel piping).

Main Steam Line

The main steam piping is constructed from ferritic steel. Stress corrosion cracking in ferritic steels commonly result from a caustic environment. A source of a caustic environment in the main steam piping would be moisture carryover from the steam generator. However, the secondary side water treatment utilizes all volatile treatment. All volatile treatment effectively precludes causticity in the steam generator bulk liquid environment. For some operating plants prior to implementing all volatile treatment, the phosphate water treatment caused a caustic chemical imbalance resulting in stress corrosion cracking of steam generator tubing. Under all volatile treatment water treatment conditions, there is no instance of caustic stress corrosion cracking on the ferritic steam lines indicating no significant caustic carryover. The operating secondary side chemistry precludes stress corrosion cracking on the ferritic main steam line.

Based on the above discussion, stress corrosion cracking does not have an adverse effect on the integrity of AP1000 leak-before-break piping systems.

3B.2.3 Water Hammer

Primary Loop Piping

The reactor coolant loop is designed to operate at a pressure greater than the saturation pressure of the coolant, thus precluding the voiding conditions necessary for water hammer to occur. The reactor coolant primary system is designed for Level A, B, C, and D (normal, upset, emergency, and faulted) service condition transients. The design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening have been considered in the system design. Other valve and pump actuations cause relatively slow transients with no significant effect on the system dynamic loads.

To provide dynamic system stability, reactor coolant parameters are controlled. Temperature during normal operation is maintained within a narrow range by control rod positioning. Pressure is controlled within a narrow range for steady-state conditions by pressurizer heaters and pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle. The operating transients of the reactor coolant system primary loop piping are such that significant water hammer loads are not expected to occur.

Auxiliary Stainless Steel Piping

The passive core cooling system and automatic depressurization system are designed to minimize the potential for water hammer induced dynamic loads. Design features include:

- Continuously sloping core makeup tank and passive residual heat exchanger inlet lines to eliminate local high points

- Inlet diffusers in the core makeup tanks to preclude adverse steam and water interactions

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- Vacuum breakers in the discharge lines of the automatic depressurization valves connected to the pressurizer

The AP1000 pressurizer spray control valve is similar to what is used in the operating plants. There is no history of water hammer caused by the spray control valve.

The normal residual heat removal system isolation valves are slow closing valves, identical to operating plants, and therefore would not be a source of water hammer.

These features minimize the potential of water hammer in the auxiliary stainless steel piping system.

Main Steam Line

The steam lines are not subject to water hammer by the nature of the fluid transported. The following system design provisions address concerns regarding steam hammer within the main steam line and identify the significant dynamic loads included in the main steam piping design.

- Design features that prevent water slug formations are included in the system design and layout. In the main steam system, these include the use of drain pots and the proper sloping of lines.

- The operating and maintenance procedures that protect against a potential occurrence of steam hammer include system operating procedures that provide for slowly heating up (to avoid condensate formation from hotter steam on colder surfaces), operating procedures that caution against fast closing of the main steam isolation valves except when necessary, and operating and maintenance procedures that emphasize proper draining.

- The stress analyses for the safety-related portion of the main steam system piping and components include the dynamic loads from rapid valve actuations, including actuation of the main steam isolation valves and the safety valves.

Based on the above discussion, water hammer does not have an adverse effect on the integrity of AP1000 leak-before-break piping systems.

3B.2.4 Fatigue

Low-Cycle Fatigue

Low-cycle fatigue due to normal operation and anticipated transients is accounted for in the design of the piping system. The Class 1 piping systems comply with the fatigue usage requirements of the ASME Code, Section III. The Class 2 and 3 piping systems comply with the stress range reduction factors of the ASME Code, Section III.

Due to the nature of operating parameters, main steam line piping (Class 2) and the Class 3 portion of the accumulator piping, are not subjected to any significant transients to cause low-cycle fatigue.

Based on the above discussion, low-cycle fatigue is not a concern of AP1000 leak-before-break piping systems.

High-Cycle Fatigue

High-cycle fatigue loads in the system result primarily from pump vibrations. The steam generator is designed so that flow-induced vibrations in the tubes are avoided (see Subsection 5.4.2). The loads from reactor coolant pump vibrations are minimized by criteria for pump shaft vibrations during hot functional testing and operation. During operation, an alarm signals when the reactor coolant pump vibration is greater than the limits.

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With these precautions taken, the likelihood of leakage due to fatigue in piping systems evaluated for leak-before-break is very small.

3B.2.5 Thermal Aging

Stainless Steel Piping

Piping used in the reactor coolant loop and other auxiliary lines are wrought stainless steel materials, rather than cast materials, so that thermal aging concerns are not expected for the AP1000 piping and fittings. The welds used in the assembly of the AP1000 are gas tungsten arc welds (GTAW). These welds are essentially as resistant to the effects of thermal aging as the base metal materials. This is due to the typically low ferrite content in welds which results in minimal impact from thermal aging. Based on this information, thermal aging of weld materials and piping used in the AP1000 is not an issue.

Main Steam Lines

The main steam piping system does not have cast materials. The welding process used on these lines is also gas tungsten arc weld (GTAW).

There are no thermal aging concerns for the carbon steel piping of the main steam line and the alloy steel of the main feedwater piping.

The material used for the main steam piping system is not susceptible to dynamic strain aging effects.

3B.2.6 Thermal Stratification

Leak-before-break analyses include consideration of the loads and stresses due to thermal stratification.

Thermal stratification occurs only in a pipe that has a susceptible geometry and low flow velocities. A temperature difference between the flowing fluid and stagnant fluid is also a prerequisite.

The design of piping and component nozzles in the AP1000 includes provisions to minimize the potential for and the effects of thermal stratification, cycling, and striping, pursuant to actions requested in several NRC bulletins, as discussed below.

Primary Loop Piping

Thermal stratification in the reactor coolant loops resulting from actuation of passive safety features is evaluated as a design transient. Stratification effects due to both Level B and Level D service conditions are considered. The criteria used in the evaluation of the stress in the loop piping due to stratification is the same as that applicable for other Level B and Level D service conditions.

Auxiliary Stainless Steel Piping

Pursuant to the actions requested in NRC Bulletin 88-11, the pressurizer surge line is analyzed to demonstrate that the applicable requirements of the ASME Code, Section III are met. This analysis includes consideration of plant operation, thermal stratification, and thermal striping using temperature distributions and transients developed from experience on existing plant monitoring programs.

Pursuant to the actions requested in NRC Bulletin 88-08 (cracking in piping connected to reactor coolant systems due to isolation valve leakage), a systems review of the AP1000 piping was performed in accordance with the criteria provided in Subsection 3.9.3.1.2.

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The unisolable sections of the following lines which are evaluated for leak-before-break have been reviewed and are not susceptible to adverse stresses as described in NRC Bulletin 88-08:

Passive residual heat removal (PRHR) line from the hot leg, through the passive residual heat removal heat exchanger, and to the steam generator channel head
The potential for leakage through the isolation valves is not a concern for the piping extending from the reactor coolant system hot leg connection to the passive residual heat removal heat exchanger inlet, since hot leakage from the reactor coolant system would be entering a hot section of piping. Leakage exiting the passive residual heat removal heat exchanger would not be a concern since the cooled leakage would be entering a cold section of piping. This leakage would then heat up in the piping directly below the steam generator. Any amount of leakage is expected to be small, since the pressure differential across the isolation valves is about 50 psi (the difference between the hot leg and reactor coolant pump suction pressures). Activation of the passive residual heat removal system following a plant scram is not a concern, since stratification will not occur due to the high flow velocity in the passive residual heat removal return flow line.

Automatic depressurization stage 4 lines from the hot legs to the stage 4 depressurization valves
Leakage is not a concern since the squib valves are leaktight and other potential leakage flow paths have double isolation.

Pressurizer safety line from the pressurizer to the safety valve
This line is steam filled and will not experience stratified loadings.

Automatic depressurization stage 2 and 3 lines from the pressurizer to the depressurization valves
Leakage is not a concern since double isolation exists in all potential leakage flow paths.

Normal residual heat removal suction lines from the hot legs to the isolation valves
Thermal stratification in the normal residual heat removal suction lines, including leakage through the isolation valves, is considered in the ASME pipe stress and fatigue analysis of these lines.

Direct vessel injection lines
Thermal stratification in the direct vessel injection lines, including leakage through the isolation valves, is considered in the ASME Code pipe stress and fatigue analysis of these lines.

Main Steam Line

The steam lines are not subjected to thermal stratification by the nature of fluid transported.

Based on the above discussion, thermal stratification does not have an adverse effect on the integrity of AP1000 leak-before-break piping systems.

3B.2.7 Other Mechanisms

The pipe evaluated for leak-before-break does not operate at temperature for which creep fatigue must be considered. Creep fatigue is a concern for ferritic steel piping operation at temperatures above 700°F and for austenitic stainless steel operation above 800°F.

Pipe degradation or failure by indirect causes such as fires, missiles, and component support failures is precluded by criteria for design, fabrication, inspection, and separation of potential hazards in the vicinity of the safety-related piping. The structures, larger pipe, and components in the vicinity of pipe evaluated for leak-before-break are safety-related and seismically designed or are seismically supported if nonsafety-related.

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Cleavage type failures are not a concern for systems operating temperature and material used in the stainless steel piping systems. The material used in the main steam line is highly ductile and resistant to cleavage type failure at operating temperatures. The resistance to failure have been demonstrated by material fracture toughness tests.

3B.3 Leak-before-Break Bounding Analysis

The methodology used for performing the bounding analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2).

Bounding leak-before-break analysis for the applicable AP1000 piping systems is performed. The analysis criteria and development techniques of the bounding analysis curves (BAC) are described below. The bounding analysis curve allows for the evaluation of the piping system in advance of the final piping analysis, incorporating leak-before-break considerations early in the piping design process. The leak-before-break bounding analysis curve is used to evaluate critical points in the piping system. A minimum of two points are required to develop the bounding analysis curve. One point for the low normal stress case and the other point for the high normal stress case. If variations in pipe size, material, pressure or temperature occur for a specific piping system, an additional bounding analysis curve is generated. These points meet the following margins for leak-before-break analysis: (References 1 and 2).

- Margin of 10 on leak detection capability
- Margin of 2 on flaw size
- Establish margin of 1 on load by using absolute combination method of maximum loads

The calculations to establish the bounding analysis curves use minimum values for wall thickness at the weld counterbore and ASME Code material properties. For the main steam line lower bound material property values determined from tests of the material are used. The use of the minimum values bounds the results of larger values. Since the piping is designed and analyzed using ASME Code minimum material properties, these are used conservatively in a consistent manner for evaluation of leak-before-break evaluations. The as-built material properties are expected to be higher than the ASME Code minimum properties. Using minimum thickness instead of a nominal thickness is conservative for the stability analysis and was also used for leak-before-break in operating plants. The use of one thickness (either nominal or minimum) for both leak rate and stability calculation gives comparable overall margins for typical plant loads. The bounding analysis curves are established using the axial load from internal pressure and neglecting other axial loads. This is an appropriate approximation because experience with leak-before-break calculations has shown that the axial load due to pressure is the dominant axial load.

3B.3.1 Procedure for Stainless Steel Piping

3B.3.1.1 Pipe Geometry, Material and Operating Conditions

The following information is identified for each of the lines:

- Piping materials - 316LN/304L, Type 304L is used for the accumulator discharge line
- Normal operating temperature
- Normal operating pressure
- Pipe outside diameter
- Pipe thickness

The number of bounding analysis curves needed for each analyzable piping system is determined by a review of the combinations of the following parameters:

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- Pipe size
- Pipe schedule
- Operating pressures (100 percent power and maximum stress condition)
- Operating temperatures (100 percent power and maximum stress condition)

3B.3.1.2 Pipe Physical Properties

The physical and metallurgical properties for each of the lines are determined in the following manner

- Minimum wall thickness is calculated at the weld counterbore
- The area (A) and section modulus (Z) are calculated using minimum wall thickness
- The yield strength is the ASME Code, Section II (Reference 5) minimum value, at temperature of interest
- The ultimate strength is the ASME Code, Section II (Reference 5) minimum value, at temperature of interest
- The modulus of elasticity is the ASME Code, Section II (Reference 5) at temperature of interest

3B.3.1.3 Low Normal Stress Case (Case 1)

To determine the first point of the bounding analysis curve the following steps are used.

- Calculate axial force Fp (for normal operating pressure)
- Assume a lower magnitude of bending stress. The magnitude selected is a very small number that is lower than the expected minimum bending stress.
- Calculate bending moment = (bending stress) x (section modulus)
- Calculate the leakage flaw size at 100 percent power condition for 10 times the leak detection capability (for 0.5 gpm leak detection capability, this is 10 x 0.5 = 5 gpm)
- Perform the stability analysis using the limit load methodology to obtain the critical flaw size. For AP1000 piping systems, there is no cast material and the weld process is gas tungsten arc welds (Z factor is 1.0 since weld process is gas tungsten arc welds, Reference 1.)
  - Determine the maximum loads for a critical flaw size of twice the leakage flaw size. The margin of 2 on flaw size is satisfied.
- Calculate the low normal stress and corresponding maximum stress by using:

$$\text{Stress} = \frac{\text{Axial Force}}{\text{Area}} + \frac{\text{Bending Moment}}{\text{Section Modulus}} \qquad \text{(3B-1)}$$

3B.3.1.4 High Normal Stress Case (Case 2)

To determine the other endpoint of the bounding analysis curve the following steps are used.

- Axial force Fp is calculated as above for normal operating pressure

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- Assume a higher magnitude of bending stress to get higher bending moment. The magnitude of bending is selected such that the corresponding maximum stress generated is close to the flow stress.

- Calculate bending moment = (bending stress) x (section modulus)

- Repeat leakage flaw size and stability calculations as outlined for the low normal stress case above

Note: For an intermediate point, calculation steps are the same as low normal or the high normal case.

3B.3.1.5 Develop the Bounding Analysis Curve

- For Case 1, normal and maximum stresses are established.

- For Case 2, normal and maximum stresses are established.

- Plot these two points with normal versus maximum stress. The curve is generated by joining these two points in a straight line. More than two points may be used if desired, to obtain a smooth curve fit between the calculated points. A typical curve is shown in Figure 3B-1.

3B.3.2 Procedure for Non-stainless Steel Piping

The procedure to develop the bounding analysis curve for the carbon steel for main steam lines is similar to that for the stainless steel and is described below.

3B.3.2.1 Pipe Geometry, Material and Operating Conditions

The following information is identified for each of the lines:

- Piping materials
- Normal operating temperature
- Normal operating pressure
- Pipe outside diameter
- Piping thickness

The number of bounding analysis curves needed for each analyzable piping system is determined by a review of the combinations of the following parameters:

  – Pipe size
  – Pipe schedule
  – Operating pressures (100 percent power and maximum stress condition)
  – Operating temperatures (100 percent power and maximum stress condition)

3B.3.2.2 Calculations Steps

- The minimum wall thickness is calculated at the weld counterbore

- The area (A) and section modulus (Z) are calculated using minimum wall thickness

- The material yield strength, ultimate strength, modulus of elasticity, stress-strain curves, and J-R curves are determined from the material tests

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3B.3.2.3 Low Normal Stress Case (Case 1)

To determine the first point of the bounding analysis curve the following steps are used.

- Calculate axial force Fp (for normal operating pressure)
- Assume a lower magnitude of bending stress
- Calculate bending moment = (bending stress) x (section modulus)
- Calculate the leakage flaw size at 100 percent power condition for 10 times the leak detection capability (for 0.5 gpm leak detection capability, this is 10 x 0.5 = 5 gpm)
- Stability analysis
  - Perform J-integral analysis
  - Determine the maximum loads for a critical flaw size of twice the leakage flaw size by satisfying the stability criteria. The margin of 2 on flaw size is satisfied.
- Stability criteria
  - Japplied ≤ JIC
  - If Japplied > JIC, then Japplied < Jmax and Tapplied < Tmat
- Calculate the low normal stress and corresponding maximum stress by using:

$$\text{Stress} = \frac{\text{Axial Force}}{\text{Area}} + \frac{\text{Bending Moment}}{\text{Section Modulus}}$$

3B.3.2.4 High Normal Stress Case (Case 2)

To determine the other endpoint of the bounding analysis curve the following steps are used.

- Axial force Fp is calculated above (for normal operating pressure)
- Assume a higher magnitude of bending stress to get higher bending moment
- Calculate bending moment = (bending stress) x (section modulus)
- Repeat leakage flaw size and stability calculations as outlined for the low normal stress case above

Note: For an intermediate point, calculation steps are the same as low normal or the high normal case.

3B.3.2.5 Develop the Bounding Analysis Curve

Follow steps as outlined for the stainless steel case in Subsection 3B.3.1.5.

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3B.3.3 Evaluation of Piping System Using Bounding Analysis Curves

To evaluate the applicability of leak-before-break, the results of the pipe stress analysis are compared to the bounding analysis curve. The critical location is the location of highest maximum stress as determined by the pipe stress results. A comparison is made with the applicable bounding analysis curves for the analyzable piping systems. As outlined in 3B.3.1.1 and 3B.3.2.1, bounding analysis curves are calculated for different combinations of pipe size, pipe schedule, operating pressures, operating temperatures.

The bounding analysis curves are used during the layout and design of the piping systems to provide a design that satisfies leak-before-break criteria. In addition, the results of the as-built piping analysis reconciliation to the bounding analysis curves to verify that the fabricated piping systems satisfy leak-before-break criteria. See Subsection 3.6.4 for the Combined License information item associated with this verification.

At the critical location, the load combination for the maximum stress calculation uses the absolute sum method. The load combination is as follows:

(1) |Pressure| + |Deadweight| + |Thermal (100% Power)*| + |Safe Shutdown Earthquake|

(2) |Pressure| + |Deadweight| + |Maximum Stratification| (Additional Case for Surge Line)

The normal stress is calculated using the algebraic sum method at critical location and the following load combination.

(1) Pressure + Deadweight + Thermal (100% Power*)

* Includes applicable 100% Power stratification loads.

3B.3.3.1 Calculation of Stresses

The stresses due to axial loads and moments are calculated by the following equation:

where:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad \text{(3B-2)}$$

σ = stress

F = axial load

M = moment

A = cross-sectional area

Z = section modulus

The moments for the desired loading combinations are calculated by the following equation:

$$M = \sqrt{M_X^2 + M_Y^2 + M_Z^2} \qquad \text{(3B-3)}$$

where,

M = moment for required loading

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M_X = torsional moment

M_Y = Y component of bending moment

M_Z = Z component of bending moment

The Y and Z-axes are lateral axes to the X-axis which is the axial axis

The axial load and moments for the normal case and maximum case are computed by the methods shown below.

3B.3.3.2 Normal Loads

The normal operating loads are calculated by the following equations:

$$F = F_{DW} + F_{Th} + F_P \quad (3B\text{-}4)$$

$$M_X = (M_X)_{DW} + (M_X)_{Th} \quad (3B\text{-}5)$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{Th} \quad (3B\text{-}6)$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{Th} \quad (3B\text{-}7)$$

The subscripts of the above equations represent the following load cases:

DW = deadweight

Th = normal thermal expansion (100 percent power, including applicable stratification loads)

P = load due to internal pressure

The method of combining loads is often referred to as the algebraic sum method.

Calculate the normal stress at the critical location.

3B.3.3.3 Maximum Loads

For the maximum case, the absolute summation method of load combination is applied which results in higher magnitude of the combined loads. Since stability is demonstrated using these loads, the leak-before-break margin on loads is satisfied. An example of the absolute summation expressions are shown below:

$$F = |F_{DW}| + |F_{Th}| + |F_P| + |F_{SSEINERTIA}| + |F_{SSEAM}| \quad (3B\text{-}8)$$

$$M_X = |(M_X)_{DW}| + |(M_X)_{Th}| + |(M_X)_{SSEINERTIA}| + |(M_X)_{SSEAM}| \quad (3B\text{-}9)$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{Th}| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \quad (3B\text{-}10)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{Th}| + |(M_Z)_{SSEINERTIA}| + |(M_Z)_{SSEAM}| \quad (3B\text{-}11)$$

where subscripts SSE, Inertia and AM mean safe shutdown earthquake, inertia and anchor motion respectively.

Note: For the surge line, see Subsection 3B.3.3 for additional case.

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3B.3.3.4 Bounding Analysis Curve Comparison – LBB Criteria

To compare the stress results with the bounding analysis curve the following process is followed. The normal and maximum stress at the critical location are calculated by using the loads defined in Subsection 3B.3.3. Plot the normal stress versus maximum stress on the bounding analysis curve for the specified system. If the point is on or below the bounding analysis curve, the leak-before-break analysis and margins are satisfied. If the point falls above the bounding analysis curve, the leak-before-break analysis criteria are not satisfied and the pipe layout or support configuration needs to be revised to meet the leak-before-break bounding analysis. Figure 3B-1 shows a typical bounding analysis curve.

3B.3.4 Bounding Analysis Results

Table 3B-1 shows a summary of piping systems and corresponding bounding analysis figures. Figures 3B-1 to 3B-22 show the bounding analysis curves. The curves satisfy the margins as indicated in Section 3B.3.

3B.4 Differences in Leak-before-Break Analysis for Stainless Steel and Ferritic Steel Pipe

The significant difference between leak-before-break analysis performed for the stainless steel (Class 1 and Class 3) systems and the ferritic steel in the Class 2 systems is in the stability analysis. In the case of stainless steel systems, stability analyses are performed by limit load approach. In the ferritic steel systems, stability analyses are performed by J-integral approach.

3B.5 Differences in Inspection Criteria for Class 1, 2, and 3 Systems

Class 1, 2 and 3 systems are subjected to in-service inspection requirements from ASME Code, Section XI. For Class 1 piping, terminal ends and dissimilar metal welds are volumetrically inspected, along with other locations, to total 25 percent of the welds. For Class 2 piping, the requirement is to volumetrically inspect the terminal ends and other locations to total 7.5 percent of the welds. For Class 3 systems (the only Class 3 piping is in the accumulator line which is always at room temperature), the system receives periodic visual examinations in conjunction with pressure testing. These requirements were developed by ASME Code, Section XI consistent with the different safety classes of these systems.

The leak-before-break evaluations are based on the ability to detect a potential leaking crack; not the ability to find cracks by inservice inspections. The criteria or methods of the leak-before-break evaluations are the same for ASME Code Class 1, 2, and 3.

3B.6 Differences in Fabrication Requirements of ASME Class 1, Class 2, and Class 3 Piping

The significant difference among Class 1, 2 and 3 seamless pipe occurs in the nondestructive examination requirements. The Class 1 seamless pipe examination requirements include an ultrasonic testing examination, whereas Class 2 and 3 do not. In addition, the Class 1 examination requirements for a circumferential butt welded joint include radioagraphic testing and magnetic particle or liquid penetrant examination where Class 2 does not. The examination requirements for Class 2 pipe require radiographic examination of the welds and normally Class 3 pipe does not. As noted in Subsection 3.2.2.5, for Class 3 lines required for emergency core cooling functions, radiography will be conducted on a random sample of welds. The Class 3 leak-before-break lines are included in the lines that are radiographed. In addition see Subsection 3.6.3.2 for augmented inspection of Class 3 leak-before-break lines.

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For the fabrication of welds in the Class 1, Class 2 and Class 3 pipes there is no significant differences.

The differences in fabrication and nondestructive examination requirements do not affect the leak-before-break analyses assumptions, criteria, or methods.

3B.7 Sensitivity Study for the Constraint Effect on LBB

Westinghouse performed a sensitivity study on a 6-inch diameter pipe to demonstrate that the leak-before-break evaluation margins are not significantly affected when constraint effects of pressure induced bending are included. The analysis used a finite element model of a 6-inch diameter pipe welded to a nozzle with a fixed end condition. This conservatively represents the bounding conditions for AP1000 piping. The normal and maximum stresses were used from a representative AP600 6-inch line bounding analysis curve. The material properties for the base metal and TIG weld were considered in the analysis. The stability analysis was performed using the J-integral method. This analysis was developed in consultation with the NRC.

The conclusion of this sensitivity study is that the leak-before-break margins for 6-inch and larger piping on AP1000 are not significantly affected by the constraint effect and application of leak-before-break to such piping is acceptable.

3B.8 References

1. Standard Review Plan 3.6.3, "Leak Before Break Evaluation Procedures," Federal Register, Volume 52, Number 167, Friday, August 28, 1987; Notice (Public Comment Solicited), pp. 32626-32633.

2. NUREG-1061, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volume 3, (prepared by the Pipe Break Task Group), November 1984.

3. "Erosion-Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Program Guidelines," EPRI NP-3944, April 1985.

4. G. Cragnolino, "Erosion-Corrosion in Nuclear Power Systems-An Overview," Corrosion '87, Paper No. 86, March 1987.

5. ASME Boiler and Pressure Vessel Code, Section II, "Materials," 1998 Edition through 2000 Addenda.

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Table 3B-1 (Sheet 1 of 2)
AP1000 Leak-Before-Break Bounding Analysis Systems and Parameters

System	Subsystem	Line No(s).	Nominal Diameter (Inches)	Material	Temp (°F)	Pressure (psig)	Figure No.
RCS	Primary Loop Hot Leg	L001A, B	31 (ID)(1)	SA-376 TP316LN	610.0	2247.0	3B-2
RCS	Primary Loop Cold Leg	L002A, B, C, D	22 (ID)(1)	SA-376 TP316LN	537.2	2323.0	3B-3
SGS	Main Steam Line	L006A, B	38	SA-335 GR P11	523.0	821.0	3B-4
RCS	Normal Residual Heat Removal	L139	20	SA-312 TP316LN	610.0	2247.0	3B-5
RCS	Surge Line	L003	18	SA-312 TP316LN	653.0	2248.0	3B-6 (Sheet 1)
RCS	Surge Line	L003	18	SA-312 TP316LN	653.0(2) 455.0(2)	2248.0(2) 430.0(2)	3B-6 (Sheet 2)
RCS	Passive Residual Heat Removal Supply/ ADS 4	L135A,B; L136A,B	18	SA-312 TP316LN	610.0	2247.0	3B-7
RCS	Passive Removal Heat Removal Supply/ ADS 4	L133A, B; L137A, B; L134	14	SA-312 TP316LN	610.0	2247.0	3B-8
PXS	Passive Residual Heat Removal Supply to Cold Trap and Vent Line	L102, L107	14	SA-312 TP316LN	610.0	2235.0	3B-8
PXS	Passive Residual Heat Removal Supply after Cold Trap to PRHR HX	L102	14	SA-312 TP316LN	120.0	2235.0	3B-9
PXS	Return – PRHR HX to Isolation Valve	L103	14	SA-312 TP316LN	120.0	2235.0	3B-9
RCS	Automatic Depressurization System Stage 2, 3	L004A,B; L006A,B; L020A,B; L030A, B; L131	14	SA-312 TP316LN	653.0	2234.0	3B-10 (Sheets 1 and 2)
PXS	Passive Residual Heat Removal Return – after Isolation Valve	L104A, B; L105	14	SA-312 TP316LN	537.0	2202.0	3B-11

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Table 3B-1 (Sheet 2 of 2)
AP1000 Leak-Before-Break Bounding Analysis Systems and Parameters

System	Subsystem	Line No(s).	Nominal Diameter (Inches)	Material	Temp (°F)	Pressure (psig)	Figure No.
RCS	Passive Residual Heat Removal Return	L113	14	SA-312 TP316LN	537.0	2202.0	3B-11
PXS	Passive Residual Heat Removal Vent Line	L107	12	SA-312 TP316LN	610.0	2248.0	3B-12 (Not Used)
PXS	Accumulator to Isolation Valve	L029A, B	8	SA-312 TP304L	120.0	700.0	3B-13
RCS	Balance Line from Cold Leg to CMT Isolation Valve	L118A, B	8	SA-312 TP316LN	537.0	2323.0	3B-14
PXS	Balance Line from CMT Isolation Valve to CMT	L007A, B; L070A, B	8	SA-312 TP316LN	537.0	2332.0	3B-14
PXS	Direct Vessel Injection Line to RV, IRWST Injection	L021A, B; L125A, B	8	SA-312 TP316LN	537.0	2332.0	3B-14
PXS	Core Makeup Tank (Injection Line, RV Side of Isolation Valve, Core Makeup Tank Side of Isolation Valve), Direct Vessel Injection (Accumulator Connection to Cold Trap), IRWST Injection	L015, L016, L017, L018, L020, L025, L123, L125, L127, L133, L134 (All A, B)	8	SA-312 TP316LN	120.0	2332.0	3B-15
PXS	Direct Vessel Injection Line to RV	L021A, B	8	SA-312-TP316LN	135.0	2332.0	3B-15
RCS	Automatic Depressurization System Stage 2, 3	L021A,B; L031A,B	8	SA-312 TP316LN	653.0	2235.0	3B-16 (Not Used)
PXS	Accumulator after Isolation Valve	L027A, B	8	SA-312 TP304L	120.0	700.0	3B-17
PXS	RNS Discharge	L019A, B	6	SA-312 TP316LN	120.0	2332.0	3B-18
RCS	Automatic Depressurization System Header to RCS Safety Valve	L005A, B	6	SA-312 TP316LN	653.0	2234.0	3B-19 (Sheets 1 and 2)
RCS	Normal Residual Heat Removal	L140	12	SA-312 TP316LN	610.0	2247.0	3B-20
RNS	Normal Residual Heat Removal	L001, L002A, B	10	SA-312 TP316LN	610.0	2235.0	3B-21
RCS	Automatic Depressurization System Stage 2, 3 (Cold Trap)	L021A, B; L031A, B	8	SA-312TP316LN	100.0	2234.0	3B-22

Note:
1. ID = Inside diameter
2. Temperature and Pressure at normal and maximum stress cases as indicated in Figure 3B-6 sheet 2.

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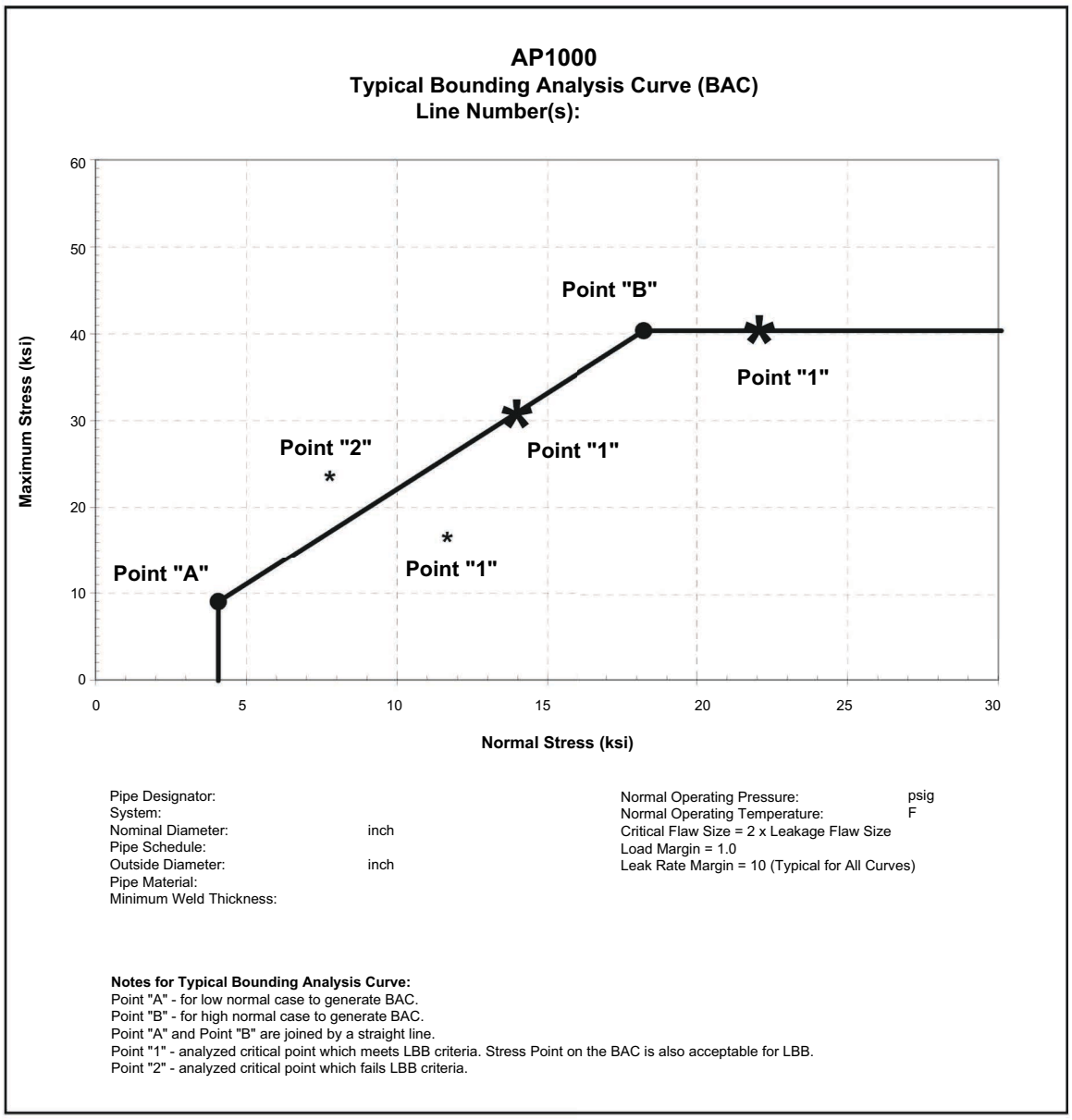

Figure 3B-1
Typical Bounding Analysis Curve (BAC)

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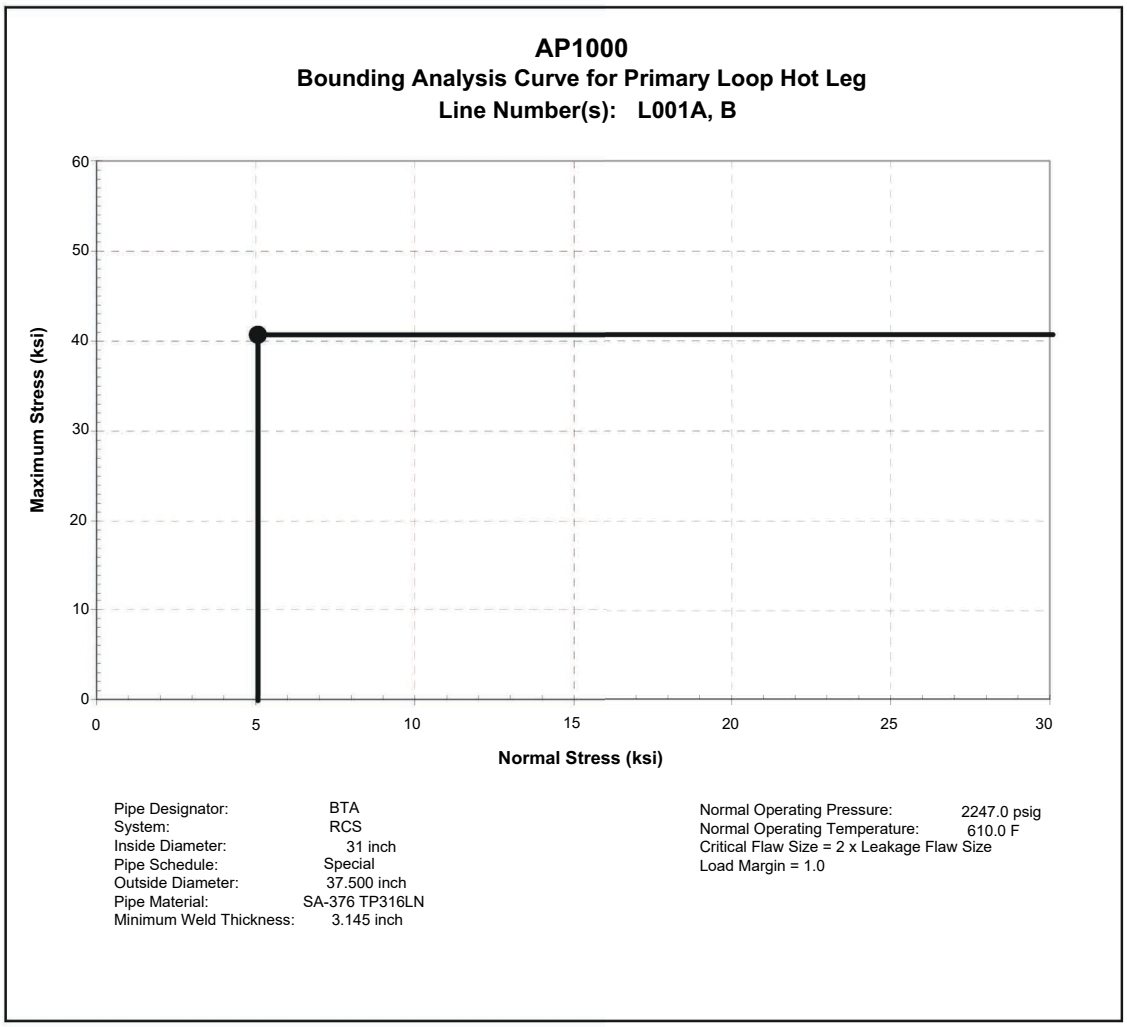

Figure 3B-2
Bounding Analysis Curve for Primary Loop Hot Leg

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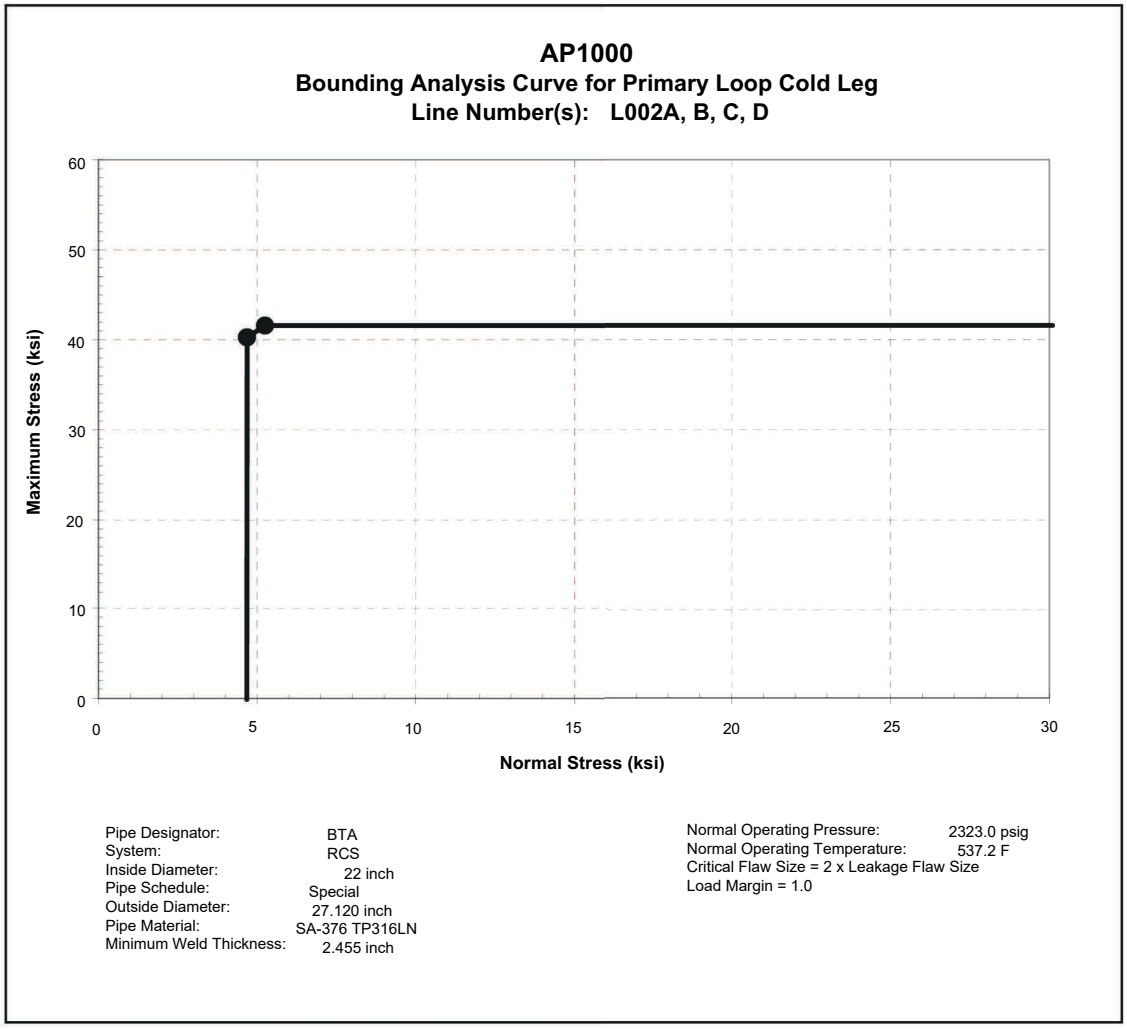

Figure 3B-3
Bounding Analysis Curve for Primary Loop Cold Leg

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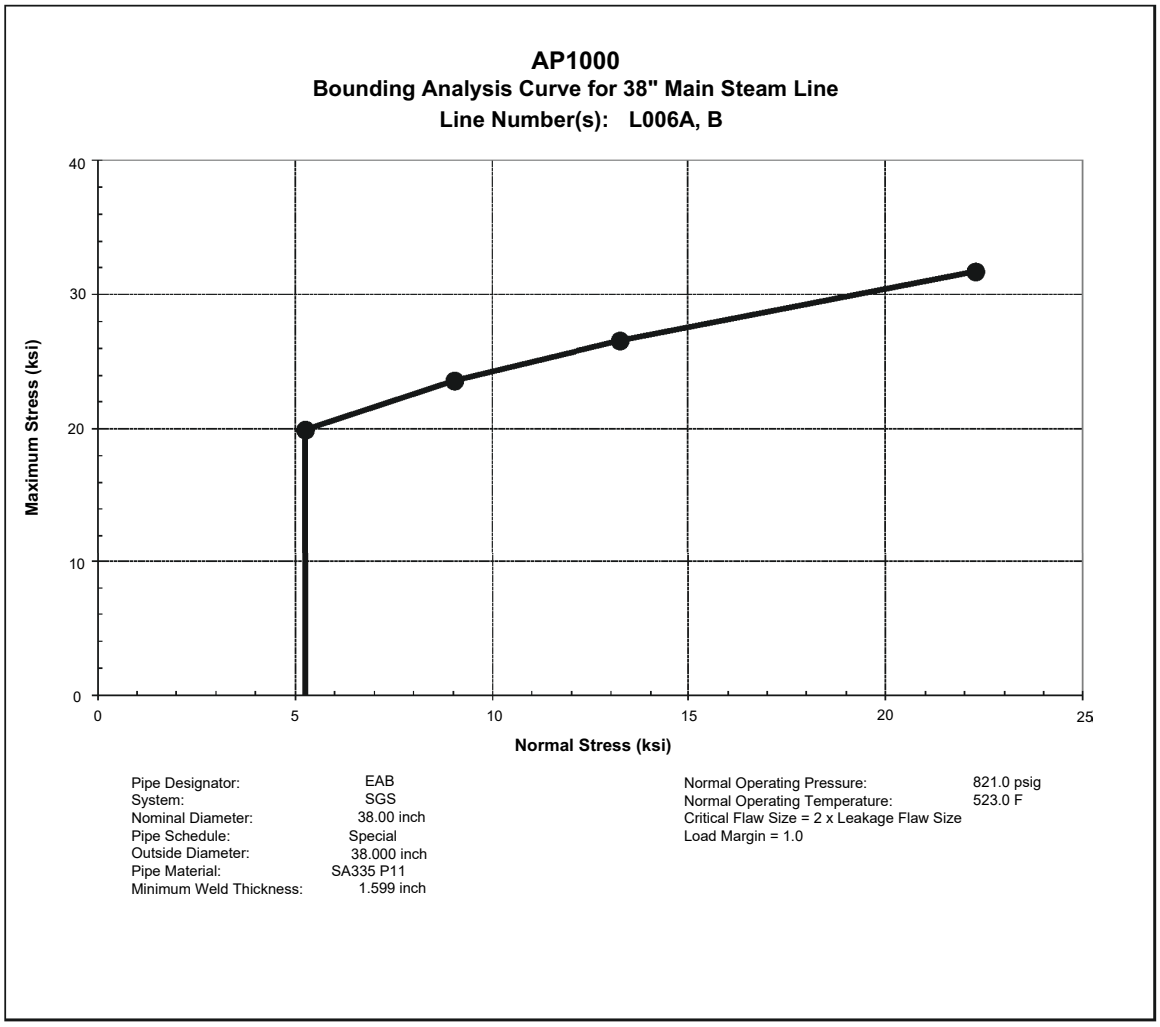

Figure 3B-4
Bounding Analysis Curve for 38″ Main Steam Line

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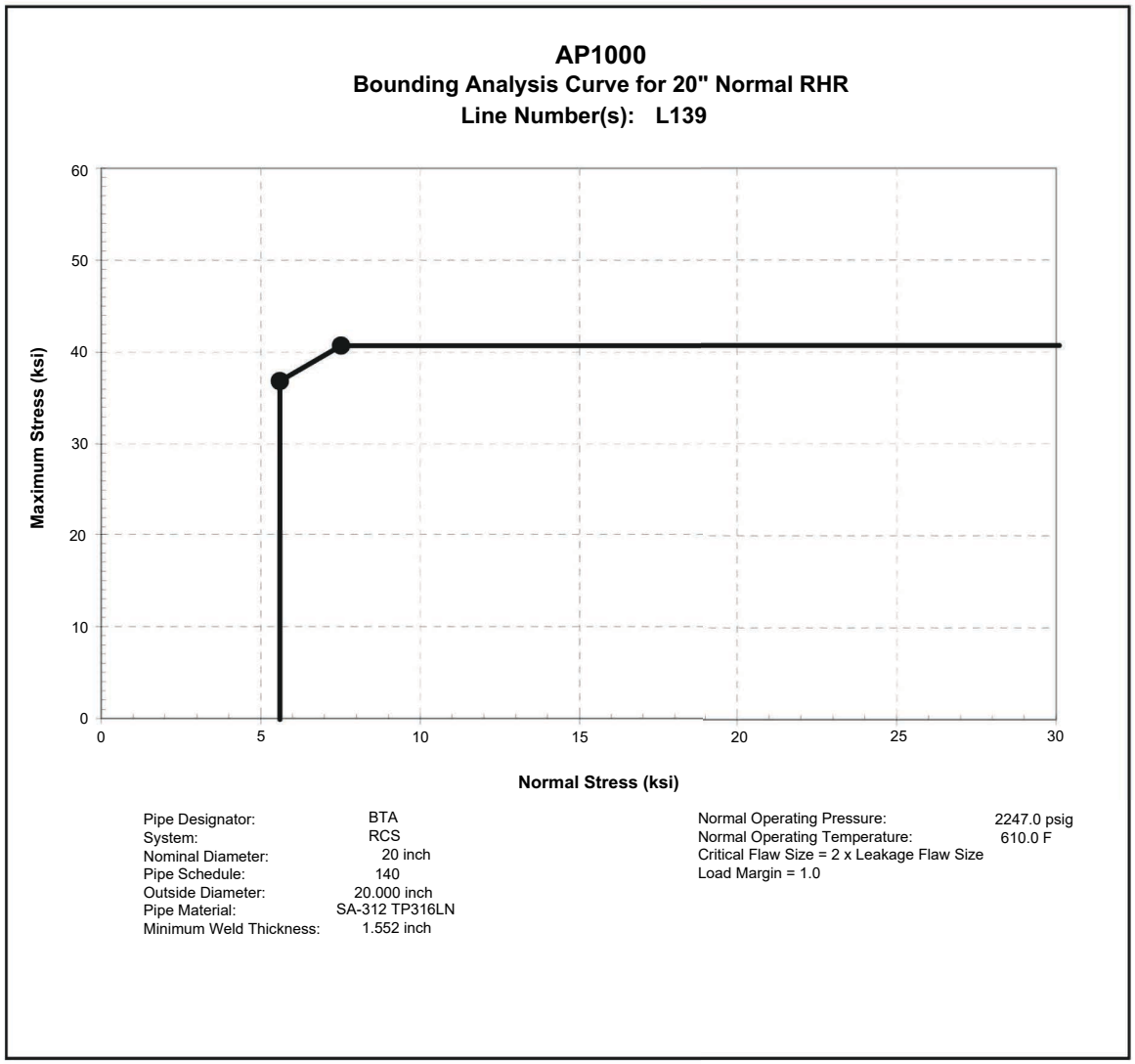

Figure 3B-5
Bounding Analysis Curve for 20″ Normal RHR

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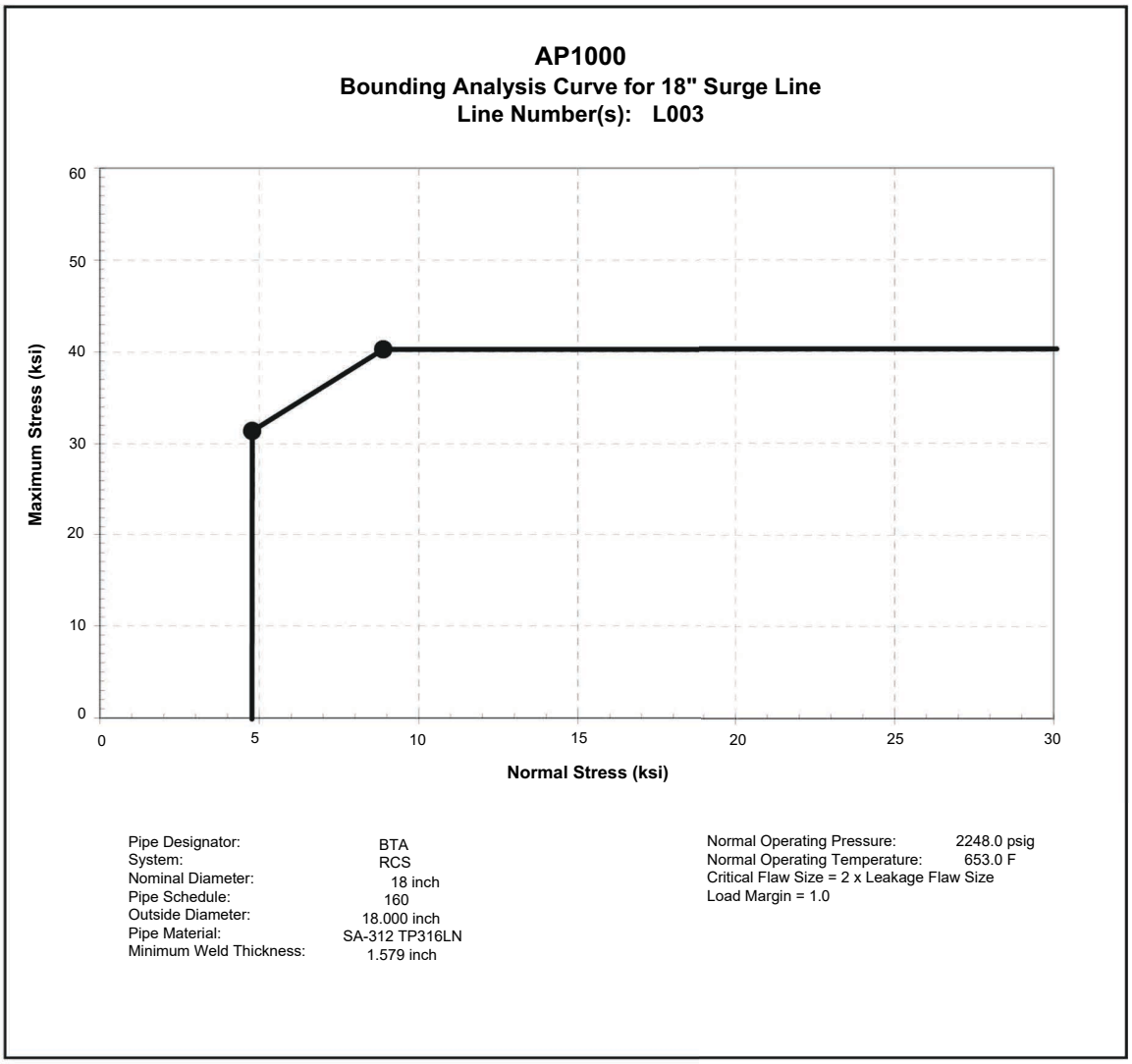

Figure 3B-6 (Sheet 1 of 2)
Bounding Analysis Curve for 18″ Surge Line

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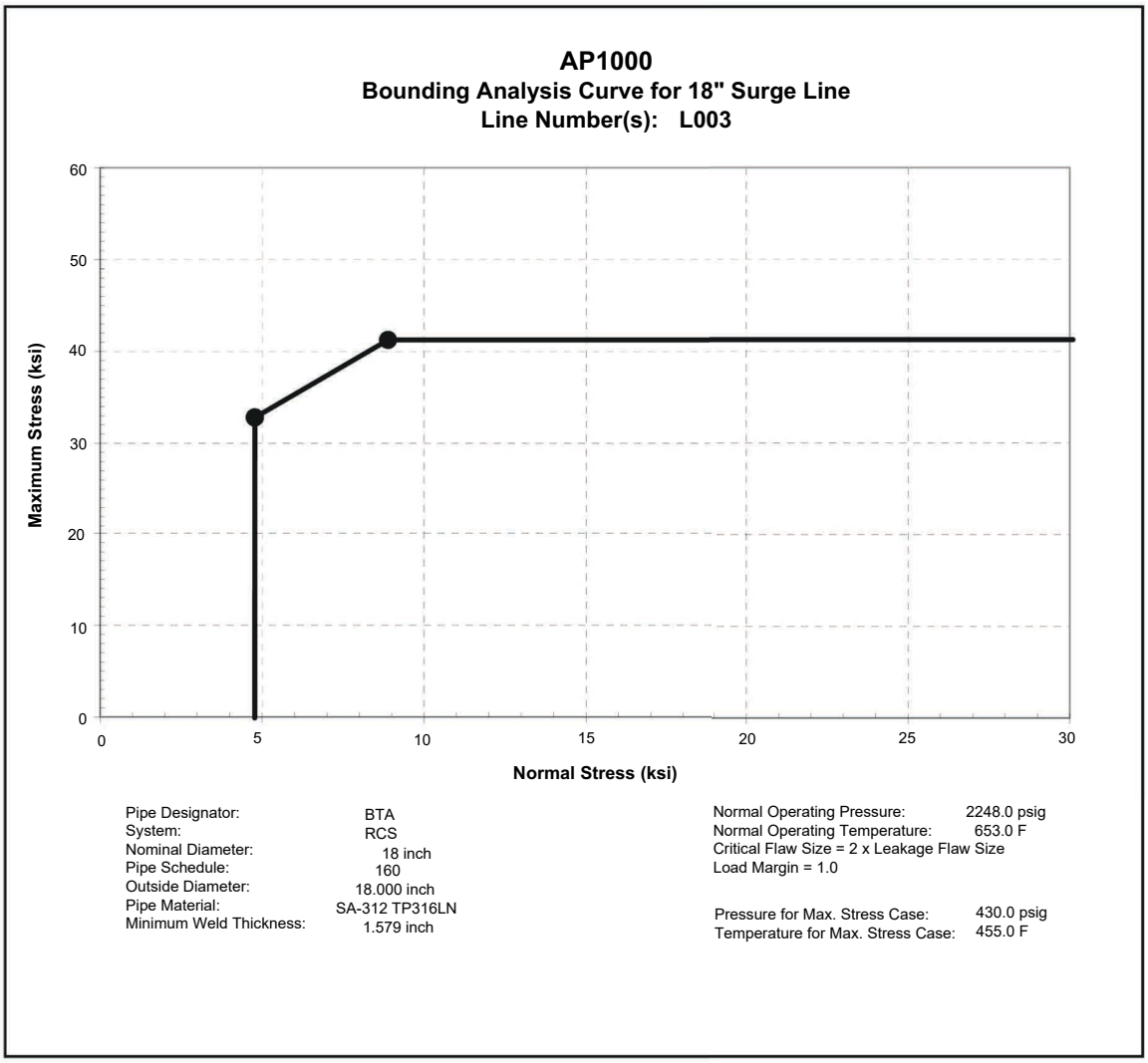

Figure 3B-6 (Sheet 2 of 2)
Bounding Analysis Curve for 18″ Surge Line

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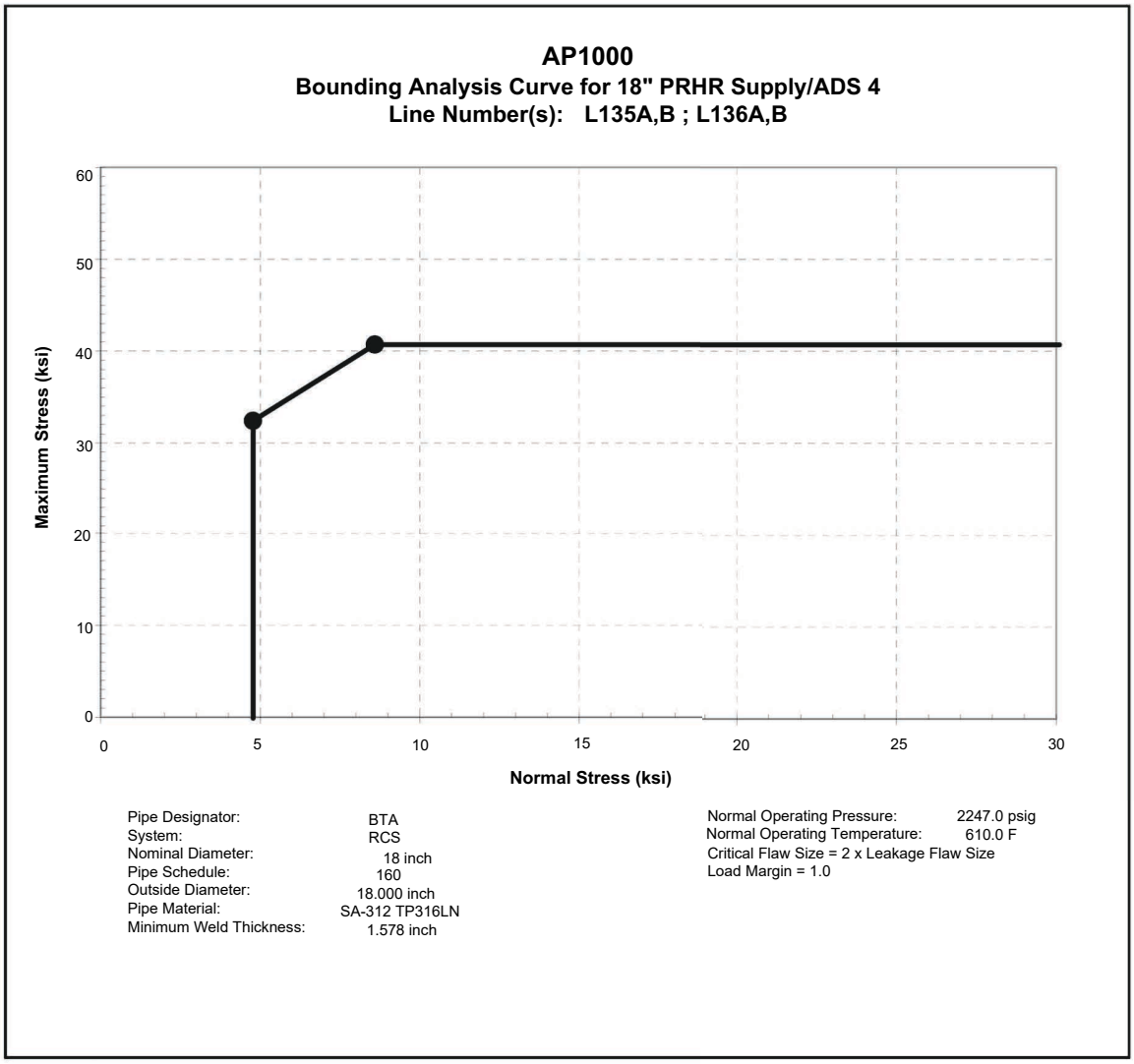

Figure 3B-7
Bounding Analysis Curve for 18″ PRHR Supply/ADS 4

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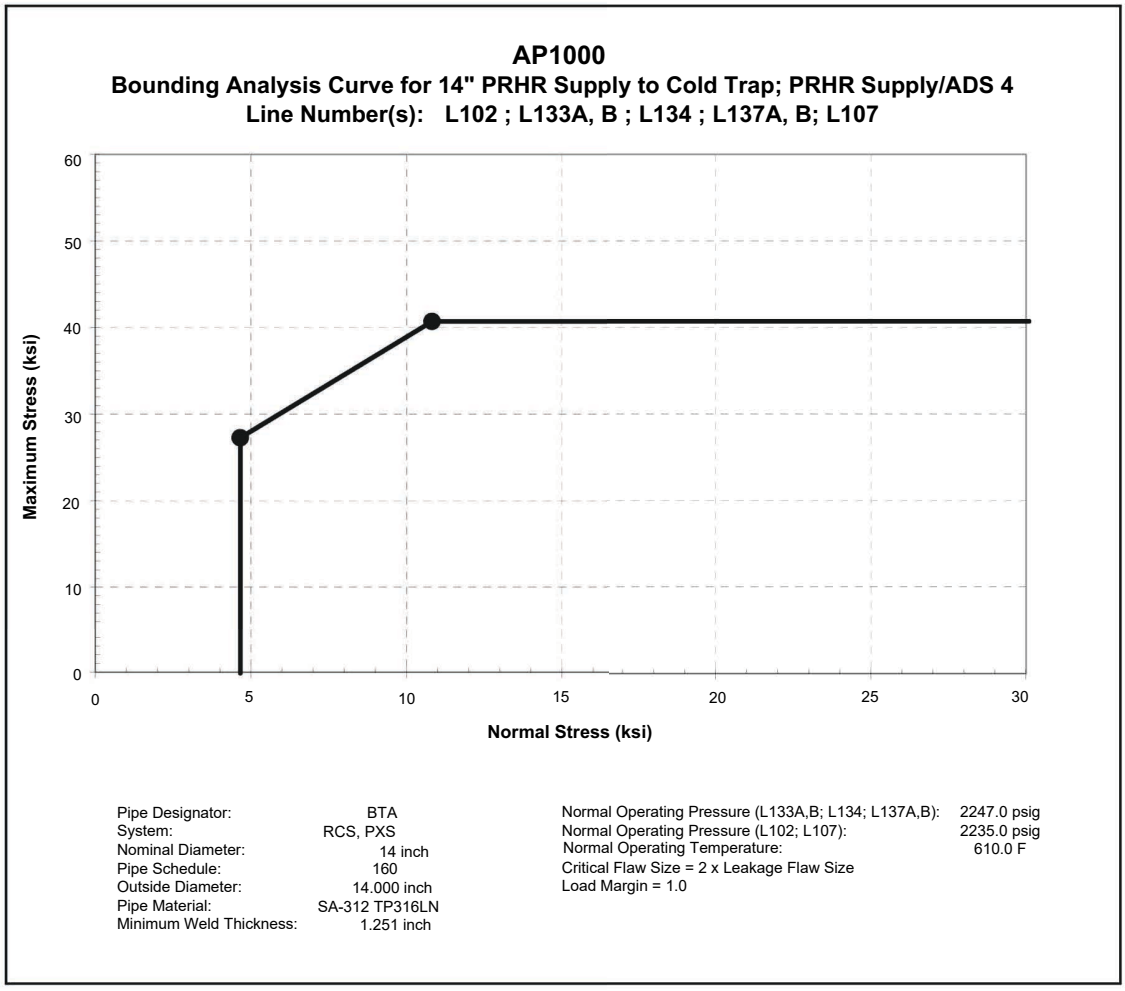

Figure 3B-8
Bounding Analysis Curve for 14″ PRHR Supply to Cold Trap, PRHR Supply/ADS4

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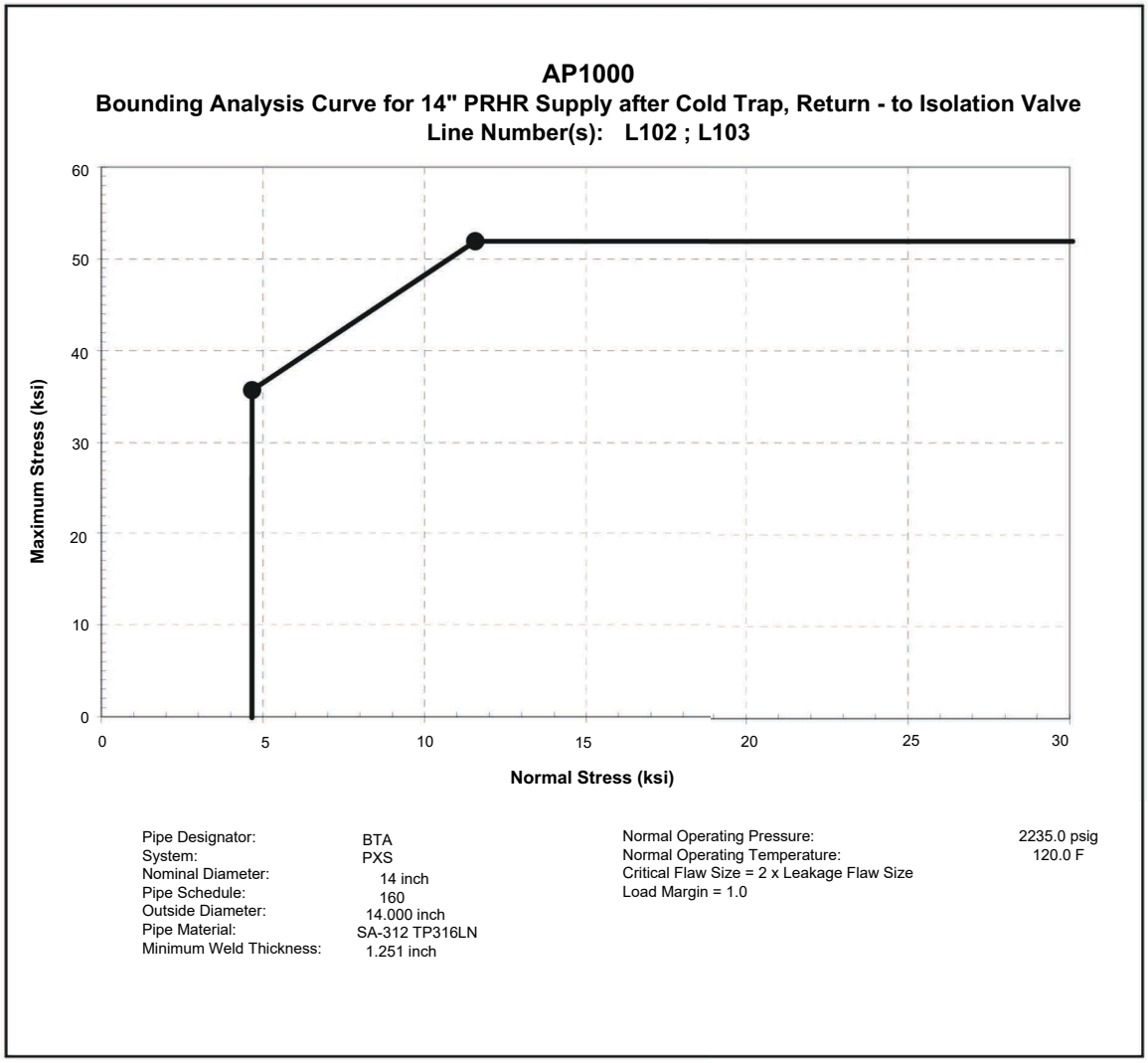

Figure 3B-9
Bounding Analysis Curve for 14″ PRHR Supply after Cold Trap, Return – to Isolation Valve

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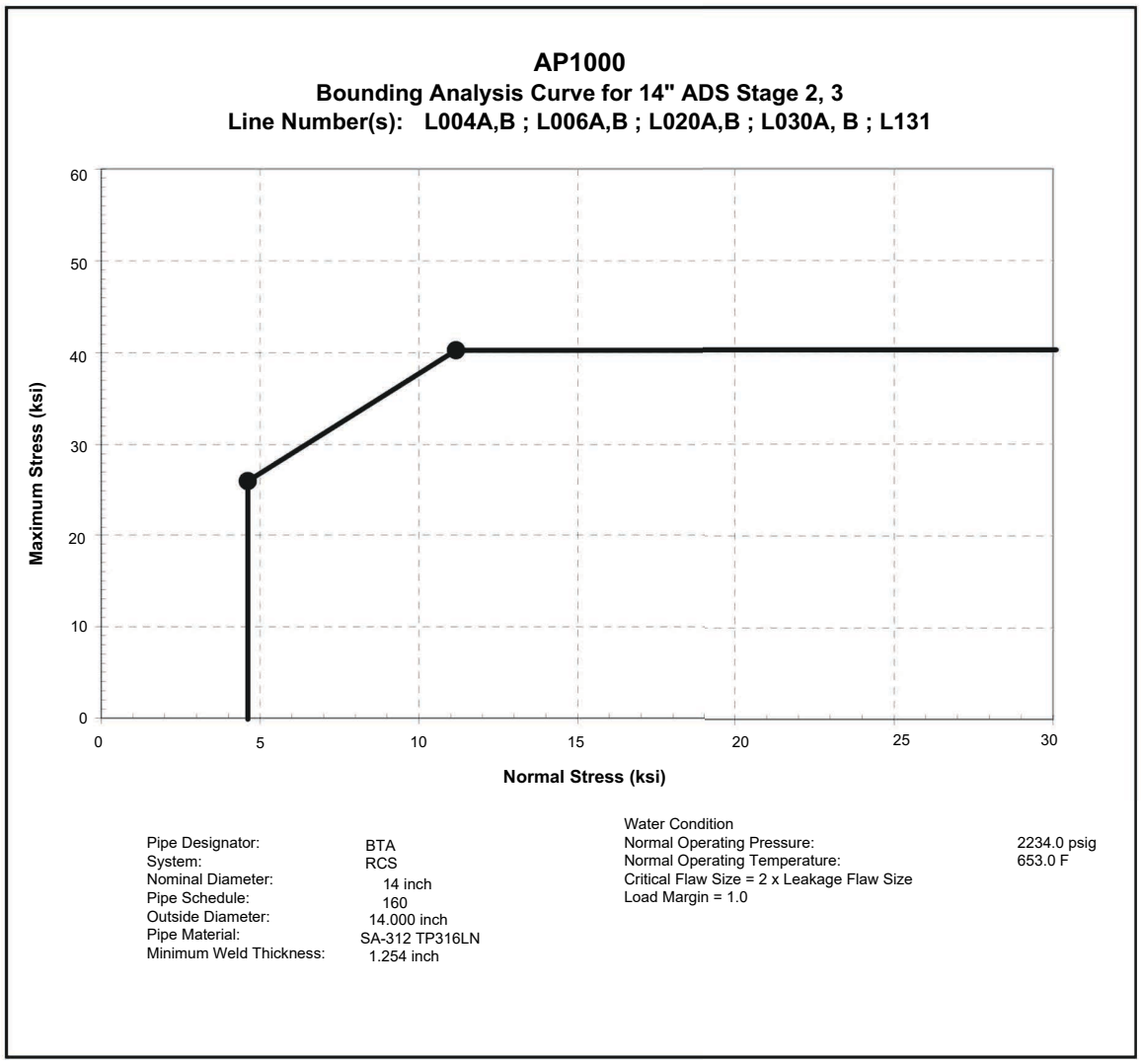

Figure 3B-10 (Sheet 1 of 2)
Bounding Analysis Curve for 14″ ADS Stage 2, 3

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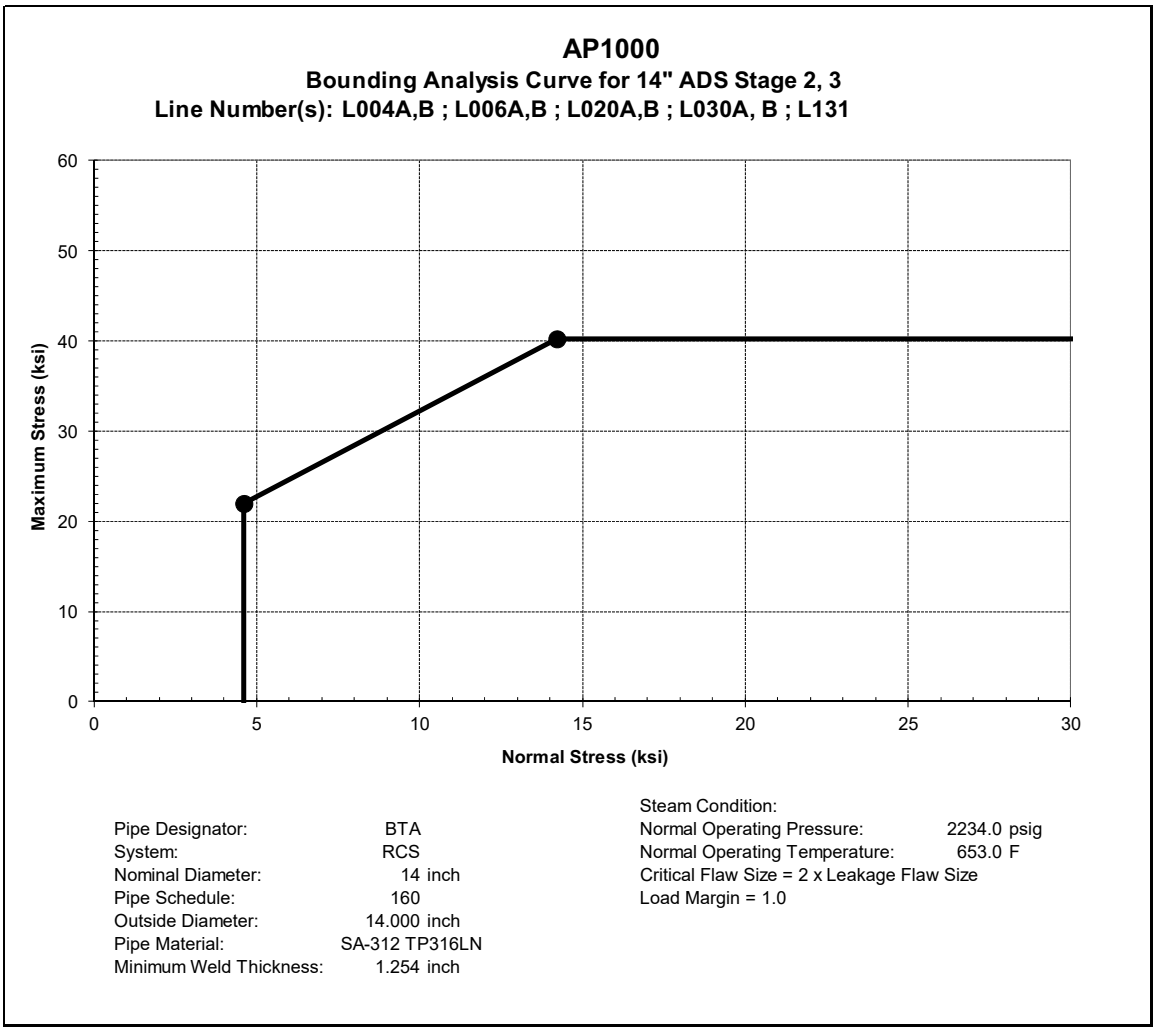

Figure 3B-10 (Sheet 2 of 2)
Bounding Analysis Curve for 14" ADS Stage 2, 3

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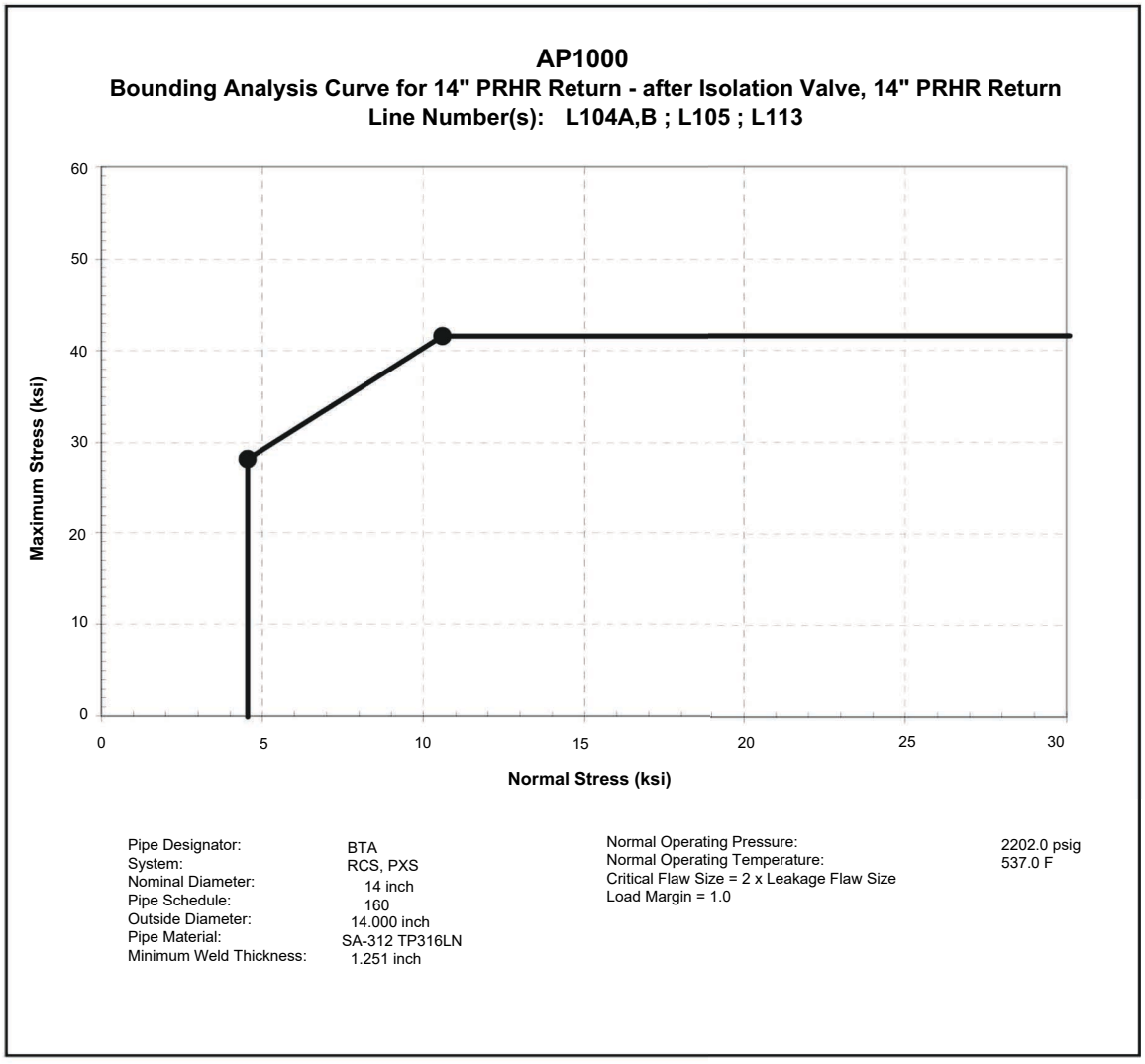

Figure 3B-11
Bounding Analysis Curve for 14″ PRHR Return – after Isolation Valve, 14″ PRHR Return

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Figure 3B-12 Not Used

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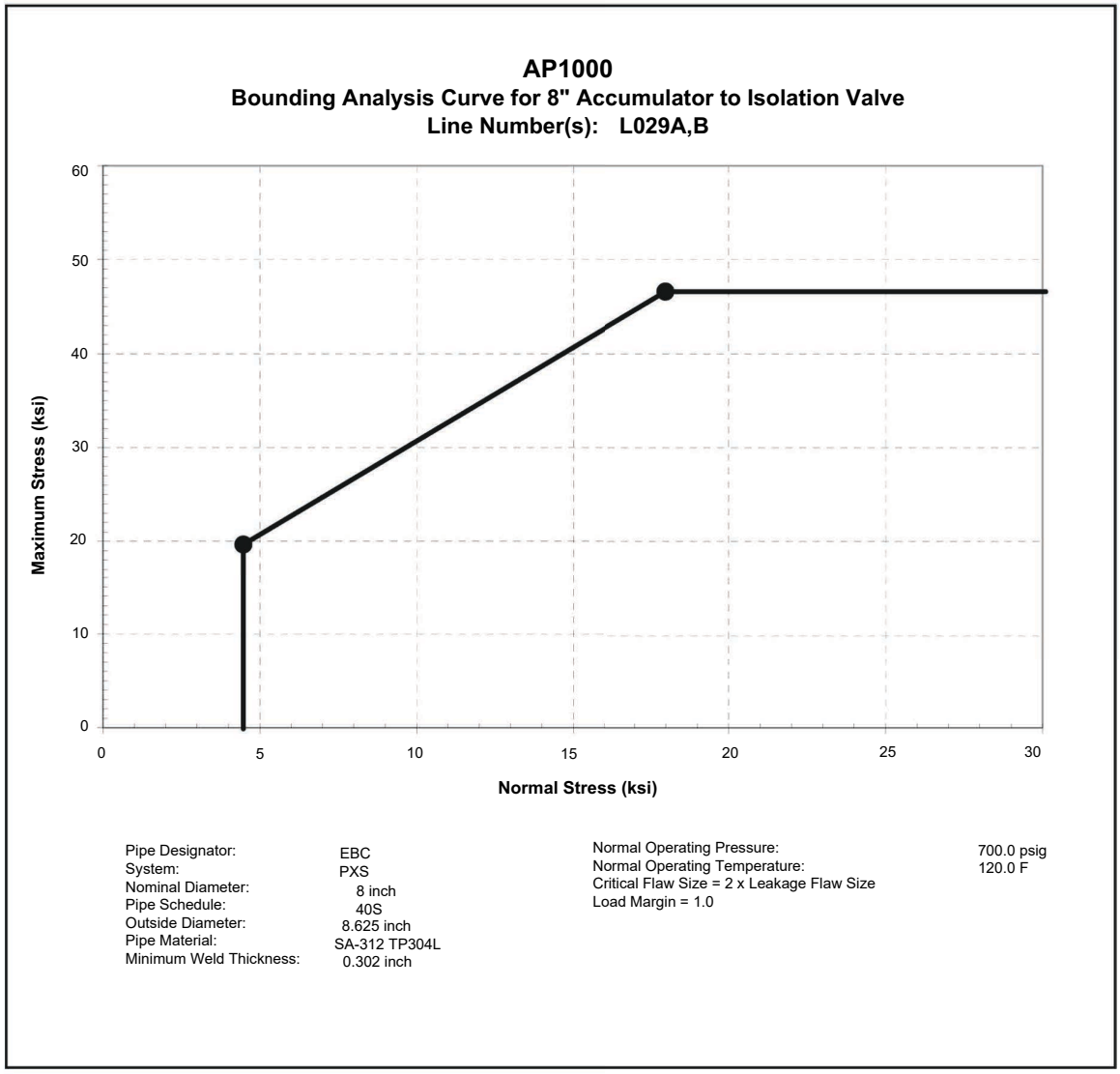

Figure 3B-13
Bounding Analysis Curve for 8″ Accumulator to Isolation Valve

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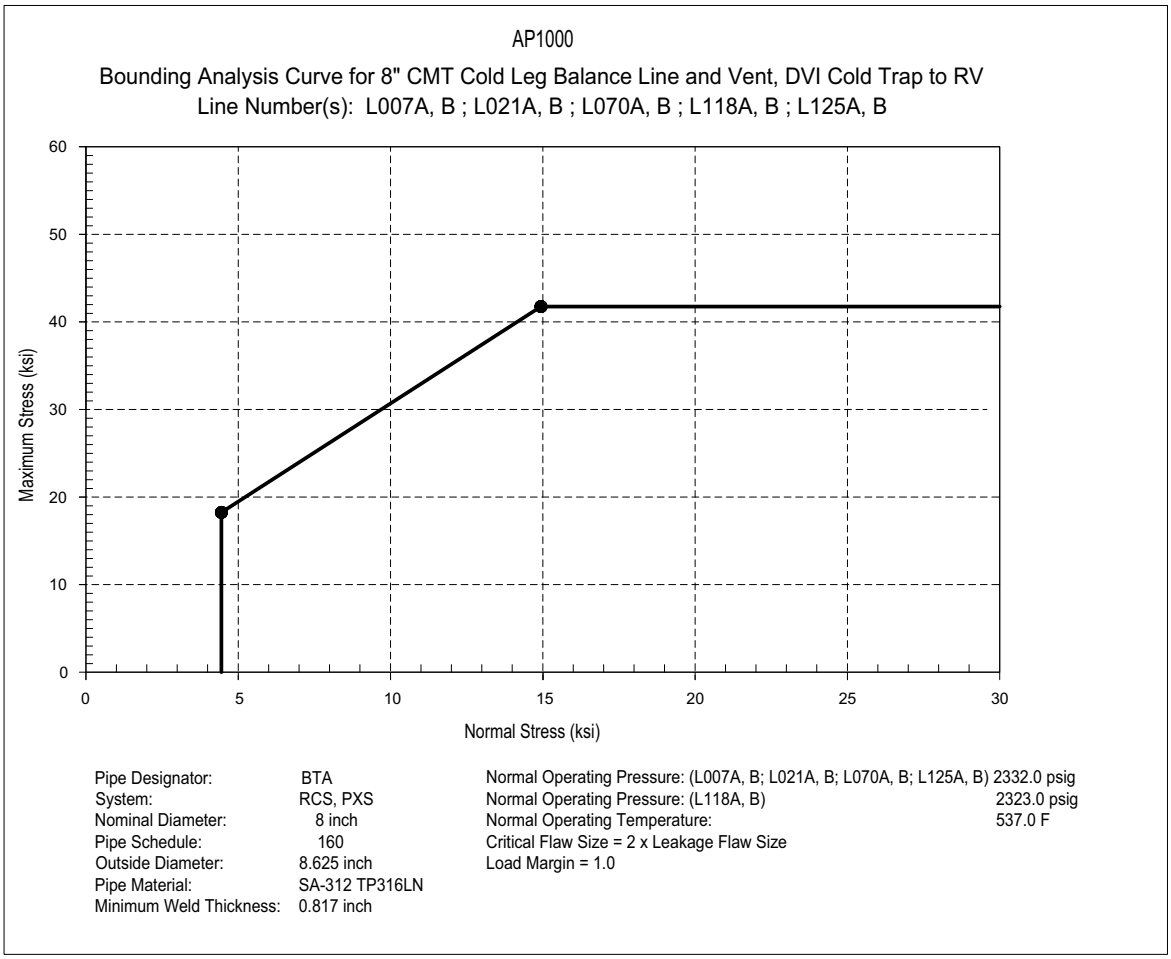

Figure 3B-14
Bounding Analysis Curve for 8″ CMT Cold Leg Balance Line and Vent, DVI Cold Trap to RV

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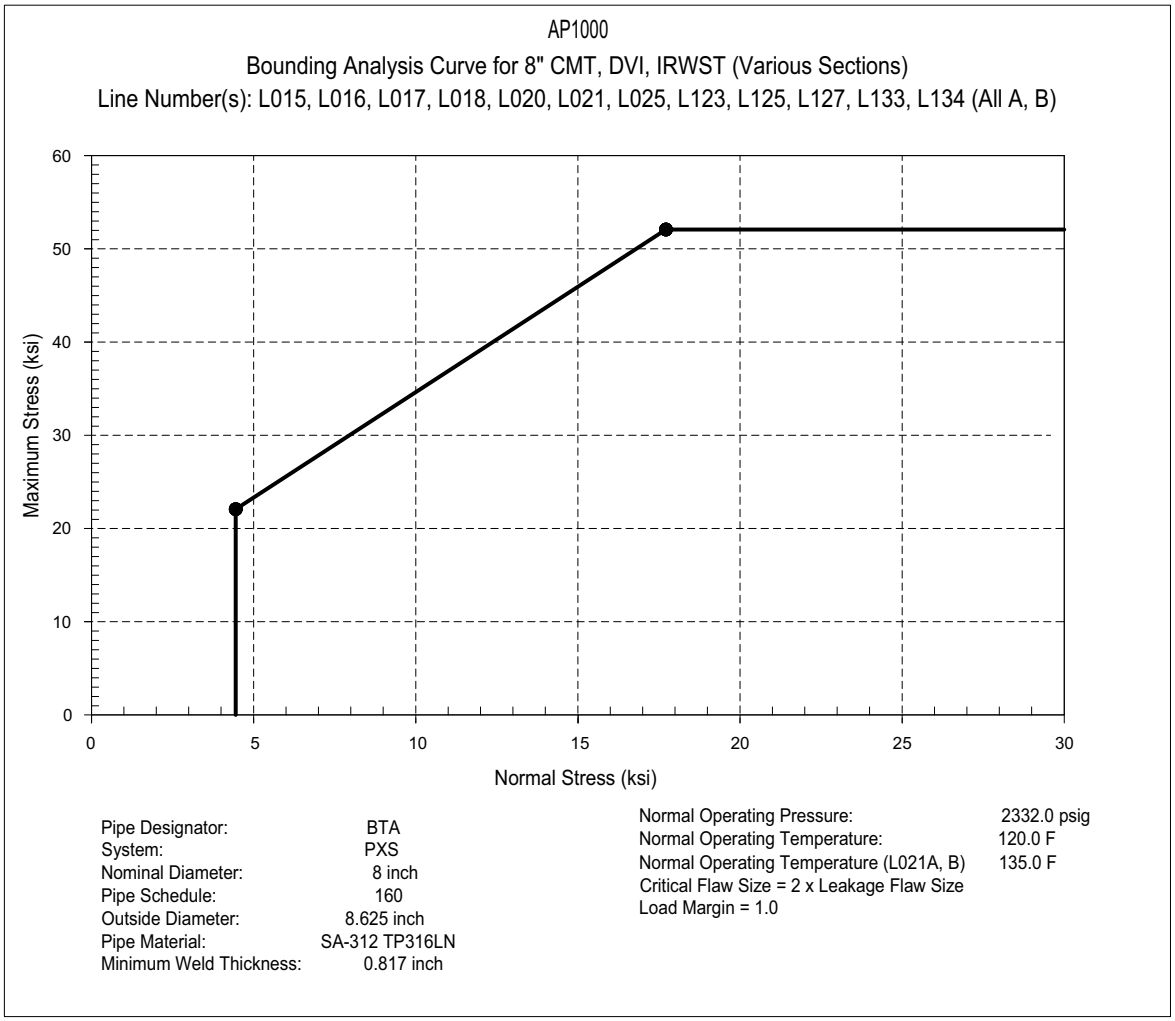

Figure 3B-15
Bounding Analysis Curve for 8″ CMT, DVI IRWST (Various Sections)

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Figure 3B-16 Not Used

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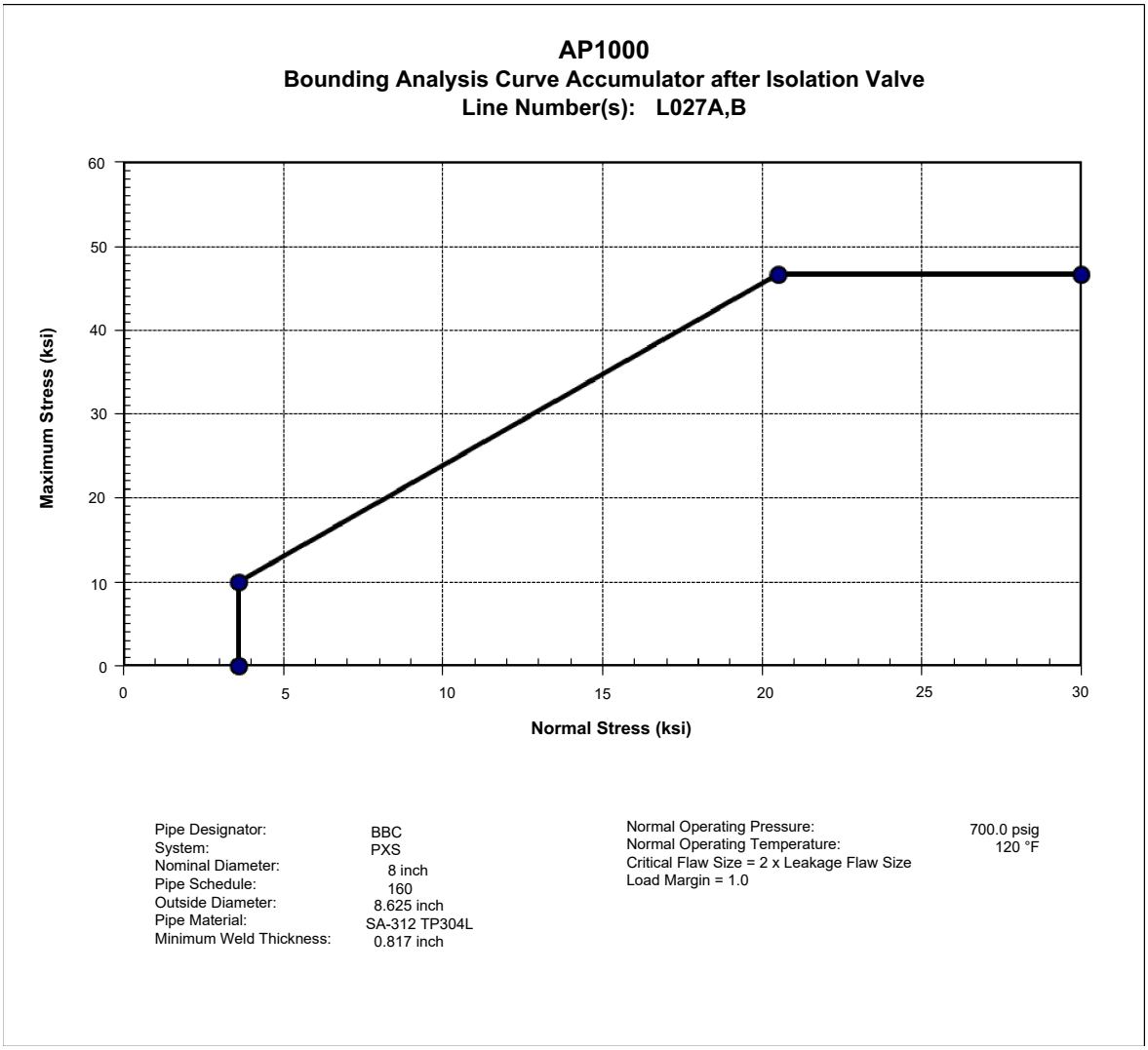

Figure 3B-17
Bounding Analysis Curve for Accumulator after Isolation Valve

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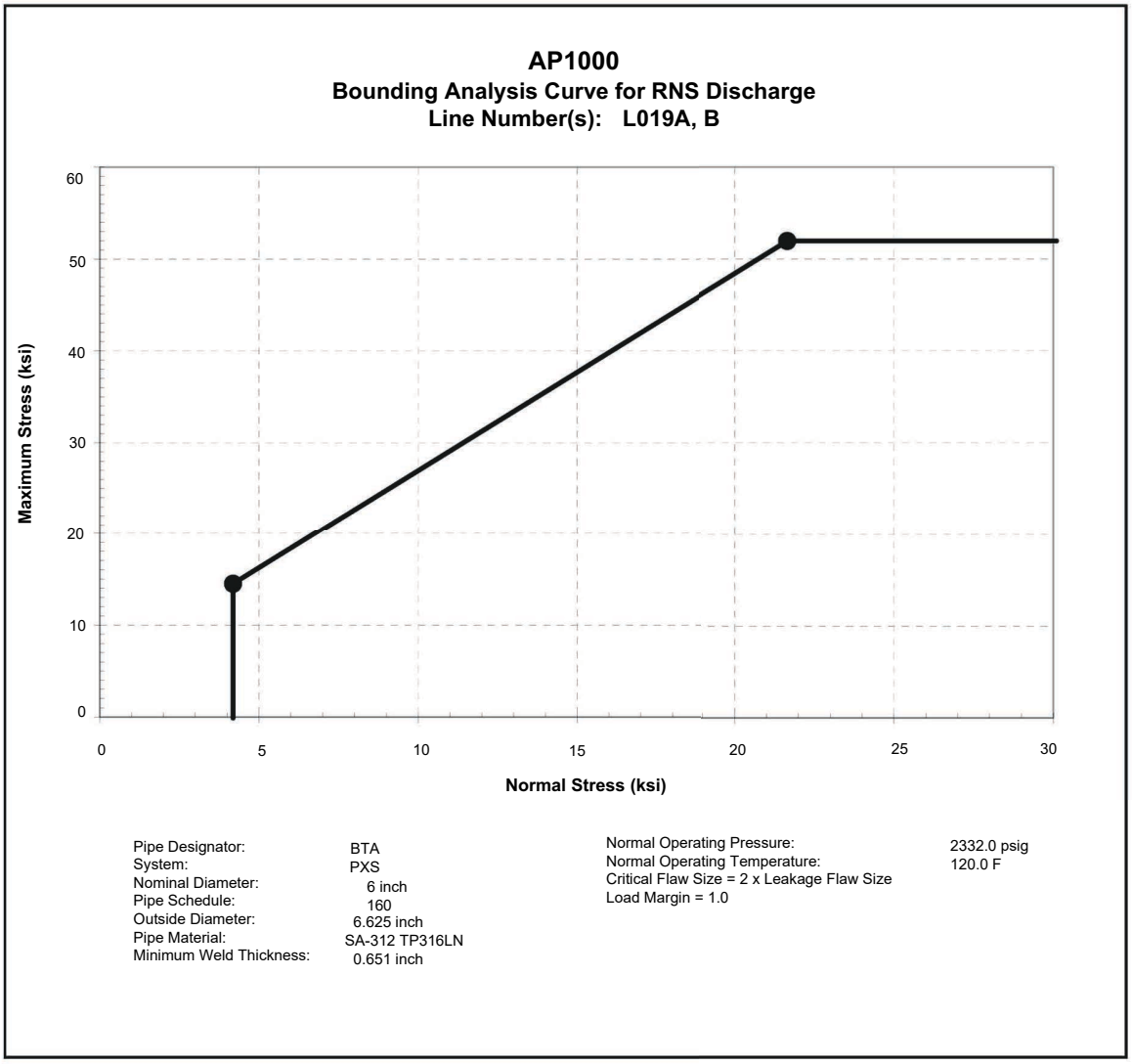

Figure 3B-18
Bounding Analysis Curve for RNS Discharge

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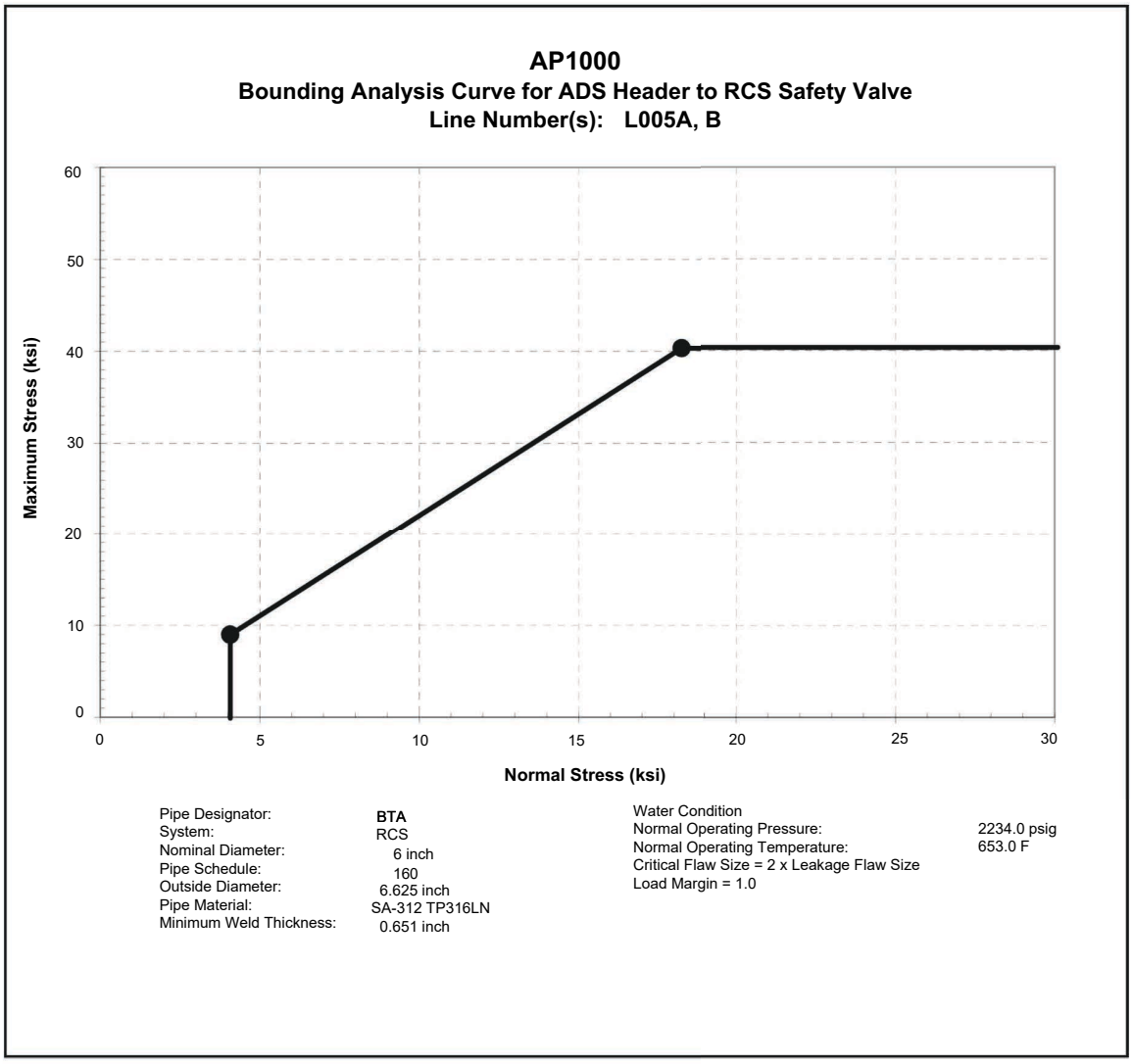

Figure 3B-19 (Sheet 1 of 2)
Bounding Analysis Curve for ADS Header to RCS Safety Valve

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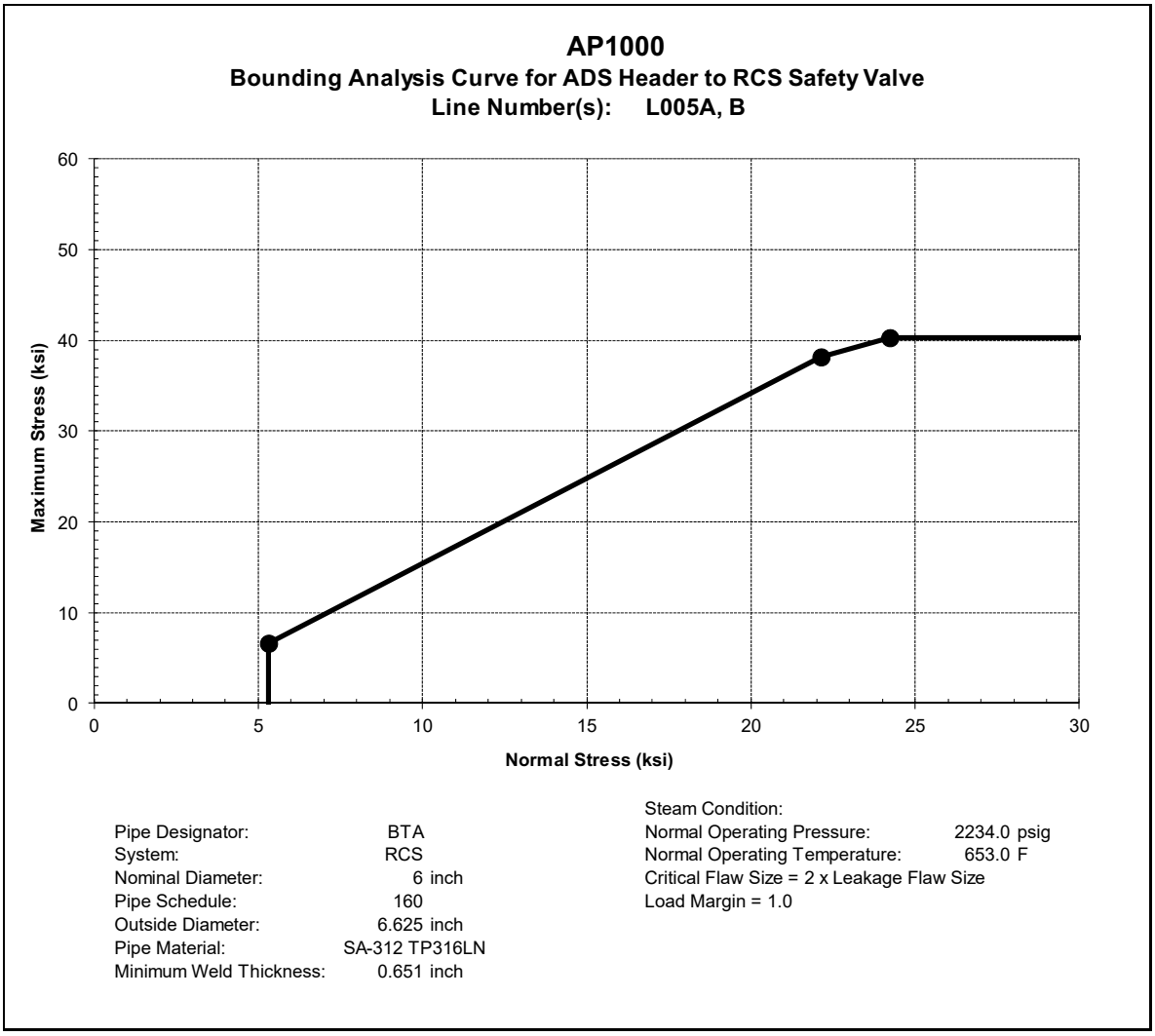

Figure 3B-19 (Sheet 2 of 2)
Bounding Analysis Curve for ADS Header to RCS Safety Valve

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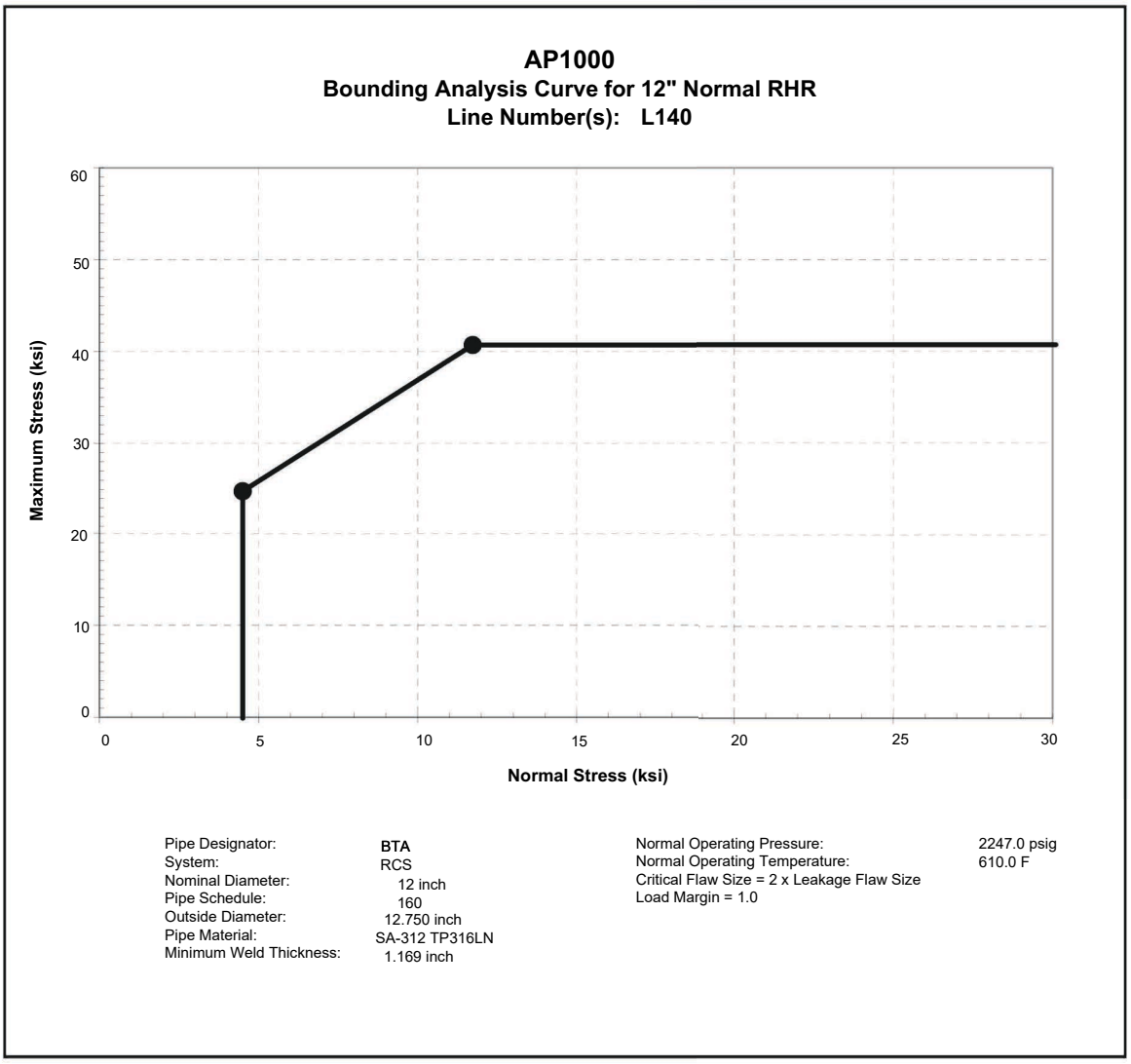

Figure 3B-20
Bounding Analysis Curve for 12″ Normal RHR

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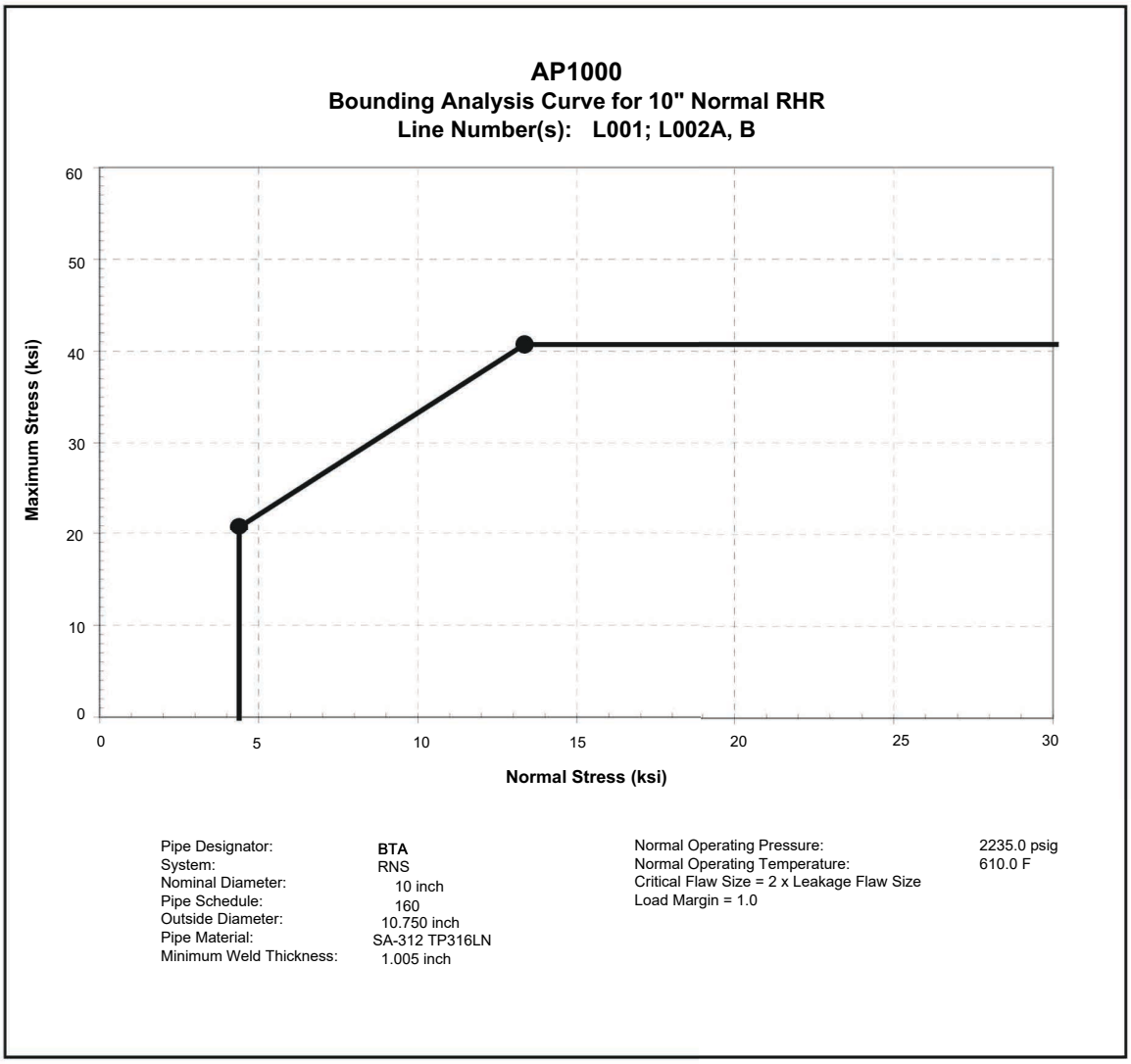

Figure 3B-21
Bounding Analysis Curve for 10″ Normal RHR

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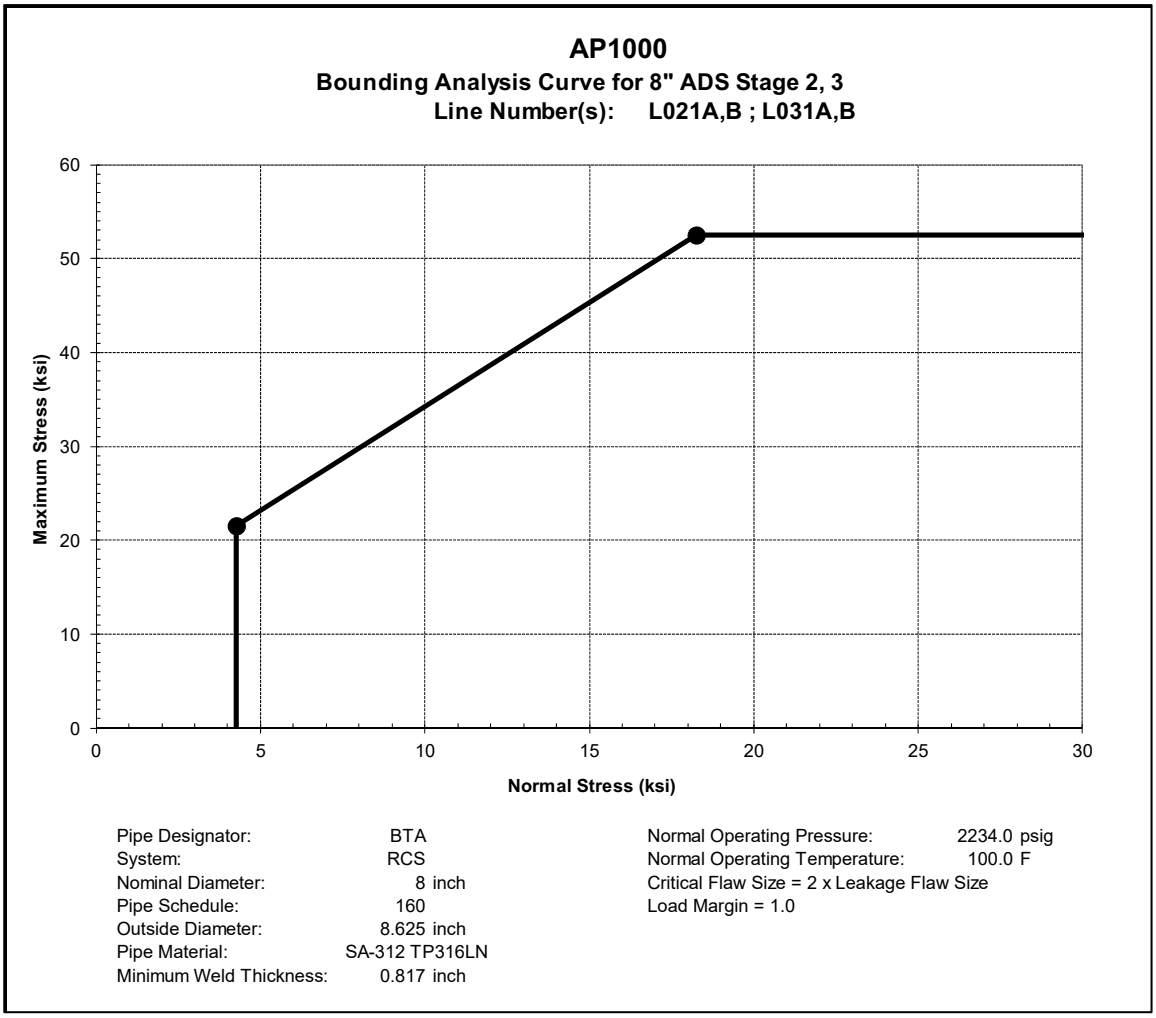

Figure 3B-22
Bounding Analysis Curve for 8″ ADS Stage 2, 3

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Appendix 3C Reactor Coolant Loop Analysis Methods

The AP1000 reactor coolant loop (RCL) model consists of three-dimensional finite elements such as pipes, beams, elbows, masses, and springs. The structural model is subjected to internal pressure, thermal expansion, weight, seismic, and pipe break loadings with imposed boundary conditions. The finite element displacement method is used for the analysis. The stiffness matrix for each element is assembled into a system of simultaneous linear equations for the entire structure. This set of equations is then solved by a variation of the Gaussian elimination method, known as the wave-front technique. This technique makes it possible to solve systems of equations with a large number of degrees of freedom using a minimum amount of computer memory.

3C.1 Reactor Coolant Loop Model Description

The piping model of the reactor coolant loop consists of a number of elements of given dimensions, sizes, and physical properties that mathematically simulate the structural response of the physical system. The system model contains the reactor pressure vessel (RPV), two steam generators (SGs), four reactor coolant pumps (RCPs), the reactor coolant loop piping, and the primary equipment supports. A two-loop model is developed for the AP1000 reactor coolant loop system.

The stiffness and mass effects of branch piping connected to the primary loop piping are considered when significant (Subsection 3.7.3.8.1).

3C.1.1 Steam Generator Model

3C.1.1.1 Steam Generator Mass and Geometrical Model

The steam generator is represented by discrete masses. The geometry of the steam generator vessel is used to determine the properties of the equivalent piping elements that join the steam generator masses for sections of the steam generator above the tubesheet. For the steam generator channel head, a super element is used to represent the stiffness characteristics that link the steam generator lower shell with the steam generator supports and nozzles. The modulus of elasticity and coefficient of thermal expansion corresponding to the thermal conditions are applied to the steam generator equivalent piping elements.

3C.1.1.2 Steam Generator Supports

The values of the steam generator support stiffnesses and locations of the supports are determined from the finite element models of the support members. The stiffness of the upper lateral supports include the steam generator shell flexibility. The local concrete building flexibility is included in the support stiffness.

3C.1.2 Reactor Coolant Pump Model

3C.1.2.1 Static Model

The reactor coolant pump is represented by a super element to represent the mass and stiffness characteristics of the pump. For a thermal expansion analysis, rigid links are modeled in parallel with a super element with the thermal expansion coefficient incorporated.

3C.1.2.2 Seismic Model

The reactor coolant pump is represented by a super element to represent the mass and stiffness characteristics of the pump. The reactor coolant pump model is a detailed model similar to that used to qualify the pump.

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3C.1.2.3 Reactor Coolant Pump Supports

There are no reactor coolant pump supports. Two reactor coolant pumps are attached to the steam generator channel head in each of the reactor coolant loops.

3C.1.3 Reactor Pressure Vessel Model

3C.1.3.1 Mass and Geometrical Model

The reactor pressure vessel model consists of equivalent pipe, stiffness, and mass elements. The elements represent the vessel shell, the vessel core barrel, the fuel assemblies, and the integrated head lift package.

The reactor pressure vessel is modeled with equivalent pipe elements and connecting stiffnesses. The equivalent pipe element properties of the vessel and barrel are those of the cylindrical structures. The beam properties of the reactor internals are adjusted to simulate their fundamental frequency. The appropriate modulus of elasticity and coefficient of thermal expansion are used for the equivalent pipe elements representing the reactor pressure vessel.

3C.1.3.2 Reactor Pressure Vessel Supports

The reactor pressure vessel is supported at the four reactor pressure vessel inlet nozzles. Each support consists of a vertical stiffness and a lateral tangential stiffness. The support is represented by a stiffness matrix. The reactor pressure vessel supports are active for the analyzed loading conditions. The reactor pressure vessel model includes the effects of the vessel shell flexibility at the inlet and outlet nozzles. The local concrete building flexibility is included in the support stiffness.

3C.1.4 Containment Interior Building Structure Model

A containment interior building structure finite element model is not required because the seismic inputs to the reactor coolant loop model are provided at all of the building attachments to the reactor coolant loop.

3C.1.5 Reactor Coolant Loop Piping Model

The reactor coolant loop piping model consists of piping elements and bends. Each reactor coolant loop has two cold legs and one hot leg. The straight runs and bends of the cold leg and hot leg are input with the nominal dimensions. Each reactor coolant loop branch connection is represented by a node point. The reactor coolant loop piping model contains distributed masses of the hot and cold leg piping for static deadweight analysis and lumped masses representing the hot and cold leg piping for dynamic analysis.

3C.2 Design Requirements

The reactor coolant piping is qualified to the requirements of the ASME Code as defined in Subsection 5.2.1.1.

The loadings for ASME Code, Section III, Class 1 components are defined in Subsection 3.9.3. The following loadings are considered in the reactor coolant loop piping analysis:

- Design pressure (P)
- Weight (DW)

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- Thermal expansion during normal operating condition
- Thermal expansion during other transient conditions (not part of this appendix)
- Safe shutdown earthquake (SSE)
- Design basis pipe break (DBPB)
- Building motions due to automatic depressurization system sparger discharge into the IRWST
- Thermal stratification during transient conditions

In addition to the analyses of these loads, the reactor coolant piping is analyzed for the effect of cyclic fatigue due to the design transients and earthquakes smaller than SSE.

3C.3 Static Analyses

3C.3.1 Deadweight Analysis

The reactor coolant loop piping system is analyzed for the effect of deadweight. The deadweight analysis is performed without considering the dry weight of the directly supported equipment. The effects of the auxiliary branch piping on the reactor coolant loop are generally negligible by the design of the auxiliary supports. A deadweight analysis is performed to include the total weight of the reactor coolant loop piping and the water weight in the components.

The reactor coolant loop deadweight model includes the corresponding active reactor coolant loop supports - reactor pressure vessel supports, and the steam generator column and lower and intermediate lateral strut supports. The steam generator upper lateral snubber supports are considered as inactive.

3C.3.2 Internal Pressure Analysis

The effects of the internal primary coolant pipe pressure are used in the calculations of forces and moments for both the reactor coolant loop piping and equipment supports. The moment stress due to pressure is considered negligible for the ASME Code pipe stress equations.

3C.3.3 Thermal Expansion Analysis

The reactor coolant loop piping is analyzed for the effects of thermal expansion. The thermal expansion analysis model considers the expansion of the reactor coolant loop piping, reactor pressure vessel, steam generator, reactor coolant pump, and the equipment supports. The stiffness effects of the auxiliary piping on the reactor coolant loop expansion are generally negligible by the design of the auxiliary lines supports.

3C.4 Seismic Analyses

The reactor coolant loop piping is analyzed for the dynamic effects of a safe shutdown earthquake (SSE).

The model used in the static analysis is modified for the dynamic analysis by including the lumped mass characteristics of the piping and equipment. The effect of the equipment motion on the reactor coolant loop piping and support system is obtained by modeling the mass and stiffness characteristics of the equipment in the overall system model. The reactor coolant loop seismic

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analysis is performed at normal full-power operation. This operating condition is considered based on the lower probability of occurrence of the earthquake at reactor coolant loop temperatures below full power.

The time history integration method of analysis is used for the reactor coolant loops. The seismic input considers the soil profiles described in Subsection 3.7.1. This input is obtained from the nuclear island seismic analysis with time history input generated from the enveloped basemat response spectra of the soil cases described in Subsection 3.7.1. The duration of the input is between 12 to 20 seconds, depending on the duration needed to envelop the design response spectra. Three runs were performed based on the envelope of the soil profiles, the building model at nominal stiffness, and at stiffness varied by + or - 30 percent to account for uncertainties. The reactor coolant loop uses separate time history displacement input from the building analysis at the primary support locations. Full direct integration is used with Rayleigh damping for loop components at 4 percent of critical damping. The steam generator snubbers have different stiffnesses in tension and compression. The mean value of the tension and compression stiffness is used in order to keep the model linear. The reactor pressure vessel vertical supports are acting downward only and are preloaded by deadweight, pressure, and thermal expansion loadings. The time history analysis is performed to evaluate the effect of lift-off of the vessel at the location of these supports.

3C.5 Reactor Coolant Loop Piping Stresses

To prevent gross rupture of the reactor coolant loop piping system, the general and local primary membrane stress criteria must be satisfied. This is accomplished by satisfying Equation (9) in paragraph NB-3652 of the ASME Code, Section III. The secondary stress caused by thermal expansion is qualified by satisfying Equation (12) in paragraph NB-3653 of the ASME Code, Section III. When the simplified requirements of ASME Section III, NB-3640 and NB-3650 are not satisfied, the more detailed methods of NB-3200 may be used as stated in NB-3630.

3C.6 Description of Computer Programs

This section provides a list of computer codes used for the AP1000 reactor coolant loop system analysis. Brief descriptions of the functions of each computer code are the following:

ANSYS – Performs Structural Analysis Using Finite Element Analysis Method. Displacements and loads are calculated at the pipe elements, supports, and equipment nozzles for pressure, deadweight, thermal, and seismic loadings.

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Appendix 3D Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment

Safety-related electrical equipment is tested under the environmental conditions expected to occur in the event of a design basis event. This testing provides a high degree of confidence in the safety-related system performance under the limiting environmental conditions. Qualification criteria were revised by IEEE 323-1974 (Reference 1) and by Regulatory Guide 1.89, which endorses this IEEE standard. The concept of aging was highlighted in IEEE 323-1974, and interpretation of the scope of aging and implementation methods were subsequently developed. 10 CFR 50.49 provides the NRC requirements for qualification of equipment located in potentially harsh environments. Therefore, the guidance provided by IEEE 323-1974 is the evolutionary root of requirements, recommended methods, and qualification procedures described in this appendix.

Specific treatment of seismic qualification, part of the qualification test sequence recommended in IEEE 323-1974, is addressed in IEEE 344-1987 (Reference 2). This appendix bases technical guidance, recommendations, and requirements for seismic qualification on IEEE 344-1987.

The AP1000 Equipment Qualification methodology addresses the expanded scope of IEEE 627-1980 (Reference 3), which encompasses the qualification of Class 1E electrical and safety-related mechanical equipment. IEEE 627 generalizes the principles and technical guidance of IEEE 323 and 344. Compliance with the IEEE 323-1974 and 344-1987 is the specific means of compliance with the intent of IEEE 627-1980 for safety-related electrical and mechanical equipment.

Safety-related electrical and mechanical equipment is typically qualified using analysis, testing, or a combination of these methods. The specific method or methods used depend on the safety-related function of the equipment type to be qualified. Safety-related mechanical equipment, such as tanks and valves, is typically qualified by analysis, with supplementary functional testing when functional operability is demonstrated only through testing, as is the case for active valves. Either testing or testing combined with analysis is the method used for environmental and seismic qualification of safety-related (Class 1E) electrical equipment.

To the extent practical, the technical discussions of this appendix follow the format headings of the equipment qualification data packages (EQDPs) to be issued as specific qualification program documentation. This formatting (see Section 3D.7) permits easy cross-reference between the methodology defined in this report and the detailed plans contained in the equipment qualification data packages. Attachment A of this appendix is the format generally used for the equipment qualification data package.

Attachment B of this appendix, "Aging Evaluation Program," describes methods for addressing potential age-related, common-mode failure mechanisms used in AP1000 equipment qualification programs. The approach conforms with current industry positions and makes maximum use of available data and experience in the evaluation, test, and analysis of aging mechanisms.

Attachment C, "Effects of Gamma Radiation Doses Below 10^4 rads on the Mechanical Properties of Materials," provides the basis that radiation aging below 10^4 rads is not a significant factor in the ability of the equipment to perform properly during a seismic event. For some devices, electrical properties are degraded above 10^3 rads. Radiation aging for safety-related equipment which is subject to lifetime doses of less than 10^4 rads (10^3 rads for certain electrical components) and not subject to a high-energy line break environment is not required to be addressed in AP1000 qualification programs.

Attachment D, "Accelerated Thermal Aging Parameters," describes the methodology employed in calculating the accelerated thermal aging parameters used in this program.

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Attachment E, "Seismic Qualification Techniques," discusses available methods for establishing a seismic qualification basis, by either test or analysis, and its application to the qualification of safety-related equipment for the AP1000.

3D.1 Purpose

The basic objectives of qualification of safety-related electrical and mechanical equipment follow:

- To reduce the potential for common cause failures due to specified environmental and seismic events
- To demonstrate that safety-related electrical and mechanical equipment is capable of performing its designated safety-related functions.

This appendix describes the methodology that has been adopted to qualify equipment according to IEEE 627-1980, "IEEE Standard for Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations." The two standards primarily used to demonstrate compliance with this standard are IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

3D.2 Scope

The qualification criteria, methods, and environmental conditions described constitute the methodology that is adopted to comply with the standards for the AP1000. This methodology applies to safety-related, seismic Category I electrical and mechanical equipment and is also used for certain monitoring equipment. Seismic Category II equipment is also within the scope of this program. The criteria used for the design of seismic Category II structures, systems, and components are discussed in Section 3.7.

Performance during abnormal environmental conditions, while not specifically designated as an industry or a regulatory qualification requirement, is also addressed by this appendix. Performance during normal service conditions is demonstrated by tests and inspections addressed by the equipment specification. Electromagnetic interference (EMI) testing or analysis is not included in the qualification process and is addressed on an individual equipment basis, as necessary.

3D.3 Introduction

This appendix identifies qualification methods used for the AP1000 to demonstrate the performance of safety-related electrical and mechanical equipment when subjected to abnormal and accident environmental conditions including loss of ventilation systems, feedline, steam line and reactor coolant system breaks, and seismic events. This appendix provides the expected conditions for various locations in the AP1000. General requirements for the development of plans/procedures/reports are also provided. Section 3D.4 identifies the various industry and regulatory criteria upon which the program is based. Section 3D.5 defines the design specifications and applicable test environments. Section 3D.6 defines the basis for the qualification method selection. Section 3D.7 outlines the general format and content of the equipment qualification data packages.

3D.4 Qualification Criteria

The environmental requirements considered in the design of safety-related equipment are embodied in GDC 2, "Design Bases for Protection Against Natural Phenomena"; GDC 4, "Environmental and Missile Design Bases"; and GDC 23, "Protection System Failure Modes." GDC 1, "Quality Standards and Records," and Criterion III, "Design Control," Criterion XI, "Test Control," and Criterion XVII,

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"Quality Assurance Record" of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to require that the environmental design of safety-related equipment is verified, documented, and controlled.

The qualification methods described in this appendix are used to verify the environmental design basis and capability of the safety-related electrical and mechanical equipment supplied for the AP1000. The results of the verification, as well as the design basis for each equipment, are documented in an equipment qualification data package (see Attachment A for sample format) and seismic analysis report for non-active mechanical equipment (with no environmental qualification requirements). Design control, test control, and quality assurance record keeping is performed through the AP1000 Quality Assurance Program. (See Chapter 17.)

3D.4.1 Qualification Guides

IEEE 323-1974 and 344-1987 serve as the basis upon which the AP1000 equipment qualification methodology demonstrates compliance with IEEE 627-1980. NRC regulations stated in 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," and NRC guidance provided in Regulatory Guide 1.89, and Regulatory Guide 1.100, endorse IEEE 323-1974 and IEEE 344-1987, respectively. The intent of the more general IEEE 627-1980 is addressed through conformance with IEEE 323 and 344.

3D.4.1.1 IEEE Standards

The following lists additional standards and guides used in developing the methodology:

- IEEE 98-1984, "IEEE Standard for the Preparation of Test Procedures for the Thermal Evaluation of Solid Electrical Insulating Materials"
- IEEE 100-1996, "IEEE Standard Dictionary of Electrical and Electronic Terms"
- IEEE 308-1991, "IEEE Standard Criteria for Class 1E Power System for Nuclear Power Generating Stations"
- IEEE 317-1983, "IEEE Standard for Electric Penetration Assemblies in Containment Structure for Nuclear Power Generating Stations"
- IEEE 381-1977, "IEEE Standard Criteria for Type Tests of Class 1E Modules Used in Nuclear Power Generating Stations"
- IEEE 382-1996, "IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Generating Stations"
- IEEE 383-1974, "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations"
- IEEE 420-1982, "IEEE Standard Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Powered Generating Stations"
- IEEE 494-1974, "IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Stations"
- IEEE 535-1986, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations"

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- IEEE 572-1985, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations"
- IEEE 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
- IEEE 649-1991, "IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations"
- IEEE 650-1990, "IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations"
- IEEE-741-1997, "IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations"
- ANSI/IEEE C37.98-1987, "IEEE Standard for Seismic Testing of Relays."

3D.4.1.2 NRC Regulatory Guides

In the area of seismic and environmental qualification of safety-related electrical and mechanical equipment, the NRC has issued the following Regulatory Guides:

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" – The guide endorses ANS and ANSI standards for quality assurance programs, but is considered here specifically for guidance in determining documentation adequacy. Appendix A of the guide, Item 9, "Procedures for Performing Maintenance," addresses procedural and documentation requirements for maintenance of safety-related equipment, preventive maintenance, repair, and replacement. This guide is a source in the development of qualification in the on-going qualification programs discussed in Subsection 3D.6.4.

Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants" – The guide prescribes acceptable values of damping used in elastic modal dynamic seismic analysis of seismic Category I structures, systems, and components. The AP1000 equipment qualification program is based on Regulatory Guide 1.61 and on values considered to be acceptable based on past NRC acceptances. The safe shutdown earthquake (SSE) damping values used for the qualification of mechanical and electrical equipment are listed in Table 3.7.1-1 of Chapter 3.

Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants" – The guide endorses, with certain qualifications, IEEE 317-1983. External circuit protection of electric penetration assemblies should meet the provisions of Section 5.4 of IEEE 741-1986, "Criteria for Protection of Class 1E Power Systems and Equipment in Nuclear Generating Stations," as these are beyond the scope of IEEE 317. The AP1000 design complies with IEEE 741-1997. The AP1000 equipment qualification program employs the recommendations of Regulatory Guide 1.63, Revision 3, in specifying qualification plans as a means of supplementing the guidance of IEEE 317 and 323.

Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants" – The guide endorses, with certain qualifications, IEEE 382-1972. The AP1000 equipment qualification program employs recommendations of Regulatory Guide 1.73, but gives preference to the guidance of IEEE 382-1996, where it is necessary to supplement the guidance of IEEE 323 or 344 in specifying qualification plans for electric valve operators.

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Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" – The guide provides guidance for conformance with 10 CFR 50.49, and endorses the procedures of IEEE 323-1974 as an acceptable means for qualifying Class 1E equipment. Implicit in the endorsement of IEEE 323 is the reference to seismic qualification methods of IEEE 344 as a part of the qualification test sequence. (See Regulatory Guide 1.100 later in this discussion.) The AP1000 equipment qualification methodology addresses the recommendations of Regulatory Guide 1.89 by the following:

- The recommendations of IEEE 323-1974 are met by the methods discussed in this appendix
- The radiation source terms used in qualification differ from those of Regulatory Guide 1.89, and are described in Section 3D.5 of this appendix
- The seismic qualification requirements employ the recommendations of IEEE 344-1987 as described in Attachment E of this appendix.

Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis" – The guide describes methods and procedures for the following:

- Combining the values of the response of individual modes in a response spectrum modal dynamic analysis to find the representative maximum value of a particular response of interest for each of the three orthogonal seismic spatial components
- Combining the maximum values (or representative maximum values) of the responses for a given element of a system or item of equipment, determined for each of the three orthogonal spatial components.

The AP1000 equipment qualification program employs methods consistent with the recommendations of Regulatory Guide 1.92 when combining individual modal response values or the response of three independent spatial components in seismic analyses.

Regulatory Guide 1.97, Revision 3, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.” The guide describes a method acceptable to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant. The AP1000 program, identified as the post-accident monitoring instrumentation system (PAMS), provides the capability to monitor plant variables and systems operating status during and following an accident. PAMS includes those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials.

Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" – The guide endorses IEEE 344-1987. Regulatory Guide 1.100 particularly notes that IEEE 344-1987 is applied in the qualification of safety-related mechanical equipment, as well as Class 1E electrical equipment. The AP1000 equipment qualification methodology employs the recommendations of Regulatory Guide 1.100, as described in Attachment E of this appendix.

Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components" – The guide describes specific methods for developing floor (and other equipment mounting locations) response spectra. Included are specific criteria for the broadening frequency amplitude peaks and smoothing of the frequency amplitude spectrum to incorporate conservatism in the seismic requirements. This is to compensate for other uncertainties of analysis. The AP1000 equipment qualification program employs methods consistent with the recommendations of Regulatory Guide 1.122.

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Regulatory Guide 1.131, "Qualification Tests of Electrical Cables, Field Splices, and Connections for Light-Water Cooled Nuclear Power Plants" – The guide endorses IEEE 383-1974. The AP1000 equipment qualification program employs the recommendations of Regulatory Guide 1.131 in specifying the qualification program plans where this guide supplements the guidance of IEEE 383 and to further demonstrate conformance with the guidance of IEEE 323.

Regulatory Guide 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants" – The guide endorses IEEE 572-1985. The AP1000 equipment qualification program employs the recommendations of Regulatory Guide 1.156 in specifying the qualification program plans where this guide supplements the guidance of IEEE 572 to demonstrate conformance with the guidance of IEEE 323.

Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants" – The guide endorses IEEE 535-1986. The AP1000 equipment qualification program employs the recommendations of Regulatory Guide 1.158 in specifying the qualification program plans where this guide supplements the guidance of IEEE 535 to demonstrate conformance with the guidance of IEEE 323.

Regulatory Guide 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems.” Regulatory Guide 1.180 provides guidance to evaluate electromagnetic and radio-frequency interference in safety-related instrumentation and control systems. The AP1000 equipment qualification program employs methods consistent with the recommendations of Regulatory Guide 1.180, where applicable.

Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactor.” The radiation dose rates and integrated doses applicable for AP1000 following a design basis accident are determined based on the criteria of NUREG-1465 and this regulatory guide.

3D.4.2 Definitions

Definitions of terms used in this appendix are contained in the referenced standards and IEEE 100, "IEEE Standard Dictionary of Electrical and Electronic Terms, Sixth Edition." Subsection 3D.4.5 clarifies the definitions of "life" (that is, design, shelf, and qualified life) as used in this methodology. The terms "design life" and "qualified life" have the meanings set forth in IEEE 323 and are used in the context of that standard.

3D.4.3 Mild Versus Harsh Environments

Qualification requirements differ for equipment located in mild and harsh environments.

IEEE 323 defines a mild environment as an environment expected as a result of normal service conditions and the extremes of abnormal service conditions where a safe shutdown earthquake is the only design basis event of consequence or conditions where thresholds of material degradation are reached. The following limits are established as the delimiting environmental parameter values for mild and harsh environments.

Typically a mild environment conforms with the environmental parameter limits of Table 3D.4-1, though others may apply to specific equipment applications or locations.

The scope of 10 CFR 50.49 is limited exclusively to equipment located in a harsh environment. The AP1000 equipment qualification program conforms with the requirements of 10 CFR 50.49 for the qualification of harsh environment equipment. A radiation-harsh environment is defined for equipment designed to operate above certain radiation thresholds where other environmental

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parameters remain bounded by normal or abnormal conditions. Any equipment that is above 10^4 rads gamma (10^3 for electronics) will be evaluated to determine if a sequential test which includes aging, radiation, and the applicable seismic event is required or if sufficient documentation exists to preclude such a test.

3D.4.4 Test Sequence

Where the test sequence deviates from that recommended by IEEE 323-1974, the deviation is justified. The test sequence employed for a given hardware item is specified in the equipment qualification data package Sections 2.1 and 3.6 (see Attachment A for example). Note that for this reference and subsequent references to Attachment A the information in Attachment A will be completed in accordance with Subsection 3.11.5. Clarifications to the IEEE 323-1974 recommended test sequence are discussed in the following:

1. Burn-In Test

   For electronic equipment, a burn-in test is completed, before operational testing of the equipment, to eliminate infant failures. The test consists of energizing the equipment for a minimum of 50 hours at nominal voltage and frequency under ambient temperature conditions. Any malfunction observed during these tests are repaired, and the 50-hour burn-in test is repeated for the repaired portion of the equipment.

2. Performance Extremes Test

   For equipment where seismic testing has previously been completed employing the recommended methods of IEEE 344-1987, seismic testing is not repeated. Testing of the equipment to demonstrate qualification at performance extremes is separately performed as permitted by IEEE 323-1974, Subsection 6.3.2(3). Additional discussion is provided in Subsection 3D.6.5.1.

3. Aging Simulation and Testing

   For equipment located in a mild environment, aging is addressed as described in Subsection 3D.6.4 and Attachment B. If there are no known aging mechanisms that significantly degrades the equipment during its service life, it is acceptable to perform seismic testing of unaged equipment. Separate testing or analysis (or both) is provided to demonstrate that the aging of components is not significant during the projected service or qualified life of the equipment.

4. Synergistic Effects

   An important consideration in the aging of equipment for harsh environment service is the possible existence of synergistic effects when multiple stress environments are applied simultaneously. This potential is addressed by conservatism inherent in the determination and use of the worst-case aging sequence and conservative accelerated aging parameters.

   The combination of effects from pressure, temperatures, humidity, and chemistry are addressed by the high-energy line break (HELB) tests. Since the test item is not exposed to radiation during this test, the effects of this parameter are conservatively addressed by subjecting the test items to the required total integrated dose before the high-energy line break. Specifically for instruments, the summing of errors for the irradiation and high-energy line break portions of the test sequence is a means of achieving conservatism.

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5. Visual Inspections/Disassembly

   The results of post-test visual inspections are not necessarily documented unless problems are discovered. Disassembly is performed only when test results or visual inspections require further investigation.

3D.4.5 Aging

3D.4.5.1 Design Life

The AP1000 equipment qualification program relies on the IEEE 323 definition for design life, particularly its distinction with respect to qualified life.

Instead of determining a qualified life for mild environment equipment for which the seismic event is the exclusive design basis event to be addressed, a design life is determined. Design lives offered in manufacturers' literature are accepted cautiously, particularly where the equipment is typically used for applications outside the nuclear industry.

An application of the design life is substantiated by sound bases in reliability theory and relevant industry standards, or experience data sources within the nuclear industry. Analyses treat the applicability and similarity of the equipment and conditions relevant for the AP1000 safety-related application. These analyses, and documentation of such, conform with guidelines of IEEE standards, as applicable, and with Sections 3D.6 and 3D.7 of this appendix.

3D.4.5.2 Shelf Life

Based on recommended storage environments, the shelf life of an equipment item is not typically a significant portion of the defined qualified life. For example, ambient temperatures during storage are typically less than the operating temperatures assumed for aging calculations. Therefore, as long as equipment is in storage and is not energized (not experiencing self-heating), a reduction in qualified life is not appropriate. However, if storage conditions differ significantly from those recommended or the storage time becomes dramatically extended, the impact to the qualified life is determined by application of the Arrhenius time-temperature relationship.

3D.4.5.3 Qualified Life

A qualified life is established for each item of safety-related equipment that is exposed to a harsh environment based on the conditions postulated at the equipment location with consideration of the equipment operability requirements.

The determination of qualified life considers potential aging mechanisms resulting from significant in-service thermal, radiation, and vibration sources, and the effects of operational cycling (mechanical or electrical or both). Generally, all aging mechanisms do not apply to each item of equipment. Relevant aging mechanisms addressed or simulated are determined jointly with the identification of the equipment's critical components, functional modes, and material characteristics, and the assessment of tolerable limits in degradation of the components. An a priori consideration in selecting equipment to qualify is the evaluation of the equipment's inherent capability to survive and operate under the conditions for which it is qualified.

Since past qualification tests have provided a substantial basis for this assessment (indeed, some may provide sufficient basis to preclude any new testing as part of the AP1000 program) specific guidance on each equipment type is not provided here. Application of the lessons of past tests, insights provided in generic industry communications (for example, technical bulletins, NRC

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Information Notices), and sound judgment in the development of test plans and analysis procedures are addressed in the documentation of qualification for each equipment type, as applicable.

Qualified life is established by the most limiting of the five aging mechanisms. Qualified life may be limited by the tolerable degradation of a single component or material critical to the equipment's capability to perform its safety function. Aging is subject to the requirement for margin. See Subsection 3D.4.8 of this appendix.

For some equipment, qualified life is established on the basis of periodic replacement of certain short-lived, age-sensitive components. The user complies with the mandatory replacement practices documented in the equipment qualification data packages (see Subsection 3D.7.2.5) to affirm the equipment qualified life.

The objective of thermal and irradiation qualified life testing is to simulate, according to the available empirical material data, the degradation effects such that the equipment is in its end-of-life condition before the application of the design basis event conditions testing.

Thermal qualified life is evaluated using the Arrhenius time-temperature relationship. (See more detailed discussions in Attachments B and D of this appendix.) The activation energy is the exclusive material-dependent parameter input into the Arrhenius time-temperature relationship. The activation energy is an empirically determined parameter indicative of the thermal degradation of a physical property of a material (for example, elasticity of silicone rubbers or insulation resistance of cross-linked polyethylene cable insulation). Each material may have more than one physical property that may be subject to thermal degradation over time. Consequently, it may have different activation energies with respect to each property. Thus, the selection of activation energy considers the material property most germane to the safety-related function of the material or component. (Also see Subsection 3D.4.5.4.)

Common practice for the evaluation of irradiation-induced degradation is to consider the sum of estimated life and the accident radiation doses before design basis event testing. When testing, the total dose is applied during the radiation aging simulation portion of the qualification test sequences. This is considered conservative because the equipment has accumulated an exposure, or total integrated dose, before the initiation of the seismic and accident environment testing. Further bases for test dose determination are provided in Subsection 3D.5.1.2. Sufficient margin must be included in test parameters (see Subsection 3D.4.8). The same margins are applied in an analysis of radiation life or design basis event radiation dosage.

The simulation of age also includes the effects of operational cycling, both electrical and mechanical. Generally, these considerations are applied specifically to electromechanical equipment such as valve operators, limit switches, motors, relays, switches, and circuit breakers. Furthermore, the simulation of these effects is waived where existing data demonstrates equipment durability greatly in excess of estimated number of operating cycles for Class 1E service. Analysis or justification is provided for any case where operational cycling is omitted in the test sequence.

It is not practicable to simultaneously simulate the aspects of aging. Development of each test plan considers known synergies and sequences the simulation of the various applicable aging mechanisms with regard for conservatism of the overall effect on the test specimens.

3D.4.5.4 Qualified Life Reevaluation

It may be possible to extend the qualified life of a particular piece of equipment at some future date by comparing the actual in-plant environments and conditions during the equipment's installed life to the values assumed for the AP1000 in establishing the qualified life. For example, the thermal qualified life might be extended by performing an analysis of actual internal or external temperatures

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(or both) experienced. Continuous temperature monitoring or use of sample devices for testing and trending materials aging may be used. These efforts reveal the conservatism of the original thermal life calculation, which assumes that the maximum value specified for the normal plant operating environment endured at all times.

Although a strict Arrhenius calculation may yield an extended qualified life, care is taken in using this extrapolation because of uncertainties in the methodology. The Arrhenius time-temperature relationship relies on empirically determined activation energies of materials. This parameter has been determined for a number of materials to at least a good approximation for small temperature extrapolations. Extrapolation of the Arrhenius model to time periods of temperature beyond the range of materials test data is questionable and may result in large errors.

Calculated qualified lives based on this methodology should be limited to 20 years unless sound technical bases can be cited. This position is consistent with industry guidelines such as IEEE 98-1984, NUREG/CR-3156 (Reference 4), and EPRI NP-1558 (Reference 5).

3D.4.6 Operability Time

The post-accident operability times specified in Section 1.7 of each equipment qualification data package (see Attachment A) are conservatively established based on the safety-related function performed by that equipment for the spectrum of design basis event conditions. These include the following:

- Trip and/or monitoring functions of sensors and instruments
- Operability requirements for electromechanical equipment
- Duration of required operability for active valves.

This evaluation also considers what consequences the failure of the device has on the operator's action or decisions and the mitigation of the event. Table 3D.4-2 lists and explains typical operability times.

For monitoring functions, simulated aging techniques are employed to shorten the test time following a high-energy line break. These also comply with the margin guidelines of Subsection 3D.4.8.

Margins for trip function requirements are contained in the high-energy line break envelopes that encompass a full spectrum of break sizes. The defined margins are also justified by the fact that the signal generated by the sensor is locked in by the protection system and does not reset should the sensor fail after completion of its designated trip time requirement.

3D.4.7 Performance Criterion

The basic performance criterion is that the qualification test program demonstrates the capability of the equipment to meet the safety-related performance requirements defined in the equipment qualification data package, Section 1.7, while subjected to the environmental conditions specified in the equipment qualification data package, Section 1.8. Where three or more specimens are tested, failure of one of three may be considered a random failure, subject to an investigation concluding that the observed failure is not indicative of a common-mode occurrence.

For equipment for which aging is addressed by evaluation of appropriate mechanism(s) through a review of available material and component information, the basic acceptance criterion is that the evaluation of test data demonstrates that the effect of aging is minor and does not affect the capability of the aged equipment to perform specified functions.

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3D.4.8 Margin

IEEE 323 (Subection 6.3.1.5) recommends that margin be applied to the most severe specified service conditions in order to establish the conditions for qualification. This margin is provided in order to account for normal variations in commercial production of equipment and for reasonable errors in defining satisfactory performance. Further guidance for determining the acceptability of margin with respect to application-specific or location-specific requirements is provided by the NRC in NUREG-0588 and Regulatory Guide 1.89, Revision 1. Margins are included in addition to conservatisms applied during the derivation of the local environmental conditions of the equipment, unless the conservatism is quantified and specifically shown to meet or exceed the guidance of IEEE 323, NUREG-0588, and Regulatory Guide 1.97.

Consistent with IEEE 323, margin is incorporated into the specification of the generic qualification parameters by either increasing the test levels, number of test cycles, test duration, or a combination of these options as appropriate. The AP1000 generic qualification parameters are selected to envelop a range of loss of coolant accident and high-energy line break sizes, and equipment locations. Margin in seismic conditions for test and analysis are addressed in Subsection 3D.4.8.4. The margins available for a specific application may be larger than the generic equipment qualification test objective for seismic events and some events outside containment and are verified on an application-specific basis.

In defining qualification parameters, the AP1000 equipment qualification program incorporates margin as described in the following subsections. Table 3D.4-3 lists margin requirements applied.

For generic testing, margin is applied at the time of testing to cover known safety-related applications of the equipment. Generally, this results in a worst-case test that provides substantial margin for applications where lesser environments apply. Application of margin for seismic qualification addresses several cases unique to the qualification approach. (See Subsection 3D.4.8.4.)

3D.4.8.1 Normal and Abnormal Extremes

As indicated in Section 7 of IEEE 323, the application of margin is directed at specifying adequate qualification requirements for the most severe service conditions represented by the design basis events (that is, high-energy line break accidents and seismic events). Consequently, the AP1000 equipment qualification methodology does not apply any systematic margin to the normal and abnormal environment parameters in defining the qualification conditions.

For electronic equipment not required to operate in a high-energy line break environment, additional margin is included by requiring that the equipment operate through the conservative normal and abnormal service conditions indicated in Figure 3D.5-1. The environmental parameters at least equal the specified range of service condition parameters. An exception occurs for transmitters where a performance verification is completed at 130°F on each transmitter to encompass the specified maximum abnormal conditions. For equipment to be qualified to operate in a high-energy line break environment, qualification to the severe high-energy line break conditions demonstrates ample margin for acceptable performance under certain specified normal and abnormal service conditions.

3D.4.8.2 Aging

No specific margin is applied to the time component in deriving appropriate aging parameters, if margin is included in deriving the accelerated aging parameters employed for simulating each applicable aging mechanism.

Margin may be addressed by demonstrating the adequacy of the aging simulated by test through the calculation of time-temperature equivalence (See Attachment B of this appendix) or the comparison

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of simulated parameters with those applicable to the intended service of the equipment. The installed life of equipment must not exceed the thermal qualified life demonstrated by this calculation. Additionally, the selection and use of the thermal aging parameters both for test and subsequent calculations are subject to criteria, including the following:

- Test temperature must endure for at least 100 hours
- Test temperature must exceed any application temperature (that is, the normal or abnormal environment in which the equipment is to be used, and for which the life is calculated)
- Test temperature must be less than state-change temperature for materials critical to the equipment safety-related function or capability to endure the subsequent design basis event testing
- A conservative activation energy is used. Activation energies for materials critical to the equipment safety-related function or capability to endure the subsequent design basis event testing are considered. Materials may have several activation energies, each for a different material property. Relevant material properties are considered.

If margin is not demonstrated through conservatism in the aging parameters or calculation, then a +10 percent time margin is included.

A margin of 10 percent in the other parameters (for example, irradiation, operational cycling) applies to both the aging simulation and the post-accident simulated aging, with few exceptions.

For equipment required by design to perform its safety-related function within a short time period into the design basis event (that is, within seconds or minutes), and having completed its function, subsequent failure is shown not to be detrimental to plant safety, margin by percentage of additional time or equivalent time-temperature is not applied. Margins for trip function requirements are contained in the worst-case high-energy line break envelope. Test parameters are simulated on a real-time basis with the transient condition margins listed in Table 3D.4-3. Trip signals, once generated by the sensors, are locked in by the protection system and do not reset in the event of subsequent sensor failure.

3D.4.8.3 Radiation

An additional 10 percent is added to the calculated total integrated dose in specifying the test requirements.

3D.4.8.4 Seismic Conditions

Required response spectra included in Subsection 3.7.2 or other AP1000 program specifications are the conditions to be enveloped. No amplitude margin is added to these conditions. Peak broadening is also discussed in Subsection 3.7.2. Seismic qualification by analysis addresses margin requirements by other methods of conservatism while using the same sets of requirements - no amplitude margin is included. For qualification tests, the test facility increases the amplitude of seismic profiles by 10 percent to incorporate margin.

For most applications, considerable margin exists with respect to the acceleration levels employed and the width of the response spectra. Further details are addressed in Attachment E.

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3D.4.8.5 High-Energy Line Break Conditions

The envelopes specified for high-energy line breaks are selected to encompass the transients resulting from a spectra of loss of coolant accidents and high-energy line break sizes and locations, and various nodes in the containment. As a consequence, these design envelopes already contain significant margin with respect to any transient corresponding to a single break.

The AP1000 equipment qualification methodology requires that the qualification envelopes be derived with a margin of 15°F and 10 psi with respect to the design envelopes in Figure 3D.5-8. The margin on dose is identified by comparing the location specific dose requirements and the AP1000 equipment qualification parameters.

Margin on chemistry is 10 percent on the bounding concentration of the chemical additives. For fluids with acidic pH, the margin is applied in chemicals that increase the acidity of the fluid. For fluids with basic pH, the margin is applied in chemicals that increase the alkalinity of the solution. Solutions simulating post-accident containment sump chemistry reflect the influence of the buffering agent and change over time from acidic to alkaline. Margin in the initial acidic sump chemistry does not have to be explicitly modeled because it is accomplished through continued low pH fluid injection. Required pH of chemistries is met or bounded in a conservative fashion.

3D.4.9 Treatment of Failures

The primary purpose of equipment qualification is to reduce the potential for common mode failures due to anticipated environmental and seismic conditions. The redundancy, diversity, and periodic testing of nuclear power plant safety-related equipment are designed to accommodate random failures of individual components.

Where an adequate test sample is available, the failure of one component or device together with a successful test of two identical components or devices indicates a random failure mechanism, subject to an investigation concluding that the observed failure is not common mode. Where insufficient test samples prevent such a conclusion, any failures are investigated to ascertain whether the failure mechanism is of common mode origin. Should a common mode failure mechanism be identified as causing the failure, either a design change is implemented to eliminate the problem or a repeat test completed to demonstrate compliance with the criteria.

For those mild environment equipment items that, through a review of available documentation, are subject to failure during a seismic event due to significant aging mechanisms, the material or component is replaced or monitored through a maintenance/surveillance program.

3D.4.10 Traceability

A system of baseline design documentation is instituted to control the design, procurement, and manufacturing of safety-related products. As part of this quality control program, critical parts are identified and assigned a level of control to reflect the estimate of potential qualification or procurement problems. In addition, levels of quality inspection are also assigned to each part. The baseline design documentation describes the equipment in sufficient detail (drawing number, part number, manufacturer) to establish traceability between equipment shipped and equipment tested in the qualification program.

3D.4.10.1 Auditable Link Document

The purchaser of equipment referencing this program requires an auditable link document that provides a tie between the specific equipment and documentation of qualification reviewed for

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acceptance under this program. This auditable link document includes one or more of the following sections, as applicable.

3D.4.10.1.1 Equipment Link

This documentation certifies that the plant specific equipment is covered by the applicable equipment test reports. This link reflects a comparison of the as-built drawings, baseline design document or other documentation of the tested equipment to the specific equipment.

3D.4.10.1.2 Component Link

This documentation certifies that the components (for example, replacement parts) used in the specific equipment are represented in the applicable test reports or via analysis under a component aging program, such as that described in Attachment B. This link applies only to equipment whose equipment qualification data package references a component testing program. This link reflects a comparison of the as-built drawings, baseline design document, or other documentation of the specific equipment to the component program listing.

3D.4.10.1.3 Material Link

This documentation certifies that the materials used in the equipment are represented in a materials aging analysis, such as that described in Attachment B. This link applies only to equipment whose equipment qualification data package references the materials aging analysis and reflects a comparison of the as-built drawings, baseline design document, or other documentation of the plant specific equipment to the materials aging analysis listing.

3D.4.10.2 Similarity

Where differences exist between items of equipment, analysis may be employed to demonstrate that the test results obtained for one piece of equipment are applicable to a similar piece of equipment. Documentation of this analysis conforms with guidelines in IEEE 323 and 627, and Subsection 3D.6.2.1 and Section 3D.7 of this appendix.

3D.5 Design Specifications

The conditions and parameters considered in the environmental and seismic qualification of AP1000 safety-related equipment are separated into three categories: normal, abnormal, and design basis event. Normal conditions are those sets and ranges of plant conditions that are expected to occur regularly and for which plant equipment is expected to perform its safety-related function, as required, on a continuous, steady-state basis. Abnormal conditions refer to the extreme ranges of normal plant conditions for which the equipment is designed to operate for a period of time without any special calibration or maintenance effort. Design basis event conditions refers to environmental parameters to which the equipment may be subjected without impairment of its defined operating characteristics for those conditions.

The following subsections define the basis for the normal, abnormal, design basis event, and post-design basis event environmental conditions specified for the qualification of safety-related equipment in the AP1000 equipment qualification program. (These are cited in Section 1.7 of each equipment qualification data package; See Attachment A.)

The service conditions simulated by the test plan are identified in equipment qualification data package Section 3.7. (See Subsection 3D.7.4.6 and Attachment A.) In general, the parameters employed are selected to be equal to (normal and abnormal) or have margin (design basis event and post-design basis event) with respect to the specified service conditions of equipment qualification

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data package, Section 1.7, as recommended by IEEE 323. These conditions are conservatively derived to allow for possible alternative locations of equipment within the plant.

3D.5.1 Normal Operating Conditions

Equipment not subject to high-energy line break environments is qualified for normal and abnormal conditions, as applicable, employing a cyclic test sequence of environmental and electrical extremes. A typical test profile, including voltage and frequency cycling, is shown in Figure 3D.5-1.

3D.5.1.1 Pressure, Temperature, Humidity

The calculated values for temperature, pressure, and humidity during normal operation are specified in Table 3D.5-1 as a function of in-plant location.

3D.5.1.2 Radiation Dose

The normal operating dose rates and consequent 60-year design expectation doses at various locations inside containment are specified in Table 3D.5-2. These values have been derived from theoretical calculations assuming an expected 60 years of continuous operation with a reactor power of 3468 MWth (including 2-percent power uncertainty) and steady-state operating conditions. Equivalent data at various locations outside containment are also specified in Table 3D.5-2.

The total integrated dose employed for testing is a combination of normal and accident doses (where applicable), and is defined to equal or exceed the maximum radiation dose contained in the equipment qualification data package. (See Section 3D.7 and Attachment A.) A margin of 10 percent is included in defining the total integrated doses for testing. Normal operating and accident gamma doses are simulated using a cobalt-60 or spent fuel source. The test dose is applied at a rate approximate to the maximum accident dose rate. Irradiation dose rates less than the maximum are considered where there is significant shielding (greater than two mm of steel) or where the peak in-containment design basis event dose rate is not expected to affect the equipment's electrical performance.

Low radiation dose rates encountered during normal operation for most equipment are not considered critical parameters because of the resultant low total integrated dose (10^4 to 10^5 rads) achieved. For equipment not required post-accident, material can be selected based on previous test results. Another test on the completed assembly is not required.

If equipment is located in an environment where the normal total integrated dose exceeds the threshold for radiation damage, then testing is required. For equipment required post-accident, the dose received during normal operation is usually an insignificant part of the total integrated dose, including accident conditions effects. The supposition that a concern over low dose rate effects diminishes as the total integrated dose decreases is supported by Sandia National Laboratories tests (References 6 and 7) on selected materials over a range of dose rates. These studies indicate that reduction in original properties is about the same (and not significant) for dose rates up to a total integrated dose in the megarad range. Although these tests were not performed at dose rates as low as those expected in a nuclear power plant and electrical properties were not evaluated, they do give some indication of the effect of varying the rate.

Based on results of research programs to date and low total integrated dose reached during normal operation, the AP1000 equipment qualification program does not consider degradation due to low dose rate effects to be a significant concern. Therefore, the program does not include any action other than inspecting organic material degradation in the plant through normal maintenance.

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3D.5.2 Abnormal Operating Conditions

Abnormal environments are defined to recognize possible plant service abnormalities that lead to short-term changes in environments at various equipment locations.

For equipment located inside containment, several abnormal environment types are considered in Subsection 3D.5.2.1. Equipment located outside containment is addressed in Subsection 3D.5.2.2.

3D.5.2.1 Abnormal Environments Inside Containment

In the AP1000 equipment qualification program there are multiple events postulated at least once over the 60 year design expectation which cause abnormal environmental conditions in the containment. These are divided into two groups of events, based on peak containment temperatures expected.

Group 1: 150°F Events

- Loss of a fan cooler
- Loss of all ac for up to 2 hours
- Pressurizer safety valve open/close during reactor coolant system transient.

Group 2: 250°F Events

- Spurious automatic depressurization system (ADS) actuation
- Passive residual heat removal (PRHR) system use (long-term)
- Reactor coolant system depressurization via pressurizer safety valve
- Small loss of coolant accident.

Only the vapor conditions (containment air temperature, pressure, and humidity) associated with these events are considered in the abnormal, group 2 environmental conditions. Other environmental conditions resulting from these events (such as submergence, or radiation) are not included in the group 2 environmental conditions, but are part of or bounded by the accident condition qualification.

Table 3D.5-3 presents the conditions associated with these abnormal environments. Best estimate analyses of AP1000 plant behavior are used to determine abnormal environmental conditions. Plant recovery occurs after each event with varying degrees of time and maintenance efforts. Thus, the conditions resulting from these events are considered in the development of aging test parameters. Event frequency, conditions, and duration are accounted for within the context of the qualified life objective of each equipment type test program.

3D.5.2.2 Abnormal Environments Outside Containment

Figure 3D.5-1 represents the assumptions made in defining potential abnormal environments due to loss of air-conditioning or ventilation systems.

Table 3D.5-4 defines the abnormal environments as a function of equipment location. The assumed durations of the abnormal conditions specified in Table 3D.5-4 are consistent with operating practices and technical specification limits. For certain plant applications, qualification for abnormal environments is not necessary when equipment is located in environmental zones that do not exceed manufacturer's design limits for equipment operation, or — for flooding/wetting — if qualified in accordance with applicable criteria in Subsection 3.11.1.2.

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3D.5.3 Seismic Events

See Attachment E.

3D.5.4 Containment Test Environment

The ASME Code, Section III, Paragraph NE-6321 specifies that containment integrity is demonstrated at not less than 1.1 times design pressure. The design pressure of the AP1000 containment is 59 psig. Consequently, the minimum pressure specified for the containment test is 59 x 1.1 = 64.9 psig. Other environmental parameters (such as temperature and humidity) of the containment test are adequately enveloped by the parameters specified for normal or abnormal plant conditions.

3D.5.5 Design Basis Event Conditions

Performance requirements are specified for those design basis events for which the equipment performs a safety-related function and which have a potential for changing the equipment environment due to increased temperature, pressure, humidity, radiation, or seismic effects. The environmental conditions for each applicable design basis event are summarized in Table 3D.5-5 and are defined in the equipment qualification data package (see Section 1.8 of Attachment A) based on considerations and assumptions described in the following subsections. The radiological conditions presented in Table 3D.5-5 represent the maximum one year TID expected to occur in that zone. Dosages used for equipment qualification will be evaluated on a room-by-room basis and may be less than this maximum dose.

3D.5.5.1 High-Energy Line Break Accidents Inside Containment

3D.5.5.1.1 Radiation Environment – Loss of Coolant Accident

The radiation dose rates and integrated doses following a design basis loss-of-coolant accident (LOCA) are determined based on the criteria and guidance provided in NUREG 1465, “Accident Source Terms for Light-Water Nuclear Power Plants – Final Report” (Reference 8) and Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” (Reference 9).

The radiation exposure inside the containment is conservatively estimated by considering the dose in the middle of the AP1000 containment. Radioactive sources are assumed to be uniformly distributed throughout the containment atmosphere, and plate out of non-gaseous activity on containment surfaces is considered. No credit is taken for the shielding provided by internal structures and equipment.

Sources are based on the emergency safeguards system core thermal power rating and the following analytical assumptions:

- Power Level (including 2-percent power uncertainty) ..... 3,468 MWt
- Fraction of total core inventory released to the containment atmosphere:

  Noble Gases (Xe, Kr) 1.0

  Halogens (I, Br) 0.40

  Alkali Metals (Cs, Rb) 0.30

  Tellurium Group (Te, Sb, Se) 0.05

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Barium, Strontium (Ba, Sr) 0.02

Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) 0.0025

Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) 0.0002

Cerium Group (Ce, Pu, Np) 0.0005

The radionuclide groups and elemental release fractions listed above are consistent with the accident source term information presented in NUREG-1465 and Regulatory Guide 1.183.

The timing of the releases are based on NUREG-1465 assumptions. The release scenario assumed in the calculations is described below.

An initial release of activity from the gaps of a number of failed fuel rods at 10 minutes into the accident is considered. The release of 3 percent of the core inventory of the volatile species (defined as noble gases, halogens, and alkali metals) is assumed. An additional release period occurs over the next 30 minutes, that is, from 10 to 40 minutes into the accident. At this point, 5 percent of the total core inventory of volatile species has been considered to be released.

Over the next 1.3 hours, releases associated with an early in-vessel release period are assumed to occur, that is, from 40 minutes to 1.97 hours into the accident. This source term is a time-varying release in which the release rate is assumed to be constant during the duration time. Additional releases during the early in-vessel release period include 95 percent of the noble gases, 35 percent of the halogens, and 25 percent of the alkali metals, as well as the fractions of the tellurium group, barium and strontium, noble metals, lanthanides, and cerium group as listed above.

There is no additional release of activity to the containment atmosphere after the in-vessel release phase. Activity removal by natural mechanisms as described in Chapter 15 Subsection 15.6.5.3.2 and Appendix B are considered only during the first 24 hours following the accident.

The above source terms are consistent with the guidance provided by the NRC in Regulatory Guide 1.183 for design basis accident (DBA) loss-of-coolant accident (LOCA) evaluations.

Based on these assumptions the instantaneous and integrated gamma and beta doses for the containment atmosphere following a loss of coolant accident are shown in Figures 3D.5-2 and 3D.5-3, respectively.

The total integrated dose of radiation employed for testing is a combination of normal and design basis event dose, as applicable. It is defined to equal or exceed the maximum radiation dose contained in the specification (Attachment A, Section 1.8). A margin of 10 percent is included in defining the total integrated dose for testing. Normal operating and design basis event gamma doses are simulated using a cobalt-60 source. The test dose is applied at a rate approximate to the initial phase of the design basis event dose rate shown in Figure 3D.5-2 as modified by shielding effects (typically 0.2 to 0.25 Mr/hr).

Where exposed organic material is evaluated by test for the effect of (accident) beta radiation, a beta source is employed. Or a cobalt-60 or spent fuel source is used to impart the same dose using gamma radiation. When doing beta equivalent testing, the total integrated dose using gamma is conservatively equal to the beta total integrated dose, or the resulting bremsstrahlung is calculated and the test item is exposed to an equivalent gamma dose.

Radiation conditions for loss of coolant accident envelop other scenarios, such as rod ejection.

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3D.5.5.1.2 Radiation Environment – Steam Line Break Accident

Sources associated with a steam line break accident are based on the release of reactor coolant system activity, assuming operation with the design basis fuel defect level of 0.25 percent. It is further assumed that an “event-initiated” iodine activity spike occurs, which increases the reactor coolant activity during the accident based on a rate of increase that is 500 times the normal activity appearance rate in the reactor coolant.

The activity inventory is instantaneously released into the containment atmosphere. The dose is conservatively estimated by considering the dose rate in the middle of the containment, with no credit for the shielding provided by the internal structures, components, and equipment. The instantaneous and integrated gamma and beta doses for the containment atmosphere following a steam line break are shown in Figures 3D.5-4 and 3D.5-5, respectively.

3D.5.5.1.3 Radiation Environment – Feedline Break

For convenience and simplicity, it is conservatively assumed that the radiation doses resulting from a feedline break are equal to the values specified in Figures 3D.5-4 and 3D.5-5 for a steam line break.

3D.5.5.1.4 Total Integrated Dose Specification

The applicable accident doses specified in equipment qualification data package Section 1.8 of Attachment A, have been derived based upon the time required to perform the specified safety function in the accident environment (Attachment A, Section 1.7) and the dose calculations described previously, subject to the following modifications:

- For equipment only required to provide trip or activation functions after accidents involving no release of radioactive material for at least one hour, the radiation dose is based on the normal dose rates (Table 3D.5-2).

3D.5.5.1.5 Temperature/Pressure Environments

The design basis events addressed are the loss of coolant accident, steam line break and feedwater line break. The WGOTHIC code is utilized to calculate the temperature and pressure conditions resulting from these breaks. To retain the option of qualifying equipment for each of these high-energy line break conditions, as applicable, separate environmental containment envelopes are specified for the higher irradiation/lower saturated temperature conditions of the loss of coolant accident as against the lower irradiation/short-term superheated temperature conditions associated with the steam line break. To limit the number of basic envelopes, this latter envelope is conservatively employed to define the containment environmental envelope following a feedline break.

Additionally, to facilitate AP1000 generic qualification and testing, the environmental envelopes have been combined to a single high-energy line break profile depicted in Figure 3D.5-8. This combined profile encompasses locations inside containment on the basis of the containment analyses for the AP1000 design. The profile, or a similar combined profile considering localized conditions resulting from the same events, is used to qualify equipment for any application or location for the AP1000 consistent with the NRC requirements in 10 CFR 50.49 and IEEE 308, 323, 603, and 627 when margin is added and via conformance with IEEE 323 guidelines.

Qualification tests to high-energy line break conditions are designed to address the applicable specified environment(s) with a margin of 15°F and 10 psi. Separate envelopes with margin are employed, or a combined loss of coolant accident/steam line break/feedwater line break envelope (Figure 3D.5-8, or its localized equivalent) may be employed for in-containment equipment

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qualification tests. Figure 3D.5-8 does not include margin from IEEE 323-1974, which will be incorporated in the environmental qualification programs. The simulated post-design basis event aging time-temperature profile (Figure 3D.5-8, or its localized equivalent from 24 hours to test conclusion) is defined consistent with the smallest value of activation energy applicable to the thermal aging sensitive components composing the test equipment or by a demonstrably conservative activation energy, as described in Attachment D.

3D.5.5.1.6 Chemical Environment

The high-energy line break qualification test for safety-related components will include chemical injection during the first 24 hours of the test, as appropriate, to simulate the reactor coolant system fluid. Initial pH is from 4 to 4.5, with the solution consisting primarily of boric acid. Components situated in locations that will not be exposed to reactor coolant fluid are not required to be subjected to chemical injection during testing.

Since there is no caustic containment spray in the AP1000, subsequent adjustments in pH may not be necessary for all tests. Sump solution chemistry is adjusted by release of alkaline chemistry, which will rise to 7.0 to 9.5 within a few hours of containment flooding. Conditions, representative of expected sump solution chemistry, are simulated for submerged equipment.

Margin in low pH value is not included, but is addressed by the continued injection through the first 24 hours. Margin in alkaline pH, where adjustment is necessary, is incorporated by a 10 percent increase in alkalinity.

3D.5.5.1.7 Submergence

Performance of equipment in a submerged condition is verified by a test that replicates the actual conditions with appropriate margin.

3D.5.5.2 High-Energy Line Break Accidents Outside Containment

For the majority of equipment located outside containment, the normal operating environment remains unchanged by a high-energy line break accident. As a consequence, qualification for such events is covered by qualification for normal conditions.

A limited amount of equipment located outside containment, near high-energy lines, could be subject to local hostile environmental conditions because of a high-energy line break outside containment. In this case, the equipment is qualified for the conditions resulting from events affecting its location and for which it is required to operate. Figure 3D.5-9 shows the design conditions for equipment that is required to perform throughout postulated events. Figure 3D.5-9 does not include margin from IEEE 323-1974, which will be incorporated in the environmental qualification programs. The maximum pressure for any event outside containment is 6 psi, with the exception of the MSIV compartments, which are designed for a 6.5 psi pressurization load.

3D.6 Qualification Methods

The recognized methods available for qualifying safety-related electrical equipment are established in IEEE 323. These are type testing, analysis, on-going qualification, or a combination of these methods. The choice of qualification method for a particular item of equipment is based upon many factors. These factors include practicability, size and complexity of equipment, economics, and availability of previous qualification to earlier standards.

The qualification method employed for each equipment type included under the AP1000 equipment qualification program is identified in the individual equipment qualification data packages whether by

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test (Attachment A, Section 3), analysis (Attachment A, Section 4), or by a combination of these methods. The AP1000 equipment qualification program may employ on-going qualification through the use of maintenance and surveillance. Guidance for such an approach is not included in this appendix.

3D.6.1 Type Test

The preferred method of environmental and seismic qualification of safety-related electrical and electromechanical equipment for the AP1000 equipment qualification program is type testing according to the guidelines and requirements of IEEE 323-1974 and 344-1987. Development of type test requirements are discussed in Section 3D.5. Documentation requirements and test plan development are addressed in Section 3D.7.

Additionally, qualification based on type tests performed according to IEEE 323 and 344, but not specifically for the AP1000, may be used as a qualification basis. Subsection 3D.6.5 of this appendix discusses the combination of qualification methods as they apply to the AP1000 equipment qualification program. (See Subsection 3D.6.5.1.)

3D.6.2 Analysis

The AP1000 equipment qualification program uses analysis for seismic qualification of equipment if the primary requirement is the demonstration of structural integrity during a seismic event. For equipment that performs an active or dynamic function, seismic qualification by analysis may also be used. (See Section E.3 of Attachment E.) However, the similarity between a qualified test unit and an as-supplied unit must be demonstrated unless otherwise justified. Subsection 3.9.2.2 describes the qualification requirements for safety-related mechanical equipment where a fluid pressure boundary is involved. For those mechanical components that are not pressure boundaries, analysis is performed in compliance with the applicable industry design standard. Where age-sensitive materials, such as gaskets and packing, are used in the assembly of mechanical equipment, the aging of these materials is normally evaluated based on an item-by-item review of the aging characteristics of the material. (See Subsection 3D.6.2.3.)

Requirements for documentation of the analysis are further treated in Section 3D.7.

3D.6.2.1 Similarity

Similarities among manufacturer's models provides several options for extending qualification to equipment without the need for a complete qualification test program.

A model series, such as that for a solenoid valve design, consists of numerous models that are identical in materials of construction and manufacturing process, but have minor variance in size, functional mode, operating voltage, electrical termination type, and mechanical interface sizing. Such variances in most cases have no impact on or relevance to the capability of the various models to perform acceptably under environmental or seismic (or both) qualification test conditions. Furthermore, the design basis document may apply equally to each member of the model series. In such cases, all members of the model series can be qualified by reference to the same testing or analysis.

There may be sufficient similarities between different model series to justify the case for similarity. A documented comparison addressing differences in the design for each, or apparent physical differences between members of each model series, may be sufficient to preclude the testing of one model series based on the testing of the other.

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Similarly, different models of a manufacturer's transmitters may be identical in some respects but different in others. The justification of similarity addresses the degree of similarity for critical characteristics. Differences that are not significant to qualification are also addressed for completeness. The mechanical and electrical functional modes and configurations must be the same. The materials of construction may be different, but must demonstrate equivalent performance. Other means of assuring accuracy may be necessary. When the devices are sufficiently similar in all attributes affecting qualification, qualification testing of one item can adequately cover another.

3D.6.2.2 Substitution

The objectives are to establish a degree of similarity and equivalence of performance for parts and materials that are different and, ultimately, to preclude the need for testing. For example, a gasket material is changed or a new type of capacitor is used because the original is no longer available, economical, or inadequate. Substitution of parts and materials is acceptable if comparison or analysis supports the conclusion that equipment performance is the same or better as a result. Consideration is given to characteristics of materials and the relative degree to which each is affected (or degraded) by the environmental parameters of qualification.

3D.6.2.3 Analysis of Safety-Related Mechanical Equipment

Environmental qualification of safety-related mechanical equipment is required to preclude common mode failures due to environmental effects of a design basis accident. Requirements are based on GDC 4 and 10 CFR 50, Appendix B. These criteria mandate that safety-related structures, systems, and components be designed to accommodate both normal and accident environmental effects.

3D.6.2.3.1 Equipment Identification

Safety-related mechanical equipment to be qualified is identified through the review of design basis documentation or the requirements of each safety-related fluid system. Only nonmetallic parts or subcomponents within the safety-related mechanical equipment are addressed for the effects of the postulated environments. The principal scope is typically valve "soft parts" that are critical to the valve safety-related function or pressure boundary integrity.

The types of components most frequently encountered in the mechanical equipment evaluations are discussed in Subsection 3D.6.2.3.3. Properties of materials that are assessed to provide confidence in safety-related function performance are also identified.

3D.6.2.3.2 Safety-Related Function

Safety-related functions and performance criteria are identified based on system and component classification. Structure, system, and component design basis documentation is reviewed to determine the specific safety functions. Components and subcomponents not involved in the equipment's safety-related function(s) are excluded from the qualification process if it is shown that their failures have no effect on the safety-related functions.

3D.6.2.3.3 Performance Criteria

Comprehensive performance criteria are established to satisfy the fundamental qualification requirements. The criterion for qualification is that the property of the nonmetallic material with regard to its application is not degraded during the specified qualified life to the point that the component is unable to perform its intended safety-related function. Properties for the component types listed in Table 3D.6-1 are discussed as examples.

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Gaskets and O-Rings

The capability of gaskets and O-rings to keep their shapes determines their ability to maintain pressure boundaries. When an O-ring or gasket loses its dimensional memory, it does not exert the necessary force on the confining surfaces. This could result in leakage. Compression set and elongation are good indicators of the dimensional memory of a material. They also reflect the extent of thermal aging and radiation-induced cross-linking. A compression set of 50 percent is chosen as a conservative end-of-life criterion even though leakage is unlikely to occur until the component takes a compression set of greater than 75 percent. When compression set data is not available for a gasket or O-ring, elongation at break is the material property evaluated because like compression set, it is an indication of dimensional memory and cross-link.

Diaphragms

Diaphragms must remain flexible yet maintain their dimensional memory throughout the estimated mechanical cycles. Retention of elongation or tensile strength is evaluated for radiation and thermal aging.

Diaphragm Support Sheets

The diaphragm support sheet prevents puncture and tearing of the diaphragm. It is not considered critical to the operability of diaphragm valves. The best indication of radiation damage and thermal aging to diaphragm support materials is retention of elongation.

Lubricants

One of the primary functions of oils and greases is to maintain a thin film barrier between moving parts to reduce friction and wear. Irradiation reduces the capability of a lubricant to perform this function by decreasing viscosity in oils and increasing penetration in greases and finally converting lubricants to hard, brittle solids if exposure is severe.

Worm Gears

Worm gears must be capable of transmitting forces without excessive deformation. Flexural strength is the material property chosen to evaluate radiation and thermal aging resistance of worm gears.

3D.6.2.3.4 Identification of Service Conditions

Service conditions are identified for the normal and accident conditions. The general design of equipment permits exemption of environmental parameters such as pressure and humidity. Where critical parts are totally enclosed and not directly exposed to potentially harsh environments, the effects of humidity and chemical spray are not addressed. The degradation of mechanical equipment due to thermal and radiation aging is typically more severe than the possible degradation due to other environments. Since most mechanical equipment interfaces with process fluid, the effect of the fluid on the environmental conditions (temperature, radiation, and chemical) is considered.

3D.6.2.3.5 Description of Potential Failure

Where applicable, potential failure modes are identified and assessed for the equipment. Assessment of equipment aging mechanisms is essential to determine if aging has a significant effect on operability. This assessment provides confidence that significant aging mechanisms are unlikely to contribute to common-mode failures adverse to the safety-related function of equipment.

3D.6.2.3.6 Qualification Procedure

The nonmetallic materials identified are evaluated to the normal and accident environmental parameters. The evaluation procedure includes the following steps:

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- Identification of the environmental effect on the material properties
- Performance of a thermal aging analysis
- Determination of the environmental effects on the equipment safety-related function.

These are detailed in the equipment qualification data package of Attachment A, Section 4.

3D.6.2.3.7 Performance Criteria

The nonmetallic subcomponents of the mechanical equipment:

a. are acceptable for the plant environment by exhibiting threshold radiation values above the postulated environmental condition, and

b. are acceptable for the plant environment by exhibiting a maximum service temperature above the maximum postulated environmental, and

c. does exhibit a service life sufficient to survive the accident duration, or

d. instead of a, b, and c, are acceptable for the plant environment by analysis that demonstrates that the safety-related function of the component is not compromised.

The mechanical equipment is considered qualified if subcomponents important to the safety function are acceptable.

Nonmetallic subcomponents not meeting the criteria must have a replacement interval specified to maintain the qualification of the affected equipment. The replacement interval is determined by analysis and documented.

3D.6.2.3.8 Equipment Qualification Maintenance Requirements

The maintenance requirements resulting from the activities described herein are identified. The qualification maintenance requirements are based on the following:

- Qualification evaluation results (for example, periodic replacement of age-susceptible parts before the end of their qualified lives)
- Equipment qualification-related maintenance activities derived from the qualification report(s)
- Vendor recommended equipment qualification maintenance. Vendor recommended maintenance is included if it is required in order to maintain qualification.

3D.6.2.3.9 Qualification Documentation

The qualification of the mechanical equipment to the postulated environments is documented in an auditable form. See Section 3D.7.

3D.6.3 Operating Experience

Qualification by experience is not employed in the AP1000 equipment qualification program as a method of qualification.

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3D.6.4 On-Going Qualification

The AP1000 equipment qualification program may employ on-going qualification through special maintenance and surveillance activities. However, this method of qualification is not suitable as a sole means for qualifying equipment for design basis event conditions. On-going qualification, as a method, is used exclusively for safety-related equipment located in a mild environment area. Such use requires supplementary test, or analysis to address equipment operability and performance during and after a seismic design basis event.

Documentation requirements for qualification that includes on-going qualification as a method are developed to conform with NRC guidance provided in Regulatory Guide 1.33, Revision 2.

3D.6.5 Combinations of Methods

Qualification by a combination of the preceding methods may be used under the AP1000 equipment qualification program.

3D.6.5.1 Use of Existing Qualification Reports

Pre-existing qualification programs and documents are used only if the seismic test program satisfies the guidelines of IEEE 344-1987 and the environmental qualification program satisfies the guidelines of IEEE 323-1974.

Qualification test and analysis reports conforming to those IEEE Standards, but not specifically performed to the AP1000 equipment qualification program parameters, may be acceptable as qualification bases. In such cases, supplementary qualification efforts described in Subsections 3D.6.2, 3D.6.3, and 3D.6.4 of this appendix may be required to validate acceptability under the AP1000 equipment qualification program. Justifications are documented as analyses, and appear in equipment qualification data package, Section 4. (See Attachment A.)

3D.6.5.1.1 Aging

Past qualification tests may provide sufficient basis to preclude new aging simulation testing as part of the AP1000 program. Also, simulation of both electrical and mechanical operational cycling may be waived where existing data demonstrates equipment durability greatly in the excess of the estimated number of operating cycles for Class 1E service. Application of past qualification and other tests is considered in the development of test plans and analysis procedures. The bases and justification is provided in qualification documentation for cases where applicable aging parameters are omitted from the test sequence.

3D.6.5.1.2 Seismic

Seismic qualification generally relies on analyses and justification to verify the adequacy or applicability of generic testing to a particular installed configuration of similar equipment. Analytical methods and documentation guidelines of IEEE 344-1987, as supplemented by Regulatory Guide 1.100, Revision 2, address these needs. Attachment E of this appendix provides the AP1000 equipment qualification program requirements regarding seismic qualification.

3D.6.5.1.3 High-Energy Line Break Conditions

Typically, existing qualification tests address conditions of high-energy line break environments occurring inside containment. These are used where it is demonstrated that the qualification envelops the applicable requirements.

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3D.7 Documentation

The AP1000 equipment qualification program documentation consists principally of four types of documents:

- "Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment" is the generic program "parent" document. It describes the methods and practices employed in the AP1000 equipment qualification program.

- Equipment qualification data packages are "daughter" documents to the methodology. Each is a summary of the qualification program for a specific equipment type (for example, a particular model or design series of a manufacturer, an as-provided system, or a family of equipment tested as a set). The equipment qualification data package defines the qualification program objectives, methods, applicable equipment performance specifications, and the qualification plan. It provides a summary of the results. Other engineering tests that are outside the scope of 10 CFR 50.49, such as Electromagnetic Compatibility (EMC) testing and ASME QME-1 qualification testing of power-operated active valves, may be included in the EQDPs for convenience, but are not addressed further in this Appendix.

- Equipment Qualification Summary Reports (EQSRs) are the reports that present specific methods used during the qualification process and the results of that process.

- Seismic analysis reports for seismic Category I and Category II mechanical equipment that is qualified by analysis only are not required to be documented in the EQDP format. Environmental qualification of this equipment is not required due to non-existence of age-sensitive components as identified in Table 3D.6-1. The seismic qualification documentation is required to meet the documentation requirements of IEEE-344-1987 or applicable industry standard, which is the basis of qualification (i.e., ASME Section III). The seismic analysis documentation should include the purpose of the analysis, general equipment description, equipment mounting and interface requirements, analysis methods and codes, acceptance criteria, seismic loading conditions, assumptions, and conclusions.

The equipment qualification data packages are developed separate from the parent document. Similarly, the equipment qualification summary reports are developed separate from the equipment qualification data packages. Equipment qualification summary reports used in the AP1000 equipment qualification program may include existing reports of testing or analysis that comply with the relevant aspects of this methodology. Information necessary to demonstrate the equipment's capability to perform its intended safety-related function(s) while exposed to normal, abnormal, accident, and post-accident environments is provided in or referenced by the equipment qualification data package. If maintenance, refurbishment, or replacement of the equipment is necessary to provide confidence in the equipment's capability to perform its safety function, this information is also included in the equipment qualification data package. Data, in raw form, cited in the equipment qualification data packages or equipment qualification summary reports is available for audit for the life of the plant.

3D.7.1 Equipment Qualification Data Package

Attachment A contains a general example of the equipment qualification data package format. To the extent practical, each equipment qualification data package consists of the following elements:

Section 1 – Specifications
Section 2 – Qualification Program
Section 3 – Qualification by Test
Section 4 – Qualification by Analysis
Section 5 – Qualification Program Conclusions

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Some organization and numbering differences are adopted where appropriate to accommodate the unique attributes of the equipment and the specific qualification requirements and methods to demonstrate the objective of the qualification program. The following paragraphs discuss the general content contained in the equipment qualification data packages.

3D.7.2 Specifications

Section 1 of Attachment A contains the performance specification of the equipment. This specification establishes the necessary parameters for which qualification is demonstrated. The basic criterion for qualification is that the safety-related functional requirements defined in Section 1 are successfully demonstrated, with margin, under the specified environmental conditions.

The following sections define the bases on which the parameters contained in Section 1 are selected.

3D.7.2.1 Equipment Identification

Equipment is identified in Section 1.1 of Attachment A by manufacturer, model or model series, and reference to other documents describing or depicting its construction, configuration, and modifications that are uniquely necessary after manufacture to its application in the AP1000 plant design. Model series (for example, a limit switch design family) and other pertinent details on items making up the equipment type qualified are compiled as a table and referenced from this section.

3D.7.2.2 Installation Requirements

So that the qualification represents the in-plant condition, the method of installation, as specified in Section 1.2 of Attachment A, is in accordance with the supplier's installation instructions. Differences unique to safety-related applications in the AP1000 design are included, with appropriate reference to drawings, technical manual supplements, or mandatory modification packages.

3D.7.2.3 Electrical Requirements

The pertinent electrical requirements are specified (for example, voltage, frequency, load) in this section. Also included is any variation in the defined parameters for which the equipment is to perform its specified functions (Section 1.3 of Attachment A).

3D.7.2.4 Auxiliary Devices

Sometimes the equipment qualified relies upon the operation of auxiliary devices in order to perform the specified safety-related functions. These devices are identified in Section 1.4 of Attachment A. Auxiliary devices include items such as electrical conductor seal assemblies that, in service, become part of the qualified equipment's pressure boundary. The applicable equipment qualification data package for the auxiliary device(s) is specified, if known.

3D.7.2.5 Preventive Maintenance

Preventive maintenance (Section 1.5 of Attachment A) to be performed includes maintenance or periodic activities assumed as part of the qualification program or necessary to support qualification. Only those activities that are required in order to support qualification or the qualified life are specified. The manufacturer's recommended maintenance activities are considered to determine that there is no adverse impact to qualification or the maintenance of qualified life. Likewise, manufacturer's recommendations for maintenance, replacement, or surveillance activities necessary to support operability are identified, or reference is made to the appropriate technical manual or supplements.

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"None" means that maintenance is not essential to qualification or the qualified life of the equipment. However, this should not preclude development of a preventive maintenance program designed to enhance equipment performance and to identify unanticipated equipment degradation as long as such a program does not compromise the qualification status of the equipment. Surveillance activities may also be considered to support a basis for and a possible extension of the qualified life.

3D.7.2.6 Safety Functions

Section 1.6 of Attachment A addresses the known safety functions of the equipment, including summarizing the basis for the aspects of the safety functions that are relevant to demonstrating the equipment's qualification. This section should address the plant conditions under which the safety function is performed, the transients or accidents required for equipment performance, a characterization of the equipment function (e.g., how the safety function is performed, whether the equipment performs an active or passive function, etc.), and other information relevant to the qualification, as applicable.

3D.7.2.7 Performance Requirements

Section 1.7 of Attachment A contains a tabulation of performance requirements for each safety-related function for which the equipment is qualified. Several such sections or tables may be necessary when the equipment is qualified for applications where the performance requirements vary. Performance requirements are stated regarding the normal and abnormal environmental conditions applicable at the location where the equipment is installed. Similarly, each design basis event and the subsequent post-event period is included in the table.

Margin is not included in the performance requirements except by conservatism in their determination.

3D.7.2.8 Environmental Conditions

Within each set of performance requirements, a set of environmental parameters is specified in Section 1.8 of Attachment A, also in tabular form. Parameters are based on the equipment location and function and include those addressed in other sections of this appendix.

Margin is not included in the environmental parameters except by conservatism in their determination. The objective is to provide the baseline reference onto which margin is added.

3D.7.3 Qualification Program

An overview of the qualification program and its objective is presented in narrative form in Section 2. Attachment A includes a table to be completed as a graphic reference. As it is assumed that tests, analyses, or some combination of the two are the principal methods of qualification, columns are included for each. Other methods, when used, are summarized in brief notes appended to the table.

References to reports of testing, analysis, or other information considered in support of the qualification program are compiled in Section 2 of Attachment A. This includes any technical manuals, drawings, and supporting material cited or referenced by text throughout the equipment qualification data package.

3D.7.4 Qualification by Test

Qualification by test is selected as the primary method of qualification for complex equipment not readily amenable to analysis or for equipment required to perform a safety-related function in a high-energy line break environment. The proposed test plan is identified in Section 3 of Attachment A.

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Where supportive analysis is claimed as an integral part of the qualification program, cross reference is provided to Attachment A, Section 4 for those aspects of the qualification not covered by the test plan. The following sections establish the basis on which the information specified in Section 3 is selected.

3D.7.4.1 Specimen Description

The equipment qualified is identified, including the baseline design document number/reference, where applicable, the equipment type, manufacturer and model number, in Section 3.1 of Attachment A. When testing a model series (or equipment families), the representative items tested are clearly identified. The basis of their representation should be included.

Section 3.1 is primarily intended to identify test specimens used in a test supporting the qualification program. But it also discusses the specimens considered for other methods used in the qualification program.

3D.7.4.2 Number Tested

The test program is based upon selectively testing a representative number of components according to type, size, or other appropriate classification, on a prototype basis. The number of items of equipment representative of the equipment type that are tested is defined in Section 3.2 of Attachment A.

3D.7.4.3 Mounting

The method of mounting the equipment for the test is identified in Section 3.3 of Attachment A. The in-plant installation requirements, as specified by the supplier under Section 1.2 of Attachment A, are fully represented.

3D.7.4.4 Connections

The equipment connections necessary to demonstrate safety-related functional operability during testing are identified in Section 3.4 of Attachment A. This includes items that are part of the installed configuration, but are not part of the test apparatus.

Particularly important are items that are included by "practice of good workmanship," such as pipe thread sealant. Another example is the use of electrical connection sealing materials. Where these items are included in the testing, they become factors in the performance of the equipment, especially under aggressive or adverse environmental conditions. Their thermal degradation and sensitivity to irradiation and chemistry environments are considered in the qualification program, both for impact to equipment performance under harsh conditions and for their contribution to equipment qualified life.

3D.7.4.5 Test Sequence

The preferred test sequence specified in Attachment A, Section 3.5 is the one recommended by IEEE 323-1974. The qualification test sequence used is specified in Section 3.6 of Attachment A. Justification for departures or additions to the preferred test sequence are included. Also, any portion of the test sequence that is supplemented by analysis or other methods is identified for completeness.

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3D.7.4.6 Simulated Service Conditions

The service conditions simulated by the test plan are identified in Attachment A, Section 3.7. In general, the parameters employed are selected to be equal to (normal and abnormal) or have margin (accident and post-accident) with respect to the specified service conditions of Attachment A, Section 1.8. Criteria for margin is detailed in Subsection 3D.4.8.

3D.7.4.7 Measured Variables

The parameters measured during the specified test sequence in order to demonstrate qualification for the performance specification (Attachment A, Section 1) are individually listed in Section 3.8 of Attachment A. This section is formatted to include parameters relevant to the test environment and the electrical and mechanical characteristics of equipment operation. Other characteristics unique to a particular test or equipment type are included, when applicable.

3D.7.4.8 Type Test Summary

Section 3.9 of Attachment A provides a narrative summary of the qualification tests and results. The applicable test reports are provided as references in Attachment A, Section 2. Test data is available for audit throughout the operation of the plant.

Each test report referenced by the equipment qualification data package should contain information cited in the preceding section, as well as the following:

- The test facility, location, and a description of the test equipment used. Monitoring equipment should have current calibration traceable to the National Institute of Standards and Technology (NIST).
- Test setup and specimen installation details.
- Description of the mounting conditions simulated during the test program and any difference between them and the mounting details shown on the equipment drawings, with qualification of any differences found.
- Description of limitations on the use and mounting of the qualified equipment found as a result of the qualification test program.
- Description of the test method and the justification that the method meets the specification test requirements.
- Description of operational settings used to demonstrate functional operability and any limitation imposed on them.
- Test records (for example, test response spectra, time history; accident transient parameters - temperature, pressure). This includes performance and operability test results, inspection results, and the monitored test and specimen and calibration records of instruments used.
- Record of compliance of test results with the seismic qualification criteria.
- Description of anomalies found during the test program, and their resolution(s).

Potential aging mechanisms resulting from significant in-service thermal, electrical, mechanical, radiation, and vibration sources are identified in Section 3.9 of Attachment A. When aging is

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addressed as part of the test sequence, the method employed for aging the equipment is indicated and is chosen to conservatively simulate the potential aging effects resulting from the operating cycles and environmental conditions specified in Attachment A, Section 1. The methods employed to address each of the potential aging mechanisms are discussed.

3D.7.5 Qualification by Analysis

Qualification by this methodology does not rely solely on analyses. Generally, analysis is permitted to support qualification testing or to establish that testing of other sufficiently similar equipment can be cited to establish or extend the qualification of equipment covered by the equipment qualification data package.

Section 4 of Attachment A is formatted to conform with the recommendations of IEEE 323-1974. Each subsection addresses a particular analysis if more than one is performed to support qualification. Not all subsections identified in the sample format apply to any particular analysis. Documentation of analyses demonstrating or supporting seismic qualification conforms with the guidelines of Attachment E and the recommendations of IEEE 344-1987.

3D.7.6 Qualification by Experience

This method of qualification is not used.

3D.7.7 Qualification Program Conclusions

Section 5 of Attachment A summarizes the conclusions of the qualification program, including and addressing methods employed and conditions upon which qualification of the equipment is based. Details regarding each aspect of simulated aging are addressed distinctly, with conclusions as to the life-limiting aspects clearly stated.

Conclusions for each design basis event are summarized. Generally, these are combined as either design basis event seismic and design basis event environmental.

3D.7.8 Combined License Information

Not used.

3D.8 References

1. IEEE-323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

2. IEEE-344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

3. IEEE-627-1980, "IEEE Standard for Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations."

4. NUREG/CR-3156, "A Survey of the State-of-the-Art in Aging of Electronics with Application to Nuclear Plant Instrumentation."

5. EPRI NP-1558, Project 890-1, "A Review of Equipment Aging Theory and Technology."

6. NUREG/CR 2156, "Radiation Thermal Degradation of PE and PVC: Mechanism of Synergism and Dose Rate Effects," Clough and Gillen, June 1981.

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7. NUREG/CR 2157, "Occurrence and Implication of Radiation Dose Rate Effects for Material Aging Studies," Clough and Gillen, June 1981.

8. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants – Final Report," L. Soffer, et al., February 1995.

9. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

Note: See Subsection 3D.4.1.1 for other IEEE references.

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Table 3D.4-1
Typical Mild Environment Parameter Limits

Parameter	Limit	Notes
Temperature	≤ 120°F	
Pressure	Atmospheric	(Nominal)
Humidity	30 – 65% ≤ 95%	(Typical) (Abnormal)
Radiation	≤ 10^4 rads gamma ≤ 10^3 rads gamma	 (IC electronics and microprocessors)
Chemistry	None	
Submergence	None	

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Table 3D.4-2
Equipment Post-Accident Operability Times

Equipment	Required Post-Accident Operability	
Equipment necessary to perform trip functions	5 minutes	(Envelops trip time requirements)
Equipment located outside containment, is accessible, and can be repaired, replaced, or recalibrated	2 weeks	
Equipment located inside containment that is inaccessible and is required for post-accident monitoring	4 months	(This number is based on an acceptable amount of time to be repaired, replaced, or recalibrated, or for an equivalent indication to be obtained.)
Equipment located inside containment, is inaccessible, or cannot be repaired, replaced, recalibrated or equivalent indication cannot be obtained	1 year	(The required post-accident operability time during submergence may be shorter if the consequences of equipment failure after submergence are deemed acceptable and/or the monitored component is administratively controlled.)
Equipment in a location that will have a mild environment following an accident or equipment that does not provide information for a Type A, B, or C primary post-accident monitoring parameter	Various	(Specific as to function, maximum of 1 year)

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Table 3D.4-3
AP1000 EQ Program Margin Requirements

Condition	Parameter	Required Margin	Notes
NORMAL:	Aging	+10%	+10% time margin, +10% radiation and/or selection of conservative test parameters. Comply with guidance of Subsection 3D.4.8.2
ABNORMAL:	Temperature/ Humidity		Margin is in "time" at abnormal test extremes.
	Pressure	None	Nominally atmospheric.
	Radiation	+10%	Include in aging doses, if applicable.
	Voltage & Frequency	+/- 10%	Simulated during temperature/humidity test.
ACCIDENT:	Transient Temperature and Pressure		Temperature (+15°F) and pressure (+10 psig peak) margins added to transient profile.
	Chemical effects	+10%	Bounding concentration of chemical additives +10%. Containment sump chemistry conditions reflect change over time from acidic to alkaline. (See Subsection 3D.4.8.5)
	Radiation	+10%	Added to calculated total integrated dose.
	Submergence	Note 1	Generally, precluded by design.
	Seismic/ Vibration	+10%	Of acceleration at equipment mounting point for either SSE or line-mounted equipment vibration. (See Subsection 3D.4.8.4.)
	Post-accident Aging	+10%	In time demonstrated via Arrhenius time/temperature relationship calculation.

Note:
1. Margin in submergence conditions is achieved by increases in temperature (+15°F), pressure (+10%), and chemistry.

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Table 3D.5-1 (Sheet 1 of 3)
Normal Operating Environments

(Notes 1 and 2)

Location/Parameter	Normal Range	Notes
Zone 1 – Containment (Room numbers: 11000 through 11999)		
Temperature	50 - 120°F (All rooms except 11105 and 11205) 50 - 135°F (Rooms 11105 and 11205)	
Pressure	-0.2 - +1.0 psig	
Humidity	0 - 100%	
Radiation	see Table 3D.5-2	
Chemistry	None	
Zone 2 - Auxiliary Building - Non-Radiological - I&C, DC Equipment, RCP Switchgear & Battery rooms, etc. (Room numbers: 12101, 12102, 12103, 12104, 12105, 12111, 12112, 12113, 12201, 12202, 12203, 12204, 12205, 12207, 12211, 12212, 12213, 12300, 12301, 12302, 12303, 12304, 12305, 12311, 12312, 12313, 12405, 12411, 12412, 12501, Stairwell S05, and 12505)		
Temperature	67 - 77°F (All rooms except 12405, 12501, and 12505) 50 - 85°F (Rooms 12405, 12501, Stairwell S05, and 12505)	
Pressure	Slightly positive to slightly negative	
Humidity	10 - 60%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	
Zone 3 - Auxiliary Building - Non-Radiological - Main Control Room (Room number: 12400, 12401)		
Temperature	67 - 78°F	
Pressure	Slightly positive	
Humidity	25 - 60%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	
Zone 4 - Auxiliary Building - Non-Radiological - Accessible (Room numbers: 12321, 12421, 12422, 12423)		
Temperature	50 - 85°F (Rooms 12321, 12421) 50 - 105°F (Rooms 12422, 12423)	
Pressure	Slightly positive	
Humidity	10 - 60%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	

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Table 3D.5-1 (Sheet 2 of 3)
Normal Operating Environments

(Notes 1 and 2)

Location/Parameter	Normal Range	Notes
Zone 5 - Auxiliary Building - Non-Radiological - MSIV Compartments (Room numbers: 12404, 12406, 12504, 12506)		
Temperature	50 - 130°F	
Pressure	Atmospheric	
Humidity	10 - 100%	
Radiation	$<10^4$ rads gamma	
Chemistry	None	
Zone 6 - Auxiliary Building - Radiological - Inaccessible (Room numbers: 12154, 12158, 12162, 12163, 12166, 12167, 12171, 12172, 12254, 12255, 12256, 12258, 12262, 12264, 12265, 12354, 12362, 12363, 12365, 12371, 12372, 12373, 12374, 12454, 12462, 12463)		
Temperature	50 - 130°F	
Pressure	Slightly negative to atmospheric	
Humidity	10 - 100%	
Radiation	See Table 3D.5-2	
Chemistry	None	
Zone 7 - Auxiliary Building - Radiological - Accessible (Room numbers: 12151, 12152, 12153, 12155, 12156, 12161, 12168, 12169, 12241, 12242, 12244, 12251, 12252, 12261, 12268, 12271, 12272, 12273, 12274, 12275, 12341, 12343, 12344, 12345, 12351, 12352, 12361, 12451, 12452, 12461, 12553, 12554, 12555, 12556, 12561, 12651)		
Temperature	50 - 104°F (All rooms except 12555) 60 - 80°F (Room 12555)	
Pressure	Slightly negative	
Humidity	10 - 100%	
Radiation	See Table 3D.5-2	
Chemistry	None	
Zone 8 - Turbine Building (Room numbers: 20300 through 20799)		
Temperature	50 - 105°F	
Pressure	Atmospheric	
Humidity	10 - 100%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	

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Table 3D.5-1 (Sheet 3 of 3)
Normal Operating Environments

(Notes 1 and 2)

Location/Parameter	Normal Range	Notes
Zone 9 - Auxiliary Building - PCS Valve Room and Upper Annulus (Room number: 12541, 12701)		
Temperature	70 - 120°F (Room 12701) -40 - 115°F (Room 12541)	
Pressure	Atmospheric	
Humidity	10 - 100%	
Radiation	See Table 3D.5-2	
Chemistry	None	
Zone 10 - Auxiliary Building - Non-Radiological - Valve/Piping Penetration Room with SG Blowdown (Room number: 12306)		
Temperature	50 - 105°F	
Pressure	Slightly positive	
Humidity	10 - 60%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	
Zone 11 - Auxiliary Building - Radiological - Fuel Handling Area (Room numbers: 12472, 12562, 12563, 12564)		
Temperature	50 - 105°F	
Pressure	Slightly negative	
Humidity	10 - 100%	
Radiation	See Table 3D.5-2	
Chemistry	None	

Notes:
1. Room numbers - see Section 1.2, General Arrangement drawings.
2. Relative humidity is not controlled except in the main control room.

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Table 3D.5-2
60-Year Normal Operating Doses

Location	Gamma Dose Rate (Rad air hour)	60-Year Gamma Dose (Rads air)
Inside Containment:		
RCS Pipe - Center	1.9×10^{3}	1.0×10^{9}
RCS Pipe - ID	1.1×10^{3}	5.7×10^{8}
RCS Pipe - OD (contact)	7.8×10^{1}	4.1×10^{7}
RCS Pipe - General Area(b)	4.0×10^{1}	2.1×10^{7}
Outside CA01 Excluding Rooms 11104 and 11204	<0.45	$<2.4 \times 10^{5}$
Adjacent to Reactor Vessel Wall	$\leq 3.6 \times 10^{4}$	$\leq 1.9 \times 10^{10}$(a)
Spent Resin and Spent Fuel Handling Areas(c)	--	$<1 \times 10^{9}$
IRWST	4.2×10^{1}	2.2×10^{7}
Outside Containment:		
Penetration Area	--	$<2 \times 10^{7}$
Pump Cubicles	--	$<2 \times 10^{7}$
Radioactive Waste Area	--	$<2 \times 10^{7}$
Radwaste Tank Cubicles	--	$<5 \times 10^{7}$
Other General Areas Not Under Radiation Control	--	$<1 \times 10^{4}$
Spent Resin and Spent Fuel Handling Areas(c)	--	$<1 \times 10^{9}$

Notes:
a. 60-year neutron fluence for E>1 MeV is 4.6×10^{18} n/cm^2
b. 12 inches from RCS pipe OD
c. Total gamma dose outside of tanks

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Table 3D.5-3
Abnormal Operating Environments
Inside Containment

Conditions/Parameter	Abnormal Extreme	Duration	Notes
Group 1 (150°F) Abnormal Events			
Temperature	150°F	4 hours	Note 1
Pressure	2 psig	4 hours	Note 1
Humidity	100%	4 hours	Note 1
Radiation	Same as normal		
Chemistry	None		
Submergence	None		
Group 2 (250°F) Abnormal Events			
Temperature	250°F	30 days	Note 1
Pressure	18 psig	30 days	Note 1
Humidity	100%	30 days	Note 1
Radiation			Note 2
Chemistry	None		
Submergence	None		

Notes:
1. Parameter value is not maximum for full duration.
2. Minor increase over normal radiation conditions expected.

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Table 3D.5-4 (Sheet 1 of 3)
Abnormal Operating Environments
Outside Containment

Conditions/Parameter	Abnormal Extreme	Duration	Notes
Zone 2			
Temperature (Rooms 12101, 12102, 12103, 12104, 12105, 12202, 12204)	120°F max	7 days	Note 3
Temperature (Rooms 12113, 12201, 12203, 12205, 12207, 12212, 12301, 12302, 12303, 12304, 12305, 12312, 12313, 12411, 12412)	120°F max	7 days	Note 3
Temperature (Rooms 12111, 12112, 12211, 12213, 12300, 12311)	120°F max	7 days	
Temperature (Rooms 12405, 12501, 12505, Stairwell S05)	120°F max	7 days	
Pressure	Atmospheric		
Humidity	40 – 95%		Note 2
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 3			
Temperature	Figure 3D.5-1 (Sheet 1)	7 days	
Pressure	Atmospheric		Note 1
Humidity	5 – 95%		Note 2
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 4			
Temperature	120°F max	10x4 hrs	
Pressure	Slightly positive		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 5			
Temperature	150°F max	10x4 hrs	
Pressure	Atmospheric		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Piping Rupture			Note 6
Submergence and Spray	12504, 12506 0" 12404, 12406 36"	2 weeks	Notes 5, 7, 8

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Table 3D.5-4 (Sheet 2 of 3)
Abnormal Operating Environments
Outside Containment

Conditions/Parameter	Abnormal Extreme	Duration	Notes
Zone 6			
Temperature	150°F max (Room 12256) 140°F max (Other Rooms)	10x4 hrs	
Pressure	Atmospheric		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Pipe Rupture			Note 6
Submergence and Spray	12259 3" 12255 36" 12258 132"	2 weeks	Notes 5, 7, 8
Submergence	12262 36" 12254 60" 12158 76.5" 12154, 12162, 12163, 12171, 12172 168" 12264 249" 12265 288"	2 weeks	Notes 7, 8
Zone 7			
Temperature	200°F max (Rooms 12241, 12341, 12343, 12344, 12345) 155°F max (Room 12252) 150°F max (Rooms 12451, 12554) 130°F max (Room 12561) 114°F max (Other Rooms)	10x4 hrs	
Pressure	Slightly negative		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Pipe Rupture			Note 6
Submergence and Spray	12241, 12244 36"	2 weeks	Notes 5, 7, 8
Submergence	12452 30" 12242, 12251, 12252, 12261, 12271, 12272, 12273, 12274, 12275 36" 12268 54" 12168 90" 12341 93" 12151, 12153, 12155 12156 168" 12169 173.5" 12152, 12161 180"	2 weeks	Notes 7, 8

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Table 3D.5-4 (Sheet 3 of 3)
Abnormal Operating Environments
Outside Containment

Conditions/Parameter	Abnormal Extreme	Duration	Notes
Zones 8			
Temperature	Same as normal		
Pressure	Same as normal		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 9			
Temperature (Room 12701)	40 to 120°F	3 days	
Temperature (Room 12541)	-40 to 150°F		
Pressure	Same as normal		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 10			
Temperature	140°F max	10x4 hrs	
Pressure	Same as normal		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Pipe Rupture			Note 6
Submergence and Spray	12306 36"	2 weeks	Notes 5, 7, 8
Zone 11			
Temperature	212°F max	7 days	
Pressure	Atmospheric		Note 4
Humidity	100%		
Radiation	Same as normal		
Chemistry/Submergence	None		

Notes:

1. Main control room air pressure is maintained above a nominal value of atmospheric during accident conditions to prevent radioactive contaminant entry.
2. Figure 3D.5-1 Sheets 1 and 2 have two curves post-72 hours. The high curve represents the introduction of outside air that is high temperature, low humidity. The low curve represents the introduction of outside air that is low temperature, high humidity. The EQ Programs will include both of these extremes.
3. Test environments resulting from rooms with equipment supplied by 24- and 72-hour batteries are shown on Figure 3D.5-1, Sheet 2, for the dc equipment rooms 12203 and 12207 and for the I&C rooms 12302 and 12304. The 24-hour battery is disconnected at 24 hours. The 72-hour battery is not disconnected. Environments resulting from rooms with equipment supplied by 24-hour batteries only, – that is, dc equipment rooms 12201 and 12205 and I&C rooms 12301 and 12305 – are enveloped by the environments shown on Figure 3D.5-1, Sheet 2. The environments shown on Figure 3D.5-1, Sheet 2 also apply to the dc equipment rooms containing batteries (i.e., the IDS battery rooms 12101, 12102, 12103, 12104, 12105, 12202 and 12204).
4. A relief panel is designed to open when the fuel handling area temperature exceeds 165°F.
5. This room may also be subjected to spray or wetting above the submergence elevation as a result of pipe ruptures.
6. Equipment in these areas is addressed using the criteria outlined in Subsection 3.11.1.2.
7. Submergence in this table is specified for various rooms, and the distances tabulated represent the bounding flood height above the floor elevation in a given zone or room.
8. Spray and submergence as applicable, based upon the methods described in Subsection 3.4.1 and Section 3.6. Equipment affected by spray or submergence required to undergo qualification is reflected in the requirements contained in Table 3.11-1.

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Table 3D.5-5
Accident Environments

(See Table 3D.5-1 for environmental zones)

Zone 1 – Inside Containment	
Temperature and pressure Submergence as applicable up to elevation 110'-6"	See Figure 3D.5-8 for most locations. Other locations may use a localized equivalent of this figure.
Radiation	See Figures 3D.5-2 through 3D.5-5.
Zones 2, 4 – Outside Containment	
Temperature (Zone 2)	Maximum: Same as Abnormal. See Table 3D.5-4. Rooms 12111, 12112, 12211, 12213, 12300, 12311 Minimum: 60°F
Temperature (Zone 4)	Same as Abnormal. See Table 3D.5-4.
Pressure	Same as Abnormal. See Table 3D.5-4.
Radiation	TID 1 year ≤ 1E+04 Rads maximum.
Zones 3, 8 – Outside Containment	
Temperature and pressure	Same as Abnormal. See Table 3D.5-4.
Radiation	TID 1 year ≤ 1E+03 Rads maximum.
Zone 5 – Outside Containment – MSIV Compartments	
Temperature	See Figure 3D.5-9.
Radiation	TID 1 year ≤ 3E+05 Rads maximum. See also Figures 3D.5-4 and 3D.5-5.
Zones 6, 11 – Outside Containment	
Temperature (Zone 6)	Same as Abnormal. See Table 3D.5-4.
Temperature (Zone 11)	Maximum: Same as Abnormal. See Table 3D.5-4. Minimum: 40°F
Pressure	Same as Abnormal. See Table 3D.5-4.
Radiation	Same as Abnormal. See Table 3D.5-4.
Zone 7 – Outside Containment	
Temperature	
Rooms 12241, 12341, 12343, 12344, and 12345	200°F max
Rooms 12156, 12244	135°F max
Other Rooms	Same as Abnormal with exceptions identified below. See Table 3D.5-4. Room 12555 Minimum: 50°F Room 12151 Maximum: 120°F
Pressure	Same as Abnormal. See Table 3D.5-4.
Radiation	TID 1 year ≤ 2E+07 Rads maximum.
Zone 9 – Outside Containment	
Pressure	Same as Abnormal. See Table 3D.5-4.
Temperature	
Room 12541	-40 - 180°F
Room 12701	40 - 125°F
Radiation	TID 1 year ≤ 6E+06 Rads maximum.
Zone 10 – Outside Containment	
Temperature	See Figure 3D.5-9.
Radiation	See Figures 3D.5-4 and 3D.5-5.

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Table 3D.6-1
Mechanical Equipment Components Requiring Environmental Qualification

Component	Material Property
Gaskets	Compression set/elongation
O-rings	Compression set/elongation
Diaphragms	Elongation/tensile strength
Diaphragm support sheets	Tensile strength/elongation
Lubricant	Viscosity/penetration
Worm gear	Flexural strength

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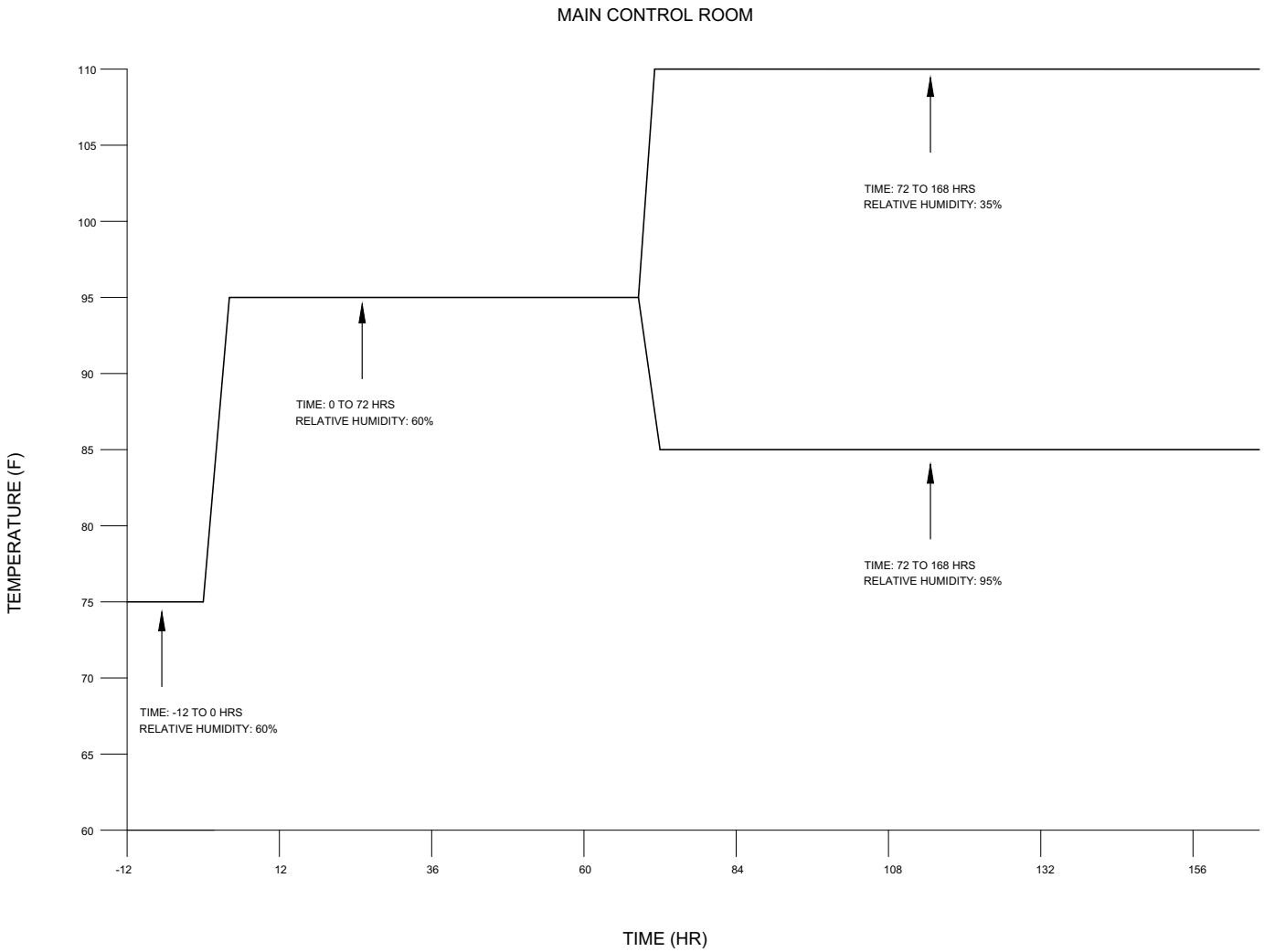

Figure 3D.5-1 (Sheet 1 of 3)
Typical Abnormal Environmental Test Profile:
Main Control Room

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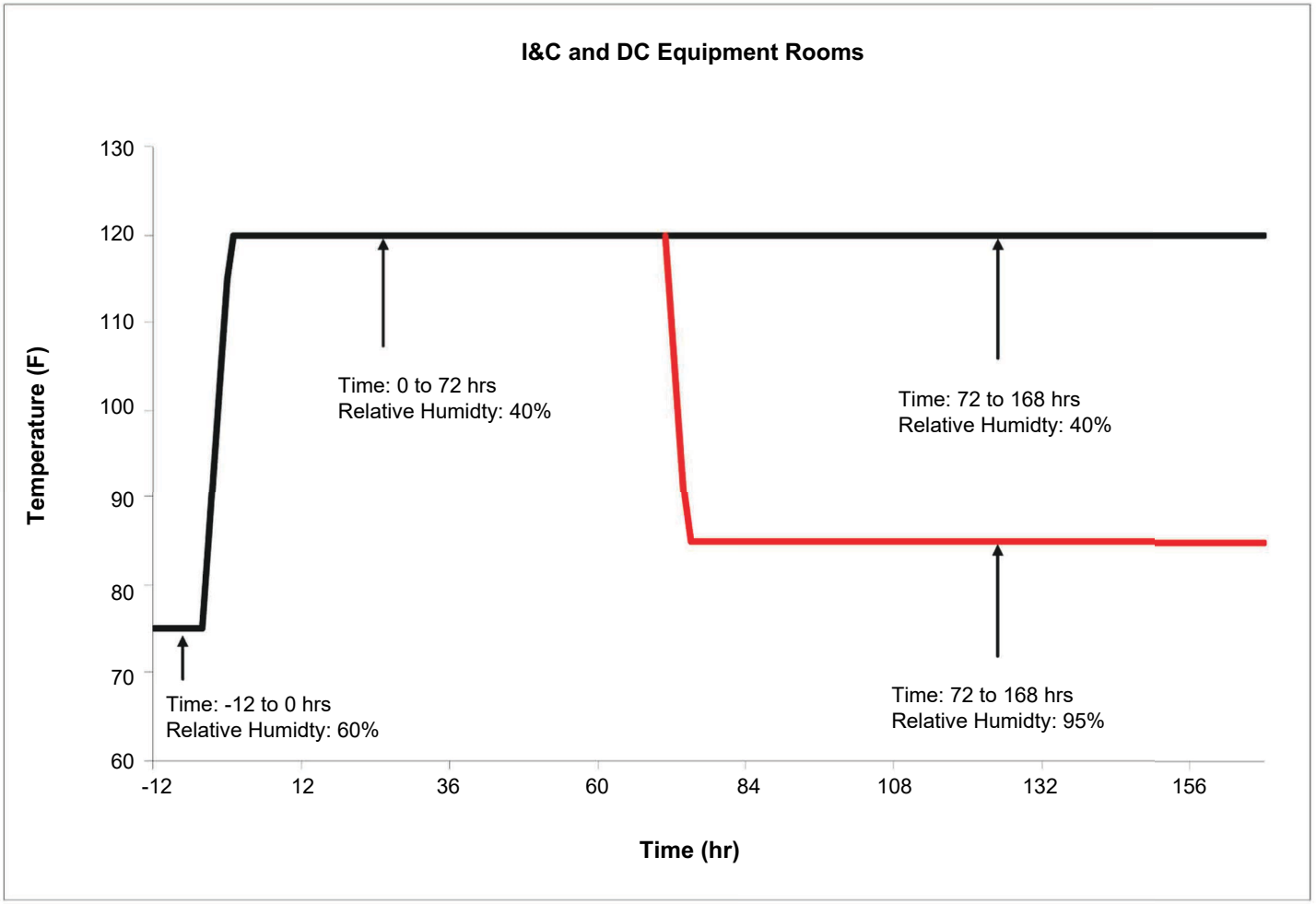

Figure 3D.5-1 (Sheet 2 of 3)
Typical Abnormal Environmental Test Profile:
I&C and DC Equipment Rooms

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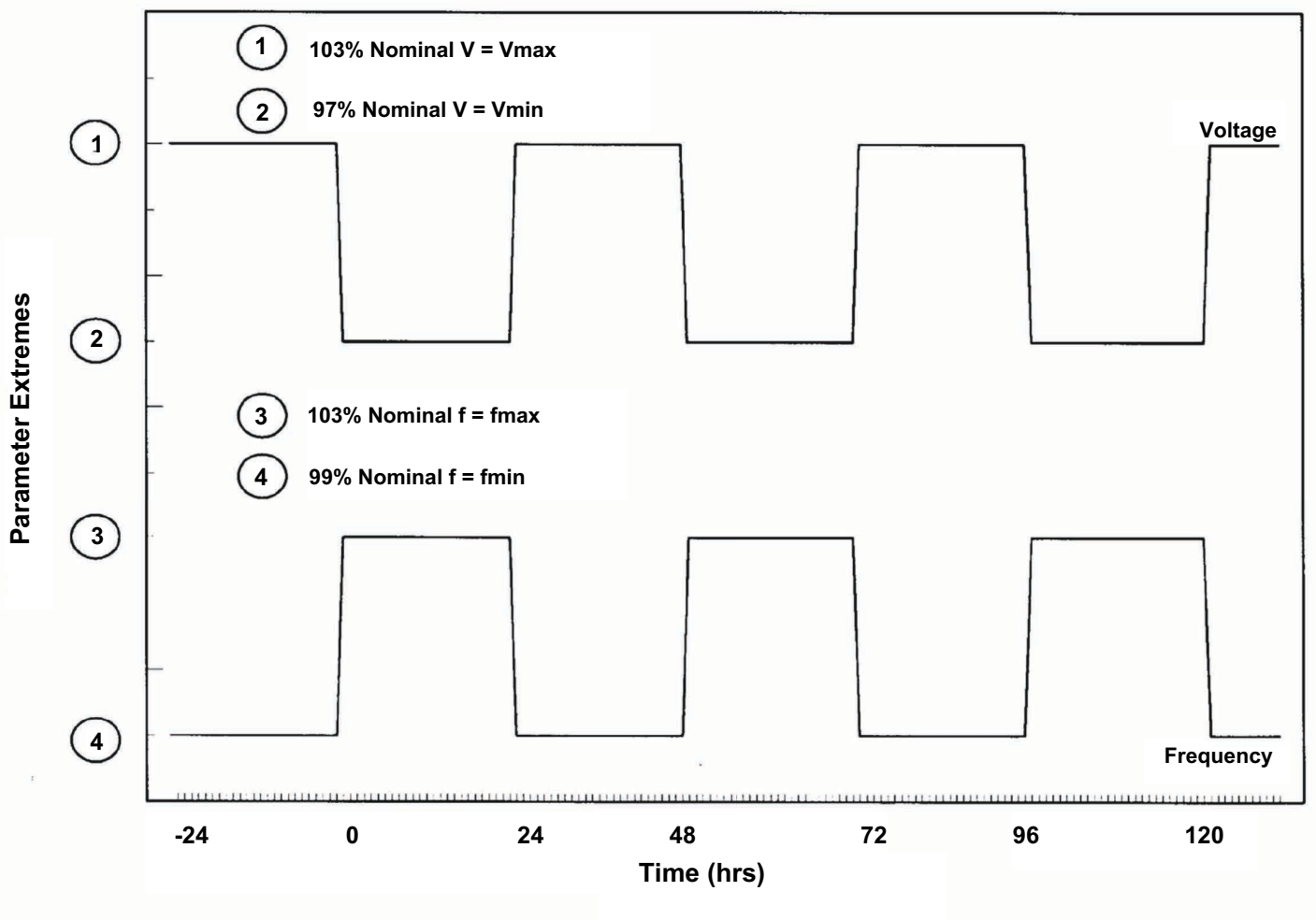

Figure 3D.5-1 (Sheet 3 of 3)
Typical Abnormal Environmental Test Profile:
Voltage and Frequency Variations

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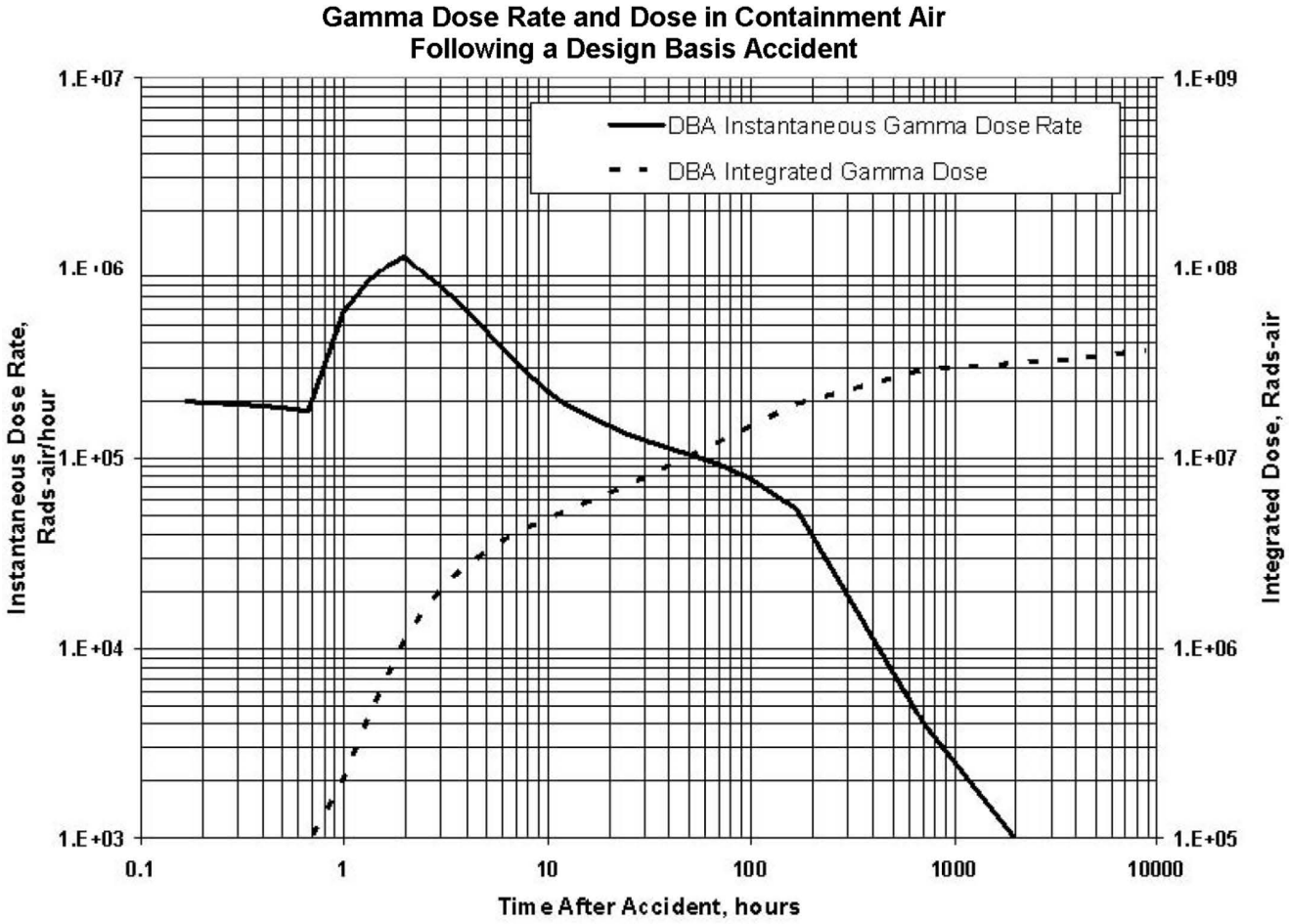

Figure 3D.5-2
Gamma Dose and Dose Rate Inside Containment After a LOCA

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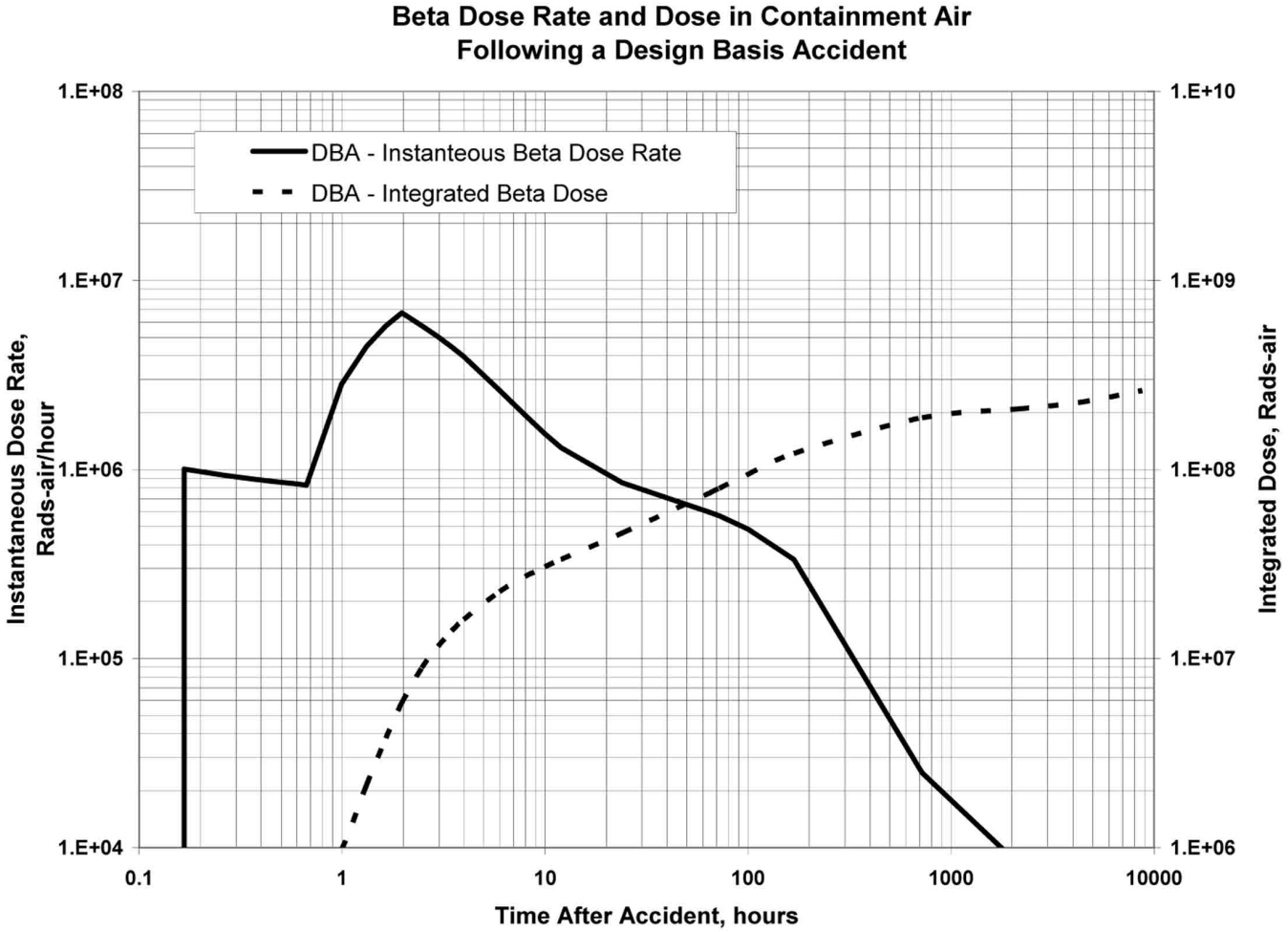

Figure 3D.5-3
Beta Dose and Dose Rate Inside Containment After a LOCA

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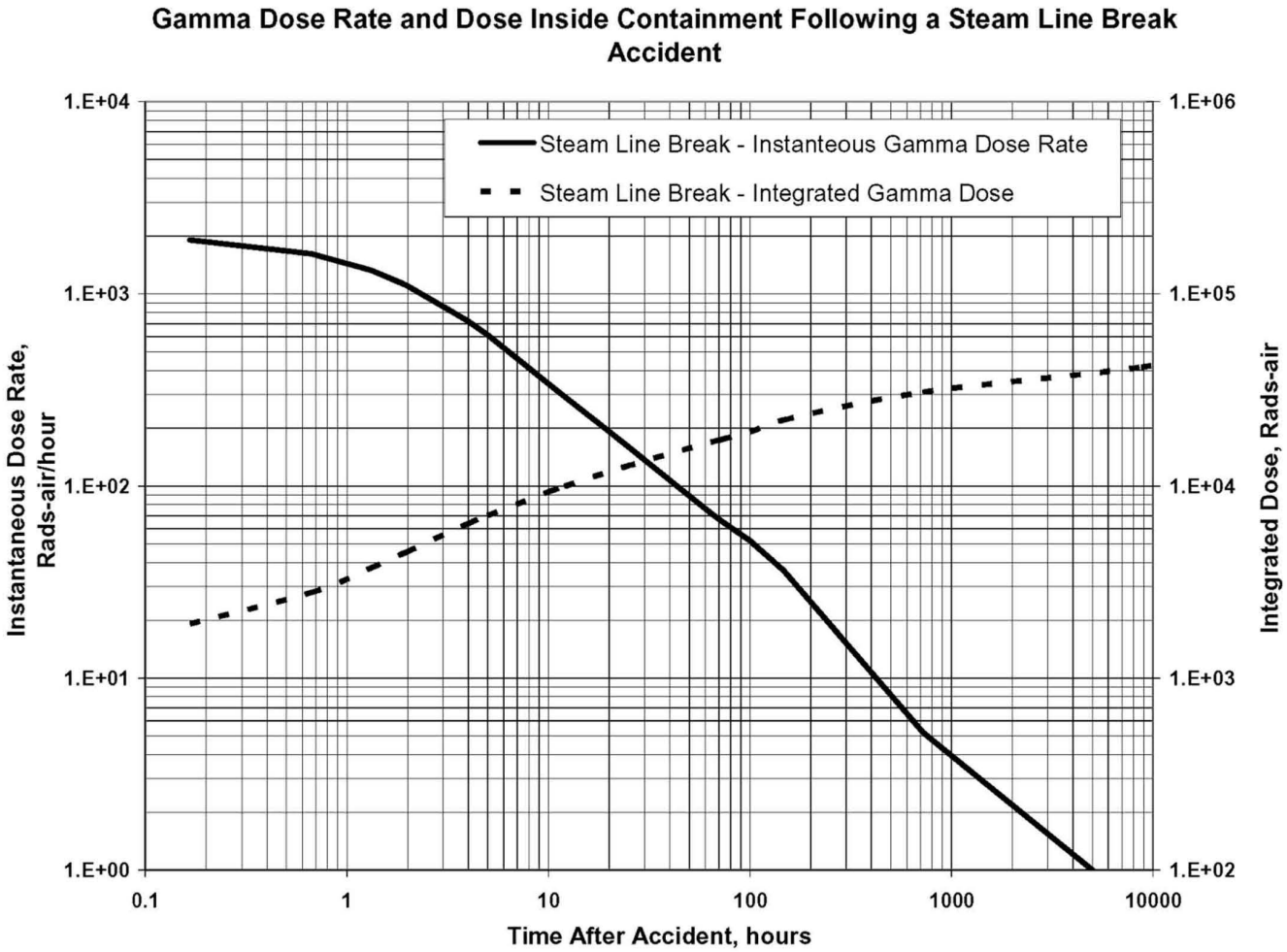

Figure 3D.5-4
Gamma Dose and Dose Rate Inside Containment After a Steam Line Break

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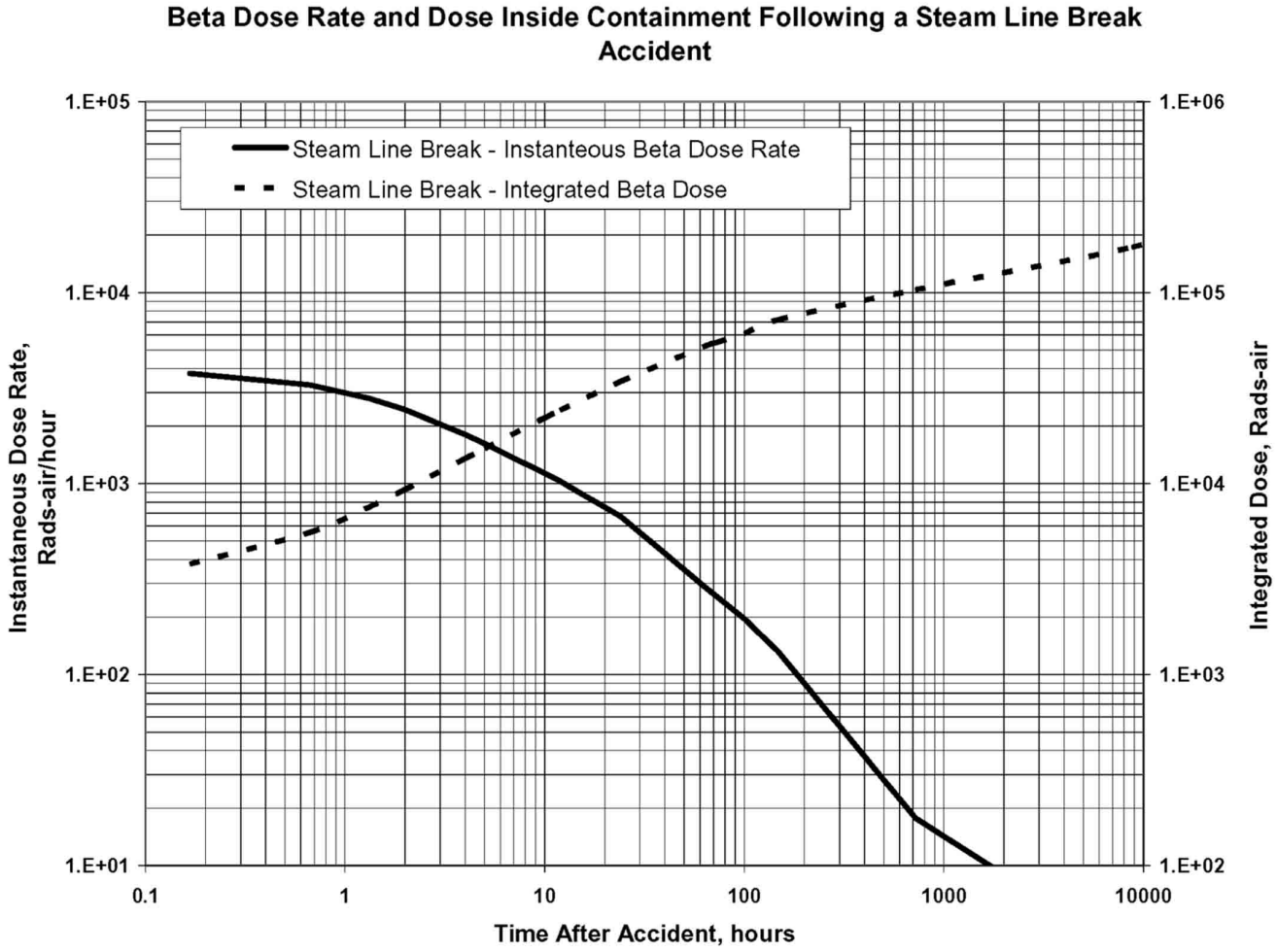

Figure 3D.5-5
Beta Dose and Dose Rate Inside Containment After a Steam Line Break

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Figures 3D.5-6–3D.5-7 Not Used

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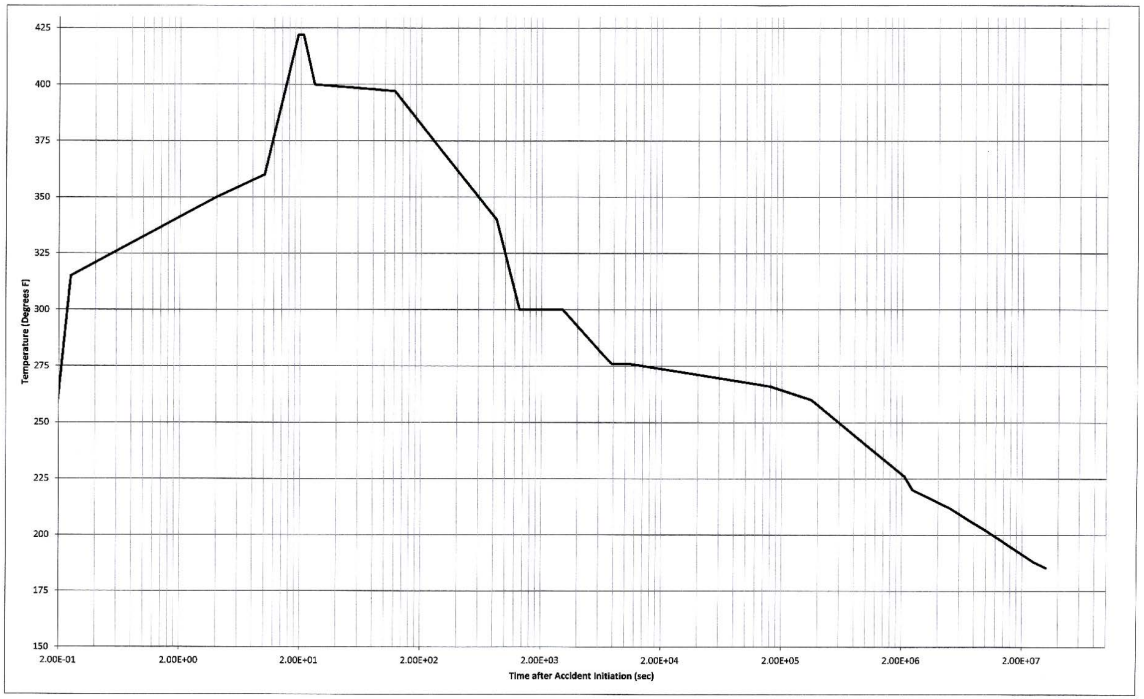

Figure 3D.5-8 (Sheet 1 of 2)
Typical Combined LOCA/SLB/FLB EQ Design Envelope for Inside Containment Temperature(1)

Note:
1. Refer to Section 6.2 for detailed post-accident containment response. This figure represents only the Environmental Qualification profile, which bounds the actual containment response.

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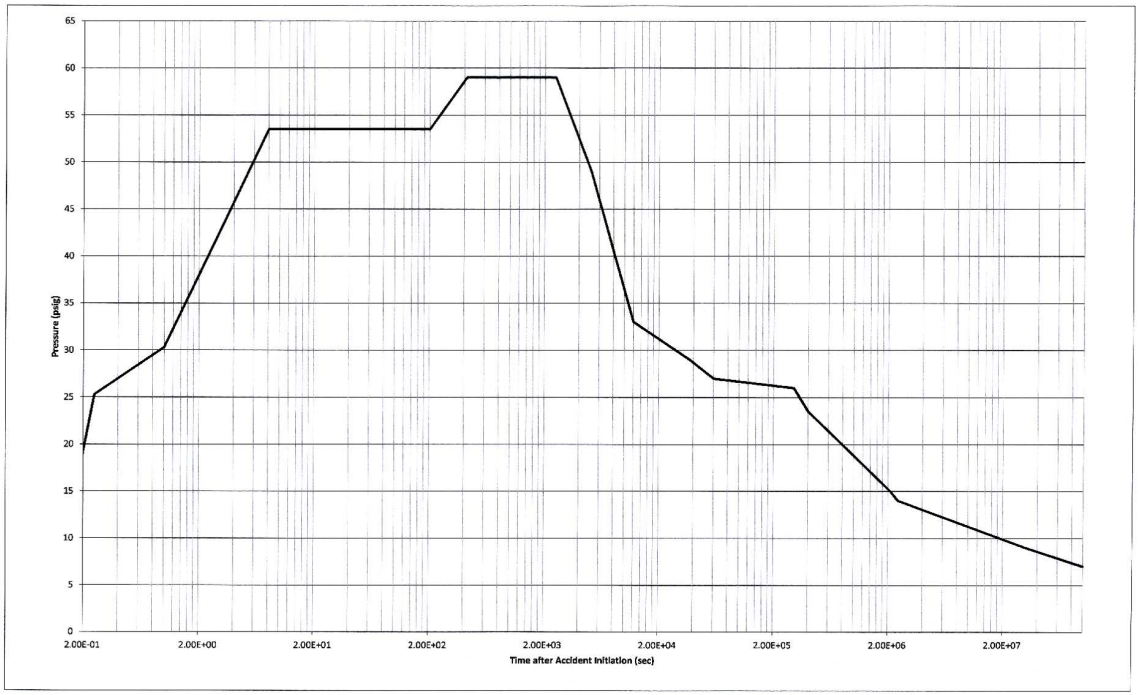

Figure 3D.5-8 (Sheet 2 of 2)
Typical Combined LOCA/SLB/FLB EQ Design Envelope for Inside Containment Pressure(1)

Note:
1. Refer to Section 6.2 for detailed post-accident containment response. This figure represents only the Environmental Qualification profile, which bounds the actual containment response.

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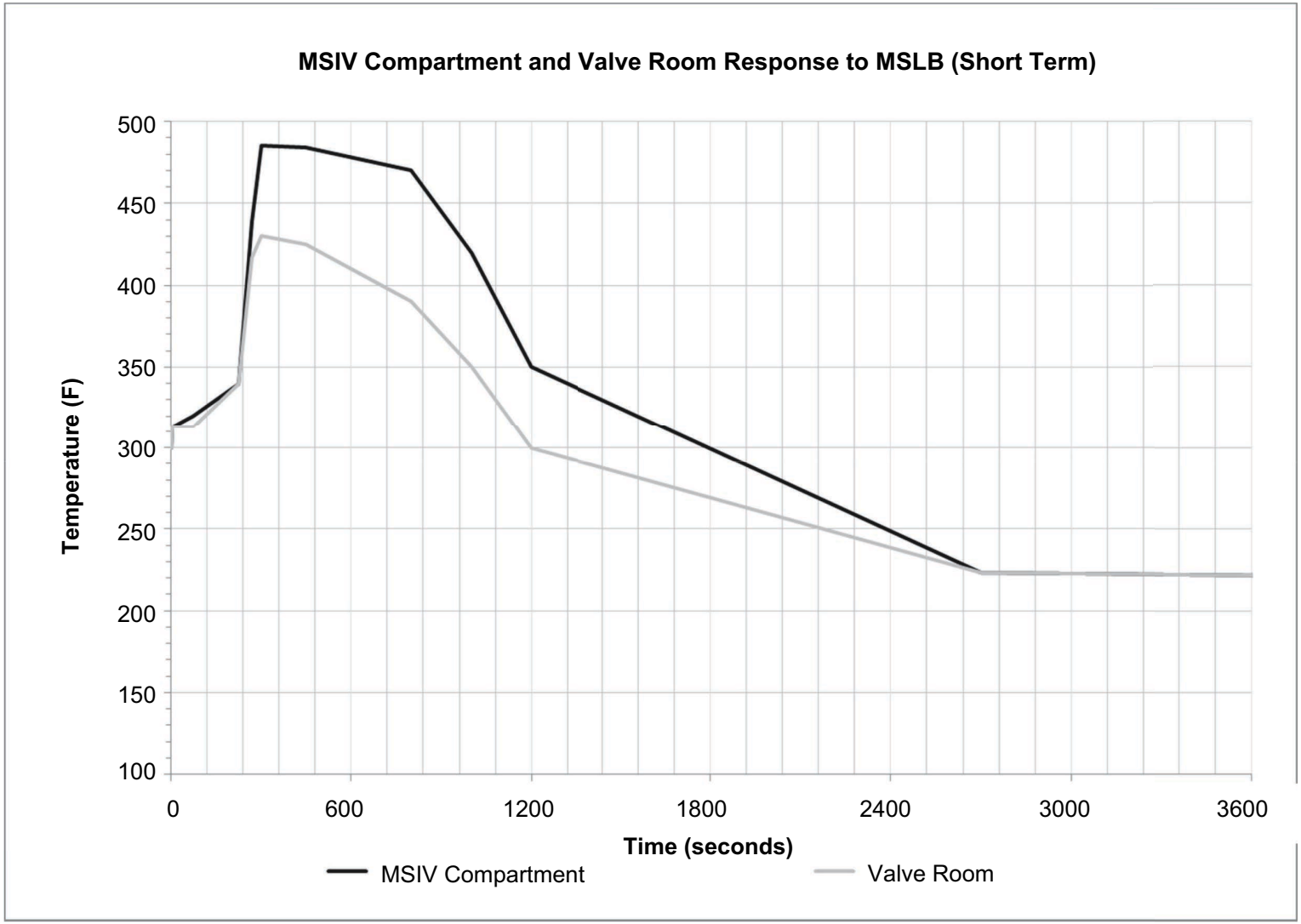

Figure 3D.5-9 (Sheet 1 of 2)
MSIV Compartment Response to MSLB (Short Term)

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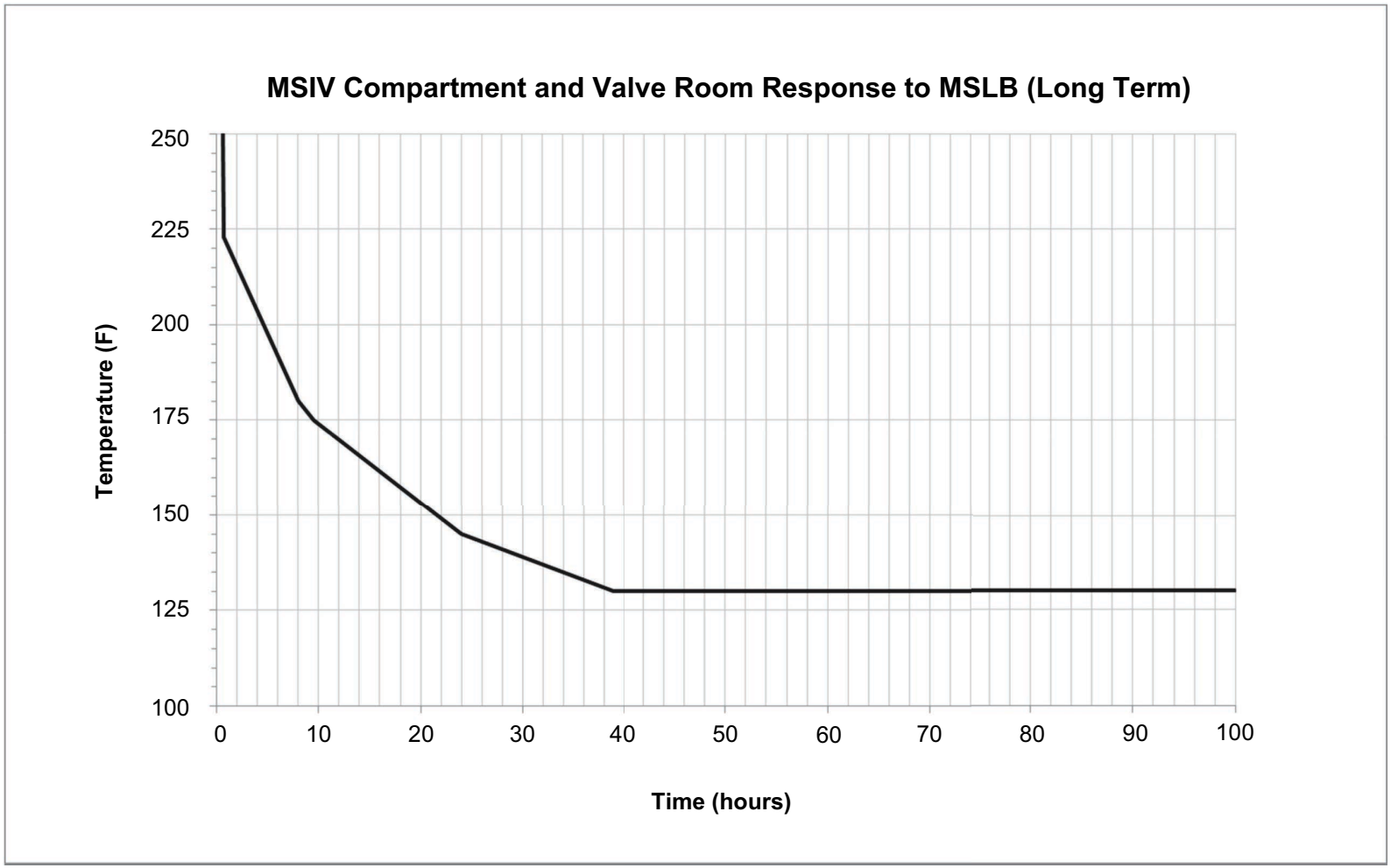

Figure 3D.5-9 (Sheet 2 of 2)
MSIV Compartment Response to MSLB (Long Term)

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ATTACHMENT A

SAMPLE EQUIPMENT QUALIFICATION DATA PACKAGE (EQDP)

To the extent practical, the equipment qualification data packages consist of the following elements:

Section 1 – Specifications
Section 2 – Qualification Program
Section 3 – Qualification by Test
Section 4 – Qualification by Analysis
Section 5 – Qualification Program Conclusions

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EQDP-_____
Rev. _____
{date issued}

EQUIPMENT QUALIFICATION DATA PACKAGE

Equipment _____

Manufacturer _____

Model _____

Application _____

Environment: ____ Harsh ____ Mild

Prepared by: _____
{name}

Reviewed by: _____
{name}

Approved by: _____
{name}

This document provides or summarizes the seismic and environmental qualification of the equipment identified above in accordance with the AP1000 EQ Program Methodology.

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1 SPECIFICATIONS

1.1 EQUIPMENT IDENTIFICATION: {create table(s) for details if a model series is to be qualified.}

Manufacturer _____
Model _____
Technical Manual _____
Drawings _____
Specification No. _____
Modifications _____

1.2 INSTALLATION REQUIREMENTS: {Cite vendor technical manual; details of mounting used for seismic test specimen(s); include any special requirements unique to Class 1E service}

1.3 ELECTRICAL REQUIREMENTS

1.3.1	Voltage:	_____	
1.3.2	Frequency:	_____	{if powered by AC}
1.3.3	Load:	_____	{as applicable}
1.3.4	Other:	_____	{identify and address as needed}

1.4 AUXILIARY DEVICES: {These are devices required to be interfaced with the subject equipment to provide qualification or operability but not specifically included or addressed in this document.}

1.5 PREVENTATIVE MAINTENANCE: {Identify manufacturer recommended maintenance activities required as part of the qualification program. Identify activities that are required to support qualification or the qualified life. "None" shall mean that maintenance is not essential to qualification or the qualified life. The following statement may be used in cases where qualification is not contingent upon maintenance or surveillance activities:

"No preventive maintenance is required to support the equipment qualified life. This does not preclude development of a preventive maintenance program designed to enhance equipment performance and identify unanticipated equipment degradation as long as this program does not compromise the qualification status of the equipment. Surveillance activities may also be considered to support the basis for, and a possible extension, of the qualified life."}

1.6 SAFETY FUNCTIONS

{Specify known safety functions for which qualification is intended to apply.}

1.7 PERFORMANCE REQUIREMENTS[a] for: {RCS Loop RTDs}

			Containment Test	DBE[b] Conditions	
Parameter	Normal Conditions	Abnormal Conditions	Abnormal	Seismic	LOCA
1.7.1 Time requirement					
1.7.2 Performance					

1.8 ENVIRONMENTAL CONDITIONS[a] for Same Function

1.8.1 Temperature (°F)

1.8.2 Pressure (psig)

1.8.3 Humidity (%RH)

1.8.4 Radiation (Rads)

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1.8.5 Chemicals

1.8.6 Vibration

1.8.7 Acceleration (g)

Notes: a: Test margin is not included in the parameters of this section.
b: DBE is the Design Basis Event.

{If more than one set of performance requirements and/or associated environmental conditions are to be specified, replicate these sections in pairs as "1.8 Performance ..." and "1.9 Environment ...", etc.}

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2 QUALIFICATION PROGRAM

2.1 PROGRAM OBJECTIVE

The objective of this qualification program is to demonstrate, employing the recommended practices of Regulatory Guides 1.89 and 1.100 and IEEE 323-1974, 344-1987, {cite others as applicable} capability of the {Equipment description} to perform its/their safety related function(s) described in EQDP Section 1.7 while exposed to the applicable conditions and events defined in EQDP Section 1.8.

{Narrative should introduce an outline of the program plan. Table below to be completed as graphic reference. Table shall not be abbreviated; items must appear and be addressed by direct response.}

2.2 REFERENCES

{List test report(s) and information sources cited in this document}

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	Qualification Method(s)		
CONDITION	**TEST**	**ANALYSIS**	**OTHER**
Aging:			
Thermal			
Radiation			
Vibrational			
Operational Cycling			
Electrical			
Mechanical			
Abnormal Environment			
Inadvertent ADS Actuation			
Seismic			
LOCA			
HELB Inside Containment			
HELB Outside Containment			
Post-accident Aging			

NOTES:

{All spaces above to be noted as "Yes," "No," or "Note #." Notes will be appended to the table. Notes will also include items "Not Applicable" with terse explanation and/or forwarding reference.}

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3 QUALIFICATION BY TEST (TEST PLAN AND SUMMARY)

3.1 SPECIMEN DESCRIPTION

{Identify the item or items to be tested}

3.2 NUMBER TESTED

{If more than one type is to be tested, identify how many of each. Subsequent Sections should clarify specifics for each.}

3.3 MOUNTING

{Identify specific seismic mounting details, referencing applicable drawings, instructions, documents. Note existence of differences from manufacturer recommendations}

3.4 CONNECTIONS

{Identify interfaces, both electrical and mechanical, identify any connectors or sealing assemblies used which are not provided with the equipment, or are not covered by this qualification.

3.5 TEST SEQUENCE PREFERRED

This section identifies the preferred test sequences as specified in IEEE 323-1974.

3.5.1 Inspection of Test Item
3.5.2 Operation (Normal Condition)
3.5.3 Operation (Performance Specifications Extremes: Section 1)
3.5.4 Simulated Aging
3.5.5 Vibration/Seismic
3.5.6 Operation (Simulated High Energy Line Break Conditions)
3.5.7 Operation (Simulated Post-HELB Conditions)
3.5.8 Inspection

3.6 TEST SEQUENCE ACTUAL

This section identifies the actual test sequence which constitutes the qualification program for this equipment. A justification for anything other than the preferred sequence is provided.

Test Sequence (from Section 3.5):

{List and explain; provide forwarding references to subsequent subsections as necessary}

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3.7 **SERVICE CONDITIONS TO BE SIMULATED BY TEST(1)**

	Normal	Abnormal	Seismic	HELB	Post-HELB
3.7.1 Temperature (°F)					
3.7.2 Pressure (psig)					
3.7.3 Humidity (% RH)					
3.7.4 Radiation (Rads)					
3.7.5 Chemicals					
3.7.6 Vibration					
3.7.7 Seismic (g)					

(1) Test parameter margins are included for the worst-case known requirements applicable to the equipment type. Margin for a specific parameter is dependent on the requirements of each application or location for the equipment; these may vary.

(2) Post-accident operability addressed through simulated thermal aging. Temperature and other parameters are selected to envelop the requirements.

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3.8 MEASURED VARIABLES

This section tabulates the variables and parameters required to be measured during each of the following tests in the qualification test sequence.

Tests: {example}
A: Thermal Aging
B: Mechanical Cycling
C: Irradiation
D: Seismic Test
E: HELB Test

	Required	Not Required
3.8.1 Category I □Environment		
3.8.1.1 Temperature	...	...
3.8.1.2 Pressure	...	...
3.8.1.3 Moisture	...	...
3.8.1.4 Gas Composition	...	...
3.8.1.5 Vibration	...	...
3.8.1.6 Time	...	...
3.8.2 Category II □Input Electrical Characteristics		
3.8.2.1 Voltage	...	...
3.8.2.2 Current	...	...
3.8.2.3 Frequency	...	...
3.8.2.4 Power	...	...
3.8.2.5 Other	...	...
3.8.3 Category III □Fluid Characteristics		
3.8.3.1 Chemical Composition	...	...
3.8.3.2 Flowrate	...	...
3.8.3.3 Spray	...	...
3.8.3.4 Temperature	...	...
3.8.4 Category IV □Radiological Features		
3.8.4.1 Energy Type	...	...
3.8.4.2 Energy Level	...	...
3.8.4.3 Dose Rate	...	...
3.8.4.4 Integrated Dose	...	...
3.8.5 Category V □Electrical Characteristics		
3.8.5.1 Insulation Resistance	...	...
3.8.5.2 Output Voltage	...	...
3.8.5.3 Output Current	...	...
3.8.5.4 Output Power	...	...
3.8.5.5 Response Time	...	...
3.8.5.6 Frequency Characteristics	...	...
3.8.5.7 Simulated Load	...	...

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	Required	Not Required
3.8.6 Category VI □Mechanical Characteristics		
3.8.6.1 Thrust	...	...
3.8.6.2 Torque	...	...
3.8.6.3 Time	...	...
3.8.6.4 Load Profile	...	...
3.8.7 Category VII □Auxiliary Equipment		
3.8.7.1 {as applicable, also see Section 1.4 of EQDP}	...	...

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3.9 TYPE TEST SUMMARY

3.9.1 Normal Environment Testing

Operation of the {equipment} under normal conditions is demonstrated by {discuss test, checks, et. al. which provide baseline performance data} ... , as reported in Reference _.

3.9.2 Abnormal Environment Testing

Operation of the {equipment} under abnormal conditions is demonstrated by {discuss test, checks, et. al. which provide baseline performance data} ... , as reported in Reference _.

3.9.3 Aging Simulation Procedure

{Describe the aging mechanisms simulated and the sequence, including justifications as necessary.}

The test units were pre-conditioned to simulate an aged condition prior to subjecting them to the Design Basis Event (DBE) seismic and environmental conditions/simulation. The aged condition was achieved by separate phases of {accelerated thermal aging, thermal cycling, and radiation exposure to a total integrated gamma dose equivalent to a twenty-year normal dose plus the design basis accident dose, and accelerated flow induced and pipe vibration simulation}. Through all the pre-conditioning phases, the {equipment, performance} were monitored to verify {continuous operation}.

3.9.3.1 Design Life: {Also, justification of the bases for a design life goal should be provided, when used in mild-environment programs. Generally inapplicable to harsh-environment programs.}

3.9.3.2 Shelf Life: {Though not typically applicable, state any limitation in life, as well as conditions which may be detrimental if known.}

3.9.3.3 Thermal Aging: The qualified life is __ years based on an ambient temperature of {__°C (__°F) and a __°C temperature rise due to _____}. Calculations are based on a test temperature of __, test duration of ___ hours, and an activation of __ eV (See References x, et al.)}.

3.9.3.4 Radiation Aging: The qualified life is limited by the expected radiation during the __-year life and the Design Basis Event. {Subtract accident TID from qualified TID; account for margin, remainder is to be compared to normal/abnormal radiation requirements to yield life limits.}

3.9.3.5 Operating Cycles: {Expected number of electrical and/or mechanical cycles, or numbers of actuations, as applicable. Estimated on the basis of the expected for the design, qualified, installed life of the equipment. Specification may be on a per annum or a per fuel cycle basis. Compare to cycle life data from test.}

3.9.3.6 Vibration Aging: {present bases; refer to test profile and/or Subsection 3.9.4}.

3.9.4 Seismic Tests

The seismic testing reported in Reference x was completed on aged equipment employing {method(s)} in accordance with Regulatory Guide 1.100 and IEEE 344-1987. ... {Summarize equipment condition and/or performance versus the acceptance criteria.} ... Actual margin should be determined for each application/location throughout the plant and verified to meet or exceed the margin requirements.

{Discuss or reference discussion of test anomalies.}

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3.9.5 High Energy Line Break/Post HELB Simulation

The {equipment} were subjected to the HELB simulation temperature/pressure profile of Figure x. Following the __°F temperature peak, the temperature gradually declines to __°F and is held at saturated steam conditions for _ days, simulating a _____ period of Post-HELB operation. The test data and activation energy specified in Subsection 3.9.3.3 can be used to determine margin in post-accident aging for each application/location of the equipment.

{Summarize equipment condition and/or performance versus the criteria}

{Discuss or reference discussion of test anomalies.}

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4 QUALIFICATION BY ANALYSIS

The AP1000 EQ Program does permit qualification solely on the basis of analyses for equipment outside the scope of 10CFR50.49. The following subsections discuss each of the analyses preformed, its test basis and justification, and summarizes conclusions documented in References x; et. al., which provided detailed accounts of each analysis.

{Each subsection will address a particular analysis, if more than one is performed to support qualification.}

4.x (EXAMPLE)

{The purpose and objective will be identified here. Subsections will provide necessary details per the following format.}

4.x.1 {Equipment, Characteristic or Aspect} Analyzed

{A general description of the equipment and its function based on applicable equipment and mounting drawings, and purchase orders.}

4.x.2 Equipment Specification(s)

{The applicable design standards shall be documented including any limitations imposed by the equipment specification. Installation detail considered or represented are to be included.}

4.x.3 Methods and Codes

{Description of analytical methods or techniques, computer program, mathematical model(s) used, and the method(s) of verification}

4.x.4 Acceptance Criteria

{The specific safety function(s), postulated failure modes, or the failure effects to be demonstrated by analysis.}

4.x.5 Model

{Description of mathematical model of equipment or feature analyzed.}

4.x.6 Assumptions and Justifications

{EXAMPLES: Description of the loading conditions to be used. Summary of stresses to be considered.}

4.x.7 Impact to Safety Function

{Summarize analytically established performance characteristics and their acceptability. Discussion and summary of the analytical results which demonstrate equipment structural integrity and, where appropriate, operability. Particular to cabinets, critical deflections should be determined and included in mounting requirements for spacing with respect to other equipment and structures.}

4.x.8 Conclusions

{Descriptive summary, including any conditions imposed on qualification or use; qualified life, limitations, surveillance/maintenance requirements, et. al.} Further discussion of this analysis is presented in Reference x.

4.Y ENVIRONMENTAL QUALIFICATION ANALYSIS FOR {VALVE SOFT PARTS}

{purpose and objective}

4.Y.1 Equipment Identification

{Per Subsection 3D.6.2.3.1}

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4.Y.2 Component Identification

{Per Subsection 3D.6.2.3.1}

4.Y.3 Safety Related Functions

{Per Subsection 3D.6.2.3.2}

4.Y.4 Component Acceptance Criteria

{Per Subsection 3D.6.2.3.3}

4.Y.5 Service Conditions

{Per Subsection 3D.6.2.3.4}

4.Y.6 Potential Failure Modes

{Per Subsection 3D.6.2.3.5}

4.Y.7 Identify the Environmental Effects on Material Properties

Each non-metallic, including lubricants, is evaluated to determine the effect of the environmental conditions on the material properties. For each non-metallic, a radiation threshold level and maximum service temperature is identified.

The radiation threshold level and the maximum service temperature are identified using materials handbooks, textbooks, government and industry reports, and laboratory data. If the evaluation indicates that the lowest levels may be exceeded for certain equipment, higher levels are identified at which varying degrees of material degradation may occur.

Mechanical equipment is highly resistive to degradation due to elevated humidity levels: therefore, relative humidity is not included as a parameter to be evaluated for environmental qualification. Pressure can be discounted for most equipment types, as there are no foreseen failures due to elevated pressure levels for most mechanical equipment. However, pressure must be addressed in the evaluation.

The susceptibility of the non-metallic material to the chemicals due to the design basis accident and exposure to the process fluid is evaluated. The material information in the chemical handbooks is an acceptable source of qualification documentation.

4.Y.7.1 Perform Thermal Aging Analysis

Aging analysis is performed for organic materials. Mineral-based subcomponents are not considered to be sensitive to thermal aging during the design life of a plant and, therefore, are not analyzed.

Aging in mechanical components is associated with corrosion, erosion, particle deposits and embrittlement. In new construction, corrosion and erosion are considered by providing additional material thickness as a corrosion or erosion allowance above the required design. The other aging phenomena are considered during inservice inspections of operating components in accordance with ASME Code, Section XI. Aging qualification of metallic parts of equipment except for corrosion and erosion is in compliance with ASME Code, Section XI, therefore aging effects on metallic components are not addressed herein.

The non-metallic material analysis for determining the expected qualified thermal life is performed using Arrhenius methodology. The thermal input during the operating time, as explained below, is deducted from the tested thermal aging of the material at service temperature to obtain the qualified life.

The component is evaluated for the specified post-accident operating time. The thermal input from the postulated accident profile (i.e., LOCA/MSLB) for the duration of the specified operating time is compared to the material thermal aging data. The Arrhenius model is used to perform this comparison. The component is evaluated for the maximum post-accident operating time unless a system analysis is performed to justify shorter operating times.

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Analysis of the non-metallics should also take into account any degradation of the part due to its use in dynamic modes (i.e., moving part).

4.Y.7.2 Evaluate the Environmental Effects on Equipment Safety-Related Function

A conservative initial screening of the non-metallic subcomponents is made by comparison of the material capabilities (threshold radiation level and maximum service temperature) with the maximum postulated environmental conditions. If the threshold radiation values and the maximum service temperatures are above the maximum postulated environmental conditions, and if the material aging analysis demonstrates a service life sufficient to survive the accident duration, then the material is considered acceptable.

Those items which are not shown to be acceptable based on the above comparison are evaluated in further detail regarding:

- extent of material degradation
- material properties affected
- equipment/subcomponent function
- extent of equipment functional degradation
- location-specific environmental conditions

4.Y.8 Conclusions

{Per subsection 3D.6.2.3.7}

4.Y.9 EQ Maintenance Requirements

{Per subsection 3D.6.2.3.8}

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5 QUALIFICATION PROGRAM CONCLUSIONS

5.1 AGING

{Discuss specifics and state on limitations or requirements; specifics with respect to:

- Design Life Goal
- Thermal Aging
- Radiation Aging
- Operating Cycles
- Vibration Aging}

5.2 DBE QUALIFICATIONS

5.3 PROGRAM CONCLUSIONS

The qualification of the {equipment} is demonstrated by the completion of the simulated aging and Design Basis Event testing described herein and reported in Reference {1}.

{State any conditions imposed on qualification or qualified life, cite any lessons learned which necessitate future user actions to preserve continued qualification}

{Refer to QUALFICATION SUMMARY TABLE}

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QUALIFICATION SUMMARY TABLE

SYSTEM	{RPS}
CATEGORY	Category(1) {a}
LOCATION	{Containment bldg.}
STRUCTURE/AREA	{Zone Number}
EQUIPMENT TYPE	{pressure transmitter }
MANUFACTURER	{_____}
MODEL	{_____}

PARAMETER	QUAL METHOD(2)	ENVIRONMENTAL EXTREMES QUALIFIED(3)	SPECIFIED(4)	NOTES
NORMAL				
ABNORMAL				
QUALIFIED LIFE				{5}
SEISMIC	{Both}	Figure x	{Ref; Fig.}	
ACCIDENT		Figure x	{Ref; Fig.}	
Temperature	{Test}	___°F		
Pressure	{Test}	___ psig		
Rel. humidity	{Test}	___%		
Radiation	{Both}	___E+06 R(γ)		
	{Both}	___E+06 R(β)		
Chemistry	{Test}	{Note 6}		
Operability	{Both}			
Accuracy	{Test}			

NOTES:

1. Equipment category as per NUREG-0588, Appendix E, Section 2.
2. Qual. Methods are: Test, Analysis, Both (Test & Anal.), or Other.
3. Qualified values are test extremes which include margin.
4. Environmental parameters for the plant location are to be inserted. If more than one applicable, most extreme are to be cited
5. Qualified life estimated on basis of maximum normal temperature of ___°C (__°F) and a temperature rise of ___°C (____°F).
6. Chemistry Conditions: {pH and composition}.

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ATTACHMENT B

AGING EVALUATION PROGRAM

B.1 Introduction

As stated in IEEE 323, aging of Class 1E equipment during normal service is considered as an integral part of the qualification program. The objective is not to address random age-induced failures that occur in-service and are detected by periodic testing and maintenance programs. The objective is to address the concern that some aging mechanisms, when considered in conjunction with the specified design basis events (DBE), may have the potential for common mode failure.

The AP1000 equipment qualification program addresses the aging concern and makes maximum use of available data and experience on aging mechanisms. This approach places primary emphasis on common mode failures due to enveloping design basis events. For example, reasonable assurance against common mode failures being induced because of a loss of heating, ventilation, and air conditioning (HVAC) is provided by adequate design, normal maintenance, and calibration procedures.

B.2 Objectives

The objectives of the aging evaluation program follow:

- To establish, where possible, the effects of the degradation due to aging mechanisms that occur before the occurrence of an accident, when safety-related equipment is called upon to function

- To provide increased confidence that safety-related equipment performs its safety-related function under the specified service condition.

B.3 Basic Approach

The general approach to addressing aging allocates equipment to one of two categories.

- Harsh environment includes electrical equipment required to perform a safety-related function in a high-energy line break (HELB) environment. For this equipment an aging simulation is included as part of the equipment qualification test sequence. The equipment is energized during the aging simulation.

- Mild environment includes equipment required to mitigate high-energy line breaks but which, due to its location, is isolated from any adverse external environment resulting from the accident. For equipment in a mild environment the single design basis event capable of producing an adverse environment at the equipment location is the seismic event. Aging, for mild environment, is not included in the equipment qualification test sequence. Significant aging mechanisms are determined by evaluation of available test data. Generally, this data is from separate programs conducted to demonstrate that aged components continue to meet manufacturer's performance specifications under applicable seismic design basis event conditions and that seismic testing of unaged equipment is not invalidated by anticipated aging mechanisms.

B.4 Harsh Environment

Electrical equipment required to perform a safety-related function in a high-energy line break (such as a loss of coolant accident, feed line break, or steam line break) environment is included in the

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harsh environment category. This category provides for an aging simulation to be included in the equipment's qualification test sequence.

B.4.1 Scope

An example of equipment scope and the aging mechanisms applied under harsh environment are shown in Tables 3D.B-1 and 3D.B-2, respectively. The equipment selected is that Class 1E equipment qualified to operate in a high-energy line break environment. The aging mechanisms discussed next are those to which the equipment may be potentially sensitive in its installed location.

B.4.2 Aging Mechanisms

The aging mechanisms that could potentially affect electrical equipment in a harsh environment are discussed under the following headings:

Time, in conjunction with:

- Operational stresses (current, voltage, operating cycles, Joulean self-heating)
- (External stresses (thermal, vibration, radiation, humidity, seismic).

The aging mechanisms considered potentially significant and to be simulated are identified in Table 3D.B-2 for each item of equipment in a harsh environment. Where applied, the aging mechanisms are simulated as described in the following discussions.

B.4.3 Time

For equipment subject to high-energy line break conditions, the most significant in-service aging mechanisms (that is, radiation and thermal) come into effect during reactor operation. Consequently, it can be assumed that the "aging clock" starts on plant startup.

B.4.4 Operational Stresses

Electrical Cycling

Electrical supplies to safety-related equipment are, in general, highly stable. So aging effects due to supply cycling during service are not anticipated. Where the equipment is anticipated to experience multiple startup and shutdown cycles, the equipment is electrically cycled to simulate the number of anticipated startup and shutdown cycles plus 10 percent.

Mechanical Cycling

Aging effects resulting from anticipated mechanical cycling of the equipment are simulated by applying, as a minimum, the number of cycles estimated to occur during the target qualified life plus 10 percent. Mechanical cycling covers such operations as switching and relay actuation.

Joulean Self-Heating

Where the equipment is not aged in a live condition, the aging effects resulting from Joulean self-heating are recognized by employing the equipment operating temperature as the datum temperature for assessing the accelerated thermal aging parameters to be employed.

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B.4.5 External Stresses

Thermal Effects

Thermal effects are considered one of the most significant aging mechanisms to address. The equipment is thermally aged to simulate an end-of-qualified-life condition using the Arrhenius model to establish the appropriate conditioning period at elevated temperature. Where data is not available to establish the model parameters for the materials employed, a verifiably conservative value of 0.5 eV is used for activation energy (Attachment D).

For each piece of equipment an appropriate normal and abnormal operating temperature and an associated time history are determined for inclusion in the Arrhenius model. The equipment temperature is determined by the addition of an appropriate equipment specific ΔT to the external ambient temperature. Attachment D also provides information concerning the determination of appropriate ambient temperatures and time-temperature histories for use in thermal aging evaluation of equipment. Post-accident thermal aging is included by recognizing the higher post-accident ambient temperatures in determining the parameters employed for the post-accident accelerated thermal aging simulation.

In-Service Vibration

The majority of safety-related electrical equipment has a proven history of in-plant service. Thus, it is unlikely that a significant, undetected, failure mechanism exists because of low-level, in-plant vibration. In addition, a simulation of earthquakes smaller than the safe shutdown earthquake (SSE) employed during equipment and component seismic testing give added confidence that this potential aging mechanism is covered (See Attachment E, Section 4.4). For line-mounted equipment, in-service pipe and flow induced vibration may be significant. As a consequence, an additional vibration aging step is included in the aging sequence as indicated for certain items of equipment in Table 3D.B-2. (See Attachment E, Subsection 5.2.2.)

Radiation

Radiation during normal operation is not considered an aging mechanism for equipment subject to in-service integrated doses less than 10^4 rads. Research has established that no aging mechanisms are measurable below 10^4 rads (Attachment C) for materials and most components supplied in safety-related electrical equipment. Some devices may have performance limitations below 10^4 rads. For radiation doses in excess of 10^4 rads, the equipment is irradiated using a gamma (γ) source to a dose equivalent to the estimated dose to be incurred during normal operation for the target qualified life. The estimated doses employed are specified in the equipment qualification data package, Subection 1.8.4, and are based on a 100 percent load factor, including appropriate margin. For harsh environment equipment, the equivalent accident dose is usually applied before design basis event testing.

Humidity

The use of materials significantly affected by humidity is avoided. For equipment subject to high energy line break environments, the aging effects due to humidity during normal operation are judged to be insignificant compared to the effects of the high-temperature steam accident simulation. Therefore, no additional humidity aging simulation is required.

Seismic Aging

The potential aging effects of low-level seismic activity and some low-level, in-plant vibration are addressed by employing a simulation of five earthquakes of 50 percent of the magnitude of a safe shutdown earthquake before seismic testing of the aged equipment.

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B.4.6 Synergism

An important consideration in aging is the possible existence of synergistic effects when multiple stress environments are applied simultaneously. The potential for significant synergistic effects is addressed by the conservatisms inherent in using the "worst-case" aging sequence, conservative accelerated aging parameters and conservative, design basis event test levels which provide confidence that any synergistic effects are enveloped.

B.4.7 Design Basis Event Testing

Design basis event testing subsequent to equipment aging is discussed in Appendix 3D as to guidelines for defining high-energy line break environments and seismic conditions. Testing for equipment specific test environments and seismic parameters is discussed in Attachment A, Section 3.

B.4.8 Aging Sequence

The aging mechanisms applied to equipment subject to high-energy line break environments are determined by definition of the aging environments at the equipment location and by a subsequent evaluation of the sensitivity of the equipment to these environments. If the sensitivity of the equipment is not known, aging mechanisms are simulated by conservative methods as previously described. Those aging mechanisms that are simulated for example equipment subject to high-energy line break environments are shown in Table 3D.B-2.

The order in which each of the aging mechanisms is applied is as shown in Table 3D.B-2. This order is considered to be conservative, as no aging mechanism is anticipated to be capable of reducing the impact of the previously applied mechanisms. As an example, thermal aging is applied before radiation aging to preclude the annealing out of radiation-induced defects. Similarly, the effects of mechanical aging are considered more significant when applied to equipment that has already been preaged to address thermal and radiation phenomena.

B.4.9 Performance Criterion

The basic acceptance criterion is that the qualification tests demonstrate the capability of the aged equipment to perform prespecified, safety-related functions consistent with meeting the performance specification of Attachment A, Section 1.7 of the applicable equipment qualification data packages while exposed to the associated environmental conditions defined in Attachment A, Section 1.8.

B.4.10 Failure Treatment

When thermal aging is simulated at an equipment level, a conservative value for the activation energy is assumed for the components composing the equipment. As a consequence, many components are grossly overaged, and failure of some of the components is expected during the aging simulation. When three test units are preaged, in the event of such failure(s), one of the following options is selected.

- when a particular component fails in one of the three test units, the failure is considered random. The failed component is replaced by a new component, and the test is continued

- when a particular component fails in more than one of the three test units, either:

    1. the failed components are replaced by new identical components and the aging simulation continued. The claimed qualified life of the unit is consistent with the minimum aging period simulated by at least two of the three units; or

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2. the failed components is replaced by identical components specifically aged to the qualified life by assuming for thermal aging a less conservative activation energy specifically determined for the component, or

3. the failed components are replaced by a different type of component which is aged for a period equal to the test units.

When less than three test samples prevent such a conclusion from being reached, any failures are investigated to ascertain whether the failure mechanism is of common mode origin. Should a common mode failure mechanism be identified as having caused the failure, a design change is implemented to eliminate the problem. Supplemental or repeat tests will be completed to demonstrate compliance with the acceptance criteria.

B.5 Mild Environment

A mild environment aging program includes Class 1E equipment not required to perform a safety related function in a high-energy line break environment. It involves a review of available information to demonstrate the absence of significant in-service aging mechanisms. For equipment allocated to this category, the single design basis event capable of producing an adverse environment at the equipment location is the seismic event. Seismic testing completed on unaged equipment is verified as valid by demonstrating that no available information suggests that aged materials and components would not continue to meet their design specification during a seismic event.

B.5.1 Scope

A mild environment aging program includes both a review of material analysis and the results of a component testing program for equipment not required to perform a safety-related function in a high-energy line break environment. Equipment is included that is required to mitigate high-energy line breaks but which, because of the equipment location, is isolated from the adverse environment resulting from the accident. An example of the equipment allocated to a mild environment is identified in Table 3D.B-1.

B.5.2 Performance Criteria

Available Material Analysis – For equipment and components for which aging is addressed by evaluation of appropriate mechanisms, the basic performance criterion is that the evaluation of test data demonstrates the effect of aging is minor and does not affect the capability of the aged equipment to perform prespecified functions. This is consistent with meeting the performance specification of Attachment A, Section 1.7 of the applicable equipment qualification data package while exposed to the associated environmental conditions defined in Attachment A, Section 1.8.

Available Component Aging Data – Random component failure or unacceptable performance due to aging is detected by routine maintenance and equipment calibration during service. The objective of a mild environment aging program is to provide reasonable assurance that a seismic event does not constitute a common mode failure mechanism capable of inducing unacceptable performance characteristics in aged components. Consequently, the single performance criterion for the aging portion of the qualification sequence requires that the component not fail to perform its general function, not that the component meets the original design and procurement specifications.

For the seismic event simulation, the component is considered acceptable if, during and after the simulation, it does not exhibit any temporary or permanent step change in performance characteristics. Failure of one of three components tested is considered a random failure, subject to an investigation concluding the observed failure is not common mode.

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B.5.3 Failure Treatment

In the event of failure to demonstrate conformance to criteria, the following options are available for resolution of qualification with respect to age:

- Establish a maintenance and surveillance program
- Replace the materials or components with those constructed of materials of known acceptable characteristics.

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Table 3D.B-1
Example Class 1E Equipment Scope and Category Allocation

Aging Method	Equipment
Harsh Environment	Valve Motor Operators Solenoid Valves Externally Mounted Limit Switches Pressure Transmitters Differential Pressure Transmitters Resistance Temperature Detectors Neutron Detectors Pressure Sensor Batteries*
Mild Environment	Pressure Transmitters Differential Pressure Transmitters Main Control Board Switch Modules Recorders (Post-Accident Monitoring) Indicators (Post-Accident Monitoring) Instrument Bus Distribution Panels Instrument Bus Power Supply (Static Inverter) Motor Control Centers Integrated Protection Cabinets (IPC) Engineered Safety Features Actuation Cabinets (ESFAC) Logic Cabinets Reactor Trip Switchgear Reactor Coolant Pump Switchgear

Note:
* To comply with R.G. 1.158

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Table 3D.B-2
Aging Mechanism Sequence

Equipment	Location	Category	Burn-in	Aging Mechanisms						DBE	
				Thermal	Radiation	Mechanical	Vibration	Electrical	Seismic	Seismic	HELB
Safety-related Valve Motor Operators	I/C O/C	Harsh Harsh		X X	X X	X X	X X		X X	X X	X
Safety-related Solenoid Valves	I/C O/C	Harsh Harsh		X X	X X	X X	X X		X X	X X	X
Safety-related Externally Mounted Limit Switches	I/C O/C	Harsh Harsh		X X	X X	X X	X X		X X	X X	X
Pressure Transmitters	I/C&OC	Harsh	X	X	X				X	X	X
Differential Pressure Transmitters	I/C&OC	Harsh	X	X	X				X	X	X
Resistance Temperature Detectors: Well Mounted	I/C	Harsh		X	X		X		X	X	X
Excore Neutron Detectors	I/C	Harsh		X	X				X	X	X
Pressure Sensor	I/C	Harsh							X	X	X

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ATTACHMENT C

EFFECTS OF GAMMA RADIATION DOSES BELOW 10^4 RADS ON THE MECHANICAL PROPERTIES OF MATERIALS

C.1 Introduction

One potential common-mode failure mechanism to consider in the qualification of safety-related equipment is gamma radiation. As part of a qualification program, the effect of gamma radiation dose is considered for two purposes: as a component of the high-energy line break environment and as a potential aging mechanism that could reduce the capability of safety-related equipment to perform safety-related functions under design basis event conditions (seismic or high-energy line break).

The scope of this attachment is limited to consideration of the effect of radiation for that substantial portion of equipment that does not experience an adverse change in external environment as a result of a high-energy line break, and for which, therefore, the only gamma radiation concern is an in-service aging mechanism.

This attachment assumes that the equipment contains devices that have been selected for performance through the total integrated dose expected in service. For example, devices such as integrated circuits may have a limit of 1000 rads established, in which case the following discussion applies for its installed life. The information in this attachment is not adequate to be applied to equipment that must perform its function in a high-energy line break.

The primary purpose of equipment qualification is to reduce the potential for common-cause failures due to environmental effects during the qualified life. Random failures that inevitably occur inservice are accommodated by the redundancy and diversity of the design of safety-related systems. Furthermore, in-service maintenance and testing programs are designed to detect such random failures. The chances of two identical components that perform identical functions failing during the same limited time period in between routine tests considered insignificant because of the following:

- General low failure rate of components used in nuclear equipment
- Minor differences in component material or geometric tolerances or both
- Minor differences in operating environment.

Therefore, failures that are induced in components by normal background gamma radiation below 10^4 rads (10^3 rads for some devices) alone are considered to be random. Thus, the only gamma radiation concern addressed for equipment not subject to an adverse high-energy line break environment is the potential for an aging mechanism resulting in a deterioration in component properties such that, when subject to seismic stress, a common-cause failure results. When considering such a failure mode, the aging mechanism of concern is not one that affects the electrical properties of components but one that reduces the mechanical strength and flexibility of components.

C.2 Scope

This report summarizes available information concerning the effects of gamma radiation on material mechanical properties. It justifies that for a gamma dose of less than 10^4 rads there are no observable radiation effects that impact material mechanical properties. Of the materials investigated, only Teflon (PTFE) is subject to an alteration of mechanical properties for a gamma dose of less than 10^5 rads. Information is drawn from several sources listed as references in Section C.5. They include various texts concerning radiation effects and damage and pertinent reports.

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C.3 Discussion

The primary effects of gamma photons on materials are ionization, material heating (primarily at high dose rates, which is of negligible significance here), and some displacement damage caused by high-energy photons. Some other types of radiation have effects similar to those induced by gamma radiation. This allows the use of data obtained from exposure of material to an alternate radiation to provide limited information concerning the effects of exposure to gamma radiation.

For example, the primary consequence of fast-neutron bombardment of material is atom displacement. Therefore, if the effect of radiation on a material property is primarily dependent on atom displacement, it is inferred that for an equivalent dose (rads) of gamma and fast-neutron radiation, data obtained from neutron irradiation provides a conservative estimate of the effect of gamma irradiation in producing displacements.

The same type of inference is drawn for the ionization effect of charged particle (for example, electron, proton, alpha particle) irradiation. Charged particles do not have the penetration capability that gamma or neutron radiations exhibit as a result of extensive interaction between charged particles and atomic charge centers.

Table 3D.C-1 summarizes information derived from the listed references. The information relates to the effect of gamma radiation on material mechanical properties. Table 3D.C-1 presents either the threshold dose (that dose at which an effect on any mechanical property can first be detected) or, the dose that results in the identified effect. This provides a general indication of the susceptibility of material mechanical properties to gamma radiation.

An evaluation of the information available on inorganic materials summarized in Table 3D.C-1 shows that the mechanical damage threshold for gamma radiation is many orders of magnitude greater than 10^4 rads. For the organic materials listed in Table 3D.C-1, a histogram comparing threshold dose level and frequency of material susceptibility is provided. In instances for which a material threshold dose is not indicated in Table 3D.C-1, a threshold value is assumed which is one order of magnitude lower than the indicated damage dose. Where information is available, referenced documents indicate that the difference between threshold dose and 25 percent damage dose is about a factor of three. Thus, a factor of 10 supplies substantial margin in estimating the threshold dose level. Figure C-1 shows that any indications of mechanical property damage thresholds below 10^4 rads would be extremely unusual.

The references listed do not identify the existence of materials whose mechanical properties are deteriorated when exposed to a gamma radiation dose up to 10^4 rads. So it can be concluded that common-cause failures do not occur in electrical equipment during or after a seismic event as a result of radiation-induced degradation up to 10^4 rads.

This is supported by NRC documentation available as an attachment to "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," which provides further justification for the use of 10^4 rads as a threshold for mechanical damage. The NRC information appears to be consistent with the information provided in Table 3D.C-1.

C.4 Conclusions

For Class 1E equipment subject to a lifetime gamma dose of up to 10^4 rads, it is not necessary to address radiation aging for qualification purposes provided that the equipment is not required to perform a safety-related function in a high-energy line break environment.

As previously noted, this appendix does not apply to electrical properties of components in safety-related equipment.

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C.5 References

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Table 3D.C-1 (Sheet 1 of 2)
Radiation-Induced Degradation
of Material Mechanical Properties

Material	Mechanical Damage	Threshold Dose for Comments
Structural Metals	10^{19} n/cm^2 (fast neutron spectrum)	Similar to cold work (10^{10} rads)
Inorganic Materials	~10^{17} n/cm^2 (fast neutron spectrum)	Borated materials have lower threshold values for neutron irradiation.
Elastomers		
Natural Rubber	2x10^6 rads(C)	
Polyurethane Rubber	9x10^5 rads(C)	
Styrene-Butadiene Rubber	2x10^6 rads(C)	
Nitrile Rubber	7x10^6 rads(C)	Compression set is 25% degraded
Neoprene Rubber	7x10^6 rads(C)	
Hypalon	~10^7 rads(C)	Variable
Acrylic Rubber	9x10^7 rads(C)	Variable
Silicone Rubber	10^7 rads(C)	~25% damage
Fluorocarbon Rubber	9x10^7 rads(C)	~25% hardness, 80% elongation
Polysulfate Rubber	10^8 rads(C)	
Butyl Rubber	10^7 rads(C)	~25% damage
One rad (C) is the field of radiation that will produce 100 ergs/gm in carbon.		
Plastic		
Teflon (PTFE)	1.7x10^4 rads(C)	
Kel-F	1.3x10^6 rads(C)	
Polyethylene	≥ 10^7 rads(C)	
Polystyrene	10^8 rads	
Mylar	10^6 rads(C)	Conservative
Polyamide (Nylon)	8.6x10^5 rads(C)	
Diallyl Phthalate	10^8 rads(C)	
Polypropylene	10^7 rads(C)	
Polyurethane	7x10^8 rads(C)	

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Table 3D.C-1 (Sheet 2 of 2)
Radiation-Induced Degradation
of Material Mechanical Properties

Material	Mechanical Damage	Threshold Dose for Comments
Plastic (Continued)		
Kynar (400)	10^7 rads(C)	
Acrylics	$8.2x10^5$ rads	
Amino Resins	10^6 rads	
Aromatic Amide-Imide	10^7 rads	
Resins	10^7 rads	
Cellulose Derivatives	$3x10^7$ rads	25% damage
Polyester, Glass Filled	$8.7x10^8$ rads	
Phenolics	$3x10^8$ rads(C)	25% damage
Silicones	10^8 rads(C)	
Polycarbonate Resins	$5x10^7$ rads	25% damage to elongation
Polyesters	~ 10^5 - 10^6 rads	
Styrene Polymers	$4x10^7$ rads(C)	
Styrene Copolymers	$4x10^7$ rads(C)	25% damage
Vinyl Polymers	$1.4x10^6$ – $8.8x10^7$ rads(C)	
Vinyl Copolymers	$1.4x10^6$ – $8.8x10^7$ rads(C)	
Encapsulating Compounds		
RTV 501	$2x10^6$ rads	
Sylgard 182	$2x10^6$ rads	
Sylgard 1383	$2x10^6$ rads	
Polyurethane Foam	$2x10^6$ rads	
Epoxies	10^9 rads	

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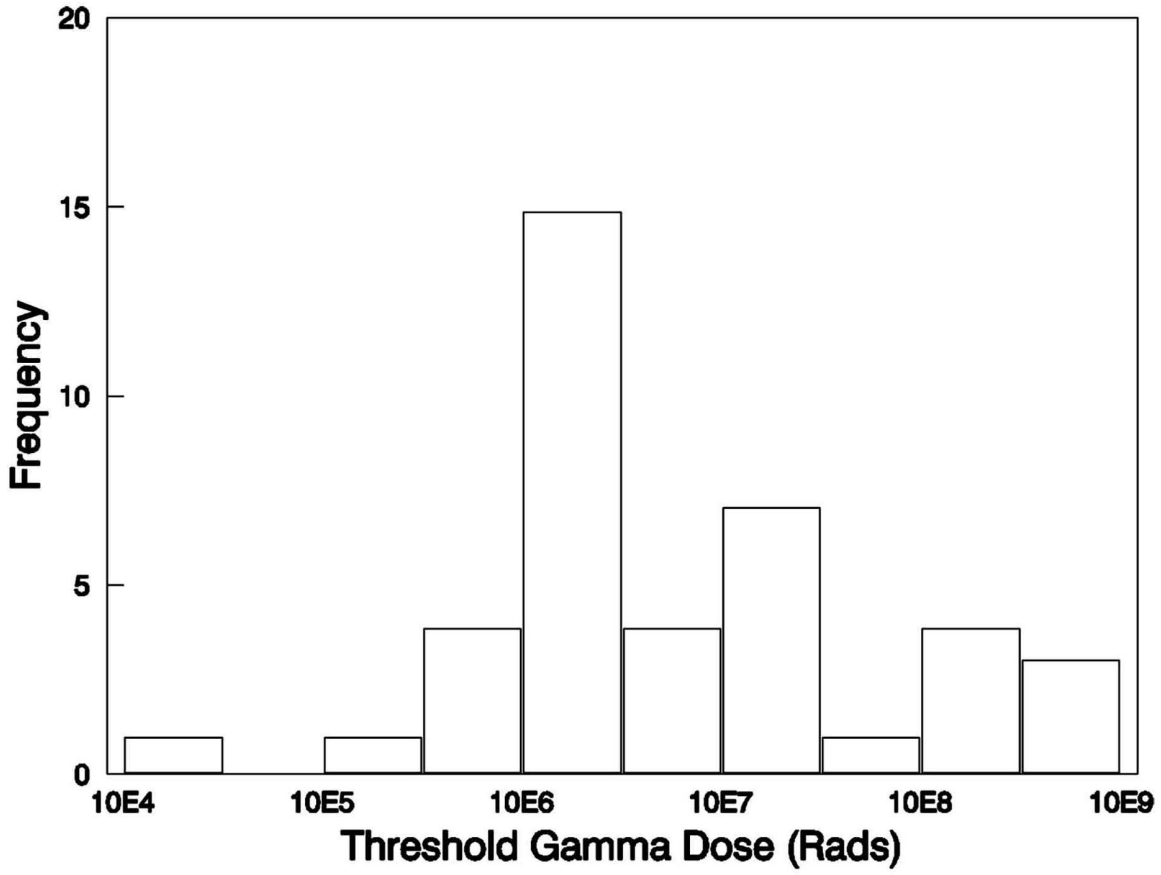

Figure 3D.C-1
Histogram of Threshold Gamma Dose for Mechanical Damage to Elastomers, Plastics, and Encapsulation Compounds

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ATTACHMENT D

ACCELERATED THERMAL AGING PARAMETERS

D.1 Introduction

Attachment B describes the approach employed in the AP1000 equipment qualification program to address the aging requirement of IEEE 323. For equipment required to perform a safety-related function in a high-energy line break environment, the AP1000 equipment qualification program includes an aging simulation as part of its qualification test sequence (Harsh Environment of Attachment B).

For equipment not required to perform a safety related function in a high-energy line break environment, the single design basis event considered is a seismic event. Aging, in this case (Mild Environment of Attachment B) is not usually included in the test sequence. Aging, where significant, is addressed by separate qualification of aged components, using conservative testing under applicable seismic design basis event conditions.

Thermal effects are one of the primary aging mechanisms addressed by the AP1000 equipment qualification program described in Attachment B for equipment containing nonmetallic or nonceramic materials. When thermal aging effects are established as potentially significant to the capability of the component or equipment to perform its safety-related function under design basis event conditions, or in the absence of evidence to the contrary, the component or equipment is thermally aged to simulate an end-of-qualified-life condition before design basis event testing. Equipment required to operate in a high-energy line break environment is also thermally aged to simulate the post-accident conditions consistent with its established functional requirements.

This attachment defines the appropriate thermal environments considered for each item of equipment in the AP1000 equipment qualification program and establishes consequent accelerated thermal aging parameters for use in the qualification programs.

D.2 Arrhenius Model

If an aging mechanism is governed by a single chemical reaction, the rate of which is dependent on temperature alone, the Arrhenius equation can be used as the basis for establishing the accelerated aging parameters:

$$\frac{dR}{dt} = Ae^{\frac{-E}{kT}} \quad (1)$$

where:

E = activation energy (eV)

k = Boltzmann's constant (8.617×10^{-5} eV/K)

A = constant factor

T = material temperature (K)

$\frac{dR}{dt}$ = reaction rate = aging rate

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Integration gives:

$$\Delta R = Be^{\frac{-E}{kT}} \Delta t \quad (2)$$

where:

ΔR = change in measured property due to aging

Δt = time for aging effect ΔR to occur

B = constant factor

If the accelerated aging process employed correctly simulates the change in properties due to aging under normal operating or post-accident temperature conditions, then:

$$\Delta R_1 = \Delta R_0 \quad (3)$$

and

$$B\, t_1\, e^{\frac{-E}{kT_1}} = B\, t_o\, e^{\frac{-E}{kT_0}}$$

and

$$Ln\, t_1 = \frac{-E}{k} \frac{T_1 - T_0}{T_1 T_0} + Ln\, t_o$$

where:

T_1 = accelerated aging material temperature (K)

t_1 = time at temperature T_1

T_0 = material temperature under normal operating or post-accident conditions (K)

t_0 = time at temperature T_0

From Equation 3, given an activation energy (E) for the material, the time required at any selected elevated temperature can be calculated to simulate the ambient aging effects.

This model has been verified to represent the thermal aging characteristics of nonmetallic and non-ceramic materials and is employed in the AP1000 equipment qualification program to derive accelerated thermal aging parameters. The only material dependent parameter input into this model, when establishing the accelerated aging parameters, is the activation energy. This parameter is a direct measure of the chemical reaction rate governing the thermal degradation of the material.

D.3 Activation Energy

A single material may have more than one physical property that thermally degrades (for example, dielectric strength, flexural strength.) As a consequence, the material exhibits different activation energies with respect to each property. The activation energy selected is the one that reflects the physical property most significant to the safety-related function performed or the stresses applied to the material by the design basis fault(s) considered.

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In actual practice, however, rarely is the choice so simple. Electrical components are invariably made up of more than one material. In many cases either the materials employed are not known in any chemical detail but just by a general organic or industrial trade name, or the appropriate activation energy is not known.

Where an activation energy is not available that reflects the material or component as well as the physical property of interest, a single conservative activation energy is used.

A distribution of activation energies (Figure 3D.D-1) was produced by EPRI (Reference 1) based on 170 materials. An independent review of materials used in Westinghouse-supplied equipment is summarized in Table 3D.D-1 and plotted in similar form in Figure 3D.D-2. A statistical analysis indicates that 95 percent of the activation energies exceed about 0.4 eV from the EPRI data and 0.6 eV from the Westinghouse data. Based on this information, a value of 0.5 eV is selected for use throughout this program whenever specific activation energies are not available. Employing a low value of activation energy in deriving the accelerated aging parameters causes materials having a high activation energy to be overaged with respect to the simulated conditions.

D.4 Thermal Aging (Normal/Abnormal Operating Conditions)

This section establishes the methodology employed and derives a typical set of accelerated aging parameters for equipment in various plant locations.

D.4.1 Normal Operation Temperature (T_0)

In determining the ambient operating temperature (T_0) of the component/material/equipment under investigation, the following is considered:

- External ambient temperature (T_a)
- Temperature rise in cabinet/enclosure (T_r)
- Self-heating effects (T_j)

where $T_o = T_a + T_r + T_j$

D.4.1.1 External Ambient Temperature (T_a)

a) For equipment located in areas supplied by an air-conditioning system, a typical value assumed for (T_a) throughout the qualified life is 68°F (20°C). For air-conditioning systems, two excursions per year to 91°F (33.3°C), each lasting 72 hours, has a negligible additional aging effect.

b) For equipment located in areas supplied by a ventilation system, a typical value assumed (T_a) throughout the qualified life is 77°F (25°C). Two excursions per year to 122°F (50°C), each lasting 72 hours, has a negligible additional aging effect.

D.4.1.2 Temperature Rise in Enclosure (T_r)

This temperature rise is estimated based on the heat generated (radiative and conductive) by equipment inside or attached to the enclosure. For example, limit switches may be affected by process heat through the valve. Temperatures measured during test runs may be available. A typical value for temperature rise inside an electronics cabinet is 10°C.

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D.4.1.3 Self-Heating Effects (T_j)

For equipment that is energized during most of its life, a self-heating effect is measured or established. If the equipment is energized only for short durations, this effect may be determined to be negligible. Temperature effects due to the solenoid of an energized valve may be significant (over 40°C). In determining junction temperatures of semiconductor devices, known operating parameters along with the thermal impedance are used. If the power dissipation is not known, a 50 percent operating stress is assumed.

D.4.2 Accelerated Aging Temperature (T_i)

Temperatures used for actual accelerated thermal aging tests are determined based on the equipment or component specifications in an attempt to prevent damage from high temperature alone and second-order (non-Arrhenius) effects such as the glass transition temperature of plastics. A maximum of 130°C is typically used for electronic component aging, but this is evaluated on a case basis. If the device is energized during the accelerated aging process, the self-heating effect as determined in the preceding section is added to the oven temperature to determine the total aging temperature (T_1).

D.4.3 Examples of Arrhenius Calculations

D.4.3.1 For a Normally Energized Component Aged Energized – The Self-Heating Effect is Added to Both (T_o) and (T_1):

Conditions: T_a = 25°C, T_r = 10°C

T_j = 25°C, eV = 0.5,

Aging time = t_i

Oven temperature = 130°C

Qualified life goal = 10 years

Therefore T_o = 25 + 10 + 25 = 60°C = 333K

T_1 = 130 + 25 = 155°C = 428K

$$t_1 = 10e^{\frac{-0.5}{K}\frac{(428-333)}{(428 \times 333)}} * 8760 = 1831 \text{ hours}$$

D.4.3.2 For a Normally De-energized Component Aged Energized – the Self-heating Effect is Added Only to T_1:

Conditions: T_a = 25°C, T_r = 10°C

T_j = 25°C, eV = 0.5, Aging time = t_1

Oven temperature = 130°C

Qualified life goal = 10 years

Therefore T_o = 25 + 10 = 35°C = 308K

T_1 = 130 + 25 = 155°C = 428K

$$t_1 = 10e^{\frac{-0.5}{K}\frac{(428-308)}{(428 \times 308)}} * 8760 = 445 \text{ hours}$$

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D.5 Post-Accident Thermal Aging

Most cases, some safety-related postaccident performance capability is specified by the functional requirements. As a consequence, to qualify equipment to IEEE 323, the effects of post-accident thermal aging must be simulated after the high-energy line break test. This section establishes the accelerated thermal aging parameters employed in performing this simulation.

D.5.1 Post-Accident Operating Temperatures

Assuming continuous operation of containment safeguards systems following an accident, the containment environment temperature is reduced to the external ambient temperature well within one year for any postulated high-energy line break. However, to allow for possible variations in plant operations following an accident, the design high-energy line break envelope extending to one year is indicated by Figure 3D.5-8 or an equivalent curve considering the location of the equipment.

For safety-related equipment located inside containment, either the self-heating effects of the operating unit, under post-accident conditions, may be insignificant compared to the heat input from the external environment (transmitters, RTDs), or the unit may not be in continuous operation during this phase (valve operators). So it may not be necessary to add a specific temperature increment to account for self-heating of these devices following an accident. The portion of Figure 3D.5-8 (or the localized equivalent) that is not addressed by DBA testing is then input at T_0 into the Arrhenius equation to calculate appropriate accelerated aging parameters for post-accident conditions. However, as noted in Section D.4, if the equipment is energized during the aging simulation period, the self-heating effect is added to both T_o and T_1.

D.5.2 Accelerated Thermal Aging Parameters for Post-Accident Conditions

The Arrhenius equation is used as the basis for establishing accelerated thermal aging. The aging temperature most often used for post-accident thermal simulation is 250°F (121°C). This temperature is selected as a maximum for electronic components and is generally used for tests. When applying the Arrhenius equation, conservative (lower bound) material activation energy is used that represents the equipment materials and material physical properties, and is essential to the safety function performance during the DBA. There are various ways to apply the Arrhenius equation to post-accident thermal aging. If the selected aging temperature is greater than the test temperature (including margin) at 24 hours, the Arrhenius equation is applied to the post-DBA temperature curve from the first day to the end of the post-accident operability time (e.g.; 72 hours, 2 weeks, 4 months or 1 year) in small time increments. The required aging times to simulate these small increments are then summed to yield a total test time. Another method is used if the selected aging temperature is less than the test temperature (including margin) after 24 hours, then the test temperature should be applied until the aging temperature is met. The Arrhenius equation is applied to the post-DBA temperature curve from the time the aging temperature is met to the end of the post-accident operability time.

D.6 References

1. EPRI NP-1558, Project 890-1, "A Review of Equipment Aging Theory and Technology," September 1980.

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Table 3D.D-1 (Sheet 1 of 2)
Activation Energies From Westinghouse Reports

Material	Electron Volts
Melamine-Glass, G5	0.29
Epoxy B-725	0.48
Ester-Glass, GPO-3	0.57
RTV Silicone	0.60
Phenolic-Asbestos, A	0.61
Nylon 33 GF	0.70
Acetal	0.73
Mineral Phenolic	0.74
Silicone Varnish	0.74
Polypropylene	0.81
Polysulfone	0.83
Phenolic-Cotton, C	0.84
Formvar	0.85
Epoxy	0.88
Epoxy Adhesive	0.89
Nylon	0.90
Pressboard	0.91
Kapton	0.93
Silicone	0.94
Phenolic-Asbestos, A	0.94
Cast Epoxy	0.98
Urethane-Nylon	0.99
Phenolic-Glass, G-3	1.01
Polycarbonate	1.01
Phenolic-Paper, X	1.02
Epoxy Wire	1.05
Epoxy-Glass, FR-4	1.05
Varnish Cotton	1.06
PVC	1.08
Ester-Glass, GPO-1	1.09
Cellulose Phenolic	1.10
X-Link Ethylene	1.11
Urethane	1.12
Ester-Glass, GPO-2	1.13
Ester-Nylon	1.14

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Table 3D.D-1 (Sheet 2 of 2)
Activation Energies From Westinghouse Reports

Material	Electron Volts
Ester-Glass, GPO-1	1.16
32102BK Varnish	1.16
Vulcanized Fiber	1.16
Cellulose Mineral Phenolic	1.17
Mylar	1.18
Cast Epoxy	1.18
32101EV Varnish	1.18
Epoxy	1.18
Silicone	1.18
Phenolic-Paper, XX	1.20
Vulanized Fiber	1.21
Cellulose Phenolic	1.24
Phenolic-Glass, G-3	1.24
Kraft Phenolic	1.25
Neoprene	1.26
Amide-Imide Varnish	1.31
Loctite 75	1.38
Acetyl. Cotton	1.39
Silicone-Asbestos	1.41
Epoxy-Glass, FR-4	1.50
Mylar	1.58
Nomex	1.59
Omega Varnish	1.59
Epoxy-Glass, G-11	1.64
Polythermaleze	1.64
Kraft Paper	1.67
Valox 310SE-0	1.75
Varnished Kraft	1.86
Nomex	1.91
Ester-Glass, GPO-3	2.03
Phenolic-Cotton, C	2.12
Melamine-Glass, G-5	2.18

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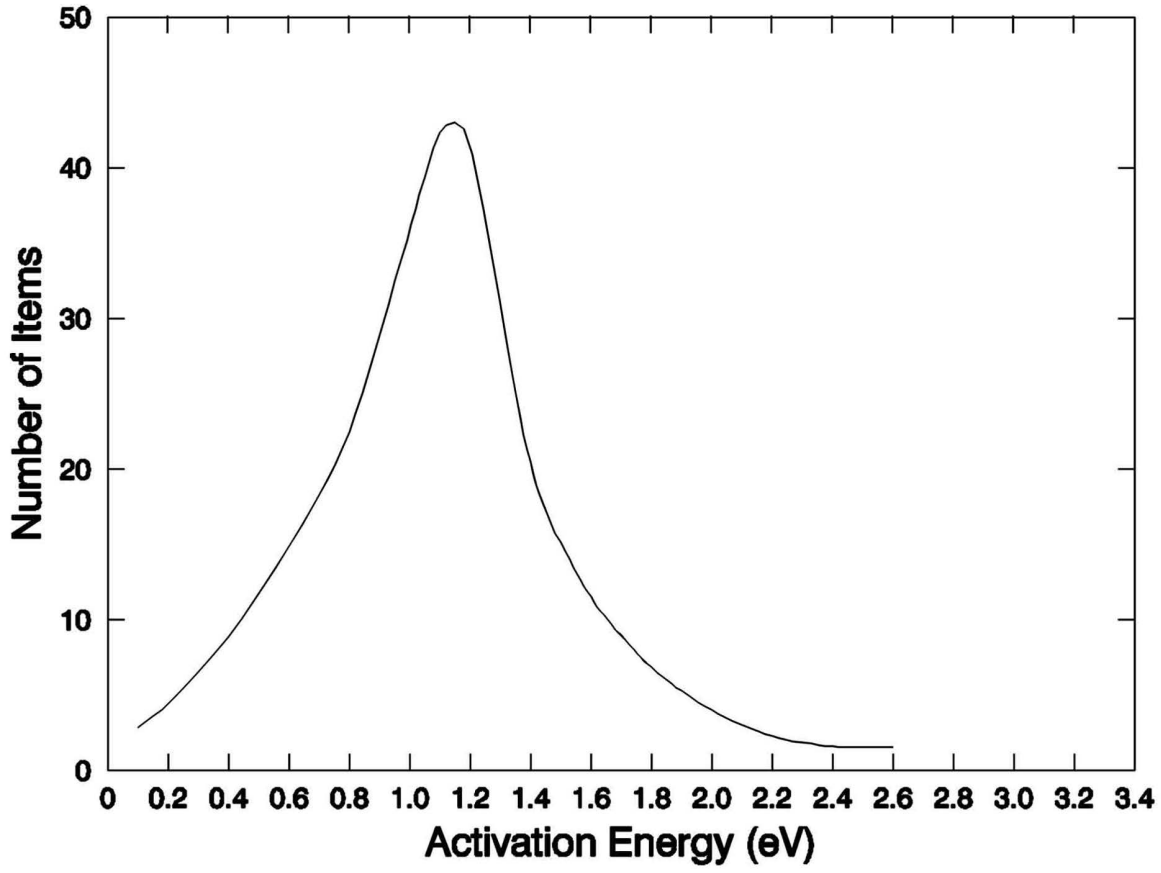

Figure 3D.D-1
Frequency Distribution of Activation Energies of Various Components/Materials (EPRI Data)

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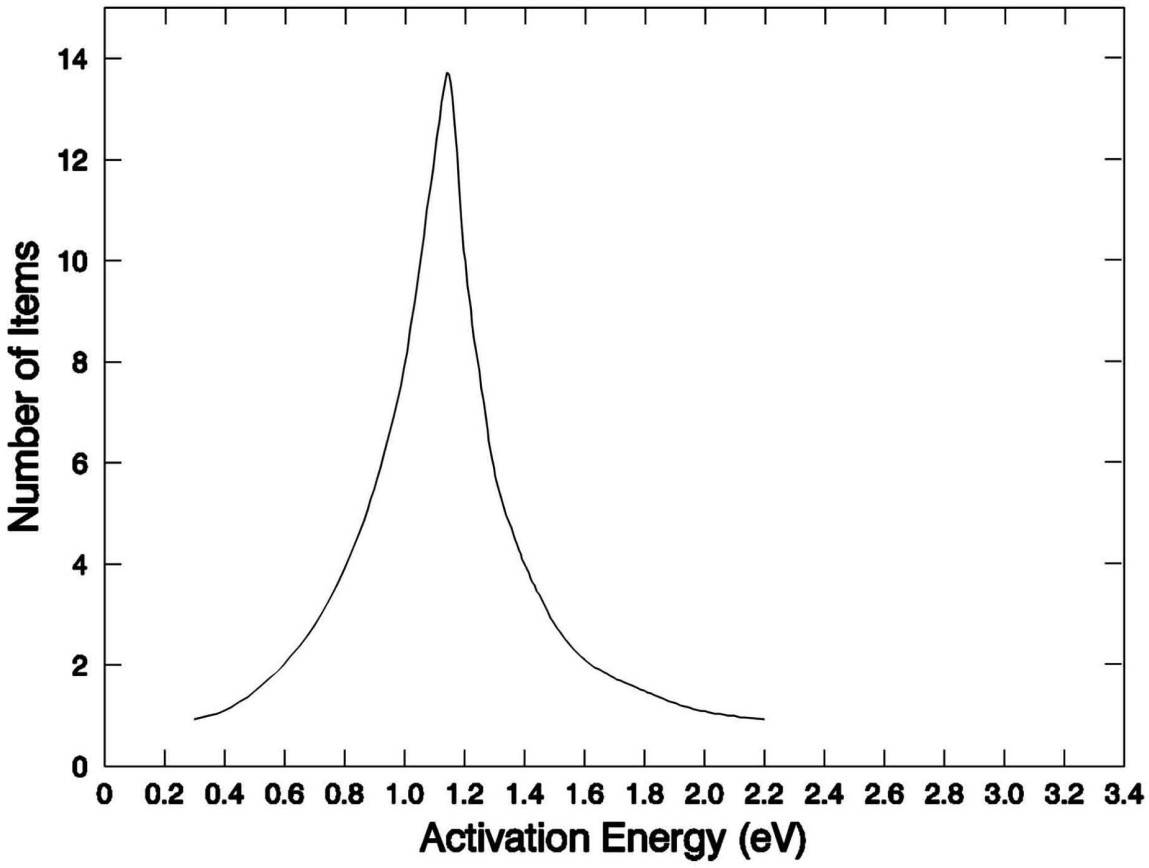

Figure 3D.D-2
Frequency Distribution of Activation Energies of Various Components/Materials
(Westinghouse Data)

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Figure 3D.D-3 Not Used

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ATTACHMENT E

SEISMIC QUALIFICATION TECHNIQUES

E.1 Purpose

The following is the methodology used to seismically qualify seismic Category I mechanical and electrical equipment for the AP1000 equipment qualification program. Qualification work covered by this appendix meets the applicable requirements of IEEE 344-1987 and 382-1996.

E.2 Definitions

The following are definitions of terms unique to or distinct from common industry usage. (See Section E.4.2.)

E.2.1 1/2 Safe Shutdown Earthquake

The 1/2 safe shutdown earth (SSE) is the earthquake level used during seismic testing to seismically age safety-related equipment before performing safe shutdown earthquake testing.

E.2.2 Seismic Category I Equipment

Seismic Category 1 equipment consists of structures, systems, and components required to withstand the effects of the safe shutdown earthquake and remain structurally intact, leak-tight (in case of pressurized systems), and functional to the extent required to perform their safety-related function.

E.2.3 Seismic Category II Equipment

Seismic Category II equipment is that equipment whose continued function is not required, but whose failure could reduce the functioning of seismic Category I structures, systems, and components to an unacceptable level. Seismic Category II equipment must be capable of maintaining structural integrity so that a seismic event up to and including an SSE would not cause such a failure.

E.2.4 Non-seismic Equipment

Equipment designated as non-seismic does not require seismic qualification.

E.2.5 Active Equipment

Equipment that must perform a mechanical or electrical operation during or after (or both) the safe shutdown earthquake in order to accomplish its safety-related function.

E.2.6 Passive Equipment

Equipment where maintenance of structural or pressure integrity is the only requirement necessary for accomplishing its safety-related function.

E.3 Qualification Methods

This section presents a general description of the seismic qualification methods used by AP1000 for the seismic qualification of seismic Category I safety-related mechanical and electrical equipment. Three methods are used: test, analysis, and a combination of the two. The approaches for qualification by testing and by analysis are discussed in Section E.5 and Section E.6, respectively. The following discussion covers the conditions under which each approach is used and the general

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requirements applicable to the use of the methods. The qualification sequence is defined in Appendix 3D.

E.3.1 Use of Qualification by Testing

The preferred method for seismic qualification of safety-related Class 1E electrical and electromechanical equipment is seismic testing. The nature of the seismic and vibrational input used depends on where the equipment is used. For equipment mounted so that the seismic environment includes frequency content between 1 and 33 hertz (hard mounted), the seismic test input is multifrequency. For equipment mounted so that seismic ground motion is filtered to contain one predominant structural mode (line mounted), single frequency testing is appropriate. This is the case for equipment mounted on piping systems, ductwork, or cable trays.

E.3.2 Use of Qualification by Analysis

Analysis is used for seismic qualification when one of the following conditions is met:

- The equipment is too large or the interface support conditions cannot adequately be simulated on the test table.
- The only requirement is to maintain structural integrity during a postulated seismic event.
- The equipment represents a linear system, or the nonlinearities can conservatively be accounted for in the analysis. This approach is also applicable to the development of the seismic environment, required response spectrum curve, at the mounting location of a component attached to a larger structure when the device is seismically qualified by separate component testing.
- The analysis is used to document the seismic similarity of the equipment provided and that previously qualified by testing.

Seismic qualification of safety-related electrical equipment by analysis alone is not recommended for complex equipment that cannot be modeled to adequately predict its response. Analysis without testing may be acceptable provided structural integrity alone can ensure the design-intended function.

E.4 Requirements

E.4.1 Damping

Damping level of a component or system describes its capability to dissipate vibrational energy during a seismic event. The damping level used defines the response magnitude of an ideal single degree of freedom linear oscillator when subjected to the specified input as documented by the required response spectrum (RRS) curve. The significance of the damping value used depends on whether qualification is by testing or analysis.

E.4.1.1 Testing

Equipment qualification by testing involves subjecting the base of the equipment to a representative seismic acceleration time history. The response characteristics of the equipment are a function of the inherent damping present in the equipment. In this case the damping value used (typically five percent) serves as a convenient means of showing the compliance of the test response spectrum (TRS) with the required response spectrum.

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E.4.1.2 Analysis

In the case of qualification by analysis, the damping level used is representative of the damping actually present in the equipment. Unless other documented equipment damping data is available, the values specified in Table 3.7.1-1 of Chapter 3 are used.

E.4.2 Interface Requirements

As part of the seismic qualification program, consideration is given to the definition of the clearances needed around the equipment mounted in the plant to permit the equipment to move during a postulated seismic event without causing impact between adjacent pieces of safety-related equipment. This is done as part of seismic testing by measuring the maximum dynamic relative displacement of the top and bottom of the equipment.

When performing qualification by analysis, the relative motion is obtained as part of the analytical results. These motions are reported in the seismic analysis report or qualification report and are used to determine the required clearance between adjacent pieces of equipment.

In addition, the qualification program takes into account the restraining effect of other interfaces, such as cables and conduits attached to the equipment, which may change the dynamic response characteristic of the equipment.

E.4.3 Mounting Simulation

The mounting conditions simulated by analysis or during seismic test are representative of the equipment as-installed mounting conditions used for the AP1000 equipment. When an interfacing structure exists between the safety-related equipment being qualified and the floor or wall at which the equipment mounting required response spectrum is specified, its flexibility is simulated as part of the qualification program. If this is not done, justification must be provided, demonstrating that the deviations in mounting conditions do not affect the applicability of qualification program.

E.4.4 1/2 Safe Shutdown Earthquake

The AP1000 makes use of a small earthquake having the intensity of one-half of the safe shutdown earthquake at the safety-related equipment mounting location to simulate the fatigue effects of smaller earthquakes that may occur before the postulated safe shutdown earthquake. These small earthquakes correspond to the operating basis earthquakes (OBEs) referenced in IEEE 344-1987. When qualification by testing is used, five of these small earthquakes are used to vibrationally age the equipment before the safe shutdown earthquake. When qualification by analysis is used, two safe shutdown earthquake events are used to simulate the fatigue aging effects. Each event contains 10 peak cycles. These stress cycles are used to verify that the equipment is not subject to failure due to low cycle fatigue.

E.4.5 Safe Shutdown Earthquake

The safe shutdown earthquake required response spectrum curve defines the seismic qualification basis for each piece of safety-related equipment. The seismic level varies according to the mounting location of the equipment. When equipment qualification is based on testing, an additional 10 percent test acceleration margin is added as specified in IEEE 323-1974.

E.4.6 Other Dynamic Loads

Hydrodynamic loads are considered as part of the qualification program, where applicable.

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E.5 Qualification by Test

Seismic qualification testing is the preferred method for electrical, mechanical, and electromechanical equipment. Seismic testing shall be performed and input generated as specified in IEEE 344-1987. The nature of the test input used depends on whether the equipment is hard mounted or line mounted. The test program consists of the following elements, as applicable: environmental aging, mechanical aging, vibrational aging, and safe shutdown earthquake testing. For those cases where the equipment is also subject to a loss of coolant or a high-energy line break accident, these accidents are simulated on the same qualification specimen after completion of the testing previously discussed. (See Subsections 3D.4.4 and 3D.7.4.)

The characteristics of the required seismic and dynamic input motions should be specified by the response spectrum or time history methods. These characteristics, derived from the structures or systems seismic and dynamic analyses, should be representative of the input motions at the equipment mounting locations.

For seismic and dynamic loads, the actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (that is, the test response spectrum should closely resemble and envelop the required response spectrum over the critical frequency range).

Since seismic and the dynamic load excitation generally have a broad frequency content, multi-frequency vibration input motion should be used. However, single frequency input motion, such as sine beats, is acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (for example, by structural filtering effects), or that the anticipated response of the equipment is adequately represented by one mode, or in the case of structural integrity assurance, that the input has sufficient intensity and duration to produce sufficiently high levels of stress for such assurance. Components that have been previously tested to IEEE-344-1971 should be reevaluated or retested to justify the appropriateness of the input motion used, and requalified if necessary.

For the seismic and dynamic portion of the loads, the test input motion should be applied to one vertical axis and one principal axis (or two orthogonal axes) simultaneously unless it can be demonstrated that the equipment response motion in the horizontal direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

E.5.1 Qualification of Hard-Mounted Equipment

Hard-mounted equipment is seismically tested mounted on a test table capable of producing multifrequency, multiaxis inputs. The waveform characteristics of the input are random and scaled in such a way that the test response spectrum equals or exceeds the required response spectrum (including margin). The input signal meets the requirements of Subsection 7.6.3 of IEEE 344-1987.

Furthermore, the test input simulates the multidirectional nature of the earthquake. The preferred method for meeting this requirement is to the use a triaxial test table capable of producing three statistically independent, orthogonal input motions. In this case the seismic testing consists of five 1/2 safe shutdown earthquake tests and one safe shutdown earthquake test in one orientation.

Using a biaxial test table is acceptable if it is justified that the horizontal and vertical test inputs conservatively simulate the three-dimensional nature of the seismic event. One acceptable approach

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is to mount the equipment on the test table with its front-to-back axis oriented at 45 degrees to the horizontal drive axis and scale the horizontal component of the input by a factor of the square root of two. Statistically independent inputs are preferred and, if used, the test can be performed in two stages, with the equipment rotated 90 degrees about the vertical axis. In this case, the five 1/2 safe shutdown earthquake inputs need to be applied only in the first orientation.

If a dependent biaxial test table is used, the test is performed in four stages. The first stage involves five 1/2 safe shutdown earthquake tests and one safe shutdown earthquake test in the first orientation. The second, third, and fourth orientations are obtained by successively rotating the equipment 90 degrees clockwise from its previous position. One safe shutdown earthquake test is performed in each of the last three orientations.

Each multifrequency test has a minimum of 15 seconds of strong motion input. The strong motion portion is preceded and followed by a period of testing where the test input is ramped up and ramped down, respectively, so that the equipment is not subjected to impact loading. The adequacy of each test run is evaluated using the criteria set forth in Subsection 7.6.3.1 of IEEE 344-1987.

E.5.2 Qualification of Line-Mounted Equipment

Line-mounted equipment, because of the dynamic filtering characteristics of its mounting, is effectively subject to single frequency input. This condition is common for valves and sensors supported by piping systems, cable trays, and duct systems. This equipment is qualified consistent with the requirements of IEEE 382-1996.

In some cases this equipment may also be used in the hard-mounted condition. In this case multifrequency, multiaxis testing is also required unless justification is provided that the previous single frequency tests demonstrate the capability of the equipment to operate under the hard-mounted seismic conditions. Because of the large size of typical valves, it may be necessary to perform separate testing of the operators and valve assembly.

E.5.2.1 Seismic Qualification Test Sequence

The seismic qualification process is broken down into the following steps:

1. Mount the equipment on a rigid test fixture and perform a resonant search test to demonstrate that the equipment is structurally rigid (fundamental frequency greater than 33 hertz) and does not amplify the seismic motions acting at the equipment mounting interface.

2. Perform single frequency testing on the line-mounted equipment.

3. Perform multifrequency, multiaxis testing on the equipment, if appropriate.

4. If an active valve assembly is to be seismically qualified, additional testing is needed as follows:

    a. Perform a static pull test on the valve.

    b. Perform a static seismic analysis using a verified model of the valve and its extended structure to demonstrate that the valve has adequate structural strength to perform its safety-related function without exceeding the design allowable stresses specified in ASME Code, Section III, Subsection NB, NC, or ND for pressure-retaining parts, as

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appropriate, and Subsection NF for non-pressure-retaining boundary parts. Limiting extended structure stress to material yield strength minimizes deflections, which could interfere with valve stroke function.

E.5.2.2 Line Vibration Aging

Line-mounted equipment may be subject to operational vibrations resulting from normal plant operations. The potential fatiguing effect of this vibrational aging is simulated as part of the qualification program. This requirement is satisfied by subjecting the equipment to a sine sweep from 5 to 100 to 5 hertz at an acceleration level of 0.75g or such reduced acceleration at low frequencies to limit the double amplitude to 0.025 inch as specified in Section 5.3.a, Part III of IEEE 382-1996.

E.5.2.3 Single Frequency Testing

The single frequency testing acceleration waveform is either sine beat or sine dwell applied at one-third octave frequency intervals as specified in IEEE 382-1996. Each dwell has a time length adequate to permit performance of functional testing, with a minimum time of 15 seconds. To account for the three-dimensional nature of the seismic event, the test input level is taken as the square root of two times the required input motion (RIM) level specified in IEEE 382. The level includes the 10 percent test margin. Each test series is performed using single axis input. The test series is performed successively in each of three orthogonal axes.

E.5.2.4 Seismic Aging

The aging effect of the five 1/2 safe shutdown earthquake earthquakes can be simulated by exposing the equipment to two sinusoidal sweeps at one-half of the safe shutdown earthquake required input motion level in each orthogonal axis. Each sweep shall go from 2 to 35 hertz to 2 hertz at a rate not to exceed one octave per minute. One sweep is performed with the equipment in its inactive mode, and the other with the equipment in its safety-related operational mode.

E.5.2.5 Static Deflection Testing of Active Valves

The seismic testing just discussed is normally performed only on the valve operator and the attached appurtenances. If the valve assembly is rigid, the operability of the valve assembly during a postulated seismic event may be demonstrated by performing a static pull test using a peak acceleration value equivalent to a triaxial acceleration of 6g. If the valve assembly is determined to be flexible, a supplemental analysis of the seismic response of the flexible valve and its supporting piping is performed to determine the actual acceleration level present at the center of gravity of the valve assembly.

The valve is placed in a suitable test fixture with the operator and appurtenances mounted and oriented as in the normal valve assembly installation. The valve is mounted so that the extended structure is freestanding and supported only by the valve nozzles. The valve is positioned so that the horizontal and vertical load components simulating the three-dimensional nature of the seismic event produce a worst-case stress condition in the valve extended structure.

During testing, the valve shall be internally pressurized and nozzle loads applied. Static loads simulating dead weight and seismic loads are applied to the extended structure. The tests are normally performed at ambient temperature. These loads simulate to the extent feasible the load distribution acting on critical parts of the valve assembly. The valve is actuated using the actuator system seismically qualified according to IEEE 382-1996. The valve assembly is cycled from its normal to the desired safety-related position within the time limits defined in the equipment specification. Leakage measurements are made, where required, and compared to the allowable values specified in the valve design specification.

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E.5.3 Operational Conditions

When equipment being qualified performs a safety-related function during the safe shutdown earthquake, the equipment is operated and monitored to demonstrate that the equipment functions properly before, during, and after the seismic event. If the test time is not long enough to complete the required functional tests, the length of the strong motion test time is increased to permit completion of the required functional testing.

Where functional testing is dependent on external electrical supply, the testing is performed using the worst-case electrical supply conditions.

E.5.4 Resonant Search Testing

Resonant search testing is performed to provide data on the natural frequency and dynamic response characteristics of the equipment qualified. For hard-mounted equipment being qualified by seismic testing, resonant search testing is done to provide additional information but is not required for qualification of the equipment. This is an important consideration because frequency testing for hard-mounted equipment is normally performed with the equipment mounted on the test table, where dynamic interaction of the table and the equipment has a significant effect on the measured natural frequency.

For qualification of line-mounted valve assemblies, it is necessary that the assemblies be rigid. To meet this requirement, the assembly mounted to a rigid test fixture so that the frequencies measured are indeed representative of the valve assembly. If it is not feasible to provide a rigid fixture, as is likely the case when testing such very large valves, as the main steam and feedwater isolation valves, additional tests and analyses may be required to determine if the apparent flexibility measured is due to the test fixture or to the characteristic of the valve assembly itself.

If the resonant search test data is being generated to verify the accuracy of an analytical modeling technique, the test specimen mounting details must accurately simulate the boundary conditions used in the analytical model.

E.6 Qualification by Analysis

Section E.3.2 defines the limits on the use of analysis to demonstrate seismic qualification of safety-related equipment. The following sections describe the analytical methods to be employed for qualification of equipment. There are two techniques, static and dynamic, used to qualify equipment. The success of either method depends on the ability of the analytical model to describe the response of the system to seismic loads. Alternative methods of analysis are accepted if their conservatism is documented.

The analysis is used to demonstrate the structural adequacy of the equipment being qualified. This is done by showing that the calculated stresses do not exceed the design allowable stresses specified in ASME Code, Section III, Subsection NB, NC, or ND for pressure-retaining equipment and Subsection NF for nonpressure-retaining equipment.

E.6.1 Modeling

Analysis may be performed by hand calculations, finite element, or mathematical models that adequately represent the mass and stiffness characteristics of the equipment. The model contains enough degrees of freedom to adequately represent the dynamic behavior over the frequency range of interest. It includes the essential features of the equipment.

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Dynamic properties reflect the in-service operating conditions, such as structural coupling, dynamic effects of contained liquids, and externally applied restraints (where appropriate). Where the modeled equipment exhibits some nonlinear behavior, this nonlinearity is modeled unless justification is provided that it is insignificant or that the linear model provides conservative results. The adequacy of the model or of the modeling techniques is shown by comparing the predicted responses to the responses predicted by benchmark problems or modal testing. Acceptable benchmark problems include hand calculations, analysis of the same problem using a comparable verified public-domain program, empirical data, or information from the technical literature.

In addition to documenting the modeling technique, a quality assurance program is in place that defines the requirements for the control, verification, and documentation for the computer programs used for qualification of safety-related equipment. The computer programs used in the qualification process are verified on the same computer on which the qualification analysis is performed.

E.6.2 Qualification by Static Analysis

For rigid equipment, the seismic forces resulting from one seismic input direction are calculated for each node point by multiplying the nodal mass in that direction by the appropriate zero period acceleration (ZPA) floor acceleration. The combined system response of the equipment to the simultaneous loads acting in all three directions is calculated by combining the three components, using the square root sum of the squares (SRSS) method. The square root sum of the squares method is used to account for the statistical independence of the individual orthogonal seismic components.

E.6.3 Qualification by Dynamic Analysis

If the lowest natural frequency of the equipment lies below the cutoff frequency, the response of the equipment to the seismic event in each orthogonal direction will be dynamically amplified and the equipment is said to be flexible. The analysis is performed in compliance with the guidelines set forth in the SSAR and in Regulatory Guides 1.92, 1.100, and 1.122.

The preferred method of analysis is the response spectrum method. In this method the responses in each equipment mode are calculated separately and combined by the square root sum of the squares method, provided the modes are not closely spaced. (Consecutive modes are said to be closely spaced if their frequencies differ from that of the first mode in the group by less than 10 percent.) The responses for each mode in a group are combined absolutely. The group response is then combined with the remaining modal responses using the square root sum of the squares method. The responses for each of the three orthogonal seismic components can then be combined as discussed in Section E.6.2. The applicable damping levels are noted in Table 3.7.1-1 of Chapter 3.

E.6.3.1 Response Spectrum Analysis

Modes up to and including the cutoff frequency are included in this summation. In some cases, the structure is basically rigid, with some of the flexible mode representing local effects. This situation is evaluated by reviewing the modal masses applicable to a given seismic input direction. If the sum of the effective modal masses used in the response spectrum analysis is greater than 0.9 times the total equipment mass, the model is assumed to adequately represent the total equipment mass. If this criterion is not satisfied, it means that a significant part of the equipment seismic response is due to the static seismic response of the higher equipment modes (above the cutoff frequency). If this situation occurs, the analyst determines the component of the response due to the higher modes and combines it with the flexible response component by square root sum of the squares. (This requirement is discussed in the SSAR, Subsection 3.7.2.)

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E.6.3.2 Static Coefficient Method

As an alternative to the response spectrum method, the static coefficient method of analysis may be used. In this method the frequencies of the equipment are not determined, but a static analysis is performed, assuming that a peak acceleration equal to 1.5 times the peak spectral acceleration given in the applicable required response spectrum acts on the structure as described in Section E.6.2.

The static coefficient of 1.5 takes into account the combined effects of multifrequency excitation and multimode response for equipment and structures that can be represented by a simple model. A lower static coefficient may be used when it can be demonstrated that it will yield conservative results.

E.6.3.3 Time History Analysis

The time-history method of analysis is the preferred method of analysis when the equipment exhibits significant nonlinear behavior or when it is necessary to generate response spectra for specific component mounting locations in the equipment. The acceptable methods that are used to develop the seismic time histories are discussed in Regulatory Guide 1.122, ASME Code, Section III, Appendix N, and in Section 6.2 of IEEE 344-1987. Other analytical methods may be used to generate in-equipment response spectra provided that they are verified to produce accurate and/or conservative results.

E.7 Qualification by Test Experience

This method of qualification is not used.

E.8 Performance Criteria

E.8.1 Equipment Qualification by Test

The performance criterion for qualification of equipment is that the equipment successfully perform its safety-related function during and after the postulated seismic event. Acceptance requires, as a minimum, that:

- No spurious or unwanted outputs occur in the circuits that could impair the safety-related functional operability of the equipment;
- No gross structural damage of the equipment occur during the seismic event that could lead to the equipment or any part thereof becoming a missile. Local inelastic deformation of the equipment is permitted; and,
- Satisfactory completion of specified baseline tests are demonstrated before, during, and after the seismic test sequence.

E.8.2 Equipment Qualification by Analysis

E.8.2.1 Structural Integrity

The analysis verifies that the equipment, when subjected to the worst case combination of operating and seismic loads, maintains its structural integrity. In addition the analysis shows that the equipment is not subject to low cycle fatigue failure when subject to postulated seismic loading. Finally the analysis verifies that seismically induced equipment motion does not lead to impacting with other nearby equipment.

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E.8.2.2 Operability

Analysis can be used to demonstrate equipment operability for those pieces of equipment where structural integrity or limitation of deformation guarantees operability. As an example the analysis of active equipment verifies that the combination of operating and postulated seismic loads do not produce stress levels or deformations that exceed established functional limits. The rationale for use of these limits is justified.

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Appendix 3E High-Energy Piping in the Nuclear Island

This appendix identifies high-energy piping in the nuclear island with a diameter larger than 1 inch. Candidate leak-before-break piping is identified in Figures 3E-1 through 3E-5 along with other piping for which high-energy pipe failures are postulated. These figures also identify piping in the break exclusion zones inside and outside containment. These figures do not include piping of 1 inch size and smaller. These figures do not include the 6-inch diameter pipe segments on the high-energy reactor vessel head vent piping, or the 3-inch diameter high-energy lines for the reactor coolant pump heat exchangers which are evaluated in accordance with Subsection 3.6.2 and to which the leak-before-break methodology is not applied. Instrumentation and instrumentation lines are not included, except where a branch connection of greater than 1 inch is required.

The selection of the failure type is based on whether the system is high or moderate energy during normal operating conditions of the system. High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig. Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high energy are considered moderate energy. Piping systems that exceed 200°F or 275 psig for 2 percent or less of the time during which the system is in operation or that experience high-energy pressures or temperatures for less than 1 percent of the plant operation time are considered moderate energy. In piping whose nominal diameter is greater than 1 inch but less than 4 inches, only circumferential breaks are postulated at each selected location. No breaks are postulated for piping whose nominal diameter is 1 inch or less.

The three-letter code included in the line numbering identifies the pipe specification. The letters define the pressure class, material specification, and AP1000 equipment classification, respectively. The symbols used in Figures 3E-1 through 3E-5 are the same as the P&ID figures. See Figure 1.7-2 for additional information on the drawing legend and for the key for the pipe specification. Section 3.2 includes additional information on the AP1000 equipment classification.

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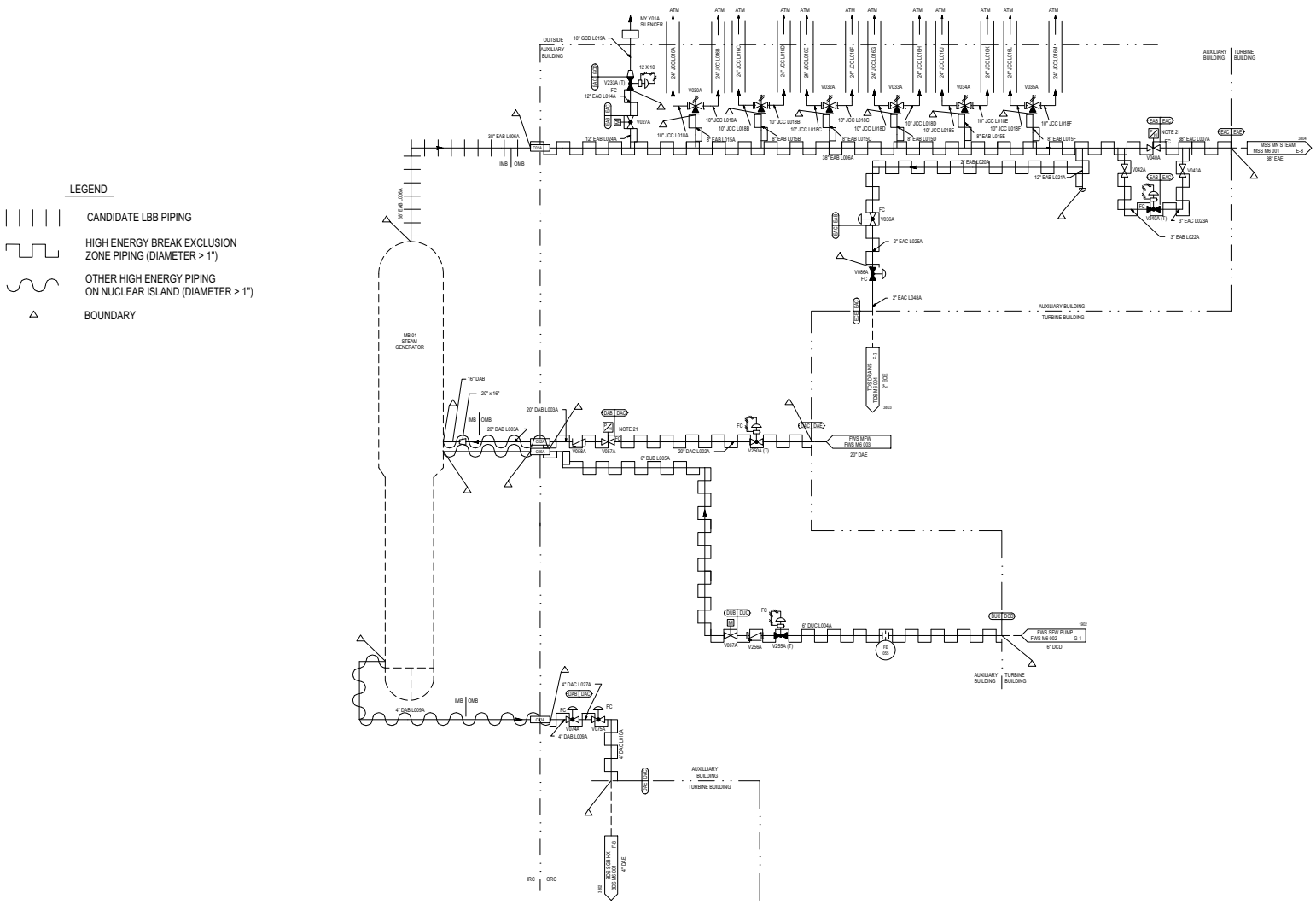

Figure 3E-1 (Sheet 1 of 2)
High Energy Piping – Steam Generator System

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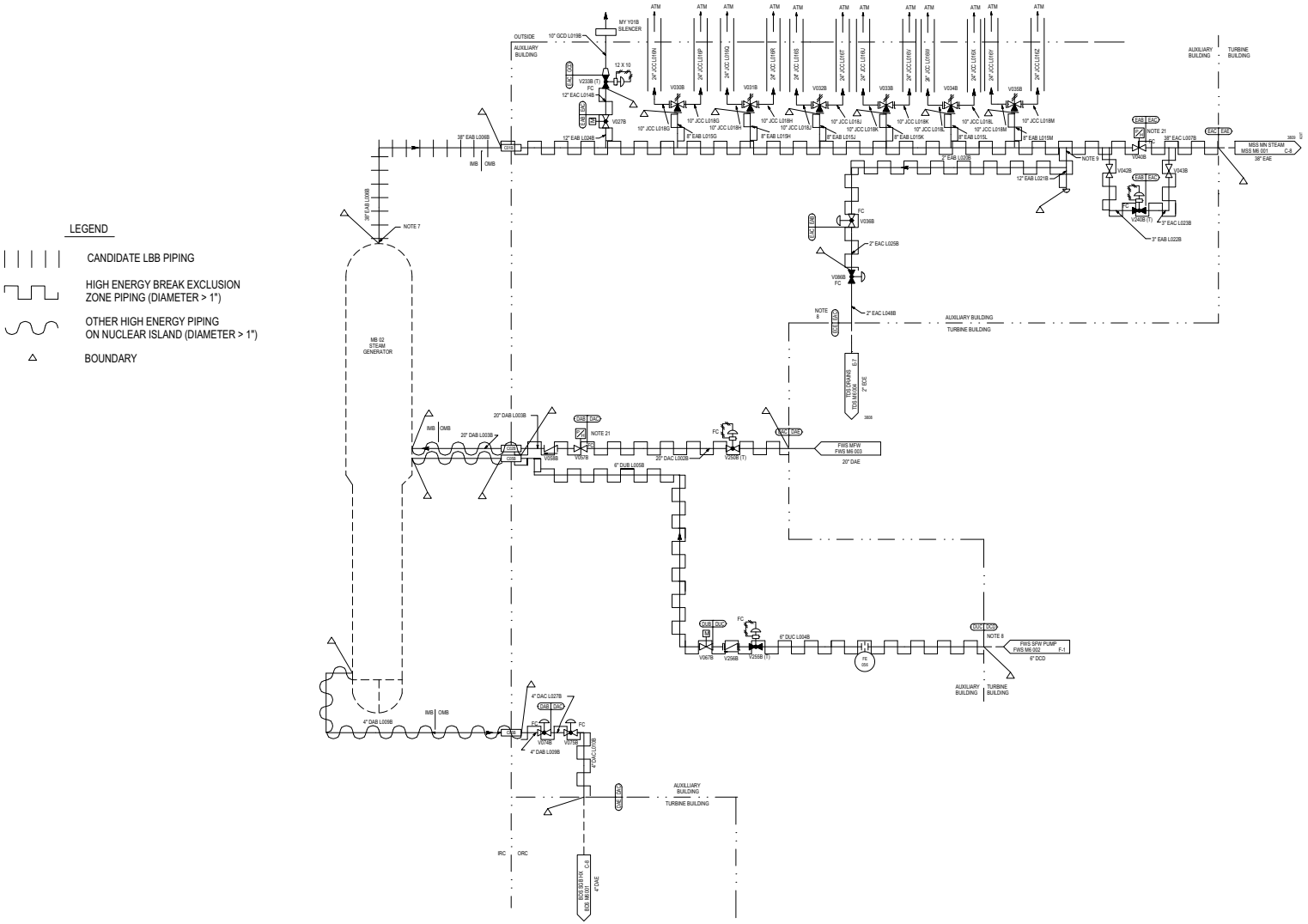

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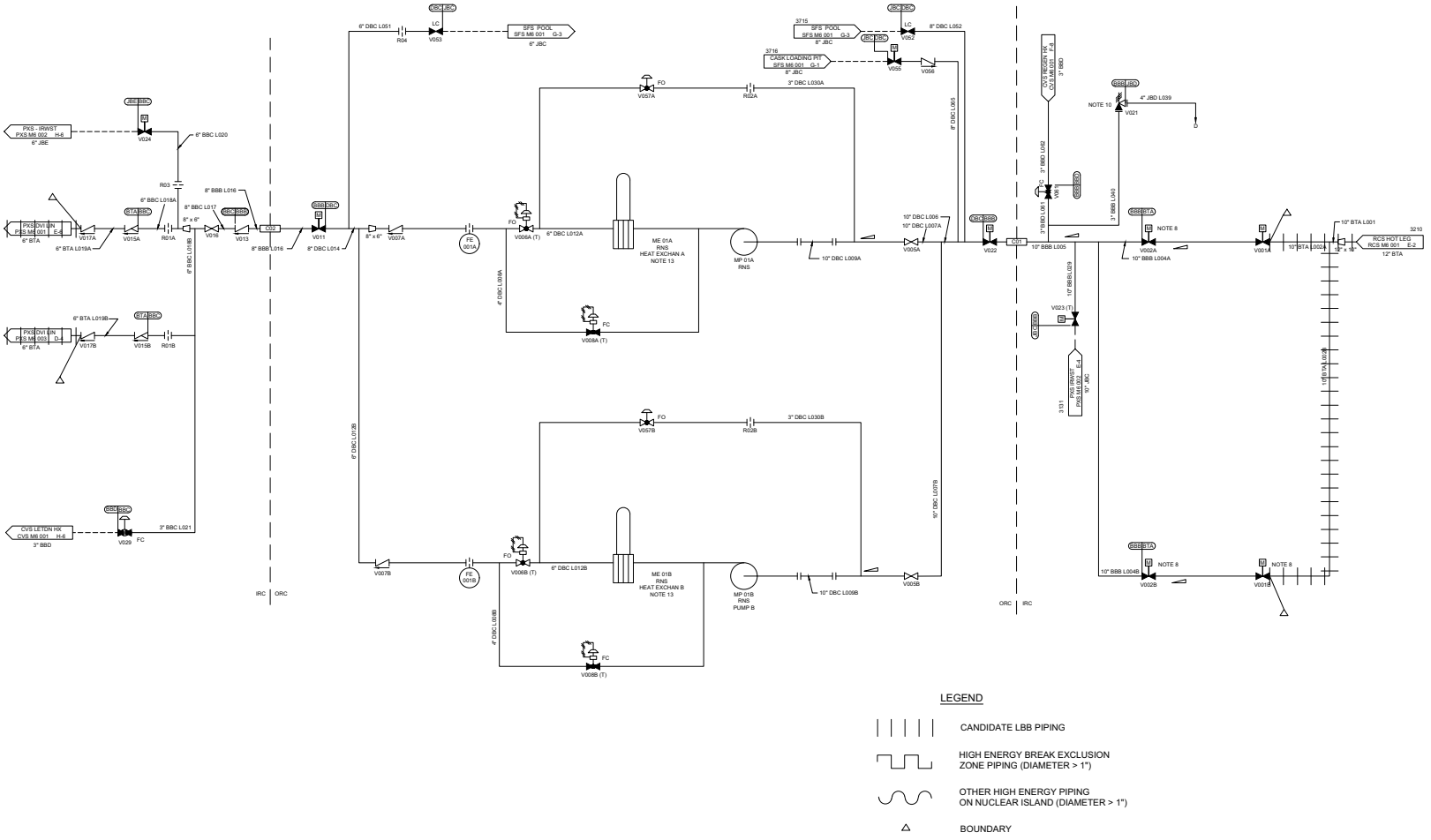

Figure 3E-2
High Energy Piping – Normal Residual Heat Removal System

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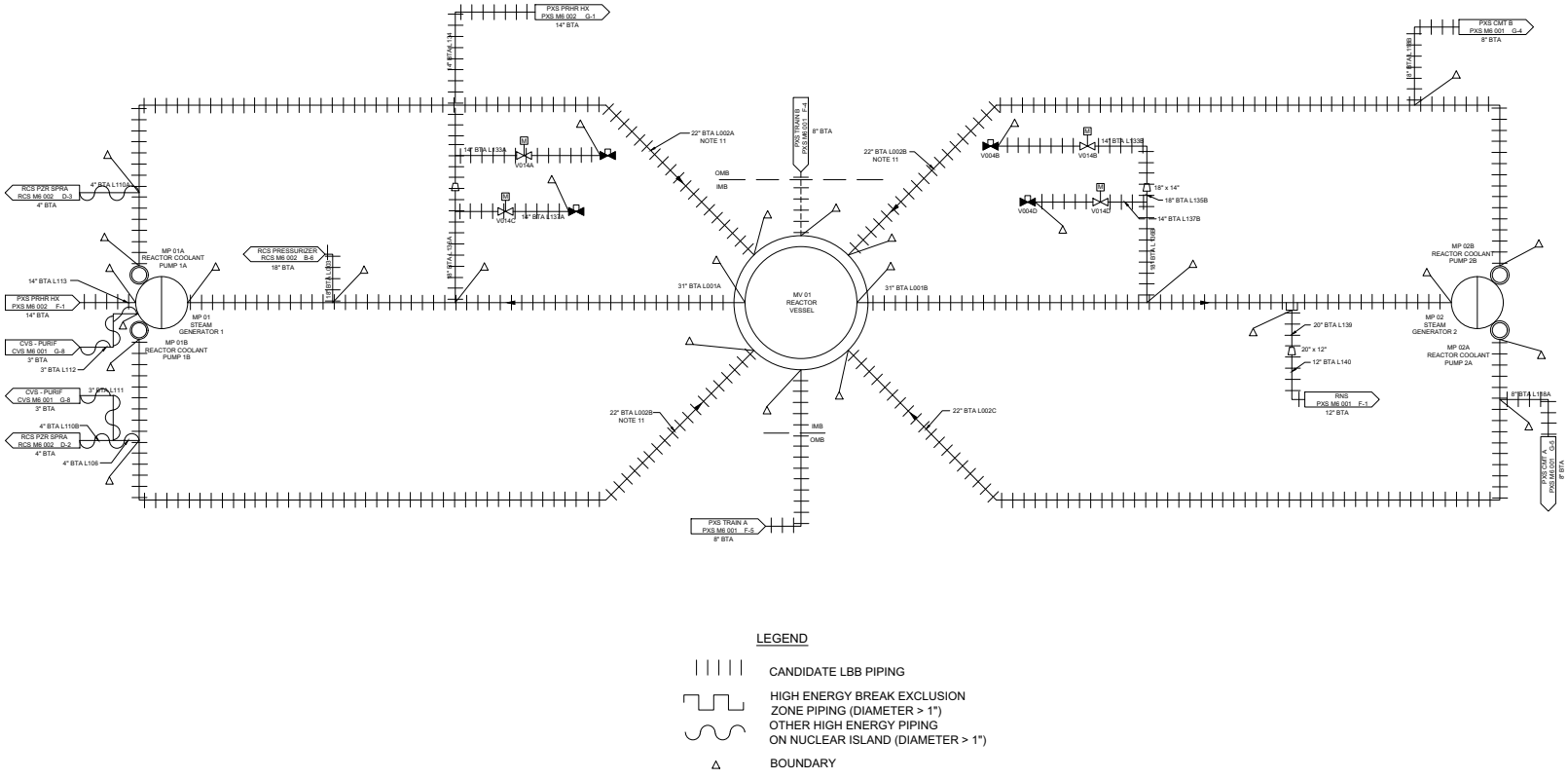

Figure 3E-3 (Sheet 1 of 2)
High Energy Piping – Reactor Coolant System

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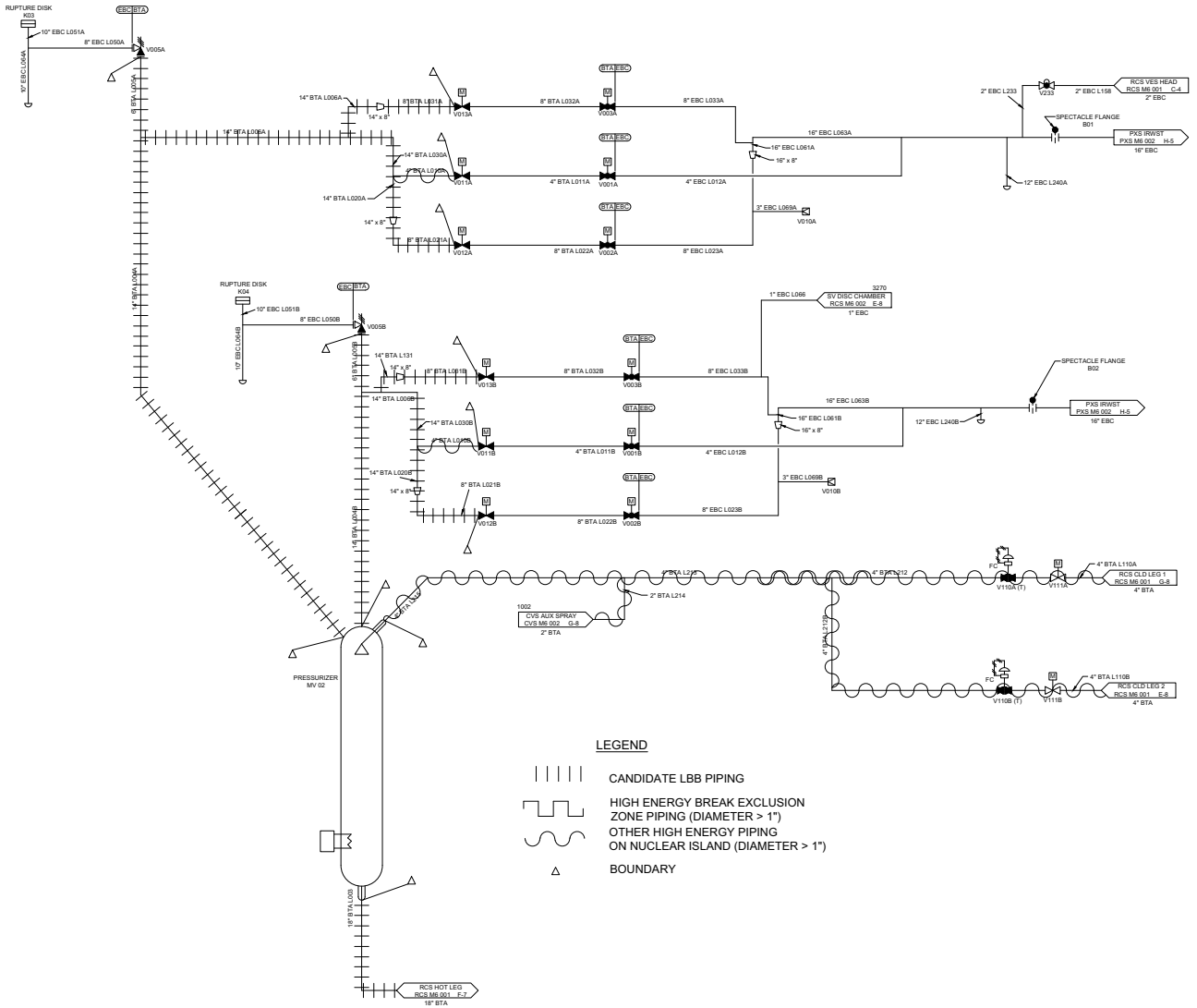

Figure 3E-3 (Sheet 2 of 2)
High Energy Piping – Reactor Coolant System

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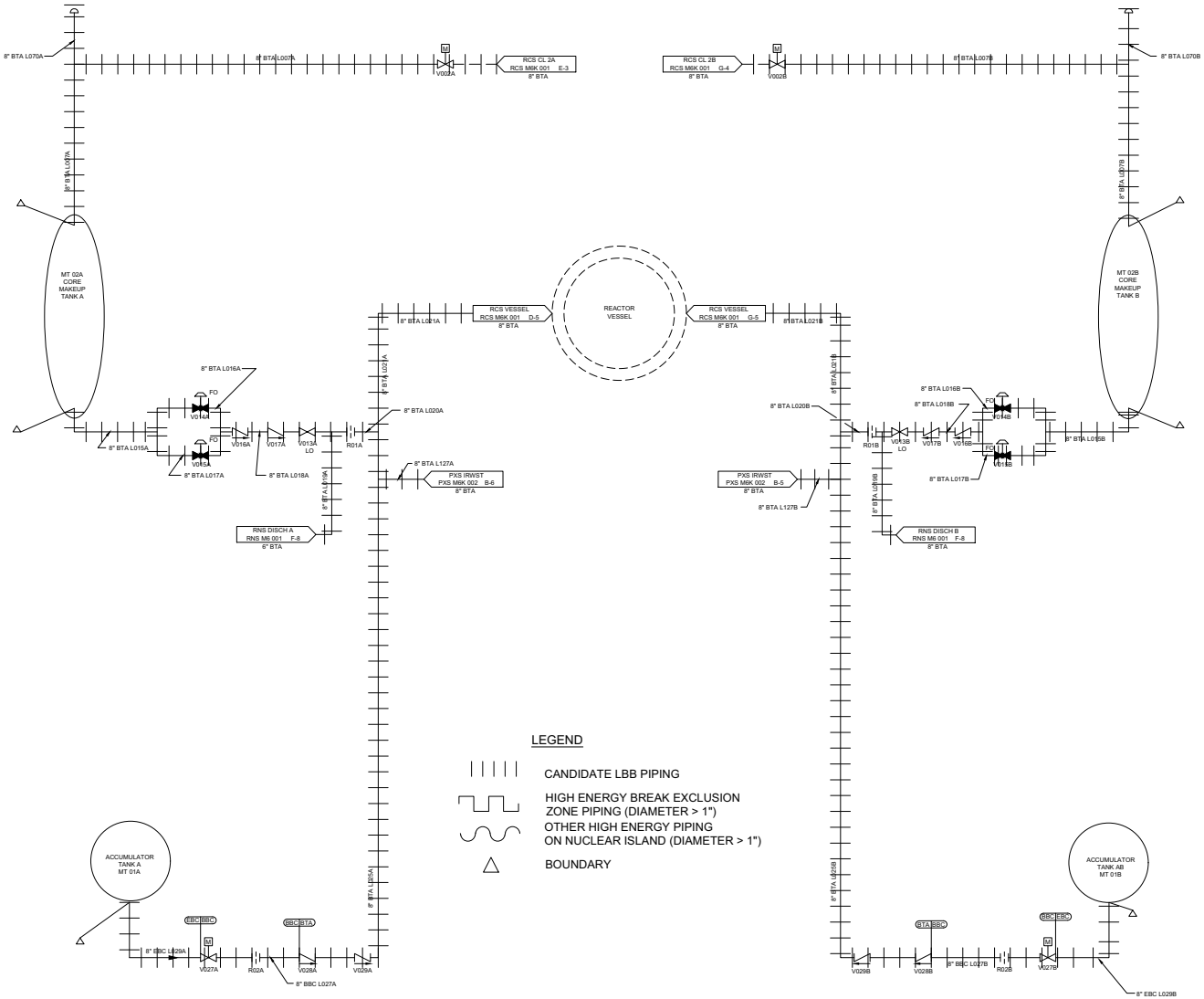

Figure 3E-4 (Sheet 1 of 2)
High Energy Piping – Passive Core Cooling System

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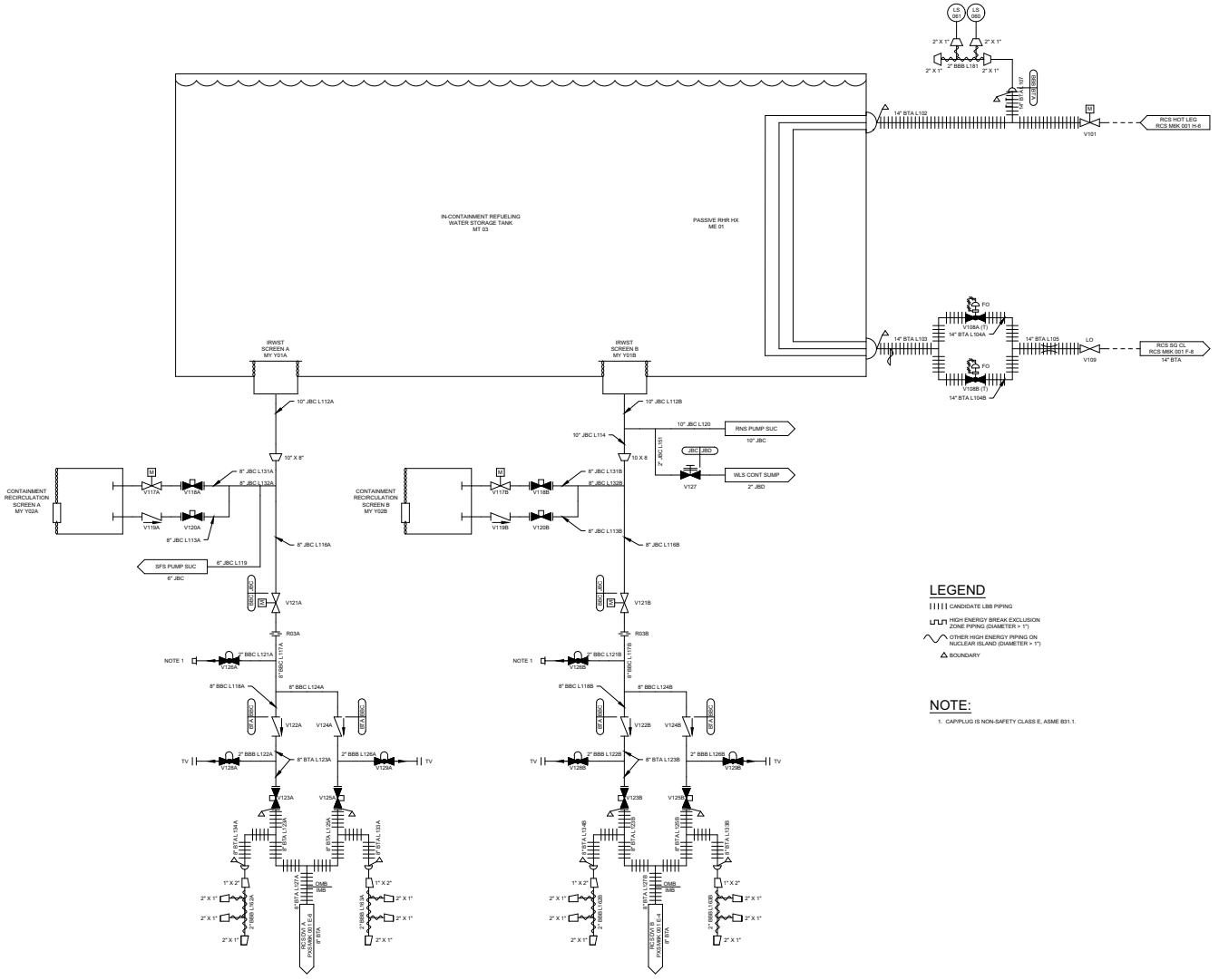

Figure 3E-4 (Sheet 2 of 2)
High Energy Piping – Passive Core Cooling System

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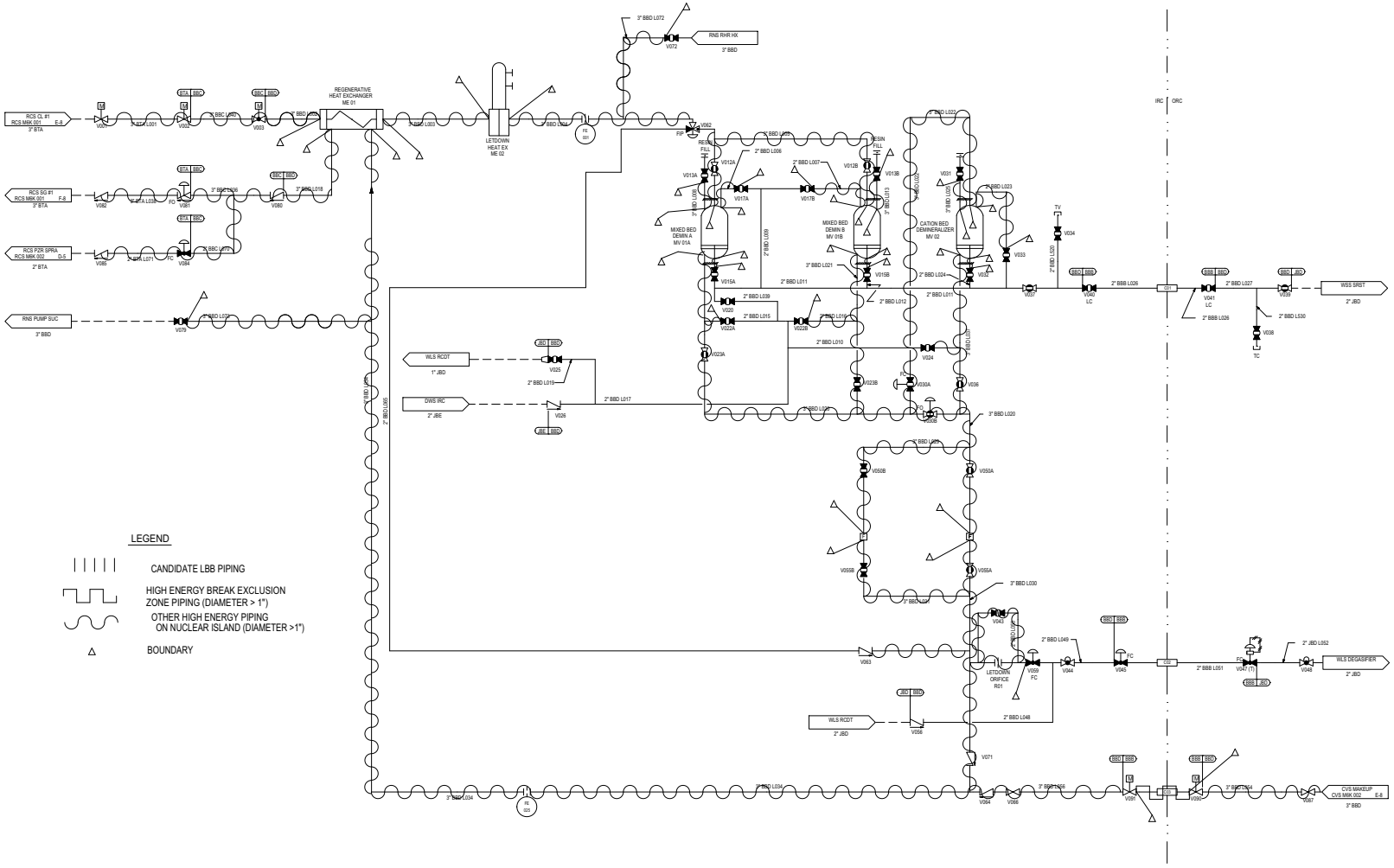

Figure 3E-5 (Sheet 1 of 2)
High Energy Piping – Chemical and Volume Control System

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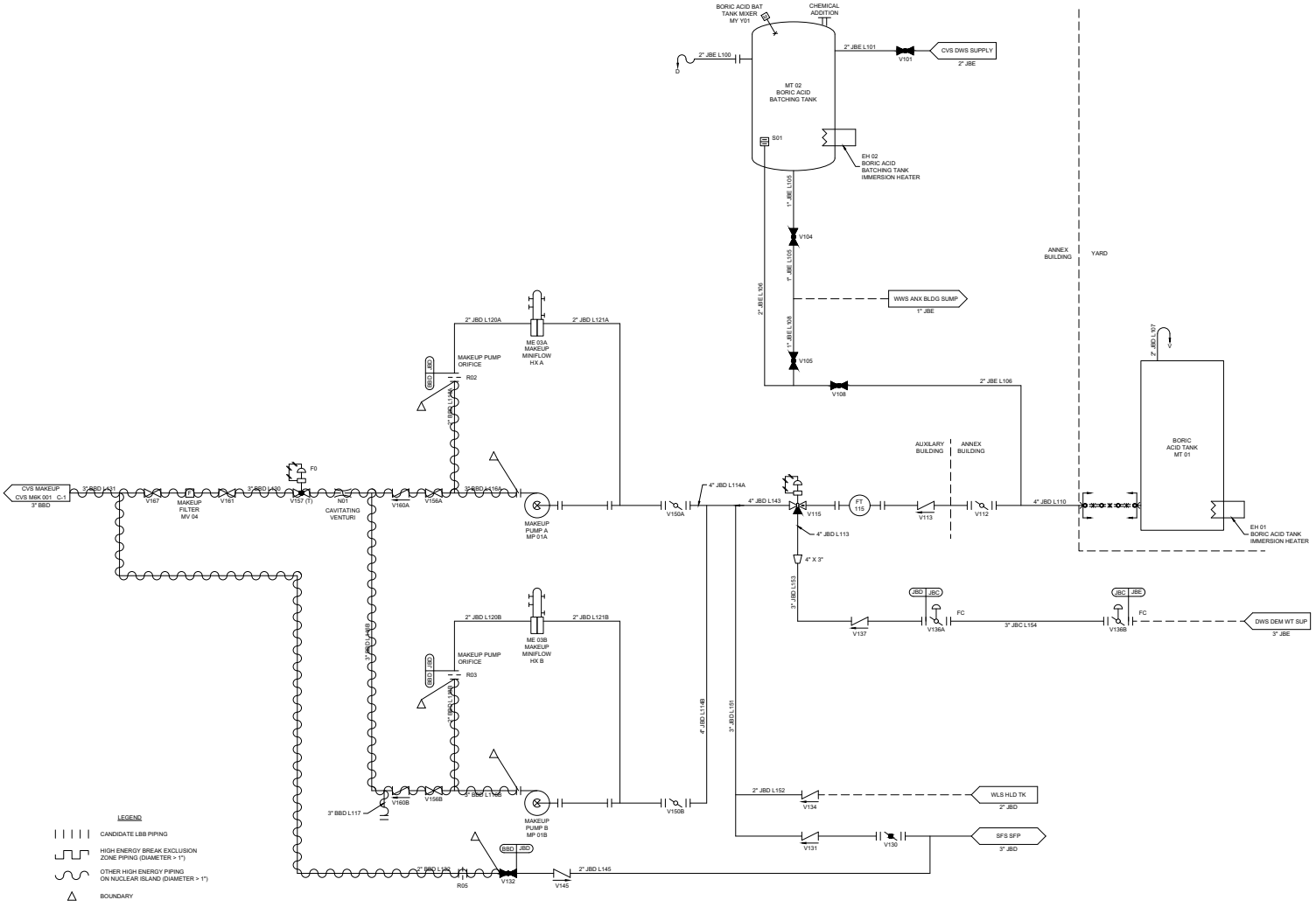

Figure 3E-5 (Sheet 2 of 2)
High Energy Piping – Chemical and Volume Control System

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Appendix 3F Cable Trays and Cable Tray Supports

This appendix provides the design criteria for seismic Category I cable trays and their supports. Seismic Category II cable trays and their supports are also designed utilizing the design criteria of this appendix.

3F.1 Codes and Standards

The design of cable trays and their supports conform to the following codes and standards:

- American Iron and Steel Institute (AISI), Specification for the Design of Cold Formed Steel Structural Members, 1996 Edition and Supplement No. 1, July 30, 1999
- American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, ANSI/AISC N690-1994. American Welding Society (AWS), Structural Welding Code - Steel, AWS D1.1-2000 provides an acceptable alternative for ANSI/AISC N690 weld requirements as described in Subsections 3.8.3.2 and 3.8.4.2.
- Institute of Electrical and Electronic Engineers (IEEE), Standard 344-1987, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
- National Electrical Manufacturers Association (NEMA), Standard Publication No. VE 1-1998, Metallic Cable Tray Systems

3F.2 Loads and Load Combinations

3F.2.1 Loads

3F.2.1.1 Dead Load (D)

Dead load includes the weight of the cable trays, their supports and the cables inside the trays and any permanently attached items. Temporary items used during construction or maintenance are removed prior to operation.

It also includes the weight of

- Cable tray covers and
- Other components and fittings

3F.2.1.2 Construction Live Load (L)

Live load consists of a load of 250 pounds to be applied only during construction on the tray at a critical location to maximize flexural and shear stresses. This load is not combined with seismic loads.

3F.2.1.3 Safe Shutdown Earthquake (E_s)

Seismic response of the cable trays and their supports are produced due to seismic excitation of the supports.

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3F.2.1.4 Thermal Load

These loads are usually not considered and trays are provided with expansion joints in accordance with NEMA.

3F.2.2 Load Combinations

The following load combinations are used for designing the cable trays and their supports:

(a) D + L

(b) D + E_s

3F.3 Analysis and Design

Cable tray systems are designed to maintain structural integrity which assures requirements for physical protection, cable separation, fire protection, and spatial interaction are met. The stresses are maintained within the allowable limits as specified in Subsection 3F.3.3. Section properties and weights of the trays are obtained from manufacturer's data.

3F.3.1 Damping

The maximum damping ratio is 10 percent unless the configuration is demonstrated to be similar to that of the tests described in (Reference 19) of Subsection 3.7.6.

As stated in Subsection 3.7.1.3, the damping ratio used for the AP1000 cable tray systems may be based on test results presented in Reference 19 (Subsection 3.7.6). The cable tray test program conducted by ANCO Engineers Inc. included more than 2000 dynamic tests of representative cable tray system design and construction. The test configurations included items such as various tray types on rigid supports, various tray hanger systems, effects of tray types, effects of strut connections and effects of bracing spacing, unbraced and braced tray systems. Cable ties were also used during the test program. Based on observations during the tests, the high damping values within the cable tray system are provided mainly by the movement, sliding or bouncing of the cables within the tray. The tests show that, for unloaded trays, the damping ratio closely approximates the 7 percent used for bolted structures, and a minimum damping value of 20 percent is maintained with cable ties at spacing greater than or equal to four feet. The tests show that for loaded trays, the damping ratio increases with increased cable loading, reaching a value of 30 percent at cable fill ratio of 50 percent to 100 percent. The major factors which affect the damping ratio of the cable tray systems are the input acceleration level, cable fill ratio, and the ability of the cables to move within the trays during a safe shutdown earthquake.

The AP1000 cable tray system design requires no sprayed-on material for fire protection. Cable ties are provided at spacing greater than 4 feet, thereby permitting cable movement within the trays. The damping ratio used for the cable tray system is dependent on the level of seismic input and the amount of cable fill within the trays. As shown in Figure 3.7.1-13, the 20 percent constant damping ratio may be used for trays loaded to more than 50 percent and subjected to input floor acceleration greater than 0.35g. For cable trays loaded to less than 50 percent and lower than 0.35g input floor acceleration, linearly interpolated lower damping values may be used.

3F.3.2 Seismic Analysis

The methodology for seismic analysis is provided in Subsection 3.7.3. Seismic loads are determined by either using the equivalent static load method of analysis or by performing dynamic analysis.

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Stresses are determined for the seismic excitation in two horizontal and one vertical direction. The stresses in the three directions are combined using the square root of the sum of the squares (SRSS) method or the 100-40-40 method as described in Subsection 3.7.3.6.

3F.3.3 Allowable Stresses

The basic stress allowables for the cable trays are based on the American Iron and Steel Institute specification. The basic stress allowables for cable tray supports utilizing light gage cold rolled channel type sections are based on the manufacturer's published catalog values. The basic stress allowables for cable tray supports utilizing rolled structural shapes are in accordance with ANSI/AISC N690 and the supplemental requirements described in Subsection 3.8.4.5.2.

The allowable stresses for the load combinations are as follows:

D + L	Basic Allowable
D + E_s	1.6 times basic allowable for tension and 1.4 times basic allowable for compression

3F.3.4 Connections

Connections are designed in accordance with the applicable codes and standards listed in Section 3F.1. For connections used with light gage cold rolled channel type sections, design is based on the manufacturer's published catalog values. Supports are attached to the building structure by bolted or welded connections. Fastening of the supports to concrete structures meets the supplemental requirements given in Subsection 3.8.4.5.1.

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Appendix 3G Nuclear Island Seismic Analyses

3G.1 Introduction

This appendix summarizes the seismic analyses of the nuclear island building structures performed to support the AP1000 design certification extension from just hard rock sites, to sites ranging from soft soils to hard rock. The seismic Category I building structures consist of the containment building (the steel containment vessel [SCV] and the containment internal structures [CIS]), the shield building, and the auxiliary building. These structures are founded on a common basemat and are collectively known as the nuclear island or nuclear island structures. Key dimensions of the seismic Category I building structures, such as thickness of the basemat, floor slabs, roofs and walls, are shown in Figures 3.7.1-14 and 3.7.2-12.

Analyses were performed in accordance with the criteria and methods described in Section 3.7. Section 3G.2 describes the development of the finite element models. Section 3G.3 describes the soil structure interaction analyses of a range of site parameters and the selection of the parameters used in the design analyses. Section 3G.4 describes the fixed base and soil structure interaction dynamic analyses and provides typical results from these dynamic analyses. References 3 and 6 provide a summary of dynamic and seismic analysis results (i.e., modal model properties, accelerations, displacements, response spectra) and the nuclear island liftoff analyses. The seismic analyses of the nuclear island are summarized in a seismic analysis summary report. Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Sections 3.7 and 3.8 provided the following acceptance criteria are met:

- The structural design meets the acceptance criteria specified in Section 3.8.
- The seismic floor response spectra (FRS) meet the acceptance criteria specified in Subsection 3.7.5.4.

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report by the Combined License applicant.

Table 3G.1-1 and Figure 3G.1-1 summarize the types of models and analysis methods that are used in the seismic analyses of the nuclear island, as well as the type of results that are obtained and where they are used in the design. Table 3G.1-2 summarizes the dynamic analyses performed and the methods used for combination of modal responses and directional input.

3G.2 Nuclear Island Finite Element Models

The AP1000 nuclear island consists of three distinct seismic Category I structures founded on a common basemat. The three building structures that make up the nuclear island are the coupled auxiliary and shield building (ASB), the SCV, and the CIS. The shield building and the auxiliary building are monolithically constructed with reinforced concrete and, therefore, considered one structure. The nuclear island is embedded approximately 40 feet with the bottom of basemat at elevation 60′-6″ and plant grade located at elevation 100′-0″. The SCV is described in Subsection 3.8.2, the CIS in Subsection 3.8.3, the ASB in Subsection 3.8.4, and the nuclear island basemat in Subsection 3.8.5.

Seismic systems are defined, according to SRP 3.7.2 (Reference 1), Section II.3.a, as the seismic Category I structures that are considered in conjunction with their foundation and supporting media to form a soil-structure interaction model. Fixed base seismic analyses are performed for the nuclear island at a rock site. Soil-structure interaction analyses are performed for soil sites. The analyses generate a set of in-structure responses (design member forces, nodal accelerations, nodal

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displacements, and floor response spectra), which are used in the design and analysis of seismic Category I structures, components, and seismic subsystems. Concrete structures are modeled with linear elastic uncracked properties. However, the modulus of elasticity is reduced to 80% of the ACI code value to reduce stiffness to simulate cracking.

A seismic response spectrum analysis is performed to develop the seismic design loads for the design of the auxiliary building, shield building, and containment internal structure, and the loads generated include the amplified load due to flexibility and the distribution of this load to the surrounding structures. Equivalent static analyses are used to design the shield building roof and radial roof beams, tension ring, air inlet structure, and PCS tank.

3G.2.1 Individual Building and Equipment Models

3G.2.1.1 Coupled Auxiliary and Shield Building

The finite element shell dynamic model of the coupled ASB is a finite element model using primarily shell elements. The portion of the model up to the elevation of the auxiliary building roof is developed using the solid model features of ANSYS, which allow definition of the geometry and structural properties. The nominal element size in the auxiliary building model is about 9 feet so that each wall has two elements for the wall height of about 18 feet between floors. This mesh size, which is the same as that of the solid model, has sufficient refinement for global seismic behavior. It is combined with a finite element model of the shield building roof and cylinder above the elevation of the auxiliary building roof. This model is shown in Figure 3G.2-1. This finite element shell dynamic model is part of the NI10 model.

Since the water in the passive containment cooling system tank responds at a very low frequency (sloshing) and does not affect building response, the passive containment cooling system tank water mass is reduced to exclude the low frequency water sloshing mass. The wall thickness of the bottom portion of the shield building (elevation 63.5′ to 81.5′) is modeled as one half (1.5′) since the CIS model is connected to this portion and extends out to the mid-radius of the shield building cylindrical wall. Local portions of the ASB floors and walls are modeled with sufficient detail to give the response of the flexible areas.

3G.2.1.2 Containment Internal Structures

The finite element shell model of the containment internal structures is a finite element model using primarily shell elements for the walls and floors and solid elements for the mass concrete. It is developed using the solid model features of ANSYS, which allow definition of the geometry and structural properties. This model is used in both static and dynamic analyses. It models the inner and outer mass concrete basemats embedding the lower portion of the containment vessel, and the concrete structures above the mass concrete inside the containment vessel. The walls and basemat inside containment for this model are shown in Figure 3G.2-2. The basemat (dish) outside the containment vessel is shown in Figure 3G.2-3. This finite element shell dynamic model is part of the NI10 model. Static analyses are also performed on the model to obtain member forces in the walls. This model is also used in the 3D finite element basemat model (see Subsection 3.8.5.4.1).

3G.2.1.3 Containment Vessel

The SCV is a freestanding, cylindrical, steel shell structure with ellipsoidal upper and lower steel domes. The finite element model of the containment vessel is an axisymmetric model fixed at elevation 100′. Static analyses are performed with this model to obtain shell stresses as described in Subsection 3.8.2.4.1.1. The model is also used to develop modal properties (frequencies and mode shapes). The three-dimensional, lumped-mass stick model of the SCV is developed based on the

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axisymmetric shell model. Figure 3G.2-4 presents the SCV stick model. In the stick model, the properties are calculated as follows:

- Members representing the cylindrical portion are based on the properties of the actual circular cross section of the containment vessel.
- Members representing the bottom head are based on equivalent stiffnesses calculated from the shell of revolution analyses for static 1.0g in vertical and horizontal directions.
- Shear, bending and torsional properties for members representing the top head are based on the average of the properties at the successive nodes, using the actual circular cross section. These are the properties that affect the horizontal modes. Axial properties, which affect the vertical modes, are based on equivalent stiffnesses calculated from the shell of revolution analyses for static 1.0g in the vertical direction.

The equivalent static acceleration analyses of the containment vessel use a finite element shell model with a refined mesh in the area adjacent to the large penetrations. Comparison of this with a time history analysis for the regions immediately surrounding the large penetrations verifies that the loads from equivalent static analysis are conservative to time history using a representative study.

The stick model is combined with the polar crane stick model as shown in Figure 3G.2-4. Modal properties of the containment vessel with and without the polar crane are shown in Table 3G.2-1. It is connected to nodes on the dish model. NI10 node numbers are shown in red and NI20 node numbers are shown in black.

The method used to construct a stick model from the axisymmetric shell model of the containment vessel is verified by comparison of the natural frequencies determined from the stick model and the shell of revolution model as shown in Table 3G.2-2. The shell of revolution vertical model (n = 0 harmonic) has a series of local shell modes of the top head above elevation 265′ between 23 and 30 hertz. These modes are predominantly in a direction normal to the shell surface and cannot be represented by a stick model. These local modes have small contribution to the total response to a vertical earthquake as they are at a high frequency where seismic excitation is small. The only seismic Category I components attached to this portion of the top head are the water distribution weirs of the passive containment cooling system. These weirs are designed such that their fundamental frequencies are outside the 23 to 30 hertz range of the local shell modes.

An evaluation was made of the connection of the bottom of the steel containment vessel stick model to the CIS finite element model. Comparisons were made between the unconstrained fully symmetric, radially constrained fully symmetric, and original asymmetric connectivity models. The response spectra at the elevation of the polar crane girder for the first two models are almost identical, and the third model had only minor differences. Based on this comparison, the unconstrained fully symmetric connectivity model is used.

3G.2.1.4 Polar Crane

The polar crane is supported on a ring girder, which is an integral part of the SCV at elevation 228′-0″, as shown in Figure 3.8.2-1. It is modeled as a multi-degree of freedom system attached to the steel containment shell at elevation 224′ (midpoint of ring girder) as shown in Figure 3G.2-4. The polar crane is modeled using a simplified and detailed model. The simplified model has five masses at the mid-height of the bridge at elevation 236′-6″ and one mass for the trolley, as shown in Figure 3G.2-5A. The polar crane model includes the flexibility of the crane bridge girders and truck assembly. The containment shell's local flexibility is considered when combined with the global model. When fixed at the center of containment, the model shows fundamental frequencies of 4.6 hertz transverse to the bridge, 6.9 hertz vertically, and 9.4 hertz along the bridge. The Detailed

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Model of the polar crane consists of 76 nodes is defined having 96 dynamic degrees of freedom. It is used to verify the accuracy of the simplified model. This model is shown in Figure 3G.2-5B.

Nodes 425 to 428 are connected to 48 additional nodes. All 52 of these nodes are used to model the truck stiffnesses. The local SCV stiffness is considered when the detailed model of the polar crane is combined with the SCV model.

1. The elements connecting nodes 433, 431/477, 435, 436, 432/478, and 434 represent the trolley.

2. Nodes 437 to 454 are located on the polar crane bridge girders and cross beams. The end nodes (437, 443, 444, and 450) are used to connect the cross beams to the girders; these nodes are also attached to the trucks (nodes 425 to 428) by rigid links.

3. When the Detailed Model of the polar crane is combined with the SCV model, nodes 508 to 511, 528 to 531, 548 to 551, and 568 to 571 are merged to SCV nodes on the polar crane ring girder.

3G.2.1.5 Major Equipment and Structures Using Stick Models

The major equipment supported by the CIS is represented by stick models connected to the CIS. These stick models are the reactor coolant loop model shown in Figure 3G.2-6, the pressurizer model shown in Figure 3G.2-7, and the core makeup tank model shown in Figure 3G.2-8. The core makeup tank model is used only in the nuclear island fine (NI10) model; the core makeup tank is represented by mass in the nuclear island coarse model (NI20).

3G.2.2 Nuclear Island Dynamic Models

Finite element shell models (3D) of the nuclear island concrete structures are used for the time history seismic analyses. Stick models are coupled to the shell models of the concrete structures for the containment vessel, polar crane, the reactor coolant loop and pressurizer. Two models are used. The fine (NI10) model is used to define the seismic response for the hard rock site. The coarse (NI20) model is used for the soil structure interaction (SSI) analyses. It is similar to the NI10 model with the exception that the mesh size for the ASB and CIS is approximately 20 feet instead of 10 feet. This model is set up in both ANSYS and SASSI. The NI05 model is used to develop amplified seismic response for the envelope of soil profiles presented in Subsection 3.7.1.4 for flexible regions not captured by the coarser NI20 model. The NI05 model is also used in response spectrum analysis of the nuclear island to develop design seismic member forces and moments. The NI10, NI20, and NI05 models are described in the subsections below.

3G.2.2.1 NI10 Model

The large solid-shell finite element model of the AP1000 nuclear island shown in Figure 3G.2-9 combines the ASB solid-shell model described in Subsection 3G.2.1.1, and the CIS solid-shell model described in Subsection 3G.2.1.2. The containment vessel and major equipment that are supported by the CIS are represented by stick models and are connected to the CIS. These stick models are the SCV and the polar crane models, the reactor coolant loop model, core makeup tank models, and the pressurizer model. The stick models are described in Subsections 3G.2.1.3 and 3G.2.1.4. The CIS and attached sticks are shown in Figure 3G.2-10. This AP1000 nuclear island model is referred to as the NI10 or fine model. The ASB portion of this model has a mesh size of approximately 10 feet.

The SCV is connected to the CIS model using constraint equations. The SCV at the bottom of the stick at elevation 100′ (node 130401) is connected to CIS nodes at the same elevation. Figure 3G.2-4 shows the SCV stick model with the constraint equation nodes. The nodes are defined using a

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cylindrical coordinate system whose origin coincides with the center of containment (node 130401). The CIS vertical displacement is tied rigidly (constrained) to the vertical displacement and RX and RY rotations of node 130401. The CIS tangential displacement is tied rigidly (constrained) to the horizontal displacement and RZ rotation of node 130401.

3G.2.2.2 NI20 Model

The NI20 coarse model has fewer nodes and elements than the NI10 model. It captures the essential features of the nuclear island configuration. The nominal shell and solid element dimension is about 20 feet. It is used in the soil-structure interaction analyses of the nuclear island performed using the program SASSI. The stick models are the same as used for the NI10 model except that the core makeup tank is not included. This model is shown in Figures 3G.2-11 and 3G.2-12. Results of fixed base analyses of the NI20 model were compared to those of the NI10 model to confirm the adequacy of the NI20 model for use in the soil-structure-interaction analyses.

3G.2.2.3 Nuclear Island Stick Model

The nuclear island lumped-mass stick model consists of the stick models of the individual buildings interconnected by rigid links. Each individual stick model is developed to match the modal properties of the finite element models described in Subsections 3G.2.1.1 and 3G.2.1.2 above. Modal analyses and seismic time history analyses were performed using this model for the hard rock design certification.

The nuclear island lumped-mass stick model has been replaced in the design analyses described in this appendix by the NI10 and NI20 finite element shell dynamic models of the nuclear island described in Subsections 3G.2.2.1 and 3G.2.2.2 above. A 2D stick model is used in the soil sensitivity analyses described in Section 3G.3.

3G.2.2.4 NI05 Model

The NI05 solid-shell finite element model of the AP1000 nuclear island is shown in Figures 3G.2-13 to 3G.2-15. The NI05 model is used for response spectrum analysis of the nuclear island auxiliary and shield building structures. The NI05 model is also used for the mode superposition time history analysis of the nuclear island for the amplified response at flexible floors. The NI05 model is used for the static analysis of the nuclear island for the basemat design. The NI05 model is a refined version of the NI10 model where the auxiliary and shield building mesh size is reduced from approximately 10 feet by 10 feet tetrahedral mesh to approximately 5 feet by 5 feet. The major equipment stick models supported by the CIS are the same as used for the NI10 model. The steel containment vessel stick model and connections are also the same as the NI10 model. The only difference between the NI05 CIS and NI10 CIS is the basemat (bowl) and dish region as shown in Figure 3G.2-15. The model is validated by a comparison of the mass participation by frequency of the fundamental modes to those of the NI10 model.

3G.2.2.5 Seismic Stability Model

The sliding stability of the nuclear island basemat is evaluated using a non-linear 2D East-West (EW) stick model of the nuclear island structures using the ANSYS program. Three concentric sticks represent ASB, CIS, and SCV, respectively. The reactor coolant loop is included as mass only. The basemat is modeled as a rigid beam, which is free in translation along the EW and vertical directions. The nuclear island combined sticks are attached to the rigid basemat at the nuclear island mass center.

Each node of the rigid basemat is connected with two spring elements in the horizontal and vertical directions, respectively. The spring elements only model the foundation media (rock or soil) damping,

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not stiffness. A layer of contact elements is added along the rigid basemat bottom to simulate the friction forces between basemat bottom and foundation media as well as foundation media stiffnesses. The friction coefficient between the basemat bottom and the soil media is set at 0.55. Figure 3G.2-19 shows the schematic of this non-linear 2D EW nuclear island stick model. The contact elements are free to uplift when the upward force (normal force) is larger than the associated dead load component. When the tangential force is larger than the friction force, sliding occurs.

3G.2.3 Static Models

Member forces in the ASB are obtained from analyses of a model that is more refined than the finite element model described in Subsection 3G.2.1.1. This model is developed by meshing one area of the solid model with four finite elements. The nominal element size in this auxiliary building model is about 4.5 feet so that each wall has four elements for the wall height of about 18 feet between floors. This finite element shell model is referred to as the NI05 model. This refinement is used to calculate the design member forces and moments using response spectra analysis of the nuclear island models with seismic input enveloping all soil conditions. The finite element shell model of the containment internal structures described in Subsection 3G.2.1.2, which includes the basemat within the shield building and the containment vessel stick model, is also included.

Finite element solid/shell models were used for the equivalent static seismic analysis. For the detailed design of the shield building roof, a finite element model of one quadrant of the roof is used as described in Subsection 3G.2.3.1. For the detailed design of the steel containment vessel, a shell mesh finite element model with a much finer mesh in the areas surrounding the major penetrations is used as described in Subsection 3G.2.3.2. For the static analysis of the containment vessel, an axisymmetric model is used as described in Subsection 3G.2.3.3. The nuclear island basemat is evaluated using the NI05 finite element model described in Subsection 3G.2.2.4.

3G.2.3.1 Quadrant Model of Shield Building Roof

The one quadrant model of the shield building roof is shown in Figure 3G.2-16. The model is constructed with solid and shell elements and contains structures from the exposed shield wall through the top of the shield building roof. The quadrant model is used for the equivalent static analysis of the shield building roof. The results from the more detailed analysis are used in the design of the shield building roof and radial roof beams, tension ring, air inlet structure, and PCS tank.

3G.2.3.2 Containment Vessel 3D Finite Element Model

The 3D finite element model of the steel containment vessel is shown in Figure 3G.2-17. The finite element model for the steel containment vessel is used for the stress analysis of the large penetrations (personnel locks and equipment hatches) of the containment vessel.

3G.2.3.3 Containment Vessel Axisymmetric Model

The axisymmetric finite element model of the steel containment vessel is shown in Figure 3G.2-18. The axisymmetric model is a two-dimensional model with added mass for the stiffeners, crane girder, equipment hatches, and air locks.

3G.3 2D SASSI Analyses

This section describes the soil structure interaction analyses performed using 2D models in SASSI to select the design soil cases for the AP1000. The AP1000 footprint, or interface to the soil medium, is identical to the AP600. The AP1000 containment and shield building are 25′ 6″ and 20′ 6″ (Reference 4) respectively taller than AP600. Results and conclusions from the AP600 soil studies (Reference 2) are considered in establishing the design soil profiles for the AP1000.

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Analyses were performed using 2D stick models of the AP1000 for horizontal seismic input with and without adjacent structures. The soil profiles included a hard rock site, a firm rock site, a soft rock site, a soft-to-medium soil site, an upper bound soft-to-medium site, and a soft soil site. Analyses were also performed without adjacent structures for a hard rock site, a firm rock site, a soft rock site, a soft-to-medium soil site, an upper bound soft-to-medium site, and a soft soil site. The soil damping and degradation curves are described in Subsection 3.7.1.4. The soil profiles selected for the AP1000 use the same parameters on depth to bedrock, depth to water table, and variation of shear wave velocity with depth as those used in the AP600 design analyses. The Poisson's ratio is 0.25 for rock sites (hard and firm rock) and 0.35 for soil sites (soft-to-medium soil, and upper bound soft-to-medium soil). For all the soil profiles defined, the base rock has been taken to be at 120 feet below grade level. The soil profiles are shown in Figure 3G.3-1. The shear wave velocity profiles and related governing parameters are as follows:

- For the hard rock site, an upper bound case for rock sites using a shear wave velocity of 8000 feet per second.

- For the firm rock site, a shear wave velocity of 3500 feet per second to a depth of 120 feet, and base rock at the depth of 120 feet.

- For the soft rock site, a shear wave velocity of 2400 feet per second at the ground surface, increasing linearly to 3200 feet per second at a depth of 240 feet, and base rock at the depth of 120 feet.

- For the upper bound soft-to-medium soil site, a shear wave velocity of 1414 feet per second at ground surface, increasing parabolically to 3394 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water at grade level. The initial soil shear modulus profile is twice that of the soft-to-medium soil site.

- For the soft-to-medium soil site, a shear wave velocity of 1000 feet per second at ground surface, increasing parabolically to 2400 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water is assumed at grade level.

- For the soft soil site, a shear wave velocity of 1000 feet per second at ground surface, increasing linearly to 1200 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water is assumed at grade level.

The analyses with and without adjacent structures demonstrated that the effect of adjacent buildings on the nuclear island response is small. Based on this, the 3D SASSI analyses of the AP1000 nuclear island can be performed without adjacent buildings similar to those performed for the AP600.

The maximum acceleration values obtained from the AP1000 analyses without adjacent structures are given in Table 3G.3-1. The soil cases giving the maximum response are shown in bold. Floor response spectra associated with nodes 41, 120, 310, 411, and 535 for the six AP1000 soil cases are shown in Figures 3G.3-2 to 3G.3-11.

Based on review of the above results, five soil conditions were selected for 3D SASSI analyses in addition to the hard rock condition evaluated in the existing AP1000 Design Certification. Thus, the following five soil and rock cases identified in Subsection 3.7.1.4 are considered: hard rock, firm rock, soft rock, soft-to-medium soil, upper bound soft-to-medium, and soft soil.

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3G.4 Nuclear Island Dynamic Analyses

3G.4.1 ANSYS Fixed Base Analysis

The NI10 model described in Subsection 3G.2.2.1 was analyzed by time history modal superposition. To perform the time history analysis of this large model, the ANSYS superelement (substructuring) techniques were applied. Substructuring is a procedure that condenses a group of finite elements into one element represented as a matrix. The reasons for substructuring are to reduce computer time of subsequent evaluations. Two sets of analyses were performed. To obtain the time history response of the ASB, the ASB finite element model was merged with the superelement of the CIS and its major components. To obtain the time history response of the CIS, the CIS finite element model was merged with the superelement of the ASB.

Deflection time history responses were obtained at selected representative locations. These locations included major wall and floor intersections and nodes at the cardinal orientations at key elevations of the shield building. Nodes were also selected at mid-span on flexible walls and floors. Typical locations are shown for the ASB at elevation 135′ on Figures 3G.4-1 and 3G.4-2. Figure 3G.4-1 shows the “rigid” locations, and Figure 3G.4-2 shows the “flexible” locations.

ANSYS is used to calculate the maximum relative deflection to the nuclear island for the envelope case that considers all of the soil and hard rock site cases. Synthesized displacement time histories are developed using the envelope seismic response spectra from the six site conditions (hard rock, firm rock, soft rock, upper-bound-soft-to-medium, soft-to-medium, and soft soil). Seismic response spectra at nine locations are used (four edge locations, one center location, and four corner locations). It is not necessary to adjust for drift since relative deflections to the basemat are calculated and the drift would be subtracted from the results.

3G.4.2 3D SASSI Analyses

The computer program SASSI 2000 is used to perform Soil-Structure Interaction analysis with the NI20 Coarse Finite Element Model. The SASSI Soil-Structure Interaction analyses are performed for the five soil conditions established from the AP1000 2D SASSI analyses. These soil conditions are firm rock, soft rock, soft-to-medium soil, upper bound soft-to-medium, and soft soil. The model includes a surrounding layer of excavated soil and the existing soil media as shown in Figures 3G.4-3 and 3G.4-4. Acceleration time histories and floor response spectra are obtained. Adjacent structures have a negligible effect on the nuclear island structures and, thus, are not considered in the 3D SASSI analyses.

Westinghouse has adopted the approach that calculates displacements internally within the ACS SASSI program based on an analytical complex frequency domain approach that uses inverse Fast-Fourier Transforms (FFT) to compute relative displacement histories instead of double numerical integration in the time domain that computes absolute displacement time histories from absolute acceleration time histories.

The relative displacement time history is calculated using ACS SASSI RELDISP module. The complex acceleration transfer functions (TF) are computed for reference and all selected output nodes. The relative acceleration transfer function is calculated by subtracting the reference node TF from the output node TF. The relative displacement transfer function is obtained by dividing the circular frequency square (ω^2) for each frequency data point. The relative displacement time history is obtained by taking the inverse FFT.

Relative displacements are calculated between adjacent buildings and the nuclear island using soft springs between the buildings. The spring stiffness is very small so that it does not affect the dynamic

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response. These calculations are performed using 2D models and SASSI 2000. The relative deflection is calculated using the maximum compressive spring force and the stiffness value.

In these analyses, the three components of ground motions (N-S, E-W, and vertical direction) are input separately. Each design acceleration time history (N-S, E-W, and vertical) is applied separately, and the time history responses are calculated at the required nodes. The resulting co-linear time history responses at a node due to the three earthquake components are then combined algebraically.

3G.4.3 Seismic Analysis

3G.4.3.1 Response Spectrum Analysis

The response spectrum methodology used in the AP1000 design employs the Complete Quadratic Combination (CQC, Section 1.1 of Reference 5) grouping method for closely spaced modes with the Der Kiureghian Correlation Coefficient (Section 1.1.3 of Reference 5) used for correlation between modes. The Lindley-Yow (Section 1.3.2, Reference 5) spectra analysis methodology is employed for modes with both periodic and rigid response components. The modal analysis performed to develop composite modal participation is used to develop input for the response spectrum analysis. Modes ranging from 0 to 33 Hz or higher are considered. For modes above the cutoff frequency, the Lindley-Yow is used. The Static ZPA Method (Section 1.4.2, Reference 5) is employed for the residual rigid response component for each mode as outlined in NRC Regulatory Guide 1.92 (Reference 5). The complete solution is developed via Combination Method B (Section 1.5.2, Reference 5). The combined effects, considering three spatial components of an earthquake (N-S, E-W, and Vertical), are combined by square root sum of the squares method (Section 2.1, Reference 5).

In Subsection 3.7.2.6, “Three Components of Earthquake Motion,” the combination of three components of earthquake motion is discussed.

3G.4.3.2 Absolute Accelerations

The seismic analyses results, which include the new shield building configuration described in Section 3.8, are given in Reference 3.

3G.4.3.3 Seismic Response Spectra

The AP1000 plant floor response spectra for the six key locations are provided in Figure 3G.4-5X to 3G.4-10Z. The key locations are defined in Table 3G.4-1. The design seismic response spectra are conservatively adjusted in the low frequency range in anticipation of future sites having a slightly higher response at the lower frequency.

The in-structure response spectra at six key locations, as defined below, are used if a site-specific 3D dynamic analysis evaluation as outlined in Subsection 2.5.2 is required. The site is acceptable if the floor response spectra from the site-specific evaluation do not exceed the AP1000 spectra for each of the locations identified below or the exceedances are justified.

[***FRS Location***	***Figure Numbers***
Containment internal structures at elevation of reactor vessel support	*Figure 3G.4-5X to 3G.4-5Z*
Containment operating floor	*Figure 3G.4-6X to 3G.4-6Z*
Auxiliary building NE corner at elevation 116'-6"	*Figure 3G.4-7X to 3G.4-7Z*
Shield building at fuel building roof	*Figure 3G.4-8X to 3G.4-8Z*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Shield building roof	*Figure 3G.4-9X to 3G.4-9Z*
Steel containment vessel at polar crane support	*Figure 3G.4-10X to 3G.4-10Z*]*

Note:
See Table 3G.4-1 for locations of six key locations.

3G.4.3.4 Bearing Pressure Demand

Bearing pressure demand was calculated using both 2D and 3D analyses. Both linear and non-linear analyses are performed with the 2D nuclear island model. The maximum bearing pressures calculated include the effect of dead, live, and seismic loading.

The 2D model was used to evaluate the effect of liftoff on the bearing pressure. Since the largest bearing pressure will result from the east-west seismic excitation because of the smaller width of the basemat in this direction, liftoff was evaluated using an east-west stick model of the nuclear island structures, supported on a rigid basemat with non-linear springs. Direct integration time history analyses were performed. The bearing pressures calculated from these analyses are summarized in Table 3G.4-2. The pressures are at the edge of the basemat. Results are given for the three cases that result in the highest bearing pressure (hard rock [HR], upper bound soft to medium [UBSM] soil, and soft to medium [SM] soil). The linear results show maximum bearing pressures on the west side of 31 to 33 ksf. Liftoff increases the subgrade pressure close to the west edge by 4 percent to 6 percent with insignificant effect beneath most of the basemat.

The SASSI soil-structure interaction analyses are performed based on the nuclear island 3D SASSI model for the hard rock and five soil conditions established from the AP1000 2D SASSI analyses. The SASSI model of the nuclear island is based on the NI20 finite element model. The bearing pressures from the 3D SASSI analyses have been obtained by combining the time history results from the north-south, east-west, and vertical earthquakes. The maximum soil-bearing pressure demand is obtained from the hard rock (HR) case equal to 35 ksf. It is noted that a maximum localized peak is obtained on the west edge of 38 ksf; a limit of 35 ksf for maximum bearing seismic demand is obtained by averaging the soil pressure over 335 ft^2 of the west edge of the shield building where the maximum stress occurs.

3G.5 References

1. NUREG-800, Review of Safety Analysis Reports for Nuclear Power Plants, Section 3.7.2, Seismic System Analysis, Revision 2.

2. GW-GL-700, AP600 Design Control Document, Appendices 2A and 2B, Revision 4.

3. APP-GW-S2R-010, “Extension of Nuclear Island Seismic Analyses to Soil Sites,” Westinghouse Electric Company LLC.

4. APP-GW-GLN-112, “Structural Verification for Enhanced Shield Building,” Westinghouse Electric Company LLC.

5. U.S. NRC Regulatory 1.92, Revision 2, “Combining Modal Responses and Spatial Components in Seismic Analysis.”

6. APP-GW-GLR-044, “Nuclear Island Basemat and Foundation,” Westinghouse Electric Company LLC.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3G.1-1 (Sheet 1 of 4)
Summary of Models and Analysis Methods

Model	Analysis Method	Program	Type of Dynamic Response/Purpose
3D (ASB) solid-shell model	-	ANSYS	Creates the finite element mesh for the ASB finite element model.
3D (CIS) solid-shell model	-	ANSYS	Creates the finite element mesh for the CIS finite element model.
3D finite element model including shield building roof (ASB10)	-	ANSYS	ASB portion of NI10.
3D finite element model including dish below containment vessel	Response spectrum analysis	ANSYS	CIS portion of NI10.
3D finite element shell model of nuclear island [NI10] (coupled auxiliary and shield building shell model, containment internal structures, steel containment vessel, polar crane, RCL, pressurizer, and CMTs)	Mode superposition time history analysis	ANSYS	Performed for hard rock profile for ASB with CIS as superelement and for CIS with ASB as superelement. To develop time histories for generating plant design floor response spectra for nuclear island structures. To obtain maximum absolute nodal accelerations (ZPA) to be used in equivalent static analyses. To obtain maximum displacements relative to basemat.
3D finite element coarse shell model of auxiliary and shield building and containment internal structures [NI20] (including steel containment vessel, polar crane, RCL, and pressurizer)	Mode superposition time history analysis	ANSYS	Performed for hard rock profile for comparisons against more detailed NI10 model.

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Table 3G.1-1 (Sheet 2 of 4)
Summary of Models and Analysis Methods

Model	Analysis Method	Program	Type of Dynamic Response/Purpose
Finite element lumped-mass stick model of nuclear island	Time history analysis	SASSI	Performed 2D parametric soil studies to help establish the bounding generic soil conditions and to develop adjustment factors to reflect all generic site conditions for seismic stability evaluation.
Finite element lumped-mass stick model of nuclear island	Direct integration time history analysis	ANSYS	Performed 2D linear and non-linear seismic analyses to evaluate effect of liftoff on Floor Response Spectra and bearing.
3D finite element coarse shell model of auxiliary and shield building and containment internal structures [NI20] (including steel containment vessel, polar crane, RCL, and pressurizer)	Time history analysis Complex frequency response analysis	SASSI	Performed for the five soil profiles of firm rock, soft rock, upper bound soft-to-medium soil, soft-to-medium soil, and soft soil. To develop time histories for generating plant design floor response spectra for nuclear island structures. To obtain maximum absolute nodal accelerations (ZPA) to be used in equivalent static analyses. To obtain maximum displacements relative to basemat. To obtain SSE bearing pressures for all generic soil cases. To obtain maximum member forces and moments in selected elements for comparison to equivalent static results.
3D shell model of auxiliary and shield building and containment internal structures [NI20] (including steel containment vessel)	Mode superposition time history analysis	ANSYS	Performed to develop loads for seismic stability evaluation.

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Table 3G.1-1 (Sheet 3 of 4)
Summary of Models and Analysis Methods

Model	Analysis Method	Program	Type of Dynamic Response/Purpose
3D shell of revolution model of steel containment vessel	Modal analysis; equivalent static analysis using accelerations from time history analyses	ANSYS	To obtain dynamic properties. To obtain SSE stresses for the containment vessel.
3D lumped-mass stick model of the SCV	-	ANSYS	Used in the NI10 and NI20 models.
3D lumped-mass stick model of the RCL	-	ANSYS	Used in the NI10 and NI20 models.
3D lumped-mass stick model of the pressurizer	-	ANSYS	Used in the NI10 and NI20 models.
3D lumped-mass stick model of the CMT	-	ANSYS	Used in the NI10 model.
3D lumped mass detailed model of the polar crane	Modal analysis	ANSYS	To obtain dynamic properties. Used with 3D finite element shell model of the containment vessel.
3D lumped mass simplified (single beam) model of the polar crane	-	ANSYS	Used in the NI10 and NI20 models.
3D finite element shell model of containment vessel	Mode superposition time history analysis Equivalent static analysis	ANSYS	Used with detailed polar crane model to obtain acceleration response of equipment hatch and airlocks. To obtain shell stresses in vicinity of the large penetrations of the containment vessel.

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Table 3G.1-1 (Sheet 4 of 4)
Summary of Models and Analysis Methods

Model	Analysis Method	Program	Type of Dynamic Response/Purpose
3D finite element refined shell model of nuclear island (NI05)	Equivalent static non-linear analysis using accelerations from time history analyses Mode superposition time history analysis for the wall and floor flexibility using synthetic time histories developed to match spectral envelopes applied at the base Response spectrum analysis with seismic input enveloping all soils cases	ANSYS	To obtain SSE member forces for the nuclear island basemat. To obtain floor and wall flexibility response characteristics. To obtain maximum displacements relative to basemat. To obtain SSE member forces for the auxiliary and shield building and the containment internal structures.
3D finite element coarse shell model of auxiliary and shield building and containment internal structures [NI20] (including steel containment vessel, polar crane, RCL, and pressurizer)	Mode superposition time history analysis with seismic input enveloping all soil cases	ANSYS	To obtain total basemat reactions for comparison to reactions in equivalent static linear analyses using NI05 model.
Quadrant model of shield building roof (See subsection 3.8.4.4.1 for information on use of the quadrant model.)	Equivalent static analysis The PCS tank is designed using the maximum accelerations at the applicable elevation resulting from time history dynamic analyses of the nuclear island. The tension ring and air inlet use maximum accelerations that are increased based on results of response spectrum analysis.	ANSYS	To obtain member forces for shield building roof and radial roof beams, air inlet structure, tension ring, and PCS tank.

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Table 3G.1-2
Summary of Dynamic Analyses and Combination Techniques

Model	Analysis Method	Program	Three Components Combination	Modal Combination
3D finite element, fixed base models, coupled auxiliary and shield building shell model, with superelement of containment internal structures (NI10 and NI20)	Mode superposition time history analysis	ANSYS	Algebraic Sum	n/a
3D finite element nuclear island model (NI20)	Complex frequency response analysis	SASSI	Algebraic Sum	n/a
3D finite element, fixed base models, coupled auxiliary and shield building and containment internal structures including shield building roof (NI05)	Response spectrum analysis	ANSYS	SRSS or 100%, 40%, 40%	Lindley-Yow
3D finite element model of the nuclear island basemat (NI05)	Equivalent static analysis using nodal accelerations from shell model	ANSYS	100%, 40%, 40%	n/a
3D shell of revolution model of steel containment vessel	Equivalent static analysis using nodal accelerations from 3D stick model	ANSYS	SRSS or 100%, 40%, 40%	n/a
PCS valve room and miscellaneous steel frame structures, miscellaneous flexible walls, and floors	Response spectrum analysis	ANSYS	SRSS or 100%, 40%, 40%	Grouping or Lindley-Yow
2D stick model analyses with liftoff	Direct integration time history	ANSYS	Algebraic Sum	n/a

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Table 3G.2-1 (Sheet 1 of 2)
Steel Containment
Vessel Lumped-Mass Stick Model (Without Polar Crane) Modal Properties

Mode	Frequency	Effective Mass		
		X Direction	Y Direction	Z Direction
1	6.28	3.51	159.81	0.01
2	6.28	160.01	3.51	0.00
3	12.87	0.03	0.00	0.00
4	16.89	0.00	0.01	173.42
5	18.87	0.29	40.91	0.00
6	18.89	40.77	0.29	0.00
7	28.13	0.00	0.00	28.45
8	31.67	0.09	2.81	0.00
9	31.80	3.05	0.09	0.00
10	37.66	1.28	0.02	0.00
11	38.34	0.04	4.98	0.02
12	38.73	3.41	0.02	0.00
13	47.37	0.00	0.00	5.26
14	53.48	4.76	0.66	0.00
15	53.53	0.64	4.84	0.00
16	59.55	0.00	0.03	3.78
17	62.60	0.15	0.00	0.05
18	63.10	0.00	0.05	6.98
19	63.23	0.00	0.00	0.04
20	65.80	0.03	0.66	0.05
Sum of Effective Masses		218.06	218.69	218.05

Notes:
1. Fixed at Elevation 100′.
2. The total mass of the containment vessel is 229.16 kip-sec^2/ft.

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Table 3G.2-1 (Sheet 2 of 2)
Steel Containment Vessel Lumped-Mass Stick Model (With Polar Crane)
Modal Properties

Mode	Frequency	Effective Mass		
		X Direction	Y Direction	Z Direction
1	4.20	0.00	78.15	0.00
2	4.98	149.97	0.00	0.17
3	6.44	6.80	0.01	23.59
4	6.47	0.00	112.05	0.00
5	8.00	35.09	0.00	1.19
6	8.61	0.00	4.06	0.00
7	12.58	0.03	0.18	0.00
8	16.00	0.06	0.08	0.22
9	16.09	5.16	0.00	146.06
10	17.21	24.58	0.00	36.27
11	18.86	0.00	40.88	0.00
12	20.64	13.30	0.00	0.71
13	23.74	0.00	0.40	0.00
14	27.04	0.13	0.00	15.64
15	29.62	2.88	0.00	3.61
16	30.21	0.00	0.07	0.00
17	31.62	0.00	2.92	0.01
18	32.85	0.58	0.00	0.38
19	34.02	0.28	0.01	4.87
20	36.97	0.20	1.61	0.00
Sum of Effective Masses		239.04	240.42	232.72

Notes:
1. Fixed at Elevation 100′.
2. The total mass of the containment vessel with the polar crane is 261.02 kip-sec^2/ft.

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Table 3G.2-2 Comparison of Frequencies for Containment Vessel Seismic Model

Mode No.	Vertical Model		Horizontal Model	
	Shell of Revolution Model	Stick Model	Shell of Revolution Model	Stick Model
1	16.51 hertz	16.97 hertz	6.20 hertz	6.31 hertz
2	23.26 hertz	28.20 hertz	18.58 hertz	18.96 hertz

Note:
1. Fixed at elevation 100′.

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Table 3G.3-1
AP1000 ZPA for 2D SASSI Cases

	North-South		Hard Rock ZPA [g]	Firm Rock ZPA [g]	Soft Rock ZPA [g]	UBSM ZPA [g]	SM ZPA [g]	Soft Soil ZPA [g]
	Node	El. feet						
ASB	21	81.5	0.326	0.326	0.345	**0.358**	0.306	0.249
	41	99	0.348	0.327	0.347	**0.361**	0.308	0.227
	120	179.6	**0.571**	0.501	0.469	0.498	0.529	0.247
	150	242.5	0.803	0.795	0.816	**0.819**	0.787	0.29
	310	333.1	1.449	1.561	**1.567**	1.524	1.226	0.453
SCV	407	138.6	0.405	**0.424**	0.408	0.387	0.407	0.232
	411	200	0.82	**0.916**	0.672	0.541	0.484	0.263
	417	281.9	1.396	**1.465**	1.031	0.723	0.598	0.372
CIS	535	134.3	**0.548**	0.45	0.347	0.368	0.355	0.229
	538	169	**1.517**	0.874	0.45	0.441	0.397	0.317
	East-West		**Hard Rock ZPA [g]**	**Firm Rock ZPA [g]**	**Soft Rock ZPA [g]**	**UBSM ZPA [g]**	**SM ZPA [g]**	**Soft Soil ZPA [g]**
	Node	**El. feet**						
ASB	21	81.5	0.309	0.318	0.359	**0.376**	0.311	0.235
	41	99	0.318	0.336	0.367	**0.385**	0.317	0.237
	120	179.6	**0.607**	0.561	0.546	0.549	0.605	0.295
	150	242.5	0.84	0.823	0.854	0.912	**0.962**	0.557
	310	333.1	1.449	1.536	1.624	**1.74**	1.506	0.891
SCV	407	138.6	0.528	0.529	**0.535**	0.513	0.38	0.247
	411	200	0.817	**0.95**	0.816	0.741	0.515	0.429
	417	281.9	1.251	**1.503**	1.136	0.985	0.716	0.675
CIS	535	134.3	**0.52**	0.404	0.391	0.404	0.365	0.259
	538	169	**1.679**	1.052	0.755	0.553	0.526	0.441

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Table 3G.4-1
Key Nodes at Location

Location	General Area	Elevation (feet)
CIS at Reactor Vessel Support Elevation	SCV Center	100.00
CIS at Operating Deck	SG West Compartment, NE	134.25
ASB NE Corner at Control Room Floor	NE Corner	116.50
ASB Corner of Fuel Building Roof at Shield Building	NW Corner of Fuel Bldg	179.19
ASB Shield Building Roof Area	South Side of Shield Bldg	327.41
SCV Near Polar Crane	SCV Stick Model	224.00

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Table 3G.4-2
Maximum Bearing Pressure from 2D Time History Analyses

Soil Case	Analysis	East Edge (ksf)	West Edge (ksf)
Hard Rock	Linear	17.18	32.77
	Liftoff	17.38	34.85
Upper-bound Soft to Medium	Linear	19.46	31.69
	Liftoff	18.42	33.51
Soft to Medium	Linear	15.84	30.82
	Liftoff	17.06	32.18

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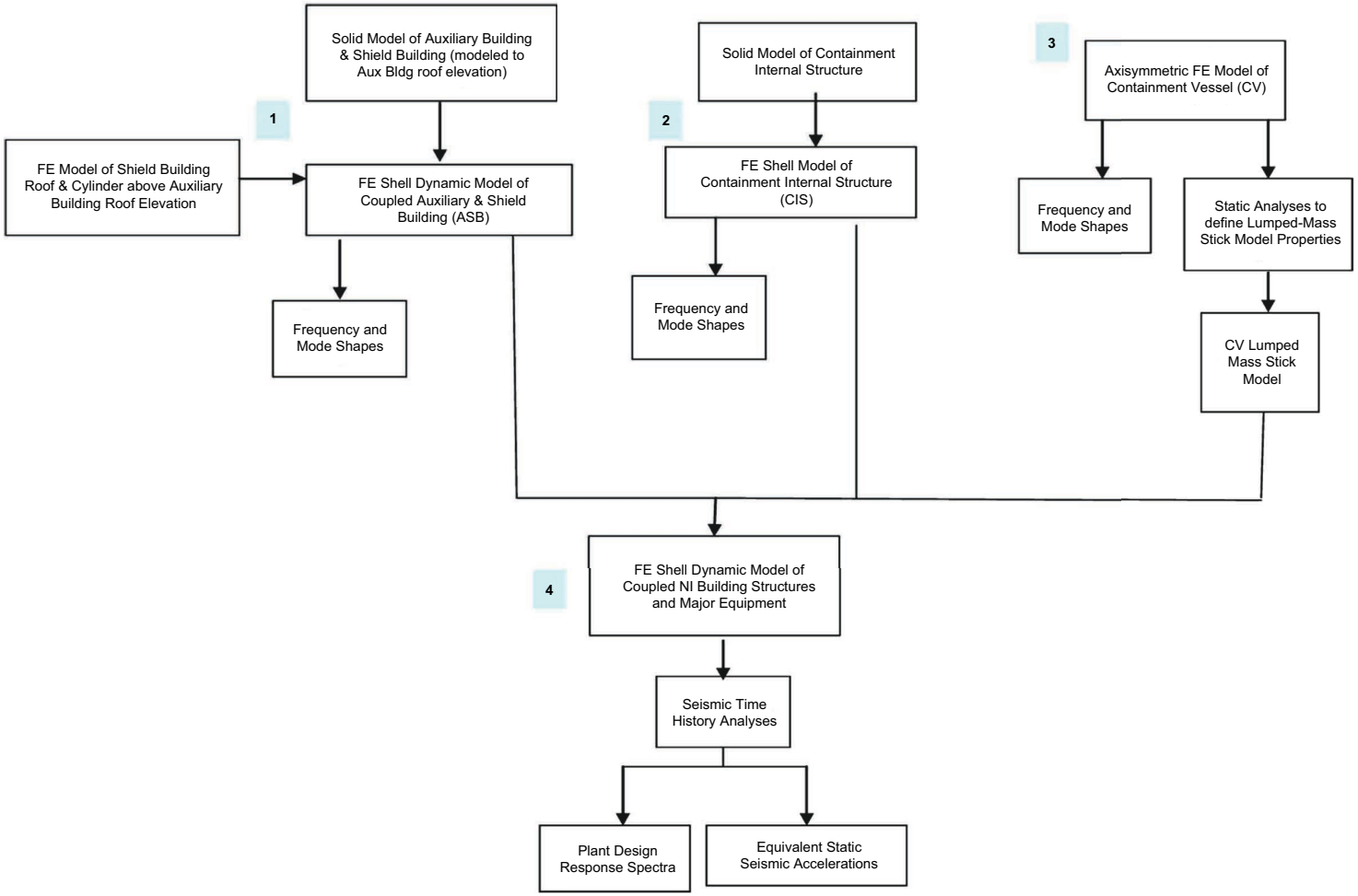

Figure 3G.1-1
Nuclear Island Seismic Analysis Models

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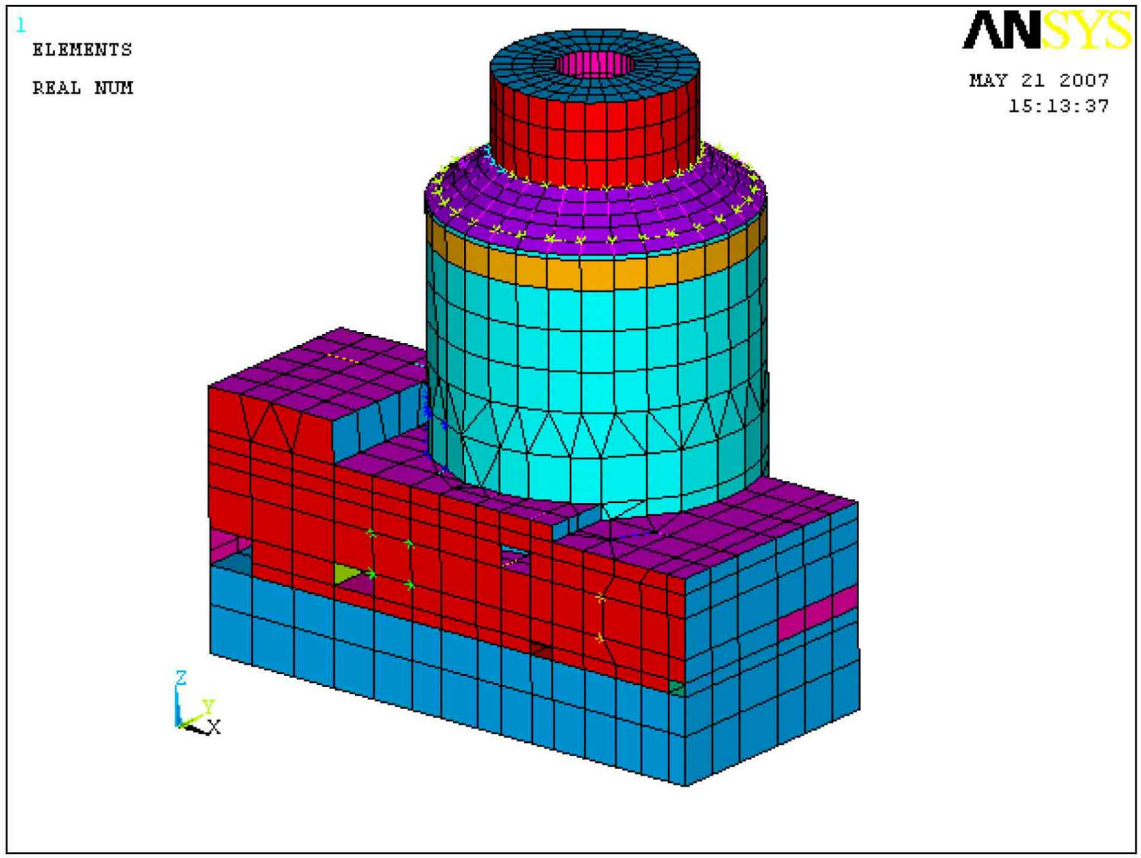

Figure 3G.2-1
3D Finite Element Model of
Coupled Shield and Auxiliary Building

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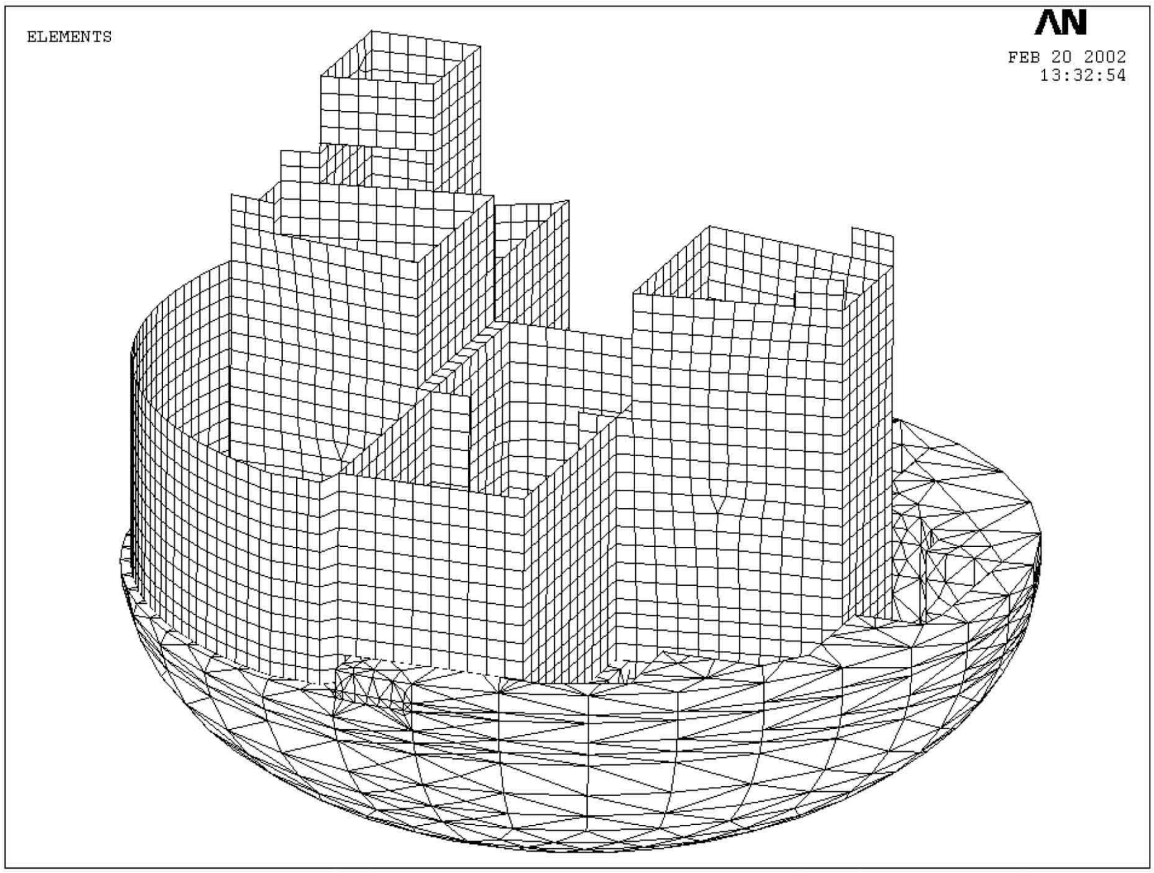

Note: This figure shows the finite element model of walls and basemat inside containment. Floors are not shown.

Figure 3G.2-2
3D Finite Element Model of
Containment Internal Structures

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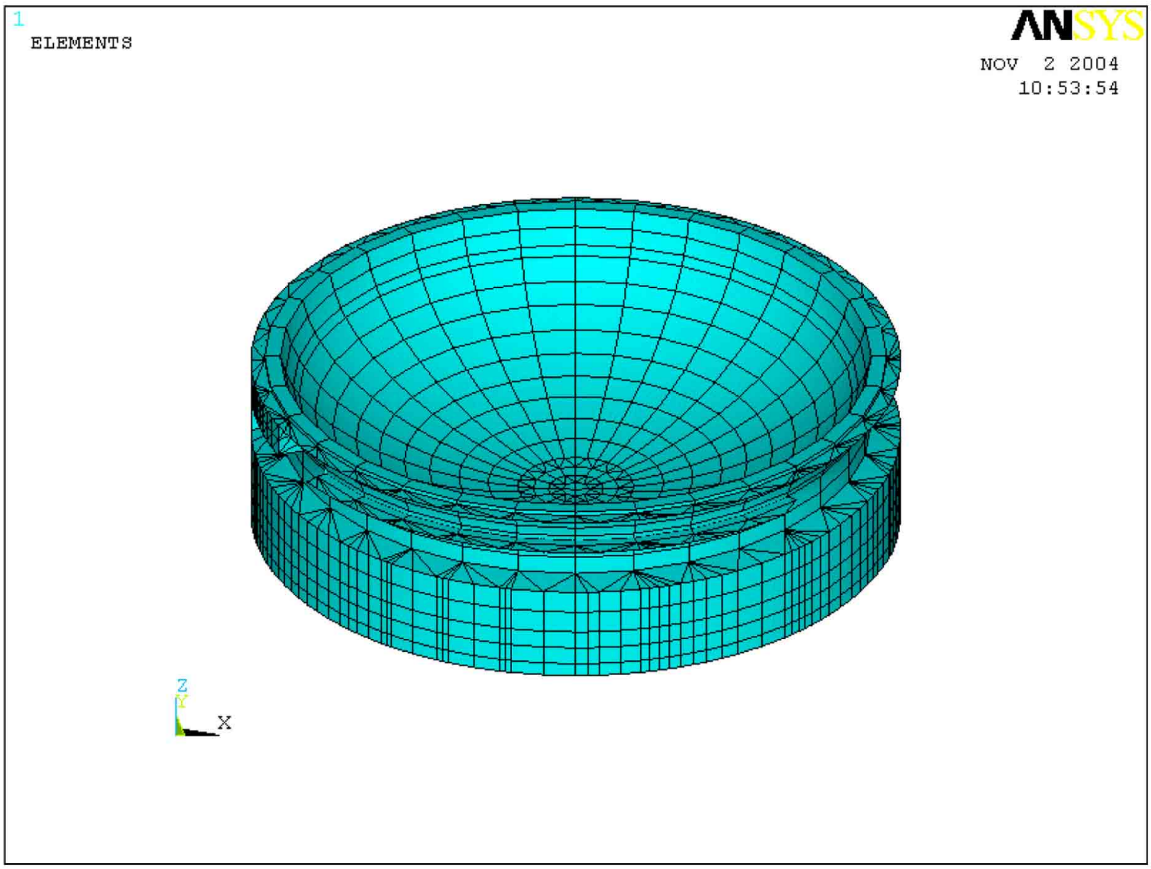

Figure 3G.2-3
3D Finite Element Model of
Containment Outer Basemat (Dish)

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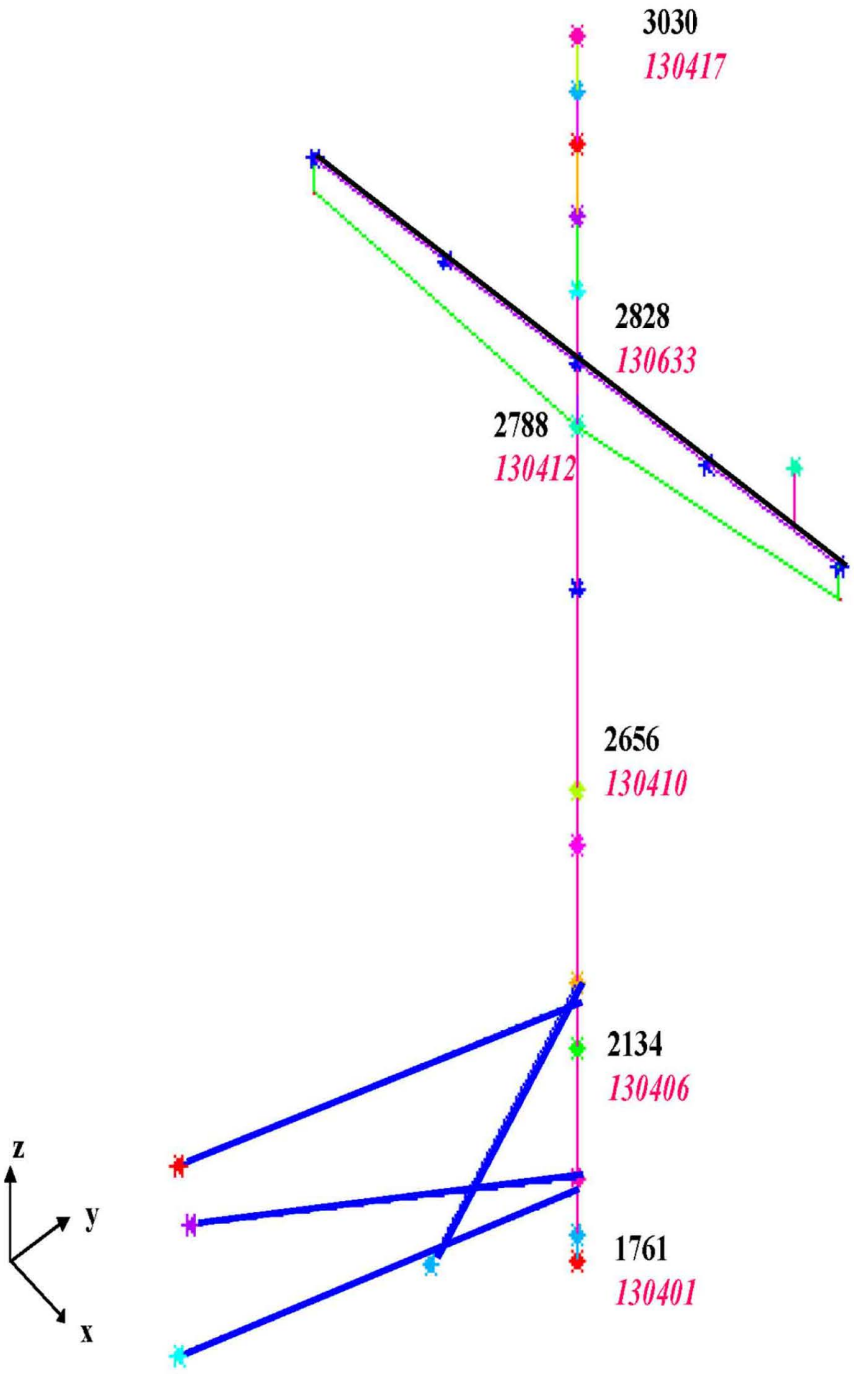

Figure 3G.2-4
Steel Containment Vessel and Polar Crane Models

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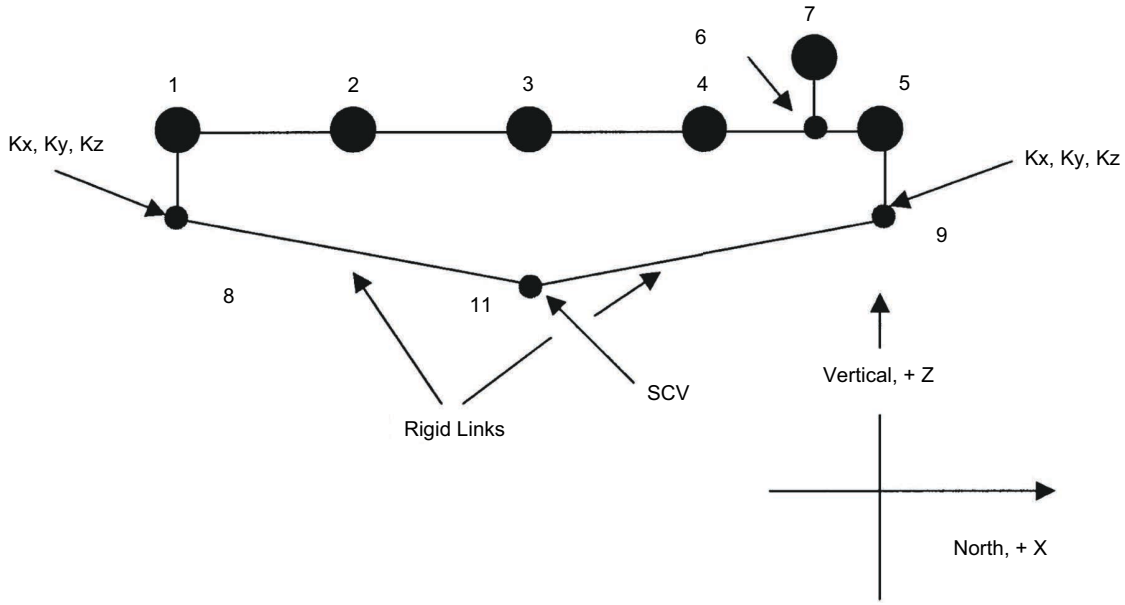

Local SCV Stiffness are Kx, Ky, Kz

Dynamic Degrees of Freedom

- Masses at nodes 1, 2, 3, 4, 5, and 7
- All Mass nodes have DOFs in X, Y, and Z directions

Comments:

1. Cross Beams between girders are represented by rotation spring constants Kxx and Kzz
2. Cross Beam rotational spring constant Kyy is negligible compared to girder stiffness

Figure 3G.2-5A
Polar Crane Model Simplified Model

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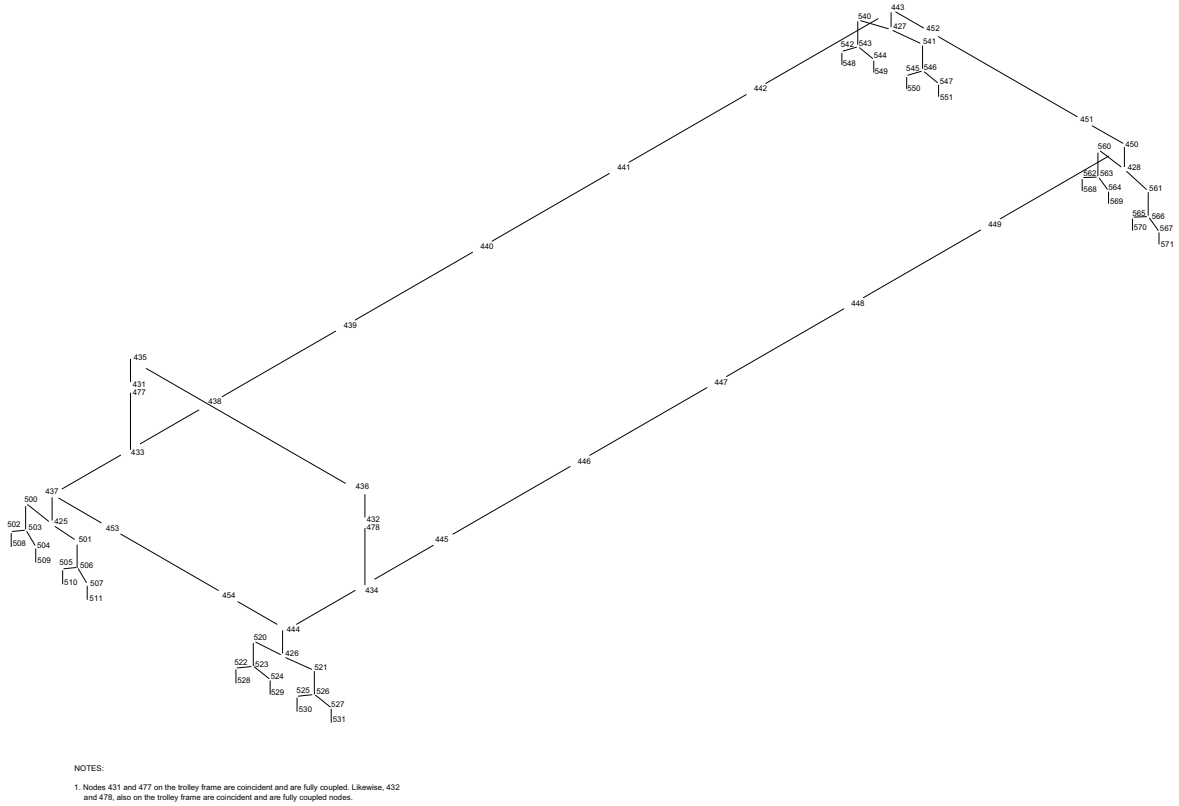

Figure 3G.2-5B
Polar Crane Model Detailed Model

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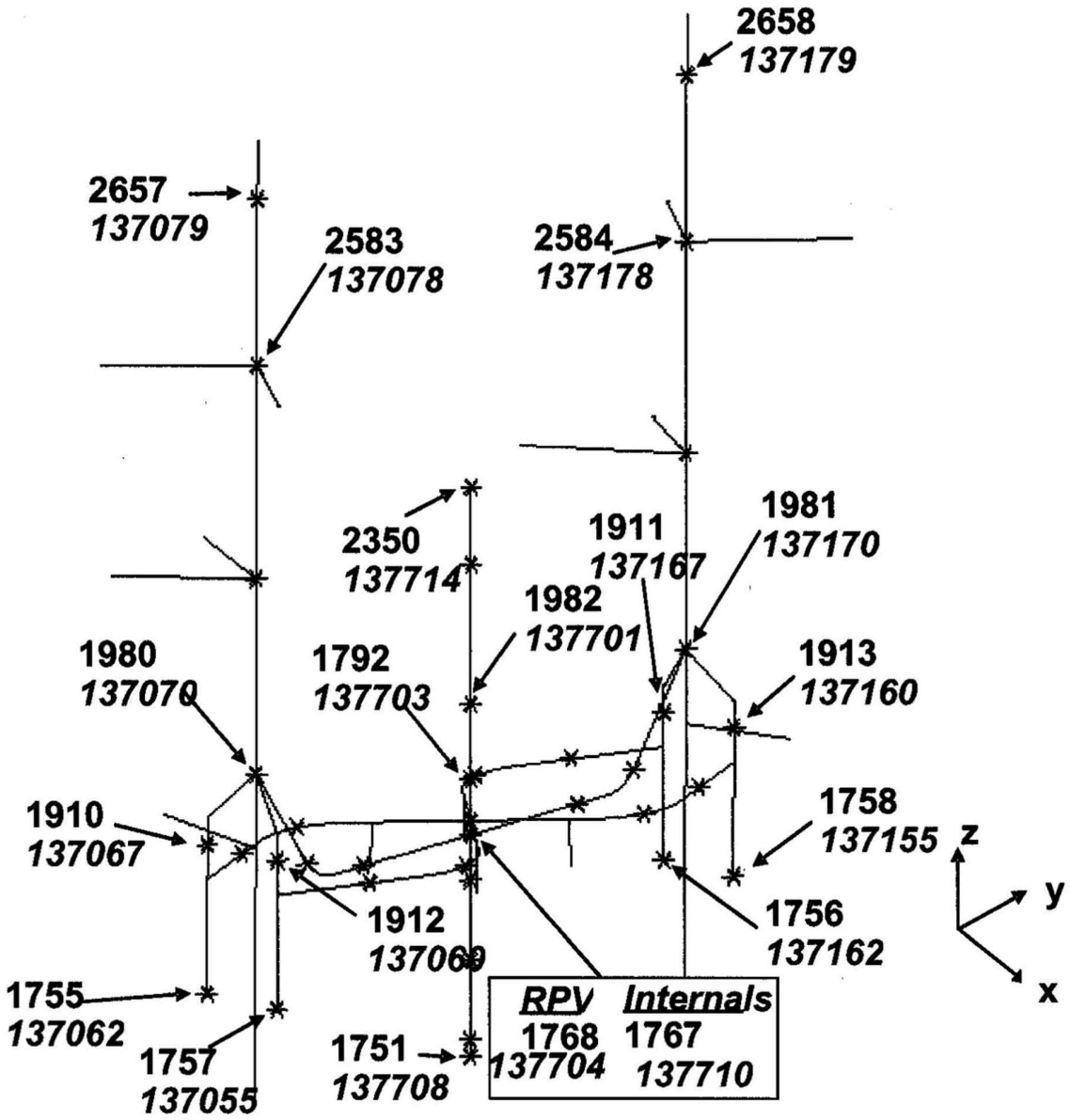

Figure 3G.2-6
Reactor Coolant Loop
Lumped-Mass Stick Model

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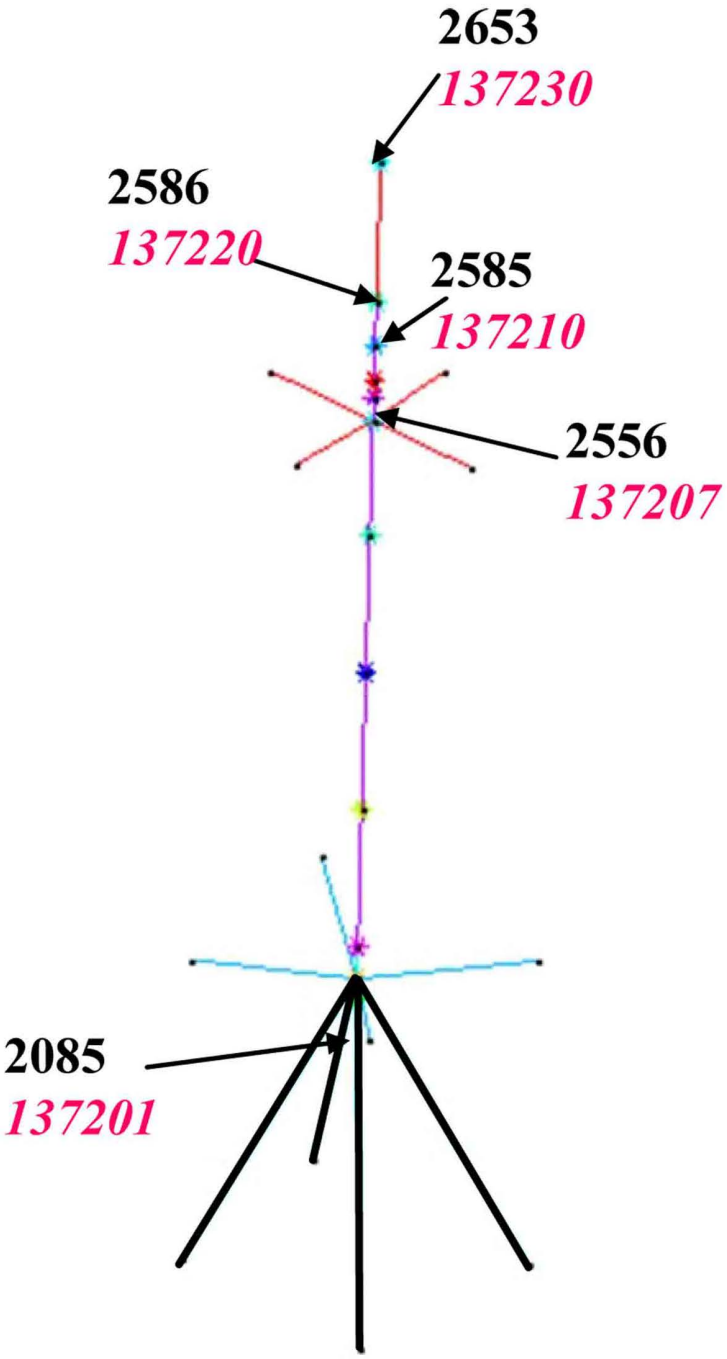

Figure 3G.2-7
Pressurizer Model

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Figure 3G.2-8
Core Makeup Tank Models

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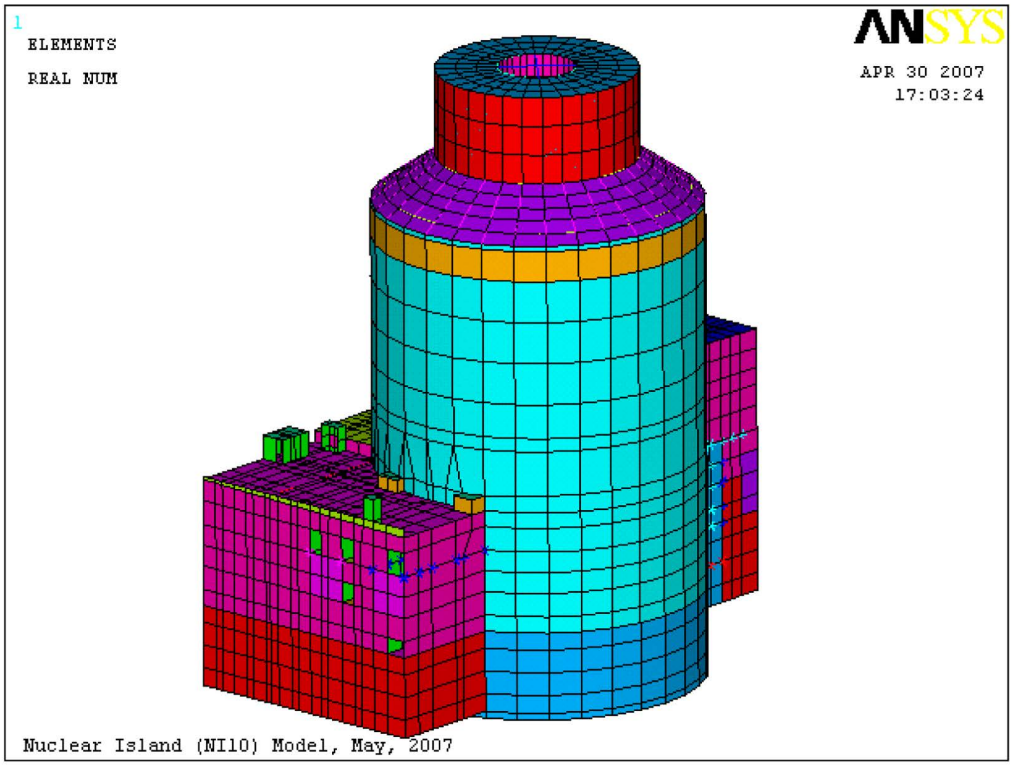

Figure 3G.2-9
AP1000 Nuclear Island Solid-Shell Model (NI10)

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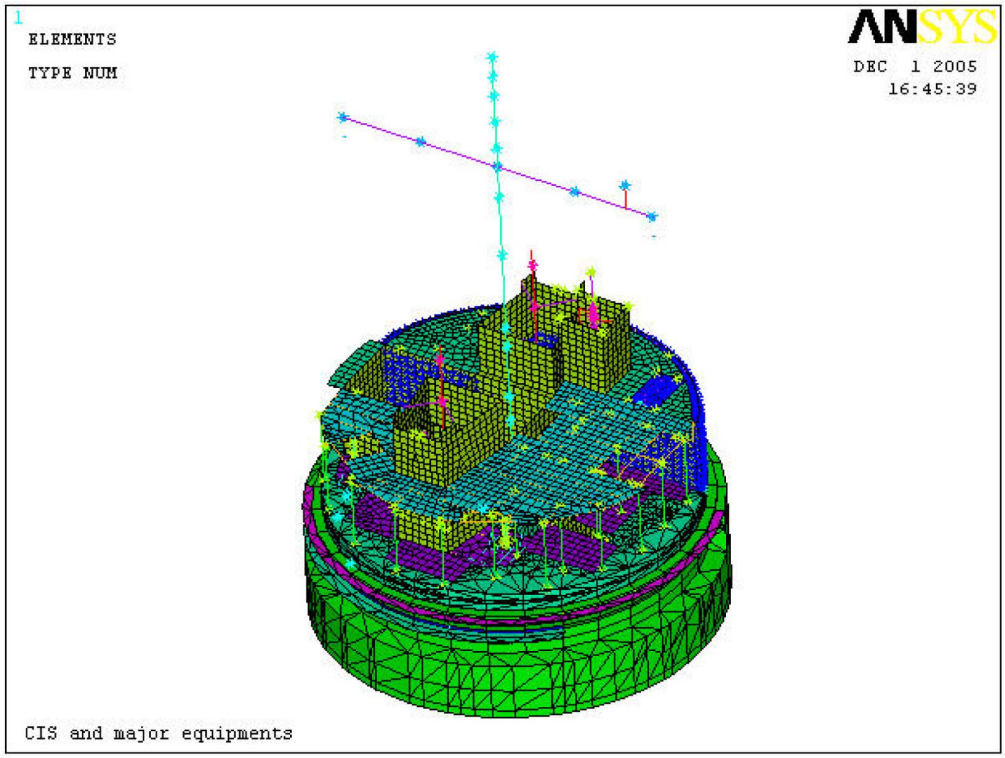

Figure 3G.2-10
Containment Internal Structure with the SCV, PC, Reactor Coolant Loop, and Pressurizer

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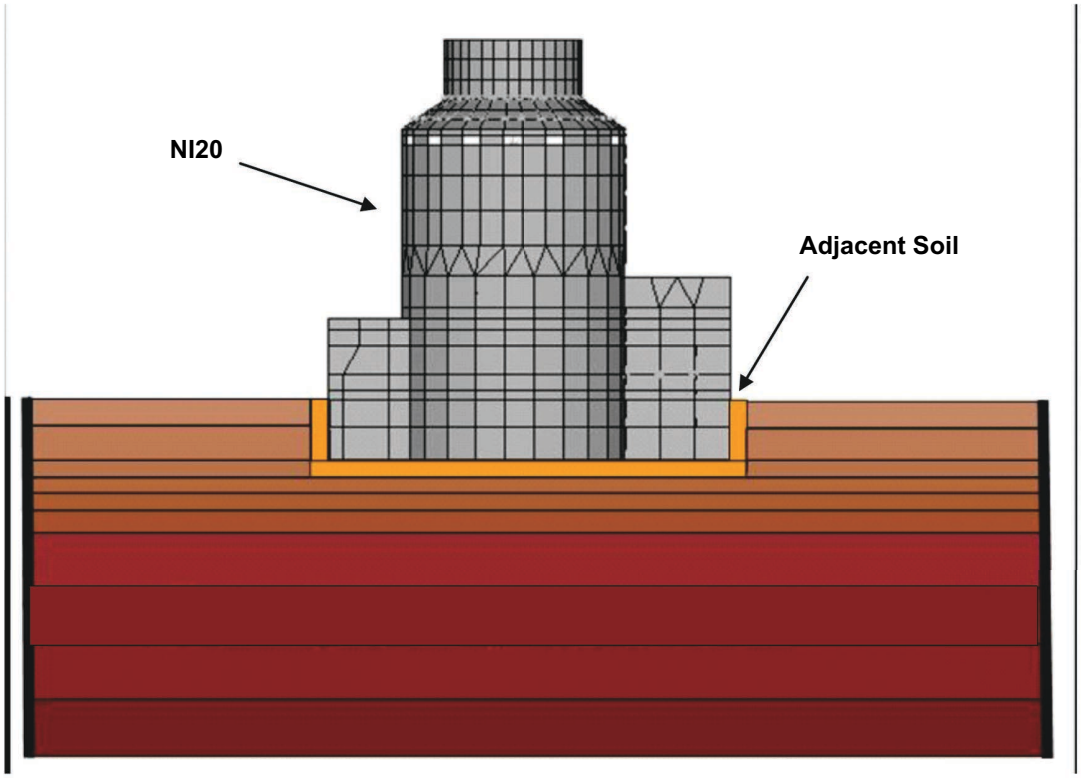

Note: The adjacent soil elements are part of the structural portion of SASSI and have the same material properties as the soil. These elements are used to obtain soil lateral and bearing soil pressures.

Figure 3G.2-11
Soil Structure Interaction Model – NI20 Looking East

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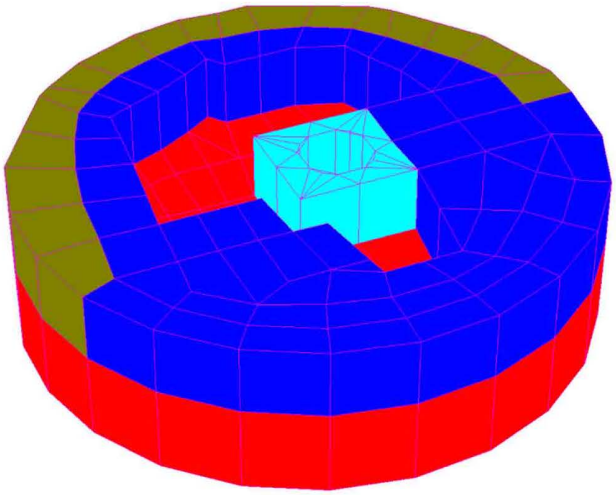

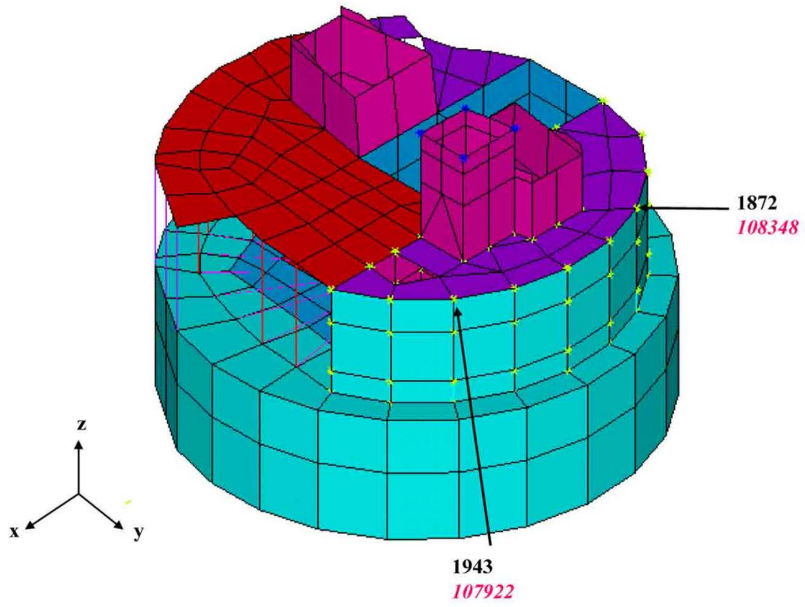

Figure 3G.2-12
Coarse Model of Containment Internal Structures

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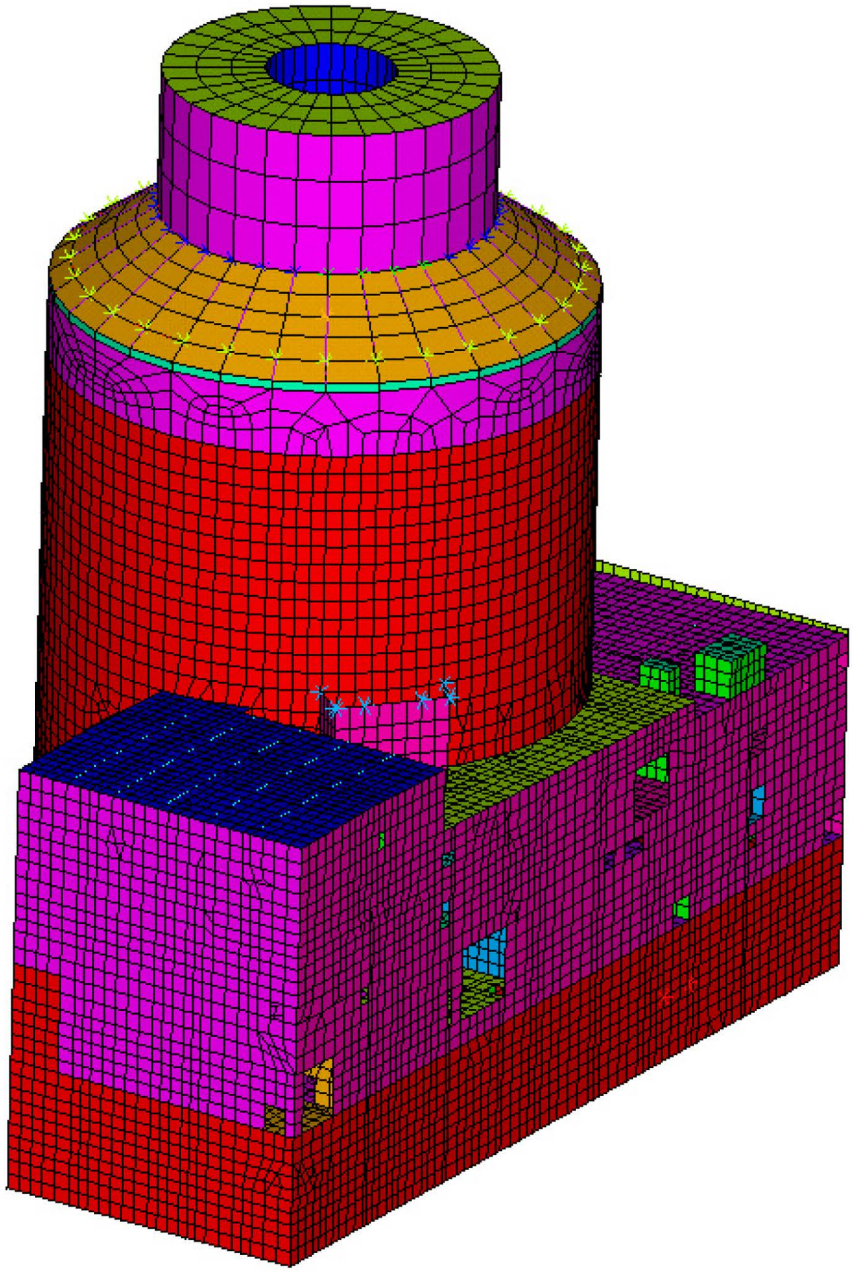

Figure 3G.2-13
Fine Mesh (NI05) Model of Auxiliary and Shield Building

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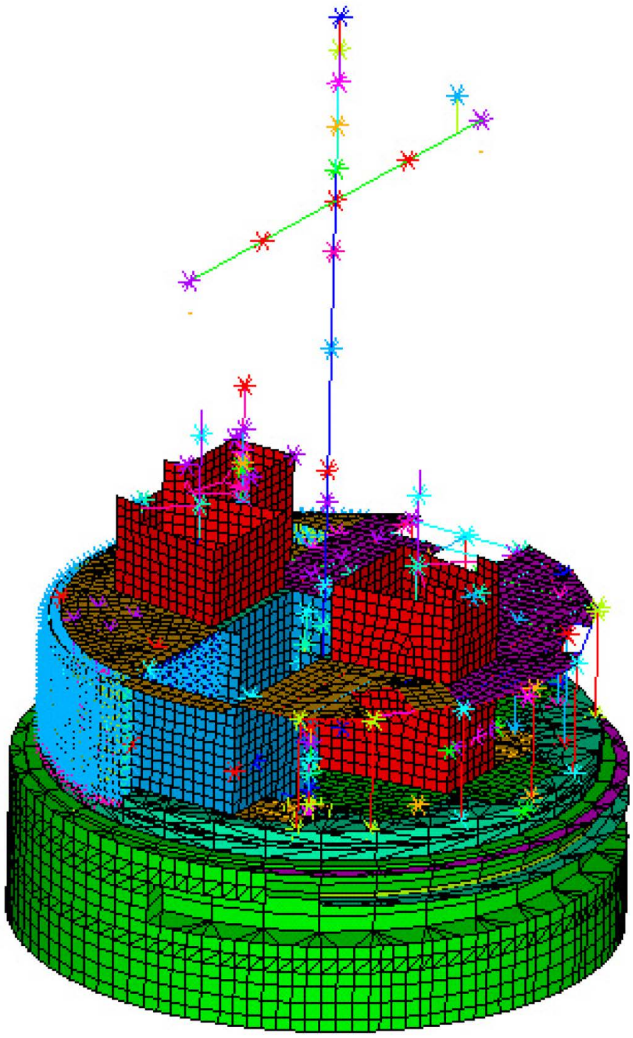

Figure 3G.2-14
NI05 Model of Containment Internal Structures

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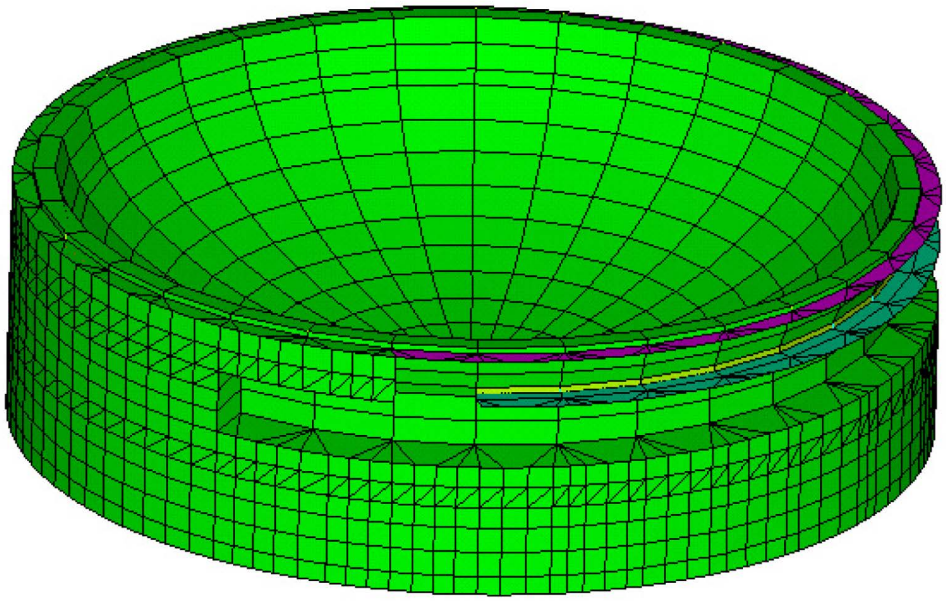

Figure 3G.2-15
3D NI05 Refined Mesh Model of Outer Containment Basemat (Dish)

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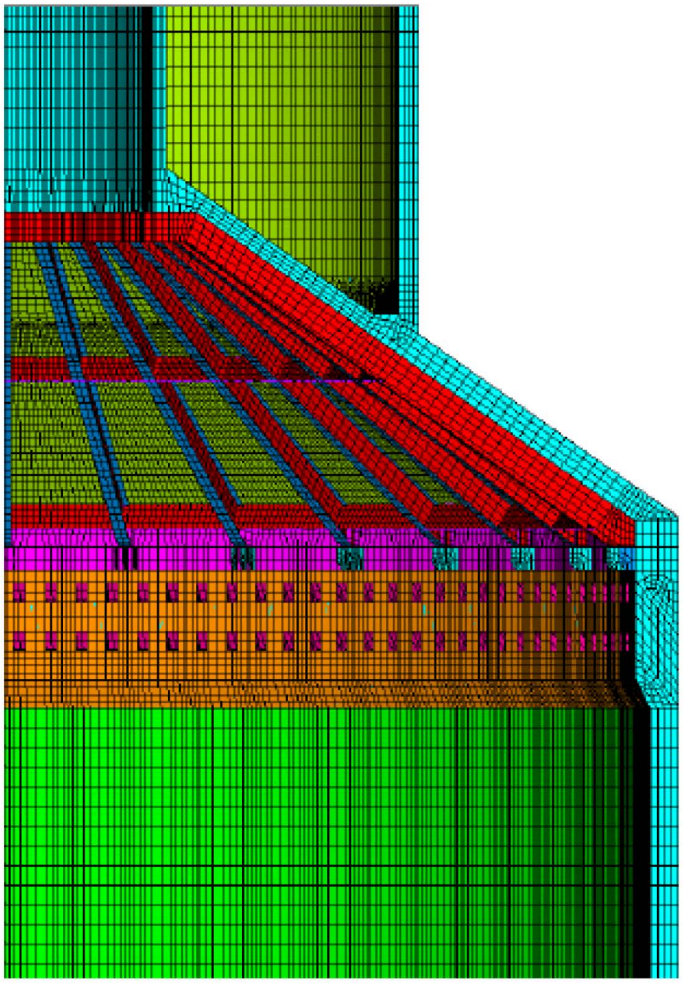

Figure 3G.2-16
Quadrant Model of Shield Building Roof

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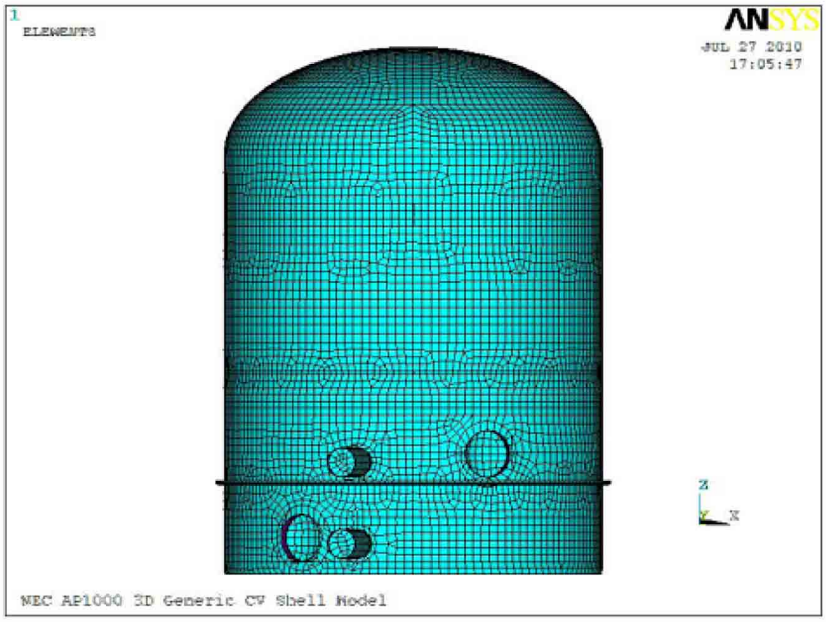
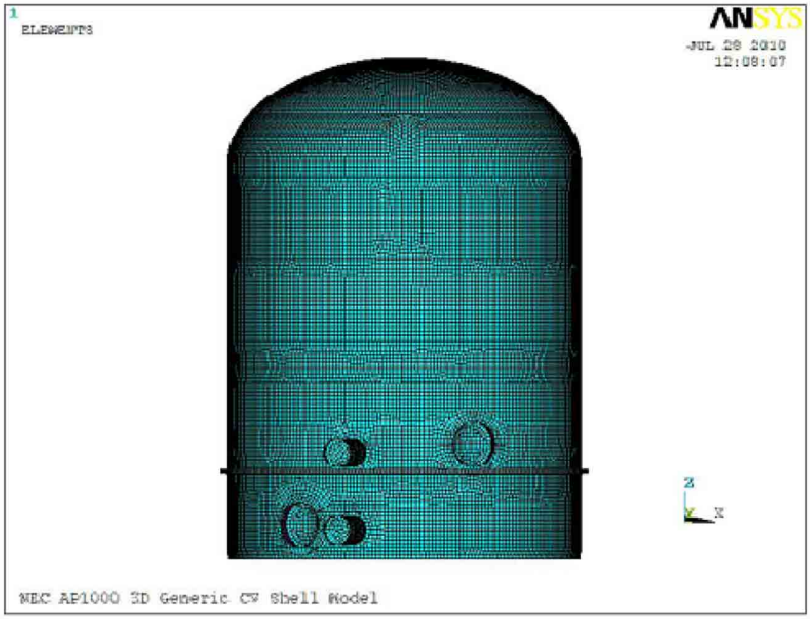


Mesh Size 37″ x 37″



Mesh Size 18″ x 18″

Figure 3G.2-17
Detailed 3D Finite Element Model of Containment Vessel Including Large Penetrations

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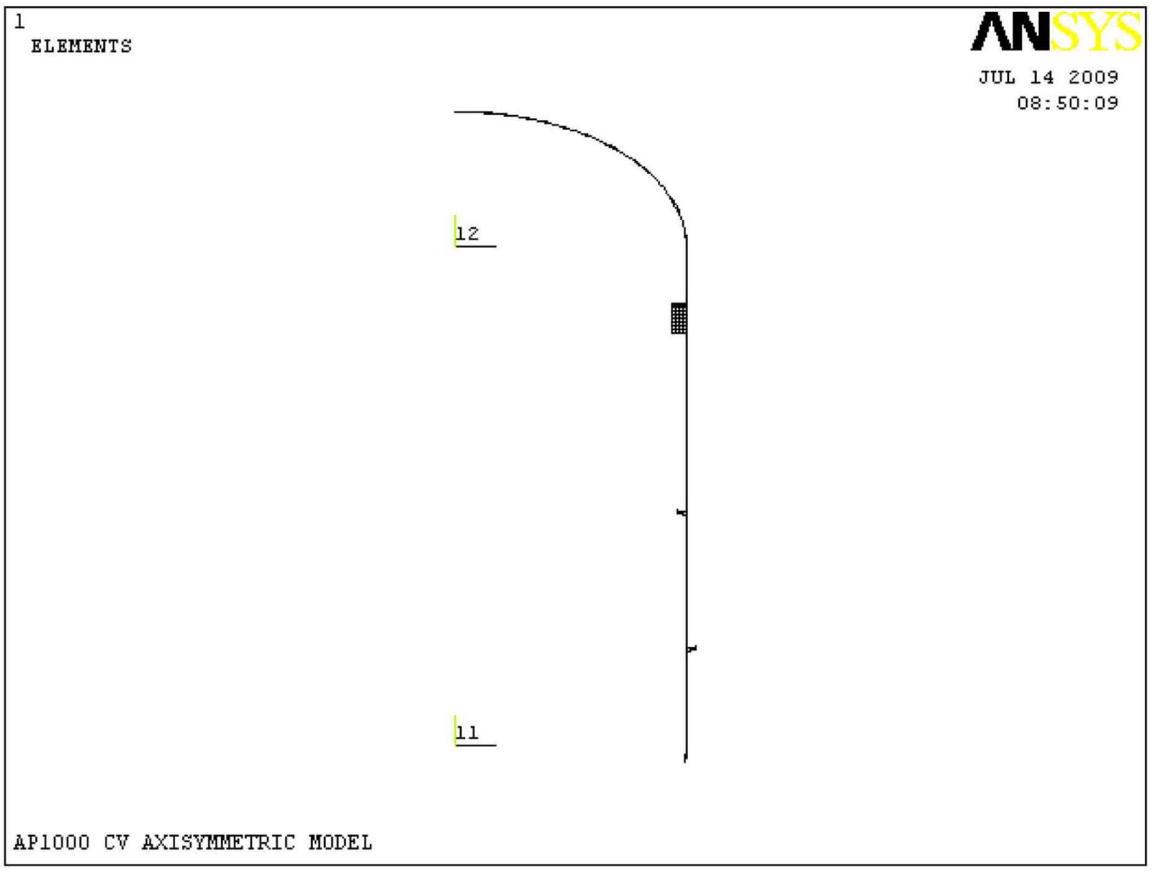

Figure 3G.2-18
Axisymmetric Model of Containment Vessel

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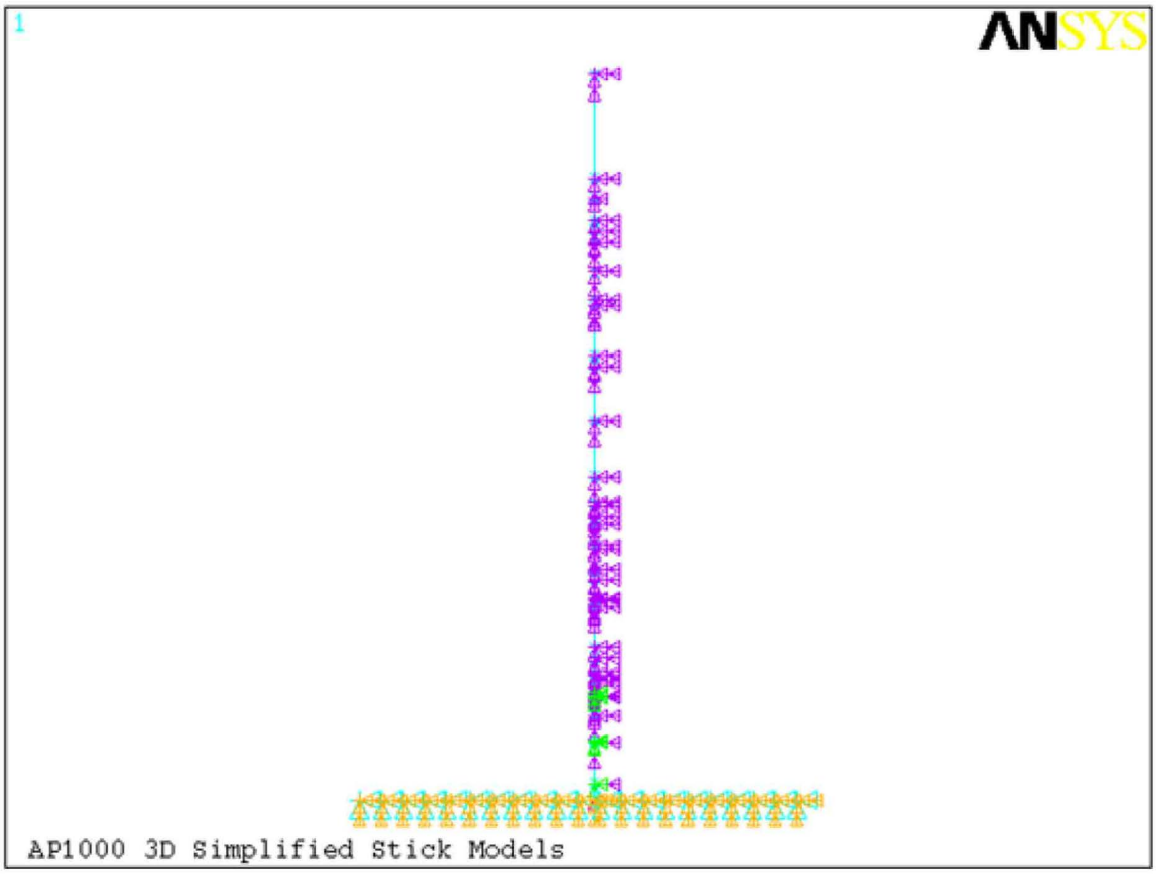

Figure 3G.2-19
Schematic of Non-linear 2D East-West Nuclear Island Stick Model Used for Stability Evaluation that Addresses Sliding and Overturning

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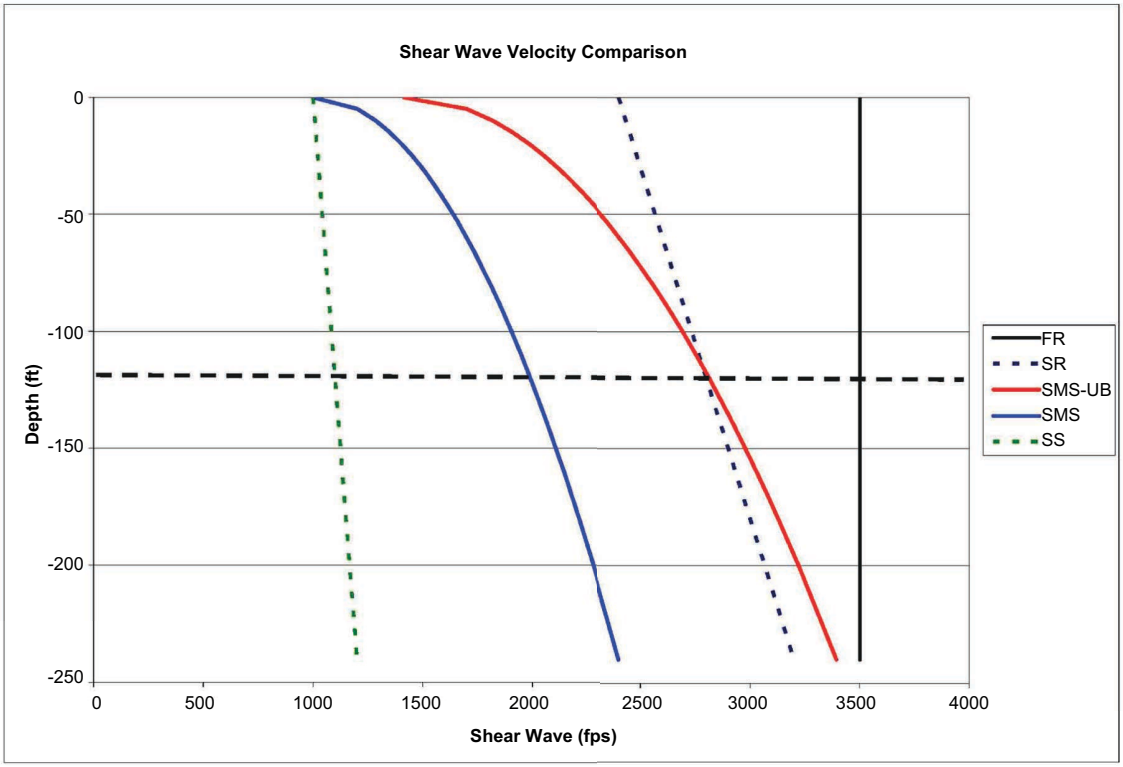

Note: Fixed base analyses were performed for hard rock sites. These analyses are applicable for shear wave velocity greater than 8000 feet per second.

Figure 3G.3-1
Generic Soil Profiles

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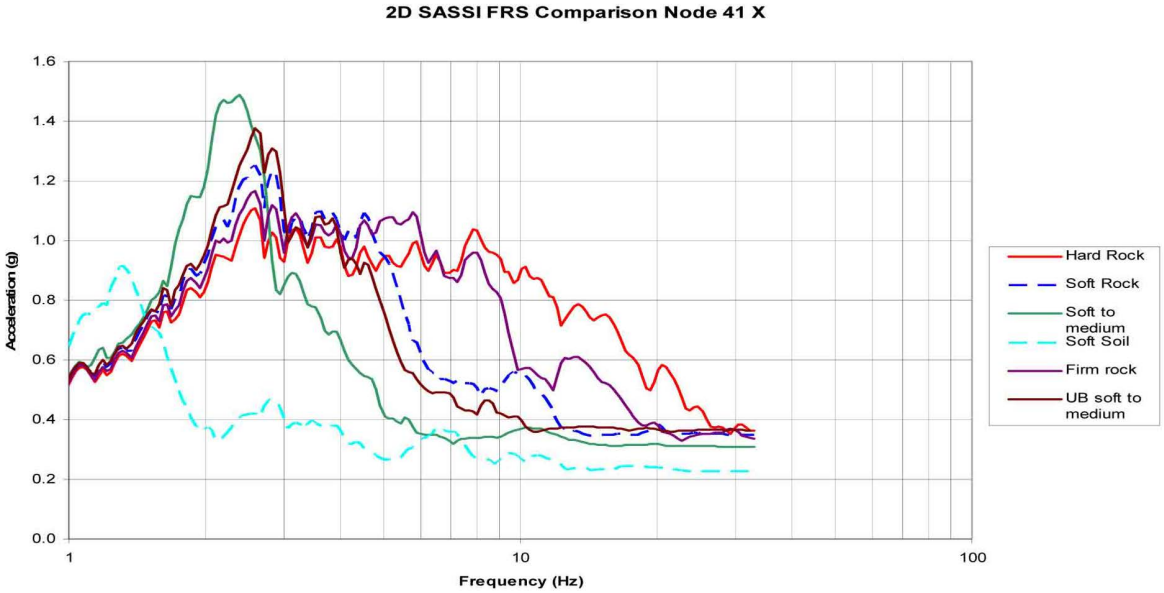

Figure 3G.3-2
2D SASSI FRS – Node 41 X (ASB El. 99′)

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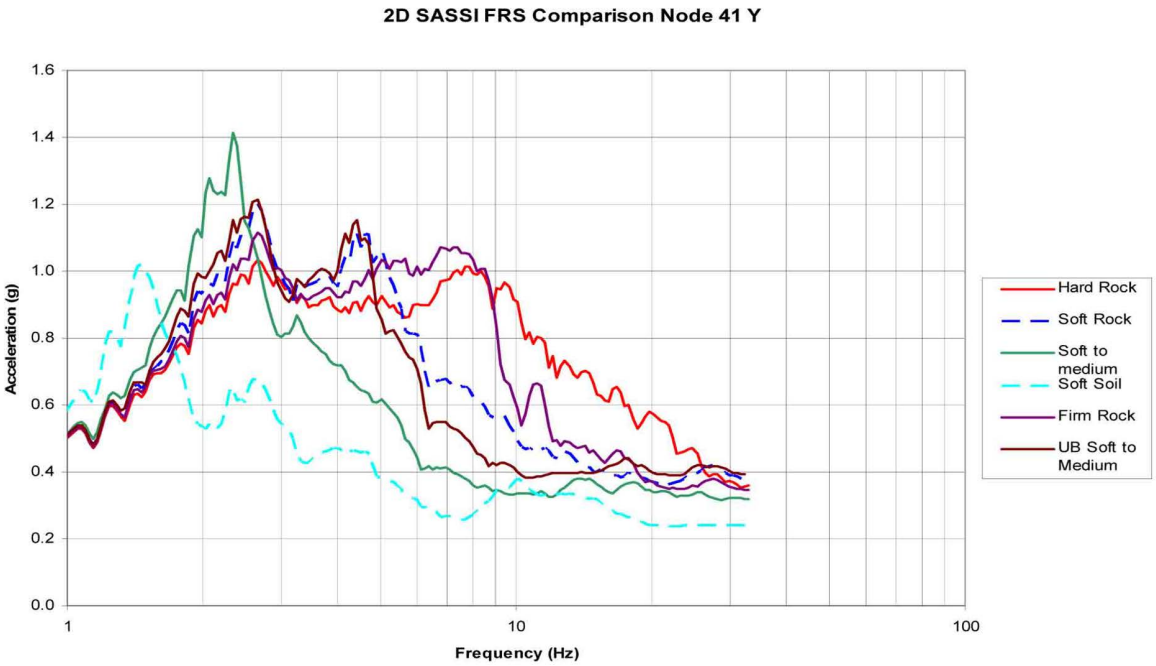

Figure 3G.3-3
2D SASSI FRS – Node 41 Y (ASB El. 99′)

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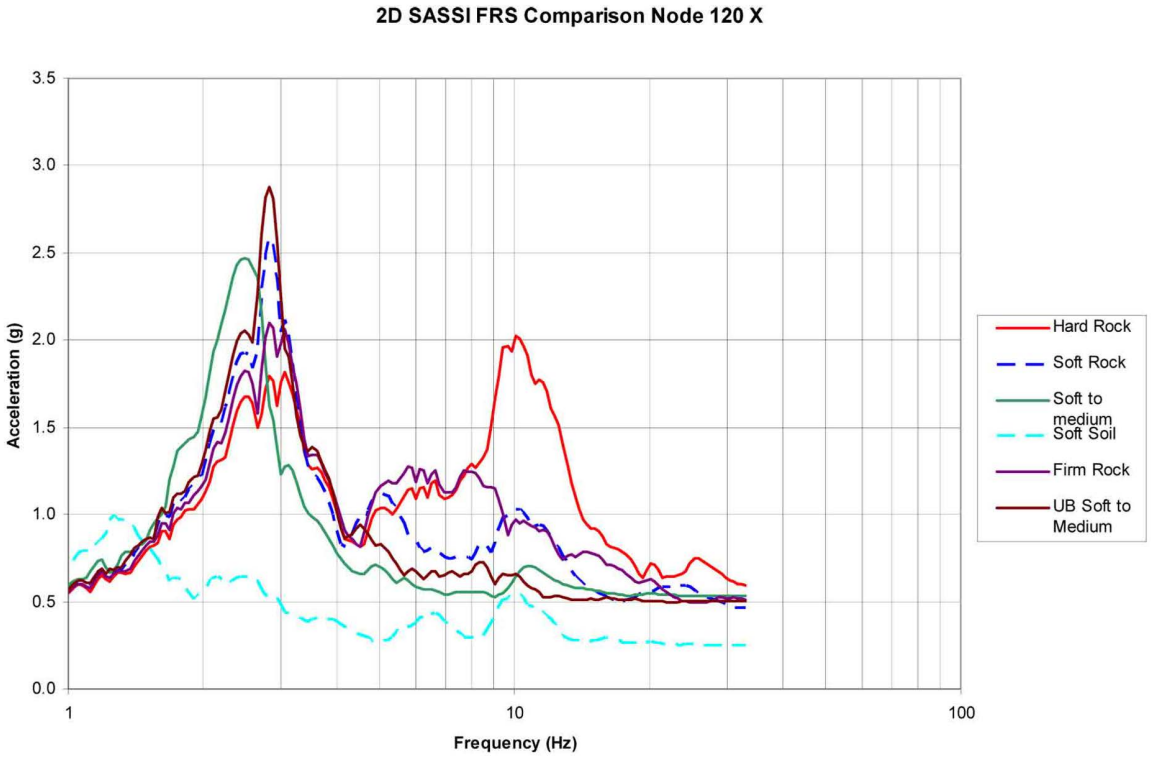

Figure 3G.3-4
2D SASSI FRS – Node 120 X (ASB El. 179.6')

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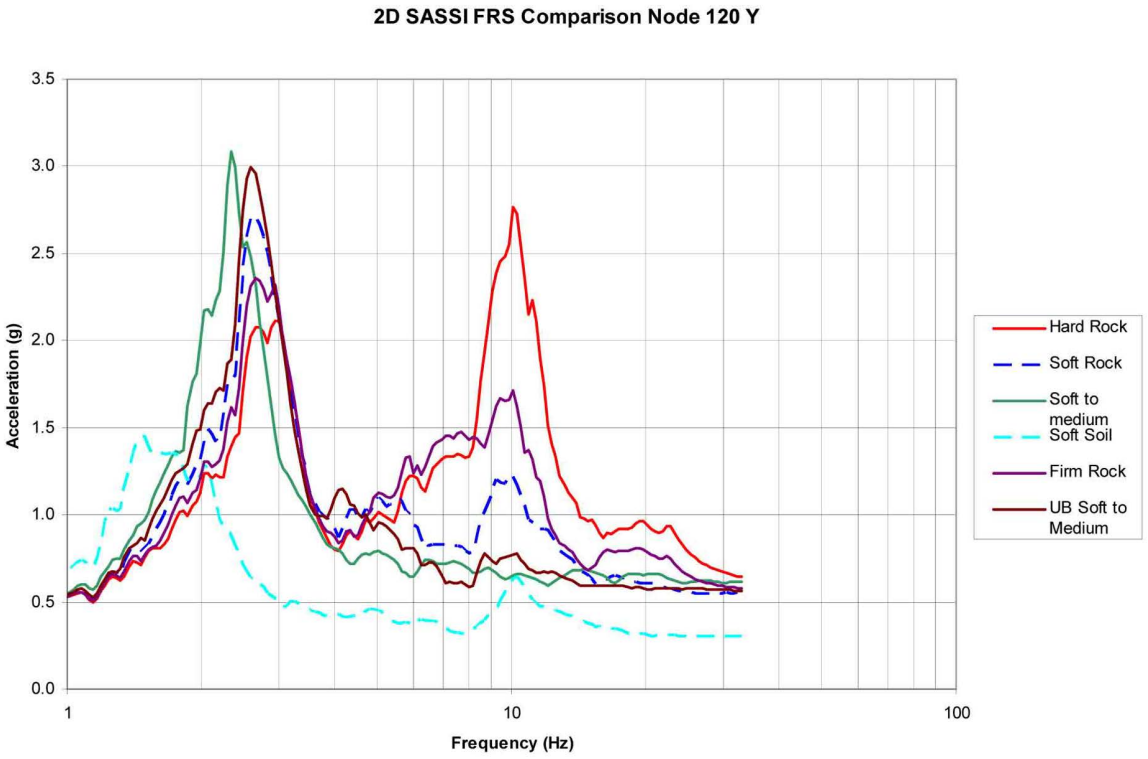

Figure 3G.3-5
2D SASSI FRS – Node 120 Y (ASB El. 179.6')

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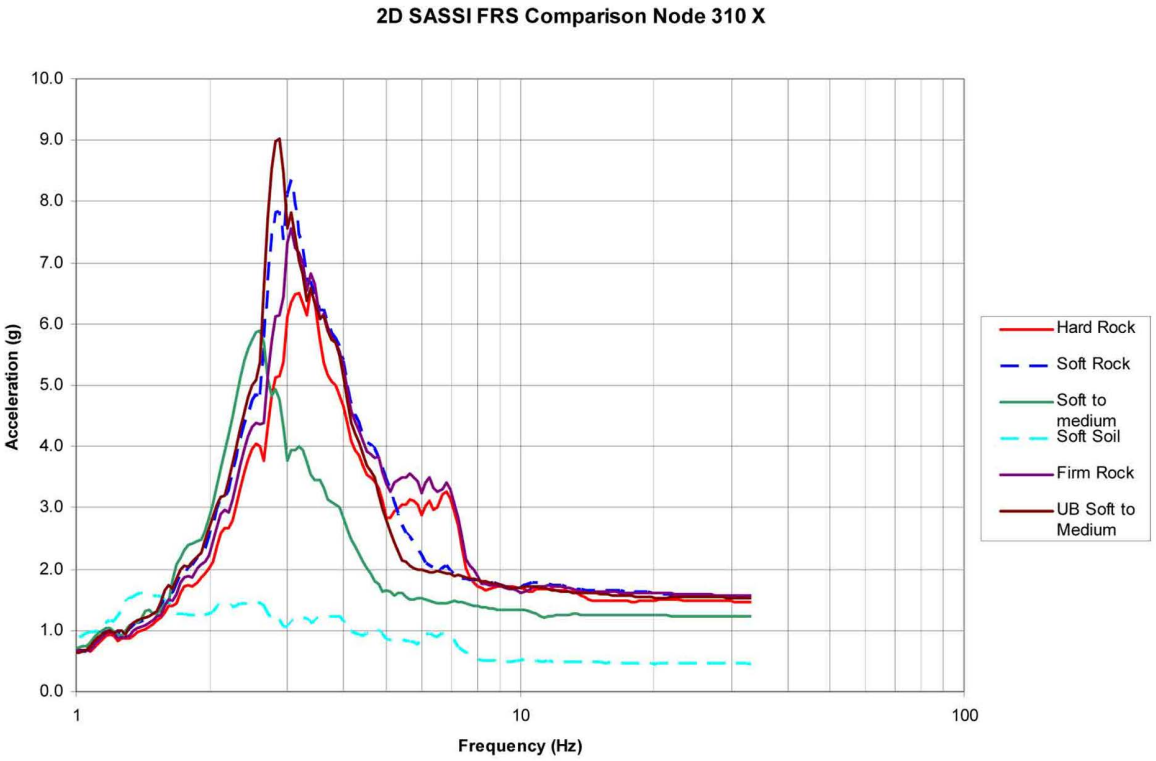

Figure 3G.3-6
2D SASSI FRS – Node 310 X (ASB El. 333.2′)

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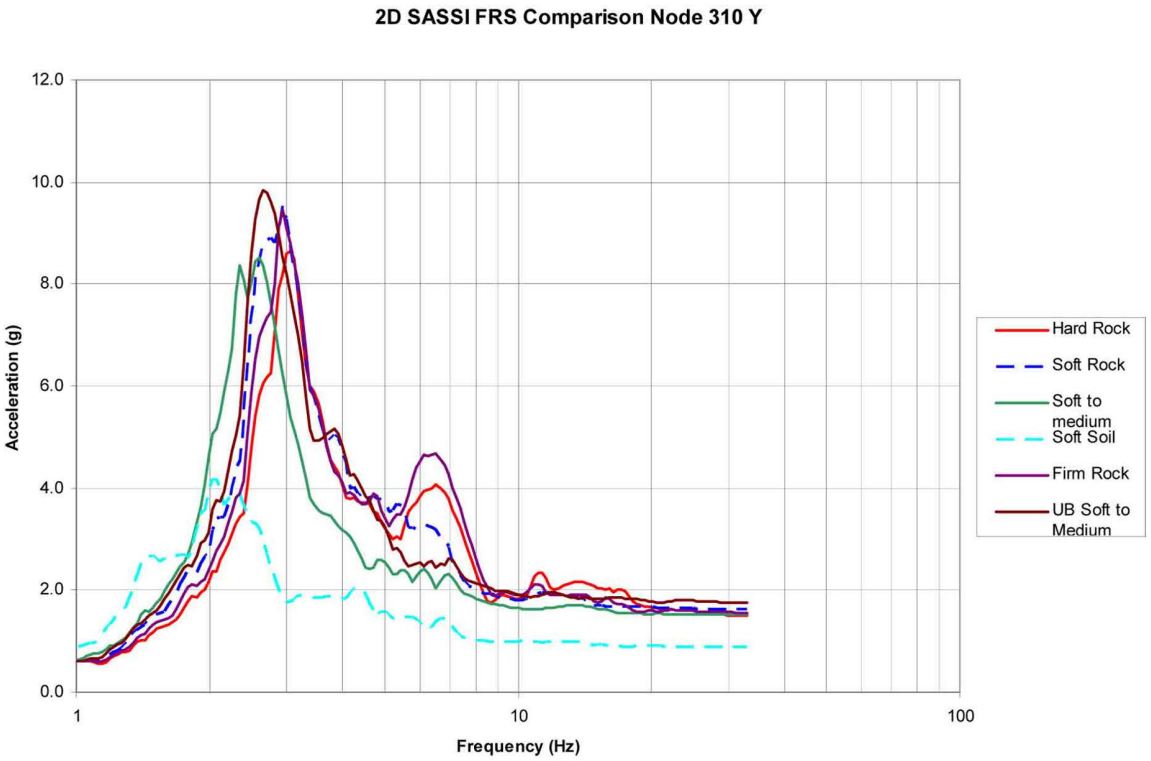

Figure 3G.3-7
2D SASSI FRS – Node 310 Y (ASB El. 333.2′)

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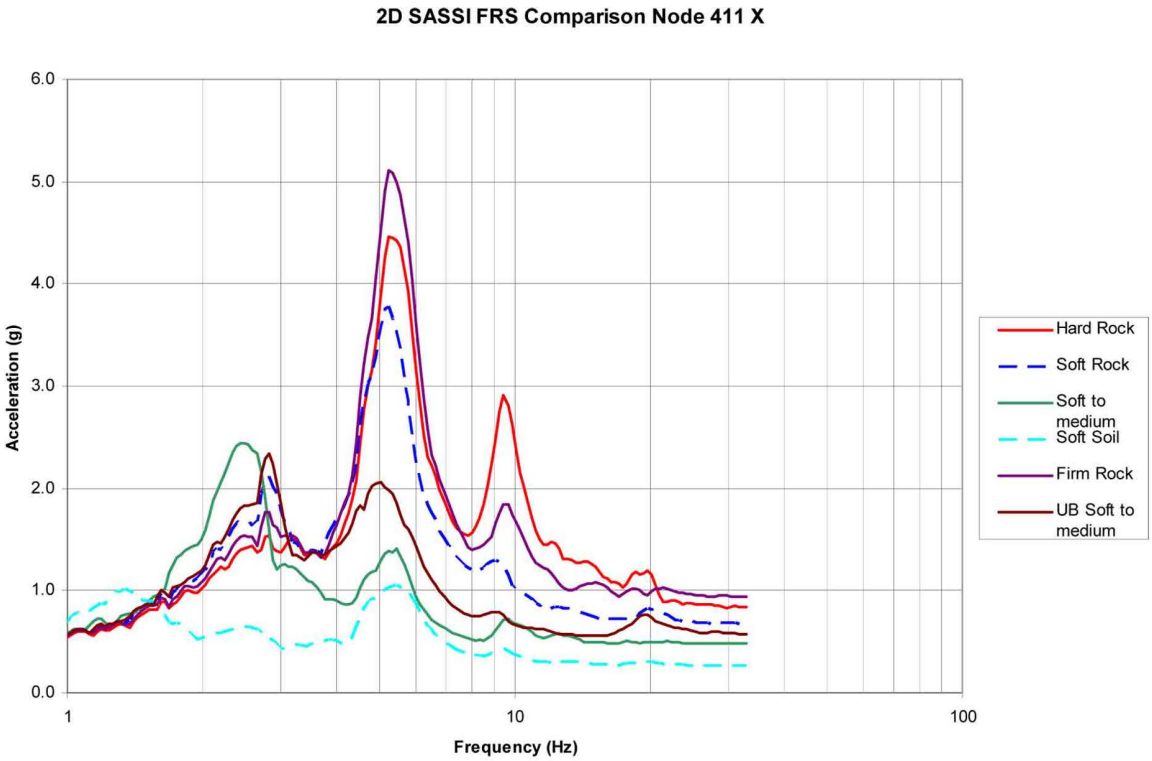

Figure 3G.3-8
2D SASSI FRS – Node 411 X (SCV El. 200.0′)

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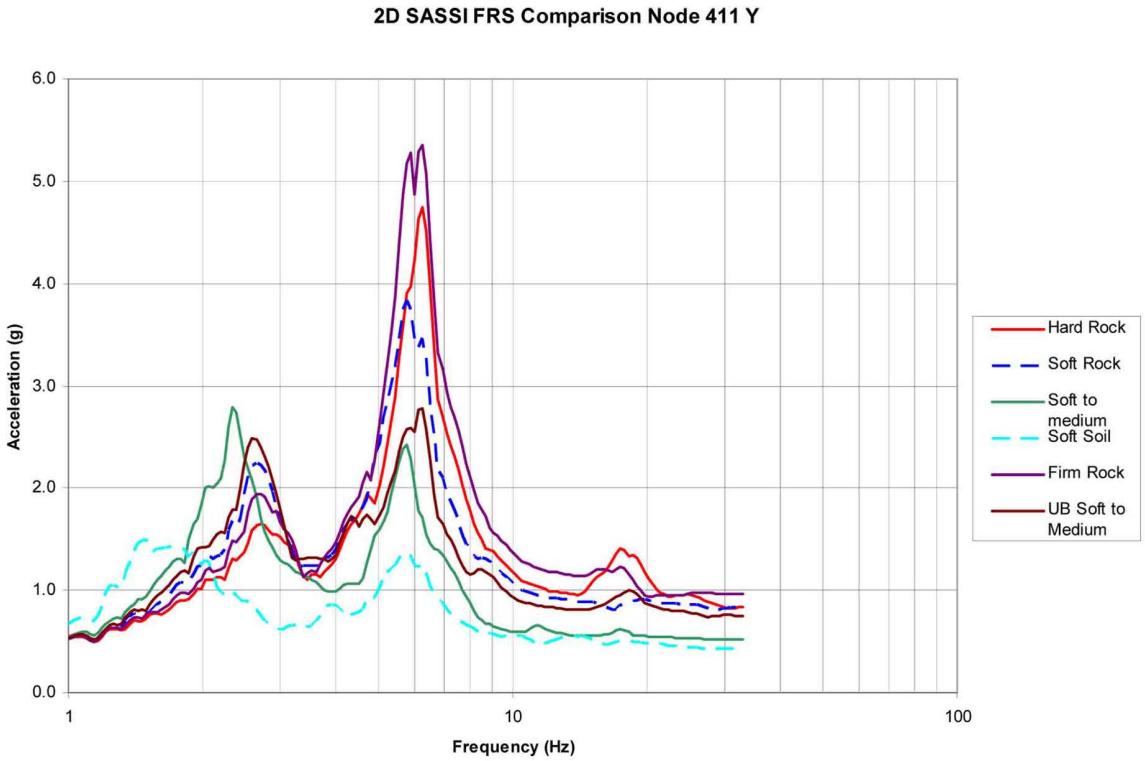

Figure 3G.3-9
2D SASSI FRS – Node 411 Y (SCV El. 200.0′)

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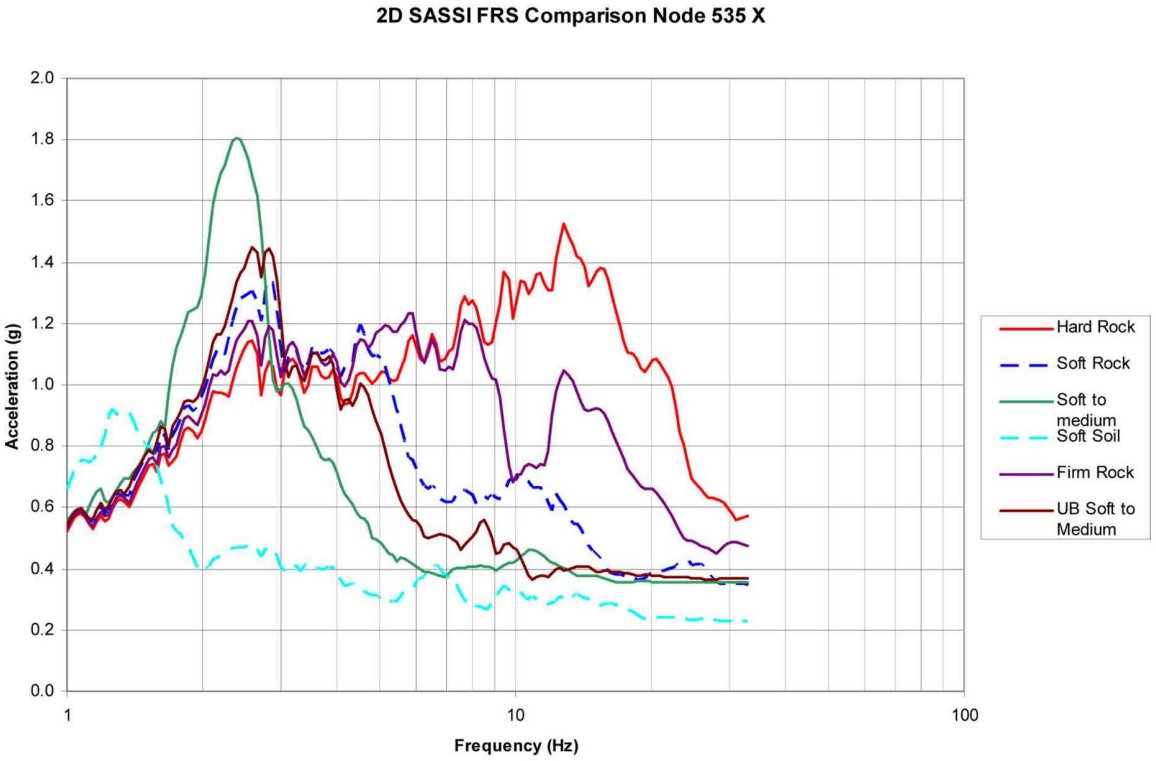

Figure 3G.3-10
2D SASSI FRS – Node 535 X (CIS El. 134.3′)

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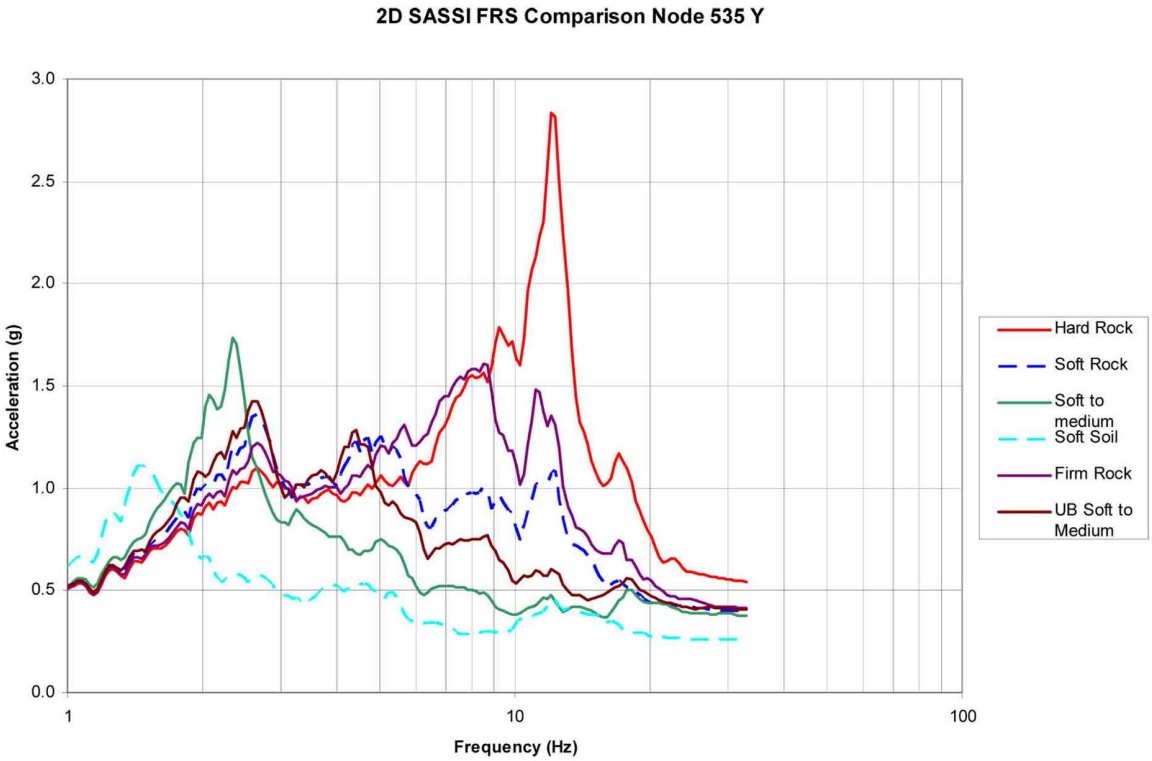

Figure 3G.3-11
2D SASSI FRS – Node 535 Y (CIS El. 134.3′)

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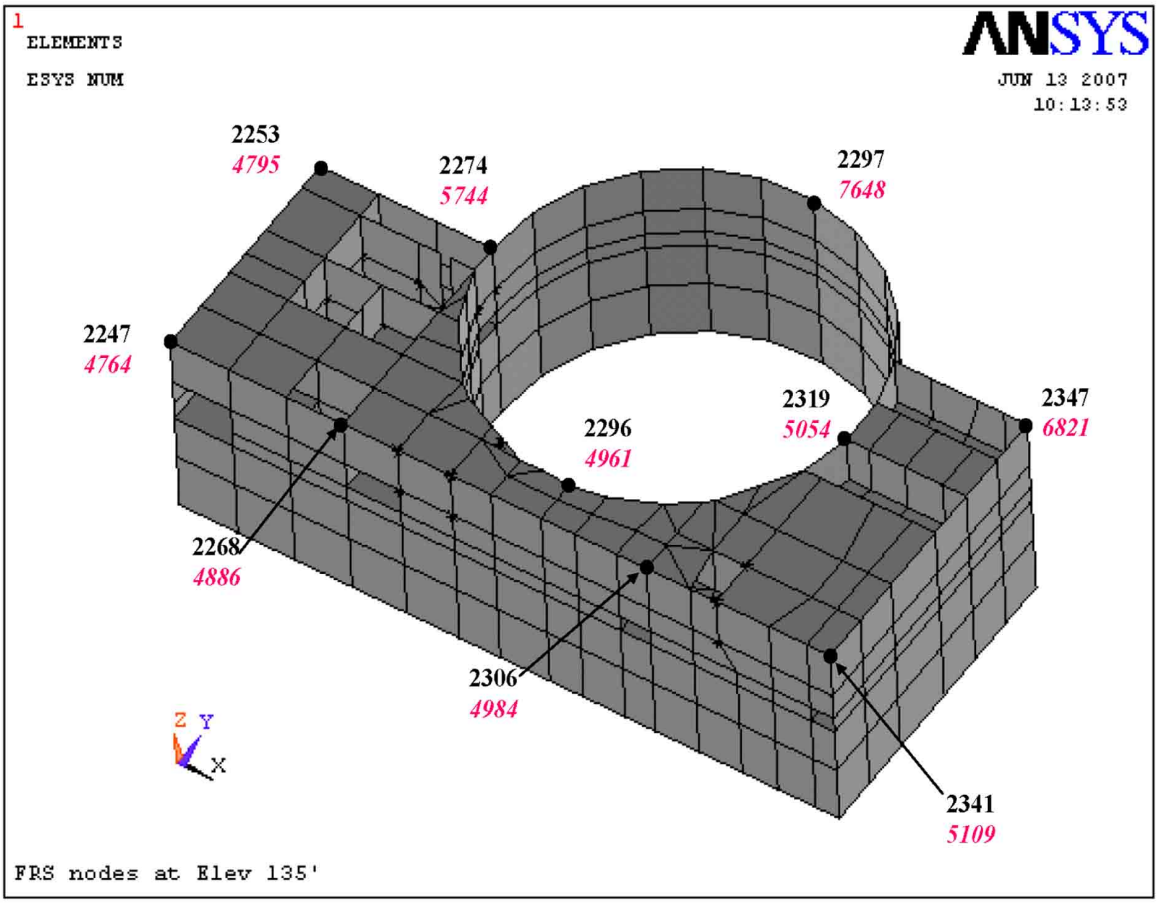

Figure 3G.4-1
Auxiliary Shield Building "Rigid" Nodes at El. 135′

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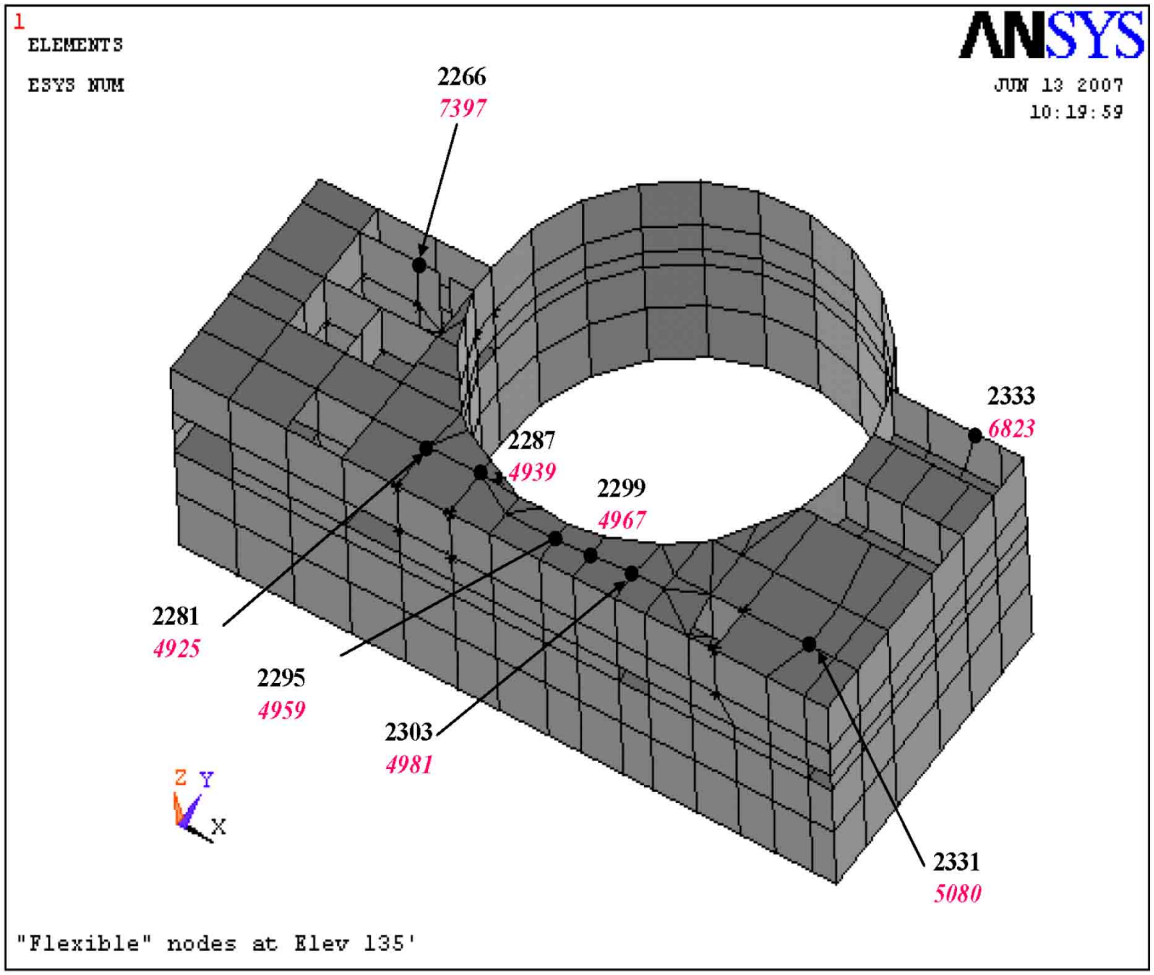

Figure 3G.4-2
Auxiliary Shield Building "Flexible" Nodes at El. 135'

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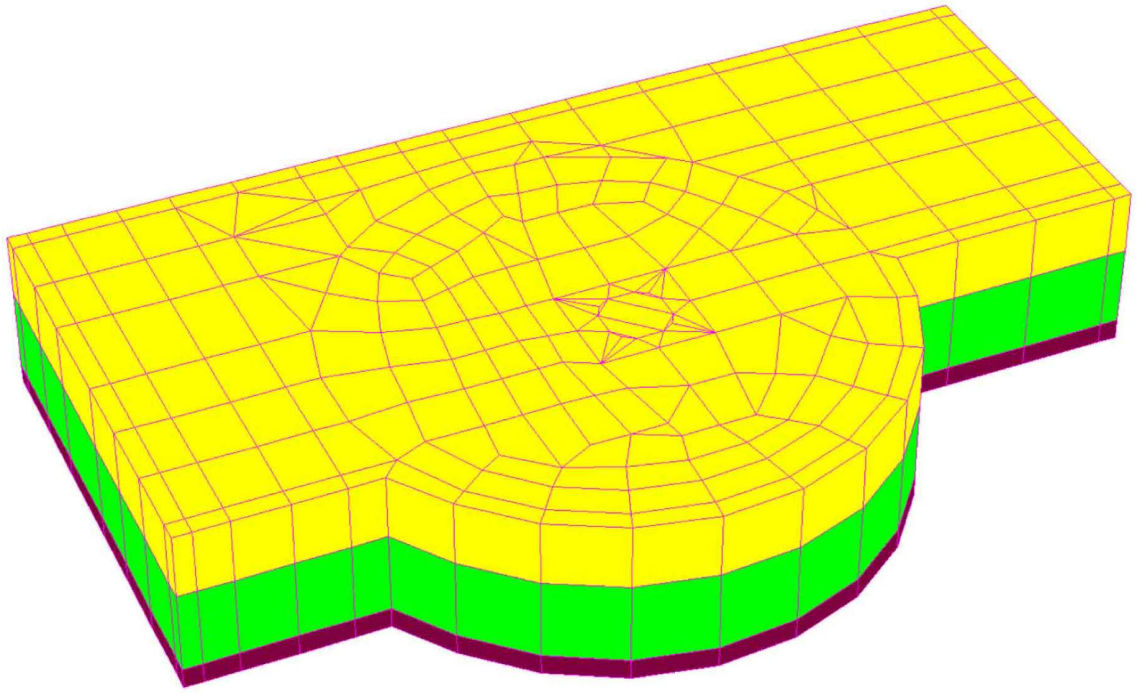

Figure 3G.4-3
Excavated Soil

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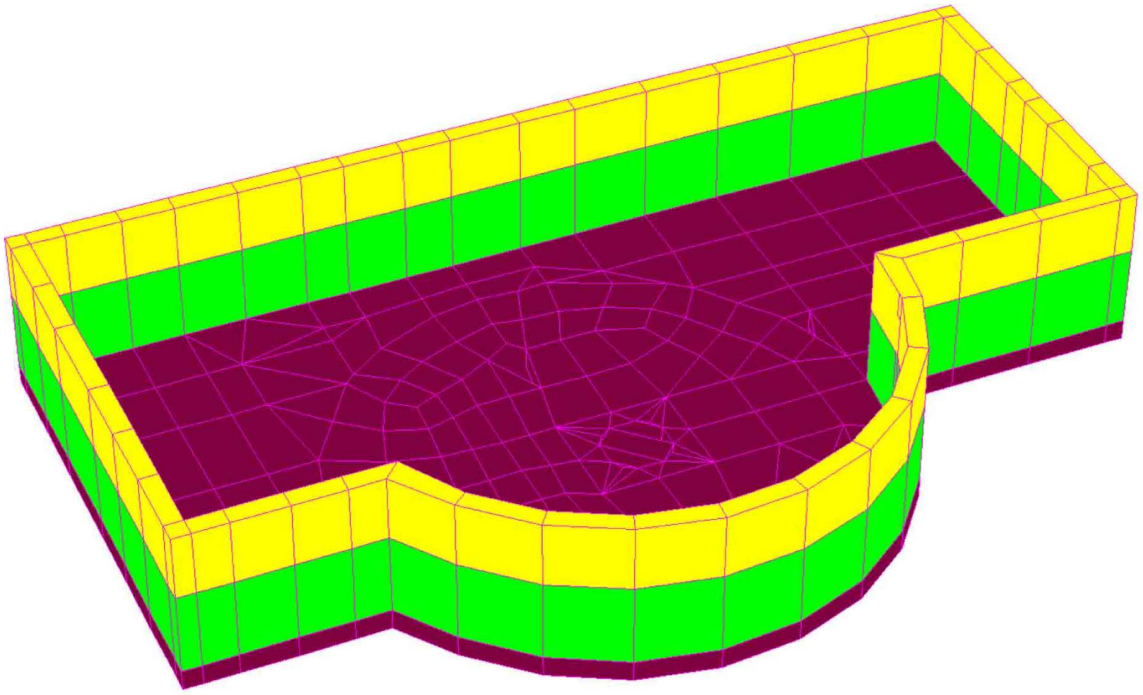

Figure 3G.4-4
Additional Elements for Soil Pressure Calculations

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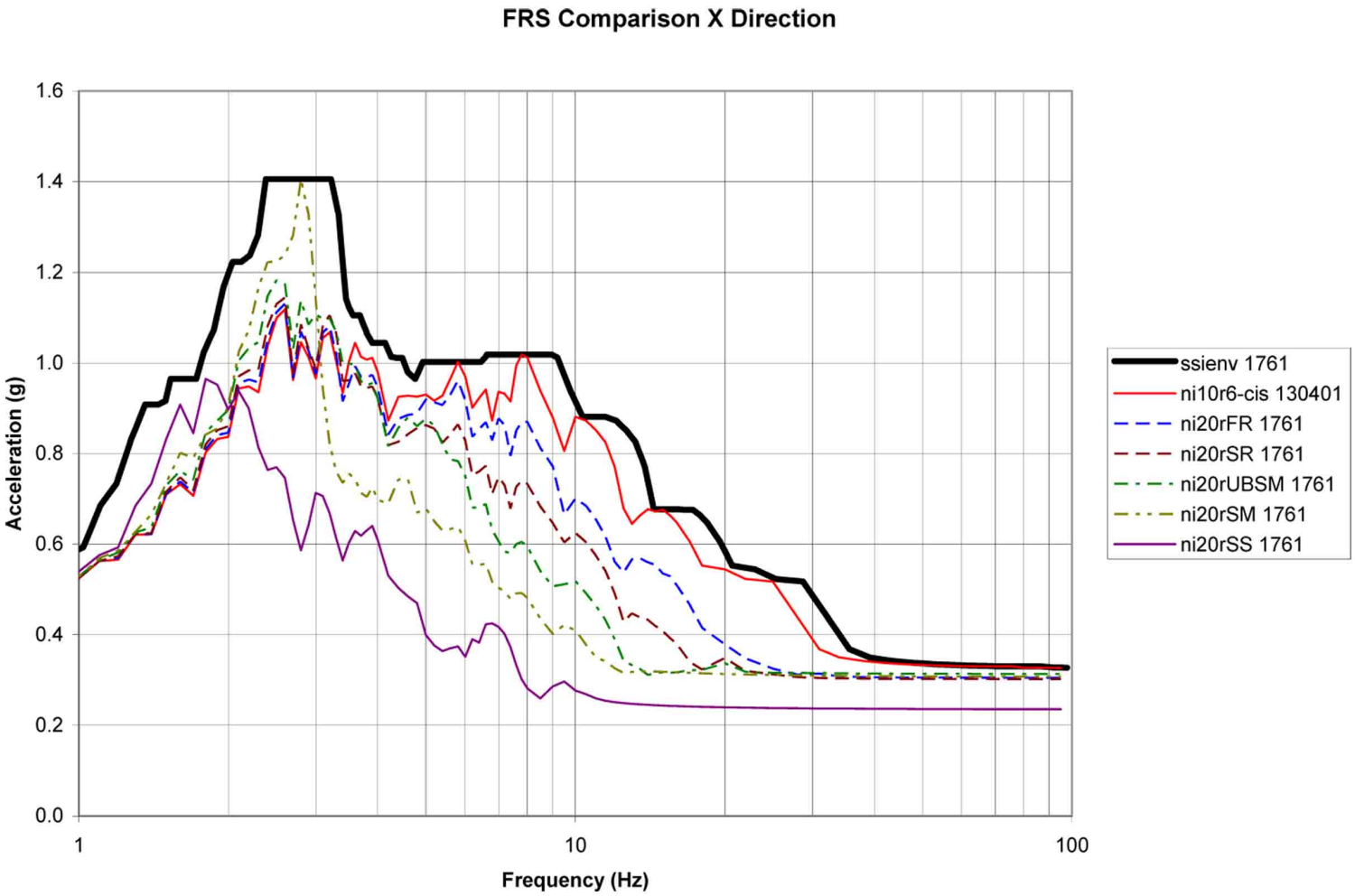

[***Figure 3G.4-5X***
X Direction FRS for Node 130401 (NI10) or 1761 (NI20)
CIS at Reactor Vessel Support Elevation of 100']*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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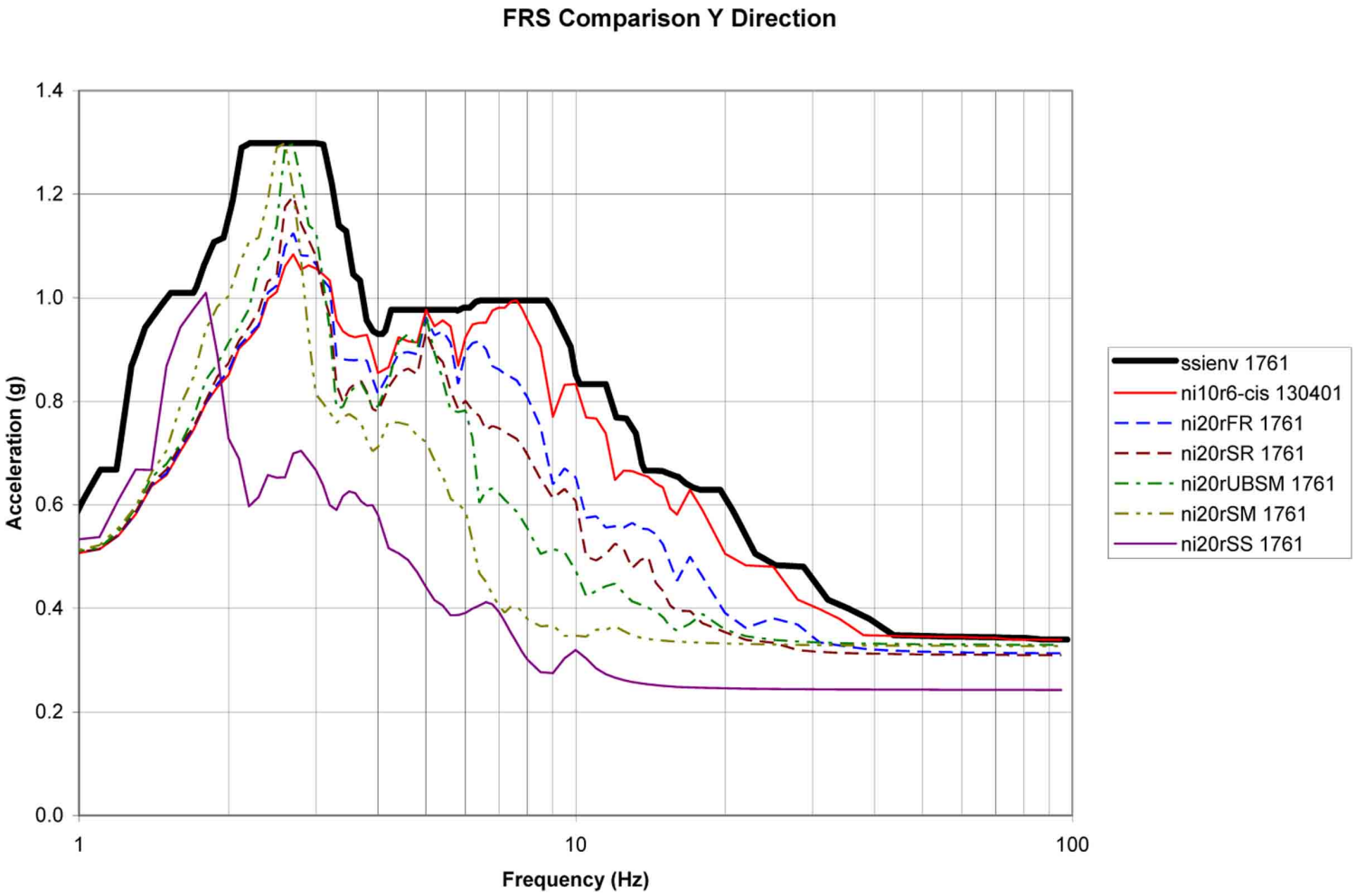

Figure 3G.4-5Y
Y Direction FRS for Node 130401 (NI10) or 1761 (NI20)
CIS at Reactor Vessel Support Elevation of 100'\]*

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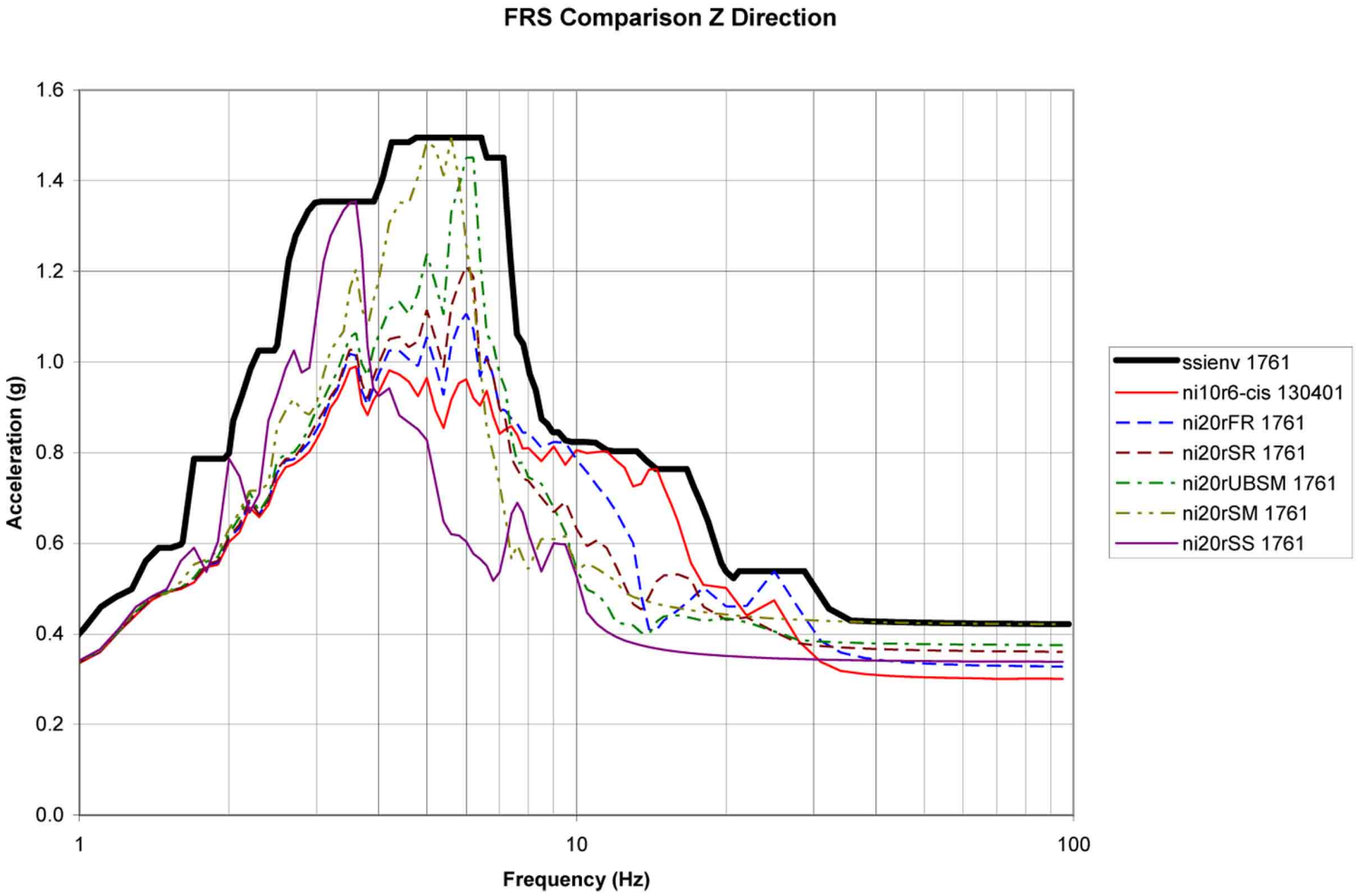

Figure 3G.4-5Z
Z Direction FRS for Node 130401 (NI10) or 1761 (NI20)
CIS at Reactor Vessel Support Elevation of 100'*

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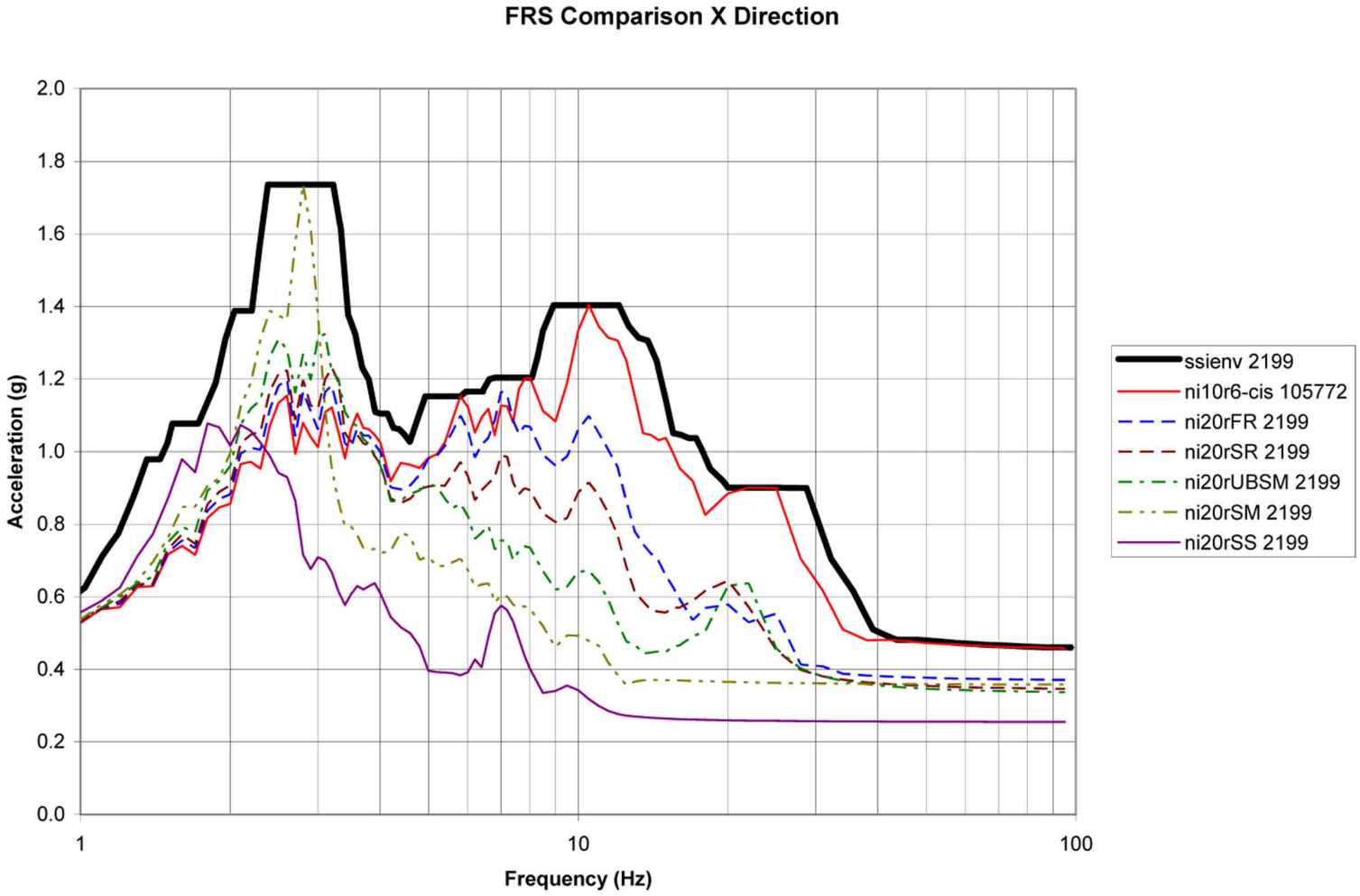

Figure 3G.4-6X
X Direction FRS for Node 105772 (NI10) or 2199 (NI20)
CIS at Operating Deck Elevation 134.25'*

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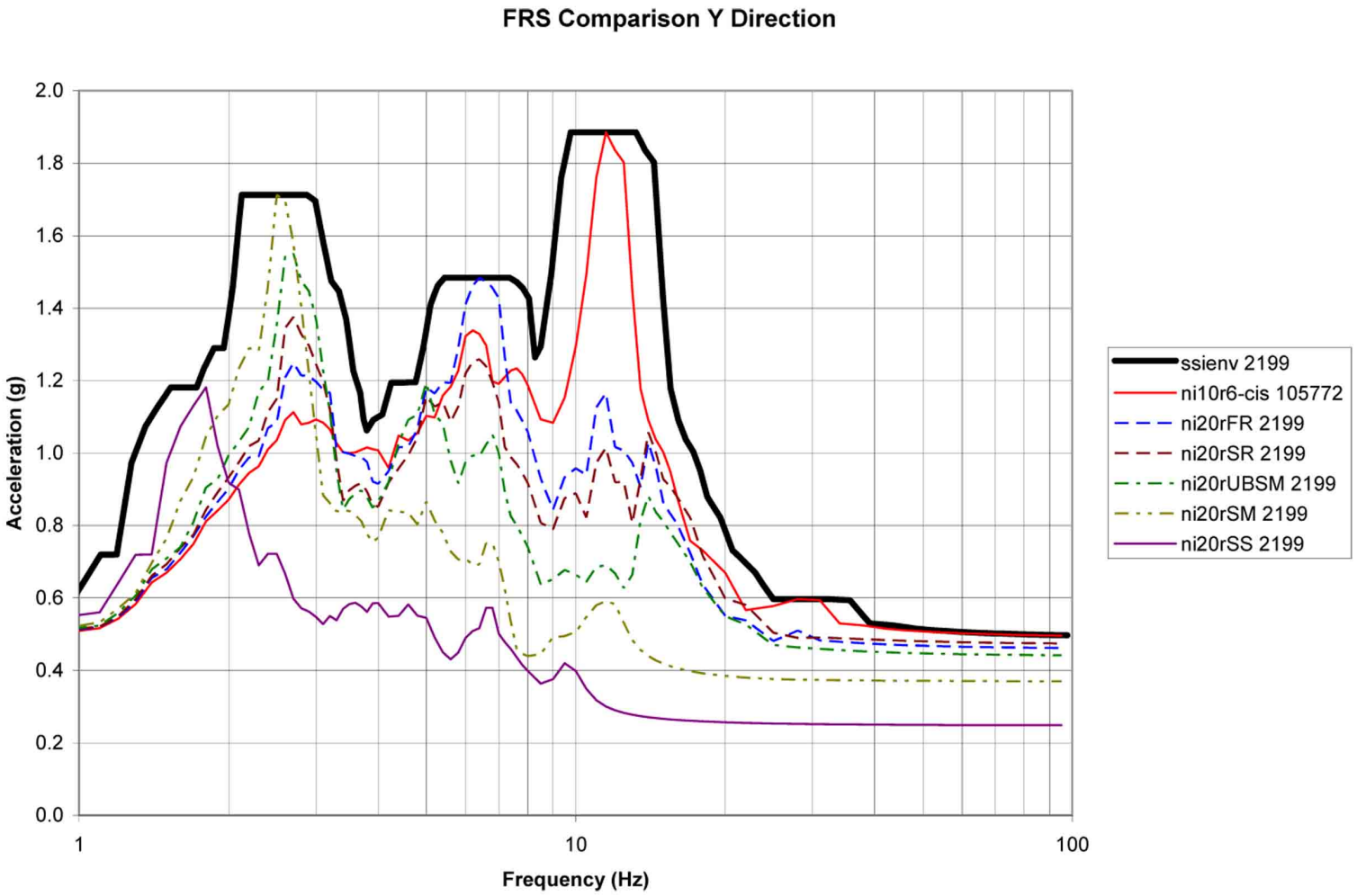

Figure 3G.4-6Y
Y Direction FRS for Node 105772 (NI10) or 2199 (NI20)
CIS at Operating Deck Elevation 134.25'*

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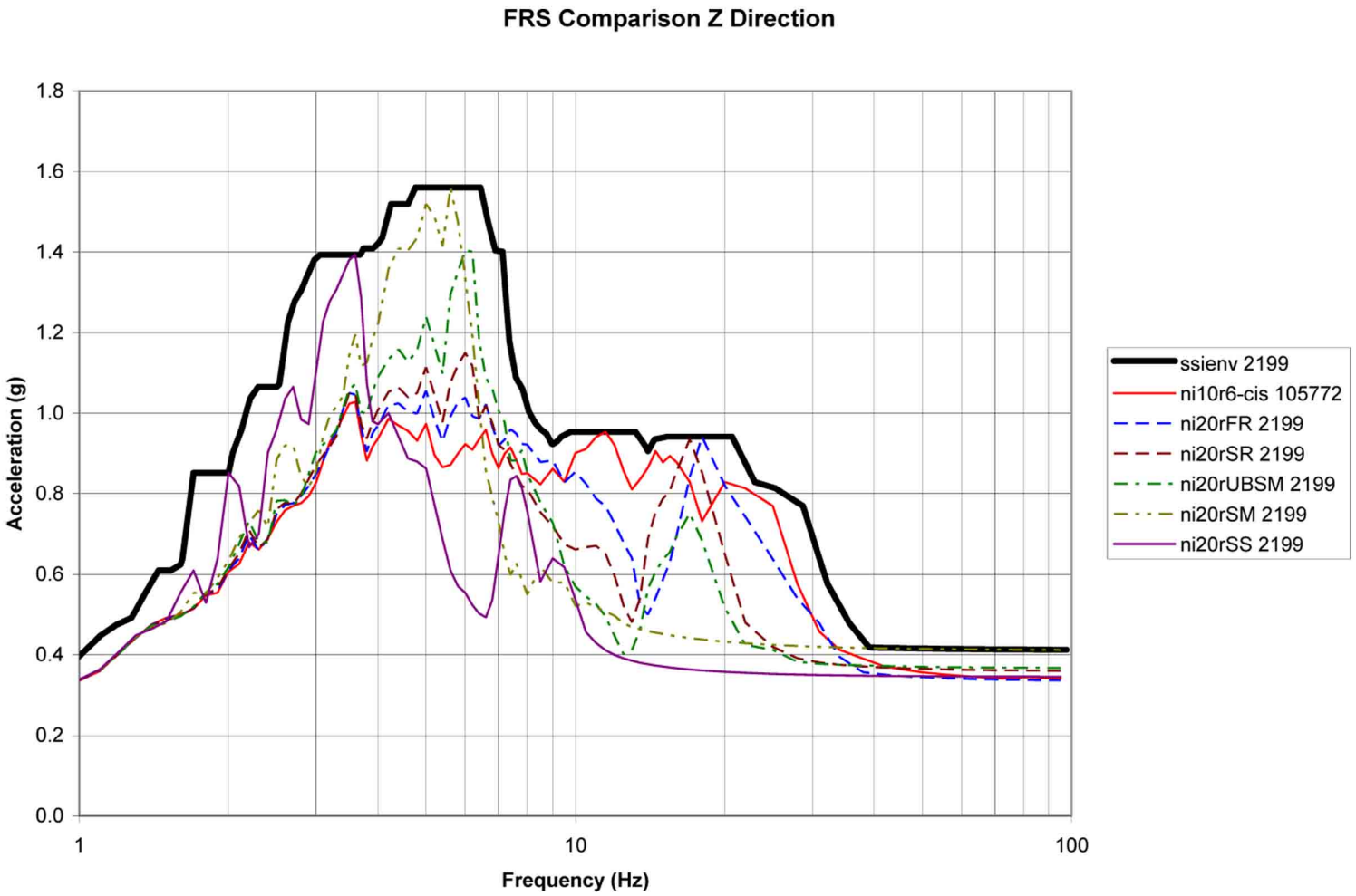

Figure 3G.4-6Z
Z Direction FRS for Node 105772 (NI10) or 2199 (NI20)
CIS at Operating Deck Elevation 134.25'*

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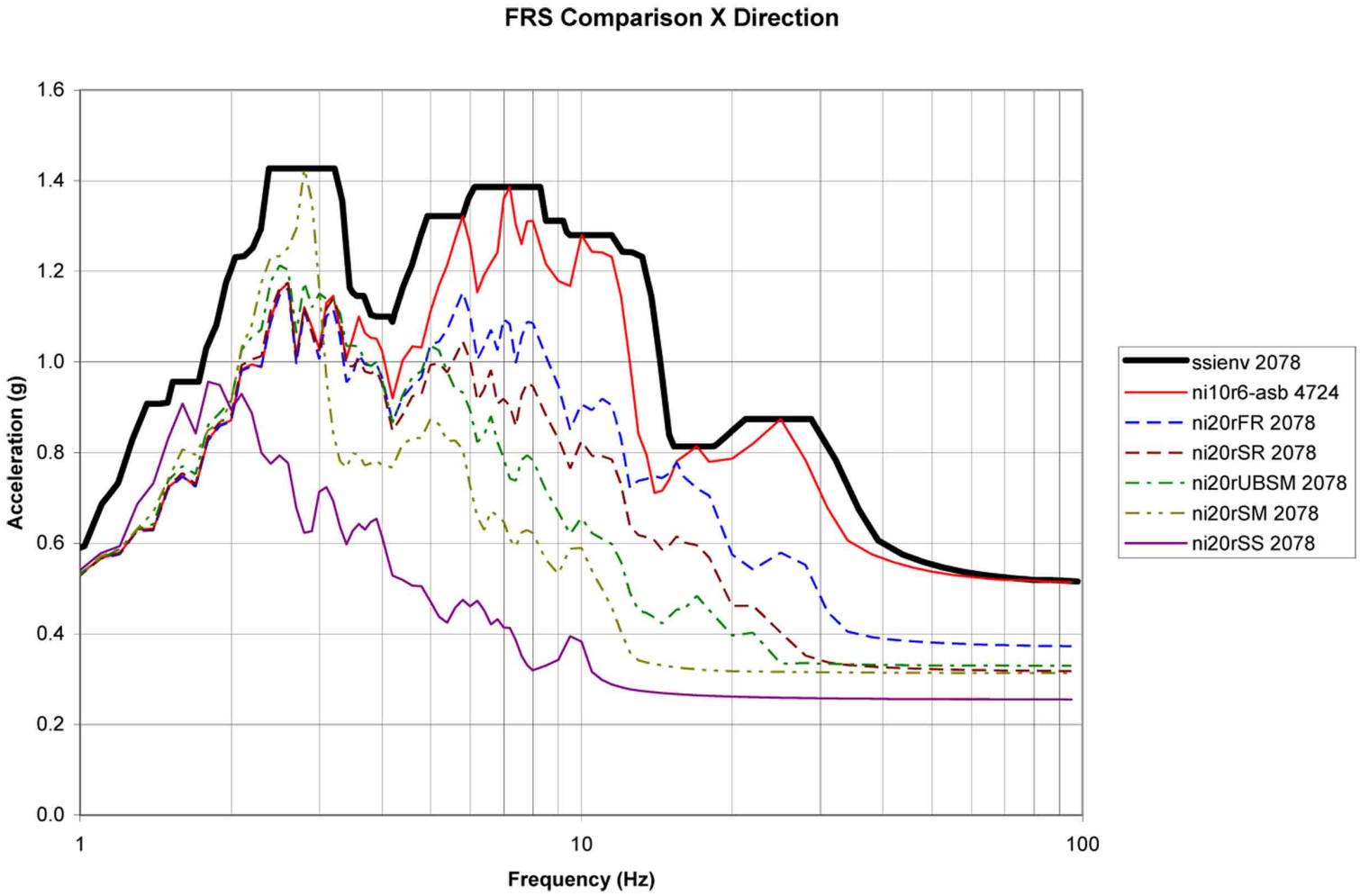

Figure 3G.4-7X
X Direction FRS for Node 4724 (NI10) or 2078 (NI20)
ASB Control Room Side Elevation 116.50'\]*

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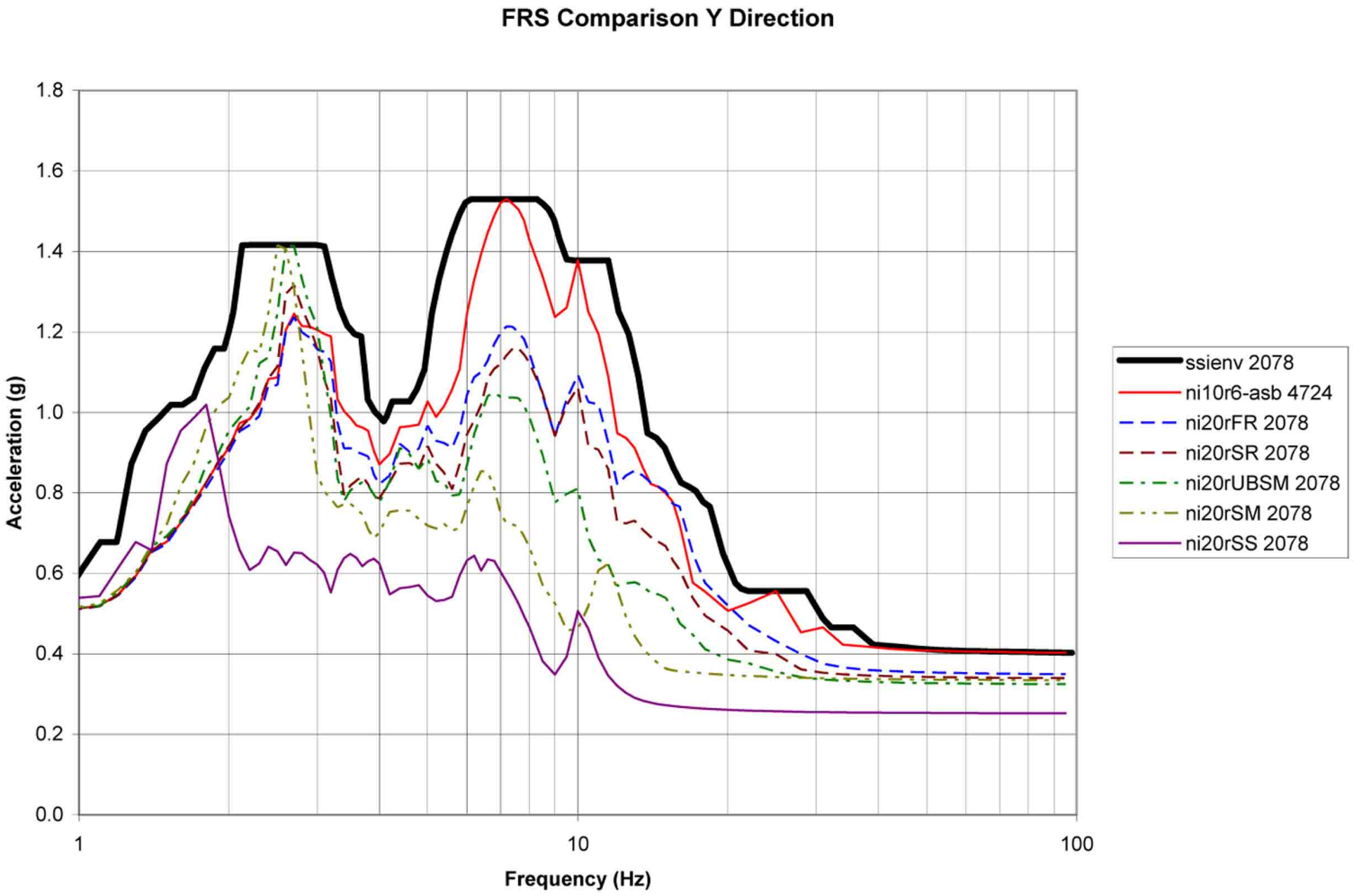

Figure 3G.4-7Y
Y Direction FRS for Node 4724 (NI10) or 2078 (NI20)
ASB Control Room Side Elevation 116.50'*]

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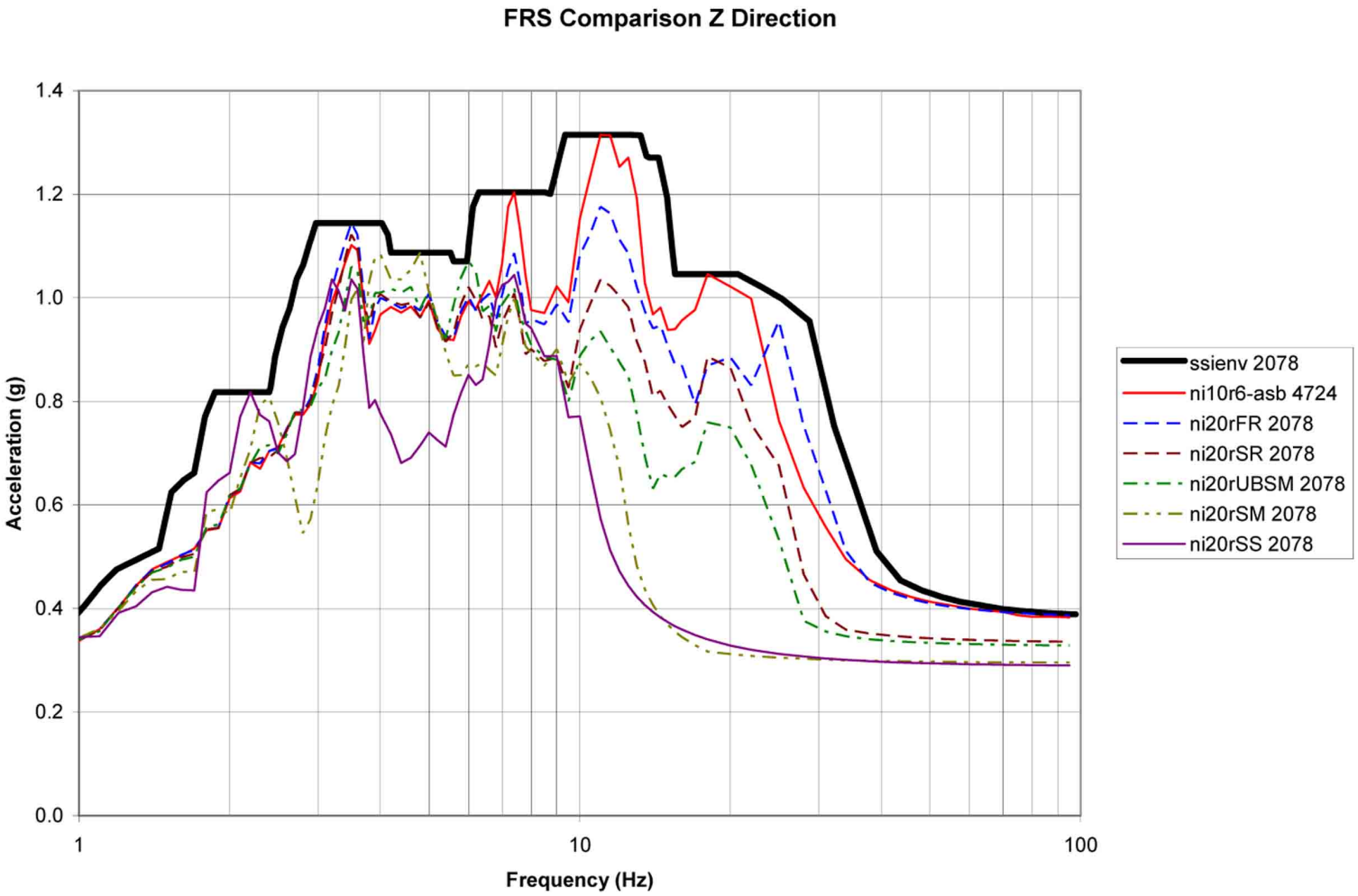

[Figure 3G.4-7Z
Z Direction FRS for Node 4724 (NI10) or 2078 (NI20)
ASB Control Room Side Elevation 116.50']*

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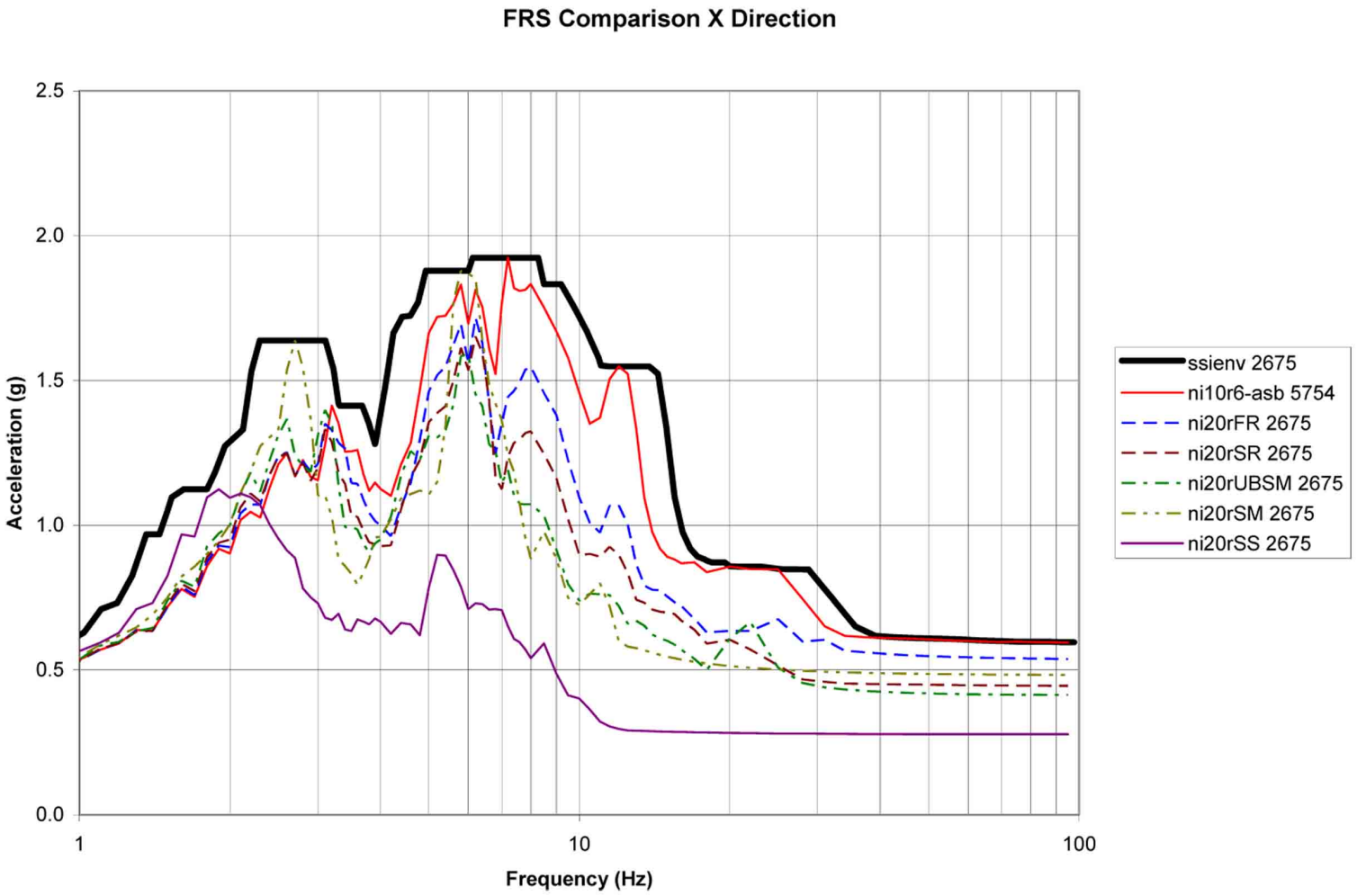

Figure 3G.4-8X
X Direction FRS for Node 5754 (NI10) or 2675 (NI20)
ASB Fuel Building Roof Elevation 179.19']*

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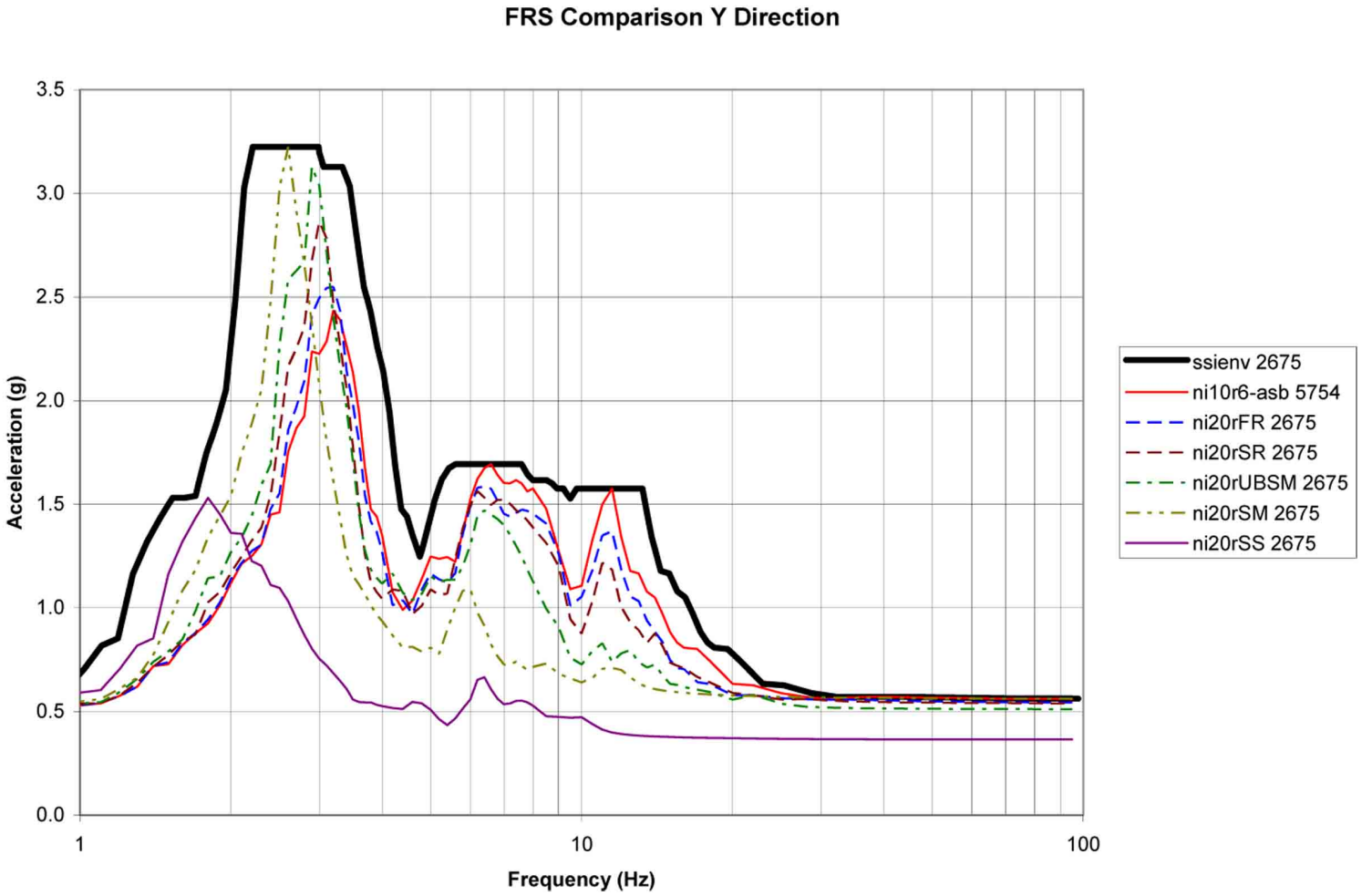

Figure 3G.4-8Y
Y Direction FRS for Node 5754 (NI10) or 2675 (NI20)
ASB Fuel Building Roof Elevation 179.19'-0"]*

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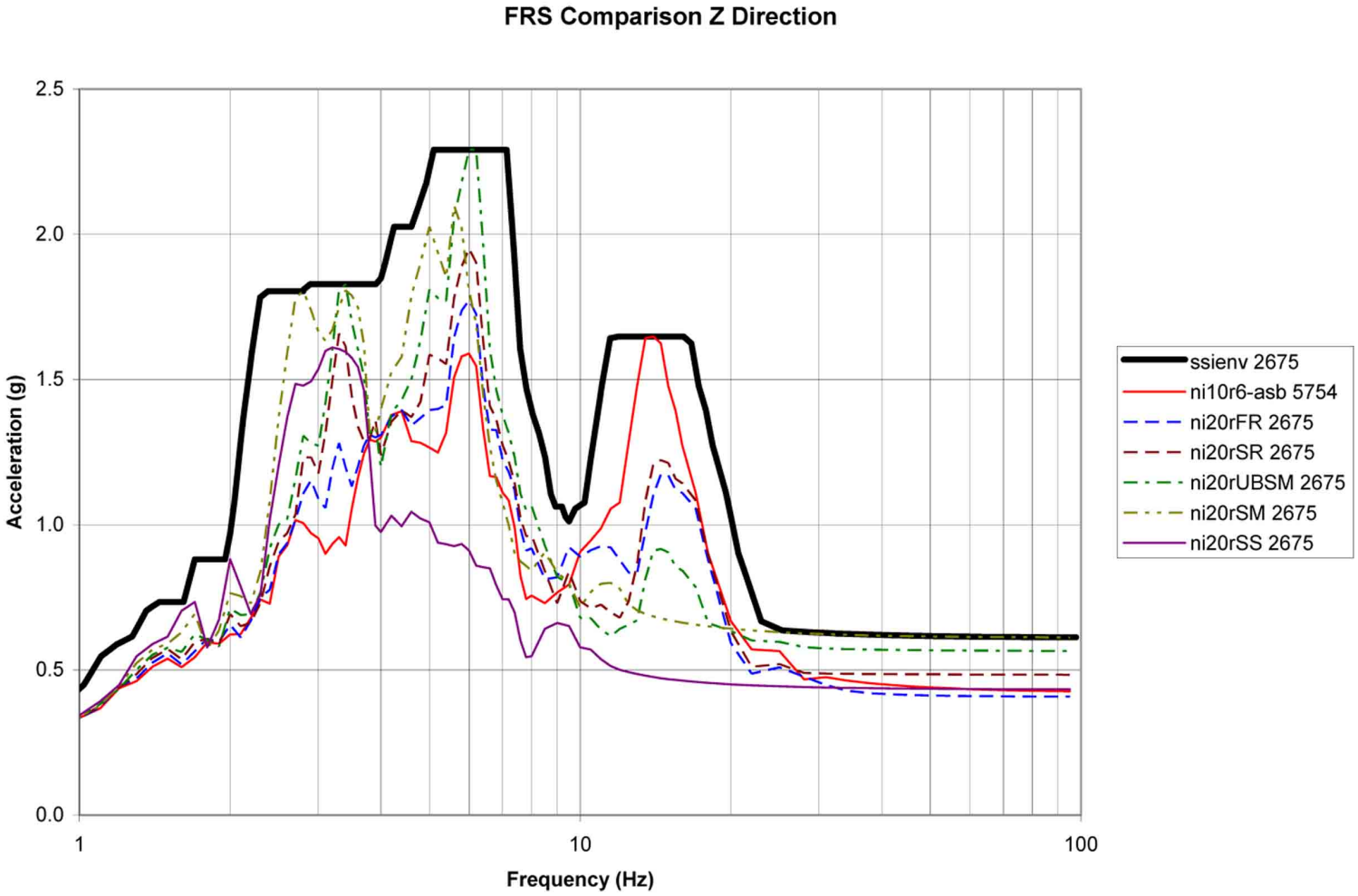

[Figure 3G.4-8Z
Z Direction FRS for Node 5754 (NI10) or 2675 (NI20)
ASB Fuel Building Roof Elevation 179.19']*

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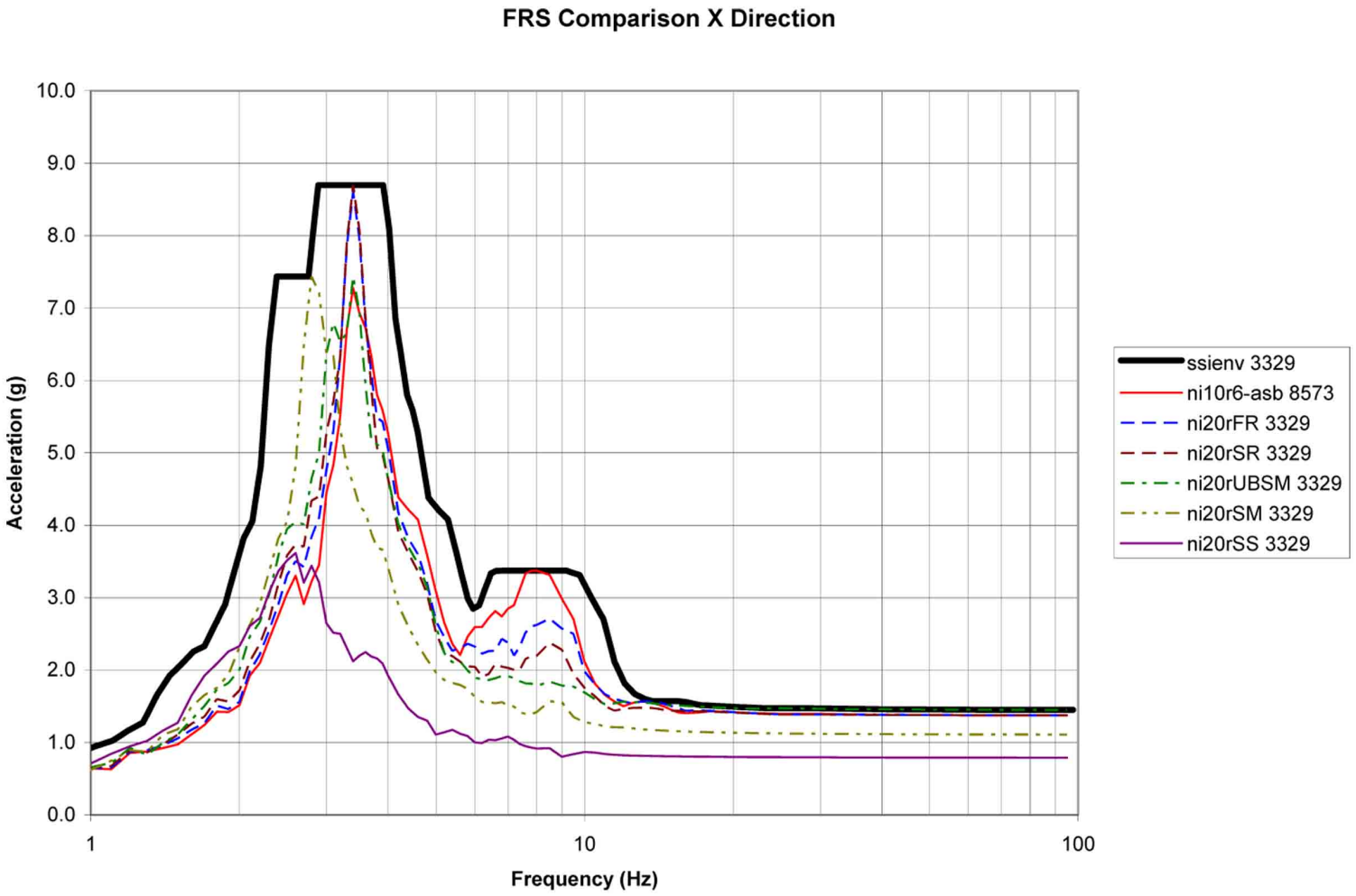

Figure 3G.4-9X
X Direction FRS for Node 8573 (NI10) or 3329 (NI20)
ASB Shield Building Roof Elevation 327.41'\]*

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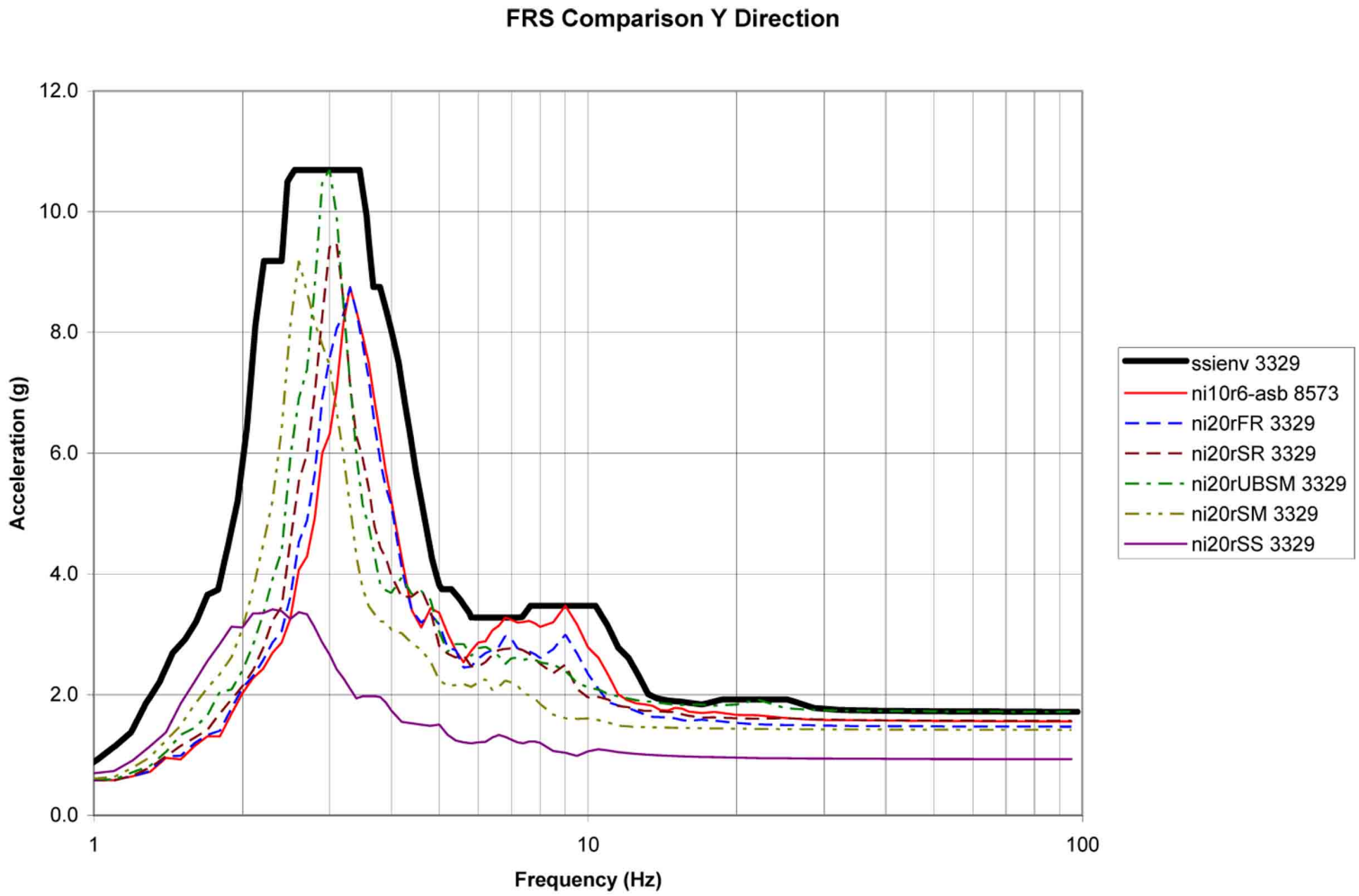

Figure 3G.4-9Y
Y Direction FRS for Node 8573 (NI10) or 3329 (NI20)
ASB Shield Building Roof Elevation 327.41'*]

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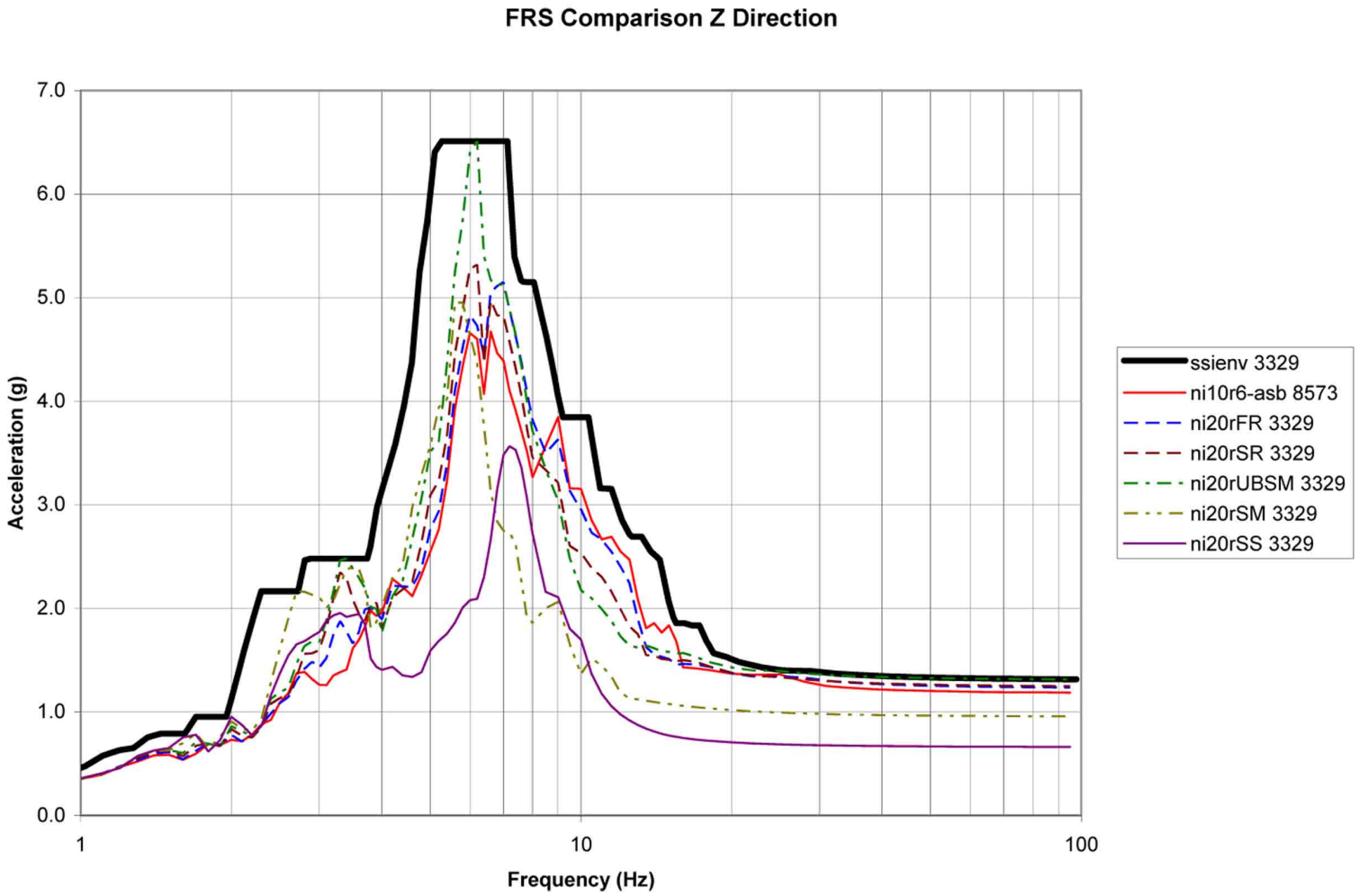

Figure 3G.4-9Z
Z Direction FRS for Node 8573 (NI10) or 3329 (NI20)
ASB Shield Building Roof Elevation 327.41'*

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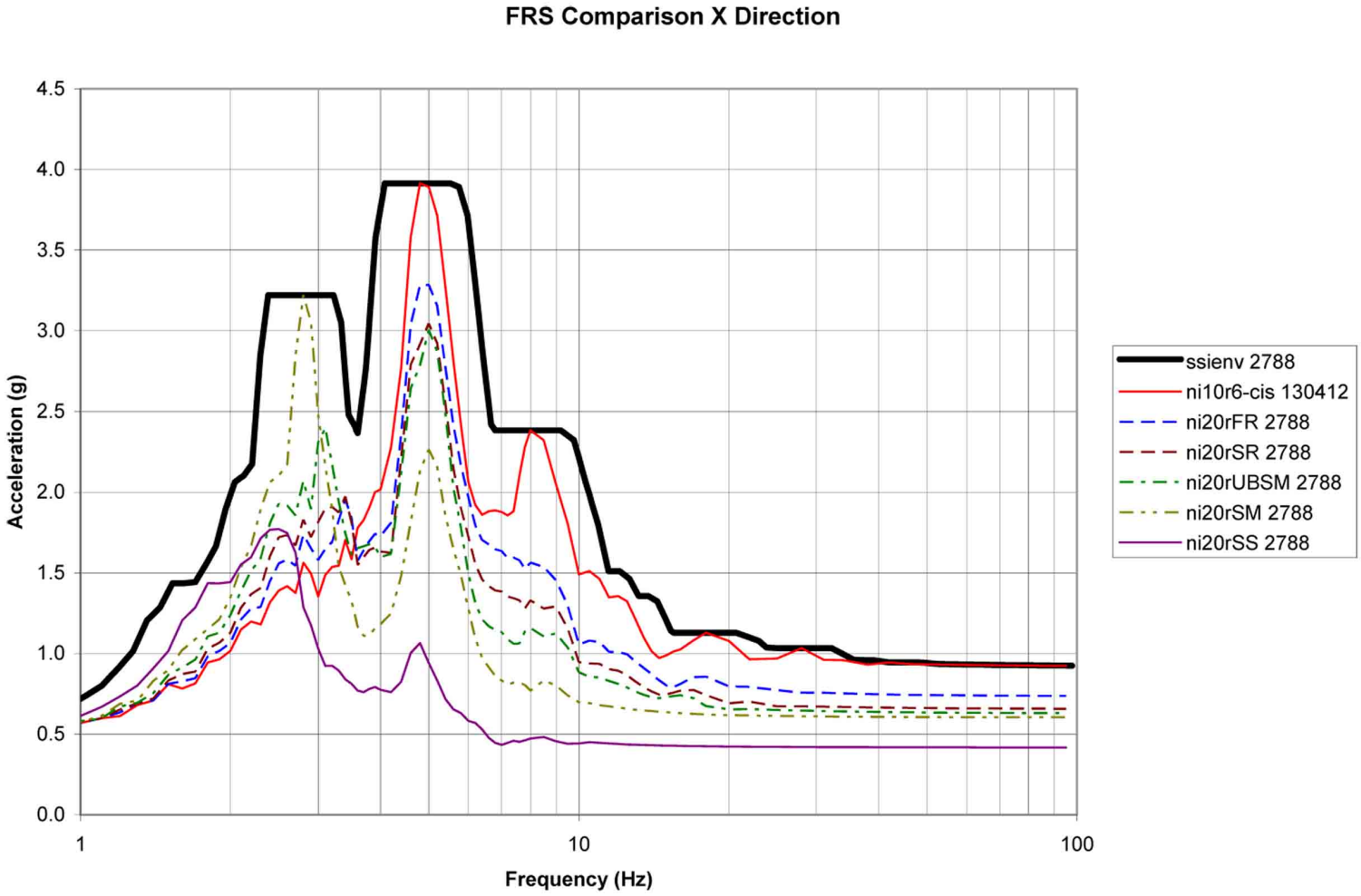

Figure 3G.4-10X
X Direction FRS for Node 130412 (NI10) or 2788 (NI20)
SCV Near Polar Crane Elevation 224.00'\]*

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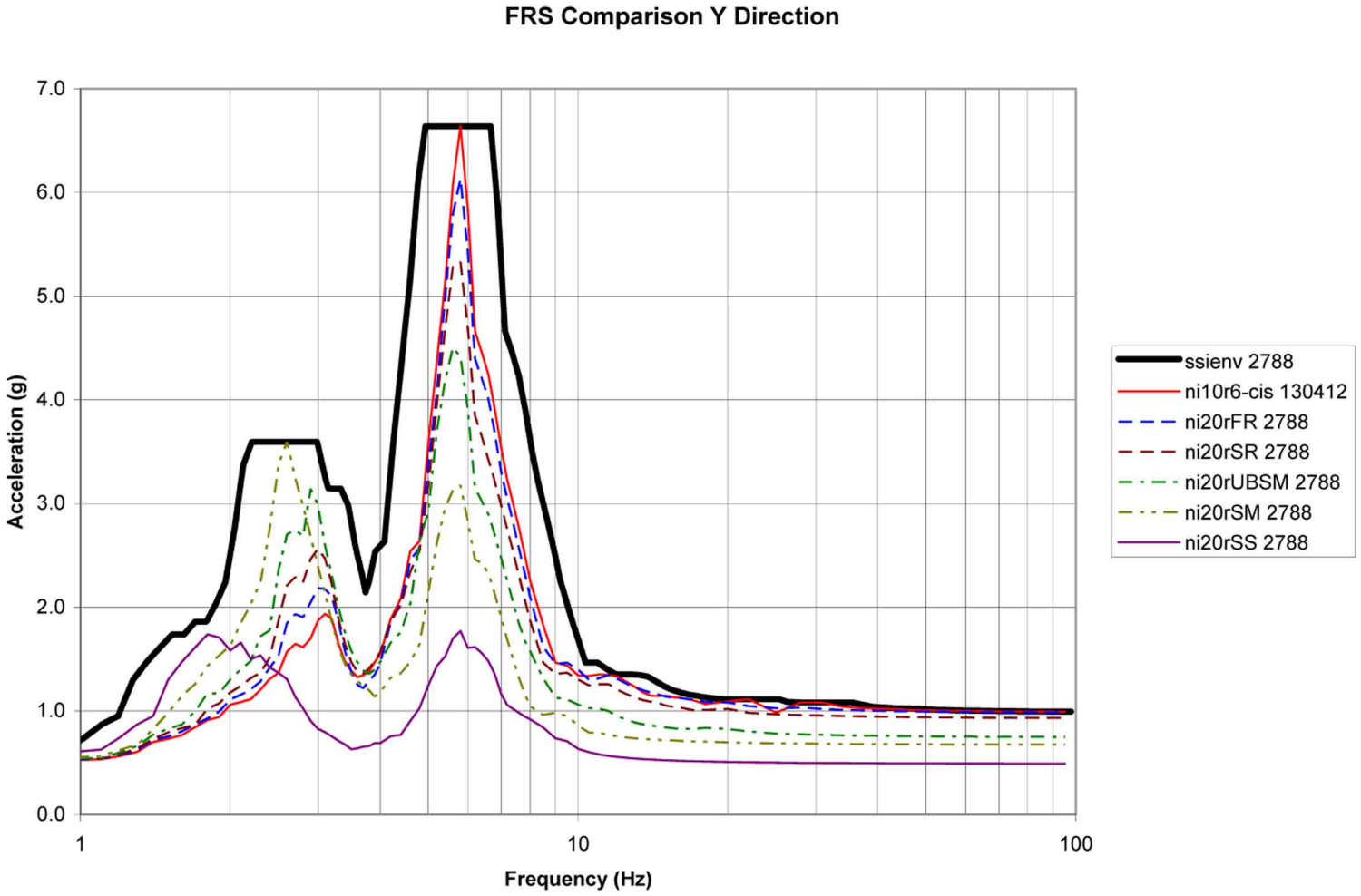

Figure 3G.4-10Y
Y Direction FRS for Node 130412 (NI10) or 2788 (NI20)
SCV Near Polar Crane Elevation 224.00'*]

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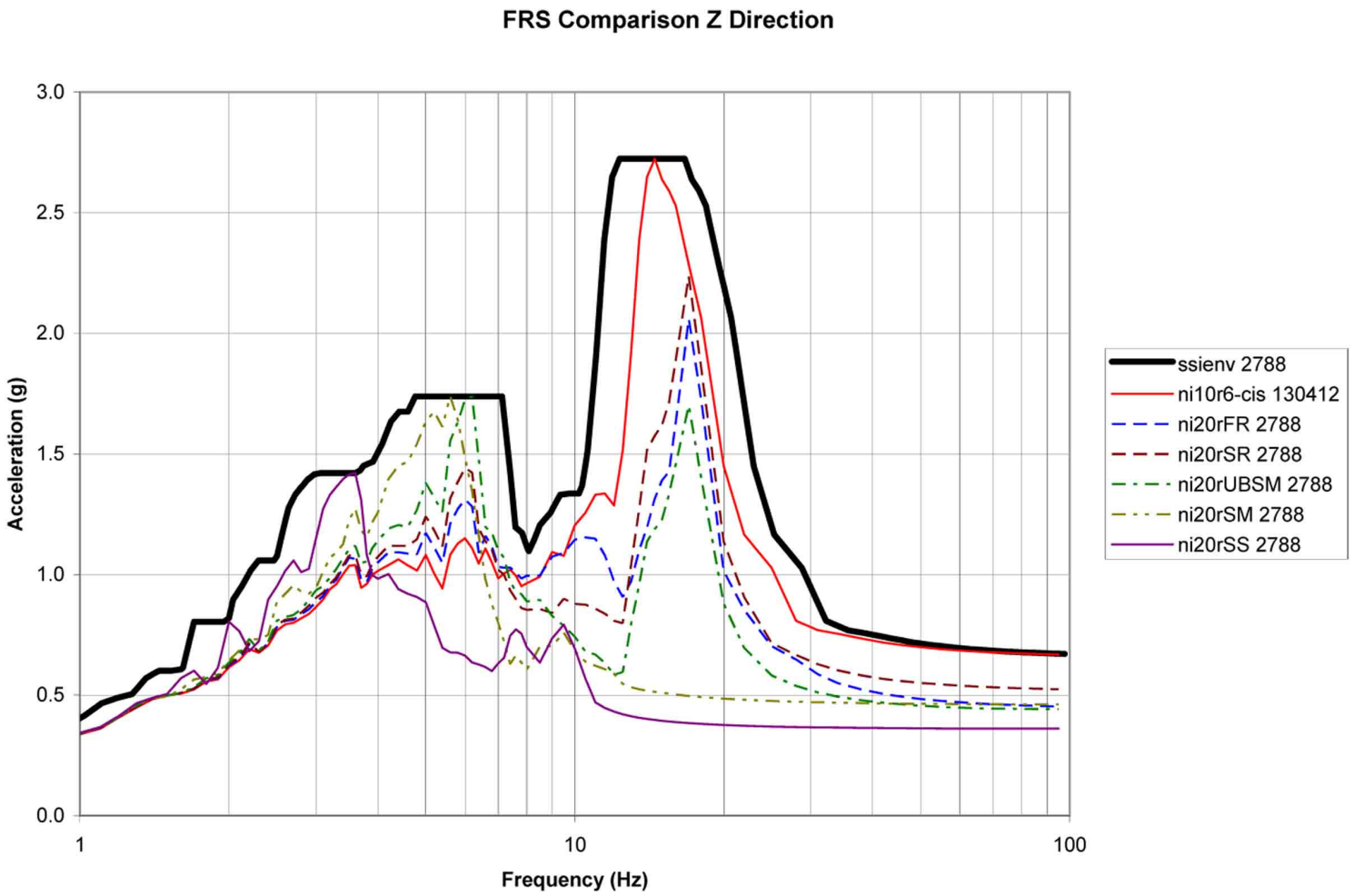

[Figure 3G.4-10Z
Z Direction FRS for Node 130412 (NI10) or 2788 (NI20)
SCV Near Polar Crane Elevation 224.00']*

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Appendix 3H Auxiliary and Shield Building Critical Sections

3H.1 Introduction

[*This appendix summarizes the structural design and analysis of structures identified as "Critical Sections" in the auxiliary and shield buildings. The design summaries include the following information:*

- *Description of buildings*
- *Governing codes and regulations*
- *Structural loads and load combinations*
- *Global analyses*
- *Structural design of critical structural elements*

Subsections 3H.2 through 3H.5 include a general description of the auxiliary building and shield building, a summary of the design criteria and the global analyses. The 3H.5 figures referenced in the descriptions show the structural designs for critical sections which are identified in subsection 3H.5 and shown in Figure 3H.5-1 (3 sheets). The exact locations of the critical sections related to the shield building cylinder are shown in Figure 3H.5-16. Representative design details are provided for these structures in subsection 3H.5.]*

3H.2 Description of Auxiliary and Shield Buildings

3H.2.1 Description of Auxiliary Building

[*The auxiliary building is a reinforced concrete structure. The auxiliary building is one of the three buildings that make up the nuclear island and shares a common basemat with the containment building and the shield building. The auxiliary building general layout is shown in Figure 3H.2-1. It is a C-shaped section of the nuclear island that wraps around approximately half of the circumference of the shield building. The building dimensions are shown on key structural dimension drawings, Figure 3.7.2-12.*

The auxiliary building is divided into six areas, which are identified in Figure 3H.2-1. It is a 5-story building; three stories are located above grade and two are located below grade. Areas 1 and 2 (Figure 3H.2-1) have five floors, including two floors below grade level. The lowest floor at elevation 66'-6" is used exclusively for housing battery racks. The next higher floor, at elevation 82'-6", also has battery racks and some electrical equipment. The floor at the grade level, elevation 100'-0", has electrical penetration areas, a remote shutdown workstation room, motor control centers, and some Division A and Division C equipment. The main control room is situated on the floor at elevation 117'-6", which also has rooms for the main steam and feedwater lines. The floor at elevation 135'-3" carries air filtration and air handling units, chiller pump equipment, and other mechanical and electrical equipment. The roof for areas 1 and 2 is at elevation 153'-0".

Areas 3 and 4 of the auxiliary building are the areas east of the containment shield building. Valve and piping areas, and some mechanical equipment, are located in the basement floor at elevation 66'-6". The floor at elevation 82'-6" has a piping penetration area, a radiation chemistry laboratory, makeup pumps, and other mechanical equipment. The floor at grade level elevation 100'-0" has an electrical penetration room, a staging area for the equipment hatch, and the access opening to the annex building. The electrical penetration area, trip switchgears, and other electrical equipment occupy most of the floor at elevation 117'-6". The floor at elevation 135'-3" is used for the

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storage of main control room air cylinders and provides access to the annex building. The roof for these areas is at elevation 160'-6".

Areas 5 and 6 include facilities for storage and handling of new and spent fuel. The spent fuel pool, fuel transfer canal, and cask loading and cask washdown pits have concrete walls and floors. They are lined on the inside surface with stainless steel plate for leak prevention. Interior structural walls and major floors are constructed using concrete filled steel plate modules, steel form modules, or reinforced concrete. The new fuel storage area is a separate reinforced concrete pit providing temporary dry storage for the new fuel assemblies. A 150-ton cask handling crane travels in the east-west direction. The location and travel of this crane prevents the crane from carrying loads over the spent fuel pool to preclude them from falling into the spent fuel pool. Mechanical equipment is also located in this area for spent fuel cooling, residual heat removal, and liquid waste processing. This equipment is generally nonsafety-related.]*

3H.2.2 Description of Shield Building

The shield building is the structure and annulus area that surrounds the containment building. It shares a common basemat with the containment building and the auxiliary building. The shield building uses concrete-filled steel plate construction (SC) as well as reinforced concrete (RC) structure. The figures in Section 1.2 show the layout of the shield building and its interface with the other buildings of the nuclear island.

Figure 3.8.4-5 shows the following significant features and the principal systems and components of the shield building:

- Shield building cylindrical structure
- Shield building roof structure
- RC/SC connections
- Air inlets and tension ring
- Passive containment cooling system (PCS) water storage tank (PCCWST)
- Knuckle region (connection to exterior wall of PCCWST)
- Compression ring (connection to interior wall of PCCWST)

The overall configuration of the shield building is established from functional requirements related to radiation shielding, missile barrier, passive containment cooling, tornado, and seismic event protection. These functional requirements led to establishing the design based on two primary design codes used for nuclear plant structures: 1) ACI 349 for reinforced concrete design, and 2) ANSI/AISC N690 for structural steel design.

The shield building SC walls are anchored to the RC basemat and shield building RC wall by mechanical connections. These RC-to-SC connections are also used in the other regions of the shield building, including:

- Auxiliary building RC roof connection to the shield building SC wall
- Auxiliary building RC wall connection to shield building SC wall
- Tension ring connection to the shield building RC roof

The connections provide for the direct transfer of forces from the RC reinforcing steel to the SC liner plates.

The cylindrical shield wall has an outside radius of 72.5 feet and a thickness of 36 inches. The cylindrical wall section that is a few feet below the auxiliary building roof line is a reinforced concrete (RC) structure. The section that is not protected by the auxiliary building is a steel concrete (SC) composite structure (see Figure 3H.5-16). The overall thickness of 36 inches is the same as the RC

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wall below. The concrete for the SC portion is standard concrete with compressive strength of 6000 psi. The SC portion is constructed with steel surface plates, which act as concrete reinforcement. The 0.75-inch tie bars are welded to the steel faceplates to develop composite behavior of the steel faceplates and concrete. The shear studs are welded to the inside surface of the steel plate. The tie bar spacing is reduced in the higher stress regions. A typical SC wall panel is shown in Figure 3H.5-13.

The tension ring is located at the interface of the shield building steel concrete composite air inlet structures and the shield building reinforced concrete roof. The top of the tension ring interfaces with the RC roof slab. The tension ring supports the roof girders that are located under the RC roof slab. The bottom of the tension ring is attached to the air inlets structure. The bottom of the air inlets structure is attached to the top of the cylindrical SC wall of the shield building. The connection of the tension ring to the roof is a mechanical connection design and is described above.

The primary function of the tension ring is to resist the thrust from the shield building roof. The air inlets structure is located directly below the tension ring and includes the air openings that provide for natural circulation of cooling air. Though its steel plates are connected to the concrete infill by studs and tie bars, the tension ring is conservatively designed as a hollow steel box girder. The concrete infill is credited only for out-of-plane shear transfer and for stability of the steel plates. The tension ring is designed to have high stiffness and to remain elastic under required load combinations.

The air inlets structure is a 4.5-foot-thick SC structure with through-wall openings for air flow. The air inlet openings consist of circular pipes at a downward inclination of 38 degrees from the vertical. Steel plates on each face, aligned with the inner and outer flanges of the tension ring, serve as primary reinforcement. The concrete infill is connected to the steel plates with tie bars and studs. The top of the air inlets structure is welded to the underside of the tension ring. The bottom of the air inlets structure is welded to the SC wall.

The shield building conical roof steel structure consists of 32 radial beams. Between each pair of radial beams there are circumferential beams. A steel plate is welded to the top flanges of each beam and forms a surface on which the concrete is placed. The steel structure forms a conical shell that spans the area from the compression ring to the tension ring.

The outside diameter of the PCS tank (passive containment cooling water storage tank) intersects with the shield building roof at the knuckle region. Outside of the PCS tank, the concrete roof slab thickness is 3 feet and at the bottom of the PCS tank, the concrete thickness is 2 feet. The wall from the PCS tank applies a load to the roof slab, and also provides stiffness and increases the strength of the roof in that region.

The inside diameter of the PCS tank intersects with the roof slab at the compression ring. The compression ring provides the compression support for the conical roof dome. It consists of a composite structure having a curved steel beam section, which supports the concrete roof directly above it. The inside wall of the PCS tank is located above the concrete roof. Studs are placed on the top flange of the steel girder to allow the steel and concrete sections to act as a composite unit. The curved girder is designed to provide support for the steel structure during construction and during the initial placement of the concrete roof before the concrete has hardened sufficiently.

The PCS tank sits on top of the shield building roof. It is supported by and acts integrally with the conical roof. The inside surface has a liner that functions to provide leak protection, but is not required to provide structural strength to the structure. Leak chase channels are provided over the liner welds. The top elevation of the water inside the tank for the PCS has sufficient freeboard to preclude impact on the roof during the SSE.

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3H.3 Design Criteria

[*The auxiliary and shield building structures are reinforced concrete structures, structural modules, and horizontal concrete slabs supported by composite structural steel framing.*

- *Seismic forces are obtained from the response spectrum analysis of the three-dimensional finite element analysis models as described in subsection 3H.4. The shear wall and floor slab design also considers out-of-plane bending and shear forces due to loading, such as live load, dead load, seismic, lateral earth pressure, hydrostatic, hydrodynamic, and wind pressure.*

- *The shield building roof and the passive containment cooling water storage tank are analyzed using three-dimensional finite element models with the ANSYS computer code*]* as described in subsection 3.8.4.4.1. [*Loads and load combinations include construction, dead, live, thermal, wind, and seismic. The response spectrum analysis of the nuclear island is supplemented by equivalent static acceleration analysis of a more detailed model of a quadrant of the shield building roof. The results from the more detailed analysis are used in the evaluation of the tension ring, air inlets, and radial beams. The seismic response of the water in the tank is analyzed in a separate analysis with seismic input defined by the floor response spectrum.*

- *The structural steel framing is used primarily to support the concrete slabs and roofs. Metal decking, supported by the steel framing, is used as form work for the concrete slabs and roofs.*

- *The finned floors for the main control room and the instrumentation and control room ceilings are designed as reinforced concrete slabs in accordance with American Concrete Institute standard ACI 349. The steel panels are designed and constructed in accordance with American Institute of Steel Construction Standard ANSI/AISC N690. For positive bending, the steel plate is in tension and the steel plate with fin stiffeners serves as the bottom reinforcement. For negative bending, compression is resisted by the stiffened plate and tension by top reinforcement in the concrete.*]*

3H.3.1 Governing Codes and Standards

[*The primary codes and standards used in the design of the auxiliary and shield buildings are listed below:*

- *ACI 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures" (refer to Subsection 3.8.4.5 for supplementary requirements and Subsection 3.8.4.4.1 for alternative requirements).*

- *ANSI/AISC N690-1994, "Specification for the Design, Fabrication and Erection of Safety-Related Steel Structures for Nuclear Facilities" (refer to subsection 3.8.4.5 for supplemental requirements). American Welding Society (AWS), Structural Welding Code - Steel, AWS D1.1-2000 provides an acceptable alternative for ANSI/AISC N690 weld requirements as described in Subsections 3.8.3.2 and 3.8.4.2.*]*

3H.3.2 Seismic Input

The SSE design response spectra are given in Figures 3.7.1-1 and 3.7.1-2. [*They are based on the Regulatory Guide 1.60 response spectra anchored to 0.30g, but are amplified at 25 Hertz to reflect larger high-frequency seismic energy content observed for eastern United States sites.*]* The nuclear island seismic analyses are summarized in Subsection 3.7.2.

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3H.3.3 Loads

[*The auxiliary and shield buildings are seismic Category I structures. The loads listed in the following subsections are used for the design of the building structures. All the listed loads are not necessarily applicable to all structures and their elements. Loads for which each structural element is designed are based on the conditions to which that particular structural element is potentially subjected.*]*

Dead Load (D):

[*The weight of all permanent construction and installations, including fixed equipment, is included as the dead load during its normal operating condition.*

The weight of minor equipment (not specifically included in the dead load), piping, cables and cable trays, ducts, and their supports was included as equivalent dead load (EDL). A minimum of 50 pounds per square foot (psf) was used as EDL. For floors with a significant number of small pieces of equipment, the total weight of miscellaneous small pieces of equipment, divided by the floor area of the room plus an additional 50 psf was used as the equivalent dead load.]*

Earth Pressure (H):

[*The static earth pressure acting on the structures during normal operation is considered in the design of exterior walls. The dynamic soil pressure, induced during a safe shutdown earthquake (SSE), is included as a seismic load.*]*

Live Loads (L):

[*The load imposed by the use and occupancy of the building is included as the live load. Live loads include floor area loads, laydown loads, fuel transfer casks, equipment handling loads, trucks, railroad vehicles, and similar items. The floor area live load is not applied on areas occupied by equipment whose weight is specifically included in the dead load. Live load is applicable on areas under equipment where access is provided, for instance, the floor under an elevated tank supported on legs.*

Floor loading diagrams are prepared for areas for component laydown. The diagrams show the location of major pieces of equipment and their foot-print loads or equivalent uniformly distributed loads.

The following live load items are considered in design:

A. Building floor loads

The following minimum values for live loads are used.

– *Structural platforms and gratings*	*100 psf*
– *Ground floors*	*250 psf*
– *All other elevated floors*	*200 psf* *(This load is reduced if the equivalent dead load for the floor is more than 50 psf. The sum of the live load and the equivalent dead load is 250 psf.)*

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B. Roof loads

The roof is designed for a uniform snow load of 63 psf calculated in accordance with ASCE 7-98. This corresponds to ground snow load of 75 psf, exposure factor of 1.0, thermal factor of 1.0, and an importance factor of 1.2.

C. Concentrated loads for the design of local members

– *Concentrated load on beams and girders (in load combinations that do not include seismic load)*	*5,000 pounds so applied as to maximize moment or shear. This load is not carried to columns or walls. It is not applied in areas where no heavy equipment will be located or transported, such as the access control areas.*
– *Concentrated load on slabs (considered with dead load only)*	*5,000 pounds so applied as to maximize moment or shear. This load is not carried to columns or walls. It is not applied in access control areas.*

In design reconciliation analysis, if actual loads are established to be lower than the above loads, the actual loads are used for reconciliation.

D. Temporary exterior wall surcharge

When applicable, a minimum surcharge outside and adjacent to subsurface wall of 250 psf is applied.

E. Construction loads

The additional construction loads produced by cranes, trucks, and the like, with their pickup loads, are considered. For steel beams supporting concrete floors, the weight of the wet concrete plus 100 psf uniform load and 5,000 pounds concentrated load, distributed near points of maximum shear and moment, is applied. A one-third increase in allowable stress is permitted.

Metal decking and precast concrete panels, used as formwork for concrete floors are designed for the wet weight of the concrete plus a construction live load of 20 psf uniform or 150 pounds concentrated. The deflection during normal operation is limited to span in inches divided by 180, or 0.75 inch, whichever is less.

F. Crane loads

The impact allowance for traveling crane supports and runway horizontal forces is in accordance with ANSI/AISC N690.

G. Elevator loads

The impact allowance used for the elevator supports is 100 percent, applied to design capacity and weight of car plus appurtenances, unless otherwise specified by the equipment supplier.

H. Equipment laydown and major maintenance

Floors are designed for planned refueling and maintenance activities as defined on equipment laydown drawings.]*

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Wind Load

[*The wind loads are as follows:*

- *Design wind (W)*

  For the design of the exterior walls, wind loads are applied in accordance with ASCE 7-98 with a basic wind speed of 145 mph. The importance factor is 1.15, and the exposure category is C. Wind loads are not combined with seismic loads.

- *Tornado load (W_t)*

  The exterior walls of the auxiliary and shield buildings are designed for tornado. A maximum wind speed of 300 mph (maximum rotational speed: 240 mph, maximum translational speed: 60 mph) is used to design the structures.]*

Seismic Loads (E_s)

[*The SSE (E_s) is used for evaluation of the structures of the auxiliary and shield buildings. E_s is defined as the loads generated by the SSE specified for the plant, including the associated hydrodynamic loads and dynamic incremental soil pressure.*]*

Operating Thermal Loads (T_o)

[*Normal thermal loads for the exterior walls and roofs are addressed in the design. These correspond to positive and negative linear temperature gradients with the inside surface at an average 70°F and the outside air temperature at -40°F and +115°F, respectively. These loads are considered for the seismic Category I structures in combination with the SSE also. All exterior walls of the nuclear island above grade not protected by adjacent buildings are designed for these thermal loads. The thermal gradient is also applied to the portion of the shield building between the upper annulus and the auxiliary building.*

Normal thermal loads for the passive containment cooling system (PCS) tank design are calculated based on the outside air temperature extremes specified for the safety-related design. The PCS tank is assumed to be at 40°F when the outside air temperature is -40°F. The structural design of the PCS tank also conservatively assumes that the temperature on the inside of the PCS tank walls is at 40°F when the outside air temperature is postulated to be at 115°F.

Normal thermal loads due to a thermal gradient in the structures below the grade level (exterior walls and basemat) are small and are not considered in the design.]*

Effects of Pipe Rupture (Y)

[*The evaluations consider the following loads:*

- *Accident design pressure load, P_a, within or across a compartment and/or building generated by the postulated pipe rupture, including the dynamic effects due to the pressure time history.*

  The main steam isolation valve (MSIV) compartments are designed for a 6.5 pounds per square inch (psi) pressurization load. The steam generator blowdown valve compartment is designed for a 6 psi pressurization load.

- *Accident thermal loads, T_a, due to thermal conditions generated by the postulated pipe break and including T_o.*

Temperature gradients are based on an exterior air temperature of -40°F.

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The structural integrity of the west wall of the main control room is also evaluated for the jet impingement (Y_j)]*

3H.3.4 Load Combinations and Acceptance Criteria

[*Concrete structures are designed in accordance with ACI 349 for the load combinations and load factors given in Table 3.8.4-2. Steel structures are designed in accordance with ANSI/AISC N690 for the load combinations and stress limit coefficients given in Table 3.8.4-1. The following supplemental requirements are applied for the use of ANSI/AISC N690:*

- *In Section Q1.0.2, the definition of secondary stress applies to stresses developed by temperature loading only.*
- *In Section Q1.3, where the structural effects of differential settlement are present, they are included with the dead load, D.*
- *In Table Q1.5.7.1, the stress limit coefficients for compression are as follows:*
  - *1.3 instead of 1.5 in load combinations 2, 5, and 6*
  - *1.4 instead of 1.6 in load combinations 7, 8, and 9*
  - *1.6 instead of 1.7 in load combination 11*
- *In Section Q1.5.8, for constrained members (rotation and/or displacement constraint such that a thermal load causes significant stresses) supporting safety-related structures, systems, or components, the stresses under load combinations 9, 10, and 11 are limited to those allowed in Table Q1.5.7.1 as modified above.*]*

3H.4 Seismic Analyses

[*A global seismic analysis of the AP1000 nuclear island structure is performed to obtain building seismic response for the seismic design of nuclear safety-related structures. The seismic loads for the design of the shear walls and the slabs in the auxiliary building are based on a response spectrum analysis of the auxiliary building and the shield building 3D finite element models.*]* This analysis is described in Subsection 3.7.2. [*For determining the out-of-plane seismic loads on flexible slabs and wall segments, spectral accelerations are obtained from time history analyses or from the relevant response spectra, using the 7 percent damping curve. Hand calculations are performed to estimate the out-of-plane seismic forces and the corresponding bending moment in each shear wall and floor slab element to supplement the loads obtained from the global seismic analysis.*]*

3H.4.1 Live Load for Seismic Design

[*Floor live loads, based on requirements during plant construction and maintenance activities, are specified varying from 50 to 250 pounds per square foot.*

For the local design of members, such as the floors and beams, seismic loads include the response due to masses equal to 25 percent of the specified floor live loads or 75 percent of the roof snow load, whichever is applicable. These seismic loads are combined with 100 percent of the specified live loads, or 75 percent of the roof snow load, whichever is applicable. These live and snow loads are included as mass in calculating the vertical seismic forces on the floors and roof. The mass of equipment and distributed systems is included in both the dead and seismic loads.]*

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3H.5 Structural Design of Critical Sections

[*This subsection summarizes the structural design of representative seismic Category I structural elements in the auxiliary building and shield building. Critical sections are listed below and the corresponding location numbers are shown on Figure 3H.5-1 for twelve of the critical sections. Items 13 and 14 in the list below are located in the shield building cylinder and are discussed in APP-GW-GLR-602 (Reference 1). The basis for their selection to this list is also provided for each* critical section.

(1) *South wall of auxiliary building (column line 1), elevation 66'-6" to elevation 180'-0". (This exterior wall illustrates typical loads such as soil pressure, surcharge, temperature gradients, seismic, and tornado.) – see subsection 3H.5.1.1 and Figures 3H.5-2 and 3H.5-3*

(2) *Interior wall of auxiliary building (column line 7.3), elevation 66'-6" to elevation 160'-6" (This is one of the most highly stressed shear walls.) – see subsection 3H.5.1.2 and Figure 3H.5-4*

(3) *West wall of main control room in auxiliary building (column line L), elevation 117'-6" to elevation 153'-0". (This illustrates design of a wall for subcompartment pressurization.) – see subsection 3H.5.1.3 and Figure 3H.5-12*

(4) *North wall of MSIV east compartment (column line 11 between column lines L and M), elevation 117'-6" to elevation 153'-0". (The main steam line is anchored to this wall segment.) – see subsection 3H.5.1.4 and Figure 3H.5-5*

(5) *Roof slab at elevation 180'-0" adjacent to shield building cylinder. (This is the connection between the two buildings at the highest elevation.) – see subsection 3H.5.2.1 and Figure 3H.5-7*

(6) *Floor slab on metal decking at elevation 135'-3". (This is a typical slab on metal decking and structural steel framing.) – see subsection 3H.5.2.2 and Figure 3H.5-6*

(7) *2'-0" slab in auxiliary building (operations work area (tagging room) ceiling) at elevation 135'-3". (This illustrates the design of a typical 2'-0" thick concrete slab.) – see subsection 3H.5.3.1 and Figure 3H.5-8. (Note: The 'Tagging Room' has been renamed as "Operations Work Area." However, to avoid changing the associated design and analysis documents, this room is referred to as the 'Tagging Room.')*

(8) *Finned floor in the main control room at elevation 135'-3". (This illustrates the design of the finned floors.) – see subsection 3H.5.4 and Figure 3H.5-9*

(9) *Shield building roof/exterior wall of PCS water storage tank. (This is a unique area of the roof and water tank.) – see subsection 3H.5.6.3*

(10) *Shield building roof/interior wall of PCS water storage tank. (This is a unique area of the roof and water tank.) – see subsection 3H.5.6.2*

(11) *Shield building roof, tension ring, and air inlet. (This is the junction between the shield building roof and the cylindrical wall of the shield building.) – see subsections 3H.5.6 and 3H.5.6.1*

(12) *Divider wall between the spent fuel pool and the fuel transfer canal. (This wall is subjected to thermal and seismic sloshing loads.) – see subsection 3H.5.5.1 and Figure 3H.5-10*

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(13) Shield building SC cylinder is the exposed portions of the shield building that are not protected by the Auxiliary Building and is a steel concrete composite structure – see subsection 3H.5.7.1, Figure 3H.5-16, and Figures 5 and 6 of APP-GW-GLR-602 (Reference 1)

(14) Shield building SC to RC connection is the region of the shield building that anchors the SC cylindrical wall modules to the RC basemat and wall of the shield building – see subsection 3H.5.7.2, Figure 3H.5-16, and Figures 1, 2, and 3 of APP-GW-GLR-602 (Reference 1)

The design implemented in fabrication and construction drawings and instructions will have the design shown, an equal design, or a better design for the key structural elements.]*

3H.5.1 Shear Walls

Structural Description

[*Shear walls in the auxiliary building vary in size, configuration, aspect ratio, and amount of reinforcement. The stress levels in shear walls depend on these parameters and the seismic acceleration level. The range of these parameters and the stress levels in various regions of the most severely stressed shear wall are described in the following paragraphs.*

The height of the major structural shear walls in the auxiliary building ranges between 30 to 120 feet. The length ranges between 40 and 260 feet. The aspect ratio of these walls (full height/full length) is generally less than 1.0 and often less than 0.25. The walls are typically 2 to 5 feet thick, and are monolithically cast with the concrete floor slabs, which are 9 inches to 2 feet thick. Exterior shear walls are several stories high and do not have many large openings. Interior shear walls, however, are discontinuous in both vertical and horizontal directions. The in-plane behavior of these shear walls, including the large openings, is adequately represented in the analytical models for the global seismic response. Where the refinement of these finite element models is insufficient for design of the reinforcement, for example in walls with a large number of openings, detailed finite element models are used.

The shear walls are used as the primary system for resisting the lateral loads, such as earthquakes. The auxiliary building shear walls are also evaluated for flexure and shear due to the out-of-plane loads.]*

Design Approach

[*The auxiliary building shear walls are designed to withstand the loads specified in subsection 3H.3.3. Beside dead, live, and other normal operating condition loads, the following loads are considered in the shear wall design:*

- *Seismic loads*
  - *The SSE loads for the wall are obtained from the seismic analyses of auxiliary/shield buildings that are described in subsection 3H.4.*
  - *Calculations are performed by considering shear wall segments bounded by the floors below and above the segment and the adjacent walls perpendicular to, on both sides of, the segment under consideration. Appropriate boundary conditions are assumed for the four edges of the segment. Natural frequencies of wall segments are determined using finite element models or text book formulas for the frequency of plate structures. Corresponding spectral acceleration is determined from the applicable response spectrum.*

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- *Exterior walls, below grade level, are also evaluated for dynamic earth pressure exerted during an SSE for two cases:*
  - *Dynamic earth pressure calculated in accordance with ASCE 4-98*
  - *Passive earth pressure*

- *Accident pressure load*
  - *Shear walls of the main steam isolation valves (MSIV) rooms are designed for 6.5 pounds per square inch (psi) differential pressure acting in conjunction with the seismic loads. Member forces due to accident pressure and SSE are combined by absolute sum.*
  - *The main control room wall of the east MSIV compartment is evaluated for the pressure and the jet load due to a postulated main steamline break.*

- *Tornado load*

  For exterior walls above grade level, tornado loads are considered.

The design temperatures for thermal gradient are included in Table 3H.5-1.

The shear walls are designed for the load combinations, as applicable, contained in Table 3.8.4-2. The wall sections are designed in accordance with the requirements of ACI 349-01.]*

3H.5.1.1 Exterior Wall at Column Line 1

[*The wall at column line 1 is the exterior wall at the south end of the nuclear island. The reinforced concrete wall extends from the top of the basemat at elevation 66'-6" to the roof at elevation 180'-0". It is 3'-0" thick below the grade. Above the grade, the wall thickness is as follows:*

- *2'-3", from column line I to 5'-7" west of column line K-2, from elevation 100'-0" to 109'-3";*
- *3'-0", from 5'-7" west of column line K-2 to column line N, from elevation 100'-0" to 109'-3";*
- *2'-3", from column line I to 5'-6" east of column line L-2, from elevation 109'-3" to 180'-0";*
- *3'-0", from 5'-6" east of column line L-2 to column line N, from elevation 109'-3" to 125'-0";*
- *2'-3", from 5'-6" east of column line L-2 to column line N, from elevation 125'-0" to 180'-0".*

The wall is designed for the applicable loads including dead load, live load, hydrostatic load, static and dynamic lateral soil pressure loads, seismic loads, and thermal loads. For various segments of this wall, Table 3H.5-2 provides the listing and magnitude of the various design loads under governing load combinations and Table 3H.5-3 presents the details of the wall reinforcement. The sections where the required reinforcement is calculated are shown in Figure 3H.5-2 (Sheet 1). Typical wall reinforcement is shown on Figure 3H.5-3.]*

3H.5.1.2 Wall at Column Line 7.3

[*The wall at column line 7.3 is a shear wall that connects the shield building and the nuclear island exterior wall at column line I. It extends from the top of the basemat at elevation 66'-6" to the top of the roof. The wall is 3 feet thick below the grade at elevation 100'-0" and 2 feet thick above the grade.*

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Out-of-plane lateral support is provided to the wall by the floor slabs on either side of it and the roof at the top.

The auxiliary building design loads are described in Section 3H.3.3, and the wall is designed for the applicable loads.

For various segments of this wall, the corresponding governing load combination and associated design loads are shown in Table 3H.5-4. Table 3H.5-5 presents the details of the wall reinforcement. The sections where the required reinforcement is calculated are shown in Figure 3H.5-2 (Sheet 2). Typical wall reinforcement is shown on Figure 3H.5-4]*

3H.5.1.3 Wall at Column Line L

[*The wall at column line L is a shear wall on the west side of the Main Control Room. It extends from the top of the basemat at elevation 66'-6" to the top of the roof. The wall is 2 feet thick. Out-of-plane lateral support is provided to the wall by the floor slabs on either side of it and the roof at the top. The segment of the wall that is a part of the main control room boundary is from elevation 117'-6" to elevation 135'-3".*

The auxiliary building design loads are described in subsection 3H.3.3, and the wall is designed for the applicable loads. In addition to the dead, live and seismic loads, the wall is designed to withstand a 6.5 pounds per square inch pressure load due to a pipe break in the MSIV room even though it is a break exclusion area. This wall segment is also designed to withstand a jet load due to the pipe break.

The governing load combination and associated design loads are those due to the postulated pipe rupture and are shown in Table 3H.5-6. Table 3H.5-7 and Figure 3H.5-12 present the details of the wall reinforcement. The sections where the required reinforcement is calculated are shown in Figure 3H.5-2 (Sheet 3).]*

3H.5.1.4 Wall at Column Line 11

[*The north wall of the MSIV east compartment, at column line 11 (Wall 11) between elevation 117'-6" and elevation 153'-0", has been identified as a critical section.*

The segment of the wall between elevation 117'-6" and elevation 135'-3" is 4 feet thick, and several pipes such as the main steam line, main feed water line, and the start-up feed water line are anchored to this wall at the interface with the turbine building.

The wall segment from elevation 135'-3" to elevation 153'-0" does not provide support to any high energy lines, and is 2 feet thick. This portion does not have to withstand reactions from high energy line breaks.

The wall (Wall 11) is designed to withstand loads such as the dead load, live load, seismic load and the thermal load. The MSIV compartments are break exclusion areas, but the design also considered the compartment pressurization load associated with a one square foot pipe rupture in the MSIV compartments. Wall 11 in the east MSIV compartment is also designed to withstand the jet loads and the reactions at the pipe anchors for main steam and main feedwater line breaks assumed to occur in the east MSIV compartment. Additionally, the Wall 11 design for both MSIV compartments considers loads from postulated breaks in the turbine building, which include turbine building first bay pressurization, jet loads, reactions at the Wall 11 pipe anchors, and pipe whip.

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The wall structure is analyzed using three dimensional finite element analyses supplemented by hand calculations. Analyses are performed for individual loads, and design loads are determined for applicable load combinations from Table 3.8.4-2.

Wall reinforcement for this location is shown in Figure 3H.5-5 illustrating conformance with ACI 349-01 requirements. The reinforcement detailing and development for the wall of the MSIV west compartment varies because the compartment is in the corner of the auxiliary building and the adjacent wall on column line Q (Wall Q) is oriented at 90 degrees to Wall 11.]*

3H.5.2 Composite Structures (Floors and Roof)

[*The floors consist of a concrete slab on metal deck, which rests on structural steel floor beams. Several floors in the auxiliary building are designed as one-way reinforced concrete slabs supported continuously on steel beams. Typically, the beams span between two reinforced concrete walls. The beams are designed as unshored, composite beams with formed metal deck spanning perpendicular to the members. For the floors, beams are predominately spaced at about 5- to 6-feet intervals and spans are between 15 feet and 25 feet. Based on local geometry considerations the intervals and spans are outside these ranges in a limited number of locations. The spacing between the beams or between beams and walls is as small as 3 feet and as large as 8 feet. The span of the beams is as small as 2 feet, 6 inches and as large as 38 feet, 6 inches. The designs of the beams satisfy the requirements in ANSI/AISC N690 for composite structures.*]*

Structural Description

[*A typical layout of these floors is shown in Figure 3H.5-6. The metal deck rests on the top flange of the structural steel floor beam, with the longitudinal axes of the metal deck ribs and floor beams placed perpendicular to each other. Figure 3H.5-6 shows the key structural elements in composite floors. The reinforcement size and spacing are based on loads and spans for this type of floor and are determined at each location based on the requirements in ACI 349 and ACI 318-11, Section 12.6. The development of the floor reinforcement in the walls can be either headed reinforcement or standard hooks. The beam size and spacing and beam support designs are based on loads and spans for this type of floor as noted on the figure. The beam support designs include beam seats or shear plates connected to the web of the beam. The detail design of the support for the beam, including the portion embedded in the concrete wall, is based on the load and structural system configuration as noted on the figure. The designs of these floors are in conformance with ANSI/AISC N690 and ACI 349. The depth of the ribs for 9-inch concrete floor slabs, 9.5-inch concrete floor slabs, 15-inch concrete floor slabs, and 15-inch deep concrete roof slabs are 3 inches, 2.5 inches, 3 inches, and 4.5 inches respectively. The concrete slab is tied to the structural steel floor beam by shear connectors, which are welded to the top flange of the floor beam. The concrete slab and the floor beams form a composite floor system. For the design loads after hardening of concrete, the transformed section is used to check the stresses.*

The construction sequence is as follows:

- *The structural steel floor (floor beam, metal deck, and shear connectors) is fabricated in the shop, brought to the floor location, and placed in position. In some cases, the beams and deck are preassembled and placed as a module.*

- *The metal deck is used as the formwork, and concrete is poured on the metal deck. Until concrete hardens, the load is carried by the metal deck and the steel floor beam.*

- *During concreting, no shoring is provided for the beams supporting the floors and roof. Local shoring of the metal deck at penetrations and other openings in the floor and supporting wall, or to act as temporary support at the location of an incomplete wall, is acceptable.*]*

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Design Approach

[*The floor design considers the dead, live, construction, extreme environmental, and other applicable loads identified in Section 3H.3.3. The design floor loading includes the equipment attached to the floor. The end condition for the steel beams is simply supported, or continuous. The seismic load is obtained using the applicable floor acceleration response spectrum (7 percent damping for the SSE loads).*

The load combinations applicable to the design of these floors are shown in Tables 3.8.4-1 and 3.8.4-2. The design of the floor system is performed in two parts:

- *Design of structural steel beams*
  - *The structural steel floor beams are evaluated to withstand the weight of wet concrete during the placement of concrete. The composite section is designed for the design loads during normal and extreme environment conditions. Shear connectors are also designed.*
- *Design of concrete slab*
  - *The concrete slab and the steel reinforcement of the composite section are evaluated for normal and extreme environmental conditions. The slab concrete and the reinforcement is designed to meet the requirements of American Concrete Institute standard ACI 349-01 "Code Requirements for Nuclear Safety Related Concrete Structures."*
  - *The slab design considers the in-plane and out-of-plane seismic forces. The global in-plane and out-of-plane forces are obtained from the response spectrum analysis of the 3D finite element model of the auxiliary and shield buildings. The out-of plane seismic forces due to floor self-excitation are determined by hand calculations using the applicable vertical seismic response spectrum and slab frequency.*]*

3H.5.2.1 Roof at Elevation 180′-0″, Area 6 (Critical Section is between Col. Lines N & K-2 and 3 & 4)

[*The layout of this segment of the roof is shown in Figure 3H.5-7 as Region "B." The concrete slab is 15 inches thick, plus 4.5-inch deep metal deck ribs. It is composite with 4 feet deep plate girders by using shear connectors. The girder spacing is a range of 13'-0" to 14'-2" center to center. The girder flanges are 26" x 2 1/2" and the web is 43" x 1". The girders span approximately 64 feet in the north-south direction and are designed as simply supported. The concrete slab between the girders behaves as a one-way slab and is designed to span between the girders.*

The roof girders are designed for dead and live loads, including construction loads (with wet concrete) with simple support end conditions. A one-third increase in allowable stress is permitted for the construction load combination.

The girders are also evaluated as part of the composite beam after drying of concrete. The composite roof structure is designed to withstand dead and live load / snow load, as well as the wind, tornado and seismic loads. The design summary for the critical section is shown in Table 3H.5-10. The design of the reinforcement and supporting beams, and girder and beam supports varies for other locations. The design is based on the loads and geometry for those sections.

A typical connection of the roof slab to the shield building is shown in Figure 3H.5-7. The figure shows the arrangement of reinforcement at the connection in the fuel building portion of the auxiliary building roof, the shield building cylindrical wall, and the walls of the auxiliary building just below the roof. The details of the connections between the auxiliary building roof and the shield building wall in other locations vary because of loads on the connection and the orientation of the wall to the roof

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reinforcement arrangement. These connection design details satisfy the requirements identified in Subsection 3.8.4.5.5.6.]*

3H.5.2.2 Floor at Elevation 135′-3″, Area 1 (Between Column Lines M and P)

[*The design of a typical composite floor is shown in Figure 3H.5-6. The design summary for the floor between column lines M and P at elevation 135'-3" is shown in Table 3H.5-11. The concrete slab is 9 inches thick, plus 3-inch deep metal deck ribs. The floor beam size is shown on the figure.*

- *The floor beams are designed for construction load (with wet concrete) with simple support end conditions. The design loads include the dead load and a construction live load of 100 pounds per square foot (psf) distributed load plus 5000 pounds concentrated load near the point of maximum shear and moment. A one-third increase in allowable stress is permitted.*

- *The floor beams are also designed as part of the composite beam after drying of the concrete. Because of continuity of rebars into the wall and the connection of the bottom flange to the support embedment, the end support condition is considered as fixed.*]*

3H.5.3 Reinforced Concrete Slabs

[*Reinforced concrete floors in auxiliary building are 24 inch or 36 inch thick. These floors are constructed with reinforced concrete placed on the top of 8 to 12 inch thick precast concrete panels. The precast concrete panels are installed at the bottom to serve as the formwork and withstand the load of wet concrete slab. The spans of the floors are predominately 13 feet to 20 feet, and the precast panels are predominately 7 to 14 feet wide. Based on local geometry considerations, the widths and spans are outside these ranges in a limited number of locations. The spans of the floor are as small as 5 feet and as large as 21 feet. The width of the precast panels is as small as 4 feet and as large as 19 feet. The number of side-by-side precast panels ranges from one to eight.*

Examples of such floors are the Operations Work Area (Tagging Room) ceiling slab at elevation 135 ft 3 inches in Area 2, and the Area 5/6 elevation 100'-0" slab between column lines 1 & 2.

Figure 3H.5-8 shows the key structural elements in reinforced concrete floor slabs. The precast panels and the cast-in-place concrete are designed to act together as a composite reinforced slab so that the floor dynamic response is consistent with the auxiliary building finite element analysis. However, the precast panels are neglected in determining floor strength and load carrying capacity. The reinforcement size and spacing are determined for each location, based on specific loads and spans, and satisfy the requirements in ACI 349 and ACI 318-11, Section 12.6. The floor thickness and precast panel thickness for this type of floor are based on specific loads and spans as noted on the figure. The type and thickness of adjacent walls and floors vary as noted on the figure. The main reinforcement is provided in the cast-in-place concrete. Reinforcement is placed in both the top and bottom layers of the cast-in-place concrete in both directions. For the design of the reinforcement in the cast-in-place floors, post-construction loads are conservatively assumed to be resisted only by the cast-in-place concrete and the reinforcement placed within it. The reinforcement in the cast-in-place portion is fully developed into supporting adjacent walls such that the connection is assumed to be a fixed connection. The development of the floor reinforcement in the walls is achieved using either headed reinforcement or standard hooks.

The precast panel reinforcement is designed to resist the weight of the panel and the wet weight of the cast-in-place concrete during construction. Reinforcement is placed within the precast panel portion in both top and bottom layers in both directions. The precast panel reinforcement is contained within the panel. The reinforcement is discontinuous with a design gap between adjacent precast panels and between precast panels and walls. The precast panels are connected to the concrete

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placed above them by shear reinforcement which satisfies the requirements of ACI 349 Chapter 17. The precast panels and the cast-in-place concrete are made to act together as a composite reinforced concrete slab by roughening the top surface of the precast panel and providing shear ties between the two elements. The detail designs of the supports for the precast panels are based on the loading and design requirements. The design of these floors is in conformance with ANSI/AISC N690 and ACI 349.

The finite element analysis model used for the auxiliary building seismic response assumes a homogenous thickness of concrete for the floor system, and includes floor-to-wall connections that are fixed over the full thickness of the reinforced concrete floor. The detailed design of the floor system includes a gap between the precast panel and the wall and between adjacent precast panels. Although the gap between the precast panels and the wall reduces the thickness of the floor in direct contact with the wall, the design of the floor system satisfies the requirements of ACI 349, including fully developing the floor reinforcement in the wall. The design of the floor system and the connection with the wall provide a fixed connection that transfers forces and moments from the floor to the wall.

Detailed analysis of the floor system connection design details, including the gap between the precast panel and wall is performed for the floor constructed with precast panels and is consistent with the nuclear island seismic model. The effects of stiffness, reinforcement anchorage, and concrete cracking are considered in the detailed analyses. The detailed analyses demonstrate that these floors have vertical response above 33 Hz and are rigid, which is consistent with the nuclear island seismic model.]*

3H.5.3.1 Operations Work Area (Tagging Room) Ceiling

The tagging room (room number 12401) location is shown on Figure 1.2-8. [*Figure 3H.5-8 shows the typical cross section and reinforcement. The design summary for this location is shown in Table 3H.5-12. Design dimensions of the Operations Work Area (Tagging Room) Ceiling are as follows:*

Room Size:	*16'-0" x 11'-10"*
Boundary Conditions:	*Fixed at Walls J and K*
Clear Span:	*16'-0"*
Slab Thickness:	*Total = 24 inches*
	Precast Panel = 8 inches
	Cast-in-Place = 16 inches

*The two precast concrete panels, each 6'-4½" wide and spanning over 16'-0" clear span, are installed to serve as the formwork.]**

3H.5.4 Concrete Finned Floors

[*The ceilings of the main control room and the instrumentation and control rooms in the auxiliary building are designed as finned-floor modules. A typical floor design is shown in Figure 3H.5-9. A finned floor consists of a 24-inch-thick concrete slab poured over a stiffened steel plate ceiling. The fins, welded to stiffen the steel plate, are half inch by 9 inch rectangular sections perpendicular to the plate. Shear studs are welded on the other side of the steel plate, and the steel and concrete act as a composite section. Figure 3H.5-9 shows the finned floor above the main control room and the connection of the floor to the wall on column line 11. The finned floors above the instrumentation and control rooms are similar, but differ with some design details, such as the size and spacing of reinforcement and the connecting dowels used in the connection of the floor to the wall. Penetrations and other interferences may cause localized variances with the design details shown in the figure.*

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The fins are exposed to the environment of the room and enhance the heat-absorbing capacity of the ceiling. Several shop-fabricated steel panels, cut to room width and placed side by side perpendicular to the room length, are used to construct the stiffened plate ceiling in a modularized fashion. The stiffened plate with fins is designed to withstand construction loads prior to concrete hardening.]*

The construction sequence for finned floors is as follows:

- Each panel (fins, steel plates, and shear connectors) is fabricated in the shop, brought to the floor location, and placed in position. In some cases, the panels are preassembled and placed as a module. The fins frame into steel shapes at the ends of the floors. The steel shapes are supported by intermittent brackets connected to the walls.

- The steel plate is used as the formwork for concrete placement. The wet concrete load is carried by the steel plates and the fins.

- During concrete placement, shoring may be provided for floors with longer spans. Local shoring of the steel plates at penetrations and other openings in the floor and supporting wall, or to act as temporary support at the location of an incomplete wall may also be provided.

[*The main control room ceiling fin floor is designed for the dead, live, and the seismic loads. The design summary is shown in Table 3H.5-13.*

The finned floor structure is evaluated for the load combinations listed in Tables 3.8.4-1 and 3.8.4-2.]*

Design Methodology

[*The finned floors are designed as reinforced concrete slabs in accordance with ACI Standard 349. For positive bending, the steel plate is in tension. The steel plate with fin stiffeners serves the function of bottom rebars. For negative bending, the potential for buckling due to compression in this element is checked by using the criteria of American National Standards Institute/American Institute of Steel Construction standards ANSI/AISC N690-94. Twisting, and therefore lateral buckling of the stiffener, is restrained by the concrete.*

The finned floors resist vertical and in-plane forces for both normal and extreme loading conditions. For positive bending, the concrete above the neutral axis carries compressive stresses and the stiffened steel plate resists tension. Negative bending compression is resisted by the stiffened plate and tension by top rebars in the concrete. The neutral axis for negative bending is located in the stiffened plate section, and the concrete in tension is assumed inactive. Horizontal in-plane forces are resisted by the stiffened plate and longitudinal rebars.

Minimum top reinforcement is provided in the slab in each direction for shrinkage and temperature crack control. In addition, top reinforcement located parallel to the stiffeners is used as tension reinforcement in negative bending. The stiffened plate provides crack control capability for the bottom of the slab in the transverse direction. Additional bottom layer reinforcing steel is provided in the finned floors at elevation 117'-6" where needed to maintain the structural integrity of the fire barrier during a fire event due to potential deterioration of mechanical properties of exposed steel fin plates during the fire.

Composite section properties, based on an all steel-transformed section, as detailed in Section Q1.11 of ANSI/AISC N690-94, are used to design the following:

- *Weld strength between stiffener and the steel plate*
- *Spacing of the shear studs for the composite action*

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The stiffened plate alone is designed to resist all construction loads prior to the concrete hardening. The plate is designed against the criteria for bending and shear, specified in ANSI/AISC N690-94, Sections Q1.5.1.4 and Q1.5.1.2. In addition, the weld between the stiffener and the steel plate is designed to satisfy the code requirements.]*

3H.5.5 Structural Modules

[*Structural modules are used for some of the structural elements on the south side of the auxiliary building. These structural modules are structural elements built up with welded steel structural shapes and plates. The modules consist of steel faceplates connected by steel trusses as shown in Figure 3.8.3-2. The primary purpose of the trusses is to stiffen and hold together the faceplates during handling, erection, and concrete placement. The thickness of the steel faceplates is 0.5 inch except in a few local areas. The nominal spacing of the trusses is 30 inches. Shear studs are welded to the inside faces of the steel faceplates. Faceplates are welded to adjacent faceplates with full penetration welds so that the weld is at least as strong as the plate. The structural wall modules are anchored to the concrete base by reinforcing steel dowels or other types of connections embedded in the reinforced concrete below. After erection, concrete is placed between the faceplates.*

These modules include the spent fuel pool, fuel transfer canal, and cask loading and cask washdown pits. The structural modules are similar to the structural modules for the containment internal structures (see description in subsection 3.8.3 and Figures 3.8.3-8, 3.8.3-14, 3.8.3-15 and 3.8.3-17). Figure 3.8.4-4 shows the location of the structural modules in the auxiliary building. The structural modules extend from elevation 66'-6" to elevation 135'-3".

The loads and load combinations applicable to the structural modules in the auxiliary building are the same as for the containment internal structures]* (Subsection 3.8.3.5.3) [*except that there are no ADS nor pressure loads due to pipe breaks.*

The design methodology of these modules in the auxiliary building is similar to the design of the structural modules in the containment internal structures]* described in Subsection 3.8.3.5.3.

3H.5.5.1 West Wall of Spent Fuel Pool

[*Figure 3H.5-10 shows an elevation of the west wall of the spent fuel pool (column line L-2), and element numbers in the finite element model. The wall is a 4 feet thick concrete filled structural wall module.*

A finite element analysis is performed for seismic, thermal, and hydrostatic loads with the following assumptions:

- *The seismic in-plane and out-of-plane forces are obtained from the response spectrum analysis of the 3D finite element model of the auxiliary and shield buildings.*
- *The thermal loads are applied as linearly varying temperatures between the inner and outer faces of the walls and floors.*
- *The hydrostatic loads are applied to the spent fuel pool walls and floors, which is considered full with water. This provides the loads for the design of the divider wall.*
- *The seismic sloshing is modeled in the spent fuel pool.*

The concrete filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349. The face plates are treated as reinforcing steel.

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Methods of analysis are based on accepted principles of structural mechanics and are consistent with the geometry and boundary conditions of the structures. Both computer codes and hand calculations are used.

Table 3H.5-8 shows the required plate thickness for certain critical locations. The steel plates are half inch thick.]*

3H.5.6 Shield Building Roof and Connections

[*The shield building roof is a reinforced concrete shell (supporting the passive containment cooling system tank and air diffuser), which is supported on a structural steel module. The structural configuration is shown on sheets 7, 8, and 9 of Figure 3.7.2-12. Air intakes are located at the top of the cylindrical portion of the shield building. The conical roof supports the passive containment cooling system tank. The conical roof is constructed as a structural steel module and lifted into place during construction. Steel beams provide permanent structural support for steel liner and concrete. The concrete is cast in place. Connection between concrete and steel liner are made using shear studs.*

The design of the shield building is shown in Figure 3H.5-11 (Sheets 1-6). These figures show the typical details of the "Tension Ring," the "Air Inlet Structure," and the "Exterior Wall of the Passive Containment Cooling System Tank." Figure 3H.5-16, Sheets 1 and 2, also shows the typical dimensions of the surface plates and the SC to RC connections on the shield building cylindrical segment.]* The column line and auxiliary building roof information in Figure 3H.5-16 is shown for reference and is not Tier 2* information. Interferences such as penetrations; openings; and embedded pipe, conduit, and anchors can cause localized variances with design details shown in the figure.

[*A detailed ANSYS model was used to represent these components of the enhanced design. Analyses were performed to determine the response of the structures for the dead weight, hydrostatic load due to PCS water, snow load, wind load, tornado load, seismic load (including seismic-induced pressure on PCS wall), and thermal loads. The design was evaluated to comply with the requirements of ANSI/AISC N690-94 and of ACI 349-01.*

The design summaries of the components are included in Table 3H.5-9.

The steel frame for the shield building roof and the concrete placed directly thereon is designed to ANSI/AISC N690.

- *In the radial direction, the steel beams, the steel surface plate, and the concrete are evaluated as a composite section using the axial and bending member forces in the steel and concrete section from the finite element analyses. The steel stresses and the end connection are calculated assuming the steel alone resists all loads applied before the concrete has reached 75 percent of its required strength and the effective composite section resists all loads applied after that time.*

- *The concrete is evaluated using all member forces in the concrete and surface steel plate from the finite element analyses (in-plane and out-of-plane forces and moments). The circumferential channels are provided for construction only and are not modeled in the finite element analysis or credited for resisting permanent loads. The concrete section is evaluated by the strength method of ACI-349. The steel plate is not considered as reinforcement in the circumferential direction.*

Additional information is provided in Table 3H.5-15.]*

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3H.5.6.1 Air Inlets and Tension Ring

[*The configuration and plate size of the air inlets enhance their structural performance. The air inlets structure (as shown on Figure 3H.5-14) is located at the top of the cylindrical wall portion of the shield building, beginning at approximately elevation 251' and rising to approximately elevation 266'. The air inlets serve as the intake for air as part of the PCS.*

Above the air inlets, at approximately elevation 266', is the connection designated as the tension ring that connects and supports the conical roof. The tension ring also contains 32 radial beam seat connections where the radial beams for the conical roof are connected.

The air inlets region is 4.5-feet thick with steel plates on each face as the primary reinforcement, which are connected using tie bars. Near the bottom of the air inlet structure, the thickness transitions to 3 feet thick to connect with the shield building cylinder. The air inlet openings are formed using pipe at a downward inclination of 38 degrees from the vertical. The pipe spacing is approximately 2.81 degrees circumferentially with shear studs welded to the outside surface of the pipes. The tie bars are located with three bars between adjacent air inlets at each elevation at maximum design spacing of 8.5 inches vertically. At approximately the same elevations as the tie bars, two 3/4-inch by 6-inch (minimum) shear studs are located between the tie bars except at elevations where there is interference with the air inlet pipes. Tie bars and studs may be omitted in local areas due to design features and other obstructions, such as penetrations, embedments and embedded pipe, conduit, and anchors.]* In the vicinity of the opening through the air inlet region, pipes and other design elements are not provided and the design details can vary from that shown on the figures.

[*The tension ring is designed as a structural steel box structure with concrete infill and shear studs. Also the connection of the RC conical roof to the tension ring is designed to be a mechanical connection. The air inlets and tension ring design methodology is supported by linear analysis and benchmarked nonlinear analysis. The tension ring is designed to ANSI/AISC N690 and is a concrete-filled box girder, with two continuous 1.5-inch-thick steel plates top and bottom, which connect the inner liner plate to the outer liner plate, as shown in Figure 3H.5-15.*]* Interferences such as penetrations; embedments; openings for concrete placement; and embedded pipe, conduit, and anchors can cause localized variances with the design details shown in the figure.

3H.5.6.2 Compression Ring and Interior Wall of Passive Containment Cooling Water Storage Tank

[*The other areas of the shield building are designed to existing industry code requirements, and include the conical roof, the passive containment cooling water storage tank, the compression ring, the knuckle region, and their related attachments. These areas are designed as RC structures in accordance with ACI 349. The steel frame for the roof is designed for the applicable building code ANSI/AISC N690. The concrete roof is designed to ACI 349 requirements without credit for the steel plate on the bottom of the concrete. The configuration and reinforcement of the compression ring and the connection to the interior wall of the passive containment cooling water storage tank is shown in Figure 3H.5-11.*]* Interferences such as penetrations; leak chases; and embedded pipe, conduit, and anchors can cause localized variances with the reinforcement design details shown in the figure.

[*Additional information is provided in Table 3H.5-15.*]*

3H.5.6.3 Knuckle Region and Exterior Wall of Passive Containment Cooling Water Storage Tank

[*The exterior wall of the passive containment cooling water storage tank is two feet thick. The wall starts at the tank floor elevation of 293' 9". There is a stainless steel liner on the inside surface of the tank. The wall liner consists of a plate with stiffeners and welded studs on the concrete side of the*

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plate. Leak chase channels are provided over the liner welds. The reinforcement in the concrete wall is designed without taking credit for the strength provided by the liner. The governing loads for design of the exterior wall are the hydrostatic pressure of the water, the in-plane and out-of-plane seismic response, and the temperature gradient across the wall. The reinforcement is shown in Figure 3H.5-11.]* Interferences such as penetrations; leak chases; and embedded pipe, conduit, and anchors can cause localized variances with the reinforcement design details shown in the figure. [*The reinforcement required and the reinforcement provided is summarized in Table 3H.5-9.*

Additional information is provided in Table 3H.5-15.]*

3H.5.7 Shield Building Cylinder (SC)

3H.5.7.1 Shield Building Cylindrical Wall

[*The shield building surrounds the containment vessel and shares a common basemat with the containment vessel and the auxiliary building. The cylindrical shield wall has an outside radius of 72.5 feet and a thickness of 36 inches. The cylindrical wall section that is below the auxiliary building roof line is a reinforced concrete structure. The section that is not protected by the auxiliary building is a steel concrete composite structure, where two 0.75-inch plates act compositely with 34.5 inches of concrete via tie bars and shear studs. The steel plate modules are connected to the reinforced concrete basemat and walls by mechanical connectors as described below.*

A typical configuration of the SC wall is shown in Figure 3H.5-13. The overall thickness of 36 inches is the same as the RC wall below. The concrete for the SC portion is standard concrete with a compressive strength of 6000 psi. The SC portion is constructed with steel surface plates, which act as concrete reinforcement. The nominal thickness of the steel faceplates is 0.75 inches. The faceplates are thicker (up to 1.0-inch nominal thickness), as necessary, to address loads from connections and attachments in the areas of these local loads. In each module, tie bars are welded to the steel faceplates to develop composite behavior of the steel faceplates and concrete. The shear studs are welded to the inside surface of the steel plate to provide composite action. The tie bars are at closer spacing in the higher stress regions. The reinforcement detailing incorporates ACI 349 requirements.

The panels of the SC wall are welded together with a complete joint penetration weld.

The wall is designed for the applicable loads described in subsection 3H.3.3. A finite element analysis is performed to determine the design forces.

Table 3H.5-14 shows the design summary for the enhanced shield SC cylindrical wall. The three sheets represent locations in the shield building cylinder that have some of the largest demands due to mechanical loads. The element on the west side at grade near the RC/SC connection has large tension forces due to overturning of the cylinder under seismic demand. This area is one of the most stressed elements in tension. The element near the fuel handling building roof at elevation 180' is an element with high out-of-plane shear due to the interaction between the fuel handling building and the cylinder during an earthquake. This element is located close to the fuel building roof. The element above wall 7.3 at elevation 175' has the largest demand for out-of-plane shear in the general part of the cylindrical wall away from the SC/RC connection and the interface with the auxiliary building roof.

Additional discussion and information are provided in Section 4 and Figures 5 and 6 of APP-GW-GLR-602 (Reference 1).]*

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3H.5.7.2 Reinforced Concrete (RC)/Steel Concrete Composite (SC) Horizontal and Vertical Connections

[*The steel plate modules are anchored to the RC basemat and walls of the shield building by mechanical rebar connections. The connectors provide for the direct transfer of forces from the RC reinforcing steel to the SC liner plates.*

At the horizontal connection at the interface with the RC structure that occurs on the bottom of the lowest SC wall module, vertical reinforcing bars in the RC basemat wall are connected to the module with a mechanical connection. A similar vertical connection occurs on the vertical edges of SC wall modules that interface with the RC portion of the shield building wall. In the vertical connection, hoop reinforcing bars in the RC wall are connected to a mechanical connection and forces are transferred directly from the hoop bars to the SC liner plate. The mechanical connections are designed to the stress limits of ANSI/AISC N690 for loads in the reinforcing bars equivalent to 125 percent of the specified yield strength of the reinforcing bar and are proven components used in existing structures. This design basis exceeds the maximum demand that occurs on the west side of the shield building at grade and is summarized in Sheet 3 of Table 3H.5-14. This connection improves the overall ductility of the RC/SC connection.

Additional discussion and information are provided in Section 4 and Figures 1, 2, 3, and 4 of APP-GW-GLR-602 (Reference 1).]*

3H.5.8 References

1. [*APP-GW-GLR-602, Revision 5, October 2014, (Proprietary) and APP-GW-GLR-603, Revision 5* (Non-Proprietary), “AP1000 Shield Building Design Details for Select Wall and RC/SC *Connections,” Westinghouse Electric Company LLC.*]*

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Table 3H.5-1
Nuclear Island: Design Temperatures for Thermal Gradient

Structure (See detail in Subsection 3H.3.3.)	Load	Temperature (°F)		Remark
PCS Tank Walls	Normal Thermal, T_o	[*(Outside)* *-40* *+115*	*(Inside)* *+40* *+40*]*	–
Roofs and Exterior Walls Above Grade Air Temperatures (Note 3)	Normal Thermal, T_o Accident Thermal, T_a	[*(Outside)* *-40* *+115* *-40* *-40*	*(Inside)* *+70* *+70* *+485* *+212*]*	– MSIV room Fuel handling area
Roofs and Exterior Walls Above Grade Concrete Temperatures (Note 3)	Normal Thermal, T_o Accident Thermal, T_a	[*(Outside)* *-21.6* *-22.8* *-25.4* *+3.2* *+109.1* *+108.0* *+107.5* *+98.6* *-40* *-40* *+63*	*(Inside)* *+47* *+48.4* *+51.5* *+46.6* *+79.2* *+80.7* *+81.3* *+81.3* *+222* *+212* *+212*]*	24″ thickness 27″ thickness 36″ thickness 15″ insulated roof 24″ thickness 27″ thickness 36″ thickness 15″ insulated roof MSIV room Fuel handling area Insulated roof (above fuel handling area)
Interior Walls/Slabs Concrete Temperatures	Normal Thermal, T_o Accident Thermal, T_a	[*(Side 1)* *N/R* *+70* *+70*	*(Side 2)* *N/R* *+222* *+212*]*	– MSIV room Fuel handling area
Exterior Walls Below Grade	Normal Thermal, T_o Accident Thermal, T_a	N/R +50	N/R +140	– Auxiliary Building
Basemat	Normal Thermal, T_o Accident Thermal, T_a	N/R +50	N/R +140	– Auxiliary Building
Shield Building (Between Upper Annulus and Auxiliary Building)	Normal Thermal, T_o Accident Thermal, T_a	[*(Outside)* *-40* *+115* *-40* *+70*	*(Inside)* *+70* *+70* *+222* *+165*]*	– MSIV room wall Rest of wall

Notes:
1. N/R means loads due to a thermal gradient are not required to be considered.
2. Based on ACI 349-01 (Appendix A), the base temperature for the construction is assumed to be 70°F.
3. For the auxiliary building exterior walls above ground, the temperature on the outside face of the walls can use 70°F instead of -40°F in winter or 115°F in summer when the exterior walls are adjacent to another building.

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Table 3H.5-2
Exterior Wall at Column Line 1 Forces and Moments in Critical Locations
(Units: kips, ft)

Load Combination	M_X	M_Y	M_{XY}	T_X	T_Y	T_{XY}
Elevation 180′-0″ to 135′-3″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		194.9	0.3		155.8	82.8
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym		266.4	18.4		27.2	34.0
D+F+L+H+Es+Pa+Ta+Ra+Yr+Yj+Ym	144.3		49.1	144.3		95.5
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	126.6		42.5	149.0		52.0
Elevation 135′-3″ to 100′-0″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		42.1	5.0		257.9	44.0
D+F+L+H+Es+Pa+Ta+Ra+Yr+Yj+Ym	59.7		1.9	149.7		47.3
D+F+L+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	83.4		37.0	72.2		53.0
Elevation 100′-0″ to 82′-6″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		16.4	13.8		194.4	130.8
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym		16.4	13.8		194.4	130.8
0.9D+0.9F+1.3L+1.2To+1.3H+1.3Ro	1.0		19.2	34.5		122.3
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	5.9		16.0	36.0		128.8
Elevation 82′-6″ to 66′-6″						
[*0.9D+F+H+Ro+To+Es*		18.1	2.3		204.2	83.4
0.9D+F+H+Ro+To+Es		105.7	22.8		230.6	25.6
0.9D+F+H+Ro+To+Es]*	24.4		2.3	11.4		83.4

Note:
X is along the horizontal direction, and Y is in the vertical direction.

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Table 3H.5-3
Exterior Wall on Column Line 1 Details of Wall Reinforcement (in ^{2}/ft)
(See Figure 3H.5-2 for Locations of Wall Sections.)

Wall Segment (See detail in Subsection 3H.5.1.1.)	Location	Required			*[Provided (Minimum)]**		
		Vertical	Horizontal	Shear(3)	*Vertical*	*Horizontal*	*Shear(3)*
Wall Section 1, 6							
Elevation 180′-0″ to 135′-3″				NR			None
	Outside Face	3.65	3.08		[*3.91*	*3.12*	
	Inside Face	2.76	2.51		*3.12*	*3.12*]*	
Wall Section 2, 3, 7							
Elevation 135′-3″ to 100′-0″				0.17			[*0.80*]*
	Outside Face	2.96	1.98		[*3.12*	*3.12*	
	Inside Face	2.96	2.10		*3.12*	*3.12*]*	
Wall Section 4, 8							
Elevation 100′-0″ to 82′-6″				0.03			[*0.44*]*
	Outside Face	3.04	1.55		[*3.12*	*1.56*	
	Inside Face	3.04	1.25		*3.12*	*1.27*]*	
Wall Section 5, 9							
Elevation 82′-6″ to 66′-6″				0.56			[*0.88*]*
	Outside Face	3.37	0.95		[*4.39*	*1.27*	
	Inside Face	2.68	0.95		*3.12*	*1.27*]*	

Notes:
1. NR = not required.
2. Not used.
3. Refer to Subsection 3.8.4.4.1 for the requirements for shear reinforcement.

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Table 3H.5-4
Interior Wall at Column Line 7.3 Forces and Moments in Critical Locations
(Units: kips, ft)

Load Combination	M_X	M_Y	M_{XY}	T_X	T_Y	T_{XY}
From Roof to Elevation 155′-6″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		21.3	15.6		135.0	233.5
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	54.3		16.9	241.8		199.5
Elevation 155′-6″ to 135′-3″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		9.1	4.0		192.2	189.3
D+F+L+H+Ro+To+Es]*	23.5		2.5	39.1		215.9
Elevation 135′-3″ to 117′-6″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		9.3	3.7		177.9	161.6
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	11.3		0.6	46.3		184.3
Elevation 117′-6″ to 100′-0″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		7.8	2.7		133.8	199.0
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	4.7		1.2	178.9		116.4
Elevation 100′-0″ to 82′-6″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		31.0	8.5		166.2	91.5
D+F+L+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	12.8		0.0	6.4		164.8
Elevation 82′-6″ to 66′-6″						
[*0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym*		24.5	3.8		105.7	79.3
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	2.5		3.8	9.7		79.3

Note:
X is along the horizontal direction, and Y is in the vertical direction.

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Table 3H.5-5
Interior Wall on Column Line 7.3 Details of Wall Reinforcement
(See Figure 3H.5-2 for Locations of Wall Sections.)

Wall Segment (See detail in Subsection 3H.5.1.2.)	Location	Wall Section	Reinforcement on Each Face (in^2/ft)	
			Required	[*Provided (Min.)*]*
From Roof to Elevation 155′-6″	Horizontal	1	4.06	[*4.12*
	Vertical	7	3.52	*3.72*
Elevation 155′-6″ to 135′-3″	Horizontal	2	2.52	*3.12*
	Vertical	8	3.64	*3.72*
Elevation 135′-3″ to 117′-6″	Horizontal	3	2.10	*2.54*
	Vertical	9	3.10	*3.12*
Elevation 117′-6″ to 100′-0″	Horizontal	4	2.34	*2.54*
	Vertical	10	2.97	*3.12*
Elevation 100′-0″ to 82′-6″	Horizontal	5	1.38	*2.54*
	Vertical	11	2.33	*3.12*
Elevation 82′-6″ to 66′-6″	Horizontal	6	0.63	*1.27*
	Vertical	12	1.52	*1.56*
Shear Reinforcement(2) (in^2/ft^2)				
From Roof to Elevation 155′-6″	Standard hook or T headed bar	7	0.22	*0.80*]*

Notes:
1. Not used.
2. Refer to Subsection 3.8.4.4.1 for the requirements for shear reinforcement

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Table 3H.5-6
Interior Wall at Column Line L Forces and Moments in Critical Locations
(Units: kips, ft)

Load Combination	M_X	M_Y	M_{XY}	T_X	T_Y	T_{XY}
Elevation 154′-2″ to 135′-3″						
[*0.9D+F+L+H+Ro+To+Es*		12.0	4.8		124.1	129.7
0.9D+F+H+Es+Pa+Ta+Ra+Yr+Yj+Ym]*	31.3		4.4	24.6		233.1
Elevation 135′-3″ to 117′-6″						
[*0.9D+F+L+H+Ro+To+Es*		4.5	5.0		124.8	127.5
0.9D+F+L+H+Ro+To+Es]*	26.1		6.2	9.3		132.6

Note:
X is along the horizontal direction, and Y is in the vertical direction.

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Table 3H.5-7
Interior Wall on Column Line L Details of Wall Reinforcement
(See Figure 3H.5-2, Sheet 3, for Locations of Wall Sections.)

Wall Segment (See detail in Subsection 3H.5.1.3.)	Location	Wall Section	Reinforcement on Each Face (in^2/ft)	
			Required	[*Provided (Min.)*]*
Elevation 154′-2″ to 135′-3″	Horizontal	1	2.09	[*2.27*
	Vertical	3	2.38	*3.12*
Elevation 135′-3″ to 117′-6″	Horizontal	2	1.46	*4.39*
	Vertical	4	2.37	*5.66*]*
Shear Reinforcement(2) (in^2/ft^2)				
Elevation 154′-2″ to 135′-3″	Standard hook or T headed bar	5	0.13	[*0.80*
Elevation 135′-3″ to 117′-6″	Standard hook or T headed bar	6	0.55	*1.76*]*

Notes:
1. Not used.
2. Refer to Subsection 3.8.4.4.1 for the requirements for shear reinforcement.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-8 (Sheet 1 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 20477

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-15.54	-11.97	-19.88	0.12	0.10	-1.72	-0.36	
Live (L)	0.78	-0.04	-1.18	-0.24	-0.13	-0.01	0.02	
Hydro (F)	30.96	4.98	-3.82	-63.41	-5.35	43.84	-5.16	
Seismic (Es)	82.85	54.22	125.46	58.55	31.63	87.52	15.70	
Thermal (To)	-493.98	-241.71	-55.16	-494.28	-151.42	59.44	-31.16	
Thermal (Ta)	-863.45	-406.83	-143.22	-466.24	-453.18	196.15	-90.75	
LC(1a)	22.92	-9.85	-35.18	-89.01	-7.58	58.96	-7.69	[*1.4D+1.7L+1.4F*
LC(3a)	111.43	49.18	99.06	-30.33	24.10	147.17	8.14	*D+L+F+Es*
LC(3b)	111.43	49.18	-151.86	-147.44	-39.15	-27.87	-23.26	*D+L+F+E's*
LC(3e)	-197.31	-101.89	64.59	-339.26	-70.53	184.32	-11.34	*D+L+F+Es+To*
LC(3f)	-197.31	-101.89	-186.34	-456.36	-133.79	9.28	-42.73	*D+L+F+E's+To*
LC(3m)	112.20	50.42	102.23	-30.11	24.23	147.35	8.15	*0.9D+F+Es*
LC(3n)	112.20	50.42	-148.70	-147.21	-39.03	-27.69	-23.24	*0.9D+F+E's*
LC(3o)	-196.53	-100.65	67.75	-339.03	-70.41	184.50	-11.32	*0.9D+F+Es+To*
LC(3p)	-196.53	-100.65	-183.17	-456.13	-133.67	9.46	-42.72	*0.9D+F+E's+To*
LC(5a)	-523.45	-261.30	-114.39	-979.93	-288.62	164.71	-62.21	*D+L+F+Ta*
LC(5b)	-624.45	-326.96	-150.38	-10.07	8.26	60.09	-7.70	*D+L+F+Ta*
LC(6a)	-541.61	-272.74	-24.91	-921.37	-256.99	252.23	-46.51	*D+L+F+Es+Ta*
LC(6b)	-707.30	-381.18	-275.84	-1038.48	-320.25	77.19	-77.91	*D+L+F+E's+Ta*
LC(7a)	-353.28	-188.68	-67.78	-437.47	-119.25	88.80	-29.14	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.24 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for the load combinations including thermal:	35.8 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for the load combinations including thermal:	74.2 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-8 (Sheet 2 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 10529

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-17.41	-97.25	-17.82	-1.28	-0.78	-0.71	-1.27	
Live (L)	-0.49	-3.11	-0.65	-0.09	-0.29	0.00	0.03	
Hydro (F)	6.13	-6.31	3.61	-1.47	-51.56	11.49	13.08	
Seismic (Es)	75.07	320.60	80.95	15.40	78.74	18.35	26.14	
Thermal (To)	-196.00	-447.61	-134.64	-144.47	-333.40	8.42	-4.77	
Thermal (Ta)	-361.26	-827.41	-251.05	-409.00	-954.60	32.29	-7.21	
LC(1a)	-16.64	-150.28	-21.00	-3.99	-73.77	15.09	16.57	[*1.4D+1.7L+1.4F*
LC(3a)	65.74	211.40	67.54	11.98	5.48	33.73	43.20	*D+L+F+Es*
LC(3b)	65.74	211.40	-94.37	-18.82	-151.99	-2.98	-9.07	*D+L+F+E's*
LC(3e)	-56.76	-68.36	-16.61	-78.31	-202.89	38.99	40.22	*D+L+F+Es+To*
LC(3f)	-56.76	-68.36	-178.52	-109.11	-360.36	2.29	-12.05	*D+L+F+E's+To*
LC(3m)	67.98	224.24	69.97	12.20	5.85	33.80	43.31	*0.9D+F+Es*
LC(3n)	67.98	224.24	-91.94	-18.60	-151.62	-2.90	-8.97	*0.9D+F+E's*
LC(3o)	-54.52	-55.52	-14.18	-78.10	-202.53	39.06	40.33	*0.9D+F+Es+To*
LC(3p)	-54.52	-55.52	-176.09	-108.90	-360.00	2.36	-11.95	*0.9D+F+E's+To*
LC(5a)	-237.57	-623.81	-171.77	-258.46	-649.25	30.96	7.33	*D+L+F+Ta*
LC(5b)	-293.85	-862.10	-215.10	7.87	-109.50	11.32	19.92	*D+L+F+Ta*
LC(6a)	-218.78	-541.51	-134.14	-243.06	-570.52	49.31	33.47	*D+L+F+Es+Ta*
LC(6b)	-368.91	-1182.70	-296.05	-273.86	-727.99	12.60	-18.81	*D+L+F+E's+Ta*
LC(7a)	-159.49	-448.50	-116.74	-111.35	-305.38	17.63	8.86	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.26 inches (maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for the load combinations including thermal:	48.2 ksi
[*Yield stress:*	*65.0 ksi (Minimum*]*
Maximum stress intensity range for the load combinations including thermal:	70.1 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-8 (Sheet 3 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 10544

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-23.27	-59.59	-41.62	3.76	-0.81	0.28	-0.71	
Live (L)	-0.82	-2.01	-1.14	0.36	-0.04	0.04	-0.02	
Hydro (F)	-4.24	-1.67	-3.22	28.34	2.08	1.19	-2.90	
Seismic (Es)	75.42	145.64	111.14	44.70	4.30	5.79	4.07	
Thermal (To)	-103.44	-329.56	-196.76	87.54	-91.81	42.74	40.36	
Thermal (Ta)	-188.49	-561.47	-351.05	198.05	-247.04	118.07	126.02	
LC(1a)	-40.05	-89.18	-64.72	45.55	1.71	2.13	-5.10	[*1.4D+1.7L+1.4F*
LC(3a)	45.29	81.70	63.87	88.50	6.37	7.78	-0.73	*D+L+F+Es*
LC(3b)	45.29	81.70	-158.42	-0.91	-2.24	-3.80	-8.87	*D+L+F+E's*
LC(3e)	-19.36	-124.27	-59.11	143.21	-51.01	34.49	24.50	*D+L+F+Es+To*
LC(3f)	-19.36	-124.27	-281.39	53.80	-59.62	22.91	16.36	*D+L+F+E's+To*
LC(3m)	48.45	89.67	69.17	87.76	6.49	7.72	-0.63	*0.9D+F+Es*
LC(3n)	48.45	89.67	-153.12	-1.64	-2.12	-3.87	-8.78	*0.9D+F+E's*
LC(3o)	-16.20	-116.31	-53.81	142.48	-50.89	34.43	24.59	*0.9D+F+Es+To*
LC(3p)	-16.20	-116.31	-276.09	53.07	-59.50	22.84	16.45	*0.9D+F+E's+To*
LC(5a)	-146.24	-414.18	-265.39	156.24	-153.16	75.31	75.13	*D+L+F+Ta*
LC(5b)	-238.80	-668.77	-433.11	76.45	-23.63	2.24	-19.91	*D+L+F+Ta*
LC(6a)	-163.37	-523.13	-321.97	200.94	-148.86	81.10	-15.83	*D+L+F+Es+Ta*
LC(6b)	-314.22	-814.40	-544.25	31.75	-157.47	69.51	-23.98	*D+L+F+E's+Ta*
LC(7a)	-107.64	-314.11	-196.14	99.83	-67.57	33.65	26.45	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.16 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for the load combinations including thermal:	51.3 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for the load combinations including thermal:	97.4 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-8 (Sheet 4 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 10524

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-26.88	-136.36	-4.55	-2.71	8.30	-1.73	-2.55	
Live (L)	-0.57	-3.79	-0.78	-0.28	-0.56	0.01	0.10	
Hydro (F)	9.65	-2.30	3.42	20.28	15.30	2.96	-1.29	
Seismic (Es)	71.56	289.90	91.93	54.83	199.98	7.42	64.34	
Thermal (To)	-318.13	-400.37	145.44	-220.52	-146.44	-5.03	-40.60	
Thermal (Ta)	-622.23	-668.85	285.16	-561.88	-521.80	-9.38	-75.34	
LC(1a)	-25.08	-200.56	-2.90	24.13	32.08	1.74	-5.20	[*1.4D+1.7L+1.4F*
LC(3a)	57.62	146.53	91.38	80.24	229.14	9.85	60.09	*D+L+F+Es*
LC(3b)	57.62	146.53	-92.47	-29.42	-170.83	-5.00	-68.59	*D+L+F+E's*
LC(3e)	-141.21	-103.70	182.28	-57.58	137.61	6.71	34.72	*D+L+F+Es+To*
LC(3f)	-141.21	-103.70	-1.57	-167.25	-262.35	-8.14	-93.96	*D+L+F+E's+To*
LC(3m)	60.88	163.95	92.61	80.79	228.87	10.01	60.24	*0.9D+F+Es*
LC(3n)	60.88	163.95	-91.24	-28.87	-171.09	-4.84	-68.44	*0.9D+F+E's*
LC(3o)	-137.95	-86.28	183.51	-57.03	137.35	6.87	34.87	*0.9D+F+Es+To*
LC(3p)	-137.95	-86.28	-0.34	-166.70	-262.62	-7.98	-93.81	*0.9D+F+E's+To*
LC(5a)	-406.69	-560.48	176.31	-333.88	-303.09	-4.62	-50.82	*D+L+F+Ta*
LC(5b)	-633.03	-744.00	335.67	-9.66	68.94	-92.69	-84.33	*D+L+F+Ta*
LC(6a)	-561.47	-454.11	427.59	-279.05	-103.11	-85.27	-19.99	*D+L+F+Es+Ta*
LC(6b)	-704.58	-1033.90	243.74	-388.71	-503.08	-100.12	-148.67	*D+L+F+E's+Ta*
LC(7a)	-257.42	-450.80	106.88	-147.30	-85.78	-2.46	-34.34	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.27 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for the load combinations including thermal:	39.6 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for the load combinations including thermal:	74.3 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-8 (Sheet 5 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 20462

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-8.17	-30.95	-0.12	-1.17	0.10	-0.16	0.54	
Live (L)	-0.14	-0.57	0.14	-0.22	-0.79	0.00	0.07	
Hydro (F)	5.35	-0.40	-1.33	8.58	86.46	1.24	-22.66	
Seismic (Es)	24.18	33.34	30.34	23.64	161.73	3.85	15.90	
Thermal (To)	-242.25	-76.28	48.77	-202.47	-136.67	0.52	-1.81	
Thermal (Ta)	-542.05	-110.19	87.80	-643.16	-500.95	5.98	3.24	
LC(1a)	-4.18	-44.88	-1.79	10.01	119.85	1.52	-30.86	[*1.4D+1.7L+1.4F*
LC(3a)	23.37	1.25	28.50	34.27	282.09	5.44	-15.22	*D+L+F+Es*
LC(3b)	23.37	1.25	-32.18	-13.01	-41.37	-2.27	-47.03	*D+L+F+E's*
LC(3e)	-128.03	-46.43	58.98	-92.27	196.68	5.76	-16.35	*D+L+F+Es+To*
LC(3f)	-128.03	-46.43	-1.70	-139.55	-126.78	-1.94	-48.16	*D+L+F+E's+To*
LC(3m)	24.32	4.92	28.37	34.60	282.87	5.45	-15.34	*0.9D+F+Es*
LC(3n)	24.32	4.92	-32.31	-12.67	-40.59	-2.25	-47.15	*0.9D+F+E's*
LC(3o)	-127.08	-42.76	58.85	-91.94	197.45	5.78	-16.48	*0.9D+F+Es+To*
LC(3p)	-127.08	-42.76	-1.82	-139.21	-126.00	-1.93	-48.28	*0.9D+F+E's+To*
LC(5a)	-341.74	-100.80	53.57	-394.78	-227.32	4.82	-20.04	*D+L+F+Ta*
LC(5b)	-539.71	-94.05	58.98	41.54	179.68	-5.14	-28.12	*D+L+F+Ta*
LC(6a)	-515.53	-67.46	89.32	-371.14	-65.59	8.68	-12.22	*D+L+F+Es+Ta*
LC(6b)	-563.90	-134.15	28.64	-418.42	-389.05	-8.99	-44.03	*D+L+F+E's+Ta*
LC(7a)	-184.82	-90.88	35.24	-144.35	-12.63	1.53	-24.50	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.20 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for the load combinations including thermal:	18.0 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for the load combinations including thermal:	23.0 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-8 (Sheet 6 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 21402

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-1.52	-18.30	4.68	-0.37	0.72	0.00	-0.10	
Live (L)	-0.21	-0.93	0.42	0.28	-0.02	-0.02	-0.06	
Hydro (F)	4.03	-0.43	-1.50	92.82	12.95	-15.46	3.60	
Seismic (Es)	36.96	21.63	18.85	145.49	29.84	11.74	17.22	
Thermal (To)	-192.49	-155.43	84.14	-251.71	-207.93	11.21	8.20	
Thermal (Ta)	-334.80	-329.90	141.91	-762.50	-659.55	44.32	29.47	
LC(1a)	3.16	-27.81	5.16	129.90	19.09	-21.67	4.80	[*1.4D+1.7L+1.4F*
LC(3a)	40.88	1.79	21.85	275.34	48.66	-9.92	22.10	*D+L+F+Es*
LC(3b)	40.88	1.79	-15.86	-15.63	-11.02	-33.39	-12.34	*D+L+F+E's*
LC(3e)	-79.43	-95.35	74.44	118.02	-81.30	-2.92	27.23	*D+L+F+Es+To*
LC(3f)	-79.43	-95.35	36.73	-172.95	-140.98	-26.39	-7.21	*D+L+F+E's+To*
LC(3m)	41.24	4.55	20.97	275.10	48.61	-9.90	22.17	*0.9D+F+Es*
LC(3n)	41.24	4.55	-16.74	-15.87	-11.07	-33.38	-12.27	*0.9D+F+E's*
LC(3o)	-79.07	-92.59	73.55	117.78	-81.35	-2.90	27.29	*0.9D+F+Es+To*
LC(3p)	-79.07	-92.59	35.85	-173.19	-141.03	-26.37	-7.15	*0.9D+F+E's+To*
LC(5a)	-206.95	-225.85	92.29	-383.84	-398.58	12.22	21.86	*D+L+F+Ta*
LC(5b)	-234.63	-256.17	103.47	-397.15	-399.55	10.60	21.12	*D+L+F+Ta*
LC(6a)	-197.67	-234.54	122.33	-251.67	-369.71	23.96	38.35	*D+L+F+Es+Ta*
LC(6b)	-271.60	-277.80	84.62	-542.64	-429.39	0.49	3.90	*D+L+F+E's+Ta*
LC(7a)	-142.00	-137.45	66.99	-91.35	-141.63	-7.85	9.75	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.20 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for the load combinations including thermal:	28.2 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for the load combinations including thermal:	34.2 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-8 (Sheet 7 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 21414

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	0.86	-10.93	-2.70	0.30	-0.02	-0.04	0.03	
Live (L)	0.15	0.06	-0.38	0.00	-0.13	-0.01	0.03	
Hydro (F)	3.53	0.73	-2.83	-24.72	-26.63	-1.07	1.12	
Seismic (Es)	23.63	12.73	32.11	21.80	30.03	2.52	5.47	
Thermal (To)	-66.65	-39.03	45.95	-118.96	-121.48	8.32	7.62	
Thermal (Ta)	-111.54	-58.96	50.88	-364.47	-375.75	24.84	24.99	
LC(1a)	6.38	-14.19	-8.39	-34.20	-37.54	-1.57	1.64	[*1.4D+1.7L+1.4F*
LC(3a)	29.57	2.87	25.06	-12.51	-7.41	0.97	7.08	*D+L+F+Es*
LC(3b)	29.57	2.87	-39.15	-56.11	-67.47	-4.06	-3.86	*D+L+F+E's*
LC(3e)	-12.09	-21.52	53.78	-86.86	-83.33	6.17	11.85	*D+L+F+Es+To*
LC(3f)	-12.09	-21.52	-10.43	-130.46	-143.39	1.14	0.91	*D+L+F+E's+To*
LC(3m)	29.34	3.91	25.72	-12.54	-7.28	0.98	7.06	*0.9D+F+Es*
LC(3n)	29.34	3.91	-38.50	-56.14	-67.33	-4.05	-3.88	*0.9D+F+E's*
LC(3o)	-12.32	-20.49	54.43	-86.89	-83.20	6.18	11.82	*0.9D+F+Es+To*
LC(3p)	-12.32	-20.49	-9.78	-130.49	-143.26	1.15	0.88	*0.9D+F+E's+To*
LC(5a)	-65.19	-47.00	25.89	-252.22	-261.63	14.41	16.78	*D+L+F+Ta*
LC(5b)	-88.36	-71.43	29.84	-36.01	-27.13	3.01	-2.28	*D+L+F+Ta*
LC(6a)	-64.73	-58.70	61.95	-230.42	-231.60	16.92	3.19	*D+L+F+Es+Ta*
LC(6b)	-111.99	-84.16	-2.27	-274.02	-291.66	11.89	-7.75	*D+L+F+E's+Ta*
LC(7a)	-45.20	-39.92	28.16	-114.86	-119.27	5.06	6.95	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.11 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for the load combinations including thermal:	27.2 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for the load combinations including thermal:	35.0 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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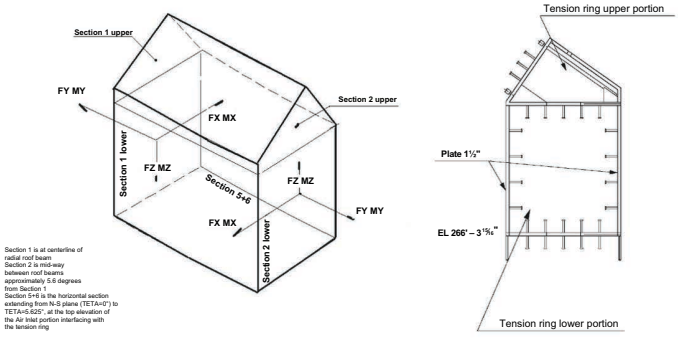
Table 3H.5-9 (Sheet 1 of 3)
Shield Building Roof Reinforcement Summary
(Tension Ring)

Tension Ring – Axial Force and Bending Verification								
Location		**Seismic Maximum Stresses**		**Maximum Stresses ksi**	**F_y ksi**	**Maximum Steel Area Required (in^2/ft)**	[***Steel Area Provided***]*	[***Design Limit for Ratio Max Required/ Provided***]*
Section	**Angles**	**Seismic L/C**	**f_a ksi**					
2 lower	5.625°	33	28.11	30.07	50	18.00	[*Liner 1 1/2" = 18 (in^2/ft) (Min)*]*	[*1.00*]*
	84.375°	41	27.59					
1 lower	0°	34	30.07					
	90°	43	29.48					
Tension Ring – Shear Force and Torsion Verification								
Location		**Seismic Maximum Stresses**		**Maximum Stresses ksi**	**F_y ksi**	**Maximum Steel Area Required (in^2/ft)**	[***Steel Area Provided***]*	[***Design Limit(1) for Ratio Max Required/ Provided***]*
Section	**Angles**	**Seismic L/C**	**f_v ksi**					
2 lower	5.625°	42	4.58	7.27	50	6.54	[*Liner 1 1/2" = 18 (in^2/ft) (Min.)*]*	[*0.36 + 2%*]*
	84.375°	35	4.24					
1 lower	0°	17	7.27					
	90°	9	6.78					



Notes:

1. [*The two percent of the ratio value that may be added to the design limit represents an allowance for minor variances in analysis results.*]*
2. Not used.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-9 (Sheet 2a of 3)
Shield Building Roof Reinforcement Summary
(Air Inlet)

AIS Reinforcement Summary – Horizontal Sections						
Locations (Figure 3H.5-11)		Steel Area (Vertical Direction – Z Local Dir.)				
		Required - Seismic Load Combinations (in^2/ft)		Maximum Required (in^2/ft)	[*Provided*]*	[*Design Limit(1) for Ratio Max Required/ Provided*]*
Sections	**Angles**	**Seismic L/C**	**Values**			
5+6	0°-5.625°	32	2.43	5.03	[*Liner 1" = 12 (in^2/ft) (Min.)*]*	[*0.42 + 2%*]*
	84.375°-90°	32	2.31			
7	0°-5.625°	16	4.19			
	84.375°-90°	24	4.27			
9	0°-5.625°	16	5.02			
	84.375°-90°	24	5.03			
11	0°-5.625°	16	3.55	3.57	[*Liner 1" = 12 (in^2/ft) (Min.)*]*	[*0.30 + 2%*]*
	84.375°-90°	24	3.57			

Notes:
1. [*The two percent of the ratio value that may be added to the design limit represents an allowance for minor variances in analysis results.*]*
2. Not used.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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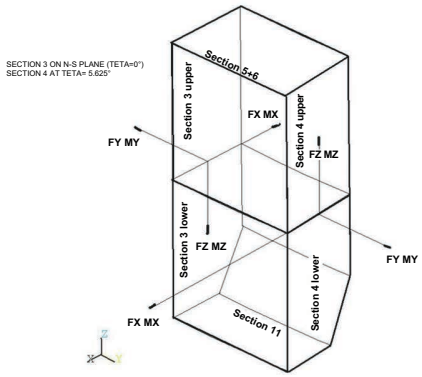
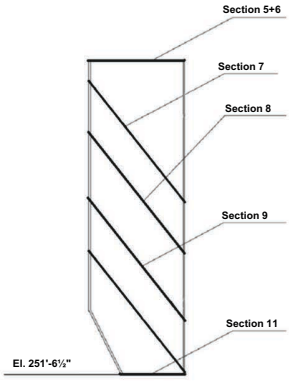
Table 3H.5-9 (Sheet 2b of 3)
Shield Building Roof Reinforcement Summary
(Air Inlet)

AIS Reinforcement Summary – Vertical Sections						
Locations (Figure 3H.5-11)		**Steel Area (Hoop Direction – Y Local Dir.)**				
		Required - Seismic Load Combinations (in^2/ft)		**Maximum Required (in^2/ft)**	[***Provided***]*	[***Design Limit for Ratio Max Required/ Provided***]*
Sections	**Angles**	**Seismic L/C**	**Values(3)**			
3 Upper	0°	33	11.66	11.91	[*Liner 1" = 12 (in^2/ft) (Min.)*]*	[*0.99*]*
	90°	41	11.00			
3 Lower	0°	33	11.66			
	90°	41	11.00			
4 Upper	5.625°	33	11.91			
	84.375°	41	11.16			
4 Lower	5.625°	33	11.91			
	84.375°	41	11.16			





Notes:
1. Not used.
2. Not used.
3. The values reflect averaged results between the upper and lower sections.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-9 Sheet 2c of 3)
(Shield Building Roof Reinforcement Summary
(Air Inlet)

Out of Plane Shear Reinforcement Summary – AIS							
Locations (Figure 3H.5-11)		**Required – Seismic Load Combinations (in^2/ft)**			**Maximum Required (in^2/ft)**	**[*Steel Area Provided*]***	**[*Design Limit(1) for Ratio Max Required/ Provided*]***
Angles	**Sections**	**Seismic L/C**	**Values**	**Sum**			
0° - 5.625°	Max of Vertical Sections 3 upper - 4 upper	25	0.16	0.16	0.22	*[(3) 3/4" TIE BAR @2.8125° (41.36") (8 1/2" in vertical direction) = 0.50 (in^2/ ft) (Min.)]**	*[0.68 + 2%]**
	Circumferential Section 5+6		0.00				
84.375° - 90°	Max of Vertical sections 3 upper - 4 upper	25	0.16	0.16			
	Circumferential Section 5+6		0.00				
0° - 5.625°	Max of Vertical Sections 3 upper – 4 upper	25	0.16	0.16			
	Circumferential Section 7		0.00				
84.375° - 90°	Max of Vertical Sections 3 upper – 4 upper	25	0.16	0.16			
	Circumferential Section 7		0.00				
0° - 5.625°	Max of Vertical Sections 3 lower - 4 lower	18	0.22	0.22			
	Circumferential Section 9		0.00				
84.375° - 90°	Max of Vertical Sections 3 lower - 4 lower	11	0.22	0.22			
	Circumferential Section 9		0.00				
0° - 5.625°	Max of Vertical Sections 3 lower - 4 lower	18	0.22	0.22			
	Circumferential Section 11		0.00				
84.375° - 90°	Max of Vertical Sections 3 lower - 4 lower	11	0.22	0.22			
	Circumferential Section 11		0.00				

Notes:
1. [*The two percent of the ratio value that may be added to the design limit represents an allowance for minor variances in analysis results.*]*
2. Not used.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-9 (Sheet 3 of 3)
Shield Building Roof Reinforcement Summary
(Exterior Wall Of Passive Containment Cooling System Tank)

Wall Segment	Location (Figure 3H.5-11 Sheet 5 of 6)	Reinforcement on Each Face, in^2/ft			Ratio Required/ Provided
		Maximum Required	Provided (Minimum)		
Bottom	Vertical	1.53	1#8@1.125° & 1#11@1.125°	[*2.75*	0.56
	Hoop	1.22	1#9@6"	*2*	0.61
	Shear	0.11	1#4@0.5625°x6"	*0.94*	0.12
Mid-height	Vertical	0.69	1#8@1.125° & 1#11@1.125°	*2.75*	0.25
	Hoop	1.99	1#9@6"	*2*	0.99
Top	Vertical	0.34	1#8@1.125° & 1#11@1.125°	*2.75*	0.12
	Hoop	0.58	1#9@6"	*2*]*	0.29

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Table 3H.5-10
Design Summary Of Roof At Elevation 180′-0″, Area 6
(Near Shield Building Interface)

Governing Load Combination (Roof Girder)	
Combination Number	7 – Abnormal Condition
Bending Moment	= 8028 kip-ft
Corresponding Stress	= 27.1 ksi
Allowable Stress	= 53.0 ksi
Shear Force	= 553 kips
Corresponding Stress	= 11.5 ksi
Allowable Stress	= 26.4 ksi
Governing Load Combination (Concrete Slab)	
Parallel to Girders	
Combination Numbers	7 – Abnormal Condition
Reinforcement (Each Face)	
Required	= 2.54 in^2/ft
[*Provided*	*= 2.54 in^2/ft (Minimum)*]*
Perpendicular to Girders	
Combination Numbers	3 – Extreme Environmental Condition
Reinforcement (Each Face)	
Required	= 2.37 in^2/ft
[*Provided*	*= 3.12 in^2/ft (Minimum)*]*

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Table 3H.5-11
Design Summary Of Floor At Elevation 135′-3″ Area 1 (Between Column Lines M And P)

Governing Load Combination (Steel Beam)	
Load Combination	3 – Extreme Environmental Condition Downward Seismic
Bending Moment	=(-) 41.1 kips-ft
Corresponding Stress	= 13.9 ksi
Allowable Stress	= 38.0 ksi
Shear Force	= 22.1 kips
Corresponding Stress	= 6.2 ksi
Allowable Stress	= 20.1 ksi
Governing Load Combination (Concrete Slab)	
Parallel to the Beams	
Load Combination	6 – Abnormal Condition
Bending Moment	=(-) 7.6 kips-ft/ft
In-plane Shear	= 31.0 kips (per foot width of the slab)
Reinforcement (Each Face)	
Required	= 0.43 in^2/ft
[*Provided*	*= 0.44 in^2/ft (Min.)*]*
Perpendicular to the Beams	
Combination Number	6 – Abnormal Condition
Bending Moment	=(-) 1.42 kips-ft (per foot width of the slab)
Reinforcement (Each Face)	
Required	= 0.43 in^2/ft
[*Provided*	*= 0.60 in^2/ft (Min.)*]*

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Table 3H.5-12
Design Summary Of Floor At Elevation 135′-3″
(Operations Work Area (Previously Known As 'Tagging Room') Ceiling))

Design of Precast Concrete Panels	
Governing Load Combination	Construction
Design Bending Moment (Midspan)	= 12.07 kip-ft/ft
Bottom Reinforcement (E/W Direction)	
Required(1)	= 0.49 in^2/ft
[*Provided*	*= 0.79 in^2/ft (Min.)*]*
Top Reinforcement (E/W Direction)	
Required(1)	= (Minimum required by Code)
[*Provided*	*= 0.20 in^2/ft (Min.)*]*
Top and Bottom Reinforcement (N/S Direction)	
Required(1)	= (Minimum required by Code)
[*Provided*	*= 0.20 in^2/ft (Min.)*]*
Design of 24-inch-Thick Slab	
Governing Load Combination	Abnormal Condition
Design Bending Moment (E/W Direction) Midspan	= 16.23 kips-ft/ft
Design In-plane Shear	= 30.19 kips/ft
Design In-plane Tension	= 37.1 kips/ft
Bottom Reinforcement (E/W Direction)	
Required(1)	= 0.71 in^2/ft
[*Provided*	*= 0.79 in^2/ft (Min.)*]*
Design Bending Moment (E/W Direction) at Support	= 17.75 kips-ft/ft
Design In-plane Shear	= 30.19 kips/ft
Design In-plane Tension	= 37.1 kips/ft
Top Reinforcement (E/W Direction)	
Required(1)	= 0.73 in^2/ft
[*Provided*	*= 1.00 in^2/ft (Min.)*]*
Design Bending Moment (N/S Direction)	= 21.55 kips-ft/ft
Design In-plane Shear	= 30.19 kips/ft
Design In-plane Tension	= 24.65 kip/ ft
Top and Bottom Reinforcement (N/S Direction)	
Required(1)	= 0.65 in^2/ft
[*Provided*	*= 0.79 in^2/ft (Min.)*]*

Note:

1. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and thermal loads are numerically combined as the thermal loads were assessed to be insignificant. When the seismic and thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

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Table 3H.5-13
Design Summary Of Floor At Elevation 135′-3″ Area 2 (Main Control Room Ceiling)

The design of the bottom plate with fins is governed by the construction (wet concrete) load.
For the composite floor, the design forces used for the evaluation of a 9-inch-wide strip of the slab are as follows: Maximum bending moment = +35.0 (-47.0) kips-ft Maximum shear force = 22.3 kips
The key structural capacities of a 9-inch wide strip of the slab are as follows: (1) • [*The actual area of the tension steel (bottom plate and fins) is 9.0 in^2 (Min.), except in the area of penetrations, openings, or other obstructions,*]* which provides a positive bending moment capacity of 518.5 kips-ft. • The minimum provided area of tension steel (top reinforcement) is 0.59 in^2, except in the area of penetrations, openings, or other obstructions, which provides a negative bending moment capacity of 56 kips-ft. • [*The design shear strength, not reduced for in-plane axial forces, is 23.22 kips.* • *The shear studs have a maximum design spacing of 9 inches c/c, in both directions except in the area of penetrations, openings, or other obstructions.*]*

Note:

1. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

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Table 3H.5-14 (Sheet 1 of 3)
Design Summary of Enhanced Shield Building Cylindrical Wall Load Combinations, and Comparison to Acceptance Criteria Elevation 180 Feet Near Fuel Handling Building Roof

Load/Combination	TX kip/ft	TY kip/ft	TXY kip/ft	MX k-ft/ft	MY k-ft/ft	MXY k-ft/ft	NX kip/ft	NY kip/ft	Comments
Dead	-7	-118	15	-25	-18	4	-6	-5	
Live	1	-1	1	0	0	0	0	0	
Seismic	157	390	163	301	211	35	71	33	
1	-8	-168	22	-35	-24	5	-8	-7	1.4 D + 1.7 L
2	152	270	179	276	194	38	66	28	D + L + Es
3	152	270	-148	-326	-228	-31	-77	-38	D + L + E's
4	-163	-509	-148	-326	-228	-31	-77	-38	D + L - Es
5	-163	-509	179	276	194	38	66	28	D + L - E's
6	152	283	177	278	195	38	66	28	0.9 D + Es
7	152	283	-150	-324	-227	-31	-77	-38	0.9 D + E's
35(2)	205	399	233	449	303	69	105	31	0.9 D + E's + αTo(W1)
37(2)	221	387	238	460	311	68	108	31	0.9 D + E's + αTo(W2)

x-direction is horizontal; y-direction is vertical.

Element number:	12164
[*Plate thickness required for load combinations excluding thermal:*	*0.50 inches + 2%(1)*]*
[*Plate thickness required for load combinations including thermal:*	*0.65 inches + 2%(1)*]*
[*Plate thickness provided:*	*0.75 inches*]*(3)
[*Shear reinforcement required for load combinations excluding thermal:*	*0.72 in^2/ft^2 + 2%(1)*]*
[*Shear reinforcement required for load combinations including thermal:*	*1.00 in^2/ft^2 + 2%(1)*]*
Shear reinforcement provided:	See [*APP-GW-GLR-602, Section 4.*]*

Notes:

1. [*The Tier 2* designation for "Plate thickness required" requires NRC approval if this value is exceeded as a result of design changes or detail design adjustments identified during preparation of fabrication or construction drawings or instructions.*]*
2. Load cases 35 and 37 are the two governing load combinations for element 12164 that include thermal and seismic loads combined numerically. W1 designates the winter conditions with the spent fuel pool at the normal operating temperature limit. W2 designates the winter conditions with the spent fuel pool and fuel transfer canal at the normal operating temperature limit. Es is SRSS (member forces are positive) of the SSE loads. E's is Es with all member forces except axial forces (TX, TY) reversed to negative.
3. The 0.75-inch plate thickness is the nominal plate size for the shield building away from connections, attachments, and other local loads. The plate may be thicker (up to 1.0-inch nominal thickness) in the area around these local loads.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-14 (Sheet 2 of 3)
Design Summary of Enhanced Shield Building Load Combinations, and Comparison to Acceptance Criteria Elevation 175 Feet Near Intersection With Wall 7.3

Load/Combination	TX kip/ft	TY kip/ft	TXY kip/ft	MX k-ft/ft	MY k-ft/ft	MXY k-ft/ft	NX kip/ft	NY kip/ft	Comments
Dead	-7	-106	12	-6	5	1	0	2	
Live	0	-1	0	0	0	0	0	0	
Seismic	34	327	178	38	25	13	2	8	
1	-9	-150	17	-9	7	1	0	3	1.4 D + 1.7 L
2	28	220	190	32	30	14	2	10	D + L + Es
3	28	220	-166	-45	-21	-12	-3	-6	D + L + E's
4	-40	-434	-166	-45	-21	-12	-3	-6	D + L - Es
5	-40	-434	190	32	30	14	2	10	D + L - E's
6	28	232	189	33	30	14	2	10	0.9 D + Es
7	28	232	-167	-44	-21	-12	-3	-7	0.9 D + E's
19(2)	75	251	168	-39	-107	-20	-2	-5	D + L + E's + αTo(W1)
37(2)	75	251	168	-39	-107	-20	-2	-5	0.9 D + E's + αTo(W2)
x-direction is horizontal; y-direction is vertical.									
Element number:						11514			
[*Plate thickness required for load combinations excluding thermal:*						*0.45 inches + 2%(1)*]*			
[*Plate thickness required for load combinations including thermal:*						*0.50 inches + 2%(1)*]*			
[*Plate thickness provided:*						*0.75 inches*]*(3)			
[*Shear reinforcement required for load combinations excluding thermal:*						*0.09 in^2/ft^2 + 2%(1)*]*			
[*Shear reinforcement required for load combinations including thermal:*						*0.12 in^2/ft^2 + 2%(1)*]*			
Shear reinforcement provided:						See [*APP-GW-GLR-602, Section 4.*]*			

Notes:

1. [*The Tier 2* designation for "Plate thickness required" requires NRC approval if this value is exceeded as a result of design changes or detail design adjustments identified during preparation of fabrication or construction drawings or instructions.*]*
2. Load cases 19 and 37 are the two governing load combinations for element 11514 that include thermal and seismic loads combined numerically. W1 designates the winter conditions with the spent fuel pool at the normal operating temperature limit. W2 designates the winter conditions with the spent fuel pool and fuel transfer canal at the normal operating temperature limit. Es is SRSS (member forces are positive) of the SSE loads. E's is Es with all member forces except axial forces (TX, TY) reversed to negative.
3. The 0.75-inch plate thickness is the nominal plate size for the shield building away from connections, attachments, and other local loads. The plate may be thicker (up to 1.0-inch nominal thickness) in the area around these local loads.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 3H.5-14 (Sheet 3 of 3)
Design Summary of Enhanced Shield Building Load Combinations, and Comparison to Acceptance Criteria Elevation Grade on West Side

Load/Combination	TX kip/ft	TY kip/ft	TXY kip/ft	MX k-ft/ft	MY k-ft/ft	MXY k-ft/ft	NX kip/ft	NY kip/ft	Comments
Dead	-4	-129	0	2	16	0	0	-2	
Live	0	1	0	0	0	0	0	0	
Seismic	80	489	234	2	20	18	4	8	
1	-5	-179	0	3	23	0	0	-2	1.4 D + 1.7 L
2	77	362	234	5	36	19	4	7	D + L + Es
3	77	362	-234	0	-4	-18	-4	-10	D + L + E's
4	-84	-617	-234	0	-4	-18	-4	-10	D + L - Es
5	-84	-617	234	5	36	19	4	7	D + L - E's
6	77	374	234	5	34	19	4	7	0.9 D + Es
7	77	374	-235	0	-5	-18	-4	-10	0.9 D + E's
23(2)	202	377	-240	113	151	-18	-4	-32	D + L + E's + αTo(W1)
41(2)	203	393	-240	112	149	-17	-4	-32	0.9 D + E's + αTo(W2)

x-direction is horizontal; y-direction is vertical.

Element number:	23752
[*Plate thickness required for load combinations excluding thermal:*	*0.65 inches + 2%(1)*]*
[*Plate thickness required for load combinations including thermal:*	*0.66 inches + 2%(1)*]*
[*Plate thickness provided:*	*0.75 inches*]*(3)
[*Shear reinforcement required for load combinations excluding thermal:*	*0.08 in^2/ft^2 + 2%(1)*]*
[*Shear reinforcement required for load combinations including thermal:*	*0.26 in^2/ft^2 + 2%(1)*]*
Shear reinforcement provided:	See [*APP-GW-GLR-602, Section 4.*]*

Notes:

1. [*The Tier 2* designation for "Plate thickness required" requires NRC approval if this value is exceeded as a result of design changes or detail design adjustments identified during preparation of fabrication or construction drawings or instructions.*]*
2. Load cases 23 and 41 are the two governing load combinations for element 23752 that include thermal and seismic loads combined numerically. W1 designates the winter conditions with the spent fuel pool at the normal operating temperature limit. W2 designates the winter conditions with the spent fuel pool and fuel transfer canal at the normal operating temperature limit. Es is SRSS (member forces are positive) of the SSE loads. E's is Es with all member forces except axial forces (TX, TY) reversed to negative.
3. The 0.75-inch plate thickness is the nominal plate size for the shield building away from connections, attachments, and other local loads. The plate may be thicker (up to 1.0-inch nominal thickness) in the area around these local loads.

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Table 3H.5-15
Shield Building Roof Reinforcement Ratio of Code Required Versus Provided

Critical Sections	Stress Component	Required in^2/ft	Provided (Minimum) in^2/ft	Reinforcement Ratio
[*Conical Roof Beams*]* (1)	Axial + Bending	-	[*Radial Beams W36 X 395*(2)]*	1.65
	Shear	-		3.85
[*Conical Roof Near Tension Ring*]*	Radial	1.81	[*2.34*]*	1.29
	Hoop	4.01	[4.68]*	1.16
[*Knuckle Region*]*	Vertical	1.53	[*2.75*]*	1.80
	Radial	3.48(3)	[*3.55*]*	1.02
	Hoop	2.67	[*3.12*]*	1.16
[*Compression Ring*]*	Vertical	1.09	[*2.40*]*	2.20
	Radial	2.08	[*3.56*]*	1.71
	Hoop	2.50	[*3.12*]*	1.25

Note:
1. Roof beams are not considered as reinforcement for the reinforced concrete roof. Ratio for conical roof beams is based on demand and allowable stresses in psi.
2. Built-up plate girders can be substituted for rolled shape. The section dimensions of the built-up plate girders are specified in Figure 3H.5-11 Sheet 1.
3. The value reflects averaged result between different angles of the stress line.

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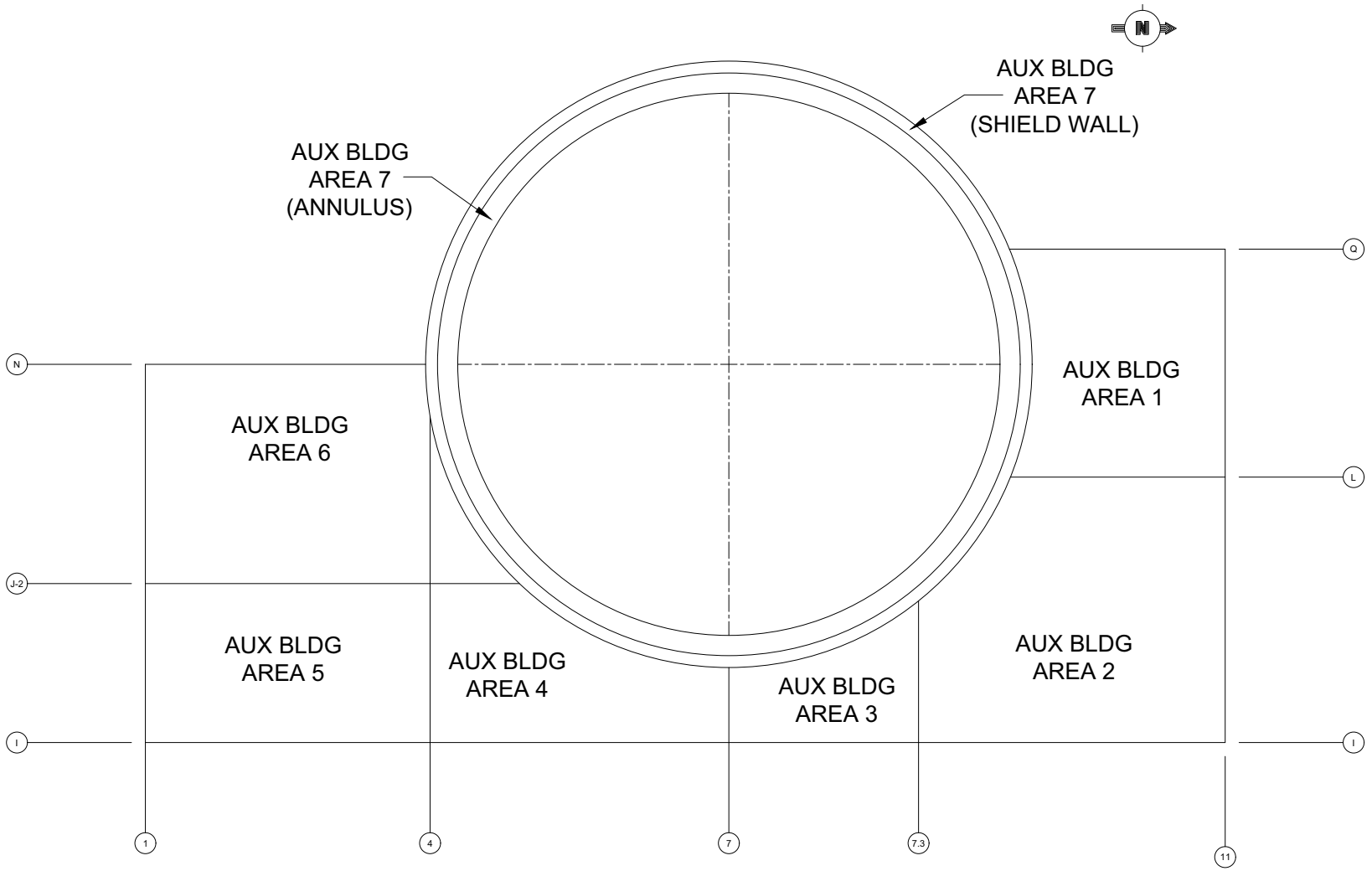

Figure 3H.2-1
[*General Layout of Auxiliary Building*]*

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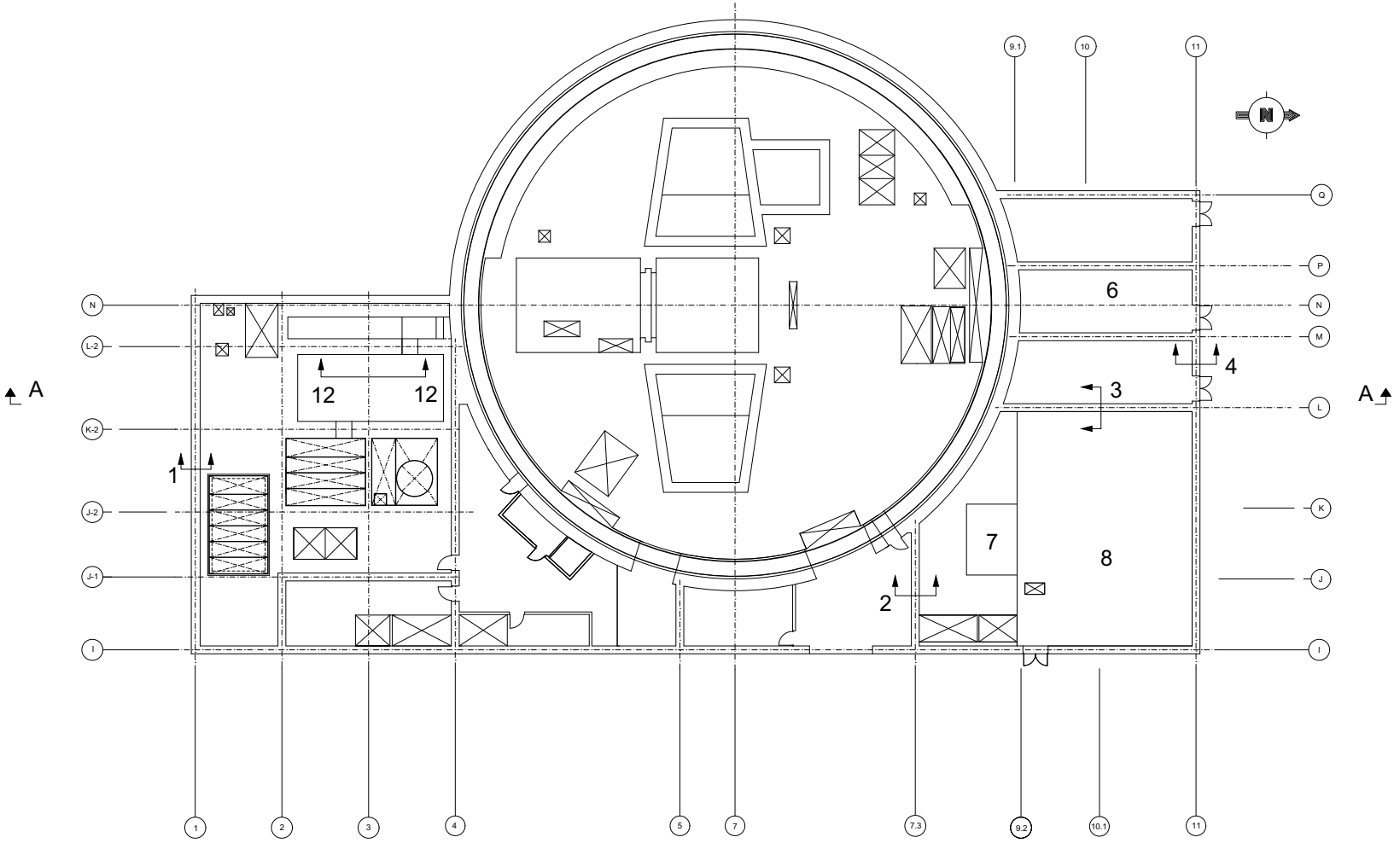

Figure 3H.5-1 (Sheet 1 of 3)
[*Nuclear Island Critical Sections Plan at El. 135'-3"*]*

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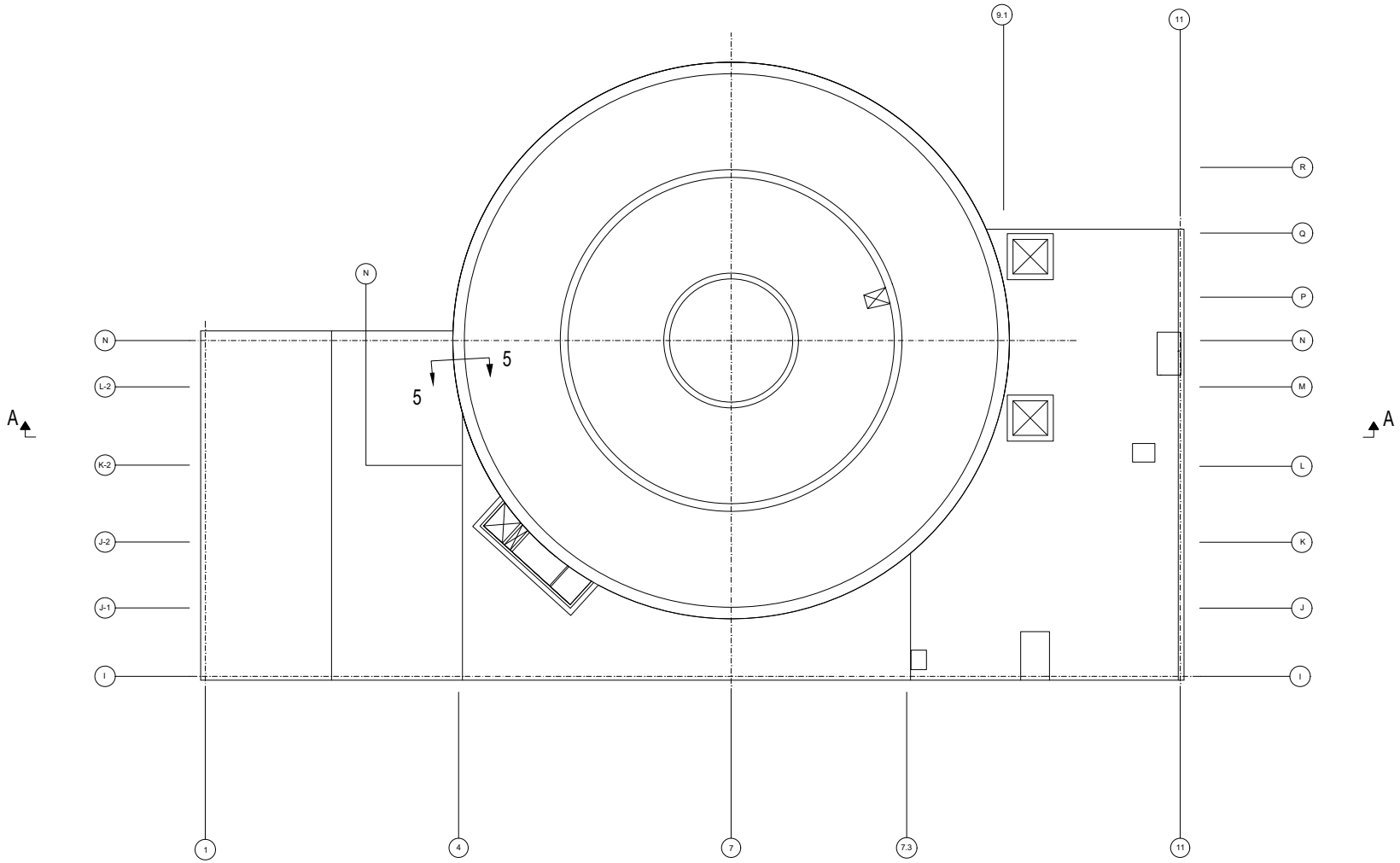

Figure 3H.5-1 (Sheet 2 of 3)
[*Nuclear Island Critical Sections Plan at El. 180'-0"*]*

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Figure 3H.5-1 (Sheet 3 of 3)
**[*Nuclear Island Critical Sections*
Section A-A*]

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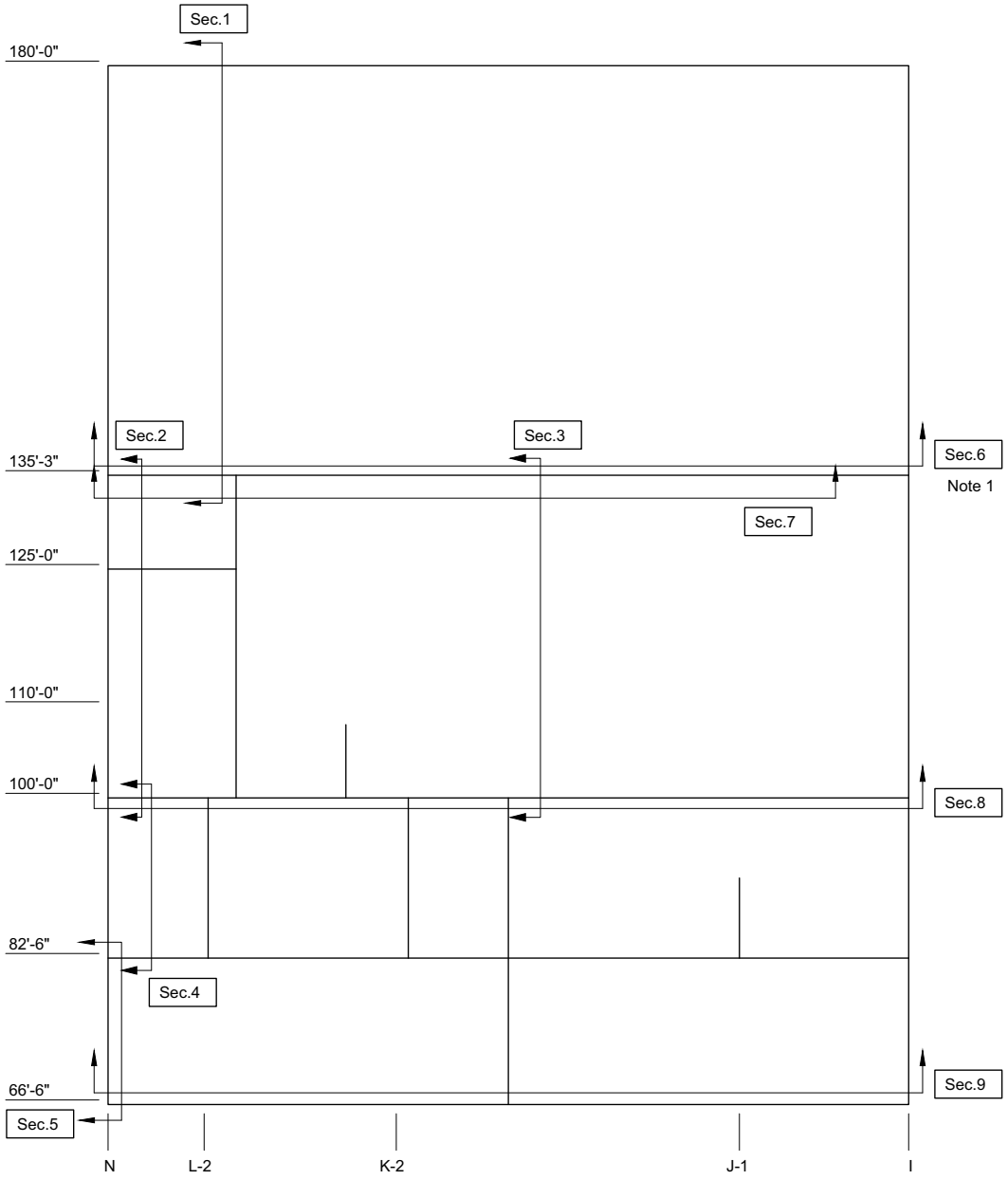

Figure 3H.5-2 (Sheet 1 of 3)
[*Wall on Column Line 1*]*

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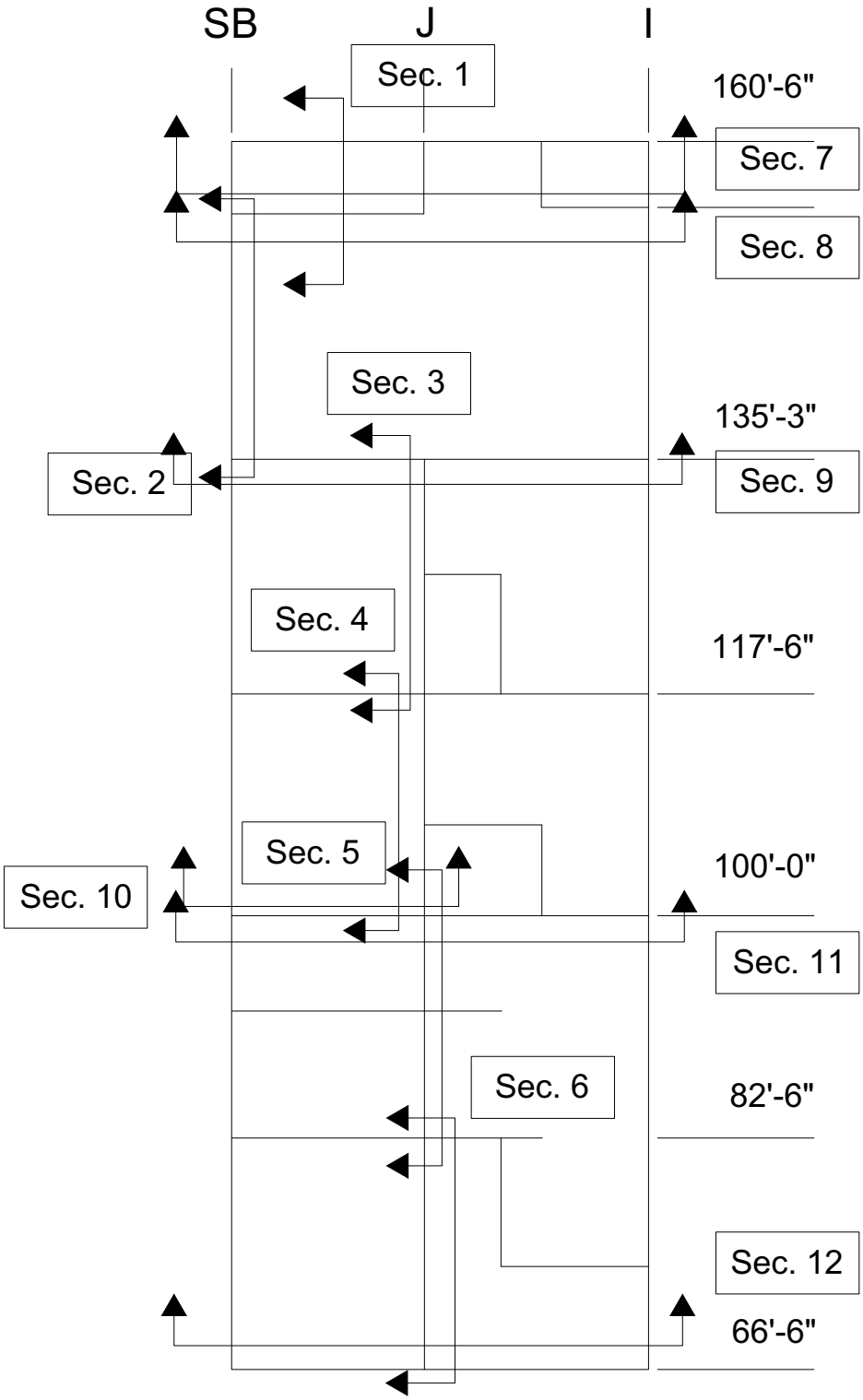

Figure 3H.5-2 (Sheet 2 of 3)
[*Wall on Column Line 7.3*]*

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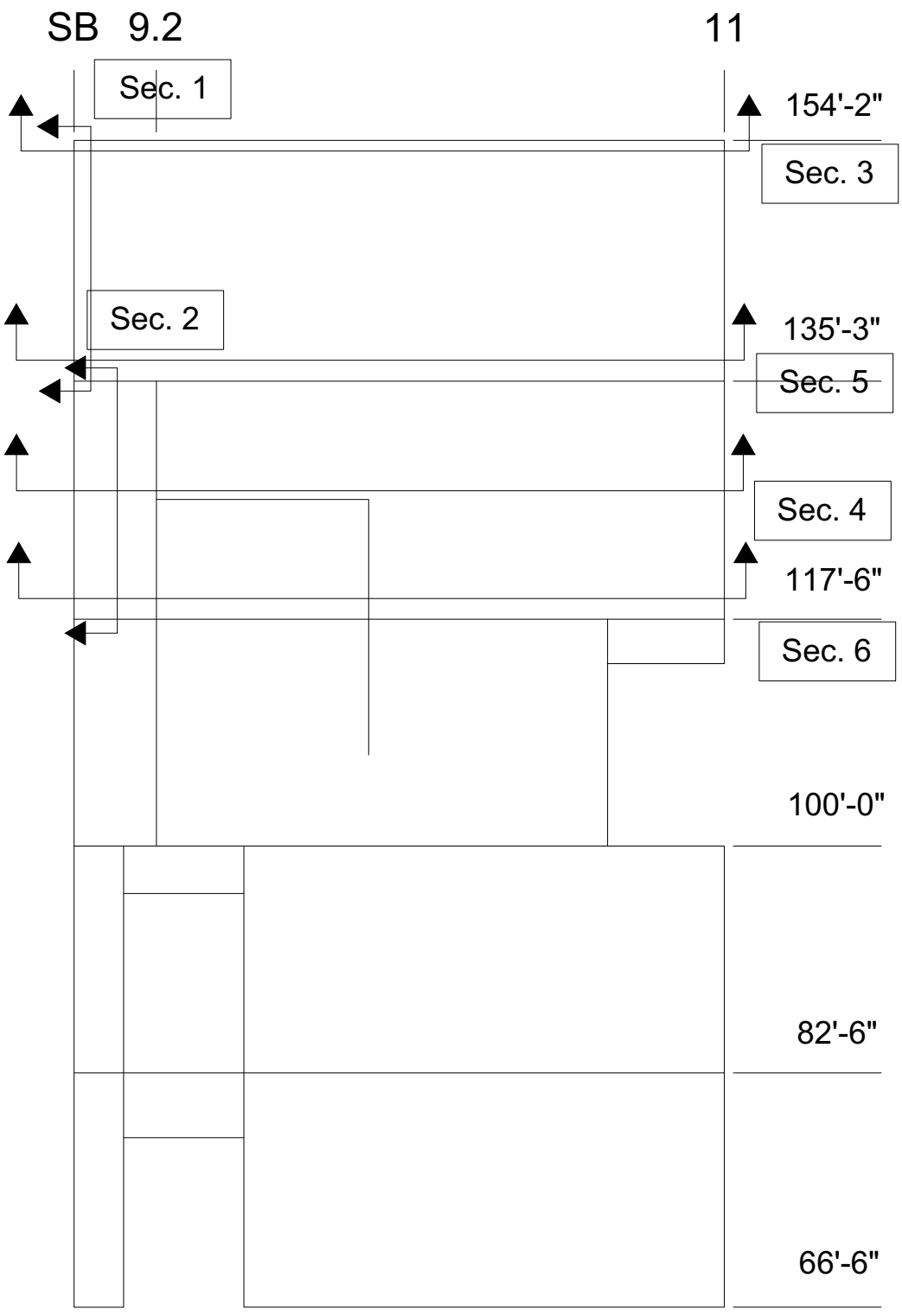

Figure 3H.5-2 (Sheet 3 of 3)
[*Wall on Column Line L*]*

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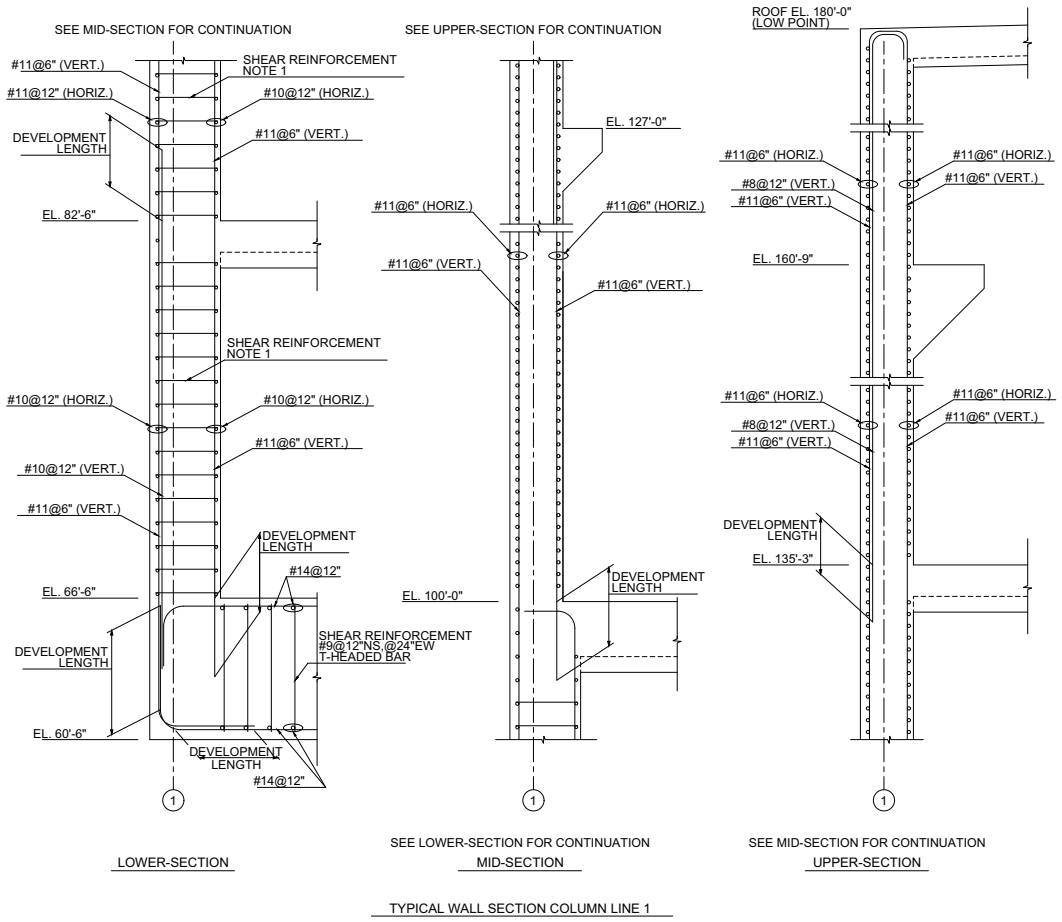

NOTE 1:
REFER TO SUBSECTION 3.8.4.4.1 FOR THE REQUIREMENTS FOR SHEAR REINFORCEMENT AND TABLE 3H.5-3 FOR SHEAR REINFORCEMENT PROVIDED.

Figure 3H.5-3
[*Typical Reinforcement in Wall on Column Line 1*]*

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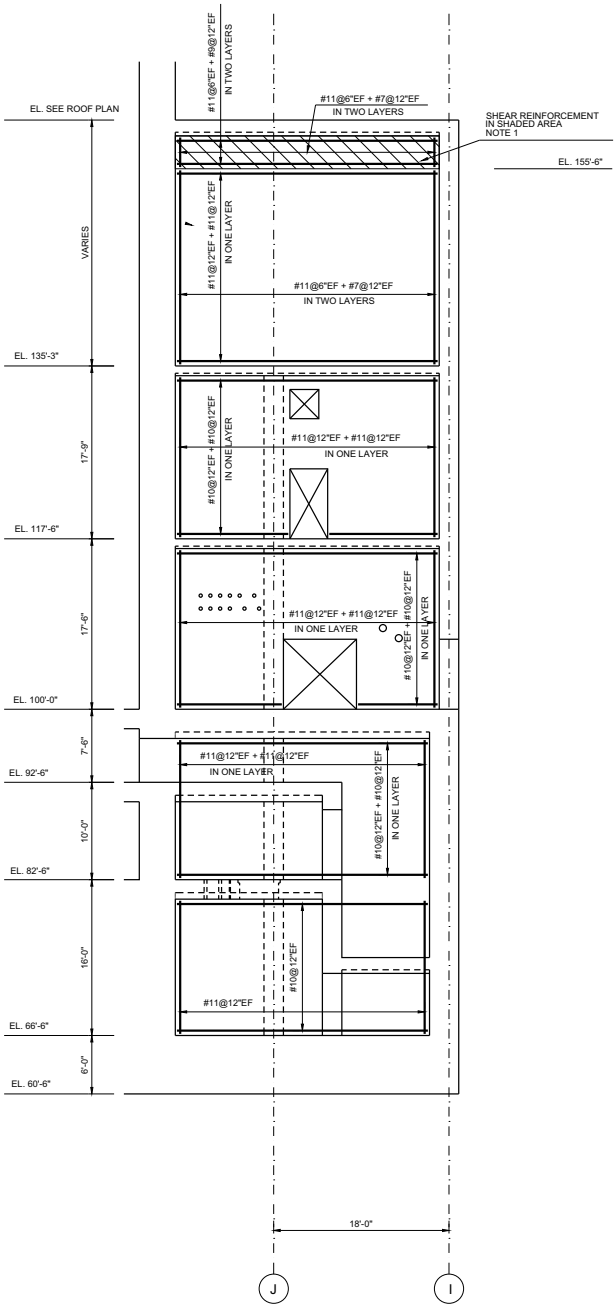

WALL 7.3

NOTE 1:
REFER TO SUBSECTION 3.8.4.4.1 FOR THE REQUIREMENTS FOR SHEAR REINFORCEMENT AND TABLE 3H.5-5 FOR SHEAR REINFORCEMENT PROVIDED.

Figure 3H.5-4
[*Typical Reinforcement in Wall 7.3*]*

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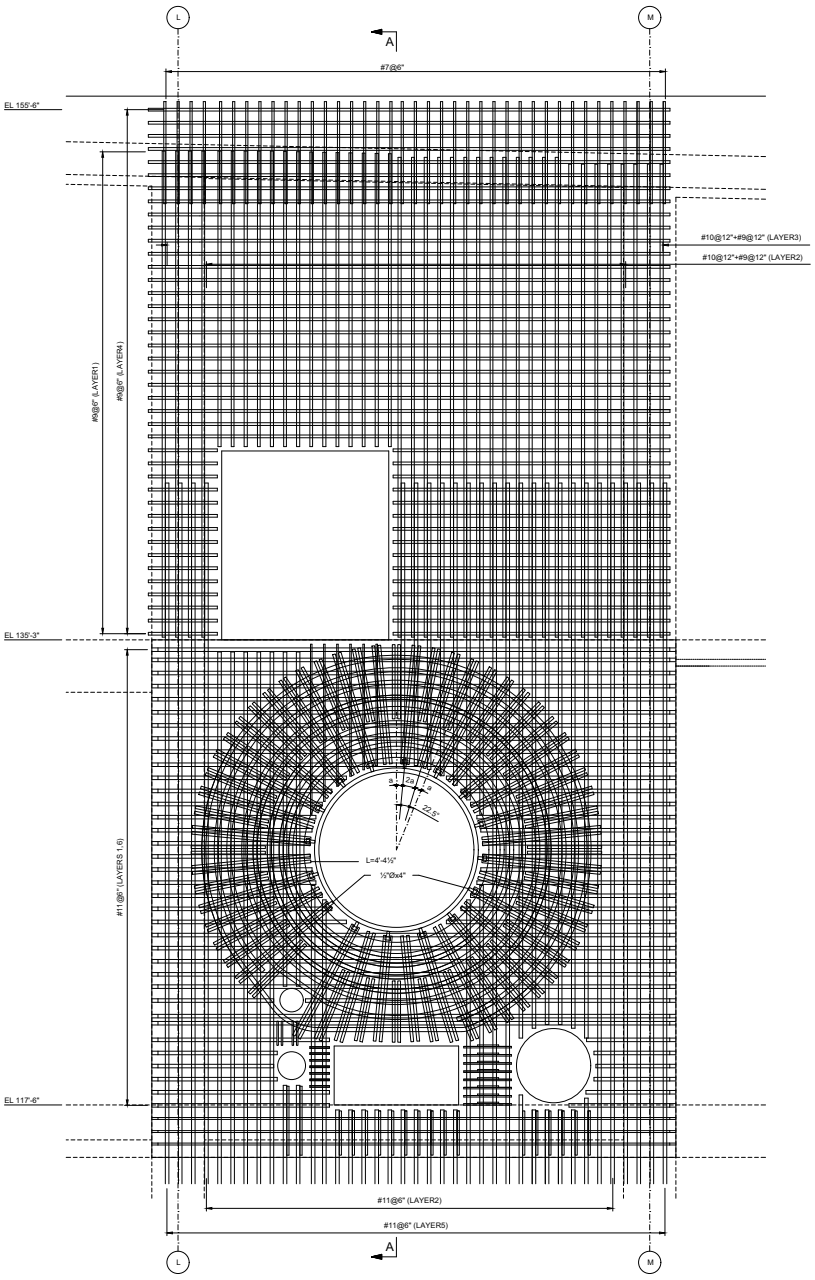

NOTES:

1. DETAIL SHOWN IS SPECIFIC TO THE WALL BETWEEN COLUMN LINE L AND COLUMN LINE M. REFER TO SUBSECTION 3H.5.1.4 AND OTHER NOTES FOR ADDITIONAL INFORMATION ABOUT DESIGN DETAILS FOR THE OTHER WALL SECTION.
2. LAYER 1, 2, 5 AND 6 REINFORCEMENT TERMINATING WITHIN THE EXTENT OF THE RADIAL AND CIRCUMFERENTIAL REINFORCEMENT AROUND THE PENETRATION IS DEVELOPED USING STANDARD HOOKS OR HEADED REINFORCEMENT (NOT SHOWN). LAYER 3 AND 4 REINFORCEMENT TERMINATING WITHIN THE 4'-0" THICK SECTION IS DEVELOPED WITH EITHER A FULL STRAIGHT DEVELOPMENT ACCORDING TO CHAPTER 12 AND 21 OF ACI 349, OR A FULL STRAIGHT BAR DEVELOPMENT ACCORDING TO CHAPTER 12 OF ACI 349 WITH ADDITIONAL HEADED ANCHORAGE.
3. REFER TO SUBSECTION 3.8.4.4.1 FOR THE REQUIREMENTS FOR DEVELOPMENT OF HEADED REINFORCEMENT.
4. THE OTHER WALL SECTION FROM COLUMN LINE P TO Q CONTAINING A MAIN STEAM LINE PENETRATION IS ON THE CORNER OF THE AUXILIARY BUILDING, NOT DETAILED IN THIS FIGURE, AND THE DETAILING AND DEVELOPMENT OF THE REINFORCEMENT VARIES CONSISTENT WITH ACI 349 REQUIREMENTS FOR AN ADJACENT WALL AT A CORNER.
5. SOME DETAILING ASSOCIATED WITH REINFORCEMENT SPLICES AND DEVELOPMENT IS NOT PRESENTED IN THE FIGURE. THE DESIGN OF THIS DETAILING CONFORMS TO THE REQUIREMENTS OF ACI 349.
6. THE CRITICAL SECTION BOUNDARIES ARE DEFINED IN SUBSECTION 3H.5.1.4.
7. FOR CLARITY, NOT ALL REINFORCING BARS PROVIDED FOR THE PURPOSE OF CRACK CONTROL ARE DEPICTED IN THIS FIGURE. THE NUMBER, SIZE, AND PLACEMENT OF REINFORCING BARS FOR CRACK CONTROL MAY DEVIATE FROM THAT DEPICTED IN THIS FIGURE AS NEEDED TO FACILITATE CONSTRUCTION. PLACEMENT OF REINFORCING BARS FOR CRACK CONTROL SHALL BE IN ACCORDANCE WITH ACI 349-01.
8. FOR CLARITY, OUT-OF-PLANE SHEAR REINFORCEMENT IS NOT DEPICTED IN THIS FIGURE. OUT-OF-PLANE SHEAR REINFORCEMENT SHALL BE IN ACCORDANCE WITH ACI 349-01.

Figure 3H.5-5 (Sheet 1 of 3)
[*Concrete Reinforcement in Wall 11*]*

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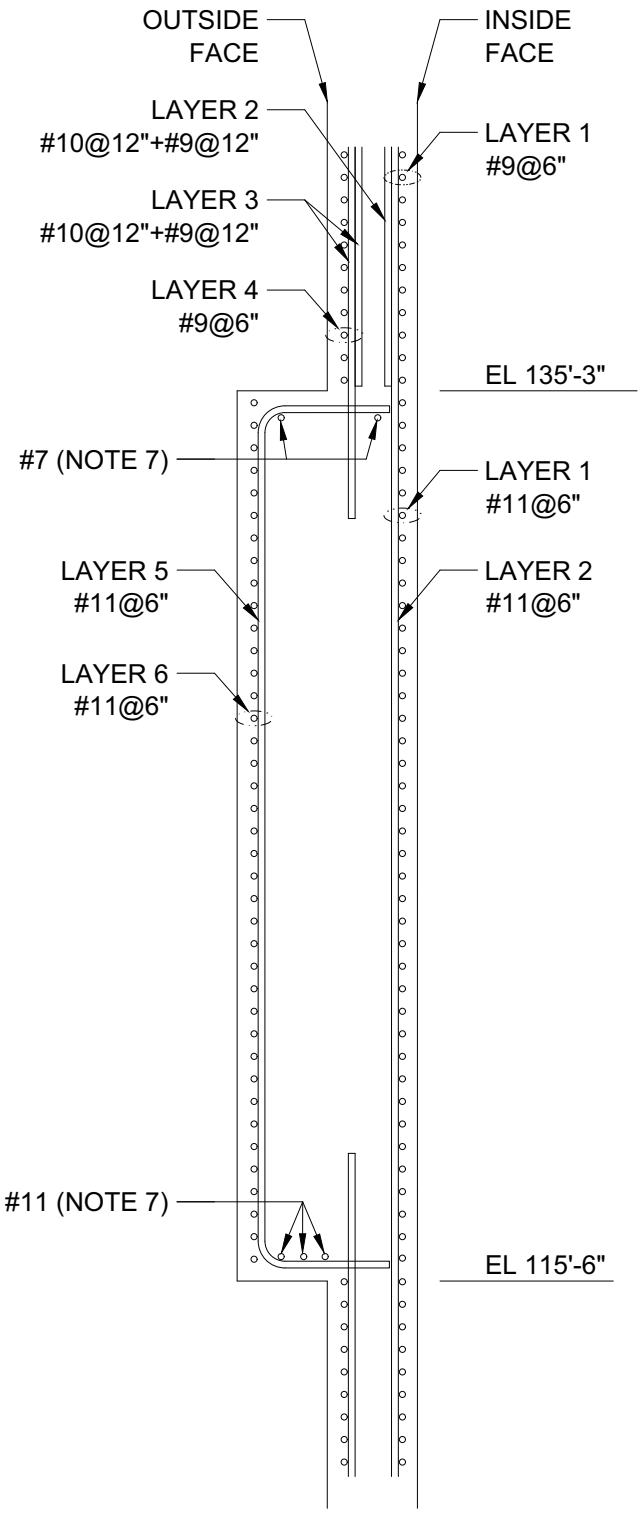

Figure 3H.5-5 (Sheet 2 of 3)
[*Concrete Reinforcement Layers in Wall 11 (Looking East)*]*

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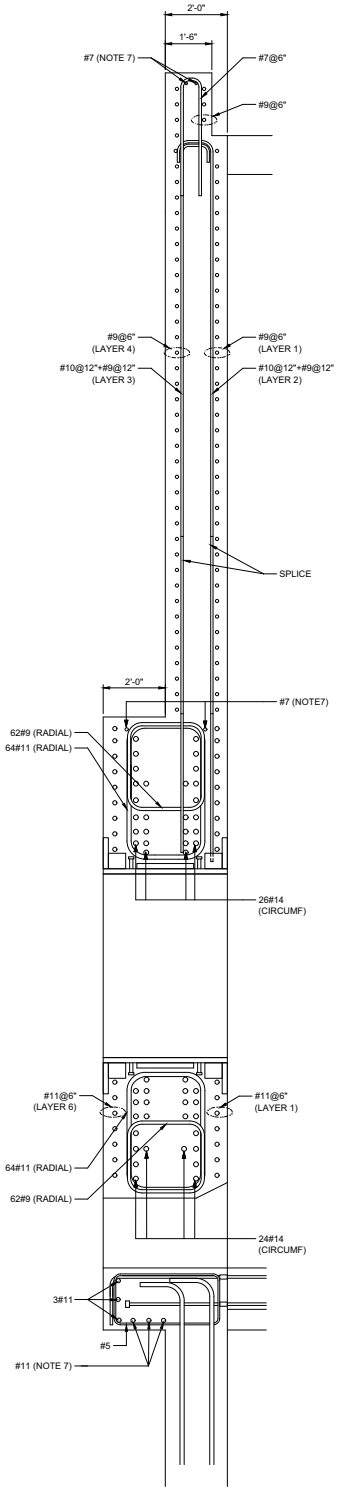

Figure 3H.5-5 (Sheet 3 of 3)
[*Wall 11 at Main Steamline Anchor Section A-A*]*

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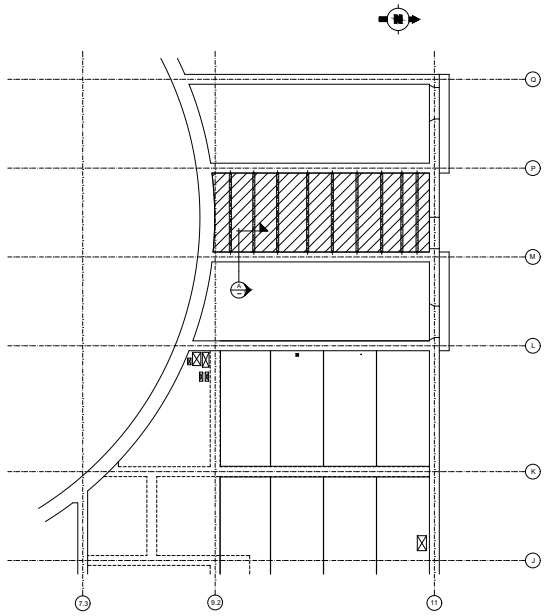
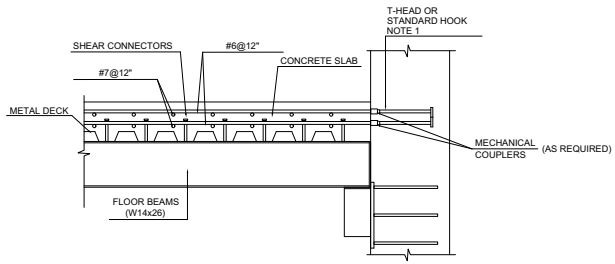


PLAN VIEW
NOTES 1, 3, 6



SECTION A-A
NOTES 1,2,3,4,5,6, & 7

NOTES:

1. DETAIL SHOWN IS SPECIFIC TO THE COMPOSITE FLOOR AT EL. 135'-3" AT INTERSECTION WITH WALL M. REFER TO SUBSECTION 3H.5.2 AND OTHER NOTES FOR ADDITIONAL INFORMATION ABOUT DESIGN DETAILS FOR OTHER FLOOR SECTIONS.

2. REFER TO SUBSECTION 3.8.4.4.1 FOR THE REQUIREMENTS FOR DEVELOPMENT OF HEADED REINFORCEMENT.

3. FLOOR BEAM SIZE AND SPACING ARE DESIGNED BASED ON FLOOR LOAD AND GEOMETRY TO SATISFY AISC N690 REQUIREMENTS. THE BEAM SIZES USED ARE PREDOMINATELY W14x26 AND W14x48. THE RANGE OF BEAM SIZES USED IN OTHER LOCATIONS IS FROM W10 TO W44. THE SPACING BETWEEN THE BEAMS IS PREDOMINATELY IN A RANGE OF 5 TO 6 FEET.

4. THE REINFORCEMENT SHOWN IS FOR LOCATIONS AWAY FROM OPENINGS, PENETRATIONS, EMBEDMENTS, AND OTHER OBSTRUCTIONS.

5. REINFORCEMENT SIZE AND SPACING ARE BASED ON THE REQUIREMENTS IN ACI 349 AND ACI 318-11, SECTION 12.6.

6. THE ADJACENT WALL MAY BE DESIGNED AS A STRUCTURAL WALL MODULE.

7. THE DETAIL DESIGN, LOCATION, AND EMBEDMENT OF THE BEAM SUPPORTS ARE DESIGNED TO THE REQUIREMENTS OF AISC N690 AND ACI 349 AS APPLICABLE. SUPPORT CONFIGURATION, INCLUDING THE USE OF PLATES, STRUCTURAL SHAPES, AND STIFFENERS, IS BASED ON LOADING AND LOCAL GEOMETRY CONSIDERATIONS.THE DESIGN OF EMBEDMENT ANCHORAGE INCLUDING TYPE, SIZE, AND SPACING SATISFIES THE REQUIREMENTS OF APPENDIX B AS WELL AS OTHER APPLICABLE SECTIONS OF ACI 349.

Figure 3H.5-6
[*Auxiliary Building Typical Composite Floor*]*

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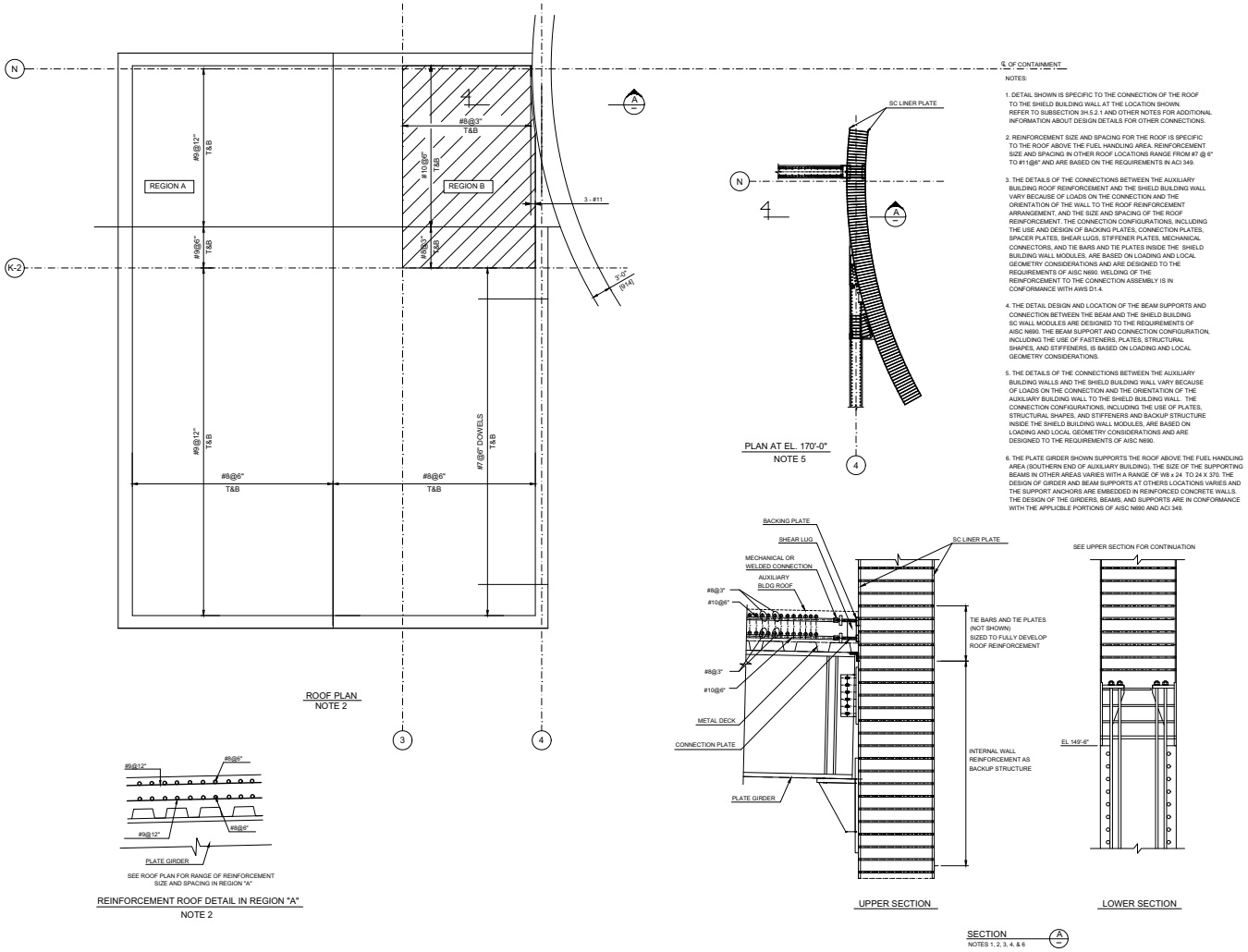

Figure 3H.5-7
[*Typical Reinforcement and Connection to Shield Building*]*

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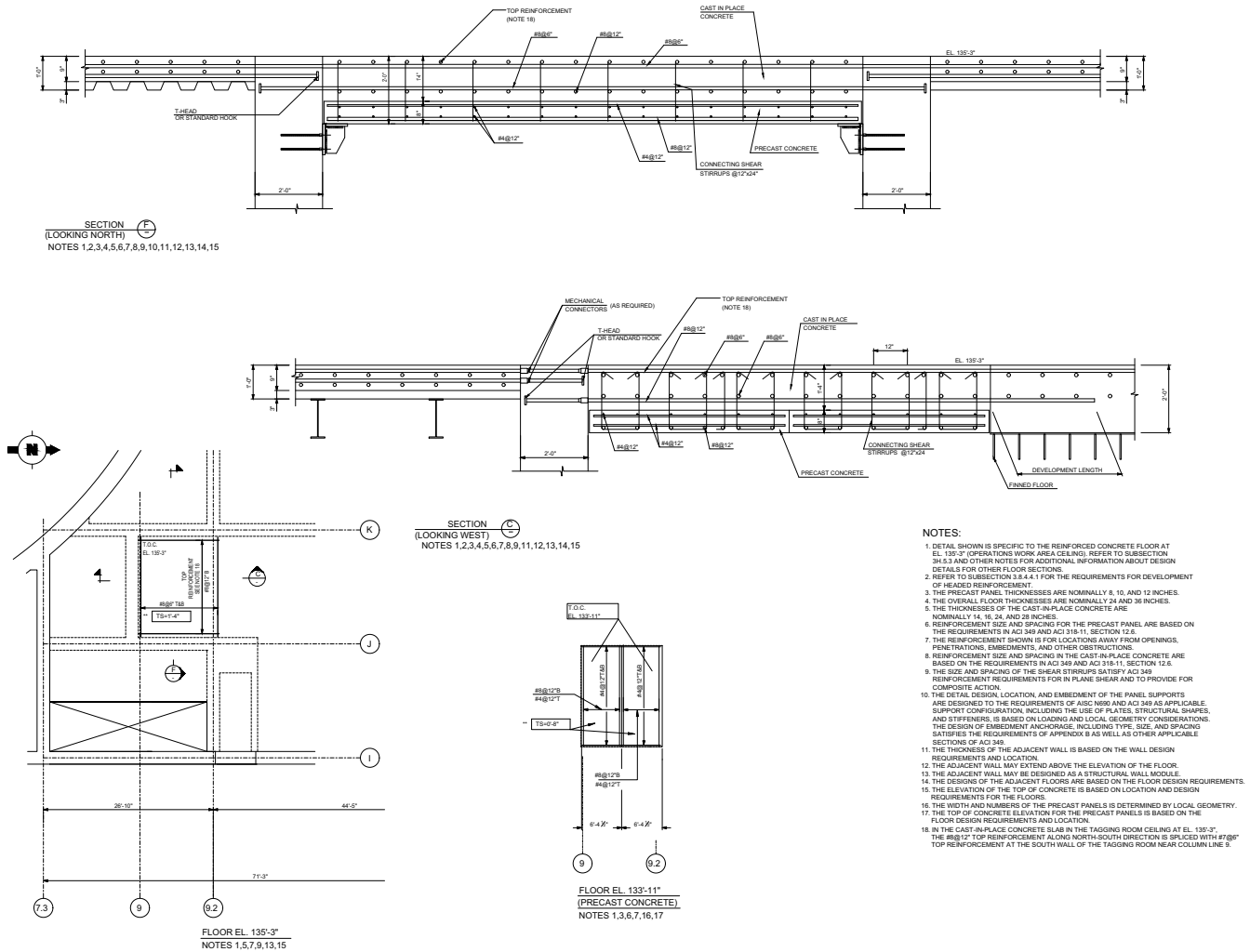

Figure 3H.5-8
[*Auxiliary Building Operations Work Area (Tagging Room) Ceiling*]*

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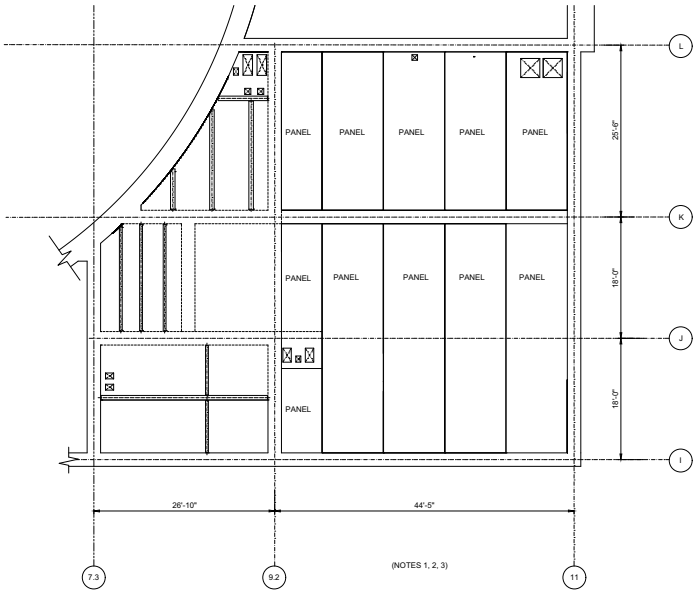

NOTES:

1. DETAIL SHOWN IS SPECIFIC TO THE REINFORCED CONCRETE FLOOR AT EL. 135'-3" (MAIN CONTROL ROOM CEILING). REFER TO THIS AND OTHER NOTES FOR ADDITIONAL INFORMATION ABOUT DESIGN DETAILS FOR OTHER FLOOR SECTIONS AND FOR CONNECTIONS TO OTHER WALLS.
2. THE NUMBER OF STEEL PANELS USED TO CONSTRUCT THE FLOORS IS DETERMINED BY THE SIZE OF THE ROOM AND FABRICATOR CAPABILITIES.
3. THE OPENINGS FOR PIPING, HVAC DUCTS, OR CABLE TRAYS MAY VARY.
4. THE DEVELOPMENT OF THE FLOOR REINFORCEMENT IN THE WALLS CAN BE HEADED REINFORCEMENT INSTEAD OF STANDARD HOOKS. REFER TO SUBSECTION 3.8.4.4.1 FOR THE REQUIREMENTS FOR DEVELOPMENT OF HEADED REINFORCEMENT.
5. THE REINFORCEMENT AND CONNECTION DOWELS SHOWN ARE FOR LOCATIONS AWAY FROM OPENINGS, PENETRATIONS, EMBEDMENTS, AND OTHER OBSTRUCTIONS.
6. REINFORCEMENT SPACING AND SIZE FOR CONNECTING DOWELS ARE BASED ON THE REQUIREMENTS IN ACI 318-11 SECTION 12.6 AND ACI 349. THE RANGE OF SPACING AND SIZE OF THE CONNECTING DOWELS VARIES FROM 6" TO 12" AND FROM #8 TO #11, RESPECTIVELY. IN CERTAIN LOCATIONS SOME CONNECTING DOWELS ARE DEVELOPED INTO ADJACENT REINFORCEMENT CONCRETE FLOORS INSTEAD OF CONNECTING TO HOOKS OR HEADED REINFORCEMENT IN THE WALL. THE HOOK ORIENTATION IN CA20 FLOORS MAY VARY FROM THAT OF FINNED FLOORS.
7. DOWEL LENGTH IS THE LONGEST OF A) ACI 349 REQUIREMENTS FOR SPLICE LENGTH, B) LENGTH TO INCORPORATE SUFFICIENT SHEAR STUDS TO DEVELOP THE CAPACITY OF THE DOWEL PER AISC N690 SHEAR STUD CAPACITIES, OR C) LENGTH TO INCORPORATE SUFFICIENT SHEAR STUDS TO DEVELOP THE DEMAND IN THE BOTTON PLATE, PER AISC N690 SHEAR STUD CAPACITIES.
8. ADDITIONAL BOTTOM LAYER REINFORCING STEEL IS PROVIDED IN THE FINNED FLOORS AT ELEVATION 117'-6" WHERE NEEDED TO MAINTAIN THE STRUCTURAL INTEGRITY OF THE FIRE BARRIER.
9. THE RANGE OF REINFORCEMENT SPACING AND SIZE VARIES FROM 6" TO 12" AND FROM #7 TO #11, RESPECTIVELY.
10. THE ELEVATION OF THE TOP OF CONCRETE IS BASED ON LOCATION AND DESIGN REQUIREMENTS FOR THE FLOOR PLATES.
11. THE DESIGN OF THE FINS VARIES AT LOCATIONS NEAR OPENINGS, PENETRATIONS, AND OTHER OBSTRUCTIONS AND DUE TO ATTACHMENTS TO THE FINS AND FLOOR PLATES.
12. THE CENTER LINE LOCATION OF THE SHEAR STUDS MAY VARY FROM THAT OF THE FINS.
13. THE GAP BETWEEN STEEL PLATE AND WALL, AND USE OF CONSTRUCTION JOINTS VARIES BASED ON FABRICATION AND CONSTRUCTION NEEDS.
14. THE NUMBER OF LAYERS OF TOP REINFORCEMENT AND TOP DOWELS ALONG EACH DIRECTION MAY VARY AS LONG AS THE MINIMUM REQUIRED REINFORCEMENT IS PROVIDED PER ACI 349.
15. THE SHEAR STUD DESIGN SHOWN IS FOR LOCATIONS AWAY FROM OPENINGS, PENETRATIONS, EMBEDMENTS, AND OTHER OBSTRUCTIONS.
16. THE DESIGN INFORMATION SHOWN ON SHEET 3 IS APPLICABLE TO FINNED FLOORS FOR LOCATIONS AWAY FROM OPENINGS, PENETRATIONS, EMBEDMENTS, AND OTHER OBSTRUCTIONS.

Figure 3H.5-9 (Sheet 1 of 3)
[*Auxiliary Building Finned Floor*]*

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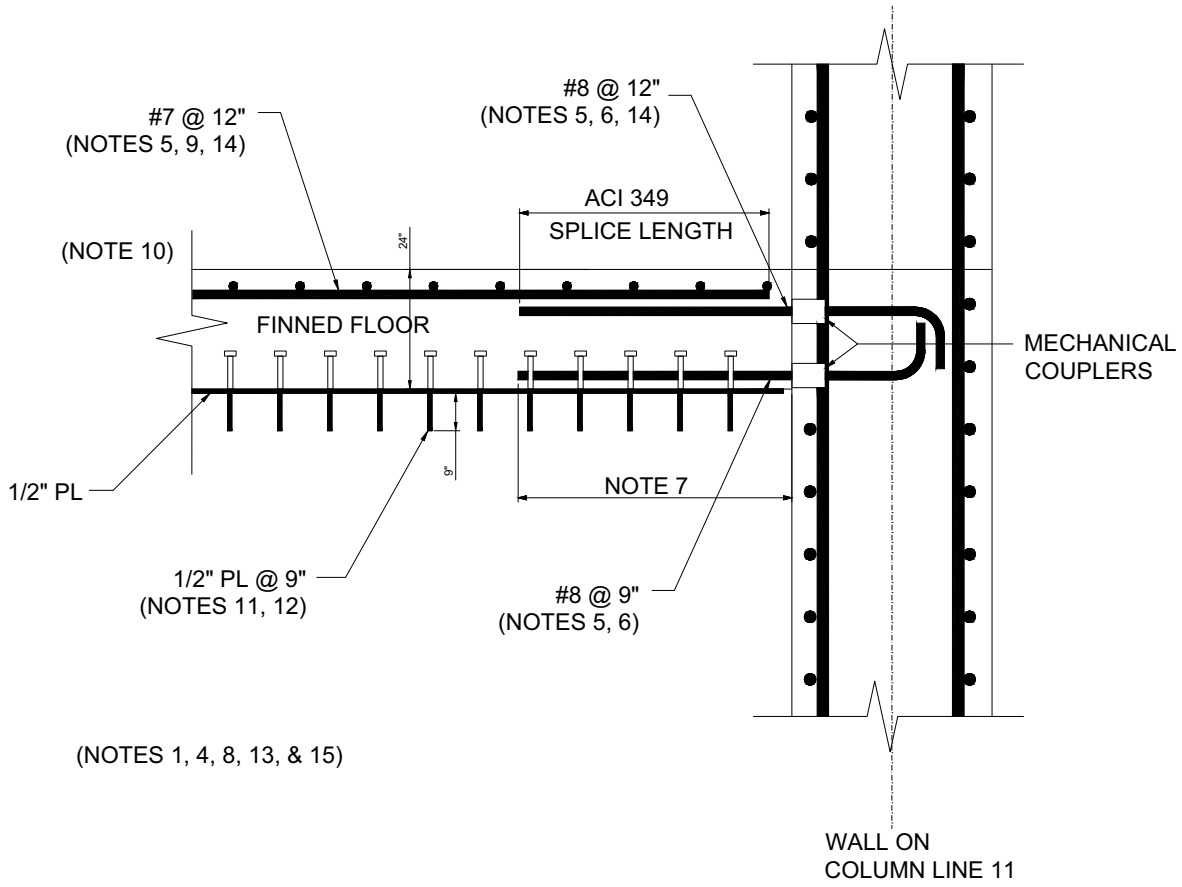

Figure 3H.5-9 (Sheet 2 of 3)
[*Auxiliary Building Finned Floor*]*

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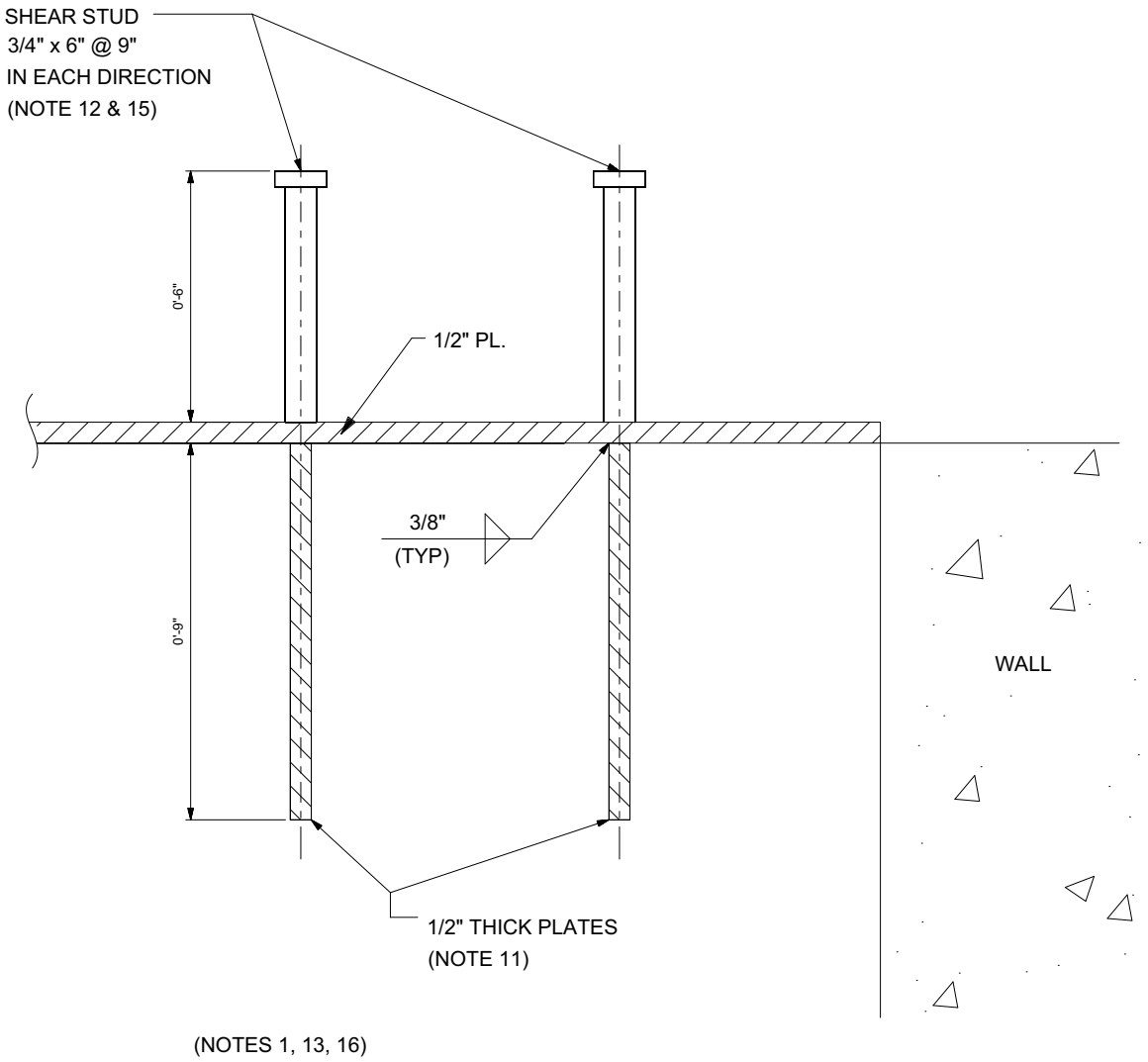

Figure 3H.5-9 (Sheet 3 of 3)
[*Auxiliary Building Finned Floor*]*

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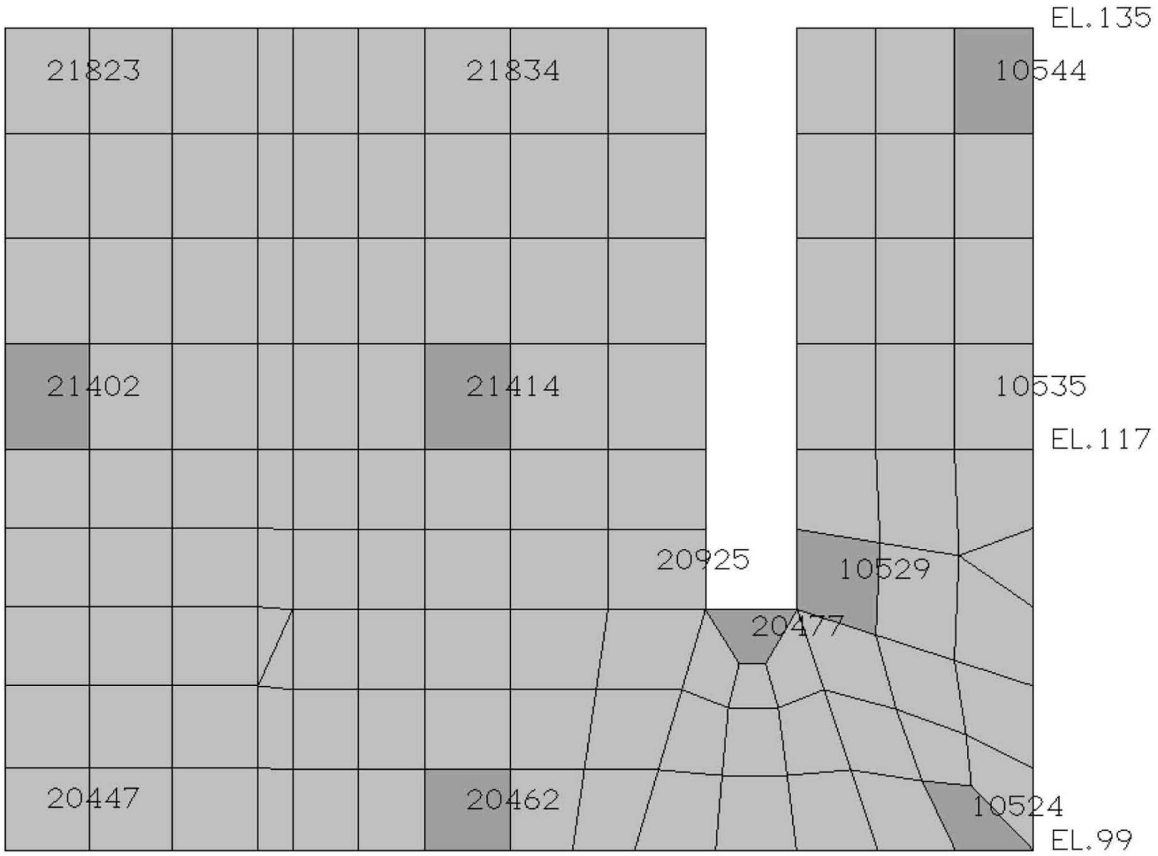

Figure 3H.5-10
[*Spent Fuel Pool Wall Divider Wall Element Locations*]*

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Figure 3H.5-11 (Sheet 1 of 6)
[*Design of Shield Building: Roof and Air Inlets*]*

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Figure 3H.5-11 (Sheet 2 of 6)
[*Design of Shield Building: Concrete Detail at Tension Ring*]*

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Figure 3H.5-11 (Sheet 3 of 6)
[*Design of Shield Building: Roof/Air Inlet Interface*]*

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Figure 3H.5-11 (Sheet 4 of 6)
[*Design of Shield Building at Air Inlets*]*

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Figure 3H.5-11 (Sheet 5 of 6)
[*Design of Shield Building: Tank/Roof Interface Reinforcement*]*

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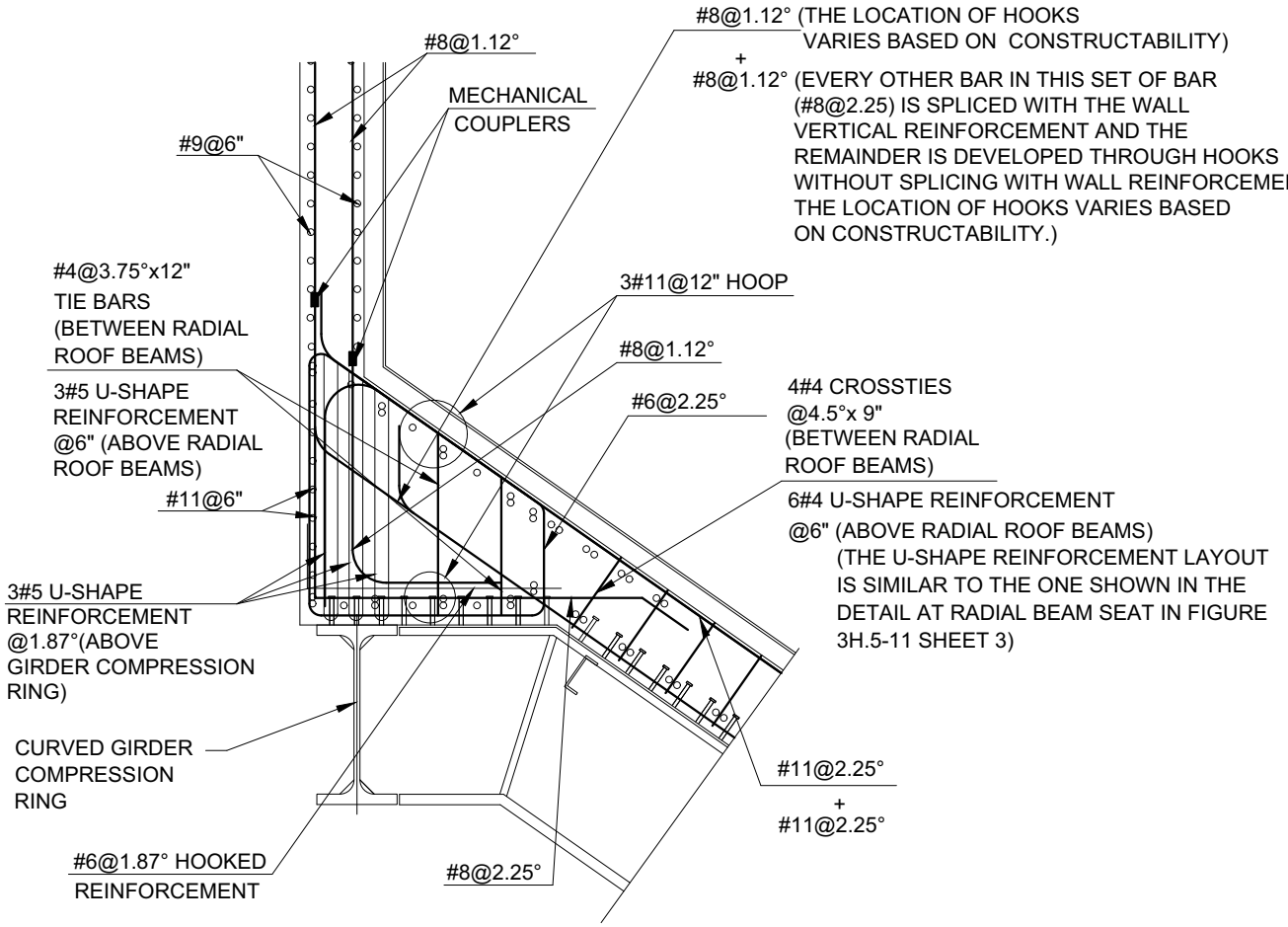

Figure 3H.5-11 (Sheet 6 of 6)
Design of Shield Building:
Tank/Compression Ring Roof Interface Reinforcement

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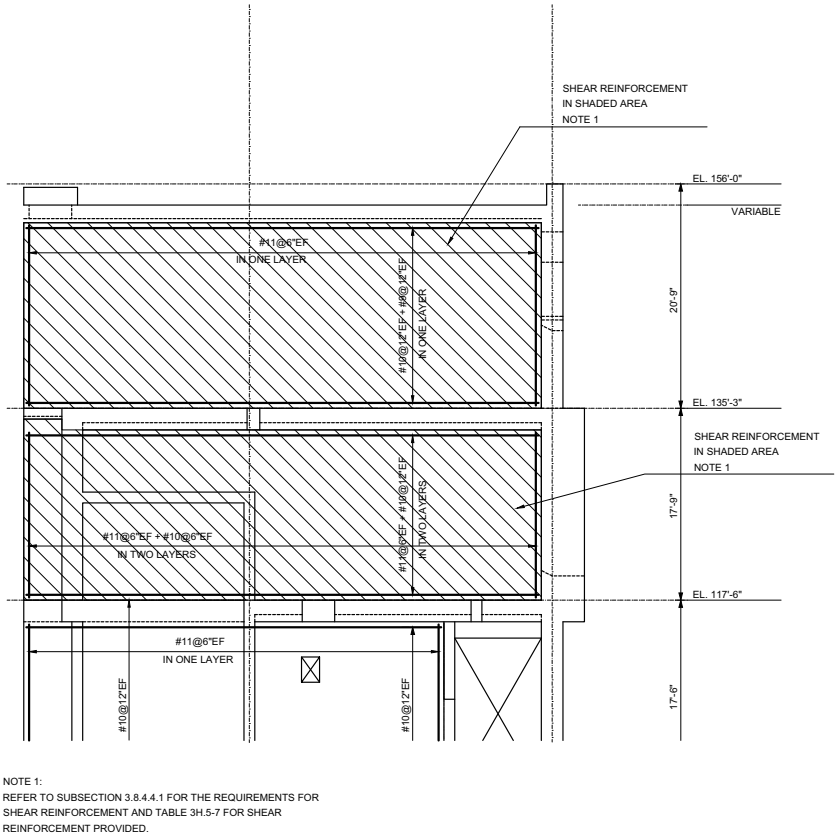

NOTE 1:
REFER TO SUBSECTION 3.8.4.4.1 FOR THE REQUIREMENTS FOR SHEAR REINFORCEMENT AND TABLE 3H.5-7 FOR SHEAR REINFORCEMENT PROVIDED.

Figure 3H.5-12
[*Typical Reinforcement in Wall L*]*

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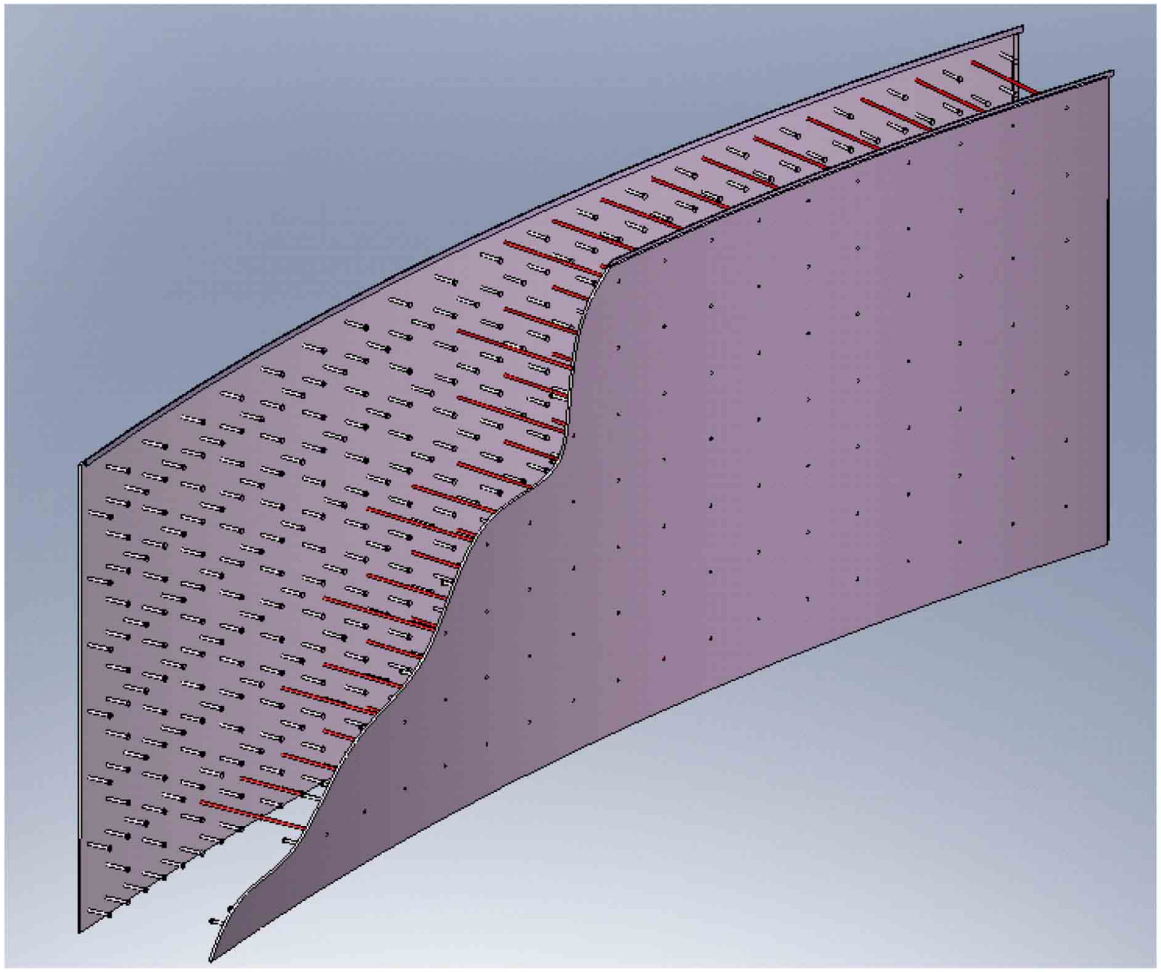

Figure 3H.5-13
Enhanced Shield Building Wall Panel Layout

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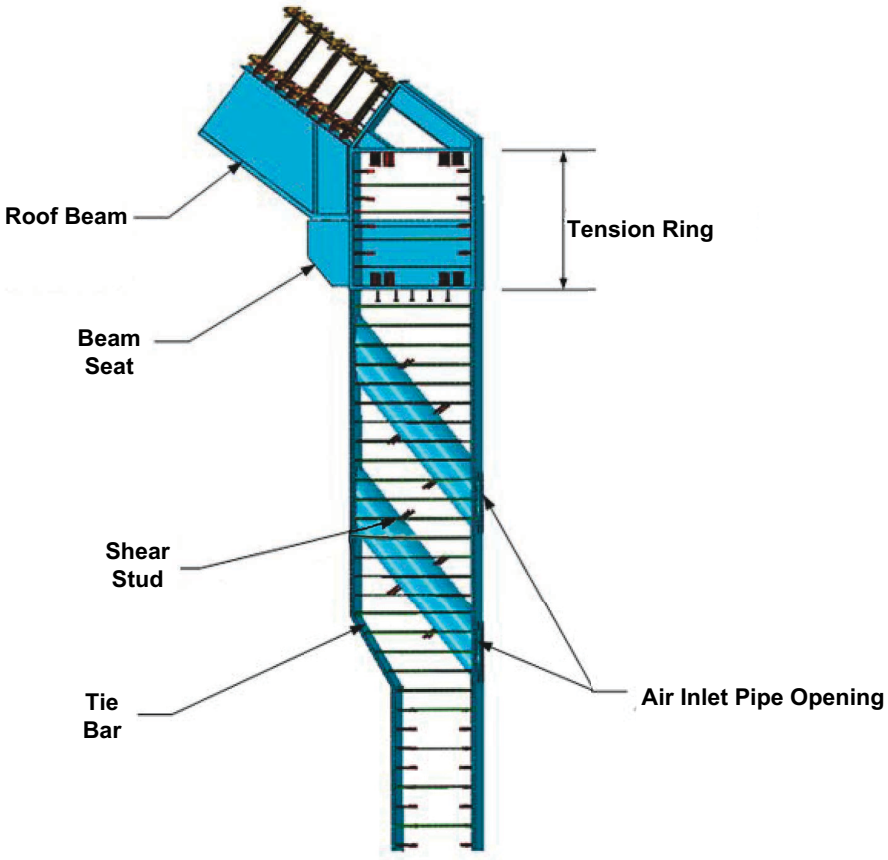

Figure 3H.5-14
Elevation View of Tension Ring and Air Inlets

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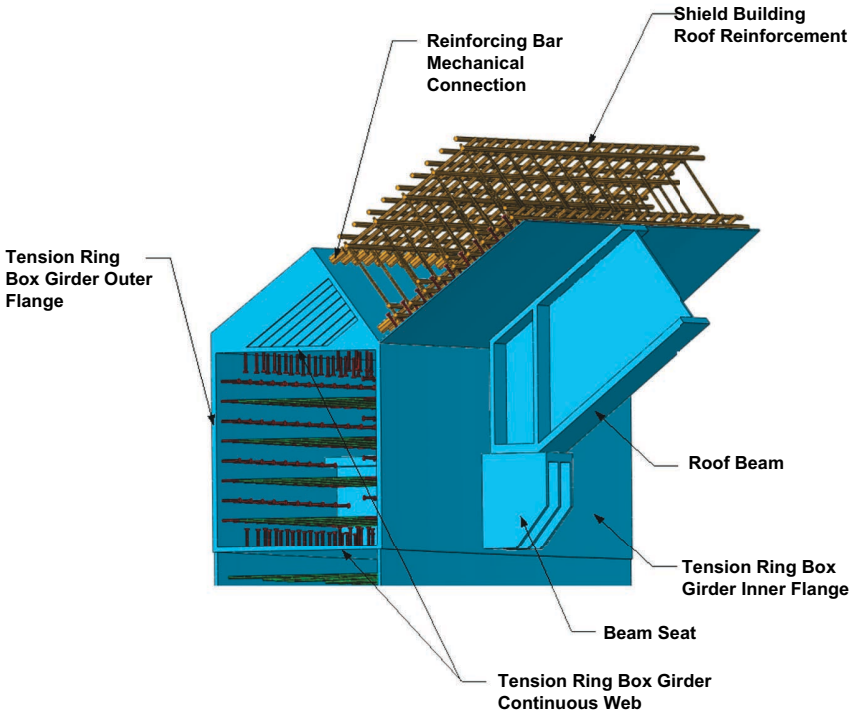

Figure 3H.5-15
[Shield Building Tension Ring]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Security-Related Information, Withheld Under 10 CFR 2.390

For additional information, see Figure 6 of APP-GW-GLR-602 (Reference 1).

Figure 3H.5-16 (Sheet 1 of 2)
**[*Design of Shield Building:*
Surface Plates on Cylindrical Section – Developed View 90-270 Degrees*]

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Security-Related Information, Withheld Under 10 CFR 2.390

For additional information, see Figure 6 of APP-GW-GLR-602 (Reference 1).

Figure 3H.5-16 (Sheet 2 of 2)
**[*Design of Shield Building:*
Surface Plates on Cylindrical Section – Developed View 270-90 Degrees*]

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Appendix 3I Evaluation for High Frequency Seismic Input

3I.1 Introduction

The seismic analysis and design of the AP1000 plant is based on the Certified Seismic Design Response Spectra (CSDRS) shown in Subsection 3.7.1.1. These spectra are based on Regulatory Guide 1.60 with an increase in the 25 hertz region. Ground Motion Response Spectra (GMRS) for some Central and Eastern United States rock sites show higher amplitude at high frequency than the CSDRS. Evaluations are described in this appendix for an envelope response spectra with high frequency for the seismic input. The resulting spectra of this site are shown in Figure 3I.1-1 and Figure 3I.1-2 and compare this hard rock high frequency (HRHF) envelope response spectra at the foundation level against the AP1000 CSDRS for both the horizontal and vertical directions for 5% damping. The HRHF envelope response spectra exceed the CSDRS for frequencies above about 15 Hz.

High frequency seismic input is generally considered to be non-damaging as described in Reference 1. The evaluation of the AP1000 nuclear island for the high frequency input is based on the analysis of a limited sample of structures, components, supports, and piping to demonstrate that the high frequency seismic response is non-damaging. The evaluation includes building structures, reactor pressure vessel and internals, primary component supports, primary loop nozzles, piping, and equipment.

This appendix describes the methodology and criteria used in the evaluation to confirm that the high frequency input is not damaging to equipment and structures qualified by analysis for the AP1000 CSDRS. It provides supplemental criteria for selection and testing of equipment whose function might be sensitive to high frequency. The results of the high frequency evaluation demonstrating that the AP1000 plant is qualified for this type of input are documented in a technical report (Reference 2). This report will provide a summary of the analysis and test results.

3I.2 High Frequency Seismic Input

Presented in Figures 3I.1-1 and 3I.1-2 is a comparison of the horizontal and vertical HRHF envelope response spectra and the AP1000 CSDRS. The HRHF envelope response spectra presented are calculated at foundation level (39.5' below grade), at the upper most competent material and treated as an outcrop for calculation purposes.

For each direction, the HRHF envelope response spectra exceed the design spectra in higher frequencies (greater than 15 Hz horizontal and 20 Hz vertical). The spectra are used for the HRHF envelope response spectra. If necessary, the HRHF envelope response spectra are enhanced at low frequencies so that HRHF envelope response spectra fully envelope all of the hard rock sites.

These HRHF envelope response spectra are further limited in that the shear wave velocity limitation is defined at the bottom of the basemat equal to or higher than 7,500 fps, while maintaining a shear wave velocity equal to or above 8,000 fps at the lower depths.

3I.3 NI Models Used To Develop High Frequency Response

The NI20 nuclear island model described in Appendix 3G is analyzed in ACS SASSI using the HRHF time histories applied at foundation level to obtain the motion at the base.

A modal analysis of the NI05 model for both the auxiliary and shield buildings and containment internal structure (CIS) has been performed for each of these regions. Specific areas within each wall or floor where out-of-plane modes, which may respond to either CSDRS or HRHF input (including structures with modes less than 33 Hz and between 33 Hz to 50 Hz), have been identified. The

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survey reveals that some regions, typically in the middle of a floor or wall, exhibit amplified behavior compared to the critical nodes at the corner and edge building locations. The amplified FRS for these regions is generated in addition to the typical set of critical nodes for building analysis by a single time history analysis of the NI05 building model subject to the HRHF time history input. Seismic response spectra for each of the “flexible” nodes are considered when selecting the pre-existing “group” spectra, which is the envelope of the entire floor in that area.

Evaluation of incoherent HRHF spectra has been performed. The CSDRS and HRHF seismic responses were compared with coherent and incoherent considerations at a number of locations in the nuclear island. There are some exceedances, mostly above the 15 hertz region, and these are typical of the plant comparative responses. The steel containment vessel (SCV) was excluded from the evaluation because the HRHF spectra at the base of the SCV are enveloped by the AP1000 CSDRS spectra at the base of the SCV.

Structures designed to the CSDRS input are adequately designed for the HRHF input because the HRHF coherent results are enveloped by the CSDRS results.

3I.4 Evaluation Methodology

The demonstration that the AP1000 nuclear power plant is qualified for the high frequency seismic response does not require the analysis of the total plant. The evaluations made are of representative systems, structures, and components, selected by screening, as potentially sensitive to high frequency input in locations where there were exceedances in the high frequency region. Acceptability of this sample is considered sufficient to demonstrate that the AP1000 is qualified.

The high frequency seismic analyses that are performed use time history or broadened response spectra. The analysis is not performed using the combination spectra of the CSDRS and the HRHF envelope response spectra. Separate analyses with each spectra are used.

The high frequency seismic analyses used the soil-structure interaction code ACS SASSI. The results presented in this report are based on the stochastic (multiple, statistical analyses) seismic incoherent soil-structure interaction analysis approach referred herein as the simulation approach.

The evaluations performed assess the ability of the system, structure, or component to maintain its safety function.

Supplementary analyses are performed as needed to show that high frequency floor response spectra exceedances are not damaging. These analyses can include: gap nonlinearities; material inelastic behavior; multi point response spectra analyses where the high frequency response excites a local part of the system. Tests on equipment are specified as needed where function cannot be demonstrated by analysis, or analysis is not appropriate.

3I.5 General Selection Screening Criteria

The following general screening criteria are used to identify representative AP1000 systems, structures, and components (SSCs) for the samples to be evaluated to demonstrate acceptability of the AP1000 nuclear power plant for the high frequency motion.

- Select systems, structures, and components based on their importance to safety. This includes the review of component safety function for the SSE event and its potential failure modes due to an SSE. Those components whose failure modes would result in safe shutdown are excluded.

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- Select systems, structures, and components that are located in areas of the plant that experience large high frequency seismic response.
- Select systems, structures, and components that have significant modal response within the region of high frequency amplification. Significance is defined by such items as modal mass; participation factor, stress and/or deflection.
- Select systems, structures, and components that have significant stress as compared to allowable when considering load combinations that include seismic.

3I.6 Evaluation

In this section the portions of structures, the components, and the systems that are evaluated for the high frequency seismic response are identified. The sample to be evaluated, based on the screening criteria applicable to the SSCs consists of the following:

- Building Structures
  - Auxiliary Building – 3 locations
  - Shield Building – 8 locations
  - CIS – 2 locations
- Primary Coolant Loop
  - Reactor Vessel and Internals
  - Primary Component Supports
  - Reactor Coolant Loop Primary Equipment Nozzles
- Piping Systems – ASME Class 1, 2, and 3 piping systems will be evaluated for the HRHF GMRS. This evaluation is within the scope of the piping DAC (see COL Information Item 3.9-7).
- Electro-Mechanical Equipment – Equipment that is potentially sensitive to high frequency input (see Table 3I.6-1)

These structures, systems, and equipment are discussed in more detail in the sections that follow.

3I.6.1 Building Structures

Maintaining the NI buildings structural integrity is important to the safety of the plant. Representative portions of the buildings that are evaluated for the effect of high frequency input are selected based on those areas that can experience high seismic shear and moment loads due to the seismic event. Areas chosen are at the base of the shield building, in the vicinity of auxiliary building floors that have fundamental frequencies in the high frequency region, and the corners of the auxiliary building. Three locations are selected on the auxiliary building that reflect the bottom of a wall where the shear and moment would be large, a wall in the vicinity of a floor that is influenced by high frequency response, and a corner intersection of walls. Eight locations are evaluated on the shield building. Four at elevation 107' and four at elevation 211'. These locations are located on the east, west, north and south sides. The south-west wall of the refueling canal is evaluated since it is a representative wall on

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the refueling canal. The CA02 wall in the CIS building is evaluated since it is a representative wall associated with the IRWST.

The evaluation consists of a comparison of the loads from the high frequency input to those obtained from the AP1000 design spectra, shown in Figures 3I.1-1 and 3I.1-2, for these representative building structures. The NI building structures are considered qualified for the high frequency input if the seismic loads from the Regulatory Guide 1.60 (modified) envelope those from the high frequency input. If there is any exceedance, this is evaluated further to confirm that the existing design is adequate.

3I.6.2 Primary Coolant Loop

A failure within the reactor coolant loop could challenge the integrity of the reactor coolant pressure boundary. Therefore, it is chosen for evaluation. The components evaluated are as follows:

- Reactor vessel and internals
- Reactor vessel supports
- Steam generator supports
- Reactor coolant loop primary equipment nozzles

The reactor vessel and internals are selected since they are important to safety and their analysis is representative of major primary components. The building structure below the reactor vessel supports is fairly stiff and there may be significant vertical amplification at the supports of the reactor pressure vessel. Further, reactor vessel internals have relatively complex structural systems including gap nonlinearities and sliding elements. Also, they may be sensitive to high frequency input as summarized below:

- Vertical and horizontal modes of the upper internals and the reactor vessel modes are in the relatively high frequency range.

- Additional high frequencies are associated with nonlinear impact

The evaluation consists of a comparison of the loads from the high frequency input to those obtained from the Regulatory Guide 1.60 (modified) input. Qualification is shown for the high frequency input if the seismic loads from the Regulatory Guide 1.60 (modified) envelope those from the high frequency input. If there is exceedance, then comparison is made for the combination of the seismic with the design basis pipe break loads and steady state loads. Qualification is then shown if the high frequency loads are relatively insignificant compared to the other loads, or there are no required design changes.

Maintaining the integrity of the reactor vessel and steam generator supports is important to preserving the primary component safety function. They are representative of supports on components, and see high loads.

The reactor coolant loop nozzles at the cold and hot leg interfaces of the reactor pressure vessel, reactor coolant pumps, and steam generators are important to include in the evaluation since these are critical areas of components.

The evaluation of the primary component supports and reactor coolant loop nozzles consists of a comparison of the loads from the high frequency input to those obtained from the Regulatory Guide 1.60 (modified) input. These items are considered qualified for the high frequency input if the seismic loads from the Regulatory Guide 1.60 (modified) envelope those from the high frequency input. If there is any exceedance, then an evaluation is made combining the high frequency loads with the other load components (e.g., thermal, pressure, dead) and a comparison made to the design loads. If

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the design loads envelope the load combinations that include the high frequency seismic input, then the nozzles and supports are considered qualified for the high frequency input.

3I.6.3 Piping Systems

ASME Class 1, 2, and 3 piping systems will be evaluated for the HRHF GMRS. This evaluation is within the scope of the piping DAC (see COL Information Item 3.9-7).

3I.6.4 Electrical and Electro-Mechanical Equipment

The groups of safety-related equipment considered for evaluation are those that may be sensitive to the high frequency input. This includes those cabinet-mounted equipment, field sensors, and appurtenants that may be sensitive to high frequency seismic inputs identified in Table 3I.6-1.

Sample safety-related cabinets have been identified that are typically sensitive to seismic input. Evaluations were performed to verify these cabinets do not have excessive seismic excitation on their mounted equipment, the cabinet designs do not require changes due to the high frequency input, and the cabinets will maintain their structural integrity during the high frequency input. Time history analyses of these cabinets were performed for both the Regulatory Guide 1.60 (modified) and the high frequency inputs so that comparisons can be made to their seismic response from both seismic inputs. This analytical study reported in APP-GW-GLR-115 (Reference 2) concluded that safety-related equipment may be screened.

The AP1000 HRHF screening program for determination and evaluation of potential high frequency sensitive equipment is in compliance with the NRC requirements in Section 4.0, “Identification and Evaluation of HF Sensitive Mechanical and Electrical Equipment/Components,” of COL/DC-ISG-1 (Reference 3). The AP1000 HRHF screening program is also consistent with the guidelines developed as part of an industry review document in the EPRI White Paper, “Seismic Screening of Components Sensitive to High Frequency Vibratory Motions” (Reference 4), transmitted to the NRC on June 28, 2007, for determining the safety-related equipment and components that may be HRHF sensitive, and screening procedures to ensure that any safety-related equipment and components sensitive to HRHF seismic excitation are screened out. This industry review of HF exceedance and further evaluations of SSCs performed by Westinghouse concluded that HRHF envelope response spectra are less harmful than the CSDRS except for the functionality of potential HRHF-sensitive components.

The AP1000 HRHF screening program is based on an HF evaluation study reported in APP-GW-GLR-115 (Reference 2). The HF evaluation study concluded that AP1000 In-Structure Response Spectra (ISRS) developed from the AP1000 CSDRS would, in the majority of cases, produce equipment stress results of the same magnitude or higher than the stress results produced from HRHF seismic excitation. The exception to this condition is when the dominant natural frequency of the equipment is in the HRHF exceedance range and there can be significantly more response because the frequency coincides with the input driving force. Under this condition, forces/ stresses generated in the equipment could be due to the acceleration exceedance; therefore, the equipment will be subjected to HRHF seismic evaluation/testing to screen out equipment by verifying its performance and acceptability under HRHF excitation. Review of seismic test data for electrical and microprocessor based cabinets performed to generic and high frequency excitation concluded that seismic testing that peaks in the lower frequency range will produce larger displacements and velocities, and will result in higher stresses in the equipment.

The goal of the AP1000 HRHF screening program is to identify the potential safety-related equipment and components that have the potential to be HRHF-sensitive and show them to be acceptable for their specific application (screened-out). The AP1000 HRHF screening program is a two step process. The first step is an HRHF susceptibility review to identify potential high frequency sensitive

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safety-related equipment. The second step is the screened-out equipment process to demonstrate its acceptability for the HRHF seismic excitation. Evaluation of screened-in equipment as defined in COL/DC-ISG-1 (Reference 3) is not performed because all safety-related equipment that is screened-in will be eliminated or shown to be acceptable through a design change process.

For the AP1000 HRHF screening program, the following conditions must exist:

1. Plant-specific HRHF GMRS exceeds the AP1000 CSDRS in the high frequency range at 5% critical damping.

2. Safety-related equipment has potential failure modes involving change of state, chatter, signal change/drift, and connection problems.

Table 3I.6-2 is a list of potential HRHF-sensitive AP1000 safety-related equipment developed based on Table 3.11-1 of Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment.” The equipment in Table 3.2-3 of Section 3.2, “AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment,” and Table 3I.6-3 is not HRHF-sensitive. The structural integrity and operability of equipment in Table 3.2-3 and Table 3I.6-3 will not be impacted by the high frequency excitation.

The HRHF susceptibility review of AP1000 safety-related equipment is not performed for potential failure modes associated with mounting, connections, fasteners, joints, and structural interface. These potential failure modes are addressed through the seismic qualification of the safety-related equipment to the AP1000 ISRS testing performed in compliance with IEEE Standard 344-1987. The AP1000 ISRS qualification testing generates higher displacements and velocities than those resulting from HRHF seismic excitation since the AP1000 ISRS is controlled by the lower frequency range. The higher displacement, velocities, and accelerations will detect these equipment structural failure modes if they exist.

At locations where HRHF response spectra show exceedance of the CSDRS and there is a likelihood of equipment damage, further evaluations would be performed to verify that the existing qualification is adequate for equipment not high frequency sensitive, as listed in Table 3I.6-3, under the following conditions:

- Safety-related equipment must have modes or natural frequencies in the range of interest.

- Evaluation will apply the same acceptance criteria and methodologies used in CSDRS qualification.

To demonstrate acceptability for both CSDRS and HRHF testing, the test response spectra must envelop the CSDRS and HRHF spectra, respectively, with margin over the frequency range of interest in compliance with IEEE Standard 344-1987. In the event that the CSDRS and/or HRHF response spectra would be revised after the qualification program has been completed, a reconciliation effort would be performed to verify that the CSDRS and HRHF testing is still valid. The reconciliation effort may result in requalification activities and qualification documentation revisions.

High Frequency Screening Process – Step 1

The potential failure modes of high frequency sensitive component types and assemblies are important considerations in the high frequency program. The following are potential failure modes of high frequency sensitive components/equipment:

- Inadvertent change of state
- Chatter
- Change in accuracy and drift in output signal or set-point

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- Electrical connection failure or intermediacy (e.g., poor quality solder joints)
- Mechanical connection failure
- Mechanical misalignment/binding (e.g., latches, plungers)
- Fatigue failure (e.g., solder joints, ceramics, self-taping screws, spot welds)
- Improperly and unrestrained mounted components
- Inadequately secured/locked mechanical fasteners and connections

Components and equipment determined to have potential failure modes involve change of state, chatter, signal change/drift, and connection problems will be demonstrated to be acceptable through the performance of a supplemental high frequency screening test. Those high frequency sensitive components having failure modes associated with mounting, connections and fasteners, joints, and interface are considered to be acceptable as a result of the AP1000 ISRS qualification testing per IEEE Standard 344-1987 and/or require quality assurance inspection and process/design controls.

High Frequency Screening Process – Step 2

The HRHF susceptibility review is to verify that the subject equipment is capable of performing its safety-related function under HRHF seismic excitation. All AP1000 safety-related equipment will be qualified to the AP1000 ISRS, and the dominant natural frequencies of the equipment will be determined. The EPRI White Paper (Reference 4) identifies the following three evaluation methods to demonstrate that potential HRHF-sensitive safety-related equipment is not HRHF vulnerable:

1. Existing seismic qualification test data for potential high frequency sensitive equipment should be reviewed for applicability and adequacy of the test method to demonstrate sufficient high frequency content.

2. Systems/circuits containing potentially sensitive items should be reviewed for inappropriate/ unacceptable system actions due to assumed change of state, contact chatter/intermittency, set point drifts, or loss of calibration.

3. HRHF vibration screening test is conducted to identify any HRHF sensitivities/abnormalities of the components. Several conventional test methods are recommended.

The first and third evaluation methods are part of the AP1000 HRHF screening program and are further detailed below. The AP1000 HRHF seismic screening evaluation will employ the AP1000 HRHF envelope response spectra as input in verifying potential HF sensitive safety-related equipment is not vulnerable to HRHF seismic excitation. Additional seismic test margin will be introduced into the HRHF seismic screening evaluation as needed.

Method 1: Review of Seismic Test Data

Available seismic test data can be used for AP1000 HRHF plant applications when:

- Seismic qualification testing performed on potential HRHF-sensitive safety-related equipment meets as a minimum the AP1000 ISRS in compliance with IEEE Standard 344-1987.

- Safe shutdown earthquake (SSE) test had sufficient energy content in the HRHF region to verify that the safety-related equipment is not vulnerable to HRHF seismic excitation.

No additional seismic testing is required for safety-related equipment previously tested and whose qualification level envelops the HRHF required response spectra (RRS).

IEEE Standard 344-1987 provides guidance to ensure that the seismic test input is generated and in compliance with the frequency range of interest. To demonstrate acceptability for frequency content, it is necessary to show that the frequency content of the test waveform is at least as broad as the

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frequency content of the amplified region of the RRS except at the low frequencies where non-enveloping is permitted under certain conditions (refer to IEEE Standard 344-1987 subclauses 7.6.3.1(10) and 7.6.3.1(13)). An evaluation of the test input waveform should be conducted per IEEE Standard 344-1987 Annex B to verify the test data has sufficient content over the frequency range of interest throughout the input time history. If an evaluation of the test input is performed, and the data demonstrates sufficient frequency content in the high frequency range throughout the time history, then the data is acceptable.

Method 3: HRHF Screening Test

The HRHF screening test is a supplemental test to the required seismic qualification methods performed in accordance with IEEE Standard 344-1987 for those plants that have high frequency exceedance of the AP1000 CSDRS. The purpose of the HRHF screening test is to demonstrate that the potential HRHF-sensitive safety-related equipment will perform its safety-related function as required under HRHF seismic excitation. The HRHF screening test is performed in conjunction with the AP1000 ISRS seismic qualification testing, or it is performed as a supplemental test after completion of the AP1000 ISRS seismic qualification testing. The AP1000 ISRS and HRHF test input time histories have 30-second durations with frequency content up to the cutoff frequency developed in accordance with subclause 7.6.3 (Multiple-Frequency Tests) and Annex B (Frequency Content and Stationarity) of IEEE Standard 344-1987. During the AP1000 ISRS and HRHF testing, the equipment will be functional and monitored to verify the safety-related function was demonstrated. Screening testing will be performed using HRHF response spectra as defined in the EPRI White Paper (Reference 4) when AP1000 HRHF inputs are not available. The HRHF response spectra will be generated based on the 5g and 15g peak spectral acceleration at 5% critical damping in the 25 Hz to 50 Hz frequency range. If the HRHF screening test cannot demonstrate the equipment to be acceptable, then the safety-related equipment will be removed or modified and additional testing or justification will be required.

The AP1000 safety-related equipment will be seismic qualified to the AP1000 ISRS associated with the mounting location of the equipment as a minimum. Seismic qualification testing will consist of five AP1000 ISRS operating basis earthquakes (OBEs) followed by one SSE as a minimum. The OBE level will be at least one-half the SSE level. The OBE testing is used to vibration age and address low-cycle fatigue of equipment prior to SSE testing. Cyclic fatiguing of equipment for HRHF exceedance area is adequately addressed by performing five OBE (one-half the SSE) and a minimum of one SSE seismic test runs in compliance with IEEE Standard 344-1987 prior to performing the supplemental HRHF screening test. Additional OBE testing in the high frequency exceedance range is adequately addressed by the demonstration that the peak stress cycles required for five one-half SSE events using the AP1000 HRHF ISRS are equivalent to or enveloped by the peak stress cycles resulting from five one-half SSE events and one full SSE event using the AP1000 CSD ISRS (Reference 5).

The test results of AP1000 seismic qualification programs with multiple operational states (for example, relays have three possible operational states: de-energized, energized, and change of state) will be used to determine the most sensitive equipment electrical operational state. The HRHF test run is performed on the equipment in its most sensitive electrical operational state to demonstrate its safety-related function under HRHF seismic excitation. If this is not possible, additional HRHF screening tests will be performed as needed to address the other most sensitive electrical operation states.

3I.7 References

1. EPRI Draft White Paper, “Considerations for NPP Equipment and Structures Subjected to Response Levels Caused by High Frequency Ground Motions,” Transmitted to NRC March 19, 2007.

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2. APP-GW-GLR-115, “Effect of High Frequency Seismic Content on SSCs,” Westinghouse Electric Company LLC.

3. COL/DC-ISG-1, “Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion,” May 19, 2008.

4. EPRI White Paper, “Seismic Screening of Components Sensitive to High Frequency Vibratory Motions,” June 2007.

5. Letter, R. Sisk (Westinghouse) to NRC, “AP1000 Response to Request for Additional Information (SRP3),” DCP/NRC2396, March 5, 2009.

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Table 3I.6-1
Potential High Frequency Sensitive Equipment List

- Equipment or components with moving parts and required to perform a switching function during the seismic event (e.g., circuit breakers, contactors, auxiliary switches, molded case circuit breakers, motor control center starters, and pneumatic control assemblies)
- Components with moving parts that may bounce or chatter such as relays and actuation devices (e.g., shunt trips)
- Unrestrained components
- Potentiometers
- Process switches and sensors (e.g., pressure/differential pressure, temperature, level, limit/position, and flow)
- Components with accuracy requirements that may drift due to seismic loading
- Interfaces such as secondary contacts
  - Connectors and connections (including circuit board connections for digital and analog equipment)

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Table 3I.6-2 (Sheet 1 of 21)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Batteries	
IDSA 125V 60 Cell Battery 1A	IDSA-DB-1A
IDSA 125V 60 Cell Battery 1B	IDSA-DB-1B
IDSB 125V 60 Cell Battery 1A	IDSB-DB-1A
IDSB 125V 60 Cell Battery 1B	IDSB-DB-1B
IDSB 125V 60 Cell Battery 2A	IDSB-DB-2A
IDSB 125V 60 Cell Battery 2B	IDSB-DB-2B
IDSC 125V 60 Cell Battery 1A	IDSC-DB-1A
IDSC 125V 60 Cell Battery 1B	IDSC-DB-1B
IDSC 125V 60 Cell Battery 2A	IDSC-DB-2A
IDSC 125V 60 Cell Battery 2B	IDSC-DB-2B
IDSD 125V 60 Cell Battery 1A	IDSD-DB-1A
IDSD 125V 60 Cell Battery 1B	IDSD-DB-1B
Spare 125V 60 Cell Battery 1A	IDSS-DB-1A
Spare 125V 60 Cell Battery 1B	IDSS-DB-1B
Battery Chargers	
IDSA Battery Charger	IDSA-DC-1
IDSB Battery Charger 1	IDSB-DC-1
IDSB Battery Charger 2	IDSB-DC-2
IDSC Battery Charger 1	IDSC-DC-1
IDSC Battery Charger 2	IDSC-DC-2
IDSD Battery Charger	IDSD-DC-1
Spare Battery Charger	IDSS-DC-1
Distribution Panels	
IDSA 250 Vdc Dist Panel	IDSA-DD-1
IDSB 250 Vdc Dist Panel	IDSB-DD-1
IDSC 250 Vdc Dist Panel	IDSC-DD-1
IDSD 250 Vdc Dist Panel	IDSD-DD-1
IDSA 120 Vac Dist Panel 1	IDSA-EA-1
IDSA 120 Vac Dist Panel 2	IDSA-EA-2
IDSB 120 Vac Dist Panel 1	IDSB-EA-1
IDSB 120 Vac Dist Panel 2	IDSB-EA-2
IDSB 120 Vac Dist Panel 3	IDSB-EA-3
IDSC 120 Vac Dist Panel 1	IDSC-EA-1
IDSC 120 Vac Dist Panel 2	IDSC-EA-2
IDSC 120 Vac Dist Panel 3	IDSC-EA-3
IDSD 120 Vac Dist Panel 1	IDSD-EA-1
IDSD 120 Vac Dist Panel 2	IDSD-EA-2

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Table 3I.6-2 (Sheet 2 of 21)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Fuse Panels	
IDSA Fuse Panel	IDSA-EA-4
IDSA Battery Monitor Fuse Panel	IDSA-EA-5
IDSB Fuse Panel	IDSB-EA-4
IDSB Fuse Panel	IDSB-EA-5
IDSB Fuse Panel	IDSB-EA-6
IDSB Battery Monitor Fuse Panel	IDSB-EA-7
IDSB Battery Monitor Fuse Panel	IDSB-EA-8
IDSC Fuse Panel	IDSC-EA-4
IDSC Fuse Panel	IDSC-EA-5
IDSC Fuse Panel	IDSC-EA-6
IDSC Battery Monitor Fuse Panel	IDSC-EA-7
IDSC Battery Monitor Fuse Panel	IDSC-EA-8
IDSD Fuse Panel	IDSD-EA-4
IDSD Battery Monitor Fuse Panel	IDSD-EA-5
IDSS Battery Monitor Fuse Panel	IDSS-EA-1
Stand-alone Electrical Isolation Devices	
Current-to-Current Transmitter 1	WLS-JD-MISX01
Current-to-Current Transmitter 1	WLS-JD-MISX02
Current-to-Current Transmitter 1	WLS-JD-MISX03
Transfer Switches	
IDSA Fused Transfer Switch Box 1	IDSA-DF-1
IDSB Fused Transfer Switch Box 1	IDSB-DF-1
IDSB Fused Transfer Switch Box 2	IDSB-DF-2
IDSC Fused Transfer Switch Box 1	IDSC-DF-1
IDSC Fused Transfer Switch Box 2	IDSC-DF-2
IDSD Fused Transfer Switch Box 1	IDSD-DF-1
IDSS Fused Transfer Switch Box 1	IDSS-DF-1
IDSS Spare Battery Termination Box	IDSS–DF-3
Motor Control Centers	
IDSA 250 Vdc MCC	IDSA-DK-1
IDSB 250 Vdc MCC	IDSB-DK-1
IDSC 250 Vdc MCC	IDSC-DK-1
IDSD 250 Vdc MCC	IDSD-DK-1
Switchboards	
IDSA 250 Vdc Switchboard 1	IDSA-DS-1
IDSB 250 Vdc Switchboard 1	IDSB-DS-1
IDSB 250 Vdc Switchboard 2	IDSB-DS-2
IDSC 250 Vdc Switchboard 1	IDSC-DS-1
IDSC 250 Vdc Switchboard 2	IDSC-DS-2
IDSD 250 Vdc Switchboard 1	IDSD-DS-1

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List of Potential High Frequency Sensitive
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Description	AP1000 Tag Number
Transformers	
IDSA Regulating Transformer 1	IDSA-DT-1
IDSB Regulating Transformer 1	IDSB-DT-1
IDSC Regulating Transformer 1	IDSC-DT-1
IDSD Regulating Transformer 1	IDSD-DT-1
Inverters	
IDSA Inverter	IDSA-DU-1
IDSB Inverter 1	IDSB-DU-1
IDSB Inverter 2	IDSB-DU-2
IDSC Inverter 1	IDSC-DU-1
IDSC Inverter 2	IDSC-DU-2
IDSD Inverter	IDSD-DU-1
Switchgear	
RCP 1A 6900V Switchgear 31	ECS-ES-31
RCP 1A 6900V Switchgear 32	ECS-ES-32
RCP 2A 6900V Switchgear 51	ECS-ES-51
RCP 2A 6900V Switchgear 52	ECS-ES-52
RCP 1B 6900V Switchgear 41	ECS-ES-41
RCP 1B 6900V Switchgear 42	ECS-ES-42
RCP 2B 6900V Switchgear 61	ECS-ES-61
RCP 2B 6900V Switchgear 62	ECS-ES-62
Junction Box Division A	PMS-EJ-001A
Junction Box Division B	PMS-EJ-001B
Junction Box Division C	PMS-EJ-001C
Junction Box Division D	PMS-EJ-001D
Junction Box Division A	PMS-EJ-002A
Junction Box Division B	PMS-EJ-002B
Junction Box Division C	PMS-EJ-002C
Junction Box Division D	PMS-EJ-002D
Reactor Trip Switchgear	PMS-JD-RTSA01
Reactor Trip Switchgear	PMS-JD-RTSA02
Reactor Trip Switchgear	PMS-JD-RTSB01
Reactor Trip Switchgear	PMS-JD-RTSB02
Reactor Trip Switchgear	PMS-JD-RTSC01
Reactor Trip Switchgear	PMS-JD-RTSC02
Reactor Trip Switchgear	PMS-JD-RTSD01
Reactor Trip Switchgear	PMS-JD-RTSD02

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Level Transmitters	
Core Makeup Tank A Narrow Range	PXS-JE-LE011A / PXS-JE-LT011A
Core Makeup Tank A Narrow Range	PXS-JE-LE011B / PXS-JE-LT011B
Core Makeup Tank A Narrow Range	PXS-JE-LE011C / PXS-JE-LT011C
Core Makeup Tank A Narrow Range	PXS-JE-LE011D / PXS-JE-LT011D
Core Makeup Tank B Narrow Range	PXS-JE-LE012A / PXS-JE-LT012A
Core Makeup Tank B Narrow Range	PXS-JE-LE012B / PXS-JE-LT012B
Core Makeup Tank B Narrow Range	PXS-JE-LE012C / PXS-JE-LT012C
Core Makeup Tank B Narrow Range	PXS-JE-LE012D / PXS-JE-LT012D
Core Makeup Tank A Narrow Range	PXS-JE-LE013A / PXS-JE-LT013A
Core Makeup Tank A Narrow Range	PXS-JE-LE013B / PXS-JE-LT013B
Core Makeup Tank A Narrow Range	PXS-JE-LE013C / PXS-JE-LT013C
Core Makeup Tank A Narrow Range	PXS-JE-LE013D / PXS-JE-LT013D
Core Makeup Tank B Narrow Range	PXS-JE-LE014A / PXS-JE-LT014A
Core Makeup Tank B Narrow Range	PXS-JE-LE014B / PXS-JE-LT014B
Core Makeup Tank B Narrow Range	PXS-JE-LE014C / PXS-JE-LT014C
Core Makeup Tank B Narrow Range	PXS-JE-LE014D / PXS-JE-LT014D
Containment Floodup Level	PXS-JE-LE050 / PXS-JE-LT050
Containment Floodup Level	PXS-JE-LE051 / PXS-JE-LT051
Containment Floodup Level	PXS-JE-LE052 / PXS-JE-LT052
Neutron Detectors	
Source Range Neutron Detector	RXS-JE-NE001A
Source Range Neutron Detector	RXS-JE-NE001B
Source Range Neutron Detector	RXS-JE-NE001C
Source Range Neutron Detector	RXS-JE-NE001D
Intermediate Range Neutron Detector	RXS-JE-NE002A
Intermediate Range Neutron Detector	RXS-JE-NE002B
Intermediate Range Neutron Detector	RXS-JE-NE002C
Intermediate Range Neutron Detector	RXS-JE-NE002D
Power Range Neutron Detector (Lower)	RXS-JE-NE003A
Power Range Neutron Detector (Lower)	RXS-JE-NE003B
Power Range Neutron Detector (Lower)	RXS-JE-NE003C
Power Range Neutron Detector (Lower)	RXS-JE-NE003D
Power Range Neutron Detector (Upper)	RXS-JE-NE004A
Power Range Neutron Detector (Upper)	RXS–JE-NE004B
Power Range Neutron Detector (Upper)	RXS-JE-NE004C
Power Range Neutron Detector (Upper)	RXS-JE-NE004D

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Radiation Monitors	
Containment High Range Area Monitor	PXS-JE-RE160
Containment High Range Area Monitor	PXS-JE-RE161
Containment High Range Area Monitor	PXS-JE-RE162
Containment High Range Area Monitor	PXS-JE-RE163
Control Room Supply Air Area Radiation Monitor - Particulate Detector	VBS-JE-RE001A
Control Room Supply Air Area Radiation Monitor - Particulate Detector	VBS-JE-RE001B
Control Room Supply Air Area Radiation Monitor - Iodine Detector	VBS-JE-RE002A
Control Room Supply Air Area Radiation Monitor - Iodine Detector	VBS-JE-RE002B
Control Room Supply Air Area Radiation Monitor - Noble Gas Detector	VBS-JE-RE003A
Control Room Supply Air Area Radiation Monitor - Noble Gas Detector	VBS-JE-RE003B
Control Room Supply Air Area Radiation Monitor - Air Flow Transmitter	VBS-JE-FT011A
Control Room Supply Air Area Radiation Monitor - Air Flow Transmitter	VBS-JE-FT011B
Control Room Supply Air Area Radiation Monitor - Sample Pump	VBS-MP-01A
Control Room Supply Air Area Radiation Monitor - Sample Pump	VBS-MP-01B
Control Room Supply Air Area Radiation Monitor - Local Radiation Processors	VBS-JE-RY001A
Control Room Supply Air Area Radiation Monitor - Local Radiation Processors	VBS-JE-RY001B
Speed Sensors	
RCP 1A Pump Speed	RCS-JE-ST281
RCP 1B Pump Speed	RCS-JE-ST282
RCP 2A Pump Speed	RCS-JE-ST283
RCP 2B Pump Speed	RCS-JE-ST284
Speed Sensor Preamplifier	
RCP 1A Speed Sensor Preamplifier Enclosure	PMS-JD-MIS01A
RCP 1B Speed Sensor Preamplifier Enclosure	PMS-JD-MIS01B
RCP 2A Speed Sensor Preamplifier Enclosure	PMS-JD-MIS02A
RCP 2B Speed Sensor Preamplifier Enclosure	PMS-JD-MIS02B
Transmitters	
PCS Water Delivery Flow	PCS-JE-FT001
PCS Water Delivery Flow	PCS-JE-FT002
PCS Water Delivery Flow	PCS-JE-FT003
PCS Water Delivery Flow	PCS-JE-FT004
PCS Storage Tank Water Level	PCS-JE-LT010
PCS Storage Tank Water Level	PCS-JE-LT011
PRHR HX Flow	PXS-JE-FT049A
PRHR HX Flow	PXS-JE-FT049B
RCS Hot Leg 1 Flow	RCS-JE-FT101A
RCS Hot Leg 1 Flow	RCS-JE-FT101B
RCS Hot Leg 1 Flow	RCS-JE-FT101C
RCS Hot Leg 1 Flow	RCS-JE-FT101D
RCS Hot Leg 2 Flow	RCS-JE-FT102A

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List of Potential High Frequency Sensitive
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Description	AP1000 Tag Number
RCS Hot Leg 2 Flow	RCS-JE-FT102B
RCS Hot Leg 2 Flow	RCS-JE-FT102C
RCS Hot Leg 2 Flow	RCS-JE-FT102D
SG1 Startup Feedwater Flow	SGS-JE-FT055A
SG1 Startup Feedwater Flow	SGS-JE-FT055B
SG2 Startup Feedwater Flow	SGS-JE-FT-056A
SG2 Startup Feedwater Flow	SGS-JE-FT056B
MCR Air Delivery Line Flow Rate – A	VES-JE-FT003A
MCR Air Delivery Line Flow Rate – B	VES-JE-FT003B
IRWST Wide Range Level	PXS-JE-LT046
IRWST Wide Range Level	PXS-JE-LT047
IRWST Wide Range Level	PXS-JE-LT048
IRWST Lower Narrow Range Level	PXS-JE-LT066
IRWST Lower Narrow Range Level	PXS-JE-LT067
IRWST Lower Narrow Range Level	PXS-JE-LT068
IRWST Lower Narrow Range Level	PXS-JE-LT069
RCS Hot Leg Water Level	RCS-JE-LT160A
RCS Hot Leg Water Level	RCS-JE-LT160B
PZR Level	RCS-JE-LT195A
PZR Level	RCS-JE-LT195B
PZR Level	RCS-JE-LT195C
PZR Level	RCS-JE-LT195D
SG1 Narrow Range Level	SGS-JE-LT001
SG1 Narrow Range Level	SGS-JE-LT002
SG1 Narrow Range Level	SGS-JE-LT003
SG1 Narrow Range Level	SGS-JE-LT004
SG2 Narrow Range Level	SGS-JE-LT005
SG2 Narrow Range Level	SGS-JE-LT006
SG2 Narrow Range Level	SGS-JE-LT007
SG2 Narrow Range Level	SGS-JE-LT008
SG1 Wide Range Level	SGS-JE-LT011
SG1 Wide Range Level	SGS-JE-LT012
SG1 Wide Range Level	SGS-JE-LT015
SG1 Wide Range Level	SGS-JE-LT016
SG2 Wide Range Level	SGS-JE-LT013
SG2 Wide Range Level	SGS-JE-LT014
SG2 Wide Range Level	SGS-JE-LT017
SG2 Wide Range Level	SGS-JE-LT018

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Spent Fuel Pool Level	SFS-JE-LT019A
Spent Fuel Pool Level	SFS-JE-LT019B
Spent Fuel Pool Level	SFS-JE-LT019C
WLS Auxiliary Building RCA Floodup Level	WLS-JE-LT400A
WLS Auxiliary Building RCA Floodup Level	WLS-JE-LT400B
Containment Pressure Normal Range	PCS-JE-PT005
Containment Pressure Normal Range	PCS-JE-PT006
Containment Pressure Normal Range	PCS-JE-PT007
Containment Pressure Normal Range	PCS-JE-PT008
Containment Pressure Extended Range	PCS-JE-PT012
Containment Pressure Extended Range	PCS-JE-PT013
Containment Pressure Extended Range	PCS-JE-PT014
RCS Wide Range Pressure	RCS-JE-PT140A
RCS Wide Range Pressure	RCS-JE-PT140B
RCS Wide Range Pressure	RCS-JE-PT140C
RCS Wide Range Pressure	RCS-JE-PT140D
PZR Pressure	RCS-JE-PT191A
PZR Pressure	RCS-JE-PT191B
PZR Pressure	RCS-JE-PT191C
PZR Pressure	RCS-JE-PT191D
Main Steam Line SG1 Pressure	SGS-JE-PT030
Main Steam Line SG1 Pressure	SGS-JE-PT031
Main Steam Line SG1 Pressure	SGS-JE-PT032
Main Steam Line SG1 Pressure	SGS-JE-PT033
Main Steam Line SG2 Pressure	SGS-JE-PT034
Main Steam Line SG2 Pressure	SGS-JE-PT035
Main Steam Line SG2 Pressure	SGS-JE-PT036
Main Steam Line SG2 Pressure	SGS-JE-PT037
Main Control Room Differential Pressure	VES-JE-PDT004A
Main Control Room Differential Pressure	VES-JE-PDT004B
Protection and Safety Monitoring Systems	
Bistable Coincident Logic Cabinet 01 Division A	PMS-JD-BCCA01
Bistable Coincident Logic Cabinet 02 Division A	PMS-JD-BCCA02
Bistable Coincident Logic Cabinet 01 Division B	PMS-JD-BCCB01
Bistable Coincident Logic Cabinet 02 Division B	PMS-JD-BCCB02
Bistable Coincident Logic Cabinet 01 Division C	PMS-JD-BCCC01
Bistable Coincident Logic Cabinet 02 Division C	PMS-JD-BCCC02
Bistable Coincident Logic Cabinet 01 Division D	PMS-JD-BCCD01
Bistable Coincident Logic Cabinet 02 Division D	PMS-JD-BCCD02
Integrated Logic Cabinet 01 Division A	PMS-JD-ILCA01
Integrated Logic Cabinet 02 Division A	PMS-JD-ILCA02
Integrated Logic Cabinet 03 Division A	PMS-JD-ILCA03
Integrated Logic Cabinet 01 Division B	PMS-JD-ILCB01
Integrated Logic Cabinet 02 Division B	PMS-JD-ILCB02

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Integrated Logic Cabinet 03 Division B	PMS-JD-ILCB03
Integrated Logic Cabinet 04 Division B	PMS-JD-ILCB04
Integrated Logic Cabinet 01 Division C	PMS-JD-ILCC01
Integrated Logic Cabinet 02 Division C	PMS-JD-ILCC02
Integrated Logic Cabinet 01 Division D	PMS-JD-ILCD01
Integrated Logic Cabinet 02 Division D	PMS-JD-ILCD02
Integrated Logic Cabinet 03 Division D	PMS-JD-ILCD03
Integrated Logic Cabinet 04 Division D	PMS-JD-ILCD04
Maintenance and Test Cabinet Division A	PMS-JD-MTCA01
Maintenance and Test Cabinet Division B	PMS-JD-MTCB01
Maintenance and Test Cabinet Division C	PMS-JD-MTCC01
Maintenance and Test Cabinet Division D	PMS-JD-MTCD01
Nuclear Instrumentation Cabinet Division A	PMS-JD-NICA01
Nuclear Instrumentation Cabinet Division B	PMS-JD-NICB01
Nuclear Instrumentation Cabinet Division C	PMS-JD-NICC01
Nuclear Instrumentation Cabinet Division D	PMS-JD-NICD01
Qualified Data Processing System Cabinet Division B	PMS-JD-QDPB01
Qualified Data Processing System Cabinet Division C	PMS-JD-QDPC01
Division A Squib Valve Controller Cabinet	PMS-JD-SVCA01
Division B Squib Valve Controller Cabinet	PMS-JD-SVCB01
Division C Squib Valve Controller Cabinet	PMS-JD-SVCC01
Division D Squib Valve Controller Cabinet	PMS-JD-SVCD01
Source Range Neutron Flux Preamplifier Panel A	PMS-JW-005A
Source Range Neutron Flux Preamplifier Panel B	PMS-JW-005B
Source Range Neutron Flux Preamplifier Panel C	PMS-JW-005C
Source Range Neutron Flux Preamplifier Panel D	PMS-JW-005D
Intermediate Range Neutron Flux Preamplifier Panel A	PMS-JW-006A
Intermediate Range Neutron Flux Preamplifier Panel B	PMS-JW-006B
Intermediate Range Neutron Flux Preamplifier Panel C	PMS-JW-006C
Intermediate Range Neutron Flux Preamplifier Panel D	PMS-JW-006D

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Main Control Room	
Operator Workstation A	N/A
Operator Workstation B	N/A
Supervisor Workstation	N/A
Primary Dedicated Safety Panel	OCS-JC-10
Secondary Dedicated Safety Panel	OCS-JC-11
MCR/RSR Transfer Switch Panel	OCS-JW-001
Qualified Safety Display Division A	PMS-JD-QSPA01
Qualified Safety Display Division B	PMS-JD-QSPB01
Qualified Safety Display Division C	PMS-JD-QSPC01
Qualified Safety Display Division D	PMS-JD-QSPD01
MCR Load Shed Panel 1	VES-EP-01
MCR Load Shed Panel 2	VES-EP-02
Active Valves	
Containment Isolation – Air Out	
Solenoid Valve	CAS-PL-V014-S2
Solenoid Valve	CAS-PL-V014-S1
Limit Switch	CAS-PL-V014-L
Containment Isolation – Inlet	
Limit Switch	CCS-PL-V200-L
Motor Operator	CCS-PL-V200-M
Containment Isolation – Outlet	
Limit Switch	CCS-PL-V207-L
Motor Operator	CCS-PL-V207-M
Containment Isolation – Outlet	
Limit Switch	CCS-PL-V208-L
Motor Operator	CCS-PL-V208-M
RCS Purification Stop Valve	
Limit Switch	CVS-PL-V001-L
Motor Operator	CVS-PL-V001-M
RCS Purification Stop Valve	
Limit Switch	CVS-PL-V002-L
Motor Operator	CVS-PL-V002-M
RCS Letdown Stop Valve	
Limit Switch	CVS-PL-V003-L
Motor Operator	CVS-PL-V003-M
WLS Letdown IRC Isolation	
Limit Switch	CVS-PL-V045-L
Solenoid Valve	CVS-PL-V045-S1
Letdown Flow ORC Isolation	
Limit Switch	CVS-PL-V047-L
Solenoid Valve	CVS-PL-V047-S1
Auxiliary PZR Spray Isolation	
Limit Switch	CVS-PL-V084-L

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Solenoid Valve	CVS-PL-V084-S
Makeup Line Containment Isolation	
Limit Switch	CVS-PL-V090-L
Motor Operator	CVS-PL-V090-M
Makeup Line Containment Isolation	
Limit Switch	CVS-PL-V091-L
Motor Operator	CVS-PL-V091-M
Zinc Addition Containment Isolation	
Limit Switch	CVS-PL-V092-L
Solenoid Valve	CVS-PL-V092-S
Zinc Addition IRC Isolation Valve	
Limit Switch	CVS-PL-V094-L
Solenoid Valve	CVS-PL-V094-S
Demineralizer Water System Isolation	
Limit Switch	CVS-PL-V136A-L
Solenoid Valve	CVS-PL-V136A-S
Demineralized Water System Isolation	
Limit Switch	CVS-PL-V136B-L
Solenoid Valve	CVS-PL-V136B-S
Hydrogen Injection Containment Isolation	
Limit Switch	CVS-PL-V219-L
Solenoid Valve	CVS-PL-V219-S
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V001A-L
Solenoid Valve	PCS-PL-V001A-S1
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V001B-L
Solenoid Valve	PCS-PL-V001B-S1
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V001C-L
Motor Operator	PCS-PL-V001C-M
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V002A-L
Motor Operator	PCS-PL-V002A-M
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V002B-L
Motor Operator	PCS-PL-V002B-M
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V002C-L
Motor Operator	PCS-PL-V002C-M

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List of Potential High Frequency Sensitive
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Description	AP1000 Tag Number
Containment Isolation – Air Sample Line	
Limit Switch	PSS-PL-V008-L
Solenoid Operator	PSS-PL-V008-S
Voltage Control Box	PSS-DJ-05
Containment Isolation – Liquid Sample Line	
Limit Switch	PSS-PL-V010A-L
Solenoid Operator	PSS-PL-V010A-S
Voltage Control Box	PSS-DJ-03
Containment Isolation – Liquid Sample Line	
Limit Switch	PSS-PL-V010B-L
Solenoid Operator	PSS-PL-V010B-S
Voltage Control Box	PSS-DJ-04
Containment Isolation – Liquid Sample Line	
Limit Switch	PSS-PL-V011A-L
Solenoid Valve	PSS-PL-V011A-S
Containment Isolation – Liquid Sample Line	
Limit Switch	PSS-PL-V011B-L
Solenoid Valve	PSS-PL-V011B-S
Containment Isolation - Sample Return Line	
Limit Switch	PSS-PL-V023-L
Solenoid Valve	PSS-PL-V023-S
Containment Isolation – Sample Return Line	
Limit Switch	PSS-PL-V024-L
Solenoid Operator	PSS-PL-V024-S
Voltage Control Box	PSS-DJ-06
Containment Isolation - Air Sample Line	
Limit Switch	PSS-PL-V046-L
Solenoid Valve	PSS-PL-V046-S
Core Makeup Tank A Discharge Isolation	
Limit Switch	PXS-PL-V014A-L
Solenoid Valve	PXS-PL-V014A-S1
Solenoid Valve	PXS-PL-V014A-S3
Core Makeup Tank B Discharge Isolation	
Limit Switch	PXS-PL-V014B-L
Solenoid Valve	PXS-PL-V014B-S1
Solenoid Valve	PXS-PL-V014B-S3
Core Makeup Tank A Discharge Isolation	
Limit Switch	PXS-PL-V015A-L
Solenoid Valve	PXS-PL-V015A-S1
Solenoid Valve	PXS-PL-V015A-S3
Core Makeup Tank B Discharge Isolation	
Limit Switch	PXS-PL-V015B-L
Solenoid Valve	PXS-PL-V015B-S1
Solenoid Valve	PXS-PL-V015B-S3

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Nitrogen Supply Outside Containment Isolation	
Limit Switch	PXS-PL-V042-L
Solenoid Valve	PXS-PL-V042-S
PRHR HX Discharge Isolation	
Limit Switch	PXS-PL-V108A-L
Solenoid Valve	PXS-PL-V108A-S1
Solenoid Valve	PXS-PL-V108A-S3
PRHR HX Discharge Isolation	
Limit Switch	PXS-PL-V108B-L
Solenoid Valve	PXS-PL-V108B-S1
Solenoid Valve	PXS-PL-V108B-S3
Recirc Sump A Isolation	
Limit Switch	PXS-PL–V118A-L
Squib Operator	PXS-PL–V118A-T
Recirc Sump B Isolation	
Limit Switch	PXS-PL-V118B-L
Squib Operator	PXS-PL-V118B-T
Recirc Sump A	
Limit Switch	PXS-PL-V120A-L
Squib Operator	PXS-PL-V120A-T
Recirc Sump B	
Limit Switch	PXS-PL-V120B-L
Squib Operator	PXS-PL-V120B-T
IRWST Injection A	
Limit Switch	PXS-PL-V123A-L
Squib Operator	PXS-PL-V123A-T
IRWST Injection B	
Limit Switch	PXS-PL-V123B-L
Squib Operator	PXS-PL-V123B-T
IRWST Injection A	
Limit Switch	PXS-PL-V125A-L
Squib Operator	PXS-PL-V125A-T
IRWST Injection B	
Limit Switch	PXS-PL-V125B-L
Squib Operator	PXS-PL-V125B-T
IRWST Gutter Drain Isolation A	
Limit Switch	PXS-PL-V130A-L
Solenoid Valve	PXS-PL-V130A-S1
IRWST Gutter Drain Isolation B	
Limit Switch	PXS-PL-V130B-L
Solenoid Valve	PXS-PL-V130B-S1

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Table 3I.6-2 (Sheet 13 of 21)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
First Stage ADS	
Limit Switch	RCS-PL-V001A-L
Motor Operator	RCS-PL-V001A-M
First Stage ADS	
Limit Switch	RCS-PL-V001B-L
Motor Operator	RCS-PL-V001B-M
Second Stage ADS	
Limit Switch	RCS-PL-V002A-L
Motor Operator	RCS-PL-V002A-M
Second Stage ADS	
Limit Switch	RCS-PL-V002B-L
Motor Operator	RCS-PL-V002B-M
Third Stage ADS	
Limit Switch	RCS-PL-V003A-L
Motor Operator	RCS-PL-V003A-M
Third Stage ADS	
Limit Switch	RCS-PL-V003B-L
Motor Operator	RCS-PL-V003B-M
Fourth Stage ADS	
Limit Switch	RCS-PL-V004A-L
Squib Operator	RCS-PL-V004A-T
Fourth Stage ADS	
Limit Switch	RCS-PL-V004B-L
Squib Operator	RCS-PL-V004B-T
Fourth Stage ADS	
Limit Switch	RCS-PL-V004C-L
Squib Operator	RCS-PL-V004C-T
Fourth Stage ADS	
Limit Switch	RCS-PL-V004D-L
Squib Operator	RCS-PL-V004D-T
First Stage ADS Isolation	
Limit Switch	RCS-PL-V011A-L
Motor Operator	RCS-PL-V011A-M
First Stage ADS Isolation	
Limit Switch	RCS-PL-V011B-L
Motor Operator	RCS-PL-V011B-M
Second Stage ADS Isolation	
Limit Switch	RCS-PL-V012A-L
Motor Operator	RCS-PL-V012A-M

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Second Stage ADS Isolation	
Limit Switch	RCS-PL-V012B-L
Motor Operator	RCS-PL-V012B-M
Third Stage ADS Isolation	
Limit Switch	RCS-PL-V013A-L
Motor Operator	RCS-PL-V013A-M
Third Stage ADS Isolation	
Limit Switch	RCS-PL-V013B-L
Motor Operator	RCS-PL-V013B-M
Reactor Vessel Head Vent	
Limit Switch	RCS-PL-V150A-L
Solenoid Operator	RCS-PL-V150A-S
Voltage Control Box	RCS-DJ-01
Reactor Vessel Head Vent	
Limit Switch	RCS-PL-V150B-L
Solenoid Operator	RCS-PL-V150B-S
Voltage Control Box	RCS-DJ-02
Reactor Vessel Head Vent	
Limit Switch	RCS-PL-V150C-L
Solenoid Operator	RCS-PL-V150C-S
Voltage Control Box	RCS-DJ-03
Reactor Vessel Head Vent	
Limit Switch	RCS-PL-V150D-L
Solenoid Operator	RCS-PL-V150D-S
Voltage Control Box	RCS-DJ-04
RCS Inner Suction Isolation	
Limit Switch	RNS-PL-V001A-L
Motor Operator	RNS-PL-V001A-M
RCS Inner Suction Isolation	
Limit Switch	RNS-PL-V001B-L
Motor Operator	RNS-PL-V001B-M
RCS Outer Suction Isolation	
Limit Switch	RNS-PL-V002A-L
Motor Operator	RNS-PL-V002A-M
RCS Outer Suction Isolation	
Limit Switch	RNS-PL-V002B-L
Motor Operator	RNS-PL-V002B-M

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
RNS Discharge Containment Isolation Valve	
Limit Switch	RNS-PL-V011-L
Motor Operator	RNS-PL-V011-M
RNS Suction Header Containment Isolation	
Limit Switch	RNS-PL-V022-L
Motor Operator	RNS-PL-V022-M
IRWST Suction Line Isolation	
Limit Switch	RNS-PL-V023-L
Motor Operator	RNS-PL-V023-M
RNS – CVS Containment Isolation	
Limit Switch	RNS-PL-V061-L
Air Operator	RNS-PL-V061-S
SDS – MCR Vent Isolation Valve	
Limit Switch	SDS-PL-V001-L
Motor Operator	SDS-PL-V001-M
SDS – MCR Vent Isolation Valve	
Limit Switch	SDS-PL-V002-L
Motor Operator	SDS-PL-V002-M
SFS Suction Line Containment Isolation	
Limit Switch	SFS-PL-V034-L
Motor Operator	SFS-PL–V034-M
SFS Suction Line Containment Isolation	
Limit Switch	SFS-PL-V035-L
Motor Operator	SFS- PL-V035-M
SFS Discharge Line Containment Isolation	
Limit Switch	SFS-PL-V038-L
Motor Operator	SFS-PL-V038-M
PORV Block Valve	
Limit Switch	SGS-PL-V027A-L
Motor Operator	SGS-PL-V027A-M
PORV Block Valve	
Limit Switch	SGS-PL-V027B-L
Motor Operator	SGS-PL-V027B-M
Main Steam Safety Valve 1 Limit Switch	SGS-PL-V030A-L
Main Steam Safety Valve 1 Limit Switch	SGS-PL-V030B-L
Main Steam Safety Valve 2 Limit Switch	SGS-PL-V031A-L
Main Steam Safety Valve 2 Limit Switch	SGS-PL-V031B-L
Main Steam Safety Valve 3 Limit Switch	SGS-PL-V032A-L
Main Steam Safety Valve 3 Limit Switch	SGS-PL-V032B-L

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Main Steam Safety Valve 4 Limit Switch	SGS-PL-V033A-L
Main Steam Safety Valve 4 Limit Switch	SGS-PL-V033B-L
Main Steam Safety Valve 5 Limit Switch	SGS-PL-V034A-L
Main Steam Safety Valve 5 Limit Switch	SGS-PL-V034B-L
Main Steam Safety Valve 6 Limit Switch	SGS-PL-V035A-L
Main Steam Safety Valve 6 Limit Switch	SGS-PL-V035B-L
Steam Line Condensate Drain Isolation	
Limit Switch	SGS-PL-V036A-L
Solenoid Valve	SGS-PL-V036A-S
Steam Line Condensate Drain Isolation	
Limit Switch	SGS-PL-V036B-L
Solenoid Valve	SGS-PL-V036B-S
Main Steam Line Isolation	
Limit Switch	SGS-PL-V040A-L
Solenoid Valve	SGS-PL-V040A-S1
Solenoid Valve	SGS-PL-V040A-S2
Solenoid Valve	SGS-PL-V040A-S3
Solenoid Valve	SGS-PL-V040A-S4
Main Steam Line Isolation	
Limit Switch	SGS-PL-V040B-L
Solenoid Valve	SGS-PL-V040B-S1
Solenoid Valve	SGS-PL-V040B-S2
Solenoid Valve	SGS-PL-V040B-S3
Solenoid Valve	SGS-PL-V040B-S4
Main Feedwater Isolation	
Limit Switch	SGS-PL-V057A-L
Solenoid Valve	SGS-PL-V057A-S1
Solenoid Valve	SGS-PL-V057A-S2
Solenoid Valve	SGS-PL-V057A-S3
Solenoid Valve	SGS-PL-V057A-S4
Main Feedwater Isolation	
Limit Switch	SGS-PL-V057B-L
Solenoid Valve	SGS-PL-V057B-S1
Solenoid Valve	SGS-PL-V057B-S2
Solenoid Valve	SGS-PL-V057B-S3
Solenoid Valve	SGS-PL-V057B-S4
Startup Feedwater Isolation	
Limit Switch	SGS-PL-V067A-L
Motor Operator	SGS-PL-V067A-M

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Startup Feedwater Isolation	
Limit Switch	SGS-PL-V067B-L
Motor Operator	SGS-PL-V067B-M
SG Blowdown Isolation	
Limit Switch	SGS-PL-V074A-L
Solenoid Valve	SGS-PL-V074A-S
SG Blowdown Isolation	
Limit Switch	SGS-PL-V074B-L
Solenoid Valve	SGS-PL-V074B-S
SG Series Blowdown Isolation	
Limit Switch	SGS-PL-V075A-L
Solenoid Valve	SGS-PL-V075A-S
SG Series Blowdown Isolation	
Limit Switch	SGS-PL-V075B-L
Solenoid Valve	SGS-PL-V075B-S
Steam Line Condensate Drain Isolation	
Level Switch	SGS-PL-V086A-L
Solenoid Valve	SGS-PL-V086A-S
Steam Line Condensate Drain Isolation	
Limit Switch	SGS-PL-V086B-L
Solenoid Valve	SGS-PL-V086B-S
Power Operated Relief Valve	
Limit Switch	SGS-PL-V233A-L
Solenoid Valve	SGS-PL-V233A-S
Power Operated Relief Valve	
Limit Switch	SGS-PL-V233B-L
Solenoid Valve	SGS-PL-V233B-S
MSIV Bypass Isolation Valve	
Limit Switch	SGS-PL-V240A-L
Solenoid Valve	SGS-PL-V240A-S1
Solenoid Valve	SGS-PL-V240A-S2
MSIV Bypass Isolation Valve	
Limit Switch	SGS-PL-V240B-L
Solenoid Valve	SGS-PL-V240B-S1
Solenoid Valve	SGS-PL-V240B-S2
Main Feedwater Control Valve	
Limit Switch	SGS-PL-V250A-L
Solenoid Valve	SGS-PL-V250A-S1
Solenoid Valve	SGS-PL-V250A-S2
Main Feedwater Control Valve	
Limit Switch	SGS-PL-V250B-L
Solenoid Valve	SGS-PL-V250B-S1
Solenoid Valve	SGS-PL-V250B-S2

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Startup Feedwater Control Valve	
Limit Switch	SGS-PL-V255A-L
Solenoid Valve	SGS-PL-V255A-S
Startup Feedwater Control Valve	
Limit Switch	SGS-PL-V255B-L
Solenoid Valve	SGS-PL-V255B-S
MCR Isolation Valve	
Limit Switch	VBS-PL-V186-L
Motor Operator	VBS-PL-V186-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V187-L
Motor Operator	VBS-PL-V187-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V188-L
Motor Operator	VBS-PL-V188-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V189-L
Motor Operator	VBS-PL-V189-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V190-L
Motor Operator	VBS-PL-V190-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V191-L
Motor Operator	VBS-PL-V191-M
Air Delivery Isolation Valve A	
Limit Switch	VES-PL-V005A-L
Solenoid Operator	VES-PL-V005A-S
Voltage Control Box	VES-DJ-01
Air Delivery Isolation Valve B	
Limit Switch	VES-PL-V005B-L
Solenoid Operator	VES-PL-V005B-S
Voltage Control Box	VES-DJ-02
Relief Isolation Valve A	
Limit Switch	VES-PL-V022A-L
Solenoid Valve	VES-PL-V022A-S
Relief Isolation Valve B	
Limit Switch	VES-PL-V022B-L
Solenoid Valve	VES-PL-V022B-S
Containment Purge Inlet Isolation	
Limit Switch	VFS-PL-V003-L
Solenoid Valve	VFS-PL-V003-S1
Containment Purge Inlet Isolation	
Limit Switch	VFS-PL-V004-L
Solenoid Valve	VFS-PL-V004-S1

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Containment Purge Discharge Isolation	
Limit Switch	VFS-PL-V009-L
Solenoid Valve	VFS-PL-V009-S1
Containment Purge Discharge Isolation	
Limit Switch	VFS-PL-V010-L
Solenoid Valve	VFS-PL-V010-S1
Vacuum Relief Containment Isolation Valve A - ORC	
Limit Switch	VFS-PL-V800A-L
Motor Operator	VFS-PL-V800A-M
Vacuum Relief Containment Isolation Valve B - ORC	
Limit Switch	VFS-PL-V800B-L
Motor Operator	VFS-PL-V800B-M
Fan Cooler Supply Isolation	
Limit Switch	VWS-PL-V058-L
Solenoid Valve	VWS-PL-V058-S
Fan Cooler Return Isolation	
Limit Switch	VWS-PL-V082-L
Solenoid Valve	VWS-PL-V082-S
Fan Cooler Return Isolation	
Limit Switch	VWS-PL-V086-L
Solenoid Valve	VWS-PL-V086-S
Sump Containment Isolation IRC	
Limit Switch	WLS-PL-V055-L
Solenoid Valve	WLS-PL-V055-S1
Sump Containment Isolation ORC	
Limit Switch	WLS-PL-V057-L
Solenoid Valve	WLS-PL-V057-S1
RCDT Gas Containment Isolation	
Limit Switch	WLS-PL-V067-L
Solenoid Valve	WLS-PL-V067-S
RCDT Gas Containment Isolation	
Limit Switch	WLS-PL-V068-L
Solenoid Valve	WLS-PL-V068-S
Nonactive Valves	
Hot Leg 1 Sample Isolation	
Limit Switch	PSS-PL-V001A-L
Solenoid Valve	PSS-PL-V001A-S
Voltage Control Box	PSS-DJ-01
Hot Leg 2 Sample Isolation	
Limit Switch	PSS-PL-V001B-L
Solenoid Valve	PSS-PL-V001B-S
Voltage Control Box	PSS-DJ-02
Core Makeup Tank A CL Inlet Isolation	
Limit Switch	PXS-PL-V002A-L

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
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Description	AP1000 Tag Number
Motor Operator	PXS-PL-V002A-M
Core Makeup Tank B CL Inlet Isolation	
Limit Switch	PXS-PL-V002B-L
Motor Operator	PXS-PL-V002B-M
Accumulator A N2 Vent Isolation	
Limit Switch	PXS-PL-V021A-L
Solenoid Valve	PXS-PL-V021A-S
Accumulator B N2 Vent Isolation	
Limit Switch	PXS-PL-V021B-L
Solenoid Valve	PXS-PL-V021B-S
Accumulator A Discharge Isolation	
Limit Switch	PXS-PL-V027A-L
Motor Operator	PXS-PL-V027A-M
Accumulator B Discharge Isolation	
Limit Switch	PXS-PL-V027B-L
Motor Operator	PXS-PL-V027B-M
PRHR HX Inlet Isolation	
Limit Switch	PXS-PL-V101-L
Motor Operator	PXS-PL-V101-M
Recirc Sump A Isolation	
Limit Switch	PXS-PL-V117A-L
Motor Operator	PXS-PL-V117A-M

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List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Recirc Sump B Isolation	
Limit Switch	PXS-PL-V117B-L
Motor Operator	PXS-PL-V117B-M
IRWST Line A Isolation	
Limit Switch	PXS-PL-V121A-L
Motor Operator	PXS-PL-V121A-M
IRWST Line B Isolation	
Limit Switch	PXS-PL-V121B-L
Motor Operator	PXS-PL-V121B-M
Core Makeup Tank A Fill Isolation	
Limit Switch	PXS-PL-V230A-L
Solenoid Valve	PXS-PL-V230A-S
Core Makeup Tank B Fill Isolation	
Limit Switch	PXS-PL-V230B-L
Solenoid Valve	PXS-PL-V230B-S
Accumulator A Fill/Drain Isolation	
Limit Switch	PXS-PL-V232A-L
Solenoid Valve	PXS-PL-V232A-S
Accumulator B Fill/Drain Isolation	
Limit Switch	PXS-PL-V232B-L
Solenoid Valve	PXS-PL-V232B-S
Fourth Stage ADS Isolation	
Limit Switch	RCS-PL-V014A-L
Motor Operator	RCS-PL-V014A-M
Fourth Stage ADS Isolation	
Limit Switch	RCS-PL-V014B-L
Motor Operator	RCS-PL-V014B-M
Fourth Stage ADS Isolation	
Limit Switch	RCS-PL-V014C-L
Motor Operator	RCS-PL-V014C-M
Fourth Stage ADS Isolation	
Limit Switch	RCS-PL-V014D-L
Motor Operator	RCS-PL-V014D-M

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Table 3I.6-3 (Sheet 1 of 23)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Resistance Temperature Detectors		
PRHR HX Outlet Temperature	RCS-JE-TE161	1
RCS Cold Leg 1A Narrow Range Temperature	RCS-JE-TE121A	1
RCS Cold Leg 1A Narrow Range Temperature	RCS-JE-TE121D	1
RCS Cold Leg 1B Narrow Range Temperature	RCS-JE-TE121B	1
RCS Cold Leg 1B Narrow Range Temperature	RCS-JE-TE121C	1
RCS Cold Leg 2A Narrow Range Temperature	RCS-JE-TE122B	1
RCS Cold Leg 2A Narrow Range Temperature	RCS-JE-TE122C	1
RCS Cold Leg 2B Narrow Range Temperature	RCS-JE-TE122A	1
RCS Cold Leg 2B Narrow Range Temperature	RCS-JE-TE122D	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE131A	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE131C	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE132A	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE132C	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE133C	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE133A	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE131B	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE131D	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE132B	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE132D	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE133B	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE133D	1
RCS Cold Leg 1A Dual Range Temperature	RCS-JE-TE125A	1
RCS Cold Leg 1B Dual Range Temperature	RCS-JE-TE125C	1
RCS Cold Leg 2A Dual Range Temperature	RCS-JE-TE125B	1
RCS Cold Leg 2B Dual Range Temperature	RCS-JE-TE125D	1
RCS Hot Leg 1 Wide Range Temperature	RCS-JE-TE135A	1
RCS Hot Leg 2 Wide Range Temperature	RCS-JE-TE135B	1
PZR Reference Leg Level Temperature	RCS-JE-TE193A	1
PZR Reference Leg Level Temperature	RCS-JE-TE193B	1
PZR Reference Leg Level Temperature	RCS-JE-TE193C	1
PZR Reference Leg Level Temperature	RCS-JE-TE193D	1

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
RCP 1A Bearing Water Temperature	RCS-JE-TE211A	1
RCP 1A Bearing Water Temperature	RCS-JE-TE211B	1
RCP 1A Bearing Water Temperature	RCS-JE-TE211C	1
RCP 1A Bearing Water Temperature	RCS-JE-TE211D	1
RCP 1B Bearing Water Temperature	RCS-JE-TE212A	1
RCP 1B Bearing Water Temperature	RCS-JE-TE212B	1
RCP 1B Bearing Water Temperature	RCS-JE-TE212C	1
RCP 1B Bearing Water Temperature	RCS-JE-TE212D	1
RCP 2A Bearing Water Temperature	RCS-JE-TE213A	1
RCP 2A Bearing Water Temperature	RCS-JE-TE213B	1
RCP 2A Bearing Water Temperature	RCS-JE-TE213C	1
RCP 2A Bearing Water Temperature	RCS-JE-TE213D	1
RCP 2B Bearing Water Temperature	RCS-JE-TE214A	1
RCP 2B Bearing Water Temperature	RCS-JE-TE214B	1
RCP 2B Bearing Water Temperature	RCS-JE-TE214C	1
RCP 2B Bearing Water Temperature	RCS-JE-TE214D	1
Thermocouples		1
Incore Thermocouples (Note 4)	IIS-JE-TE001-TE042	1
Penetrations		
Penetrations (Mechanical)		1
Penetrations (Electrical)		1
Active Valves		
Containment Isolation – Air Out	CAS-PL-V014	2
Containment Isolation – Air In	CAS-PL-V015	2
Containment Isolation – Inlet	CCS-PL-V200	2
Service Air Supply Inside Containment Isolation	CAS-PL-V205	2
Containment Isolation – Inlet	CCS-PL-V201	2
Containment Isolation – Outlet	CCS-PL-V207	2
Containment Isolation – Outlet	CCS-PL-V208	2
CCS Containment Isolation Relief	CCS-PL-V220	2
CCS IRC Relief Valve	CCS-PL-V270	2
CCS IRC Relief Valve	CCS-PL-V271	2
RCS Purification Stop Valve	CVS-PL-V001	2
RCS Purification Stop Valve	CVS-PL-V002	2
RCS Letdown Stop Valve	CVS-PL-V003	2
Demineralizer Flush Line Relief Valve	CVS-PL-V042	2

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Description	AP1000 Tag Number	Comment
WLS Letdown IRC Isolation	CVS-PL-V045	2
Letdown Flow ORC Isolation	CVS-PL-V047	2
Letdown Line Containment Isolation Thermal Relief	CVS-PL-V058	2
CVS Makeup Line Bypass Check Valve	CVS-PL-V067	2
RCS Purification Check Valve	CVS-PL-V080	2
RCS Purification Stop Valve	CVS-PL-V081	2
RCS Purification Check Valve	CVS-PL-V082	2
Auxiliary PZR Spray Isolation	CVS-PL-V084	2
Auxiliary PZR Spray Isolation	CVS-PL-V085	2
Makeup Line Containment Isolation	CVS-PL-V090	2
Makeup Line Containment Isolation	CVS-PL-V091	2
Zinc Addition Containment Isolation	CVS-PL-V092	2
Zinc Addition Containment Isolation	CVS-PL-V094	2
Zinc Injection Containment Isolation Thermal Overpressurization Relief Valve	CVS-PL-V098	2
Makeup Containment Isolation	CVS-PL-V100	2
Demineralizer Water System Isolation	CVS-PL-V136A	2
Demineralized Water System Isolation	CVS-PL-V136B	2
Hydrogen Injection Containment Isolation Check Valve	CVS-PL-V217	2
Hydrogen Injection Containment Isolation	CVS-PL-V219	2
DWS Containment Penetration Thermal Relief Valve	DWS-PL-V241	2
Demin Water Supply Containment Isolation – Inside	DWS-PL-V245	2
Fuel Transfer Tube Gate Valve	FHS-PL-V001	2
Fire Water Containment Supply Isolation – Inside	FPS-PL-V052	2
FPS Containment Penetration Thermal Relief Valve	FPS-PL-V702	2
PCCWST Isolation Valve	PCS-PL-V001A	2
PCCWST Isolation Valve	PCS-PL-V001B	2
PCCWST Isolation Valve	PCS-PL-V001C	2
PCCWST Isolation Valve	PCS-PL-V002A	2
PCCWST Isolation Valve	PCS-PL-V002B	2
PCCWST Isolation Valve	PCS-PL-V002C	2
PCCWST Fire Protection Isolation	PCS-PL-V005	2
PCCWST Emergency Spent Fuel Pool Makeup Isolation	PCS-PL-V009	2
Water Bucket Makeup Line Drain Valve	PCS-PL-V015	2
Water Bucket Makeup Line Isolation Valve	PCS-PL-V020	2
PCS Recirculation Isolation	PCS-PL-V023	2
PCCWST Long-Term Makeup Check Valve	PCS-PL-V039	2
PCCWST Long Term Makeup Isolation Drain Valve	PCS-PL-V042	2
PCCWST Long Term Makeup Isolation Valve	PCS-PL-V044	2
Emergency Makeup to the Spent Fuel Pool Isolation Valve	PCS-PL-V045	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
PCCWST Recirculation Return Isolation Valve	PCS-PL-V046	2
Emergency Makeup to the Spent Fuel Pool Drain Isolation Valve	PCS-PL-V049	2
Spent Fuel Pool Long Term Makeup Isolation Valve	PCS-PL-V050	2
Spent Fuel Pool Emergency Makeup Lower Isolation Valve	PCS-PL-V051	2
Shutoff Valve for Leakage Sensor	PCS-PL-V060A	2
Shutoff Valve for Leakage Sensor	PCS-PL-V060B	2
Containment Isolation – Air Sample Line	PSS-PL-V008	2
Containment Isolation – Liquid Sample Line	PSS-PL-V010A	2
Containment Isolation – Liquid Sample Line	PSS-PL-V010B	2
Containment Isolation – Liquid Sample Line	PSS-PL-V011A	2
Containment Isolation – Liquid Sample Line	PSS-PL-V011B	2
Containment Isolation – Sample Return Line	PSS-PL-V023	2
Containment Isolation Sample Return	PSS-PL-V024	2
Containment Isolation – Air Sample Line	PSS-PL-V046	2
PWS MCR Isolation Valve	PWS-PL-V418	2
PWS MCR Isolation Valve	PWS-PL-V420	2
PWS MCR Vacuum Relief	PWS-PL-V498	2
Core Makeup Tank A Discharge Isolation	PXS-PL-V014A	2
Core Makeup Tank B Discharge Isolation	PXS-PL-V014B	2
Core Makeup Tank A Discharge Isolation	PXS-PL-V015A	2
Core Makeup Tank B Discharge Isolation	PXS-PL-V015B	2
Core Makeup Tank A Discharge	PXS-PL-V016A	2
Core Makeup Tank B Discharge	PXS-PL-V016B	2
Core Makeup Tank A Discharge	PXS-PL-V017A	2
Core Makeup Tank B Discharge	PXS-PL-V017B	2
Accumulator A Pressure Relief	PXS-PL-V022A	2
Accumulator B Pressure Relief	PXS-PL-V022B	2
Accumulator A Discharge	PXS-PL-V028A	2
Accumulator B Discharge	PXS-PL-V028B	2
Accumulator A Discharge	PXS-PL-V029A	2
Accumulator B Discharge	PXS-PL-V029B	2
Nitrogen Supply Outside Containment Isolation	PXS-PL-V042	2
Nitrogen Supply Containment Isolation IRC Check Valve	PXS-PL-V043	2
PRHR HX Discharge Isolation	PXS-PL-V108A	2
PRHR HX Discharge Isolation	PXS-PL-V108B	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Recirc Sump A Isolation	PXS-PL-V118A	2
Recirc Sump B Isolation	PXS-PL-V118B	2
Recirc Sump A Isolation	PXS-PL-V119A	2
Recirc Sump B Isolation	PXS-PL-V119B	2
Recirc Sump A Isolation	PXS-PL-V120A	2
Recirc Sump B Isolation	PXS- PL-V120B	2
IRWST Injection A	PXS-PL-V122A	2
IRWST Injection B	PXS-PL-V122B	2
IRWST Injection A	PXS-PL-V123A	2
IRWST Injection B	PXS-PL-V123B	2
IRWST Injection A	PXS-PL-V124A	2
IRWST Injection B	PXS-PL-V124B	2
IRWST Injection A	PXS-PL-V125A	2
IRWST Injection B	PXS-PL-V125B	2
IRWST Gutter Drain Isolation A	PXS-PL-V130A	2
IRWST Gutter Drain Isolation B	PXS-PL-V130B	2
First Stage ADS	RCS-PL-V001A	2
First Stage ADS	RCS-PL-V001B	2
Second Stage ADS	RCS-PL-V002A	2
Second Stage ADS	RCS-PL-V002B	2
Third Stage ADS	RCS-PL-V003A	2
Third Stage ADS	RCS-PL-V003B	2
Fourth Stage ADS	RCS-PL-V004A	2
Fourth Stage ADS	RCS-PL-V004B	2
Fourth Stage ADS	RCS-PL-V004C	2
Fourth Stage ADS	RCS-PL-V004D	2
PZR Safety Valve	RCS-PL-V005A	2
PZR Safety Valve	RCS-PL-V005B	2
ADS Discharge Header A Relief	RCS-PL-V010A	2
ADS Discharge Header B Relief	RCS-PL-V010B	2
First Stage ADS Isolation	RCS-PL-V011A	2
First Stage ADS Isolation	RCS-PL-V011B	2
Second Stage ADS Isolation	RCS-PL-V012A	2
Second Stage ADS Isolation	RCS-PL-V012B	2
Third Stage ADS Isolation	RCS-PL-V013A	2
Third Stage ADS Isolation	RCS-PL-V013B	2

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Description	AP1000 Tag Number	Comment
Reactor Vessel Head Vent	RCS-PL-V150A	2
Reactor Vessel Head Vent	RCS-PL-V150B	2
Reactor Vessel Head Vent	RCS-PL-V150C	2
Reactor Vessel Head Vent	RCS-PL-V150D	2
RCS Inner Suction Isolation	RNS-PL-V001A	2
RCS Inner Suction Isolation	RNS-PL-V001B	2
RCS Outer Suction Isolation	RNS-PL-V002A	2
RCS Outer Suction Isolation	RNS-PL-V002B	2
RCS Thermal Relief	RNS-PL-V003A	2
RCS Thermal Relief	RNS-PL-V003B	2
RNS Discharge Containment Isolation Valve	RNS-PL-V011	2
RNS Discharge Containment Isolation Valve Test Connection	RNS-PL-V012	2
RNS Discharge Containment Isolation	RNS-PL-V013	2
RNS Discharge RCS Pressure Boundary	RNS-PL-V015A	2
RNS Discharge RCS Pressure Boundary	RNS-PL-V015B	2
RNS Discharge RCS Pressure Boundary	RNS-PL-V017A	2
RNS Discharge RCS Pressure Boundary	RNS-PL-V017B	2
RNS Hot Leg Suction Relief	RNS-PL-V020	2
RNS Hot Leg Suction Relief	RNS-PL-V021	2
RNS Suction Header Containment Isolation	RNS-PL-V022	2
IRWST Suction Line Isolation	RNS-PL-V023	2
RNS Pump Discharge Relief	RNS-PL-V045	2
RNS – CVS Containment Isolation	RNS-PL-V061	2
SDS MCR Vent Isolation Valve	SDS-PL-V001	2
SDS MCR Vent Isolation Valve	SDS-PL-V002	2
SFS Suction Line Containment Isolation	SFS-PL-V034	2
SFS Suction Line Containment Isolation	SFS-PL-V035	2
SFS Discharge Containment Isolation	SFS-PL-V037	2
SFS Discharge Line Containment Isolation	SFS-PL-V038	2
SFS Cask Loading Pit to SFS Pump	SFS-PL-V042	2
SFS Pump to Cask Loading Pit	SFS-PL-V045	2
Cask Loading Pit to WLS	SFS-PL-V049	2
Spent Fuel Pool to Cask Washdown Pit Isolation	SFS-PL-V066	2
SFS Containment Isolation Relief	SFS-PL-V067	2
Cask Washdown Pit Drain Isolation	SFS-PL-V068	2
Refueling Cavity to SG Compartment	SFS-PL-V071	2
Refueling Cavity to SG Compartment	SFS-PL-V072	2
Refueling Cavity Drain Line Test Connection	SFS-PL-V117	2
PORV Block Valve	SGS-PL-V027A	2
PORV Block Valve	SGS-PL-V027B	2

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Description	AP1000 Tag Number	Comment
Steam Safety Valve SG01	SGS-PL-V030A	2
Steam Safety Valve SG02	SGS-PL-V030B	2
Steam Safety Valve SG01	SGS-PL-V031A	2
Steam Safety Valve SG02	SGS-PL-V031B	2
Steam Safety Valve SG01	SGS-PL-V032A	2
Steam Safety Valve SG02	SGS-PL-V032B	2
Steam Safety Valve SG01	SGS-PL-V033A	2
Steam Safety Valve SG02	SGS-PL-V033B	2
Steam Safety Valve SG01	SGS-PL-V034A	2
Steam Safety Valve SG02	SGS-PL-V034B	2
Steam Safety Valve SG01	SGS-PL-V035A	2
Steam Safety Valve SG02	SGS-PL-V035B	2
Steam Line Condensate Drain Isolation	SGS-PL-V036A	2
Steam Line Condensate Drain Isolation	SGS-PL-V036B	2
Main Steam Line Isolation	SGS-PL-V040A	2
Main Steam Line Isolation	SGS-PL-V040B	2
Main Feedwater Isolation	SGS-PL-V057A	2
Main Feedwater Isolation	SGS-PL-V057B	2
Startup Feedwater Isolation	SGS-PL-V067A	2
Startup Feedwater Isolation	SGS-PL-V067B	2
SG Blowdown Isolation	SGS-PL-V074A	2
SG Blowdown Isolation	SGS-PL-V074B	2
SG Series Blowdown Isolation	SGS-PL-V075A	2
SG Series Blowdown Isolation	SGS-PL-V075B	2
Steam Line Condensate Drain Isolation	SGS-PL-V086A	2
Steam Line Condensate Drain Isolation	SGS-PL-V086B	2
Power-Operated Relief Valve	SGS-PL-V233A	2
Power-Operated Relief Valve	SGS-PL-V233B	2
MSIV Bypass Isolation Valve	SGS-PL-V240A	2
MSIV Bypass Isolation Valve	SGS-PL-V240B	2
Main Feedwater Control Valve	SGS-PL-V250A	2
Main Feedwater Control Valve	SGS-PL-V250B	2
Startup Feedwater Control Valve	SGS-PL-V255A	2
Startup Feedwater Control Valve	SGS-PL-V255B	2
Main Feedwater Thermal Relief Valve	SGS-PL-V257A	2
Main Feedwater Thermal Relief Valve	SGS-PL-V257B	2
Startup Feedwater Thermal Relief Valve	SGS-PL-V258A	2
Startup Feedwater Thermal Relief Valve	SGS-PL-V258B	2
MCR Isolation Valve	VBS-PL-V186	2
MCR Isolation Valve	VBS-PL-V187	2
MCR Isolation Valve	VBS-PL-V188	2
MCR Isolation Valve	VBS-PL-V189	2
MCR Isolation Valve	VBS-PL-V190	2
MCR Isolation Valve	VBS-PL-V191	2

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Description	AP1000 Tag Number	Comment
Air Delivery Isolation Valve	VES-PL-V001	2
Pressure Regulator Valve A	VES-PL-V002A	2
Pressure Regulator Valve B	VES-PL-V002B	2
Air Delivery Isolation Valve A	VES-PL-V005A	2
Air Delivery Isolation Valve B	VES-PL-V005B	2
Temporary Instrument Isolation Valve A	VES-PL-V018	2
Temporary Instrument Isolation Valve B	VES-PL-V019	2
Relief Isolation Valve A	VES-PL-V022A	2
Relief Isolation Valve B	VES-PL-V022B	2
Air Tank Safety Relief Valve A	VES-PL-V040A	2
Air Tank Safety Relief Valve B	VES-PL-V040B	2
Air Tank Safety Relief Valve C	VES-PL-V040C	2
Air Tank Safety Relief Valve D	VES-PL-V040D	2
Main Air Flow Path Isolation Valve	VES-PL-V044	2
Containment Purge Inlet Isolation	VFS-PL-V003	2
Containment Purge Inlet Isolation	VFS-PL-V004	2
Containment Purge Discharge Isolation	VFS-PL-V009	2
Containment Purge Discharge Isolation	VFS-PL-V010	2
Vacuum Relief Containment Isolation Valve A - ORC	VFS-PL-V800A	2
Vacuum Relief Containment Isolation Valve B - ORC	VFS-PL-V800B	2
Vacuum Relief Containment Isolation Check Valve A - IRC	VFS-PL-V803A	2
Vacuum Relief Containment Isolation Check Valve B - IRC	VFS-PL-V803B	2
VWS Containment Penetration Thermal Relief Valve	VWS-PL-V053	2
VWS Containment Penetration Thermal Relief Valve	VWS-PL-V057	2
Fan Cooler Supply Isolation	VWS-PL-V058	2
Fan Coolers Supply Containment Isolation Check Valve	VWS-PL-V062	2
VWS Containment Isolation Relief	VWS-PL-V080	2
Fan Cooler Return Isolation	VWS-PL-V082	2
Fan Cooler Return Isolation	VWS-PL-V086	2
Sump Containment Isolation IRC	WLS-PL-V055	2
Sump Containment Isolation ORC	WLS-PL-V057	2
WLS Containment Isolation Relief	WLS-PL-V058	2
RCDT Gas Containment Isolation	WLS-PL-V067	2
RCDT Gas Containment Isolation	WLS-PL-V068	2
CVS To Sump	WLS-PL-V071 A	2
PXS A To Sump	WLS-PL-V071 B	2
PXS B To Sump	WLS-PL-V071 C	2
CVS To Sump	WLS-PL-V072 A	2
PXS A To Sump	WLS-PL-V072 B	2
PXS B To Sump	WLS-PL-V072 C	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Miscellaneous		
Nonactive Valves		
Containment Penetration Test Connection Isolation	CAS-PL-V027	2
Service Air Supply Outside Containment Isolation	CAS-PL-V204	2
Containment Penetration Test Connection Isolation	CAS-PL-V219	2
Containment Isolation Valve Test Connection – Outlet Line	CCS-PL-V209	2
CCS Supply Containment Isolation – IRC	CCS-PL-V214	2
CCS Supply Containment Isolation Valve Test Connection – IRC	CCS-PL-V215	2
Containment Leak Test Outlet Line – IRC	CCS-PL-V216	2
Containment Isolation Valve V207 Body Test Connection Valve	CCS-PL-V217	2
Containment Isolation Valve Test Connection – Inlet Line	CCS-PL-V257	2
Resin Flush IRC Isolation	CVS-PL-V040	2
Resin Flush ORC Isolation	CVS-PL-V041	2
Letdown PZR Instrument Root	CVS-PL-V046	2
Zinc Addition – IRC Shutoff	CVS-PL-V065	2
Zinc Add Cont Isolation Test Connection	CVS-PL-V095	2
Zinc Addition Containment Isolation Test Connection	CVS-PL-V096	2
Hydrogen Injection Containment Isolation Test Connection Valve	CVS-PL-V218	2
Demin Water Supply Containment Isolation – Outside	DWS-PL-V244	2
Containment Penetration Test Connection Isolation	DWS-PL-V248	2
Fire Water Containment Supply Isolation	FPS-PL-V050	2
Fire Water Containment Test Connection Isolation	FPS-PL-V051	2
Flow Transmitter FT001 Root Valve	PCS-PL-V010A	2
Flow Transmitter FT001 Root Valve	PCS-PL-V010B	2
Flow Transmitter FT002 Root Valve	PCS-PL-V011A	2
Flow Transmitter FT002 Root Valve	PCS-PL-V011B	2
Flow Transmitter FT003 Root Valve	PCS-PL-V012A	2
Flow Transmitter FT003 Root Valve	PCS-PL-V012B	2
Flow Transmitter FT004 Root Valve	PCS-PL-V013A	2
Flow Transmitter FT004 Root Valve	PCS-PL-V013B	2
PCCWST Drain Isolation Valve	PCS-PL-V016	2
Makeup to Distribution Bucket Isolation Valve	PCS-PL-V026	2
PCCWST Isolation Valve Leakage Detection Drain	PCS-PL-V029	2
PCCWST Isolation Valve Leakage Detection Crossconn	PCS-PL-V030	2
PCCWST Level Instrument Root Valve	PCS-PL-V031A	2
PCCWST Level Instrument Root Valve	PCS-PL-V031B	2
Recirculation Pump Suction from Long Term Makeup Isolation Valve	PCS-PL-V033	2
Spent Fuel Pool Emergency Makeup Isolation	PCS-PL-V052	2
Recirculation Header Discharge to SFS Pool Vent Isolation Valve	PCS-PL-V303	2
Recirculation Header Discharge to SFS Pool Drain Isolation Valve	PCS-PL-V304	2
PCCWST Recirculation Return Drain Isolation Valve	PCS-PL-V305	2

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Description	AP1000 Tag Number	Comment
Hot Leg 1 Sample Isolation	PSS-PL-V001A	2
Hot Leg 2 Sample Isolation	PSS-PL-V001B	2
Pressurizer Sample Isolation	PSS-PL-V003	2
PXS Accumulator Sample Isolation	PSS-PL-V004A	2
PXS Accumulator Sample Isolation	PSS-PL-V004B	2
PXS CMT A Sample Isolation	PSS-PL-V005A	2
PXS CMT B Sample Isolation	PSS-PL-V005B	2
PXS CMT A Sample Isolation	PSS-PL-V005C	2
PXS CMT B Sample Isolation	PSS-PL-V005D	2
Liquid Sample Isolation Valve	PSS-PL-V012A	2
Liquid Sample Check Valve	PSS-PL-V012B	2
RCS Pressurizer Sample Isolation Valve	PSS-PL-V013	2
RCS Hot Leg 1 Sample Isolation Valve	PSS-PL-V014A	2
RCS Hot Leg 2 Sample Isolation Valve	PSS-PL-V014B	2
PXS Accumulator Sample Isolation Valve	PSS-PL-V015A	2
PXS Accumulator Sample Isolation Valve	PSS-PL-V015B	2
PXS CMT A Sample Isolation Valve	PSS-PL-V016A	2
PXS CMT B Sample Isolation Valve	PSS-PL-V016B	2
PXS CMT A Sample Isolation Valve	PSS-PL-V016C	2
PXS CMT B Sample Isolation Valve	PSS-PL-V016D	2
Containment Testing Boundary Isolation Valve	PSS-PL-V076A	2
Containment Testing Boundary Isolation Valve	PSS-PL-V076B	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V082	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V083	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V085	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V086	2
PXS CMT A Sample Isolation Valve	PSS-PL-V902A	2
PXS CMT B Sample Isolation Valve	PSS-PL-V902B	2
PXS CMT B Sample Isolation Valve	PSS-PL-V903	2
Core Makeup Tank A CL Inlet Isolation	PXS-PL-V002A	2
Core Makeup Tank B CL Inlet Isolation	PXS-PL-V002B	2
Core Makeup Tank A Upper Sample	PXS-PL-V010A	2
Core Makeup Tank B Upper Sample	PXS-PL-V010B	2
Core Makeup Tank A Lower Sample	PXS-PL-V011A	2
Core Makeup Tank B Lower Sample	PXS-PL-V011B	2
Core Makeup Tank A Drain	PXS-PL-V012A	2
Core Makeup Tank B Drain	PXS-PL-V012B	2
Core Makeup Tank A Discharge Manual Isolation	PXS-PL-V013A	2
Core Makeup Tank B Discharge Manual Isolation	PXS-PL-V013B	2
RNS to CMT Injection Line A Drain	PXS-PL-V019A	2
RNS to CMT Injection Line B Drain	PXS-PL-V019B	2
IRWST Injection Line A Drain	PXS-PL-V020A	2
IRWST Injection Line B Drain	PXS-PL-V020B	2

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Description	AP1000 Tag Number	Comment
Accumulator A N_2 Vent	PXS-PL-V021A	2
Accumulator B N_2 Vent	PXS-PL-V021B	2
Accumulator A PZR Transmitter Isolation	PXS-PL-V023A	2
Accumulator B PZR Transmitter Isolation	PXS-PL-V023B	2
Accumulator A PZR Transmitter Isolation	PXS-PL-V024A	2
Accumulator B PZR Transmitter Isolation	PXS-PL-V024B	2
Accumulator A Sample	PXS-PL-V025A	2
Accumulator B Sample	PXS-PL-V025B	2
Accumulator A Drain	PXS-PL-V026A	2
Accumulator B Drain	PXS-PL-V026B	2
Accumulator A Discharge Isolation	PXS-PL-V027A	2
Accumulator B Discharge Isolation	PXS-PL-V027B	2
Core Makeup Tank A Highpoint Vent	PXS-PL-V030A	2
Core Makeup Tank B Highpoint Vent	PXS-PL-V030B	2
Core Makeup Tank A Highpoint Vent	PXS-PL-V031A	2
Core Makeup Tank B Highpoint Vent	PXS-PL-V031B	2
Accumulator A Check Valve Drain	PXS-PL-V033A	2
Accumulator B Check Valve Drain	PXS-PL-V033B	2
Accumulator N_2 Containment Penetration Test Connection	PXS-PL-V052	2
CMT A Wide Level Upper Root	PXS-PL-V080A	2
CMT B Wide Level Upper Root	PXS-PL-V080B	2
CMT A Wide Level Lower Root	PXS-PL-V081A	2
CMT B Wide Level Lower Root	PXS-PL-V081B	2
CMT A Upper Level A Isolation 1	PXS-PL-V082A	2
CMT B Upper Level A Isolation 1	PXS-PL-V082B	2
CMT A Upper Level A Isolation 2	PXS-PL-V083A	2
CMT B Upper Level A Isolation 2	PXS-PL-V083B	2
CMT A Upper Level A Vent	PXS-PL-V084A	2
CMT B Upper Level A Vent	PXS-PL-V084B	2
LT-011 A/C Drain Valve	PXS-PL-V085A	2
LT-012 A/C Drain Valve	PXS-PL-V085B	2
CMT A Upper Level B Isolation 1	PXS-PL-V086A	2
CMT B Upper Level B Isolation 1	PXS-PL-V086B	2
CMT A Upper Level B Isolation 2	PXS-PL-V087A	2
CMT B Upper Level B Isolation 2	PXS-PL-V087B	2
CMT A Upper Level B Vent	PXS-PL-V088A	2
CMT B Upper Level B Vent	PXS-PL-V088B	2
CMT A Upper Level B Drain	PXS-PL-V089A	2
CMT B Upper Level B Drain	PXS-PL-V089B	2

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Description	AP1000 Tag Number	Comment
CMT A Lower Level A Isolation 1	PXS-PL-V092A	2
CMT B Lower Level A Isolation 1	PXS-PL-V092B	2
CMT A Lower Level A Isolation 2	PXS-PL-V093A	2
CMT B Lower Level A Isolation 2	PXS-PL-V093B	2
CMT A Lower Level A Vent	PXS-PL-V094A	2
CMT B Lower Level A Vent	PXS-PL-V094B	2
CMT A Lower Level A Drain	PXS-PL-V095A	2
CMT B Lower Level A Drain	PXS-PL-V095B	2
CMT A Lower Level B Isolation 1	PXS-PL-V096A	2
CMT B Lower Level B Isolation 1	PXS-PL-V096B	2
CMT A Lower Level B Isolation 2	PXS-PL-V097A	2
CMT B Lower Level B Isolation 2	PXS-PL-V097B	2
CMT A Lower Level B Vent	PXS-PL-V098A	2
CMT B Lower Level B Vent	PXS-PL-V098B	2
CMT A Lower Level B Drain	PXS-PL-V099A	2
CMT B Lower Level B Drain	PXS-PL-V099B	2
PRHR HX Inlet Isolation	PXS-PL-V101	2
PRHR HX Inlet Head Vent	PXS-PL-V102A	2
PRHR HX Inlet Head Drain	PXS-PL-V102B	2
PRHR HX Outlet Head Vent	PXS-PL-V103A	2
PRHR HX Outlet Head Drain	PXS-PL-V103B	2
PRHR HX Flow Transmitter A Isolation	PXS-PL-V104A	2
PRHR HX Flow Transmitter B Isolation	PXS-PL-V104B	2
PRHR HX Flow Transmitter A Isolation	PXS-PL-V105A	2
PRHR HX Flow Transmitter B Isolation	PXS-PL-V105B	2
Containment Recirculation A Highpoint Vent	PXS-PL-V106	2
Containment Recirculation A Highpoint Vent	PXS-PL-V107	2
PRHR HX/RCS Return Isolation	PXS-PL-V109	2
PRHR HX Highpoint Vent	PXS-PL-V111A	2
PRHR HX Highpoint Vent	PXS-PL-V111B	2
PRHR HX Pressure Transmitter Isolation	PXS-PL-V113	2
Containment Recirculation A Drain	PXS-PL-V115A	2
Containment Recirculation B Drain	PXS-PL-V115B	2
Containment Recirculation A Drain	PXS-PL-V116A	2
Containment Recirculation B Drain	PXS-PL-V116B	2
Recirc Sump A Isolation	PXS-PL-V117A	2
Recirc Sump B Isolation	PXS-PL-V117B	2
IRWST Line A Isolation	PXS-PL-V121A	2
IRWST Line B Isolation	PXS-PL-V121B	2

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Description	AP1000 Tag Number	Comment
IRWST Injection Check Test	PXS-PL-V126A	2
IRWST Injection Check Test	PXS-PL-V126B	2
IRWST Injection Line A Drain	PXS-PL-V127	2
IRWST Injection Check Test	PXS-PL-V128A	2
IRWST Injection Check Test	PXS-PL-V128B	2
IRWST Injection Check Test	PXS-PL-V129A	2
IRWST Injection Check Test	PXS-PL-V129B	2
IRWST Injection Line A Drain	PXS-PL-V131A	2
IRWST Injection Line B Drain	PXS-PL-V131B	2
IRWST Injection Line A Drain	PXS-PL-V132A	2
IRWST Injection Line B Drain	PXS-PL-V132B	2
IRWST Injection Line A Highpoint Vent	PXS-PL-V133A	2
IRWST Injection Line B Highpoint Vent	PXS-PL-V133B	2
IRWST Injection Line A Highpoint Vent	PXS-PL-V134A	2
IRWST Injection Line B Highpoint Vent	PXS-PL-V134B	2
IRWST Injection Line A Highpoint Vent Isolation	PXS-PL-V135A	2
IRWST Injection Line B Highpoint Vent Isolation	PXS-PL-V135B	2
RNS Suction Pump Line Drain	PXS-PL-V149	2
IRWST Wide Range Level Transmitter B Isolation	PXS-PL-V151B	2
IRWST Wide Range Level Transmitter C Isolation	PXS-PL-V151C	2
IRWST Wide Range Level Transmitter D Isolation	PXS-PL-V151D	2
IRWST Lower Narrow Range Level Transmitter A Isolation	PXS-PL-V161A	2
IRWST Lower Narrow Range Level Transmitter B Isolation	PXS-PL-V161B	2
IRWST Lower Narrow Range Level Transmitter C Isolation	PXS-PL-V161C	2
IRWST Lower Narrow Range Level Transmitter D Isolation	PXS-PL-V161D	2
IRWST Lower Narrow Range Level Transmitter A Reference Leg Isolation	PXS-PL-V162A	2
IRWST Lower Narrow Range Level Transmitter B Reference Leg Isolation	PXS-PL-V162B	2
IRWST Lower Narrow Range Level Transmitter C Reference Leg Isolation	PXS-PL-V162C	2
IRWST Lower Narrow Range Level Transmitter D Reference Leg Isolation	PXS-PL-V162D	2
PRHR Flow Transmitter A Vent	PXS-PL-V170A	2
PRHR Flow Transmitter B Vent	PXS-PL-V170B	2
PRHR Flow Transmitter A Vent	PXS-PL-V171A	2
PRHR Flow Transmitter B Vent	PXS-PL-V171B	2
Accumulator A Leak Test	PXS-PL-V201A	2
Accumulator B Leak Test	PXS-PL-V201B	2
Accumulator A Leak Test	PXS-PL-V202A	2
Accumulator B Leak Test	PXS-PL-V202B	2
RNS Discharge Leak Test	PXS-PL-V205A	2
RNS Discharge Leak Test	PXS-PL-V205B	2
RNS Discharge Leak Test	PXS-PL-V206	2
RNS Suction Leak Test	PXS-PL-V207A	2
RNS Suction Leak Test	PXS-PL-V207B	2

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Description	AP1000 Tag Number	Comment
RNS Suction Leak Test	PXS-PL-V208A	2
Core Makeup Tank A Fill Isolation	PXS-PL-V230A	2
Core Makeup Tank B Fill Isolation	PXS-PL-V230B	2
Core Makeup Tank A Fill Check	PXS-PL-V231A	2
Core Makeup Tank B Fill Check	PXS-PL-V231B	2
Accumulator A Fill/Drain Isolation	PXS-PL-V232A	2
Accumulator B Fill/Drain Isolation	PXS-PL-V232B	2
Accumulator A to PXS Test Header Iso Valve	PXS-PL-V233A	2
Accumulator B to PXS Test Header Iso Valve	PXS-PL-V233B	2
CMT A Check Valve Test Valve	PXS-PL-V250A	2
CMT B Check Valve Test Valve	PXS-PL-V250B	2
CMT A Check Valve Test Valve	PXS-PL-V251A	2
CMT B Check Valve Test Valve	PXS-PL-V251B	2
CMT A Check Valve Test Valve	PXS-PL-V252A	2
CMT B Check Valve Test Valve	PXS-PL-V252B	2
ADS Test Valve	RCS-PL-V007A	2
ADS Test Valve	RCS-PL-V007B	2
ADS Test Valve	RCS-PL-V007C	2
ADS Valve Leakage Check Valve	RCS-PL-V008	2
Fourth Stage ADS Isolation	RCS-PL-V014A	2
Fourth Stage ADS Isolation	RCS-PL-V014B	2
Fourth Stage ADS Isolation	RCS-PL-V014C	2
Fourth Stage ADS Isolation	RCS-PL-V014D	2
Hot Leg 2 Level Instrument Root	RCS-PL-V095	2
Hot Leg 2 Level Instrument Root	RCS-PL-V096	2
Hot Leg 1 Level Instrument Root	RCS-PL-V097	2
Hot Leg 1 Level Instrument Root	RCS-PL-V098	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101A	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101B	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101C	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101D	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101E	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101F	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102A	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102B	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102C	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102D	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102E	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102F	2
PRHR HX Outlet Line Drain	RCS-PL-V103	2
Hot Leg 1 Sample Isolation	RCS-PL-V108A	2
Hot Leg 2 Sample Isolation	RCS-PL-V108B	2
PZR Spray Valve	RCS-PL-V110A	2
PZR Spray Valve	RCS-PL-V110B	2

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Description	AP1000 Tag Number	Comment
PZR Spray Block Valve	RCS-PL-V111A	2
PZR Spray Block Valve	RCS-PL-V111B	2
Cold Leg 1A Bend Instrument Root	RCS-PL-V171A	2
Cold Leg 1A Bend Instrument Root	RCS-PL-V171B	2
Cold Leg 1B Bend Instrument Root	RCS-PL-V172A	2
Cold Leg 1B Bend Instrument Root	RCS-PL-V172B	2
Cold Leg 2A Bend Instrument Root	RCS-PL-V173A	2
Cold Leg 2A Bend Instrument Root	RCS-PL-V173B	2
Cold Leg 2B Bend Instrument Root	RCS-PL-V174A	2
Cold Leg 2B Bend Instrument Root	RCS-PL-V174B	2
PZR Manual Vent	RCS-PL-V204	2
PZR Manual Vent	RCS-PL-V205	2
PZR Spray Bypass	RCS-PL-V210A	2
PZR Spray Bypass	RCS-PL-V210B	2
PZR Level Steam Space Instrument Root	RCS-PL-V225A	2
PZR Level Steam Space Instrument Root	RCS-PL-V225B	2
PZR Level Steam Space Instrument Root	RCS-PL-V225C	2
PZR Level Steam Space Instrument Root	RCS-PL-V225D	2
PZR Level Liquid Space Instrument Root	RCS-PL-V226A	2
PZR Level Liquid Space Instrument Root	RCS-PL-V226B	2
PZR Level Liquid Space Instrument Root	RCS-PL-V226C	2
PZR Level Liquid Space Instrument Root	RCS-PL-V226D	2
Pressurizer Level Reference Leg Tubing Instrument Root	RCS-PL-V227A	2
Pressurizer Level Reference Leg Tubing Instrument Root	RCS-PL-V227B	2
Pressurizer Level Reference Leg Tubing Instrument Root	RCS-PL-V227C	2
Pressurizer Level Reference Leg Tubing Instrument Root	RCS-PL-V227D	2
Pressurizer Liquid Sample Isolation	RCS-PL-V228	2
Wide Range PZR Level Liquid Space Instrument Root	RCS-PL-V229	2
Manual Head Vent	RCS-PL-V232	2
Head Vent Isolation	RCS-PL-V233	2
ADS Valve Discharge Header Drain Isolation	RCS-PL-V241	2
ADS Discharge Line Isolation Valve	RCS-PL-V250	2
RCP 1A Flush	RCS-PL-V260A	2
RCP 1B Flush	RCS-PL-V260B	2
RCP 2A Flush	RCS-PL-V260C	2
RCP 2B Flush	RCS-PL-V260D	2
RCP 1A Drain	RCS-PL-V261A	2
RCP 1B Drain	RCS-PL-V261B	2
RCP 2A Drain	RCS-PL-V261C	2
RCP 2B Drain	RCS-PL-V261D	2
RCS Pressure Boundary Valve Thermal Relief Isolation	RNS-PL-V004A	2
RCS Pressure Boundary Valve Thermal Relief Isolation	RNS-PL-V004B	2
RNS Pump A Suction Isolation	RNS-PL-V005A	2
RNS Pump B Suction Isolation	RNS-PL-V005B	2
RNS HX A Outlet Flow Control	RNS-PL-V006A	2

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Description	AP1000 Tag Number	Comment
RNS HX B Outlet Flow Control	RNS-PL-V006B	2
RNS Pump A Discharge Isolation	RNS-PL-V007A	2
RNS Pump B Discharge Isolation	RNS-PL-V007B	2
RNS HX A Bypass Flow Control	RNS-PL-V008A	2
RNS HX B Bypass Flow Control	RNS-PL-V008B	2
RNS Discharge Containment Isolation Valve Test	RNS-PL-V010	2
RNS Discharge Containment Isolation Valve Test Connection	RNS-PL-V014	2
RNS Discharge Containment Penetration Isolation Valves Test	RNS-PL-V016	2
RNS Discharge to IRWST Isolation	RNS-PL-V024	2
RNS Suction from IRWST – Bonnet Relief Isolation	RNS-PL-V025	2
RNS Suction from IRWST – Containment Isolation Test	RNS-PL-V026	2
RNS Discharge to CVS	RNS-PL-V029	2
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V031A	2
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V031B	2
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V032A	2
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V032B	2
RNS Pump A Suction Pressure Instrument Isolation	RNS-PL-V033A	2
RNS Pump B Suction Pressure Instrument Isolation	RNS-PL-V033B	2
RNS Pump A Discharge Pressure Instrument Isolation	RNS-PL-V034A	2
RNS Pump B Discharge Pressure Instrument Isolation	RNS-PL-V034B	2
RNS Pump A Suction Piping Drain Isolation	RNS-PL-V036A	2
RNS Pump B Suction Piping Drain Isolation	RNS-PL-V036B	2
RNS Pump Seal Cooler A Vent Isolation	RNS-PL-V048A	2
RNS Pump Seal Cooler B Vent Isolation	RNS-PL-V048B	2
RNS Pump Seal Cooler A Drain Isolation	RNS-PL-V049A	2
RNS Pump Seal Cooler B Drain Isolation	RNS-PL-V049B	2
RNS Pump A Casing Drain Isolation	RNS-PL-V050	2
RNS Pump B Casing Drain Isolation	RNS-PL-V051	2
RNS Suction from SFP Isolation	RNS-PL-V052	2
RNS Discharge to SFP Isolation	RNS-PL-V053	2
RNS Suction from Cask Loading Pit Isolation Valve	RNS-PL-V055	2
RNS Pump Suction to Cask Loading Pit Isolation	RNS-PL-V056	2
RNS Train A Miniflow Isolation Valve	RNS-PL-V057A	2
RNS Train B Miniflow Isolation Valve	RNS-PL-V057B	2
RNS Pump Suction Containment Isolation Test Connection	RNS-PL-V059	2
RNS Discharge Drain Valve	RNS-PL-V065	2
RNS Discharge to DVI Line A Drain	RNS-PL-V066A	2
RNS Discharge to DVI Line B Drain	RNS-PL-V066B	2
RNS Discharge to DVI Line A Drain	RNS-PL-V067A	2
RNS Discharge to DVI Line B Drain	RNS-PL-V067B	2
RNS Discharge to IRWST Drain	RNS-PL-V068	2

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Description	AP1000 Tag Number	Comment
RNS Pump A Miniflow Vent	RNS-PL-V069A	2
RNS Pump B Miniflow Vent	RNS-PL-V069B	2
RNS HX A Channel Head Drain Isolation	RNS-PL-V071A	2
RNS HX B Channel Head Drain Isolation	RNS-PL-V071B	2
RNS HX A Channel Head Drain Isolation	RNS-PL-V072A	2
RNS HX B Channel Head Drain Isolation	RNS-PL-V072B	2
RNS HX A Channel Head Drain Isolation	RNS-PL-V073A	2
RNS HX B Channel Head Drain Isolation	RNS-PL-V073B	2
RNS HX A Channel Head Drain Isolation	RNS-PL-V074A	2
RNS HX B Channel Head Drain Isolation	RNS-PL-V074B	2
RNS HX A Channel Head Drain Isolation	RNS-PL-V075A	2
RNS HX B Channel Head Drain Isolation	RNS-PL-V075B	2
IRWST Suction Line Vent	RNS-PL-V080	2
RNS Cask Loading Pit Suction Line Vent	RNS-PL-V081	2
RNS Discharge Drain	RNS-PL-V082	2
LT019A Root Isolation Valve	SFS-PL-V024A	2
LT019B Root Isolation Valve	SFS-PL-V024B	2
LT019C Root Isolation Valve	SFS-PL-V024C	2
LT020 Root Isolation Valve	SFS-PL-V028	2
SFS Refueling Cavity Drain To SGS Compartment Isolation	SFS-PL-V031	2
SFS Refueling Cavity Suction Isolation	SFS-PL-V032	2
SFS Refueling Cavity Drain to Containment Sump Isolation	SFS-PL-V033	2
SFS Suction Line from IRWST Isolation	SFS-PL-V039	2
SFS Fuel Transfer Canal Suction Isolation	SFS-PL-V040	2
SFS Cask Loading Pit Suction Isolation	SFS-PL-V041	2
Cask Loading Pit Level Transmitter Root Isolation Valve	SFS-PL-V043	2
SFS Containment Penetration Test Connection	SFS-PL-V048	2
SFS Containment Penetration Test Connection Isolation	SFS-PL-V056	2
SFS Containment Isolation Valve V034 Test	SFS-PL-V058	2
SFS Containment Floodup Isolation Valve	SFS-PL-V075	2
LT001, LT011 Root Isolation Valve	SGS-PL-V001A	2
LT005, LT013 Root Isolation Valve	SGS-PL-V001B	2
LT001 Root Isolation Valve	SGS-PL-V002A	2
LT005 Root Isolation Valve	SGS-PL-V002B	2
LT002, LT012 Root Isolation Valve	SGS-PL-V003A	2
LT006, LT014 Root Isolation Valve	SGS-PL-V003B	2
LT002 Root Isolation Valve	SGS-PL-V004A	2
LT006 Root Isolation Valve	SGS-PL-V004B	2
LT003, LT015, LT044 Root Isolation Valve	SGS-PL-V005A	2
LT007, LT017, LT046 Root Isolation Valve	SGS-PL-V005B	2

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Description	AP1000 Tag Number	Comment
LT003 Root Isolation Valve	SGS-PL-V006A	2
LT007 Root Isolation Valve	SGS-PL-V006B	2
LT004, LT016, LT045 Root Isolation Valve	SGS-PL-V007A	2
LT008, LT018, LT047 Root Isolation Valve	SGS-PL-V007B	2
LT004 Root Isolation Valve	SGS-PL-V008A	2
LT008 Root Isolation Valve	SGS-PL-V008B	2
LT011 Root Isolation Valve	SGS-PL-V010A	2
LT013 Root Isolation Valve	SGS-PL-V010B	2
LT011 Root Isolation Valve	SGS-PL-V011A	2
LT013 Root Isolation Valve	SGS-PL-V011B	2
LT012 Root Isolation Valve	SGS-PL-V012A	2
LT014 Root Isolation Valve	SGS-PL-V012B	2
LT012 Root Isolation Valve	SGS-PL-V013A	2
LT014 Root Isolation Valve	SGS-PL-V013B	2
FT021 Root Isolation Valve	SGS-PL-V015A	2
FT023 Root Isolation Valve	SGS-PL-V015B	2
FT020 Root Isolation Valve	SGS-PL-V016A	2
FT022 Root Isolation Valve	SGS-PL-V016B	2
FT021 Root Isolation Valve	SGS-PL-V017A	2
FT023 Root Isolation Valve	SGS-PL-V017B	2
FT020 Root Isolation Valve	SGS-PL-V018A	2
FT022 Root Isolation Valve	SGS-PL-V018B	2
Main Steam Line Vent Isolation	SGS-PL-V019A	2
Main Steam Line Vent Isolation	SGS-PL-V019B	2
FT024 Root Isolation Valve	SGS-PL-V020A	2
FT025 Root Isolation Valve	SGS-PL-V020B	2
FT024 Root Isolation Valve	SGS-PL-V021A	2
FT025 Root Isolation Valve	SGS-PL-V021B	2
PT030 Root Isolation Valve	SGS-PL-V022A	2
PT034 Root Isolation Valve	SGS-PL-V022B	2
PT031 Root Isolation Valve	SGS-PL-V023A	2
PT035 Root Isolation Valve	SGS-PL-V023B	2
PT032 Root Isolation Valve	SGS-PL-V024A	2
PT036 Root Isolation Valve	SGS-PL-V024B	2
PT033 Root Isolation Valve	SGS-PL-V025A	2
PT037 Root Isolation Valve	SGS-PL-V025B	2
Steam Line 1 Nitrogen Supply Isolation	SGS-PL-V038A	2
Steam Line 2 Nitrogen Supply Isolation	SGS-PL-V038B	2
MSIV Bypass Control Isolation	SGS-PL-V042A	2
MSIV Bypass Control Isolation	SGS-PL-V042B	2
MSIV Bypass Control Isolation	SGS-PL-V043A	2
MSIV Bypass Control Isolation	SGS-PL-V043B	2

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Description	AP1000 Tag Number	Comment
SG1 Condensate Pipe Drain Valve	SGS-PL-V045A	2
SG2 Condensate Pipe Drain Valve	SGS-PL-V045B	2
LT015 Root Isolation Valve	SGS-PL-V046A	2
LT017 Root Isolation Valve	SGS-PL-V046B	2
LT015, LT044 Root Isolation Valve	SGS-PL-V047A	2
LT017, LT046 Root Isolation Valve	SGS-PL-V047B	2
LT016 Root Isolation Valve	SGS-PL-V048A	2
LT018 Root Isolation Valve	SGS-PL-V048B	2
LT016, LT045 Root Isolation Valve	SGS-PL-V049A	2
LT018, LT047 Root Isolation Valve	SGS-PL-V049B	2
LT044 Root Isolation Valve	SGS-PL-V050A	2
LT046 Root Isolation Valve	SGS-PL-V050B	2
LT044 Root Isolation Valve	SGS-PL-V051A	2
LT046 Root Isolation Valve	SGS-PL-V051B	2
LT045 Root Isolation Valve	SGS-PL-V052A	2
LT047 Root Isolation Valve	SGS-PL-V052B	2
LT045 Root Isolation Valve	SGS-PL-V053A	2
LT047 Root Isolation Valve	SGS-PL-V053B	2
PT062 Root Isolation Valve	SGS-PL-V056A	2
PT063 Root Isolation Valve	SGS-PL-V056B	2
Main Feedwater Check	SGS-PL-V058A	2
Main Feedwater Check	SGS-PL-V058B	2
FT055A Root Isolation Valve	SGS-PL-V062A	2
FT056A Root Isolation Valve	SGS-PL-V062B	2
FT055A Root Isolation Valve	SGS-PL-V063A	2
FT056A Root Isolation Valve	SGS-PL-V063B	2
FT055B Root Isolation Valve	SGS-PL-V064A	2
FT056B Root Isolation Valve	SGS-PL-V064B	2
FT055B Root Isolation Valve	SGS-PL-V065A	2
FT056B Root Isolation Valve	SGS-PL-V065B	2
FT055 Root Isolation Valve	SGS-PL-V066A	2
FT056 Root Isolation Valve	SGS-PL-V066B	2
FT055 Root Isolation Valve	SGS-PL-V068A	2
FT056 Root Isolation Valve	SGS-PL-V068B	2
SG 1 Blowdown Vent Line Isolation	SGS-PL-V076A	2
SG 2 Blowdown Vent Line Isolation	SGS-PL-V076B	2
SG1 Nitrogen Sparging Isolation	SGS-PL-V084A	2
SG2 Nitrogen Sparging Isolation	SGS-PL-V084B	2
Orifice Isolation Valve	SGS-PL-V093A	2
Orifice Isolation Valve	SGS-PL-V093B	2
Orifice Cleanout Line Isolation Valve	SGS-PL-V094A	2
Orifice Cleanout Line Isolation Valve	SGS-PL-V094B	2
Orifice Isolation Valve	SGS-PL-V095A	2
Orifice Isolation Valve	SGS-PL-V095B	2

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Description	AP1000 Tag Number	Comment
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V096A	2
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V096B	2
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V097A	2
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V097B	2
Startup Feedwater Drain Isolation Valve	SGS-PL-V100A	2
Startup Feedwater Drain Isolation Valve	SGS-PL-V100B	2
Main Feedwater Drain Isolation Valve	SGS-PL-V101A	2
Main Feedwater Drain Isolation Valve	SGS-PL-V101B	2
Startup Feedwater Vent Isolation Valve	SGS-PL-V102A	2
Startup Feedwater Vent Isolation Valve	SGS-PL-V102B	2
Main Feedwater Vent Isolation Valve	SGS-PL-V103A	2
Main Feedwater Vent Isolation Valve	SGS-PL-V103B	2
Main Feedwater Drain Isolation Valve	SGS-PL-V104A	2
Main Feedwater Drain Isolation Valve	SGS-PL-V104B	2
Startup Feedwater Check Valve	SGS-PL-V256A	2
Startup Feedwater Check Valve	SGS-PL-V256B	2
SG 1 NR LT001 Upper Isolation Valve	SGS-PL-V300	2
SG 1 NR LT001 Lower Isolation Valve	SGS-PL-V301	2
SG 1 NR LT002 Upper Isolation Valve	SGS-PL-V302	2
SG 1 NR LT002 Lower Isolation Valve	SGS-PL-V303	2
SG 1 NR LT003 Upper Isolation Valve	SGS-PL-V304	2
SG 1 NR LT003 Lower Isolation Valve	SGS-PL-V305	2
SG 1 NR LT004 Upper Isolation Valve	SGS-PL-V306	2
SG 1 NR LT004 Lower Isolation Valve	SGS-PL-V307	2
SG 1 WR LT011 Lower Isolation Valve	SGS-PL-V308	2
SG 1 WR LT012 Lower Isolation Valve	SGS-PL-V309	2
SG 1 WR LT015 Lower Isolation Valve	SGS-PL-V310	2
SG 1 WR LT016 Lower Isolation Valve	SGS-PL-V311	2
SG 1 Main Steam FT020 Isolation Valve	SGS-PL-V312	2
SG 1 Main Steam FT021 Isolation Valve	SGS-PL-V313	2
SG 1 Main Steam FT024 Isolation Valve	SGS-PL-V314	2
SG 2 NR LT005 Upper Isolation Valve	SGS-PL-V320	2
SG 2 NR LT005 Lower Isolation Valve	SGS-PL-V321	2
SG 2 NR LT006 Upper Isolation Valve	SGS-PL-V322	2
SG 2 NR LT006 Lower Isolation Valve	SGS-PL-V323	2
SG 2 NR LT007 Upper Isolation Valve	SGS-PL-V324	2
SG 2 NR LT007 Lower Isolation Valve	SGS-PL-V325	2
SG 2 NR LT008 Upper Isolation Valve	SGS-PL-V326	2
SG 2 NR LT008 Lower Isolation Valve	SGS-PL-V327	2
SG 2 WR LT013 Lower Isolation Valve	SGS-PL-V328	2
SG 2 WR LT014 Lower Isolation Valve	SGS-PL-V329	2

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Description	AP1000 Tag Number	Comment
SG 2 WR LT017 Lower Isolation Valve	SGS-PL-V330	2
SG 2 WR LT018 Lower Isolation Valve	SGS-PL-V331	2
SG 2 Main Steam FT022 Isolation Valve	SGS-PL-V332	2
SG 2 Main Steam FT023 Isolation Valve	SGS-PL-V333	2
SG 2 Main Steam FT025 Isolation Valve	SGS-PL-V334	2
Air Delivery Line Pressure Instrument Isolation Valve A	VES-PL-V006A	2
Air Delivery Line Pressure Instrument Isolation Valve B	VES-PL-V006B	2
Air Delivery Line Maintenance Isolation Valve A	VES-PL-V010A	2
Air Delivery Line Maintenance Isolation Valve B	VES-PL-V010B	2
Air Delivery Line Maintenance Isolation Valve A	VES-PL-V011A	2
Air Delivery Line Maintenance Isolation Valve B	VES-PL-V011B	2
Temporary Instrument Isolation Valve A	VES-PL-V016	2
Temporary Instrument Isolation Valve B	VES-PL-V020	2
Air Bank 1 Isolation Valve A	VES-PL-V024A	2
Air Bank 2 Isolation Valve B	VES-PL-V024B	2
Air Bank 3 Isolation Valve C	VES-PL-V024C	2
Air Bank 4 Isolation Valve D	VES-PL-V024D	2
Air Bank 1 Isolation Valve A	VES-PL-V025A	2
Air Bank 2 Isolation Valve B	VES-PL-V025B	2
Air Bank 3 Isolation Valve C	VES-PL-V025C	2
Air Bank 4 Isolation Valve D	VES-PL-V025D	2
Air Bank 1 Fill/Vent Isolation Valve A	VES-PL-V026A	2
Air Bank 2 Fill/Vent Isolation Valve B	VES-PL-V026B	2
Air Bank 3 Fill/Vent Isolation Valve C	VES-PL-V026C	2
Air Bank 4 Fill/Vent Isolation Valve D	VES-PL-V026D	2
DP Instrument Line Isolation Valve A	VES-PL-V043A	2
DP Instrument Line Isolation Valve B	VES-PL-V043B	2
Containment Isolation Test Connection	VFS-PL-V006	2
RCS Ejector Discharge Isolation	VFS-PL-V007	2
Containment Isolation Test Connection	VFS-PL-V008	2
Containment Isolation Test Connection	VFS-PL-V012	2
Containment Isolation Test Connection	VFS-PL-V015	2
Containment Isolation Test Connection	VFS-PL-V101	2
Containment Atmosphere to Filtration Units Isolation	VFS-PL-V202	2
Filtration Units to Containment Atmosphere Manual Isolation	VFS-PL-V587	2
Main Equipment Hatch Test Connection	VUS-PL-V013	2
Maintenance Equipment Hatch Test Connection	VUS-PL-V014	2
Main Equipment Hatch Test Connection	VUS-PL-V015	2

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Table 3I.6-3 (Sheet 22 of 23)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Maintenance Equipment Hatch Test Connection	VUS-PL-V016	2
Personnel Hatch Test Connection	VUS-PL-V017	2
Personnel Hatch Test Connection	VUS-PL-V018	2
Personnel Hatch Test Connection	VUS-PL-V019	2
Personnel Hatch Test Connection	VUS-PL-V020	2
Personnel Hatch Test Connection	VUS-PL-V021	2
Personnel Hatch Test Connection	VUS-PL-V022	2
Fuel Transfer Tube Test Connection	VUS-PL-V023	2
Spare Penetration Test Connection	VUS-PL-V140	2
Spare Penetration Test Connection	VUS-PL-V141	2
Spare Penetration Test Connection	VUS-PL-V142	2
VWS Supply Containment Penetration IRC Test Connection/Vent	VWS-PL-V424	2
VWS Return Containment Penetration ORC Test Connection/Vent	VWS-PL-V425	2
WWS MCR Isolation Valve	WWS-PL-V506	2
VBS-PDT-032A Isolation Valve	VBS-PL-V184A	2
VBS-PDT-032B Isolation Valve	VBS-PL-V184B	2
VBS-PDT-032C Isolation Valve	VBS-PL-V184C	2
Heat Exchangers		
Normal Residual Heat Removal Heat Exchanger A	RNS-ME-01A	3
Normal Residual Heat Removal Heat Exchanger B	RNS-ME-01B	3
Tanks		
Spent Fuel Pool	FHS-MT-01	3
Fuel Transfer Canal	FHS-MT-02	3
Refueling Cavity	FHS-MT-03	3
Spent Fuel Cask Loading Pit	FHS-MT-05	3
Spent Fuel Cask Washdown Pit	FHS-MT-06	3
Passive Containment Cooling Water Storage Tank	PCS-MT-01	3
Water Distribution Bucket	PCS-MT-03	3
Water Collection Troughs	PCS-MT-04	3
Passive RHR Heat Exchanger	PXS-ME-01	3
Accumulator Tank A	PXS-MT-01A	3
Accumulator Tank B	PXS-MT-01B	3
Core Makeup Tank A	PXS-MT-02A	3
Core Makeup Tank B	PXS-MT-02B	3
In-Containment Refueling Water Storage Tank	PXS-MT-03	3
Emergency Air Storage Tank 01	VES-MT-01	3
Emergency Air Storage Tank 02	VES-MT-02	3
Emergency Air Storage Tank 03	VES-MT-03	3
Emergency Air Storage Tank 04	VES-MT-04	3
Emergency Air Storage Tank 05	VES-MT-05	3
Emergency Air Storage Tank 06	VES-MT-06	3
Emergency Air Storage Tank 07	VES-MT-07	3
Emergency Air Storage Tank 08	VES-MT-08	3

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Table 3I.6-3 (Sheet 23 of 23)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Emergency Air Storage Tank 09	VES-MT-09	3
Emergency Air Storage Tank 10	VES-MT-10	3
Emergency Air Storage Tank 11	VES-MT-11	3
Emergency Air Storage Tank 12	VES-MT-12	3
Emergency Air Storage Tank 13	VES-MT-13	3
Emergency Air Storage Tank 14	VES-MT-14	3
Emergency Air Storage Tank 15	VES-MT-15	3
Emergency Air Storage Tank 16	VES-MT-16	3
Emergency Air Storage Tank 17	VES-MT-17	3
Emergency Air Storage Tank 18	VES-MT-18	3
Emergency Air Storage Tank 19	VES-MT-19	3
Emergency Air Storage Tank 20	VES-MT-20	3
Emergency Air Storage Tank 21	VES-MT-21	3
Emergency Air Storage Tank 22	VES-MT-22	3
Emergency Air Storage Tank 23	VES-MT-23	3
Emergency Air Storage Tank 24	VES-MT-24	3
Emergency Air Storage Tank 25	VES-MT-25	3
Emergency Air Storage Tank 26	VES-MT-26	3
Emergency Air Storage Tank 27	VES-MT-27	3
Emergency Air Storage Tank 28	VES-MT-28	3
Emergency Air Storage Tank 29	VES-MT-29	3
Emergency Air Storage Tank 30	VES-MT-30	3
Emergency Air Storage Tank 31	VES-MT-31	3
Emergency Air Storage Tank 32	VES-MT-32	3

Notes:

1. Rugged AP1000 safety-related equipment with no moving parts required in demonstrating functional operability during a seismic event is considered to be not sensitive to HRHF seismic loadings. Seismic qualification is based on the seismic loads associated with the mounting location of the safety-related equipment as a minimum. AP1000 CSDRS seismic loads at the mounting location of the safety-related equipment produces comparable or higher equipment stresses and deflections than the HRHF seismic loadings based on the work reported in APP-GW-GLR-115, “Effect of High Frequency Seismic Content on SSCs.” For rugged safety-related line-mounted equipment being qualified by test, seismic testing will be performed in compliance with IEEE Standard 382-1996 with a required input motion (RIM) curve extended to 64 Hz typically to a peak acceleration of 6g.
2. AP1000 safety-related valves are seismic qualified in accordance with ASME code for structural integrity to a maximum acceleration of 6g in all three principal orthogonal axes. AP1000 CSDRS seismic loads at the mounting location of the safety-related equipment produce comparable or higher equipment stresses and deflections than the HRHF seismic loadings based on the work reported in APP-GW-GLR-115, “Effect of High Frequency Seismic Content on SSCs.” For rugged safety-related line-mounted equipment being qualified by test, seismic testing will be performed in compliance with IEEE Standard 382-1996 with a required input motion (RIM) curve extended to 64 Hz typically to a peak acceleration of 6g.
3. Seismic qualification is based on structural integrity alone to the seismic loadings associated with the mounting location of the safety-related equipment as a minimum. AP1000 CSDRS seismic loads at the mounting location of the safety-related equipment produce comparable or higher equipment stresses and deflections than the HRHF seismic loadings based on the work reported in APP-GW-GLR-115, “Effect of High Frequency Seismic Content on SSCs.”
4. IIS-JE-TE009, IIS-JE-TE013, IIS-JE-TE030, and IIS-JE-TE034 are not safety-related and therefore excluded from this list.

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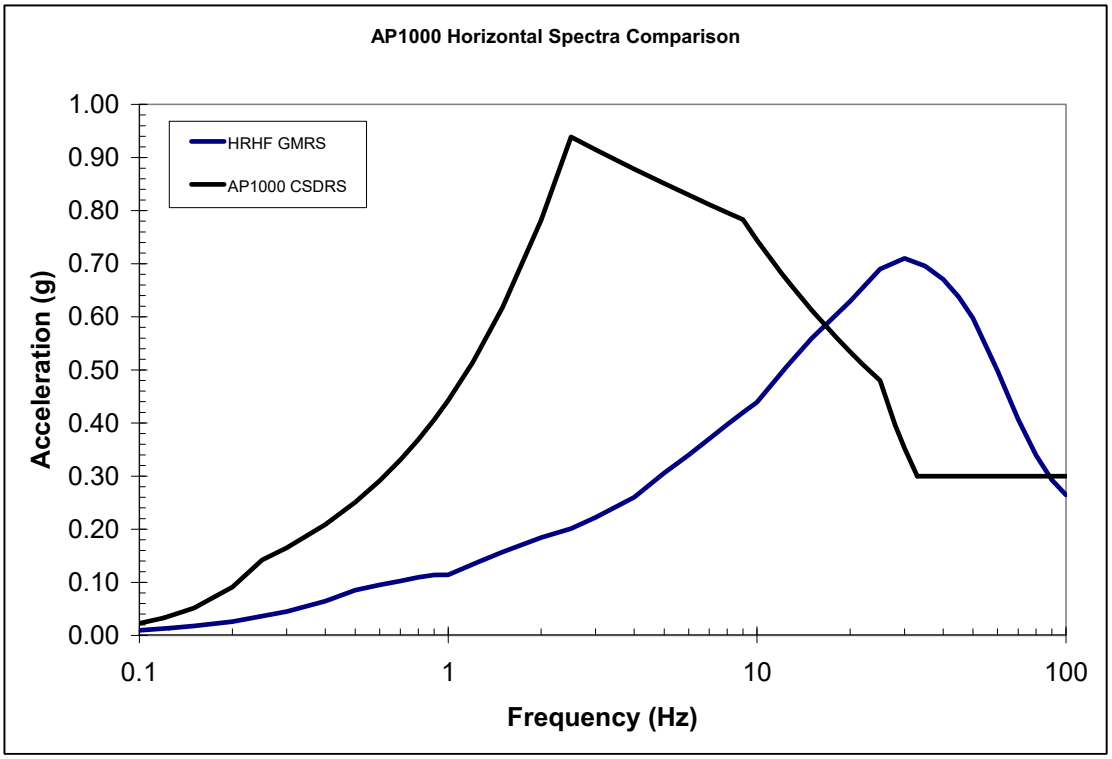

Figure 3I.1-1
Comparison of Horizontal AP1000 CSDRS and HRHF Envelope Response Spectra

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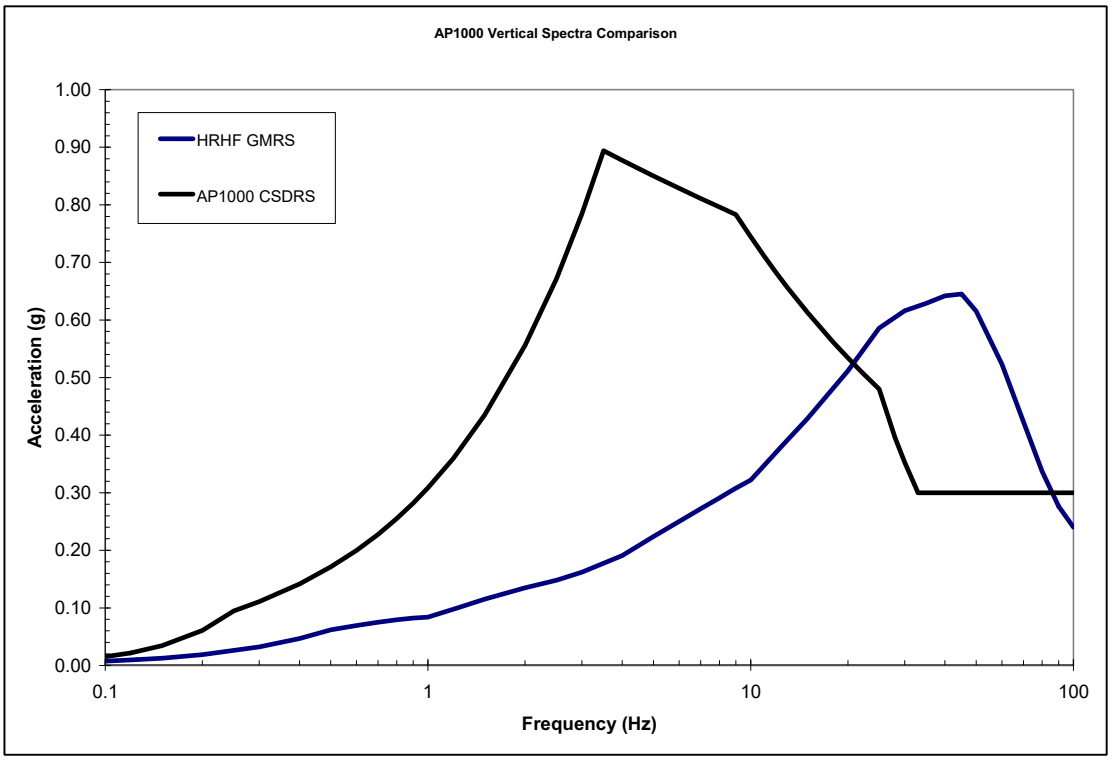

Figure 3I.1-2
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Chapter 4 Reactor

4.1 Summary Description

This chapter describes the mechanical components of the reactor and reactor core, including the fuel rods and fuel assemblies, the nuclear design, and the thermal-hydraulic design.

The reactor contains a matrix of fuel rods assembled into mechanically identical fuel assemblies along with control and structural elements. The assemblies, containing various fuel enrichments, are configured into the core arrangement located and supported by the reactor internals. The reactor internals also direct the flow of the coolant past the fuel rods. The coolant and moderator are light water at a normal operating pressure of 2250 psia. The fuel, internals, and coolant are contained within a heavy walled reactor pressure vessel. An AP1000 fuel assembly consists of 264 fuel rods in a 17x17 square array. The center position in the fuel assembly has a guide thimble that is reserved for in-core instrumentation. The remaining 24 positions in the fuel assembly have guide thimbles. The guide thimbles are joined to the top and bottom nozzles of the fuel assembly and provide the supporting structure for the fuel grids.

The fuel grids consist of an egg-crate arrangement of interlocked straps that maintain lateral spacing between the rods. The grid straps have spring fingers and dimples that grip and support the fuel rods. The intermediate mixing vane grids also have coolant mixing vanes. In addition, there are four intermediate flow mixing (IFM) grids. The IFM grid straps contain support dimples and coolant mixing vanes only. The top and bottom grids and protective grid do not contain mixing vanes.

The AP1000 fuel assemblies are similar to the 17x17 Robust and 17x17 XL Robust fuel assemblies. The 17x17 Robust fuel assemblies have an active fuel length of 12 feet and three intermediate flow mixing grids in the top mixing vane grid spans. The 17x17 XL Robust fuel assemblies have an active fuel length of 14 feet (same as AP1000 fuel assemblies) and no intermediate flow mixing grids. The AP1000 fuel assemblies have four intermediate flow mixing grids in the top mixing vane grid spans.

There is substantial operating experience with the 17x17 Robust and 17x17 XL Robust fuel assemblies. The 17x17 Robust fuel assemblies are described in References 1, 2 and 3. The 17x17 XL Robust fuel assemblies are described in References 4 and 5.

The XL Robust fuel assembly evolved from the previous VANTAGE+, VANTAGE 5 and VANTAGE 5 HYBRID designs. The XL Robust fuel assembly is based on the substantial design and operating experience with those designs. The design is described and evaluated in References 2, 3, 6 through 10.

A number of proven design features have been incorporated in the AP1000 fuel assembly design. The AP1000 fuel assembly design includes: low pressure drop intermediate grids, four intermediate flow mixing (IFM) grids, a reconstitutable Westinghouse integral nozzle (WIN), and extended burnup capability. The bottom nozzle is a debris filter bottom nozzle (DFBN) that minimizes the potential for fuel damage due to debris in the reactor coolant. The AP1000 fuel assembly design also includes a protective grid for enhanced debris resistance.

The fuel rods consist of enriched uranium, in the form of cylindrical pellets of uranium dioxide, contained in ZIRLO®[1] (Reference 8) tubing. The tubing is plugged and seal welded at the ends to encapsulate the fuel. An axial blanket comprised of fuel pellets with reduced enrichment may be placed at each end of the enriched fuel pellet stack to reduce the neutron leakage and to improve fuel utilization.

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Other types of fuel rods may be used to varying degrees within some fuel assemblies. One type uses an integral fuel burnable absorber (IFBA) containing a thin boride coating on the surface of the fuel pellets. The boride-coated fuel pellets provide a burnable absorber integral to the fuel.

Fuel rods are pressurized internally with helium during fabrication to reduce clad creepdown during operation and thereby prevent clad flattening. The fuel rods in the AP1000 fuel assemblies contain additional gas space below the fuel pellets, compared to the 17x17 Robust, 17x17 XL Robust and other previous fuel assembly designs to allow for increased fission gas production due to high fuel burnups.

Depending on the position of the assembly in the core, the guide thimbles are used for rod cluster control assemblies (RCCAs), gray rod cluster assemblies (GRCAs), neutron source assemblies, wet annular burnable absorber (WABA) assemblies, or thimble plugs.

For the initial core design, WABAs and IFBAs are used. Discrete burnable absorber designs, integral fuel burnable absorber designs or combinations may be used in subsequent reloads.

The bottom nozzle is a box-like structure that serves as the lower structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The size of flow passages through the bottom nozzle limits the size of debris that can enter the fuel assembly. The top nozzle assembly serves as the upper structural element of the fuel assembly and provides a partial protective housing for the rod cluster control assembly or other components.

The rod cluster control assemblies consist of 24 absorber rodlets fastened at the top end to a common hub, or spider assembly. Each absorber rod consists of an alloy of silver-indium-cadmium, which is clad in stainless steel. The rod cluster control assemblies are used to control relatively rapid changes in reactivity and to control the axial power distribution.

The gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub, or spider assembly (Reference 11). Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that the absorber rods consists of tungsten contained within a nickel-chromium-iron Alloy 718 sleeve. Stainless steel spacers are provided at the bottom of the rodlet between the bottom of the sleeve and the bottom end plug of the stainless steel cladding.

The gray rod cluster assemblies are used in base load operation and load follow maneuvering. The assemblies provide a mechanical shim reactivity mechanism to minimize the need for changes to the concentration of soluble boron.

The reactor core is cooled and moderated by light water at a pressure of 2250 psia. Soluble boron in the moderator/coolant serves as a neutron absorber. The concentration of boron is varied to control reactivity changes that occur relatively slowly, including the effects of fuel burnup. Burnable absorbers are also employed in the initial cycle to limit the amount of soluble boron required and, thereby maintain the desired negative reactivity coefficients.

The nuclear design analyses establish the core locations for control rods and burnable absorbers. The analyses define design parameters, such as fuel enrichments and boron concentration in the coolant.

The nuclear design establishes that the reactor core and the reactor control system satisfy design criteria, even if the rod cluster control assembly of the highest reactivity worth is in the fully withdrawn position.

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The core has inherent stability against diametral and azimuthal power oscillations. Axial power oscillations, which may be induced by load changes, and resultant transient xenon may be suppressed by the use of the rod cluster control assemblies.

The control rod drive mechanisms used to withdraw and insert the rod cluster control assemblies and the gray rod cluster assemblies are described in Subsection 3.9.4.

The thermal-hydraulic design analyses establish that adequate heat transfer is provided between the fuel clad and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution, and mixing. The mixing vanes incorporated in the fuel assembly spacer grid design and the fuel assembly intermediate flow mixers induce additional flow mixing between the various flow channels within a fuel assembly, as well as between adjacent assemblies.

The reactor internals direct the flow of coolant to and from the fuel assemblies and are described in Subsection 3.9.5.

The performance of the core is monitored by fixed neutron detectors outside the core, fixed neutron detectors within the core, and thermocouples at the outlet of selected fuel assemblies. The ex-core nuclear instrumentation provides input to automatic control functions.

Table 4.1-1 presents a summary of the principal nuclear, thermal-hydraulic, and mechanical design parameters of the AP1000 fuel. A comparison is provided to the fuel design used in AP1000, AP600 and in a licensed Westinghouse-designed plant using XL Robust fuel. For the comparison with a plant containing XL Robust fuel, a 193 fuel assembly plant is used, since no domestic, Westinghouse-designed 157 fuel assembly plants use 17x17 XL Robust fuel.

Table 4.1-2 tabulates the analytical techniques employed in the core design. The design basis must be met using these analytical techniques. Enhancements may be made to these techniques provided that the changes are bounded by NRC-approved methods, models, or criteria. In addition, application of the process described in WCAP-12488-A, (Reference 9) allows the Combined License holder to make fuel mechanical changes. Table 4.1-3 tabulates the mechanical loading conditions considered for the core internals and components. Specific or limiting loads considered for design purposes of the various components are listed as follows: fuel assemblies in Subsection 4.2.1.5; control rods (RCCAs and GRCAs), burnable absorber rods, and neutron source rods, in Subsection 4.2.1.6. The dynamic analyses, input forcing functions, and response loadings for the control rod drive system and reactor vessel internals are presented in Subsections 3.9.4 and 3.9.5.

4.1.1 Principal Design Requirements

The fuel and control rod mechanism are designed so the performance and safety criteria described in Chapter 4 and Chapter 15 are met. [*The mechanical design and physical arrangement of the reactor components, together with the corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) are designed to achieve these criteria, referred to as Principal Design Requirements:*

- *Fuel damage, defined as penetration of the fuel cladding, is predicted not to occur during normal operation and anticipated operational transients.*
- *Materials used in the fuel assembly and in-core control components are selected to be compatible in a pressurized water reactor environment.*
- *For normal operation and anticipated transient conditions, the minimum DNBR calculated using the WRB-2M correlation is greater than or equal to 1.14.*

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- *Fuel melting will not occur at the overpower limit for Condition I or II events.*
- *The maximum fuel rod cladding temperature following a loss-of-coolant accident is calculated to be less than 2200°F.*
- *For normal operation and anticipated transient conditions, the calculated core average linear power, including densification effects, is less than or equal to 5.718 kw/ft for the initial fuel cycle.*
- *For normal operation and anticipated transient conditions, the calculated total heat flux hot channel factor, F_Q, is less than or equal to 2.60 for the initial fuel cycle.*
- *Calculated rod worths provide sufficient reactivity to account for the power defect from full power to zero power and provide the required shutdown margin, with allowance for the worst stuck rod.*
- *Calculations of the accidental withdrawal of two control banks using the maximum reactivity change rate predict that the peak linear heat rate and DNBR limits are met.*
- *The maximum rod cluster control assembly and gray rod speed (or travel rate) is 45 inches per minute.*
- *The control rod drive mechanisms are hydrotested after manufacture at a minimum of 125 percent of system design pressure.*
- *For the initial fuel cycle, the fuel rod temperature coefficient is calculated to be negative for power operating conditions.*
- *For the initial fuel cycle, the moderator temperature coefficient is calculated to be negative for power operating conditions.*]*

4.1.2 Combined License Information

This section contained no requirement for additional information.

4.1.3 References

1. Letter from N. J. Liparulo (Westinghouse) to J. E. Lyons (NRC), “Transmittal of Response to NRC Request for Information on Wolf Creek Fuel Design Modifications,” NSD-NRC-97-5189, June 30, 1997.

2. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), “Transmittal of Presentation Material for NRC/Westinghouse Fuel Design Change Meeting on April 15, 1996,” NSD-NRC-96-4964, April 22, 1996.

3. Letter from Westinghouse to NRC, “Fuel Criteria Evaluation Process Notification for the 17x17 Robust Fuel Assembly with IFM Grid Design,” NSD-NRC-98-5796, October 13, 1998.

4. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), “Notification of FCEP Application for WRB-1 and WRB-2 Applicability to the 17x17 Modified LPD Grid Design for Robust Fuel Assembly Application,” NSD-NRC-98-5618, March 25, 1998.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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5. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), “Fuel Criteria Evaluation Process Notification for the Revised Guide Thimble Dashpot Design for the 17x17 XL Robust Fuel Assembly Design,” NSD-NRC-98-5722, June 23, 1998.

6. Davidson, S. L., and Kramer, W. R., (Ed.), “Reference Core Report Vantage 5 Fuel Assembly,” WCAP-10444-P-A (Proprietary), September 1985 and WCAP-10445-A (Non-Proprietary), December 1983.

7. Davidson, S. L., (Ed.), “VANTAGE 5H Fuel Assembly,” Addendum 2-A, WCAP-10444-P-A (Proprietary) and WCAP-10445-NP-A (Non-Proprietary), February 1989.

8. Davidson, S. L., and Nuhfer, D. L., (Ed.), “VANTAGE+ Fuel Assembly Reference Core Report,” WCAP-12610-P-A (Proprietary) and WCAP-14342-A (Non-Proprietary), April 1995.

9. [*Davidson, S. L. (Ed.), “Fuel Criteria Evaluation Process,” WCAP-12488-A (Proprietary) and WCAP-14204-A (Non-Proprietary), October 1994.*]*

10. Letter, Peralta, J. D. (NRC) to Maurer, B. F. (Westinghouse), “Approval for Increase in Licensing Burnup Limit to 62,000 MWD/MTU (TAC No. MD1486),” May 25, 2006.

11. Conner, M. E., et al., “Enhanced GRCA Rodlet Design,” WCAP-16943-P-A (Proprietary) and WCAP-16943-NP-A (Non-Proprietary), September, 2012.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 4.1-1 (Sheet 1 of 3)
Reactor Design Comparison Table

Thermal and Hydraulic Design Parameters	AP1000[j]	AP600	Typical XL Plant
Reactor core heat output (MWt)	3400	1933	3800
Reactor core heat output (10^6 Btu/hr)	11,601	6596	12,969
Heat generated in fuel (%)	97.4	97.4	97.4
System pressure, nominal (psia)	2250	2250	2250
System pressure, minimum steady-state (psia)	2190	2200	2204
Minimum departure from nuclear boiling (DNBR) for design transients Typical flow channel Thimble (cold wall) flow channel	 >1.25[d] >1.25[d]	 >1.23 >1.22	 >1.26 >1.24
Departure from nucleate boiling (DNB) correlation[b]	WRB-2M[b]	WRB-2	WRB-1[a]
Coolant Flow[c]			
Total vessel thermal design flow rate (10^6 lbm/hr) Effective flow rate for heat transfer (10^6 lbm/hr) Effective flow area for heat transfer (ft^2) Average velocity along fuel rods (ft/s) Average mass velocity (10^6 lbm/hr-ft^2)	113.5 106.8 41.8 15.8 2.55	72.9 66.3 38.5 10.6 1.72	145.0 132.7 51.1 16.6 2.60
Coolant Temperature[c][e]			
Nominal inlet (°F) Average rise in vessel (°F) Average rise in core (°F) Average in core (°F) Average in vessel (°F)	535.0 77.2 81.4 578.1 573.6	532.8 69.6 75.8 572.6 567.6	561.2 63.6 68.7 597.8 593.0
Heat Transfer			
Active heat transfer surface area (ft^2) Avg. heat flux (BTU/hr-ft^2) Maximum heat flux for normal operation (BTU/hr-ft^2)[f] Average linear power (kW/ft)[g] Peak linear power for normal operation (kW/ft)[f][g] Peak linear power (kW/ft)[f][h] (Resulting from overpower transients/operator errors, assuming a maximum overpower of 118%)	56,700 199,300 518,200 5.72 14.9 ≤22.45	44,884 143,000 372,226 4.11 10.7 22.5	69,700 181,200 489,200 5.20 14.0 ≤22.45

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Table 4.1-1 (Sheet 2 of 3)
Reactor Design Comparison Table

Thermal and Hydraulic Design Parameters	AP1000[j]	AP600	Typical XL Plant
Heat flux hot channel factor (F_Q)	2.60	2.60	2.70
Peak fuel center line temperature (°F) (For prevention of center-line melt)	4700	4700	4700
Fuel assembly design	17x17 XL Robust Fuel	17x17	17x17 XL Robust Fuel/ No IFM
Number of fuel assemblies	157	145	193
Uranium dioxide rods per assembly	264	264	264
Rod pitch (in.)	0.496	0.496	0.496
Overall dimensions (in.)	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426
Fuel weight, as uranium dioxide (lb)	211,588	167,360	261,000
Clad weight (lb)	43,105	35,555	63,200
Number of grids per assembly Top and bottom - (Ni-Cr-Fe Alloy 718) Intermediate Intermediate flow mixing Protective Grid - (Ni-Cr-Fe Alloy 718)	 2[i] 8 ZIRLO 4 ZIRLO 1	 2[i] 7 Zircaloy-4 or 7 ZIRLO 4 Zircaloy-4 or 5 ZIRLO 1	 2 8 ZIRLO 0 1
Loading technique, first cycle	5 region nonuniform	3 region nonuniform	3 region nonuniform
Fuel Rods			
Number	41,448	38,280	50,952
Outside diameter (in.)	0.374	0.374	0.374
Diametral gap (non-IFBA) (in.)	0.0065	0.0065	0.0065
Clad thickness (in.)	0.0225	0.0225	0.0225
Clad material	ZIRLO	Zircaloy-4 or ZIRLO	Zircaloy-4/ ZIRLO
Fuel Pellets			
Material	UO_2 sintered	UO_2 sintered	UO_2 sintered
Density (% of theoretical)	95.5	95	95
Diameter (in.)	0.3225	0.3225	0.3225
Length (in.)	0.387	0.387	0.387

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Table 4.1-1 (Sheet 3 of 3)
Reactor Design Comparison Table

Rod Cluster Control Assemblies	AP1000[j]	AP600	Typical XL Plant
Neutron Absorber			
RCCA GRCA	24 Ag-In-Cd rodlets 24 Tungsten rodlets	24 Ag-In-Cd rodlets 20 304 SS rodlets 4 Ag-In-Cd rodlets	24 Hafnium or Ag-In-Cd
Cladding material	Type 304 or 304L SS, cold-worked	Type 304 SS, cold-worked	Type 304 SS, cold-worked
Clad thickness (in.) - RCCA - GRCA	0.0185 0.0255	0.0185 0.0185	0.0185
Number of clusters	53 RCCAs 16 GRCAs	45 RCCAs 16 GRCAs	57 RCCAs 0 GRCAs
Core Structure			
Core barrel, ID/OD (in.)	133.75/137.75	133.75/137.75	148.0/152.5
Thermal shield	Neutron Panel	None	Neutron Panel
Baffle thickness (in.)	Core Shroud	Radial reflector	0.875
Structure Characteristics			
Core diameter, equivalent (in.)	119.7	115.0	132.7
Core height, cold, active fuel (in.)	168.0	144.0	168.0
Fuel Enrichment First Cycle (Weight Percent)			
Region 1	0.74 / --	1.90	Typical
Region 2	1.58 / --	2.80	3.8 to 4.4
Region 3	3.20 / 1.58	3.70	(5.0 Max)
Region 4	3.776 / 3.20		
Region 5	4.376 / 3.20		

Notes:
a. WRB-2M will be used in future reloads
b. See Subsection 4.4.2.2.1 for the use of the ABB-NV, WLOP, WRB-2 and WRB-2M correlations
c. Flow rates and temperatures are based on 10 percent steam generator tube plugging for the AP600 and AP1000 designs
d. The Design Limit DNBR is 1.25
e. Coolant temperatures based on thermal design flow (for AP600 and AP1000)
f. Based on FQ of 2.60 for AP600 and AP1000
g. Based on densified active fuel length. The value for AP1000 is rounded to 5.72 kW/ft.
h. See Subsection 4.3.2.2.6
i. The top grid will be fabricated of nickel-chromium-iron Alloy 718.
j. For the AP1000 design, the assembly average enrichments are given for the mid-zone and axial blanket regions.

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Table 4.1-2
Analytical Techniques in Core Design

Analysis	Technique	Computer Code	Subsection Referenced
Mechanical design of core internals loads, deflections, and stress analysis	Static and dynamic modeling	BLOWDOWN code, FORCE, finite element structural analysis code, and others	3.7.2.1 3.9.2 3.9.3
Fuel rod design Fuel performance characteristics (such as, temperature, internal pressure, and clad stress)	Semi-empirical thermal model of fuel rod with considerations such as fuel density changes, heat transfer, and fission gas release.	Westinghouse fuel rod design model	4.2.1.1 4.2.3.2 4.2.3.3 4.3.3.1 4.4.2.11
Nuclear design Cross-sections and group constants	Microscopic data; macroscopic constants for homogenized core regions	Modified ENDF/B library with PHOENIX-P or PARAGON	4.3.3.2 4.3.3.3
X-Y and X-Y-Z power distributions, fuel depletion, critical boron concentrations, X-Y and X-Y-Z xenon distributions, reactivity coefficients	2-group diffusion theory, 2-group nodal theory	ANC (2-D or 3-D)	4.3.3.3
Axial power distributions, control rod worths, and axial xenon distribution	1-D, 2-group diffusion theory	APOLLO	4.3.3.3
Fuel rod power	Integral transport theory	LASER	4.3.3.1
Effective resonance temperature	Monte Carlo weighing function	REPAD	4.3.3.1
Criticality of reactor and fuel assemblies	3-D, Monte Carlo theory	MCNP4a	4.3.2.6.1
Vessel irradiation	Multigroup spatial dependent transport theory	DOT	4.3.2.8
Thermal-hydraulic design steady state	Subchannel analysis of local fluid conditions in rod bundles, including inertial and cross-flow resistance terms; solution progresses from core-wide to hot assembly to hot channel.	VIPRE-01	4.4.4.5.2
Transient departure from nucleate boiling	Subchannel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservation equations; solution progresses from core-wide to hot assembly to hot channel.	VIPRE-01	4.4.4.5.4

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Table 4.1-3
Design Loading Conditions for Reactor Core Components

- Fuel assembly weight and core component weights (burnable absorbers, sources, RCCA, GRCA, thimble plug)
- Fuel assembly holddown spring forces and core component spring forces
- Internals weight
- Control rod trip (equivalent static load)
- Differential pressure
- Spring preloads
- Coolant flow forces (static)
- Temperature gradients
- Thermal expansion
- Interference between components
- Vibration (mechanically or hydraulically induced)
- Operational transients listed in Table 3.9-1
- Pump overspeed
- Seismic loads (safe shutdown earthquake)
- Blowdown forces (due to pipe rupture)

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4.2 Fuel System Design

The plant conditions for design are divided into four categories.

- Condition I - normal operation and operational transients
- Condition II - events of moderate frequency
- Condition III - infrequent incidents
- Condition IV - limiting faults

Chapter 15 describes bases and plant operation and events involving each condition.

The reactor is designed so that its components meet the following performance and safety criteria:

- The mechanical design and physical arrangement of the reactor core components, together with corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) provide that:
  - Fuel damage, that is, breach of fuel rod clad pressure boundary, is not expected during Condition I and Condition II events. A very small amount of fuel damage may occur. This is within the capability of the plant cleanup system and is consistent with the plant design bases.
  - The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged. The fraction of fuel rods damaged must be limited to meet the dose guidelines identified in Chapter 15 although sufficient fuel damage might occur to preclude immediate resumption of operation.
  - The reactor can be brought to a safe state and the core kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.
- The fuel assemblies are designed to withstand non-operational loads induced during shipping, handling, and core loading without exceeding the criteria of Subsection 4.2.1.5.1.
- The fuel assemblies are designed to accept control rod insertions to provide the required reactivity control for power operations and reactivity shutdown conditions.
- The fuel assemblies have provisions for the insertion of in-core instrumentation.
- The reactor vessel and internals, in conjunction with the fuel assembly structure, directs reactor coolant through the core. Because of the resulting flow distribution and bypass flow, the heat transfer performance requirements are met for the modes of operation.

The following subsection provides the fuel system design bases and design limits. It is consistent with the criteria of the Standard Review Plan, Section 4.2.

Consistent with the growth in technology, Westinghouse modifies fuel system designs. These modifications utilize NRC approved methods. [*A set of design fuel criteria to be satisfied by new fuel designs was issued to the NRC in WCAP-12488-A and its addendums (Reference 1)*]* and also presented below in Subsection 4.2.1.

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4.2.1 Design Basis

The fuel rod and fuel assembly design bases are established to satisfy the general performance and safety criteria presented in Section 4.2 of the Standard Review Plan. [*The design bases and acceptance limits used by Westinghouse are also described in the Westinghouse Fuel Criteria Evaluation Process, WCAP-12488-A and its addendums (Reference 1).*]*

The fuel rods are designed to satisfy the fuel rod design criteria for rod burnup levels up to the design discharge burnup using the extended burnup design methods described in the Extended Burnup Evaluation report, WCAP-10125-P-A (Reference 2).

The AP1000 fuel rod design considers effects such as fuel density changes, fission gas release, clad creep, and other physical properties which vary with burnup. The integrity of the fuel rods is provided by designing to prevent excessive fuel temperatures as discussed in Subsection 4.2.1.2.1; excessive internal rod gas pressures due to fission gas releases as discussed in Subsections 4.2.1.3.1 and 4.2.1.3.2; and excessive cladding stresses, strains, and strain fatigue, as discussed in Subsections 4.2.1.1.2 and 4.2.1.1.3. The fuel rods are designed so that the conservative design bases of the following events envelope the lifetime operating conditions of the fuel. For each design basis, the performance of the limiting fuel rod, with appropriate consideration for uncertainties, does not exceed the limits specified by the design basis. The detailed fuel rod design also establishes such parameters as pellet size and density, clad/pellet diametral gap, gas plenum size, and helium pre-pressurization level.

Integrity of the fuel assembly structure is provided by setting limits on stresses and deformations due to various loads and by preventing the assembly structure from interfering with the functioning of other components. Three types of loads are considered:

- Non-operational loads, such as those due to shipping and handling
- Normal and abnormal loads, which are defined for Conditions I and II
- Abnormal loads, which are defined for Conditions III and IV

The design bases for the in-core control components are described in Subsection 4.2.1.6.

4.2.1.1 Cladding

4.2.1.1.1 Mechanical Properties

The ZIRLO cladding material combines neutron economy (low absorption cross-section); high corrosion resistance to coolant, fuel, and fission products; and high strength and ductility at operating temperatures. ZIRLO is an advanced zirconium based alloy that has the same or similar properties and advantages as Zircaloy-4 and was developed to support extended fuel burnup. WCAP-12610-P-A (Reference 5) provides a discussion of chemical and mechanical properties of the ZIRLO cladding material and a comparison to Zircaloy-4.

4.2.1.1.2 Stress-Strain Limits

Clad Stress

[*The volume average effective stress calculated with the Von Mises equation (considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling and uniform clad creep, and pressure differences) is less than the 0.2 percent offset yield stress with due consideration to temperature and irradiation effects for Condition I and II events, WCAP-12488-A (Reference 1).*]* While the clad has some capability for accommodating plastic

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strain, the yield stress has been accepted as a conservative design limit. The allowable stress limits due to Condition III and IV loadings, described in Subsection 4.2.1.5.3, are also applied to the fuel rod.

Clad Strain

[*The total plastic tensile creep strain due to uniform clad creep, and uniform cylindrical fuel pellet expansion associated with fuel swelling and thermal expansion is less than one percent from the unirradiated condition, WCAP-12488-A (Reference 1).*]* The acceptance limit for fuel rod clad strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than one percent from the pre-transient value. These limits are consistent with proven practice.

4.2.1.1.3 Fatigue and Vibration

Fatigue

[*The usage factor due to cycle fatigue is less than 1.0, WCAP-12488-A (Reference 1).*]* That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure. The fatigue curve is based on a safety factor of two on the stress amplitude or a safety factor of 20 on the number of cycles, whichever is more conservative.

Vibration

Potential fretting wear due to vibration is prevented, giving confidence that the stress-strain limits are not exceeded during design life. Fretting of the clad surface can occur due to flow-induced vibration between the fuel rods and fuel assembly grid springs. Vibration and fretting forces may vary during the fuel life due to clad diameter creep down combined with grid spring relaxation.

4.2.1.1.4 Chemical Properties

Chemical properties of the ZIRLO cladding are discussed in WCAP-12610 (Reference 5).

4.2.1.2 Fuel Material

4.2.1.2.1 Thermal-Physical Properties

The center temperature of the hottest pellet is below the melting temperature of the uranium dioxide. The melting temperature of unirradiated uranium dioxide, 5080°F, decreases by 58°F per 10,000 megawatt days per metric ton of uranium, as discussed in WCAP-9179 (Reference 4). Fuel melting will not occur at the overpower limit for Condition I or II events. This provides sufficient margin for uncertainties as described in Subsection 4.4.2.9.

The nominal design density of the fuel is approximately 95.5 percent of the theoretical density. Additional information on fuel properties is provided in WCAP-9179 (Reference 4).

4.2.1.2.2 Fuel Densification and Fission Product Swelling

The design bases and models used for fuel densification and swelling are provided in WCAP-10851-P-A (Reference 7), and WCAP-15063-P-A, Revision 1 (Reference 21).

4.2.1.2.3 Chemical Properties

WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5) provide the basis for justifying that no adverse chemical interactions occur between the fuel and its adjacent material.

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4.2.1.3 Fuel Rod Performance

4.2.1.3.1 Fuel Rod Models

The basic fuel rod models and the ability to predict fuel rod operating characteristics are given in WCAP-15063-P-A, Revision 1 (Reference 21) and Subsection 4.2.3.

4.2.1.3.2 Mechanical Design Limits

Cladding collapse is precluded during the fuel rod design lifetime. Current generation Westinghouse fuel is sufficiently stable with respect to fuel densification. Significant axial gaps in the pellet stack necessary for clad flattening do not occur and therefore, clad flattening will not occur. Clad flattening methodologies are described in WCAP-13589-A, (Reference 8) and WCAP-8377 (Reference 22).

The rod internal gas pressure remains below the value which causes the fuel/clad diametral gap to increase due to outward cladding creep during steady-state operation. Rod pressure is also limited such that extensive departure from nucleate boiling propagation does not occur as discussed in WCAP-8963-P-A (Reference 9).

4.2.1.4 Spacer Grids

4.2.1.4.1 Mechanical Limits and Materials Properties

The grid component strength criteria are based on experimental tests. The limit is established at the 95-percent confidence level on the true mean crush strength at operating temperature. This limit is sufficient to provide that, under worst-case combined seismic and pipe rupture event, the core will maintain a geometry amenable to cooling. As an integral part of the fuel assembly structure, the grids satisfy the applicable fuel assembly design bases and limits defined in Subsection 4.2.1.5.

The grid material and chemical properties are given in WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5).

4.2.1.4.2 Vibration and Fatigue

The grids provide sufficient fuel rod support to limit fuel rod vibration and maintain clad fretting wear within acceptable limits (defined in Subsection 4.2.1.1).

4.2.1.5 Fuel Assembly Structural Design

As discussed in Subsection 4.2.1, the structural integrity of the fuel assemblies is provided by setting design limits on stresses and deformations due to various non-operational, operational, and accident loads. These limits are applied to the design and evaluation of the top and bottom nozzles, guide thimbles, grids, and thimble joints. [*Design changes to the fuel assembly structure qualify for evaluation in WCAP-12488-A (Reference 1).*]*

The design bases for evaluating the structural integrity of the fuel assemblies are discussed in Subsections 4.2.1.5.1 through 4.2.1.5.3.

4.2.1.5.1 Non-Operational

The non-operational load is a loading of 4 g axial (longitudinal) and 6 g lateral (transverse) with dimensional stability.

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4.2.1.5.2 Normal Operation and Operational Transients (Condition I) and Events of Moderate Frequency (Condition II)

For the normal operation (Condition I) and upset (Condition II) conditions, the fuel assembly component structural design criteria are established for the two primary material categories, austenitic steels and zirconium alloys. The stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide. The maximum shear theory (Tresca criterion) for combined stresses is used to determine the stress intensities for the austenitic steel components. The stress intensity is defined as the largest numerical difference between the various principal stresses in a three-dimensional field. The design stress intensity value, S_m, for austenitic steels and zirconium alloys is given by the lowest of the following:

- One-third of the specified minimum tensile strength or two-thirds of the specified minimum yield strength at room temperature
- One-third of the tensile strength or 90 percent of the yield strength at temperature, but not to exceed two-thirds of the specified minimum yield strength at room temperature

The stress limits for the austenitic steel components are given below. Stress nomenclature follows the ASME Code, Section III.

Stress Intensity Limits

Categories	Limit
General primary membrane stress intensity	S_m
Local primary membrane stress intensity	1.5 S_m
Primary membrane plus bending stress intensity	1.5 S_m
Total primary plus secondary stress intensity	3.0 S_m

The zirconium alloy structural components, which consist of guide thimbles and fuel tubes, are in turn subdivided into two categories because of material difference and functional requirements. The fuel tube design criteria are covered separately in Subsection 4.2.1.1. The maximum shear theory is used to evaluate the guide thimble design. For conservative purposes, the zirconium alloy unirradiated properties are used to define the stress limits.

4.2.1.5.3 Infrequent Incidents (Condition III) and Limiting Faults (Condition IV)

Typical worse case abnormal loads during Conditions III and IV are represented by seismic and pipe rupture loadings. The design criteria for this category of loadings are as follows:

- Deflections or excessive deformation of components cannot interfere with capability of insertion of the control rods or emergency cooling of the fuel rods.
- The fuel assembly structural components stresses under faulted conditions are evaluated primarily using the methods outlined in Appendix F of the ASME Code, Section III. Since the current analytical methods use linear elastic analysis, the stress allowables are defined as the smaller value of 2.4 S_m or 0.70 S_u for primary membrane and 3.6 S_m or 1.05 S_u for primary membrane plus primary bending. For the austenitic steel fuel assembly components, the stress intensity is defined in accordance with the rules described in the previous section

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for normal operating conditions. For the zirconium alloy components, the stress intensity limits are set at two-thirds of the material yield strength, S_y, at reactor operating temperature. This results in zirconium alloy stress limits being the smaller value of 1.6 S_y or 0.70 S_u for primary membrane and 2.4 S_y or 1.05 S_u for primary membrane plus bending.

- For conservative purposes, the zirconium alloy unirradiated properties are used to define the stress limits.

The material and chemical properties of the fuel assembly components are given in WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5). Subsection 4.2.3.4 discusses the spacer grid crush testing.

Thermal-hydraulic design is discussed in Section 4.4.

4.2.1.6 In-core Control Components

The in-core control components are subdivided into permanent and temporary devices. The permanent components are the rod cluster control assemblies, gray rod cluster assemblies, thimble plugs and secondary neutron source assemblies. The temporary components are the primary neutron source assemblies (which are normally used only in the initial core), and the burnable absorber assemblies. For some reloads, the use of burnable absorbers may be necessary for power distribution control and/or to achieve an acceptable moderator temperature coefficient throughout core life (See Subsection 4.3.1.2.2). [*Design changes to the in-core control components qualify for evaluation using the criteria defined in WCAP-12488-A (Reference 1).*]*

Materials are selected for:

- Compatibility in a pressurized water reactor environment
- Adequate mechanical properties at room and operating temperatures
- Resistance to adverse property changes in a radioactive environment
- Compatibility with interfacing components

Material properties are given in WCAP-9179 (Reference 4).

The design bases for the in-core control components are given in Subsections 4.2.1.6.1 through 4.2.1.6.3.

4.2.1.6.1 Control Rods

For Conditions I and II, the stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide in the design of the RCCA and GRCA structural parts in addition to absorber cladding.

Design conditions considered under the ASME Code, Section III, are as follows:

- External pressure equal to the reactor coolant system operating pressure with appropriate allowance for overpressure transients
- Wear allowance equivalent to 1000 reactor trips
- Bending of the rod due to a misalignment in the guide thimble

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- Forces imposed on the rods during rod drop
- Loads imposed by the accelerations of the control rod drive mechanism
- Radiation exposure during maximum core life. The RCCA absorber material temperature does not exceed its melting temperature (1454°F for silver-indium-cadmium [Ag-In-Cd]), (see WCAP-9179, Reference 4). Similarly, the GRCA absorber material temperature does not exceed its melting temperature (6116°F for tungsten), (see WCAP-16943-P-A, Reference 25).
- Temperature effects at operating conditions

4.2.1.6.2 Burnable Absorber Rods

For Conditions I and II, the stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide in the design of the burnable absorber cladding. For abnormal loads during Conditions III and IV, code stresses are not considered limiting. Failures of the burnable absorber rods during these conditions must not interfere with reactor shutdown or emergency cooling of the fuel rods. The burnable absorber material is nonstructural. The structural elements of the burnable absorber rod are designed to maintain the absorber geometry even if the absorber material is fractured.

To reduce the dissolved boron requirement for control of excess reactivity, wet annular burnable absorber (WABA) rods have been incorporated in the first core design. The burnable absorber material is boron carbide contained in an alumina matrix. Thermal-physical and gas release properties of alumina-boron carbide are described in WCAP-9179 (Reference 4) and WCAP-10021-P-A (Reference 10). Discrete burnable absorber rods are designed so that the absorber temperature does not exceed 1200°F during normal operation or an overpower transient. The 1200°F maximum temperature helium gas release in a discrete burnable absorber rod will not exceed 30 percent of theoretical. See WCAP-10021-P-A (Reference 10).

4.2.1.6.3 Neutron Source Rods

The neutron source rods are designed to withstand the following:

- The external pressure equal to reactor coolant system operating pressure with appropriate allowance for overpressure transients
- An internal pressure equal to the pressure generated by released gases over the source rod life

4.2.1.7 Surveillance Program

Subsection 4.2.4.6 discusses the testing and fuel surveillance operation experience program that has been and is being conducted to verify the adequacy of the fuel performance and design bases. Fuel surveillance and testing results, as they become available, are used to improve fuel rod design and manufacturing processes and to confirm that the design bases and safety criteria are satisfied.

4.2.2 Description and Design Drawings

The fuel assembly, fuel rod, and in-core control component design data is given in Table 4.3-1.

Each fuel assembly consists of 264 fuel rods, 24 guide thimbles, and 1 instrumentation tube arranged within a supporting structure. The instrumentation thimble is located in the center position and

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provides a channel for insertion of an in-core neutron detector, if the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly, a gray rod cluster assembly, a neutron source assembly, a burnable absorber assembly, or a thimble plug, depending on the position of the particular fuel assembly in the core. Figure 4.2-1 shows a cross-section of the fuel assembly array, and Figure 4.2-2 shows a fuel assembly full-length view.

The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles. The fuel rods are supported within the fuel assembly structure by fourteen grids (top grid (1), bottom grid (1), intermediate grids (8) and intermediate flow mixer (IFM) grids (4)), plus one protective grid. The top grid is fabricated from nickel-chromium-iron Alloy 718. The bottom grid is fabricated from nickel-chromium-iron Alloy 718. The intermediate grids and the IFM grids are fabricated from ZIRLO (see WCAP-12610-P-A, Reference 5). Top, bottom, and intermediate grids provide axial and lateral support to the fuel rods. In addition, the four IFM grids located near the center of the fuel assembly and between the intermediate grids provide additional fuel rod restraint. The protective grid, in combination with the debris filter bottom nozzle (DFBN), the protective zirconium oxide coated fuel cladding, and the long, solid fuel rod bottom end plug, provide debris failure mitigation.

Fuel assemblies are installed vertically in the reactor vessel and stand upright on the lower core plate, which is fitted with alignment pins to locate and orient each assembly. After the fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears down against the hold-down springs on the top nozzle of each fuel assembly to hold the fuel assemblies in place.

Improper orientation of fuel assemblies within the core is prevented by the use of an indexing hole in one corner of the top nozzle top plate. The assembly is oriented with respect to the handling tool and the core by means of a pin inserted into this indexing hole. Visual confirmation of proper orientation is also provided by an engraved identification number on the opposite corner clamp.

4.2.2.1 Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in cold-worked and stress relieved ZIRLO tubing, which is plugged and seal-welded at the ends to encapsulate the fuel. ZIRLO is an advanced zirconium based alloy selected for its mechanical properties and low neutron absorption cross-section (see WCAP-12610-P-A, Reference 5). Figure 4.2-3 shows a schematic of the fuel rod. The fuel pellets are right circular cylinders consisting of slightly enriched uranium dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly, to allow greater axial expansion at the pellet centerline and to increase the void volume for fission gas release. The ends of each pellet also have a small chamfer at the outer cylindrical surface which improves manufacturability, and mitigates potential pellet damage due to fuel rod handling.

Void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the clad and the fuel, and fuel density changes during irradiation. To facilitate the extended burnup capability necessitated by longer operating cycles, the fuel rod is designed with two plenums (upper and lower) to accommodate the additional fission gas release. The upper plenum volume is maintained by a fuel pellet hold-down spring. The lower plenum volume is maintained by a standoff assembly.

Shifting of the fuel within the clad during handling or shipping, prior to core loading, is prevented by a stainless steel helical spring which bears on top of the fuel pellet stack. Assembly consists of plugging and welding the bottom of the cladding, installing the bottom plenum spacer assembly, fuel

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pellets and top plenum spring, and then plugging and welding the top of the rod. The solid bottom end plug has an internal grip feature and tapered end to facilitate fuel rod loading during fuel assembly fabrication and reconstitution. Additionally, the bottom end plug is designed to be sufficiently long to extend through the protective grid. The bottom section of the fuel rod has a protective zirconium oxide coated surface feature. Use of the protective grid with a longer end plug and the debris filter bottom nozzle, in addition to the coated cladding surface, constitutes a three-level debris protection package, which enhances the fuel reliability performance against trapped debris. This precludes any breach in the fuel rod pressure boundary due to clad fretting wear induced by debris trapped at the bottom section of the fuel assembly.

The fuel rods are internally pressurized with helium during the welding process to minimize compressive clad stresses and prevent clad flattening under reactor coolant operating pressures. The fuel rods are pre-pressurized and designed so that:

- The internal gas pressure mechanical design limit referred to in Subsection 4.2.1.3 is not exceeded
- The cladding stress-strain limits (Subsection 4.2.1.1) are not exceeded for Condition I and II events
- Clad flattening will not occur during the fuel core life

The AP1000 fuel rod design may also include axial blankets. The axial blankets consist of fuel pellets of a reduced enrichment at each end of the fuel rod pellet stack. Axial blankets reduce neutron leakage axially and improve fuel utilization. The axial blankets use chamfered pellets that are longer than the enriched pellets to help prevent accidental mixing during manufacturing. Furthermore, axial blankets have no impact on the source range detector response, since the reduction in power from the axial blanket is limited to the top and bottom 0.67 feet of the core, while the source range detectors are centered typically about three feet from the bottom of the core.

The AP1000 fuel rod design may also include annular fuel pellets in the top and bottom 8 inches of the fuel stack. These pellets can be either fully enriched or partially enriched. The annular fuel pellets provide additional void volume in the fuel rod to accommodate fission gas release.

The AP1000 fuel rods include integral fuel burnable absorbers. The integral fuel burnable absorbers are boride-coated fuel pellets. The boride-coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin boride coating less than 0.001 inch in thickness on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of integral fuel burnable absorber rods within an assembly may vary depending on specific application. See WCAP-12610-P-A (Reference 5).

4.2.2.2 Fuel Assembly Structure

As shown in Figure 4.2-2, the fuel assembly structure consists of a bottom nozzle, top nozzle, fuel rods, guide thimbles, and grids.

4.2.2.2.1 Bottom Nozzle

The bottom nozzle serves as the bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The nozzle is fabricated from Type 304 and Grade CF-3 stainless steel and consists of a perforated plate, and casting which incorporates a skirt and four angle legs with bearing pads. Figure 4.2-2 illustrates this concept. The legs and skirt form a plenum to direct the inlet coolant flow to the fuel assembly. The perforated plate also prevents accidental downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel

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assembly guide thimbles by locked thimble screws, which penetrate through the nozzle and engage with a threaded plug in each guide thimble.

Coolant flows from the plenum in the bottom nozzle, upward through the penetrations in the plate, to the channels between the fuel rods. The penetrations in the plate are positioned between the rows of the fuel rods.

In addition to serving as the bottom structural element of the fuel assembly, the bottom nozzle also functions as a debris filter. The bottom nozzle perforated plate contains a multiplicity of flow holes which are sized to minimize passage of detrimental debris particles into the active fuel region of the core while maintaining sufficient hydraulic and structural margins. Furthermore, the skirt provides improved bottom nozzle structural stability and increased design margins to reduce damage due to abnormal handling.

Axial loads (from top nozzle hold-down springs) imposed on the fuel assembly and the weight of the fuel assembly are transmitted through the bottom nozzle to the lower core plate. Indexing and positioning of the fuel assembly is controlled by alignment holes in two diagonally opposite bearing pads that mate with locating pins in the lower core plate. Lateral loads on the fuel assembly are transmitted to the lower core plate through the locating pins.

The AP1000 bottom nozzle also has a reconstitution design feature which facilitates the easy removal of the nozzle from the fuel assembly. This design incorporates a thimble screw with a circular locking cup located around the screw head. The locking cup is crimped into a local spherical radius relief on the bottom nozzle. To remove the bottom nozzle, a counterclockwise torque is applied to the thimble screw until the locking cup (detents) is relaxed and the thimble screw is removed. This reconstitutable design permits the remote unlocking, the removal, and the relocking of the thimble screws, as the same or a new bottom nozzle is reattached to the fuel assembly.

4.2.2.2.2 Top Nozzle

The reconstitutable top nozzle functions as the upper structural component of the fuel assembly and, in addition, provides a partial protective housing for the rod cluster control assembly, discrete burnable absorber, or other core components. As shown in Figure 4.2-2, the top nozzle assembly includes four sets of hold-down springs, which are secured to the top nozzle top plate. The springs are made of nickel-chromium-iron Alloy 718. The other top nozzle components are made of Type 304L and Grade CF-3 stainless steel.

The adapter plate is provided with round penetrations and slots (with semicircular ends) to permit the flow of coolant upward through the top nozzle. Other round holes are provided in the adapter plate to accept (guide thimble) inserts which are mechanically locked to the adapter plate using a lock tube. The unique design of the insert joint and lock tube are the key design features of the reconstitutable top nozzle.

The ligaments in the adapter plate cover the top of the fuel rods precluding any upward ejection of the fuel rods from the fuel assembly. The enclosure is a box-like structure which establishes the distance between the adapter plate and the top plate. The top plate has a large square hole in the center to permit access for the rod cluster control assembly, burnable absorber assembly, or other components. Hold-down springs are mounted on the top plate and are retained by retaining pins located at diagonally opposite corners of the top plate.

The top plate also contains integral pads located on the two remaining top nozzle corners. The pads include alignment holes which, when fully engaged with the reactor internals upper core plate guide pins, provide proper alignment to the fuel assembly, reactor internals, and rod cluster control assembly.

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As shown in Figure 4.2-4, to remove the top nozzle assembly a tool is first inserted through a lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool which overrides local lock tube deformations and withdraws the lock tubes from the inserts. After the lock tubes have been removed, the nozzle assembly is removed by raising it off the upper slotted ends of the nozzle inserts, which deflect inwardly under the axial lift load.

With the top nozzle assembly removed, direct access is provided for fuel rod examination or replacement. Reconstitution is completed by the remounting of the nozzle assembly and the insertion of lock tubes. Details of this design feature, the design bases and evaluation of the reconstitutable top nozzle are given in WCAP-10444-P-A (Reference 11).

4.2.2.2.3 Guide Thimbles and Instrument Tube

The guide thimbles are structural members that provide channels for the neutron absorber rods, burnable absorber rods, neutron source rods, or other assemblies. Each guide thimble is fabricated from Zircaloy-4 or ZIRLO with constant OD and ID over the entire length. Separate dashpot tubes, which are made from Zircaloy-4 or ZIRLO tubing, are inserted into the bottom portion of the guide thimble tubes. The larger tube diameter at the top section provides a relatively large annular area necessary to permit rapid control rod insertion during a reactor trip, as well as to accommodate the flow of coolant during normal operation. Holes are provided on the guide thimble above the dashpot to reduce the rod drop time. The lower portion of the guide thimble with the dashpot tube results in a dashpot action near the end of the control rod travel during normal trip operation. The dashpot is closed at the bottom by means of an end plug, which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation.

As stated previously, the AP1000 fuel assembly includes a reconstitutable top nozzle as a standard feature. To accommodate the reconstitutable feature, the top end of the zirconium alloy guide thimble is fastened to a tubular sleeve, or insert, by a three tier expansion bulge joint. An expansion tool is inserted inside the nozzle insert and guide thimble to the proper elevation. The four lobes on the expansion tool force the guide thimble and insert outward locally to a predetermined diameter, therefore joining the two components.

Upon installation of the top nozzle assembly, the bulge near the top of the nozzle insert is captured in a corresponding groove in the hole of the top nozzle adapter plate. As shown in Figure 4.2-4, the mechanical connection between the nozzle insert-guide thimble and top nozzle is made by insertion of a lock tube into the insert. The design of the top grid sleeve-guide thimble and top nozzle insert-guide thimble bulge joint connections have been mechanically tested and found to meet applicable design criteria.

The fuel rod support grids are secured to the guide thimbles using a similar bulge joint connection to create an integral structure. Attachment of the intermediate mixing vane and intermediate flow mixer (IFM) zirconium alloy grids to the guide thimbles is performed using the fastening technique depicted in Figures 4.2-5 and 4.2-6.

The intermediate mixing vane and intermediate flow mixer grids employ a single tier bulge connection between the grid sleeve and guide thimble as compared to the two tier bulge connection used for the top grid. The design of the single tier bulge joint connection has also been mechanically tested and meets the design requirements.

The bottom nickel-chromium-iron Alloy 718 grid is secured to the guide thimble assembly by a double tier bulge connection between the grid sleeve and guide thimble. The design of the double tier bulge joint connection has also been mechanically tested and meets the design requirements.

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The lower end of the guide thimble is fitted with a welded end plug. The nickel-chromium-iron Alloy 718 protective grid is secured to the guide thimble assembly by nickel-chromium-iron Alloy 718 spacers that are spot-welded to the grid. As shown in Figure 4.2-7, the spacer is captured between the guide thimble end plug and the bottom nozzle by means of a (thimble) locking screw.

The described methods of grid fastening are standard and have been used successfully since the introduction of zirconium alloy guide thimbles in 1969.

The central instrumentation tube in each fuel assembly is constrained by seating in counterbores located in both top and bottom nozzles. The instrumentation tube has a constant diameter and provides an unrestricted passageway for the in-core neutron detector which enters the fuel assembly from the top nozzle. Furthermore, the instrumentation tube is secured to the top, bottom, and mid-grids with bulge joint connections similar to those previously discussed for securing the grids to the guide thimbles.

4.2.2.2.4 Grid Assemblies

As shown in Figure 4.2-2, the fuel rods are supported at intervals along their lengths by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is given support at six contact points within each grid by the combination of support dimples and springs. The grid assembly consists of individual slotted straps assembled and interlocked into an egg-crate type arrangement with the straps permanently joined at their points of intersection. The straps may contain springs, support dimples, and mixing vanes; or any such combination.

Two types of structural grid assemblies are used on the AP1000 fuel assembly. One type, with mixing vanes projecting from the edges of the straps into the coolant stream, is used in the high heat flux region of the fuel assemblies to promote mixing of the coolant. The other type, located at the top and bottom of the assembly, does not contain mixing vanes on the internal straps. The outside straps on the grids contain mixing vanes that, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core.

Because of its corrosion resistance and high strength properties, the bottom grid material chosen for the AP1000 fuel assembly design is nickel-chromium-iron Alloy 718. The top grid is fabricated from nickel-chromium-iron Alloy 718. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies are designed to allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

The eight intermediate (mixing vane), or structural grids on the AP1000 fuel assembly are made of ZIRLO. This material was selected to take advantage of the material's inherent low neutron capture cross-section. The zirconium alloy grids have thicker straps than the nickel-chromium-iron alloy grids. The zirconium alloy grid incorporates the same grid cell support configuration as the nickel-chromium-iron alloy grid. The zirconium alloy interlocking strap joints and grid/sleeve joints are fabricated by laser welding, whereas the nickel-chromium-iron alloy grid joints (except the protective grid) are brazed. The interlocking strap joints for the protective grid are also fabricated by laser welding. The mixing vanes incorporated in the zirconium alloy intermediate grids induce additional flow mixing among the various flow channels in a fuel assembly as well as between adjacent fuel assemblies. This additional flow mixing enhances thermal performance.

As shown in Figure 4.2-2, the intermediate flow mixer grids are located at selected spans between the zirconium alloy mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hotter fuel assembly spans. Each intermediate flow mixer grid cell contains four dimples that are designed to prevent mid-span channel closure in the

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spans containing intermediate flow mixers and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the intermediate flow mixer grid can accomplish its flow mixing objective with minimal pressure drop.

The intermediate flow mixer (IFM) grids, like the mixing vane grids, are fabricated from ZIRLO. The intermediate flow mixer grids are manufactured using the same basic techniques as the zirconium alloy structural grid assemblies and are joined to the guide thimbles via sleeves which are welded at the bottom of appropriate grid cells.

Grid impact testing has been performed on zirconium alloy structural grids and the intermediate flow mixer grids indicative of the AP1000 design. The purpose of the testing was to determine the dynamic buckling, or crush, strength of the grids. The grid impact testing was performed at an elevated temperature of 600°F. This temperature is a conservative value representing the core average temperature at the mid-grid locations.

The intermediate flow mixer grids are not intended to be structural members. The intermediate flow mixer grids do, however, share the loads of the structural grids during faulted loading and, as such, contribute to enhance the load carrying capability of the AP1000 fuel assembly.

The dynamic crush strength of the AP1000 structural grids and intermediate flow mixer grids envelope the calculated grid impact loading during combined seismic and pipe rupture events. A coolable geometry is, therefore, provided at the intermediate flow mixer grid elevations, as well as at the structural grid elevations.

4.2.2.3 In-core Control Components

Reactivity control is provided by neutron absorbing rods, gray rods, burnable absorber rods, and a soluble chemical neutron absorber (boric acid). The boric acid concentration is varied to control long-term reactivity changes such as:

- Fuel depletion and fission product buildup
- Cold to hot, zero power reactivity changes
- Reactivity change produced by intermediate-term fission products such as xenon and samarium
- Burnable absorber depletion

The chemical and volume control system, which is used to adjust the level of boron in the coolant, is discussed in Section 9.3.

The rod cluster control assemblies provide reactivity control for:

- Shutdown
- Reactivity changes due to coolant temperature changes in the power range
- Reactivity changes associated with the power coefficient of reactivity
- Reactivity changes due to void formation

A negative power coefficient is maintained at hot, full-power conditions throughout the entire cycle to reduce possible deleterious effects caused by a positive coefficient during pipe rupture or loss-of-flow

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accidents. The first fuel cycle needs more excess reactivity than subsequent cycles due to the loading of fresh (unburned) fuel. Since soluble boron alone is insufficient to provide a negative moderator coefficient, burnable absorber assemblies are also used. Use of burnable absorber assemblies during reloads is discussed in Subsection 4.3.1.2.2.

The most effective reactivity control components are the rod cluster control assemblies and the corresponding drive rod assemblies, which along with the gray rod cluster assemblies, are the only kinetic parts in the reactor. Figure 4.2-8 identifies the rod cluster control and drive rod assembly, in addition to the arrangement of these components in the reactor relative to the interfacing fuel assembly, guide thimbles, and control rod drive mechanism. The arrangement for the gray rod cluster assemblies is the same.

As shown in Figure 4.2-8, the guidance system for the rod cluster control assembly is provided by the guide thimbles. The guide thimbles provide two regimes of guidance: first, in the lower section, a continuous guidance system provides support immediately above the core, which protects the rod against excessive deformation and wear caused by hydraulic loading. Second, the region above the continuous section provides support and guidance at uniformly spaced intervals.

As shown in Figure 4.2-9, the envelope of support is determined by the pattern of the control rod cluster. The guide thimbles provide alignment and support of the control rods, spider body, and drive rod while maintaining trip times at or below required limits.

Subsections 4.2.2.3.1 through 4.2.2.3.4 describe each reactivity control component in detail. The control rod drive mechanism assembly is described in Subsection 9.3.4. The neutron source assemblies provide a means of monitoring the core during periods of low neutron activity.

4.2.2.3.1 Rod Cluster Control Assemblies

The rod cluster control assemblies are divided into two categories: control and shutdown. The control groups compensate for reactivity changes due to variations in operating conditions of the reactor, that is, power and temperature variations. Two nuclear design criteria have been employed for selection of the control group. First, the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to confirm that the power capability is met. The control and shutdown groups provide adequate shutdown margin.

As illustrated in Figure 4.2-9, a rod cluster control assembly is comprised of a group of individual neutron absorber rods fastened at the top end to a common spider assembly.

The absorber material used in these rods is silver-indium-cadmium alloy, which is essentially “black” to thermal neutrons and has sufficient additional resonance absorption to significantly increase worth. As such, these rods are sometimes referred to as “black” rods. As shown in Figure 4.2-10, the absorber material is in the form of solid bars sealed in cold-worked stainless steel tubes. Sufficient diametral and end clearance is provided to accommodate relative thermal expansions.

The control rods have bottom plugs with bullet-like tips to reduce the hydraulic drag during reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles.

The material used in the absorber rod end plugs is Type 308 or 308L stainless steel. The design stresses used for these materials are the same as those defined in the ASME B&PV Code for Type 304 or 304L stainless steel, which have essentially the same strength properties as Type 308 and 308L stainless steel, respectively.

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The allowable stresses used as a function of temperature are listed in Table 2A of the ASME Code, Section II, Part D. The fatigue strength for the Type 308 or 308L material is based on the S-N curve for austenitic stainless steels in Figure I-9.2 of the ASME Code, Section III.

The spider assembly is in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Internal groove-like profiles to facilitate handling tool and drive rod assembly connection are machined into the upper end of the hub. Coil springs inside the spider body absorb the impact energy at the end of a trip insertion. The radial vanes may either be joined to the hub by welding and brazing, and the fingers joined to the vanes by brazing, or the vanes and fingers may be integral with the spider body. A bolt, which holds the springs and retainer, is threaded into the hub within the skirt and welded to prevent loosening while in service.

The components of the spider assembly are made from Types 304, 304L and/or CF-3 (casting equivalent of 304L) stainless steel except for the retainer, which is of Type 630 material, and the springs, which are nickel-chromium-iron Alloy 718.

The absorber rods are fastened securely to the spider. The rods are first threaded into the spider fingers and then secured with a locking device. The end plug below the thread is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

The overall length of the rod cluster control assembly is such that, when the assembly is withdrawn through its full travel, the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble.

4.2.2.3.2 Gray Rod Cluster Assemblies

Externally the mechanical design of the gray rod cluster assembly is identical to the rod cluster control assembly. In addition, the control rod drive mechanism and the interface with the fuel assemblies and guide thimbles are identical to those of the rod cluster control assembly.

As shown in Figure 4.2-11, the gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub or spider. Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that the absorber material consists of tungsten encapsulated in a nickel-chromium-iron Alloy 718 sleeve and clad with stainless steel cladding which has the same outer diameter as the rod cluster control assembly cladding. The lower portion of the rodlets consists of a stainless steel spacer.

The gray rod cluster assemblies are used in base load operation and load follow maneuvering and provide a mechanical shim to replace the use of changes in the concentration of soluble boron, that is, a chemical shim, normally used for this purpose. The AP1000 uses 53 rod cluster control assemblies and 16 gray rod cluster assemblies.

4.2.2.3.3 Burnable Absorber Assembly

Each burnable absorber assembly consists of discrete burnable absorber rods attached to a hold-down assembly. Figure 4.2-12 shows the burnable absorber assemblies. When needed for nuclear considerations, burnable absorber assemblies are inserted into selected thimbles within fuel assemblies.

The wet annular burnable absorber rods (WABA) consist of pellets of alumina-boron carbide material contained within zirconium alloy tubes. These zirconium alloy tubes, which form the outer clad for the burnable absorber rod, are plugged, pressurized with helium, and seal-welded at each end to

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encapsulate the stack of absorber material. The absorber stack length, shown in Figure 4.2-12, is positioned axially within the burnable absorber rod by the use of a zirconium alloy bottom-end spacer as necessary.

The burnable absorber rods in each fuel assembly are grouped and attached together at the top end of the rods to a hold-down assembly by a flat, perforated retaining plate, which fits within the fuel assembly top nozzle and rests on the adapter plate.

The retaining plate and the burnable absorber rods are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals assembly is lowered into the reactor. With this arrangement, the burnable absorber rods cannot be ejected from the core by flow forces. Each rod is attached to the baseplate by a nut that is crimped into place.

4.2.2.3.4 Neutron Source Assemblies

The purpose of a neutron source assembly is to provide a base neutron level to give confidence that the detectors are operational and responding to core multiplication neutrons. For the first core, neutron sources are placed in the reactor to provide a positive neutron count of at least two counts per second on the source range detectors attributable to core neutrons. The detectors, called source range detectors, are used primarily during subcritical modes of core operation.

The source assembly also permits detection of changes in the core multiplication factor during core loading, refueling, and approach to criticality. This can be done since the multiplication factor is related to an inverse function of the detector count rate. Changes in the multiplication factor can be detected during addition of fuel assemblies while loading the core, changes in control rod positions, and changes in boron concentration.

Both primary and secondary neutron source rods are used. The primary source rods, containing a radioactive material, spontaneously emit neutrons during initial core loading, reactor startup, and initial operation of the first core. After the primary source rods decay beyond the desired neutron flux level, neutrons are then supplied by the secondary source rods. The secondary source rods contain a stable material, which is activated during reactor operation. The activation results in the subsequent release of neutrons.

Four source assemblies are typically installed in initial load of the reactor core: two primary source assemblies and two secondary source assemblies. Each primary source assembly contains one primary source rod and a number of burnable absorber rods. Each secondary source assembly contains a symmetrical grouping of secondary source rodlets. Figure 4.2-14 shows the primary source assembly. Figure 4.2-15 shows the secondary source assembly.

Neutron source assemblies are employed at opposite sides of the core. The source assemblies are inserted into the rod cluster control guide thimbles in fuel assemblies at selected locations.

As shown in Figures 4.2-14 and 4.2-15, the source assemblies contain a hold-down assembly identical to that of the burnable absorber assembly.

The primary and secondary source rods both use the same cladding material as the absorber rods. The secondary source rods contain antimony-beryllium pellets stacked to a height of approximately 88 inches. The primary source rods contain capsules of californium (plutonium-beryllium possible alternate) source material and alumina spacers to position the source material within the cladding. The rods in each assembly are fastened at the top end to a hold-down assembly.

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The other structural members, except for the springs, are constructed of Type 304, 304L, and 308L stainless steel. The springs exposed to the reactor coolant are nickel-chromium-iron Alloy 718.

4.2.3 Design Evaluation

[*The fuel assemblies, fuel rods, and in-core control components are designed to satisfy the performance and safety criteria of*]* Section 4.2 of the Standard Review Plan, the mechanical design bases of Subsection 4.2.1 and [*the Fuel Criteria Evaluation Process per WCAP-12488-A (Reference 1)*]*, and other interfacing nuclear and thermal and hydraulic design bases specified in Sections 4.3 and 4.4.

Effects of Conditions II, III, IV or anticipated transients without trip on fuel integrity are presented in Chapter 15.

The initial step in fuel rod design evaluation for a region of fuel is to determine the limiting rod(s). Limiting rods are defined as those rods whose predicted performance provides the minimum margin to each of the design criteria. For a number of design criteria, the limiting rod is the lead burnup rod of a fuel region. In other instances, it may be the maximum power or the minimum burnup rod. For the most part, no single rod is limiting with respect to all the design criteria.

After identifying the limiting rod(s), an analysis is performed to consider the effects of rod operating history, model uncertainties, and dimensional variations. To verify adherence to the design criteria, the evaluation considers the effects of postulated transient power changes during operation consistent with Conditions I and II. These transient power increases can affect both rod average and local power levels. Parameters considered include rod internal pressure, fuel temperature, clad stress, and clad strain. In fuel rod design analyses, these performance parameters provide the basis for comparison between expected fuel rod behavior and the corresponding design criteria limits.

Fuel rod and assembly models used for the performance evaluations are documented and maintained under an appropriate control system. Material properties used in the design evaluations are given in WCAP-12610 (Reference 5).

4.2.3.1 Cladding

4.2.3.1.1 Vibration and Wear

Fuel rod vibrations are flow induced. The effect of vibration on the fuel assembly and individual fuel rods is minimal. The cyclic stress range associated with deflections of such small magnitude is insignificant and has no effect on the structural integrity of the fuel rod.

The reaction force on the grid supports, due to rod vibration motions, is also small and is much less than the spring preload. Adequate fuel clad spring contact is maintained. No significant wear of the clad or grid supports is predicted during the life of the fuel assembly based on out-of-pile flow tests, performance of similarly designed fuel in operating reactors, and design analyses.

Clad fretting and fuel vibration has been experimentally investigated, as shown in WCAP-8278 (Reference 13).

4.2.3.1.2 Fuel Rod Internal Pressure and Cladding Stresses

A burnup-dependent fission gas release model (WCAP-15063-P-A, Revision 1 [Reference 21]) is used in determining the internal gas pressure as a function of irradiation time. The plenum volume of the fuel rod has been designed to provide that the maximum internal pressure of the fuel rod will not exceed the value which would cause:

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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- The fuel/clad diametral gap to increase during steady-state operation
- Extensive departure from nucleate boiling propagation to occur (Reference 26)

The clad stresses at a constant local fuel rod power are low. Compressive stresses are created by the pressure differential between the coolant pressure and the rod internal gas pressure. Because of the pre-pressurization with helium, the volume average effective stresses are always less than approximately 14,000 psi at the pressurization level used in the AP1000 fuel rod design. Stresses due to the temperature gradient are not included in this average effective stress because thermal stresses are, in general, negative at the clad inside diameter and positive at the clad outside diameter, and their contribution to the clad volume average stress is small. Furthermore, the thermal stress decreases with time during steady-state operation due to stress relaxation. The stress due to pressure differential is highest in the minimum power rod at beginning-of-life due to low internal gas pressure and decreases as rod power increases. Thermal stresses are maximum in the maximum power rod due to the larger temperature gradient and decrease as the rod power is decreased.

The internal gas pressure at beginning-of-life ranges from approximately 200 to 750 psi for typical lead burnup fuel rods. The total tangential stress at the clad inside diameter at beginning-of-life is approximately 19,500 psi compressive (approximately 18,500 psi due to ΔP and approximately 1,000 due to ΔT) for a low-power rod operating at four kilowatts/foot. Total tangential stress is approximately 20,500 psi compressive (approximately 18,000 psi due to ΔP and approximately 2,500 psi due to ΔT) for a high-power rod operating at 10 kilowatts/foot. However, the volume average effective stress at beginning-of-life is between approximately 13,500 psi (high-power rod) and approximately 14,000 psi (low-power rod). These stresses are substantially below even the unirradiated clad yield strength (approximately 55,500 psi) at a typical clad mean operating temperature of 700°F.

Tensile stresses could be created once the clad has come in contact with the pellet. These stresses would be induced by the fuel pellet swelling during irradiation. Swelling of the fuel pellet can result in small clad strains (less than one percent) for expected discharge burnups, but the associated clad stresses are very low because of clad creep (thermal- and irradiation-induced creep). The one percent strain criterion is extremely conservative for fuel-swelling driven clad strain because the strain rate associated with solid fission products swelling is very slow. A detailed discussion of fuel rod performance is given in Subsection 4.2.3.3.

4.2.3.1.3 Material and Chemical Evaluation

ZIRLO clad has a high corrosion resistance to the coolant, fuel, and fission products. As shown in WCAP-8183 (Reference 3), there is considerable pressurized water reactor operating experience on the capability of Zircaloy-4 as a clad material. ZIRLO, an advanced zirconium based alloy, has equal or better corrosion resistance than Zircaloy-4 (see WCAP-12610-P-A, [Reference 5]). Controls on fuel fabrication specify maximum moisture levels to preclude clad hydriding.

Metallographic examination of irradiated commercial fuel rods has shown occurrences of fuel/clad chemical interaction. Reaction layers of less than one mil in thickness have been observed between fuel and clad at limited points around the circumference. Metallographic data indicates that this interface layer remains very thin even at high burnup. Thus, there is no indication of propagation of the layer and eventual clad penetration.

Stress corrosion cracking is another postulated phenomenon related to fuel/clad chemical interaction. Out-of-pile tests have shown that in the presence of high clad tensile stresses, large concentrations of iodine can chemically attack the zirconium alloy tubing and may lead to eventual clad cracking. Extensive post-irradiation examination has produced no evidence that this mechanism has been operative in Westinghouse commercial pressurized water reactor fuel.

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4.2.3.1.4 Rod Bowing

WCAP-8691 (Reference 14) presents the model used for evaluation of AP1000 fuel rod bowing. This model has been used for bow assessment in 14x14, 15x15, and 17x17 type cores.

4.2.3.1.5 Consequences of Power Coolant Mismatch

Consequences of power coolant mismatch are discussed in Chapter 15.

4.2.3.1.6 Creep Collapse and Creepdown

This subject and the associated irradiation stability of cladding have been evaluated. In WCAP-13589-A (Reference 8), it is shown that current generation Westinghouse fuel is sufficiently stable with respect to fuel densification. Significant axial gaps do not form in the pellet stack, preventing clad collapse from occurring. The design basis of no clad collapse during planned core life is therefore satisfied. Cladding collapse analyses, if required, would be performed using the methods described in WCAP-8377 (Reference 22).

4.2.3.2 Fuel Materials Considerations

Sintered, high-density uranium dioxide fuel reacts only slightly with the clad at core operating temperatures and pressures. In the event of clad defects, the high resistance of uranium dioxide to attack by water protects against fuel deterioration, although limited fuel erosion can occur. The consequences of defects in the clad are greatly reduced by the ability of uranium dioxide to retain fission products, including those which are gaseous or highly volatile.

Observations from several early Westinghouse pressurized water reactors as discussed in WCAP-8218-P-A (Reference 6) have shown that fuel pellets can densify under irradiation to a density higher than the manufactured values. Fuel densification and subsequent settling of the fuel pellets can result in local and distributed gaps in the fuel rods. The densification process is related to the elimination of very small as-fabricated porosity in the fuel during irradiation. Early fuels were intentionally manufactured to low initial density and were undersintered, which resulted in a large fraction of very small pores. Densification behavior in current fuel is controlled by improved manufacturing process controls and by specifying a nominal 95.5 percent initial fuel density, which results in reduced levels of small, densifying porosity.

The evaluation of fuel densification effects and the treatment of fuel swelling and fission gas release are described in WCAP-13589-A (Reference 8) and WCAP-15063-P-A, Revision 1 (Reference 21).

4.2.3.3 Fuel Rod Performance

In the calculation of the steady-state performance of a nuclear fuel rod, the following interacting factors are considered:

- Clad creep and elastic deflection
- Pellet density changes, thermal expansion, gas release, and thermal properties as a function of temperature and fuel burnup
- Internal pressure as a function of fission gas release, rod geometry, and temperature distribution

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These effects are evaluated using fuel rod design models, as discussed in WCAP-15063-P-A, Revision 1 (Reference 21), that include appropriate models for time dependent fuel densification. With these interacting factors considered, the model determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and clad temperatures, and clad deflections are calculated. The fuel rod is divided into several axial sections and radially into a number of annular zones. Fuel density changes are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure.

The initial rod internal pressure is selected to delay fuel/clad mechanical interaction and to avoid the potential for clad flattening. It is limited, however, by the design criteria for the rod internal pressure, as discussed in Subsection 4.2.1.3.

The gap conductance between the pellet surface and the clad inner diameter is calculated as a function of the composition, temperature and pressure of the gas mixture, and the gap size or contact pressure between the clad and pellet. After computing the fuel temperature for each pellet zone, the fractional fission gas release is assessed using an empirical model derived from experimental data, as detailed in WCAP-15063-P-A, Revision 1 (Reference 21). The total amount of gas released is based on the average fractional release within each axial and radial zone and the gas generation rate, which, in turn, is a function of burnup. Finally, the gas released is summed over the zones, and the pressure is calculated.

The model shows close agreement in fit for a variety of published and proprietary data on fission gas release, fuel temperatures, and clad deflections, as detailed in WCAP-15063-P-A, Revision 1 (Reference 21). These data include variations in power, time, fuel density, and geometry.

4.2.3.3.1 Fuel/Cladding Mechanical Interaction

One factor in fuel element duty is potential mechanical interaction of the fuel and clad. This fuel/clad interaction produces cyclic stresses and strains in the clad, and these, in turn, reduce clad life. The reduction of fuel/clad interaction is therefore a goal of design. The technology for using pre-pressurized fuel rods in Westinghouse pressurized water reactors has been developed to further this objective.

The gap between the fuel and clad is initially sufficient to prevent hard contact between the two. However, during power operation a gradual compressive creep of the clad onto the fuel pellet occurs due to the external pressure exerted on the rod by the coolant. Clad compressive creep eventually results in fuel/clad contact. Once fuel/clad contact occurs, changes in power level result in changes in clad stresses and strains. By using pre-pressurized fuel rods to partially offset the effect of the coolant external pressure, the rate of clad creep toward the surface of the fuel is reduced. Fuel rod pre-pressurization delays the time at which fuel/clad contact occurs and, hence, significantly reduces the extent of cyclic stresses and strains experienced by the clad both before and after fuel/clad contact. These factors result in an increase in the fatigue life margin of the clad and lead to greater clad reliability.

A two-dimensional (r,θ) finite element model has been established to investigate the effects of radial pellet cracks on stress concentrations in the clad. Stress concentration herein is defined as the difference between the maximum clad stress in the θ direction and the mean clad stress. The first case has the fuel and clad in mechanical equilibrium; and, as a result, the stress in the clad is close to zero. In subsequent cases the pellet power is increased in steps and the resultant fuel thermal expansion imposes tensile stress in the clad.

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In addition to uniform clad stresses, stress concentrations develop in the clad adjacent to radial cracks in the pellet. These radial cracks have a tendency to open during a power increase, but the frictional forces between fuel and clad oppose the opening of these cracks and result in localized increases in clad stress. As the power is further increased, large tensile stresses exceed the ultimate tensile strength of uranium dioxide and additional cracks in the fuel pellet are created, limiting the magnitude of the stress concentration in the clad.

As part of the standard fuel rod design analysis, the maximum stress concentration evaluated from finite element calculations is added to the volume-averaged effective stress in the clad as determined from one-dimensional stress/strain calculations. The resultant clad stress is then compared to the temperature-dependent cladding yield stress to confirm that the stress/strain criteria are satisfied.

The transient evaluation method is described in the following paragraphs.

Pellet thermal expansion due to power increases is considered the only mechanism by which significant stresses and strains can be imposed on the clad.

Power increases in commercial reactors can result from fuel shuffling (for example, a fuel assembly positioned near the core center for cycle 2 operation after operating near the periphery during cycle 1), reactor power escalation following extended reduced power operation, and full-length control rod movement. In the mechanical design model, lead rods are depleted using best-estimate power histories as determined by core physics calculations. During burnup, the amount of diametral gap closure is evaluated based upon the pellet expansion cracking model, clad creep model, and fuel swelling model. At various times during the depletion, the power is increased locally in the rod to the burnup-dependent attainable power density as determined by core physics calculations. The radial, tangential, and axial clad stresses resulting from the power increase are combined into a volume average effective clad stress.

The von Mises criterion is used to determine whether the clad yield stress has been exceeded. This criterion states that an isotropic material in multi-axial stress will begin to yield plastically when the effective stress exceeds the yield stress as determined by an axial tensile test. The yield stress correlation is that for irradiated cladding, since fuel/clad interaction occurs at high burnup. In applying this criterion, the effective stress is increased by an allowance which accounts for stress concentrations in the clad adjacent to radial cracks in the pellet, prior to the comparison with the yield stress. This allowance was evaluated using a two-dimensional (r,θ) finite element model.

Slow transient power increases can result in large clad strains without exceeding the clad yield stress because of clad creep and stress relaxation. Therefore, in addition to the yield stress criterion, a criterion on allowable clad strain is necessary. Based upon high strain rate burst and tensile test data on irradiated tubing, one percent strain was determined to be a conservative lower limit on irradiated clad ductility and that was adopted as a design criterion.

In addition to the mechanical design models and design criteria, the AP1000 fuel rod design relies on performance data accumulated through transient power test programs in experimental and commercial reactors, and through normal operation in commercial reactors.

It is recognized that a possible limitation to the satisfactory behavior of the fuel rods in a reactor subjected to daily load follow is the failure of the cladding by low-cycle strain fatigue. During their normal residence time in the reactor, the fuel rods may be subjected to on the order of 1000 load follow cycles, with typical changes in power level from 50 to 100 percent of their steady-state values.

The assessment of the fatigue life of the fuel rod cladding is subjected to considerable uncertainty because of the difficulty of evaluating the strain range which results from the cyclic interaction of the fuel pellets and cladding. This difficulty arises, for example, from such highly unpredictable

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phenomena as pellet cracking, fragmentation, and relocation. Westinghouse investigated this particular phenomenon both analytically and experimentally. Strain fatigue tests on irradiated and nonirradiated hydrided Zircaloy-4 cladding were performed. These tests permitted the definition of a conservative fatigue-life limit and recommendation of a methodology to treat the strain fatigue evaluation of the Westinghouse-referenced fuel rod designs. (See WCAP-9500-P-A, Reference 15.)

Successful load follow operation has been performed on several reactors. There was no significant coolant activity increase that could be associated with the load follow mode of operation.

The Westinghouse analytical approach to strain fatigue is based on a comprehensive review of the available strain fatigue models. The review included the Langer-O'Donnell model (Reference 16) the Yao-Munse model, and the Manson-Halford model. Upon completion of this review, and using the results of the Westinghouse experimental programs as documented in WCAP-9500-P-A (Reference 15), it was concluded that the approach defined by Langer-O'Donnell would be retained and the empirical factors of their correlation modified to conservatively bound the results of the Westinghouse testing program.

The design equations followed the concept for the fatigue design criterion according to the ASME Code, Section III:

- The calculated pseudo stress amplitude (Sa) has to be multiplied by a factor of two to obtain the allowable number of cycles (Nf).
- The allowable cycles for a given Sa is five percent of Nf or a safety factor of 20 on cycles.

The lesser of the two allowable numbers of cycles is selected. The cumulative fatigue life fraction is then computed as:

$$\sum_{1}^{k} \frac{n_k}{N_{f\,k}} \leq 1$$

where:

n_k = number of diurnal cycles of mode k.

$N_{f\,k}$ = number of allowable cycles.

4.2.3.3.2 Irradiation Experience

Westinghouse fuel operational experience is presented in CENPD-404-P-A (Reference 27). Additional test assembly and test rod experience is given in WCAP-10125-P-A (Reference 2).

4.2.3.3.3 Fuel and Cladding Temperature

The methods used for evaluation of fuel rod temperatures are presented in Subsection 4.4.2.11.

4.2.3.3.4 Potentially Damaging Temperature Effects During Transients

The fuel rod experiences many operational transients (intentional maneuvers) during its residence in the core. A number of thermal effects must be considered when analyzing the fuel rod performance.

The clad can be in contact with the fuel pellet at some time in the fuel lifetime. Clad/pellet interaction occurs if the fuel pellet temperature is increased after the clad is in contact with the pellet. Clad/pellet interaction is discussed in Subsection 4.2.3.3.1.

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Clad flattening has been observed in some operating power reactors. This is no longer a concern because clad flattening is precluded during the fuel residence in the core (Subsection 4.2.3.1) by the use of stable fuel.

Potential differential thermal expansion between the fuel rods and the guide thimbles during a transient is considered in the design. Excessive bowing of the fuel rods is precluded because the grid assemblies allow axial movement of the fuel rods relative to the grids. Specifically, thermal expansion of the fuel rods is considered in the grid design so that axial loads imposed on the fuel rods during a thermal transient will not result in excessively bowed fuel rods.

4.2.3.3.5 Fuel Element Burnout and Potential Energy Release

As discussed in Subsection 4.4.2.2, the core is protected from departure from nucleate boiling over the full range of possible operating conditions. In the extremely unlikely event that departure from nucleate boiling should occur, the clad temperature will rise due to the steam blanketing at the rod surface and the consequent degradation in heat transfer. During this time there is a potential for chemical reaction between the cladding and the coolant. However, because of the relatively good film boiling heat transfer following departure from nucleate boiling, the energy release resulting from this reaction is insignificant compared to the power produced by the fuel.

4.2.3.3.6 Coolant Flow Blockage Effects on Fuel Rods

The coolant flow blockage effects on fuel rods are presented in Subsection 4.4.4.7.

4.2.3.4 Spacer Grids

The coolant flow channels are established and maintained by the structure composed of grids and guide thimbles. The lateral spacing between fuel rods is provided and controlled by the support dimples of adjacent grid cells. Contact of the fuel rods on the dimples is maintained through the clamping force of the grid springs. Lateral motion of the fuel rods is opposed by the spring force and the internal moments generated between the spring and the support dimples. Grid testing is discussed in WCAP-8236 (Reference 17) and WCAP-10444-P-A (Reference 11).

4.2.3.5 Fuel Assembly

4.2.3.5.1 Stresses and Deflections

The fuel assembly component stress levels are limited by the design. For example, stresses in the fuel rod due to thermal expansion and zirconium alloy irradiation growth are limited by the relative motion of the rod as it slips over the grid spring and dimple surfaces. Clearances between the fuel rod ends and nozzles are provided so that zirconium alloy irradiation growth does not result in rod end interference. Stresses in the fuel assembly caused by tripping of the rod cluster control assembly have little influence on fatigue usage margin because of the small number of events during the life of an assembly. Assembly components and prototype fuel assemblies made from production parts have been subjected to structural tests to verify that the design bases requirements are met.

The fuel assembly design loads for shipping have been established at 4 g axial and 6 g lateral. Accelerometers are permanently placed in the shipping cask to monitor and detect fuel assembly accelerations that would exceed the criteria. Experience indicates that loads that exceed the allowable limits rarely occur. Exceeding the limits requires reinspection of the fuel assembly for damage. Tests on various fuel assembly components, such as the grid assembly, sleeves, inserts, and structure joints, have been performed to confirm that the shipping design limits do not result in impairment of fuel assembly function. Seismic analysis methodology of the fuel assembly

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is presented in WCAP-8236 (Reference 17), WCAP-9401-P-A (Reference 18), and WCAP-10444-P-A (Reference 11).

To demonstrate that the fuel assemblies will maintain a geometry that is capable of being cooled under the worst-case accident Condition IV event, a plant specific or bounding seismic analysis is performed.

The fuel assembly response resulting from safe shutdown earthquake condition is analyzed using time-history numerical techniques. The vessel motion for this type of event primarily causes lateral loads on the reactor core. Consequently, the methodology and analytical procedures as described in WCAP-8236 (Reference 17) and WCAP-9401-P-A (Reference 18) are used to assess the fuel assembly deflections and impact forces.

The motions of the reactor internals upper and lower core plates and the core barrel at the upper core plate elevation, which are simultaneously applied to simulate the reactor core input motion, are obtained from the time-history analysis of the reactor vessel and internals. The fuel assembly response, namely the displacements and impact forces, is obtained with the reactor core model. Similar dynamic analyses of the core were performed using reactor internals motions indicative of the postulated pipe rupture. Scenarios regarding breaches in the pressure boundary are investigated to determine the most limiting structural loads for the fuel assembly. The application of leak-before-break limits the size of the pipe rupture loads for which the fuel assemblies must be analyzed. The pipe rupture used in the fuel assembly analysis is the largest pipe connected to the reactor coolant system which does not satisfy the leak-before-break criteria. Subsection 3.6.3 discusses mechanistic pipe break.

4.2.3.5.1.1 Grid Analyses

The maximum grid impact force obtained from seismic analyses is less than the allowable grid strength. With respect to the guidelines of Appendix A of the Standard Review Plan, Section 4.2, Westinghouse has demonstrated that a simultaneous safe shutdown earthquake and pipe rupture event is highly unlikely. The fatigue cycles, crack initiation, and crack growth due to normal operating and seismic events will not realistically lead to a pipe rupture. More information is available in WCAP-9283 (Reference 19).

Based on the deterministic fracture mechanics evaluation of small flaws in piping components, Westinghouse has demonstrated that the dynamic affects of a large pipe rupture in the primary coolant piping system for the AP1000 design does not have to be considered.

A design basis for the piping design in the AP1000 is that the reactor coolant loop and surge lines will satisfy the leak-before-break criteria for mechanistic pipe break. In addition, the piping connected to the reactor coolant system that is six inch nominal diameter or larger is evaluated for leak-before-break. The result of a pipe leakage event consistent with the mechanistic pipe break evaluation would be to impose insignificant asymmetric loadings on the reactor core system. Thus, fuel assembly grid loads due to large pipe ruptures are unrealistic and, as such, are not included in the analysis.

The pressure boundary integrity for numerous branch lines is analyzed to determine the most limiting break of a line not qualified for leak-before-break for the dynamic loading of the reactor core. Grid loads resulting from a combined seismic and pipe rupture event do not cause unacceptable grid deformation as to preclude a core coolable geometry.

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4.2.3.5.1.2 Nongrid Analyses

The stresses induced in the various fuel assembly nongrid components are assessed based on the most limiting seismic condition. The fuel assembly axial forces resulting from the hold-down spring load together with its own weight distribution are the primary sources of the stresses in the guide thimbles and fuel assembly nozzles. The fuel rod accident induced stresses, which are generally very small, are caused by bending due to the fuel assembly deflections during a seismic event. The seismic-induced stresses are compared with the allowable stress limits for the fuel assembly major components. The component stresses, which include normal operating stresses, are below the established allowable limits. Consequently, the structural designs of the fuel assembly components are acceptable for the design basis accident conditions for the AP1000.

4.2.3.5.2 Dimensional Stability

Localized yielding and slight deformation in some fuel assembly components are allowed to occur during a Condition III or IV event. The maximum permanent deflection, or deformations, do not result in any violation of the functional requirements of the fuel assembly.

4.2.3.6 Reactivity Control Assemblies and Burnable Absorber Rods

4.2.3.6.1 Internal Pressure and Cladding Stresses during Normal, Transient, and Accident Conditions

The designs of the burnable absorber, source, and gray rods provide a sufficient cold void volume to accommodate the internal pressure increase during operation. This is not a concern for the rod cluster control assembly absorber rodlets because no significant amount of gas is released by the silver-indium-cadmium absorber material.

For the discrete burnable absorber rod, there is sufficient cold void volume to limit the internal pressure to a value, which satisfies the design criteria. For the source rods and gray rods, a void volume is provided within the rod to limit the maximum internal pressure increase at end-of-life. Figures 4.2-14 and 4.2-15 detail the primary and secondary source assemblies and Figure 4.2-11 details the gray rod cluster assembly.

During normal transient and accident conditions, the void volume limits the internal pressures to values that satisfy the criteria in Subsection 4.2.1.6. These limits are established not only to prevent the peak stresses from reaching unacceptable values, but also to limit the amplitude of the oscillatory stress component in consideration of the fatigue characteristics of the materials.

Rod, guide thimble, and dashpot flow analyses indicate that the flow is sufficient to prevent coolant boiling within the guide thimble. Therefore, clad temperatures at which the clad material has adequate strength to resist coolant operating pressures and rod internal pressures are maintained.

4.2.3.6.2 Thermal Stability of the Absorber Material, Including Changes and Thermal Expansion

The radial and axial temperature profiles within the source and absorber rods are determined by considering gap conductance, thermal expansion, neutron or gamma heating of the contained material as well as gamma heating of the clad.

The maximum temperatures of the silver-indium-cadmium RCCA or tungsten GRCA absorber materials are calculated and found to be significantly less than the material melting point and found to occur axially at only the highest flux region. The mechanical and thermal expansion properties of the silver-indium-cadmium absorber material are discussed in WCAP-9179 (Reference 4).

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The mechanical and thermal expansion properties of the tungsten absorber material are discussed in WCAP-16943-P-A (Reference 25).

The wet annular burnable absorber (WABA) assemblies are used in the first core. The maximum temperature of the alumina-boron carbide burnable absorber pellet is expected to be less than 1200°F which takes place following the initial power ascent. As the operating cycle proceeds, the burnable absorber pellet temperature decreases due to a reduction in heat generation due to boron depletion and better gap conduction as the helium produced diffuses into the gap.

Sufficient diametral and end clearances have been provided in the neutron absorber, burnable absorber, and source rods to accommodate the relative thermal expansions between the enclosed material and the surrounding clad and end plug.

4.2.3.6.3 Irradiation Stability of the Absorber Material, Taking into Consideration Gas Release and Swelling

The irradiation stability of the silver-indium-cadmium absorber material is discussed in WCAP-9179 (Reference 4). Irradiation produces no deleterious effects in the absorber material. The irradiation stability of the tungsten absorber material is discussed in WCAP-16943-P-A (Reference 25).

As mentioned in Subsection 4.2.3.6.1, gas release is not a concern for the rod cluster control rod material because no gas is produced by the absorber material. Sufficient diametral and end clearances are provided to accommodate any potential expansion and/or swelling of the absorber material for both RCCA and GRCA absorber rods.

Irradiation produces no deleterious effects in the tungsten absorber material of the gray rodlets. Some minor cracking of the tungsten material may occur, but this cracking does not affect the absorber column geometric stability due to the small clearance between absorber and sleeve (Reference 25).

The alumina-boron carbide burnable absorber pellets are designed such that gross swelling or crumbling of the pellets is not predicted to occur during reactor operation. Some minor cracking of the pellets may occur, but this cracking should not affect the overall absorber and stack integrity.

4.2.3.6.4 Potential for Chemical Interaction, Including Possible Waterlogging Rupture

The structural materials selected have good resistance to irradiation damage and are compatible with the reactor environment.

Corrosion of the materials exposed to the coolant is quite low, and proper control of chloride and oxygen in the coolant minimizes potential for the occurrence of stress corrosion. The potential for the interference with rod cluster control assembly movement due to possible corrosion phenomena is very low.

Waterlogging rupture is not a failure mechanism associated with the AP1000 control rods. Furthermore, a breach of the cladding for any postulated reason does not result in serious consequences.

The silver-indium-cadmium absorber material is relatively inert and will remain inert even when subjected to high coolant velocity regions. Rapid loss of reactivity control material will not occur. Test results detailed in WCAP-9179 (Reference 4) concluded that additions of indium and cadmium to silver, in the amounts to form the silver-indium-cadmium absorber material composition, result in small corrosion rates.

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In the unlikely event of GRCA rod cladding breach, loss of absorber material will not occur because the inner sleeve encapsulates the tungsten absorber (WCAP-16943-P-A, Reference 25).

For the discrete burnable absorber, in the unlikely event that the zirconium alloy clad is breached, the boron carbide in the affected rod(s) could be leached out by the coolant water. If this occurred early, in-core instruments could detect large peaking factor changes, and corrective action would be taken, if warranted. A postulated clad breach after substantial irradiation would have no significant effect on peaking factors since the boron will have been depleted. Breaching of the zirconium alloy clad by internal hydriding is not expected due to moisture controls employed during fabrication. Rods of this design have performed very well with no failures observed.

4.2.4 Testing and Inspection Plan

4.2.4.1 Quality Assurance Program

The Quality Assurance Program Plan of the Westinghouse Commercial Nuclear Fuel Division for the AP1000 is summarized in Chapter 17.

The program provides for control over activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage, and transportation. The program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

Westinghouse drawings and product, process, and material specifications identify the inspections to be performed.

4.2.4.2 Quality Control

Quality control philosophy is generally based on the following inspections being performed to a 95 percent confidence that at least 95 percent of the product meets specification, unless otherwise noted.

4.2.4.2.1 Fuel System Components and Parts

The characteristics inspected depend on the component parts. The quality control program includes dimensional and visual examinations, check audits of test reports, material certification, and nondestructive examination, such as X-ray and ultrasonic.

The material used in the AP1000 core is accepted and released by Quality Control.

4.2.4.2.2 Pellets

Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips, and surface conditions according to approved standards.

Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are taken on a specified sample basis throughout pellet production.

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4.2.4.2.3 Rod Inspection

Fuel rod, rod cluster control rod, discrete burnable absorber rod, and source rod inspections consists of the following nondestructive examination techniques and methods, as applicable:

- Each rod is leak tested using a calibrated mass spectrometer, with helium being the detectable gas.
- Rod welds are inspected by ultrasonic test or X-ray in accordance with a qualified technique and Westinghouse specifications meeting the requirements of ASTM-E-142-86 (Reference 20).
- Rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.
- Fuel rods are inspected by gamma scanning or other approved methods, as discussed in Subsection 4.2.4.5, to confirm proper plenum dimensions.
- Fuel rods are inspected by gamma scanning, or other approved methods, as discussed in Subsection 4.2.4.5, to confirm that no significant gaps exist between pellets.
- Fuel rods are actively and/or passively gamma scanned to verify enrichment control prior to acceptance for assembly loading.
- Traceability of rods and associated rod components is established by quality control.

4.2.4.2.4 Assemblies

Each fuel rod, control rod, burnable absorber rod and source rod assembly is inspected for compliance with drawing and/or specification requirements. Other in-core control component inspection and specification requirements are given in Subsection 4.2.4.4.

4.2.4.2.5 Other Inspections

The following inspections are performed as part of the routine inspection operation:

- Tool and gauge inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on serialized tools. Complete records are kept of calibration and conditions of tools.
- Audits are performed of inspection activities and records to confirm that prescribed methods are followed and that records are correct and properly maintained.
- Surveillance inspection, where appropriate, and audits of outside contractors are performed to confirm conformance with specified requirements.

4.2.4.2.6 Process Control

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

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The uranium dioxide powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and unique barcode numbers. A quality control identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by analysis.

Powder withdrawal from storage can be made by only one authorized group, which directs the powder to the correct pellet production line. The pellet production lines are physically separated from each other, and pellets of only a single nominal enrichment and density are produced in a given production line at any given time.

Finished pellets are placed on trays identified with the same color code as the powder containers and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by quality control. Physical barriers are used to prevent mixing of pellets of different nominal densities and enrichments in the pellet storage area. Unused powder and substandard pellets are returned to storage in the original color-coded containers.

Loading of pellets into the clad is performed in isolated production lines; only one density and enrichment (with possible exception for top and bottom (axial blanket) zones) are loaded on a line at a time.

A serialized traceability code is placed on each fuel tube, which identifies the contract and enrichment. The end plugs are inserted and then welded (in an inert gas atmosphere) to seal the tube. The fuel tube remains coded and traceability identified until just prior to installation in the fuel assembly.

Similar traceability is provided for wet annular burnable absorber, source, and control rods, as required.

4.2.4.3 Letdown Radiation Monitoring

Radiation monitoring of the reactor coolant is made by grab samples and laboratory analysis of the primary coolant. Refer to information presented in Subsections 9.3.3 and 9.3.6, and Table 9.3.3-1.

4.2.4.4 In-core Control Component Testing and Inspection

Tests and inspections are performed on each reactivity control component to verify the mechanical characteristics. In the case of the rod cluster control assembly, prototype testing has been conducted. Manufacturing test/inspections and functional testing at the plant site are both performed.

During the component manufacturing phase, the following requirements apply to the reactivity control components to provide the proper functioning during reactor operation:

- Materials are procured to specifications to attain the desired standard of quality.
- Spider assemblies with brazed and welded vanes and fingers are proof-tested by applying a 5000-pound load to the spider body, so that approximately 310 pounds is applied to each vane. This proof load provides a bending moment at the spider body approximately equivalent to 1.4 times the load caused by the acceleration imposed by the control rod drive mechanism.
- Rods are checked for integrity by the applicable nondestructive methods described in Subsection 4.2.4.2.3.

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- To confirm proper fit with the fuel assembly, the rod cluster control, discrete burnable absorber, and source assemblies are installed in the fuel assembly and checked for binding in the dry condition.

The rod cluster control assemblies and gray rod cluster assemblies are also functionally tested, following core loading but prior to criticality, to demonstrate reliable operation of the assemblies. Each assembly is operated (and tripped) one time at full-flow/hot conditions. In addition, any assembly that has a drop time greater than a two sigma limit from the average rod drop time is subjected to additional rod drops to confirm drop time. Thus, each assembly is sufficiently tested to confirm proper functioning and operation.

To demonstrate continuous free movement of the rod cluster control assemblies, and gray rod cluster assemblies, and to provide acceptable core power distributions during operations, partial movement checks are performed as required by the technical specifications. In addition, periodic drop tests of the rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements.

If a rod cluster control assembly and/or gray rod cluster assembly cannot be moved by its mechanism, and is determined to be untrippable, adjustments in the boron concentration of the coolant provide that adequate shutdown margin will be achieved following a trip. Thus, inability to move one assembly can be tolerated until the reactor can be safely taken to Mode 3.

4.2.4.5 Tests and Inspections by Others

For tests and inspections performed by others, Westinghouse reviews and approves the quality control procedures, and inspection plans to be utilized to confirm that they are equivalent to the description provided in Subsections 4.2.4.1 through 4.2.4.4 and are performed properly to meet Westinghouse requirements.

4.2.4.6 Inservice Surveillance

As detailed in CENPD-404-P-A (Reference 27), significant 17x17 fuel assembly operating experience has been obtained. A surveillance program is expected to be established for the AP1000 for inspection of post-irradiated fuel assemblies. This surveillance program will establish the schedule, guidelines, and inspection criteria for conducting visual inspection of post-irradiated fuel assemblies and/or insert components. The surveillance program includes a visual examination of some discharged fuel assemblies from each refueling. This program also includes criteria for additional inspection requirements for post-irradiated fuel assemblies if unusual characteristics are noticed in the visual inspection or if plant instrumentation and subsequent laboratory analysis indicates gross failed fuel. The post-irradiated fuel surveillance program will address disposition of fuel assemblies and/or insert components receiving an unsatisfactory visual inspection. Those post-irradiated fuel assemblies receiving an unsatisfactory visual inspection are not reinserted into the core until a more detailed inspection and/or evaluation can be performed.

4.2.4.7 Onsite Inspection

Written procedures are used for the post-shipment inspection of the new fuel assemblies in addition to reactivity control and source components. Fuel handling procedures specify the sequence in which handling and inspection take place.

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Loaded fuel containers, when received onsite, are externally inspected to confirm that labels and markings are intact and security seals are unbroken. After the containers are opened, the shock indicators attached to the suspended internals are inspected to determine whether movement during transit exceeded design limitations.

Following removal of the fuel assembly from the container in accordance with detailed procedures, the fuel assembly plastic wrapper is examined for evidence of damage. The polyethylene wrapper is then removed, and a visual inspection of the entire fuel assembly is performed.

Control rod, gray rod, secondary source rod and discrete burnable absorber rod assemblies are usually shipped in fuel assemblies. They are inspected either prior to removal of the fuel assembly from the container or after the fuel assemblies are placed in the new fuel storage racks. The control rod assembly is withdrawn a few inches from the fuel assembly to confirm free and unrestricted movement, and the exposed section is visually inspected for mechanical integrity, replaced in the fuel assembly, and stored with the fuel assembly. Control rod, secondary source or discrete burnable absorber assemblies may be stored separately or within fuel assemblies in the new fuel storage area.

4.2.5 Combined License Information

Changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the DCD are addressed in APP-GW-GLR-059 (Reference 24).

4.2.6 References

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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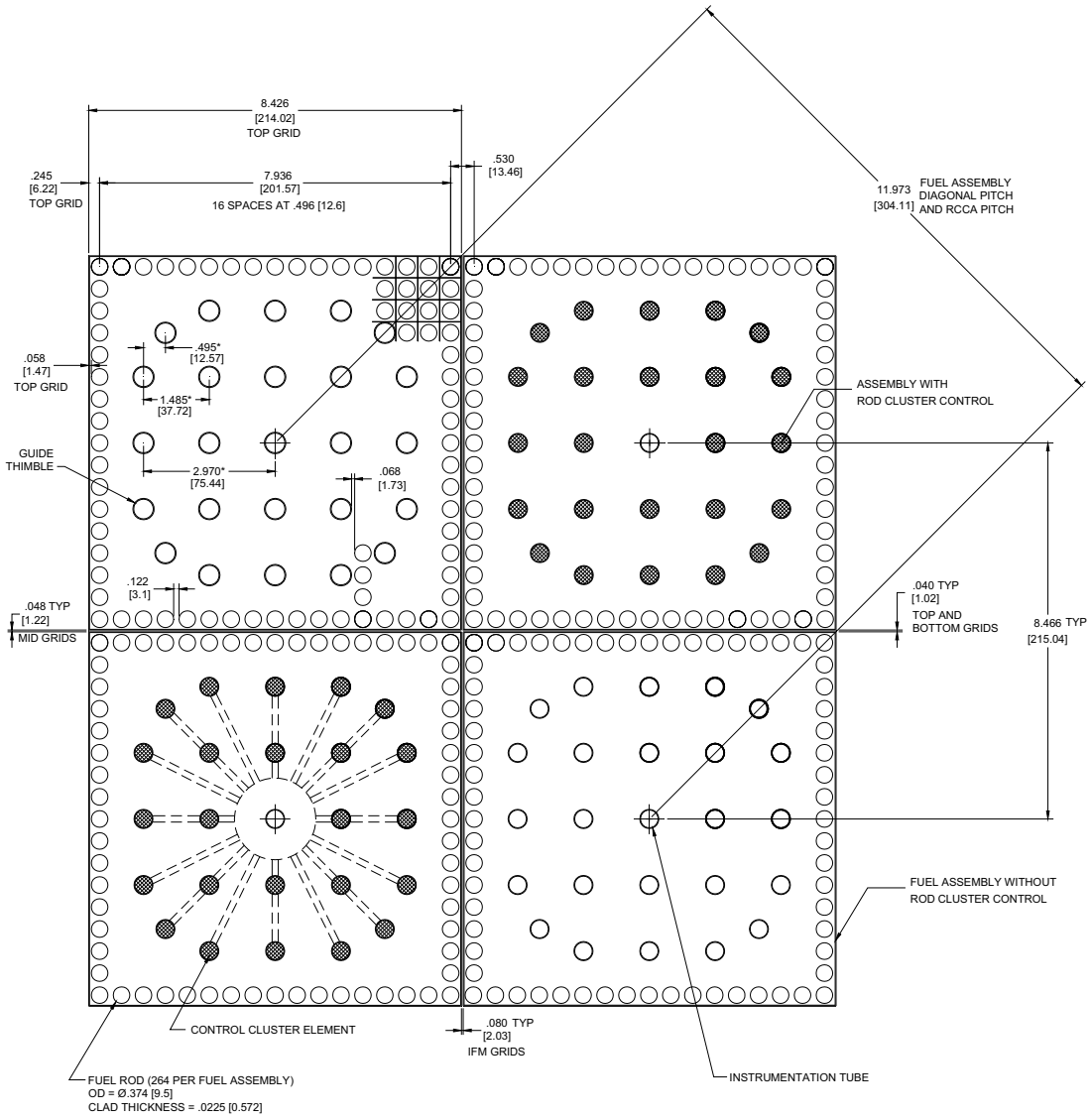

PRIMARY DIMENSIONS ARE IN INCHES (NOMINAL)
SECONDARY DIMENSIONS ARE IN MILLIMETERS

* GUIDE THIMBLE LOCATIONS
AT TOP NOZZLE ADAPTER PLATE

Figure 4.2-1
Fuel Assembly Cross-Section

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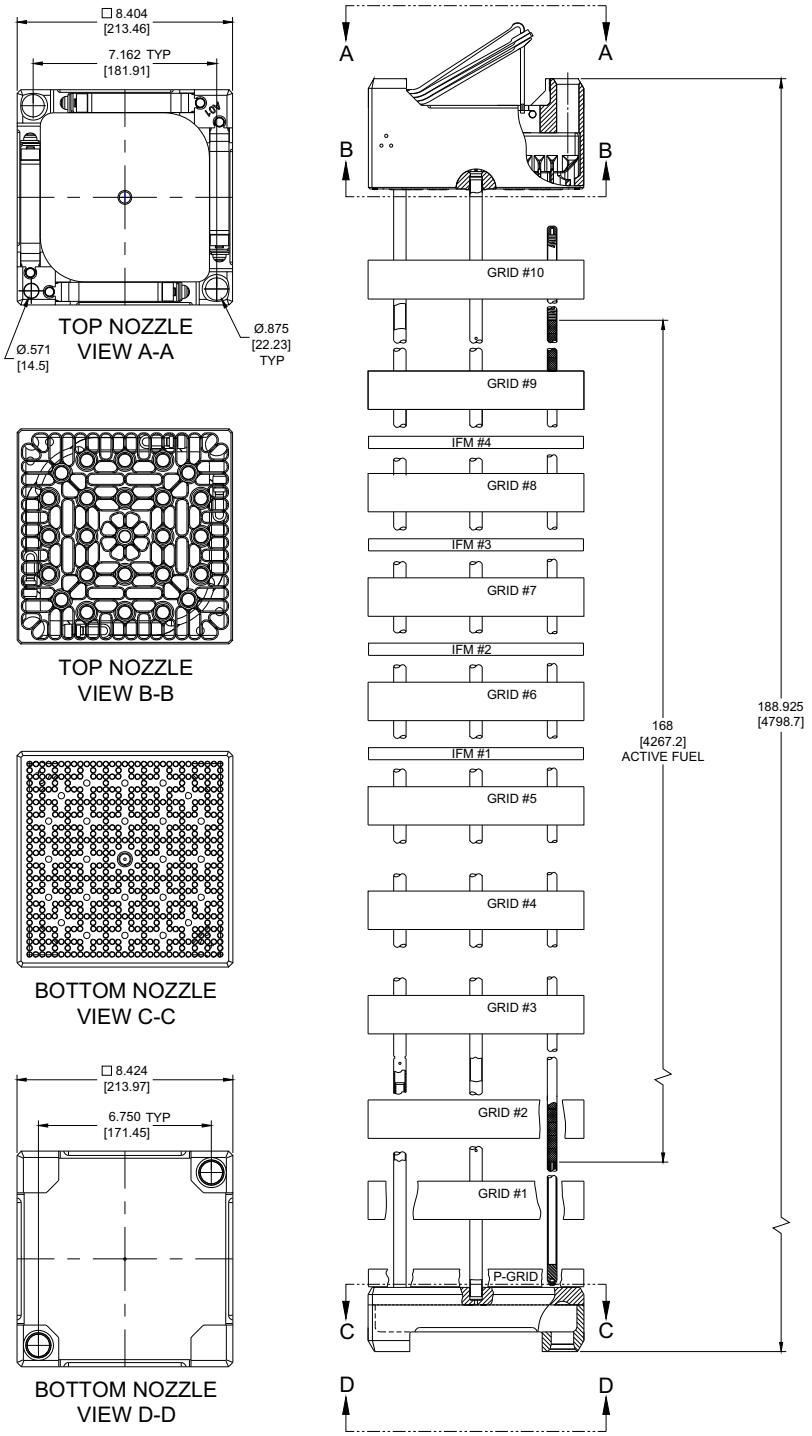

Figure 4.2-2
Fuel Assembly Outline

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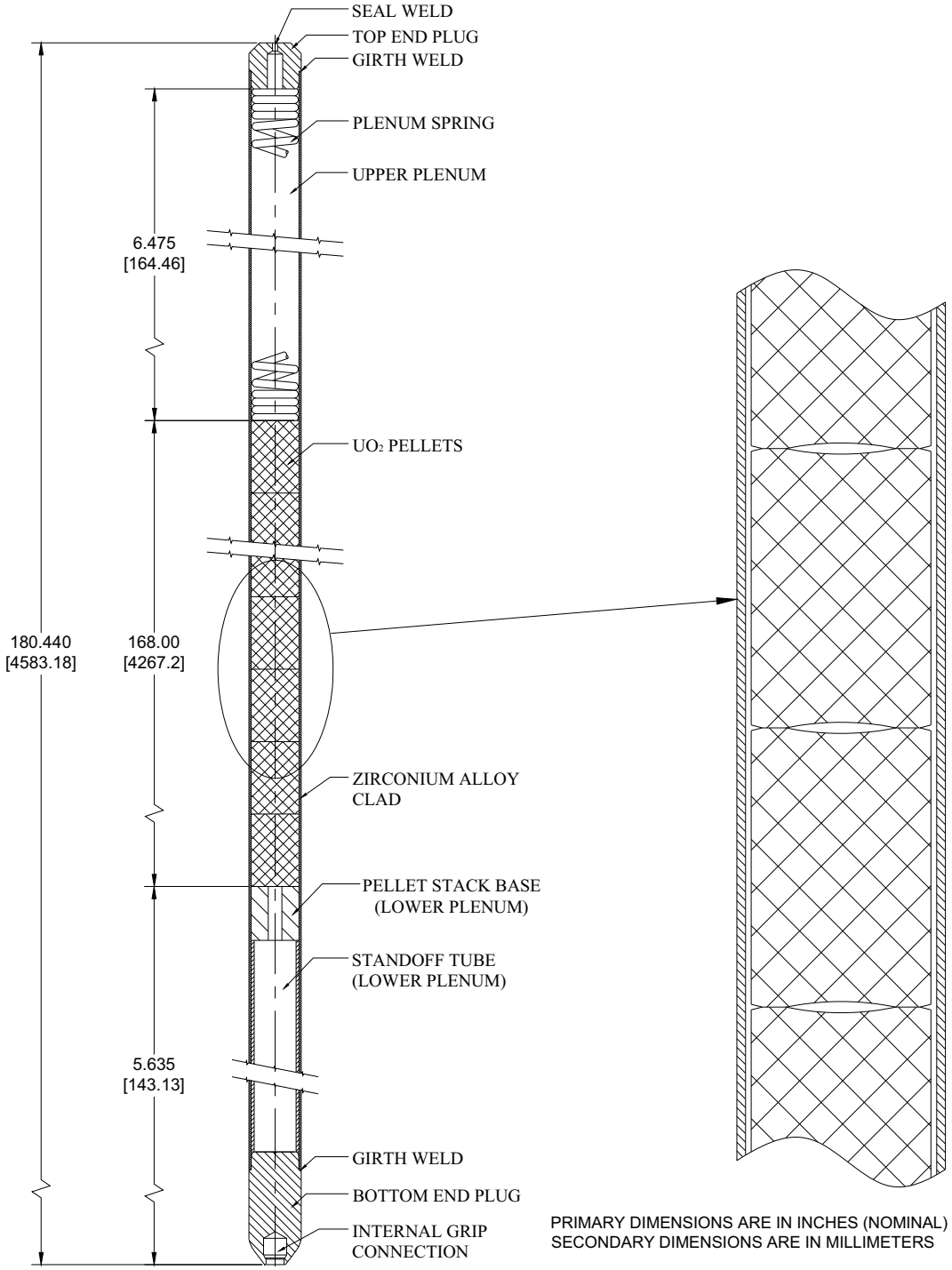

Figure 4.2-3
Fuel Rod Schematic

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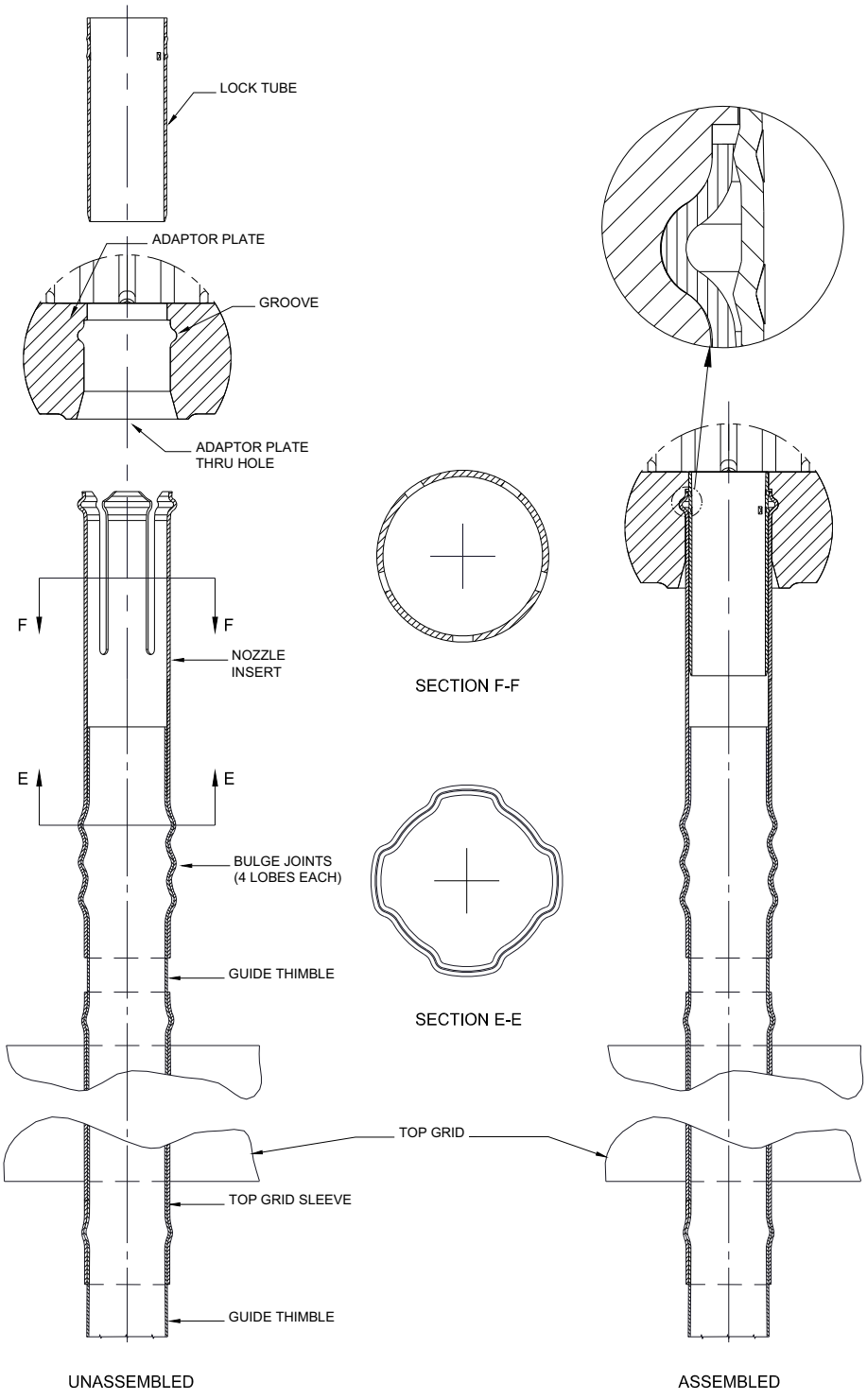

Figure 4.2-4
Top Grid Sleeve Detail

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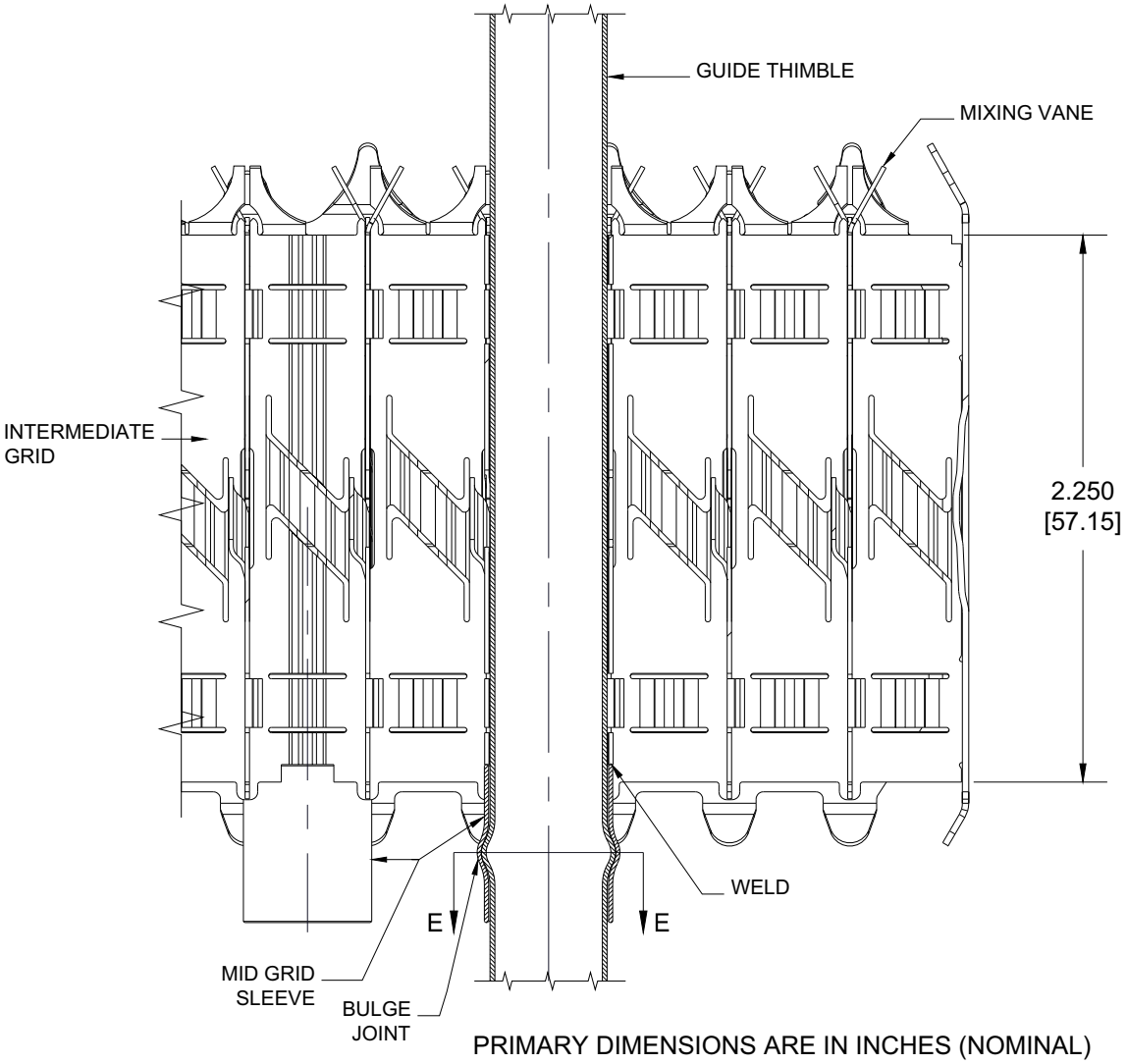

Figure 4.2-5
Intermediate Grid to Thimble Attachment Joint

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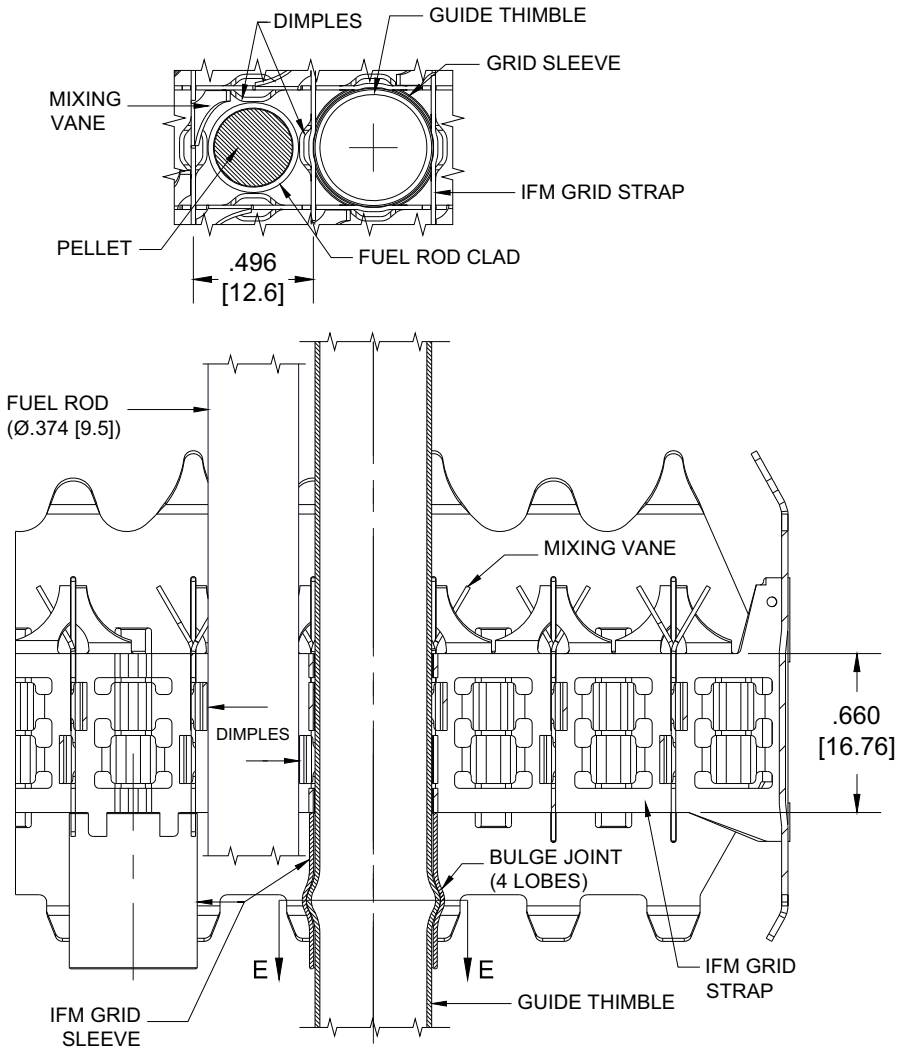

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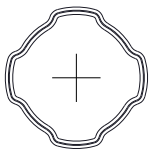
SECONDARY DIMENSIONS ARE IN MILLIMETERS



SECTION E-E
(TYPICAL)

Figure 4.2-6
Intermediate Flow Mixer
Grid to Thimble Attachment

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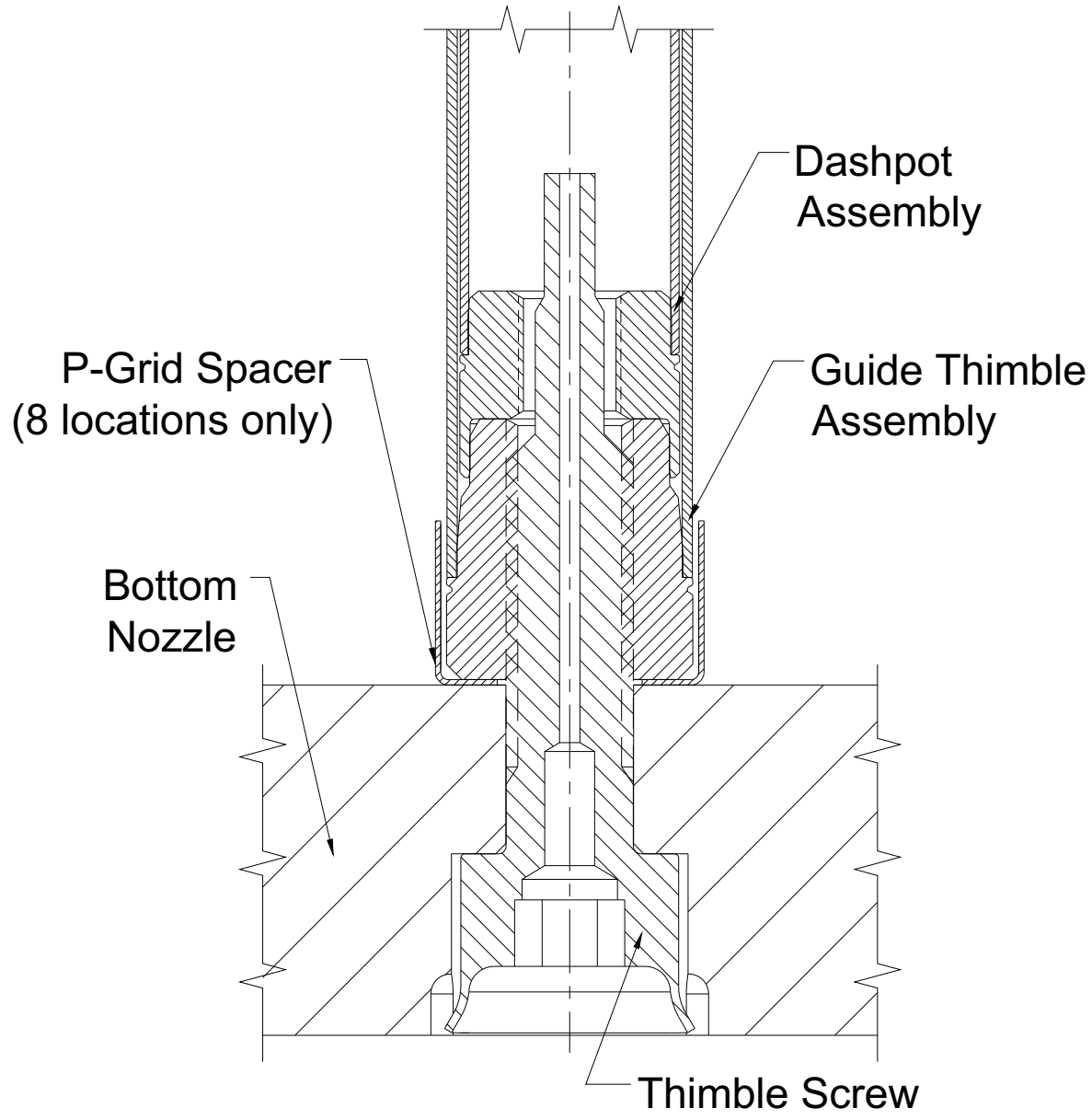

Figure 4.2-7
Grid Thimble to Bottom Nozzle Joint

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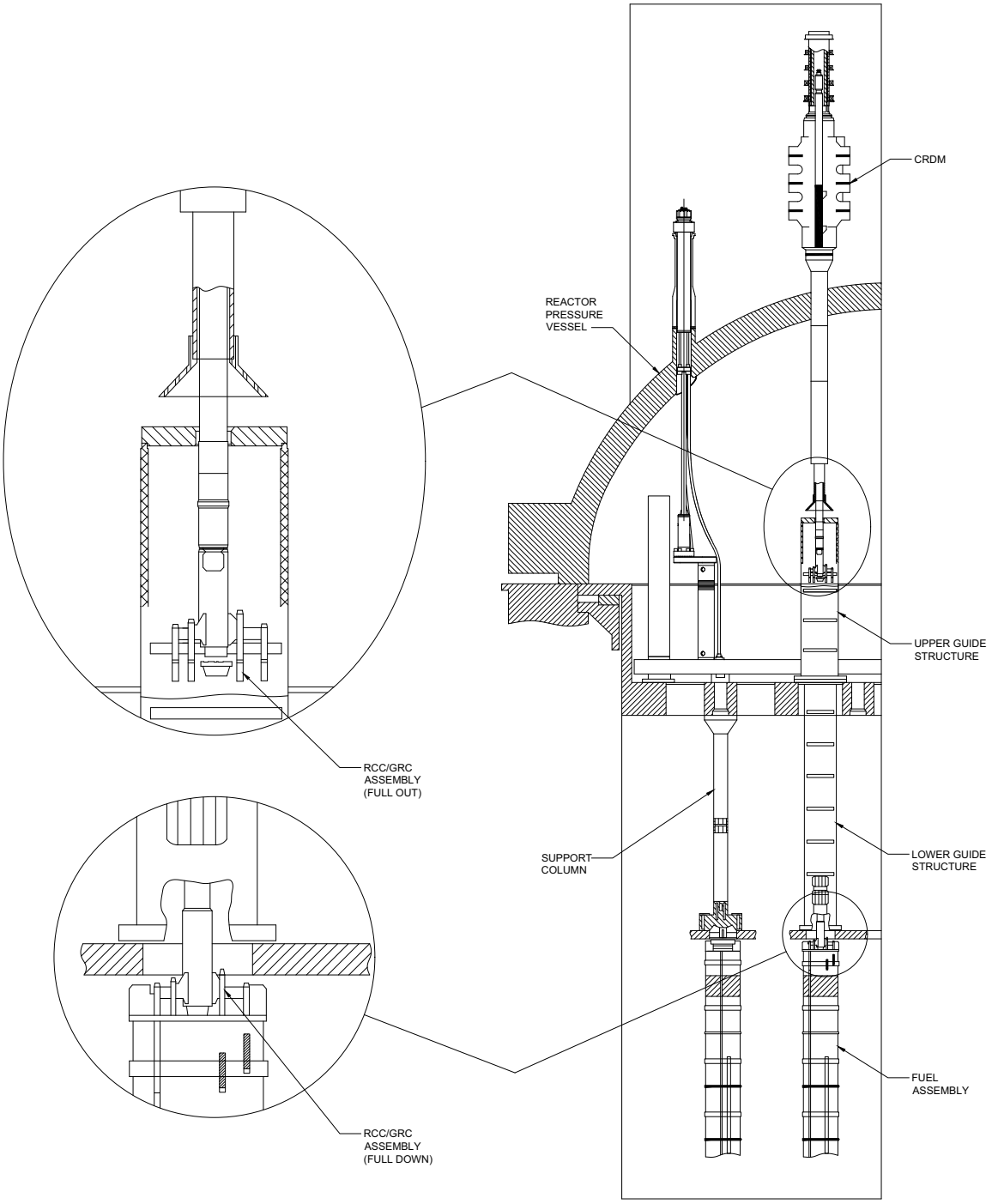

Figure 4.2-8
Rod Cluster Control and Drive Rod Assembly With Interfacing Components

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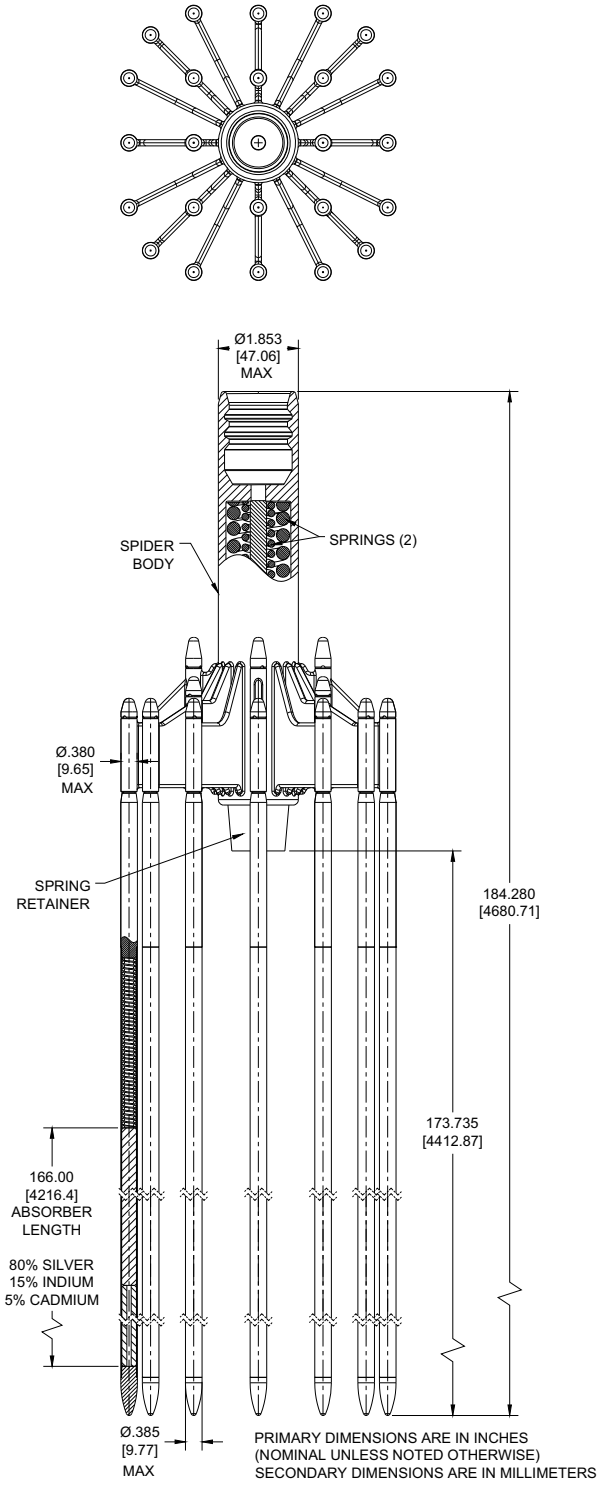

Figure 4.2-9
Rod Cluster Control Assembly

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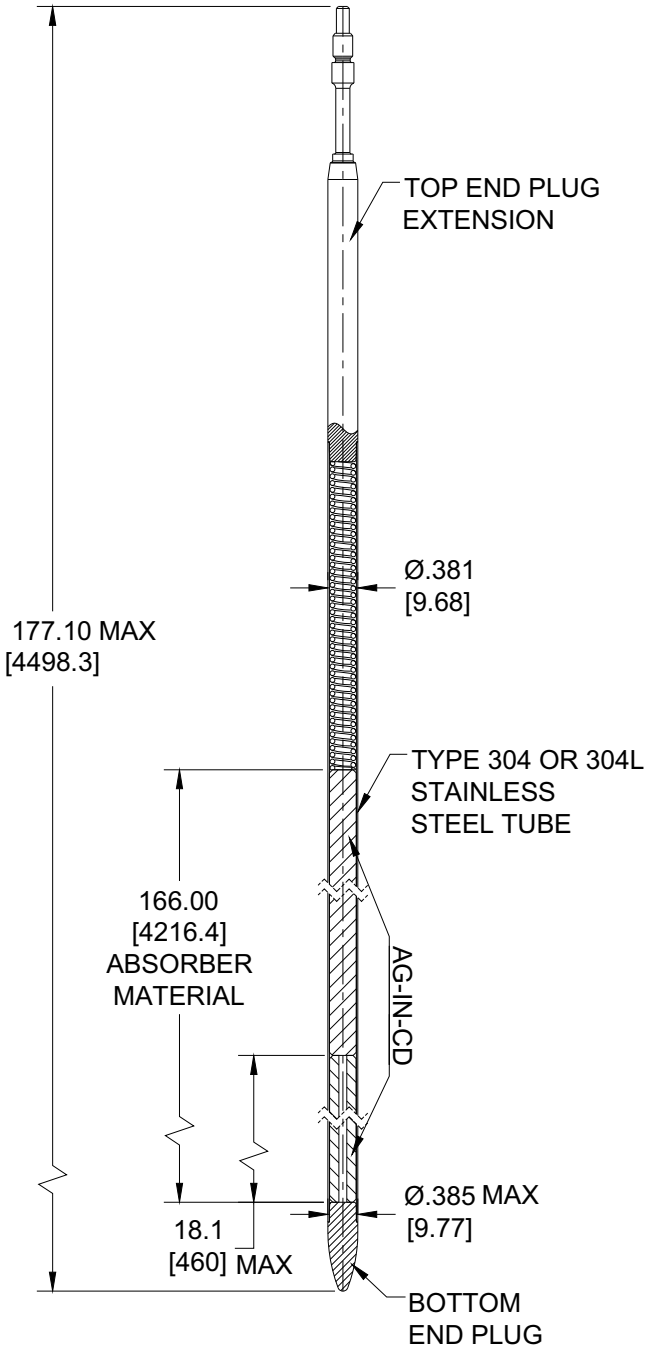

PRIMARY DIMENSIONS ARE IN INCHES
(NOMINAL UNLESS NOTED OTHERWISE)
SECONDARY DIMENSIONS ARE IN MILLIMETERS

Figure 4.2-10
Absorber Rod Detail

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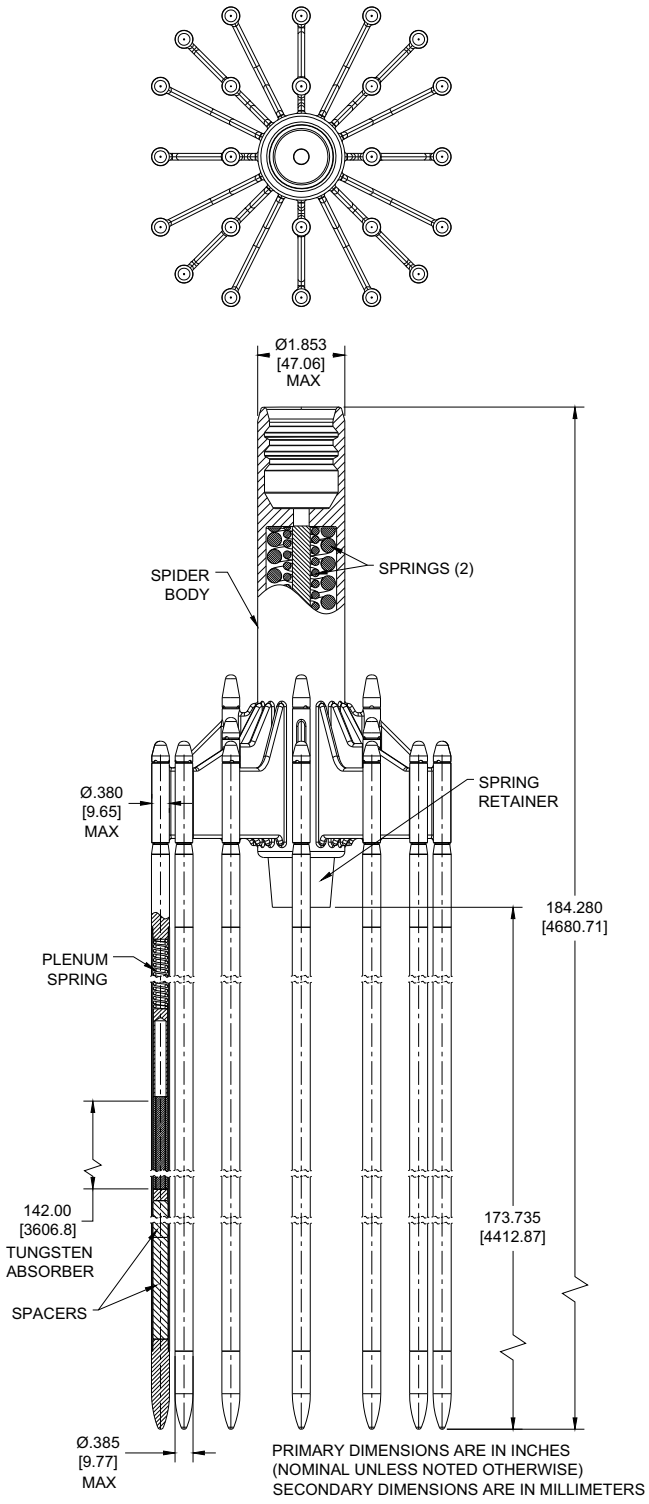

Figure 4.2-11
Gray Rod Cluster Assembly

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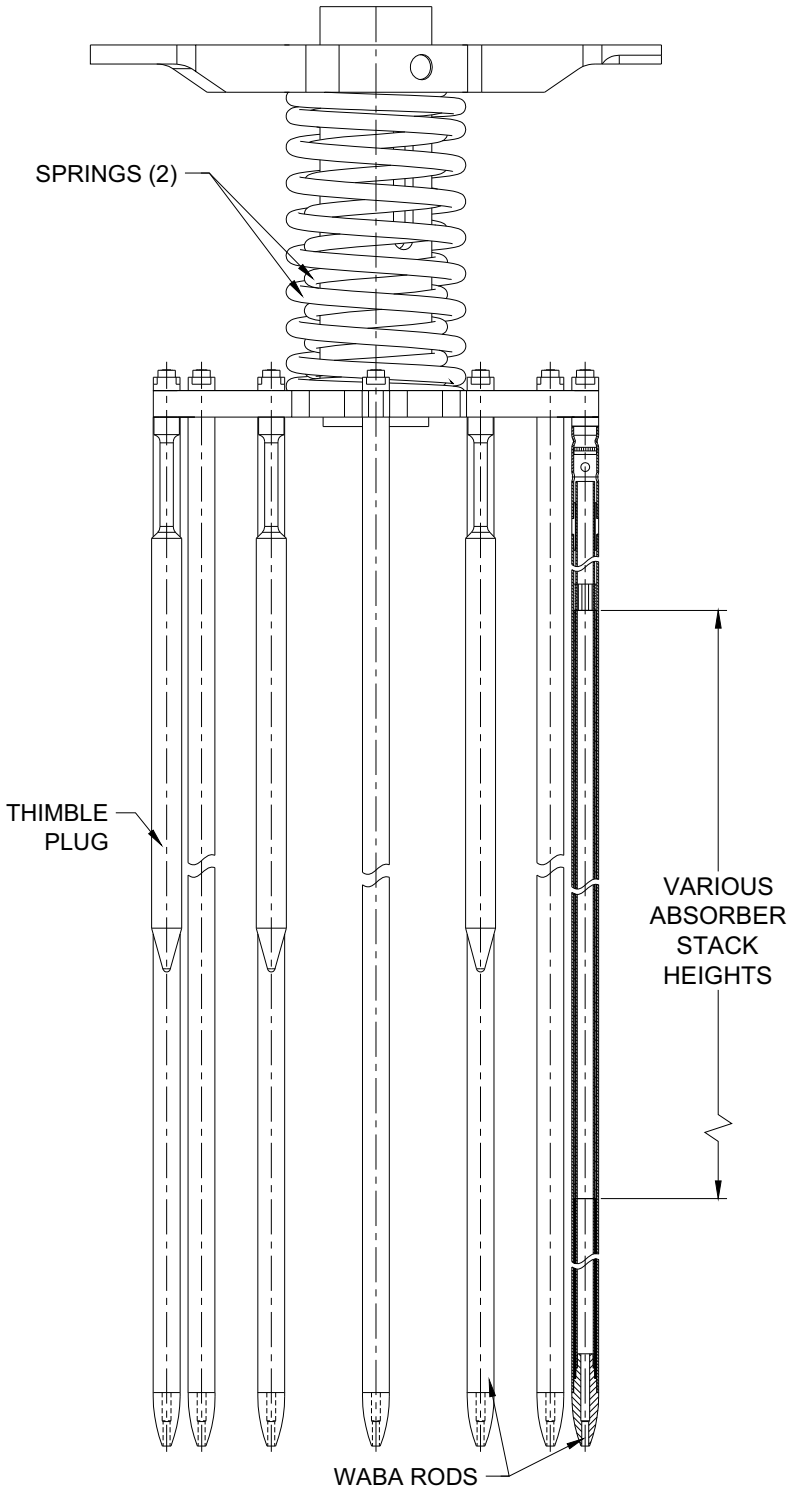

Figure 4.2-12
Wet Annular Burnable Absorber Assembly

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Figure 4.2-13 Not Used

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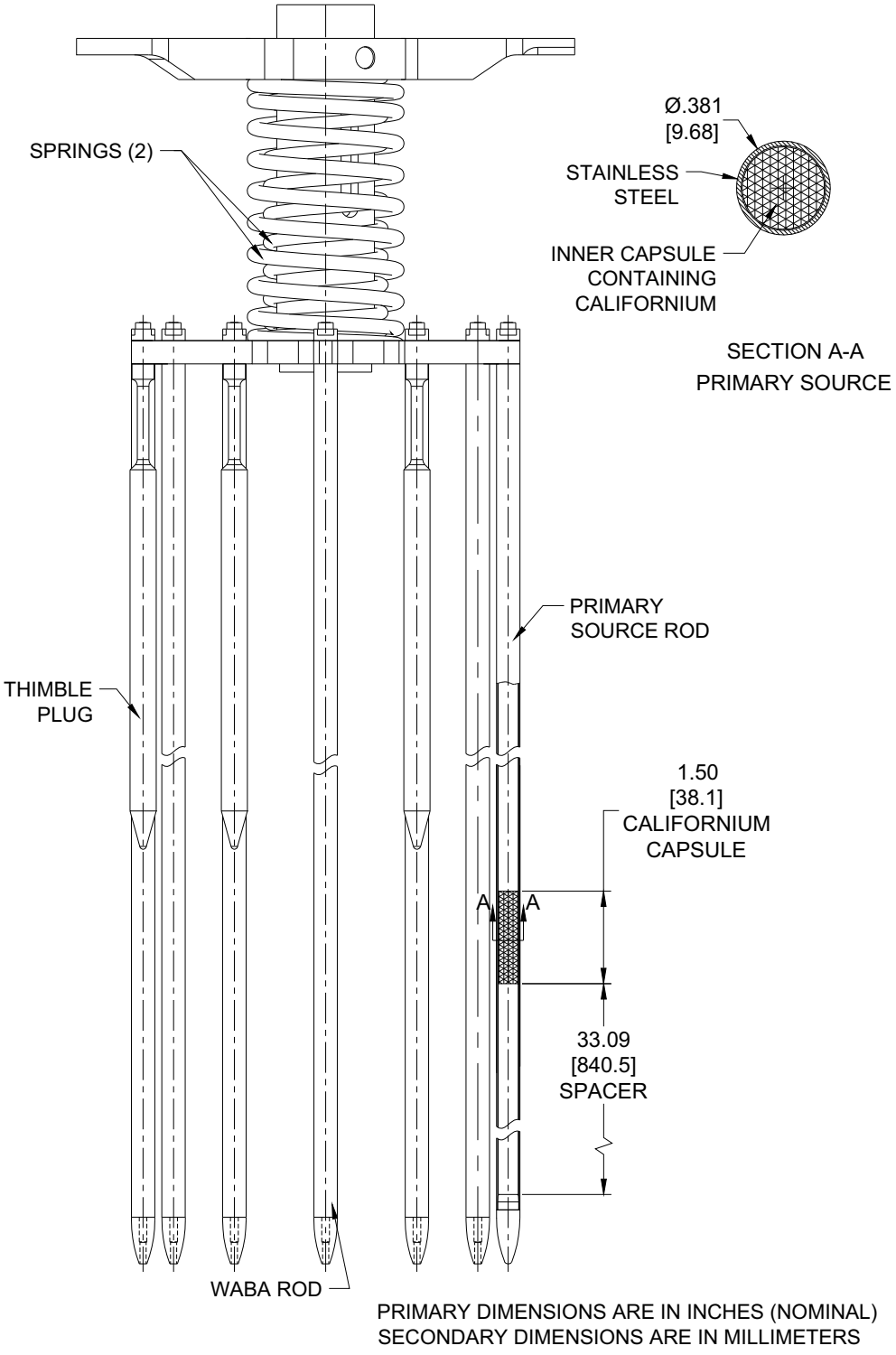

Figure 4.2-14
Primary Source Assembly

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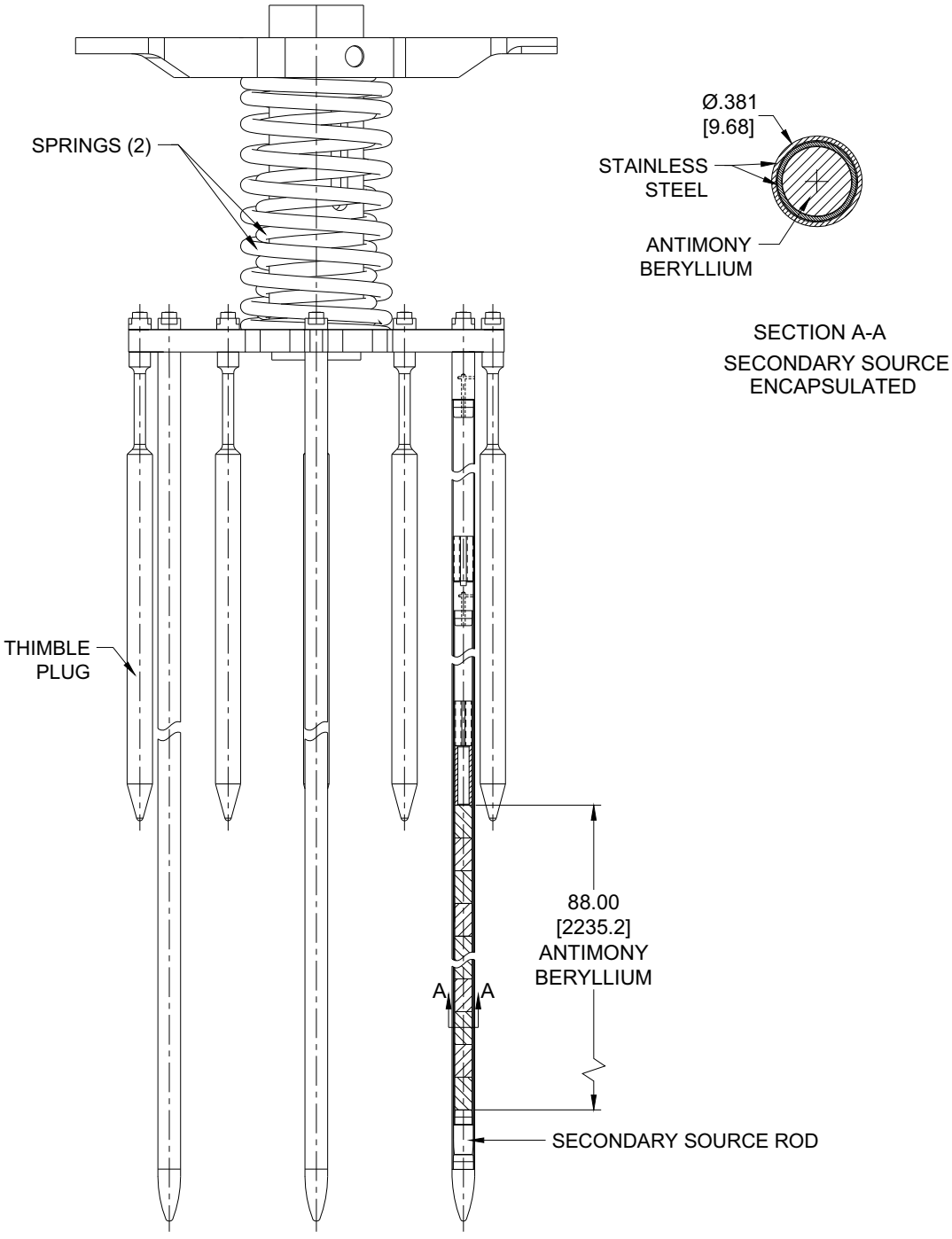

PRIMARY DIMENSIONS ARE IN INCHES (NOMINAL)
SECONDARY DIMENSIONS ARE IN MILLIMETERS

Figure 4.2-15
Secondary Source Assembly

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4.3 Nuclear Design

4.3.1 Design Basis

This section describes the design bases and functional requirements used in the nuclear design of the fuel and reactivity control system and relates these design bases to the General Design Criteria (GDC). The design bases are the fundamental criteria that must be met using approved analytical techniques. [*Enhancements to these techniques may be made provided that the changes are founded by NRC approved methodologies as discussed in*]* WCAP-9272-P-A (Reference 1) and [*WCAP-12488-P-A (Reference 2).*]*

The plant conditions for design are divided into four categories:

- Condition I — Normal operation and operational transients
- Condition II — Events of moderate frequency
- Condition III — Infrequent incidents
- Condition IV — Limiting faults

The reactor is designed so that its components meet the following performance and safety criteria:

- In general, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.
- Condition II occurrences are accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action.
- Fuel damage, that is, breach of fuel rod clad pressure boundary, is not expected during Condition I and Condition II occurrences. A very small amount of fuel damage may occur. This is within the capability of the chemical and volume control system (CVS) and is consistent with the plant design basis.
- Condition III occurrences do not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude immediate resumption of operation.
- The release of radioactive material due to Condition III occurrences is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary.
- A Condition III occurrence does not by itself generate a Condition IV occurrence or result in a consequential loss of function of the reactor coolant or reactor containment barriers.
- Condition IV faults do not cause a release of radioactive material that results in exceeding the dose limits identified in Chapter 15. Condition IV occurrences are faults that are not expected to occur but are defined as limiting faults which are included in the design.

The core design power distribution limits related to fuel integrity are met for Condition I occurrences through conservative design and are maintained by the action of the control system.

The requirements for Condition II occurrences are met by providing an adequate protection system which monitors reactor parameters.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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The control and protection systems are described in Chapter 7.

The consequences of Condition II, III, and IV occurrences are described in Chapter 15.

4.3.1.1 Fuel Burnup

4.3.1.1.1 Basis

A limitation on initial installed excess reactivity or average discharge burnup is not required other than as is quantified in terms of other design bases, such as overall negative power reactivity feedback discussed below. [*The NRC has approved, in WCAP-12488-P-A (Reference 2), maximum fuel rod average burnup of 60,000 MWD/MTU. Extended burnup to 62,000 MWD/MTU has been established in Reference 61.*]*

4.3.1.1.2 Discussion

Fuel burnup is a measure of fuel depletion which represents the integrated energy output of the fuel in megawatt-days per metric ton of uranium (MWD/MTU) and is a useful means for quantifying fuel exposure criteria.

The core design lifetime, or design discharge burnup, is achieved by installing sufficient initial excess reactivity in each fuel region and by following a fuel replacement program (such as that described in Subsection 4.3.2) that meets the safety-related criteria in each cycle of operation.

Initial excess reactivity installed in the fuel, although not a design basis, must be sufficient to maintain core criticality at full-power operating conditions throughout cycle life with equilibrium xenon, samarium, and other fission products present. Burnable absorbers, control rod insertion, and/or chemical shim are used to compensate for the excess reactivity. The end of design cycle life is defined to occur when the chemical shim concentration is essentially zero with control rods present to the degree necessary for operational requirements. In terms of soluble boron concentration, this corresponds to approximately 10 ppm with the control and gray rods essentially withdrawn.

4.3.1.2 Negative Reactivity Feedbacks (Reactivity Coefficients)

4.3.1.2.1 Basis

For the initial fuel cycle, the fuel temperature coefficient will be negative, and the moderator temperature coefficient of reactivity will be negative for power operating conditions, thereby providing negative reactivity feedback characteristics. The design basis meets General Design Criterion 11.

4.3.1.2.2 Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption (Doppler) effects associated with changing fuel temperature and the neutron spectrum and reactor composition change effects resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium results in a Doppler coefficient of reactivity that is negative. This coefficient provides the most rapid reactivity compensation. The initial core is also designed to have an overall negative moderator temperature coefficient of reactivity during power operation so that average coolant temperature changes or void content provides another, slower compensatory effect. For some core designs, if the compensation for excess reactivity is provided only by chemical shim, the moderator temperature coefficient could become positive. Nominal power operation is permitted only in a range of overall negative moderator temperature coefficient. The negative moderator temperature coefficient can be achieved through the use of discrete burnable absorbers (BAs) and/or

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integral fuel burnable absorbers and/or control rods by limiting the reactivity controlled by soluble boron.

Burnable absorber content (quantity and distribution) is not stated as a design basis. However, for some reloads, the use of burnable absorbers may be necessary for power distribution control and/or to achieve an acceptable moderator temperature coefficient throughout core life. The required burnable absorber loading is that which is required to meet design criteria.

4.3.1.3 Control of Power Distribution

4.3.1.3.1 Basis

The nuclear design basis is that, with at least a 95 percent confidence level:

- The fuel will not operate with a power distribution that would result in exceeding the departure from nucleate boiling (DNB) design basis (i.e., the departure from nucleate boiling ratio (DNBR) shall be greater than the design limit departure from nucleate boiling ratio as discussed in Subsection 4.4.1) under Condition I and II occurrences, including the maximum overpower condition.

- Under abnormal conditions, including the maximum overpower condition, the peak linear heat rate (PLHR) will not cause fuel melting, as defined in Subsection 4.4.1.2.

- Fuel management will be such as to produce values of fuel rod power and burnup consistent with the assumptions in the fuel rod mechanical integrity analysis of Section 4.2.

- The fuel will not be operated at Peak Linear Heat Rate (PLHR) values greater than those found to be acceptable within the body of the safety analysis under normal operating conditions, including an allowance of one percent for calorimetric error (calorimetric uncertainty calculation will be provided per Subsection 15.0.15.1).

The above basis meets General Design Criterion 10.

4.3.1.3.2 Discussion

Calculation of extreme power shapes which affect fuel design limits are performed with proven methods. The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state. Even though there is close agreement between calculated peak power and measurements, a nuclear uncertainty is applied (Subsection 4.3.2.2.1) to calculated power distribution. Such margins are provided both for the analysis for normal operating states and for anticipated transients.

4.3.1.4 Maximum Controlled Reactivity Insertion Rate

4.3.1.4.1 Basis

The maximum reactivity insertion rate due to withdrawal of rod cluster control assemblies (RCCAs) or gray rod cluster assemblies (GRCAs) or by boron dilution is limited by plant design, hardware, and basic physics. During normal power operation, the maximum controlled reactivity insertion rate is limited. The maximum reactivity change rate for accidental withdrawal of two control banks is set such that PLHR and the departure from nucleate boiling ratio limitations are not challenged. This satisfies General Design Criterion 25.

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The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to preclude rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity due to a rod withdrawal or an ejection accident. (See Chapter 15).

Following any Condition IV occurrence, such as rod ejection or steam line break, the reactor can be brought to the shutdown condition, and the core maintains acceptable heat transfer geometry. This satisfies General Design Criterion 28.

4.3.1.4.2 Discussion

Reactivity addition associated with an accidental withdrawal of a control bank (or banks) is limited by the maximum rod speed (or travel rate) and by the worth of the bank(s). For this reactor, the maximum control and gray rod speed is 45 inches per minute.

The reactivity change rates are conservatively calculated, assuming unfavorable axial power and xenon distributions. The typical peak xenon burnout rate is significantly lower than the maximum reactivity addition rate for normal operation and for accidental withdrawal of two banks.

4.3.1.5 Shutdown Margins

4.3.1.5.1 Basis

Minimum shutdown margin as specified in the technical specifications is required in all operating modes.

In analyses involving reactor trip, the single, highest worth rod cluster control assembly is postulated to remain untripped in its full-out position (stuck rod criterion). This satisfies General Design Criterion 26.

4.3.1.5.2 Discussion

Two independent reactivity control systems are provided: control rods and soluble boron in the coolant. The control rods provide reactivity changes which compensate for the reactivity effects of the fuel and water density changes accompanying power level changes over the range from full load to no load. The control rods provide the minimum shutdown margin under Condition I occurrences and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits (very small number of rod failures), assuming that the highest worth control rod is stuck out upon trip.

The boron system can compensate for xenon burnout reactivity changes and maintain the reactor in the cold shutdown condition. Thus, backup and emergency shutdown provisions are provided by mechanical and chemical shim control systems which satisfy General Design Criterion 26. Reactivity changes due to fuel depletion are accommodated with the boron system.

4.3.1.5.3 Basis

When fuel assemblies are in the pressure vessel and the vessel head is not in place, keff will be maintained at or below 0.95 with control rods and soluble boron. Further, the fuel will be maintained sufficiently subcritical that removal of the rod cluster control assemblies will not result in criticality.

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4.3.1.5.4 Discussion

ANSI N18.2 (Reference 3) specifies a keff not to exceed 0.95 in spent fuel storage racks and transfer equipment flooded with pure water and a keff not to exceed 0.98 in normally dry new fuel storage racks, assuming optimum moderation. No criterion is given for the refueling operation. However, a five percent margin, which is consistent with spent fuel storage and transfer and the new fuel storage, is adequate for the controlled and continuously monitored operations involved.

The boron concentration required to meet the refueling shutdown criteria is specified in the Core Operating Limits Report (COLR). Verification that these shutdown criteria are met, including uncertainties, is achieved using standard design methods. The subcriticality of the core is continuously monitored as described in the technical specifications.

4.3.1.6 Stability

4.3.1.6.1 Basis

The core will be inherently stable to power oscillations at the fundamental mode. This satisfies General Design Criterion 12.

Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

4.3.1.6.2 Discussion

Oscillations of the total power output of the core, from whatever cause, are readily detected by the loop temperature sensors and by the nuclear instrumentation. The core is protected by these systems; a reactor trip occurs if power increases unacceptably, thereby preserving the design margins to fuel design limits. The combined stability of the turbine, steam generator and the reactor power control systems are such that total core power oscillations are not normally possible. The redundancy of the protection circuits results in a low probability of exceeding design power levels.

The core is designed so that diametral and azimuthal oscillations due to spatial xenon effects are self-damping; no operator action or control action is required to suppress them. The stability to diametral oscillations is so great that this excitation is highly improbable. Convergent azimuthal oscillations can be excited by prohibited motion of individual control rods.

Indications of power distribution anomalies are continuously available from an online core monitoring system. The online monitoring system processes information provided by the fixed in-core detectors, in-core thermocouples, and loop temperature measurements. Radial power distributions are therefore continuously monitored, thus power oscillations are readily observable and alarmed. The ex-core long ion chambers also provide surveillance and alarms of anomalous power distributions. In proposed core designs, these horizontal plane oscillations are self-damping by virtue of reactivity feedback effects inherent to the basic core physics.

Axial xenon spatial power oscillations may occur during core life, especially late in the cycle. The online core monitoring system provides continuous surveillance of the axial power distributions. The control rod system provides both manual and automatic control systems for controlling the axial power distributions.

Confidence that fuel design limits are not exceeded is provided by reactor protection system overpower ΔT ($OP\Delta T$) and overtemperature ΔT ($OT\Delta T$) trip functions, which use the loop temperature sensors, pressurizer pressure indication, and measured axial offset as an input. Detection and suppression of xenon oscillations are discussed in Subsection 4.3.2.7.

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4.3.1.7 Anticipated Transients Without Scram (ATWS)

The AP1000 diverse reactor trip actuation system is independent of the reactor trip breakers used by the protection monitoring system. The diverse reactor trip reduces the probability and consequences of a postulated ATWS. The effects of anticipated transients with failure to trip are not considered in the design bases of the plant. Analysis has shown that the likelihood of such a hypothetical event is negligibly small. Furthermore, analysis of the consequences of a hypothetical failure to trip following anticipated transients has shown that no significant core damage would result, system peak pressures should be limited to acceptable values, and no failure of the reactor coolant system would result. (See WCAP-8330, Reference 5). The process used to evaluate the ATWS risk in compliance with 10 CFR 50.62 is described in Section 15.8.

4.3.2 Description

4.3.2.1 Nuclear Design Description

The reactor core consists of a specified number of fuel rods held in bundles by spacer grids and top and bottom fittings. The fuel rods are fabricated from cylindrical tubes made of zirconium based alloy(s) containing uranium dioxide fuel pellets. The bundles, known as fuel assemblies, are arranged in a pattern which approximates a right circular cylinder.

Each fuel assembly contains a 17 x 17 rod array composed nominally of 264 fuel rods, 24 rod cluster control thimbles, and an in-core instrumentation thimble. Figure 4.2-1 shows a cross-sectional view of a 17 x 17 fuel assembly and the related rod cluster control guide thimble locations. Detailed descriptions of the AP1000 fuel assembly design features are given in Section 4.2.

Both the initial and reload core loading patterns can employ various fuel management techniques including “low-leakage” designs, and are anticipated to operate approximately 18 months between refueling. For reload core loading patterns, the initial and final positions of assemblies, and the number of fresh assemblies and their placement are dependent on the energy requirement for the reload cycle and burnup and power histories of the previous cycles.

For the initial core loading, the fuel rods within certain assemblies contain varying uranium enrichments in both the radial and axial planes. Fuel containing up to five average enrichments will be used in the initial core load to establish a favorable radial power distribution simulating the reactivity distribution of a low leakage reload core. Figure 4.3-1 shows the fuel loading pattern used in the initial cycle. The higher enriched regions will be configured in the core interior consistent with the feed fuel placement in a reload core, and the lower enriched regions will approximate the reactivity of the burned fuel assemblies of a reload core. The enrichments for the initial cycle are shown in Table 4.3-1.

The core average enrichment is determined by the amount of fissionable material required to provide the desired energy requirements. The physics of the burnout process is such that operation of the reactor depletes the amount of fuel available due to the absorption of neutrons by the U-235 atoms and their subsequent fission. In addition, the fission process results in the formation of fission products, some of which readily absorb neutrons. These effects, the depletion and the buildup of fission products, are partially offset by the buildup of plutonium shown in Figure 4.3-2 for a typical 17 x 17 fuel assembly, which occurs due to the parasitic absorption of neutrons in U-238. Therefore, at the beginning of any cycle a reactivity reserve equal to the depletion of the fissionable fuel and the buildup of fission product poisons less the buildup of fissile fuel over the specified cycle life is built into the reactor. This excess reactivity is controlled by removable neutron-absorbing material in the form of boron dissolved in the primary coolant, control rod insertion, burnable absorber rods, and/or integral fuel burnable absorbers (IFBA). The stack length of the burnable absorber rods and/or integral absorber bearing fuel may vary for different core designs, with the optimum length

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determined on a design specific basis. Figure 4.3-3 is a plot of the core soluble boron concentration versus core depletion for the first operating cycle.

The concentration of the soluble neutron absorber is varied to compensate for reactivity changes due to fuel burnup, fission product poisoning including xenon and samarium, burnable absorber depletion, and the cold-to-operating moderator temperature change. Throughout the operating range, the CVS is designed to provide changes in reactor coolant system (RCS) boron concentration to compensate for the reactivity effects of fuel depletion, peak xenon burnout and decay, and cold shutdown boration requirements.

Burnable absorbers are strategically located to provide a favorable radial power distribution and provide for negative reactivity feedback. Figures 4.3-4a and 4.3-4a show the burnable absorber distributions within a fuel assembly for the several patterns used in a 17 x 17 array. The initial core burnable absorber loading pattern is shown in Figure 4.3-5.

Tables 4.3-1 through 4.3-3 contain summaries of reactor core design parameters including reactivity coefficients, delayed neutron fraction, and neutron lifetimes. Sufficient information is included to permit an independent calculation of the nuclear performance characteristics of the core.

4.3.2.2 Power Distribution

The accuracy of power distribution calculations has been confirmed through approximately 1000 flux maps under conditions very similar to those expected. Details of this confirmation are given in WCAP-7308-L-P-A (Reference 7) and in Subsection 4.3.2.2.7.

4.3.2.2.1 Definitions

Relative power distributions within the reactor are quantified in terms of hot channel factors. These hot channel factors are normalized ratios of maximal absolute power generation rates and are a measure of the peak pellet power within the reactor core relative to the average pellet (F_Q) and the energy produced in a coolant channel relative to the core average channel ($F_{\Delta H}$). Absolute power generation rates are expressed in terms of quantities related to the nuclear or thermal design; more specifically, volumetric power density (qvol) is the thermal power produced per unit volume of the core (kW/liter).

Linear heat rate (LHR) is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, LHR is the unit of absolute power density most commonly used. For practical purposes, LHR differs from qvol by a constant factor which includes geometry effects and the heat flux deposition fraction. The peak linear heat rate (PLHR) is defined as the maximum linear heat rate occurring throughout the reactor. PLHR directly impacts fuel temperatures and decay power levels thus being a significant safety analysis parameter.

Average linear heat rate (ALHR) is the total thermal power produced in the fuel rods expressed as heat flux divided by the total active fuel length of the rods in the core.

Local heat flux is the heat flux at the surface of the cladding (Btu/hr-ft^2). For nominal rod parameters, this differs from linear heat rate by a constant factor.

Rod power is the total power generated in one rod (kW).

Average rod power is the total thermal power produced in the fuel rods divided by the number of fuel rods (assuming the rods have equal length).

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The hot channel factors used in the discussion of power distributions in this section are defined as follows:

F_Q, **heat flux hot channel factor**, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F^N_Q, **nuclear heat flux hot channel factor**, is defined as the maximum local fuel rod linear heat rate divided by the average fuel rod linear heat rate, assuming nominal fuel pellet and rod parameters.

F^E_Q, **engineering heat flux hot channel factor**, is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, burnable absorber content, surface area of the fuel rod, and eccentricity of the gap between pellet and clad. Combined statistically, the net effect is a factor of 1.03 to be applied to the fuel rod surface heat flux.

$F^N_{\Delta H}$, **nuclear enthalpy rise hot channel factor**, is defined as the ratio of the maximum integrated rod power within the core to the average rod power.

Manufacturing tolerances, hot channel power distribution, and surrounding channel power distributions are treated explicitly in the calculation of the departure from nucleate boiling ratio described in Section 4.4.

It is convenient for the purposes of discussion to define subfactors of F_Q. However, design limits are set in terms of the total peaking factor.

F_Q = total peaking factor or heat flux hot channel factor = $\frac{PLHR}{ALHR}$

Without densification effects:

$$F_Q = F^N_Q \times F^E_Q = F^N_{XY} \times F^N_Z \times F^N_U \times F^E_Q$$

where F^N_Q and F^E_Q are defined above and:

F^N_U = factor for calculational uncertainty, assumed to be 1.05.

F^N_{XY} = ratio of peak power density to average power density in the horizontal plane of peak local power.

F^N_Z = ratio of the power per unit core height in the horizontal plane of peak local power to the average value of power per unit core height. If the plane of peak local power coincides with the plane of maximum power per unit core height, then F^N_Z is the core average axial peaking factor.

4.3.2.2.2 Radial Power Distributions

The power shape in horizontal sections of the core at full power is a function of the fuel assembly and burnable absorber loading patterns, the control rod pattern, and the fuel burnup distribution. Thus, at any time in the cycle, a horizontal section of the core can be characterized as unrodded or with control rods. These two situations combined with burnup effects determine the radial power shapes which can exist in the core at full power. Typical first cycle values of $F^N_{\Delta H}$, the nuclear enthalpy rise hot channel factors from beginning of life (BOL) to end of life (EOL) are given in Table 4.3-2. The effects on radial power shapes of power level, xenon, samarium, and moderator density effects are

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also considered, but these are quite small. The effect of nonuniform flow distribution is negligible. While radial power distributions in various planes of the core are often illustrated, since the moderator density is directly proportional to enthalpy, the core radial enthalpy rise distribution, as determined by the integral of power up each channel, is of greater interest. Figures 4.3-6 through 4.3-11 show typical normalized power density distributions for one-quarter of the core for representative operating conditions. These conditions are as follows:

- Hot full power (HFP) near beginning of life, unrodded, no xenon
- Hot full power near beginning of life, unrodded, equilibrium xenon
- Hot full power near beginning of life, gray bank MA+MB in, equilibrium xenon
- Hot full power near middle of life (MOL), unrodded equilibrium xenon
- Hot full power near end of life, unrodded, equilibrium xenon
- Hot full power near end of life, gray bank MA+MB in, equilibrium xenon

Since the position of the hot channel varies from time to time, a single-reference radial design power distribution is selected for departure from nucleate boiling calculations. This reference power distribution is chosen conservatively to concentrate power in one area of the core, minimizing the benefits of flow redistribution. Assembly powers are normalized to core average power. The radial power distribution within a fuel rod and its variation with burnup as utilized in thermal calculations and fuel rod design are discussed in Section 4.4.

4.3.2.2.3 Assembly Power Distributions

For the purpose of illustration, typical rodwise power distributions from the beginning of life and end of life conditions corresponding to Figures 4.3-7 and 4.3-10, are given for representative hot channel assemblies in Figures 4.3-12 and 4.3-13, respectively.

Since the detailed power distribution surrounding the hot channel varies from time to time, a conservatively flat radial assembly power distribution is assumed in the departure from nucleate boiling analysis, described in Section 4.4, with the rod of maximum integrated power artificially raised to the design value of $F^N_{\Delta H}$. Care is taken in the nuclear design of the fuel cycles and operating conditions to confirm that a flatter assembly power distribution does not occur with limiting values of $F^N_{\Delta H}$.

4.3.2.2.4 Axial Power Distributions

The distribution of power in the axial or vertical direction is largely under the control of the operator through either the manual operation of the control rods or the automatic motion of control rods in conjunction with manual operation of the chemical and volume control system. The automated mode of operation is referred to as mechanical shim (MSHIM) and is discussed in Subsection 4.3.2.4.16. The rod control system automatically modulates the insertion of the axial offset (AO) control bank controlling the axial power distribution simultaneous with the MSHIM gray and control rod banks to maintain programmed coolant temperature. Operation of the chemical and volume control system is initiated manually by the operator to compensate for fuel burnup and maintain the desired MSHIM bank insertion. Nuclear effects which cause variations in the axial power shape include moderator density, Doppler effect on resonance absorption, spatial distribution of xenon, burnup, and axial distribution of fuel enrichment and burnable absorber. Automatically controlled variations in total power output and rod motion are also important in determining the axial power shape at any time.

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The online core monitoring system provides the operator with detailed power distribution information in both the radial and axial sense continuously using signals from the fixed in-core detectors. Signals are also available to the operator from the ex-core ion chambers, which are long ion chambers outside the reactor vessel running parallel to the axis of the core. Separate signals are taken from each ion chamber. The ion chamber signals are processed and calibrated against in-core measurements such that an indication of the power in the top of the core less the power in the bottom of the core is derived. The calibrated difference in power between the core top and bottom halves, called the flux difference (ΔI), is derived for each of the four channels of ex-core detectors and is displayed on the control panel. The principal use of the flux difference is to provide the shape penalty function to the OTΔT DNB protection and the OPΔT overpower protection.

4.3.2.2.5 Local Power Peaking

Fuel densification occurred early in the evolution of pressurized water reactor fuel manufacture under irradiation in several operating reactors. This caused the fuel pellets to shrink both axially and radially. The pellet shrinkage combined with random hang-up of fuel pellets can result in gaps in the fuel column when the pellets below the hung-up pellet settle in the fuel rod. The gaps vary in length and location in the fuel rod. Because of decreased neutron absorption in the vicinity of the gap, power peaking occurs in the adjacent fuel rods, resulting in an increased power peaking factor. A quantitative measure of this local peaking is given by the power spike factor S(Z), where Z is the axial location in the core. The power spike factor S(z) is discussed in References 8, 9, and 10.

Modern PWR fuel manufacturing practices have essentially eliminated significant fuel densification impacts on reactor design and operation. It has since been concluded and accepted that a densification power spike factor of 1.0 is appropriate for Westinghouse fuel as described in WCAP-13589-A (Reference 59).

4.3.2.2.6 Limiting Power Distributions

According to the ANSI classification of plant conditions (Chapter 15), Condition I occurrences are those expected frequently or regularly in the course of power operation, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Condition I occurrences are considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). Analysis of each fault condition described is based on a conservative set of corresponding initial conditions.

The list of steady-state and shutdown conditions, permissible deviations, and operational transients is given in Chapter 15. Implicit in the definition of normal operation is proper and timely action by the reactor operator; that is, the operator follows recommended operating procedures for maintaining appropriate power distributions and takes any necessary remedial actions when alerted to do so by the plant instrumentation.

The online monitoring system evaluates the consequences of limiting power distributions based upon the conditions prevalent in the reactor at the current time. Operating space evaluations performed by the online monitoring system include the most limiting power distributions that can be generated by inappropriate operator or control system actions given the current core power level, xenon distribution, MSHIM or AO bank insertion and core burnup. Thus, as stated, the worst or limiting power distribution which can occur during normal operation is considered as the starting point for analysis of Conditions II, III, and IV occurrences.

Improper procedural actions or errors by the operator are assumed in the design as occurrences of moderate frequency (Condition II). Some of the consequences which might result are discussed in Chapter 15. Therefore, the limiting power shapes which result from such Condition II occurrences are

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those power distributions which deviate from the normal operating condition within the allowable operating space as defined in the core operating limits; e.g., due to lack of proper action by the operator during a xenon transient following a change in power level brought about by control rod motion. Power distributions which fall in this category are used for determination of the reactor protection system setpoints to maintain margin to overpower or departure from nucleate boiling limits.

The means for maintaining power distributions within the required absolute power generation limits are described in the technical specifications. The online core monitoring system provides the operator with the current allowable operating space, detailed current power distribution information, thermal margin assessment and operational recommendations to manage and maintain required thermal margins. As such, the online monitoring system provides the primary means of managing and maintaining required operating thermal margins during normal operation.

In the unlikely event that the online monitoring system is out of service, power distribution controls based on bounding, precalculated analysis are also provided to the operator such that the online monitoring system is not a required element for short term reactor operation. Limits are placed on the axial flux difference so that the heat flux hot channel factor F_Q is maintained within acceptable limits. A discussion of precalculated power distribution control in Westinghouse pressurized water reactors (PWRs) is included in WCAP-7811 (Reference 11). Detailed background information on the design constraints on local power density in a Westinghouse PWR, on the defined operating procedures, and on the measures taken to preclude exceeding design limits is presented in the Westinghouse topical report on power distribution control and load following procedures WCAP-8385 (Reference 12) and the Westinghouse topical report on improved F_Q surveillance methodologies WCAP-17661 (Reference 15). The following paragraphs summarize these reports and describe the calculations used to establish the upper bound on peaking factors.

The calculations used to establish the upper bound on peaking factors, F_Q and $F^N_{\Delta H}$, include the nuclear effects which influence the radial and axial power distributions throughout core life for various modes of operation, including load follow, reduced power operation, and axial xenon transients.

Power distributions are calculated for the full-power condition. Fuel and moderator temperature feedback effects are included within these calculations in each spatial dimension. The steady-state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution effects neglected. The effect of flow redistribution is calculated explicitly where it is important in the departure from nucleate boiling analysis of accidents. The effect of xenon on radial power distribution is small (compare Figures 4.3-6 and 4.3-7) but is included as part of the normal design process.

The core axial profile can experience significant changes, which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. For the study of points of closest approach to thermal margin limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the parameters which are readily observed on the plant. Specifically, the nuclear design parameters significant to the axial power distribution analysis are as follows:

- Core power level
- Core height
- Coolant temperature and flow
- Coolant temperature program as a function of reactor power
- Fuel cycle lifetimes

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- Rod bank worth
- Rod bank overlaps

Normal operation of the plant assumes compliance with the following conditions:

- Control rods in a single bank move together with no individual rod insertion differing from the bank demand position by more than the number of steps identified in the technical specifications.
- Control banks are sequenced with overlapping banks.
- The control bank insertion limits are not violated.
- Axial power distribution control procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures followed in normal operation with the online monitoring system out of service. In service, the online core monitoring system provides continuous indication of power distribution, shutdown margin, and margin to design limits.

The constant axial offset control (CAOC) procedures described in WCAP-8385 (Reference 12) were developed to allow the use of a low upper bound on F_Q during normal operation (hence maximizing the margin to core safety limits) while permitting the maintenance of necessary load follow flexibility. The CAOC procedures require control of the AO at all power levels within a permissible operating band about a target value corresponding to the equilibrium full power value. Control to a target AO minimizes transient xenon effects on the axial power distributions.

The F_Q Surveillance Technical Specification responsible for providing assurance that F_Q will remain within the limits assumed in the plant safety analyses when the core is operated within its allowed operating space is based on the CAOC methodology presented in WCAP-8385 (Reference 12). The key operating space limits include RTP, control bank rod insertion limits (RILs), and the axial flux difference (AFD) limits. Together, these operating space limits restrict the range of potential non-equilibrium core power shapes during normal operation, thereby limiting the maximum non-equilibrium $F_Q(Z)$ and the maximum local power density.

The F_Q Surveillance formulation relies on a combination of analytical factors and periodic measurements to provide assurance that core operation within the allowed operating space will be acceptable. When an F_Q surveillance is performed, the equilibrium $F_Q(Z)$ is measured at or near steady-state conditions. This is then multiplied by an analytical factor, W(Z), which characterizes the increase in $F_Q(Z)$ for non-equilibrium operation. The result, when uncertainties are included, is the maximum postulated transient $F_Q(Z)$, which is then compared to the $F_Q(Z)$ limit.

WCAP-17661-P-A (Reference 15) improves upon the CAOC F_Q Surveillance formulation by permitting adjustment of the surveillance to the target AO core conditions. Also, the improved CAOC F_Q Surveillance formulation incorporates the concept of operating spaces by providing the option of implementing a more restrictive CAOC operating space, which is defined as a unique combination of AFD band and control bank RILs, if the F_Q limit is exceeded rather than having to reduce power.

The relaxed axial offset control (RAOC) procedures described in WCAP-10216-P-A (Reference 13) were developed to provide wide control band widths and consequently, more operating flexibility. These wide operating limits, particularly at lower power levels, increase plant availability by allowing quicker plant startup and increased maneuvering flexibility without trip. This procedure has been

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modified to accommodate AP1000 MSHIM operation. It is applied to analysis of axial power distributions under MSHIM control for the purpose of defining the allowed normal operating space such that Condition I thermal margin limits are maintained and Condition II occurrences are adequately protected by the reactor protection system when the online monitoring system is out of service.

The purpose of this analysis is to find the widest permissible ΔI versus power operating space by analyzing a wide range of achievable xenon distributions, MSHIM/AO bank insertion, and power level.

The bounding analyses performed off line in anticipation of the online monitoring system being out of service is similar to that based on the relaxed axial offset control analysis, which uses a xenon reconstruction model described in WCAP-10216-P-A (Reference 13). This is a practical method which is used to define the power operating space allowed with AP1000 MSHIM operation. Each resulting power shape is analyzed to determine if loss-of-coolant accident constraints are met or exceeded.

The online monitoring system evaluates the effects of radial xenon distribution changes due to operational parameter changes continuously and therefore eliminates the need for overly conservative bounding evaluations when the online monitoring system is available. A detailed discussion of this effect may be found in WCAP-8385 (Reference 12). The calculated values have been increased by a factor of 1.05 for method uncertainty and a factor of 1.03 for the engineering factor F^{E}_{Q}.

The envelope drawn in Figure 4.3-14 represents an upper bound envelope on local power density versus elevation in the core. This envelope is a conservative representation of the bounding values of local power density.

The online monitoring system measures the core condition continuously and evaluates the thermal margin condition directly in terms of peak linear heat rate and margin to departure from nucleate boiling limitations directly.

Allowing for fuel densification effects, the average linear power at 3400 MW is 5.72 kW/ft. From Figure 4.3-14, the conservative upper bound value of normalized local power density, including uncertainty allowances, is 2.60 corresponding to a peak linear heat rate of 15.0 kW/ft at each core elevation at 101 percent power.

To determine reactor protection system setpoints with respect to power distributions, three categories of events are considered: rod control equipment malfunctions and operator errors of commission or omission. In evaluating these three categories of events, the core is assumed to be operating within the four constraints described above.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence) for both AO and MSHIM banks. Also included are motions of the AO and MSHIM banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions are calculated throughout these occurrences, assuming short-term corrective action; that is, no transient xenon effects are considered to result from the malfunction. The event is assumed to occur from typical normal operating situations, which include normal xenon transients. It is further assumed in determining the power distributions that total core power level would be limited by reactor trip to below the overpower protection setpoint of nominally 118 percent rated thermal power. Since the study is to determine protection limits with respect to power and axial offset, no credit is taken for OTΔT or OPΔT trip setpoint reduction due to flux difference. The peak power density which can occur in such events, assuming reactor trip at or below

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118 percent, is less than that required for fuel centerline melt, including uncertainties and densification effects.

The second category assumes that the operator mispositions the AO and/or MSHIM rod banks in violation of the insertion limits and creates short-term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct a power distribution limit violation (such as boration/dilution transient) assuming automatic operation of the rod control system which will maintain constant reactor power.

For each of the above categories, the trip setpoints are designed so as not to exceed fuel centerline melt criteria as well as fuel mechanical design criteria.

The appropriate hot channel factors F_Q and $F^N_{\Delta H}$ for peak local power density and for DNB analysis at full power are based on analyses of possible operating power shapes and are addressed in the technical specifications.

The maximum allowable F_Q can be increased with decreasing power, as shown in the technical specifications. Increasing $F^N_{\Delta H}$ with decreasing power is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits, as described in Subsection 4.4.4.3. The allowance for increased $F^N_{\Delta H}$ permitted is addressed in the technical specifications.

This becomes a design basis criterion which is used for establishing acceptable control rod patterns and control bank sequencing. Likewise, fuel loading patterns for each cycle are selected with consideration of this design criterion. The worst values of $F^N_{\Delta H}$ for possible rod configurations occurring in normal operation are used in verifying that this criterion is met. The worst values generally occur when the rods are assumed to be at their insertion limits. Operation with rod positions above the allowed rod insertion limits provides increased margin to the $F^N_{\Delta H}$ criterion. As discussed in Section 3.2 of WCAP-7912-P-A (Reference 14), it has been determined that the technical specifications limits are met, provided the above conditions are observed. These limits are taken as input to the thermal-hydraulic design basis, as described in Subsection 4.4.4.3.1.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the precondition for a subsequent hypothetical accident, but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition. These alarms are described in Chapter 7.

The independence of the various individual uncertainties constituting the uncertainty factor on F_Q enables the uncertainty (F^U_Q) to be calculated by statistically combining the individual uncertainties on the limiting rod. The standard deviation of the resultant distribution of F^U_Q is determined by taking the square root of the sum of the variances of each of the contributing distributions WCAP-7308-L-P-A (Reference 7). The values for F^E_Q and F^N_U are 1.03 and 1.05, respectively. The value for the rod bow factor, F^B_Q, is 1.056, which accounts for the maximum F_Q penalty as a function of burnup due to rod bow effects.

4.3.2.2.7 Experimental Verification of Power Distribution Analysis

This subject is discussed in WCAP-7308-L-P-A (Reference 7) and WCAP-12472-P-A (Reference 4). A summary of these reports and the extension to include the fixed in-core instrumentation system is given below. Power distribution related measurements are incorporated into the evaluation of calculated power distribution information using the in-core instrumentation processing algorithms contained within the online monitoring system. The processing algorithms contained within the online

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monitoring system are functionally identical to those historically used for the evaluation of power distribution measurements in Westinghouse PWRs. Advances in technology allow a complete functional integration of reaction rate measurement algorithms and the expected reaction rate predictive capability within the same software package. The predictive software integrated within the online monitoring system supplies accurate, detailed information of current reactor conditions. The historical algorithms are described in detail in WCAP-12472-P-A (Reference 4).

The measured versus calculational comparison is performed continuously by the online monitoring system throughout the core life. The online monitoring system operability requirements are specified in the technical specifications.

In a measurement of the reactor power distribution and the associated thermal margin limiting parameters, with the in-core instrumentation system described in Subsections 7.7.1 and 4.4.6, the following uncertainties must be considered:

A. Reproducibility of the measured signal

B. Errors in the calculated relationship between detector current and local power generation within the fuel bundle

C. Errors in the detector current associated with the depletion of the emitter material, manufacturing tolerances and measured detector depletion

D. Errors due to the inference of power generation some distance from the measurement thimble

The appropriate allowance for category A has been accounted for through the imposition of strict manufacturing tolerances for the individual detectors. This approach is accepted industry practice and has been used in PWRs with fixed in-core instrumentation worldwide. Errors in category B above are quantified by calculation and evaluation of critical experiment data on arrays of rods with simulated guide thimbles, control rods, burnable absorbers, etc. These critical experiments provide the quantification of errors of categories A and D above. Errors in category C due to depletion of emitter material are accounted for by direct depletion of the detectors as modeled by the monitoring system. A bounding variability is used in the uncertainty calculation to account for the difference between the actual and the tracked detector depletion in the methodology.

WCAP-7308-L-P-A (Reference 7) describes critical experiments performed at the Westinghouse Reactor Evaluation Center and measurements taken on two Westinghouse plants with movable fission chamber in-core instrumentation systems. The measurement aspects of the movable fission chamber share the previous uncertainty categories less category C which is independent of the other sources of uncertainty. WCAP-7308-L-P-A (Reference 7) concludes that the uncertainty associated with peak linear heat rate (F_Q*P) is less than five percent at the 95 percent confidence level with only five percent of the measurements greater than the inferred value.

In comparing measured power distributions (or detector currents) with calculations for the same operating conditions, it is not possible to isolate the detector reproducibility. Thus, a comparison between measured and predicted power distributions includes some measurement error. Such a comparison is given in Figure 4.3-15 for one of the maps used in WCAP-7308-L-P-A (Reference 7). Since the first publication of WCAP-7308-L-P-A, hundreds of measurements have been taken on reactors all over the world. These results confirm the adequacy of the five percent uncertainty allowance on the calculated peak linear heat rate (ALHR*F_Q*P).

A similar analysis for the uncertainty in hot rod integrated power $F_{\Delta H}$*P measurements results in an allowance of four percent at the equivalent of a 95 percent confidence level.

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A measurement in the fourth cycle of a 157-assembly, 12-foot core is compared with a simplified one-dimensional core average axial calculation in Figure 4.3-16. This calculation does not give explicit representation to the fuel grids.

The accumulated data on power distributions in actual operation are basically of three types:

- Much of the data is obtained in steady-state operation at constant power in the normal operating configuration.
- Data with unusual values of axial offset are obtained as part of the ex-core detector calibration exercise performed monthly.
- Special tests have been performed in load follow and other transient xenon conditions which have yielded useful information on power distributions.

These data are presented in detail in WCAP-7912-P-A (Reference 14). Figure 4.3-17 contains a summary of measured values of F_Q as a function of axial offset for five plants from that report.

4.3.2.2.8 Testing

A series of physics tests are planned to be performed on the first core. These tests and the criteria for satisfactory results are described in Chapter 14. Since not all limiting situations can be created at beginning of life, the main purpose of the tests is to provide a check on the calculational methods used in the predictions for the conditions of the test. Tests performed at the beginning of each reload cycle are limited to verification of the selected safety-related parameters of the reload design.

4.3.2.2.9 Monitoring Instrumentation

The adequacy of instrument numbers, spatial deployment, required correlations between readings and peaking factors, calibration, and errors are described in WCAP-12472-P (Reference 4). The relevant conclusions are summarized in Subsection 4.3.2.2.7 and Subsection 4.4.6.

Provided the limitations given in Subsection 4.3.2.2.6 on rod insertion and flux difference are observed, the in-core and ex-core detector systems provide adequate monitoring of power distributions when the online monitoring system is out of service. Further details of specific limits on the observed rod positions and flux difference are given in the technical specifications, together with a discussion of their bases.

Limits for alarms and reactor trip are given in the technical specifications. Descriptions of the systems provided are given in Section 7.7.

4.3.2.3 Reactivity Coefficients

The kinetic characteristics of the reactor core determine the response of the core to changing plant conditions or to operator adjustments made during normal operation, as well as the core response during abnormal or accidental transients. These kinetic characteristics are quantified in reactivity coefficients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions, such as thermal power, moderator and fuel temperatures, coolant pressure, or void conditions, although the latter are relatively unimportant. Since reactivity coefficients change during the life of the core, ranges of coefficients are employed in transient analysis to determine the response of the plant throughout life. The results of such simulations and the reactivity coefficients used are presented in Chapter 15.

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The reactivity coefficients are calculated with approved nuclear methods. The effect of radial and axial power distribution on core average reactivity coefficients is implicit in those calculations and is not significant under normal operating conditions. For example, a skewed xenon distribution which results in changing axial offset by five percent typically changes the moderator and Doppler temperature coefficients by less than 0.01 pcm/°F. An artificially skewed xenon distribution which results in changing the radial $F^{N}_{\Delta H}$ by three percent typically changes the moderator and Doppler temperature coefficients by less than 0.03 pcm/°F and 0.001 pcm/°F, respectively. The spatial effects are accentuated in some transient conditions, for example, in postulated rupture of the main steam line and rupture of a rod cluster control assembly mechanism housing described in Subsections 15.1.5 and 15.4.8, and are included in these analyses.

The analytical methods and calculational models used in calculating the reactivity coefficients are given in Subsection 4.3.3. These models have been confirmed through extensive qualification efforts performed for core and lattice designs.

Quantitative information for calculated reactivity coefficients including fuel-Doppler coefficient, moderator coefficients (density, temperature, pressure, and void), and power coefficient, is given in the following sections.

4.3.2.3.1 Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature and is primarily a measure of the Doppler broadening of U-238 and Pu-240 resonance absorption peaks. Doppler broadening of other isotopes is also considered, but their contribution to the Doppler effect is small. An increase in fuel temperature increases the effective resonance absorption cross sections of the fuel and produces a corresponding reduction in reactivity.

The fuel temperature coefficient is calculated using approved nuclear methods. Moderator temperature is held constant, and the power level is varied. Spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of power density, as discussed in Subsection 4.3.3.1.

A typical Doppler temperature coefficient is shown in Figure 4.3-18 as a function of the effective fuel temperature (at beginning of life and end of life conditions). The effective fuel temperature is lower than the volume-averaged fuel temperature, since the neutron flux distribution is non-uniform through the pellet and gives preferential weight to the surface temperature. A typical Doppler-only contribution to the power coefficient, defined later, is shown in Figure 4.3-19 as a function of relative core power. The integral of the differential curve in Figure 4.3-19 is the Doppler contribution to the power defect and is shown in Figure 4.3-20 as a function of relative power. The Doppler temperature coefficient becomes more negative as a function of life as the Pu-240 content increases, thus increasing the Pu-240 resonance absorption. The upper and lower limits of Doppler coefficient used in accident analyses are given in Chapter 15.

4.3.2.3.2 Moderator Coefficients

The moderator coefficient is a measure of the change in reactivity due to a change in specific coolant parameters, such as density/temperature, pressure, or void. The coefficients obtained are moderator density/temperature, pressure, and void coefficients.

4.3.2.3.2.1 Moderator Density and Temperature Coefficients

The moderator temperature coefficient is defined as the change in reactivity per degree change in the moderator temperature. The moderator density coefficient is defined as the change in reactivity per

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unit change in moderator density. Generally, the effects of the changes in moderator density and the temperature are considered together.

The soluble boron used in the reactor as a means of reactivity control also has an effect on the moderator temperature and density coefficients, since the soluble boron density and the water density are decreased when the coolant temperature rises. A decrease in the soluble boron density introduces a positive component in the moderator density coefficient. If the concentration of soluble boron is large enough, the net value of the moderator temperature coefficient may be positive.

The initial core hot boron concentration is sufficiently low that the moderator temperature coefficient is negative at operating temperatures with the burnable absorber loading specified. Discrete or integral fuel burnable absorbers can be used in reload cores to confirm the moderator temperature coefficient is negative over the range of power operation. The effect of control rods is to make the moderator temperature coefficient more negative, since the thermal neutron mean free path, and hence the volume affected by the control rods, increase with an increase in temperature.

Except for a period early in the fuel cycle when boron density is increased briefly to compensate for burnable poison rod depletion, with burnup, the moderator temperature coefficient becomes more negative, primarily as a result of boric acid dilution, but also to a significant extent from the effects of the buildup of plutonium and fission products.

The moderator temperature and density coefficients are calculated for a range of plant conditions by performing two group two- or three-dimensional calculations, in which the moderator temperature is varied by about ±5°F about each of the mean temperatures, resulting in density changes consistent with the temperature change. The moderator temperature coefficient is shown as a function of core temperature and boron concentration for the core in Figures 4.3-21 through 4.3-23. The temperature range covered is from cold, about 70°F, to about 550°F. Figure 4.3-24 shows the unrodded hot full-power moderator temperature coefficient plotted as a function of burnup for the initial cycle. The temperature coefficient corresponds to the unrodded critical boron concentration present at hot full power operating conditions.

The moderator coefficients presented here are calculated to describe the core behavior in normal and accident situations when the moderator temperature changes can be considered to affect the entire core.

4.3.2.3.2.2 Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is of much less significance than the moderator temperature coefficient. A change of 50 psi in pressure has approximately the same effect on reactivity as a one half degree change in moderator temperature. This coefficient can be determined from the moderator temperature coefficient by relating change in pressure to the corresponding change in density. The typical moderator pressure coefficient may be negative over a portion of the moderator temperature range at beginning of life (BOL) (-0.004 pcm/psi) but is always positive at operating conditions and becomes more positive during life (+0.3 pcm/psi, at end of life).

4.3.2.3.2.3 Moderator Void Coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. In a PWR, this coefficient is not very significant because of the low void content in the coolant. The core void content is less than one-half of one percent and is due to local or statistical boiling. The typical void coefficient varies from 50 pcm/percent void at BOL and at low temperatures

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to minus 250 pcm/percent void at EOL and at operating temperatures. The void coefficient at operating temperature becomes more negative with fuel burnup.

4.3.2.3.3 Power Coefficient

The combined effect of moderator temperature and fuel temperature change as the core power level changes is called the total power coefficient and is expressed in terms of reactivity change per percent power change. Since a three-dimensional calculation is performed in determining total power coefficients and total power defects, the axial redistribution reactivity component described in Subsection 4.3.2.4.3 is implicitly included. A typical power coefficient at beginning of life (BOL) and end of life (EOL) conditions is given in Figure 4.3-25.

The total power coefficient becomes more negative with burnup, reflecting the combined effect of moderator and fuel temperature coefficients with burnup. The power defect (integral reactivity effect) at BOL and EOL is given in Figure 4.3-26.

4.3.2.3.4 Comparison of Calculated and Experimental Reactivity Coefficients

Subsection 4.3.3 describes the comparison of calculated and experimental reactivity coefficients in detail.

Experimental evaluation of the reactivity coefficients will be performed during the physics startup tests described in Chapter 14.

4.3.2.3.5 Reactivity Coefficients Used in Transient Analysis

Table 4.3-2 gives the limiting values as well as typical best-estimate values for the reactivity coefficients for the initial cycle. The limiting values are used as design limits in the transient analysis. The exact values of the coefficient used in the analysis depend on whether the transient of interest is examined at the BOL or EOL, whether the most negative or the most positive (least negative) coefficients are appropriate, and whether spatial non-uniformity must be considered in the analysis. Conservative values of coefficients, considering various aspects of analysis, are used in the transient analysis. This is described in Chapter 15.

The reactivity coefficients shown in Figures 4.3-18 through 4.3-26 are typical best-estimate values calculated for the initial cycle. Limiting values are chosen to encompass the best-estimate reactivity coefficients, including the uncertainties given in Subsection 4.3.3.3 over appropriate operating conditions. The most positive, as well as the most negative, values are selected to form the design basis range used in the transient analysis. A direct comparison of the best-estimate and design limit values for the initial cycle is shown in Table 4.3-2. In many instances the most conservative combination of reactivity coefficients is used in the transient analysis even though the extreme coefficients assumed may not simultaneously occur at the conditions assumed in the analysis. The need for a reevaluation of any accident in a subsequent cycle is contingent upon whether the coefficients for that cycle fall within the identified range used in the analysis presented in Chapter 15 with due allowance for the calculational uncertainties given in Subsection 4.3.3.3. Control rod requirements are given in Table 4.3-3 for the initial cycle and for a hypothetical equilibrium cycle, since these are markedly different. These latter numbers are provided for information only.

4.3.2.4 Control Requirements

To establish the required shutdown margin stated in the COLR under conditions where a cooldown to ambient temperature is required, concentrated soluble boron is added to the coolant. Boron concentrations for several core conditions are listed in Table 4.3-2 for the initial cycle. For core conditions including refueling, the boron concentration is well below the solubility limit. The rod

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cluster control assemblies are employed to bring the reactor to the shutdown condition. The minimum required shutdown margin is given in the COLR.

The ability to meet the shutdown margin requirements for hot conditions is demonstrated for the initial cycle and for an equilibrium reload cycle by performing a bounding calculation, the results of which are shown in Table 4.3-3. Table 4.3-3 compares the difference between the rod cluster control assembly reactivity available with an allowance for the worst stuck rod with that required for control and protection purposes. The shutdown margin includes an allowance of seven percent for analytic uncertainties which assumes the use of silver-indium-cadmium rod cluster control assemblies. Use of a seven percent uncertainty allowance on rod cluster control assembly worth is discussed and shown to be acceptable in WCAP-9217 (Reference 17). The largest reactivity control requirement appears at the EOL when the moderator temperature coefficient reaches its peak negative value as reflected in the larger power defect.

Any available negative reactivity insertion from withdrawn tungsten GRCAs is conservatively excluded when determining the available shutdown margin at hot operating conditions, even though all GRCAs are released into the core on a reactor trip (Reference 70). Only silver-indium-cadmium control rods are assumed to insert when the reactor is tripped for purposes of demonstrating that adequate shutdown margin is available at hot operating conditions. As such, the use of a seven percent uncertainty allowance for the credited trip rod worth remains appropriate in Table 4.3-3. After the reactor is brought to a shut down condition, the presence of GRCAs which are confirmed to be inserted and which have met the applicable physics testing acceptance criteria may be credited in confirming that the required shutdown margin is maintained during any cooldown period and as the result of long term xenon decay (Reference 70).

During plant operation, the available shutdown margin for hot operating conditions is continuously confirmed by the online monitoring system, by comparing the operating soluble boron concentration at current core conditions to the shutdown boron concentration that would be required immediately following a reactor trip from those conditions. The required shutdown boron concentration used in this type of calculation is conservatively determined at the target shutdown reactivity condition, assuming that all control rods insert except for the 16 tungsten GRCAs and the highest worth silver-indium-cadmium RCCA.

The control rods are required to provide sufficient reactivity to account for the power defect from full power to zero power and to provide the required shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler effect, moderator temperature, flux redistribution, and reduction in void content as discussed below.

4.3.2.4.1 Doppler Effect

The Doppler effect arises from the broadening of U-238 and Pu-240 resonance cross-sections with an increase in effective pellet temperature. This effect is most noticeable over the range of zero power to full power due to the large pellet temperature increase with power generation. The Doppler effect is implicitly included in the total power defects shown in Table 4.3-3.

4.3.2.4.2 Variable Average Moderator Temperature

When the core is shut down to the hot zero-power condition, the average moderator temperature changes from the equilibrium full-load value determined by the steam generator and turbine characteristics (such as steam pressure, heat transfer, tube fouling) to the equilibrium no-load value, which is based on the steam generator shell side design pressure. The design change in temperature is conservatively increased to account for the control system dead band and measurement errors.

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When the moderator coefficient is negative, there is a reactivity addition with power reduction. The moderator coefficient becomes more negative as the fuel depletes because the boron concentration is reduced. This effect is the major contributor to the increased requirement at EOL. The change in average moderator temperature is implicitly included in the total power defects shown in Table 4.3-3.

4.3.2.4.3 Redistribution

During full-power operation, the coolant density decreases with core height. This, together with partial insertion of control rods, results in less fuel depletion near the top of the core. Under steady-state conditions, the relative power distribution will be slightly asymmetric toward the bottom of the core. On the other hand, at hot zero-power conditions, the coolant density is uniform up the core, and there is no flattening due to Doppler effect. The result will be a flux distribution which at zero power can be skewed toward the top of the core. Since a three-dimensional calculation is performed in determining total power defect, flux redistribution is implicitly included in this calculation. The three-dimensional total power defects specified in Table 4.3-3 were calculated including the use of a conservatively skewed adverse axial xenon distribution which increases the redistribution effect.

4.3.2.4.4 Void Content

A small void content in the core is due to nucleate boiling at full power. The void collapse coincident with power reduction makes a small positive reactivity contribution which has been added to the calculated total power defects shown in Table 4.3-3.

4.3.2.4.5 Rod Insertion Allowance

The MSHIM and AO banks are operated within a prescribed band of travel to compensate for changes in temperature and axial offset which are caused by fuel depletion and power maneuvers as described in Subsection 4.3.2.4.16. In calculating the available shutdown margin at hot operating conditions, the pre-trip control rod insertion can affect both the available trip rod worth and the total power defect control requirements. In addition, since the tungsten GRCAs are assumed not to insert on a reactor trip (Reference 70), the initial gray rod positions assumed prior to the trip can also have a small effect on the worth of the silver-indium-cadmium control rods that insert after the trip. In the bounding calculations shown in Table 4.3-3, the effect of the most limiting allowed control rod insertion is implicitly included in the calculated trip rod worth and total power defect values reported in the table. The most limiting allowed control rod insertion was determined by performing a series of three-dimensional shutdown margin calculations over the range of allowed control rod motion, and selecting the conditions which resulted in the minimum calculated shutdown margin.

4.3.2.4.6 Installed Excess Reactivity for Depletion

Excess reactivity is installed at the beginning of each cycle to provide sufficient reactivity to compensate for fuel depletion and fission product buildup throughout the cycle. This reactivity is controlled by the addition of soluble boron to the coolant, control rod insertion, and by burnable absorbers when necessary. The soluble boron concentration for several core configurations and the unit boron worth are given in Tables 4.3-1 and 4.3-2 for the initial cycle. Since the excess reactivity for burnup is balanced during operation by negative reactivity from the above sources, it is not included in control rod requirements.

4.3.2.4.7 Xenon and Samarium Poisoning

Changes in xenon and samarium concentrations in the core occur at a sufficiently slow rate, even following rapid power level changes, that the resulting reactivity change can be controlled by changing the gray and/or control rod insertion. (Also see Subsection 4.3.2.4.16).

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4.3.2.4.8 pH Effects

Changes in reactivity due to a change in coolant pH, if any, are sufficiently small in magnitude and occur slowly enough to be controlled by the boron system WCAP-3696-8 (Reference 18).

4.3.2.4.9 Experimental Confirmation

Following a normal shutdown, the total core reactivity change during cooldown with a stuck rod has been measured on a 121-assembly, 10-foot-high core and a 121-assembly, 12-foot-high core. In each case, the core was allowed to cool down until it reached criticality simulating the steam line break accident. For the 10-foot core, the total reactivity change associated with the cooldown is over predicted by about 0.3-percent Δρ with respect to the measured result. This represents an error of about five percent in the total reactivity change and is about half the uncertainty allowance for this quantity. For the 12-foot core, the difference between the measured and predicted reactivity change is an even smaller 0.2 percent Δρ. These measurements and others demonstrate the capability of the methods described in Subsection 4.3.3.

4.3.2.4.10 Control

Core reactivity is controlled by means of a chemical poison dissolved in the coolant, rod cluster control assemblies, gray rod cluster assemblies and burnable absorbers as described below.

4.3.2.4.11 Chemical Shim

Boron in solution as boric acid is used to control relatively slow reactivity changes associated with:

- The moderator temperature defect in going from cold shutdown at ambient temperature to the hot operating temperature at zero power
- Transient xenon and samarium reactivity effects following power changes
- The reactivity effects of fissile inventory depletion and buildup of long-life fission products
- The depletion of the burnable absorbers

The boron concentrations for various core conditions are presented in Table 4.3-2 for the initial cycle.

4.3.2.4.12 Rod Cluster Control Assemblies

The number of rod cluster control assemblies is shown in Table 4.3-1. The rod cluster control assemblies are used for shutdown and control purposes to offset fast reactivity changes associated with:

- The required shutdown margin in the hot zero power, stuck rod condition
- The reactivity compensation as a result of an increase in power above hot zero power (power defect, including Doppler and moderator reactivity changes)
- Unprogrammed fluctuations in boron concentration, coolant temperature, or xenon concentration (with rods not exceeding the allowable rod insertion limits)
- Reactivity changes resulting from load changes

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The allowed control bank reactivity insertion is limited at full power to maintain shutdown capability. As the power level is reduced, control rod reactivity requirements are also reduced, and more rod insertion is allowed. The control bank position is monitored, and the operator is notified by an alarm if the limit is approached. The determination of the insertion limit uses conservative xenon distributions and axial power shapes. In addition, the rod cluster control assembly withdrawal pattern determined from the analyses is used in determining power distribution factors and in determining the maximum worth of an inserted rod cluster control assembly ejection accident. For further discussion, refer to the technical specifications on rod insertion limits.

Power distribution, rod ejection, and rod misalignment analyses are based on the arrangement of the shutdown and control groups of the rod cluster control assemblies shown in Figure 4.3-27. Shutdown rod cluster control assemblies are withdrawn before withdrawal of the control and AO banks is initiated. The approach to critical is initiated by using the chemical and volume control system to establish an appropriate boron concentration based upon the estimated critical condition then withdrawing the AO bank above the zero power insertion limit and finally withdrawing the control banks sequentially. The limits of rod insertion and further discussion on the basis for rod insertion limits are provided in the COLR.

4.3.2.4.13 Gray Rod Cluster Assemblies

The rod cluster control assembly control banks include four gray rod banks consisting of gray rod cluster assemblies (GRCAs). Gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub or spider. Geometrically, it is the same as a rod cluster control assembly except that the GRCA design uses tungsten encapsulated in a nickel-chromium-iron Alloy 718 sleeve as an absorber. The term gray rod refers to the reduced reactivity worth relative to that of a rod cluster control assembly consisting of 24 silver-indium-cadmium rodlets. The gray rod cluster assemblies are used in base load operation and load follow maneuvering and provide a mechanical shim reactivity mechanism which reduces the need for changes to the concentration of soluble boron (that is, chemical shim).

4.3.2.4.14 Burnable Absorbers

Discrete burnable absorber rods and integral fuel burnable absorber rods will be used to provide partial control of the excess reactivity available during the first operating cycle. In doing so, the burnable absorber loading controls peaking factors and prevents the moderator temperature coefficient from being positive at normal operating conditions. The burnable absorbers perform this function by reducing the requirement for soluble boron in the moderator at the beginning of the fuel cycle, as described previously. For purposes of illustration, the initial cycle burnable absorber pattern is shown in Figure 4.3-5. Figures 4.3-4a and 4.3-4b show the burnable absorber distribution within a fuel assembly for several burnable absorber patterns used in the 17 x 17 array. The boron in the rods is depleted with burnup but at a slow rate so that the peaking factor limits are not exceeded and the resulting critical concentration of soluble boron is such that the moderator temperature coefficient remains within the limits stated above for power operating conditions.

4.3.2.4.15 Peak Xenon Startup

Compensation for the peak xenon buildup may be accomplished using the boron control system. Startup from the peak xenon condition is accomplished with a combination of rod motion and boron dilution. The boron dilution can be made at any time, including during the shutdown period, provided the shutdown margin is maintained.

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4.3.2.4.16 Load Follow Control and Xenon Control

During load follow maneuvers, power changes are primarily accomplished using control rod motion alone, as required. Control rod motion is limited by the control rod insertion limits as provided in the COLR and discussed in Subsections 4.3.2.4.12 and 4.3.2.4.5. The power distribution is maintained within acceptable limits through limitations on control rod insertion. Reactivity changes due to the changing xenon concentration are also controlled by rod motion. The soluble boron concentration may also be changed during large load change maneuvers or during extended reduced power operation to maintain the control rods in a more optimum range for power distribution control.

Rapid power increases (five percent/min) from part power during load follow operation are accomplished with rod motion.

The rod control system is designed to automatically provide the power and temperature control described above 30 percent rated power for most of the cycle length without the need to change boron concentration as a result of the load maneuver. The automated mode of operation is referred to as mechanical shim (MSHIM) because of the usage of mechanical means to control reactivity and power distribution simultaneously. MSHIM operation allows load maneuvering without boron change because of the degree of allowed insertion of the control banks in conjunction with the independent power distribution control of the axial offset (AO) control bank. The worth and overlap of the MA, MB, MC, MD, M1, and M2 control banks are designed such that the AO control bank insertion will always result in a monotonically decreasing axial offset. MSHIM operation uses the MA, MB, MC, MD, M1, and M2 control banks to maintain the programmed coolant average temperature throughout the operating power range. The AO control bank is independently modulated by the rod control system to maintain a nearly constant axial offset throughout the operating power range. The target axial offset used during MSHIM operation is established at a more negative value than the axial offset associated with the all rods out condition. The negative bias is necessary to maintain both positive and negative axial offset control effectiveness by the AO-bank. Operation with gray control rod banks (MA, MB, MC, and MD) inserted has less of an effect on the core axial power distribution than insertion of the black control rods banks (M1 and M2) and results in a smaller negative bias in the target axial offset. Load change operations that are large enough to require a black control rod bank to enter the core may require a more negative target axial offset to accomplish. However, the boron system can optionally be used to maintain operation in the more optimum range of gray rod motion during such maneuvers. The degree of control rod insertion under MSHIM operation allows rapid return to power without the need to change boron concentration.

Extended base load operation is performed by controlling axial offset to the target value using the AO control bank, and by controlling the coolant average temperature to the programmed value with the M-banks. Boron concentration changes are made periodically as the fuel depletes to reposition the M-banks and allow for a periodic exchange of the gray rod bank insertion sequence. MSHIM load follow and base load operations (including the gray rod bank insertion sequence exchanges) are considered Condition I normal operations.

4.3.2.4.17 Burnup

Control of the excess reactivity for burnup is accomplished using soluble boron, control rod insertion, and/or burnable absorbers. The boron concentration is limited during operating conditions to maintain the moderator temperature coefficient within its specified limits. A sufficient burnable absorber loading is installed at the beginning of a cycle to give the desired cycle lifetime, without exceeding the boron concentration limit. The end of a fuel cycle is reached when the soluble boron concentration approaches the practical minimum boron concentration in the range of 0 to 10 ppm.

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4.3.2.4.18 Rapid Power Reduction System

The reactor power control system is designed with the capability of responding to full load rejection without initiating a reactor trip using the normal rod control system, reactor control system, and the rapid power reduction system. Load rejections requiring greater than a fifty percent reduction of rated thermal power or the loss of a main feedwater pump train as indicated by the pump motor breaker positions initiates the rapid power reduction system when above a predetermined power level. The rapid power reduction system utilizes selected (either automatically or manually) control rod banks which are intentionally tripped to rapidly reduce reactor power into a range where the rod control and reactor control systems are sufficient to maintain stable plant operation. The consequences of accidental or inappropriate actuation of the rapid power reduction system are included in the cycle specific safety analysis and licensing process.

4.3.2.5 Control Rod Patterns and Reactivity Worth

The rod cluster control assemblies are designated by function as the control groups and the shutdown groups. The terms group and bank are used synonymously to describe a particular grouping of control assemblies. The rod cluster control assembly patterns are displayed in Figure 4.3-27. The control banks are labeled MA, MB, MC, MD, M1, M2, and AO with the MA, MB, MC, and MD banks comprised of gray rod cluster assemblies; and the shutdown banks are labeled S1, S2, S3, and S4. Each bank of more than four rod cluster control assemblies, although operated and controlled as a unit, is composed of two or more subgroups. The axial position of the rod cluster control assemblies may be controlled manually or automatically. The rod cluster control assemblies are dropped into the core following actuation of reactor trip signals.

Two criteria have been employed for selection of the control groups. First, the total reactivity worth must be adequate to meet the requirements specified in Table 4.3-3. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to meet the power capability requirements. Analyses indicate that the first requirement can be met either by a single group or by two or more banks whose total worth equals at least the required amount. The axial power shape is more peaked following movement of a single group of rods worth three to four percent $\Delta\rho$. Therefore, control bank rod cluster control assemblies have been separated into several bank groupings. Typical control bank worth for the initial cycle are shown in Table 4.3-2.

The position of control banks for criticality under any reactor condition is determined by the concentration of boron in the coolant. On an approach to criticality, boron is adjusted so that criticality will be achieved with control rods above the insertion limit set by shutdown and other considerations. (See the technical specifications and COLR). Early in the cycle, there may also be a withdrawal limit at low power to maintain the moderator temperature coefficient within the specified limits for that power level.

Ejected rod worths for several different conditions are given in Subsection 15.4.8.

Allowable deviations due to misaligned control rods are discussed in the technical specifications.

A representative differential rod worth calculation for two banks of control rods withdrawn simultaneously (rod withdrawal accident) is given in Figure 4.3-28.

Calculation of control rod reactivity worth versus time following reactor trip involves both control rod velocity and differential reactivity worth. A representative example of the rod position versus time of travel after rod release is given in Figure 4.3-29. The actual rod position versus time of travel used in the safety analysis is given in Chapter 15. For nuclear design purposes, the reactivity worth versus rod position is calculated by a series of steady-state calculations at various control positions,

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assuming the rods out of the core as the initial position in order to minimize the initial reactivity insertion rate. Also, to be conservative, the rod of highest worth is assumed stuck out of the core, and the flux distribution (and thus reactivity importance) is assumed to be skewed to the bottom of the core. A representative result of these calculations is shown in Figure 4.3-30.

The shutdown groups provide additional negative reactivity to establish adequate shutdown margin. Shutdown margin is the amount by which the core would be subcritical at hot shutdown if the rod cluster control assemblies were tripped, but assuming that the highest worth assembly remained fully withdrawn and no changes in xenon or boron took place. The loss of control rod worth due to the depletion of the absorber material is negligible.

The values given in Table 4.3-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design bases minimum shutdown margin, allowing for the highest worth cluster to be at its fully withdrawn position. An allowance for the uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.6 Criticality of the Reactor During Refueling

The basis for maintaining the reactor subcritical during refueling is presented in Subsection 4.3.1.5, and a discussion of how control requirements are met is given in Subsections 4.3.2.4 and 4.3.2.5.

4.3.2.6.1 Criticality Design Method Outside the Reactor

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping, and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies. The details of the methodology used for the new fuel rack and spent fuel rack criticality analysis are included in the Chapter 9.1 references.

The design criteria are consistent with General Design Criterion (GDC) 62, Reference 19, and NRC guidance given in Reference 20. The applicable 10 CFR Part 50.68 requirements are as follows:

1. The maximum K-effective value, including all biases and uncertainties, must be less than 0.95 with soluble boron credit and less than 1.0 with full density unborated water. Note this design criterion is provided in 10 CFR Part 50.68, Item 4 of Paragraph b. Note that the specific terminology is:

   “If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.”

2. The maximum enrichment of fresh fuel assemblies must be less than or equal to 5.0 weight-percent U-235. Note this design criterion is provided in 10 CFR Part 50.68, Item 7 of Paragraph b. Note that the specific terminology is:

   “The maximum nominal U–235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.”

The following conditions are assumed in meeting this design bases:

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- The fuel assembly contains the highest enrichment authorized without any control rods or non-integral burnable absorber(s) and is at its most reactive point in life.
- For flooded conditions, the moderator is pure water at the temperature within the design limits which yields the largest reactivity.
- The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design.
- Mechanical uncertainties are treated by combining both the worst-case bounding value and sensitivity study approaches.
- Credit is taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption.

Fuel depletion analyses during core operation were performed with CASMO-4 (using the 70-group cross-section library), a two-dimensional multigroup transport theory code based on capture probabilities (Reference 53). CASMO-4 is used to determine the isotopic composition of the spent fuel. In addition, the CASMO-4 calculations are restarted in the storage rack geometry, yielding the two-dimensional infinite multiplication factor (k_{inf}) for the storage rack to determine the reactivity effect of fuel and rack tolerances, temperature variation, and to perform various studies.

The design method which determines the criticality safety of fuel assemblies outside the reactor uses the MCNP4a code (Reference 21), with continuous energy cross-sections based on ENDF/B-V and ENDF/B-VI.

A set of 62 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty. The benchmark experiments cover a wide range of geometries, materials, and enrichments, all of them adequate for qualifying methods to analyze light water reactor lattices (References 22 to 28, and 65 to 68).

The analysis of the 62 critical experiments results in an average K_{eff} of 0.9991. Comparison with the measured values results in a method bias of 0.0009. The standard deviation of the set of reactivities is 0.0011. The 95/95 tolerance factor is conservatively set to 2.0.

The analytical methods employed herein conform with ANSI N18.2 (Reference 3), Section 5.7, Fuel Handling System; ANSI N16.9 (Reference 29), NRC Standard Review Plan, Subsection 9.1.2, the NRC guidance, “OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications” (Reference 30).

4.3.2.6.2 Soluble Boron Credit Methodology

The minimum soluble boron requirement under normal and accident conditions must be determined to show that the reactivity of the spent fuel racks remains below 0.95. This is achieved by crediting a discrete amount of soluble boron and then determining by linear interpolation the appropriate amount of soluble boron necessary to reduce the maximum K_{eff} to 0.95 with all uncertainties and biases included.

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4.3.2.7 Stability

4.3.2.7.1 Introduction

The stability of the PWR cores against xenon-induced spatial oscillations and the control of such transients are discussed extensively in References 11, 31, 32, and 33. A summary of these reports is given in the following discussion, and the design bases are given in Subsection 4.3.1.6.

In a large reactor core, xenon-induced oscillations can take place with no corresponding change in the total power of the core. The oscillation may be caused by a power shift in the core which occurs rapidly by comparison with the xenon-iodine time constants. Such a power shift occurs in the axial direction when a plant load change is made by control rod motion and results in a change in the moderator density and fuel temperature distributions. Such a power shift could occur in the diametral plane of the core as a result of abnormal control action.

Due to the negative power coefficient of reactivity, PWR cores are inherently stable to oscillations in total power. Protection against total power instabilities is provided by the control and protection system, as described in Section 7.7. Hence, the discussion on the core stability will be limited to xenon-induced spatial oscillations.

4.3.2.7.2 Stability Index

Power distributions, either in the axial direction or in the X-Y plane, can undergo oscillations due to perturbations introduced in the equilibrium distributions without changing the total core power. The harmonics and the stability of the core against xenon-induced oscillations can be determined in terms of the eigenvalue of the first flux harmonics. Writing the eigenvalue ξ of the first flux harmonic as:

$$\xi = b + ic \tag{1}$$

Then b is defined as the stability index and $T = 2\pi/c$ as the oscillation period of the first harmonic. The time dependence of the first harmonic $\delta\phi$ in the power distribution can now be represented as:

$$\delta\phi(t) = A e^{\xi t} = a e^{bt} \cos ct \tag{2}$$

where A and a are constants. The stability index can also be obtained approximately by:

$$b = \frac{1}{T} \ln \frac{A_{n+1}}{A_n} \tag{3}$$

where A_n and A_{n+1} are the successive peak amplitudes of the oscillation and T is the time period between the successive peaks.

4.3.2.7.3 Prediction of the Core Stability

The core described in this report has an active fuel length that is 24 inches longer (nominal) than that for previous Westinghouse PWRs licensed in the U.S. with 157 fuel assemblies. For this reason, it is expected that this core will be as stable as the 12-foot designs with respect to radial and diametral xenon oscillations since the radial core dimensions have not changed. This core will be slightly less stable than the 12-foot, 157 assembly cores with respect to axial xenon oscillations because the active core height has been increased by 24 inches. The effect of this increase will be to decrease the burnup at which the axial stability index becomes zero (Subsection 4.3.2.7.4 below). The moderator temperature coefficients and the Doppler temperature coefficients of reactivity will be similar to those of previous designs. Control banks included in the core design are sufficient to

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dampen any xenon oscillations that may occur. Free axial xenon oscillations are not allowed to occur for a core of any height, except during special tests as described in Subsection 4.3.2.7.4.

4.3.2.7.4 Stability Measurements

4.3.2.7.4.1 Axial Measurements

Two axial xenon transient tests conducted in a PWR with a core height of 12 feet and 121 fuel assemblies are reported in WCAP-7964 (Reference 34) and are discussed here. The tests were performed at approximately 10 percent and 50 percent of cycle life.

Both a free-running oscillation test and a controlled test were performed during the first test. The second test at mid-cycle consisted of a free-running oscillation test only. In each of the free-running oscillation tests, a perturbation was introduced to the equilibrium power distribution through an impulse motion of the lead control bank and the subsequent oscillation period was monitored. In the controlled test conducted early in the cycle, the part-length rods were used to follow the oscillations to maintain an axial offset within the prescribed limits. The axial offset of power was obtained from the ex-core ion chamber readings (which had been calibrated against the in-core flux maps) as a function of time for both free-running tests, as shown in Figure 12 of WCAP-7964 (Reference 34).

The total core power was maintained constant during these spatial xenon tests, and the stability index and the oscillation period were obtained from a least-square fit of the axial offset data in the form of equation 2. The axial offset of power is the quantity that properly represents the axial stability in the sense that it essentially eliminates any contribution from even-order harmonics, including the fundamental mode. The conclusions of the tests follow:

- The core was stable against induced axial xenon transients, at the core average burnups of both 1550 MWD/MTU and 7700 MWD/MTU. The measured stability indices are -0.041 h^{-1} for the first test and - 0.014 h^{-1} for the second test. The corresponding oscillation periods are 32.4 and 27.2 hours, respectively.

- The reactor core becomes less stable as fuel burnup progresses, and the axial stability index is essentially zero at 12,000 MWD/MTU. However, the movable control rod systems can control axial oscillations, as described in Subsection 4.3.2.7.3.

4.3.2.7.4.2 Measurements in the X-Y Plane

Two X-Y xenon oscillation tests were performed at a PWR plant with a core height of 12 feet and 157 fuel assemblies. The first test was conducted at a core average burnup of 1540 MWD/MTU and the second at a core average burnup of 12,900 MWD/MTU. Both of the X-Y xenon tests show that the core was stable in the X-Y plane at both burnups. The second test shows that the core became more stable as the fuel burnup increased, and Westinghouse PWRs with 121 and 157 assemblies are stable throughout their burnup cycles. The results of these tests are applicable to the 157-assembly AP1000 core, as discussed in Subsection 4.3.2.7.3.

In each of the two X-Y tests, a perturbation was introduced to the equilibrium power distribution through an impulse motion of one rod cluster control unit located along the diagonal axis. Following the perturbation, the uncontrolled oscillation was monitored, using the movable detector and thermocouple system and the ex-core power range detectors. The quadrant tilt difference (QTD) is the quantity that properly represents the diametral oscillation in the X-Y plane of the reactor core in that the differences of the quadrant average powers over two symmetrically opposite quadrants essentially eliminates the contribution to the oscillation from the azimuthal mode. The quadrant tilt difference data were fitted in the form of equation 2 of Subsection 4.3.2.7.2 through a least-square

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method. A stability index of - 0.076 hr^{-1} (per hour) with a period of 29.6 hr was obtained from the thermocouple data shown in Figure 4.3-31.

It was observed in the second X-Y xenon test that the PWR core with 157 fuel assemblies had become more stable due to an increased fuel depletion, and the stability index was not determined.

4.3.2.7.5 Comparison of Calculations with Measurements

The direct simulation of axial offset data was carried out using a licensed one-dimensional code (WCAP-7048-P-A (Reference 35)). The analysis of the X-Y xenon transient tests was performed in an X-Y geometry, using a licensed few group two-dimensional code (WCAP-7213-A (Reference 36)). Both of these codes solve the two-group, time-dependent neutron diffusion equation with time-dependent xenon and iodine concentrations. The fuel temperature and moderator density feedback is limited to a steady-state model. The X-Y calculations were performed in an average enthalpy plane.

The detailed experimental data during the tests, including the reactor power level, the enthalpy rise, and the impulse motion of the control rod assembly, as well as the plant follow burnup data, were closely simulated in the study.

The results of the stability calculation for the axial tests are compared with the experimental data in Table 4.3-5. The calculations show conservative results for both of the axial tests with a margin of approximately 0.01 hr^{-1} in the stability index.

An analytical simulation of the first X-Y xenon oscillation test shows a calculated stability index of - 0.081 hr^{-1}, in good agreement with the measured value of - 0.076 hr^{-1}. As indicated earlier, the second X-Y xenon test showed that the core had become more stable compared to the first test, and no evaluation of the stability index was attempted. This increase in the core stability in the X-Y plane due to increased fuel burnup is due mainly to the increased magnitude of the negative moderator temperature coefficient.

Previous studies of the physics of xenon oscillations, including three-dimensional analysis, are reported in a series of topical reports (References 31, 32, and 33). A more detailed description of the experimental results and analysis of the axial and X-Y xenon transient tests is presented in WCAP-7964 (Reference 34) and Section 1 of WCAP-8768 (Reference 37).

4.3.2.7.6 Stability Control and Protection

The online monitoring system provides continuous indication of current power distributions and provides guidance to the plant operator as to the timing and most appropriate action(s) to maintain stable axial power distributions. In the event the online monitoring system is out of service, the ex-core detector system is utilized to provide indications of xenon-induced spatial oscillations. The readings from the ex-core detectors are available to the operator and also form part of the protection system.

4.3.2.7.6.1 Axial Power Distribution

The rod control system automatically maintains axial power distribution within very tight axial offset bands as part of normal operation. The AO control bank is specifically designed with sufficient worth to be capable of maintaining essentially constant axial offset over the power operating range. The rod control system is also allowed to be operated in manual control in which case the operator is instructed to maintain an axial offset within a prescribed operating band, based on the ex-core detector readings. Should the axial offset be permitted to move far enough outside this band, the

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protection limit is encroached, and the turbine power is automatically reduced or a reactor trip signal generated, or both.

As fuel burnup progresses, PWR cores become less stable to axial xenon oscillations. However, free xenon oscillations are not allowed to occur, except for special tests. The AO control bank is sufficient to dampen and control any axial xenon oscillations present. Should the axial offset be inadvertently permitted to move far enough outside the allowed band due to an axial xenon oscillation or for any other reason, the OTΔT and/or OPΔT protection setpoint including the axial offset compensation is reached and the turbine power is automatically reduced and/or a reactor trip signal is generated.

4.3.2.7.6.2 Radial Power Distribution

The core described herein is calculated to be stable against X-Y xenon-induced oscillations during the core life.

The X-Y stability of large PWRs has been further verified as part of the startup physics test program for PWR cores with 193 fuel assemblies. The measured X-Y stability of the cores with 157 and 193 assemblies was in close agreement with the calculated stability, as discussed in Subsections 4.3.2.7.4 and 4.3.2.7.5. In the unlikely event that X-Y oscillations occur, backup actions are possible and would be implemented, if necessary, to increase the natural stability of the core. This is based on the fact that several actions could be taken to make the moderator temperature coefficient more negative, which would increase the stability of the core in the X-Y plane.

Provisions for protection against non-symmetric perturbations in the X-Y power distribution that could result from equipment malfunctions are made in the protection system design. This includes control rod drop, rod misalignment, and asymmetric loss of coolant flow.

A more detailed discussion of the power distribution control in PWR cores is presented in WCAP-7811 (Reference 11) and WCAP-8385 (Reference 12).

4.3.2.8 Vessel Irradiation

A review of the methods and analyses used in the determination of neutron and gamma ray flux attenuation between the core and the pressure vessel is provided below. A more complete discussion on the pressure vessel irradiation and surveillance program is given in Section 5.3.

The materials that serve to attenuate neutrons originating in the core and gamma rays from both the core and structural components consist of the core shroud, core barrel and associated water annuli. These are within the region between the core and the pressure vessel.

In general, few group neutron diffusion theory codes are used to determine fission power density distributions within the active core, and the accuracy of these analyses is verified by in-core measurements on operating reactors. Region and rodwise power-sharing information from the core calculations is then used as source information in two-dimensional transport calculations which compute the flux distributions throughout the reactor.

The neutron flux distribution and spectrum in the various structural components vary significantly from the core to the pressure vessel. Representative values of the neutron flux distribution and spectrum are presented in Table 4.3-6.

As discussed in Section 5.3, the irradiation surveillance program utilizes actual test samples to verify the accuracy of the calculated fluxes at the vessel.

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4.3.3 Analytical Methods

Calculations required in nuclear design consist of three distinct types, which are performed in sequence:

1. Determination of effective fuel temperatures

2. Generation of few-group cross sections

3. Space-dependent, few-group diffusion calculations

These calculations are carried out by computer codes which can be executed individually. Most of the codes required have been linked to form an automated design sequence which minimizes design time, avoids errors in transcription of data, and standardizes the design methods.

4.3.3.1 Fuel Temperature (Doppler) Calculations

Temperatures vary radially within the fuel rod, depending on the heat generation rate in the pellet; the conductivity of the materials in the pellet, gap, and clad; and the temperature of the coolant.

The fuel temperatures for use in most nuclear design Doppler calculations are obtained from a simplified version of the Westinghouse fuel rod design model described in Subsection 4.2.1.3, which considers the effect of radial variation of pellet conductivity, expansion coefficient and heat generation rate, elastic deflection of the clad, and a gap conductance which depends on the initial fill gas, the hot open gap dimension, and the fraction of the pellet over which the gap is closed. The fraction of the gap assumed closed represents an empirical adjustment used to produce close agreement with observed reactivity data at beginning of life. Further gap closure occurs with burnup and accounts for the decrease in Doppler defect with burnup which has been observed in operating plants. For detailed calculations of the Doppler coefficient, such as for use in xenon stability calculations, a more sophisticated temperature model is used, which accounts for the effects of fuel swelling, fission gas release, and plastic clad deformation.

Radial power distributions in the pellet as a function of burnup are obtained from LASER (WCAP-6073, Reference 38) calculations.

The effective U-238 temperature for resonance absorption is obtained from the radial temperature distribution by applying a radially dependent weighing function. The weighing function was determined from REPAD (WCAP-2048, Reference 39) Monte Carlo calculations of resonance escape probabilities in several steady-state and transient temperature distributions. In each case, a flat pellet temperature was determined which produced the same resonance escape probability as the actual distribution. The weighing function was empirically determined from these results.

The effective Pu-240 temperature for resonance absorption is determined by a convolution of the radial distribution of Pu-240 densities from LASER burnup calculations and the radial weighing function. The resulting temperature is burnup dependent, but the difference between U-238 and Pu-240 temperatures, in terms of reactivity effects, is small.

The effective pellet temperature for pellet dimensional change is that value which produces the same outer pellet radius in a virgin pellet as that obtained from the temperature model. The effective clad temperature for dimensional change is its average value.

The temperature calculational model has been validated by plant Doppler defect data, as shown in Table 4.3-7, and Doppler coefficient data, as shown in Figure 4.3-32. Stability index measurements also provide a sensitive measure of the Doppler coefficient near full power (Subsection 4.3.2.7).

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4.3.3.2 Macroscopic Group Constants

PHOENIX-P (WCAP-11596-P-A, Reference 40) and PARAGON (WCAP-16045-P-A, Reference 69) have been used for generating the macroscopic cross sections needed for the spatial few group codes. PHOENIX-P, PARAGON, or other NRC approved lattice codes will be used for reload designs.

PHOENIX-P has been approved by the NRC as a lattice code for the generation of macroscopic and microscopic few group cross sections for PWR analysis. (See WCAP-11596-P-A, Reference 40). PHOENIX-P is a two-dimensional, multigroup, transport-based lattice code capable of providing necessary data for PWR analysis. Since it is a dimensional lattice code, PHOENIX-P does not rely on pre-determined spatial/spectral interaction assumptions for the heterogeneous fuel lattice and can provide a more accurate multigroup spatial flux solution than versions (ARK) of LEOPARD/CINDER.

The solution for the detailed spatial flux and energy distribution is divided into two major steps in PHOENIX-P (See References 40 and 41). First, a two-dimensional fine energy group nodal solution is obtained, coupling individual subcell regions (e.g., pellet, clad and moderator) as well as surrounding pins, using a method based on Carlvik's collision probability approach and heterogeneous response fluxes which preserve the heterogeneous nature of the pin cells and their surroundings. The nodal solution provides an accurate and detailed local flux distribution, which is then used to homogenize the pin cells spatially to few groups.

Then, a standard S_4 discrete ordinates calculation solves for the angular distribution, based on the group-collapsed and homogenized cross sections from the first step. These S_4 fluxes normalize the detailed spatial and energy nodal fluxes, which are then used to compute reaction rates, power distributions and to deplete the fuel and burnable absorbers. A standard B1 calculation evaluates the fundamental mode critical spectrum, providing an improved fast diffusion coefficient for the core spatial codes.

PHOENIX-P employs a 70 energy group library derived mainly from the ENDF/B-VI files (Reference 71). This library was designed to capture the integral properties of the multigroup data properly during group collapse and to model important resonance parameters properly. It contains neutronics data necessary for modelling fuel, fission products, cladding and structural materials, coolant, and control and burnable absorber materials present in PWRs.

Group constants for burnable absorber cells, RCCA cells, guide thimbles and instrumentation thimbles, or other non-fuel cells, can be obtained directly from PHOENIX-P without any adjustments such as those required in the cell or 1D lattice codes.

PHOENIX-P has been validated through an extensive qualification effort which includes calculation-measurement comparison of the Strawbridge-Barry critical experiments (See References 42 and 43), the KRITZ high temperature criticals (Reference 44), the AEC sponsored B&W criticals (References 45 through 47) and measured actinide isotopic data from fuel pins irradiated in the Saxton and Yankee Rowe cores (References 48 through 52). In addition, calculation-measurement comparisons have been made to operating reactor data measured during startup tests and during normal power operation.

Validation of the cross section method is based on analysis of critical experiments, isotopic data, plant critical boron concentration data, and control rod worth measurement data such as that shown in Table 4.3-8.

Confirmatory critical experiments on burnable absorber rods are described in WCAP-7806 (Reference 42).

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Group constants for tungsten GRCAs are generated using the PARAGON lattice code. Like PHOENIX-P, PARAGON is a two-dimensional, multigroup, transport-based lattice code capable of providing necessary data for PWR analysis and is approved by the NRC in Reference 69. WCAP-16943-P-A (Reference 70) contains a description of the nuclear methods for modeling tungsten and PARAGON benchmark results to Monte-Carlo simulations for assemblies containing tungsten GRCAs.

The PARAGON lattice code is also capable of generating all of the group constants generated by PHOENIX-P, and has been benchmarked and qualified to the same degree as PHOENIX-P. The NRC has approved the use of PARAGON as an alternative method for generating all macroscopic and microscopic group constants for uranium fueled cores (Reference 69). The primary difference between PARAGON and PHOENIX-P is that PARAGON uses Collision Probability theory with the interface current method to solve the integral transport equation. PARAGON also allows increased flexibility in modeling the exact assembly and pin cell geometry. The group constants generated by PARAGON are coupled to the spatial few-group code using the NEXUS nuclear data methodology (Reference 72).

4.3.3.3 Spatial Few-Group Diffusion Calculations

The 3D ANC code (see WCAP-10965-P-A, References 57 and 73) permits the introduction of advanced fuel designs with axial heterogeneities, such as axial blankets and part-length burnable absorbers, and allows such features to be modeled explicitly. The three dimensional nature of this code provides both radial and axial power distribution. For some applications, the updated version APOLLO (see WCAP-13524 Reference 60) of the PANDA code (see WCAP-7048-P-A Reference 35) may be used for axial calculations, and a two-dimensional collapse of 3D ANC that properly accounts for the three-dimensional features of the fuel may be used for X-Y calculations.

Spatial few group calculations are carried out to determine the critical boron concentrations and power distributions. The moderator coefficient is evaluated by varying the inlet temperature in the same kind of calculations as those used for power distribution and reactivity predictions.

Validation of the reactivity calculations is associated with validation of the group constants themselves, as discussed in Subsection 4.3.3.2. Validation of the Doppler calculations is associated with the fuel temperature validation discussed in Subsection 4.3.3.1. Validation of the moderator coefficient calculations is obtained by comparison with plant measurements at hot zero power conditions, similar to that shown in Table 4.3-9.

Axial calculations may be used in place of the full three-dimensional nodal model to determine differential control rod worth curves (reactivity versus rod insertion) and to demonstrate load follow capability. Group constants are obtained from the three-dimensional nodal model by flux-volume weighing on an axial slicewise basis. Radial bucklings are determined by varying parameters in the buckling model while forcing the one-dimensional model to reproduce the axial characteristics (axial offset, midplane power) of the three-dimensional model.

Validation of the spatial codes for calculating power distributions involves the use of in-core and ex-core detectors and is discussed in Subsection 4.3.2.2.7.

As discussed in Subsection 4.3.3.2, calculation-measurement comparisons have been made to operating reactor data measured during startup tests and during normal power operation. These comparisons include a variety of core geometries and fuel loading patterns, and incorporate a reasonable extreme range of fuel enrichment, burnable absorber loading, and cycle burnup. Qualification data identified in References 40, 69, and 72 indicate small mean and standard deviations relative to measurement which are equal to or less than those found in previous reviews of

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similar or parallel approved methodologies. For the reload designs the spatial codes described above, other NRC approved codes, or both are used.

4.3.4 Combined License Information

Changes to the reference design of the fuel, burnable absorber rods, rod cluster assemblies, or initial core design from that presented in the DCD are addressed in APP-GW-GLR-059 (Reference 64).

4.3.5 References

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Table 4.3-1 (Sheet 1 of 3)
[*REACTOR CORE DESCRIPTION (FIRST CYCLE)*]*

Active core	
Equivalent diameter (in.)	*119.7*
Active fuel height first core (in.), cold	*168*
Height-to-diameter ratio	*1.40*
Total cross section area (ft^2)	*78.14*
H_2O/U molecular ratio, cell, cold	*2.40*
Reflector thickness and composition	
Top - water plus steel (in.)	*~10*
Bottom - water plus steel (in.)	*~10*
Side - water plus steel (in.)	*~15*
Fuel assemblies	
Number	*157*
Rod array	*17 x 17*
Rods per assembly	*264*
Rod pitch (in.)	*0.496*
Overall transverse dimensions (in.)	*8.426 x 8.426*
Fuel weight, as UO_2 (lb)	*211,588*
Zircaloy clad weight (lb)	*43,105*
Number of grids per assembly	
Top and bottom - (Ni-Cr-Fe Alloy 718)	*2[(a)]*
Intermediate	*8 ZIRLO®*
Intermediate flow mixing (IFM)	*4 ZIRLO*
Protective	*1 (Ni-Cr-Fe Alloy 718)*
Number of guide thimbles per assembly	*24*
Composition of guide thimbles	*ZIRLO*
Diameter of guide thimbles, upper part (in.)	*0.442 ID x 0.482 OD*
Diameter of guide thimbles, lower part (in.)	*0.397 ID x 0.482 OD*
Diameter of instrument guide thimbles (in.)	*0.442 ID x 0.482 OD*

Note:
(a) The top and bottom grids will be fabricated of nickel-chromium-iron Alloy 718.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 4.3-1 (Sheet 2 of 3)
[*REACTOR CORE DESCRIPTION (FIRST CYCLE)*]*

Fuel rods	
Number	*41,448*
Outside diameter (in.)	*0.374*
Diameter gap (in.)	*0.0065*
Clad thickness (in.)	*0.0225*
Clad material	*ZIRLO*
Fuel pellets	
Material	*UO_2 sintered*
Density (% of theoretical) (nominal)	*95.5*
Fuel enrichments, first core (average weight %, midzone / blanket)	
Region 1	*0.74 / ---*
Region 2	*1.58 / ---*
Region 3	*3.20 / 1.58*
Region 4	*3.776 / 3.20*
Region 5	*4.376 / 3.20*
Diameter (in.)	*0.3225*
Length (in.)	*0.387*
Mass of UO_2 per ft of fuel rod (lb/ft)	*0.366*
Rod Cluster Control Assemblies	
Neutron absorber	*Ag-In-Cd*
Diameter (in.)	*0.341*
Density (lb/in.^{3})	*Ag-In-Cd 0.367*
Cladding material	*Type 304 or 304L, cold-worked SS*
Cladding OD (in.)	*0.381*
Cladding thickness (in.)	*0.0185*
Number of clusters, full-length	*53*
Number of absorber rods per cluster	*24*
Gray Rod Cluster Assemblies	
Neutron absorber	*Tungsten/Alloy 718*
Diameter (in.)	*Tungsten 0.197 / Alloy 718 0.310*
Density (lb/in.^{3})	*Tungsten 0.695 / Alloy 718 0.296*
Cladding material	*Type 304 or 304L, cold-worked SS*
Cladding OD (in.)	*0.381*
Cladding thickness (in.)	*0.0255*
Number of clusters, full-length	*16*
Number of absorber rods per cluster	*24*

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Table 4.3-1 (Sheet 3 of 3)
[*REACTOR CORE DESCRIPTION (FIRST CYCLE)*]*

Discrete Burnable absorber rods (first core)	
Number	*592*
Material	*Alumina Boron-Carbide*
OD (in.)	*0.381*
Inner tube, OD (in.)	*0.267*
Clad material	*Zircaloy*
Inner tube material	*Zircaloy*
B_{10} content (mg/cm)	*6.03*
Absorber length (in.)	*See Figure 4.3-4b*
Integral Fuel Burnable Absorbers (first core)	
Number	*5632*
Type	*IFBA*
Material	*Boride Coating*
B_{10} Content (Mg/cm)	*0.773*
Absorber length (in.)	*152*
Excess reactivity	
Maximum fuel assembly $K\infty$ (cold, clean, unborated water)	*1.392*
Maximum core reactivity K_{eff} (cold, zero power, beginning of cycle, zero soluble boron)	*1.201*

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Table 4.3-2 (Sheet 1 of 2)
[*NUCLEAR DESIGN PARAMETERS (FIRST CYCLE)*]*

Core average linear power, including densification effects (kW/ft)		*5.72*
Total heat flux hot channel factor, F_Q		*≤2.60*
Nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$		*≤1.72*
Reactivity coefficients [(a)]	*Design Limits*	*Typical Best Estimate*
Doppler-only power coefficients (see Figure 15.0.4-1) (pcm/% power)[(b)]		
Upper curve	*-19.4 to -12.6*	*-14.6 to -9.0*
Lower curve	*-10.2 to -6.7*	*-12.4 to -8.9*
Doppler temperature coefficient (pcm/°F)[(b)]	*-3.5 to -1.0*	*-2.1 to -1.4*
Moderator temperature coefficient (pcm/°F)[(b)]	*0 to -40*	*0 to -35*
Boron coefficient (pcm/ppm)[(b)]	*-13.5 to -5.0*	*-11.3 to -7.2*
Rodded moderator density coefficient (pcm/g/cm^3)[(b)]	*≤ 0.47x10^5*	*≤ 0.45x10^5*
Delayed neutron fraction and lifetime, β_{eff}		*0.0075(0.0044)[(c)]*
Prompt Neutron Lifetime, ℓ^*, μs		*19.8*
Control rods		
Rod requirements		*See Table 4.3-3*
Maximum ejected rod worth		*See Chapter 15*
Typical Bank worth HZP no overlap (pcm)[(b)]	*BOL, Xe Free*	*EOL, Eq. Xe*
MA Bank	*238*	*257*
MB Bank	*248*	*327*
MC Bank	*232*	*194*
MD Bank	*239*	*271*
M1 Bank	*686*	*757*
M2 Bank	*1363*	*1031*
AO Bank	*1627*	*1544*

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Table 4.3-2 (Sheet 2 of 2)
[*NUCLEAR DESIGN PARAMETERS (FIRST CYCLE)*]*

Typical Hot Channel Factors $F_{\Delta H}^{N}$ [(g)]	*BOL*	*EOL*
Unrodded	*1.44*	*1.38*
MA bank	*1.48*	*1.44*
MA + MB banks	*1.51*	*1.43*
MA + MB + MC banks	*1.51*	*1.42*
MA + MB + MC + MD banks	*1.54*	*1.47*
MA + MB + MC + MD + M1 banks	*1.63*	*1.53*
AO bank	*1.68*	*1.61*

Typical Boron concentrations (ppm)	
Zero power, k_{eff} = 0.99, cold[(d)] RCCAs out	*1427*
Zero power, k_{eff} = 0.99, hot[(e)] RCCAs out	*1429*
Design basis refueling boron concentration	*2700*
Zero power, $k_{eff} \leq 0.95$, cold[(d)] RCCAs in	*1061*
Zero power, k_{eff} = 1.00, hot[(e)] RCCAs out	*1321*
Full power, no xenon, k_{eff} = 1.0, hot RCCAs out	*1160*
Full power, equilibrium xenon, k = 1.0, hot RCCAs out	*844*
Reduction with fuel burnup	
First cycle (ppm/(GWD/MTU))[(f)]	*See Figure 4.3-3*
Reload cycle (ppm/(GWD/MTU))	*~40*

Notes:

(a) Uncertainties are given in Subsection 4.3.3.3.

(b) 1 pcm = 10^{-5} Δρ where Δρ is calculated from two statepoint values of k_{eff} by ln (k_1/k_2).

(c) Bounding lower value used for safety analysis.

(d) Cold means 68°F, 1 atm.

(e) Hot means 557°F, 2250 psia.

(f) 1 GWD = 1000 MWD. During the first cycle, a large complement of burnable absorbers is present which significantly reduce the boron depletion rate compared to reload cycles.

(g) Rodded hot channel factors reflect full insertion of each bank at hot full power conditions. Rod Insertion limits for the first cycle prohibit full insertion of the M1 and AO-banks during full power operation. The Rodded hot channel factors for these conditions are therefore not indicative of permitted operating conditions at full rated thermal power.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 4.3-3
[*REACTIVITY REQUIREMENTS FOR ROD CLUSTER CONTROL ASSEMBLIES*]*

Requirement	*First Cycle BOL Worths (%Δρ)*	*First Cycle EOL Worth (%Δρ)*	*Equilibrium Cycle EOL Representative Worths (%Δρ)*
(1) Total power defect (%Δρ)[(a)]	*1.66*	*3.14*	*3.50*
Trip rod worth[(b)]	*7.24*	*6.02*	*6.41*
(2) Less 7%[(c)]	*6.73*	*5.60*	*5.96*
Shutdown Margin			
Calculated margin (2) – (1)	*5.07*	*2.46*	*2.46*
Required shutdown margin[(d)]	*1.60*	*1.60*	*1.60*

Notes:

(a) Includes Doppler, Moderator Temperature, Redistribution, and Void collapse reactivity effects associated with reducing power from full power to zero. Also includes the effect of inserted control rods at the most limiting allowed insertion point on the total power defect.

(b) Negative reactivity inserted by RCCAs on the reactor trip. Assumes RCCAs start at most limiting allowed insertion point and fully insert on the reactor trip except for the highest worth stuck RCCA. Also conservatively excludes negative reactivity from withdrawn GRCAs which are designed to insert on the reactor trip.

(c) 7 percent adjustment to accommodate uncertainties (this assumes the use of Ag-In-Cd RCCAs).

(d) The design basis minimum shutdown margin is 1.60 percent.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 4.3-4 Not Used

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Table 4.3-5
Stability Index for Pressurized Water Reactor Cores with a 12-Foot Height

Burnup (MWD/MTU)	F_Z	C_B (ppm)	Axial Stability Index (h^{-1})	
			Experiment	Calculated
1550	1.34	1065	-0.0410	-0.0320
7700	1.27	700	-0.0140	-0.0060
5090[(a)]			-0.0325	-0.0255
			Radial Stability Index (h^{-1})	
			Experiment	Calculated
2250[(b)]			-0.0680	-0.0700

Notes:
(a) Four-loop plant, 12-foot core in cycle 1, axial stability test
(b) Four-loop plant, 12-foot core in cycle 1, radial (X-Y) stability test

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Table 4.3-6
Typical Neutron Flux Levels (n/cm^2/s) at Full Power

	E ≥ 1.0 MeV	**1.00 MeV > E ≥5.53 KeV**	**5.53 KeV > E ≥0.625 eV**	**E < 0.625 eV**
Core center	1.12×10^{14}	1.76×10^{14}	1.28×10^{14}	5.47×10^{13}
Core outer radius at midheight	3.86×10^{13}	6.08×10^{13}	4.42×10^{13}	1.83×10^{13}
Core top, on axis	3.02×10^{13}	4.75×10^{13}	3.46×10^{13}	2.17×10^{13}
Core bottom, on axis	2.92×10^{13}	4.59×10^{13}	3.34×10^{13}	2.40×10^{13}
Pressure vessel ID azimuthal peak	4.71×10^{10}	8.4×10^{10}	5.56×10^{10}	5.32×10^{10}

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Table 4.3-7
Comparison of Measured and Calculated Doppler Defects

Plant	Fuel	Core Burnup (MWD/MTU)	Measured (pcm)[(a)]	Calculated (pcm)
1	Air filled	1800	1700	1710
2	Air filled	7700	1300	1440
3	Air and helium filled	8460	1200	1210

Note:
(a) pcm = 10^5 x ln (k_2/k_1)

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Table 4.3-8
Comparison of Measured and Calculated AG-in-CD Rod Worth

2-Loop Plant, 121 Assemblies, 10-ft Core		Measured (pcm)	Calculated (pcm)
Group B		1885	1893
Group A		1530	1649
Shutdown group		3050	2917
ESADA critical, 0.69-in. pitch[a] 2 w/o PuO_2, 8% Pu-240, 9 control rods			
6.21-in. rod separation		2250	2250
2.07-in. rod separation		4220	4160
1.38-in. rod separation		4100	4019
Benchmark Critical Experiment Hafnium Control Rod Worth			
Control Rod Configuration	**No. of Fuel Rods**	**Measured[b] Worth (Δppm B-10)**	**Calculated[b] Worth (Δppm B-10)**
9 hafnium rods	1192	138.3	141.0

Notes:
(a) Report in WCAP-3726-1 (Reference 58).
(b) Calculated and measured worth are given in terms of an equivalent charge in B-10 concentration.

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Table 4.3-9
Comparison of Measured and Calculated Moderator Coefficients at HZP, BOL

Plant Type/ Control Bank Configuration	Measured α_{iso} [(a)] (pcm/°F)	Calculated α_{iso} (pcm/°F)
3-loop, 157-assembly, 12-ft core		
D at 160 steps	-0.50	-0.50
D in, C at 190 steps	-3.01	-2.75
D in, C at 28 steps	-7.67	-7.02
B, C, and D in	-5.16	-4.45
2-loop, 121-assembly, 12-ft core		
D at 180 steps	+0.85	+1.02
D in, C at 180 steps	-2.40	-1.90
C and D in, B at 165 steps	-4.40	-5.58
B, C, and D in, A at 174 steps	-8.70	-8.12
4-loop, 193-assembly, 12-ft core		
ARO	-0.52	-1.2
D in	-4.35	-5.7
D and C in	-8.59	-10.0
D, C, and B in	-10.14	-10.55
D, C, B, and A in	-14.63	-14.45

Note:
(a) Isothermal coefficients, which include the Doppler effect in the fuel

$$\alpha_{iso} = 10^5 \ln \frac{k_2}{k_1} / \Delta T\,^{\circ}F$$

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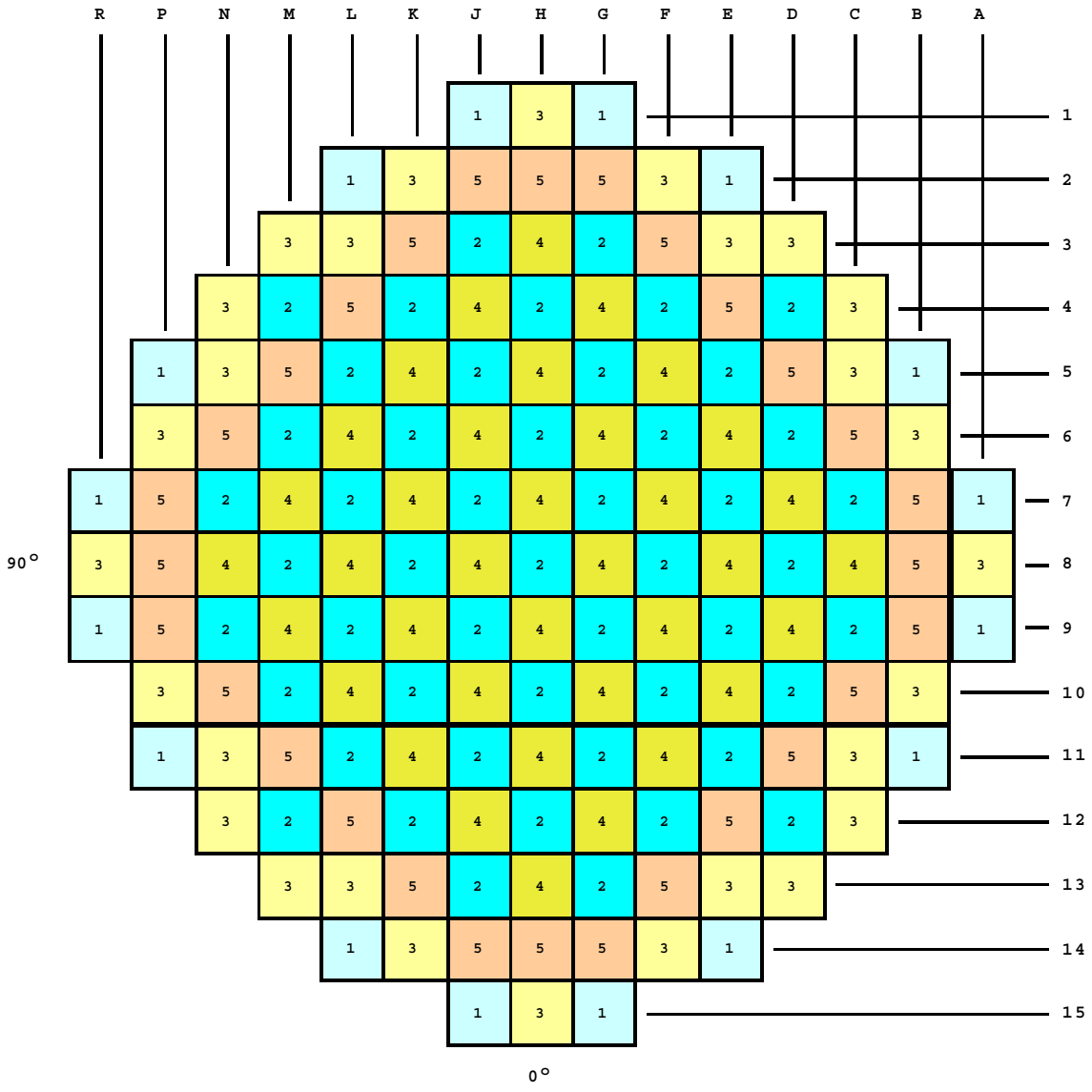

Region	Enrichment
1	0.74 w/o
2	1.58 w/o
3	3.20 w/o
4	3.776 w/o
5	4.376 w/o

Figure 4.3-1
Fuel Loading Arrangement for First Core

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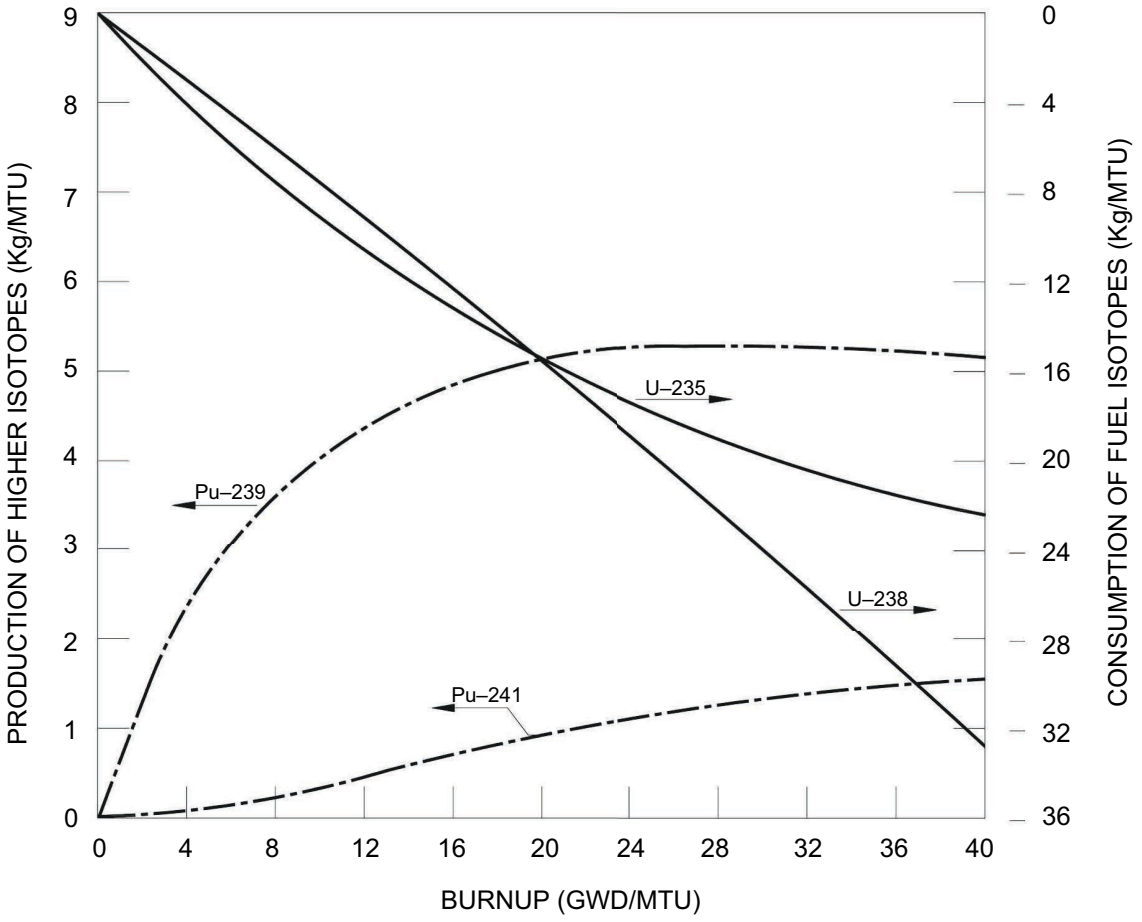

Figure 4.3-2
Typical Production and Consumption of Higher Isotopes

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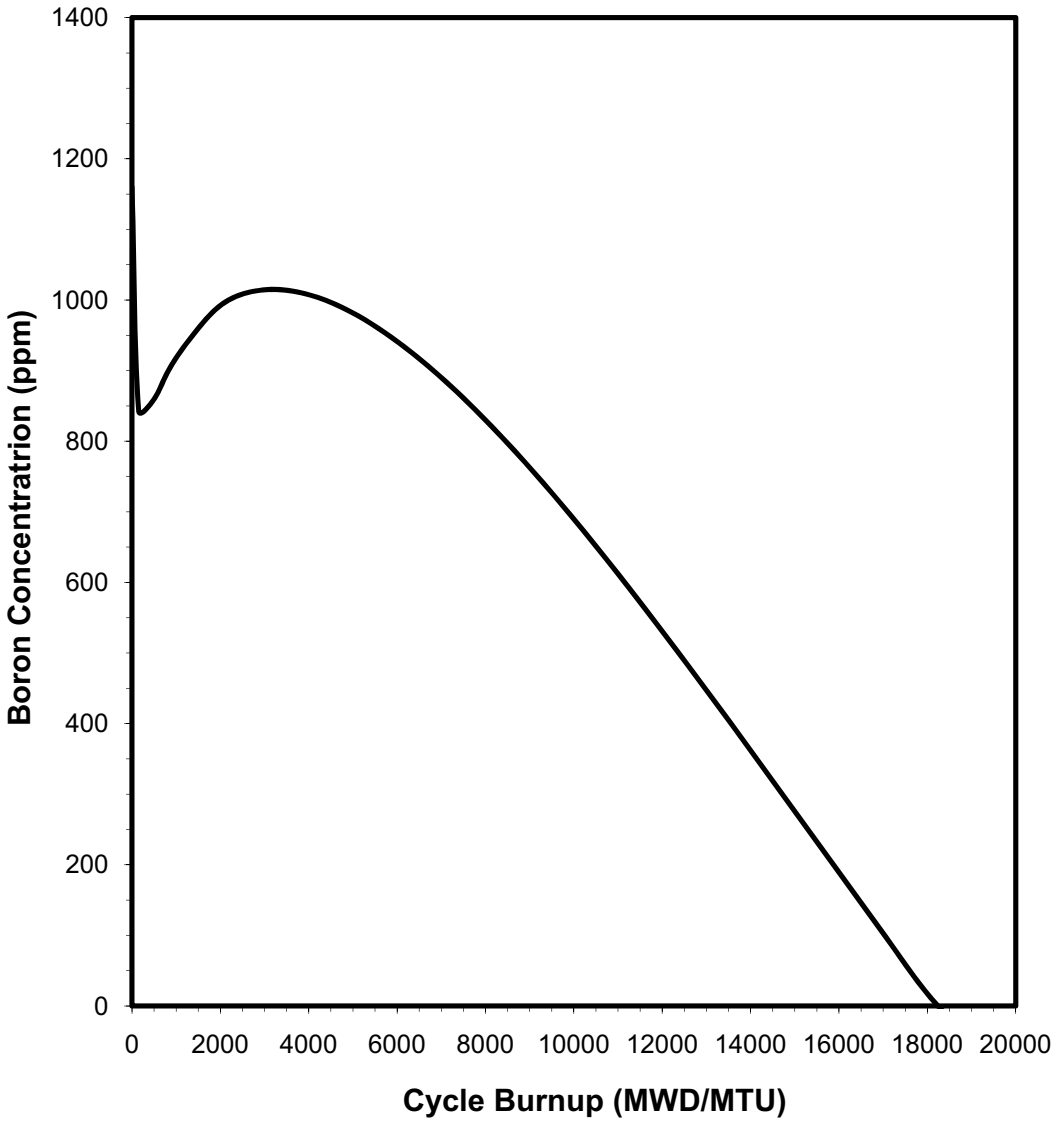

Figure 4.3-3
Cycle 1 Soluble Boron Concentration Versus Burnup

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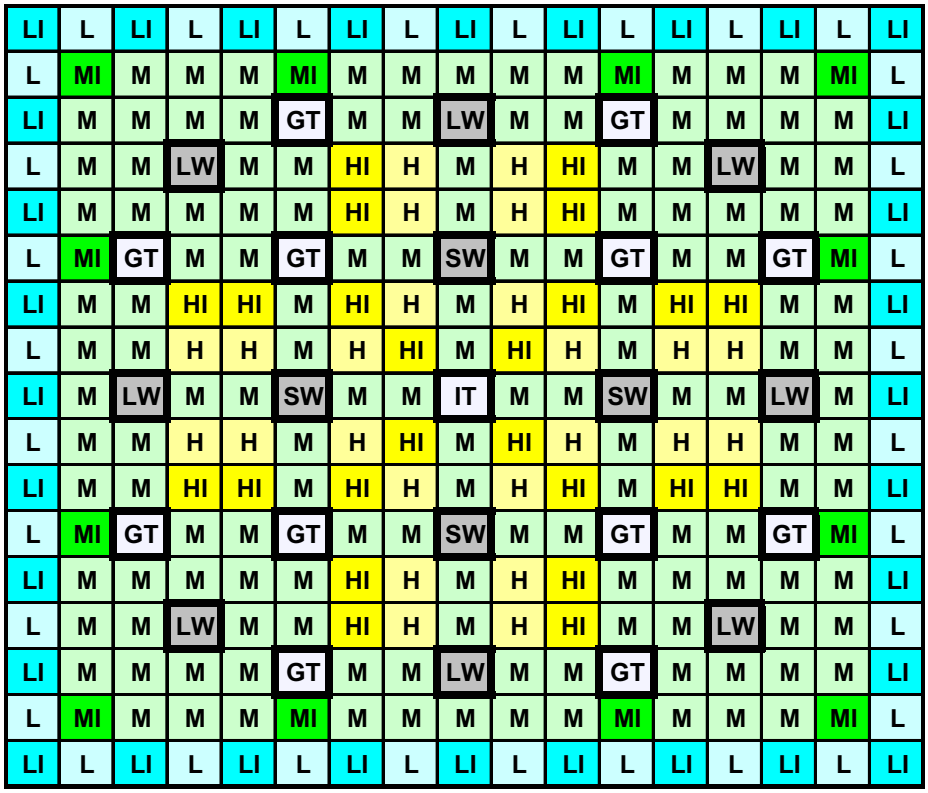

Region D 68 IFBA + 12 WABA

	No. Pins	Enr.	BA
L	32	3.40	No BA
LI	32	3.40	IFBA
M	140	3.80	No BA
MI	12	3.80	IFBA
H	24	4.20	No BA
HI	24	4.20	IFBA
SW	4		WABA
LW	8		WABA
GT	12		
IT	1		

Figure 4.3-4a (Sheet 1 of 4)
Cycle 1 Assembly Integral and Wet Annular Burnable Absorber Patterns

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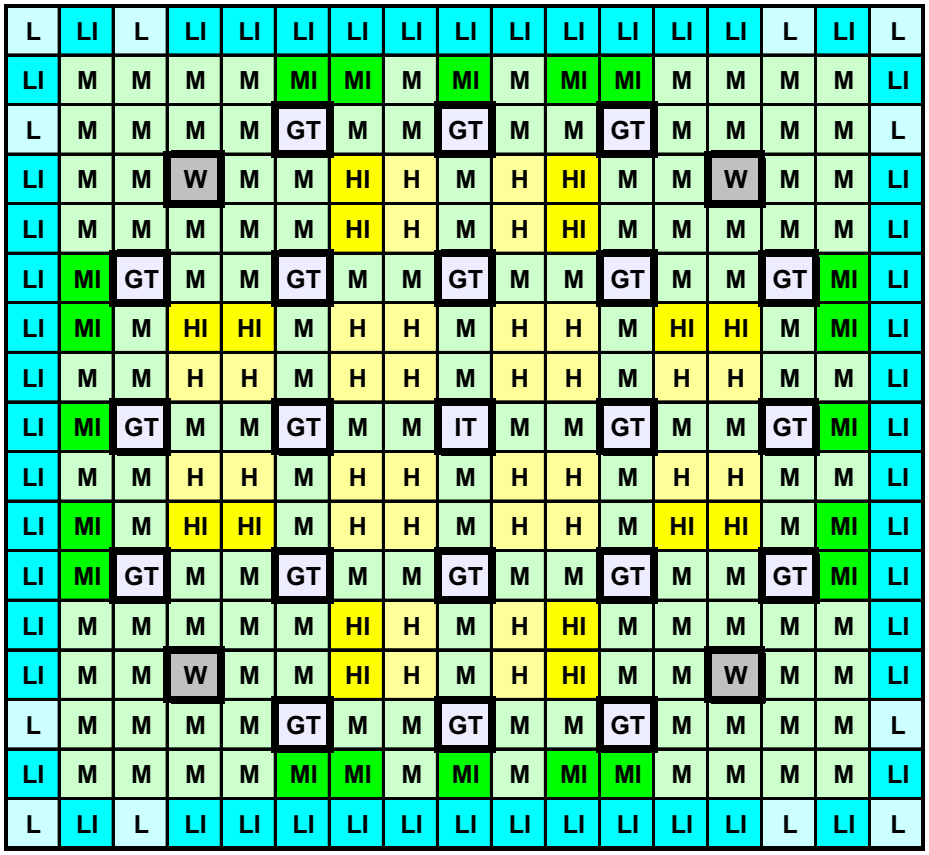

Region E 88 IFBA + 4 WABA

	No. Pins	Enr.	BA
L	12	4.00	No BA
LI	52	4.00	IFBA
M	132	4.40	No BA
MI	20	4.40	IFBA
H	32	4.80	No BA
HI	16	4.80	IFBA
W	4		WABA
GT	20		
IT	1		

Figure 4.3-4a (Sheet 2 of 4)
Cycle 1 Assembly Integral and Wet Annular Burnable Absorber Patterns

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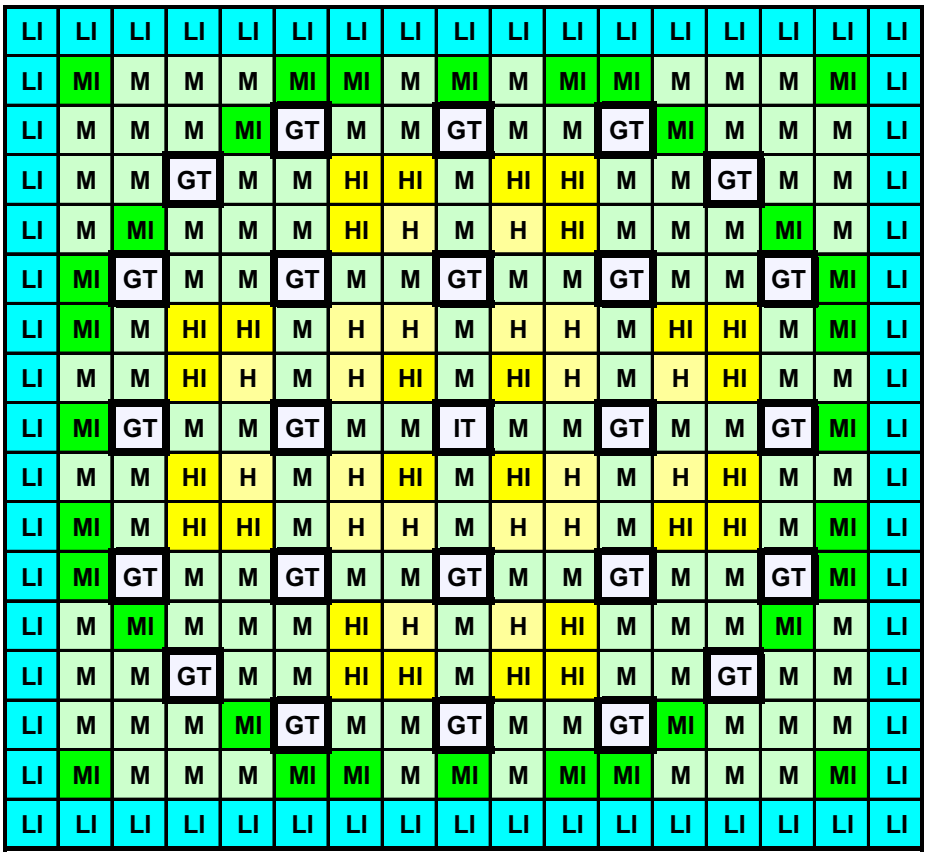

Region E 124 IFBA

	No. Pins	Enr.	BA
L	0	4.00	No BA
LI	64	4.00	IFBA
M	120	4.40	No BA
MI	32	4.40	IFBA
H	20	4.80	No BA
HI	28	4.80	IFBA
W	0		WABA
GT	24		
IT	1		

Figure 4.3-4a (Sheet 3 of 4)
Cycle 1 Assembly Integral and Wet Annular Burnable Absorber Patterns

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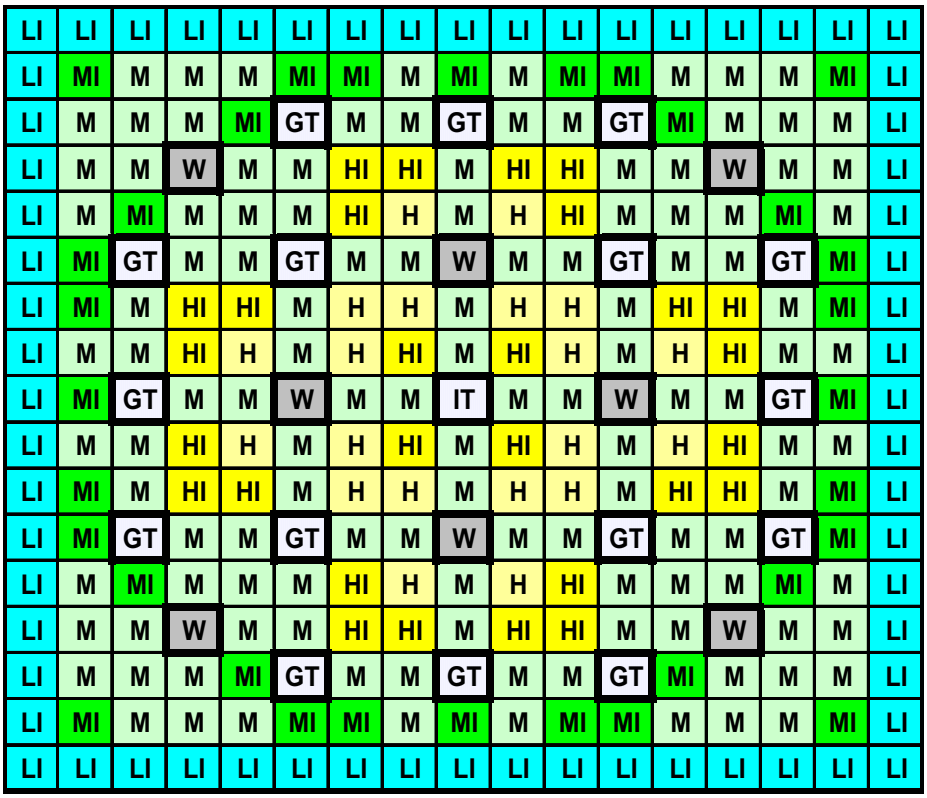

Region E 124 IFBA + 8 WABA

	No. Pins	Enr.	BA
L	0	4.00	No BA
LI	64	4.00	IFBA
M	120	4.40	No BA
MI	32	4.40	IFBA
H	20	4.80	No BA
HI	28	4.80	IFBA
W	8		WABA
GT	16		
IT	1		

Figure 4.3-4a (Sheet 4 of 4)
Cycle 1 Assembly Integral and Wet Annual Burnable Absorber Patterns

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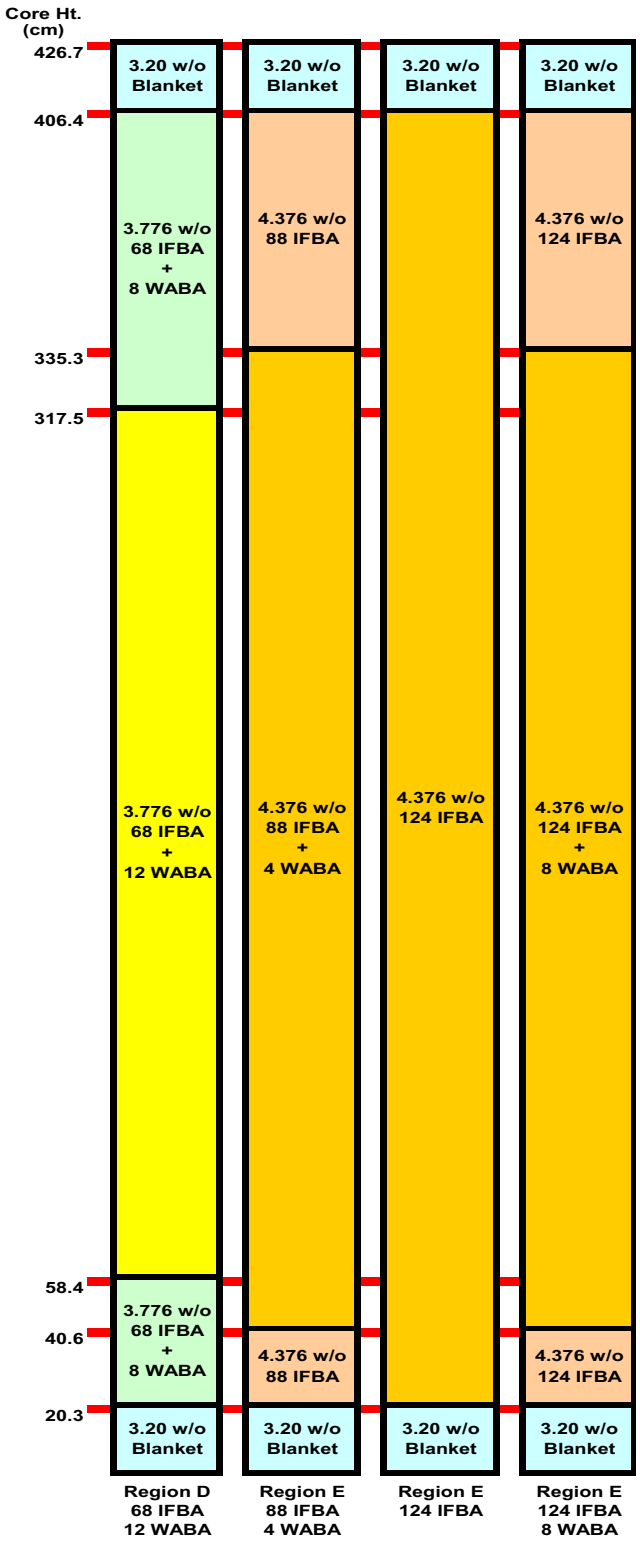

Figure 4.3-4b
Cycle 1 Assembly Integral and Wet Annular Burnable Absorber Axial Configurations

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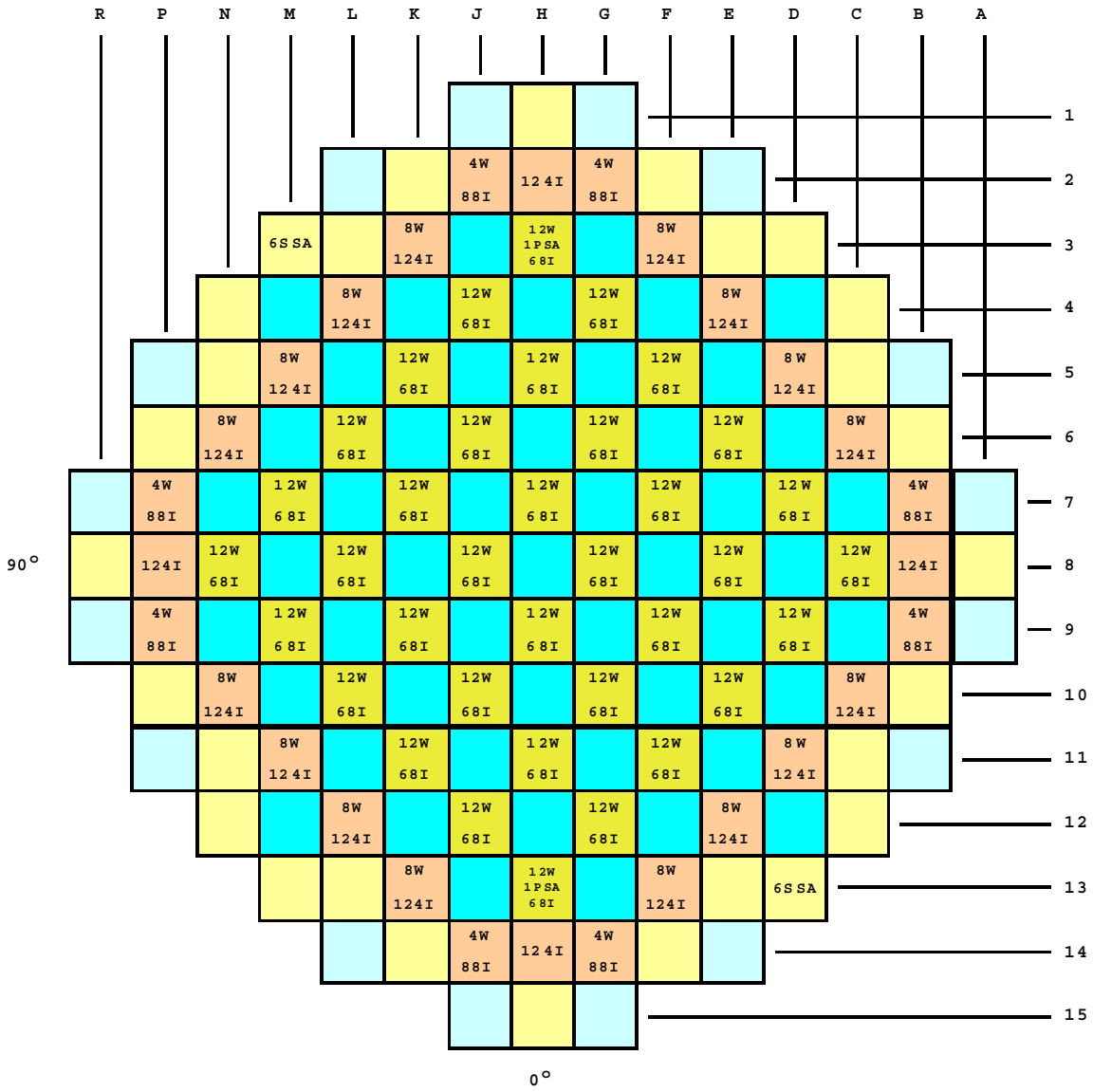

TYPE	TOTAL
##W...(NUMBER OF WABA RODLETS).....	592
##I...(TOTAL NUMBER OF FRESH IFBA RODS).....	5632
#SSA..(NUMBER OF SECONDARY SOURCE RODLETS)...	12
#PSA..(NUMBER OF PRIMARY SOURCE RODLETS).....	2

Figure 4.3-5
Cycle 1 Burnable Absorber, Primary, and Secondary Source Assembly Locations

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0.962	1.296	0.964	1.299	0.966	1.277	1.205	0.653
1.296	0.963	1.299	0.967	1.301	0.947	1.149	0.201
0.964	1.299	0.967	1.307	0.970	1.290	0.874	
1.299	0.967	1.307	0.963	1.348	1.164	0.232	
0.966	1.301	0.970	1.348	0.829	0.765		
1.277	0.947	1.290	1.164	0.765			
1.205	1.149	0.874	0.232				
0.653	0.201						

Calculated $F_{\Delta H}^{N} = 1.460$

Key: Values Represent Assembly Relative Power

Figure 4.3-6
Cycle 1
Normalized Power Density Distribution
Near Beginning of Life, Unrodded Core,
Hot Full Power, No Xenon

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1.002	1.327	0.998	1.316	0.982	1.258	1.171	0.647
1.327	1.001	1.323	0.992	1.300	0.947	1.119	0.207
0.998	1.323	0.996	1.314	0.977	1.262	0.856	
1.316	0.992	1.314	0.974	1.326	1.136	0.238	
0.982	1.300	0.977	1.326	0.829	0.752		
1.258	0.947	1.262	1.136	0.752			
1.171	1.119	0.856	0.238				
0.647	0.207						

Calculated $F^N_{\Delta H} = 1.441$

Key: Values Represent Assembly Relative Power

Figure 4.3-7
Cycle 1
Normalized Power Density Distribution
Near Beginning of Life, Unrodded Core,
Hot Full Power, Equilibrium Xenon

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1.029	1.348	1.009	1.338	1.004	1.224	0.928	0.566
1.348	1.001	1.283	0.997	1.330	0.949	1.063	0.191
1.009	1.283	0.794	1.288	1.013	1.306	0.874	
1.338	0.997	1.288	0.995	1.391	1.204	0.253	
1.004	1.330	1.013	1.391	0.888	0.808		
1.224	0.949	1.306	1.204	0.808			
0.928	1.063	0.874	0.253				
0.566	0.191						

Calculated $F^N_{\Delta H} = 1.505$

Key: Values Represent Assembly Relative Power

Figure 4.3-8
Cycle 1
Normalized Power Density Distribution
Near Beginning of Life, Gray Bank MA+MB Inserted,
Hot Full Power, Equilibrium Xenon

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1.017	1.350	1.017	1.349	1.014	1.339	1.260	0.660
1.350	1.017	1.349	1.014	1.338	0.979	1.144	0.275
1.017	1.349	1.013	1.338	0.991	1.269	0.790	
1.349	1.014	1.338	0.987	1.303	0.963	0.267	
1.014	1.338	0.991	1.303	0.756	0.599		
1.339	0.979	1.269	0.963	0.599			
1.260	1.144	0.790	0.267				
0.660	0.275						

Calculated $F^N_{\Delta H} = 1.428$

Key: Values Represent Assembly Relative Power

Figure 4.3-9
Cycle 1
Normalized Power Density Distribution
Near Middle of Life, Unrodded Core,
Hot Full Power, Equilibrium Xenon

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0.984	1.253	0.990	1.266	1.006	1.293	1.263	0.738
1.253	0.987	1.260	0.997	1.278	0.998	1.170	0.366
0.990	1.260	0.995	1.272	1.001	1.266	0.832	
1.266	0.997	1.272	0.998	1.287	0.981	0.346	
1.006	1.278	1.001	1.287	0.814	0.645		
1.293	0.998	1.266	0.981	0.645			
1.263	1.170	0.832	0.346				
0.738	0.366						

Calculated $F_{\Delta H}^{N} = 1.378$

Key: Values Represent Assembly Relative Power

Figure 4.3-10
Cycle 1
Normalized Power Density Distribution
Near End of Life, Unrodded Core,
Hot Full Power, Equilibrium Xenon

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1.017	1.284	1.006	1.296	1.027	1.253	0.989	0.645
1.284	0.993	1.234	1.006	1.310	0.997	1.106	0.335
1.006	1.234	0.802	1.258	1.039	1.312	0.848	
1.296	1.006	1.258	1.023	1.356	1.041	0.366	
1.027	1.310	1.039	1.356	0.872	0.695		
1.253	0.997	1.312	1.041	0.695			
0.989	1.106	0.848	0.366				
0.645	0.335						

Calculated $F^{N}_{\Delta H} = 1.431$

Key: Values Represent Assembly Relative Power

Figure 4.3-11
Cycle 1
Normalized Power Density Distribution
Near End of Life, Gray Bank MA+MB Inserted,
Hot Full Power, Equilibrium Xenon

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1.215	1.187	1.200	1.215	1.228	1.234	1.239	1.258	1.265	1.273	1.265	1.269	1.269	1.261	1.251	1.250	1.310
1.139	1.194	1.279	1.296	1.312	1.277	1.248	1.345	1.310	1.355	1.267	1.304	1.348	1.341	1.338	1.272	1.263
1.133	1.258	1.260	1.255	1.246		1.347	1.360		1.368	1.362		1.273	1.295	1.317	1.350	1.274
1.143	1.270	1.250		1.283	1.343	1.281	1.281	1.322	1.287	1.290	1.360	1.307		1.305	1.364	1.293
1.159	1.288	1.245	1.286	1.317	1.356	1.287	1.339	1.258	1.343	1.294	1.369	1.336	1.317	1.293	1.381	1.311
1.170	1.261		1.353	1.363		1.341	1.266		1.269	1.347		1.378	1.380		1.345	1.320
1.181	1.239	1.359	1.298	1.301	1.349	1.396	1.365	1.278	1.367	1.400	1.355	1.311	1.317	1.402	1.315	1.325
1.203	1.339	1.378	1.304	1.359	1.279	1.371	1.309	1.354	1.311	1.374	1.284	1.369	1.321	1.417	1.417	1.342
1.213	1.307		1.348	1.280		1.287	1.358		1.361	1.292		1.290	1.366		1.378	1.345
1.220	1.354	1.392	1.317	1.372	1.291	1.383	1.319	1.364	1.321	1.384	1.293	1.377	1.329	1.425	1.424	1.349
1.213	1.269	1.389	1.325	1.327	1.374	1.421	1.388	1.299	1.388	1.420	1.374	1.328	1.333	1.418	1.329	1.338
1.217	1.306		1.397	1.405		1.379	1.300		1.300	1.377		1.406	1.405		1.367	1.340
1.218	1.350	1.302	1.343	1.373	1.411	1.337	1.389	1.303	1.388	1.335	1.409	1.373	1.350	1.324	1.412	1.338
1.212	1.345	1.323		1.353	1.413	1.344	1.342	1.382	1.342	1.342	1.411	1.353		1.346	1.404	1.329
1.208	1.344	1.347	1.341	1.329		1.431	1.440		1.440	1.429		1.329	1.348	1.368	1.399	1.319
1.217	1.284	1.381	1.400	1.416	1.376	1.341	1.440	1.398	1.441	1.341	1.376	1.418	1.407	1.401	1.330	1.320
1.294	1.282	1.305	1.326	1.341	1.346	1.348	1.362	1.364	1.365	1.351	1.349	1.345	1.334	1.322	1.322	1.388

Figure 4.3-12
Rodwise Power Distribution in a Typical Assembly (M-5)
Near Beginning of Life
Hot Full Power, Equilibrium Xenon, Unrodded Core

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1.115	1.080	1.078	1.095	1.120	1.148	1.167	1.187	1.213	1.217	1.224	1.232	1.232	1.235	1.244	1.269	1.319
1.085	1.108	1.126	1.153	1.188	1.234	1.222	1.246	1.284	1.276	1.277	1.318	1.300	1.293	1.291	1.296	1.286
1.076	1.119	1.144	1.208	1.233		1.271	1.282		1.312	1.327		1.343	1.347	1.307	1.304	1.275
1.081	1.134	1.195		1.255	1.268	1.289	1.297	1.295	1.326	1.344	1.350	1.365		1.359	1.318	1.277
1.091	1.154	1.205	1.240	1.230	1.269	1.291	1.308	1.296	1.337	1.345	1.350	1.337	1.376	1.366	1.336	1.285
1.103	1.183		1.238	1.253		1.279	1.289		1.318	1.333		1.361	1.373		1.366	1.296
1.107	1.156	1.212	1.242	1.260	1.264	1.298	1.312	1.304	1.341	1.353	1.344	1.368	1.378	1.373	1.335	1.299
1.113	1.166	1.210	1.237	1.263	1.260	1.299	1.308	1.314	1.337	1.353	1.341	1.372	1.372	1.370	1.346	1.303
1.123	1.188		1.220	1.237		1.277	1.300		1.329	1.331		1.345	1.355		1.368	1.312
1.113	1.166	1.210	1.237	1.263	1.260	1.299	1.308	1.314	1.337	1.353	1.341	1.372	1.372	1.370	1.346	1.303
1.107	1.156	1.212	1.242	1.260	1.264	1.298	1.312	1.304	1.341	1.353	1.344	1.368	1.378	1.373	1.335	1.299
1.103	1.183		1.238	1.253		1.279	1.289		1.318	1.333		1.361	1.373		1.366	1.296
1.091	1.154	1.205	1.240	1.230	1.269	1.291	1.308	1.296	1.337	1.345	1.350	1.337	1.376	1.366	1.336	1.285
1.081	1.134	1.195		1.255	1.268	1.288	1.297	1.295	1.326	1.344	1.350	1.365		1.359	1.318	1.277
1.076	1.119	1.144	1.208	1.233		1.271	1.282		1.312	1.327		1.343	1.347	1.307	1.304	1.275
1.085	1.108	1.126	1.153	1.188	1.234	1.222	1.246	1.284	1.276	1.277	1.318	1.300	1.293	1.291	1.296	1.286
1.115	1.080	1.078	1.095	1.120	1.148	1.167	1.187	1.213	1.217	1.224	1.232	1.232	1.235	1.244	1.269	1.319

Figure 4.3-13
Rodwise Power Distribution in a Typical Assembly (P-8)
Near End of Life
Hot Full Power, Equilibrium Xenon, Unrodded Core

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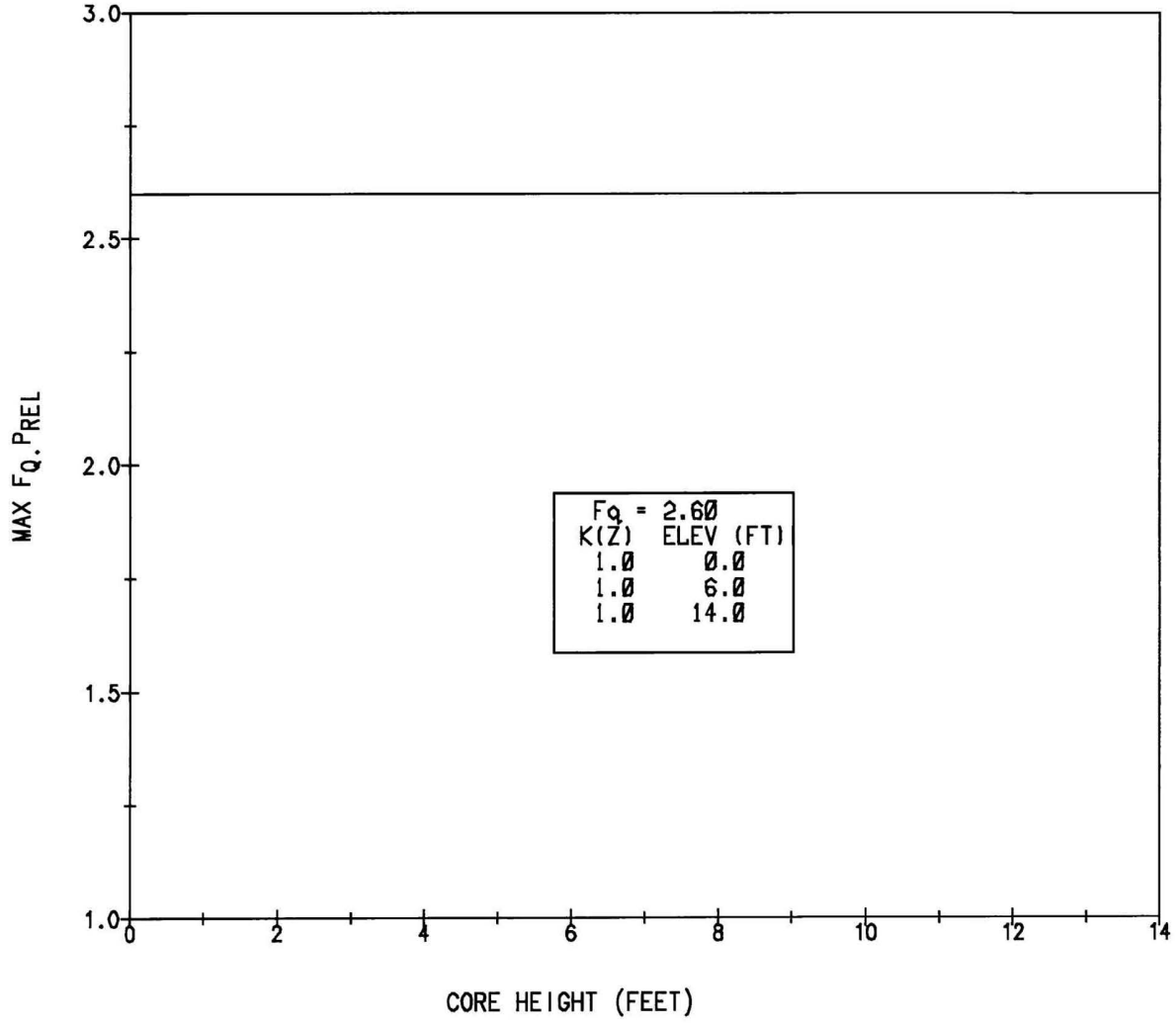

Figure 4.3-14
Maximum F_Q x Power Versus Axial Height
During Normal Operation

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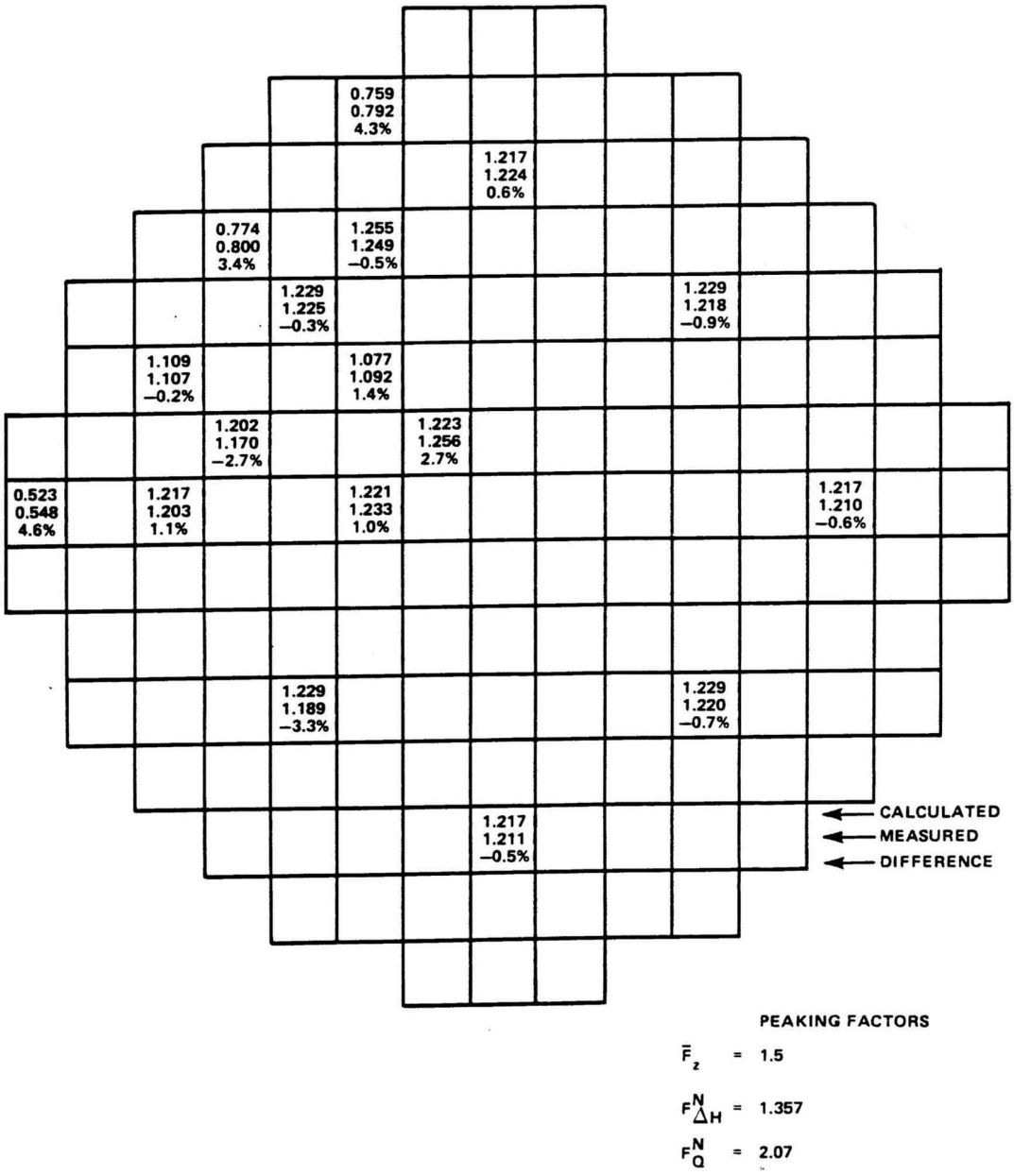

Figure 4.3-15
Typical Comparison Between Calculated and Measured Relative Fuel Assembly Power Distribution

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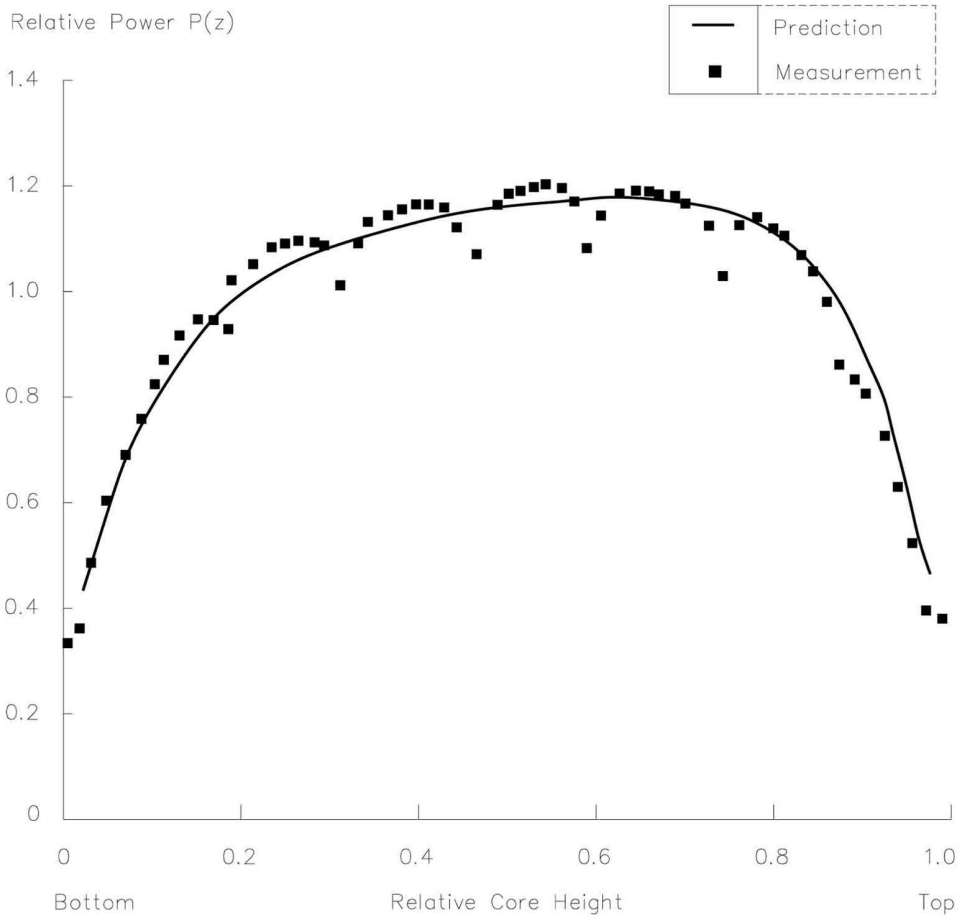

Figure 4.3-16
Typical Calculated Versus Measured Axial Power Distribution

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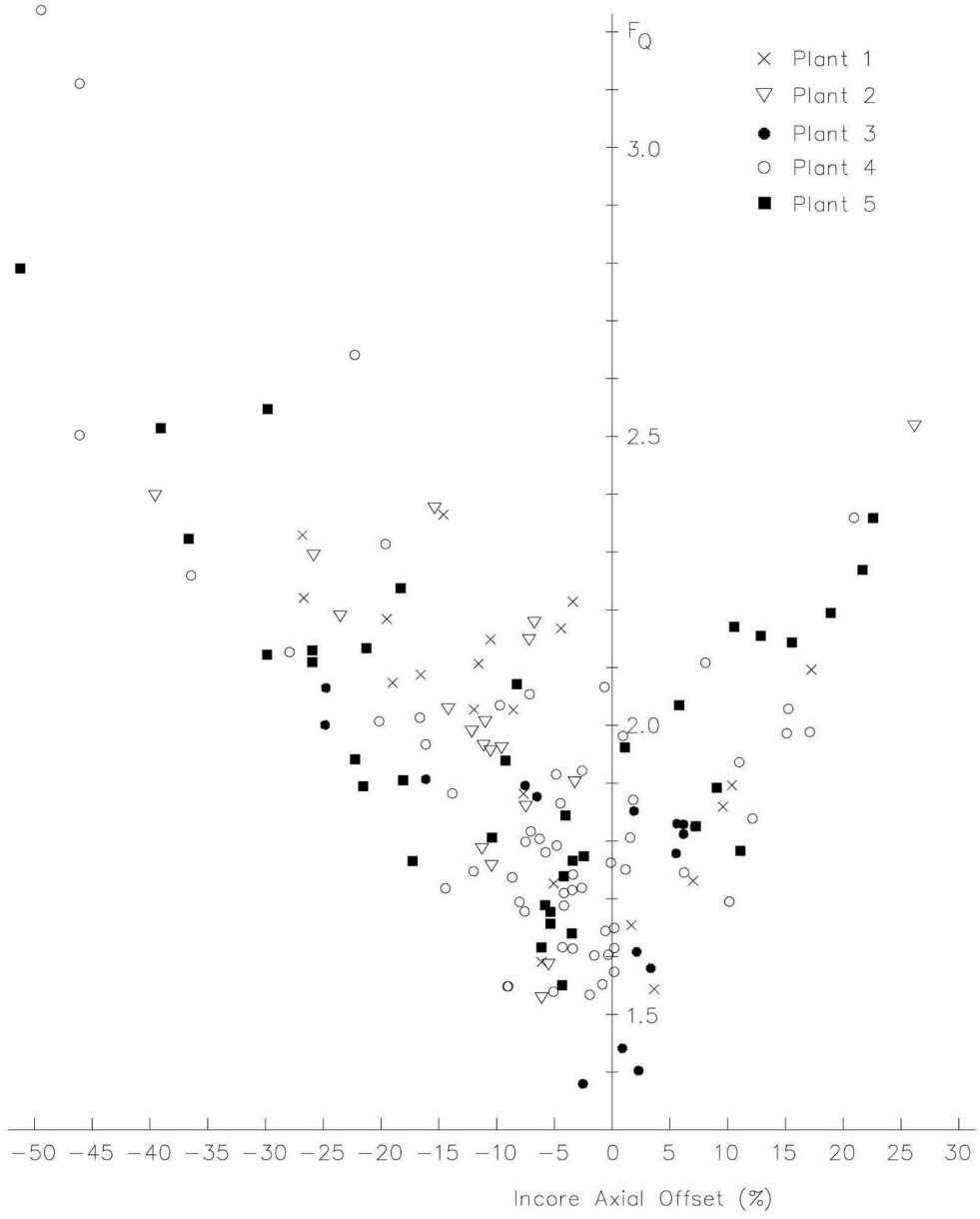

Figure 4.3-17
Measured F_Q Values Versus Axial Offset for Full Power Rod Configurations

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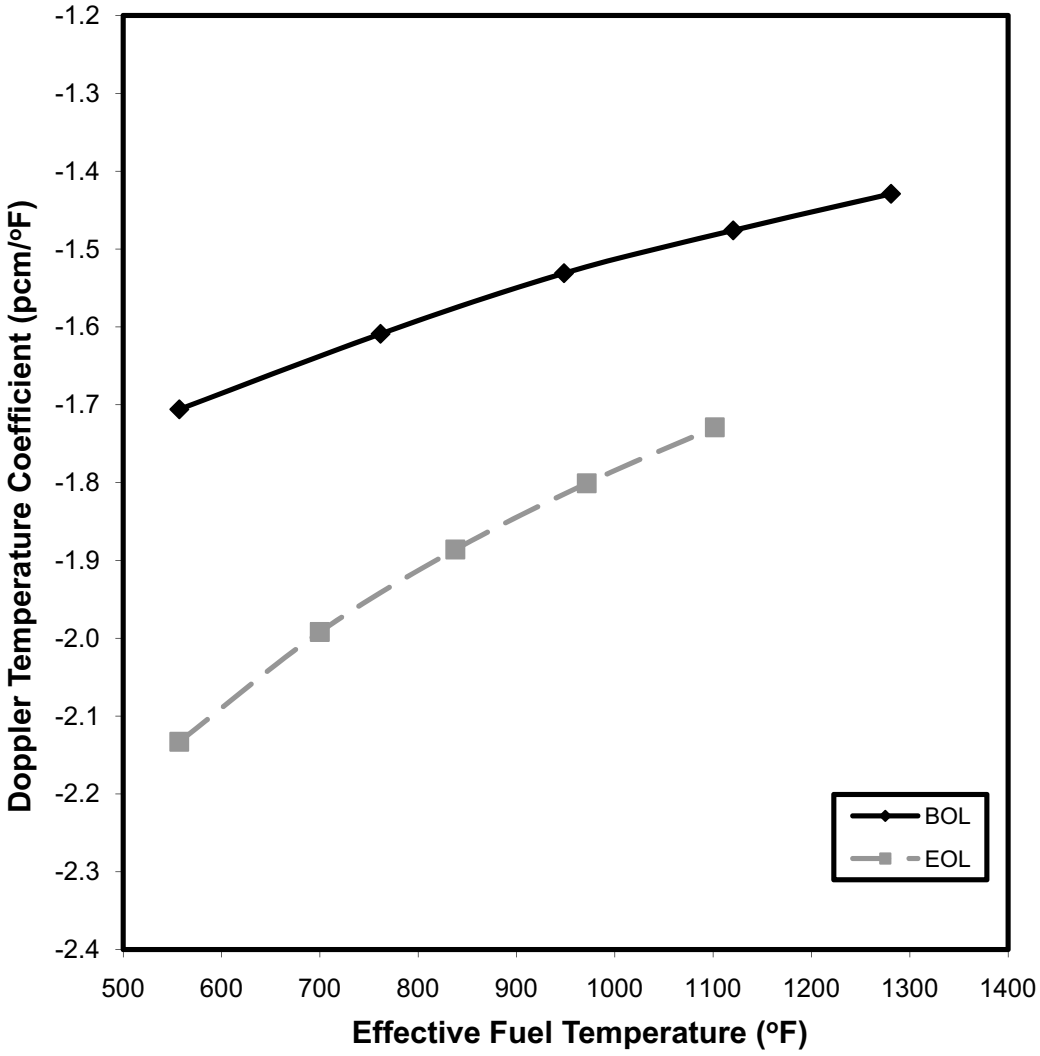

Figure 4.3-18
Typical Doppler Temperature Coefficient at BOL and EOL

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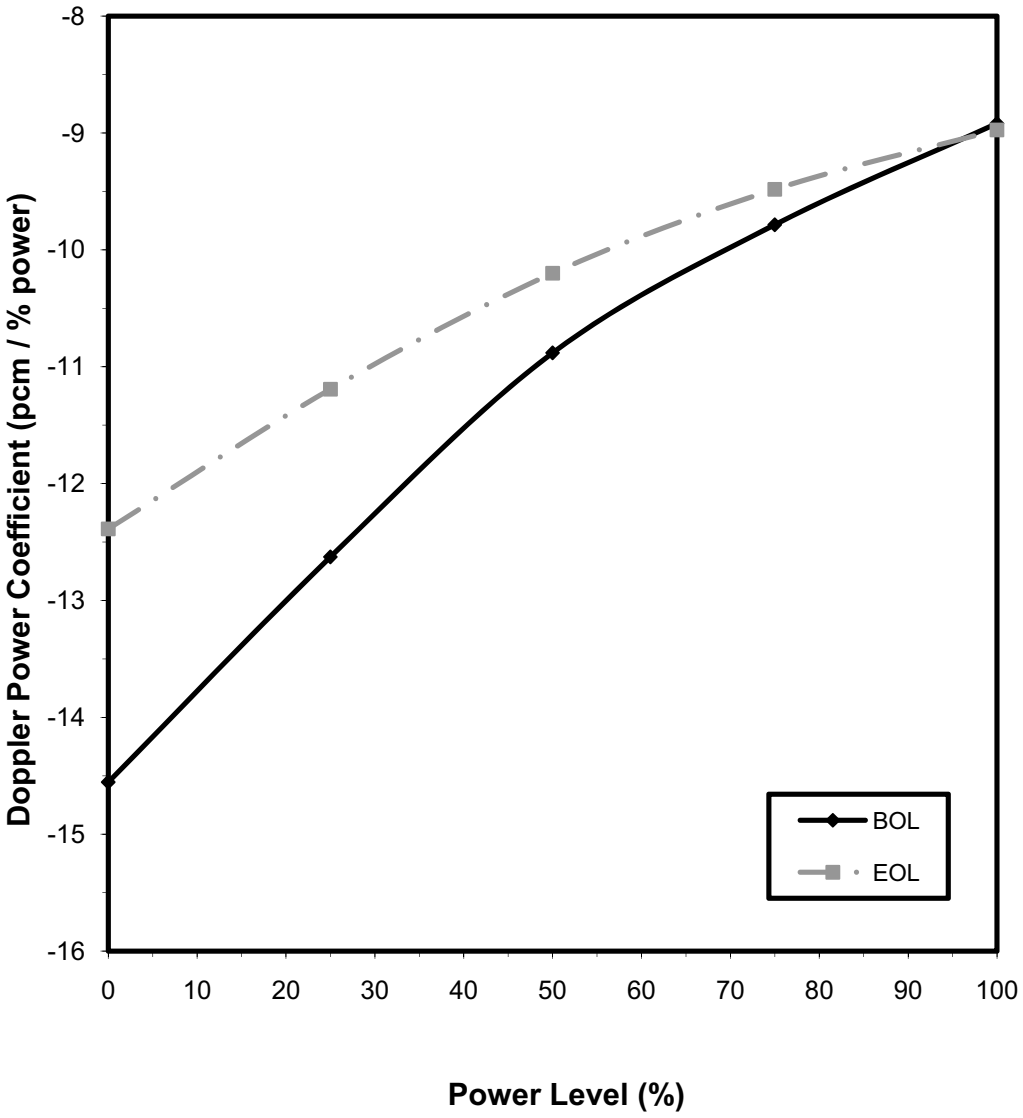

Figure 4.3-19
Typical Doppler-Only Power Coefficient at BOL and EOL

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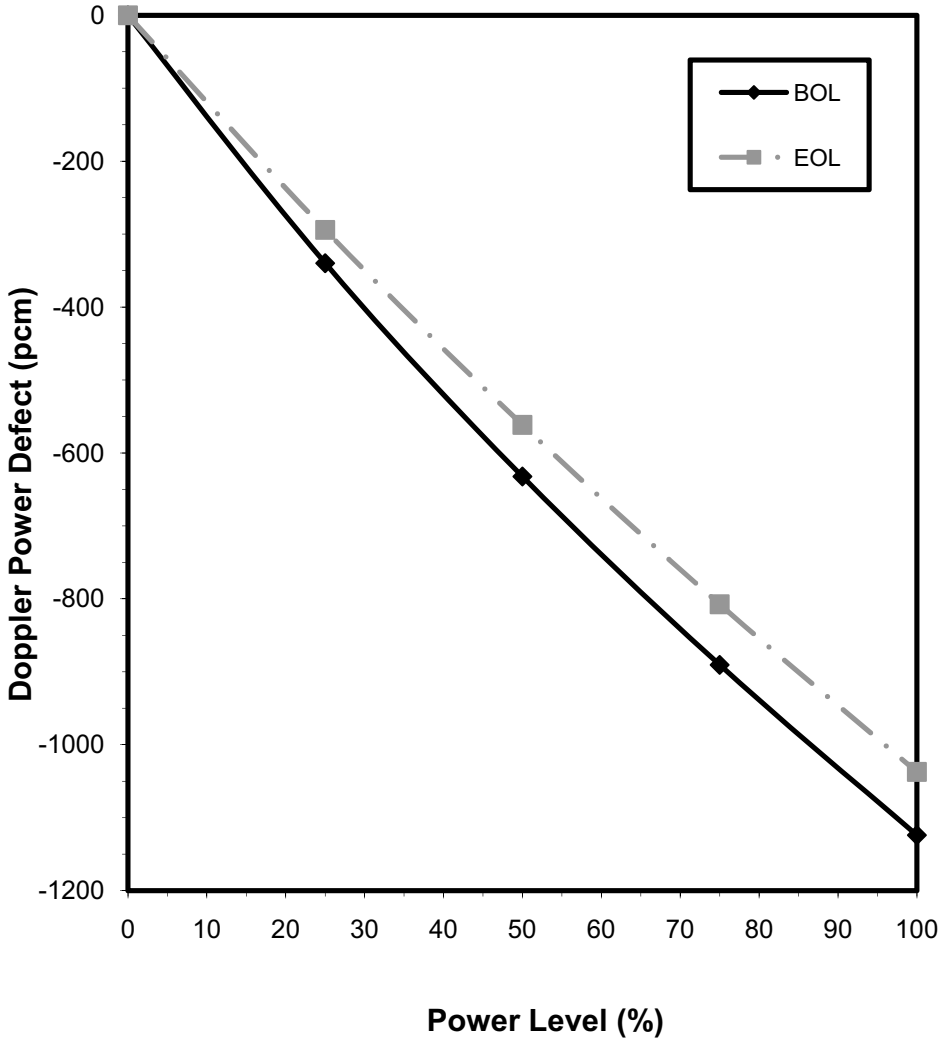

Figure 4.3-20
Typical Doppler-Only Power Defect at BOL and EOL

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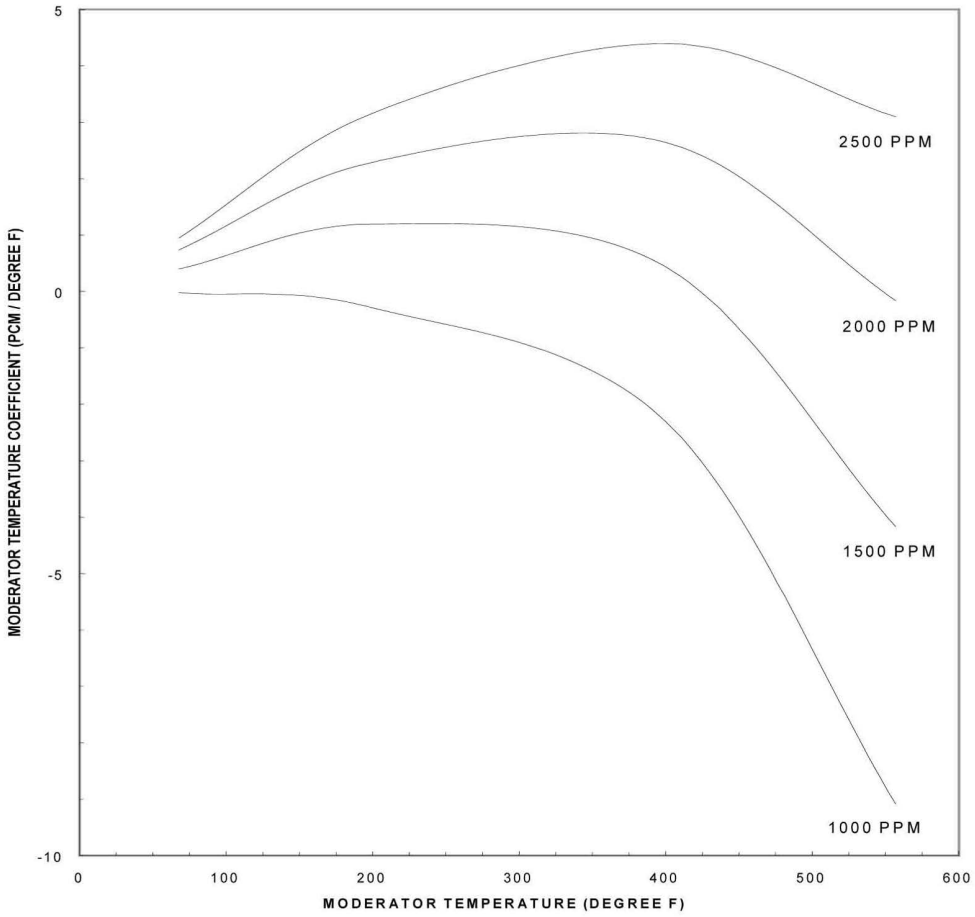

Figure 4.3-21
Typical Moderator Temperature Coefficient at BOL, Unrodded

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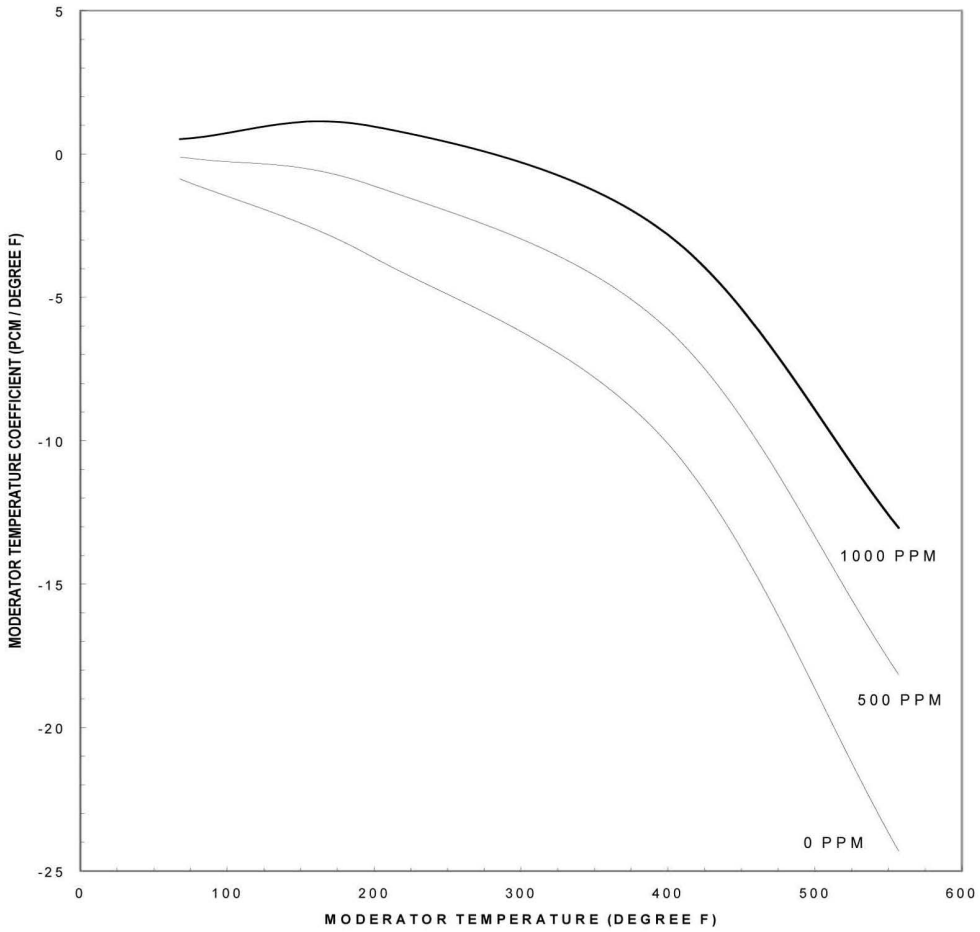

Figure 4.3-22
Typical Moderator Temperature Coefficient at EOL

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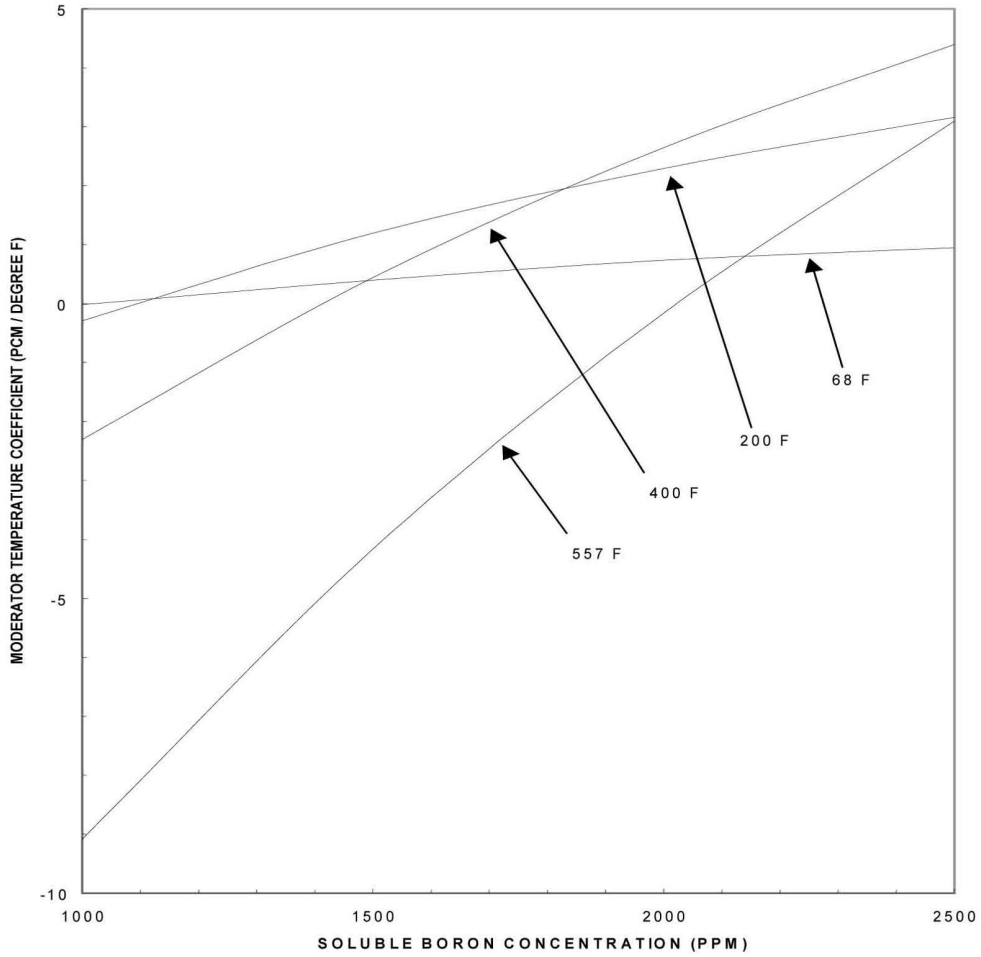

Figure 4.3-23
Typical Moderator Temperature Coefficient as a Function of Boron Concentration at BOL, Unrodded

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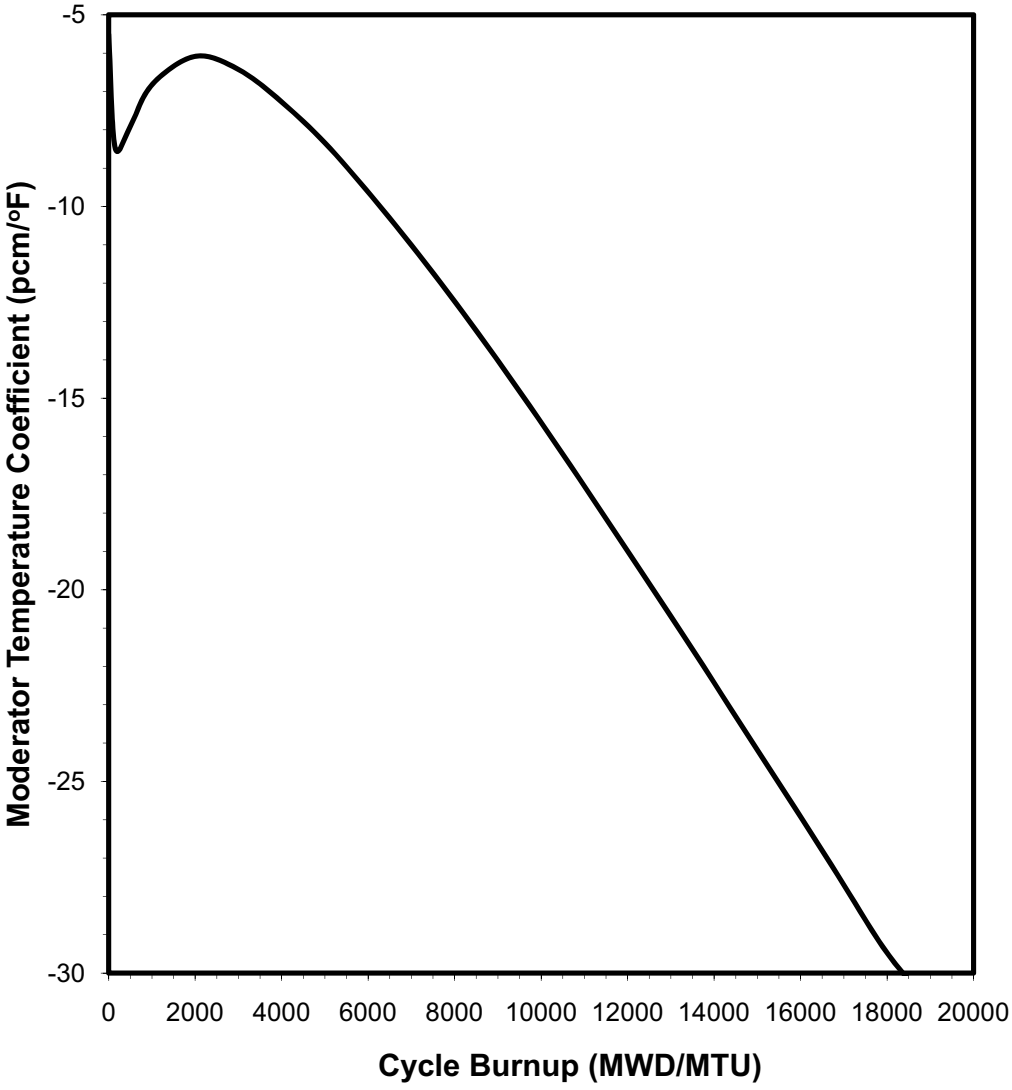

Figure 4.3-24
Typical Hot Full Power Moderator Temperature Coefficient versus Cycle Burnup

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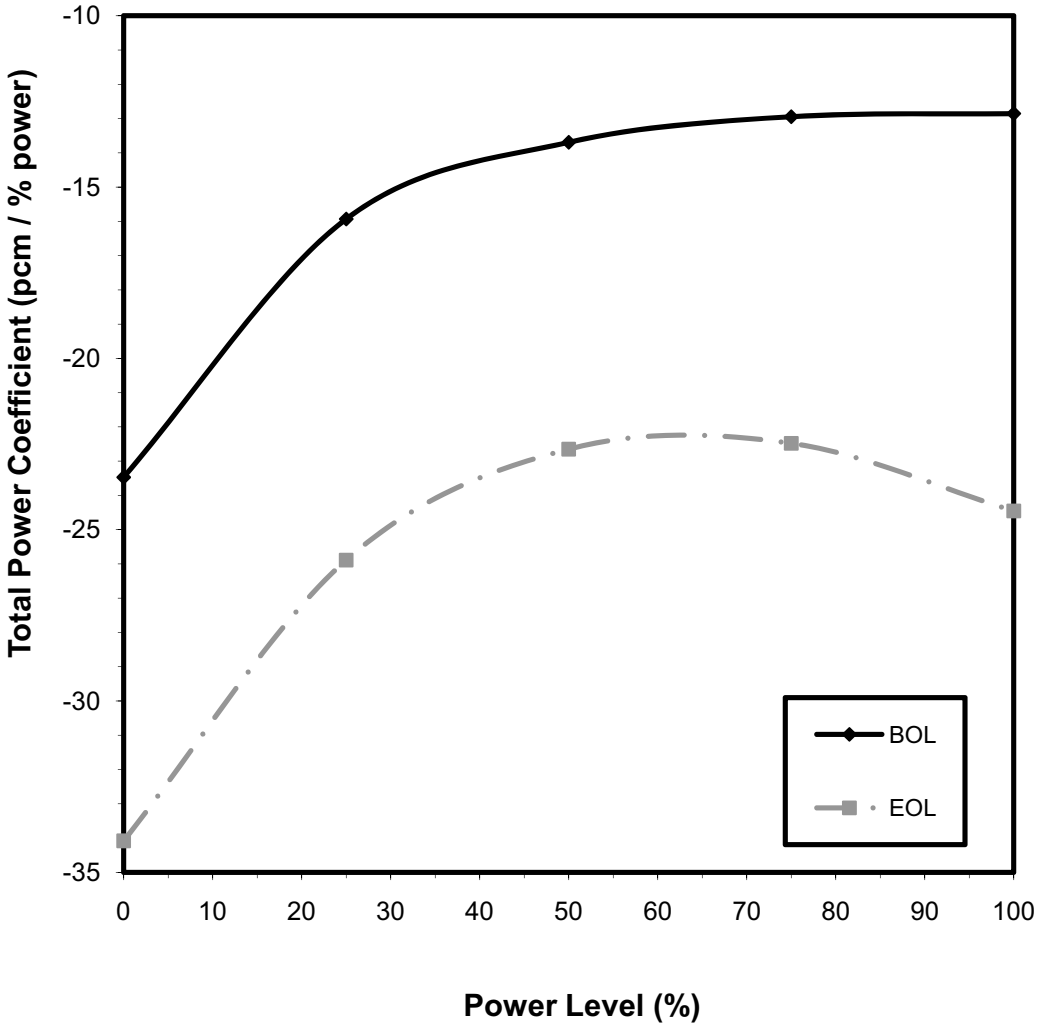

Figure 4.3-25
Typical Total Power Coefficient at BOL and EOL

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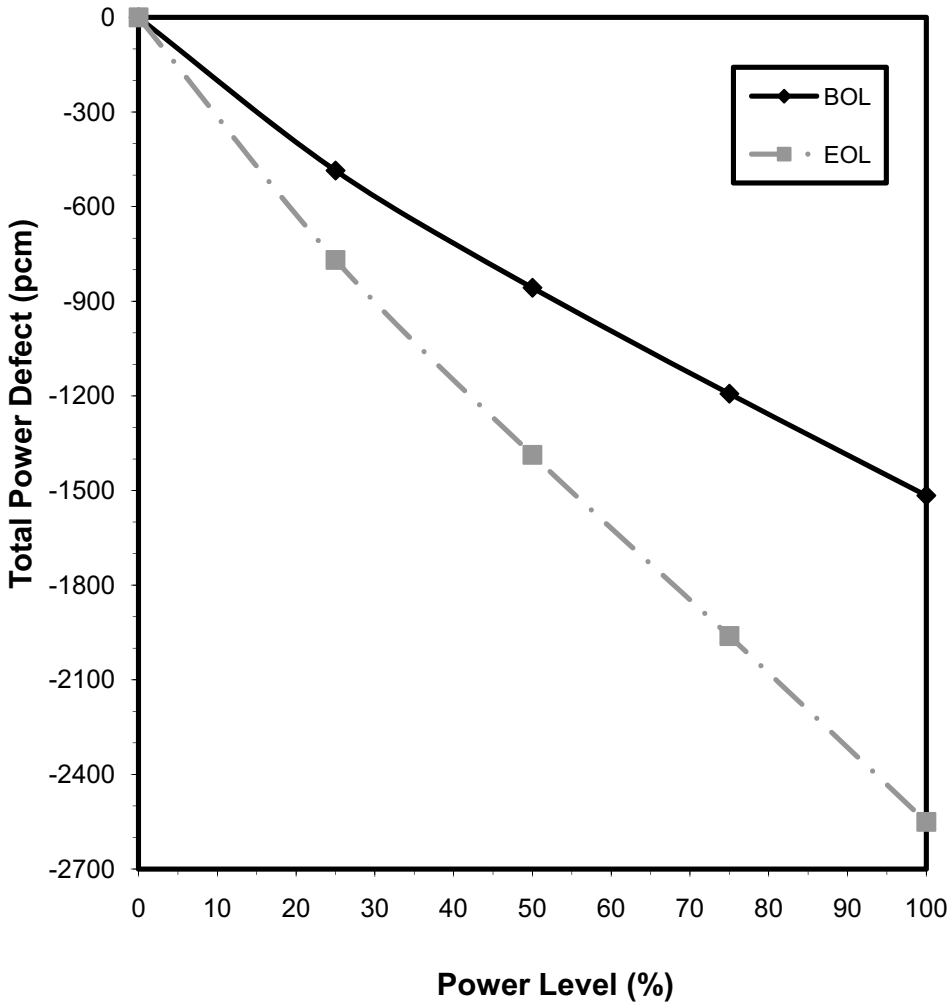

Figure 4.3-26
Typical Total Power Defect at BOL and EOL

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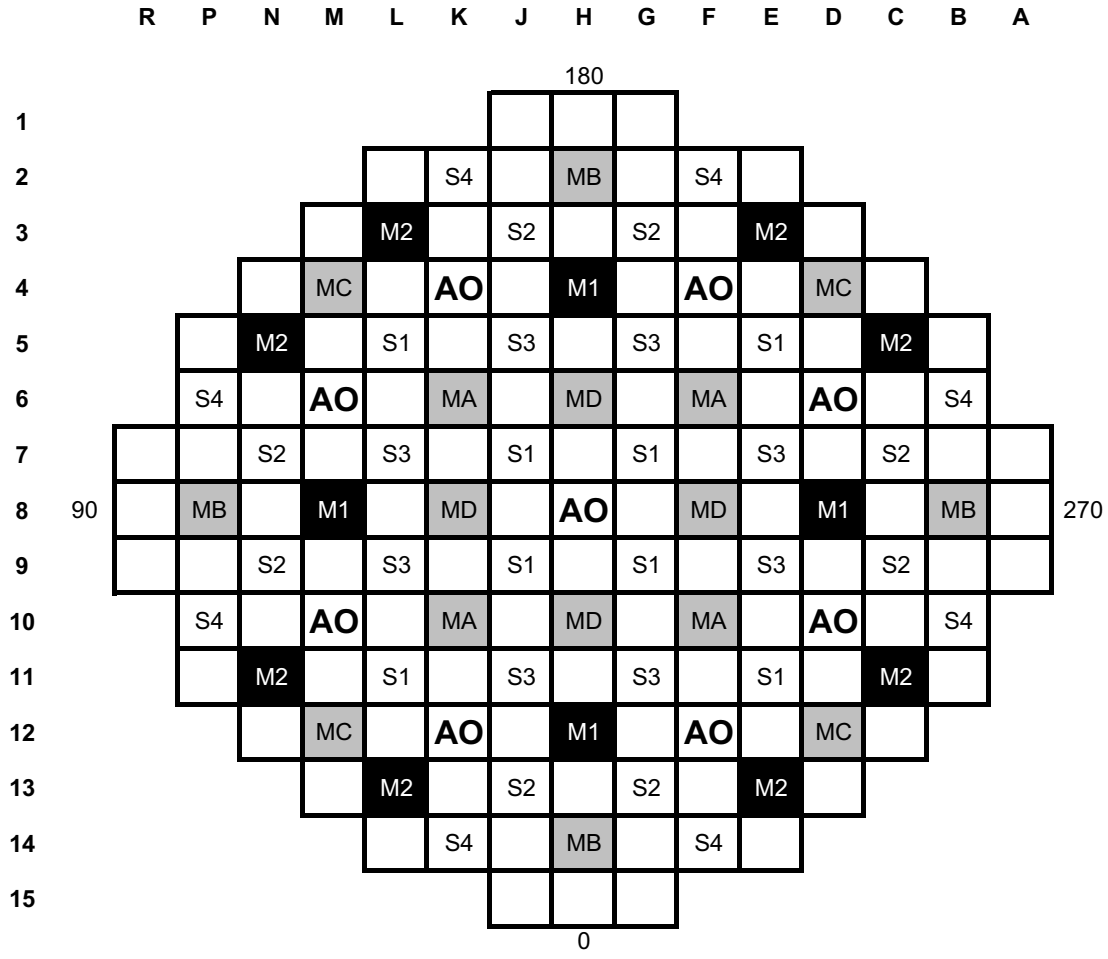

Bank ID	Group Association	Cluster Design Type	# of Clusters
MA	MSHIM Control	Gray	4
MB	MSHIM Control	Gray	4
MC	MSHIM Control	Gray	4
MD	MSHIM Control	Gray	4
M1	MSHIM Control	Black	4
M2	MSHIM Control	Black	8
AO	Axial Offset Control	Black	9
S1	Shutdown	Black	8
S2	Shutdown	Black	8
S3	Shutdown	Black	8
S4	Shutdown	Black	8
Total			**69**

Figure 4.3-27
Rod Cluster Control/Gray Rod Cluster Assembly (RCCA/GRCA) Assembly Pattern

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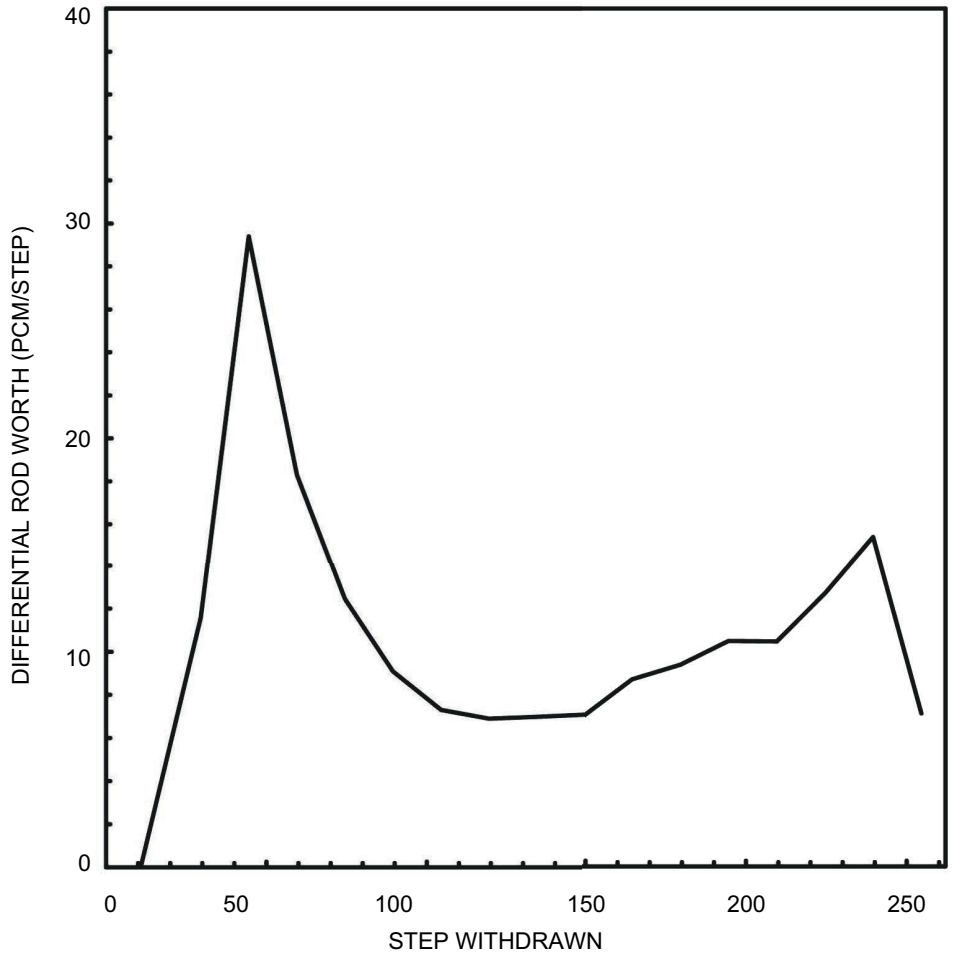

Figure 4.3-28
Typical Accidental Simultaneous Withdrawal of Two Control Banks at EOL, HZP, Moving in the Same Plane

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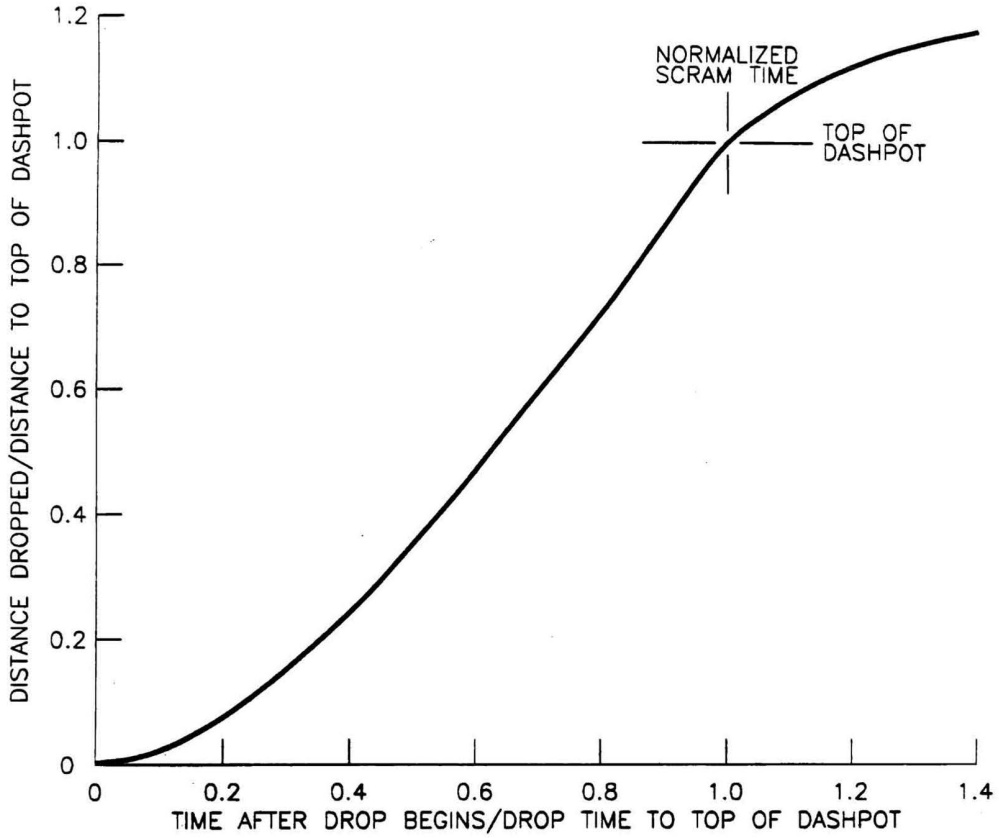

Figure 4.3-29
Typical Design Trip Curve

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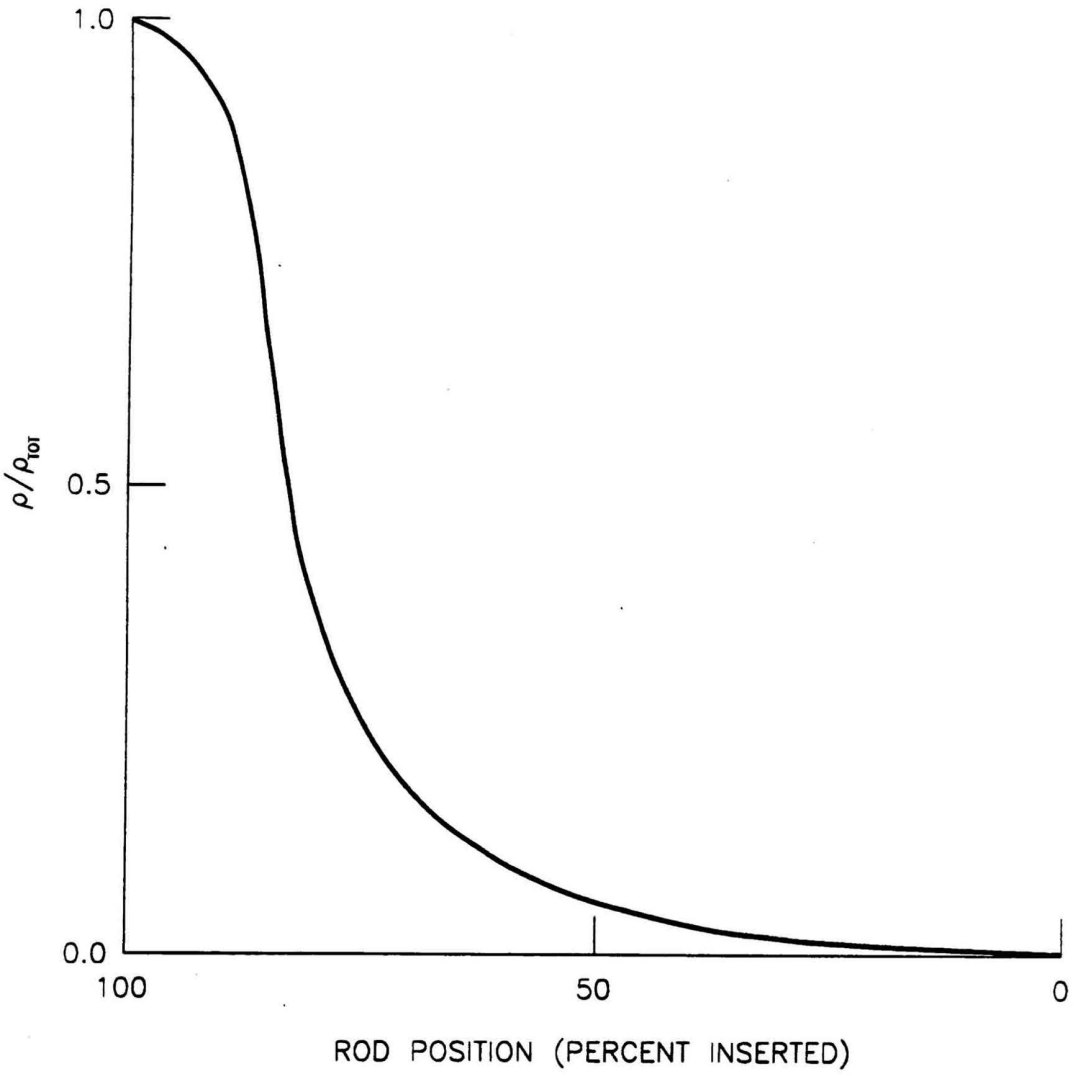

Figure 4.3-30
Typical Normalized Rod Worth Versus Percent Insertion
All Rods Inserting Less Most Reactive Stuck Rod

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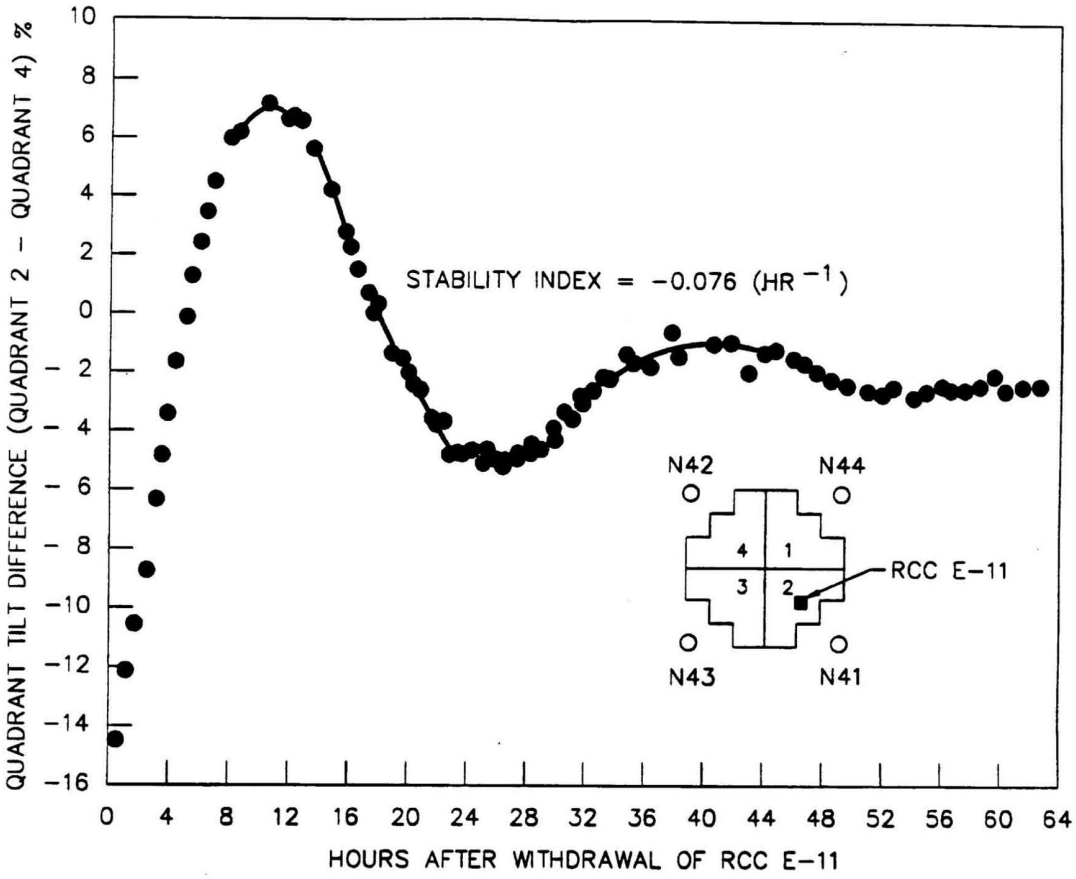

Figure 4.3-31
X-Y Xenon Test Thermocouple Response
Quadrant Tilt Difference Versus Time

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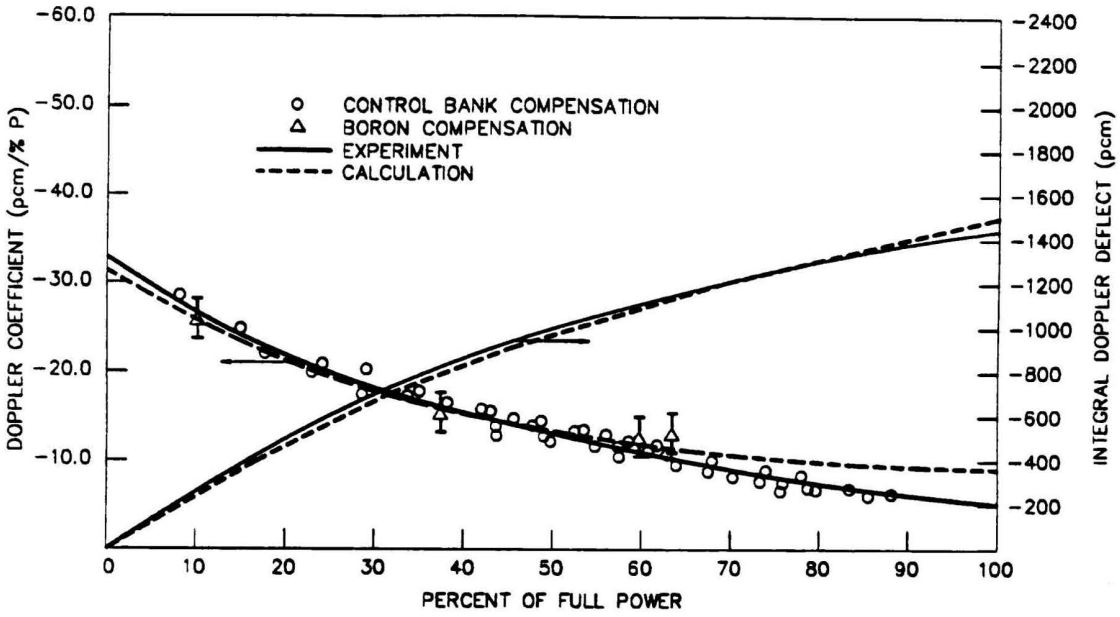

Figure 4.3-32
Calculated and Measured Doppler Defect and Coefficients at BOL, 2-Loop Plant, 121 Assemblies, 12-foot Core

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4.4 Thermal and Hydraulic Design

The thermal and hydraulic design of the reactor core provides adequate heat transfer compatible with the heat generation distribution in the core. This provides adequate heat removal by the reactor coolant system, the normal residual heat removal system, or the passive core cooling system.

4.4.1 Design Basis

The following performance and safety criteria requirements are established for the thermal and hydraulic design of the fuel. Condition I, II, III, and IV transients and events through out this section are as defined in ANSI N18.2a-75 (Reference 1).

- Fuel damage (defined as penetration of the fission product barrier; that is, the fuel rod clad) is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant cleanup system and are consistent with the plant design bases.

- The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged (as defined in the above definition), although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

- The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

To satisfy these requirements, the following design bases have been established for the thermal and hydraulic design of the reactor core.

4.4.1.1 Departure from Nucleate Boiling Design Basis

4.4.1.1.1 Design Basis

There is at least a 95-percent probability at a 95-percent confidence level that departure from nucleate boiling (DNB) does not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events).

4.4.1.1.2 Discussion

The design method employed to meet the DNB design basis for the AP1000 fuel assemblies is the Revised Thermal Design Procedure, WCAP-11397-P-A (Reference 2). With the Revised Thermal Design Procedure methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, Revised Thermal Design Procedure design limits departure from nucleate boiling ratio (DNBR) values are determined such that there is at least a 95-percent probability at a 95-percent confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events).

Assumed uncertainties in the plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and reactor coolant system flow) are evaluated. Only the random portion of the plant operating parameter uncertainties is included in the statistical combination. Instrumentation bias is treated as a direct DNBR penalty. Since the parameter uncertainties are

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considered in determining the Revised Thermal Design Procedure design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

For those transients that use the VIPRE-01 computer program (Subsection 4.4.4.5.2) and the WRB-2M correlation (Subsection 4.4.2.2.1), the Revised Thermal Design Procedure design limits are 1.25 for the typical cell and 1.25 for the thimble cell. These values may be revised (slightly) when plant specific uncertainties are available.

To maintain DNBR margin to offset DNB penalties such as those due to fuel rod bow (as described in Subsection 4.4.2.2.5), the safety analyses are performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in DNBR margin. A portion of this margin is used to offset rod bow and unanticipated DNBR penalties.

The Standard Thermal Design Procedure is used for those analyses where the Revised Thermal Design Procedure is not applicable. In the Standard Thermal Design Procedure method the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety analyses input values to give the lowest minimum DNBR. The DNBR limit for Standard Thermal Design Procedure is the appropriate DNB correlation limits increased to give sufficient margins to cover any DNBR penalties associated with the analysis.

By preventing DNB, adequate heat transfer is provided from the fuel clad to the reactor coolant, thereby preventing clad damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis, since it is within a few degrees of coolant temperature during operation in the nucleate boiling region. Limits provided by the nuclear control and protection systems are such that this design basis is met for transients associated with Condition II events including overpower transients. There is an additional large DNBR margin at rated power operation and during normal operating transients.

4.4.1.2 Fuel Temperature Design Basis

4.4.1.2.1 Design Basis

During modes of operation associated with Condition I and Condition II events, there is at least a 95-percent probability at a 95-percent confidence level that the peak kW/ft fuel rods will not exceed the uranium dioxide melting temperature. The melting temperature of uranium dioxide is 5080°F (Reference 3) unirradiated and decreasing 58°F per 10,000 MWD/MTU. By precluding uranium dioxide melting, the fuel geometry is preserved and possible adverse effects of molten uranium dioxide on the cladding are eliminated. Design evaluations for Condition I and II events have shown that fuel melting will not occur for achievable local burnups up to 75,000 MWD/MTU (Reference 81). The NRC has approved design evaluations up to 60,000 MWD/MTU in Reference 81 and up to 62,000 MWD/MTU in References 9 and 88.

4.4.1.2.2 Discussion

Fuel rod thermal evaluations are performed at rated power, at maximum overpower, and during transients at various burnups. These analyses confirm that this design basis and the fuel integrity design bases given in Section 4.2 are met. They also provide input for the evaluation of Condition III and IV events given in Chapter 15.

The center-line temperature limit has been applied to reload cores with a lead rod average burnup of up to 60,000 MWD/MTU. For higher burnups, the peak kilowatt-per-foot experienced during Condition I and II events is limited to that maximum value which is sufficient to provide that the fuel

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center-line temperatures remain below the melting temperature for the fuel rods. Thus, the fuel rod design basis that fuel rod damage not occur due to fuel melting continues to be met.

4.4.1.3 Core Flow Design Basis

4.4.1.3.1 Design Basis

Typical minimum value of 94.1 percent of the thermal flow rate is assumed to pass through the fuel rod region of the core and is effective for fuel rod cooling. Coolant flow through the thimble and instrumentation tubes and the leakage between the core barrel and core shroud, head cooling flow, and leakage to the vessel outlet nozzles are not considered effective for heat removal.

4.4.1.3.2 Discussion

Core cooling evaluations are based on the thermal flow rate (minimum flow) entering the reactor vessel. A typical maximum value of 5.9 percent of this value is allotted as bypass flow. This includes rod cluster control guide thimble and instrumentation tube cooling flow, leakage between the core barrel and the core shroud, head cooling flow, and leakage to the vessel outlet nozzles. The shroud core cavity flow is considered as active flow that is effective for fuel rod cooling.

The maximum bypass flow fraction of 5.9 percent assumes the use of thimble plugging devices in the rod cluster control guide thimble tubes that do not contain any other core components.

4.4.1.4 Hydrodynamic Stability Design Basis

Modes of operation associated with Condition I and II events do not lead to hydrodynamic instability.

4.4.1.5 Other Considerations

The design bases described in Subsections 4.4.1 through 4.4.1.4 together with the fuel clad and fuel assembly design bases given in Subsection 4.2.1 are sufficiently comprehensive that additional limits are not required.

Fuel rod diametral gap characteristics, moderator coolant flow velocity and distribution, and moderator void are not inherently limiting. Each of these parameters is incorporated into the thermal and hydraulic models used to confirm that the above-mentioned design criteria are met. For instance, the fuel rod diametral gap characteristics change with time, as described in Subsection 4.2.3, and the fuel rod integrity is evaluated on that basis. The effect of the moderator flow velocity and distribution described in Subsection 4.4.2.2 and the moderator void distribution described in Subsection 4.4.2.4 are included in the core thermal evaluation and thus affect the design basis.

Meeting the fuel clad integrity criteria covers the possible effects of clad temperature limitations. Clad surface temperature limits are imposed on Condition I and Condition II operation to preclude conditions of accelerated oxidation. A clad temperature limit is applied to the loss-of-coolant accident described in Subsection 15.6.5; control rod ejection accident described in Subsection 15.4.8; and locked rotor accident described in Subsection 15.3.3.

4.4.2 Description of Thermal and Hydraulic Design of the Reactor Core

4.4.2.1 Summary Comparison

Table 4.4-1 provides a comparison of the design parameters for the AP1000, the AP600, and a licensed Westinghouse-designed plant using XL Robust fuel. For the comparison with a plant

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containing XL Robust fuel, a 193 fuel assembly plant is used, since no domestic Westinghouse designed 157 fuel assembly plants use 17x17 XL Robust fuel.

4.4.2.2 Critical Heat Flux Ratio or DNBR and Mixing Technology

The minimum DNBRs for the rated power and anticipated transient conditions are given in Table 4.4-1. The minimum DNBR in the limiting flow channel is typically downstream of the peak heat flux location (hotspot) due to the increased downstream enthalpy rise.

DNBRs are calculated by using the correlation and definitions described in Subsections 4.4.2.2.1 and 4.4.2.2.2. The VIPRE-01 computer code described in Subsection 4.4.4.5, is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is described in Subsections 4.4.4.3.1 (nuclear hot channel factors) and 4.4.2.2.4 (engineering hot channel factors).

4.4.2.2.1 DNB Technology

The primary DNB correlation used for the analysis of the AP1000 fuel is the WRB-2M correlation (References 82 and 82a). The WRB-2M correlation applies to the Robust Fuel Assemblies, which are planned to be used in the AP1000 core. This correlation applies to most AP1000 conditions.

A correlation limit of 1.14 is applicable for the WRB-2M correlation.

The applicable range of parameters for the WRB-2M correlation is:

Pressure	$1495 \le P \le 2425$ psia
Local mass velocity	$0.97 \le G_{loc}/10^6 \le 3.1$ lb/ft^2-hr
Local quality	$-0.1 \le X_{loc} \le 0.29$
Heated length, inlet to CHF location	$L_H \le 14$ feet
Grid spacing	$10 \le g_{sp} \le 20.6$ inches
Equivalent hydraulic diameter	$0.37 \le D_e \le 0.46$ inches
Equivalent heated hydraulic diameter	$0.46 \le D_h \le 0.54$ inches

The WRB-2 (Reference 4), ABB-NV (References 89 and 90), or WLOP (Reference 90) correlation is used wherever the WRB-2M correlation is not applicable. The WRB-2 correlation limit is 1.17.

The applicable range of parameters for the WRB-2 correlation is:

Pressure	$1440 \le P \le 2490$ psia
Local mass velocity	$0.9 \le G_{loc}/10^6 \le 3.7$ lb/ft^2-hr
Local quality	$-0.1 \le X_{loc} \le 0.3$
Heat length, inlet to DNB location	$L_h \le 14$ feet
Grid spacing	$10 \le g_{sp} \le 26$ inches
Equivalent hydraulic diameter	$0.37 \le D_e \le 0.51$ inches
Equivalent heated hydraulic diameter	$0.46 \le D_h \le 0.59$ inches

The WRB-2 correlation was developed based on mixing vane data and, therefore, is only applicable in the heated rod spans above the first mixing vane grid.

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In the heated region below the first mixing vane grid, the ABB-NV correlation, References 89 and 90, which is based on CHF data from fuel assemblies without mixing vane grids, is used to calculate DNBR values. For system pressures and flow rates where the above correlations are not applicable, the WLOP correlation, Reference 90, is used to calculate DNBR values.

4.4.2.2.2 Definition of DNBR

The DNB heat flux ratio, DNBR, as applied to typical cells (flow cells with all walls heated) and thimble cells (flow cells with heated and unheated walls) is defined as:

$$\text{DNBR} = \frac{q''_{\text{DNB, predicted}}}{q''\text{actual}}$$

where:

$$q''_{\text{DNB, predicted}} = \frac{q''_{\text{WRB-2M}}}{F} \quad \text{or} \quad q''_{\text{DNB, predicted}} = \frac{q''_{\text{WRB-2}}}{F}$$

$q''_{\text{WRB-2M}}$ = the uniform DNB heat flux as predicted by the WRB-2M DNB correlation

$q''_{\text{WRB-2}}$ = the uniform DNB heat flux as predicted by the WRB-2 DNB correlation

F = the flux shape factor to account for nonuniform axial heat flux distributions (Reference 10) with the term “C” modified as in Reference 5

q''_{actual} = the actual local heat flux

Adjusted F factors are used for WRB-2M, Reference 82a, ABB-NV, References 89 and 90, and WLOP, Reference 90.

4.4.2.2.3 Mixing Technology

The rate of heat exchange by mixing between flow channels is proportional to the difference in the local mean fluid enthalpy of the respective channels, the local fluid density, and the flow velocity. The proportionality is expressed by the dimensionless thermal diffusion coefficient (TDC) which is defined as:

$$\text{TDC} = \frac{w'}{\rho V a}$$

where:

w' = flow exchange rate per unit length (lbm/ft-s)

ρ = fluid density (lbm/ft^3)

V = fluid velocity (ft/s)

a = lateral flow area between channels per unit length (ft^2/ft)

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The application of the thermal diffusion coefficient in the VIPRE-01 analysis for determining the overall mixing effect or heat exchange rate is presented in Reference 83.

As discussed in WCAP-7941-P-A (Reference 12) those series of tests, using the “R” mixing vane grid design on 13-, 26-, and 32-inch grid spacing, were conducted in pressurized water loops at Reynolds numbers similar to that of a pressurized water reactor core under the following single- and two-phase (subcooled boiling) flow conditions:

- Pressure 1500 to 2400 psia
- Inlet temperature 332 to 642°F
- Mass velocity 1.0 to 3.5 x 10^6 lbm/hr-ft^2
- Reynolds number 1.34 to 7.45 x 10^5
- Bulk outlet quality -52.1 to -13.5 percent

The thermal diffusion coefficient is determined by comparing the THINC code predictions with the measured subchannel exit temperatures. Data for 26-inch axial grid spacing are presented in Figure 4.4-1, where the thermal diffusion coefficient is plotted versus the Reynolds number. The thermal diffusion coefficient is found to be independent of the Reynolds number, mass velocity, pressure, and quality over the ranges tested. The two-phase data (local, subcooled boiling) falls within the scatter of the single-phase data. The effect of two-phase flow on the value of the thermal diffusion coefficient is demonstrated in WCAP-7941-P-A (Reference 12), by Rowe and Angle (References 13 and 14), and Gonzalez-Santalo and Griffith (Reference 15). In the subcooled boiling region, the values of the thermal diffusion coefficient are indistinguishable from the single-phase values. In the quality region, Rowe and Angle show that in the case with rod spacing similar to that in pressurized water reactor core geometry, the value of the thermal diffusion coefficient increased with quality to a point and then decreased, but never below the single-phase value. Gonzalez-Santalo and Griffith show that the mixing coefficient increased as the void fraction increased.

The data from these tests on the R-mixing vane grid show that a design thermal diffusion coefficient value of 0.038 (for 26-inch grid spacing) can be used in determining the effect of coolant mixing in the THINC analysis. An equivalent value of the mixing coefficient is used in the VIPRE-01 evaluations (Reference 83). A mixing test program similar to the one just described was conducted for the current 17 x 17 geometry and mixing vane grids on 26-inch spacing, as described in WCAP-8298-P-A (Reference 16). The mean value of the thermal diffusion coefficient obtained from these tests is 0.059.

The inclusion of intermediate flow mixer grids in the upper spans of the fuel assembly results in a grid spacing of approximately 10 inches giving higher values of the thermal diffusion coefficient. A conservative value of the thermal diffusion coefficient, 0.038, is used to determine the effect of coolant mixing in the core thermal performance analysis.

4.4.2.2.4 Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core-average ratios of these quantities. The heat flux hot channel factor considers the local maximum linear heat generation rate at a point (the hotspot), and the enthalpy rise hot channel factor involves the maximum integrated value along a channel (the hot channel).

Each of the total hot channel factors is composed of a nuclear hot channel factor, Subsection 4.4.4.3, describing the neutron power distribution and an engineering hot channel factor, which allows for

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variations in flow conditions and fabrication tolerances. The engineering hot channel factors are made up of subfactors which account for the influence of the variations of fuel pellet diameter, density, enrichment, and eccentricity; inlet flow distribution; flow redistribution; and flow mixing.

Heat Flux Engineering Hot Channel Factor, F_Q^E

The heat flux engineering hot channel factor is used to evaluate the maximum linear heat generation rate in the core. This subfactor is determined by statistically combining the fabrication variations for fuel pellet diameter, density, and enrichment. As shown in WCAP-8174 (Reference 17), no DNB penalty needs to be taken for the short, relatively low-intensity heat flux spikes caused by variations in the above parameters, as well as fuel pellet eccentricity and fuel rod diameter variation.

Enthalpy Rise Engineering Hot Channel Factor, $F_{\Delta H}^E$

The effect of variations in flow conditions and fabrication tolerances on the hot channel enthalpy rise is directly considered in the VIPRE-01 core thermal subchannel analysis, described in Subsection 4.4.4.5.1 under any reactor operating condition. The following items are considered as contributors to the enthalpy rise engineering hot channel factor:

- Pellet diameter, density, and enrichment

  Variations in pellet diameter, density, and enrichment are considered statistically in establishing the limit DNBRs, described in Subsection 4.4.1.1.2, for the Revised Thermal Design Procedure (Reference 2). Uncertainties in these variables are determined from sampling of manufacturing data.

- Inlet flow maldistribution

  The consideration of inlet flow maldistribution in core thermal performances is described in Subsection 4.4.4.2.2. A design basis of five-percent reduction in coolant flow to the hot assembly is used in the VIPRE-01 analyses.

- Flow redistribution

  The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the nonuniform power distribution is inherently considered in the VIPRE-01 analyses for every operating condition evaluated.

- Flow mixing

  The subchannel mixing model incorporated in the VIPRE-01 code and used in reactor design is based on experimental data, as detailed in WCAP-7667-P-A (Reference 18) and discussed in Subsections 4.4.2.2.3 and 4.4.4.5.1. The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances. The VIPRE-01 mixing model is discussed in Reference 83.

4.4.2.2.5 Effects of Rod Bow on DNBR

The phenomenon of fuel rod bowing, as described in WCAP-8691 (Reference 19), is accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR and/or margin obtained form measured plant operating parameters (such as $F_{\Delta H}^N$ or core flow), which are

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less limiting than those required by the plant safety analysis, can be used to offset the effect of rod bow.

For the safety analysis of the AP1000, sufficient DNBR margin was maintained, as described in Subsection 4.4.1.1.2, to accommodate the full and low flow rod bow DNBR penalties identified in Reference 20. The referenced penalties are applicable to the analyses using the WRB-2M or WRB-2 DNB correlations.

The maximum rod bow penalties (less than about 2 percent DNBR) accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWD/MTU. At burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required (Reference 21).

In the upper spans of the fuel assembly, additional restraint is provided with the intermediate flow mixer grids such that the grid-to-grid spacing in those spans with intermediate flow mixer grids is approximately 10 inches compared to approximately 20 inches in the other spans. Using the NRC approved scaling factor [see WCAP 8691 (Reference 19) and Reference 21], results in predicted channel closure in the limiting 10 inch spans of less than 50 percent closure. Therefore, no rod bow DNBR penalty is required in the 10 inch spans in the safety analyses.

4.4.2.3 Linear Heat Generation Rate

The core average and maximum linear heat generation rates are given in Table 4.4-1. The method of determining the maximum linear heat generation rate is given in Subsection 4.3.2.2.

4.4.2.4 Void Fraction Distribution

The calculated core average and the hot subchannel maximum and average void fractions are presented in Table 4.4-2 for operation at full power. The void models used in the VIPRE-W code are described in Subsection 4.4.2.7.3.

4.4.2.5 Core Coolant Flow Distribution

The VIPRE-01 code is used to calculate the flow and enthalpy distribution in the core for use in safety analysis. Extensive experimental verification of VIPRE-01 is presented in Reference 84.

4.4.2.6 Core Pressure Drops and Hydraulic Loads

4.4.2.6.1 Core Pressure Drops

The analytical model and experimental data used to calculate the pressure drops shown in Table 4.4-1 are described in Subsection 4.4.2.7. The core pressure drop includes the fuel assembly, lower core plate, and upper core plate pressure drops. The full-power operation pressure drop values shown in Table 4.4-1 are the unrecoverable pressure drops across the vessel, including the inlet and outlet nozzles, and across the core. These pressure drops are based on the best-estimate flow for actual plant operating conditions as described in Subsection 5.1.4. This subsection also defines and describes the thermal design flow (minimum flow) that is the basis for reactor core thermal performance and the mechanical design flow (maximum flow) that is used in the mechanical design of the reactor vessel internals and fuel assemblies. Since the best-estimate flow is that flow which is most likely to exist in an operating plant, the calculated core pressure drops in Table 4.4-1 are based on this best-estimate flow rather than the thermal design flow.

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The uncertainties associated with the core pressure drop values are presented in Subsection 4.4.2.9.2.

4.4.2.6.2 Hydraulic Loads

Figure 4.2-2 shows the fuel assembly hold-down springs. These springs are designed to keep the fuel assemblies in contact with the lower core plate under Condition I and II events, except for the turbine overspeed transient associated with a loss of external load. The hold-down springs are designed to tolerate the possibility of an overdeflection associated with fuel assembly lift-off for this case and to provide contact between the fuel assembly and the lower core plate following this transient. More adverse flow conditions occur during a loss-of-coolant accident. These conditions are presented in Subsection 15.6.5.

Hydraulic loads at normal operating conditions are calculated considering the best-estimate flow, described in Section 5.1, and accounting for the minimum core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold plant startup conditions are based on the cold best-estimate flow, but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which could possibly create a flow rate 18-percent greater than the best estimate flow, are evaluated to be approximately twice the fuel assembly weight.

Hydraulic verification tests for the fuel assembly are described in Reference 86.

4.4.2.7 Correlation and Physical Data

4.4.2.7.1 Surface Heat Transfer Coefficients

Forced convection heat transfer coefficients are obtained from the Dittus-Boelter correlation (Reference 24), with the properties evaluated at bulk fluid conditions:

$$\frac{hD_e}{K} = 0.023\left(\frac{D_e G}{\mu}\right)^{0.8}\left(\frac{C_p \mu}{K}\right)^{0.4}$$

where:

h = heat transfer coefficient (btu/h-ft^2-°F)

D_e = equivalent diameter (ft)

K = thermal conductivity (Btu/h-ft-°F)

G = mass velocity (lbm/h-ft^2)

μ = dynamic viscosity (lbm/ft-h)

C_p = heat capacity (Btu/lb-°F)

This correlation has been shown to be conservative (Reference 25) for rod bundle geometries with pitch-to-diameter ratios in the range used by pressurized water reactors.

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The onset of nucleate boiling occurs when the clad wall temperature reaches the amount of superheat predicted by Thom's correlation (Reference 26). After this occurrence, the outer clad wall temperature is determined by:

$$\Delta T_{sat} = [0.072\exp(-P/1260)](q'')^{0.5}$$

where:

ΔT_{sat} = wall superheat, $T_W - T_{sat}$ (°F)

q'' = wall heat flux (Btu/h-ft^2)

P = pressure (psia)

T_w = outer clad wall temperature (°F)

T_{sat} = saturation temperature of coolant at pressure P (°F)

4.4.2.7.2 Total Core and Vessel Pressure Drop

Unrecoverable pressure losses occur as a result of viscous drag (friction) and/or geometry changes (form) in the fluid flow path. The flow field is assumed to be incompressible, turbulent, single-phase water. Those assumptions apply to the core and vessel pressure drop calculations for the purpose of establishing the primary loop flow rate. Two-phase considerations are neglected in the vessel pressure drop evaluation because the core average void is negligible, as shown in Table 4.4-2. Two-phase flow considerations in the core thermal subchannel analysis are considered and the models are described in Subsection 4.4.4.2.3. Core and vessel pressure losses are calculated by equations of the form:

$$\Delta P_L = (K + f\frac{L}{D_e})\frac{\rho V^2}{2 g_c (144)}$$

where:

ΔP_L = unrecoverable pressure drop (lb/in.^{2})

ρ = fluid density (lbm/ft^3)

L = length (ft)

D_e = equivalent diameter (ft)

V = fluid velocity (ft/s)

g_c = 32.174 (lbm-ft/lbf-s^2)

K = form loss coefficient (dimensionless)

f = friction loss coefficient (dimensionless)

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form

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and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

Values are quoted in Table 4.4-1 for unrecoverable pressure loss across the reactor vessel, including the inlet and outlet nozzles, and across the core. The results of full-scale tests of core components and fuel assemblies are used in developing the core pressure loss characteristic.

Tests of the primary coolant loop flow rates are made prior to initial criticality as described in Subsection 4.4.5.1, to verify that the flow rates used in the design, which are determined in part from the pressure losses calculated by the method described here, are conservative. See Section 14.2 for preoperational testing.

4.4.2.7.3 Void Fraction Correlation

VIPRE-01 considers two-phase flow in two steps. First, a quality model is used to compute the flowing vapor mass fraction (true quality) including the effects of subcooled boiling. Then, given the true void quality, a bulk void model is applied to compute the vapor volume fraction (void fraction).

VIPRE-01 uses a profile fit model (Reference 83) for determining subcooled quality. It calculates the local vapor volumetric fraction in forced convection boiling by: 1) predicting the point of bubble departure from the heated surface and 2) postulating a relationship between the true local vapor fraction and the corresponding thermal equilibrium value.

The void fraction in the bulk boiling region is predicted by using homogeneous flow theory and assuming no slip. The void fraction in this region is therefore a function only of the thermodynamic quality.

4.4.2.8 Thermal Effects of Operational Transients

DNB core safety limits are generated as a function of coolant temperature, pressure, core power, and axial power imbalance. Steady-state operation within these safety limits provides that the DNB design basis is met. Subsection 15.0.6 discusses the overtemperature ΔT trip (based on DNBR limit) versus T_{avg}. This system provides protection against anticipated operational transients that are slow with respect to fluid transport delays in the primary system. In addition, for fast transients (such as uncontrolled rod bank withdrawal at power incident as described in Subsection 15.4.2), specific protection functions are provided as described in Section 7.2. The use of these protection functions is described in Chapter 15.

4.4.2.9 Uncertainties in Estimates

4.4.2.9.1 Uncertainties in Fuel and Clad Temperatures

As described in Subsection 4.4.2.11, the fuel temperature is a function of crud, oxide, clad, pellet-clad gap, and pellet conductances. Uncertainties in the fuel temperature calculation are essentially of two types: fabrication uncertainties, such as variations in the pellet and clad dimensions and the pellet density; and model uncertainties, such as variations in the pellet conductivity and the gap conductance. These uncertainties have been quantified by comparison of the thermal model to the in-pile thermocouple measurements (References 30 through 36), by out-of-pile measurements of the fuel and clad properties (References 37 through 48), and by measurements of the fuel and clad dimensions during fabrication. The resulting uncertainties are then used in the evaluations involving the fuel temperature. The effect of densification on fuel temperature uncertainties is also included in the calculation of the total uncertainty.

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In addition to the temperature uncertainty described above, the measurement uncertainty in determining the local power and the effect of density and enrichment variations on the local power are considered in establishing the heat flux hot channel factor. These uncertainties are described in Subsection 4.3.2.2.1.

Reactor trip setpoints, as specified in the technical specifications, include allowance for instrument and measurement uncertainties such as calorimetric error, instrument drift and channel reproducibility, temperature measurement uncertainties, noise, and heat capacity variations.

Uncertainty in determining the cladding temperature results from uncertainties in the crud and oxide thicknesses. Because of the excellent heat transfer between the surface of the rod and the coolant, the film temperature drop does not appreciably contribute to the uncertainty.

4.4.2.9.2 Uncertainties in Pressure Drops

Core and vessel pressure drops based on the best-estimate flow, as described in Section 5.1, are quoted in Table 4.4-1. The uncertainties quoted are based on the uncertainties in both the test results and the analytical extension of these values to the reactor application.

A major use of the core and vessel pressure drops is to determine the primary system coolant flow rates, as described in Section 5.1. In addition, as described in Subsection 4.4.5.1, tests on primary system prior to initial criticality, are conducted to verify that a conservative primary system coolant flow rate has been used in the design and analysis of the plant.

4.4.2.9.3 Uncertainties Due to Inlet Flow Maldistribution

The effects of uncertainties in the inlet flow maldistribution criteria used in the core thermal analyses are described in Subsection 4.4.4.2.2.

4.4.2.9.4 Uncertainty in DNB Correlation

The uncertainty in the DNB correlation described in Subsection 4.4.2.2, is written as a statement on the probability of not being in DNB based on the statistics of the DNB data. This is described in Subsection 4.4.2.2.2.

4.4.2.9.5 Uncertainties in DNBR Calculations

The uncertainties in the DNBRs calculated by the VIPRE-01 analyses, discussed in Subsection 4.4.4.5.1, due to uncertainties in the nuclear peaking factors are accounted for by applying conservatively high values of the nuclear peaking factors. Measurement error allowances are included in the statistical evaluation of the limit DNBR described in Subsection 4.4.1.1 using the Revised Thermal Design Procedure. More information is provided in WCAP-11397-P-A (Reference 2). In addition, conservative values for the engineering hot channel factors are used as presented in Subsection 4.4.2.2.4. The results of a sensitivity study, WCAP-8054-P-A (Reference 22), with THINC-IV, a VIPRE-01 equivalent code, show that the minimum DNBR in the hot channel is relatively insensitive to variations in the core-wide radial power distribution (for the same value of $F^{N}_{\Delta H}$).

The ability of the VIPRE-01 computer code to accurately predict flow and enthalpy distributions in rod bundles is discussed in Subsection 4.4.4.5.1 and in Reference 83. Studies (Reference 84) have been performed to determine the sensitivity of the minimum DNBR to the void fraction correlation (see also Subsection 4.4.2.7.3) and the inlet flow distributions. The results of these studies show that the minimum DNBR is relatively insensitive to variation in these parameters. Furthermore, the VIPRE-01 flow field model for predicting conditions in the hot channels is consistent with that used in

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the derivation of the DNB correlation limits including void/quality modeling, turbulent mixing and crossflow and two phase flow (Reference 83).

4.4.2.9.6 Uncertainties in Flow Rates

The uncertainties associated with reactor coolant loop flow rates are discussed in Section 5.1. A thermal design flow is defined for use in core thermal performance evaluations accounting for both prediction and measurement uncertainties. In addition, another 5.9 percent of the thermal design flow is assumed to be ineffective for core heat removal capability because it bypasses the core through the various available vessel flow paths described in Subsection 4.4.4.2.1.

4.4.2.9.7 Uncertainties in Hydraulic Loads

As described in Subsection 4.4.2.6.2, hydraulic loads on the fuel assembly are evaluated for a pump overspeed transient which creates flow rates 18 percent greater than the best estimate flow. The best estimate flow is the most likely flow rate value for the actual plant operating condition.

4.4.2.9.8 Uncertainty in Mixing Coefficient

A conservative value of the mixing coefficient, that is, the thermal diffusion coefficient, is used in the VIPRE-01 analyses.

4.4.2.10 Flux Tilt Considerations

Significant quadrant power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation. A dropped or misaligned rod cluster control assembly could cause changes in hot channel factors. These events are analyzed separately in Chapter 15.

Other possible causes for quadrant power tilts include X-Y xenon transients, inlet temperature mismatches, enrichment variations within tolerances, and so forth.

In addition to unanticipated quadrant power tilts as described above, other readily explainable asymmetries may be observed during calibration of the ex-core detector quadrant power tilt alarm. During operation, in-core maps are taken at least one per month and additional maps are obtained periodically for calibration purposes. Each of these maps is reviewed for deviations from the expected power distributions.

Asymmetry in the core, from quadrant to quadrant, is frequently a consequence of the design when assembly and/or component shuffling and rotation requirements do not allow exact symmetry preservation. In each case, the acceptability of an observed asymmetry, planned or otherwise, depends solely on meeting the required accident analyses assumptions. In practice, once acceptability has been established by review of the incore maps, the quadrant power tilt alarms and related instrumentation are adjusted to indicate zero quadrant power tilt ratio as the final step in the calibration process. This action confirms that the instrumentation is correctly calibrated to alarm in the event an unexplained or unanticipated change occurs in the quadrant-to-quadrant relationships between calibration intervals.

Proper functioning of the quadrant power tilt alarm is significant. No allowances are made in the design for increased hot channel factors due to unexpected developing flux tilts, since likely causes are presented by design or procedures or are specifically analyzed.

Finally, in the event that unexplained flux tilts do occur, the Technical Specifications provide appropriate corrective actions to provide continued safe operation of the reactor.

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4.4.2.11 Fuel and Cladding Temperatures

Consistent with the thermal-hydraulic design bases described in Subsection 4.4.1, the following discussion pertains mainly to fuel pellet temperature evaluation. A description of fuel clad integrity is presented in Subsection 4.2.3.1.

The thermal-hydraulic design provides that the maximum fuel temperature is below the melting point of uranium dioxide, Subsection 4.4.1.2. To preclude center melting and to serve as a basis for overpower protection system setpoints, a calculated center-line fuel temperature of 4700°F is selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations, as described in Subsection 4.4.2.9.1. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the uranium dioxide thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad gap, and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, have been combined into a semi-empirical thermal model, discussed in Subsection 4.2.3.3, that includes a model for time-dependent fuel densification, as given in WCAP-10851-P-A (Reference 49) and WCAP-15063-P-A, Revision 1 (Reference 85). This thermal model enables the determination of these factors and their net effects on temperature profiles. The temperature predictions have been compared to in-pile fuel temperature measurements (References 30 through 36, 50 and 85) and melt radius data (References 51 and 52) with good results.

Fuel rod thermal evaluations (fuel centerline, average and surface temperatures) are performed at several times in the fuel rod lifetime (with consideration of time-dependent densification) to determine the maximum fuel temperatures.

The principal factors employed in the determination of the fuel temperature follow.

4.4.2.11.1 Uranium Dioxide Thermal Conductivity

The thermal conductivity of uranium dioxide was evaluated from data reported in References 37 through 48 and 53. At the higher temperatures, thermal conductivity is best obtained by using the integral conductivity to melt. From an examination of the data, it has been concluded that the best estimate is:

$$\int_{o}^{2800} K\,dt = 93\ W/cm$$

This conclusion is based on the integral values reported in References 51 and 53 through 57.

The design curve for the thermal conductivity is shown in Figure 4.4-2. The section of the curve at temperatures between 0° and 1300°C is in agreement with the recommendation of the International Atomic Energy Agency (IAEA) panel (Reference 58). The section of the curve above 1300°C is derived for an integral value of 93 W/cm. (References 51, 53, and 57).

Thermal conductivity for uranium dioxide at 95-percent theoretical density can be represented by the following equation:

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$$K = \frac{1}{11.8 + 0.0238\,T} + 8.775 \times 10^{-13}\,T^3$$

where:

K = W/cm-°C

T = °C.

4.4.2.11.2 Radial Power Distribution in Uranium Dioxide Fuel Rods

An accurate description of the radial power distribution as a function of burnup is needed for determining the power level for incipient fuel melting and other important performance parameters, such as pellet thermal expansion, fuel swelling, and fission gas release rates. Radial power distribution in uranium dioxide fuel rods is determined with the neutron transport theory code, LASER. The LASER code has been validated by comparing the code predictions on radial burnup and isotopic distributions with measured radial microdrill data, as detailed in WCAP-6069 (Reference 59) and WCAP-3385-56 (Reference 60). A radial power depression factor, f, is determined using radial power distributions predicted by LASER. The factor, f, enters into the determination of the pellet centerline temperature, T_c, relative to the pellet surface temperature, T_g, through the expression:

$$\int_{T_i}^{T_c} K(T)\,dT = \frac{q'\,f}{4\pi}$$

where:

K(T) = the thermal conductivity for uranium dioxide with a uniform density distribution

q' = the linear power generation rate

The corresponding correlation for an annular fuel pellet is:

$$\int_{T_s}^{T_c} K(T)\,dT = \frac{q'\,f}{4\pi}\left[1 - \frac{2\ln(R_0/R_i)}{(R_0/R_i)^2 - 1}\right]$$

where:

R_0 = outer radius of fuel pellet

R_i = radius of the central void

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4.4.2.11.3 Gap Conductance

The temperature drop across the pellet-clad gap is a function of the gap size and the thermal conductivity of the gas in the gap. The gap conductance model is selected so that when combined with the uranium dioxide thermal conductivity model, the calculated fuel center-line temperature reflect the in-pile temperature measurements. A more detailed description of the gap conductance model is presented in WCAP-10851-P-A (Reference 49) and WCAP-15063-P-A (Reference 85).

4.4.2.11.4 Surface Heat Transfer Coefficients

The fuel rod surface heat transfer coefficients during subcooled forced convection and nucleate boiling are presented in Subsection 4.4.2.7.1.

4.4.2.11.5 Fuel Clad Temperatures

The outer surface of the fuel rod at the hotspot operates at a temperature a few degrees above fluid temperature for steady-state operation at rated power throughout core life due to the onset of nucleate boiling. At beginning of life this temperature is the same as the clad metal outer surface.

During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the clad surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. Since the thermal-hydraulic design basis limits DNB, adequate heat transfer is provided between the fuel clad and the reactor coolant so that the core thermal output is not limited by considerations of clad temperature.

4.4.2.11.6 Treatment of Peaking Factors

The total heat flux hot channel factor, F_Q, is defined by the ratio of the maximum-to-core-average heat flux. The design value of F_Q, as presented in Table 4.3-2 and described in Subsection 4.3.2.2.6, is 2.6 for normal operation.

As described in Subsection 4.3.2.2.6, the peak linear power resulting from overpower transients/ operator errors (assuming a maximum overpower of 118 percent) is less than or equal to 22.45 kW/ft. The centerline fuel temperature must be below the uranium dioxide melt temperature over the lifetime of the rod, including allowances for uncertainties. The fuel temperature design basis is described in Subsection 4.4.1.2 and results in a maximum allowable calculated center-line temperature of 4700°F. The peak linear power for prevention of center-line melt is 22.5 kW/ft. The center-line temperature at the peak linear power resulting from overpower transients/operator errors (assuming a maximum overpower of 118 percent) is below that required to produce melting.

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

4.4.3.1 Plant Configuration Data

Plant configuration data for the thermal-hydraulic and fluid systems external to the core are provided as appropriate in Chapters 5, 6, and 9. Areas of interest are as follows:

- Total coolant flow rates for the reactor coolant system and each loop are provided in Table 5.1-3. Flow rates employed in the evaluation of the core are presented throughout Section 4.4.

- Total reactor coolant system volume including pressurizer and surge line and reactor coolant system liquid volume, including pressurizer water at steady-state power conditions, are given in Table 5.1-2.

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- The flow path length through each volume may be calculated from physical data provided in Table 5.1-2.
- Line lengths and sizes for the passive core cooling system are determined to provide a total system resistance which will provide, as a minimum, the fluid delivery rates assumed in the safety analyses described in Chapter 15.
- The parameters for components of the reactor coolant system are presented in Section 5.4.
- The steady-state pressure drops and temperature distributions through the reactor coolant system are presented in Table 5.1-1.

4.4.3.2 Operating Restrictions on Pumps

The minimum net positive suction head is established before operating the reactor coolant pumps. The operator verifies that the system pressure satisfies net positive suction head requirements prior to operating the pumps.

4.4.3.3 Power-Flow Operating Map (Boiling Water Reactor BWR)

This subsection is not applicable to AP1000.

4.4.3.4 Temperature-Power Operating Map (PWR)

The relationship between reactor coolant system temperature and power is a linear relationship between zero and 100-percent power.

The effects of reduced core flow due to inoperative pumps is described in Subsections 5.4.1 and 15.2.6 and Section 15.3. The AP1000 does not include power operation with one pump out of service. Natural circulation capability of the system is described in Subsection 5.4.2.3.2.

4.4.3.5 Load Following Characteristics

Load follow using control rod and gray rod motion is described in Subsection 4.3.2.4.16. The reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load, as described in Section 7.7.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal and hydraulic characteristics are given in Tables 4.1-1, 4.4-1, and 4.4-2.

4.4.4 Evaluation

4.4.4.1 Critical Heat Flux

The critical heat flux correlations used in the core thermal analysis are explained in Subsection 4.4.2.

4.4.4.2 Core Hydraulics

4.4.4.2.1 Flow Paths Considered in Core Pressure Drop and Thermal Design

The following flow paths for core bypass are considered:

A. Flow through the spray nozzles into the upper head for head cooling purposes

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B. Flow entering into the rod cluster control and gray rod cluster guide thimbles

C. Leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel

D. Flow introduced through the core shroud for the purpose of cooling and not considered available for core cooling

The above contributions are evaluated to confirm that the design value of the core bypass flow is met.

Of the total allowance, one part is associated with the core and the remainder is associated with the internals (items A, C, and D above). Calculations have been performed using drawing tolerances in the worst direction and accounting for uncertainties in pressure losses. Based on these calculations, the core bypass is no greater than the 5.9 percent design value.

Flow model test results for the flow path through the reactor are described in Subsection 4.4.2.7.2.

4.4.4.2.2 Inlet Flow Distributions

A core inlet flow distribution reduction of five percent to the hot assembly inlet is used in the VIPRE-01 analyses of DNBR in the AP1000 core. Studies shown in WCAP-8054-P-A (Reference 22), made with THINC-IV, a VIPRE-01 equivalent code, show that flow distributions significantly more nonuniform than five percent have a very small effect on DNBR, which is accounted for in the DNB analysis.

4.4.4.2.3 Empirical Friction Factor Correlations

The friction factor for VIPRE-01 in the axial direction, parallel to the fuel rod axis, is evaluated using a correlation for a smooth tube (Reference 83). The effect of two-phase flow on the friction loss is expressed in terms of the single-phase friction pressure drop and a two-phase friction multiplier. The multiplier is calculated using the homogenous equilibrium flow model.

The flow in the lateral directions, normal to the fuel rod axis, views the reactor core as a large tube bank. Thus, the lateral friction factor proposed by Idel'chik (Reference 64) is applicable. This correlation is of the form:

$$F_L = A\, Re_L^{-0.2}$$

where:

A = a function of the rod pitch and diameter as given in Idel'chik (Reference 64)

Re_L = the lateral Reynolds number based on the rod diameter

The comparisons of predictions to data given in Reference 83 verify the applicability of the VIPRE-01 correlations in PWR design.

4.4.4.3 Influence of Power Distribution

The core power distribution, which is largely established at beginning of life by fuel enrichment, loading pattern, and core power level, is also a function of variables such as control rod worth and

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position, and fuel depletion through lifetime. Radial power distributions in various planes of the core are often illustrated for general interest. However, the core radial enthalpy rise distribution, as determined by the integral of power up each channel, is of greater importance for DNBR analyses. These radial power distributions, characterized by $F_N^{\Delta H}$ (defined in Subsection 4.3.2.2.1), as well as axial heat flux profiles are discussed in the Subsections 4.4.4.3.1 and 4.4.4.3.2

4.4.4.3.1 Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^N$

Given the local power density q′ (kW/ft) at a point x, y, z in a core with N fuel rods and height H, then:

$$F_{\Delta H}^N = \frac{\text{hot rod power}}{\text{average rod power}} = \frac{\underset{o}{\overset{H}{\text{Max}}} \int q'(x_o, y_o, z_o)\, dz}{\frac{1}{N} \sum_{\text{all rods}} \int_o^H q'(x, y, z)\, dz}$$

The way in which $F_{\Delta H}^N$ is used in the DNBR calculation is important. The location of minimum DNBR depends on the axial profile, and the value of DNBR depends on the enthalpy rise to that point. Basically, the maximum value of the rod integral power is used to identify the most likely rod for minimum DNBR. An axial power profile is obtained that, when normalized to the design value of $F_{\Delta H}^N$, recreates the axial heat flux along the limiting rod. The surrounding rods are assumed to have the same axial profile with rod average powers which are typical distributions found in hot assemblies. In this manner, worst-case axial profiles can be combined with worst-case radial distributions for reference DNBR calculations.

It should be noted again that $F_{\Delta H}^N$ is an integral and is used as such in DNBR calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal power shapes throughout the core.

For operation at a fraction of full power, the design $F_{\Delta H}^N$ used is given by:

$$F_{\Delta H}^N = F_{\Delta H}^{RTP}[1 + 0.3(1 - P)]$$

where:

$F_{\Delta H}^N$ is the limit at rated thermal power (RTP):

P is the fraction of rated thermal power and $F_{\Delta H}^{RTP}$ = 1.654 (= 1.72 / 1.04).

The permitted relaxation of $F_{\Delta H}^N$ is included in the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits, as detailed in WCAP-7912-P-A (Reference 65). This allows greater flexibility in the nuclear design.

4.4.4.3.2 Axial Heat Flux Distributions

As described in Subsection 4.3.2.2, the axial heat flux distribution can vary as a result of rod motion or power change or as a result of a spatial xenon transient which may occur in the axial direction. The ex-core nuclear detectors, as described in Subsection 4.3.2.2.7, are used to measure the axial power imbalance. The information from the ex-core detectors is used to protect the core from excessive axial power imbalance. The reference axial shape used in establishing core DNB limits (that is,

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overtemperature ΔT protection system setpoints) is a chopped cosine with a peak-to-average value of 1.61. The reactor trip system provides automatic reduction of the trip setpoints on excessive axial power imbalance. To determine the magnitude of the setpoint reduction, the reference shape is supplemented by other axial shapes skewed to the bottom and top of the core.

The course of those accidents in which DNB is a concern is analyzed in Chapter 15 assuming that the protection setpoints have been set on the basis of these shapes. In many cases, the axial power distribution in the hot channel changes throughout the course of the accident due to rod motion, coolant temperature, and power level changes.

The initial conditions for the accidents for which DNB protection is required are assumed to be those permissible within the specified axial offset control limits described in Subsection 4.3.2.2. In the case of the loss-of-flow accident, the hot channel heat flux profile is very similar to the power density profile in normal operation preceding the accident. It is therefore possible to illustrate the calculated minimum DNBR for conditions representative of the loss-of-flow accident as a function of the flux difference initially in the core. The power shapes are evaluated with a full-power radial peaking factor ($F^N_{\Delta H}$) of 1.654 (= 1.72 / 1.04). The radial contribution to the hot rod power shape is conservative both for the initial condition and for the condition at the time of minimum DNBR during the loss-of-flow transient. The minimum DNBR is calculated for the design power shape for non-overpower/ overtemperature DNB events. This design shape results in calculated DNBR that bounds the normal operation shapes.

4.4.4.4 Core Thermal Response

A general summary of the steady-state thermal-hydraulic design parameters including thermal output and flow rates is provided in Table 4.4-1.

As stated in Subsection 4.4.1, the design bases of the application are to prevent DNB and to prevent fuel melting for Condition I and II events. The protective systems described in Chapter 7 are designed to meet these bases. The response of the core to Condition II transients is given in Chapter 15.

4.4.4.5 Analytical Methods

4.4.4.5.1 Core Analysis

The objective of reactor core thermal design is to determine the maximum heat removal capability in all flow subchannels and to show that the core safety limits, as presented in the technical specifications, are not exceeded while combining engineering and nuclear effects. The thermal design takes into account local variations in dimensions, power generation, flow redistribution, and mixing. The Westinghouse version of VIPRE-01, a three-dimensional subchannel code that has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core and hot channels, is described in Reference 83, VIPRE-01 modeling of a PWR core is based on a one-pass modeling approach (Reference 83). In the one-pass modeling, hot channels and their adjacent channels are modeled in detail, while the rest of the core is modeled simultaneously on a relatively coarse mesh. The behavior of the hot assembly is determined by superimposing the power distribution upon the inlet flow distribution while allowing for flow mixing and flow distribution between flow channels. Local variations in fuel rod power, fuel rod and pellet fabrication, and turbulent mixing are also considered in determining conditions in the hot channels. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow, and pressure drop.

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4.4.4.5.2 Steady State Analysis

The VIPRE-01 core model as approved by the NRC (Reference 83) is used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core under all expected operating conditions. The VIPRE-01 code is described in detail in Reference 84, including discussions on code validation with experimental data. The VIPRE-01 modeling method is described in Reference 83, including empirical models and correlations used. The effect of crud on the flow and enthalpy distribution in the core is not directly accounted for in the VIPRE-01 evaluations. However, conservative treatment by the Westinghouse VIPRE-01 modeling method has been demonstrated to bound this effect in DNBR calculations (Reference 83).

Estimates of uncertainties are discussed in Subsection 4.4.2.9.

4.4.4.5.3 Experimental Verification

Extensive additional experimental verification of VIPRE-01 is presented in Reference 84.

The VIPRE-01 analysis is based on a knowledge and understanding of the heat transfer and hydrodynamic behavior of the coolant flow and the mechanical characteristics of the fuel elements. The use of the VIPRE-01 analysis provides a realistic evaluation of the core performance and is used in the thermal hydraulic analyses as described above.

4.4.4.5.4 Transient Analysis

VIPRE-01 is capable of transient DNB analysis. The conservation equations in the VIPRE-01 code contain the necessary accumulation terms for transient calculations. The input description can include one or more of the following time dependent arrays:

1. Inlet flow variation
2. Core heat flux variation
3. Core pressure variation
4. Inlet temperature or enthalpy variation

At the beginning of the transient, the calculation procedure is carried out as in the steady state analysis. The time is incremented by an amount determined either by the user or by the time step control options in the code itself. At each new time step the calculations are carried out with the addition of the accumulation terms which are evaluated using the information from the previous time step. This procedure is continued until a preset maximum time is reached.

At time intervals selected by the user, a complete description of the coolant parameter distributions as well as DNBR is printed out. In this manner the variation of any parameter with time can be readily determined.

4.4.4.6 Hydrodynamic and Flow Power Coupled Instability

Boiling flow may be susceptible to thermohydrodynamic instabilities (Reference 68). These instabilities are undesirable in reactors, since they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components. Therefore, a thermo-hydraulic design criterion was developed which states that modes of operation under Condition I and II events shall not lead to thermohydrodynamic instabilities.

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Two specific types of flow instabilities are considered for AP1000 operation. These are the Ledinegg (or flow excursion) type of static instability and the density wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flow rate from one steady state to another. This instability occurs (Reference 68) when the scope of the reactor coolant system pressure drop-flow rate curve:

$$\left(\frac{\partial \Delta P}{\partial G} \mid \text{internal}\right)$$

becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve:

$$\left(\frac{\partial \Delta P}{\partial G} \mid \text{external}\right)$$

The criterion for stability is thus:

$$\left(\frac{\partial \Delta P}{\partial G}\Big|_{\text{internal}} \geq \frac{\partial \Delta P}{\partial G}\Big|_{\text{external}}\right)$$

The reactor coolant pump head curve has a negative slope ($\partial\Delta P/\partial G$ external less than zero), whereas the reactor coolant system pressure drop-flow curve has a positive slope ($\partial\Delta P/\partial G$ internal greater than zero) over the Condition I and Condition II operational ranges. Thus, the Ledinegg instability does not occur.

The mechanism of density wave oscillations in a heated channel has been described by R. T. Lahey and F. J. Moody (Reference 69). Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single-phase region and causes quality or void perturbations in the two-phase regions that travel up the channel with the flow. The quality and length perturbations in the two-phase region create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, then the two-phase pressure drop perturbation feeds back to the single-phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by M. Ishii (Reference 70) for parallel closed-channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability. This method had been used to assess the stability of typical Westinghouse reactor designs, including the design outlined in References 71, 72, and 73, under Condition I and II operation. The results indicate that a large margin-to-density wave instability exists. Increases on the order of 150 percent of rated reactor power would be required for the predicted inception of this type of instability.

The application of the Ishii method (Reference 70) to Westinghouse reactor designs is conservative due to the parallel open-channel feature of Westinghouse pressurized water reactor cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high-power density. There is also energy transfer from channels of high-power density to lower power density channels. This

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coupling with cooler channels leads to the conclusion that an open-channel configuration is more stable than the above closed-channel analysis under the same boundary conditions.

Flow stability tests (Reference 74) have been conducted where the closed channel systems were shown to be less stable than when the same channels were cross-connected at several locations. The cross-connections were such that the resistance to channel cross-flow and enthalpy perturbations would be greater than would exist in a pressurized water reactor core which has a relatively low resistance to cross-flow.

Flow instabilities that have been observed have occurred almost exclusively in closed-channel systems operating at low pressures relative to the Westinghouse pressurized water reactor operating pressures. H. S. Kao, T. D. Morgan, and W. B. Parker (Reference 75) analyzed parallel closed-channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 2200 psia. The results showed that, for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Additional evidence that flow instabilities do not adversely affect thermal margin is provided by the data from the rod bundle DNB tests. Many Westinghouse rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or inconsistent data which might be indicative of flow instabilities in the rod bundle.

In summary, it is concluded that thermohydrodynamic instabilities will not occur under Condition I and II for Westinghouse pressurized water reactor designs. A large power margin, greater than 150 percent of rated power, exists to predicted inception of such instabilities. Analysis has been performed which shows that minor plant-to-plant differences in Westinghouse reactor designs such as fuel assembly arrays, power-to-flow ratios, and fuel assembly length do not result in gross deterioration of the above power margins.

4.4.4.7 Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior are more pronounced than external blockages of the same magnitude. In both cases, the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where the reduction occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs similar to the VIPRE-01 program. Inspection of the DNB correlation (Subsection 4.4.2.2 and References 4 and 5) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

The VIPRE-01 code is capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs. Reference 84 shows that, for a fuel assembly similar to the Westinghouse design, VIPRE-01 accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked. Full recovery of the flow was found to occur about 30 inches downstream of the blockage. With the reactor operating at the nominal full-power conditions specified in Table 4.4-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the fuel rods reaching the DNBR limit.

The open literature supports the conclusion that flow blockage in open-lattice cores, similar to the Westinghouse cores, causes flow perturbations which are local to the blockage. For example, A. Ohstubo and S. Uruwashi (Reference 76) show that the mean bundle velocity is approached asymptomatically about four inches downstream from the flow blockage in a single flow cell. Similar

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results were also found for two and three cells completely blocked. P. Basmer, et al., (Reference 77) tested an open-lattice fuel assembly in which 41 percent of the subchannels were completely blocked in the center of the test bundle between spacer grids. Their results show that the stagnant zone behind the flow blockage essentially disappears after 1.65 L/De or about five inches for their test bundle. They also found that leakage flow through the blockage tended to shorten the stagnant zone or, in essence, the complete recovery length. Thus, local flow blockages within a fuel assembly have little effect on subchannel enthalpy rise. In reality, a local flow blockage would be expected to promote turbulence and, therefore would not likely affect DNBR at all.

Coolant flow blockages induce local cross-flows as well as promote turbulence. Fuel rod behavior is changed under the influence of a sufficiently high cross-flow component. Fuel rod vibration could occur, caused by this cross-flow component, through vortex shedding or turbulent mechanisms. If the cross-flow velocity exceeds the limit established for fluid elastic stability, large amplitude whirling results. The limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. The cross-flow velocity required to exceed fluid elastic stability limits is dependent on the axial location of the blockage and the characterization of the cross-flow (jet flow or not). These limits are greater than those for vibratory fuel rod wear. Cross-flow velocity above the established limits can lead to mechanical wear of the fuel rods at the grid support locations. Fuel rod wear due to flow-induced vibration is considered in the fuel rod fretting evaluation as discussed in Section 4.2.

4.4.5 Testing and Verification

4.4.5.1 Tests Prior to Initial Criticality

A reactor coolant flow test is performed, as discussed in Chapter 14, following fuel loading but prior to initial criticality. Coolant loop pressure data is obtained in this test. This data allows determination of the coolant flow rates at reactor operating conditions. This test verifies that proper coolant flow rates have been used in the core thermal and hydraulic analysis.

4.4.5.2 Initial Power and Plant Operation

Core power distribution measurements are made at several core power levels, as discussed in Chapter 14. These tests are used to confirm that conservative peaking factors are used in the core thermal and hydraulic analysis.

Additional demonstration of the overall conservatism of the THINC analysis was obtained by comparing THINC predictions to in-core thermocouple measurements, as detailed WCAP-8453-A (Reference 78). VIPRE-01 has been confirmed to be as conservative as the THINC code in Reference 83.

4.4.5.3 Component and Fuel Inspections

Inspections performed on the manufactured fuel are described in Subsection 4.2.4. Fabrication measurements critical to thermal and hydraulic analysis are obtained to verify that the engineering hot channel factors in the design analyses (Subsection 4.4.2.2.4) are met.

4.4.6 Instrumentation Requirements

4.4.6.1 Incore Instrumentation

The primary function of the incore instrumentation system is to provide a three-dimensional flux map of the reactor core. This map is used to calibrate neutron detectors used by the protection and safety monitoring system as well as to optimize core performance. A secondary function of the incore

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instrumentation system is to provide the protection and safety monitoring system with the signals necessary for monitoring core exit temperatures. This secondary function is the result of the mechanical design that groups the detectors used for generating the flux map in the same thimble as the core exit thermocouples.

The incore instrumentation system consists of incore instrument thimble assemblies, which house fixed incore detectors, core exit thermocouple assemblies contained within an inner and outer sheath assembly, and associated signal processing and data processing equipment. There are 42 incore instrument thimble assemblies: each is composed of multiple fixed incore detectors and one thermocouple.

The thimbles are inserted into the active core through the upper head and internals of the reactor vessel. The signals output from the fixed incore detectors are digitized inside containment and multiplexed out of the containment. The signal processing software integral to the incore instrumentation system allows the fixed incore detector signals to be used to calculate an accurate three-dimensional core power distribution suitable for developing calibration information for the excore nuclear instrumentation input to the overtemperature and overpower ΔT reactor trip setpoints. The system is also capable of accurately determining whether the reactor power distribution is currently within the operating limits defined in the technical specifications while the reactor is operating above approximately 20 percent of rated thermal power.

The incore instrument system data processor receives the transmitted digitized fixed incore detector signals from the signal processor and combines the measured data with analytically-derived constants, and certain other plant instrumentation sensor signals, to generate a full three-dimensional indication of nuclear power distribution in the reactor core. It also edits the three-dimensional indication of power distribution to extract pertinent power distribution parameters outputs for use by the plant operators and engineers. The data processor also generates hardcopy representations of the detailed three-dimensional nuclear power indications.

The hardware and software which perform the three-dimensional power distribution calculation are capable of executing the calculation algorithms and constructing graphical and tabular displays of core conditions at intervals of one minute or less. The software provides information to enable the reactor operator to ascertain how the measured peaking factor performance agrees with the peaking factor performance predicted by the design model used to determine the acceptability of the fuel loading pattern. The analysis software provides information required to activate a visual alarm display to alert the reactor operator about the current existence of, or the potential for, reactor operating limit violations. The calculation algorithms are capable of determining the three-dimensional reactor core power distribution using a minimum set of the total 42 in-core instrumentation thimble assemblies. Each in-core instrumentation thimble assembly consists of multiple fixed in-core detector elements that start at the top of the active fuel and have sequentially increasing lengths such that the longest element reaches the bottom of the active fuel in the fuel assembly. The calculation algorithms utilize the measured signal from detectors of different lengths within the assembly. The difference in signal from two operable detectors in the same assembly is defined as a detector segment. The minimum number of in-core monitor assemblies detectors required for operability of the system is at least 75% operating detector segments during the initial power distribution measurement required in each operating cycle; and at least 40% operating detector segments following the cycle initial power distribution measurement. A minimum of 15 operating detector segments in each quadrant with at least 6 detector segments in each quadrant below the core mid-plane and 6 detector segments per quadrant above the core midplane is required both prior to and following the cycle initial power distribution measurement. The hardware which performs the online power distribution monitoring is configured such that a single hardware failure will not necessitate a reactor maximum power reduction or restrict normal reactor operations.

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During plant operation, the incore instrument thimble assembly is positioned within the fuel assembly and exits through the top of the reactor vessel QuickLoc seal connection. The fixed incore detector and core exit thermocouple signal exit the detector through a multipin connector to the incore instrument thimble tube cables. The fixed incore detector and core exit thermocouple cables are then routed to different data conditioning and processing stations. The data is processed and the results are available for display in the main control room.

4.4.6.2 Overtemperature and Overpower ΔT Instrumentation

The overtemperature ΔT trip protects the core against low DNBR. The overpower ΔT trip protects against excessive power (fuel rod rating protection).

As described in Subsection 7.2.1.1.3, factors included in establishing the overtemperature ΔT and overpower ΔT trip setpoints include the reactor coolant temperature in each loop and the axial distribution of core power as seen by excore neutron detectors.

4.4.6.3 Instrumentation to Limit Maximum Power Output

The signals from the three ranges (source, intermediate, and power) of neutron flux detectors, are used to limit the maximum power output of the reactor within their respective ranges.

There are twelve radial locations containing a total of twelve neutron flux detectors installed around the reactor between the vessel and the primary shield. Four proportional counters for the source range are located at the highest fluence portions of the core containing the primary startup sources at an elevation approximately one-fourth of the core height. Four pulse fission chambers for the intermediate range are positioned at an elevation corresponding to one-half of the core height. Four uncompensated ionization chamber assemblies for the power range are installed vertically at the four corners of the core. These assemblies are located equidistant from the reactor vessel along the length and, to minimize neutron flux pattern distortions, within approximately one foot of the reactor vessel. Each power range detector provides two signals corresponding to the neutron flux in the upper and in the lower sections of a core quadrant. The three ranges of detectors are used as inputs to monitor neutron flux from a completely shutdown condition to 120 percent of full power, with the capability of recording overpower excursions up to 200 percent of full power.

The output of the power range channels is used for:

- Protecting the core against the consequences of rod ejection accidents
- Protecting the core against the consequences of adverse power distributions resulting from dropped rods
- The rod speed control function
- Alerting the operator to an excessive power imbalance between the quadrants

The intermediate range detectors also provide signals for the post-accident monitoring system.

Details of the neutron detectors and nuclear instrumentation design and the control and trip logic are given in Chapter 7. The limits on neutron flux operation and trip setpoints are given in the technical specifications.

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4.4.6.4 Digital Metal Impact Monitoring System

The digital metal impact monitoring system is the subsystem of the nonsafety-related special monitoring system that monitors the reactor coolant system for metallic loose parts. It consists of several active instrumentation channels, each comprising a piezoelectric accelerometer (sensor), signal conditioning, and diagnostic equipment. The digital metal impact monitoring system conforms with Regulatory Guide 1.133.

The digital metal impact monitoring system is designed to detect loose parts that weigh from 0.25 to 30 pounds, and can also detect impact with a kinetic energy of 0.5 foot-pounds on the inside surface of the reactor coolant system pressure boundary within three feet of a sensor.

The digital metal impact monitoring system consists of several redundant instrumentation channels, each comprised of a piezoelectric accelerometer (sensor), preamplifier, and signal conditioning equipment. The output signal from each accelerometer is amplified by the preamplifier and signal conditioning equipment before it is processed by a discriminator to eliminate noise and signals which are not indicative of loose part impacts. The system starts up and operates automatically.

The system facilitates performance tests, hardware integrity tests, and the recognition, location, replacement, repair and adjustment of malfunctioning components. System performance tests are made using a hammer as a tool to simulate an impact. Additional system performance testing is performed using special test modules. These modules simulate impacts and test performance of the signal processing equipment. Hardware integrity tests are also performed to verify equipment operation.

The impact detect algorithm, used by the signal processing equipment, is designed to minimize the number of false alarms. False impact detection, attributable to normal hydraulic, mechanical and electrical noise, is minimized by a number of techniques including:

- Utilizing a floating level within the impact detection algorithm. The floating level is based on signal levels not characteristic of an impact, and is generally a function of the background noise level.

- Comparing the impact event with the times and type of normally occurring plant operation events received from plant control system such as a control rod stepping.

- Comparing the number of events detected within a given time interval.

The sensors of the digital metal impact monitoring system are fastened mechanically to the reactor coolant system at potential loose part collection regions including the upper and lower head region of the reactor pressure vessel, and the reactor coolant inlet region of each steam generator.

The equipment inside the containment is designed to remain functional through an earthquake of a magnitude equal to 50 percent of the calculated safe shutdown earthquake and normal environments (radiation, vibration, temperature, humidity) anticipated during the operating lifetime. The instrument channels associated with the sensors at each reactor coolant system location are physically separated from each other starting at the sensor locations to a point in the plant that is always accessible for maintenance during full-power operation.

The digital metal impact monitoring system is calibrated prior to plant startup. Capabilities exist for subsequent periodic online channel checks and channel functional tests and for offline channel calibrations at refueling outages.

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4.4.7 Combined License Information

4.4.7.1 Changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the DCD are addressed in APP-GW-GLR-059 (Reference 87).

4.4.7.2 Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in Subsection 7.1.6, the design limit DNBR values will be calculated. The calculations will be completed using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in this section remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty. This will be completed prior to fuel load.

4.4.8 References

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 4.4-1 (Sheet 1 of 2)
Thermal and Hydraulic Comparison Table
(AP1000, AP600 and a Typical Westinghouse XL Plant)

Design Parameters	AP1000[a]	AP600	Typical XL Plant
Reactor core heat output (MWt)	3400	1933	3800
Reactor core heat output (10^6 BTU/hr)	11601	6596	12,969
Heat generated in fuel (%)	97.4	97.4	97.4
System pressure, nominal (psia)	2250	2250	2250
System pressure, minimum (psia)	2190	2200	2204
Minimum DNBR at nominal conditions			
Typical flow channel	2.59	3.48	2.20
Thimble (cold wall) flow channel)	2.60	3.33	2.12
Minimum DNBR for design transients			
Typical flow channel	>1.25[b]	>1.23	>1.26
Thimble (cold wall) flow channel	>1.25[b]	>1.22	>1.24
DNB correlation[c]	WRB-2M	WRB-2	WRB-1
Coolant conditions[d]			
Vessel minimum measured flow rate (MMF)			
10^6 lbm/hr	115.55	74.4	148.9
gpm	301,670	193,200	403,000
Vessel thermal design flow rate (TDF)			
10^6 lbm/hr	113.5	72.9	145.0
gpm	296,000	189,600	392,000
Effective flow rate for heat transfer[e]			
10^6 lbm/hr	106.8	66.3	132.7
gpm	278,500	172,500	358,700
Effective flow area for heat transfer (ft^2)	41.8	38.5	51.1
Average velocity along fuel rods (ft/s)[e]	15.8	10.6	16.6
Average mass velocity, 10^6 lbm/hr-ft^2[e]	2.55	1.72	2.60
Coolant temperature[d][e]			
Nominal inlet (°F)	535.0	532.8	561.2
Average rise in vessel (°F)	77.2	69.6	63.6
Average rise in core (°F)	81.4	75.8	68.7
Average in core (°F)	578.1	572.6	597.8
Average in vessel (°F)	573.6	567.6	593.0

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Table 4.4-1 (Sheet 2 of 2)
Thermal and Hydraulic Comparison Table
(AP1000, AP600 and a Typical Westinghouse XL Plant)

Design Parameters	AP1000[a]	AP600	Typical XL Plant
Heat transfer			
Active heat transfer surface area (ft^2)[f]	56,700	44,884	69,700
Average heat flux (BTU/hr-ft^2)	199,300	143,000	181,200
Maximum heat flux for normal operation (BTU/hr-ft^2)[g]	518,200	372,226	498,200
Average linear power (kW/ft)[f][m]	5.72	4.11	5.20
Peak linear power for normal operation (kW/ft)[g,h]	14.9	10.7	14.0
Peak linear power resulting from overpower transients/operator errors, assuming a maximum overpower of 118% (kW/ft)[h]	$\leq$22.45	22.5	$\leq$22.45
Peak Linear power for prevention of center-line melt (kW/ft)[i]	22.5	22.5	22.45
Power density (kW/liter of core)[j]	109.7	78.82	98.8
Specific power (kW/kg uranium)[j]	40.2	28.89	36.6
Fuel central temperature			
Peak at peak linear power for prevention of centerline melt (°F)	4700	4,700	4700
Pressure drop[k]			
Across core (psi)	38.7 $\pm$ 3.9[l]	17.5 $\pm$ 1.7	38.8 $\pm$ 3.9
Across vessel, including nozzle (psi)	64.8 $\pm$ 6.5[l]	45.3 $\pm$ 4.5	59.7 $\pm$ 6.0

Notes:
(a) Robust Fuel Assembly.
(b) The design limit DNBR is 1.25.
(c) WRB-2M is used for AP1000. WRB-2, ABB-NV, or WLOP is used for AP1000 where WRB-2M is not applicable. See Subsection 4.4.2.2.1 for use of ABB-NV, WLOP, WRB-2 and WRB-2M correlations.
(d) Based on vessel average temperature equal to 573.6°F. Flow rates and temperatures based on 10 percent steam generator tube plugging.
(e) Based on thermal design flow and 5.9 percent bypass flow.
(f) Based on densified active fuel length. The value for AP1000 is rounded to 5.72 kW/ft.
(g) Based on 2.60 F_Q peaking factor.
(h) See Subsection 4.3.2.2.6.
(i) See Subsection 4.4.2.11.6.
(j) Based on cold dimensions and 95.5 percent of theoretical density fuel for AP1000; 95 percent for others.
(k) These are typical values based on best-estimate reactor flow rate as discussed in Section 5.1.
(l) Inlet temperature = 536.8°F.
(m) The value for AP1000 is rounded to 5.72 kW/ft.

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Table 4.4-2
Void Fractions At Nominal Reactor Conditions
With Design Hot Channel Factors
(Based On VIPRE-01)

	Average	Maximum
Core, %	0.0	-
Hot Subchannel, %	0.3	2.1

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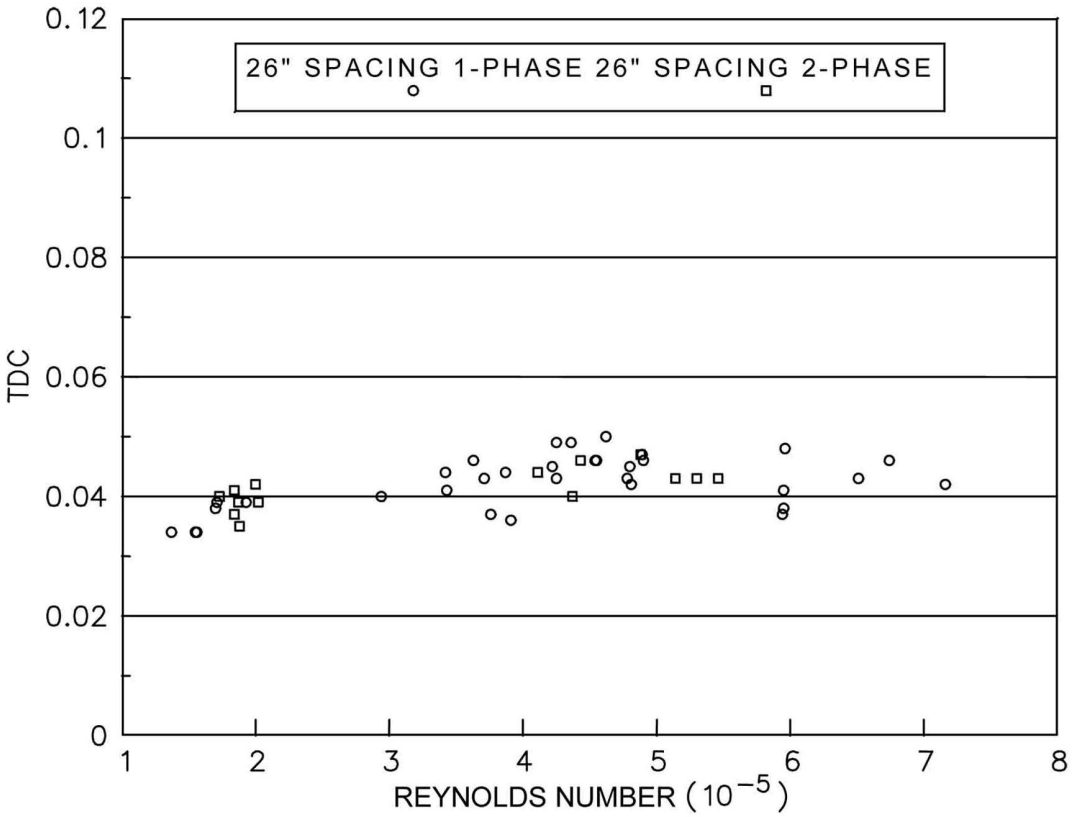

Figure 4.4-1
Thermal Diffusion Coefficient (TDC)
as a Function of Reynolds Number

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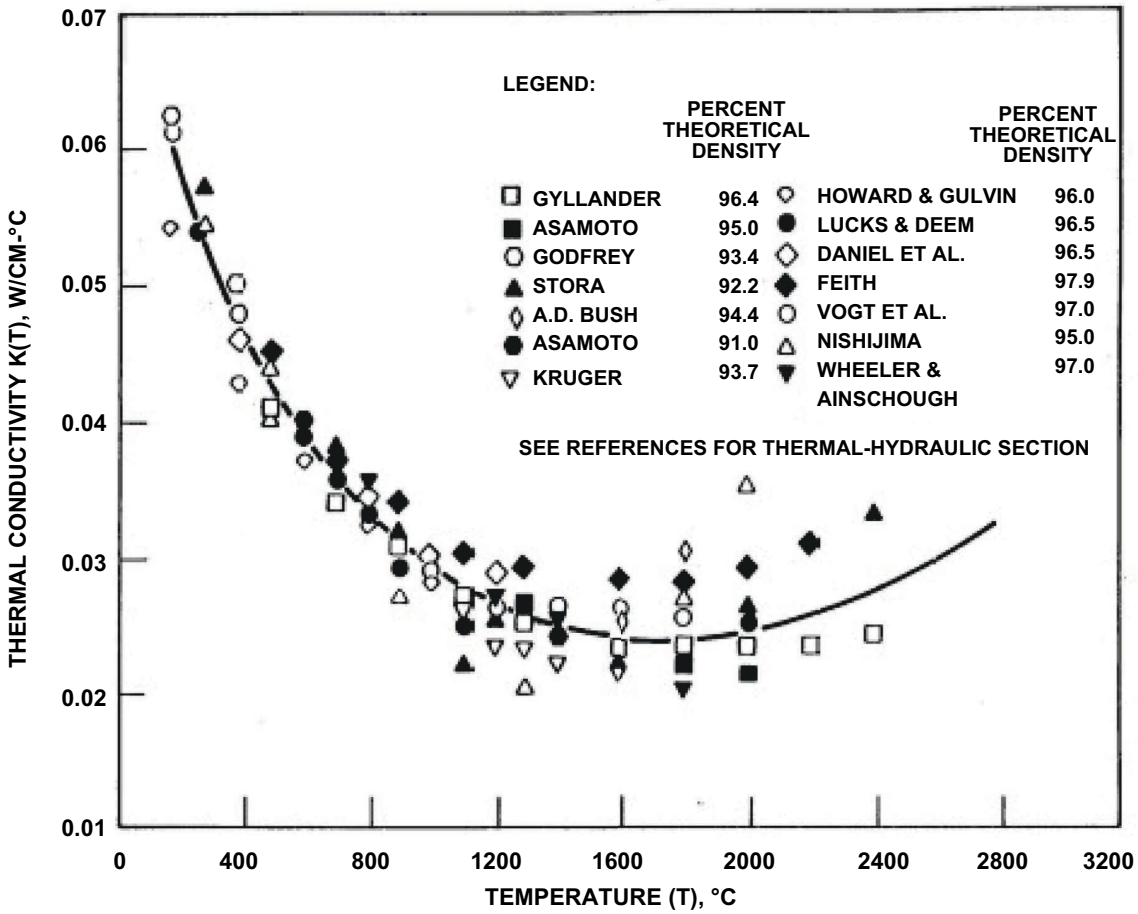

Figure 4.4-2
Thermal Conductivity of Uranium Dioxide
(Data Corrected to 95% Theoretical Density)

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4.5 Reactor Materials

4.5.1 Control Rod and Drive System Structural Materials

4.5.1.1 Materials Specifications

The parts of the control rod drive mechanisms and control rod drive line exposed to reactor coolant are made of metals that resist the corrosive action of the coolant. Three types of metals are used exclusively: stainless steels, nickel-chromium-iron alloys, and, to a limited extent, cobalt-based alloys. These materials have provided many years of successful operation in similar control rod drive mechanisms. In the case of stainless steels, only austenitic and martensitic stainless steels are used.

Pressure-containing materials comply with the ASME Code, Section III. The material specifications for portions of the control rod drive mechanism that are reactor coolant pressure boundary are included in Table 5.2-1. These parts are fabricated from austenitic (Type 316, 316L, 316LN and Type 304, 304L, 304LN) stainless steel. Nickel-chromium-iron alloy (Alloy 690) is used for the reactor vessel head penetration. For pressure boundary parts, austenitic stainless steels are not used in the heat-treated conditions which can cause susceptibility to stress-corrosion cracking or accelerated corrosion in pressurized water reactor coolant chemistry and temperature environments. Pressure boundary parts and components made of stainless steel do not have specified minimum yield strength greater than 90,000 psi.

The material selection is based in part on the duty cycle specified for the control rod drive mechanisms and control rods. The materials are specified so that the components do not suffer adverse effects, such as excessive wear or galling, as a result of a minimum 300 trips from full power and 60 coupling and decoupling cycles of the drive rod coupling assembly. The material for the control rod drive mechanisms and the control rod assemblies are selected for acceptable performance. That is, the design goal is to achieve a service life of 9×10^6 full-step cycles. Inspection or changes in operation indicate the need for replacement or refurbishment. The worst case result of undetected wear of a control rod drive mechanism or drive rod is a rod assembly drop or a failure to drop an assembly during a trip. Both events are accounted for in safety analyses. The pressure boundary components are not subject to significant wear due to stepping cycles.

Internal latch assembly parts are fabricated of heat-treated martensitic and austenitic stainless steel. Heat treatment is such that stress-corrosion cracking is not initiated. Components and parts made of stainless steel do not have specified minimum yield strength greater than 90,000 psi. Magnetic pole pieces are immersed in the reactor coolant and are fabricated from Type 410 stainless steel. Nonmagnetic parts, except shims, pins, and springs, are fabricated from Type 304 stainless steel. A cobalt alloy or qualified substitute is used to fabricate the latch, link, and link pins. Springs and shims are made from nickel-chromium-iron alloy (Alloy X-750 and Alloy 625). Lock screws are fabricated of Type 316 stainless steel. Latch arm tips fabricated of stainless steel may be surfaced with a suitable hard facing material to provide improved resistance to wear. Hard chrome plate is used selectively for bearing and wear surfaces.

The drive rod assembly is also immersed in the reactor coolant and uses a Type 410 stainless steel drive rod. The drive rod coupling is machined from Type 403 or 410 stainless steel. The protective sleeve and disconnect button are also Type 410 stainless steel. The remaining parts are Type 304 or Type 304L stainless steel with the exception of the springs, button retainer, and locking button, which are fabricated of nickel-chromium-iron alloy.

The absorber rodlets in the rod cluster control assemblies and the gray rod cluster assemblies are closed stainless steel tubes (cladding) containing absorber material. The stainless steel cladding isolates from the reactor coolant, the absorber material, and other substances inside the tubes. The containment function of the control rod cladding and the effects of neutron flux in the control rod

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materials are addressed in Section 4.2. The outside surface of the absorber rodlet is chromium plated or ion nitrided to enhance resistance to wear due to the stepping motion and vibration of the rods. The rods included in one rod cluster control assembly or gray rod cluster assembly are attached at the top to a common hub which connects with the drive rod of the control rod drive mechanism. The hub is fabricated from Type 304, Type 304L, or Grade CF-3 stainless steel.

The coil housing is exposed to containment atmosphere and requires a magnetic material. Low carbon cast steel and ductile iron are qualified by tests or other evaluations for this application. The finished housings are electroless nickel plated to provide resistance against general corrosion.

Coils are wound on composite bobbins, with double glass-insulated copper wire. Coils are vacuum impregnated with silicone varnish. A wrapping of mica sheet is secured to the coil outside diameter. The result is a well-insulated coil capable of sustained operation at 392°F (200°C).

4.5.1.2 Fabrication and Processing of Austenitic Stainless Steel Components

The discussions provided in Subsection 5.2.3.4 concerning the processes, inspections, and tests on austenitic stainless steel components to prevent increased susceptibility to intergranular corrosion caused by sensitization are applicable to the austenitic stainless steel pressure-housing components of the control rod drive mechanism. The discussions provided in Subsection 5.2.3.4, concerning the control of welding of austenitic stainless steels especially control of delta ferrite are also applicable. Subsection 5.2.3.4 discusses the compliance with the guidelines of Regulatory Guides 1.31, 1.34, and 1.44. The welded control rod drive mechanism austenitic stainless steels that come into contact with the primary reactor coolant meet the guidance of Regulatory Guide 1.44.

4.5.1.3 Other Materials

For the cobalt alloy used to fabricate the latch, link, and link pins in latch assemblies, stress-corrosion cracking has not been observed in this application. Where hardfacing material is used in the latch assembly, a cobalt base alloy equivalent to Stellite-6 is used.

The springs in the control rod drive mechanism are made from nickel-chromium-iron alloy (Alloy 750), ordered to Aerospace Material Specification (AMS) 5698 or AMS 5699 with additional restrictions on prohibited materials. Operating experience has shown that springs made of this material are not subject to stress-corrosion cracking in pressurized water reactor primary water environments. Alloy 750 is not used for bolting applications in the control rod drive mechanisms.

4.5.1.4 Contamination Protection and Cleaning of Austenitic Stainless Steel

The control rod drive mechanisms are cleaned prior to delivery in accordance with the guidance provided in NQA-1 (see Chapter 17). Process specifications in packaging and shipment are discussed in Subsection 5.2.3. Westinghouse personnel conduct surveillance of these operations to verify that manufacturers and installers adhere to appropriate requirements as described in Subsection 5.2.3.

Tools used in abrasive work operations on austenitic stainless steel, such as grinding or wire brushing, do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress-corrosion cracking.

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4.5.2 Reactor Internal and Core Support Materials

4.5.2.1 Materials Specifications

The major core support material for the reactor internals is SA-182, SA-336, SA-376, SA-479, or SA-240 Types 304, 304L, 304LN, or 304H stainless steels. Fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with Regulatory Guide 1.44. For threaded structural fasteners the material used is strain hardened Type 316 stainless steel (XM-19 as large diameter alternate) and for the clevis insert-to-vessel bolts either UNS N07718 or N07750. Remaining internals parts not fabricated from Types 304, 304L, 304LN, or 304H stainless steels typically include wear surfaces such as hardfacing on the radial keys, clevis inserts, alignment plates (Stellite™ 6 or 156 or low cobalt hardfaces); dowel pins (Type 316, XM-19 as large diameter alternate); and SB-166 for the clevis inserts.

Additional product forms that may be used by the reactor internals structures are SA-193, SA-213, SA-312, SA-358, and SB-637. Reactor internals structures materials that are not Types 304, 304L, 304LN, or 304H stainless steel are the guide bushings locating and support pins, instrumentation adapter, instrument tube tip, instrument tube sleeve top, guide stud (all UNS S21800), compression fitting (Type 316), hold down spring (SA-182, F6NM), irradiation specimen access plug springs and instrument stalk springs (UNS N07750), instrument tube sleeve spring (UNS N07718), and SB-168 for the flow skirt. Additionally, reactor internals structures will use either threaded structural fasteners of strain hardened Type 316 stainless steel, UNS N07718 or N07750 (XM-19 as large diameter alternate). ASTM A240 or ASTM A479 Type 300-series stainless steel may be used for small components classified as internal structures such as locking cups, locking caps, and shims.

Core support structure, internal structure, and threaded structural fastener materials are specified in the ASME Code, Section II, Part D, Subpart 1, Tables 2A, 2B, and 4 as supplemented by Code Case N-60, except for ASTM A240 and ASTM A479 which may be used for small components classified as internal structures. The qualification of cobalt free wear resistant alloys for use in reactor coolant is addressed in Subsection 4.5.1.3.

The use of cast austenitic stainless steel (CASS) is minimized in the AP1000 reactor internals. If used, CASS will be limited in carbon (low carbon grade: L grade) and ferrite contents and will be evaluated in terms of thermal aging effects.

The estimated peak neutron fluence for the AP1000 reactor internals has been considered in the design. Susceptibility to irradiation-assisted stress corrosion cracking or void swelling in reactor internals identified in the current pressurized water reactor fleet are being addressed in reactor internals material reliability programs. The selection of materials for the AP1000 reactor internals considers information developed by these programs. Ni-Cr-Fe Alloy 600 is not used in the AP1000 reactor internals.

4.5.2.2 Controls on Welding

The discussions provided in Subsection 5.2.3.4 are applicable to the welding of reactor internals and core support components.

4.5.2.3 Nondestructive Examination of Tubular Products and Fittings

The nondestructive examination and acceptance during fabrication of wrought seamless tubular products and fitting material are in accordance with ASME Code, Section III, Article NG-2500. The acceptance standards for welding post-assembly are in accordance with the requirements of ASME Code, Section III, Article NG-5300.

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4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel Components

The discussions provided in Subsection 5.2.3.4 and Section 1.9 describes the conformance of reactor internals and core support structures with Regulatory Guides 1.31 and 1.44.

The discussion provided in Section 1.9 describes the conformance of reactor internals with Regulatory Guides 1.34 and 1.71.

4.5.2.5 Contamination Protection and Cleaning of Austenitic Stainless Steel

The discussions provided in Subsection 5.2.3 and Section 1.9 are applicable to the reactor internals and core support structures describe the conformance of the process specifications with Regulatory Guide 1.37. The process specifications follow the guidance of NQA-1.

4.5.3 Combined License Information

This section contained no requirement for additional information.

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4.6 Functional Design of Reactivity Control Systems

4.6.1 Information for Control Rod Drive System

The control rod drive mechanism (CRDM) and operation of the control rod drive system are described in Subsection 3.9.4. Figure 3.9-4 provides the details of the control rod drive mechanisms. Figure 4.2-8 provides the configuration of the driveline, including the control rod drive mechanism. No hydraulic system is associated with the functioning of the control rod drive system. The instrumentation and controls for the reactor trip system are described in Section 7.2. The reactor control system is described in Section 7.7.

The control rod drive mechanisms are contained within an integrated head package located on top of the reactor vessel head as described in Subsection 3.9.7. This assembly provides the support required for seismic restraint in conjunction with the attachment of the control rod drive mechanisms to the reactor vessel head. An outer shroud and the seismic restraint structure isolate the control rod drive mechanisms from the effects of ruptures of high-energy lines outside the shroud, and from missiles. The shroud also is used to direct air from the cooling fans past the control rod drive mechanisms. The cooling system maintains the temperatures of the coils in the control rod drive mechanisms below the design operating temperature. The integrated head package provides the proper support and required separation for electrical lines providing power to the control rod drive mechanisms and signals from the rod position sensors.

The line for the reactor head vent system is located among the control rod drive mechanisms and is supported by the integrated head package. This line is pressurized to reactor coolant system pressure and considered to be a high-energy line. This line is constructed to the appropriate requirements of the ASME Code. Figure 3.9-7 shows elements of the integrated head package surrounding the control rod drive mechanisms.

4.6.2 Evaluations of the Control Rod Drive System

Rod control systems of the type used in the AP1000 have been analyzed in detailed reliability studies. These studies include fault tree analysis and failure mode and effects analyses. These studies, and the analyses presented in Chapter 15, demonstrate that the control rod drive system performs its intended safety-related function – a reactor trip. The control rod drive system puts the reactor in a subcritical condition when a safety-related system setting is reached with an assumed credible failure of a single active component.

The essential elements of the control rod drive system (those required to provide reactor trip) are isolated from nonessential portions of the rod control system by the reactor trip switchgear, as described in Section 7.2. The essential portion of the control rod drive system is shielded from the direct effects of postulated moderate- and high-energy line breaks by the integrated head package. The dynamic effects of pipe ruptures do not have to be considered for those pipes that satisfy the requirements for mechanistic pipe break, as outlined in Subsection 3.6.3.

The reactor vessel head vent lines and instrumentation conduits are one inch nominal diameter or smaller. Breaks in lines of this size do not have to be postulated for dynamic effects, pressurization, and spray wetting. The pressure boundary housing of the control rod drive mechanisms is constructed to the requirements of the ASME Code and a break in this pressure boundary is not credible.

The only instrumentation required of the control rod drive mechanism and supporting systems to operate safely is the rod position indicator. A break in the cables connected to the rod position indicators would neither preclude a reactor trip, nor would it result in an unplanned withdrawal of a rod assembly. A break in the power cable to the control rod drive mechanism coils results in a drop of

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the rod assembly. Information on the pressure and temperature of the control rod drive mechanisms and surrounding areas is not required for safe operation. The design pressure and temperature of the control rod drive mechanism housing is the same as the reactor coolant system, which is protected by safety valves. Overheating of the control rod drive mechanism coils due to a failure of the cooling system would in the worst case result in a drop of one or more rod assemblies. The reactor and reactor protection system is designed to accommodate and protect against rod drop events. Additional information is provided in Subsection 3.9.1, and Sections 7.2, and 15.4.

4.6.3 Testing and Verification of the Control Rod Drive System

The control rod drive system is extensively tested prior to its operation. These tests may be subdivided into five categories:

- Prototype tests of components
- Prototype control rod drive system tests
- Production tests of components following manufacture and prior to installation
- Onsite pre-operational and initial startup tests
- Periodic in-service tests

These tests, which are described in Subsection 3.9.4.4 and Sections 4.2 and 14.2, are conducted to verify the operability of the control rod drive system when called upon to function.

4.6.4 Information for Combined Performance of Reactivity Systems

As indicated in Chapter 15, there are only three postulated events that assume credit for reactivity control systems, other than a reactor trip to render the plant subcritical. These events are the steam-line break, feedwater line break, and small break loss of coolant accident. The reactivity control systems in these accidents are the reactor trip system and the passive core cooling system (PXS). Additional information on the control rod drive system is presented in Subsection 3.9.4. The passive core cooling system is discussed further in Section 6.3.

No credit is taken for the boration capabilities of the chemical and volume control system (CVS) as a system in the analysis of transients presented in Chapter 15. Information on the capabilities of the chemical and volume control system is provided in Subsection 9.3.6. The adverse boron dilution possibilities due to the operation of the chemical and volume control system are investigated in Subsection 15.4.6. Prior proper operation of the chemical and volume control system has been presumed as an initial condition to evaluate transients. Appropriate technical specifications promote the correct operation or remedial action.

The AP1000 instrumentation and control system includes a diverse actuation system (DAS). This system provides for automatic control rod insertion, turbine trip, passive residual heat removal heat exchanger start, core makeup tank start, isolation of critical containment penetrations, and start of the passive containment cooling system as appropriate upon conditions indicative of an anticipated transient without scram or other failure of the plant control and reactor protection system. This system is diverse and independent from the reactor trip system from the sensor through actuation devices.

In addition to the above, the AP1000 plant systems provide for operator response to an anticipated transient without scram (ATWS) event that includes core reactivity control followed by core decay heat removal. Core reactivity control is provided by a manual trip of the control rods, insertion of the

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control rods, the chemical and volume control system, or by the core makeup tank injection. The decay heat removal can be performed by the startup feedwater system or the passive residual heat removal system.

4.6.5 Evaluation of Combined Performance

The evaluations of the steam-line break, the feedwater line break, and the small break loss of coolant accident, which presume the combined actuation of the reactor trip system and the control rod drive system and the passive safety injection, are presented in Subsections 15.1.5 and 15.2.8 and Section 15.6. Reactor trip signals and signals to actuate passive safety features for these events are generated from functionally diverse sensors. These signals actuate diverse means of reactivity control; that is, control rod insertion and injection of soluble neutron absorber.

Non-diverse but redundant types of equipment are used only in the processing of the incoming sensor signals into appropriate logic which initiates the protective action. This equipment is described in Sections 7.2 and 7.3. In particular, protection from equipment failures is provided by redundant equipment and periodic testing. Effects of failures of this equipment have been extensively investigated. Reliability studies, including failure mode and effects analysis for this type of equipment verify that a single failure does not have an adverse effect upon the engineered safety features actuation system. Adequacy of the passive core cooling system performance under faulted conditions is verified in Section 6.3.

In addition to the automatic actuations provided for by the diverse actuation system, that system also provides for manual actuation of the reactor trip.

The probability of a common mode failure impairing the ability of the reactor trip system to perform its safety-related function is extremely low. However, analyses are performed to demonstrate compliance with the requirements of 10 CFR 50.62. These analyses demonstrate that safety criteria would not be exceeded even if the control rod drive system were rendered incapable of functioning during anticipated transients for which its function would normally be expected. The evaluation demonstrates that borated water from the core makeup tank shuts down the reactor with no rods required, and the passive residual heat removal system provides sufficient core heat removal.

4.6.6 Combined License Information

This section contained no requirement for additional information.

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Chapter 5 Reactor Coolant System and Connected Systems

5.1 Summary Description

This section describes the reactor coolant system (RCS) and includes a schematic flow diagram of the reactor coolant system (Figure 5.1-1), an isometric view of the reactor coolant loops and major components (Figure 5.1-2), a sketch of the loop layout (Figure 5.1-3), and a sketch of the elevation of the reactor coolant system (Figure 5.1-4). The piping and instrumentation diagram (Figure 5.1-5, sheets 1, 2, and 3) shows additional details of the design of the reactor coolant system.

5.1.1 Design Bases

The performance and safety design bases of the reactor coolant system and its major components are interrelated. These design bases are listed as follows:

- The reactor coolant system transfers to the steam and power conversion system the heat produced during power operation as well as the heat produced when the reactor is subcritical, including the initial phase of plant cooldown.
- The reactor coolant system transfers to the normal residual heat removal system the heat produced during the subsequent phase of plant cooldown and cold shutdown.
- During power operation and normal operational transients (including the transition from forced to natural circulation), the reactor coolant system heat removal maintain fuel condition within the operating bounds permitted by the reactor control and protection systems.
- The reactor coolant system provides the water used as the core neutron moderator and reflector conserving thermal neutrons and improving neutron economy. The reactor coolant system also provides the water used as a solvent for the neutron absorber used in chemical shim reactivity control.
- The reactor coolant system maintains the homogeneity of the soluble neutron poison concentration and the rate of change of the coolant temperature so that uncontrolled reactivity changes do not occur.
- The reactor coolant system pressure boundary accommodates the temperatures and pressures associated with operational transients.
- The reactor vessel supports the reactor core and control rod drive mechanisms.
- The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, the pressurizer accommodates volume changes in the reactor coolant.
- The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.
- The steam generators provide high-quality steam to the turbine. The tubes and tubesheet boundary prevent the transfer of radioactivity generated within the core to the secondary system.

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- The reactor coolant system piping contains the coolant under operating temperature and pressure conditions and limits leakage (and activity release) to the containment atmosphere. The reactor coolant system piping contains demineralized and borated water that is circulated at the flow rate and temperature consistent with achieving the reactor core thermal and hydraulic performance.

- The reactor coolant system is monitored for loose parts, as described in Subsection 4.4.6.

- Applicable industry standards and equipment classifications of reactor coolant system components are identified in Tables 3.2-1 and 3.2-3 of Subsection 3.2.2.

- The reactor vessel head is equipped with suitable provisions for connecting the head vent system, which meets the requirements of 10 CFR 50.34 (f)(2)(vi) (TMI Action Item II.B.1). (See Subsection 5.4.12.)

- The pressurizer surge line and each loop spray line connected with the reactor coolant system are instrumented with resistance temperature detectors (RTDs) attached to the pipe to detect thermal stratification.

5.1.2 Design Description

Figure 5.1-1 shows a schematic of the reactor coolant system. Table 5.1-1 provides the principal pressures, temperatures, and flow rates of the system at the locations noted in Figure 5.1-1 under normal steady-state, full-power operating conditions. These parameters are based on the best-estimate flow at the pump discharge. Table 5.1-2 contains a summary of nominal system design and operating parameters under normal steady-state, full-power operating conditions. These parameters are based on the best-estimate conditions at nominal full power. The reactor coolant system volume under these conditions is also provided.

The reactor coolant system consists of two heat transfer circuits, each with a steam generator, two reactor coolant pumps, and a single hot leg and two cold legs for circulating reactor coolant. In addition, the system includes the pressurizer, interconnecting piping, valves, and instrumentation for operational control and safeguards actuation. All reactor coolant system equipment is located in the reactor containment.

During operation, the reactor coolant pumps circulate pressurized water through the reactor vessel then the steam generators. The water, which serves as coolant, moderator, and solvent for boric acid (chemical shim control), is heated as it passes through the core. It is transported to the steam generators where the heat is transferred to the steam system. Then it is returned to the reactor vessel by the pumps to repeat the process.

The reactor coolant system pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to provide a high degree of integrity throughout operation of the plant.

The reactor coolant system pressure is controlled by operation of the pressurizer, where water and steam are maintained in equilibrium by the activation of electrical heaters or a water spray, or both. Steam is formed by the heaters or condensed by the water spray to control pressure variations due to expansion and contraction of the reactor coolant.

Spring-loaded safety valves are installed above and connected to the pressurizer to provide overpressure protection for the reactor coolant system. These valves discharge into the containment atmosphere. Three stages of reactor coolant system automatic depressurization valves are also connected to the pressurizer. These valves discharge steam and water through spargers to the

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in-containment refueling water storage tank (IRWST) of the passive core cooling system (PXS). Most (initially all) of the steam and water discharged to the spargers is condensed and cooled by mixing with the water in the tank.

The fourth-stage automatic depressurization valves are connected by two redundant paths to each reactor coolant loop hot leg and discharge directly to the containment atmosphere.

The reactor coolant system is also served by a number of auxiliary systems, including the chemical and volume control system (CVS), the passive core cooling system (PXS), the normal residual heat removal system (RNS), the steam generator system (SGS), the primary sampling system (PSS), the liquid radwaste system (WLS), and the component cooling water system (CCS).

The reactor coolant system includes the following:

- The reactor vessel, including control rod drive mechanism housings.
- The reactor coolant pumps, consisting of four sealless pumps that pump fluid through the entire reactor coolant and reactor systems. Two pumps are coupled with each steam generator.
- The portion of the steam generators containing reactor coolant, including the channel head, tubesheet, and tubes.
- The pressurizer which is attached by the surge line to one of the reactor coolant hot legs. With a combined steam and water volume, the pressurizer maintains the reactor system within a narrow pressure range.
- The safety and automatic depressurization system valves.
- The reactor vessel head vent isolation valves.
- The interconnecting piping and fittings between the preceding principal components.
- The piping, fittings, and valves leading to connecting auxiliary or support systems.

The piping and instrumentation diagram of the reactor coolant system (Figure 5.1-5) shows the extent of the systems located within the containment and the interface between the reactor coolant system and the secondary (heat utilization) system.

Figures 5.1-3 and 5.1-4 show the plan and section of the reactor coolant loops. These figures show reactor coolant system components in relationship to supporting and surrounding steel and concrete structures. The figures show the protection provided to the reactor coolant system by its physical layout.

5.1.3 System Components

The major components of the reactor coolant system are described in the following subsections. Additional details of the design and requirements of these components are found in other sections of this safety analysis report.

5.1.3.1 Reactor Vessel

The reactor vessel is cylindrical, with a hemispherical bottom head and removable, flanged, hemispherical upper head. The vessel contains the core, core support structures, control rods, and

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other parts directly associated with the core. The vessel interfaces with the reactor internals, the integrated head package, and reactor coolant loop piping and is supported on the containment building concrete structure.

The design of the AP1000 reactor vessel closely matches the existing vessel designs of Westinghouse three-loop plants. New features for the AP1000 have been incorporated without departing from the proven features of existing vessel designs.

The vessel has inlet and outlet nozzles positioned in two horizontal planes between the upper head flange and the top of the core. The nozzles are located in this configuration to provide an acceptable cross-flow velocity in the vessel outlet region and to facilitate optimum layout of the reactor coolant system equipment. The inlet and outlet nozzles are offset, with the inlet positioned above the outlet, to allow mid-loop operation for removal of a main coolant pump without discharge of the core.

Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

5.1.3.2 AP1000 Steam Generator

The AP1000 steam generator (SG) is a vertical shell and U-tube evaporator with integral moisture separating equipment. The basic steam generator design and features have been proven in tests and in previous steam generators including replacement steam generator designs.

Design enhancements include nickel-chromium-iron Alloy 690 thermally treated tubes on a triangular pitch, improved antivibration bars, single-tier separators, enhanced maintenance features, and a primary-side channel head design that allows for easy access and maintenance by robotic tooling. The AP1000 steam generator employs tube supports utilizing a broached hole support plate design. All tubes in the steam generator are accessible for sleeving, if necessary. The design enhancements are based on proven technology.

The basic function of the AP1000 steam generator is to transfer heat from the single-phase reactor coolant water through the U-shaped heat exchanger tubes to the boiling, two-phase steam mixture in the secondary side of the steam generator. The steam generator separates dry, saturated steam from the boiling mixture, and delivers the steam to a nozzle from which it is delivered to the turbine. Water from the feedwater system replenishes the steam generator water inventory by entering the steam generator through a feedwater inlet nozzle and feedring.

In addition to its steady-state performance function, the steam generator secondary side provides a water inventory which is continuously available as a heat sink to absorb primary side high temperature transients.

5.1.3.3 Reactor Coolant Pumps

The AP1000 reactor coolant pumps are high-inertia, high-reliability, low-maintenance, sealless pumps of canned motor design that circulate the reactor coolant through the reactor vessel, loop piping, and steam generators. The pumps are integrated into the steam generator channel head.

The integration of the pump suction into the bottom of the steam generator channel head eliminates the cross-over leg of coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support system for the steam generator, pumps, and piping; and reduces the potential for uncovering of the core by eliminating the need to clear the loop seal during a small loss of coolant accident.

The AP1000 design uses four pumps. Two pumps are coupled with each steam generator.

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Each AP1000 reactor coolant pump is a vertical, single-stage centrifugal pump designed to pump large volumes of main coolant at high pressures and temperatures. Because of its sealless design, it is more tolerant of off-design conditions that could adversely affect shaft seal designs. The main impeller attaches to the rotor shaft of the driving motor, which is an electric induction motor. The stator and rotor of the motor are both encased in corrosion-resistant cans constructed and supported to withstand full system pressure.

Primary coolant circulates between the stator and rotor which obviates the need for a seal around the motor shaft. Additionally, the motor bearings are lubricated by primary coolant. The motor is thus an integral part of the pump. The basic pump design has been proven by many years of service in other applications.

The pump motor size is minimized through the use of a variable frequency drive to provide speed control in order to reduce motor power requirements during pump startup from cold conditions. The variable frequency drive is used only during heatup and cooldown when the reactor trip breakers are open. During power operations, the drive is isolated and the pump is run at constant speed.

To provide the rotating inertia needed for flow coast-down, bi-metallic flywheel assemblies are attached to the pump shaft.

5.1.3.4 Primary Coolant Piping

Reactor coolant system piping is configured with two identical main coolant loops, each of which employs a single 31-inch inside diameter hot leg pipe to transport reactor coolant to a steam generator. The two reactor coolant pump suction nozzles are welded directly to the outlet nozzles on the bottom of the steam generator channel head. Two 22-inch inside diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the reactor vessel to complete the circuit.

The loop configuration and material have been selected such that pipe stresses are sufficiently low for the primary loop and large auxiliary lines to meet the requirements to demonstrate "leak-before-break." Thus, pipe rupture restraints are not required, and the loop is analyzed for pipe ruptures only for small auxiliary lines that do not meet the leak-before-break requirements.

5.1.3.5 Pressurizer

The AP1000 pressurizer is a principal component of the reactor coolant system pressure control system. It is a vertical, cylindrical vessel with hemispherical top and bottom heads, where liquid and vapor are maintained in equilibrium saturated conditions.

One spray nozzle and two nozzles for connecting the safety and depressurization valve inlet headers are located in the top head. Electrical heaters are installed through the bottom head. The heaters are removable for replacement. The bottom head contains the nozzle for attaching the surge line. This line connects the pressurizer to a hot leg, and provides for the flow of reactor coolant into and out of the pressurizer during reactor coolant system thermal expansions and contractions.

5.1.3.6 Pressurizer Safety Valves

The pressurizer safety valves are spring loaded, self-actuated with back-pressure compensation. Their set pressure and combined capacity is based on not exceeding the reactor coolant system maximum pressure limit during the Level B service condition loss of load transient.

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5.1.3.7 Reactor Coolant System Automatic Depressurization Valves

Some of the functions of the AP1000 passive core cooling system (PXS) are dependent on depressurization of the reactor coolant system. This is accomplished by the automatically actuated depressurization valves. The automatic depressurization valves connected to the pressurizer are arranged in six parallel sets of two valves in series opening in three stages.

A set of fourth-stage automatic depressurization valves is connected to each reactor coolant hot leg. Each set of valves consists of two parallel paths of two valves in series.

To mitigate the consequences of the various accident scenarios, the controls are arranged to open the valves in a prescribed sequence based on core makeup tank level and a timer as described in Section 6.3.

5.1.4 System Performance Characteristics

Table 5.1-3 lists the nominal thermal hydraulic parameters of the reactor coolant system. The system performance parameters are also determined for an assumed 10 percent uniform steam generator tube plugging condition.

Reactor coolant flow is established by a detailed design procedure supported by operating plant performance data and component hydraulics experimental data. The procedure establishes a best-estimate flow and conservatively high and low flows for the applicable mechanical and thermal design considerations. In establishing the range of design flows, the procedure accounts for the uncertainties in the component flow resistances and the pump head-flow capability, established by analysis of the available experimental data. The procedure also accounts for the uncertainties in the technique used to measure flow in the operating plant.

Definitions of the four reactor coolant flows applied in various plant design considerations are presented in the following paragraphs.

5.1.4.1 Best-Estimate Flow

The best-estimate flow is the most likely value for the normal full-power operating condition. This flow is based on the best estimate of the fuel, reactor vessel, steam generator, and piping flow resistances, and on the best estimate of the reactor coolant pump head and flow capability. The best-estimate flow provides the basis for the other design flows required for the system and component design. The best-estimate flow and head also define the performance requirement for the reactor coolant pump. Table 5.1-1 lists system parameters based on best-estimate flow.

The best-estimate flow analysis is based on extensive experimental data, including accurate flow and pressure drop data from an operating plant, flow resistance measurements from several fuel assembly hydraulics tests, and hydraulic performance measurements from several pump impeller model tests. Since operating plant flow measurements are in close agreement with the calculated best-estimate flows, the flows established with this design procedure can be applied to the plant design with a high level of confidence.

Although the best-estimate flow is the most likely value to be expected in operation, more conservative flow rates are applied in the thermal and mechanical designs.

5.1.4.2 Minimum Measured Flow

The minimum measured flow is specified in the technical specifications as the flow that must be confirmed or exceeded by the flow measurements obtained during plant startup. This is the flow used

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in reactor core departure from nucleate boiling (DNB) analysis for the thermal design procedure used in the AP1000. In the thermal design procedure methodology for DNB analysis, flow measurement uncertainties are combined statistically with fuel design and manufacturing uncertainties.

The measured reactor coolant flow will most likely differ from the best-estimate flow because of uncertainties in the hydraulics analysis and the inaccuracies in the instrumentation used to measure flow. The measured flow is expected to fall within a range around the best-estimate flow. The magnitude of the expected range is established by statistically combining the system hydraulics uncertainty with the total flow rate within the expected range, less any excess flow margin that may be provided to account for future changes in the hydraulics of the reactor coolant system.

5.1.4.3 Thermal Design Flow

The thermal design flow is the conservatively low value used for thermal-hydraulic analyses where the design and measurement uncertainties are not combined statistically, and additional flow margin must therefore be explicitly included. The thermal design flow is derived by subtracting the plant flow measurement uncertainty from the minimum measured flow. The thermal design flow is approximately 4.5 percent less than the best-estimate flow. The thermal design flow is confirmed when the plant is placed in operation. Table 5.1-3 provides tabulations of important design parameters based on the thermal design flow.

5.1.4.4 Mechanical Design Flow

Mechanical design flow is the conservatively high flow used as the basis for the mechanical design of the reactor vessel internals, fuel assemblies, and other system components. Mechanical design flow is established at 104 percent of best-estimate flow.

5.1.5 Combined License Information

This section contained no requirement for additional information.

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Table 5.1-1
Principal System Pressures, Temperatures, and Flow Rates
(Nominal Steady-State, Full Power Operating Conditions)

Location (Fig. 5.1-1)	Description	Fluid	Pressure (psig)	Nominal Temp. (°F)	Flow[(a)] (gpm)
1	Hot Leg 1	Reactor Coolant	2247	610	177,645
2	Hot Leg 2	Reactor Coolant	2247	610	177,645
3	Cold Leg 1A	Reactor Coolant	2323	537.2	78,750
4	Cold Leg 1B	Reactor Coolant	2323	537.2	78,750
5	Cold Leg 2A	Reactor Coolant	2323	537.2	78,750
6	Cold Leg 2B	Reactor Coolant	2323	537.2	78,750
7	Surge Line Inlet	Reactor Coolant	2248	610	-
8	Pressurizer Inlet	Reactor Coolant	2241	653.0	-
9	Pressurizer Liquid	Reactor Coolant	2235	653.0	-
10	Pressurizer Steam	Steam	2235	653.0	-
11	Pressurizer Spray 1A	Reactor Coolant	2317	537.2	1 - 2
12	Pressurizer Spray 1B	Reactor Coolant	2317	537.2	1 - 2
13	Common Spray Line	Reactor Coolant	2317	537.2	2 - 4
14	ADS Valve Inlet	Steam	2234	653.0	-
15	ADS Valve Inlet	Steam	2234	653.0	-

Note:
(a) At the conditions specified.

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Table 5.1-2
Nominal System Design and Operating Parameters

General	
Plant design objective, years	60
NSSS power, MWt	3415
Reactor coolant pressure, psia	2250
Reactor coolant liquid volume at power conditions (including 1000 ft^3 pressurizer liquid), ft^3	9600
Loops	
Number of cold legs	4
Number of hot legs	2
Hot leg ID, in.	31
Cold leg ID, in.	22
Reactor Coolant Pumps	
Type of reactor coolant pumps	Sealless
Number of reactor coolant pumps	4
Estimated motor rating, hp	7300
Effective pump power to coolant, MWt	15
Pressurizer	
Number of units	1
Total volume, ft^3	2100
Water volume, ft^3	1000
Spray capacity, gpm	700
Inside diameter, in.	100
Height, in.	503
Steam Generator	
Steam generator power, MWt/unit	1707.5
Type	Vertical U-tube Feedring-type
Number of units	2
Surface area, ft^2/unit	123,540
Shell design pressure, psia	1200
Zero load temperature, °F	557
Feedwater temperature, °F	440
Exit steam pressure, psia	836
Steam flow, lb/hr per steam generator	7.50×10^6
Total steam flow, lb/hr	14.99×10^6

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Table 5.1-3
Thermal-Hydraulic Parameters
(Nominal)

Detailed Thermal-Hydraulic Parameters		
Best-Estimate Flow (BEF)	**Without Plugging**	**With 10% Tube Plugging**
Flow rate, gpm/loop	157,500	155,500
Reactor vessel outlet temperature, °F	610.0	610.4
Reactor vessel inlet temperature, °F	537.2	536.8
Minimum Measured Flow (MMF)		
Flow rate, gpm/loop	152,775	150,835
Thermal Design Flow (TDF)		
Flow rate, gpm/loop	149,940	148,000
Reactor vessel outlet temperature, °F	611.7	612.2
Reactor vessel inlet temperature, °F	535.5	535.0
Mechanical Design Flow (MDF)		
Flow rate, gpm/flow	163,800	
Best-Estimate Reactor Core and Vessel Thermal-Hydraulic Parameters		**Without Plugging**
NSSS power, MWt		3415
Reactor power, MWt		3400
Best-Estimate loop flow, gpm/loop		157,500
Best-Estimate vessel flow, lb/hr		120.5×10^6
Best-Estimate core flow, lb/hr		113.4×10^6
Reactor coolant pressure, psia		2250
Vessel/core inlet temperature, °F		537.2
Vessel average temperature, °F		573.6
Vessel outlet temperature, °F		610.0
Average core outlet temperature, °F		615.9
Total core bypass flow, (percent of total flow)		5.9
Core barrel nozzle flow		1.0
Head cooling flow		1.5
Thimble flow		1.9
Core shroud cooling flow		0.5
Unallocated bypass flow		1.0

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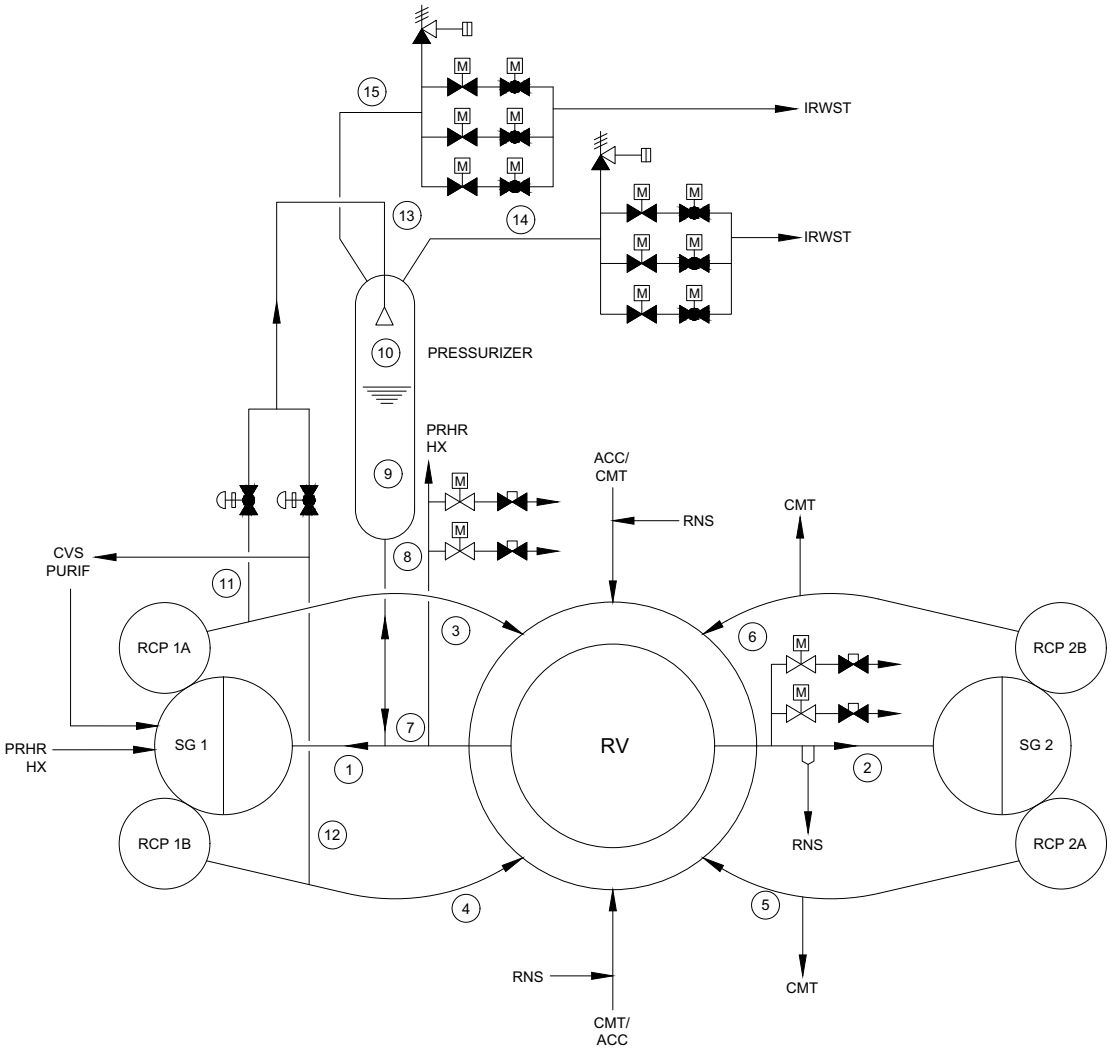

Figure 5.1-1
Reactor Coolant System Schematic Flow Diagram

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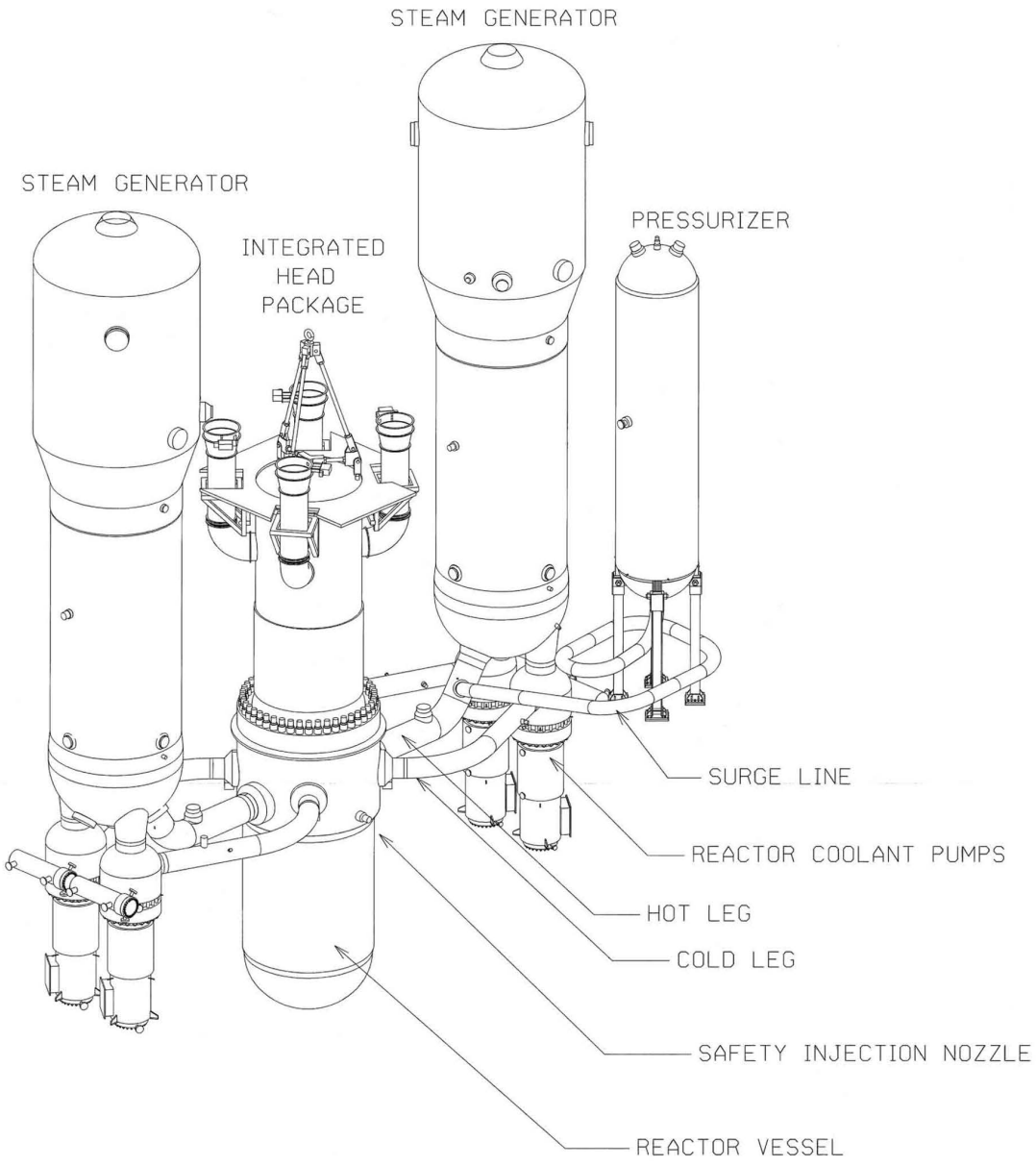

Figure 5.1-2
Reactor Coolant Loops – Isometric View

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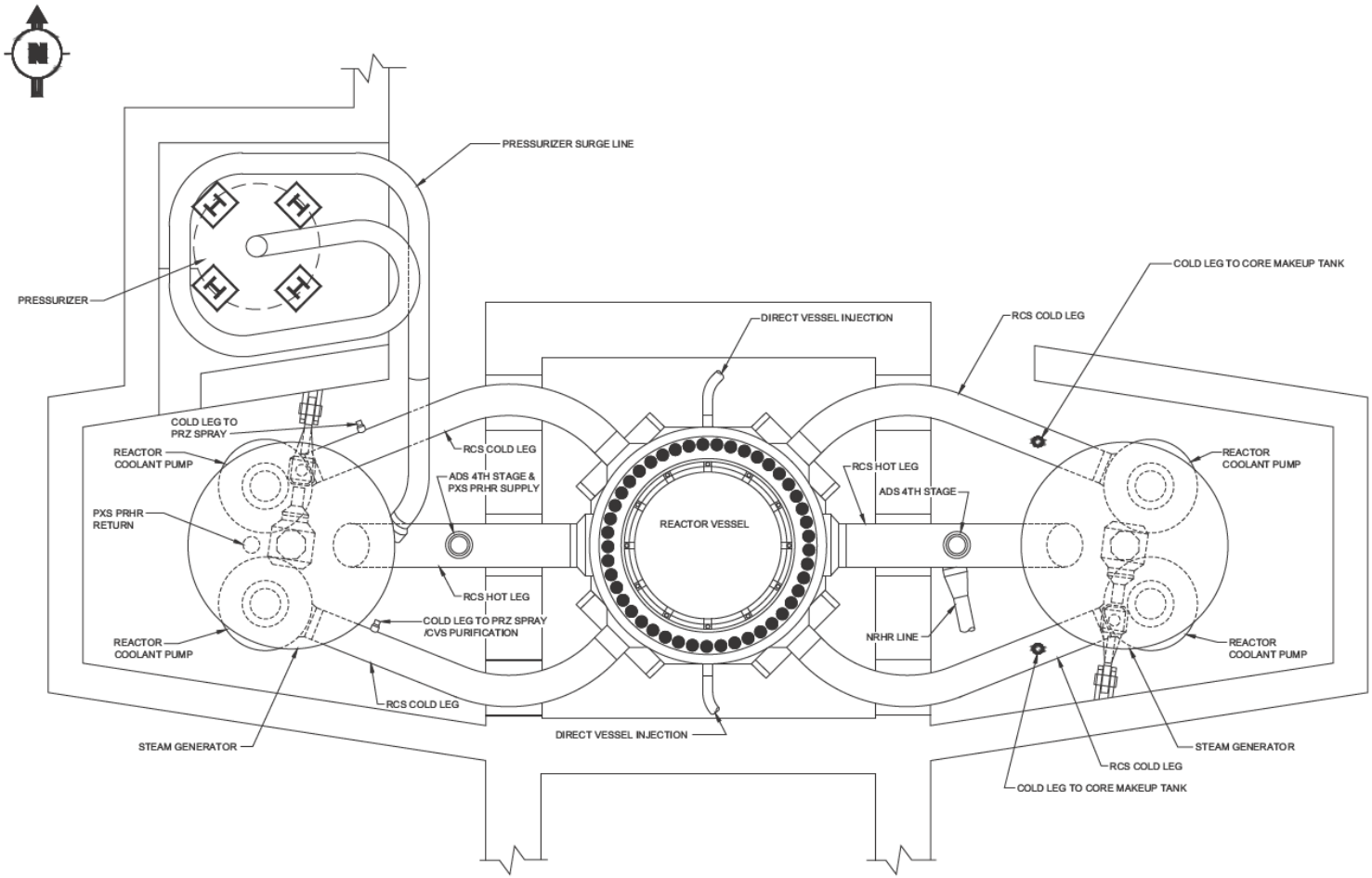

Figure 5.1-3
Reactor Coolant System – Loop Layout

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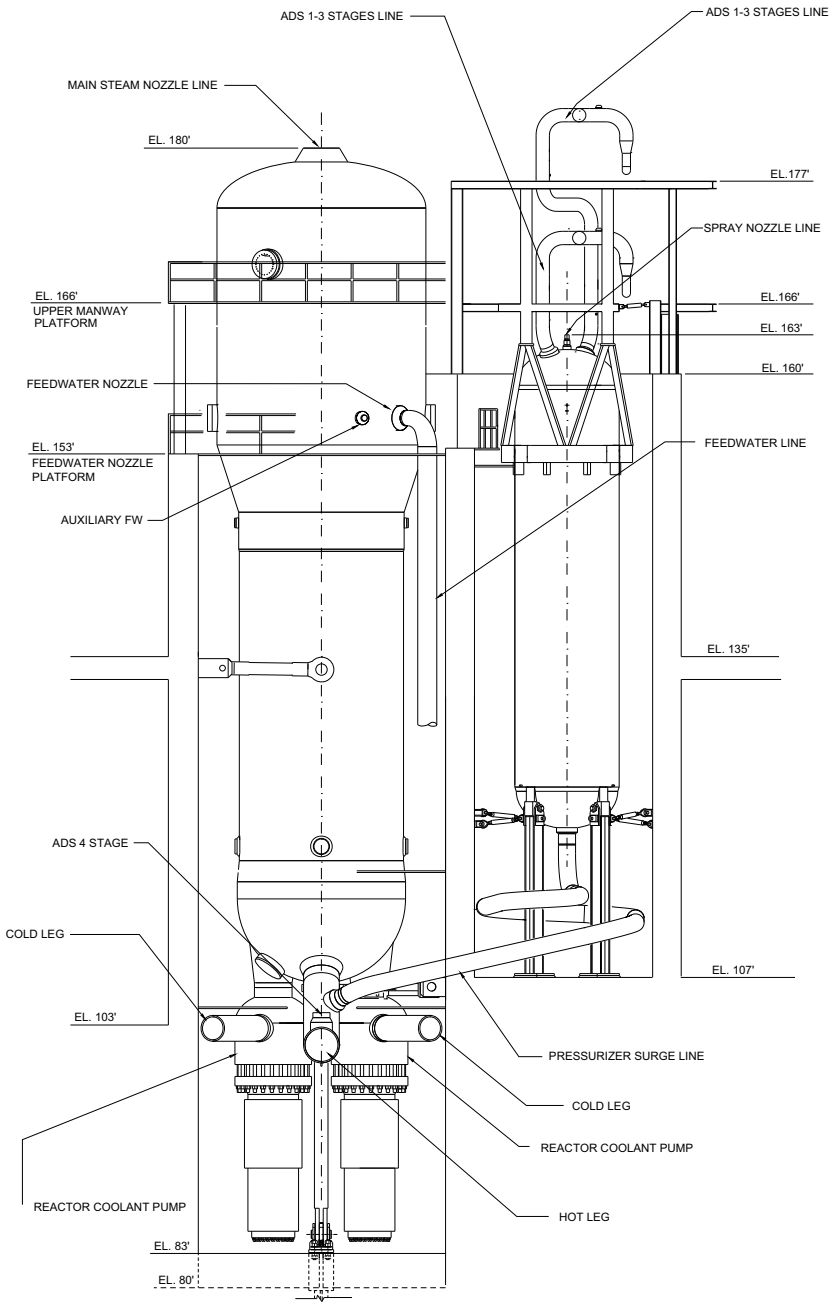

NOTE: THE ELEVATIONS INCLUDED IN THIS FIGURE
ARE ROUNDED TO THE NEAREST FOOT.

Figure 5.1-4
Reactor Coolant System – Elevation

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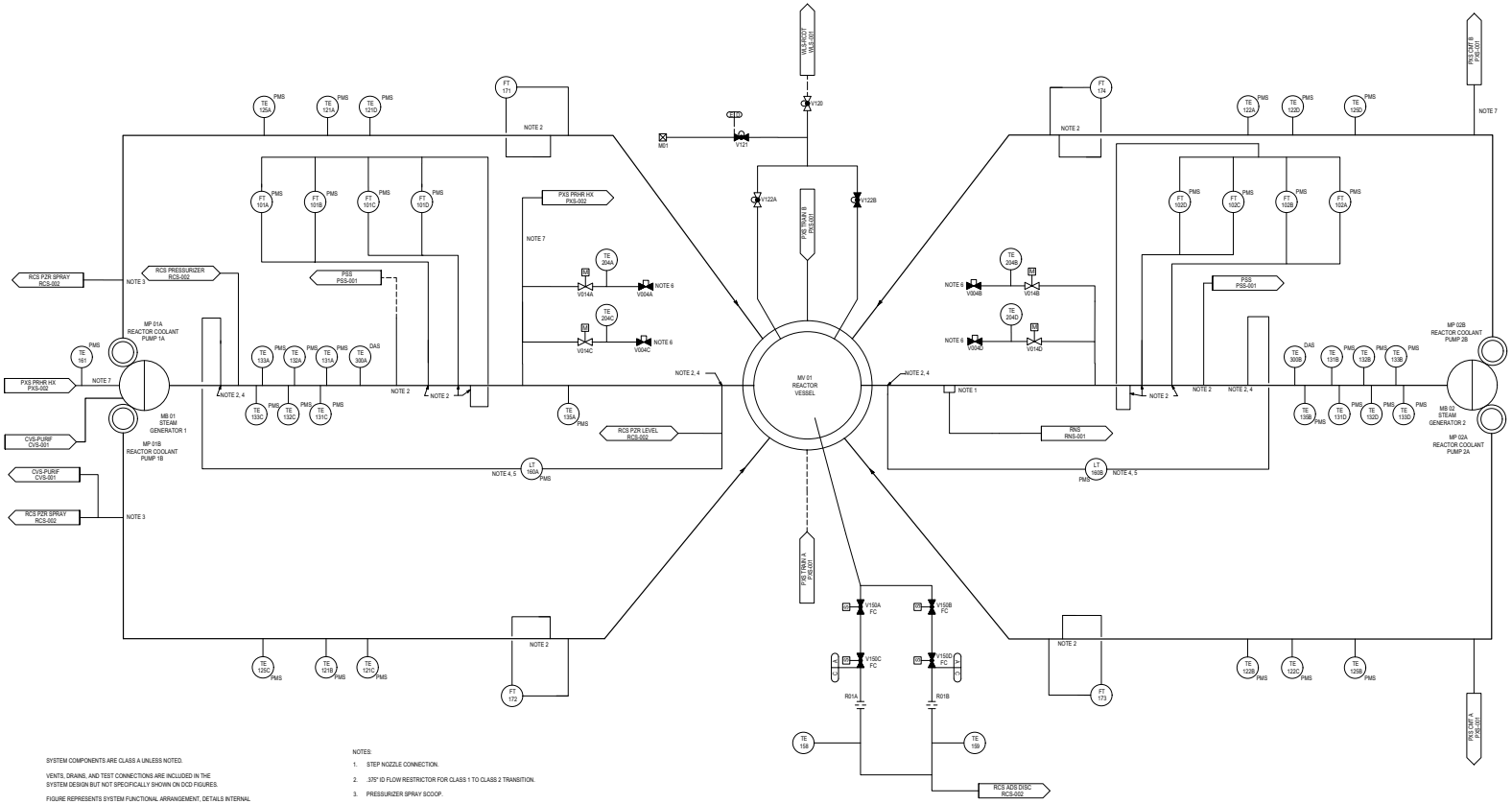

SYSTEM COMPONENTS ARE CLASS A UNLESS NOTED.

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

NOTES:

1. STEP NOZZLE CONNECTION.
2. .375" ID FLOW RESTRICTOR FOR CLASS 1 TO CLASS 2 TRANSITION.
3. PRESSURIZER SPRAY SCOOP.
4. LEVEL INSTRUMENTATION PROVIDED TO MONITOR RCS WATER LEVEL DURING MID-LOOP OPERATION FOLLOWING SHUTDOWN.
5. PMS - OPENS IRWST INJECTION VALVES ON LOW-4 (EMPTY) HOT LEG LEVEL PLUS 30 MINUTE TIME DELAY.
6. THE 4TH STAGE ADS VALVES DISCHARGE TO THE STEAM GENERATOR COMPARTMENTS AT AN ELEVATION GREATER THAN 110'.
7. ROUTE ALL ASSOCIATED PIPING TO HIGH POINT VENT WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE VENT LOCATION FOR THOSE PIPING SEGMENTS.

Inside Reactor Containment

Figure 5.1-5 (Sheet 1 of 3)
Simplified Reactor Coolant System Piping and Instrumentation Diagram (REF) RCS 001

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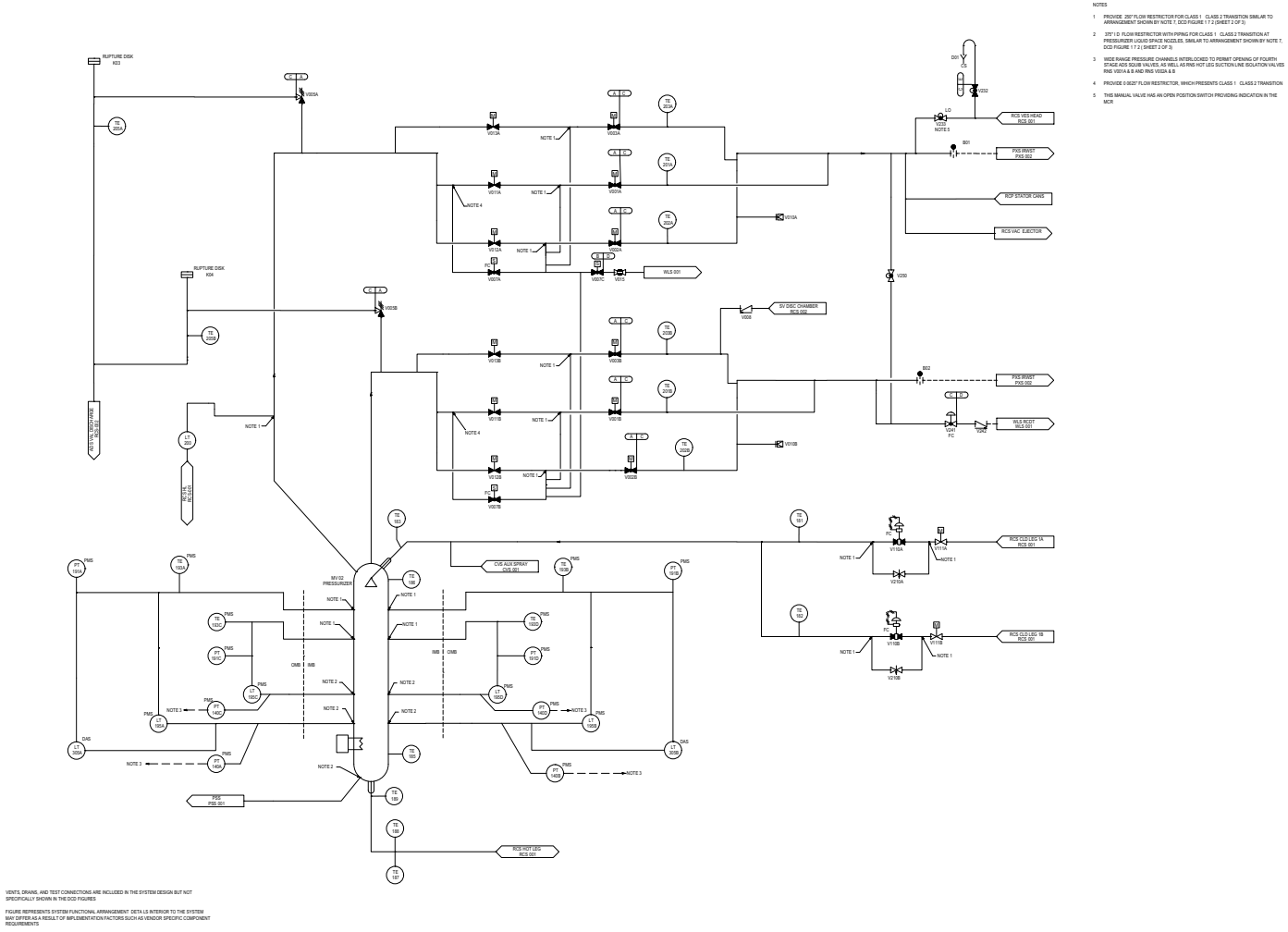

Inside Reactor Containment

Figure 5.1-5 (Sheet 2 of 3)
Simplified Reactor Coolant System Piping and Instrumentation Diagram (REF) RCS 002

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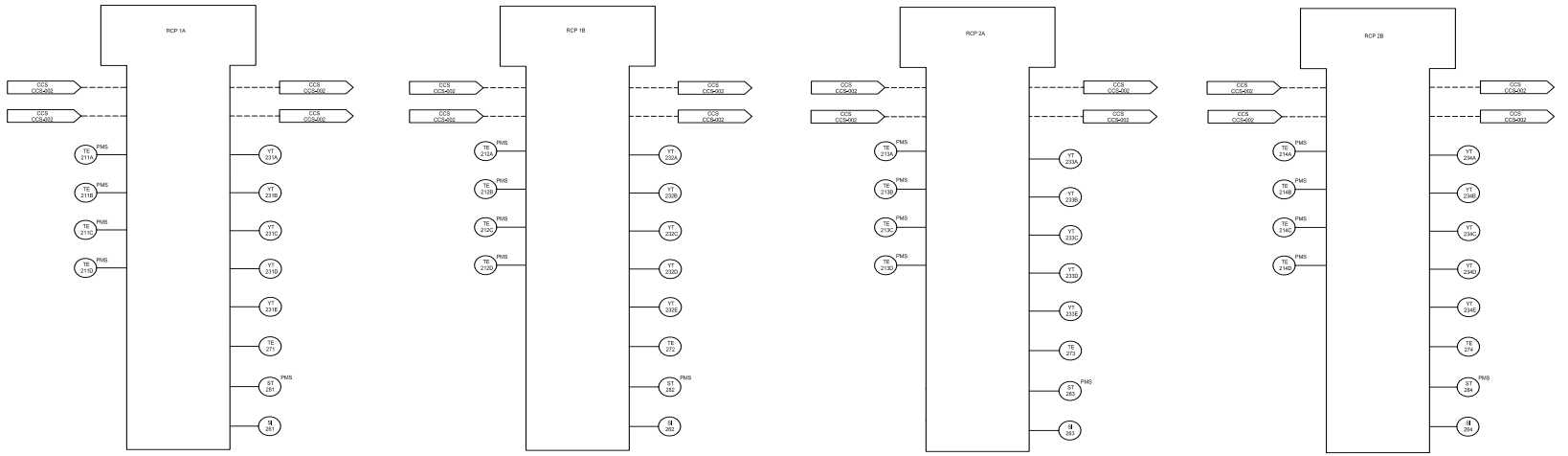

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Inside Reactor Containment

Figure 5.1-5 (Sheet 3 of 3)
Simplified Reactor Coolant System Piping and Instrumentation Diagram (REF) RCS 003

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5.2 Integrity of Reactor Coolant Pressure Boundary

This section discusses the measures to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) during plant operation. Section 50.2 of 10 CFR 50 defines the reactor coolant pressure boundary as vessels, piping, pumps, and valves that are part of the reactor coolant system (RCS), or that are connected to the reactor coolant system up to and including the following:

- The outermost containment isolation valve in system piping that penetrates the containment
- The second of two valves closed during normal operation in system piping that does not penetrate containment
- The reactor coolant system overpressure protection valves

The design transients used in the design and fatigue analysis of ASME Code Class 1 and Class CS components, supports, and reactor internals are provided in Subsection 3.9.1. The loading conditions, loading combinations, evaluation methods, and stress limits for design and service conditions for components, core support structures, and component supports are discussed in Subsection 3.9.3.

The term reactor coolant system, as used in this section, is defined in Section 5.1. The AP1000 reactor coolant pressure boundary is consistent with that of 10 CFR 50.2.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

Reactor coolant pressure boundary components are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III. A portion of the chemical and volume control system inside containment that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(z). Systems other than the reactor coolant system connecting to the chemical and volume control system have required isolation and are not classified as reactor coolant pressure boundary. The alternate classification is discussed in Subsection 5.2.1.3. The quality group classification for the reactor coolant pressure boundary components is identified in Subsection 3.2.2. The quality group classification is used to determine the appropriate sections of the ASME Code or other standards to be applied to the components.

The edition and addenda of the ASME Code applied in the design and manufacture of each component are the edition and addenda established by the requirements of the Design Certification. The use of editions and addenda issued subsequent to the Design Certification is permitted or required based on the provisions in the Design Certification. If a later Code edition/addenda than the Design Certification Code edition/addenda is used by the material and/or component supplier, then a code reconciliation to determine acceptability is performed as required by the ASME Code, Section III, NCA-1140. The later Code edition/addenda must be authorized in 10 CFR 50.55a or in a specific authorization as provided in 50.55a(z). Code Cases to be used in design and construction are identified in this document.

Inservice inspection of the reactor coolant pressure boundary is conducted in accordance with the applicable edition and addenda of the ASME Boiler and Pressure Vessel Code Section XI, as described in Subsection 5.2.4. Inservice testing of the reactor coolant pressure boundary components is in accordance with the edition and addenda of the ASME OM Code as discussed in Subsection 3.9.6 for pumps and valves, and as discussed in Subsection 3.9.3.4.4 for dynamic restraints. The baseline used for the evaluations done to support this safety analysis report and the

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Design Certification is the 1998 Edition, 2000 Addenda, [*with an additional restriction for piping design.*

The restriction on piping design is that the treatment of dynamic loads, including seismic loads, in pipe stress analysis will satisfy the requirements of the ASME Code, Section III, Subarticles NB-3210, NB-3220, NB-3620, NB-3630, NB-3650, NC-3620, NC-3650, ND-3620, and ND-3650 1989 Edition, 1989 Addenda. The requirements shown below for fillet welds are also applicable.

The criteria below are used in place of those in paragraph NB-3683.4(c)(1) and Footnote 11 to Figures NC/ND-3673.2(b)-1 of the 1989 Addenda to the 1989 Edition of ASME Code, Section III. This criteria is based on the criteria included in the 1989 Edition of the ASME Code, Section III.

For girth fillet welds between the piping and socket welded fittings, valves and flanges, and slip on flanges in ASME III Class 1, 2, and 3 piping, the primary stress indices and stress intensification factors are as follows:

Primary Stress Indices

B_1 = *0.75*

B_2 = *1.5*

Stress Intensification Factor

i = *2.1*(t_n/C_x), but not less than 1.3*

C_x = *fillet weld leg length based on ASME III 1989 Edition, Figures NC/ND-4427-1, sketches (c-1), (c-2), and (c-3). For unequal leg length, use smaller leg length for C_x.]**

Seismic Integrity of the CVS System Inside Containment

To provide for the seismic integrity and pressure boundary [*integrity of the nonsafety-related (ASME B31.1, Piping Class D) CVS piping located inside containment, a seismic analysis will be performed and a CVS Seismic Analysis Report prepared with a faulted stress limit equal to the smaller of 4.5 S_h and 3.0 S_y and based on the following additional criteria:*

Additional loading combinations and stress limits for nonsafety-related chemical and volume control system piping systems and components inside containment]*

Condition	***Loading Combination(3)***	***[Equations (ND3650)***	***Stress Limit***
Level D	*PMAX(1) + DW + SSE + SSES*	*9*	*Smaller of 4.5 S_h or 3.0 S_y*
	SSES	*F_{AM}/A_M(4)*	*1.0 S_h*
	TNU + SSES	*i (M1 + M2)/Z(2)*	*3.0 S_h*

Notes:

1. *For earthquake loading, PMAX is equal to normal operating pressure at 100% power.*
2. *Where: M1 is range of moments for TNU, M2 is one half the range of SSES moments, M1 + M2 is larger of M1 plus one half the range of SSES, or full range of SSES.*
3. *See Table 3.9-3 for description of loads.*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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4. F_{AM} is amplitude of axial force for SSES; A_M is nominal pipe metal area.]*

Component supports, equipment, and structural steel frame are evaluated to demonstrate that they do not fail under seismic loads. Design methods and stress criteria are the same as for corresponding Seismic Category I components. The functionality of the chemical and volume control system does not have to be maintained to insure structural integrity of the components.

[*Dimensional fabrication, assembly, erection, inspection, examination, and testing requirements as defined in Chapters IV, V, and VI of the ASME B31.1 Code are applicable and used for the ASME B31.1 (Piping Class D) CVS piping systems, valves, and equipment inside containment.*]*

5.2.1.2 Applicable Code Cases

[*ASME Code Cases used in the AP1000 are listed in Table 5.2-3.*]* In addition, other ASME Code Cases found in Regulatory Guide 1.84, as discussed in Section 1.9 and Appendix 1A, in effect at the time of the Design Certification may be used for pressure boundary components. Use of Code Cases approved in revisions of the Regulatory Guides issued subsequent to the Design Certification may be used as discussed in Subsection 5.2.6.1 by using the process outlined above for updating the ASME Code edition and addenda. Use of any Code Case not approved in Regulatory Guide 1.84 on Class 1 components is authorized as provided in 50.55a(z) and the requirements of the Design Certification.

The use of any Code Case conditionally approved in Regulatory Guide 1.84 used on Class 1 components meets the conditions established in the Regulatory Guide.

ASME Code Cases required for Section XI inspections will be identified in Plant Owner provided Inspection Plans as referenced in Subsection 5.2.6.2. See Subsection 5.2.4, “Inservice Inspection and Testing of Class 1 Components,” and Section 6.6, “Inservice Inspection of Class 2, 3, and MC Components,” for discussion of inservice examinations and procedures.

5.2.1.3 Alternate Classification

The Code of Federal Regulations, Section 10 CFR 50.55a requires the reactor coolant pressure boundary be class A (ASME Boiler and Pressure Vessel Code Section III, Class 1). Components which are connected to the reactor coolant pressure boundary that can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open) with automatic actuation to close can be classified as class C (ASME Section III, class 3) according to 50.55a.

A portion of the chemical and volume control system inside containment is not classified as safety-related. The classification of the AP1000 reactor coolant pressure boundary deviates from the requirement that the reactor coolant pressure boundary be classified as safety related and be constructed using the ASME Code, Section III as provided in 10 CFR 50.55a. The safety-related classification of the AP1000 reactor coolant pressure boundary ends at the third isolation valve between the reactor coolant system and the chemical and volume control system. The nonsafety-related portion of the chemical and volume control system inside containment provides purification of the reactor coolant and includes heat exchangers, demineralizers, filters and connecting piping. For a description of the chemical and volume control system, refer to Subsection 9.3.6. The portion of the chemical and volume control system between the inside and outside containment isolation valves is classified as Class B and is constructed using the ASME Code, Section III.

The nonsafety-related portion of the chemical and volume control system is designed using ASME B31.1 and ASME Code, Section VIII for the construction of the piping, valves, and components. The nonsafety-related portion of the CVS inside containment is analyzed seismically.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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The methods and criteria used for the seismic analysis are similar to those used of seismic Category II pipe and are defined in the Subsection 5.2.1.1. The chemical and volume control system components are located inside the containment which is a seismic Category I structure.

The alternate classification of the nonsafety-related purification subsystems satisfies the purpose of 10 CFR 50.55a that structures, systems, and components of nuclear power plants which are important to safety be designed, fabricated, erected, and tested to quality standards that reflect the importance of the safety functions to be performed.

The AP1000 chemical and volume control system is not required to perform safety-related functions such as emergency boration or reactor coolant makeup. Safety-related core makeup tanks are capable of providing sufficient reactor coolant makeup for shutdown and cooldown without makeup supplied by the chemical and volume control system. Safe shutdown of the reactor does not require use of the chemical and volume control system makeup. AP1000 safe shutdown is discussed in Section 7.4.

The isolation valves between the reactor coolant system and the chemical and volume control system are active safety-related valves that are designed, qualified, inspected and tested for the isolation requirements. These isolation valves also function to isolate the reactor coolant system in the event of a break in the nonsafety portion of the chemical and volume control system purification loop. The isolation valves between the reactor coolant system and chemical and volume control system are designed and qualified for design conditions that include closing against blowdown flow with full system differential pressure. These valves are qualified for adverse seismic and environmental conditions. The valves are subject to inservice testing including operability testing.

The potential for release of activity from a break or leak in the chemical and volume control system is minimized by the location of the purification subsystem inside containment and the design and test of the isolation valves. Chemical and volume control system leakage inside containment is detectable by the reactor control leak detection function as potential reactor coolant pressure boundary leakage. This leakage must be identified before the reactor coolant leak limit is reached. The nonsafety-related classification of the system does not impact the need to identify the source of a leak inside containment.

5.2.2 Overpressure Protection

Reactor coolant system and steam system overpressure protection during power operation are provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the reactor protection system. Combinations of these systems provide compliance with the overpressure protection requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized water reactor systems.

Low temperature overpressure protection is provided by relief valves in the suction line of the normal residual heat removal (RNS) system. The sizing and use of the relief valves for low temperature overpressure protection is consistent with the guidelines of Branch Technical Position RSB 5-2.

5.2.2.1 Design Bases

Overpressure protection during power operation is provided for the reactor coolant system by the pressurizer safety valves. This protection is afforded for the following events to envelop those credible events that could lead to overpressure of the reactor coolant system if adequate overpressure protection were not provided:

- Loss of electrical load and/or turbine trip

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- Uncontrolled rod withdrawal at power
- Loss of reactor coolant flow
- Loss of normal feedwater
- Loss of offsite power to the station auxiliaries

The sizing of the pressurizer safety valves is based on the analysis of a complete loss of steam flow to the turbine, with the reactor operating at 102 percent of rated power. In this analysis, feedwater flow is also assumed to be lost. No credit is taken for operation of the pressurizer level control system, pressurizer spray system, rod control system, steam dump system, or steam line power-operated relief valves. The reactor is maintained at full power (no credit for direct reactor trip on turbine trip and for reactivity feedback effects), and steam relief through the steam generator safety valves is considered. The total pressurizer safety valve capacity is required to be at least as large as the maximum surge rate into the pressurizer during this transient.

This sizing procedure results in a safety valve capacity well in excess of the capacity required to prevent exceeding 110 percent of system design pressure for the events previously listed. The discharge of the safety valve is routed through a rupture disk to containment atmosphere. The rupture disk is to contain leakage past the valve. The rupture disk pressure rating is substantially less than the set pressure of the safety valve. See Subsection 5.4.11 for additional information on the safety valve discharge system. Subsection 5.4.5 describes the connection of the safety valves to the pressurizer.

Administrative controls and plant procedures aid in controlling reactor coolant system pressure during low-temperature operation. Normal plant operating procedures maximize the use of a steam or gas bubble in the pressurizer during periods of low pressure, low-temperature operation. For those low-temperature modes of operation when operation with a water solid pressurizer is possible, relief valves in the residual heat removal system provide low-temperature overpressure protection for the reactor coolant system. The valves are sized to prevent overpressure during the following credible events with a water-solid pressurizer:

- Makeup/letdown flow mismatch
- Inadvertent actuation of the pressurizer heaters
- Loss of residual heat removal with reactor coolant system heatup due to decay heat and pump heat
- Inadvertent start of one reactor coolant pump
- Inadvertent hydrogen addition

Of those events the makeup/letdown flow mismatch is the limiting mass input condition. Inadvertent start of an inactive reactor coolant pump is the limiting heat input condition to size the relief valves. The flow rate postulated for mass input condition is based on the flow from one or both makeup pumps at the set pressure of the relief valves. The heat input condition is based on a 50-degree temperature difference between the reactor coolant system and the steam generator secondary side.

The set pressures for the normal residual heat removal system relief valves are established based on the lower value of the normal residual heat removal system design pressure and the low-temperature pressure limit for the reactor vessel based on ASME Code, Section III, Appendix G, analyses. The

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pressure-temperature limits for the reactor vessel, based on expected material properties and the vessel design, are discussed in Subsection 5.3.3.

The capacity of the residual heat removal relief valves can maintain the pressure in the reactor coolant system and the residual heat removal system to a pressure less than the lesser of 110 percent of the design pressure of the normal residual heat removal system or the pressure limit from the Appendix G analyses for the limiting event.

Overpressure protection for the steam system is provided by steam generator safety valves. The capacity of the steam system safety valves limits steam system pressure to less than 110 percent of the steam generator shell side design pressure. See Section 10.3 for details.

Section 10.3 discusses the steam generator relief valves and connecting piping.

5.2.2.2 Design Evaluation

The relief capacities of the pressurizer safety valves, steam generator safety valves, and the normal residual heat removal system relief valves are determined from the postulated overpressure transient conditions in conjunction with the action of the reactor protection system. An overpressure protection report is prepared according to Article NB-7300 of Section III of the ASME Code. WCAP-7907 (Reference 1) describes the analytical model used in the analysis of the overpressure protection system and the basis for its validity.

Chapter 15 includes a design description of certain initiating events and describes assumptions made, method of analysis, conclusions, and the predicted response of the AP1000 to those events. The performance characteristics of the pressurizer safety valves are included in the analysis of the response. The incidents evaluated include postulated accidents not included in the compilation of credible events used for valve sizing purposes.

Subsection 5.4.9 discusses the capacities of the pressurizer safety valves and residual heat removal system relief valves used for low temperature overpressure protection. The setpoints and reactor trip signals which occur during operational overpressure transients are discussed in Subsection 5.4.5. With the current AP1000 pressure-temperature limits (Subsection 5.3.3), the set pressures for the relief valves in the normal residual heat removal system are based on a sizing analysis performed to prevent the reactor coolant system pressure from exceeding the applicable low temperature pressure limit for the reactor vessel based on ASME Code, Section III, Appendix G. The limiting mass and energy input transients are assumed for the sizing analysis.

5.2.2.3 Piping and Instrumentation Diagrams

The connection of the pressurizer safety valves to the pressurizer is incorporated into the pressurizer safety and relief valve module and is discussed in Subsection 5.4.9. The pressurizer safety and relief valve module configuration appears in the piping and instrumentation drawing for the reactor coolant system (Figure 5.1-5). The normal residual heat removal system (Subsection 5.4.7) incorporates the relief valves for low-temperature overpressure protection. The valves which isolate the normal residual heat removal system from the reactor coolant system do not have an autoclosure interlock. Figure 5.4-6 shows a simplified sketch of the normal residual heat removal system. Figure 5.4-7 shows the piping and instrumentation drawing for the residual heat removal system.

Section 10.3 discusses the safety valves for the main steam system. Figure 10.3.2-1 shows the piping and instrumentation drawing for the main steam system.

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5.2.2.4 Equipment and Component Description

Subsection 5.4.9 discusses the design and design parameters for the safety valves providing operating and low-temperature overpressure protection. The pressurizer safety valves are ASME Boiler and Pressure Vessel Code Class 1 components. These valves are tested and analyzed using the design transients, loading conditions, seismic considerations, and stress limits for Class 1 components as described in Subsections 3.9.1, 3.9.2, and 3.9.3.

The relief valves included in the normal residual heat removal system provide containment boundary function since they are connected to the piping between the containment isolation valves for the system. Containment isolation requirements are discussed in Subsection 6.2.3. Based on the containment boundary function, the relief valves are ASME Code Class 2 components and are analyzed to the appropriate requirements.

In addition to the testing and analysis required for ASME Code requirements, the pressurizer safety valves are of a type which has been verified to operate during normal operation, anticipated transients, and postulated accident conditions. The verification program (Reference 2) was established by the Electric Power Research Institute to address the requirements of 10 CFR 50.34 (f)(2)(x). These requirements do not apply to relief valves of the size and type represented by the relief valves on the normal residual heat removal system.

Section 10.3 discusses the equipment and components that provide the main steam system overpressure protection.

5.2.2.5 Mounting of Pressure Relief Devices

Subsection 5.4.9 describes the design and installation of the pressure relief devices for the reactor coolant system. Section 3.9 describes the design basis for the assumed loads for the primary- and secondary-side pressure relief devices. Subsection 10.3.2, discusses the main steam safety valves and the power-operated atmospheric steam relief valves.

5.2.2.6 Applicable Codes and Classification

The requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 (Overpressure Protection Report) and NC-7300 (Overpressure Protection Analysis), are met.

Piping, valves, and associated equipment used for overpressure protection are classified according to the classification system discussed in Subsection 3.2.2. These safety-class designations are delineated in Table 3.2-3.

5.2.2.7 Material Specifications

See Subsection 5.2.3 for the material specifications for the pressurizer safety valves. The piping in the pressurizer safety and relief valve module up to the safety valve is considered reactor coolant system. See Subsection 5.2.3 for material specifications. The discharge piping is austenitic stainless steel. Subsection 5.4.7 specifies the materials used in the normal residual heat removal system.

5.2.2.8 Process Instrumentation

Each pressurizer safety valve discharge line incorporates a main control room temperature indicator and alarm to notify the operator of steam discharge due to either leakage or actual valve operation.

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5.2.2.9 System Reliability

ASME Code safety valves and relief valves have demonstrated a high degree of reliability over many years of service. The in-service inspection and testing required of safety valves and relief valves (Subsections 3.9.6 and 5.2.4 and Section 6.6) provides assurance of continued reliability and conformance to setpoints. The assessment of reliability, availability, and maintainability which is done to evaluate the estimated availability for the AP1000 includes estimates for the contribution of safety valves and relief valves to unavailability. These estimates were based on experience for operating units.

5.2.2.10 Testing and Inspection

Subsections 3.9.6 and 5.4.8 and Section 6.6 discuss the preservice and in-service testing and inspection required for the safety valves and relief valves. The testing and inspection requirements are in conformance with industry standards, including Section XI of the ASME Code.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Materials Specifications

Table 5.2-1 lists material specifications used for the principal pressure-retaining applications in Class 1 primary components and reactor coolant system piping. Material specifications with grades, classes or types are included for the reactor vessel components, steam generator components, reactor coolant pump, pressurizer, core makeup tank, and the passive residual heat removal heat exchanger. Table 5.2-1 lists the application of nickel-chromium-iron alloys in the reactor coolant pressure boundary. The use of nickel-chromium-iron alloy in the reactor coolant pressure boundary is limited to Alloy 690, or its associated weld metals Alloys 52, 52M, and 152, and similar alloys developed for improved weldability as allowed by ASME Boiler and Pressure Vessel Code rules. Steam generator tubes use Alloy 690 in the thermally treated form. Nickel-chromium-iron alloys are used where corrosion resistance of the alloy is an important consideration and where the use of nickel-chromium-iron alloy is the choice because of the coefficient of thermal expansion. Subsection 5.4.3 defines reactor coolant piping. See Subsection 4.5.2 for material specifications used for the core support structures and reactor internals. See appropriate sections for internals of other components. Engineered safeguards features materials are included in Subsection 6.1.1. The nonsafety-related portion of the chemical and volume control system inside containment in contact with reactor coolant is constructed of or clad with corrosion resistant material such as Type 304 or Type 316 stainless steel or material with equivalent corrosion resistance. The materials are compatible with the reactor coolant. The nonsafety-related portion of the chemical and volume control system is not required to conform to the process requirements outlined below.

Table 5.2-1 material specifications are the materials used in the AP1000 reactor coolant pressure boundary. The materials used in the reactor coolant pressure boundary conform to the applicable ASME Code rules. Cast austenitic stainless steel does not exceed a ferrite content of 20 FN. Calculation of ferrite content is based on Hull's equivalent factors.

The welding materials used for joining the ferritic base materials of the reactor coolant pressure boundary conform to or are equivalent to ASME Material Specifications SFA 5.1, 5.5, 5.17, 5.18, 5.20, 5.23, 5.28, 5.29, and 5.30. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining the austenitic stainless steel base materials of the reactor coolant pressure boundary conform to ASME Material Specifications SFA 5.4, 5.9, 5.22, and 5.30. They are qualified to the requirements of the ASME Code, Section III.

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The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14, or are similar welding alloys to those in SFA-5.11 or SFA-5.14 developed for improved weldability as allowed by the ASME Boiler and Pressure Vessel Code rules. They are qualified to the requirements of the ASME Code, Section III.

The fabrication and installation specifications for partial penetration welds with Alloy 52/52M/152, within the ASME Class 1 reactor coolant pressure boundary, require successive dye penetrant examinations after the first pass and after every 1/4-inch of weld metal. The specifications for J-groove welds, which join ASME Class 1 reactor coolant pressure boundary penetrations require ultrasonic examination of the interface where the weld joins the penetration tube. The specifications for butt welds used for nozzle safe-end welds require these welds to be radiographically inspected. These weld specifications are applicable to the ASME Class 1 reactor coolant pressure boundary portions of the reactor vessel (Section 5.3), the reactor coolant pumps (Subsection 5.4.1), the steam generators (Subsection 5.4.2), the reactor coolant system piping (Subsection 5.4.3), the pressurizer (Subsection 5.4.5), the core makeup tanks (Subsection 5.4.13), and the passive residual heat removal heat exchanger (Subsection 5.4.14).

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Chemistry of Reactor Coolant

The reactor coolant system chemistry specifications conform to the recommendation of Regulatory Guide 1.44, with exceptions as stated in Subsection 1.9.1, and are shown in Table 5.2-2.

The reactor coolant system water chemistry is selected to minimize corrosion. Routinely scheduled analyses of the coolant chemical composition are performed to verify that the reactor coolant chemistry meets the specifications. Other additions, such as those to reduce activity transport and deposition, may be added to the system.

The chemical and volume control system (CVS) provides a means for adding chemicals to the reactor coolant system. The chemicals perform the following functions:

- Control the pH of the coolant during prestartup testing and subsequent operation
- Scavenge oxygen from the coolant during heatup
- Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during power operations following startup

Table 5.2-2 shows the normal limits for chemical additives and reactor coolant impurities for power operation.

The pH control chemical is lithium hydroxide monohydrate, enriched in the lithium-7 isotope to 99.9 percent. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/nickel-chromium-iron systems. In addition, lithium-7 is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The lithium-7 hydroxide is introduced into the reactor coolant system via the charging flow. The concentration of lithium-7 hydroxide in the reactor coolant system is maintained in the range specified for pH control.

During reactor startup from the cold condition, hydrazine is used as an oxygen-scavenging agent. The hydrazine solution is introduced into the reactor coolant system in the same manner as described for the pH control agent.

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The reactor coolant is treated with dissolved hydrogen to control the net decomposition of water by radiolysis in the core region. The hydrogen reacts with oxygen introduced into the reactor coolant system by the radiolysis effect of radiation on molecules. Hydrogen makeup is supplied to the reactor coolant system by direct injection of high pressure gaseous hydrogen, which can be adjusted to provide the correct equilibrium hydrogen concentration. Subsection 1.9.1 indicates the degree of conformance with Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," with the exceptions as stated in Subsection 1.9.1.

Boron, in the chemical form of boric acid, is added to the reactor coolant system for long-term reactivity control of the core.

Suspended solid (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the chemical and volume control system.

The water chemistry program is based on industry guidelines as described in EPRI “Pressurized Water Reactor Primary Water Chemistry” (Reference 201). The program includes periodic monitoring and control of chemical additives and reactor coolant impurities listed in Table 5.2-2. Detailed procedures implement the program requirements for sampling and analysis frequencies, and corrective actions for control of reactor water chemistry.

The frequency of sampling water chemistry varies (e.g. continuous, daily, weekly, or as needed) based on plant operating conditions and the EPRI water chemistry guidelines. Whenever corrective actions are taken to address an abnormal chemistry condition, increased sampling is utilized to verify the effectiveness of these actions. When measured water chemistry parameters are outside the specified range, corrective actions are taken to bring the parameter back within the acceptable range and within the time period specified in the EPRI water chemistry guidelines. Following corrective actions, additional samples are taken and analyzed to verify that the corrective actions were effective in returning the concentrations of contaminants to within the specified range.

Chemistry procedures will provide guidance for the sampling and monitoring of primary coolant properties.

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

Ferritic low-alloy and carbon steels used in principal pressure-retaining applications have corrosion-resistant cladding on surfaces exposed to the reactor coolant. The corrosion resistance of the cladding material is at least equivalent to the corrosion resistance of Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy, martensitic stainless steel, and precipitation-hardened stainless steel. These clad materials may be subjected to the ASME Code-required postweld heat treatment for ferritic base materials.

Ferritic low-alloy and carbon steel nozzles have safe ends of stainless steel-wrought materials welded to nickel-chromium-iron alloy-weld metal F-number 43 buttering. The safe end is welded to the F 43 buttering after completion of postweld heat treatment of the buttering when the nozzle is larger than a 4-inch nominal inside diameter and/or the wall thickness is greater than 0.531 inch.

Austenitic stainless steel and nickel-chromium-iron alloy base materials with primary pressure-retaining applications are used in the solution-annealed or thermally treated conditions. These heat treatments are as required by the material specifications.

During later fabrications, these materials are not heated above 800°F other than locally by welding operations. The solution-annealed surge line material is subsequently formed by hot-bending followed by a resolution-annealing heat treatment.

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Components using stainless steel sensitized in the manner expected during component fabrication and installation operate satisfactorily under normal plant chemistry conditions in pressurized water reactor (PWR) systems because chlorides, fluorides, and oxygen are controlled to very low levels. Subsection 1.9.1 indicates the degree of conformance with Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," with the exceptions as stated in Subsection 1.9.1.

The use of cobalt base alloy is minimized. Restrictions on the use of cobalt in the base metal for material in contact with reactor coolant are discussed in Subsection 12.3.1.1.1.

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

In general, materials that are used in principal pressure-retaining applications and are subject to elevated temperature during system operation are in contact with thermal insulation that covers their outer surfaces.

The thermal insulation used on the reactor coolant pressure boundary is reflective stainless steel-type.

The compounded materials in the form of blocks, boards, cloths, tapes, adhesives, cements, etc., are silicated to provide protection of austenitic stainless steels against stress corrosion that may result from accidental wetting of the insulation by spillage, minor leakage, or other contamination from the environmental atmosphere. Subsection 1.9.1 indicates the degree of conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is considered possible based on service experience (such as valve packing, pump seals, etc.), only materials compatible with the coolant are used. Table 5.2-1 shows examples. Ferritic materials exposed to coolant leakage can be readily observed as part of the inservice visual and/or nondestructive inspection program to confirm the integrity of the component for subsequent service.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness properties of the reactor coolant pressure boundary components meet the requirements of the ASME Code, Section III, Subarticle NB-2300. Those portions of the reactor coolant pressure boundary that meet the requirements of ASME Code, Section III, Class 2 per the criteria of 10 CFR 50.55a, meet the fracture toughness requirements of the ASME Code, Section III, Subarticle NC-2300. The fracture toughness properties of the reactor coolant pressure boundary components also meet the requirements of Appendix G of 10 CFR 50.

The fracture toughness properties of the reactor vessel materials are discussed in Section 5.3.

Limiting steam generator and pressurizer reference temperatures for a nil ductility transition (RT_{NDT}) temperatures are guaranteed at 10°F for the base materials and the weldments.

These materials meet the 50-foot-pound absorbed energy and 35-mils lateral expansion requirements of the ASME Code, Section III, at 70°F. The actual results of these tests are provided in the ASME material data reports which are supplied for each component and submitted to the owner at the time of shipment of the component.

Temperature instruments and Charpy impact test machines are calibrated to meet the requirements of the ASME Code, Section III, Paragraph NB-2360.

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Westinghouse has conducted a test program to determine the fracture toughness of low-alloy ferritic materials with specified minimum yield strengths greater than 50,000 psi to demonstrate compliance with Appendix G of the ASME Code, Section III. In this program, fracture toughness properties were determined and shown to be adequate for base metal plates and forgings, weld metal, and heat-affected zone metal for higher-strength ferritic materials used for components of the reactor coolant pressure boundary. WCAP-9292 (Reference 3) documents the program results.

5.2.3.3.2 Control of Welding

Welding is conducted using procedures qualified according to the rules of Sections III and IX of the ASME Code. Control of welding variables (as well as examination and testing) during procedure qualification and production welding is performed according to ASME Code requirements.

The practices for storing and handling welding electrodes and fluxes comply with ASME Code, Section III, Paragraphs NB-2400 and NB-4400.

Subsection 1.9.1 indicates the degree of conformance of the ferritic materials components of the reactor coolant pressure boundary with Regulatory Guides 1.31, "Control of Ferrite Content in Stainless Steel Welds"; 1.34, "Control of Electroslag Weld Properties"; 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"; 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"; and 1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

Subsections 5.2.3.4.1 through 5.2.3.4.5 address Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and present the methods and controls to avoid sensitization and to prevent intergranular attack (IGA) of austenitic stainless steel components. Also, Subsection 1.9.1 indicates the degree of conformance with Regulatory Guide 1.44.

5.2.3.4.1 Cleaning and Contamination Protection Procedures

Austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems are handled, protected, stored, and cleaned according to recognized, accepted methods designed to minimize contamination that could lead to stress corrosion cracking. The procedures covering these controls are stipulated in process specifications. Tools used in abrasive work operations on austenitic stainless steel, such as grinding or wire brushing, do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress-corrosion cracking.

These process specifications supplement the equipment specifications and purchase order requirements of every individual austenitic stainless steel component or system procured for the AP1000, regardless of the ASME Code classification.

The process specifications define these requirements and follow the guidance of ASME NQA-1.

Subsection 1.9.1 indicates the degree of conformance of the austenitic stainless steel components of the reactor coolant pressure boundary with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

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5.2.3.4.2 Solution Heat Treatment Requirements

The austenitic stainless steels listed in Table 5.2-1 are used in the final heat-treated condition required by the respective ASME Code, Section II, materials specification for the particular type or grade of alloy.

5.2.3.4.3 Material Testing Program

Austenitic stainless steel materials of product forms with simple shapes need not be corrosion-tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as plates, sheets, bars, pipe, and tubes, as well as forgings, fittings, and other shaped products that do not have inaccessible cavities or chambers that would preclude rapid cooling when water-quenched. This characterization of cavities or chambers as inaccessible is in relation to the entry of water during quenching and is not a determination of the component accessibility for inservice inspection.

When testing is required, the tests are performed according to a process specification following the guidelines of ASTM A 262, Practice A or E.

5.2.3.4.4 Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels

Unstabilized stainless steels can be subject to intergranular attack if the steels are sensitized, if certain species are present, such as chlorides and oxygen, and if they are exposed to a stressed condition. In the reactor coolant system, reliance is placed on the elimination or avoidance of these conditions. This is accomplished by the following:

- Control of primary water chemistry to provide a benign environment
- Use of materials in the final heat-treated condition and the prohibition of subsequent heat treatments from 800°F to 1500°F
- Control of welding processes and procedures to avoid heat-affected zone sensitization
- Confirmation that the welding procedures used for the manufacture of components in the primary pressure boundary and the reactor internals do not result in the sensitization of heat-affected zones

Further information on each of these steps is provided in the following paragraphs.

The water chemistry in the reactor coolant system is controlled to prevent the intrusion of aggressive elements. In particular, the maximum permissible oxygen and chloride concentrations are 0.005 ppm and 0.15 ppm, respectively. Table 5.2-2 lists the recommended reactor coolant water chemistry specifications.

The precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage are stipulated in the appropriate process specifications. The use of hydrogen overpressure precludes the presence of oxygen during operation.

The effectiveness of these controls has been demonstrated by both laboratory tests and operating experience. The long-term exposure of severely sensitized stainless steels to reactor coolant environments in early Westinghouse pressurized water reactors has not resulted in any sign of intergranular attack. WCAP-7477 (Reference 4) describes the laboratory experimental findings and reactor operating experience. The additional years of operations since Reference 4 was issued have provided further confirmation of the earlier conclusions that severely sensitized stainless steels do

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not undergo any intergranular attack in Westinghouse pressurized water reactor coolant environments.

Although there is no evidence that pressurized water reactor coolant water attacks sensitized stainless steels, it is good metallurgical practice to avoid the use of sensitized stainless steels in the reactor coolant system components.

Accordingly, measures are taken to prohibit the use of sensitized stainless steels and to prevent sensitization during component fabrication. The wrought austenitic stainless steel stock used in the reactor coolant pressure boundary is used in one of the following conditions:

- Solution-annealed and water-quenched
- Solution-annealed and cooled through the sensitization temperature range within less than about 5 minutes

Westinghouse has verified that these practices will prevent sensitization by performing corrosion tests on wrought material as it was received.

The heat-affected zones of welded components must, of necessity, be heated into the sensitization temperature range (800°F to 1500°F). However, severe sensitization (that is, continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion) can be avoided by controlling welding parameters and welding processes. The heat input and associated cooling rate through the carbide precipitation range are of primary importance. Westinghouse has demonstrated this by corrosion-testing a number of weldments.

The heat input in austenitic pressure boundary weldments is controlled by the following:

- Limiting the maximum interpass temperature to 350°F
- Exercising approval rights on welding procedures
- Requiring qualification of processes

5.2.3.4.5 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures

If during the course of fabrication, steel is inadvertently exposed to the sensitization temperature range, the material may be tested according to a process specification, following the guidelines of ASTM A 262, to verify that it is not susceptible to intergranular attack. Testing is not required for the following:

- Cast metal or weld metal with a ferrite content of 5 percent or more
- Material with a carbon content of 0.03 percent or less
- Material exposed to special processing, provided the following:
  - Processing is properly controlled to develop a uniform product
  - Adequate documentation exists of service experience and/or test data to demonstrate that the processing will not result in increased susceptibility to intergranular attack

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If such material is not verified to be not susceptible to intergranular attack, the material is resolution-annealed and water-quenched or rejected.

5.2.3.4.6 Control of Welding

The following paragraphs address Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." They present the methods used, and the verification of these methods, for austenitic stainless steel welding.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring, or hot cracking, in the weld.

Also, it has been well documented that delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. The minimum delta ferrite level below which the material will be prone to hot cracking lies between 0 and 3 percent delta ferrite.

The following paragraphs discuss welding processes used to join stainless steel parts in components designed, fabricated, or stamped according to the ASME Code, Section III, Classes 1 and 2, and core support components. Delta ferrite control is appropriate for the preceding welding requirements, except where no filler metal is used or where such control is not applicable, such as the following: electron beam welding; autogenous gas shielded tungsten arc welding; explosive welding; welding using fully austenitic welding materials.

The fabrication and installation specifications require welding procedures and welder qualification according to Section III of the ASME Code. They also include the delta ferrite determinations for the austenitic stainless steel welding materials used for welding qualification testing and for production processing.

Specifically, the undiluted weld deposits of the "starting" welding materials must contain at least 5 percent delta ferrite. (The equivalent ferrite number may be substituted for percent delta ferrite.) This is determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III of the ASME Code or magnetic measurement by calibrated instruments.

When new welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, they are performed according to the requirements of Sections III and IX of the ASME Code.

The results of the destructive and nondestructive tests are recorded in the procedure qualification record, in addition to the information required by Section III of the ASME Code.

The welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of Section III of the ASME Code. For applications using austenitic stainless steel welding material, the material conforms to ASME weld metal analysis A-8, Type 308, 308L, 309, 309L, 316, or 316L.

Delta ferrite determinations of austenitic stainless steel weld filler materials to be used with gas tungsten arc welding (GTAW) and plasma arc welding (PAW) processes and any other welding material to be used with any GTAW, PAW, or gas metal arc welding (GMAW) process, including consumable insert material, shall be made using a magnetic measuring instrument and weld deposits made in accordance with ASME Code, Section III, NB-2432.1(c) or (d) or, alternatively, the delta ferrite determinations for welding materials may be performed by the use of chemical analysis performed either on the filler metal or on an undiluted weld deposit made in accordance with NB-2432. The allowable delta ferrite range shall be 5 FN to 20 FN for the weld material with low

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molybdenum content, and 5 FN to 16 FN for weld materials with higher molybdenum content such as Types 316/316L, which contain 2.0 to 3.0% molybdenum.

Delta ferrite determinations of austenitic stainless steel weld filler materials to be used with flux welding processes, such as shielded metal arc welding (SMAW), submerged arc welding (SAW) or for electro-slag weld (ESW) deposited cladding and other welding material to be used with other than the GTAW, PAW, or GMAW process shall be made using a magnetic measuring instrument and weld deposits made in accordance with ASME Code, Section III, B-2432.1(c) or (d) or, alternatively, the delta ferrite determinations may be performed by the use of chemical analysis of the undiluted weld deposit of NB-2432 in conjunction with Figure NB-2433.1-1. The allowable delta ferrite range shall be 5 FN to 20 FN for the weld material with low molybdenum content, and 5 FN to 16 FN for weld materials with higher molybdenum content such as types 316/316L, which contain 2.0 to 3.0% molybdenum.

Welding materials are tested using the welding energy inputs employed in production welding.

Combinations of approved heats and lots of welding materials are used for welding processes. The welding quality assurance program includes identification and control of welding material by lots and heats as appropriate. Weld processing is monitored according to approved inspection programs that include review of materials, qualification records, and welding parameters. Welding systems are also subject to the following:

- Quality assurance audit, including calibration of gauges and instruments
- Identification of welding materials
- Welder and procedure qualifications
- Availability and use of approved welding and heat-treating procedures
- Documentary evidence of compliance with materials, welding parameters, and inspection requirements

Fabrication and installation welds are inspected using nondestructive examination methods according to Section III of the ASME Code rules.

To verify the reliability of these controls, Westinghouse has completed a delta ferrite verification program, described in WCAP-8324-A (Reference 5). This program has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative for conformance with the NRC Interim Position on Regulatory Guide 1.31. The regulatory staff's acceptance letter and topical report evaluation were received on December 30, 1974. The program results, which support the hypothesis presented in WCAP-8324-A (Reference 5), are summarized in WCAP-8693 (Reference 6).

Subsection 1.9.1 indicates the degree of conformance of the austenitic stainless steel components of the reactor coolant pressure boundary with Regulatory Guides 1.34, "Control of Electroslag Weld Properties," and 1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.3.4.7 Control of Cold Work in Austenitic Stainless Steels

The use of cold worked austenitic stainless steels is limited to small parts including pins and fasteners where proven alternatives are not available and where cold worked material has been used successfully in similar applications. Cold work control of austenitic stainless steels in pressure boundary applications is provided by limiting the hardness of austenitic stainless steel raw material

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and controlling the hardness during fabrication by process control of bending, cold forming, straightening or other similar operation. Grinding of material in contact with reactor coolant is controlled by procedures. Ground surfaces are finished with successively finer grit sizes to remove the bulk of cold worked material.

5.2.3.5 Threaded Fastener Lubricants

The lubricants to be used on threaded fasteners which maintain pressure boundary integrity in the reactor coolant and related systems and in the steam, feed, and condensate systems; threaded fasteners used inside those systems; and threaded fasteners used in component structural support for those systems are specified in the design specification. Field selection of thread lubricants is not permitted. The thread lubricants are selected based on experience and test data which show them to be effective, but not to cause or accelerate corrosion of the fastener. Where leak sealants are used on threaded fasteners or can be in contact with the fastener in service, their selection is based on satisfactory experience or test data. Selection considers possible adverse interaction between sealants and lubricants. Lubricants containing molybdenum sulphide are prohibited.

5.2.4 Inservice Inspection and Testing of Class 1 Components

Preservice and inservice inspection and testing of ASME Code Class 1 pressure-retaining components (including vessels, piping, pumps, valves, bolting, and supports) within the reactor coolant pressure boundary are performed in accordance with Section XI of the ASME Code including addenda according to 10 CFR 50.55a(g). This includes all ASME Code Section XI mandatory appendices.

The specific edition and addenda of the Code used to determine the requirements for the inspection and testing plan for the initial and subsequent inspection intervals is to be delineated in the inspection program. The Code includes requirements for system pressure tests and functional tests for active components. The requirements for system pressure tests are defined in Section XI, IWA-5000 and IWB-5000. These tests verify the pressure boundary integrity in conjunction with inservice inspection. Section 6.6 discusses Classes 2 and 3 component examinations.

Subsection 3.9.6 discusses the in-service functional testing of valves for operational readiness. Since none of the pumps in the AP1000 are required to perform an active safety function, the operational readiness test program for pumps is controlled administratively.

In conformance with ASME Code and NRC requirements, the preparation of inspection and testing programs is discussed in Subsection 5.2.6. A preservice inspection program (nondestructive examination) for the AP1000 will be developed and submitted to the NRC. The in-service inspection program will be submitted to the NRC as discussed in Subsection 5.2.6. These programs will comply with applicable in-service inspection provisions of 10 CFR 50.55a(b)(2).

The preservice program provides details of areas subject to examination, as well as the method and extent of preservice examinations. The in-service program details the areas subject to examination and the method, extent, and frequency of examinations. Additionally, component supports and examination requirements are included in the inspection programs.

5.2.4.1 System Boundary Subject to Inspection

ASME Code Class 1 components (including vessels, piping, pumps, valves, bolting, and supports) are designated AP1000 equipment Class A (see Subsection 3.2.2). Class 1 pressure-retaining components and their specific boundaries are included in the equipment designation list and the line designation list. Both of these lists are contained in the inspection programs.

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The Class 1 system boundary for both preservice and inservice inspection programs and the system pressure test program includes those items within the Class 1 and Quality Group A (Equipment Class A per Subsection 3.2.2 and Table 3.2-3) boundary. Based on 10 CFR Part 50 and Regulatory Guide 1.26, the Class 1 boundary includes the following:

- Reactor pressure vessel;
- Portions of the Reactor System (RXS);
- Portions of the Chemical and Volume Control System (CVS);
- Portions of the Incore Instrumentation System (IIS);
- Portions of the Passive Core Cooling System (PXS);
- Portions of the Reactor Coolant System (RCS); and
- Portions of the Normal Residual Heat Removal System (RNS).

Those portions of the above systems within the Class 1 boundary are those items that are part of the reactor coolant pressure boundary as defined in Section 5.2.

Exclusions

Portions of the systems within the reactor coolant pressure boundary (RCPB), as defined above, that are excluded from the Class 1 boundary in accordance with 10 CFR Part 50, Section 50.55a, are as follows:

- Those components where, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only; or
- Components that are or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve is capable of automatic actuation and, assuming the other valve is open, its closure time is such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

The description of portions of systems excluded from the RCPB does not address Class 1 components exempt from inservice examinations under ASME Code Section XI rules. The Class 1 components exempt from inservice examinations are defined by ASME Section XI, IWB-1220, except as modified by 10 CFR 50.55a.

The inservice inspection program is augmented for reactor vessel top head inspections by use of the ASME Code Case(s) as specified in 10 CFR 50.55a(g).

Boric acid corrosion control procedures require inspection of the reactor coolant pressure boundary subject to leakage that can cause boric acid corrosion of the reactor coolant pressure boundary materials. The procedures determine the principal locations where leaks can cause degradation of the primary pressure boundary by boric acid corrosion. Potential paths of the leaking coolant are established. The boric acid corrosion control procedures also contain methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located.

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The boric acid corrosion control procedures consist of:

1. Visual inspections of component surfaces that are potentially exposed to borated water leakage.

2. Discovery of leak path and removal of boric acid residue.

3. Assessment of the corrosion.

4. Follow-up inspection for adequacy of corrective actions, as appropriate.

The inservice inspection program, along with the boric acid corrosion control procedures, provides guidance for inspecting the integrity of bolting and threaded fasteners.

The in-service inspection program is augmented to include the performance of a 100 percent volumetric examination of the weld build-up on the reactor vessel head for the instrumentation penetrations (Quickloc) conducted once during each 120-month inspection interval in accordance with the ASME Code, Section XI. The weld build-up acceptance standards are those provided in ASME Code, Section XI, IWB-3514. Personnel performing examinations and the ultrasonic examination systems are qualified in accordance with ASME Code, Section XI, Appendix VIII. Alternatively, an alternative inspection may be developed in conjunction with the voluntary consensus standards bodies (i.e., ASME) and submitted to the NRC for approval.

5.2.4.2 Arrangement and Inspectability

ASME Code Class 1 components are designed so that access is provided in the installed condition for visual, surface, and volumetric examinations specified by the ASME Code Section XI (1998 Edition) and mandatory appendices. Design provisions, in accordance with Section XI, Article IWA-1500, are incorporated in the design processes for Class 1 components.

The AP1000 design activity includes a design for inspectability program. The goal of this program is to provide for the inspectability access and conformance of component design with available inspection equipment and techniques. Factors such as examination requirements, examination techniques, accessibility, component geometry and material selection are used in evaluating component designs. Examination requirements and examination techniques are defined by inservice inspection personnel. Inservice inspection review as part of the design process provides component designs that conform to inspection requirements and establishes recommendations for enhanced inspections.

Considerable experience is utilized in designing, locating, and supporting pressure-retaining components to permit preservice and in-service inspection required by Section XI of the ASME Code. Factors such as examination requirements, examination techniques, accessibility, component geometry, and material selections aid in establishing the designs. The inspection design goals are to eliminate uninspectable components, reduce occupational radiation exposure, reduce inspections times, allow state-of-the-art inspection system, and enhance flaw detection and the reliability of flaw characterization.

As one example of component geometry that reduces inspection requirements, the reactor pressure vessel has no longitudinal welds requiring in-service inspection. No Quality Group A (ASME Code Class 1) components require in-service inspection during reactor operation.

Removable insulation and shielding are provided on those piping systems requiring volumetric and surface examination. Removable hangers and pipe whip restraints are provided as necessary and practical to facilitate inservice inspection. Working platforms are provided in areas requiring

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inspection and servicing of pumps and valves. Permanent or temporary working platforms, scaffolding, and ladders facilitate access to piping and component welds. The components and welds requiring in-service inspection allow for the application of the required in-service inspection methods. Such design features include sufficient clearances for personnel and equipment, maximized examination surface distances, two-sided access (except for valve to pipe welds), favorable materials, weld-joint simplicity, elimination of geometrical interferences, and proper weld surface preparation.

Some of the ASME Class 1 components are included in modules fabricated offsite and shipped to the site. (See Subsection 3.9.1.5.) The modules are designed and engineered to provide access for in-service inspection and maintenance activities. The attention to detail engineered into the modules before construction provides the accessibility for inspection and maintenance. Relief from Section XI requirements should not be required for Class 1 pressure retaining components in the AP1000. Future unanticipated changes in the ASME Code, Section XI requirements could, however, necessitate relief requests. Relief from the inspection requirements of ASME Code, Section XI will be requested when full compliance is not practical according to the requirements of 10 CFR 50.55a(g)(5)(iv). In such cases, specific information will be provided which identifies the applicable Code requirements, justification for the relief request, and the inspection method to be used as an alternative.

Space is provided to handle and store insulation, structural members, shielding, and other materials related to the inspection. Suitable hoists and other handling equipment, lighting, and sources of power for inspection equipment are installed. The integrated head package provides for access to inspect the reactor vessel head and the weld of the control rod drive mechanisms to the reactor vessel head. Closure studs, nuts, and washers are removed to a dry location for direct inspection.

5.2.4.3 Examination Techniques and Procedures

The visual, surface, and volumetric examination techniques and procedures agree with the requirements of Subarticle IWA-2200 and Table IWB-2500-1 of the ASME Code, Section XI. Qualification of the ultrasonic inspection equipment, personnel and procedures is in compliance with Appendix VII of the ASME Code, Section XI. Approved Code Cases listed in Regulatory Guide 1.147 are applied as the need arises during the pre-service inspection. Approved Code Cases determined as necessary to accomplish pre-service inspection activities are used. The liquid penetrant method or the magnetic particle method is used for surface examinations. Radiography, ultrasonic, or eddy current techniques (manual or remote) are used for volumetric examinations.

The reactor vessel is designed so that the reactor pressure vessel (RPV) inspections can be performed primarily from the vessel internal surfaces. These inspections can be done remotely using existing inspection tool designs to minimize occupational radiation exposure and to facilitate the inspections. Access is also available for the application of inspection techniques from the outside of the complete reactor pressure vessel. Reactor pressure vessel welds are examined to meet the requirements of Appendix VIII of ASME Code, Section XI, which has been incorporated into the guidance of Regulatory Guide 1.150, as defined in Subsection 1.9.1.

5.2.4.3.1 Examination Methods

Ultrasonic Examination of the Reactor Vessel

Ultrasonic examination for the RPV is conducted in accordance with the ASME Code, Section XI. The design of the RPV considered the requirements of the ASME Code Section XI with regard to performance of preservice inspection. For the required preservice examinations, the reactor vessel meets the acceptance standards of Section XI, IWB-3510. The RPV shell welds are designed for 100% accessibility for both preservice and inservice inspection. RPV shell welds may be examined from the inside or outside diameter surfaces (or a combination of those techniques) using automated

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ultrasonic examination equipment. The RPV nozzle-to-shell welds are 100% accessible for preservice inspection but might have limited areas that may not be accessible from the outer surface for inservice examination techniques. If accessibility is limited, an inservice inspection program relief request is prepared and submitted for review approval by the NRC.

Inner radius examinations are performed from the outside of the nozzle using several compound angle transducer wedges to obtain complete coverage of the required examination volume. Alternatively, nozzle inner radius examinations may be performed using enhanced visual techniques, as allowed by 10 CFR 50.55a(b)(2)(xxi).

Visual Examination

Visual examination methods VT-1, VT-2 and VT-3 are conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations meet the requirements of IWA-5240.

Where direct visual VT-1 examinations are conducted without the use of mirrors or with other viewing aids, clearance is provided where feasible for the head and shoulders of a man within a working arm's length of the surface to be examined.

Surface Examination

Examination methods are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

Volumetric Ultrasonic Direct Examination

Examination methods are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

Alternative Examination Techniques

Examination methods are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

5.2.4.3.2 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII. Qualification to ASME Section XI, Appendix VIII, is in compliance with the provisions of 10 CFR 50.55a. Alternatively, qualification of personnel and examination systems may use other methods when authorized in accordance with 10 CFR 50.55a(z).

5.2.4.4 Inspection Intervals

Inspection intervals are established as defined in Subarticles IWA-2400 and IWB-2400 of the ASME Code, Section XI. The interval may be extended by as much as one year so that inspections are concurrent with plant outages. It is intended that in-service examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval.

5.2.4.5 Examination Categories and Requirements

The examination categories and requirements are established according to Subarticle IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI. Class 1 piping supports will be examined in

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accordance with ASME Section XI, IWF-2500. Preservice examinations required by design specification and preservice documentation are in accordance with ASME Section III, NB-5280. Components exempt from preservice examination are described in ASME Section III, NB-5283.

The preservice examinations comply with IWB-2200. The preservice examination is performed once in accordance with ASME XI, IWB-2200, on all of the items selected for inservice examination, with the exception of the examinations specifically excluded by ASME Section XI from preservice requirements, such as VT-3 examination of valve body and pump casing internal surfaces (B-M-2 and B-L-2 examination categories, respectively) and the visual VT-2 examinations for category B-P.

5.2.4.6 Evaluation of Examination Results

Evaluation of examination results are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System pressure tests comply with IWA-5000 and IWB-5000 of the ASME Code, Section XI. These system pressure tests are included in the design transients defined in Subsection 3.9.1. This subsection discusses the transients included in the evaluation of fatigue of Class 1 components due to cyclic loads.

5.2.4.8 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the initial examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(z) or 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.

5.2.4.9 Preservice Inspection of Class 1 Components

Preservice examinations required by design specification and preservice documentation are in accordance with ASME Section III, NB-5281. Volumetric and surface examinations are performed as specified in ASME Section III, NB-5282. Components described in ASME Section III, NB-5283 are exempt from preservice examination.

5.2.4.10 Program Implementation

The milestones for preservice and inservice inspection program implementation are identified in Table 13.4-201.

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary (RCPB) leakage detection monitoring provides a means of detecting and to the extent practical, identifying the source and quantifying the reactor coolant leakage. The detection monitors perform the detection and monitoring function in conformance with the requirements of General Design Criteria 2 and 30 and the recommendations of Regulatory Guide 1.45. Leakage detection monitoring is also maintained in support of the use of leak-before-break criteria for high-energy pipe in containment. See Subsection 3.6.3 for the application of leak-before-break criteria.

Leakage detection monitoring is accomplished using instrumentation and other components of several systems. Diverse measurement methods including level, flow, and radioactivity

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measurements are used for leak detection. The equipment classification for each of the systems and components used for leak detection is generally determined by the requirements and functions of the system in which it is located. There is no requirement that leak detection and monitoring components be safety-related. See Figure 5.2-1 for the leak detection approach. The descriptions of the instrumentation and components used for leak detection and monitoring include information on the system.

To satisfy position 1 of Regulatory Guide 1.45, reactor coolant pressure boundary leakage is classified as either identified or unidentified leakage. Identified leakage includes:

- Leakage from closed systems such as reactor vessel seal or valve leaks that are captured and conducted to a collecting tank

- Leakage into auxiliary systems and secondary systems (intersystem leakage) (This leakage is considered to be part of the 10 gpm limit identified leakage in the bases of the technical specification 3.4.7. This additional leakage must be considered in the evaluation of the reactor coolant inventory balance.)

Other leakage is unidentified leakage.

5.2.5.1 Collection and Monitoring of Identified Leakage

Identified leakage other than intersystem leakage is collected in the reactor coolant drain tank. The reactor coolant drain tank is a closed tank located in the reactor cavity in the containment. The tank vent is piped to the gaseous radwaste system to prevent release of radioactive gas to the containment atmosphere. For positions 1 and 7 of Regulatory Guide 1.45, the liquid level in the reactor coolant drain tank and total flow pumped out of the reactor coolant drain tank are used to monitor the identified leakage rate. The identified leakage rate is automatically calculated by the data display and processing system using the liquid level in the reactor coolant drain tank. A leak as small as 0.1 gpm can be detected in one hour. The design leak of 10 gpm will be detected in less than a minute. These parameters are available in the main control room. The reactor coolant drain tank, pumps, and sensors are part of the liquid radwaste system. The following sections outline the various sources of identified leakage other than intersystem leakage.

5.2.5.1.1 Valve Stem Leakoff Collection

Valve stem leakoff connections are not provided in the AP1000.

5.2.5.1.2 Reactor Head Seal

The reactor vessel flange and head flange are sealed by two concentric seals. Seal leakage is detected by two leak-off connections: one between the inner and outer seal, and one outside the outer seal. These lines are combined in a header before being routed to the reactor coolant drain tank. An isolation valve is installed in the common line. During normal plant operation, the leak-off valves are aligned so that leakage across the inner seal drains to the reactor coolant drain tank.

A surface-mounted resistance temperature detector installed on the bottom of the common reactor vessel seal leak pipe provides an indication and high temperature alarm signal in the main control room indicating the possibility of a reactor pressure vessel head seal leak. The temperature detector and drain line downstream of the isolation valve are part of the liquid radwaste system.

The reactor coolant pump closure flange is sealed with a welded canopy seal and does not require leak-off collection provisions.

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Leakage from other flanges is discussed in Subsection 5.2.5.3, Collection and Monitoring of Unidentified Leakage.

5.2.5.1.3 Pressurizer Safety Relief and Automatic Depressurization Valves

Temperature is sensed downstream of each pressurizer safety relief valve and each automatic depressurization valve mounted on the pressurizer by a resistance temperature detector on the discharge piping just downstream of each globe valve. High temperature indications (alarms in the main control room) identify a reduction of coolant inventory as a result of seat leakage through one of the valves. These detectors are part of the reactor coolant system. This leakage is drained to the reactor coolant drain tank during normal plant operation and vented to containment atmosphere or the in-containment refueling water storage tank during accident conditions. This identified leakage is measured by the change in level of the reactor coolant drain tank.

5.2.5.1.4 Other Leakage Sources

In the course of plant operation, various minor leaks of the reactor coolant pressure boundary may be detected by operating personnel. If these leaks can be subsequently observed, quantified, and routed to the containment sump, this leakage will be considered identified leakage.

5.2.5.2 Intersystem Leakage Detection

Substantial intersystem leakage from the reactor coolant pressure boundary to other systems is not expected. However, possible leakage points across passive barriers or valves and their detection methods are considered. In accordance with position 4 of Regulatory Guide 1.45, auxiliary systems connected to the reactor coolant pressure boundary incorporate design and administrative provisions that limit leakage. Leakage is detected by increasing auxiliary system level, temperature, flow, or pressure, by lifting the relief valves or increasing the values of monitored radiation in the auxiliary system.

The normal residual heat removal system and the chemical and volume control system, which are connected to the reactor coolant system, have potential for leakage past closed valves. For additional information on the control of reactor coolant leakage into these systems, see Subsections 5.4.7 and 9.3.6 and the intersystem LOCA discussion in Subsection 1.9.5.1.

5.2.5.2.1 Steam Generator Tubes

An important potential identified leakage path for reactor coolant is through the steam generator tubes into the secondary side of the steam generator. Identified leakage from the steam generator primary side is detected by one, or a combination, of the following:

- High condenser air removal discharge radioactivity, as monitored and alarmed by the turbine island vent discharge radiation monitor
- Steam generator secondary side radioactivity, as monitored and alarmed by the steam generator blowdown radiation monitor
- Secondary side radioactivity, as monitored and alarmed by the main steam line radiation monitors
- Radioactivity, boric acid, or conductivity in condensate as indicated by laboratory analysis

Details on the radiation monitors are provided in Section 11.5, Radiation Monitoring.

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5.2.5.2.2 Component Cooling Water System

Leakage from the reactor coolant system to the component cooling water system is detected by high component cooling water system radiation, by increasing surge tank level, by high flow downstream of selected components, by a high temperature condition at reactor coolant pump bearing water temperature RTDs, or by some combination of the preceding. Refer to Section 11.5, Radiation Monitoring, and Subsection 9.2.2, Component Cooling Water System.

5.2.5.2.3 Passive Residual Heat Removal Heat Exchanger Tubes

A potential identified leakage path for reactor coolant is through the passive residual heat removal heat exchanger into the in-containment refueling water storage tank. Identified leakage from the passive residual heat removal heat exchanger tubes is detected as follows:

- High temperature in the passive residual heat removal heat exchanger, as monitored and alarmed by a temperature detector in the heat exchanger inlet cold trap, alerts the operators to potential leakage. The location of the instrument is selected to provide early indication of leakage considering the potential for thermal stratification. The alarm setpoint is selected to provide early indication of leakage.

- With the increase in temperature in the passive residual heat removal heat exchanger inlet cold trap coinciding with an increase in the reactor coolant system inventory balance leakage rate and no corresponding increase in containment sump level, containment pressure, temperature or humidity, or containment atmosphere radiation, then passive residual heat removal heat exchanger leakage is confirmed. If no increase in reactor coolant system inventory balance is present, then the increase in passive residual heat removal cold trap temperature is indicative of leakage through the outlet isolation valves.

- If residual heat removal heat exchanger tube leakage is present the operator then compares the current reactor coolant system inventory leakage to the reactor coolant inventory leakage before passive residual heat removal heat exchanger tube leakage was present. The difference in the reactor coolant system leakage is the passive residual heat removal heat exchanger tube leakage rate.

5.2.5.3 Collection and Monitoring of Unidentified Leakage

Position 3 of Regulator Guide 1.45 identifies three diverse methods of detecting unidentified leakage. AP1000 use two of these three and adds a third method. To detect unidentified leakage inside containment, the following diverse methods may be utilized to quantify and assist in locating the leakage:

- Containment Sump Level

- Reactor Coolant System Inventory Balance

- Containment Atmosphere Radiation

Other methods that can be employed to supplement the above methods include:

- Containment Atmosphere Pressure, Temperature, and Humidity

- Containment Water Level

- Visual Inspection

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The reactor coolant system is an all-welded system, except for the connections on the pressurizer safety valves, reactor vessel head, explosively actuated fourth stage automatic depressurization system valves, pressurizer and steam generator manways, reactor vessel head vent, the RCP drain, the RCP heat exchanger piping, and the RCP heat exchanger vent, which are flanged. During normal operation, variations in airborne radioactivity, containment pressure, temperature, or specific humidity above the normal level signify a possible increase in unidentified leakage rates and alert the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage. The following sections outline the methods used to collect and monitor unidentified leakage.

These methods also allow for identification of main steam line leakage inside containment. The primary method of identifying steam line leakage is redundant containment sump level monitoring. A diverse backup method is provided by containment water level monitoring. The safety-related class 1E containment water level sensors use a different measuring process than the containment sump level sensors.

5.2.5.3.1 Containment Sump Level Monitor

In conformance with position 2 of Regulatory Guide 1.45, leakage from the reactor coolant pressure boundary and other components not otherwise identified inside the containment will condense and flow by gravity via the floor drains and other drains to the containment sump.

A leak in the primary system would result in reactor coolant flowing into the containment sump. Leakage is indicated by an increase in the sump level. The containment sump level is monitored by three seismic Category I level sensors. Position 6 of Regulatory Guide 1.45 requires two sensors. The third sensor is provided for redundancy in detecting main steam line leakage. The level sensors are powered from a safety-related Class 1E electrical source. These sensors remain functional when subjected to a safe shutdown earthquake and are considered the primary method of RCPB leakage detection in containment after an SSE. This provides conformance to position 6 of Regulatory Guide 1.45 although using different technology than envisioned in that guidance (sump level rather than airborne radioactivity). The containment sump level and sump total flow sensors located on the discharge of the sump pump are part of the liquid radwaste system.

Failure of two of the level sensors will still allow the calculation of a 0.5 gpm in-leakage rate within 1 hour. The data display and processing system (DDS) computes the leakage rate and the plant control system (PLS) provides an alarm in the main control room if the average change in leak rate for any given measurement period exceeds 0.5 gpm for unidentified leakage. The minimum detectable leak rate is no greater than 0.19 gpm. Unidentified leakage is the total leakage minus the identified leakage. The leakage rate algorithm subtracts the identified leakage directed to the sump.

Control room display and functionality for the level sensors is not seismically qualified; however, local readout (outside containment) of the instrumentation is provided and is qualified to be operable post-SSE.

To satisfy positions 2 and 5 of Regulatory Guide 1.45, the measurement interval must be long enough to permit the measurement loop to adequately detect the increase in level that would correspond to 0.5 gpm leak rate, and yet short enough to ensure that such a leak rate is detected within an hour. The measurement interval is less than or equal to 1 hour.

When the sump level increases to the high level setpoint, one of the sump pumps automatically starts to pump the accumulated liquid to the waste holdup tanks in the liquid radwaste system. The sump discharge flow is integrated and available for display in the control room, in accordance with position 7 of Regulatory Guide 1.45.

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Procedures to identify the leakage source upon a change in the unidentified leakage rate into the sump include the following:

- Check for changes in containment atmosphere radiation monitor indications,
- Check for changes in containment humidity, pressure, and temperature,
- Check makeup rate to the reactor coolant system for abnormal increases,
- Perform an RCS inventory balance,
- Check for changes in water levels and other parameters in systems which could leak water into the containment, and
- Review records for maintenance operations which may have discharged water into the containment.

This procedure allows identification of main steam line leakage as well as RCS leakage.

5.2.5.3.2 Reactor Coolant System Inventory Balance

Reactor coolant system inventory monitoring provides an indication of system leakage. Net level change in the pressurizer is indicative of system leakage. Monitoring net makeup from the chemical and volume control system and net collected leakage provides an important method of obtaining information to establish a water inventory balance. An abnormal increase in makeup water requirements or a significant change in the water inventory balance can indicate increased system leakage.

The reactor coolant system inventory balance is a quantitative inventory or mass balance calculation. This approach allows determination of both the type and magnitude of leakage. Steady-state operation is required to perform a proper inventory balance calculation. Steady-state is defined as stable reactor coolant system pressure, temperature, power level, pressurizer level, and reactor coolant drain tank and in-containment refueling water storage tank levels. The reactor coolant inventory balance is done on a periodic basis and when other indication and detection methods indicate a change in the leak rate. The minimum detectable leak is 0.13 gpm.

The mass balance involves isolating the reactor coolant system to the extent possible and observing the change in inventory which occurs over a known time period. This involves isolating the systems connected to the reactor coolant system. System inventory is determined by observing the level in the pressurizer. Compensation is provided for changes in plant conditions which affect water density. The change in the inventory determines the total reactor coolant system leak rate. Identified leakages are monitored (using the reactor coolant drain tank) to calculate a leakage rate and by monitoring the intersystem leakage. The unidentified leakage rate is then calculated by subtracting the identified leakage rate from the total reactor coolant system leakage rate.

Since the pressurizer inventory is controlled during normal plant operation through the level control system, the level in the pressurizer will be reasonably constant even if leakage exists. The mass contained in the pressurizer may fluctuate sufficiently, however, to have a significant effect on the calculated leak rate. The pressurizer mass calculation includes both the steam and water mass contributions.

Changes in the reactor coolant system mass inventory are a result of changes in liquid density. Liquid density is a strong function of temperature and a lesser function of pressure. A range of temperatures exists throughout the reactor coolant system all of which may vary over time. A simplified, but

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acceptably accurate, model for determining mass changes is to assume all of the reactor coolant system is at $T_{Average}$.

The inventory balance calculation is done by the data display and processing system with additional input from sensors in the protection and safety monitoring system, chemical and volume control system, and liquid radwaste system. The use of components and sensors in systems required for plant operation provides conformance with the regulatory guidance of position 6 in Regulatory Guide 1.45 that leak detection should be provided following seismic events that do not require plant shutdown.

5.2.5.3.3 Containment Atmosphere Radioactivity Monitor

Leakage from the reactor coolant pressure boundary will result in an increase in the radioactivity levels inside containment. The containment atmosphere is continuously monitored for airborne particulate radioactivity. Air flow through the monitor is provided by the suction created by a vacuum pump. An F18 particulate concentration monitor indicates radiation concentrations in the containment atmosphere.

F18 particulate is a neutron activated product, which is proportional to power levels. An increase in activity inside containment would, therefore, indicate a leakage from the reactor coolant pressure boundary. Based on the concentration of F18 in particulate form and the power level, reactor coolant pressure boundary leakage can be estimated.

The F18 particulate monitor is seismic Category I. The remaining tubing is not seismically qualified, and safety-related Class 1E power is not provided. The leakage detection system can be reasonably expected to remain functional following seismic events of lesser severity than the SSE. However, no special qualification program is used to ensure operability under such conditions. Conformance to the intent of position 6 guidance of Regulatory Guide 1.45 — that is, leak detection should be provided following seismic events that do not require plant shutdown — is provided by the seismic Category I containment sump level monitor leakage detection system.

The F18 particulate monitor is operable when the plant is above 20-percent power, and can detect a 0.5 gpm leak within 1 hour when the plant is at full power.

Radioactivity concentration indication and alarms for loss of sample flow, high radiation, and loss of indication are provided. Sample collection connections permit sample collection for laboratory analysis. The radiation monitor can be calibrated during power operation.

5.2.5.3.4 Containment Pressure, Temperature and Humidity Monitors

Reactor coolant pressure boundary leakage increases containment pressure, temperature, and humidity, values available to the operator through the plant control system.

An increase in containment pressure is an indication of increased leakage or a high energy line break. Containment pressure is monitored by redundant Class 1E pressure transmitters. For additional discussion see Subsection 6.2.2, Passive Containment Cooling System.

The containment average temperature is monitored using temperature instrumentation at the inlet to the containment fan cooler as an indication of increased leakage or a high energy line break. This instrumentation as well as temperature instruments within specific areas, including steam generator areas, pressurizer area, and containment compartments, are part of the containment recirculation cooling system.

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An increase in the containment average temperature combined with an increase in containment pressure indicate increased leakage or a high energy line break. The individual compartment area temperatures can assist in identifying the location of the leak.

Containment humidity is monitored using temperature-compensated humidity detectors which determine the water-vapor content of the containment atmosphere. An increase in the containment atmosphere humidity indicates release of water vapor within the containment. The containment humidity monitors are part of the containment leak rate test system.

The humidity monitors supplement the containment sump level monitors and are most sensitive under conditions when there is no condensation. A rapid increase of humidity over the ambient value by more than 10 percent is indication of a probable leak.

Containment pressure, temperature and humidity can assist in identifying and locating a leak. They are not relied on to quantify a leak.

5.2.5.3.5 Response to Reactor Coolant System Leakage

Operating procedures specify operator actions in response to prolonged low level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the Technical Specification (TS) limits to provide operators sufficient time to take action before the TS limit is reached. The procedures include identifying, monitoring, trending, and addressing prolonged low level leakage. The procedures for effective management of leakage, including low level leakage, are developed including the following operations related activities:

- Trends in the unidentified leakage rates are periodically analyzed. When the leakage rate increases noticeably from the baseline leakage rate, the safety significance of the leak is evaluated. The rate of increase in the leakage is determined to verify that plant actions can be taken before the plant exceeds TS limits.

- Procedures are established for responding to leakage. These procedures address the following considerations to prevent adverse safety consequence results from the leakage:

  – Plant procedures specify operator actions in response to leakage rates less than the limits set forth in the Technical Specifications. The procedures include actions for confirming the existence of a leak, identifying its source, increasing the frequency of monitoring, verifying the leakage rate (through a water inventory balance), responding to trends in the leakage rate, performing a walkdown outside containment, planning a containment entry, adjusting alarm setpoints, limiting the amount of time that operation is permitted when the sources of the leakage are unknown, and determining the safety significance of the leakage.

  – Plant procedures specify the amount of time the leakage detection and monitoring instruments (other than those required by Technical Specifications) may be out of service to effectively monitor the leakage rate during plant operation (i.e., hot shutdown, hot standby, startup, transients, and power operation).

- The output and alarms from leakage monitoring systems are provided in the main control room. Procedures are readily available to the operators for converting the instrument output to a common leakage rate. (Alternatively, these procedures may be part of a computer program so that the operators have a real-time indication of the leakage rate as determined from the output of these monitors.) Periodic calibration and testing of leakage monitoring

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systems are conducted. The alarm(s), and associated setpoint(s), provide operators an early warning signal so that they can take corrective actions, as discussed above, i.e., before the plant exceeds TS limits.

- During maintenance and refueling outages, actions are taken to identify the source of any unidentified leakage that was detected during plant operation. In addition, corrective action is taken to eliminate the condition resulting in the leakage.

The procedures described above will be available prior to fuel load.

5.2.5.4 Safety Evaluation

Leak detection monitoring has no safety-related function. Therefore, the single failure criterion does not apply and there is no requirement for a nuclear safety evaluation. The containment sump level monitors and the containment atmosphere monitor are seismic Category I. The components used to calculate reactor coolant system inventory balance are both safety-related and nonsafety-related components. The containment sump level monitors are powered from the Class 1E dc and UPS system (IDS). Measurement signals are processed by the data display and processing system and the plant control system (PLS).

5.2.5.5 Tests and Inspections

To satisfy position 8 of Regulatory Guide 1.45, periodic testing of leakage detection monitors verifies the operability and sensitivity of detector equipment. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks in conformance with regulatory guidance.

5.2.5.6 Instrumentation Applications

The parameters tabulated below satisfy position 7 of Regulatory Guide 1.45 and are provided in the main control room to allow operating personnel to monitor for indications of reactor coolant pressure boundary leakage. The containment sump level, containment atmosphere radioactivity, reactor coolant system inventory balance, and the flow measurements are provided as gallon per minute leakage equivalent.

Parameter	System(s)	Alarm or Indication
Containment sump level and sump total flow	WLS	Both
Reactor coolant drain tank level and drain tank total flow	WLS	Both
Containment atmosphere radioactivity	PSS	Both
Reactor coolant system inventory balance parameters	CVS, PCS, PXS, RCS, WLS	Both
Containment humidity	VUS	Indication
Containment atmospheric pressure	PCS	Both
Containment atmosphere temperature	VCS	Both
Containment water level	PXS	Both
Reactor vessel head seal leak temperature	WLS	Both
Pressurizer safety relief valve leakage temperature	RCS	Both
Reactor coolant pump bearing water RTDs	RCS	Both

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Steam generator blowdown radiation	BDS	Both
Turbine island vent discharge radiation	TDS	Both
Component cooling water radiation	CCS	Both
Main steam line radiation	SGS	Both
Component cooling water surge tank level	CCS	Both

5.2.5.7 Technical Specification

Limits which satisfy position 9 of Regulatory Guide 1.45 for identified and unidentified reactor coolant leakage are identified in the Technical Specifications. LCO 3.4.7 addresses RCS leakage limits. LCO 3.7.8 addresses main steam line leakage limits. LCO 3.4.9 addresses leak detection instrument requirements.

5.2.6 Combined License Information Items

5.2.6.1 ASME Code and Addenda

The ASME Code editions and addenda to be used are addressed in Subsection 5.2.1.1.

5.2.6.2 Plant-Specific Inspection Program

The plant-specific preservice inspection and inservice inspection program is addressed in Subsections 5.2.4, 5.2.4.1, 5.2.4.3.1, 5.2.4.3.2, 5.2.4.4, 5.2.4.5, 5.2.4.6, 5.2.4.8, 5.2.4.9, and 5.2.4.10.

5.2.6.3 Response to Unidentified Reactor Coolant System Leakage Inside Containment

Prolonged low-level unidentified reactor coolant leakage inside containment is addressed in Subsection 5.2.5.3.5.

5.2.7 References

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3. Logsdon, W. A., Begley, J. A., and Gottshall, C. L., "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals," WCAP-9292, March 1978.

4. Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970, and WCAP-7735 (Nonproprietary), August 1971.

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201. EPRI, “Pressurized Water Reactor Primary Water Chemistry Guidelines.”

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Table 5.2-1 (Sheet 1 of 6)
Reactor Coolant Pressure Boundary Materials Specifications

Component	Material Specification	Class, Grade, or Type
Reactor Vessel Components		
Head plates (other than core region)	SA-533 SA-508	Type B, CL 1 GR 3 CL 1
Shell courses	SA-508	GR 3 CL 1
Shell, flange, and nozzle forgings	SA-508	GR 3 CL 1
Nozzle safe ends	SA-182	F316, F316L, F316LN
CRDM latch housing head penetration nozzle	SB-167 SB-166	N06690 N06690
Instrumentation nozzles, upper head	SB-167 SB-166 SA-182 SA-479	N06690 N06690 F304, F304L, F304LN, F316, F316L, F316LN 304, 304L, 304LN 316, 316L, 316LN, S21800
Closure studs	SA-540	GR B23 CL 3, GR B24 CL 3
Monitor tubes	SA-312(1) SA-376 SA-182	TP304, TP304L, TP304LN, TP316, TP316L, TP316LN TP304, TP304LN, TP316, TP316LN F304, F304L, F304LN, F316, F316L, F316LN
Vent pipe	SB-166 SB-167 SA-312(1) SA-376	N06690 N06690 TP304, TP304L, TP304LN, TP316, TP316L, TP316LN TP304, TP304LN, TP316, TP316LN

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Table 5.2-1 (Sheet 2 of 6)
Reactor Coolant Pressure Boundary Materials Specifications

Component	Material Specification	Class, Grade, or Type
Steam Generator Components		
Pressure plates	SA-533	Type B, CL 1 or CL 2
Pressure forgings (including primary side nozzles and tube sheet)	SA-508	GR 3, CL 2
Nozzle safe ends	SA-182 SA-336 SB-564	F316, F316L, F316LN F316LN N06690
Channel heads	SA-508	GR 3, CL 2
Tubes	SB-163	N06690
Manway studs/ Nuts	SA-193 SA-194	GR B7 GR 7
Pressurizer Components		
Pressure plates	SA-533	Type B, CL 1
Pressure forgings	SA-508	GR 3, CL 2
Nozzle safe ends	SA-182 SA-336 SB-564 SB-163	F316, F316L, F316LN F316, F316L, F316LN N06690 N06690
Manway studs/ Nuts	SA-193 SA-194	GR B7 GR 7

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Table 5.2-1 (Sheet 3 of 6)
Reactor Coolant Pressure Boundary Materials Specifications

Component	Material Specification	Class, Grade, or Type
Reactor Coolant Pump		
Pressure forgings	SA-182	F304, F304L, F304LN, F304N F316, F316L, F316LN, F316N, F6 per Code Case N-4-11
	SA-508	GR1[4]
	SA-336	F304, F304L, F304LN, F316, F316L, F316LN, F316N
Pressure casting	SA-351	CF3A or CF8A
Tube and pipe	SA-213	TP304, TP304L, TP304LN, TP316, TP316L, TP316LN
	SA-376	TP304, TP304LN, TP316, TP316LN
	SA-312[1]	TP304, TP304L, TP304LN, TP316, TP316L, TP316LN
	SA-403	316LN
Pressure bar stock	SA-479	304
Pressure plates	SA-240	304, 304L, 304LN, 304N 316, 316L, 316LN
Closure bolting	SA-193	GR B7
	SA-194	GR 2H or 7
	SA-540	GR B24, CL2 & CL 4, or GR B23, CL 2, CL 3 & 4
Reactor Coolant Piping		
Reactor coolant pipe	SA-376	TP304, TP304LN, TP316, TP316LN
	SA-182[2]	F304, F304L, F304LN, F316, F316L, F316LN
Reactor coolant fittings, branch nozzles	SA-376	TP304, TP304LN, TP316, TP316LN
	SA-182	F304, F304L, 304LN, F316, F316L, F316LN
Surge line	SA-376	TP304, TP304LN, TP316, TP316LN
	SA-312[1]	TP304, TP304L, TP304LN, TP316, TP316L, TP316LN

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Table 5.2-1 (Sheet 4 of 6)
Reactor Coolant Pressure Boundary Materials Specifications

Component	Material Specification	Class, Grade, or Type
RCP piping other than loop and surge line	SA-312(1)	TP304, TP304L, TP304LN, TP316, TP316L, TP316LN
	SA-376	TP304, TP304L, TP304LN, TP316, TP316L, TP316LN
CRDM		
Latch housing except for head penetration nozzle	SA-336	F304, F304L, F304LN, F316, F316L, F316LN
	SA-182	F304, F304L, F304LN, F316, F316L, F316LN
Rod travel housing	SA-336	F304, F304L, F304LN, F316, F316L, F316LN
	SA-182	F304, F304L, F304LN, F316, F316L, F316LN
Valves		
Bodies	SA-182	F304, F304L F316, F316L
	SA-240	304, 304L 316, 316L
	SA-479	304, 304L 316, 316L
	SA-351	CF3A, CF3M, CF8, CF3, CF8M
Bonnets	SA-182	F304, F304L F316, F316L
	SA-240	304, 304L 316, 316L
	SA-351	CF3A, CF3M, CF8, CF3, CF8M
	SA-479	304, 304L 316, 316L XM-19

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Table 5.2-1 (Sheet 5 of 6)
Reactor Coolant Pressure Boundary Materials Specifications

Component	Material Specification	Class, Grade, or Type
Discs	SA-182	F304, F304L, F304LN, F316, F316L, F316LN
	SA-240	304, 304L 316, 316L
	SA-479	XM-19 304, 304L 316, 316L
	SB-637	N07750 (limited application)(9) N07718 (limited application)(9)
	SA-564	Type 630 (H1100 or H1150),
	SB-564	N06690
	A-565	GR 616 (limited application)(10)
	SA-351	CF3A, CF3M, CF8, CF3, CF8M
Stems(11)		
Pressure retaining bolting	SA-453	GR 660
	SA-564	Type 630 (H1100)
	SB-637	N07718
Pressure retaining nuts	SA-453	GR 660
	SA-194	GR 6 or 8
Core Makeup Tank		
Pressure plates	SA-533	Type B, CL 1
	SA-240	304, 304L, 304LN, 316, 316L, 316LN
Pressure forgings	SA-508	GR 3 CL 1
	SA-182	F304, F304L, F316, F316L
	SA-336	F304, F304L, F316, F316L
Passive Residual Heat Removal Heat Exchanger		
Pressure plates	SA-533	Type B CL1
	SA-240	304, 304L, 304LN

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Table 5.2-1 (Sheet 6 of 6)
Reactor Coolant Pressure Boundary Materials Specifications

Component	Material Specification	Class, Grade, or Type
Pressure forgings	SA-508	GR 3 CL 2
	SA-336	F304, F304L, F304LN
Tubing	SB-163	N06690
Welding Consumables		
Austenitic stainless steel corrosion-resistant cladding, buttering, and welds(5)	SFA 5.4	E308, E308L, E309, E309L, E316, E316L
	SFA 5.9(6)	ER308, ER308L, ER309, ER309L, ER316, ER316L, EQ308L, EQ309L
	SFA 5.22(3)	E308LTX-Y, E308TX-Y, E309LTX-Y, E309TX-Y, E316LTX-Y, E316TX-Y
	SFA 5.30	IN308, IN308L, IN316, IN316L
Ni-Cr-Fe corrosion-resistant cladding, buttering, and welds(7)	SFA 5.11	ENiCrFe-7
	SFA 5.14	ERNiCrFe-7, ERNiCrFe-7A, EQNiCrFe-7, EQNiCrFe-7A
Carbon steel pressure boundary welds(8)(4)	SFA 5.1, 5.17, 5.18, 5.20, 5.30	To be compatible with base material
Low alloy pressure boundary welds(8)	SFA 5.5, 5.23, 5.28, 5.29	To be compatible with base material

Notes:
1. Limited to seamless form only.
2. Subject to manufacturing sequence and final finish condition review.
3. Only gas-shielded electrodes for use with the FCAW process are permitted. These electrodes shall not be used for root passes except for joints welded from two sides where the root is back-gouged to sound metal as evidenced by magnetic particle or liquid penetrant testing.
   X=Position, acceptable values 0 (flat and horizontal) and 1 (all positions)
   Y=Shield Gas, acceptable values 1 (100% CO_2) and 4 (75-80% Argon, remainder CO_2)
4. GR1 material (carbon steel) and associated filler material is used only for reactor coolant pump components that are not exposed to the reactor coolant. These components are limited to the stator main flange, stator shell, and external heat exchanger supports.
5. Austenitic stainless steel filler metals that are exposed to temperatures within the 800°F to 1500°F temperature range after welding, and are not subsequently solution annealed, do not contain more than 0.03% or 0.04% carbon by weight (depending on the maximum carbon content of the corresponding low-carbon classification in the SFA specification), or have demonstrated nonsensitization per Regulatory Guide 1.44.
6. In addition to ER, EC (composite) rod/electrodes may also be used.
7. These materials are UNS N06052, N06054, and W86152, where F43 grouping is allowed by codes cases 2143-1 and 2142-2. Note that UNS N06054 is only in ASME Section II part C 2004 with 2006 addenda and later. Similar welding alloys developed for improved weldability may be used as allowed by ASME Boiler and Pressure Vessel Code rules.
8. These weld metals are compatible with the base metal mechanical requirements and meet applicable ASME Section III, Section II part C, and Section IX requirements. Their use is limited to applications in which the welds are not exposed to reactor coolant. These weld metals used with a flux bearing welding process are also not used for root passes of single-sided welds.
9. Valve material is limited by valve component per NB-2121(c)(1).
10. Valve material is limited by valve size per NB-2121(c)(3).
11. Stems are not pressure boundary material per NB-2121(b).

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Table 5.2-2
Reactor Coolant Water Chemistry Specifications

Electrical conductivity	Determined by the concentration of boric acid and alkali present. Expected range is <1 to 40 μmhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) and 10.5 (low boric acid concentration) at 25°C. Values will be 5.0 or greater at normal operating temperatures.
Oxygen(1)	0.1 ppm, maximum
Chloride(2)	0.15 ppm, maximum
Fluoride(2)	0.15 ppm, maximum
Hydrogen(3)	25 to 50 cm^3 (STP)/kg H_2O
Suspended solids(4)	0.2 ppm, maximum
pH control agent (Li7OH)(5)	Lithium is coordinated with boron per fuel warranty contract.
Boric acid	Variable from 0 to 4000 ppm as boron
Silica(6)	1.0 ppm, maximum
Aluminum(6)	0.05 ppm, maximum
Calcium(6) + magnesium	0.05 ppm, maximum
Magnesium(6)	0.025 ppm, maximum
Zinc(7)	0.04 ppm, maximum

Notes:
1. Oxygen concentration must be controlled to less than 0.1 ppm in the reactor coolant by scavenging with hydrazine prior to plant operation above 250°F. During power operation with the specified hydrogen concentration maintained in the coolant, the residual oxygen concentration will not exceed 0.005 ppm.
2. Halogen concentrations must be maintained below the specified values regardless of system temperature.
3. Hydrogen must be maintained in the reactor coolant for plant operations with nuclear power above 1 MW. The normal operating range should be 30-40 cm^3 (STP) H_2/kg H_2O.
4. Solids concentration determined by filtration through filter having 0.45-μm pore size.
5. The specified lithium concentrations must be established for startup testing prior to heatup beyond 150°F. During cold hydrostatic testing and hot functional testing in the absence of boric acid, the reactor coolant limits for lithium hydroxide must be maintained to inhibit halogen stress corrosion cracking.
6. These limits are included in the table of reactor coolant specifications as recommended standards for monitoring coolant purity. Establishing coolant purity within the limits shown for these species is judged desirable with the current data base to minimize fuel clad crud deposition, which affects the corrosion resistance and heat transfer of the clad.
7. Specification is applicable during power operation when zinc is being injected. The zinc concentration is maintained at the lower of 0.04 ppm or that specified in the reload safety analyses.

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Table 5.2-3
ASME Code Cases

Code Case Number	Title
N-4-11	Special Type 403 Modified Forgings or Bars, Section III, Division 1, Class 1 and Class CS
N-20-4	SB-163 Nickel-Chromium-Iron Tubing (Alloys 600 and 690) and Nickel-Iron-Chromium Alloy 800 at a Specified Minimum Yield Strength of 40.0 ksi and Cold Worked Alloy 800 at Yield Strength of 47.0 ksi, Section III, Division 1, Class 1
N-60-5	Material for Core Support Structures, Section III, Division 1[(a)]
N-71-18	Additional Material for Subsection NF, Class 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III Division 1
[*N-122-2*	*Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 1 Piping Section III, Division 1*]*
N-192-3	Use of Braided Flexible Connectors, Section III, Division 1, Class 2 and 3
N-249-14	Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated Without Welding, Section III, Division 1[(b)]
[*N-284-1*	*Metal Containment Shell Buckling Design Methods, Section III, Division 1 Class MC*]*
[*N-318-5*	*Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping Section III, Division*]*
[*N-319-3*	*Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping Section III, Division 1*]*
[*N-391-2*	*Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping Section III, Division 1*]*
[*N-392-3*	*Procedure for Valuation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping Section III, Division 1[(c)]*]*
N-405-1	Socket Welds, Section III, Division 1
N-474-2	Design Stress Intensities and Yield Strength Values for UNS06690 With a Minimum Yield Strength of 35 ksi, Class 1 Components, Section III, Division 1
2142-2	F-Number Grouping for Ni-Cr-Fe Filler Metals, Section IX (Applicable to all Sections, including Section III, Division 1, and Section XI)
2143-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode, Section IX
N-655-1	Use of SA-738, Grade B, for Metal Containment Vessels, Class MC, Section III, Division 1
N-757-1	Alternative Rules for Acceptability for Class 2 and 3 Valves, NPS 1 (DN25) and Smaller with Welded and Nonwelded End Connections other than Flanges, Section III, Division 1[(d)]
N-759-2	Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Class 1, 2, and 3, Section III, Division 1
N-782	Use of Code Editions, Addenda, and Cases Section III, Division 1
N-801	Rules for Repair of N-Stamped Class 1, 2, and 3 Components by Organization Other Than the N Certificate Holder That Originally Stamped the Component Being Repaired, Section III, Division 1[(e)]
N-802	Rules for Repair of Stamped Components by the N Certificate Holder That Originally Stamped the Component, Section III, Division 1[(e)]
N-844	Alternatives to the Requirements of NB-4250(c) Section III, Division 1[(f)]
N-870-1	Rules for the Elimination of External Surface Defects on Class 1, 2, and 3 Piping, Pumps, or Valves After Component Stamping and Prior to Completion of the N-3 Data Report, Section III, Division 1[(f)]
N-887	Alternatives to the Requirements of NB-4424.2(a), Figure NB-4250-2, and Figure NB-4250-3, Section III, Division 1[(f)]

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 5.2-3
ASME Code Cases

Notes:

(a) Use of this code case will meet the conditions for Code Case N-60-4 in Reg. Guide 1.85 Revision 30.
(b) Use of this code case will meet the conditions for Code Case N-249-10 in Reg. Guide 1.85 Revision 30.
(c) Use of this code case will meet the conditions for Code Case N-392-1 in Reg. Guide 1.84 Revision 30.
(d) Use of this code case is subject to the condition that the design provisions of ASME Code, Section III, Division I, Appendix XIII not be used for the design of Code Class 3 (ND) valves.
(e) Use of this code case is acceptable per Reg. Guide 1.84 Revision 37.
(f) Use of this code case is acceptable per Reg. Guide 1.84 Revision 40.

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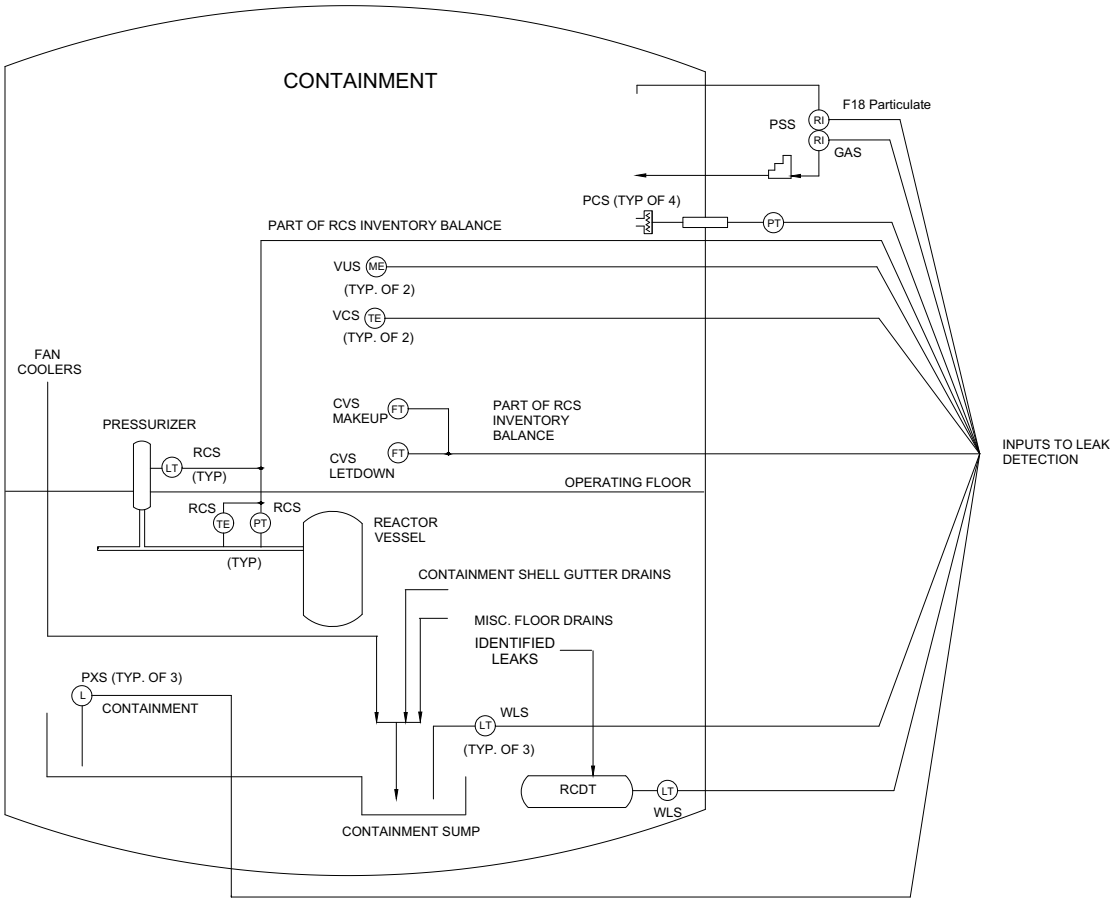

Figure 5.2-1
Leak Detection Approach

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5.3 Reactor Vessel

5.3.1 Reactor Vessel Design

5.3.1.1 Safety Design Bases

The reactor vessel, as an integral part of the reactor coolant pressure boundary will be designed, fabricated, erected and tested to quality standards commensurate with the requirements set forth in 10 CFR 50, 50.55a and General Design Criterion 1. Design and fabrication of the reactor vessel is carried out in accordance with ASME Code, Section III, Class 1 requirements. Subsections 5.2.3 and 5.3.2 provide further details.

The performance and safety design bases of the reactor vessel follow:

- The reactor vessel provides a high integrity pressure boundary to contain the reactor coolant, heat generating reactor core, and fuel fission products. The reactor vessel is the primary pressure boundary for the reactor coolant and the secondary barrier against the release of radioactive fission products.
- The reactor vessel provides support for the reactor internals, flow skirt, and core to ensure that the core remains in a coolable configuration.
- The reactor vessel directs main coolant flow through the core by close interface with the reactor internals and flow skirt.
- The reactor vessel provides for core internals location and alignment.
- The reactor vessel provides support and alignment for the control rod drive mechanisms and in-core instrumentation assemblies.
- The reactor vessel provides support and alignment for the integrated head assembly.
- The reactor vessel provides an effective seal between the refueling cavity and sump during refueling operations.
- The reactor vessel supports and locates the main coolant loop piping.
- The reactor vessel provides support for safety injection flow paths.
- The reactor vessel serves as a heat exchanger during core meltdown scenario with water on the outside surface of the vessel.

5.3.1.2 Safety Description

The reactor vessel consists of a cylindrical section with a transition ring, hemispherical bottom head, and a removable flanged hemispherical upper head (Figure 5.3-1). Key dimensions are shown in Figures 5.3-5 and 5.3-6. The cylindrical section consists of two shells, the upper shell and the lower shell. The upper and lower shells and the lower hemispherical head are fabricated from low alloy steel and clad with austenitic stainless steel. The upper shell forging is welded to the lower shell forging, and the lower shell is welded to the transition ring, which is welded to the hemispherical bottom head. The removable flanged hemispherical upper head consists of a single forging, which includes the closure head flange and the closure head dome. The closure head is fabricated from a low alloy steel forging and clad with austenitic stainless steel. Specifics of the processes used in base materials, clad material, and weld materials are discussed in Subsection 5.2.3. The removable

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flanged hemispherical closure head is attached to the vessel (consisting of the upper shell-lower shell-transition ring-bottom hemispherical head dome) by studs. Two metal o-rings are used for sealing the two assemblies. Inner and outer monitor tubes are provided through the upper shell to collect any leakage past the o-rings. Details of the head gasket monitoring connections are included in Subsection 5.2.5.1.2.

The reactor vessel supports the internals. An internal ledge is machined into the top of the upper shell section. The core barrel flange rests on the ledge. A large circumferential spring is positioned on the top surface of the core barrel flange. The upper support plate rests on the top surface of the spring. The spring is compressed by installation of the reactor vessel closure head and the upper and lower core support assemblies are restrained from any axial movements.

Four core support pads are located on the bottom hemispherical head just below the transition ring-to-lower shell circumferential weld. The core support pads function as a clevis. At assembly, as the lower internals are lowered into the vessel, the keys at the bottom of the lower internals engage the clevis in the axial direction. With this design, the internals are provided with a lateral support at the furthest extremity and may be viewed as a beam supported at the top and bottom.

The interfaces between the reactor vessel and the lower internals core barrel are such that the main coolant flow enters through the inlet nozzle and is directed down through the annulus between the reactor vessel and core barrel and through the flow skirt and flows up through the core. The annulus is designed such that the core remains in a coolable configuration for all design conditions.

Prior to installation of the internals into the reactor vessel, guide studs are assembled into the upper shell. Dimensional relationships are established between the guide studs and the core support pads such that when the lower internals lifting rig engages the guide studs, the keys at the bottom of the lower internals are in relative circumferential position to enter the core support pads.

There are 69 penetrations in the removable flanged hemispherical upper head (closure head) that are used to provide access for the control rod drive mechanisms. Each control rod drive mechanism is positioned in its opening and welded to the closure head penetration. In addition there are eight penetrations in the closure head used to provide access for in-core and core exit instrumentation.

Lugs are welded to the outside surface of the closure head along the outer periphery of the dome section. The purpose of these lugs is to provide support and alignment for the integrated head package.

Attached to the top surface and along the outer periphery of the upper shell is a permanent cavity seal ring. During field assembly the permanent cavity seal ring is welded to the refueling cavity liner. This permanent cavity seal ring provides an effective water seal between the refueling cavity and sump during refueling operations.

A support pad is integral to each of the four inlet nozzles. The reactor vessel is supported by the pads. The pads rest on steel base pads atop a support structure, which is attached to the concrete foundation wall. Thermal expansion and contraction of the vessel are accommodated by sliding surfaces between the support pads and the base plates. Side stops on these plates keep the vessel centered and resist lateral loads.

The reactor vessel primary and direct vessel injection (DVI) nozzles are located in the upper shell. These nozzles are either forged as part of the upper shell forging or are fabricated by “set in” construction such that the welding is through the vessel shell forging. A stainless steel safe end is shop welded to each of the four inlet, two outlet and two DVI nozzles to facilitate field welding without heat treatment to the stainless steel reactor coolant piping system. The primary coolant nozzles

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support one end of the primary coolant system. Reaction loads are transferred into the nozzles and eventually into the support pads. The inlet and outlet elevation nozzles are offset in different planes by 17.5 inches. This allows pump maintenance without discharging the core.

There are no penetrations in the reactor vessel below the core. This eliminates the possibility of a loss-of-coolant accident by leakage from the reactor vessel that would allow the core to be uncovered.

5.3.1.3 System Safety Evaluation

The reactor vessel is part of the reactor coolant system. Load and stress evaluation for operating loads and mechanical transients of safe shutdown earthquake (SSE), and pipe ruptures appear in Subsection 3.9.3.

5.3.1.4 Inservice Inspection/Inservice Testing

Inservice surveillance is discussed in Subsection 5.3.4.7.

5.3.2 Reactor Vessel Materials

5.3.2.1 Material Specifications

Material specifications are in accordance with the ASME Code requirements and are given in Subsection 5.2.3. All ferritic reactor vessel materials comply with the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR 50.

The ferritic materials of the reactor vessel beltline are restricted to the maximum limits shown in Table 5.3-1. Copper, nickel, and phosphorus content is restricted to reduce sensitivity to irradiation embrittlement in service.

5.3.2.2 Special Processes Used for Manufacturing and Fabrication

The reactor vessel is classified as AP1000 Class A. Design and fabrication of the reactor vessel is carried out in accordance with ASME Code, Section III, Class 1 requirements. The shell sections, flange, and nozzles are manufactured as forgings. The hemispherical heads are made from dished plates or forgings. The reactor vessel parts are joined by welding, using the single or multiple wire submerged arc and the shielded metal arc processes. Gas metal arc welding and plasma arc welding are acceptable methods of applying buttering for dissimilar metal welds.

The use of severely sensitized stainless steel as a pressure boundary material is prohibited and is eliminated by either a select choice of material or by programming the method of assembly.

At locations in the reactor vessel where stainless steel and nickel-chromium-iron alloy are joined, the final joining beads are nickel-chromium-iron alloy weld metal in order to prevent cracking.

The location of full penetration weld seams in the upper closure head and vessel bottom head are restricted to areas that permit accessibility during in-service inspection.

The stainless steel clad surfaces are sampled to demonstrate that composition requirements are met.

Freedom from underclad cracking is provided by special evaluation of the procedure qualification for cladding applied on low-alloy steel (SA-508, GR 3 CL 1).

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Minimum preheat requirements have been established for pressure boundary welds using low-alloy material. The preheat is maintained until either a low temperature (400°F – 500°F) post heat treatment, an intermediate postweld heat treatment or a full postweld heat treatment is performed.

A field weld is made, after the reactor vessel has been set, to install the permanent cavity seal ring. This stainless steel filler weld joins the permanent cavity seal ring to the reactor vessel seal ledge. A minimum preheat is specified for this weld in compliance with the ASME Code requirements.

The flow skirt is also welded to support lugs in the field after the reactor vessel/internals system is set.

5.3.2.3 Special Methods for Nondestructive Examination

The nondestructive examination (NDE) of the reactor vessel and its appurtenances is conducted in accordance with ASME Code, Section III requirements; also, numerous examinations are performed in addition to ASME Code, Section III requirements. The nondestructive examination of the vessel is discussed in the following paragraphs, and the reactor vessel quality assurance program is given in Table 5.3-2.

5.3.2.3.1 Ultrasonic Examination

In addition to the required ASME Code straight beam ultrasonic examination, angle beam inspection over 100 percent of one major surface of plate material is performed during fabrication to detect discontinuities that may be undetected by the straight beam examination.

In addition to the ASME Code, Section III nondestructive examination, the full penetration ferritic pressure boundary welds in the reactor vessel, the nozzle to safe end welds, the primary and direct vessel injection (DVI) nozzle inner radii and inner protrusions, the Quickloc instrument nozzle (QIN) weld buildups, and the QIN dissimilar metal welds are ultrasonically examined during fabrication. These tests are performed upon completion of the welding and intermediate heat treatment but prior to the final postweld heat treatment.

After hydrotesting, the full penetration ferritic pressure boundary welds in the reactor vessel, the nozzle to safe end welds, the primary and DVI nozzle inner radii and inner protrusions, the QIN weld buildups, and the QIN dissimilar metal welds are ultrasonically examined. These inspections are performed in addition to the ASME Code, Section III nondestructive examination requirements.

5.3.2.3.2 Penetrant Examinations

The partial penetration welds between the control rod drive mechanism latch housing nozzle and reactor vessel closure head are inspected by dye penetrant after the root pass, in addition to ASME code requirements. The QIN full penetration dissimilar metal weld is examined by dye penetrant on the back chip and final surfaces to ASME code requirements. Core support block attachment welds are inspected by dye penetrant after the first layer of weld metal and after each 0.5 inch of weld metal. Clad surfaces and other vessel and head internal surfaces are inspected by dye penetrant after the hydrostatic test.

5.3.2.3.3 Magnetic Particle Examination

Magnetic particle examination requirements below are in addition to the magnetic particle examination requirements of Section III of the ASME Code. All magnetic particle examinations of materials and welds are performed in accordance with the following:

- Prior to the final postweld heat treatment, by the prod, coil, direct contact, or yoke method

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- After the final postweld heat treatment, only by the yoke method

The following surfaces and welds are examined by magnetic particle methods. The acceptance standards are in accordance with Section III of the ASME Code.

Surface Examinations

- Magnetic particle examination of exterior vessel and head surfaces after the hydrostatic test.
- Magnetic particle examination of exterior closure stud surfaces and all nut surfaces after final machining or rolling. Continuous circular and longitudinal magnetization is used.
- Magnetic particle examination of inside diameter surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. Liquid (dye) penetrant examination may be used as an alternative method, in accordance with ASME B&PV, Section III, NB-2541(a), when the inspection surface and surrounding geometry do not permit effective use of magnetic particle examination equipment. Both dye penetrant examination and magnetic particle examination are capable to detect surface flaws in carbon and low alloy steels regardless of the cooling process of the material. This inspection is performed after forming and machining and prior to cladding.

Weld Examination

Magnetic particle examination of the welds attaching the closure head lifting lugs and refueling seal ledge to the reactor vessel after the first layer and each 0.5 inch of weld metal is deposited. All pressure boundary welds are examined after back-chipping or back-grinding operations.

5.3.2.4 Special Controls for Ferritic and Austenitic Stainless Steels

Welding of ferritic steels and austenitic stainless steels is discussed in Subsection 5.2.3. Subsection 5.2.3 includes discussions on the degree of conformance with Regulatory Guide 1.44. Section 1.9 discusses the degree of conformance with Regulatory Guides, including 1.31 and 1.34 (if applicable), as well as 1.37, 1.43, 1.50, 1.71, and 1.99.

5.3.2.5 Fracture Toughness

Assurance of adequate fracture toughness of ferritic materials in the reactor vessel (ASME Code, Section III, Class 1 component) is provided by compliance with the requirements for fracture toughness testing included in NB-2300 to Section III of the ASME Code and Appendix G of 10 CFR 50.

The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline base metal transverse direction and welds are 75 foot-pounds, as required by Appendix G of 10 CFR 50. The vessel fracture toughness data are given in Table 5.3-3. The AP1000 end-of-life RT_{NDT} and upper shelf energy projections were estimated using Regulatory Guide 1.99 for the end-of-life neutron fluence at the 1/4-thickness (T) and ID reactor vessel locations.

5.3.2.6 Material Surveillance

In the surveillance program, the evaluation of radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch, tensile, and 1/2-T compact tension (CT) fracture mechanics test specimens. The program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach. The program conforms to ASTM E-185 (Reference 1) and 10 CFR 50, Appendix H.

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Surveillance test materials are prepared from the actual materials used in fabricating the beltline region of the reactor vessel. Records are maintained of the chemical analyses, fabrication history, mechanical properties and other essential variables pertinent to the fabrication process of the shell forging and weld metal from which the surveillance test materials are prepared. The test materials are processed so that they are representative of the material in the completed reactor vessel.

Three metallurgically different materials prepared from sections of reactor vessel shell forging are used for test specimens. These include base metal, weld metal and heat affected zone (HAZ) material.

Base metal test material is manufactured from a section of ring forging, either the intermediate shell course, the lower shell course, or the transition ring of the reactor pressure vessel. Selection is based on an evaluation of initial toughness (characterized by the reference temperature (RT_{NDT}) and Upper Shelf Energy (USE)), and the predicted effect of chemical composition (nickel and residual copper) and neutron fluence on the toughness (RT_{NDT} shift and decrease in USE) during reactor operation. The ring forging with the highest predicted adjusted RT_{NDT} temperature (initial RT_{NDT} plus RT_{NDT} shift) or that with USE predicted to approach close to the minimum limit of 50 ft-lb at end-of-license (EOL) is selected as the surveillance base metal test material. The means for measuring initial toughness and for predicting irradiation induced toughness changes is consistent with applicable procedures in force at the time the material is being selected. The section of shell forging used for the base metal test block is adjacent to the test material used for fracture toughness tests.

Weld metal and HAZ test material is produced by welding together sections of the forgings from the beltline of the reactor vessel. The HAZ test material is manufactured from a section of the same shell course forging used for base metal test material. The sections of shell course forging used for weld metal and HAZ test material are adjacent to the test material used for fracture toughness tests. The heat of wire or rod and lot of flux are from the same heat and lot used in making the beltline region welds. Welding parameters duplicate those used for the beltline region welds. The procedures for inspection of the reactor vessel welds are followed for the inspection of the welds in test materials. The surveillance weld and HAZ material are heat-treated to metallurgical conditions which are representative of the final metallurgical conditions of similar materials in the completed reactor vessel.

Test Specimens are marked to identify the type of materials and the orientation with respect to the test materials. Drawings specify the identification system to be used and include plant identification, type of material, orientation of specimen and sequential number.

Baseline test specimens are provided for establishing the baseline (unirradiated) properties of the reactor vessel materials. The data from tests of these specimens provides the basis for determining the radiation induced property changes of the reactor vessel materials.

Drop weight test specimens of each of base metal, weld metal, and HAZ metal are provided for establishing the nil-ductility transition temperature (NDTT) of the unirradiated surveillance materials. These data form the basis for RT_{NDT} determination from which subsequent radiation induced changes are determined.

Standard Charpy impact test specimens each of base metal (longitudinal (tangential) and transverse (axial)), weld metal, and HAZ material are provided for developing a Charpy impact energy transition curve from fully brittle to fully ductile behavior for defining specific index temperatures for these materials. These data, together with the drop weight NDTT, are used to establish an RT_{NDT} for each material.

Tensile test specimens each of base metal (longitudinal (tangential) and transverse (axial)), weld metal, and HAZ metal are provided to permit a sufficient number of tests for accurately establishing

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the tensile properties for these materials at a minimum of three test temperatures (e.g., ambient, operating and one intermediate temperature) to define the strength of the material.

The above described test specimens are to be used for determining changes in the strength and toughness of the surveillance materials resulting from neutron irradiation. Sufficient Charpy impact, compact tension and tensile test specimens are provided for establishing the changes in the properties of the surveillance materials over the lifetime of the reactor vessel. The type, quantity, and storage conditions (e.g., surveillance capsules backfilled with inert gas) of test specimens meet or exceed the minimum requirements of ASTM E-185.

Reactor materials do not begin to be affected by neutron fluence until the reactor begins critical operation. Table 13.4-201 provides milestones for reactor vessel material surveillance program implementation.

The reactor vessel surveillance program incorporates eight specimen capsules. The capsules are located in guide baskets bolted to the outside of the core barrel as shown in Figure 5.3-4 and positioned directly opposite the center portion of the core. The capsules can be removed when the vessel head is removed. To meet the guidelines of ASTM E-185 (lead factors less than three), the specimen guide baskets are located azimuthally near the lowest fluence locations at 135, 225, and 315 degrees. The 45 degree location is also a low fluence azimuthal location; however, there is a Roto-Lock insert for the internals lifting rig, which would prevent access for removal of the capsules from the baskets. Therefore, there are no guide baskets at the 45 degree location. Eight specimen capsules are provided by including three guide baskets at the 135 and 315 degree azimuthal locations and two baskets at the 225 degree location.

The capsules contain reactor vessel weld metal, base metal, and heat-affected zone metal specimens. The base metal specimens are oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting base material located in the core region of the reactor vessel. The 8 capsules contain 72 tensile specimens, 480 Charpy V-notch specimens, and 48 compact tension specimens. Archive material sufficient for two additional capsules and heat-affected-zone (HAZ) materials is retained.

Dosimeters, as described below, are placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys are included to monitor the maximum temperature of the specimens. The specimens are enclosed in a tight-fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity. The complete capsule is helium leak tested. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent is made for surveillance material and as deposited weld metal. Each of the eight capsules contains the specimens shown in Table 5.3-4.

The following dosimeters and thermal monitors are included in each of the eight capsules:

- Dosimeters
  - Iron
  - Copper
  - Nickel
  - Niobium-93 (cadmium shielded)
  - Cobalt-aluminum (0.15-percent cobalt)

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  - Cobalt-aluminum (cadmium shielded)

- Thermal Monitors
  - 97.5-percent lead, 2.5-percent silver, (579°F melting point)
  - 97.5-percent lead, 1.75-percent silver, 0.75-percent tin (590°F melting point)

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall, with the specimens being located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition temperature shift measurements are representative of the vessel at a later time in life. The lead factors for the eight specimen capsule locations based on the reference neutron flux distribution (flux distribution that results in the maximum fluence on the reactor vessel inner surface) vary between approximately 1.8 and 2.3. These lead factors will change over the life of the plant due to changes in core design and operating parameters. Data from CT fracture toughness specimens are expected to provide additional information for use in determining allowable stresses for irradiated material.

Correlations between the calculations and measurements of the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Subsection 5.3.2.6.1. The anticipated degree to which the specimens perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure is made by the use of data on capsules withdrawn. The recommended program schedule for removal of the capsules for post-irradiation testing includes five capsules to be withdrawn instead of four as specified in ASTM E-185 (Reference 1) and Appendix H of 10 CFR 50. The following is the recommended withdrawal schedule of capsules for AP1000.

Capsule	Withdrawal Time
1st	When the accumulated neutron fluence of the capsule is 5×10^{18} n/cm^2.
2nd	When the accumulated neutron fluence of the capsule corresponds to the approximate end of life fluence at the reactor vessel 1/4T location.
3rd	When the accumulated neutron fluence of the capsule corresponds to the approximate end of life fluence at the reactor vessel inner wall location.
4th	When the accumulated neutron fluence of the capsule corresponds to a fluence not less than once or greater than twice the peak end of vessel life fluence.
5th	End of plant design objective of 60 years
6th	Standby
7th	Standby
8th	Standby

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5.3.2.6.1 Measurement of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

The use of passive neutron sensors such as those included in the internal surveillance capsule dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period. An accurate estimate of the average neutron flux level, and hence, time integrated exposure (fluence) experienced by the sensors may be derived from the activation measurements only if the parameters of the irradiation are well known. In particular, the following variables are of interest:

- The measured specific activity of each sensor
- The physical characteristics of each sensor
- The operating history of the reactor
- The energy response of each sensor
- The neutron energy spectrum at the sensor location

The procedures used to determine sensor specific activities, to develop reaction rates for individual sensors from the measured specific activities and the operating history of the reactor, and to derive key fast neutron exposure parameters from the measured reaction rates are described below.

5.3.2.6.1.1 Determination of Sensor Reaction Rates

The specific activity of each of the radiometric sensors is determined using established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor is determined by means of a high purity germanium gamma spectrometer. In the case of the surveillance capsule multiple foil sensor sets, these analyses are performed by direct counting of each of the individual wires or, as in the case of niobium monitors, by appropriate methods as described in ASTM E 1297.

The irradiation history of the reactor over its operating lifetime is determined from plant power generation records. In particular, operating data are extracted on a monthly basis from reactor startup to the end of the capsule irradiation period. For the sensor sets utilized in the surveillance capsule irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation are determined from the following equation:

$$R = \frac{A}{N_0 F Y \sum_j \frac{P_j}{P_{ref}} C_j [1 - e^{-\lambda t_j}] e^{-\lambda t_d}}$$

where:

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A = measured specific activity provided in terms of disintegrations per second per gram of target material (dps/gm).

R = reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} expressed in terms of reactions per second per nucleus of target isotope (rps/nucleus).

N_0 = number of target element atoms per gram of sensor.

F = weight fraction of the target isotope in the sensor material.

Y = number of product atoms produced per reaction.

P_j = average core power level during irradiation period j (MW).

P_{ref} = maximum or reference core power level of the reactor (MW).

C_j = calculated ratio of ϕ(E > 1.0 MeV) during irradiation period j to the time weighted average ϕ(E > 1.0 MeV) over the entire irradiation period.

λ = decay constant of the product isotope (sec-1).

t_j = length of irradiation period j (sec).

t_d = decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the total irradiation period.

In the above equation, the ratio P_j/P_{ref} accounts for month-by-month variation of power level within a given fuel cycle. The ratio C_j is calculated for each fuel cycle and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. Since the neutron flux at the measurement locations within the surveillance capsules is dominated by neutrons produced in the peripheral fuel assemblies, the change in the relative power in these assemblies from fuel cycle to fuel cycle can have a significant impact on the activation of neutron sensors. For a single-cycle irradiation, $C_j = 1.0$. However, for multiple-cycle irradiations, particularly those employing low leakage fuel management, the additional C_j correction must be utilized in order to provide accurate determinations of the decay corrected reaction rates for the dosimeter sets contained in the surveillance capsules.

5.3.2.6.1.2 Least Squares Adjustment Procedure

Least squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as neutron fluence (E > 1.0 MeV) or iron atom displacements (dpa) along with their uncertainties are then easily obtained from the adjusted spectrum. The use of measurements in combination with the analytical results reduces the uncertainty in the calculated spectrum and acts to remove biases that may be present in the analytical technique.

In general, the least squares methods, as applied to pressure vessel fluence evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

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$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, Ri, to a single neutron spectrum, ϕg, through the multigroup dosimeter reaction cross-section, σig, each with an uncertainty δ.

The use of least squares adjustment methods in LWR dosimetry evaluations is not new. The American Society for Testing and Materials (ASTM) has addressed the use of adjustment codes in ASTM Standard E944, “Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance” and many industry workshops have been held to discuss the various applications. For example, the ASTM-EURATOM Symposia on Reactor Dosimetry holds workshops on neutron spectrum unfolding and adjustment techniques at each of its bi-annual conferences.

The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement. The analytical method alone may be deficient because it inherently contains uncertainty due to the input assumptions to the calculation. Typically these assumptions include parameters such as the temperature of the water in the peripheral fuel assemblies, by-pass region, and downcomer regions, component dimensions, and peripheral core source. Industry consensus indicates that the use of calculation alone results in overall uncertainties in the neutron exposure parameters in the range of 15-20% (1σ).

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For a given application, the calculated neutron spectrum is obtained from the results of plant specific neutron transport calculations applicable to the irradiation period experienced by the dosimetry sensor set. This calculation is performed using the benchmarked transport calculational methodology described in Subsection 5.3.2.6.2. The sensor reaction rates are derived from the measured specific activities obtained from the counting laboratory using the specific irradiation history of the sensor set to perform the radioactive decay corrections. The dosimetry reaction cross-sections and uncertainties that are utilized in LWR evaluations comply with ASTM Standard E1018, “Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB).”

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum are input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties also follows the guidance provided in ASTM Standard E 944.

5.3.2.6.2 Calculation of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

A generalized set of guidelines for performing fast neutron exposure calculations within the reactor configuration, and procedures for analyzing measured irradiation sample data that can be correlated to these calculations, has been promulgated by the Nuclear Regulatory Commission (NRC) in

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Regulatory Guide 1.190, or RG-1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” [Reference 2]. Since different calculational models exist and are continuously evolving along with the associated model inputs, e.g., cross-section data, it is worthwhile summarizing the key models, inputs, and procedures that the NRC staff finds acceptable for use in determining fast neutron exposures within the reactor geometry. This material is highlighted in the subsection of material that is provided below.

5.3.2.6.2.1 Calculation and Dosimetry Measurement Procedures

The selection of a particular geometric model, the corresponding input data, and the overall methodology used to determine fast neutron exposures within the reactor geometry are based on the needs for accurately determining a solution to the problem that must be solved and the data/ resources that are currently available to accomplish this task. Based on these constraints, engineering judgment is applied to each problem based on an analyst’s thorough understanding of the problem, detailed knowledge of the plant, and due consideration to the strengths and weaknesses associated with a given calculational model and/or methodology. Based on these conditions, RG-1.190 does not recommend using a singular calculational technique to determine fast neutron exposures. Instead, RG-1.190 suggests that one of the following neutron transport tools be used to perform this work.

- Discrete Ordinates Transport Calculations
  - Adjoint calculations benchmarked to a reference-forward calculation, or stand-alone forward calculations.
  - Various geometrical models utilized with suitable mesh spacing in order to accurately represent the spatial distribution of the material compositions and source.
  - In performing discrete ordinates transport calculations, RG-1.190 also suggests that a P_3 angular decomposition of the scattering cross-sections be used, as a minimum.
  - RG-1.190 also recommends that discrete ordinates transport calculations utilize S_8 angular quadrature, as a minimum.
  - RG-1.190 indicates that the latest version of the Evaluated Nuclear Data File, or ENDF/B, should be used for determining the nuclear cross-sections; however, cross-sections based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked are also acceptable.
- Monte Carlo Transport Calculations

  A complete description of the Westinghouse pressure vessel neutron fluence methodology, which is based on discrete ordinates transport calculations, is provided in Reference 3. The Westinghouse methodology adheres to the guidelines set forth in Regulatory Guide 1.190.

5.3.2.6.2.2 Plant-Specific Calculations

The location, selection, and evaluation of neutron dosimetry and the associated radiometric monitors, as well as fast (E > 1.0 MeV) neutron fluence assessments of the AP1000 reactor pressure vessel, are conducted in accordance with the guidelines that are specified in Regulatory Guide 1.190.

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5.3.2.6.3 Report of Test Results

A summary technical report for each capsule withdrawn with the test results is submitted, as specified in 10 CFR 50.4, within one year of the date of capsule withdrawal unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

The report includes the data required by ASTM E185-82, as specified in paragraph III.B.1 of 10 CFR Part 50, Appendix H, and includes the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

If the test results indicate a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specification is provided with the report.

5.3.2.7 Reactor Vessel Fasteners

The reactor vessel closure studs, nuts, and washers are designed and fabricated in accordance with the requirements of the ASME Code, Section III. The closure studs are fabricated of SA-540. The closure stud material meets the fracture toughness requirements of the ASME Code, Section III, and 10 CFR 50, Appendix G. Conformance with Regulatory Guide 1.65, Materials and Inspections for Reactor Vessel Closure Studs, is discussed in Section 1.9. Nondestructive examinations are performed in accordance with the ASME Code, Section III. See Subsection 5.2.3 for restrictions on lubricants.

Refueling procedures require that the reactor vessel closure studs, nuts, and washers are lifted out of their respective holes and a stud support collar be put in place prior to the lift of the integrated head assembly during preparation for refueling. In this way the studs are lifted with and stored on the head. An alternative method is to remove the reactor vessel closure studs, nuts, and washers from the reactor closure and place them in storage racks during preparation for refueling. In this method, the storage racks are removed from the refueling cavity and stored at convenient locations on the containment operating deck prior to removal of the reactor closure head and refueling cavity flooding. In either case, the reactor closure studs are not exposed to the borated refueling cavity water. Additional protection against the possibility of incurring corrosion effects is provided by the use of a manganese base phosphate surfacing treatment.

The stud holes in the reactor flange are sealed with special plugs before removing the reactor closure, thus preventing leakage of the borated refueling water into the stud holes.

5.3.3 Pressure-Temperature Limits

5.3.3.1 Limit Curves

Heatup and cooldown pressure-temperature limit curves are required as a means of protecting the reactor vessel during startup and shut down to minimize the possibility of fast fracture. The methods outlined in Appendix G of Section III of the ASME Code are employed in the analysis of protection against nonductile failure. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil ductility temperature, which includes a reference nil ductility temperature shift (ΔRT_{NDT}), initial RT_{NDT} and margin. The extent of the RT_{NDT} shift is enhanced by certain chemical elements (such as copper and nickel).

Predicted ΔRT_{NDT} values are derived considering the effect of fluence and copper and nickel content for the reactor vessel steels exposed to 550°F temperature. U.S. NRC Regulatory Guide 1.99 is used in calculating adjusted reference temperature. Since the cold leg temperature exceeds 525°F (minimum steady-state temperature is 535°F at 100% power, thermal design flow, and 10% tube

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plugging), the procedures of Regulatory Guide 1.99 for nominal embrittlement apply. The heatup and cooldown curves are developed considering a sufficient magnitude of radiation embrittlement so that no unirradiated ferritic materials in other components of the reactor coolant system will be limiting in the analysis.

The pressure-temperature curves are developed considering a radiation embrittlement of up to 54 effective full power years (EFPY) consistent with the plant design objective of 60 years with 90 percent availability. Copper, nickel contents and initial RT_{NDT} for materials in the reactor vessel beltline region and the reactor vessel flange and the closure head flange region are shown in Tables 5.3-1 and 5.3-3. The operating curves are developed with the methodology given in Reference 6, which is in accordance with 10 CFR 50, Appendix G.

The curves are applicable up to 54 effective full-power years. These curves, shown in Figures 5.3-2 and 5.3-3, are generic curves for the AP1000 reactor vessel design and they are limiting curves based on copper and nickel material composition (Reference 9). These curves are applicable as long as the following criteria are met:

- 10 CFR 50, Appendix G as related to pressure-temperature remains unchanged,
- Adjusted Reference Temperatures at 1/4T and 3/4T locations remain below the bases of Figures 5.3-2 and 5.3-3

The results of the material surveillance program described in Subsection 5.3.2.6 will be used to verify the validity of ΔRT_{NDT} used in the calculation for the development of heatup and cooldown curves. The projected fluence, copper, and nickel contents along with the RT_{NDT} calculation will be adjusted if necessary, from time to time using the surveillance capsule results. This may require the development of new heatup and cooldown curves.

Higher rates of temperature changes when the reactor coolant system pressure is at or above the operating pressure do not impact the determination of the proper curve to use. Figure 5.3-2 also includes a curve for the leak test limit at steady-state temperature and curves for the criticality limit for nuclear heatup.

Temperature limits for core operation, inservice leak and hydrotests are calculated in accordance with the ASME Code, Section III, Appendix G.

5.3.3.2 Operating Procedures

Plant operating procedures are developed and maintained to prevent exceeding the pressure-temperature limits identified in reactor coolant system pressure and temperature limits report, as required by Technical Specification 5.6.4, during normal and abnormal operating conditions and system tests.

5.3.4 Reactor Vessel Integrity

5.3.4.1 Design

The reactor vessel is the high pressure containment boundary used to support and enclose the reactor core. It provides flow direction with the reactor internals through the core and maintains a volume of coolant around the core. The vessel is cylindrical, with a transition ring, hemispherical bottom head, and removable flanged hemispherical upper head. The vessel is fabricated by welding together the lower head dome, the transition ring, the lower shell, and the upper shell. The upper shell contains the penetrations from the inlet and outlet nozzles and direct vessel injection nozzles. The closure head is fabricated with a head dome and bolting flange. The upper head has

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penetrations for the control rod drive mechanisms, the incore instrumentation, head vent, and support lugs for the integrated head package.

The reactor vessel (including closure head) is approximately 40 feet long and has an inner diameter at the core region of 159 inches. The total weight of the vessel (including closure head and CRDMs) is approximately 417 tons. Surfaces which can become wetted during operation and refueling are clad to a nominal 0.22 inches of thickness with stainless steel welded overlay which includes the upper shell top surface but not the stud holes. The AP1000 reactor vessel's design objective is to withstand the design environment of 2500 psi and 650°F for 60 years. The major factor affecting vessel life is radiation degradation of the lower shell.

As a safety precaution, there are no penetrations below the top of the core. This eliminates the possibility of a loss of coolant accident by leakage from the reactor vessel which could allow the core to be uncovered. The core is positioned as low as possible in the vessel to limit reflood time in an accident. The main radial support system of the lower end of the reactor internals is accomplished by key and keyway joints to the vessel wall. At equally spaced points around the circumference, a clevis block is located on the reactor vessel inner diameter. A permanent cavity seal ring is attached to the top of the vessel shell for welding to the refueling cavity liner. To decrease outage time during refueling, access to the stud holes is provided to allow stud hole plugging with the head in place. By the use of a ring forging with an integral flange, the number of welds is minimized to decrease inservice inspection time.

The lower head has an approximate 6.5 feet inner spherical radius. The lower radial supports are located on the head at the elevation of the lower internals lower core support plate. The transition ring is welded to the lower shell course with the weld located outside the highest fluence area of the beltline region of the reactor vessel. The lower shell is a ring forging about 8 inches thick with an inner diameter of 159 inches. The length of the shell is greater than 168 inches to place the upper shell weld outside of the active fuel region. The upper shell is a large ring forging. Included in this forging are four 22-inch inner diameter inlet nozzles, two 31-inch inner diameter outlet nozzles and two 6.81-inch inner diameter direct vessel injection nozzles (8-inch schedule 160 pipe connections). These nozzles are forged into the ring or are fabricated by "set in" construction. The inlet and outlet nozzles are offset axially in different planes by 17.5 inches. The injection nozzles are 100 inches down from the main flange and the outlet nozzles are 80 inches down and the inlet nozzles are 62.5 inches below the mating surface.

The closure head has a 77.5-inch inner spherical radius and a 188.0-inch O.D. outer flange. Cladding is extended across the bottom of the flange for refueling purposes. Forty-five, seven-inch diameter studs attach the head to the lower vessel (which consists of the lower shell, transition ring, and bottom head) and two metal o-rings are used for sealing. The upper head has sixty-nine 4-inch outer diameter penetrations for the control rod drive mechanism housings and eight penetrations for the incore instrumentation tubes.

The eight penetrations for the incore instrumentation tubes are Quickloc instrument nozzles, which are welded to the reactor vessel head. Up to six instrument thimble assemblies pass through each Quickloc instrument nozzle. The reactor vessel head penetration diameter is approximately 4.75 inches to the cladding. The material of the pressure boundary parts of the Quickloc are SA-182 – Type F304, SA-479 – Type 304 and Type 316, and UNS S21800. The Quickloc instrument nozzle is welded to the Ni-Cr-Fe buttering on the weld buildup on the reactor vessel closure head. The Quickloc provides two pressure boundaries: 1) between the Quickloc plug and the incore instrument thimble assemblies (this pressure boundary is disassembled only when the instrument thimble is replaced) and 2) between the Quickloc plug and the instrument nozzle (this pressure boundary is disassembled when the reactor vessel closure head is removed). The Quickloc instrument nozzle pressure boundary parts are designed and fabricated to the requirements of the ASME Code, Section III Division 1, Subsection NB. The Quickloc internal, non-pressure boundary parts are

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designed to Subsection NG. The Quickloc assembly is a proven design, which was first installed in an operating plant in 1995. Since then, it has been installed on four additional operating plants.

The vessel is manufactured from low alloy steel plates and forgings to minimize size. The chemical content of the core region base material is specifically controlled. A surveillance program is used to monitor the radiation damage to the vessel material.

The four vessel supports are located beneath the inlet nozzles and the internals support ledge is machined into the top of the upper shell. The top of the upper shell contains the stud holes and has the sealing surface for the closure head. Inner and outer monitor tubes are provided through the shell to collect any leakage past the closure region o-rings.

The reactor vessel is designed and fabricated in accordance with the quality standards set forth in 10 CFR 50, General Design Criteria 1, 14, 30, and 31, and 50.55a; and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section III. Principal design parameters of the reactor vessel are given in Table 5.3-5. The vessel design and construction enables inspection in accordance with the ASME Code, Section XI.

Cyclic loads are introduced by normal power changes, reactor trips, and startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the design life. Thermal stratification during passive core cooling system operation and natural circulation cooldown is considered by performing a thermal/flow analysis using computational fluid dynamics techniques. This analysis includes thermally-induced fluid buoyancy, heat transfer between the coolant and the metal of the vessel and internals and uses thermal/flow boundary conditions based on an existing thermal/hydraulic transient analysis of the primary reactor coolant system. This analysis provides temperature maps that are used to evaluate thermal stresses.

Analysis proves that the vessel is in compliance with the fatigue and stress limits of the ASME Code, Section III. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heatup and cooldown rates imposed by plant operating limits are 100°F per hour for normal operations.

5.3.4.2 Materials of Construction

The materials used in the fabrication of the reactor vessel are discussed in Subsection 5.2.3.

5.3.4.3 Fabrication Methods

The fabrication methods used in the construction of the reactor vessel are discussed in Subsection 5.3.2.2.

5.3.4.4 Inspection Requirements

The nondestructive examinations performed on the reactor vessel are described in Subsection 5.3.2.3.

5.3.4.5 Shipment and Installation

The reactor vessel is shipped in a horizontal position on a shipping skid with a vessel-lifting truss assembly. All vessel openings are sealed to prevent the entrance of moisture, and an adequate quantity of desiccant bags is placed inside the vessel. These are usually placed in a wire mesh basket attached to the vessel cover. All carbon steel surfaces are protected with a temporary protective covering before shipment.

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The closure head is also shipped with a shipping cover and skid. An enclosure attached to the reactor vessel closure head flange protects the control rod mechanism housings. All head openings are sealed to prevent the entrance of moisture, and an adequate quantity of desiccant bags is placed inside the head. These are placed in a wire mesh basket attached to the head cover. All carbon steel surfaces are protected with a temporary protective covering before shipment.

5.3.4.6 Operating Conditions

Operating limitations for the reactor vessel are presented in Subsection 5.3.3 and in the technical specifications.

In addition to the analysis of primary components discussed in Subsection 3.9.1.4, the reactor vessel is further qualified to ensure against unstable crack growth under faulted conditions. Safeguard actuation following a loss-of-coolant, tube rupture or other similar emergency or faulted event produces relatively high thermal stresses in regions of the reactor vessel which come into contact with water from the passive core cooling system. Primary consideration is given to these areas, including the reactor vessel beltline region and the reactor vessel primary coolant nozzles, to ensure the integrity of the reactor vessel under these severe postulated transients. TMI Action Item II.K.2.13, is satisfied upon submittal of RT_{NDT} values which are below the pressurized thermal shock (PTS) rule screening values. The results given in Table 5.3-3 show that the issue is resolved.

For the beltline region, the NRC staff concluded that conservatively calculated screening criterion values of RT_{NDT} less than 270°F for plate material and axial welds, and less than 300°F for circumferential welds, present an acceptably low risk of vessel failure from pressurized thermal shock events. These values were chosen as the screening criterion in the pressurized thermal shock rule for 10 CFR 50.34 (new plants) and 10 CFR 50.61 (operating plants). The conservative methods chosen by the NRC staff for the calculation of RT_{PTS} for the purpose of comparison with the screening criterion is presented in paragraph (b)(2) of 10 CFR 50.61. Details of the analysis method and the basis for the pressurized thermal shock rule can be found in SECY-82-465 (Reference 4).

The revised pressurized thermal shock rule, (10 CFR 50.61), effective June 14, 1991 makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99.

The reactor vessel beltline materials are specified in Subsection 5.3.2. Evaluation of the AP1000 reactor vessel material showed that even at the fluence level which results in the highest RT_{PTS} value, this value is well below the screening criteria of 270°F. RT_{PTS} is RT_{NDT}, the reference nil ductility transition temperature as calculated by the method chosen by the NRC staff as presented in paragraph (b)(2) of 10 CFR 50.61, and the pressurized thermal shock rule. The pressurized thermal shock rule states that this method of calculating RT_{PTS} should be used in reporting values used to compare pressurized thermal shock to the above screening criterion set in the pressurized thermal shock rule. The screening criteria will not be exceeded using the method of calculation prescribed by the pressurized thermal shock rule for the vessel design objective. The material properties, and initial RT_{NDT} and end-of-life RT_{PTS} requirements and predictions are in Tables 5.3-1 and 5.3-3. The materials that are exposed to high fluence levels at the beltline region of the reactor vessel are subject to the pressurized thermal shock rule. These materials are a subset of the reactor vessel materials identified in Subsection 5.3.2.

The principles and procedures of linear elastic fracture mechanics (LEFM) are used to evaluate thermal effects in the regions of interest. The linear elastic fracture mechanics approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in linear elastic fracture mechanics is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

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5.3.4.7 Inservice Surveillance

The internal surfaces of the reactor vessel are accessible for periodic inspection. Visual and/or nondestructive techniques are used. During refueling, the vessel cladding is capable of being inspected in certain areas of the upper shell above the primary coolant inlet nozzles, and if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

Reactor vessel closure head examination is performed periodically as required by 10 CFR 50.55a. Optical devices permit a selective inspection of the cladding, control rod drive mechanism nozzles, and the gasket seating surface. Access to the top head surface is provided by 7 ports around the circumference of the integrated head package shroud and by multiple removable insulation panels, which interface with the head under the integrated head package shroud. Both the ports and the insulation panels provide access to the bare vessel head, and CRDM and instrumentation penetrations for use of a remote, mobile visual inspection manipulator to perform a 360° inspection around each penetration. The head insulation is a stand-off design with a minimum offset from the head surface of 3 inches.

The knuckle transition piece, which is the area of highest stress of the closure head, is accessible on the outer surface for visual inspection, dye penetrant or magnetic particle testing, and ultrasonic testing. The closure studs and nuts can be inspected periodically using visual, magnetic particle, and ultrasonic techniques.

The closure studs, nuts, washers, and the vessel flange seal surface, as well as the full-penetration welds in the following areas of the installed reactor vessel, are available for nondestructive examination:

- Vessel shell, from the inside surface.
- Primary coolant nozzles, from the inside surface. Only partial outside diameter coverage is provided.
- Closure head, from the inside surface; bottom head, from the inside surface.
- Field welds between the reactor vessel nozzle safe ends and the main coolant piping, from the inside surface.

The design considerations which have been incorporated into the system design to permit the above inspection are as follows:

- Reactor internals are completely removable. The tools and storage space required to permit removal of the reactor internals are provided.
- The closure head is stored on a stand on the reactor operating deck during refueling to facilitate direct visual inspection.
- Reactor vessel studs, nuts, and washers can be removed to dry storage during refueling.
- Access is provided to the reactor vessel nozzle safe ends. The insulation covering the nozzle-to-pipe welds may be removed.

Because radiation levels and remote underwater accessibility limits access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation

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for the periodic nondestructive tests which are required by the ASME Code inservice inspection requirements. These are as follows:

- Shop ultrasonic examinations are performed on internally clad surfaces to an acceptance and repair standard to provide an adequate cladding bond to allow later ultrasonic testing of the base metal from the inside surface. The size of cladding bond defect allowed is 0.25 inch by 0.75 inch with the greater direction parallel to the weld in the region bounded by 2T (T = wall thickness) on both sides of each full-penetration pressure boundary weld. Unbounded areas exceeding 0.442 square inches (0.75-inch diameter) in other regions are rejected.

- The design of the reactor vessel shell is an uncluttered cylindrical surface to permit future positioning of the test equipment without obstruction.

- The weld-deposited clad surface on both sides of the welds to be inspected is specifically prepared to ensure meaningful ultrasonic examinations.

- During fabrication, full-penetration ferritic pressure boundary welds are ultrasonically examined in addition to code examinations.

- After the shop hydrostatic testing, full-penetration ferritic pressure boundary welds (with the exception of the closure head welds), as well as the nozzles to safe end welds, are ultrasonically examined from both the inside and outside diameters in addition to ASME Code, Section III requirements.

- Preservice examinations for the closure head will include a baseline top-of-the head visual examination; ultrasonic examinations of the inside diameter surface of each vessel head penetration; eddy current examinations of the surface of head penetration welds, the outside diameter surface of the vessel penetrations, and the inside diameter surface of the penetrations; and post-hydro liquid penetrant examinations of accessible surfaces that have undergone preservice inspection eddy current examinations.

The vessel design and construction enables inspection in accordance with the ASME Code, Section XI.

5.3.5 Reactor Vessel Insulation

5.3.5.1 Reactor Vessel Insulation Design Bases

Reactor vessel insulation is provided to minimize heat losses from the primary system. Nonsafety-related reflective (or suitable equivalent, as described within Subsection 6.3.2.2.7) insulation similar to that in use in current pressurized water reactors is utilized. The AP1000 reactor vessel insulation contains design features to promote in-vessel retention following severe accidents. In the unlikely event of a beyond design basis accident, the reactor cavity is flooded with water, and the reactor vessel insulation allows heat removal from core debris via boiling on the outside surface of the reactor vessel. The reactor vessel insulation permits a water layer next to the reactor vessel to promote heat transfer from the reactor vessel. This is accomplished by providing:

- A means of allowing water free access to the region between the reactor vessel and insulation.

- A means to allow steam generated by water contact with the reactor vessel to escape from the region surrounding the reactor vessel.

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- The insulation support frame and the insulation panels form a structurally reliable flowpath for the water and steam.

The reactor vessel insulation and its supports are designed to withstand bounding pressure differentials across the reactor vessel insulation panels during the period that the reactor vessel is externally flooded with water and the core heat is removed from the vessel wall by water and generated steam is vented. This is accomplished by providing steam vents with a minimum flow area of 12 ft^2 from the vessel insulation annular space. The flow path from the reactor loop compartment to the reactor cavity provides an open flow path for water to flood the reactor cavity. The reactor vessel insulation water inlets are designed to minimize the pressure drop during ex-vessel cooling to permit water inflow to cool the vessel.

5.3.5.2 Description of Insulation

A schematic of the reactor vessel, the vessel insulation and the reactor cavity is shown in Figure 5.3-7. The insulation is mounted on a structural frame that is supported from the wall and floor of the reactor cavity. The insulation panels are designed to have a minimum gap between the insulation and reactor vessel not less than 2 inches when subjected to the dynamic loads in the direction towards the vessel that result during ex-vessel cooling.

The bottom portion of the vessel insulation is constructed to provide a flow channel conducive for heat removal.

The structural frame supporting the insulation is designed to withstand the bounding severe accident loads while maintaining the flow path. The fasteners holding the insulation panels to the frame are also designed for these loads.

At the bottom of the insulation are water inlet assemblies. Each water inlet assembly is normally closed to prevent an air circulation path through the vessel insulation. The inlet assemblies are self-actuating passive devices. The inlet assemblies open when the cavity is filled with water. This permits ingress of water during a severe accident, while preventing excessive heat loss during normal operation.

The total flow area of the water inlet assemblies has sufficient margin to preclude significant pressure drop during ex-vessel cooling during a severe accident. The minimum total flow area for the water inlets assemblies is 6 ft^2. Due to the relatively low approach velocities in the flow paths leading to the reactor cavity, the grating over the vertical access tunnel, the design of the doorway between the reactor coolant drain tank compartment and the reactor cavity, the low flow velocities approaching the water inlet assemblies, and the relatively large minimum flow area through each water inlet assembly, at least 7 in^2, the water inlet assemblies and the steam flow path are not susceptible to clogging from debris inside containment.

Multiple steam vents in the nozzle gallery provide a flow path for the steam/water within the reactor vessel/insulation annular space to flow back to the containment flood-up region. The steam vents provide 12 ft^2 minimum flow area for steam/water to exit the annular space. Each of the steam vents has a door that will be opened by the steam/water flow generated under the insulation with the cavity filled with water, but which remains in place when only normal air cooling flow is operating.

Extensive maintenance of the vessel insulation is not normally required. Periodic verification of the vessel insulation moving parts can be performed during refueling outages.

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5.3.5.3 Description of External Vessel Cooling Flooded Compartments

Ex-vessel cooling during a severe accident is provided by flooding the reactor coolant system loop compartment including the vertical access tunnel, the reactor coolant drain tank room, and the reactor cavity. Water from these compartments replenishes the water that comes in contact with the reactor vessel and is boiled and vented to containment. The opening between the vertical access tunnel and the reactor coolant drain tank room is approximately 100 ft^2. Removable steel grating is provided over the inlet to the vertical access tunnel to restrict access to the lower compartments. This grating precludes large debris from being transported into the reactor cavity during ex-vessel cooling scenarios. Figure 5.3-8 depicts the flooded compartments that provide the water for ex-vessel cooling. The doorway between the reactor cavity compartment and the reactor coolant drain tank room consists of a normally closed door and a damper above the door. The door and damper arrangement, shown in Figure 5.3-9, maintains the proper air flow through the reactor cavity during normal operation. The damper prevents air from flowing into the reactor coolant drain tank compartment, but opens to permit flooding of the reactor cavity from the reactor coolant drain tank compartment. The damper opening has a minimum flow area of 8 ft^2 and is not susceptible to clogging from debris that can pass through the grating over the inlet to the vertical access tunnel. It is constructed of light-weight material to minimize the force necessary to open the damper and permit flooding and continued water flow through the opening during ex-vessel cooling. The damper provides an acceptable pressure drop through the opening during ex-vessel cooling.

Subsection 6.3.2.1.3 discusses post-accident operation of the passive core cooling system, which operates to flood the reactor cavity following an accident. Subsection 9.1.3 discusses the connections from the refueling cavity to the steam generator compartment that facilitate flooding of the reactor cavity following an accident.

5.3.5.4 Determination of Forces on Insulation and Support System

The forces that may be expected in the reactor cavity region of the AP1000 plant during a core damage accident in which the core has relocated to the lower head and the reactor cavity is reflooded can be based on test results from the ULPU test program (Reference 5). The particular configuration (Configuration V) reviewed closely models the full-scale AP1000 geometry of water in the region near the reactor vessel, between the reactor vessel and the reactor vessel insulation. The ULPU tests provide data on the pressure generated in the region between the reactor vessel and reactor vessel insulation. These data, along with observations and conclusions from heat transfer studies, are used to develop the functional requirements with respect to in-vessel retention for the reactor vessel insulation and support system. Interpretation of data collected from ULPU Configuration V experiments in conjunction with the static head of water that would be present in the AP1000 is used to estimate forces acting on the rigid sections of insulation. The ULPU V test results indicate that the pressure variations in the flow channel between the vessel and the insulation are on the order of plus/minus 0.5 meters of water. Fast Fourier Transform analysis of the ULPU V pressure data is also included in the ULPU V test report. This analysis shows that the dominant frequency of the pressure variations is less than about 2 Hz. The natural frequency of the insulation structure is expected to be well above 2 Hz.

5.3.5.5 Design Evaluation

A structural analysis of the AP1000 reactor cavity insulation system was performed that demonstrates that it meets the functional requirements discussed above. The analysis encompasses the insulation and support system and includes a determination of the stresses in support members, bolts, insulation panels and welds, as well as deflection of support members and insulation panels.

Loads on the insulation and the support structure include hydrostatic loads and dynamic loads from boiling. These loads are of the same order as those analyzed for AP600, and the results of the

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AP1000 analysis show that the insulation is able to meet its functional requirements. The reactor vessel insulation provides an engineered pathway for water-cooling the vessel and for venting steam from the reactor cavity. These results were also compared to the available test data.

The reactor vessel insulation is purchased equipment. The purchase specification for the reactor vessel insulation design required confirmatory static load analyses.

5.3.6 Combined License Information

5.3.6.1 Pressure-Temperature Limit Curves

The pressure-temperature curves shown in Figures 5.3-2 and 5.3-3 are generic curves for AP1000 reactor vessel design, and they are the limiting curves based on copper and nickel material composition. However, for a specific AP1000, these curves will be plotted based on material composition of copper and nickel. Use of plant-specific curves will be addressed by the Combined License holder during procurement and fabrication of the reactor vessel. As noted in the bases to Technical Specification 3.4.14, use of plant-specific curves requires evaluation of the LTOP analysis. This includes an evaluation of the setpoint pressures for the RNS relief valves by the Combined License holder to determine if the setpoint pressures need to be changed based on the plant-specific pressure-temperature curves. The development of the plant-specific curves and evaluation of the setpoint pressures are required prior to fuel load.

5.3.6.2 Reactor Vessel Materials Surveillance Program

The reactor vessel reactor material surveillance program is addressed in Subsections 5.3.2.6 and 5.3.2.6.3.

5.3.6.3 Surveillance Capsule Lead Factor and Azimuthal Location Confirmation

The surveillance capsule lead factors and azimuthal locations are addressed in APP-GW-GLR-023 (Reference 7), and the applicable changes are incorporated.

5.3.6.4 Reactor Vessel Materials Properties Verification

5.3.6.4.1 The verification of plant-specific belt line material properties consistent with the requirements in Subsection 5.3.3.1 and Tables 5.3-1 and 5.3-3 will be completed prior to fuel load. The verification will include a pressurized thermal shock evaluation based on as procured reactor vessel material data and the projected neutron fluence for the plant design objective of 60 years. This evaluation report will be submitted for NRC staff review.

5.3.6.4.2 The structural analysis of the AP1000 reactor vessel insulation and support structure is addressed in APP-GW-GLR-060 (Reference 8), and the applicable changes are incorporated.

5.3.6.5 Reactor Vessel Insulation

The verification that the reactor vessel insulation is consistent with the design bases established for in-vessel retention is addressed in Reference 8 (APP-GW-GLR-060).

5.3.6.6 Inservice Inspection of Reactor Vessel Head Weld Buildup

The inservice inspection program of the weld buildup on the reactor vessel head for the instrumentation penetrations (Quickloc) is addressed in Subsection 5.2.4.1.

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5.3.7 References

1. ASTM E-185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.”

2. Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” United States Nuclear Regulatory Commission, Office of Nuclear Reactor Research, March, 2001.

3. WCAP-15557, “Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology,” S. L. Anderson, August 2000.

4. NRC Policy Issue, “Pressurized Thermal Shock,” SECY-82-465, November 23, 1982.

5. Theofanous, T.G., et al., “Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility,” CRSS-03/06, June 2003.

6. WCAP-14040-NP-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004.

7. APP-GW-GLR-023, “Surveillance Capsule Lead Factor and Azimuthal Location Confirmation,” Westinghouse Electric Company LLC.

8. APP-GW-GLR-060, “Reactor Vessel Insulation System – Verification of In-Vessel Retention Design Bases,” Westinghouse Electric Company LLC.

9. APP-RXS-Z0R-001, Revision 3, “AP1000 Generic Pressure Temperature Limits Report,” April 2018.

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Table 5.3-1
Maximum Limits for Elements of the Reactor Vessel

Element	Beltline Forging (percent)	As Deposited Weld Metal (percent)
Copper	0.06	0.06
Phosphorus	0.01	0.01
Vanadium	0.05	0.05
Sulfur	0.01	0.01
Nickel	0.85	0.85

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Table 5.3-2
Reactor Vessel Quality Assurance Program

	RT[a]	UT[a]	PT[a]	MT[a][b]
Forgings				
Flanges		Yes		Yes
Studs and nuts		Yes		Yes
CRDM latch housing nozzle		Yes	Yes	
Quickloc instrument nozzle (QIN)		Yes	Yes	
Main nozzles		Yes		Yes
Nozzle safe ends		Yes	Yes	
Shell sections		Yes		Yes
Heads		Yes		Yes
Plates		Yes		Yes
Weldments				
Head and shell	Yes	Yes		Yes
CRDM latch housing nozzle to closure head connection			Yes	
QIN to QIN weld buildup connection	Yes	Yes	Yes	
QIN weld buildup		Yes		Yes
Main nozzle	Yes	Yes		Yes
Cladding		Yes	Yes	
Nozzle to safe ends	Yes	Yes	Yes	
All full-penetration ferritic pressure boundary welds accessible after hydrotest		Yes		Yes
Full-penetration nonferritic pressure boundary welds accessible after hydrotest a. Nozzle to safe ends		Yes	Yes	
Seal ledge				Yes
Head lift lugs				Yes
Core pad welds			Yes	
Flow skirt support lugs weld buildup		Yes	Yes	

Notes:

a. RT - Radiographic
UT - Ultrasonic
PT - Dye penetrant
MT - Magnetic particle

b. Dye penetrant (PT) examination may be used in lieu of magnetic particle (MT) examination in accordance with Subsection 5.3.2.3.3.

Base metal weld repairs as a result of UT, MT, RT, and/or PT indications are cleared by the same nondestructive examination technique/procedure by which the indications were found. The repairs meet applicable Section III requirements.

In addition, UT examination in accordance with the in process/posthydro UT requirements is performed on base metal repairs in the core region and base metal repairs in the inservice inspection zone (1/2 T).

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Table 5.3-3
End-of-Life RT_{NDT} and Upper Shelf Energy Projections

	Unirradiated		End-of-life (54 EFPY)	
	RT_{NDT} (°F)	USE (ft-lb)	USE (ft-lb) 1/4T	RT_{PTS} (°F)
Beltline Forging	-10	> 75	> 50	< 270(2)
Head	10	N/A	N/A	N/A
Flange	10	N/A	N/A	N/A
Weld	10	N/A	N/A	N/A
Beltline Weld	-20	> 75	> 50	< 300(2)

Notes:
1. The minimum unirradiated upper shelf energy for beltline base metal is for the transverse direction.
2. End-of-Life RT_{PTS} requirements shown. End-of-Life RT_{PTS} (also equals RT_{NDT}) will be determined for as-built material. The preliminary RT_{PTS} for the AP1000 reactor vessel beltline forging and beltline weld are 94°F and 148°F, respectively.

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Table 5.3-4
Reactor Vessel Material Surveillance Program

Capsules S, T, U, V, W, X, Y, and Z			
Material	**Charpy Specimens**	**Tensile Specimens**	**1/2T-CT Specimens**
Forging (tangential)	15	3	2
Forging (axial)	15	3	2
Weld Metal	15	3	2
Heat Affected Zone (HAZ)	15	–	–

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Table 5.3-5
Reactor Vessel Design Parameters

(approximate values)	
Design pressure (psig)	2485
Design temperature (°F)	650
Overall height of vessel and closure head, bottom head outside diameter to the control rod drive mechanism rod position indicator mating surface	45-9
Number of reactor closure head studs	45
Diameter of reactor closure head/studs, (in.)	7
Outside diameter of closure head flange (in.)	188
Inside diameter of flange (in.)	148.81
Outside diameter at shell (in.)	176
Inside diameter at shell (in.)	159
Inlet nozzle inside diameter (in.)	22
Outlet nozzle inside diameter (in.)	31
Clad thickness, nominal (in.)	0.22
Lower head thickness, minimum (in.)	6
Vessel beltline thickness, minimum (in.)	8
Closure head thickness (in.)	6.25

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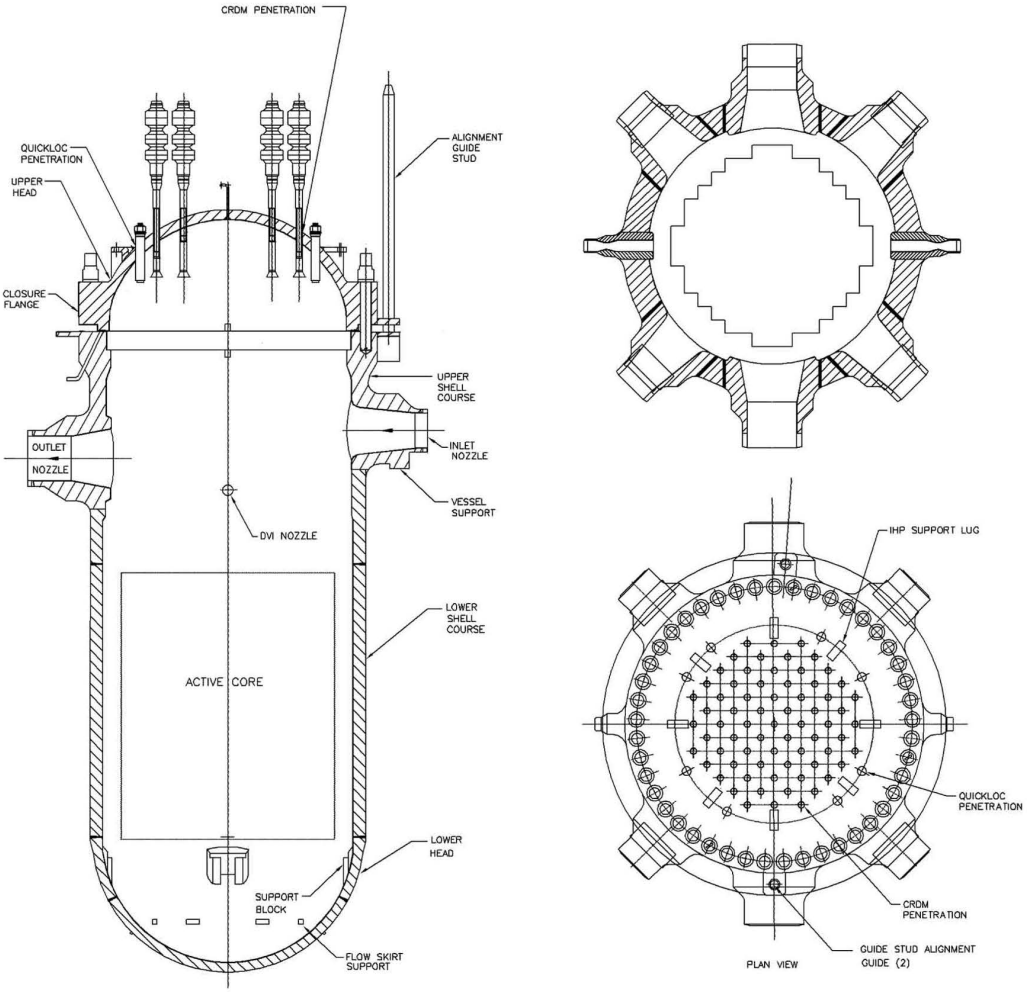

Figure 5.3-1
Reactor Vessel

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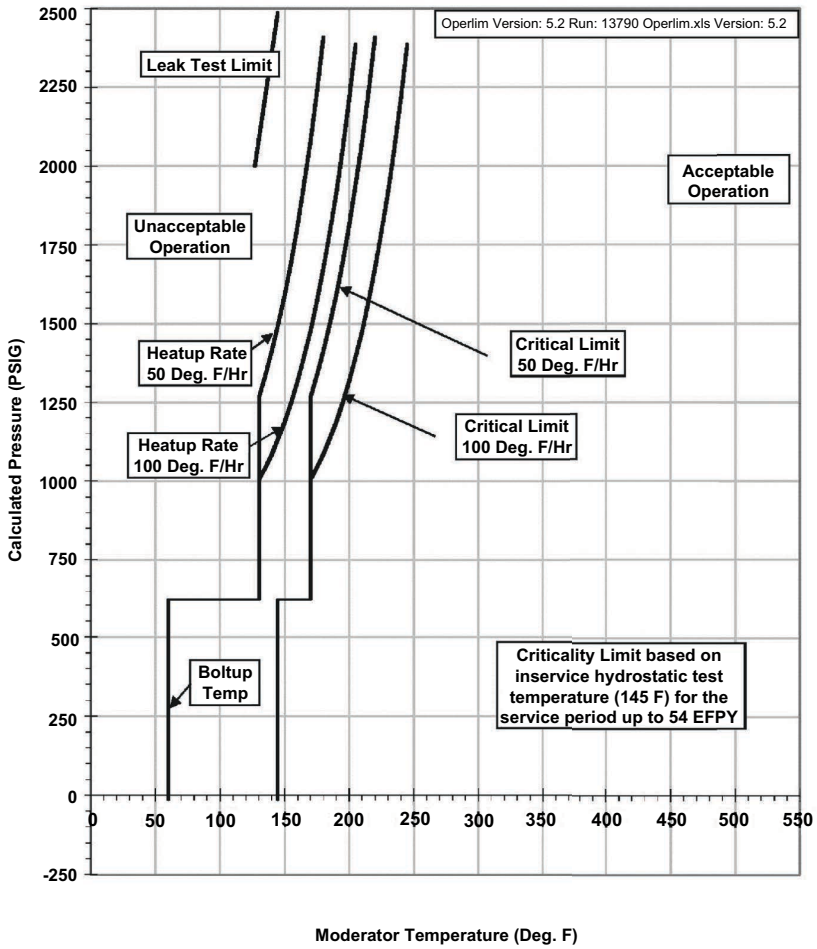

Figure 5.3-2
AP1000 Reactor Coolant System Heatup Limitations (Heatup Rate Up to 50° and 100°F/hour) Representative for the First 54 EFPY (Without Margins for Instrumentation Errors)

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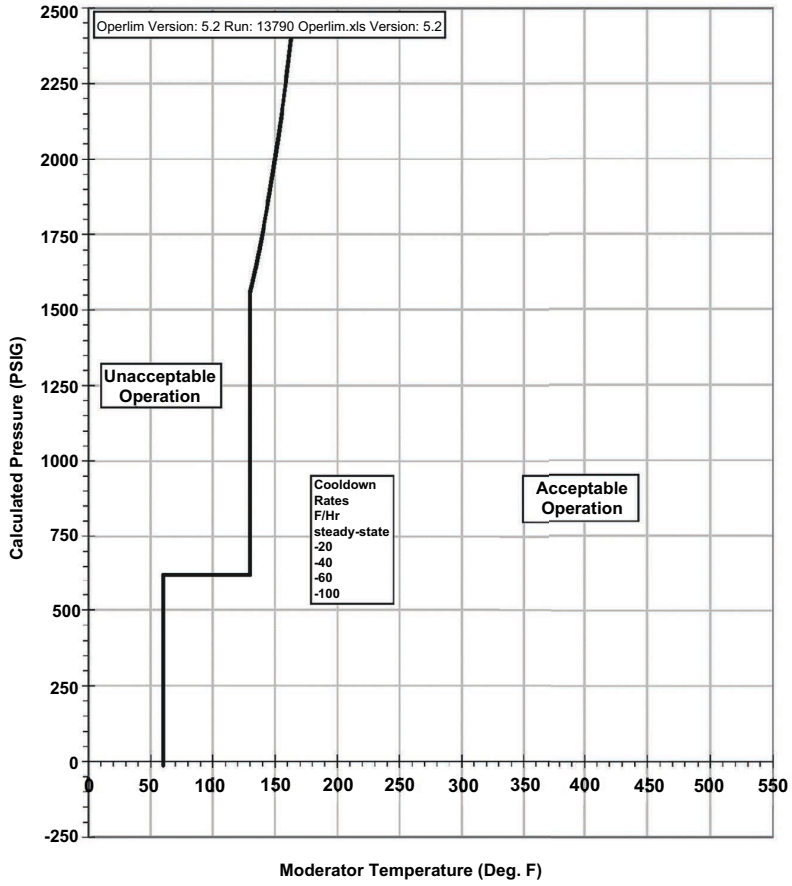

Figure 5.3-3
AP1000 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 50° and 100°F/hour) Representative for the First 54 EFPY (Without Margins for Instrumentation Errors)

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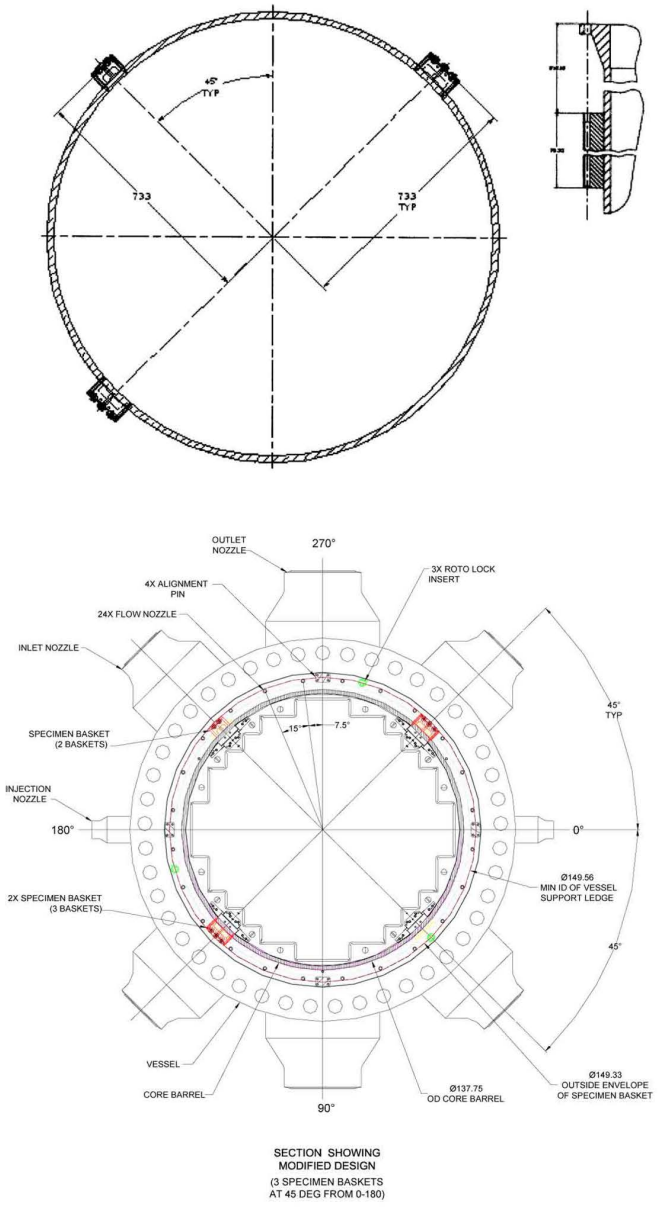

Figure 5.3-4
AP1000 Reactor Vessel Surveillance Capsules Locations

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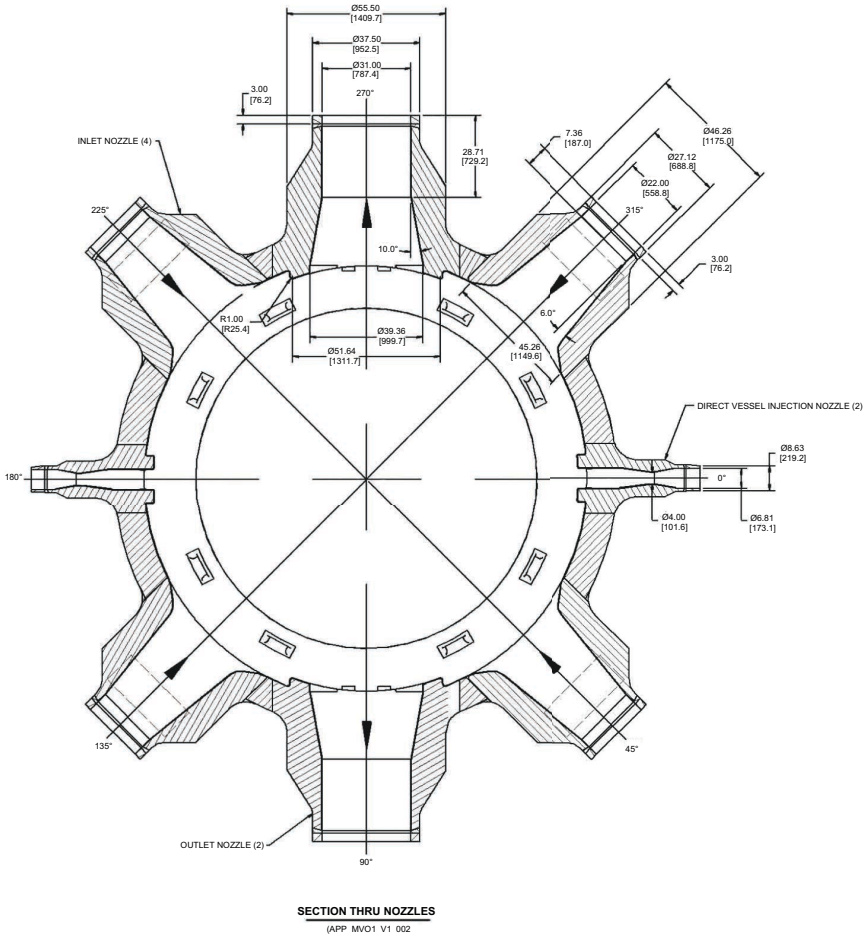

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Figure 5.3-5
Reactor Vessel Key Dimensions
Plan View

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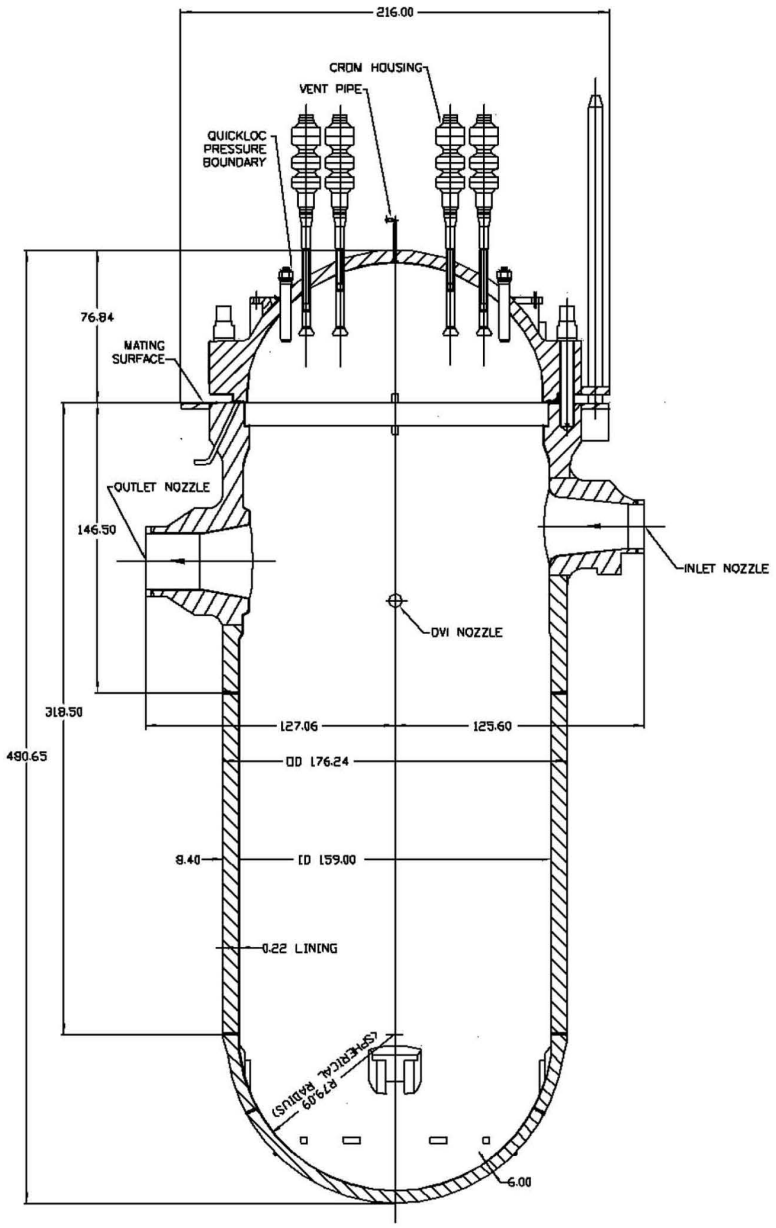

Figure 5.3-6
Reactor Vessel Key Dimensions, Side View

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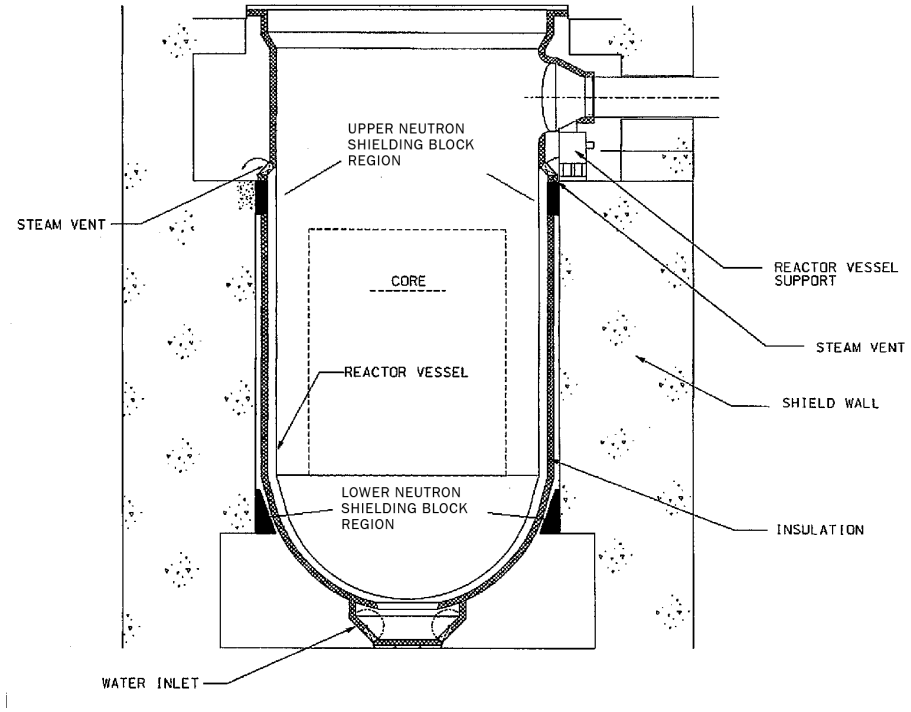

Figure 5.3-7
Schematic of Reactor Vessel Insulation

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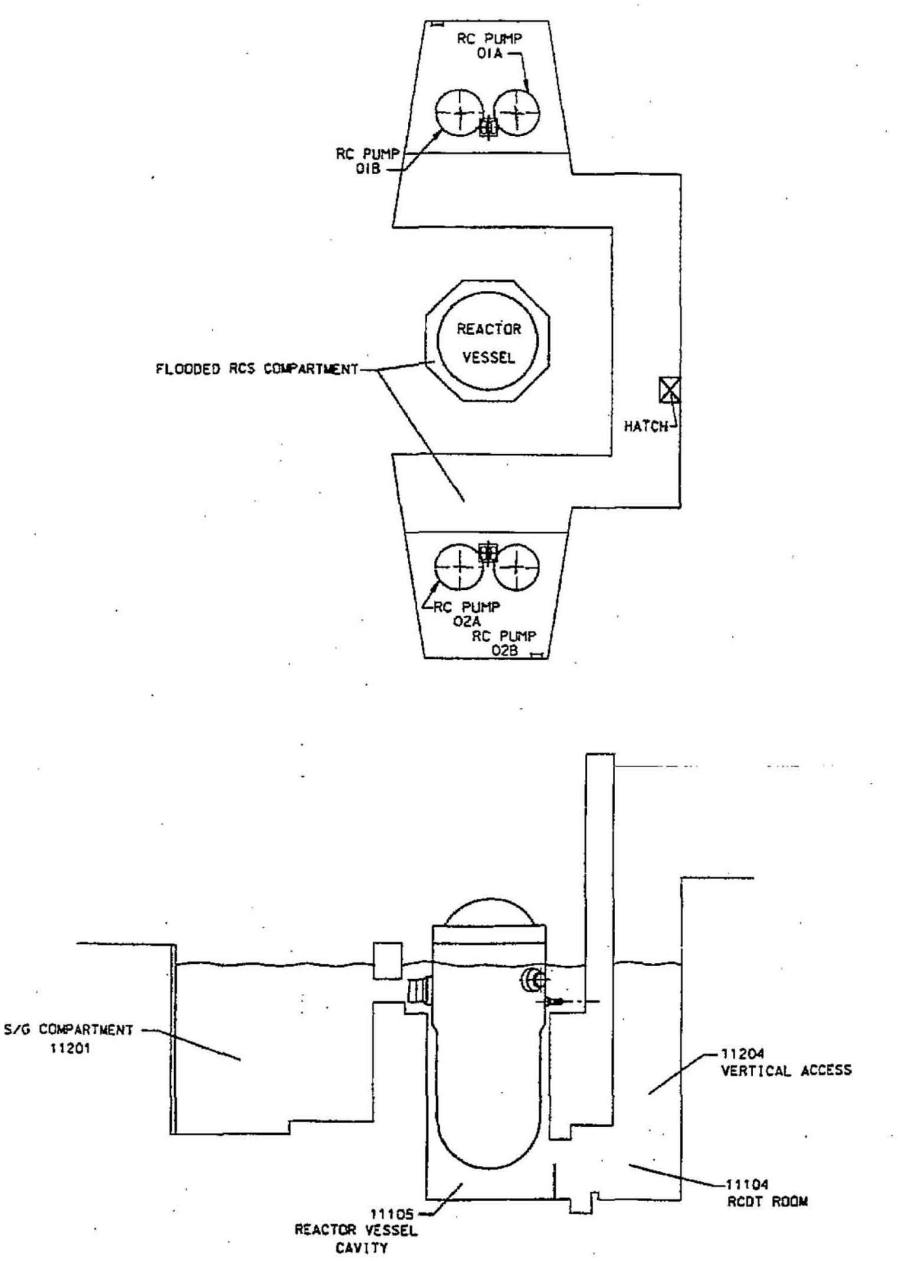

Figure 5.3-8
RCS Flooded Compartments During Ex-Vessel Cooling

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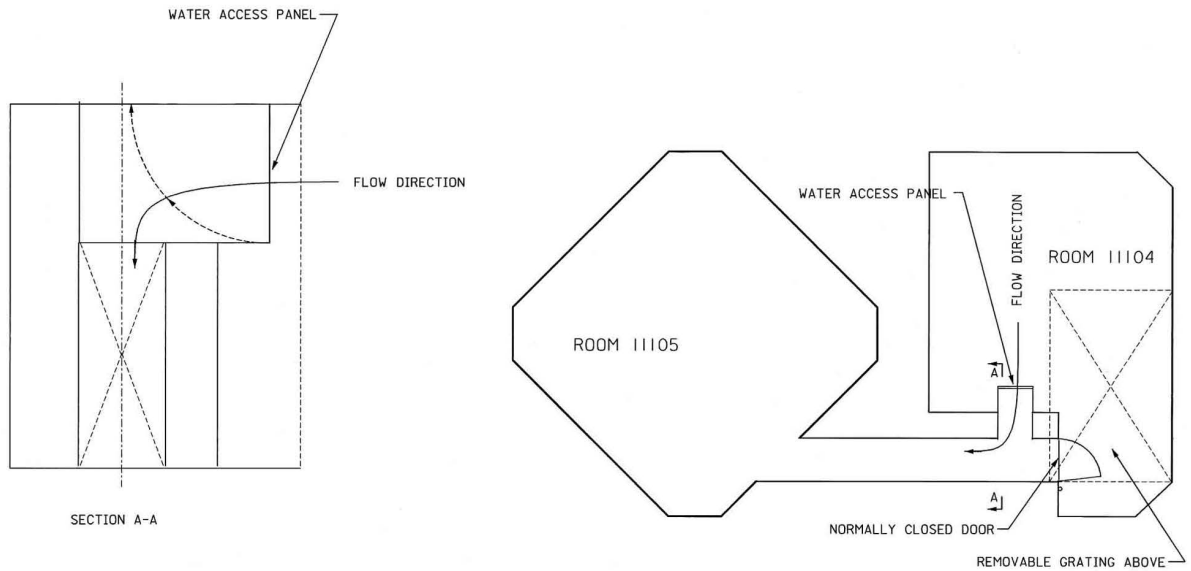

Figure 5.3-9
Door Between RCDT Room and Reactor Cavity Compartment

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5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pump Assembly

5.4.1.1 Design Bases

The reactor coolant pump (RCP) is an integral part of the reactor coolant pressure boundary. It is designed, fabricated, erected, and tested to quality standards consistent with the requirements set forth in 10 CFR 50, 50.55a and General Design Criterion 1. The reactor coolant pump casing and stator shell provide a barrier to the release of reactor coolant and other radioactive materials to the containment atmosphere.

The reactor coolant pump provides an adequate core cooling flow rate for sufficient heat transfer to maintain a departure from nucleate boiling ratio (DNBR) greater than the limit established in the safety analysis. Pump assembly rotational inertia is provided by a flywheel (inside the pump pressure boundary) motor rotor, and other rotating parts. This rotational inertia provides flow during coastdown conditions. This forced flow following an assumed loss of electrical power and the subsequent natural circulation effect in the reactor coolant system (RCS) adequately cools the core. The net positive suction head (NPSH) required for operation is by conservative pump design always less than that available by system design and operation.

The reactor coolant pump pressure boundary shields the balance of the reactor coolant pressure boundary from theoretical worst-case flywheel failures. The reactor coolant pump pressure boundary is analyzed to demonstrate that a fractured flywheel cannot breach the reactor coolant system boundary (impacted pressure boundary components are stator closure, stator main flange, and lower stator flange) and impair the operation of safety-related systems or components. This meets the requirements of General Design Criteria 4. The reactor coolant pump flywheel is designed, manufactured, and inspected to minimize the potential for the generation of high-energy fragments (missiles) under any anticipated operating or accident condition consistent with the intent of the guidelines set forth in Standard Review Plan Subsection 5.4.1.1 and Regulatory Guide 1.14. Each flywheel is tested at an overspeed condition to verify the flywheel design and construction.

5.4.1.2 Pump Assembly Description

5.4.1.2.1 Design Description

[*The reactor coolant pump is a single-stage, hermetically sealed, high-inertia, centrifugal sealless pump of canned motor design.*]* It pumps large volumes of reactor coolant at high pressures and temperature. Figure 5.4-1 shows a reactor coolant pump. Table 5.4-1 gives the design parameters.

A reactor coolant pump is directly connected to each of two outlet nozzles on the steam generator channel head. The two pumps on a steam generator turn in the same direction.

A sealless pump contains the motor and all rotating components inside a pressure vessel. The pressure vessel consists of the pump casing, stator closure, stator main flange, stator shell, stator lower flange, and stator cap, which are designed for full reactor coolant system pressure. In a canned motor pump, the stator and rotor are encased in corrosion-resistant cans that prevent contact of the rotor bars and stator windings by the reactor coolant. Because the shaft for the impeller and rotor is contained within the pressure boundary, seals are not required to restrict leakage out of the pump into containment. The connection between the pump casing and the stator closure is provided with a welded canopy type seal assembly, which provides definitive leak protection for the pump closure. Access to the internals of the pump and motor is by severing the canopy seal weld. When the pump is reassembled, a canopy seal is rewelded. Canned motor reactor coolant pumps have a long history of safe, reliable performance in military and commercial nuclear plant service.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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The reactor coolant pump driving motor is a vertical, water-cooled, squirrel-cage induction motor with a canned rotor and stator. It is designed for removal from the casing for inspection, maintenance, and replacement, if required. The stator can protects the stator (windings and insulation) from the controlled portion of the reactor coolant circulating inside the motor and bearing cavity. The can on the rotor isolates the copper rotor bars from the system and minimizes the potential for the copper to plate out in other areas.

The motor is cooled by primary reactor coolant system coolant circulating through the motor cavity and by component cooling water circulating through a cooling jacket on the outside of the motor housing. Primary coolant used to cool the motor enters the lower end of the rotor and passes axially through the motor cavity to remove heat from the rotor and stator. An auxiliary impeller provides the motive force for circulating the coolant. Heat from the primary coolant is transferred to component coolant water in an external heat exchanger.

Each pump motor is driven by a variable speed drive, which is used for pump startup and operation when the reactor trip breakers are open. When the reactor trip breakers are closed, the variable frequency drives are bypassed and the pumps run at constant speed.

Flywheel assemblies provide rotating inertia that increases the coastdown time for the pump. Each flywheel assembly is of bi-metallic design consisting of a tungsten heavy metal alloy for mass with Type 403 stainless steel and 18Mn-18Cr alloy steel structural components. The upper flywheel assembly is located between the motor and pump impeller. The lower assembly is located below the canned motor, with the thrust bearing. Surrounding the flywheel assemblies are the heavy walls of the stator closure, casing, thermal barrier, or stator lower flange.

The materials in contact with the reactor coolant and cooling water (with the exception of the bearing material) are austenitic stainless steel, nickel-chromium-iron alloy, or equivalent corrosion-resistant material.

There are two journal bearings, one at the bottom of the rotor shaft and the other between the upper flywheel assembly and the motor. The bearings are a hydrodynamic film-riding design. During rotor rotation, a thin film of water forms between the journal and pads, providing lubrication.

The thrust bearing assembly is at the bottom of the rotor shaft. The pivoted pad hydrodynamic bearing provides positive axial location of the rotating assembly regardless of operating conditions.

The reactor coolant pump is equipped with a vibration monitoring system that continuously monitors pump structure (frame) vibrations. Five vibration monitors provide pump vibration information. The readout equipment includes warning alarms and high-vibration level alarms, as well as output for analytical instruments.

Four resistance temperature detectors (RTDs) monitor motor cooling circuit water temperature. These detectors provide indication of anomalous bearing or motor operation, as well as leakage through the pump labyrinth seal into the stator cavity as a result of a leak in the pump external heat exchanger. They also provide a system for automatic shutdown in the event of a prolonged loss of component cooling water or a large tube leak from the external heat exchanger into the CCS.

A speed sensor monitors rotor rpm’s. Additionally, voltage and current sensors provide information on motor load and electrical input.

5.4.1.2.2 Description of Operation

Reactor coolant is pumped by the main impeller. It is drawn through the eye of the impeller and discharged via the diffuser out through the radial discharge nozzle in the side of the casing. Once the

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motor housing is filled with coolant, the labyrinth seals around the shaft between the impeller and the thermal barrier minimize the flow of coolant into the motor during operation.

An auxiliary impeller at the lower part of the rotor shaft circulates a controlled volume of the primary coolant through the motor cavity and external heat exchanger. The coolant is cooled to about 150°F by component cooling water circulating on the shell side of the external heat exchanger. The cooled reactor coolant then passes through the motor cavity, where it removes heat from the rotor and stator and lubricates the motor's hydrodynamic bearings.

The variable frequency drives enable the startup of the reactor coolant pumps at slow speeds to decrease the power required from the pump motor during operation at cold conditions. The variable frequency drive provides operational flexibility during pump startup and reactor coolant system heatup. During a plant startup, the general startup procedure for the pumps is for the operator to start the pumps at a low speed. During reactor coolant system heatup, the pumps are run at the highest speed that is within the allowable motor current limits. As the reactor coolant temperature increases, the allowable pump speed also increases. Before the reactor trip breakers are closed, the variable frequency controllers are bypassed and the pumps run at constant speed.

During all power operations (Modes 1 and 2), the variable frequency drives are bypassed and the pumps run at constant speed.

5.4.1.3 Design Evaluation

5.4.1.3.1 Pump Performance

The reactor coolant pump is sized to deliver a flow rate that equals or exceeds the required flow rate. Testing prior to plant startup confirms the total delivery capability of the reactor coolant pump. See Section 14.2. Thus, adequate forced circulation coolant flow is confirmed prior to initial plant operation.

The required net positive suction head is provided with ample margin to provide operational integrity and minimize the potential for cavitation. The AP1000 does not require reactor coolant pump operation to achieve safe shut down. Minimum net positive suction head requirements are not required to provide safe operation of the AP1000.

5.4.1.3.2 Overspeed Conditions

Reactor coolant pump overspeed can be postulated for either a fault in the connected electrical system that results in an increase in the frequency of the supplied current or due to a pipe rupture which results in an increase in the flow through the pump as the coolant exits the pipe.

For grid disconnect transients or turbine trips actuated by either the reactor trip system or the turbine protection system, the turbine overspeed control system acts to limit the reactor coolant pump overspeed. The turbine control system acts to rapidly close the turbine governor and intercept valves.

An electrical fault requiring immediate generator trip (with resulting turbine trip) will result in an overspeed condition in the electrically coupled reactor coolant pump no greater than that described previously for the grid disconnect/turbine trip transient.

Pump overspeed from high coolant flow rates associated with pipe rupture events are mitigated by the inertia of the pump, flywheel, and motor and by the connection of the motor to the electrical grid. Because of the application of mechanistic pipe break criteria, dynamic effects such as pump overspeed are not evaluated for breaks in piping in which leak-before-break is demonstrated.

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5.4.1.3.3 Pressure Boundary Integrity

The pressure boundary integrity is verified for normal, anticipated transients, and postulated accident conditions. The pressure boundary components (pump casing, stator closure, stator main flange, stator shell, stator lower flange, stator cap, and external piping and tube side of the external heat exchanger) meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III. These components are designed, analyzed, and tested according to the requirements in Paragraph NB-3400 of the ASME Code, Section III. Wells provided for resistance temperature detectors and a phase reference sensor, and speed sensor penetrations also satisfy the requirements of the ASME Code, Section III.

The motor terminals form part of the pressure boundary in the event of a stator-can failure. The ASME Code does not include criteria or methods for completely designing or analyzing such terminals. Motor terminals are designed, analyzed, and tested using criteria established and validated based on many years of service. Where applicable, ASME Code requirements and criteria are used. Individual terminals are hydrostatically tested to test the integrity prior to performance testing.

5.4.1.3.4 Coastdown Capability

It is important to reactor protection that the reactor coolant continues to flow for a time after reactor trip and loss of electrical power. To provide this flow, each reactor coolant pump has a high-density flywheel and high-inertia rotor. The rotating inertia of the pump, motor, and flywheel is used during the coastdown period to continue the reactor coolant flow. The reactor coolant pump is designed for the safe shutdown earthquake. The coastdown capability of the pump is maintained even for the case of loss of offsite and onsite electrical power coincident with the safe shutdown earthquake. Core flow transients and figures are provided in Subsections 15.3.1 and 15.3.2.

A loss of component cooling water has no impact on coastdown capability. The reactor coolant pump can operate without cooling water until a reactor coolant pump trip occurs on High-2 bearing water temperature. This prevents damage that could potentially affect coastdown.

The reactor trip system maintains the pump operation within the assumptions used for loss of coolant flow analyses. This also provides that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation.

The reactor coolant pump coastdown occurs on a power loss to the plant. The following conditions are assumed to occur simultaneously:

- Reactor coolant system at normal operation temperature and pressure,
- Loss of cooling water,
- Loss of pump power,
- Reactor trip

If the stator can should leak during operation, the reactor coolant may cause a short in the stator windings. In such a case, the result would be the same as a loss of power to that pump. With either a rotor or a stator can failure, no fluid would be lost to the containment.

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5.4.1.3.5 Bearing Integrity

The design requirements for the reactor coolant pump bearings provide long life with negligible wear. The vibration warning level and high-vibration level alarm set-points are, in part, based on evaluation of the effect of vibration on bearing life.

The bearings provide adequate stiffness to control shaft motion, protect the pump impeller and shaft labyrinths from wear, and avoid contact between the motor stator and rotor. The bearing loads are maintained within the load capabilities of hydrodynamic journal bearings even under the severe conditions experienced during seismic events. The bearing/shaft design and loadings are established by analysis and testing.

The frame vibration detectors provide indication of bearing performance. Control room indicators and alarms provide indication for operator action.

The bearing cooling provisions include a temperature monitoring system. The system operates continuously and has at least four redundant indicators per pump. Upon initiation of failure, the system indicates and alarms in the control room as a high bearing water temperature. All of the pumps trip when the High-2 temperature setpoint is reached.

5.4.1.3.6 Integrity of Rotating Components

The rotating components of the pump and motor are analyzed for dynamic characteristics, including natural frequencies, stability, and forced responses to normal operation loads, and for several postulated fault conditions associated with the rotating masses. The fault conditions include seized rotor events, and integrity of the rotating components, including the flywheel.

5.4.1.3.6.1 Natural Frequency and Critical Speeds

The damped natural frequency of the reactor coolant pump rotating assembly is greater than 120 percent of the normal operating speed.

Determination of the damped natural frequency of the reactor coolant pump rotor bearing system model includes the effects of the bearing films, can annular fluid interaction, motor magnetic phenomena, and pump structure. The damped natural frequencies for the AP1000 reactor coolant pump exhibit sufficient energy dissipation to be stable. The high degree of damping provides smooth pump operation.

The pump is analyzed for the response of the rotor and stator to external forcing functions. The support and connection of the pump to the steam generator and piping are considered in the analysis. The responses are evaluated using criteria including critical loads, stress deformation, wear, and displacement limits to establish the actual system critical speeds.

5.4.1.3.6.2 Rotor Seizure

The design of the pump is such as to preclude the instantaneous stopping of any rotating component of the pump or motor. The rotating inertia and power supplied to the motor would overcome interference between the impeller, bearings, flywheel assemblies, motor rotor, or rotor can and the surrounding components for a period of time. A change in the condition of any of the components sufficient to cause an interference would be indicated by the instrumentation monitoring speed, vibration, temperature, or current.

The reactor coolant system and reactor coolant pump are analyzed for a locked rotor event. To analyze the mechanical and structural effects of a rapid slow down of the rotating assembly, a failure

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of the rotating assembly is postulated that results in deformation that causes an interference with the surrounding reactor coolant pump components. For such an interference, the pump and motor are postulated to come to a complete stop in a very short time period. This assumption bounds other postulated mechanisms for a rapid slowdown of the rotor, including impeller rub and rotor or stator can failure. The connection of the pump with the steam generator and discharge piping is analyzed for the vibration of the pump, hydraulic effects, and the torque due to the rapid slow down of the rotating assembly. The stresses in the pump casing, motor housing, steam generator channel head, and piping are analyzed using ASME Code, Section III, Service Level D limits for this condition.

The transient analysis of thermal and hydraulic effects of a postulated locked rotor event is based on a nonmechanistic, instantaneous stop of the impeller. This conservative assumption bounds any slower stop and provides a comparison with the same analysis done for other nuclear power plants. The transient analysis considers the effect of the locked rotor on the reactor core and the reactor coolant system pressure. The results of the transient analysis are found in Subsection 15.3.3 and show that the peak reactor coolant system pressure does not exceed the licensed safety analysis RCS pressure limit.

5.4.1.3.6.3 Flywheel Integrity

The reactor coolant pump in the AP1000 complies with the requirement of General Design Criterion (GDC) Number 4. That Criterion states that components important to safety be protected against the effects of missiles.

The flywheel assemblies are located within and surrounded by the heavy walls of the stator closure, stator main flange, casing, thermal barrier, or lower stator flange. In the event of a postulated worst-case flywheel assembly failure, the surrounding structure can, by a large margin, contain the energy of the fragments without causing a rupture of the pressure boundary. The analysis in Reference 10 of the capacity of the housing to contain the fragments of the flywheel is done using the energy absorption equations of Hagg and Sankey (Reference 2).

Compliance with the requirement of GDC 4 related to missiles can be demonstrated without reference to flywheel integrity, nevertheless, the intent of the guidelines of Regulatory Guide 1.14 is followed in the design and fabrication of the flywheel. The guidelines in Regulatory Guide 1.14 apply to steel flywheels. Since the bi-metallic design of the AP1000 reactor coolant pump flywheel does not respond in the same manner as homogeneous steel, many of the guidelines in the Regulatory Guide are not directly applicable.

The reactor coolant pump flywheel assemblies are fabricated from a tungsten heavy alloy, Type 403 stainless steel, and 18Mn-18Cr alloy steel (ASTM A289, Grade 8). Heavy alloy segments (ASTM B777, Class 4) are fitted to a stainless steel hub modified (ASTM A336, Grade F6); these segments are not relied upon structurally. The modifications to the ASTM A336, Grade F6 specification consist of allowing slightly more carbon and sulfur content. The segments are held into place by an interference fit retainer cylinder of 18Mn-18Cr alloy steel placed over the outside of the assembly. The assembly is hermetically sealed from primary coolant by endplates and an outer thin shell of Alloy 625. Ni/Fe/Cr Alloy 600 base material is not used in the flywheel assembly; however, Alloy 600 type weld filler material may be used for welds on the thin Alloy 625 enclosure.

The bi-metallic flywheel design will be manufactured using multiple processes and materials. In accordance with Regulatory Guide 1.14, each structural component of the bi-metallic flywheel will be inspected prior to final assembly according to its fabrication and the procedures outlined below. The Type 403 stainless steel inner hub material will be subject to impact testing using one Charpy V-notch test (three specimens) in accordance with Section III, NB-2300 of the ASME Code, magnetic particle examination, and ultrasonic examination in accordance with Section III, NB-2500 of the ASME Code. The retainer ring will be subject to one Charpy V-notch test (three specimens) per material

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specification ASTM A289, Supplemental Requirement S1, liquid penetrant examination, and ultrasonic examination in accordance with Section III, NB-2500 of the ASME Code. Following finishing operations on the flywheel assembly, the outside circumference of the retainer ring and the inside surface of the inner hub are subject to liquid penetrant inspections in accordance with Section III, NB-2500 of the ASME Code.

The design speed of the flywheel is defined as 125 percent of the synchronous speed of the motor. The design speed envelopes all expected overspeed conditions. At the normal speed the calculated maximum primary stress in the flywheel assemblies is less than one third of minimum yield strength. At the design speed the calculated maximum primary stress in the flywheel assemblies is less than two thirds of minimum yield strength.

An analysis of the flywheel failure modes of ductile failure, nonductile failure and excessive deformation of the flywheel is performed to evaluate the flywheel design. The analysis is performed to determine that the critical flywheel failure speeds, based on these failure modes, are greater than the design speed. The critical flywheel failure speeds are not the same as the critical speed identified for the rotor. The critical flywheel failure speeds are greater than the design speed. The overspeed condition for a postulated pipe rupture accident is less than the critical flywheel failure speeds.

The flywheel assemblies are sealed within a welded nickel-chromium-iron alloy enclosure to prevent contact with the reactor coolant or any other fluid. The enclosure minimizes the potential for corrosion of the flywheel and contamination of the reactor coolant. The enclosure material specifications are ASTM-B-443 and ASTM-B-564. Even though the welds of the flywheel enclosure are not external pressure boundary welds, these welds are made using procedures and specifications that follow the rules of the ASME Code. A dye penetrant test of the enclosure welds is performed in conformance with these requirements. The final assemblies are leak tested using a leak test hole located in the inner hub.

No credit is taken in the analysis of the flywheel missile generation for the retention of the fragments by the enclosure. A leak in the enclosure during operation could result in an out-of-balance flywheel assembly. An out-of-balance flywheel exhibits an increase in vibration, which is monitored by vibration instrumentation.

The flywheel enclosure contributes only a small portion of the energy in a rotating flywheel assembly.

The stress in the welds of the flywheel enclosure components for normal and design speeds are within the criteria in subsection NG of the ASME Code, which is used as a guideline.

Pipe rupture overspeed is based on a break of the largest branch line pipe connected to the reactor coolant system piping that is not qualified for leak-before-break criteria. The exclusion of the reactor coolant loop piping and branch line piping of 6 inches or larger size from the basis of the pump loss of coolant accident overspeed condition is based on the provision in GDC 4 to exclude dynamic effects of pipe rupture when a leak-before-break analysis demonstrates that appropriate criteria are satisfied. See Subsection 3.6.3 for a discussion of leak-before-break analyses. The criteria of Subsection 3.6.2 are used to determine pipe break size and location for those piping systems that do not satisfy the requirements for mechanistic pipe break criteria.

In addition to material specification and non destructive testing requirement, each flywheel is subject to a spin test at 125 percent overspeed, followed by visual inspection, during manufacture. This demonstrates quality of the flywheel. Since the basis for the safety of the flywheel is retention of the fragments within the reactor coolant pump pressure boundary, periodic inservice inspections of the flywheel assemblies are not required to ensure that the basis for safe operation is maintained.

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Because of the configuration of the flywheel assemblies, inservice inspection of the flywheel assemblies may not result in significant inspection results. Inspection of the flywheel assemblies would require removal of the assemblies from the shaft, removal of the enclosures, rewelding of the enclosure, reassembly, and balancing of the pump shaft. Opening of the pump assembly for a periodic inspection of the enclosure would result in an increased occupational radiation exposure and would not be consistent with goals relative to maintaining exposure as low as reasonably achievable. Also, opening the pump may increase the potential for entry of foreign objects into the motor area. For these reasons, routine, periodic inspection of the flywheel assemblies in the AP1000 reactor coolant pump is not recommended.

5.4.1.3.6.4 Other Rotating Components

The rotating components (other than the flywheel), including the impeller, auxiliary impeller, rotor, and rotor can, are evaluated for potential missile generation. In the event of fracture, the fragments from these components are contained by the surrounding pressure housing. The impeller is contained by the pump casing. The rotor and rotor can are contained by the stator, stator can, and motor housing. The auxiliary impeller is contained by the motor housing. In each case, the energy of the postulated fragments is less than that required to penetrate through the pressure boundary.

5.4.1.4 Tests and Inspections

Reactor coolant pump construction is subject to a quality assurance program. The pressure boundary components meet requirements established by the ASME Code. In addition, the flywheel is subject to quality assurance requirements. Table 5.4-3 outlines the inspection included in the reactor coolant pump quality assurance program.

The reactor coolant pump inservice inspection program is according to the ASME Code, Section XI.

The design enables removal of the pump internals for inspection of the pump casing, if required. As noted earlier, routine inspections of the impeller, flywheel, and motor internals are not required for safe operation of the pump.

5.4.1.4.1 Reactor Coolant System Flow Rate Verification

Initial verification of the reactor coolant system flow rate is made during the plant initial test program. Reactor coolant system flow rates are measured during the pre-core load hot functional tests, and during the startup tests. The objective of these tests is to verify that the reactor coolant system flow rate meets the flow rate range of Technical Specification 3.4.1.

After the pre-core reactor coolant system flow rate measurement is taken, analytical adjustments are made to the pre-core measured reactor coolant system flow rate to predict a post-core reactor coolant system flow rate. Calculations of the reactor coolant system flow rate with and without the core are performed. The calculation of the pre-core load reactor coolant system flow rate is compared with results of the pre-core load flow testing, and this information will be used in the calculation of the post-core load reactor coolant system flow rate as appropriate. The predicted post-core load reactor coolant system flow rate is checked to verify that it satisfies Technical Specification 3.4.1. Verifications are also made that the post-core load reactor coolant system flow rates satisfy Technical Specification 3.4.1 flow limits during startup testing.

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5.4.2 Steam Generators

5.4.2.1 Design Bases

The steam generator channel head, tubesheet, and tubes are a portion of the reactor coolant pressure boundary. The tubes transfer heat to the steam system while retaining radioactive contaminants in the primary system. The steam generator removes heat from the reactor coolant system during power operation and anticipated transients and under natural circulation conditions. The steam generator heat transfer function and associated secondary water and steam systems are not required to provide a safety-related safe shutdown of the plant.

The steam generator secondary shell functions as containment boundary during operation and during shutdown when access opening closures are in place.

Tables 5.4-4 and 5.4-5 give steam generator design data. AP1000 equipment, seismic and ASME Boiler and Pressure Vessel Code classifications of the steam generator components are discussed in Section 3.2. ASME Code and Code Case compliance are discussed in Subsection 5.2.1. The ASME Code classification for the secondary side is specified as Class 2. The pressure-retaining parts of the steam generator, including the primary and secondary pressure boundaries, are designed to satisfy the criteria specified in Section III of the ASME Code for Class 1 components.

Subsection 3.9.3 discusses the design stress limits, loads, and combined loading conditions. Subsection 3.9.1 discusses the transient conditions applicable to the steam generator. The number of transients is based on 60 years of operation.

In addition to the loading conditions associated with pressure and temperature variations for transient and anticipated accident conditions, the steam generator is evaluated for fluid borne and structural vibration originating with the reactor coolant pump. The steam generator is also evaluated for the load on the primary outlet nozzles resulting from a postulated locked reactor coolant pump rotor. See Subsection 5.4.1.3.6 for a discussion of the locked rotor postulation.

Chapter 11 gives estimates of radioactivity levels anticipated in the secondary side of the steam generators during normal operation and the bases for the estimates. Chapter 15 discusses the accident analysis of a steam generator tube rupture.

The water chemistry on the primary side, selected to provide the necessary boron content for reactivity control, should minimize corrosion of reactor coolant system surfaces. The effectiveness of the water chemistry in the control of the secondary side corrosion is discussed in Chapter 10. Compatibility of steam generator tubing with both primary and secondary coolants is discussed further in Subsection 5.4.2.4.3.

The steam generator is designed to minimize the potential for mechanical or flow-induced vibration. Tube support adequacy is discussed in Subsection 5.4.2.3.3. The tubes and tubesheet are analyzed and confirmed to withstand the maximum accident loading conditions defined in Subsection 3.9.3. Further consideration is given in Subsection 5.4.2.3.4 to the effect of tube-wall thinning on accident condition stresses.

5.4.2.2 Design Description

The AP1000 steam generator is a vertical-shell U-tube evaporator with integral moisture separating equipment. Figure 5.4-2 shows the steam generator, indicating several of its design features.

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The design of the Model Delta-125 steam generator, except for the configuration of the channel head, is similar to an upgraded Model Delta-75 steam generator. The Delta-75 steam generator has been placed in operation as a replacement steam generator.

Steam generator design features are described in the following paragraphs.

On the primary side, the reactor coolant flow enters the primary chamber via the hot leg nozzle. The lower portion of the primary chamber is elliptical and merges into a cylindrical portion, which mates to the tubesheet. This arrangement provides enhanced access to all tubes, including those at the periphery of the bundle, with robotics equipment. This feature enhances the ability to inspect, replace and repair portions of the AP1000 unit compared to the more spherical primary chamber of earlier designs. The head is divided into inlet and outlet chambers by a vertical divider plate extending from the apex of the head to the tubesheet.

The reactor coolant flow enters the inverted U-tubes, transferring heat to the secondary side during its traverse, and returns to the cold leg side of the primary chamber. The flow exits the steam generator via two cold leg nozzles to which the reactor coolant pumps are directly attached. A high-integrity, nickel-chromium-iron (Alloy 690) weld is made to the nickel-chromium-iron alloy buttered ends of these nozzles.

A passive residual heat removal (PRHR) nozzle attaches to the bottom of the channel head of the loop 1 steam generator on the cold leg portion of the head. This nozzle provides recirculated flow from the passive residual heat removal heat exchanger to cool the primary side under emergency conditions. A separate nozzle on one of the steam generator channel heads is connected to a line from the chemical and volume control system. The nozzle provides for purification flow and makeup flow from the chemical and volume control system to the reactor coolant system.

The AP1000 steam generator channel head has provisions to drain the head. To minimize deposits of radioactive corrosion products on the channel head surfaces and to enhance the decontamination of these surfaces, the channel head cladding is machined or electropolished for a smooth surface. The primary manways provide enhanced primary chamber access compared to previous model steam generators.

Should steam generator replacement using a channel head cut be required, the arrangement of the AP1000 steam generator channel head facilitates steam generator replacement in two ways. It is completely unobstructed around its circumference for mounting cutting equipment. And is long enough to permit post-weld heat treatment with minimal effect of tubesheet acting as a heat sink.

The tubes are fabricated of nickel-chromium-iron Alloy 690. The tubes undergo thermal treatment following tube-forming operations. The tubes are tack-expanded, welded, and expanded over the full depth of the tubesheet. Full depth expansion was selected because of its capability to minimize secondary water access to the tube-to-tube-sheet crevice. The method by which the tubes are expanded into the tubesheet is determined based on consideration of the residual stresses and the resultant susceptibility of the tube to degradation. Residual stresses (and the expanded tube's susceptibility to degradation) are limited, in part, through tight control of the pre-expansion clearance between the tube and tubesheet hole.

Support of the tubes is provided by ferritic stainless steel tube support plates. The holes in the tube support plates are broached with a hole geometry to promote flow along the tube and to provide an appropriate interface between the tube support plate and the tube. Figure 5.4-3 shows the support plate hole geometry. Anti-vibration bars installed in the U-bend portion of the tube bundle minimize the potential for excessive vibration.

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Steam is generated on the shell side, flows upward, and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes through a feedwater nozzle. The feedwater enters a feedring via a welded thermal sleeve connection and leaves it through nozzles attached to the top of the feedring. The nozzles are fabricated of an alloy that is very resistant to erosion and corrosion with the expected secondary water chemistry and flow rate through the nozzles. After exiting the nozzles, the feedwater flow mixes with saturated water removed by the moisture separators. The flow then enters the downcomer annulus between the wrapper and the shell.

Fluid instabilities and water hammer phenomena are important considerations in the design of steam generators. Water level instabilities can occur from density wave instabilities which could affect steam generator performance. Density wave instability is avoided in the AP1000 steam generator by including appropriate pressure losses in the downcomer and the risers that lead to negative damping factors.

Steam generator bubble collapse water hammer has occurred in certain early pressurized water reactor steam generator designs having feedrings equipped with bottom discharge holes. Prevention and mitigation of feedline-related water hammer has been accomplished through an improved design and operation of the feedwater delivery system. The AP1000 steam generator and feedwater system incorporate features designed to eliminate the conditions linked to the occurrence of steam generator water hammer. The steam generator features include introducing feedwater into the steam generator at an elevation above the top of the tube bundle and below the normal water level by a top discharge feedring. The top discharge of the feedring helps to reduce the potential for vapor formation in the feedring. This minimizes the potential for conditions that can result in water hammer in the feedwater piping. The feedwater system features (Subsection 10.4.7 discusses in more detail) designed to prevent and mitigate water hammer include a short, horizontal or downward sloping feedwater pipe at steam generator inlet.

These features minimize the potential for trapping pockets of steam which could lead to water hammer events.

Stratification and striping are reduced by an upturning elbow inside the steam generator which raises the feedring relative to the feedwater nozzle. The elevated feedring reduces the potential for stratified flow by allowing the cooler, more dense feedwater to fill the nozzle/elbow arrangement before rising into the feedring.

The potential for water hammer, stratification, and striping is additionally reduced by the use of a separate startup feedwater nozzle. The startup feedwater nozzle is located at an elevation that is the same as the main feedwater nozzle and is rotated circumferentially away from the main feedwater nozzle. A startup feedwater spray system independent of the main feedwater feedring is used to introduce startup feedwater into the steam generator. The layout of the startup feedwater piping includes the same features as the main feedwater line to minimize the potential for waterhammer. The startup feedwater system is used to introduce water into the secondary side of the steam generator as described in Subsection 10.4.7.2.3.

At the bottom of the wrapper, the water is directed toward the center of the tube bundle by the lowest tube support plate. This recirculation arrangement serves to minimize the low-velocity zones having the potential for sludge deposition.

As the water passes the tube bundle, it is converted to a steam-water mixture. Subsequently, the steam-water mixture from the tube bundle rises into the steam drum section, where centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary separators, or dryers, for further moisture removal, increasing its quality to a designed minimum of 99.75 percent (0.25 percent by weight maximum moisture). Water separated from the

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steam combines with entering feedwater and recirculates through the steam generator. A sludge collector located amidst the inner primary separator risers provides a preferred region for sludge settling away, from the tubesheet and tube support plates. The dry, saturated steam exits the steam generator through the outlet nozzle, which has a steam-flow restrictor. (See Subsection 5.4.4.)

5.4.2.3 Design Evaluation

Integrity of the pressure retaining function of the steam generator is provided by compliance with the ASME Code. The evaluation of the stress levels and fatigue usage for the steam generator pressure boundary is calculated for the specified loading conditions and demonstrates that the values are less than the allowable limits. These calculations are documented in a stress report as required by the ASME Code. Corrosion allowances which are consistent with material erosion/corrosion resistance and service environment (velocity, chemistry, etc.) are employed throughout the design.

Meeting the heat transfer requirements and tube vibration and tube wall integrity requirements in addition to the ASME Code requirements is discussed in the following subsections:

5.4.2.3.1 Forced Convection

The steam generator transfers to the secondary coolant loop the heat generated during power operation in the reactor and by the reactor coolant pumps. The evaluation of the steam generator thermal performance, including required heat transfer area and steam flow, uses conservative assumptions for parameters such as primary flow rates and heat transfer coefficients. The effective heat transfer coefficient is determined by the physical characteristics of the AP1000 steam generator and the fluid conditions in the primary and secondary systems for the nominal 100 percent design case. It includes a conservative allowance for fouling and uncertainty. Tables 5.4-4 and 5.4-5 show the nominal design requirements and parameters. Table 5.1-1 lists additional parameters used to evaluate the steam generator design.

5.4.2.3.2 Natural Circulation Flow

When the normal feedwater supply is not available, water may be supplied to the steam generators by the startup feedwater system. The startup feedwater system is a nonsafety-related system that provides a nonsafety-related source of decay heat removal. In addition, the system is used during startup and shutdown and other times when the normal feedwater system is not available.

When the steam generator is supplied with water from the startup feedwater system, the steam generator has enough surface area and a small enough primary-side hydraulic resistance to remove decay heat from the reactor coolant by natural circulation without operation of the reactor coolant pumps.

If the passive residual heat removal system activates, the passive residual heat removal nozzle connection to the steam generator passes coolant flow from the passive residual heat removal heat exchanger into the cold leg side of the channel head. Coolant is drawn through the reactor coolant pumps into the cold legs and then into the reactor vessel.

5.4.2.3.3 Mechanical and Flow-Induced Vibration under Normal Operating Conditions

Potential sources of tube excitation are considered, including primary fluid flow within the U-tubes, mechanically induced vibration, and secondary fluid flow on the outside of the U-tubes. The effects of primary fluid flow and mechanically induced vibration, including those developed by the reactor coolant pump, are acceptable during normal operation. The primary source of potential tube degradation due to vibration is the hydrodynamic excitation of the tubes by the secondary fluid. This

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area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience.

Three potential tube vibration mechanisms related to hydrodynamic excitation of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluid-elastic vibration mechanisms.

Nonuniform, two-phase turbulent flow exists throughout most of the tube bundle. Therefore, vortex shedding is possible only for the outer few rows of the inlet region. Moderate tube response caused by vortex shedding is observed in some carefully controlled laboratory tests on idealized tube arrays. However, no evidence of tube response caused by vortex shedding is observed in steam generator scale model tests simulating the inlet region. Bounding calculations consistent with laboratory test parameters confirmed that vibration amplitudes would be acceptably small, even if the carefully controlled laboratory conditions were unexpectedly reproduced in the steam generator.

Flow-induced vibrations due to flow turbulence are also small: Root mean square amplitudes are less than allowances used in tube sizing. These vibrations cause stresses that are two orders of magnitude below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation due to secondary flow turbulence is anticipated.

Tube fluid elastic excitation is potentially more significant than either vortex shedding or turbulence. Relatively large tube amplitudes can feed back proportionally large tube driving forces if an instability threshold is exceeded. Tube support spacing, in both the tube support plates in the straight leg region and the anti-vibration bars in the U-bend region, provides tube response frequencies such that the instability threshold is not exceeded. This approach provides large margins against initiation of fluid elastic vibration for tubes effectively supported by the tube support system.

Small clearances between the tubes and the supporting structure are required for steam generator fabrication. These clearances introduce the potential that any given tube support location may not be totally effective in restraining tube motion if there is a finite gap around the tube at that location. Fluid-elastic tube response within available support clearances is therefore theoretically possible if secondary flow conditions exceed the instability threshold when no support is assumed at the location with a gap around the tube. This potential has been investigated both with tests and analyses for both the U-bend and straight leg regions.

AP1000 steam generator tube wear potential is expected to be within available design margins even for limiting tube fit-up conditions, based on previous experience. The AP1000 steam generator includes a number of features that minimize the potential for tube wear at tube supports and antivibration bars. Provisions to minimize the potential for wear include optimal spacing between the tube supports and the configuration of the anti-vibration bar assemblies. Tube wear is minimized in the tube support plate design by the configuration of the broached hole through the support plate, the surface finish of the broached hole in the tube support plate, the clearance between the tube and the hole in the tube support plate, and tube support plate material selection.

Tube bending stresses corresponding to tube vibration response remain more than two orders of magnitude below fatigue limits as a consequence of vibration amplitudes constrained by the tube supports. These analyses and tests for limiting postulated fit-up conditions include simultaneous contributions from flow turbulence.

As outlined, analyses and tests demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the AP1000 steam generators. Operating experience with steam generators having the same size tubes and similar flow conditions supports this conclusion.

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The U-bend fatigue (discussed in NRC Bulletin 88-02) is not a consideration in the AP1000 steam generators. The mechanism considered in Bulletin 88-02 requires denting of the top tube support plate. But this is not expected with the stainless steel tube support plates in the AP1000 steam generator. Additionally, the location of anti-vibration bars is controlled by in-process dimensional inspection.

5.4.2.3.4 Allowable Tube Wall Thinning under Accident Conditions

An evaluation determined the extent of tube wall thinning that can be tolerated under accident conditions. The worst-case loading conditions are assumed to be imposed upon uniformly thinned tubes at the most critical location in the steam generator. Under such a postulated design basis accident, vibration is short enough duration that there is no endurance issue to be considered.

The steam generator tubes, existing originally at their minimum wall thickness and reduced by a conservative general corrosion and erosion loss, provide an adequate safety margin (sufficient wall thickness) in addition to the minimum required for a maximum stress less than the allowable stress limit, as defined by the ASME Code.

Studies have been made on AP1000 sized tubing under accident loadings. The results show that the maximum Level D Service condition stress due to combined pipe rupture and safe shutdown earthquake loads is less than the allowable limit. The tube thickness required to achieve the acceptable stress is less than the minimum AP1000 steam generator tube wall thickness, which is reduced to account for assumed general corrosion and erosion rate. Thus, an adequate safety margin is exhibited. The general corrosion rate is based on a conservative weight-loss rate for Alloy 690 TT tubing in flowing, 650°F primary-side reactor coolant fluid. The estimated weight loss, based on testing when equated to a thinning rate and projected over a 60-year design objective, is much less than the assumed corrosion allowance of 3 mils. This leaves the remainder of the general corrosion allowance for thinning on the secondary side.

5.4.2.4 Steam Generator Materials

5.4.2.4.1 Selection and Fabrication of Materials

The pressure boundary materials used in the steam generator are selected and fabricated in accordance with the requirements of Section II and III of the ASME Code. Subsection 5.2.3 contains a general discussion of material specifications. Table 5.2-1 lists the types of materials. Fabrication of reactor coolant pressure boundary materials is also discussed in Subsection 5.2.3, particularly in Subsections 5.2.3.3 and 5.2.3.4.

Industry-wide corrosion testing and specification development programs have justified the selection of thermally treated Alloy 690, a nickel-chromium-iron alloy (ASME SB-163), for the steam generator tubes. The channel head divider plate is also Alloy 690 (ASME SB-168). The interior surfaces of the reactor coolant channel head, nozzles, and manways are clad with austenitic stainless steel. The primary side of the tubesheet is weld clad with nickel-chromium-iron alloy (ASME SFA-5.14). The tubes are then seal welded to the tubesheet cladding. These fusion welds, comply with Sections III and IX of the ASME Code. The welds are dye-penetrant inspected and leak-tested before each tube is expanded the full depth of the tubesheet bore.

Nickel-chromium-iron alloy in various forms is used for parts where high velocities could otherwise lead to erosion/corrosion. These include the nozzles on the feedwater ring and startup feedwater sparger.

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Subsection 5.2.1 discusses authorization for use of ASME Code cases used in material selection. Subsection 1.9.1 discusses the extent of conformance with Regulatory Guides 1.84, Design and Fabrication Code Case Acceptability ASME Section III, Division 1.

During manufacture, the primary and secondary sides of the steam generator are cleaned according to written procedures following the guidance of Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants, and ASME NQA-1 Part II. Onsite cleaning and cleanliness control also follow the guidance of Regulatory Guide 1.37 (discussed in Subsection 1.9.1). Cleaning process specifications are discussed in Subsection 5.2.3.4.

Subsection 5.2.3.3 discusses the fracture toughness of the materials. Adequate fracture toughness of ferritic materials in the reactor coolant pressure boundary is provided by compliance with 10 CFR 50, Appendix G, Fracture Toughness Requirements, and Paragraph NB-2300 of Section III of the ASME Code.

The heat and lot of tubing material for each steam generator tube is recorded and documented as part of the quality assurance records. Archive samples of each heat and lot of steam generator tubing material are provided for use in future materials testing programs or as inservice inspection calibration standards. A minimum of 7 feet of tubing in the final heat treat condition is supplied.

The exterior of the steam generator surface may be submerged following a postulated actuation of the automatic depressurization system (ADS). During this event, water may be present on the outside of the steam generator without affecting the heat transfer or pressure boundary capabilities of the AP1000 steam generator.

5.4.2.4.2 Steam Generator Design Effects on Materials

Several features in the AP1000 steam generator minimize crevice areas and the deposition of contaminants from the secondary-side flow. Such crevices and deposits could otherwise produce a local environment allowing potential chemical concentration and material corrosion.

The portion of the tube within the tubesheet is expanded to close the crevice between the tube and tubesheet. The length of the expansion is carefully controlled to minimize the potential of an over-expanded condition above the tubesheet and to minimize the extent of unexpanded tube at the top of the tubesheet.

The tube support plates are made of corrosion resistant Type 405 stainless steel alloy. A three-lobed, or trifoil, tube hole design provides flow adjacent to the tube outer surface. This provides high sweeping velocities at the tube and tube support plate intersections. The trifoil tube support plate provides in-plane and out-of-plane strength. The sweeping velocities through the support plate reduce sludge accumulation in the tube-to-tube support crevices. Figure 5.4-3 shows the trifoil broached holes. This support plate design contributes to a high circulation ratio. The increased flow from a high circulation ratio circulation results in increased flow in the interior of the bundle, as well as horizontal velocity across the tubesheet, which reduces the tendency for sludge deposition.

The effect of the total bundle flow on the vibrational stability of the tube bundle has been analyzed, with consideration given to flow-induced excitation frequencies. The maximum unsupported span length of tubing in the U-bend region and the optimal number of anti-vibration bars has been determined, using advanced statistical techniques and vibration modeling. The anti-vibration bars are fabricated from Type 405 stainless steel. The construction minimizes the gaps between the anti-vibration bars and tubes.

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Additional measures in the AP1000 steam generator design minimize areas of dryout in the steam generator and sludge accumulations in low-velocity areas. The wrapper design results in significant water velocities across the tubesheet.

A high capacity blowdown system is capable of continuous blowdown of the steam generators at a moderate volume and intermittent flow. The intakes of the blowdown system are located on the top surface (secondary side) of the tubesheet.

A passive sludge collector, which provides a low flow settling zone, is in the upper shell region located among the inner primary moisture separator risers. The sludge collector, or mud drum, provides a location for particulate to settle remote from the tubesheet and tube support plates. The mud drum can be cleaned during a plant shutdown.

Several methods can be used to clean operating steam generators of secondary-side deposits. Sludge lancing is a procedure in which a hydraulic jet inserted through an access opening (handhole) loosens deposits and the loose material is flushed out of the steam generator. A minimum of four 6-inch access ports are provided for sludge lancing, inspection of the tube bundle by portable inspection equipment, and retrieval of loose objects. They are located above the tubesheet 90° apart (two on the tubelane and two at 90° from the tube lane) to provide access to the secondary face of the tubesheet. Also, a minimum of two 4-inch ports located on the secondary shell in line with the tubelane and above the top tube support plate provide access to the U-Bend area. A blowdown hole, located at the bottom of the secondary side drain channel permits continuous blowdown and monitoring of secondary water chemistry. The materials of the secondary side of the steam generator are also compatible with chemical cleaning.

5.4.2.4.3 Compatibility of Steam Generator Tubing with Primary and Secondary Coolants

The industry corrosion tests mentioned in Subsection 5.4.2.4.1, subjected the steam generator tubing material thermally treated Alloy 690 ASME SB-163, to simulated steam generator water chemistry. These tests indicated that the loss due to general corrosion over the 60-year operating design objective is small compared to the tube wall thickness. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions indicate that Alloy 690 TT provides as good or better corrosion resistance as either Alloy 600 TT or nickel-iron-chromium Alloy 800. Alloy 690 TT also resists general corrosion in severe operating water conditions.

Some operating experience has revealed areas on secondary surfaces where localized corrosion rates were significantly greater than the low general corrosion rates. Both intergranular stress corrosion and tube wall thinning were experienced in localized areas, although not simultaneously at the same location or under the same environmental conditions (water chemistry, sludge composition).

The all volatile treatment (AVT) control program minimizes the possibility of the tube wall thinning phenomenon. Successful AVT operation requires maintenance of low concentrations of impurities in the steam generator water. This reduces the potential for formation of highly concentrated solutions in low-flow zones, which is a precursor of corrosion. By restricting the total alkalinity in the steam generator and prohibiting extended operation with free alkalinity, the all volatile treatment program minimizes the possibility for intergranular corrosion in localized areas due to excessive levels of free caustic.

Laboratory testing shows that Alloy 690 TT tubing is compatible with the AVT environment. Isothermal corrosion testing in high-purity water shows that Alloy 690 TT exhibiting normal microstructure tested at normal engineering stress levels is not susceptible to intergranular stress

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corrosion cracking in extended exposure to high-temperature water. These tests also show that no general type corrosion occurred. Field experience with Alloy 690 TT tubing in operation since 1989 has been excellent.

Model boiler tests evaluate similar AVT chemistry guidelines adopted by Westinghouse and EPRI. Conformance to the guidelines enhances tube corrosion performance. The secondary water chemistry guidelines for AP1000 are found in Chapter 10. Action levels for secondary side water chemistry during power operation are addressed in Subsection 10.3.5.5. Extensive operating data has been accumulated for all volatile treatment chemistry.

A comprehensive program of steam generator inspections, including the recommendations of Regulatory Guide 1.83, Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes, with the exceptions as stated in Subsection 1.9.1, provides for detection of any degradation that might occur in the steam generator tubing.

Included with the standard operating condition water chemistry controls are chemistry controls during zero power (including shutdown, no-load, heatup, cooldown, and refueling operations). The startup feedwater nozzle may be used to supply hydrazine, ammonia, and other chemicals to control secondary pH and oxygen during wet layup. Sparging the steam generator with nitrogen through the blowdown line also promotes secondary recirculation at zero power. This recirculation can be used, in conjunction with the addition of cleaning agents into the secondary side, to remove magnetite, copper, or other deposited contaminants. The AP1000 steam generator is also configured for pressure pulse cleaning and water slap methods to remove deposits on the secondary side.

High margins against primary water stress corrosion cracking exist with the specification of thermally treated Alloy 690 tubing. Alloy 690 TT is resistant to primary water stress corrosion cracking over the range of anticipated operating environments. The tubing is thermally treated according to a laboratory-derived treatment process and is generally consistent with industry-accepted and EPRI procedures.

The tube support plates are fabricated of ferritic stainless steel. Laboratory tests show that this material is resistant to corrosion in the AVT environment. If corrosion of ferritic stainless steel were to occur because of the concentration of contaminants, the volume of the corrosion products is essentially equivalent to the volume of the parent material consumed. This would be expected to preclude denting. The support plates are also designed with trifoil tube holes rather than cylindrical holes. The trifoil tube hole (see Figure 5.4-3) design promotes high velocity flow along the tube and is expected to minimize the accumulation of impurities at the support plate location.

5.4.2.5 Steam Generator Inservice Inspection

The steam generator is designed to permit inspection of pressure boundary parts, including individual tubes. Preservice inspection of the AP1000 steam generators is performed according to the ASME Code. Inservice inspection complies with the requirements of 10 CFR 50.55a.

The design includes a number of openings to provide access to both the primary and secondary sides of the steam generator. The openings include four 18-inch diameter manways, one for access to each chamber of the reactor coolant channel head and two in the steam drum for inspection and maintenance of the upper shell internals. In addition, there are a minimum of four 6-inch diameter handholes in the shell, located just above the tubesheet secondary surface are provided. A minimum of two 4-inch diameter inspection openings are provided at each end of the tubelane between the upper tube support plate and the row 1 tubes. Additional access to the tube bundle U-bend is provided through the internal deck plate at the bottom of the primary separators. For proper functioning of the steam generator, some of the deck-plate openings are covered with hatch plates welded in place that are removable by grinding, gouging, or other methods to cut off the welds.

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Regulatory Guide 1.83 provides recommendations on the inspection of tubes. The recommendations cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, and required actions based on findings. Any eddy current inspection performed in the manufacturing facility is conducted by personnel qualified to the requirements for inspectors performing inservice inspection of operating units. The manufacturing facility inspection is conducted using the same equipment as, or equipment similar to, that used during inservice inspection of operating units. Exceptions to Regulatory Guide 1.83 are noted in Subsection 1.9.1.

The steam generators permit access to tubes for inspection, repair, or plugging, if necessary, per the guidelines described in Regulatory Guide 1.83. Tooling to install mechanical and welded plugs, tube repair sleeves, or effect other repair processes remotely can be delivered robotically. The AP1000 steam generator includes features to enhance robotics inspection of steam generator tubes. These include a cylindrical section of the channel head and primary manways. Computer simulation using designs of existing robotically delivered inspection and maintenance equipment verifies that tubes can be accessed. To facilitate tube identification for manual activities, the tube location for a fraction of the tubes is scribed on the tubesheet.

The minimum requirements for inservice inspection of steam generators, including tube repair criteria, are discussed in Subsection 5.4.15 considering NRC requirements and industry recommendations. The steam generator tube integrity is verified in accordance with a Steam Generator Tube Surveillance Program. The Steam Generator Tube Surveillance Program is discussed in Subsection 5.4.15. Section XI of the ASME Code provides general acceptance criteria for indications of tube degradation in the steam generator.

A steam generator tube surveillance program is implemented in accordance with the recommendations and guidance of Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines” (Reference 201). A program for periodic monitoring of degradation of steam generator internals is also implemented in accordance with NEI 97-06. Applicable Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP) guidelines are followed as described in the NEI 97-06. The Programs are in compliance with applicable sections of ASME Section XI.

NEI 97-06 and the referenced EPRI SGMP guidelines provide recommendations concerning the inspection of tubes, which cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, required actions based on findings, and tube plugging. The minimum requirements for inservice inspection of steam generators, including plugging criteria, are established in Technical Specification 5.5.4.

The tube surveillance and degradation monitoring programs include provisions to maintain the compatibility of steam generator tubing with primary and secondary coolant to limit the steam generators’ susceptibility to corrosion. These provisions are in accordance with NEI 97-06.

5.4.2.6 Quality Assurance

The steam generator is constructed to a quality assurance program that meets the requirements of the ASME Code and ASME NQA-1. Table 5.4-6 outlines the testing included in the steam generator quality assurance program.

The radiographic inspection and acceptance standard comply with the requirements of Section III of the ASME Code per applicable Code Year and Addenda.

Liquid penetrant inspection and acceptance standards comply with the requirements of Section III of the ASME Code per applicable Code Year and Addenda. Liquid penetrant inspection is performed on

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weld-deposited tubesheet cladding, channel head cladding, divider-plate-to-tubesheet and to channel head weldments, tube-to-tubesheet weldments, and weld-deposit cladding.

Magnetic particle inspection and acceptance standards comply with the requirements of Section III of the ASME Code per applicable Code Year and Addenda. Magnetic particle inspection is performed on the tubesheet forging, channel head forging, nozzle forging, and the following weldments:

- Nozzle to shell (if not integral)
- Support brackets
- Instrument connection (secondary)
- Temporary attachments, after removal
- Accessible pressure retaining welds after hydrostatic test

Ultrasonic inspection and acceptance standards comply with the requirements of Section III of the ASME Code per applicable Code Year and Addenda. Ultrasonic tests are performed on the tubesheet forgings, tubesheet cladding, secondary shells and heads plates and forgings, and nozzle forgings.

The heat transfer tubing is subjected to eddy current testing and ultrasonic examination.

Hydrostatic tests comply with Section III of the ASME Code.

Non-destructive examination of pressure boundary and associated weldments will be performed in accordance with the applicable Code Year and Addenda of ASME Section III, Subsections NB and NC.

5.4.3 Reactor Coolant System Piping

5.4.3.1 Design Bases

The reactor coolant system piping accommodates the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. The piping in the reactor coolant system is AP1000 equipment Class A (ANS Safety Class 1, Quality Group A) (see Subsection 3.3.2) and is designed and fabricated according to ASME Code, Section III, Class 1 requirements. Lines with a 3/8-inch or less flow restricting orifice qualify as AP1000 equipment Class B (ANS Safety Class 2, Quality Group B) and are designed and fabricated with ASME Code, Section III, Class 2 requirements. If one of these lines breaks, the chemical and volume control system charging pumps are capable of providing makeup flow while maintaining pressurizer water level. Stresses are maintained within the limits of Section III of the ASME Code. Code and material requirements are provided in Section 5.2. Inservice inspection of Class 1 components is discussed in Subsection 5.2.4.

Materials of construction are specified to minimize corrosion/erosion and to provide compatibility with the operating environment including the expected radiation level. The welding, cutting, heat treating and other processes used to minimize sensitization of stainless steel are discussed in Subsection 5.2.3.

The thickness of reactor coolant system piping satisfies the design requirements of the ASME Code, Section III, Subsection NB. The analysis of piping of nominal pipe size of 6 inches or greater which demonstrates leak-before-break characteristics, as outlined in Subsection 3.6.3, does not include

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loads due to the dynamic effects of pipe rupture. The minimum pipe bend radius is 1.5-nominal pipe diameters, and ovality meets the requirements of the ASME Code.

Butt welds, branch connection nozzle welds, and boss welds are of a full-penetration design. Flanges conform to ANSI B16.5. Socket weld fittings and socket joints conform to ANSI B16.11.

5.4.3.2 Design Description

5.4.3.2.1 Piping Elements

The reactor coolant system piping includes those sections of reactor coolant hot leg and cold leg piping interconnecting the reactor vessel, steam generators, and reactor coolant pumps. It also includes piping connected to the reactor coolant loop piping and primary components. Figure 5.1-5 shows the Piping and Instrumentation Drawing (P&ID) of the reactor coolant system. The boundary of the reactor coolant system includes the second of two isolation or shut off valves and the piping between those valves. A single ASME Code safety valve may also represent the boundary of the reactor coolant system. The connected piping in the reactor coolant system includes the following:

- Chemical and volume control system (CVS) purification return line from the system isolation valve up to a nozzle on the steam generator channel head
- Chemical and volume control system purification line from the branch connection on the pressurizer spray line to the system isolation valve
- Pressurizer spray lines from the reactor coolant cold legs up to the spray nozzle on the pressurizer vessel
- Normal residual heat removal system (RNS) pump suction line from one reactor coolant hot leg up to the designated isolation valve
- Normal residual heat removal system discharge line from the designated check valve to the connection to the core makeup tank return lines to the reactor vessel direct injection nozzle
- Accumulator lines from the designated check valve to the reactor vessel direct injection nozzle
- Passive core cooling system (PXS) lines from the cold legs to the core make-up tanks and back to the reactor vessel direct injection nozzles
- Drain, sample and instrument lines to the designated isolation valve.
- Pressurizer surge line from one reactor coolant loop hot leg to the pressurizer vessel surge nozzle
- Pressurizer spray scoop, reactor coolant temperature element installation boss, and the temperature element well itself
- All branch connection nozzles attached to reactor coolant loops
- Pressure relief lines in the pressurizer safety and relief valve module from nozzles on top of the pressurizer vessel up to and including the pressurizer safety valves
- Automatic depressurization system (ADS) lines from the pressurizer relief lines to the stages 1, 2, and 3 automatic depressurization system valves

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- Automatic depressurization system lines from the connection with the hot leg up to the fourth stage valves
- Auxiliary spray line from the isolation valve up to the main pressurizer spray line
- Passive core cooling system lines from the hot leg to the passive residual heat removal heat exchanger, and back to the nozzle on the steam generator channel head
- Vent line from the reactor vessel head to the system isolation valves
- In-containment refueling water storage tank injection lines from the designated valves to the reactor vessel direct injection nozzle

Table 5.4-7 gives principal design data for the reactor coolant piping.

A discussion of the codes used in the fabrication of reactor coolant piping and fittings appears in Section 5.2.

Reactor coolant system piping is fabricated of austenitic stainless steel. The piping is forged seamless without longitudinal or electroslag welds. It complies with the requirements of the ASME Code, Section II (Parts A and C), Section III, and Section IX. The reactor coolant system piping does not contain any cast fittings. Changes in direction are accomplished in most cases using bent pipe instead of elbows to minimize the number of welds, fittings, and short radius turns.

5.4.3.2.2 Piping Connections

Joints and connections are welded, except for the pressurizer safety valves, the reactor head vent line, miscellaneous vents and drains, orifice flanges, and the RCP heat exchanger piping, where flanged joints are used. Fillet welds may be used to connect small instrument lines to socket weld connections. RCS loop piping branch lines and penetrations are above the horizontal centerline of the reactor coolant loop piping, except for the following:

- The residual heat removal pump suction line, which is located at the bottom of a hot leg pipe. This enables the water level in the reactor coolant system to be lowered in the reactor coolant loop pipe while continuing to operate the residual heat removal system, should this be required for maintenance.
- The pressurizer level channel nozzles with a 0.375-inch or less flow restrictor and the hot leg level channel nozzle with a 0.375-inch flow restrictor located in the hot leg piping.
- Eight of twelve cold leg-narrow range thermowells.
- Three of the six narrow-range thermowell taps in each hot leg.
- Two pressurizer spray scoop nozzles, one each located in cold legs L002A and L002B.
- One flow measurement tap on the outside bend of each hot leg elbow (total of two).
- One flow measurement tap on the inside and outside of each cold leg elbow (total of eight).

5.4.3.2.3 Encroachment into Coolant Flow

Parts encroaching into the primary coolant loop flow path are limited to the following:

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- The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.

- The narrow-range and wide-range temperature detectors are in resistance temperature detector wells that extend into both the hot and cold legs of the reactor coolant loop piping.

5.4.3.3 Design Evaluation

The loading combinations, stress limits, and analytical methods for the structural evaluation of the reactor coolant system piping and supports for design conditions, normal conditions, anticipated transients, and postulated accident conditions are discussed in Subsection 3.9.3. The requirements for dynamic testing and analysis are discussed in Subsection 3.9.2. The reactor coolant system design transients for normal operation, anticipated transients and postulated accident conditions are discussed in Subsection 3.9.1.

The pressurizer surge line has been specifically designed and instrumented to minimize the potential for thermal stratification that could increase cyclic stresses and fatigue usage. At the connection of the surge line to the hot leg, the surge line is sloped 24 degrees from horizontal. The connection to the reactor coolant hot leg is in the portion of the loop piping that is at an angle with horizontal and adjacent to the steam generator inlet nozzle. The run between the hot leg and pressurizer continuously slopes up. The surge line has an angle of at least 2.5 degrees to horizontal. The pressurizer surge line is shown in Figure 5.4-4. Changes of direction in the surge line are made using pipe bends instead of elbow fittings.

The surge line temperature is monitored for indication of thermal stratification. The temperature is monitored at three locations using strap-on resistance temperature detectors. One location is on the vertical section of pipe directly under the pressurizer. The other two locations are on the top and bottom of the pipe at the same diameter on a more horizontal section of pipe near the pressurizer.

Temperatures in the spray lines from the cold legs of one loop are measured and indicated. Alarms from these signals actuate to warn the operator of low spray water temperature or to indicate insufficient flow in the spray lines.

5.4.3.4 Material Corrosion/Erosion Evaluation

The pipe material is selected to minimize corrosion in the reactor coolant water chemistry. (See Subsection 5.2.3.) A periodic analysis of the coolant chemistry is performed to verify that the reactor coolant water quality meets the specifications. Water quality is maintained to minimize corrosion by using the chemical and volume control system and sampling system, described in Chapter 9.

Contamination of stainless steel and nickel-chromium-iron alloys by copper, low-melting-temperature alloys, mercury, and lead is prohibited during fabrication, installation, and operation.

The austenitic stainless steel surfaces are cleaned to an appropriate halogen limit. The austenitic stainless steel piping is very resistant to erosion due to single-phase fluid flow. The flow rate in the reactor coolant loop piping and branch connections during normal operation and anticipated transients is significantly below the threshold value for erosion due to water for austenitic stainless steel.

The material selection, water chemistry specification, and residual stress in the piping minimize the potential for stress corrosion cracking. (See Subsection 5.2.3.) Reactor coolant system piping is stress-relieved subsequent to bending or other fabrication operations which could result in significant residual stress in the pipe. Processes such as welding or heat treating which apply heat to stainless steel are controlled to minimize the potential for sensitization of the stainless steel.

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Pressure boundary welds out to the second valve that delineates the reactor coolant system boundary are accessible for inservice examination as required by ASME Code, Section XI, and are fitted with removable insulation. Reactor coolant system piping is seamless and does not have any longitudinal welds.

5.4.3.5 Test and Inspections

The reactor coolant system piping construction is subject to a quality assurance program. The pressure boundary components meet requirements established by the ASME Code and ASME NQA-1. The testing included in the reactor coolant system piping quality assurance program is outlined in Table 5.4-8.

A transverse tension test conforming with the supplementary requirements S2 of material specification ASME SA-376 applies to each heat of pipe material.

Ultrasonic examination is performed throughout 100 percent of the wall volume of each pipe, fitting, and other forgings according to the applicable requirements of Section III of the ASME Code for reactor coolant system piping. Unacceptable defects are eliminated according to the requirements of the ASME Code. The surfaces of weld areas are smooth enough to permit preservice and inservice non-destructive examination.

The ends of pipe sections and branch ends are machined to provide a smooth weld transition adjacent to the weld.

A liquid penetrant examination is performed on accessible surfaces, including weld surfaces, of each finished pipe and fitting according to the criteria of the ASME Code, Section III. Acceptance standards are according to the applicable requirements of the ASME Code, Section III. Liquid penetrant examinations are done on the area of pipe bends before the bending operation and after the subsequent heat treatment. Since reactor coolant system piping is austenitic stainless steel, magnetic particle testing for surface examination is not an option. Fillet weld joints are examined by liquid penetrant examination method.

Radiographic examination is performed on circumferential butt welds and on branch connection nozzle welds exceeding 4-inch nominal pipe size.

The examination of a weld repair is repeated as required for the original weld. Except, when the defect was originally detected by the liquid penetrant method, and when the repair cavity does not exceed the lesser of 0.38 inch or 10 percent of the thickness, it need be re-examined only by the liquid penetrant method.

5.4.4 Main Steam Line Flow Restriction

5.4.4.1 Design Bases

The outlet nozzle of the steam generator has a flow restrictor that limits steam flow in the unlikely event of a break in the main steam line. A large increase in steam flow results in choked flow in the restrictor which limits further increase in flow. In a steam line qualified for mechanistic pipe break, a sudden rupture resulting in a large increase in steam flow is not expected. The flow restrictor performs the following functions:

- Limits rapid rise in containment pressure
- Limits the rate of heat removal from the reactor to keep the cooldown rate within acceptable limits

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- Reduces thrust forces on the main steam line piping
- Limits pressure differentials on internal steam generator components, particularly the tube support plates

The restrictor is configured to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

The flow restrictor consists of seven nickel-chromium-iron Alloy 690 (ASME SB-564) venturi inserts which are installed in holes in an integral steam outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle, and the other six are equally spaced around it. After insertion into the nozzle forging holes, the venturi inserts are welded to the nickel-chromium-iron alloy buttering on the inner surface of the forging.

5.4.4.3 Design Evaluation

The flow restrictor design has been analyzed to determine its structural adequacy. The equivalent throat area of the steam generator outlet is 1.4 square feet. The resultant pressure drop through the restrictor at 100 percent steam flow is approximately 20 psi. This is based on a design flow rate of 7.50×10^6 pounds per hour. The flow restrictor is AP1000 Equipment Class B, ANS Safety Class 2, and seismic Category I; but is constructed to ASME Code, Section III, Class 1 requirements. Materials of construction of the flow restrictor are in accordance with Section II of the ASME Code. The material of the inserts is not an ASME Code pressure boundary, nor is it welded to an ASME Code pressure boundary. The method for seismic analysis is dynamic.

5.4.4.4 Inspections

Since the restrictor is not part of the steam system pressure boundary, inservice inspections are not required.

5.4.5 Pressurizer

The pressurizer provides a point in the reactor coolant system where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control of the reactor coolant system during steady-state operations and transients. The pressurizer provides a controlled volume from which level can be measured.

The pressurizer contains the water inventory used to maintain reactor coolant system volume in the event of a minor system leak for a reasonable period without replenishment. The pressurizer surge line connects the pressurizer to one reactor coolant hot leg. This allows continuous coolant volume and pressure adjustments between the reactor coolant system and the pressurizer.

5.4.5.1 Design Bases

The pressurizer is sized to meet following requirements:

- The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.
- The water volume is sufficient to prevent a reactor trip during a step-load increase of 10 percent of full power, with automatic reactor control.

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- The water volume is sufficient to prevent uncovering of the heaters following reactor trip and turbine trip, with normal operation of control systems and no failures of nuclear steam supply systems.

- The steam volume is large enough to accommodate the surge resulting from a step load reduction from 100 percent power to house loads without reactor trip, assuming normal operation of control systems.

- The steam volume is large enough to prevent water relief through the safety valves following a complete loss of load with the high water level initiating a reactor trip, without steam dump.

- A Low-3 pressurizer pressure engineered safety features actuation signal will not be activated because of a reactor trip and turbine trip, assuming normal operation of control and makeup systems and no failures of the nuclear steam supply systems.

The pressurizer is sized to have sufficient volume to accomplish the preceding requirements without power-operated relief valves. The AP1000 pressurizer has approximately 40 percent more volume than the pressurizers for previous plants with similar power levels. This increased volume provides plant operating flexibility and minimizes challenges to the safety relief valves.

The pressurizer and surge line provide the connection of the reactor coolant system to the safety relief valves and the automatic depressurization system valves. The safety relief valves provide overpressure protection for the reactor coolant system. The automatic depressurization system is provided to reduce reactor coolant system pressure in stages to allow stored water in the in-containment refueling water storage tank to flow into the reactor coolant system to provide cooling.

The pressurizer surge nozzle and the surge line between the pressurizer and one hot leg are sized to maintain the pressure drop between the reactor coolant system and the safety valves within allowable limits during a design discharge flow from the safety valves or the valves of the automatic depressurization system. Requirements for the surge line and piping connecting the pressurizer to safety and automatic depressurization valves is discussed in Subsection 5.4.3.

Section 3.2 discusses the AP1000 equipment classification, seismic category and ASME Code classification of the pressurizer. ASME Code and Code Case compliance is discussed in Subsection 5.2.1.

The design stress limits, loads, and combined loading conditions are discussed in Subsection 3.9.3. Design transients for the components of the reactor coolant system are discussed in Subsection 3.9.1. The pressurizer surge nozzle and surge line are designed to withstand the thermal stresses resulting from volume surges occurring during operation. The evaluation of design transients for the pressurizer addresses the potential for thermal stratification at the surge nozzle.

The pressurizer provides a location for high point venting of noncondensable gases from the reactor coolant system. The gas accumulations in the pressurizer can be removed by remote manual operation of the first-stage automatic depressurization system valves following an accident. Degassing of the pressurizer using the automatic depressurization valves will not be required on a routine basis for normal and moderate frequency events. See Subsection 5.4.12 for a discussion of high-point vents.

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5.4.5.2 Design Description

5.4.5.2.1 Pressurizer

The pressurizer is a vertical, cylindrical vessel having hemispherical top and bottom heads constructed of low alloy steel. Internal surfaces exposed to the reactor coolant are clad austenitic stainless steel. Material specifications are provided in Table 5.2-1 for the pressurizer.

The general configuration of the pressurizer is shown in Figure 5.4-5. The design data for the pressurizer are given in Table 5.4-9. Codes and material requirements are provided in Section 5.2. Nickel-chromium-iron alloys are not used for heater wells or instrument nozzles.

The spray line nozzles and the automatic depressurization and safety valve connections are located in the top head of the pressurizer vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves can also be operated manually from the control room. In the bottom head at the connection of the surge line to the surge nozzle a thermal sleeve protects the nozzle from thermal transients.

A retaining screen above the surge nozzle prevents passage of any foreign matter from the pressurizer to the reactor coolant system. A baffle in the lower section of the pressurizer prevents an insurge of cold water from flowing directly to the steam/water interface. The baffle also assists in mixing the incoming water with the water in the pressurizer. The retaining screen and baffle also act as a diffuser. The baffle also supports the heaters to limit vibration.

Electric direct-immersion heaters are installed in vertically oriented heater wells located in the pressurizer bottom head. The heater wells are welded to the bottom head and form part of the pressure boundary. The heaters can be removed for maintenance or replacement.

The heaters are grouped into a control group and backup groups. The heaters in the control group are proportional heaters which are supplied with continuously variable power to match heating needs. The heaters in the backup group are either off or at full power. The power supply to the heaters is a 480-volt 60 Hz three-phase circuit. Each heater is connected to one leg of a delta-connected circuit and is rated at 460 volts with one-phase current. The capacity of the control and backup groups is defined in Table 5.4-10.

A manway in the upper shell provides access to the internal space of the pressurizer in order to inspect or maintain the spray nozzle. The manway closure is a gasketed cover held in place with threaded fasteners. Periodic planned inspections of the pressurizer interior are not required.

Brackets on the upper shell attach the structure (a ring girder) of the pressurizer safety and relief valve (PSARV) module. The pressurizer safety and relief valve module includes the safety valves and the first three stages of automatic depressurization system valves. The support brackets on the pressurizer represent the primary vertical load path to the building structure. Sway struts between the ring girder and pressurizer compartment walls also provide lateral support to the upper portion of the pressurizer. See Subsection 5.4.10 for additional details.

Four steel columns attach to pads on the lower head to provide vertical support for the vessel. The columns are based at elevation 107'-2". Lateral support for the lower portion of the vessel is provided by sway struts between the columns and compartment walls.

The AP1000 pressurizer has metallic reflective insulation (MRI) installed on the external surfaces; the insulation is designed to minimize heat losses from the pressurizer, to reduce heat load on the containment cooling system, and to limit temperatures in nearby concrete or components. During

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normal operating conditions, the insulation has an average maximum heat transfer rate of 65 Btu/hr-ft^2 at a containment design temperature of 120°F.

5.4.5.2.2 Instrumentation

Instrument connections are provided in the pressurizer shell to measure important parameters. Eight level taps are provided for four channels of level measurement. Level taps are also used for connection to the pressure measurement instrumentation. Two temperature taps monitor water/steam temperature. A sample tap connection is provided for connection to the sampling system to monitor coolant chemistry. The instrument and sample taps are constructed of stainless steel and designed for a socket weld of the connecting lines to the taps. The sample and instrument taps incorporate an integral flow restrictor with a diameter of 0.38 inch or smaller.

See Chapter 7 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

5.4.5.2.3 Operation

During steady-state operation at 100 percent power, approximately 50 percent of the pressurizer volume is water and 50 percent is steam. Electric immersion heaters in the bottom of the vessel keep the water at saturation temperature. The heaters also maintain a constant operating pressure.

A small continuous spray flow is provided through a manual bypass valve around each power-operated spray valve to minimize the boron concentration difference between the pressurizer liquid and the reactor coolant. This continuous flow also prevents excessive cooling of the spray piping. Proportional heaters in the control group are continuously on during normal operation to compensate for the continuous introduction of cooler spray water and for losses to ambient.

These conditions result in a continuous outsurge in most cases during normal operation and anticipated transients. The outsurge minimizes the potential for thermal stratification in the surge line.

During an outsurge of water from the pressurizer, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the Low-3-pressure engineered safety features actuation setpoint. During an insurge from the reactor coolant system, the spray system (which is fed from two cold legs) condenses steam in the pressurizer. This prevents the pressurizer pressure from reaching the High-2-pressure reactor trip setpoint.

During heatup and cooldown of the plant, when the potential for thermal stratification in the pressurizer is the greatest, the pressurizer may be operated with a continuous outsurge of water from the pressurizer. This is achieved by continuous maximum spray flow and energizing of all of the backup pressurizer heater groups. The temperature difference between the pressurizer and hot leg is minimized by maintaining the lowest reactor coolant system pressure possible consistent with operation of a reactor coolant pump. This mode of operation minimizes the frequency and magnitude of thermal shock to the surge line nozzle and lower pressurizer head, and the potential for stratification in the pressurizer and surge line. The design analyses of the pressurizer include consideration of transients on the lower head and shell regions to account for these possible insurge/outsurge events.

The pressurizer is the initial source of water to keep the reactor coolant system full of water in the event of a small loss of coolant. Pressurizer level and pressure measurements indicate if other sources of water, including the chemical and volume control system and passive safety systems, must be used to supply additional reactor coolant.

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Power to the pressurizer heaters is blocked when the core makeup tanks are actuated. This action reduces the potential for steam generator overfill for a steam generator tube rupture accident.

5.4.5.3 Design Evaluation

5.4.5.3.1 System Pressure Control

The reactor coolant system pressure is controlled by the pressurizer whenever a steam volume is present in the pressurizer.

A design basis safety limit has been set so that the reactor coolant system pressure does not exceed the maximum transient value based on the design pressure as allowed under the ASME Code, Section III. Evaluation of plant conditions of operation considered for design indicates that this safety limit is not reached. The safety valves provide overpressure protection. See Subsection 5.2.2.

During startup and shutdown, the rate of temperature change in the reactor coolant system is controlled automatically by the steam dump system. Heatup rate is controlled by energy input from the reactor coolant pumps and by the modulation of the steam dump valves. Pressurizer heatup rate is controlled by the electrical heaters in the pressurizer.

When the pressurizer is filled with water, i.e., during initial system heatup or near the end of the second phase of plant cooldown, reactor coolant system pressure is controlled by the letdown flowrate.

The AP1000 pressurizer heaters are powered from the 480 V ac system. During loss of offsite power events concurrent with a turbine trip, selected pressurizer heater buses are capable of being powered from the onsite diesel generators via manual alignment. This permits use of the pressurizer for control purposes when power is supplied by the diesel-generators. The power supplied by the diesel-generators is sufficient to establish and maintain natural circulation in hot standby condition in conformance with the requirement of 10 CFR 50.34 (f)(2)(xiii).

If loss of offsite power occurs and onsite power is available, the pressurizer heaters and startup feedwater pumps can operate to provide natural circulation and cooling through the steam generators.

Should the onsite diesel generators not be available during loss of offsite power events, core decay heat is removed from the reactor coolant system using the passive residual heat removal heat exchanger. The decay heat is transferred to the in-containment refueling water storage tank (IRWST) water. The passive core cooling system does not require the use of pressurizer heaters to maintain pressure control. The passive containment cooling system functions to maintain the plant in a safe condition.

NUREG-0737, Action Item II.E.3.1, outlines four requirements for emergency power supply to the pressurizer heaters for purposes of establishing natural circulation in the reactor coolant system during a loss of offsite power. NUREG-0737 does not address scenarios involving natural circulation cooling for a loss of all ac power, which is a design basis for the AP1000. Under these circumstances, cooling is provided by the passive residual heat removal system. Upon a loss of all ac power, the heaters are not available to maintain the pressurizer inventory in a saturated condition. That condition is not needed for the plant to be maintained in a safe condition. On this basis, compliance with the requirements of the action item is not required to provide for the safety of the AP1000. Nevertheless, AP1000 compliance with the intent of these requirements is summarized in the following paragraphs.

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The heaters are powered from separate electrical buses for each heater group. Two groups of heaters can be administratively loaded onto the non-Class 1E diesel-generator-backed buses (Figure 8.3.1-1).

Analysis of AP1000 steady-state heat losses indicates that a heater capacity of about 166 kW is sufficient to provide saturated conditions in the pressurizer. Each AP1000 heater group has a capacity greater than 166 kW (see Table 5.4-10). One group alone can maintain control over reactor coolant system pressure and subcooling.

Established administrative procedures are followed for re-energizing groups. Associated actions can be controlled from either the main control room or the shutdown panel. It is not necessary to shed other loads in order to manually load a heater group.

Based on analysis of other pressurizer water reactors, the reactor coolant system sensible heat capacity is such that adequate subcooling can be maintained in the reactor coolant system for four hours without heat input from the pressurizer heaters. Thus, the time required to accomplish connection of the heaters to the emergency buses is consistent with timely initiation of natural circulation conditions.

Since the buses supplying the heaters for the diesel generators are not Class 1E, the 480 V breakers supplying the heaters are not required to be “qualified in accordance with safety-related requirements.”

5.4.5.3.2 Pressurizer Level Control

The normal operating water volume at full-load conditions is approximately 50 percent of the free internal vessel volume. Under part-load conditions the water volume in the pressurizer is reduced proportionally with reductions in plant load to approximately 25 percent of the free internal vessel volume at the zero-power condition.

5.4.5.3.3 Pressure Setpoints

The reactor coolant system design and operating pressure, together with the safety valve setpoints, heater actuation setpoints, pressurizer spray valve setpoints, and protection system pressure setpoints, are listed in Table 5.4-11. When operating in load regulation mode, the pressurizer spray and backup heaters are on continuously. This continuous operation decreases the number of actuations of the backup heaters and spray valves, thereby extending the component lifetimes.

The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressure allows for operating transient pressure changes.

The Low-3 pressurizer pressure engineered safety features actuation signal does not require a coincident low pressurizer water level signal.

5.4.5.3.4 Pressurizer Spray

Two separate, automatically controlled spray valves with remote manual overrides are used to initiate pressurizer spray.

In parallel with each spray valve is a manual throttle valve. The throttle permits a small, continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open. Flow through this valve helps to maintain uniform water chemistry and temperature in the

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pressurizer. Temperature sensors with low temperature alarms are located in each spray line to alert the operator to insufficient bypass flow.

The layout of the common spray line piping routed to the pressurizer forms a water seal that prevents steam buildup back to the control valves. The design spray rate is selected to prevent the pressurizer pressure from reaching the reactor trip setpoint during a step reduction in power level of 10 percent of full load.

The pressurizer spray lines and valves are large enough to provide the required spray flowrate under the driving force of the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop in order to use the velocity head of the reactor coolant loop flow to add to the spray driving force. The spray line also assists in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

A flowpath from the chemical and volume control system to the pressurizer spray line is also provided. This path provides auxiliary spray to the vapor space of the pressurizer during cooldown, hot standby, and hot shutdown when the reactor coolant pumps are not operating. The pressurizer spray connection and the spray piping can withstand the thermal stresses resulting from the introduction of cold spray water.

5.4.5.4 Tests and Inspections

The pressurizer construction is subject to a quality assurance program. The pressure boundary components meet requirements established by the ASME Code and ASME NQA-1. Table 5.4-12 outlines the testing included in the pressurizer quality assurance program.

The design of the pressurizer permits the inspection program prescribed by the ASME Code, Section XI. To implement the requirements of the ASME Code, Section XI, the following welds, when present, are designed and constructed to present a smooth transition surface between the parent metal and the weld metal. The weld surface is ground smooth for ultrasonic inspection.

- Surge nozzle to the lower head
- Safety and spray nozzles to the upper head
- Nozzle to safe end attachment welds
- The girth full-penetration welds

The liner within the safe end nozzle region extends beyond the weld region to maintain a uniform geometry for ultrasonic inspection.

Peripheral support rings are furnished for the removable insulation modules.

5.4.6 Automatic Depressurization System Valves

The automatic depressurization system (ADS) valves are part of the reactor coolant system and interface with the passive core cooling system (PXS). Twenty valves are divided into four depressurization stages. These stages connect to the reactor coolant system at three different locations. The automatic depressurization system first, second, and third stage valves are included as part of the pressurizer safety and relief valve (PSARV) module and are connected to nozzles on top of the pressurizer. The fourth stage valves connect to the hot leg of each reactor coolant loop. The reactor coolant system P&ID, Figure 5.1-5, shows the arrangement of the valves.

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Opening of the automatic depressurization system valves is required for the passive core cooling system to function as required to provide emergency core cooling following postulated accident conditions. Operation of the passive core cooling system, including setpoints for the opening of the automatic depressurization system valves is discussed in Section 6.3.

The first stage valves may also be used, as required following an accident, to remove noncondensable gases from the steam space of the pressurizer. (See Subsection 5.4.11.)

5.4.6.1 Design Bases

Subsection 5.4.8 discusses the general design basis, design evaluation, and testing and inspection for reactor coolant system valves, including the automatic depressurization system valves. The automatic depressurization system valves are seismic Category 1, AP1000 equipment Class A components. (See Subsection 3.2.2.) The fourth stage valves are interlocked so that they can not be opened until reactor coolant system pressure has been substantially reduced. The design criteria and bases, functional requirements, mechanical design, and testing and inspection of the passive core cooling system are included in Section 6.3. The design requirements for the passive core cooling system also apply to automatic depressurization valves except where the requirements for reactor coolant system valves are more restrictive.

5.4.6.2 Design Description

The first stage automatic depressurization system valves are motor-operated 4-inch valves. The second and third stage automatic depressurization system valves are motor-operated 8-inch valves. The fourth stage automatic depressurization system valves are 14 inch squib valves arranged in series with normally-open, dc powered, motor-operator valves. See Section 6.3 for a discussion of the sizing of the automatic depressurization system valves.

The control system for the opening of the automatic depressurization system valves, as part of the passive core cooling system, has an appropriate level of diverse and redundant features to minimize the inadvertent opening of the valves.

For each stage 1-3 discharge path a pair of valves are placed in series to minimize the potential for an inadvertent discharge of the automatic depressurization system valves. The fourth stage valves are interlocked so that they cannot be opened until reactor coolant system pressure has been substantially reduced.

The first, second, and third stage valves are located on the pressurizer safety and relief valve module clustered into two groups. Each group has one pair of valves for each stage. The two groups are on different elevations and are separated by a steel plate.

Vacuum breakers are provided in the AP1000 ADS discharge lines to help prevent water hammer following ADS operation. The vacuum breakers limit the pressure reduction that could be caused by steam condensation in the discharge line and thus limit the potential for liquid backflow from the in-containment refueling water storage tank following ADS operation.

A bypass test line is connected to the inlet and outlet of the first, second, and third stage upstream isolation valves. This bypass line can control the differential pressure across the upstream valves during inservice testing. The bypass test solenoid valves do not have a safety-related function to open.

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5.4.6.3 Design Verification

The automatic depressurization system valves are verified to meet their safety-related functional requirements by the following:

- Valve equipment qualification
- Pre-operational valve operational verification
- In-service valve operational verification

Automatic depressurization system valve qualification is addressed in Subsection 5.4.8.1.2 for the stage 1/2/3 motor operated valves and in Subsection 5.4.8.1.3 for the stage 4 squib valves. The equipment qualification includes type testing which verifies the automatic depressurization system valve operability and flow capacity. Automatic depressurization system valve pre-operational valve operational verification is addressed in Subsection 14.2.9.1. Automatic depressurization system valve in-service valve operational verification is addressed in Subsection 3.9.6.

5.4.6.4 Inspection and Testing Requirements

The requirements for tests and inspections for reactor coolant system valves is found in Subsection 5.4.8.4. In addition, tests for the automatic depressurization system valves and piping are conducted during preoperational testing of the passive core cooling system, as discussed in Sections 6.3 and 14.2.

5.4.6.4.1 Flow Testing

Initial verification of the resistance of the automatic depressurization system piping and valves is performed during the plant initial test program. A low pressure flow test and associated analysis is conducted to determine the total piping flow resistance of each automatic depressurization system valve group connected to the pressurizer (i.e. stages 1-3) from the pressurizer through the outlet of the downstream valve. The reactor coolant system shall be at cold conditions with the pressurizer full of water. The normal residual heat removal pumps will be used to provide injection flow into the reactor coolant system, discharging through the ADS valves.

Inspections and associated analysis of the piping flow paths from the discharge of the automatic depressurization system valve groups connected to the pressurizer (i.e., stages 1-3) to the spargers are conducted to verify the line routings are consistent with the line routings used for design flow resistance calculations. The calculated piping flow resistances from the pressurizer through the sparger, with valves of each group open are bounded by the resistances used in the Chapter 15 safety analysis.

Inspection of the piping flow paths from each hot leg through the automatic depressurization stage 4 valves is conducted. The calculated flow resistances with valves in each group open are bounded by the resistances used in the Chapter 15 safety analysis.

5.4.7 Normal Residual Heat Removal System

The normal residual heat removal system (RNS) performs the following major functions:

- **Reactor Coolant System Shutdown Heat Removal** - Remove heat from the core and the reactor coolant system during shutdown operations.

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- **Shutdown Purification** - Provide reactor coolant system and refueling cavity purification flow to the chemical and volume control system during refueling operations.
- **In-containment Refueling Water Storage Tank Cooling** - Provide cooling for the in-containment refueling water storage tank.
- **Reactor Coolant System Makeup** - Provide low pressure makeup to the reactor coolant system.
- **Post-Accident Recovery** - Remove heat from the core and the reactor coolant system following successful mitigation of an accident by the passive core cooling system.
- **Low Temperature Overpressure Protection** - Provide low temperature overpressure protection (LTOP) for the reactor coolant system during refueling, startup, and shutdown operations.
- **Long-Term, Post-Accident Containment Inventory Makeup Flowpath** - Provide long-term, post-accident makeup flowpath to the containment inventory.
- **Spent Fuel Pool Cooling** - Provide backup for cooling the spent fuel pool.

5.4.7.1 Design Bases

5.4.7.1.1 Safety Design Bases

The safety-related functions provided by the normal residual heat removal system include containment isolation of normal residual heat removal system lines penetrating containment, preservation of the reactor coolant system pressure boundary and a flow path for long term post-accident makeup to the containment inventory. The containment isolation valves perform the containment isolation function according to the criteria specified in Subsection 6.2.3. The system preserves the reactor coolant system pressure boundary according to the criteria specified in Subsection 5.4.8.

The normal residual heat removal system piping and components outside containment are an AP1000 Class C, Seismic Category I pressure boundary. This classification recognizes the importance of pressure boundary integrity even though these components have no safety-related functions.

5.4.7.1.2 Nonsafety Design Bases

Subsection 5.4.7 outlines the principal functions of the normal residual heat removal system. The normal residual heat removal system is designed to be reliable. This reliability is achieved by using redundant equipment and a simplified system design. The normal residual heat removal system is not a safety-related system. It is not required to operate to mitigate design basis events.

The normal residual heat removal system is designed for a single nuclear power unit and is not shared between units. The normal residual heat removal system is operated from the main control room.

The normal residual heat removal system provides the capability to cool the spent fuel pool during times when it is not needed for removing heat from the reactor coolant system.

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5.4.7.1.2.1 Shutdown Heat Removal

The normal residual heat removal system removes both residual and sensible heat from the core and the reactor coolant system. It reduces the temperature of the reactor coolant system during the second phase of plant cooldown. The first phase of cooldown is accomplished by transferring heat from the reactor coolant system via the steam generators to the main steam system (MSS).

Following cooldown, the normal residual heat removal system removes heat from the core and the reactor coolant system during the plant shutdown, until the plant is started up.

The normal residual heat removal system reduces the temperature of the reactor coolant system from 350° to 125°F within 96 hours after shutdown. The system maintains the reactor coolant temperature at or below 125°F for the plant shutdown. The system performs this function based on the following:

- Operation of the system with both subsystems of normal residual heat removal system pumps and heat exchangers available.

- Initiation of normal residual heat removal system operation at four hours following reactor shutdown, after the first phase of cooldown by the main steam system has reduced the reactor coolant system temperature to less than or equal to 350°F and 450 psig.

- The component cooling water system supply temperature to the normal residual heat removal system heat exchangers is based on maximum normal ambient wet bulb temperature as defined in Chapter 2, Table 2-1. The maximum normal ambient temperature is assumed for shutdown cooling.

- Operation of the system is consistent with reactor coolant system cooldown rate limits and consistent with maintaining the component cooling water below design limits during cooldown.

- Core decay heat generation is based on the decay heat curve for a three-region core having burnups consistent with a 24-month or 18-month refueling schedule and based on the ANSI/ANS-5.1-1994 decay heat curve (Reference 5).

- A failure of an active component during normal cooldown does not preclude the ability to cool down, but lengthens the time required to reach 125°F. Furthermore, if such a single failure occurs while the reactor vessel head is removed, the reactor coolant temperature remains below boiling temperature.

- The system operates at a constant normal residual heat removal flow rate throughout refueling operations. This includes the time when the level in the reactor coolant system is reduced to a midloop level to facilitate draining of the steam generators or removal of a reactor coolant pump. Operation of the system at the minimum level that the reactor coolant system can attain using the normal reactor coolant system draining connections and procedures results in no incipient vortex formation which would cause air entrainment into the pump suction.

- The pump suction line is self-venting to a condition acceptable for RNS pump operation such that no manual venting is required to start the pumps if stopped on cavitation due to excessive air entrainment. This is achieved by sloping most of the RNS pump suction piping upward from the RNS pump suction to the reactor coolant system hot leg. In the level portions of piping, there are no local high points. This arrangement minimizes entrapment of air and system venting efforts for startup.

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- Features are included that permit mid-loop operations to be performed from the main control room.

5.4.7.1.2.2 Shutdown Purification

The normal residual heat removal system provides reactor coolant system flow to the chemical and volume control system during refueling operations. The purification flow rate is consistent with the purification flow rate specified in Table 9.3.6-1.

5.4.7.1.2.3 In-Containment Refueling Water Storage Tank Cooling

The normal residual heat removal system provides cooling for the in-containment refueling water storage tank during operation of the passive residual heat removal heat exchanger or during normal plant operations when required. The system is manually initiated by the operator. The normal residual heat removal system limits the in-containment refueling water storage tank water temperature to less than boiling temperature during extended operation of the passive residual heat removal system and not greater than 120°F during normal operation. The system performs this function based on the following:

- Operation of the system with both subsystems of normal residual heat removal system pumps and heat exchangers available.

- The component cooling water system supply temperature to the normal residual heat removal system heat exchangers is based on an ambient design wet bulb temperature of no greater than 86.1°F (0 percent exceedance). The 86.1°F value is assumed for normal conditions and transients that start at normal conditions.

Since the normal residual heat removal system is not a safety-related system, its operation is not credited in Chapter 15 Accident Analyses.

5.4.7.1.2.4 Low Pressure Reactor Coolant System Makeup and Cooling

The normal residual heat removal system provides low pressure makeup from the cask loading pit to the reactor coolant system. The system is manually initiated by the operator following receipt of an automatic depressurization signal. If the system is available, it provides reactor coolant system makeup once the pressure in the reactor coolant system falls below the shutoff head of the normal residual heat removal system pumps. The system provides makeup from the cask loading pit to the reactor coolant system and provides additional margin for core cooling. The normal residual heat removal system is not required to mitigate design basis accidents, and therefore its operation is not considered in Chapter 15 Accident Analyses.

5.4.7.1.2.5 Low Temperature Overpressure Protection

The normal residual heat removal system provides a low temperature overpressure protection function for the reactor coolant system during refueling, startup, and shutdown operations. The system is designed to limit the reactor coolant system pressure to the lower of either the limits specified in 10 CFR 50, Appendix G, or 110 percent of the normal residual heat removal system design pressure.

5.4.7.1.2.6 Spent Fuel Pool Cooling

The normal residual heat removal system has the capability to supplement or take over the cooling of the spent fuel pool when it is not needed for normal shutdown cooling.

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5.4.7.2 System Description

Figure 5.4-6 shows a simplified sketch of the normal residual heat removal system. Figure 5.4-7 shows the piping and instrumentation diagram for the normal residual heat removal system. Table 5.4-13 gives the important system design parameters.

The inside containment portions of the system from the reactor coolant system up to and including the containment isolation valves outside containment are designed for full reactor coolant system pressure. The portion of the system outside containment, including the pumps, valves and heat exchangers, has a design pressure and temperature such that full reactor coolant system pressure is below the ultimate rupture strength of the piping.

The normal residual heat removal system consists of two mechanical trains of equipment. Each train includes one residual heat removal pump and one residual heat removal heat exchanger. The two trains of equipment share a common suction line from the reactor coolant system and a common discharge header. The normal residual heat removal system includes the piping, valves and instrumentation necessary for system operation.

The normal residual heat removal system suction header is connected to a reactor coolant system hot leg with a single step-nozzle connection. The step-nozzle connection is employed to minimize the likelihood of air ingestion into the residual heat removal pumps during reactor coolant system mid-loop operations. The suction header then splits into lines with two parallel sets of two normally closed, motor-operated isolation valves in series. This arrangement allows for normal residual heat removal system operation following a single failure of an isolation valve to open and also allows for normal residual heat removal system isolation following a single failure of an isolation valve to close.

The lines join into a common suction line inside containment. A single line from the in-containment refueling water storage tank is connected to the suction header before it leaves containment.

Once outside containment, the suction header contains a single normally closed, motor-operated isolation valve. Downstream of the suction header isolation valve, the header branches into two separate lines, one to each pump. Each branch line has a normally open, manual isolation valve upstream of the residual heat removal pumps. These valves are provided for pump maintenance.

The normal residual heat removal system suction header is designed to be self-venting to a condition acceptable for RNS pump operation such that no manual venting is required to start the pumps if stopped on cavitation due to excessive air entrainment. This is achieved by sloping most of the RNS pump suction piping upward from the RNS pump suction to the reactor coolant system hot leg. In the level portions of piping, there are no local high points. This eliminates any locations where a sufficient volume of air could collect to cause low net positive suction head, pump binding, and a loss of residual heat removal capability.

The discharge of each residual heat removal pump is directed to its respective residual heat removal heat exchanger. The outlet of each residual heat removal heat exchanger is routed to the common discharge header, which contains a normally closed, motor-operated isolation valve. For pump protection, a miniflow line with an orifice is included from downstream of the residual heat removal heat exchanger to upstream of the residual heat removal pump suction. This line is sized to provide sufficient pump flow when the pressure in the reactor coolant system is above the residual heat removal pump shutoff head.

Once inside containment, the common discharge header contains a check valve that acts as a containment isolation valve. Downstream of the check valve, the discharge header branches into two lines, one to each passive core cooling system direct vessel injection nozzle. These branch lines each contain a stop check valve and check valve in series that serve as the reactor coolant system

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pressure boundary. A line to the chemical and volume control system demineralizers branches from one of the direct vessel injection lines. This line is used for shutdown purification of the reactor coolant system. Another line branches from the same direct vessel injection line to the in-containment refueling water storage tank which is used when cooling the tank.

Two relief valves are located on the normal residual heat removal system suction header inside containment. These valves provide low temperature overpressure protection of the reactor coolant system. Subsection 5.4.9 describes the sizing basis of these valves. Another relief valve outside of containment provides protection against excess pressure for the piping and components.

When the normal residual heat removal system is in operation, the water chemistry is the same as that of the reactor coolant. Sampling may be performed using the normal residual heat removal heat exchangers channel head drain connections. Sampling of the reactor coolant system using these connections is available at shutdown. Sampling of the in-containment refueling water storage tank is available during normal plant operation.

5.4.7.2.1 Design Features Addressing Shutdown and Mid-Loop Operations

The following is a summary of the specific AP1000 design features that address Generic Letter (GL) 88-17 regarding mid-loop operations. In addition, these features support improved safety during shutdown.

Loop Piping Offset - As shown in Figure 5.3-6, the reactor coolant system hot legs and cold legs are vertically offset. This permits draining of the steam generators for nozzle dam insertion with hot leg level much higher than traditional designs. The reactor coolant system must be drained to a level which is sufficient to provide a vent path from the pressurizer to the steam generators. This is nominally 80 percent level in the hot leg. This loop piping offset also allows a reactor coolant pump to be replaced without removing a full core.

Step-nozzle Connection - The normal residual heat removal system employs a step-nozzle connection to the reactor coolant system hot leg. The step-nozzle connection has two effects on mid-loop operation. One effect is to substantially lower the RCS hot leg level at which a vortex occurs in the residual heat removal pump suction line due to the lower fluid velocity in the hot leg nozzle. This increases the margin from the nominal mid-loop level to the level where air entrainment into the pump suction begins.

Another effect of the step-nozzle is that, if a vortex should occur, the maximum air entrainment into the pump suction has been shown experimentally to be no greater than 5 percent. This level of air ingestion will make air binding of the pump much less likely.

Normal Residual Heat Removal Throttling During Mid-Loop - The normal residual heat removal pumps are designed to minimize susceptibility to cavitation. Normally, the normal residual heat removal system operates without the need for throttling a residual heat removal control valve when the level in the reactor coolant system is reduced to a mid-loop level. If the reactor coolant system is at saturated conditions and mid-loop level, some throttling of a flow control valve is necessary to maintain adequate net positive suction head.

Self-Venting Suction Line - The residual heat removal pump suction line is designed to be self-venting to a condition acceptable for RNS pump operation such that no manual venting is required to restart the pumps if stopped on cavitation due to excessive air entrainment. This is achieved by sloping most of the RNS pump suction piping, upward from the RNS pump to the reactor coolant system hot leg. In the level portions of piping, there are no local high points. This eliminates potential problems with refilling the pump suction line if a residual heat removal pump is stopped when cavitating due to excessive air entrainment. With the self-venting suction line, the line will refill

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without manual venting and the pumps can be immediately (no additional operator action required) restarted once an adequate level in the hot leg is re-established.

Wide Range Pressurizer Level - A nonsafety-related independent pressurizer level transmitter, calibrated for low temperature conditions, provides water level indication during startup, shutdown, and refueling operations in the main control room and at the remote shutdown workstation. The upper level tap is connected to an ADS valve inlet header above the top of the pressurizer. The lower level tap is connected to the bottom of the hot leg. This provides level indication for the entire pressurizer and a continuous reading as the level in the pressurizer decreases to mid-loop levels during shutdown operations.

Hot Leg Level Instrumentation - The AP1000 reactor coolant system contains level instrumentation in each hot leg with indication in the main control room. In addition to the wide-range pressurizer level instrumentation (used during cold plant operation) which provides continuous level indication in the main control room from the normal level in the pressurizer, two narrow-range hot leg level instruments are available. Alarms are provided to alert the operator when the reactor coolant system hot leg level is approaching a low level. The isolation valves in the line used to drain the reactor coolant system close on a low reactor coolant system level during shutdown operations. Operations required during mid-loop are performed by the operator in the main control room. The level monitoring and control features significantly improve the reliability of the AP1000 during mid-loop operations.

Reactor Vessel Outlet Temperature - Reactor coolant system hot leg wide range temperature instruments are provided in each hot leg for normal residual heat removal system operation with normal inventory. The normal residual heat removal temperature instruments, upstream of the heat exchangers, indicate reactor coolant system hot leg temperature when in reduced inventory conditions. In addition, at least two incore thermocouple channels are available to measure the core exit temperature during midloop residual heat removal operation. These two thermocouple channels are associated with separate electrical divisions.

ADS Valves - The automatic depressurization system first-, second-, and third-stage valves, connected to the top of the pressurizer, are open whenever the core makeup tanks are blocked during shutdown conditions while the reactor vessel upper internals are in place. This provides a vent path that, when combined with actuation of the automatic depressurization system fourth-stage valves, sufficient steam venting area is provided to reduce reactor coolant system pressure after decay heat removal is lost. Actuation of ADS stage 4, either automatically or manually (as described in Subsection 19E.4.8.3) allows the in-containment water storage tank to provide injection flow following a loss of decay heat removal.

The capability to restore containment integrity during shutdown conditions is provided. The containment equipment hatches are equipped with guide rails that allow reinstallation of the hatches to re-establish containment integrity. The containment design also includes penetrations for temporary cables and hoses needed for shutdown operations.

Procedures direct the operator in the proper conduct of midloop operation and aid in identifying and correcting abnormal conditions that might occur during shutdown operations.

5.4.7.2.2 Design Features Addressing Intersystem LOCA

The AP1000 has addressed the intersystem LOCA section of SECY 90-016 with a number of design features. These design features are:

Codes and Standards/Seismic Protection - The portions of the normal residual heat removal system located outside containment (that serve no active safety functions) are classified as AP1000

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Equipment Class C so that the design, manufacture, installation, and inspection of this pressure boundary is in accordance with the following industry codes and standards and regulatory requirements: 10 CFR 50, Appendix B; Regulatory Guide 1.26 Quality Group C; and ASME Boiler and Pressure Vessel Code, Section III, Class 3. The pressure boundary is classified as Seismic Category I.

Increased Design Pressure - The portions of the normal residual heat removal system from the reactor coolant system to the containment isolation valves outside containment are designed to the operating pressure of the reactor coolant system. The portions of the system downstream of the suction line containment isolation valve and upstream of the discharge line containment isolation valve are designed so that its ultimate rupture strength is not less than the operating pressure of the reactor coolant system. Specifically, the piping is designed as schedule 80S, and the flanges, valves, and fittings are specified to be greater than or equal to ANS class 900. The design pressure of the normal residual heat removal system is 900 psi, which is approximately 40 percent of operating reactor coolant system pressure.

Reactor Coolant System Isolation Valve - The AP1000 normal residual heat removal system contains an isolation valve in the pump suction line from the reactor coolant system. This motor-operated containment isolation valve is designed to the reactor coolant system pressure. It provides an additional barrier between the reactor coolant system and lower pressure portions of the normal residual heat removal system.

Normal Residual Heat Removal System Relief Valves - The inside containment AP1000 normal residual heat removal system relief valves are connected to the residual heat removal pump suction line. These valves are designed to provide low-temperature overpressure protection of the reactor coolant system as described in Subsection 5.2.2. They are connected to the high pressure portion of the pump suction line and reduces the risk of overpressurizing the low pressure portions of the system.

Features Preventing Inadvertent Opening of Isolation Valves - The reactor coolant system isolation valves are interlocked to prevent their opening at reactor coolant system pressures above 450 psig. Section 7.6 discusses this interlock. The power to these valves is administratively blocked during normal power operation.

RCS Pressure Indication and High Alarm - The AP1000 Normal residual heat removal system contains an instrumentation channel that indicates pressure in each normal residual heat removal pump suction line. A high pressure alarm is provided in the main control room to alert the operator to a condition of rising RCS pressure that could eventually exceed the design pressure of the normal residual heat removal system.

Closed valves connecting to spent fuel pool - The cross-connecting piping between the normal residual heat removal system and the spent fuel pool cooling system is isolated by normally closed valves.

5.4.7.3 Component Description

The descriptions of the normal residual heat removal system components are provided in the following subsections. Table 5.4-14 lists the key equipment parameters for the normal residual heat removal system components.

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5.4.7.3.1 Normal Residual Heat Removal Pumps (MP01 A&B)

Two residual heat removal pumps are provided. These pumps are single stage, vertical in-line, bottom suction centrifugal pumps. They are coupled with a motor shaft driven by an ac powered induction motor.

Each pump is sized to provide the flow required by its respective heat exchanger for removal of its design basis heat load. Redundant pumps and heat exchangers provide sufficient cooling to prevent RCS boiling if one subsystem is inoperative. A continuously open miniflow line is also provided to protect the pump from operation at low flow conditions.

5.4.7.3.2 Normal Residual Heat Removal Heat Exchangers (ME01 A&B)

Two residual heat removal heat exchangers are installed to provide redundant residual heat removal capability. These heat exchangers are vertically mounted, shell and U-tube design. Reactor coolant flow circulates through the stainless steel tubes while component cooling water circulates through the carbon steel shell. The tubes are welded to the tubesheet.

5.4.7.3.3 Normal Residual Heat Removal Valves

The normal residual heat removal system packed valves designated for radioactive service are provided with stem packing designs that provide enhanced resistance to leakage. Leakage to the atmosphere is essentially zero for these valves.

Manual and motor-operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. The basic material of construction for valves is stainless steel.

Two manually operated valves that individually isolate each of the RNS trains are locked-open to prevent thermal overpressurization of the RNS piping in the auxiliary building. There is one additional valve locked-open to maintain a path for the RNS discharge containment isolation check valve to provide overpressure protection for the containment penetration piping and provide a flow path for post-accident makeup to passive core cooling.

5.4.7.3.3.1 Reactor Coolant System Inner/Outer Isolation Valves (V001 A&B, V002 A&B)

There are two parallel sets of two valves in series for a total of four valves. These valves are normally closed, motor-operated valves and are located inside the containment. These valves form the reactor coolant pressure boundary. They are opened only for normal cooldown after reactor coolant system depressurization to 450 psig. They are controlled from the main control room and fail in the “as-is” position. These valves are protected from inadvertently opening when the reactor coolant system pressure is above 450 psig by an interlock. Power to these valves is administratively blocked during normal power operations. The outer isolation valves, V002A & V002B, also act as containment isolation valves.

5.4.7.3.3.2 In-Containment Refueling Water Storage Tank Suction Line Isolation Valve (V023)

There is one motor-operated valve located inside containment in the line from the in-containment refueling water storage tank to the pump suction header. This valve is designed for full reactor coolant system pressure. It also acts as a containment isolation valve.

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5.4.7.3.3.3 Residual Heat Removal Isolation Valve (V011)

There is one motor-operated valve in the pump discharge header outside of containment. This valve is designed for full reactor coolant system pressure. It also acts as a containment isolation valve.

5.4.7.3.3.4 In-Containment Refueling Water Storage Tank Return Isolation Valve (V024)

There is one normally closed motor-operated valve located inside containment in the discharge line to the in-containment refueling water storage tank. This valve is aligned for full-flow testing of the residual heat removal pumps or for operations involving cooling of the in-containment refueling water storage tank.

5.4.7.3.3.5 Cask Loading Pit Isolation Valve (V055)

There is one normally closed motor-operated valve in the line between the cask loading pit and the residual heat removal pump suction line. This valve can be opened by the operator to provide low pressure injection from the cask loading pit to the reactor coolant system during an accident.

5.4.7.3.3.6 Normal Residual Heat Removal Pump Miniflow Isolation Valves (V057A&B)

There is one normally open air-operated valve in each of the residual heat removal pump miniflow lines. During plant cooldown the operator can close these valves to increase the circulating flow rate of the reactor coolant through the residual heat removal heat exchangers to decrease the reactor coolant system cooldown time. These valves automatically open on low flow in the residual heat removal heat exchanger discharge line.

5.4.7.4 System Operation and Performance

Operation of the normal residual heat removal system is described in the following sections. System operations are controlled and monitored from the main control room, including mid-loop operations. The reactor coolant system is equipped with mid-loop level instrumentation with remote readout in the main control room. This instrumentation is used for monitoring mid-loop operations from the main control room.

5.4.7.4.1 Plant Startup

Plant startup includes the operations that bring the reactor plant from a cold shutdown condition to no-load operating temperature and pressure, and subsequently to power operation.

During cold shutdown conditions, both residual heat removal pumps and heat exchangers operate to circulate reactor coolant and remove decay heat. The residual heat removal pumps are switched off when plant startup begins. The normal residual heat removal system remains aligned to the reactor coolant system to maintain a low pressure letdown path to the chemical and volume control system. This alignment provides reactor coolant system purification flow and low temperature over-pressure protection of the reactor coolant system. As the reactor coolant pumps are started, their thermal input begins heating the reactor coolant inventory. Once the pressurizer steam bubble formation is complete, the normal residual heat removal system suction header isolation valve and the discharge header isolation valve are closed and tested for leakage. The valve arrangement is then set for normal operation, as shown in Figure 5.4-6.

5.4.7.4.2 Plant Cooldown

Plant cooldown is the operation that brings the reactor plant from normal operating temperature and pressure to refueling conditions.

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The initial phase of plant cooldown consists of reactor coolant cooldown and depressurization. Heat is transferred from the reactor coolant system via the steam generators to the main steam system. Depressurization is accomplished by spraying reactor coolant into the pressurizer, which cools and condenses the pressurizer steam bubble.

When the reactor coolant temperature and pressure have been reduced to 350°F and 450 psig, respectively (approximately four hours after reactor shutdown), the second phase of plant cooldown is initiated with the normal residual heat removal system being placed in service.

Before starting the residual heat removal pumps, the in-containment refueling water storage tank isolation valve is closed. Then the normal residual heat removal system suction header isolation valve and the discharge header isolation valve are opened. When the pressure in the reactor coolant system has been reduced to below 450 psig, the inner/outer isolation valves are opened.

Once the proper valve alignment has been performed and component cooling water flow has been initiated to both residual heat removal heat exchangers, normal residual heat removal system operation may begin. The pumps are started and the cooldown proceeds. The cooldown rate is controlled by throttling the flow through the heat exchanger based on reactor coolant temperature.

This mode of operation continues for the duration of the cooldown until the reactor coolant system temperature is reduced to 140°F and the system is depressurized. The reactor coolant system may then be opened for either maintenance or refueling. Cooldown continues until the reactor coolant system temperature is lowered to 125°F (about 96 hours after reactor shutdown).

During the cooldown operations, the reactor coolant system water level is drained to a “mid-loop” level to facilitate steam generator draining and maintenance activities. For normal refuelings, the level to which the reactor coolant system is drained is that which allows air to be vented into the steam generators from the pressurizer. This level is nominally an 80 percent water level in the hot leg. The design of the AP1000 normal residual heat removal system is such that throttling of the residual heat removal pump flow during mid-loop operations to avoid air-entrainment into the pump suction is not required.

At the appropriate time during the cooldown, the operator lowers the water level in the reactor coolant system by manually positioning the chemical and volume control system letdown control valve. At this time the makeup pumps are turned off; and the letdown flow control valve controls the drain rate to the liquid waste processing system. The drain rate proceeds initially at the maximum letdown rate and is substantially reduced as the level in the reactor coolant system is lowered to the top of the hot leg. The letdown flow control valve as well as the letdown line containment isolation valve receives a signal to automatically close once the appropriate level is attained. Alarms actuate in the main control room if the level continues to drop to alert the operator to manually isolate the letdown line.

5.4.7.4.3 Refueling

Both residual heat removal pumps and heat exchangers remain operating during refueling. Water transfers from the in-containment refueling water storage tank to the refueling cavity are performed by the spent fuel pool cooling system (SFS). This function has traditionally been performed by residual heat removal systems. That capability still exists if the need arises. To improve clarity in the refueling cavity and reduce operational radiation exposure, the spent fuel pool cooling system is used to flood the refueling cavity without flooding through the reactor vessel.

As decay heat decreases and as fuel is moved to the spent fuel pool, one residual heat removal pump and heat exchanger may be taken out of service. However, the valves remain aligned should the need arise to start this pump quickly in case of a failure of the operating residual heat removal pump.

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5.4.7.4.4 Accident Recovery Operations

Upon actuation of automatic depressurization, the normal residual heat removal system can be employed to provide low-pressure reactor coolant system makeup. Provided that radiation levels inside containment are below a high radiation value and after resetting the safeguards actuation signal to the valves as necessary, the operator may open the cask loading pit suction valves and the residual heat removal discharge isolation valve and start the residual heat removal pumps. Water is pumped from the cask loading pit to the direct vessel injection lines. Operation of the normal residual heat removal system will not prevent the passive core cooling system from performing its safety functions.

5.4.7.4.5 Spent Fuel Pool Cooling

The normal residual heat removal system has the capability of being connected to supplement or take over the cooling function of the spent fuel pool cooling system. The normally closed valves in the cross-connecting piping are opened. One normal residual heat removal pump is started. Spent fuel pool water is drawn through the pump, passed through a heat exchanger and returned to the pool.

This mode of cooling is available when the normal residual heat removal system is not needed for normal shutdown cooling. The spent fuel pool water flow path between the spent fuel pool and the normal residual heat removal system is independent of the flow path used for spent fuel pool cooling by the spent fuel pool cooling system.

5.4.7.4.6 Fire Leading to MODE 5, Cold Shutdown

In the event of loss of normal component cooling system function where it is desired to transfer to MODE 5, Cold Shutdown, to facilitate maintenance, the fire protection system can provide the source of cooling water for a normal residual heat removal system pump and heat exchanger as described in Subsection 9.2.2.4.5.5.

5.4.7.5 Design Evaluation

Since the normal residual heat removal system is connected to the reactor coolant system, portions of the system that create the reactor coolant system pressure boundary are designed according to ANSI/ANS 51.1 (Reference 6) with regards to maintaining the reactor coolant system pressure boundary integrity.

Since the normal residual heat removal system penetrates the containment boundary, the containment penetration lines are designed according to the containment isolation criteria identified in Subsection 6.2.3.

Makeup water can be provided through the normal residual heat removal system for long-term post-accident containment makeup. This makeup is provided through the manual containment isolation test connection valve in the discharge of the normal residual heat removal system. This test connection provides a containment-inventory makeup flow path to be used after a loss-of-coolant accident (LOCA). With the maximum allowable containment leak rate, makeup to containment is not needed for about one month.

The normal residual heat removal system components and piping are compatible with the radioactive fluids they contain.

The design of the normal residual heat removal system has been compared with the acceptance criteria set forth in Subsection 5.4.7, “Residual Heat Removal System,” Revision 3, of the NRC’s Standard Review Plan. The specific General Design Criteria identified in the Standard Review Plan

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section are General Design Criteria 2, 4, 5, 19, and 34. Additionally, positions of Regulatory Guides 1.1, 1.29, and 1.68 were also reviewed to determine the degree of compliance between the AP1000 and the acceptance criteria. Branch Technical Position RSB 5-1 was also reviewed as appropriate.

Discussions of the conformance with Regulatory Guides and Branch Technical Positions are found in Section 1.9. Compliance with General Design Criteria is found Section 3.1.

5.4.7.6 Inspection and Testing Requirements

5.4.7.6.1 Preoperational Inspection and Testing

Preoperational tests are conducted to verify proper operation of the normal residual heat removal system (RNS). The preoperational tests include valve inspection and testing, flow testing, and verification of heat removal capability.

5.4.7.6.1.1 Valve Inspection and Testing

The inspection requirements of the normal residual heat removal system valves that constitute the reactor coolant pressure boundary are consistent with those identified in Subsection 5.2.4. The inspection requirements of the normal residual heat removal system valves that isolate the lines penetrating containment are consistent with those identified in Section 6.6.

The low temperature overpressure protection relief valves, RNS-V020 and RNS-V021, located on the normal residual heat removal system suction relief lines, are bench tested with water. Each valve's set pressure is verified to be less than or equal to the value assumed in the low temperature overpressure protection analysis. Relieving capacity of each valve is certified in accordance with the ASME code, Section III, NC-7000.

5.4.7.6.1.2 Flow Testing

Each installed normal residual heat removal system pump is tested to measure the flow through the normal residual heat removal system heat exchangers when aligned to cool the reactor coolant system. Testing will be performed with the pump suction aligned to the reactor coolant system hot leg and the discharge aligned to the passive core cooling system direct vessel injection lines. Flow will be measured using instrumentation in the pump discharge line. Testing will confirm that each pump provides at least the required flow rate shown in Table 5.4-14. This is the minimum flow rate required to ensure that the normal residual heat removal system can meet its functional requirement of cooling the reactor during shutdown operations.

Each installed normal residual heat removal system pump is also tested to measure the flow when aligned to deliver low pressure makeup to the reactor coolant system. Testing will be performed with the pump suction aligned to the cask loading pit and the discharge aligned to the passive core cooling system direct vessel injection lines. Flow will be measured using instrumentation in the pump discharge line. The reactor coolant system will be at atmospheric pressure for this test. Testing will confirm that each pump provides at least the required flow rate shown in Table 5.4-14. This is the minimum flow rate required to ensure that the normal residual heat removal system can meet its functional requirement to prevent 4th stage ADS actuation for small breaks.

5.4.7.6.1.3 Heat Removal Capability Analysis

Heat exchanger manufacturer's heat exchanger data will be used to perform an analysis to verify that the heat removal capability of each normal residual heat removal system heat exchanger, as calculated by the product of the heat transfer coefficient and the effective heat transfer area, UA, is equal to or greater than the required value shown in Table 5.4-14. This is the minimum value required

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to ensure that the normal residual heat removal system can meet its functional requirement of cooling the reactor during shutdown operations.

5.4.7.6.2 System Gas Accumulation Assessment and Mitigation

Subsection 6.3.6.3.1 describes the assessment method to address the potential for gas intrusion for the passive core cooling system in response to Interim Staff Guidance (ISG-019) and Generic Letter 2008-01. The same assessment methodology was used to assess the normal residual heat removal system.

The normal residual heat removal system locations equipped with manual vent valves will be inspected according to the system surveillance and venting procedures described in Subsection 6.3.6.3 to eliminate identified gas accumulations.

5.4.7.7 Instrumentation Requirements

The normal residual heat removal system contains instrumentation to monitor system performance. System parameters necessary for system operation are monitored in the main control room including the following:

- Residual heat removal flow;
- Residual heat removal heat exchanger inlet and system outlet temperatures; and,
- Residual heat removal pump discharge pressure.

In addition, the reactor coolant system contains instrumentation to control and monitor the operations of the normal residual heat removal system. These include the following:

- Reactor coolant system wide range pressure; and,
- Reactor coolant system hot leg level.

Instrumentation is also provided to enable mid-loop operations to be performed from the main control room.

The motor-operated valves connected to the reactor coolant system hot leg are interlocked to prevent them from opening when reactor coolant system pressure exceeds 450 psig. These valves are also interlocked to prevent their being opened unless the isolation valve from the in-containment refueling water storage tank to the residual heat removal pump suction header is closed. Section 7.6 describes this interlock.

5.4.8 Valves

Valves in the reactor coolant system and safety-related valves in connecting systems provide the primary means for the flow of water into and out of the reactor coolant system. In the following paragraphs the design basis, description, evaluation and testing of ASME Code Class 1, 2 and 3 valves is discussed. This discussion includes safety-related valves not in the reactor coolant system because the valve requirements are independent of the system.

5.4.8.1 Design Bases

Valves within the reactor coolant system and safety-related valves in connected systems are designed, manufactured, and tested to meet the requirements of the ASME Code, Section III. As

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noted in Section 5.2, valves out to and including the second valve that is normally closed or capable of automatic or remote closure are part of the reactor coolant system. The reactor coolant pressure boundary valves are manufactured to the ASME Code Class 1 requirements. Valves of 1 inch and smaller in lines connected to the reactor coolant system are manufactured to Class 2 requirements when the flow is limited by a flow-limiting orifice.

Containment isolation valves are manufactured to ASME Code, Class 2 requirements. Other AP1000 equipment Class C safety-related valves are manufactured to ASME Code, Class 3 requirements. Safety-related valves in auxiliary systems are manufactured to ASME Code Class 2 and 3 requirements depending on their function and classification as outlined in Subsection 3.2.2.

Table 5.4-15 provides design data for the reactor coolant pressure boundary valves. Valves and operators are sized to provide valve operation under the full range of design basis flow and pressure drop conditions, including recovery from potential mispositioning of the valves. Operating modes, normal operating and worst-case differential pressures, fluid temperature ranges, and environmental effects are considered in sizing valves and valve operators. Table 5.4-16 gives the normal and maximum differential pressure expected during opening and closing of motor-operated valves in the reactor coolant pressure boundary. Check valves considered part of the reactor coolant system are located inside the containment.

5.4.8.1.1 Check Valves Design and Qualification

Design basis and required operating conditions for safety-related check valves are established based on design conditions including the required system operating cycles to be experienced by the valve, environmental conditions under which the valve is required to function, and severe transient loadings expected during the life of the valve. The design conditions considered may include water hammer and pipe break transients, sealing and leakage requirements, operating fluid conditions (including flow, velocity, temperature, and temperature gradient), maintenance requirements, time between major refurbishments, corrosion requirements, vibratory loading, planned testing methods, and test frequency, and periods of idle operation. Design conditions may include other requirements identified during plant detail design. The maximum loading resulting from the design conditions and transients are evaluated in accordance with the ASME Code, Section III Class 1 design requirements.

Active safety-related check valves include the capability to verify the movement of each check valve's obturator during inservice testing by observing a direct instrumentation indication of the valve position or by using non-intrusive test methods. This instrumentation provides nonintrusive check valve indication and may be either permanently or temporarily installed.

Check valve model and size selection are based on the systems flow conditions, installed location of the valve with respect to flow disturbance, and orientation of the valve in the piping system. Design features, surface finish, and materials can accommodate provisions for nonintrusive determination of disk position and potential valve degradation over time. Valve internal parts are designed with self-aligning features for the purpose of assured alignment after each valve opening. Qualification testing provides for the adequacy of the safety-related check valves under design conditions. This testing includes test data from the manufacturer, field test data and empirical test data supported by test or test (such as prototype) of similar valves where similarity is justified by technical data. Sampling size for the qualification test is justified by technical data.

For safety-related active check valves with extended structures functional qualification will be performed to demonstrate by test, by analysis or by a combination thereof, the ability to operate at the safety-related design conditions. This functional qualification will demonstrate the valve operability during and after loads representative of the maximum seismic and vibratory event. Check valve internal parts are analyzed for maximum design basis loading conditions in accordance with the requirements in ASME Code, Section III.

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5.4.8.1.2 Motor-Operated Valves Design and Qualification

[*Design basis and required operating conditions are established for active safety-related motor-operated valves. Based on the design conditions the motor-operated valves will have a structural analysis performed to demonstrate their components are within the structural limits at the design conditions. The motor-operated valves are designed for a range of conditions up to the design conditions which includes fluid flow, differential pressure (including line break, if necessary), system pressure and temperature, ambient temperature, operating voltage range and stroke time. The sizing of the motor operators on the valves take into account diagnostic equipment accuracies, changes in output capability for increasing differential pressures and flow and ambient temperature and reduction in motor voltage, control switch repeatability, friction variations and other changes in parameters that could result in an increase in operating loads or a decrease in operator output.*]* Valves that are subjected to large temperature changes during operation and can have water or high pressure fluid trapped in the bonnet cavity are evaluated for pressure locking. Provisions are provided, as required to reduce the susceptibility to bonnet overpressurization, pressure locking, and thermal binding.

[*The motor-operated valves have a functional qualification performed to demonstrate by test, by analysis or by a combination thereof, the ability to operate over a range up to the design conditions. This functional qualification will demonstrate the motor-operated valve capability during and after loads representative of the maximum seismic or vibratory event (as required to perform their intended function), demonstrate the valve sealing capability, demonstrate capability under cold and hot operating conditions, demonstrate capability under maximum pipe end loads and demonstrate flow interruption and functional capability. The testing includes test data provided by the manufacturer, field test data, empirical data supported by testing or analysis of prototype tests of similar motor-operated valves that support the qualification where similarity must be justified by technical data. The qualification must be used for validating the required thrust and torque as applicable to operate the valve and the output capability of the motor operator.*]*

Motor-operated valves are designed to be able to change their position from an improper position (mis-positioned) either prior to or during accidents. The recovery from mis-positioning is considered a nonsafety-related function. The nonsafety-related capability to recover from valve mis-positioning is provided for plant operational availability considerations. Systems with safety-related functions that contain motor-operated valves are designed to tolerate mis-positioning as a single failure or redundant features are provided to preclude mis-positioning. These features include multiple position indicators and alarms, technical specification surveillance, power lock-out, and confirmatory open or close signals.

Since recovery from mis-positioning is a nonsafety-related function, equipment qualification testing and inservice testing is not required for the recovery from mis-position function.

Provisions are made, where possible, for in-situ testing of motor-operated valves at a range of conditions up to the maximum design basis operating conditions in the safety-related design direction (open or close). Where an alternative to in-situ testing is required, the justification of the alternative method to design condition differential testing is documented as part of the valve test program.

5.4.8.1.3 Other Power-Operated Valves Including Explosively Actuated Valves Design and Qualification

Design basis and required operating conditions are established for power-operated (POV) and explosively actuated valve assemblies with an active safety-related function. Power-operated valve assemblies include pneumatic-hydraulic-, air piston-, and solenoid-operated assemblies. Explosively-actuated valves have shear caps and are actuated by an explosive charge fired by an electrical signal.

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[*The power-operated safety related valves will have a structural analysis performed to demonstrate their components are within the structural limits at the design conditions. Power operated valve assemblies and explosively actuated valves are designed to accept the maximum compression, tension, and torsional loads which the assembly is capable of producing in combination with other loads such as pressure, thermal, or externally applied loads. The maximum loading resulting from the design conditions and transients is evaluated in accordance with the ASME Code, Section III Class 1 design requirements. Packing adjustment limits are identified to reduce the potential for stem binding.*

The power-operated valves are designed to operated at design operating conditions which include fluid flow, differential pressure (including pipe break, if necessary), system pressure, fluid temperature, ambient temperature, fluid supply conditions (or electrical power supply), spring force and stroke time requirements. The power operated valves, depending on their design and actuation mode, have the operators sized to account for diagnostic equipment accuracies, changes in output capability for increasing differential pressures and flow, friction variations and changes in other parameters that could result in an increase in operating loads or a decrease in operator output.

The power-operated, safety-related valves have a functional qualification performed to demonstrate by test, by analysis or by a combination thereof, the ability to operate at the design conditions. Qualification testing of each size, type, and model is performed under a range of differential pressures and maximum achievable flow conditions up to the design conditions. This functional qualification will demonstrate the power-operated valves capability during and after loads representative of the maximum seismic or vibratory event (as required to perform their intended function), demonstrate the valve sealing capability, demonstrate capability under cold and hot operating conditions, demonstrate capability under maximum pipe end loads and demonstrate flow interruption and functional capability. The testing includes test data from the manufacturer, field test data, empirical data supported by test, or analysis of prototype tests of similar power-operated valves that support qualification of the power-operated valve. Similarity must be justified by technical data. Solenoid-operated valves are verified to satisfy the applicable requirements for Class 1E components. Solenoid-operated valves are verified to perform their safety-related design requirements over a range of electrical power supply conditions including minimum and maximum voltage.]*

5.4.8.2 Design Description

The materials of construction are selected to minimize the effects of corrosion and erosion and are compatible with the environment. The valves in contact with reactor coolant fluid shall be constructed of stainless steel materials or alloys acceptable for the fluid chemistry.

Safety-related valves do not have full penetration welds within the valve body walls except that explosive actuated valves may be fabricated using full penetration welds of the valve bodies.

Valves and actuators are furnished as a matched system capable of operating over the entire range of design basis conditions. The function of the valve and operator including switch settings for motor-operated valves are qualified by testing, analysis or a combination thereof.

Valves that have stem packing are constructed with packing material compatible with the system fluid and stem material. Valves greater than 2 inch diameter may have live load packing to maintain a compressive packing force. Variations in valve packing configurations are bounded in occupational radiation exposure dose assessments. Valves supplied with stem packing are supplied with a backseat which may be utilized to minimize stem leakage. The backseat capability does not rely on system pressure to achieve a satisfactory seal. Valve designs such as main steam isolation valves, safety relief valves, packless valves and small solenoid valves by nature of the design of these valves do not have backseat capability. Motor operated valves are not backseated during normal operation. The backseating of the valve must not compromise the structural integrity of the valve and

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the backseats are capable of retaining the valve stem against full system pressure and maximum thrust produced by the actuator.

Gate valves at the interface with the reactor coolant system and connected safety-related systems are either of the wedge or parallel disc design and have essentially straight through flow. The wedge design is flex-wedge; solid wedge designs are not used. Gate valves have backseats. Gate valves that are susceptible to overpressurization as the result of the heatup of trapped fluid shall be provided with venting capability to alleviate the issue. The valve shall be of outside screw and yoke design. Gate valves are not used in flow regulation or throttling service.

Globe valves are either T or Y type of either a standard or balanced plug design. Valves that are used for throttling service are designed with a disc or disc/cage assembly that will provide the required flow characteristic. Motor operated and manual valves are of the outside screw and yoke design.

Check valves are typically swing type, but tilt disk, nozzle check, and lift check may be used. Check valves containing radioactive fluid are fabricated of stainless steel. These valves do not have body penetration other than the inlet, outlet and bonnet. The check hinge is serviced through the bonnet. Operating parts are contained within the body. The disc of swing check valves has limited rotation to provide a change of seating surface and alignment after each valve opening.

5.4.8.3 Design Evaluation

ASME Code, Class 1 valves meet the design requirements of ASME Code, Section III, Article NB-3000. ASME Code, Class 2 valves meet the design requirements of ASME Code, Section III, Article NC-3000. ASME Code Class 3 valves meet the design requirements of ASME Code, Section III, Article ND-3000. The AP1000 equipment Classes A, B, and C valves, which are manufactured to ASME Code Classes 1, 2, and 3 respectively, meet established functional requirements. The functional requirements include operability, differential pressure during opening or closure, and seat leakage. The functional requirements are based on ASME QME-1 (Reference 8).

The design transients for the valves including the number and the duration of each type of cycle are identified in Subsection 3.9.1.1.

Valves with extended structures have testing or analysis performed to demonstrate that the natural frequency is greater than 33 hz. In addition, a structural analysis is performed to verify the design loading will not effect the intended operation of the valve.

Qualification testing of each power operated valve which includes motor-operated, air operated, hydraulic operated, solenoid operated and explosive actuated valves demonstrates the capability of the operator to operate over the full range of expected plant operating conditions. Qualification testing also demonstrates the closing, opening, and seating capability of the valve against the maximum pressure differential and flow within a specified time over the entire operating range. Requirements for qualification testing of power-operated active valves are based on ASME QME-1 (Reference 8). The testing programs in Section 3.10 demonstrate the capability of the valves to operate, as required, during anticipated and postulated plant conditions.

Reactor coolant chemistry parameters are compatible with valve construction materials.

5.4.8.4 Tests and Inspections

The nondestructive examinations for the reactor coolant pressure boundary valves meet the more stringent requirements of the ASME Code, Section III, and ASME B16.34 (Reference 9). The nondestructive examination required is evaluated for each type and class of valve. The examinations consist of the following:

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- **Radiographic Examination** - Classes 1 and 2 valve bodies, bonnets, and discs which are made of cast material are radiographically examined in accordance with the ASME Code, Section III. The procedure and acceptance standards are according to the requirements for Class 1 in the ASME Code, Section III.

- **Ultrasonic Examination -** Classes 1 and 2 valve bodies, bonnets, and discs and Classes 1, 2, and 3 valve stems of 1 inch nominal diameter or larger fabricated of wrought or forged material are ultrasonically examined, except for stems for valves with inlet pipe connections of 2 inches nominal pipe size and less. The procedures and acceptance standards are according to the requirements for Class 1 in the ASME Code, Section III.

- **Liquid Penetrant Examination** - Bodies, bonnets, discs, and stems, including machined surfaces on these parts, are liquid-penetrant examined in accordance with the ASME Code, Section III. The procedures and acceptance standards are according to the requirements for Class 1 in the ASME Code, Section III.

Hydrostatic pressure boundary test and seat leakage are performed on the reactor coolant pressure boundary valves. The valves are subjected to the following tests as appropriate following manufacture: hydrostatic pressure boundary test, disc hydrostatic test, backseat leakage test, packing leakage test, stem leakage test, and main seat leakage test. Valves used for containment isolation are subjected to a pneumatic seat leakage test. Each diaphragm actuator assembly is subjected to a pneumatic leakage test.

Preoperational testing is performed on the valves to verify operability during design basis operating conditions. The preoperational testing is described in the following sections. The requirements of NRC Generic Letter 89-10 are used as guidelines to develop the preservice test program for valve operability. Except when test alternatives are justified, design conditions are used for the operability testing.

Subsection 5.2.4 discusses inservice inspection for ASME Code Class 1 valves. Section 6.6 discusses inservice inspection for ASME Code Class 2 and 3 components. Valves are accessible for disassembly and internal visual inspection to the extent practical. Subsection 3.9.6 discusses the inservice testing program for active valves.

5.4.8.5 Preoperational Testing

Results of preoperational testing will be used to demonstrate that the results of testing under in situ or installed conditions can be used to confirm the capacity of the valve to operate under design conditions as discussed in Section 14.4.

5.4.8.5.1 Check Valves

Active check valves are tested in the open and close direction. Testing a check valve confirms the valve operability to move to the position to fulfill the safety-related mission during applicable plant modes. The test shows that the check valve opens in response to flow and closes when the flow is stopped. Operability testing of the valves is described in Subsection 3.9.3.2.2. Full-flow testing during applicable plant modes of check valves or sufficient flow to fully open the check valve to demonstrate valve operability under design conditions is permitted in most cases by the system design. Where this testing cannot be accomplished, an alternate method of demonstrating operability is developed, and justified. A demonstration of reverse-flow isolation of the check valves that is that the check valve closes when the flow is stopped is performed using direct means or diagnostics. The testing includes the effects of rapid pump starts and stops as required by expected system operating conditions.

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The valves to be tested, the safety-related functions of the valves, and the type of testing to be done to verify the capability of the valves to perform the safety-related functions are outlined in valve inservice test requirements found in Subsection 3.9.6. The valves to be tested, safety-related functions, and test requirements for preoperational testing are the same as outlined in inservice test requirements.

During pre-operational testing the following is verified to demonstrate the acceptability of the functional performance.

- The valves are verified to fully open or fully closed under design flow conditions.
- The disc movement from full open to full close is free.
- The valve leakage when fully closed is within established limits, as applicable.
- The disc is stable in the full open position at the system operating flow, conditions.
- The valve disc position can be verified without disassembly of the valve.
- The valve design features, surface finish and materials can accommodate nonintrusive diagnostic testing methods.
- The testing requirements in the inservice test plan can be accommodated in the piping system design.

5.4.8.5.2 Motor-Operated Valves

[*Active safety-related motor-operated valves are tested to verify that the valves open and close under static and safety-related design conditions. Where the safety-related design conditions cannot be achieved, the testing is performed at the maximum achievable dynamic conditions. During the testing critical parameters needed to determine the required closing and opening loads are measured These parameters include thrust, torque, travel, differential pressure, system pressure, fluid flow, voltage, temperature, operating time and thrust/torque at seating, unwedging and at control switch trip. The data collected during the testing on the parameters is used to determine the required operator loads and output capability for the design operating conditions in conjunction with the diagnostic equipment inaccuracies, load changes for increasing differential pressures and flow and ambient temperature and reduction in motor voltage, control switch repeatability, friction variations and changes in other parameters that could result in an increase in operating loads or decrease in operator output capability. The resulting operating loads including uncertainties are then compared to the structural capabilities of the motor-operated valve.*]* Active safety-related motor-operated valves are tested prior to operation for operability as described in Subsection 3.9.3.2.2.

Pre-operational testing and evaluation is used to demonstrate the acceptability of the valves functional performance including the following.

- The valves are verified to open and close as applicable at a range of safety-related conditions up to the design conditions to perform their safety function.
- The control switch settings must be adequate to provide margin for diagnostic accuracy, control switch repeatability, load sensitive behavior and degradation.

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- The motor operator capability at degraded voltage must exceed the required operating loads and the loads at the control switch settings including diagnostic equipment inaccuracies, load changes for increasing differential pressures and flow, control switch repeatability, friction variations and other parameters that could result in an increase in operating loads or decrease in operator output capability.

- The maximum operating loads including diagnostic equipment inaccuracies, load changes for increasing differential pressures and flow, control switch repeatability, friction variations and other parameters that could result in an increase in operating loads or decrease in operator output capability are verified not to exceed the allowable structural capability limits of the motor-operated valve components.

- The stroke time measurements during opening and closing must be within the design requirements if stroke time is important to the safety function.

- The remote position indication is verified against the local position indication.

- The valve leakage when fully closed is within established limits, as applicable.

5.4.8.5.3 Power-Operated Valves

[*Active safety related power-operated valve assemblies are tested to verify that the valve opens and closes under static and design conditions. Where the design conditions cannot be achieved, the testing is performed at the maximum achievable dynamic conditions. During the testing, critical parameters needed to determine the required closing and opening loads are measured. These parameters include seat load, torque or thrust, travel, spring rate, differential pressure, system pressure, fluid flow, temperature, power supply, operating time and minimum supply pressure. The data collected during the testing on the parameters is used to determine the required operating loads for the design operating conditions in conjunction with the diagnostic equipment inaccuracies and other parameters that could result in an increase in operating loads or decrease in operator output capability. The resulting operating loads including uncertainties are then compared to the structural capabilities of the power-operated valve.*]*

During pre-operational testing the following are verified to demonstrate the acceptability of the functional performance.

- The valves are verified to open and close as applicable at a range of conditions up to the design conditions to perform its safety function.

- For air-operated valves and hydraulically-operated valves the operator capability at minimum supply pressure, power supply or loss of motive force exceed the required operating loads including diagnostic equipment inaccuracies and other parameters that could result in an increase in operating loads or decrease in operator output capability.

- For solenoid-operated valves the valve must be capable of opening or closing the valve at the minimum power supply.

- For air-operated valves and hydraulically-operated valves the maximum operating loads including diagnostic equipment inaccuracies and other parameters that could result in an increase in operating loads are verified not to exceed the allowable structural capability limits of the power-operated valve components.

- The stroke time measurements during opening and closing must be within the design requirements for safety-related functions.

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- The remote position indication is verified against the local position indication.
- The valve leakage when fully closed is within established limits, as applicable.

5.4.9 Reactor Coolant System Pressure Relief Devices

Safety valves connected to the pressurizer provide overpressure protection for the reactor coolant system during power operation. The relief valves on the suction line of the normal residual heat removal system (RNS) provide low temperature overpressure protection consistent with the guidelines of NRC Branch Technical Position RSB 5-2. The following discusses the requirements for the valves. Sizing of the safety and relief valves is discussed in Subsection 5.2.2.

Power-operated relief valves are not provided in the AP1000 reactor coolant system. Non-reclosing pressure relief devices are not used for pressure relief on the AP1000 reactor coolant system. Section 10.3 discusses safety valves for the main steam system. The automatic depressurization valves which are also connected to the pressurizer and are the interface with the passive core cooling system, are not pressure relief devices. (See Subsection 5.4.6.)

5.4.9.1 Design Bases

The combined capacity of the pressurizer safety valves can accommodate the maximum pressurizer surge resulting from complete loss of load. The relief valves on the suction line of the normal residual heat removal system can accommodate the flow from one or both makeup pumps with no letdown and a water-solid reactor coolant system during low-temperature modes. Table 5.4-17 gives design parameters for the pressurizer safety valves and the residual heat removal system relief valves.

Use of the pressurizer safety valves and the normal residual heat removal relief valves at elevated temperatures in post-accident environments is not anticipated.

5.4.9.2 Design Description

The pressurizer safety valves and the normal residual heat removal system relief valves are spring loaded, self-actuated by direct fluid pressure, and have backpressure compensation features. These valves are designed to reclose and prevent further flow of fluid after normal conditions have been restored. The pressurizer safety valves are of the totally enclosed pop type. The normal residual heat removal relief valves are designed for water relief.

The pressurizer safety valves are incorporated in the pressurizer safety and relief valve (PSARV) module, which provides the connection to the pressurizer nozzles. The routing of pipe between the pressurizer and the safety valves does not include a loop seal. Any condensation of steam in the connecting pipe up to the valve rains back to the pressurizer. Condensate does not collect as a slug of water to be discharged during the initial opening of the valve. The discharge of the safety valve is routed through a rupture disk to containment atmosphere. The rupture disk is provided to contain leakage past the valve, is designed for a substantially lower set pressure than the safety valve set pressure, and does not function as a relief device. Because the rupture disks have relatively low set pressures and the header directing the safety valve discharge to the rupture disk also drains into the ADS discharge, with ADS discharge having a potential rupture disk breaking pressure, for economic reasons being to reduce rupture disk replacement costs, a check valve is installed in the safety valve discharge drain line to reduce the potential for ADS backpressure actuating the rupture disks while ADS is discharging. The reactor coolant system Piping and Instrumentation Drawing (Figure 5.1-5) shows the arrangement of the safety valves, rupture disks, and check valve.

The relief valves in the normal residual heat removal system are located between the suction line of the pump and the valve that isolates the residual heat removal system from the reactor coolant

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system. The discharge from the valves is directed to the containment sump. Subsection 5.4.7 discusses the residual heat removal system. Figure 5.4-6 shows a simplified sketch of the normal residual heat removal system.

In accordance with the requirements of 10 CFR 50.34(f)(2)(xi), positive position indication is provided for the pressurizer safety valves and the normal residual heat removal system relief valves, which provide overpressure protection for the reactor coolant pressure boundary.

Temperatures in the safety valve discharge lines are measured, and an indication and a high temperature alarm are provided in the control room. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve. Leakage past the pressurizer safety valve during normal operation is collected and directed to the reactor coolant drain tank. Section 7.5 discusses the functional requirements for the instrumentation required to monitor the safety valves.

Temperature in the common relief valve discharge line is measured, and an indication and a high alarm are provided in the control room. A rate of change in the discharge line temperature is an indication of leakage or relief through the associated valves. Leakage past the RNS relief valves is directed to the containment sump.

5.4.9.3 Design Evaluation

The pressurizer safety valves prevent reactor coolant system pressure from exceeding 110 percent of system design pressure, in compliance with the ASME Code, Section III. The relief valves on the suction line of the normal residual heat removal system protect that system from exceeding 110 percent of the design pressure of the system and from exceeding the pressure-temperature limits determined from ASME Code, Appendix G, analyses.

The reactor coolant system pressure transients are described in Subsection 15.2.3 and are the basis for the ASME Code Overpressure Protection Report. In the analysis of overpressure events, the pressurizer safety valves are assumed to actuate at 2500 psia. The safety valve flowrate assumed is based on full flow at 2575 psia, assuming 3 percent accumulation.

In certain design basis events described in Chapter 15, including the inadvertent CMT actuation, loss of normal feedwater, and CVS malfunction events, the reactor coolant system experiences slow pressurization rates up to the pressurizer safety valve lift setpoint and pressure is relieved through the safety valves for an extended period of time. For these events, the reactor coolant system pressure is slowly increasing as a result of the mismatch between the decay heat removal rate from the passive residual heat removal heat exchanger and the core decay heat. This slow pressurization of the reactor coolant system may be relieved through a small amount of steam flow through the safety valves without either safety valve fully opening. However, the pressurizer safety valves are designed to minimize leakage at pressures approaching lift pressure. These valves are qualified to relieve pressure for the duration of the anticipated transients assuming the valves fully open when their lift setpoint is reached. Operation of the safety valves under these conditions could result in small leakage from the valve (much less than the capacity of the normal makeup system), but does not impair the valve overpressure protection capability.

The relief valves on the normal residual heat removal system have an accumulation of 10 percent of the set pressure. The set pressures are the lower of the pressure based on the design pressure of the residual heat removal system and the pressure based on the reactor vessel low temperature pressure limit. The pressure limit determined based on the design pressure includes the effect of the pressure rise across the pump. The set pressures in Table 5.4-17 are based on the reactor vessel low temperature pressure limit. The lowest permissible set pressure is based on the required net positive suction head for the reactor coolant pump.

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5.4.9.4 Tests and Inspections

The safety and relief valves are the subject of a variety of tests to validate the design and to verify pressure boundary and functional integrity. For valves that are required to function during a Service Level D condition, static deflection tests are performed to demonstrate operability. Section 3.10 describes these tests.

Safety valves similar to those connected to the pressurizer have been tested within the Electric Power Research Institute (EPRI) safety and relief valve test program. Capacity data for the specific AP1000 safety valve size has been correlated with the EPRI test data to demonstrate that the valve is adequate for steam flow and water flow, even though water flow is not anticipated through the pressurizer safety valves. The completion of this program addresses the requirements of 10 CFR 50.34(f)(2)(x) as related to reactor coolant system relief and safety valve testing. The normal residual heat removal system relief valves are designed for water relief and are not reactor coolant system pressure relief devices since each valve has a set pressure less than reactor coolant system design pressure. Therefore, the valves selected for the normal residual heat removal system relief valves are independent from the Electric Power Research Institute safety and relief valve test program.

Reactor coolant system pressure relief devices are subjected to preservice and inservice hydrostatic tests, seat leakage tests, operational tests, and inspections, as required. The preservice and inservice inspection and testing programs for valves are described in Subsections 3.9.6 and 5.2.4 and Section 6.6. The test program for the safety valves complies with the requirements of ASME Appendix I of the OM Code.

The pressure boundary portion of the valves are required to be inservice inspected according to the rules of Section XI of the ASME Code. There are no full-penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection.

Type testing of the pressurizer safety valves is performed to verify that the pressurizer safety valves operate stably during slow pressurization transients at pressures near the valve set pressure. The valve full-open pressure is the pressure at which the safety valve opens with significant blowdown flow. The duration of the testing need not duplicate the times indicated in the accident analysis results but should last for a sufficient time to demonstrate stable valve operation. Stable valve performance which may include extended periods of valve simmering or typical valve opening (popping fully open) without excessive valve cycling or chattering for a 15 minute time duration is sufficient. The testing is performed to verify the pressurizer safety valves are capable of performing their intended design function throughout the duration of the slow pressurization transients by either simmering or popping fully open. Following this testing, the valve integrity is demonstrated, and the valve leakage is required to be less than the makeup capability of the chemical and volume control system makeup pumps.

5.4.10 Component Supports

5.4.10.1 Design Bases

Component supports provide deadweight support for the piping and equipment, allow lateral thermal movement of the loop during plant operation, and restrain the loops and components during accident and seismic conditions. Subsection 3.9.3 discusses the loading combinations and design stress limits. Support design is according to the ASME Code, Section III, Subsection NF.

The design provides for the integrity of the reactor coolant pressure boundary for normal, seismic, and accident conditions. The design also maintains the piping stresses less than ASME Code limits and less than the limits required to support mechanistic pipe break discussed in Subsection 3.6.3.

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Section 3.9 presents the results of piping and supports stress evaluations. The loads associated with the dynamic effects of postulated pipe rupture for pipes 6" and larger, which satisfied the requirements for mechanistic pipe break, are not included. See Subsection 3.6.3.

The edition of the ASME code, Section III, subsection NF, which is used as the baseline requirement, address the guidance of Regulatory Guides 1.124 and 1.130. The plant design is in conformance with these requirements of the ASME Code. Conformance with Regulatory Guides 1.124 and 1.130 is discussed in detail in Section 1.9. The embedded portions of the component supports are designed according to AISC N690 and ACI 349, as discussed in Subsection 3.8.3.

5.4.10.2 Design Description

The support structures are welded, structural steel sections. Linear structures (tension and compression struts, columns, and beams) are used except for the reactor vessel supports, which are plate-and-shell-type structures. Attachments to the supported equipment are either integral (welded to the component) or non-integral (pinned to, bolted to, or borne against the components). The supports-to-concrete attachments are either brackets welded to heavy embedded plates or anchor bolts or are embedded fabricated assemblies.

The supports permit thermal growth of the supported systems but restrain vertical and lateral movement resulting from seismic and pipe-break loadings. This is accomplished by using pinned ends in the vertical support columns, girders, bumper pedestals, and hydraulic snubbers, and lateral struts.

Because of manufacturing and construction tolerances, ample adjustment for the support structures provides proper erection alignment and fit-up. This is accomplished by shimming or grouting at the supports-to-concrete interface and by shimming at the supports-to-equipment interface.

The supports for the various components are described in the following paragraphs.

5.4.10.2.1 Reactor Pressure Vessel

The reactor vessel supports consist of four individual, air-cooled steel box structures located beneath the inlet nozzles (See Figure 3.8.3-4). The boxes are air-cooled to achieve a concrete design temperature of 200°F. To reduce heat transfer from the nozzle to the concrete, cooled air is baffled vertically through the support, and the heated air is vented at the top.

Vertical and horizontal loads are transmitted from the reactor vessel nozzle pad to the box structure through an integral recessed “pocket” at the top of the box. The nozzle pad bears on permanently lubricated wear plates that allow radial thermal movements of the nozzle with minimal friction resistance to the movement. The vessel support boxes transfer loads from the reactor pressure vessel to vertical and horizontal embedments in the primary shield wall concrete.

5.4.10.2.2 Steam Generator

As shown in Figure 3.8.3-5, each steam generator support consists of the following:

Vertical Support

The vertical support consists of a single vertical column extending from the steam generator compartment floor to the bottom of each steam generator channel head. The column is constructed of a heavy wide flange section, and is pinned at both ends to permit thermal movement of each steam generator during plant heatup and cooldown. The column is located so that it allows full access to the steam generator for routine maintenance activities. It is located far enough from the reactor coolant pump motors to permit pump maintenance and inservice inspection.

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Lower Lateral Support

The lower horizontal support is located at the bottom of the channel head. It consists of a tension/compression link bar and extension bar oriented nearly perpendicular to the hot leg. The link bar/extension bar is pinned at both the wall bracket and the steam generator channel head to permit movement of the steam generator during plant heatup and cooldown.

Upper Lateral Support

The upper horizontal support in the direction of the hot leg is located on the upper shell just above the transition cone. It consists of two large hydraulic snubbers oriented parallel with the hot leg centerline. One snubber is mounted on each side of the generator on top of the steam generator compartment wall. The hydraulic snubbers are valved to permit relatively unrestricted steam generator movement during thermal transient conditions, and to “lock up” and act as a rigid strut under dynamic loads.

The upper steam generator horizontal support in the direction normal to the hot leg is located on the lower shell just below the transition cone. It consists of two rigid struts oriented perpendicular to the hot leg. The two rigid struts are mounted on the steam generator compartment wall at the elevation of the operating deck. The steam generator loads are transferred to the struts and snubbers through trunnions on the generator shell.

5.4.10.2.3 Reactor Coolant Pump

The reactor coolant pumps are supported entirely by the steam generators; consequently, there are no reactor coolant pump supports.

5.4.10.2.4 Pressurizer

The supports for the pressurizer, as shown in Figure 3.8.3-6, consist of the following:

- Four steel columns attached to the lower head to provide vertical support for the pressurizer. Struts connected to the lower head and surrounding walls provide lateral support.

- The upper lateral support consists of a box-type ring girder that surrounds the pressurizer. The support connects to the corners of the pressurizer cubicle walls with eight standard sway struts. The girder rests on and is supported vertically by the pressurizer valve support brackets. The pressurizer upper support also supports the pressurizer safety relief piping and valve module, in addition to providing lateral support to the pressurizer.

5.4.10.2.5 Control Rod Drive Mechanism Supports

The support for the control rod drive mechanism is provided by the integrated head package, as described in Subsection 3.9.7.

5.4.10.3 Design Evaluation

An evaluation verifies the design adequacy and structural integrity of the reactor coolant loop and the primary equipment supports system. This evaluation compares the analytical results with established criteria for acceptability. Structural analyses demonstrate design adequacy for safety and reliability of the plant in case of a seismic disturbance, and/or loss of coolant accident conditions. Loads that the system is expected to encounter during its lifetime (thermal, weight, and pressure) are applied, and stresses are compared to allowable values. Subsection 3.9.3 discusses the modeling and analysis methods.

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5.4.10.4 Tests and Inspections

Nondestructive examinations are performed according to the procedures of the ASME Code, Section V, except as modified by the ASME Code, Section III, Subsection NF.

5.4.11 Pressurizer Relief Discharge

The AP1000 does not have a pressurizer relief discharge system. The AP1000 has neither power operated pressurizer relief valves nor a pressurizer relief discharge tank. Some of the functions provided by the pressurizer relief discharge system in previous nuclear power plants are provided by portions of other systems in the AP1000.

The safety valves connected to the top of the pressurizer provide for overpressure protection of the reactor coolant system. First-, second-, and third-stage automatic depressurization system valves provide for depressurization of the reactor coolant system and venting of noncondensable gases in the pressurizer following an accident. These functions are discussed in Subsections 5.2.2, 5.4.12, and in Section 6.3. The AP1000 does not have power operated relief valves connected to the pressurizer.

The discharge of the safety valves is directed through a rupture disk to containment atmosphere.

The discharge of the first-, second-, and third-stage automatic depressurization system valves is directed to the in-containment refueling water storage tank. For the automatic depressurization system valves, the following discussion considers only the gas venting function. Only the first stage automatic depressurization valves are used to vent non-condensible gases following an accident. The sizing considerations and design basis for the in-containment refueling water storage tank for the depressurization function are discussed throughout Section 6.3. The provisions to minimize the differential pressure between the containment atmosphere and the interior of the in-containment refueling water storage tank are also discussed in Subsection 6.3.2.

The relief valves on the normal residual heat removal system, which provide low temperature overpressure protection, discharge into the containment sump. See Subsection 5.4.7 for a discussion of the connections to and location of the relief valves in the normal residual heat removal system.

5.4.11.1 Design Bases

The containment has the capability to absorb the pressure increase and heat load resulting from the discharge of the safety valves to containment atmosphere. The in-containment refueling water storage tank has the capability to absorb the pressure increase and heat load from the discharge, including the water seal, steam and gases, from a first-stage automatic depressurization system valve when used to vent noncondensable gases from the pressurizer following an accident. The venting of noncondensable gases from the pressurizer following an accident is not a safety-related function.

5.4.11.2 System Description

Each safety valve discharge is directed to a rupture disk at the end of the discharge piping. A small pipe is connected to the discharge piping to drain away condensed steam leaking past the safety valve. The discharge is directed away from any safety related equipment, structures, or supports that could be damaged to the extent that emergency plant shutdown is prevented by such a discharge.

The discharge from each of two groups of automatic depressurization system valves is connected to a separate sparger below the water level in the in-containment refueling water storage tank. The piping and instrumentation diagram for the connection between the automatic depressurization

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system valves and the in-containment refueling water storage tank is shown in Figure 6.3-1. The in-containment refueling water storage tank is a stainless steel lined compartment integrated into the containment interior structure. The discharge of water, steam, and gases from the first-stage automatic depressurization system valves when used to vent noncondensable gases does not result in pressure in excess of the in-containment refueling water storage tank design pressure. Additionally, vents on the top of the tank protect the tank from overpressure, as described in Subsection 6.3.2.

Overflow provisions prevent overfilling of the tank. The overflow is directed into the refueling cavity. The in-containment refueling water storage tank does not have a cover gas and does not require a connection to the waste gas processing system. The normal residual heat removal system provides nonsafety-related cooling of the in-containment refueling water storage tank.

5.4.11.3 Safety Evaluation

The design of the control for the reactor coolant system and the volume of the pressurizer is such that a discharge from the safety valves is not expected. The containment design pressure, which is based on loss of coolant accident considerations, is greatly in excess of the pressure that would result from the discharge of a pressurizer safety valve. The heat load resulting from a discharge of a pressurizer safety valve is considerably less than the capacity of the passive containment cooling system or the fan coolers. See Section 6.2.

Venting of noncondensable gases, including entrained steam and water from the loop seals in the lines to the automatic depressurizations system valves, from the pressurizer into spargers below the water line in the in-containment refueling water storage tank does not result in a significant increase in the pressure or water temperature. The in-containment refueling water storage tank is not susceptible to vacuum conditions resulting from the cooling of hot water in the tank, as described in Subsection 6.3.2. The in-containment refueling water storage tank has capacity in excess of that required for venting of noncondensable gases from the pressurizer following an accident.

5.4.11.4 Instrumentation Requirements

The instrumentation for the safety valve discharge pipe, containment, and in-containment refueling water storage tank are discussed in Subsections 5.2.5, 5.4.9, and in Sections 6.2 and 6.3, respectively. Separate instrumentation for the monitoring of the discharge of noncondensable gases is not required.

5.4.11.5 Inspection and Testing Requirements

Sections 6.2 and 6.3 discuss the requirements for inspection and testing of the containment and in-containment refueling water storage tank, including operational testing of the spargers. Separate testing is not required for the noncondensable gas venting function.

5.4.12 Reactor Coolant System High Point Vents

The requirements for high point vents are provided for the AP1000 by the reactor vessel head vent valves and the automatic depressurization system valves. The primary function of the reactor vessel head vent is for use during plant startup to properly fill the reactor coolant system and vessel head. Both reactor vessel head vent valves and the automatic depressurization system valves may be activated and controlled from the main control room. The AP1000 does not require use of a reactor vessel head vent to provide safety-related core cooling following a postulated accident.

The reactor vessel head vent valves (Figure 5.4-8) can remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired

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natural circulation through the steam generators resulting from the accumulation of noncondensable gases in the reactor coolant system. The design of the reactor vessel head vent system is in accordance with the requirements of 10 CFR 50.34 (f)(2)(vi).

The reactor vessel head vent valves can be operated from the main control room to provide an emergency letdown path which is used to prevent pressurizer overfill following long-term loss of heat sink events. An orifice is provided downstream of each set of head vent valves to limit the emergency letdown flow rate.

The first stage valves of the automatic depressurization system are attached to the pressurizer and provide the capability of removing noncondensable gases from the pressurizer steam space following an accident. Venting of noncondensable gases from the pressurizer steam space is not required to provide safety-related core cooling following a postulated accident. Gas accumulations are removed by remote manual operation of the first stage automatic depressurization system valves.

The discharge of the automatic depressurization system valves is directed to the in-containment refueling water storage tank. Subsection 5.4.6 and Section 6.3 discuss the automatic depressurization system valves and discharge system.

The passive residual heat removal heat exchanger piping and the core makeup tank inlet piping in the passive core cooling system include high point vents that provide the capability of removing noncondensable gases that could interfere with heat exchanger or core makeup tank operation. These gases are normally expected to accumulate when the reactor coolant system is refilled and pressurized following refueling shutdown. Any noncondensable gases that collect in these high points can be manually vented.

The discharge of the passive residual heat removal heat exchanger high point vent is directed to the in-containment refueling water storage tank. The discharge of the core makeup tank high point vent is directed to the reactor coolant drain tank. Section 6.3 discusses the passive residual heat removal heat exchanger and venting capability, which is part of the passive core cooling system.

5.4.12.1 Design Bases

The reactor vessel head vent arrangement is designed to remove noncondensable gases or steam from the reactor coolant system via remote manual operations from the main control room through a pair of valves. The system discharges to the in-containment refueling water storage tank (IRWST).

The reactor vessel head vent system is designed to provide an emergency letdown path that can be used to prevent long-term pressurizer overfill following loss of heat sink events. The reactor vessel head vent is designed to limit the emergency letdown flow rate to within the capabilities of the normal makeup system. The reactor vessel head vent system can also vent noncondensable gases from the reactor head in case of a severe accident.

The system vents the reactor vessel head by using only safety-related equipment. The reactor vessel head vent system satisfies applicable requirements and industry standards, including ASME Code classifications, safety classifications, single-failure criteria, and environmental qualification.

The piping and equipment from the vessel head vent up to and including the second isolation valve are designed and fabricated according to ASME Codes Section III, Class 1 requirements. The remainder of the piping and equipment are design and fabricated in accordance with ASME Code, Section III, Class 3 requirements.

The supports and support structures conform with the applicable requirements of the ASME Code.

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The Class 1 piping used for the reactor vessel head vent is 1-inch schedule 160 and 6-inch schedule 160. In accordance with ASME Section III it is analyzed following the procedures of NC-3600 for Class 2 piping.

The piping stresses meet the requirements of ASME Code, Section III, NC-3600, with a design temperature of 650°F and a design pressure of 2485 psig.

The automatic depressurization system functions as a part of the passive core cooling system. The first stage automatic depressurization system valves are connected to the pressurizer. The valves are designed, constructed, and inspected to ASME Code Class 1 and seismic Category I requirements. Subsection 5.4.6 and Section 6.3 discuss the design bases for the automatic depressurization system and automatic depressurization system valves.

The primary function of the passive residual heat removal heat exchanger and core makeup tank high point vents is to prevent accumulation of noncondensable gases from the reactor coolant system that could interfere with operation of the passive core cooling system. Section 6.3 discusses the design bases for the passive residual heat removal heat exchanger, the core makeup tanks, and their vent lines.

5.4.12.2 System Description

The reactor vessel head vent arrangement consists of two flow paths, each with redundant isolation valves. Orifices are located downstream of each set of head vent isolation valves to limit the reactor vessel head vent flow rate. Table 5.4-18 lists the equipment design parameters. The reactor vessel head vent arrangement is shown on the reactor coolant system piping and instrumentation diagram (Figure 5.1-5).

The head vent arrangement consists of two parallel paths of two 1-inch, open/close, solenoid-operated isolation valves connected to a 1-inch vent pipe located near the center of the reactor vessel head and two sections of 6-inch pipe along this pipe. The system design with two valves in series in each flow path minimizes the possibility of reactor coolant pressure boundary leakage. The solenoid-operated isolation valves are powered by the safety-related Class 1E DC and UPS system. The solenoid-operated isolation valves are fail-closed, normally closed valves. The valves are included in the valve operability program and are qualified to IEEE-323, IEEE-344, and IEEE-382.

The vent system piping is supported such that the resulting loads and stresses on the piping and on the vent connection to the vessel head are acceptable.

The automatic depressurization system valves are included as part of the pressurizer safety and relief valve module attached to the top of the pressurizer and are connected to the pressurizer nozzles. The automatic depressurization system includes a group of valves attached to the reactor coolant system hot leg that are not used to vent noncondensable gases. The pressurizer safety and relief valve module is supported by an attachment to the top of the pressurizer and provides support for the automatic depressurization system valves. The automatic depressurization system valves are active valves required to provide safe shutdown or to mitigate the consequences of postulated accidents. Subsection 5.4.6 discusses the function control and power requirements for the automatic depressurization system valves.

5.4.12.3 Safety Evaluation

The reactor vessel head vent system is designed so that a single failure of the remotely operated vent valves, power supply, or control system does not prevent isolation of the vent path. The two isolation valves in the active flow path provide a redundant method of isolating the venting

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system. With two valves in series, the failure of any one valve does not inadvertently open a vent path or prevent isolation of a flow path. The Chapter 15 accident analysis and supporting analyses are performed consistent with the reactor vessel head vent system design parameters provided in Table 5.4-18.

The reactor vessel head vent system has two normally de-energized valves in series in each flow path. This arrangement eliminates the possibility of opening a flow path due to the spurious movement of one valve.

A break of the reactor vessel head vent system line would result in a small loss of coolant accident of not greater than one-inch diameter. Such a break is similar to those analyzed in Subsection 15.6.5. Since a break in the head vent line would behave similarly to the hot leg break case presented in Subsection 15.6.5, the results presented therein apply to a reactor vessel head vent system line break. This postulated vent line results in no calculated core uncovery.

Subsection 5.4.6 and Section 6.3 discuss the evaluation of the automatic depressurization system valves. Inadvertent opening of an automatic depressurization system valve is included in the transients considered for mechanical system and component design in Subsection 3.9.1.

Section 6.3 discusses the evaluation of the passive residual heat removal heat exchanger and core makeup tanks. These high point vent lines contain two manual isolation valves in series, so that a single failure of either valve to reclose following venting operation does not prevent isolation of the flow path. The high point vent line from the passive residual heat removal heat exchanger to the in-containment refueling water storage tank contains a flow-restricting orifice such that postulated break flow is within the makeup capability of the chemical and volume control system and therefore would not normally require actuation of the passive safety systems.

5.4.12.4 Inspection and Testing Requirements

Inservice inspection of ASME Code Classes 2 and 3 components is conducted according to Section 6.6. Subsection 3.9.6 discusses inservice testing and inspection of valves. Subsection 5.2.4 discusses inservice inspection and testing of ASME Code, Class 1 components that are part of the reactor coolant pressure boundary.

The requirements for tests and inspections for reactor coolant system valves is found in Subsection 5.4.8.4. In addition, tests for the reactor vessel head vent valves and piping are conducted during preoperational testing of the reactor coolant system, as discussed in Section 14.2.

5.4.12.4.1 Flow Testing

Initial verification of the capacity of the reactor vessel head vent valves is performed during the plant initial test program. A low pressure flow test and associated analysis is conducted to determine the capacity of each reactor vessel head vent flow path. The reactor coolant system is at cold conditions with the pressurizer full of water. The normal residual heat removal pumps are used to provide injection flow into the reactor coolant system, discharging through the reactor vessel head vent valves. The measured flow rate at low pressure is such that the head vent flow capacity is at least 9.0 lbm/sec at an RCS pressure of 2500 psia.

5.4.12.5 Instrumentation Requirements

The reactor head vent valves can be operated from the control room or the remote shutdown workstation. The isolation valves in the vent line and automatic depressurization system valves have position sensors. The position indication from each solenoid-operated isolation valve is monitored in the control room.

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5.4.13 Core Makeup Tank

The core makeup tank (CMT) in the passive core cooling system stores cold borated water under system pressure for high pressure reactor coolant makeup. See Section 6.3 for a discussion of the operation of the core makeup tank in the passive core cooling system and the connections to the core makeup tank.

5.4.13.1 Design Bases

The core makeup tank is designed and fabricated according to the ASME Code, Section III as a Class 1 component. See Subsection 5.2.1. The boundaries of the ASME Code include the pressure-containing materials up to, but excluding, the circumferential welds at nozzle safe ends. The manway cover and bolting materials are included within this boundary. The core makeup tank is AP1000 equipment Class A (ANS Safety Class 1, Quality Group A). Stresses are maintained within the limits of the ASME Code, Section III. Section 5.2 provides the ASME Code and material requirements. Subsection 5.2.4 discusses inservice inspection.

Materials of construction are specified to minimize corrosion/erosion and to provide compatibility with the operating environment, including the expected radiation level. Subsection 5.2.3 discusses the welding, cutting, heat treating and other processes used to minimize sensitization of stainless steel.

Instrumentation nozzles are welded to the clad inside wall of the vessel according to ASME Code, Section III. Butt welds, branch connection nozzle welds, and boss welds are of a full-penetration design. Flanges conform to ANSI B16.5.

The transients used to evaluate the core makeup tank are based on the system design transients described in Subsection 3.9.1.1. In addition to normal reactor coolant system transients, two additional Service Level B transients affect only the core makeup tank. There are an assumed 30 occurrences of the first transient, leakage at power, in the plant lifetime. This event covers situations which a small leak draws in hot reactor coolant system fluid. There are an assumed 10 occurrences in the plant lifetime of the second transient, increase in containment temperature above normal operating range.

5.4.13.2 Design Description

The core makeup tank is a low-alloy steel vessel with 308L stainless steel internal cladding. The minimum free internal volume for the core makeup tank is 2487 cubic feet. The normal full-power temperature and pressure in the core makeup tank are 70° to 120°F and 2250 psia, respectively. The tank is designed to withstand the design environment of 2500 psia and 650°F. The core makeup tank is a vertically mounted, cylindrical pressure vessel with hemispherical top and bottom heads.

The core makeup tank is supported on columns. One nozzle on the lower head connects the tank to the reactor vessel direct vessel injection (DVI) piping. One nozzle in the center of the upper head connects the tank to a line connected to one of the RCS cold legs. The top nozzle incorporates a diffuser inside the tank. The diffuser has the same diameter and thickness as the connecting piping. The bottom of the diffuser is plugged (with the exception of a small drain hole) and holes are drilled in the side. The diffuser forces the steam flow to turn 90 degrees which limits the steam penetration into the coolant in the core makeup tank. The core makeup tank includes a manway and cover in the shell to allow access to the tank interior.

To maintain system pressure, the flowpath from the reactor coolant system cold leg to the upper head of the core makeup tank is normally open. The core makeup tank discharge piping flow path from the lower head to the reactor vessel is blocked by two normally closed, fail-open, parallel isolation valves. See Section 6.3 for a description of the system operation.

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The tank includes nozzles and flanges for connection to level detection instrumentation.

Three sample lines, one in the upper head, one in the vessel shell, and one in the lower head, are provided for sampling the solution in the core makeup tank. A fill connection is provided for core makeup tank make up water from the chemical and volume control system.

5.4.13.3 Design Evaluation

Subsection 3.9.3 discusses the loading combinations, stress limits, and analytical methods for the structural evaluation of the reactor coolant system core makeup tank for design conditions, normal conditions, anticipated transients, and postulated accident conditions. Subsection 3.9.2 discusses the requirements for dynamic testing and analysis. The reactor coolant system design transients for normal operation, anticipated transients and postulated accident conditions are discussed in Subsection 3.9.1.

Stress intensities resulting from design loads do not exceed the limits specified in ASME Code, Section III. The rules for the evaluation of the faulted conditions are defined in Appendix F of the ASME Code, Section III. Only those stress limits applicable for an elastic system analysis are used for the external load analysis.

5.4.13.4 Material Corrosion/Erosion Evaluation

Those portions of the core makeup tank in contact with reactor coolant are fabricated from or clad with stainless steel. The water chemistry of the core makeup tank, comparable to reactor coolant, causes minimal corrosion of the stainless steel. Erosion is not an issue, since there is normally no flow. A periodic analysis of the coolant chemical composition verifies that the reactor coolant quality meets the specifications, as discussed in Subsection 5.2.3.

Contamination of stainless steel and nickel-chromium-iron alloys by copper, low-melting-temperature alloys, mercury, and lead is prohibited. The material selection, water chemistry specification, and residual stress in the piping minimize the potential for stress corrosion cracking, as discussed in Subsection 5.2.3.

5.4.13.5 Test and Inspections

Charpy V-notch tests and drop-weight fracture toughness tests are performed as required. Orientation of test specimens is according to the ASME Code, Section III, except that the material is not considered to be subjected to high irradiation.

Compliance with the sensitization requirement is demonstrated by passing the susceptibility to intergranular attack test of ASTM A-262, Practice A or E. Inservice inspection requirements for Class 1 are discussed in Subsection 5.2.4.

In addition, materials and welds are inspected according to the requirements of the ASME Code, Section III Class 1.

5.4.14 Passive Residual Heat Removal Heat Exchanger

The passive residual heat removal heat exchanger (PRHR HX) is the component of the passive core cooling system that removes core decay heat for any postulated non-loss of coolant accident event where a loss of cooling capability via the steam generators occurs. Section 6.3 discusses the operation of the passive residual heat removal heat exchanger in the passive core cooling system.

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5.4.14.1 Design Bases

The passive residual heat removal heat exchanger automatically actuates to remove core decay heat for 72 hours as discussed in Section 6.3, assuming the condensate from steam generated in the in-containment refueling water storage tank (IRWST) is returned to the tank. The passive residual heat removal heat exchanger is designed to withstand the design environment of 2500 psia and 650°F.

The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour. The passive residual heat removal heat exchanger will prevent water relief from the pressurizer and remove sufficient decay heat from the reactor coolant system to satisfy the applicable post-accident safety evaluation criteria detailed in Chapter 15 for at least 72 hours. In addition, the passive residual heat removal heat exchanger will cool the reactor coolant system, with reactor coolant pumps operating or in the natural circulation mode, so that the reactor coolant system pressure can be lowered to reduce stress levels in the system if required. See Section 6.3 for a discussion of the capability of the passive core cooling system.

The passive residual heat removal heat exchanger is designed and fabricated according to the ASME Code, Section III, as a Class 1 component. Those portions of the passive residual heat exchanger that support the primary-side pressure boundary and falls under the jurisdiction of ASME Code, Section III, Subsection NF are AP1000 equipment Class A (ANS Safety Class 1, Quality Group A). Stresses for ASME Code, Section III equipment and supports are maintained within the limits of Section III of the Code. Section 5.2 provides ASME Code, Section III and material requirements. Subsection 5.2.4 discusses inservice inspection.

Materials of construction are specified to minimize corrosion/erosion and to provide compatibility with the operating environment, including the expected radiation level. Subsection 5.2.3 discusses the welding, cutting, heat treating and other processes used to minimize sensitization of stainless steel.

5.4.14.2 Design Description

The passive residual heat removal heat exchanger consists of an upper and lower tubesheet mounted through the wall of the in-containment refueling water storage tank. A series of 0.75-inch outer diameter C-shaped tubes connect the tubesheets shown in Figure 6.3-5. The primary coolant passes through the tubes, which transfer decay heat to the in-containment refueling water storage tank water and generate enough thermal driving head to maintain the flow through the heat exchanger during natural circulation. The design minimizes the diameter of the tubesheets and allows ample flow area between the tubes in the in-containment refueling water storage tank.

The horizontal lengths of the tubes and lateral support spacing in the vertical section allow for the potential temperature difference between the tubes at cold conditions and the tubes at hot conditions. The tubes are supported in the in-containment refueling water storage tank interior with a frame structure.

The passive residual heat removal heat exchanger is welded to the in-containment refueling water storage tank.

The design of the passive residual heat removal heat exchanger incorporates a flexible member at the heat exchanger to in-containment refueling water storage tank interface to minimize the load imposed on the wall of the in-containment refueling water storage tank resulting from thermal expansion on the tubesheet.

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5.4.14.3 Design Evaluation

Subsection 3.9.3 discusses the loading combinations, stress limits, and analytical methods for the structural evaluation of the passive residual heat removal heat exchanger for design conditions, normal conditions, anticipated transients, and postulated accident conditions. Operation of passive residual heat removal heat exchanger is evaluated using Service Levels B, C, and D plant conditions. In addition to loads due to conditions in the reactor coolant system and operation of the passive residual heat removal heat exchanger, the passive residual heat removal heat exchanger is evaluated for hydraulic loads due to discharge of steam from the automatic depressurization system valves into a sparger in the in-containment refueling water storage tank. These loads are evaluated using Service Level B limits and are not combined with any other Service Level C or D conditions.

Seismic, loss of coolant accident, sparger activation and flow-induced vibration loads are derived using dynamic models of the passive residual heat removal heat exchanger. The dynamic analysis considers the hydraulic interaction between the coolant (steam or water) and the system structural elements.

Subsection 3.9.2 discusses the requirements for dynamic testing and analysis. Subsection 3.9.1 discusses the reactor coolant system design transients for normal operation, anticipated transients, and postulated accident conditions. In addition to reactor coolant system design transients, there are two additional Service Level B transients that affect only the passive residual heat removal heat exchanger. In the plant lifetime, there are an assumed 30 occurrences of the first transient, leakage at power. This event covers situations in which a small leak in the manway cover draws in hot reactor coolant system fluid. There are an assumed 10 occurrences in the plant lifetime of the second transient, increase in in-containment refueling water storage tank temperature, due an event which activates passive core cooling.

Stress intensities resulting from design loads do not exceed the limits specified in ASME Code, Section III. The rules evaluating the Service Level D conditions are defined in Appendix F of the ASME Code, Section III. Only those stress limits applicable for an elastic system analysis are used for the external load analysis.

During normal plant operation the system is pressurized to the reactor coolant system hot leg pressure at the temperature of the in-containment refueling water storage tank. The pressure transients during normal plant operation are the same as those for the reactor coolant system hot leg. There is no flow through the passive residual heat removal heat exchanger during normal plant operation. The tubesheet temperatures are calculated to provide sufficient temperature drop between the tubesheet and the attachment to the tank. Section 6.3 describes the passive residual heat removal heat exchanger performance characteristics.

5.4.14.4 Material Corrosion/Erosion Evaluation

Those portions of the passive residual heat removal heat exchanger in contact with reactor coolant are fabricated from or clad with corrosion-resistant material. The use of severely sensitized austenitic stainless steel in the pressure boundary of the reactor coolant system is prohibited. A periodic analysis of the coolant chemical composition verifies that the reactor coolant quality meets the specifications discussed in Subsection 5.2.3.

Sulphur, lead, copper, mercury, aluminum, antimony, arsenic, and other low-melting-point elements and their alloys and compounds are restricted in their use as construction materials, erection aids, cleaning agents, and coatings for finished surfaces of the passive residual heat removal heat exchanger that are in contact with reactor coolant system fluid or in-containment refueling water storage tank. Contamination of stainless steel and nickel-chromium-iron alloys by copper, low-melting-temperature alloys, mercury, and lead is prohibited. The material selection, water chemistry

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specification, and residual stress in the piping minimize the potential for stress corrosion cracking, as discussed in Subsection 5.2.3.

Stainless steel and nickel-chromium-iron alloys used in the passive residual heat removal heat exchanger are procured to ASME specifications.

5.4.14.5 Testing and Inspections

The passive residual heat removal heat exchanger is designed and manufactured to permit inservice testing and examination as specified in the ASME Code, Section XI. Methods and techniques developed for steam generator tube eddy current inspection can be used for the passive residual heat removal heat exchanger tubes.

Preservice and inservice inspection of the PRHR HX is performed according to Subsection 5.2.4. PRHR HX tube integrity is verified using inspection procedures that consider the methods and techniques developed for steam generator tube eddy current inspection.

Access for inspection and maintenance is possible through manways in the top and bottom channel heads without draining the in-containment refueling water storage tank.

Hydrostatic tests are performed in accordance with the requirements of the ASME Code, Section III, using working fluids meeting the appropriate water chemistry specifications.

5.4.15 Combined License Information

The Steam Generator Tube Surveillance Program is addressed in Subsection 5.4.2.5.

5.4.16 References

1. Not used.

2. Hagg, A. C. and Sankey, G. O., “The Containment of Disk Burst Fragments by Cylindrical Shells,” ASME Journal of Engineering for Power, April 1974, pp. 114-123.

3. Not used.

4. Not used.

5. ANSI/ANS-5.1-1994, “Decay Heat Power in Light Water Reactors.”

6. ANSI/ANS-51.1-1983, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.”

7. Not used.

8. ASME QME-1-2007 Edition, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.”

9. ASME B16.34-1996, “Valves – Flanged, Threaded, and Welding End.”

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10. Curtiss-Wright Electro-Mechanical Corporation Report AP1000RCP-06-009-P, Revision 3 (Proprietary), and AP1000RCP-06-009-NP, Revision 3 (Non-Proprietary), “Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel,” APP-MP01-GLR-001P (Proprietary) and APP-MP01-GLR-001NP (Non-Proprietary), November 2016.

201. Nuclear Energy Institute, “Steam Generator Program Guidelines,” NEI 97-06.

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Table 5.4-1
Reactor Coolant Pump Design Parameters

Unit design pressure (psia)	2500
Unit design temperature (°F)	650
Estimated Unit overall height (ft)	22
Component cooling water flow (gpm)	650
Maximum continuous component cooling water inlet temperature (°F)(1)	95
Total estimated weight motor and casing, dry (lb)	200,000
Pump	
Design flow (gpm)	78,750
Developed head (feet)	365
Pump discharge nozzle, inside diameter (inches)	22
Pump suction nozzle, inside diameter (inches)	26
Speed (synchronous)(rpm)	1800
Motor	
Type	Squirrel Cage Induction
Voltage (V)	6900
Phase	3
Frequency (Hz)	60
Insulation class	Class H or N
Current (amp) Starting Nominal input, cold reactor coolant	 Variable Variable
Motor/pump rotor minimum required moment of inertia	Sufficient to provide flow coastdown as given in Figure 15.3.2-1

Note:
1. An elevated component cooling water supply temperature of up to 110°F may occur for a 6-hour period.

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Table 5.4-2 Not Used

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Table 5.4-3
Reactor Coolant Pump Quality Assurance Program

	RT[a]	UT[a]	PT[a]	MT[a]
Castings – Casing (or pressure boundary)	X		X	
Flywheel		X	X	X
Forgings		X		X
Plate			X	
Weldments Circumferential Instrument connections	X[c]		 X X	
Motor terminals[b]		X	X	

Notes:
(a) RT - radiographic, UT - ultrasonic, PT - dye penetrant, MT - magnetic particle
(b) See Subsection 5.4.1.3.3.
(c) RT does not apply to flywheel enclosure.

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Table 5.4-4
Steam Generator Design Requirements

Type	Vertical U-tube Feedring-type
Design pressure, reactor coolant side (psia)	2500
Design pressure, steam side (psia)	1200
Design pressure, primary to secondary (psi)	1650
Design temperature, reactor coolant side (°F)	650
Design temperature, steam side (°F)	600
S/G Power, MWt/unit	1707.5
Total heat transfer surface area (ft^2)	123,538
Steam nozzle outlet pressure, psia	836
Steam flow, lb/hr per S/G	$7.50x10^6$
Total steam flow, lb/hr	$14.99x10^6$
Maximum moisture carryover (weight percent) maximum	0.25[(a)]
No load temperature, °F	557
Feedwater temperature, °F	440
Number of tubes per unit	10,025
Tube outer diameter, inch	0.688
Tube wall thickness, inch	0.040
Tube pitch, inches	0.980 (triangular)

Note:
a. Moisture content at the outlet of the steam generator secondary separators (dryers).

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Table 5.4-5
Steam Generator Design Parameters
(Nominal Values)

Tube pitch, inches	0.980 (triangular)
Overall length, inches	884.29*
Upper shell I.D., inches	210
Lower shell I.D., inches	165
Tubesheet thickness, inches	31.417**
Primary water volume, ft^3	2077
Water volume in tubes, ft^3	1489
Water volume in plenums, ft^3	588
Secondary water volume, ft^3	3517.3
Secondary steam volume, ft^3	5543.4
Secondary water mass, lbm	169,251
Design fouling factor, hr-°F-ft^2/BTU	9.0×10^{-5}

Notes:
* Measured from steam nozzle to the flat, exterior portion of the channel head.
** Base metal thickness.

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Table 5.4-6
Steam Generator Quality Assurance Program

	RT[(a)]	UT[(a)]	PT[(a)]	MT[(a)]	ET[(a)]
Base Metals					
Tubesheet Forging		Yes		Yes	
Channel Head Forging Plate Casting	 Yes	 Yes Yes 	 	 Yes Yes	
Secondary Shell and Head Forgings Plate		 Yes Yes		 Yes	
Tubes		Yes			Yes
Nozzles (Forgings)		Yes		Yes	
Safe ends		Yes	Yes		
Welds					
Channel head if fabricated	Yes			Yes	
Pressure boundary, longitudinal if fabricated	Yes			Yes	
Pressure boundary, circumferential	Yes			Yes	
Primary nozzles to fabricated head	Yes			Yes	
Primary nozzles to forged head	Yes			Yes	
Manways to fabricated head or shell	Yes			Yes	
Manways to forged head or shell	Yes			Yes	
Steam and feedwater nozzles to fabricated shell	Yes			Yes	
Steam and feedwater nozzles to forged shell	Yes			Yes	
Support brackets				Yes	
Tube to tubesheet			Yes		
Instrument connections (secondary)				Yes	
Temporary attachments after removal				Yes	
After hydrostatic test (all major pressure boundary welds and complete cast channel head where accessible)				Yes	
Weld deposit on primary nozzles	Yes		Yes		
Safe end to nozzle	Yes		Yes		
Cladding					
Tubesheet		Yes[(b)]	Yes		
Channel head		Yes	Yes		
Cladding (channel head-tubesheet joint cladding restoration)		Yes	Yes		

Notes:
(a) RT – Radiographic, UT – Ultrasonic, PT – Dye penetrant, MT – Magnetic particle, ET – Eddy current.
(b) Flat surfaces only

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Table 5.4-7
Reactor Coolant System Piping Design

Reactor Coolant Loop Piping	
Design Pressure (psig)	2485
Design Temperature (°F)	650
Reactor Inlet Piping	
Inside Diameter (ID)	22
Nominal Wall Thickness	2.56
Reactor Outlet Piping	
Inside Diameter (ID)	31
Nominal Wall Thickness	3.25
Pressurizer Surge Line	
Design Pressure (psig)	2485
Design Temperature (°F)	680
Pressurizer Surge Line Piping	
Nominal Pipe Size	18
Nominal Wall Thickness	1.78
Pressurizer Safety Valve and ADS Valve Inlet Line	
Design Pressure (psig)	2485
Design Temperature (°F)	680
Other Rector Coolant Branch Lines	
Design Pressure (psig)	2485
Design Temperature (°F)	650

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Table 5.4-8
Reactor Coolant System Piping Quality Assurance Program

	RT[(a)]	UT[(a)]	PT[(a)]
Pipe (Forged Seamless)		Yes	Yes
Fittings		Yes	Yes
Weldments			
Circumferential Butt Welds	Yes		Yes
Branch Nozzle Connections	Yes[(b)]		Yes
Fillet Weld Instrument Connections			Yes

Notes:
(a) RT - Radiographic; UT - Ultrasonic; PT - Dye Penetrant
(b) No RT is required for branch nozzle connections of 4 inch nominal size smaller.

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Table 5.4-9
Pressurizer Design Data

Design pressure (psig)	2485
Design temperature (°F)	680
Surge line nozzle nominal diameter (in.)	18
Spray line nozzle nominal diameter (in.)	4
Safety valve nozzle nominal diameter (in.)	14
Internal volume (ft^3)	2100

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Table 5.4-10
Pressurizer Heater Group Parameters

Power Supply Voltage (Vac)	480
Heater Rated Voltage (Vac)	460
Frequency (Hz.)	60
Power Capacity (kW)	
Control Group	369
Backup Group A	246
Backup Group B	246
Backup Group C	369
Backup Group D	369

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Table 5.4-11
Reactor Coolant System Design Pressure Settings

	Base Load Mode (Psig)
Hydrostatic test pressure	3106
Design pressure	2485
Safety valves (begin to open)	2485
High pressure reactor trip	2405
High pressure alarm	2335
Pressurizer spray valves (full open)	2310
Pressurizer spray valves (begin to open)	2260
Proportional heaters (begin to operate)	2250
Operating pressure	2235
Proportional heater (full operation)	2220
Backup heaters on	2210
Low pressure alarm	2085
Low pressure safeguards actuation	1855

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Table 5.4-12
Pressurizer Quality Assurance Program

	RT[a]	UT[a]	PT[a]	MT[a][c]
Heads				
Forged head		Yes		Yes
Cladding		Yes	Yes	
Shell				
Forgings		Yes		Yes
Cladding		Yes	Yes	
Heaters				
Tubing		Yes[b]	Yes	
Centering of element	Yes			
Nozzle (Forgings)		Yes	Yes[c]	Yes[c]
Weldments				
Shell, circumferential	Yes			Yes
Nozzle to head (if fabricated)	Yes			Yes
Cladding		Yes	Yes	
Nozzle safe end	Yes		Yes	
Instrument nozzle			Yes	
Temporary attachments (after removal)				Yes
Boundary welds (after shop hydrostatic tests)				Yes
Support brackets				Yes

Notes:
(a) RT - Radiographic, UT - Ultrasonic, PT - Dye Penetrant, MT - Magnetic Particle.
(b) Eddy current testing can be used in lieu of UT.
(c) Dye penetrant (PT) examination is acceptable when accessibility limitations prohibit the use of magnetic particle (MT) test equipment.

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Table 5.4-13
Design Bases for Normal Residual Heat Removal System Operation

RNS initiation, hours after reactor shutdown	4
RCS initial pressure (psig)	450
RCS initial temperature (°F)	350
CCS Design Temperature (°F)[(a)]	95
Cooldown time, (hours after shutdown)	96
RCS temperature at end of cooldown (°F)	125

Note:
(a) The maximum CCS temperature during cooldown is 110°F.

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Table 5.4-14
Normal Residual Heat Removal System Component Data

Normal RHR Pumps (per pump)		
Minimum Flow Required for Shutdown Cooling (gpm)	1400	
Minimum Flow Required for Low Pressure Makeup (gpm)	1100	
Design Flow (gpm)	1500	
Design Head (ft)	360	
Normal RHR Heat Exchangers		
Minimum UA Required for Shutdown Cooling (BTU/hr-°F)	2.2×10^6	
Design Heat Removal Capacity (BTU/hr)(1)	23×10^6	
	Tube Side	**Shell Side**
Design Flow (lb/hr)	750,000	1,405,000
Inlet Temperature (°F)	125	87.5
Outlet Temperature (°F)	94	104
Fluid	Reactor Coolant	CCS

Note:
(1) Design heat removal capacity is based on decay heat at 96 hours after reactor shutdown.

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Table 5.4-15
Reactor Coolant System Valve Design Parameters

Design pressure (psig)	2485
Preoperational plant hydrotest (psig)	3106
Design temperature (°F)	
Reactor coolant system	650
Pressurizer safety valves and ADS valves	680

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Table 5.4-16
Reactor Coolant System Motor-operated Valves Design Opening and Closing Pressures

	Normal ΔP (PSIG)[a]		Design ΔP (PSIG)	
	OPEN	**CLOSE**	**OPEN**	**CLOSE**
First Stage ADS Valves (RCS-PL-V001A & B)	2235	2235[b,c]	2485	2485
First Stage ADS Isolation Valves (RCS-PL-V011A & B)	2235	2235	2485	2485
Second Stage ADS Valves (RCS-PL-V002A & B)	1200	100[b]	2485	1200
Second Stage ADS Isolation Valves (RCS-PL-V012A & B)	1200	100	2485	1200
Third Stage ADS Valves (RCS-PL-V003A & B)	500	100	2485	1200
Third Stage ADS Isolation Valves (RCS-PL-V013A & B)	500	100	2485	1200
Fourth Stage ADS Isolation Valves (RCS-PL-V014A & B)	N/A[e]	0	200[e]	200
CVS Purification Isolation Valves (CVS-PL-V001,-V002,-003)	2235	2235	2485	2485
Normal RHR Inner/Outer Isolation Valves (RNS-PL-V001A,B -V002A,B)[d]	450	450	600	600

Notes:
(a) Normal expected operating pressures.
(b) Valves are prevented from closing until ADS signal is reset.
(c) First stage ADS valve can be manually actuated for controlled depressurizations or gas venting.
(d) Valves are administratively blocked from opening at the motor control center.
(e) Fourth stage ADS block valves are normally open.

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Table 5.4-17
Pressurizer Safety Valves – Design Parameters

Number	2
Minimum required relieving capacity per valve (lb/hr)	750,000 at 3% accumulation
Set pressure (psig)	2485 ±25 psi
Design temperature (°F)	680
Fluid	Saturated steam
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	500
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100
Residual Heat Removal Relief Valve - Design Parameters	
Number	2 (RNS-V021/RNS-V020)
Required relieving capacity per valve, Reactor Vessel Low Temperature Overpressure Protection (LTOP) flowrate (gpm)	1061/50
Required relieving capacity per valve, Normal Residual Heat Removal System (RNS) ASME Section III Overpressure Event flowrate (gpm)	944/50
Nominal set pressure (psig)	500*/470*
Full-open pressure, with accumulation (psig)	550*/517*
RNS design temperature (°F)	400
Fluid	Reactor coolant
Pressure-Relieving Temperature LTOP (°F)	300
Pressure-Relieving Temperature RNS ASME Section III Overpressure Protection (°F)	400
Backpressure	
Normal (psig)	3 to 5
Expected maximum during LTOP event discharge (psig)	40.2
Expected maximum during RNS ASME Section III overpressure event discharge (psig)	203
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100

Note:
* See text (Subsection 5.4.9.3) for discussion of set pressure

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Table 5.4-18
Reactor Vessel Head Vent System Design Parameters

System design pressure, psig	2485
System design temperature, °F	650
Number of remotely-operated valves	4
Vent line, nominal diameter, inches	1
Head vent capacity, lbm/sec (assuming a single failure, RCS pressure at 2500 psia)	9.0

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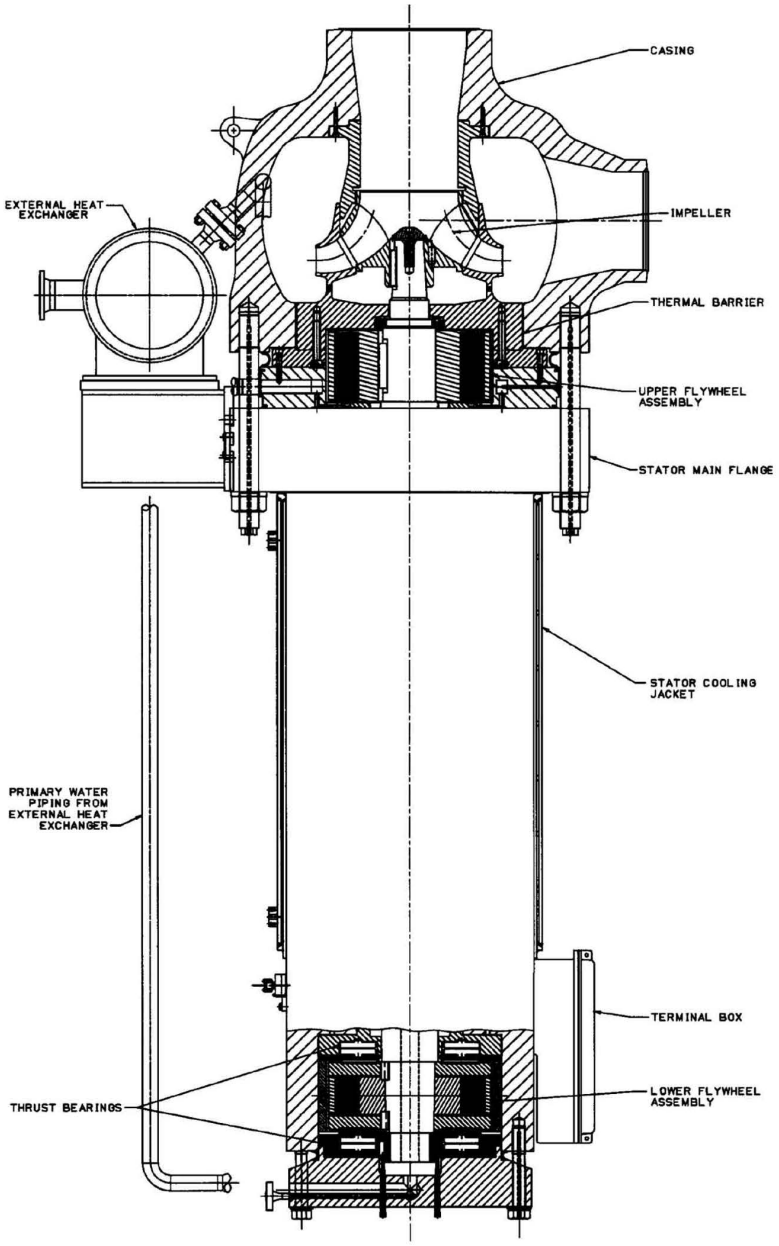

Figure 5.4-1
Reactor Coolant Pump

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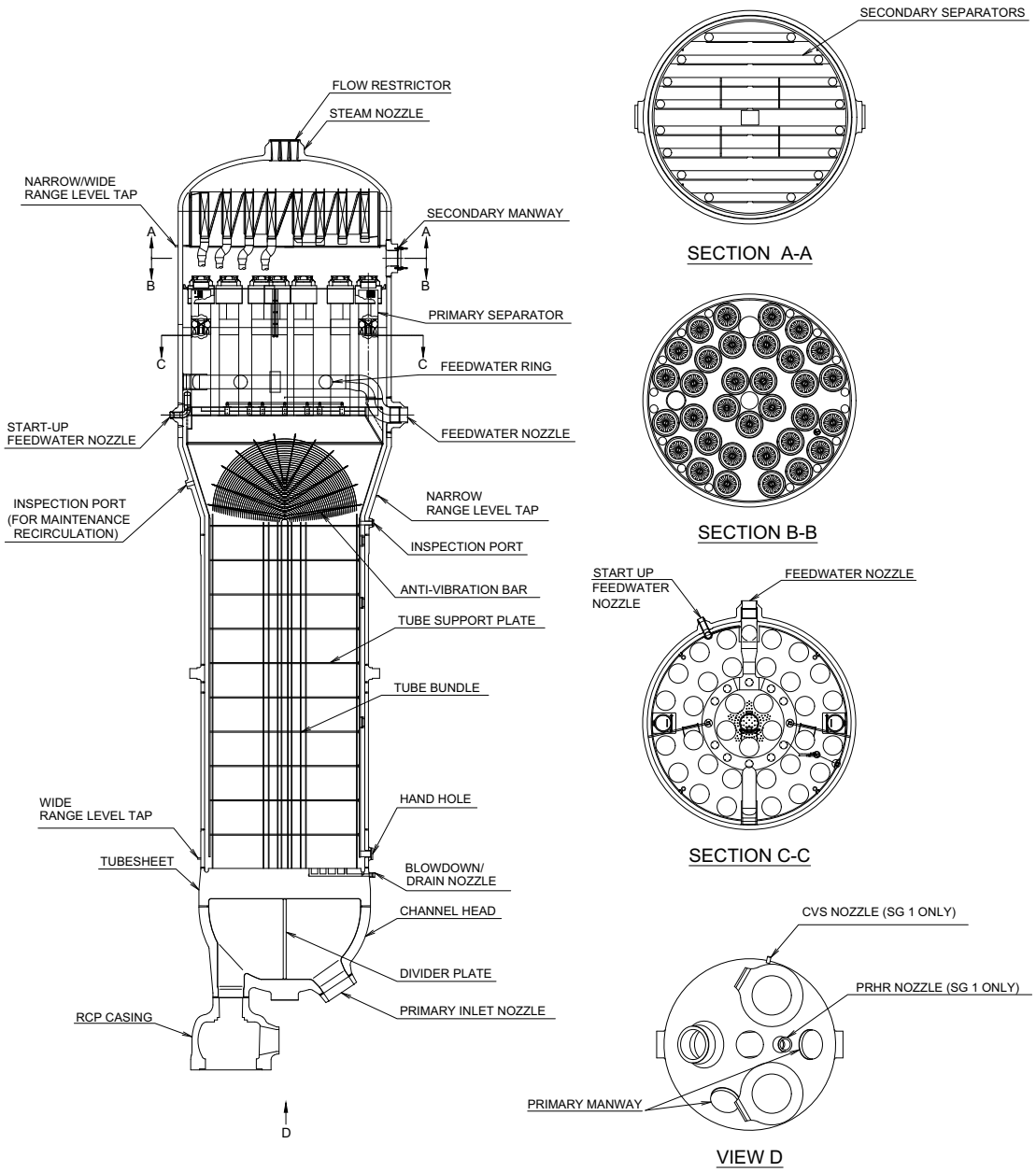

Figure 5.4-2
Steam Generator

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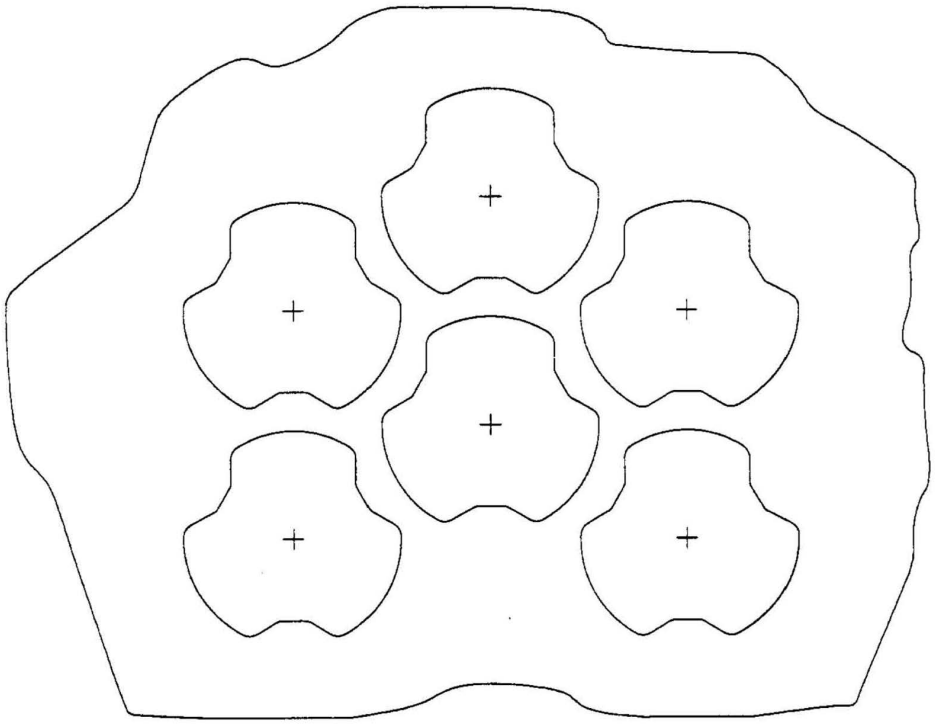

Figure 5.4-3
Support Plate Geometry
(Trifoil Holes)

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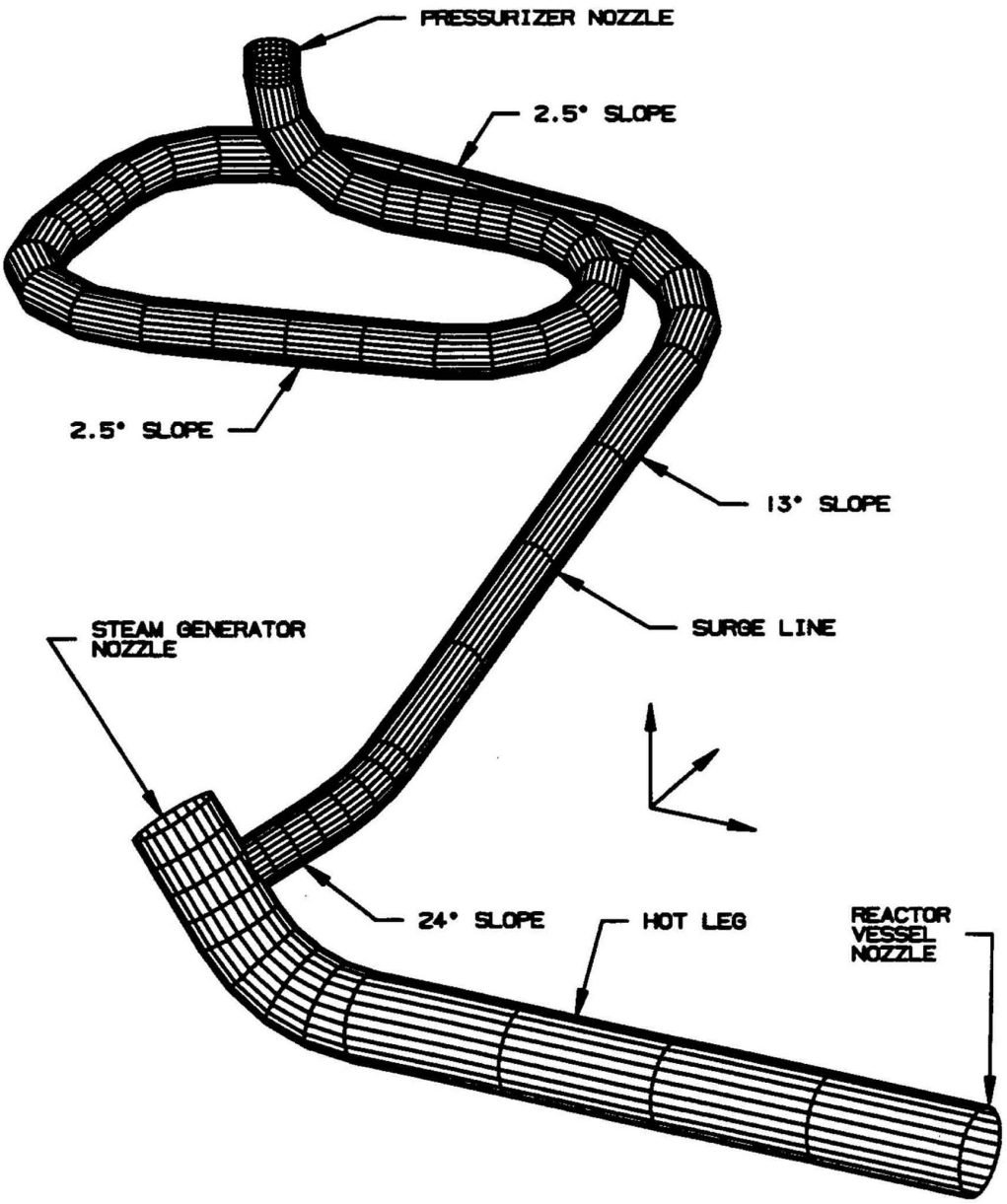

Figure 5.4-4
Surge Line

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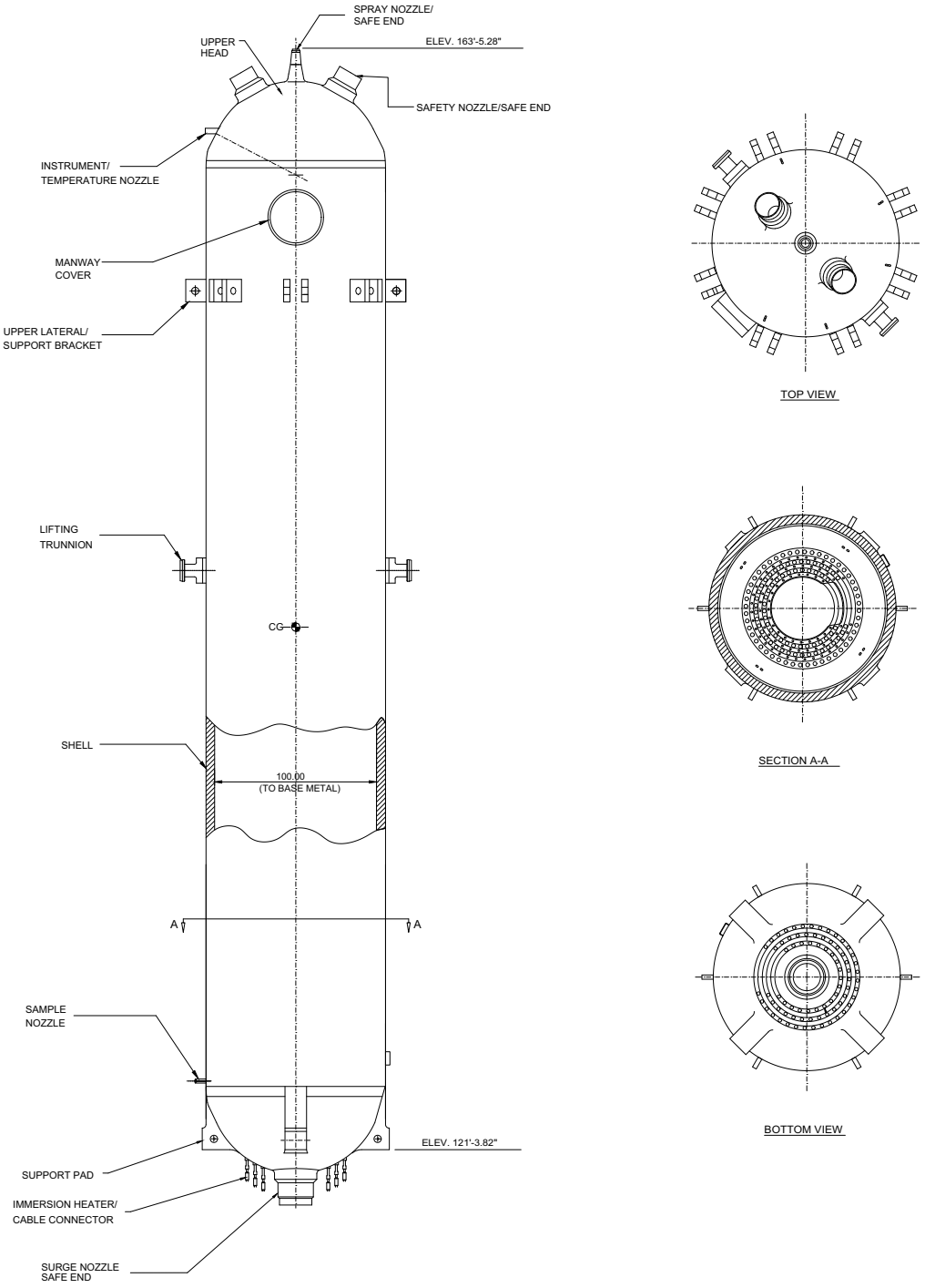

Figure 5.4-5
Pressurizer

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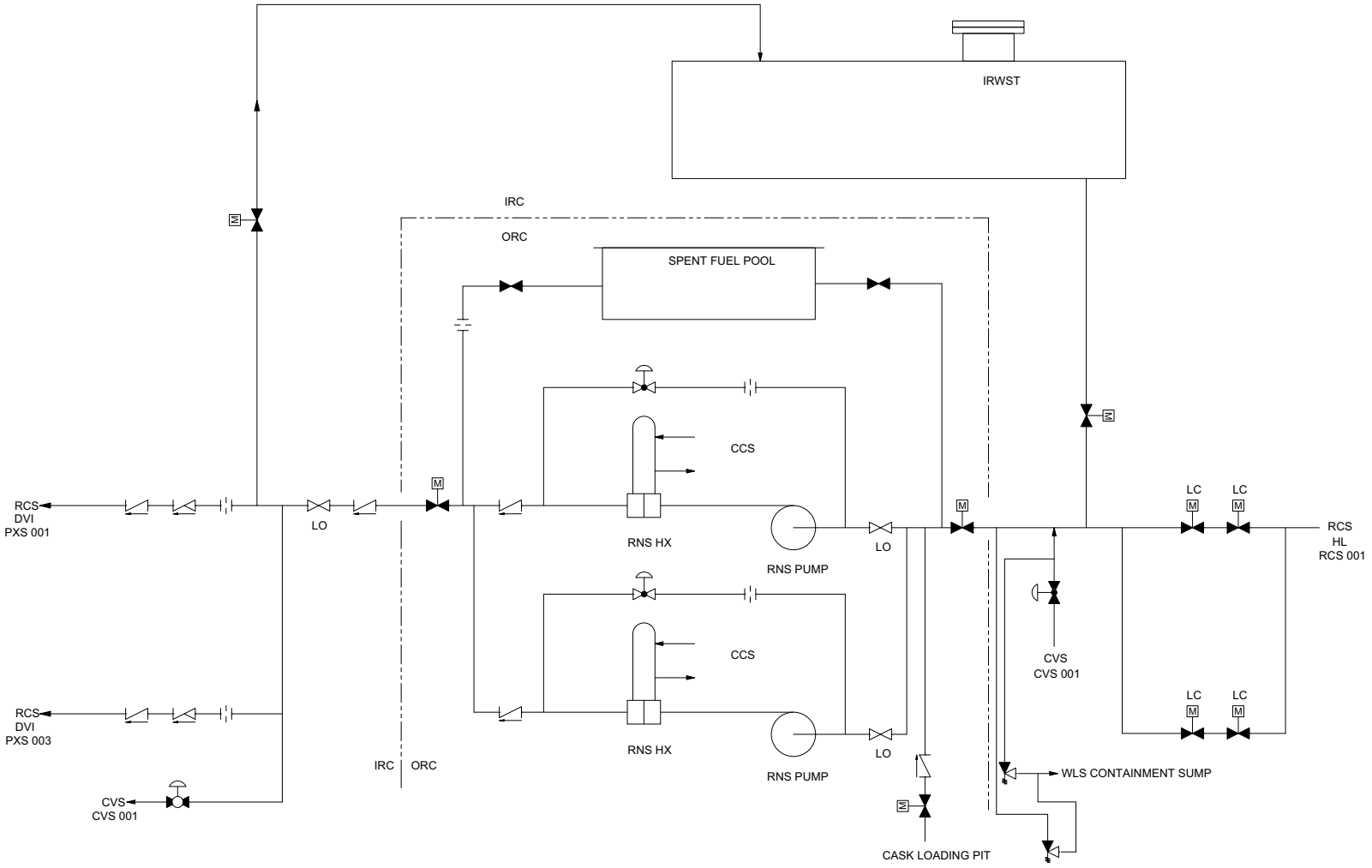

Figure 5.4-6
Normal Residual Heat Removal System

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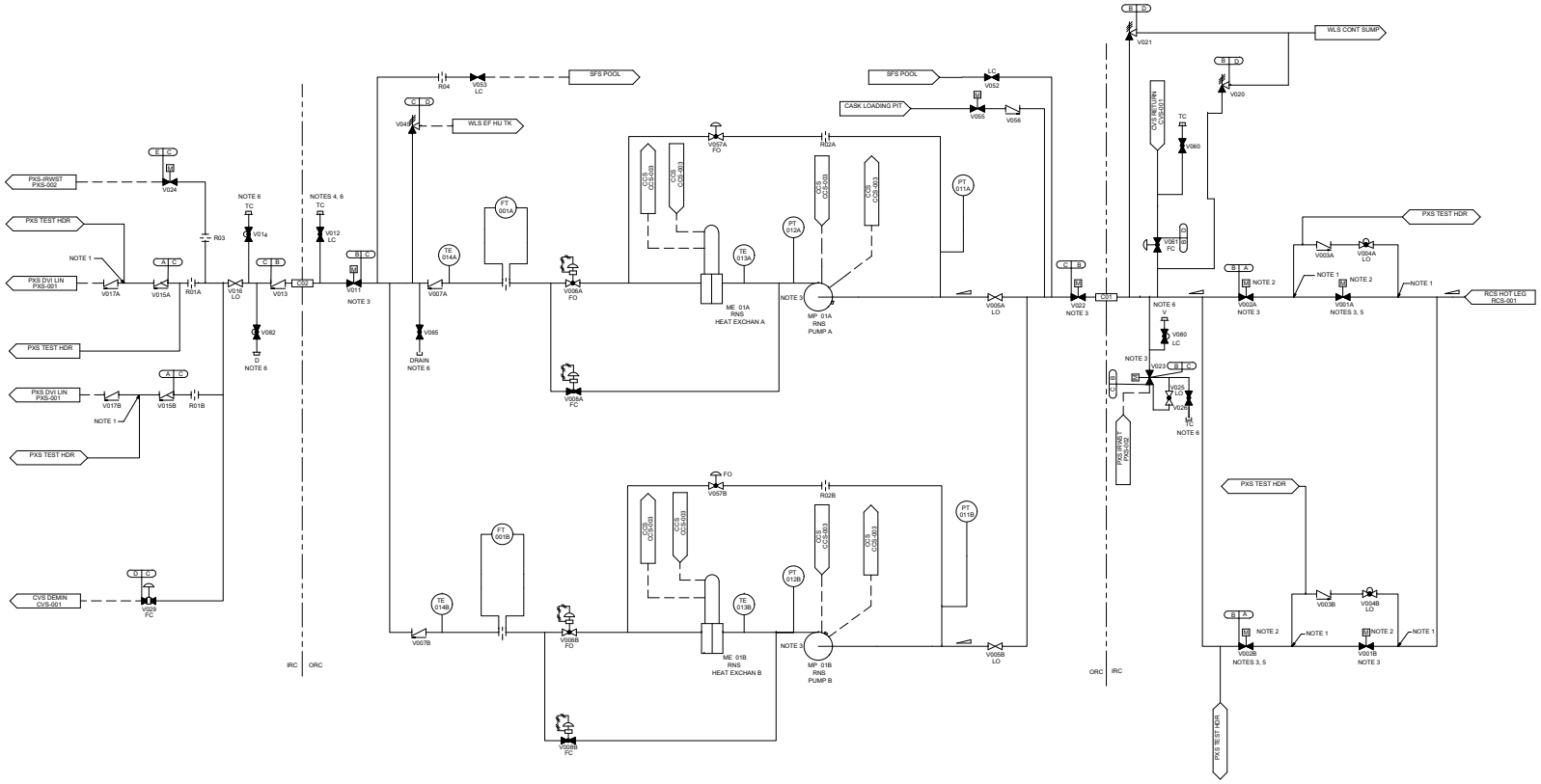

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

NOTES:

1. PROVIDE .375" ID FLOW RESTRICTOR FOR CLASS 1 AND CLASS 2 TRANSITION
2. POWER TO VALVE LOCKED OUT AT THE MOTOR CONTROL CENTER.
3. VALVES CLOSE AND PUMPS STOP ON HIGH CONTAINMENT RADIATION, SAFEGUARDS ACTUATION SIGNAL, OR ADS STAGE 4 ACTUATION SIGNAL.
4. POST-ACCIDENT CONTAINMENT MAKEUP IS ADDED THROUGH TEST CONNECTION.
5. VALVE TO BE ACTUATED ON TIME DELAY AFTER PMS ACTUATION SIGNAL.
6. CAP/PLUG IS NON-SAFETY CLASS E, ASME B31.1.

Figure 5.4-7
Simplified Normal Residual Heat Removal System Piping and Instrument Diagram (REF) RNS 001

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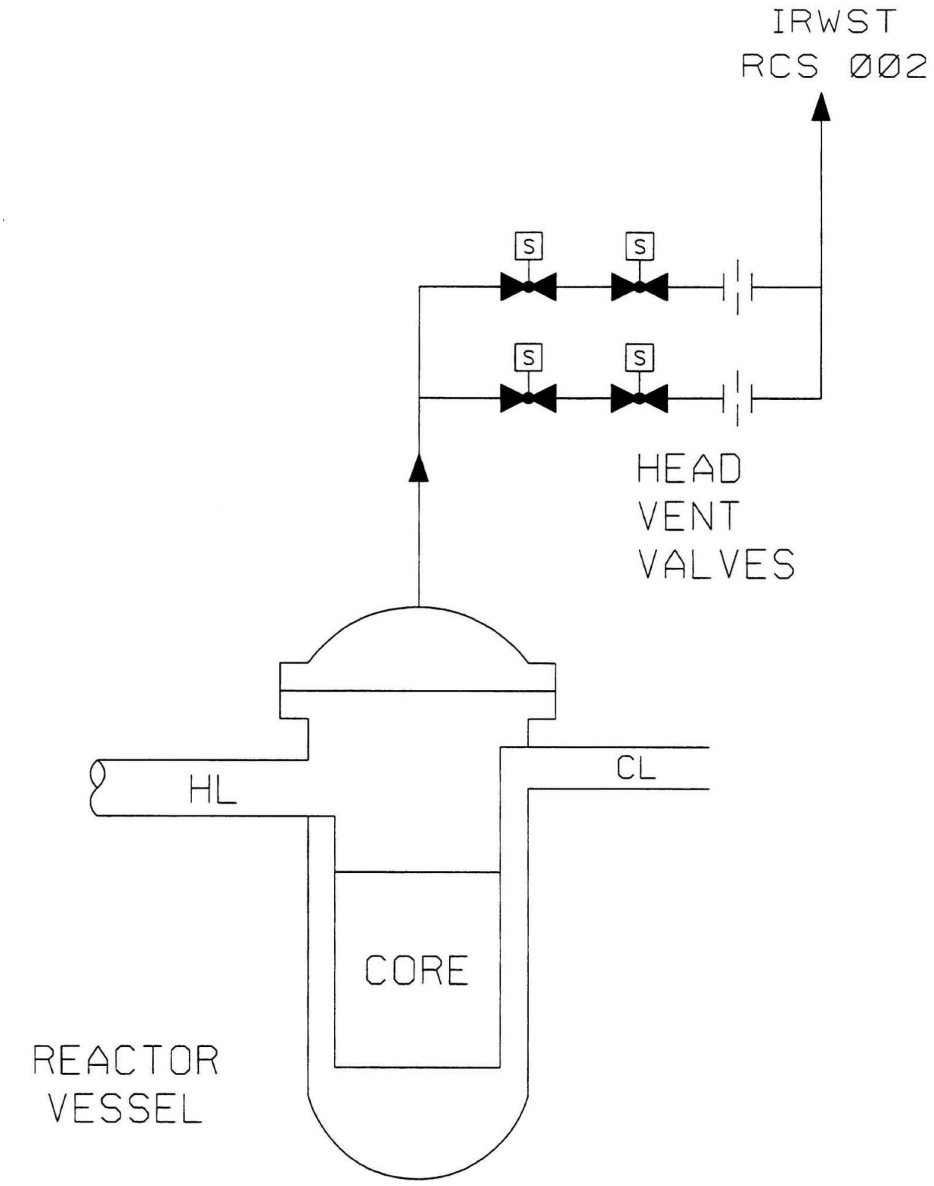

Figure 5.4-8
Reactor Vessel Head Vent System

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Chapter 6 Engineered Safety Features

6.0 Engineered Safety Features

Engineered safety features (ESF) protect the public in the event of an accidental release of radioactive fission products from the reactor coolant system. The engineered safety features function to localize, control, mitigate, and terminate such accidents and to maintain radiation exposure levels to the public below applicable limits and guidelines, such as 10 CFR 50.34. The following are defined as engineered safety features:

Containment

The containment vessel, discussed in Subsection 6.2.1, is a free standing cylindrical steel vessel with ellipsoidal upper and lower heads. It is surrounded by a seismic Category I shield building. The function of the containment vessel, as part of the overall containment system, is to contain the release of radioactivity following postulated design basis accidents. The containment vessel also functions as the safety-related ultimate heat sink by transferring the heat associated with accident sources to the surrounding environment. The following paragraph details this safety-related feature.

Passive Containment Cooling System

The function of the passive containment cooling system, discussed in Subsection 6.2.2, is to maintain the temperature below a maximum value and to reduce the containment temperature and pressure following a postulated design-basis event. The passive containment cooling system removes thermal energy from the containment atmosphere. The passive containment cooling system also serves as the safety-related ultimate heat sink for other design basis events and shutdowns. The passive containment cooling system limits the release of radioactive material to the environment by reducing the pressure differential between the containment atmosphere and the external environment. This diminishes the driving force for leakage of fission products from the containment to the atmosphere.

Containment Isolation System

The major function of the containment isolation system of the AP1000, discussed in Subsection 6.2.3, is to provide containment isolation to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary, if required. This prevents or limits the escape of fission products that may result from postulated accidents. Containment isolation provisions are designed so that fluid lines penetrating the primary containment boundary are isolated in the event of an accident. This minimizes the release of radioactivity to the environment.

Passive Core Cooling System

The primary function of the passive core cooling system, discussed in Section 6.3, is to provide emergency core cooling following postulated design-basis events. The passive core cooling system provides reactor coolant system makeup and boration during transients or accidents where the normal reactor coolant system makeup supply from the chemical and volume control system is lost or is insufficient. The passive core cooling system provides safety injection to the reactor coolant system to provide adequate core cooling for the complete range of loss of coolant accident events up to, and including, the double ended rupture of the largest primary loop reactor coolant system piping. The passive core cooling system provides core decay heat removal during transients, accidents, or whenever the normal heat removal paths are lost.

Main Control Room Emergency Habitability System

The main control room emergency habitability system, discussed in Section 6.4, is designed so that the main control room remains habitable following a postulated design basis event. With a loss of all

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ac power sources, the habitability system will maintain an acceptable environment for continued operating staff occupancy.

Fission Product Control

Post-accident safety-related fission product control for the AP1000, discussed in Section 6.5, is provided by natural removal processes inside containment, the containment boundary, and the containment isolation system. The natural removal processes, including various aerosol removal processes and pool scrubbing, remove airborne particulates and elemental iodine from the containment atmosphere following a postulated design basis event.

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6.1 Engineered Safety Features Materials

This section provides a description of the materials used in the fabrication of engineered safety features components and of the provisions to avoid material interactions that could potentially impair the operation of the engineered safety features. A list of engineered safety features was given previously in Section 6.0. Reactor coolant system materials, including branch piping connected to the reactor coolant system, are described in Subsection 5.2.3.

6.1.1 Metallic Materials

Materials for use in engineered safety features are selected for their compatibility with the reactor coolant system and refueling water.

The edition and addenda of the ASME Code applied in the design and manufacture of each component are the edition and addenda established by the requirements of the Design Certification. The use of editions and addenda issued subsequent to the Design Certification is permitted or required based on the provisions in the Design Certification. The baseline used for the evaluations done to support this safety analysis report and the Design Certification is the 1998 Edition, through the 2000 Addenda. When material is procured to later editions or addenda, the design of the component is reconciled to the new material properties in accordance with the rules of the ASME Code, provided that the later edition and addenda are authorized in 10 CFR 50.55a or in a specific authorization as provided in 50.55a(z).

6.1.1.1 Specifications for Principal Pressure-Retaining Materials

The pressure-retaining materials in engineered safety features system components comply with the corresponding material specification permitted by the ASME Code, Section III, Division 1. The material specifications used for pressure-retaining valves in contact with reactor coolant are the specifications used for reactor coolant pressure boundary valves and piping. See Table 5.2-1 for a listing of these specifications. The material specifications for pressure-retaining materials in each component of an engineered safety features system meet the requirements of Article NC-2000 of the ASME Code, Section III, Class 2, for Quality Group B; Article ND-2000 of the ASME Code, Section III, Class 3, for Quality Group C components; and Article NE-2000 of the ASME Code, Section III for containment pressure boundary components.

Containment penetration materials meet the requirements of Articles NC-2000 or NE-2000 of the ASME Code, Section III, Division 1. The quality groups assigned to each component are given in Section 3.2. The pressure-retaining materials are indicated in Table 6.1-1. Materials for ASME Class 1 equipment are provided in Subsection 5.2.3.

The following subsection provides information on the selection and fabrication of the materials in the engineered safety features of the plant.

Components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or equivalent corrosion-resistant material. The use of nickel-chromium-iron alloy in the engineered safety features is limited to Alloy 690 or its associated weld metals Alloys 52, 52M, 152, and similar alloys developed for improved weldability as allowed by ASME Boiler and Pressure Vessel Code rules. Nickel-chromium-iron alloy is used where the corrosion resistance of the alloy is an important consideration and where the use of nickel-chromium-iron alloy is the choice because of the coefficient of thermal expansion.

The material for the air storage tanks in the main control room emergency habitability system is tested for lateral expansion by the Charpy V-Notch impact testing method per supplementary requirement S3 of material specification SA-372. The lateral expansion shall be 20 mils or greater

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from an average of three tests at or below the lowest service temperature. The lowest expansion value of the three tests shall be 15 mils or greater. The material is not permitted to be weld repaired.

6.1.1.2 Fabrication Requirements

The welding materials used for joining the ferritic base materials of the pressure-retaining portions of the engineered safety features conform to, or are equivalent to, ASME Material Specifications SFA 5.1, 5.5, 5.17, 5.18, 5.20, 5.23, 5.28, 5.29, and 5.30. The welding materials used for joining nickel-chromium-iron alloy in similar base material combination, and in dissimilar ferritic or austenitic base material combination, conform to ASME Material Specifications SFA 5.11 and 5.14, or are similar welding alloys to those in SFA-5.11 or SFA-5.14 developed for improved weldability as allowed by the ASME Boiler and Pressure Vessel Code rules.

The welding materials used for joining the austenitic stainless steel base materials for the pressure-retaining portions of engineered safety features conform to, or are equivalent to, ASME Material Specifications SFA 5.4, 5.9, 5.22, and 5.30. These materials are qualified to the requirements of the ASME Code, Section III and Section IX, and are used in procedures qualified to these same rules. The methods used to control delta ferrite content in austenitic stainless steel weldments in engineered safety features components are the same as those for ASME Code Class 1 components, described in Subsection 5.2.3.4.

The integrity of the safety-related components of the engineered safety features is maintained during component manufacture. Austenitic stainless steel is used in the final heat-treated condition as required by the respective ASME Code, Section II, material specification for the particular type or grade of alloy. Also, austenitic stainless steel materials used in the engineered safety features components are handled, protected, stored, and cleaned according to recognized and accepted methods designed to minimize contamination, which could lead to stress corrosion cracking. These controls for engineered safety features components are the same as those for ASME Code Class 1 components, discussed in Subsection 5.2.3.4. Sensitization avoidance, intergranular attack prevention, and control of cold work for engineered safety features components are the same as the ASME Code Class 1 components discussed in Subsection 5.2.3.4. Cold-worked austenitic stainless steels having a minimum specified yield strength greater than 90,000 psi are not used for pressure boundary components of the engineered safety features in contact with reactor coolant.

Information is provided in Section 1.9 concerning the degree of conformance with the following Regulatory Guides:

- Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal
- Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel

Lead, antimony, cadmium, indium, mercury, and tin metals and their alloys are not allowed to come in contact with engineered safety features component parts made of stainless steel or high alloy metals during fabrication or operation. Bearing alloys containing greater than 1 percent of lead, antimony, cadmium, or indium are not used in contact with reactor coolant.

In accordance with Appendix B to 10 CFR Part 50, the quality assurance program establishes measures to provide control of special processes. One element of control is the review and acceptance of vendor procedures that pertain to the fabrication, welding, and other quality assurance methods for safety related component to determine both code and regulatory conformance. Included in this review and acceptance process are those vendor procedures necessary to provide conformance with the requirements of Regulatory Guides 1.31 and 1.44 for engineered safety features components as discussed in Section 6.1 and reactor coolant system components as discussed in Subsection 5.2.3.

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6.1.1.3 Specifications for Nonpressure-Retaining Materials

Materials for nonpressure-retaining portions of engineered safety features in contact with borated water or other fluids may be procured under ASTM designation. The principle examples of these items are the in-containment refueling water storage tank liner and the passive containment cooling system storage tank liner.

The wetted plate walls of the in-containment refueling water storage tank are fabricated of ASTM A240 UNS S32101. ASTM A240 UNS S32101 is a chromium, manganese, and nitrogen-strengthened duplex stainless steel with higher ultimate tensile and yield strengths than type 304 and 316 stainless steel. This material can be welded using a matching Duplex 2101 (2304 or 2209) filler metal by any of the commonly used stainless steel welding methods, including shielded metal arc welding (SMAW), gas tungsten arc welding TIG (GTAW), gas metal arc welding MIG (GMAW), flux-cored arc welding (FCW), plasma arc welding (PAW), and submerged arc welding (SAW). This material is used for applications where the higher strength allows reductions in weight and material costs. The material has a resistance to intergranular stress corrosion cracking similar to or better than type 304 and 304L stainless steel.

6.1.1.4 Material Compatibility with Reactor Coolant System Coolant and Engineered Safety Features Fluids

Engineered safety features components materials are manufactured primarily of stainless steel or other corrosion-resistant material. Protective coatings are applied on carbon steel structures and equipment located inside the containment, as discussed in Subsection 6.1.2.

Austenitic stainless steel plate conforms to ASME SA-240. Austenitic stainless steel is confined to those areas or components which are not subject to post-weld heat treatment. Carbon steel forgings conform to ASME SA-350. Austenitic stainless steel forgings conform to ASME SA-182. Nickel-chromium-iron alloy pipe conforms to ASME SB-167. Carbon steel castings conform to ASME SA-352. Austenitic stainless steel castings conform to ASME SA-351.

The use of cobalt-base alloys is minimized. Restrictions on the use of cobalt in the base metal for material in contact with reactor coolant are discussed in Subsection 12.3.1.1.1.

In post-accident situations where the containment is flooded with water containing boric acid, pH adjustment is provided by the release of trisodium phosphate into the water. The trisodium phosphate is held in baskets located in the floodable volume that includes the steam generator compartments and contains the reactor coolant loop. The addition of trisodium phosphate to the solution is sufficient to raise the pH of the fluid to above 7.0. This pH is consistent with the guidance of NRC Branch Technical Position MTEB-6.1 for the protection of austenitic stainless steel from chloride-induced stress corrosion cracking. Section 6.3 describes the design of the trisodium phosphate baskets.

In the post-accident environment, both aluminum and zinc surfaces in the containment are subject to chemical attack resulting in the production of hydrogen and/or chemical precipitants that can affect long-term core cooling. The amount of aluminum allowed in the containment below the maximum flood level of a design basis loss-of-coolant accident (LOCA) (refer to Subsection 6.3.2.2.7.1, item 3) will be limited to less than 40 pounds with a surface area of no more than 13.33 square feet during operating conditions. A large potential source of aluminum in the AP1000 containment are the excore detectors described in Subsection 7.1.2.7.2. To avoid sump water contact with the excore detectors, they are enclosed in stainless steel or titanium housings. The non-flooded surfaces would be wetted by condensing steam, but they would not be subjected to the boric acid or trisodium phosphate solutions since there is no containment spray. For this reason, the amount of aluminum in the excore detectors is not applied to the 40-pound weight limit restriction as they are not subject to the post-design basis accident (DBA) environment as a result of stainless steel or titanium encasement.

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Furthermore, other aluminum, within containment encased in stainless steel/titanium that can ensure interaction with the boric acid or trisodium phosphate solutions does not occur, should not be applied to the 40-pound weight limit. Nonsafety-related passive autocatalytic recombiners are provided to limit hydrogen buildup inside containment.

6.1.1.5 Integrity of Safety-Related Components

The pH adjustment baskets provide for long-term pH control. In the case of inadvertent short-term flooding when the pH adjustment baskets remain above the flood level, the condition of the material in contact with the fluid is evaluated prior to return to operation. Based on previous industry testing and experience, the behavior of austenitic stainless steels in the post-design basis accident environment is acceptable. Cracking is not anticipated, provided that the core cooling pH is maintained at an adequate level.

6.1.1.6 Thermal Insulation

The majority of the engineered safety features insulation used in the AP1000 containment is reflective metallic insulation. Fibrous insulation may be used if it is enclosed in stainless steel cans. The selection, procurement, testing, storage, and installation of nonmetallic thermal insulation provides confidence that the leachable concentrations of chloride, fluoride, and silicate are in conformance with Regulatory Guide 1.36. Conformance with Regulatory Guide 1.36 is summarized in Section 1.9.

6.1.1.7 Component and System Cleaning

See Subsection 1.9.1 for a discussion on the provisions of Regulatory Guide 1.37 for the cleaning of components and systems.

6.1.2 Organic Materials

6.1.2.1 Protective Coatings

6.1.2.1.1 General

The AP1000 is divided into four areas with respect to the use of protective coatings. These four areas are:

- Inside containment
- Exterior surfaces of the containment vessel
- Radiologically controlled areas outside containment
- Remainder of plant

The considerations for protective coatings differ for these four areas and the coatings selection process accounts for these differing considerations. The AP1000 design considers the function of the coatings, their potential failure modes, and their requirements for maintenance. Table 6.1-2 lists different areas and surfaces inside containment and on the containment shell that have coatings, their functions, and to what extent their coatings are related to plant safety.

Coatings used outside containment do not provide functions related to plant safety except for the coating on the outside of the containment shell. The coating on the outside of the containment shell

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above elevation 135′ 3″ shell supports passive containment cooling system heat transfer and is classified as a Service Level III coating.

The coating used on the inside surface of the containment shell supports the transfer of thermal energy from the post-accident atmosphere inside containment to the containment shell. Passive containment cooling system testing and analysis have been performed with a coating. This coating is classified as a Service Level I coating.

Coatings are not used in the vicinity of the containment recirculation screens to minimize the possibility of debris clogging the screens. Subsection 6.3.2.2.7.3 defines the area in the vicinity of the recirculation screens where coatings are not used.

Coatings used inside containment, except for inorganic zinc used on the inside surface of the containment shell and on other components, and epoxy top coat used on the inside surface of the containment shell are classified as Service Level II coatings because their failure does not prevent functioning of the engineered safety features. If the Service Level I epoxy or Service Level II coatings delaminate, the solid debris they may form will not have a negative impact on the performance of safety-related post-accident cooling systems. See Subsection 6.1.2.1.5 for a discussion of the factors including plant design features and low water flows that permit the use of Service Level II coatings inside containment. Protective coatings are maintained to provide corrosion protection for the containment pressure boundary and for other system components inside containment.

The corrosion protection of the containment shell is a safety-related function. Good housekeeping and decontamination functions of the coatings are nonsafety-related functions.

For information on coating design features, quality assurance, material and application requirements, and performance monitoring requirements, see Subsection 6.1.2.1.6.

6.1.2.1.2 Inside Containment

Carbon Steel

Inorganic zinc is the basic coating applied to the containment vessel. Below the operating floor, most of the inorganic zinc coating is top coated with epoxy where enhanced decontamination is desired. Where practical, carbon steel and structural modules within the containment are coated with self-priming high solids epoxy (SPHSE). When coating use is not practical, carbon steel items (such as items with threaded surfaces, bolted connections, surfaces that could wear due to moving parts, and surfaces with temperature or radiation conditions that exceed the coatings' qualification) are left uncoated or use a rust preventative. Where practical, miscellaneous carbon steel items (such as stairs, ceilings, gratings, ladders, railings, conduit, duct, and cable tray) are hot-dip galvanized. Steel surfaces subject to immersion during normal plant operation (such as sumps and gutters) are stainless steel or are coated with SPHSE applied directly to the carbon steel without an inorganic zinc primer. Carbon steel structures and equipment are assembled in modules, and the modules are coated in the fabrication shop under controlled conditions.

Concrete

Concrete surfaces inside containment are coated primarily to prevent concrete from dusting, to protect it from chemical attack, and to enhance decontaminability. In keeping with ALARA goals, the exposed concrete surfaces are made as decontaminable as practical in areas of frequent personnel access and areas subject to liquid spray, splash, spillage, or immersion.

Exposed concrete surfaces inside containment are coated with epoxy. Concrete floors inside containment are coated. Exposed concrete walls inside containment are coated to a minimum height of 7 feet.

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6.1.2.1.3 Exterior of Containment Vessel

The exterior of the containment vessel is coated with the same inorganic zinc as is used inside of the containment vessel. The inorganic zinc coating enhances heat transfer by providing good heat conduction and by enhancing surface wetting of the exterior surface of the containment vessel. The inorganic zinc also provides corrosion protection.

6.1.2.1.4 Radiologically Controlled Areas Outside Containment and Remainder of Plant

The coatings used in the radiologically controlled areas outside containment and in the remainder of the plant are also classified as Service Level II coatings. However, these coatings are selected, specified, and applied in a manner that optimizes performance and standardization within the AP1000 design. Therefore, wherever practical, the same coating systems are used in radiologically controlled areas outside containment as are used inside containment. The ALARA concept is carried through in areas subject to radiation exposure and possible radiological contamination. Consistent with ALARA goals, the surfaces are made as decontaminable as practical in areas of frequent personnel access and areas subject to liquid spray, splash, spillage, or immersion. The remainder of the plant coating systems are commercial grade materials that are selected and applied according to the expected conditions in the specific areas where the coatings are applied.

6.1.2.1.5 Safety Evaluation

This subsection describes the basis for classifying coatings as Service Level I, II, or III. Table 6.1-2 identifies which coatings are classified as Service Level I and Service Level III.

The inorganic zinc coating on the outside of the containment shell above elevation 135' 3" supports passive containment cooling system heat transfer and is classified as a Service Level III coating.

The inorganic zinc coating used on the inside surface of the containment shell supports the transfer of thermal energy from the post-accident atmosphere inside containment to the containment shell. Passive containment cooling system testing and analysis have been performed with an inorganic zinc coating. This coating is classified as Service Level I coating.

The AP1000 has a number of design features that facilitate the use of Service Level II coatings inside containment. These features include a passive safety injection system that provides a long delay time between a LOCA and the time recirculation starts. This time delay provides time for settling of debris. These passive systems also flood the containment to a high level which allows the use of containment recirculation screens that are located well above the floor and are relatively tall. Significant volume is provided for the accumulation of coating debris without affecting screen plugging. These screens are protected by a plate located above the screens that extends out in front and to the side of the screens. Coatings are not used under this plate in the vicinity of the screens. The protective plate, together with low recirculation flow, approach velocity and the screen size preclude postulated coating debris above the plate from reaching the screens. Refer to Subsection 6.3.2.2.7.3 for additional discussion of these screens, the protective plate and the areas where coatings are prohibited from being used.

The recirculation inlets are screened enclosures located near the northwest and southwest corners of the east steam generator compartment (refer to the figures in Subsection 6.3.2.2.7.3). The enclosure bottoms are located above the surrounding floor, which prevent ingress of heavy debris (density ≥ 100 lb_m/ft^3). Additionally, the screens are oriented vertically and are protected by a large plate located above the screens, further enhancing the capability of the screens to function with debris in the water. The screen mesh size and the surface area of the containment recirculation screens in the AP1000, in conjunction with the large floor area for debris to settle on, can accommodate failure of

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coatings inside containment during a design basis accident even though the residue of such a failure is unlikely to be transported to the vicinity of the enclosures.

The AP1000 does not have a safety-related containment spray system. The containment spray system provided in the AP1000 is only used for beyond design basis events. This reduces the chance that coatings will peel off surfaces inside containment because the thermal shock of cold spray water on hot surfaces combined with the rapid depressurization following spray initiation are recognized as contributors to coating failure. Parts of the containment below elevation 110′ are flooded and water is recirculated through the passive core cooling system. However, the volume of water moved in this manner is relatively small and the flow velocity is very low.

The coating systems used inside containment also include epoxy and/or self-priming high solids epoxy coatings. These are applied to concrete substrates, as top coats over the inorganic zinc coating, and directly to steel, as noted in Subsection 6.1.2.1.2. The failure modes of these systems could include delamination or peeling if the epoxy coatings are not properly applied (References 1, 2, 3). The epoxies applied to concrete and carbon steel surfaces are sufficiently heavy (dry film density greater than or equal to 100 lb/ft^3) so that transport of small chips with the low water velocity in the AP1000 containment is limited.

Inside containment, there are components coated with various manufacturers' standard coating systems. These coating systems are generally not required to have Class I or III safety classification as delineated in Table 6.1-2; however, those located below the maximum flood level of a design basis LOCA, or where there is sufficient water flow to transport debris, are required to be sufficiently heavy (dry film density greater than or equal to 100 lb/ft^3) so that transport of small chips with the low water velocity in the AP1000 containment is limited.

If a coating on walls, structures, or components has a dry film density less than 100 lb/ft^3, then testing and/or analysis must be performed to demonstrate that the debris is not transported to an AP1000 screen or into the core through a flooded break. The testing and/or analysis must be approved by the NRC.

In addition, inorganic zinc should be used only on surfaces that may be exposed to temperatures that are above the limits of epoxy coatings during normal operating conditions; inorganic zinc coatings used in such applications are required to be Safety – Service Level I to prevent detachment during a LOCA since such debris is not likely to settle out.

Requirements related to production of hydrogen as a result of zinc corrosion in design basis accident conditions, including the zinc in paints applied inside containment, were eliminated by the final rule, effective October 16, 2003, amending 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."

Nonconforming coatings that cannot be qualified as Service Level I are evaluated on a case basis relative to impact on plant safety. The total inventory of unqualified coatings (coatings which cannot be qualified as Safety-Service Level I) within Service Level I areas combined with the total amount of coating debris fines that can be generated by a LOCA jet is restricted to the limits established in Subsection 6.3.2.2.7.1.

6.1.2.1.6 Quality Assurance Features

A number of quality assurance features provide confidence that the coating systems inside the containment, on the exterior of the containment vessel and in potentially contaminated areas outside containment will perform as intended. These features enhance the ALARA program and enhance corrosion resistance. The features are discussed in the following paragraphs.

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Service Level I and Service Level III Coatings

The quality assurance program for Service Level I and Service Level III coatings conforms to the requirements of ASME NQA-1 as identified in the position statement of Regulatory Guide 1.28 provided in Appendix 1A. Safety related coatings meet the pertinent provisions of 10CFR Part 50 Appendix B to 10CFR Part 50. The service level classification of coatings is consistent with the positions given in Regulatory Guide 1.54 (revision and exceptions as specified in Appendix 1A), “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants.” Service Level I and Service Level III coatings used in the AP1000 are tested for radiation tolerance and for performance under design basis accident conditions. Where decontaminability is desired, the coatings are evaluated for decontaminability. The coating applicator submits and follows acceptable procedures to control surface preparation, application of coatings and inspection of coatings. The painters are qualified and certified, and the inspectors are qualified and certified.

The inorganic zinc coating used on the inside surface (Service Level I coatings) and outside surface (Service Level III coatings) of the containment shell is inspected using a non-destructive dry film thickness test and a MEK rub test. These inspections are performed after the initial application and after recoating. Long term surveillance of the coating is provided by visual inspections performed during refueling outages. Other inspections are not required.

During the design and construction phase, the coatings program associated with selection, procurement and application of safety related coatings is performed to applicable quality standards. The requirements for the coatings program are contained in certified drawings and/or standards and specifications controlling the coating processes of the designer (Westinghouse) (these design documents will be available prior to the procurement and application of the coating material by the constructor of the plant). Regulatory Guide 1.54 and ASTM D5144 (Reference 201) form the basis for the coating program.

During the operations phase, the coatings program is administratively controlled in accordance with the quality assurance program implemented to satisfy 10 CFR Part 50, Appendix B, and 10 CFR Part 52 requirements. The coatings program provides direction for the procurement, application, inspection, and monitoring of safety related coating systems. Prior to initial fuel loading, a consolidated plant coatings program will be in place to address procurement, application, and monitoring (maintenance) of those coating system(s) for the life of the plant.

Coating system monitoring requirements for the containment coating systems are based on ASTM D5163 (Reference 202), "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant," and ASTM D7167 (Reference 203), "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality assurance requirements. Management of unqualified coatings inside containment is performed in accordance with the guidance of ASTM D7491 (Reference 204), "Standard Guide for Management of Non-Conforming Coatings in Coating Service Level I Areas of Nuclear Power Plants."

Refer to Table 6.1-2 for identification of Service Level I and Service Level III coating applications in the AP1000.

Service Level II Coatings

The use of Service Level II coatings inside containment is based on the use of selected types of coatings and the properties of the coatings. To preclude the use of inappropriate coatings, the procurement of Service Level II coatings used inside containment is considered a safety-related activity whereas the Service Level II coatings used outside the containment are nonsafety-related.

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Such Service Level II coatings used inside containment are procured to the same standards as Service Level I coatings with regard to radiation tolerance and performance under design basis accident conditions as discussed below.

Service Level II coatings used on manufactured components are not subject to the procurement under 10 CFR 50, Appendix B, because their high density limits the transport with the low water velocity in the AP1000 containment.

Service Level II coatings used inside containment are tested for radiation tolerance and for performance under design basis accident conditions to the standards referenced in Appendix 1A. Service Level II coating application configurations such as variations in substrate, surface preparation, cleanliness, profile, color, and dry film thickness are not required to be qualified by DBA testing. Delamination of Service Level II coatings during DBA testing is acceptable if the Service Level II coating meets the criteria identified in Subsection 6.1.2.1.1. Where decontaminability is desired, the coatings are selected to be as decontaminable as practical.

Service Level II coatings used inside containment on internal structures, including walls, floor slabs, structural steel, and the polar crane are subject to procurement and DBA testing under 10 CFR 50 Appendix B, except in the following areas: a) surfaces located inside the chemical and volume control system room # 11209, and b) the area outside of the annular wall of the IRWST. Service Level II coatings used in the chemical and volume control system room are not subject to procurement under 10 CFR 50, Appendix B and DBA testing, because the room is connected to the containment in a limited way through a drain line. In addition, the drain line is routed to the waste liquid processing system sump which is located well below and separate from the recirculation screens. Service Level II coatings used on the outside of the annular wall of the IRWST are not subject to procurement under 10 CFR 50, Appendix B and DBA testing, because their high density and location limit the transportation in the accident conditions of the low water velocity in the AP1000 containment.

The Service Level II coatings used inside containment are as shown in Table 6.1-2. Coating system application, inspection and monitoring requirements for the Service Level II coatings used inside containment will be performed in accordance with a program based on ASTM D5144 (Reference 201), “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants,” and the guidance of ASTM D5163 (Reference 202), “Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant.” Application anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality requirements.This program is not subject to 10 CFR 50, Appendix B, quality assurance requirements.

Due to the use of modularized construction, a significant portion of the containment coatings are shop applied to the containment vessel and to piping, structural and equipment modules. This application of coatings under controlled shop conditions provides additional confidence that the coatings will perform as designed and as expected.

The coatings used in radiologically controlled areas outside containment, consistent with ALARA goals, are selected to be as decontaminable as practical; they are not specified to be radiation tolerance tested or design basis accident tested, and they are not procured to Appendix B to 10 CFR 50. Where practical, the same coating materials are used in radiologically controlled areas outside containment as are used inside containment. This provides a high level of quality and optimizes maintenance painting over the life of the plant.

6.1.2.2 Other Organic Materials

A listing of other organic materials in the containment is developed based on the specific type of equipment and the supplier selected to provide it. Materials are evaluated for potential interaction

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with engineered safety features to provide confidence that the performance of the engineered safety features is not unacceptably affected.

6.1.3 Combined License Information Items

6.1.3.1 Procedure Review

The review of vendor fabrication and welding procedures or other quality assurance methods to judge conformance of austenitic stainless steels with Regulatory Guides 1.31 and 1.44 is addressed in Subsection 6.1.1.2.

6.1.3.2 Coating Program

The programs to control procurement, application, inspection, and monitoring of Service Level I, Service Level II, and Service Level III coatings are addressed in Subsection 6.1.2.1.6.

6.1.4 References

1. NUREG-0797, “Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2.”

2. Bolt, R. O. and J. G. Carroll, “Radiation Effects on Organic Materials,” Academic Press, New York, 1963, Chapter 12.

3. Parkinson, W. W. and O. Sisman, “The Use of Plastics and Elastomers in Nuclear Radiation,” Nuclear Engineering and Design 17 (1971), pp 247-280, North-Holland Publishing Co., Amsterdam.

201. ASTM D5144-08, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants.”

202. ASTM D5163-05a, “Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant.”

203. ASTM D7167-05, “Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant.”

204. ASTM D7491-08, “Standard Guide for Management of Non-Conforming Coatings in Coating Service Level I Areas of Nuclear Power Plants.”

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Table 6.1-1 (Sheet 1 of 2)
Engineered Safety Features Pressure-Retaining Materials

Component	Materials
Passive Core Cooling System (safety-related portion)	
Core makeup tank	Refer to Subsection 5.2.3
Passive residual heat removal heat exchanger	Refer to Subsection 5.3.4, Table 5.2-1
In-containment refueling water storage tank	ASTM A240 S32101 or TP304L (or dual grade TP-304/304L)
Accumulators	Refer to Subsections 6.1.1, 6.3.2.2, and 6.3.2.4
Spargers	
Piping	SA-358 Grade 304 or 316 or SA-312 Grade TP304 or TP316
Fittings	SA-182 Grade F304 or SA-403 Grade WP304 or WP316
Tube and piping, valves, and fittings	Refer to Subsections 6.1.1 and 6.3.2.4
Passive containment cooling system (safety-related portion)	
Passive containment cooling system water storage tank	ASTM A240 TP304
PCCWST Discharge Path	
Valves	SA-182 Grade F304 or F304L or F316 or F316L or SA-351 Grade CF3 or CF3A or CF3M or CF8 or CF8M
Tube and piping	SA-312 Grade TP304L or SA-213 Grade TP316
Fittings	SA-182 Grade F304L or F316 or F316L or dual-marked F304/F304L or F316/F316L or SA-403 Grade WP304L or SA-479 Type 316
PCS Recirculation Subsystem	
Valves	SA-217 Grade WC6 or SA-182 Grade F304 or F304L or F316 or F316L or F11 Class 2 or SA-351 Grade CF8M
Piping	SA-335 Grade P11 or SA-312 Grade TP304L
Fittings	SA-234 Grade WP11 Class 1 or SA-182 Grade F304L or F316L or dual-marked F304/F304L or F316/F316L or F11 Class 2 or SA-403 Grade WP304L or SA-479 Type 316
Containment vessel and penetrations	Refer to Subsection 3.8.2.1
Valves in contact with borated water	Refer to Subsection 5.2.3, Table 5.2-1

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Table 6.1-1 (Sheet 2 of 2)
Engineered Safety Features Pressure-Retaining Materials

Component	Materials
Main control room emergency habitability system	
Valves	SA-182 Grade F11 or F304 or F316 or F304L or F316L, or SA-217 Grade WC6
Pipe	SA-335 Grade P11
Air storage tanks	SA-372

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Table 6.1-2 (Sheet 1 of 2)
AP1000 Coated Surfaces, Containment Shell and Surfaces Inside Containment

Surface	Boundary	Surface Material	Coating	Coating Functions/Safety Classifications		Coating Classification (1)
Containment Shell, Outside Surface	Shell surfaces above elevation 135′ 3″	Carbon Steel	Inorganic Zinc Coating	1 Promote wettability 2 Heat conduction 3 Nondetachable 4 Inhibit corrosion	1 Safety 2 Safety 3 Safety 4 Safety	Safety – Service Level III
Containment Shell, Inside Surface	Shell surfaces above the operating floor	Carbon Steel	Inorganic Zinc Coating	1 Promote wettability 2 Heat conduction 3 Nondetachable 4 Inhibit corrosion	1 Safety (2) 2 Safety 3 Safety 4 Safety	Safety – Service Level I
	Shell surfaces below the operating floor	Carbon Steel	Inorganic Zinc Coating with Epoxy Top Coat	1 Nondetachable (inorganic zinc only) 2 Inhibit corrosion 3 Enhance radioactive decontamination (epoxy only) 4 Ensure settling (epoxy only)	1 Safety 2 Safety 3 Nonsafety 4 Safety	Safety – Service Level I
Components Inside Containment	(6)	Material of Component (6)	NA (6)	1 Ensure settling 2 Inhibit corrosion	1 Safety (7) 2 Non-safety	Non-safety (7) Service Level II
Inside Containment	Areas surrounding the containment recirculation screens (3)	NA	NA	NA	NA	NA
	Concrete walls, ceilings and floors (4)	Concrete	Epoxy	1 Ensure settling 2 Prevent dusting 3 Protect from chemical attack 4 Enhance radioactive decontamination 5 Heat conduction	1 Safety (5) 2 Nonsafety 3 Nonsafety 4 Nonsafety 5 Safety (5)	Nonsafety (5) Service Level II

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Table 6.1-2 (Sheet 2 of 2)
AP1000 Coated Surfaces, Containment Shell and Surfaces Inside Containment

Surface	Boundary	Surface Material	Coating	Coating Functions/Safety Classifications		Coating Classification (1)
	Steel walls, ceilings, floors, columns, beams, braces, plates (4)	Carbon Steel	Self-Priming High Solid Epoxy	1 Ensure settling 2 Inhibit corrosion 3 Enhance radioactive decontamination 4 Heat conduction	1 Safety (5) 2 Nonsafety 3 Nonsafety 4 Safety (5)	Nonsafety (5) Service Level II

Notes:

1. The applicability of 10 CFR 50, Appendix B, and other codes and standards to coatings and their application are discussed in Subsection 6.1.2.1.6.
2. An inorganic zinc coating on the inside of the containment shell is not required to promote wettability, however it has been included in PCS testing and analysis and as a result is considered safety-related.
3. Areas around PXS recirculation screens do not require coatings as defined in Subsection 6.3.2.2.7.3.
4. 10 CFR 50, Appendix B, does not apply to DBA testing and procurement of coatings in the CVS room inside containment as discussed in Subsection 6.1.2.1.6.
5. 10 CFR 50, Appendix B, applies to DBA testing and procurement of these Service Level II coatings as discussed in Subsection 6.1.2.1.6.
6. The explicit coating material is not required to be specified. However, the coating material must comply with the restrictions set forth in Subsection 6.1.2.1.5 and Table 6.1-2 for components located below the maximum flood level for a design basis LOCA or where there is sufficient water flow to transport debris. If a coating on walls, structures, or components has a dry film density less than 100 lb/ft^3, then testing and/or analysis must be performed to demonstrate that the debris is not transported to an AP1000 screen or into the core through a flooded break. The testing and/or analysis must be approved by the NRC. Inorganic zinc should be used only on surfaces that may be exposed to temperatures that are above the limits of epoxy coatings during normal operating conditions; inorganic zinc coatings used in such applications are required to be Safety – Service Level I to prevent detachment during a LOCA since such debris is not likely to settle out.
7. 10 CFR 50, Appendix B does not apply to DBA testing and procurement of coatings used on manufactured components as discussed in Subsection 6.1.2.1.6.

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6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

6.2.1.1.1 Design Basis

The containment system is designed such that for all break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary side pipe, the containment peak pressure is below the design pressure. A summary of the results is presented in Table 6.2.1.1-1.

This capability is maintained by the containment system assuming the worst single failure affecting the operation of the passive containment cooling system (PCS). For primary system breaks, loss of offsite power (LOOP) is assumed. For secondary system breaks, offsite power is assumed to be available when it maximizes the mass and energy released from the break. Additional discussion of the assumptions made for secondary side pipe breaks may be found in Subsection 6.2.1.4.

The single failure postulated for the containment pressure/temperature calculations is the failure of one of the valves controlling the cooling water flow for the PCS. Failure of one of these valves would lead to cooling water flow being delivered to the containment vessel through two of three delivery headers. This results in reduced cooling flow for PCS operation. However, as stated in Table 6.2.2-3, only one flowpath is necessary to meet the minimum containment cooling requirements, and only one flowpath is assumed in the containment integrity analysis.

The containment integrity analyses for the AP1000 employ a multivolume lumped parameter model to study the long-term containment response to postulated Loss of Coolant Accidents (LOCA) and Main Steam Line Break (MSLB) accidents.

The analyses presented in this section are based on assumptions that are conservative with respect to the containment and its heat removal systems, such as minimum heat removal, and maximum initial containment pressure.

The containment design for the Safe Shutdown Earthquake (SSE) is discussed in Subsection 3.8.2.

The minimum containment backpressure used in the Passive Core Cooling System (PXS) analysis is discussed in Subsection 6.2.1.5.

6.2.1.1.2 Design Features

The operation of the PCS is discussed in Subsection 6.2.2. The arrangement of the containment and internal structures is described in Section 1.2.

The reactor coolant loop is surrounded by structural walls of the containment internal structures. These structural walls are a minimum of 2-feet - 6-inches thick and enclose the reactor vessel, steam generators, reactor coolant pumps, and the pressurizer.

The containment vessel is designed and constructed in accordance with the ASME Code, Section III, Subsection NE, Metal Containment, as described in Subsection 3.8.2.

Structural steel non-pressure retaining parts such as ladders, walkways, and handrails are designed to the requirements for steel structures defined in Subsection 3.8.4.

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The design features provide adequate containment sump levels following a design basis event as described in Section 3.4.

Containment and subcompartment atmospheres are maintained during normal operation within prescribed pressure, temperature, and humidity limits by means of the containment air recirculation system (VCS), and the central chilled water system (VWS). The recirculation system cooling coils are provided with chilled water for temperature control. The filtration supply and exhaust subsystem can be utilized periodically to purge the containment air for pressure control. Periodic inspection and maintenance verify functional capability.

6.2.1.1.3 Design Evaluation

The Westinghouse-GOTHIC (WGOTHIC) computer code (Reference 20) is a computer program for modeling multiphase flow in a containment transient analysis. It solves the conservation equations in integral form for mass, energy, and momentum for multicomponent flow. The momentum conservation equations are written separately for each phase in the flow field (drops, liquid pools, and atmosphere vapor). The following terms are included in the momentum equation: storage, convection, surface stress, body force, boundary source, phase interface source, and equipment source.

To model the passive cooling features of the AP1000, several assumptions are made in creating the plant decks. The external cooling water does not completely wet the containment shell, therefore, both wet and dry sections of the shell are modeled in the WGOTHIC analyses. The analyses use conservative coverage fractions to determine evaporative cooling as explained in Chapter 7 of Reference 20. The analyses conservatively assume that the external cooling water is not initiated until 400 seconds (Chapter 7, Reference 20) into the transient, allowing time to initiate the signal and to fill the headers and weirs and to develop the flow down the containment side walls. The effects of water flowing down the shell from gravitational forces are explicitly considered in the analysis.

The containment initial conditions of pressure, temperature, and humidity are provided in Table 6.2.1.1-2.

For the LOCA events, two double-ended guillotine reactor coolant system pipe breaks are analyzed. The breaks are postulated to occur in either a hot or a cold leg of the reactor coolant system. The hot leg break is limiting during the blowdown phase of the accident, and is only analyzed for this time period. The cold leg break is the most limiting scenario after the blowdown, and is analyzed for the long term containment response. The cold leg break analysis includes the long term contribution to containment pressure from the sources of stored energy, such as the steam generators. The LOCA mass and energy releases described in Subsection 6.2.1.3 are used for these calculations.

For the MSLB event, a double-ended pipe rupture is analyzed at several power levels with the WGOTHIC code. The MSLB mass and energy releases described in Subsection 6.2.1.4 are used for these calculations.

Table 6.2.1.1-10 summarizes the required volume of steel inside containment to validate the heat sink input and methodology for the WGOTHIC containment analyses. It includes requirements for analyses that minimize heat sinks (the peak containment pressure analyses) and ones that maximize heat sinks (the minimum containment pressure analyses).

The methodology for calculating the internal containment heat sink data used in the WGOTHIC analyses is presented in Reference 20, Section 13. Heat sinks include both metallic and concrete structures. Table 6.2.1.1-11 lists the heat sinks that are credited for heat transfer in the containment evaluation model for peak containment pressure analyses. The physical properties of the heat sink materials are presented in Table 6.2.1.1-8. These properties are inputs to the containment peak

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pressure evaluation, consistent with the methodology specified in Reference 20 Section 13, including:

- The thermal conductivity and volumetric heat capacity of the steel materials are reduced by 10% per the ASME standard to account for variability of alloy content.
- The thermal conductivity of inorganic zinc is reduced by a factor of two relative to the minimum test requirement. The conditions for use of this reduction factor identified in Section 4 of Reference 37 are met.
- Emissivity is reduced by 10% and is only credited on the outer surface of the containment vessel, baffle, and shield building. Emissivity inside containment is ignored.

Carbon steel and concrete heat sinks inside containment are modeled with a coating of epoxy on the exposed surface. The containment vessel is modeled with an inorganic zinc coating inside and outside containment. The containment vessel below the operating deck has both inorganic zinc coating and an epoxy top coating.

The free volume inside the containment is calculated using nominal, cold dimensions. This conservatively neglects the thermal expansion of the containment vessel at elevated temperatures.

The containment pressure and temperature responses to a double-ended cold leg guillotine break are presented in Figures 6.2.1.1-5 and 6.2.1.1-6 for the first 5000 seconds, which shows the peak portion of the containment transients and Figures 6.2.1.1-7 and 6.2.1.1-8 for the 3-day transients. The containment pressure and temperature response to a double-ended hot leg guillotine break are presented in Figures 6.2.1.1-9 and 6.2.1.1-10. A separate analysis for the double-ended cold leg guillotine LOCA event, without considering heat conduction from the dry to wet section, results in somewhat higher containment pressure in the long term, but still below 50 percent of design pressure at 24 hours. This separate analysis confirms the assumption in Subsection 15.6.3.3.3 of reducing the containment leakage to half its design value at 24 hours as shown in Table 6.2.1.1-3.

The peak containment pressure and temperature results of the LOCA and MSLB postulated accidents are provided in Table 6.2.1.1-1. The peak value is the maximum of any location within the multi-node WGOTHIC model. A comparison of the containment integrity acceptance criteria to General Design Criteria is provided in Table 6.2.1.1-3.

The containment pressure response for the peak pressure steam line break case is provided in Figure 6.2.1.1-1. The containment temperature response for the peak temperature steam line break case is provided in Figure 6.2.1.1-2.

The instrumentation provided outside containment to monitor and record the containment pressure and the instrumentation provided inside containment to monitor and record temperature are found in Section 7.5.

6.2.1.1.4 External Pressure Analysis

Certain design basis events and credible inadvertent systems actuation have the potential to result in containment external pressure loads. Evaluations of these events show that a loss of all ac power sources during cold ambient conditions has the potential for creating the worst-case external pressure load on the containment vessel. This event leads to a reduction in the internal containment heat loads from the reactor coolant system and other active components, thus resulting in a temperature reduction within the containment and an accompanying pressure reduction. Evaluations are performed to determine the maximum external pressure to which the containment may be

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subjected, and to develop the operating differential air temperature limit presented in LCO 3.6.9 of the Technical Specifications.

The bounding scenario results from a postulated loss of ac power sources (station blackout). This scenario, along with bounding assumptions and initial conditions, are used to determine the maximum expected external pressure transient. The containment pressure response from the bounding transient is used for sizing the containment vacuum relief system and verifies that the vacuum relief system is capable of mitigating the most bounding external pressure scenario.

The evaluation assumes a 25°F ambient temperature with no outside wind blowing to maximize the containment internal temperature and corresponding containment vessel shell temperatures. The initial internal containment temperature is in equilibrium at the maximum allowable value of 120°F. A 25°F outside temperature coupled with a 120°F internal temperature exceeds the maximum allowable internal/external temperature differential depicted in the AP1000 Technical Specifications (LCO 3.6.9). However, this is conservative and bounding as described below. Once the equilibrium temperature and humidity values are determined, the bounding analysis is performed with an initial containment internal relative humidity set to a value that is higher than the equilibrium conditions. A conservatively large value for humidity coupled with the assumed maximum containment internal temperature creates the largest potential for external pressure as this maximizes the partial pressure of steam vapor, vapor concentration, and corresponding condensation rate. These parameters represent the dominant effect for the determination of the bounding external pressure scenario. A negative 0.2 psig initial containment pressure is used for this evaluation. At transient initiation, the external wind is assumed to instantaneously accelerate to 48 mph (24.9 ft/s in annulus riser region) and the external temperature is assumed to begin decreasing at a rate of 30°F/hr. It is also conservatively assumed that no air leakage occurs into the containment during the transient. The key assumptions for containment initial conditions and containment transient conditions are listed in Table 6.2.1.1-9.

The external pressure evaluations are performed using WGOTHIC with conservatively low estimates of the containment heat loads and conservatively high heat removal through the containment vessel consistent with the limiting assumptions stated above. Results of these evaluations are used to develop the maximum depressurization rate of containment to determine the adequacy of the active safety grade containment vacuum relief system. Figure 6.2.1.1-11 shows that the performance of the vacuum relief system is sufficient to mitigate the maximum expected external pressure scenario.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Basis

Subcompartments within containment are designed to withstand the transient differential pressures of a postulated pipe break. These subcompartments are vented so that differential pressures remain within structural limits. The subcompartment walls are challenged by the differential pressures resulting from a break in a high energy line. Therefore, a high energy line is postulated, with a break size chosen consistent with the position presented in Section 3.6, for analyzing the maximum differential pressures across subcompartment walls.

Section 3.6 describes the application of the mechanistic pipe break criteria, commonly referred to as leak-before-break (LBB), to the evaluation of pipe ruptures. This eliminates the need to consider the dynamic effects of postulated pipe breaks for pipes which qualify for LBB. However, the LBB leakage cracks are considered for subcompartment pressurization. The LBB evaluation of leakage from a crack (10 times larger than the minimum leakage the detection system is capable of sensing) determined that the flow magnitude of an LBB leakage crack would not challenge the structural integrity of any of the walls evaluated for subcompartment pressurization. The analyses of containment pressure and temperature, emergency core cooling, and environmental qualification of

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equipment are based on double-ended guillotine (DEG) reactor coolant system breaks and through-wall cracks.

6.2.1.2.1.1 Summary of Subcompartment Pipe Break Analyses

Each subcompartment is analyzed for effects of differential pressures resulting from the break of the most limiting line in the subcompartment which has not been evaluated for LBB.

The subcompartment analysis demonstrates that the wall differential pressures resulting from the most limiting high energy line break within the subcompartments are within the design capability.

Table 6.2.1.2-1 provides the design and maximum calculated subcompartment differential pressures for the breaks considered for short-term subcompartment pressurization.

6.2.1.2.2 Design Features

The plant general arrangement drawings shown in Section 1.2 include descriptions of the containment sub-compartments and surrounding areas. The general arrangement drawings are used in assembling the subcompartment analysis model.

Vent paths considered in the analyses are shown in the general arrangement drawings and consist of floor gratings and openings through walls. In the AP1000 subcompartment analyses, no credit is taken for vent paths that become available only after the occurrence of the postulated break (such as blowout panels, doors, hinged panels and insulation collapsing).

6.2.1.2.3 Design Evaluation

The Transient Mass Distribution (TMD) computer code (Reference 2) is used in the subcompartment analysis to calculate the differential pressures across subcompartment walls. The TMD code has been reviewed by the NRC and approved for use in subcompartment differential pressure analyses.

Specific information relative to details on the analysis, such as noding diagrams, volumes, vent areas, and initial conditions, are provided in Reference 26. The nodalization has been chosen so that substantial pressure gradients do not exist within a node, and 100 percent entrainment is assumed. Vent flow is based on homogeneous mixture in thermal equilibrium with 100 percent water entrainment. In order to model the various compartments of containment, a control volume technique is used to spatially represent these regions. The conservations of mass, momentum, energy, and equation of state are solved within each control volume as a function of time and space. The model includes moisture entrainment effects in the momentum and energy equations. The peak pressures in each element, and the peak differential pressures across structures within the containment, are determined from these calculations. The pressure differentials occur because of the propagation of the pressure disturbance around the containment subsequent to the pipe break.

The methodology used to generate the short term mass and energy releases is described in Subsection 6.2.1.3.1.

The initial atmospheric conditions used in the TMD subcompartment analysis are selected so that the calculated differential pressures are maximized. These conditions are chosen according to criteria identified in Subsection 6.2.1.2 of NUREG-0800. Preliminary runs were included on a case by case basis to justify the initial conditions selected to maximize differential pressures.

The containment and subcompartment atmospheres during normal operating conditions are maintained within prescribed pressure, temperature, and humidity limits by means of the containment air recirculation system (VCS), and the central chilled water system (VWS). The recirculation system

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cooling coils are provided with chilled water to provide sufficient temperature control. The filtration supply and exhaust subsystem can be utilized to purge the containment air for pressure control. Periodic inspection and maintenance are performed to verify functional capability.

6.2.1.2.3.1 Flow Equation

The flow equations used by the TMD code to calculate the flow between nodes are described in Reference 2. These flow equations are based on the unaugmented critical flow model, which demonstrate conservatively low critical flow velocity predictions compared to experimental test data. Due to the TMD calculation methods presented in Subsection 1.3.1 of Reference 2, 100 percent entrainment results in the highest calculated differential pressures and therefore this degree of entrainment is conservatively assumed in the subcompartment analysis.

6.2.1.2.3.2 Pipe Breaks

Subcompartment pressurization analyses were performed for break locations found within various subcompartments inside the containment. Table 6.2.1.2-1 provides the locations that were analyzed. Break locations were excluded if:

- A pair of train specific breaks occurs in the same location.
- They occur in a subcompartment that cannot pressurize.
- They do not contain high energy fluid capable of pressurizing subcompartment.
- They are bounded by another break in the same subcompartment.

The subcompartment analysis for the steam generator compartments (Rooms 11201, 11301, 11401 and 11402) is performed assuming double-ended ruptures in the chemical and volume control system (CVS) 3-inch purification return line, 4-inch steam generator blowdown line, or 4-inch pressurizer spray line. The breaks can be assumed to occur between the 83-foot elevation and the 135-foot, 3-inch elevation of the steam generator compartment. Because the TMD code assumes homogeneous mixtures within a node, the specific location of the break within the node is not critical to the differential pressure calculation. No flow restrictions exist that limit the flow out of the break.

The analysis for the steam generator vertical access area (Room 11204) is performed assuming double-ended ruptures in the 3-inch purification line between the CVS-PL-V003 outlet to the regenerative heat exchanger (RHX) tube inlet, or the 3-inch purification line between the RHX shell outlet to the CVS-PL-V080 check valve inlet. These breaks are assumed to occur between the 83-foot elevation and the 103-foot, 5.5-inch elevation of the steam generator vertical access area compartment.

The analysis for the main CVS room (Room 11209) is performed assuming double-ended ruptures in the 3-inch purification line from the RHX tube outlet to the letdown heat exchanger (LHX) inlet (center and south locations), the 3-inch purification line between the CVS-PL-V003 outlet and the RHX tube inlet, or the 3-inch purification line from the RHX shell outlet to the CVS-PL-V080 check valve inlet. These breaks can be assumed to occur between the 80-foot, 6-inch elevation and the 105-foot, 2-inch elevation of the CVS room compartment.

The analysis for the pipe tunnel in the CVS compartment (Room 11209) is performed assuming double-ended ruptures in the 3-inch purification line from the CVS-PL-V003 outlet to RHX tube inlet, the 3-inch purification return line between the RHX shell outlet and the CVS-PL-V080 check valve inlet, or the 4-inch steam generator blowdown line. The breaks can be assumed to occur between the 100-foot elevation and the 105-foot, 5-inch elevation of the CVS room pipe tunnel.

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The analysis for the steam generator access room (Room 11304) is performed for immediate break ruptures in the CVS return line piping assembly from the regenerative heat exchanger, bypass of CVS-PL-V081, 3-inch by 1-inch weldolet, downstream of CVS-PL-V081 and in the CVS return line piping assembly from the regenerative heat exchanger, valve transition, CVS-PL-V081 and CVS-PL-V082. The break at the downstream side of CVS-PL-V082 is a double-ended rupture break; the others are single-ended ruptures. The breaks can be assumed to occur between the 107-foot, 2-inch elevation and the 116-foot, 6-inch elevation of the steam generator access room.

The analysis for the pressurizer spray valve room (Room 11403) is performed for immediate break ruptures in the pressurizer spray piping valve module, 4-inch by 2-inch schedule 160 reducing tee to the auxiliary spray line. These breaks are double-ended ruptures and are assumed to occur between the 118-foot, 6-inch elevation and the 133-foot, 3-inch elevation of the pressurizer spray valve room.

6.2.1.2.3.3 Node Selection

The nodalization for the sub-compartments is analyzed in sufficient detail such that nodal boundaries are at the location of flow obstructions or geometrical changes within the subcompartment. These discontinuities create pressure differentials between adjoining nodes. There are no significant discontinuities within each node, and hence the pressure gradient is negligible within any node.

6.2.1.2.3.4 Vent Flowpath Flow Conditions

The flow characteristics for each of the subcompartments are such that, at no time during the transient does critical flow exist through vent paths.

6.2.1.3 Mass and Energy Release Analyses for Postulated Pipe Ruptures

Mass and Energy releases are documented in this section for two different types of transients.

The first section describes the methodology used to calculate the releases for the subcompartment differential pressure analysis using the TMD code (referred to as the short term analysis). These releases are used for the subcompartment response in Subsection 6.2.1.2.

The second section describes the methodology used to determine the releases for the containment pressure and temperature calculations using the WGOTHIC code (Reference 20) (referred to as the long term analysis). These releases are used for the containment integrity analysis in Subsection 6.2.1.1.

The short term analysis considers only the initial stages of the blowdown transient, and takes into consideration the application of LBB methodology. LBB is discussed in Subsection 3.6.3. Since LBB is applicable to reactor coolant system piping that is 6 inches in diameter and greater, the mass and energy release analysis for sub-compartments postulates the complete DEG severance of 3-inch and 4-inch pipe. The mass and energy release postulated for a ruptured steam line is for a one square foot break.

Conversely, the limiting break size for containment integrity analysis considers as its LOCA design basis the complete DEG severance of the largest reactor coolant system pipe.

The containment system receives mass and energy releases following a postulated rupture of the reactor coolant system. The release rates are calculated for pipe failure at two locations: the hot leg and the cold leg. These break locations are analyzed for both the short-term and the long-term transients. Because the initial operating pressure of the reactor coolant system is approximately 2250 psi, the mass and energy are released extremely rapidly when the break occurs. As the water exits from the broken pipe, a portion of it flashes to steam because of the differences in pressure and

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temperature between the reactor coolant system and containment. The reactor coolant system depressurizes rapidly since break flow exits from both sides of the pipe in a DEG severance.

6.2.1.3.1 Short Term Mass and Energy Release Data

The AP1000 short term LOCA mass and energy releases are predicted for the first ten seconds of the blowdown from a postulated DEG break of the largest non-LBB high energy line in each compartment. The density of the fluid released from a postulated pipe rupture has a direct effect on the magnitude of the differential pressures that results across subcompartment walls.

Short-term transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The modified Zaloudek correlation (Reference 3), which models this condition, is used to calculate the critical mass flux for the LOCA breaks analyzed for subcompartment pressurization. Using this correlation maximizes the mass released, which is conservative for the subcompartment analysis. This is the same critical flow correlation utilized from the short-term LOCA SATAN-VI computer program.

The Moody correlation (Reference 34), a model developed for predicting the maximum flow rate of a single component two-phase mixture, is used to calculate the critical mass flow for the steam generator blowdown line breaks analyzed for subcompartment pressurization. No reduction in break flow is credited due to the depressurization of the system. The limiting plant condition in terms of initial secondary-system pressure is obtained when the plant is at hot zero power.

The initial conditions and inputs used for the LOCA mass and energy releases are given in Table 6.2.1.3-1. The limiting plant condition in terms of initial secondary-system pressure used for the steam generator blowdown line break mass and energy releases are obtained when the plant is at hot zero power. The short term LOCA mass and energy releases are affected by the initial density of the fluid. A lower density yields a more conservative maximum compartment differential pressure. The break specific mass and energy releases used in the TMD code for calculating subcompartment pressurization are identified in Table 6.2.1.2-1.

6.2.1.3.2 LOCA Long Term Mass and Energy Release Data

A long term LOCA analysis calculational model is typically divided into four phases: blowdown, which includes the period from the accident initiation (when the reactor is in a steady-state full power operation condition) to the time that the broken loop pressure equalizes to the containment pressure; refill, which is the time from the end of the blowdown to the time when the passive core cooling system (PXS) refills the vessel lower plenum; reflood, which begins when the water starts to flood the core and continues until the core is completely quenched; and post-reflood, which is the period after the core has been quenched and energy is released to the reactor coolant system primary system by the reactor coolant system metal, core decay heat, and the steam generators.

The long-term analysis considers the blowdown, reflood, and post-reflood phases of the transient. The refill period is conservatively neglected so that the releases to the containment are conservatively maximized.

The AP1000 long-term LOCA mass and energy releases are predicted for the blowdown phase for postulated DECLG and DEHLG breaks. The blowdown phase mass and energy releases are calculated using the NRC approved SATAN-VI computer code (Reference 4). The post blowdown phase mass and energy releases for the postulated DECLG break are calculated considering the energy released from the available energy sources described below. The energy release rates are conservatively modeled so that the energy is released quickly. The higher release rates result in a conservative containment pressure calculation. The releases are provided in Tables 6.2.1.3-9 and 6.2.1.3-10.

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6.2.1.3.2.1 Mass and Energy Sources

The following are accounted for in the long-term LOCA mass and energy calculation:

- Decay heat
- Core stored energy
- Reactor coolant system fluid and metal energy
- Steam Generator fluid and metal energy
- Accumulators, core make-up tanks (CMTs), and the in-containment refueling water storage tank (IRWST) fluid energy
- Zirconium-water reaction

The methods and assumptions used to release the various energy sources during the blowdown phase are given in Reference 4.

The following parameters are used to conservatively analyze the energy release for maximum containment pressure (calorimetric uncertainty calculation will be provided per Subsection 15.0.15.1):

- Maximum expected operating temperature
- Allowance in temperature for instrument error and dead band
- Margin in volume (+1.4 percent)
- Allowance in volume for thermal expansion (+1.6 percent)
- 100 percent full power operation
- Allowance for calorimetric error (+1.0 percent of full power)
- Conservatively modified coefficients of heat transfer
- Allowance in core stored energy for effect of fuel densification
- Margin in core stored energy (+15.0 percent)
- Allowance in pressure for instrument error and dead band
- Margin in steam generator mass inventory (+10.0 percent)
- One percent of the Zirconium surrounding the fuel is assumed to react

6.2.1.3.2.2 Description of Blowdown Model

A description of the SATAN-VI model that is used to determine the mass and energy released from the reactor coolant system during the blowdown phase of a postulated LOCA is provided in Reference 4. Significant correlations are discussed in this reference. The only exception to the Reference 4 methodology is that the overall heat transfer coefficient between the primary side of the

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reactor coolant system metal and the fluid will be adjusted when there is no liquid in the system to prevent an unrealistically high temperature for the steam release.

6.2.1.3.2.3 Description of Post-Blowdown Model

The remaining reactor coolant system and SG mass and energy inventories at the end of blowdown are used to define the initial conditions for the beginning of the reflood portion of the transient. The broken and unbroken loop SG inventories are kept separate to account for potential differences in the cooldown rate between the loops. In addition, the mass added to the reactor coolant system from the IRWST is returned to containment as break flow so that no net change in system mass occurs.

Energy addition due to decay heat is computed using the 1979 ANS standard (plus 2 sigma) decay heat table from Reference 4. The energy release rates from the reactor coolant system metal and steam generators are modelled using exponential decay rates. This modelling is consistent with analyses for current generation design analyses that are performed with the models described in Reference 4.

The accumulator, CMT, and IRWST mass flow rates are computed from the end of blowdown to the time the tanks empty. The rate of reactor coolant system mass accumulation is assumed to decrease exponentially during the reflood phase. More CMT and accumulator flow is spilled from the break as the system refills. The break flow rate is determined by subtracting the reactor coolant system mass addition rate from the sum of the accumulator, CMT and IRWST flow rates.

Mass which is added to, and which remains in, the vessel is assumed to be raised to saturation. Therefore, the actual amount of energy available for release to the containment for a given time period is determined from the difference between the energy required to raise the temperature of the incoming flow to saturation and the sum of the decay heat, core stored energy, reactor coolant system metal energy and SG mass and metal energy release rates. The energy release rate for the available break flow is determined from a comparison of the total energy available release rate and the energy release rate assuming that the break flow is 100-percent saturated steam following IRWST draindown. Saturated steam releases maximize the calculated containment pressurization.

6.2.1.3.2.4 Single Failure Analysis

The assumptions for the containment mass and energy release analysis are intended to maximize the calculated release. A single failure could reduce the flow rate of water to the RCS, but would not disable the passive core cooling function. The effects of a single failure are taken into account in the containment analysis of Subsection 6.2.1.1.

6.2.1.3.2.5 Metal-Water Reaction

Consistent with 10 CFR 50, Appendix K criteria, the energy release associated with the zirconium-water exothermic reaction has been considered. The LOCA peak cladding temperature analysis, presented in Chapter 15, demonstrates compliance with the Appendix K criteria demonstrates that no appreciable level of zirconium oxidation occurs. This level of reaction has been bounded in the containment mass and energy release analysis by incorporating the heat of reaction from 1 percent of the zirconium surrounding the fuel. This exceeds the level predicted by the LOCA analysis and results in additional conservatism in the mass and energy release calculations.

6.2.1.3.2.6 Energy Inventories

The amount of mass and energy released to containment during a postulated LOCA is provided in summary Tables 6.2.1.3-9 and 6.2.1.3-10.

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6.2.1.3.2.7 Additional Information Required for Confirmatory Analysis

System parameters and hydraulic characteristics needed to perform confirmatory analysis are provided in Table 6.2.1.3-8 and Figures 6.2.1.3-1 through 6.2.1.3-4.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary-System Pipe Rupture Inside Containment

Steam line ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steam line rupture is dependent upon the configuration of the plant steam system, the containment design as well as the plant operating conditions and the size of the rupture. This section describes the methods used in determining the containment responses to a variety of postulated pipe breaks encompassing variations in plant operation.

6.2.1.4.1 Significant Parameters Affecting Steam Line Break Mass and Energy Releases

Four major factors influence the release of mass and energy following a steam line break: steam generator fluid inventory, primary-to-secondary heat transfer, protective system operation and the state of the secondary fluid blowdown. The following is a list of those plant variables which have significant influence on the mass and energy releases:

- Plant power level
- Main feedwater system design
- Startup feedwater system design
- Postulated break type, size, and location
- Availability of offsite power
- Safety system failures
- Steam generator reverse heat transfer and reactor coolant system metal heat capacity.

The following is a discussion of each of these variables.

6.2.1.4.1.1 Plant Power Level

Steam line breaks are postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power generally result in a greater total mass release to the containment. Because of increased energy storage in the primary plant, increased heat transfer in the steam generators and additional energy generation in the nuclear fuel, the energy released to the containment from breaks postulated to occur during power operation may be greater than for breaks occurring with the plant in a hot shutdown condition. Additionally, steam pressure and the dynamic conditions in the steam generators change with increasing power. They have significant influence on the rate of blowdown from the break following a steam break event.

Because of the opposing effects of changing power level on steam line break releases, no single power level can be pre-defined as a worst case initial condition for a steam line break event.

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Therefore, several different power levels (101%, 70%, 30%, 0%) spanning the operating range as well as the hot shutdown condition are analyzed.

6.2.1.4.1.2 Main Feedwater System Design

The rapid depressurization that occurs following a rupture may result in large amounts of water being added to the steam generators through the main feedwater system. Rapid closing isolation valves are provided in the main feedwater lines to limit this effect. The piping layout downstream of the isolation valves determine the volume in the feedwater lines that cannot be isolated from the steam generators. As the steam generator pressure decreases, some of the fluid in this volume will flash into the steam generator, providing additional secondary fluid that may exit out the rupture. This unisolated feedwater mass between the steam generator and isolation valve is accounted for within the results in Subsection 6.2.1.4.3.2. The assumed unisolable volume bounds the volume to either the feedwater control valve or the feedwater isolation valve on the faulted loop, so that no additional feedwater mass could be postulated due to a single failure of one of the valves.

The feedwater addition that occurs prior to closing of the feedwater line isolation valves is conservatively calculated based on the depressurization of the faulted steam generator, and assuming that the feedwater control valve is fully open in response to the increased steam flow rate.

6.2.1.4.1.3 Startup Feedwater System Design

Within the first minute following a steam line break, the startup feedwater system may be initiated on a low steam generator water level signal in combination with low main feedwater flow. The addition of startup feedwater to the steam generators would increase the secondary mass available for release to the containment through the steam line break if the actuation signal occurs. Maximum startup feedwater flow to the faulted steam generator is conservatively assumed after the main feedwater flow is isolated until the startup feedwater is automatically terminated on a Low-2 RCS Tcold signal.

6.2.1.4.1.4 Postulated Break Type, Size and Location

The steam line break is postulated as a full double-ended pipe rupture immediately downstream of the integral flow restrictor on the faulted steam generator. The forward break flow from the faulted steam generator is limited by the flow restrictor area (1.4 ft^2). The faulted steam generator is unisolable from the break location, and the forward break flow continues until the steam generator is empty. The reverse break flow consists of the initial steam within the steam line piping and steam that comes from the intact steam generator. The reverse break flow continues until main steam line isolation valve (MSIV) closure. The modeling of the reverse break flow does not differentiate the location of the MSIVs, and all steam that has exited the intact steam generator prior to MSIV closure is assumed to be released out the break. This bounds the possible effects of an MSIV failed open.

No liquid entrainment is credited in the break effluent from the double-ended pipe rupture. The release of dry saturated steam from the largest possible break size maximizes the mass and energy release to the containment.

6.2.1.4.1.5 Availability of Offsite Power

The effects of the assumption of the availability of offsite power are enveloped in the analysis.

Offsite power is assumed to be available where it maximizes the mass and energy released from the break because of the following:

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- The continued operation of the reactor coolant pumps until automatically tripped as a result of core makeup tank (CMT) actuation. This maximizes the energy transferred from the reactor coolant system to the steam generator.

- The continued operation of the feedwater pumps and actuation of the startup feedwater system until they are automatically terminated. This maximizes the steam generator inventories available for release.

- The AP1000 is equipped with the passive safeguards system including the CMT and the passive residual heat removal (PRHR) heat exchanger. Following a steam line rupture, these passive systems are actuated when their setpoints are reached. This decreases the primary coolant temperatures. The actuation and operation of these passive safeguards systems do not require the availability of offsite power.

  When the PRHR is in operation, the core-generated heat is dissipated to the in-containment refueling water storage tank (IRWST) via the PRHR heat exchanger. This causes a reduction of the heat transfer from the primary system to the steam generator secondary system and causes a reduction of mass and energy releases via the break.

Thus, the availability of ac power maximizes the mass and energy releases via the break. Therefore, blowdown occurring with the availability of offsite power is more limiting than cases where offsite power is not available.

6.2.1.4.1.6 Safety System Failures

The calculation of the mass and energy release following a steam line rupture is done to conservatively bound the possible increase of mass release due to safety system failures. Two failures, which are bounded are:

- Failure of one main steam isolation valve, as discussed in Subsection 6.2.1.4.1.4

- Failure of one main feedwater isolation valve, as discussed in Subsection 6.2.1.4.1.2

6.2.1.4.1.7 Steam Generator Reverse Heat Transfer and Reactor Coolant System Metal Heat Capacity

Once steam line isolation is complete, the steam generator in the intact steam loop becomes a source of energy that can be transferred to the steam generator with the broken line. This energy transfer occurs through the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes drops below the temperature of the secondary fluid in the intact unit, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steam line.

Similarly, the heat stored in the metal of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps is transferred to the primary coolant as the plant cooldown progresses. This energy also is available to be transferred to the steam generator with the broken line. Energy from the metal of the faulted steam generator is also transferred to the fluid in the faulted steam generator as the fluid temperature drops below the metal temperature.

The effects of the reactor coolant system metal, the steam generator metal, and the energy in the fluid from the intact steam generator are included in the results presented.

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6.2.1.4.2 Description of Blowdown Model

The steam line blowdown is calculated with the LOFTRAN code (Reference 6) which has been modified to include simulation of the AP1000 passive residual heat removal heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. Documentation of the code changes for the passive models is provided in Reference 31. The methodology for the steam line break analysis is based on Reference 5. The applicability of the LOFTRAN code to AP1000, and the applicability of the methodology used to analyze the steam line break blowdown are discussed in Reference 32.

6.2.1.4.3 Containment Response Analysis

The WGOTHIC Computer Code (Reference 20) is used to determine the containment responses following the steam line break, which is described in Subsection 6.2.1.1.

6.2.1.4.3.1 Initial Conditions

The initial containment conditions are discussed in Subsection 6.2.1.1.3.

6.2.1.4.3.2 Mass and Energy Release Data

Using References 5, 6, 31 and 32 as a basis, mass and energy release data are developed to determine the containment pressure-temperature response for the cases analyzed. Table 6.2.1.4-2 provides the mass and energy release analysis results for the steam line break case that produces the highest containment pressure and Table 6.2.1.4-3 provides the mass and energy release analysis results for the steam line break case that produces the highest containment temperature. Table 6.2.1.4-4 provides nominal plant data used in the steam line break mass and energy releases determination.

6.2.1.4.3.3 Containment Pressure-Temperature Results

The results of the containment pressure-temperature analyses for the postulated secondary system pipe ruptures that produce the highest peak containment pressure and temperature are presented in Subsection 6.2.1.1.3.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System (PWR)

An analysis is performed to establish a minimum containment pressure boundary condition applied to the WCOBRA/TRAC code for the cold leg guillotine break for the emergency core cooling system (ECCS) analysis presented in Subsection 15.6.5.4A. A single-node containment model is used with the WGOTHIC computer code to assess the containment pressure response. The calculated containment backpressure is provided in Figure 6.2.1.5-1. Results of the WCOBRA/TRAC analyses demonstrate that the AP1000 meets 10 CFR 50.46 requirements (Reference 7).

Minimum containment pressure analyses are also performed to provide the containment backpressure boundary condition for small break LOCA (Subsection 15.6.5.4B) and post-LOCA long-term cooling analyses (Subsection 15.6.5.4C). A multi-node WGOTHIC containment model described in WCAP-15846 (Reference 20) is used.

6.2.1.5.1 Mass and Energy Release Data

The mass and energy releases to the containment for the double-ended cold-leg guillotine break (DECLG) transient are presented in Table 6.2.1.5-1, as computed by the WCOBRA/TRAC code.

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The mathematical models which calculate the mass and energy releases to the containment are described in Subsection 15.6.5. A break spectrum analysis is performed (see references in Subsection 15.6.5) that considers various break sizes and Moody discharge coefficients for the double-ended cold leg guillotines and splits. Mixing of steam and accumulator water injected into the vessel reduces the available energy released to the containment vapor space, thereby minimizing calculated containment pressure.

6.2.1.5.2 Initial Containment Internal Conditions

Initial containment conditions were biased for the emergency core cooling system backpressure analysis to predict a conservatively low containment backpressure. Initial containment conditions include an initial pressure of 14.5 psia, initial containment temperature of 50°F, and a relative humidity of 100 percent. An air annulus temperature of 0°F is assumed. The initial through-thickness metal temperature of the containment shell is assumed to also be 0°F.

6.2.1.5.3 Other Parameters

Containment parameters, such as containment volume and passive heat sinks, are biased to predict a conservative low containment backpressure. The containment volume used in the calculation is conservatively set to 1.1 times the free volume of the AP1000 containment Evaluation Model. Containment internal heat sinks used heat transfer correlations of 4 times Tagami (Reference 9) during the blowdown phase followed by 1.2 times Uchida (Reference 10) for the post-blowdown phase. Passive heat sink surface areas were increased by a factor of 1.35 relative to the heat sinks developed for the peak containment evaluation model described in Reference 20. The thermal conductivity and volumetric heat capacity of the steel materials were increased by 10% from the ASME standard to increase heat removal. No air gap was modeled between the steel liner and base concrete of jacketed concrete heat sinks. The outside surface of the containment shell was maintained at 0°F throughout the calculation. To further minimize containment pressure, containment purge was assumed to be in operation at time zero and air is vented through both the containment purge supply and exhaust lines until the isolation valves have fully closed. These valves were modeled to close after the High-2 containment pressure setpoint was reached and after an associated delay to account for valve closure time.

6.2.1.6 Testing and Inspection

This section describes the functional testing of the containment vessel. Testing and in-service inspection of the containment vessel are described in Subsection 3.8.2.6. Isolation testing and leak testing are described in Subsection 6.2.5. Testing and inspection are consistent with regulatory requirements and guidelines.

The valves of the passive containment cooling system are stroke tested periodically. Subsection 6.2.2 provides a description of testing and inspection.

The baffle between the containment vessel and the shield building is equipped with removable panels to allow for inspection of the containment surface. See Subsection 3.8.2 for the requirements for in-service inspection of the steel containment vessel. Subsection 6.2.2 provides a description of testing and inspection to be performed.

Testing is not required on any subcompartment vent or on the collection of condensation from the containment shell. The collection of condensate from the containment shell and its use in leakage detection are discussed in Subsection 5.2.5.

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6.2.1.7 Instrumentation Requirements

Instrumentation is provided to monitor the conditions inside the containment and to actuate the appropriate engineered safety features, should those conditions exceed the predetermined levels. The instruments measure the containment pressure, containment atmosphere radioactivity, and containment hydrogen concentration. Instrumentation to monitor reactor coolant system leakage into containment is described in Subsection 5.2.5.

The containment pressure is measured by four independent pressure transmitters. The signals are fed into the engineered safety features actuation system, as described in Subsection 7.3.1. Upon detection of high pressure inside the containment, the appropriate safety actuation signals are generated to actuate the necessary safety-related systems. Low pressure is alarmed but does not actuate the safety-related systems.

The physically separated pressure transmitters are located outside the containment. Section 7.3 provides a description.

The containment atmosphere radiation level is monitored by four independent area monitors located above the operating deck inside the containment building. The measurements are continuously fed into the engineered safety features actuation system logic. Section 11.5 provides information on the containment area radiation monitors. The engineered safety features actuation system operation is described in Section 7.3.

The containment hydrogen concentration is measured by hydrogen monitors, as described in Subsection 6.2.4. Hydrogen concentrations are monitored by three sensors located in the upper containment dome to provide a representative indication of bulk containment hydrogen concentration.

These indications are used by the plant operators to monitor hydrogen concentrations. High hydrogen concentration is alarmed in the main control room.

6.2.2 Passive Containment Cooling System

The passive containment cooling system (PCS) is an engineered safety features system. Its functional objective is to reduce the containment temperature and pressure following a loss of coolant accident (LOCA) or main steam line break (MSLB) accident inside the containment by removing thermal energy from the containment atmosphere. The passive containment cooling system also serves as the means of transferring heat to the safety-related ultimate heat sink for other events resulting in a significant increase in containment pressure and temperature.

The passive containment cooling system limits releases of radioactivity (post-accident) by reducing the pressure differential between the containment atmosphere and the external environment, thereby diminishing the driving force for leakage of fission products from the containment to the atmosphere. This subsection describes the safety design bases of the safety-related containment cooling function. Nonsafety-related containment cooling, a function of the containment recirculation cooling system, is described in Subsection 9.4.6.

The passive containment cooling system also provides a source of makeup water to the spent fuel pool in the event of a prolonged loss of normal spent fuel pool cooling.

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6.2.2.1 Safety Design Basis

- The passive containment cooling system is designed to withstand the effects of natural phenomena such as ambient temperature extremes, earthquakes, winds, tornadoes, or floods.

- Passive containment cooling system operation is automatically initiated upon receipt of a High-2 containment pressure signal.

- The passive containment cooling system is designed so that a single failure of an active component, assuming loss of offsite or onsite ac power sources, will not impair the capability of the system to perform its safety-related function.

- Active components of the passive containment cooling system are capable of being tested during plant operation. Provisions are made for inspection of major components in accordance with the intervals specified in the ASME Code, Section XI.

- The passive containment cooling system components required to mitigate the consequences of an accident are designed to remain functional in the accident environment and to withstand the dynamic effects of the accident.

- The passive containment cooling system is capable of removing sufficient thermal energy including subsequent decay heat from the containment atmosphere following a design basis event resulting in containment pressurization such that the containment pressure remains below the design value with no operator action required for 72 hours.

- The passive containment cooling system is designed and fabricated to appropriate codes consistent with Regulatory Guides 1.26 and 1.32 and in accordance with Regulatory Guide 1.29 as described in Section 1.9.

6.2.2.2 System Design

6.2.2.2.1 General Description

The passive containment cooling system and components are designed to the codes and standards identified in Section 3.2; flood design is described in Section 3.4; missile protection is described in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is described in Section 3.6. Seismic and environmental design and equipment qualification are described in Sections 3.10 and 3.11. The actuation system is described in Section 7.3.

6.2.2.2.2 System Description

The passive containment cooling system is a safety-related system which is capable of transferring heat directly from the steel containment vessel to the environment. This transfer of heat prevents the containment from exceeding the design pressure and temperature following a postulated design basis accident, as identified in Chapters 6 and 15. The passive containment cooling system makes use of the steel containment vessel and the concrete shield building surrounding the containment. The major components of the passive containment cooling system are: the passive containment cooling water storage tank (PCCWST) which is incorporated into the shield building structure above the containment; an air baffle, located between the steel containment vessel and the concrete shield building, which defines the cooling air flowpath; air inlets and an air exhaust, also incorporated into the shield building structure; and a water distribution system, mounted on the outside surface of the steel containment vessel, which functions to distribute water flow on the containment. A passive containment cooling ancillary water storage tank and two recirculation pumps are provided for onsite

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storage of additional passive containment cooling system cooling water, to transfer the inventory to the passive containment cooling water storage tank, and to provide a back-up supply to the fire protection system (FPS) seismic standpipe system as discussed in Subsection 9.5.1.

A normally isolated, manually-opened flow path is available between the passive containment cooling system water storage tank and the spent fuel pool.

The passive containment cooling water outlet valves are located in the PCS valve room. Due to the unique location of this room, insulation is required to allow the room to passively keep PCS valve room components and piping above freezing for 72 hours following loss of AC power in cold weather. Following seismic and environmental design basis events, the safety related function of the insulation is the R-value. The insulation is supported as seismic Category I.

A recirculation path is provided to control the passive containment cooling water storage tank water chemistry and to provide heating for freeze protection. Passive containment cooling water storage tank filling operations and normal makeup needs are provided by the demineralized water transfer and storage system discussed in Subsection 9.2.4.

The system piping and instrumentation diagram is shown in Figure 6.2.2-1. System parameters are shown in Table 6.2.2-1. A simplified system sketch is included as Figure 6.2.2-2.

6.2.2.2.3 Component Description

The mechanical components of the passive containment cooling system are described in this subsection. Table 6.2.2-2 provides the component design parameters.

Passive Containment Cooling Water Storage Tank – The passive containment cooling water storage tank is incorporated into the shield building structure above the containment vessel. The inside wetted walls of the tank are lined with stainless steel plate. It is filled with demineralized water and has the minimum required useable volume for the passive containment cooling function as defined in Table 6.2.2-2. The passive containment cooling system functions as the safety-related ultimate heat sink. The passive containment cooling water storage tank is seismically designed and missile protected.

The surrounding reinforced concrete supporting structure is designed to ACI 349 as described in Subsection 3.8.4.3. The welded seams of the plates forming part of the leak tight boundary are examined by liquid penetrant after fabrication to confirm that the boundary does not leak.

The tank also has redundant level measurement channels and alarms for monitoring the tank water level and redundant temperature measurement channels to monitor and alarm for potential freezing. To maintain system operability, a recirculation loop that provides chemistry and temperature control is connected to the tank.

The tank is constructed to provide sufficient thermal inertia and insulation such that draindown can be accomplished without heater operation.

In addition to its containment heat removal function, the passive containment cooling water storage tank also serves as a source of makeup water to the spent fuel pool and a seismic Category I water storage reservoir for fire protection following a safe shutdown earthquake.

The PCCWST suction pipe for the fire protection system is configured so that actuation of the fire protection system will not infringe on the usable capacity allocated to the passive containment cooling function as defined in Table 6.2.2-2.

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Passive Containment Cooling Water Storage Tank Isolation Valves – The passive containment cooling system water storage tank outlet piping is equipped with three sets of redundant isolation valves. In two sets, air-operated butterfly valves are normally closed and open upon receipt of a High-2 containment pressure signal. These valves fail-open, providing a fail-safe position, on the loss of air or loss of 1E dc power. In series with these valves are normally-open motor-operated gate valves located upstream of the butterfly valves. They are provided to allow for testing or maintenance of the butterfly valves. A third set of motor-operated gate valves is provided. One valve is normally closed, and the other is normally open. Based on PRA insights, diversity requirements are adopted for these valves to minimize the consequences of common-mode failure of motor-operated valves to cause a loss of containment cooling in multiple failure scenarios.

The storage tank isolation valves, along with the passive containment cooling water storage tank discharge piping and associated instrumentation between the passive containment cooling water storage tank and the downstream side of the isolation valves, are contained within a temperature-controlled valve room to prevent freezing. Valve room heating is provided to maintain the room temperature above 70°F to prevent freezing for 72 hours following a loss of ac power during seismic and design basis events.

Flow Control Orifices – Orifices are installed in each of the four passive containment cooling water storage tank outlet pipes. They are used, along with the different elevations of the outlet pipes, to control the flow of water from the passive containment cooling water storage tank as a function of water level. The orifices are located within the temperature-controlled valve room.

Water Distribution Bucket – A water distribution bucket is provided to deliver water to the outer surface of the containment dome. The redundant passive containment cooling water delivery pipes and auxiliary water source piping discharge into the bucket, below its operational water level, to prevent excessive splashing. A set of circumferentially spaced distribution slots are included around the top of the bucket. The bucket is hung from the shield building roof and suspended just above the containment dome for optimum water delivery. The structural requirements for safety-related structural steel identified in Subsection 3.8.4 apply to the water distribution bucket. ANSI/ASCE-8-90 (Reference 24) is used for design and analysis of stainless steel cold formed parts. The water distribution bucket is fabricated from one or more of the materials included in Table 3.8.4-6, ASTM-A240 austenitic stainless steel, or ASTM-A276 austenitic stainless steel.

Water Distribution Weir System – A weir-type water delivery system is provided to optimize the wetted coverage of the containment shell during passive containment cooling system operation. The water delivered to the center of the containment dome by the water distribution bucket flows over the containment dome, being distributed evenly by slots in the distribution bucket. Vertical divider plates are attached to the containment dome and originate at the distribution bucket extending radially along the surface of the dome to the first distribution weir. The divider plates limit maldistribution of flow which might otherwise occur due to variations in the slope of the containment dome. At the first distribution weir set, the water in that sector is collected and then redistributed onto the containment utilizing channeling walls and collection troughs equipped with distribution weirs. A second set of weirs are installed on the containment dome at a greater radius to again collect and then redistribute the cooling water to enhance shell coverage. The system includes channeling walls and collection troughs, equipped with distribution weirs. The distribution system is capable of functioning during extreme low- or high-ambient temperature conditions. The structural requirements for safety-related structural steel and cold formed steel structures identified in Subsection 3.8.4 apply to the water distribution weir system. ANSI/ASCE-8-90, (Reference 24) is used for design and analysis of stainless steel cold formed parts. The water distribution weir system is fabricated from one or more of the materials included in Table 3.8.4-6, ASTM-A240 austenitic stainless steel, or ASTM-A276 austenitic stainless steel.

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Air Flow Path – An air flow path is provided to direct air along the outside of the containment shell to provide containment cooling. The air flow path includes a screened shield building inlet, an air baffle that divides the outer and inner flow annuli, and a chimney to increase buoyancy. Subsection 3.8.4.1.3 includes information regarding the air baffle. The general arrangement drawings provided in Section 1.2 provide layout information of the air flow path.

Passive Containment Cooling Ancillary Water Storage Tank – The passive containment cooling ancillary water storage tank is a cylindrical steel tank located at ground level near the auxiliary building. It is filled with demineralized water and has a volume of greater than required for makeup to the passive containment cooling water storage tank and the spent fuel pool as defined in Table 6.2.2-2. The PCCAWST volume includes the base amount of water needed to maintain PCCWST and spent fuel pool makeup inventory by providing 135 gpm for four days (780,000 gal.) plus an additional volume intended to provide margin for potential instrument inaccuracies, flow control valve mis-positioning, and makeup inventory for the fire protection system. The tank is analyzed, designed and constructed using the method and criteria for Seismic Category II components defined in Subsections 3.2.1 and 3.7.2, except that 3D SASSI models are used for the tank due to its location relative to the Nuclear Island. The tank is designed and analyzed for Category 5 hurricanes including the effects of sustained winds, maximum gusts, and associated wind-borne missiles.

The tank has a level measurement, an alarm for monitoring the tank water level and a temperature measurement channel to monitor and alarm for potential freezing. To maintain system operability, an internal heater, controlled by the temperature instrument, is provided to maintain water contents above freezing. Chemistry can be adjusted by passive containment cooling water storage tank recirculation loop.

The tank is insulated to assure sufficient thermal inertia of the contents is available to prevent freezing for 7 days without heater operation. The transfer piping is maintained dry also to preclude freezing.

Chemical Addition Tank – The chemical addition tank is a small, vertical, cylindrical tank that is sized to inject a solution of hydrogen peroxide to maintain a passive containment cooling water storage tank concentration for control of algae growth.

Recirculation Pumps – Each recirculation pump is a 100 percent capacity centrifugal pump with wetted components made of austenitic stainless steel. The pump is sized to recirculate the entire volume of PCCWST water once every week. Each pump is capable of providing makeup flow to both the PCCWST and the spent fuel pool simultaneously. Both pumps are operated in parallel to meet fire protection system requirements.

Recirculation Heater – The recirculation heater is provided for freeze protection. The heater is sized based on heat losses from the passive containment cooling water storage tank and recirculation piping at the minimum site temperature, as defined in Section 2.3.

6.2.2.2.4 System Operation

Operation of the passive containment cooling system is initiated upon receipt of two out of four High-2 containment pressure signals. Manual actuation by the operator is also possible from either the main control room or remote shutdown workstation. System actuation consists of opening the passive containment cooling water storage tank isolation valves. This allows the passive containment cooling water storage tank water to be delivered to the top, external surface of the steel containment shell. The flow of water, provided entirely by the force of gravity, forms a water film over the dome and side walls of the containment structure.

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The flow of water to the containment outer surface is initially established for short-term containment cooling following a design basis loss of coolant accident. The flow rate is reduced over a period of not less than 72 hours. This flow provides the desired reduction in containment pressure over time and removes decay heat. The flow rate change is dependent only upon the decreasing water level in the passive containment cooling water storage tank. Prior to 72 hours after the event, operator actions are taken to align the passive containment cooling ancillary water storage tank to the suction of the passive containment cooling system recirculation pumps to replenish the cooling water supply to the passive containment cooling water storage tank. Sufficient inventory is available within the passive containment cooling ancillary water storage tank to maintain the minimum flow rate for an additional 4 days. The passive containment cooling system performance parameters are identified in Table 6.2.2-1.

To adequately wet the containment surface, the water is delivered to the distribution bucket above the center of the containment dome which subsequently delivers the water to the containment surface. A weir-type water distribution system is used on the dome surface to distribute the water for effective wetting of the dome and vertical sides of the containment shell. The weir system contains radial arms and weirs located considering the effects of tolerances of the containment vessel design and construction. A corrosion-resistant paint or coating for the containment vessel is specified to enhance surface wetability and film formation.

The cooling water not evaporated from the vessel wall flows down to the bottom of the inner containment annulus into annulus drains. The redundant annulus drains route the excess water out of the upper annulus. The annulus drains are located in the shield building wall slightly above the floor level to minimize the potential for clogging of the drains by debris. The drains are horizontal or have a slight slope to promote drainage. The drains are always open (without isolation valves) and each is sized to accept maximum passive containment cooling system flow. The outside ends of the drains are located above catch basins or other storm drain collectors.

A path for the natural circulation of air upward along the outside walls of the containment structure is always open. The natural circulation air flow path begins at the shield building inlet, where atmospheric air is turned upward from the horizontal by louvers outside the concrete structure. Air flows past the set of fixed louvers and is forced to turn downward into an outer annulus. This outer shield building annulus is encompassed by the concrete shield building on the outside and a removable baffle on the inside. This inner annulus is encompassed by the baffle wall on the outside and the steel containment vessel on the inside. Air flows up through the inner annulus to the top of the containment vessel and then exhausts through the shield building chimney.

As the containment structure heats up in response to high containment temperature, heat is removed from within the containment via conduction through the steel containment vessel, convection from the containment surface to the water film, convection and evaporation from the water film to the air, and radiation from the water film to the air baffle. As heat and water vapor are transferred to the air space between the containment structure and air baffle, the air becomes less dense than the air in the outer annulus. This density difference causes an increase in the natural circulation of the air upward between the containment structure and the air baffle, with the air finally exiting at the top center of the shield building.

The passive containment cooling water storage tank provides water for containment wetting for at least 72 hours following system actuation. Operator action can be taken to replenish this water supply from the passive containment cooling ancillary water storage tank or to provide an alternate water source directly to the containment shell through an installed safety-related seismic piping connection. In addition, water sources used for normal filling operations can be used to replenish the water supply.

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The arrangement of the air inlet and air exhaust in the shield building structure has been selected so that wind effects aid the natural air circulation. The air inlets are placed at the top, outside of the shield building, providing a symmetrical air inlet that reduces the effect of wind speed and direction or adjacent structures. The air/water vapor exhaust structure is elevated above the air inlet to provide additional buoyancy and reduces the potential of exhaust air being drawn into the air inlet. The air flow inlet and chimney regions are both designed to protect against ice or snow buildup and to prevent foreign objects from entering the air flow path.

Inadvertent actuation of the passive containment cooling system is terminated through operator action by closing either of the series isolation valves from the main control room. Subsection 6.2.1.1.4 provides a discussion of the effects of inadvertent system actuation.

The passive containment cooling system provides for makeup water to the spent fuel pool to provide for continued spent fuel pool inventory and heat removal. The passive containment cooling water storage tank provides makeup to the spent fuel pool when the inventory is not required for passive containment cooling system operation. An installed long term makeup connection for the passive containment cooling system and the spent fuel pool is provided as a part of the passive containment cooling system. The passive containment cooling ancillary water storage tank and the passive containment cooling system recirculation pumps may also be utilized for makeup to the spent fuel pool.

The passive containment cooling system provides spray water to the spent fuel pool spray header. Use of the PCCWST to provide water to the spent fuel pool spray header is controlled by the Extensive Damage Mitigation Guidelines (EDMG) per NEI 06-12 (Reference 33).

6.2.2.3 Safety Evaluation

The safety-related portions of the passive containment cooling system are located within the shield building structure. This building (including the safety-related portions of the passive containment cooling system) is designed to withstand the effects of natural phenomena such as earthquakes, winds, tornadoes, or floods. Components of the passive containment cooling system are designed to withstand the effects of ambient temperature extremes.

The portions of the passive containment cooling system which provide for long term (post 72-hour) water supply for containment wetting are located in Seismic Category I or Seismic Category II structures excluding the passive containment cooling ancillary water storage tank and associated valves located outside of the auxiliary building. The water storage tank and the anchorage for the associated valves are Seismic Category II. The features of these structures which protect this function are analyzed and designed for Category 5 hurricanes including the effects of sustained winds, maximum gusts, and associated wind-borne missiles. The resistance of the full-scale, as-designed passive containment cooling air flow path is verified by scale testing and analysis (Reference 38).

Operation of the containment cooling system is initiated automatically following the receipt of a High-2 containment pressure signal. The use of this signal provides for system actuation during transients, resulting in mass and energy releases to containment, while avoiding unnecessary actuations. System actuation requires the opening of any of the three normally closed isolation valves, with no other actions required to initiate the post-accident heat removal function since the cooling air flow path is always open. Operation of the passive containment cooling system may also be initiated from the main control room and from the remote shutdown workstation. A description of the actuation system is contained in Section 7.3.

The active components of the passive containment cooling system, the isolation valves, are located in three redundant pipe lines. Failure of a component in one train does not affect the operability of the

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other mechanical train or the overall system performance. The fail-open, air-operated valves require no electrical power to move to their safe (open) position. The normally open motor-operated valves are powered from separate redundant Class 1E dc power sources. Table 6.2.2-3 presents a failure modes and effects analysis of the passive containment cooling system.

Capability is provided to periodically test actuation of the passive containment cooling system. Active components can be tested periodically during plant operation to verify operability. The system can be inspected during unit shutdown. Additional information is contained in Subsections 3.9.6 and 6.2.2.4, as well as in the Technical Specifications.

There are four instrument lines that penetrate containment and are required to remain functional following an accident. The lines are used to sense the pressure of the containment atmosphere and convey it to pressure transmitters outside containment. The pressure transmitters, tubing, and pressure sensors inside containment comprise a sealed, fluid-filled assembly forming a double barrier between inside and outside containment. If the instrument line breaks outside containment, leakage of containment atmosphere is prevented by the pressure sensor and the sealed tubing boundary inside containment. If a break occurs inside containment, leakage is prevented by the transmitter and tubing boundary outside containment. The pressure sensors, tubing, and pressure transmitters are designed and tested for seismic Category I service.

The containment pressure analyses are based on an ambient air temperature of 115°F dry bulb and 86.1°F coincident wet bulb. The passive containment cooling water storage tank water temperature basis is 120°F. Results of the analyses are provided in Subsection 6.2.1.

The shield building air inlets were changed as part of the enhanced shield building design. The impact of these changes on the containment pressure analyses is small, and the conclusions remain valid. The analyses provided in Subsection 6.2.1 include the air inlet changes.

6.2.2.4 Testing and Inspection

6.2.2.4.1 Inspections

The passive containment cooling system is designed to permit periodic testing of system readiness as specified in the Technical Specifications.

The portions of the passive containment cooling system from the isolation valves to the passive containment cooling water storage tank are accessible and can be inspected during power operation or shutdown for leaktightness. Examination and inspection of the pressure retaining piping welds is performed in accordance with ASME Code, Section XI. The design of the containment vessel and air baffle retains provisions for the inspection of the vessel during plant shutdowns.

6.2.2.4.2 Preoperational Testing

Preoperational testing of the passive containment cooling system is verified to provide adequate cooling of the containment. The flow rates are confirmed at the minimum initial tank level, an intermediate step with all but one standpipe delivering flow and at a final step with all but two standpipes delivering to the containment shell. The flow rates are measured utilizing the differential pressure across the orifices within each standpipe and will be consistent with the flow rates specified in Table 6.2.2-1.

The containment coverage will be measured at the base of the upper annulus in addition to the coverage at any elevation between elevation 266 ft. and the spring line for the full flow case using the PCS water storage tank delivering to the containment shell and a lower flow case with both PCS recirculation pumps delivering to the containment shell. For the low flow case, a throttle valve is used

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to obtain a low flow rate less than the full capacity of the PCS recirculation pumps. This flow rate is then re-established for subsequent tests using the throttle valve. These benchmark values will be used to develop acceptance criteria for the Technical Specifications. The full flow condition is selected since it is the most important flow rate from the standpoint of peak containment pressure and the lower flow rate is selected to verify wetting characteristics at less than full flow conditions.

The standpipe elevations are verified to be at the values specified in Table 6.2.2-2.

The inventory within the tank is verified to provide 72 hours of operation from the minimum initial operating water level with a minimum flow rate over the duration in excess of 100.7 gpm or a report exists and concludes that the as-measured flow rates bound the 72 hours containment peak pressure and temperature results. The flow rates are measured utilizing the differential pressure across the orifices within each standpipe.

The containment vessel exterior surface is verified to be coated with an inorganic zinc coating.

The passive containment cooling air flow path will be verified at the following locations:

- Air inlets
- Base of the outer annulus
- Base of the inner annulus
- Discharge structure

With either a temporary water supply or the passive containment cooling ancillary water storage tank connected to the suction of the recirculation pumps and with either of the two pumps operating, flow must be provided simultaneously to the passive containment cooling water storage tank and to the spent fuel pool for the following cases:

- Greater than or equal to 100 gpm to the passive containment cooling water storage tank and greater than or equal to 35 gpm to the spent fuel pool
- Greater than or equal to 80 gpm to the passive containment cooling water storage tank and greater than or equal to 50 gpm to the spent fuel pool
- Greater than or equal to 50 gpm to the passive containment cooling water storage tank and greater than or equal to 80 gpm to the spent fuel pool

The passive containment cooling water storage tank provides makeup water to the spent fuel pool. When aligned to the spent fuel pool the flow rate is verified to exceed 118 gpm. Installed instrumentation will be utilized to verify the flow rate. The volume of the passive containment cooling water storage tank is verified to exceed the minimum usable volume defined in Table 6.2.2-2.

Additional details for preoperational testing of the passive containment cooling system are provided in Chapter 14.

6.2.2.4.3 Operational Testing

Operational testing is performed to:

- Demonstrate that the sequencing of valves occurs on the initiation of High-2 containment pressure and demonstrate the proper operation of remotely operated valves.

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- Verify valve operation during plant operation. The normally open motor-operated valves, in series with each normally closed air-operated isolation valve, are temporarily closed. This closing permits isolation valve stroke testing without actuation of the passive containment cooling system.

- Verify water flow delivery and containment water coverage, consistent with the accident analysis.

- Verify visually that the path for containment cooling air flow is not obstructed by debris or foreign objects.

- Test frequency is consistent with the plant Technical Specifications and inservice testing program (Subsection 3.9.6).

6.2.2.4.4 System Gas Accumulation Assessment and Mitigation

Subsection 6.3.6.3.1 describes the assessment method to address the potential for gas intrusion for the passive core cooling system in response to Interim Staff Guidance (ISG-019) and Generic Letter 2008-01. The same assessment methodology was used to assess the passive containment cooling system.

The passive containment cooling system locations equipped with manual vent valves will be inspected according to the system surveillance and venting procedures described in Subsection 6.3.6.3 to eliminate identified gas accumulations.

6.2.2.5 Instrumentation Requirements

The status of the passive containment cooling system is displayed in the main control room. The operator is alerted to problems with the operation of the equipment within this system during both normal and post-accident conditions.

Normal operation of the passive containment cooling system is demonstrated by monitoring the recirculation pump discharge pressure, flow rate, water storage tank level and temperature, and valve room temperature. Post-accident operation of the passive containment cooling system is demonstrated by monitoring the passive containment cooling water storage tank level, passive containment cooling system cooling water flow rate, containment pressure, and external cooling air discharge temperature.

The information on the activation signal-generating equipment is found in Chapter 7.

The protection and safety monitoring system providing system actuation is discussed in Chapter 7.

6.2.3 Containment Isolation System

The major function of the containment isolation system of the AP1000 is to provide containment isolation to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary, if required. This prevents or limits the escape of fission products that may result from postulated accidents. Containment isolation provisions are designed so that fluid lines which penetrate the primary containment boundary are isolated in the event of an accident. This minimizes the release of radioactivity to the environment.

The containment isolation system consists of the piping, valves, and actuators that isolate the containment. The design of the containment isolation system satisfies the requirements of NUREG 0737, as described in the following paragraphs.

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6.2.3.1 Design Basis

6.2.3.1.1 Safety Design Basis

A. The containment isolation system is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (General Design Criterion 2).

B. The containment isolation system is designed to remain functional after a safe shutdown earthquake (SSE) and to perform its intended function following the postulated hazards of fire, internal missiles, or pipe breaks (General Design Criteria 3 and 4).

C. The containment isolation system is designed and fabricated to codes consistent with the quality group classification, described in Section 3.2. Conformance with Regulatory Guide 1.26, 1.29, and 1.32 is described in Section 1.9.

D. The containment isolation system provides isolation of lines penetrating the containment for design basis events requiring containment integrity.

E. Upon failure of a main steam line, the containment isolation system isolates the steam generators as required to prevent excessive cooldown of the reactor coolant system or overpressurization of the containment.

F. The containment isolation system is designed in accordance with General Design Criterion 54.

G. Each line that penetrates the containment that is either a part of the reactor coolant pressure boundary or that connects directly to the containment atmosphere, and does not meet the requirements for a closed system (as defined in paragraph H below), satisfies the requirements of General Design Criteria 55 and 56. For most lines, the safety design basis is isolation valve(s) in one of the configurations described in GDC 55 and GDC 56. The acceptable basis for isolation of instrument lines for containment pressure measurements is as specified in NUREG-0800, Standard Review Plan, Subsection 6.2.4:

   “Regulatory Guide (RG) 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines closed both inside and outside containment are designed to withstand pressure and temperature conditions following a loss-of-coolant accident (LOCA) and dynamic effects are acceptable without isolation valves.”

H. Each line that penetrates the containment, that is neither part of the reactor coolant pressure boundary nor connected directly to the atmosphere of the containment, and that satisfies the requirements of a closed system is provided with a containment isolation valve according to General Design Criterion 57. A closed system is not a part of the reactor coolant pressure boundary and is not connected directly to the atmosphere of the containment. A closed system also meets the following additional requirements:

   - The system is protected against missiles and the effects of high-energy line break.
   - The system is designed to Seismic Category I requirements.
   - The system is designed to ASME Code, Section III, Class 2 requirements.
   - The system is designed to withstand temperatures at least equal to the containment design temperature.

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- The system is designed to withstand the external pressure from the containment structural acceptance test.

- The system is designed to withstand the design basis accident transient and environment.

I. The containment isolation system is designed so that no single failure in the containment isolation system prevents the system from performing its intended functions.

J. Fluid penetrations supporting the engineered safety features functions have remote manual isolation valves. These valves can be closed from the main control room or from the remote shutdown workstation, if required.

K. The containment isolation system is designed according to 10 CFR 50.34, so that the resetting of an isolation signal will not cause any valve to change position.

6.2.3.1.2 Power Generation Design Basis

The containment isolation system has no power generation design basis. Power generation design bases associated with individual components of the containment isolation system are discussed in the section describing the system of which they are an integral part.

6.2.3.1.3 Additional Requirements

The AP1000 containment isolation system is designed to meet the following additional requirements:

A. The containment isolation elements are designed to minimize the number of isolation valves which are subject to Type C tests of 10 CFR 50, Appendix J. Specific requirements are the following:

- The number of pipe lines which provide a direct connection between the inside and outside of primary containment during normal operation are minimized.

- Closed systems outside of containment that may be open to the containment atmosphere during an accident are designed for the same conditions as the containment itself, and are testable during Type A leak tests.

- The total number of penetrations requiring isolation valves are minimized by appropriate system design. For example:

  – In the component cooling system, a single header with branch lines inside of containment is employed instead of providing a separate penetration for each branch line.

  – Consistent with other considerations, such as containment arrangement and exposure of essential safety equipment to potentially harsh environments, the equipment is located inside and outside of containment so as to require the smallest number of penetrations.

- Consistent with current practice, Type C testing is not required for pressurized water reactor main steam, feedwater, startup feedwater, or steam generator blowdown isolation valves. The steam generator tubes are considered to be a suitable boundary to prevent

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release of radioactivity from the reactor coolant system following an accident. The steam generator shell and pipe lines, up to and including the first isolation valve, are considered a suitable boundary to prevent release of containment radioactivity.

B. Personnel hatches and equipment hatches are sealed by closures with double gaskets. The fuel transfer tube is sealed with a hatch/blind flange inside containment.

C. Containment isolation is actuated on a two-out-of-four logic from within the protection and safety monitoring system. The safeguards signals provided to each isolation valve are selected to enhance plant safety. Provisions are provided for manual containment isolation from the main control room.

D. Penetration lines with automatic isolation valves are isolated by engineered safety features actuation signals.

E. Isolation valves are designed to provide leaktight service against the medium to which the valves are exposed in the short and long-term course of any accident. For example, a valve is gas-tight if the valve is exposed to the containment atmosphere.

F. Isolation valves are designed to have the capacity to close against the conditions that may exist during events requiring containment isolation.

G. Isolation valve closure times are designed to limit the release of radioactivity to within regulation and are consistent with standard valve operators, except where a shorter closure time is required.

H. The position of each power-operated isolation valve (fully closed or open), whether automatic or remote manual, is indicated in the main control room and is provided as input to the data display and processing system. Such position indication is based on actual valve position, for example, by a limit switch which directly senses the actual valve stem position, rather than demanded valve position.

I. Normally closed manual containment isolation valves have provisions for locking the valves closed. Locking devices are designed such that the valves can be locked only in the fully closed position. Administrative control provides verification that manual isolation valves are maintained locked closed during normal operation. Position locks provide confidence that valves are placed in the correct position prior to locking.

J. Automatic containment isolation valves are powered by Class 1E dc power. Air-operated valves fail in the closed position upon loss of a support system, such as instrument air or electric power.

K. Valve alignments used for fluid system testing during operation are designed so that either: containment bypass does not occur during testing, assuming a single failure; or exceptions are identified, and remotely operated valves provide timely isolation from the control room. Containment isolation provisions can be relaxed during system testing. The intent of the design is to provide confidence that operators are aware of any such condition and have the capability to restore containment integrity.

L. A diverse method of initiating closure is provided for those containment isolation valves associated with penetrations representing the highest potential for containment bypass. Diverse actuation is discussed in Section 7.7.

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M. Containment penetrations with leaktight barriers, both inboard and outboard, are designed to limit pressure excursion between the barriers due to heatup of fluid between the barriers. The penetration will either be fitted with relief or check valves to relieve internal pressure or one of the valves has been designed or oriented to limit pressures to an acceptable value. For example, a penetration which incorporates two air-operated globe valves –one of the globe valves will be oriented such that pressure between the two valves will lift the plug from the seat to relieve the pressure, then reseat.

6.2.3.2 System Description

6.2.3.2.1 General Description

Piping systems penetrating the containment have containment isolation features. These features serve to minimize the release of fission products following a design basis accident. SRP Subsection 6.2.4 provides acceptable alternative arrangements to the explicit arrangements given in General Design Criteria 55, 56 and 57. Table 6.2.3-1 lists each penetration and provides a summary of the containment isolation characteristics. For isolation valves outside containment, Table 6.2.3-1 also lists the nominal length of pipe between the containment penetration and the isolation valve. The Piping and Instrumentation Diagrams of the applicable systems show the functional arrangement of the containment penetration, isolation valves, test and drain connections. Section 1.7 contains a list of the Piping and Instrumentation Diagrams.

As discussed in Subsection 6.2.3.1, the AP1000 containment isolation design satisfies the NRC requirements including post-Three Mile Island requirements. Two barriers are provided -- one inside containment and one outside containment. Usually these barriers are valves, but in some cases they are closed piping systems not connected to the reactor coolant system or to the containment atmosphere.

The AP1000 has fewer mechanical containment penetrations (including hatches) and a higher percentage of normally closed isolation valves than current plants. The majority of the penetrations that are normally open incorporate fail closed isolation valves that close automatically with the loss of support systems such as instrument air. Table 6.2.3-1 lists the AP1000 containment mechanical penetrations and the isolation valves associated with them. Provisions for leak testing are discussed in Subsection 6.2.5.

For those systems having automatic isolation valves or for those provided with remote-manual isolation, Subsection 6.2.3.5 describes the power supply and associated actuation system. Power-operated (air, motor, or pneumatic) containment isolation valves have position indication in the main control room.

The actuation signal that occurs directly as a result of the event initiating containment isolation is designated in Table 6.2.3-1. If a change in valve position is required at any time following primary actuation, a secondary actuation signal is generated which places the valve in an alternative position. The closure times for automatic containment isolation valves are provided in Table 6.2.3-1.

The containment air filtration system is used to purge the containment atmosphere of airborne radioactivity during normal plant operation. The containment vacuum relief system is a safety grade system, used to mitigate a containment external pressure scenario, and is part of the containment air filtration system. The containment air filtration system is designed in accordance with Branch Technical Position CSB 6-4. The purge component of the air filtration system uses 16-inch supply and exhaust lines and containment isolation valves. The vacuum relief component of the air filtration system uses 6-inch supply lines and containment isolation valves. These valves close automatically on a containment isolation signal. The entire containment air filtration system is described in Subsection 9.4.7.

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Section 3.6 describes dynamic effects of pipe rupture. Section 3.5 discusses missile protection, and Section 3.8 discusses the design of Category I structures including any structure used as a protective device. Lines associated with those penetrations that are considered closed systems inside the containment are protected from the effects of a pipe rupture and missiles. The actuators for power-operated isolation valves inside the containment are either located above the maximum containment water level or in a normally nonflooded area. The actuators are designed for flooded operation or are not required to function following containment isolation and designed and qualified not to spuriously open in a flooded condition.

Other defined bases for containment isolation are provided in SRP Subsection 6.2.4.

6.2.3.2.2 Component Description

Codes and standards applicable to the piping and valves associated with containment isolation are those for Class B components, as discussed in Section 3.2. Containment penetrations are classified as Quality Group B and Seismic Category I.

Section 3.11 provides the normal, abnormal, and post-loss-of-coolant accident environment that is used to qualify the operability of power-operated isolation valves located inside the containment.

The containment penetrations which are part of the main steam system and the feedwater system are designed to meet the stress requirements of NRC Branch Technical Position MEB 3-1, and the classification and inspection requirements of NRC Branch Technical Position ASB 3-1, as described in Section 3.6. Section 3.8 discusses the interface between the piping system and the steel containment.

As discussed in Subsection 6.2.3.5, the instrumentation and control system provides the signals which determine when containment isolation is required. Containment penetrations are either normally closed prior to the isolation signal or the valves automatically close upon receipt of the appropriate engineered safety features actuation signal.

6.2.3.2.3 System Operation

During normal system operation, approximately 25 percent of the penetrations are not isolated. These lines are automatically isolated upon receipt of isolation signals, as described in Subsections 6.2.3.3 and 6.2.3.4 and Chapter 7. Lines not in use during power operation are normally closed and remain closed under administrative control during reactor operation.

6.2.3.3 Design Evaluation

A. Engineered safeguards and containment isolation signals automatically isolate process lines which are normally open during operation. The containment isolation system uses diversity in the parameters sensed for the initiation of redundant train-oriented isolation signals. The majority of process lines are closed upon receipt of a containment isolation signal. This safeguards signal is generated by any of the following initiating conditions.

- Low-3 pressurizer pressure
- Low-2 steam-line pressure
- Low-2 T_{cold}
- High-2 containment pressure

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- Manual containment isolation actuation

The component cooling water lines penetrating containment provide cooling water to the reactor coolant pumps and chemical and volume control system and liquid radwaste system heat exchangers. The reactor coolant pumps are interlocked to trip following a safeguards actuation (S) signal but will continue to operate (if in service) following a containment isolation (T) signal. In order to provide reliable cooling to the reactor coolant pumps the component cooling lines are isolated on a safeguards actuation signal rather than on a containment isolation signal. The safeguards actuation signal is generated by any of the following conditions.

- Low-3 pressurizer pressure
- Low-2 steam line pressure
- Low-2 reactor coolant inlet temperature
- High-2 containment pressure
- Manual initiation

The chemical and volume control system charging line, normal residual heat removal system reactor coolant and IRWST cooling lines, and containment air filtration system containment purge lines are isolated on high containment radiation signals. Closure of the containment air filtration system isolation valves is based on providing rapid response to elevated activity conditions in containment to limit offsite doses and is initiated on either a high radiation signal or a containment isolation signal consistent with the requirements of NUREG-0737 (Reference 22) and NUREG-0718 Rev 2 (Reference 23). The isolation of the chemical and volume control system charging line on a High-2 radiation signal and normal residual heat removal system cooling lines on a High-2 radiation or safeguards actuation signal with provisions to reset safeguards actuation signal for the normal residual heat removal system valves permits a defense in depth response to a postulated accident by providing for normal residual heat removal system and chemical and volume control system operation unless there is a high radiation level present.

The remainder of the containment isolation valves are closed on parameters indicative of the need to isolate.

B. Upon failure of a main steam line, the steam generators are isolated, and the main steam-line isolation valves, main steam-line isolation bypass valves, power operated relief block valves, and the main steam-line drain are closed to prevent excessive cooldown of the reactor coolant system or overpressurization of the containment.

The two redundant train-oriented steam-line isolation signals are initiated upon receipt of any of the following signals:

- Low-2 steam-line pressure
- High steam pressure negative rate
- High-2 containment pressure
- Manual actuation
- Low-2 T_{cold}

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The main steam-line isolation valves, main steam line isolation valve bypass valves, main feedwater isolation valves, steam generator blowdown system isolation valves, and piping are designed to prevent uncontrolled blowdown from more than one steam generator. The main steam-line isolation valves and main feedwater isolation valves close fully within 5 seconds after an isolation is initiated. The blowdown rate is restricted by steam flow restrictors located within the steam generator outlet steam nozzles in each blowdown path. For main steam-line breaks upstream of an isolation valve, uncontrolled blowdown from more than one steam generator is prevented by the main steam-line isolation valves on each main steam line.

Failure of any one of these components relied upon to prevent uncontrolled blowdown of more than one steam generator does not permit a second steam generator blowdown to occur. No single active component failure results in the failure of more than one main steam isolation valve to operate. Redundant main steam isolation signals, described in Section 7.3, are fed to redundant parallel actuation vent valves to provide isolation valve closure in the event of a single isolation signal failure.

The effects on the reactor coolant system after a steam-line break resulting in single steam generator blowdown and the offsite radiation exposure after a steam line break outside containment are discussed in Chapter 15. The containment pressure transient following a main steam-line break inside containment is discussed in Section 6.2.

C. The containment isolation system is designed according to General Design Criterion 54. Leakage detection capabilities and leakage detection test program are discussed in Subsection 6.2.5. Valve operability tests are also discussed in Subsection 3.9.6. Redundancy of valves and reliability of the isolation system are provided by the other safety design bases stated in Section 6.2. Redundancy and reliability of the actuation system are covered in Section 7.3.

   The use of motor-operated valves that fail as-is upon loss of actuating power in lines penetrating the containment is based upon the consideration of what valve position provides the plant safety. Furthermore, each of these valves, is provided with redundant backup valves to prevent a single failure from disabling the isolation function. Examples include: a check valve inside the containment and motor-operated valve outside the containment or two motor-operated valves in series, each powered from a separate engineered safety features division.

D. Lines that penetrate the containment and which are either part of the reactor coolant pressure boundary, connect directly to the containment atmosphere, or do not meet the requirements for a closed system are provided with one of the following valve arrangements conforming to the requirements of General Design Criteria 55 and 56, as follows:

   - One locked-closed isolation valve inside and one locked-closed isolation valve outside containment
   - One automatic isolation valve inside and one locked-closed isolation valve outside containment
   - One locked-closed isolation valve inside and one automatic isolation valve outside containment. (A simple check valve is not used as the automatic isolation valve outside containment.)
   - One automatic isolation valve inside and one automatic isolation valve outside containment. (A simple check valve is not used as the automatic isolation valve outside containment).

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Isolation valves outside containment are located as close to the containment as practical. Upon loss of actuating power, air-operated automatic isolation valves fail closed.

In accordance with GDC 56, isolation of instrument lines for containment pressure transmitters is demonstrated on a different basis. The lines are closed inside and outside containment, and are designed to withstand pressure and temperature conditions following a loss-of-coolant accident (LOCA) and dynamic effects.

E. Each line penetrating the containment that is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere, and that satisfies the requirements of a closed system, has at least one containment isolation valve. This containment isolation valve is either automatic, locked-closed, or capable of remote-manual operation. The valve is outside the containment and located as close to the containment as practical. A simple check valve is not used as the automatic isolation valve. This design is in compliance with General Design Criterion 57.

F. The containment isolation system is designed according to seismic Category I requirements as specified in Section 3.2. The components (and supporting structures) of any system, equipment, or structure that are non-seismic and whose collapse could result in loss of a required function of the containment isolation system through either impact or resultant flooding are evaluated to confirm that they will not collapse when subjected to seismic loading resulting from a safe shutdown earthquake.

Air-operated isolation valves fail in the closed position upon loss of air or power. Containment isolation system valves required to be operated after a design basis accident or safe shutdown earthquake are powered by the Class 1E dc electric power system.

6.2.3.4 Tests and Inspections

6.2.3.4.1 Preoperational Testing

Preoperational testing is described in Chapter 14. The containment isolation system is testable through the operational sequence that is postulated to take place following an accident, including operation of applicable portions of the protection system and the transfer between normal and standby power sources.

The safety related function of containment boundary integrity is verified by an integrated leakage rate test. The integrated leakage rate is verified to be less than L_a as defined in Table 6.5.3-1. The integrated containment leakage rate system is utilized to measure the containment leak rate for determination of the integrated leakage rate. The containment isolation valves are verified to close within the time specified in Table 6.2.3-1.

The piping and valves associated with the containment penetration are designed and located to permit pre-service and in-service inspection according to ASME Section XI, as discussed in Subsection 3.9.6 and Section 6.6.

6.2.3.4.2 In-service Testing

Each line penetrating the containment is provided with testing features to allow containment leak rate tests according to 10 CFR 50, Appendix J, as discussed in Subsection 6.2.5.

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6.2.3.5 Instrumentation and Control Application

Instrumentation and control necessary for containment isolation, and the sensors used to determine that containment isolation is required, are described in Section 7.3.

Engineered safeguards actuation signals which initiate containment isolation will be initiated using two out of four logic. Containment isolation signals can also be initiated manually from the main control room. Containment isolation valves requiring isolation close automatically on receipt of a safeguards actuation signal.

Containment isolation valves that are equipped with power operators and are automatically actuated may also be controlled individually from the main control room. Also, a manual override of an automatic isolation signal is installed to permit manual control of the associated valve. For all valves except the vacuum relief containment isolation valves, the override control function can be performed only subsequent to resetting of the actuation signal. That is, deliberate manual action is required to change the position of containment isolation valves in addition to resetting the original actuation signal. Resetting of the actuation signal does not cause any valve to change position. The design does not allow ganged reopening of the containment isolation valves. Reopening of the isolation valves is performed on a valve-by-valve basis, or on a line-by-line basis. Safeguards actuation signals take precedence over manual overrides of other isolation signals. For example, a containment isolation signal causes isolation valve closure even though the high containment radiation signal is being overridden by the operator. Containment isolation valves with power operators are provided with open/closed indication, which is displayed in the main control room. The valve mechanism also provides a local mechanical indication of valve position.

As discussed in Subsection 9.4.7.2.3, the containment vacuum relief path includes normally closed motor-operated isolation valves, which are located outside the containment and open automatically to provide a flow path to allow atmospheric air into the containment to equalize differential pressure across the containment vessel shell. These valves also perform a containment isolation function when vacuum relief is not required. As discussed in Subsection 7.6.2.4, an interlock ensures the availability of the engineered safety features for the vacuum relief isolation valves to perform their vacuum relief and containment isolation functions.

If a negative containment pressure condition occurs that causes the vacuum relief isolation valves to automatically actuate open, there will not be a simultaneous need to close for containment isolation. The negative pressure inside the containment prevents expulsion of air from inside the containment when vacuum relief is actuated so that there are no challenges to the offsite dose limits or main control room habitability. Passive cooling system operations with low core decay heat may significantly delay containment pressurization.

Containment isolation is typically required for events that pressurize the containment with steam, such as a primary system or steam generator system line break, or operation of the passive core cooling systems. An event that causes containment pressurization precludes the need for vacuum relief actuation.

If containment conditions change following vacuum relief actuation so that the containment pressure increases, then the vacuum relief actuation signal (which is not latched) would clear and allow the containment isolation signal to automatically close the vacuum relief isolation valves. Since these valves would have recently opened for vacuum relief actuation during the event, it is expected that they would close. A relatively low containment pressure differential and mild containment conditions would be expected when the valves close for containment isolation during this event. Additionally, there are self-actuated vacuum relief valves inside the containment that are in series with the vacuum relief isolation valves, which provide single failure protection in the event that one of the motor-operated valves fails to close.

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There is a valve interlock between the inside containment purge exhaust isolation valve and the vacuum relief isolation valves, which limits the potential release of radioactivity from the containment while the containment isolation valves are being closed.

The valve interlock prevents having two parallel vent paths out of the containment in the event of an accident where a negative pressure condition inside the containment does not exist.

The valve interlock preserves the assumptions of the dose analyses, which are bounded by closure of the normal containment purge isolation valves. Having the vacuum relief flow path open, in parallel with the normal containment purge isolation valves without a negative pressure condition in the containment, would provide simultaneous air flow discharge paths. The valve interlock prevents both paths from being open simultaneously. The potential radioactivity release out through the larger containment purge system piping bounds the potential radioactivity release out of the smaller vacuum relief piping during the closure of the vacuum relief isolation valves.

Power supplies and control functions necessary for containment isolation are Class 1E, as described in Chapters 7 and 8.

6.2.4 Containment Hydrogen Control System

The containment hydrogen control system is provided to limit the hydrogen concentration in the containment so that containment integrity is not endangered.

Following a severe accident, it is assumed that 100 percent of the fuel cladding reacts with water. Although hydrogen production due to radiolysis and corrosion occurs, the cladding reaction with water dominates the production of hydrogen for this case. The hydrogen generation from the zirconium-steam reaction could be sufficiently rapid that it may not be possible to prevent the hydrogen concentration in the containment from exceeding the lower flammability limit. The function of the containment hydrogen control system for this case is to promote hydrogen burning soon after the lower flammability limit is reached in the containment. Initiation of hydrogen burning at the lower level of hydrogen flammability prevents accidental hydrogen burn initiation at high hydrogen concentration levels and thus provides confidence that containment integrity can be maintained during hydrogen burns and that safety-related equipment can continue to operate during and after the burns.

The containment hydrogen control system serves the following functions:

- Hydrogen concentration monitoring
- Hydrogen control during and following a degraded core or core melt scenarios (provided by hydrogen igniters). In addition, two nonsafety-related passive autocatalytic recombiners (PARs) are provided for defense-in-depth protection against the buildup of hydrogen following a loss of coolant accident.

6.2.4.1 Design Basis

A. The hydrogen control system is designed to provide containment atmosphere cleanup (hydrogen control) in accordance with General Design Criterion 41, 42 and 43.

B. The hydrogen control system is designed in accordance with the requirements of 10 CFR 50.44 and meets the NRC staff's position related to hydrogen control of SECY-93-087.

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C. The hydrogen control system is designed in compliance with the recommendations of NUREG 0737 and 0660 as detailed in Section 1.9.

D. The hydrogen control system is designed in accordance with the recommendations of Regulatory Guide 1.7 as discussed in Appendix 1A. The containment recirculation system discussed in Subsection 9.4.7 provides the controlled purge capability for the containment as specified in position C.4 of Regulatory Guide 1.7

E. The hydrogen control system is designed and fabricated to codes consistent with the quality group classification, described in Section 3.2. Conformance with Regulatory Guide 1.26, 1.29, and 1.32 is described in Section 1.9.

F. The hydrogen control system complies with the intent of Regulatory Guide 1.82 “The Water Sources For Long-Term Recirculation Cooling Following A Loss-Of-Coolant Accident” as it could be applied to concerns for blockage of recombiner air flow paths.

6.2.4.1.1 Containment Mixing

Containment structures are arranged to promote mixing via natural circulation. The physical mechanisms of natural circulation mixing that occur in the AP1000 are discussed in Appendix 6A and summarized below. For a postulated break low in the containment, buoyant flows develop through the lower compartments due to density head differences between the rising plume and the surrounding containment atmosphere, tending to drive mixing through lower compartments and into the region above the operating deck. There is also a degree of mixing within the region above the operating deck, which occurs due to the introduction of and the entrainment into the steam-rich plume as it rises from the operating deck openings. Thus, natural forces tend to mix the containment atmosphere.

Two general characteristics have been incorporated into the design of the AP1000 to promote mixing and eliminate dead-end compartments. The compartments below deck are large open volumes with relatively large interconnections, which promote mixing throughout the below deck region. All compartments below deck are provided with openings through the top of the compartment to eliminate the potential for a dead pocket of high-hydrogen concentration. In addition, if forced containment air-circulation is operated during post-accident recovery, then nonsafety-related fan coolers contribute to circulation in containment.

In the event of a hydrogen release to the containment, passive autocatalytic recombiners act to recombine hydrogen and oxygen on a catalytic surface (see Subsection 6.2.4.2.2). The enthalpy of reaction generates heat within a passive autocatalytic recombiner, which further drives containment mixing by natural circulation. Catalytic recombiners reduce hydrogen concentration at very low hydrogen concentrations (less than 1 percent) and very high steam concentrations, and may also promote convection to complement passive containment cooling system natural circulation currents to inhibit stratification of the containment atmosphere (Reference 17). The implementation of passive autocatalytic recombiners has a favorable impact on both containment mixing and hydrogen mitigation.

For events involving actuation of ADS Stages 1-3, hydrogen has the potential to accumulate in the IRWST airspace where concentrations will be at elevated levels for a limited timeframe and may exceed detonation limits. A detonation in the IRWST has a low probability of occurrence because there are no credible ignition sources in the IRWST airspace during the timeframe of the elevated hydrogen concentration. In the event a hydrogen detonation does occur within the IRWST airspace, the structure and safety-related components within the IRWST maintain their ability to perform their safety-related function. After ADS Stages 1-3 blowdown is complete, the containment hydrogen control system contains design features to address the remaining hydrogen, if needed.

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6.2.4.1.2 Validity of Hydrogen Monitoring

The hydrogen monitoring function monitors bulk containment hydrogen concentration.

6.2.4.1.3 Hydrogen Control for Severe Accident

The containment hydrogen concentration is limited by operation of the distributed hydrogen ignition subsystem. Ignition causes deflagration of hydrogen (burning of the hydrogen with flame front propagation at subsonic velocity) at hydrogen concentrations between the flammability limit and 10 volume percent and thus prevents the occurrence of hydrogen detonation (burning of hydrogen with supersonic flame front propagation).

6.2.4.2 System Design

6.2.4.2.1 Hydrogen Concentration Monitoring Subsystem

The hydrogen concentration monitoring subsystem consists of three hydrogen sensors, each paired with a pressure sensor. The sensors are placed in the upper dome where bulk hydrogen concentration can be monitored. The paired pressure sensors are used for pressure compensation for the hydrogen sensors.

The subsystem is comprised of three hydrogen concentration monitors, one corresponding to each pair of hydrogen and pressure sensors, designated as non-Class 1E serving to provide a post accident monitoring function. See Section 7.5 for additional information.

The hydrogen monitors are powered by the Non-Class 1E dc and UPS System. Monitor parameters are provided in Table 6.2.4-1. Hydrogen concentration is continuously indicated in the main control room. Additionally, high hydrogen concentration alarms are provided in the main control room.

6.2.4.2.2 Hydrogen Recombination Subsystem

The hydrogen recombination subsystem is designed to accommodate the hydrogen production rate anticipated for loss of coolant accident. The hydrogen recombination subsystem consists of two nonsafety-related passive autocatalytic recombiners installed inside the containment above the operating deck at approximate elevations of 170 feet and 181 feet respectively, each a minimum of 13 feet inboard from the containment shell. The locations provide placement within a homogeneously mixed region of containment as supported by Subsection 6.2.4.1.1 and Appendix 6A. The location is in a predominately upflow natural convection region. Additionally, the PARs are located azimuthally away from potential high upflow regions such as the direct plume above the loop compartment.

The passive autocatalytic recombiners are simple and passive in nature without moving parts and independent of the need for electrical power or any other support system. The recombiners require no power supply and are self-actuated by the presence of the reactants (hydrogen and oxygen).

Normally, oxygen and hydrogen recombine by rapid burning only at elevated temperatures (greater than about 1100°F [600°C]). However, in the presence of catalytic materials such as the palladium group, this “catalytic burning” occurs even at temperatures below 32°F (0°C). Adsorption of the oxygen and hydrogen molecules occurs on the surface of the catalytic metal because of attractive forces of the atoms or molecules on the catalyst surface. Passive autocatalytic recombiner devices use palladium or platinum as a catalyst to combine molecular hydrogen with oxygen gases into water vapor. The catalytic process can be summarized by the following steps (Reference 15):

1. Diffusion of the reactants (oxygen and hydrogen) to the catalyst

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2. Reaction of the catalyst (chemisorption)

3. Reaction of intermediates to give the product (water vapor)

4. Desorption of the product

5. Diffusion of the product away from the catalyst

The reactants must get to the catalyst before they can react and subsequently the product must move away from the catalyst before more reactants will be able to react.

The passive autocatalytic recombiner device consists of a stainless steel enclosure providing both the structure for the device and support for the catalyst material. The enclosure is open on the bottom and top and extends above the catalyst elevation to provide a chimney to yield additional lift to enhance the efficiency and ventilation capability of the device. The catalyst material is either constrained within screen cartridges or deposited on a metal plate substrate material and supported within the enclosure. The spaces between the cartridges or plates serve as ventilation channels for the throughflow. During operation, the air inside the recombiner is heated by the recombination process, causing it to rise by natural convection. As it rises, replacement air is drawn into the recombiner through the bottom of the passive autocatalytic recombiner and heated by the exothermic reaction, forming water vapor, and exhausted through the chimney where the hot gases mix with containment atmosphere. The device is a molecular diffusion filter and thus the open flow channels are not susceptible to fouling.

Passive autocatalytic recombiners begin the recombination of hydrogen and oxygen almost immediately upon exposure to these gases when the catalyst is not wetted. If the catalyst material is wet, then a short delay is experienced in passive autocatalytic recombiner startup (References 19 and 29). The delay is short with respect to the time that the PARs have to control hydrogen accumulation rates (days to weeks) following a design basis accident. The recombination process occurs at room or elevated temperature during the early period of accidents prior to the buildup of flammable gas concentrations. Passive autocatalytic recombiners are effective over a wide range of ambient temperatures, concentrations of reactants (rich and lean, oxygen/hydrogen less than 1 percent) and steam inerting (steam concentrations greater than 50 percent). Although the passive autocatalytic recombiner depletion rate reaches peak efficiency within a short period of time, the rate varies with hydrogen concentration and containment pressure, (Reference 19).

Passive autocatalytic recombiners have been shown to be effective at minimizing the buildup of hydrogen inside containment following loss of coolant accidents (Reference 16). They are provided in the AP1000 as defense-in-depth protection against the buildup of hydrogen following a loss of coolant accident. A summary of component data for the hydrogen recombiners is provided in Table 6.2.4-2.

6.2.4.2.3 Hydrogen Ignition Subsystem

The hydrogen ignition subsystem is provided to address the possibility of an event that results in a rapid production of large amounts of hydrogen such that the rate of production exceeds the capacity of the recombiners. Consequently, the containment hydrogen concentration will exceed the flammability limits. This massive hydrogen production is postulated to occur as the result of a degraded core or core melt accident (severe accident scenario) in which up to 100 percent of the zirconium fuel cladding reacts with steam to produce hydrogen.

The hydrogen ignition subsystem consists of 66 hydrogen igniters strategically distributed throughout the containment. Since the igniters are incorporated in the design to address a low-probability severe accident, the hydrogen ignition system is not Class 1E. Although not class 1E, the igniter coverage,

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distribution and power supply has been designed to minimize the potential loss of igniter protection globally for containment and locally for individual compartments. The igniters have been divided into two power groups. Power to each group will be normally provided by offsite power, however should offsite power be unavailable, then each of the power groups is powered by one of the onsite non-essential diesels and finally should the diesels fail to provide power then approximately 4 hours of igniter operation is supported by the non-Class 1E batteries for each group through the use of appropriate load shedding. Assignment of igniters to each group is based on providing coverage for each compartment or area by at least one igniter from each group.

The locations of the igniters are based on evaluation of hydrogen transport in the containment and the hydrogen combustion characteristics. Locations include compartmented areas in the containment and various locations throughout the free volume, including the upper dome.

For enclosed areas of the containment at least two igniters are installed. The separation between igniter locations is selected to prevent the velocity of a flame front initiated by one igniter from becoming significant before being extinguished by a similar flame front propagating from another igniter. The number of hydrogen igniters and their locations are selected considering the behavior of hydrogen in the containment during severe accidents. The likely hydrogen transport paths in the containment and hydrogen burn physics are the two important aspects influencing the choice of igniter location.

The primary objective of installing an igniter system is to promote hydrogen burning at a low concentration and, to the extent possible, to burn hydrogen more or less continuously so that the hydrogen concentration does not build up in the containment. To achieve this goal, igniters are placed in the major regions of the containment where hydrogen may be released, through which it may flow, or where it may accumulate. The criteria utilized in the evaluation and the application of the criteria to specific compartments is provided in Table 6.2.4-6. The location of igniters throughout containment is provided in Figures 6.2.4-5 through 6.2.4-13. The location of igniters is also summarized in Table 6.2.4-7 identifying subcompartment/regions and which igniters by power group provide protection. The locations identified are considered approximations (± 2.5 feet) with the final locations governed by the installation details.

The igniter assembly is designed to reach a minimum surface temperature of 1700°F in the anticipated containment environment following a loss of coolant accident. A spray shield is provided to protect the igniter from falling water drops (resulting from condensation of steam on the containment shell and on nearby equipment and structures). Design parameters for the igniters are provided in Table 6.2.4-3.

6.2.4.2.4 Containment Purge

Containment purge is not part of the containment hydrogen control system. The purge capability of the containment air filtration system (see Subsection 9.4.7) can be used to provide containment venting prior to post-loss of coolant accident cleanup operations.

6.2.4.3 Design Evaluation (Design Basis Accident)

A design basis accident evaluation is not required.

6.2.4.4 Design Evaluation (Severe Accident)

Although a severe accident involving major core degradation or core melt is not a design basis accident, the containment hydrogen control system contains design features to address this potential occurrence. The hydrogen monitoring subsystem has sufficient range to monitor concentrations up to 20 percent hydrogen. The hydrogen ignition subsystem is provided so that hydrogen is burned off in

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a controlled manner, preventing the possibility of deflagration with supersonic flame front propagation which could result in large pressure spikes in the containment.

It is assumed that 100 percent of the active fuel cladding zirconium reacts with steam. This reaction may take several hours to complete. The igniters initiate hydrogen burns at concentrations less than 10 percent by volume and prevent the containment hydrogen concentration from exceeding this limit. Further evaluation of hydrogen control by the igniters is presented in the AP1000 Probabilistic Risk Assessment.

6.2.4.5 Tests and Inspections

6.2.4.5.1 Preoperational Inspection and Testing

Hydrogen Monitoring Subsystem

Pre-operational testing is performed either before or after installation but prior to plant startup to verify performance.

Hydrogen Recombination Subsystem

The performance of the autocatalytic recombiner plates (or cartridges) is tested by the manufacturer for each lot or batch of catalyst material. The number of plates tested is based on the guidance provided in ANSI/ASQC Z1.4-1993, “Sampling Procedures and Tables for Inspection by Attributes,” (formerly Military Standard 105), required to achieve Inspection Level III quality level.

Hydrogen Ignition Subsystem

Pre-operational testing and inspection is performed after installation of the hydrogen ignition system and prior to plant startup to verify operability of the hydrogen igniters. It is verified that 66 igniter assemblies are installed at the locations defined by Figures 6.2.4-5 through 6.2.4-11. Operability of the igniters is confirmed by verification of the surface temperature in excess of the value specified in Table 6.2.4-3. This temperature is sufficient to ensure ignition of hydrogen concentrations above the flammability limit.

Pre-operational inspection is performed to verify the location of openings through the ceilings of the passive core cooling system valve/accumulator rooms with respect to the containment pressure boundary. The primary openings are those that constitute at least 98% of the opening area. The primary openings in Room 11206 that vent to Room 11300 are the equipment access opening and CMT-A opening. These openings are verified to be a minimum distance of 24.3 feet and 9.4 feet, respectively, from the containment shell. The primary opening in Room 11207 that vents to Room 11300 is the CMT-B opening, which is verified to be a minimum distance of 24.6 feet from the containment shell. Other openings must be at least 3 feet from the containment shell.

Pre-operational inspection is performed to verify the orientation of the vents from the IRWST that are located along the side of the IRWST next to the containment. The discharge of each of these IRWST vents must be oriented generally away from the containment shell.

6.2.4.5.2 In-service Testing

Hydrogen Monitoring Subsystem

The system is normally in service. Periodic testing and calibration are performed to provide ongoing confirmation that the hydrogen monitoring function can be reliably performed.

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Hydrogen Recombination Subsystem

Periodic inspection and testing are performed on the passive autocatalytic recombiners. The testing is performed by testing a sample of the catalyst plates as specified in Subsection 6.2.4.5.1.

Hydrogen Ignition Subsystem

Periodic inspection and testing are performed to confirm the continued operability of the hydrogen ignition system. Operability testing consists of energizing the igniters and confirming the surface temperature as specified in Table 6.2.4-3.

6.2.4.6 Combined License Information

This section contained no requirement for additional information.

6.2.5 Containment Leak Rate Test System

The reactor containment, containment penetrations and isolation barriers are designed to permit periodic leak rate testing in accordance with General Design Criteria 52, 53, and 54. The containment leak rate test system is designed to verify that leakage from the containment remains within limits established in the Technical Specifications.

6.2.5.1 Design Basis

Leak rate testing requirements are defined by 10 CFR 50 Appendix J, “Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors,” (Reference 14) which classifies leak tests as Types A, B and C. The system design provides testing capability consistent with the testing requirements of ANSI-56.8 (Reference 13). The system design accommodates the test methods and frequencies consistent with requirements of 10 CFR 50 Appendix J, Option A or Option B.

The Containment Leak Rate Test Program using 10 CFR Part 50, Appendix J Option B is established in accordance with NEI 94-01 (Subsection 6.2.7, Reference 30), as modified and endorsed by the NRC in Regulatory Guide 1.163. Table 13.4-201 provides milestones for containment leak rate testing implementation.

6.2.5.1.1 Safety Design Basis

The containment leak rate test system serves no safety-related function other than containment isolation, and therefore has no nuclear safety design basis except for containment isolation. See Subsection 6.2.3 for the containment isolation system.

6.2.5.1.2 Power Generation Design Basis

The containment leak rate test system is designed to verify the leak tightness of the reactor containment. The specified maximum allowable containment leak rate is 0.10 weight percent of the containment air mass per day at the calculated peak accident pressure, P_a, identified in Subsection 6.2.1. The system is specifically designed to perform the following tests in accordance with the provisions of ANSI-56.8 (Reference 13):

- Containment integrated leak rate testing (Type A): The containment is pressurized with clean, dry air to a pressure of P_a. Measurements of containment pressure, dry bulb temperature, and dew point temperature are used to determine the decrease in the mass of air in the containment over time, and thus establish the leak rate.

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- Local leak rate testing of containment penetrations with a design that incorporates features such as resilient seals, gaskets, and expansion bellows (Type B): The leakage limiting boundary is pressurized with air or nitrogen to a pressure of P_a and the pressure decay or the leak flow rate is measured.

- Local leak rate testing of containment isolation valves (Type C): The piping test volume is pressurized with air or nitrogen to a pressure of P_a and pressure decay or the leak flow rate is measured. For valves sealed with a fluid such as water, the test volume is pressurized with the seal fluid to a pressure of not less than 1.1 P_a.

The containment leak rate test system piping is also designed for use during the performance of the containment structural integrity test. The instrumentation used for the structural integrity test may be different than that used for the integrated leak rate test.

6.2.5.1.3 Codes and Standards

The containment leak rate test system is designed to conform to the applicable codes and standards listed in Section 3.2. The containment leak testing program satisfies 10 CFR 50, Appendix J requirements.

6.2.5.2 System Description

6.2.5.2.1 General Description

The containment leak rate test system is illustrated on Figure 6.2.5-1. Unless otherwise indicated on the figure, piping and instrumentation is permanently installed. Fixed test connections used for Type C testing of piping penetrations are not shown on Figure 6.2.5-1. These connections are not part of the containment leak rate test system and are shown on the applicable system piping and instrument diagram figure.

Air compressor assemblies used for Type A testing are temporarily installed and are connected to the permanent system piping. The number and capacity of the compressors is sufficient to pressurize the containment with air to a pressure of P_a at a maximum containment pressurization rate of about 5 psi/hour. The compressor assemblies include additional equipment, such as air coolers, moisture separators and air dryers to reduce the moisture content of the air entering containment.

Temperature and humidity sensors are installed inside containment for Type A testing. Data acquisition hardware and instrumentation is available outside containment. Instrumentation not required during normal plant operation may be installed temporarily for the Type A tests.

The system is designed to permit depressurization of the containment at a maximum rate of 10 psi/hour.

Portable leak rate test panels are used to perform Type C containment isolation valve leak testing using air or nitrogen. The panels are also used for Type B testing of penetrations, for which there is no permanently installed test equipment. The panels include pressure regulators, filters, pressure gauges and flow instrumentation, as required to perform specific tests.

6.2.5.2.2 System Operation

Containment Integrated Leak Rate Test (Type A)

An integrated leak rate test of the primary reactor containment is performed prior to initial plant operation, and periodically thereafter, to confirm that the total leakage from the containment does not

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exceed the maximum allowable leak rate. The allowable leak rate specified in the test criteria is less than the maximum allowable containment leak rate, in accordance with 10 CFR 50, Appendix J.

Following construction of the containment and satisfactory completion of the structural integrity test, described in Subsection 3.8.2.7, a preoperational Type A test is performed as described in Chapter 14. Additional Type A tests are conducted during the plant life, at intervals in accordance with the Technical Specifications.

- Pretest Requirements

Prior to performing an integrated leak rate test, a number of pretest requirements must be satisfied as described in this subsection.

A general inspection of the accessible interior and exterior surfaces of the primary containment structure and components is performed to uncover any evidence of structural deterioration that could affect either the containment structural integrity or leak tightness. If there is evidence of structural deterioration, corrective action is taken prior to performing the Type A test. The structural deterioration and corrective action are reported in accordance with 10 CFR 50, Appendix J. Except as described above, during the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments are made so that the containment can be tested in as close to the “as-is” condition as practical.

Containment isolation valves are placed in their post-accident positions, identified in Table 6.2.3-1, unless such positioning is impractical or unsafe. Test exceptions to post-accident valve positioning are identified in Table 6.2.3-1 or are discussed in the test report. Closure of containment isolation valves is accomplished by normal operation and with no preliminary exercising or adjustments (such as tightening of a valve by manual handwheel after closure by the power actuator). Valve closure malfunctions or valve leakage that requires corrective action before the test is reported in conjunction with the Type A test report.

Those portions of fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment, are opened or vented to the containment atmosphere prior to and during the test.

Portions of systems inside containment that penetrate containment and could rupture as a result of a loss of coolant accident are vented to the containment atmosphere and drained of water to the extent necessary to provide exposure of the containment isolation valves to containment air test pressure and to allow them to be subjected to the full differential test pressure, except that:

- Systems that are required to maintain the plant in a safe condition during the Type A test remain operable and are not vented.

- Systems that are required to establish and maintain equilibrium containment conditions during Type A testing remain operable and are not vented.

- Systems that are normally filled with water and operating under post-accident conditions are not vented.

Systems not required to be vented and drained for Type A testing are identified in Table 6.2.3-1. The leak rates for the containment isolation valves in these systems, measured by Type C testing, are reported in the Type A test report.

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Tanks inside the containment are vented to the containment atmosphere as necessary to protect them from the effects of external test pressure and/or to preclude leakage which could affect the accuracy of the test results. Similarly, instrumentation and other components that could be adversely affected by the test pressure are vented or removed from containment.

The containment atmospheric conditions are allowed to stabilize prior to the start of the Type A test consistent with the guidance of ANSI-56.8. The containment recirculation cooling system and central chilled water system are operated as necessary prior to, and during, the test to maintain stable test conditions.

- Test Method

The Type A test is conducted in accordance with ANSI-56.8, using the absolute method. The test duration is established consistent with ANSI-56.8 following the stabilization period. Periodic measurements of containment pressure, dry bulb temperatures and dew point temperatures (water vapor pressure) are used to determine the decrease in the mass of air in the containment over time. A standard statistical analysis of the data is conducted consistent with recommendations of ANSI-56.8.

The accuracy of the Type A test results is then verified by a supplemental verification test. The supplemental verification test is performed using methodology consistent with the recommendations described in ANSI-56.8.

Test criteria for the Type A test are given in the technical specifications. If any Type A test fails to meet the criteria, the test schedule for subsequent tests is adjusted in accordance with 10 CFR 50, Appendix J as defined in the Containment Leakage Rate Testing Program.

During the period between the completion of one Type A test and the initiation of the containment inspection for the subsequent Type A test, repairs or adjustments are made to components identified as exceeding individual leakage limits, as soon as practical after such leakage is identified.

Containment Penetration Leak Rate Tests (Type B)

The following containment penetrations receive preoperational and periodic Type B leak rate tests in accordance with ANSI-56.8 with test intervals as defined by NEI 94-01 (Reference 30):

- Penetrations whose design incorporates resilient seals, gaskets or sealant compounds
- Air locks and associated door seals
- Equipment and access hatches and associated seals
- Electrical penetrations
- Expansion bellows for main steam and feedwater piping penetrations

Figure 6.2.5-1 provides the piping and instrumentation diagram for the containment leak rate test system and illustrates examples of containment penetrations subject to Type B tests.

The fuel transfer tube penetration is sealed with a blind flange inside containment. The flanged joint is fitted with testable seals as shown in Figure 3.8.2-4. The two expansion bellows used on the fuel transfer tube penetration are not part of the leakage-limiting boundary of the containment.

The personnel hatches (airlocks) are designed to be tested by internal pressurization. The doors of the personnel hatches have testable seals as shown in Figure 3.8.2-3. Mechanical and electrical

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penetrations on the personnel hatches are also equipped with testable seals. The hatch cover flanges for the main equipment and maintenance hatches have testable seals as shown in Figure 3.8.2-2. Containment electrical penetrations have testable seals as shown in Figure 3.8.2-4.

Type B leak tests are performed by local pressurization using the test connections shown on Figure 6.2.5-1. Unless otherwise noted in Table 6.2.3-1, the test pressure is not less than the calculated containment peak accident pressure, P_a. Either the pressure decay or the flowmeter test method is used. These test methods and the test criteria are presented below for Type C tests.

Containment Isolation Valve Leak Rate Tests (Type C)

Containment isolation valves receive preoperational and periodic Type C leak rate tests in accordance with ANSI-56.8 with test intervals as defined by NEI 94-01 (Reference 30). A list of containment isolation valves subject to Type C tests is provided in Table 6.2.3-1. Containment isolation valve arrangement and test connections provided for Type C testing are illustrated on the applicable system piping and instrument diagram figure.

Type C leak tests are performed by local pressurization. Each valve to be tested is closed by normal means without any preliminary exercising or adjustments. Piping is drained and vented as needed and a test volume is established that, when pressurized, will produce a differential pressure across the valve. Table 6.2.3-1 identifies the direction in which the differential pressure is applied.

Isolation valves whose seats may be exposed to the containment atmosphere subsequent to a loss of coolant accident are tested with air or nitrogen at a pressure not less than P_a. Valves in lines which are designed to be, or remain, filled with a liquid for at least 30 days subsequent to a loss of coolant accident are leak rate tested with that liquid at a pressure not less than 1.1 times P_a. Isolation valves tested with liquid are identified in Table 6.2.3-1.

Isolation valves are tested using either the pressure decay or flowmeter method. For the pressure decay method the test volume is pressurized with air or nitrogen. The rate of decay of pressure in the known volume is monitored to calculate the leak rate. For the flowmeter method pressure is maintained in the test volume by supplying air or nitrogen through a calibrated flowmeter. The measured makeup flow rate is the isolation valve leak rate.

The leak rates of penetrations and valves subject to Type B and C testing are combined in accordance with 10 CFR 50, Appendix J. As each Type B or C test, or group of tests, is completed the combined total leak rate is revised to reflect the latest results. Thus, a reliable summary of containment leaktightness is maintained current. Leak rate limits and the criteria for the combined leakage results are described in the technical specifications.

Scheduling and Reporting of Periodic Tests

Schedules for the performance of periodic Type A, B, and C leak rate tests are in accordance with the Technical Specifications as specified in the Containment Leakage Rate Testing Program. Provisions for reporting test results are described in the Containment Leakage Rate Testing Program.

Type B and C tests may be conducted at any time that plant conditions permit, provided that the time between tests for any individual penetration or valve does not exceed the maximum allowable interval specified in the Containment Leakage Rate Testing Program.

Schedules for the performance of periodic Type A, B, and C leak rate tests are in accordance with NEI 94-01, as endorsed and modified by Regulatory Guide 1.163, and described below:

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Type A Tests

A preoperational Type A test is conducted prior to initial fuel load. If initial fuel load is delayed longer than 36 months after completion of the preoperational Type A test, a second preoperational Type A test shall be performed prior to initial fuel load. The first periodic Type A test is performed within 48 months after the successful completion of the last preoperational Type A test. Periodic Type A tests are performed at a frequency of at least once per 48 months, until acceptable performance is established. The interval for testing begins at initial reactor operation. Each test interval begins upon completion of a Type A test and ends at the start of the next test. The extension of the Type A test interval is determined in accordance with NEI 94-01.

Type A testing is performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as successful completion of two consecutive Type A tests where the calculated performance leakage rate was less than 1.0 L_a. A preoperational Type A test may be used as one of the two Type A tests that must be successfully completed to extend the test interval, provided that an engineering analysis is performed to document why a preoperational Type A test can be treated as a periodic test. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Type B Tests (Except Containment Airlocks)

Type B tests are performed prior to initial entry into Mode 4. Subsequent periodic Type B tests are performed at a frequency of at least once per 30 months, until acceptable performance is established. The test intervals for Type B penetrations may be increased based upon completion of two consecutive periodic as-found Type B tests where results of each test are within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the component prior to implementing Option B of 10 CFR Part 50, Appendix J. An extended test interval for Type B tests may be increased to a specific value in a range of frequencies from greater than once per 30 months up to a maximum of once per 120 months. The extension of specific test intervals for Type B penetrations is determined in accordance with NEI 94-01.

Type B Tests (Containment Airlocks)

Containment airlock(s) are tested at an internal pressure of not less than P_{ac}. (Prior to a preoperational Type A test P_{ac}= P_a.) Subsequent periodic tests are performed at a frequency of at least once per 30 months. In addition, equalizing valves, door seals, and penetrations with resilient seals (i.e., shaft seals, electrical penetrations, view port seals and other similar penetrations) that are testable, are tested at a frequency of once per 30 months.

For periods of multiple containment entries where the airlock doors are routinely used for access more frequently than once every seven days (e.g., shift or daily inspection tours of the containment), door seals may be tested once per 30 days during this time period.

Airlock door seals are tested prior to a preoperational Type A test. When containment integrity is required, airlock door seals are tested within seven days after each containment access.

Type C Tests

Type C tests are performed prior to initial entry into Mode 4. Subsequent periodic Type C tests are performed at a frequency of at least once per 30 months, until adequate performance has been established. Test intervals for Type C valves may be increased based upon completion of two consecutive periodic as-found Type C tests where the result of each test is within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive passing

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tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the valve prior to implementing Option B of 10 CFR Part 50, Appendix J. Intervals for Type C testing may be increased to a specific value in a range of frequencies from 30 months up to a maximum of 60 months. Test interval extensions for Type C valves are determined in accordance with NEI 94-01.

Reporting

A post-outage report is prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The report is available on-site for NRC review. The report shows that the applicable performance criteria are met, and serves as a record that continuing performance is acceptable.

Special Testing Requirements

AP1000 does not have a subatmospheric containment or a secondary containment. There are no containment isolation valves which rely on a fluid seal system. Thus, there are no special testing requirements.

Acceptance Criteria

Acceptance criteria for Type A, B and C Tests are established in Technical Specification 5.5.8.

6.2.5.2.3 Component Description

The system pressurization equipment is temporarily installed for Type A testing. In addition to one or more compressors, this hardware includes components such as aftercoolers, moisture separators, filters and air dryers. The hardware characteristics may vary from test to test.

The flow control valve in the pressurization line is a leaktight valve capable of throttling to a low flow rate.

6.2.5.2.4 Instrumentation Applications

For Type A testing, instruments are provided to measure containment absolute pressure, dry bulb temperature, dew point temperature, air flow rate, and atmospheric pressure. Data acquisition equipment scans, processes and records data from the individual sensors. For Type B and C testing, instruments are provided to measure pressure, dry bulb temperature, and flow rate.

The quantity and location of Type A instrumentation and permanently installed Type B instrumentation, is indicated on Figure 6.2.5-1. The type, make and range of test instruments may vary from test to test. The instrument accuracy must meet the criteria of Reference 13.

6.2.5.3 Safety Evaluation

The containment leak rate test system has no safety-related function, other than containment isolation and therefore requires no nuclear safety evaluation, other than containment isolation which is described in Subsection 6.2.3.

6.2.5.4 Inservice Inspection/Inservice Testing

There are no special inspection or testing requirements for the containment leak rate test system. Test equipment is inspected and instruments are calibrated in accordance with ANSI-56.8 criteria and the requirements of the test procedure.

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6.2.6 Combined License Information for Containment Leak Rate Testing

The Containment Leakage Rate Testing Program is addressed in Subsections 6.2.5.1 and 6.2.5.2.2.

6.2.7 References

1. Not used.

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3. Shepard, R. M., et al., "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, June 1975 (Proprietary), and WCAP-8312-A, Revision 2, August 1975 (Non-Proprietary).

4. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," WCAP-10325-P-A (Proprietary) and WCAP-10326-A (Non-Proprietary), May 1983.

5. Land, R. E., "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822 (Proprietary) and WCAP-8860 (Non-Proprietary), September 1976; "Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture," WCAP-8822-P-S1 (Proprietary), January 1985; "Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Subatmospheric Containment Designs," WCAP-8822-S2-P-A (Proprietary), September 1986.

6. Burnett, T. W. T., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), April 1984.

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8. Not used.

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14. 10 CFR 50, Appendix J, "Containment Leak Rate Testing," September 26, 1995.

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16. AP600 Standard Safety Analysis Report, Section 6.2.

17. J. Rohde, et al., “Hydrogen Mitigation by Catalytic Recombiners and Ignition During Severe Accidents,” Third International Conference on Containment Design and Operation, Canadian Nuclear Society, Toronto, Ontario, October 19-21, 1994.

18. Not used.

19. EPRI Report, “NIS passive autocatalytic recombiner Depletion Rate Equation for Evaluation of Hydrogen Recombination During AP600 Design Basis Accident,” EPRI ALWR Program, November 15, 1995.

20. WCAP-15846 (Proprietary) and WCAP-15862 (Non-Proprietary) “WGOTHIC Application to AP600 and AP1000,” Revision 5, September 2016.

21. Not used.

22. NUREG-737, “Clarification of TMI Action Plan Requirements,” October, 1980

23. NUREG-718. Rev. 2, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License,” January, 1982.

24. ANSI/ASCE-8-90, Specification for the Design of Cold Formed Stainless Steel Structural Members

25. Not used.

26. WCAP-15965-P (Proprietary) and WCAP-15965-NP (Non-Proprietary), “AP1000 Subcompartment Models,” Revision 1, September 2014.

27. Not used.

28. Not used.

29. EPRI Report TR-107517, Volumes 1, 2, and 3, “Generic Model Tests of Passive Autocatalytic Recombiners (PARs) for Combustible Gas Control in Nuclear Power Plants,” June 1997.

30. Nuclear Energy Institute Report, NEI 94-01, “Industry Guidelines for Implementing Performance Based Option of 10 CFR 50, Appendix J,” Revision 0.

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33. NEI 06-12, “B.5.b Phase 2 & 3 Submittal Guidelines,” Revision 2.

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38. APP-PCS-M3C-033, “AP1000 Passive Containment Cooling System (PCS) Air Flow Path Resistance Analysis by Using Computational Fluid Dynamics (CFD).”

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Table 6.2.1.1-1
Summary of Calculated Peak Containment Pressures and Vapor Temperatures

Break	Peak Pressure (psig)	Available[1] Margin (psi)	Peak Temperature (°F)
Double-ended hot leg guillotine	49.8	9.2	385.8
Double-ended cold leg guillotine	58.1	0.9	357.8
Full main steam line DER, 30% power, MSIV failure	57.2	1.8	381.8
Full main steam line DER, 101% power, MSIV failure	54.4	4.6	383.6

Note:
1. Design Pressure is 59 psig

Table 6.2.1.1-2
Containment Integrity Analysis Initial Conditions

Internal Temperature (°F)	120
Pressure (psia)	15.7
Internal Relative Humidity (%)	0
Net Free Volume (ft^3)	2.06E+06
External Temperature (°F)	115 dry bulb 86.1 wet bulb

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Table 6.2.1.1-3
Postulated Containment Accident Analyses Relative to GDC

Criterion	Acceptance Criterion Value	Lumped DEHLG LOCA Value	Lumped DECLG LOCA Value	30% Power MSLB Value	External Pressurization Value
GDC 16 & GDC 50 Design Pressure	<59.0 psig	49.8 psig	58.1 psig	57.2 psig	
GDC 38 Rapidly Reduce Containment Pressure	< 29.5 psig		< 27 psig after 24 hrs(1)		
GDC 38 & 50 External Pressure	< 1.7 psid				1.63
GDC 38 & GDC 50 Containment Heat Removal Single Failure	Most Severe	Two of Three PCS Water Supply Flowpaths(2)	Two of Three PCS Water Supply Flowpaths(2)	Two of Three PCS Supply Flowpaths(2)	

Notes:
1. Analysis without crediting heat conduction from dry to wet sections on containment shell.
2. Table 6.2.2-3 notes that operation of only one of three flowpaths is necessary to meet minimum containment cooling requirements.

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Tables 6.2.1.1-4–6.2.1.1-7 Not Used

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Table 6.2.1.1-8
Material Properties Used in the WGOTHIC Containment Evaluation Model

Material		Thermal Conductivity (BTU/hr-ft-°F)	Volumetric Heat Capacity (BTU/ft^3-°F)	Emissivity
Epoxy		0.1875	26.25	N/A
Inorganic zinc		0.302	22.825	0.54(1)
Baffle aluminum		13.4	28.443	0.81(1)
Concrete		0.83	26.6	0.81(1)
Air	0°F	0.0131	0.020736	N/A
	250°F	0.0192	0.020988	N/A
	500°F	0.0246	0.021427	N/A
Group A Carbon Steel	70°F	31.4	44.9	N/A
	100°F	31.2	46.2	N/A
	150°F	30.8	48.0	N/A
	200°F	30.3	49.6	N/A
	250°F	29.7	50.8	N/A
	300°F	29.1	51.9	N/A
Group B Carbon Steel	70°F	24.6	46.4	N/A
	100°F	24.8	47.8	N/A
	150°F	25.0	49.6	N/A
	200°F	25.0	51.4	N/A
	250°F	24.8	52.7	N/A
	300°F	24.6	54.0	N/A
Group C Carbon Steel	70°F	21.3	46.5	N/A
	100°F	21.2	47.1	N/A
	150°F	21.2	48.4	N/A
	200°F	21.2	49.9	N/A
	250°F	21.1	51.1	N/A
	300°F	21.1	52.5	N/A
Group D Carbon Steel	70°F	18.9	46.3	N/A
	100°F	18.9	47.1	N/A
	150°F	19.1	48.7	N/A
	200°F	19.2	49.9	N/A
	250°F	19.3	51.1	N/A
	300°F	19.4	52.2	N/A
Group J Stainless Steel	70°F	7.7	51.3	N/A
	100°F	7.8	51.5	N/A
	150°F	8.1	52.6	N/A
	200°F	8.4	53.7	N/A
	250°F	8.6	54.7	N/A
	300°F	8.8	55.1	N/A
Group K Stainless Steel	70°F	7.5	53.1	N/A
	100°F	7.5	53.4	N/A
	150°F	7.7	54.5	N/A
	200°F	7.9	54.6	N/A
	250°F	8.2	55.7	N/A
	300°F	8.4	55.8	N/A

Notes:
1. Emissivity is only credited outside containment

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Table 6.2.1.1-9
Containment External Pressure Analysis Major Assumptions

Initial	
Parameter	**Value**
Containment External Temperature	25°F
Containment Wind Speed	Natural convection
Containment Internal Temperature	120°F
Containment Initial Humidity	72%
IRWST Temperature	120°F
Containment Internal Pressure	14.5 psia
Transient Conditions	
Containment External Temperature	Decreasing at 30°F/hr
Containment Wind Speed	Forced convection at 24.9 ft/s in the riser region
Containment Heat Rate	Sensible heat from Mode 3 with 0 decay heat considered
Safety Analysis Limit Assumed for Vacuum Relief System Actuation	-1.2 psig

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Table 6.2.1.1-10
[*Plant Requirements for Metal Heat Sinks Inside Containment*]*

Region	*Metal Volume (ft^3) (minimum – maximum)*
Above the operating deck	*4973 – 7279*
Inside SG/Pressurizer compartments	*1002 – 1677*
Below the operating deck	*5246 – 8255*
Total	*11,811(3) – 17,211*

Notes:
1. *Does not include the containment vessel.*
2. *Only includes structures and equipment that are typically less than or equal to containment ambient temperature.*
3. *The total minimum volume includes an additional 590 ft^3 of metal inside containment.*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 6.2.1.1-11 (Sheet 1 of 2)
Heat Sinks Credited in the Peak Containment Pressure Analyses

Description	Material	Surface Area (ft^2)	Metal Volume (ft^3)	Metal Effective Thickness (in)
Above the Operating Deck				
Containment vessel above the operating deck	Carbon Steel C	62,988	8,986	1.625 and 1.750
Polar crane bridge and trolley	Carbon Steel B	9,000	1,936	2.581
Crane girder	Carbon Steel C	11,509	1,190	1.241
Circular walkway at 162’	Carbon Steel B	11,903	349	0.351
Upper manway platforms at 166’ above SG rooms and ADS platform at 166’ and 176.88’ over pressurizer compartment	Carbon Steel B	13,060	238	0.219
Internal stiffener	Carbon Steel C	2,320	232	1.198
Integrated head stands	Carbon Steel B	692	170	2.940
Recirculation unit platforms at 153’ (CH53, CH57)	Carbon Steel B	9,056	176	0.242
Outer SG compartment walls (loop B)	Carbon Steel B, Concrete	1,808	139	0.922
Outer SG compartment walls (loop A)	Carbon Steel B, Concrete	1,572	105	0.800
Outer pressurizer walls	Carbon Steel B, Concrete	1,117	77	0.832
Stairs outside SG compartments	Carbon Steel B	4,394	74	0.201
Attachment plates above the operating deck	Carbon Steel C	678	62	1.090
Ring duct	Carbon Steel A	9,024	117	0.155
Recirculation duct and supports	Carbon Steel A	3,823	53	0.166
ADS module	Carbon Steel C	1,856	58	0.373
Inside SG Compartments and Pressurizer Compartment				
Inner SG room walls (loop B)	Carbon Steel B, Concrete	6,838	364	0.639
Inner SG room walls (loop A)	Carbon Steel B, Concrete	5,738	314	0.656
Inner pressurizer compartment walls	Carbon Steel B, Concrete	2,958	150	0.608
Lower manway platforms at 104’-7” in SG compartments	Carbon Steel B	3,631	113	0.375
Tubesheet platforms at 116’-6” in SG compartments	Carbon Steel B	3,979	33	0.100
Steel plate at 95’-6” in SG compartment (loop B)	Stainless Steel K	239	27	1.370
Flow-Through Compartments Below the Operating Deck				
Containment vessel below operating deck	Carbon Steel C	10,381	1,514	1.750
CMTs	Carbon Steel C	2,109	1,187	6.754
Floor framing below operating deck	Carbon Steel A	9,236	495	0.643
Platform at 118’-6” between maintenance floor and maintenance floor mezzanine	Carbon Steel B	27,703	439	0.190
Columns in maintenance floor room	Carbon Steel B	2,255	330	1.754
Curved IRWST wall	Stainless Steel K	4,110	219	0.639
Walls in maintenance floor and maintenance floor mezzanine	Carbon Steel B, Concrete	9,029	372	0.493

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Table 6.2.1.1-11 (Sheet 2 of 2)
Heat Sinks Credited in the Peak Containment Pressure Analyses

Description	Material	Surface Area (ft^2)	Metal Volume (ft^3)	Metal Effective Thickness (in)
Walls in reactor coolant drain tank (RCDT) room and vertical access (VA) room	Carbon Steel B, Concrete	2,966	133	0.538
Platforms at 83’ and 107’, and stairs in VA room	Carbon Steel B	3,131	37	0.140
Ceiling of maintenance floor mezzanine, Q-deck	Carbon Steel B, Concrete	5,552	28	0.060
Ceiling of maintenance floor mezzanine, steel plate (south)	Carbon Steel A, Concrete	514	13	0.293
Ceiling of pressurizer spray valve room	Carbon Steel B, Concrete	199	8	0.500
Dead-end Compartments Below the Operating Deck				
Accumulators	Carbon Steel C	1,616	264	1.959
PXS compartments walls	Carbon Steel B, Concrete	5,492	261	0.569
Upper refueling room walls	Stainless Steel K, Concrete	5,191	218	0.503
Platforms in PXS compartments	Carbon Steel B	6,404	158	0.296
CVS compartment walls	Carbon Steel B, Concrete	2,869	120	0.500
Demineralizers, reactor coolant filters in CVS compartment	Stainless Steel J	364	109	3.599
IRWST ceiling	Stainless Steel K, Concrete	2,174	87	0.479
PXS compartments ceiling	Carbon Steel A, Concrete	1,668	70	0.500
Lower refueling room walls	Stainless Steel K, Concrete	1,164	66	0.681
CVS compartment ceiling	Carbon Steel A, Concrete	679	28	0.500
Hatches over PXS, CVS Rooms	Carbon Steel D	62	17	3.294

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Table 6.2.1.2-1 (Sheet 1 of 3)
Calculated Maximum Differential Pressures of High Energy Line Breaks for Short-Term Subcompartment Pressurization[3]

AP1000 Room #	Possible[1] Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential Pressure (psi)	Table for Mass and Energy Data
11104	None	5.0	NA	NA
11105	None	5.0	NA	NA
11201	4" Cold Leg 1A Pressurizer Spray Line	5.0	<4.0	6.2.1.3-6
	4" Cold Leg 1B Pressurizer Spray Line	5.0	<4.0	6.2.1.3-6
11202	None	5.0	NA	NA
11204	3" Purification Line from CVS-PL-V003 Outlet to RHX Tube Inlet	5.0	<2.0	6.2.1.3-3
	3" Purification Return Line from RHX Shell Outlet to Check Valve CVS-PL-V080 Inlet	5.0	<2.0	6.2.1.3-4
11205	None	5.0	NA	NA
11206	None	5.0	NA	NA
11207	None	5.0	NA	NA
11208	None	5.0	NA	NA
11209 North	None	5.0	NA	NA
11209 Center	3" Purification Line from RHX Tube Outlet to LHX Tube Inlet	5.0	<4.0	6.2.1.3-2
	3" Purification Line from CVS-PL-V003 Outlet to RHX Tube Inlet	5.0	<4.9	6.2.1.3-3
	3" Purification Return Line from RHX Shell Outlet to Check Valve CVS-PL-V080 Inlet	5.0	<3.5	6.2.1.3-4
11209 South	3" Purification Line from RHX Tube Outlet to LHX Tube Inlet	5.0	<3.5	6.2.1.3-2

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Table 6.2.1.2-1 (Sheet 2 of 3)
Calculated Maximum Differential Pressures of High Energy Line Breaks for Short-Term Subcompartment Pressurization[3]

AP1000 Room #	Possible[1] Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential Pressure (psi)	Table for Mass and Energy Data
11209 Pipe Tunnel	3" Purification Line from CVS-PL-V003 Outlet to RHX Tube Inlet	7.5[2]	<5.5	6.2.1.3-3
	3" Purification Return Line from RHX Shell Outlet to Check Valve CVS-PL-V080 Inlet	7.5[2]	<5.0	6.2.1.3-4
	4" Blowdown Line of SG2 to SGS-PL-V074B	7.5[2]	<5.5	6.2.1.3-7
11300	None	5.0	NA	NA
11301	3" CVS Purification Return Line	5.0	<4.0	6.2.1.3-5
11302	None	5.0	NA	NA
11303	None	5.0	NA	NA
11304	3" CVS-PL-V081 Inlet to CVS-PL-V082 Outlet	5.0	<2.5	6.2.1.3-11
	3" CVS Purification Return Line	5.0	<3.0	6.2.1.3-12
11305	None	5.0	NA	NA
11400	None	5.0	NA	NA
11401	4" Line to Outlet Nozzle of SG1	5.0	<3.0	6.2.1.3-7
11402	4" Blowdown Line to Outlet Nozzle of SG2	5.0	<3.0	6.2.1.3-7
11403	2" Aux Spray Piping to Pressurizer Spray Nozzle	5.0	<2.5	6.2.1.3-14
	4" Aux Spray Piping to Pressurizer Spray Nozzle	5.0	<4.8	6.2.1.3-13
11500	None	5.0	NA	NA
11501	None	5.0	NA	NA
11502	None	5.0	NA	NA
11503	None	5.0	NA	NA

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Table 6.2.1.2-1 (Sheet 3 of 3)
Calculated Maximum Differential Pressures of High Energy Line Breaks for Short-Term Subcompartment Pressurization(3)

AP1000 Room #	Possible(1) Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential Pressure (psi)	Table for Mass and Energy Data
11504	None	5.0	NA	NA
11601	None	5.0	NA	NA
11602	None	5.0	NA	NA
11603	None	5.0	NA	NA
11701	None	5.0	NA	NA
11702	None	5.0	NA	NA
11703	None	5.0	NA	NA

Notes:
1. "None" indicates that there are no breaks analyzed within that specific subcompartment with respect to the break criteria provided in Subsection 6.2.1.2.3.2.
2. Structures are designed to a differential pressurization load of 5.0 psi except the CVS room pipe tunnel is designed to a differential pressurization load of 7.5 psi as discussed in Subsection 3.8.3.5.
3. "NA" indicates that no calculation was performed because no rupture was postulated or that the line was postulated to rupture in a region with a large free volume so compartment differential pressures would be negligible.

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Table 6.2.1.3-1
Short-term Mass and Energy Inputs

Analyzed Break	Pressure (psia)	Temperature (°F)
3" CVS Purification Return Line	2266.7	462
4" Cold Leg 1A Pressurizer Spray Line	2381.7	537
4" Cold Leg 1B Pressurizer Spray Line	2381.7	537
3" Purification Line from CVS-PL-V003 outlet to RHX tube inlet	2376.4	537
3" Purification Return Line from RHX shell outlet to check valve CVS-PL-V080 inlet	2254.9	441
3" Purification Return Line from RHX tube outlet to LHX tube inlet	2366.7	240
3" CVS-PL-V081 inlet to CVS-PL-V082 outlet	2183.1	441
3" CVS Purification Return Line	2183.1	441
4" Auxiliary Spray Piping to Pressurizer Spray Nozzle	2317	537
2" Auxiliary Spray Piping to Pressurizer Spray Nozzle	2317	537

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Table 6.2.1.3-2
Short-term Mass and Energy Releases - 3" Purification Line from RHX Tube Outlet to LHX Tube Inlet

Break Location	Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
RHX Tube Outlet	0.00	0.00	0.00
	0.001	1926.0	6.810E+05
	0.4385	1926.0	6.810E+05
	0.4386	860.9	4.589E+05
	End	860.9	4.589E+05
LHX Tube Inlet	0.00	0.00	0.00
	0.001	2130.2	4.442E+05
	0.0290	2130.2	4.442E+05
	0.0291	1926.0	6.810E+05
	0.4094	1926.0	6.810E+05
	0.4095	860.9	4.589E+05
	End	860.9	4.589E+05

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Table 6.2.1.3-3
Short-term Mass and Energy Releases - 3" Purification Line from CVS-PL-V003 Outlet to RHX Tube Inlet

Break Location	Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
Penetration between Rooms 11204 and 11209	0.00	0.00	0.00
	0.001	1721.8	9.178E+05
	0.3611	1721.8	9.178E+05
	0.3612	1926.0	6.810E+05
	0.7998	1926.0	6.810E+05
	0.7999	860.9	4.589E+05
	End	860.9	4.589E+05
RHX Tube Inlet	0.00	0.00	0.00
	0.001	1721.8	9.178E+05
	0.2895	1721.8	9.178E+05
	0.2896	1926.0	6.810E+05
	0.7281	1926.0	6.810E+05
	0.7282	860.9	4.589E+05
	End	860.9	4.589E+05

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Table 6.2.1.3-4
Short-term Mass and Energy Releases - 3" Purification Return Line from RHX Shell Outlet to Check Valve CVS-PL-V080 Inlet

Break Location	Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
Penetration between Rooms 11204 and 11209	0.00	0.00	0.00
	0.001	1921.8	8.074E+05
	0.0231	1921.8	8.074E+05
	0.0232	960.9	4.037E+05
	0.6614	960.9	4.037E+05
	0.6615	0.000	0.000
	End	0.000	0.000
RHX Shell Outlet	0.00	0.00	0.00
	0.001	1921.8	8.074E+05
	0.0867	1921.8	8.074E+05
	0.0868	960.9	4.037E+05
	0.5978	960.9	4.037E+05
	0.5979	0.000	0.000
	End	0.000	0.000

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Table 6.2.1.3-5
Short-term Mass and Energy Releases - 3" CVS Purification Return Line

Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
0.00	0.000	0.000
0.001	2341.5	1.134E+06
End	2341.5	1.134E+06

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Table 6.2.1.3-6
Short-term Mass and Energy Releases - 4" Cold Leg Pressurizer Spray Line

Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
0.00	0.000	0.000
0.001	4282.0	2.282E+06
End	4282.0	2.282E+06

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Table 6.2.1.3-7
Short-term Mass and Energy Releases - 4" Steam Generator Blowdown Line

Break Location	Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
At Steam Generator Outlet Nozzle	0.00	0.00	0.00
	0.001	1479.88	8.303E+05
	4.49	1479.88	8.303E+05
	4.50	739.94	4.152E+05
	End	739.94	4.152E+05
In Penetration of Pipe Tunnel	0.00	0.00	0.00
	0.001	1479.88	8.303E+05
	0.71	1479.88	8.303E+05
	0.72	1257.89	7.058E+05
	3.78	1257.89	7.058E+05
	3.79	517.96	2.906E+05
	End	517.96	2.906E+05

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Table 6.2.1.3-8
Parameters for LOCA Long-Term Containment Analysis

Number of Loops	2
Active Core Length (ft)	14.0
Core Power, license application (MWt)	3400
Nominal Vessel Inlet Temperature (°F)	535.5
Nominal Vessel Outlet Temperature (°F)	612.2
Steam Pressure (psia)	870.0
Rod Array	17 x 17
Accumulator Temperature (°F)	120.0
Containment Design Pressure (psia)	73.7

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Table 6.2.1.3-9 (Sheet 1 of 7)
Long-Term DECL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.00000	0.00	0.00	0.00	1172.85
0.00108	40052.87	533.75	0.00	1172.85
0.10110	59368.22	536.62	0.00	1172.85
0.20117	66300.29	536.89	0.00	1172.85
0.30142	66447.67	536.95	0.00	1172.85
0.40126	65675.65	537.00	0.00	1172.85
0.50145	65274.95	537.18	0.00	1172.85
0.60133	65290.12	537.55	0.00	1172.85
0.70130	65183.67	538.11	0.00	1172.85
0.80144	64916.35	538.88	0.00	1172.85
0.90103	64703.12	539.95	0.00	1172.85
1.00122	64778.19	541.41	0.00	1172.85
1.10107	63859.23	543.18	0.00	1172.85
1.20134	62704.81	545.26	0.00	1172.85
1.30148	61608.59	547.41	0.00	1172.85
1.40117	60696.70	549.57	0.00	1172.85
1.50101	59585.89	551.90	0.00	1172.85
1.60141	58359.48	554.43	0.00	1172.85
1.70131	56787.10	556.83	0.00	1172.85
1.80125	55305.59	558.86	0.00	1172.85
1.90137	54805.09	560.74	0.00	1172.85
2.00120	53523.86	562.60	0.00	1172.85
2.10115	51899.58	564.36	0.00	1172.85
2.20135	50470.83	566.26	0.00	1172.85
2.30124	48836.67	568.63	0.00	1172.85
2.40170	46760.05	570.30	0.00	1172.85
2.50110	45835.40	570.67	0.00	1172.85
2.60114	45719.77	570.44	0.00	1172.85
2.70114	43403.61	570.20	0.00	1172.85
2.80148	42788.61	571.01	0.00	1172.85
2.90154	41342.36	571.84	0.00	1172.85
3.00113	40211.11	572.79	0.00	1172.85
3.10107	38847.30	573.56	0.00	1172.85
3.20165	37459.63	574.16	0.00	1172.85
3.30125	36707.64	574.64	0.00	1172.85
3.40140	35554.13	574.74	0.00	1172.85
3.50164	33912.53	574.51	0.00	1172.85
3.60102	32445.79	574.15	0.00	1172.85
3.70124	31442.01	573.68	0.00	1172.85
3.80133	30850.86	573.12	0.00	1172.85
3.90137	30538.16	572.52	0.00	1172.85
4.00213	30402.98	572.04	0.00	1172.85
4.20089	30215.90	571.36	0.00	1172.85

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Table 6.2.1.3-9 (Sheet 2 of 7)
Long-Term DECL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
4.40010	29884.56	570.88	0.00	1172.85
4.60029	29237.21	569.91	0.00	1172.85
4.80091	28863.69	569.12	0.00	1172.85
5.00095	28500.04	568.75	0.00	1172.85
5.20029	27991.45	568.29	0.00	1172.85
5.40047	27617.14	567.86	0.00	1172.85
5.60072	26880.06	567.43	0.00	1172.85
5.80010	25514.01	567.46	0.00	1172.85
6.00037	24438.77	568.10	0.00	1172.85
6.20107	23755.47	568.80	0.00	1172.85
6.40077	23353.56	570.18	0.00	1172.85
6.60037	24088.35	574.19	0.00	1172.85
6.80035	22164.93	578.28	0.00	1172.85
7.00004	22614.22	581.76	0.00	1172.85
7.20050	22660.07	587.32	0.00	1172.85
7.40063	21703.27	593.51	0.00	1172.85
7.60053	21990.12	598.58	0.00	1172.85
7.80031	21794.29	601.95	0.00	1172.85
8.00010	20967.93	606.91	0.00	1172.85
8.20042	20881.43	612.68	0.00	1172.85
8.40037	20555.65	617.72	0.00	1172.85
8.60062	20196.66	623.44	0.00	1172.85
8.80051	19882.25	628.99	0.00	1172.85
9.00018	19753.14	632.83	0.00	1172.85
9.20122	19392.43	635.04	0.00	1172.85
9.40127	18955.06	636.58	0.00	1172.85
9.60057	18125.94	643.16	0.00	1172.85
9.80034	17308.72	655.66	0.00	1172.85
10.00008	16528.22	674.84	0.00	1172.85
10.20015	15595.19	700.74	0.00	1172.85
10.40041	15312.81	707.39	0.00	1172.85
10.40102	15311.00	707.41	0.00	1172.85
10.40164	15309.15	707.43	0.00	1172.85
10.60060	14926.83	708.59	0.00	1172.85
10.80003	14921.74	707.14	0.00	1172.85
11.00059	14498.78	713.57	0.00	1172.85
11.20005	14065.62	725.27	0.00	1172.85
11.40030	13746.99	726.15	0.00	1172.85
11.60079	13378.70	725.68	0.00	1172.85
11.80077	13108.47	723.62	0.00	1172.85
12.00002	12817.73	725.37	0.00	1172.85
12.20002	12608.56	728.61	0.00	1172.85
12.40202	12228.65	731.29	0.00	1172.85
12.60080	11936.85	732.84	0.00	1172.85

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Table 6.2.1.3-9 (Sheet 3 of 7)
Long-Term DECL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
12.80058	11684.11	734.81	0.00	1172.85
13.00048	11398.79	742.13	0.00	1172.85
13.20050	11185.74	745.24	0.00	1172.85
13.40083	10918.00	748.36	0.00	1172.85
13.60040	10702.65	750.83	0.00	1172.85
13.80071	10588.34	754.34	0.00	1172.85
14.00069	10461.08	753.58	0.00	1172.85
14.20110	10350.79	748.85	0.00	1172.85
14.40163	10338.33	743.29	0.00	1172.85
14.60172	10339.87	739.95	0.00	1172.85
14.80216	10168.76	736.39	0.00	1172.85
15.00154	10107.50	731.33	0.00	1172.85
15.20132	10037.27	730.64	0.00	1172.85
15.40209	9805.80	731.18	0.00	1172.85
15.60198	9668.46	729.18	0.00	1172.85
15.80174	9572.84	729.82	0.00	1172.85
16.00165	9332.89	732.42	0.00	1172.85
16.20099	9168.69	732.14	0.00	1172.85
16.40111	9045.69	733.75	0.00	1172.85
16.60064	8823.68	736.75	0.00	1172.85
16.80135	8656.43	737.69	0.00	1172.85
17.00245	8501.97	740.04	0.00	1172.85
17.20278	8305.69	743.04	0.00	1172.85
17.40007	8132.45	745.16	0.00	1172.85
17.60291	7951.18	748.11	0.00	1172.85
17.80012	7753.09	751.70	0.00	1172.85
18.00106	7538.37	755.57	0.00	1172.85
18.20078	7305.90	759.71	0.00	1172.85
18.40101	7057.74	763.21	0.00	1172.85
18.60187	6816.10	765.94	0.00	1172.85
18.80066	6587.07	768.24	0.00	1172.85
19.00116	6420.48	774.60	0.00	1172.85
19.20154	6201.37	780.97	0.00	1172.85
19.40130	5931.44	787.59	0.00	1172.85
19.60026	5703.61	791.54	0.00	1172.85
19.80112	5474.96	795.15	0.00	1172.85
20.00183	5247.80	797.78	0.00	1172.85
20.20207	5037.97	798.71	0.00	1172.85
20.40037	4844.34	796.50	0.00	1172.85
20.60103	4673.24	792.34	0.00	1172.85
20.80154	4507.68	785.42	0.00	1172.85
21.00033	4359.90	775.95	0.00	1172.85
21.20061	4213.90	765.34	0.00	1172.85
21.40038	4075.02	753.86	0.00	1172.85

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Table 6.2.1.3-9 (Sheet 4 of 7)
Long-Term DECL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
21.60041	3941.72	741.54	0.00	1172.85
21.80051	3813.73	728.79	0.00	1172.85
22.00092	3691.21	715.52	0.00	1172.85
22.20055	3573.81	702.02	0.00	1172.85
22.40015	3460.34	688.27	0.00	1172.85
22.60085	3349.67	674.29	0.00	1172.85
22.80028	3240.79	660.43	0.00	1172.85
23.00123	3134.87	646.66	0.00	1172.85
23.20025	3032.48	632.75	0.00	1172.85
23.40130	2934.25	618.77	0.00	1172.85
23.60062	2839.94	605.26	0.00	1172.85
23.80059	2785.80	590.53	0.00	1172.85
24.00032	2738.35	576.48	0.00	1172.85
24.20023	2690.76	563.10	0.00	1172.85
24.40037	2646.22	550.81	0.00	1172.85
24.60079	2595.15	538.42	0.00	1172.85
24.80099	2539.13	526.24	0.00	1172.85
25.00086	2484.05	514.43	0.00	1172.85
25.20027	2428.62	503.09	0.00	1172.85
25.40063	2373.34	492.10	0.00	1172.85
25.60067	2316.52	481.47	0.00	1172.85
25.80048	2260.36	471.09	0.00	1172.85
26.00093	2204.96	461.22	0.00	1172.85
26.20012	2150.27	451.77	0.00	1172.85
26.40032	2095.19	442.78	0.00	1172.85
26.60063	2041.33	433.95	0.00	1172.85
26.80110	1987.93	425.53	0.00	1172.85
27.00110	1932.88	417.43	0.00	1172.85
27.20147	1875.86	409.77	0.00	1172.85
27.40054	1816.27	402.49	0.00	1172.85
27.60094	1763.81	394.91	0.00	1172.85
27.80020	1734.80	386.34	0.00	1172.85
28.00090	1703.80	378.17	0.00	1172.85
28.20081	1659.91	370.23	0.00	1172.85
28.40000	1602.77	362.72	0.00	1172.85
28.60056	1529.80	355.96	0.00	1172.85
28.80117	1438.61	351.27	0.00	1172.85
29.00084	1301.55	331.87	0.00	1172.85
29.20008	1046.95	320.22	0.00	1172.85
29.40028	846.42	293.61	0.00	1172.85
29.60048	740.78	369.07	0.00	1172.85
29.80038	390.68	290.99	0.00	1172.85
30.00017	76.52	291.27	0.00	1172.85
30.20022	0.00	291.27	0.00	1172.85

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Long-Term DECL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
30.30000	0.00	1172.85	0.00	1172.85
30.70000	0.00	1172.85	0.00	1172.85
31.60000	0.00	1172.85	110.15	1172.85
39.98990	0.00	1172.85	277.76	1172.85
44.26170	44.77	332.98	309.91	1172.85
51.11270	151.51	201.99	306.69	1172.85
55.33040	206.47	188.62	304.53	1172.85
60.08680	260.73	181.68	302.04	1172.85
64.61610	303.35	178.72	300.29	1172.85
69.75980	344.91	177.00	298.26	1172.85
75.64790	383.82	176.25	295.90	1172.85
79.69800	404.01	176.38	294.29	1172.85
86.42560	430.39	177.08	292.30	1172.85
91.00040	444.22	177.71	290.93	1172.85
95.00040	453.98	178.34	289.72	1172.85
101.00040	464.97	179.37	287.88	1172.85
105.00040	474.33	179.63	286.81	1172.85
111.00040	474.91	181.29	285.44	1172.85
119.00040	479.52	182.62	283.41	1172.85
132.23300	471.78	185.81	280.19	1172.85
150.00000	458.14	189.29	275.62	1172.85
153.03100	452.32	190.28	274.92	1172.85
159.50000	446.63	191.35	273.67	1172.85
185.00000	407.43	197.56	269.12	1172.85
190.00000	397.85	199.07	268.28	1172.85
192.00000	210.98	247.43	277.12	1172.85
201.00000	336.81	1172.85	178.25	1172.85
210.10000	0.00	1172.85	203.18	1172.85
215.04000	0.00	1172.85	306.42	1172.85
225.14500	5.35	1172.85	294.70	1172.85
251.34600	0.00	1172.85	290.12	1172.85
262.10700	0.00	1172.85	301.34	1172.85
278.62500	0.00	1172.85	288.52	1172.85
299.44900	78.64	339.18	272.67	1172.85
319.81500	0.00	1172.85	282.61	1172.85
341.55800	0.00	1172.85	288.11	1172.85
357.38100	0.00	1172.85	281.99	1172.85
380.08900	0.00	1172.85	213.98	1172.85
401.34000	66.05	333.10	206.10	1172.85
422.89000	101.14	289.75	207.95	1172.85
439.29700	60.86	337.13	198.55	1172.85
461.72200	111.06	277.12	199.40	1172.85
482.52000	70.43	317.00	197.45	1172.85
503.31800	58.46	333.71	192.82	1172.85

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Table 6.2.1.3-9 (Sheet 6 of 7)
Long-Term DECL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
518.91600	75.95	305.41	189.44	1172.85
539.71400	76.76	302.01	188.86	1172.85
560.51200	54.02	335.33	183.08	1172.85
581.30900	76.29	297.85	179.47	1172.85
602.10700	81.14	289.73	173.60	1172.85
648.90200	76.26	290.30	166.19	1172.85
701.67700	82.14	277.68	160.57	1172.85
749.38800	79.12	275.91	154.02	1172.85
801.38200	84.06	264.97	149.41	1172.85
848.61900	78.26	266.39	143.76	1172.85
898.37400	85.11	253.96	139.52	1172.85
947.83100	76.92	257.74	133.90	1172.85
1002.89100	83.57	245.17	122.89	1172.85
1150.00000	72.37	243.43	112.79	1172.85
1201.00000	303.14	146.76	112.79	1172.85
1251.00000	314.82	157.69	111.17	1172.85
1300.00000	314.73	155.46	108.24	1172.85
1310.00000	314.96	154.98	107.65	1172.85
1531.16000	321.76	145.34	95.87	1172.85
1750.00000	330.97	137.47	86.56	1172.85
1984.63000	332.53	131.38	78.21	1172.85
3997.77000	321.57	112.70	47.86	1172.85
6009.01000	294.32	110.59	39.01	1172.85
6512.70000	284.52	110.60	37.89	1172.85
7518.20000	270.41	110.58	35.95	1172.85
8022.81000	262.54	110.62	35.06	1172.85
9980.83000	232.36	111.06	32.69	1172.85
10000.00000	232.06	111.06	32.67	1172.85
10500.00000	224.18	111.25	32.30	1172.85
11000.00000	216.31	111.46	31.95	1172.85
11500.00000	208.39	111.69	31.59	1172.85
11756.12000	204.33	111.81	31.42	1172.85
11806.12000	198.63	207.23	36.30	1172.85
11856.12000	0.00	1171.70	36.26	1171.70
13000.00000	0.00	1171.70	35.35	1171.70
15005.00000	0.00	1171.70	33.78	1171.70
20005.80000	0.00	1171.70	31.27	1171.70
26007.30000	0.00	1171.70	29.77	1171.70
30007.90000	0.00	1171.70	28.77	1171.70
36008.10000	0.00	1171.70	27.11	1171.70
40000.00000	0.00	1171.70	26.02	1171.70
60000.00000	0.00	1171.70	23.27	1171.70
1000000.00	0.00	1171.70	8.63	1171.70
1500000.00	0.00	1171.70	7.25	1171.70

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Long-Term DECL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
2000000.00	0.00	1171.70	6.39	1171.70
4000000.00	0.00	1171.70	4.53	1171.70

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Table 6.2.1.3-10 (Sheet 1 of 3)
Blowdown DEHL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Average Enthalpy (Btu/lbm)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
0.00000	0.00	0.00	0.00	1172.85
0.00109	105,570.69	640.36	0.00	1172.85
0.10123	84,276.91	643.74	0.00	1172.85
0.20147	69,536.27	648.87	0.00	1172.85
0.30212	66,459.53	648.51	0.00	1172.85
0.40115	63,767.37	647.50	0.00	1172.85
0.50117	61,780.84	646.07	0.00	1172.85
0.60163	60,568.67	644.59	0.00	1172.85
0.70199	59,836.05	643.37	0.00	1172.85
0.80156	59,097.35	642.66	0.00	1172.85
0.90127	58,145.00	642.15	0.00	1172.85
1.00142	57,238.52	642.77	0.00	1172.85
1.10124	56,197.23	644.53	0.00	1172.85
1.20146	55,077.32	645.55	0.00	1172.85
1.30111	54,566.21	646.09	0.00	1172.85
1.40180	54,386.15	647.10	0.00	1172.85
1.50106	53,769.79	647.80	0.00	1172.85
1.60162	52,685.67	647.41	0.00	1172.85
1.70104	51,506.02	645.71	0.00	1172.85
1.80114	50,609.82	644.03	0.00	1172.85
1.90174	50,184.90	643.59	0.00	1172.85
2.00150	49,725.24	643.29	0.00	1172.85
2.10118	49,161.41	643.89	0.00	1172.85
2.20155	48,229.62	645.79	0.00	1172.85
2.30103	47,263.74	647.48	0.00	1172.85
2.40113	46,518.36	648.54	0.00	1172.85
2.50111	46,092.72	648.26	0.00	1172.85
2.60141	45,621.89	646.13	0.00	1172.85
2.70168	45,100.39	643.63	0.00	1172.85
2.80172	44,605.49	641.15	0.00	1172.85
2.90173	44,144.54	638.98	0.00	1172.85
3.00164	43,675.82	637.47	0.00	1172.85
3.10145	43,218.53	636.62	0.00	1172.85
3.20168	42,709.16	636.15	0.00	1172.85
3.30109	42,224.10	635.20	0.00	1172.85
3.40161	41,851.30	633.69	0.00	1172.85
3.50202	41,572.59	632.18	0.00	1172.85
3.60144	41,230.11	631.98	0.00	1172.85
3.70164	40,821.18	633.63	0.00	1172.85
3.80150	40,275.80	635.07	0.00	1172.85
3.90150	39,842.81	638.19	0.00	1172.85
4.00100	39,426.69	638.75	0.00	1172.85
4.20138	39,081.20	636.41	0.00	1172.85

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Table 6.2.1.3-10 (Sheet 2 of 3)
Blowdown DEHL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Average Enthalpy (Btu/lbm)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
4.40048	39,248.60	629.12	0.00	1172.85
4.60221	39,552.56	621.60	0.00	1172.85
4.80261	40,044.99	614.28	0.00	1172.85
5.00037	34,131.29	647.30	0.00	1172.85
5.20026	34,081.91	641.02	0.00	1172.85
5.40054	34,072.49	632.55	0.00	1172.85
5.60033	33,996.78	628.40	0.00	1172.85
5.80062	33,868.08	624.58	0.00	1172.85
6.00047	33,694.00	621.26	0.00	1172.85
6.20044	33,486.38	619.01	0.00	1172.85
6.40091	32,744.20	620.51	0.00	1172.85
6.60074	32,303.09	621.00	0.00	1172.85
6.80014	31,604.74	621.00	0.00	1172.85
7.00181	31,339.32	618.53	0.00	1172.85
7.20150	30,952.43	616.40	0.00	1172.85
7.40211	30,470.86	615.61	0.00	1172.85
7.60157	30,019.92	614.03	0.00	1172.85
7.80114	29,402.10	613.35	0.00	1172.85
8.00274	28,766.97	613.62	0.00	1172.85
8.20091	28,022.33	614.27	0.00	1172.85
8.40064	27,170.14	616.22	0.00	1172.85
8.60026	26,248.56	618.43	0.00	1172.85
8.80142	25,251.81	623.03	0.00	1172.85
9.00012	24,695.39	624.65	0.00	1172.85
9.20300	23,232.10	632.39	0.00	1172.85
9.40008	22,325.42	638.48	0.00	1172.85
9.60128	21,729.93	641.25	0.00	1172.85
9.80116	20,355.91	652.80	0.00	1172.85
10.00109	19,420.14	662.24	0.00	1172.85
10.20236	18,471.76	672.91	0.00	1172.85
10.40117	17,602.76	682.03	0.00	1172.85
10.60151	16,610.39	694.91	0.00	1172.85
10.80100	15,628.61	713.20	0.00	1172.85
11.00240	14,721.95	731.75	0.00	1172.85
11.20062	13,821.05	752.20	0.00	1172.85
11.40229	12,889.32	775.73	0.00	1172.85
11.60122	11,975.66	802.31	0.00	1172.85
11.80212	11,077.56	833.25	0.00	1172.85
12.00010	10,210.87	868.52	0.00	1172.85
12.20059	9,346.99	910.42	0.00	1172.85
12.40039	8,498.21	955.06	0.00	1172.85
12.60095	7,826.42	977.10	0.00	1172.85
12.80059	7,371.91	986.44	0.00	1172.85
13.00077	6,967.45	997.64	0.00	1172.85

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Table 6.2.1.3-10 (Sheet 3 of 3)
Blowdown DEHL Break
Mass and Energy Releases

Time (sec)	Two-Phase		Steam	
	Mass (lbm/sec)	Average Enthalpy (Btu/lbm)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
13.20054	6,562.39	1,014.23	0.00	1172.85
13.40073	6,181.01	1,032.01	0.00	1172.85
13.60048	5,830.31	1,047.88	0.00	1172.85
13.80018	5,505.98	1,060.91	0.00	1172.85
14.00022	5,206.85	1,065.11	0.00	1172.85
14.20054	4,944.17	1,065.64	0.00	1172.85
14.40083	4,611.19	1,078.60	0.00	1172.85
14.60032	4,336.69	1,092.23	0.00	1172.85
14.80049	4,095.19	1,103.57	0.00	1172.85
15.00012	3,818.80	1,124.39	0.00	1172.85
15.20081	3,569.39	1,147.25	0.00	1172.85
15.40035	3,336.41	1,165.94	0.00	1172.85
15.60040	3,120.28	1,182.81	0.00	1172.85
15.80093	2,913.39	1,199.27	0.00	1172.85
16.00065	2,726.75	1,211.13	0.00	1172.85
16.20058	2,526.63	1,204.33	0.00	1172.85
16.40032	2,350.66	1,209.98	0.00	1172.85
16.60035	2,222.36	1,210.26	0.00	1172.85
16.80092	2,096.17	1,205.95	0.00	1172.85
17.00019	1,946.13	1,201.30	0.00	1172.85
17.20081	1,797.79	1,203.22	0.00	1172.85
17.40079	1,663.12	1,206.03	0.00	1172.85
17.60039	1,531.49	1,205.92	0.00	1172.85
17.80081	1,399.92	1,209.32	0.00	1172.85
18.00075	1,246.94	1,214.97	0.00	1172.85
18.20019	1,095.75	1,222.13	0.00	1172.85
18.40124	984.45	1,219.68	0.00	1172.85
18.60070	850.38	1,218.35	0.00	1172.85
18.80042	687.47	1,216.49	0.00	1172.85
19.00044	535.47	1,222.79	0.00	1172.85
19.20009	426.42	1,231.82	0.00	1172.85
19.40032	269.60	1,236.30	0.00	1172.85
19.60049	0.00	0.00	0.00	1172.85

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Table 6.2.1.3-11
Short-term Mass and Energy Releases - 3" CVS-PL-V081 Inlet to CVS-PL-V082 Outlet

Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
0.00	1110.9	5.838E+05
End	1110.9	5.838E+05

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Table 6.2.1.3-12
Short-term Mass and Energy Releases - 3" CVS Purification Return Line

Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
0.00	2367.7	1.114E+06
End	2367.7	1.114E+06

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Table 6.2.1.3-13
Short-term Mass and Energy Releases - 4" Auxiliary Spray Piping to Pressurizer Spray Nozzle

Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
0.00	4282.0	2.282E+06
End	4282.0	2.282E+06

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Table 6.2.1.3-14
Short-term Mass and Energy Releases - 2" Auxiliary Spray Piping to Pressurizer Spray Nozzle

Time (sec)	Total Mass Release Rate (lbm/sec)	Total Energy Release Rate (Btu/sec)
0.000	951.6	5.072E+05
0.0362	951.6	5.072E+05
0.0363	475.8	2.536E+05
End	475.8	2.536E+05

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Table 6.2.1.4-1 Not Used

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Table 6.2.1.4-2 (Sheet 1 of 2)
Main Steam Line Double Ended Rupture
Mass and Energy Releases from 30% Power Level

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.00	11468.30	1188.58
0.01	17637.10	1188.60
0.50	11865.20	1189.62
0.60	10839.60	1189.81
2.39	9088.22	1192.77
4.76	6977.98	1195.62
7.10	4973.83	1197.61
7.11	2402.15	1197.62
9.51	2311.41	1198.55
19.01	1881.93	1202.24
23.81	1699.56	1203.37
28.61	1543.11	1204.06
33.31	1411.22	1204.38
39.41	1267.86	1204.47
44.61	1169.42	1204.38
49.81	1084.97	1204.18
55.11	1009.94	1203.88
60.31	945.58	1203.51
65.51	889.21	1203.10
70.71	839.97	1202.66
81.11	759.10	1201.75
91.51	694.97	1200.81
101.91	642.42	1199.87
112.31	598.45	1198.95
122.81	560.75	1198.06
133.21	528.62	1197.22
143.61	500.83	1196.42
164.41	455.33	1194.93
185.31	419.91	1193.56
206.11	391.98	1192.33
226.91	369.44	1191.27
247.81	350.46	1190.33
289.41	319.78	1188.67
331.11	295.40	1187.26
372.81	276.49	1186.05
414.51	260.24	1184.91
456.21	246.00	1183.82
539.51	222.09	1181.80
622.81	201.03	1179.82
706.21	183.96	1178.08
753.51	167.56	1180.19
770.01	160.58	1181.96
786.61	151.83	1184.83
796.21	145.40	1187.30
803.21	139.83	1189.63

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Table 6.2.1.4-2 (Sheet 2 of 2)
Main Steam Line Double Ended Rupture
Mass and Energy Releases from 30% Power Level

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
815.61	128.28	1195.13
823.81	117.54	1199.98
828.01	110.64	1202.88
836.21	94.84	1209.36
844.51	77.41	1216.98
848.71	60.78	1221.36
850.51	46.64	1223.19
851.41	38.93	1224.05
852.01	32.47	1224.58
852.71	19.65	1225.19
852.81	11.58	1225.30
852.91	2.62	1150.35
1000.01	2.62	1150.35

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Table 6.2.1.4-3 (Sheet 1 of 2)
Main Steam Line Double Ended Rupture
Mass and Energy Releases from Full Power Level

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.00	9488.34	1195.03
0.01	14673.17	1195.04
0.49	10265.53	1195.51
0.53	10146.34	1195.54
2.12	8728.07	1196.69
7.11	4914.85	1197.03
7.12	2470.30	1196.90
9.11	2431.47	1197.32
11.11	2344.55	1198.22
18.91	1939.20	1201.83
22.91	1767.69	1202.99
26.91	1621.85	1203.75
30.81	1499.85	1204.20
40.31	1267.30	1204.47
45.31	1173.99	1204.39
50.11	1097.11	1204.22
59.81	975.89	1203.69
69.61	882.49	1203.04
79.41	809.49	1202.35
89.21	751.10	1201.65
99.01	702.44	1200.93
118.61	624.39	1199.51
138.21	564.93	1198.16
157.81	518.99	1196.95
177.31	483.30	1195.89
196.91	454.74	1194.91
236.11	411.54	1193.20
275.31	379.33	1191.74
314.41	353.51	1190.48
353.61	331.84	1189.34
376.81	305.86	1191.91
388.41	289.54	1194.76
400.01	269.11	1199.84
405.81	256.10	1203.69
411.61	240.27	1208.68
417.41	220.56	1215.05
423.21	196.05	1222.95
429.01	167.41	1232.28
437.51	119.88	1248.03
439.11	98.02	1251.23
439.81	90.27	1252.49
440.61	82.50	1253.87
443.51	57.13	1258.36
444.31	49.36	1259.44
445.01	41.86	1260.34

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Table 6.2.1.4-3 (Sheet 2 of 2)
Main Steam Line Double Ended Rupture
Mass and Energy Releases from Full Power Level

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
445.71	33.21	1261.19
446.01	28.91	1261.54
446.21	25.55	1261.78
446.31	23.51	1261.89
446.41	20.64	1262.01
446.51	2.62	1150.35
500.01	2.62	1150.35

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Table 6.2.1.4-4
Parameters for Steam Line Break Mass and Energy Releases Inside Containment Analysis

Plant data for all cases:	
Power, Nominal Rating (MWt)	3415
Nominal RCS Flow (GPM)	299,880
Nominal Full Load T_{avg} (°F)	573.6
Nominal RCS Pressure (psia)	2250
Nominal Steam Temperature (°F)	525.5
Nominal Feedwater Enthalpy (BTU/lbm)	419.4

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Table 6.2.1.5-1 (Sheet 1 of 12)
Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
0.00	0.00	0
0.01	7942.87	4197068
0.51	57956.40	30282500
1.01	56472.40	29608220
1.51	54084.40	28449280
2.01	48999.30	25903610
2.51	44669.40	23773360
3.01	40136.20	21554050
3.51	36059.80	19555650
4.01	32343.80	17726170
4.51	30080.20	16651800
5.01	28154.80	15805980
5.51	26326.81	15014160
6.01	24527.91	14256040
6.51	22846.03	13563570
7.01	21582.36	13025710
7.51	20192.85	12407740
8.01	18986.50	11890100
8.51	17871.42	11359050
9.01	17076.27	10930030
9.51	16328.68	10479370
10.01	15748.25	10165630
10.51	15138.17	9823398
11.01	14561.62	9486835
11.51	13474.22	9024197
12.01	13680.55	8910145
12.51	13179.21	8612196
13.01	12104.41	8178315
13.51	11928.12	7962325
14.01	11331.71	7630674
14.51	10285.89	7140065
15.01	9612.06	6800880
15.51	8890.77	6437936
16.01	8397.32	6117610
16.51	8110.99	5827232
17.01	7989.51	5551450
17.51	7776.17	5249230
18.01	7400.69	4907268
18.51	7127.20	4601066
19.01	6709.37	4279012
19.51	6386.74	3978181
20.01	6115.01	3705914
20.51	5776.52	3425902
21.01	5517.91	3180462
21.51	5273.02	2963178

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Table 6.2.1.5-1 (Sheet 2 of 12)
Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
22.01	4939.00	2748973
22.51	4605.24	2551896
23.01	4470.27	2376817
23.51	3941.49	2159592
24.01	3651.01	2028490
24.51	3112.96	1831021
25.01	2948.30	1674552
25.51	2559.77	1528877
26.01	2458.65	1408223
26.51	2948.44	1371068
27.01	2746.30	1290115
27.51	1786.07	1043598
28.01	2712.07	1012782
28.51	2910.33	1083228
29.01	2233.45	918945
29.51	1450.78	729919
30.01	1211.72	620407
30.51	1589.06	589646
31.01	2324.34	657795
31.51	511.11	385601
32.01	327.07	257183
32.51	229.93	170637
33.01	138.97	118111
33.51	38.78	49703
34.01	30.85	39557
34.51	24.61	28213
35.01	32.26	28358
35.51	24.41	20639
36.01	7.67	6232
36.51	63.33	50679
37.01	17.81	22853
37.51	13.02	16607
38.01	9.73	12441
38.51	0.00	0
39.01	3.26	2928
39.51	65.55	53687
40.01	67.34	72602
40.51	62.30	67887
41.01	49.03	58542
41.51	43.45	52364
42.01	17.96	22932
42.51	15.20	19131
43.01	33.83	37590
43.51	25.45	21467
44.01	29.94	24662
44.51	33.10	29509

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Table 6.2.1.5-1 (Sheet 3 of 12)
Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
45.01	28.30	26896
45.51	24.91	23758
46.01	72.50	78140
46.51	56.34	68242
47.01	40.06	48849
47.51	106.92	118490
48.01	26.54	28205
48.51	57.76	61793
49.01	166.99	188706
49.51	142.10	153683
50.01	122.34	133270
50.51	104.57	118432
51.01	94.50	106009
51.51	114.97	125648
52.01	304.27	327915
52.51	438.18	353398
53.01	217.91	238944
53.51	208.85	243504
54.01	196.84	229968
54.51	170.84	200237
55.01	143.16	168668
55.51	108.35	123758
56.01	96.07	110132
56.51	145.71	166951
57.01	184.00	199601
57.51	148.14	159282
58.01	111.16	129618
58.51	103.14	115094
59.01	59.11	70920
59.51	121.13	133279
60.01	192.10	221407
60.51	274.36	259082
61.01	217.48	222919
61.51	140.77	164224
62.01	113.46	131276
62.51	81.97	95753
63.01	125.53	146907
63.51	277.75	267057
64.01	279.66	259800
64.51	202.16	222185
65.01	181.19	194112
65.51	125.69	141746
66.01	97.86	111750
66.51	155.16	176333
67.01	224.14	242392
67.51	230.40	234200

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Table 6.2.1.5-1 (Sheet 4 of 12)
Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
68.01	180.98	207383
68.51	155.58	181044
69.01	129.63	148391
69.51	134.89	158826
70.01	179.41	202659
70.51	189.19	217939
71.01	189.94	212600
71.51	200.16	215380
72.01	155.87	182233
72.51	144.58	165043
73.01	152.39	171976
73.51	193.73	218892
74.01	213.02	246071
74.51	207.90	243728
75.01	285.04	276724
75.51	218.60	234337
76.01	174.99	194687
76.51	168.67	191156
77.01	230.41	262908
77.51	220.33	257748
78.01	328.67	299957
78.51	269.62	271423
79.01	194.11	220457
79.51	190.95	218188
80.01	213.99	244892
80.51	236.23	267305
81.01	260.53	279713
81.51	223.30	258393
82.01	217.44	254964
82.51	205.88	242910
83.01	238.23	259684
83.51	243.58	275560
84.01	234.26	269102
84.51	228.11	268944
85.01	220.56	262949
85.51	211.33	251832
86.01	218.36	259571
86.51	248.80	284213
87.01	260.76	296727
87.51	262.52	299547
88.01	264.89	298182
88.51	820.35	304725
89.01	1658.45	387314
89.51	419.25	332530
90.01	465.12	307753
90.51	798.98	323657

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Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
91.01	1564.18	392925
91.51	1681.45	488011
92.01	524.80	328926
92.51	715.87	302892
93.01	916.00	331695
93.51	1609.64	395641
94.01	1483.75	445986
94.51	796.83	367070
95.01	567.18	323061
95.51	846.66	343109
96.01	1008.32	354410
96.51	1583.02	418731
97.01	846.20	359939
97.51	559.47	324035
98.01	651.65	321456
98.51	736.91	339803
99.01	916.43	329098
99.51	1519.14	410810
100.01	720.12	338623
100.51	508.98	313605
101.01	751.29	332987
101.51	709.66	337150
102.01	1026.40	342413
102.51	1501.55	485867
103.01	1509.55	443226
103.51	1615.99	389899
104.01	709.60	343936
104.51	298.47	290818
105.01	405.18	307500
105.51	236.00	257365
106.01	214.57	244634
106.51	196.03	225879
107.01	211.49	243737
107.51	209.45	241604
108.01	212.12	244237
108.51	211.72	243033
109.01	231.93	255782
109.51	221.98	254250
110.01	230.22	261121
110.51	221.28	253187
111.01	220.14	254614
111.51	219.87	253850
112.01	222.09	256551
112.51	228.52	255866
113.01	225.70	254820
113.51	230.06	256396

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Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
114.01	232.07	259134
114.51	221.75	256236
115.01	220.54	255640
115.51	208.33	246119
116.01	217.96	249376
116.51	203.44	239316
117.01	233.75	262602
117.51	231.74	261453
118.01	212.96	251032
118.51	945.18	291332
119.01	1704.73	351147
119.51	1072.30	370636
120.01	805.66	318168
120.51	1629.05	366554
121.01	1601.92	368274
121.51	876.85	344974
122.01	953.84	345943
122.51	1061.37	345113
123.01	2077.05	361856
123.51	1155.47	400824
124.01	735.68	308794
124.51	1068.45	328982
125.01	1486.66	389182
125.51	1532.30	376918
126.01	1054.35	378053
126.51	355.16	300169
127.01	985.78	370905
127.51	1033.92	331664
128.01	1622.16	383545
128.51	1509.10	437329
129.01	886.90	351995
129.51	335.52	289093
130.01	567.94	261593
130.51	1353.63	358127
131.01	1597.91	353608
131.51	250.38	256082
132.01	183.70	215997
132.51	182.98	211803
133.01	188.64	218584
133.51	189.25	220139
134.01	187.08	215705
134.51	208.59	225750
135.01	189.82	213258
135.51	186.80	214116
136.01	182.34	209390
136.51	183.58	209779

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Table 6.2.1.5-1 (Sheet 7 of 12)
Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
137.01	181.74	207033
137.51	176.95	201679
138.01	195.32	207464
138.51	169.57	191895
139.01	199.69	206952
139.51	175.63	196657
140.01	174.02	195324
140.51	169.66	190497
141.01	154.86	176102
141.51	166.87	176534
142.01	142.95	162974
142.51	145.92	165928
143.01	142.22	163605
143.51	140.98	163470
144.01	135.00	157359
144.51	130.69	152796
145.01	143.00	154391
145.51	129.44	150373
146.01	131.26	152180
146.51	126.54	149938
147.01	120.20	146381
147.51	121.14	147390
148.01	118.77	145198
148.51	137.33	152053
149.01	125.19	149500
149.51	125.34	151325
150.01	123.63	151903
150.51	121.18	149714
151.01	123.87	153456
151.51	142.83	161818
152.01	123.52	152431
152.51	137.71	164398
153.01	138.42	165917
153.51	143.37	169283
154.01	144.04	170853
154.51	146.94	172510
155.01	157.46	180972
155.51	182.18	194506
156.01	182.68	199083
156.51	240.23	226854
157.01	186.17	201397
157.51	188.78	207869
158.01	177.53	197204
158.51	182.14	199744
159.01	170.94	193739
159.51	175.02	193991

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Table 6.2.1.5-1 (Sheet 8 of 12)
Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
160.01	183.89	202184
160.51	178.53	200576
161.01	180.58	202006
161.51	176.12	199661
162.01	178.24	202356
162.51	183.29	206040
163.01	187.97	205289
163.51	192.60	207365
164.01	188.51	207204
164.51	180.04	204996
165.01	177.20	203222
165.51	177.24	203394
166.01	179.82	205213
166.51	191.01	210737
167.01	202.91	222624
167.51	185.70	209773
168.01	201.58	219384
168.51	700.89	213031
169.01	1432.73	284625
169.51	722.96	282295
170.01	575.47	275907
170.51	705.57	272191
171.01	1397.11	321115
171.51	1406.78	306333
172.01	1180.97	318445
172.51	1451.68	326298
173.01	847.43	302801
173.51	751.32	271505
174.01	1484.65	298434
174.51	1442.95	361325
175.01	306.30	258039
175.51	754.45	290941
176.01	1234.32	321950
176.51	1756.73	314456
177.01	846.52	320401
177.51	750.78	285970
178.01	1352.84	375378
178.51	1523.54	329154
179.01	1610.91	489664
179.51	539.67	286078
180.01	747.77	308665
180.51	1370.97	351393
181.01	1627.21	324214
181.51	355.67	296933
182.01	748.64	299398
182.51	1313.10	372028

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Table 6.2.1.5-1 (Sheet 9 of 12)
Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
183.01	1602.96	368136
183.51	1550.98	345203
184.01	1200.60	358644
184.51	1566.80	362057
185.01	567.66	310741
185.51	759.46	306641
186.01	1582.28	358212
186.51	805.19	309261
187.01	1606.31	377818
187.51	753.33	326692
188.01	848.18	318199
188.51	1765.10	384565
189.01	754.14	309859
189.51	1345.61	397944
190.01	1756.46	369226
190.51	925.16	368732
191.01	670.60	329144
191.51	784.74	311315
192.01	2049.53	372920
192.51	1470.02	508289
193.01	611.38	320418
193.51	756.78	328536
194.01	879.90	312220
194.51	1098.97	378667
195.01	1744.50	365532
195.51	701.02	343283
196.01	662.32	323679
196.51	832.72	334816
197.01	1646.10	379766
197.51	893.81	335562
198.01	1714.79	373751
198.51	611.19	333258
199.01	609.68	320394
199.51	859.92	338769
200.01	1607.56	389751
200.51	847.96	336163
201.01	1726.12	374152
201.51	469.76	312506
202.01	487.49	317122
202.51	613.81	324092
203.01	833.89	325131
203.51	1259.87	410487
204.01	1625.88	385435
204.51	1483.54	513440
205.01	334.19	297833
205.51	254.25	275754

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Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
206.01	251.45	276027
206.51	240.26	268651
207.01	242.13	267065
207.51	245.12	268164
208.01	243.47	265902
208.51	237.91	263708
209.01	234.71	258062
209.51	231.02	257222
210.01	226.45	253802
210.51	232.31	254552
211.01	223.84	249586
211.51	216.30	243441
212.01	213.96	238951
212.51	215.87	240067
213.01	215.05	237629
213.51	219.00	238554
214.01	226.80	244094
214.51	223.98	241160
215.01	214.93	234096
215.51	210.14	230933
216.01	202.37	225065
216.51	196.68	219516
217.01	200.69	219035
217.51	184.67	208066
218.01	185.29	209360
218.51	181.72	207222
219.01	186.53	210264
219.51	193.14	213581
220.01	190.57	208299
220.51	178.65	199749
221.01	175.16	196941
221.51	164.34	188476
222.01	163.08	188061
222.51	158.73	183780
223.01	160.78	186829
223.51	167.31	185445
224.01	160.04	181597
224.51	154.22	179787
225.01	152.87	177243
225.51	163.79	186478
226.01	159.82	182491
226.51	169.10	190870
227.01	166.42	186909
227.51	169.19	189283
228.01	167.97	188685
228.51	162.80	183799

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Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
229.01	163.17	183882
229.51	166.72	185166
230.01	163.60	183083
230.51	166.08	185501
231.01	185.01	198040
231.51	171.17	190371
232.01	170.02	189413
232.51	167.32	187192
233.01	165.87	187133
233.51	162.23	184680
234.01	162.12	184239
234.51	163.65	185256
235.01	170.11	190998
235.51	172.03	191517
236.01	176.57	195559
236.51	177.90	195105
237.01	179.11	196781
237.51	177.36	194796
238.01	184.55	203196
238.51	175.43	193499
239.01	180.92	198347
239.51	186.38	201952
240.01	187.65	204156
240.51	206.22	215759
241.01	189.12	203103
241.51	193.57	210875
242.01	193.46	207113
242.51	198.81	213650
243.01	201.22	217114
243.51	197.74	212561
244.01	206.62	221578
244.51	203.31	217715
245.01	208.59	223702
245.51	203.91	219671
246.01	208.89	222208
246.51	212.31	225805
247.01	212.32	225398
247.51	214.35	227594
248.01	214.15	226651
248.51	217.36	228825
249.01	222.31	234267
249.51	217.86	229171
250.01	223.83	235041
250.51	220.16	231207
251.01	240.82	243903
251.51	253.81	247226

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Table 6.2.1.5-1 (Sheet 12 of 12)
Minimum Containment Pressure Mass and Energy Releases for Large Break LOCA

Time (sec)	Mass Release (lbm/s)	Energy Release (BTU/s)
252.01	436.89	265598
252.51	276.84	260077
253.01	829.62	316490
253.51	314.26	270224
254.01	236.27	246359
254.51	314.74	268735
255.01	272.12	261657
255.51	252.32	253658
256.01	277.28	262941
256.51	342.11	256883
257.01	755.60	308477
257.51	411.16	280874
258.01	646.77	293049
258.51	592.41	299836
259.01	276.26	264083
259.51	697.14	280918
260.01	1081.61	351122
260.51	708.00	303184
261.01	917.67	334397
261.51	542.69	321954
262.01	249.78	263198
262.51	347.78	287660
263.01	560.89	302627
263.51	682.74	309234
264.01	837.97	332817

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Table 6.2.2-1
Passive Containment Cooling System Performance Parameters

PCCWST(1) useable capacity for PCS (gal) - Minimum	756,700
PCCWST useable capacity for FPS(2) (gal) - Minimum	18,000
Flow duration from PCCWST (days) - Minimum	3
PCCWST minimum temperature (°F)	40
PCCWST maximum temperature (°F)	120
Upper annulus drain rate (per drain) - Minimum	525 gpm
PCCAWST(4) long-term makeup rate to containment - Minimum(7)	100 gpm
PCCAWST long-term makeup to spent fuel pool – Minimum(7)	35 gpm
PCCAWST long-term makeup duration - Minimum	4 days
PCCWST long-term makeup to spent fuel pool – Minimum	118 gpm

PCCWST Water Elevation (Note 3) (feet)	Nominal Flow (Note 5) (gpm)	Minimum Flow (Note 6) (gpm)	Safety Analysis Flow (gpm)	Wetted Coverage (Note 3) (% of circumference)
At initiating height	489	484	N/A	N/A
27.5	484	N/A	469.1	90
24.1	242	241	226.6	90
20.3	186	187	176.3	72.9
16.8	153	154	144.2	59.6
At 72 hours	105	102	100.7 @ 72 hours	41.7

Notes:

1. PCCWST = passive containment cooling water storage tank
2. FPS = fire protection system
3. PCCWST Water Elevation corresponds to the nominal standpipe elevations in feet above the tank floor (Reference Plant Elevation 293′-9″, see Figure 3.8.4-2). Wetted coverage is measured as the linear percentage of the containment shell circumference wetted measured at any elevation between elevation 266 ft. and the upper spring line for the safety analysis flow rate conditions.
4. PCCAWST = passive containment cooling ancillary water storage tank
5. The nominal flow is based on a nominal PCCWST water elevation between the overflow and the minimum useable level.
6. The minimum flow is based on initiating the system at the PCCWST minimum useable capacity.
7. These flow rates apply when the plant is not refueling. The minimum makeup flow rates are adjusted following a design basis loss of normal spent fuel pool cooling that occurs during a refueling because more decay heat is located in the spent fuel pool and less decay heat is located in the reactor. See Subsections 6.2.2.4.2 and 9.1.3 for additional details.

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Table 6.2.2-2
Component Data
Passive Containment Cooling System
(Nominal)

Passive Containment Cooling Water Storage Tank	
Volume (gal) - Minimum	756,700
Design temperature (°F)	125
Design pressure (psig)	Atmospheric
Material	Concrete with stainless steel liner
Standpipe Elevations Above Bottom of Tank Floor (Plant Elevation 293′-9″)	
Overflow (ft) – Nominal	28.5
Top standpipe (ft) - Nominal	24.1
Second standpipe (ft) - Nominal	20.3
Third standpipe (ft) - Nominal	16.8
Bottom standpipe (ft)	0.5
Passive Containment Ancillary Cooling Water Storage Tank	
Volume (gal) - Nominal	885,000
Design temperature (°F)	125
Design pressure (psig)	Atmospheric
Material	Carbon steel
Water Distribution Bucket	
Volume (gal) - Nominal	42
Design temperature (°F)	300
Design pressure (psig)	Atmospheric
Material	Stainless steel
Water Distribution Collection Troughs and Weirs	
Design temperature (°F)	N/A
Design pressure (psig)	Atmospheric
Material	Stainless steel
Passive Containment Cooling Recirculation Pump	
Quantity	2
Type	Centrifugal
Design capacity (gpm)	135
Design total differential head (ft) at 135 gpm	308 to 341

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Table 6.2.2-3
Failure Mode and Effects Analysis -
Passive Containment Cooling System Components

Component	Failure Mode	PCS Operation Phase	Effect on System Operation	Failure Detection Method	Remarks
Air-operated butterfly valve PCS-PL-V001A (PCS-PL-V001B and motor-operated valve PCS-PL-V001C analogous)	Failure to open on demand	Passive containment cooling water delivery to containment	Failure blocks flow of containment cooling water through one path of PCS which reduces system redundancy. No safety effect on system operation. Minimum containment cooling requirements will be met by the flow of cooling water through operation of one of three flowpaths.	Valve position indication (closed to open position change) in main control room and at the remote shutdown workstation	Valve is normally closed during power operations. Valve opens on actuation by a High-2 containment pressure signal or loss of air or loss of 1E power.
Motor-operated gate valve PCS-PL-V002A (PCS-PL-V002B and PCS-PL-V002C analogous)	Spurious valve closure	Passive containment cooling water delivery to containment	Spurious closure blocks flow of containment cooling water through associated flowpath of PCS which reduces system redundancy. No safety effect on system operation. Minimum containment cooling requirements will be met by the flow of cooling water through operation of one of three flowpaths.	Valve position indication (open to closed position change) in main control room and at the remote shutdown workstation	Valve is normally open during power operations. Valve receives confirmatory open signal on High-2.
Air-operated butterfly valve PCS-PL-V001A (PCS-PL-V001B and motor-operated valve PCS-PL-V001C analogous)	Spurious valve opening	Normal idle condition	Failure initiates flow of containment cooling water through associated flow path of PCS when not required. No safety effect on system operation. Flow will be terminated through operator action by closing the series isolation valves via the main control room.	Valve position indication (closed to open) in main control room or at the remote shutdown workstation. Also by PCS flow indication and decreasing PCCWST level.	Valve is normally closed during power operations to isolate PCS water.

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Table 6.2.3-1 (Sheet 1 of 4)
Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration			Isolation Device						Test		
	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	Pipe Length	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1, 14] & Note	Medium	Direction
CAS	Service air in	In	No	CAS-PL-V204 CAS-PL-V205	11 -	9.3.1	C-O-C C-O-C	None None	N/A N/A	C	Air or N2	Forward
	Instrument air in	In	No	CAS-PL-V014 CAS-PL-V015	8 -	9.3.1	O-O-C O-O-C	T None	std. N/A	C	Air or N2	Forward
CCS	IRC loads in	In	No	CCS-PL-V200 CCS-PL-V201	9 -	9.2.2	O-O-C O-O-C	S, HRCP2 None	std. N/A	C	Air or N2	Forward
	IRC loads out	Out	No	CCS-PL-V208 CCS-PL-V207 CCS-PL-V220	8 - -	9.2.2	O-O-C O-O-C C-C-C	S, HRCP2 S, HRCP2 None	std. std. N/A	C	Air or N2	Forward Forward Reverse
CVS	Spent resin flush out	Out	No	CVS-PL-V041 CVS-PL-V040 CVS-PL-V042	19 - -	9.3.6	C-C-C C-C-C C-C-C	None None None	N/A N/A N/A	C	Air or N2	Forward Forward Reverse
	Letdown	Out	No	CVS-PL-V047 CVS-PL-V045 CVS-PL-V058	38 - -	9.3.6	C-O-C C-O-C C-C-C	T, DAS T, DAS None	std. std. N/A	C C, 13 C	Air or N2	Forward Forward Reverse
	Makeup	In	No	CVS-PL-V090 CVS-PL-V091 CVS-PL-V100	31 - -	9.3.6	O-O-C O-O-C C-C-C	HR2, PL2, S+PL1, SGL3, R+SGL HR2, PL2, S+PL1, SGL3, R+SGL None	30 sec. 30 sec. N/A	C	Air or N2	Forward
	Zinc injection to RCS	In	No	CVS-PL-V092 CVS-PL-V094 CVS-PL-V098	22 -	9.3.6	O-C-C O-C-C C-C-C	T, LPL1 T, LPL1 None	std. std. N/A	C	Air or N2	Forward Forward Reverse
	Hydrogen Injection to RCS	In	No	CVS-PL-V219 CVS-PL-V217	38	9.3.6	O-C-C O-C-C	T, LPL1 None	std. N/A	C	Air or N2	Forward
DWS	Demin. water supply	In	No	DWS-PL-V244 DWS-PL-V245	29 -	9.2.4	C-O-C C-O-C	None None	N/A N/A	C	Air or N2	Forward
FHS	Fuel transfer	N/A	No	FHS-FT-01	-	6.2.5	C-O-C	None	N/A	B	Air or N2	Forward
FPS	Fire protection standpipe sys.	In	No	FPS-PL-V050 FPS-PL-V052	58 -	9.5.1	C-C-C C-C-C	None None	N/A N/A	C	Air or N2	Forward
PSS	RCS/PXS/CVS samples out	Out	No	PSS-PL-V010A PSS-PL-V010B PSS-PL-V011A PSS-PL-V011B	- - 19 15	9.3.3	O-C-C C-C-C O-C-C C-C-C	T T T T	std. std. std. std.	C	Air or N2	Forward
	Cont. air samples out	Out	No	PSS-PL-V046 PSS-PL-V008	13 -	9.3.3	O-C-C O-C-C	T T	std. std.	C	Air or N2	Forward
	RCS/Cont. air sample return	In	No	PSS-PL-V023 PSS-PL-V024	16 -	9.3.3	O-C-C O-C-C	T T	std. std.	C	Air or N2	Forward

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Table 6.2.3-1 (Sheet 2 of 4)
Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration			Isolation Device						Test		
	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	Pipe Length	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1, 14] & Note	Medium	Direction
PXS	N_2 to accumulators	In	No	PXS-PL-V042 PXS-PL-V043	9 -	6.3	O-O-C C-C-C	T None	std. N/A	C	Air or N2	Forward
RNS	RCS to RHR pump	Out	No	RNS-PL-V002A RNS-PL-V002B RNS-PL-V023 RNS-PL-V022 RNS-PL-V020 RNS-PL-V021 RNS-PL-V061 PXS-PL-V208A	- - - 46 - - - -	5.4.7 5.4.7 5.4.7 5.4.7 5.4.7 5.4.7 5.4.7 6.3	C-O-C C-O-C C-O-C C-O-C C-C-C C-C-C C-O-C C-C-C	HR2, S, ADS4 HR2, S, ADS4 HR2, S, ADS4 HR2, S, ADS4 None None T None	std. std. (see Note 12) std. std. N/A N/A std. N/A	A,5	H2O	Forward
	RHR pump to RCS	In	No	RNS-PL-V011 RNS-PL-V012 RNS-PL-V013	26 -	5.4.7	C-O-C C-C-C (11) C-O-C	HR2, S, ADS4 None None	std. N/A N/A	A,5	H2O	Forward
SFS	IRWST/Ref. cav. SFP pump discharge	In	No	SFS-PL-V038 SFS-PL-V037	20 -	9.1.3	C-O-C C-O-C	T None	std. N/A	C	Air or N2	Forward
	IRWST/Ref. cav. purif. out	Out	No	SFS-PL-V035 SFS-PL-V034 SFS-PL-V067	30 - -	9.1.3	C-O-C C-O-C C-C-C	T T None	std. std. N/A	C	Air or N2	Forward Forward Reverse
SGS	Main steam line 01	Out	Yes	SGS-PL-V040A SGS-PL-V027A(7) SGS-PL-V030A,31A,32A, 33A,34A,35A SGS-PL-V036A SGS-PL-V240A	29 26 13, 16, 20, 23, 26, 29 40 47	10.3	O-C-C O-O-C C-C-C O-O-C C-C-C	MS LSL2 None MS MS	5 sec std. N/A std. std.	A,2	N2	Forward
	Main steam line 02	Out	Yes	SGS-PL-V040B SGS-PL-V027B(7) SGS-PL-V030B,31B,32B, 33B,34B,35B SGS-PL-V036B SGS-PL-V240B	29 26 13, 16, 20, 23, 26, 29 40 47	10.3	O-C-C O-O-C C-C-C O-O-C C-C-C	MS LSL2 None MS MS	5 sec std. N/A std. std.	A,2	N2	Forward
	Main feedwater 01	In	Yes	SGS-PL-V057A SGS-PL-V101A SGS-PL-V103A	28 - -	10.3	O-C-C C-C-C C-C-C	MF None None	5 sec N/A N/A	A,2	H2O	Forward
	Main feedwater 02	In	Yes	SGS-PL-V057B SGS-PL-V101B SGS-PL-V103B	28 - -	10.3	O-C-C C-C-C C-C-C	MF None None	5 sec N/A N/A	A,2	H2O	Forward
	SG blowdown 01	Out	Yes	SGS-PL-V074A	13	10.3	O-O-C	PRHR, T, LSGL	std.	A,2	H2O	Forward
	SG blowdown 02	Out	Yes	SGS-PL-V074B	19	10.3	O-O-C	PRHR, T, LSGL	std.	A,2	H2O	Forward
	Startup feedwater 01	In	Yes	SGS-PL-V067A	29	10.3	O-O-C	LTC2, SGL3, R+SGL	std.	A,2	H2O	Forward
	Startup feedwater 02	In	Yes	SGS-PL-V067B	27	10.3	O-O-C	LTC2, SGL3, R+SGL	std.	A,2	H2O	Forward

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Table 6.2.3-1 (Sheet 3 of 4)
Containment Mechanical Penetrations and Isolation Valves

	Containment Penetration			Isolation Device						Test		
System	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	Pipe Length	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1, 14] & Note	Medium	Direction
VFS	Cont. air filter supply	In	No	VFS-PL-V003 VFS-PL-V004	33 -	9.4.7	C-O-C C-O-C	T,HR1,DAS T,HR1,DAS	10 sec 10 sec	C	Air or N2	Forward Forward
	Cont. air filter exhaust	Out	No	VFS-PL-V010 VFS-PL-V009 VFS-PL-V008 VFS-PL-V800A VFS-PL-V800B VFS-PL-V803A VFS-PL-V803B	85 - - 99 102 - -	9.4.7	C-O-C C-O-C C-C-C C-C-C C-C-C C-C-C C-C-C	T,HR1,DAS T,HR1,DAS N/A T, HR1 (Note 8) T, HR1 (Note 8) None None	10 sec 10 sec N/A 30 sec 30 sec N/A N/A	C C N/A C C C C	Air or N2	Forward
VWS	Fan Coolers out	Out	No	VWS-PL-V086 VWS-PL-V082 VWS-PL-V080	9 - -	9.2.7	O-O-C O-O-C C-C-C	T T None	std. std. N/A	C,3,4	Air or N2	Forward Forward Reverse
	Fan coolers in	In	No	VWS-PL-V058 VWS-PL-V062	8 -	9.2.7	O-O-C O-O-C	T N/A	std. N/A	C,3,4	Air or N2	Forward
WLS	Reactor coolant drain tank gas	Out	No	WLS-PL-V068 WLS-PL-V067	49 -	11.2	C-C-C C-C-C	T T	std. std.	C	Air or N2	Forward
	Normal cont. sump	Out	No	WLS-PL-V057 WLS-PL-V055 WLS-PL-V058	39 - -	11.2	C-C-C C-C-C C-C-C	T,DAS T,DAS None	std. std. N/A	C, 13 C, 13 C	Air or N2	Forward Forward Reverse
SPARE		N/A	No	P40	-	6.2.5	C-C-C	N/A	N/A	B	Air or N2	Forward
SPARE		N/A	No	P41	-	6.2.5	C-C-C	N/A	N/A	B	Air or N2	Forward
SPARE		N/A	No	P42	-	6.2.5	C-C-C	N/A	N/A	B	Air or N2	Forward
CNS	Main equipment hatch	N/A	No	CNS-MY-Y01	-	6.2.5	C-C-C	None	N/A	B	Air or N2	Forward
	Maintenance hatch	N/A	No	CNS-MY-Y02	-	6.2.5	C-C-C	None	N/A	B	Air or N2	Forward
	Upper Personnel Hatch	N/A	No	CNS-MY-Y03	-	6.2.5	C-C-C	None	N/A	B	Air or N2	Forward
	Lower Personnel Hatch	N/A	No	CNS-MY-Y04	-	6.2.5	C-C-C	None	N/A	B	Air or N2	Forward
PCS	Containment pressure instrumentation lines (four)	N/A	Yes	P46, P47, P48, P49	-	6.2.3.1	N/A	N/A	N/A	A,10	Capillary Fluid	Forward

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Table 6.2.3-1 (Sheet 4 of 4)
Containment Mechanical Penetrations and Isolation Valves

Explanation of Heading and Acronyms for Table 6.2.3-1

System:	Fluid system penetrating containment
Containment Penetration:	These fields refer to the penetration itself
Line:	Fluid system line
Flow:	Direction of flow in or out of containment
Closed Sys IRC:	Closed system inside containment as defined in Subsection 6.2.3.1.1
Isolation Device:	These fields refer to the isolation devices for a given penetration
Valve/Hatch ID:	Identification number on P&ID or system figure
Pipe Length:	Nominal length of pipe to outboard containment isolation valve, feet
Subsection Containing Figure:	Safety analysis report containing the system P&ID or figure
Position N-S-A:	Device position for N (normal operation)
	S (shutdown)
	A (post-accident)
Signal:	Device closure signal
	MS: Main steam line isolation
	ADS4: ADS Stage 4 Actuation
	LSL2: Low-2 steam line pressure
	MF: Main feedwater isolation
	LTC2: Low-2 T_{cold}
	PRHR: Passive residual heat removal actuation
	T: Containment isolation
	S: Safeguards actuation signal
	R: Reactor Trip signal
	HR1: High-1 containment radiation
	HR2: High-2 containment radiation
	DAS: Diverse actuation system signal
	PL2: High 2 pressurizer level signal
	S+PL1: Safeguards actuation signal plus high 1 pressurizer level
	R+SGL: Reactor trip plus High steam generator level
	SGL3: High-3 steam generator level
	HRCP2: High-2 reactor coolant pump bearing water temperature trip
	LSGL: Low steam generator level
	LPL1: Low-1 Pressurizer Water Level
Closure Time:	Required valve closure stroke time
	std: Industry standard for valve type (≤ 60 seconds)
	N/A: Not Applicable
Test:	These fields refer to the penetration testing requirements
Type:	Required test type
	A: Integrated Leak Rate Test
	B: Local Leak Rate Test -- penetration
	C: Local Leak Rate Test -- fluid systems
Note:	See notes below
Medium:	Test fluid on valve seat
Direction:	Pressurization direction
	Forward: High pressure on containment side
	Reverse: High pressure on outboard side

Notes:

1. Containment leak rate tests are designated Type A, B, or C according to 10CFR50, Appendix J.
2. The secondary side of the steam generator, including main steam, feedwater, startup feedwater, blowdown and sampling piping from the steam generators to the containment penetration, is considered an extension of the containment. These systems are not part of the reactor coolant pressure boundary and do not open directly to the containment atmosphere during post-accident conditions. During Type A tests, the secondary side of the steam generators is vented to the atmosphere outside containment to ensure that full test differential pressure is applied to this boundary.
3. The central chilled water system remains water-filled and operational during the Type A test in order to maintain stable containment atmospheric conditions.
4. The containment isolation valves for this penetration are open during the Type A test to facilitate testing.
5. RNS CIVs do not constitute potential primary containment atmospheric pathways during and following a Design Basis Accident (DBA).
6. Deleted.
7. Refer to Table 15.0-4b for PORV block valve closure time.
8. These valves also receive a signal to open on Low-2 containment pressure.
9. Not used.
10. The containment pressure instrumentation lines are sealed, fluid-filled, and closed inside and outside the containment, without containment isolation valves. They are not vented or drained during Type A testing.
11. Valve is opened and administratively controlled for post-72-hour accident actions to permit long-term containment makeup.
12. A 60-second time delay is imposed on the actuation signal for RNS-PL-V002B. This is not a radiological concern since the valve is normally closed with power removed in normal plant operation, only in service during shutdown conditions, and is flooded during post-accident operation. Once the time delay expires, the stroke time of the valve from receipt of the actuation signal is <60 seconds.
13. A closure time of less than or equal to 150 seconds is required when actuated from DAS.
14. Alternate Type tests can be conducted for repair / modification activities as applicable in accordance with regulatory guidance.

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Table 6.2.4-1
Component Data - Hydrogen Monitors
(Nominal)

Number	3
Range (% hydrogen)	0 - 20
Response time	90% in 180 seconds

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Table 6.2.4-2
Component Data - Hydrogen Recombiner
(Nominal)

Number Full Size PAR	2
Average efficiency (percent)	85
Depletion rate	Reference 19

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Table 6.2.4-3
Component Data - Hydrogen Igniter
(Nominal)

Number	66
Surface Temperature (°F)	≥1700

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Tables 6.2.4-4–6.2.4-5 Not Used

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Table 6.2.4-6 (Sheet 1 of 3)
Igniter Location

Criteria
• A sufficient number of igniters are placed in the major transport paths (including dominant natural circulation pathways) of hydrogen so that hydrogen can be burned continuously close to the release point. This prevents hydrogen from preferentially accumulating in a certain region of the containment. • Igniters (minimum of 2) are located in major regions or compartments where hydrogen may be released, through which it may flow, or where it may accumulate. • It is preferable to ignite a hydrogen-air mixture at the bottom so that upward flame propagation can be promoted at lean hydrogen concentrations. Igniters within each subcompartment are located in the vicinity of, and above, the highest potential release location within the subcompartment. • In compartments with relatively small openings in the ceiling, the potential may exist for the hydrogen-air mixture to rise and to collect near the ceiling. Therefore, one or more igniters are placed near the ceiling of such compartments. Igniter coverage is provided within the upper 10 percent of the vertical height subcompartments or 10 feet from the ceiling whichever is less. In cases where the highest potential release point is low in the compartment, both this and the previous criteria are considered. • To the extent possible, igniters are placed away from walls and other large surfaces so that a flame front created by ignition at the bottom of a compartment can travel unimpeded up to the top. • A sufficient number of igniters are installed in long, narrow compartments (corridors) so that the flame fronts created by the igniters need to travel only a limited distance before they merge. This limits the potential for significant flame acceleration. • Igniter coverage is provided to control combustion in areas where oxygen rich air may enter into an inerted region with combustible hydrogen levels during an accident scenario. • Igniters are located above the flood level, if possible. Those which may be flooded have redundant fuses to protect the power supply. • In locations where the potential hydrogen release location can be defined, i.e. above the IRWST spargers, at IRWST vents, etc igniter coverage is provided as close to the source as feasible. • Provisions for installation, maintenance, and testing are to be considered.

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Table 6.2.4-6 (Sheet 2 of 3)
Igniter Location

Implementation

Note: The igniter locations and power groups are delineated in Table 6.2.4-7. For each subcompartment/area, the igniters are split between the two power groups and provide backup control for each other.

- **Reactor Cavity –** Hydrogen releases within the reactor cavity will flow either through the vertical access tunnel, through the opening around the RCS hot and cold legs into the loop compartments or if the permanent cavity seal ring fails then potentially through the refueling cavity. The potential flow paths have igniters. No igniters have been located within the reactor cavity since this region would always be flooded, adequate igniter coverage is available in hydrogen pathways from the reactor cavity and any maintenance or inspection would result in elevated personnel exposure.
- **Loop Compartments –** Hydrogen releases from the hot or cold legs or from the reactor cavity would flow up through the loop compartment to the dome region. Igniter coverage provided within the loop compartment consists of igniters at different elevations covering the perimeter of the compartment. Additional igniter coverage is provided above each loop compartment.
- **Pressurizer Compartment –** Hydrogen releases within the pressurizer compartment would flow up through the compartment toward the dome region. Igniter coverage provided within the compartment consists of igniters at different elevations covering the perimeter of the compartment. Additional igniter coverage is provided above the pressurizer compartment.
- **Tunnel Connection Loop Compartments –** The tunnel between the loop compartments and extending downward into the reactor coolant drain tank cavity is provided with igniters for hydrogen control. Releases within the reactor cavity or from the loop compartment may flow through this vertical access tunnel. Igniter coverage is provided over the width of the tunnel at separate elevations.
- **Refueling Cavity –** Hydrogen releases from the reactor cavity or potentially from the reactor coolant loops may flow up past the permanent cavity seal ring and through the refueling cavity to the dome region. Igniter coverage provided within the refueling cavity consists of igniters at different elevations covering the perimeter of the compartment. Additional igniter coverage is provided above the refueling cavity.
- **Southeast Valve and Accumulator Rooms –** Hydrogen releases within the southeast valve or accumulator rooms will rise with the mass and energy releases to near the ceiling and exit either through the opening on the west wall or through piping penetration holes in the ceiling. The hydrogen control protection is provided by igniters, located near the ceiling of each of the adjoining rooms.

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Table 6.2.4-6 (Sheet 3 of 3)
Igniter Location

- **East Valve, Northeast Accumulator, and Northeast Valve Room –** Hydrogen releases within the east valve, northeast accumulator or valve rooms will rise with the mass and energy releases to near the ceiling and exit either through the enlarged vent area surrounding the discharge piping from the core makeup tank and through other piping penetration holes in the ceiling. The hydrogen control protection is provided by igniters, located near the ceiling of each of the adjoining rooms.
- **North CVS Equipment Room –** Hydrogen releases within the CVS equipment room will rise from the piping or equipment located on the CVS module to near the ceiling, pass over the outer barrier wall and flow up through the stairwell or ceiling grating. Hydrogen control is provided by igniters located near the ceiling of the equipment room between the equipment module and the major relief paths from the compartment.
- **IRWST –** Hydrogen releases into the IRWST are controlled by the distribution of igniters internal to the IRWST and near the vents from and into the IRWST.
  - IRWST Sparger Side – Igniters are located within the IRWST just below the tank roof of the IRWST and near the spargers. In the event of hydrogen releases via the spargers, the igniters near the release points will provide the most immediate point of recombination.
  - IRWST Roof Vents – Should the environment within the IRWST be inerted or otherwise not be ignited by the assemblies near the sparger, the hydrogen will be ignited as it exhausts from the IRWST at any of the vents with igniter assemblies. The igniters are located in the upper compartment just above the vents.
  - IRWST Vacuum Breaker Vents – In the event that the IRWST is hydrogen rich and air is drawn into the IRWST the mixture will become flammable. In order to provide this recombination, the two inlet vents on the other side of the IRWST from the sparger side are fitted with an igniter.
  - IRWST Hooded Vents – Should the environment within the IRWST be inerted or otherwise not be ignited by the assemblies near the sparger side, the hydrogen will be ignited as it exhausts from the IRWST by the igniters at the vent hoods. The igniters are located in the upper compartment just outside the vent hoods.
- **Lower Compartment Area –** Hydrogen releases within the lower compartment will rise with the mass and energy releases to near the ceiling and exit either through the north stairwell or along the circumferential gap between the operating deck and the containment shell. The hydrogen control protection is provided by igniters spread over the potential release areas and located either near the mezzanine deck elevation or near the ceiling. This approach provides wide coverage over the entire compartment area at two separate elevations.
- **Upper Compartment –** Hydrogen control is provided at three separate regions within the upper compartment. At the lower region, igniters are distributed over the area primarily above the major release flow paths including the loop compartments, refueling cavity, pressurizer compartment and above the stairwell from the lower compartment area. An igniter is provided in each quadrant at the mid region of the upper compartment. At the upper region, igniters are located to initiate recombination of hydrogen not ignited at either the source or along its flow path.

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Table 6.2.4-7
Subcompartment/Area Igniter Coverage

	Igniter Coverage (Elevation)[1]	
Subcompartment	**Power Group 1**	**Power Group 2**
Reactor Cavity	1 (El 89'-3") 3 (El 95') 13 (El 121'-6"), 5, 55 (El 120') 58 (El 132') 8, 12 (El 139')	4 (El 96'-6") 2 (El 99') 11, 7 (El 119'), 56 (El 120') 57 (El 132') 6, 14 (El 139')
Loop Compartment 01	13 (El 121'-6") 12 (El 139')	11 (El 120') 14 (El 139')
Loop Compartment 02	5 (El 120') 8 (El 139')	7 (El 119') 6 (El 139')
Pressurizer Compartment	49 (El 154'-3") 60 (El 135')	50 (El 154') 59 (El 135')
Tunnel connecting Loop Compartments	1 (El 89'-3") 3 (El 95') 31 (El 120')	4 (El 96'-6") 2 (El 99') 30 (El 132'-8")
Southeast Valve Room	21 (El 105'-2")	20 (El 105'-2")
Southeast Accumulator Room	21 (El 105'-2")	20 (El 105'-2")
East Valve Room	18 (El 105'-2")	19 (El 105'-2")
Northeast Accumulator Room	18 (El 105'-2")	17, 19 (El 105'-2")
Northeast Valve Room	18 (El 105'-2")	17 (El 105'-2")
North CVS Equipment Room	34 (El 105'-2")	33 (El 105'-2")
Lower Compartment Area (CMT and Valve area)	22 (El 134') 27 (El 117'-3") 28 (El 118'-11") 29, 31, 32 (El 120')	23, 24, 25 (El 134') 26 (El 119'-6") 30 (El 132'-8")
IRWST Hooded Vents	35, 37 (El 137'-6")	36, 38 (El 137'-6")
IRWST Sparger Side	9 (El 133'-3")	10 (El 133'-3")
IRWST Vacuum Breaker Vents	16 (El 132'-7")	15 (El 132'-7")
IRWST Roof Vents	65 (El 137')	66 (El 137')
Refueling Cavity	55 (El 120') 58 (El 132')	56 (El 120') 57 (El 132')
Upper Compartment		
Lower Region	39, 42, 44, 43 (El 167'-7"), 47 (El 166'-9")	40, 41, 45 (El 161'-11"), 46, 48 (El 166'-9")
Mid Region	51, 54 (El 218')	52, 53 (El 218')
Upper Region	61, 63 (El 258')	62, 64 (El 258')

Note:
1. Elevations are approximate (± 2.5 feet).

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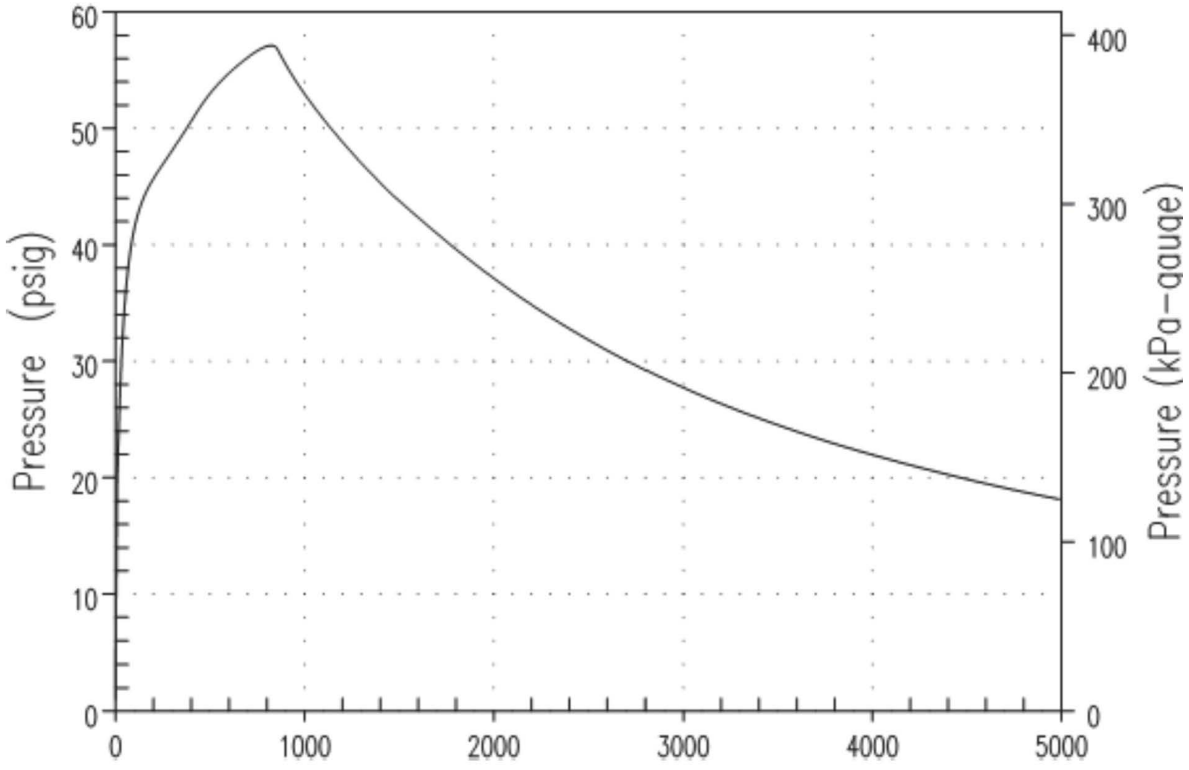

Figure 6.2.1.1-1
AP1000 Containment Pressure Response for Full DER MSLB – 30% Power

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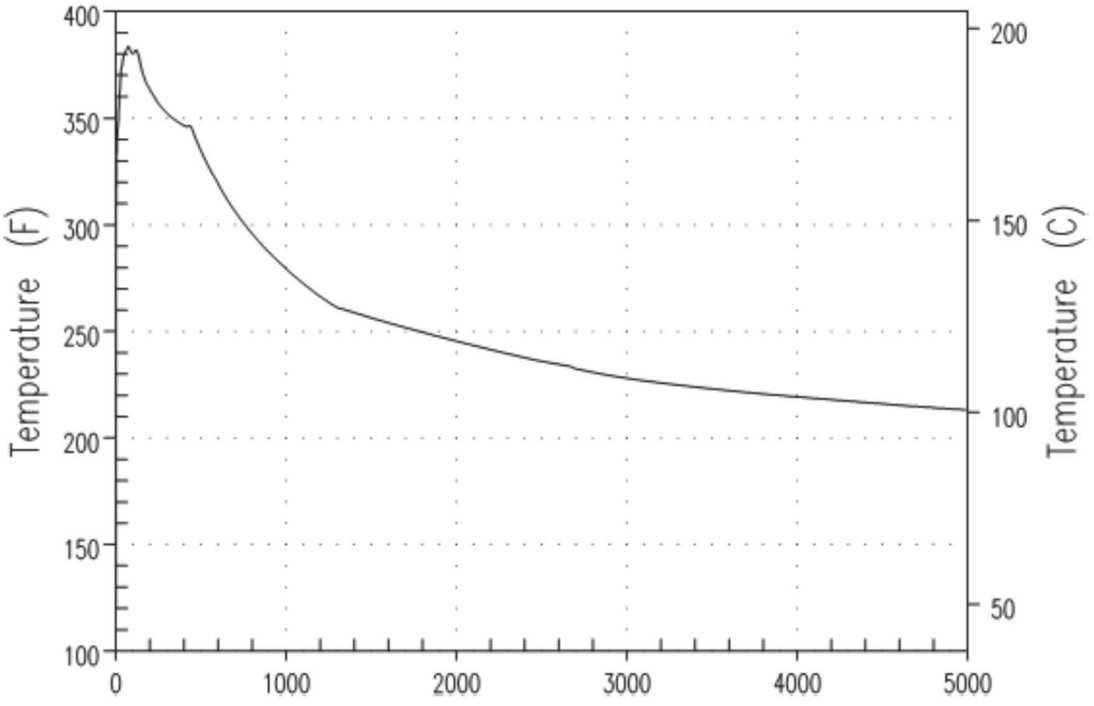

Figure 6.2.1.1-2
AP1000 Containment Temperature Response for Full DER MSLB – 101% Power

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Figures 6.2.1.1-3–6.2.1.1-4 Not Used

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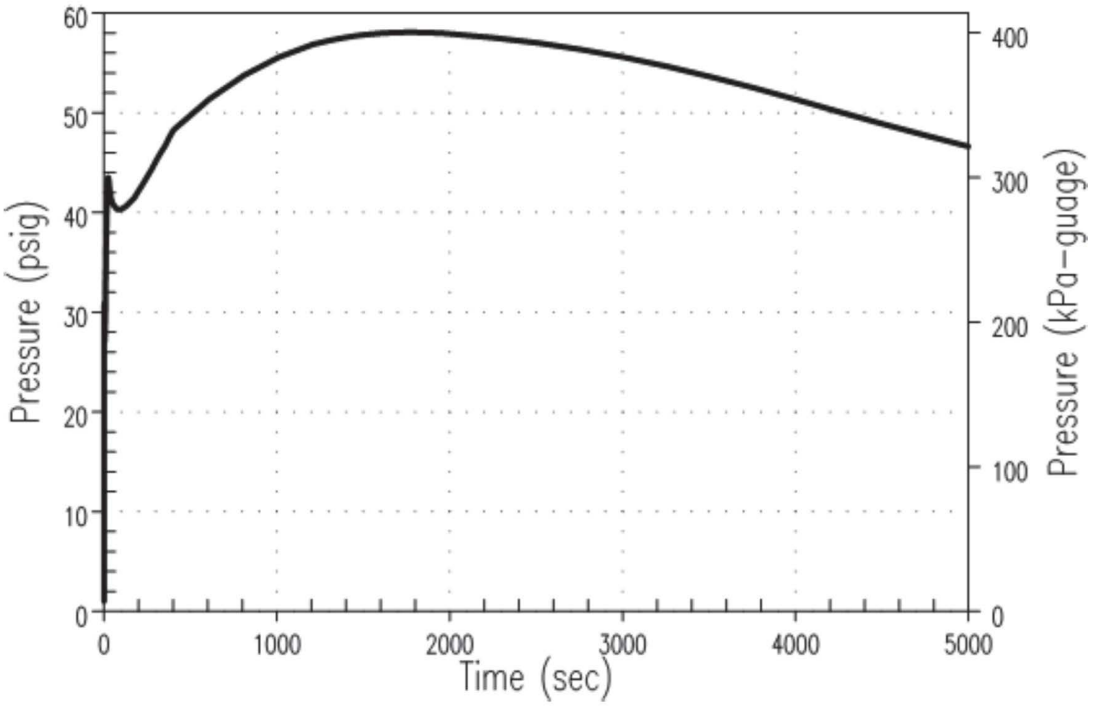

Figure 6.2.1.1-5
AP1000 Containment Pressure Response for DECLG LOCA – 5000 sec

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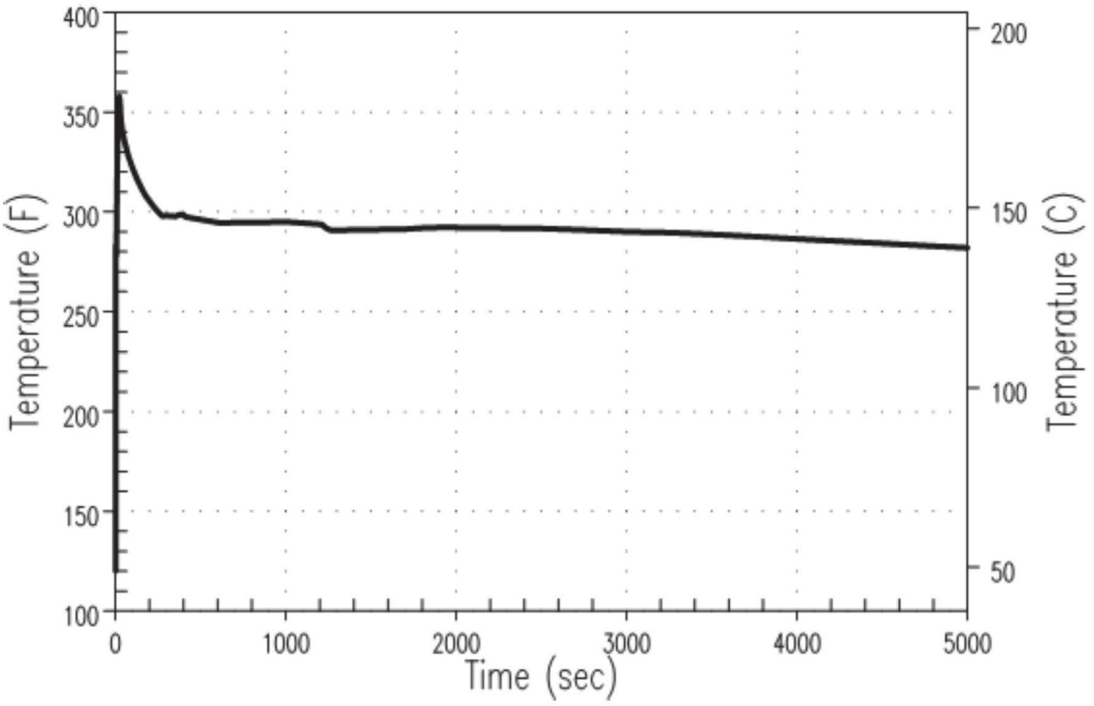

Figure 6.2.1.1-6
AP1000 Containment Temperature Response to DECLG LOCA – 5000 sec

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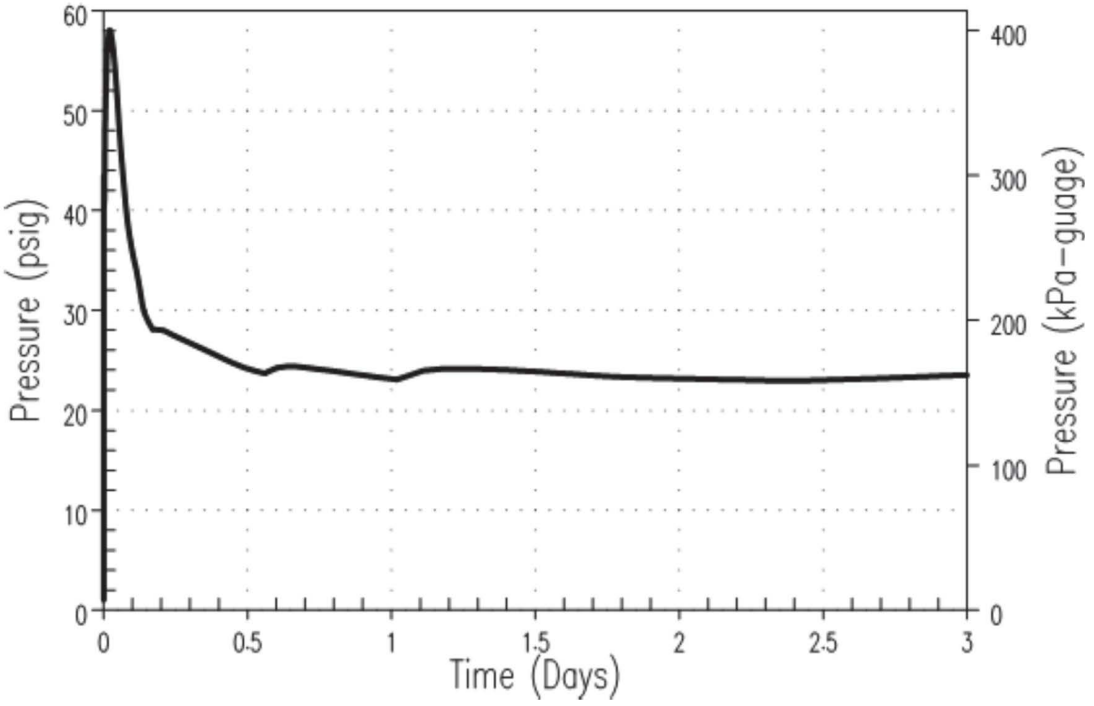

Figure 6.2.1.1-7
AP1000 Containment Pressure Response for DECLG LOCA – 3 Days

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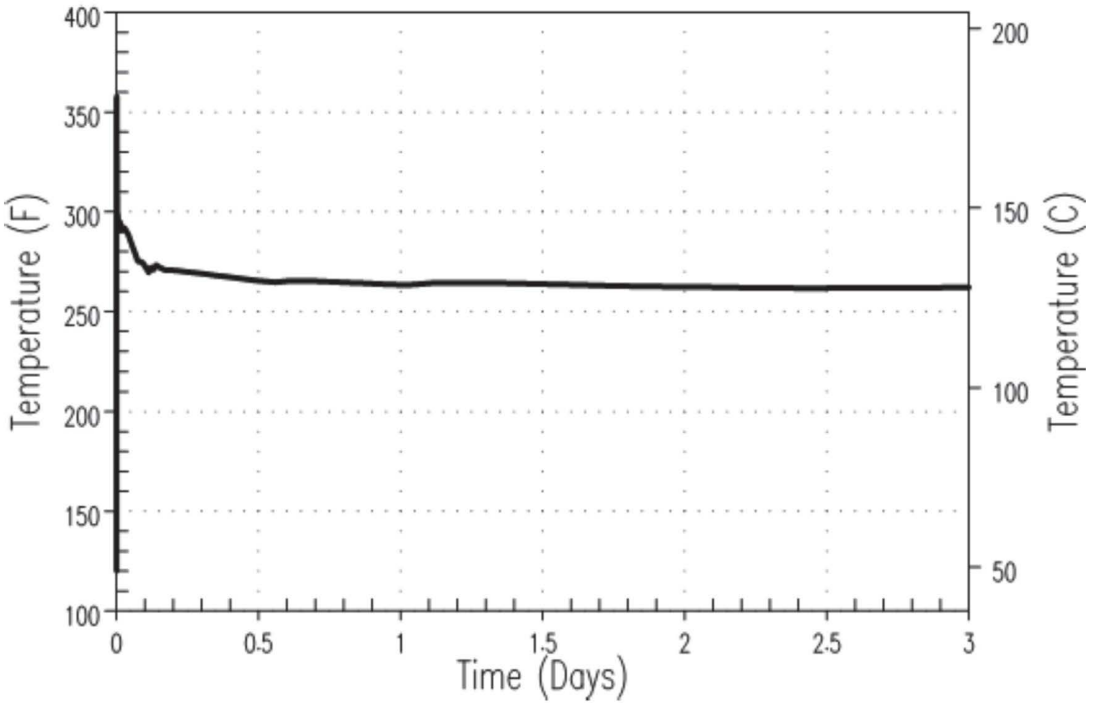

Figure 6.2.1.1-8
AP1000 Containment Temperature Response for DECLG LOCA – 3 Days

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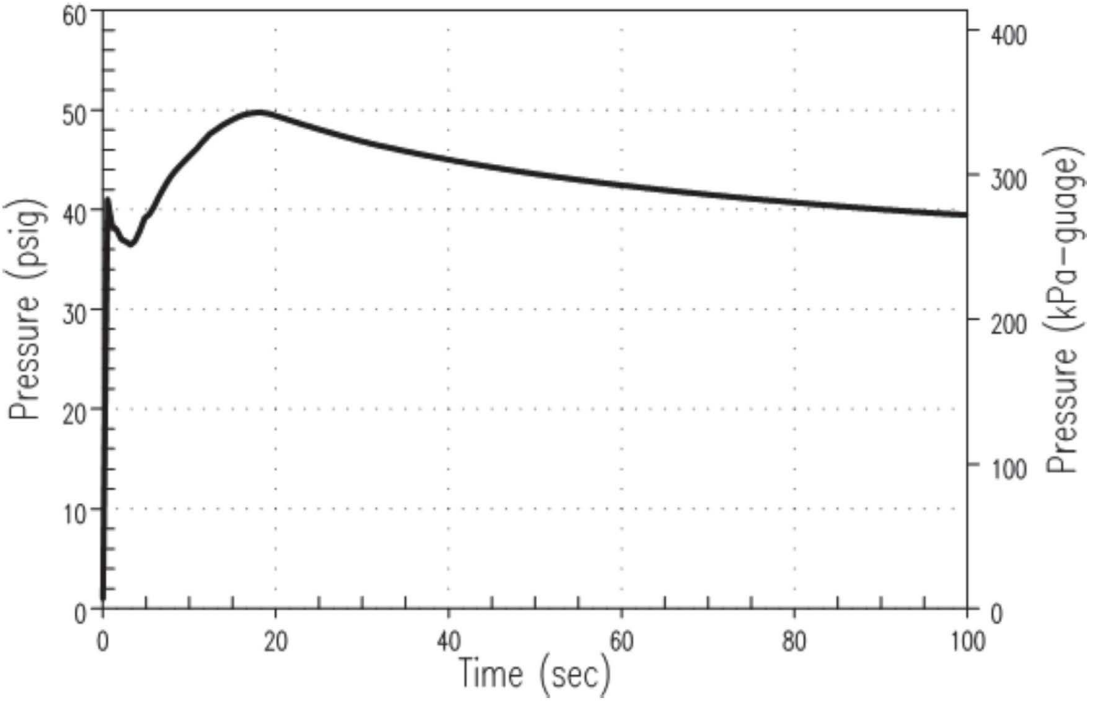

Figure 6.2.1.1-9
AP1000 Containment Pressure Response – DEHLG LOCA

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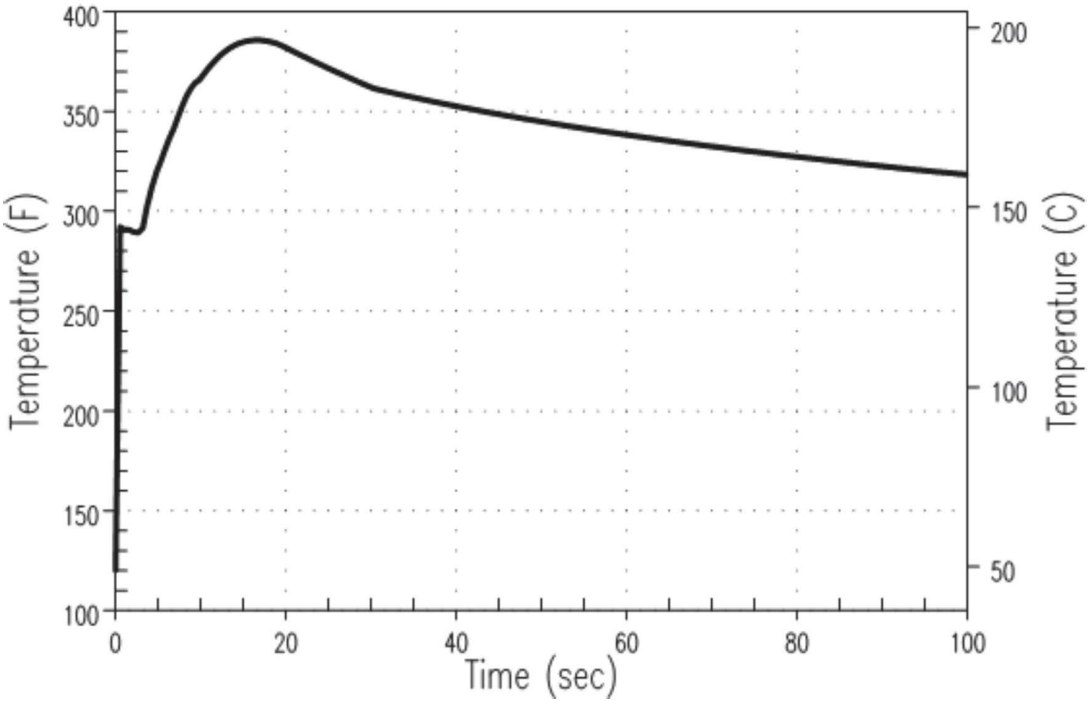

Figure 6.2.1.1-10
AP1000 Containment Temperature Response for DEHLG LOCA

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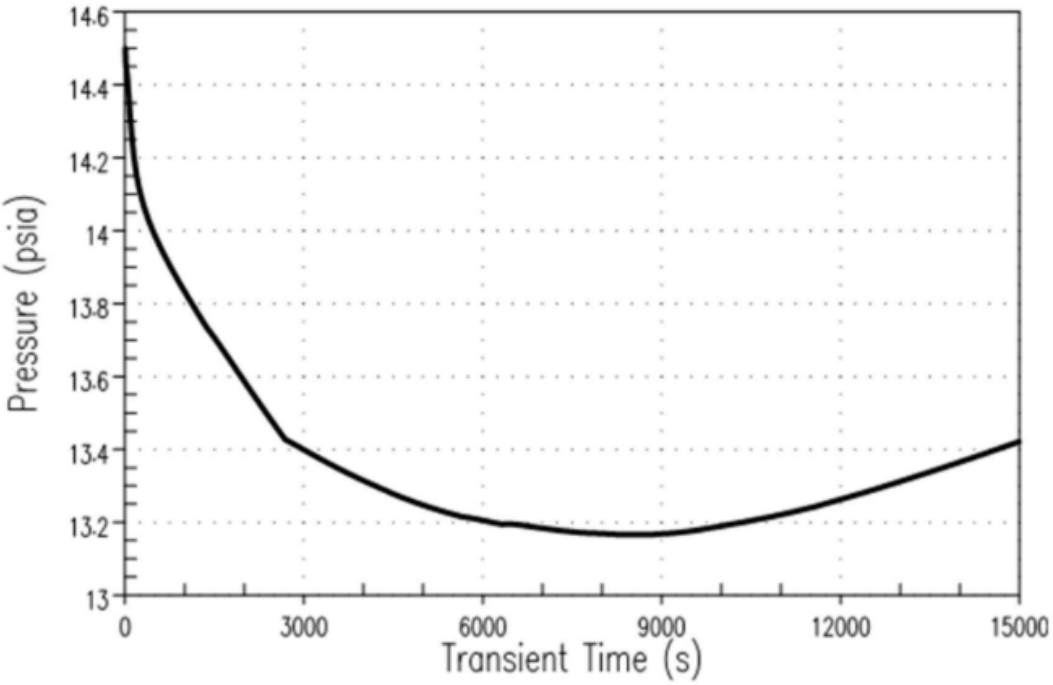

Figure 6.2.1.1-11
AP1000 Design External Pressure Analysis
Containment Pressure vs. Time

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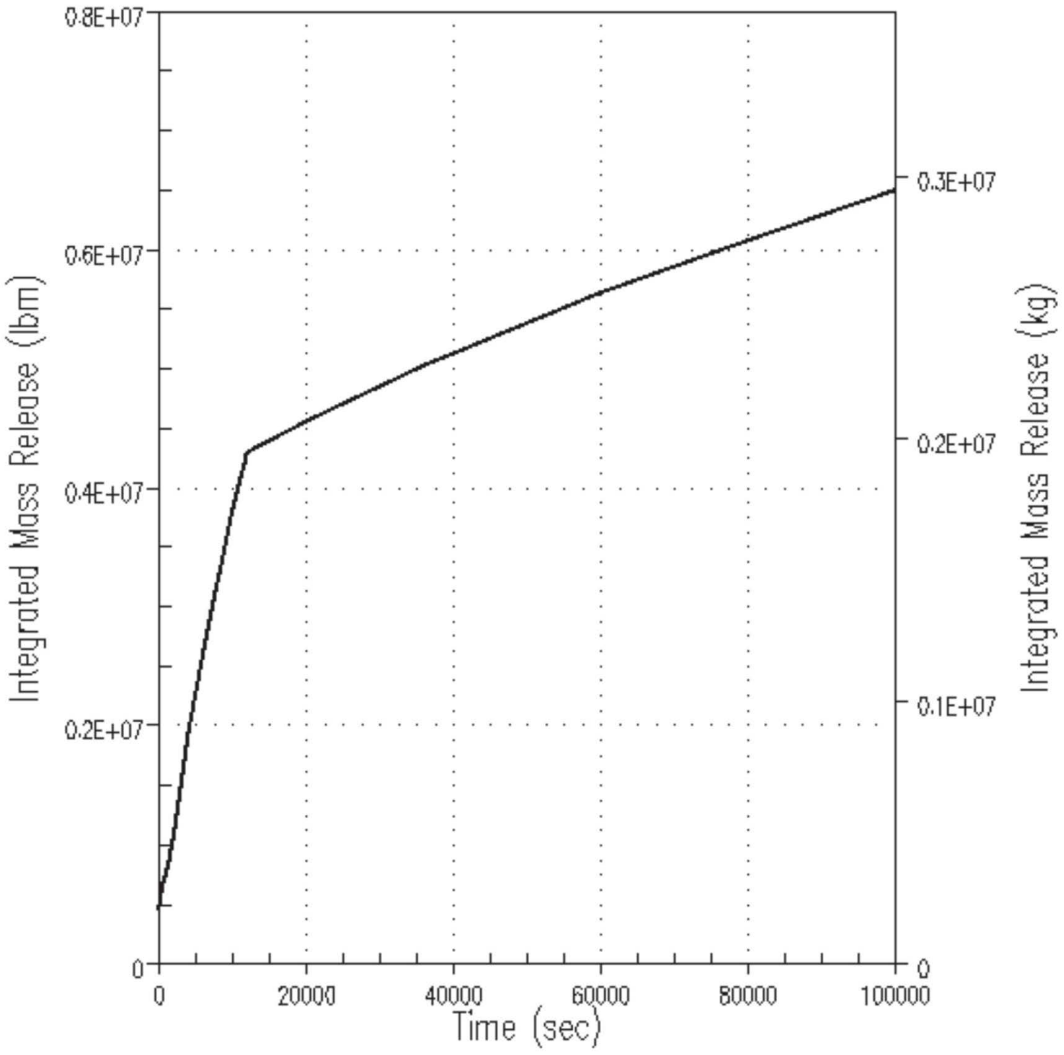

Figure 6.2.1.3-1
AP1000 DECLG Integrated Break Flow

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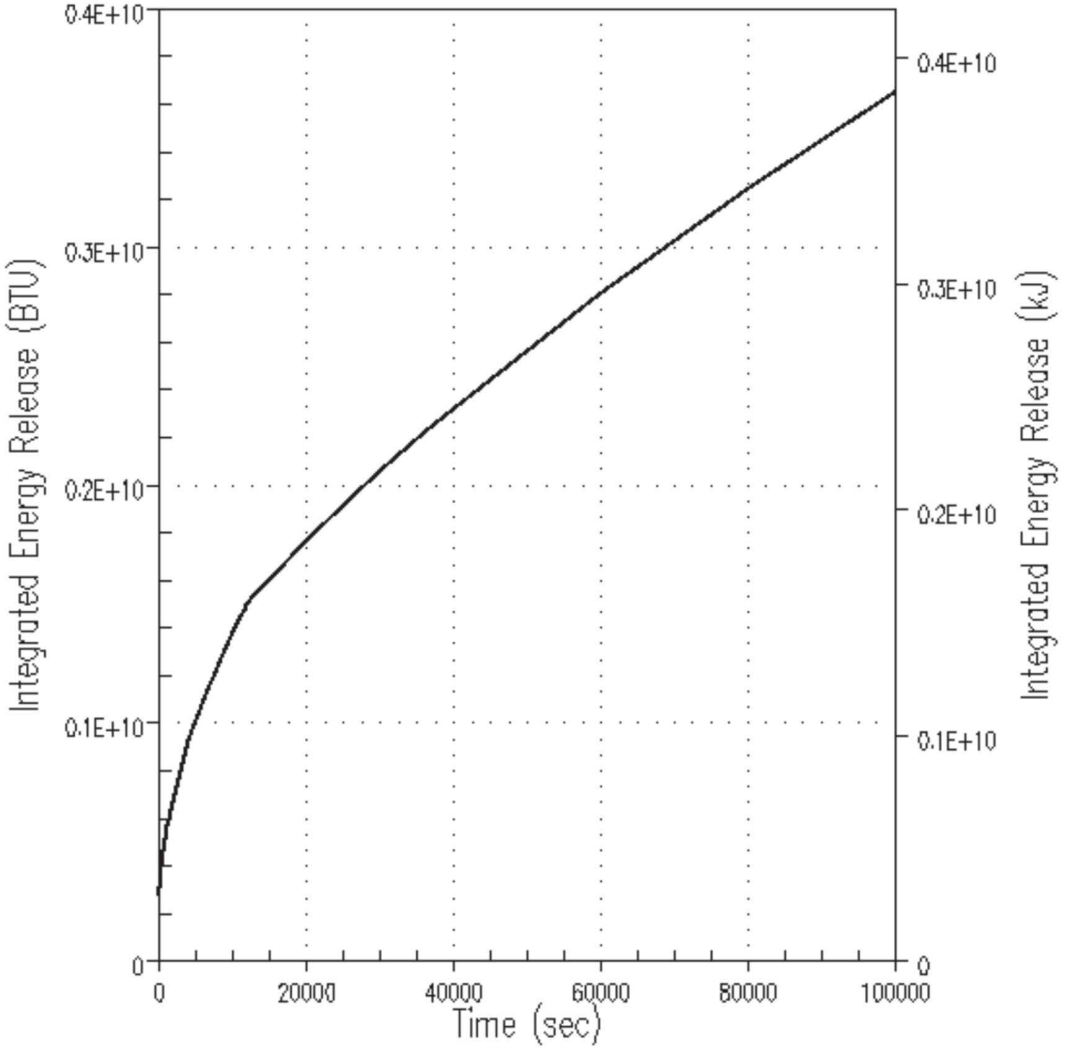

Figure 6.2.1.3-2
AP1000 DECLG LOCA Integrated Energy Released

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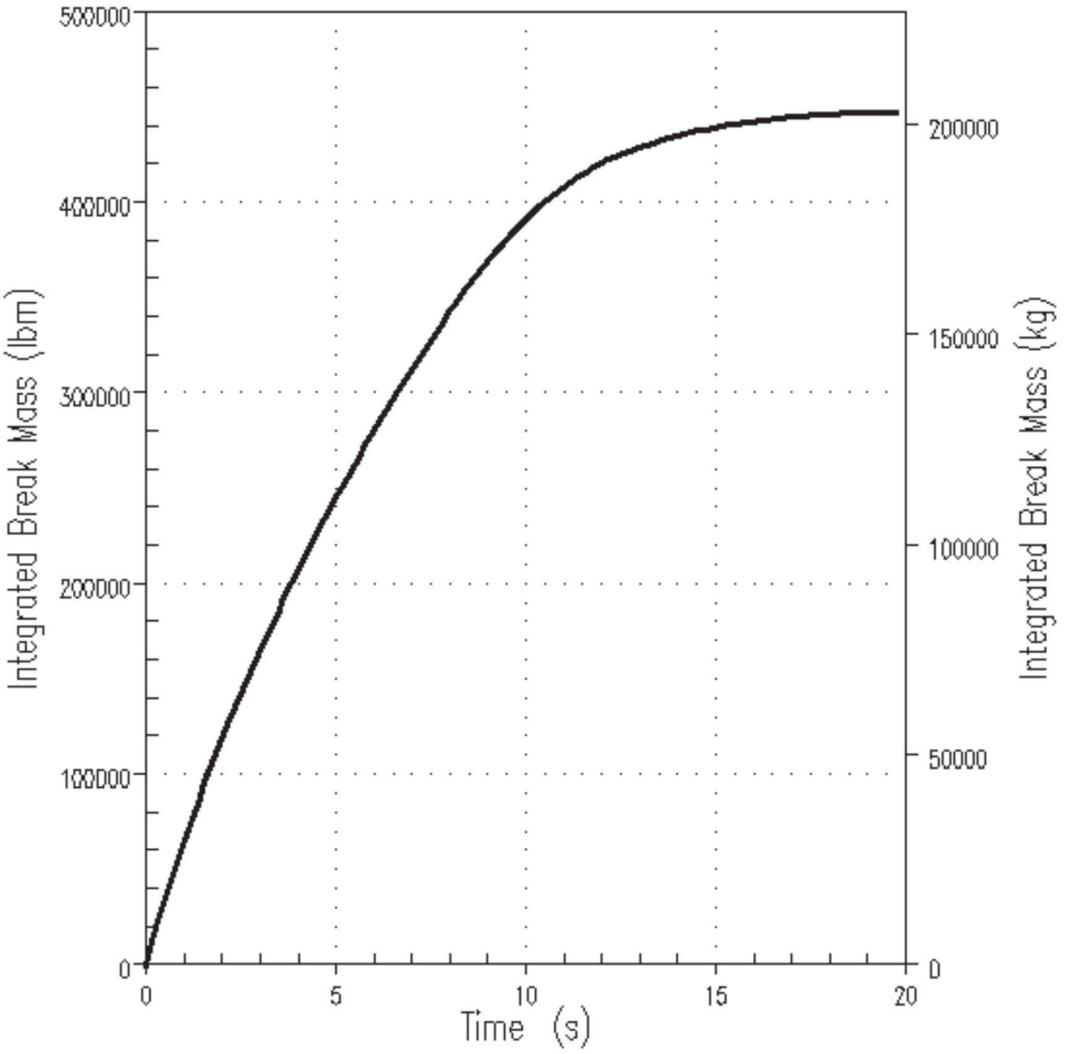

Figure 6.2.1.3-3
AP1000 DEHLG Integrated Break Flow

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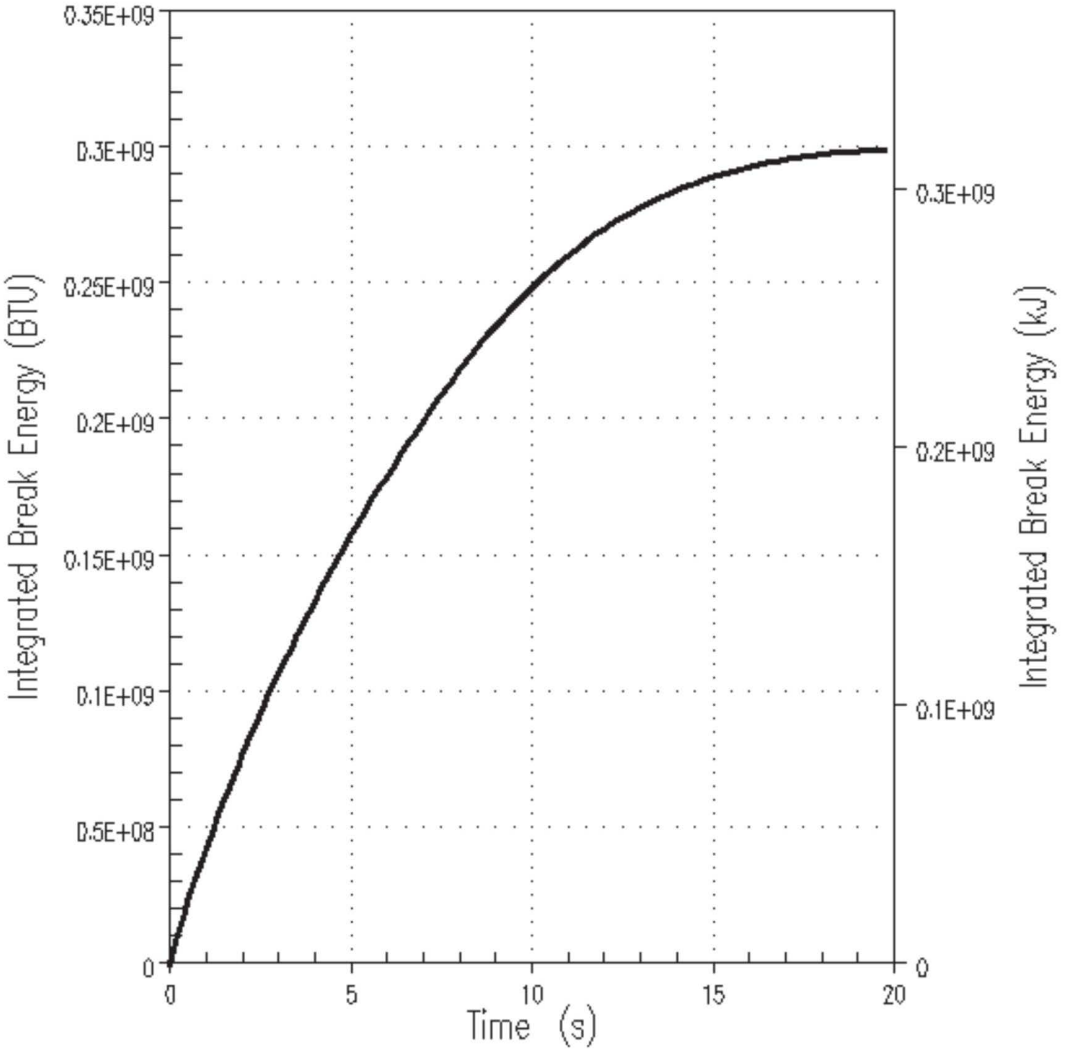

Figure 6.2.1.3-4
AP1000 DEHLG LOCA Integrated Energy Released

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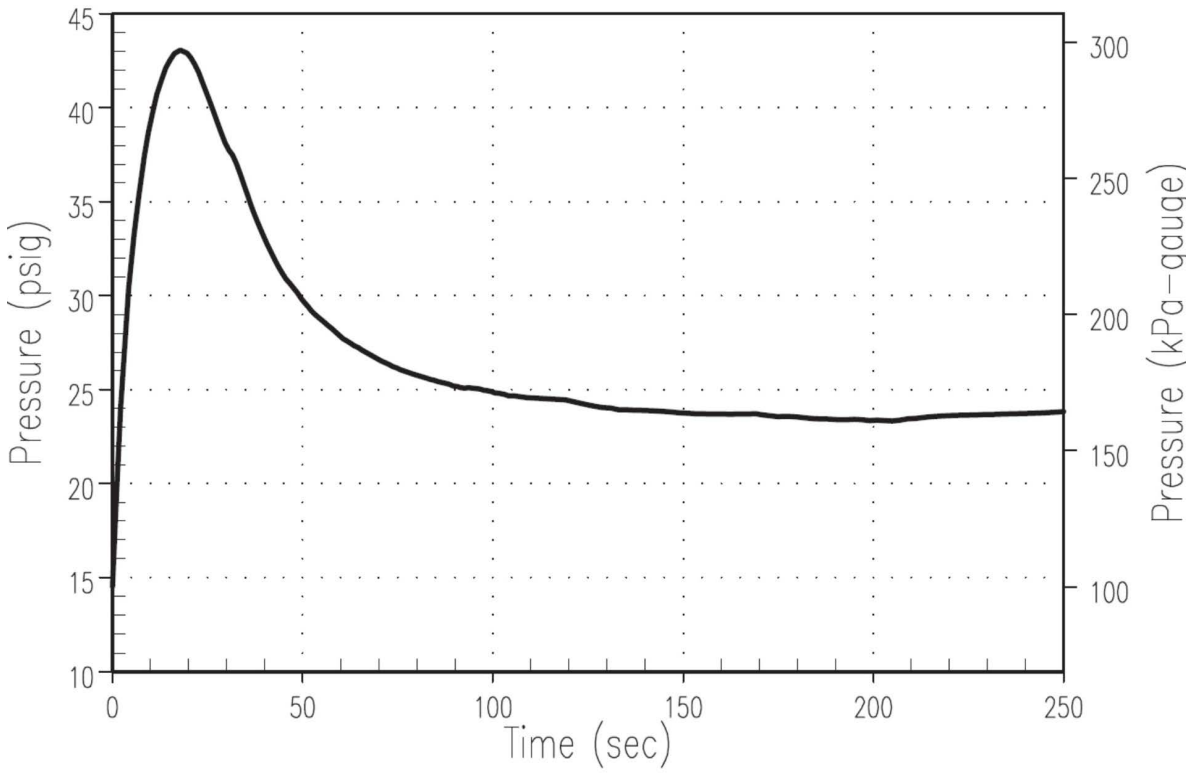

Figure 6.2.1.5-1
AP1000 Minimum Containment Pressure for DECLG LOCA

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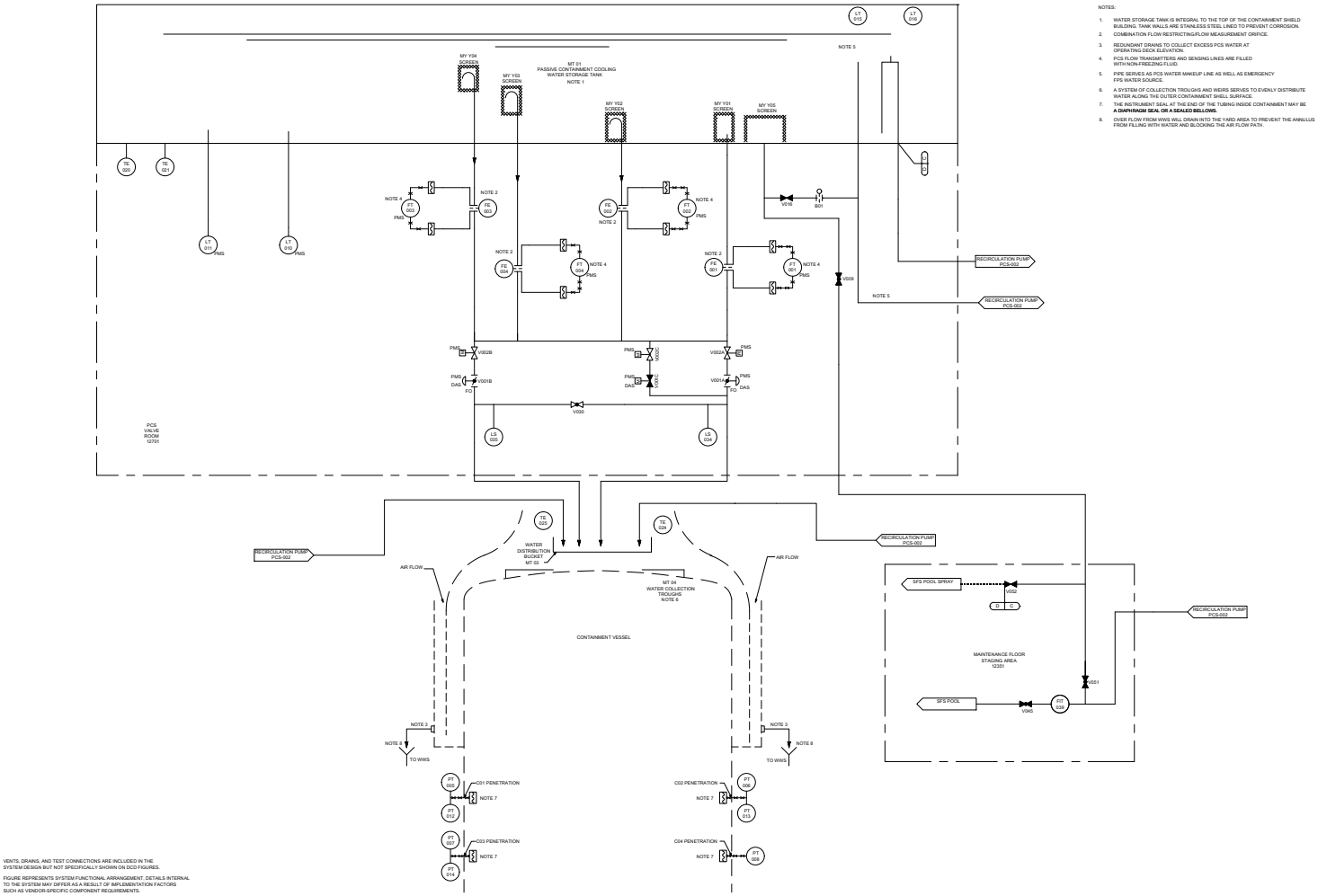

Figure 6.2.2-1 (Sheet 1 of 2)
Simplified Passive Containment Cooling System Piping and Instrumentation Diagram
(REF) PCS 001

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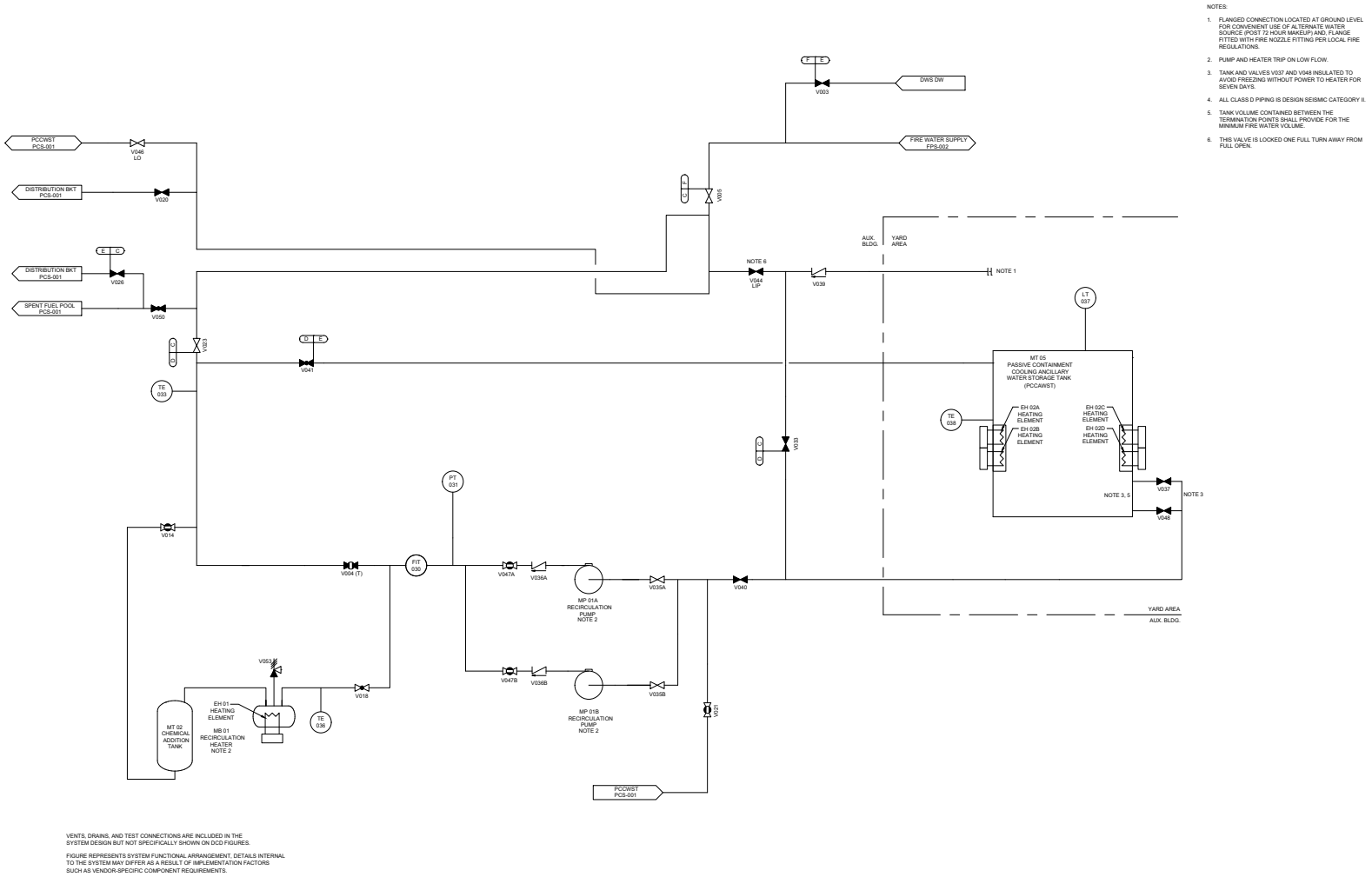

Figure 6.2.2-1 (Sheet 2 of 2)
Simplified Passive Containment Cooling System Piping and Instrumentation Diagram (REF) PCS 002

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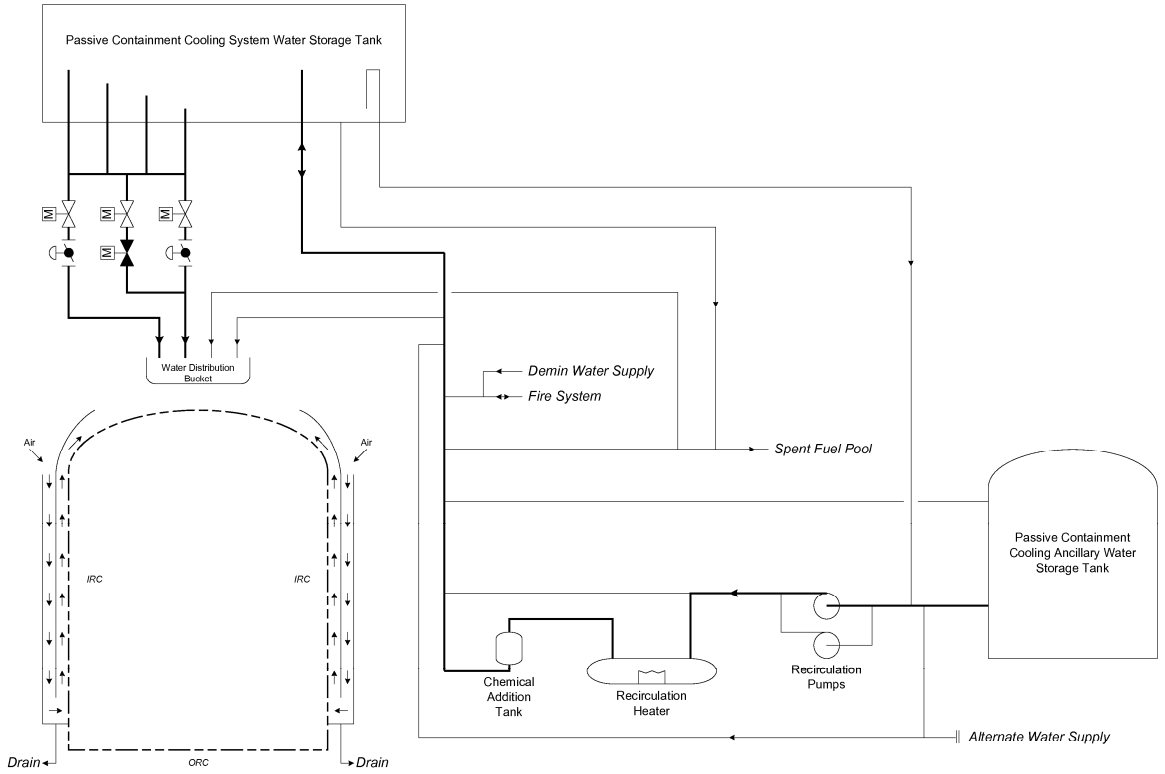

Figure 6.2.2-2
Simplified Sketch of Passive Containment Cooling System

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Figures 6.2.4-1–6.2.4-4 Not Used

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Figure 6.2.4-5
Hydrogen Igniter Locations – Section View

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Figure 6.2.4-6
Hydrogen Igniter Locations
Plan View Elevation 82′-6″

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Figure 6.2.4-7
Hydrogen Igniter Locations – Section View

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Figure 6.2.4-8
Hydrogen Igniter Locations
Plan View Elevation 96′-6″

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Figure 6.2.4-9
Hydrogen Igniter Locations
Plan View Elevation 118′-6″

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Figure 6.2.4-10
Hydrogen Igniter Locations
Plan View Elevation 135′-3″

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Figure 6.2.4-11
Hydrogen Igniter Locations
Plan View Elevation 162′-0″

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Figure 6.2.4-12
Hydrogen Igniter Locations
Plan View Elevation 210′-0″

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Figure 6.2.4-13
Hydrogen Igniter Locations Section A-A

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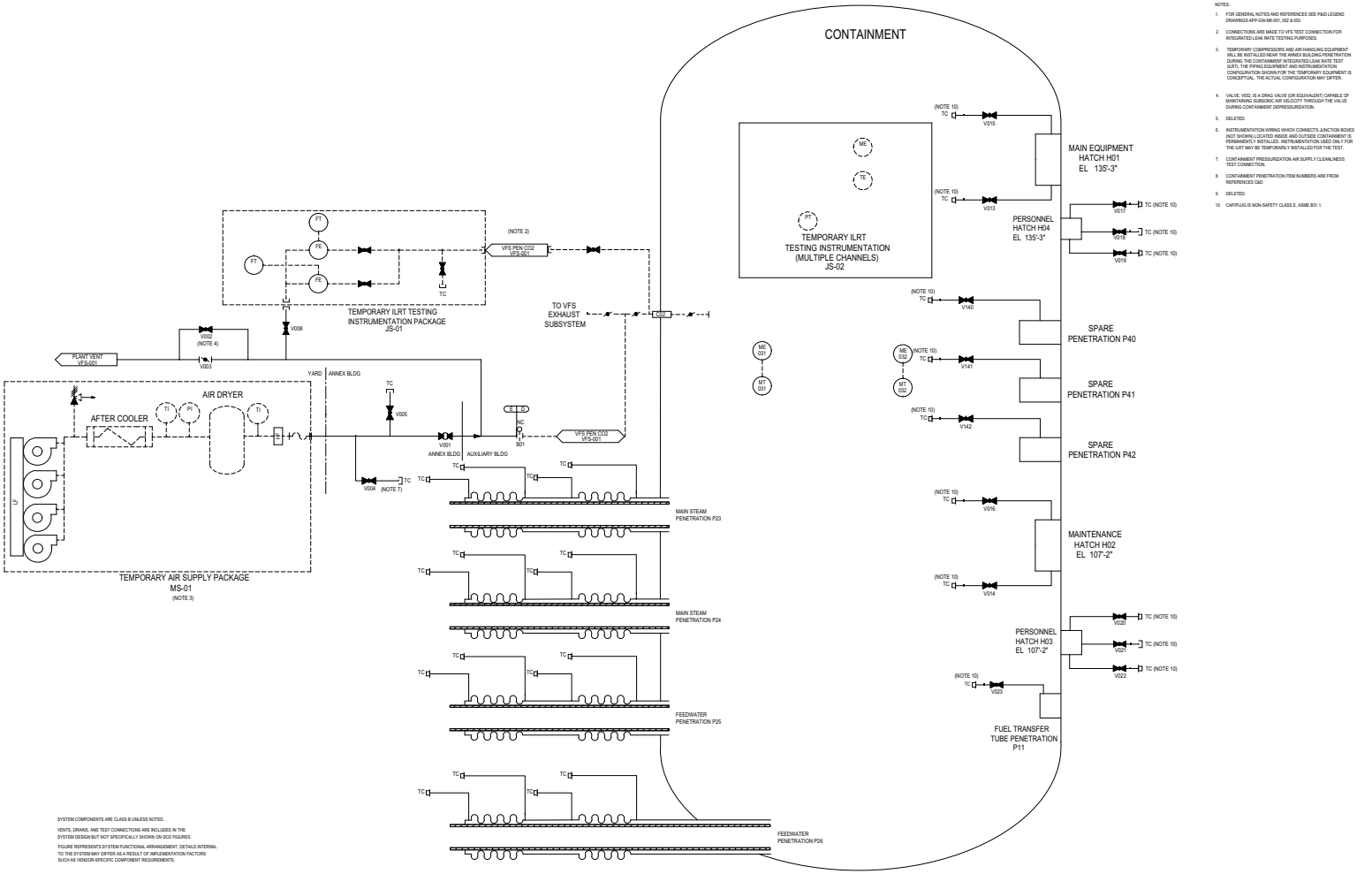

Figure 6.2.5-1
Simplified Containment Leak Rate Test System Piping and Instrumentation Diagram (REF) VUS 001

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6.3 Passive Core Cooling System

The primary function of the passive core cooling system is to provide emergency core cooling following postulated design basis events. To accomplish this primary function, the passive core cooling system is designed to perform the following functions:

- Emergency core decay heat removal

  Provide core decay heat removal during transients, accidents or whenever the normal heat removal paths are lost. This heat removal function is available at reactor coolant system conditions including shutdowns. During refueling operations, when the IRWST is drained into the refueling cavity, other passive means of core decay heat removal are utilized. Subsection 6.3.3.4.4 provides a description of how this is accomplished.

- Reactor coolant system emergency makeup and boration

  Provide reactor coolant system makeup and boration during transients or accidents when the normal reactor coolant system makeup supply from the chemical and volume control system is unavailable or is insufficient.

- Safety injection

  Provide safety injection to the reactor coolant system to provide adequate core cooling for the complete range of loss of coolant accidents, up to and including the double-ended rupture of the largest primary loop reactor coolant system piping.

- Containment pH control

  Provide for chemical addition to the containment during post-accident conditions to establish floodup chemistry conditions that support radionuclide retention with high radioactivity in containment and to prevent corrosion of containment equipment during long-term floodup conditions.

The passive core cooling system is designed to operate without the use of active equipment such as pumps and ac power sources. The passive core cooling system depends on reliable passive components and processes such as gravity injection and expansion of compressed gases. The passive core cooling system does require a one-time alignment of valves upon actuation of the specific components.

6.3.1 Design Basis

The passive core cooling system is designed to perform its safety-related functions based on the following considerations:

- It has component redundancy to provide confidence that its safety-related functions are performed, even in the unlikely event of the most limiting single failure occurring coincident with postulated design basis events.

- Components are designed and fabricated according to industry standard quality groups commensurate with its intended safety-related functions.

- It is tested and inspected at appropriate intervals, as defined by the ASME Code, Section XI, and by technical specifications.

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- It performs its intended safety-related functions following events such as fire, internal missiles or pipe breaks.
- It is protected from the effects of external events such as earthquakes, tornadoes, and floods.
- It is designed to be sufficiently reliable, considering redundancy and diversity, to support the plant core melt frequency and significant release frequency goals.

6.3.1.1 Safety Design Basis

The passive core cooling system is designed to provide emergency core cooling during events involving increases and decreases in secondary side heat removal and decreases in reactor coolant system inventory. Subsection 6.3.3 provides a description of the design basis events. The performance criteria are provided in Subsection 6.3.1 and also described in Chapter 15, under the respective event sections.

6.3.1.1.1 Emergency Core Decay Heat Removal

For postulated non-LOCA events, where a loss of capability to remove core decay heat via the steam generators occurs, the passive core cooling system is designed to perform the following functions for at least 72 hours:

- The passive residual heat removal heat exchanger automatically actuates to provide reactor coolant system cooling and to prevent water relief through the pressurizer safety valves.
- The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate collection features, and the passive containment cooling system, is designed to remove decay heat following a design basis event. Automatic depressurization actuation is not expected, but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to Subsection 6.3.1.1.4).
- The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions following a non-LOCA event. The applicable post-accident safety evaluation criteria are discussed in Chapter 15.
- The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.
- During a steam generator tube rupture event, the passive residual heat removal heat exchanger removes core decay heat and reduces reactor coolant system temperature and pressure, equalizing with steam generator pressure and terminating break flow, without overfilling the steam generator.

System operation beyond 72 hours is described in Subsection 6.3.1.2.1.

6.3.1.1.2 Reactor Coolant System Emergency Makeup and Boration

For postulated non-LOCA events, sufficient core makeup water inventory is automatically provided to keep the core covered and to allow for decay heat removal. In addition, this makeup prevents actuation of the automatic depressurization system for a significant time.

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For postulated events resulting in an inadvertent cooldown of the reactor coolant system, such as a steam line break, sufficient borated water is automatically provided to makeup for reactor coolant system shrinkage. The borated water also counteracts the reactivity increase caused by the resulting system cooldown.

For a Condition III steam line break described in Chapter 15, return to power is acceptable if there is no core damage. For this event, the automatic depressurization system is not actuated.

For a large steam line break, the peak return to power is limited so that the offsite dose limits are satisfied. Following either of these events, the reactor is automatically brought to a subcritical condition.

For safe shutdown, the passive core cooling system is designed to supply sufficient boron to the reactor coolant system to maintain the technical specification shutdown margin for cold, post-depressurization conditions, with the most reactive rod fully withdrawn from the core. The automatic depressurization system is not expected to actuate for these events.

6.3.1.1.3 Safety Injection

The passive core cooling system provides sufficient water to the reactor coolant system to mitigate the effects of a loss of coolant accident. In the event of a large loss of coolant accident, up to and including the rupture of a hot or cold leg pipe, where essentially all of the reactor coolant volume is initially displaced, the passive core cooling system rapidly refills the reactor vessel, refloods the core, and continuously removes the core decay heat. A large break is a rupture with a total cross-sectional area equal to or greater than one square foot. Although the criteria for mechanistic pipe break are used to limit the size of pipe rupture considered in the design and evaluation of piping systems, as described in Subsection 3.6.3, such criteria are not used in the design of the passive core cooling system.

Sufficient water is provided to the reactor vessel following a postulated loss of coolant accident so that the performance criteria for emergency core cooling systems, described in Chapter 15, are satisfied.

The automatic depressurization system valves, provided as part of the reactor coolant system, are designed so that together with the passive core cooling system they:

- Satisfy the small loss of coolant accident performance requirements
- Provide effective core cooling for loss of coolant accidents from when the passive core cooling system is actuated through the long-term cooling mode.

6.3.1.1.4 Safe Shutdown

The functional requirements for the passive core cooling system specify that the plant be brought to a safe, stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in Subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the passive core cooling system, in conjunction with the passive containment cooling system and the automatic depressurization system, has the capability to establish long-term safe shutdown conditions in the reactor coolant system as identified in Subsection 7.4.1.1.

The core makeup tanks automatically provide injection to the reactor coolant system after they are actuated on low reactor coolant temperature or low pressurizer pressure or level. The passive core

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cooling system can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of normal systems. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available.

In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system automatically actuates. However, after the initial plant cooldown following a non-LOCA event, operators assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend the passive residual heat removal heat exchanger operation by de-energizing the loads on the 24-hour Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to maintain safe, stable conditions for at least 72 hours. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

In most sequences, the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. For loss of coolant accidents, when the core makeup tank level reaches the automatic depressurization system actuation setpoint and other postulated events where the passive residual heat removal heat exchanger operation is not extended or is exhausted, the automatic depressurization system may be initiated. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 250°F within 24 hours. The passive core cooling system can maintain this safe shutdown condition indefinitely for the plant as identifed in Subsection 7.4.1.1.

The passive core cooling system functional requirements are met over the range of anticipated events and single failure assumptions. The primary function of the passive core cooling system during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Details of the safe shutdown design bases are presented in Subsection 5.4.7 and Section 7.4. The performance of the passive residual heat removal exchanger to bring the plant to 420°F in 36 hours is summarized in Subsection 19E.4.10.2.

6.3.1.1.5 Containment pH Control

The passive core cooling system is capable of maintaining the desired post-accident pH conditions in the recirculation water after containment floodup. The pH adjustment is capable of maintaining containment pH within a range of 7.0 to 9.5, to enhance radionuclide retention in the containment and to prevent stress corrosion cracking of containment components during long-term containment floodup.

6.3.1.1.6 Reliability Requirements

The passive core cooling system satisfies a variety of reliability requirements, including redundancy (such as for components, power supplies, actuation signals, and instrumentation), equipment testing to confirm operability, procurement of qualified components, and provisions for periodic maintenance. In addition, the system provides protection in a number of areas including:

- Single active and passive component failures
- Spurious failures

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- Physical damage from fires, flooding, missiles, pipe whip, and accident loads
- Environmental conditions such as high-temperature steam and containment floodup

Subsection 6.3.1.3 includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Post Accident Core Decay Heat Removal

The passive residual heat removal heat exchanger is designed to cool the reactor coolant system to 420°F in 36 hours, with or without reactor coolant pumps operating. This allows the reactor coolant system to be depressurized and the stress in the reactor coolant system and connecting piping to be reduced to low levels. This non-bounding, conservative evaluation is discussed in Subsection 19E.4.10.2.

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features, and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition of 420°F for greater than 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators during the extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in Subsection 7.4.1.

Eventually, if pressurizer heaters are not available, the pressurizer subcools due to ambient heat loss. When this happens, the steam void within the pressurizer is transferred to the reactor coolant system. It has been determined that this condition is safe as long as the passive residual heat removal performance is not affected.

If passive residual heat removal performance is affected by subcooling (or other plant conditions) and non-safety systems to control core cooling are not reestablished, then the final, long-term safe shutdown conditions may be achieved and maintained using the automatic depressurization system as discussed in Subsection 7.4.1.1.

6.3.1.2.2 PRHR HX Modulation Control on Loss of Startup Feedwater

Should the main feedwater (MFW) and SFW be unavailable, the PRHR HX is used for decay heat removal. Nonsafety-related PLS PRHR HX modulation control system opens one of the two PRHR outlet valves to throttle PRHR flow. The PRHR outlet valve used for the throttling function can be manually selected by the operator. This non-safety defense in depth function provides protection against an unnecessary safety-related PRHR actuation resulting in full PRHR flow. If Low-2 narrow range SG level and Low-2 SFW flow are present in at least one SG, and at least one RCP is running in loop 1 (indicated by speed greater than the specified setpoint), then the automatic PRHR HX throttling actuation signal is latched in until manually reset by the operator. PRHR HX modulation control system then throttles flow with closed-loop feedback to maintain T_{cold} below the High T_{cold} setpoint that is used to fully actuate PRHR. The PRHR flow demand is also inversely proportional to SFW flow to automatically transition off the PRHR throttling if SFW becomes available. If PMS PRHR HX actuation logic is met both PRHR valves actuate to the full open position.

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6.3.1.3 Power Generation Design Basis

The passive core cooling system is designed to be sufficiently reliable to support the probabilistic risk analysis goals for core damage frequency and severe release frequency. In assessing the reliability for probabilistic risk analysis purposes, more realistic analysis is used for both the passive core cooling system performance and for plant response.

In the event of a small loss of coolant accident, the passive core cooling system limits the increase in peak clad temperature and core uncovery with design basis assumptions. For pipe ruptures of eight-inch nominal diameter size and less, the passive core cooling system is designed to prevent core damage with best estimate assumptions.

The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour.

The frequency of automatic depressurization system actuation is limited to a low probability to reduce safety risks and to minimize plant outages. Equipment is located so that it is not flooded or it is designed so that it is not damaged by the flooding. Major plant equipment is designed for multiple occurrences without damage.

The pH control equipment is designed to minimize the potential for and the impact of inadvertent actuation.

The passive core cooling system is capable of supporting the required testing and maintenance, including capabilities to isolate and drain equipment.

6.3.2 System Design

The passive core cooling system is a seismic Category I, safety-related system. It consists of two core makeup tanks, two accumulators, the in-containment refueling water storage tank, the passive residual heat removal heat exchanger, pH adjustment baskets, and associated piping, valves, instrumentation, and other related equipment. The automatic depressurization system valves and spargers, which are part of the reactor coolant system, also provide important passive core cooling functions.

The passive core cooling system is designed to provide adequate core cooling in the event of design basis events. The redundant onsite safety-related class 1E dc and UPS system provides power such that protection is provided for a loss of ac power sources, coincident with an event, assuming a single failure has occurred.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Figure 6.3-1 shows the piping and instrumentation drawings of the passive core cooling system. Simplified flow diagrams are shown in Figures 6.3-3 and 6.3-4. The accident analysis results of events analyzed in Chapter 15 provide a summary of the expected fluid conditions in the passive core cooling system for the various locations shown on the simplified flow diagrams, for the specific plant conditions identified -- safety injection and decay heat removal.

The passive core cooling system is designed to supply the core cooling flow rates to the reactor coolant system specified in Chapter 15 for the accident analyses. The accident analyses flow rates and heat removal rates are calculated by assuming a range of component parameters, including best estimate and conservatively high and low values.

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The passive core cooling system design is based on the six major components, listed in Subsection 6.3.2.2, that function together in various combinations to support the four passive core cooling system functions:

- Emergency decay heat removal
- Emergency reactor makeup/boration
- Safety injection
- Containment pH control

6.3.2.1.1 Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions

For events not involving a loss of coolant, the emergency core decay heat removal is provided by the passive core cooling system via the passive residual heat removal heat exchanger. The heat exchanger consists of a bank of C-tubes, connected to a tubesheet and channel head arrangement at the top (inlet) and bottom (outlet). The passive residual heat removal heat exchanger connects to the reactor coolant system through an inlet line from one reactor coolant system hot leg (through a tee from one of the fourth stage automatic depressurization lines) and an outlet line to the associated steam generator cold leg plenum (reactor coolant pump suction).

The inlet line is normally open and connects to the upper passive residual heat removal heat exchanger channel head. The inlet line is connected to the top of the hot leg and is routed continuously upward to the high point near the heat exchanger inlet. The normal water temperature in the inlet line will be hotter than the discharge line.

The outlet line contains normally closed air-operated valves that open on loss of air pressure or on control signal actuation. The alignment of the passive residual heat removal heat exchanger (with a normally open inlet motor-operated valve and normally closed outlet air-operated valves) maintains the heat exchanger full of reactor coolant at reactor coolant system pressure. The water temperature in the heat exchanger is about the same as the water in the in-containment refueling water storage tank, so that a thermal driving head is established and maintained during plant operation.

The heat exchanger is elevated above the reactor coolant system loops to induce natural circulation flow through the heat exchanger when the reactor coolant pumps are not available. The passive residual heat removal heat exchanger piping arrangement also allows actuation of the heat exchanger with reactor coolant pumps operating. When the reactor coolant pumps are operating, they provide forced flow in the same direction as natural circulation flow through the heat exchanger. If the pumps are operating and subsequently trip, then natural circulation continues to provide the driving head for heat exchanger flow.

The heat exchanger is located in the in-containment refueling water storage tank, which provides the heat sink for the heat exchanger.

Although gas accumulation is not expected, there is a vertical pipe stub on the top of the inlet piping high point that serves as a gas collection chamber. Level detectors indicate when gases have collected in this area. There are provisions to allow the operators to open manual valves to locally vent these gases to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features, and the passive containment cooling system, can provide core cooling for at least 72 hours. After the in-containment refueling water storage tank water

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reaches its saturation temperature (in several hours), the process of steaming to the containment initiates. Containment pressure increases as steam is released from the in-containment refueling water storage tank. As containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the incontainment refueling water storage tank drains to the containment sump.

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near the operating deck elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank. The gutter normally drains to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for greater than 14 days.

The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system condition. It transfers decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment. This occurs after in-containment refueling water storage tank saturation is reached and steaming to containment initiates.

The duration the passive residual heat removal heat exchanger can continue to remove decay heat is affected by the efficiency of the return of condensate to the in-containment refueling water storage tank. The in-containment refueling water storage tank water level is affected by the amount of steam that leaves the tank and does not return. Resources are typically recovered within 72 hours, which allows the operators to place active, defense-in-depth systems into service and to terminate passive system operation. If resources are not recovered within this time frame, cooling can be extended as described in Subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event.

6.3.2.1.2 Reactor Coolant System Emergency Makeup and Boration

The core makeup tanks provide reactor coolant system makeup and boration during events not involving loss of coolant when the normal makeup system is unavailable or insufficient. There are two core makeup tanks located inside the containment at an elevation slightly above the reactor coolant loops. During normal operation, the core makeup tanks are completely full of cold, borated water. The boration capability of these tanks provides adequate core shutdown margin following a steam line break.

The core makeup tanks are connected to the reactor coolant system through a discharge injection line and an inlet pressure balance line connected to a cold leg. The discharge line is blocked by two normally closed, parallel air-operated isolation valves that open on a loss of air pressure or electrical power, or on control signal actuation. The core makeup tank discharge isolation valves are diverse from the passive residual heat removal heat exchanger outlet isolation valves discussed above. They use different valve body styles and different air operator types.

The pressure balance line from the cold leg is normally open to maintain the core makeup tanks at reactor coolant system pressure, which prevents water hammer upon initiation of core makeup tank injection.

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The cold leg pressure balance line is connected to the top of the cold leg and is routed continuously upward to the high point near the core makeup tank inlet. The normal water temperature in this line will be hotter than the discharge line.

The outlet line from the bottom of each core makeup tank provides an injection path to one of the two direct vessel injection lines, which are connected to the reactor vessel downcomer annulus. Upon receipt of a safeguards actuation signal, the two parallel valves in each discharge line open to align the associated core makeup tank to the reactor coolant system.

There are two operating processes for the core makeup tanks, steam-compensated injection and water recirculation. During steam-compensated injection, steam is supplied to the core makeup tanks to displace the water that is injected into the reactor coolant system. This steam is provided to the core makeup tanks through the cold leg pressure balance line. The cold leg line only has steam flow if the cold legs are voided.

During water recirculation, hot water from the cold leg enters the core makeup tanks, and the cold water in the tank is discharged to the reactor coolant system. This results in reactor coolant system boration and a net increase in reactor coolant system mass.

The operating process for the core makeup tanks depends on conditions in the reactor coolant system, primarily voiding in the cold leg. When the cold leg is full of water, the cold leg pressure balance line remains full of water and the injection occurs via water recirculation. If reactor coolant system inventory decreases sufficiently to cause cold leg voiding, then steam flows through the cold leg balance lines to the core makeup tanks.

Following an event such as steam-line break, the reactor coolant system experiences a decrease in temperature and pressure due to an increase of energy removed by the secondary system as a consequence of the break. The cooldown results in a reduction of the core shutdown margin due to the negative moderator temperature coefficient. There is a potential return to power, assuming the most reactive rod cluster control assembly is stuck in its fully withdrawn position. The actuation of the core makeup tanks following this event provides injection of borated water via water recirculation to mitigate the reactivity transient and provide the required shutdown margin.

In case of a steam generator tube rupture, core makeup tank injection together with the steam generator overfill prevention logic terminates the reactor coolant system leak into the steam generator. This occurs without actuation of the automatic depressurization system and without operator action. In a steam generator tube rupture, the core makeup tanks operate in the water recirculation mode to provide borated water to compensate for reactor coolant system inventory losses and to borate the reactor coolant system. In case of a leak rate of 10 gallons per minute, the passive core cooling system can delay the automatic depressurization system actuation for at least 10 hours while providing makeup water to the reactor coolant system. After the actuation of the automatic depressurization system, the passive core cooling system provides sufficient borated water to compensate for reactor coolant system shrinkage and to provide the reactor coolant system boration.

6.3.2.1.3 Safety Injection During Loss of Coolant Accidents

The passive core cooling system uses four different sources of passive injection during loss of coolant accidents.

- Accumulators provide a very high flow for a limited duration of several minutes.
- The core makeup tanks provide a relatively high flow for a longer duration.

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- The in-containment refueling water storage tank provides a lower flow, but for a much longer time.

- The containment is the final long-term source of water. It becomes available following the injection of the other three sources and floodup of containment.

The operation of the core makeup tanks is described in the Subsection 6.3.2.1.2. During a loss of coolant accident, they provide injection rates commensurate with the severity of the loss of coolant accident. For a larger loss of coolant accident, and after the automatic depressurization system has been actuated, the cold legs are expected to be voided. In this situation, the core makeup tanks operate at their maximum injection rate with steam entering the core makeup tanks through the cold leg pressure balance lines.

Downstream of the parallel discharge isolation valves, the core makeup tank discharge line contains two check valves, in series, that normally remain open with or without flow in the line. These valves prevent reverse flow through this line, from the accumulator, that would bypass the reactor vessel in the event of a larger loss of coolant accident in the cold leg or the cold leg pressure balance line.

For smaller loss of coolant accidents the core makeup tanks initially operate in the water recirculation mode since the cold legs are water filled. During this water recirculation, the core makeup tanks remain full, but the cold, borated water is purged with hot, less borated cold leg water. The water recirculation provides reactor coolant system makeup and also effectively borates the reactor coolant system. As the accident progresses, when the cold legs void, the core makeup tanks switch to the steam displacement mode which provides higher flow rates.

The two accumulators contain borated water and a compressed nitrogen cover gas to provide rapid injection. They are located inside the reactor containment and the discharge from each tank is connected to one of the direct vessel injection lines. These lines connect to the reactor vessel downcomer. A deflector in the annulus directs the water flow downward to minimize core bypass flow. The water and gas volumes and the discharge line resistance provide several minutes of injection in a large loss of coolant accident.

The in-containment refueling water storage tank is located in the containment at an elevation slightly above the reactor coolant system loop piping. Reactor coolant system injection is possible only after the reactor coolant system has been depressurized by the automatic depressurization system or by a loss of coolant accident. Squib valves in the in-containment refueling water storage tank injection lines open automatically on a 4th stage automatic depressurization signal. Check valves, arranged in series with the squib valves, open when the reactor pressure decreases to below the in-containment refueling water storage tank injection head.

After the accumulators, core makeup tanks, and the in-containment refueling water storage tank inject, the containment is flooded up to a level sufficient to provide recirculation flow through the gravity injection lines back into the reactor coolant system.

The time that it takes until the initiation of containment recirculation flow varies greatly, depending on the specific event. With a break in a direct vessel injection line, the in-containment refueling water storage tank spills out through the break and floods the containment, along with reactor coolant system leakage, and recirculation can occur in several hours. In the event of automatic depressurization without a reactor coolant system break and with condensate return, the in-containment refueling water storage tank level decreases very slowly. Recirculation may not initiate for 72 hours. Containment recirculation may be impacted by shutdown maintenance activities which remove access hatches or curbs located at the 107'-2" plant elevation. Subsection 3.4.1.2.2.1 discusses containment flooding events in further detail.

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Containment recirculation initiates when the recirculation line valves are open and the containment floodup level is sufficiently high. When the in-containment refueling water storage tank level decreases to the Low-3 level, the containment recirculation squib valves automatically open to provide redundant flow paths from the containment to the reactor.

These recirculation flow paths can also provide a suction flow path from the containment to the normal residual heat removal pumps, when they are operating after containment flood up. In addition, the squib valves in the recirculation paths containing normally open motor-operated valves can be manually opened to intentionally drain the in-containment refueling water storage tank to the reactor cavity during severe accidents. This action is modeled in the AP1000 probabilistic risk assessment.

A range of break sizes and locations are analyzed to verify the adequacy of passive core cooling system injection. These events include a no-break case, a complete severance of one (eight-inch) direct vessel injection line case, and other smaller break cases. Successful reactor coolant system depressurization to in-containment refueling water storage tank injection is achieved, as shown in Chapter 15.

In larger loss of coolant accidents, including double ended ruptures in reactor coolant system piping, the passive core cooling system can provide a large flow rate, from the accumulators, to quickly refill the reactor vessel lower plenum and downcomer. The accumulators provide the required injection flow during the first part of the event including refilling the downcomer and lower plenum and partially reflooding the core. After the accumulators empty, the core makeup tanks complete the reflooding of the core. The subsequent in-containment refueling water storage tank injection and recirculation provide long-term cooling. Both injection lines are available since the injection lines are not the source of a large pipe break.

6.3.2.1.4 Containment pH Control

Control of the pH in the containment sump water post-accident is achieved through the use of pH adjustment baskets containing granulated trisodium phosphate (TSP). The baskets are located below the minimum post-accident floodup level, and chemical addition is initiated passively when the water reaches the baskets. The baskets are placed at least a foot above the floor to reduce the chance that water spills in containment will dissolve the TSP.

The TSP is designed to maintain the pH of the containment sump water in a range from 7.0 to 9.5. This chemistry reduces radiolytic formation of elemental iodine in the containment sump, consequently reducing the aqueous production of organic iodine, and ultimately reducing the airborne iodine in containment and offsite doses.

The chemical addition also helps to reduce the potential for stress corrosion cracking of stainless steel components in a post floodup condition, where chlorides can leach out of the containment concrete and potentially affect these components during a long-term floodup event.

6.3.2.1.5 Passive Core Cooling System Actuation

Table 6.3-1 lists the remotely actuated valves used by the various passive core cooling system components. The engineered safeguards features actuation signals used for these valves are described in Section 7.3. Table 6.3-1 shows the normal valve position, the valve position to actuate the associated component, and the failure position of the valve. The failed position represents the position that the valve fails upon loss of electrical power or other motive sources, such as instrument air.

Table 6.3-3 contains the failure mode and effects analysis of the passive core cooling system.

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6.3.2.2 Equipment and Component Descriptions

Table 6.3-2 contains a summary of equipment parameters for major components of the passive core cooling system.

6.3.2.2.1 Core Makeup Tanks

The two core makeup tanks are vertical, cylindrical tanks with hemispherical upper and lower heads. They are made of carbon steel, clad on the internal surfaces with stainless steel. The core makeup tanks are AP1000 Equipment Class A and are designed to meet seismic Category I requirements. They are located inside containment on the 107-foot floor elevation. The core makeup tanks are located above the direct vessel injection line connections to the reactor vessel, which are located at an elevation near the bottom of the hot leg.

During normal operation the core makeup tanks are completely filled with borated water and are maintained at reactor coolant system pressure by the cold leg pressure balance line. The temperature of the borated water in the core makeup tanks is about the same as the containment ambient temperature since the tanks are not insulated or heated.

The inlet line from the cold leg is sized for loss of coolant accidents, where the cold legs become voided and higher core makeup tank injection flows are required. The discharge line from each core makeup tank contains a flow-tuning orifice that provides a mechanism for the field adjustment of the injection line resistance. The orifice is used to establish the required flow rates assumed in the core makeup tank design. The core makeup tanks provide injection for an extended time after core makeup tank actuation. The duration of injection will be much longer when the core makeup tanks operate in the water recirculation mode as compared to the steam condensation mode.

Connections are provided for remotely adjusting the boron concentration of the borated water in each core makeup tank during normal plant operation, as required. Makeup water for the core makeup tank is provided by the chemical and volume control system. Samples from the core makeup tanks are taken periodically to check boron concentration.

Each core makeup tank has an inlet diffuser which is designed to reduce steam velocities entering the core makeup tank; thereby minimizing potential water hammer and reducing the amount of mixing that occurs during initial core makeup tank operation. The inlet diffuser flow area is ≥ 165 in^2.

The core makeup tanks are located inside the containment but outside the secondary shield wall. This facilitates maintenance and inspection.

Core makeup tank level and inlet and outlet line temperatures are monitored by indicators and alarms. The operator can take action as required to meet the technical specification requirements for core makeup tank operability. Core makeup tank temperature is monitored and trended to investigate potential leaks prior to alarm setpoints.

6.3.2.2.2 Accumulators

The two accumulators are spherical tanks made of carbon steel and clad on the internal surfaces with stainless steel. The accumulators are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. They are located inside the containment on the floor just below the core makeup tanks.

The accumulators are mostly filled with borated water and pressurized with nitrogen gas. The temperature of the borated water in the accumulators is about the same as the containment ambient temperature since the tanks are not insulated or heated. Each accumulator is connected to one of the

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direct vessel injection lines. During normal operation, the accumulator is isolated from the reactor coolant system by two check valves in series. When the reactor coolant system pressure falls below the accumulator pressure, the check valves open and borated water is forced into the reactor coolant system by the gas pressure. Mechanical operation of the check valves is the only action required to open the injection path from the accumulators to the core.

The accumulators are designed to deliver a high flow of borated water to the reactor vessel in the event of a large loss of coolant accident. This large flow rate is used to quickly establish core cooling following the large loss of reactor coolant system inventory.

The injection line from each accumulator contains a flow-tuning orifice that provides a mechanism for the field adjustment of the injection line resistance. The orifice is used to establish the required flow rates assumed in the accumulator design. The accumulator provides injection for several minutes after reactor coolant system pressure drops below the static accumulator pressure.

Connections are provided for remotely adjusting the level and boron concentration of the borated water in each accumulator during normal plant operation, as required. Accumulator water level may be adjusted either by draining or by pumping borated water from the chemical and volume control system to the accumulator. Samples from the accumulators are taken periodically to check the boron concentration.

Accumulator pressure is provided by a supply of nitrogen gas and can be adjusted as required during normal plant operation. However, the accumulators are normally isolated from the nitrogen supply. Gas relief valves on the accumulators protect them from overpressurization. The system also includes the capability to remotely vent gas from the accumulator, if required.

The accumulators are located inside the containment and outside the secondary shield wall. This facilitates maintenance and inspection.

Accumulator level and pressure are monitored by indication and alarms. The operator can take action, as required, to meet the technical specification requirements for accumulator operability.

6.3.2.2.3 In-Containment Refueling Water Storage Tank

The in-containment refueling water storage tank is a large, stainless-steel lined tank located underneath the operating deck inside the containment. The in-containment refueling water storage tank is AP1000 Equipment Class C and is designed to meet seismic Category I requirements. The tank is constructed as an integral part of the containment internal structures, and is isolated from the steel containment vessel except for the bottom portion of the tank wall, separated from the containment vessel by concrete. See Subsection 3.8.3 for additional information.

The bottom of the in-containment refueling water storage tank is above the reactor coolant system loop elevation so that the borated refueling water can drain by gravity into the reactor coolant system after it is sufficiently depressurized. The in-containment refueling water storage tank is connected to the reactor coolant system through both direct vessel injection lines. The in-containment refueling water storage tank contains borated water, at the existing temperature and pressure in containment.

Vents are installed in the roof of the in-containment refueling water storage tank. These vents are provided with normally closed covers in order to contain water vapor and radioactive gases within the tank during normal operation and to prevent debris from entering the tank from the containment operating deck. The vent covers open with a slight pressurization of the in-containment refueling water storage tank. These vents provide a path to vent steam released by the spargers or generated by the passive residual heat removal heat exchanger, into the containment atmosphere. Other vent covers also open on small pressure differentials to allow air/steam to enter the in-containment

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refueling water storage tank from containment, such as during a loss of coolant accident, to prevent damage to the tank. Overflow weirs with covers are provided from the in-containment refueling water storage tank to the refueling cavity to accommodate volume and mass increases during passive residual heat removal heat exchanger or automatic depressurization system operation, while minimizing the floodup of the containment.

The IRWST is stainless steel lined and does not contain material either in the tank or the recirculation path that could plug the outlet screens.

The in-containment refueling water storage tank contains one passive residual heat removal heat exchanger and two depressurization spargers. The top of the passive residual heat removal heat exchanger tubes are located underwater and extend down into the in-containment refueling water storage tank. The spargers are also submerged in the in-containment refueling water storage tank, with the spargers midarms located below the normal water level.

The in-containment refueling water storage tank is sized to provide the flooding of the refueling cavity for normal refueling, the post-loss of coolant accident flooding of the containment for reactor coolant system long-term cooling mode, and to support the passive residual heat removal heat exchanger operation. Flow out of the in-containment refueling water storage tank during the injection mode includes conservative allowances for spill flow during a direct vessel injection line break.

The in-containment refueling water storage tank can provide sufficient injection until the containment sump floods up high enough to initiate recirculation flow. The injection duration varies greatly, depending upon the specific event. A direct vessel injection line break more rapidly drains the in-containment refueling water storage tank and speeds containment floodup.

The containment floodup volume for a LOCA in PXS room B is less than 71,960 ft^3 (excluding the in-containment refueling water storage tank) below a containment elevation of 107.68 feet.

Connections to the in-containment refueling water storage tank provide for transfer to and from the reactor coolant system/refueling cavity via the normal residual heat removal system, purification and sampling via the spent fuel pool cooling system, and remotely adjusting boron concentration to the chemical and volume control system. Also, the normal residual heat removal system can provide cooling of the in-containment refueling water storage tank.

In-containment refueling water storage tank level and temperature are monitored by indicators and alarms. The operator can take action, as required, to meet the technical specification requirements for in-containment refueling water storage tank operability.

The protection and safety monitoring system isolates the spent fuel pool cooling system by closing the spent fuel pool cooling system containment isolation valves on low IRWST wide range level to prevent loss of inventory in the event of a leak in the nonsafety spent fuel pool cooling system.

The actuation can be manually blocked while the plant is in Mode 6, below P-9, to allow the spent fuel pool cooling system to transfer the IRWST to the refueling cavity and perform cooling and purification of the refueling cavity. A manual block can also be used during other modes when the plant is below P-9 so that if required, operators can continue to use the spent fuel pool cooling system to cool, purify, sample, or transfer water to the IRWST when the level is below the low setpoint.

6.3.2.2.4 pH Adjustment Baskets

The passive core cooling system utilizes pH adjustment baskets for control of the pH level in the containment sump. The baskets are made of stainless steel with a mesh front that readily permits

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contact with water. The baskets are designated AP1000 Equipment Class C, and are designed to meet seismic Category I requirements.

The total weight of TSP contained in the baskets is at least 26,460 pounds. The TSP, in granular form, is provided to raise the pH of the borated water in the containment following an accident to at least 7.0. After extended plant operation, the granular TSP may cake into a solid form as it absorbs moisture. Assuming that the TSP has caked, the dissolution time of the TSP is approximately 3 hours. Good mixing with the sump water is expected due to both basket construction and because the baskets are placed in locations conducive to recirculation flows post-accident. The baskets are designed for ease of replacement of the TSP.

6.3.2.2.5 Passive Residual Heat Removal Heat Exchanger

The passive residual heat removal exchanger consists of inlet and outlet channel heads connected together by vertical C-shaped tubes. The tubes are supported inside the in-containment refueling water storage tank. The top of the tubes is several feet below the in-containment refueling water storage tank water surface. The component data for the passive residual heat removal heat exchanger is shown in Table 6.3-2. The passive residual heat removal heat exchanger is AP1000 Equipment Class A and is designed to meet seismic Category I requirements.

The heat exchanger inlet piping connects to an inlet channel head located near the outside top of the tank. The inlet channel head and tubesheet are attached to the tank wall via an extension flange. The heat exchanger is supported by a frame which is attached to the IRWST floor and ceiling. The heat exchanger supports are designed to ASME Code, Section III, subsection NF. The extended flange is designed to accommodate thermal expansion. Figure 6.3-5 illustrates the relationship between these parts and the boundaries of design code jurisdiction. The heat exchanger outlet piping is connected to the outlet channel head, which is vertically below the inlet channel head, near the tank bottom. The outlet channel head has an identical structural configuration to the inlet channel head. Both channel head tubesheets are similar to the steam generator tubesheets and they have manways for inspection and maintenance access.

The passive residual heat removal heat exchanger is designed to remove sufficient heat so that its operation, in conjunction with available inventory in the steam generators, provide reactor coolant system cooling and prevents water relief through the pressurizer safety valves during loss of main feedwater or main feedline break events.

Passive residual heat removal heat exchanger flow and inlet and outlet line temperatures are monitored by indicators and alarms. The operator can take action, as required, to meet the technical specification requirements or follow emergency operating procedures for control of the passive residual heat removal heat exchanger operation.

6.3.2.2.6 Depressurization Spargers

Two reactor coolant depressurization spargers are provided. Each one is connected to an automatic depressurization system discharge header (shared by three automatic depressurization system stages) and submerged in the in-containment refueling water storage tank. Each sparger has four branch arms inclined downward. The connection of the sparger branch arms to the sparger hub are submerged below the in-containment refueling water storage tank overflow level by $\leq$11.5 feet. The component data for the spargers is shown in Table 6.3-2. The spargers are AP1000 Equipment Class C and are designed to meet seismic Category I requirements.

The spargers perform a nonsafety-related function -- minimizing plant cleanup and recovery actions following automatic depressurization. They are designed to distribute steam into the in-containment refueling water storage tank, thereby promoting more effective steam condensation.

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The first three stages of automatic depressurization system valves discharge through the spargers and are designed to pass sufficient depressurization venting flow, with an acceptable pressure drop, to support the depressurization system performance requirements. The installation of the spargers prevents undesirable and/or excessive dynamic loads on the in-containment refueling water storage tank and other structures.

Each sparger is sized to discharge at a flow rate that supports automatic depressurization system performance, which in turn, allows adequate passive core cooling system injection.

6.3.2.2.7 IRWST and Containment Recirculation Screens

The passive core cooling system has two different sets of screens that are used to prevent debris from entering the reactor and blocking core cooling passages during a LOCA: IRWST screens and containment recirculation screens. The screens are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. The structural frames, attachment to the building structure, and attachment of the screen modules use the criteria of ASME Code, Section III Subsection NF. The screen modules are fabricated of sheet metal and are designed and fabricated to a manufacturer's standard. The IRWST screens and containment recirculation screens are designed to comply with applicable licensing regulations including:

- GDC 35 of 10 CFR 50 Appendix A
- Regulatory Guide 1.82
- NUREG-0897

The operation of the passive core cooling system following a LOCA is described in Subsection 6.3.2.1.3. Proper screen design, plant layout, and other factors prevent clogging of these screens by debris during accident operations.

6.3.2.2.7.1 General Screen Design Criteria

The IRWST screens and containment recirculation screens are designed with the following criteria.

1. Screens are designed to Regulatory Guide 1.82, including:
    - Separate, large screens are provided for each function.
    - Containment recirculation screens are located well below containment floodup level. Each screen provides the function of a trash rack and a fine screen. A debris curb is provided for the containment recirculation screens to prevent high density debris from being swept along the floor to the screen face. The IRWST screens are arranged such that the lowest screening surface is 6 inches above the floor of the IRWST to prevent debris on the floor from entering the lowest levels of the screen face.
    - Floors slope away from screens (not required for AP1000).
    - Drains do not impinge on screens.
    - Screens can withstand accident loads and credible missiles.
    - Screens have conservative flow areas to account for plugging. Operation of the non-safety-related normal residual heat removal pumps with suction from the IRWST and the containment recirculation lines is considered in sizing screens.

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- System and screen performance are evaluated.
- Screens have solid top cover. Containment recirculation screens have a protective plate that is located no more than 1 foot, 3 inches above the top of the face of the screens, which extends at least 8 feet, 3 inches perpendicular to the front and at least 7 feet to the side of the face of the screens. The plate dimensions are relative to the portion of the screens where water flow enters the screen openings. The protective plate maximum height dimension is the distance between the top of the screens and the underside of the protective plate module top plate at the exposed edges of the protective plate, which extend into the containment recirculation water flow (east toward steam generator 2, and north toward the corridor). Coating debris, from coatings located outside of the ZOI, is not transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break that becomes submerged during recirculation considering the use of high density coatings discussed in Subsection 6.1.2.1.5.
- Screens are seismically qualified.
- Screen openings are sized to prevent blockage of core cooling.
- Screens are designed for adequate pump performance. AP1000 has no active ASME Section III safety-related pumps.
- Corrosion resistant materials are used for screens.
- Access openings in screens are provided for screen inspection.
- Screens are inspected each refueling.

2. Low screen approach velocities limit the transport of heavy debris even with operation of normal residual heat removal pumps.

3. [*Metal reflective insulation is used on ASME class 1 lines because they are subject to loss-of-coolant accidents. Metal reflective insulation is also used on the reactor vessel, the reactor coolant pumps, the steam generators, and on the pressurizer because they have relatively large insulation surface areas and they are located close to large ASME class 1 lines. As a result, they are subject to jet impingement during loss-of-coolant accidents.*]* A suitable equivalent insulation to metal reflective may be used. A suitable equivalent insulation is one that is encapsulated in stainless steel that is seam welded so that LOCA jet impingement does not damage the insulation and generate debris. Another suitable insulation is one that may be damaged by LOCA jet impingement as long as the resulting insulation debris is not transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break that becomes submerged during recirculation. In order to qualify as a suitable equivalent insulation, testing must be performed that subjects the insulation to conditions that bound the AP1000 conditions and demonstrates that debris would not be generated. If debris is generated, testing and/or analysis must be performed to demonstrate that the debris is not transported to an AP1000 screen or into the core through a flooded break. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing and/or analysis must be approved by the NRC.

   [*In order to provide additional margin, metal reflective insulation is used inside containment where it would be subject to jet impingement during loss-of-coolant accidents that are not otherwise shielded from the blowdown jet.*]* As a result, fibrous debris is not generated by loss-of-coolant accidents. Insulation located within the zone of influence (ZOI), which is a spherical

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region within a distance equal to 29 inside diameters (for Min-K, Koolphen-K, or rigid cellular glass insulation) or 20 inside diameters (for other types of insulation) of the LOCA pipe break is assumed to be affected by the LOCA when there are intervening components, supports, structures, or other objects.

[*The ZOI in the absence of intervening components, supports, structures, or other objects includes insulation in a cylindrical area extending out a distance equal to 45 inside diameters from the break along an axis that is a continuation of the pipe axis and up to 5 inside diameters in the radial direction from the axis.*]* A suitable equivalent insulation to metal reflective may be used as discussed in the previous paragraph.

[*Insulation used inside the containment, outside the ZOI, but below the maximum post-DBA LOCA floodup water level (plant elevation 110.2 feet), is metal reflective insulation, jacketed fiberglass, or a suitable equivalent.*]* A suitable equivalent insulation is one that would be restrained so that it would not be transported by the flow velocities present during recirculation and would not add to the chemical precipitates. In order to qualify as a suitable equivalent insulation, testing must be performed that subjects the insulation to conditions that bound the AP1000 conditions and demonstrates that debris would not be generated. If debris is generated, testing and/or analysis must be performed to demonstrate that the debris is not transported to an AP1000 screen or into the core through a flooded break. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing and/or analysis must be approved by the NRC.

[*Insulation used inside the containment, outside the ZOI, but above the maximum post-design basis accident (DBA) LOCA floodup water level, is jacketed fiberglass, rigid cellular glass, or a suitable equivalent.*]* A suitable equivalent insulation is one that when subjected to dripping of water from the containment dome would not add to the chemical precipitates; suitable equivalents include metal reflective insulation.

The non-metallic insulation used in the AP1000 reactor vessel neutron shield blocks and the water inlet doors have been determined to be a suitable equivalent for the locations bounded by testing and analysis (Reference 5).

4. Coatings are not used on surfaces located close to the containment recirculation screens. The surfaces considered close to the screens are defined in Subsection 6.3.2.2.7.3. Refer to Subsection 6.1.2.1.6. These surfaces are constructed of materials that do not require coatings.

5. The IRWST is enclosed which limits debris egress to the IRWST screens.

6. Containment recirculation screens are located above lowest levels of containment.

7. Long settling times are provided before initiation of containment recirculation.

8. Air ingestion by safety-related pumps is not an issue in the AP1000 because there are no active ASME Section III safety-related pumps. The normal residual heat removal system pumps are evaluated to show that they can operate with minimum water levels in the IRWST and in the containment.

9. A commitment for cleanliness program to limit debris in containment is provided in Subsection 6.3.8.1.

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10. [*Other potential sources of fibrous material, such as ventilation filters or fiber-producing fire barriers, are not located in jet impingement damage zones or below the maximum post-DBA LOCA floodup water level.*]*

11. Other potential sources of transportable material, such as caulking, signs, and equipment tags installed inside the containment are located:

    - Below the maximum flood level, or
    - Above the maximum flood level and not inside a cabinet or enclosure.

    Tags and signs in these locations are made of stainless steel or another metal that has a density ≥ 100 lbm/ft^3. Caulking in these locations is a high density (≥ 100 lbm/ft^3).

    The use of high-density metal prevents the production of debris that could be transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break location that is submerged during recirculation. If a high-density material is not used for these components, then the components must be located inside a cabinet or other enclosure, or otherwise shown not to transport; the enclosures do not have to be watertight, but need to prevent water dripping on them from creating a flow path that would transport the debris outside the enclosure. For light-weight (< 100 lb$_m$/ft^3) caulking, signs or tags that are located outside enclosures, testing must be performed that subjects the caulking, signs, or tags to conditions that bound the AP1000 conditions and demonstrates that debris would not be transported to an AP1000 screen or into the core through a flooded break. Note that in determining if there is sufficient water flow to transport these materials, consideration needs to be given as to whether they are within the ZOI (for the material used) because that determines whether they are in their original geometry or have been reduced to smaller pieces. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing must be approved by the NRC.

12. An evaluation consistent with Regulatory Guide 1.82, Revision 3, and subsequently approved NRC guidance, has been performed (Reference 3) to demonstrate that adequate long-term core cooling is available considering debris resulting from a LOCA together with debris that exists before a LOCA. As discussed in Subsection 6.3.2.2.7.1, a LOCA in the AP1000 does not generate fibrous debris due to damage to insulation or other materials included in the AP1000 design. The evaluation considered resident fibers and particles that could be present considering the plant design, location, and containment cleanliness program. The determination of the characteristics of such resident debris was based on sample measurements from operating plants. The evaluation also considered the potential for the generation of chemical debris (precipitants). The potential to generate such debris was determined considering the materials used inside the AP1000 containment, the post-accident water chemistry of the AP1000, and the applicable research/testing.

    The evaluation considered the following conservative considerations:

    - [*The COL cleanliness program will limit the total amount of resident debris inside the containment to ≤ 130 pounds and the amount of the total that might be fiber to ≤ 6.6 pounds.*]*
    - In addition to the resident debris, the LOCA blowdown jet may impinge on coatings and generate coating debris fines, which because of their small size, might not settle. The amount of coating debris fines that can be generated in the AP1000 by a LOCA jet will be limited to less than 70 pounds for double-ended cold leg and double-ended direct vessel injection LOCAs. In evaluating this limit, a ZOI of 4 IDs for epoxy and 10 IDs for inorganic

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zinc will be used. A DEHL LOCA could generate more coating debris; however, with the small amount of fiber available in the AP1000 following a LOCA, the additional coating debris fines that may be generated in a DEHL LOCA are not limiting.

- For ZOIs used in the debris evaluation, the diameter can be determined from the double ended guillotine break of primary system piping or the alternate debris generation size (Reference 5) developed in accordance with the alternate evaluation approach from NEI 04-07 (Reference 6).

- The total resident and ZOI coating debris available for transport following a LOCA is ≤ 193.4 pounds of particulate and ≤ 6.6 pounds of fiber. The percentage of this debris that could be transported to the screens or to the core is as follows:

  – Containment recirculation screens is ≤ 100 percent fiber and particles

  – IRWST screens is ≤ 50 percent fiber and 100 percent particles

  – Core (via a direct vessel injection or a cold leg LOCA break that becomes submerged) is ≤ 90 percent fiber and 100 percent particles

- Fibrous insulation debris is not generated and transported to the screens or into the core as discussed in item 3.

- Metal reflective insulation, including accident generated debris, is not transported to the screens or into the core.

- Coating debris is not transported to the screens or into the core as discussed in item 1.

- Debris from other sources, including caulking, signs, and tags, is not generated and transported to the screens or into the core as discussed in item 11.

- A ZOI radius of 4 IDs is applicable to the AP1000 in-containment cables bounded by testing and analysis, consistent with Reference 5.

- The total amount of chemical precipitates that could form in 30 days is ≤ 57 pounds.

- The percentage of the chemical precipitates that could be transported to the:

  – Containment recirculation screens is ≤ 100 percent.

  – IRWST screens is ≤ 100 percent.

  – Core is ≤ 100 percent.

- The range of flow rates during post-LOCA injection and recirculation is as follows:

  – CR screens: 2600 to 539 gpm

  – IRWST screens: 2600 to 464 gpm

  – Core: 2012 to 484 gpm

These flows bound operation of the PXS and the RNS. Note that if the RNS operates during post-LOCA injection or recirculation, the RNS flow is limited to 2600 gpm. This limit ensures that the

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operation of the plant is consistent with screen head loss testing. In addition, the screens will be designed structurally to withstand much higher flow rates and pressure losses to provide appropriate margin during PXS and RNS operation.

No chemical precipitates are expected to enter the IRWST because the primary water input to the IRWST is steam condensed on the containment vessel. However, during a direct vessel injection LOCA, recirculation can transport chemical debris through the containment recirculation screens and to the IRWST screens. As a result, 100 percent of the chemical debris is conservatively assumed to be transported to the IRWST screens.

The AP1000 containment recirculation screens and IRWST screens have been shown to have acceptable head losses. The head losses for these screens were determined in testing performed using the above conservative considerations. It has been shown that a head loss of 0.25 psi at the maximum screen flows is acceptable based on long-term core cooling sensitivity analysis.

Considering downstream effects as well as potential bypass through a cold leg LOCA, the core was shown to have acceptable head losses. The head losses for the core were determined in testing performed using the above conservative considerations. It has been shown that a head loss of 4.1 psi at these flows is acceptable based on long-term core cooling sensitivity analysis.

6.3.2.2.7.2 IRWST Screens

The IRWST screens are located inside the IRWST at the bottom of the tank. Figure 6.3-6 shows a plan view and Figure 6.3-7 shows a section view of these screens. Three separate screens are provided in the IRWST, one at either end of the tank and one in the center. A cross-connect pipe connects all three IRWST screens to distribute flow. The IRWST is closed off from the containment; its vents and overflow weirs are normally closed by louvers. The potential for introducing debris inadvertently during plant operations is limited. A cleanliness program (refer to Subsection 6.3.8.1) controls foreign debris from being introduced into the tank during maintenance and inspection operations. The Technical Specifications require visual inspections of the screens during every refueling outage.

The IRWST design eliminates sources of debris from inside the tank. Insulation is not used in the tank. Air filters are not used in the IRWST vents or overflow weirs. Wetted surfaces in the IRWST are corrosion resistant such as stainless steel or nickel alloys; the use of these materials prevents the formation of significant amounts of corrosion products. In addition, the water is required to be clean because it is used to fill the refueling cavity for refueling; filtering and demineralizing by the spent fuel pool cooling system is provided during and after refueling.

During a LOCA, steam vented from the reactor coolant system condenses on the containment shell and drains down the shell to the polar crane girder or internal stiffener where it is drained via downspouts to the IRWST. Steam that condenses below the internal stiffener drains down the shell and is collected in a gutter near the operating deck elevation. It is very unlikely that debris generated by a LOCA can reach the downspouts or the gutter because of their locations. Each downspout inlet is covered with a coarse screen that prevents larger debris from entering the downspout. The gutter is covered with a trash rack which prevents larger debris from clogging the gutter or entering the IRWST through the two 4-inch drain pipes. The inorganic zinc coating applied to the inside surface of the containment shell is safety – Service Level I, and will stay in place and will not detach.

The design of the IRWST screens reduces the chance of debris reaching the screens. The screens are oriented vertically such that debris that settles out of the water does not fall on the screens. The lowest screening surface of the IRWST screens is located 6 inches above the IRWST floor to prevent debris on the floor from entering the lowest levels of the IRWST screens. The screen design provides

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the trash rack function. This is accomplished by the screens having a large surface area to prevent a single object from blocking a large portion of the screen and by the screens having a robust design to preclude an object from damaging the screen and causing by-pass. The screen prevents debris larger than 0.0625 inch from being injected into the reactor coolant system and blocking fuel cooling passages. The screen is a type that has sufficient surface area to accommodate debris that could be trapped on the screen. The design of the IRWST screens is described further in APP-GW-GLN-147 (Reference 4).

The screen flow area is conservatively designed considering the operation of the nonsafety-related normal residual heat removal system pumps which produce a higher flow than the safety-related gravity driven IRWST injection/recirculation flows. As a result, when the normal residual heat removal system pumps are not operating, there is a large margin to screen clogging.

6.3.2.2.7.3 Containment Recirculation Screens

The containment recirculation screens are oriented vertically along walls above the loop compartment floor (elevation 83 feet). Figure 6.3-8 shows a plan view and Figure 6.3-9 shows a section view of these screens. Two separate screens are provided as shown in Figure 6.3-3. The loop compartment floor elevation is significantly above (11.5 feet) the lowest level in the containment, the reactor vessel cavity. A two-foot-high debris curb is provided in front of the screens.

During a LOCA, the reactor coolant system blowdown will tend to carry debris created by the accident (pipe whip/jets) into the cavity under the reactor vessel which is located away from and below the containment recirculation screens. As the accumulators, core makeup tanks and IRWST inject, the containment water level will slowly rise above the 108 foot elevation. The containment recirculation line opens when the water level in the IRWST drops to the Low-3 level setpoint a few feet above the final containment floodup level. When the recirculation lines initially open, the water level in the IRWST is higher than the containment water level and water flows from the IRWST backwards through the containment recirculation screen. This back flow tends to flush debris located close to the recirculation screens away from the screens. A flow connection between Screen A and Screen B is provided so that both recirculation screens will operate. This connection increases the reliability of the PXS in a PRA sequence where there are multiple failures of valves in one of the PXS subsystems.

The water level in the containment when recirculation begins is well above (~ 10 feet) the top of the recirculation screens. During the long containment floodup time, floating debris does not move toward the screens and heavy materials settle to the floors of the loop compartments or the reactor vessel cavity. During recirculation operation, the containment water level will not change significantly nor will it drop below the top of the screens.

The amount of debris that may exist following an accident is limited. Reflective insulation (or a suitable equivalent, as demonstrated per the criteria in Subsection 6.3.2.2.7.1) is used to preclude fibrous debris that can be generated by a loss of coolant accident and be postulated to reach the screens during recirculation. The nonsafety-related coatings used in the containment are designed to withstand the post accident environment. The containment recirculation screens are protected by a plate located above them. The protective plate prevents debris from the failure of nonsafety-related coatings from getting into the water close to the screens (closer than 8 feet, 3 inches from the front of the face of the screens and 7 feet from the side of the face of the screens) where the recirculation flow could cause the debris to be swept to the screens before it settles to the floor. The north edge of the north containment recirculation screen face is located at least 3 feet, 6 inches from the north corner of the west wall of steam generator compartment 2 (Room 11202), to which the containment recirculation screens are attached. Placement of the north containment recirculation screen prevents debris falling into the vertical access corridor (Room 11204) from entering the water closer than 3 feet, 6 inches from the face of the north containment recirculation screen. Stainless steel is used on

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the underside of the plate and on surfaces located below the plate, above the bottom of the screen face, extending at least 8 feet, 3 inches perpendicular to the front and at least 7 feet to the side of the face of the screens to prevent coating debris from reaching the screens.

A cleanliness program (refer to Subsection 6.3.8.1) controls foreign debris introduced into the containment during maintenance and inspection operations. The Technical Specifications require visual inspections of the screens during every refueling outage.

The design of the containment recirculation screens reduces the chance of debris reaching the screens. The screens are orientated vertically such that debris settling out of the water will not fall on the screens. The protective plate described above provides additional protection to the screens from debris. A 2-foot-high debris curb is provided to prevent high density debris from being swept along the floor by water flow to the containment recirculation screens. The screen design provides the trash rack function. This is accomplished by the screens having a large surface area to prevent a single object from blocking a large portion of the screen and by the screens having a robust design to preclude an object from damaging the screen and causing by-pass. The screen prevents debris larger than 0.0625 inch from being injected into the reactor coolant system and blocking fuel cooling passages. The screen is a type that has more surface area to accommodate debris that could be trapped on the screen. The design of the containment recirculation screens is further described in APP-GW-GLN-147 (Reference 4).

The screen flow area is conservatively designed, considering the operation of the normal residual heat removal system pumps, which produce a higher flow than the gravity driven IRWST injection/ recirculation flows. As a result, when the normal residual heat removal system pumps are not operating there is even more margin in screen clogging.

6.3.2.2.8 Valves

Design features used to minimize leakage for valves in the passive core cooling system include:

- Hermetically sealed valves are used for manual isolation valves that are 2 inches or smaller, except for some valves in the passive core cooling system test header, which have stems that are not normally pressurized.

- Valves which are normally open, except check valves and those which perform control function, are provided with back seats to limit stem leakage.

6.3.2.2.8.1 Manual Globe, Gate, and Check Valves

Gate valves have backseats and external screw and yoke assemblies.

Globe valves, both “T” and “Y” styles, are full-ported with external screw and yoke construction.

Stainless steel check valves have no penetration welds other than the inlet, outlet, and bonnet. The check valve hinge is serviced through the bonnet.

The gasket of the stainless steel manual globe and gate valves is similar to those described in Subsection 6.3.2.2.8.3 for motor-operated valves.

6.3.2.2.8.2 Manual Valves

Manual valves are generally used as maintenance isolation valves. When used for this function they are under administrative control. They are located so that no single valve can isolate redundant

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passive core cooling system equipment or they are provided with alarms in the main control room to indicate mispositioning.

To help preclude the possibility of passive core cooling system degradation due to valve mispositioning, line connections such as vent and drain lines, test connections, pressure points, flow element test points, flush connections, local sample points, and bypass lines are provided with double isolation or sealed barriers. The isolation is provided by one of the following methods:

- Two valves in series
- A single valve with a blind flange or pipe/tubing plug
- A single locked-closed valve
- A blind flange

6.3.2.2.8.3 Motor-Operated Valves

The motor operators for gate valves are conservatively sized, considering the frictional component of the hydraulic unbalance on the valve disc, the disc face friction, and the packing box friction. For motor-operated valves, the valve disc is guided throughout the full disc travel to prevent chattering and to provide ease of gate movement. The seating surfaces are hard-faced to prevent galling and to reduce wear.

Where a gasket is employed for the body to bonnet joint, it is either a fully trapped, controlled compression, spiral wound gasket with provisions for seal welding or it is of the pressure seal design with provisions for seal welding.

6.3.2.2.8.4 Motor-Operated Valve Controls

Remotely operated valves which do not receive a safeguards actuation signal, have their positions indicated on the main control board. When one of these valves is not in the ready position for injection during plant operation, this condition is indicated and alarmed in the main control room.

Spurious movement of a motor-operated valve due to an electrical fault in the motor actuation circuitry, coincident with loss of coolant accident, has been analyzed (Reference 1) and found to be an acceptably low probability event. In addition, power lockout in accordance with Branch Technical Position ICSB-18 is provided for those valves whose spurious movement could result in degraded passive core cooling system performance.

Table 6.3-1 provides a list of the remotely operated isolation valves in the passive core cooling system. These valves have various interlocks, automatic features, and position indication. Some valves have their control power locked out during normal plant operation. Periodic visual inspection and operability testing of the motor-operated valves in the passive core cooling system confirm valve operability. In addition, the location of the motor-operated valves within the containment, which are identified in Table 6.3-1, has been examined to identify remotely operated valves which may be submerged following a postulated loss of coolant accident.

See Section 3.4 for additional information on containment flooding effects.

6.3.2.2.8.5 Automatic Depressurization Valves

The automatic depressurization system consists of four different stages of valves. The first three stages each have two lines and each line has two valves in series; both normally closed. The fourth

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stage has four lines with each line having two valves in series; one normally open and one normally closed. The four stages, therefore, include a total of 20 valves. The four valve stages open sequentially.

The first stage, second-stage and third-stage valves have dc motor operators. The stage 1/2/3 control valves are normally closed globe valves; the isolation valves are normally closed gate valves. The fourth-stage valves are interlocked so that they can not open until reactor coolant system pressure has been substantially reduced. The fourth stage control valves are squib valves. There is a normally open motor-operated gate valve in series with each squib valve.

The first three stages have a common inlet header connected to the top of the pressurizer. The outlet of the first to third stages then combine to a common discharge line to one of the spargers in the in-containment refueling water storage tank. There is a second identical group of first- to third-stage valves with its own inlet and outlet line and sparger.

The fourth-stage valves connect directly to the top of the reactor coolant hot leg and vent directly to the steam generator compartment. There are also two groups of fourth stage valves, with one group in each steam generator compartment.

The automatic depressurization valves are designed to automatically open when actuated and to remain open for the duration of an automatic depressurization event. Valve stages 1 and 4 actuate at discrete core makeup tank levels, as either tank's level decreases during injection or from spilling out a broken injection line. Valve stages 2 and 3 actuate based upon a timed delay after actuation of the preceding stage. This opening sequence provides a controlled depressurization of the reactor coolant system. The valve opening sequence prevents simultaneous opening of more than one stage, to allow the valves to sequentially open. The valve actuation logic is based on two-of-four level detectors, in either core makeup tank for automatic depressurization system stages 1 and 4.

The stage 1/2/3 automatic depressurization control valves are designed to open relatively slowly. During the actuation of each stage, the isolation valve is sequenced open before the control valve. Therefore, there is some time delay between stage actuation and control valve actuation.

The operators can manually open the first-stage valves to a partially open position to perform a controlled depressurization of the reactor coolant system. Additional information on the automatic depressurization valves is provided in Subsection 5.4.6.

6.3.2.2.8.6 Low Differential Pressure Opening Check Valves

Several applications in the passive core cooling system gravity injection piping use check valves that open with low differential pressures. These check valves are installed in the following locations:

- The gravity injection line flow paths from the in-containment refueling water storage tank
- The containment recirculation lines that connect to the gravity injection lines

The check valves selected for these applications incorporate a simple swing-check design with a stainless steel body and hardened valve seats. The passive core cooling system check valves are safety-related, designed with their operating parts contained within the body, and with a low pressure drop across each valve. The valve internals are exposed to low temperature reactor coolant or borated refueling water.

During normal plant operation, these check valves are closed, with essentially no differential pressure across them. Confidence in the check valve operability is provided by operation at no

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differential pressure clean/cold fluid environment, the simple valve design, and the specified seat materials.

The check valves normally remain closed, except for testing or when called upon to open following an event to initiate passive core cooling system operation. The valves are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating, and they do not experience significant wear of the moving parts.

These check valves are periodically tested during shutdown conditions to demonstrate valve operation. These check valves are equipped with nonintrusive position sensors to indicate when the valves are open or closed.

6.3.2.2.8.7 Accumulator Check Valves

The accumulator check valve design is similar to the accumulator check valves in current pressurized water reactor applications. It is also similar to the low differential pressure opening check valve design described in Subsection 6.3.2.2.8.6. The accumulator check valves are diverse from the core makeup tank valves because they use different check valve types.

During normal operation, the check valves are in the closed position with a nominal differential pressure across the disc of about 1550 psid. The valves remain in this position, except for testing or when called upon to open following an event. They are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating. They do not experience significant wear of the moving parts and they are expected to function with minimal backleakage.

The accumulators can accept some inleakage from the reactor coolant system without affecting availability. Continuous inleakage requires that the accumulator water volume and boron concentration be adjusted periodically to meet technical specification requirements.

The AP1000 accumulator check valves are periodically tested during shutdown conditions to demonstrate their operation.

6.3.2.2.8.8 Relief Valves

Relief valves are installed for passive core cooling system accumulators to protect the tanks from overpressure.

The passive core cooling system piping is reviewed to identify those lengths of piping that are isolated by normally closed valves and that do not have pressure relief protection in the piping section between the valves.

These piping sections include:

- Portions of in-containment passive core cooling system test lines that are not passive core cooling system accident mitigation flow paths and are not needed to achieve safe shutdown
- Piping vents, drains, and test connections that typically have two closed valves or one closed valve and a blind flange, or pipe/tubing plug
- Check valve test lines with sections isolated by two normally closed valves.

The piping vents, drains, test connections, and check valve lines have design pressure/ temperature conditions compatible with the process piping to which they connect. Valve leakage does not overpressurize the isolated piping sections and pressure relief provisions are not required.

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6.3.2.2.8.9 Explosively Opening (Squib) Valves

Squib valves are used in several passive core cooling system lines in order to provide the following:

- Zero leakage during normal operation
- Reliable opening during an accident
- Reduced maintenance and associated personnel radiation exposure

Squib valves are used to isolate the incontainment refueling water storage tank injection lines and the containment recirculation lines. In these applications, the squib valves are not expected to be opened during normal operation and anticipated transients. In addition, after they are opened it is not necessary that they re-close.

In the incontainment refueling water storage tank injection lines, the squib valves are in series with normally closed check valves. Inadvertent opening of the injection squib valves is considered beyond design basis. However, the consequence of this inadvertent opening could lead to a loss of reactor coolant. Therefore, a blocking device, as described in Subsection 7.3.1.2.4.1, is utilized to preclude the inadvertent opening of the injection squib valves. In the containment recirculation lines, the squib valves are in series with normally closed check valves in two lines and with normally open motor operated valves in the other two lines. Inadvertent opening of these recirculation squib valves is a beyond design basis event that will not result in loss of reactor coolant but will drain the incontainment refueling water storage tank. In the event of inadvertent actuation, the operator would manage the plant in accordance with the appropriate operating procedures.

The type of squib valve used in these applications provides zero leakage in both directions. It also allows flow in both directions. A valve open position sensor is provided for these valves. The IRWST injection squib valves and the containment recirculation squib valves in series with check valves are diverse from the other containment recirculation squib valves. They are designed to different design pressures. The IRWST injection and the containment recirculation squib valves are qualified to operate after being submerged; this capability adds margin to the performance of the PXS in handling debris during long-term core cooling following a LOCA.

Squib valves are also used to isolate the fourth stage automatic depressurization system lines. These squib valves are in series with normally open motor operated gate valves. Actuation of these squib valves requires signals from two separate protection logic cabinets. This helps to prevent spurious opening of these squib valves. The type of squib valve used in this application provides zero leakage of reactor coolant out of the reactor coolant system. The reactor coolant pressure acts to open the valve. A valve open position sensor is provided for these valves.

See Section 13.6 with regard to the use of these special explosive devices on-site.

6.3.2.3 Applicable Codes and Classifications

Sections 5.2 and 3.2 list the equipment ASME Code and seismic classification for the passive core cooling system. Most of the piping and components of the passive core cooling system within containment are AP1000 Equipment Class A, B, or C and are designed to meet seismic Category I requirements. Equipment Class C components and piping, that provide an emergency core cooling function, have augmented weld inspection requirements (see Subsection 3.2.2.5). Some system piping and components that do not perform safety-related functions are nonsafety-related.

The requirements for the control, actuation, and Class 1E devices are presented in Chapters 7 and 8.

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6.3.2.4 Material Specifications and Compatibility

Materials used for engineered safety feature components are given in Section 6.1. Materials for passive core cooling system components are selected to the meet the applicable material requirements of the codes in Section 5.2, as well as the following additional requirements:

- Parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or an equivalent corrosion-resistant material.
- Internal parts of components in contact with containment emergency sump solution during recirculation are fabricated of austenitic stainless steel or an equivalent corrosion resistant material.
- Valve seating surfaces are hard-faced to prevent failure and to reduce wear.
- Valve stem materials are selected for their corrosion resistance, high-tensile properties, and their resistance to surface scoring by the packing.

Section 6.1 summarizes the materials used for passive core cooling system components.

6.3.2.5 System Reliability

The reliability of the passive core cooling system is considered including periodic testing of the components during plant operation. The passive core cooling system is a redundant, safety-related system. The system is designed to withstand credible single active or passive failures.

The initiating signals for the passive core cooling system are derived from independent sources as measured from process parameters (pressurizer Low-3 pressure) or environmental (containment High-2 pressure) variables. Redundant, as well as functionally independent variables, are measured to initiate passive core cooling system operation.

Redundant passive core cooling system components are physically separated and protected so that a single event cannot initiate a common failure.

Power sources for the passive core cooling system are divided into four independent divisions that are supplied from the Class 1E dc and UPS system. Sufficient battery capacity is maintained to provide required power to the emergency loads when onsite and offsite ac power sources are not available. Section 8.3 provides additional information.

The preoperational testing program confirms that the systems, as designed and constructed meet the functional design requirements. Section 14.2 provides additional information. The passive core cooling system is designed with the capability for on-line testing of its active components so the availability and operation status can be readily determined. Testing of passive components such as check valves, tanks, heat exchanger, and flow paths can be conducted during shutdown conditions. In addition, the integrity of the passive core cooling system is verified through examination of critical components during the routine in-service inspection. Subsection 3.9.6 provides additional information.

The reliability assurance program described in Section 16.2, extends to the procurement of passive core cooling system components. The procurement quality assurance program is described in Chapter 17.

The passive core cooling system is a redundant, safety-related system. During the long-term cooling period following a loss of coolant accident, once the passive core cooling system equipment has

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actuated, there is no long-term maintenance required. Components actuate to the safeguards actuation alignment and do not need subsequent position changes for long-term operation.

For long-term cooling, the reactor coolant system is depressurized to containment ambient pressure following a loss of coolant accident. During this period, the heat generated in the reactor core is the residual decay heat and the passive core cooling system provides the required decay heat removal.

Proper initial filling and venting of the passive core cooling system prevents water hammer from occurring in the passive core cooling system lines. In addition, the head of water provided by the various tanks keeps system lines full. The arrangement of the core makeup tank pressure equalization line design also reduces the potential for water hammer. High-point vents in the passive core cooling system lines are provided as a means for venting of lines. Fill and venting procedures for the passive core cooling system provide for the removal of air from the system.

The existence of high-point vents and the positive head of water provide means by which the operator can confirm water-solid passive core cooling system lines, where required.

6.3.2.5.1 Response to Active Failure

Treatment of active failures is described in Subsection 15.0.12.

An active failure is the failure of a powered component, a component of the electrical supply system, or instrumentation and control equipment to act on command to perform its function. One example is the failure of a motor-operated valve to move to its intended safeguards actuation position.

One change in the definition of active failures has been incorporated into the passive core cooling system design. The system has been specifically designed to treat check valve failures to reposition as active failures. More specifically, it is assumed that normally closed check valves may fail to open and normally open check valves may fail to close. Check valves that remain in the same position before and after an event are not considered active failures.

There are two exceptions to this treatment of check valve failures in the passive core cooling system. One exception is made for the accumulator check valves, which is consistent with the treatment of these specific check valves in currently licensed plant designs. The other exception is made for the core makeup tank check valves failure to re-open after they have closed during an accident. The valves are normally open, biased-opened check valves. This exception is based on the low probability of these check valves not re-opening within a few minutes after they have cycled closed during accumulator operation.

The failure mode and effects analysis provided in Table 6.3-3 provides a summary of the passive core cooling system response to single failure of the various components.

The following passive core cooling system motor-operated valves are not included in this analysis:

- Both accumulator discharge line motor-operated valves
- Both in-containment refueling water storage tank gravity injection line motor-operated valves
- Both containment recirculation line motor-operated valves
- Both core makeup tank inlet line motor-operated valves
- The passive residual heat removal heat exchanger inlet line motor-operated valve

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These valves are normally in the required position for actuation of the associated component, they have redundant position indications and alarms, and they also receive confirmatory open actuation signals. The accumulator, incontainment refueling water storage tanks and passive residual heat removal heat exchanger valves have their power removed and locked out. The core makeup tank line has redundant series controllers. Therefore, these valves are not considered in the failure modes and effects analysis.

The analysis illustrates that the passive core cooling system can sustain an active failure in either the short-term or long-term and meet the required level of performance for core cooling. The short-term operation of the active components of the passive core cooling system following a steam line rupture or a steam generator tube rupture is similar to that following a loss of coolant accident. The same analysis is applicable and the passive core cooling system can sustain the failure of a single active component and meet the level of performance for the addition of shutdown reactivity.

Portions of the passive core cooling system are also relied upon to provide boration and makeup during a safety-related shutdown. The passive core cooling system can sustain an active failure and perform the required functions necessary to establish safe shutdown conditions. Safe shutdown operation of the passive core cooling system is described in Section 7.4.

6.3.2.5.2 Response to Passive Failure

Treatment of passive failures is described in Subsection 15.0.12.

A passive failure is the structural failure of a static component which limits the component's effectiveness in carrying out its design function. Examples include cracking of pipes, sprung flanges, or valve packing leaks. The passive core cooling system can sustain a single passive failure during the long-term phase and still retain an intact flow path to the core to supply sufficient flow to keep the core covered and to remove decay heat.

Since the passive core cooling system equipment is inside the containment, offsite dose caused by passive failures is not a concern. Also, with actuation of the automatic depressurization system, the reactor coolant system pressure is very close to containment pressure. Therefore, it is not necessary to isolate or realign the passive core cooling system following a passive failure.

The passive core cooling system flow paths are separated into redundant lines, either of which can provide minimum core cooling functions and return spilled water from the floor of the containment back to the reactor coolant system. For the long-term passive core cooling system function, adequate core cooling capacity exists with one of the two redundant flow paths.

6.3.2.5.3 Lag Times

Lag times for initiation and operation of the passive core cooling system are controlled by repositioning of valves. Some valves are normally in the position required for safety-related system function and therefore, their valve operation times are not considered. For those valves that reposition to initiate safety-related system functions, the valve repositioning times are less than the times assumed in the accident analyses. These lag times refer to the time after initiation of the safeguards actuation signal.

6.3.2.5.4 Potential Boron Precipitation

Boron precipitation in the reactor vessel is prevented by sufficient flow of passive core cooling system water through the core to limit the increase in boron concentration of the water remaining in the reactor vessel. Water along with steam leaves the core and exits the RCS through the fourth stage

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ADS lines. These valves connect to the hot leg and open in about 20 minutes after a loss of coolant accident or an automatic depressurization system actuation.

6.3.2.5.5 Safe Shutdown

During a safe shutdown, the passive core cooling system provides redundancy for boration, makeup, and heat removal functions. Section 7.4 provides additional information about safe shutdown.

6.3.2.6 Protection Provisions

The measures taken to protect the system from damage that might result from various events are described in other sections, as listed below.

- Protection from dynamic effects is presented in Section 3.6.
- Protection from missiles is presented in Section 3.5.
- Protection from seismic damage is presented in Sections 3.7, 3.8, 3.9, and 3.10.
- Protection from fire is presented Subsection 9.5.1.
- Environmental qualification of equipment is presented in Section 3.11.
- Thermal stresses on the reactor coolant system are presented in Section 5.2.

6.3.2.7 Provisions for Performance Testing

The passive core cooling system includes the capability for determination of the integrity of the pressure boundary formed by series passive core cooling system check valves. Additional information on testing can be found in Subsection 6.3.6.

6.3.2.8 Manual Actions

The passive core cooling system is automatically actuated for those events as presented in Subsection 6.3.3. Following actuation, the passive core cooling system continues to operate in the injection mode until the transition to recirculation initiates automatically following containment floodup.

Although the passive core cooling system operates automatically, operator actions would be beneficial, in some cases, in reducing the consequences of an event. For example, in a steam generator tube rupture with no operator action, the protection and safety monitoring system automatically terminates the leak, prevents steam generator overfill, and limits the offsite doses. However, the operator can initiate actions, similar to those taken in current plants, to identify and isolate the faulted steam generator, cool down and depressurize the reactor coolant system to terminate the break flow to the steam generator, and stabilize plant conditions.

The operator can take action to avoid actuation of the automatic depressurization system when it is not needed. For non-LOCA events during which ac power has been lost for more than 22 hours, the protection and safety monitoring system will automatically open the automatic depressurization system valves to begin a controlled depressurization of the reactor coolant system and, eventually, containment floodup and recirculation prior to depletion of the 24-hour Class 1E actuation batteries. However, the operators can take action to block actuation of the automatic depressurization system should actuation be deemed unnecessary based on reactor coolant system conditions. This action

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allows closed loop passive residual heat removal heat exchanger operation to continue as long as acceptable reactor coolant system conditions are maintained.

Section 7.4 describes the anticipated operator actions to block the unnecessary automatic depressurization system actuation and to achieve recovery using available systems to remove decay heat. Section 7.5 describes the post-accident monitoring instrumentation available to the operator in the main control room following an event.

6.3.3 Performance Evaluation

The events described in Subsection 6.3.1 result in passive core cooling system actuation and are mitigated within the performance criteria. For the purpose of evaluation in Chapters 15 and 19, the events that result in passive core cooling system actuation are categorized as follows:

A. Increase in heat removal by the secondary system

1. Inadvertent opening of a steam generator power-operated atmospheric steam relief or safety valve

2. Steam system piping failure

B. Decrease in heat removal by the secondary system

1. Loss of Main Feedwater Flow

2. Feedwater system piping failure

C. Decrease in reactor coolant system inventory

1. Steam generator tube rupture

2. Loss of coolant accident from a spectrum of postulated reactor coolant system piping failures

3. Loss of coolant due to a rod cluster control assembly ejection accident

   (This event is enveloped by the reactor coolant system piping failures.)

D. Shutdown Events (Chapter 19)

1. Loss of Startup Feedwater

2. Loss of normal residual heat removal system with reactor coolant system pressure boundary intact

3. Loss of normal residual heat removal system during mid-loop operation

4. Loss of normal residual heat removal system with refueling cavity flooded

The events listed in groups A and B are non-LOCA events where the primary protection is provided by the passive core cooling system passive residual heat removal heat exchanger. For these events, the passive residual heat removal heat exchanger is actuated by the protection and monitoring system for the following conditions:

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- Low-2 steam generator narrow range water level, coincident with startup feedwater Low-2 flow coincident with T_{cold} - High
- Low-2 steam generator wide range water level
- Core makeup tank actuation
- Automatic depressurization actuation
- Pressurizer water level - High 3
- Manual actuation

The events listed in group C above are events involving the loss of reactor coolant where the primary protection is by the core makeup tanks and accumulators. For these events the core makeup tanks are actuated by the protection and monitoring system for the following conditions:

- Pressurizer Low-3 pressure
- Pressurizer Low-2 level
- Steam line Low-2 pressure
- Containment High-2 pressure
- Cold leg Low-2 temperature
- Low-2 steam generator wide range water level, coincident with reactor coolant system high hot leg temperature
- Manual actuation

In addition to initiating passive core cooling system operation, these signals initiate other safeguards automatic actions including reactor trip, reactor coolant pump trip, feedwater isolation, and containment isolation. The passive core cooling system actuation signals are described in Section 7.3.

The core makeup tanks and passive residual heat removal heat exchangers are also actuated by the Diverse Actuation System as described in Subsection 7.7.1.11.

Upon receipt of an actuation signal, the actions described in Subsection 6.3.2.1 are automatically initiated to align the appropriate features of the passive core cooling system.

For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III, and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in Subsection 15.0.11.2. In this evaluation, it is assumed that the operators power down the protection and safety monitoring actuation cabinets in the 22-hour time frame prior to the automatic timer actuating the automatic depressurization system.

In addition to mitigating the initiating events, the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to the specified safe shutdown condition of 420°F

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within 36 hours as described in Subsection 19E.4.10.2. A non-bounding, conservative analysis of the plant response during operator-initiated, extended operation of the passive residual heat removal heat exchanger is demonstrated in the shutdown temperature evaluation of Subsection 19E.4.10.2. The closed-loop cooling mode allows the reactor coolant system pressure to decrease and reduces the stress in the reactor coolant system and connecting pipe.

For loss of coolant accidents, the core makeup tanks deliver borated water to the reactor coolant system via the direct vessel injection nozzles. The accumulators deliver flow to the direct vessel injection line whenever reactor coolant system pressure drops below the tank static pressure. The in-containment refueling water storage tank provides gravity injection once the reactor coolant system pressure is reduced to below the injection head from the in-containment refueling water storage tank. The passive core cooling system flow rates vary depending upon the type of event and its characteristic pressure transient.

As the core makeup tanks drain down, the automatic depressurization system valves are sequentially actuated. The depressurization sequence establishes reactor coolant pressure conditions that allow injection from the accumulators, and then from the in-containment refueling water storage tank and the containment recirculation path. Therefore, an injection source is continually available. If onsite or offsite ac power has not been restored after 72 hours, the post-72 hour support actions described in Subsection 1.9.5.4 maintain this mode of core cooling and provide adequate decay heat removal for an unlimited time.

The transient analyses summarized in Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in Subsection 6.3.1.1.4, the passive systems are capable of bringing the plant to a safe, stable condition for at least 72 hours in closed loop cooling mode and for longer in an open loop mode.

The events listed in group D occur during shutdown conditions that are characterized by slow plant responses and mild thermal-hydraulic transients. In addition, some of the passive core cooling system features need to be isolated to allow the plant to be in these conditions or to perform maintenance on the system. The protection and monitoring system automatically actuates gravity injection from the IRWST to provide core cooling during shutdown conditions prior to refueling cavity floodup. In addition, the operator can also manually actuate other passive core cooling system equipment, such as the passive residual heat removal heat exchanger, to provide core cooling during shutdown conditions when the equipment does not automatically actuate.

6.3.3.1 Increase in Heat Removal by the Secondary System

A number of events that could result in an increase in heat removal from the reactor coolant system by the secondary system have been postulated. For each event, consideration has been given to operation of nonsafety-related systems that could affect the event results. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.1. For those events resulting in passive core cooling system actuation, the following summarizes passive core cooling system performance.

6.3.3.1.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Subsection 15.1.4 provides a description of an inadvertent opening of a steam generator relief or safety valve, including criteria and analytical results.

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For this event, upon generation of a safeguards actuation signal the reactor is tripped, the core makeup tanks are actuated, and the reactor coolant pumps are tripped. Since the core makeup tanks are actuated, the passive residual heat removal heat exchanger is also actuated. The main steam lines are also isolated to prevent blowdown of more than one steam generator. The core makeup tanks operate with water recirculation injection to provide borated water to the reactor vessel downcomer plenum for reactor coolant system inventory and reactivity control. The trip of the reactor initially brings the reactor sub-critical. The rapid reactor coolant system cool down may result in the reactor returning to critical because the rate of positive reactivity addition (reactor coolant system temperature reduction) exceeds the rate of negative reactivity addition (boron from the core makeup tank). As the event continues, the reactor coolant system cooldown will slow down such that the continued core makeup tank boration will return the reactor sub-critical. The departure from nucleate boiling design basis is met, thereby preventing fuel damage.

During this event, the startup feedwater system is assumed to malfunction so that it injects water at the maximum flow rate. This injection continues until feedwater isolation occurs on low reactor coolant system temperature. The feedwater isolation signal terminates the feedwater addition from the startup feedwater system. The passive residual heat removal heat exchanger is also assumed to function in this event. This heat removal mechanism continues throughout the duration of the event.

For this event, the core makeup tanks operate in the water recirculation mode, providing boration and injection flow without draining. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level.

Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.1.2 Steam System Pipe Failure

The most severe core conditions resulting from a steam system piping failure are associated with a double-ended rupture of a main steam line, occurring at zero power. Effects of smaller piping failures at higher power levels are bounded by the double-ended rupture at zero power. Subsection 15.1.5 provides a description of this event, including criteria and analytical results.

For this event, the passive core cooling system functions as described in Subsection 6.3.3.1.1 for the inadvertent opening of a steam generator relief or safety valve. However, this piping failure constitutes a more severe cooldown transient. The malfunctioning of the startup feedwater system is considered as it was in the inadvertent steam generator depressurization. The trip of the reactor initially brings the reactor sub-critical. The rapid reactor coolant system cool down may result in the reactor returning to critical because the rate of positive reactivity addition (reactor coolant system temperature reduction) exceeds the rate of negative reactivity addition (boron from the core makeup tank). As the event continues, the reactor coolant system cooldown will slow down such that the continued core makeup tank boration will return the reactor sub-critical. The departure from nucleate boiling design basis is met.

For this event, the reactor coolant system may depressurize sufficiently to permit the accumulators to deliver makeup water to the reactor coolant system. The core makeup tanks inject via water recirculation without draining. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates a normal plant shutdown.

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6.3.3.2 Decrease in Heat Removal by the Secondary System

A number of events have been postulated that could result in a decrease in heat removal from the reactor coolant system by the secondary system. For each event, consideration has been given to operation of nonsafety-related systems that could affect the consequences of an event. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.2. For those events resulting in passive core cooling system actuation, the following summarizes passive core cooling system performance.

6.3.3.2.1 Loss of Main Feedwater

The most severe core conditions resulting from a loss of main feedwater system flow are associated with a loss of flow at full power. The heat-up transient effects of loss of flow at reduced power levels are bounded by the loss of flow at full power. Subsection 15.2.7 provides a description of this event, including criteria and analytical results.

For this event, the passive residual heat removal heat exchanger is actuated. If the core makeup tanks are not initially actuated, they actuate later when passive residual heat exchanger cooling sufficiently reduces pressurizer level. The passive residual heat removal heat exchanger serves to remove core decay heat and the core makeup tanks inject a borated water solution directly into the reactor vessel downcomer annulus. Since the reactor coolant pumps are tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation conditions. The core makeup tanks operate via water recirculation, without draining, to maintain reactor coolant system inventory. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level. Since the event is characterized by a heat-up transient, the injection of negative reactivity is not required and is not taken credit for in the analysis to control core reactivity.

The reactor coolant system does not depressurize to permit the accumulators to deliver makeup water to the reactor coolant system. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates a normal plant shutdown.

6.3.3.2.1.1 Loss of AC Power to Plant Auxiliaries

The most severe conditions resulting from a loss of ac power to the plant auxiliaries are associated with loss of offsite power with a loss of main feedwater system flow at full power. A loss of main feedwater with a loss of ac power lasting longer than a few hours presents the highest demand on passive residual heat removal heat exchanger operation. Subsection 15.2.6 provides a description of this short-term event, including criteria and analytical results.

During most events, the passive systems would be terminated in hours. When an ac power source is restored and passive core cooling system termination criteria are satisfied, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations (as discussed in Subsection 6.3.1.2.1).

However, if normal systems are not recovered as expected, the passive residual heat removal heat exchanger removes core decay heat and maintains acceptable reactor coolant system conditions for at least 72 hours. For a non-LOCA event where ac power is lost, the automatic depressurization system will actuate in approximately 22 hours if operators do not act to avoid actuation when it is not needed. For this long-term transient, it is assumed operators extend passive residual heat exchanger operation as described in the Subsection 7.4.1.1.

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The loss of main feedwater with loss of ac power event is analyzed for a 72-hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries as described in Subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of the passive residual heat removal heat exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met. If non-safety systems capable of removing decay heat are not recovered, operator action to actuate automatic depressurization system is eventually required. This condition would then be bounded by the Condition III event of inadvertent automatic depressurization system actuation.

Reactor coolant system leakage could limit closed-loop capacity. A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the 24-hour Class 1E batteries, or could require the operators to re-energize the buses powered by the Class 1E batteries before 72 hours so that the automatic depressurization system valves could be actuated.

6.3.3.2.2 Feedwater System Pipe Failure

The most severe core conditions resulting from a feedwater system piping failure are associated with a double-ended rupture of a feed line at full power. Depending on break size and power level, a feedwater system pipe failure could cause either a reactor coolant system cooldown transient or a reactor coolant system heat-up transient. Only the reactor coolant system heat-up transient is evaluated as a feedwater system pipe failure, since the spectrum of cooldown transients is bounded by the steam system pipe failure analyses. The heat-up transient effects of smaller piping failures at reduced power levels are bounded by the double-ended feed line rupture at full power. Subsection 15.2.8 provides a description of this event, including criteria and analytical results.

For this event, the passive residual heat removal heat exchanger and the core makeup tanks are actuated. The passive residual heat removal heat exchanger serves to remove core decay heat, and the core makeup tanks inject a borated water solution directly into the reactor vessel downcomer. Since the reactor coolant pumps are tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation conditions. The core makeup tanks operate via water recirculation to maintain reactor coolant system inventory. Since the event is characterized by a heat-up transient, the injection of negative reactivity is not required and is not taken credit for in the analysis to control core reactivity.

The reactor coolant system does not depressurize to permit the accumulators to deliver makeup water to the reactor coolant system. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.3 Decrease in Reactor Coolant System Inventory

A number of events have been postulated that could result in a decrease in reactor coolant system inventory. For each event, consideration has been given to operation of nonsafety-related systems that could affect the consequences of the event. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.6. For those events which result in passive core cooling system actuation, the following summarizes passive core cooling system performance.

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6.3.3.3.1 Steam Generator Tube Rupture

Although a steam generator tube rupture is an event that results in a decrease in reactor coolant system inventory, severe core conditions do not result from a steam generator tube rupture. The event analyzed is a complete severance of a single steam generator tube that occurs at power with the reactor coolant contaminated with fission products, corresponding to continuous operation with a limited amount of defective fuel rods. Effects of smaller breaks are bounded by the complete severance. Subsection 15.6.3 provides a description of this event, including criteria and analytical results.

For this event, the nonsafety-related makeup pumps are automatically actuated when reactor coolant system inventory decreases and a reactor trip occurs, followed by actuation of the startup feedwater pumps. The startup feedwater flow initiates on low steam generator level following the reactor trip and automatically throttles feedwater flow to maintain programmed steam generator level, limiting overfill of the faulted steam generator. The makeup pumps automatically function to maintain the programmed pressurizer level. The operators are expected to take actions similar to those in current plants to identify and isolate the faulted steam generator, cooldown and depressurize the reactor coolant system to terminate the break flow into the steam generator, and stabilize plant conditions.

If the operator fails to take timely or correct actions in response to the leak, or if the makeup pumps and/or the startup feedwater pumps malfunction with excessive flow, then the water level in the faulted steam generator continues to increase. This actuates safety-related overfill protection and automatically isolates the startup feedwater pumps and the chemical and volume control system makeup pumps. The core makeup tanks subsequently actuate on Low-2 pressurizer level, if they are not already actuated. Actuation of the core makeup tanks automatically actuates the passive residual heat removal system heat exchanger.

The core makeup tanks operate via water recirculation to provide borated water directly into the reactor vessel downcomer to maintain reactor coolant system inventory. The passive residual heat removal heat exchanger serves to remove core decay heat. Since the reactor coolant pumps are automatically tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation flow conditions. The passive residual heat removal heat exchanger, in conjunction with the core makeup tanks, remove core decay heat and reduce reactor coolant system temperature. As the reactor coolant system cools and the inventory contracts, pressurizer level and pressure decrease, equalizing with steam generator pressure and terminating break flow.

If the nonsafety-related systems fail to start, the core makeup tanks and the passive residual heat removal heat exchangers automatically actuate. Their response is similar to that previously described, except that the faulted steam generator level is lower.

In these events, the plant conditions are stabilized without actuating the automatic depressurization system. Once plant conditions are stable, the operator completes a normal plant shutdown.

6.3.3.3.2 Loss of Coolant Accident

A loss of coolant accident is a rupture of the reactor coolant system piping or branch piping that results in a decrease in reactor coolant system inventory that exceeds the flow capability of the normal makeup system. Ruptures resulting in break flow within the capability of the normal makeup system do not result in decreasing reactor coolant system pressure and actuation of the passive core cooling system. The maximum break size for which the normal makeup system can maintain reactor coolant system pressure is obtained by comparing the calculated flow from the reactor coolant system through the postulated break with the charging pump makeup flow at a reactor coolant system pressure that is above the low pressure safeguards actuation setpoint. The makeup flow rate

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from one makeup pump is adequate to maintain pressurizer pressure for a break through a 0.375-inch diameter hole. Therefore, the normal makeup system can maintain reactor coolant system pressure and permit the operator to execute an orderly shutdown.

For the purpose of evaluation, the spectrum of postulated piping breaks in the reactor coolant system is divided into major pipe breaks (large break) and minor pipe breaks (small breaks). The large break is a rupture with a total cross-sectional area equal to or greater than one square foot. The small break is defined as a rupture with a total cross-sectional area less than one square foot. Section 15.6 provides a description of this event, including criteria and analytical results.

For either event, the core makeup tanks are actuated upon receipt of a safeguards actuation signal. These tanks provide high-pressure injection. For large breaks, or after the automatic depressurization system is actuated, the accumulators also provide injection. After automatic depressurization system actuation, the in-containment refueling water storage tank, and the containment recirculation sump, provide low pressure injection.

The core makeup tanks can operate via water recirculation or steam-compensated injection during LOCAs. For smaller loss of coolant accidents, the reactor coolant system inventory is sufficient to establish water recirculation. For larger break sizes, when the pressurizer empties and voiding occurs in the cold legs steam-compensated injection initiates. When the cold legs void, the core makeup tank flow increases.

As the core makeup tanks drain, their level sequences the automatic depressurization system valve stages. As the level drops in the core makeup tank, the first-stage actuates. The first-stage valves are connected to the top of the pressurizer and discharge to the in-containment refueling water storage tank via the automatic depressurization system spargers. After a time delay, the second-stage is actuated. The second stage valves are connected with the same flow path as the first-stage valves. After an additional time delay, the third-stage is actuated. The third stage valves are identical to the second-stage valves. As the core makeup tank drops to a low level the fourth-stage is actuated. The fourth stage valves are connected to both hot legs and they discharge directly to the reactor coolant system loop compartments at an elevation just above the maximum containment floodup level.

The in-containment refueling water storage tank line squib valves are opened on the fourth stage actuation signal. Check valves arranged in series with the squib valves remain closed until the reactor depressurizes. After depressurization, the in-containment refueling water storage tank provides injection flow. The flow continues until containment floodup initiates containment recirculation.

For large breaks or following automatic depressurization system initiation, the accumulators provide rapid injection to the reactor vessel through the same connections used by the core makeup tanks and the in-containment refueling water storage tank injection. The accumulators begin to inject when the reactor coolant system depressurizes to about 700 psig. During the loss of coolant accident transient, flow to the reactor coolant system is dependent on the reactor coolant system pressure transient. The passive core cooling system water injected into the reactor coolant system provides for heat transfer from the core, prevents excessive core clad temperatures, and refloods the core (for large loss of coolant accidents) or keeps the core covered (for small loss of coolant accidents).

For small loss of coolant accidents, the control rods provide the initial core shutdown and the boron in the passive core cooling system tanks add negative reactivity to provide adequate shutdown at low temperatures.

Following the initial thermal-hydraulic transient for a loss of coolant accident event, the passive core cooling system continues to supply water to the reactor coolant system for long-term cooling. When the water level in the in-containment refueling water storage tank drops to the Low-3 level, the water

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level in the containment has increased to a sufficient level to provide recirculation flow. The in-containment refueling water storage tank Low-3 level signal opens the squib valves in the lines between the containment and the gravity injection line. Initially, some of the water remaining in the tank drains to the containment until the water levels equalize. During this drain, injection to the core continues. The redundant flow paths provide continued cooling of the core by recirculation of the water in the containment. Figure 6.3-3 provides process flow information illustrating passive core cooling system performance for the various modes of system operation.

6.3.3.3.3 Passive Residual Heat Removal Heat Exchanger Tube Rupture

Although a passive residual heat removal heat exchanger tube rupture is an event that results in a decrease in reactor coolant system inventory, severe core conditions do not result from this event. There is a spectrum of heat exchanger tube leak sizes that are possible. For a small initiating leak, the passive core cooling system temperature instrumentation for the heat exchanger is used to identify that this is a heat exchanger leak. If the leak rate is less than the Technical Specification limits, plant operation can continue indefinitely. If the leak rate exceeds the Technical Specification limits the plant would be shut down to repair the heat exchanger.

If a severe tube leak occurs, the operators can use available instrumentation to identify the leak source. Action can then be taken to remotely isolate the heat exchanger by closing the motor-operated inlet isolation valve, which is normally open. The plant would be shut down to repair the heat exchanger.

This event is addressed in Section 15.6.

6.3.3.4 Shutdown Events

The passive core cooling system components are available whenever the reactor is critical and when reactor coolant energy is sufficiently high to require passive safety injection. During low-temperature physics testing, the core decay heat levels are low and there is a negligible amount of stored energy in the reactor coolant. Therefore, an event comparable in severity to events occurring at operating conditions is not possible and passive core cooling system equipment is not required. The possibility of a loss of coolant accident during plant startup and shutdown has been considered.

During shutdown conditions, some of the passive core cooling system equipment is isolated. In addition, since the normal residual heat removal system is not a safety-related system, its loss is considered. Containment recirculation may be impacted by shutdown maintenance activities which remove access hatches or curbs located at the 107'-2" plant elevation. Subsection 3.4.1.2.2.1 discusses containment flooding events in further detail.

As a result, gravity injection is automatically actuated when required during shutdown conditions prior to refueling cavity floodup, as discussed in Subsection 6.3.3.3.2. The operator can also manually actuate other passive core cooling system equipment, such as the passive residual heat removal heat exchanger, if required for accident mitigation during shutdown conditions when the equipment does not automatically actuate.

6.3.3.4.1 Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heat-ups

During normal cooldowns, the steam generators are supplied by the startup feedwater pumps and steam from the steam generator is directed to either the main condenser or to the atmosphere. There are two nonsafety-related startup feedwater pumps, each of which is capable of providing sufficient feedwater flow to both steam generators to remove decay heat. These pumps are also automatically loaded on the nonsafety-related diesel-generators in the event normal ac power and offsite power are lost. Since these pumps are nonsafety-related, their failure is considered.

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Should the main feedwater (MFW) and SFW be unavailable, the PRHR HX modulation control system automatically actuates on a Low-2 narrow range steam generator level and Low-2 SFW flow present in at least one steam generator, coincident with at least one RCP running in loop 1, per Subsection 6.3.1.2.2. If the event progresses such that the steam generator narrow range water level setpoint Low-2 for two-out-of-two steam generators coincident with startup feedwater Low-2 flow setpoint in two-out-of-two steam generators coincident with T_{cold} - High in either loop are met then PRHR HX actuation signal will be generated. The passive residual heat removal heat exchanger can maintain the reactor coolant system temperature, as well as provide for reactor coolant system cooldown to conditions where the normal residual heat removal system can be operated.

Since the chemical and volume control system makeup pumps are nonsafety-related, they may not be available. In this case, the core makeup tanks automatically actuate as the cooldown continues and the pressurizer level decreases. The core makeup tanks operate in a water recirculation mode to maintain reactor coolant system inventory while the passive residual heat removal heat exchanger is operating.

The in-containment refueling water storage tank provides the heat sink for the passive residual heat removal heat exchanger. Initially, the heat addition increases the water temperature. Within one to two hours, the water reaches saturation temperature and begins to boil. The steam generated in the in-containment refueling water storage tank discharges to containment. Because the containment integrity is maintained during cooldown Modes 3 and 4, the passive containment cooling system provides the safety-related ultimate heat sink. Therefore, most of the steam generated in the in-containment refueling water storage tank is condensed on the inside of the containment vessel and drains back into the in-containment refueling water storage tank via the condensate return gutter arrangement. This allows it to function as a heat sink for greater than 14 days, as discussed in Subsection 6.3.1.2.1.

6.3.3.4.2 Loss of Normal Residual Heat Removal Cooling With The Reactor Coolant System Pressure Boundary Intact

During normal shutdown conditions, the normal residual heat removal system is placed into service at about 350°F to accomplish reactor coolant system cooldown to refueling temperatures. The normal residual heat removal system piping is safety-related and meets seismic Category I requirements to prevent pipe breaks that could result in a significant loss of reactor coolant during system operation. The pump motors and the electrical power supplies are nonsafety-related.

The system is designed so that with single failure of an active system component, it can maintain the plant in a hot shutdown condition (<350°F). It is also possible to perform a reactor coolant system cooldown, but at a slower rate than with full system capability. Heat removed by the normal residual heat removal system is transferred to the component cooling water system and then to the service water system. The heat removal path is powered by the nonsafety-related diesel-generators in the event that normal ac power and offsite power are lost.

Since the normal residual heat removal pumps are nonsafety-related, they may not be available. In this case, the reactor coolant system pressure boundary remains intact and the passive residual heat removal heat exchanger provides the safety-related heat removal flow path.

The normal residual heat removal system is operated once the reactor coolant system temperature is too low to support sufficient steam production for decay heat removal. With a loss of shutdown cooling, the reactor coolant system temperature does not increase sufficiently to initiate steam generator steaming and to reduce steam generator level. This is because the steam generators are normally filled, with a nitrogen purge established, during shutdown conditions. The loss of cooling would result in the heat up of the reactor coolant system and a pressure increase resulting in one or both of the normal residual heat removal system relief valves opening. This loss of fluid would result

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in a decrease in the pressurizer level; a Low-2 pressurizer level signal automatically actuates the core makeup tanks and the passive residual heat removal heat exchanger. The passive residual heat removal heat exchanger could also be manually actuated.

The passive residual heat removal heat exchanger is capable of functioning at low reactor coolant system temperatures and pressures, but it may not be able to maintain the initial reactor coolant system temperature. It can remove sufficient heat to maintain the reactor coolant system within the normal residual heat removal system design limits (400°F). This permits the normal residual heat removal system to be placed back in operation when it becomes available.

For this event, the reactor coolant system temperature is expected to increase and expand into the pressurizer. Reactor coolant system injection should not be required. The makeup pumps are aligned for automatic operation in the event that pressurizer level decreases, due to leakage. However, since they are nonsafety-related, they are considered unavailable for reactor coolant system makeup. Therefore should safety-related makeup be required, the core makeup tanks would automatically actuate and operate via water recirculation injection. For some scenarios, the core makeup tanks could drain down and actuate the automatic depressurization system valves. This would lead to injection via the in-containment refueling water storage tank and containment recirculation paths.

6.3.3.4.3 Loss of Normal Residual Heat Removal Cooling During Reduced Inventory

During reactor coolant system maintenance, the most limiting shutdown condition anticipated is with the reactor coolant level reduced and the reactor coolant system pressure boundary opened. It is normal practice to open the steam generator channel head manway covers to install the hot leg and cold leg nozzle dams during a refueling outage. In this situation, the normal residual heat removal system is used to cool the reactor coolant system. The AP1000 incorporates many features to reduce the probability of losing the normal residual heat removal system. However, since the normal residual heat removal system is nonsafety-related, its failure has been considered. The normal residual heat removal system is described Subsection 5.4.7.

In reduced inventory operation with the reactor coolant system depressurized and the pressure boundary opened, the passive residual heat removal heat exchanger is unable to remove the decay heat because the reactor coolant system cannot heat sufficiently above the in-containment refueling water storage tank temperature.

In this situation, core cooling is provided by the safety-related passive core cooling system, using gravity injection from the in-containment refueling water storage tank, while venting through the automatic depressurization system valves (and possibly through other openings in the reactor coolant system).

Prior to draining the reactor coolant system inventory below the no-load pressurizer level, the core makeup tanks are isolated to preclude inadvertent draining into the reactor coolant system while preparing for midloop operation. During plant shutdown, at 1000 psig, the accumulators are isolated to prevent inadvertent injection. In this configuration, the core makeup tanks and accumulators are isolated from the reactor coolant system, however these valves can be remotely opened with operator action to provide additional makeup water injection, if required.

Before the core makeup tanks are isolated, the automatic depressurization first-, second-, and third-stages valves are opened manually be the operators. The automatic depressurization system first-, second- and third-stage valves are required to remain open whenever the reactor coolant inventory is reduced or the upper core internals are in place. During an extended loss of normal residual heat removal system operation the stage one, two and three vent paths may not provide sufficient vent capability to allow gravity injection of water from the in-containment refueling water storage tank because of pressurizer surge line flooding. As a result, the automatic depressurization stage four

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paths are required. The stage four valves are opened manually as described in Subsection 19E.4.8.3 or automatically by a signal from the protection and monitoring system on a Low-4 hot leg level signal following a time delay.

The in-containment refueling water storage tank injection squib valves automatically open via the same Low-4 hot leg level signal that opens the automatic depressurization stage four valves. The operators can also open these injection and depressurization valves via the diverse actuation system. Once these valves open, injection from the in-containment refueling water storage tank provides gravity injection for core cooling. When the in-containment refueling water storage tank level drops to the Low-3 level, the squib valves in the containment recirculation line automatically open. This action initiates containment recirculation flow, with flow passing through the in-containment refueling water storage tank gravity injection lines, which provides long-term core cooling.

This arrangement provides automatic core cooling protection, while in reduced inventory operation while also providing protection (an evacuation alarm and sufficient time to evacuate) for maintenance personnel in containment during midloop operation. The time delay also provides the operators with time to take actions to restore nonsafety-related decay heat removal prior to actuating the passive core cooling system.

During reduced inventory conditions the capability of closing the containment is required. After the containment is closed, containment recirculation can continue indefinitely, with the decay heat generating steam which condenses on the containment vessel and drains back into the in-containment refueling water storage tank.

6.3.3.4.4 Loss of Normal Residual Heat Removal Cooling During Refueling

The normal residual heat removal system is normally used for decay heat removal during refueling operation. Its failure is considered because it is not a safety-related system. In this case, it is assumed that the reactor vessel head is removed and the water from the in-containment refueling water storage tank has been transferred to the refueling cavity, which is flooded to its high level condition. The passive residual heat removal heat exchanger is not available and containment integrity is expected to be relaxed with air locks and/or equipment hatches open.

Assuming that the refueling cavity was just flooded when the normal residual heat removal system fails, the refueling cavity water heats up to saturation temperature in about nine hours. With the slow heat-up of the refueling cavity water, there is ample time to close containment before significant steaming to the containment begins. The Technical Specifications require that containment closure capability be maintained during refueling MODES such that closure of the containment can be assumed. With the containment closed, water will not be lost from containment and long-term cooling can be maintained without subsequent need for cooling water makeup. Without closing the containment, boiling would reduce the water level to the top of the fuel assemblies in about five days.

6.3.4 Post-72 Hour Actions

The AP1000 passive core cooling system design includes safety-related equipment that is sufficient to automatically establish and maintain safe shutdown conditions for the plant following design basis events. The passive core cooling system can maintain safe shutdown conditions for 72 hours after an event without operator action and without both nonsafety-related onsite and offsite power.

There is only one action that may be required to provide long-term core cooling. There is a potential need for containment inventory makeup. The need for makeup to containment is directly related to the leakrate from the containment. With the maximum allowable containment leakrate, makeup to containment is not needed for about one month. A safety-related connection is available in the normal residual heat removal system to align a temporary makeup source to containment.

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6.3.5 Limits on System Parameters

The analyses show that the design basis performance of the passive core cooling system is sufficient to meet the core cooling requirements following an event, with the minimum engineered safety features equipment operating. To provide this capability in the event of the single failure of components, technical specifications are established for reactor operation.

The passive core cooling system equipment is not required to operate to support either normal power operation or shutdown operation of the plant. This reduces the probability that the passive core cooling system equipment is unavailable due to maintenance. Planned maintenance on the passive core cooling system equipment is accomplished during shutdown operations when the core temperatures are low, decay heat levels are low, and the Technical Specifications do not require availability of the equipment.

The principal system parameters and the number of components that may be out of operation during testing, quantities and concentrations of coolant available, and allowable time for operation in a degraded status are provided in the technical specifications.

If efforts to restore the operable status of the passive core cooling system equipment are not accomplished within technical specification requirements, the plant is required to be placed in a lower operational mode.

6.3.6 Inspection and Testing Requirements

6.3.6.1 Preoperational Inspection and Testing

Preoperational inspections and tests of the passive core cooling system are performed to verify the operability of the system prior to loading fuel. This testing includes valve inspection and testing, flow testing, and verification of heat removal capability.

Preoperational testing of the passive core cooling system is completed in conjunction with testing of the reactor coolant system following flushing and hydrostatic testing, with the system cold and the reactor vessel head removed. The passive core cooling system is aligned for normal power operation. This testing provides the following information:

- Satisfactory safeguards actuation signal generation and transmission
- Valve operating times
- Injection starting times
- Injection delivery rates

The preoperational testing program includes testing of the following passive core cooling system components:

- Core makeup tanks
- Accumulators
- In-containment refueling water storage tank
- Containment recirculation

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- Passive residual heat removal heat exchanger

Conformance with the recommendations of Regulatory Guide 1.79 is described in Subsection 1.9.1. Preoperational testing of the passive core cooling system is conducted in accordance with the requirements presented in Subsection 14.2.9.1.3.

6.3.6.1.1 Flow Testing

Initial verification of the resistance of the passive core cooling injection lines is performed by conducting a series of flow tests for the core makeup tanks, accumulators, in-containment refueling water storage tank, and containment recirculation piping. The calculated flow resistances are bounded by the resistances used in the Chapter 15 safety analyses.

6.3.6.1.2 Heat Transfer Testing

Initial verification of the heat transfer capability of the passive residual heat removal heat exchanger is performed by conducting a forced flow test. This test is conducted during hot functional testing of the reactor coolant system and performed as described in Subsection 14.2.9.1.3 item g).

6.3.6.1.3 Preoperational Inspections

Preoperational inspections are performed to verify that important elevations associated with the passive core cooling system components are consistent with the accident analyses presented in Chapter 15. The following elevations are verified:

- The bottom inside surface of each core makeup tank is at least 7.5 feet above the direct vessel injection nozzle centerline.
- The bottom inside surface of the in-containment refueling water storage tank is at least 3.4 feet above the direct vessel injection nozzle centerline.
- The centerline of the upper passive residual heat removal heat exchanger channel head is at least 26.3 feet above the hot leg centerline.
- The pH baskets are located below plant elevation 107 feet, 2 inches.

Inspections of the passive core cooling system tanks and pH adjustment baskets are conducted to verify that the actual tank volumes are greater than or equal to volume assumed in the Chapter 15 accident analyses. Inspections to determine dimensions of the core makeup tanks, accumulators, in-containment refueling water storage tank, and pH adjustment baskets are conducted, and calculations are performed to verify that actual volume is not less than the corresponding minimum required volume listed in Table 6.3-2.

6.3.6.2 In-Service Testing and Inspection

In-service testing and inspection of the passive core cooling system components and the associated support systems are planned. The passive core cooling system components and systems are designed to meet the intent of the ASME Operations and Maintenance (OM) Code, for in-service testing. A description of the in-service testing program is provided in Subsection 3.9.6.

Two basic types of in-service testing are performed on the passive core cooling system components:

- Periodic exercise testing of active components during power operation (for example, cycling of specific valves)

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- Operability testing of specific passive core cooling system features during plant shutdown (for example, accumulator injection flow to the reactor vessel or leak testing of containment isolation valves during selected plant shutdown.

The passive core cooling system includes specific features to support in-service test performance:

- Remotely operated valves can be exercised during routine plant maintenance
- Level, pressure, flow, and valve position instrumentation is provided for monitoring required passive core cooling system equipment during plant operation and testing
- Permanently installed test lines and connections are provided for operability testing

6.3.6.3 Mitigation of Gas Accumulation

Periodic system surveillance and venting procedures, in addition to specific design features, are implemented that aim to prevent gas accumulation and minimize or eliminate gas whenever found. Locations identified by the gas accumulation assessment have been equipped with manual vent valves or continuously monitored and alarmed pipe stubs with manual vent valves. These locations are specified within the periodic system surveillance and venting procedures as locations of high importance. Locations outfitted with pipe stub collection and alarm features have Technical Specifications including Surveillance Requirements to continuously monitor for gas accumulation and Required Actions subsequent to identifying gas accumulation in those locations to vent the identified gas accumulation. Plant startup and operational procedures include venting and surveillance steps that provide a means to track and trend accumulated gas such that problem areas can be systematically identified, monitored, and corrected.

6.3.6.3.1 System Gas Accumulation Assessment

Reviews of pipe layout and routing drawings to identify high-point vent and low-point drain locations are included as part of system design finalization activities. This existing design activity was expanded for the AP1000 passive safety systems, to integrate the draft Interim Staff Guidance (ISG) document ISG-019 regarding gas intrusion assessment guidance into the design process, helping to confirm that the potential issues identified in Generic Letter 2008-01 have been addressed within the design. Westinghouse also performed a comprehensive assessment for gas intrusion within the passive safety systems, consistent with the methodology in NEI 09-10 as applied in current operating plants, and consistent with the additional guidance in the ISG.

6.3.6.3.2 System Design Features to Mitigate Gas Intrusion

The gas intrusion assessment described in Subsection 6.3.6.3.1 helped to identify:

- Potential gas accumulation locations in the passive core cooling system piping.
- Potential gas intrusion mechanisms during various plant conditions (including plant startup, shutdown, post-maintenance system restoration and filling, power operation, and accident conditions).
- Passive core cooling system design features to provide the capability to perform system high-point venting and to continuously monitor several high-point locations.

These passive core cooling system design features help to eliminate the potential for significant gas accumulation in specific passive safety system injection lines that could adversely impact passive

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safety system operation. System venting capabilities are provided for the following passive safety system locations:

- IRWST injection line squib valve inlet lines
  - Vents located at the piping high points upstream of the parallel paths in both IRWST safety injection lines
  - Vents located between the check and squib valves in each line of the parallel paths in both IRWST safety injection lines
- Core makeup tank outlet lines
  - Vents located on bodies of both in-series check valves in both core makeup tank outlet lines
- Containment recirculation Line A
  - Vents located at the high points in the common containment recirculation Line A path between the recirculation squib valves and the IRWST injection line (before and after the spent fuel system connection tee)

The potential for gas accumulation in passive safety system IRWST injection lines following accumulator injection is precluded by connecting the accumulator injection line in the direct vessel injection line riser section vertically above the IRWST injection line connection to the direct vessel injection line riser.

In addition, passive safety system design features are provided to monitor for gas accumulation at several specific locations. These design features include pipe stub gas collection chambers with redundant instrumentation at each high point, are continuously monitored and alarmed, have hard piped vent lines, are accessible during power operation, and include Technical Specifications and Surveillance Requirements specifically intended to identify unintended gas accumulations that could potentially challenge passive safety system operability for the following locations:

- Core makeup tank inlet highpoints
- Passive residual heat removal heat exchanger inlet high point
- IRWST injection line squib valve outlet high points

To ensure that all of the vent locations identified above function properly, notes are included on the system piping and instrumentation diagrams that specify layout sloping requirements. The intent of the layout requirements is to help ensure that the installed vents can effectively vent accumulated gases from the associated line segments. These notes also appear on the isometric drawings to make certain that the layout sloping requirements are observed during fabrication, construction, and installation.

The continuously monitored and alarmed pipe stub gas collection chamber, including Technical Specifications and Surveillance Requirements, was not utilized for the high points in the containment recirculation Line A because there are no credible postulated gas intrusion mechanisms by which gas is expected to migrate into these lines (except in the event of improper venting during line filling operations such as after maintenance). This isolated piping section is maintained in a standby condition prior to passive safety system actuation. This local high point is located between the containment recirculation squib valves and the IRWST injection line squib valves, and it remains

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connected to the IRWST so that the tank water elevation head maintains the pressure in this line prior to actuation.

Passive safety system locations equipped with manual vent valves will be inspected according to the system surveillance and venting procedures to eliminate any identified gas accumulation. Locations equipped with pipe stub gas collection and alarm features will be continuously monitored via alarm indications in the main control room. Because the locations with pipe stub collection and alarm features are continuously monitored and have Surveillance Requirements and Required Actions, the potential exists that these locations will be vented at RCS pressure. Consequently, these locations have manual vent valves and are hard-piped to either the IRWST or the reactor coolant drain tank for potential venting at RCS pressure.

For the AP1000, the structures, systems, and components (SSCs) of the passive safety systems that are used to establish and maintain safe shutdown conditions for the plant are identified and discussed in Subsections 7.4.1.1 and 7.4.2, and listed in Table 7.4-1. These same SSCs that provide the AP1000 safe shutdown capability also provide the passive, safety-related accident mitigation functions, including those that are equivalent to the emergency core cooling system, decay heat removal, and containment spray system functions for active plants specified in the generic letter.

6.3.7 Instrumentation Requirements

Instrumentation channels employed for actuation of passive core cooling system operation are described in Section 7.3. This subsection describes the instrumentation provided for monitoring passive core cooling system components during normal plant operation and also during passive core cooling system post-accident operation. Alarms are annunciated in the main control room.

6.3.7.1 Pressure Indication

6.3.7.1.1 Accumulator Pressure

Two pressure channels are installed on each accumulator. The pressure indications are used to confirm that accumulator pressure is within bounds of the assumptions used in the safety analysis. Each channel provides pressure indication in the main control room and also provides high-pressure and low-pressure alarms.

6.3.7.1.2 Passive Residual Heat Removal Heat Exchanger Pressure

One pressure indicator is installed on the passive residual heat removal heat exchanger inlet line. The instrument provides pressure indication in the main control room.

6.3.7.2 Temperature Indication

6.3.7.2.1 Core Makeup Tank Inlet Line Temperature

Individual temperature channels are installed on the inlet line for each core makeup tank. The temperature indication is used to determine if there is a sufficient thermal gradient for system operation. Each channel provides temperature indication in the main control room and also provides a low-temperature alarm.

6.3.7.2.2 Passive Residual Heat Removal Heat Exchanger Inlet Temperature

One temperature channel is installed on the inlet line to the passive residual heat removal heat exchanger. The temperature indication is used to detect reactor coolant system leakage into the passive residual heat removal heat exchanger, either through the discharge valves or from tube

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leakage into the in-containment refueling water storage tank, and to identify the leakage path. The channel provides temperature indication in the main control room and also provides a high-temperature alarm.

6.3.7.2.3 In-Containment Refueling Water Storage Tank Temperature

Four temperature channels are installed on the in-containment refueling water storage tank. The temperature indications are used to confirm that in-containment refueling water storage tank temperature is within the bounds of the assumptions used in the safety analysis. The temperature indications are also used to monitor in-containment refueling water storage tank temperature during passive core cooling system operation. Each channel provides temperature indication and high-temperature alarms in the main control room.

6.3.7.2.4 Core Makeup Tank Outlet Line Temperature

Two temperature channels are installed, one on each core makeup tank outlet line. The temperature indication is used to detect reactor coolant system leakage into the core makeup tanks. Each channel provides temperature indication in the main control room and also provides a high-temperature alarm.

6.3.7.2.5 Direct Vessel Injection Line Temperature

Two temperature channels are installed, one on each direct vessel injection line. The temperature indication is used to detect reactor coolant system leakage back through the direct vessel injection lines to the core makeup tanks, accumulator, or in-containment refueling water storage tank. Each channel provides temperature indication in the main control room.

6.3.7.2.6 Passive Residual Heat Removal Heat Exchanger Inlet High Point Temperature

One temperature channel is installed on the passive residual heat removal heat exchanger inlet line. The temperature indication is used to determine that the temperature in the inlet is within the bounds of the assumptions used in the safety analysis. The channel provides temperature indication and a low temperature alarm in the main control room.

6.3.7.3 Passive Residual Heat Removal Heat Exchanger Outlet Flow Indication

Two flow channels are installed on the passive residual heat removal outlet line. The flow indications are used to monitor and control passive residual heat removal heat exchanger operation. Each channel provides flow indication in the main control room.

6.3.7.4 Level Indication

6.3.7.4.1 Core Makeup Tank Level

Ten level channels are installed on each core makeup tank. There are 2 wide range level channels which are used to confirm that the core makeup tanks are maintained at full water level during normal operation. There are four narrow range level channels which are used to control the actuation of the automatic depressurization system stages 1, 2, and 3 valves. There are four narrow range level channels which are used to control the actuation of the automatic depressurization system stage 4 valves. Each wide range channel provides level indication and alarms in the main control room. Each narrow range channel provides level indication and alarms in the main control room and actuation of the automatic depressurization system. Each set of two narrow range channels share upper and lower level tap connections with the core makeup tanks; a failure modes and effects analysis

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confirms the ability of this arrangement to tolerate single failures (Reference 2). Each upper level tap line has a downward slope of ≥ 2.4 degrees from the centerline of the connection to the core makeup tank to the centerline of the connection to the standpipe.

6.3.7.4.2 Accumulator Level

Two level channels are installed on each accumulator. The level indications are used to confirm that accumulator level is within bounds of the assumptions used in the safety analysis. Each channel provides level indication and both high and low level alarms in the main control room.

6.3.7.4.3 In-Containment Refueling Water Storage Tank Level

Nine level channels are installed on the in-containment refueling water storage tank. There are two upper narrow range level channels. These level indications are used to confirm that in-containment refueling water storage tank normal level is within the bounds of the assumptions used in the safety analysis. There are three wide range level channels. Two of these level indications are used to provide refueling cavity and spent fuel pool cooling system isolation. There are four lower narrow range level channels. These level indications are used to provide containment recirculation valve repositioning. Each channel provides level indication in the main control room and provides level alarms.

The in-containment refueling water storage tank is sized and the level alarm setpoints selected to provide adequate in-containment refueling water storage tank injection (and spill flow to containment for a direct vessel injection line break) until containment floodup is sufficient to provide recirculation flow.

6.3.7.4.4 Containment Level

Three level channels are installed on the containment. The level indications are used to monitor containment level from the reactor vessel cavity up to the maximum containment floodup elevation. Each channel provides level indication and alarms in the main control room.

6.3.7.5 Containment Radiation Level

Four channels are installed for the containment radiation. The radiation indications are used to monitor containment conditions. Each channel provides radiation indication and high radiation alarms in the main control room. Section 11.5 provides additional information.

6.3.7.6 Valve Position Indication and Control

6.3.7.6.1 Valve Position Indication

Individual valve position is provided for the safety-related, remotely actuated valves listed in Table 6.3-1. In addition, valve position is provided for certain manually operated valves, as described in Subsection 6.3.2.2.8.2, that can isolate redundant passive core cooling equipment, if mispositioned. The incontainment refueling water injection check valves and containment recirculation check valves have nonintrusive position indication.

For certain passive core cooling system valves with position indication, alarms in the main control room are provided to alert the operators to valve mispositioning. For the passive residual heat removal heat exchanger discharge valves, valve position indication is used to initiate a reactor trip upon opening of these valves while the reactor is at power.

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6.3.7.6.2 Valve Position Control

Valve controls are provided for remotely operated passive core cooling system valves. Table 6.3-1 provides a list of the passive core cooling system remotely operated valves. These remotely operated valves have controls in the main control room.

6.3.7.6.2.1 Accumulator Motor-Operated Valve Controls

As part of the plant shutdown procedures, the operator is required to close the accumulator motor-operated valves. This prevents a loss of accumulator water inventory to the reactor coolant system when the reactor coolant system is depressurized. The valves are closed after the reactor coolant system has been depressurized to below the setpoint to block the safeguards actuation signal. The redundant pressure and level alarms on each accumulator function to alert the operator to close these valves, if any are inadvertently left open. Power is locked out after the valves are closed. During plant startup, the operator is directed by plant procedures to energize and open these valves prior to reaching the reactor coolant system pressure setpoint that unblocks the safeguards actuation signal. Redundant indication and alarms are available to alert the operator if a valve is inadvertently left closed once the reactor coolant system pressure increases beyond the setpoint. Power is also locked out after these valves are opened.

The accumulator isolation valves are not required to move during power operation. For a description of limiting conditions for operation and surveillance requirements of these valves, refer to the technical specifications. The accumulator isolation valves receive a safeguards actuation signal to confirm that they are open in the event of an accident. As a result of the power lock out, technical specifications, and the redundant position indication and alarms, the valve controls are nonsafety-related.

6.3.7.6.2.2 In-Containment Refueling Water Storage Tank Injection Motor-Operated Valve Controls

The motor-operated valves in each in-containment refueling water storage tank injection line are normally open during all modes of normal plant operation. Power to these valves is locked out. Redundant valve position indication and alarms are provided to alert the operator if a valve is inadvertently closed. The technical specifications specify surveillances to show that these valves are open. These valves also receive a safeguards actuation signal to confirm that they are open in the event of an accident. As a result of the power lock out, the redundant position indication and alarms and the technical specifications the valve controls are nonsafety-related.

6.3.7.6.2.3 Passive Residual Heat Removal Heat Exchanger Inlet Motor-Operated Valve Control

The motor-operated valve in the passive residual heat removal heat exchanger inlet line is normally open during normal plant operation. Power to this valve is locked out. Redundant valve position indications and alarms are provided to alert the operator if the valve is open. This valve also receives an actuation signal to confirm that it is open in the event of an accident.

6.3.7.7 Automatic Depressurization System Actuation at 24 Hours

A timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This prevents discharging the Class 1E dc power sources such that they are no longer able to operate the automatic depressurization system valves. If power becomes available to the dc batteries and they are no longer discharging prior to activation of the timer, then the automatic depressurization system actuation would be delayed. If the plant does not need actuation of the automatic depressurization system based on having stable pressurizer level,

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full core makeup tanks, and high and stable in-containment refueling water storage tank levels, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the automatic depressurization system and allow for its actuation later should the plant conditions unexpectedly degrade.

6.3.8 Combined License Information

6.3.8.1 Containment Cleanliness Program

A program to limit the amount of debris that might be left in the containment following refueling and maintenance outages is described below. The cleanliness program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation to items that do not produce debris (physical or chemical), which could be transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break that becomes submerged during recirculation. The cleanliness program shall limit the amount of latent debris and fibrous material located within the containment, as identified in Subsection 6.3.2.2.7.1, item 12.

Administrative procedures implement the containment cleanliness program. Implementation of the program minimizes the amount of debris left in containment following personnel entry and exits. The program includes, as a minimum, the following:

Responsibilities

The program defines the organizational responsibilities for implementing the program; defines personnel and material controls; and defines the inspection and reporting requirements.

Implementation

Containment Entry/Exit

- Controls to account for the quantities and types of materials introduced into the containment.
- Limits on the types and quantities of materials, including scaffolding and tools, to ensure adequate accountability controls. This may be accomplished by the work management process. Storage of aluminum is prohibited without engineering authorization. Cardboard boxes or miscellaneous packing material is not brought into containment without approval.
- If entries are made at power, prohibited materials and limits on quantities of materials that may generate hydrogen are established.
- Controls for loose items, such as keys and pens, which could be inadvertently left in containment.
- Methods and controls for securing any items and materials left unattended in containment.
- Administrative controls for accounting for tools, equipment and other material are established.
- Administrative controls for accounting of the permanent removal of materials previously introduced into the containment.

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- Limits on the types and quantities of materials, including scaffolding and tools, that may be left unattended in containment during outages and power operation. Types of materials considered are tape, labels, plastic film, and paper and cloth products.
- Requirements and actions to be taken for unaccounted for material.
- Requirements for final containment cleanliness inspections.
- Record keeping requirements for entry/exit logs.

Housekeeping

Housekeeping procedures require that work areas be maintained in a clean and orderly fashion during work activities and returned to original conditions (or better) upon completion of work.

Sampling Program

A sampling program is implemented consistent with NEI Guidance Report 04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology” as supplemented by the NRC in the “Safety Evaluation by The Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), ‘Pressurized Water Reactor Sump Performance Evaluation Methodology.’” Latent debris sampling is implemented before startup. The sampling is conducted after containment exit cleanliness inspections to provide reasonable assurance that the plant latent debris design bases are met. Sampling frequency and scope may be adjusted based on sampling results. Results are evaluated post-start up and any nonconforming results will be addressed in the Corrective Action Program.

6.3.8.2 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA

The long-term core cooling considering debris resulting from a LOCA together with debris that exists before a LOCA is addressed in APP-GW-GLR-079 (Reference 3), and the applicable changes are incorporated.

6.3.9 References

1. WCAP-8966, “Evaluation of Mispositioned ECCS Valves,” September 1977.
2. WCAP-13594 (P), WCAP-13662 (NP), “FMEA of Advanced Passive Plant Protection System,” Revision 1, June 1998.
3. APP-GW-GLR-079, “AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA,” Westinghouse Electric Company LLC.
4. APP-GW-GLN-147, “AP1000 Containment Recirculation and IRWST Screen Design,” Westinghouse Electric Company LLC.
5. WCAP-17938 (P-A/NP-A), “AP1000 In-Containment Cables and Non-Metallic Insulation Debris Assessment,” Revision 3, June 2018.
6. NEI 04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” Revision 0.

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Table 6.3-1
Passive Core Cooling System - Remote Actuation Valves

	Normal Position	Actuation Position	Failed Position	Notes
Core Makeup Tanks				
CMT inlet isolation MOV (V002A/B)	Open	Open	As is	(1,4)
CMT outlet isolation AOV (V014A/B,V015A/B)	Closed	Open	Open	
CMT fill/drain isolation AOV (V230A/B)	Closed	N/A	Closed	(4)
Accumulators				
Accumulators vent isolation SOV (V021A/B)	Closed	N/A	Closed	(4)
Accumulator discharge MOV (V027A/B)	Open	Open	As is	(2,4)
Nitrogen supply isolation AOV (V042)	Open	Closed	Closed	
Accumulator fill/drain isolation AOV (V232A/B)	Closed	N/A	Closed	(4)
In-Containment Refueling Water Storage Tank				
IRWST injection line MOV (V121A/B)	Open	Open	As is	(2,4)
IRWST injection line squib (V123A/B, V125A/B)	Closed	Open	As is	
Containment Recirculation Sump Valves				
Recirculation line MOVs (V117A/B)	Open	Open	As is	(3,4)
Recirculation line squib valves (V118A/B, V120A/B)	Closed	Open	As is	
Passive Residual Heat Removal Heat Exchanger				
PRHR HX inlet MOV (V101)	Open	Open	As is	(2,4)
PRHR HX outlet AOVs (V108A/B)	Closed	Open	Open	
IRWST gutter isolation AOVs (V130A/B)	Open	Closed	Closed	
Automatic Depressurization System Valves				
ADS Stage 1 MOVs (V001A/B, V011A/B)	Closed	Open	As is	
ADS Stage 2 MOVs (V002A/B, V012A/B)	Closed	Open	As is	
ADS Stage 3 MOVs (V003A/B, V013A/B)	Closed	Open	As is	
ADS Stage 4 MOVs (V014A/B/C/D)	Open	Open	As is	(3,4)
ADS Stage 4 squib valves (V004A/B/C/D)	Closed	Open	As is	

Notes:
(1) These valves are normally in the correct post-accident position, but receive confirmatory actuation signals to redundant controllers.
(2) These valves are normally in the correct post-accident position with their power locked out. They also receive confirmatory actuation signals.
(3) These valves are normally in the correct post-accident position, but receive confirmatory actuation signals.
(4) The operation of these valves is not safety-related.

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Table 6.3-2 (Sheet 1 of 2)
Component Data - Passive Core Cooling System

Passive RHR HX			
Number		1	
Type		Vertical C-Tube	
Case		Design	
Heat transfer (BTU/hr)		2.01 E+08	
		Tube side	Shell side
Fluid		Reactor coolant	IRWST water
Design flow (lb/hr)		5.03 E+05	N/A
Temperature	in (°F)	567	120
	out (°F)	199	N/A
Design pressure (psig)		2485	N/A
Design temperature (°F)		650	N/A
Material		Alloy 690	N/A
AP1000 equipment class		A	N/A
Core Makeup Tanks			
Number		2	
Type		Vertical, cylindrical, hemispherical heads	
Minimum volume (cubic feet)		2487	
Design pressure (psig)		2485	
Design temperature (°F)		650	
Material		Carbon-steel, stainless steel clad	
AP1000 equipment class		A	
Accumulators			
Number		2	
Type		Spherical	
Volume (cubic feet)		2000	
Design pressure (psig)		800	
Design temperature (°F)		300	
Material		Carbon-steel, stainless steel clad	
AP1000 equipment class		C	

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Table 6.3-2 (Sheet 2 of 2)
Component Data - Passive Core Cooling System

IRWST		
Number	1	
Type	Integral to containment internal structure	
Volume, minimum water (cubic feet)	73,100	
Design pressure (psig)	5	
Design temperature (°F)	150 *	
Material	Wetted surfaces are stainless steel	
AP1000 equipment class	C	
Spargers		
Number	2	
Type	Cruciform	
Flow area of holes (in^2)	274	
Design pressure (psig)	600	
Design temperature (°F)	500	
Material	Stainless Steel	
AP1000 equipment class	C	
pH Adjustment Baskets		
Number	4 Total	
Type	Rectangular	
Volume minimum total (cubic feet)	560	
Material	Stainless steel	
AP1000 equipment class	C	
Screens	IRWST	Containment Recirculation
Number	3	2
Surface area, screen (square feet)	IRWST Screens A and B: ≥ 575 per screen IRWST Screen C: ≥ 1150	≥ 2,500 per screen
Material	Stainless steel	Stainless steel
AP1000 equipment class	C	C

Note:
* Several times during plant life, the refueling water could reach 250°F.

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Table 6.3-3 (Sheet 1 of 4)
Failure Mode and Effects Analysis -
Passive Core Cooling System Components

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
CMT outlet isolation AOVs V014A/B, V015A/B Normally closed/ fail open	Failure to open on demand	All design basis events	No safety-related effect since each valve has a redundant, parallel isolation AOV, which provides flow through a parallel branch line for the affected CMT. The other CMT is unaffected.[(a)]	Valve position indication alarm in MCR and at RSW	
CMT discharge line check valves V016A/B, V017A/B Normally open	Failure to close on reverse flow	All design basis events	No safety-related effect since each valve has a redundant, series check valve which closes to prevent reverse flow, during a cold leg (large) LOCA or cold leg balance line break, preventing accumulator flow from bypassing the reactor vessel.	No valve position indication	
Accumulator nitrogen supply/ vent valves V021A/B, V045 Normally closed/ fail closed	Spurious opening	All design basis events	No safety-related effect since each valve has either a normally closed redundant, series isolation SOV or a check valve in each vent flow path, that prevents accumulator nitrogen from leaking out of the accumulator, which could degrade accumulator injection.	Valve position indication in MCR and at RSW Accumulator low pressure alarm in MCR and at RSW	
Accumulator nitrogen supply containment isolation AOV V042 Normally open/ fail closed	Failure to close on demand	All design basis events	No safety-related effect since each valve has a redundant, series isolation check valve which independently closes on reverse flow in the line, preventing reactor coolant from leaking out of containment.	Valve position indication alarm in MCR and at RSW	
Accumulator nitrogen supply containment isolation check valve V043 Normally open	Failure to close on reverse flow	All design basis events	No safety-related effect since each valve has a redundant, series isolation AOV, actuated by a separate division, which closes to prevent reactor coolant from leaking out of containment.	No valve position indication	

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Table 6.3-3 (Sheet 2 of 4)
Failure Mode and Effects Analysis -
Passive Core Cooling System Components

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
PRHR HX outlet line isolation AOVs V108A/B Normally closed/ fail open	Failure to open	All design basis events	No safety-related effect since each valve has a redundant, parallel isolation AOV, actuated by a separate division, which opens to provide PRHR HX flow through a parallel branch line.	Valve position indication alarm in MCR and at RSW PRHR HX flow indication in MCR & RSW	
IRWST gravity injection line check valves V122A/B, V124A/B Normally closed	Failure to open	All design basis events	No safety-related effect since each valve has a redundant flow path through a check valve and a squib valve that open to provide gravity injection through a parallel branch line. The other IRWST gravity injection line is unaffected.	Valve position indication in MCR and at RSW	
IRWST gravity injection line squib valves V123A/B, V125A/B Normally closed/ fail as is	Failure to open	All design basis events	No safety-related effect since each valve has a redundant flow path through a check valve and a squib valve that open to provide gravity injection through a parallel branch line. The other IRWST gravity injection line is unaffected.	Valve position indication in MCR and at RSW	
IRWST gutter isolation valves V130A/B Normally open/ fail closed	Failure to close	All design basis events	No safety-related effect since each valve has a redundant, series isolation AOV, actuated by a separate division, which closes to divert the gutter flow into the IRWST.	Valve position indication alarm in MCR and at RSW	
Containment recirculation line check valves V119A/B Normally closed	Failure to open	All design basis events	No safety-related effect since each valve has a redundant flow path through a MOV and a squib valve, actuated by separate divisions, that open to provide recirculation through a parallel branch line. The other containment recirculation line is unaffected.	Valve position indication in MCR and at RSW	

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Table 6.3-3 (Sheet 3 of 4)
Failure Mode and Effects Analysis -
Passive Core Cooling System Components

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
Containment recirculation line squib valves V120A/B Normally closed/ fail as is	Failure to open	All design basis events	No safety-related effect since each valve has a redundant flow path through a MOV and a squib valve, actuated by separate divisions, that open to provide recirculation through a parallel branch line. The other containment recirculation line is unaffected.	Valve position indication in MCR and at RSW	
Containment recirculation line squib valves V118A/B Normally closed/ fail as is	Failure to open	All design basis events	No safety-related effect since each valve has a redundant flow path through a check valve and a squib valve, actuated by separate divisions, that independently open to provide recirculation through a parallel branch line. The other containment recirculation line is unaffected.	Valve position indication in MCR and at RSW	
Accumulator fill/ drain line isolation AOVs V232A/B Normally closed/ fail closed	Spurious opening	All design basis events	No safety-related effect since each valve has a normally closed redundant, series isolation valve in each drain flow path, which prevents draining water from the accumulator.	Valve position indication alarm in MCR and at RSW	
CMT fill line isolation AOVs V230A/B Normally closed/ fail closed	Spurious opening	All design basis events	No safety-related effect since each valve has a redundant, series check valve that closes on reverse flow and prevents draining water from the CMT.	Valve position indication alarm in MCR and at RSW	
CMT fill line check valves V231A/B Normally closed	Failure to close on reverse flow	All design basis events	No safety-related effect since each valve has a normally closed redundant, series AOV that prevents draining water from the CMT.	No valve position indication CMT low level indication alarm in MCR and at RSW	

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Table 6.3-3 (Sheet 4 of 4)
Failure Mode and Effects Analysis -
Passive Core Cooling System Components

Component	Failure Mode	Plant Condition	Effect on System Operation	Failure Detection Method	Remarks
ADS Stage 1 to 3 MOVs V001A/B, V011A/B, V002A/B, V012A/B, V003A/B, V013A/B, and ADS Stage 4 squib valves V004A/B/C/D Normally closed/ fail as is applies to all of the above	Failure to open on demand	All design basis events	Failure to open blocks reactor coolant system vent flow through the one of two parallel branch lines of the affected ADS valve stage. Failure of a Stage 4 ADS valve is the most limiting single valve failure from the standpoint of ADS performance, based on this stage being the largest valve size. With the failure of ADS path, the ADS vent flow capacity is reduced, but safety analysis has demonstrated that the limiting Stage 4 ADS valve failure still meets design basis reactor coolant system venting requirements.	ADS Stage 1 to 3 MOVs have valve position indication alarm in the MCR and at RSW. ADS Stage 4 squib valves have valve position indication in the MCR and at RSW.	
Class 1E direct current and UPS system distribution switchgear division IDSA DS 1 IDSB DS 1 IDSC DS 1 IDSD DS 1	Failure of a dc power source	All design basis events	Failure of a single dc power source from either Division A or Division B is the most limiting dc failure. The limiting PXS components are the IRWST injection/containment recirc. valves and the ADS valves. Failure of either of these dc power sources can prevent actuation of the ADS Stage 1 and Stage 3 MOVs in one group of ADS valves. The other ADS valves are unaffected by this failure. This dc power failure can also cause failure of one (of 4) IRWST injection squib valves and one (of 4) squib recirculation valves. The ADS vent flow and IRWST injection/containment recirculation capacity is reduced, but safety analysis has demonstrated that this limiting valve failure combination still meets design basis reactor coolant system venting/injection requirements.	Valves position indication alarm in MCR and at RSW	For other PXS components, the loss of a Class 1E division either actuates the affected AOVs to a fail-safe position, or does not affect MOVs which are already in appropriate positions

Notes

a: The same two PMS divisions are needed to actuate both CMT outlet valves for a given CMT (divisions B and D for V014A and V015A and divisions A and C for V014B and V015B). Redundant LCLs within each PMS division ensure CMT injection will occur when necessary with the single failure of an LCL.

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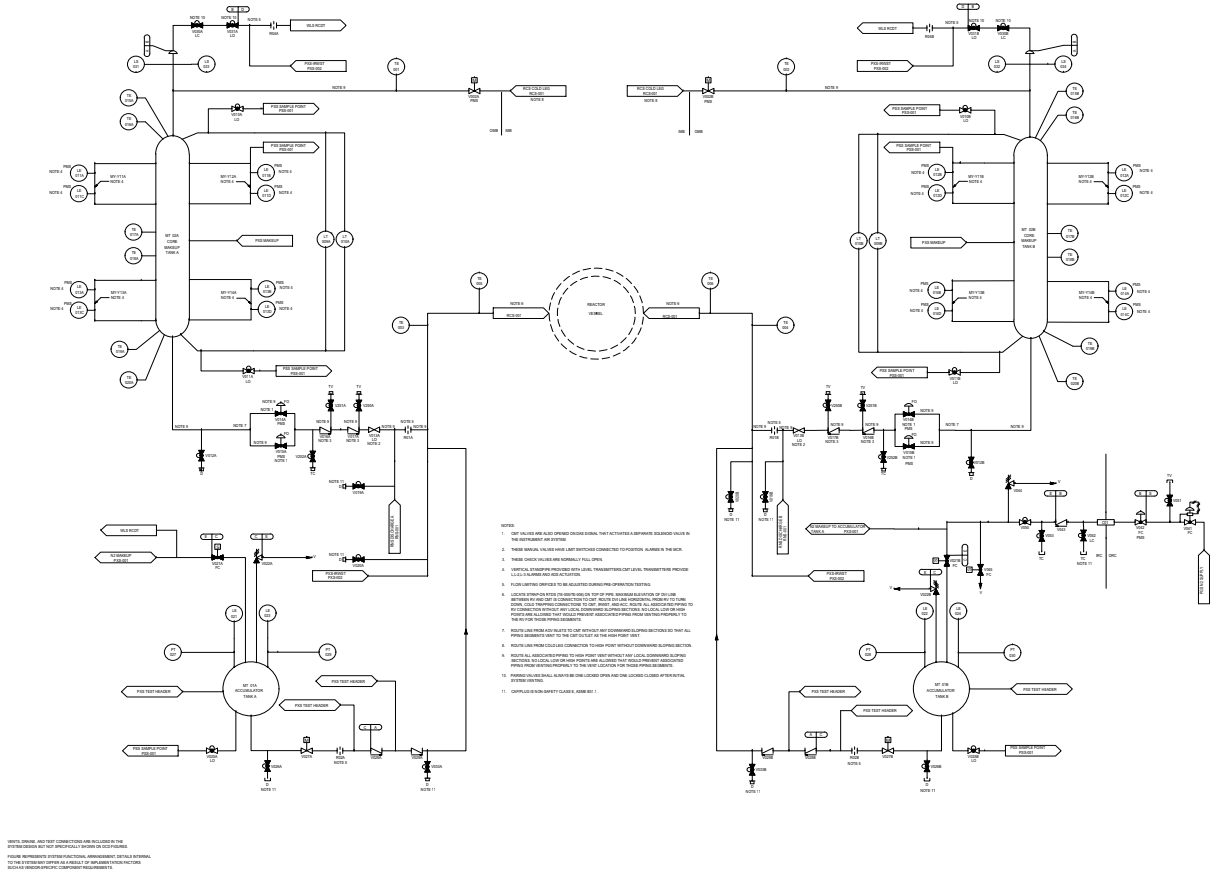

Figure 6.3-1 (Sheet 1 of 3)
Simplified Passive Core Cooling System
Piping and Instrumentation Diagram
(REF) PXS 001

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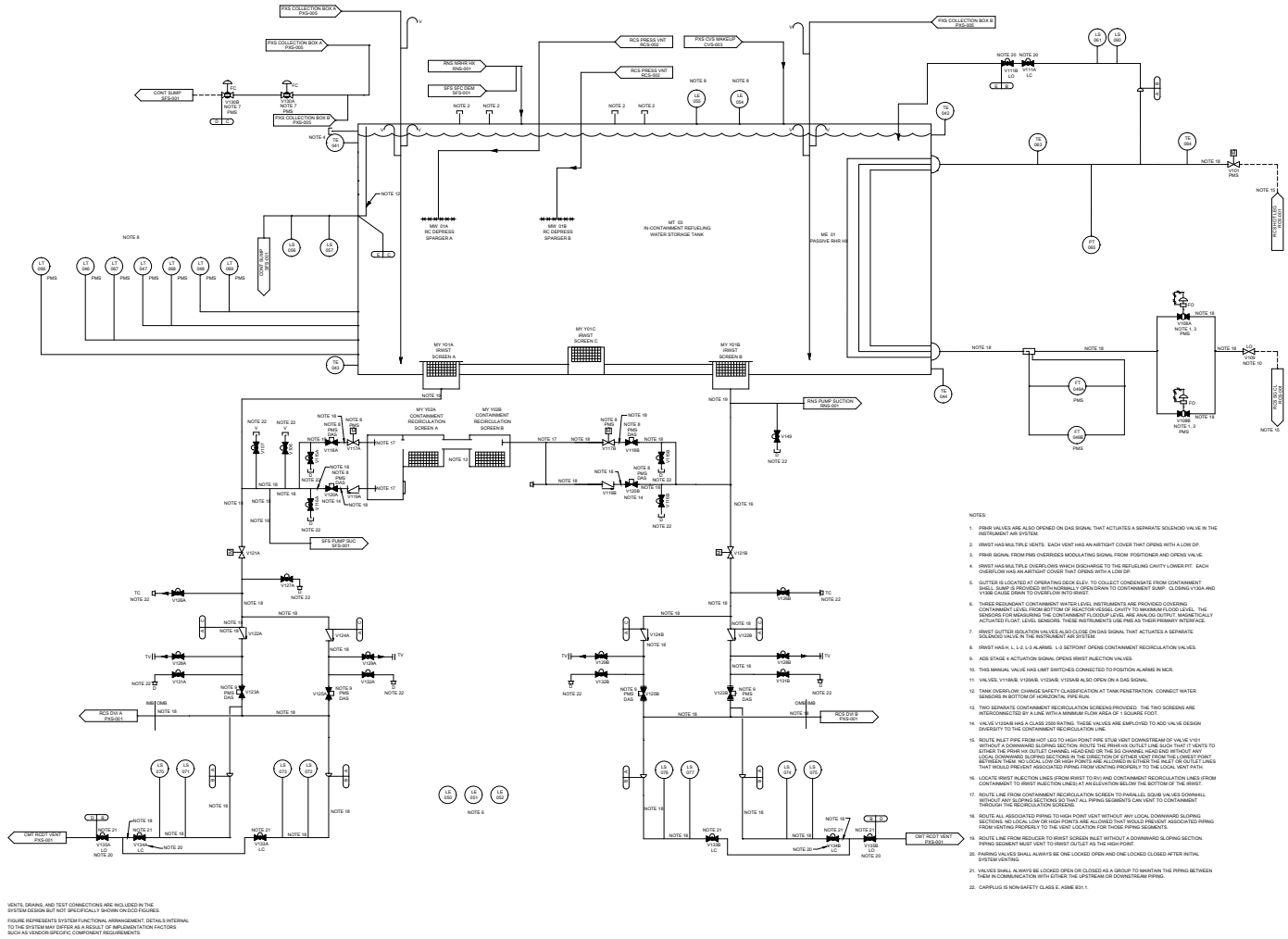

Figure 6.3-1 (Sheet 2 of 3)
Simplified Passive Core Cooling System Piping and Instrumentation Diagram (REF) PXS 002

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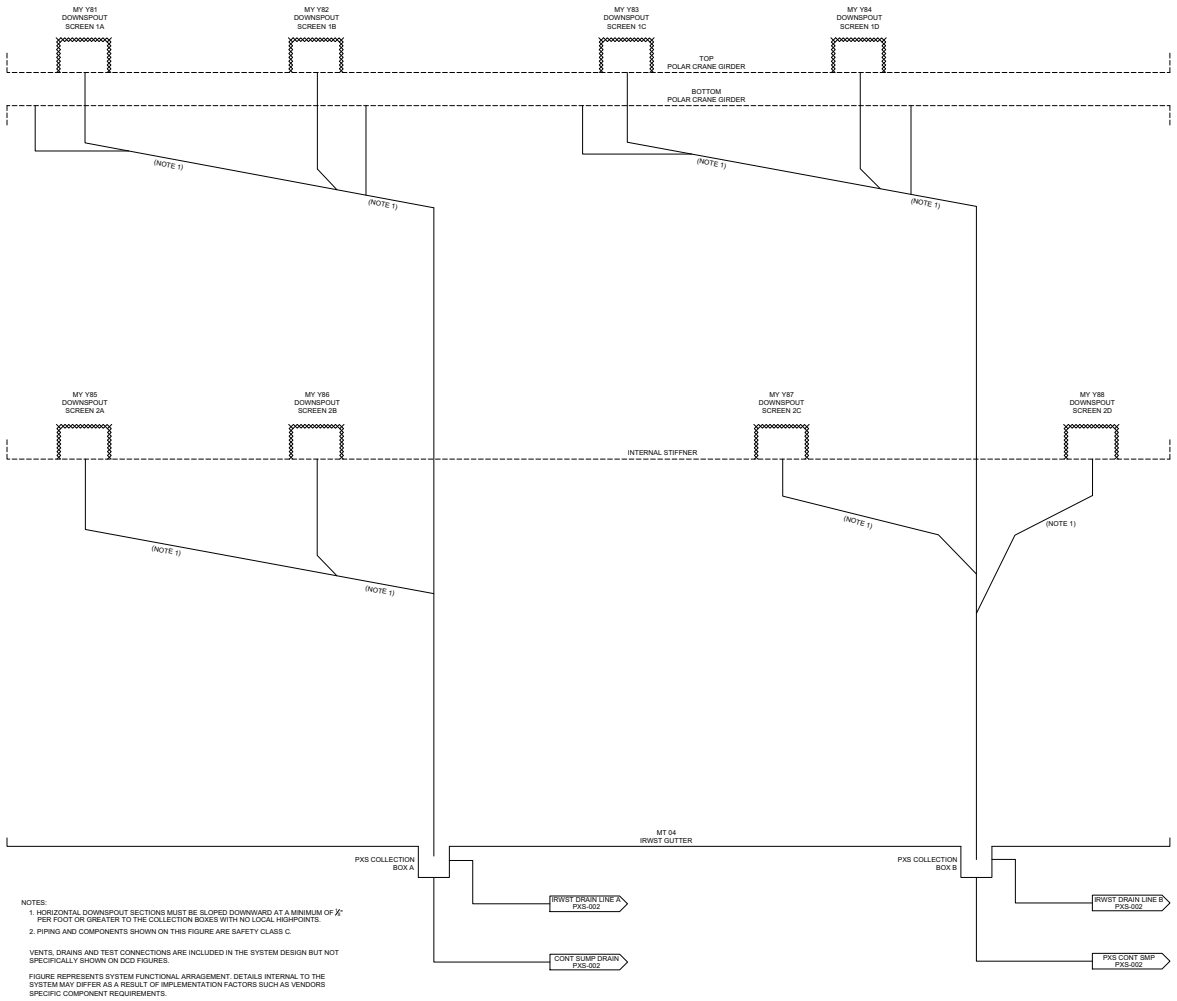

Figure 6.3-1 (Sheet 3 of 3)
Simplified Passive Core Cooling System
Piping and Instrumentation Diagram
(REF) PXS 003

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Figure 6.3-2 Not Used

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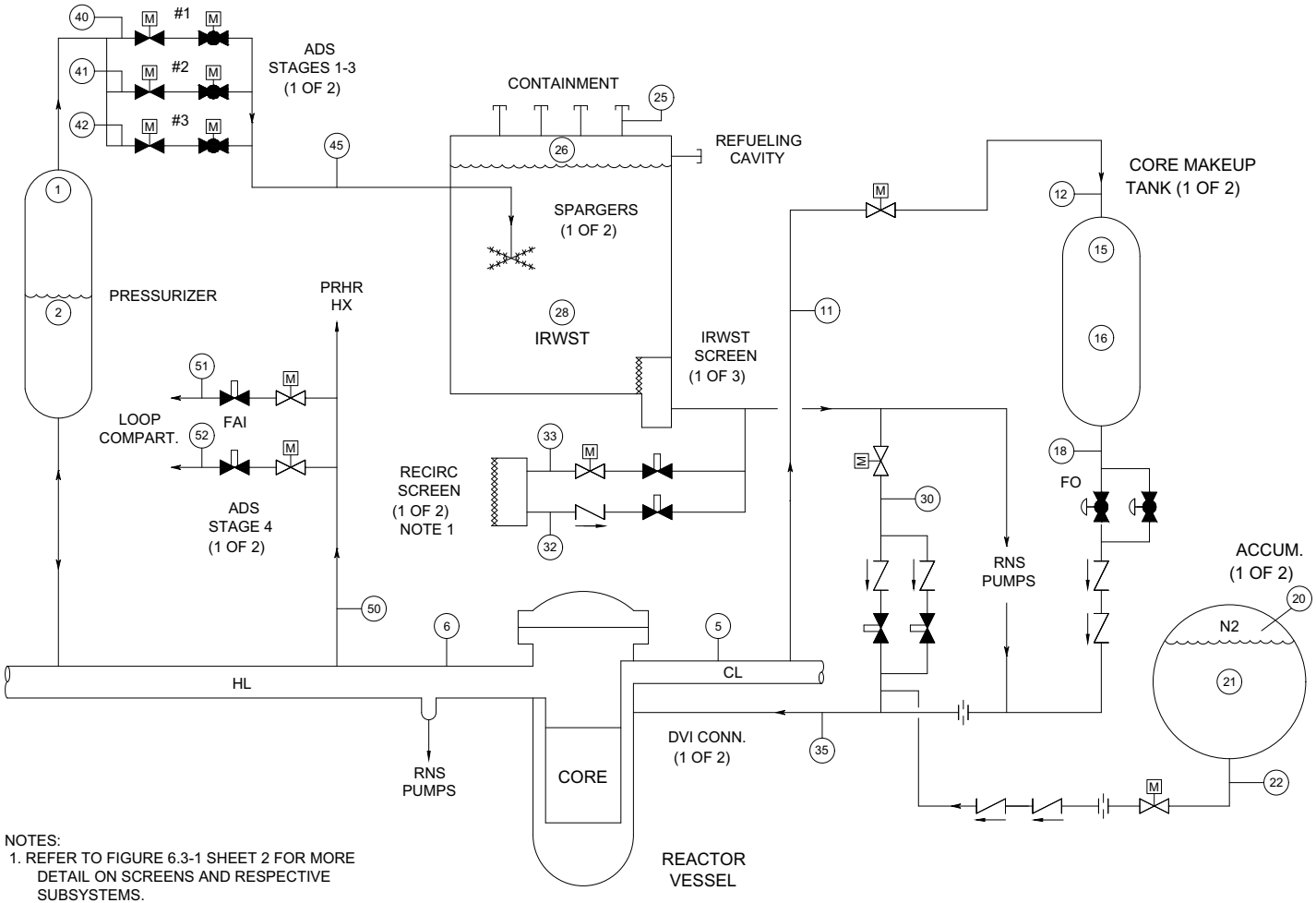

Inside Reactor Containment
Figure 6.3-3
Passive Safety Injection
(REF) RCS & PXS

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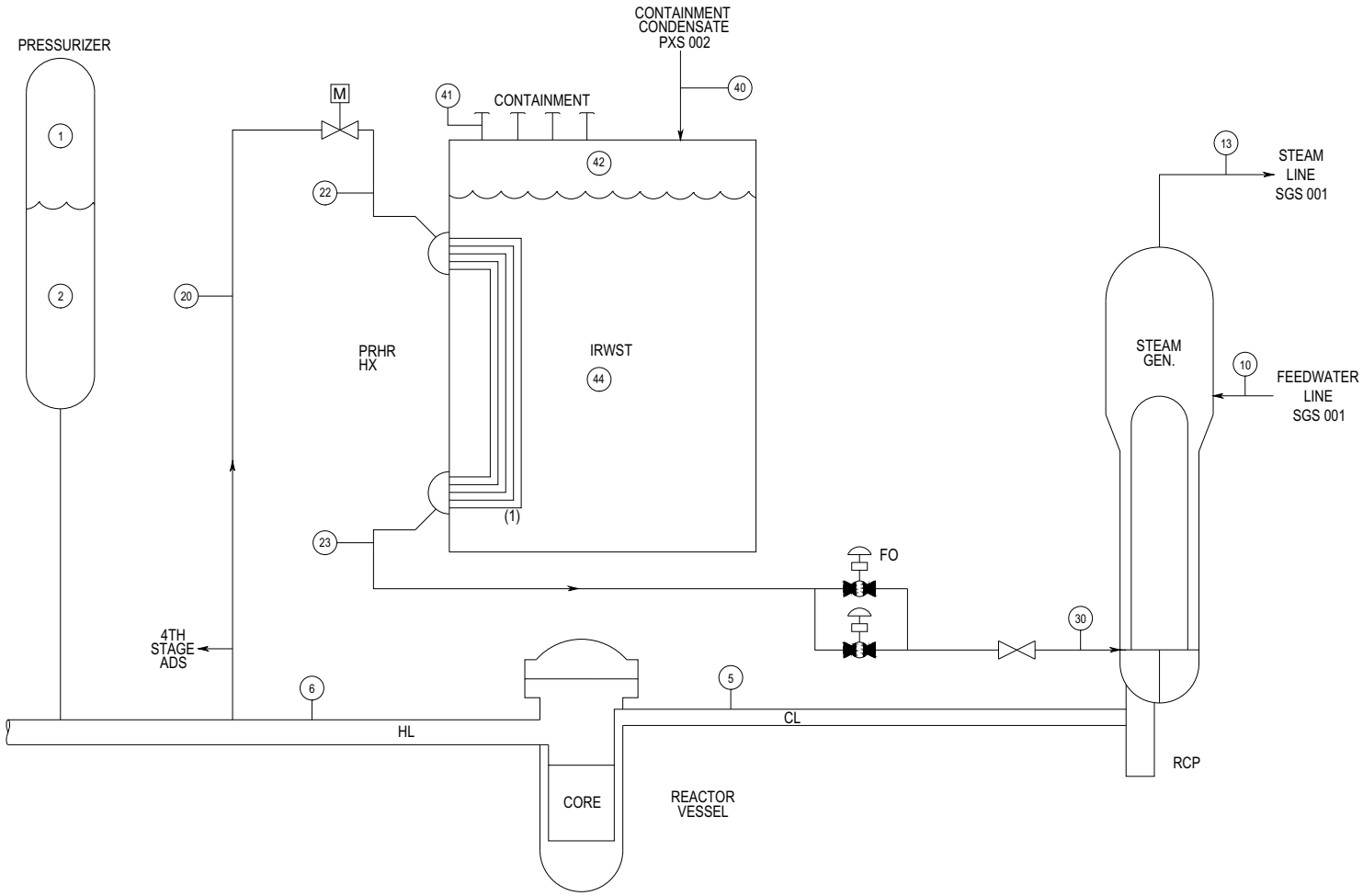

Inside Reactor Containment
Figure 6.3-4
Passive Decay Heat Removal
(REF) RCS & PXS

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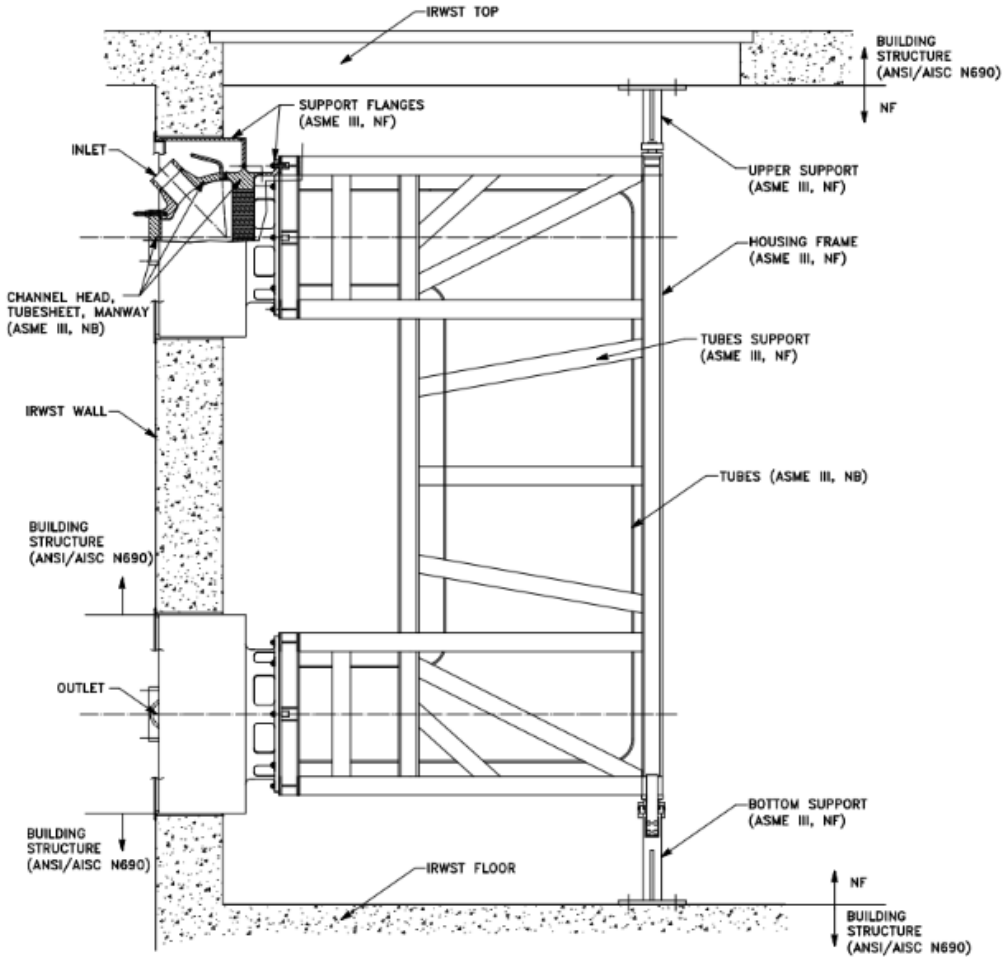

Figure 6.
Passive Heat Removal Heat Exchang

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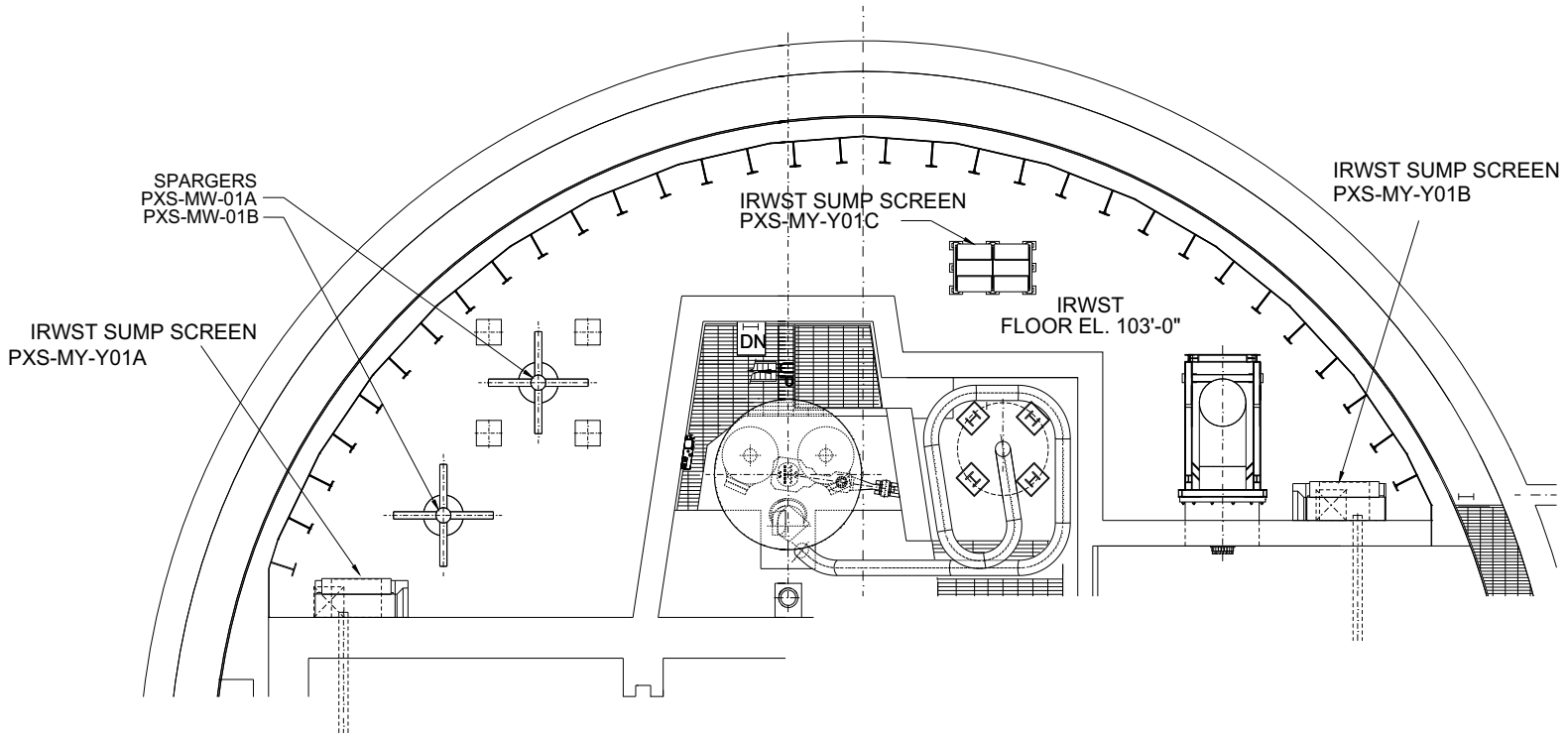

Figure 6.3-6
IRWST Screen Plan Location

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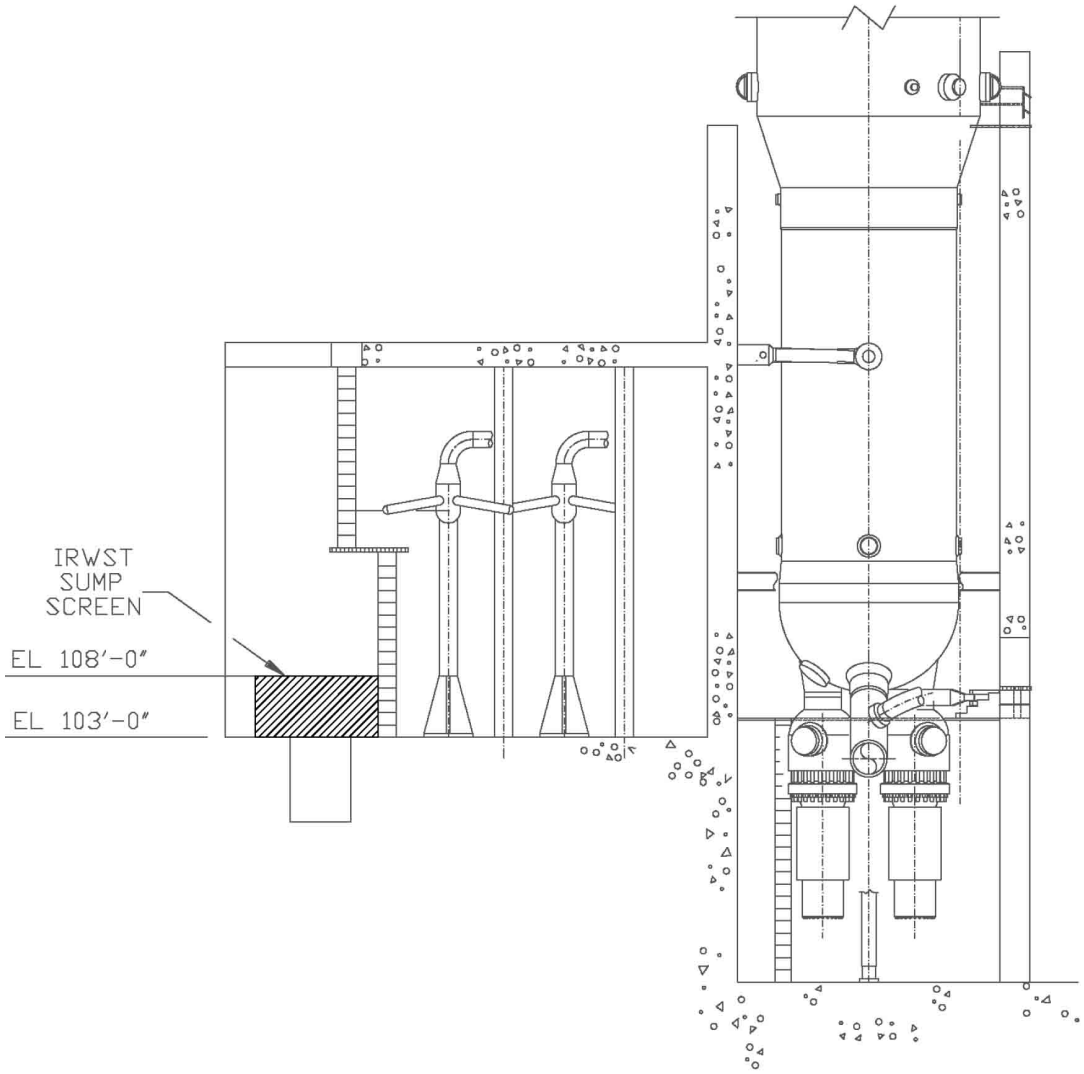

Figure 6.3-7
IRWST Screen Section Location

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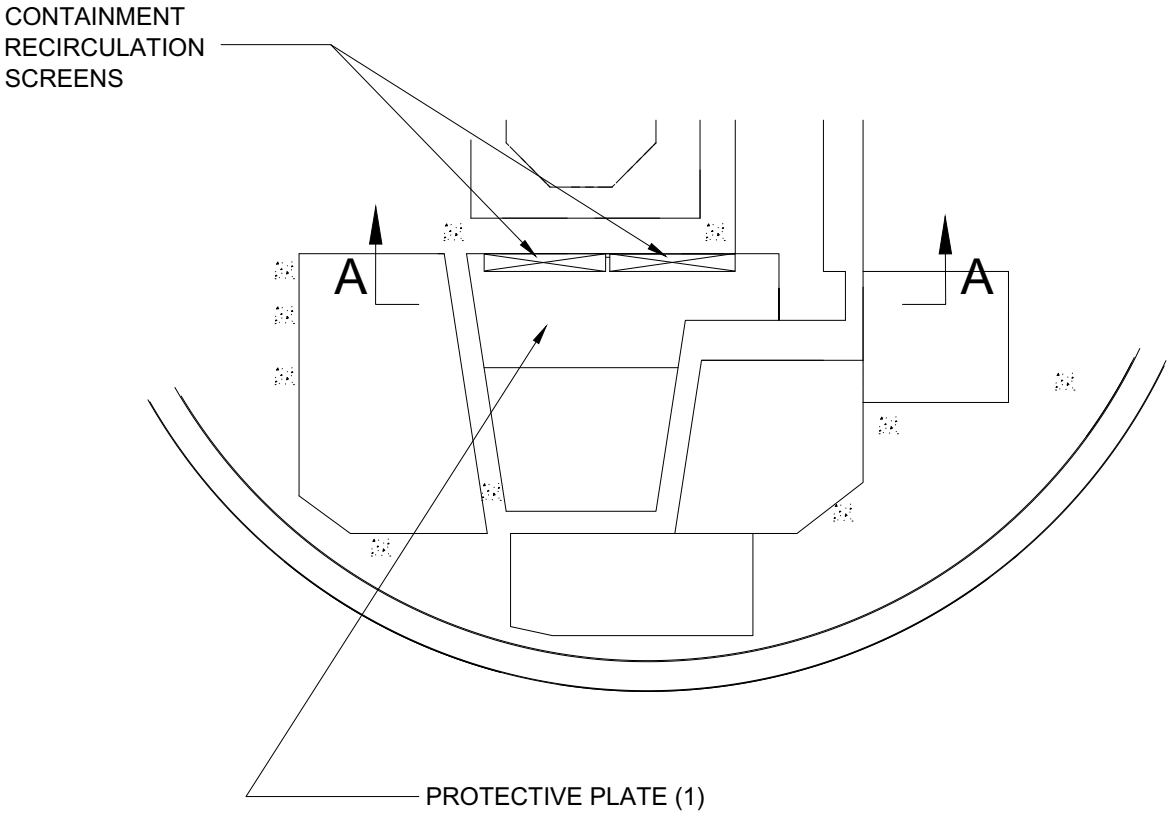

NOTE 1 - MINIMUM PLATE SIZE AND ELEVATION LIMITS ARE DEFINED IN SUBSECTION 6.3.2.2.7.1.

Figure 6.3-8
Containment Recirculation Screen Location Plan

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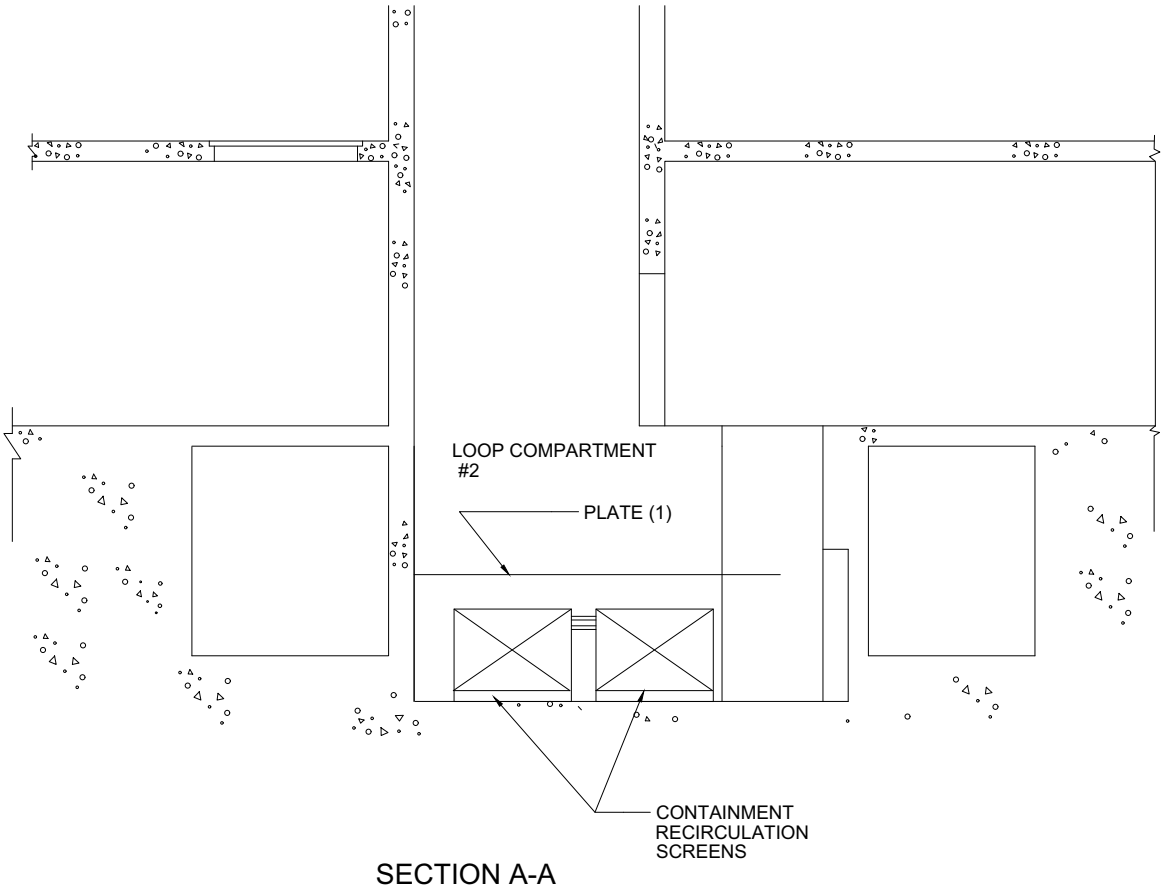

SECTION A-A

NOTE 1 - MINIMUM PLATE SIZE AND ELEVATION LIMITS
ARE DEFINED IN SUBSECTION 6.3.2.2.7.1.

Figure 6.3-9
Containment Recirculation Screen Location Elevation

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6.4 Habitability Systems

The habitability systems are a set of individual systems that collectively provide the habitability functions for the plant. The systems that make up the habitability systems are the:

- Nuclear island nonradioactive ventilation system (VBS)
- Main control room emergency habitability system (VES)
- Radiation monitoring system (RMS)
- Plant lighting system (ELS)
- Fire Protection System (FPS)

When a source of ac power is available, the nuclear island nonradioactive ventilation system (VBS) provides normal and abnormal HVAC service to the main control room (MCR), control support area (CSA), instrumentation and control rooms, dc equipment rooms, battery rooms, and the nuclear island nonradioactive ventilation system equipment room as described in Subsection 9.4.1.

If ac power is unavailable for more than 10 minutes or if main control room differential pressure is below the “Low” setpoint for more than 10 minutes or if “High-2” particulate or iodine radioactivity is detected in the main control room supply air duct, which may lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The main control room emergency habitability system is capable of providing emergency ventilation and pressurization for the main control room. The main control room emergency habitability system also provides emergency passive heat sinks for the main control room, instrumentation and control rooms, and dc equipment rooms.

Radiation monitoring of the main control room environment is provided by the radiation monitoring system. Smoke detection is provided in the VBS system. Emergency lighting is provided by the plant lighting system. Storage capacity is provided in the main control room for personnel support equipment. Manual hose stations outside the MCR and portable fire extinguishers are provided to fight MCR fires.

6.4.1 Safety Design Basis

The safety design bases discussed here apply only to the portion of the individual system providing the specified function. The range of applicability is discussed in Subsection 6.4.4.

6.4.1.1 Main Control Room Design Basis

The habitability systems provide coverage for the main control room pressure boundary as defined in Subsection 6.4.2.1. The following discussion summarizes the safety design bases with respect to the main control room:

- The habitability systems are capable of maintaining the main control room environment suitable for prolonged occupancy throughout the duration of the postulated accidents discussed in Chapter 15 that require protection from the release of radioactivity. Refer to Section 3.1 and Subsections 6.4.4 and 15.6.5.3 for a discussion on conformance with General Design Criterion 19 and to Section 1.9 for a discussion on conformance with Generic Issue B-66.

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- The main control room is designed to withstand the effects of an SSE and a design-basis tornado.
- A maximum main control room occupancy of up to 11 persons can be accommodated.
- The radiation exposure of main control room personnel throughout the duration of the postulated limiting faults discussed in Chapter 15 does not exceed the limits set by General Design Criterion 19.
- The emergency habitability system maintains CO_2 concentration to less than 0.5 percent for up to 11 main control room occupants.
- The habitability systems provide the capability to detect and protect main control room personnel from external fire, smoke, and airborne radioactivity.
- Automatic actuation of the individual systems that perform a habitability systems function is provided. Smoke detectors, radiation detectors, and associated control equipment are installed at various plant locations as necessary to provide the appropriate operation of the systems.
- The habitability system provides the capability to provide passive air filtration for the main control room during VES operation. The filtration portion of the systems meets the intent of Regulatory Guide 1.52 (Reference 10).

6.4.1.2 Instrumentation and Control Room/DC Equipment Rooms Design Basis

The habitability systems are also designed to service the instrumentation and control rooms and dc equipment rooms. The habitability systems are capable of maintaining the temperature in the instrumentation and control rooms and dc equipment rooms below the equipment qualification temperature limit throughout the duration of the postulated accidents discussed in Chapter 15, an SSE, or design-basis tornado.

6.4.2 System Description

Only the main control room emergency habitability system is discussed in this subsection. The remaining systems are described only as necessary to define their functions in meeting the safety-related design bases of the habitability systems. Descriptions of the nuclear island nonradioactive ventilation system, fire protection system, plant lighting system, and radiation monitoring system are found in Subsections 9.4.1, 9.5.1, 9.5.3, and Section 11.5, respectively.

6.4.2.1 Definition of the Main Control Room Pressure Boundary

The main control room pressure boundary is located on elevation 117′-6″ in the auxiliary building, on the nuclear island. As shown in Figure 6.4-1, the pressure boundary encompasses the main control area, operations work area, operations break room, shift supervisor's office, kitchen, and toilet facilities. The pressure boundary is represented by the line around the periphery of the boundary in the figure. The stairwell leading down to elevation 100′ and the area within the vestibule are specifically excluded from the boundary.

The areas, equipment, and materials to which the main control room operator requires access during a postulated accident are shown in Figure 6.4-1. This figure is a subset of Figure 1.2-8. Areas adjacent to the main control room are shown in Figures 1.2-19 and 1.2-24. The layout, size, and ergonomics of the operator workstations and wall panel information system depicted in Figure 6.4-1 do not reflect the results of the design process described in Chapter 18. The actual size, shape,

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ergonomics, and layout of the operator workstations and wall panel information system is an output of the design process in Chapter 18.

6.4.2.2 General Description

The main control room emergency habitability system air storage tanks are sized to deliver the required air flow to the main control room and induce sufficient air flow through the passive filtration line to meet the ventilation and pressurization requirements for 72 hours based on the performance requirements of Subsection 6.4.1.1. Normal system makeup is provided by a connection to the breathable quality air compressor in the compressed and instrument air system (CAS). See Subsection 9.3.1 for a description of the CAS. A connection for refilling operation is provided in the CAS.

Flow from the air storage tanks induces a filtration flow of at least 600 cfm. Testing was conducted to validate that the passive filtration line is capable of inducing a filtration flow of at least 600 cfm greater than the design flow rate from the VES emergency air storage tanks. The testing is documented in TR-SEE-III-09-03 (Reference 12). The filtration flow passes through a series of silencers to maintain acceptable main control room noise levels. The passive filtration portion of the system includes a HEPA filter, a charcoal adsorber, and a downstream postfilter. The filters are configured to satisfy the guidelines of Regulatory Guide 1.52 (Reference 10). The air intake to the passive filtration ductwork is located near the operations work area. The ductwork is routed behind the main control area through the operations break room to reduce the overall noise level in the main control area. The filtered air supply is then distributed to three supply locations that are sufficiently separated from the air intake to avoid short circuiting of the air flow. Two of the supply locations are located inside the main control area. Flow dampers ensure the filtered air is properly distributed throughout the main control room envelope.

The function of providing passive heat sinks for the main control room, instrumentation and control rooms, and dc equipment rooms is part of the main control room emergency habitability system. The heat sinks for each room are designed to limit the temperature rise inside each room during the 72-hour period following a loss of nuclear island nonradioactive ventilation system operation. The heat sinks consist primarily of the thermal mass of the concrete that makes up the ceilings and walls of these rooms.

To enhance the heat-absorbing capability of the ceilings, a metal form is attached to the interior surface of the concrete at selected locations. Metallic plates are attached perpendicular to the form. These plates extend into the room and act as thermal fins to enhance the heat transfer from the room air to the concrete. The specifics of the fin construction for the main control room and I&C room ceilings are described in Subsection 3.8.4.1.2.

The normal operating temperatures in the main control room, instrumentation and control rooms, dc equipment rooms, and adjacent rooms are kept within a specified range by the nuclear island nonradioactive ventilation system in order to maintain a design basis initial heat sink capacity of each room. See Subsection 9.4.1 for a description of the nuclear island nonradioactive ventilation system.

In the unlikely event that power to the nuclear island nonradioactive ventilation system is unavailable for more than 72 hours, MCR habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCR such that the maximum average Wet Bulb Globe Temperature (WBGT) Index for the MCR is less than 90°F. See Subsection 9.4.1 for a description of this cooling mode of operation. Doors and ducts may be opened to provide a supply pathway and an exhaust pathway. Likewise, outside air is supplied to division B and C instrumentation and control rooms in order to maintain the ambient temperature below the qualification temperature of the equipment.

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The main control room emergency habitability system piping and instrumentation diagram is shown in Figure 6.4-2.

6.4.2.3 Component Description

The main control room emergency habitability system compressed air supply contains a set of storage tanks connected to a main and an alternate air delivery line and equipment to provide electrical load de-energization. Components common to both lines include a manual isolation valve and a pressure regulating valve. Single active failure protection is provided by the use of redundant, remotely operated isolation valves, which are located within the MCR pressure boundary. In the event of insufficient or excessive flow in the main delivery line, the main delivery line is isolated and the alternate delivery line is manually actuated. The alternate delivery line contains the same components as the main delivery line with the exception of the remotely operated isolation valves, and thus is capable of supplying compressed air to the MCR pressure boundary at the required air flowrate. The VES piping and penetrations for the MCR envelope are designated as equipment Class C. Additional details on Class C designation are provided in Subsection 3.2.2.5. The classification of VES components is provided in Table 3.2-3, as appropriate.

- Emergency Air Storage Tanks

  There are a total of 32 air storage tanks. The air storage tanks are constructed of forged, seamless pipe, with no welds, and conform to Section VIII and Appendix 22 of the ASME Code. The design pressure of the air storage tanks is 4000 psi. The storage tanks collectively contain a minimum storage capacity of 327,574 scf.

- MCR Load Shed Panels

  The de-energization of the MCR electrical loads is performed using Class 1E equipment. Equipment within each of the two electrical panels is actuated from the “main control room isolation, air supply initiation, and electrical load de-energization” engineered safety feature. The de-energization is separated into two stages to provide operators with the maximum available nonsafety-related equipment while maintaining the MCR heat load within the requirements of the VES.

  Each electrical panel has redundant relays and timers controlled by both protection and safety monitoring system (PMS) Division A and PMS Division C. Either division is capable of actuating the timers and relays associated with each electrical panel independent of one another. This configuration prevents routine maintenance or single failures of a PMS cabinet from creating a spurious loss of MCR electrical loads while still providing for single failure protection. To accomplish the De-energize MCR Electrical Loads function, one set of Stage 1 and Stage 2 timers in each electrical panel must receive the PMS command.

  Relays in both electrical panels must be actuated to carry out the overall function. However, overall actuation may occur via different combinations of Division A and Division C commands.

- Pressure Regulating Valve

  Each compressed air supply line contains a pressure regulating valve located downstream of the common header. The pressure at the outlet of the valve is controlled via a two-staged self-contained pressure control operator. A failure of either stage of the pressure regulating valve will not cause the valve to fail completely open. A failure of the second stage of the pressure regulating valve will increase flow from the emergency air storage tanks. There is adequate margin in the emergency air storage tanks such that an operator has time to isolate the line and manually actuate the alternate delivery line.

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- Flow Metering Orifice

  The flow rate of air delivered to the main control room pressure boundary is limited by an orifice located downstream of the pressure regulating valve in the eductor and in the eductor bypass line. The orifice is sized to provide the required air flow rate to the main control room pressure boundary.

- Air Delivery Main Isolation Valve

  The pressure boundary of the compressed air storage tanks is maintained by normally closed remotely operated isolation valves in the main supply line. These valves are located within MCR pressure boundary downstream of the pressure regulating valve and automatically initiate air flow upon receipt of a signal to open (see Subsection 6.4.3.2).

- Pressure Relief Isolation Valve

  To limit the pressure increase within the main control room, isolation valves are provided, one in each of redundant flowpaths, which open on a time delay after receipt of an emergency habitability system actuation signal. The valves provide a leak tight seal to protect the integrity of the main control room pressure boundary during normal operation, and are normally closed to prevent interference with the operation of the nonradioactive ventilation system.

- Main Air Flowpath Isolation Valve

  The main air flowpath contains a normally open, manually operated valve located within the MCR pressure boundary, downstream of the remotely operated air delivery main isolation valves. The valve is provided as a means of isolating and preserving the air storage tank's contents in the event of a pressure regulating valve malfunction.

- Air Delivery Alternate Isolation Valve

  The alternate air delivery flowpath contains a normally closed, manually operated valve, located within the MCR pressure boundary. The valve is provided as a means of manually activating the alternate air delivery flowpath in the event the main air delivery flowpath is inoperable.

- Pressure Relief Damper

  Pressure relief dampers are located downstream of the butterfly isolation valves, and are set to open on a differential pressure of at least 1/8-inch water gauge with respect to the surrounding areas. The differential pressure between the control room and the surrounding area location is monitored to ensure that a positive pressure is maintained in the control room with respect to its surroundings.

  The pressure relief dampers discharge through the MCR vestibule in order to reduce the amount of radioactivity that can be transported into the MCR when operators enter. Two vestibule discharge openings provide a purge flow path from the vestibule to the electrical penetration room division A (room 12412).

- Eductor

  An eductor is connected to the discharge of the VES makeup line from the emergency air storage tanks and to ductwork located inside the main control room envelope that comprises the passive filtration portion of the VES. The eductor works by directing compressed air from the VES storage tanks through a specially designed nozzle to create a powerful vacuum that draws air from the

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main control room through the surrounding ductwork into the passive air filtration line. The eductor is designed to create a vacuum capable of drawing at least 600 scfm of flow into the passive air filtration system. This flow rate is based on a VES makeup flow of 65 ± 5 scfm at an approximate pressure of 110 psig at the discharge of the bottled air supply to the eductor. The eductor has no electrical power requirements, contains no moving parts, and requires no maintenance such as adjusting setpoints or lubricating bearings.

- High-Efficiency Particulate (HEPA) Filter, Charcoal Adsorber, and Postfilter

  The main control room passive filtration flowpath contains a HEPA filter in series with a charcoal adsorber and a postfilter. They work to remove particulate and iodine from the air to reduce potential control room dose during VES operation.

  HEPA filters are constructed, qualified, and tested in accordance with UL-586 (Reference 9) and ASME AG-1 (Reference 7), Section FC. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol in accordance with ASME AG-1 (Reference 7), Section TA.

  The charcoal adsorber is designed, constructed, qualified, and tested in accordance with ASME AG-1 (Reference 7), Section FD; and Regulatory Guide 1.52. Each charcoal adsorber is an assembly with 2-inch deep Type II adsorber cells, conforming to IE Bulletin 80-03 (Reference 8).

  Postfilters downstream of the charcoal filters have a minimum DOP efficiency of 95 percent. The filters are UL listed in accordance with UL 900 (Reference 11).

- Silencers

  Two silencers are located in the passive air filtration line. One silencer is located downstream of the eductor, and the other silencer is located upstream of the eductor. The silencers are designed to reduce the noise created by the passive air filtration line.

- Control Room Access Doors

  Two sets of doors, with a vestibule between, are provided at the access to the main control room.

- Breathing Apparatus

  Self-contained portable breathing equipment with air bottles is stored inside the main control room pressure boundary. The amount of stored air is sufficient to provide a 6-hour supply of breathable air for up to 11 main control room occupants. This is backup protection to the permanently installed habitability systems.

6.4.2.4 Leaktightness

The main control room pressure boundary is designed for low leakage. It consists of cast-in-place reinforced concrete walls and slabs, and is constructed to minimize leakage through construction joints and penetrations. The following features are applied as needed in order to achieve this objective:

- The outside surface of penetrations sleeves in contact with concrete are sealed with a crack sealer. The piping and electrical cable penetrations are sealed with qualified pressure-resistant material compatible with penetration materials and/or cable jacketing.

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- The interior or exterior surfaces of the main control room envelope (walls, floor, and ceiling) are coated with low permeability paint/epoxy sealant.

- Inside surfaces of penetrations and sleeves in contact with commodities (i.e., pipes and conduits, etc.) are sealed. Main control room pressure boundary HVAC isolation valves are qualified to shut tight against control room pressure.

- Penetration sealing materials are designed to withstand at least 1/4-inch water gauge pressure differential in an air pressure barrier. Penetration sealing material is a silicone-based material or equivalent.

- There is no HVAC duct that penetrates the main control room pressure boundary. Ventilation flow paths that pass through the main control room pressure boundary consist of safety-related cast sleeves and attached piping, including redundant safety-related seismic Category I isolation valves that are physically located within the main control room envelope.

The piping, conduits, and electrical cable trays penetrating through any combination of main control room pressure boundary are sealed with seal assembly compatible with the materials of penetration commodities. Penetration sealing materials are selected to meet barrier design requirements and are designed to withstand specific area environmental design requirements and remain functional and undamaged during and following an SSE. There are no adverse environmental effects on the MCR sealant materials resulting from postulated spent fuel pool boiling events.

The main control room pressure boundary main entrance is designed with a double-door vestibule, which is purged by the pressure relief damper discharge flow during main control room emergency habitability system operation. The emergency exit door (stairs to elevation 100′) is normally closed, and remains closed under design basis source term conditions. Administrative controls prohibit the emergency exit door to the remote shutdown workstation from being used for normal ingress and egress during VES operation.

When the main control room pressure boundary is isolated in an accident situation, there is no direct communication with the outside atmosphere, nor is there communication with the normal ventilation system. Leakage from the main control room pressure boundary is the result of an internal pressure of at least 1/8-inch water gauge provided by emergency habitability system operation.

The exfiltration and infiltration analysis for nuclear island nonradioactive ventilation system operation is discussed in Subsection 9.4.1.

6.4.2.5 Interaction with Other Zones and Pressurized Equipment

The main control room emergency habitability system is a self-contained system. There is no interaction between other zones and pressurized equipment.

For a discussion of the nuclear island nonradioactive ventilation system, refer to Subsection 9.4.1.

6.4.2.6 Shielding Design

The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the main control room. Main control room shielding design bases are discussed in Section 12.3. In addition to shielding provided by building structural features, consideration is given to shielding provided by the VES filter shielding. Descriptions of the design basis LOCA source terms, main control room shielding parameters, and evaluation of doses to main control room personnel are presented in Section 15.6.

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The main control room and its location in the plant are shown in Figure 12.3-1.

6.4.3 System Operation

This subsection discusses the operation of the main control room emergency habitability system.

Generic Issue 83 addresses the importance of maintaining control room habitability following an accidental release of external toxic or radioactive material or smoke and the capability of the control room operators to safely control the reactor. Procedures and training for control room habitability are written in accordance with Section 13.5 for control room operating procedures, and Section 13.2 for operator training. The procedures and training are verified to be consistent to the intent of Generic Issue 83.

The procedures and training address the toxic chemical events addressed in Sections 2.2 and 6.4 consistent with the guidance provided in regulatory position C.5 of Regulatory Guide 1.78, including arrangements with Federal, State, and local agencies or other cognizant organizations for the prompt notification of the nuclear power plant when accidents involving hazardous chemicals occur within five miles of the plant. The procedures include the conduct of periodic surveys of stationary and mobile sources of hazardous chemicals affecting the evaluations consistent with the guidance provided in regulatory position 2.5 of Regulatory Guide 1.196. The procedures include appropriate reviews of the configuration of the control room envelope and habitability systems consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196. The procedures also include periodic assessments of the control room habitability systems' material condition, configuration controls, safety analyses, and operating and maintenance procedures consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196.

Procedures for testing and maintenance are consistent with the design requirements of the DCD including the guidance provided in regulatory position 2.7.1 of Regulatory Guide 1.196.

6.4.3.1 Normal Mode

The main control room emergency habitability system is not required to operate during normal conditions. The nuclear island nonradioactive ventilation system maintains the air temperature of a number of rooms within a predetermined temperature range. The rooms with this requirement include the rooms with a main control room emergency habitability system passive heat sink design and their adjacent rooms.

6.4.3.2 Emergency Mode

Operation of the main control room emergency habitability system is automatically initiated by either of the following conditions:

- "High-2" particulate or iodine radioactivity in the main control room supply air duct
- Loss of ac power for more than 10 minutes
- "Low" main control room differential pressure for more than 10 minutes

Operation can also be initiated by manual actuation.

The nuclear island nonradioactive ventilation system is isolated from the main control room pressure boundary by automatic closure of the isolation devices located in the nuclear island nonradioactive ventilation system ductwork if radiation levels in the main control room supply air duct exceed the "High-2" setpoint or if ac power is lost for more than 10 minutes or if main control room differential

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pressure is below the “Low” setpoint for more than 10 minutes. At the same time, the main control room emergency habitability system begins to deliver air from the emergency air storage tanks to the main control room by automatically opening the isolation valves located in the supply line. The relief damper isolation valves also open allowing the pressure relief dampers to function and discharge the damper flow to purge the vestibule.

After the main control room emergency habitability system isolation valves are opened, the air supply pressure is regulated by a self-contained regulating valve. This valve maintains a constant downstream pressure regardless of the upstream pressure. Downstream of the pressure regulating valve, a constant air flow rate is maintained by the flow metering orifice in the eductor and in the eductor bypass line. This flow rate is sufficient to maintain the main control room pressure boundary at least 1/8-inch water gauge positive differential pressure with respect to the surroundings and induce a flow rate of at least 600 cfm into the passive air filtration line. The main control room emergency habitability system air flow rate is also sufficient to maintain the carbon dioxide levels below 0.5 percent concentration for 11 occupants and to maintain air quality within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1.

The emergency air storage tanks are sized to provide the required air flow to the main control room pressure boundary for 72 hours. After 72 hours, the main control room is cooled by drawing in outside air and circulating it through the room, as discussed in Subsection 6.4.2.2.

The temperature and humidity in the main control room pressure boundary following a loss of the nuclear island nonradioactive ventilation system remain within limits for reliable human performance (Reference 14) over a 72-hour period. The bounding initial values of temperature/relative humidity in the MCR are 75°F/60 percent, the relative humidity in the MCR varies between 5% and 95% with a corresponding dry bulb temperature variance between 75°F to under 95°F. The temperature/relative humidity values calculated during the 72 hours following a design basis accident equate to a maximum average WBGT Index for the MCR of less than 90°F. The 90°F WBGT Index is the design limit for minimizing performance decrements and potential harm, and preserving well-being and effectiveness of the MCR staff for an unlimited duration (Reference 14). Non-Class 1E MCR heat loads are de-energized by PMS automatic actions, and the 24-hour battery heat loads are terminated or exhausted at 24 hours to maintain the occupied zone of the MCR and the zones containing qualified safety-related equipment within the constraints of the heat loads in Table 6.4-3 (to maintain temperature below the WBGT limit) at 72 hours after VES actuation. The occupied zone is considered to be the area between the raised floor and 7 feet above the floor, which encompasses the reactor operator and the senior reactor operator consoles.

Sufficient thermal mass is provided in the walls and ceiling of the main control room to absorb the heat generated by the equipment, lights, and occupants. The temperature in the instrumentation and control rooms and dc equipment rooms following a loss of the nuclear island nonradioactive ventilation system remains below acceptable limits as discussed in Subsection 6.4.4. As in the main control room, sufficient thermal mass is provided surrounding these rooms to absorb the heat generated by the equipment. After 72 hours, the instrumentation and control rooms will be cooled by drawing in outside air and circulating it through the room, as discussed in Subsection 6.4.2.2.

In the event of a loss of ac power or “Low” main control room differential pressure for more than 10 minutes, the nuclear island nonradioactive ventilation system isolation valves automatically close and the main control room emergency habitability system isolation valves automatically open. These actions protect the main control room occupants from a potential radiation release. In instances in which there is no radiological source term present, the compressed air storage tanks are refilled via a connection to the breathable quality air compressor in the compressed and instrument air system (CAS). The compressed air storage tanks can also be refilled from portable supplies by an installed connection in the CAS.

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6.4.4 System Safety Evaluation

In the event of an accident involving the release of radioactivity to the environment, the nuclear island nonradioactive ventilation system (VBS) is expected to switch from the normal operating mode to the supplemental air filtration mode to protect the main control room personnel. Although the VBS is not a safety-related system, it is expected to be available to provide the necessary protection for realistic events. However, the design basis accident doses reported in Chapter 15 utilize conservative assumptions, and the main control room doses are calculated based on operation of the safety-related emergency habitability system (VES) since this is the system that is relied upon to limit the amount of activity the personnel are exposed to. The analyses assume that the VBS is initially in operation, but fails to enter the supplemental air filtration mode on a High-1 radioactivity indication in the main control room atmosphere. VES operation is then assumed to be initiated once the “High-2” level for control room atmosphere iodine or particulate radioactivity is reached.

Doses are also calculated assuming that the VBS does operate in the supplemental air filtration mode as designed, but with no switchover to VES operation. This VBS operating case demonstrates the defense-in-depth that is provided by the system and also shows that, in the event of an accident with realistic assumptions, the VBS is adequate to protect the control room operators without depending on VES operation.

Doses were determined for the following design basis:

	VES Operating	VBS Operating
Large Break LOCA	4.33 rem TEDE	4.84 rem TEDE
Fuel Handling Accident	1.5 rem TEDE	1.1 rem TEDE
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	3.5 rem TEDE	2.8 rem TEDE
(Accident-initiated iodine spike)	1.0 rem TEDE	0.8 rem TEDE
Steam Line Break		
(Pre-existing iodine spike)	1.1 rem TEDE	0.6 rem TEDE
(Accident-initiated iodine spike)	1.4 rem TEDE	1.6 rem TEDE
Rod Ejection Accident	3.6 rem TEDE	2.2 rem TEDE
Locked Rotor Accident		
(Accident without feedwater available)	0.4 rem TEDE	0.5 rem TEDE
(Accident with feedwater available)	0.2 rem TEDE	0.6 rem TEDE
Small Line Break Outside Containment	0.4 rem TEDE	0.2 rem TEDE

For all events the doses are within the dose acceptance limit of 5.0 rem TEDE. The details of analysis assumptions for modeling the doses to the main control room personnel are delineated in the LOCA dose analysis discussion in Subsection 15.6.5.3 for VES operating cases. The analysis assumptions are provided in Subsection 9.4.1.2.3.1 for the VBS operating case.

No radioactive materials are stored or transported near the main control room pressure boundary (this does not apply to installed equipment, such as radiation monitors described in Section 11.5, nor to portable equipment that is in use, such as calibration equipment, survey instrumentation, tools, and other such transient items).

As discussed and evaluated in Subsection 9.5.1, the use of noncombustible construction and heat and flame resistant materials throughout the plant reduces the likelihood of fire and consequential

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impact on the main control room atmosphere. Operation of the nuclear island nonradioactive ventilation system in the event of a fire is discussed in Subsection 9.4.1.

The exhaust stacks of the onsite standby power diesel generators are located in excess of 150 feet away from the fresh air intakes of the main control room. The onsite standby power system fuel oil storage tanks are located in excess of 300 feet from the main control room fresh air intakes. These separation distances reduce the possibility that combustion fumes or smoke from an oil fire would be drawn into the main control room.

The protection of the operators in the main control room from offsite toxic gas releases is discussed in Section 2.2. The sources of onsite chemicals are described in Table 6.4-1, and their locations are shown on Figure 1.2-2. Analysis of these sources is in accordance with Regulatory Guide 1.78 (Reference 5) and the methodology in NUREG-0570, “Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release” (Reference 6), and the analysis shows that these sources do not represent a toxic or flammability hazard to control room personnel.

Table 6.4-201 provides additional details regarding the evaluated onsite chemicals.

A supply of protective clothing, respirators, and self-contained breathing apparatus adequate for 11 persons is stored within the main control room pressure boundary.

The main control room emergency habitability system components discussed in Subsection 6.4.2.3 are arranged as shown in Figure 6.4-2. The location of components and piping within the main control room pressure boundary provides the required supply of compressed air to the main control room pressure boundary, as shown in Figure 6.4-1.

During emergency operation, the main control room emergency habitability system passive heat sinks are designed to limit the temperature inside the main control room to remain within limits for reliable human performance (Reference 14) over 72 hours. The passive heat sinks limit the air temperature inside the instrumentation and control rooms to 120°F and dc equipment rooms to 120°F. The walls and ceilings that act as the passive heat sinks contain sufficient thermal mass to accommodate the heat sources from equipment, personnel, and lighting for 72 hours.

The main control room emergency habitability system nominally provides 65 scfm of ventilation air to the main control room from the compressed air storage tanks. Sixty scfm of supplied ventilation flow is sufficient to induce a filtration flow of at least 600 cfm into the passive air filtration line located inside the main control room envelope. This ventilation flow is also sufficient to pressurize the control room to at least positive 1/8-inch water gauge differential pressure with respect to the surrounding areas in addition to limiting the carbon dioxide concentration below one-half percent by volume for a maximum occupancy of 11 persons and maintaining air quality within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1.

Automatic transfer of habitability system functions from the main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system to the main control room emergency habitability system is initiated by any of the following conditions:

- “High-2” particulate or iodine radioactivity in MCR air supply duct
- Loss of ac power for more than 10 minutes
- “Low” main control room differential pressure for more than 10 minutes

The airborne fission product source term in the reactor containment following the postulated LOCA is assumed to leak from the containment and airborne fission products are assumed to result from

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spent fuel pool steaming. The concentration of radioactivity, which is assumed to surround the main control room, after the postulated accident, is evaluated as a function of the fission product decay constants, the containment leak rate, and the meteorological conditions assumed. The assessment of the amount of radioactivity within the main control room takes into consideration the radiological decay of fission products and the infiltration/exfiltration rates to and from the main control room pressure boundary.

A single active failure of a component of the main control room emergency habitability system or nuclear island nonradioactive ventilation system does not impair the capability of the systems to accomplish their intended functions. The Class 1E components of the main control room emergency habitability system are connected to independent Class 1E power supplies. Both the main control room emergency habitability system and the portions of the nuclear island nonradioactive ventilation system which isolates the main control room are designed to remain functional during an SSE or design-basis tornado.

In accordance with SECY-77-439 (Reference 13), a single passive failure of a component in the passive filtration line in the main control room emergency habitability system does not impair the capability of the system to accomplish its intended function. There is no source that could create line blockage in the VES line from the air bottles to the eductor. Thus potential blockage in the filtration line does not preclude breathable air from the emergency air storage tanks from being delivered to the main control room envelope for 72 hours during VES operation. Passive filtration using the main control room habitability system is not required to maintain operator dose rates below the acceptance limit of 5.0 rem TEDE 24 hours after the initiation of a design basis event. The dose rates for the following limiting cases were determined to demonstrate that passive filtration is not required 24 hours after the initiation of a design basis event. The following cases are evaluated since they involve releases that extend beyond 24 hours after the initiation of the event:

Large Break LOCA	4.4 rem TEDE
Steam Line Break	
(Pre-existing iodine spike)	1.2 rem TEDE
(Accident-initiated iodine spike)	2.0 rem TEDE

For all events, the doses are within the dose acceptance limit of 5.0 rem TEDE. The details of analysis assumptions for modeling the doses to the main control room personnel are the same as those delineated in the LOCA dose analysis discussion in Subsection 15.6.5.3 assuming a passive failure disables the passive filtration flow path after 24 hours. Potential blockage in the filtration line does not preclude breathable air from the emergency air storage tanks from being delivered to the main control room envelope for 72 hours during VES operation. An eductor bypass line with a flow control orifice provides the operators with the ability to ensure that the breathable air from the emergency air storage tanks is delivered to the MCR.

6.4.4.1 Dual Unit Analysis

Credible events that could put the control room operators at risk from a dose standpoint at a single AP1000 unit have been evaluated and addressed in the DCD. The dose to the control room operators at an adjacent AP1000 unit due to a radiological release from another unit is bounded by the dose to control room operators on the affected unit. While it is possible that a unit may be downwind in an unfavorable location, the dose at the downwind unit would be bounded by what has already been evaluated for a single unit AP1000. Simultaneous accidents at multiple units at a common site are not considered to be a credible event.

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6.4.5 Inservice Inspection/Inservice Testing

A program of preoperational and inservice testing requirements is implemented to confirm initial and continued system capability. The VES system is tested and inspected at appropriate intervals, as defined by the technical specifications. Emphasis is placed on tests and inspections of the safety-related portions of the habitability systems.

6.4.5.1 Preoperational Inspection and Testing

Preoperational testing of the main control room emergency habitability system is performed to verify that the air flow rate of 65 ± 5 scfm is sufficient to induce a flow rate of at least 600 cfm into the passive air filtration line and maintain pressurization of the main control room envelope of at least 1/8-inch water gauge with respect to the adjacent areas. The positive pressure within the main control room is confirmed via the differential pressure transmitters within the control room. The installed flow meters are utilized to verify the system flow rates. The preoperational testing also verifies that the VES pressure regulating valves are capable of maintaining the VES flow rate of 65 ± 5 scfm over the operating range of expected valve inlet pressures. The pressurization of the control room limits the ingress of radioactivity, and the recirculation through the passive air filtration line maintains operator dose limits below regulatory limits. Air quality within the MCR environment is confirmed to be within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1 by analyzing air samples taken during the pressurization test.

The storage capacity of the compressed air storage tanks is verified to be in excess of 327,574 scf of compressed air. This amount of compressed air will assure 72 hours of air supply to the main control room.

Temperatures within the MCR are verified by analysis and/or testing to remain within the limits for reliable human performance (Reference 14) for a 72-hour period following a bounding scenario with MCR isolation and nonsafety-related ac power available (see Table 6.4-3 for heat loads) and a station blackout (battery-backed loads only).

Preoperational testing of the main control room isolation valves in the nuclear island nonradioactive ventilation system is performed to verify the leaktightness of the valves.

Preoperational testing for main control room envelope habitability during VES operation will be conducted in accordance with ASTM E741 (Reference 4). Where possible, inleakage testing is performed in conjunction with the VES system level operability testing since the VES must be in operation to perform the inleakage testing.

Testing and inspection of the radiation monitors is discussed in Section 11.5. The other tests noted above are discussed in Chapter 14.

6.4.5.2 Inservice Testing

Inservice testing of the main control room emergency habitability system and nuclear island nonradioactive ventilation system is conducted in accordance with the surveillance requirements specified in the Technical Specifications.

ASTM E741 testing of the main control room pressure boundary is conducted in accordance with the frequency specified in the technical specifications.

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6.4.5.3 Air Quality Testing

Connections are provided for sampling the air supplied from the compressed and instrument air system and for periodic sampling of the air stored in the storage tanks. Air samples of the compressed air storage tanks are taken quarterly and analyzed for acceptable air quality within the guidelines of CGA G-7.1-1997, “Commodity Specification for Air,” Grade E (Reference 15), with a pressure dew point of 40°F or lower at 3,400 psig or greater.

6.4.5.4 Main Control Room Envelope Habitability

Testing for main control room envelope habitability during VES operation will be conducted in accordance with ASTM E741 (Reference 4).

The main control room envelope must undergo an analysis of inleakage into the control room envelope to determine the integrity of the control room envelope boundary during a design basis accident, hazardous chemical release, or smoke event. Baseline control room envelope habitability testing will be performed as discussed in Subsection 6.4.5.1, followed by a self-assessment at three (3) years after successful baseline testing, and a periodic test at six (6) years in conjunction with other ASME inservice testing requirements. The self-assessment of the ability to maintain main control room habitability includes a review of procedures, boundaries, design changes, maintenance activities, safety analyses, and other related determinations.

If periodic testing is successful, then the assessment/testing cycle continues with a self-assessment three (3) years later and periodic testing three (3) years after the self-assessment. If a periodic testing is unsuccessful, then a periodic test is required three (3) years after repair and successful re-testing, following the unsuccessful periodic testing, to ensure there is no accelerated degradation of the main control room boundary or discrepancies in control of the main control room habitability.

In addition to periodic tests, control room envelope testing will also be performed when changes are made to structures, systems, and components that could impact control room envelope integrity, including systems internal and external to the control room envelope. The tests must be commensurate with the types and degrees of modifications and repairs and the potential impact upon integrity. Additional control room envelope testing will also be performed if a new limiting condition or alignment arises for which no inleakage data is available. Test failure is considered to be inleakage in excess of the licensing basis value for the particular challenge to control room envelope integrity.

Where possible, inleakage testing is performed in conjunction with the VES system level operability testing since the VES must be in operation to perform the inleakage testing.

6.4.6 Instrumentation Requirements

The indications in the main control room used to monitor the main control room emergency habitability system and nuclear island nonradioactive ventilation system are listed in Table 6.4-2.

Instrumentation required for actuation of the main control room emergency habitability system and nuclear island nonradioactive ventilation system are discussed in Subsection 7.3.1.

Details of the radiation monitors used to provide the main control room indication of actuation of the nuclear island nonradioactive ventilation system supplemental filtration mode of operation and actuation of main control room emergency habitability system operation are given in Section 11.5.

A description of initiating circuits, logic, periodic testing requirements, and redundancy of instrumentation relating to the habitability systems is provided in Section 7.3.

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6.4.7 Combined License Information

Combined License applicants referencing the AP1000 certified design are responsible for the amount and location of possible sources of hazardous chemicals in or near the plant and for Seismic Category I Class 1E hazardous chemical monitoring, as required. Regulatory Guide 1.78 (Reference 5) addresses control room protection for hazardous chemicals and evaluation of offsite hazardous chemical releases (including the potential for hazardous chemical releases beyond 72 hours) in order to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19.

Procedures and training for control room envelope habitability consistency with the intent of Generic Issue 83 (see Section 1.9) are addressed in Subsection 6.4.3.

The testing frequency for the main control room envelope habitability is discussed in Subsection 6.4.5.4.

6.4.8 References

1. “Ventilation for Acceptable Indoor Air Quality,” ASHRAE Standard 62 - 1989.

2. Not used.

3. Not used.

4. “Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution,” ASTM E741, 2000.

5. “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” Regulatory Guide 1.78, Revision 1, December 2001.

6. NUREG-0570, “Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release,” June 1979.

7. “Code on Nuclear Air and Gas Treatment,” ASME/ANSI AG-1-1997.

8. “Loss of Charcoal Adsorber Cells,” IE Bulletin 80-03, 1980.

9. “High-Efficiency, Particular, Air-Filter Units,” UL 586, 9th Edition.

10. “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Regulatory Guide 1.52, Revision 3, 2001.

11. “Standard for Air Filter Units,” UL 900, 8th Edition.

12. “AP1000 VES Air Filtration System Test Report,” TR-SEE-III-09-03.

13. “Single Failure Criterion,” SECY-77-439.

14. NUREG-0700, “Human-System Interface Design Review Guidelines,” Revision 2, 2002.

15. “Commodity Specification for Air,” CGA G-7.1-1997.

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Table 6.4-1
Onsite Chemicals[b]

Material	State	Location
Hydrogen	Liquid/Gas	Gas storage/Yard at Turbine Building
Nitrogen	Liquid/Gas	Gas storage
CO_2	Liquid/Gas	Gas storage
Oxygen Scavenger	Liquid	SWS chemical treatment building/area
pH Addition	Liquid	Turbine building[a], SWS chemical treatment building/ area
Sulfuric Acid	Liquid	CWS area, SWS chemical treatment building/area
Sodium Hydroxide	Liquid	Turbine building
Dispersant	Liquid	Turbine building[a], CWS area[a], SWS chemical treatment building/area
Fuel Oil	Liquid	DG fuel oil storage tank/DG building/Annex building/FPS Pump Enclosure/Yard West of Turbine Building
Corrosion Inhibitor	Liquid	Turbine building[a], CWS area[a], SWS chemical treatment building/area
Scale Inhibitor	Liquid	Turbine building
Biocide/Disinfectant	Liquid	Turbine building[a], CWS area[a], SWS chemical treatment building/area
Nonoxidizing Biocide	Liquid	SWS chemical treatment building/area
Sodium Bisulfite	Liquid	Turbine building

Note:
a. Site-specific
b. Table 6.4-201 supplements Table 6.4-1.

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Table 6.4-2
Main Control Room Habitability Indications and Alarms

VES emergency air storage tank pressure (indication and low and low-low alarms)
VES MCR pressure boundary differential pressure (indication and high and low alarms)
VES air delivery line flowrate (indication and high and low alarms)
VES passive filtration flow rate (indication and high and low alarms)
VBS main control room supply air radiation level (“High-1” and “High-2” alarms)
VBS outside air intake smoke level (high alarm)
VBS isolation valve position
VBS MCR pressure boundary differential pressure

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Table 6.4-3
Loss of AC Power Heat Load Limits

Room Name	Room Numbers	Heat Load 0 to 24 Hours (Btu/sec)	Heat Load 24 to 72 Hours (Btu/sec)
MCR Envelope	12401	23.5 (Hour 0 to 0.5) 14.5 (Hour 0.5 to 3.5) 4.75 (Hour 3.5 through 24)	3.95
I&C Rooms	12301	3.50	1.10
	12302	3.20	2.50
	12304	4.30	3.20
	12305	4.20	1.50
dc Equipment Rooms	12201	2.00	2.00
	12203	3.30	3.30
	12205	2.40	2.40
	12207	3.45	3.45

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Table 6.4-201 (Sheet 1 of 2)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals(1)

A — Standard Onsite Toxic Chemicals

Evaluated Material(2)	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR Intake	Evaluated Location	MCR Habitability Impact Evaluation
Hydrogen	Gas	2200 scf	126.3 ft	Yard at turbine building	MCR
Hydrogen	Liquid(3)	1500 gal	577 ft	Gas storage	MCR
Nitrogen	Liquid/Gas(3)	3000 gal	577 ft	Gas storage	MCR
Carbon Dioxide (CO_2)	Liquid/Gas(3)	6 tons	577 ft	Gas storage	MCR
Oxygen Scavenger [Hydrazine, or Carbohydrazide]	Liquid	1600 gal	203 ft	SWS chemical treatment building	IH
pH Addition [Ammonium Hydroxide, Morpholine, Ethanolamine or Methoxypropylamine]	Liquid	1600 gal	203 ft	SWS chemical treatment building	IH
Sulfuric Acid	Liquid	800 gal	203 ft	SWS chemical treatment building	IH
Sulfuric Acid	Liquid	20,000 gal	436 ft	CWS area	IH
Sodium Hydroxide	Liquid	800 gal	203 ft	Turbine building	S
Sodium Bisulfite	Liquid	450 gal	203 ft	Turbine Building	S
Fuel Oil	Liquid	94,000 gal	197 ft	DG fuel oil storage tank, DG building, Annex building	IH

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Table 6.4-201 (Sheet 2 of 2)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals(1)

A — Standard Onsite Toxic Chemicals

Evaluated Material(2)	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR Intake	Evaluated Location	MCR Habitability Impact Evaluation
Corrosion Inhibitor/Silt Dispersant [Nalco Trasar 3DT187 or Nalco Trasar 3DT487]	Liquid	800 gal	203 ft	SWS chemical treatment building	S
Scale Inhibitor [Polyacrylic Acid]	Liquid	800 gal	203 ft	Turbine building	S
Biocide/Disinfectant [Sodium Hypochlorite or Nalco Stabrex™ ST70]	Liquid	800 gal	203 ft	SWS chemical treatment building	S
Biocide/Disinfectant [Sodium hypochlorite]	Liquid	10,000 gal	436 ft	CWS area	S
Nonoxidizing Biocide [Nalco 77352NA or Ammonium Chloride]	Liquid	800 gal	203 ft	SWS chemical treatment building	S

1) This table supplements Table 6.4-1. Quantities are by largest evaluated container content for the evaluated location per unit. Quantities and distances are bounding evaluation values and may not be actual amounts and distances. Smaller quantities of a chemical at further distances from the MCR air intake are not shown on this table. Actual site locations are confirmed to be at or beyond the evaluated distance.

S - Chemicals with an Impact Evaluation designation of “S” for the MCR Habitability Impact Evaluation were evaluated and screened out based on the chemical properties, distance, and quantities.

IH - Chemicals with an Impact Evaluation designation of “IH” indicates the evaluation of this chemical considered the design detail of the main control room intake height.

MCR - Chemicals with an Impact Evaluation designation of “MCR” indicates the evaluation of this chemical considered design details of the main control room such as volume, envelope boundaries, ventilation systems, and occupancy factor.

2. Evaluated chemicals listed in [brackets] are typical of the class of compounds evaluated. The exact chemicals vary based on site needs and technologyavailable. The bracketed chemicals are substituted for similar chemical performing a similar function only after review by site personnel to ensure the new chemical does not impact the previous MCR Habitability Impact Evaluation.

3. Evaluated state is analyzed as a mass. Equivalent mass of gas is bounded by liquid.

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Figure 6.4-1
Main Control Room Envelope

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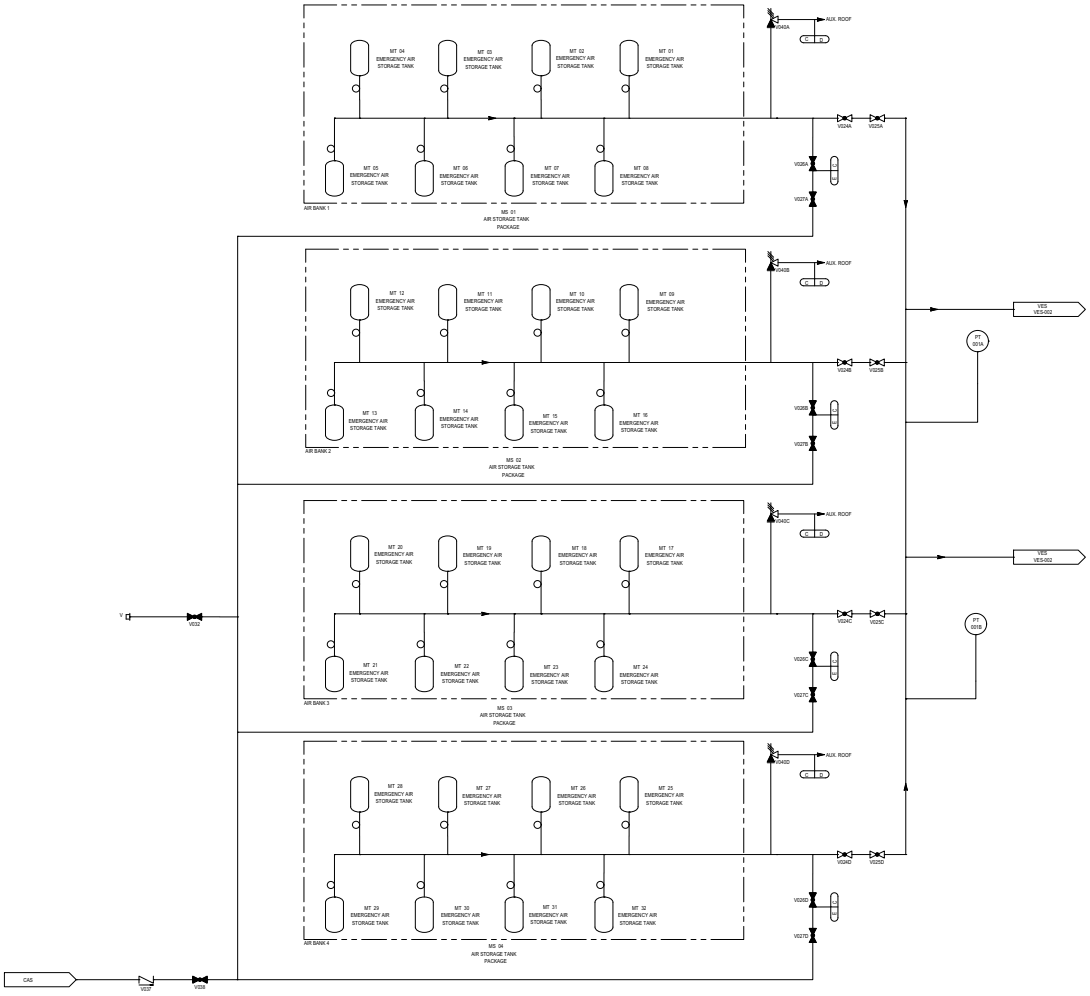

Figure 6.4-2 (Sheet 1 of 2)
Simplified Main Control Room Habitability System Piping and Instrumentation Diagram (REF) VES 001

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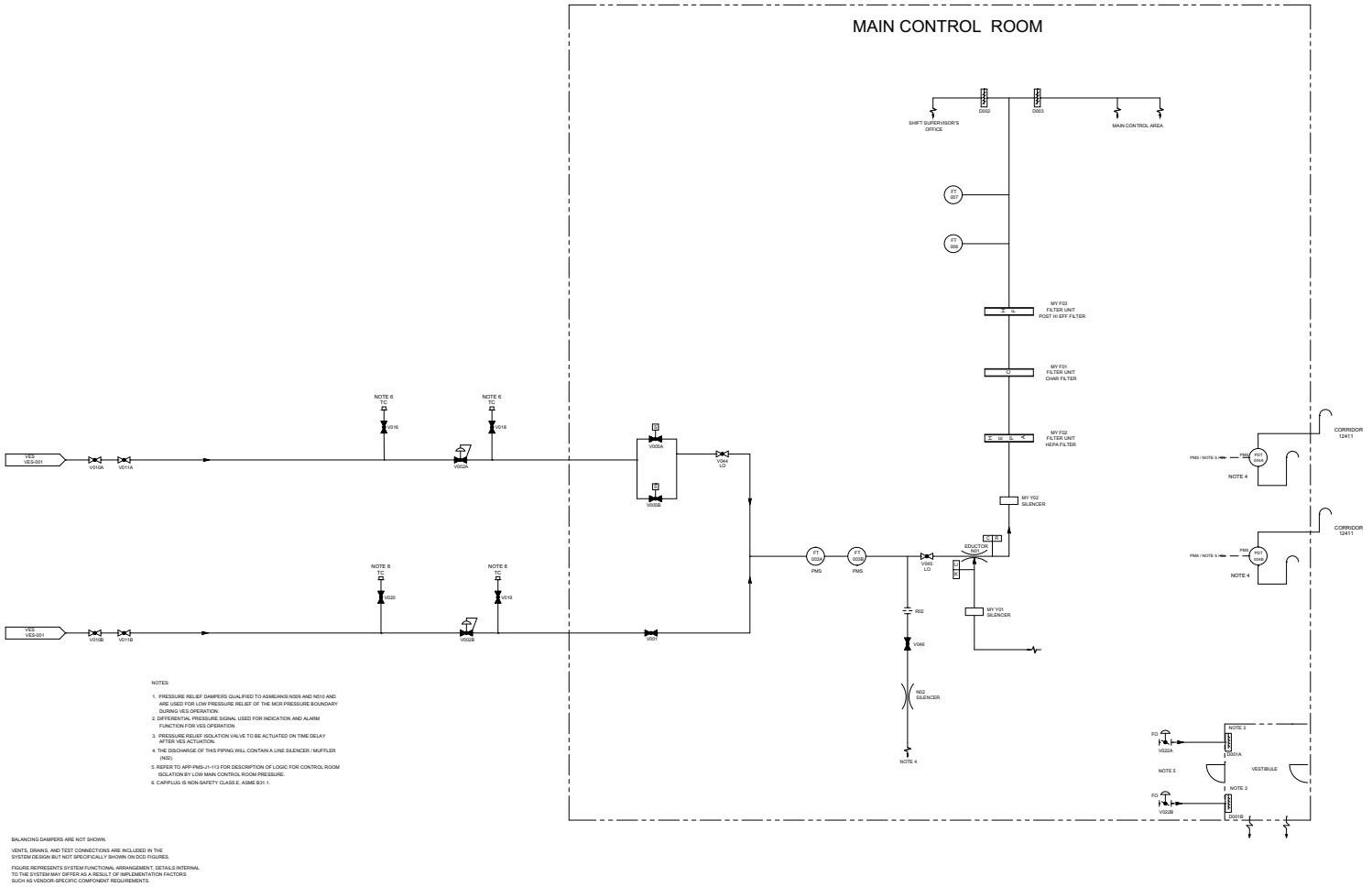

Figure 6.4-2 (Sheet 2 of 2)
Simplified Main Control Room Habitability System Piping and Instrumentation Diagram
(REF) VES 002

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6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Feature (ESF) Filter Systems

This subsection is not applicable to the AP1000.

6.5.2 Containment Spray System

In the event of a design basis LOCA there is an assumed core degradation that results in a significant release of radioactivity to the containment atmosphere. This activity would consist of noble gases, particulates, and a small amount of elemental and organic iodine (as discussed in Subsection 15.6.5.3, most of the iodine would be in the particulate form). The AP1000 does not include a safety-related containment spray system to remove airborne particulates or elemental iodine. Removal of airborne activity is by natural processes that do not depend on sprays (that is sedimentation, diffusiophoresis, and thermophoresis). These removal mechanisms are discussed in Appendix 15B.

Much of the non-gaseous airborne activity would eventually be deposited in the containment sump solution. Long-term retention of iodine in the containment sump following design basis accidents requires adjustment of the sump solution pH to 7.0 or above. This pH adjustment is accomplished by the passive core cooling system and is discussed in Subsection 6.3.2.1.4.

In accordance with Reference 1, the fire protection system provides a nonsafety-related containment spray function for accident management following a severe accident. This design feature is not safety-related and is not credited in any accident analysis including the dose analysis provided in Subsection 15.6.5. Dose reduction following a severe accident may be enhanced over the natural removal mechanisms via the nonsafety-related containment spray. Subsection 15.6.5.3.2 provides additional discussion of the natural removal mechanisms. The following subsections provide a discussion of the nonsafety-related containment spray function provided by the fire protection system.

6.5.2.1 System Description

The fire protection system provides a nonsafety-related containment spray function for severe accident management. Subsection 9.5.1 provides a description of the fire protection system including equipment and valves that support the containment spray function such as the fire pumps and fire main header. This section provides the description of the portion of the fire protection system designed specifically to provide the containment spray function.

The source of water for the containment spray function is provided by the secondary fire protection system water tank. Either the motor driven or diesel driven fire protection system pump may be used to deliver fire water to the containment spray header. The flow path to containment is via the normal fire main header as shown in Figure 9.5.1-1, sheets 1 through 3. The containment spray flow path is from the fire main extension, through the fire protection system line that penetrates containment, to the containment spray riser that connects to the fire protection system header inside containment. This riser supplies two ring headers located above the containment polar crane.

6.5.2.1.1 Valves

The containment spray flow path from the fire main header contains one normally open manual valve (FPS-V048), one normally closed manual valve (FPS-V101), one locked closed manual containment isolation valve outside containment (FPS-V050), a containment isolation check valve inside containment (FPS-V052), a normally open manual isolation valve in the spray riser (FPS-V700), and

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a normally closed remotely-operated valve (FPS-V701) downstream of the manual isolation valve in the spray riser.

Containment spray is initiated by first closing the passive containment cooling water system fire header isolation valve (PCS-V005) isolating the passive containment cooling water storage tank, opening the manual valves outside containment, and by opening the remotely-operated valve inside containment. The manual valves outside containment are located in valve / piping penetration room 12306. The valves are located close to the entrance door such that radiation exposures to an individual required to enter the room and align the valves would not exceed the prescribed post-accident dose limits discussed in Subsection 12.4.1.8.

Valve FPS-V701 is a fail-open air-operated valve such that the containment spray flow path can be opened following a loss of the nonsafety-related compressed and instrument air system. During shutdown operations, the fire protection system header inside containment is pressurized from the passive containment cooling water storage tank for fire protection and manual isolation valve FPS-V700 is closed.

6.5.2.1.2 Containment Spray Header and Nozzles

The containment spray header consists of a single header that feeds two ring headers located above the containment polar crane. The containment spray ring headers and spray nozzles are oriented to maximize containment volume coverage. A lower ring header is located at plant elevation 260 feet, and contains 44 spray nozzles. An upper ring header is located at plant elevation 275 feet, and contains 24 spray nozzles.

The nozzles within the spray ring header are conventional containment spray nozzles utilized in past Westinghouse pressurized water reactors. The spray nozzles are selected on the basis of drop size to provide adequate absorption of fission products from the containment atmosphere.

6.5.2.1.3 Applicable Codes and Classifications

The containment spray function is not safety-related, and therefore the valves and piping in the containment spray flow path are not required to be safety-related for the containment spray function. However, the containment isolation piping and valves are safety-related (AP1000 Equipment Class B) to perform the safety-related function of containment isolation. The classification of the remaining portions of the fire header are nonsafety-related, and are classified as Class F as discussed in Subsections 3.2.2.7 and 9.5.1. The containment spray header and valve, downstream of the manual isolation valve inside containment is nonsafety-related and classified as Class E. The containment spray header is classified as Seismic category II.

6.5.2.1.4 System Operation

During normal operation, the fire protection system header inside containment is isolated from the fire main header by closed isolation valves, including a locked closed containment isolation valve. The containment spray piping is therefore not pressurized during normal operation. During plant shutdown modes, personnel access to containment is required, and as such, the fire protection system standby header inside containment is pressurized by the water in the passive containment cooling water storage tank. During these modes, the manual isolation valve located between the header and the spray ring is closed to further isolate the containment spray header from the passive containment cooling water storage tank. Inadvertent actuation of the containment spray system during power operation and shutdown is not credible. Inadvertent actuation of the containment spray would require multiple failures of closed valves.

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Severe accident management guidelines provide the operator with guidance to initiate the containment spray feature of the fire protection system. Operator action to open two manual isolation valves outside of containment followed by remotely opening the containment spray isolation valve within containment from either the main control room or the remote shutdown workstation will initiate the spray function. Containment spray may be terminated at any time by closing the remotely operated isolation valve within containment, or by closing any of the manual valves in the containment spray flow path outside containment. Operation of the containment spray will have no effect on the availability of the remainder of the fire protection system other than the loss of inventory from the secondary fire water tank due to the sprayed water. To preserve inventory for firefighting, the primary fire water tank is isolated during containment spray operation. Since the fire protection system operates in the active standby mode, i.e. the supply piping is kept full and pressurized, once the remotely operated isolation valve is opened the system will perform the containment spray function.

When water pressure in the fire main begins to fall, due to a demand for water from containment spray, the motor-driven pump starts automatically on a low-pressure signal. If the motor-driven pump fails to start, the diesel-driven pump starts upon a lower pressure signal. The pump continues to run until it is stopped manually.

6.5.2.2 Design Evaluation

6.5.2.2.1 Containment Coverage

The containment spray nozzles are the Lechler (SPRACO Company) spray nozzles or equivalent, which provide a drop size distribution which has been established by testing and found suitable for fission product removal. The fire protection system header provides a containment spray nozzle differential pressure of 40 psid, which fixes the drop size distribution. The mass mean drop size produced at this differential pressure is conservatively assumed to be 1000 microns.

The fire protection system header can provide the design flow rate of 15.2 gpm to each spray nozzle at a containment backpressure of 20 psig for a total containment spray flow of approximately 1034 gpm. Analyses of severe accident sequences show that containment backpressure is less than 20 psig after containment spray flow is initiated.

Figure 6.5-1 is a diagram of containment which shows the developed spray patterns for the containment spray ring headers. The overlay of the spray patterns on the containment is useful in illustrating the completeness of spray coverage in the sprayed region. Furthermore, as discussed in Reference 2, there is significant momentum exchange between the spray droplets and the closed air volume of the containment, which provides far greater mixing within the sprayed region than the idealized spray patterns would indicate. Therefore, even though small areas of the sprayed region are not directly sprayed by the developed spray patterns, the sprayed region of the containment is well-mixed.

The sprayed regions of containment include the region of containment above the operating deck, and the refueling cavity, which is open at the operating deck. The total free volume of the sprayed region is approximately 1.7×10^6 cubic feet which represents approximately 84% of the total containment free volume.

6.5.2.2.2 Aerosol Removal Effectiveness of Sprays

The removal of aerosol activity from the containment atmosphere by sprays is simply described by:

$$C_t = C_o e^{-\lambda t}$$

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where:

C_t = concentration of aerosols at time "t"

C_o = initial concentration of aerosols

λ = aerosol removal coefficient for sprays (hr^{-1})

t = elapsed time (hr)

However, to fully model the removal of aerosols from the containment atmosphere in a severe accident, the analysis also needs to take into account mixing between the sprayed and unsprayed regions and the rate of release of activity from the core into the containment atmosphere.

6.5.2.2.3 Aerosol Removal Coefficient for Sprays

The aerosol removal coefficient for sprays is calculated by the following equation from the Standard Review Plan (Reference 2):

$$\lambda = 3hfE / 2Vd$$

where:

h = average spray drop fall height (ft)

f = spray flow rate (ft^3/hr)

E = collection efficiency

V = volume of the containment exposed to sprays

d = average spray drop diameter (ft)

Reference 2 identifies a value for E/d of 10 m^{-1} (3.05 ft^{-1}) as being conservative until the air concentration is reduced by a factor of 50. Using this together with a nominal spray fall height of 125 feet and a nominal flow rate of 1000 gpm (8022 ft^3/hr), the aerosol removal coefficient for the containment sprays is approximately 2.7 hr^{-1} in the sprayed volume. This spray removal coefficient is significantly greater than that associated with the natural removal mechanisms assumed in the design basis analysis (see Appendix 15B) and would enhance dose reduction following a severe accident.

The decontamination factor (DF) that would be achieved at any point in time is dependent on the timing of spray operation. Additionally, the continuing release of activity must be factored into the determination of DF (i.e., the DF would be based on the integrated activity release to the containment at a point in time, not on the amount of activity present in the containment atmosphere at the time spray operation is initiated). After a DF of 50 is reached, the value of E/D would be reduced by a factor of ten (Reference 2) and the aerosol removal coefficient would also be reduced by the same factor to a value of 0.27 hr^{-1}. Based on an assumed spray actuation shortly after the onset of core melt and a nominal spray duration of three hours, the DF of 50 would not be reached until after spray operation was terminated.

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6.5.3 Fission Product Control Systems

The containment atmosphere is depleted of elemental iodine and particulates as a result of the passive removal processes discussed in Appendix 15B. No active fission product control systems are required in the AP1000 design to meet regulatory requirements. The passive removal processes and the limited leakage from the containment of less than L_a as defined in the Containment Leakage Rate Testing Program, result in doses less than the regulatory guideline limits. (See Subsection 15.6.5.3.)

6.5.3.1 Primary Containment

The containment consists of a freestanding cylindrical steel vessel with ellipsoidal heads. The containment structural design is presented in Subsection 3.8.2.

The containment vessel, penetrations, and isolation valves function to limit the release of radioactive materials following postulated accidents. The resulting offsite doses are less than regulatory guideline limits. Containment parameters affecting fission product release accident analyses are given in Table 6.5.3-1.

Long-term containment pressure and temperature response to the design basis accident are presented in Section 6.2.

The containment air filtration system may be operated for personnel access to the containment when the reactor is at power, as presented in Subsection 9.4.7. For this reason, the radiological assessment of a loss-of-coolant accident assumes that both trains of the air filtration system are in service at the initiation of the event. The isolation valves receive automatic signals to close from diverse parameters. The valves are designed to close automatically as described in Subsection 6.2.3.

Containment hydrogen control systems are presented in Subsection 6.2.4.

6.5.3.2 Secondary Containment

There is no secondary containment provided for the fission product control following design basis accident.

The annulus between containment and shield building from the elevation 100′-0″ to the elevation 132′-3″ acts as a holdup volume to limit the spread of fission products following severe accident. Most containment penetrations are located within this holdup volume. It is served by the radiologically controlled area ventilation system (VAS) described in Subsection 9.4.3. Isolation dampers are provided to reduce the air interchange between the holdup volume and environment. Fission product control via holdup within the annulus is considered in severe accident dose analysis but excluded from consideration for design basis accident dose evaluations presented in Chapter 15.

6.5.4 Combined License Information

This section contained no requirement for additional information.

6.5.5 References

1. SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," June 30, 1997.

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2. NUREG-0800, Section 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System."

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Table 6.5.3-1
Primary Containment Operation
Following a Design Basis Accident

Type of structure	Freestanding cylindrical steel vessel with ellipsoidal heads
Containment free volume (ft^3)	2.06×10^6
Design basis containment leak rate	0.10% containment air weight per day

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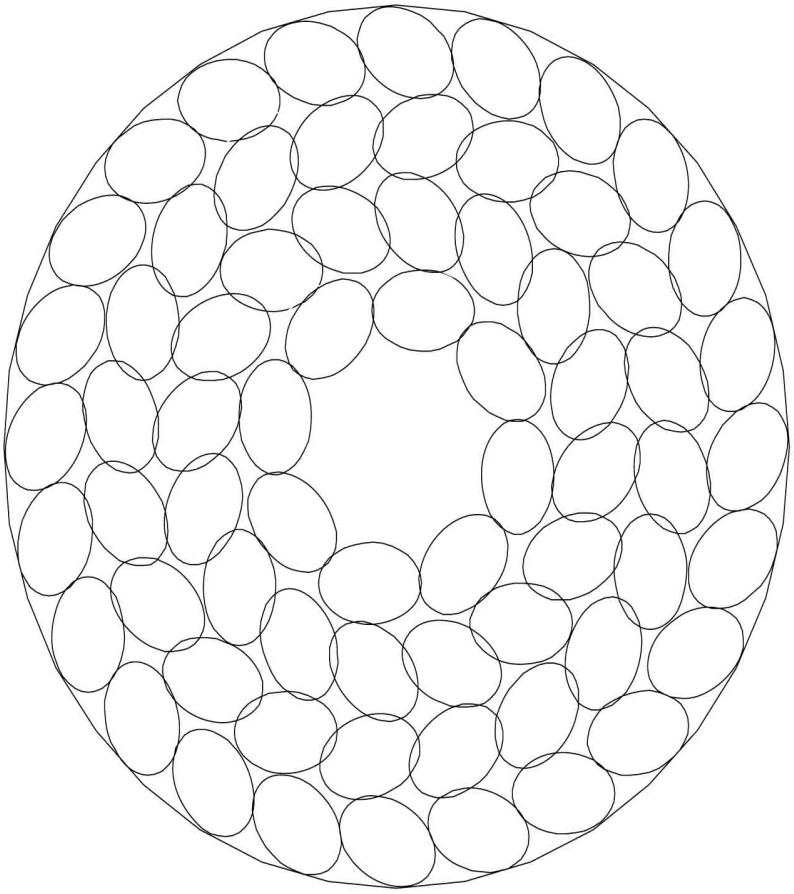

Figure 6.5-1
Containment Spray Coverage at Operating Deck

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6.6 Inservice Inspection of Class 2, 3, and MC Components

The inservice inspection program incorporates the ASME Boiler and Pressure Vessel Code approved in 10 CFR 50.55a(b) subject to the limitations and modifications listed in 10 CFR 50.55a(b).

6.6.1 Components Subject to Examination

Preservice and inservice inspections of Quality Group B and C pressure retaining components (ASME Code, Section III Class 2 and 3 components) such as vessels, piping, pumps, valves, bolting, and supports as identified in Subsection 3.2.2 are performed in accordance with the ASME Code, Section XI, as required by 10 CFR 50.55a(g). This includes the ASME Code Section XI Mandatory Appendices. Preservice and inservice inspections of Quality Group B components that are ASME Class MC (metallic containment) pressure-retaining components and integral attachments are performed in accordance with the ASME Code, Section XI, as required by 10 CFR 50.55a. Refer to Subsection 3.8.2, “Steel Containment” for design details, including accessibility of primary containment.

The responsibility for preparation of the pre-service inspection program (nondestructive examination) is described in Subsection 6.6.9. The responsibility for the inservice inspection program that is required prior to commercial operation is described in Subsection 6.6.9. These programs will address applicable inservice inspection provisions of 10 CFR 50.55a(g). The pre-service program will provide details of areas subject to inspection, as well as the method and extent of pre-service inspection. The inservice inspection program will detail the areas subject to inspection and method, extent, and frequency of inspection.

Class 2 and 3 components are included in the equipment designation list and the line designation list contained in the inservice inspection program.

6.6.2 Accessibility

ASME Code Class 2, 3, and MC components are designed so that access is provided in the installed condition for visual, surface and volumetric examinations specified by the ASME Code. See Subsection 5.2.1.1 for a discussion of the baseline ASME Code edition and Addenda. Design provisions, in accordance with Section XI, IWA-1500, are formally implemented in the Class 2, 3, and MC component design processes.

The goal of designing for inspectability is to provide for the inspectability access and conformance of component design with available inspection equipment and techniques. Factors such as examination requirements, examination techniques, accessibility, component geometry and material selection are used in evaluating component designs. Examination requirements and examination techniques are defined by inservice inspection personnel. Inservice inspection review as part of the design process provides component designs that conform to inspection requirements and establishes recommendations for enhanced inspections.

Considerable experience has been drawn on in designing, locating, and supporting Quality Group B and C (ASME Class 2 and 3) and Class MC pressure-retaining components to permit pre-service and inservice inspection required by Section XI of the ASME Code. Factors such as examination requirements, examination techniques, accessibility, component geometry, and material selections are used in establishing the designs. The inspection design goals are to eliminate uninspectable components, reduce occupational radiation exposure, reduce inspection times, allow state-of-the-art inspection systems, and enhance detection and the reliability of flaw characterization. There are no Quality Group B and C components or Class MC components, which require inservice inspection during reactor operation.

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Removable insulation is provided on piping systems requiring volumetric and surface inspection. Removable hangers and pipe whip restraints are provided, as necessary and practical, to facilitate inservice inspection. Working platforms are provided in areas requiring inspection and servicing of pumps and valves. Temporary or permanent platforms, scaffolding, and ladders are provided to facilitate access to piping welds. The components and welds requiring inservice inspection are designed to allow for the application of the required inservice inspection methods, that is, sufficient clearances for personnel and equipment, maximized examination surface distances, two-sided access (except for valve to pipe welds), favorable materials, weld joint simplicity, elimination of geometrical interferences, and proper weld surface preparation.

Many of the ASME Code, Section III, Class 2 and 3 components are included in modules which are fabricated offsite and shipped to the site, as described in Subsection 3.9.1.5. The modules are designed and engineered to provide access for in-service inspection and maintenance activities. The attention to detail that is engineered into the modules prior to construction improves the accessibility for inspection and maintenance.

Future unanticipated changes in the Section XI requirements could, however, necessitate relief requests. Relief from the inspection requirements of Section XI will be requested when full compliance is not practical according to the requirements of 10 CFR 50.55a. In such cases, specific information will be provided to identify the applicable ASME Code requirements, justification for the relief request, and the inspection method to be used as an alternative.

Space is provided to handle and store insulation, structural members, shielding, and other material related to the inspection. Suitable hoists and other handling equipment, lighting, and sources of power for inspection equipment are installed at appropriate locations.

During the construction phase of the project, anomalies and construction issues are addressed using change control procedures. Modifications reviewed following design certification adhere to the same level of review as the certified design per 10 CFR Part 50, Appendix B as implemented by the Westinghouse Quality Management System (QMS). The QMS requires that changes to approved design documents, including field changes, are subject to the same review and approval process as the original design. This explicitly requires the field change process to follow the same level of review that was required during the design process. Accessibility and inspectability are key components of the design process.

Control of accessibility for inspectability and testing during post-design certification activities is provided via procedures for design control and plant modifications.

6.6.3 Examination Techniques and Procedures

The visual, surface, and volumetric examination techniques and procedures are in accordance with the requirements of ASME Code, Section XI, Article IWA-2000. Approved Code Cases listed in Regulatory Guide 1.147 are applied as the need arises during the pre-service inspection. Approved Code Cases determined as necessary to accomplish pre-service inspection activities are used.

6.6.3.1 Examination Methods

Visual Examination

Examination methods are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

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Surface Examination

Examination methods are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

Ultrasonic Examination

Examination methods are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

Alternative Examination Techniques

Examination methods are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

6.6.3.2 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII.

6.6.3.3 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(z) or 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.

6.6.4 Inspection Intervals

Inspection intervals included in the inspection program are as defined in Subarticles IWA-2400, IWC-2400, IWD-2400, IWE-2400, and IWF-2400 of the ASME Code, Section XI. The periods within each inspection interval may be extended by as much as one year to permit inspections to be concurrent with plant outages. It is intended that inservice examinations be performed during normal plant outages, such as refueling shutdown or maintenance shutdowns occurring during the inspection interval.

Inspection intervals meet the requirements of ASME Section XI as specified in 10 CFR 50.55a.

6.6.5 Examination Categories and Requirements

Examination categories and examination requirements (examination methods, acceptance criteria, extent of examination, and frequency of examination) for Class 2 components are in accordance with Subsection IWC and Table IWC-2500-1 of the ASME Code, Section XI. Similar information for Class 3 components is in conformance with Subsection IWD and Table IWD-2500-1. For component supports, examination categories and examination requirements are in conformance with Subsection IWF and Table IWF-2500-1; and for Class MC components, examination categories and examination requirements are in conformance with Subsection IWE and Table IWE-2500-1 of ASME Code, Section XI.

The pre-service examination of Class 2 components is according to the requirements of Subarticle IWC-2200. The pre-service examination of Class MC components is in accordance with the requirements of Subarticle IWE-2200. The pre-service examination requirements for component

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supports are in accordance with the requirements of Subarticle IWF-2200. The pre-service examination of Class 3 components is according to the requirements of Subarticle IWD-2200.

As provided in ASME Section XI, IWC-1220, IWD-1220, and IWE-1220, certain portions of Class 2, 3, and MC systems are exempt from the volumetric, surface, and visual examination requirements of IWC-2500, IWD-2500, and IWE-2500. Supports associated with Class 2, 3, and MC components are also exempt in accordance with the requirements of IWF-1230.

6.6.6 Evaluation of Examination Results

Examination results are evaluated per the acceptance standards found in IWA-3000, IWC-3000, IWD-3000, IWE 3000, and IWF-3000 of the ASME Code, Section XI. Repair and replacement procedures are in accordance with ASME Code, Section XI, Article IWA-4000.

Evaluation of examination results are conducted in accordance with ASME Section XI, as modified or required by 10 CFR 50.55a.

6.6.7 System Pressure Tests

System pressure tests comply with IWA-5000, IWC-5000, IWD-5000, and IWE-5000 of the ASME Code, Section XI, for Class 2, 3, and MC components. Pressure testing of Class MC components is performed per the 10 CFR 50 Appendix J “Containment Leak Rate Testing” Program.

6.6.8 Augmented Inservice Inspection to Protect against Postulated Piping Failures

An augmented inspection program is developed for high-energy fluid systems piping between containment isolation valves. Such a program is also developed where no isolation valve is used inside containment between the first rigid pipe connection to the containment penetration or the first pipe whip restraint inside containment and the outside isolation valve. This program provides for 100 percent volumetric examination of welds in the affected piping during each inspection interval, conducted according to the ASME Code, Section XI. The program covers the break exclusion portion of high-energy fluid systems described in Subsections 3.6.1 and 3.6.2.

There is no requirement for an augmented inspection of ASME Code, Section III Class 1, 2, or 3 pipe to address erosion-corrosion-induced pipe wall thinning. Class 1, 2, and 3 pipe containing single-phase water or two-phase steam and water is fabricated of erosion-corrosion resistant material. See Section 10.1 for information on monitoring of nonsafety-related pipe for erosion-corrosion.

6.6.9 Combined License Information Items

6.6.9.1 Inspection Programs

The pre-service inspection program (nondestructive examination) and an inservice inspection program for ASME Code, Section III Class 2, 3, and MC systems, components, and supports are addressed in Section 6.6 introduction, and in Subsections 6.6.1, 6.6.3.1, 6.6.3.2, 6.6.3.3, 6.6.4, and 6.6.6.

6.6.9.2 Construction Activities

The controls to preserve accessibility and inspectability for ASME Code, Section III, Class 2, 3, and MC components and piping during construction or other post design certification activities are addressed in Subsection 6.6.2.

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Appendix 6A Fission Product Distribution in the AP1000 Post-Design Basis Accident Containment Atmosphere

The AP1000 design-basis analyses for hydrogen control (Subsection 6.2.4.3) and natural aerosol removal coefficient (Appendix 15B) assume that the fission products and hydrogen released to the containment following a postulated design basis loss of coolant accident (LOCA) are homogeneously distributed in the containment atmosphere within the open compartments that participate in natural circulation. The purpose of this discussion is to justify the homogeneous assumption for aerosol natural deposition calculations.

The following evaluation includes:

- Identification of the accident sequence assumptions and boundary conditions in the reactor coolant system and containment prior to the fission product and hydrogen releases
- Identification of the limiting steam and fission product release location from the reactor coolant system to the containment
- Discussion of containment natural circulation in quasi-steady conditions
- Discussion of AP1000 passive containment cooling system (PCS) large-scale test (LST) insights that support the well-mixed assumption

6A.1 Design Basis Sequence Assumptions

The design-basis fission product source term (Subsection 15.6.5.3.1) is superimposed onto thermal-hydraulic conditions of the design-basis accident sequence for the evaluation of fission product deposition. The following assumptions define the design basis conditions. The AP1000 design-basis sequence consists of a LOCA which drains the reactor coolant system (RCS) and core makeup tanks (CMTs) sufficiently to activate the automatic depressurization system (ADS). Both trains of all four stages of automatic depressurization system open sequentially. During the depressurization, the core makeup tanks and accumulators inject into the reactor vessel downcomer. The final reactor coolant system pressure is essentially equal to the containment pressure which allows gravity injection of the IRWST water. Steam is produced in the vessel at the rate dictated by decay heat minus the heat in the volatile fission products which have been released from the core. The passive containment cooling system water flow is initiated based on High-2 containment pressure from the blowdown or the automatic depressurization system prior to the release of fission products.

Fission product release occurs from a fully depressurized reactor coolant system. The aerosols are carried into the containment in a buoyancy-driven steam flow. The earliest time of fission product release from core degradation is well past the time of the blowdown and automatic depressurization. The containment condition during and following the release is quasi-steady-state. Internal heat sinks are assumed to be essentially thermally-saturated and no longer effective, and the condensation rate of steam on the containment dome and shell is equivalent to the decay heat steaming rate.

6A.1.1 Break Size and Fission Product Release Location in Containment

This section discusses each of the postulated fission product release locations from the reactor coolant system, the containment location for each, the size limitations and the phenomena associated with the break locations. It is shown that it is appropriate to assume that the steam and fission products are released from the reactor coolant system hot leg to the containment above the maximum water flood-up elevation in the steam generator compartment gas space.

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6A.1.1.1 Releases From Depressurization System Lines

Any design-basis LOCA which can be postulated to produce a large core activity release to containment will actuate the four stages of the automatic depressurization system. The stage 1, 2 and 3 automatic depressurization system lines, which relieve from the top of the pressurizer (see Figure 6A-1), deliver flow to the containment through the in-containment refueling water storage tank (IRWST). This is not considered to be a major fission product release pathway because the IRWST is a cold, effectively closed system with no leakage pathway to the environment. The IRWST is nearly full of water during the depressurization blowdown which would trap any postulated fission products released to the IRWST. At the time the water is drained below the spargers, the reactor coolant system is depressurized with stage 4 automatic depressurization system open, and the IRWST vents, which are closed with flappers, are not expected be significantly opened by the small buoyancy-driven flows. Aerosols released from stages 1, 2 and 3, either before or after the draining of the IRWST, would essentially be trapped in the water or in the IRWST compartment. Therefore, this pathway is conservatively neglected as a release pathway from the reactor coolant system to maximize the activity entering the containment atmosphere.

Stage 4 automatic depressurization system lines relieve reactor coolant system coolant, steam, and fission products from the hot legs (see Figure 6A-1) to the steam generator compartments above the maximum water flood-up level. The stage 4 lines consist of four 14-inch schedule 160 lines. Two lines are connected to each of the two hot legs. Each of these trains relieves at the 112-foot elevation to a steam generator compartment.

Of the postulated release locations in the reactor coolant system, openings in the hot-side piping, such as the stage 4 automatic depressurization system, provide the lowest resistance pathway for fission product releases to the containment because of the large flow area, high temperatures, short resident time and low surface area for aerosol deposition in the reactor coolant system. To reach openings in the cold side piping when stage 4 automatic depressurization system valves are open, the reactor coolant system low-pressure natural circulation flow must pass through the steam generator tubes (see Figure 6A-1). At the superheated steam temperature of the gas which accompanies the fission product flow, significant heat transfer would take place in the steam generator tubes which are cooled on the secondary side by water. Aerosol deposition to the tubes would remove fission products from the release before the flow reached the containment. Therefore, releases from cold-side breaks are less severe than hot side breaks with the stage 4 automatic depressurization system open.

6A.1.1.2 Releases From Coolant Loop Breaks

Breaks in the reactor coolant system loop piping (hot legs or cold legs) relieve primary coolant, steam and fission products to the steam generator compartments. Assuming double-ended guillotine breaks, the hot-leg break has a diameter of 31 inches (78.7 cm) and the cold-leg break has a diameter of 22 inches (55.9 cm). Breaks in the hot leg piping are more limiting than breaks in the cold leg with respect to the fission product releases to the containment because of the larger break area, higher temperatures, shorter resident time and lower surface area for aerosol deposition in the reactor coolant system. Therefore, of the coolant loop breaks, hot leg breaks to the steam generator compartment provide the more conservative magnitude of fission product release to the containment. Because of the similar fission product flow path, release magnitude and release location, the hot leg breaks can be lumped with the stage 4 automatic depressurization system releases.

6A.1.1.3 Direct Vessel Injection Line Breaks

A break in one of the two direct vessel injection lines can relieve steam and fission products outside the steam generator compartments to one of the two dead-ended accumulator compartments below the core makeup tank room. The piping is 8-inch diameter schedule 160 piping, but an orifice at the

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reactor vessel wall limits the break size to a 4-inch diameter. The nozzle connects to the reactor vessel downcomer (see Figure 6A-1), so all direct vessel injection line breaks relieve from the cold-side of the reactor coolant system. The accumulator compartments have significant heat sink surfaces (equipment, grating, support structures and compartment walls) for aerosol deposition to trap fission products within the dead-ended compartment. Given the small break size, cold-side location of the break, the compartment retention capacity, and the large relief flow area associated with the open stage 4 automatic depressurization system valves, very little fission product release is expected from the direct vessel injection line. The steam release to an accumulator compartment is negligible with respect to that from the stage 4 automatic depressurization system.

6A.1.1.4 Core Makeup Tank Balance Line Breaks

Breaks in the core makeup tank balance lines can relieve steam and fission products to the core makeup tank room. The balance line piping is 8-inch diameter schedule 160 piping. The balance line nozzle is attached to a cold leg (see Figure 6A-1). Given the small break size, cold-side location of the break, the compartment retention capacity, and the large relief flow area associated with the open stage 4 automatic depressurization system valves, very little fission product release is expected from the balance line. The steam, hydrogen and fission product releases to the core makeup tank room is negligible with respect to the release from the stage 4 automatic depressurization system.

6A.1.1.5 Chemical and Volume Control System Line Breaks

A break in the chemical and volume control system (CVS) line relieves to the dead-ended chemical and volume control system compartment below the core makeup tank room. The chemical and volume control system piping is 3-inch diameter schedule 160 piping. The inlet of the chemical and volume control system draws from the cold leg and the outlet discharges to the reactor coolant pump suction, both on the cold-side of the reactor coolant system (see Figure 6A-1). Given the small break size, cold-side location of the break, the compartment retention capacity, and the large relief flow area associated with the open stage 4 automatic depressurization system valves, very little fission product release is expected from the chemical and volume control system piping. The steam release to the chemical and volume control system compartment is negligible with respect to that from the stage 4 automatic depressurization system.

6A.1.1.6 Fission Product Release Location Conclusion

The fission product releases are expected to discharge mainly from the stage 4 automatic depressurization system lines, which relieve from the hot legs to the steam generator compartments. Stage 4 automatic depressurization system is open in all design-basis LOCA sequences that can be postulated to produce large core activity releases to the containment. For a coolant loop break, the release would also go to the steam generator compartments along with the releases from the stage 4 automatic depressurization system lines. Fission products released to other postulated containment locations are negligible by comparison because the releases are from the cold-side of the reactor coolant system through comparatively long and narrow piping pathways. Therefore, the bounding release pathway is a hot-side break into the steam generator compartments with fission product and steam releases through the break and stage 4 automatic depressurization system.

6A.2 Containment Natural Circulation and Mixing

This section describes the natural circulation flow path and the entrainment processes in the containment atmosphere. Figure 6A-2 graphically depicts the containment natural circulation flow paths and the entrainment processes.

The steam plume, rising from a point low in the containment, and the condensation on the containment surface and wall entrainment rates provide the driving forces for natural circulation in the

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containment. Based on the sequence timing, the containment conditions at the time of the fission product releases are quasi-steady-state. Therefore, it is assumed:

$Q_{st} \approx$ constant

$Q_{cond} = Q_{st}$

where:

Q_{st} = steam volumetric flowrate

Q_{cond} = condensation volumetric flowrate.

Steam and fission products are released low in the containment through stage 4 automatic depressurization system at the 112-foot elevation as hot, buoyant plumes from the low pressure primary system into the steam generator compartments. Entrainment into the rising plume drives circulation of surrounding atmosphere into the bottom of the steam generator compartment through the openings to the core makeup tank room. The fission products are released from the reactor coolant system with the steam plumes. The plumes rise through the steam generator compartments, mix with the flow entrained from below and are released into the upper compartment at the top of the steam generator doghouses (153-foot elevation). The plumes rise unconstrained for over 100 feet in the containment. As the plumes rise, the surrounding upper compartment gas mixture is entrained. The steam, fission products and any non-condensable gases (e.g. hydrogen and air) in the plumes are mixed with a large volume of entrained atmosphere in the rising plume.

An estimate of the volume entrained into the plume above the operating deck is made conservatively neglecting entrainment into the lower steam generator compartment, and assuming the plumes from the two steam generator compartments behave as one:

$Q_{ent} = 0.15 * B^{1/3} * Z^{5/3}$ (Reference 1)

where:

Q_{ent} = volumetric flowrate of entrained gas in the rising plume above the operating deck

Z = height of rising plume

B = $g * Q_{ST} * (\rho_{amb} - \rho_{st})/\rho_{amb}$

g = gravitational acceleration

The fission product releases occur at approximately 1 hour when the best estimate (no uncertainty) 1979 ANS decay heat rate is 1.4%. At one hour, the volatile fission products which are released from the core contribute 30% of the decay heat, so the decay heat fraction is 1.0% and 34 MW of steam is generated in the reactor vessel. At a containment pressure of approximately 50 psia, the source flow is approximately 295 ft^3/sec and $\Delta\rho/\rho$ is approximately 1/4. Thus, $B^{1/3}$ = 13.3 $ft^{4/3}$/sec. For a release into the upper compartment where Z = 125 ft, Q_{ent} = 6250 ft^3/sec and Q_{ent}/Q_{st} = 21.2.

At 24 hours, best estimate (no uncertainty) 1979 ANS decay heat is 0.6%, and the volatile fission products released from the core contribute 15% of the decay heat. The heat generated in the vessel, generating steam is 17.3 MW, assuming the containment pressure is 34 psia and $\Delta\rho/\rho$ = 0.32. So the source flow is approximately 216 ft^3/sec, $B^{1/3}$ = 13.1 $ft^{4/3}$/sec, Q_{ent} = 6142 ft^3/sec and Q_{ent}/Q_{st} = 28.4.

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Therefore, for the AP1000 height above the operating deck, a conservative entrainment ratio for times greater than 1 hour after accident initiation is:

$$Q_{ent}/Q_{st} > 20$$

The application of water to the external surface of the containment shell maintains the containment shell at a cool temperature. The condensation of steam on the containment shell creates a heavy, air-rich downward flowing gas boundary layer on the wall. Fission products are carried along in the wall layer flow. As it flows downward along the wall, the wall layer also entrains surrounding mixture. Thus, the circulation flow rate in the above-deck volume generates significant circulation flow.

A review of literature on circulation within enclosures (appendix 9.C of (Reference 2) shows that as long as there is cooling on the inner surface of the containment shell, there are no regions of stratification in the containment including under the containment dome. There are significant recirculation flows in the stratified regions between the plume and the wall layer. Thus concentration gradients are small and there are no stagnant regions above the operating deck.

The circulation time constant due to entrainment above the operating deck for the AP1000 can be estimated by $V/(20*Q_{st})$, where V is the containment volume above the operating deck, and the steam generator compartments and core makeup tank room above the 108′ elevation, $2.0x10^6$ ft^3. Therefore, the circulation time constant at 1 hour is approximately 340 seconds. At 24 hours it is 462 seconds. The time constant is estimated to be conservatively large as it does not include entrainment into the downward flowing wall layer. At 1 hour, during the fission product release, the time constant of 340 seconds is very short compared to the 1.3 hour fission product release duration. Therefore, the fission products can be assumed to be homogeneous within the gas volume as soon as they are released. There is no stagnant region in the upper compartment as the entire volume participates in the rising plume, entrainment flow and wall layer. Stratification exists in the form of a relatively shallow, continuous vertical steam gradient as discussed in section 3.0.

Over the time period of interest, no mechanisms exist to separate the non-condensable gases (air and hydrogen) once they are mixed in the rising plumes. The molecular weight difference is so overwhelmed by natural circulation it does not lead to gravitational separation. The terminal gravitational settling rate of hydrogen in air at 1 atm and 25°C is less than 10^{-6} cm/sec (Reference 4). Over the height of the upper compartment, 125 ft, the average separation length is 62.5 ft (1588 cm) so the time for gravitational separation of the hydrogen and air is $1.6x10^9$ seconds. By comparing the separation time to the time constant for the plume entrainment circulation (463 seconds) it is determined that the separation rate is orders of magnitude less effective than the convective mixing forces. Thus gravity effects do not lead to separation of hydrogen from the non-condensable mixture.

As the downward boundary layer flow reaches the operating deck (135-foot elevation), it has been cooled and somewhat depleted of steam. The air, hydrogen and fission products remain well-mixed in the flow. Vents in the operating deck (135′ elevation, see Figure 6A-2) and a gap between the operating deck and the containment wall allow the denser gases to “drain” down into the maintenance floor area and vertical access tunnel through two large vertical openings which empty to the steam generator compartments. Little condensation is expected below the operating deck in the quasi-steady-state condition as the metal heat sinks are essentially thermally-saturated. The condensation on heat sinks below the operating deck is small compared to that on the steel shell. The maintenance floor area and vertical access tunnel communicates with the steam generator compartments such that air flow will freely pass to the steam generator compartments. In the steam generator compartment, the circulation flow is entrained by the initial steam source, and the circuit begins again.

The accumulator and chemical and volume control system compartments and the reactor cavity, including the reactor coolant drain tank room, do not participate in the large-scale natural circulation

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flow as they are dead-ended or filled with water. The IRWST compartment is essentially sealed at the vents by flappers after blowdown. The accumulator and chemical and volume control system compartments, IRWST, reactor cavity and reactor coolant drain tank compartments are not considered in the calculation of the aerosol deposition.

6A.3 Insights From the Passive Containment Cooling System Large Scale Test and AP1000 Stratification Studies

The AP600 passive containment cooling system Large Scale Test (LST) provides insight into the circulation and stratification behavior in the AP1000 containment. The following results are consistent with international test data from various scales (Reference 2, Appendix 9.C). Since the large scale test did not include a flow path into the simulated steam generator compartment, the degree of mixing of injected light non-condensable gases with the existing air throughout the test vessel is conservatively underestimated. This is because the extra flow path would allow density-driven circulation through the path into the compartment, introducing an additional mixing mechanism which exists in AP1000.

In the large scale test rising plume, large amounts of surrounding air-steam mixture were entrained with the released gases. Estimates of entrainment above the deck in large scale test show that about one times the break volumetric flow is entrained. In several large scale test tests, 217.1, 218.1, 219.1, and 221.1, in which helium (a hydrogen simulant) was released in an amount equal to 10-20 volume percent, non-condensable gas concentrations were measured (Reference 3). The helium fraction was reduced from 100% at the release point to 50% of the non-condensable gas in the dome during the initial period of injection. For design basis hydrogen releases, the hydrogen concentration as a fraction of the non-condensable gas in the dome would be much less due to the increased height for entrainment.

The existence of circulation under the dome in the large scale test can be seen based on the reduction of helium non-condensable fraction over time after the helium release stops. The mixing of helium above the deck establishes homogeneous concentrations in only a few minutes in the large scale test. Note that it was seen to take hours for the circulation to mix the injected helium with the non-condensable gases in the compartment below the deck, however, this was due to a lack of a flow path in the simulated steam generator compartments. Because of the additional height for entrainment in the AP1000, circulation is about 10 times greater than in the large scale test based on plume entrainment alone. Wall layer entrainment and circulation through the steam generator compartment would further increase the circulation in AP1000. This result indicates that in the AP1000 circulation distributes the injected non-condensable gases with the air throughout the containment quickly compared to the rate of release.

The effect of external cooling on non-condensable gas distributions was studied in large scale test 219.1 which started out with a dry external shell, injected helium, and then initiated the external water cooling. Non-condensable gas data showed that the application of external cooling acts to accelerate the mixing of non-condensable gases, which is probably due to the higher wall layer entrainment rate from the higher condensation rate on the cooler shell.

As discussed above, the fluid dynamics of entrainment into a buoyant plume and wall boundary layers generate large amounts of circulation within the above deck region. Thus, the region is not a static, layered stratification, and there are no stagnant pockets of gases that do not participate in the circulation. The physics do, however, lead to a standing vertical steam density gradient in the circulating stratified region, which will tend to be slightly richer in steam at the top due to the lower density of the injected steam.

Based on the above, at quasi-steady conditions, the decay heat steaming and heat and mass transfer to the steel shell create natural circulation in the containment that mixes the fission products

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and hydrogen quickly throughout the circulating volume. Circulation time constants indicate that it is reasonable to assume non-condensable gases and fission products can be assumed to be homogeneous in the volumes participating in the circulation. The rising plume and the cooling of the shell create a vertical steam density gradient and a vertical temperature gradient in the upper compartment circulating stratified region. The density and temperature gradients result from a balance between the forces that drive the natural circulation. In the evaluation, no credit is taken for cold plumes falling from the containment dome which cause further circulation above the operating deck.

In Reference 5, studies were performed to demonstrate that the AP1000 containment is at least as well-mixed as the AP600 containment. Studies performed indicate the increase in containment height slightly improves the steady state mixing for the AP1000 when compared to the AP600, and therefore the conclusions regarding the mixing characteristics of the AP600 containment can be applied to the AP1000 containment.

Based on the above, condensation and sensible heat transfer occur over the entire steel shell, albeit at different rates over the height of the shell. As shown in Appendix 15B, thermophoresis and diffusiophoresis are directly related to the heat and mass transfer. Fission products are present at all sites of steam condensation and sensible heat transfer in the containment. In Appendix 15B, the processes are modeled by assuming homogeneous aerosol mass distribution throughout the circulating volume and averaging the steam condensation and sensible heat transfer over the entire upper shell. This treatment provides a valid estimate of the aerosol deposition rates.

6A.4 Conclusions

Based on first principal arguments and insight from testing at various scales, the following conclusions are made with respect to mixing in the AP1000 containment during quasi-steady conditions:

- As long as there is cooling on the inner surface of the containment shell, downward wall flow will prevent stagnation under the dome
- No unmixed pockets develop as the doorways extend to the floor and vents are in the ceiling. For the rooms participating in the natural circulation flow, the entire volume participates in the circulation
- The rising plume, condensation of steam on the containment shell, and downward flowing wall layer create vertical steam density and temperature gradients above the operating deck
- Fission products and hydrogen are quickly and uniformly mixed, relative to the duration of the release, in the containment volumes participating in the natural circulation
- For the purpose of calculating long-term aerosol deposition, it is reasonable to assume that aerosols and non-condensable gases are homogeneous throughout the major compartments participating in the containment natural circulation: the steam generator compartments, upper compartment and core makeup tank room.

6A.5 References

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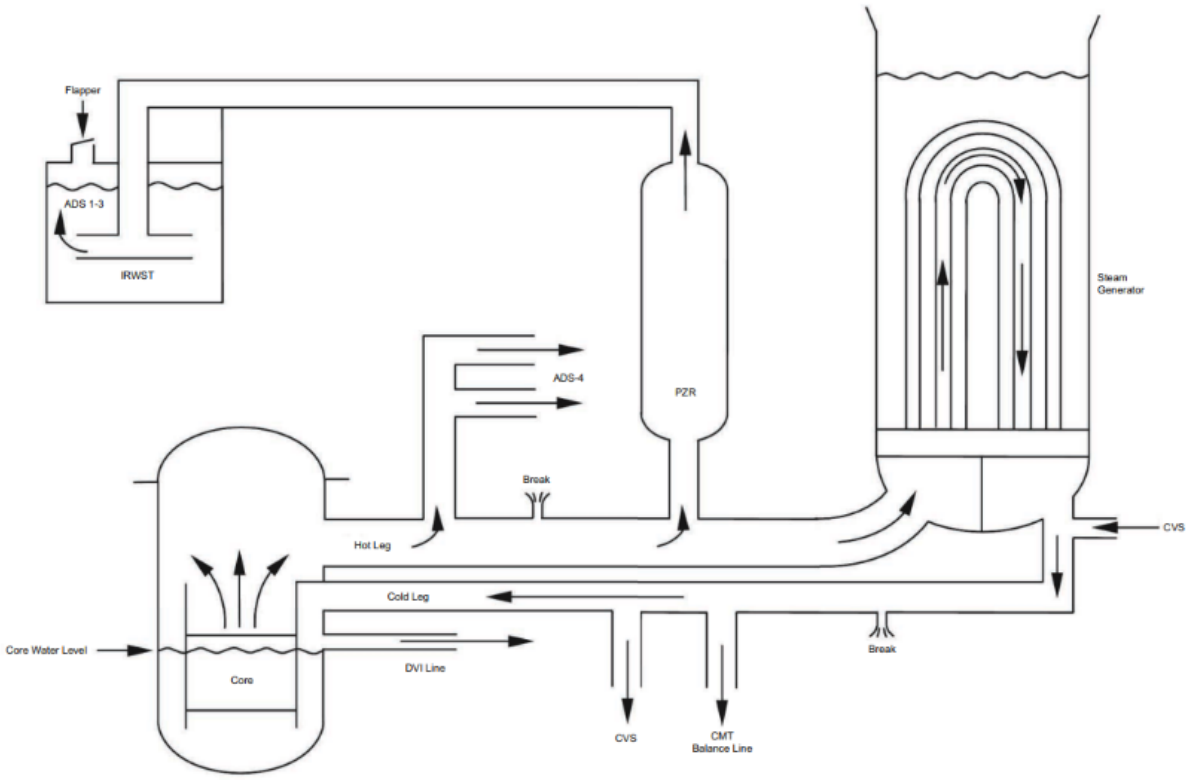

Figure 6A-1
RCS Release Locations

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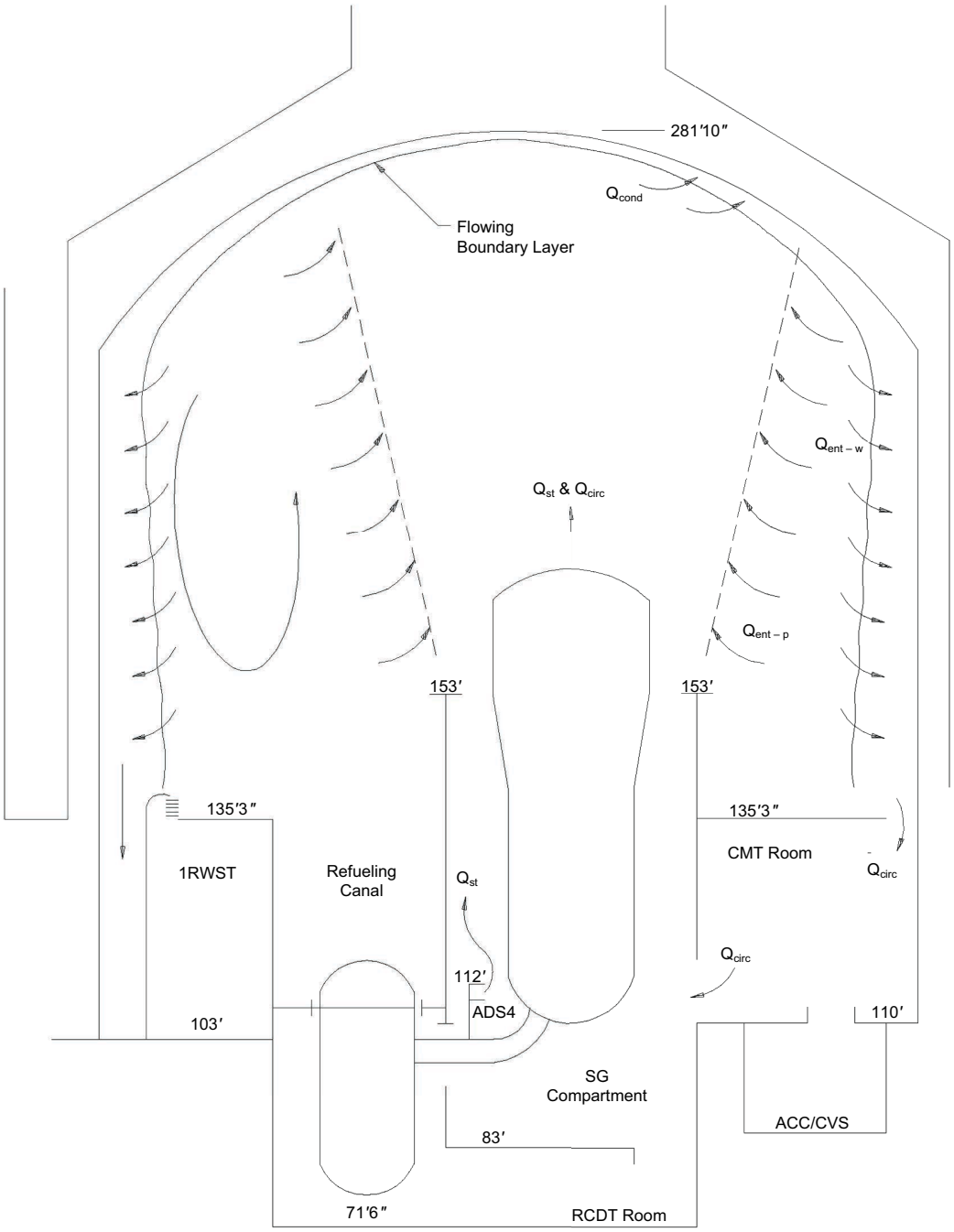

Figure 6A-2
Containment Natural Circulation

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Chapter 7 Instrumentation and Controls

7.1 Introduction

The instrumentation and control systems presented in this chapter provide protection against unsafe reactor operation during steady-state and transient power operations. They initiate selected protective functions to mitigate the consequences of design basis events. This chapter relates the functional performance requirements, design bases, system descriptions, and safety evaluations for those systems. The safety evaluations show that the systems can be designed and built to conform to the applicable criteria, codes, and standards concerned with the safe generation of nuclear power.

Because of the rapid changes that are taking place in the digital computer and graphic display technologies employed in a modern human system interface, design certification of the AP1000 focuses upon the process used to design and implement instrumentation and control systems for the AP1000, rather than on the specific implementation. The design specifics provided here are included as an example for illustration.

Chapter 7 for the AP1000 has been written to describe the protection system hardware utilizing the Common Qualified Platform (Common Q) described in Reference 8 (which includes the NRC SER). The I&C functional requirements of the AP600, which has received Design Certification, have been retained to the maximum extent compatible with the Common Q hardware and software.

The terminology used for Chapter 7 is intended to be independent of any product, but when this is not possible, Common Q terminology is used.

This chapter also discusses the instrumentation portions of the safety-related systems which function to achieve the system responses assumed in the accident analysis, and those needed to shutdown the plant. Section 7.1 describes the AP1000 instrumentation and control architecture, with specific emphasis on the protection and safety monitoring system. The plant control system is discussed briefly. Other systems are discussed in more detail in relevant sections or chapters. Section 7.2 discusses the reactor trip function, and Section 7.3 addresses the engineered safety features (ESF). Systems required for safe shutdown are discussed in Section 7.4 in support of other chapters. Safety-related display instrumentation is discussed in Section 7.5 and interlocks important to safety are presented in Section 7.6. Control systems and the diverse actuation system are discussed in Section 7.7.

Definitions

Terminology used in this chapter reflects an interdisciplinary approach to safety-related systems similar to that proposed in IEEE 603 (Reference 1).

Safety System – The aggregate of electrical and mechanical equipment necessary to mitigate the consequences of design basis events.

Protection and Safety Monitoring System – The aggregate of electrical and mechanical equipment which senses generating station conditions and generates the signals to actuate reactor trip and ESF, and which provides the equipment necessary to monitor plant safety-related functions during and following designated events.

Protective Function – Any one of the functions necessary to mitigate the consequences of a design basis event. Protective functions are initiated by the protection and safety monitoring system logic and will be accomplished by the trip and actuation subsystems. Examples of protective functions are reactor trip and engineered safety features (such as valve alignment and containment isolation).

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Actuated Equipment – The assembly of prime movers and driven equipment used to accomplish a protective function (such as solenoids, shutdown rods, and valves).

Actuation Device – A component that directly controls the motive power for actuated equipment (such as circuit breakers, relays, and pilot valves).

Division – One of the four redundant segments of the safety system. A division includes its associated sensors, field wiring, cabinets, and electronics used to generate one of the redundant actuation signals for a protective function. It also includes the power source and actuation signals.

Channel – One of the several separate and redundant measurements of a single variable used by the protection and safety monitoring system in generating the signal to initiate a protective function. A channel can lose its identity when it is combined with other inputs in a division.

Degree of Redundancy – The number of redundant channels monitoring a single variable, or the number of redundant divisions which can initiate a given protective function or accomplish a given protective function. Redundancy is used to maintain protection capability when the safety-related system is degraded by a single random failure.

System-Level Actuation – Actuation of a sufficient number of actuation devices to effect a protective function.

Component-Level Actuation – Actuation of a single actuation device (component).

7.1.1 The AP1000 Instrumentation and Control Architecture

Figure 7.1-1 illustrates the instrumentation and control architecture for the AP1000. The figure shows two major sections separated by the real-time data network. Figure 7.1-1 depicts the real-time data highway as a single network.

The lower portion of the figure includes the plant protection, control, and monitoring functions. At the lower right-hand side is the protection and safety monitoring system. It performs the reactor trip functions, the engineered safety features (ESF) actuation functions, and the Qualified Data Processing (QDPS) functions. The I&C equipment performing reactor trip and ESF actuation functions, their related sensors, and the reactor trip switchgear are, for the most part, four-way redundant. This redundancy permits the use of bypass logic so that a division or individual channel out of service can be accommodated by the operating portions of the protection system.

The ESF coincidence logic performs system-level logic calculations, such as initiation of the passive residual heat removal system. It receives inputs from the plant protection subsystem bistables and the main control room.

The ESF actuation subsystems provide the capability for on-off control of individual safety-related plant loads. They receive inputs from the ESF coincidence logic, remote shutdown workstation and the main control room.

The plant control system performs nonsafety-related instrumentation and control functions using both discrete (on/off) and modulating (analog) type actuation devices.

The nonsafety-related real-time data network, which horizontally divides Figure 7.1-1, is a high speed, redundant communications network that links systems of importance to the operator. Safety-related systems are connected to the network through gateways and qualified isolation devices so that the safety-related functions are not compromised by failures elsewhere. Plant

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protection, control, and monitoring systems feed real-time data into the network for use by the control room and the data display and processing system.

The upper portion of the figure depicts the control rooms and data display and processing system. The main control room is implemented as a set of compact operator consoles featuring color graphic displays and soft control input devices. The graphics are supported by a set of graphics workstations that take their input from the real-time data network. An advanced alarm system, implemented in a similar technology, is also provided.

The data display and processing system is implemented in a distributed architecture. The working elements of the distributed computer system are graphics workstations, although their graphics capability is secondary to their computing performance. The distributed computer system obtains its input from the real-time data network and delivers its output over the network to other users.

WCAP-15775 (Reference 7), Sections 1 through 5, describes the diversity and defense-in-depth features of the AP1000 instrumentation and control architecture.

Protection and Safety Monitoring System

The protection and safety monitoring system provides detection of off-nominal conditions and actuation of appropriate safety-related functions necessary to achieve and maintain the plant in a safe shutdown condition. The protection and safety monitoring system controls safety-related components in the plant that are operated from the main control room or remote shutdown workstation. Secure development and operational environments for the protection and safety monitoring system are used during design as described in Section 2 of APP-GW-J0R-012 (Reference 22).

In addition, the protection and safety monitoring system provides the equipment necessary to monitor the plant safety-related functions during and following an accident as required by Regulatory Guide 1.97.

Special Monitoring System

The special monitoring system does not perform any safety-related or defense-in-depth functions. The special monitoring system consists of specialized subsystems that interface with the instrumentation and control architecture to provide diagnostic and long-term monitoring functions.

The digital metal impact monitoring system is the subsystem of the special monitoring system that detects the presence of metallic debris in the reactor coolant system when the debris impacts against the internal parts of the reactor coolant system. The digital metal impact monitoring system is composed of digital circuit boards, indicators, power supplies and remotely located sensors and related signal processing devices. A minimum of two sensors are located at each natural collection region, connected to separate instrumentation channels, to maintain the impact monitoring function if a sensor fails in service. The digital metal impact monitoring system is described in Subsection 4.4.6.4.

Plant Control System

The plant control system provides the functions necessary for normal operation of the plant from cold shutdown through full power. The plant control system controls nonsafety-related components in the plant that are operated from the main control room or remote shutdown workstation.

The plant control system contains nonsafety-related control and instrumentation equipment to change reactor power, control pressurizer pressure and level, control feedwater flow, and perform other plant functions associated with power generation. The plant control system is described in Subsections 7.1.3 and 7.7.1.

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Diverse Actuation System

The diverse actuation system is a nonsafety-related, diverse system that provides an alternate means of initiating reactor trip and actuating selected engineered safety features, and providing plant information to the operator. The diverse actuation system is described in Subsection 7.7.1.11.

Operation and Control Centers System

The operation and control centers system includes the main control room, the technical support center, the remote shutdown room, emergency operations facility, local control stations and associated workstations for these centers. With the exception of the control console structures, the equipment in the control room is part of the other systems (for example, protection and safety monitoring system, plant control system, data display and processing system).

The boundaries of the operation and control centers system for the main control room and the remote shutdown workstation are the signal interfaces with the plant components. These interfaces are via the plant protection and safety monitoring system processor and logic circuits, which interface with the reactor trip and ESF plant components; the plant control system processor and logic circuits, which interface with the nonsafety-related plant components; and the plant real-time data network, which provides plant parameters, plant component status, and alarms.

Data Display and Processing System

The data display and processing system provides the equipment used for processing data that result in nonsafety-related alarms and displays for both normal and emergency plant operations, generating these displays and alarms, providing analysis of plant data, providing plant data logging and historical storage and retrieval, and providing operational support for plant personnel. DDS also provides the manual actuation switches in the remote shutdown room (RSR).

The data display and processing system also contains the real-time data network, which is a redundant data highway that links the elements of the AP1000 instrumentation and control architecture.

Incore Instrumentation System

The primary function of the incore instrumentation system is to provide a three-dimensional flux map of the reactor core. This map is used to calibrate neutron detectors used by the protection and safety monitoring system, as well as to optimize core performance. A secondary function of the incore instrumentation system is to provide the protection and safety monitoring system with the thermocouple signals necessary for the post-accident inadequate core cooling monitor. The incore instrument assemblies house both fixed incore flux detectors and core exit thermocouples. The incore instrumentation system is described in Subsection 4.4.6.1.

7.1.2 Protection and Safety Monitoring System

Reference 19, Section 2.1 provides an overview description of the protection and safety monitoring system.

7.1.2.1 Plant Protection Subsystems

Reference 19, Section 2.2 describes the plant protection subsystems.

7.1.2.1.1 Reactor Trip Functions

Section 7.2 describes the reactor trip functions.

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7.1.2.1.2 Reactor Trip Switchgear Interface and Logic

Reference 19, Subsection 2.2.3.1.1 describes the reactor trip switchgear interface and logic.

7.1.2.1.3 Manual Reactor Trip

Reference 19, Subsection 2.2.3.1.3 describes the manual reactor trip.

7.1.2.2 Engineered Safety Features Coincidence Logic

Reference 19, Subsection 2.2.3.2.1 describes the Engineered Safety Features Coincidence Logic.

7.1.2.3 Engineered Safety Features Actuation Subsystems

Reference 19, Subsection 2.2.3.2.2 describes the Engineered Safety Features Actuation Subsystems.

7.1.2.4 Reactor Trip Switchgear

Reference 19, Subsection 2.2.3.1.2 describes the reactor trip switchgear, which is also referred to as the reactor trip circuit breaker (RTCB).

7.1.2.5 Qualified Data Processing System

Reference 19, Section 4.2 describes the Qualified Data Processing System (QDPS).

7.1.2.6 Main Control Room Multiplexers

The protection and safety monitoring system does not use multiplexers to provide a signal path between the protection system equipment and the main control room. Each division's safety display communicates with the protection system equipment via that division's communications network as shown in Figure 2-2 of Reference 19.

7.1.2.7 Sensors

The protection and safety monitoring system monitors key variables related to equipment mechanical limitations, and variables directly affecting the heat transfer capability of the reactor. Some limits, such as the overtemperature ΔT setpoint, are calculated in the plant protection subsystem from other parameters because direct measurement of the variable is not possible. This subsection provides a description of the sensors which monitor the variables for the protection and safety monitoring system. For convenience the discussions are grouped into the following three categories:

- Process sensors
- Nuclear instrumentation detectors
- Status inputs from field equipment

The inputs described are those required to generate the initiation signals for the protective functions. The use of each parameter is discussed in the sections that deal with each protective function. For example, reactor trip is discussed in Section 7.2 and ESF actuation is described in Section 7.3.

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7.1.2.7.1 Process Sensors

The process sensors are devices which measure temperature, pressure, fluid flow, and fluid level. Process instrumentation excludes nuclear and radiation measurements.

Additional information on these process variables is included as part of the description of each process system provided in other chapters. The process variables measured by the protection and safety monitoring system are listed in Sections 7.2, 7.3, and 7.5.

7.1.2.7.2 Nuclear Instrumentation Detectors

Three types of neutron detectors are used to monitor the leakage neutron flux from a completely shutdown condition to 120 percent of full power. The intermediate range channels are capable of measuring overpower excursions up to 200 percent of full power.

The lowest range (source range) covers six decades of leakage neutron flux. The lowest observed count rate depends on the strength of the neutron sources in the core and the core multiplication associated with the shutdown reactivity. This generally is greater than two counts per second. The next range (intermediate range) covers eight decades. Detectors and instrumentation are chosen to provide overlap between the higher portion of the source range and the lower portion of the power range. The highest range of instrumentation (power range) covers approximately two decades of the total instrumentation range. This is a linear range that overlaps the higher portion of the intermediate range. The neutron detectors are installed in tubes located around the reactor vessel in the primary shield. Detector types for these three ranges are:

- Source range – proportional counter or pulse fission chamber
- Intermediate range – pulse fission chamber
- Power range – uncompensated ionization chamber

7.1.2.7.3 Equipment Status Inputs

Some inputs to the protection system are not measurements of process or nuclear variables, but are discrete indications of the status of certain equipment. Examples include manual switch positions, contact status inputs, and indications provided by valve limit switches.

7.1.2.8 Communication Functions

Reference 19, Section 3, and Reference 25, Sections 1 through 8, describe the communication functions.

7.1.2.9 Fault Tolerance, Maintenance, Test, and Bypass

Reference 19, Section 7 describes the fault tolerance features, and Section 6 describes the maintenance, test, and bypass features of the protection and safety monitoring system.

7.1.2.10 Isolation Devices

Isolation devices are used to maintain the electrical independence of divisions, and to prevent interaction between nonsafety-related systems and the safety-related system.

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Isolation devices are incorporated into selected interconnections to maintain division independence. Isolation devices serve to prevent credible faults (such as open circuits, short circuits, or applied credible voltages) in one circuit from propagating to another circuit.

7.1.2.11 Test Subsystem

Reference 19, Section 6 describes the test subsystem.

The automatic continuous self-diagnostic features of the PMS and application software provide sufficient assurance of PMS component operability where diagnostics are credited. The faults detected by diagnostics are alarmed and displayed in the Main Control Room. Operators assess operability of the PMS using the plant alarm response and abnormal operating procedures.

7.1.2.12 Safety-Related Display Instrumentation

Safety-related display (PMS safety display subsystem) instrumentation provides the operator with information to determine the effect of automatic and manual actions taken following reactor trip due to a Condition II, III, or IV event as defined in Chapter 15. This instrumentation also provides for operator display of the information necessary to meet Regulatory Guide 1.97. A description of the equipment used to provide this function is provided in Reference 19, Section 4.1. A description of the data provided to the operator by this instrumentation is provided in Section 7.5. A description of the equipment used to provide data to the safety-related display instrumentation is provided in Subsection 7.1.2.5.

In addition to the functions of the PMS and the post-accident monitoring identified above, the PMS provides water level indication for the flooding events as discussed in Sections 3.4 and 11.2. The two sensors used for water level indication provide input into the bistable processor logic (BPL) subsystem and then into the integrated communications processor (ICP) subsystem. The information is then sent to the safety displays. These sensors are categorized as Class 1E and are on the safety-related displays for flood indication, but do not perform any safety-related function.

7.1.2.13 Auxiliary Supporting Systems

The safety-related system equipment is supported by the supply of uninterruptible electrical power. This electrical power is supplied by the Class 1E dc and UPS system discussed in Chapter 8.

7.1.2.14 Verification and Validation

[*Adequacy of the hardware and software is demonstrated for the protection and safety monitoring system through a verification and validation (V&V) program. Details on the verification and validation program for the Common Q portion of the protection and safety monitoring system are provided in WCAP-16096-P-A (Reference 9).*]* WCAP-16096-P-A defines a software development process consistent with appropriate industry standards.

7.1.2.14.1 Design Process

[*WCAP-16096-P-A (Reference 9) provides a planned design process for Common Q software development during life cycle stages:*

- *Conceptual phase (may also be referred to as design requirements phase)*
- *Requirements phase (may also be referred to as system definition phase)*
- *Design phase (may also be referred to as hardware and software development phase)*

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- *Implementation phase (may also be referred to as hardware and software development phase)*
- *Test phase (may also be referred to as system integration and test phase)*
- *Installation and checkout phase (may also be referred to as installation phase)*

WCAP-16096-P-A (Reference 9), WCAP-15927 (Reference 20), and NRC-approved Westinghouse Quality Management System (Reference 21) describe design processes that will be used for the Common Q portion of the AP1000 protection and safety monitoring system.]*

Reference 22 describes how the protection and safety monitoring system meets the control of access requirements of IEEE 603.

The planning (or design requirements) phase documents are listed below. Figure 7.1-2 shows the relationship of the same documents.

Document 1: WNA-PD-00206-WAPP, “AP1000 Engineering Project Execution Program”

Document 2: WNA-PD-00051-WAPP, “AP1000 I&C Programs Plan”

Document 3: WNA-PQ-00283-WAPP, “AP1000 I&C Programs Project Quality Plan”

Document 4: WNA-PD-00283-WAPP, “U.S. AP1000 I&C Protection and Safety Monitoring System”

Document 5: WNA-PD-00042-WAPP, “AP1000 Protection and Safety Monitoring System Software Development Plan”

Document 6: WCAP-16096-P-A, “Software Program Manual for Common Q™ Systems”

Document 7: NABU-DP-00014-GEN, “Design Process for Common Q Safety Systems”

Document 8: WNA-PV-00009-GEN, “Verification & Validation Process for the Common Q Safety Systems”

Document 9: WNA-PT-00058-GEN, “Testing Process for Common Q Safety Systems”

Document 10: NABU-DP-00015-GEN, “Common Q Software Configuration Management Guidelines”

Document 11: 00000-ICE-3889, “Coding Standards & Guidelines for Common Q Systems”

Document 12: APP-PMS-GER-020, “Protection and Safety Monitoring System Concept Phase V&V Summary Report”

Document 13: APP-PMS-T5-001, “AP1000 Protection and Safety Monitoring System Test Plan”

Document 14: WCAP-15927, “Design Process for AP1000 Common Q Safety Systems”

Document 15: APP-GW-J0R-012, “AP1000™ Protection and Safety Monitoring System Computer Security Plan”

Document 16: WCAP-17179 (APP-GW-GLR-143), “AP1000 Component Interface Module Technical Report,” Revision 6

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7.1.2.14.2 Commercial Dedication

[*WCAP-16097-P-A (Reference 8) provides for the use of commercial off-the-shelf hardware and software through a commercial dedication process.*]* Control of the hardware and software during the operational and maintenance phase is the responsibility of the Combined License applicant as described in Subsection 13.5.3.

7.1.2.14.3 Operational Process

The protection and safety monitoring system has a software operations plan or procedure. The software operations plan addresses the secure operating environment of the protection and safety monitoring system. This includes precluding unauthorized changes to hardware, software, and system parameters. The software operations plan requires monitoring to detect penetration, or attempted penetration, of the system. There are also provisions on how to respond to unauthorized access.

The software operations plan includes administrative controls for changing setpoints in the PMS. The controls include declaring the channel associated with the setpoint change inoperable and not relying on its function until declared operable again. The controls for an inoperable channel would then comply with any applicable Technical Specification action. A software operations plan also includes administrative controls to require that the PMS is in the proper configuration and room environmental conditions are acceptable for cabinet access prior to making setpoint changes. The administrative controls prevent the protection and safety monitoring system division room environment from interfering with protection and safety monitoring system equipment when plant personnel are making setpoint changes. The administrative controls account for local emissions from nearby equipment and activities (e.g., welding) while the maintenance test cabinet (MTC) door is open.

7.1.3 Plant Control System

The plant control system is a nonsafety-related system that provides control and coordination of the plant during startup, ascent to power, power operation, and shutdown conditions. The plant control system integrates the automatic and manual control of the reactor, reactor coolant, and various reactor support processes for required normal and off-normal conditions. The plant control system also provides control of the nonsafety-related decay heat removal systems during shutdown. The plant control system accomplishes these functions through use of the following:

- Rod control
- Pressurizer pressure and level control
- Steam generator water level control
- Steam dump (turbine bypass) control
- Rapid power reduction
- PRHR HX Modulation Control

The plant control system provides automatic regulation of reactor and other key system parameters in response to changes in operating limits (load changes). The plant control system acts to maximize margins to plant safety limits and maximize the plant transient performance. The plant control system also provides the capability for manual control of plant systems and equipment. Redundant control logic is used in some applications to increase single-failure tolerance.

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The plant control system includes the equipment from the process sensor input circuitry through to the modulating and nonmodulating control outputs as well as the digital signals to other plant systems. Modulating control devices include valve positioners, pump speed controllers, and the control rod equipment. Nonmodulating devices include motor starters for motor-operated valves and pumps, breakers for heaters, and solenoids for actuation of air-operated valves. The plant control system cabinets contain the process sensor inputs and the modulating and nonmodulating outputs. The plant control system also includes equipment to monitor and control the control rods.

The functions of the plant control system are performed by system assemblies including:

- Distributed controllers
- Signal selector algorithms
- Operator controls and indication
- Real-time data network
- Rod control system
- Rod position indication
- Rod drive motor-generator sets

7.1.3.1 Distributed Controllers

Each distributed controller processes inputs, performs system-level and component-level control calculations, provides capability for an operator interface to the controlled components, transmits control signals to discrete, modulating, and networked interfaced control components, and provides plant status and plant parameter information to the real-time data network.

The distributed controllers receive process inputs and implement the system-level logic and control algorithms appropriate for the plant operating mode. The distributed controllers receive process inputs from, and transmit process control outputs to, the actuated components. The distributed controller also transmits and receives process signals via the real-time data network. The real-time data network also provides for two-way communication between the distributed controllers and between the distributed controllers and the main control room and remote shutdown workstation.

Control functions are distributed across multiple distributed controllers so that single failures within a controller do not degrade the performance of control functions performed by other controllers. The major control functions which are implemented in different distributed controllers include reactor power control, feedwater control, pressurizer control, and turbine control.

7.1.3.2 Signal Selector Algorithms

Signal selector algorithms provide the plant control system with the ability to obtain inputs from the protection and safety monitoring system. The signal selector algorithms select those protection system signals that represent the actual status of the plant and reject erroneous signals. Therefore, the control system does not cause an unsafe control action to occur even if one of four redundant protection channels is degraded by random failure simultaneous with another of the four channels bypassed for test or maintenance.

Each signal selector algorithm receives data from each of the redundant divisions of the protection and safety monitoring system. The data is received from each division through an isolation device.

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The signal selector algorithms provide validated process values to the plant control system. They also provide the validation status; the average, the median, or an individual input value of the valid process values; the number of valid process values; an alarm (if one process value has been rejected); and another alarm (if the algorithm is not able to provide a valid process value).

For the logic values received from the protection and safety monitoring system, such as permissives, two-out-of-four (2/4) voting is used to provide a valid logic value to the plant control system.

The signal selection algorithm is executed in the PLS, and the results are not available to PMS or DAS. Therefore, PMS and DAS performance, controls, and displays are independent of the signal selector algorithm.

7.1.3.3 Operator Controls and Indication

The plant control system operator interface is a set of soft control devices that replace conventional switch/light or potentiometer/meter assemblies used for operator interface with control systems. These soft control devices provide consistent operator interfaces for the plant control system. The soft controls are accessible from the visual display unit (VDU) based operator workstations in the main control room and the remote shutdown room. Soft control devices are used to control safety-related and nonsafety-related equipment.

The implementation of the soft controls is consistent with the following functional requirements:

- The soft control function does not affect the electrical or functional isolation of the safety-related and nonsafety-related equipment. This isolation is maintained upon a single failure of any equipment performing or supporting the soft control function.

- Failure of the operator displays does not prevent an operator from being able to safely shutdown the plant.

Section 18.8 presents the design implementation plan for the human system interfaces, including the soft controls. When the operator desires to operate a component from the nonsafety interface, the graphical operator display, which represents the component status, is selected by the operator on the VDU-based workstation. The operator then selects the component symbol to open the soft control template. The operator then selects the desired control action on the template via placing the cursor over the desired action. After the operator verifies that the desired control action is properly selected, the operator then actuates the control action, causing the selected control action to be transmitted to the component.

7.1.3.4 Real-Time Data Network

The real-time data network is a redundant data highway that supports both periodic and aperiodic data transfers of nonsafety-related signals and data. Periodic transfers consist of process data that is broadcast over the network at fixed intervals and is available to all destinations. Aperiodic data transfer is generally used for messages or file transfers.

The real-time data network provides communications among the distributed controllers, the plant protection and safety monitoring system gateways, the incore instrumentation, and the special monitoring system.

7.1.3.5 Rod Control System

The primary means of regulating the reactor power and power distribution is to position clusters of control rods in the reactor core using the rod control system.

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The control rods are moved into and out of the reactor core by means of electromagnetic jacking mechanisms, called control rod drive mechanisms, located on the reactor vessel head. Each control rod drive mechanism consists of two gripper mechanisms, one stationary and one movable, that hold a notched driveline attached to the upper end of the control rod. The grippers and the lift armature are controlled by coils mounted external to the mechanism, concentric with the rod driveline. By controlling the sequence of energizing these coils, the mechanism can be made to step into, or out of, the reactor in increments. The rod control equipment provides this sequence control.

The control rods are arranged into symmetrical groups. The groups of control rods are divided into two categories: shutdown rods that are normally held fully withdrawn from the reactor, and control rods that are positioned to some intermediate insertion. In addition, there is a subcategory of control rods (low worth gray rods). If a rapid shutdown is necessary, the control, shutdown, and gray rods are dropped into the reactor by de-energizing their drive mechanisms.

Interlocks are provided to prevent the motion of the control rods outside of planned sequences.

7.1.3.6 Rod Position Indication

The position of each control rod is continuously monitored by the rod position indication system. This information is detected by the rod position detector assemblies. The signals from the detectors are processed by the data cabinets and transmitted to the distributed controllers. The distributed controllers further process the rod position information and transmit this information to the real-time data network.

7.1.3.7 Rod Drive Motor-Generator Sets

The rod drive motor-generator sets provide the power to the control rod drive mechanisms through the reactor trip switchgear. The rod drive motor-generator sets are included in the plant control system. The safety-related reactor trip switchgear is included in the plant protection and safety monitoring system.

There are two motor-generator sets with flywheels and two control cabinets. Each motor-generator is a three-phase induction motor, direct-coupled to a flywheel, and a synchronous generator.

During normal operating conditions, both motor generator sets are operating in parallel and equally sharing the total load demand. Each motor-generator set is capable of supplying the entire load requirements when the other set is out of service.

7.1.4 Identification of Safety Criteria

7.1.4.1 Conformance of the Safety System Instrumentation to Applicable Criteria

The safety-related system instrumentation described in Subsection 7.1.1 is designed and built to conform to the applicable criteria, codes, and standards concerned with the safe generation of nuclear power. Applicable General Design Criteria are listed in Section 3.1, NRC Regulatory Guides in Subsection 1.9.1, and Branch Technical Positions in Subsection 1.9.2. Industry Standards are cited as references.

The instrumentation and control portion of the safety-related system meets the requirements of IEEE 603-1991 as discussed in WCAP-15776 (Reference 12) Sections 1 through 5. The references to WCAP-15776 in Section 7.2 and Section 7.3 include Sections 1 through 5. Therefore, only these sections are incorporated by reference. The topics are listed in the same order as they appear in Sections 4 through 8 of IEEE 603-1991. IEEE 603 provides the design bases of the instrumentation

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and control portion of the safety system. Other criteria related to the IEEE 603-1991 requirements are also identified.

7.1.4.2 Conformance With Industry Standards

The instrumentation and control systems are designed in accordance with guidance provided in applicable portions of the following standards. The portions of the standards which are considered to be applicable are the portions of the standards which apply to instrumentation and control systems performing protection and control functions in an industrial environment:

- IEEE 323-1974; “IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations”
- IEEE 344-1987; “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations”
- IEEE 379-2000; “IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems”
- IEEE 383-1974; “IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations”
- IEEE 384-1981; “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits” (See discussion for Regulatory Guide 1.75 in Appendix 1A)
- IEEE 420-1982; “IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations”
- IEEE 603-1991; “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations”
- IEEE 627-1980; “IEEE Standard for Design Qualification of Safety Systems Equipment Used in Nuclear Power Generating Stations”
- IEEE 1050-1996; “IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations”
- IEEE 1074-1995; “IEEE Standard for Developing Software Life Cycle Processes” (See Discussion for Regulatory Guide 1.173 in Appendix 1A)
- Regulatory Guide 1.180, Revision 1, “Guidelines For Evaluating Electromagnetic and Radio-Frequency Interference In Safety-Related Instrumentation And Control Systems”

7.1.5 AP1000 Protective Functions

Protective functions are those necessary to achieve the system responses assumed in the safety analyses, and those needed to shut down the plant safely. The protective functions are grouped into two classes, reactor trip and ESF actuation. The software associated with these functions is considered a basic component as defined in 10 CFR 21 (Reference 6).

Reactor trip is discussed in Section 7.2. ESF actuation is discussed in Section 7.3.

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7.1.6 Combined License Information

7.1.6.1 The calculation of setpoints for protective functions is addressed in WCAP-16361-P (Reference 17), and the applicable changes are incorporated.

The Setpoint Program described in Technical Specifications Section 5.5 provides the appropriate controls for update of the instrumentation setpoints following completion of the calculation of setpoints for protective functions and the reconciliation of the setpoints against the final design.

7.1.6.2 Common Q Platform generic open items and plant specific action items identified during AP1000 design certification are addressed in APP-GW-GLR-017 (Reference 18), and the applicable changes are incorporated.

7.1.7 References

1. IEEE 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations."

2. Not used.

3. Not used.

4. Not used.

5. Not used.

6. 10 CFR 21, "Reporting of Defects and Noncompliance."

7. WCAP-15775, "AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report," Revision 8, May 2017.[1]

8. [*WCAP-16097-P-A (Proprietary) and WCAP-16097-NP-A (Non-Proprietary), Revision 3, "Common Qualified Platform Topical Report," February 2013. (Note: as modified by the Topical Report alternative in WCAP-15927, Revision 8).*]*

9. [*WCAP-16096-P-A (Proprietary) and WCAP-16096-NP-A (Non-Proprietary), Revision 4, "Software Program Manual for Common Q™ Systems," February 2013. (Note: as modified by the Software Program Manual alternatives in WCAP-15927, Revision 8)*]*

10. Not used.

11. Not used.

12. WCAP-15776, "Safety Criteria for the AP1000 Instrument and Control Systems," Revision 1, July 2019.

13. Not used.

14. Not used.

15. Not used.

1. The text incorporates by reference portions or all of this document.

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16. Not used.

17. WCAP-16361-P (Proprietary) and WCAP-16361-NP (Non-Proprietary), “Westinghouse Setpoint Methodology for Protection Systems – AP1000,” February 2011.

18. APP-GW-GLR-017, AP1000 Standard Combined License Technical Report, “Resolution of Common Q NRC Items,” Revision 0, Westinghouse Electric Company LLC.

19. WCAP-16675-P (Proprietary) and WCAP-16675-NP (Non-Proprietary), “AP1000 Protection and Safety Monitoring System Architecture Technical Report,” Revision 10, September 2020.[1]

20. [*WCAP-15927-P (Proprietary) and WCAP-15927-NP (Non-Proprietary), Revision 8, September 2020 “Design Process for AP1000 Common Q Safety Systems.”*]*

21. Westinghouse Electric Company Quality Management System (QMS), Revision 7 (Non-Proprietary), October 1, 2013.

22. APP-GW-J0R-012, “AP1000 Protection and Safety Monitoring System Computer Security Plan,” Revision 5, September 2020, Westinghouse Electric Company LLC.[1]

23. Not used.

24. [*WCAP-17179-P (Proprietary) and WCAP-17179-NP (Non-Proprietary), “AP1000 Component Interface Module Technical Report,” Revision 6, April 2016.*]*

25. WCAP-16674-P (Proprietary) and WCAP-16674-NP (Non-Proprietary), “AP1000 I&C Data Communication and Manual Control of Safety Systems and Components,” Revision 9, June 2019.[1]

1. The text incorporates by reference portions or all of this document.

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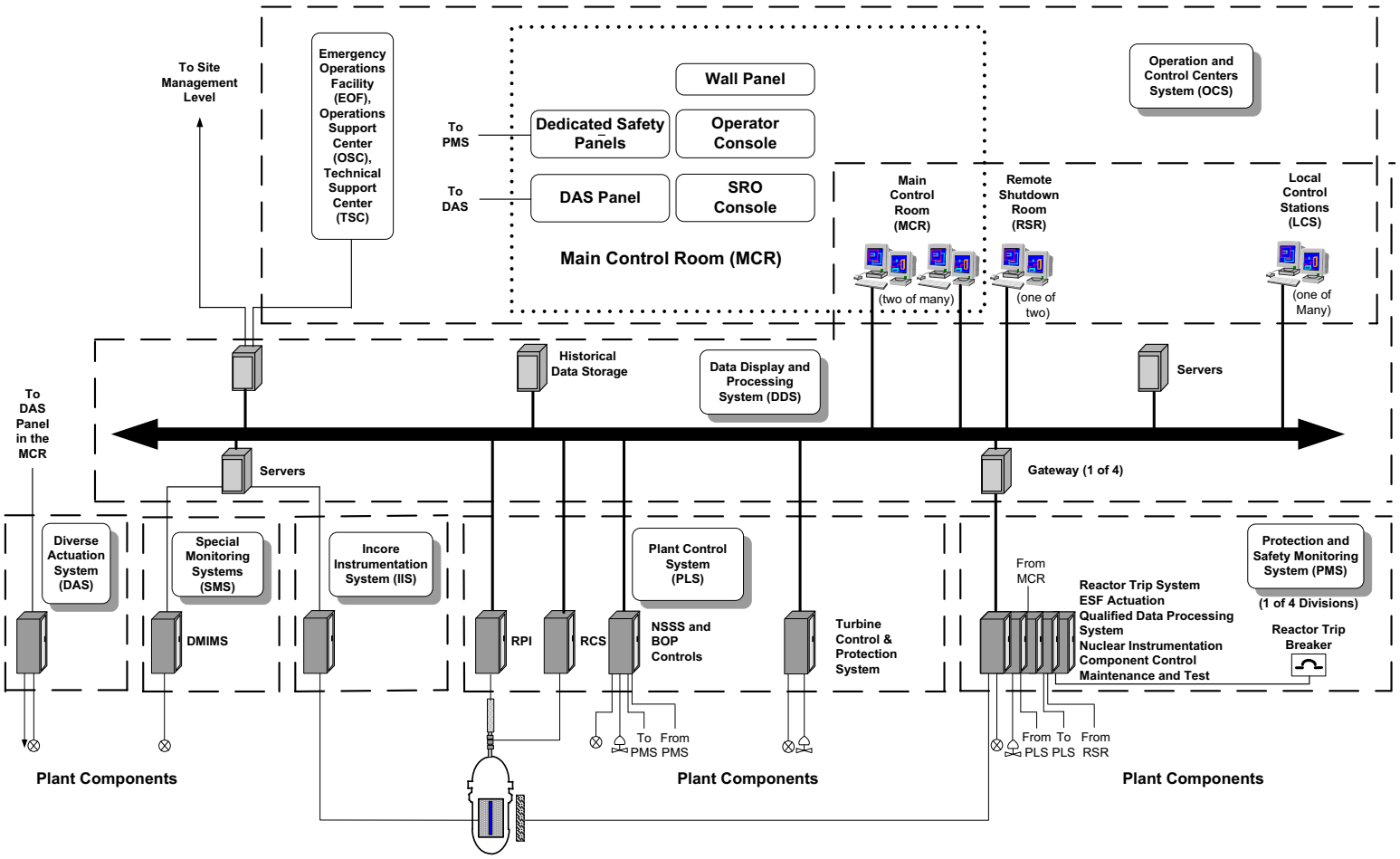

Figure 7.1-1
Instrumentation and Control Architecture

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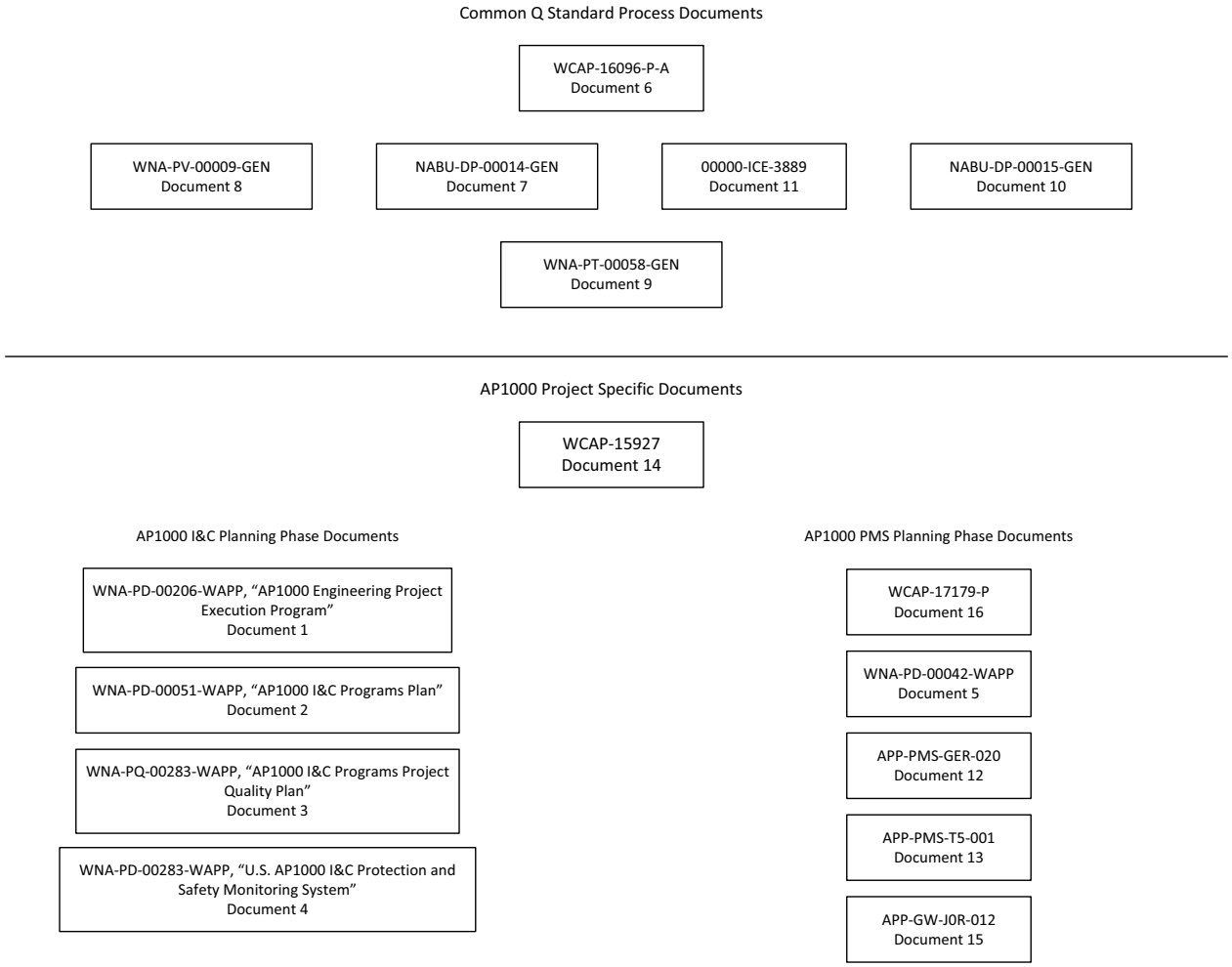

Figure 7.1-2
Common Q Standard Process and AP1000 Project-Specific Documents

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Figures 7.1-3–7.1-11 Not Used

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7.2 Reactor Trip

7.2.1 Description

Considerations, such as mechanical or hydraulic limitations on equipment or heat transfer requirements on the reactor core, define a safe operating region for the plant. Maneuvering of the plant within this safe operating region is permitted in response to normal power generation demands. The plant design provides margin to the safety limits so that an unsafe condition is not caused by the transients induced by normal operating changes. The plant control system attempts to keep the reactor operating away from any safety limit. Excursions toward a limit occur because of abnormal demands, malfunctions in the control system, or by severe transients induced by occurrence of a Condition II or III event, as discussed in Chapter 15. Hypothetical events (Condition IV) are analyzed with respect to plant safety limits. The safety system keeps the reactor within the safe region by shutting down the reactor whenever safety limits are approached. Reactor trip is a protective function performed by the protection and safety monitoring system when it anticipates an approach of a parameter to its safety limit. Reactor shutdown occurs when electrical power is removed from the rod drive mechanism coils, allowing the rods to fall by gravity into the reactor core.

Section 7.1 provides a description of the reactor trip equipment. The equipment involved is:

- Sensors and manual inputs
- Protection and safety monitoring system cabinets
- Reactor trip switchgear

The plant protection subsystems maintain surveillance of key process variables directly related to equipment mechanical limitations (such as pressure), and of variables which directly affect the heat transfer capability of the reactor (such as flow and temperature). Some limits, such as the overtemperature ΔT setpoint, are calculated in the protection and safety monitoring system from other parameters when direct measurement of the variable is not possible. Table 7.2-1 lists variables monitored for reactor trip.

Four redundant measurements, using four separate sensors, are made for each variable used for reactor trip. Analog signals are converted to digital form by analog-to-digital converters within the protection and safety monitoring system. Signal conditioning is applied to selected inputs following the conversion to digital form. Following necessary calculations and processing, the measurements are compared against the applicable setpoint for that variable. Steam generator level, pressurizer level, power range neutron flux, and reactor coolant flow are compensated per Subsection 7.2.1.2.2. A partial trip signal for a parameter is generated if one channel's measurement exceeds its predetermined or calculated limit. Processing of variables for reactor trip is identical in each of the four redundant divisions of the protection system. Each division sends its partial trip status to each of the other three divisions over isolated data links. Each division is capable of generating a reactor trip signal if two or more of the redundant channels of a single variable are in the partial trip state.

The reactor trip signal from each of the four divisions of the protection and safety monitoring system is sent to the corresponding reactor trip switchgear breakers.

Each of the four reactor trip actuation divisions consists of two reactor trip circuit breakers. The reactor is tripped when two or more actuation divisions output a reactor trip signal. This automatic trip demand initiates the following two actions. It deenergizes the under-voltage trip attachments on the reactor trip breakers, and it energizes the shunt trip devices on the reactor trip breakers. Either action causes the breakers to trip. Opening the appropriate trip breakers removes power to the rod drive mechanism coils, allowing the rods to fall into the core. This rapid negative reactivity insertion causes the reactor to shutdown.

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Bypasses of parameter channels used to generate reactor trip signals and of reactor trip actuation divisions are permitted as described in Subsection 7.2.1.1.13. Single failure criterion is met even when one channel or division is bypassed. Bypassing two or more redundant channels or divisions is not allowed.

Subsection 7.2.1.1 provides a description of each of the reactor trip functions. Figure 7.2-1 shows the functional diagrams for reactor trips, as well as functional diagrams for other related plant functions. Figure 7.2-1 sheets are derived from the APP-PMS-J1 drawings and the references shown on Figure 7.2-1 sheets are tied to these parent drawings and not the sheets. Table 7.2-5 provides a cross reference to match the APP-PMS-J1 drawing to its corresponding Figure 7.2-1 sheet.

7.2.1.1 Functional Description

The following subsections describe the specific reactor trip functions and are grouped according to the following 11 conditions:

- Subsection 7.2.1.1.1 Nuclear Startup Trips
- Subsection 7.2.1.1.2 Nuclear Overpower Trips
- Subsection 7.2.1.1.3 Core Heat Removal Trips
- Subsection 7.2.1.1.4 Primary Overpressure Trips
- Subsection 7.2.1.1.5 Loss of Heat Sink Trips
- Subsection 7.2.1.1.6 Feedwater Isolation Trip
- Subsection 7.2.1.1.7 Automatic Depressurization Systems Actuation Reactor Trip
- Subsection 7.2.1.1.8 Core Makeup Tank Injection Trip
- Subsection 7.2.1.1.9 Reactor Trip on Passive Residual Heat Removal Actuation
- Subsection 7.2.1.1.10 Reactor Trip on Safeguards Actuation
- Subsection 7.2.1.1.11 Manual Reactor Trip

Table 7.2-2 lists the reactor trips and summarizes the coincidence logic to trip. Table 7.2-3 provides the interlocks for each trip. Table 7.2-4, lists system level manual inputs to reactor trip functions.

7.2.1.1.1 Nuclear Startup Trips

Source Range High Neutron Flux Trip

Source range high neutron flux trips the reactor when two of the four source range channels exceed the trip setpoint. This trip provides protection during reactor startup and plant shutdown. This function is delayed from actuating each time the source range detector's high voltage power is energized to prevent a spurious trip due to the short term instability of the processed source range values. This trip function may be manually blocked and the high voltage source range detector power supply de-energized when the intermediate range neutron flux is above the P-6 setpoint value. It is automatically blocked and the high voltage source range detector power supply is automatically de-energized by the power range neutron flux interlock above the P-10 setpoint value. The block may be manually reset when neutron flux is between P-6 and P-10. The reset and energization of the high voltage source range detector power supply both occur automatically when the intermediate range flux decreases below P-6. The channels can be individually bypassed to permit channel testing during plant shutdown or prior to startup. This bypass action is indicated in the main control room.

Figure 7.2-1, sheet 3 shows the logic for this trip. This sheet also shows the development of permissive P-6 while P-10 is shown in Figure 7.2-1, sheet 4.

Intermediate Range High Neutron Flux Trip

Intermediate range high neutron flux trips the reactor when two of the four intermediate range channels exceed the trip setpoint. This trip, which provides protection during reactor startup, can be

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manually blocked if the power range channels are above approximately 10-percent power (P-10). The block is automatically reset when the power range channels indicate less than 10-percent power. The intermediate range channels, including detectors, are separate from the power range channels. The intermediate range channels can be individually bypassed to permit channel testing during plant shutdown or prior to startup. This bypass action is indicated in the main control room.

Figure 7.2-1, sheet 3 shows the logic for this trip. The development of permissive P-10 is shown in Figure 7.2-1, sheet 4.

Power Range High Neutron Flux Trip (Low Setpoint)

Power range high neutron flux (low setpoint) trips the reactor when two of the four power range channels exceed the trip setpoint. The flux level is compensated for cold leg density, using cold leg temperature and pressurizer pressure as inputs.

The trip, which provides protection during startup, can be manually blocked when the power range channels are above approximately 10-percent power (P-10). The block is automatically reset when the power range channels indicate less than 10-percent power.

Figure 7.2-1, sheet 3 shows the logic for this trip. The development of permissive P-10 is shown on Figure 7.2-1, sheet 4.

7.2.1.1.2 Nuclear Overpower Trips

Power Range High Neutron Flux Trip (High Setpoint)

Power range high neutron flux (high setpoint) trips the plant when two of the four power range channels exceed the trip setpoint. The flux level is compensated for cold leg density, using cold leg temperature and pressurizer pressure as inputs. It provides a backup trip to the overtemperature ΔT, overpower ΔT, and power range high positive flux rate trips for protection against excessive core power generation during normal operation and is always active. Figure 7.2-1, sheet 4 shows the logic for this trip.

Power Range High Positive Flux Rate Reactor Trip

This trip protects the reactor against a rapid increase in core power generation during normal operation and is always active. The flux level is compensated for cold leg density, using cold leg temperature and pressurizer pressure as inputs. The reactor is tripped when nuclear flux increases rapidly in excess of a predetermined setpoint in two out of the four power range channels.

Figure 7.2-1, sheet 4 shows the logic for this trip.

7.2.1.1.3 Core Heat Removal Trips

Overtemperature ΔT Reactor Trip

The overtemperature ΔT trip provides core protection to prevent departure from nucleate boiling for combinations of pressure, power, coolant temperature, and axial power distribution. The protection is provided if the transient is slow with respect to piping transient delays from the core to the temperature detectors and pressure is within the range between the high and low pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit. If axial peaks are greater than design, as indicated by the difference between upper and lower power range nuclear detectors, the reactor trip limit is automatically reduced. Two hot leg

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temperature measurements per loop are combined with individual cold leg temperature measurements to form four ΔT power signals, $q_{\Delta T}$.

The ΔT power signal, $q_{\Delta T}$, is the calculated core power based on the properties of compressed water at the measured hot leg T_H, cold leg temperature, T_C, and pressurizer pressure, P_{PZR}:

$q_{\Delta T} = f(T_H, T_C, P_{PZR})$

$q_{\Delta T} = \rho(T_C, P_{ZR})[h(T_H, P_{ZR}) - h(T_C, P_{ZR}) - C]/\Delta T°$

Where:

T_C = $(1+\tau_1 s)/[(1+\tau_2 s)(1+\tau_3 s)]T_{COLD}$, where T_{COLD} is the measured cold leg temperature (lead/lag compensation with time constants τ_1, τ_2, and τ_3 applied to compensate for cold leg-to-core transit time)

T_H = $(1+\tau_4 s)/[(1+\tau_5 s)(1+\tau_6 s)]T_{HOT}$, where T_{HOT} is the measured hot leg temperature (lead/lag compensation with time constants τ_4, τ_5, and τ_6 applied to compensate for core-to-hot leg transit time)

$\rho(T_C,P_{PZR})$ = density of water at cold leg temperature in the cold leg (T_C) and pressurizer pressure, P_{PZR}

$h(T,P_{PZR})$ = enthalpy of water at the specified temperature (T_H or T_C) and pressurizer pressure P_{PZR}

ΔT° = a conversion factor, such that the value of $q_{\Delta T}$ is 100 percent at normal rated thermal power

C = a bias coefficient that permits zeroing $q_{\Delta T}$ at zero power (to compensate for small errors in RTD calibration)

s = Laplace transform operator

The ΔT setpoint for the overtemperature trip is continuously calculated, with one set of temperature measurements per loop by interpolating from tabulated core safety limits, with correction, (if needed) for adverse axial power distribution.

$OT\Delta T_{SP} = OT\Delta T^{\circ}_{SP} - f_1(\Delta I)$

Where:

$f_1(\Delta I)$ = the penalty associated with an adverse axial power distribution

$OT\Delta T^{\circ}_{SP}$ = the core DNB thermal design limit with design axial power distribution

$OT\Delta T^{\circ}_{SP}$ = $f(P_{PZR}, TC)$. The function, f(P,T), is determined by interpolation from specified tables of allowable core thermal power as a function of core inlet temperature at various pressures.

P_{PZR} and T_C, pressurizer pressure and cold leg temperature, are as defined previously.

A reactor trip is initiated if $q_{\Delta T} \geq OT\Delta T_{SP}$ in two of the four divisions.

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Two separate ionization chambers supply the upper and lower flux signal for each overtemperature ΔT channel.

Increases in ΔI beyond a predefined deadband results in a decrease in trip setpoint.

The required one pressurizer pressure parameter per loop is obtained from four separate sensors connected to pressure taps at the top of the pressurizer.

Figure 7.2-1, sheet 5, shows the logic for the overtemperature ΔT trip function.

A more detailed description of the Overtemperature ΔT reactor trip is provided in Reference 5.

Overpower ΔT Trip

The Overpower ΔT reactor trip provides confidence of fuel integrity during overpower conditions and limits the required range for overtemperature ΔT protection.

A reactor trip is initiated if the ΔT power signal, $q_{\Delta T}$, exceeds the setpoint in two of the four divisions; that is, if:

$$q_{\Delta T} \geq OP\Delta T_{SP} = C_{OP}^{\circ} - f_2(\Delta I),$$

Where:

$q_{\Delta T}$ is the same ΔT power signal used for the Overtemperature ΔT reactor trip

C_{OP}° = A preset bias

$f_2(\Delta I)$ = A function of the neutron flux difference between upper and lower ionization chamber flux signals; to correct, if necessary, for an adverse axial flux shape.

Increases in ΔI beyond a predefined deadband results in a decrease in trip setpoint.

The source of temperature and neutron flux information is identical to that of the overtemperature ΔT trip, and the resultant ΔT setpoint is compared to the same measured ΔT power signal. Figure 7.2-1, sheet 5, shows the logic for this trip function.

A more detailed description of the Overpower ΔT reactor trip is provided in Reference 5.

Reactor Trip on Low-2 Pressurizer Pressure

This trip protects against low pressure, which could lead to departure from nucleate boiling. The parameter sensed is reactor coolant pressure as measured in the pressurizer. This trip is automatically blocked when reactor power is below the P-10 permissive setpoint to allow control rod testing during cold, depressurized conditions. The block is automatically reset when reactor power is above the P-10 setpoint.

Figure 7.2-1, sheet 5, shows the logic for this trip. The development of the P-10 permissive is shown in Figure 7.2-1, sheet 4.

Reactor Trip on Low-2 Reactor Coolant Flow

This trip protects against departure from nucleate boiling in the event of Low-2 reactor coolant flow. Flow in each hot leg is measured at the hot leg elbow. Two first-order lag filters in series are used to reduce noise on the input signals. Each flow signal is compensated for changes in reactor coolant density using the ΔT power signal. The trip on Low-2 flow in the hot legs is automatically blocked

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when reactor power is below the P-10 permissive setpoint. This enhances reliability by preventing unnecessary reactor trips. The block function is automatically reset when reactor power is above the P-10 setpoint.

Figure 7.2-1, sheet 5 shows the logic for this trip. The development of permissive P-10 is shown in Figure 7.2-1, sheet 4.

Reactor Trip on Low-2 Reactor Coolant Pump Speed

This trip protects the reactor core from departure from nucleate boiling in the event of a loss of flow in more than one loop. This protection is provided by tripping the reactor when the speed on two out of the four reactor coolant pumps falls below the setpoint. Loss of flow in more than one loop could be caused by a voltage or frequency transient in the plant power supply such as would occur during a station blackout. It could be caused by inadvertent opening of more than one reactor coolant pump circuit breaker. There is one speed detector mounted on each reactor coolant pump. The trip is automatically blocked when reactor power is below the P-10 permissive setpoint to enhance reliability by preventing unnecessary reactor trips. The block is automatically reset when reactor power is above the P-10 setpoint.

Figure 7.2-1, sheet 5, shows the logic for this trip. The development of P-10 is shown in Figure 7.2-1, sheet 4.

Reactor coolant pump speed is detected by a probe mounted on the reactor coolant pump frame. The speed signal is transmitted to the protection and safety monitoring system to provide the input to the trip logic function.

The Low-2 reactor coolant pump speed trip provides a direct measurement of the parameter of interest. It permits the plant to ride through many postulated voltage or frequency dip transients without reactor trip if safety limits are not violated. Selection of the Low-2 reactor coolant pump speed trip setpoint and time response provide for the timely initiation of reactor trip during the complete loss of flow accident and the limiting frequency decay event, consistent with the analysis results reported in Chapter 15.

The reactor coolant pump speed sensor response time is not dependent on moving parts and produces a nearly instantaneous response delayed only by the transit time for the flow of electric current in the sensor. The preamplifier also provides a prompt response compared to the overall channel response time and is designed such that there are no components that can contribute to changes in the response time. The response time for the reactor coolant pump speed sensor and preamplifier is allocated in lieu of direct measurement. This is conservative with respect to the response time identified through benchmark testing (Reference 6).

The reactor coolant pump speed detectors perform their protective function (during the complete loss of flow accident and the limiting frequency decay event) in an environment (temperature, humidity, pressure, chemical, and radiation) that is not changed by the event. Therefore, it is not necessary to impose environmental qualification requirements on these detectors more restrictive than those imposed for use under rated conditions. The reactor coolant pump speed detectors are qualified for use under rated conditions with their performance verified by operation in the plant. The reactor coolant pump speed detectors are qualified to the most limiting vibrations experienced by pump operation.

Reactor Trip on High-2 Reactor Coolant Pump Bearing Water Temperature

This trip is an anticipatory trip and is lead-lag compensated based on the expectation of a complete loss of reactor coolant flow if cooling water is lost to any of the reactor coolant pumps. This trip occurs before the reactor coolant pumps are tripped on the same measurement.

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Figure 7.2-1, sheet 5, shows the logic for this trip.

7.2.1.1.4 Primary Overpressure Trips

Pressurizer High-2 Pressure Reactor Trip

This trip protects the reactor coolant system against system overpressure. The same sensors used for the pressurizer Low-2 pressure reactor trip are used for the High-2 pressure trip except that separate setpoints are used. The High-2 pressurizer pressure protection trips the reactor when two out of the four pressurizer pressure channels exceed the trip setpoint. There are no interlocks or permissives associated with this trip function.

Figure 7.2-1, sheet 6, shows the logic for this trip.

High-3 Pressurizer Water Level Reactor Trip

This trip is provided as backup to the High-2 pressurizer pressure reactor trip and serves to prevent water relief through the pressurizer safety valves. The high-3 pressurizer water level protection trips the reactor when two out of the four pressurizer water level channels exceed the trip setpoint. The level signal is compensated for pressurizer reference leg temperature, pressurizer pressure, and RCS wide range pressure. The trip is automatically blocked when reactor power is below the P-10 permissive setpoint. This permits control rod testing with the plant cold and the pressurizer water solid. The block is automatically reset when reactor power is above the P-10 setpoint.

Figure 7.2-1, sheet 6, shows the logic for the trip. The development of P-10 is shown in Figure 7.2-1, sheet 4.

7.2.1.1.5 Loss of Heat Sink Trip

Reactor Trip on Low-2 Steam Generator Narrow Range Water Level in any Steam Generator

This trip protects the reactor from loss of heat sink in the event of a loss of feedwater to the steam generators. The reactor is tripped when two out of the four narrow range water level sensors in any steam generator produce signals below the setpoint value. The level signal is compensated for steam line pressure. This trip can be manually blocked when the reactor coolant average temperature is below the P-9 permissive setpoint. This block is automatically reset when reactor coolant average temperature is above the P-9 permissive setpoint.

Figure 7.2-1, sheet 7, shows the logic for the trip. The development of the P-9 permissive is shown on Figure 7.2-1, sheet 7.

7.2.1.1.6 Feedwater Isolation Trip

High-3 Steam Generator Water Level in Any Steam Generator

This function is an anticipatory trip based on the expectation that a reactor trip would occur after steam generator feedwater is isolated. The plant control system uses a lower steam generator water level setpoint, High-1, to close the feedwater control valves. This provides an interval for operator action to prevent total isolation of the steam generator and a reactor trip before the High-3 setpoint is exceeded. The level signal is compensated for steam line pressure. The trip on High-3 steam generator water level may be manually blocked below the P-11 permissive setpoint to allow control rod testing. The block is automatically reset when the pressurizer pressure is above the P-11 setpoint.

Figure 7.2-1, sheet 10, shows the logic for this trip function.

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7.2.1.1.7 Automatic Depressurization Systems Actuation Reactor Trip

A reactor trip is initiated if an automatic depressurization system actuation occurs either automatically or manually. This provides a reactor trip if the system is depressurized and a trip is not initiated from another source. The automatic depressurization system actuation function is discussed in Subsection 7.3.1.2.4.

Manual automatic depressurization system actuation is initiated from either of two sets of controls in the main control room. Operating either of the two sets of controls also sends a reactor trip signal to the reactor trip switchgear breakers. Outputs on the control sets, physically and electrically separated, send their position status to the protection and safety monitoring system. These outputs de-energize the interposing relays and the undervoltage trip attachments and energize the shunt trip attachments on the reactor trip breakers. Either action causes the breakers to open. The shunt trip attachment provides a backup to the undervoltage trip of the breakers.

Figure 7.2-1, sheet 15 shows the logic for this trip function. There are no interlocks or bypasses associated with this trip.

7.2.1.1.8 Core Makeup Tank Injection Trip

A reactor trip is initiated if core makeup injection occurs either automatically or manually. Since core makeup tank injection results in a trip of the reactor coolant pumps, providing a reactor trip upon core makeup tank injection maximizes the margin to DNB at all power levels. The core makeup tank injection function is discussed in Subsection 7.3.1.2.3.

Manual core makeup tank injection is initiated from either of two controls in the main control room. Operating either of the two controls also sends a reactor trip signal to the reactor trip switchgear breakers. Outputs on each control, physically and electrically separated, send their position status to the protection and safety monitoring system. These outputs de-energize the interposing relays and the undervoltage trip attachments and energize the shunt trip attachments on the reactor trip breakers. Either action causes the breakers to open. The shunt trip attachment provides a backup to the undervoltage trip of the breakers.

Figure 7.2-1, sheets 2 and 12 show the logic for this trip function. There are no interlocks or bypasses associated with this trip.

7.2.1.1.9 Reactor Trip on Passive Residual Heat Removal System Actuation

A reactor trip is initiated when either of the passive residual heat removal (PRHR) system's discharge valves is not closed, allowing flow through the PRHR heat exchanger (PRHR HX). This reactor trip will prevent the fuel design limits from being exceeded in the event the PRHR HX is inadvertently aligned to allow flow when the reactor is being operated at power conditions. The PRHR HX alignment is discussed in Subsection 7.3.1.2.7.

Figure 7.2-1, sheets 2 and 4 show the logic for this trip function. There are no interlocks or bypasses associated with this trip.

7.2.1.1.10 Reactor Trip on Safeguards Actuation

A reactor trip is initiated with any signal that causes a safeguards actuation. This reactor trip occurs whether the safeguards actuation is commanded automatically or manually. The means for actuating safeguards automatically are described in Section 7.3. This trip protects the core against a loss of reactor coolant or a steam line rupture.

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Manual safeguards actuation is initiated from either of two controls in the main control room. Operating either of the two controls also sends a reactor trip signal to the reactor trip switchgear breakers. Outputs on each control, physically and electrically separated, send their position status to the protection and safety monitoring system. These outputs de-energize the interposing relays and the undervoltage trip attachments and energize the shunt trip attachments on the reactor trip breakers. Either action causes the breakers to open. The shunt trip attachment provides a backup to the undervoltage trip of the breakers.

Figure 7.2-1, sheets 2 and 11, show the logic for this trip function. There are no interlocks or bypasses associated with this trip.

7.2.1.1.11 Manual Reactor Trip

The manual reactor trip consists of 2 controls in the main control room, either of which trip all 8 of the reactor trip switchgear breakers. The reactor trip circuit breakers contain both undervoltage and shunt trip attachments. The shunt trip acts as a diverse backup to the undervoltage trip in the breakers. Contacts on each control, physically and electrically separated, are in series with the undervoltage trip attachment on the reactor trip breakers, the shunt trip attachment, and the power outputs at the protection and safety monitoring system cabinet. Actuating either control interrupts power from the voting logic to the interposing relays and undervoltage trip attachments, releasing them. It also supplies power to shunt trip attachments, actuating the shunt trip attachments. The breakers trip when either the shunt trip attachments are energized or the undervoltage trip attachments are de-energized. Actuating either manual trip control causes each breaker to trip by initiating both of these actions.

Figure 7.2-1, sheets 2 and 13, show the logic for the manual trip. There are no interlocks or bypasses associated with this trip.

7.2.1.1.12 Reactor Trip System Interlocks

The interlocks used in the reactor trip functions are designated as P-xx permissives. Table 7.2-3 provides a listing of these interlocks. These permissives are implemented at the channel level rather than at the logic level because plant availability has been determined to be improved using this technique of integrating permissives into each channel.

Manual blocks to reactor trip are listed on Table 7.2-4 and are described in the following subsections. The source, intermediate, power (low setpoint), and steam generator water level manual blocks, when used in conjunction with the applicable permissives, are implemented during startup.

Source Range Block (One Control for each Division)

The source range reactor trip may be manually blocked upon the occurrence of the P-6 permissive and the block is automatically reset when the permissive condition is not met. The channel is automatically blocked upon the occurrence of the P-10 permissive. The block of the trip function also de-energizes the source range neutron flux detectors. Figure 7.2-1, sheet 3, shows these blocks.

Intermediate Range Block (One Control for each Division)

The intermediate range reactor trip may be manually blocked upon the occurrence of the P-10 permissive and the block is automatically reset when the permissive condition is not met. Figure 7.2-1, sheet 3, shows this block.

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Power Range (Low Setpoint) Block (One Control for each Division)

The power range low setpoint reactor trip may be manually blocked upon the occurrence of the P-10 permissive and the block is automatically reset when the permissive condition is not met. Figure 7.2-1, sheet 3, shows this block.

Steam Generator Low-2 Water Level Block (One Control for each Division)

The steam generator Low-2 reactor trip may be manually blocked upon the occurrence of the P-9 permissive and the block is automatically reset when the permissive condition is not met.

Steam Generator High-3 Water Level Block (One Control for each Division)

The steam generator High-3 reactor trip may be manually blocked upon the occurrence of the P-11 permissive. This block function is automatically reset when the permissive condition is not met. Figure 7.2-1, sheets 9, 10, and 11, illustrates the functional logic relating to this function.

7.2.1.1.13 Bypasses of Reactor Trip Functions

Each channel used in reactor trip can be bypassed, as discussed in Subsection 7.1.2.9, except for reactor trips resulting from manual initiations. One channel can be bypassed for an indefinite period of time with the normal two-out-of-four trip logic automatically reverting to a two-out-of-three trip logic. Bypassing two or more channels is not allowed.

7.2.1.2 Design Basis for Reactor Trips

This section provides the design bases information on the reactor trip function, including the information required by Section 4 of IEEE-603-1991. Reactor trip is a protective function generated as part of the protection and safety monitoring system. Those design bases relating to the equipment that initiates and accomplishes reactor trips are contained in WCAP-15776 (Reference 2). The design bases presented here concern the variables monitored for reactor trips, the minimum performance requirements in generating the trips, and the requirements placed on reactor trips during various reactor operating modes.

7.2.1.2.1 Design Basis: Generating Station Conditions Requiring Reactor Trip (Paragraph 4.1 of IEEE-603-1991)

The generating station conditions requiring protective actions are analyzed in Chapter 15. Conditions that result in a reactor trip are listed in Table 15.0-6. This table correlates the accident conditions (II, III, or IV events) to each reactor trip.

7.2.1.2.2 Design Basis: Variables, Levels, Ranges, and Accuracies Used in Reactor Trip Functions (Paragraphs 4.1, 4.2, and 4.4 of IEEE-603-1991)

The variables monitored for reactor trips are:

- Neutron flux (with cold leg temperature and pressurizer pressure compensation)
- Reactor coolant pump bearing water temperature
- Pressurizer pressure
- Water level in the pressurizer (with pressurizer reference leg temperature, pressurizer pressure, and RCS wide range pressure compensation)
- Reactor coolant flow in each loop (with thermal power compensation)
- Speed of each reactor coolant pump
- Water level in each steam generator (with steam line pressure compensation)
- Cold leg temperature (T_{cold}) in each loop
- Hot leg temperature (T_{hot}) in each loop

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- Status of each manual reactor trip control
- Status of PRHR HX discharge valves

The ranges and response times for each variable are listed on Table 7.2-1. Channel accuracies are accounted for in the setpoint methodology. Some PMS response times are allocated rather than measured (Reference 6).

A discussion on levels that require reactor trip is contained in Subsection 7.2.1.1.

The reactor trip setpoints are maintained by the setpoint program, which is described in the Technical Specifications.

7.2.1.2.3 Design Basis: Spatially Dependent Parameters Used in Reactor Trip (Paragraph 4.6 of IEEE-603-1991)

The hot and cold leg temperature signals required for input to the protection and control functions are obtained using thermowell-mounted RTDs installed in each reactor coolant loop. The hot leg temperature measurement in each loop is accomplished using six fast-response, narrow-range RTDs each in its own thermowell; three thermowells are RTDs for each of the two divisions monitoring that hot leg. The three thermowells for each division are mounted approximately 120 degrees apart in the cross-sectional plane of the piping, to obtain a representative temperature sample. The temperatures measured by the three RTDs are different due to hot leg temperature streaming and vary as a function of thermal power. Therefore, these signals are averaged using electronic weighting to generate a hot leg average temperature. Provisions are incorporated into the process electronics to allow for operation with only two RTDs in service. The two RTD measurements can be biased to compensate for the loss of the third RTD.

Radially varying cold leg temperature is not a concern since the resistance temperature detectors are located downstream of the reactor coolant pumps. The pumps provide mixing of the coolant so that radial temperature variations do not exist.

Radial neutron flux is not a spatially dependent concern because of core radial symmetry. Axial variation in neutron flux is used for calculations involving overtemperature and overpower ΔT. Excore detectors furnish this axially-dependent information to the overtemperature and overpower calculators. See Subsection 7.2.1.1.3.

7.2.1.2.4 Design Basis: Operational Limits for Variables in Various Reactor Operating Modes (Paragraph 4.3 of IEEE-603-1991)

During startup or shutdown, reactor trips are provided for three ranges of neutron flux (source, intermediate, and power range). The source range, intermediate range, and power range (low setpoint) trips are manually blocked when the appropriate power escalation permissives are present. The blocks are automatically reset during power de-escalation. Subsection 7.2.1.1.1 describes these reactor trips. Their interlocks are described in Subsection 7.2.1.1.12.

During testing or maintenance, functions are provided to bypass a channel monitoring a variable for reactor trip. Although no setpoints need to be changed for bypassing, the coincidence logic is automatically adjusted as described in Subsection 7.2.1.1.13. The logic provides that the remaining redundant channels for that variable meet the single failure criterion. The two-out-of-four logic is automatically reinstated when the bypass is removed.

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7.2.1.2.5 Design Basis: Reactor Trips for Malfunctions, Accidents, Natural Phenomena, or Credible Events (Paragraph 4.7 and 4.8 of IEEE-603-1991)

There are no reactor trip functions that directly shutdown the reactor on occurrence of either natural phenomena (such as seismic flood or wind) or internal events (such as fire or pipe whip). The operator can trip the reactor at any time by actuating the manual reactor trip.

Functional diversity is used to determine the reactor trips for accident conditions. Generally, two or more reactor trips occur for the transients analyzed in the accident analyses.

For example, protection is provided for the complete loss of coolant flow event by Low-2 reactor coolant pump speed and by Low-2 coolant flow reactor trips. Complete reliance is not made on a single reactor trip terminating a given event. Table 15.0-6 lists the reactor trips and the conditions which normally result in each trip.

Redundancy provides confidence that reactor trips are generated on demand, even when the protection system is degraded by a single failure. Reactor trips are four-way redundant. The single failure criterion is met even if one channel is bypassed, as discussed in Subsection 7.2.1.1.13. More than one bypass is not allowed.

7.2.1.3 System Drawings

Functional diagrams of the reactor trip function are provided in Figure 7.2-1.

7.2.2 Analyses

7.2.2.1 Failure Modes and Effects Analysis (FMEA)

A failure modes and effects analysis was performed on the AP1000 protection and safety monitoring system. Through the process of examining the feasible failure modes, it was concluded that the AP1000 protection system maintains safety functions during single point failures. The AP1000 failure modes and effects analysis is documented in Appendix 7B.

7.2.2.2 Conformance of the Reactor Trip Function to Applicable Criteria

Reactor trip is a protective function generated by the AP1000 protection and safety monitoring system. Requirements addressing equipment in the protection and safety monitoring system are presented in WCAP-15776 (Reference 2). The discussions presented in this subsection address only the functional aspects of reactor trip.

7.2.2.2.1 Conformance to the General Functional Requirements for Reactor Trip (Section 5 of IEEE-603-1991, GDC-13, GDC-20)

The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level. The reactor trips are listed in Table 7.2-2 and are discussed in Subsection 7.2.1.1. The variables which are monitored for these trips are listed in Subsection 7.2.1.2.2. Table 7.2-1 lists the ranges and response times for these variables. Channel accuracies are accounted for in the setpoint methodology. The reactor trip setpoints are maintained by the setpoint program, which is described in the Technical Specifications.

As discussed in WCAP-15776 (Reference 2), the setpoints set into the protection and safety monitoring system equipment provide a margin to the safety limits which are assumed in the accident analyses. The safety limits are based on mechanical or hydraulic limitations of equipment or on heat transfer characteristics of the reactor core. While most setpoints used for reactor trip are fixed, there

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are continuously calculated setpoints for the overtemperature and overpower ΔT trips. Setpoints for reactor trip are selected on the basis of engineering design and safety studies. The setpoints provide a margin to allow for uncertainties and instrument errors.

The overtemperature and overpower conditions are not directly measurable quantities. However, the process variables that determine overtemperature and overpower conditions are sensed and evaluated. Small isolated changes in various process variables may not individually result in reaching a core safety limit. However, the combined variations over time may cause the overtemperature or overpower limit to be exceeded. The design concept for reactor trips takes cognizance of this situation by providing reactor trips associated with individual process variables in addition to the overtemperature and overpower ΔT safety limit trips. Process variable trips prevent reactor operation when a monitored value reaches a core or safety limit. Overtemperature and overpower ΔT trips provide protection for slow transients. Other trips, such as low flow or high flux, trip the reactor for rapid changes in flow or flux respectively.

Table 15.0-6 summarizes events which normally result in reactor trips.

7.2.2.2.2 Conformance to the Single Failure Criterion for Reactor Trip (Paragraph 5.1 of IEEE 603-1991, IEEE 379-2000)

A single failure in the protection and safety monitoring system or the reactor trip actuation divisions does not prevent a reactor trip, even when a reactor trip channel is bypassed for test or maintenance. Conformance of the equipment to this requirement is discussed in WCAP-15776 (Reference 2). In addition to the redundancy of equipment, diversity of reactor trip functions is incorporated. Most Condition II, III, or IV events requiring a reactor trip are protected by trips from diverse parameters. For example, reactor trip, because of an uncontrolled rod cluster control assembly bank withdrawal at power, may occur on power range high neutron flux, overtemperature, overpower, pressurizer High-2 pressure or pressurizer High-3 water level. Reactor trip on complete loss of reactor coolant flow may occur on Low-2 flow or from the diverse parameter of Low-2 reactor coolant pump speed.

7.2.2.2.3 Conformance to the Requirements Covering Control and Reactor Trip Interactions (Paragraphs 5.6 and 6.3 of IEEE 603-1991, GDC-24)

The AP1000 is designed to permit maneuvering of the plant in response to normal power generation demands without causing a reactor trip. The plant control system attempts to keep the reactor operating away from any safety limit. However, the selection of the reactor trip setpoints does not take credit for such control actions. The accident analyses in Chapter 15 assume that the plant is at normal operation commensurate with the operating mode at the onset of the accident. If a control system action leads to more conservative results, that assumption is made. If failure of a control system to work leads to more conservative results, that assumption is made. In this way, reactor trips do not depend on control system actions.

As stated in Subsection 7.7.1.12, it is considered advantageous to use certain protection data for control functions. Isolation devices are incorporated into the protection system to prevent control system failures from degrading the performance of the protection system.

Failures in a protection channel monitoring a variable that is also used for control do not result in control system actions requiring protection by the redundant channels monitoring that variable. This is discussed in WCAP-15776 (Reference 2).

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7.2.2.2.4 Conformance to Requirements on the Derivation of System Inputs for Reactor Trip (Paragraph 6.4 of IEEE 603-1991)

To the extent feasible, inputs used for reactor trip are derived from signals that are direct measurements of the desired variables. Two exceptions exist, overtemperature and overpower, which cannot be directly measured. The process variables that do affect these parameters can be measured and they are used to continuously calculate the setpoints.

The overtemperature ΔT trip setpoint is calculated from pressurizer pressure, reactor coolant temperature, and nuclear axial power shape. The setpoint is compared against the measured ΔT power signal.

Overpower ΔT is calculated from reactor coolant temperature and the nuclear axial power shape in the core. This value is compared against the measured ΔT power signal.

The overtemperature and overpower ΔT trips are described in Subsection 7.2.1.1.3.

7.2.2.2.5 Conformance to Requirements on Bypassing of Reactor Trip Functions (Paragraph 5.8, 5.9, 6.6, and 6.7 of IEEE 603-1991)

With the exception of the manual reactor trips, reactor trip channels and the reactor trip actuation divisions are permitted to be bypassed as described in WCAP-15776 (Reference 2).

Operating bypasses for reactor trips are described in Subsection 7.2.1.1.13.

7.2.2.2.6 Conformance to Requirements on Multiple Setpoints Used for Reactor Trips (Paragraph 6.8.2 of IEEE 603-1991)

For monitoring neutron flux, multiple setpoints are used. When a more restrictive trip setting becomes necessary to provide adequate protection for a particular mode of operation or set of operating conditions, the protection and safety monitoring system hardware and software are designed to provide positive means or administrative control to ensure that the more restrictive trip setpoint is used. The hardware and software used to prevent improper use of less restrictive trip settings are considered part of the protection and safety monitoring system.

7.2.2.2.7 Conformance to the Requirement for Completion of Reactor Trip Once Initiated (Paragraph 5.2 of IEEE 603-1991, Regulatory Guide 1.62)

Once initiated, reactor trips proceed to completion. Return to operation requires deliberate operator action to reset the reactor trip circuit breakers that are opened by the reactor trip signal. The circuit breakers cannot be closed while the reactor trip signals are present from the respective protection and safety monitoring system division. A manual control is provided in the main control room to reset (close) the reactor trip breakers when all reactor trip signals have been cleared. The reset feature is not a safety function. Refer also to WCAP-15776 (Reference 2).

7.2.2.2.8 Conformance to the Requirement to Provide for Manual Initiation of Reactor Trip (Paragraph 6.2 of IEEE 603-1991, Regulatory Guide 1.62)

The reactor is tripped by actuating one of two manual reactor trip controls from the main control room. The reactor is also tripped upon manual actuation of the automatic depressurization system, manual core makeup tank injection, or upon manual safeguards actuation. These reactor trips are described in Subsections 7.2.1.1.7, 7.2.1.1.8, 7.2.1.1.10, and 7.2.1.1.11. Refer also to WCAP-15776 (Reference 2).

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7.2.3 Combined License Information

The FMEA for the protection and safety monitoring system is addressed in WCAP-16438-P (Reference 1) and WCAP-16592-P (Reference 4), and the applicable changes are incorporated.

7.2.4 References

1. WCAP-16438-P (Proprietary), WCAP-16438-NP (Non-Proprietary), “FMEA of AP1000 Protection and Safety Monitoring System,” Revision 3, February 2011.

2. WCAP-15776, “Safety Criteria for the AP1000 Instrument and Control Systems,” Revision 1, July 2019.

3. Not Used.

4. WCAP-16592-P (Proprietary), WCAP-16592-NP (Non-Proprietary), “Software Hazards Analysis of AP1000 Protection and Safety Monitoring System,” Revision 2.

5. APP-GW-GLR-137, “Bases of Digital Overpower and Overtemperature Reactor Trips,” Revision 1.

6. APP-GW-GLR-804, “AP1000® Technical Report to Support Modification to Response Time and Channel Calibration Testing Requirements for Reactor Coolant Pump Speed,” Revision 0, Westinghouse Electric Company, LLC.

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Table 7.2-1 (Sheet 1 of 3)
Reactor Trip Variables, Limits, and Ranges
(Design Basis for Reactor Trip)
(Nominal)

Protective Functions	Variable	Range of Variables	Response Time Assumed in Accident Analysis (Sec)(1)
Source Range High Neutron Flux	Neutron flux	6 decades of neutron flux: 1 to 10^6 counts per second	N/A(2)
Intermediate Range High Neutron Flux	Neutron flux	8 decades of neutron flux overlapping source range by 2 decades and including 100% power	N/A(2)
Power Range Neutron Flux (Low Setpoint)	Neutron flux	1 to 120% of full power	0.9
Power Range Neutron Flux (High-Setpoint)	Neutron flux	1 to 120% of full power	N/A(2)
Power Range High Positive Flux Rate	Neutron flux	1 to 120% of full power	0.9
Overtemperature ΔT	Cold leg temp. (T_{cold})	490° to 610°F	2.0(3)
	Hot leg temp. (T_{hot})	530° to 650°F	2.0(3)
	Pressurizer pressure	1500 to 2500 psig	2.0
	Neutron flux (difference between top and bottom power range detectors)	-60 to +60% (Δφ)	N/A(2)
Overpower ΔT	Cold leg temp. (T_{cold})	490° to 610°F	1.0(3)
	Hot leg temp. (T_{hot})	530° to 650°F	1.0(3)
	Pressurizer pressure	1500 to 2500 psig	1.0
	Neutron flux (difference between top and bottom power range detectors)	-60 to +60% (Δφ)	N/A(2)
Pressurizer Low Pressure	Pressurizer pressure	1500 to 2500 psig	2.0

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Table 7.2-1 (Sheet 2 of 3)
Reactor Trip Variables, Limits, and Ranges
(Design Basis for Reactor Trip)
(Nominal)

Protective Functions	Variable	Range of Variables	Response Time Assumed in Accident Analysis (Sec)(1)
Pressurizer High Pressure	Pressurizer pressure	1500 to 2500 psig	2.0
Pressurizer High Water Level	Pressurizer water level	0-100% of entire cylindrical portion of pressurizer	2.0
Low Reactor Coolant Flow	Coolant flow	0 to 120% of rated flow	1.451
Low Reactor Coolant Pump Speed	Pump speed	0 to 120% of rated speed	0.65
Low Steam Generator Water Level	Steam generator water level	0-100% of span (narrow range taps)	2.0
High Steam Generator Water Level	Steam generator water level	0-100% of span (narrow range taps)	2.0
Reactor Coolant Pump High Bearing Water Temperature	Reactor coolant pump bearing water temperature	50° to 400°F	N/A(2)

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Table 7.2-1 (Sheet 3 of 3)
Reactor Trip Variables, Limits, and Ranges
(Design Basis for Reactor Trip)
(Nominal)

Protective Functions	Variable	Range of Variables	Response Time Assumed in Accident Analysis (Sec)(1)
Automatic or Manual Safeguards Actuation	See Table 7.3-4	See Table 7.3-4	2.0(3)
Manual Reactor Trip	Control status	N/A	N/A
Automatic or Manual Depressurization System Actuation	See Table 7.3-4	See Table 7.3-4	N/A(4)
Automatic or Manual Core Makeup Tank Injection	See Table 7.3-4	See Table 7.3-4	N/A(4)
Automatic or Manual PRHR Actuation	PRHR discharge valve position	Valve closed/valve not-closed	1.25

Notes:

1. Delay from the time that the process variable exceeds the setpoint until the time that the control rods are free to fall into the core (includes reactor trip breaker opening delay and control rod drive mechanism gripper release delay). This response time reflects the total delay from both the PMS equipment and the associated sensor.
2. The following functions are not explicitly assumed to initiate a reactor trip in the accident analysis. However, they are subject to response time test surveillance requirements in the technical specifications. Therefore, the following maximum allowable response times are allocated:

   Source Range High Neutron Flux = 2.0 seconds
   Intermediate Range High Neutron Flux = 2.0 seconds
   Power Range Neutron Flux (High-Setpoint) = 2.0 seconds
   Neutron flux (difference between top and bottom power range detectors) for Overtemperature ΔT = 2.0 seconds
   Neutron flux (difference between top and bottom power range detectors) for Overpower ΔT = 2.0 seconds
   Reactor Coolant Pump High Bearing Water Temperature = 8.5 seconds

3. In addition to the response time given, a 4.0 second lag time is used in the accident analysis for this function to account for the associated RTDs. For automatic safeguards actuation, the 4.0 second lag time is only used for the temperature inputs.
4. This function is not explicitly assumed to initiate a reactor trip in the accident analysis and does not have any response time test requirements in the technical specifications.

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Table 7.2-2 (Sheet 1 of 2)
Reactor Trips

Reactor Trip[1]	No. of Channels	Division Trip Logic	Bypass Logic	Permissives and Interlocks (See Table 7.2-3)
Source Range High Neutron Flux Reactor Trip	4	2/4	Yes[2]	P-6, P-10
Intermediate Range High Neutron Flux Reactor Trip	4	2/4	Yes[2]	P-10
Power Range High Neutron Flux (Low Setpoint) Trip	4	2/4	Yes[2]	P-10
Power Range High Neutron Flux (High Setpoint) Trip	4	2/4	Yes[2]	----
High Positive Flux Rate Trip	4	2/4	Yes[2]	----
High-2 Reactor Coolant Pump Bearing Water Temperature	16 (4/pump)	2/4 in any single pump	Yes[2]	----
Overtemperature ΔT	4 (2/loop)	2/4	Yes[2]	----
Overpower ΔT	4 (2/loop)	2/4	Yes[2]	----
Pressurizer Low-2 Pressure Trip	4	2/4	Yes[2]	P-10
Pressurizer High-2 Pressure Trip	4	2/4	Yes[2]	----
High-3 Pressurizer Water Level Trip	4	2/4	Yes[2]	P-10
Low-2 Reactor Coolant Flow	8 (4/hot leg)	2/4 in either hot leg	Yes[2]	P-10
Low-2 Reactor Coolant Pump speed	4 (1/pump)	2/4	Yes[2]	P-10
Low-2 Steam Generator Narrow Range Water Level	4/steam generator	2/4 in any steam generator	Yes[2]	P-9
High-3 Steam Generator Water Level	4/steam generator	2/4 in any steam generator	Yes[2]	P-11

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Table 7.2-2 (Sheet 2 of 2)
Reactor Trips

Reactor Trip[1]	No. of Channels	Division Trip Logic	Bypass Logic	Permissives and Interlocks (See Table 7.2-3)
Automatic Safeguards Actuation	4	2/4	Yes[2]	—
Automatic Depressurization System Actuation	4	2/4	Yes[2]	—
Automatic Core Makeup Tank Injection	4	2/4	Yes[2]	—
PRHR Actuation	4	2/4	Yes[2]	—
Manual Safeguards Actuation	2 controls	1/2 controls	No	—
Manual Depressurization System Actuation	4 controls	2/4 controls	No	—
Manual Core Makeup Tank Injection	2 controls	1/2 controls	No	—
Manual Reactor Trip	2 controls	1/2 controls	No	—

Notes:
1. Reactor Trip divisions are also bypassed with the logic as defined in 2. below.
2. Bypass Logic = 2/4 with no bypasses; 2/3 with 1 bypass; more than one bypass is not allowed.

— No permissive or interlock.

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Table 7.2-3 (Sheet 1 of 2)
Reactor Trip Permissives and Interlocks

Designation	Derivation	Function
P-6	Intermediate range neutron flux above setpoint	Allows manual block of source range reactor trip
$\overline{\text{P-6}}$	Intermediate range neutron flux below setpoint	Automatically resets source range reactor trip
P-9	Reactor coolant average temperature above setpoint	Prevents the block of Low-2 steam generator narrow range water level reactor trip
$\overline{\text{P-9}}$	Reactor coolant average temperature below setpoint	Allows manual block of Low-2 steam generator narrow range water level reactor trip
P-10	Power range nuclear power above setpoint	(a) Allows manual block of power range (low setpoint) reactor trip (b) Allows manual block of intermediate range reactor trip (c) Automatically blocks source range reactor trip (back-up to P-6) (d) Allows reactor trip on Low-2 coolant flow (e) Allows reactor trip on Low-2 reactor coolant pump speed (f) Allows reactor trip on High-3 pressurizer water level (g) Allows reactor trip on Low-2 pressurizer pressure

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Table 7.2-3 (Sheet 2 of 2)
Reactor Trip Permissives and Interlocks

Designation	Derivation	Function
$\overline{\text{P-10}}$	Power range nuclear power below setpoint	(a) Prevents the block of power range (low setpoint) reactor trip (b) Prevents the block of intermediate range reactor trip (c) Permits manual reset of each source range channel reactor trip (d) Blocks reactor trip on Low-2 coolant flow (e) Blocks reactor trip on Low-2 reactor coolant pump speed (f) Blocks reactor trip on High-3 pressurizer water level (g) Blocks reactor trip on Low-2 pressurizer pressure
P-11	Pressurizer pressure below setpoint	Allows manual block of High-3 steam generator water level reactor trip
$\overline{\text{P-11}}$	Pressurizer pressure above setpoint	Automatically resets High-3 steam generator water level reactor trip

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Table 7.2-4
System-Level Manual Inputs to the Reactor Trip Functions

Manual Control	To Divisions				Figure 7.2-1 Sheet
Manual Reactor Trip Control #1	A	B	C	D	13
Manual Reactor Trip Control #2	A	B	C	D	13
Source Range High Neutron Flux Block, Division A	A				3
Source Range High Neutron Flux Block, Division B		B			3
Source Range High Neutron Flux Block, Division C			C		3
Source Range High Neutron Flux Block, Division D				D	3
Intermediate Range High Neutron Flux Block, Division A	A				3
Intermediate Range High Neutron Flux Block, Division B		B			3
Intermediate Range High Neutron Flux Block, Division C			C		3
Intermediate Range High Neutron Flux Block, Division D				D	3
Power Range High Neutron Flux Block (Low Setpoint), Division A	A				3
Power Range High Neutron Flux Block (Low Setpoint), Division B		B			3
Power Range High Neutron Flux Block (Low Setpoint), Division C			C		3
Power Range High Neutron Flux Block (Low Setpoint), Division D				D	3
Low Steam Generator Level PRHR Actuation Block, Division A	A				7
Low Steam Generator Level PRHR Actuation Block, Division B		B			7
Low Steam Generator Level PRHR Actuation Block, Division C			C		7
Low Steam Generator Level PRHR Actuation Block, Division D				D	7
Manual Safeguards Actuation Control #1	A	B	C	D	11
Manual Safeguards Actuation Control #2	A	B	C	D	11
Manual Core Makeup Tank Injection Control #1	A	B	C	D	12
Manual Core Makeup Tank Injection Control #2	A	B	C	D	12
Manual Depressurization System Stages 1, 2 & 3 Actuation Controls #1 & 2	A	B	C	D	15
Manual Depressurization System Stages 1, 2 & 3 Actuation Controls #3 & 4	A	B	C	D	15
Steam Line/Feedwater Isolation and Safeguards Block, Division A	A				9
Steam Line/Feedwater Isolation and Safeguards Block, Division B		B			9
Steam Line/Feedwater Isolation and Safeguards Block, Division C			C		9
Steam Line/Feedwater Isolation and Safeguards Block, Division D				D	9

Note:
Controls are located in the main control room except as noted on the applicable sheet of Figure 7.2-1.

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Table 7.2-5
Figure 7.2-1 Cross References

APP-PMS-J1 Drawing Number	Figure 7.2-1 Sheet Number
APP-PMS-J1-101	1
APP-PMS-J1-102	2
APP-PMS-J1-103	3
APP-PMS-J1-104	4
APP-PMS-J1-105	5
APP-PMS-J1-106	6
APP-PMS-J1-107	7
APP-PMS-J1-108	8
APP-PMS-J1-109	9
APP-PMS-J1-110	10
APP-PMS-J1-111	11
APP-PMS-J1-112	12
APP-PMS-J1-113	13
APP-PMS-J1-114	14
APP-PMS-J1-115	15
APP-PMS-J1-116	16
Not Used	17
APP-PMS-J1-119	18
APP-PMS-J1-120	19
APP-DAS-J1-102	20
APP-DAS-J1-103	21

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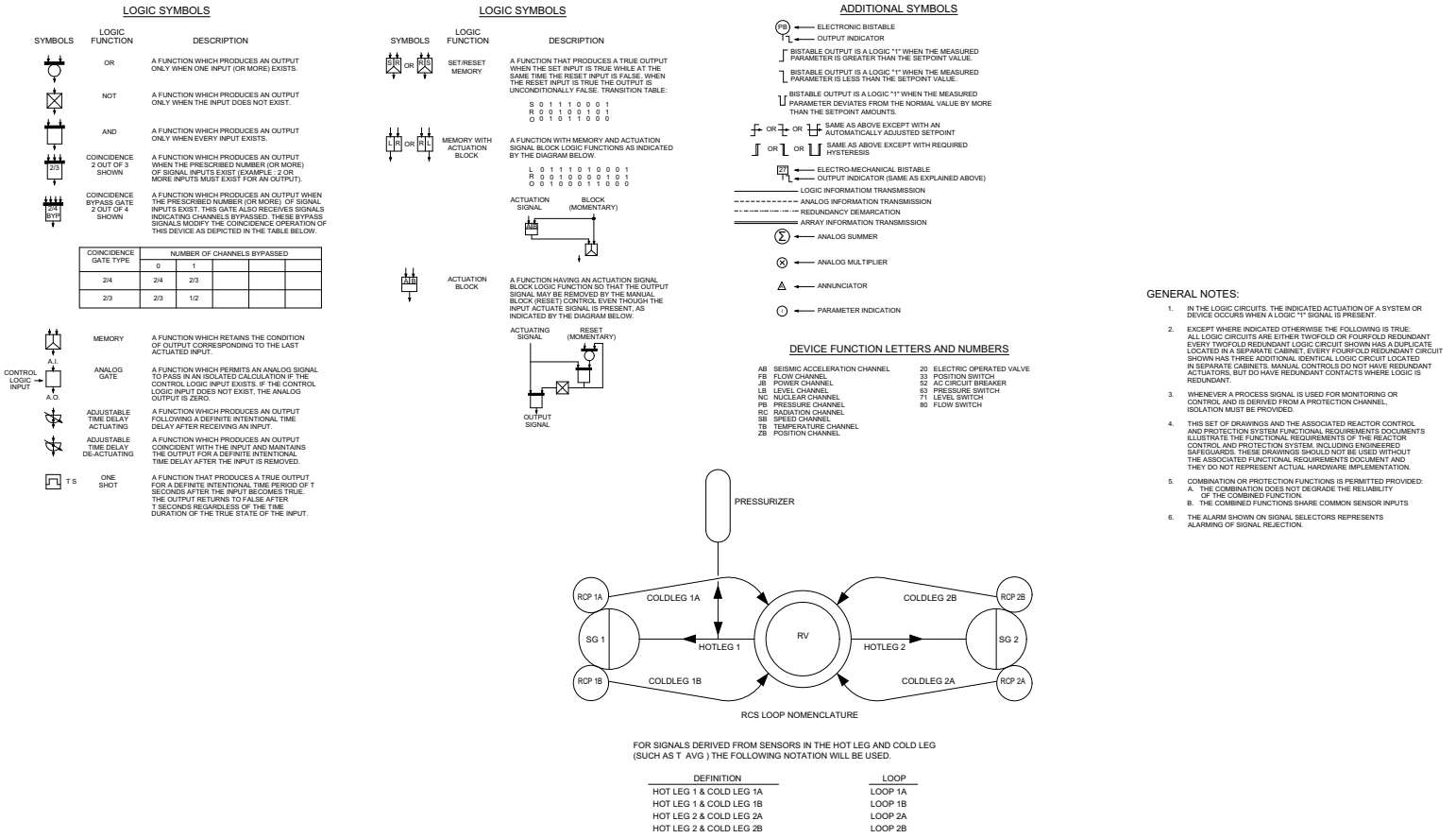

Figure 7.2-1 (Sheet 1 of 21)
Functional Diagram Index and Symbols

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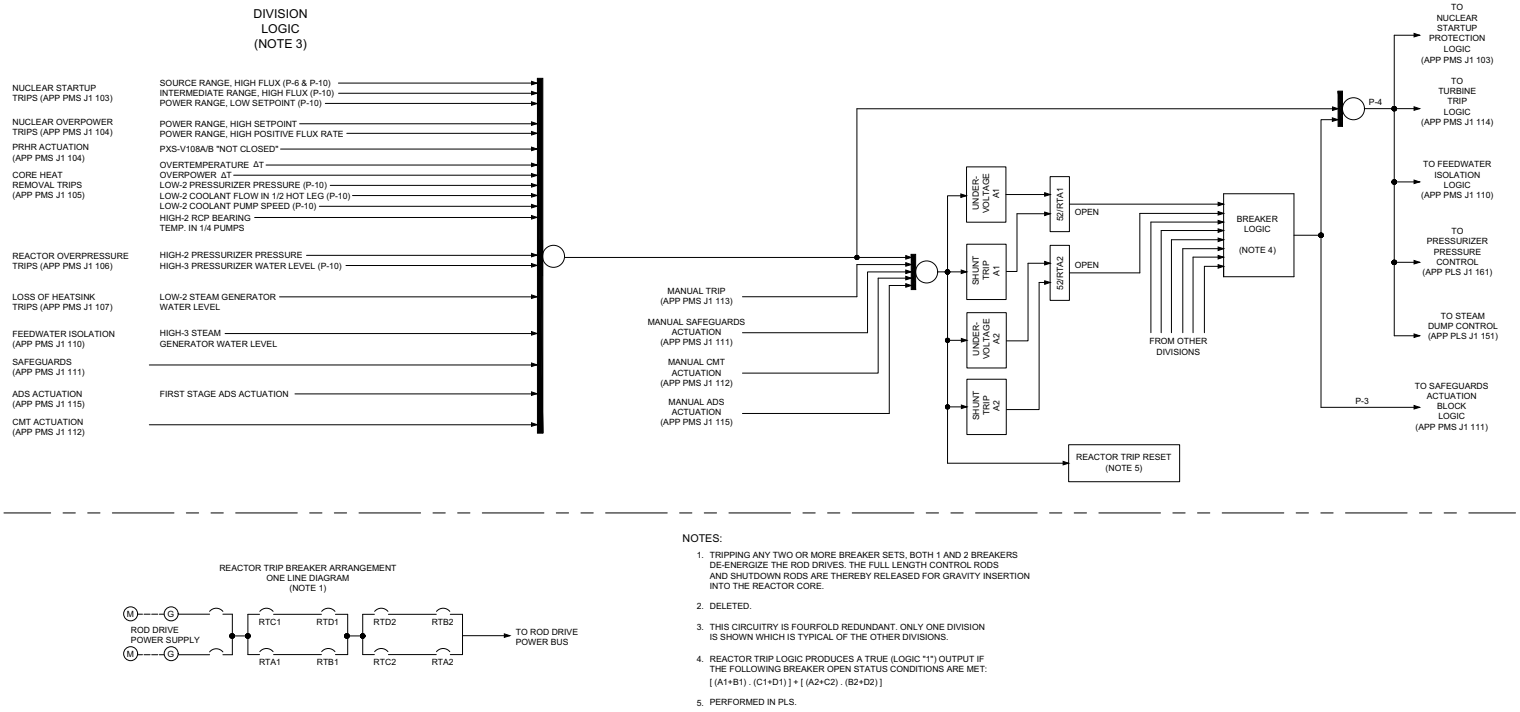

NOTES:

1. TRIPPING ANY TWO OR MORE BREAKER SETS, BOTH 1 AND 2 BREAKERS DE-ENERGIZE THE ROD DRIVES. THE FULL LENGTH CONTROL RODS AND SHUTDOWN RODS ARE THEREBY RELEASED FOR GRAVITY INSERTION INTO THE REACTOR CORE.
2. DELETED.
3. THIS CIRCUITRY IS FOURFOLD REDUNDANT. ONLY ONE DIVISION IS SHOWN WHICH IS TYPICAL OF THE OTHER DIVISIONS.
4. REACTOR TRIP LOGIC PRODUCES A TRUE (LOGIC "1") OUTPUT IF THE FOLLOWING BREAKER OPEN STATUS CONDITIONS ARE MET:
   [(A1+B1) . (C1+D1)] + [(A2+C2) . (B2+D2)]
5. PERFORMED IN PLS.

Figure 7.2-1 (Sheet 2 of 21)
Functional Diagram
Reactor Trip Functions

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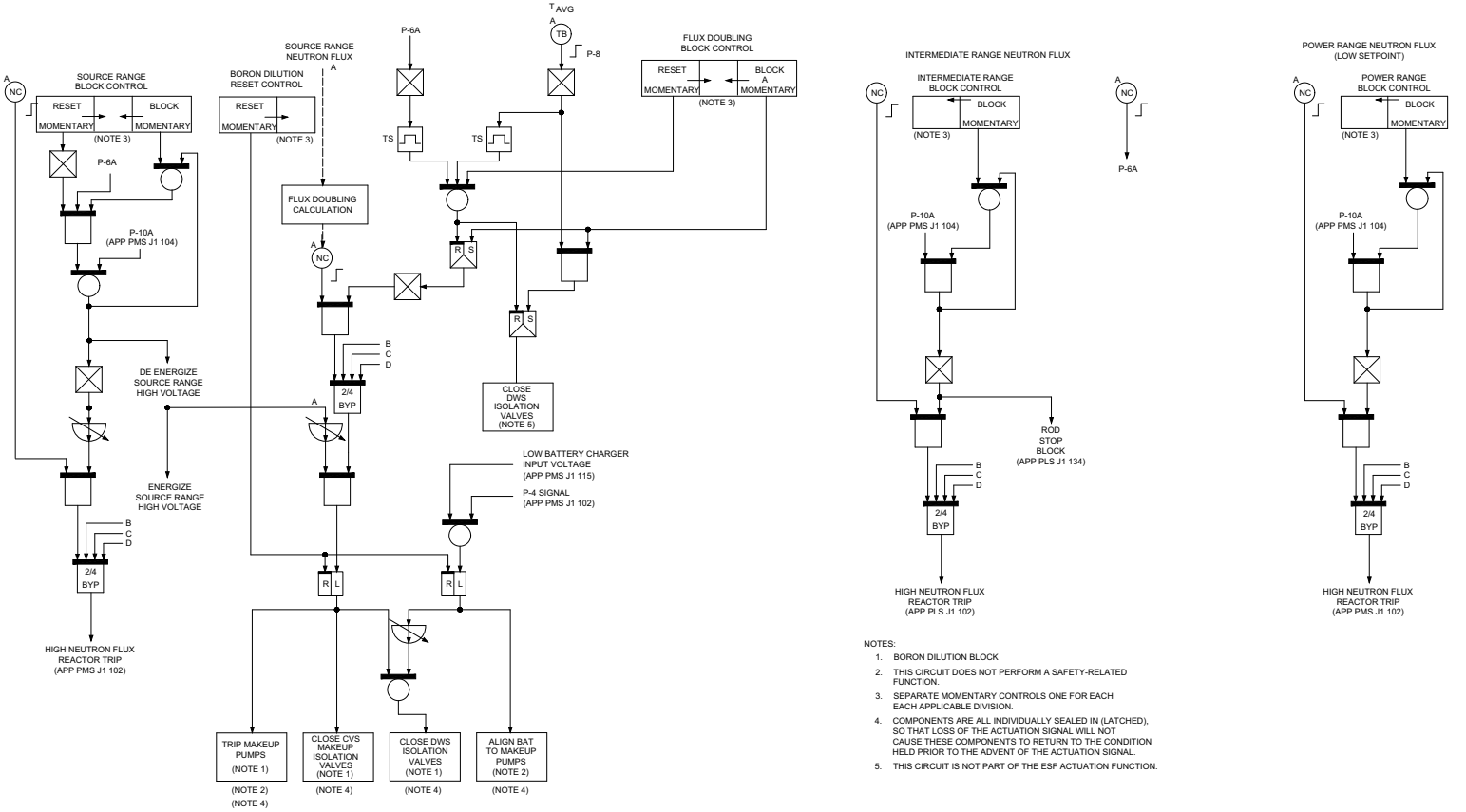

NOTES:

1. BORON DILUTION BLOCK
2. THIS CIRCUIT DOES NOT PERFORM A SAFETY-RELATED FUNCTION.
3. SEPARATE MOMENTARY CONTROLS ONE FOR EACH EACH APPLICABLE DIVISION.
4. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
5. THIS CIRCUIT IS NOT PART OF THE ESF ACTUATION FUNCTION.

Figure 7.2-1 (Sheet 3 of 21)
Functional Diagram
Nuclear Startup Protection

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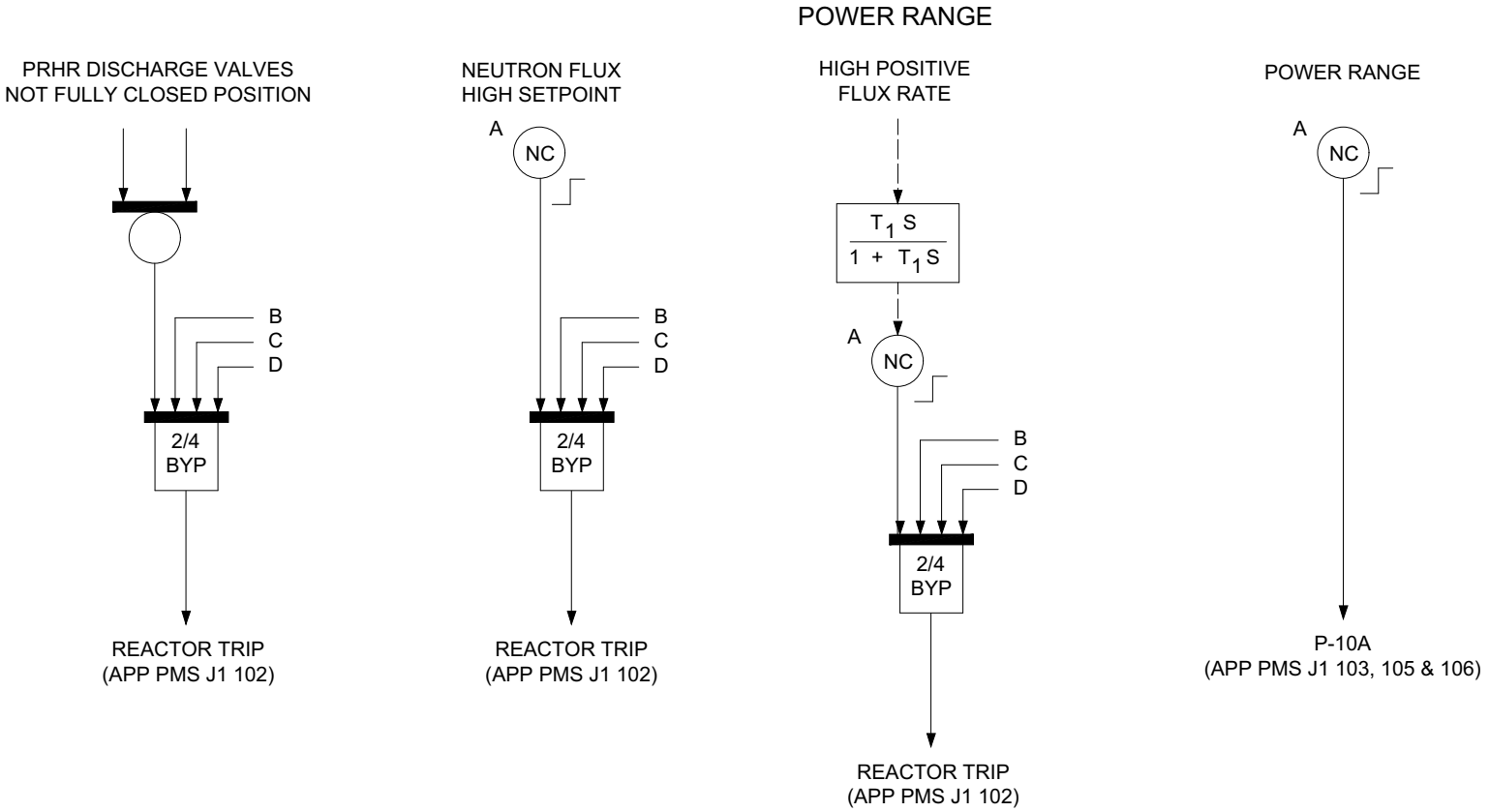

Figure 7.2-1 (Sheet 4 of 21)
Functional Diagram
Nuclear Overpower Protection

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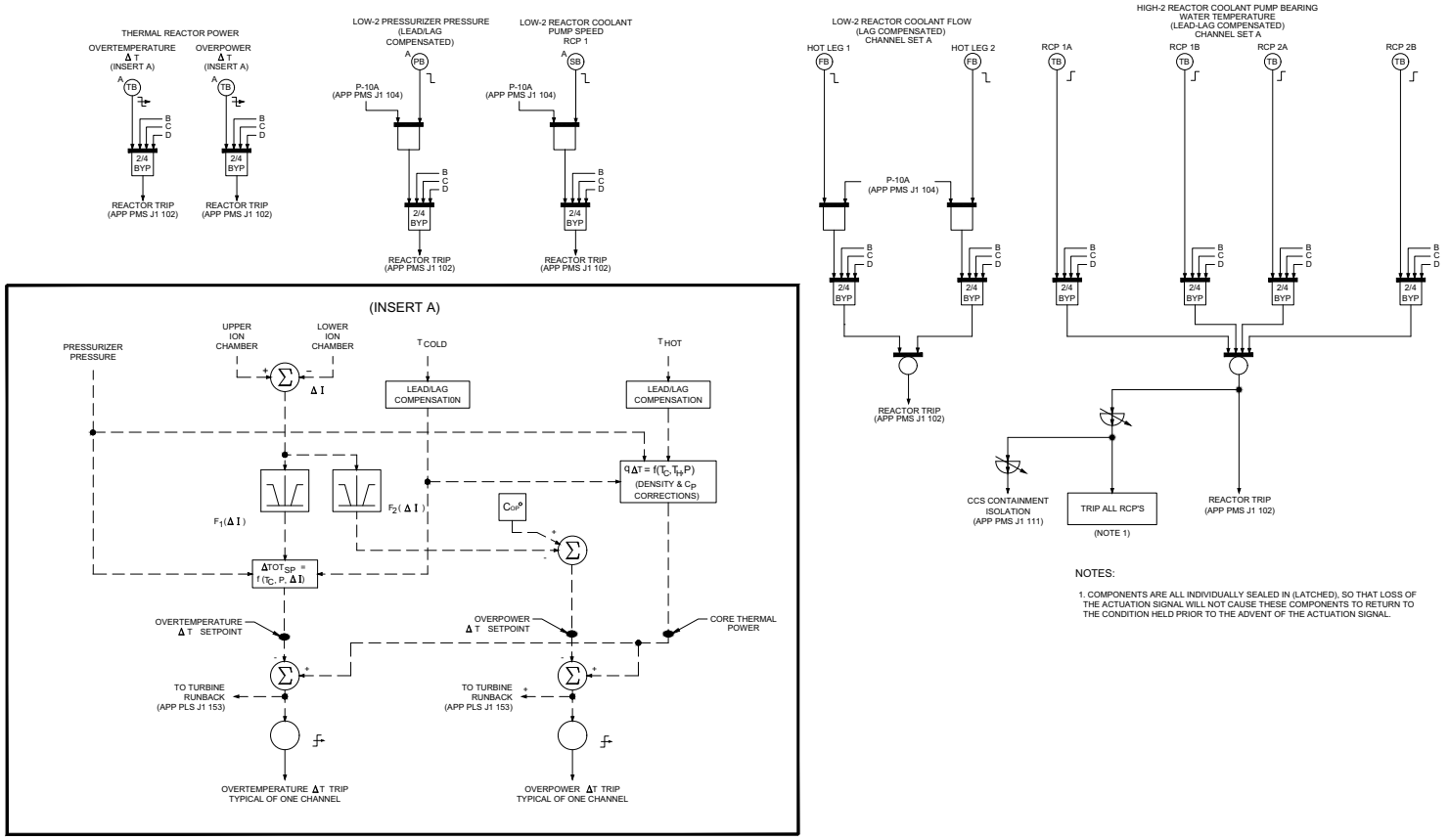

Figure 7.2-1 (Sheet 5 of 21)
Functional Diagram
Core Heat Removal Protection
and Reactor Coolant Pump Trip

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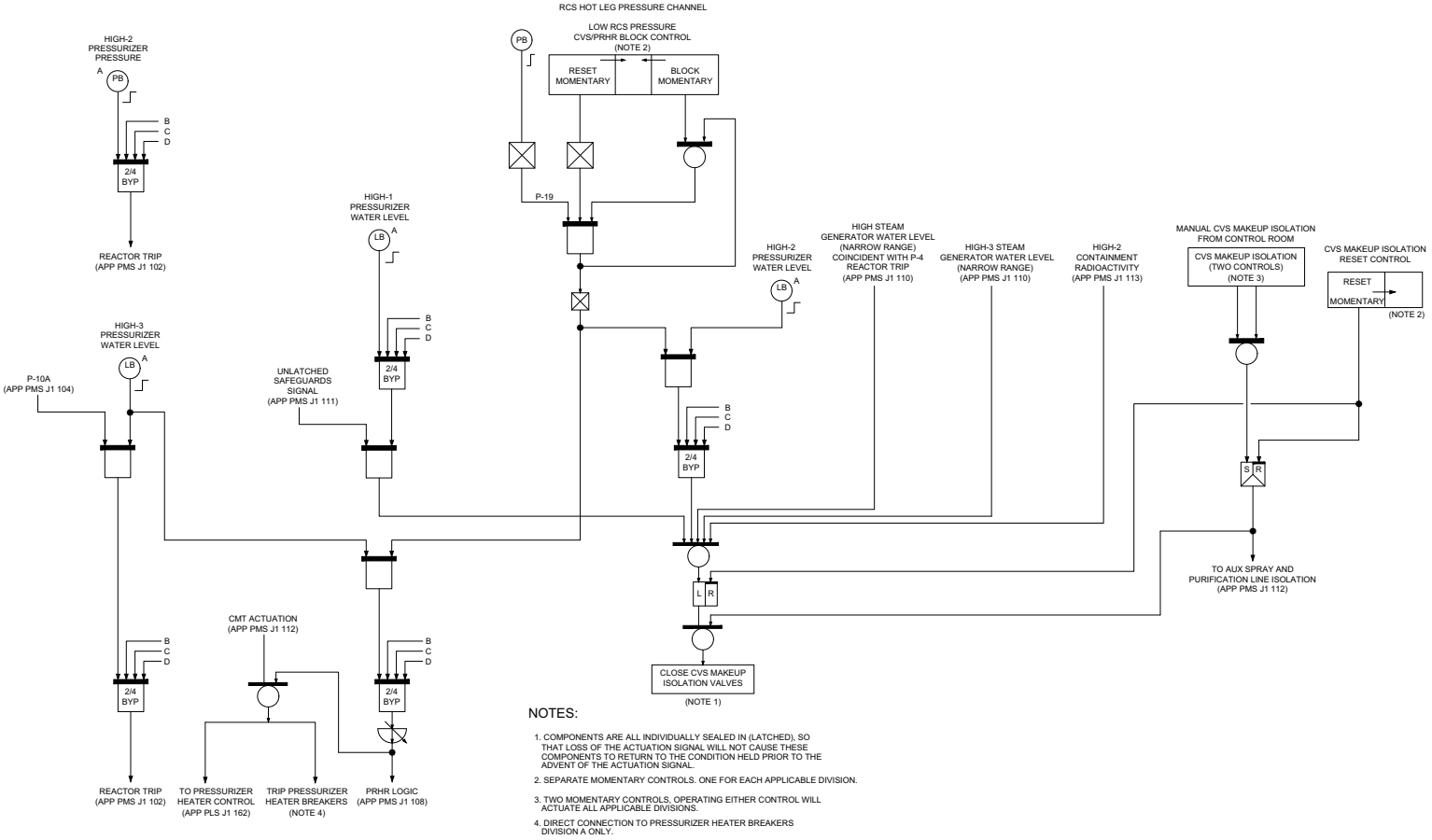

Figure 7.2-1 (Sheet 6 of 21)
Functional Diagram
Primary Overpressure & Loss of Heat Sink Protection

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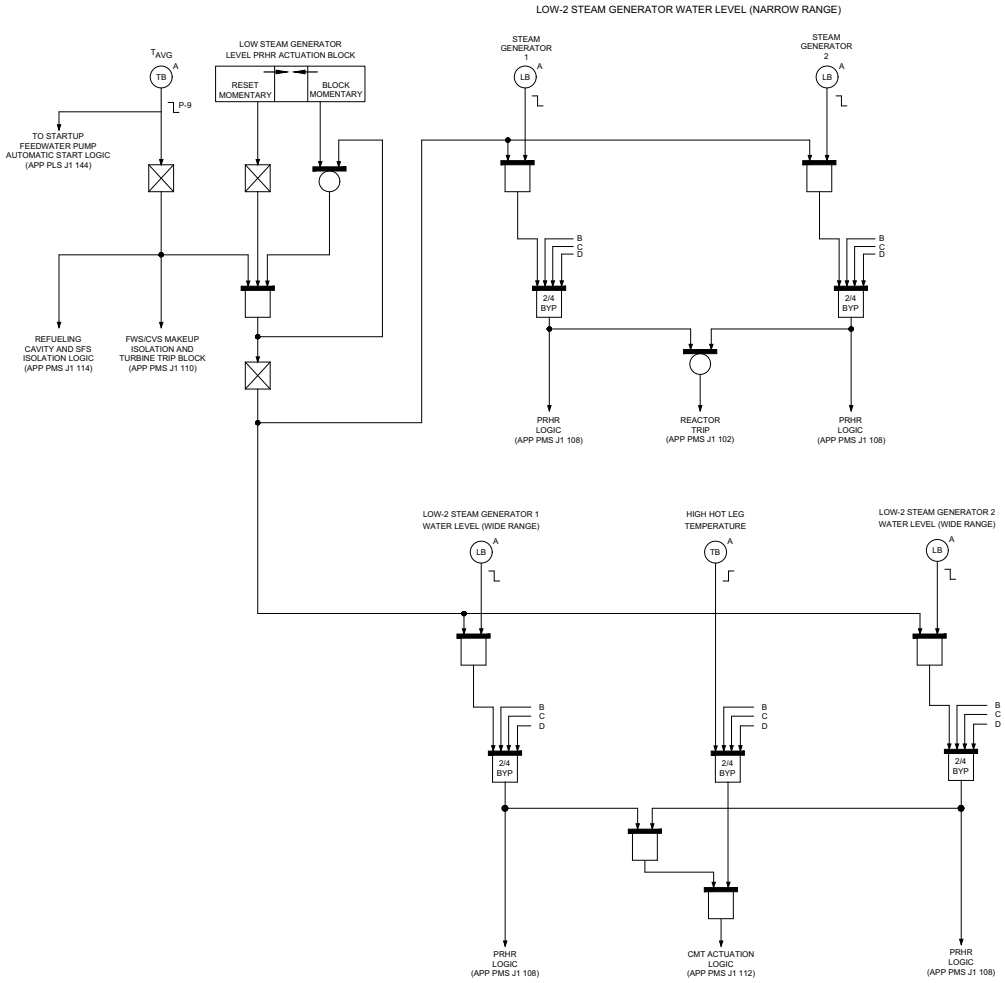

Figure 7.2-1 (Sheet 7 of 21)
Functional Diagram
Loss of Heat Sink Protection

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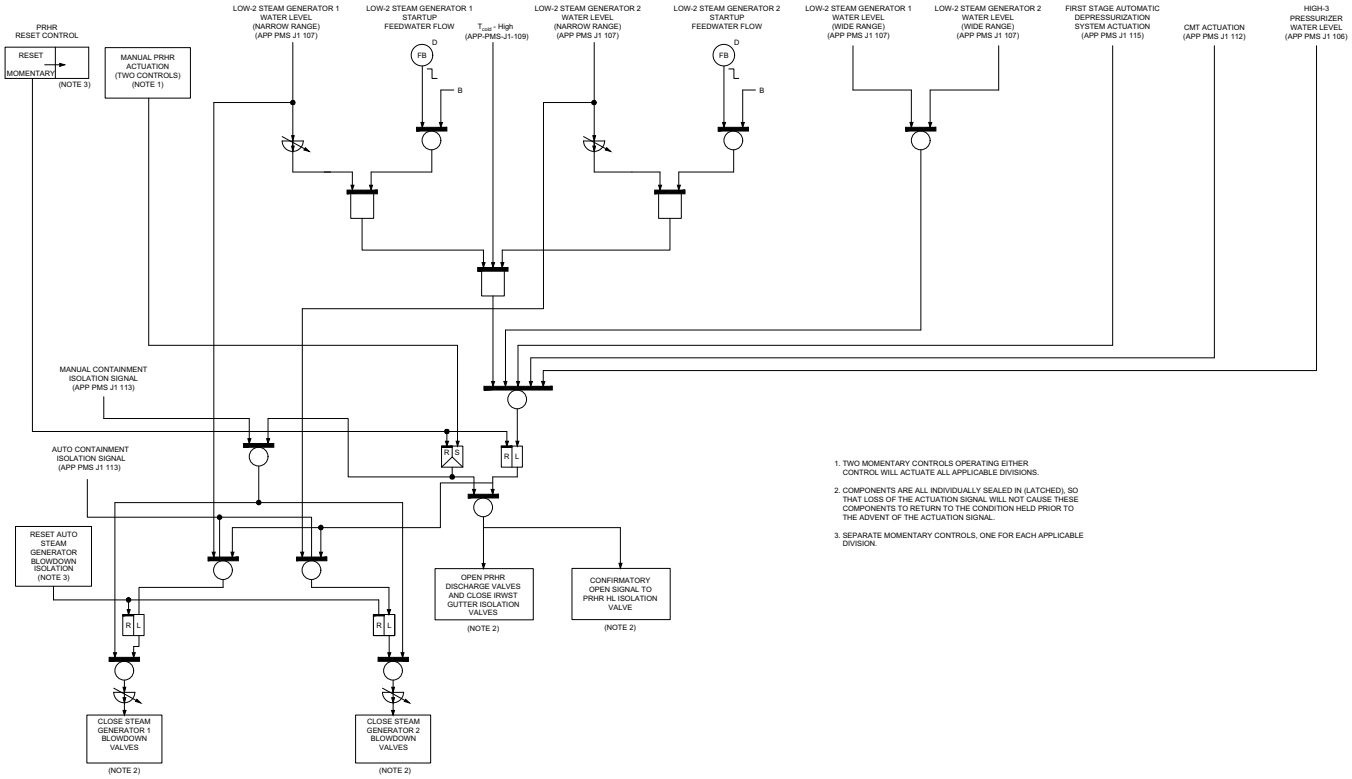

Figure 7.2-1 (Sheet 8 of 21)
Functional Diagram
Loss of Heat Sink Protection

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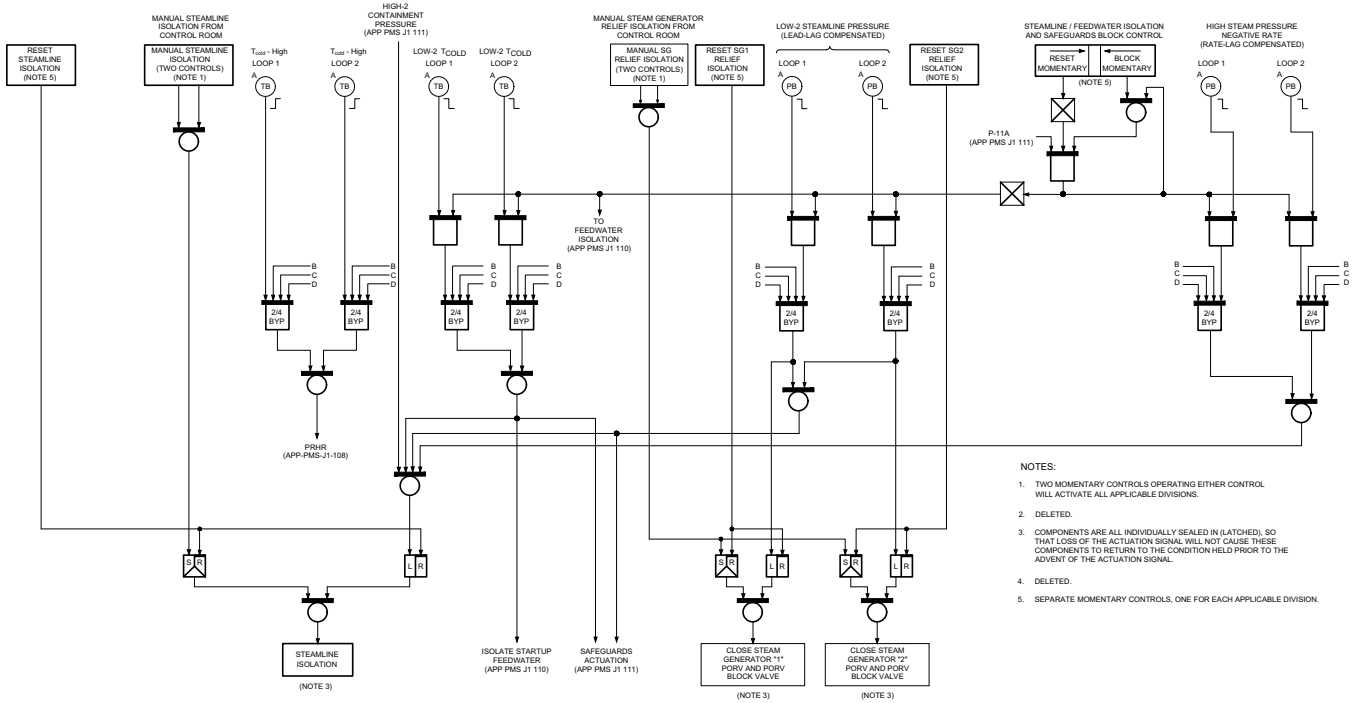

Figure 7.2-1 (Sheet 9 of 21)
Functional Diagram
Steam Line Isolation

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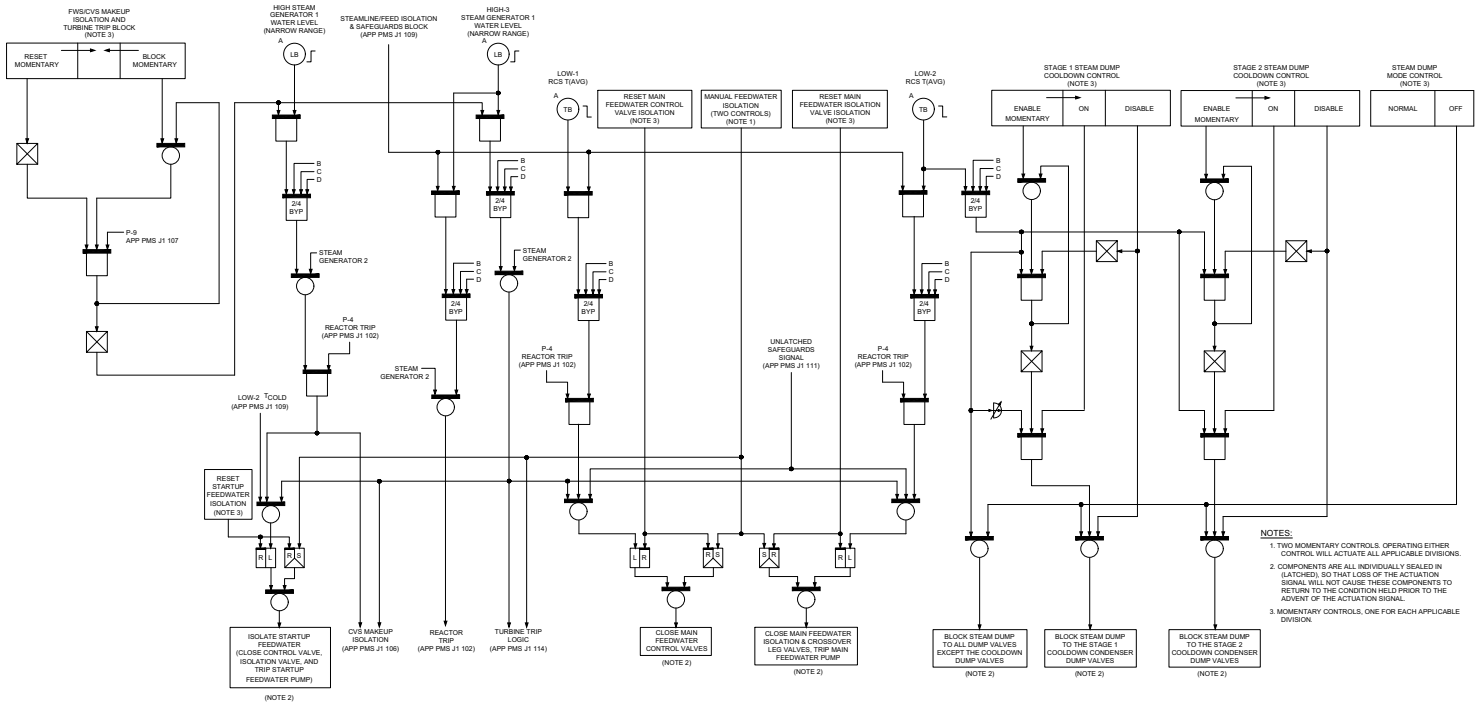

Figure 7.2-1 (Sheet 10 of 21)
Functional Diagram
Feedwater Isolation

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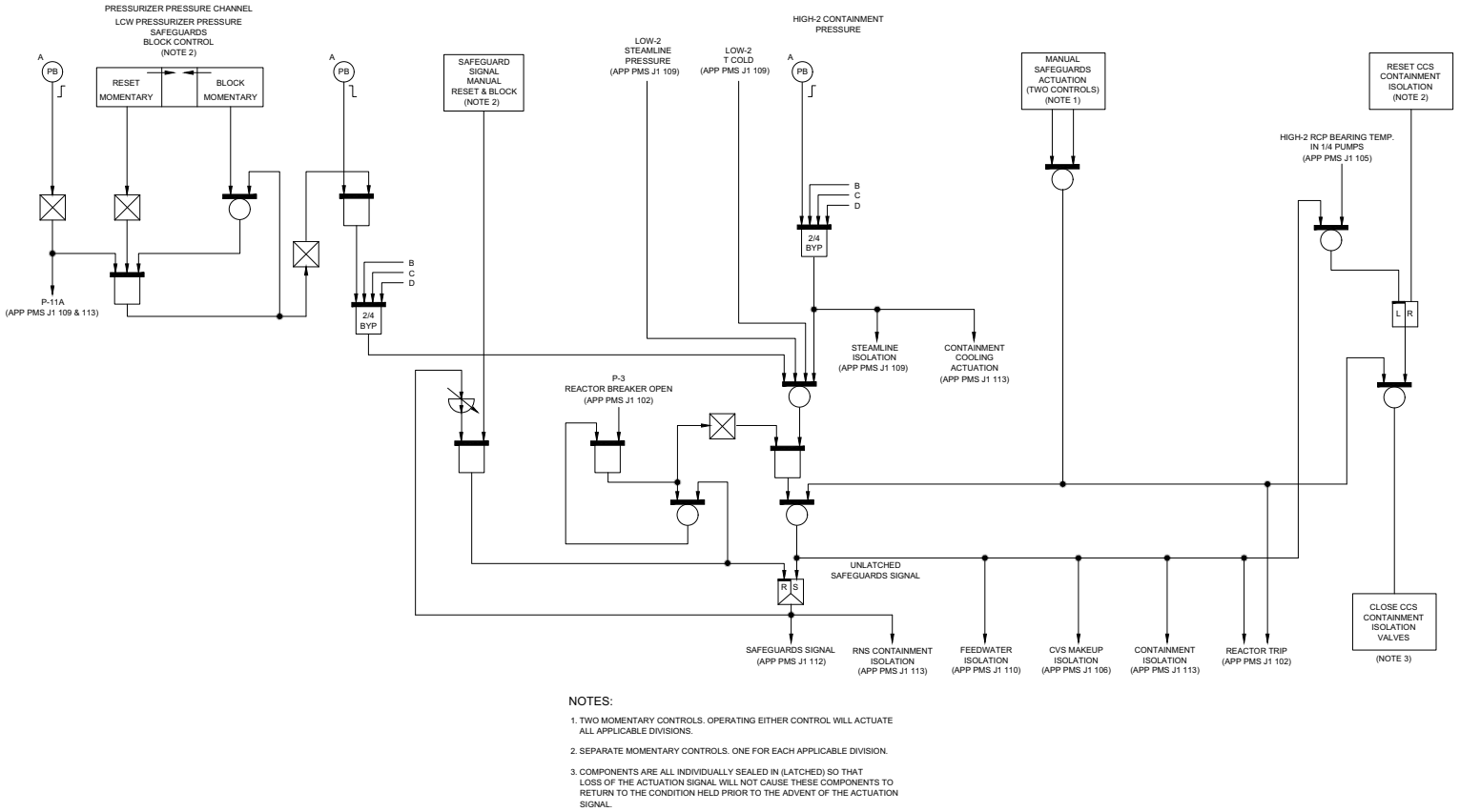

Figure 7.2-1 (Sheet 11 of 21)
Functional Diagram
Safeguards Actuation

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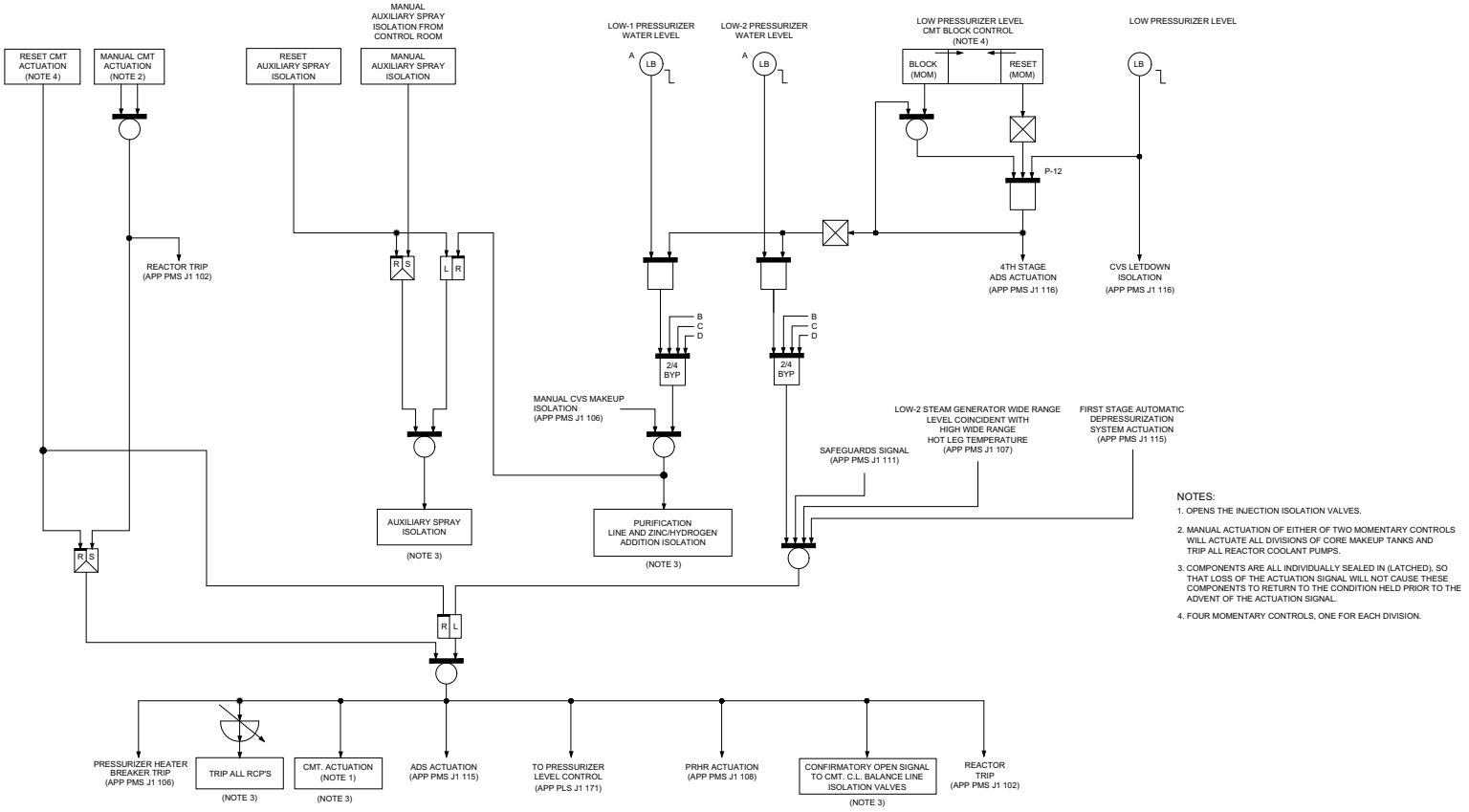

Figure 7.2-1 (Sheet 12 of 21)
Functional Diagram
Core Makeup Tank Actuation

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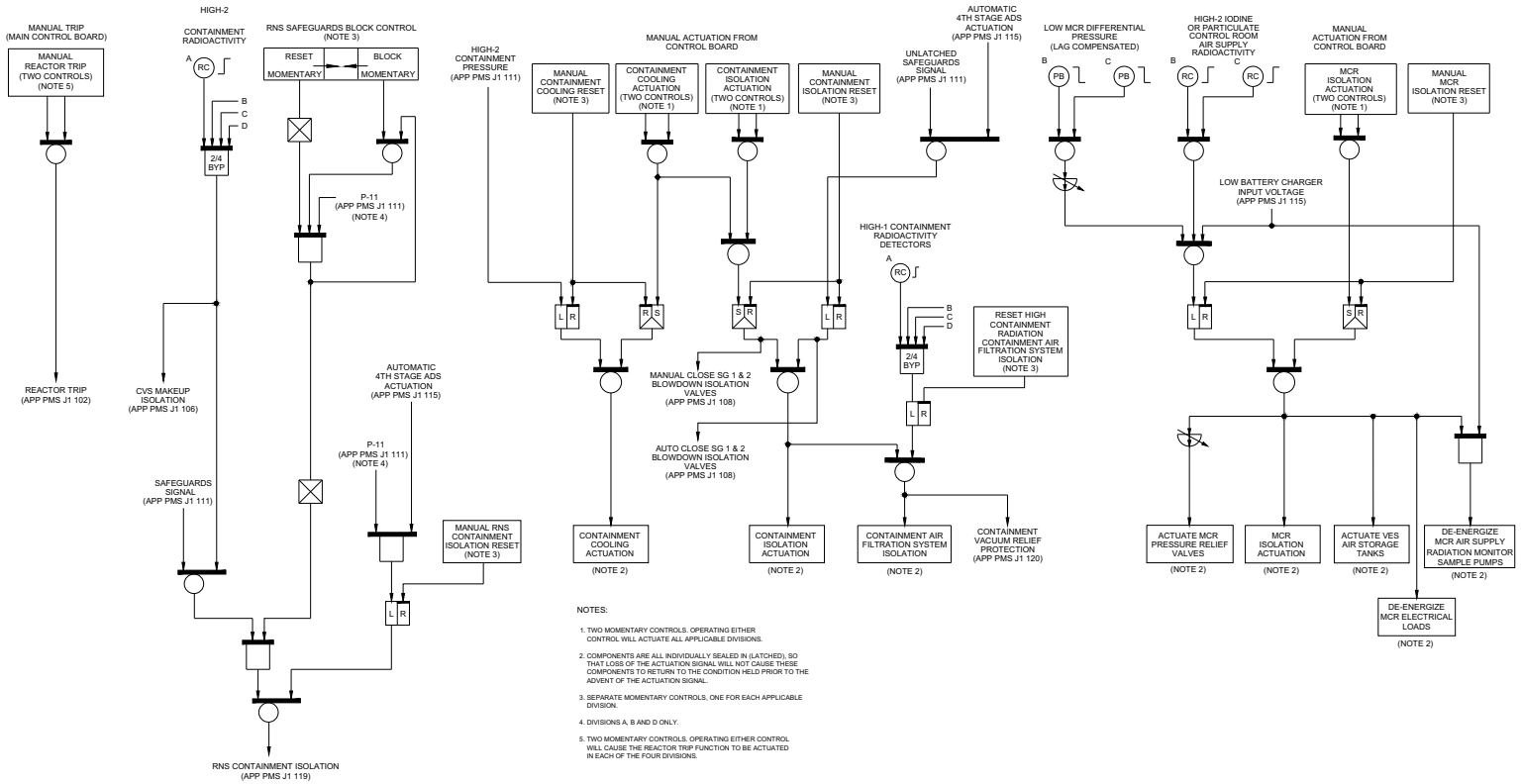

Figure 7.2-1 (Sheet 13 of 21)
Functional Diagram
Containment and Other Protection

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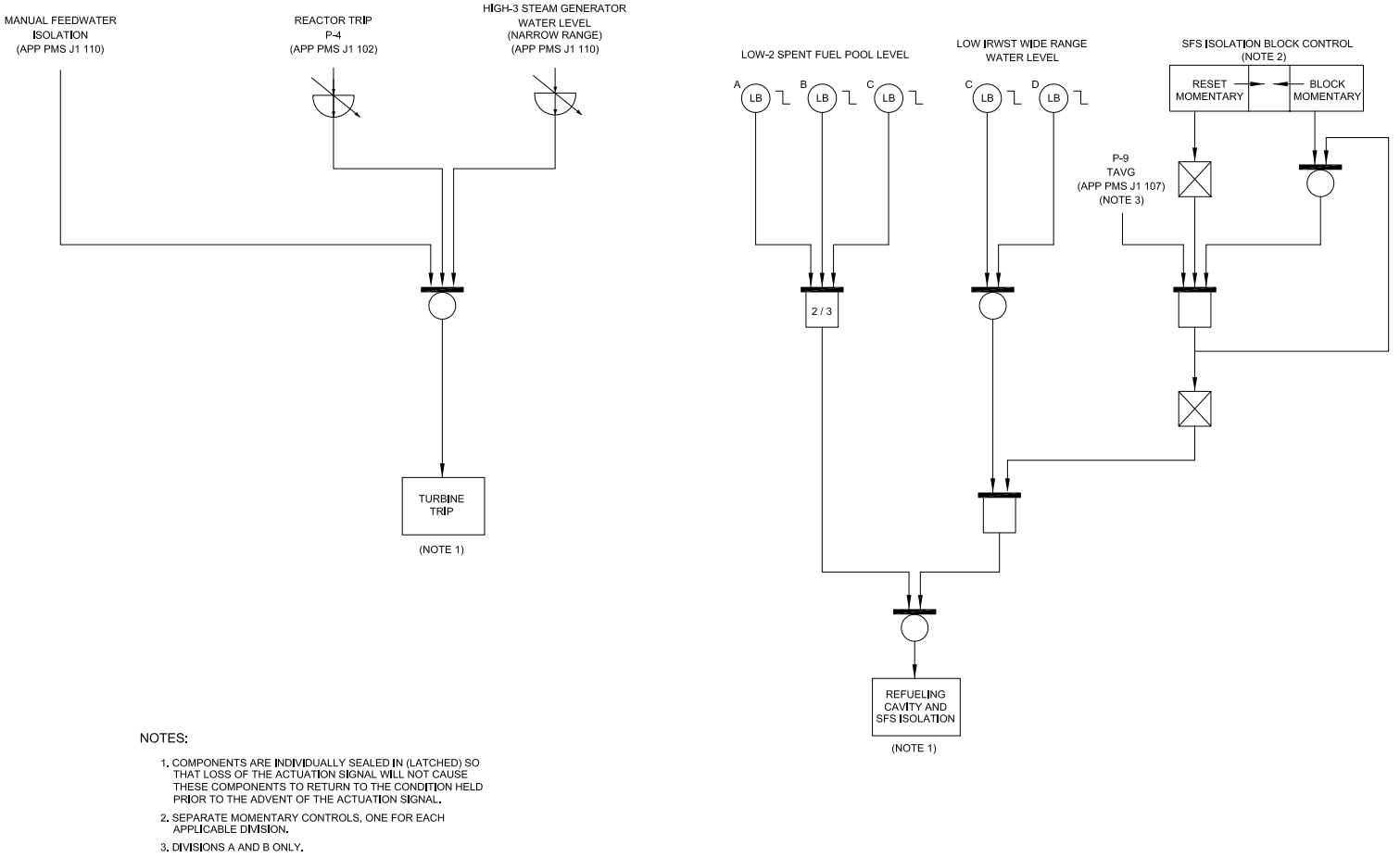

NOTES:

1. COMPONENTS ARE INDIVIDUALLY SEALED IN (LATCHED) SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
2. SEPARATE MOMENTARY CONTROLS, ONE FOR EACH APPLICABLE DIVISION.
3. DIVISIONS A AND B ONLY.

Figure 7.2-1 (Sheet 14 of 21)
Functional Diagram
Turbine Trip and Refueling Cavity and SFS Isolation

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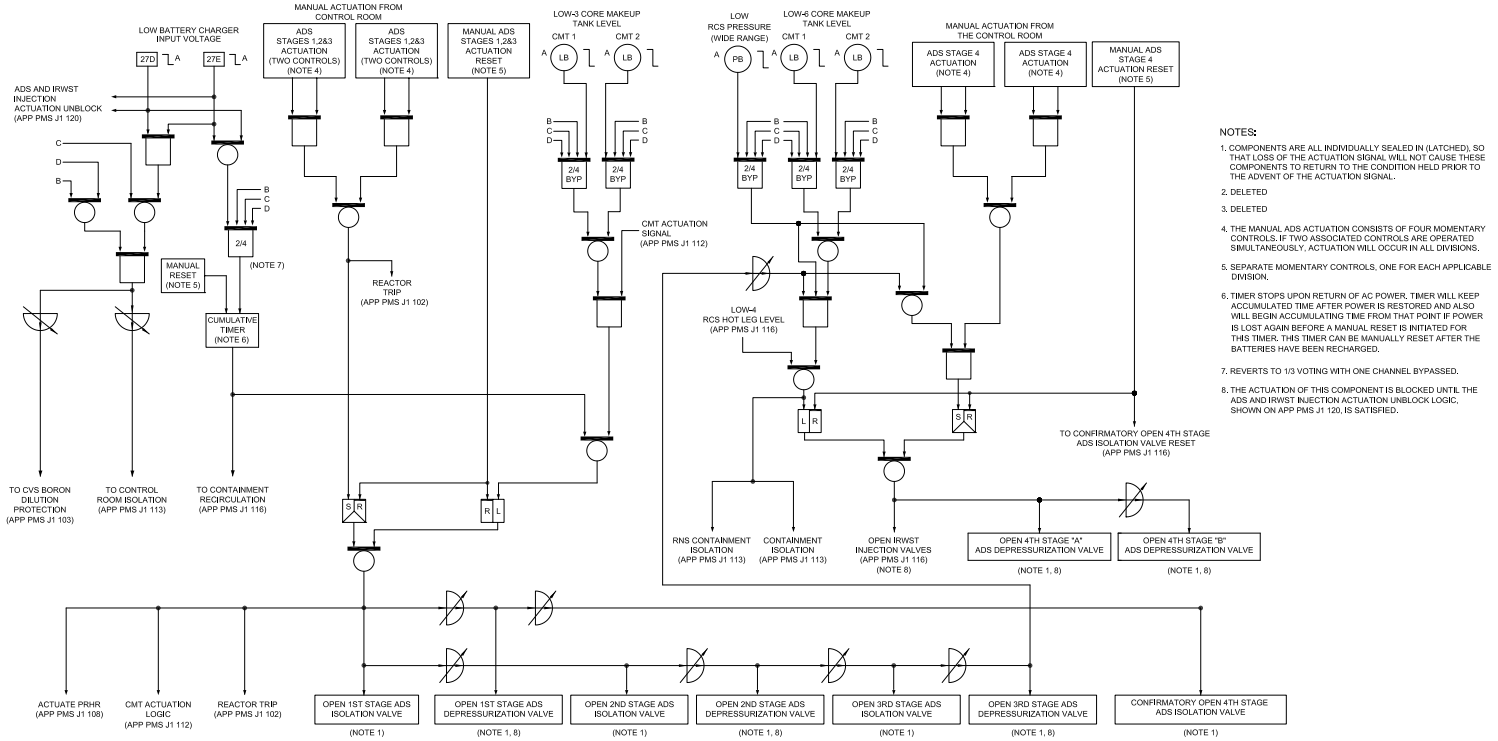

Figure 7.2-1 (Sheet 15 of 21)
Functional Diagram
Automatic RCS Depressurization Valve Sequencing

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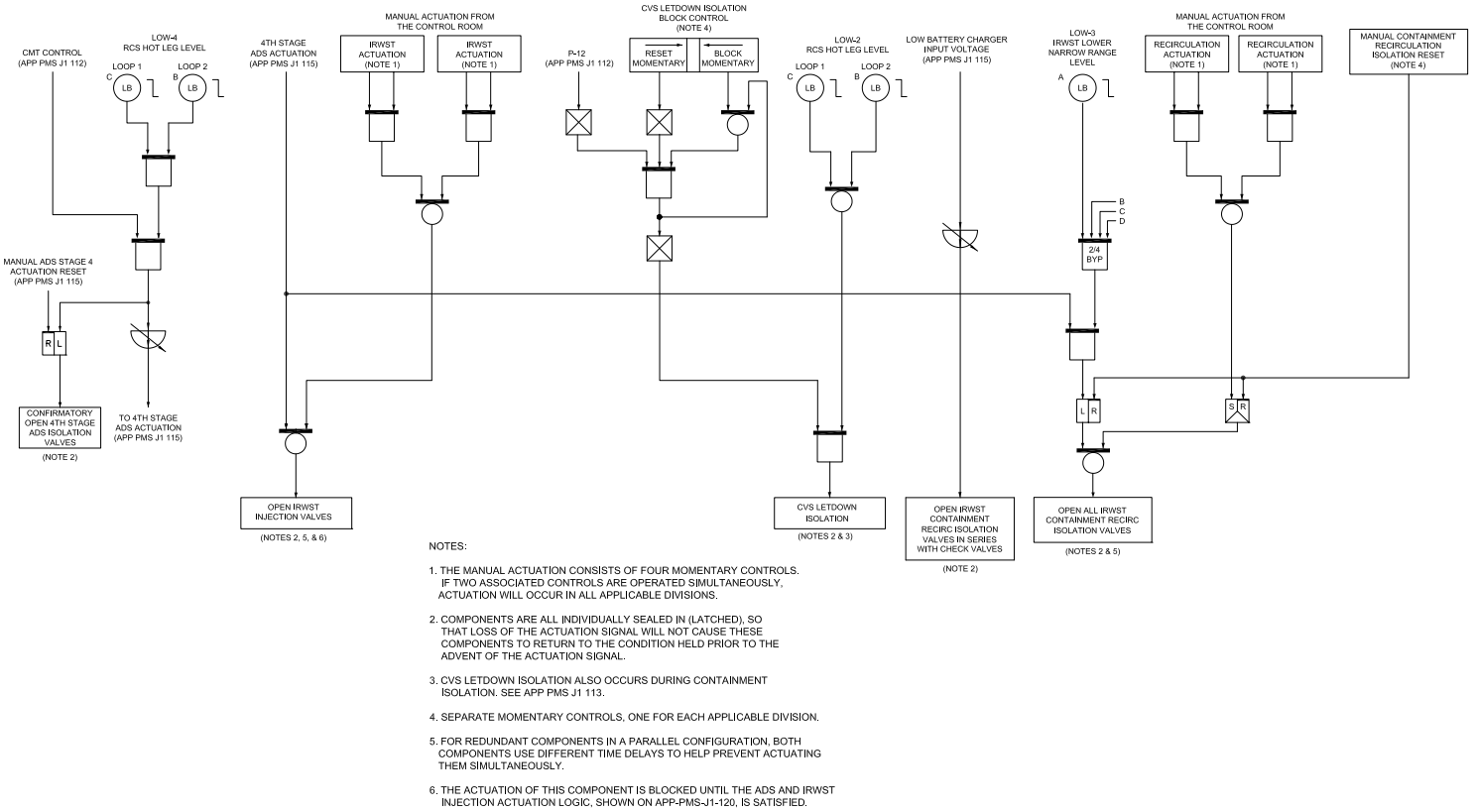

NOTES:

1. THE MANUAL ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS. IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY, ACTUATION WILL OCCUR IN ALL APPLICABLE DIVISIONS.
2. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
3. CVS LETDOWN ISOLATION ALSO OCCURS DURING CONTAINMENT ISOLATION. SEE APP PMS J1 113.
4. SEPARATE MOMENTARY CONTROLS, ONE FOR EACH APPLICABLE DIVISION.
5. FOR REDUNDANT COMPONENTS IN A PARALLEL CONFIGURATION, BOTH COMPONENTS USE DIFFERENT TIME DELAYS TO HELP PREVENT ACTUATING THEM SIMULTANEOUSLY.
6. THE ACTUATION OF THIS COMPONENT IS BLOCKED UNTIL THE ADS AND IRWST INJECTION ACTUATION LOGIC, SHOWN ON APP-PMS-J1-120, IS SATISFIED.

Figure 7.2-1 (Sheet 16 of 21)
Functional Diagram
In-containment Refueling Water Storage Tank Actuations

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Figure 7.2-1 (Sheet 17 of 21) Not Used

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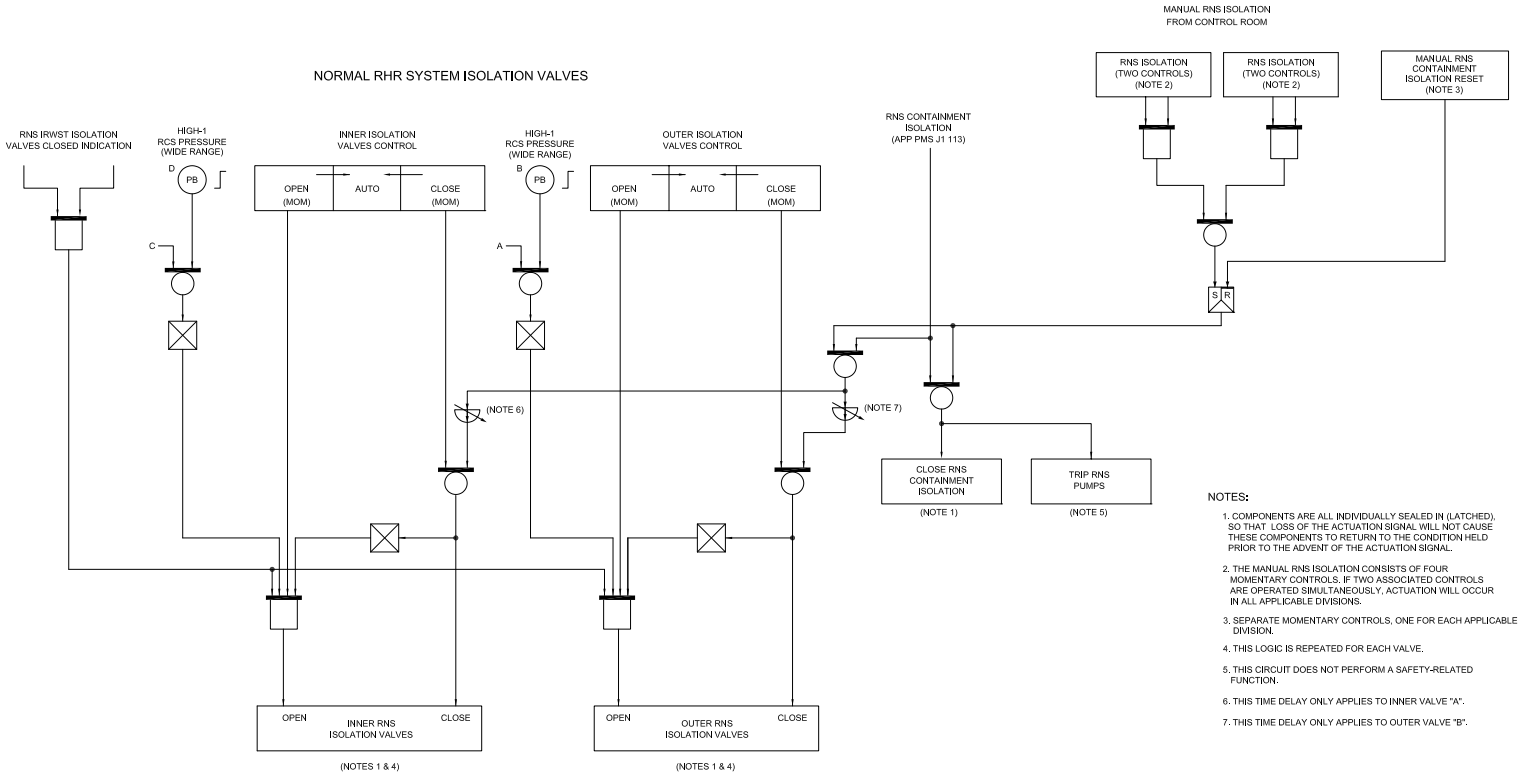

Figure 7.2-1 (Sheet 18 of 21)
Functional Diagram
Normal Residual Heat Removal System Isolation Valve Interlocks

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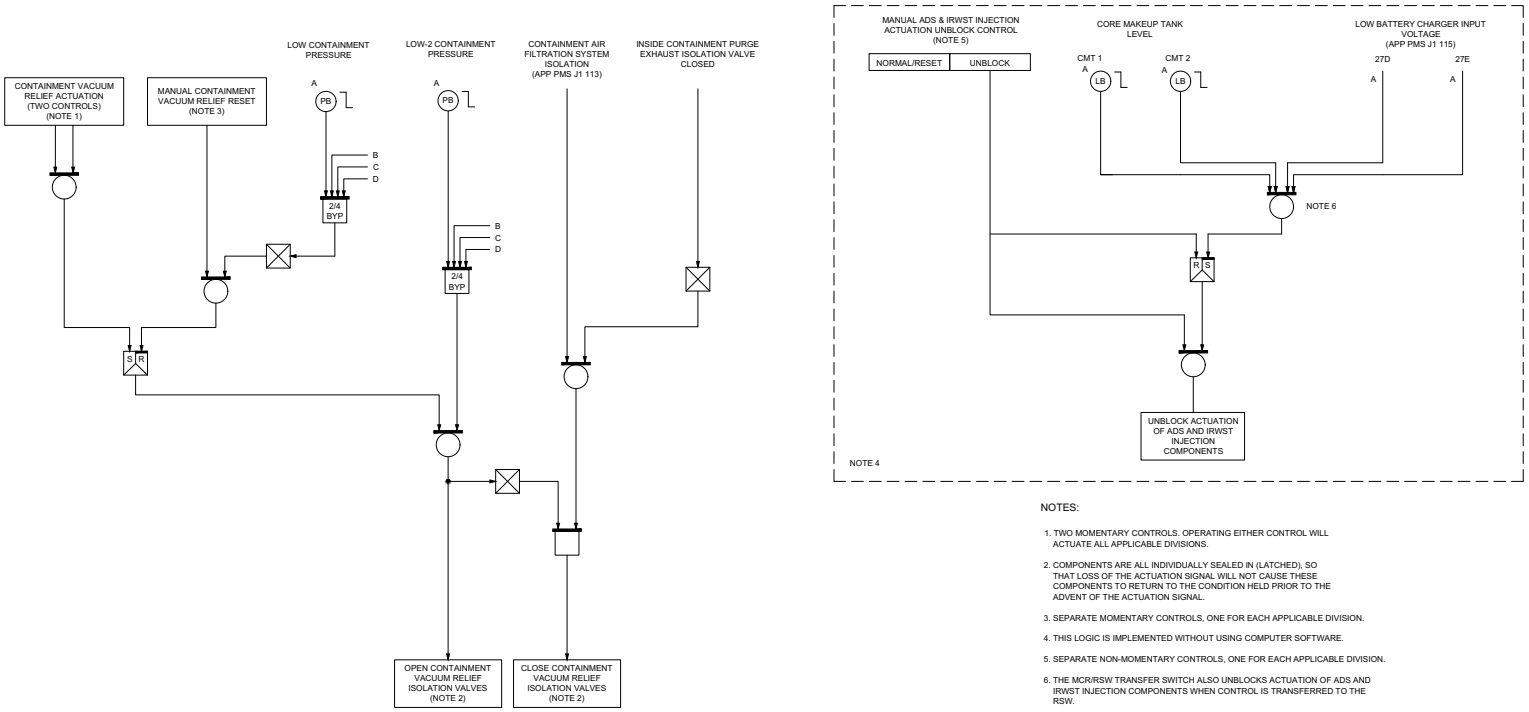

NOTES:

1. TWO MOMENTARY CONTROLS. OPERATING EITHER CONTROL WILL ACTUATE ALL APPLICABLE DIVISIONS.
2. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
3. SEPARATE MOMENTARY CONTROLS, ONE FOR EACH APPLICABLE DIVISION.
4. THIS LOGIC IS IMPLEMENTED WITHOUT USING COMPUTER SOFTWARE.
5. SEPARATE NON-MOMENTARY CONTROLS, ONE FOR EACH APPLICABLE DIVISION.
6. THE MCR/RSW TRANSFER SWITCH ALSO UNBLOCKS ACTUATION OF ADS AND IRWST INJECTION COMPONENTS WHEN CONTROL IS TRANSFERRED TO THE RSW.

Figure 7.2-1 (Sheet 19 of 21)
Functional Diagram
Containment Vacuum Relief Protection

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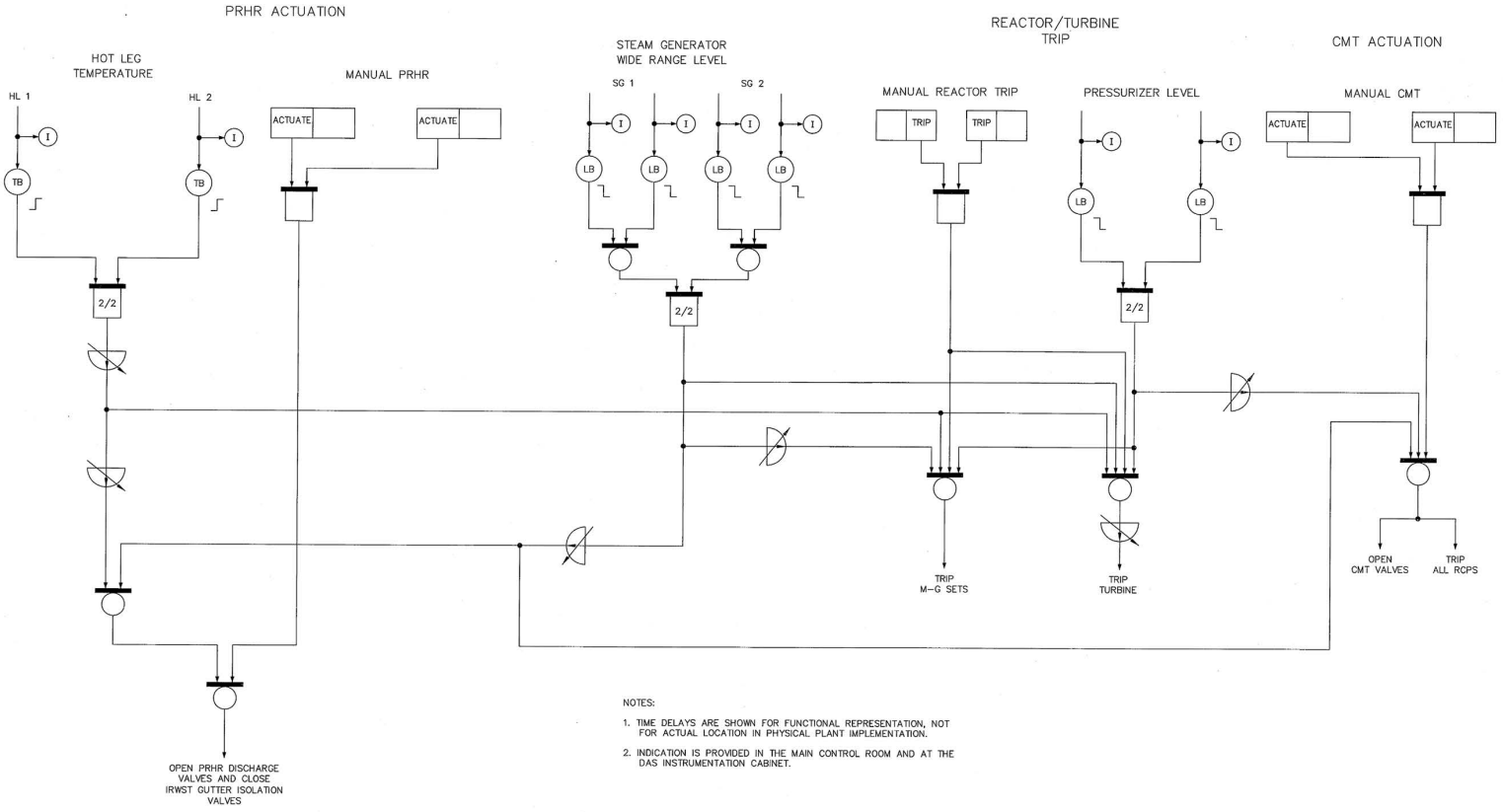

Figure 7.2-1 (Sheet 20 of 21)
Functional Diagram
Diverse Actuation System Logic Automatic Actuations

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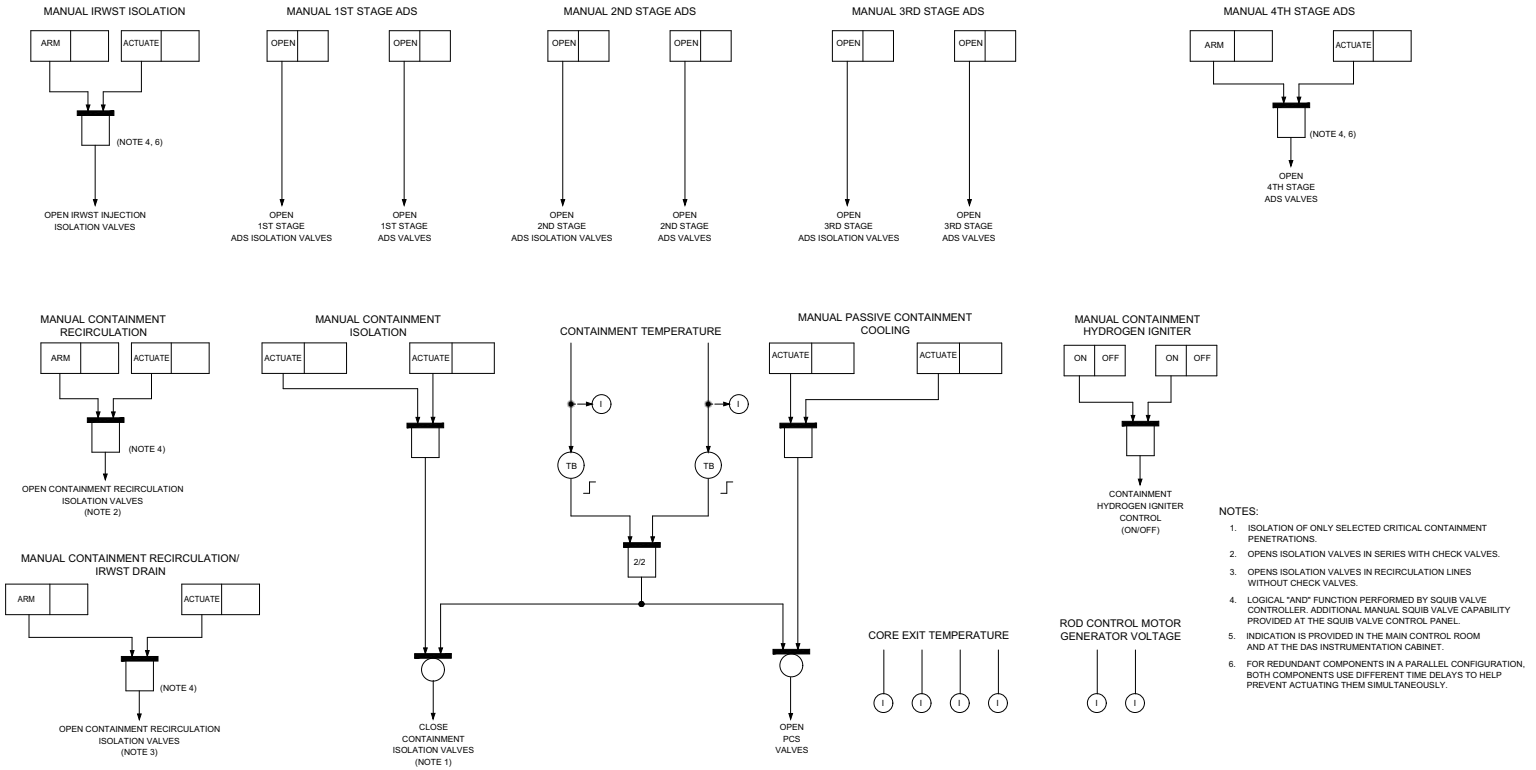

Figure 7.2-1 (Sheet 21 of 21)
Functional Diagram
Diverse Actuation System Logic, Manual Actuations

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7.3 Engineered Safety Features

AP1000 provides instrumentation and controls to sense accident situations and initiate engineered safety features (ESF). The occurrence of a limiting fault, such as a loss of coolant accident or a secondary system break, requires a reactor trip plus actuation of one or more of the engineered safety features. This combination of events prevents or mitigates damage to the core and reactor coolant system components, and provides containment integrity.

7.3.1 Description

The protection and safety monitoring system is actuated when safety system setpoints are reached for selected plant parameters. The selected combination of process parameter setpoint violations is indicative of primary or secondary system boundary ruptures. Once the required logic combination is generated, the protection and safety monitoring system equipment sends the signals to actuate appropriate engineered safety features components.

The following paragraphs summarize the major functional elements of the protection and safety monitoring system that are involved in generating an actuation signal to an engineered safety features component.

Four sensors normally monitor each variable used for an engineered safety feature actuation. (These sensors may monitor the same variable for a reactor trip function.) Analog measurements are converted to digital form by analog-to-digital converters within each of the four divisions of the protection and safety monitoring system. Following required signal conditioning or processing, the measurements are compared against the setpoints for the engineered safety feature to be generated. Steam generator level and pressurizer level are compensated per Subsection 7.3.1.5.2. When the measurement exceeds the setpoint, the output of the comparison results in a channel partial trip condition. The partial trip information is transmitted to the ESF coincidence logic to form the signals that result in an engineered safety features actuation. The voting logic is performed twice within each division. Each voting logic element generates an actuation signal if the required coincidence of partial trips exists at its inputs.

The signals are combined within each division of ESF coincidence logic to generate a system-level signal. System-level manual actions are also processed by the logic in each division.

The system-level signals are then broken down to the individual actuation signals to actuate each component associated with a system-level engineered safety feature. For example, a single safeguards actuation signal must trip the reactor and the reactor coolant pumps, align core makeup tank and in-containment refueling water storage tank valves, and initiate containment isolation. The interposing logic accomplishes this function and also performs necessary interlocking so that components are properly aligned for safety. Component-level manual actions are also processed by this interposing logic. The power interface transforms the low level signals to voltages and currents commensurate with the actuation devices they operate. The actuation devices, in turn, control motive power to the final engineered safety feature component.

Subsection 7.3.1.2 provides a functional description of the signals and initiating logic for each of the engineered safety features. Figure 7.2-1 presents the functional diagrams for engineered safety features actuation.

Table 7.3-1 summarizes the signals and initiating logic for each of the engineered safety features initiated by the protection and safety monitoring system. Most of the functions provide protection against design basis events which are analyzed in Chapter 6 and Chapter 15. However, not all the functions listed in Table 7.3-1 are necessary to meet the assumptions used in performing the safety analysis. For example, the design provides features which provide automatic actuations which are

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not required for performing the safety analysis. In addition, some functions are provided to support assumptions used in the probabilistic risk assessment, but are not used to mitigate a design basis accident. Only those functions which meet the 10 CFR 50.36(c)(2)(ii) criteria are included in the Technical Specifications. This accounts for any difference between functions listed in Table 7.3-1 and functions which are included in the Technical Specifications.

7.3.1.1 Safeguards Actuation (S) Signal

A safeguards actuation (S) signal is used in the initiation logic of many of the engineered safety features discussed in Subsection 7.3.1.2. In addition, as described in Section 7.2, the safeguards actuation signal also initiates a reactor trip. The variables that are monitored and used to generate a safeguards actuation signal are typically those that provide indication of a significant plant transient that requires a response by several engineered safety features.

The safeguards actuation signal is generated from any of the following initiating conditions:

1. Low-3 pressurizer pressure
2. Low-2 lead-lag compensated steam line pressure
3. Low-2 cold leg temperature
4. High-2 containment pressure
5. Manual initiation

Condition 1 results from the coincidence of pressurizer pressure below the Low-3 setpoint in any two of the four divisions.

Condition 2 results from the coincidence of two of the four divisions of compensated steam line pressure below the Low-2 setpoint in either of the two steam lines. The steam line pressure signal is lead-lag compensated to improve system response.

Condition 3 results from the coincidence of two of the four divisions of reactor coolant system cold leg temperature below the Low-2 setpoint in any loop.

Condition 4 results from the coincidence of two of the four divisions of containment pressure above the High-2 setpoint.

Condition 5 consists of two momentary controls. Manual actuation of either of the two controls will trip the reactor and generate a safeguards actuation signal.

To permit startup and cooldown, the safeguards actuation signals generated from Low-3 pressurizer pressure, Low-2 steam line pressure, or Low-2 reactor coolant inlet temperature can be manually blocked when pressurizer pressure is below the P-11 setpoint. The signal is automatically unblocked when the pressurizer pressure is above the P-11 setpoint.

Separate momentary controls are provided, each of which will manually reset the safeguards actuation signal in a single division. Manual reset of a safeguards actuation signal in coincidence with reactor trip breaker open (P-3) blocks the safeguards actuation signal. Absence of P-3 automatically resets the blocking function. The safeguards actuation signal is manually reset based on a preset delay following initiation. Resetting the signal does not reposition any safeguards actuated equipment, since individual components are required to latch in and seal on the safeguards actuation signal.

The logic relating to the development of the safeguards actuation signal is illustrated in Figure 7.2-1, sheets 9 and 11.

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7.3.1.2 Engineered Safety Feature Descriptions

The following subsections provide a functional description of the signals and initiating logic for each engineered safety feature. Table 7.3-1 lists the signals and summarizes the coincidence logic used to generate the safeguards actuation signal or initiate each engineered safety feature. Table 7.3-2 describes the permissives and interlocks relating to the engineered safety features. Table 7.3-3 lists the system-level manual input to the engineered safety features.

7.3.1.2.1 Containment Isolation

A signal to actuate containment isolation is generated from any of the following conditions:

1. Automatic or manual safeguards actuation signal (Subsection 7.3.1.1)
2. Automatic fourth stage ADS actuation (Subsection 7.3.1.2.4)
3. Manual initiation
4. Manual actuation of passive containment cooling (Subsection 7.3.1.2.12)

Conditions 1, 2, and 4 are discussed in other subsections as noted.

Condition 3 consists of the manual actuation of either of two momentary controls in the main control room. Either control actuates all divisions and closes the nonessential fluid system paths from the containment.

Manual reset is provided to block the automatic actuation signal for containment isolation. Separate momentary controls are provided for resetting each division.

No other interlocks or permissive signals apply directly to the containment isolation function. Automatic actuation originates from a safeguards actuation (S) signal that does contain interlock and permissive inputs.

The functional logic that actuates containment isolation is illustrated in Figure 7.2-1, sheets 11, 13, and 15.

7.3.1.2.2 In-Containment Refueling Water Storage Tank Injection

Signals to align the in-containment refueling water storage tank for injection are generated from the following conditions:

1. Actuation of the fourth stage of the automatic depressurization system (Subsection 7.3.1.2.4)

2. Coincidence loop 1 and loop 2 hot leg levels below Low-4 setpoint for a duration exceeding an adjustable time delay

3. Manual initiation

Each of the above conditions opens the in-containment refueling water storage tank injection valves, thereby providing a flow path to the reactor coolant system. As described in Subsection 7.3.1.2.4.1, a blocking device prevents ADS spurious actuation as well as in-containment refueling water storage tank injection.

In addition to initiating in-containment refueling water storage tank injection, condition 2 also initiates the opening sequence of the fourth stage of the automatic depressurization system. This is discussed in Subsection 7.3.1.2.4.

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Condition 3 consists of two sets of two momentary controls. Manual actuation of both controls of either of the two control sets generates signals that open the in-containment refueling water storage tank injection valves. A two-control simultaneous actuation prevents inadvertent actuation.

In-containment refueling water storage tank injection on Low-4 hot leg level is automatically blocked when the pressurizer water level is above the P-12 setpoint. This reduces the probability of a spurious injection. This block is removed when the core makeup tank actuation on Low-2 pressurizer level function is manually blocked to allow mid-loop operation. As described in Subsection 7.3.1.2.3, this core makeup tank actuation function can be manually blocked when the pressurizer water level is below the P-12 setpoint.

The functional logic relating to in-containment refueling water storage tank injection is illustrated in Figure 7.2-1, sheets 12 and 16.

7.3.1.2.3 Core Makeup Tank Injection

Signals to align the core makeup tanks for injection are generated from the following conditions:

1. Automatic or manual safeguards actuation (Subsection 7.3.1.1)

2. Automatic or manual actuation of the first stage of the automatic depressurization system (Subsection 7.3.1.2.4)

3. Low-2 pressurizer level

4. Low-2 steam generator wide range water level coincident with High hot leg temperature

5. Manual initiation

Conditions 1 through 5 initiate a block of the pressurizer heaters; trip the reactor and reactor coolant pumps; initiate alignment of the core makeup tank isolation valves for passive injection to the reactor coolant system; and provide a confirmatory open signal to the cold leg balance line isolation valves. The balance line isolation valves are normally open but can be closed by the operator. The confirmatory open signal automatically overrides any cases when the cold leg balance line isolation valves are closed for short periods of time. The motive force for core makeup tank injection is provided by density differences between the fluids in the cold leg balance line and the core makeup tank water.

Condition 3 results from the coincidence of pressurizer level below the Low-2 setpoint in any two of the four divisions. This function can be manually blocked when the pressurizer water level is below the P-12 setpoint. This function is automatically unblocked when the pressurizer water level is above the P-12 setpoint.

Condition 4 is derived from a coincidence of:

- Both steam generator 1 and steam generator 2 wide range level below the Low-2 setpoint (derived from two of the four wide range water level measurement divisions for each steam generator), and

- Two of the four divisions of hot leg temperature above the High (T_{hot}) setpoint

Low-2 steam generator wide range water level can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint, and is automatically unblocked when the reactor coolant average temperature is above the P-9 permissive.

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Condition 5 consists of two momentary controls. Manual actuation of either of the two controls will align the core makeup tanks for injection.

The functional logic relating to core makeup tank injection is illustrated in Figure 7.2-1, sheets 7, 12 and 15. The development of the P-9 permissive is shown on Figure 7.2-1, sheet 7.

7.3.1.2.4 Automatic Depressurization System Actuation

A signal to actuate the first stage of the automatic depressurization system is generated from any of the following conditions:

1. Core makeup tank injection alignment signal (Subsection 7.3.1.2.3) coincident with core makeup tank level less than the Low-3 setpoint in either core makeup tank in two of the four divisions

2. Extended loss of ac power sources (low Class 1E battery charger input voltage)

3. Manual initiation

Any actuation of the first stage of the automatic depressurization system also trips the reactor and reactor coolant pumps, align the core makeup tanks for injection, and actuates the passive residual heat removal heat exchanger.

The automatic depressurization system is arranged to sequentially open four parallel stages of valves. Each of the first three stages consists of two parallel paths with each path containing an isolation valve and a depressurization valve. The first three stages are connected to the pressurizer and discharge into the in-containment refueling water storage tank. The fourth stage paths are connected to the hot legs of the reactor coolant system and discharge to containment.

The first stage isolation valves open on any actuation of the first stage of the automatic depressurization system. The first stage depressurization valves are opened following a preset time delay after the isolation valves are sent a signal to open. No interlocks or permissive signals apply directly to the first stage depressurization. However, some safeguards actuation signals, from which the core makeup tank injection actuation signal is derived, do contain interlock and permissive inputs.

The second stage isolation valves are sent a signal to open following a preset time delay after the first stage isolation valves are sent a signal to open. The second stage depressurization valves are sent a signal to open following a preset time delay after the second stage isolation valves are sent a signal to open, similar to stage one.

Similar to the second stage, the third stage isolation valves are sent a signal to open following a preset time delay after the second stage depressurization valves are sent a signal to open. The third stage depressurization valves are sent a signal to open following a preset time delay after the third stage isolation valves are sent a signal to open.

The fourth stage of the automatic depressurization system consists of four parallel paths. Each of these paths consists of a normally open isolation valve and a depressurization valve. The four paths are divided into two redundant groups with two paths in each group. Within each group, one path is designated to be substage A and the second path is designated to be substage B.

The fourth stage isolation valves receive a confirmatory open signal (nonsafety-related function) following a preset time delay after the first stage depressurization valves are sent a signal to open.

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The fourth stage is actuated upon the coincidence of a Low-6 core makeup tank level and Low reactor coolant system pressure following a preset time delay after the third stage depressurization valves are sent a signal to open. The Low-6 core makeup tank level input is based on the core makeup tank level being less than the Low-6 setpoint in two of the four divisions in either core makeup tank. Upon a fourth stage actuation signal, the substage-A depressurization valves are opened. The signal to open the substage-B depressurization valve is provided following a preset time delay after the substage-A depressurization valves are sent a signal to open. The net effect is to provide a controlled depressurization of the reactor coolant system. In addition to initiating this controlled depressurization sequence, the fourth stage actuation signal also provides a signal that aligns the in-containment refueling water storage tank for injection, as discussed in Subsection 7.3.1.2.2.

A signal to initiate the opening sequence of the fourth stage is also generated upon the occurrence of coincident loop 1 and loop 2 hot leg levels below the Low-4 setpoint. First, the fourth stage isolation valves receive a confirmatory open signal (a nonsafety-related function) following a preset time delay after the first stage depressurization valves are sent a signal to open. Then, the depressurization valves receive an open signal if the coincident loop 1 and loop 2 hot leg levels remain below this setpoint for a duration exceeding a preset time delay. This signal also initiates in-containment refueling water storage tank injection. As discussed in Subsection 7.3.1.2.2, this signal is automatically blocked when the pressurizer water level is above the P-12 setpoint. This reduces the probability of a spurious signal. The block is removed when the core makeup tanks actuation on Low-2 pressurizer level function is manually blocked to allow mid-loop operation.

The automatic actuation of fourth stage ADS isolates the containment and when pressurizer pressure is below the P-11 permissive setpoint also isolates the RNS, as discussed in Subsections 7.3.1.2.1 and 7.3.1.2.20. This function isolates a potential break in the RNS when it is in service.

The fourth stage can also be manually initiated. In this case the manual initiation signal is interlocked to prevent actuation until either the reactor coolant system pressure has decreased below a preset setpoint, or until the signals which control the opening sequence of the first, second, and third stage valves have been generated. As discussed above, the signals to the first, second, and third stage valves are generated based on preset time delays.

The core makeup tank injection alignment signal, which is part of condition 1, is latched-in upon its occurrence. A deliberate operator action is required to reset this latch. This feature is provided so that an automatic depressurization system actuation signal is not cleared by the reset of the safeguards actuation signal as discussed in Subsection 7.3.1.1.

Condition 2 results from the loss of all ac power for a period of time that approaches the 24-hour Class 1E dc battery capability to activate the automatic depressurization system valves. The timed output holds upon restoration of ac power and is manually reset after the batteries are recharged. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. Two sensors are connected to each of the four battery charger inputs. The loss of ac power signal is based on the detection of an undervoltage condition by either of the two sensors connected to two of the four battery chargers.

Condition 3 is achieved via either of two sets of two momentary controls. If both controls of either set are operated simultaneously, actuation of the automatic depressurization system occurs. A two-control simultaneous actuation prevents inadvertent actuation.

The functional logic relating to automatic depressurization operation is illustrated in Figure 7.2-1, sheet 15.

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7.3.1.2.4.1 Block to Prevent ADS and IRWST Injection Spurious Actuation

A number of measures have been taken to reduce the likelihood of spurious actuation of ESF functions in the AP1000 Protection and Safety Monitoring System (PMS) design. Special attention has been given to prevention of spurious Automatic Depressurization System (ADS) and In-containment Refueling Water Storage Tank (IRWST) Injection valve action, since a spurious actuation could result in a release of reactor coolant to containment. In order to prevent such spurious actuations, an ADS and IRWST Injection blocking device is provided that is independent of PMS processor hardware and software. Each division of the PMS contains an independent blocking device that prevents the following ADS Stage 1-4 valves and IRWST Injection valves from actuating unless there is a confirmatory process condition separate from the PMS ADS and IRWST Injection actuation logic:

ADS Stage 1: Depressurization Valve for Stage 1A Valve Set and Depressurization Valve for Stage 1B Valve Set

ADS Stage 2: Depressurization Valve for Stage 2A Valve Set and Depressurization Valve for Stage 2B Valve Set

ADS Stage 3: Depressurization Valve for Stage 3A Valve Set and Depressurization Valve for Stage 3B Valve Set

ADS Stage 4: Block “ARM” signal for all squib valves

IRWST Injection Isolation Valves: Block “ARM” signal for all squib valves

Independence

The ADS and IRWST Injection blocking device is a Class 1E module physically located within each of the PMS divisions. The blocking device is diverse from the PMS processor hardware and software that is used to create the automatic ADS and IRWST Injection actuation signal, which provides the input to the component interface modules for the ADS and IRWST Injection valves. There are no inter-divisional connections between the blocking devices nor will there be any coincidence voting among the blocking devices.

Removing the ADS and IRWST Injection Block

The ADS and IRWST injection blocking device uses Core Makeup Tank (CMT) level and Class 1E Battery Charger (IDS) input voltage to automatically remove this block. The ADS and IRWST injection block in each division receives an input from a level sensor on each CMT that removes the block if the water level in either CMT is below a predetermined setpoint. CMT level is also used to actuate the ADS, therefore the unblock CMT level setpoint is set at a level with enough margin, including instrument uncertainty, to ensure the unblock function occurs prior the need for ADS actuation. Similarly, the block is removed if either of the two undervoltage relay inputs indicates that the voltage has dropped below a predetermined setpoint. The use of two CMT level sensors and two IDS undervoltage inputs in each ADS and IRWST injection blocking device ensures that the blocking device does not significantly affect the reliability of ADS and IRWST injection to actuate when required. In addition, the main control room (MCR)/remote shutdown workstation (RSW) switch unblocks actuation of the ADS and IRWST injection signal when control is transferred to the remote shutdown room (RSR) by the main control room (MCR)/remote shutdown workstation (RSW) switch. The MCR also contains one switch for each division that allows operators to manually remove the ADS and IRWST injection blocks. The logic for the ADS blocking device is shown on Figure 7.2-1 (Sheet 19). The ADS and IRWST injection block must be manually removed if less than two CMTs are allowed to be operable.

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7.3.1.2.5 Reactor Coolant Pump Trip

A signal to trip reactor coolant pumps is generated from any one of the following conditions:

1. Automatic or manual safeguards actuation signal (Subsection 7.3.1.1)
2. Automatic or manual actuation of the first stage of the automatic depressurization system (Subsection 7.3.1.2.4)
3. Low-2 pressurizer level
4. Low-2 steam generator wide range water level coincident with High hot leg temperature
5. Manual initiation of core makeup tank injection (Subsection 7.3.1.2.3)
6. High-2 reactor coolant pump bearing water temperature

Once a signal to trip the reactor coolant pump is generated, the actual tripping of the pump is delayed by a preset time delay.

Condition 3 results from the coincidence of pressurizer level below the Low-2 setpoint in any two of the four divisions. This function can be manually blocked when the pressurizer water level is below the P-12 setpoint. This function is automatically unblocked when the pressurizer water level is above the P-12 setpoint.

Condition 4 is derived from a coincidence of:

- Both steam generator 1 and steam generator 2 wide range level below the Low-2 setpoint (derived from two of the four wide range water level measurement divisions for each steam generator), and
- Two of the four divisions of hot leg temperature above the High (T_{hot}) setpoint

Condition 4 can be manually blocked when reactor coolant system average temperature is below the P-9 setpoint. This function is automatically unblocked when the reactor coolant system average temperature is above the P-9 setpoint.

Condition 6 is derived from a coincidence of two of the four divisions of High-2 reactor coolant pump bearing water temperature for any reactor coolant pump. All of the reactor coolant pumps are tripped simultaneously if Condition 6 is met for the bearing water temperature of any reactor coolant pump. This function is included for equipment protection. The high temperature setpoint and dynamic compensation are the same as used in the High-2 reactor coolant pump bearing water temperature reactor trip (Subsection 7.2.1.1.3) but with the inclusion of preset time delay.

The functional logic relating to the tripping of the reactor coolant pumps is illustrated in Figure 7.2-1, sheets 5, 7, 12, and 15.

7.3.1.2.6 Main Feedwater Isolation

Signals to isolate the main feedwater supply to the steam generators are generated from any of the following conditions:

1. Automatic or manual safeguards actuation (Subsection 7.3.1.1)
2. Manual initiation

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3. High-3 steam generator narrow range water level
4. Low-1 reactor coolant system average temperature coincident with P-4 permissive
5. Low-2 reactor coolant system average temperature coincident with P-4 permissive

Conditions 1, 2, and 3 isolate the main feedwater supply by tripping the main feedwater pumps and closing the main feedwater control, isolation and crossover valves. These conditions also initiate a turbine trip.

Condition 2 consists of two momentary controls. Manual actuation of either of the two controls will trip the turbine and isolate the main feedwater supply. This action also initiates isolation of startup feedwater (Subsection 7.3.1.2.13).

Condition 3 is derived from a coincidence of two of the four divisions of narrow range steam generator water level above the High-3 setpoint for either steam generator. In addition to tripping the turbine and isolating the main feedwater supply, condition 3 also initiates a reactor trip, isolates the startup feedwater supply (Subsection 7.3.1.2.13), and isolates the chemical and volume control system. The functions of turbine trip, main feedwater supply isolation, startup feedwater supply isolation, and chemical volume control system isolation can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint. This function is automatically unblocked when the reactor coolant system average temperature is above the P-9 permissive setpoint.

Condition 4 results from a coincidence of two of the four divisions of reactor loop average temperature (T_{avg}) below the Low-1 setpoint coincident with the P-4 permissive (reactor trip). This condition results in the closure of the main feedwater control valves. The feedwater isolation resulting from this condition may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint.

Condition 5 results from a coincidence of two of the four divisions of reactor loop average temperature (T_{avg}) below the Low-2 setpoint coincident with the P-4 permissive (reactor trip). This condition results in the tripping of the main feedwater pumps and closure of the main feedwater isolation and crossover valves. The feedwater isolation resulting from this condition may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint.

The functional logic relating to the isolation of the main feedwater is illustrated in Figure 7.2-1, sheet 10.

7.3.1.2.7 Passive Residual Heat Removal Heat Exchanger Alignment

A signal to align the passive heat removal heat exchanger to passively remove core heat is generated from any of the following conditions:

1. Core makeup tank injection alignment signal (Subsection 7.3.1.2.3)
2. First stage automatic depressurization system actuation (Subsection 7.3.1.2.4)
3. Low-2 steam generator wide range water level
4. Low-2 steam generator narrow range water level coincident with Low-2 startup feedwater flow coincident with High cold leg temperature (T_{cold} - High)
5. High-3 pressurizer water level
6. Manual initiation

Each of these conditions opens the passive residual heat removal discharge isolation valves, closes the in-containment refueling water storage tank gutter isolation valves, and provides a confirmatory

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open signal to the inlet isolation valve. The inlet isolation valve is normally open but can be closed by the operator. These conditions override any closure signal to this valve and also close the blowdown isolation valves in both steam generators.

Condition 3 results from the coincidence of two of the four divisions of steam generator wide range water level below the Low-2 setpoint in either of the two steam generators. Low-2 steam generator wide range water level can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint, and is automatically unblocked when the reactor coolant average temperature is above the P-9 permissive.

Condition 4 results from the coincidence of two of the four divisions of steam generator narrow range water level below the Low-2 setpoint, after a preset time delay, coincident with a Low-2 startup feedwater flow for both steam generators coincident with High cold leg temperature (T_{cold} - High) in either loop. The Low-2 steam generator narrow range water level also isolates blowdown in the affected steam generator. Low-2 steam generator narrow range water level can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint, and is automatically unblocked when the reactor coolant average temperature is above the P-9 permissive.

Condition 5 results from the coincidence of pressurizer level above the High-3 setpoint in any two of four divisions. This function can be manually blocked when the reactor coolant system pressure is below the P-19 permissive setpoint to permit pressurizer water solid conditions with the plant cold. This function is automatically unblocked when reactor coolant system pressure is above the P-19 setpoint. In addition to actuating the passive residual heat removal heat exchanger, condition 5 initiates a block of the pressurizer heaters.

Condition 6 consists of two momentary controls. Manual actuation of either of the two controls will align the passive residual heat removal heat exchanger initiating heat removal by this path.

The functional logic relating to alignment of the passive residual heat removal heat exchanger is illustrated in Figure 7.2-1, sheet 8. The development of the P-9 permissive is shown on Figure 7.2-1, sheet 7.

7.3.1.2.8 Turbine Trip

A signal to initiate turbine trip is generated from any of the following conditions:

1. Reactor trip (Table 7.3-2, interlock P-4)
2. High-3 steam generator narrow-range water level
3. Manual feedwater isolation (Subsection 7.3.1.2.6)

Each of these conditions initiates a turbine trip to prevent or terminate an excessive cooldown of the reactor or minimizes the potential for equipment damage caused by loss of steam supply to the turbine.

Condition 2 results from a coincidence of two of the four divisions of narrow range steam generator water level above the High-3 setpoint for either steam generator. This function can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint. This function is automatically unblocked when the reactor coolant system average temperature is above the P-9 permissive setpoint.

The functional logic relating to the tripping of the turbine is illustrated in Figure 7.2-1, sheet 14.

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7.3.1.2.9 Containment Recirculation

Signals to align the containment recirculation isolation valves are generated from the following conditions:

1. Low-3 in-containment refueling water storage tank water level in coincidence with fourth stage automatic depressurization system actuation (Subsection 7.3.1.2.4)

2. Manual initiation

3. Extended loss of ac power sources

There are four parallel containment recirculation paths provided to permit the recirculation of the water provided by the in-containment refueling water storage tank. Two of these paths are provided with two isolation valves in series while the remaining two paths are provided with a single isolation valve in series with a check valve.

Conditions 1 and 2 result in the opening of all isolation valves in all four parallel paths. Condition 3 results in the opening of the two isolation valves that are in series with the check valves.

Condition 1 results from the coincidence of two of the four divisions of in-containment refueling water storage tank water level below the Low-3 setpoint, coincident with an automatic fourth stage automatic depressurization system signal.

Condition 2 consists of two sets of two momentary controls. Manual actuation of both controls of either of the two control sets initiates recirculation in all four parallel paths. A two-control simultaneous actuation prevents inadvertent actuation.

Condition 3 results from the loss of all ac power for a period of time that approaches the 24-hour Class 1E dc battery capability to activate the in-containment refueling water storage tank containment recirculation isolation valves. The timed output holds on restoration of ac power and is manually reset after the batteries are recharged. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. Two sensors are connected to each of the four battery charger inputs. The loss of ac power signal is based on the detection of an undervoltage condition by either of the two sensors connected to two of the four battery chargers.

The functional logic relating to activation of the containment recirculation isolation valves is illustrated in Figure 7.2-1, sheets 15 and 16.

7.3.1.2.10 Steam Line Isolation

A signal to isolate the steam line is generated from any one of the following conditions:

1. Manual initiation
2. High-2 containment pressure
3. Low-2 lead-lag compensated steam line pressure
4. High steam line pressure negative rate
5. Low-2 cold leg temperature

The steam line isolation signal closes the main steam line isolation valves and the stop and bypass valves. In addition to manual system-level steam line isolation, steam line isolation valves can be closed individually via the non-safety plant control system.

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Condition 1 consists of two momentary controls. Manual actuation of either of the two controls initiates steam line isolation for both steam generators.

Condition 2 results from the coincidence of two of the four divisions of containment pressure above the High-2 setpoint.

Condition 3 results from the coincidence of two of the four divisions of compensated steam line pressure below the Low-2 setpoint. The steam line pressure signal is lead-lag compensated to improve system response. If the pressure is below this setpoint, in either steam line, both main steam lines are isolated.

Condition 4 results from the coincidence in either steam line of two of the four divisions of rate-lag compensated steam line pressure exceeding the High negative rate setpoint.

Condition 5 results from the coincidence of reactor coolant system cold leg temperature below the Low-2 T_{cold} setpoint in any loop.

Steam line isolation for conditions 3 and 5 may be manually blocked when pressurizer pressure is below the P-11 setpoint and is automatically unblocked when pressurizer pressure is above P-11. Steam line isolation on condition 4 is automatically blocked when pressurizer pressure is above P-11 and is automatically unblocked on the manual blocking of the steam line isolation for conditions 3 and 5. Under all plant conditions, steam line isolation is automatically provided on either Condition 3 or 5, or Condition 4.

The functional logic relating to main steam isolation is illustrated in Figure 7.2-1, sheet 9.

7.3.1.2.11 Steam Generator Blowdown System Isolation

Signals to close the isolation valves of the steam generator blowdown system in both steam generators are generated from the following conditions:

1. Passive residual heat removal heat exchanger alignment signal (Subsection 7.3.1.2.7)
2. Low-2 steam generator narrow range water level
3. Manual and automatic containment isolation (Subsection 7.3.1.2.1)

Condition 2 results from the coincidence of two of the four divisions of steam generator narrow range water level below the Low-2 setpoint. This condition only closes the blowdown system isolation valves of the affected steam generator. Low-2 steam generator narrow range water level can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint, and is automatically unblocked when the reactor coolant average temperature is above the P-9 permissive.

The functional logic relating to steam generator blowdown isolation is illustrated in Figure 7.2-1, sheets 7 and 8. The development of the P-9 permissive is shown on Figure 7.2-1, sheet 7.

7.3.1.2.12 Passive Containment Cooling Actuation

A signal to actuate the passive containment cooling system is generated from either of the following conditions:

1. Manual initiation
2. High-2 containment pressure

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The passive containment cooling actuation signal opens valves that initiate gravity flow of cooling water from the passive containment cooling system water storage tank to the top of the containment shell. The evaporation of the water on the containment shell provides the passive cooling.

Condition 1 consists of two momentary controls. Manual actuation of either of the two controls results in manual actuation of the passive containment cooling system. This action also initiates containment isolation (Subsection 7.3.1.2.1) and isolation of the containment air filtration system (Subsection 7.3.1.2.19).

Condition 2 results from a coincidence of two of the four divisions of containment pressure above the High-2 setpoint. Manual reset is provided to block this actuation signal for passive containment cooling. Separate momentary controls are provided for resetting each division.

The functional logic relating to actuation of the passive containment cooling system is illustrated in Figure 7.2-1, sheet 13.

7.3.1.2.13 Startup Feedwater Isolation

Signals to isolate the startup feedwater supply to the steam generators are generated from either of the following conditions:

1. Low-2 cold leg temperature
2. High-3 steam generator narrow range water level
3. Manual actuation of main feedwater isolation (Subsection 7.3.1.2.6)
4. High steam generator narrow range water level (coincident with P-4 permissive)

Any of these conditions isolates the startup feedwater supply by tripping the startup feedwater pumps and closing the startup feedwater isolation and control valves.

Condition 1 results from the coincidence of reactor coolant system cold leg temperature below the Low-2 T_{cold} setpoint in any loop. Startup feedwater isolation on this condition may be manually blocked when the pressurizer pressure is below the P-11 setpoint. This function is automatically unblocked when the pressurizer pressure is above the P-11 setpoint.

Condition 2 results from a coincidence of two of the four divisions of narrow range steam generator water level above the High-3 setpoint for either steam generator. This function can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint. This function is automatically unblocked when the reactor coolant system average temperature is above the P-9 permissive setpoint.

Condition 3 is discussed in other subsections as noted.

Condition 4 results from a coincidence of two of the four divisions of narrow range steam generator water level above the High setpoint for either steam generator coincident with the P-4 permissive (reactor trip). This function can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint. This function is automatically unblocked when the reactor coolant system average temperature is above the P-9 permissive setpoint.

The functional logic relating to the isolation of the startup feedwater is illustrated in Figure 7.2-1, sheets 9 and 10.

7.3.1.2.14 Boron Dilution Block

Signals to block boron dilution are generated from any of the following conditions:

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1. Excessive increasing rate of source range flux doubling signal
2. Loss of ac power sources (low Class 1E battery charger input voltage)
3. Reactor trip (Table 7.3-2, interlock P-4)

In the event of an excessive increasing rate of source range flux doubling signal, the block of boron dilution is accomplished by closing the chemical and volume control system makeup isolation valves and closing the makeup pump suction valves to the demineralized water storage tanks. This signal also provides a non-safety trip of the makeup pumps. These actions terminate the supply of potentially unborated water to the reactor coolant system as quickly as possible.

In the event of a loss of ac power sources or a reactor trip (as indicated by P-4), the block of boron dilution is accomplished by closing the makeup pump suction valves to the demineralized water storage tanks and aligning the boric acid tank to the suction of the makeup pumps. This permits makeup as needed but ensures that it will be from a borated source that will not reduce the available shutdown margin in the reactor core.

Condition 1 is an average of the source range count rate, sampled at least N times over the most recent time period T_1, compared to a similar average taken at time period T_2 earlier. If the ratio of the current average count rate to the earlier average count rate is greater than a preset value, a partial trip is generated in the division. On a coincidence of excessively increasing source range neutron flux in two of the four divisions, boron dilution is blocked. The Flux Doubling function is also delayed from actuating each time the source range detector's high voltage power is energized to prevent a spurious dilution block due to the short term instability of the processed source range values. This source range flux doubling signal may be manually blocked to permit plant startup and normal power operation when reactor coolant average temperature is above the P-8 setpoint. It is automatically reinstated when reactor power is decreased below the P-6 power level during shutdown or reactor coolant average temperature decreases below the P-8 setpoint.

Below the P-6 setpoint, the source range neutron flux detectors are automatically energized and the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions is automatically reinstated. Above the P-10 setpoint, the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation is automatically blocked and the source range neutron flux detectors are de-energized. Above the P-6 setpoint and below the P-10 setpoint, automatic Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions can be manually blocked or manually unblocked.

The source range flux doubling function can also be manually blocked during shutdown conditions when below the P-8 setpoint. Prior to manually blocking the source range flux doubling function, the operator isolates unborated water source flow paths. When blocked during shutdown conditions, an automatic close signal is also sent to the CVS demineralized water system isolation valves to prevent boron dilution.

Condition 2 results from the loss of ac power. A short, preset time delay is provided to prevent actuation upon momentary power fluctuations; however, actuation occurs before ac power is restored by the onsite diesel generators. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. Two sensors are connected to each of the four battery charger inputs. The loss of ac power signal is based on the detection of an undervoltage condition by each of the two sensors connected to two of the four battery chargers. The two-out-of-four logic is based on an undervoltage to the battery chargers for divisions A or C coincident with an undervoltage to the battery chargers for divisions B or D.

Condition 3 results from a reactor trip as indicated by the P-4 interlock.

The functional logic relating to the boron dilution block is illustrated in Figure 7.2-1, sheets 3 and 15.

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7.3.1.2.15 Chemical and Volume Control System Makeup Isolation

A signal to close the makeup line containment isolation valves of the chemical and volume control system is generated from any of the following conditions:

1. High-2 pressurizer level
2. High-3 steam generator narrow range water level
3. Automatic or manual safeguards actuation signal (Subsection 7.3.1.1) coincident with High-1 pressurizer level
4. High-2 containment radioactivity
5. Manual initiation
6. High steam generator narrow range water level (coincident with P-4 permissive)
7. Excessive increasing rate of source range flux doubling signal

Condition 1 results from the coincidence of pressurizer level above the High-2 setpoint in any two of the four divisions. This function can be manually blocked when the reactor coolant system pressure is below the P-19 permissive setpoint to permit pressurizer water solid conditions with the plant cold and to permit pressurizer level makeup during plant cooldowns. This function is automatically unblocked when reactor coolant system pressure is above the P-19 setpoint.

Condition 2 results from a coincidence of two of the four divisions of narrow range steam generator water level above the High-3 setpoint for either steam generator. This function can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint. This function is automatically unblocked when the reactor coolant system average temperature is above the P-9 permissive setpoint.

Condition 3 results from the coincidence of two of the four divisions of pressurizer level above the High-1 setpoint, coincident with an automatic or manual safeguards actuation.

Condition 4 results from the coincidence of containment radioactivity above the High-2 setpoint in any two of the four divisions.

Condition 5 consists of two momentary controls. This action also initiates auxiliary spray and purification line and zinc/hydrogen addition isolation (Subsection 7.3.1.2.18).

Condition 6 results from a coincidence of two of the four divisions of narrow range steam generator water level above the High setpoint for either steam generator coincident with the P-4 permissive (reactor trip). This function can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint. This function is automatically unblocked when the reactor coolant system average temperature is above the P-9 permissive setpoint.

Condition 7 is an average of the source range count rate, sampled at least N times over the most recent time period T_1, compared to a similar average taken at time period T_2 earlier. If the ratio of the current average count rate to the earlier average count rate is greater than a preset value, a partial trip is generated in the division. On a coincidence of excessively increasing source range neutron flux in two of the four divisions, chemical and volume control system makeup is isolated. The flux doubling function is also delayed from actuating each time the source range detector's high voltage power is energized to prevent a spurious chemical and volume control system makeup isolation due

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to the short-term instability of the processed source range values. This source range flux doubling signal may be manually blocked to permit plant startup and normal power operation when reactor coolant average temperature is above the P-8 setpoint. It is automatically reinstated when reactor power is decreased below the P-6 power level during shutdown or reactor coolant average temperature decreases below the P-8 setpoint. The source range flux doubling function can also be manually blocked during shutdown conditions when below the P-8 setpoint. Prior to manually blocking the source range flux doubling function, the operator isolates unborated water source flow paths. When blocked during shutdown conditions, an automatic close signal is also sent to the CVS demineralized water system isolation valves to prevent inadvertent boron dilution.

Below the P-6 setpoint, the source range neutron flux detectors are automatically energized and the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions is automatically reinstated. Above the P-10 setpoint, the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation is automatically blocked and the source range neutron flux detectors are de-energized. Above the P-6 setpoint and below the P-10 setpoint, automatic Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions can be manually blocked or manually unblocked.

The functional logic relating to chemical and volume control system isolation is illustrated in Figure 7.2-1, sheets 3, 6, and 11.

7.3.1.2.16 Steam Dump Block

Signals to block steam dump (turbine bypass) are generated from either of the following conditions:

1. Low-2 reactor coolant system average temperature
2. Manual initiation

Condition 1 results from a coincidence of two of the four divisions of reactor loop average temperature (T_{avg}) below the Low-2 setpoint. This blocks the opening of the steam dump valves. This signal also becomes an input to the steam dump interlock selector switch for unblocking the steam dump valves used for plant cooldown.

Condition 2 consists of three sets of controls. The first set of two controls selects whether the steam dump system has its normal manual and automatic operating modes available or is turned off. The second set of two controls enables or disables the operations of the Stage 1 cooldown steam dump valves if the reactor coolant average temperature (T_{avg}) is below the Low-2 setpoint. The third set of two controls enables or disables the operation of the Stage 2 cooldown steam dump valves.

The functional logic relating to the steam dump block is illustrated in Figure 7.2-1, sheet 10.

7.3.1.2.17 Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization

Signals to initiate isolation of the main control room, to initiate the air supply, to open the main control room pressure relief isolation valves, and to de-energize nonessential main control room electrical loads are generated from any of the following conditions:

1. High-2 main control room air supply radioactivity level
2. Loss of ac power sources (low Class 1E battery charger input voltage)
3. “Low” main control room differential pressure
4. Manual initiation

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Condition 1 is the occurrence one of two main control room air supply radioactivity monitors detecting the iodine or particulate radioactivity level above the High-2 setpoint.

Condition 2 results from the loss of normal control room ventilation due to a loss of all ac power sources. A preset time delay is provided to permit the restoration of ventilation and ac power from the offsite sources or from the onsite diesel generators before initiation. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. Two sensors are connected to each of the four battery charger inputs. The loss of ac power signal is based on the detection of an undervoltage condition by each of the two sensors connected to two of the four battery chargers. The logic is based on an undervoltage to the battery chargers for divisions A or C coincident with an undervoltage to the battery chargers for divisions B or D.

Condition 3 results from the loss of main control room differential pressure as detected by the pressure boundary differential sensors. One out of two logic is based on main control room differential pressure below the “Low” setpoint for greater than 10 minutes.

In addition, the loss of all ac power sources coincident with main control room isolation will de-energize the main control room air supply radiation monitor sample pumps in order to conserve the battery capacity, maintain room temperature below the equipment qualification limitation, and maintain the assumed heat load for MCR habitability.

Condition 4 consists of two momentary controls. Manual actuation of either of the two controls will result in main control room isolation, air supply initiation, and electrical load de-energization.

The functional logic relating to main control room isolation, air supply initiation, and electrical load de-energization is illustrated in Figure 7.2-1, Sheet 13.

7.3.1.2.18 Auxiliary Spray and Purification Line and Zinc/Hydrogen Addition Isolation

A signal to isolate the auxiliary spray purification, zinc addition, and hydrogen addition lines is generated upon the coincidence of pressurizer level below the Low-1 setpoint in any two of four divisions. This helps to maintain reactor coolant system inventory. This function can be manually blocked when the pressurizer water level is below the P-12 setpoint. This function is automatically unblocked when the pressurizer water level is above the P-12 setpoint. The automatic auxiliary spray isolation signal can be reset by the operator, after actuation of the auxiliary spray isolation valve, by using the reset control. This will allow the operators to use the auxiliary spray to rapidly depressurize the reactor coolant system. The operator can also manually initiate auxiliary spray isolation. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 12.

The auxiliary spray and purification line and zinc/hydrogen addition isolation signal is also generated upon manual actuation of chemical and volume control system makeup isolation (Subsection 7.3.1.2.15).

The zinc addition and hydrogen addition isolation valves are also closed by the containment isolation function as described in Subsection 7.3.1.2.1.

7.3.1.2.19 Containment Air Filtration System Isolation

A signal to isolate the containment air filtration system is generated from any of the following conditions:

1. Automatic or manual safeguards actuation signal (Subsection 7.3.1.1)
2. Manual actuation of containment isolation (Subsection 7.3.1.2.1)
3. Manual actuation of passive containment cooling (Subsection 7.3.1.2.12)

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4. High-1 containment radioactivity

Conditions 1, 2, and 3 are discussed in other subsections as noted.

Condition 4 results from the coincidence of containment radioactivity above the High-1 setpoint in any two of the four divisions.

The manual reset which is provided to block the automatic actuation signal for containment isolation (Subsection 7.3.1.2.1) also resets the containment air filtration system isolation signal generated as a result of condition 1.

No other interlocks or permissive signals apply directly to the containment air filtration system isolation function. Automatic actuation originates from a safeguards actuation (S) signal that does contain interlock and permissive inputs.

The functional logic relating to air filtration system isolation is illustrated in Figure 7.2-1, sheets 11 and 13.

7.3.1.2.20 Normal Residual Heat Removal System Isolation

Signals for isolating the normal residual heat removal system lines are generated from any of the following conditions:

1. Automatic or manual safeguards actuation signal (Subsection 7.3.1.1)
2. High-2 containment radioactivity
3. Automatic fourth stage ADS actuation (Subsection 7.3.1.2.4)
4. Manual initiation

The isolation signal generated as a result of Condition 1 can be manually reset to block the isolation of the normal heat removal system lines. This is done to permit the normal residual heat removal system to operate after the occurrence of a safeguards actuation signal. Separate momentary controls are provided for resetting each division.

Condition 2 results from the coincidence of containment radioactivity above the High-2 setpoint in any two of the four divisions.

Actuation signals for conditions 1 and 2 can be manually blocked when pressurizer pressure is below the P-11 permissive setpoint and can be manually reset to allow use of the system in accident mitigation and plant recovery if desired. These signals are automatically unblocked when the pressurizer pressure is above the P-11 permissive setpoint.

The condition 3 actuation signal is automatically enabled when pressurizer pressure is below the P-11 permissive setpoint and can be manually reset to assist with accident mitigation and plant recovery processes.

Condition 4 consists of two sets of two momentary controls. Manual actuation of both controls of either of two control sets initiates closure of RNS isolation valves. A two-control simultaneous actuation prevents inadvertent actuation.

The functional logic relating to normal residual heat removal system isolation is illustrated in Figure 7.2-1, sheets 13, 15, and 18.

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7.3.1.2.21 Refueling Cavity and Spent Fuel Pool Cooling System Isolation

Signals for isolating the spent fuel pool cooling system lines by closing the spent fuel pool cooling system containment isolation valves are generated from either of the following conditions:

1. Low-2 spent fuel pool level
2. Low IRWST wide range level

Condition 1 results from the coincidence of spent fuel pool level below the Low-2 setpoint in two of three divisions. This helps to maintain the water inventory in the refueling cavity due to line leakage.

Condition 2 results from IRWST wide range level below the Low setpoint in one of two divisions. This helps to maintain the water inventory in the IRWST in the event of a leak in the nonsafety-related spent fuel pool cooling system. Low IRWST wide range level can be manually blocked when the reactor coolant system average temperature is below the P-9 permissive setpoint and the plant is in Mode 6, to allow the spent fuel pool cooling system to transfer the IRWST inventory to the refueling cavity and perform cooling and purification of the refueling cavity. A manual block can also be used when the reactor coolant system average temperature is below the P-9 permissive setpoint to use the spent fuel pool cooling system to cool, purify, sample, or transfer water to or from the IRWST when the level is below the Low IRWST wide range level setpoint. Low IRWST wide range level is automatically unblocked when the reactor coolant average temperature is above the P-9 permissive.

The functional logic relating to refueling cavity and spent fuel pool cooling system isolation is illustrated in Figure 7.2-1, sheet 14. The development of the P-9 permissive is shown on Figure 7.2-1, sheet 7.

7.3.1.2.22 Chemical and Volume Control System Letdown Isolation

A signal to isolate the letdown valves of the chemical and volume control system is generated upon the occurrence of a Low-2 hot leg level in either of the two hot leg loops. This helps to maintain reactor coolant system inventory during mid-loop operation. This signal may be manually blocked by the operator when pressurizer level is above the P-12 setpoint. The functional logic relating to this is illustrated in Figure 7.2-1, sheet 16. These letdown valves are also closed by the containment isolation function as described in Subsection 7.3.1.2.1.

7.3.1.2.23 Pressurizer Heater Trip

Signals for disabling the operation of the pressurizer heaters are generated from any of the following conditions:

1. Core makeup tank injection alignment signal (Subsection 7.3.1.2.3)
2. High-3 pressurizer water level

Division A of the protection and safety monitoring system provides actuation signals to five load center circuit breakers which provide the power feed to five pressurizer heater electrical control centers. When these five power feed breakers are opened, the electrical power is removed from the pressurizer heaters. In addition, each division of the protection and safety monitoring system provides a separate signal to the plant control system. Within the plant control system, these signals are voted two-out-of four and the result is used to open the pressurizer heater circuits at the motor control center. This arrangement provides for complete disabling of the pressurizer heaters, even if a single component failure occurs. Pressurizer heater trip on condition 2 may be manually blocked when wide range RCS pressure is below the P-19 setpoint.

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The functional logic relating to the pressurizer heater block is illustrated in Figure 7.2-1, sheets 6 and 12.

7.3.1.2.24 Steam Generator Relief Isolation

A signal for closing the steam generator power operated relief valves and their block valves is generated from any of the following conditions:

1. Manual initiation
2. Low-2 lead-lag compensated steam line pressure

Condition 2 results from the coincidence of two of the four divisions of compensated steam line pressure below the Low-2 setpoint. The steam line pressure signal is lead-lag compensated to improve system response. The signal closes the steam generator power-operated relief valve and the associated block valve for the affected steam generator. Steam generator relief isolation for condition 2 may be manually blocked when pressurizer pressure is below the P-11 setpoint and is automatically unblocked when pressurizer pressure is above P-11.

The functional logic relating to steam generator relief isolation is illustrated in Figure 7.2-1, sheet 9.

7.3.1.2.25 Component Cooling System Containment Isolation Valve Closure

A closure signal for the component cooling system containment isolation valves is generated from the following:

1. Safeguards actuation signal
2. Reactor coolant pump High-2 bearing water temperature

For condition 1, the safeguards acutation signal is generated based on the incidence of any of the following conditions:

- Automatic or manual safeguards actuation (Subsection 7.1.3.1)

For condition 2, a signal to close the component cooling system containment isolation valves is derived from a coincidence of two of the four divisions of High-2 reactor coolant pump bearing water temperature for any reactor coolant pump. The high temperature setpoint and dynamic compensation are the same as used in the High-2 reactor coolant pump bearing water temperature reactor coolant pump trip (Subsection 7.3.1.2.5, condition 6), but with the inclusion of preset time delay.

The functional logic relating to the tripping of the reactor pumps and the closure of the component cooling system isolation valves is illustrated in Figure 7.2-1, Sheets 5 and 11.

7.3.1.2.26 Containment Vacuum Relief

A signal for opening the containment vacuum relief valves is generated from the following conditions:

- Low-2 containment pressure
- Manual initiation coincident with low containment pressure

Condition 1 results from the incidence of containment pressure reaching the Low-2 setpoint in any two of the four divisions.

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Condition 2 consists of two momentary controls. Manual actuation of either of the two controls will result in opening of the containment vacuum relief valves. Manual containment vacuum relief valve actuation for Condition 2 is interlocked to prevent actuation until the Containment Pressure Low setpoint is reached in any two of the four divisions. After manual containment vacuum relief is initiated, the signal is automatically reset when containment pressure rises above the Low setpoint. Following signal reset, the valves will remain open until closed by a manual or automatic actuation signal.

Either signal will actuate two motor-operated containment isolation valves to break the containment vacuum.

The functional logic relating to containment vacuum relief is illustrated in Figure 7.2-1, Sheet 19.

7.3.1.3 Blocks, Permissives, and Interlocks for Engineered Safety Features Actuation

The interlocks used for engineered safety features actuation are designated as "P-xx" permissives and are listed in Table 7.3-2.

7.3.1.4 Bypasses of Engineered Safety Features Actuation

The channels used in engineered safety features actuation that can be manually bypassed are indicated in Table 7.3-1. A description of this bypass capability is provided in Subsection 7.1.2.9. The actuation logic is not bypassed for test. During tests, the actuation logic is fully tested by blocking the actuation logic output before it results in component actuation.

7.3.1.5 Design Basis for Engineered Safety Features Actuation

The following subsections provide the design bases information for engineered safety features actuation, including the information required by Section 4 of IEEE 603-1991. Engineered safety features are initiated by the protection and safety monitoring system. Those design bases relating to the equipment that initiates and accomplishes engineered safety features are given in WCAP-15776 (Reference 1). The design bases presented here concern the variables monitored for engineered safety features actuation and the minimum performance requirements in generating the actuation signals.

7.3.1.5.1 Design Basis: Generating Station Conditions Requiring Engineered Safety Features Actuation (Paragraph 4.1 of IEEE 603-1991)

The generating station conditions requiring protective action are identified in Table 15.0-6, which summarizes the engineered safety features as they relate to the Condition II, III, or IV events analyzed in Chapter 15.

7.3.1.5.2 Design Basis: Variables, Ranges, Accuracies, and Response Times Used in Engineered Safety Features Actuation (Paragraphs 4.1, 4.2, and 4.4 of IEEE 603-1991)

The variables monitored for engineered safety features actuation are:

- Pressurizer pressure
- Pressurizer water level (with reference leg temperature, pressurizer pressure, and RCS wide range pressure compensation)
- Reactor coolant temperature (T_{hot} and T_{cold}) in each loop
- Containment pressure
- Containment radioactivity level

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- Steam line pressure in each steam line
- Water level in each steam generator (narrow and wide ranges) (with steam line pressure compensation)
- Source range neutron flux
- Core makeup tank level
- Reactor coolant level in each of the two hot legs
- Loss of ac power sources (low Class 1E battery charger input voltage)
- In-containment refueling water storage tank level
- Main control room supply air radioactivity level
- Main control room differential pressure
- Reactor coolant pump bearing water temperature
- Startup feedwater flow
- Spent fuel pool level
- Reactor coolant pressure in each of the two hot legs

Subsections 7.3.1.1 and 7.3.1.2 discuss levels that result in engineered safety features actuation. The engineered safety features actuation and trip setpoints are maintained by the setpoint program, which is described in the Technical Specifications.

Ranges and response times for the variables used in engineered safety features actuation are listed in Table 7.3-4. Channel accuracies are accounted for in the setpoint methodology. The time response is the maximum allowable time for an actuation signal to reach the necessary components. Any intentional time delays in the system are accounted for separately in the accident analyses. Some PMS response times are allocated rather than measured.

7.3.1.5.3 Design Basis: Spatially Dependent Variables Used for Engineered Safety Features Actuation (Paragraph 4.6 of IEEE 603-1991)

Spatially dependent variables are discussed in Subsection 7.2.1.2.3.

7.3.1.5.4 Design Basis: Limits for Engineered Safety Features Parameters in Various Reactor Operating Modes (Paragraph 4.3 of IEEE 603-1991)

During startup or shutdown, various engineered safety features actuation can be manually blocked. These functions are listed in Table 7.3-1.

During testing or maintenance of the protection and safety monitoring system, certain channels used for engineered safety features may be bypassed. Although no setpoints are changed for bypassing, the logic is automatically adjusted, as described in Subsection 7.3.1.4. The safeguards channels that can be bypassed in the protection and safety monitoring system are listed in Table 7.3-1.

7.3.1.5.5 Design Basis: Engineered Safety Features for Malfunctions, Accidents, Natural Phenomena, or Credible Events (Paragraph 4.7 and 4.8 of IEEE 603-1991)

The accidents that the various engineered safety features are designed to mitigate are detailed in Chapter 6 and Chapter 15. Table 15.0-6 contains a summary listing of the engineered safety features actuated for various Condition II, III, or IV events. It relies on provisions made to protect equipment against damage from natural phenomena and credible internal events. Consequently, there are no engineered safety features actuated by the protection and safety monitoring system to mitigate the consequences of events such as fires.

Functional diversity is used in determining the actuation signals for engineered safety features. For example, a safeguards actuation signal is generated from High-2 containment pressure, Low-3

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pressurizer pressure, and Low-2 compensated steam line pressure. Engineered safety features are not normally actuated by a single signal. The extent of this diversity is seen from the initiating signals presented in Subsections 7.3.1.1 and 7.3.1.2. Table 7.3-1 also lists the engineered safety features signals and the conditions that result from their actuation.

Redundancy provides confidence that engineered safety features are actuated on demand, even when the protection and safety monitoring system is degraded by a single random failure. The single-failure criterion is met even when engineered safety features channels are bypassed.

7.3.1.6 System Drawings

Functional diagrams are provided in Figure 7.2-1.

7.3.2 Analysis for Engineered Safety Features Actuation

7.3.2.1 Failure Modes and Effects Analyses

The AP1000 failure modes and effects analysis (Appendix 7B) examines failures of the protection and safety monitoring system. This analysis concludes that the protection system maintains safety functions during single point failures.

7.3.2.2 Conformance of Engineered Safety Features to the Requirements of IEEE 603-1991

The discussions presented in this subsection address only the functional aspects of actuating engineered safety features. Requirements addressing equipment in the protection and safety monitoring system are presented in WCAP-15776 (Reference 1).

7.3.2.2.1 Conformance to the General Functional Requirements for Engineered Safety Features Actuation (Section 5 of IEEE 603-1991)

The protection and safety monitoring system automatically generates an actuation signal for an engineered safety feature whenever a monitored condition reaches a preset value. The specific engineered safety features actuation functions are listed in Table 7.3-1 and are discussed in Subsection 7.3.1.2.

Table 7.3-4 lists the ranges and response times of the parameters monitored. Channel accuracies are accounted for in the setpoint methodology. The engineered safety features, in conjunction with a reactor trip, protect against damage to the core and reactor coolant system components, as well as maintain containment integrity following a Condition II, III, or IV event. Table 15.0-6 summarizes the events that normally result in the initiation of engineered safety features. The setpoints that actuate engineered safety features are maintained by the setpoint program, which is described in the Technical Specifications.

7.3.2.2.2 Conformance to the Single Failure Criterion for Engineered Safety Features Actuation (Paragraph 5.1 of IEEE 603-1991)

A single failure in the protection and safety monitoring system does not prevent an actuation of the engineered safety features when the monitored condition reaches the preset value that requires the initiation of an actuation signal. The single failure criterion is met even when one division of the ESF coincidence logic is being tested, as discussed in Subsection 7.1.2.9, or when there is a bypass condition in connection with test or maintenance of the protection and safety monitoring system.

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7.3.2.2.3 Conformance to the Requirements for Channel Independence of the Engineered Safety Features Actuation (Paragraph 5.6.1 of IEEE 603-1991)

A discussion of channel independence is presented in WCAP-15776 (Reference 1). The signals to initiate division A of the engineered safety features are electrically isolated from the signals to initiate the redundant divisions (B, C, and D). Divisions of the safeguards actuation system are electrically independent and redundant, as are the power supplies for the divisions up to and including the final actuated equipment.

7.3.2.2.4 Conformance to the Requirements Governing Control and Protection System Interaction of the Engineered Safety Features Actuation (Paragraphs 5.6.3.1, 5.6.3.3, and 6.3.1 of IEEE 603-1991)

Discussions on this subject are presented in WCAP-15776 (Reference 1).

7.3.2.2.5 Derivation of System Input for Engineered Safety Features Actuation (Paragraph 6.4 of IEEE 603-1991)

To the extent feasible and practical, the protection and safety monitoring system inputs used to actuate engineered safety features are derived from signals that are direct measures of the desired parameters. The parameters are listed in Table 7.3-4.

7.3.2.2.6 Capability for Sensor Checks and Equipment Test and Calibration of the Engineered Safety Features Actuation (Paragraphs 5.7 and 6.5 of IEEE 603-1991)

The discussion of system testability provided in Section 7.1 is applicable to the sensors, signal processing, and actuation logic that initiate engineered safety features actuation.

The testing program meets Regulatory Guide 1.22 as discussed in WCAP-15776 (Reference 1). The program is as follows:

- Prior to initial plant operations, engineered safety features tests are conducted.
- Subsequent to initial startup, engineered safety features tests are conducted during each regularly scheduled refueling outage.
- During operation of the reactor, the protection and safety monitoring system is tested as described in Subsection 7.1.2.11. In addition, the engineered safety features final actuators, whose operation is compatible with continued plant operation, are tested periodically at power.
- Continuity of the wiring is verified for devices that cannot be tested at power without damaging or upsetting the plant. Operability of the final actuated equipment is demonstrated at shutdown.

During reactor operation, the basis for acceptability of engineered safety features actuation includes the successful completion of the tests performed on the protection and safety monitoring system. Process indications are used to verify operability of sensors.

7.3.2.2.7 Conformance to Requirements on Bypassing Engineered Safety Features Actuation Functions (Paragraph 5.8, 5.9, 6.6, and 6.7 of IEEE 603-1991)

Discussions on bypassing are provided in WCAP-15776 (Reference 1) and Subsection 7.3.1.4.

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7.3.2.2.8 Conformance to the Requirement for Completion of Engineered Safety Features Actuation Once Initiated (Paragraph 5.2 of IEEE 603-1991)

Once initiated, engineered safety features proceed to completion.

Equipment actuated on a safeguards actuation signal cannot be returned to its previous position for a predetermined time period following initiation of the safeguards actuation signal. A block of the automatic safeguards signal is permitted at this time, if the reactor is tripped. This interlock is shown in Figure 7.2-1, sheet 11.

Resetting a system-level safeguards signal does not terminate any safeguards function. Rather, it permits the operator to individually reposition equipment. Equipment cannot be reset until the system-level signal is reset.

7.3.2.2.9 Conformance to the Requirement to Provide Manual Initiation at the System-Level for All Safeguards Actuation (Paragraph 6.2 of IEEE 603-1991)

Manual initiation at the system-level exists for the engineered safety features actuation. These system-level manual initiations are discussed in Subsections 7.3.1.1 and 7.3.1.2.

As a minimum, two controls are provided for each system-level manual initiation so that the protective function can be manually initiated at the system-level, despite a single random failure in one control. In certain applications, such as automatic depressurization, two pairs of controls are provided. One pair must be actuated simultaneously. This reduces the likelihood of inadvertent actuation while providing a design that meets the single failure criterion.

7.3.3 Combined License Information

This section contained no requirement for information.

7.3.4 References

1. WCAP-15776, “Safety Criteria for the AP1000 Instrument and Control Systems,” Revision 1, July 2019.

2. Not Used.

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Table 7.3-1 (Sheet 1 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
1. Safeguards Actuation Signal (Figure 7.2-1, Sheets 9 and 11)			
a. Low-3 pressurizer pressure	4	2/4-BYP[1]	Can be manually blocked on presence of P-3 Block automatically removed on absence of P-3 Manual block permitted below P-11 Automatically unblocked above P-11
b. Low-2 lead-lag compensated steam line pressure	4/steam line	2/4-BYP[1] in either steam line	Can be manually blocked on presence of P-3 Block automatically removed on absence of P-3 Manual block permitted below P-11 Automatically unblocked above P-11
c. Low-2 cold leg temperature (Low-2 T_{cold})	4/loop	2/4-BYP[1] either loop[6]	Can be manually blocked on presence of P-3 Block automatically removed on absence of P-3 Manual block permitted below P-11 Automatically unblocked above P-11
d. High-2 containment pressure	4	2/4-BYP[1]	Can be manually blocked on presence of P-3 Block automatically removed on absence of P-3
e. Manual safeguards initiation	2 controls	1/2 controls	None
2. Containment Isolation (Figure 7.2-1 Sheets 11, 13, and 15)			
a. Automatic or manual safeguards actuation signal	(See items 1a through 1e)		
b. Automatic fourth stage ADS actuation[(8)]	(See items 3d through 3f)		
c. Manual initiation	2 controls	1/2 controls	None
d. Manual initiation of passive containment cooling	(See item 10a)		

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Table 7.3-1 (Sheet 2 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
3. Automatic Depressurization System[9] (Figure 7.2-1, Sheets 15 and 19) **(Initiate Stages 1, 2, and 3)**			
a. Core makeup tank injection coincident with	(See items 6a through 6e)		
Core makeup tank level less than Low-3 setpoint	4/tank	2/4-BYP[1] either tank[2]	None
b. Extended undervoltage to Class 1E battery chargers[8]	2/charger	1/2 per charger and 2/4 chargers	None
c. Stages 1, 2, and 3 manual initiation	4 controls	2/4 controls[3]	None
(Initiate Stage 4)			
d. Stage 4 manual initiation coincident with one of the following two conditions:	4 controls	2/4 controls[3]	None
Low[10] reactor coolant system pressure or	4	2/4 BYP[1]	None
3rd stage depressurization			
e. Core makeup tank level less than Low-6 setpoint coincident with	4/tank	2/4 BYP[1] either tank[2]	None
Low[10] reactor coolant system pressure and coincident with	4	2/4 BYP[1]	None
3rd stage depressurization			
f. Coincident loop 1 and loop 2 Low-4 hot leg level (after delay)	1 per loop	2/2[1]	Manual unblock permitted below P-12 Automatically blocked above P-12

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Table 7.3-1 (Sheet 3 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
4. Main Feedwater Isolation (Figure 7.2-1, Sheet 10) (Closure of Control Valves)			
a. Safeguards actuation signal (automatic or manual)	(See items 1a through 1e)		
b. Manual initiation	2 controls	1/2 controls	None
c. High-3 steam generator narrow range level	4/steam generator	2/4-BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
d. Low[10] reactor coolant temperature (Low T_{avg}) coincident with	2/loop	2/4 -BYP[1]	Manual block permitted below P-11 Automatically unblocked above P-11
Reactor trip (P-4)	1/division	2/4	None
(Trip of Main Feedwater Pumps and Closure of Isolation and Crossover Valves)			
a. Safeguards actuation signal (automatic or manual)	(See items 1a through 1e)		
b. Manual initiation	2 controls	1/2 controls	None
c. High-3 steam generator narrow range level	4/steam generator	2/4-BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
d. Low reactor coolant temperature (Low-2 T_{avg}) coincident with	2/loop	2/4 - BYP[1]	Manual block permitted below P-11 Automatically unblocked above P-11
Reactor trip (P-4)	1/division	2/4	None
5. Reactor Coolant Pump Trip (Figure 7.2-1, Sheets 5, 7, 12, and 15) **(Trips All Reactor Coolant Pumps)**			
a. Safeguards actuation signal (automatic or manual)	(See items 1a through 1e)		
b. Automatic reactor coolant system depressurization (first stage)	(See items 3a through 3c)		
c. Low-2 pressurizer level	4	2/4-BYP[1]	Manual block permitted below P-12 Automatically unblocked above P-12
d. Low-2 steam generator wide range water level coincident with	4/steam generator	2/4-BYP[1] in both steam generators	Manual block permitted below P-9 Automatically unblocked above P-9
High hot leg temperature (High T_{hot})[(8)]	2/loop	2/4-BYP[1]	None
e. Manual core makeup tank initiation	(See item 6e)		
f. High-2 (lead-lag compensated) reactor coolant pump bearing water temperature	4/pump	2/4-BYP[1] in affected pump	None

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Table 7.3-1 (Sheet 4 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
6. Core Makeup Tank Injection (Figure 7.2-1, Sheets 7, 12 and 15)			
a. Safeguards actuation signal (automatic or manual)	(See items 1a through 1e)		
b. Automatic reactor coolant system depressurization (first stage)	(See items 3a through 3c)		
c. Low-2 pressurizer level	4	2/4-BYP[1]	Manual block permitted below P-12 Automatically unblocked above P-12
d. Low-2 steam generator wide range water level coincident with	4/steam generator	2/4-BYP[1] in both steam generators	Manual block permitted below P-9 Automatically unblocked above P-9
High hot leg temperature (High T_{hot})[(8)]	2/loop	2/4-BYP[1]	None
e. Manual initiation	2 controls	1/2 controls	None
7. Turbine Trip (Figure 7.2-1, Sheet 14)			
a. Manual feedwater isolation	(See item 4b)		
b. Reactor trip (P-4)	1/division	2/4	None
c. High-3 steam generator narrow range level	4/steam generator	2/4-BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
8. Steam Line Isolation (Figure 7.2-1, Sheet 9)			
a. Manual initiation	2 controls	1/2 controls	None
b. High-2 containment pressure	4	2/4-BYP[1]	None
c. Low-2 lead-lag compensated steam line pressure[4]	4/steam line	2/4-BYP[1] in either steam line	Manual block permitted below P-11 Automatically unblocked above P-11
d. High steam line negative pressure rate	4/steam line	2/4-BYP[1] in either steam line[7]	Manual unblock permitted below P-11 Automatically blocked above P-11
e. Low-2 cold leg temperature (Low-2 T_{cold})	4/loop	2/4-BYP[1] either loop[6]	Manual block permitted below P-11 Automatically unblocked above P-11
9. Steam Generator Blowdown System Isolation (Figure 7.2-1 Sheets 7 and 8)			
a. Passive residual heat removal heat exchanger actuation	(See items 12a through 12f)		
b. Low-2 steam generator narrow range water level	4/steam generator	2/4 BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
c. Containment isolation	(See items 2a through 2c)		
10. Passive Containment Cooling Actuation (Figure 7.2-1, Sheet 13)			
a. Manual initiation	2 controls	1/2 controls	None
b. High-2 containment pressure	4	2/4-BYP[1]	None

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Table 7.3-1 (Sheet 5 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
11. Startup Feedwater Isolation (Figure 7.2-1, Sheets 9 and 10)			
a. Low-2 cold leg temperature (Low-2 T_{cold})	4/loop	2/4-BYP[1] either loop[6]	Manual block permitted below P-11 Automatically unblocked above P-11
b. High-3 steam generator narrow range water level	4/steam generator	2/4-BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
c. Manual initiation of main feedwater isolation		(See item 4b)	
d. High[10] steam generator narrow range level coincident with	4/steam generator	2/4-BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
Reactor trip (P-4)	1/division	2/4	None
12. Passive Residual Heat Removal (Figure 7.2-1, Sheet 8)			
a. Manual initiation	2 controls	1/2 controls	None
b. Low-2 steam generator narrow range water level coincident with	4/steam generator	2/4-BYP[1] in both steam generators	Manual block permitted below P-9 Automatically unblocked above P-9
Low-2 startup feedwater flow	2/feedwater line	1/2[1] in both feedwater lines	None
High cold leg temperature (T_{cold} - High)	4/loop	2/4 BYP[1] either loop[6]	None
c. Low-2 steam generator wide range water level	4/steam generator	2/4-BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
d. Core makeup tank injection	(See Items 6a through 6e)		
e. Automatic reactor coolant system depressurization (first stage)	(See items 3a through 3c)		
f. High-3 pressurizer level	4	2/4-BYP[1]	Manual block permitted below P-19 Automatically unblocked above P-19
13. Block of Boron Dilution (Figure 7.2-1, Sheets 3 and 15)			
a. Flux doubling calculation	4	2/4-BYP[1]	Manual block permitted above P-6 and above P-8 Automatically unblocked below P-6 or below P-8 Manual block permitted below P-8; demineralized water system isolation valves signaled closed when blocked below P-8
b. Undervoltage to Class 1E battery chargers[(8)]	2/charger	2/2 per charger and 2/4 chargers[5]	None
c. Reactor trip (P-4)	1/division	2/4	None

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Table 7.3-1 (Sheet 6 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
14. Chemical and Volume Control System Makeup Isolation (See Figure 7.2-1, Sheets 3, 6, and 11)			
a. High-2 pressurizer water level	4	2/4-BYP[1]	Automatically unblocked above P-19 Manual block permitted below P-19
b. High-3 steam generator narrow range level	4/steam generator	2/4-BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
c. Automatic or manual safeguards actuation signal coincident with	(See items 1a through 1e)		
High[10] pressurizer water level	4	2/4-BYP[1]	None
d. High-2 containment radioactivity	4	2/4-BYP[1]	None
e. Manual initiation	2 controls	1/2 controls	None
f. Flux doubling calculation	4	2/4-BYP[1]	Manual block permitted above P-6 and above P-8 Automatically unblocked below P-6 or below P-8 Manual block permitted below P-8; demineralized water system isolation valves signaled closed when blocked below P-8
g. High[10] steam generator narrow range level coincident with	4/steam generator	2/4-BYP[1] in either steam generator	Manual block permitted below P-9 Automatically unblocked above P-9
Reactor trip (P-4)	1/division	2/4	None
15. Steam Dump Block (Figure 7.2-1, Sheet 10)[(8)]			
a. Low reactor coolant temperature (Low-2 T_{avg})	2/loop	2/4-BYP[1]	None
b. Mode control	2 controls	1/division	None
c. Manual stage 1 cooldown control	2 controls	1/division	None
d. Manual stage 2 cooldown control	2 controls	1/division	None
16. Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization (Figure 7.2-1, Sheet 13)[(13)]			
a. High-2 main control room air supply iodine or particulate radiation	2	1/2[1]	None
b. Extended undervoltage to Class 1E 24-hour battery chargers[(13)]	2/charger	2/2 per charger and 2/4 chargers[5]	None
c. Extended Low main control room differential pressure	2	1/2[1]	None
d. Manual initiation[(8)]	2 controls	1/2 controls	None

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Table 7.3-1 (Sheet 7 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
17. Auxiliary Spray and Purification Line and Zinc/Hydrogen Addition Isolation (Figure 7.2-1, Sheet 12)			
a. Low[10] pressurizer level	4	2/4-BYP[1]	Manual block permitted below P-12 Automatically unblocked above P-12
b. Manual initiation of chemical and volume control system makeup isolation	(See item 14e)		
c. Manual initiation of auxiliary spray isolation[(8,12)]	1	1/1	None
d. Containment isolation[(11)]	(See Items 2a through 2c)		
18. Containment Air Filtration System Isolation (Figure 7.2-1, Sheets 11 and 13)			
a. Containment isolation	(See items 2a through 2c)		
b. High[10] containment radioactivity	4	2/4-BYP[1]	None
c. N/A	2	N/A	For containment vacuum relief valves only – close on inside containment purge isolation valve not closed
19. Normal Residual Heat Removal System Isolation (Figure 7.2-1, Sheets 13, 15, and 18)			
a. Automatic or manual safeguards actuation signal	(See items 1a through 1e)		
b. High-2 containment radioactivity	4	2/4-BYP[1]	Manual block permitted below P-11 Automatically unblocked above P-11
c. Automatic fourth stage ADS actuation[(8)]	4	2/4-BYP[1]	Automatically enabled below P-11 Automatically blocked above P-11
d. Manual initiation	4 controls	2/4 controls[3]	None
20. Refueling Cavity and SFS Isolation (Figure 7.2-1, Sheet 14)			
a. Low-2 spent fuel pool level	3	2/3[1]	None
b. Low IRWST wide range level	2	1/2[1]	Manual block permitted below P-9 Automatically unblocked above P-9
21. Open In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valves[(9)] (Figure 7.2-1, Sheets 12, 16, and 19)			
a. Automatic reactor coolant system depressurization (fourth stage)	(See items 3d and 3e)		
b. Manual initiation	4 controls	2/4 controls[3]	None

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Table 7.3-1 (Sheet 8 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
22. Open Containment Recirculation Valves In Series with Check Valves (Figure 7.2-1, Sheet 15 and 16)			
a. Extended undervoltage to Class 1E battery chargers(8)	2/charger	1/2 per charger and 2/4 chargers	None
23. Open All Containment Recirculation Valves (Figure 7.2-1, Sheet 16)			
a. Automatic reactor coolant system depressurization (fourth stage) coincident with		(See items 3d through 3f)	
Low IRWST lower narrow range level (Low-3 setpoint)	4	2/4-BYP[1]	None
b. Manual initiation	4 controls	2/4 controls[3]	None
24. Chemical and Volume Control System Letdown Isolation (Figure 7.2-1, Sheet 16)			
a. Low-2 hot leg level	1 per loop	1/2[1]	Manual block permitted above P-12 Automatically unblocked below P-12
25. Pressurizer Heater Trip (Figure 7.2-1, Sheets 6 and 12)			
a. Core makeup tank injection	(See items 6a through 6e)		
b. High-3 pressurizer level	4	2/4-BYP[1]	Manual block permitted below P-19 Automatically unblocked above P-19
26. Steam Generator Relief Isolation (Figure 7.2-1, Sheet 9)			
a. Manual initiation	2 controls	1/2 controls	None
b. Low-2 lead-lag compensated steam line pressure[4]	4/steam line	2/4-BYP[1] in either steam line	Manual block permitted below P-11 Automatically unblocked above P-11
27. Close Component Cooling System Containment Isolation Valves (Figure 7.2-1, Sheets 5 and 11)			
a. High-2 (lead-lag compensated) reactor coolant pump bearing water temperature	4/pump	2/4-BYP[1] in affected pump	None
b. Safeguards actuation signal (automatic or manual)	(See items 1.a through 1.e)		

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Table 7.3-1 (Sheet 9 of 9)
Engineered Safety Features Actuation Signals

Actuation Signal	No. of Divisions/ Controls	Actuation Logic	Permissives and Interlocks
28. Containment Vacuum Relief (Figure 7.2-1, Sheet 19)			
a. Low-2 containment pressure	4	2/4-BYP[1]	None
b. Manual initiation coincident with the following condition:	2 controls	1/2 controls	None
Low containment pressure	4	2/4-BYP[1]	None

Notes:

1. Capable of bypass (examples: The 2 out of 4 logic becomes 2 out of 3, the 2 out of 3 logic becomes 2 out of 2, and the 1 out of 2 logic is 1 out of 1 with one bypass. Note that 2 out of 2 logic becomes 2 out of 1 which renders the function inoperable). 2/4-BYP, where identified, indicates bypass logic specifically for 2/4 coincidence logic functions.
2. Any two channels from either tank not in same division.
3. Two associated controls must be actuated simultaneously.
4. Also, closes power-operated relief block valve of respective steam generator.
5. The logic is based on undervoltage to the battery chargers for divisions A or C coincident with an undervoltage to the battery chargers for divisions B or D.
6. Any two channels from either loop not in same division.
7. Any two channels from either line not in same division.
8. This function does not meet the 10 CFR 50.36(c)(2)(ii) criteria and is not included in the Technical Specifications.
9. Spurious ADS and IRWST injection actuations, both automatic and manual, are blocked as described in Subsection 7.3.1.2.4.1.
10. Low and Low-1 are considered to be equivalent terms when referring to the first low setpoint designator. High and High-1 are considered to be equivalent terms when referring to the first high setpoint designator.
11. Applies to only the Zinc/Hydrogen Addition Isolation.
12. Applies to only the Auxiliary Spray Isolation.
13. De-energization of Main Control Room air supply radiation monitor sample pumps occurs on an extended undervoltage to Class 1E 24-hour battery chargers as shown in Figure 7.2-1, Sheet 13.

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Table 7.3-2 (Sheet 1 of 7)
Interlocks for Engineered Safety Features Actuation System

Designation	Derivation	Function
P-3	Reactor trip breaker open	Permits manual reset of safeguards actuation signal to block automatic safeguards actuation
$\overline{\text{P-3}}$	Reactor trip breakers closed	Automatically resets the manual block of automatic safeguards actuation
P-4	Reactor trip initiated or reactor trip breakers open	(a) Permits closing of the main feedwater control valves on low reactor coolant temperature (b) Permits closing of the main feedwater isolation valves, closing of the crossover leg valves, and tripping of the main feedwater pumps on Low-2 reactor coolant temperature (c) Trips turbine (d) Blocks boron dilution (e) Permits isolation of the startup feedwater valves on high narrow range water level in either steam generator (f) Permits isolation of the CVS makeup valves on high narrow range water level in either steam generator
$\overline{\text{P-4}}$	No reactor trip initiated and reactor trip breakers closed	(a) Prevents closing of the main feedwater control valves on low reactor coolant temperature (b) Prevents closing of the main feedwater isolation valves, closing of the crossover leg valves, and tripping of the main feedwater pumps on Low-2 reactor coolant temperature (c) Removes demand for turbine trip (d) Removes demand for boron dilution block (e) Prevents isolation of the startup feedwater valves on high narrow range water level in either steam generator (f) Prevents isolation of the CVS makeup valves on high narrow range water level in either steam generator
P-6	Intermediate range neutron flux channels above setpoint	Permits manual block of flux doubling actuation of the boron dilution block and CVS isolation Functions
$\overline{\text{P-6}}$	Intermediate range neutron flux channels below setpoint	Automatically resets the manual block of flux doubling actuation of the boron dilution block and CVS isolation Functions
P-8	Reactor coolant average temperature above setpoint	Permits manual block of flux doubling actuation of the boron dilution block and CVS isolation Functions

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Table 7.3-2 (Sheet 2 of 7)
Interlocks for Engineered Safety Features Actuation System

Designation	Derivation	Function
$\overline{\text{P-8}}$	Reactor coolant average temperature below setpoint	(a) Automatically resets the manual block of flux doubling actuation of the boron dilution block and CVS isolation Functions (b) Permits manual block of flux doubling actuation of the boron dilution block and CVS isolation Functions; signals the demineralized water system isolation valves closed if flux doubling actuation of the boron dilution block is blocked below P-8
P-9	Reactor coolant average temperature above setpoint	(a) Prevents manual block of core makeup tank actuation on Low-2 steam generator wide range water level and automatically unblocks (b) Prevents manual block of steam generator blowdown system isolation on Low-2 steam generator narrow range water level and automatically unblocks (c) Prevents manual block of passive residual heat removal actuation on Low-2 steam generator wide range water level and on Low-2 steam generator narrow range water level and automatically unblocks (d) Prevents manual block of refueling cavity and spent fuel pool cooling system isolation on low IRWST wide range level and automatically unblocks (e) Prevents manual block of startup feedwater isolation on high narrow range steam generator level coincident with reactor trip (P-4) and automatically unblocks (f) Prevents manual block of CVS isolation on high narrow range steam generator level coincident with reactor trip (P-4) and automatically unblocks (g) Prevents manual block of startup feedwater isolation on High-3 narrow range steam generator level and automatically unblocks (h) Prevents manual block of CVS isolation on High-3 narrow range steam generator level and automatically unblocks (i) Prevents manual block of main feedwater isolation on High-3 narrow range steam generator level and automatically unblocks (j) Prevents manual block of turbine trip on High-3 narrow range steam generator level and automatically unblocks (k) Prevents manual block of reactor coolant pump trip on Low-2 steam generator wide range water level and automatically unblocks

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Table 7.3-2 (Sheet 3 of 7)
Interlocks for Engineered Safety Features Actuation System

Designation	Derivation	Function
$\overline{\text{P-9}}$	Reactor coolant average temperature below setpoint	(a) Permits manual block of core makeup tank actuation on Low-2 steam generator wide range water level (b) Permits manual block of steam generator blowdown system isolation on Low-2 steam generator narrow range water level (c) Permits manual block of passive residual heat removal actuation on Low-2 steam generator wide range water level and on Low-2 steam generator narrow range water level (d) Permits manual block of refueling cavity and spent fuel pool cooling system isolation on low IRWST wide range level (e) Permits manual block of startup feedwater isolation on high narrow range steam generator level coincident with reactor trip (P-4) (f) Permits manual block of CVS isolation on high narrow range steam generator level coincident with reactor trip (P-4) (g) Permits manual block of startup feedwater isolation on High-3 narrow range steam generator level (h) Permits manual block of CVS isolation on High-3 narrow range steam generator level (i) Permits manual block of main feedwater isolation on High-3 narrow range steam generator level (j) Permits manual block of turbine trip on High-3 narrow range steam generator level (k) Permits manual block of reactor coolant pump trip on Low-2 steam generator wide range water level

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Table 7.3-2 (Sheet 4 of 7)
Interlocks for Engineered Safety Features Actuation System

Designation	Derivation	Function
P-11	Pressurizer pressure below setpoint	(a) Permits manual block of safeguards actuation on Low-3 pressurizer pressure, Low-2 compensated steam line pressure, or Low-2 reactor coolant inlet temperature (b) Permits manual block of steam line isolation on Low-2 reactor coolant inlet temperature (c) Permits manual block of steam line isolation and steam generator power-operated relief valve block valve closure on Low-2 compensated steam line pressure (d) Coincident with manual actions of (b) or (c), automatically unblocks steam line isolation on high negative steam line pressure rate (e) Permits manual block of main feedwater control valve isolation on low reactor coolant temperature coincident with reactor trip (P-4) (f) Permits manual block of main feedwater isolation actuation, crossover leg valve isolation, and main feedwater pump trip on Low-2 reactor coolant temperature coincident with reactor trip (P-4) (g) Permits manual block of startup feedwater isolation on Low-2 reactor coolant inlet temperature (h) Permits manual block of normal residual heat removal system isolation on High-2 containment radioactivity. (i) Permits RNS containment isolation on automatic ADS stage 4 actuation.

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Table 7.3-2 (Sheet 5 of 7)
Interlocks for Engineered Safety Features Actuation System

Designation	Derivation	Function
$\overline{\text{P-11}}$	Pressurizer pressure above setpoint	(a) Prevents manual block of safeguards actuation on Low-3 pressurizer pressure, Low-2 compensated steam line pressure, or Low-2 reactor coolant inlet temperature (b) Prevents manual block of steam line isolation on Low-2 reactor coolant inlet temperature (c) Prevents manual block of steam line isolation and steam generator power-operated relief valve block valve closure on Low-2 compensated steam line pressure (d) Automatic block of steam line isolation on high negative steam line pressure rate (e) Prevents manual block of main feedwater control valve isolation on low reactor coolant temperature coincident with reactor trip (P-4) (f) Prevents manual block of main feedwater isolation actuation, crossover leg valve isolation, and main feedwater pump trip on Low-2 reactor coolant temperature coincident with reactor trip (P-4) (g) Prevents manual block of startup feedwater isolation on Low-2 reactor coolant inlet temperature (h) Prevents manual block of normal residual heat removal system isolation on High-2 containment radioactivity (i) Prevents RNS containment isolation on automatic ADS stage 4 actuation.

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Table 7.3-2 (Sheet 6 of 7)
Interlocks for Engineered Safety Features Actuation System

Designation	Derivation	Function
P-12	Pressurizer level below setpoint	(a) Permits manual block of core makeup tank actuation on Low-2 pressurizer level to allow mid-loop operation (b) Permits manual block of reactor coolant pump trip on Low-2 pressurizer level to allow mid-loop operation (c) Permits manual block of auxiliary spray and purification line isolation, and zinc and hydrogen addition isolation valves isolation on low pressurizer level to allow mid-loop operation (d) Coincident with manual action of (a), automatically unblocks fourth stage automatic depressurization system initiation on Low-4 hot leg level to provide protection during mid-loop operation. (e) Automatically unblocks chemical and volume control system letdown isolation on Low-2 hot leg level
$\overline{\text{P-12}}$	Pressurizer level above setpoint	(a) Prevents manual block of core makeup tank actuation on Low-2 pressurizer level (b) Prevents manual block of reactor coolant pump trip on Low-2 pressurizer level (c) Prevents manual block of auxiliary spray and purification line isolation, and zinc and hydrogen addition isolation valves isolation on low pressurizer level (d) Automatically blocks fourth stage automatic depressurization system initiation on Low-4 hot leg level to reduce the probability of spurious actuation. (e) Permits manual block of chemical and volume control system letdown isolation on Low-2 hot leg level

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Table 7.3-2 (Sheet 7 of 7)
Interlocks for Engineered Safety Features Actuation System

Designation	Derivation	Function
P-19	Reactor coolant system pressure below setpoint	(a) Permits manual block of chemical and volume control system makeup isolation on High-2 pressurizer water level (b) Permits manual block of passive residual heat removal heat exchanger alignment on High-3 pressurizer water level (c) Permits manual block of the pressurizer heater trip on High-3 pressurizer water level
$\overline{\text{P-19}}$	Reactor coolant system pressure above setpoint	(a) Prevents manual block of chemical and volume control system makeup isolation on High-2 pressurizer water level (b) Prevents manual block of passive residual heat removal heat exchanger alignment on High-3 pressurizer water level (c) Prevents manual block of the pressurizer heater trip on High-3 pressurizer water level

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Table 7.3-3 (Sheet 1 of 2)
System-Level Manual Input to the
Engineered Safety Features Actuation System

Manual Control	To Divisions				Figure 7.2-1 Sheet
Manual safeguards actuation #1	A	B	C	D	11
Manual safeguards actuation #2	A	B	C	D	11
Manual chemical and volume control system makeup isolation #1	A		C	D	6
Manual chemical and volume control system makeup isolation #2	A		C	D	6
Manual passive residual heat removal heat exchanger alignment #1	A	B	C	D	8
Manual passive residual heat removal heat exchanger alignment #2	A	B	C	D	8
Manual steam line isolation #1		B		D	9
Manual steam line isolation #2		B		D	9
Manual steam generator relief isolation #1		B		D	9
Manual steam generator relief isolation #2		B		D	9
Steam/feedwater isolation and safeguards block control #1	A				9
Steam/feedwater isolation and safeguards block control #2		B			9
Steam/feedwater isolation and safeguards block control #3			C		9
Steam/feedwater isolation and safeguards block control #4				D	9
Manual feedwater isolation #1		B		D	10
Manual feedwater isolation #2		B		D	10
Manual steam dump mode control #1		B			10
Manual steam dump mode control #2				D	10
Manual Stage 1 steam dump cooldown control #1		B			10
Manual Stage 1 steam dump cooldown control #2				D	10
Manual Stage 2 steam dump cooldown control #1		B			10
Manual Stage 2 steam dump cooldown control #2				D	10
Pressurizer pressure safeguards block control #1	A				11
Pressurizer pressure safeguards block control #2		B			11
Pressurizer pressure safeguards block control #3			C		11
Pressurizer pressure safeguards block control #4				D	11
Manual auxiliary spray isolation			C		12
Manual core makeup tank injection actuation #1	A	B	C	D	12
Manual core makeup tank injection actuation #2	A	B	C	D	12
Core makeup tank injection actuation block control #1	A				12
Core makeup tank injection actuation block control #2		B			12
Core makeup tank injection actuation block control #3			C		12
Core makeup tank injection actuation block control #4				D	12
Manual passive containment cooling actuation #1	A	B	C	D	13
Manual passive containment cooling actuation #2	A	B	C	D	13
Manual passive containment isolation actuation #1	A	B	C	D	13
Manual passive containment isolation actuation #2	A	B	C	D	13
Manual depressurization system stages 1, 2, and 3 actuation #1 & #2	A	B	C	D	15

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Table 7.3-3 (Sheet 2 of 2)
System-Level Manual Input to the
Engineered Safety Features Actuation System

Manual Control	To Divisions				Figure 7.2-1 Sheet
Manual depressurization system stages 1, 2, and 3 actuation #3 & #4	A	B	C	D	15
Manual depressurization system stage 4 actuation #1 & #2	A	B	C	D	15
Manual depressurization system stage 4 actuation #3 & #4	A	B	C	D	15
Manual IRWST injection actuation #1 & #2	A	B	C	D	16
Manual IRWST injection actuation #3 & #4	A	B	C	D	16
Manual containment recirculation actuation #1 & #2	A	B	C	D	16
Manual containment recirculation actuation #3 & #4	A	B	C	D	16
Manual main control room isolation, air supply initiation, and electrical load de-energization #1	A	B	C		13
Manual main control room isolation, air supply initiation, and electrical load de-energization #2	A	B	C		13
RCS pressure CVS/PRHR block control #1	A				6
RCS pressure CVS/PRHR block control #2		B			6
RCS pressure CVS/PRHR block control #3			C		6
RCS pressure CVS/PRHR block control #4				D	6
SFS isolation block control #1	A				14
SFS isolation block control #2		B			14
Normal residual heat removal system isolation safeguards block control #1	A				13
Normal residual heat removal system isolation safeguards block control #2		B			13
Boron dilution block control #1	A				3
Boron dilution block control #2		B			3
Boron dilution block control #3			C		3
Boron dilution block control #4				D	3
Manual RNS isolation #1 & #3	A	B		D	18
Manual RNS isolation #2 & #4	A	B		D	18
CVS letdown isolation block control #1	A				16
CVS letdown isolation block control #2				D	16
Manual containment vacuum relief actuation #1	A		C		19
Manual containment vacuum relief actuation #2	A		C		19
FWS/CVS isolation and turbine trip block control #1	A				10
FWS/CVS isolation and turbine trip block control #2		B			10
FWS/CVS isolation and turbine trip block control #3			C		10
FWS/CVS isolation and turbine trip block control #4				D	10

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Table 7.3-4 (Sheet 1 of 2)
Engineered Safety Features Actuation, Variables, Limits, and Ranges (Nominal)

Variable	Range of Variable	Response Time Assumed in Accident Analysis (Sec)[1]
Pressurizer pressure	1500 to 2500 psig	2.0
Steam line pressure	500 to 1300 psig	2.0
Steam line negative pressure rate	0 to 1300 psig	N/A[2]
Cold leg temperature (T_{cold})	490 to 610°F	2.0[3]
Hot leg temperature (T_{hot})	530 to 650°F	N/A[4]
Containment pressure	-5 to 10 psig	2.0
Reactor coolant system hot leg level	0 to 100% of span	N/A[2]
In-containment refueling water storage tank level	0 to 100% of span	N/A[2]
Undervoltage on input of 1E battery charger	0 to 500 V	N/A[4]
Steam generator narrow range water level	0 to 100% of span (narrow range taps)	2.0
Steam generator wide range water level	0 to 100% of span (wide range taps)	2.0
Core makeup tank narrow range upper water level	0 to 100% of span	2.0
Core makeup tank narrow range lower water level	0 to 100% of span	2.0
Reactor coolant pump bearing temperature	50 to 400°F	N/A[2]
Spent fuel pool level	0 to 26 feet	N/A[2]
Reactor coolant system wide range pressure	0 to 3300 psig	N/A[2]

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Table 7.3-4 (Sheet 2 of 2)
Engineered Safety Features Actuation, Variables, Limits, and Ranges (Nominal)

Variable	Range of Variable	Response Time Assumed in Accident Analysis (Sec)(1)
Pressurizer water level	0 to 100% of cylindrical portion of pressurizer	2.0
Startup feedwater flow	0 to 600 gpm	2.0
Neutron flux (flux doubling calculation)	1 to 10^6 c/sec	80.0(5)
Control room supply air radiation level	Particulate: 10^{-12} to 10^{-7} µ Ci/cc Iodine: 10^{-11} to 10^{-5} µ Ci/cc	200
Control room differential pressure	+1.00 to -1.00 in. w.g.	N/A(2)
Containment radioactivity	10^0 to 10^7 R/hr	N/A(2)

Notes:

1. Delay from the time that the process variable exceeds the setpoint until the time that an output is provided to the actuated device. This response time reflects the total delay from both the PMS equipment and the associated sensor.
2. The following functions are not explicitly assumed to initiate an ESF in the accident analysis. However, they are subject to response time test surveillance requirements in the technical specifications. Therefore, the following maximum allowable response times are allocated:

   Steam line negative pressure rate = 2.0 seconds
   Reactor coolant system hot leg level = 3.0 seconds
   In-containment refueling water storage tank level (Narrow Range) = 3.0 seconds
   In-containment refueling water storage tank level (Wide Range) = 3.0 seconds
   Reactor coolant pump bearing temperature = 8.5 seconds
   Spent Fuel Pool Level = 2.0 seconds
   Reactor coolant system wide range pressure = 2.0 seconds
   Control room differential pressure = 3.0 seconds
   Containment Radioactivity = 25.0 seconds

3. In addition to the response time given, a 4.0 second lag time is used in the accident analysis for this function to account for the associated RTDs.
4. This function is not explicitly assumed to initiate an ESF in the accident analysis and does not have any response time test requirements in the technical specifications.
5. Response time depends on flux doubling calculation.

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7.4 Systems Required for Safe Shutdown

Systems to establish safe shutdown conditions perform two basic functions. First, they provide the necessary reactivity control to maintain the core in a subcritical condition. Boration capability is provided to compensate for xenon decay and to maintain the required core shutdown margin. Second, these systems must provide residual heat removal capability to maintain adequate core cooling.

The designation of systems required for safe shutdown depends on identifying those systems that provide the following capabilities for maintaining a safe shutdown:

- Decay heat removal
- Reactor coolant system inventory control
- Reactor coolant system pressure control
- Reactivity control

There are two different safe shutdown conditions that are expected following a transient or accident condition. Short-term safe shutdown refers to the plant conditions from the start of an event until about 36 hours later. Long-term safe shutdown refers to the plant conditions after this 36-hour period.

The short-term safe shutdown conditions include maintaining the reactor subcritical, the reactor coolant average temperature less than or equal to no load temperature, and adequate coolant inventory and core cooling. These shutdown conditions shall be achieved following any of the design basis events using safety-related equipment. The specific safe shutdown condition achieved is a function of the particular accident sequence.

The long-term safe shutdown conditions are the same as the short-term conditions except that the core average temperature shall be less than 420°F. This long-term condition must be achieved within 36 hours following a non-LOCA event using the passive residual heat removal heat exchanger as shown in Appendix 19E. These safe shutdown conditions can be maintained by the passive residual heat removal heat exchanger for greater than 14 days based on a non-bounding, conservative analysis that only credits using safety-related equipment. In addition, these safe shutdown conditions can be maintained indefinitely using the automatic depressurization system and passive injection/ recirculation as discussed in Subsection 7.4.1.1.

There are no systems specifically and solely dedicated as safe shutdown systems. However, there are a number of plant systems that are available to establish and maintain safe shutdown conditions. Normally, in the event of a turbine or reactor trip, nonsafety-related plant systems automatically function to place the plant in short-term safe shutdown, as described in Subsection 7.4.1.2. During the short-term safe shutdown condition, an adequate heat sink is provided to remove reactor core residual heat and boration control is available. Redundancy of systems and components is provided to enable continued maintenance of the short-term safe shutdown condition. Additional redundant nonsafety-related systems are normally available to manually perform a plant depressurization and cooldown.

The engineered safety systems are designed to establish and maintain safe shutdown conditions for the plant. Nonsafety-related systems are not required for safe shutdown of the plant.

This section focuses on safety-related systems used to establish and maintain safe shutdown conditions. The discussion of safe shutdown does not include accident response and/or mitigation since the standard review plan for this section addresses safe shutdown not related to accident

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mitigation. However, safe shutdown conditions are also established and maintained by these safety-related systems following accident conditions. For example, the control rods are released to initially place the plant in a shutdown condition to mitigate the consequences of various accidents. The passive core cooling system, on the other hand, is used to provide core cooling in an accident, but it is also one of the principal systems used for safe shutdown. Only those specific engineered safety systems listed in Table 7.4-1 are used to establish and maintain safe shutdown of the plant. These engineered safety systems automatically function to place the plant in a safe shutdown condition without operator action.

The instrumentation functions necessary for safe shutdown are available through instrumentation channels associated with the safety-related systems in the primary plant. These channels automatically actuate the protective functions provided by the safety-related systems. Manual actuation of the associated safety-related systems is also provided.

The instrumentation systems discussed in this section are those which are required during nonaccident conditions to align the safety-related systems and perform the specified safe shutdown functions.

The specific systems available for safe shutdown are discussed in Subsection 7.4.2 and are listed in Table 7.4-1.

Maintenance of safe shutdown conditions with these systems, and the associated instrumentation and controls, includes consideration of the accident consequences that might challenge safe shutdown conditions. The accident consequences that are germane are those that tend to degrade the capabilities for coolant circulation, boration, heat removal, and depressurization. Safe shutdown is achieved following any of the accidents analyzed in Chapter 15. The specific safe shutdown condition reached is a function of the particular accident sequence.

The instrumentation and controls discussed in Subsection 7.4.1 are used to control and/or monitor shutdown. These safety-related systems allow the maintenance of safe shutdown, even under accident conditions that tend toward a return to criticality or a loss of heat sink.

In addition to the operation of safety-related systems used for safe shutdown, as described in Subsection 7.4.1, the following are part of the safe shutdown provisions:

- The turbine is tripped. (This can be accomplished at the turbine as well as from the main control room.)
- The reactor is tripped. (This can be accomplished at the reactor trip switchgear as well as from the main control room.)
- Support of engineered safety systems actuation is provided by safety-related onsite dc power.

7.4.1 Safe Shutdown

7.4.1.1 Safe Shutdown Using Safety-Related Systems

The following describes the process that establishes safe shutdown conditions for the plant, based on a conservative, non-bounding analysis using the safety-related systems, and no operator action. The reactor coolant system is assumed to be intact for this discussion of safe shutdown.

Since this discussion only considers the use of safety-related systems, offsite electrical power sources are assumed to be lost at the start of the event. This results in a loss of the reactor coolant

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pumps. Even though the reactor coolant pumps are tripped during the initiation of certain engineered safety system actuation, it is assumed that no engineered safety system actuation signal is generated for this initiating event. With loss of the reactor coolant pumps, reactor coolant system natural circulation flow initiates and transfers core heat to the steam generators. Since feedwater flow is lost, the existing steam generator water inventory provides initial decay heat removal capability.

The initial loss of main ac power results in the Class 1E dc batteries automatically supplying power to the Class 1E dc power distribution network and the four Class 1E 120 Vac instrumentation divisions via the inverters.

The initial response of the passive safety systems is to actuate the passive residual heat removal heat exchanger due to Low-2 steam generator wide range water level. The passive residual heat removal heat exchanger removes decay heat from the core by transferring this heat to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger removes core decay heat, cooling the reactor coolant system. As reactor coolant system cooldown continues, the reactor coolant system pressure decreases due to contraction of the reactor coolant system inventory since the pressurizer heaters are de-energized. An engineered safety system actuation signal occurs when reactor coolant system pressure decreases below a setpoint. This actuates the core makeup tanks, if they had not been previously actuated due to Low-2 pressurizer level. The core makeup tanks provide borated water injection to the reactor coolant system.

The engineered safety system actuation signal generated on Low-3 pressurizer pressure also actuates containment isolation. This prevents loss of water inventory from containment and permits operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank for greater than 14 days.

The in-containment refueling water storage tank starts to boil about one to two hours after passive residual heat removal operation is initiated. Once boiling occurs, the in-containment refueling water storage tank begins steaming to containment, transferring heat to the air flowing on the outside of the containment shell. As steaming to containment continues, containment pressure slowly increases. As containment pressure slowly increases, an engineered safety system actuation signal is generated on containment High-2 pressure, resulting in the initiation of passive containment cooling. This provides water flow on the outside of the containment shell to improve the heat removal performance from containment through evaporative cooling to the outside air.

A gutter located at the operating deck elevation collects condensate from the inside of the containment shell. Valves located in drain lines from the gutter to the containment waste sump close on a passive residual heat removal heat exchanger actuation signal. This action diverts the condensate to the in-containment refueling water storage tank. The system provides core decay heat removal in this configuration for greater than 14 days with a limited increase in the containment water level.

Once the reactor coolant system and the safety systems are in this configuration, the plant is in a safe, stable shutdown condition. The reactor coolant system temperatures and pressures continue to slowly decrease. The passive residual heat removal heat exchanger has the capacity to maintain a safe, stable reactor coolant system condition during a design basis event for at least 72 hours in a closed-loop mode of operation. A non-bounding, conservative analysis of extended operation in this mode shows the passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours.

Operation in this configuration may be limited in time duration by reactor coolant system leakage. The core makeup tanks can only supply a limited amount of makeup in the event there is reactor

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coolant system leakage. Eventually the volume of the water in the core makeup tanks will decrease to the first stage automatic depressurization setpoint. The time to reach this setpoint depends upon the reactor coolant system leak rate and the reactor coolant cooldown.

The 24-hour Class 1E dc batteries that power the automatic depressurization system valves provide power for at least 24 hours. There is a timer that measures the time that ac power sources are unavailable. This timer provides for automatic actuation of the automatic depressurization system before the 24-hour Class 1E dc batteries are discharged. The emergency response guidelines direct the operator to assess the need for automatic depressurization before the timer completes its count (approximately 22 hours). The operator assessment considers core makeup tank levels, reactor coolant system hot leg level, temperature, and pressure. If automatic depressurization is not needed, the operator is directed to de-energize all loads on the 24-hour Class 1E dc batteries. This action preserves the capability for the operator to initiate automatic depressurization at a later time based on assessment of the same parameters.

The automatic depressurization system can be manually initiated by the operator at any time, but no operator action is needed to provide safe shutdown conditions. Once the automatic depressurization system sequence initiates, the plant automatically transitions to lower pressure and temperature conditions that establish and maintain long-term safe shutdown of the plant.

When the automatic depressurization system is actuated, the first stage depressurization valves open and the reactor coolant system depressurization starts. The second and third stage depressurization valves open in sequence, based on automatic timers that are started upon the actuation of the first stage depressurization valves. As reactor coolant inventory continues to be lost, the core makeup tanks continue to inject. If the volume of the water in the core makeup tanks decrease to the fourth stage automatic depressurization setpoint, the fourth stage depressurization valves open. The water and steam vented from the reactor coolant system initially flows into the in-containment refueling water storage tank and overflows into the refueling canal. Eventually this overflows into the reactor vessel cavity, where any moisture from the fourth stage automatic depressurization system valves also collects from discharge in the loop compartments. This overflow initiates the floodup of containment, along with condensate from the containment shell and other cool surfaces in containment.

As the reactor coolant system pressure decreases, the accumulators inject borated water into the reactor coolant system. After the fourth stage automatic depressurization system valves open, the reactor coolant system pressure is reduced sufficiently so that in-containment refueling water storage tank injection can begin as the core makeup tanks empty.

The drain down of the in-containment refueling water storage tank is relatively slow, depending on the injection rates and the reactor coolant system pressure. As the in-containment refueling water storage tank continues to inject, the containment floodup also continues and eventually the floodup volume is sufficient to initiate flow from the recirculation sump.

As the reactor coolant system voids during the cooldown and depressurization process, water flow through the passive residual heat removal heat exchanger is replaced by steam flow, which also provides core cooling. As the in-containment refueling water storage tank empties and uncovers the passive residual heat removal heat exchanger, heat transfer via this path decreases. Eventually, the passive residual heat removal heat exchanger is uncovered, heat removal by the passive residual heat removal heat exchanger stops, and decay heat is removed by automatic depressurization system venting.

The final long-term safe shutdown plant conditions are maintained with the reactor coolant system depressurized to about 10 psig at saturated conditions, venting steam through the automatic depressurization system valves to containment, with heat transferred to the outside atmosphere via

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the passive containment cooling system. With containment isolation established, the water inventory inside containment provides an indefinite cooling water supply for core decay heat removal.

7.4.1.2 Safe Shutdown Using Safety-Related and Nonsafety-Related Systems

This subsection describes situations where nonsafety-related features of the plant are used together with safety-related systems to establish safe shutdown conditions. As discussed in Subsection 7.4.1.1, the AP1000 can be placed in a safe shutdown condition and maintained there using safety-related systems and no operator actions. Section 6.3 provides additional discussion of these situations.

Following passive residual heat removal heat exchanger actuation, the in-containment refueling water storage tank heats up and starts to boil after several hours of operation. If normal steam generator heat removal is not re-established, the operators align the normal residual heat removal system to cool the in-containment refueling water storage tank. This operation prevents significant steaming to the containment.

In case the automatic depresurization system is actuated, the operators align the normal residual heat removal system to provide injection to the reactor coolant system. This action causes the core makeup tank level to remain above the fourth stage valve actuation setpoint and prevents significant steaming to and flooding of the containment.

7.4.1.3 Safe Shutdown Using Nonsafety-Related Systems

This subsection describes the process to establish and maintain safe shutdown conditions using the nonsafety-related systems. As discussed in Section 7.4, the review of the plant safe shutdown capability, including the capabilities provided by the nonsafety-related systems, does not include accident response or mitigation. The nonsafety-related systems normally used to support plant shutdown operations are expected to be available. Offsite power is also expected to be available to support safe shutdown operations, although the nonsafety-related systems can establish and maintain safe shutdown conditions using only onsite electrical power.

For the purposes of this discussion, the nonsafety-related system operation following a reactor trip is described. As assumed in the discussion in Subsection 7.4.1.1 on safe shutdown using safety-related systems, the reactor coolant system is assumed to be intact during plant safe shutdown operations.

The nonsafety-related systems and equipment used to establish and maintain safe shutdown conditions are the same systems and equipment that are operated during normal plant startup and shutdown evolutions. The safe shutdown capability using the safety-related systems, described in Subsection 7.4.1.1, is only expected to be used in the event that the nonsafety-related systems are not available.

The nonsafety-related systems operate to establish and maintain safe shutdown conditions by providing the safe shutdown functions described in Section 7.4, except that reactivity control is only needed for long-term safe shutdown. If offsite power is available, the operation of these nonsafety-related systems is automatic.

The nonsafety-related systems actuate to establish and maintain the short-term safe shutdown conditions. The systems can also establish and maintain long-term safe shutdown conditions within the time limits discussed in Section 7.4. The operational philosophy following any event is to maintain appropriate safe shutdown conditions based on the duration of the shutdown, until the plant is able to re-start.

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Cold shutdown conditions would only be established if it becomes necessary for equipment repair or due to limitations of the nonsafety-related systems in maintaining safe shutdown conditions (such as feedwater system water inventory). This philosophy reduces unnecessary challenges to plant safety due to the transition from operating systems to infrequently-operated standby systems.

Normally, offsite electrical power is available and the nonsafety-related systems automatically maintain short-term safe shutdown conditions as follows:

- Reactor coolant system forced flow to the steam generators by the reactor coolant pumps
- Feedwater from the main or startup feedwater systems
- Heat removal by the steam generators to the main condenser using turbine bypass valves
- Condenser heat removal provided by the main circulating water system
- Reactor coolant system inventory and boration control by the chemical and volume control system
- Reactor coolant system pressure control using pressurizer heaters and normal spray

If offsite power is not available, the reactor coolant pumps, main feedwater pumps, and main circulating water pumps will not be operating. However, the nonsafety-related systems maintain short-term safe shutdown conditions without offsite electrical power as follows:

- Electrical power provided to the required nonsafety-related systems by the diesel-generators of the onsite standby power system
- Heat removal by the steam generators directly to the atmosphere through the power-operated relief valves
- Feedwater from the startup feedwater system
- Reactor coolant system flow to the steam generators via natural circulation
- Reactor coolant system inventory and boration control by the chemical and volume control system
- Reactor coolant system pressure control using pressurizer heaters and auxiliary spray

In case the main feedwater is unavailable, the initial response of the nonsafety-related systems following a reactor trip is to automatically actuate the startup feedwater system, on low steam generator water level, to provide decay heat removal. The steam generators can remove decay heat from the core by either forced or natural circulation in the reactor coolant system. If offsite electrical power is available, the reactor coolant pumps continue to provide forced circulation in the reactor coolant system and the circulating water system continues to operate to provide a heat sink for the steam discharged from the steam generators to the main condenser.

With offsite power and the main condenser available, the turbine bypass valves automatically actuate after the reactor trip to control reactor coolant system temperature, based on the pre-set steam generator pressure control set point that is normally established for standby turbine bypass valve operation. The main feedwater system or the startup feedwater system automatically maintains steam generator water level as the turbine bypass valves continue to throttle steam flow to match the

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decreasing core decay heat levels. The pressurizer heaters and spray automatically maintain reactor coolant system subcooling with pressure at normal reactor coolant system conditions.

The chemical and volume control system makeup pumps automatically actuate as required to provide borated makeup water to maintain pressurizer level in the programmed band for no-load conditions. The makeup source is the boric acid tank which provides long-term reactivity control. The makeup pumps are expected to operate infrequently during these conditions to compensate for normal reactor coolant system inventory losses such as valve leakage.

Operation of the nonsafety-related systems in this mode maintains short-term safe shutdown conditions and reactor coolant system temperature and pressure remain near no-load conditions. If it becomes necessary to perform a plant cooldown and depressurization to establish long-term safe shutdown conditions, the nonsafety-related systems are used, following the normal plant cooldown procedures. Manual boration to the cold shutdown boron concentration is provided by the chemical and volume control system by initiating reactor coolant system letdown in combination with makeup pump operation. After the boration is completed and letdown is secured, the makeup pumps automatically maintain reactor coolant system inventory throughout the remainder of the cooldown process.

After the required boration is completed the turbine bypass valves are used to initiate the cooldown, with manual control of pressurizer heaters and spray to maintain the reactor coolant system pressure, temperature, and cooldown rate within the limits specified in the technical specifications. The main feedwater system automatically provides feedwater and maintains steam generator level throughout the cooldown process.

When the reactor coolant system temperature and pressure are reduced to within the capabilities of the normal residual heat removal system, at approximately 350°F and 400 psig, the system is manually aligned to the reactor coolant system and started to continue the cooldown process. The final long-term safe shutdown conditions established would be dependent upon the specific maintenance required.

The use of the nonsafety-related systems and equipment for both short-term and long-term safe shutdown also requires the operation of associated support systems. These normally operating support systems include component cooling water, chilled water, compressed air, area ventilation, and nonsafety-related instrumentation and control power. These systems are started as required following a loss of offsite power, once the nonsafety-related diesel-generators are started.

If offsite electrical power is unavailable, the nonsafety-related systems actuate to establish and maintain safe shutdown conditions. There are some differences in the decay heat discharge flow path and the reactor coolant system remains at a slightly higher temperature resulting from the natural circulation flow conditions. With the loss of offsite electrical power, the nonsafety-related diesel-generators provide electrical power for the required nonsafety-related equipment. However, the reactor coolant pumps, main feedwater pumps, and main circulating water pumps are not available. Therefore, core decay heat is transferred to the steam generators using natural circulation in the reactor coolant system, the startup feedwater pumps supply the steam generators, and the steam generators discharge directly to the atmosphere to remove decay heat.

When offsite electrical power is unavailable, reactor coolant temperature is automatically maintained by the steam generator atmospheric power-operated relief valves instead of the turbine bypass valves. The steam generator power-operated relief valves maintain a pre-set steam generator pressure by throttling the steam discharged directly from the steam generators to the atmosphere. The relief valve operation maintains a slightly higher steam generator pressure than the pressure maintained with turbine bypass valve standby operation, resulting in a slight increase in the reactor coolant system temperature. The automatic operation of the startup feedwater subsystem maintains

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steam generator inventory with the pumps powered from the diesel-generators. In addition, the direct discharge of steam to the atmosphere prevents condensate recovery, which limits the water inventory for the startup feedwater system.

Following a loss of offsite power, the reactor coolant system temperature is slightly higher than for a reactor trip when offsite electrical power is available, resulting from natural circulation flow and steam generator power-operated relief valve operation. Since the transition to natural circulation flow is relatively slow, the reactor coolant system pressure remains stable without operator action. Operator action is not required to maintain reactor coolant system pressure.

Without offsite electrical power, the pressurizer heaters are manually re-energized after the diesel-generators start. Without reactor coolant pump operation, normal pressurizer spray is unavailable to counteract system pressure increases. Therefore, auxiliary spray provided by the chemical and volume control system makeup pumps is manually initiated to decrease reactor coolant system pressure, if necessary. The operation of the chemical and volume control system makeup pumps to maintain reactor coolant system inventory is similar to their operation when offsite power is available, except that the pumps are manually controlled and powered from the diesel-generators.

The nonsafety-related systems are normally expected to maintain short-term safe shutdown conditions when offsite power is not available. If it is required to establish long-term safe shutdown conditions for equipment maintenance, the cooldown would normally be delayed until offsite power is recovered.

However, the nonsafety-related systems can be used to perform a natural circulation cooldown, if necessary. When performing a natural circulation plant cooldown and depressurization, the operation of the nonsafety-related systems is similar to the normal cooldown operation except that they are powered from the diesel-generators. The primary difference in operation is the use of the steam generator power-operated relief valves to control the cooldown process.

7.4.2 Safe Shutdown Systems

To effect a safe shutdown, with safety-related systems, the plant is initially brought to a stable condition with heat removal provided by the passive residual heat removal heat exchanger. For safe shutdown conditions, control is possible from either the main control room or the remote shutdown workstation. To accomplish a safe shutdown, the functions required are: coolant circulation, boration, heat removal, and depressurization. The portions of the protection and safety monitoring system required to achieve the safe shutdown condition are described in Sections 7.2 and 7.3. The minimum systems required to maintain safe shutdown conditions under a nonaccident condition are listed and discussed in the following paragraphs.

7.4.2.1 Passive Core Cooling System

A description of the passive core cooling system and its operation is provided in Section 6.3. The passive residual heat removal heat exchanger, the core makeup tanks, the in-containment refueling water storage tank, the containment recirculation, and the automatic depressurization system actuate automatically. They can also be manually initiated. Actuation controls are located at the remote shutdown workstation as well as in the main control room.

The safety injection flow from the accumulators, initiates automatically by the reactor coolant system depressurization process. The operation of the accumulator is integrated with the automatic actuation of the other passive core cooling subsystems.

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7.4.2.2 Passive Containment Cooling System

A description of the passive containment cooling system and its operation is provided in Subsection 6.2.2. The passive containment cooling system actuates automatically. It also can be manually initiated. Actuation controls are located at the remote shutdown workstation as well as in the main control room.

7.4.2.3 Containment Isolation

A description of containment isolation valves and their operation is provided in various subsections. Each system that has piping that penetrates the containment vessel and therefore, requires containment isolation valves is discussed in its own subsection. Most of these systems are nonsafety-related; however, the containment isolation valves and the associated piping are safety-related and automatically close on a safeguards actuation (S) signal. The containment isolation system is discussed in Subsection 6.2.3.

7.4.2.4 Reactor Coolant System Circulation

The preferred method of coolant circulation is forced circulation with the reactor coolant pumps supplying the driving head. Upon the loss of main ac power, or when the reactor coolant pumps are tripped during engineered safety system actuation, the reactor coolant pumps are not available. However, the reactor coolant system is designed to provide sufficient natural circulation to achieve safe shutdown conditions with the steam generators and passive residual heat removal heat exchanger removing decay heat. Natural circulation flow is verified by monitoring the reactor coolant system temperatures.

7.4.2.5 Other Systems Required for Safe Shutdown

The other safety-related equipment and systems used to maintain the plant in safe shutdown are identified in Table 7.4-1. They are also listed below, with a reference to the respective section or subsection which discusses their operation in more detail:

- Protection and safety monitoring system — Sections 7.2, 7.3, and 7.5
- Class 1E dc and UPS system — Subsection 8.3.2

These systems are either normally operating or they start automatically when required. The instrumentation for these systems is described in the particular section containing the system description.

The monitoring instrumentation available in the main control room for safe shutdown is safety-related and is part of the protection and safety monitoring system. The instrumentation available for safe shutdown monitoring is listed in Section 7.5.

7.4.3 Safe Shutdown from Outside the Main Control Room

7.4.3.1 Description

If temporary evacuation of the main control room is required because of some abnormal main control room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the main control room through the use of controls and monitoring located at the remote shutdown workstation. Safe shutdown is a stable plant condition that can be maintained for an extended period of time. In the event that access to the main control room is restricted, the plant is maintained in safe shutdown until the main control room can be re-entered.

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7.4.3.1.1 Remote Shutdown Room/Remote Shutdown Workstation

Safe shutdown can be established and maintained from the remote shutdown room. The I&C equipment in the room is collectively referred to as the remote shutdown workstation. The workstation is designed to allow control of a shutdown following an evacuation of the control room, coincident with the loss of offsite power and a single active failure. No other design basis event is postulated. Subsection 9.5.1 provides a discussion of shutdown in the event of a fire. The design basis for the remote shutdown workstation does not require safety-related displays, alarms, and controls.

The remote shutdown workstation contains nonsafety controls, displays, and alarms for the safety-related equipment required to establish and maintain safe shutdown. Additionally, control of nonsafety-related components is available.

The remote shutdown workstation includes operator workstations that are similar to the operator workstations in the main control room and are designed to the same standards. The remote shutdown workstation also includes dedicated nonsafety controls that provide the minimum inventory of controls listed in Table 18.12.2-1. The dedicated nonsafety controls interface to the plant safety and monitoring system via qualified isolators within that system.

The operator workstations have the same capabilities as the reactor operator's workstation in the main control room. The displays and alarms listed in Table 18.12.2-1 are retrievable from the operator workstations. Subsection 18.12.3 provides more discussion on the remote shutdown workstation displays, alarms, and controls.

The remote shutdown workstation is provided for use only following an evacuation of the main control room. No actions are anticipated from the remote shutdown workstation during normal, routine shutdown, refueling, or maintenance operations.

The remote shutdown workstation has sufficient communication circuits to allow the operator to effectively establish safe shutdown conditions. As detailed in Subsection 9.5.2, communication is available between the following stations:

- Main control room
- Remote shutdown workstation
- Onsite technical support center
- Diesel generator local control station

Operator control capability at the remote shutdown workstation is normally disabled, and operator control functions are normally performed from workstations located inside the main control room; however, operator control capability can be transferred from the main control room workstations to the remote shutdown workstation if the control room requires evacuation. Procedures will instruct the operator to trip the reactor prior to evacuating the control room and transferring control to the remote shutdown workstation. This operator control transfer capability cannot be disabled by any single active failure coincident with the loss of offsite power.

The control transfer function is implemented by multiple transfer switches. Each individual transfer switch is associated with only a single safety-related or single nonsafety-related group. These switches are located behind an unlocked access panel. Entry into this access panel will result in alarms at the main control room and remote shutdown workstation. The access panel is located within a fire zone which is separate from the main control room. Actuation of these transfer switches

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results in additional alarms at the main control room and remote shutdown workstation, the activation of operator control capability from the remote workstation, and the deactivation of operator control capability from the main control room workstations. This deactivation of operator control capability includes deactivation of all operator control capability provided by the soft control devices described in Subsection 7.1.3.3 and deactivation of all operator control capability provided by dedicated switches. This includes deactivation of operator control capability using manual actuation functions provided by the diverse actuation system as described in Subsection 7.7.1.11. The manual reactor trip switches located in the main control room are not affected by this control transfer function. The operator displays, located in the main control room and on the remote shutdown workstation, are also not affected by this control transfer function. The displays on the remote shutdown workstation are operational during normal operation (from the main control room) so that they can be used with no delay if transfer to the remote shutdown workstation is required.

7.4.3.1.2 Controls at Other Locations

In addition to the controls and indicators provided at the remote shutdown workstation, the following controls are provided outside the main control room:

- Reactor trip capability at the reactor trip switchgear
- Turbine trip capability at the turbine
- Start/stop controls for the diesel generators, located at each diesel generator local control panel
- Local control at motor control centers and electrical switchgear.

7.4.3.1.3 Design Bases Information

According to GDC 19, the capability of establishing a shutdown condition and maintaining the station in a safe status in that mode is an essential function. The controls and indications necessary for this function are identified in Subsection 7.4.2. To provide the availability of the remote shutdown workstation after control room evacuation, the following design features are provided:

- The remote shutdown workstation conforms with the guidelines provided by ANSI 58.6 1996 (Reference 1).
- The remote shutdown workstation achieves and maintains safe shutdown conditions from full power conditions and maintains safe shutdown conditions thereafter.
- The remote shutdown workstation achieves safe shutdown when offsite power is available and when offsite power is not available.
- The remote shutdown workstation operates safety-related systems, independent from the main control room.
- The remote shutdown workstation is designed with redundancy. When a random event, such as a fire, or an allowable technical specification maintenance results in one safety-related division being unavailable, a single failure in a redundant division is not postulated. When a random event other than fire causes a main control room evacuation, a coincident single failure in the safety systems controlled from the remote shutdown workstation is considered.
- Access to the remote shutdown workstation is under administrative control.

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7.4.3.2 Analysis

The analysis of the systems required for safe shutdown is provided in Subsection 7.4.1. The following discussion is limited to the remote shutdown workstation.

Conformance to NRC General Design Criteria

General Design Criterion 19 – The remote shutdown workstation provides adequate controls and indications located outside the main control room to establish and maintain the reactor and the reactor coolant system in a safe shutdown condition in the event that the main control room must be evacuated.

Conformance to NRC Regulatory Guides

Regulatory Guide 1.22 – The remote shutdown workstation is tested periodically during station operation.

Regulatory Guide 1.29 – The remote shutdown workstation is designed as seismic Category II to prevent compromising the function of safety-related devices during or after a safe shutdown earthquake.

Conformance to IEEE 603-1991

The remote shutdown workstation and the design features which provide for the transfer of control capability from the main control room to the remote shutdown workstation conform to applicable portions of IEEE 603-1991. The circuits which perform the control transfer function are designed so that a single failure does not prevent maintaining safe shutdown. This is accomplished by redundant components in the systems required for safe shutdown, using independent safety-related power divisions.

To prevent interaction between the redundant systems, the redundant control channels are wired independently and are separated from each other. Nonsafety-related circuits available for (but not required for) safe shutdown are electrically isolated from safety-related circuits.

7.4.4 Combined License Information

This section contained no requirement for information.

7.4.5 References

1. ANSI 58.6 1996, "Criteria for Remote Shutdown for Light Water Reactors."

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Table 7.4-1
Systems Required for Safe Shutdown

Protection and Safety Monitoring System
Passive Core Cooling System
Passive Residual Heat Removal Heat Exchanger
Core Makeup Tanks
Accumulators
In-Containment Refueling Water Storage Tank
Containment Sump Recirculation
Automatic Depressurization Valves
Passive Containment Cooling System
Class 1E dc and UPS System
Containment Isolation Valves
Reactor System
Control Rods

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7.5 Safety-Related Display Information

7.5.1 Introduction

An analysis is conducted to identify the appropriate variables and to establish the appropriate design bases and qualification criteria for instrumentation employed by the operator for monitoring conditions in the reactor coolant system, the secondary heat removal system, the containment, and the systems used for attaining a safe shutdown condition. This selection of monitored variables is based on the guidance provided in Regulatory Guide 1.97. The variables and instrument design criterion selected for the AP1000 is described in Subsections 7.5.2 and 7.5.3.

The safety-related display information is used by the operator to monitor and maintain the safety of the AP1000 throughout operating conditions that include anticipated operational occurrences and accident and post-accident conditions. The equipment which processes the safety-related display information and makes it available to the operator is discussed in Subsection 7.5.4.

7.5.2 Variable Classifications and Requirements

Accident monitoring instrumentation is necessary to permit the operator to take actions to address design basis accident situations and for unforeseen situations (should plant conditions evolve differently than predicted by the safety analyses, the control room operating staff has sufficient information to evaluate and monitor the course of the event). Additional instrumentation is needed to indicate to the operating staff whether the integrity of the fuel cladding, the reactor coolant pressure boundary, or the reactor containment has degraded beyond the prescribed limits defined in the plant safety analyses and other evaluations.

Six types of variables are classified to provide this instrumentation:

- Variables that provide information needed by the operator to perform manual actions associated with design basis accident events, for which no automatic control is provided and that are required for the safety systems to accomplish their safety function, are designated as Type A.

- Variables needed to assess that the plant critical safety functions are accomplished or maintained, as identified in the plant safety analysis and other evaluations, are designated as Type B.

- Variables used to monitor for the gross breach or the potential for gross breach of the fuel cladding, the reactor coolant pressure boundary, or the containment are designated as Type C.

- Variables needed to assess the operation of individual safety-related systems are designated as Type D.

- Variables used in determining the magnitude of the postulated releases and continually assessing releases of radioactive materials are designated as Type E.

- Variables that provide information to manually actuate and to monitor the performance of nonsafety-related systems to prevent unnecessary actuation of safety-related systems following plant events are designated as Type F.

The six classifications of variables are not mutually exclusive. When a variable is included in one or more of the six classifications, the equipment monitoring this variable meets the requirements of the highest category identified.

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Three categories of design and qualification criteria are used. This classification is made to identify the importance of the information and to specify the requirements placed on the accident monitoring instrumentation. Category 1 instrumentation has the highest performance requirements and is used for information that cannot be lost. Category 2 and Category 3 instruments are of lesser importance in determining the state of the plant and do not require the same level of operational assurance.

The primary differences between category requirements are in qualification, application of single failure, power supply, and display requirements. Category 1 requires seismic and environmental qualification, the application of a single-failure criterion, use of emergency power, and an immediately accessible display. Category 2 requires environmental qualification commensurate with the required function. It may require emergency power, but does not require the single failure criterion or an immediately accessible display. Category 2 requires a rigorous performance verification for a single instrument channel. Category 3, which is high quality commercial grade, does not require qualification, single failure criterion, emergency power, or an immediately accessible display.

The results of the post-accident monitoring system analysis are summarized in Table 7.5-1, which lists the variables selected for accident monitoring and reflects design requirements imposed on the instrumentation to satisfy post-accident monitoring guidance in accordance with the variable type and category. The instruments used for accident monitoring may have other design functions that impose requirements more stringent than those identified by the post-accident monitoring system.

Table 7.5-1 summarizes the following information for each variable identified:

- Instrument range or status
- Type and category
- Environmental qualification
- Seismic qualification
- Number of required channels
- Power supply
- Qualified data processing system (QDPS) indication

7.5.2.1 Variable Types

Accident monitoring variables and information display channels are those that enable the control room operating staff to perform the functions defined by the Types A, B, C, D, E, and F classifications.

Type A

Type A variables provide the primary information to permit the control room operating staff to:

- Perform the diagnosis in the AP1000 emergency operating instructions

- Take the specified, preplanned, manually-controlled actions, for which automatic controls are not provided, and that are required for safety-related systems to mitigate design basis accidents

There are no specific preplanned, manually-controlled actions for safety-related systems to mitigate design basis accidents in the AP1000 design. This includes the diagnosis of plant conditions required to take preplanned manual action. Variables used for contingency actions and additional variables that might be utilized are Types B, C, D, E, and F.

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Type B

Type B variables provide the control room operating staff with information to assess the process of accomplishing or maintaining critical integrity safety-related functions (that is, reactivity control, reactor coolant system integrity, reactor coolant system inventory control, reactor core cooling, heat sink maintenance, and reactor containment environment).

Type C

Type C variables provide the control room operating staff information to monitor:

- The extent to which variables that indicate the potential for causing a gross breach of a fission product barrier have exceeded the design basis values
- The in-core fuel cladding, the reactor coolant pressure boundary, or the primary reactor containment that may have been subject to gross breach

These variables include those required to initiate the early phases of an emergency plan. Excluded are those associated with monitoring of radiological release from the plant that are included in Type E.

Type C variables used to monitor the potential for breach of a fission product barrier have an extended range. The extended range is chosen to minimize the probability of instrument saturation even if conditions exceed those predicted by the safety analysis.

Although variables selected to fulfill Type C functions may rapidly approach the values that indicate an actual gross failure, it is the final steady-state value reached that is important. Therefore, a high degree of accuracy and a rapid response time are not necessary for Type C instrument channels.

Type D

Type D variables provide the control room operating staff with sufficient information to:

- Monitor the performance of plant safety-related systems used for mitigating the consequences of an accident and subsequent plant recovery to attain a safe shutdown condition, including verification of the automatic actuation of safety-related systems
- Take specified, preplanned, manually controlled actions using safety-related systems for establishing and maintaining a safe shutdown condition

Type E

Type E variables provide the control room operating staff with information to:

- Monitor the plant areas where access may be required to service equipment necessary to monitor or mitigate the consequences of an accident
- Estimate the magnitude of release of radioactive material through identified pathways and continually assess such releases
- Monitor radiation levels and radioactivity in the environment surrounding the plant
- Monitor the habitability of the main control room

Type F

Type F variables provide the information that allows the control room operating staff to:

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- Take specified, preplanned, manually controlled actions using nonsafety-related systems to prevent the unnecessary actuation of safety-related systems

- Monitor the performance of plant nonsafety-related systems used for mitigating the consequences of an accident and subsequent plant recovery to establish shutdown conditions, including verification of the automatic actuation of nonsafety-related systems

- Operate other nonsafety-related systems normally used for plant cooldown and to maintain plant shutdown conditions

7.5.2.2 Variable Categories

The qualification requirements of the Types A, B, C, D, E, and F accident monitoring instrumentation are subdivided into three categories. Descriptions of the three categories are given below. Table 7.5-2 summarizes the selection criteria for Types A, B, C, D, E, and F variables into each of the three categories. Table 7.5-3 summarizes the design and qualification requirements of the three designated categories.

7.5.2.2.1 Category 1

Selection Criteria for Category 1

The selection criteria for Category 1 variables are subdivided according to the variable type. For Type A, those primary variables used for providing information for preplanned operator action, required for the safety-related systems to accomplish their safety function for design basis accidents, are designated as Category 1. For Type B, those primary variables used for monitoring the process of accomplishing or maintaining critical safety functions are designated Category 1. For Type C, those primary variables used for monitoring the potential for breach of a fission product barrier are designated as Category 1. There are no Types D, E, or F Category 1 variables.

Qualification Criteria for Category 1

The Category 1 instrumentation is seismically and environmentally qualified as described in Sections 3.10 and 3.11. Instrumentation continues to read within the required accuracy following, but not necessarily during, a seismic event.

Each instrumentation channel is qualified from the sensor up to, and including, the display. Subsection 7.5.2.2.4 details the extended range instrumentation qualification.

Design Criteria for Category 1

The following design criteria apply to Category 1:

- No single failure (within either the accident monitoring instrumentation, its auxiliary supporting features, or its power sources), concurrent with the failures that are a cause of or result from a specific accident, prevents the control room operating staff from receiving the required information. Where failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree), additional information is provided to allow the control room operating staff to analyze the actual conditions in the plant. This is accomplished by providing additional independent channels of information of the same variable (an identical channel), or by providing independent channels which monitor different variables which bear known relationships to the channels (a diverse channel(s)). Redundant or diverse channels are electrically independent and physically separated from each other and from equipment not classified as safety-related.

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If ambiguity does not result from failure of the channel, then a redundant or diverse channel is not provided.

- The instrumentation is energized from the uninterruptible power supply inverter subsystem from the Class 1E dc system.
- Servicing, testing, and calibration programs are specified to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between testing is less than the normal time interval between shutdowns, a capability for testing during power operation is provided.
- The design provides administrative control of the access for removing channels from service.
- The design provides administrative control of the access to setpoint adjustments, module calibration adjustments, and test points.
- The monitoring instrumentation design minimizes the development of conditions that cause displays to give anomalous indications that are potentially confusing to the control room operating staff.
- The instrumentation is designed to promote the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.
- To the extent practicable, monitoring instrumentation inputs are from sensors that directly measure the desired variables. An indirect measurement is made only when it is shown by analysis to provide unambiguous information.
- Periodic checking, testing, calibration, and calibration verification is performed.
- The range selected for the instrumentation encompasses the expected operating range of the monitored variable.

Information Processing and Display Interface Criteria for Category 1

The following interface criteria are implemented in the processing and displaying of the information:

- The control room operating staff has immediate access to the information from redundant or diverse channels in familiar units of measure. For example, degrees are used, not volts, for temperature readings. Where two or more instruments are needed to cover a particular range, overlapping instrument spans are provided.
- Continuous recording of these channels is provided following an accident until continuous recording of such information is not necessary. The term continuous recording does not exclude the use of discrete time sample data storage systems. This recording is available when required and does not need to be immediately accessible. The recording function is provided by the non-Class 1E data display and processing system.

7.5.2.2.2 Category 2

Selection Criteria for Category 2

The selection criteria for Category 2 variables are subdivided according to the variable type. For Types A, B, and C, some variables that provide backup information are designated Category 2. For Type D, those primary variables that are used for monitoring the performance of safety systems are designated as Category 2. For Type E, those primary parameters monitored for use in determining

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the magnitude of the release of radioactive materials and for continuously assessing such releases are designated as Category 2. For Type F, those primary parameters monitored for use in implementing preplanned actions using nonsafety-related systems or for monitoring the status of nonsafety-related system operation are designated as Category 2.

Qualification Criteria for Category 2

Category 2 instrumentation is qualified from the sensor up to, and including, the channel isolation device for the environment in which it operates to serve its intended function.

Design Criteria for Category 2

The following design criteria apply to Category 2:

- Category 2 instrumentation that is required for operation of a safety-related component is energized from the Class 1E dc uninterruptible power supply system. Otherwise, the instrumentation is energized from the non-Class 1E dc uninterruptible power system.
- The out-of-service interval is based on the technical specification requirements on out-of-service for the system the instrument serves where applicable.
- Servicing, testing, and calibration programs are implemented to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between testing is less than the time interval between shutdowns, a capability for testing during power operation is provided.
- The design provides administrative control of the access for removing channels from service.
- The design provides administrative control of the access to setpoint adjustments, module calibration adjustments, and test points.
- The monitoring instrumentation design minimizes the potential for the development of conditions that cause displays to give anomalous indications that are potentially confusing to the control room operating staff.
- The instrumentation is designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.
- To the extent practicable, monitoring instrumentation inputs are from sensors that directly measure the desired variables. An indirect measurement is made only when it can be shown by analysis to provide unambiguous information.
- Periodic checking, testing, calibration, and calibration verification is performed.
- The range selected for the instrumentation encompasses the expected operating range of the monitored variable.

Information Processing and Display Interface Criteria for Category 2

The instrumentation signal is processed for display on demand. Recording requirements are determined on a case-by-case basis.

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7.5.2.2.3 Category 3

Selection Criteria for Category 3

The selection criteria for Category 3 variables are subdivided according to the variable type. Types B, C, D, E, and F variables which provide backup information are designated as Category 3.

Qualification Criteria for Category 3

The instrumentation is high quality, commercial grade which is not required to provide information when exposed to a post-accident adverse environment.

Design Criteria for Category 3

The following design criteria apply to Category 3:

- Servicing, testing, and calibration programs are implemented to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between testing is less than the normal time interval between plant shutdowns, a capability for testing during power operation is provided.
- The design provides administrative control of the access for removing channels from service.
- The design provides administrative control of the access to setpoint adjustments, module calibration adjustments, and test points.
- The monitoring instrumentation design minimizes the potential for the development of conditions that cause displays to give anomalous indications that are potentially confusing to the control room operating staff.
- The instrumentation is designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.
- To the extent practicable, monitoring instrumentation inputs are from sensors that directly measure the desired variables. An indirect measurement is made only when it can be shown by analysis to provide unambiguous information.

Information Processing and Display Interface Criteria for Category 3

The instrumentation signal is processed for display on demand. Recording requirements are determined on a case-by-case basis.

7.5.2.2.4 Extended Range Instrumentation Qualification Criteria

The qualification environment for extended range instrumentation is based on the design basis accident events. The qualification value of the monitored variable is equal to the maximum range for the variable. The monitored variable is assumed to approach this peak by extrapolating the most severe initial ramp associated with the design basis accident events. The decay is considered proportional to the decay for this variable associated with the design basis accidents. No additional qualification margin is added to the extended range variable. Since extended variable ranges are nonmechanistically determined, extension of associated parameter levels is not justifiable and is, therefore, not implemented. For example, a sensor measuring containment pressure is qualified for the measured process variable range (that is, four times design pressure for steel containments), but the corresponding ambient temperature is not mechanistically linked to that pressure. Rather, the ambient temperature value is the bounding value for design basis accident events analyzed in

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Chapter 15. The extended range instrument provides information if conditions degrade beyond those postulated in the safety analysis.

7.5.3 Description of Variables

7.5.3.1 Type A Variables

Type A variables provide primary information to permit the control room operating staff to:

- Perform the diagnosis in the AP1000 emergency operating procedures.
- Take specified preplanned, manually-controlled actions, for which automatic controls are not provided, and that are required for safety-related systems to mitigate design basis accidents.

There are no specific preplanned, manually-controlled actions for safety-related systems to mitigate design basis accidents in the AP1000 design. This includes the diagnosis of plant conditions required to take preplanned manual action. Therefore, as reflected in Table 7.5-4, there are no Type A variables.

7.5.3.2 Type B Variables

Type B variables provide information to the control room operating staff to assess the process of accomplishing or maintaining critical safety functions, including the following:

- Reactivity control
- Reactor coolant system integrity
- Reactor coolant system inventory control
- Reactor core cooling
- Heat sink maintenance
- Containment environment.

Variables which provide the most direct indication (primary variable) to assess each of the six critical safety functions are designated as Category 1. Backup variables are designated as Category 2 or Category 3. These variables are listed in Table 7.5-5.

7.5.3.3 Type C Variables

Type C variables provide the control room operating staff with information to monitor the potential for breach or the actual gross breach of:

- Incore fuel cladding
- Reactor coolant system boundary
- Containment boundary.

Variables associated with monitoring radiological release from the plant are included in Type E.

Those Type C variables that provide the most direct measure of the potential for breach of one of the three fission product boundaries are designated as Category 1. Backup information that indicates potential for breach or actual breach is designated as Category 2 or Category 3. These variables are listed in Table 7.5-6.

7.5.3.4 Type D Variables

Type D variables provide sufficient information to the control room operating staff to:

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- Monitor the performance of plant safety-related systems used for mitigating the consequences of an accident and subsequent plant recovery to attain a safe shutdown condition, including verification of the automatic actuation of safety-related systems

- Take specified, preplanned, manually controlled actions using safety-related systems used for establishing and maintaining a safe shutdown condition

Primary Type D variables are designated as Category 2. Backup information is designated as Category 3. These variables are listed in Table 7.5-7.

7.5.3.5 Type E Variables

Type E variables provide the control room operating staff with information to:

- Monitor the plant areas where access may be required to service equipment to monitor or mitigate the consequences of an accident

- Estimate the magnitude of release of radioactive materials through identified pathways

- Monitor radiation levels and radioactivity in the environment surrounding the plant

- Monitor the habitability of the main control room

Primary Type E variables are designated as Category 2. Backup variables are designated as Category 3. These variables are listed in Table 7.5-8.

7.5.3.6 Type F Variables

Type F variables provide the control room operating staff with information to:

- Take preplanned manual actions using nonsafety-related systems to prevent unnecessary actuation of the safety-related systems

- Monitor the performance of the nonsafety-related systems used to mitigate the consequences of an accident

- Operate other nonsafety-related systems normally used for plant cooldown and to maintain plant shutdown conditions

Primary Type F variables are designated as Category 2. Backup variables are designated as Category 3. These variables are listed in Table 7.5-9.

7.5.4 Processing and Display Equipment

The AP1000 processing and display function is performed by equipment which is part of the protection and safety monitoring system, plant control system, and the data display and processing system. A description of each of these processing systems is provided in Section 7.1.

The protection and safety monitoring system provides signal conditioning, communications, and display functions for Category 1 variables and for Category 2 variables that are energized from the Class 1E dc uninterruptible power supply system. The plant control system and the data display and processing system provides signal conditioning, communications and display functions for Category 3 variables and for Category 2 variables that are energized from the non-Class 1E dc uninterruptible power system. The data display and processing system also provides an alternate display of the

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variables which are displayed by the protection and safety monitoring system. Electrical separation of the data display and processing system and the protection and safety monitoring system is maintained through the use of isolation devices in the interconnections connecting the two systems, as discussed in Subsection 7.1.2.10. The portion of the protection and safety monitoring system which is dedicated to providing post-accident monitoring is referred to as the qualified data processing system and is discussed in Subsection 7.1.2.5. Post-accident information is displayed in the MCR on the four divisions of safety displays of the PMS safety display subsystem.

The qualified data processing system is divided into two separate electrical divisions. Each of the two electrical divisions is connected to a Class 1E dc uninterruptible power system with sufficient battery capacity to provide necessary electrical power for at least 72 hours. If all ac power sources are lost for a period of time that exceeds 72 hours, the power supply system will be energized from the ancillary diesel generator or from ac power sources which are brought to the site from other locations. See Section 8.3.

Instrumentation associated with primary variables that are energized from the Class 1E dc uninterruptible power supply system are powered from one of the two electrical divisions with 72 hour battery capacity. Instrumentation associated with other variables that are energized from the Class 1E dc uninterruptible power supply system are powered from one of four electrical divisions with 24 hour battery capacity. If a variable exists only to provide a backup to a primary variable, it may be powered by an electrical division with a 24 hour battery capacity. In such cases, provisions are provided to enable this variable to be powered by an alternate source if it is needed to resolve a discrepancy between two primary variables in the event that all ac power sources are lost for a period in excess of 24 hours.

Class 1E position indication signals for valves and electrical breakers may be powered by an electrical division with 24 hour battery capacity. This is necessary to make full use of all four Class 1E electrical divisions to enhance fire separation criteria. The power associated with the actuation signal for each of these valves or electrical breakers is provided by an electrical division with 24 hour battery capacity, so there is no need to provide position indication beyond this period. The operator will verify that the valves or electrical breakers have achieved the proper position for long-term stable plant operation before position indication is lost. Once the position indication is lost, there is no need for further monitoring since the operator does not have any remote capability for changing the position of these components.

Electrically operated valves, which have the electrical power removed to meet the single failure criterion, are provided with redundant valve position sensors. Each of the two position sensors is powered from a different non-Class 1E power source.

7.5.5 Combined License Information

Combined License applicants referencing the AP1000 certified design will provide the information for variables listed as site specific in DCD Tables 7.5-1 and 7.5-8.

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Table 7.5-1 (Sheet 1 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
RCS wide range pressure	0-3300 psig	B1, D2, C1, F2	Harsh	Yes	3 (Note 4)	1E	Yes	Located inside containment
RCS T_H (Wide Range)	50-700°F	B1, B2, D2, F2	Harsh	Yes	2	1E	Yes	Diverse Measurement: Core exit temperature
RCS T_C (Wide Range)	50-700°F	B1, B2, D2, F2	Harsh	Yes	3 (Note 4)	1E	Yes	
Steam generator wide range water level	0-100% of span	D2, F3	Harsh	Yes	1/steam generator	1E	Yes	
Steam generator narrow range water level	0-100% of span	D2, F2	Harsh	Yes	1/steam generator	1E	Yes	
Pressurizer level (Note 11)	0-100% of span	B1, D2, F2	Harsh	Yes	3 (Note 4)	1E	Yes	
Pressurizer reference leg temperature	50-420°F	B1, D2	Harsh	Yes	3 (Note 4)	1E	Yes	
Neutron flux	10^{-6}-200% power	B1	Harsh	Yes	3 (Note 4)	1E	Yes	
Control rod position	0-264 steps	B3, D3	None	None	1/control rod	Non-1E	No	
Containment water level	El. 72'-11" to 112'-2 1/2" in discrete steps	B1, C1, F2	Harsh	Yes	3 (Note 4)	1E	Yes	

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Table 7.5-1 (Sheet 2 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Core exit temperature	200-2300°F	B1, C1, F2	Harsh	Yes	2/quadrant per Division	1E	Yes	
PRHR HX inlet temperature	50-650°F	D3	None	None	1	Non-1E	No	Primary indication is RCS T_H
PRHR HX outlet temperature	50-500°F	B2, D2	Harsh	Yes	1	1E	Yes	
PRHR flow	700-3000 gpm	B1, D2, F2	Harsh	Yes	2	1E	Yes	Diverse measurement: PRHR HX inlet isolation valve status together with PRHR HX control valve status
IRWST wide range water level	0-100% of span	B1, D2, F2	Harsh	Yes	3 (Note 4)	1E	Yes	
IRWST lower narrow range water level	0-100% of span	D2	Harsh	Yes	1	1E	No	
RCS subcooling (Note 6)	200°F Sub-cooling to 35°F super heat	B1, F2	Harsh	Yes	2 calculated variables	1E	Yes	
Passive containment cooling water flow	0-150 gpm	B1, D2	Mild	Yes	1 (Note 1)	1E	Yes	
PCS storage tank water level	5-100% of tank height	B1, D2	Mild	Yes	2	1E	Yes	Diverse measurement: PCS flow
IRWST surface temperature	50-300°F	D3	None	None	1	Non-1E	No	
IRWST bottom temperature	50-300°F	D3	None	None	1	Non-1E	No	
Steam line pressure	0-1300 psig	F2	Harsh/ Mild (Note 8)	Yes	1/steam generator	1E	Yes	

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Table 7.5-1 (Sheet 3 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Startup feedwater flow	0-600 gpm	F2	Mild	Yes	1/steam generator	1E	Yes	
Startup feedwater control valve status	Open/Closed	D2, F3	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Containment pressure	-5 to 10 psig	B1, C2, D2, F2	Mild	Yes	3 (Note 4)	1E	Yes	
Containment pressure (extended range)	0 to 240 psig	C1	Mild	Yes	3 (Note 4)	1E	Yes	
Containment area radiation (high range)	10^0-10^7 R/hr	C1, E2, F2	Harsh	Yes	3 (Note 4)	1E	Yes	
Reactor vessel hot leg water level	0-100% of span	B2, B3	Harsh	Yes	1	1E	Yes	Two instruments are provided
Plant vent noble gas radiation level	10^{-7} to 10^5 µCi/cc	C2, E2	Mild	None	1	Non-1E	No	Several instruments can fulfill range
Plant vent particulate and iodine radiation level	(Note 3)	E3	Mild	None	1/detector type	Non-1E	No	
Remotely operated containment isolation valve status (Note 13)	Open/Closed	B1, D2	Harsh/Mild/ Radiation Harsh	Yes	1/valve (Note 7)	1E	Yes	Separate divisions on series valves
Containment vacuum relief valve status	Open/Closed	B1, D2	Mild	Yes	1/valve (Note 7)	1E	Yes	Input to remotely operated containment isolation valve status variable
Boundary environs radiation	N/A	C3, E3	None	None	N/A	Non-1E	No	Site specific
Hydrogen concentration	0-20%	C3	None	None	1	Non-1E	No	Three instruments are provided
Class 1E dc switchboard voltages	0-300 Vdc	D2	Mild	Yes	1/ switchboard	1E	Yes	
Diesel generator status	On/Off	F3	None	None	1/diesel generator	Non-1E	No	
Diesel generator ready to pick up load	On/Off	F3	None	None	1/generator	Non-1E	No	

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Table 7.5-1 (Sheet 4 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Diesel generator load	0-6000 kW	F3	None	None	1/diesel generator	Non-1E	No	
Voltage for diesel-backed buses	0-8600V	F3	None	None	3/bus	Non-1E	No	
Power supply to diesel-backed buses	On/Off	F3	None	None	1/supply source/bus	Non-1E	No	
RCP bearing water temperature	50-400°F	F3	None	None	1/RCP (Note 10)	Non-1E	No	
RCP breaker status	Open/Closed	D2, F3	Mild	Yes	1/breaker (Note 10)	1E	No	
Reactor trip breaker status	Open/Closed	D2	Mild	Yes	1/breaker (Note 10)	1E	No	
MCR air storage bottle pressure	0-5000 psig	D2	Mild	None	1	Non-1E	No	Two instruments are provided
Turbine stop valve status	Open/Closed	D2	None (Note 12)	None	1/valve	Non-1E	No	
Turbine control valve status	Open/Closed	D2	None (Note 12)	None	1/valve	Non-1E	No	
Pressurizer pressure	1500-2500 psig	B1, D2	Harsh	Yes	3 (Note 4)	1E	Yes	
Pressurizer safety valve status	Open/Closed	D2	Harsh	None	1/valve	Non-1E	No	
Pressurizer heater power (current)	0-800 amps	F3	None	None	1/group	Non-1E	No	
Steam generator PORV status	Open/Closed	D2, F3	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Steam generator PORV block valve status	Open/Closed	D2, F3	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Steam generator safety valve status	Open/Closed	D2	Harsh	None	1/valve	Non-1E	No	
Main feedwater isolation valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Main feedwater flow	0-9x10^6 lb/hr	F3	None	None	1/feedline	Non-1E	No	
Main feedwater control valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	

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Table 7.5-1 (Sheet 5 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Steam generator blowdown isolation valve status	Open/Closed	B1, D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	Input to remotely operated containment isolation valve status variable
Steam flow	0-9x10^6 lb/hr	F3	None	None	1/steam generator	Non-1E	No	
Main steam line isolation valve status	Open/Closed	D2, F3	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Main steam line isolation bypass valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Main feedwater pump status	On/Off	D2, F3	Mild	None	1/pump	Non-1E	No	
Main to startup feedwater crossover valve status	Open/Closed	D2, F3	Mild	None	1	Non-1E	No	
Startup feedwater pump status	On/Off	F3	None	None	1/pump	Non-1E	No	
Circulating water pump status	On/Off	F3	None	None	1/pump	Non-1E	No	
Condenser backpressure	0-30 in. Hg(A)	F3	None	None	1	Non-1E	No	
Steam line condensate drain control valve status	Open/Closed	D2, F3	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Steam line condensate drain isolation valve status	Open/Closed	D2, F3	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Startup feedwater Isolation valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Condenser steam dump valve status	Open/Closed	D2, F3	Mild	None	1/valve	Non-1E	No	
Condensate storage tank water level	0-100% of span	F3	None	None	1	Non-1E	No	
PCS water storage tank isolation valve status (AOV)	Open/Closed	D2	Radiation Harsh	Yes	1/valve (Note 7)	1E	Yes	
PCS water storage tank isolation valve status (MOV)	Open/Closed	D2	Radiation Harsh	Yes	1 (Note 7)	1E	Yes	

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Table 7.5-1 (Sheet 6 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
PCS water storage tank series isolation valve status (MOV)	Open/Closed	D2	Radiation Harsh	Yes	1/valve (Note 7)	1E	Yes	
Containment temperature	32-400°F	D2, F3	Harsh	None	1	Non-1E	No	
CCS surge tank level	0-100% of span	F3	None	None	1	Non-1E	No	
CCS flow	0-20,000 gpm	F3	None	None	1	Non-1E	No	
CCS pump status	On/Off	F3	None	None	1/pump	Non-1E	No	
CCS flow to RNS valve status	Open/Closed	F3	None	None	1/valve	Non-1E	No	
CCS containment isolation valve status	Open/Closed	D2	Mild/Harsh	Yes	1/valve (Note 7)	1E	Yes	Harsh applies only to the valve located inside containment
CCS flow to RCPs valve status	Open/Closed	F3	None	None	1/valve	Non-1E	No	
CCS pump inlet temperature	50-200°F	F3	None	None	1	Non-1E	No	
CCS heat exchanger outlet temperature	50-130°F	F3	None	None	1	Non-1E	No	
Containment fan cooler status	On/Off	F3	None	None	1/fan	Non-1E	No	

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Table 7.5-1 (Sheet 7 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Water-cooled chiller status	On/Off	F3	None	None	1/chiller	Non-1E	No	
Water-cooled chilled water pump status	On/Off	F3	None	None	1/pump	Non-1E	No	
Water-cooled chilled water valve status	Open/Closed	F3	None	None	1/valve	Non-1E	No	
Spent fuel pool pump flow	0-1500 gpm	F3	None	None	1/pump	Non-1E	No	
Spent fuel pool temperature	50-250°F	F3	None	None	1	Non-1E	No	
Spent fuel pool water level	0-100% of span	D2, F3	Radiation Harsh	Yes	3 (Note 4)	1E	Yes	
SFS to SGS compartment valve status	Open/Closed	D2	Harsh	Yes	1 (Note 7)	1E	Yes	
SFS to cont. sump valve status	Open/Closed	D2	Harsh	Yes	1 (Note 7)	1E	Yes	
SFS floodup valve status	Open/Closed	D2	Harsh	Yes	1 (Note 7)	1E	Yes	
SFS suction and discharge line isolation valve status	Open/Closed	B1, D2	Radiation Harsh/Harsh	Yes	1/valve (Note 7)	1E	Yes	Input to remotely operated containment isolation valve status variable. Harsh applies only to the valve located inside containment
CMT discharge isolation valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
CMT inlet isolation valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
CMT upper water level	0%-100% of span	D2, F2	Harsh	Yes	1/tank	1E	Yes	
CMT lower water level	0%-100% of span	D2, F2	Harsh	Yes	1/tank	1E	Yes	
IRWST injection isolation valve (Squib)	Open/Not Open	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
IRWST line isolation valve status (MOV)	Open/Closed	D3	None	None	1/valve	Non-1E	No	

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Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
ADS: first, second and third stage valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
ADS fourth stage valve status (Squib)	Open/Not Open	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
ADS fourth stage valve status (MOV)	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Pressurizer spray valve status	Open/Closed	F3	None	None	1/valve	Non-1E	No	
PRHR HX inlet isolation valve status	Open/Closed	B1, D2	Harsh	Yes	1 (Note 7)	1E	Yes	Together with PRHR HX control valve status as diverse variables to PRHR flow
PRHR HX control valve status	Open/Closed	B1, D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	Together with PRHR HX inlet isolation valve status as diverse variables to PRHR flow
IRWST gutter bypass isolation valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Accumulator pressure	100-800 psig	D2	Harsh	None	1/tank	Non-1E	No	
Accumulator isolation valve status	Open/Closed	D3	None	None	1/valve	Non-1E	No	
Accumulator vent valve status	Open/Closed	F3	None	None	1/valve	Non-1E	No	
Auxiliary spray line isolation valve status	Open/Closed	D2, F3	Harsh	Yes	1 (Note 7)	1E	Yes	
Purification stop valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Purification line containment isolation valve status	Open/Closed	B1, D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	Input to remotely operated containment isolation valve status variable

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Table 7.5-1 (Sheet 9 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Containment recirculation isolation valve status (Squib)	Open/Not Open	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Containment recirculation isolation valve status (MOV)	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
Boric acid tank level	0-100%	F3	None	None	1	Non-1E	No	
Demineralized water isolation valve status	Open/Closed	D2	Radiation Harsh	Yes	1/valve (Note 7)	1E	Yes	
Boric acid flow	0-175 gpm	F3	None	None	1	Non-1E	No	
Demineralized water flow	0-175 gpm	F3	None	None	1	Non-1E	No	
Makeup blend valve status	Position	F3	None	None	1	Non-1E	No	
Makeup flow	0-175 gpm	F3	None	None	1	Non-1E	No	
Makeup pump status	On/Off	F3	None	None	1/pump	Non-1E	No	
Makeup flow control valve status	Position	F3	None	None	1	Non-1E	No	
CVS makeup line containment isolation valve status	Open/Closed	D2	Radiation Harsh/Harsh	Yes	1/valve (Note 7)	1E	Yes	Harsh applies only to the valve located inside containment
Letdown flow	0-120 gpm	F3	None	None	1	Non-1E	No	
RNS hot leg suction isolation valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
RNS pump discharge isolation valve status	Open/Closed	D2	Radiation Harsh	Yes	1 (Note 7)	1E	Yes	
RNS flow	0-3000 gpm	F3	None	None	1/pump	Non-1E	No	
RCS sampling line isolation valve status	Open/Closed	E3	Harsh	None	1/valve	Non-1E	No	

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Table 7.5-1 (Sheet 10 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
IRWST to RNS suction valve status	Open/Closed	B1, D2, F3	Harsh/Radiation Harsh	Yes	1/valve (Note 7)	1E	Yes	Harsh applies only to the valve located inside containment
RNS discharge to IRWST valve status	Open/Closed	F3	None	None	1	Non-1E	No	
RNS pump status	On/Off	F3	None	None	1/pump	Non-1E	No	
Reactor vessel head vent valve status	Open/Closed	D2	Harsh	Yes	1/valve (Note 7)	1E	Yes	
MCR return air isolation valve status	Open/Closed	D2, F3	Mild	Yes	1/valve (Note 7)	1E	Yes	
MCR toilet exhaust isolation valve status	Open/Closed	D2	Mild	Yes	1/valve (Note 7)	1E	Yes	
MCR supply air isolation valve status	Open/Closed	D2, F3	Mild	Yes	1/valve (Note 7)	1E	Yes	
MCR differential pressure	-1" to +1" wg	D2	Mild	Yes	1	1E	Yes	
MCR air delivery flowrate	0-80 cfm	D2	Mild	Yes	1	1E	Yes	
MCR pressure relief isolation valve status	Open/Closed	D2	Mild	Yes	1/valve	1E	Yes	
MCR vent isolation valve status	Open/Closed	D2	Mild	Yes	1/valve (Note 7)	1E	Yes	
MCR air delivery isolation valve status	Open/Closed	D2	Mild	Yes	1/valve (Note 7)	1E	Yes	
MCR electrical load status	Open/Closed	D2	Mild	Yes	1/contactor	1E	Yes	
Instrument air header pressure	0-125 psig	F3	None	None	1	Non-1E	No	
Service water flow	0-15,000 gpm	F3	None	None	1/pump	Non-1E	No	
Service water pump status	On/Off	F3	None	None	1/pump	Non-1E	No	

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Table 7.5-1 (Sheet 11 of 12)
Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Service water pump discharge valve status	Open/Closed	F3	None	None	1/valve	Non-1E	No	
Service water pump discharge temperature	50-150°F	F3	None	None	1/pump	Non-1E	No	
Main control room supply air radiation	Note 5	E3, F3	Mild	Yes	2 (Note 9)	1E	Yes	
Plant vent air flow	0-110% design flow	E2	Mild	None	1	Non-1E	No	
Turbine island vent discharge radiation level	10^{-6}-10^{+5} µCi/cc	C2, E2	Mild	None	1	Non-1E	No	
Turbine island vent flow	0-6,500 scfm	C2, E2	Mild	None	1	Non-1E	No	
Steam generator blowdown discharge radiation	10^{-7} to 10^{-2} µCi/cc	C2	Mild	None	1	Non-1E	No	
Steam generator blowdown electrodeionization reject flow radiation level	10^{-7} to 10^{-2} µCi/cc	C2	Mild	None	1	Non-1E	No	
Main steam line radiation level	10^{-1}-10^{3} µCi/cc	C2, E2	Mild	None	1/line	Non-1E	No	
Control support area radiation	10^{-1}-10^{4} mR/hr	E3	None	None	1	Non-1E	No	
Meteorological parameters	N/A	E3	None	None	N/A	Non-1E	No	Site specific

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Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Primary sampling station area radiation level	10^{-1}-10^{7} mR/hr	E3	None	None	1	Non-1E	No	
VES passive air filtration flow	0-2000 cfm	E3	None	None	1	Non-1E	No	

Notes:

1. The flow instrument on the lowest standpipe is the Category 1 instrument; and is diverse indication for PCS storage tank water level.
2. The same information is available in the control support area via the monitor bus. Information available on the qualified data processing system is also available at the remote shutdown workstation.
3. Extended range fixed particulate and iodine sample filters are provided for removal for onsite laboratory analysis meeting range 10^{-3} to 10^{2} µCi/cc. This extended range radiation level is not displayed in the control room. Radiation level from the normal range particulate monitor (range: 10^{-12} to 10^{-7} µCi/cc) and normal range iodines monitor (range: 10^{-11} to 10^{-6} µCi/cc) is indicated in the control room.
4. The number of instruments required after stable plant conditions is two. A third channel is available through temporary connections to resolve information ambiguity if necessary (See Subsection 7.5.4).
5. Noble gas: 10^{-7} to 10^{-1} µCi/cc
   Particulate: 10^{-12} to 10^{-7} µCi/cc
   Iodines: 10^{-11} to 10^{-5} µCi/cc
6. Degree of subcooling is calculated from RCS wide range pressure and core exit temperature.
7. This instrument is only required to be provided with 24 hour battery capacity.
8. Two steam line pressure instruments per SG are located inside containment, and are qualified for a harsh environment. Two steam line pressure instruments per SG are located outside containment (not in MSIV compartment), and are qualified for a mild environment.
9. MCR supply air radiation monitoring is not required after MCR has been isolated.
10. This instrument is only required when non-safety power is available.
11. The pressurizer level signal is compensated by pressurizer reference leg temperature, pressurizer pressure, and RCS wide range pressure.
12. These devices are backup verification to QDPS status parameters. These devices are purchased to perform in their anticipated service environments for the plant conditions for which they must function.
13. Remotely operated containment isolation valve status applies to components that receive the ESF containment isolation signal (T signal). SGS-PL-V075A/B are not containment isolation valves, but are conservatively included because they receive the T signal. SGS-PL-V075A/B are in series with containment isolation valves SGS-PL-V074A/B that also receive the T signal, and are functionally equivalent.

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Table 7.5-2
Summary of Selection of Criteria

Type	Category 1	Category 2	Category 3
A	Primary variables that are used for diagnosis or providing information necessary for operator action	Variables that provide backup information	None
B	Primary variables that are used for monitoring the process of accomplishing or maintaining critical safety functions	Variables that provide backup information	Variables that provide backup information
C	Primary variables that are used for monitoring the potential for breach of a fission product barrier	Variables that provide backup information	Variables that provide backup information
D	None	Primary variables used for monitoring the performance of plant safety-related systems	Variables that provide backup information and monitor the performance of plant safety-related systems
E	None	Primary variables to be monitored in determining the magnitude of the release of radioactive materials and for continuously assessing such releases.	Variables that provide backup information in determining the magnitude of the release of radioactive materials and for continuously assessing such releases
F	None	Primary variables to be monitored to implement preplanned manual actions using nonsafety-related systems	Variables that provide backup information and for monitoring the performance of nonsafety-related systems

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Table 7.5-3
Summary of Qualification, Design, and Interface Requirements

	Category 1	Category 2	Category 3
Qualification			
Environmental	Yes	Yes	No
Seismic	Yes	As appropriate (See Subsection 7.5.2.2.2.)	No
Design			
Single failure	Yes	No	No
Power supply	Class 1E dc battery	Class 1E dc or Non-Class 1E dc battery onsite (As appropriate, see Subsection 7.5.2.2.2.)	Non-Class 1E
Channel out of service	Technical Specifications	As appropriate (See Subsection 7.5.2.2.2.)	No specific requirement
Interface			
Minimum indication	Immediately accessible	On demand	On demand
Recording	Yes	As required (See Subsection 7.5.2.2.2.)	As required (See Subsection 7.5.2.2.3.)

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Table 7.5-4
Summary of Type A Variables

There are no Type A variables for AP1000.

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Table 7.5-5
Summary of Type B Variables

Function Monitored	Variable	Type/Category
Reactivity Control	Neutron flux Control rod position	B1 B3
Reactor Coolant System Integrity	RCS wide range pressure RCS wide range T_{hot} RCS wide range T_{cold} Containment water level Containment pressure	B1 B1 B1 B1 B1
Reactor Coolant Inventory Control	Pressurizer level Pressurizer reference leg temperature Pressurizer pressure RCS wide range pressure Reactor vessel - hot leg water level	B1 B1 B1 B1 B3
Reactor Core Cooling	Core exit temperature RCS subcooling RCS wide range pressure RCS wide range T_{hot} RCS wide range T_{cold} Reactor vessel - hot leg water level	B1 B1 B1 B2 B2 B2
Heat Sink Maintenance	IRWST wide range water level PRHR flow PRHR HX outlet temperature PRHR HX inlet isolation valve status PRHR HX control valve status PCS storage tank water level Passive containment cooling water flow IRWST to RNS suction valve status	B1 B1 B2 B1 B1 B1 B1 B1
Containment Environment	Containment pressure Remotely operated containment isolation valve status Containment water level	B1 B1 B1

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Table 7.5-6
Summary of Type C Variables

Function Monitored	Variable	Type/Category
Incore Fuel Clad	Core exit temperature	C1
RCS Boundary	RCS wide range pressure	C1
	Containment pressure	C2
	Containment water level	C1
	Containment area radiation (high range)	C1
	Turbine island vent discharge radiation level	C2
	Turbine island vent flow	C2
	Steam generator blowdown discharge radiation	C2
	Steam generator blowdown electrodeionization reject flow radiation level	C2
	Main steam line radiation level	C2
Containment Boundary	Containment pressure (extended range)	C1
	Plant vent noble gas radiation level	C2
	Hydrogen concentration	C3
	Boundary environs radiation	C3

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Table 7.5-7 (Sheet 1 of 5)
Summary of Type D Variables

System	Variable	Type/Category
Reactivity Control System	Reactor trip breaker status	D2
	Control rod position	D3
Pressurizer Level and Pressure Control	Pressurizer safety valve status	D2
	Pressurizer level	D2
	RCS wide range pressure	D2
	Pressurizer pressure	D2
	Pressurizer reference leg temperature	D2
RCS Loops	RCS wide range T_{hot}	D2
	RCS wide range T_{cold}	D2
	RCP breaker status	D2
Secondary Pressure and Level Control	Steam generator PORV status	D2
	Steam generator PORV block valve status	D2
	Steam generator safety valve status	D2
	Main feedwater isolation valve status	D2
	Steam generator wide range water level	D2
	Steam generator narrow range water level	D2
	Steam generator blowdown isolation valve status	D2

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Table 7.5-7 (Sheet 2 of 5)
Summary of Type D Variables

System	Variable	Type/Category
Secondary Pressure and Level Control (continued)	Main feedwater pump status	D2
	Main feedwater control valve status	D2
	Main steam line isolation valve status	D2
	Main steam line isolation bypass valve status	D2
Startup Feedwater	Startup feedwater control valve status	D2
	Startup feedwater isolation valve status	D2
	Main to startup feedwater crossover valve status	D2
Safeguards	Containment pressure	D2
	Accumulator pressure	D2
	Core makeup tank upper water level	D2
	Core makeup tank lower water level	D2
	IRWST/line isolation valve status (MOV)	D3
	IRWST/injection isolation valve status (Squib)	D2
	ADS: first, second, and third stage valve status	D2
	ADS fourth stage valve status (MOV)	D2
	ADS fourth stage valve status (Squib)	D2
	PRHR heat exchanger inlet isolation valve status	D2
	PRHR heat exchanger control valve status	D2
	Reactor vessel head vent valve status	D2
	CMT discharge isolation valve status	D2
	CMT inlet isolation valve status	D2

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Table 7.5-7 (Sheet 3 of 5)
Summary of Type D Variables

System	Variable	Type/Category
Safeguards (continued)	Accumulator isolation valve status	D3
	PRHR flow	D2
	Containment recirculation isolation valve status (MOV)	D2
	Containment recirculation isolation valve status (Squib)	D2
	PRHR HX inlet temperature	D3
	PRHR HX outlet temperature	D2
	IRWST surface temperature	D3
	IRWST bottom temperature	D3
	IRWST wide range water level	D2
	IRWST lower narrow range water level	D2
	IRWST gutter bypass isolation valve status	D2
	CCS containment isolation valve status	D2
	Remotely operated containment isolation valve status	D2
Chemical and Volume Control	Auxiliary spray line isolation valve status	D2
	Purification stop valve status	D2
	Purification line containment isolation valve status	D2
	Demineralized water isolation valve status	D2
	CVS makeup line containment isolation valve status	D2
Normal Residual Heat Removal	RNS hot leg suction isolation valve status	D2
	RNS pump discharge isolation valve status	D2
	IRWST to RNS suction valve status	D2

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Table 7.5-7 (Sheet 4 of 5)
Summary of Type D Variables

System	Variable	Type/Category
Electric Power	Class 1E dc switchboard voltage	D2
Spent Fuel Pool	Spent fuel pool water level	D2
	SFS to SGS compartment valve status	D2
	SFS to containment valve status	D2
	SFS floodup valve status	D2
	SFS suction and discharge line isolation valve status	D2
Containment Cooling	Containment temperature	D2
	PCS water storage tank series isolation valve status (MOV)	D2
	PCS water storage tank isolation valve status (MOV)	D2
	PCS water storage tank isolation valve status (AOV)	D2
	Passive containment cooling water flow	D2
	PCS storage tank water level	D2
Containment	Containment vacuum relief valve status	D2

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Table 7.5-7 (Sheet 5 of 5)
Summary of Type D Variables

System	Variable	Type/Category
HVAC System Status	MCR return air isolation valve status	D2
	MCR toilet exhaust isolation valve status	D2
	MCR supply air isolation valve status	D2
	MCR air delivery isolation valve status	D2
	MCR pressure relief isolation valve status	D2
	MCR electrical load status	D2
	MCR vent isolation valve status	D2
	MCR air storage bottle pressure	D2
	MCR differential pressure	D2
	MCR air delivery flowrate	D2
Main Steam	Turbine stop valve status	D2
	Turbine control valve status	D2
	Condenser steam dump valve status	D2
	Steam line condensate drain control valve status	D2
	Steam line condensate drain isolation valve status	D2

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Table 7.5-8
Summary of Type E Variables

Function Monitored	Variable	Type/Category
Containment Radiation	Containment area radiation (high level)	E2
Area Radiation	Control support area radiation	E3
	Primary sampling station area radiation level	E3
Airborne Radioactivity Released from Plant	Turbine island vent discharge radiation level	E2
	Turbine island vent flow	E2
	Plant vent noble gas radiation level	E2
	Plant vent air flow	E2
	Main steam line radiation level	E2
	Plant vent particulate and iodine radiation level	E3
	Main control room supply air radiation level	E3
Environs Radiation and Radioactivity	Site specific	E3
Meteorology	Site specific	E3
Accident Sampling	RCS sampling line isolation valve status	E3
	Containment air (grab sample analysis)	E3
MCR Filtration Flow	VES passive air filtration flow	E3

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Table 7.5-9 (Sheet 1 of 3)
Summary of Type F Variables

Variable	Type/Category
Monitoring for preplanned manual nonsafety-related system actions	
RCS wide range pressure	F2
RCS wide range T_{hot}	F2
RCS wide range T_{cold}	F2
Steam generator narrow range water level	F2
Pressurizer level	F2
Containment pressure	F2
Steam line pressure	F2
Containment water level	F2
IRWST wide range water level	F2
Startup feedwater flow	F2
Containment area radiation (high range)	F2
Core exit temperature	F2
RCS subcooling	F2
PRHR flow	F2
Core makeup tank upper water level	F2
Core makeup tank lower water level	F2
Monitoring for nonsafety-related system performance	
Pressurizer heater power (current)	F3
Steam generator PORV status	F3
Steam generator PORV block valve status	F3
Startup feedwater control valve status	F3
Main feedwater flow	F3
Steam generator wide range water level	F3
Steam flow	F3
Main steam line isolation valve status	F3
Main feedwater pump status	F3
Startup feedwater pump status	F3
Condenser steam dump valve status	F3
Condensate storage tank water level	F3
Pressurizer spray valve status	F3

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Table 7.5-9 (Sheet 2 of 3)
Summary of Type F Variables

Variable	Type/Category
Auxiliary spray line isolation valve status	F3
Makeup flow	F3
Makeup pump status	F3
Letdown flow	F3
Circulating water pump status	F3
Condenser backpressure	F3
Accumulator vent valve status	F3
Boric acid tank level	F3
Boric acid flow	F3
Demineralized water flow	F3
Makeup blend valve status	F3
Makeup flow control valve status	F3
RNS flow	F3
RNS pump status	F3
IRWST to RNS suction valve status	F3
RNS discharge to IRWST valve status	F3
CCS surge tank level	F3
CCS flow	F3
CCS pump status	F3
CCS flow to RNS valve status	F3
CCS flow to RCPs valve status	F3
CCS pump inlet temperature	F3
CCS heat exchanger outlet temperature	F3
Diesel generator status	F3
Diesel generator ready to pick up load	F3
Diesel generator load	F3
Voltage for diesel-backed buses	F3
Power supply to diesel-backed buses	F3
RCP bearing water temperature	F3
RCP breaker status	F3
Containment fan cooler status	F3
Water-cooled chiller status	F3

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Table 7.5-9 (Sheet 3 of 3)
Summary of Type F Variables

Variable	Type/Category
Water-cooled chilled water pump status	F3
Water-cooled chilled water valve status	F3
Containment temperature	F3
MCR supply air isolation valve status	F3
MCR return air isolation valve status	F3
Main control room supply air radiation	F3
Service water flow	F3
Service water pump status	F3
Service water pump discharge valve status	F3
Service water pump discharge temperature	F3
Instrument air header pressure	F3
Spent fuel pool pump flow	F3
Spent fuel pool temperature	F3
Spent fuel pool water level	F3
Main to startup feedwater crossover valve status	F3
Steam line condensate drain control valve status	F3
Steam line condensate drain isolation valve status	F3

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7.6 Interlock Systems Important to Safety

This section discusses interlock systems which operate to reduce the probability of occurrence of specific events or to verify the state of a safety system. These include interlocks to prevent overpressurization of low-pressure systems and interlocks to verify availability of engineered safety features.

7.6.1 Prevention of Overpressurization of Low-Pressure Systems

7.6.1.1 Description of Normal Residual Heat Removal Isolation Valve Interlocks

An interlock is provided for the normally closed, motor-operated normal residual heat removal system (RNS) inner and outer suction isolation valves. The interlock prevents the suction valves for the normal residual heat removal system from being opened by operator action unless the reactor coolant system pressure is less than a preset pressure and both the suction and discharge valves for the in-containment refueling water storage tank are in a closed position.

There are two parallel sets of two motor-operated valves in series in the normal residual heat removal system pumps suction line from the reactor coolant system hot leg. The two valves nearest the reactor coolant system are designated as the inner isolation valves. The two valves nearest the normal residual heat removal system pumps are designated as the outer isolation valves. Logic for the outer valves is similar to that provided for the inner isolation valves, except that equipment diversity is provided by virtue of the fact that the pressure transmitters used for valve interlocks on the inner valves are diverse from the pressure transmitters used for the outer valve interlocks. Typically, this diversity is achieved by procuring wide range pressure transmitters either with similar measurement principles from different vendors, or with different measurement principles (from either the same or different vendors).

Each valve is interlocked so that it cannot be opened unless the reactor coolant system pressure is below a preset pressure. This interlock prevents the valve from being opened when the reactor coolant system pressure is above the normal residual heat removal system design pressure.

Figure 7.2-1, Sheet 18 illustrates the interlock logic that applies to these valves. The logic, shown on Sheet 18, is replicated twice, once for each parallel path. This interlock logic prevents the two series isolation valves from being opened while the reactor coolant system is pressurized above a set pressure.

The valves may be closed by operator action from the main control room at any time. To prevent an inadvertent closure of these valves, no auto-closure interlock that would close the valves on high reactor coolant system pressure is included.

The normal residual heat removal system relief valves provide adequate system pressure protection for conditions after the valves have been opened. (This is discussed in Subsection 5.2.2.1). Alarms are provided in the main control room and on the remote shutdown workstation to alert the operator if reactor coolant system pressure exceeds the normal residual heat removal system design pressure after the valves are opened.

7.6.1.2 Analysis of Normal Residual Heat Removal Valve Interlocks

IEEE 603-1991 and IEEE 338-1987 criteria do not apply to the normal residual heat removal isolation valve interlocks. Their function is not required during, or after, a design basis event. However, because of the possible severity of the consequences of loss of function, the requirements of IEEE 603-1991 are applied with the following comments:

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- For the purpose of applying IEEE 603-1991, the protection system is the two parallel sets of two valves in series and the components of their interlock circuitry. The inner valve is powered by a separate power supply from the outer valve of each series combination.

- Online testability; IEEE 603-1991, Paragraph 5.7: The pressure interlock signals and logic are tested on line to the maximum extent possible without adversely affecting safety. This test includes the initiating signals for the interlocks from the protection and safety monitoring system cabinets.

- IEEE 603-1991, Paragraph 6.8.2: This requirement does not apply, as the setpoints are independent of mode of operation and are not changed.

7.6.2 Availability of Engineered Safety Features

7.6.2.1 Passive Residual Heat Removal Heat Exchanger Inlet Isolation Valve

The passive core cooling system passive residual heat removal heat exchanger inlet line includes a normally open motor-operated isolation valve that can be manually controlled from either the main control room or the remote shutdown workstation. The generation of the confirmatory open signal to this valve is described in Subsection 7.3.1.2.7.

The use of a confirmatory open signal to this valve provides a means to automatically override cases when this isolation valve is closed for short periods of time.

The operation of the valve is controlled by an actuation control circuit that functions in the following manner:

- The protection and safety monitoring system provides a confirmatory open signal that will automatically open the valve in the case of a passive residual heat removal heat exchanger actuation signal if power is available to the valve.

- The plant control system provides control features to allow the valve to be operated from the normal operator workstations.

- The control circuit has a valve close actuation function that closes the valve when a control on the operator workstation is manually actuated. This function is required when performing periodic inservice testing of the passive residual heat removal heat exchanger discharge valves when the reactor is shut down. The automatic confirmatory open signal makes the passive residual heat exchanger available during inservice testing of the discharge valves.

- The maximum permissible time that a passive residual heat removal heat exchanger inlet isolation valve can be closed is specified in technical specifications.

The valve is interlocked so that:

- If the valve is closed and power not removed, it opens automatically on receipt of a confirmatory open signal.

- It cannot be manually closed when a confirmatory open signal is present.

When the passive residual heat removal heat exchanger is required to be functional, the inlet isolation valve is open with power removed to prevent inadvertent closure due to a fire.

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This normally open motor-operated valve has alarms, indicating valve mispositioning (with regard to its passive core cooling function). The alarm actuates in the main control room and the remote shutdown workstation. Redundant valve position sensors are used to generate the alarm when the valve is not fully open.

7.6.2.2 Core Makeup Tank Cold Leg Balance Line Isolation Valves

Each core makeup tank has a cold leg balance line which is provided with a normally open, motor-operated, isolation valve. The balance line isolation valves, for each core makeup tank, may be manually controlled from either the main control room or the remote shutdown workstation. The generation of the confirmatory open signal to these valves is described in Subsection 7.3.1.2.3.

A confirmatory open signal to these valves automatically overrides any cases when the balance line isolation valve is closed for short periods of time. As a result of the confirmatory open signal, isolation of the core makeup tank cold leg balance line to permit inservice testing of the core makeup tank discharge valves, is acceptable.

The operation of each valve is controlled by an actuation control circuit that functions in the following manner:

- The protection and safety monitoring system provides a confirmatory open signal from automatic or manual core makeup tank actuation for cases when a balance line isolation valve is closed to support testing, maintenance, or outage related activities.
- The plant control system provides control features to allow the valve to be operated from the normal operator workstations.
- The control circuit has a valve close actuation function that closes the valve when a control on the operator workstation is manually actuated. This function is provided for performing periodic operability tests of the core makeup tank discharge valves when the reactor is operating. The automatic confirmatory open signal makes the balance line isolation valves available during operability testing of the valves.
- An administrative procedure can be invoked to maintain the valve in the closed position. This administrative procedure allows the valve to be maintained closed if needed for leakage isolation. The maximum permissible time that a core makeup tank cold leg balance line isolation valve can be closed is specified in technical specifications.

Each valve is interlocked so that:

- It receives a confirmatory open signal on a core makeup tank actuation signal.
- It cannot be manually closed when a confirmatory open signal is present.
- There is an additional control circuit that will automatically open the valve when the pressurizer water level increases above the P-12 interlock.

During power and shutdown operations, the core makeup tank cold leg balance line isolation valve remains open. To prevent an inadvertent closure of the valve, redundant output cards are used in the protection and safety monitoring system cabinet. As a result, it is not necessary to lock out control circuit power during normal operation. However, motive power is removed from this valve for risk reduction purposes noted in Chapter 19.

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These normally open motor-operated valves have alarms, indicating valve mispositioning (with regard to their passive core cooling function). The alarms actuate in the main control room and the remote shutdown workstation. Redundant position sensors are used to generate the alarm when the valve is not fully open.

7.6.2.3 Interlocks for the Accumulator Isolation Valve and IRWST Discharge Valve

The accumulator isolation and in-containment refueling water storage tank injection isolation valves are safety-related in order to retain their pressure boundary and remain in their open position. The accumulator isolation and in-containment refueling water storage tank injection valve operators are nonsafety-related since the valves are not required to change position to mitigate an accident. The Chapter 15 safety analyses assume that these valves are not subject to valve mispositioning (prior to an accident) or spurious closure (during an accident). Valve mispositioning and spurious closure are prevented by the following:

- The Technical Specifications require these valves to be open and power locked out whenever these injection paths are required to be available. The accumulators are required to be available when the reactor coolant system pressure is above 1000 psig. Both in-containment refueling water storage tank injection lines are required to be available in Modes 1, 2, 3, and 4. One in-containment refueling water storage tank injection line is required to be available in Mode 5 and in Mode 6.

- The Technical Specifications require verification that the motor-operated valves are open every 12 hours. They also require verification that power is removed every 31 days.

- With power locked out, redundant (nonsafety-related) valve position indication is provided in the main control room and remote shutdown workstation. Valve position indication and alarm are provided to alert the operator if these valves are mispositioned. These indications are powered by different nonsafety-related power supplies.

In addition, the valves have a confirmatory open signal during an accident (safeguards actuation signal for accumulator motor-operated valves and automatic depressurization system stage 4 signal for in-containment refueling water storage tank motor-operated valves). The valves also have an automatic open signal when their close permissives (P-11 for accumulator motor-operated valves and P-12 for in-containment refueling water storage tank motor-operated valves) clear during plant startup. The confirmatory open and the automatic open control signals are provided to the valve operator by the nonsafety-related plant control system.

7.6.2.4 Interlock for Containment Vacuum Relief Isolation Valves

The containment vacuum relief path includes normally closed motor-operated isolation valves, which are located outside the containment and open automatically to provide a flow path to allow atmospheric air into the containment to equalize a negative differential pressure across the containment vessel shell. These valves also perform a containment isolation function when vacuum relief is not required. An interlock ensures the availability of the engineered safety features for the vacuum relief isolation valves to perform their vacuum relief and containment isolation functions. The opening of these valves at any time other than the mitigation of a negative pressure condition or for required testing is controlled by the following:

- Technical Specifications for containment isolation valves require verification every 31 days that the containment purge isolation valves are closed with a few appropriate exceptions.

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- Safety-related position indication is available in the main control room and in the remote shutdown workstation on the non-safety-related operator work stations for the containment vacuum relief isolation valves and for the containment purge exhaust containment isolation valves. An alarm is provided if these valves are incorrectly positioned.

There is also a valve interlock between the inside containment purge exhaust isolation valve and the vacuum relief isolation valves, which limits the potential release of radioactivity from the containment while the containment isolation valves are being closed. The purge and vacuum relief valve interlock automatically closes the vacuum relief isolation valves any time that the inside containment purge exhaust valve is not fully closed and a coincident vacuum relief actuation signal is not present. If a vacuum relief actuation signal is present, it takes priority over the valve closure interlock.

As discussed in Subsections 6.2.3.5 and 9.4.7.2.3, the vacuum relief subsystem is used to mitigate a condition where the atmospheric air pressure outside is higher than the reactor containment air pressure (so that a negative containment pressure condition exists). The valves can be opened either automatically or manually when an excessively low containment pressure exists to mitigate the event. The actuation signal includes a control interlock provided by the protection and safety monitoring system. The interlock is a part of the logic for the control of the valves shown on Figure 7.2-1 (Sheet 19 of 21).

7.6.3 Combined License Information

This section contained no requirement for information.

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7.7 Control and Instrumentation Systems

The function of the AP1000 control systems is to establish and maintain the plant operating conditions within prescribed limits. The control system improves plant safety by minimizing the number of situations for which some protective response is initiated and relieves the operator from routine tasks.

The AP1000 control systems share a common hardware design and implementation philosophy. They are also functionally integrated to enhance responsiveness during plant transients. Specific design requirements are imposed that limit the impact of individual equipment failures. (See Subsection 7.1.3).

The control systems regulate the operating conditions in the plant automatically in response to changing plant conditions and changes in plant load demand. These operating conditions include the following:

Reactor Coolant System Temperature - The control systems function to maintain the reactor coolant system temperature at or near a programmed value. This value is a function of plant load or other operating conditions. Steam conditions for the turbine depend on the temperature maintained in the reactor coolant. Reactor coolant system temperature is also used for controlling core reactivity.

Nuclear Power Distribution - Operating limits include the distribution of nuclear energy production within the core as well as its average value. The axial distribution of the nuclear power is controlled within prescribed limits.

Reactor Coolant System Pressure - The reactor coolant system is pressurized to prevent significant boiling at operating temperatures. This pressure is controlled within limits that prevent reductions which expose the fuel to possible departure from nucleate boiling or from increases that would challenge the reactor coolant system design pressure.

Pressurizer Water Level - To provide a sufficient buffer for plant transients, the reactor coolant system pressurizer contains a prescribed volume of water and steam which depends on plant load and operating temperature.

Steam Generator Water Level - The steam generator water level is maintained within limits to provide adequate energy removal capability and to avoid moisture carryover.

Steam Dump (Turbine Bypass) - For fast and large transients such as load rejections, an additional thermal load (designated steam dump or turbine bypass) functions until nuclear power is reduced. This steam dump is also used to maintain hot no-load or hot low-load conditions prior to turbine loading. It provides a means for plant cooldown.

7.7.1 Description

The plant control and instrumentation systems described in this section perform the following functions:

Reactor Power Control System - The reactor power control system coordinates the responses of the various reactivity control mechanisms. The system enables daily load follow operation with a minimum of manual control by the operator. Load regulation and frequency control are compatible with the reactor power control system operation. Axial nuclear power distribution control is also performed by the reactor power control system.

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Rod Control System - The rod control system, in conjunction with the reactor power control system, maintains nuclear power and reactor coolant temperature, without challenges to the protection systems, during normal operating transients.

Pressurizer Pressure Control - The pressurizer pressure control system maintains or restores the pressurizer pressure to the nominal operating value following normal operating transients. The control system reacts to avoid challenges to the protection systems during these operating transients.

Pressurizer Water Level Control - The pressurizer water level control system establishes, and maintains or restores pressurizer water level to its programmed operating region. The required water level operating region is programmed as a function of reactor coolant system temperature to minimize charging and letdown requirements. No challenges to the protection system result from normal operational transients.

Feedwater Control System - The feedwater control system maintains the steam generator water level at a predetermined setpoint during steady-state operation. It also maintains the water level within operating limits during normal transient operation. The feedwater control system restores normal water level following a unit trip. The various modes of feedwater addition are automated to require a minimum of operator involvement.

Steam Dump Control - The steam dump control system reacts to prevent a reactor trip following a sudden loss of electrical load. The steam dump control system also removes stored energy and residual heat following a reactor trip so that the plant can be brought to equilibrium no-load conditions without actuation of the steam generator safety valves. The steam dump control system also provides for maintaining the plant at no-load or low-load conditions to facilitate a controlled cooldown of the plant.

Rapid Power Reduction - For large, rapid load rejections (turbine trip or grid disconnect from 50-percent power or greater) or the loss of a main feedwater pump train, a rapid nuclear power cutback is implemented. This results in a reduction of thermal power to a level that can be handled by the steam dump system in conjunction with the reactor power control system for the large load rejection. For a loss of main feedwater pump transient, the actuation of the rapid power reduction reduces power to a level that can be handled by the remaining feedwater pumps.

Defense-In-Depth Control - The plant control system provides control of systems performing defense-in-depth functions. Table 7.7-3 provides a listing of the defense-in-depth functions that are supported by the plant control system and provides a cross reference to the applicable information located in other sections of this document.

PRHR HX Modulation Control System - In an event of a loss of feedwater the PRHR HX modulation control system maintains T_{cold} to avoid an unnecessary PMS PRHR HX actuation, or safeguards actuation signal. The control system modulates the PRHR HX outlet control valve to control T_{cold}.

7.7.1.1 Reactor Power Control System

Automatic reactor power and power distribution control are the basic functions of the reactor power control system. They are achieved by varying the position of the control rods. Separate control rod banks are used to regulate reactor power and power distribution.

The reactor power control system enables the plant to respond to the following load change transients:

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- Step load changes of plus or minus 10 percent
- Ramp load increases and decreases of 5 percent per minute
- Daily load follow operations with the following profile:
  - Power ramps from 100 percent to 50 percent in 2 hours
  - Power remains at 50 percent for 2 to 10 hours
  - Power ramps back up to 100 percent in 2 hours
  - Power remains at 100 percent for the remainder of the 24-hour cycle
- Grid frequency response (denoted load regulation) resulting in a maximum of 10-percent power change at 2 percent per minute

These capabilities are accomplished without a reactor trip or steam dump actuation. During daily load follow and load regulation transients, automatic control of axial offset is provided. The system restores coolant average temperature to a value which is within the programmed temperature band following a change in load. Manual control of either the power control rods (M banks) or the axial offset control rods (AO bank) is performed within the range of defined insertion limits.

The reactor power control system uses a different control strategy for the rods used to regulate core power (M banks) from the control strategy used to regulate axial offset (AO bank). Reactor coolant system boron concentration is adjusted by the operator to account for long-term core burnup. The adjustment also inserts the gray M banks to targeted positions with both black M banks (M1 and M2) remaining in a near fully withdrawn position and the AO bank within its normal operating range. During load follow or load regulation response transients, the power control and the axial offset control subsystems jointly function to control both core power and axial offset. The following two subsections provide a description of each control subsystem.

7.7.1.1.1 Power Control

The power control subsystem controls the reactor coolant average temperature by regulating the M control rod bank positions. The reactor coolant loop average temperatures are determined from hot and cold leg measurements in each reactor coolant loop. The average coolant temperature (T_{avg}) is computed for each loop, where:

$$T_{avg} = \frac{T_{hot} + T_{cold}}{2}$$

The error between the programmed reference temperature (based on turbine first stage pressure) and the measured signal selected median loop T_{avg} constitutes the primary control signal. The programmed coolant temperature increases linearly with turbine load from the zero-power to the full-power condition.

The temperature input signals for the power control subsystem are fed from protection channels via isolation devices and the signal selector function.

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An additional control input signal is derived from the reactor power versus turbine load mismatch signal. This additional control input signal improves system performance by enhancing response and reducing transient peaks.

The deviation of the reactor coolant temperature from the programmed value is the basic control variable for reactor power control. A deadband is included in the power control subsystem so that no rod motion is demanded if the temperature error is within the deadband. As the temperature error becomes greater, the demanded rod speed becomes greater.

Separate reactor control deadbands are used for various modes of control. If the plant is in a load regulation mode of operation, then the deadband is widened from that used for base load or load follow operation. This allows the core reactivity feedbacks to assist in stabilizing the plant at the conclusion of the maneuver and reduces the total control rod movement and subsequent wear on the control rods.

A different control strategy is used at low-power levels, principally when the turbine is off-line and the steam dump system is used to regulate coolant temperature. In this mode, nuclear power is controlled directly. For this mode, a nuclear power setpoint calculator allows the operator to enter a desired power level above or below the current power level along with a desired rate of change (limited to fixed predetermined maximum limits). The nuclear power setpoint calculator then supplies a changing setpoint that provides for a linear ramp change in core power at the selected rate.

7.7.1.1.2 Axial Offset Control

The axial offset control subsystem controls the core axial offset (power difference between the top and bottom halves of the core) to a value that is within the desired control range for load follow and grid frequency change transients. This is accomplished by using a control rod bank separate from those used for the reactor power control described in Subsection 7.7.1.1.1. Measurements of axial offset are input into the axial offset control subsystem and then compared to an axial offset control "window." This window is calculated from measurements of ΔT power, along with operator inputs for the desired axial offset target value and target bandwidth. If the plant is in a load regulation mode of control, then lag compensation is applied to both the ΔT power and the axial offset signals. This provides smoothed input to the axial offset controller to avoid unnecessary axial offset control. When the axial offset error is outside the acceptable control window, the axial offset rods are actuated until the axial offset error is back inside the control window.

In order to minimize the potential for adverse interactions and optimize coordination between the power and axial offset rod control subsystems, an interface rule is established such that control of T_{avg} always takes precedence over control of power distribution.

7.7.1.2 Rod Control System

The rod control system receives rod speed and direction signals from the power control and axial offset control subsystems. The portion of the rod control system associated with the power and axial offset control subsystems each operate their own sets of control rod banks as follows:

- The power control portion operates the MA, MB, MC, MD, M1 and M2 control rod banks.
- The axial offset control portion operates the AO control rod bank.

For power control, the rod speed demand signals vary over the range of 5 to 45 inches per minute (8 to 72 steps per minute), depending on the magnitude of the input signal and the need for axial offset control. Manual control is provided to move a bank in or out at a prescribed fixed speed. In the

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automatic mode, the rods are withdrawn (or inserted) in a predetermined sequence within the limits imposed by the control interlocks as shown in Table 7.7-1.

For axial offset control, the rod speed demand signals are set to a fixed constant speed of approximately 5 inches per minute (8 steps per minute). Manual control is provided to move a bank in or out at a prescribed fixed speed. In the automatic mode, the rods are withdrawn (or inserted) within the limits imposed by the control interlocks, as shown in Table 7.7-2.

The shutdown control rod banks are always in the fully withdrawn position during normal operation and are moved to this position at a constant speed by manual control prior to criticality. A reactor trip signal causes them to fall by gravity into the core. There are four shutdown control rod banks.

The power and axial offset control rod banks are the only rods that can be manipulated under automatic control. Each bank contains one or more groups of four control rod assemblies with the exception of the axial offset control rod bank, which contains an additional group with a single assembly. Each control rod assembly in a group is electrically paralleled to move simultaneously. There is individual position indication for each control rod assembly.

Power to the rod drive mechanisms is supplied by two motor-generator sets operating from two separate 480-volt, 3-phase busses. Each generator is the synchronous type, and is driven by an induction motor. The ac power is distributed to the rod control system cabinets through the reactor trip switchgear.

The variable speed rod drive programmer used in the power control subsystem inserts small amounts of reactivity at low speed. This permits fine control of reactor coolant average temperature about a small temperature deadband, as well as furnishing control at high speed for transients such as load rejections. A summary of the control rod assembly sequencing characteristics is given below:

- The control rod groups within the same bank are stepped so that the relative position of the groups do not differ by more than one step.

- The control rod banks are programmed so that withdrawal of the banks is sequenced in a prescribed order. The programmed insertion sequence is the opposite of the withdrawal sequence. That is, the last control bank withdrawn is the first control bank inserted.

- The control bank withdrawals are programmed so that, when the first bank reaches a preset position, the next bank begins to move out simultaneously with the first bank. This preset position is determined by the maximum allowable overlap between banks (approximately 50 to 100 steps). This withdrawal sequence continues until the control banks reach the fully withdrawn position. The control bank insertion sequence is the opposite of the withdrawal sequence.

- Overlap between successive control banks is adjustable between 0 to 50 percent (0 to 135 steps), with an accuracy of ±1 step.

The constant rod speed used in the axial offset control subsystem provides a slow stable control of core axial offset. This is acceptable since axial offset changes for the design basis load follow transients generally occur over several hours and rapid response is not needed. The slow response of the axial offset control system also allows the rods used by the power control subsystem to counteract the core power reactivity changes that are induced by the axial offset rods.

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7.7.1.3 Control Rod Position Monitoring

Digital Rod Position - The digital rod position indication system measures the position of each control rod assembly using a detector consisting of discrete coils mounted concentric with the rod drive pressure housing. The coils are located axially along the pressure housing and magnetically sense the entry and presence of the rod drive shaft through its center line.

Demand Position System - The demand position system counts the pulses generated in the rod control system to provide a digital readout of the demanded bank position. The demanded and measured rod position signals are displayed in the main control room. An alarm is generated whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. The alarm is set with appropriate allowance for instrument error and within sufficiently narrow limits to prevent exceeding core design hot channel factors.

Alarms are also generated if any shutdown rod is detected to have left its fully withdrawn position, or if any M bank control rods are detected at the bottom position, except as part of the normal insertion sequence.

7.7.1.4 Control Rod Insertion Limits

With the reactor critical, the normal indication of reactivity status in the core is the position of the control rod bank in relation to reactor power (as indicated by the ΔT power monitors). The ΔT power signal is used to calculate insertion limits for the banks. The following two alarms are provided for each bank.

- A "low" alarm alerts the operator of an approach to the M bank or AO bank insertion limits. Further approach is avoided by following appropriate plant procedure.

- A "low-low" alarm alerts the operator to take immediate action to restore margin to the M bank or AO bank insertion limits. Interlocks will terminate automatic AO bank withdrawal (to prevent further insertion of the M banks) or insertion (to avoid the AO bank insertion limits).

The purpose of the control bank rod insertion alarms and interlocks is to provide warning to the operator of excessive rod insertion and to terminate the insertion. The insertion limit maintains sufficient core reactivity shutdown margin following reactor trip. It also provides a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection. Insertion limits provide confidence that acceptable nuclear peaking factors are maintained. Since the amount of shutdown reactivity required for the design shutdown margin following a reactor trip increases with increasing power, the allowable rod insertion limits are decreased (the rods must be withdrawn further) with increasing power. The insertion limits for the M banks and the AO bank are calculated from the reactor power, as measured by the ΔT power monitor, according to the following equations:

$$Z_{LL}^{M} = A + B \cdot P_{EF} + C \cdot Z_{AO} + D \cdot P_{EF} \cdot Z_{AO}$$

$$Z_{LL}^{AO} = E + F \cdot P_{EF}$$

where:

Z_{LL}^{M} = Maximum permissible insertion limit for the affected M control bank

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Z_{LL}^{AO} = Maximum permissible insertion limit for the affected AO control bank

Z_{AO} = Current AO bank position (fraction inserted)

P_{EF} = Max(P_{TH},P^{LP}), effective power for each bank

P_{TH} = ΔT Power (thermal power)

P^{LP} = Low power threshold for each bank

A,B,C,D,E,F = Constants chosen to maintain Z_{LL} ≥ the actual limit based on physics calculations

The control rod bank demand position (Z) for the M banks and the AO bank is compared to the respective Z_{LL} as follows:

- If Z - Z_{LL} ≤ G, a low alarm is actuated.
- If Z - Z_{LL} ≤ H, a low-low alarm and interlock is actuated.

Since nuclear peaking factors can be aggravated by the opposite movement of the M banks and the AO bank, the interlocks on the AO bank are different, depending on whether the M bank or the AO bank insertion limit setpoint is actuated. If an M bank insertion limit is reached, this stops AO bank withdrawal and reduces the increases in the core peaking factor. If an AO bank insertion limit is reached, this stops AO bank insertion. If the M banks are at their C-18 withdrawal limit, AO bank automatic insertion is blocked.

7.7.1.5 Control Rod Stops

Rod stops are provided to prevent abnormal power conditions that could result from excessive control rod withdrawal initiated by either a control system malfunction or operator violation of administrative procedures.

7.7.1.6 Pressurizer Pressure Control System

The primary system pressure is closely regulated during operation to prevent pressure from increasing to the point where an engineered safety features actuation is required to prevent overstressing the pressure boundary; or from decreasing to a condition where engineered safety features actuation is required to prevent the possibility of departure from nucleate boiling. Fine control of pressure to the desired setpoint is accomplished by regulating the power to a bank of heaters located in the pressurizer. Large decreases in pressure are accommodated by turning on additional heater banks and by the inherent flashing from the water mass in the pressurizer, which is at saturation. Large pressure increases are controlled by actuating pressurizer spray to condense steam.

Pressurizer pressure control is designed to provide stable and accurate control of pressure to its predetermined setpoint. Automatic pressure control is available from the point at which nominal pressure is established in the startup cycle to 100-percent power. During steady-state operating conditions, the pressurizer heater output is regulated to compensate for pressurizer heat loss and a small continuous pressurizer spray. During normal transient operation, the pressure is regulated to provide adequate margin to safety systems actuation or reactor trip. The pressurizer pressure control system is designed to minimize equipment duty (such as spray nozzle thermal cycling due to spray actuation) due to load regulation operation.

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Small changes in pressure are regulated by modulation of the variable heater control. Reset (integral) action is included to maintain pressure at its setpoint. Decreases in pressure larger than that which can be accommodated by the variable heater control results in the actuation of the backup heaters. The backup heaters are deactivated when the variable heaters alone are capable of restoring pressure.

7.7.1.7 Pressurizer Water Level Control System

The pressurizer water inventory, or level control, provides a reservoir for the reactor coolant system inventory changes that occur due to changes in reactor coolant system density. As the reactor coolant system temperature is increased from hot zero-load to full-load values, the reactor coolant system fluid expands. The pressurizer level is programmed to absorb this change. A deadband is provided around the nominal pressurizer level program to intermittently control charging and letdown. When the pressurizer water level reaches the lower limit of the deadband, it actuates the charging system. The charging system continues to operate until the level is restored to a limit above the nominal program value. When the pressurizer water level reaches the upper limit of the deadband, it actuates letdown to the liquid waste processing system.

Pressurizer water level control provides stable and accurate control of pressurizer level within a prescribed deadband around the programmed setpoint value, as derived from the plant operating parameters. Automatic level control is supplied from the point in the startup cycle where the hot zero-load level is established through 100-percent power.

7.7.1.8 Feedwater Control System

The feedwater control system consists of those controllers and associated hardware whose primary function is to regulate the flow of feedwater into the steam generator. The feedwater control system consists of two separate subsystems. The feedwater control subsystem regulates the flow of feedwater into the steam generators via the main feedwater line. The startup feedwater control subsystem regulates the flow of feedwater into the steam generators via the startup feedwater line. Flow to the startup feedwater line may be supplied by either the main or startup feedwater pump. The following two subsections provide a description of each control subsystem.

7.7.1.8.1 Feedwater Control

The feedwater control subsystem maintains a programmed water level in the shell side of the steam generator during steady-state operation, and limits the water level shrink and swell during normal plant transients. This prevents an undesirable reactor trip actuation. Indication is provided for monitoring system operation. Alarms and indications are provided to alert the plant operator of control system malfunctions or abnormal operating conditions.

Two modes of feedwater control are incorporated in the feedwater control subsystem. In the high-power control mode, the feedwater flow is regulated in response to changes in steam flow and proportional plus integral (PI)-compensated steam generator narrow range water level deviation from setpoint. In the low-power control mode, the feedwater flow is regulated in response to changes in steam generator wide-range water level and PI-compensated steam generator narrow range water level deviation from setpoint.

The transition from the low to the high-power control mode is initiated on the basis of the filtered high range feedwater flow signal. The transition point is set at a feedwater flow corresponding to a power at which reliable steam flow indication is expected. The transition point is also low enough to allow effective feedforward control using wide range water level. Tracking is provided to allow a smooth transition between control modes and between manual and automatic control.

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A high steam generator water level signal reduces the feedwater flow demand signal and closes the feedwater control valves.

7.7.1.8.2 Startup Feedwater Control

During no-load or very low power conditions, the main feedwater control subsystem is not intended to be used for automatic control of the steam generator water level. The startup feedwater control subsystem performs this function.

The startup feedwater control subsystem maintains a programmed water level in the shell side of the steam generator during low power (below approximately 10 percent of plant rated thermal power), no-load, and plant heatup and cooldown modes. During low feedwater flow demand, feedwater is controlled by the startup feedwater control subsystem. Transition between the main and startup feedwater line is automatically controlled based on flow measurements within the respective lines. The startup feedwater is also automatically actuated on signals which indicate a loss of water inventory or heat sink in the secondary side of the steam generator and will attempt to recover the inventory loss and return the steam generator water level to the programmed value. If the startup feedwater cannot recover the inventory deficit, reactor cooling is initiated by the passive residual heat removal system.

The startup feedwater control subsystem regulates the flow of feedwater in a manner which is similar to the way (main) feedwater is controlled in the low-power control mode. Feedwater flow is regulated in response to changes in steam generator wide-range water level and PI-compensated steam generator narrow range water level deviation from setpoint. Tracking is provided to allow a smooth transition between control modes and between manual and automatic control.

7.7.1.9 Steam Dump Control System

The AP1000 is designed to sustain a 100-percent load rejection, or a turbine trip from 100-percent power, without generating a reactor trip, requiring atmospheric steam relief, or actuating a pressurizer or steam generator safety valve. The automatic steam dump control system, in conjunction with other control systems, is provided to accommodate this abnormal load rejection and to reduce the effects of the transient imposed on the reactor coolant system. By bypassing main steam to the condenser, an artificial load is maintained on the primary system. This artificial load makes up the difference between the reactor power and the turbine load for load rejections and turbine trips. It also removes latent and decay heat following a reactor trip.

The steam dump system is sized to pass up to 40 percent of rated main steam flow. This capacity, in conjunction with the performance of the reactor power control system, is sufficient to handle reactor trips from any power level, turbine trips from up to a predetermined power level, and load rejections up to a predetermined magnitude. For turbine trips above a threshold power level, or load rejections exceeding a given magnitude, the steam dump system operates in conjunction with the rapid power reduction system described in Subsection 7.7.1.10 to meet the performance described in the previous paragraph.

The steam dump control system has two main modes of operation:

- The T_{avg} mode uses the difference between measured signal selected median loop T_{avg} and a reference temperature derived from turbine first-stage pressure, to generate a steam dump demand signal. This mode is largely used for at-power transients requiring steam dump, such as load rejections and turbine trips (where the load rejection T_{avg} mode is used). The load rejection controller is discussed in Subsection 7.7.1.9.1.

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- The pressure mode uses the difference between measured steam header pressure and a pressure setpoint to generate a steam dump demand signal. This mode is used for low-power conditions (up through turbine synchronization), plant cooldown, and reactor trip. It is described in Subsection 7.7.1.9.3.

The Class 1E process variable input signals to the steam dump control system are fed from protection channels via isolation devices and the signal selector function. Each input (T_{avg}, turbine load, and steam header pressure) is obtained from multiple transmitters of the same parameter. The signal selector rejects any signal which is bad in comparison with the remaining transmitter outputs and allows only valid measurements to be used by the control system. This makes the steam dump system tolerant of single transmitter failures or input signal failures and eliminates interaction between the control and the protection system.

To prevent actuation of steam dump on small load perturbations, an independent load rejection sensing circuit is provided. This circuit senses the rate of decrease in the turbine load as detected by the turbine first stage pressure. It unblocks the dump valves when the rate of a load rejection exceeds a preset value corresponding to a 10-percent step load decrease or a sustained ramp load decrease of greater than 5 percent per minute.

7.7.1.9.1 Load Rejection Steam Dump Controller

This controller prevents a large increase in reactor coolant temperature following a large, sudden load decrease. The error signal is a difference between the lead-lag compensated selected T_{avg} and the selected reference T_{avg} (designated T_{ref}), based on turbine first stage pressure.

The T_{avg} input signals are the same as those used in the reactor power control system, although a signal selector algorithm in a separate controller is employed. The lead-lag compensation for the T_{avg} signal compensates for lags in the plant thermal response and in valve positioning. The lead-lag compensation in the T_{ref} signal is used to compensate for hangup effects noted in the turbine first stage pressure measurement on turbine trips and grid disconnects. It allows for a decrease in gain in the steam dump controller, thereby increasing stability. Following a sudden load decrease, T_{ref} is immediately decreased and T_{avg} tends to increase. This generates an immediate demand signal for steam dump. Following the initial steam dump opening, the reactor power control system in conjunction with the rod control system commands the control rods to insert in a controlled manner to reduce the reactor power to match turbine load. On a load rejection resulting in a turbine runback, the steam dump terminates when the reactor power matches the turbine load and the temperature error is within the maneuvering capability of the control rods. On a grid disconnect, the steam dump modulates closed in response to the control rods reducing nuclear power to match the steam demand of plant auxiliaries. At this point, rod insertion stops and the plant stabilizes in preparation for a turbine/generator restart and/or grid synchronization with the steam dumps partially open. On a turbine trip the steam dump and reactor power control system reduce the plant to hot zero power conditions.

7.7.1.9.2 Not Used

7.7.1.9.3 Steam Header Pressure Controller

Decay heat removal between hot standby and residual heat removal system cut-in conditions is maintained by the steam header pressure controller. This controller uses the difference between steam header pressure and a pressure setpoint to control the steam flow to the condensers. Reset action is used to eliminate steady-state error. This controller uses the same steam dump valves as the load rejection controller described in Subsection 7.7.1.9.1. Following a reactor trip, the steam dump control system automatically switches to steam header pressure control mode with a fixed pressure setpoint. Otherwise, the steam header pressure control mode is manually selected by the

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operator. The pressure setpoint is manually adjusted by the operator based on the desired reactor coolant system temperature. In addition, the controller has a feature that allows automatically controlled plant cooldowns at a chosen rate (within limits). The operator can enter the desired cooldown rate and the desired target reactor coolant system temperature. The control system then dumps the required steam to achieve the setpoint cooldown rate and stops at the target setpoint.

7.7.1.10 Rapid Power Reduction System

The rapid power reduction system rapidly reduces the nuclear power to a level capable of being handled by the steam dump system for a large load rejection (greater than 50-percent power reduction at a rapid rate). Upon the detection of a large and rapid turbine power reduction (via a rate/lag circuit, similar to that used for steam dump control), the circuit provides a signal demanding the release of a selected number of control rods when power is above a predetermined level.

The large load rejection also actuates the steam dump system and the reactor power control system. Following the initiation of the load rejection, the power control rods insert in a controlled manner due to the mismatch between the programmed reference average coolant temperature (based on turbine first stage pressure) and the compensated average coolant temperature measured in the reactor coolant loops. In a similar manner, the load rejection steam dump controller controls the steam dump valves to prevent a large increase in reactor coolant temperature. Following the release of the selected control rods, the power control system continues to insert the remaining control rods to reduce power (by temperature control channel trying to match T_{avg} to T_{ref}). Following the initial opening, the steam dump valves modulate closed based upon the (T_{avg} - T_{ref}) signal.

Following rapid power reduction actuation, controlled rod insertion, steam dump modulation, and/or additional turbine runbacks continue as necessary until the plant is stabilized at a reduced power level. At this time, the operators can then recover the released control rods to restore normal overlap. The system can then be returned to automatic, and a normal power escalation will be performed to the desired power level. The rapid power reduction system is also used to rapidly reduce the nuclear power in conjunction with a turbine runback upon loss of one of the three operating main feedwater pumps as indicated by motor breaker positions when power is above a predetermined level. Power reduction is achieved in the same manner as described above.

7.7.1.10.1 Rod Block Interlock

To avoid the potential for a withdrawal of the normally functioning power control rods following the rod release by the rapid power reduction system, a rod withdrawal block (P-17) is actuated. The P-17 interlock is actuated by the rapid reduction in nuclear power caused by the insertion of power control rods (M banks) into the core after the initiation of the rapid power reduction system as discussed in Subsection 7.7.1.10. The rod withdrawal block does not adversely impact the performance of the rapid power reduction system. The demand of the power control subsystem is a continuous rod insertion. Rod withdrawal during the power reduction phase is not required.

7.7.1.10.2 Rapid Power Reduction Rod Selection

The number of rods needed to obtain the required power reduction is dependent on the initiating transient and the transient initial conditions, including the core burnup during the fuel cycle, the initial power level, and the initial position of the M-banks. Therefore, bank selection logic is provided to automatically determine which rods will be released by the rapid power reduction system based on these factors. Alternatively, the operator can override the automatic selection and manually select the rods for release.

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7.7.1.11 Diverse Actuation System

The diverse actuation system is a nonsafety-related system that provides a diverse backup to the protection and safety monitoring system. This backup is included to support the aggressive AP1000 risk goals by reducing the probability of a severe accident which potentially results from the unlikely coincidence of postulated transients and postulated common mode failure in the protection and control systems.

The protection and safety monitoring system is designed to prevent common mode failures. However, in the low probability case where a common mode failure does occur, the diverse actuation system provides diverse protection. The specific functions performed by the diverse actuation system are selected based on the PRA evaluation. The diverse actuation system functional requirements are based on an assessment of the protection system instrumentation common mode failure probabilities combined with the event probability. The plant-specific PRA model reflects DAS functionality testing at the frequency specified in the TRM.

The functional logic for the diverse actuation system is shown in Figure 7.2-1, sheets 20 and 21.

The DAS is developed using a planned design process, which provides for specific design documentation during the following life cycle stages:

- Design Requirements Phase
- System Definition Phase

These life cycle stages are completed by developing a number of specific design documents. The following documents are developed to address the Design Requirements and System Definition Phases:

- WCAP-17184-P, “AP1000™ Diverse Actuation System Planning and Functional Design Summary Technical Report,” including Appendix A, “DAS Setpoint Methodology Description,” and Appendix B, “PRA Performance Requirements Associated with DAS Manual Actuation.”

  WCAP-17184-P, “AP1000™ Diverse Actuation System Planning and Functional Design Summary Technical Report,” (Reference 1) identifies the DAS architecture and associated licensing basis at the functional design level. The overall DAS detailed design is not identified in the report. Select design details are identified only for the purpose of architectural completeness or licensing compliance. The content of WCAP-17184 Sections 1 through 11, and Appendices A and B are incorporated by reference. Appendix A of the Technical Report describes the DAS setpoint methodology and provides a representative basis for DAS nominal trip setpoints. Appendix B addresses operator actions taken through DAS that are modeled in the “AP1000 Probabilistic Risk Assessment” (PRA). These manual actions are not required to mitigate design basis accidents but instead are modeled in the PRA to provide insights into sequences that involve multiple failures. This appendix lists the operator actions used in the PRA for manual DAS actions from the control room DAS actuation panel.

- WCAP-15775, “AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report”

  Diversity is a principle in instrumentation of sensing different variables, using different technology, using different logic or algorithms, or using different actuation means to provide different ways of responding to postulated plant conditions. NUREG/CR-6303 segregates the types of diversity into six different areas: human, design, software, functional, signal, and equipment. WCAP-15775, “AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report,” (Reference 2) describes the type of diversity that exists among the four echelons of defense for

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AP1000 and identifies dependencies among the echelons. The content of WCAP-15775 Sections 1 through 5 is incorporated by reference.

A number of additional design process documents are submitted to the U.S. NRC for review and audit. They include the following:

1. “AP1000 Diverse Actuation System Logic Drawings”

2. “AP1000 Diverse Actuation System Sub-System Requirements”

3. “DAS Requirements Traceability Report”

Automatic Actuation Function

The automatic actuation signals provided by the diverse actuation system are generated in a functionally diverse manner from the protection system actuation signals. The common-mode failure of sensors of a similar design is also considered in the selection of these functions.

The automatic actuation function is accomplished by redundant logic subsystems. Input signals are received from the sensors by an input signal conditioning block, which consists of one or more electronic modules. This block converts the signals to standardized levels, provides a barrier against electromagnetic and radio frequency interference, and presents the resulting signal to the input signal conversion block. The conversion block continuously performs analog to digital signal conversions and stores the value for use by the signal processing block.

The signal processing block polls the various input signals, evaluates the input signals against stored setpoints, executes the logic when thresholds are exceeded, and issues actuation commands.

The resulting output signals are passed to the output signal conversion block, whose function is to convert logic states to parallel, low-level dc signals. These signals are passed to the output signal conditioning block. This block provides high-level signals capable of switching the traditional power plant loads, such as breakers and motor controls. It also provides a barrier against electromagnetic and radio frequency interference.

The DAS automatic actuation signals are generated in a functionally diverse manner from the PMS signals. Diversity between DAS and PMS is achieved by the use of different architectures, different hardware implementations, and different software, if any.

Software diversity between the DAS and PMS will be achieved through the use of different algorithms, logic, program architecture, executable operating system, and executable software/logic.

The AP1000 plant Component Interface Module (CIM) provides the priority logic between PMS and plant control for component control. Section 2.9 of WCAP-17179, “AP1000 Component Interface Module Technical Report,” (Reference 3) identifies how diversity is maintained between the DAS and the CIM.

The diverse automatic actuations are:

- Trip rods via the motor generator set, trip turbine, initiate the passive residual heat removal, actuate core makeup tanks, close the in-containment refueling water storage tank gutter isolation valves, and trip the reactor coolant pumps on low wide-range steam generator water level

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- Trip rods via the motor generator set, trip turbine, open the passive heat removal discharge isolation valves, and close the in-containment refueling water storage tank gutter isolation valves on high hot leg temperature
- Trip rods via the motor generator set, trip turbine, actuate the core makeup tanks, and trip the reactor coolant pumps on low pressurizer water level
- Isolate selected containment penetrations and start passive containment cooling water flow on high containment temperature

The selection of setpoints and time responses determine that the automatic functions do not normally actuate unless the protection and safety monitoring system has failed to actuate to control plant conditions. Capability is provided for testing and calibrating the channels of the diverse actuation system.

Manual Actuation Function

[*The manual actuation function of the diverse actuation system is implemented by hard-wiring the controls located in the main control room directly to the final loads in a way that completely bypasses the normal path through the protection and safety monitoring system cabinets, and the diverse actuation system automatic logic.*]*

The diverse manual functions are:

- Reactor and turbine trip
- Passive containment cooling actuation
- Core makeup tank actuation and reactor coolant pump trip
- Open stage 1 automatic depressurization system valves
- Open stage 2 automatic depressurization system valves
- Open stage 3 automatic depressurization system valves
- Open stage 4 automatic depressurization system valves
- Open the passive residual heat removal discharge isolation valves and close the in-containment refueling water storage tank gutter isolation valves
- Selected containment penetration isolation
- Containment hydrogen igniter actuation
- Initiate in-containment refueling water storage tank injection
- Initiate containment recirculation
- Initiate in-containment refueling water storage tank drain to containment

In addition to the above functions, a redundant method of actuating the following components is provided at DAS Processor Cabinet 1, Processor Cabinet 2, and the Squib Valve Control Cabinet:

- Open stage 4 automatic depressurization system valves

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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- Initiate in-containment refueling water storage tank injection
- Initiate containment recirculation
- Initiate in-containment refueling water storage tank drain to containment

The specific DAS controls were selected based on PRA risk importance (Reference 4). Although the DAS manual controls are non-Class 1E, they have been shown to be PRA risk-important (Reference 4).

Actuation Logic Function

There are two actuation logic modes, automatic and manual. The automatic actuation logic mode functions to logically combine the automatic signals from the two redundant automatic subsystems in a two-out-of-two basis. The combined signal operates a power switch with an output drive capability that is compatible, in voltage and current capacity, with the requirements of the final actuation devices. The two-out-of-two logic is implemented by connecting the outputs in series. The manual actuation mode operates in parallel to independently actuate the final devices.

Actuation signals are output to the loads in the form of normally de-energized, energize-to-actuate signals. The normally de-energized output state, along with the dual, two out of two redundancy reduces the probability of inadvertent actuation.

After an actuation signal is output to a load, subsequent return to operation requires deliberate operator action.

Indication

To support the diverse manual actuations, sensor outputs are displayed in the main control room in a manner that is diverse from the protection system display functions. The instrument sensor output displayed in the main control room is repeated at DAS Processor Cabinet 1. The indications that are provided from at least two sensors per function are:

- Steam generator water level – for reactor trip and passive residual heat removal actuations, and for overfill prevention by manual actuation of the automatic depressurization system valves
- Hot leg temperature – for passive residual heat removal actuation and reactor trip
- Core exit temperature – for automatic depressurization system actuation and subsequent initiation of in-containment refueling water storage tank injection and also containment hydrogen igniter actuation
- Pressurizer level – for core makeup tank actuation and reactor coolant pump trip
- Containment temperature – for containment isolation and passive containment cooling system actuation
- Rod control motor generator voltage – for confirmation of reactor trip

Isolation

The diverse actuation system uses sensors that are separate from those being used by the protection and safety monitoring system and the plant control system. This prohibits failures from propagating to the other plant systems through the use of shared sensors.

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There is signal isolation between the two subsystems within the diverse actuation system, one for each input and output path. These isolators are characterized by a high common mode voltage withstand capability to provide the necessary isolation against faults. The configuration is set up such that the isolation devices are capable of protecting against fault propagation between the diverse actuation system subsystems.

Actuation interfaces are shared between the diverse actuation system and the protection and safety monitoring system. The diverse actuation system actuation devices are isolated from the protection and safety monitoring system actuation devices, so as to avoid adverse interactions between the two systems. The actuation devices of each system are capable of independent operation that is not affected by the operation of the other. The diverse actuation system is designed to actuate components only in a manner that initiates the safety function. This type of interface also prevents the failure of an actuation device in one system from propagating a failure into the other system.

The diverse actuation system and the protection and safety monitoring system use independent and separate uninterruptible power supplies.

Operability, Availability, and Testing

The diverse actuation system is designed to provide protection under all plant operating conditions in which the reactor vessel head is in place and non-Class 1E UPS power is available. The automatic actuation processors, in each of the two redundant automatic subsystems of the diverse actuation system, are provided with the capability for channel calibration and testing while the plant is operating. To prevent inadvertent DAS actuations during online calibration, testing activities or maintenance, the normal activation function is bypassed. Testing of the diverse actuation system is performed on a periodic basis.

Equipment Qualification and Quality Standards

The diverse actuation system is located in a controlled environment, but is capable of functioning during and after normal and abnormal events and conditions that include:

- Wide temperature range of 40° to 120°F
- Noncondensing relative humidity up to 95 percent
- Radio frequency and electromagnetic interference

The diverse actuation system processor cabinets are located in the seismic Category I Auxiliary Building. The diverse actuation system equipment, including actuated devices, is designed and tested in accordance with industry standards. The adequacy of the hardware and software (if any) is demonstrated through a diverse form of the verification and validation program discussed in Subsection 7.1.2.14. This program provides for the use of commercial off-the-shelf hardware and software. As the diverse actuation system performs many of the protection functions associated within the ATWS systems used in existing plants, the diverse actuation system is designed to meet the quality guidelines established by Generic Letter 85-06, "Quality Assurance Guidelines for ATWS Equipment that is not Safety-Related."

7.7.1.12 Signal Selector Algorithm

The plant control system for the AP1000 derives some of its control inputs from signals that are also used in the protection and safety monitoring system. The advantages of this design are:

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- The nonsafety-related plant systems are controlled from the same measurements which provide protection. This permits the control system to function in a manner which maintains margin between operating conditions and safety limits, and reduces the likelihood of spurious trips.

- Reducing the number of redundant measurements for any single process variable reduces the overall plant complexity at critical pressure boundary penetrations. This leads to a reduction in separation requirements within the containment, as well as to a decrease in plant cost and maintenance requirements.

To obtain these advantages, measures are taken to provide the independence of the protection and control systems. The criteria for these measures are contained in IEEE 603-1991, Section 5.6.3. Isolation devices are provided to guard the protection system against possible electrical faults in the control system.

To avoid a single component failure or spurious signal causing an inadvertent plant trip while a channel is in test or maintenance, the protection and safety monitoring system uses the bypass logic discussed in Subsection 7.1.2.9. This necessitates a different mechanism for achieving the functional independence of control and protection.

Functional independence of control and protection is obtained by signal selector algorithms. The purpose of the signal selector algorithm is to prevent a failed signal, caused by the failure of a protection channel, from initiating a control action that could lead to a plant condition requiring that protective action. The signal selector function provides this capability by comparing the redundant signals and automatically eliminating an aberrant signal from use in the control system. This capability exists for bypassed sensors or for sensors whose signals have diverged from the expected error tolerance.

The operation of the signal selector algorithm is described in Subsection 7.1.3.2.

7.7.1.13 PRHR HX Modulation Control System

Should the main feedwater (MFW) and SFW be unavailable, the passive residual heat removal (PRHR) heat exchanger (HX) is used for decay heat removal. Nonsafety-related PLS PRHR HX modulation control system opens one of the two PRHR outlet valves to throttle PRHR flow. The PRHR outlet valve used for the throttling function can be manually selected by the operator. This non-safety defense in depth function provides protection against an unnecessary safety-related PRHR HX actuation resulting in full PRHR flow. If Low-2 narrow range SG level and Low-2 SFW flow are present in at least one SG, and at least one RCP is running in loop 1 (indicated by speed greater than the specified setpoint), then the automatic PRHR HX modulation control system actuation signal is latched in until manually reset by the operator. PRHR HX modulation control system then throttles PRHR flow with closed-loop feedback to maintain T_{cold} below the High T_{cold} setpoint that is used to fully actuate PRHR HX. The PRHR flow demand is also inversely proportional to SFW flow to automatically transition off the PRHR HX throttling if SFW becomes available. If PMS PRHR HX actuation logic is met both PRHR discharge isolation valves actuate to the full open position regardless of PRHR HX modulation control system demand.

7.7.2 Analysis

The control system is capable of maneuvering the plant through certain reference transients. This maneuvering is done without the need for manual intervention and without violating plant protection or component limits. The plant control systems provide high reliability during these anticipated operational occurrences and meet the following objectives:

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- The capability to accept 10-percent step load decreases from an initial power level between 100-percent and 25-percent of full power, and step load increase of 10-percent from an initial power level between 15-percent and 90-percent of full power without reactor trip or steam dump actuation.

- The capability to accept ramp load changes at 5-percent power per minute while operating in the range of 15-percent to 100-percent of full power without reactor trip or steam dump system actuation, subject to core power distribution limits.

- The capability to accept the design full-load rejection without reactor trip.

- The capability to accept a turbine trip from full-power operation without reactor trip. This capability is provided with the normally available systems (such as steam dump and feedwater control).

- The capability to follow the design basis network load follow pattern for 90-percent of the fuel cycle. The design basis load follow pattern is defined as the daily (24-hour period) cycle consisting of 10 to 18 hours of operation at 100-percent power, followed by a 2-hour linear ramp to 50-percent power, followed by 2 to 10 hours of operation at 50-percent power and then a 2-hour linear ramp back to 100-percent power.

- The capability to satisfy a 20-percent power increase or decrease within 10 minutes.

- The capability of handling grid frequency changes equivalent to 10-percent peak-to-peak power changes at a two percent per minute rate. This capability is provided over a 15- to 100-percent power range throughout the plant operating life. A total of 35 peak-to-peak swings per day are allowed.

The control system permits maneuvering the plant through the transients without actuation of the following:

- Steam generator safety valves

- Steam generator power operated relief valves

- Pressurizer safety valves

In addition, these valves are not actuated during a normal plant trip.

7.7.3 Combined License Information

This section contained no requirement for information.

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7.7.4 References

1. WCAP-17184-P (Proprietary) and WCAP-17184-NP (Non-Proprietary), “AP1000™ Diverse Actuation System Planning and Functional Design Summary Technical Report,” Revision 13, May 2019.[1]

2. WCAP-15775, “AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report,” Revision 8, May 2017.[1]

3. [*WCAP-17179-P (Proprietary) and WCAP-17179-NP (Non-Proprietary), “AP1000 Component Interface Module Technical Report,” Revision 6, April 2016.*]*[1]

4. WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," Revision 2. August 2003.

1. The text incorporates by reference portions or all of this document.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 7.7-1
Rod Control System Interlocks - Power Control Subsystem

Designation	Derivation	Function
C-1	Intermediate range neutron flux (output of signal selector) above setpoint (when 2/4 intermediate range neutron flux is not blocked)	Blocks automatic and manual control rod withdrawal
C-2	Nuclear power (output of signal selector) above setpoint	Blocks automatic and manual control rod withdrawal
C-3	Margin to overtemperature ΔT (output of signal selector) below setpoint	Blocks automatic and manual control rod withdrawal
		Actuates turbine runback via load reference
C-4	Margin to overpower ΔT (output of signal selector) below setpoint	Blocks automatic and manual control rod withdrawal
		Actuates turbine runback via load reference
C-5	Turbine first stage pressure (output of signal selector) below setpoint (blocked if in low-power rod control mode)	Blocks automatic control rod withdrawal
C-11	1/1M bank control rod position above setpoint	Blocks automatic rod withdrawal
C-16	Reactor coolant system T_{avg} or (T_{avg} minus T_{ref}) signal (output of signal selector) below setpoint	Stops automatic turbine loading until condition clears
P-17	Negative flux rate (calculated from signal selected nuclear power) below setpoint	Blocks automatic rod withdrawal

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Table 7.7-2
Rod Control System Interlocks – Axial Offset Control Subsystem

Designation	Derivation	Function
C-1	Intermediate range neutron flux (output of signal selector) above setpoint (when 2/4 intermediate range neutron flux is not blocked)	Blocks automatic and manual axial offset control rod withdrawal
C-2	Nuclear power (output of signal selector) above setpoint	Blocks automatic and manual axial offset control rod withdrawal
C-5	Turbine first stage pressure (output of signal selector) below setpoint	Blocks automatic axial offset control rod withdrawal and insertion
C-15	1/1 bank AO control rod position below setpoint	Blocks automatic axial offset control rod insertion
C-17	1/6M bank control rod positions below setpoint	Blocks automatic axial offset control rod withdrawal
C-18	1/1M bank control rod position above setpoint	Blocks automatic axial offset control rod insertion
---	Power control rods moving in	Blocks automatic axial offset control rod withdrawal
---	Power control rods moving out	Blocks automatic axial offset control rod insertion
P-17	Negative flux rate (calculated from signal selected nuclear power) below setpoint	Blocks automatic axial offset control rod withdrawal

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Table 7.7-3 (Sheet 1 of 3)
Cross Reference Table for Defense-in-Depth Functions
Supported by the Plant Control System

Supported System	Defense-in-Depth Function	DCD Section	DCD Figure
Component Cooling Water (CCS)	Provides cooling for normal residual heat removal system heat exchangers and pumps when the reactor coolant system pressure and temperature are below 450 psig and 350°F.	9.2.2.1.2.2 9.2.2.4.3	9.2.2-2
Component Cooling Water (CCS)	Provides cooling for the miniflow heat exchangers of the chemical and volume control system makeup pumps.	9.3.6.3.1	9.2.2-2
Component Cooling Water (CCS)	Provides cooling for the spent fuel pool heat exchangers for heat removal from the spent fuel pool.	9.2.2.1.2.3	9.2.2-2
Chemical and Volume Control (CVS)	Supply makeup and boration to the reactor coolant system.	9.3.6.7	9.3.6-1
Chemical and Volume Control (CVS)	Supply coolant to the pressurizer auxiliary spray line.	9.3.6.4.5	9.3.6-1
Standby Diesel Fuel Oil (DOS)	Supply fuel to the onsite standby power diesel generators.	9.5.4	9.5.4-1
Main and Startup Feedwater (FWS)	Provide startup feedwater for heat removal from the reactor coolant system (startup feedwater).	10.4.9.1.2	10.4.7-1 10.3.2-1
Normal Residual Heat Removal (RNS)	Remove heat from the reactor coolant system during shutdown operation at reduced pressure and temperature.	5.4.7.1.2.1	5.4-7
Normal Residual Heat Removal (RNS)	Provide low temperature overpressure protection for the reactor coolant system.	5.4.7.1.2.5	5.4-7
Normal Residual Heat Removal (RNS)	Provide low-pressure makeup to the reactor coolant system and remove heat from the reactor coolant system following actuation of the automatic depressurization system.	5.4.7.1.2.4 5.4.7.4.4	5.4-7

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Table 7.7-3 (Sheet 2 of 3)
Cross Reference Table for Defense-in-Depth Functions Supported by the Plant Control System

Supported System	Defense-in-Depth Function	DCD Section	DCD Figure
Spent Fuel Pool Cooling (SFS)	Provide for heat removal from the spent fuel stored in the spent fuel pool by pumping the water from the pool through a heat exchanger, and then returning the water to the pool.	9.1.3.2	9.1-8
Steam Generator (SGS)	Provide decay heat removal capability during shutdown operations by delivery of startup feedwater flow to the steam generator and venting of steam from the steam generators to the atmosphere via the power-operated relief valves.	10.4.9 10.3	10.4.7-1 10.3.2-1
Service Water (SWS)	Provide the capability for removing heat from the component cooling water system.	9.2.1.1.2	9.2.1-1
Service Water (SWS)	Provide the capability for removing heat from the spent fuel pool via the spent fuel cooling and component cooling water systems.	9.2.2 and Table 9.2.2-2	9.2.2-1 9.2.2-2
Service Water (SWS)	Provide the capability for decay heat removal at shutdown conditions through the normal residual heat removal and component cooling systems.	9.2.2 and Table 9.2.2-2	9.2.2-1 9.2.2-2
Nuclear Island Nonradioactive Ventilation (VBS)	Provide ventilation and cooling to the main control room envelope, Class 1E instrumentation and control rooms, Class 1E dc equipment rooms, and Class 1E battery rooms.	9.4.1	9.4.1-1 all sheets
Containment Hydrogen Control (VLS)	Provide hydrogen igniters to control hydrogen concentration in excess of the recombiner capability.	6.2.4	N/A
Central Chilled Water (VWS)	Provide chilled water to support the nuclear island nonradioactive ventilation system cooling of the main control room envelope, Class 1E instrumentation and control rooms, Class 1E dc equipment rooms, and the Class 1E battery rooms.	9.2.7	9.2.7-1 sheet 2
Central Chilled Water (VWS)	Provide chilled water to support the cooling functions of the compartment unit coolers for the normal residual heat removal system pump.	9.2.7	9.2.7-1 sheet 2

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Table 7.7-3 (Sheet 3 of 3)
Cross Reference Table for Defense-in-Depth Functions Supported by the Plant Control System

Supported System	Defense-in-Depth Function	DCD Section	DCD Figure
Central Chilled Water (VWS)	Provide chilled water to support the cooling functions of the compartment unit coolers for the chemical and volume control system makeup pump.	9.2.7	9.2.7-1 sheet 2
Annex/Auxiliary Building Nonradioactive Heating and Ventilation (VXS)	Provide ventilation of the electrical switchgear rooms that contain the diesel bus switchgear. Provide ventilation of the equipment room that contains the switchgear room air-handling units.	9.4.2	9.4.2-1 sheets 3, 4, 5, and 6
Diesel Generator Building Heating and Ventilation (VZS)	Provide ventilation and cooling of the diesel generator building, and ventilation and heating of the diesel oil transfer module enclosure to support operation of the onsite standby power system.	9.4.10	9.4.10-1
Onsite Standby Power (ZOS)	Supply ac power to the Class 1E dc and UPS system.	8.3 and Table 8.3.1-2	8.3.1-2
Onsite Standby Power (ZOS)	Supply ac power to selected electrical components of the plant defense-in-depth, nonsafety-related systems.	8.3 and Table 8.3.1-2	8.3.1-2
Passive Core Cooling System (PXS)	Remove decay heat in the event of a loss of all feedwater	6.3.1.1.1	6.3-1 Sheet 2

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APPENDIX 7A NOT USED

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APPENDIX 7B PROTECTION AND SAFETY MONITORING SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

The protection and safety monitoring system failure modes and effect analysis is provided in Sections 7B.1 through 7B.13.

Each failure mode identified in the failure modes and effect analysis is assigned a failure classification to facilitate identification of significant failures with regard to their criticality to the ability of the system to accomplish its required safety functions and to the effectiveness of diagnostics and/or surveillance tests to reveal the failures. The first part of this classification will be a number indicating criticality, as defined in Table 7B-1. The second part of the classification identifies the detectability of the failure, according to the definitions provided in Table 7B-2.

Based on this analysis, it is concluded that the AP1000 protection system maintains safety functions during single point failures.

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7B.1 BPL PROCESS STATION

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7B.2 LCL PROCESS STATION

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7B.3 REACTOR TRIP INTERFACE

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7B.4 ILP PROCESS STATION

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7B.5 NIS SUBSYSTEM

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7B.6 ICP PROCESS STATION

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7B.7 ITP PROCESS STATION

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7B.8 MTP PROCESS STATION

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7B.9 SAFETY DISPLAY SUBSYSTEM

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7B.10 QDPS PROCESS STATION

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7B.11 POWER DISTRIBUTION

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7B.12 ADS AND IRWST BLOCKING DEVICE

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7B.13 SQUIB VALVE CONTROL TERMINATION UNIT

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7B.14 ACRONYMS

ADS	Automatic Depressurization System
AF100	Advant Fieldbus 100
ASIC	Application Specific Integrated Circuit
BCC	Bistable Coincidence Cabinet
BIOB	Backplane Input Output Bus
BPL	Bistable Processor Logic
CB	Circuit Breaker
CI	Communications Interface
CIM	Component Interface Module
CS	Communication Section (of the PM646A Module)
DO	Digital Output
ESF	Engineered Safeguards Features
FPGA	Field Programmable Gate Array
HIM	HSL Interface Module
HSL	High Speed Link
HV	High Voltage
I/O	Input/Output
I&C	Instrumentation and Control
ICP	Interface Communication Processor
ILC	Interposing Logic Cabinet
ILP	Integrated Logic Processor
IR	Intermediate Range
IRWST	In-containment Refueling Water Storage Tank
ITP	Interface and Test Processor
LCL	Local Coincidence Logic
LCO	Limiting Condition for Operation
MCR	Main Control Room
MOV	Motor Operated Valve
MSV	Mean Square Voltage
MTP	Maintenance and Test Processor
NIS	Nuclear Instrumentation Subsystem
NISPA	Nuclear Instrumentation Signal Processing Assembly
PC	Personal Computer
PLS	Plant Control System
PM	Processor Module
PMS	Protection and Safety Monitoring System
PR	Power Range
PS	Processor Section (of the PM Module)
QDP	Qualified Data Processor
QDPS	Qualified Data Processing System
RCP	Reactor Coolant Pump
RSR	Remote Shutdown Room
SIR	Shunt Trip Interface Relay
SR	Source Range
SRNC	Safety Remote Node Controller
STA	Shunt Trip Attachment
TU	Termination Unit

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UIR	Undervoltage Interface Relay
UVTA	Undervoltage Trip Attachment
WDT	Watchdog Timer
XIR	Auxiliary Interposing Relay

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Chapter 8 Electric Power

8.1 Introduction

8.1.1 Utility Grid Description

The operating company grid system and interconnections to other grid systems and generating stations are as described.

8.1.2 Onsite Power System Description

The onsite power system is comprised of the main ac power system and the dc power system. The main ac power system is a non-Class 1E system. The dc power system consists of two independent systems: Class 1E dc system and non-Class 1E dc system. The ac and dc onsite power system configurations are shown on Figures 8.3.1-1 and 8.3.2-1, 8.3.2-2 and 8.3.2-3, respectively.

The normal ac power supply to the main ac power system is provided from the station main generator. When the main generator is not available, plant auxiliary power is provided from the switchyard by backfeeding through the main stepup and unit auxiliary transformers. This is the preferred power supply. When neither the normal or the preferred power supply (see Subsection 8.2.1) is available due to an electrical fault, fast bus transfer is initiated to transfer the loads to the reserve auxiliary transformers powered by maintenance sources of power. Both the normal and the preferred power supply will be unavailable due to an electrical fault at either the main stepup transformer, unit auxiliary transformer, AC portion of the excitation system, isophase bus, 6.9kv nonsegregated bus duct, or switchyard zone including the switchyard leads and grid breakers to the high side of the main stepup transformer. In addition, two non-Class 1E onsite standby diesel generators supply power to selected loads in the event of loss of the normal, preferred, and maintenance power sources. The reserve auxiliary transformers also serve as a source of maintenance power. The maintenance sources are as described.

The main generator is connected to the offsite power system by three single-phase stepup transformers. The normal power source for the plant auxiliary ac loads comes from the generator bus through two unit auxiliary transformers of identical rating and a third unit auxiliary transformer of lesser rating. In the event of a loss of the main generator, the power is maintained without interruption from the preferred power supply by an autotrip of the main generator breaker. Power then flows from the switchyard to the auxiliary loads through the main and unit auxiliary transformers.

A spare single-phase main stepup transformer is provided in the transformer area. The spare can be placed in service upon failure of one phase of the main stepup transformers.

The onsite standby power system, powered by the two onsite standby diesel generators, supplies power to selected loads in the event of loss of other ac power sources. Loads that are priority loads for investment protection due to their specific functions (permanent nonsafety loads) are selected for access to the onsite standby power supply. Availability of the standby power source is not required to accomplish any safety function.

The maintenance power supplies are provided at the medium voltage (6.9 kV) buses through normally open circuit breakers. Bus transfer to maintenance source either is automatic under fast bus transfer logic or may be initiated manually.

Four independent divisions of Class 1E 250 Vdc battery systems are provided for the Class 1E dc and UPS system. Divisions B and C have two battery banks; one battery bank is sized to supply power to safety-related loads for at least 24 hours and the other battery bank is sized to supply power to a second set of safety-related loads for at least 72 hours following a design basis event (including

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the loss of all ac power). Divisions A and D have one 24-hour battery bank. The loads are assigned to each battery bank, depending on their required function, during the 72 hour coping period so that no manual or automatic load shedding is required for the first 24 hours. Two ancillary diesel generators are provided for power for Class 1E post-accident monitoring, MCR lighting, MCR and I&C room ventilation, and power to refill the PCS water storage tank and spent fuel pool if no other sources of ac power are available.

An IDS spare battery bank (IDSS-DB-1A and IDSS-DB-1B) with a spare charger (IDSS-DC-1) is provided for the Class 1E battery system. In order to preserve independence of each Class 1E dc system division, plug-in locking type disconnects are permanently installed to prevent connection of more than one battery bank to the spare. In addition, kirk-key interlock switches are provided to prevent transfer operation of more than one switchboard at a time. The spare battery bank is located in a separate room and is capable of supplying power to the required loads on any battery being temporarily replaced with the spare.

The non-Class 1E 125 Vdc power system provides continuous, reliable power to the plant nonsafety-related dc loads. Operation of the non-Class 1E dc system is not required to accomplish any safety function.

Uninterruptible power supplies (UPS) to the four independent divisions of the Class 1E 120 Vac instrument buses are included in the Class 1E dc system. The normal power to the uninterruptible power supply comes from the respective Class 1E 250 Vdc bus. The backup power comes from the main ac power system through Class 1E 480-208Y/120V voltage regulating transformers. The same configuration applies for the uninterruptible power to the non-divisional, non-Class 1E 120 Vac instrument buses. The normal power to the non-Class 1E uninterruptible power supply comes from the non-Class 1E 125 Vdc bus and the backup power comes from the main ac power system through a voltage regulating transformer.

8.1.3 Safety-Related Loads

The safety-related loads requiring Class 1E power are listed in Tables 8.3.2-1, 8.3.2-2, 8.3.2-3 and 8.3.2-4. Safety-related loads are powered from the Class 1E 250 Vdc batteries and the associated Class 1E 120 Vac instrument buses.

8.1.4 Design Basis

8.1.4.1 Offsite Power System

Offsite power has no safety-related function due to the passive design of the AP1000. Therefore, redundant offsite power supplies are not required. The design provides a reliable offsite power system that minimizes challenges to the passive safety system.

8.1.4.2 Onsite Power System

8.1.4.2.1 Safety Design Basis

- The Class 1E dc and UPS power system meets the single failure criterion (GDC 17).
- The Class 1E dc and UPS system has sufficient capacity to achieve and maintain safe shutdown of the plant for 72 hours following a complete loss of all ac power sources without requiring load shedding for the first 24 hours.
- The Class 1E dc and UPS system is divided into four independent divisions. Any three-out-of-four divisions can shut down the plant safely and maintain it in a safe shutdown condition.

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- Separation criteria preserve the independence of redundant Class 1E circuits as described in Subsection 8.3.2.4 and no single credible event is capable of disabling redundant safety-related systems.

- Special identification criteria are applied for Class 1E equipment, Class 1E cabling, and raceways that route Class 1E cables as described in Subsection 8.3.2.3.

- The Class 1E systems and equipment are designed to permit periodic inspection and testing (GDC-18).

- The Class 1E dc and UPS power system permits connection of any one 250 Vdc switchboard at a time to the spare battery and the spare battery charger. The spare battery and charger have sufficient capacity to permit continuous plant operation at 100-percent power in case of a failure or unavailability of one Class 1E battery bank and the associated battery charger, as described in Subsection 8.3.2.1.1.

- Two ancillary diesel generators provide ac power for Class 1E post-accident monitoring, MCR lighting, MCR and I&C room ventilation, and power to refill the PCS water storage tank and spent fuel pool if no other sources of power are available. The equipment used to perform this function is not safety-related because it is not needed for a prolonged period following a loss of ac and it is easily replaced with transportable generators.

8.1.4.2.2 Power Generation Design Basis

- The main ac power system is a non-Class 1E system and nonsafety-related. The normal power supply to the main ac power system comes from the station main generator through two identically rated unit auxiliary transformers and an additional unit auxiliary transformer for the electric auxiliary boiler and as described site-specific loads.

- The onsite standby power system supplies ac power to the selected permanent nonsafety loads in the event of a main generator trip concurrent with the loss of preferred power source and maintenance power source when under fast bus transfer conditions. The onsite standby diesel generators are automatically connected to the associated 6.9 kV buses upon loss of bus voltage only after the generator rated voltage and frequency is established. Loads that are important for orderly plant shutdown are sequentially connected as shown in Subsection 8.3.1 during this event.

  The permanent nonsafety loads are not required for the plant safe shutdown; therefore, the onsite standby power system is a nonsafety-related system and non-Class 1E.

- For continued operation of the plant, a spare single-phase main transformer can be placed in service upon failure of one phase of the main stepup transformers.

8.1.4.3 Design Criteria, Regulatory Guides, and IEEE Standards

Refer to Table 8.1-1 for guidelines, and their applicability to Chapter 8.

The offsite and onsite ac power systems have no safety function and, therefore, their conformance to General Design Criteria, Regulatory Guides and IEEE Standards is not required, except as indicated in Table 8.1-1.

The Class 1E dc power system design is based on the following:

- General Design Criteria (GDC)

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See Section 3.1 for a discussion of conformance to the General Design Criterion.

- Nuclear Regulatory Commission (NRC) Regulatory Guides

  See Section 1.9 for the list and details of conformance to the regulatory guides.

- IEEE Standards.

  The Class 1E dc power system design is based on the following IEEE Standards that are generally acceptable to the NRC as stated in the referenced Regulatory Guides:

  – IEEE 308-1991, IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations. Refer to Regulatory Guide 1.32.

  – IEEE 317-1983, IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations. Refer to Regulatory Guide 1.63.

  – IEEE 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations. Refer to Regulatory Guide 1.89.

  – IEEE 338-1987, IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems. Refer to Regulatory Guide 1.118.

  – IEEE 344-1987, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations. Refer to Regulatory Guide 1.100.

  – IEEE 379-2000, IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems. Refer to Regulatory Guide 1.53.

  – IEEE 382-1996, IEEE Standard for Qualification of Actuators for Power Operated Valve Assemblies with Safety Related Functions for Nuclear Power Plants. Refer to Regulatory Guide 1.73.

  – IEEE 383-1974, IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations. Refer to Regulatory Guide 1.131.

  – IEEE 384-1981, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits. Refer to Regulatory Guide 1.75.

  – IEEE 450-1995, IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications. Refer to Regulatory Guide 1.32.

  – IEEE 484-1996, IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications. Refer to Regulatory Guide 1.128.

  – IEEE 741-1997, IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations. Refer to Regulatory Guide 1.63.

  – IEEE 1202-1991, IEEE Standard for Flame Testing of Cables for Use in Cable Tray in Industrial and Commercial Occupancies.

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8.1.5 Combined License Information

This section contained no requirement for information.

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Table 8.1-1 (Sheet 1 of 5)
Criteria and Guidelines for Electric Power Systems

Criteria	Applicability ([a] Section/Subsection)			Remarks
	8.2	8.3.1	8.3.2	
1. 10CFR50 Appendix A – General Design Criteria (GDC) (See Section 3.1 for a discussion of conformance to each of the GDC).				
a. GDC 2 Design Bases for Protection Against Natural Phenomena			A	
b. GDC 4 Environmental and Missile Design Basis			A	
c. GDC 5 Sharing of Structures, Systems, and Components				not applicable
d. GDC 17 Electric Power Systems			A	
e. GDC 18 Inspection and Testing of Electric Power Systems			A	
f. GDC 50 Containment Design Basis		A	A	applicable to penetration design

Note:
(a) "A" denotes applicable to AP1000, and "G" denotes guidelines as defined in NUREG-0800, Rev. 2, Table 8-1 (SRP). No letter denotes "Not Applicable."

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Table 8.1-1 (Sheet 2 of 5)
Criteria and Guidelines for Electric Power Systems

Criteria	Applicability ([a] Section/Subsection)			Remarks
	8.2	8.3.1	8.3.2	
2. Regulatory Guide (See Section 1.9 for list and discussion of conformance to the Regulatory Guides).				
a. RG 1.6 Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems			G	
b. RG 1.9 Selection, Design, and Qualification of Diesel Generator Units Used as Stand-by (Onsite) Electric Power Systems at Nuclear Power Plants				not applicable
c. RG 1.32 Criteria for Safety-Related Electric Power Systems for Nuclear Power Generating Stations			G	
d. RG 1.47 Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems			G	
e. RG 1.63 Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants		G	G	
f. RG 1.75 Physical Independence of Electric Systems			G	
g. RG 1.81 Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants				not applicable

Note:
(a) "A" denotes applicable to AP1000, and "G" denotes guidelines as defined in NUREG-0800, Rev. 2, Table 8-1 (SRP). No letter denotes "Not Applicable."

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Table 8.1-1 (Sheet 3 of 5)
Criteria and Guidelines for Electric Power Systems

Criteria			Applicability ([a] Section/Subsection)			Remarks
			8.2	8.3.1	8.3.2	
h.	RG 1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves			G	
i.	RG 1.108	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants				not applicable
j.	RG 1.118	Periodic Testing of Electric Power and Protection Systems			G	
k.	RG 1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants			G	
l.	RG 1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants				site-specific
m.	RG 1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants			G	The insulating and jacketing material for electrical cables are selected to meet the fire and flame test requirements of IEEE Standard 1202, IEEE Standard 383, or the standards for the corresponding test types as listed in Tables K-1 and K-2 in NEI 04-02, Rev 3 excluding the option to use the alternate flame source, oil or burlap.

Note:
(a) "A" denotes applicable to AP1000, and "G" denotes guidelines as defined in NUREG-0800, Rev. 2 , Table 8-1 (SRP). No letter denotes "Not Applicable."

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Table 8.1-1 (Sheet 4 of 5)
Criteria and Guidelines for Electric Power Systems

Criteria	Applicability ([a] Section/Subsection)			Remarks
	8.2	8.3.1	8.3.2	
3. Branch Technical Position (BTP)				
a. BTP ICSB 4 (PSB) Requirements on Motor-Operated Valves in the ECCS Accumulator Lines			G	see 1.9.2
b. BTP ICSB 8 (PSB) Use of Diesel-Generator Sets for Peaking				not applicable
c. BTP ICSB 11 (PSB) Stability of Offsite Power Systems				site-specific
d. BTP ICSB 18 (PSB) Application of the Single Failure Criterion to Manually Controlled Electrically-Operated Valves			G	see 1.9.2
e. BTP ICSB 21 Guidance for Application of Regulatory Guide 1.47			G	see also 7.5
f. BTP PSB 1 Adequacy of Station Electric Distribution System Voltages				not applicable
g. BTP PSB 2 Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status				not applicable

Note:
(a) "A" denotes applicable to AP1000, and "G" denotes guidelines as defined in NUREG-0800, Rev. 2, Table 8-1 (SRP). No letter denotes "Not Applicable."

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Table 8.1-1 (Sheet 5 of 5)
Criteria and Guidelines for Electric Power Systems

Criteria	Applicability ([a] Section/Subsection)			Remarks
	8.2	8.3.1	8.3.2	
4. NUREG Reports				
a. NUREG CR0660 Enhancement of Onsite Diesel Generator Reliability				not applicable

Note:
(a) "A" denotes applicable to AP1000, and "G" denotes guidelines as defined in NUREG-0800, Rev. 2, Table 8-1 (SRP). No letter denotes "Not Applicable."

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8.2 Offsite Power System

8.2.1 System Description

A transmission system to supply offsite ac energy for startup and normal shutdown through a site-specific transmission switchyard is as described in subsection 8.2.5. This offsite ac power system is not required for plant safety.

[[A transformer area containing stepup transformers, unit auxiliary transformers, and the reserve auxiliary transformer is located next to the turbine building.]]

The normal ac power supply to the main ac power system is provided from the main generator. When the main generator is not available, plant auxiliary power is provided from the switchyard by backfeeding through the main stepup and unit auxiliary transformers. This is the preferred power supply. When neither the normal or the preferred power supply is available due to an electrical fault, the loads are transferred to onsite sources as identified in Subsection 8.1.2. In addition, two non-Class 1E onsite standby diesel generators supply power to selected plant loads in the event of loss of the normal, preferred, and maintenance power sources. The reserve auxiliary transformers also serve as a source of maintenance power.

The maintenance sources are site-specific. Maintenance power is provided at the medium voltage level (6.9 kV) through normally open circuit breakers. Bus transfer to the maintenance source is automatic under fast bus transfer logic or may be initiated manually. Connection of the preferred and maintenance power supplies to the utility grid or other power sources is site-specific.

The main generator is connected to the offsite power system via three single-phase main stepup transformers. The normal power source for the plant auxiliary ac loads is provided from the isophase generator bus through the two unit auxiliary transformers of identical ratings and a third unit auxiliary transformer of lesser rating. In the event of a loss of the main generator, the power is maintained without interruption from the preferred power supply by an auto-trip of the main generator breaker. Power then flows from the transformer area to the auxiliary loads through the main and unit auxiliary transformers.

The transmission line structures associated with the plant are designed to withstand standard loading conditions for the specific-site as provided in Reference 1.

Automatic load dispatch is not used at the plant and does not interface with safety-related action required of the reactor protection system.

8.2.1.1 Transmission Switchyard

The transmission switchyard is site-specific, and the responsibility for the transmission system is described in subsection 8.2.5.

8.2.1.2 Transformer Area

The transformer area contains the main stepup transformers, the unit auxiliary transformers, and the reserve auxiliary transformers. Protective relaying and metering required for this equipment is located in the turbine building. The necessary power sources (480 Vac, 120 Vac, and 125 Vdc) to the equipment are supplied from the turbine building. See Subsection 9.5.1 for a discussion of fire protection associated with plant transformers.

One feeder connects the transformer area with the switchyard to supply power to/from the main stepup transformers for the unit. An arrangement is shown in Figure 8.3.1-1.

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8.2.2 Grid Stability

The AP1000 is designed with passive safety-related systems for core cooling and containment integrity and, therefore, does not depend on the electric power grid for safe operation. This feature of the AP1000 significantly reduces the importance of the grid connection and the requirement for grid stability. The AP1000 safety analyses assume that the reactor coolant pumps can receive power from either the main generator or the grid for a minimum of 3 seconds following a turbine trip.

The AP1000 main generator is connected to the generator bus through the generator circuit breaker. The grid is connected to the generator bus through the main step-up transformers and the grid breakers. The reactor coolant pumps are connected to the generator bus through the reactor coolant pump breakers, the 6.9 kV switchgear, and the unit auxiliary transformers. During normal plant operation the main generator supplies power to the generator bus. Some of this power is used by the plant auxiliary systems (including the reactor coolant pumps); the rest of the power is supplied to the grid.

If, during power operation of the plant, a turbine trip occurs, the motive power (steam) to the turbine will be removed. The generator will attempt to keep the shaft rotating at synchronous speed (governed by the grid frequency) by acting like a synchronous motor. The reverse-power relay monitoring generator power will sense this condition and, after a time delay of at least 15 seconds, open the generator breaker. During this delay time the generator will be able to provide voltage support to the grid if needed. The reactor coolant pumps will receive power from the grid for at least 3 seconds following the turbine trip. A grid stability analysis to show that, with no electrical system failures, the grid will remain stable and the reactor coolant pump bus voltage will remain above the voltage required to maintain the flow assumed in the Chapter 15 analyses for a minimum of 3 seconds following a turbine trip is as addressed in Subsection 8.2.5. In the Chapter 15 analyses, if the initiating event is an electrical system failure (such as failure of the isophase bus), the analyses do not assume operation of the reactor coolant pumps following the turbine trip. The responsibility for setting the protective devices controlling the switchyard breakers with consideration given to preserving the plant grid connection following a turbine trip is discussed in Subsection 8.2.1.

If the turbine trip occurs when the grid is not connected (generator supplying plant house loads only), the main turbine-generator shaft will begin to slow down as the energy stored in the rotational inertia of the shaft is used to supply the house loads (including reactor coolant pumps). The system will coast down until the generator exciter can no longer maintain generator terminal voltage and the generator breaker is tripped on either generator under-voltage or exciter over-current. This coast down will last at least 3 seconds before the generator breaker trips.

The sequence of events following a loss-of-offsite-power event is the same as those described for grid-disconnected operation.

8.2.3 Conformance to Criteria

The offsite sources are not Class 1E. Commercial equipment is manufactured to the industrial standards listed in Subsection 8.2.6. The design meets General Design Criterion 1. Unit trips occur at the generator breaker and do not cause the loss of the preferred power source to the plant electrical systems. The AP1000 does not require ac power sources for mitigating design basis events; Chapter 15.0 describes the design bases assumptions utilized for analysis of these events.

The AP1000 plant design supports an exemption to the requirement of GDC 17 for two physically independent offsite circuits by providing safety-related passive systems for core cooling and containment integrity, and multiple nonsafety-related onsite and offsite electric power sources for other functions. See Section 6.3 for additional information on the systems for core cooling.

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A reliable dc power source supplied by batteries provides power for the safety-related valves and instrumentation during transient and accident conditions.

The Class 1E dc and UPS system is the only safety-related power source required to monitor and actuate the safety-related passive systems. Otherwise, the plant is designed to maintain core cooling and containment integrity, independent of nonsafety-related ac power sources indefinitely. The only electric power source necessary to accomplish these safety-related functions is the Class 1E dc and UPS power system which includes the associated safety-related 120V ac distribution switchgear.

Although the AP1000 is designed with reliable nonsafety-related offsite and onsite ac power that are normally expected to be available for important plant functions, nonsafety-related ac power is not relied upon to maintain the core cooling or containment integrity.

The nonsafety-related ac power system is designed such that plant auxiliaries can be powered from the grid under all modes of operation. During loss of normal ac power and offsite power, the ac power is supplied by the onsite standby diesel-generators. Preassigned loads and equipment are automatically loaded on the diesel-generators in a predetermined sequence. Additional loads can be manually added as required. The onsite standby power system is not required for safe shutdown of the plant.

Conformance with General Design Criterion 18 is provided by the test and inspection capability of the system.

8.2.4 Standards and Guides

In addition to the General Design Criteria, the industry guides and standards listed as Reference 2 through 4 are used as guides in the design and procurement of the offsite power system.

8.2.5 Combined License Information for Offsite Electrical Power

Combined License applicants referencing the AP1000 certified design will address the design of the ac power transmission system and its testing and inspection plan.

The Combined License applicant will address the technical interfaces for this nonsafety-related system listed in Table 1.8-1 and subsection 8.2.2. These technical interfaces include those for ac power requirements from offsite and the analysis of the offsite transmission system and the setting of protective devices.

8.2.6 References

1. ANSI C2-1997, “National Electrical Safety Code.”

2. ANSI C37.010-1999, “IEEE Application Guide for AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis.”

3. ANSI C37.90-1989, “IEEE Standard for Relays and Relay Systems Associated with Electric Power Apparatus.”

4. ANSI C57.12.00-2000, “IEEE Standard General Requirements for Liquid-Immersed Distribution, Power and Regulating Transformers.”

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8.3 Onsite Power Systems

8.3.1 AC Power Systems

8.3.1.1 Description

The onsite ac power system is a non-Class 1E system comprised of a normal, preferred, maintenance and standby power supplies. The normal, preferred, and maintenance power supplies are included in the main ac power system. The standby power is included in the onsite standby power system. The Class 1E and non-Class 1E 208/120 Vac instrumentation power supplies are described in Subsection 8.3.2 as a part of uninterruptible power supply in the dc power systems.

8.3.1.1.1 Onsite AC Power System

The main ac power system is a non-Class 1E system and does not perform any safety-related functions (other than tripping the reactor coolant pumps (RCPs) as noted below). It has nominal bus voltage ratings of 6.9 kV, 480 V, 277 V, 208 V, and 120 V.

Figure 8.3.1-1 shows the main generator, transformers, feeders, buses, and their connections. The ratings of major ac equipment are listed in Table 8.3.1-3.

During power generation mode, the turbine generator normally supplies electric power to the plant auxiliary loads through the unit auxiliary transformers. The plant is designed to sustain a load rejection from 100 percent power with the turbine generator continuing stable operation while supplying the plant house loads. The load rejection feature does not perform any safety function.

During plant startup, shutdown, and maintenance the generator breaker remains open. The main ac power is provided by the preferred power supply from the high-voltage switchyard (switchyard voltage is site-specific) through the plant main stepup transformers and three unit auxiliary transformers. Two of the unit auxiliary transformers each supply power to about 50 percent of the principal plant loads. A third auxiliary transformer is provided to power the electric boiler and other nonsafety-related loads as described below.

A maintenance source is provided to supply power through two reserve auxiliary transformers. The maintenance source and the associated reserve auxiliary transformers primary voltage are site specific. The reserve auxiliary transformers are sized so that it can be used in place of the unit auxiliary transformers.

Two unit auxiliary transformers (UAT 2A, 2B) are three winding transformers with two identically rated 6.9 kV secondary windings. The third unit auxiliary transformer (UAT 2C) is a two winding transformer sized to accommodate the electric boiler, the turbine turning gear, a portion of the turbine building space heating loads, and nonsafety-related site-specific loads. Secondaries of the auxiliary transformers are connected to the 6.9 kV switchgear buses by nonsegregated phase buses. The primary of the unit auxiliary transformer is connected to the main generator isolated phase bus duct tap. The 6.9 kV switchgear designation, location, connection, and connected loads are shown in Figure 8.3.1-1. The buses tagged with odd numbers (ES1, ES3, etc.) are connected to one unit auxiliary transformer and the buses tagged with even numbers (ES2, ES4, etc.) are connected to the other unit auxiliary transformer. ES7 is connected to the third unit auxiliary transformer. 6.9 kV buses ES1-ES6 are provided with an access to the maintenance source through normally open circuit breakers connecting the bus to the reserve auxiliary transformer. ES7 is not connected to the maintenance source. Bus transfer to the maintenance source is manual or automatic through a fast bus transfer scheme.

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The arrangement of the 6.9 kV buses permits feeding functionally redundant pumps or groups of loads from separate buses and enhances the plant operational flexibility. The 6.9 kV switchgear powers large motors and the load center transformers. There are two switchgear (ES1 and ES2) located in the annex building, and five (ES3, ES4, ES5, ES6, and ES7) in the turbine building.

The main stepup transformers have protective devices for sudden pressure, neutral overcurrent, and differential current. The unit auxiliary transformers have protective devices for sudden pressure, overcurrent, differential current, and neutral overcurrent. The excitation transformer has ground fault protection. The isophase bus duct has ground fault protection. If these devices sense a fault condition the following actions will be automatically taken:

- Trip high-side (grid) breaker
- Trip generator breaker
- Trip exciter field breaker
- Trip the 6.9 kV buses connected to the faulted transformer
- Initiate a fast bus transfer of ES1-ES6 6.9kV buses ES1-ES6.

The reserve auxiliary transformers have protective devices for sudden pressure, overcurrent, differential current and neutral overcurrent. The reserve auxiliary transformers protective devices trip the reserve supply breaker and any 6.9 kV buses connected to the reserve auxiliary transformers.

The onsite standby power system powered by the two onsite standby diesel generators supplies power to selected loads in the event of loss of normal ac power and offsite power supplies followed by a fast bus transfer to the reserve auxiliary transformers. Those loads that are priority loads for defense-in-depth functions based on their specific functions (permanent nonsafety loads) are assigned to buses ES1 and ES2. These plant permanent nonsafety loads are divided into two functionally redundant load groups (degree of redundancy for each load is described in the sections for the respective systems). Each load group is connected to either bus ES1 or ES2. Each bus is backed by a non-Class 1E onsite standby diesel generator. In the event of a loss of voltage on these buses, the diesel generators are automatically started and connected to the respective buses. In the event where a fast bus transfer initiates but fails to complete, the diesel generator will start on an undervoltage signal; however, if a successful residual voltage transfer occurs, the diesel generator will not be connected to the bus because the successful residual voltage transfer will provide power to the bus before the diesel connection time of 2 minutes. The source incoming breakers on switchgear ES1 and ES2 are interlocked to prevent inadvertent connection of the onsite standby diesel generator and preferred/maintenance ac power sources to the 6.9 kV buses at the same time. The diesel generator, however, is capable of being manually paralleled with the preferred or reserve power supply for periodic testing. Design provisions protect the diesel generators from excessive loading beyond the design maximum rating, should the preferred power be lost during periodic testing. The control scheme, while protecting the diesel generators from excessive loading, does not compromise the onsite power supply capabilities to support the defense-in-depth loads. See Subsection 8.3.1.1.2 for starting and load sequencing of standby diesel generators.

The reactor coolant pumps (RCPs) are powered from the four switchgear buses located in the turbine building, one RCP per bus. Variable-speed drives are provided for RCP startup and for RCP operation when the reactor trip breakers are open. During normal power operation (reactor trip breakers are closed), 60 Hz power is provided directly to the RCPs and the variable-speed drives are not connected.

Each RCP is powered through two Class 1E circuit breakers connected in series. These are the only Class 1E circuit breakers used in the main ac power system for the specific purpose of satisfying the safety-related tripping requirement of these pumps. The reactor coolant pumps connected to a common steam generator are powered from two different unit auxiliary transformers. The bus assignments for the reactor coolant pumps are shown in Figure 8.3.1-1.

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The 480 V load centers supply power to selected 460 V motor loads and to motor control centers. Bus tie breakers are provided between two 480 V load centers each serving predominantly redundant loads. This intertie allows restoration of power to selected loads in the event of a failure or maintenance of a single load center transformer. The bus tie breakers are interlocked with the corresponding bus source incoming breakers so that one of the two bus source incoming breakers must be opened before the associated tie breaker is closed. Load center 71, associated with ES-7, does not have an equivalent match.

The 480 V motor control centers supply power to 460 V motors not powered directly from load centers, while the 480/277 V, and 208/120 V distribution panels provide power for miscellaneous loads such as unit heaters, space heaters, and lighting system. The motor control centers also provide ac power to the Class 1E battery chargers for the Class 1E dc power system as described in Subsection 8.3.2.

Two ancillary ac diesel generators, located in the annex building, provide ac power for Class 1E post-accident monitoring, MCR lighting, MCR and I&C room ventilation, and pump power to refill the PCS water storage tank and the spent fuel pool, when all other sources of power are not available.

Each ancillary ac generator output is connected to a distribution panel. The distribution panel is located in the room housing the diesel generators. The distribution panel has incoming and outgoing feeder circuit breakers as shown on Figure 8.3.1-3. The outgoing feeder circuit breakers are connected to cables which are routed to the divisions B and C voltage regulating transformers and to the PCS pumps. Each distribution panel has the following outgoing connections:

- Connection for Class 1E voltage regulating transformer to power the post-accident monitoring loads, the lighting in the main control room, and ventilation in the main control room and divisions B and C I&C rooms.

- Connection for PCS recirculation pump to refill the PCS water storage tank and the spent fuel pool.

- Connection for local loads to support operation of the ancillary generator (lighting and fuel tank heating).

- Temporary connection for a test load device (e.g., load resistor).

- The connection from the ancillary diesel generators to the voltage regulating transformers includes a transfer switch to allow the regulating transformers to be powered by post-72 hour portable equipment.

  See Figure 8.3.1-3 for connections to post-72-hour loads.

8.3.1.1.1.1 Electric Circuit Protection

Protective relay schemes and direct acting trip devices on circuit breakers:

- Provide safety of personnel
- Minimize damage to equipment
- Minimize system disturbances
- Isolate faulted equipment and circuits from unfaulted equipment and circuits
- Maintain (selected) continuity of the power supply

Major types of protection systems employed for AP1000 include the following:

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Medium Voltage Switchgear

Differential Relaying

Each medium voltage switchgear bus is provided with a bus differential relay (device 87B) to protect against a bus fault. The actuation of this relay initiates tripping of the source incoming circuit breaker and all branch circuit load breakers. The differential protection scheme employs high-speed relays.

Motors rated 1500 hp and above are generally provided with a high dropout overcurrent relay (device 50D) for differential protection.

Overcurrent Relaying

To provide backup protection for the buses, the source incoming circuit breakers are equipped with an inverse time overcurrent protection on each phase and a residually connected inverse time ground overcurrent protection.

Each medium voltage motor feeder breaker is equipped with a motor protection relay which provides protection against various types of faults (phase and ground) and abnormal conditions such as locked rotor and phase unbalance. Motor overload condition is annunciated in the main control room.

Each medium voltage power feeder to a 480 V load center has a multifunction relay. The relay provides overcurrent protection on each phase for short circuit and overload, and an instantaneous overcurrent protection for ground fault.

Undervoltage Relaying

Medium voltage buses are provided with a set of three undervoltage relays (device 27B) which trip motor feeder circuit breakers connected to the bus upon loss of bus voltage using two-out-of-three logic to prevent spurious actuation. In addition, a protective device is provided on the line side of incoming supply breakers of buses ES1 and ES2 to initiate an alarm in the main control room if a sustained low or high voltage condition occurs on the utility supply system. The alarm is provided so that the operator can take appropriate corrective measures.

480-V Load Centers

Each motor-feeder breaker in load centers is equipped with a trip unit which has long time, instantaneous, and ground fault tripping features. Overload condition of motors is annunciated in the main control room.

The circuit breakers feeding the 480V motor control centers and other non-motor loads have long time, short time, and ground fault tripping features.

Each load center bus has an undervoltage relay which initiates an alarm in the main control room upon loss of bus voltage.

Load center transformers have transformer winding temperature relays (device 49T) which give an alarm on transformer overload.

480-V Motor Control Center

Motor control center feeders for low-voltage (460 V) motors have molded case circuit breakers (magnetic or motor circuit protectors) and motor starters. Motor starters are provided with thermal units (overload heaters) or current sensors. Other feeders have molded case circuit breakers with thermal and magnetic trip elements for overload and short circuit protection.

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Non-Class 1E ac motor-operated valves are protected by thermal overload devices. Thermal overload devices are selected and sized so as to provide the necessary protection while minimizing the probability of spurious interruptions of valve actuation.

8.3.1.1.2 Standby AC Power Supply

8.3.1.1.2.1 Onsite Standby Diesel Generators

Two onsite standby diesel generator units, each furnished with its own support subsystems, provide power to the selected plant nonsafety-related ac loads. Power supplies to each diesel generator subsystem components are provided from separate sources to maintain reliability and operability of the onsite standby power system. These onsite standby diesel generator units and their associated support systems are classified as AP1000 Class D, defense-in-depth systems.

The onsite standby diesel generator function to provide a backup source of electrical power to onsite equipment needed to support decay heat removal operation during reduced reactor coolant system inventory, midloop, operation is identified as an important nonsafety-related function. The standby diesel generators are included in the Investment Protection Short-Term Availability Controls described in Section 16.3 and the Design Reliability Assurance Program described in Section 17.4.

Each of the generators is directly coupled to the diesel engine. Each diesel generator unit is an independent self-contained system complete with necessary support subsystems that include:

- Diesel engine starting subsystem
- Combustion air intake and engine exhaust subsystem
- Engine cooling subsystem
- Engine lubricating oil subsystem
- Engine speed control subsystem
- Generator, exciter, generator protection, monitoring instruments, and controls subsystems

The diesel-generator starting air subsystem consists of an ac motor-driven, air-cooled compressor, a compressor inlet air filter, an air-cooled aftercooler, in-line air filters, a refrigerant dryer (with dew point at least 10°F less than the lowest normal diesel generator room temperature), and an air receiver with sufficient storage capacity for three diesel engine starts. The starting air subsystem is consistent with manufacturer's recommendations regarding the devices to crank the engine, duration of the cranking cycle, the number of engine revolutions per start attempt, volume and design pressure of the air receivers, and compressor size. The interconnecting stainless steel piping from the compressor to the diesel engine dual air starter system includes air filters, moisture drainers, and pressure regulators to provide clean dry compressed air at normal diesel generator room temperature for engine starting.

The diesel-generator combustion air intake and engine exhaust subsystem provides combustion air directly from the outside to the diesel engine while protecting it from dust, rain, snow and other environmental particulates. It then discharges exhaust gases from the engine to the outside of the diesel generator building more than 20 feet higher than the air intake. The combustion air circuit is separate from the ventilation subsystems and includes weather protected dry type inlet air filters piped directly to the inlet connections of the diesel engine-mounted turbochargers. The combustion air filters are capable of reducing airborne particulate material, assuming the maximum expected airborne particulate concentration at the combustion air intake. Each engine is provided with two

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filters as shown in Figure 8.3.1-4. A differential pressure gauge is installed across each filter to determine the need for filter replacement. The engine exhaust gas circuit consists of the engine exhaust gas discharge pipes from the turbocharger outlets to a single vertically mounted outdoor silencer which discharges to the atmosphere. Manufacturer's recommendations are considered in the design of features to protect the silencer module and other system components from possible clogging due to adverse atmospheric conditions, such as dust storms, rain, ice, and snow.

The diesel-generator engine cooling system is an independent closed loop cooling system, rejecting engine heat through two separate roof-mounted, fan-cooled radiators. The system consists of two separate cooling loops each maintained at a temperature required for optimum engine performance by separate engine-driven coolant water circulating pumps. One circuit cools the engine cylinder block, jacket, and head area, while the other circuit cools the oil cooler and turbocharger aftercooler. The cooling water in each loop passes through a three-way self-contained temperature control valve which modulates the flow of water through or around the radiator, as necessary, to maintain required water temperature. The temperature control valve has an expanding wax-type temperature-sensitive element or equivalent. The cooling circuit, which cools the engine cylinder blocks, jacket, and head areas, includes a keep-warm circuit consisting of a temperature controlled electric heater and an ac motor-driven water circulating pump.

The diesel-generator engine lubrication system is contained on the engine skid and includes an engine oil sump, a main engine-driven oil pump and a continuous engine prelube system consisting of ac and dc motor-driven pumps, an air-driven pump, and a keep warm circuit consisting of a motor driven pump and an electric heater. The prelube system maintains the engine lubrication system in service when the diesel engine is in standby mode. The lube oil is circulated through the engine and various filters and a cooler to maintain the lube oil properties suitable for engine lubrication.

The diesel generator engine fuel oil system consists of two engine-mounted, engine-driven fuel oil pumps that take fuel from the fuel oil day tank, and pump through inline oil filters to the engine fuel injectors and a separate recirculation circuit with a fuel oil cooler. The recirculation circuit discharges back to the fuel oil day tank that is maintained at the proper fuel level by the diesel fuel oil storage and transfer system.

The onsite standby diesel generators are provided with necessary controls and indicators for local or remote monitoring of the operation of the units. Essential parameters are monitored and alarmed in the main control room via the plant data display and processing system as described in Chapter 7. Indications and alarms that are available locally and in the main control room are listed in Table 8.3.1-5.

The design of the onsite standby diesel generators does not ensure functional operability or maintenance access or support plant recovery following design basis events. Maintenance accessibility is provided consistent with the system nonsafety-related functions and plant availability goals.

The simplified piping and instrumentation diagrams for the onsite standby diesel generator units and the associated subsystems are shown on Figures 8.3.1-4 and 8.3.1-5.

The onsite standby power supply system is shown schematically on one line diagram, Figure 8.3.1-1.

The onsite diesel generators were procured in accordance with an equipment specification which includes requirements based upon the manufacturer's standards and applicable recommendations from documents such as NUREG/CR-0660 (Reference 15). Capability to detect system leakage and to prevent crankcase explosions will be based upon manufacturer's recommendations. Control of moisture in the starting air system by the equipment described above will be based upon manufacturer's recommendations. Dust and dirt in the diesel generator room is controlled by the

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diesel generator building ventilation system described in Subsection 9.4.10. Personnel training is addressed as part of overall plant training in Subsection 13.2.6. Automatic engine prelube by the equipment described above will be based upon manufacturer's recommendations. Testing, test loading and preventive maintenance is addressed as part of overall plant testing and maintenance in Chapter 13. Instrumentation to support diagnostics during operation is shown on Figure 8.3.1-4. The overall diesel building ventilation design is described in Subsection 9.4.10 and the combustion air systems are described above. The fuel oil storage and handling system is described in Subsection 9.5.4. High temperature insulation will be based upon manufacturer's recommendations. Response to the effects of engine vibration will be based upon manufacturer's recommendations. Diesel building floor coatings are described in Subsections 6.1.2.1.4 and 6.1.3.2. The diesel generator building HVAC system, as described in Subsection 9.4.10, will function to provide adequate heating, ventilation, and cooling to support the operation of the diesel generators.

8.3.1.1.2.2 Generator

Each generator is a direct-shaft driven, air-cooled self ventilated machine. The generator enclosure is open drip-proof type that facilitates free movement of ventilation air. The generator component design is in compliance with the NEMA MG-1 (Reference 1) requirements.

Each generator produces its rated power at 6900 V, 60 Hz. Each generator continuous rating is based on supplying the electrical ac loads listed in Tables 8.3.1-1 or 8.3.1-2. The loads shown on Tables 8.3.1-1 and 8.3.1-2 represent a set of nonsafety-related loads which provide shutdown capability using nonsafety-related systems. The generators can also provide power for additional investment protection ac loads. The plant operator would normally provide power to these loads by de-energizing one of those system components that are redundantly supplied by both the diesel generators. The diesel generator design is compatible with the step loading requirements identified in Tables 8.3.1-1 and 8.3.1-2. The generator exciter and voltage regulator systems are capable of providing full voltage control during operating conditions including postulated fault conditions.

Each generator has a set of potential and current transformers for protective relaying and metering purposes.

The following generator protection functions are provided via relays that are mounted on the local generator control panel:

Differential (87), overcurrent (50/51), reverse power (32), underfrequency (81), under/over voltage (27/59), loss of excitation (40), ground fault (59N), negative sequence (46), synchronization check (25), voltage balance (60).

Note: The number in the parentheses identifies the ANSI device designation.

8.3.1.1.2.3 Onsite Standby Power System Performance

The onsite standby power system provides reliable ac power to the various plant system electrical loads shown on Tables 8.3.1-1 and 8.3.1-2. These loads represent system components that enhance an orderly plant shutdown under emergency conditions. Additional loads that are for investment protection can be manually loaded on the standby power supply after the loads required for orderly shutdown have been satisfied. The values listed in the "Operating Load (kW)" column of Tables 8.3.1-1 and 8.3.1-2 represent nominal values of the actual plant loads.

Both the diesel engine and the associated generator are rated based on 104°F ambient temperature at 1000 ft elevation as standard site conditions. The selected unit rating has a design margin to accommodate possible derating resulting from other site conditions.

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The diesel generator unit is able to reach the rated speed and voltage and be ready to accept electrical loads within 120 seconds after a start signal.

Each generator has an automatic load sequencer to enable controlled loading on the generator. The automatic load sequencer connects selected loads at predetermined intervals (i.e., load blocks). This feature allows recuperation of generator voltage and frequency to rated values prior to the connection of the next load.

For sequential and manual loading of the onsite standby diesel generator, see Tables 8.3.1-1 and 8.3.1-2.

To enable periodic testing, each generator has synchronizing equipment at a local panel as well as in the main control room.

The logic diagram for diesel generator initiating circuit is shown in Figure 8.3.1-2.

8.3.1.1.2.4 Operation, Inspection, and Maintenance

Operation, inspection and maintenance (including preventive, corrective, and predictive maintenance) procedures consider both the diesel generator manufacturer's recommendations and industry diesel working group recommendations.

8.3.1.1.3 Ancillary Diesel Generators

Power for Class 1E post-accident monitoring, MCR lighting, MCR and divisions B and C I&C room ventilation and for refilling the PCS water storage tank and the spent fuel pool when no other sources of power are available is provided by two ancillary diesel generators located in the annex building. The ancillary diesel generators are not needed for refilling the PCS water storage tank, spent fuel pool makeup, post-accident monitoring or lighting for the first 72 hours following a loss of all other ac sources.

The generators are classified as AP1000 Class D. The generators are commercial, skid-mounted, packaged units and can be easily replaced in the event of a failure. Generator control is manual from a control integral with the diesel skid package. These generators are located in the portion of the Annex Building that is a seismic Category II structure.

For these systems and components, the design of equipment anchorages is consistent with the SSE design of equipment anchorages of seismic Category I items and there should be no spatial interaction with any other non-seismic SSC that could adversely interact to prevent the functioning of the post-72 hour SSCs following an SSE; no dynamic qualification of the active equipment is necessary. Features of this structure which protect the function of the ancillary diesel generators are analyzed and designed for Category 5 hurricanes, including the effects of sustained winds, maximum gusts, and associated wind-borne missiles. The function of the ancillary diesel generators can be maintained through a temporary hose to exhaust fumes in the event of loss of ancillary diesel generator stacks.

The fuel for the ancillary generators is stored in a tank located in the same room as the generators. The fuel tank, piping, and valves are analyzed to show that they withstand an SSE. The tank includes provisions for venting to the outside atmosphere and for refilling from a truck or other mobile source of fuel. The tank is seismic Category II and holds sufficient fuel for 4 days of operation.

Each ancillary diesel generator output is connected to a distribution panel located in the same room as the generators. Each distribution panel has an incoming circuit breaker and outgoing feeder circuit breakers. The outgoing feeder circuit breakers are connected to cables that are routed to the

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divisions B and C voltage regulating transformers and to the passive containment cooling system recirculation pumps (see Figure 8.3.1-3).

8.3.1.1.4 Electrical Equipment Layout

The main ac power system distributes ac power to the reactor, turbine, and balance of plant (BOP) auxiliary electrical loads for startup, normal operation, and normal/emergency shutdown.

The medium voltage switchgear ES1 and ES2 are located in the electrical switchgear rooms 1 and 2 of the annex building. The incoming power is supplied from the unit auxiliary transformers ET2A and ET2B (X windings) via nonsegregated buses. The nonsegregated buses are routed from the transformer yard to the annex building in the most direct path practical.

The switchgear ES3, ES4, ES5, and ES6 are located in the turbine building electrical switchgear rooms. The incoming power is supplied from the unit auxiliary transformers ET2A and ET2B (Y windings) via nonsegregated buses to ES3 and ES4 and from ET2A and ET2B (X windings) to ES5 and ES6. Switchgear ES7 is located in the auxiliary boiler room in the turbine building.

The Class 1E medium voltage circuit breakers, ES31, ES32, ES41, ES42, ES51, ES52, ES61, and ES62, for four reactor coolant pumps are located in the auxiliary building.

The 480 V load centers are located in the turbine building electrical switchgear rooms 1 and 2 and in the annex building electrical switchgear rooms 1 and 2 based on the proximity of loads and the associated 6.9 kV switchgear. Load center 71 is located in the auxiliary boiler room in the turbine building.

The 480 V motor control centers are located throughout the plant to effectively distribute power to electrical loads. The load centers and motor control centers are free standing with top or bottom cable entry and front access. The number of stacks/cubicles varies for each location.

8.3.1.1.5 Heat Tracing System

The electric heat tracing system is nonsafety-related and provides electrical heating where temperature above ambient is required for system operation and freeze protection.

The electric heat tracing system is part of the AP1000 permanent nonsafety-related loads and is powered from the diesel backed 480 V ac motor control centers through 480 V - 208Y/120V transformers and distribution panels.

8.3.1.1.6 Containment Building Electrical Penetrations

The electrical penetrations are in accordance with IEEE 317 (Reference 2).

The penetrations conform to the same functional service level as the cables, (for example, low-level instrumentation is in a separate nozzle from power and control). The same service class separation requirements apply within inboard/outboard terminal boxes.

Individual electrical penetrations are provided for each electrical service level and follow the same raceway voltage grouping described in Subsection 8.3.1.3.4. Optical fibers are installed in instrumentation and control or low voltage power electric penetrations.

The electrical penetrations conductor modules are in penetrations of the same Service Class. Modules for instrumentation signals will be in instrumentation penetrations; modules for control

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power (e.g., 120/125/250V) will be in control power penetrations; modules for low voltage power (e.g., 600 Vac) will be in low voltage power penetrations.

It is possible to combine low voltage power with control power in the same electrical penetration assembly.

Penetrations carrying medium voltage power cables have thermocouples to monitor the temperature within the assembly at the spot expected to have the hottest temperature.

Electrical circuits passing through electrical penetrations have primary and backup protective devices. These devices coordinate with the thermal capability curves (I^2t) of the penetration assemblies. The penetrations are rated to withstand the maximum short-circuit currents available either continuously without exceeding their thermal limit, or at least longer than the field cables of the circuits so that the fault or overload currents are interrupted by the protective devices prior to a potential failure of a penetration. Penetrations are protected for the full range of currents up to the maximum short circuit current available.

Primary and backup protective devices protecting Class 1E circuits are Class 1E in accordance with IEEE 741 (Reference 10). Primary and backup protective devices protecting non-Class 1E circuits are non-Class 1E.

Penetration overcurrent protection coordination curves are generated based on the protection requirements specified by the penetration equipment manufacturer. When necessary, penetrations are protected for instantaneous overcurrent by current limiting devices such as current-limiting fuses, current-limiting breakers, or reactors.

Procedures implement periodic testing of protective devices that provide penetration overcurrent protection. A sample of each different type of overcurrent device is selected for periodic testing during refueling outages. Testing includes:

- Verification of thermal and instantaneous trip characteristics of molded case circuit breakers.
- Verification of long time, short time, and instantaneous trips of medium voltage vacuum circuit breakers.
- Verification of long time, short time, and instantaneous trips of low voltage air circuit breakers.
- Verification of Class 1E and non-Class 1E dc protective device characteristics (except fuses) per manufacturer recommendations, including testing for overcurrent interruption and/or fault current limiting.

Penetration protective devices are maintained and controlled under the plant configuration control program. A fuse control program, including a master fuse list, is established based on industry operating experience.

8.3.1.1.7 Grounding System

The AP1000 grounding system complies with the guidelines provided in IEEE 665 (Reference 18) and IEEE 1050 (Reference 20). The grounding system consists of the following four subsystems:

- Station grounding grid
- Electrical system grounding (neutral grounding)
- Equipment grounding
- Instrument/computer grounding

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The station grounding grid subsystem consists of buried, interconnected bare copper conductors and ground rods forming a plant ground grid matrix. The subsystem maintains a uniform ground potential and limits the step and touch potentials to safe values under all fault conditions.

The electrical system grounding (neutral grounding) subsystem provides grounding of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, and onsite standby diesel generators. The main and diesel generator neutrals are grounded through grounding transformers providing high-impedance grounding. The main step-up and load center transformer neutrals are grounded solidly. The auxiliary (unit and reserve) transformer secondary winding neutrals are resistance grounded except for the auxiliary transformer UAT 2C, which is solidly grounded to meet the requirements of the electrode type auxiliary boiler.

The equipment grounding subsystem provides grounding of the equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, MCCs, and control cabinets. The equipment grounding subsystem is connected to the station grounding grid.

The instrument/computer grounding subsystem provides plant instrument/computer grounding through separate radial grounding systems consisting of isolated instrumentation ground buses and insulated cables. The radial grounding systems are connected to the station grounding grid at one point only and are insulated from all other grounding circuits.

The design of the grounding grid system and the lightning protection system depends on the soil resistivity and lightning activity in the area. Therefore, the design of both systems is site-specific. See Subsection 8.3.3 for the responsibility for the grounding and lightning protection systems.

8.3.1.1.8 Lightning Protection

The lightning protection system, consisting of air terminals and ground conductors, is provided for the protection of exposed structures and buildings housing safety-related and fire protection equipment in accordance with NFPA 780 (Reference 19). Also, lightning arresters are provided in each phase of the transmission lines and at the high-voltage terminals of the outdoor transformers. The isophase bus connecting the main generator and the main transformer and the medium-voltage switchgear is provided with lightning arresters. In addition, surge suppressors are provided to protect the plant instrumentation and monitoring system from lightning-induced surges in the signal and power cables connected to devices located outside.

Direct-stroke lightning protection for facilities is accomplished by providing a low-impedance path by which the lightning stroke discharge can enter the earth directly. The direct-stroke lightning protection system, consisting of air terminals, interconnecting cables, and down conductors to the ground grid, are provided external to the facility in accordance with the guidelines included in NFPA 780. The system is connected to the station grounding grid subsystem to facilitate dissipation of the large current of a direct lightning stroke. The lightning arresters and the surge suppressors connected directly to ground provide a low-impedance path to ground for the surges caused or induced by lightning. Thus, fire or damage to facilities and equipment resulting from a lightning stroke is avoided.

The design of direct-stroke lightning protection and the associated grounding depends on the lightning activity at the plant site and the soil resistivity of the ground. It is site specific, and the design responsibility is as described in Subsection 8.3.3.

8.3.1.2 Analysis

The ac power system is non-Class 1E and is not required for safe shutdown. Compliance with existing regulatory guides and General Design Criteria is covered in Table 8.1-1 of Section 8.1.

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8.3.1.3 Raceway/Cable

8.3.1.3.1 General

The raceway system for non-Class 1E ac circuits complies with IEEE 422 (Reference 3) in respect to installation and support of cable runs between electrical equipment including physical protection. A raceway system is the complete assembly of raceways (e.g., conduit, cable tray, or wireway) and raceway supports.

8.3.1.3.2 Load Groups Segregation

There are two nonsafety-related load groups associated with different transformers, buses, and onsite standby diesel generators. No physical separation is required as these two ac load groups are non-Class 1E and nonsafety-related.

8.3.1.3.3 Cable Derating and Conduit and Cable Tray Fill

Cable Derating

The power and control cable insulation is designed for a conductor temperature of 90°C during normal operation. The allowable current carrying capacity of the cable is based on the insulation design temperature while the surrounding air is at an ambient temperature of 65°C for the containment and 40 to 50°C for other areas. The power and control cables in the integrated head package (IHP) CRDM cable tray are designed to an ambient air temperature of 50°C. Power cables, feeding loads from switchgear, load centers, motor control centers, and distribution panels are sized at 125 percent of the full-load current at a 100-percent load factor. Temperature and full-load current conditions for high-temperature cables, specialty cables, and cables within interfacing commodities are determined based on application-specific conditions.

The power cable ampacities are in accordance with the Insulated Cable Engineers Association publications (Reference 4 and 11), and National Electric Code (Reference 5). The derating is based on the type of installation, the conductor and ambient temperature, the number of cables in a raceway, and the grouping of the raceways. A further derating of the cables is applied for those cables which pass through a fire barrier. The method of calculating these derating factors is determined from the Insulated Cable Engineers Association publications and other applicable standards.

Instrumentation cable insulation is also designed for a conductor temperature of 90°C during normal operation. The operating power of these cables is low (usually mV or mA) and does not cause cable overheating at the maximum design ambient temperature.

For circuits that are routed partly through conduit and partly through trays or underground ducts, the cable size is based on the ampacity in that portion of the circuit with the lowest indicated current carrying capacity.

Conduit and Cable Tray Fill

Cable tray design is based on random cable fill of 40 percent of usable tray depth. Conduit fill design is in compliance with Chapter 9 of National Electrical Code (Reference 5). If tray or conduit fill exceeds the above stated maximum fill, the fill will be analyzed and the acceptability documented.

8.3.1.3.4 Raceway and Cable Routing

When cable trays are arranged in a vertical array they are arranged physically from top to bottom, in accordance with the function and voltage class of the cables as follows:

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- Medium voltage power (6.9 kV)
- Low voltage power (480 Vac, 208Y/120 Vac, 125 Vdc/250 Vdc)
- 120 Vac/125 Vdc/250 Vdc signal and control (if used)
- Instrumentation (analog and digital)

480 Vac, 208Y/120 Vac, 125 Vdc/250 Vdc power cables may be mixed with 120 Vac/125 Vdc/ 250 Vdc signal and control cables in random fill cable trays, subject to the following additional criteria:

- Insulation ratings are the same voltage level
- Respective sizes do not differ greatly
- Operating temperatures are compatible
- The fault current on any cable prior to the breaker or fuse clearing time is not sufficient to heat the cable insulation to the ignition point
- The signal or control function served by the Y cable is not susceptible to EMI/RFI levels generated in the low voltage power trays (XA)
- The “Y” cable does not directly interface with any inputs/outputs of a core digital control system

Separate raceways are provided for medium voltage power, low voltage power and control, as well as instrumentation cables.

Raceways that route non-Class 1E cables and their supports installed in seismic Category I structures are designed and/or physically arranged so that the safe shutdown earthquake could not cause unacceptable structural interaction or failure of seismic Category I components.

Raceways are kept at a reasonable distance from heat sources such as steam piping, steam generators, boilers, high and low pressure heaters, and any other actual or potential heat source. Cases of heat source crossings are evaluated and supplemental heat shielding is used if necessary.

See Subsection 8.3.2 for raceway and cable routing for Class 1E cables.

8.3.1.4 Inspection and Testing

Preoperational tests are conducted to verify proper operation of the ac power system. The preoperational tests include operational testing of the diesel load sequencer and diesel generator capacity testing.

Procedures are established for periodic verification of proper operation of the Onsite AC Power System capability for automatic and manual transfer from the preferred power supply to the maintenance power supply and return from the maintenance power supply to the preferred power supply.

8.3.1.4.1 Diesel Load Sequencer Operational Testing

The load sequencer for each standby diesel generator is tested to verify that it produces the appropriate sequencing signals within five (5) seconds of the fixed duration portion of the intervals specified in Tables 8.3.1-1 and 8.3.1-2. The five second margin is sufficient for proper diesel generator transient response.

The time sequence values defined in Tables 8.3.1-1 and 8.3.1-2 are the allowable initiation times for individual load blocks (starting from the maximum 120 seconds allowed for the DG to start and be ready to pickup load). The tables reflect the allowable interval that each load block has to initiate the subsequent load block. This interval is the time difference between the load blocks identified in Tables 8.3.1-1 and 8.3.1-2. For example, if the load block 0 interval begins at 120 seconds, it ends at

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140 seconds with the initiation of load block 1, thus the table establishes the allowable load block 0 interval to be 20 seconds.

The automatic load sequencer produces a load block initiation signal for each load block. The active load block duration is terminated by the initiation of the subsequent load block. The subsequent load block initiation requires the active load block signal to be available for a fixed duration (less than or equal to the intervals in Tables 8.3.1-1 and 8.3.1-2) and the DG ready to pickup load signal to be present after the fixed duration. The "ready to pickup load" signal can extend the interval beyond the fixed duration to allow the DG to recover from the current load block transient response. Testing verifies that each load block initiates within ± 5 seconds of the fixed duration in each interval (i.e., the predetermined interval).

8.3.1.4.2 Standby Diesel Generator Capacity Testing

Each standby diesel generator is tested to verify the capability to provide 4000 kW while maintaining the output voltage and frequency within the design tolerances of 6900±10% Vac and 60±5% Hz. The 4000 kW capacity is sufficient to meet the loads listed in Tables 8.3.1-1 and 8.3.1-2. The test duration will be the time required to reach engine temperature equilibrium plus 2.5 hours. This duration is sufficient to demonstrate long-term capability.

8.3.1.4.3 Ancillary Diesel Generator Capacity Testing

Each ancillary diesel generator is tested to verify the capability to provide 35 kW while maintaining the output voltage and frequency within the design tolerances of 480±10% Vac and 60±5% Hz. The 35 kW capacity is sufficient to meet the loads listed in Table 8.3.1-4. The test duration will be the time required to reach engine temperature equilibrium plus 2.5 hours. This duration is sufficient to demonstrate long-term capability.

8.3.2 DC Power Systems

8.3.2.1 Description

The plant dc power system is comprised of independent Class 1E and non-Class 1E dc power systems. Each system consists of ungrounded stationary batteries, dc distribution equipment, and uninterruptible power supply (UPS).

The Class 1E dc and UPS system provides reliable power for the safety-related equipment required for the plant instrumentation, control, monitoring, and other vital functions needed for shutdown of the plant. In addition, the Class 1E dc and UPS system provides power to the normal and emergency lighting in the main control room and at the remote shutdown workstation.

The Class 1E dc and UPS system is capable of providing reliable power for the safe shutdown of the plant without the support of battery chargers during a loss of all ac power sources coincident with a design basis accident (DBA). The system is designed so that no single failure will result in a condition that will prevent the safe shutdown of the plant.

The non-Class 1E dc and UPS system provides continuous, reliable electric power to the plant non-Class 1E control and instrumentation loads and equipment that are required for plant operation and investment protection and to the hydrogen igniters located inside containment. Operation of the non-Class 1E dc and UPS system is not required for nuclear safety. See Subsection 8.3.2.1.2.

The batteries for the Class 1E and non-Class 1E dc and UPS systems are sized in accordance with IEEE 485 (Reference 6). The operating voltage range of the Class 1E batteries and of the EDS5 turbine generator motor load support batteries is 210 to 280 Vdc. The maximum equalizing charge

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voltage for the Class 1E and EDS5 batteries is 280 Vdc. The nominal system voltage is 250 Vdc. The operating voltage range of non-Class 1E EDS1 through EDS4 batteries is 105 to 140 Vdc. The maximum equalizing charge voltage for non-Class 1E EDS1 through EDS4 batteries is 140 Vdc. The nominal system voltage is 125 Vdc for non-Class 1E EDS1 through EDS4.

The qualification test program for AP1000 24-hour and 72-hour class 1E batteries meets or exceeds the requirements of IEEE Standard 323, IEEE Standard 344, and IEEE Standard 535, including required and recommended margins, and it is in regulatory compliance with Regulatory Guides 1.89, 1.100, and 1.158. The test program requires that the battery be subjected to accelerated thermal aging and discharge cycling (wear aging) in accordance with IEEE Standard 323 and IEEE Standard 535 over its qualified life objective followed by the DBE seismic event performed in accordance with IEEE Standard 344. In addition, following the aging process, the test specimens shall be subjected to environmental testing to verify the equipment's ability to operate in postulated abnormal environmental conditions during plant operation. Discharge cycling will be performed as a potential aging mechanism prior to seismic testing using Type 3 modified performance test method in accordance with IEEE Standard 450-2002 at intervals representative of the AP1000 surveillance test requirements of the batteries with 10% margin in the number of discharge cycles, which establishes margin for the expected life of the battery. Thus, magnitude/duration (modified performance test versus service and performance tests) and test interval envelop the AP1000 and industry cycling requirements. If new battery failure modes are detected during the qualification testing, these failure modes will be evaluated for any potential changes to the technical specification's surveillance requirements and revision to maintenance procedures required to ensure identification of degradation prior to reaching those failure modes during plant operation. Following the qualification process, a report that uniquely describes step-by-step the tests performed and results, and addresses any deficiencies and repairs, including photographs, drawings, and other materials, will be maintained for records.

8.3.2.1.1 Class 1E DC and UPS System

8.3.2.1.1.1 Class 1E DC Distribution

The Class 1E dc distribution is in compliance with applicable General Design Criteria, IEEE standards, and Regulatory Guides listed in Subsection 8.1.4.3. The scope of compliance encompasses physical separation, electrical isolation, equipment qualification, effects of single active component failure, capacity of battery and battery charger, instrumentation and protective devices, and surveillance test requirements. The Class 1E dc components are housed in seismic Category I structures. For system configuration and equipment rating, see Class 1E dc one-line diagram, Figure 8.3.2-1. Nominal ratings of major Class 1E dc equipment are listed in Table 8.3.2-5.

There are four independent, Class 1E 250 Vdc divisions, A, B, C, and D. Divisions A and D are each comprising one battery bank, one switchboard, and one battery charger. The battery bank is connected to Class 1E dc switchboard through a set of fuses and a disconnect switch. Divisions B and C are each composed of two battery banks, two switchboards, and two battery chargers. The first battery bank in the four divisions, designated as 24-hour battery bank, provides power to the loads required for the first 24 hours following an event of loss of all ac power sources concurrent with a design basis accident (DBA). The battery charger for each 24-hour division battery bank provides two Class 1E battery charger input voltage status signals to the protection and safety monitoring system for actuation of engineered safety features. The second battery bank in divisions B and C, designated as 72-hour battery bank, is used for those loads requiring power for 72 hours following the same event. Each switchboard connected with a 24-hour battery bank supplies power to an inverter, a 250 Vdc distribution panel, and a 250 Vdc motor control center. Each switchboard connected with a 72-hour battery bank supplies power to an inverter. No load shedding or load management program is needed to maintain power during the required 24-hour safety actuation period.

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A single spare battery bank with a spare battery charger is provided for the Class 1E dc and UPS system. In the case of a failure or unavailability of the normal battery bank and the battery charger, permanently installed cable connections allow the spare to be connected to the affected bus by plug-in locking type disconnect along with kirk-key interlock switches. The plug-in locking type disconnect and kirk-key interlock switches permit connection of only one battery bank and battery charger at a time so that the independence of each battery division is preserved. The spare battery and the battery charger can also be utilized as a substitute when offline testing, maintenance, and equalization of an operational battery bank are desired. If the spare battery charger is substituted for a 24-hour division battery charger and the 24-hour division battery charger is disconnected from its normal power supply, the two undervoltage relays on the 24-hour division battery charger each provide a low input voltage signal to the protection and safety monitoring system for the division.

Each 250 Vdc Class 1E battery division and the spare battery bank are separately housed as described in Subsection 8.3.2.1.3.

Each battery bank, including the spare, has a battery monitor system that detects battery open-circuit conditions and monitors battery voltage. The battery monitor provides a trouble alarm in the main control room. The battery monitors are not required to support any safety-related function. The battery monitor connection to the Class 1E batteries are provided with safety-related fuse isolation panels that provide the required isolation between non-Class 1E and Class 1E circuits and equipment. The fuse panels are qualified as isolation devices in accordance with IEEE 384 (Reference 7) and Regulatory Guide 1.75, with the exceptions noted in Appendix 1A. Monitoring and alarming of dc current and voltages are through the plant control system which includes a battery discharge rate alarm. AP1000 generally uses fusible disconnect switches in the Class 1E dc system. If molded-case circuit breakers are used for dc applications, they will be sized to meet the dc interrupting rating requirements.

The Class 1E dc switchboards employ fusible disconnect switches and have adequate short circuit and continuous-current ratings. The main bus bars are braced to withstand mechanical forces resulting from a short-circuit current. Fused transfer switch boxes, equipped with double pole single throw transfer switches, are provided to facilitate battery testing, and maintenance.

Battery chargers are connected to dc switchboard buses. The input ac power for the Class 1E dc battery chargers is supplied from non-Class 1E 480 Vac diesel generator backed motor control centers. The battery chargers provide the required isolation between the non-1E ac and the Class 1E dc electrical systems. The battery chargers are qualified as isolation devices in accordance with IEEE 384 (Reference 7) and Regulatory Guide 1.75. Each battery charger has an input ac and output dc circuit breaker for the purpose of power source isolation and required protection. Each battery charger prevents the ac supply from becoming a load on the battery due to a power feedback as a result of the loss of ac power to the chargers. Each battery charger has a built-in current limiting circuit, adjustable between 110 to 125 percent of its rating to hold down the output current in the event of a short circuit or overload on the dc side. The output of the charger is ungrounded and filtered. The output float and equalizing voltages are adjustable. The battery chargers have an equalizing timer and a manual bypass switch to permit periodic equalizing charges. Each charger is capable of providing the continuous demand on its associated dc system while providing sufficient power to charge a fully discharged battery (as indicated by the loads in Tables 8.3.2-1 through 8.3.2-4) within a 24-hour period. The battery chargers are provided with a common failure/trouble alarm.

The Class 1E dc motor control centers operate at 250 Vdc nominal two wire, ungrounded system. The dc motor control centers provide branch circuit protection for the dc motor-operated valves. Motor-operated valves are protected by thermal overload devices in accordance with Regulatory Guide 1.106. Motor overload condition is annunciated in the main control room. The loads fed from the motor control centers are protected against a short-circuit fault by fusible disconnect switches.

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The Class 1E dc distribution panels provide power distribution and tripping capability between the 250 Vdc power sources and the assigned safeguard loads indicated on Figure 8.3.2-1.

No site-specific non-Class 1E dc loads are connected to the Class 1E dc system.

8.3.2.1.1.2 Class 1E Uninterruptible Power Supplies

The Class 1E UPS provides power at 208 Y/120 Vac to four independent divisions of Class 1E instrument and control power buses. Divisions A and D each consist of one Class 1E inverter associated with an instrument and control distribution panel and a backup voltage regulating transformer with a distribution panel. The inverter is powered from the respective 24-hour battery bank switchboard. Divisions B and C each consist of two inverters, two instrument and control distribution panels, and a voltage regulating transformer with a distribution panel. One inverter is powered by the 24-hour battery bank switchboard and the other, by the 72-hour battery bank switchboard. For system configuration and equipment rating, see Figures 8.3.2-1 and 8.3.2-2. The nominal ratings of the Class 1E inverters and the voltage regulating transformers are listed in Table 8.3.2-5. Under normal operation, the Class 1E inverters receive power from the associated battery bank. If an inverter is inoperable or the Class 1E 250 Vdc input to the inverter is unavailable, the power is transferred automatically to the backup ac source by a static transfer switch featuring a make-before-break contact arrangement. The backup power is received from the diesel generator backed non-Class 1E 480 Vac bus through the Class 1E voltage regulating transformer. In addition, a manual mechanical bypass switch is provided to allow connection of backup power source when the inverter is removed from service for maintenance.

In order to supply power during the post-72-hour period following a design basis accident, provisions are made to connect a ancillary ac generator to the Class 1E voltage regulating transformers (divisions B and C only). This powers the Class 1E post-accident monitoring systems and the lighting in the main control room and ventilation in the MCR and divisions B and C I&C rooms. See Subsection 8.3.1.1.1 for post-72-hour power distribution details, Subsection 9.4.1 for post-72-hour ventilation, and Subsection 9.5.3 for post-72-hour lighting details respectively.

8.3.2.1.2 Non-Class 1E DC and UPS System

The non-Class 1E dc and UPS system consists of the electric power supply and distribution equipment that provide dc and uninterruptible ac power to the plant non-Class 1E dc and ac loads that are critical for plant operation and investment protection and to the hydrogen igniters located inside containment. The non-class 1E dc and UPS system is comprised of two subsystems representing two separate power supply trains. The subsystems are located in separate rooms in the annex building. Figure 8.3.2-3, non-Class 1E dc and UPS system one line diagram represents the distribution configuration.

Each of the EDS1 and 3, and 2 and 4 subsystems consists of separate dc distribution buses. These two buses can be connected by a normally open circuit breaker to enhance the power supply source availability.

Each dc subsystem includes battery chargers, stationary batteries, dc distribution equipment, and associated monitoring and protection devices.

DC buses 1, 2, 3, and 4 (See Figure 8.3.2-3) provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E uninterruptible power supply ac system. An alternate regulated ac power source for the UPS buses is supplied from the associated regulating transformers. DC bus 5 supplies large dc motors. This configuration isolates the large motors.

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The onsite standby diesel generator backed 480 Vac distribution system provides the normal ac power to the battery chargers. Industry standard stationary batteries that are similar to the Class 1E design are provided to supply the dc power source in case the battery chargers fail to supply the dc distribution bus system loads. The batteries are sized to supply the system loads for a period of at least two hours after loss of all ac power sources. Hydrogen igniter power requirements as stated in Subsection 6.2.4.2.3 are provided through the use of appropriate load shedding.

The dc distribution switchboard houses the dc feeder protection device, dc bus ground fault detection, and appropriate metering. The component design and the current interrupting device selection follow the circuit coordination principles.

The non-Class 1E dc and UPS system is designed to meet the quality guidelines established by Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment that is not Safety-Related."

An EDS spare battery bank (EDSS-DB-1A and EDSS-DB-1B) with spare battery chargers (EDSS-DC-1 and EDSS-DC-2), is provided for the non-Class1E dc and UPS system (Figure 8.3.2-3). This 125/250 Vdc EDS spare battery and the EDS buses are completely independent of and not connectable to the IDS buses. In the case of an EDS failure or unavailability of the normal battery bank and the battery charger, permanently installed cable connections allow the spare to be connected to the affected bus with kirk-key interlock switches. The kirk-key interlock switches permit connection of only one battery bank and battery charger at a time. The spare battery bank and the battery chargers can also be utilized as a substitute when offline testing, maintenance, and equalization of an operational battery bank is desired. A center tap bus arrangement consisting of a +125 Vdc bus, a 0 bus, and a -125 Vdc bus is used in the spare fused transfer switchbox (EDSS-DF-1) in order to obtain either 125 Vdc or 250 Vdc. Spare battery EDSS-DB-1A and spare battery charger EDSS-DC-1 are connected to the +125 Vdc bus and 0 bus, which feed EDS2 and EDS4. Spare battery EDSS-DB-1B and spare battery charger EDSS-DC-2 are connected to the 0 bus and -125 Vdc bus, which feed EDS1 and EDS3. EDS5 is connected to the +125 Vdc and -125 Vdc bus, which puts the two battery strings in series, providing 250 Vdc.

8.3.2.1.3 Separation and Ventilation

For the Class 1E dc system, the 24-hour and the 72-hour battery banks are housed in the auxiliary building in ventilated rooms apart from chargers and distribution equipment. The battery rooms are ventilated to limit hydrogen accumulation. Subsection 9.4.1 describes the ventilation system in the battery rooms. Each of the four divisions of dc systems are electrically isolated and physically separated to prevent an event from causing the loss of more than one division.

8.3.2.1.4 Maintenance and Testing

Components of the 125 Vdc and 250 Vdc systems undergo periodic maintenance tests to determine the condition of the system. Batteries are checked for electrolyte level, specific gravity, and cell voltage, and are visually inspected.

The surveillance testing of the Class 1E 250 Vdc system is performed as required by the Technical Specifications.

The inverter DC input protection will be set at least 10% higher than the battery charger trip setpoints to prevent the inverter tripping before the battery charger. The time delay for the inverter high dc input voltage trip will be set higher than the time delay for the battery charger to prevent the inverter tripping before the battery charger.

Procedures are established for inspection and maintenance of Class 1E and non-Class 1E batteries. Class 1E battery maintenance, testing, and replacement are performed in conformance with

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approved Technical Specifications. Batteries are inspected periodically to verify proper electrolyte levels, specific gravity, cell temperature and battery float voltage. Cells are inspected in conformance with IEEE 450 and vendor recommendations.

The clearing of ground faults on the Class 1E dc system is also addressed by procedure. The battery testing procedures are written in conformance with IEEE 450 and the Technical Specifications.

Procedures are established for periodic testing of the Class 1E battery chargers and Class 1E voltage regulating transformers in accordance with the manufacturer recommendations.

- Circuit breakers in the Class 1E battery chargers and Class 1E voltage regulating transformers that are credited for an isolation function are tested through the use of breaker test equipment. This verification confirms the ability of the circuit to perform the designed coordination and corresponding isolation function between Class 1E and non-Class 1E components. Circuit breaker testing is done as part of the Maintenance Rule program and testing frequency is determined by that program.

- Fuses / fuse holders that are included in the isolation circuit are visually inspected.

- Class 1E battery chargers are tested to verify current limiting characteristic utilizing manufacturer recommendation and industry practices. Testing frequency is in accordance with that of the associated battery.

8.3.2.2 Analysis

Compliance with General Design Criteria (GDC) and Regulatory Guides is discussed in Sections 3.1 and 1.9, respectively. Refer to Table 8.1-1 of Section 8.1 for guidelines and applicability of GDC, Regulatory Guides and IEEE Standards. A failure modes and effects analysis for the Class 1E dc and UPS system is provided in Table 8.3.2-7.

In the event of a loss of offsite power coincident with a main generator trip, ac power to the battery charger is provided from two separate non-Class 1E onsite standby diesel generators. Divisions A and C chargers receive their ac power from one diesel generator, ZOS MG 02A, and division B and D chargers from the second diesel generator, ZOS MG 02B. Provisions are also made to power the post accident monitoring systems and the main control room lighting loads in divisions B and C from ancillary ac generators during the post 72-hour period as described in Subsection 8.3.2.1.1.2.

The Class 1E battery chargers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side, however, the voltage regulating transformers do not have active components to limit current; therefore, the Class 1E voltage regulating transformer maximum current is determined by the impedance of the transformer. They have built-in circuit breakers at the input and output sides for protection and isolation. The circuit breakers are coordinated and periodically tested to verify their current-limiting characteristics. They are qualified as isolation devices between Class 1E and non-Class 1E circuits in accordance with IEEE 384 and Regulatory Guide 1.75.

The four divisions are independent, located in separate rooms, cannot be interconnected, and their circuits are routed in dedicated, physically separated raceways. This level of electrical and physical separation prevents the failure or unavailability of a single battery, battery charger, or inverter from affecting adversely a redundant division.

The Class 1E dc and UPS system is designed in accordance with IEEE 308 (Reference 8) and IEEE 946 (Reference 9). Important system component failures are annunciated. The battery monitoring system detects battery open circuit condition and monitors battery voltage. The Class 1E

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208Y/120Vac distribution panels are equipped with undervoltage protection. The set of fuses located in the 250 Vdc switchboards provide selective tripping of circuits for a fault to limit the effects of the abnormal condition, minimize system disturbance and protect the battery from complete accidental discharge through a short circuit fault. The Class 1E dc system is ungrounded, thus, a single ground fault does not cause immediate loss of the faulted system. Ground detections with alarms are provided for each division of power so that ground faults can be located and removed before a second ground fault could disable the affected circuit. A spare battery bank and charger enables testing, maintenance, and equalization of battery banks offline. This configuration provides the capability for each battery bank or battery charger to be separately tested and maintained (including battery discharge tests, battery cell replacement, battery charger replacement) without limiting continuous plant operation at 100-percent power. Substitution of the spare battery charger for a division 24-hour division battery charger sends two low Class 1E battery charger input voltage signals to the protection and safety monitoring system from the division removed from service if the division 24-hour division battery charger is disconnected from the normal power supply. The low Class 1E battery charger input voltage signal removes the automatic depressurization system and in-containment refueling water storage tank injection block in the affected division and generates a partial trip for other engineered safety features supported by the signal as described in Subsection 7.3.1.2.

Short circuit analyses will be performed in accordance with IEEE 946 (Reference 9) and/or other acceptable industry standards or practices to determine fault currents. Circuit interrupting device coordination analyses will be performed in accordance with IEEE 141, 242 (References 16 and 17), and/or other acceptable industry standards or practices.

8.3.2.3 Physical Identification of Safety-Related Equipment

Each safety-related circuit and the raceway that routes it is given a unique identification number to distinguish between circuits and raceways of different voltage level or separation groups. Each raceway is color coded with indelible ink, paint (consistent with the considerations identified in Subsection 6.1.2.1.1), or adhesive markers (adhesive markers are not used in the containment) at intervals of 15 feet or less along the length of the raceway and on both sides of floor or wall penetrations. Each cable is color coded at a maximum of 5 feet intervals along the length of the cable and cable markers showing the cable identification number are applied at each end of the cable.

The following color coding is used for identification purposes:

Separation Group	Color Code
A	Brown
B	Green
C	Blue
D	Yellow
S	Orange
N	Black

8.3.2.4 Independence of Redundant Systems

8.3.2.4.1 General

The routing of cable and the design of raceways prevents a single credible event from disabling a redundant safety-related plant function.

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8.3.2.4.2 Raceway and Cable Routing

There are six separation groups for the cable and raceway system: group A, B, C, D, S, and N. Separation group A contains safety-related circuits from division A. Similarly, separation group B contains safety-related circuits from division B; group C from division C; group D from division D; group S from the spare separation group; and group N from nonsafety-related circuits.

Cables of one separation group are run in separate raceway and physically separated from cables of other separation groups. Group N raceways are separated from safety-related groups A, B, C and D. Separation between safety-related groups, and between safety-related groups and nonsafety-related cables, are routed according to spatial separation stipulated in Regulatory Guide 1.75 and IEEE 384 with the exceptions listed below.

Note: If not explicitly specified below, “enclosed raceway” and “enclosed configuration” refer to conduit, enclosed trays or transition boxes designed and used for supporting or enclosing wires, cables or busbars. “Open configuration” refers to free air cable and open cable trays. The exceptions are limited to low-voltage power system applications and below (instrumentation and control).

Non-hazard Area Exceptions

- The minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch between open to enclosed configurations for configurations involving low-voltage power cables.

- The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch between open configurations, and between open to closed configurations, for configurations that involve exclusively limited energy content cables (instrumentation and control).

- For configurations involving low-voltage power cables within an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway.

- Within panels and control switchboards, the minimum horizontal separation between components or cables of different separation groups (both field-routed and vendor-supplied internal wiring) is 1 inch, and the minimum vertical separation is 1 inch.

Limited Hazard Area Exceptions

- The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch with configurations that involve exclusively limited energy content cables (instrumentation and control).

- The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches with configurations that involve low-voltage power cables for cable sizes less than and equal to 2/0 AWG. The minimum vertical separation may be reduced to 3 inches for the configuration with a conduit above and crossing an open tray at an angle equal to or greater than 45 degrees.

- The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches between a conduit and an open configuration for low-voltage power cables greater than 2/0 AWG but not greater than 750 kcmil. The vertical separation may be reduced to 3 inches if the conduit is above and crossing an open tray at an angle equal to or greater than 45 degrees.

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- The minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway for low-voltage power cables.

- The minimum vertical separation is 1 inch and the minimum horizontal separation is 1 inch for configurations with a non-safety conduit and a free air safety cable with low-voltage power cables and below.

- Within panels and control switchboards, the minimum horizontal separation between components or cables of different separation groups (both field-routed and vendor-supplied internal wiring) is 1 inch, and the minimum vertical separation is 1 inch.

Hazard Area Exceptions

- Within hazard areas, the minimum spatial separation between safety-related groups, and between safety-related groups and nonsafety-related cables is the same as the spatial separation applicable in limited hazard areas, unless otherwise specified by IEEE 384 for an external hazard.

The exceptions to the guidance in Regulatory Guide 1.75 are based on test results used to support exceptions to the separation guidance for operating nuclear power plants. A summary of test results from ten electrical separation test programs is documented in Reference 13. These test programs support the AP1000 exceptions.

In addition to the listed exceptions, other cable configurations with lesser separation distances than those specified in IEEE 384 Sections 6.1.3, 6.1.4, and 6.1.5 can be established by appropriate analysis of the proposed cable installation, per IEEE 384 Section 6.1.1.3. Additional exceptions where separation by analysis has been established are documented in Reference 202.

Fiber optic cables are not capable of generating a fire due to an internally generated fault. Therefore, they do not require separation from other fiber optic cables or from current carrying cables when the fiber optic cable is considered the source of a potential fire.

Non-Class 1E circuits are electrically isolated from Class 1E circuits, and Class 1E circuits from different separation groups are electrically isolated by isolation devices, shielding and wiring techniques, physical separation (in accordance with Regulatory Guide 1.75 for circuits in raceways), or an appropriate combination thereof.

When isolation devices are used to isolate Class 1E circuits from non-Class 1E circuits, the circuits within or from the Class 1E equipment or devices are identified as Class 1E and are treated as such. Beyond the isolation device(s) these circuits are identified as non-Class 1E and are separated from Class 1E circuits in accordance with the above separation criteria.

Power and control cables are installed in conduits, solid bottom trays, or ventilated bottom trays (ladder-type). Solid tray covers are used in outdoor locations and indoors where trays run in areas where falling debris is a problem. Instrumentation cables are routed in conduit or solid bottom cable tray with solid tray covers as required. The cables are derated for specific application in the location where they are installed as stated in Subsection 8.3.1.3.3. The environmental design of electrical equipment including Class 1E cables under normal and abnormal operating conditions is discussed in Section 3.11.

Separate trays are provided for each voltage service level: 6.9 kV, low voltage power (480 Vac, 208Y/120 Vac, 125 Vdc, 250 Vdc), high-level signal and control (120 Vac, 125 Vdc, 250 Vdc), and low level signal (instrumentation). A tray designed for a single class of cables shall contain only cables of the same class except that low voltage power cables may be routed in raceways with high level signal

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and control cables if their respective sizes do not differ greatly and if they have compatible operating temperatures. When this is done in trays, the power cable ampacity is calculated as if all cables in the tray are power cable. Low voltage power cable and high level signal and control cable will not be routed in common raceways if the fault current, within the breaker or fuse clearing time, is sufficient to heat the insulation to the ignition point. Vertically stacked trays are arranged from top to bottom as stated in Subsection 8.3.1.3.4. In general, a minimum of 12 inches vertical spacing is maintained between trays of different service levels within the stack.

The electrical penetrations are in accordance with IEEE 317 (Reference 2). Class 1E and non-Class 1E electrical penetration assemblies are maintained in a separate nozzle. The physical separation of the Class 1E electrical penetration assemblies are in accordance with Regulatory Guide 1.75. The containment building penetrations are described in Subsection 8.3.1.1.6.

A raceway system is the complete assembly of raceways (e.g., conduit, cable tray, or wireway) and raceway supports. Raceways installed in seismic Category I structures have seismically designed supports or are shown not to affect safety-related equipment should they fail. Trays are not attached rigidly to seismic Category I equipment. Conduits may be attached to seismic Category I equipment with flexible type connections.

8.3.2.4.3 Hazard Protection

See Subsection 8.3.2.4.2 for separation requirements in hazard areas.

Where hazards to safety-related raceway systems are identified, a predetermined minimum separation is maintained between the break and/or missile source and any safety-related raceway systems, or a barrier designed to withstand the effects of the hazard is placed to prevent damage to raceway of redundant systems. For details of missile protection and high-energy line break protection, see Sections 3.5 and 3.6, respectively.

Where redundant circuits, devices, or equipment (different separation groups) are exposed to the same external hazard(s), predetermined spatial separation is provided. Where the spatial separation cannot be met, qualified barriers are installed. For details on fire protection, see Subsection 9.5.1.

See Section 3.4 for protection of raceways and the associated equipment against flooding.

8.3.2.4.4 Control of Compliance with Separation Criteria during Design and Installation

The separation group identification described in Subsection 8.3.2.3 provides for the maintenance of separation in the routing of cables and the connection of control boards and panels. The separation group designation on the cable or raceway is assigned to maintain compatibility with a single line diagram channel designation and other cables or raceways routed. The routing is verified during installation. Color identification of equipment and cabling (discussed in Subsection 8.3.2.3) assist field personnel in this effort.

8.3.2.5 Inspection and Testing

Preoperational tests are conducted to verify proper operation of the dc power systems. The preoperational tests include MOV terminal voltage testing and capacity testing of the batteries, chargers, inverters, and regulating transformers.

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8.3.2.5.1 Class 1E 24-Hour Battery Capacity Testing

Each Class 1E 24-hour battery is tested to verify the capability to provide its load for 24 hours while maintaining the battery terminal voltage above the minimum voltage specified in Table 8.3.2-5. Analysis will be performed based on the design duty cycle, and testing will be performed with loads which envelope the analyzed battery bank design duty cycle. Each battery is connected to a charger maintained at 270±2 V for a period of at least 24 hours prior to the test to assure the battery is fully charged.

8.3.2.5.2 Class 1E 72-Hour Battery Capacity Testing

Each Class 1E 72-hour battery is tested to verify the capability to provide its load for 72 hours while maintaining the battery terminal voltage above the minimum voltage specified in Table 8.3.2-5. Analysis will be performed based on the design duty cycle, and testing will be performed with loads which envelope the analyzed battery bank design duty cycle. Each battery is connected to a charger maintained at 270±2 V for a period of at least 24 hours prior to the test to assure the battery is fully charged.

8.3.2.5.3 Class 1E Spare Battery Capacity Testing

The Class 1E spare battery is tested to the same requirements as the most severe of the six division batteries.

8.3.2.5.4 Class 1E 24-Hour Inverter Capacity Testing

Each Class 1E 24-hour inverter is tested to verify the capability to provide 12 kW while maintaining the output voltage and frequency within the tolerances specified in Table 8.3.2-5. The 12 kW capacity is sufficient to meet the 24-hour inverter loads listed in Tables 8.3.2-1, 8.3.2-2, 8.3.2-3, and 8.3.2-4. The inverter input voltage will be no more than 210 Vdc during the test to represent the conditions at the battery end of life.

8.3.2.5.5 Class 1E 72-Hour Inverter Capacity Testing

Each Class 1E 72-hour inverter is tested to verify the capability to provide 7 kW while maintaining the output voltage and frequency within the tolerances specified in Table 8.3.2-5. The 7 kW capacity is sufficient to meet the 72-hour inverter loads listed in Tables 8.3.2-2 and 8.3.2-3. The inverter input voltage will be no more than 210 Vdc during the test to represent the conditions at the battery end of life.

8.3.2.5.6 Class 1E 24-Hour Charger Capacity Testing

Each Class 1E 24-hour charger is tested to verify the capability to provide 150 A while maintaining the output voltage within the range specified in Table 8.3.2-5. The 150 A is sufficient to meet the 24-hour loads listed in Tables 8.3.2-1, 8.3.2-2, 8.3.2-3, and 8.3.2-4 while maintaining the corresponding battery charged.

8.3.2.5.7 Class 1E 72-Hour Charger Capacity Testing

Each Class 1E 72-hour charger is tested to verify the capability to provide 125 A while maintaining the output voltage within the range specified in Table 8.3.2-5. The 125 A is sufficient to meet the 72-hour loads listed in Tables 8.3.2-2 and 8.3.2-3 while maintaining the corresponding battery charged.

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8.3.2.5.8 Class 1E Regulating Transformer Capacity Testing

Each Class 1E regulating transformer is tested to verify the capability to provide 30 kW while maintaining the output voltage within the tolerance specified in Table 8.3.2-5. The 30 kW capacity is sufficient to meet the inverter loads listed in Tables 8.3.2-1, 8.3.2-2, 8.3.2-3 and 8.3.2-4.

8.3.2.5.9 Class 1E Motor-Operated Valves Terminal Voltage Testing

As-built testing and analysis representing design conditions are used to verify that a voltage is provided from the IDS batteries to the Class 1E motor-operated valve motor terminals that is greater than or equal to the minimum design voltage necessary for the valves to operate. Class 1E motor-operated valves are tested by measuring the voltage during motor starting at both the IDS battery and motor-operated valve motor terminals while each motor-operated valve is stroked. The results are compared in a voltage analysis which verifies that the voltage at the motor-operated valve motor terminals is greater than or equal to the minimum design voltage of each motor-operated valve with an IDS battery terminal voltage of 210 Vdc.

8.3.2.5.10 Non-Class 1E Battery Capacity Testing

Each load group 1, 2, 3, and 4 non-Class 1E battery is tested to verify the capability to provide 850 A for two hours while maintaining the battery terminal voltage above the minimum voltage specified in Table 8.3.2-6. The 850 A is sufficient to meet the loads described in Subsection 8.3.2.1.2. Each battery is connected to a charger maintained at 135±1 V for a period of at least 24 hours prior to the test to assure the battery is fully charged.

8.3.2.5.11 Non-Class 1E Inverter Capacity Testing

Each load group 1, 2, 3, and 4 non-Class 1E inverter is tested to verify the capability to provide 55 kW while maintaining the output voltage and frequency within the tolerances specified in Table 8.3.2-6. The 55 kW capacity is sufficient to meet the loads described in Subsection 8.3.2.1.2.

8.3.2.5.12 Non-Class 1E Charger Capacity Testing

Each load group 1, 2, 3, and 4 non-Class 1E charger is tested to verify the capability to provide 900 A while maintaining the output voltage within the range specified in Table 8.3.2-6. The 900 A is sufficient to meet the loads described in Subsection 8.3.2.1.2 while maintaining the corresponding battery charged.

8.3.3 Combined License Information for Onsite Electrical Power

Combined License applicants referencing the AP1000 certified design will address the design of grounding and lightning protection.

The plant procedures for the following are addressed in Subsections 8.3.1.1.2.4, 8.3.1.1.6 and 8.3.2.1.4.

- Clearing ground fault on the Class 1E dc system
- Checking sulfated battery plates or other anomalous conditions through periodic inspections
- Battery maintenance and surveillance (for battery surveillance requirements, refer to the Technical Specifications, Section 3.8)
- Periodic testing of penetration protective devices

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- Diesel generator operation, inspection, and maintenance in accordance with manufacturers' recommendations.
- Periodic testing on the battery chargers and voltage regulating transformers.

8.3.4 References

1. NEMA MG-1, "Motors and Generators," 1998.

2. IEEE Standard 317, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," 1983.

3. IEEE Standard 422, "Guide for the Design and Installation of Cable Systems in Power Generating Stations," 1986.

4. ICEA Standard Publication P-54-440, "Ampacities of Cables in Open-Top Cable Trays," 1986.

5. NFPA 70, "National Electrical Code (NEC)," 1999.

6. IEEE Standard 485, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," 1997.

7. IEEE Standard 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," 1981.

8. IEEE Standard 308, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," 1991.

9. IEEE Standard 946, "IEEE Recommended Practice for the Design of dc Auxiliary Power Systems for Generating Stations," 1992.

10. IEEE Standard 741, "IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," 1997.

11. IPCEA Standard Publication P-46-426-1962, "Power Cable Ampacities, Volume I - Copper Conductors."

12. IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Vented Lead-Acid Batteries for Stationary Applications," 1995.

13. DeYoung, G. L. et al., "Cable Separation - What Do Industry Programs Show?," IEEE Transactions of Energy Conversion, September 1990, Volume 5, Number 3, pp 585-602.

14. Not used.

15. NUREG/CR-0660, "Enhancement of On-Site Emergency Diesel Generator Reliability," February 1979.

16. IEEE Standard 141, "IEEE Recommended Practice for Electric Power Distribution for Industrial Plants" (IEEE Red Book), 1993.

17. IEEE Standard 242, "IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems" (IEEE Buff Book), 1986.

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18. IEEE Standard 665, "IEEE Guide for Generating Station Grounding," 1995.

19. NFPA 780, "Standard for the Installation of Lightning Protection Systems," 2000.

20. IEEE Standard 1050, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations," 1996

201. Not Used.

202. APP-GW-EOR-006, "IEEE 384 Design Compliance Description."

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Table 8.3.1-1 (Sheet 1 of 5)
Onsite Standby Diesel Generator ZOS MG 02A Nominal Loads

Automatic Loads (Notes 1 and 2)					
Item No.	Time Seq. (sec) (Note 3)	Event or Load Description	Rating (hp/kW)	Operating Load (kW) (Note 4)	
				At Power (Note 10)	Shutdown (Note 10)
1.	0	D/G Start Signal is Initiated	-	-	-
2.		Not used.	-	-	-
3.		Not used.	-	-	-
4.	120	D/G Breaker Closes, Load Sequencer Starts (Note 6)	-	-	-
5.	120	Load Center Transformer EK11 (Note 7)	2500 kVA	7.5	7.5
6.	120	Load Center Transformer EK12 (Note 7)	2500 kVA	7.5	7.5
7.	120	Load Center Transformer EK14 (Note 7)	2500 kVA	7.5	7.5
8.	120	Annex Bldg Lighting Panels (Note 8)	120 kVA	22	22
9.	120	Aux Bldg Lighting Panels (Note 8)	270 kVA	23	23
10.	120	Turbine Bldg Lighting Panels (Note 8)	120 kVA	30.5	30.5
11.	120	D/G Bldg Lighting Panel (Note 8)	30 kVA	8	8
12.	120	D/G 2A AC/OC Radiator Fan	30 hp	25	25
13.	120	Diesel Oil Transfer Module Unit Heater A	4 kW	4	4

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Table 8.3.1-1 (Sheet 2 of 5)
Onsite Standby Diesel Generator ZOS MG 02A Nominal Loads

Automatic Loads (Notes 1 and 2)					
Item No.	Time Seq. (sec)	Event or Load Description	Rating (hp/kW)	Operating Load (kW) (Note 4)	
				At Power (Note 10)	Shutdown (Note 10)
14.	120	Diesel Oil Transfer Module Exhaust Fan A	0.5 hp	0.5	0.5
15.	120	D/G A Jacket Water Radiator Fans	60 hp	50	50
16.	120	Class 1E Div. A Regulating XFMR 1	45 kVA	0	0
17.	120	Class 1E Div. C Regulating XFMR 1	45 kVA	4	4
18.	120	Class 1E Div. A Battery Charger 1	81.8 kVA	6.5	6.5
19.	120	Class 1E Div. C Battery Charger 1	81.8 kVA	4	4
20.	120	Class 1D Div. C Battery Charger 2	81.8 kVA	4.5	4.5
21.	120	Motor-Operated Valves (Note 5)	-	-	-
22.	120	D/G A Fuel Oil Transfer Pump	5 hp	4	4
23.	120	D/G A Bldg Stdby Exhaust Fan 1A	7.5 hp	5.6	5.6
24.	120	D/G A Bldg Stdby Exhaust Fan 2A	7.5 hp	5.6	5.6
25.	120	D/G A Bldg Primary AHU MS 01A Fan	5 hp	4	4
26.	120	D/G A Fuel Oil Cooler Fan	7.5 hp	6.2	6.2
27.	120	D/G Bldg Engine AHU MS 03A Fan	15 hp	11.2	11.2
28.	120	Load Center Transformer EK13 (Note 7)	2500 kVA	7.5	7.5
29.	120	ECS Panel Transformers	330 kVA	50	50
30.	120	D/G A Lube Oil Cooling Motor - Front	0.5 hp	0.5	0.5
31.	120	D/G A Lube Oil Cooling Motor - Rear	0.5 hp	0.5	0.5
32.	120	D/G A Transformer	45 kVA	2	2
33.	120	Air Cooled Chiller #2 Piping Heat Trace	0.5 kW	0.5	0.5
34.	120	Air Cooled Chiller #2 Control and Heat Trace (Notes 11 and 14)	6 kW	6	6
35.	140	Start-up Feed Water Pump A	800 hp	665	0
36.	160	Fuel Oil Day Tank Vault Exhaust Fan A	1 hp	0.8	0.8
37.	160	Diesel Fuel Oil Transfer Heater A	90 kW	90	90
38.	160	Service Water Pump A	500 hp	350	350
39.	180	Service Water Cooling Tower Cell Fan A	200 hp	122	122
40.	200	Component Cooling Water Pump A	800 hp	580	580

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Table 8.3.1-1 (Sheet 3 of 5)
Onsite Standby Diesel Generator ZOS MG 02A Nominal Loads

Automatic Loads (Notes 1 and 2)					
Item No.	Time Seq. (sec)	Event or Load Description	Rating (hp/kW)	Operating Load (kW) (Note 4)	
				At Power (Note 10)	Shutdown (Note 10)
41.	240	Normal Residual Heat Removal Pump A	250 hp	0	207
42.	240	RNS Pump Room Fan A	1.5 hp	0	1.5
43.	240	Annex Bldg Equipment Room Return/Exhaust Fan A (Note 12)	15 hp	12.4	12.4
44.	240	Annex Bldg Equipment Room AHU MS02A Supply Fan (Note 12)	30 hp	25.0	25.0
45.	240	Annex Bldg Swgr Rm AHU MS 05A Supply Fan (Note 12)	25 hp	21.0	21.0
46.	240	Annex Bldg Swgr Rm Ret/Exhaust Fan 06A (Note 12)	15 hp	12.4	12.4
47.	240	Instrument Air Compressor A (Note 15)	250 hp	185	185
48.	240	Instrument Air Dryer Package A	0.008 kW	0.008	0.008
49.	300	Non-1E Battery Charger EDS1-DC-1	204.4 kVA	72	72
50.	300	Non 1E Battery Room A Exhaust Fan	1 hp	0.8	0.8
51.	300	Day Tank Heater Pad A	4 kW	4	4
52.	300	Containment Recirculation Fan A	200 hp	149	149
53.	360	Containment Recirculation Fan D	200 hp	149	149
54.	360	Non-1E Battery Charger EDS3-DC-1	204.4 kVA	60	60
		Total Automatically Sequenced Loads (kW)		**2806.5**	**2350**

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Table 8.3.1-1 (Sheet 4 of 5)
Onsite Standby Diesel Generator ZOS MG 02A Nominal Loads

Manual Loads (Notes 1 and 2)				
Item No.	**Time Seq. (sec)**	**Event or Load Description**	**Rating (hp/kW)**	**Operating Load (kW)**
55.	--	Supplemental Air Filtration System Fan A	15 hp	12.4
56.	--	Supplemental Air Filtration System Electric Heater A	20 kW	20
57.	--	Backup Group 4A Pressurizer Heaters	246 kW	246
58.	--	CRDM Fan 01A	75 hp	62
59.	--	CRDM Fan 01B	75 hp	62
60.	--	Spent Fuel Cooling Pump A	150 hp	124
61.	--	CVS Make-Up Pump A	600 hp	498
62.	--	Non-1E Regulating XFMR EDS1-DT-1	100 kVA	17
63.	--	Non-1E Regulating XFMR EDS3-DT-1	100 kVA	17
64.	--	Main Control Room AHU Supply Fan A (Note 11)	40 hp	34
65.	--	Main Control Room AHU Return Fan A (Note 11)	20 hp	17
66.	--	Div A/C Class 1E Electrical Room AHU Supply Fan A (Note 11)	40 hp	34
67.	--	Div A/C Class 1E Electrical Room Return Fan A (Note 11)	25 hp	21
68.	--	Div. A/C Class 1E Battery Room Exhaust Fan A	1.5 hp	1.2
69.	--	So. Bay Eq. Area AHU MS03A Supply Fan	40 hp	33

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Table 8.3.1-1 (Sheet 5 of 5)
Onsite Standby Diesel Generator ZOS MG 02A Nominal Loads

Manual Loads (Notes 1 and 2)				
Item No.	**Time Seq. (sec)**	**Event or Load Description**	**Rating (hp/kW)**	**Operating Load (kW)**
70.	--	Div B/D Class 1E Electrical Room AHU Supply Fan D (Note 11)	20 hp	17
71.	--	Div B/D Class 1E Electrical Room Return Fan D (Note 11)	10 hp	8
72.	--	Div. B/D Class 1E Battery Room Exhaust Fan D	1.5 hp	1.2
73.	--	Air Cooled Chiller Pump 2 (Note 11)	20 hp	17
74.	--	Air Cooled Chiller 2 (Notes 11 and 14)	444 kW	444
75.	--	CVS Pump Room Fan A (Note 11)	1.5 hp	1.5
		Total Manually Sequenced Loads (kW)		**1687.3**

Notes:

1. Loads listed are for diesel generator ZOS MG 02A.
2. Loads identified in the first portion of the table (AUTOMATIC LOADS) will be loaded without operator action. Loads identified in the second portion of the table (MANUAL LOADS) will be energized at operator discretion based on system needs. Automatic loads may not be started until there is a system need. Not all manually sequenced loads will be operated simultaneously.
3. Time Sequence is counted from the time a diesel generator receives the start signal and represents the end of the previous load block interval and the beginning of the following load block interval.
4. The "Operating Load" column shows the load input power requirement from the diesel generator.
5. Motor-operated valves (MOVs) pertaining to various systems will be energized in the first load block interval. Normally the MOV power requirement is for a very short duration (a few seconds); hence, the MOV load will not affect the diesel generator capacity rating.
6. On receipt of the diesel generator start signal, the engine accelerates to a set idle speed. The engine operates at the idle speed for a time to allow bearing oil pressure buildup, proper lubrication of the moving parts, and engine warmup. The engine ramps up to the rated operating speed and is ready to accept electrical loads within 120 seconds.
7. On restoring the power supply to the diesel backed bus ES1 by closing the diesel generator incoming breaker, the associated unit substation ECS EK 11, 12, 13, and 14 load center transformers are energized in the first load block interval. The transformers draw magnetizing current and the no load losses (approx. 0.3 percent of the rating) from the bus.
8. Only a part of the building lighting load is automatically connected to the diesel generator bus. The remaining lighting load is connected via manual action at the operator's discretion.
9. Not used.
10. The ‘At Power’ loads are those loads that would be automatically sequenced on the diesel generator following a loss of offsite power and reactor trip from power; i.e., reactor coolant pressure above the residual heat removal system operating pressure. The ‘Shutdown’ loads are those loads that would be automatically sequenced on the diesel generator following a loss of offsite power during a plant shutdown; i.e., reactor coolant pressure below the residual heat removal system operating pressure and the RNS isolation valves open.
11. Air cooled chiller VWS MS 03 is automatically loaded on diesel generator ZOS MG 02B along with the VAS and VBS fans associated with the cooling coils served by this chiller. The redundant air cooled chiller VWS MS 02 and its associated VAS and VBS fans can be manually loaded on diesel generator ZOS MG 02A in case of failures of VWS MS 03 or ZOS MG 02B.
12. Annex building ventilation fans are automatically loaded on diesel generator ZOS MG 02A. The redundant fans can be manually loaded on diesel generator ZOS MG 02B in case of diesel generator or fan failures.
13. Not used.
14. To support operation of Air Cooled Chiller VWS-MS-02 at very cold conditions, the breaker to the package is loaded automatically in order to energize the skid mounted heat tracing and motor-heaters for the package.The chiller compressor will not be started. Start of the chiller compressor is a manual operator action. The skid control power and heat trace load for the chiller is 6 kW and is included in the Auto Load section of the table and is thus subtracted from the 450 kW total chiller load in the Manual Load section of the table.
15. Feeder breaker for instrument air compressor package to be closed in the first load block interval and PLS enable/disable command used to delay actual start of compressor to a later load block interval to allow instrument air compressor package programmable logic controller (PLC) time to initialize.

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Table 8.3.1-2 (Sheet 1 of 5)
Onsite Standby Diesel Generator ZOS MG 02B Nominal Loads

Automatic Loads (Notes 1 and 2)					
Item No.	Time Seq. (sec) (Note 3)	Event or Load Description	Rating (hp/kW)	Operating Load (kW)	
				At Power (Note 10)	Shutdown (Note 10)
1.	0	D/G Start Signal is Initiated	-	-	-
2.		Not used.	-	-	-
3.		Not used.	-	-	-
4.	120	D/G Breaker Closes, Load Sequencer Starts (Note 6)	-	-	-
5.	120	Load Center Transformer EK21 (Note 7)	2500 kVA	7.5	7.5
6.	120	Load Center Transformer EK22 (Note 7)	2500 kVA	7.5	7.5
7.	120	Load Center Transformer EK24 (Note 7)	2500 kVA	7.5	7.5
8.	120	Annex Bldg Lighting Panels (Note 8)	120 kVA	33	33
9.	120	Radwaste Bldg Lighting Panel (Note 8)	30 kVA	8.5	8.5
10.	120	Aux Bldg Lighting Panels (Note 8)	270 kVA	20.5	20.5
11.	120	Turbine Bldg Lighting Panels (Note 8)	120 kVA	27	27
12.	120	D/G Bldg Lighting Panel (Note 8)	30 kVA	8	8
13.	120	D/G 2B AC/OC Radiator Fan	30 hp	25	25
14.	120	Diesel Oil Transfer Module Unit Heater B	4 kW	4	4
15.	120	Diesel Oil Transfer Module Exhaust Fan B	0.5 hp	0.5	0.5
16.	120	D/G B Jacket Water Radiator Fans	60 hp	50	50

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Table 8.3.1-2 (Sheet 2 of 5)
Onsite Standby Diesel Generator ZOS MG 02B Nominal Loads

Automatic Loads (Notes 1 and 2)					
Item No.	Time Seq. (sec)	Event or Load Description	Rating (hp/kW)	Operating Load (kW)	
				At Power (Note 10)	Shutdown (Note 10)
17.	120	Class 1E Div. B Regulating XFMR 1	45 kVA	4	4
18.	120	Class 1E Div. D Regulating XFMR 1	45 kVA	0	0
19.	120	Class 1E Div. B Battery Charger 1	81.8 kVA	7.3	7.3
20.	120	Class 1E Div. B Battery Charger 2	81.8 kVA	4.5	4.5
21.	120	Class 1E Div. D Battery Charger 1	81.8 kVA	8.2	8.2
22.	120	Motor-Operated Valves (Note 5)	-	-	-
23.	120	D/G B Fuel Oil Transfer Pump	5 hp	4	4
24.	120	D/G B Bldg Stdby Exhaust Fan 1B	7.5 hp	5.6	5.6
25.	120	D/G B Bldg Stdby Exhaust Fan 2B	7.5 hp	5.6	5.6
26.	120	D/G B Bldg. Primary AHU MS 01B Fan	5 hp	4	4
27.	120	D/G B Fuel Oil Cooler Fan	7.5 hp	6.2	6.2
28.	120	D/G Bldg Engine AHU MS 03B Fan	15 hp	11.2	11.2
29.	120	Load Center Transformer EK23 (Note 7)	2500 kVA	7.5	7.5
30.	120	ECS Panel Transformers	420 kVA	60	60
31.	120	D/G B Lube Oil Cooling Motor - Front	0.5 hp	0.5	0.5
32.	120	D/G B Lube Oil Cooling Motor - Rear	0.5 hp	0.5	0.5
33.	120	D/G B Transformer	45 kVA	2	2
34.	120	Air Cooled Chiller #3 Piping Heat Trace	0.5 kW	0.5	0.5
35.	140	Start-up Feed Water Pump B	800 hp	665	0
36.	160	Fuel Oil Day Tank Vault Exhaust Fan B	1 hp	0.8	0.8
37.	160	Diesel Fuel Oil Transfer Heater B	90 kW	90	90
38.	160	Service Water Pump B	500 hp	350	350
39.	180	Service Water Cooling Tower Cell Fan B	200 hp	122	122
40.	180	Main Control Room AHU Supply Fan B	40 hp	34	34
41.	180	Main Control Room AHU Return Fan B	20 hp	17	17
42.	180	Div. B/D Class 1E Electrical Room AHU Supply Fan B	20 hp	17	17

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Table 8.3.1-2 (Sheet 3 of 5)
Onsite Standby Diesel Generator ZOS MG 02B Nominal Loads

Automatic Loads (Notes 1 and 2)					
Item No.	Time Seq. (sec)	Event or Load Description	Rating (hp/kW)	Operating Load (kW)	
				At Power (Note 10)	Shutdown (Note 10)
43.	180	Div B/D Class 1E Electrical Room Return Fan B	10 hp	8	8
44.	180	Div A/C Class 1E Electrical Room AHU Supply Fan C	40 hp	34	34
45.	180	Div A/C Class 1E Electrical Room Return Fan C	25 hp	21	21
46.	180	Air Cooled Chiller Pump 3	20 hp	17	17
47.	200	Component Cooling Water Pump B	800 hp	580	580
48.	220	Air Cooled Chiller 3	450 kW	450	450
49.	240	CVS Pump Room Fan B	1.5 hp	1.5	1.5
50.	240	Instrument Air Compressor B (Note 14)	250 hp	185	185
51.	240	Instrument Air Dryer Package B	0.008 kW	0.008	0.008
52.	300	Normal Residual Heat Removal Pump B	250 hp	0	207
53.	300	RNS Pump Room Fan B	1.5 hp	0	1.5
54.	300	Non-1E Battery Charger EDS2-DC-1	204.4 kVA	80	80
55.	300	Non-1E Battery Room B Exhaust Fan 09B	1 hp	0.8	0.8
56.	300	Day Tank Heater Pad B	4 kW	4	4
57.	360	Containment Recirculation Fan B	200 hp	149	149
58.	360	Containment Recirculation Fan C	200 hp	149	149
59.	360	Non-1E Battery Charger EDS4-DC-1	204.4 kVA	50	50
60.	420	Div. B/D Class 1E Battery Room Exhaust Fan B	1.5 hp	1.5	1.5
61.	420	Div. A/C Class 1E Battery Room Exhaust Fan C	1.5 hp	1.4	1.4
		Total Automatically Sequenced Loads (kW)		**2908.6**	**2452.1**

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Table 8.3.1-2 (Sheet 4 of 5)
Onsite Standby Diesel Generator ZOS MG 02B Nominal Loads

Manual Loads (Notes 1 and 2)				
Item No.	**Time Seq. (sec)**	**Event or Load Description**	**Rating (hp/kW)**	**Operating Load (kW)**
62.	--	Supplemental Air Filtration System Fan B	15 hp	12.4
63.	--	Supplemental Air Filtration System Electric Heater B	20 kW	20
64.	--	Backup Group 4B Pressurizer Heaters	246 kW	246
65.	--	CRDM Fan 01C	75 hp	62
66.	--	CRDM Fan 01D	75 hp	62
67.	--	Spent Fuel Cooling Pump B	150 hp	124
68.	--	CVS Make-Up Pump B	600 hp	498
69.	--	Non-1E Regulating XFMR EDS2-DT-1	100 kVA	17
70.	–	Non-1E Regulating XFMR EDS4-DT-1	100 kVA	17
71.	--	Annex Bldg Equipment Room Return/Exhaust Fan B	15 hp	12.4
72.	--	Annex Bldg Equipment Room AHU MS02B Supply Fan	30 hp	25.0
73.	--	Annex Bldg Swgr Rm AHU MS 05B Supply Fan	25 hp	21.0
74.	--	Annex Bldg Swgr Rm Ret/Exhaust Fan 06B	15 hp	12.4
75.	--	So. Bay Eq. Area AHU MS03B Supply Fan	40 hp	33
		Total Manually Sequenced Loads (kW)		**1162.2**

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Table 8.3.1-2 (Sheet 5 of 5)
Onsite Standby Diesel Generator ZOS MG 02B Nominal Loads

Notes:

1. Loads listed are for diesel generator ZOS MG02B.
2. Loads identified in the first portion of the table (AUTOMATIC LOADS) will be loaded without operator action. Loads identified in the second portion of the table (MANUAL LOADS) will be energized at operator discretion based on system needs. Automatic loads may not be started until there is a system need. Not all manually sequenced loads will be operated simultaneously.
3. Time Sequence is counted from the time a diesel generator receives the start signal and represents the end of the previous load block interval and the beginning of the following load block interval.
4. The "Operating Load" column shows the load input power requirement from diesel generator.
5. Motor-operated valves (MOVs) pertaining to various systems will be energized in the first load block interval. Normally the MOV power requirement is for a very short duration (few seconds); hence the MOV load will not affect the diesel generator capacity rating.
6. On receipt of the diesel generator start signal, the engine accelerates to a set idle speed. Engine operates at the idle speed for a time period to allow bearing oil pressure build up, proper lubrication of the moving parts, and engine warmup. The engine ramps up to the rated operating speed and is ready to accept electrical loads within 120 seconds.
7. On restoring the power supply to the diesel backed bus ES2 by closing diesel generator incoming breaker, the associated unit substation ECS EK 21, 22, 23, and 24 load center transformers are energized in the first load block interval. The transformers draw magnetizing current and the no load losses (approx. 0.3 percent of the rating) from the bus.
8. Only a part of the building lighting load is automatically connected to the diesel generator bus. The remaining lighting load is connected via manual action at the operator's discretion.
9. Not used.
10. The ‘At Power’ loads are those loads that would be automatically sequenced on the diesel generator following a loss of offsite power and reactor trip from power; i.e., reactor coolant pressure above the residual heat removal system operating pressure. The ‘Shutdown’ loads are those loads that would be automatically sequenced on the diesel generator following a loss of offsite power during a plant shutdown; i.e., reactor coolant pressure below the residual heat removal system operating pressure and the RNS isolation valves open.
11. Air cooled chiller VWS MS 03 is automatically loaded on diesel generator ZOS MG 02B along with the VAS and VBS fans associated with the cooling coils served by this chiller. The redundant air cooled chiller VWS MS 02 and its associated VAS and VBS fans can be manually loaded on diesel generator ZOS MG 02A in case of failures of VWS MS 03 or ZOS MG 02B.
12. Annex building ventilation fans are automatically loaded on diesel generator ZOS MG 02A. The redundant fans can be manually loaded on diesel generator ZOS MG 02B in case of diesel generator or fan failures.
13. Not used.
14. Feeder breaker for instrument air compressor package to be closed in the first load block interval and PLS enable/disable command used to delay actual start of compressor to a later load block interval to allow instrument air compressor package programmable logic controller (PLC) time to initialize.
15. To support operation of Air Cooled Chiller VWS-MS-03 at very cold conditions, the breaker to the package is closed automatically at Load Step 0 in order to energize the skid mounted heat tracing and motor-heaters for the package. The chiller compressor is not started until Load Step 5.

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Table 8.3.1-3
Component Data - Main AC Power System
(Nominal Values)

1.	**Main Stepup Transformer**	
		3 single phase, FOA, 65°C rise, liquid filled
2.	**Unit Auxiliary Transformers (UAT 2A/2B)**	
		3 phase, 3 winding H = 76 MVA, OA, 65°C X = 38 MVA, OA, 65°C Y = 38 MVA, OA, 65°C
	Unit Auxiliary Transformer (UAT 2C)	
		3 phase, 2 winding 33 MVA, OA, 65°C
	Reserve Auxiliary Transformers (RAT)	
		3 phase, 3 winding H = 76 MVA, OA, 65°C X = 38 MVA, OA, 65°C Y = 38 MVA, OA, 65°C
3.	**6.9 kV Switchgear**	
		medium voltage metal-clad switchgear Interrupting current rating – 63kA vacuum-type circuit breaker
4.	**480 V Load Centers**	
	Transformers - Indoor, Air-Cooled Ventilated Dry-Type, Fire Retardant:	2500 kVA, AA 3 phase, 60 Hz 6900 - 480 V
		3000/4000 kVA, AA/FA (Load Centers 31 and 41) 3 phase, 60 Hz 6900 - 480 V
		1500 kVA, AA (Load Center 71) 3 phase, 60 Hz 6900 - 480 V
	Main Bus Ampacity	4000 amperes continuous
		5000 amperes continuous (Load Centers 31 and 41)
		3200 amperes continuous (Load Center 71)
	480V Breakers	metal enclosed draw-out circuit breaker or motor-starter (contactor) 65,000 A RMS symmetrical interrupting rating
5.	**480 V Motor Control Centers**	
	Horizontal Bus	800 A and 1200 A continuous rating 65,000 A RMS symmetrical bracing
	Vertical Bus	300 A continuous rating
		65,000 A RMS symmetrical bracing
	Breakers (molded case)	65,000 A RMS symmetrical interrupting rating

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Table 8.3.1-4
Post-72 Hours Nominal Load Requirements

Item No.	Description of Loads	Ancillary AC Generator 1	Ancillary AC Generator 2
		Load (kW)	Load (kW)
1.	Post-Accident Monitoring (PAM) Emergency and Panel Lighting (Division B) in Main Control Room and ancillary fans		6.5
2.	Post-Accident Monitoring (PAM) Emergency and Panel Lighting (Division C) in Main Control Room and ancillary fans	6.5	
3[(a)]	PCS Recirculation Pumps	23.6	23.6
4.	Ancillary Generator Room Lights	0.5	0.5
5.	Ancillary Generator Fuel Tank Heater	1.5 kW	1.5 kW
	Total	32.1 kW	32.1 kW

Note:

a. There are two PCS pumps; however, only one pump will be operating at any point in time on each generator. In case of fire fighting, two pumps (one on each generator) may be used.

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Table 8.3.1-5
Indication and Alarm Points
Standby Diesel Generators

Parameter	Indication		Alarm	
	Control Room	Local	Control Room	Local
Lube Oil Pressure Low (PT014A/B, PT015A/B)	No	Yes	Yes	Yes
Lube Oil Temperature High (TE029A/B)	No	Yes	Yes	Yes
Lube Oil Sump Level Low (LS02A/B)	No	Yes	Yes	Yes
Cooling Water Temperature High (TE005A/B, TE003A/B, TE002A/B)	No	Yes	Yes	Yes
Cooling Water Pressure Low (PT04A/B, PT05A/B, PT06A/B)	No	Yes	Yes	Yes
DG Starting Air Pressure Low (PT103A/B)	Yes	Yes	Yes	Yes

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Table 8.3.2-1
250V DC Class 1E Division A Battery Loads

Load Description	Power Required (kW)	
	Momentary	**Continuous**
Bus IDSA DS 1 (24 hr Battery Bank)		
Inverter		
Protection and Safety Monitoring System	0.6	2.6
Emergency Lighting	0	0.1
Containment High Range Monitor	0	0.3
Subtotal	0.6	3.0
250 Vdc Panel		
Reactor Trip Swgr & Solenoid Valves	5	0.9
250 Vdc MCC		
Motor-operated Valves	356	
Total	362	3.9

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Table 8.3.2-2
250V DC Class 1E Division B Battery Loads

Load Description	Power Required (kW)	
	Momentary	**Continuous**
Bus IDSB DS 1 (24 hr Battery Bank)		
Inverter		
Protection and Safety Monitoring System	0	1.2
Emergency Lighting and Panel Lighting	0	0.2
Subtotal	0	1.4
250 Vdc Panel		
Reactor Trip Swgr, RCP Trip & Solenoid Valves	11	3.5
250 Vdc MCC		
Motor-operated Valves	520	
Total	531	4.9
BUS IDSB DS 2 (72 hr Battery Bank)		
Inverter		
Protection and Safety Monitoring System	0.6	1.9
Emergency Lighting and Panel Lighting	0	0.3
Containment High Range Monitor	0	0.3
MCR Supply Duct Radiation Monitor	0.9	0.3
Total	1.5	2.8

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Table 8.3.2-3
250V DC Class 1E Division C Battery Loads

Load Description	Power Required (kW)	
	Momentary	**Continuous**
Bus IDSC DS 1 (24 hr Battery Bank)		
Inverter		
Protection and Safety Monitoring System	0	0.7
Emergency Lighting and Panel Lighting	0	0.2
Subtotal	0	0.9
250 Vdc Panel		
Reactor Trip Swgr, RCP Trip & Solenoid Valves	11	1.4
250 Vdc MCC		
Motor-operated Valves	293	
Total	304	2.3
BUS IDSC DS 2 (72 hr Battery Bank)		
Inverter		
Protection and Safety Monitoring System	0.6	1.9
Emergency Lighting and Panel Lighting	0	0.3
Containment High Range Monitor	0	0.3
MCR Supply Duct Radiation Monitor	0.9	0.3
Total	1.5	2.8

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Table 8.3.2-4
250V DC Class 1E Division D Battery Loads

Load Description	Power Required (kW)	
	Momentary	**Continuous**
Bus IDSD DS 1 (24 hr Battery Bank)		
Inverter		
Protection and Safety Monitoring System	0.6	2.9
Emergency Lighting	0	0.4
Containment High Range Monitor	0	0.3
Subtotal	0.6	3.6
250 Vdc Panel		
Reactor Trip Swgr & Solenoid Valves	5	2.3
250 Vdc MCC		
Motor-operated Valves	481	
Total	487	5.9

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Table 8.3.2-5
Component Data - Class 1E DC System
(Nominal Values)

a.	**Battery Bank** 250 Vdc (2 - 125 Vdc) 60 lead calcium cells, 2400 Ah. (8 hrs to 1.75 V per cell @ 77°F).
b.	**Charger** AC input - 480 V, 3-phase, 60 Hz; dc output - 250 Vdc, 200 A continuous; float voltage 2.20 to 2.25 V/cell; equalizing charge voltage 2.33 V/cell.
c.	**Switchboard and Fused Transfer Switch Box** Main bus 600 A (72-hr) and 1600 A (24-hr) continuous, 40,000 A short circuit bracing; fuse disconnect switch 40,000 A interrupting rating, continuous ratings 600 A and 1600 A.
d.	**Spare Fused Transfer Switch Box** Main bus 1600 A continuous, 40,000 A short circuit bracing; fuse disconnect switch, 40,000 A interrupting rating, continuous rating 1600 A.
e.	**Motor Control Center** Main bus 800 A continuous, vertical bus 800 A continuous, 40,000 A short circuit bracing.
f.	**Spare Battery Bank** 2-125V dc 60 lead calcium cells, 2400 Ah. (8 hrs to 1.75 V per cell @ 77°F).
g.	**Spare Charger** AC input - 480 V, 3-phase, 60 Hz; dc output - 250 Vdc, 200 A continuous; float voltage 2.20 to 2.25 V/cell; equalizing charge voltage - 2.33 V/cell.
h.	**Uninterruptible Power Supply (UPS)** i. Inverter 15 kVA with 250 Vdc input and 208Y/120 Vac, 3-phase, 4-wire, 60 Hz output; ac output voltage regulation of ±2% steady state; output frequency variation within 0.5% of nominal 60 Hz. ii. Voltage Regulating Transformer 45 kVA, 480 V - 208Y/120V, 3-phase, 4-wire.

Note:
Refer to Figures 8.3.2-1 and 8.3.2-2 for the system component configuration.

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Table 8.3.2-6 (Sheet 1 of 3)
Component Data - Non-Class 1E DC System EDS1 – EDS4
(Nominal Values)

a.	**Battery Bank** 125 Vdc 60 lead calcium cells, minimum 3500 Ah. (8 hrs to 1.75 V per cell @ 77°F).
b.	**Charger** AC input - 480 V, 3-phase, 60 Hz; dc output - 125 Vdc, 1000 A continuous; float voltage - 2.20 to 2.25 V/cell; equalizing charge voltage - 2.33 V/cell.
c.	**Switchgear** Main bus 3000 A continuous, 40,000 A short circuit bracing; breaker 3000A frame size.
d.	**Uninterruptible Power Supply (UPS)** i. Inverter 75 kVA with 125 Vdc input and 208 Y/120 Vac, 3-phase, 4-wire, 60 Hz output; ac output voltage regulation of ±2% steady state; output frequency variation within 0.5% of nominal 60 Hz. ii. Voltage Regulating Transformer 100 kVA, 480 V - 208 Y/120 V, 3-phase, 4-wire.

Note:
Refer to Figure 8.3.2-3 for the system component configuration.

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Table 8.3.2-6 (Sheet 2 of 3)
Component Data - Non-Class 1E DC System EDS5
(Nominal Values)

a.	**Battery Bank** 250 Vdc (2 - 125 Vdc) 60 lead calcium cells, minimum 3500 Ah. (8 hrs to 1.75 V per cell @ 77°F).
b.	**Charger** AC input - 480 V, 3-phase, 60 Hz; dc output - 250 Vdc, 200 A continuous; float voltage - 2.20 to 2.25 V/cell; equalizing charge voltage - 2.33 V/cell.
c.	**Switchgear** Main bus 3000 A continuous, 40,000 A short circuit bracing; breaker 3000 A frame size.

Note:
Refer to Figure 8.3.2-3 for the system component configuration.

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Table 8.3.2-6 (Sheet 3 of 3)
Component Data - Non-Class 1E DC System EDSS
(Nominal Values)

a.	**Spare Battery Bank** 250 Vdc (2 - 125 Vdc) 60 lead calcium cells, minimum 3500 Ah. (8 hrs to 1.75 V per cell @ 77°F).
b.	**Spare Charger - 1** AC input - 480 V, 3-phase, 60 Hz; dc output - 125 Vdc, 1000 A continuous; float voltage - 2.20 to 2.25 V/cell; equalizing charge voltage - 2.33 V/cell.
c.	**Spare Charger - 2** AC input - 480 V, 3-phase, 60 Hz; dc output - 125 Vdc, 1000 A continuous; float voltage - 2.20 to 2.25 V/cell; equalizing charge voltage - 2.33 V/cell.

Note:
Refer to Figure 8.3.2-3 for the system component configuration.

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Table 8.3.2-7 (Sheet 1 of 5)
Class 1E 250V DC and Class 1E Uninterruptible Power Supplies
Failure Modes and Effects Analysis

Item No.	Description of Components	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
1.	Battery Charger Division A, IDSA DC 1 Division B, IDSB DC 1, 2 Division C, IDSC DC 1, 2 Division D, IDSD DC 1	Provide IEEE 384 isolation between the non-1E AC power supply and the IDS.	A,B,C	Electrical fault propagation	Failure alarms for downstream divisional equipment annunciated in MCR	None; Other divisions available	Failure of only one div. chgr. falls into single failure criteria and the other three div. are still available.
		Provide ac input undervoltage signals to the protection and safety monitoring system. (24-hr charger only)	A,B,C	No indication	Undervoltage detection by only one of the two sensors connected to the battery charger	None; Other division(s) available to complete 2/4 div. voting logic	1/div. required for containment recirc., ADS actuation. 2/div. required for MCR isolation, boron dilution block.
			A,B	Spurious undervoltage signal	ADS and IRWST Injection unblocked alarm annunciated in MCR	None; 2/4 div. voting logic prevents spurious ESF. Spare battery charger available for connection.	ADS and IRWST injection block is described in Subsection 7.3.1.2.4.1.
2.	Battery Division A, IDSA DB 1A,1B Division B, IDSB DB 1A,1B,2A,2B Division C, IDSC DB 1A,1B,2A,2B Division D, IDSD DB 1A,1B	Provide dc power for 24 and 72 hours without battery charger (Item 1).	A,B	No output or low voltage	Battery monitor provides annunciation in main control room; switchboard failure alarm in main control room for ground detection and bus undervoltage.	None; Other divisions available. Spare battery available for connection.	Power still available with a single ground. Loss of entire battery function is single failure and the other divisions are available.
			C	No output or low voltage	Same as above.	None; Other divisions available; spare battery available.	

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Table 8.3.2-7 (Sheet 2 of 5)
Class 1E 250V DC and Class 1E Uninterruptible Power Supplies
Failure Modes and Effects Analysis

Item No.	Description of Components	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
3.	Fused transfer switch box Division A, IDSA DF 1 Division B, IDSB DF 1,2 Division C, IDSC DF 1,2 Division D, IDSD DF 1	Provide circuit continuity between Item 2 battery and Item 4 switchboard.	A,B,C	Inadvertent opening (blown fuse)	Switchboard failure alarm in main control room for ground detection and bus undervoltage.	None; Other divisions available.	
4.	250V DC Switchboard Division A, IDSA DS 1 Division B, IDSB DS 1,2 Division C, IDSC DS 1,2 Division D, IDSD DS 1	Distribute power via fusible disconnects to loads from chargers and battery.	A,B,C	Bus ground fault	Switchboard failure alarm in main control room for ground detection and bus undervoltage.	None; Other divisions available.	
5.	Fusible disconnect Division A, for Charger 1 Division B, for Charger 1,2 Division C, for Charger 1,2 Division D, for Charger 1	Provide circuit continuity between Item 1 and 4.	A,B	Inadvertent opening (blown fuse)	Alarm in main control room for charger failure (dc no charge).	None; Battery can provide power for 24 and 72 hours without chargers. Other divisions available.	
			C	Inadvertent opening (blown fuse)	Same as above.	None; Battery can provide power for 24 and 72 hours without chargers.	
6.	Fusible disconnect Division A, for Inverter 1 Division B, for Inverter 1,2 Division C, for Inverter 1,2 Division D, for Inverter 1	Provide circuit continuity between Item 4 switchboard and Item 9 inverters.	A,B,C	Inadvertent opening (blown fuse)	Inverter trouble alarm in main control room for loss of dc input, loss of ac output, input, output and backup power supply breaker open.	None; System safety function can be met with loss of one division.	

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Table 8.3.2-7 (Sheet 3 of 5)
Class 1E 250V DC and Class 1E Uninterruptible Power Supplies
Failure Modes and Effects Analysis

Item No.	Description of Components	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
7.	Fusible disconnect for DC MCC Division A Division B Division C Division D	Provide circuit continuity between Item 4 switchboard and Item 13 DC MCC.	A,B,C	Inadvertent opening (blown fuse)	DC MCC trouble alarm in main control room for bus undervoltage.	None; Other divisions available.	
8.	Fusible disconnect for DC dist panel Division A Division B Division C Division D	Provide circuit continuity between Item 4 switchboard and Item 14 dc panel.	A,B,C	Inadvertent opening (blown fuse)	DC dist. panel trouble alarm in main control room for bus undervoltage.	None; Other divisions available.	
9.	Inverter Division A, IDSA DU 1 Division B, IDSB DU 1,2 Division C, IDSC DU 1,2 Division D, IDSD DU 1	Convert 250V DC to 208Y/120V AC and provide 120V AC power.	A,B	No output	Alarm in main control room for common UPS trouble, for loss of dc input, loss of ac output; input and backup power supply breakers open.	None; Static switching function transfers to ac input on loss of inverter output function.	The UPS function is located in the inverter cabinet and switches power input between the inverter and the VRT.
			C	No output	Alarm in main control room for common UPS trouble, for loss of dc input, loss of ac output; input and backup power supply breakers open.	None; System safety function can be met with loss of one division.	Backup VRT is unavailable during station blackout.
10.	Voltage regulating transformer Division A, IDSA DT 1 Division B, IDSB DT 1 Division C, IDSC DT 1 Division D, IDSD DT 1	Provide IEEE 384 isolation between the non-1E AC power supply and AC distribution panel (Item 11).	A,B	Electrical fault propagation	Failure alarms for downstream divisional equipment annunciated in MCR.	None; Other divisions available.	Failure of only one div. VRT falls into single failure criteria; the other divisions are available.

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Table 8.3.2-7 (Sheet 4 of 5)
Class 1E 250V DC and Class 1E Uninterruptible Power Supplies
Failure Modes and Effects Analysis

Item No.	Description of Components	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
11.	208Y/120V AC distr. panel Division A, IDSA EA 1 Division B, IDSB EA 1,3 Division C, IDSC EA 1,3 Division D, IDSD EA 1	Distribute power via breakers to loads	A,B,C	Ground and bus fault	Alarm in main control room for undervoltage.	None; System safety function can be met with loss of one division.	
12.	Class 1E Fuse Panel: Division A, IDSA EA 4 Division B, IDSB EA 4,5 Division C, IDSC EA 4,5 Division D, IDSD EA 4	Provide IEEE 384 isolation between the AC distribution panel (Item 11) and the associated load.	A,B,C	Electrical fault propagation	Failure alarms for downstream divisional equipment annunciated in MCR.	None; Other divisions available.	
13.	Class 1E Fuse Panel: Division B, IDSB EA 6 Division C, IDSC EA 6	Provide IEEE 384 isolation between the AC distribution panel EA-2 and the associated load.	A,B,C	Electrical fault propagation	Failure alarms for downstream divisional equipment annunciated in MCR.	None; Other divisions available.	
14.	DC MCC DIV. A, IDSA DK 1 DIV. B, IDSB DK 1 DIV. C, IDSC DK 1 DIV. D, IDSD DK 1	Distribute power via fusible disconnect to loads.	A,B,C	Ground and bus fault	MCC trouble alarm per MCC in main control room for bus undervoltage and ground detection.	None; Other divisions available.	Power still available with a single ground.
15.	DC Distr. Panel Div. A, IDSA DD1 Div. B, IDSB DD1 Div. C, IDSC DD1 Div. D, IDSD DD1	Distribute power via fusible disconnect to loads.	A,B,C	Ground and bus fault	Panel trouble alarm per panel in main control room for bus undervoltage and ground detection.	None; Other divisions available.	Power still available with a single ground.
16.	Battery monitor fuse panel: Division A, IDSA EA 5 Division B, IDSB EA 7,8 Division C, IDSC EA 7,8 Division D, IDSD EA 5 Spare, IDSS EA 1	Provide IEEE 384 isolation between the battery (Item 2) and the battery monitor.	A,B	Electrical fault propagation	Failure alarms for associated divisional equipment annunciated in MCR.	None; Other divisions available.	

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Table 8.3.2-7 (Sheet 5 of 5)
Class 1E 250V DC and Class 1E Uninterruptible Power Supplies
Failure Modes and Effects Analysis

Item No.	Description of Components	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks

- Plant operating modes are represented as follows:

  A – Normal or preferred power available.
  B – Loss of normal power and loss of preferred power and onsite standby diesel generator available.
  C – Blackout (loss of all ac systems, except 208Y/120-V AC UPS system).

  System success criteria are as follows:

  250-V DC System – Three out of four (Division A, B, C or D) divisions required.
  208Y/120-V AC UPS System – Three out of four divisions required.

- The failure of any one fusible disconnect or opening of one circuit breaker under a fault condition results in only the loss of the associated division. The other redundant divisions still remain available.

- This table does not address interfaces between the IDS and non-Class 1E instrumentation and control systems. These interfaces include those signals that provide bus parameters or switch status to the non-Class 1E plant control system. Interfaces between the IDS and non-Class 1E instrumentation and control systems are protected by IEEE 384 qualified isolation devices. In addition, the 24 hour battery banks of IDS Division A, Division B, and Division C are isolated from the non-Class 1E plant control system by current-to-current transmitters. These isolation devices provide IEEE 384 isolation between the non-Class 1E plant control system and the IDS. If one of the isolation devices fail to prevent an electrical fault propagation from the plant control system to the IDS, a faulted channel will be provided to the plant control system and will be indicated in the MCR. Should this electrical fault cause a failure of the downstream divisional equipment, then failure alarms are annunciated in the MCR. This has no effect on the IDS safety function capability because the other IDS divisions are available.

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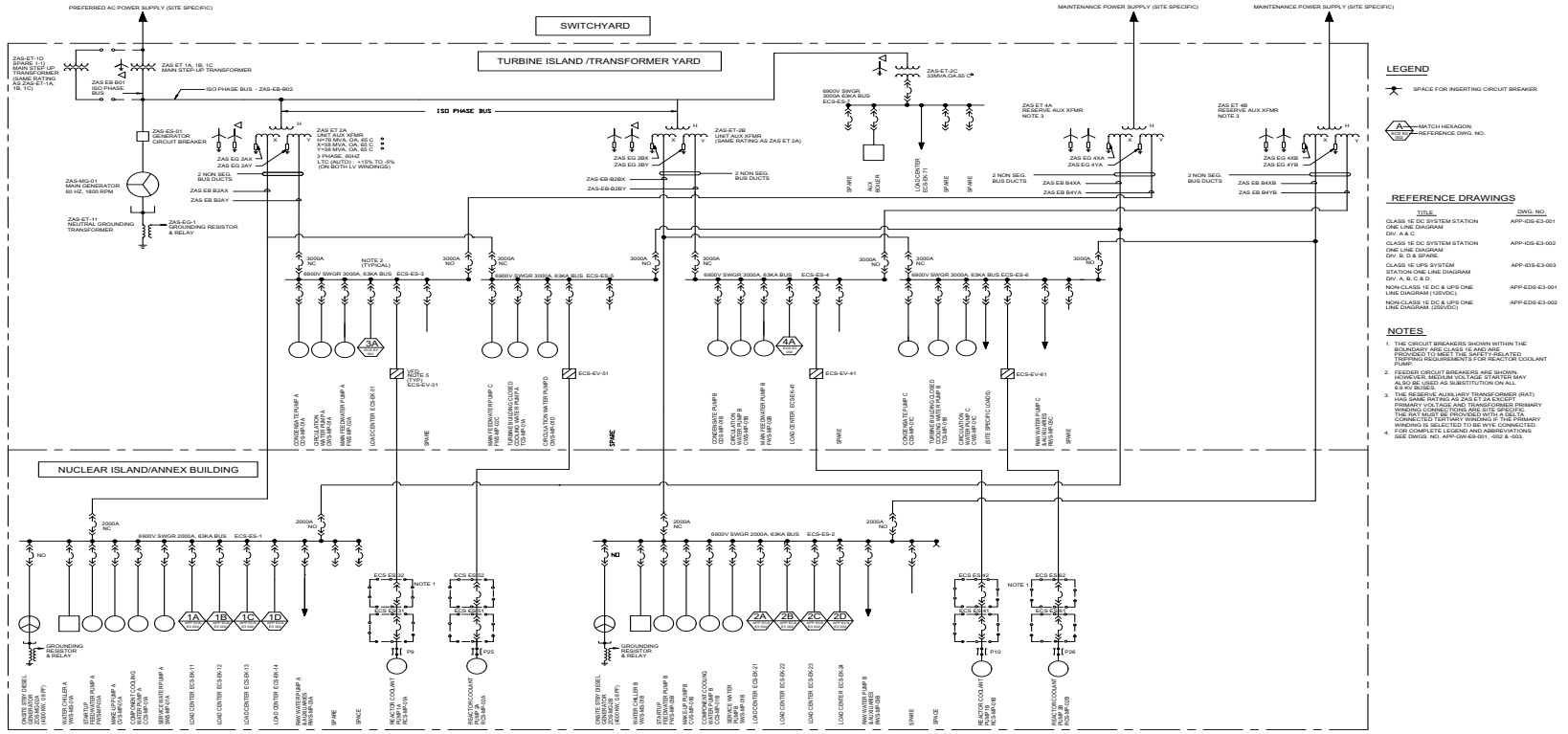

Figure 8.3.1-1
AC Power Station One Line Diagram

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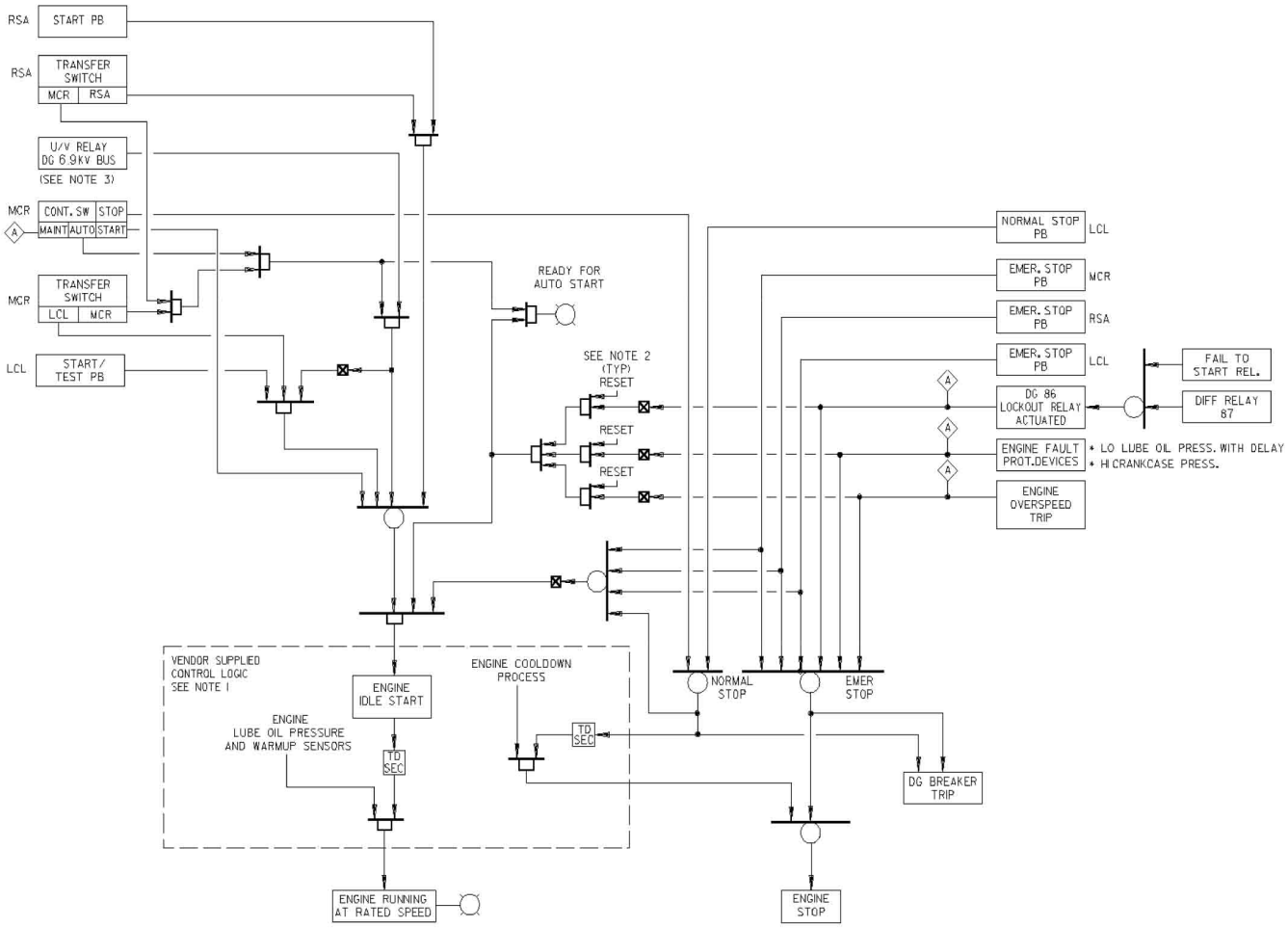

Figure 8.3.1-2
Onsite Standby Diesel Generator Initiating Circuit Logic Diagram

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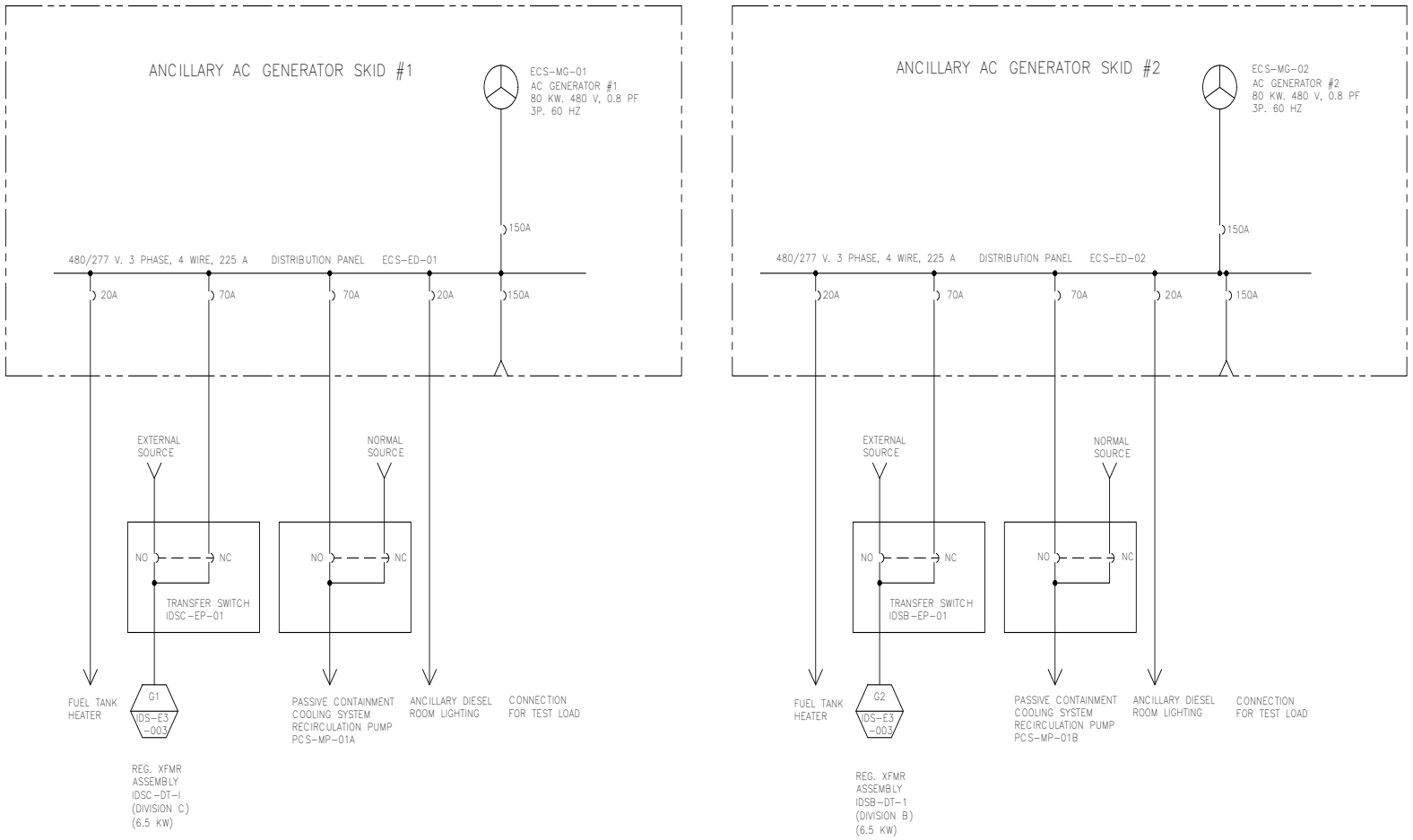

Figure 8.3.1-3
Ancillary Diesel Generator
One Line Diagram

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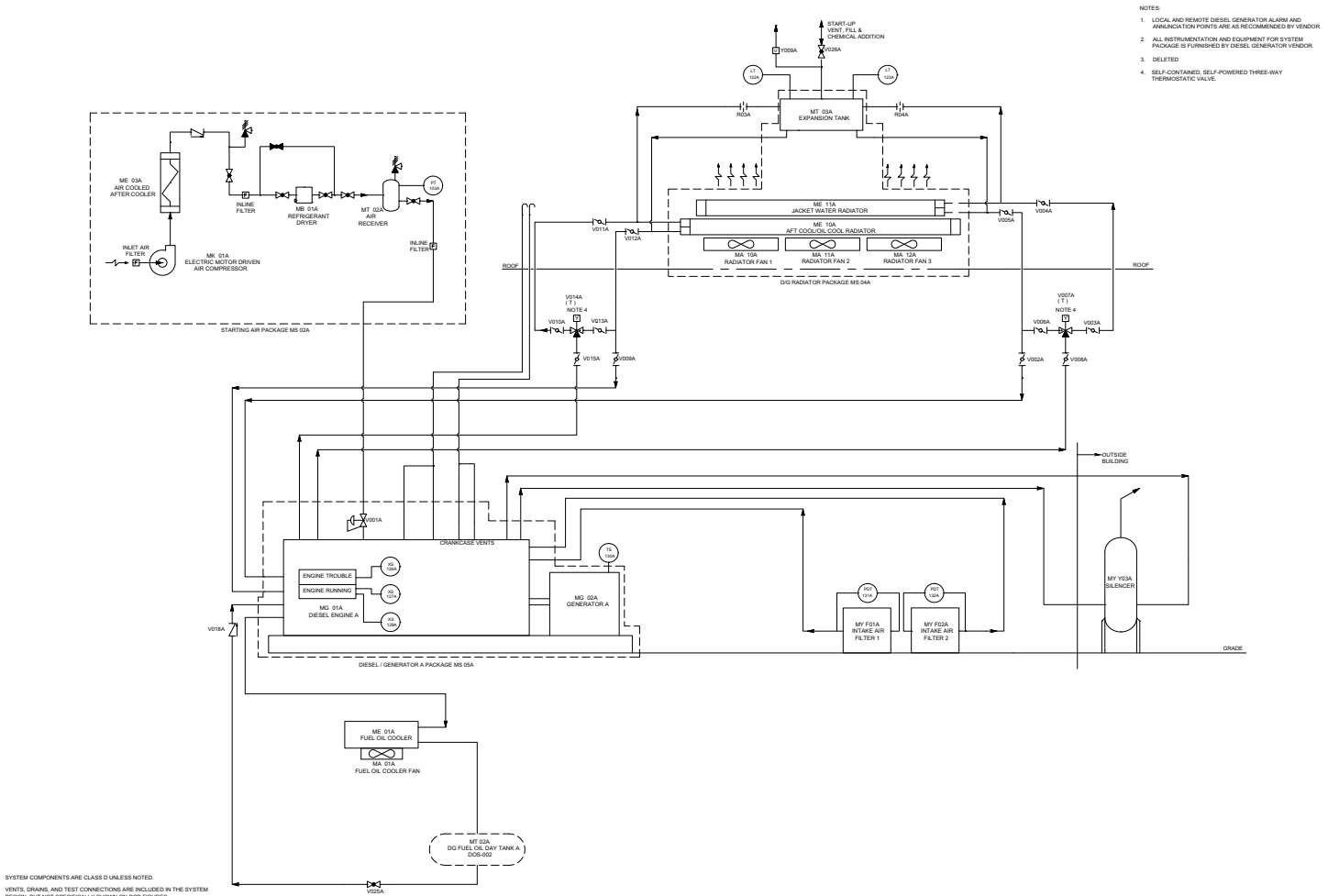

Inside Diesel Generator Building

Figure 8.3.1-4 (Sheet 1 of 2)
Simplified Diesel Generator System Piping and Instrumentation Diagram
(REF) ZOS 001

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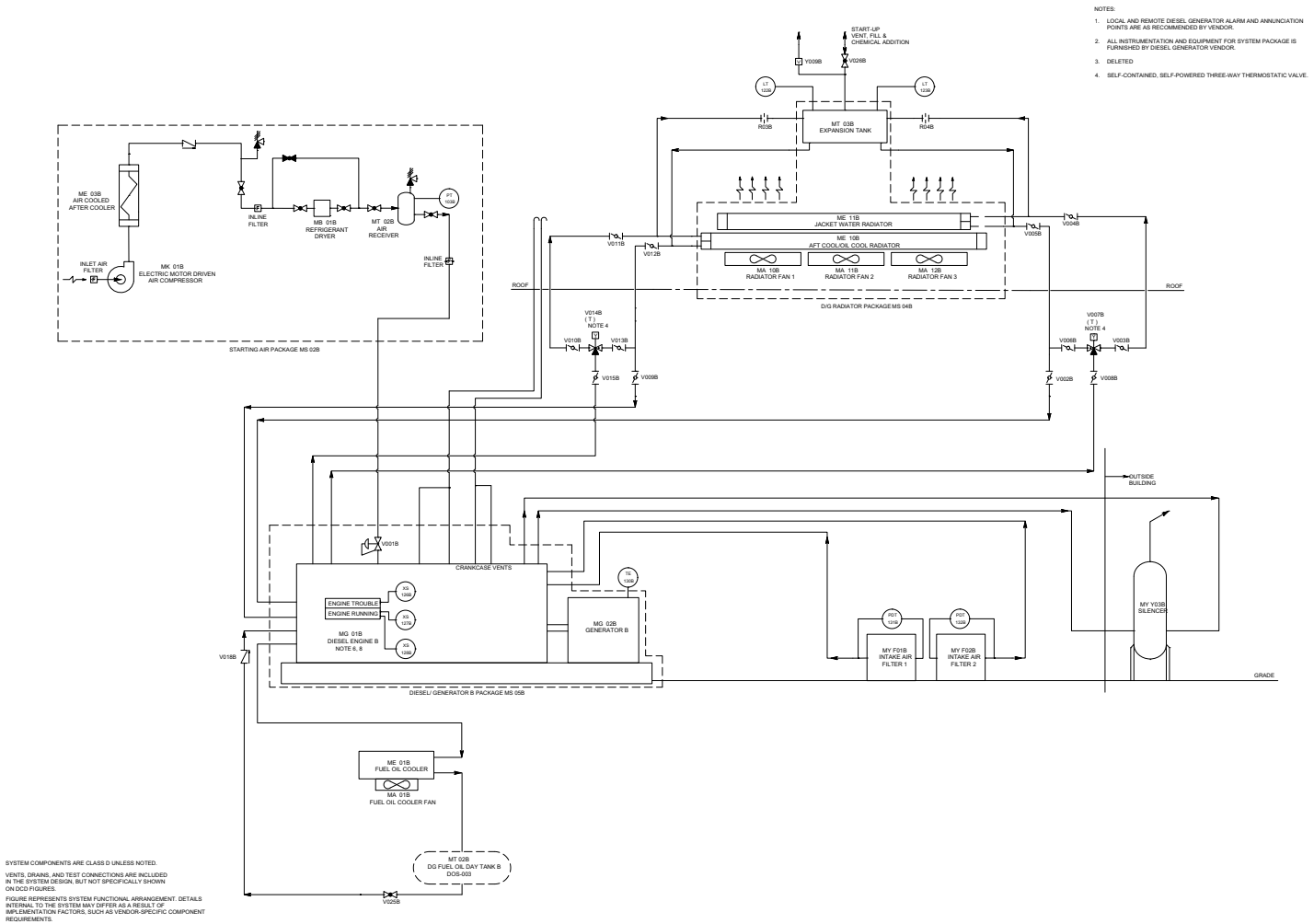

Inside Diesel Generator Building

Figure 8.3.1-4 (Sheet 2 of 2)
Simplified Diesel Generator System Piping and Instrumentation Diagram
(REF) ZOS 002

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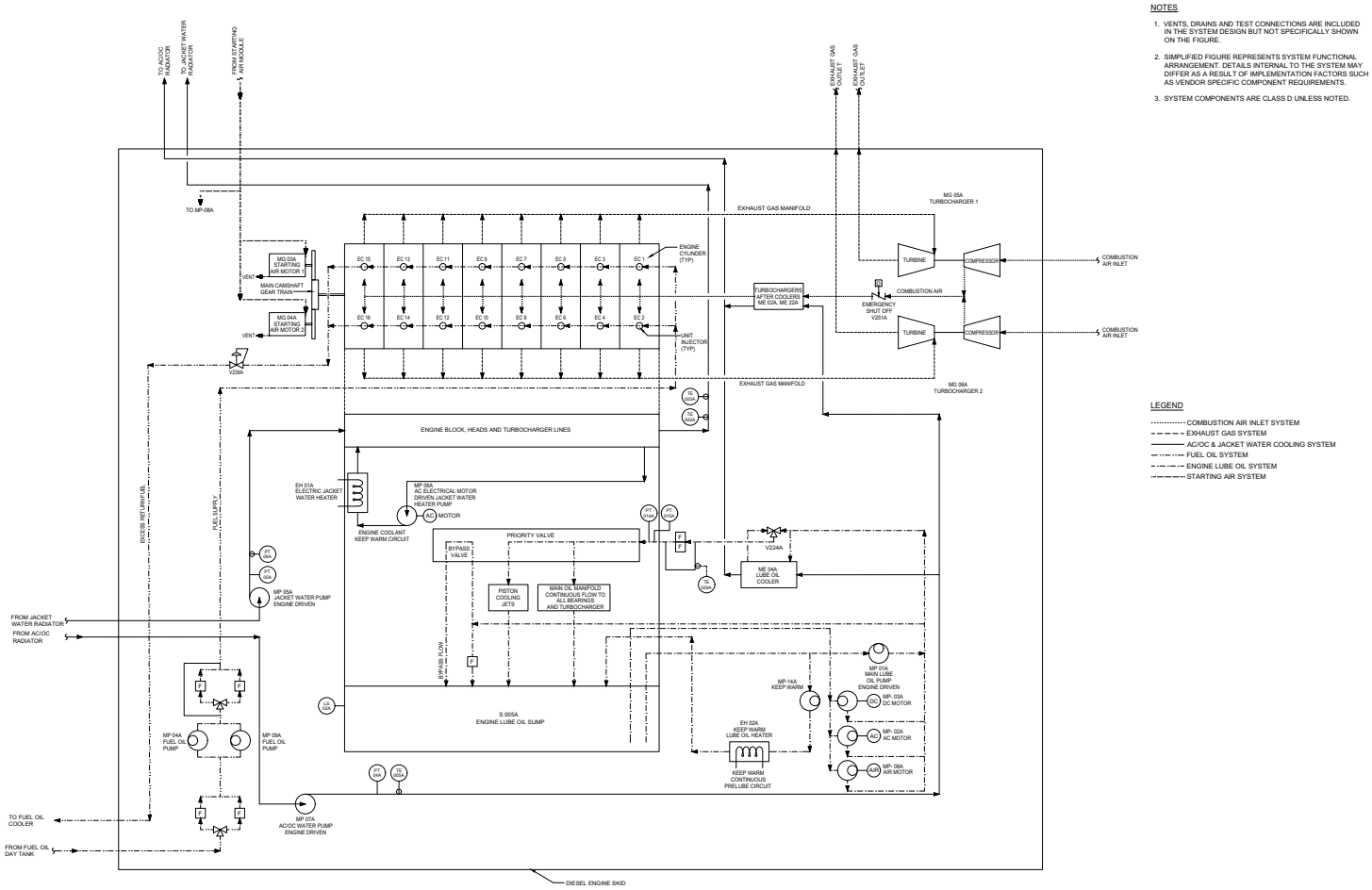

Inside Diesel Generator Building

Figure 8.3.1-5 (Sheet 1 of 2)

Diesel Engine Skid Mounted System

(REF) ZOS K001

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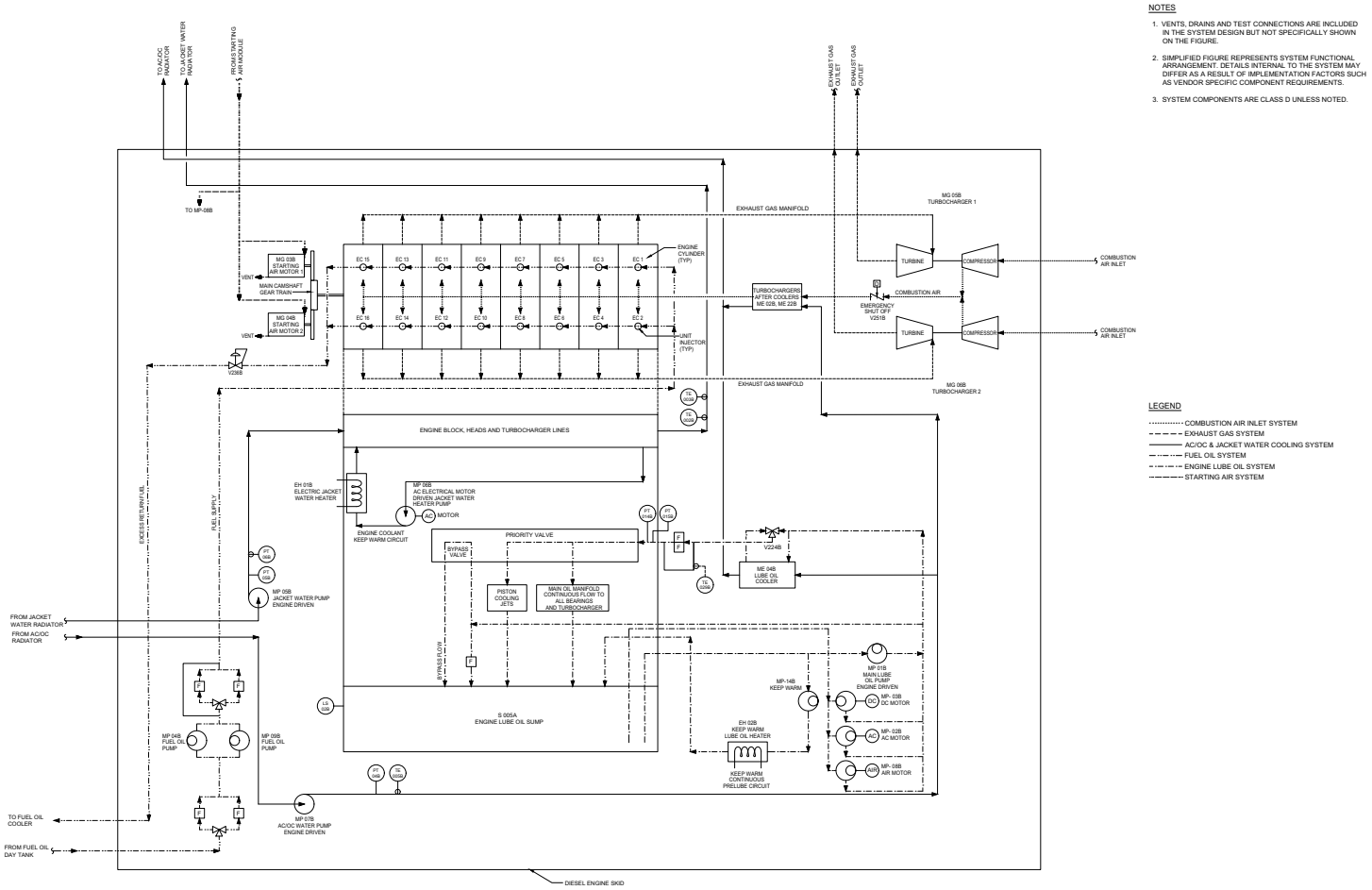

Inside Diesel Generator Building

Figure 8.3.1-5 (Sheet 2 of 2)

Diesel Engine Skid Mounted System

(REF) ZOS K002

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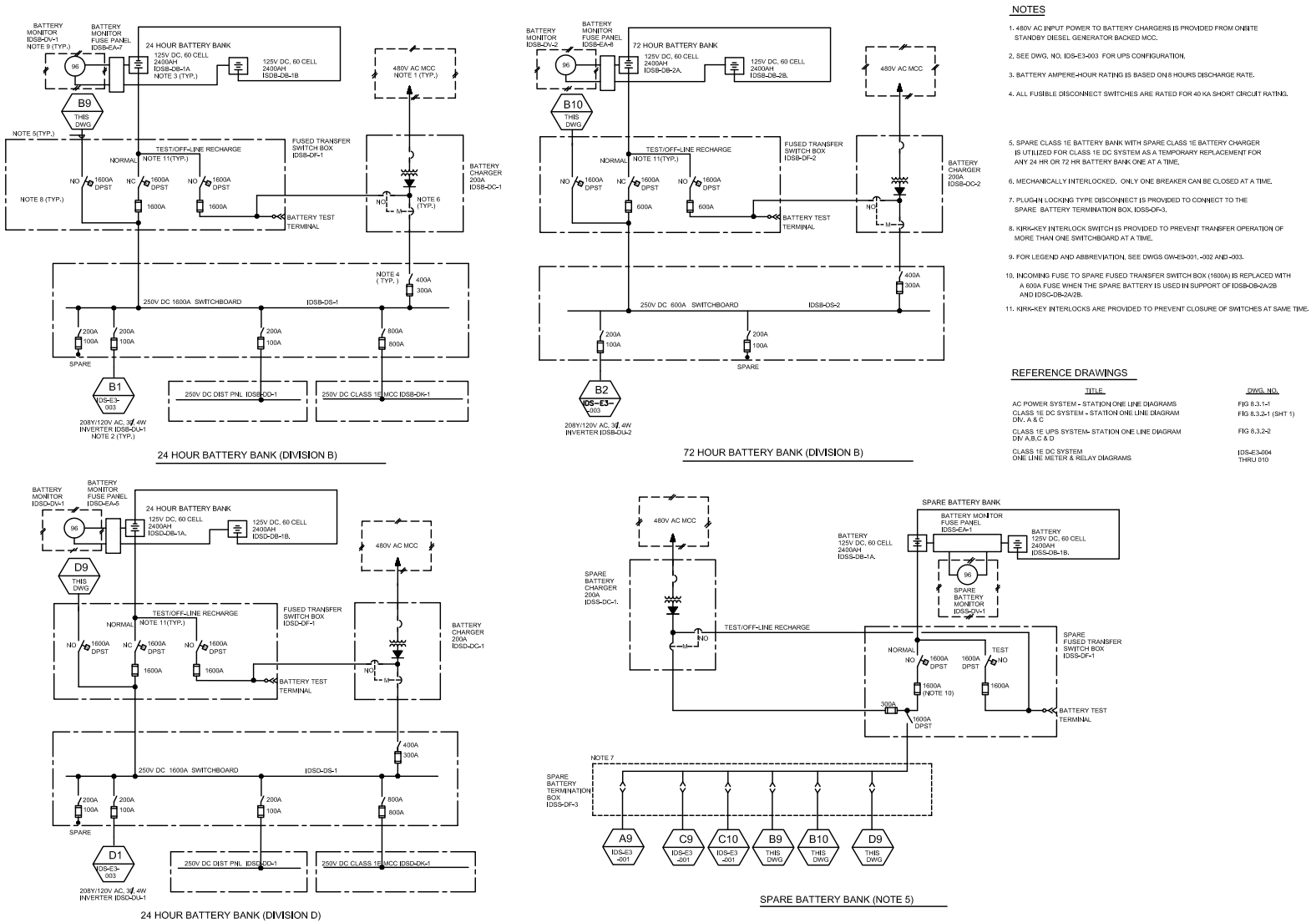

Figure 8.3.2-1 (Sheet 1 of 2)
Class 1E DC System One Line Diagram

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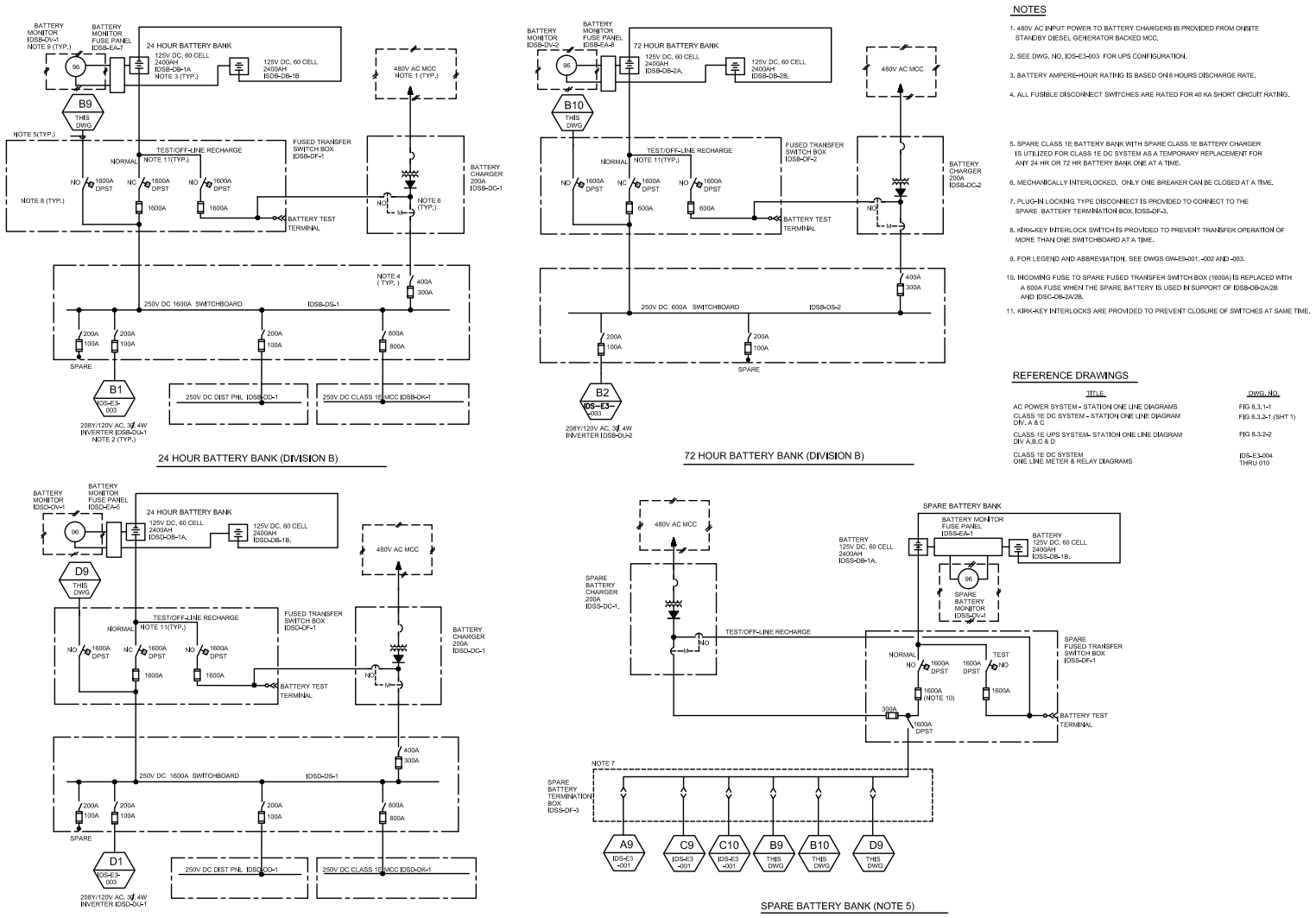

Figure 8.3.2-1 (Sheet 2 of 2)
Class 1E DC System One Line Diagram

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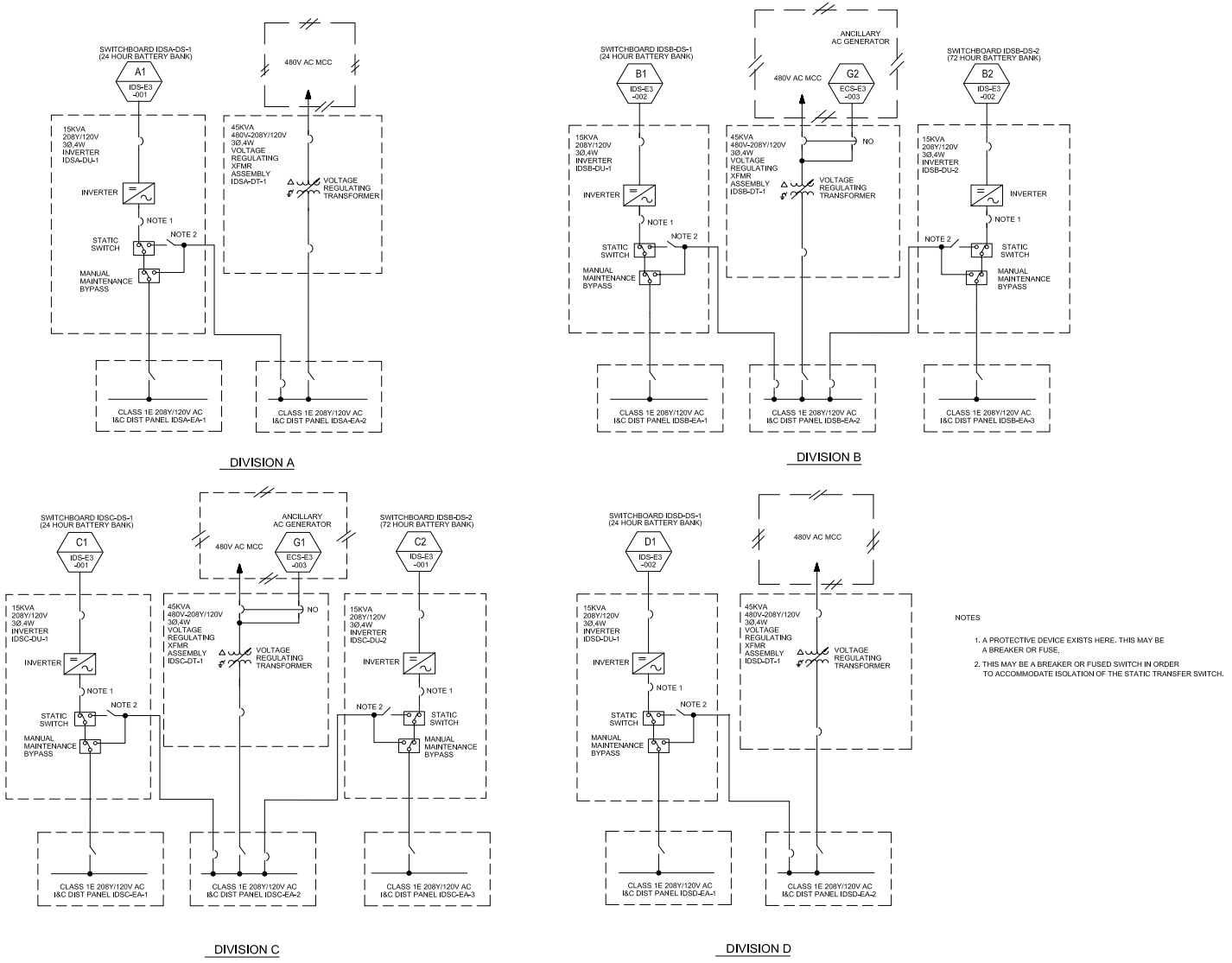

Figure 8.3.2-2
Class 1E 208y/120V UPS One Line Diagram

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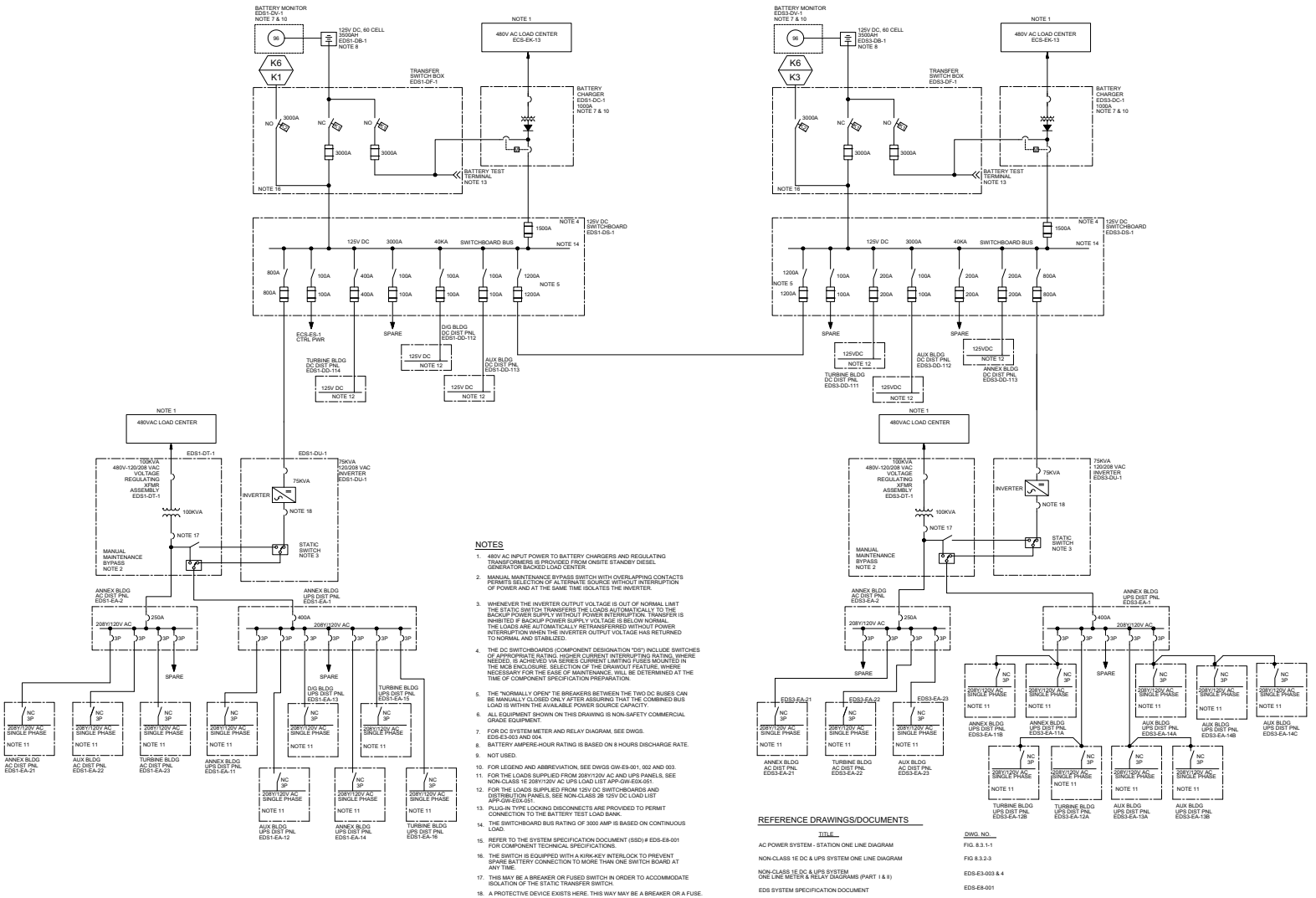

NOTES

1. 480V AC INPUT POWER TO BATTERY CHARGERS AND REGULATING TRANSFORMERS IS PROVIDED FROM ONSITE STANDBY DIESEL GENERATOR BACKED LOAD CENTER.
2. MANUAL MAINTENANCE BYPASS SWITCH WITH OVERLAPPING CONTACTS PERMITS SELECTION OF ALTERNATE SOURCE WITHOUT INTERRUPTION OF POWER AND AT THE SAME TIME ISOLATES THE INVERTER.
3. WHENEVER THE INVERTER OUTPUT VOLTAGE IS OUT OF NORMAL LIMIT THE STATIC SWITCH TRANSFERS THE LOADS AUTOMATICALLY TO THE BACKUP POWER SUPPLY WITHOUT POWER INTERRUPTION. TRANSFER IS INHIBITED IF BACKUP POWER SUPPLY VOLTAGE IS BELOW NORMAL. THE LOADS ARE AUTOMATICALLY RETRANSFERRED WITHOUT POWER INTERRUPTION WHEN THE INVERTER OUTPUT VOLTAGE HAS RETURNED TO NORMAL AND STABILIZED.
4. THE DC SWITCHBOARDS (COMPONENT DESIGNATION "DS") INCLUDE SWITCHES OF APPROPRIATE RATING. HIGHER CURRENT INTERRUPTING RATING, WHERE NEEDED, IS ACHIEVED VIA SERIES CURRENT LIMITING FUSES MOUNTED IN THE MCB ENCLOSURE. SELECTION OF THE DRAWOUT FEATURE, WHERE NECESSARY FOR THE EASE OF MAINTENANCE, WILL BE DETERMINED AT THE TIME OF COMPONENT SPECIFICATION PREPARATION.
5. THE "NORMALLY OPEN" TIE BREAKERS BETWEEN THE TWO DC BUSES CAN BE MANUALLY CLOSED ONLY AFTER ASSURING THAT THE COMBINED BUS LOAD IS WITHIN THE AVAILABLE POWER SOURCE CAPACITY.
6. ALL EQUIPMENT SHOWN ON THIS DRAWING IS NON-SAFETY COMMERCIAL GRADE EQUIPMENT.
7. FOR DC SYSTEM METER AND RELAY DIAGRAM, SEE DWGS. EDS-E3-003 AND 004.
8. BATTERY AMPERE-HOUR RATING IS BASED ON 8 HOURS DISCHARGE RATE.
9. NOT USED.
10. FOR LEGEND AND ABBREVIATION, SEE DWGS GW-E9-001, 002 AND 003.
11. FOR THE LOADS SUPPLIED FROM 208Y/120V AC AND UPS PANELS, SEE NON-CLASS 1E 208Y/120V AC UPS LOAD LIST APP-GW-E0X-051.
12. FOR THE LOADS SUPPLIED FROM 125V DC SWITCHBOARDS AND DISTRIBUTION PANELS, SEE NON-CLASS 2E 125V DC LOAD LIST APP-GW-E0X-051.
13. PLUG-IN TYPE LOCKING DISCONNECTS ARE PROVIDED TO PERMIT CONNECTION TO THE BATTERY TEST LOAD BANK.
14. THE SWITCHBOARD BUS RATING OF 3000 AMP IS BASED ON CONTINUOUS LOAD.
15. REFER TO THE SYSTEM SPECIFICATION DOCUMENT (SSD) # EDS-E8-001 FOR COMPONENT TECHNICAL SPECIFICATIONS.
16. THE SWITCH IS EQUIPPED WITH A KIRK-KEY INTERLOCK TO PREVENT SPARE BATTERY CONNECTION TO MORE THAN ONE SWITCH BOARD AT ANY TIME.
17. THIS MAY BE A BREAKER OR FUSED SWITCH IN ORDER TO ACCOMMODATE ISOLATION OF THE STATIC TRANSFER SWITCH.
18. A PROTECTIVE DEVICE EXISTS HERE. THIS WAY MAY BE A BREAKER OR A FUSE.

REFERENCE DRAWINGS/DOCUMENTS

TITLE	DWG. NO.
AC POWER SYSTEM - STATION ONE LINE DIAGRAM	FIG. 8.3.1-1
NON-CLASS 1E DC & UPS SYSTEM ONE LINE DIAGRAM	FIG 8.3.2-3
NON-CLASS 1E DC & UPS SYSTEM ONE LINE METER & RELAY DIAGRAMS (PART I & II)	EDS-E3-003 & 4
EDS SYSTEM SPECIFICATION DOCUMENT	EDS-E8-001

Figure 8.3.2-3 (Sheet 1 of 3)
Non-Class 1E DC & UPS System One Line Diagram

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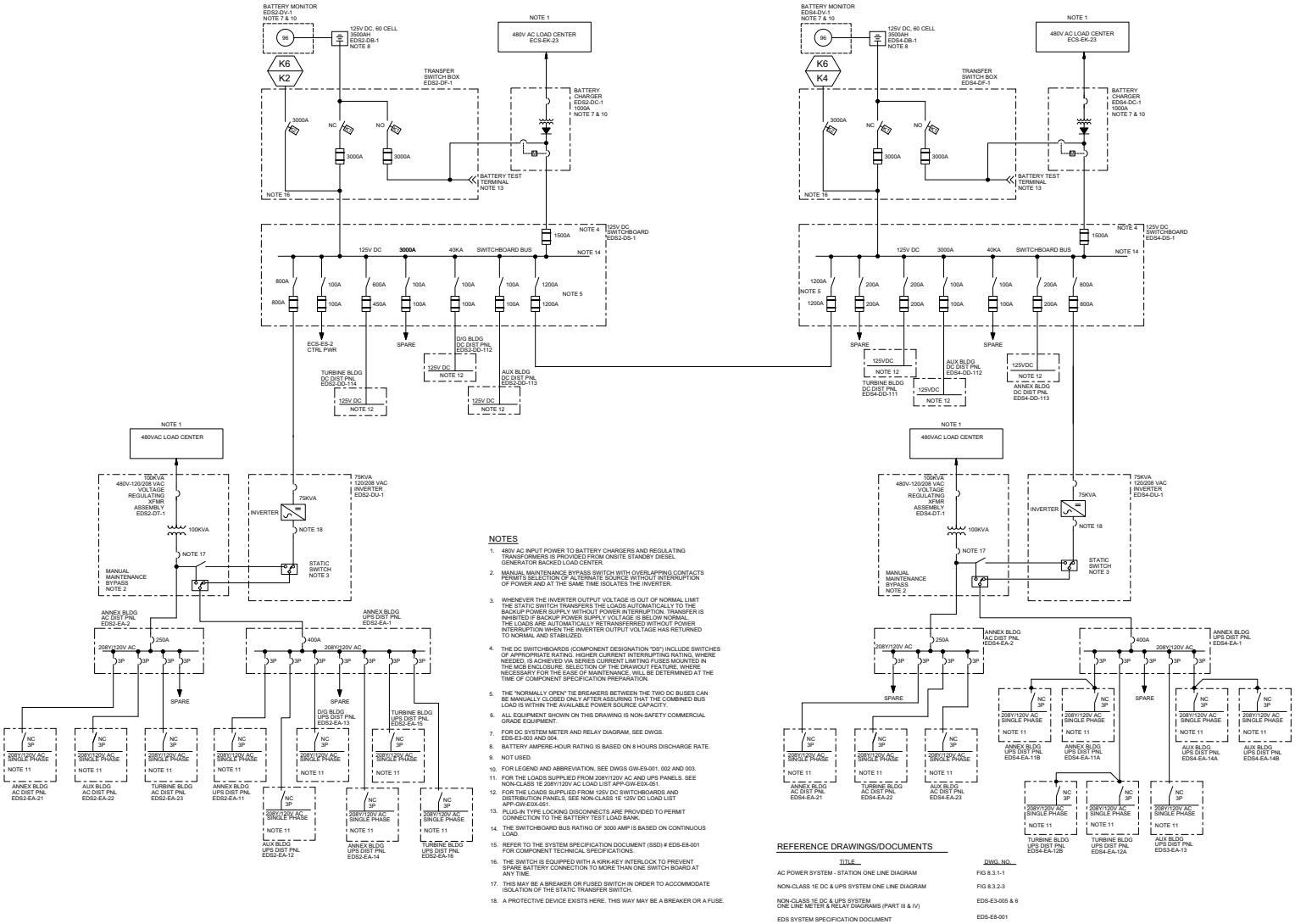

Figure 8.3.2-3 (Sheet 2 of 3)
Non-Class 1E DC & UPS System One Line Diagram

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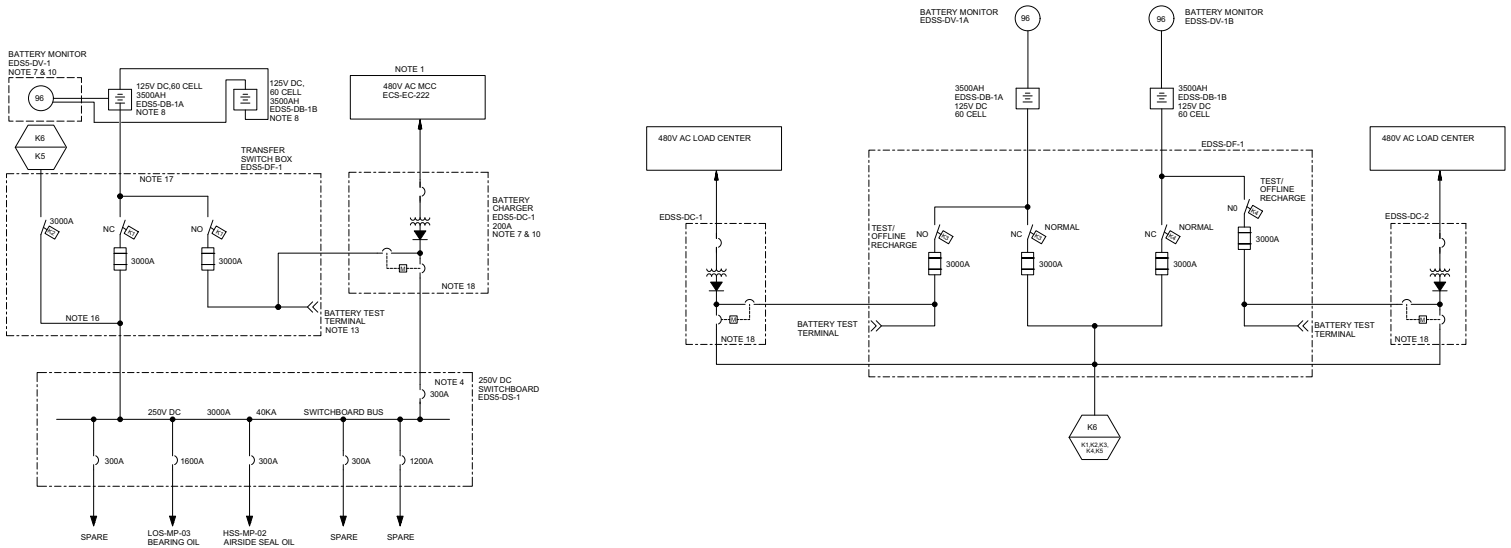

NOTES

1. 480V AC INPUT POWER TO BATTERY CHARGERS IS PROVIDED FROM ONSITE STANDBY DIESEL GENERATOR BACKED MOTOR CONTROL CENTERS.
2. NOT USED
3. NOT USED
4. THE DC SWITCHBOARDS (COMPONENT DESIGNATION "DS") INCLUDE MOLDED CASE CIRCUIT BREAKERS OF APPROPRIATE RATING. HIGHER CURRENT INTERRUPTING RATING, WHERE NEEDED, IS ACHIEVED VIA SERIES CURRENT LIMITING FUSES MOUNTED IN THE MCB ENCLOSURE. SELECTION OF THE DRAWOUT FEATURE, WHERE NECESSARY FOR THE EASE OF MAINTENANCE, WILL BE DETERMINED AT THE TIME OF COMPONENT SPECIFICATION PREPARATION.
5. NOT USED
6. ALL EQUIPMENT SHOWN ON THIS DRAWING IS NNS COMMERCIAL GRADE EQUIPMENT.
7. FOR NON-CLASS 1E DC SYSTEM METER AND RELAY DIAGRAM. SEE DWGS. EDS-E3-003 AND 004.
8. BATTERY AMPERE-HOUR RATING IS BASED ON 8 HOURS DISCHARGE RATE.
9. NOT USED
10. FOR LEGEND AND ABBREVIATION, SEE DWGS GW-E9-001, 002 AND 003.
11. NOT USED
12. NOT USED
13. PLUG-IN TYPE LOCKING DISCONNECTS ARE PROVIDED TO PERMIT CONNECTION TO THE BATTERY TEST LOAD BANK.
14. THE SWITCHBOARD BUS RATING OF 3000 AMP IS BASED ON CONTINUOUS LOAD.
15. REFER TO THE SYSTEM SPECIFICATION DOCUMENT (SSD) # APP-EDS-E8-001 FOR COMPONENT TECHNICAL SPECIFICATIONS.
16. THE SWITCH IS EQUIPPED WITH A KIRK-KEY INTERLOCK TO PREVENT SPARE BATTERY CONNECTION TO MORE THAN ONE SWITCH BOARD AT ANY TIME.
17. KIRK-KEY INTERLOCKS ARE PROVIDED TO PREVENT CLOSURE OF SWITCHES AT SAME TIME.
18. MECHANICALLY INTERLOCKED, ONLY ONE BREAKER CAN BE CLOSED AT A TIME.

REFERENCE DRAWINGS/DOCUMENTS

TITLE	DWG. NO.
AC POWER SYSTEM - STATION ONE LINE DIAGRAM	FIG. 8.3.1-1
NON-CLASS 1E DC & UPS SYSTEM ONE LINE DIAGRAM	FIG. 8.3.2-3
NON-CLASS 1E DC & UPS SYSTEM ONE LINE METER & RELAY DIAGRAMS (PART I & II)	EDS-E3-003 & 4
EDS SYSTEM SPECIFICATION DOCUMENT	EDS-E8-001

Figure 8.3.2-3 (Sheet 3 of 3)
Non-Class 1E DC & UPS System One Line Diagram

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Chapter 9 Auxiliary Systems

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.1 Design Bases

New fuel is stored in a high density rack which includes integral neutron absorbing material to maintain the required degree of subcriticality. The rack is designed to store fuel of the maximum design basis enrichment. The rack in the new fuel pit consists of an array of cells interconnected to each other at several elevations and to a thick base plate at the bottom elevation. This rack module is not anchored to the pit floor.

The new fuel rack includes storage locations for 72 fuel assemblies. The rack layout and array center-to-center spacing is shown in Figure 9.1-1. This spacing provides a minimum separation between adjacent fuel assemblies which is sufficient to maintain a subcritical array even in the event the building is flooded with unborated water or fire extinguishant aerosols or during any design basis event. The design of the rack is such that a fuel assembly cannot be inserted into a location other than a location designed to receive an assembly. Surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel.

The requirements of ANS 57.1 are addressed in Subsection 9.1.4. The rack is designed to withstand nominal operating loads and safe shutdown earthquake seismic loads defined in Table 9.1-1. The new fuel storage rack is designed to meet seismic Category I requirements of Regulatory Guide 1.29. Refer to Subsection 1.9.1 for compliance with Regulatory Guides. The rack is also designed to withstand the maximum uplift force of the fuel handling machine.

AP1000 equipment, seismic and ASME Code classifications are discussed in Section 3.2. The requirements of ASME Code Section III, Division I, Article NF3000 are used as the criteria for evaluation of stress analysis. The materials are procured in accordance with the applicable material specifications defined in ASME Section II, Part A as described in APP-GW-GLR-026 (Reference 16). Criticality analyses are performed in accordance with the requirements of ANSI N16.1-75, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (Reference 1); and analysis codes are validated against the requirements of ANSI N16.9-75, Validation of Calculational Methods for Nuclear Criticality Safety (Reference 2).

The stress analysis of the new fuel rack satisfies all of the applicable provisions in NRC Regulatory Guide 1.124, Revision 1 for components design by the linear elastic method (Reference 22).

9.1.1.2 Facilities Description

The new fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. The facility is protected from the effects of natural phenomena such as earthquakes, wind, tornados, floods, and external missiles by the external walls of the auxiliary building. See Section 3.5 for additional discussion on protection from missiles. The facility is designed to maintain its structural integrity following a safe shutdown earthquake and to perform its intended function following a postulated event such as fire, internal missiles, or pipe break. The walls surrounding the fuel handling area and new fuel storage pit protect the fuel from missiles generated inside the auxiliary building. The fuel handling area does not contain a credible source of missiles. Refer to Subsection 1.2.4.3 for a discussion of the auxiliary building. Refer to Section 3.8 for a discussion of the structural design of the new fuel storage area. Refer to Subsection 3.5.1 for a discussion of missile sources and protection.

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The dry, unlined, approximately 17-feet deep reinforced concrete pit is designed to provide support for the new fuel storage rack. The rack is supported by the pit floor. The walls of the new fuel pit are seismic Category I. The new fuel pit is normally covered to prevent foreign objects from entering the new fuel storage rack. Since the only hoist that can access the new fuel pit is single failure proof and no heavy loads are carried over the new fuel storage pit, a load drop is not likely, and the new fuel pit cover is not designed to protect the fuel assemblies from the effects of a dropped load. Figures 1.2-7 through 1.2-10 show the relationship between the new fuel storage facility and other features of the fuel handling area.

The new fuel storage pit is drained by gravity drains that are part of the radioactive waste drain system (Subsection 9.3.5), draining to the auxiliary building sump and pumped to the waste holdup tanks which are part of the liquid radwaste system (Section 11.2). These drains preclude flooding of the pit by an accidental release of water.

Nonseismic equipment in the vicinity of the new fuel storage rack is evaluated to confirm that its failure could not result in an increase of K_{eff} beyond the maximum allowable K_{eff}. Refer to Subsection 3.7.3.13 for a discussion of the nonseismic equipment evaluation.

The fuel handling machine is used to handle new fuel assemblies in the rail car bay, new fuel rack, and new fuel elevator. The capacity of the fuel handling machine, while over the new fuel storage rack, is limited to lifting a fuel assembly, control rod assembly, and handling tool. The new fuel storage rack is not accessed by the cask handling crane. This precludes the movement of loads greater than fuel components over stored new fuel assemblies.

During fuel handling operations, a ventilation system removes gaseous radioactivity from the atmosphere above the new fuel pit. Refer to Subsection 9.4.3 for a discussion of the fuel handling area HVAC system and Section 11.5 for process radiation monitoring. Security for the new fuel assemblies is described in separate security documents referred to in Section 13.6.

9.1.1.2.1 New Fuel Rack Design

A. Design and Analysis of the New Fuel Rack

The new fuel storage rack array center-to-center spacing of nominally 10.9 inches provides a minimum separation between adjacent fuel assemblies sufficient with neutron absorbing material to maintain a subcritical array. The seismic and stress analyses of the new fuel rack consider the conditions of full and partially filled fuel assembly loadings. The rack is evaluated for the safe shutdown earthquake condition against the seismic Category I requirements. A stress analysis is performed to verify the acceptability of the critical load components and paths under normal and faulted conditions. The rack rests on the pit floor.

The dynamic response of the fuel rack assembly during a seismic event is the condition which produces the governing loads and stresses on the structure. The new fuel storage rack is designed to meet the seismic Category I requirements of Regulatory Guide 1.29.

Loads and Load Combinations

The applied loads to the new fuel rack are:

- Dead loads
- Live loads - effect of lifting the empty rack during installation
- Seismic forces of the safe shutdown earthquake
- Fuel handling machine uplift while over the new fuel rack - postulated stuck new fuel assembly

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Table 9.1-1 shows loads and load combinations considered in the analyses of the new fuel rack.

The margins of safety for the new fuel rack in the multi-direction seismic event are produced using loads obtained from the seismic analysis based on the simultaneous application of three statistically independent, orthogonal accelerations.

B. Fuel Handling Machine Uplift Analysis

An analysis is performed to demonstrate that the rack can withstand a maximum uplift load of 4000 pounds. This load is applied to a postulated stuck fuel assembly. Resultant new fuel rack stresses are evaluated against the stress limits and are demonstrated to be acceptable. It is demonstrated that there is no change in new fuel rack geometry of a magnitude which causes the criticality criteria to be violated.

C. New Fuel Assembly Drop Accident Analysis

During normal fuel handling operations, a single failure-proof hoist, designed to meet the requirements of NUREG-0554, is the only hoist capable of moving new fuel above the operating floor. Per the design criteria contained in NUREG-0554, drops from a single failure-proof hoist are deemed unlikely and do not require further analysis. The consequences of such a drop are minimal since no safety-related equipment would be impacted and there are no radiological releases with new unirradiated fuel. Because the likelihood of a new fuel assembly being dropped into the new fuel pit and onto the new fuel racks is minimal, it is unnecessary to evaluate drop scenarios for the new fuel storage rack.

D. Failure of the Fuel Handling Machine

The fuel handling machine is a seismic Category II component. The fuel handling machine is evaluated to show that the machine does not fall into the new fuel pit during a seismic event. The fuel handling machine is also designed to maintain its load carrying and structural integrity functions during a safe shutdown earthquake.

E. Internally Generated Missiles

The fuel handling area does not contain any credible sources of internally generated missiles.

Stress analyses are performed by the vendor using loads developed by the dynamic analysis. Stresses are calculated at critical sections of the rack and compared to acceptance criteria referenced in ASME Section III, Division I, Article NF3000.

9.1.1.3 Safety Evaluation

The new fuel rack, being a seismic Category I structure, is designed to withstand normal and postulated dead loads, live loads, loads resulting from thermal effects, and loads caused by the safe shutdown earthquake event.

The design of the new fuel rack is such that K_{eff} (with all biases and uncertainties) remains less than or equal to 0.95 with full density unborated water and less than or equal to 0.98 with optimum moderation and full reflection conditions.

The criticality evaluation considers the inherent neutron absorbing effect of the materials of construction, including fixed neutron absorbing "poison" material.

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The new fuel rack is located in the new fuel storage pit, which has a large cover to protect the new fuel from debris. This large cover contains smaller openings/ports that access each new fuel assembly cell and have their own individual cover. Each small cover is opened/removed sequentially during new fuel transfer operations to gain access to the affected assembly and is then replaced. Both covers typically remain closed to prevent debris and foreign materials from entering the new fuel pit, new fuel rack, and new fuel assemblies. No loads are required to be carried over the new fuel storage pit while the large cover is in place. The large cover is designed such that it will not fall and damage the new fuel or new fuel rack during a seismic event. Administrative controls are utilized when either of the covers is removed for new fuel transfer operations to limit the potential for dropped object damage.

Based on the conservative design and operation of the single failure-proof FHM hoist and associated lifting tools to handle unirradiated new fuel assemblies, dropping a new fuel assembly is deemed unlikely, poses no safety or radiological consequences, and therefore, does not require analysis. Handling equipment (cask handling crane) capable of carrying loads heavier than fuel components is prevented from traveling over the fuel storage area. The new fuel storage rack can withstand an uplift force of 4000 pounds.

Materials used in new fuel rack construction are compatible with the storage pit environment, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. Structural materials are corrosion resistant and will not contaminate the fuel assemblies or storage pit environment. Neutron absorbing "poison" material used in the new fuel rack design has been qualified for the storage environment. Venting of the neutron absorbing material is considered in the detailed design of the new fuel storage rack.

The new fuel assemblies are stored dry. The rack structure is designed to maintain a safe geometric array for normal and postulated accident conditions. The fixed neutron absorbing "poison" material maintains the required degree of subcriticality for normal and postulated accident conditions such as flooding with pure water and low density optimum moderator "misting."

A discussion of the methodology used in the new fuel rack criticality analysis is provided in APP-GW-GLR-030 (Reference 17).

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

Spent fuel is stored in high density racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The racks are designed to store fuel of the maximum design basis enrichment. Each rack in the spent fuel pool consists of an array of cells interconnected to each other at several elevations and to a thick base plate at the bottom elevation. These rack modules are free-standing, neither anchored to the pool floor nor braced to the pool wall. The spent fuel storage racks include storage locations for 884 fuel assemblies and five defective fuel assemblies. The Region 1 spent fuel rack layout is shown in Figure 9.1-2. The Region 2 spent fuel rack layout is shown in Figure 9.1-3. The overall spent fuel pool rack layout is presented in Figure 9.1-4. All spent fuel racks will be in place whenever fuel is stored in the spent fuel racks. See Subsection 3.7.5.2, for discussion of site-specific procedures for activities following an earthquake. An activity will be to address measurement of the post-seismic event gaps between spent fuel racks and to take appropriate corrective actions.

The design of the spent fuel racks is such that a fuel assembly cannot be inserted into a location (i.e., between racks) other than a location designed to receive an assembly. Insertion of a fuel assembly into a fuel pool (vs. rack) location not designed to receive it (i.e., tool storage area, adjacent to the defective fuel cells and wall) is prevented administratively and addressed in Subsection 9.1.2.3.

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AP1000 equipment, seismic and ASME Code classifications are discussed in Section 3.2. The requirements of ASME Section III, Division I, Article NF3000 are used as the criteria for evaluation of stress analyses. The materials are procured in accordance with the applicable material specifications defined in ASME Section II, Part A as described in APP-GW-GLR-033 (Reference 18). Criticality analyses are performed in accordance with the requirements of ANSI N16.1-75, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (Reference 1); analysis codes are validated against the requirements of ANSI N16.9-75, Validation of Calculational Methods for Nuclear Criticality Safety (Reference 2); and overall requirements for fuel storage are in accordance with ANSI N210-76, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations (Reference 3).

The stress analysis of the spent fuel racks satisfies all of the applicable provisions in NRC Regulatory Guide 1.124, Revision 1 for components designed by the linear elastic analysis method (Reference 22).

The spent fuel pool is designed to preclude inadvertent draining of the water from the pool.

9.1.2.2 Facilities Description

The spent fuel storage facility is designed to the guidelines of ANS 57.2 (Reference 4). The spent fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. The walls of the spent fuel pool are an integral part of the seismic Category I auxiliary building structure. The facility is protected from the effects of natural phenomena such as earthquakes (Subsection 3.7.2), wind and tornados (Section 3.3), floods (Section 3.4), and external missiles (Section 3.5).

The facility is designed to maintain its structural integrity following a safe shutdown earthquake and to perform its intended function following a postulated event such as a fire. Refer to Subsection 1.2.4.3 for further discussions of the auxiliary building fuel handling area.

Nonseismic equipment in the vicinity of the spent fuel storage racks is evaluated to confirm that its failure could not result in an increase of K_{eff} beyond the maximum allowable K_{eff}. Refer to Subsection 3.7.3.13 for a discussion of the nonseismic equipment evaluation.

The spent fuel pool provides storage space for spent fuel. The pool is approximately 42.5 feet deep and constructed of reinforced concrete and concrete filled structural modules as described in Subsection 3.8.4. The portion of the structural modules in contact with the water in the pool is stainless steel and the reinforced concrete portions are lined with a stainless steel plate. The minimum water volume of the pool is about 190,500 gallons of borated water (including racks without fuel at a water level 15 inches below the operating deck) with a nominal boron concentration of 2700 ppm. Figures 1.2-7 through 1.2-10 show the spent fuel pool and other features of the fuel handling area.

The connections for the drain and makeup lines are located to preclude the draining of the spent fuel pool due to a break in a line or failure of a pump to stop. The connection for the spent fuel cooling pumps' suction is located below normal water level and above the level needed to provide sufficient water for shielding and for cooling of the fuel if the spent fuel pool cooling system is unavailable. Skimmers that normally follow the water level surface do not travel below the level of the spent fuel cooling suction. Connections for suction to the chemical and volume control system are located between the normal water level and the spent fuel cooling system pumps' suction connection level. Pipes which discharge into the spent fuel pool include a siphon break between the normal water level and the level of the spent fuel cooling system pumps' suction connection.

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The piping which returns the water to the spent fuel pool from the spent fuel pool cooling system enters the pool at the opposite end from the spent fuel pool cooling system pumps' suction connection. The piping arrangement and location ensure thorough mixing of the cooled water into the pool to prevent stagnant or hot regions.

A gated opening connects the spent fuel pool and fuel transfer canal. The fuel transfer canal is connected to the in-containment refueling cavity by a fuel transfer tube. The spent fuel transfer operation is completed underwater, and the waterways are of sufficient depth to maintain a minimum of 8.75 feet of shielding water above the active fuel height of spent fuel assemblies. A metal gate with gasket assembly separates the spent fuel pool and fuel transfer canal. This allows the fuel transfer canal to be drained without reducing the water level in the spent fuel pool. During normal operation, this gate remains open and is only closed to drain the canal. The bottom of the fuel transfer canal has a drain connected to safety-related piping and isolation valves which prevents inadvertent draining after a seismic event. Subsection 9.1.3 further addresses the minimum water level in the spent fuel pool.

Next to the spent fuel pool and accessible by another gated, gasketed opening is a cask loading pit. The cask pit is a lined reinforced concrete structure of the auxiliary building fuel handling area. It is provided for underwater loading of fuel into a shipping cask and cask draining/decontamination prior to cask transshipment from the AP1000 site. The bottom of the cask loading pit has a drain connected to safety-related piping and isolation valve which prevents inadvertent draining after a seismic event. The gate between the spent fuel pool and the cask loading pit is normally closed and opened during refueling and cask loading options. The cask loading pit can be used as a source of water for low pressure injection to the reactor coolant system via the normal residual heat removal pumps during an event in which the reactor coolant system pressure and inventory decrease and the cask loading pit gate is closed.

The fuel handling machine traverses the spent fuel pool, the fuel transfer canal, the cask loading pit, the new fuel storage pit, and the rail car bay. It is used in the movement of both new and spent fuel assemblies. The fuel handling machine is used to transfer new fuel assemblies from the new fuel storage rack into the spent fuel pool. A new fuel elevator in the spent fuel pool lowers the new fuel to an elevation accessible by the fuel handling machine and its associated fuel handling tool.

The cask handling crane is used for operations involving the spent fuel shipping cask. The cask handling crane traverses the auxiliary building and a portion of the fuel handling area. The cask handling crane's path is designed such that the cask cannot pass over the spent fuel pool, new fuel pit, or fuel transfer canal. This precludes the movement of loads greater than fuel components over stored fuel in accordance with Regulatory Guide 1.13.

During fuel handling operations, a ventilation system removes gaseous radioactivity from the atmosphere above the spent fuel pool. Refer to Subsection 9.4.3 for a discussion of the radiologically controlled area ventilation system, Section 11.5 for process radiation monitoring, Subsection 9.1.3 for the spent fuel pool cooling system, and Subsection 12.2.2 for airborne activity levels in the fuel handling area.

9.1.2.2.1 Spent Fuel Rack Design

A. Design and Analysis of Spent Fuel Racks

The spent fuel pool rack layout contains both Region 1 rack modules with a center-to-center spacing of nominally 10.93 inches and Region 2 rack modules with a center-to-center spacing of nominally 9.04 inches. Additionally, there are five defective fuel assembly storage cells with a center-to-center spacing of nominally 11.65 inches. These rack module configurations provide

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adequate separation between adjacent fuel assemblies with neutron absorbing material to maintain a subcritical array.

The material used in the AP1000 fuel storage racks is Metamic®, a metal matrix composite material consisting of a Type 6061 aluminum alloy matrix reinforced with boron carbide (B_4C). The Metamic is in the form of sheets having a nominal thickness of 0.106 inches and a minimum ^{10}B areal density of 0.0304 gm/cm^2 (minimum 30.5 wt% B_4C). The panels are not anodized, but will be cleaned via glass bead blasting and washing with demineralized water to ensure removal of surface contamination prior to installation.

No credit is taken for Metamic in the rack structural analysis, and the Metamic panels are completely encased in (and supported by) stainless steel panels (which are vented) so the mechanical properties of the Metamic do not affect the performance of the panels. Nevertheless, mechanical properties (obtained from Reference 24) are summarized below:

Property	Value
Density	2.646 gm/cm^3
Yield Strength	33,000 psi
Ultimate Strength	40,000 psi
Elongation	1.8%

With the exception of density, which is given at 30 wt% B_4C, the properties in this table are all specified for 31 wt% B_4C. Metamic has been evaluated by the NRC for use in spent fuel pool applications (Reference 25).

The Metamic panels are suitable for long-term use in the spent fuel pool environment. The Metamic panels are potentially affected by the pool's temperature, aqueous environment, and radiation field. The effects of each of these parameters are discussed separately in the following paragraphs.

The pool's temperature will exceed the ambient temperature as a result of the stored fuel heat, but will be maintained at or below 120°F. Elevated temperature testing of 31 wt% B_4C Metamic was performed at 750°F in air for nearly a year (Reference 24), with no reduction in thickness, no change in weight, no reduction on ^{10}B content, and no change in density. The complete lack of dimensional or chemical changes in these elevated rate tests is sufficient to show that temperatures up to 120°F, even for 60 years or more, will not detrimentally affect the condition of the Metamic panels.

The aqueous environment of the pool with a nominal dissolved Boron concentration of 2,700 ppm will be slightly acidic. Elevated temperature (200°F) corrosion rate testing of 32 wt% B_4C Metamic (Reference 24) for 90 days indicated that "no corrosion was observed" and there was "no significant change in ^{10}B areal density." The complete lack of any chemical changes in the tests, combined with the knowledge of the effects of temperature and pH on corrosion rate, is sufficient to show that the aqueous spent fuel pool environment, even for 60 years, will not detrimentally affect the condition of the Metamic panels.

Samples of 31 wt% B_4C Metamic were subjected to a radiation field with both gamma (1.5×10^{11} rads) and fast neutron (1.7×10^{18} nvt to 5.8×10^{19} nvt) components. Conclusions of post-irradiation testing were: Metamic exhibits excellent dimensional stability after irradiation and there was no change in Boron-10 areal density (Reference 24). The complete lack of dimensional or chemical changes as a result of these high radiation exposures is sufficient to show that the expected radiation field in the pool, even for 60 years, will not detrimentally affect the condition of the Metamic panels. A coupon tree with 14 coupons is provided with the spent fuel racks.

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Coupons are nominally 6 inches wide by 8 inches long by 0.106 inches thick. Coupons of a different width and length are acceptable when boron testing is based on a weight percentage. Each coupon is representative of the panel from which it is cut, including the presence of any scratches or other surface irregularities. The initial B_4C content of Metamic is determined from the amounts of B_4C and aluminum powder mixed together in the manufacturing process. If for example, 1000 pounds of aluminum and 250 pounds of B_4C are used to manufacture one lot of material, it would be 20 percent B_4C. This approach has been validated by Metamic LLC, the supplier of the Metamic poison material, and Holtec International, the rack designer.

The recommended Metamic monitoring schedule is as follows:

- Prior to Refueling Outage 1 – Install Coupon Tree
- Prior to Refueling Outage 2 – Remove First Coupon
- Prior to Refueling Outage 3 – Remove Second Coupon
- Prior to Refueling Outage 5 – Remove Third Coupon
- Prior to Refueling Outage 10 – Remove Fourth Coupon
- Prior to Refueling Outage 20 – Remove Fifth Coupon
- Prior to Refueling Outage 30 – Remove Sixth Coupon
- Prior to Refueling Outage 40 – Remove Seventh Coupon

If the plant is operated on an 18-month cycle, seven coupons provide 60 years of Metamic surveillance. There are seven additional coupons provided.

This coupon tree will be used to monitor the condition of the Metamic over the 60-year life of the spent fuel racks.

The seismic and stress analyses of the spent fuel racks consider the various conditions of full, partially filled, and empty fuel assembly loadings. The racks are evaluated for the safe shutdown earthquake condition and seismic Category I requirements. A detailed stress analysis is performed to verify the acceptability of the critical load components and paths under normal and faulted conditions. The racks rest on the pool floor and are evaluated to determine that under loading conditions the rack-to-rack and rack-to-wall impacts are acceptable on both the racks and the pool walls.

The dynamic response of the fuel rack assembly during a seismic event is the condition which produces the governing loads and stresses on the structure.

Loads and Load Combinations

The applied loads to the spent fuel racks are:

- Dead loads
- Live loads - effect of lifting the empty rack during installation
- Seismic forces of the safe shutdown earthquake
- Fuel assembly drop analysis
- Fuel handling machine uplift - postulated stuck fuel assembly
- Thermal loads

Table 9.1-1 shows loads and load combinations that are considered in the analyses of the spent fuel racks including those given in Reference 5.

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The margins of safety for the racks in the multi-direction seismic event are produced using loads obtained from the seismic analysis based on the simultaneous application of three statistically independent, orthogonal accelerations.

B. Fuel Handling Machine Uplift Analysis

An analysis is performed to demonstrate that the racks can withstand a maximum uplift load of 5000 pounds. This load is applied to a postulated stuck fuel assembly. Resultant rack stresses are evaluated against the stress limits and are demonstrated to be acceptable. It is also demonstrated that there is no change in rack geometry of a magnitude which causes the criticality criteria to be violated.

C. Fuel Assembly Drop Accident Analysis

In the unlikely event of dropping a fuel assembly, accidental deformation of the rack is determined and evaluated in the criticality analysis to demonstrate that it does not cause the criticality criterion to be violated. The analysis considers only the case of a dropped spent, irradiated fuel assembly in a flooded pool and takes credit for dissolved boron in the water.

For the analysis of a dropped fuel assembly, two accident conditions are postulated. The first accident condition conservatively assumes that the weight of a fuel assembly, control rod assembly, and handling tool (3100 pounds total) impacts the top of the fuel rack from a drop height of 3 feet above the top of the rack. Both a straight drop and an inclined drop are included in the assessment. Calculations are performed to demonstrate that the impact energy is absorbed by the dropped fuel assembly, the rack cells, and the rack base plate assembly. Under these faulted conditions, credit is taken for dissolved boron in the pool water.

The second accident condition assumes that the dropped fuel assembly, control rod assembly, and handling tool (3100 pounds) fall straight through an empty cell and impact the rack base plate from a drop height of 3 feet above the top of the rack. The analysis is performed and demonstrates that the impact energy is absorbed by the fuel assembly and the rack base plate. At an interior rack location, base plate deformation is limited so that the pool liner is not impacted. At a support pad location, the stresses developed in the pool liner are evaluated to be within allowable limits such that the liner integrity is maintained. Under these faulted conditions, credit is taken for dissolved boron in the pool water.

D. Fuel Rack Sliding and Overturning Analysis

Consistent with the criteria of Reference 5, the racks are evaluated for overturning and sliding displacement due to earthquake conditions under the various conditions of full, partially filled, and empty fuel assembly loadings.

E. Failure of the Fuel Handling Machine

The fuel handling machine is a seismic Category II component. The fuel handling machine is evaluated to show that it does not collapse into the spent fuel pool as a result of a seismic event. The fuel handling machine is also designed to maintain its load carrying and structural integrity functions during a safe shutdown earthquake.

F. Internally Generated Missiles

The spent fuel handling area does not contain any credible sources of internally generated missiles.

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Stress analyses are performed by the vendor using loads developed by the dynamic analysis. Stresses are calculated at critical sections of the rack and compared to acceptance criteria referenced in ASME Section III, Division I, Article NF3000.

9.1.2.3 Safety Evaluation

The design and safety evaluation of the spent fuel racks is in accordance with Reference 5. The racks, being Equipment Class D and seismic Category I structures, are designed to withstand normal and postulated dead loads, live loads, loads resulting from thermal effects, and loads caused by the safe shutdown earthquake event.

The design of the racks is such that K_{eff} remains less than or equal to 0.95 under design basis conditions, including fuel handling accidents. Inadvertent insertion of a fuel assembly between the rack periphery and the pool wall or placement of a fuel assembly across the top of a fuel rack is considered a postulated accident, and as such, realistic initial conditions such as boron in the pool water are assumed. These accident conditions have an acceptable K_{eff} of less than 0.95. The criticality evaluation, which meets the requirements of 10 CFR 50.68, Paragraph b (Reference 21), considers the inherent neutron absorbing effect of the materials of construction, including fixed neutron absorbing "poison" material. Soluble boron in the spent fuel pool and assembly burnup is used as reactivity credits.

The racks are also designed with adequate energy absorption capabilities to withstand the impact of a dropped fuel assembly from the maximum lift height of the fuel handling machine. Handling equipment (cask handling crane) capable of carrying loads heavier than fuel components is prevented by design from carrying loads over the fuel storage area. The fuel storage racks can withstand an uplift force greater than or equal to the uplift capability of the fuel handling machine (5000 pounds).

Materials used in rack construction are compatible with the storage pool environment, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. Structural materials are corrosion resistant and will not contaminate the fuel assemblies or pool environment. Neutron absorbing "poison" material used in the rack design has been qualified for the storage environment. Venting of the neutron absorbing material is considered in the detailed design of the storage racks.

Design of the spent fuel storage facility is in accordance with Regulatory Guide 1.13. A discussion of the methodology used in the spent fuel pool criticality analysis is provided in APP-GW-GLR-029 (Reference 27).

9.1.3 Spent Fuel Pool Cooling System

The spent fuel pool cooling system (SFS) is designed to remove decay heat which is generated by stored fuel assemblies from the water in the spent fuel pool. This is done by pumping the high temperature water from within the fuel pool through a heat exchanger, and then returning the water to the pool. A secondary function of the spent fuel pool cooling system is clarification and purification of the water in the spent fuel pool, the transfer canal, and the refueling water. A listing of the major functions of the spent fuel pool cooling system and the corresponding modes of operation is provided below:

- **Spent fuel pool cooling** - Remove heat from the water in the spent fuel pool during operation to maintain the pool water temperature within acceptable limits.

- **Spent fuel pool purification** - Provide purification and clarification of the spent fuel pool water during operation.

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- **Refueling cavity purification** - Provide purification of the refueling cavity during refueling operations.
- **Water transfers** - Transfer water between the in-containment refueling water storage tank (IRWST) and the refueling cavity during refueling operations.
- **In-containment refueling water storage tank purification** - Provide purification and cooling of the in-containment refueling water storage tank during normal operation.

9.1.3.1 Design Basis

9.1.3.1.1 Safety Design Basis

The spent fuel pool cooling system has the safety-related function of containment isolation. See Subsection 6.2.3 for the containment isolation system. Safety-related makeup to the spent fuel pool is discussed in Subsection 9.1.3.4.3.

9.1.3.1.2 Power Generation Basis

The principal functions of the spent fuel pool cooling system are outlined above. The spent fuel pool cooling system is designed to perform its function in a reliable and failure tolerant manner. This reliability is achieved with the use of rugged and redundant equipment. The spent fuel pool cooling system is not a safety-related system and is not required to operate following events such as earthquake, fire, passive failures or multiple active failures.

9.1.3.1.3 Spent Fuel Pool Cooling

9.1.3.1.3.1 Partial Core

The spent fuel pool cooling system is designed to remove heat from the spent fuel pool such that the spent fuel pool water temperature will be ≤ 120°F following a partial core fuel shuffle refueling. The system is designed to perform this function based on the following:

- The assumed heat load is based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the fuel pool, which includes 44% of a core (69 assemblies) being placed into the pool at 120 hours after shutdown.
- Both trains of the spent fuel pool cooling system are assumed to be operating.
- While the reactor is shut down, the component cooling water system (CCS) supply temperature used to determine the performance of the spent fuel pool cooling system (SFS) heat exchangers is based on a service water system heat sink with a maximum normal ambient design wet bulb temperature as defined in Chapter 2, Table 2-1.
- While the reactor is at power, the CCS supply temperature to the SFS heat exchangers is based on a service water system heat sink with a maximum safety ambient design wet bulb temperature as defined in Chapter 2, Table 2-1.

9.1.3.1.3.2 Full Core Off-Load

The AP1000 normal refueling basis heat load is from a full core off-load. The spent fuel pool cooling system is designed to remove heat from the spent fuel pool such that the spent fuel pool water temperature will be ≤ 120°F following a full core off-load based upon a service water heat sink at a

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maximum normal ambient wet bulb temperature as defined by Chapter 2, Table 2-1. The system is designed to perform this function based on the following:

- The assumed heat load is based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the fuel pool, plus one full core placed in the pool at 120 hours after shutdown. The time during the plant operating cycle at which the full core off-load occurs is chosen to maximize the required spent fuel pool cooling system heat load.

- The spent fuel pool cooling system is assumed to function with its full set of equipment available. One train of the normal residual heat removal system (RNS) is also connected to the spent fuel pool and provides cooling as described in Subsection 5.4.7.4.5.

- While the reactor is shut down, the CCS supply temperature used to determine the performance of the SFS heat exchangers is based on a service water system heat sink with a maximum normal ambient design wet bulb temperature as defined in Chapter 2, Table 2-1.

- While the reactor is at power, the CCS supply temperature to the SFS heat exchangers is based on a service water system heat sink with a maximum safety ambient design wet bulb temperature as defined in Chapter 2, Table 2-1.

- Assuming the heat load scenario identified above, the spent fuel pool water temperature will be < 140°F with one SFS cooling train and one RNS train in operation.

9.1.3.1.4 Spent Fuel Pool Purification

The spent fuel pool cooling system removes radioactive corrosion products, fission product ions and dust to maintain low spent fuel pool (SFP) activity levels and to maintain water clarity during all modes of plant operation. The spent fuel pool cooling system purification capability is such that the occupational radiation exposure (ORE) is minimized to support as-low-as-reasonably-achievable (ALARA) goals. The spent fuel pool cooling system clarification capability is sufficient to permit necessary operations that must be conducted in the spent fuel pool area. The spent fuel pool cooling system is designed to perform its purification function in accordance with the following additional criteria:

- The spent fuel pool cooling system is designed to limit exposure rates to personnel on the spent fuel pool fuel handling machine to less than 2.5 millirem per hour.

- The spent fuel pool cooling system flow rate for one train shall be more than that necessary to provide two water volume changes in 24 hours for the spent fuel pool water.

9.1.3.1.5 Refueling Cavity Purification

The spent fuel pool cooling system removes radioactive corrosion products, fission product ions and dust to maintain low refueling cavity activity levels and to maintain water clarity during refueling operations. The spent fuel pool cooling system purification capability is such that the occupational radiation exposure is minimized to support ALARA goals. Furthermore, the spent fuel pool cooling system clarification capability is sufficient to permit necessary refueling operations that must be conducted in the refueling cavity. The spent fuel pool cooling system is designed to perform its purification function in accordance with the following additional criterion:

- The spent fuel pool cooling system is designed to limit exposure rates to personnel on the refueling machine to less than 2.5 millirem per hour. This corresponds to an activity level in the water of approximately 0.005 microcurie per gram for the dominant gamma-emitting isotopes at the time of refueling.

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9.1.3.1.6 Water Transfers

The spent fuel pool cooling system is designed to transfer water from the in-containment refueling water storage tank to the refueling cavity prior to a refueling and then back to the in-containment refueling water storage tank upon completion of the refueling operations. The spent fuel pool cooling system is designed to perform this function in accordance with the AP1000 refueling schedule.

9.1.3.1.7 In-Containment Refueling Water Storage Tank Purification

The spent fuel pool cooling system removes radioactive corrosion products and fission ions to maintain low in-containment refueling water storage tank activity levels during normal plant operation prior to a scheduled refueling. The spent fuel pool cooling system is designed to maintain the water in the in-containment refueling water storage tank consistent with activity requirements of the water in the refueling cavity during a refueling.

9.1.3.1.8 Spent Fuel Pool Water Tritium Concentration Control

The concentration of tritium in the spent fuel pool water is maintained at less than 0.5 μCi/g to provide confidence that the airborne concentration of tritium in the fuel handling area is within 10 CFR 20, Appendix B limits (see Subsection 12.2.2). The tritium concentration in the spent fuel pool is reduced, if necessary, by transferring a portion of the spent fuel pool water to the liquid radwaste system for discharge and replacing it with non-tritiated water.

9.1.3.2 System Description

The spent fuel pool cooling system is a non-safety-related system. The safety-related function of cooling and shielding the fuel in the spent fuel pool is performed by the water in the pool. A simplified sketch of the spent fuel pool cooling system is included as Figure 9.1-5. The piping and instrumentation diagram for the spent fuel pool cooling system is Figure 9.1-6.

The spent fuel pool cooling system consists of two mechanical trains of equipment. Each train includes one spent fuel pool pump, one spent fuel pool heat exchanger, one spent fuel pool demineralizer and one spent fuel pool filter. The two trains of equipment share common suction and discharge headers. In addition, the spent fuel pool cooling system includes the piping, valves, and instrumentation necessary for system operation.

The spent fuel pool cooling system is designed such that either train of equipment can be operated to perform any of the functions required of the spent fuel pool cooling system independently of the other train. One train is continuously cooling and purifying the spent fuel pool while the other train is available for water transfers, in-containment refueling water storage tank purification, or aligned as a backup to the operating train of equipment.

Each train is designed to process spent fuel pool water. Each pump takes suction from the common suction header and discharges directly to its respective heat exchanger. The outlet piping branches into parallel lines. The purification branch is designed to process approximately 20% of the cooling flow while the bypass branch passes the remaining.

Each purification branch is routed directly to a spent fuel pool demineralizer. The outlet of the demineralizer is to a spent fuel pool filter. The outlet of the filter is then connected to the bypass branch which forms a common line that connects to the discharge header.

The spent fuel pool cooling system suction header is connected to the spent fuel pool at two locations. The main suction line connects to the spent fuel pool at an elevation 6 feet below the operating deck. Two skimmer connections take suction from the water surface of the spent fuel pool.

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This suction arrangement prevents the spent fuel pool from inadvertently being drained below a level that would prevent the water in the spent fuel pool from performing its safety-related function. This arrangement also eliminates the need for a separate skimmer circuit arrangement.

The spent fuel pool pump suction header is connected to the in-containment refueling water storage tank and the refueling cavity. This enables purification of the in-containment refueling water storage tank or the refueling cavity and allows for the transfer of water between the in-containment refueling water storage tank and the refueling cavity.

The spent fuel pool pump suction header is also connected to the fuel transfer canal and the cask loading pit. These connections are provided primarily for the transfer of water from the fuel transfer canal to the cask loading pit. Water that is normally stored in the fuel transfer canal can be sent to the cask loading pit and vice versa.

The spent fuel pool is initially filled for use with water having a nominal boron concentration of 2700 ppm. Demineralized water can be added for makeup purposes, including replacement of evaporative losses, from the demineralized water transfer and storage system. Boron may be added to the spent fuel pool from the chemical and volume control system.

The spent fuel pool water may be separated from the water in the transfer canal by a gate. The gate enables the transfer canal to be drained to permit maintenance of the fuel transfer equipment.

9.1.3.3 Component Description

The general descriptions and summaries of the design requirements for the spent fuel pool cooling system components are provided below. See Table 9.1-2. The key equipment parameters for the spent fuel pool cooling system components are contained in Table 9.1-3. Additional information regarding the applicable codes and classifications is also available in Section 3.2.

9.1.3.3.1 Spent Fuel Pool Pumps

Two spent fuel pool pumps are provided. These pumps are single stage, horizontal, centrifugal pumps having a coupled pump motor shaft driven by an ac powered induction motor. A mechanical seal is used to prevent leakage to the atmosphere. The pumps have flanged suction and discharge nozzles.

Each pump is sized to provide the flow required by its respective heat exchanger for removal of its design basis heat load. The pumps are redundant for normal refueling heat loads.

9.1.3.3.2 Spent Fuel Pool Heat Exchangers

Two spent fuel pool heat exchangers are installed to provide redundant spent fuel heat removal capability for normal refueling heat loads. These heat exchangers are plate type heat exchangers constructed of austenitic stainless steel. Spent fuel pool water circulates through one side of the heat exchanger while component cooling water (CCW) circulates through the other side.

9.1.3.3.3 Spent Fuel Pool Demineralizers

Two mixed bed type demineralizers are provided to maintain spent fuel pool purity. The demineralizers are initially charged with a hydrogen type cation resin and hydroxyl type anion resin to remove fission and corrosion products. The demineralizers will be borated during initial operation with boric acid. Each demineralizer is sized to accept the maximum purification flow from its respective cooling train. The vessels are constructed of austenitic stainless steel.

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9.1.3.3.4 Spent Fuel Pool Filters

Two spent fuel pool filters are provided, one downstream of each demineralizer in the purification branch line of each mechanical train. The filters are sized to collect small particulates and resin fines passed by the demineralizer. They are also sized to pass the maximum design purification flow. The filter assembly is constructed of austenitic stainless steel with disposable filter cartridges.

9.1.3.3.5 Spent Fuel Pool Cooling System Valves

Spent fuel pool cooling system valves operate in low temperature and pressure service. Commercially available valves are used in accordance with the codes and standard of Section 3.2. The basic material of construction is stainless steel.

9.1.3.3.5.1 Locked-In-Position Valves

Refueling Cavity Drain Isolation Valve

There is one locked-open valve in the line from the refueling cavity to the steam generator 2 compartment. This valve is provided so that water in the refueling cavity cannot be trapped and be unavailable for passive recirculation cooling by the passive core cooling system (PXS) following an accident. This valve is locked-closed during refueling operations when the refueling cavity is flooded.

Refueling Cavity Connection for Containment Flooding Isolation Valve

There is one locked-open valve in the line that goes through the wall of the refueling canal to provide a water flow path between the refueling canal and the containment floodup water volume following an accident. This valve is locked open so that as the containment floods, the refueling canal will flood before the compartments that contain passive core cooling system components, which are used for safe shutdown. This valve is locked-closed during refueling operations when the refueling cavity is flooded.

Fuel Transfer Canal Drain Valve

There is one locked-closed valve in the bottom connection to the fuel transfer canal. This valve is provided to prevent inadvertent lowering of the spent fuel pool water level in the event that the gate between the fuel transfer canal and spent fuel pool is open during a seismic event that causes a break in the downstream piping.

Spent Fuel Pool Cooling Pump to Passive Core Cooling Discharge Valves

There are two locked-open valves in the spent fuel pool cooling system pump discharge to the passive core cooling system. These locked-open valves allow the containment isolation check valve to open and prevent a potential pressure buildup inside the containment penetration piping. These valves are locked open during normal and shutdown operations except for in-service testing of the containment isolation valves.

9.1.3.3.5.2 Remotely-Operated Valves

Containment Isolation Valves

The spent fuel pool cooling system contains two lines which penetrate containment. They are the lines from the refueling cavity/in-containment refueling water storage tank to the spent fuel pool cooling system suction header and the return line to the refueling cavity/in-containment refueling water storage tank. Two remotely operated valves, one located inside and one outside containment, are provided in the line to the suction header. One remotely operated valve located outside containment and one check valve located inside containment are provided in the return line. These valves are normally closed and are opened only for purification or water transfers between the in-

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containment refueling water storage tank and the refueling cavity. They are controlled from the main control room. See Subsection 6.2.3.

9.1.3.3.6 Piping Requirements

The portions of the spent fuel pool cooling system rigid piping shown in Figure 9.1-6 are made of austenitic stainless steel. The piping joints and connections in this portion of the system are welded, except where flanged connections or threaded caps are required as indicated on the spent fuel pool cooling system piping and instrumentation diagram (Figure 9.1-6).

9.1.3.3.7 Permanent Cavity Seal Ring

The AP1000 permanent cavity seal ring is part of the fuel handling system and is a permanent welded seal ring used to provide the seal between the vessel flange and the refueling cavity floor. The permanent cavity seal ring does not use pneumatic seals and is not subject to a gross failure due to loss of a seal.

Leakage is not expected with this design. Leakage past or through the seal would not significantly affect the water level in the refueling canal and would be detected as an increase in water level in the containment sump. Water level in the sump is a key parameter in reactor coolant leak detection.

9.1.3.3.8 Reactor Cavity Connections

The spent fuel pool cooling system contains connections to the refueling cavity to prevent excessive holdup of water in the reactor cavity following an accident. The piping connection facilitates draining of the reactor cavity to the steam generator compartment following a postulated accident. The line connects at the bottom of the reactor cavity and discharges to a steam generator compartment, and contains a manual locked-open isolation valve and two check valves in series. The isolation valve is closed during refueling operations to facilitate flooding of the reactor cavity for refueling operations.

The spent fuel pool cooling system also contains a connection between the refueling cavity and Room 11300 to provide a water flow path following an accident. This connection is a single pipe through the wall of the refueling cavity and contains a manual locked-open isolation valve. This connection is provided so that as the containment floods, the refueling canal will flood before the compartments that contain passive core cooling system components, which are used for safe shutdown. Subsection 3.4.1.2.2.1 provides a discussion of post-accident containment flooding. The isolation valve is locked-closed during refueling operations to enable flooding of the reactor cavity for refueling operations.

Other connections are provided to the refueling cavity to facilitate proper draining, filling, and purification of the reactor cavity to support refueling operations.

9.1.3.4 System Operation and Performance

The operation of the spent fuel pool cooling system for the pertinent phases of plant operation are described in the following paragraphs.

9.1.3.4.1 Normal Operation

During normal plant operation, one spent fuel pool cooling system mechanical train of equipment is operating. The operating train is aligned to provide spent fuel pool cooling and purification. The other train is available to perform the other functions of the spent fuel pool cooling system such as water transfers or in-containment refueling water storage tank purification.

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9.1.3.4.1.1 Ion Exchange Media Replacement

The initial and subsequent fill of ion exchange media is made through a resin fill nozzle on the top of the ion exchange vessel. When the media is ready to be transferred to the solid radwaste system, the vessel is isolated from the process flow. The flush water line is opened to the sluice piping and demineralized water is pumped into the vessel through the normal process outlet connection upward through the media retention screen. The media fluidizes in the upward, reverse flow. When the bed has been fluidized, the sluice connection is opened and the bed is sluiced to the spent resin tanks in the solid radwaste system (WSS). Demineralized water flow continues until the bed has been removed and the sluice lines are flushed clean of spent resin.

9.1.3.4.1.2 Filter Cartridge Replacement

Replacement of spent filter cartridges is performed as described in Subsection 11.4.2.3.2.

9.1.3.4.2 Refueling

Both spent fuel pool mechanical trains are in operation during refueling. One train is aligned for spent fuel pool cooling and purification throughout the refueling. The other train performs various support functions during the refueling.

Initially the standby mechanical train is used to purify the water in the in-containment refueling water storage tank to prepare for the refueling. When the refueling cavity is ready to be flooded, the pump aligned for in-containment refueling water storage tank purification is stopped and valves are aligned to gravity drain the in-containment refueling water storage tank to the refueling cavity. Eventually the drain rate slows down and the in-containment refueling water storage tank and the refueling cavity have the same water level. At this time, the standby spent fuel pool pump is aligned to transfer the additional in-containment refueling water storage tank water into the refueling cavity.

This water transfer method improves water clarity in the refueling cavity during refueling operations as compared to conventional pressurized water reactors that have performed this function with their residual heat removal system by flooding up through the reactor vessel into the refueling cavity.

Once the refueling cavity is flooded, the standby mechanical train is re-aligned to cool and purify the refueling cavity. This mode of operation continues as needed. If the heat load is such that both pumps and heat exchangers are needed to cool the spent fuel pool, then the spent fuel pool cooling system can be aligned for that operation.

At the completion of the refueling, the standby spent fuel pool pump is used to transfer the water in the refueling cavity back to the in-containment refueling water storage tank. Once this is complete, the standby train can be aligned to cool the spent fuel pool or may be placed in standby.

9.1.3.4.3 Abnormal Conditions

The AP1000 spent fuel pool cooling system is not required to operate to mitigate design basis events. In the event the spent fuel pool cooling system is unavailable, spent fuel cooling is provided by the heat capacity of the water in the spent fuel pool and its connected safety-related spent fuel pool makeup water sources for the first 72 hours following a loss of normal spent fuel pool cooling. The spent fuel pool connected safety-related makeup water sources include the fuel transfer canal, the cask loading pit, and the cask washdown pit and can include PCS when it is not required to be dedicated for containment cooling. Connections to the spent fuel pool are made at an elevation to preclude the possibility of inadvertently draining the water in the pool to an unacceptable level.

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In the unlikely event of an extended loss of normal spent fuel pool cooling, the water level will drop. Low spent fuel pool level alarms in the control room will indicate to the operator the need to initiate makeup water to the pool. These alarms are provided from safety-related level instrumentation in the spent fuel pool. With the use of makeup water, the pool level is maintained above the spent fuel assemblies for at least 7 days. Initial spent fuel pool water level is controlled by technical specifications. During the first 72 hours any required makeup water is supplied from safety related sources. If makeup water beyond the safety related sources is required between 72 hours and 7 days, water from the passive containment cooling ancillary water storage tank is provided to the spent fuel pool. The amount of makeup required to provide the 7 day capability depends on the decay heat level of the fuel in the spent fuel pool and is provided as follows:

- When the calculated decay heat level in the spent fuel pool is less than or equal to 4.0 MWt, only the safety-related makeup from the fuel transfer canal is needed to achieve spent fuel pool cooling for at least 72 hours.

- When the calculated decay heat level in the spent fuel pool is greater than 4.0 MWt and less than or equal to 5.0 MWt, safety related makeup from the fuel transfer canal and cask washdown pit is sufficient to achieve spent fuel pool cooling for at least 72 hours. A minimum level of 13.75 feet in the cask washdown pit is provided for this purpose. Availability of the makeup source is controlled by technical requirements manual.

- When the calculated decay heat level in the spent fuel pool is greater than 5.0 MWt and less than or equal to 7.0 MWt, safety-related makeup from the fuel transfer canal, cask washdown pit and cask loading pit is sufficient to achieve spent fuel pool cooling for at least 72 hours. A minimum level of 13.75 feet in the cask washdown pit and 43.9 feet in the cask loading pit is provided for this purpose. Availability of the makeup sources is controlled by technical requirements manual.

- When calculated decay heat level in the spent fuel pool is greater than 7.0 MWt makeup from the passive containment cooling water storage tank or passive containment cooling ancillary water storage tank, or combination of the two tanks, is sufficient to achieve spent fuel pool cooling for at least 7 days.

- When the decay heat level in the reactor is less than or equal to 7.0 MWt, the passive containment cooling water storage tank is not needed for containment cooling and this water can be used for makeup to the spent fuel pool. This tank provides safety related makeup for at least 72 hours. Between 72 hours and 7 days the tank continues to provide makeup water as required until it is empty. If the passive containment cooling water storage tank empties in less than 7 days, non-safety makeup water can be provided from the passive containment cooling ancillary water storage tank.

- When the decay heat level in the reactor is greater than 7.0 MWt, the water in the passive containment cooling water storage tank is reserved for containment cooling. Safety related spent fuel pool cooling is provided for at least 72 hours from the pool itself and makeup water from the cask washdown pit and cask loading pit. After 72 hours, non-safety related makeup can be provided from the passive containment cooling ancillary water storage tank.

- Minimum volume in the passive containment cooling water storage tank for spent fuel pool makeup is 756,700 gallons. Availability of this makeup source for the first 72 hours is controlled by technical requirements manual. Minimum volume in the passive containment cooling ancillary water storage tank for spent fuel pool makeup is 201,600 gallons.

Table 9.1-4 provides the calculated timing and spent fuel pool water levels for several limiting event scenarios which would require makeup to the spent fuel pool.

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Alignment of the fuel transfer canal is accomplished by opening the gate, shown in Figure 9.1-6, located between the fuel transfer canal and spent fuel pool. Alignment of the cask washdown pit is accomplished by positioning manual valves located in the waste monitor tank room B (12365) in the auxiliary building. Alignment of the passive containment cooling water storage tank is accomplished by positioning manual valves located in the maintenance floor staging area (12351) and in the passive containment cooling valve room in the upper shield building. Because these alignments are made by positioning manual valves, they are not susceptible to active failures.

Alignment of the cask loading pit is accomplished by opening the gate, shown in Figure 9.1-6, located between the cask loading pit and the spent fuel pool. The cask loading pit gate should be opened prior to exceeding 5.0 MWt in the spent fuel pool.

Gravity driven flow from the cask washdown pit to the spent fuel pool is provided as the cask washdown pit water level will follow the spent fuel pool level. The cask washdown pit is connected to both the spent fuel pool for spent fuel pool cooling needs and to the liquid radwaste system (WLS) for temporary storage of WLS effluents (Section 11.2). However, the cask washdown pit can only perform either spent fuel pool cooling or temporary storage of WLS effluents at a time. Figure 9.1-6 shows the connection of the cask washdown pit to the spent fuel pool and WLS.

Gravity driven flow from the passive containment cooling water storage tank is controlled by a manual throttle valve with local flow indication which is set to achieve the desired flow when the makeup is initiated. Figure 6.2.2-1 shows the flow path from the passive containment cooling water storage tank leading to the spent fuel pool and the tie-in to the spent fuel pool is also shown in Figure 9.1-6.

The flow from the passive containment cooling water storage tank (PCCWST) to the spent fuel pool, required to provide sufficient makeup to the spent fuel pool to keep the fuel covered as the pool water boils off, is 118 gpm. This is the maximum flow required at the initiation of makeup flow from the PCCWST during the worst case conditions in the pool, which is a full core offload. The makeup flow rate required decreases with time as the decay heat decreases.

After 72 hours, makeup water from the passive containment cooling ancillary water storage tank can either be pumped (with the passive containment cooling recirculation pumps) to the passive containment cooling water storage tank and then gravity fed to the spent fuel pool as discussed above, or the water can be pumped directly to the spent fuel pool. When the makeup water is pumped directly to the pool, the flow rate is controlled by the same manual throttle valve which is used to set the flow rate when providing gravity driven flow from the passive containment cooling water storage tank.

The flow rates provided from the passive containment cooling ancillary water storage tank (PCCAWST) to the spent fuel pool by the recirculation pumps depend on the plant condition and spent fuel decay heat. Postulated events require SFP makeup water from the PCCAWST 72 hours following a seismic event combined with a station blackout. The required flow rates provide sufficient makeup to the spent fuel pool to keep the fuel covered as the water boils off. A 35 gpm flow rate is required when the plant has been operating at full power, shortly after startup from a refueling outage and the spent fuel pool decay heat is ≤ 5.0 MWt. A 50 gpm flow rate is required when the plant is being refueled and the spent fuel pool decay heat is > 5.0 MWt and ≤ 7.0 MWt. An 80 gpm flow rate is required when the plant is being refueled and the spent fuel pool decay heat level is > 7.0 MWt. The refueling scenarios consider the time between completion of plant cooldown and just prior to plant startup once the refueling is complete. With a refueling scenario, additional decay heat is located in the spent fuel pool because of the recent offload and enough decay heat remains in the reactor vessel such that the PCCWST is still required for containment cooling and cannot be used for spent fuel pool makeup. These conditions result in the maximum flow required from the PCCAWST because cooling water must be supplied to both the PCCWST and the spent fuel pool to provide both

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containment and spent fuel cooling for a period of four days following the initial three days of passive systems operation.

Spent fuel pool level instrumentation is discussed in Subsection 9.1.3.7.

9.1.3.4.3.1 Failure of a Spent Fuel Pool Cooling System Pump

If a spent fuel pool cooling system pump fails when only one pump is operating, an alarm is actuated. Due to the heat capacity of the water in the spent fuel pool, sufficient time exists for the operators to manually align the standby spent fuel pool cooling system train of equipment (pump/heat exchanger) to cool the spent fuel pool.

9.1.3.4.3.2 Leakage from the Spent Fuel Pool Cooling System

The connections from the spent fuel pool cooling system to the pool are such that leakage in the spent fuel pool cooling system will not result in the pool water level falling to unacceptable levels. The heat capacity of the water in the pool is sufficient to allow the operators enough time to locate the leak and repair it. In the most probable scenario, cooling will be maintained by operation of the standby train of equipment. However, if spent fuel pool cooling must be terminated, sufficient time exists to allow for repairs of a leak in the system.

9.1.3.4.3.3 Loss of Normal AC Power Operation and Offsite Power

The spent fuel pool cooling system pumps can be manually loaded on the respective onsite standby diesel generator in the event of a loss of normal ac power and offsite power. The spent fuel pool cooling system is capable of providing spent fuel pool cooling following this event.

9.1.3.4.3.4 Station Blackout

Following a loss of all ac power, the heat capacity of the water in the pool is such that cooling of the fuel is maintained. Table 9.1-4 provides the times before boiling would occur in the pool following station blackout for various scenarios as well as the minimum levels of water that would be reached. Water vapor that evaporates from the surface of the spent fuel pool is vented to the outside environment through an engineered relief panel. This vent path maintains the fuel handling area at near atmospheric pressure conditions. The doses resulting from spent fuel pool boiling have been calculated and are included in Chapter 15. The release concentrations at the site boundary are small fractions of the limits specified in 10 CFR 20, Appendix B with no credit for removal of activity by building ventilation systems (which are not available during loss of ac power situations). The equipment in the fuel handling area, rail car bay\filter storage area, and spent resin equipment and piping areas exposed to elevated temperature and humidity conditions as a result of pool boiling does not provide safety-related mitigation of the effects of spent fuel pool boiling or station blackout. The fuel handling area, rail car bay, and spent resin area do not have connecting ductwork with other areas of the radioactively controlled area of the auxiliary building and connecting floor drains have a water seal which prevents steam migration. The environment in these other areas during spent fuel pool steaming is mild with respect to safety-related equipment qualification and affords access for post-accident actions.

Spent fuel pool makeup for long term station blackout can be provided through seismically qualified safety-related makeup connections from the passive containment cooling system. These connections are located in an area of the auxiliary building that can be accessed without exposing operating personnel to excessive levels of radiation or adverse environmental conditions during boiling of the pool. Operating personnel are not required to enter the fuel handling area when normal cooling is not available, and are not required to enter the area to recover normal cooling.

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9.1.3.4.3.5 Reactor Coolant System Makeup

During an event in which the reactor coolant system pressure and inventory decrease the normal residual heat removal system pumps are started to provide makeup water to the reactor coolant system when the primary system pressure is sufficiently reduced for injection to start. The AP1000 procedure for post-accident operation of the normal residual heat removal pumps is that the operators align the pumps to the cask loading pit. This is accomplished by the operator opening a motor operated isolation valve (see Subsection 5.4.7.3.3.5) between the cask loading pit and the normal residual heat removal pump suction line. When the water in this pit nears empty, the pump suction is re-aligned to the IRWST/containment recirculation connection so that the pumps can continue to provide injection to the reactor coolant system. The refueling water from the cask loading pit provides additional water into containment (and thus additional driving head) for the post accident containment recirculation. The AP1000 emergency operating procedures will include a restriction on use of this injection method if the gate between the spent fuel pool and the cask loading pit is open at the initiation of the event. In this case the operators will be instructed to close the gate, if possible, before initiating the makeup flow with the normal residual heat removal pumps. Injection from the cask loading pit will only be initiated if the gate can be closed. The gate is normally in the closed position unless cask loading or refueling operations are in progress.

9.1.3.5 Safety Evaluation

The only spent fuel pool cooling system safety-related functions are containment isolation and emergency makeup connections to the spent fuel pool. Containment isolation evaluation is described in Subsection 6.2.3. The following provides the evaluation of the design of the spent fuel pool as well as the spent fuel pool cooling system:

- The spent fuel pool is designed such that a water level is maintained above the spent fuel assemblies for at least 7 days following a loss of the spent fuel pool cooling system, using only onsite makeup water (see Table 9.1-4). The minimum water level to achieve sufficient cooling is the subcooled, collapsed level (without vapor voids) required to cover the top of the fuel assemblies.

- The maximum heat load is assumed to be the heat load for a full core off load immediately following a refueling in which 44 percent of the fuel assemblies were replaced.

- Safety-related makeup water can be supplied to the fuel pool from the fuel transfer canal, cask washdown pit, cask loading pit, and passive containment cooling water storage tank.

- The spent fuel pool cooling system includes safety-related connections from the passive containment cooling water storage tank in the passive containment cooling system to establish safety-related makeup to the spent fuel pool following a design basis event including a seismic event.

- In addition to the safety-related water sources, makeup water is also obtained from the passive containment cooling ancillary water storage tank. Water from this tank can be pumped by the passive containment cooling system recirculation pumps either to the passive containment cooling water storage tank (and then gravity fed to the spent fuel pool), or directly to the spent fuel pool.

Radiation shielding normally provided by the water above the fuel is not required when normal spent fuel pool cooling is not available. Personnel are not permitted in the area when the level in the pool is below the minimum level.

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The acceptability of the design of the spent fuel pool cooling system is based on specific General Design Criteria (GDCs) and Regulatory Guides as described in Sections 3.1 and 1.9.

9.1.3.6 Inspection and Testing Requirements

9.1.3.6.1 Preoperational Testing, Analysis, and Inspection

9.1.3.6.1.1 Pump Flow Capability Testing

Each spent fuel pool cooling system pump will be tested. The flow paths will be aligned for normal spent fuel pool cooling by one train of spent fuel pool cooling system components. The flow delivered to each spent fuel pool cooling system heat exchanger will be measured by a flow instrument at the spent fuel pool cooling system pump discharge. The testing confirms that the pumped flow is equal to or greater than the minimum value shown in Table 9.1-3. This is the minimum value for the spent fuel pool cooling system to meet its functional requirement of normal spent fuel pool cooling.

9.1.3.6.1.2 Heat Transfer Capability Analysis

An analysis will be performed on the spent fuel pool cooling system heat exchangers during heat exchanger design. The analysis is to confirm that the product of the overall heat transfer coefficient and effective heat transfer area, UA, of each heat exchanger is equal to or greater than the minimum value shown in Table 9.1-3. This is the minimum value for the spent fuel pool cooling system to meet its functional requirement of normal spent fuel pool cooling.

9.1.3.6.1.3 Dimensional Inspections

The contained volumes of water in the spent fuel pool, fuel transfer canal, the cask washdown pit, and the cask loading pit are used for cooling the spent fuel by boiling after a prolonged loss of normal spent fuel pool cooling. The inspections are to confirm that the contained volumes are equal to or greater than the minimum values shown in Table 9.1-2. These are the minimum values for the spent fuel pool cooling system to meet its safety-related requirement of spent fuel pool cooling for 3 days after loss of normal cooling.

9.1.3.6.2 Routine Testing

Active components of the spent fuel pool cooling system are either in continuous or intermittent use during normal system operation. Periodic visual inspection and preventive maintenance are conducted.

No specific equipment tests are required since system components are normally in operation when spent fuel is stored in the fuel pool. Sampling of the fuel pool water for gross activity, tritium and particulate matter is conducted periodically.

9.1.3.6.3 System Gas Accumulation Assessment and Mitigation

Subsection 6.3.6.3.1 describes the assessment method to address the potential for gas intrusion for the passive core cooling system in response to Interim Staff Guidance (ISG-019) and Generic Letter 2008-01. The same assessment methodology was used to assess the spent fuel pool cooling system.

The spent fuel pool cooling system locations equipped with manual vent valves will be inspected according to the system surveillance and venting procedures described in Subsection 6.3.6.3 to eliminate identified gas accumulations.

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9.1.3.7 Instrumentation Requirements

The instrumentation provided for the spent fuel pool cooling system is discussed in the following paragraphs. Alarms and indications are provided as noted.

A. Temperature

Instrumentation is provided to measure the temperature of the water in the spent fuel pool and to give indication as well as annunciation in the main control room when normal temperatures are exceeded.

Instrumentation is also provided to give indication of the temperature of the spent fuel pool water as it leaves either heat exchanger.

B. Pressure

Instrumentation is provided to measure and give indication of the pressures in the spent fuel pool pump suction and discharge lines. Instrumentation is also provided at locations upstream and downstream from the spent fuel pool filter and demineralizer so that pressure differential across this equipment can be determined. High differential pressure across the spent fuel pool filter and demineralizer is annunciated in the main control room.

C. Flow

Instrumentation is provided to measure and give remote indication of the spent fuel pool cooling loop flow downstream of the spent fuel pool pumps. Purification flow is also continuously measured.

D. Level

Safety-related instrumentation is provided to give an alarm in the main control room when the water level in the spent fuel pool reaches the Low-2 level setpoint. This instrumentation is used for post-accident monitoring on the spent fuel pool level. (See Table 7.5-1) All three safety-related spent fuel pool level instruments and associated instrument tubing lines are located below the fuel handling area operating deck and the cask washdown pit. This location provides protection from missiles that may result from damage to the structure over the spent fuel pool. The SFP level instruments associated with PMS divisions A and C are physically separated from the SFP level instrument associated with PMS division B. The safety-related spent fuel pool level instruments measure the water level from the top of the spent fuel pool to the top of the fuel racks. These instruments are conservatively calibrated at a reference temperature suitable for normal spent fuel pool operation on a regular basis and accuracy is not affected by power interruptions.

Non-safety-related instrumentation is provided to give an alarm in the main control room when the water level in the spent fuel pool and the cask loading pit reaches either the High level or Low level setpoint. Instrumentation in the spent fuel pool is used to alert the operator that the pool is nearing limits for fuel movement. Instrumentation in the cask loading pit is used to alert the operator to a Low-2 level when injecting water from the pit into the reactor coolant system with the normal residual heat removal pumps.

9.1.4 Light Load Handling System (Related to Refueling)

The fuel handling and refueling system consists of equipment and structures used for conducting the refueling operation. This system conforms to General Design Criteria as defined in Section 3.1. The

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light load handling system meets the guidelines of American Nuclear Society (ANS) 57.1 (Reference 6). Figures 1.2-9 and 1.2-14 indicate the relationship between the light load handling system and the fuel handling areas.

9.1.4.1 Design Basis

9.1.4.1.1 Safety Design Basis

The following safety design basis apply to the light load handling system:

A. Fuel handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operation.

B. Handling equipment has provisions to avoid dropping of fuel handling devices during the fuel transfer operation.

C. Handling equipment used to raise and lower spent fuel has a limited maximum lift height so that the minimum required depth of water shielding is maintained.

D. The fuel transfer system, where it penetrates the containment, has provisions to preserve the integrity of the containment pressure boundary.

E. Criticality during fuel handling operations is prevented by the geometrically safe configuration of the fuel handling equipment.

F. In the event of a safe shutdown earthquake (SSE), handling equipment cannot fail in such a manner as to prevent required function of seismic Category 1 equipment. The refueling machine is designed to maintain its load carrying and structural integrity functions during a safe shutdown earthquake.

G. The inertial loads imparted to the fuel assemblies or core components during handling operations are less than potential damage causing loads.

H. Physical safety features are provided for personnel who operate handling equipment.

9.1.4.1.2 Power Generation Design Basis

Design criteria for the light load handling system are as follows:

A. The primary design requirement of the equipment is reliability. A conservative design approach is used for load bearing parts.

B. The refueling machine and fuel handling machine are designed and constructed in accordance with applicable portions of the Crane Manufacturers Association of America, Inc. (CMAA), Specification 70 for Class A-1 service (Reference 7).

C. The static design loads for the crane structures and lifting components are normal dead and live loads plus the fuel assembly weight.

D. The allowable stresses for the refueling machine and fuel handling machine structures supporting the weight of a fuel assembly are as specified in the American Institute of Steel Construction (AISC) Manual.

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E. The design load on the wire rope hoisting cables does not exceed 0.20 times the average breaking strength. Two independent cables are used, and each is assumed to carry one half the load.

F. Components critical to the operation of the equipment are assembled with the fasteners restrained from loosening under vibration.

9.1.4.2 System Description

The light load handling system consists of the equipment and structures needed for the refueling operation. This equipment is comprised of fuel assemblies, core component and reactor component hoisting equipment, handling equipment, and a fuel transfer system. The structures associated with the fuel handling equipment are the refueling cavity, the transfer canal, the fuel transfer tube, the spent fuel pool, the cask loading area, the new fuel storage area, and the new fuel receiving and inspection area.

9.1.4.2.1 Fuel Handling Description

The fuel handling equipment is designed to handle the spent fuel assemblies underwater from the time they leave the reactor vessel until they are placed in a container for shipment from the site. Underwater transfer of spent fuel assemblies provides an effective and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. The boric acid concentration in the water is sufficient to preclude criticality.

The associated fuel handling structures may be generally divided into two areas: the refueling cavity which is flooded only during plant shutdown for refueling, and the spent fuel pool and transfer canal, which is kept full of water. See Subsection 9.1.1.3 for new fuel assembly storage. The new and spent fuel storage areas are accessible to operating personnel. The refueling cavity and the fuel storage area are connected by the fuel transfer tube which is fitted with a quick opening hatch/blind flange on the canal end and a valve on the fuel storage area end. The hatch/blind flange is in place except during refueling to provide containment integrity. Fuel is carried through the tube on an underwater transfer car.

Fuel is moved between the reactor vessel and the fuel transfer system by the refueling machine. The fuel transfer system is used to move a fuel assembly and its associated core component between the containment building and the auxiliary building fuel handling area. After a fuel assembly is placed in the fuel container, the lifting arm pivots the fuel assembly to the horizontal position for passage through the fuel transfer tube. After the transfer car transports the fuel assembly through the transfer tube, the lifting arm at that end of the tube pivots the assembly to a vertical position so that the assembly can be lifted out of the fuel container.

In the fuel handling area, fuel assemblies are moved about by the fuel handling machine. Initially, a short tool is used to handle new fuel assemblies, but the new fuel elevator must be used to lower the assembly to a depth at which the fuel handling machine can place the new fuel assemblies into or out of the spent fuel storage racks.

Decay heat, generated by the spent fuel assemblies in the fuel pool is removed by the spent fuel pool cooling and cleanup system. After a sufficient decay period, the spent fuel assemblies are removed from the fuel racks and loaded into a spent fuel shipping cask for removal from the site.

9.1.4.2.2 Refueling Procedure

New fuel assemblies received for refueling are removed one at a time from the shipping container and moved into the new fuel assembly inspection area. After inspection, the accepted new fuel

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assemblies are stored in the new fuel storage rack. For the initial core load, some new fuel assemblies may be stored in the spent fuel pool.

Prior to initiating the refueling operation, the reactor coolant system (RCS) is borated and cooled down to refueling shutdown conditions as specified in the Technical Specifications. Criticality protection for refueling operations is specified in the Technical Specifications. The following significant points are addressed by the refueling procedure:

- The refueling water and the reactor coolant contain a nominal boron concentration of 2700 ppm boron. This concentration is sufficient to keep the core five percent Δk/k subcritical during the refueling operations.

- The water level in the refueling cavity is high enough to keep the radiation levels within acceptable limits when the fuel assemblies are removed from the core. Radiation monitoring is described in Section 11.5.

- Continuous communications are established and maintained between the main control room and the personnel engaged in fuel handling operations. One or more of the systems described in Subsection 9.5.2 are used for this communication.

The refueling operation is divided into four major phases: preparation, reactor disassembly, fuel handling, and reactor assembly. A general description of a typical refueling operation through these phases is provided below.

9.1.4.2.2.1 Phase I - Preparation

The reactor is shut down, borated, and cooled to refueling conditions (≤ 140°F) with a final k_{eff} less than 0.95 (all rods in). Following a radiation survey, the containment building is entered. At this time, the coolant level in the reactor vessel is lowered to a point slightly below the vessel flange. The refueling machine console is removed from storage and placed on the refueling machine and cables are connected. Then, the fuel transfer equipment and refueling machine are checked for operation (Subsection 9.1.4.4).

9.1.4.2.2.2 Phase II - Reactor Disassembly

Head cables are disconnected at the integrated head package (IHP) connector plate to allow removal of the vessel head. See Subsection 3.9.7 for a discussion of the integrated head package. The refueling cavity is prepared for flooding by checking the underwater lights, tools, and fuel transfer system; closing the refueling cavity drain lines; and removing the hatch/blind flange from the fuel transfer tube. With the refueling cavity prepared for flooding, the vessel head is unseated and raised above the vessel flange using the containment polar crane. The vessel head is then transported to its storage pedestal. See Subsection 9.1.5 for requirements for the polar crane. Water from the in-containment refueling water storage tank (IRWST) is transferred into the refueling cavity by gravity and the spent fuel pool cooling system (See Subsection 9.1.3). When the refueling cavity water level is at a depth to support the control rod drive shaft uncoupling, the control rod drive shafts are disconnected. The refueling cavity is then flooded to a height to support the installation of the internals lift rig, then the internals lift rig is installed and the upper internals are removed from the vessel. See Subsection 9.1.5 for discussion of lifting rig requirements and design. The fuel assemblies are now free from obstructions, and the core is ready for refueling after the refueling cavity is filled to the safe shielding depth Subsection 9.1.4.3.7.

9.1.4.2.2.3 Phase III - Fuel Handling

The refueling sequence is started with the refueling machine.

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The general fuel handling sequence is as follows:

1. The refueling machine is positioned over a fuel assembly in the core.

2. The refueling machine mast is lowered over a fuel assembly and engages it.

3. The refueling machine withdraws a spent fuel assembly from the core and raises it to a pre-determined height sufficient to clear the vessel flange and still leave sufficient water covering the fuel assembly.

4. The fuel transfer system car is moved into the refueling cavity from the fuel storage area, and the fuel basket is pivoted to the vertical position by the lifting arm.

5. The refueling machine is moved to line up the fuel assembly with the empty fuel basket.

6. The refueling machine loads the spent fuel assembly into the empty fuel basket of the transfer car.

7. The refueling machine then moves back over the core area, and it is aligned over the next fuel assembly to be removed in the core offload sequence.

8. In parallel with item 7 above, the fuel basket is pivoted to the horizontal position and the fuel transfer system container is moved through the fuel transfer tube to the fuel handling area by the transfer car and pivoted to the vertical position.

9. The spent fuel assembly is then unloaded from the fuel basket by the fuel handling machine.

10. The spent fuel assembly is placed in the spent fuel storage rack.

11. The fuel basket is pivoted to the horizontal position, moved back into the containment building and pivoted to the vertical position.

12. This procedure is repeated until the core is offloaded.

13. Core reload is essentially the reverse of the offload sequence described above.

9.1.4.2.2.4 Phase IV - Reactor Reassembly

Reactor reassembly, following refueling, is achieved by reversing the operations given in Phase II - Reactor Disassembly.

9.1.4.2.3 Spent Fuel Cask Loading

The spent fuel assemblies are normally stored in the spent fuel pool, until fission product activity is low enough to permit shipment. The spent fuel assemblies are then transferred to a shipping cask which is designed to shield radiation. Provisions for handling the spent fuel cask are discussed in Subsection 9.1.5.

The following procedure briefly outlines the typical steps of this operation, assuming that the cask loading pit has been previously filled with water and the gate between the cask loading pit and the spent fuel pool has been removed or opened:

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1. The transfer cask containing a clean, empty spent fuel canister is brought into the cask washdown pit. The spent fuel canister is removed as necessary and prepared for cask loading.

2. The transfer cask/spent fuel canister are placed into the cask loading pit.

3. The fuel handling machine is positioned over the specific fuel assembly to be shipped out of the spent fuel storage rack. The fuel assembly is picked up and transported into the cask loading pit. During the transfer process the fuel assembly is always maintained with the top of the active fuel at least 8.75 feet below the water surface. This provides confidence that the direct radiation from the fuel at the surface of the water is minimal.

4. Once the fuel transfer process is complete, the lid is placed on top of the cask to provide the required shielding.

5. The cask is then moved to the washdown pit and cleaned with demineralized water. Decontamination procedures can be started at this time.

6. When the spent fuel canister closure and drying processes are complete, the transfer cask is prepared for transfer into a storage or shipping container as applicable.

During the operations, sufficient water is maintained between plant personnel and fuel assemblies that are being moved to limit dose levels to those acceptable for continuous occupational exposure.

9.1.4.2.4 Component Description

A. Fuel Transfer Tube

The fuel transfer tube penetrates the containment and spent fuel area and provides a passageway for the conveyor car during refueling. During reactor operation, the fuel transfer tube is sealed at the containment end and acts as part of the containment pressure boundary. See Subsection 3.8.2.1.5 for discussion of the fuel transfer penetration.

B. Fuel Handling Machine

The fuel handling machine performs fuel handling operations in the new and spent fuel handling area. It also provides a means of tool support and operator access for long tools used in various services and handling functions. The fuel handling machine is equipped with two 2-ton hoists, one of which is single failure proof and is designed according to NUREG-0554.

The nonsingle-failure-proof hoist is primarily used for submerged handling activities. However, there are areas in the fuel handling area of the auxiliary building that the single-failure-proof hoist is not capable of accessing due to travel limitations. Therefore, it is necessary for the nonsingle-failure-proof hoist to be used in areas other than the spent fuel pool. The nonsingle-failure-proof hoist will be restricted from handling a load above the operating floor within 15 feet of the spent fuel pool. The nonsingle-failure-proof hoist is also restricted from handling new fuel above the operating floor.

The single-failure-proof hoist will be capable of handling loads in the new fuel handling area and the spent fuel handling area with operator warnings associated with the handling of spent fuel.

C. New Fuel Assembly Handling Tool

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The new fuel assembly handling tool is used to lift and transfer new fuel assemblies from the new fuel shipping containers to the new fuel storage rack. The tool is also used to transfer new fuel assemblies from the new fuel storage rack to the new fuel elevator.

D. Spent Fuel Assembly Handling Tool

The spent fuel assembly handling tool is used to lift and transfer spent fuel assemblies from the fuel transfer system to the spent fuel racks and new fuel from the elevator to the spent fuel racks.

E. New Fuel Elevator Hoist

The new fuel elevator lowers new fuel assemblies from the fuel handling area operating floor into the spent fuel pool where they can be picked up by the fuel handling machine.

F. New Rod Cluster Control Handling Tool

The new rod cluster control handling tool is used to lift and transfer new control rods from their shipping containers to the new fuel assemblies, and between new assemblies in the new fuel storage racks.

G. Refueling Machine

The refueling machine performs fuel handling operations in the containment building. It also provides a means of tool support and operator access for long tools used for service, control rod latching and unlatching, and for various handling functions.

H. Wet Annular Burnable Absorber (WABA) Handling Tool

The WABA handling tool is used to lift and transfer burnable absorbers between fuel assemblies and/or storage fixtures.

I. Spent Fuel Rack Storage Insert (SFRSI)

The SFRSI is used to store WABA assemblies or control rods in the spent fuel storage racks.

J. Fuel Transfer System

The fuel transfer system conveys fuel assemblies between the containment building and the auxiliary building fuel handling area.

K. Not used.

L. Control Rod Drive Shaft Unlatching Tool

The control rod drive shaft unlatching tool is used to latch and unlatch the control rod drive shafts from the rod cluster control assemblies. It is operated from the refueling machine walkway.

M. Control Rod Drive Shaft Handling Tool

The control rod drive shaft handling tool is used to latch and unlatch the control rod drive shafts (CRDS) from the rod cluster control assemblies.

N. Irradiation Sample Handling Tool

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The irradiation sample handling tool is used to remove irradiated reactor vessel surveillance capsules in the holders located in the reactor internals. It is also used for removing and installing the irradiation sample access plugs in the reactor internals.

O. Irradiation Tube End Plug Seating Jack

The irradiation tube end plug seating jack is used to push the irradiation samples into the specimen guides for the last few inches.

P. Control Rod Drive Shaft Storage Racks

The control rod drive shaft storage racks are located on the refueling cavity wall and are used to store spare control rod drive shafts and any ones that might be removed from the upper internals during refueling.

9.1.4.3 Safety Evaluation

9.1.4.3.1 Refueling Machine

The refueling machine design includes the following provisions to provide for safe handling of fuel assemblies. These safety features include the safety requirements listed in (ANS) 57.1:

A. Safety Interlocks

Operations which could endanger the operator or damage the fuel, designated below by an asterisk (*), are prevented by mechanical or failure tolerant electrical interlocks or by redundant electrical interlocks. Other interlocks are intended to provide equipment protection and may be implemented either mechanically or by electrical interlock and are not required to be fail safe.

Fail safe electrical design of a control system interlock is applied according to the following rules:

1. Fail safe operation of an electrically operated brake is such that the brake engages on loss of power.

2. Fail safe operation of a relay is such that the de-energized state of the relay inhibits unsafe operation.

3. Fail safe operation of a switch, termination, or wire is such that breakage or high resistance of the circuit inhibits unsafe operation. The dominant failure mode of the mechanical operation of a cam-operated limit switch is sticking of the plunger in its depressed position. Therefore, use of the plunger-extended position (on the lower part of the operating cam) to energize a relay is consistent with fail safe operation.

Those parts of a control system interlock which are not or cannot be operated in a fail safe mode as defined in the preceding rules are supplemented by a redundant component or components to provide the requisite protection. Required fail safe operations are:

*1. The refueling machine can only place a fuel assembly in the core, in the in-containment fuel storage rack, or in the fuel transfer system.

*2. When the refueling machine gripper is engaged, the machine cannot traverse unless the fuel assembly bottom nozzle is clear of the lower core plate alignment pins.

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*3. When the refueling machine gripper is disengaged, the machine cannot traverse unless the gripper is withdrawn into the mast.

*4. Simultaneous traversing and hoisting operations are prevented.

*5. The refueling machine hoist up travel stops at a predetermined height to prevent a spent fuel assembly from being raised above the minimum water depth for shielding.

*6. When a fuel assembly is raised or lowered, interlocks provide confidence that the refueling machine can only apply loads which are within safe operating limits.

*7. The fuel gripper is monitored by devices to confirm operation to the fully engaged or fully disengaged position. Alarms are actuated if both engage and disengage switches are actuated at the same time or if neither is actuated.

8. Lowering of the gripper is not permitted if slack cable exists in the hoist.

9. The inner mast is prevented from lowering completely out of the mast.

10. Before the fuel gripper can release a fuel assembly, the fuel gripper must be in its down position in the core, in the in-containment fuel storage rack, or in the fuel transfer system.

*11. The weight of the fuel assembly must be off the gripper before the fuel gripper can release a fuel assembly.

12. The refueling machine hoist is prevented from moving in the transfer machine zone unless the upender is vertical. An interlock is provided from the fuel transfer system to the refueling machine to accomplish this.

B. Bridge and Trolley Hold-Down Devices

Both refueling machine bridge and trolley are horizontally restrained on the rails by guide rollers on either side of the rail. Hold down devices are used to prevent the bridge or trolley from leaving the rails in the event of a seismic event.

C. Main Hoist Braking System

The main hoist is equipped with two independent braking systems. The winch has a mechanically-operated load brake to prevent overhauling, and a solenoid activated motor brake. Both brakes are rated at 125 percent of the hoist design load.

D. Fuel Assembly Support System.

The main hoist system is supplied with redundant paths of load support such that failure of any one component will not result in free fall of the fuel assembly. Two wire ropes are anchored to the winch drum and carried to a load equalizing mechanism on the top of the inner mast.

The fuel assembly gripper has four fingers gripping the fuel, any two of which will support the fuel assembly weight.

During each refueling outage and prior to removing fuel, the gripper and hoist system are routinely load tested to 125 percent of the maximum setting on the hoist load limit switch.

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9.1.4.3.2 Fuel Transfer System

The following personnel safety features are provided for in the fuel transfer system. These safety features include the safety requirements listed in (ANS) 57.1:

A. Transfer Car Permissive Switch

The transfer car controls are located in the fuel handling area, and conditions in the containment are therefore not visible to the operator. The transfer car permissive switch allows the fuel transfer system containment operator to exercise some control over car movement if conditions visible to him warrant such control.

B. Lifting Arm - Transfer Car Position

An interlock on the fuel transfer system prevents the upender from being moved from the horizontal to the vertical position if the transfer car has not reached the end of its travel.

C. Transfer Car - Valve Open

An interlock on the transfer tube valve permits transfer car operation only when the transfer tube valve position switch indicates the valve is fully open.

D. Fuel Container - Refueling Machine

The fuel transfer system is interlocked with the refueling machine. Whenever the transfer car is located in the refueling cavity, the fuel transfer system cannot be operated unless the refueling machine mast is in the fully retracted position or the refueling machine is over the core.

E. Lifting Arm - Fuel Handling Machine

On the spent fuel pool side, the fuel transfer system is interlocked with the fuel handling machine. The fuel transfer system cannot be operated until the loaded fuel handling machine hoist is at the up limit, the empty tool is clear of the upender, or the fuel handling machine is moved away from the fuel transfer system area. An interlock is provided from the fuel handling machine to the fuel transfer system to accomplish this.

9.1.4.3.3 Fuel Handling Machine

The fuel handling machine design includes the following provisions to provide for safe handling of fuel assemblies and other components within the auxiliary building fuel handling area. These safety features include, but are not limited to the safety requirements listed in (ANS) 57.1 with the exception of grapple release, as this feature is the manual operation of a fuel handling tool:

A. Safety Interlocks

Operations that could endanger the operator or damage the fuel, designated below by an asterisk (*), are prevented by mechanical or failure tolerant electrical interlocks, or by redundant electrical interlocks. Other interlocks are intended to provide equipment protection and may be implemented either mechanically or by electrical interlock and are not required to be fail safe.

Fail safe electrical design of a control system interlock is applied according to the following rules:

1. Fail safe operation of an electrically operated brake is such that the brake engages on loss of power.

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2. Fail safe operation of a relay is such that the de-energized state of the relay inhibits unsafe operation.

3. Fail safe operation of a switch, termination, or wire is such that breakage or high resistance of the circuit inhibits unsafe operation.

Those parts of a control system interlock that are not or cannot be operated in a fail safe mode, as defined in the preceding rules, are supplemented by a redundant component or components to provide the requisite protection. Required fail safe operations are as follows:

*1. The fuel handling machine, and its associated fuel handling tool, can only place a fuel assembly in the new fuel rack, spent fuel racks, fuel transfer system, new fuel elevator, spent fuel cask, fuel inspection/repair station, or rail car bay traveler.

*2. When the hoist load weighing system detects a load greater than the spent fuel assembly handling tool, the machine cannot traverse unless the hoist is at the up limit. For new fuel handling, the load is required to be clear of a structure prior to traversing.

*3. Simultaneous traversing and hoisting operations are prevented.

*4. The fuel handling machine hoist up travel stops at a predetermined height to prevent a spent fuel assembly from being raised above the minimum water depth for shielding. This hoist “up” travel stop predetermined height is met by the mechanical “up” limit of the non-single failure-proof hoist combined with the geometry of the spent fuel assembly handling tool (SFAHT). Together these design features assure the bottom nozzle of the spent fuel assembly remains less than 3 feet above the top of the spent fuel storage rack.

*5. When a fuel assembly is raised or lowered, interlocks provide confidence that the fuel handling machine can apply only loads that are within safe operating limits.

*6. Lowering of the hoist is not permitted if slack cable exists.

*7. The fuel handling machine hoist is prevented from moving in the transfer machine zone unless the fuel transfer machine upender is vertical. An interlock is provided from the fuel transfer system to the fuel handling machine to accomplish this.

*8. The fuel handling machine is prevented from transporting new fuel above the operating floor over the spent fuel racks.

B. Bridge Hold-Down Devices

The fuel handling machine bridge is horizontally restrained on the rails by guide rollers on either side of the rail. Hold-down devices are used to prevent the bridge from leaving the rails in a seismic event.

C. Hoist Braking System

The hoists are equipped with a solenoid-activated motor brake. The brake is rated at 125 percent of the hoist design load.

D. Fuel Assembly Support System

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The hoists are supplied with redundant paths of load support so that failure of any one component will not result in a free fall of the fuel assembly. When redundant paths are not practical, conservative safety factors shall be applied.

9.1.4.3.4 Fuel Handling Tools and Equipment

Fuel handling tools and equipment handled over an open reactor vessel or spent fuel handling area are designed to prevent inadvertent decoupling from machine hooks; i.e., lifting rigs are pinned to the machine hook, and safety latches are provided on hook supporting tools.

Tools required for handling internal reactor components are designed with fail safe features that prevent disengagement of the component in the event of operating mechanism malfunction.

These safety features apply to the following tools:

A. Control Rod Drive Shaft Unlatching Tool

The air cylinders actuating the gripper mechanism are equipped with backup springs which close the gripper in the event of loss of air to the cylinder. Air-operated valves are equipped with safety locking rings to prevent inadvertent actuation.

B. New Fuel Assembly Handling Tool

When the fingers are latched, the actuating handle is positively locked, preventing inadvertent actuations. The tool is preoperationally tested at 125 percent of the weight of one fuel assembly and the heaviest core component.

C. Spent Fuel Assembly Handling Tool

When the fingers are latched, the actuating handle is positively locked to prevent inadvertent actuations. The tool is preoperationally tested at 125 percent of the weight of one fuel assembly and the heaviest core component.

9.1.4.3.5 Seismic Considerations

The equipment classifications for fuel handling and storage equipment are listed in Section 3.2, which provides criteria for the seismic design of the various components.

For safety and non-safety equipment, design for the safe shutdown earthquake (SSE) is considered if failure might adversely affect safety-related equipment.

9.1.4.3.6 Containment Pressure Boundary Integrity

The fuel transfer tube which connects the refueling cavity (inside the containment) and the fuel storage area (outside the containment) is closed on the refueling cavity side by a hatch/blind flange except during refueling operations. Two seals are located around the periphery of the hatch/blind flange with leak-check provisions between them.

9.1.4.3.7 Radiation Shielding

During spent fuel transfer, the exposure rate to the operator is 2.5 millirem per hour or less. This is accomplished by maintaining a minimum of 8.75 feet of water above the top of the active fuel height during handling operations.

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The fuel handling devices used to lift spent fuel assemblies are the refueling machine and the fuel handling machine. The fuel handling machine nonsingle-failure-proof hoist requires the use of the spent fuel handling tool to lift spent fuel. Both the refueling machine and the fuel handling machine nonsingle-failure-proof hoist are equipped with positive stops, which prevent the spent fuel assembly from being raised above the minimum water depth for radiation shielding.

9.1.4.3.8 Radiation Monitoring

Plant procedures require that an operating radiation monitor is mounted on any machine when it is handling fuel. Refer to Subsection 11.5.6.4 for a discussion of augmented radiation monitoring during fuel handling operations.

9.1.4.4 Inspection and Testing Requirements

The test and inspection requirements for the equipment in the light load handling system are as follows:

A. Fuel Handling Machine, Refueling Machine, and New Fuel Elevator

The minimum acceptable tests include the following:

- Hoist and cable are load tested at 125 percent of the rated load.
- The equipment is assembled and checked for function and operation.

The following maintenance and checkout tests are recommended to be performed prior to refueling:

- Visual inspection for loose or foreign parts; maintenance to keep free of dirt and grease.
- Lubrication of exposed gears with proper lubricant.
- Visual inspection of hoist cables for worn or broken strands.
- Visual inspection of limit switches and limit switch actuators for any sign of damaged or broken parts.
- Inspection of the equipment for function and operation.

B. Fuel Assembly Handling Tools

The minimum acceptable tests are as follows:

- The tool shall be load tested to 125 percent of the rated load.
- The tool is assembled and checked for operation.

The following maintenance and checkout tests are recommended to be performed prior to use of the tools:

- Visual inspection of the tool for dirt and loose hardware and for any signs of damage such as nicks and burrs.
- Check of the tool for operation.

C. Fuel Transfer System

The minimum acceptable test is that the system is assembled and checked for function and operation.

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The following maintenance and checkout tests are recommended to be performed prior to refueling:

- Visual inspection for loose or foreign parts; maintenance to keep free of dirt and grease.
- Lubrication of exposed gears.
- Visual inspection of limit switches and limit switch actuators for any sign of damaged or broken parts.
- Check of system for function and operation.

The above requirements are part of the plant inspection program for the light load handling system, which is implemented through procedures. In addition to the above inspections, the procedures reflect the manufacturers' recommendations for inspection.

The light load handling program, including system inspections, is implemented prior to receipt of fuel onsite.

9.1.5 Overhead Heavy Load Handling Systems

Heavy load handling systems consist of equipment which lift loads whose weight is greater than the combined weight of a single spent fuel assembly and its handling device. This equipment is part of the mechanical handling system (MHS) and is located throughout the plant. The principal equipment is the containment polar crane and the cask handling crane. Other such equipment includes the reactor coolant pump handling machine, bridge cranes, miscellaneous monorail hoists and fixed hoists. Table 9.1-5 lists the heavy load handling systems located in the safety-related areas of the plant, specifically the nuclear island.

For AP1000, a heavy load is a load whose weight is greater than the combined weight of a fuel assembly with rod cluster control, and the associated handling device. This combined weight is about 3100 pounds. Thus, a heavy load is defined as a load weighing more than 3100 pounds.

The COL Combined License applicants referencing the AP1000 certified design will also provide a heavy load handling program description. Implementation of this program will include safe load paths for movement of heavy loads, to be referenced in procedures and shown on equipment layout drawings. This will minimize the potential to impact irradiated fuel in the reactor vessel and in the spent fuel pool, and safe shutdown equipment from movement of heavy loads.

The heavy loads handling program is based on NUREG 0612 and vendor recommendations. The key elements of the program are:

- Listing of heavy loads to be lifted during operation of the plant. This list will be provided once magnitudes have been accurately formalized but no later than three (3) months prior to fuel receipt.

- Listing of heavy load handling equipment as outlined in Table 9.1-5 and whose characteristics are described in this section.

- Heavy load handling safe load paths and routing plans including descriptions of interlocks, (automatic and manual) safety devices and procedures to assure safe load path compliance. Anticipated heavy load movements are analyzed and safe load paths defined. Safe load path considerations are based on comparison with analyzed cases, previously defined safe movement areas, and previously defined restricted areas. The analyses are in accordance with Appendix A of NUREG 0612.

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- Heavy load handling equipment maintenance manuals and procedures as described in Subsection 9.1.5.5.
- Heavy load handling equipment inspection and test plans, as outlined in Subsections 9.1.5.4 and 9.1.5.5.
- Heavy load handling personnel qualifications, training, and control procedures as described in Subsection 9.1.5.5.
- QA programs to monitor, implement, and ensure compliance with the heavy load-handling procedures as described in Subsection 9.1.5.5.

A quality assurance program, consistent with Paragraph 10 of NUREG-0554, is established and implemented for the procurement, design, fabrication, installation, inspection, testing, and operation of the crane. The program, as a minimum, includes the following elements:

- design and procurement document control
- instructions, procedures, and drawings
- control of purchased material, equipment, and services
- inspection
- testing and test control
- non-conforming items
- corrective action
- records

9.1.5.1 Design Basis

9.1.5.1.1 Safety Design Basis

Section 3.2 identifies safety and seismic classifications for mechanical handling system equipment. Heavy load handling systems are generally classified as nonsafety-related, nonseismic systems. The design function and important-to-safety function of the mechanical handling system is to prevent uncontrolled lowering of a critical load. The design requirements for components accomplishing this important-to-safety function are provided in NUREG-0554 (Reference 11). Those cranes and hoists compliant with NUREG-0554 are considered single failure proof. The components of a single failure proof system (polar crane and cask handling crane) necessary to prevent uncontrolled lowering of a critical load are classified as AP1000 Equipment Class D and seismic Category I. The seismic Category I classification requires fabrication and testing in of these components in compliance with the requirements listed in 10 CFR 50 Appendix B and 10 CFR 21.

The polar crane, cask handling crane, containment equipment hatch hoist, and containment maintenance hatch hoist are single-failure-proof systems and are classified as seismic Category I. They are designed to support a critical load during and after a safe shutdown earthquake. The equipment and maintenance hatch hoist systems are required to be operational after a safe shutdown earthquake.

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A critical load is a heavy load that, if dropped, could cause unacceptable damage to reactor fuel elements, or loss of safe shutdown or decay heat removal capability. The consequences of a postulated load drop are considered to be acceptable when the four evaluation criteria of NUREG-0612 (Reference 8), Paragraph 5.1, are satisfied.

Heavy loads handled in safety-related areas of the plant are classified as critical loads unless the consequences of a load drop have been evaluated and found to be within acceptable limits. (See Subsection 9.1.5.3.)

Plant arrangement and the design of heavy load handling systems are based on the following criteria:

- To the extent practicable, heavy loads are not carried over or near safety-related components, including irradiated fuel and safe shutdown components. Safe load paths are designated for heavy load handling in safety-related areas.
- The likelihood of a load drop is extremely small (that is, the handling system is single failure proof), or the consequences of a postulated load drop are within acceptable limits.
- Single-failure-proof systems can stop and hold a critical load following the credible failure of a single component.
- Single-failure-proof systems can support a critical load during and after a safe shutdown earthquake.

9.1.5.1.2 Codes and Standards

The mechanical handling system conforms to the applicable codes and standards listed in Section 3.2. The polar crane and cask handling cranes are designed according to NUREG-0554 (Reference 11) supplemented by ASME NOG-1 (Reference 12) for a Type I single failure proof crane. Other overhead cranes and hoists handling heavy loads are designed according to ASME NOG-1, CMAA 70 (Reference 7), and to the applicable ANSI standard.

NUREG-0612 references ANSI B30.2 (Reference 9) and CMAA-70 (Reference 7) for the design of cranes in safety-related areas, and references NUREG-0554 (Reference 11) for the design of single-failure-proof cranes. ASME NOG-1 also provides design guidance consistent with that provided by NUREG-0554 for the design of single-failure-proof cranes. The design of AP1000 cranes complies with the requirements of NUREG-0612. The polar crane, cask handling crane, equipment hatch hoist, and maintenance hatch hoist meet the NRC accepted single failure proof requirements of topical report EDR-1-P-A (Reference 26). Because the components necessary to prevent uncontrolled lowering of a critical load are fabricated and tested to seismic Category I requirements, the requirements in 10 CFR 50 Appendix B and 10 CFR 21 are applicable.

As noted in the NRC Safety Evaluation Report for EDR-1-P-A (included as part of Reference 26), additional actions necessary to incorporate a complete X-SAM crane or compact X-SAM hoist in a new facility design are the development of a detailed girder design for the trolley or hoist and the performance of the structural and seismic analyses of the girder or monorail design. Also noted in the NRC Safety Evaluation Report is the requirement for the completion of Appendices B and C of EDR-1-P-A to identify the plant-specific information needed to verify conformance with NUREG-0554 guidelines.

9.1.5.2 System Description

Table 9.1-5 lists the heavy load handling systems located in the safety-related areas of the plant, specifically the nuclear island. The polar crane and cask handling crane are designed according to

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the requirements of NUREG-0554 supplemented by ASME NOG-1 for a Type I, single-failure-proof crane. A description of these cranes is provided in this subsection. The containment equipment hatch hoist and maintenance hatch hoist are designed according to the requirements of NUREG-0554 supplemented by ASME NOG-1 for a Type I, single-failure-proof hoist. Based on the conservative design of these heavy load handling systems and associated special lifting devices, slings, and load lift points (see Subsection 9.1.5.3), a load drop of the critical loads handled by the polar crane, cask handling crane, containment equipment hatch hoist, and maintenance hatch hoist is unlikely. Except for the containment polar crane, cask handling crane, containment equipment hatch hoist, and containment maintenance hatch hoist, the heavy load handling systems are not single-failure-proof.

9.1.5.2.1 Polar Crane General Description

The containment polar crane is a bridge crane mounted on a circular runway rail supported by the containment structure. The bridge consists of two welded steel box girders held together with structural end beams. The two end beams are supported by wheeled trucks that travel on top of the runway rail.

The trolley is mounted on wheeled trucks which move by tractive power over rails secured to the crane girders. The trolley provides structural support for the crane hoisting machinery. Devices are installed to preclude derailment of the bridge or trolley under seismic loading.

Two electric-powered hoists are provided, a main hoist and an auxiliary hoist. Each hoist raises and lowers loads by reeving wire rope through upper and lower sheaves. The lower sheaves are an integral part of the load block. A hook is attached to each load block.

9.1.5.2.1.1 System Operation

The polar crane lifts a variety of loads for refueling and maintenance, such as the reactor vessel integrated head package, reactor internals, and the reactor coolant pump components. The crane is designed to withstand the containment environmental conditions during all modes of plant operation, including pressurization and depressurization of the containment. The crane is designed to operate only during shutdown periods.

Movements of the bridge, trolley, main, and auxiliary hoists can be controlled from the operator's cab or from a remote control. Both the cab and remote controls include a main power control switch. The polar crane has a keylock switch in the operators cab. The keylock switch can lock out control from the hand held remote. Motion control push buttons in the cab and on the remote return to the OFF position when released.

Bridge, trolley, and hoist speeds, and speed controls are in accordance with ASME NOG-1. All speeds are variable. Speed controls permit precise positioning of the load.

The crane can be used for steam generator replacement. The structural design of the bridge is sufficient to support the steam generator, which is a noncritical load. A special hoist on a temporary trolley may be used for the steam generator replacement. Steam generator replacement is not intended to be accomplished with single-failure-proof equipment.

9.1.5.2.1.2 Component Descriptions

The polar crane is designed according to NUREG-0554 supplemented by ASME NOG-1. Table 9.1.5-3 lists the design characteristics of this crane. This subsection describes how the code requirements are implemented in the design of key hoisting and braking components. Associated lifting devices and load lift points are also described.

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Main Hoist Systems

The hoisting rope is wound around the drum in a single layer. If the rope becomes dislodged from its proper groove, the crane drives are automatically shut down and the brakes are set. Features are also provided to contain the drum and prevent disengagement of the gearing in the event of drum shaft or bearing failure. A control brake and two redundant holding brakes are provided.

Two separate, redundant reeving systems are used, so that a single rope failure will not result in the dropping of the load. Two wire ropes are reeved side-by-side through the upper and lower sheaves. Each cable passes through an equalizer that adjusts for unequal cable length. The equalizer is also a load transfer safety system, eliminating sudden load displacement and shock to the crane in the unlikely event of a cable break. In the event of hook overtravel to the point where the load block contacts the crane structure, the ropes cannot be cut or crushed.

The load block provides two separate load attachment points; the main hook is a two-pronged, sister hook with safety latches.

Auxiliary Hoist System

The auxiliary hoist system is similar to that of the main hoist.

Special Lifting Devices

Special lifting devices for critical and non-critical loads are designed to meet the applicable requirements of ANSI N14.6 (Reference 14). The stress design safety factors are based on the combined maximum static and dynamic loads that could be imparted to the handling device, based on the characteristics of the crane. Special lifting devices used for the handling of critical loads are listed in Table 9.1.5-2.

Lifting Devices Not Specially Designed

Slings or other lifting devices not specially designed are selected in accordance with ANSI B30.9 (Reference 15), except that the load rating is based on the combined maximum static and dynamic loads that could be imparted to the sling.

For the handling of critical loads, dual or redundant slings are used, or a sling having a load rating twice that required for a non-critical load is used and shall be constructed of metallic material (chain or wire rope) per NRC Regulatory Issue Summary 2005-25, Supplement 1 (Reference 23).

Load Lift Points

The design stress safety factors for heavy load lift points, such as lifting lugs or cask trunnions, are consistent with the safety factors used for special lifting devices. The design of lift points for critical loads is in accordance with NUREG-0612, Paragraph 5.1.6.(3).

9.1.5.2.1.3 Instrumentation Applications

Limit switches are used to initiate protective responses to:

- Hoist overtravel
- Hoist overspeed
- Hoist overload or unbalanced load
- Improper winding of hoist rope on the drum
- Bridge or trolley overtravel

Redundant limit switches are used with the main hoist and the auxiliary hoists to limit the extent of travel in both the hoisting and lowering directions. The primary protection for each hoist in each

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direction is a limit switch which interrupts power to the hoist motor via the control circuitry. Interruption of power to the hoist motor causes the hoist brakes to set. The hoist may be operated in the safe direction to back out of the overtravel condition.

The secondary protection for each hoist in the raising direction is a block-actuated limit switch which directly interrupts power to the hoist motor and causes the brake(s) to set. The secondary protection for each hoist in the lowering direction is a limit switch which is mechanically and electrically independent of the primary switch but also interrupts power to the hoist motor via the control circuitry. Actuation of the secondary limit switches prevents further hoisting or lowering until specific corrective action is taken.

A centrifugal-type limit switch, located on the drum shaft, provides overspeed protection for each hoist. Hoist speeds in excess of 115 percent of the rated lowering speed for a critical load cause the hoist motor to stop and the holding brakes to set.

A load-sensing system is used to detect overloading of the hoists. Hoisting motion is stopped when the overload setpoint is exceeded. Similarly, an unbalanced load is detected by a system that stops the hoist motion when there is excessive movement of the equalizer mechanism.

A level wind limit switch is provided to detect improper threading of the hoist rope in the drum grooves. This switch stops crane drive motors and sets the brakes. Further hoisting or lowering is prevented until specific corrective action is taken.

End-of-travel limit switches are provided for the trolley. These switches are set to trip just before the trolley comes into contact with the bumper, thus providing confidence that the kinetic energy of the trolley is within the energy-absorbing capacity of the bumpers.

9.1.5.2.2 Cask Handling Crane General Description

The cask handling crane is a bridge crane mounted on two runway rails supported by the auxiliary building fuel handling area east and west wall structures. The bridge consists of two welded steel box girders held together with structural end beams. The two end beams are supported by wheeled trucks that travel on top of the runway rail.

The trolley is mounted on wheeled trucks which move by tractive power over rails secured to the crane girders. The trolley provides structural support for the crane hoisting machinery. Devices are installed to preclude derailment of the bridge or trolley under seismic loading.

The hoist is electrically powered and raises and lowers loads by reeving wire rope through sheaves that are an integral part of the load block. A hook is attached to the load block.

9.1.5.2.2.1 System Operation

The cask handling crane lifts the spent fuel shipping cask from the cask transporter in the loading bay, into the fuel handling area of the auxiliary building, places the cask in the cask washdown and cask loading pits, is used to remove and replace the cask lid, and lowers the loaded cask onto the cask transporter. The crane is designed to operate in the fuel handling area environmental conditions, and is typically used only when fuel movement activities associated with refueling the reactor are not in progress.

Movements of the bridge, trolley, main, and auxiliary hoists can be controlled from a radio remote control or from a pendant suspended from the crane. Both the pendant and radio remote controls include a main power control switch. The pendant is equipped with a keylock switch that inhibits

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control from the radio remote control. Motion control push buttons on the radio remote control and on the pendant return to the OFF position when released.

Bridge, trolley, and hoist speeds, and speed control are in accordance with ASME NOG-1. All speeds are variable. Speed controls permit precise positioning of the load.

9.1.5.2.2.2 Component Descriptions

The cask handling crane is designed according to NUREG-0554 supplemented by ASME NOG-1. Table 9.1.5-1 lists the design characteristics of this crane. This subsection describes how the code requirements are implemented in the design of key hoisting and braking components. Associated lifting devices and load lift points are also described.

Hoist System

The hoisting rope is wound around the drum in a single layer. If the rope becomes dislodged from its proper groove, the crane drives are automatically shut down and the brakes are set. Features are also provided to contain the drum and prevent disengagement of the gearing in the event of drum shaft or bearing failure. A control brake and two redundant holding brakes are provided.

Two separate, redundant reeving systems are used, so that a single rope failure will not result in the dropping of the load. Two wire ropes are reeved side-by-side through the sheave. Each cable passes through an equalizer that adjusts for unequal cable length. The equalizer is also a load transfer safety system, eliminating sudden load displacement and shock to the crane in the unlikely event of a cable break. Overtravel protection is provided (see Subsection 9.1.5.2.2.3); however, even in the event of hook overtravel in the raising direction to the point the load block contacts the crane structure, the ropes cannot be cut or crushed.

The load block provides two separate load attachment points; the main hook is a two-pronged sister hook with safety latches.

Auxiliary Hoist System

The auxiliary hoist system is similar to that of the main hoist.

Special Lifting Devices

Special lifting devices for critical and non-critical loads are designed to meet the applicable requirements of ANSI N14.6 (Reference 14). The stress design safety factors are based on the combined maximum static and dynamic loads that could be imparted to the handling device, based on the characteristics of the crane. Special lifting devices used for the handling of critical loads are listed in Table 9.1.5-2.

Lifting Devices Not Specially Designed

Slings or other lifting devices not specially designed are selected in accordance with ANSI B30.9 (Reference 15), except that the load rating is based on the combined maximum static and dynamic loads that could be imparted to the sling.

For the handling of critical loads, dual or redundant slings are used, or a sling having a load rating twice that required for a non-critical load is used and shall be constructed of metallic material (chain or wire rope) per NRC Regulatory Issue Summary 2005-25, Supplement 1 (Reference 23).

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Load Lift Points

The design stress safety factors for heavy load lift points, such as lifting lugs or cask trunnions, are consistent with the safety factors used for special lifting devices. The design of lift points for critical loads is in accordance with NUREG-0612, Paragraph 5.1.6.(3).

9.1.5.2.2.3 Instrumentation Applications

Limit switches are used to initiate protective responses to:

- Hoist overtravel.
- Hoist overspeed.
- Hoist overload or unbalance load.
- Improper winding of hoist rope on the drum.
- Bridge or trolley overspeed.
- Bridge or trolley overtravel.

Redundant limit switches are used with the main hoist and the auxiliary hoists to limit the extent of travel in both the hoisting and lowering directions. The primary protection for each hoist in each direction is a limit switch which interrupts power to the hoist motor via the control circuitry. Interruption of power to the hoist motor causes the hoist brakes to set. The hoist may be operated in the safe direction to back out of the overtravel condition.

The secondary protection for each hoist in the raising direction is a block-actuated limit switch, which is mechanically and electrically independent of the primary limit switch and interrupts power to the hoist motor and causes the brake(s) to set. The secondary protection for each hoist in the lowering direction is a limit switch, which is mechanically and electrically independent of the primary switch, but also interrupts power to the hoist motor via the control circuitry. Actuation of the secondary limit switches prevents further hoisting or lowering until specific corrective action is taken.

A centrifugal-type limit switch, located on the drum shaft, provides overspeed protection for each hoist. Hoist speeds in excess of 115 percent of the rated lowering speed for a critical load causes the hoist motor to stop and the holding brakes to set.

A load-sensing system is used to detect overloading of the hoists. Hoisting motion is stopped when the overload setpoint is exceeded. Similarly, an unbalanced load is detected by a system that stops the hoist motion when there is excessive movement of the equalizer mechanism.

A level wind limit switch is provided to detect improper threading of the hoist rope in the drum grooves. This switch stops crane drive motors and sets the brakes. Further hoisting or lowering is prevented until specific corrective action is taken.

End-of-travel limit switches are provided for the trolley. These switches are set to trip just before the trolley comes into contact with the bumper. This provides confidence that the kinetic energy of the trolley is within the energy-absorbing capacity of the bumpers.

9.1.5.2.3 Equipment Hatch Hoist General Description

The equipment hatch hoist is a hoist that is foot-mounted on a platform supported by the containment structure.

The hoist is electrically powered, and it raises and lowers loads by reeving wire rope through sheaves that are an integral part of the load block. The load block is equipped with a hook or lifting device.

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9.1.5.2.3.1 System Operation

The equipment hatch hoist lifts the equipment hatch. The hoist is designed to withstand the containment environmental conditions during all modes of plant operation, including pressurization and depressurization of the containment. The hoist is designed to operate only during shutdown periods.

Movements of the hoist can be controlled from a local pushbutton control station. The pushbutton control station includes a main power control switch. Motion control push buttons on the control station return to the OFF position when released.

Hoist speed is in accordance with ASME NOG-1.

9.1.5.2.3.2 Component Descriptions

The equipment hatch hoist is designed according to NUREG-0554 supplemented by ASME NOG-1. Table 9.1.5-4 lists the design characteristics of this hoist. This subsection describes how the code requirements are implemented in the design of key safety-related components.

Hoist System

The hoisting rope is wound around the drum in a single layer. If the rope becomes dislodged from its proper groove, the hoist drive is automatically shut down and the holding brakes are set. Features are also provided to contain the drum and prevent disengagement of the gearing in the event of drum shaft or bearing failure. Two holding brakes and a third brake on the wire rope drum with dynamic lowering capability are provided.

Two separate, redundant reeving systems are used so that a single rope failure will not result in the dropping of the load. Two wire ropes are reeved side-by-side through the sheave blocks. Each rope connects to an equalizer that adjusts for unequal rope length. The equalizer is also a load transfer safety system, eliminating sudden load displacement and shock to the hoist in the unlikely event of a rope break. Overtravel protection is provided (see Subsection 9.1.5.2.3.3); however, even in the event of overtravel in the raising direction to the point the load block contacts the hoist structure, the ropes cannot be cut or crushed.

The load block provides a load attachment point that has double the normal design factor in lieu of redundancy.

Special Lifting Devices

Special lifting devices shall not be used with the equipment hatch hoist.

Lifting Devices Not Specially Designed

Slings or other lifting devices not specially designed are selected in accordance with ANSI B30.9 (Reference 15), except that the load rating is based on the combined maximum static and dynamic loads that could be imparted to the sling.

For the handling of critical loads, dual or redundant slings are used, or a sling having a load rating twice that required for a non-critical load is used and shall be constructed of metallic material (chain or wire rope) per NRC Regulatory Issue Summary 2005-25, Supplement 1 (Reference 23).

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Load Lift Points

The design stress safety factors for heavy load lift points, such as lifting lugs, are consistent with the safety factors used for special lifting devices. The design of lift points for critical loads is in accordance with NUREG-0612, Paragraph 5.1.6.(3).

9.1.5.2.3.3 Instrumentation Applications

Limit switches are used to initiate protective responses to:

- Hoist overtravel.
- Hoist overspeed.
- Hoist overload or unbalanced load.
- Improper winding of hoist rope on the drum.

Redundant limit switches are used with the hoist to limit the extent of travel in both the hoisting and lowering directions. The primary protection for the hoist in each direction is a limit switch, which interrupts power to the hoist motor via the control circuitry. Interruption of power to the hoist motor causes the hoist brakes to set. The hoist may be operated in the safe direction to back out of the overtravel condition.

The secondary protection for the hoist in the raising direction is a block-actuated limit switch, which is mechanically and electrically independent of the primary limit switch and interrupts power to the hoist motor and causes the brake(s) to set. The secondary protection for the hoist in the lowering direction is a limit switch, which is mechanically and electrically independent of the primary switch, but also interrupts power to the hoist motor via the control circuitry. Actuation of the secondary limit switches prevents further hoisting or lowering until specific corrective action is taken.

A centrifugal-type limit switch, located on the drum shaft, provides overspeed protection for the hoist. Hoist speeds in excess of 125 percent of the rated lowering speed for a critical load causes the hoist motor to stop and the holding brakes to set.

A load-sensing system is used to detect overloading of the hoists. Hoisting motion is stopped when the overload setpoint is exceeded. Similarly, an unbalanced load is detected by a system that stops the hoist motion when there is excessive movement of the equalizer mechanism.

A level wind limit switch is provided to detect improper threading of the hoist rope in the drum grooves. This switch stops crane drive motors and sets the brakes. Further hoisting or lowering is prevented until specific corrective action is taken.

9.1.5.2.4 Maintenance Hatch Hoist General Description

The maintenance hatch hoist system is the same as that of the equipment hatch hoist system.

9.1.5.3 Safety Evaluation

The design and arrangement of heavy load handling systems promotes the safe handling of heavy loads by one of the following means:

- A single-failure-proof system is provided so that a load drop is unlikely.

- The arrangement of the system in relationship to safety-related plant components is such that the consequences of a load drop are acceptable per NUREG 0612. Postulated load drops are evaluated in the heavy loads analysis.

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The polar crane, the cask handling crane, the containment equipment hatch, and the maintenance hatch hoists are single failure proof. These systems stop and hold a critical load following the credible failure of a single component. A double design factor is provided for hooks where used as load bearing components. Redundancy is provided for load bearing components other than hooks, such as the hoisting ropes, sheaves, equalizer assembly, and holding brakes. These systems are designed to support a critical load during and after a safe shutdown earthquake. The seismic Category I equipment and maintenance hatch hoist systems are designed to remain operational following a safe shutdown earthquake. The polar crane is designed to withstand rapid pressurization of the containment during a design basis loss of coolant accident or main steam line break, without collapsing.

The cask loading pit is separated from the spent fuel pool. The cask handling crane cannot move over the spent fuel pool because the crane rails do not extend over the pool. Mechanical stops prevent the cask handling crane from going beyond the ends of the rails.

A heavy loads analysis is performed to evaluate postulated load drops from heavy load handling systems located in safety-related areas of the plant, specifically the nuclear island. No evaluations are required for critical loads handled by the containment polar crane, the cask handling crane, the containment equipment hatch hoist, and the containment maintenance hatch hoist since a load drop is unlikely.

The heavy loads analysis is to confirm that a postulated load drop does not cause unacceptable damage to reactor fuel elements, or loss of safe shutdown or decay heat removal capability.

For the MSIV monorail hoists, equipment and components required for decay heat removal during MODES 5 or 6 are not located in the load path.

There are no planned heavy load lifts outside those already described. However, over the plant life there may be occasions when heavy loads not presently addressed need to be lifted (i.e., in support of special maintenance/repairs). For these occasions, special procedures are generated that address, as a minimum, the following:

- The special procedure complies with NUREG-0612.
- A safe load path is determined. Mechanical and/or electrical stops are incorporated in the hardware design to prohibit travel outside the safe load path. Maximum lift heights are specified to minimize the impact of an unlikely load drop.
- Where a load drop could occur over irradiated fuel or safe shutdown equipment, the consequence of the load drop is evaluated. If the evaluation concludes that the load drop is not acceptable, an alternate path is evaluated, or the lift is prohibited.
- The lifting equipment is in compliance with applicable ANSI standards and has factors of safety that meet or exceed the requirements of the applicable standards.
- Operator training is provided prior to actual lifts.
- Inspection of crane components is performed in accordance with the manufacturer recommendations.

Plant procedures require that an operating radiation monitor is mounted on any crane when it is handling fuel. Refer to Subsection 11.5.6.4 for a discussion of augmented radiation monitoring during fuel handling operations.

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9.1.5.4 Inservice Inspection/Inservice Testing

Preoperational inspection and testing of overhead cranes is governed by NUREG-0554. Tests include operational testing with 100 percent load to demonstrate function and speed controls for bridge, trolley, and hoist drives and proper functioning of limit switches, locking, and safety devices. A rated load test is performed with a 125 percent load.

Following plant startup, inservice inspection of overhead cranes is governed by site-specific procedures in accordance with ANSI B30.2. Testing of crane modifications is governed by NUREG-0554 supplemented by ASME NOG-1. Inservice inspection and testing of other cranes and hoists is in accordance with manufacturer's recommendations and applicable industry standards.

In-service inspection and testing of special lifting devices and slings used in safety-related areas of the plant are in accordance with ANSI N14.6 and ANSI B30.9.

The above requirements are part of the plant inspection program for the overhead heavy load handling system, which is implemented through procedures. In addition to the above inspections, the procedures reflect the manufacturers' recommendations for inspection and the NUREG-0612 recommendations.

The overhead heavy load handling equipment inservice inspection procedures, as a minimum, address the following:

- Identification of components to be examined
- Examination techniques
- Inspection intervals
- Examination categories and requirements
- Evaluation of examination results

The overhead heavy load handling program, including system inspections, is implemented prior to receipt of fuel onsite.

9.1.5.5 Load Handling Procedures

Load handling operations for heavy loads that are handled over, could be handled over or are in the proximity of irradiated fuel or safe shutdown equipment are controlled by written procedures. As a minimum, procedures are used for handling loads with the spent fuel cask bridge and polar cranes, and for those loads listed in Table 3.1-1 of NUREG 0612. The procedures include and address the following elements:

- The specific equipment required to handle load (e.g., special lifting devices, slings, shackles, turnbuckles, clevises, load cells, etc.).
- Qualification and training of crane operators and riggers in accordance with chapter 2-3.1 of ASME B30.2, "Overhead and Gantry Cranes."
- The requirements for inspection and acceptance criteria prior to load movement.
- The defined safe load path and provisions to provide visual reference to the crane operator and/or signal person of the safe load path envelope.

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- Specific steps and proper sequence to be followed for handling load.
- Precautions, limitations, prerequisites, and/or initial conditions associated with movement of heavy loads.
- The testing, inspection, acceptance criteria and maintenance of overhead heavy load handling systems. These procedures are in accordance with the manufacturer recommendations and are consistent with ANSI B30.2 or with other appropriate and applicable ANSI standards.

Safe load paths are defined for movement of heavy loads to minimize the potential for a load drop on irradiated fuel in the reactor vessel, spent fuel pool or safe shutdown equipment. Paths are defined clearly in procedures and equipment layout drawings. Equipment layout drawings showing the safe load path are used to define safe load paths in load handling procedures. Deviation from defined safe load paths requires a written alternative procedure approved by a plant safety review committee.

9.1.6 Combined License Information for Fuel Storage and Handling

9.1.6.1 Structural Dynamic and Stress Analysis for New Fuel Rack

The confirmatory structural dynamic and stress analysis for the new fuel rack, as described in Subsection 9.1.1.2.1, is addressed in APP-GW-GLR-026 (Reference 16), and the applicable changes are incorporated.

9.1.6.2 Criticality Analysis for New Fuel Rack

The confirmatory criticality analysis for the new fuel rack, as described in Subsection 9.1.1.3, is addressed in APP-GW-GLR-030 (Reference 17), and the applicable changes are incorporated.

This report (Reference 17) addresses the degradation of integral neutron absorbing material in the new fuel pool storage racks as identified in GL-96-04, and assesses the integral neutron absorbing material capability to maintain a 5-percent subcriticality margin.

9.1.6.3 Structural Dynamic and Stress Analysis for Spent Fuel Racks

The confirmatory structural dynamic and stress analysis for the spent fuel racks, as described in Subsection 9.1.2.2.1, is addressed in APP-GW-GLR-033 (Reference 18), and the applicable changes are incorporated.

9.1.6.4 Criticality Analysis for Spent Fuel Racks

The confirmatory criticality analysis for the spent fuel racks, as described in Subsection 9.1.2.3, is addressed in APP-GW-GLR-029P (Reference 20), and the applicable changes are incorporated.

9.1.6.5 Inservice Inspection Load Handling Systems

The program for inservice inspection of the light load handling system and the overhead heavy load handling system are addressed in Subsections 9.1.4.4 and 9.1.5.4.

9.1.6.6 Operating Radiation Monitor

The radiation monitor mounted on any crane or fuel handling machine when it is handling fuel is addressed in Subsections 9.1.4.3.8 and 9.1.5.3.

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9.1.6.7 Coupon Monitoring Program

A spent fuel rack Metamic coupon monitoring program will be implemented when the plant is placed into commercial operation. This program will include tests to monitor bubbling, blistering, cracking, or flaking; and a test to monitor for corrosion, such as weight loss measurements and / or visual examination. The program will also include testing to monitor changes in physical properties of the absorber material, including neutron attenuation and thickness measurements.

The program will include the methodology and acceptance criteria for the tests listed and provide corrective action requirements based on vendor recommendations and industry operating experience. The program will be implemented through plant procedures.

Metamic Monitoring Acceptance Criteria:

- Verification of continued presence of the boron is performed by neutron attenuation measurement. A decrease of no more than 5% in Boron-10 content, as determined by neutron attenuation, is acceptable. This is equivalent to a requirement for no loss in boron within the accuracy of the measurement.

- Coupons are monitored for unacceptable swelling by measuring coupon thickness. An increase in coupon thickness at any point of no more than 10% of the initial thickness at that point is acceptable.

Changes in excess of either of the above two acceptance criteria are investigated under the corrective action program and may require early retrieval and measurement of one or more of the remaining coupons to provide validation that the indicated changes are real. If the deviation is determined to be real, an engineering evaluation is performed to identify further testing or any corrective action that may be necessary.

Additional parameters are examined for early indications of the potential onset of Metamic degradation that would suggest a need for further attention and possibly a change in the coupon withdrawal schedule. These include visual inspection for surface pitting, blistering, cracking, corrosion or edge deterioration, or unaccountable weight loss in excess of the measurement accuracy.

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9.1.7 References

1. ANSI N16.1-75, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.

2. ANSI N16.9-75, Validation of Calculational Methods for Nuclear Criticality Safety.

3. ANSI N210-76, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations.

4. ANS 57.2-1983, Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.

5. USNRC NUREG-0800, SRP 3.8.4, Revision 1, Appendix D, “Technical Position on Spent Fuel Pool Racks,” July 1981.

6. ANS 57.1-1992, Design Requirements for Light Water Reactor Fuel Handling Systems.

7. Specifications for Electric Overhead Travelling Cranes CMAA, Specification 70 – 2000.

8. USNRC, “Control of Heavy Loads at Nuclear Power Plants,” NUREG-0612, July 1980.

9. “Overhead and Gantry Cranes,” ANSI/ASME B30.2-1990.

10. Not used.

11. USNRC, “Single-Failure-Proof Cranes for Nuclear Power Plants,” NUREG-0554, May 1979.

12. “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” ASME NOG-1-1998.

13. Not used.

14. “Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More,” ANSI N14.6-1993.

15. “Slings,” ASME/ANSI B30.9-1996.

16. APP-GW-GLR-026, “New Fuel Storage Rack Structural/Seismic Analysis,” Westinghouse Electric Company LLC.

17. APP-GW-GLR-030, “New Fuel Storage Rack Criticality Analysis,” Westinghouse Electric Company LLC.

18. APP-GW-GLR-033, “Spent Fuel Storage Rack Structural/Seismic Analysis,” Westinghouse Electric Company LLC.

19. Not used.

20. APP-GW-GLR-029P, Revision 3, “AP1000 Spent Fuel Storage Racks Criticality Analysis,” Westinghouse Electric Company LLC (Westinghouse Proprietary).

21. USNRC, 10 CFR 50.68, “Criticality Accident Requirements,” January 2003.

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22. USNRC, Regulatory Guide 1.124, Revision 1, “Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports,” January 1978.

23. USNRC, Regulatory Issue Summary 2005-25, Supplement 1, “Clarification of NRC Guidelines for Control of Heavy Loads,” May 2007.

24. “Source Book for Metamic Performance Assessment,” Holtec Report HI-2043215, Revision 2, Holtec International, September 2006.

25. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Holtec International Report HI-2022871 Regarding the Use of Metamic® in Fuel Pool Applications Facility Operating License NOS DPR-51 and NPF-6 Entergy Operations, Inc Arkansas Nuclear One, Unit Nos. 1 and 2 Docket Nos. 50-313.

26. EDR-1-P-A, Revision 7, “eXtra Safety and Monitoring (X-SAM®) Single Failure Proof Cranes,” September 2015 (The NRC Safety Evaluation Report is included as part of the EDR-P-A, Revision 7 document)

27. APP-GW-GLR-029, Revision 5, “AP1000 Spent Fuel Racks Criticality Analysis,” Westinghouse Electric Company LLC (Westinghouse Proprietary).

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Table 9.1-1
Loads and Load Combinations for Fuel Racks

Load Combination	Service Level
D + L D + L + T_o	Level A (Note 1, Note 2)
D + L + T_a D + L + T_o + P_f	Level B (Note 1, Note 2)
D + L + T_a + E'	Level D (Note 2)
D + L + F_d	The functional capability of the fuel racks should be demonstrated. (Note 3)

Note:

1. There is no operating basis earthquake (OBE) for the AP1000 plant.
2. The fuel racks are freestanding; thus, there is minimal or no restraint against free thermal expansion at the base of the rack. As a result, thermal loads applied to the rack (T_o and T_a) produce only local (secondary) stresses.
3. This load combination is not required for the new fuel rack since a load drop is deemed an unlikely accident with a single failure-proof hoist.

Abbreviations are those used in NUREG-0800, Section 3.8.4 (including Appendix D) of the Standard Review Plan (SRP):

D = Dead weight induced loads (including fuel assembly weight)
L = Live load (not applicable to fuel racks since there are no moving objects in the rack load path)
F_d Force caused by the accidental drop of the heaviest load from the maximum possible height
P_f = Upward force on the racks caused by postulated stuck fuel assembly
E' = Safe shutdown earthquake (SSE)
T_o = Differential temperature induced loads based on the most critical transient or steady-state condition under normal operation or shutdown conditions
T_a = Differential temperature induced loads based on the postulated abnormal design conditions

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Table 9.1-2
Spent Fuel Pool Cooling and Purification System Design Parameters

Spent fuel pool storage capacity	889 total fuel assemblies
Spent fuel pool water volume (including racks without fuel at water level of 15 inches below the operating deck) (gallons)	190,500
Fuel transfer canal water volume (including gate area) at water level of 15 inches below the operating deck (gallons)	63,500
Minimum combined volume of spent fuel pool, fuel transfer canal, and both gate areas (above the fuel assemblies to the bottom of the spent fuel cooling suction piping) (gallons)	130,350
Minimum volume of the cask washdown pit from the cask washdown pit floor to 13.75 feet above the cask washdown pit floor (gallons)	34,100
Minimum volume of the cask loading pit from the bottom of the cask loading pit gate to the bottom of the spent fuel pool cooling suction piping (gallons)	46,050
Nominal boron concentration of water (ppm)	2,700
Maximum normal refueling case (full core offload) Water temperature with one spent fuel cooling system cooling train and one normal residual heat removal system cooling train in operation (°F)	<140
Maximum emergency core unload case Water temperature with both spent fuel cooling system cooling trains and one normal residual heat removal system cooling train in operation (°F)	<140

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Table 9.1-3 (Sheet 1 of 2)
Component Data – Spent Fuel Pool Cooling and Purification System

Spent Fuel Pool Pump		
Number		2
Design pressure (psig)		150
Design temperature (°F)		250
Nominal flow (gallons/minute)		1200
Minimum flow to support normal cooling (gpm)		900
Material		Stainless Steel
Spent Fuel Pool Heat Exchangers		
Number		2
Type		Plate
Design heat transfer (Btu/hour)		14.75×10^6
Design capacity (Btu/hour-°F)		24.0×10^5
Minimum capacity to support normal cooling (Btu/hour-°F)		22.0×10^5
	Side 1	**Side 2**
Design pressure (psig)	200	150
Design temperature (°F)	200	250
Nominal flow (pounds/hour)	6.23×10^5	5.94×10^5
Inlet temperature (°F), typ.	89.5	120
Outlet temperature (°F), typ.	113.3	95.1
Fluid circulated	Component cooling water	Spent fuel pool water
Material	Stainless steel	Stainless steel
Spent Fuel Pool Demineralizers		
Number		2
Design pressure (psig)		160
Design temperature (°F)		250
Nominal flow (gallons/minute)		200
Nominal resin volume (cubic feet)		75
Material		Stainless steel

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Table 9.1-3 (Sheet 2 of 2)
Component Data – Spent Fuel Pool Cooling and Purification System

Spent Fuel Pool Filter	
Number	2
Design pressure (psig)	160
Design temperature (°F)	250
Nominal flow (gallons/minute)	200
Filtration requirement	98% retention of particles above 5 μm
Material, vessel	Stainless steel

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Table 9.1-4
Station Blackout/Seismic Event Times(1)(9)

Event	Time to Saturation(1) (hours)	Height of Water Above Fuel at 72 Hours(4) (feet)	Height of Water Above Fuel at 7 Days(4) (feet)
Seismic Event(2) – Power Operation Immediately Following a Refueling(7)	6.0	0.5	0.5(6)
Seismic Event(8) – Refueling, Immediately Following Spent Fuel Region Offload(3)(7)	4.5	1.0(5)	1.0(6)
Seismic Event(8) – Refueling, Emergency Full Core Off-Load(3) Immediately Following Refueling(7)	2.0	8.0(5)	8.0(6)

Notes:

1. Times calculated neglect heat losses to the passive heat sinks in the fuel area of the auxiliary building.
2. Seismic event assumes water in the pool is initially drained to the level of the spent fuel pool cooling system connection simultaneous with a station blackout. Fuel cooling water sources are the spent fuel pool, fuel transfer canal, both gate areas, and cask washdown pit for 72 hours. Between 72 hours and 7 days fuel cooling water provided from passive containment cooling ancillary water storage tank.
3. Fuel movement complete, 150 hours after shutdown.
4. See Subsection 9.1.3.5 for minimum water level.
5. Alignment of PCS water storage for supply of makeup water permits maintaining pool level at this elevation. Decay heat in reactor vessel is at or below 7.0 MWt, thus no PCS water is required for containment cooling for 72 hours.
6. Alignment of the passive containment cooling ancillary water storage tank and initiation of PCS recirculation pumps provide a makeup water supply to maintain this pool level or higher above the top of the fuel.
7. The number of fuel assemblies refueled has been conservatively established to include the worst case for an 18-month fuel cycle plus 5 defective fuel assemblies (69 total assemblies or 44% of the core).
8. Seismic event assumes water in the pool is initially drained to the level of the spent fuel pool cooling system connection simultaneous with a station blackout. Fuel cooling water sources are the spent fuel pool, fuel transfer canal, both gate areas, cask washdown pit, cask loading pit, and passive containment cooling system water storage tank for 72 hours.
9. A minimum of 18 hours is available for operator action to align makeup water to the spent fuel pool after a seismic event.

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Table 9.1-5
Nuclear Island Heavy Load Handling Systems(1)

Name	Crane/Hoist Type	Location (Building)	Maximum Load Rating (tons)
Containment Polar Crane	Overhead bridge	Containment	300(2)
Equipment Hatch Hoist	Fixed hoist	Containment	13.5
Maintenance Hatch Hoist	Fixed hoist	Containment	13.5
Cask Handling Crane	Overhead bridge	Auxiliary	150
MSIV Monorails Hoist A	Monorail hoists	Auxiliary	12
MSIV Monorails Hoist B	Monorail hoists	Auxiliary	12

Notes:
1. Nuclear island elevators are discussed in the heavy loads analysis.
2. Trolley maximum load rating for a critical load.

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Table 9.1.5-1
Cask Handling Crane Component Data

Bridge	
Bridge span	60 ft - 2 in
Travel speed	See Note 1.
Service/parking/emergency braking system (type and number)	Friction (one) for all three functions
Trolley	
Travel speed	See Note 1.
Service/parking/emergency braking system (type and number)	Friction (one) for all three functions
Main Hoist	
Approximate capacity	See Table 9.1-5.
Hook speed	See Note 1.
Approximate hook travel (elevation)	To cask transport in loading bay (at grade elevation)
Main hoist braking system (diverse systems)	
Control brakes (type and number)	Electric (one)
Holding brakes (type and number)	Friction (one)
Emergency drum brake (type and number)	Friction (one)
Auxiliary Hoist	
Approximate capacity	10 tons
Hook speed	See Note 1
Approximate hook travel (elevation)	To cask transport in loading bay (at grade elevation)
Auxiliary hoist braking system (diverse systems)	
Control brakes (type and number)	Electric (one)
Holding brakes (type and number)	Friction (one)
Emergency drum brake (type and number)	Friction (one)

Note:
1. Bridge, trolley, and hoist speeds are within the recommended ranges of ASME NOG-1.

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Table 9.1.5-2
Special Lifting Devices Used for the Handling of Critical Loads

Polar Crane Special Lifting Devices	Description
Integrated head package (IHP)	The IHP combines several separate components into an integral unit. It incorporates the lifting device that provides the interface between the polar crane and the reactor vessel head.
Reactor internals lifting rig	The reactor internals lifting rig is a three-legged carbon steel and stainless steel structure that is attached to the main hook for handling of the upper and lower reactor internals packages. The lower reactor internals are not considered a critical load.
Reactor coolant pump (RCP)	The RCP handling machine is used for removal of the RCP motor and hydraulic elements from the pump casing. The pump/motor shell includes lifting lugs which are attached to a lifting device to allow the RCP motor and hydraulic elements to be handled by the polar crane main hook.
Cask Handling Crane Special Lifting Devices	**Description**
Cask lift yoke, cask lift yoke extension, and loaded canister handling equipment	These devices are used for the handling of the casks and loaded canisters, which provide the interface between the cask handling crane and the shipping cask or loaded canister.

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Table 9.1.5-3
Polar Crane Component Data

Bridge	
Bridge span	See Figure 1.2-14 and Figure 1.2-16
Travel speed	See Note 1
Service/parking/emergency braking system (type and number)	Friction (one) for all three functions
Trolley	
Travel speed	See Note 1
Service/parking/emergency braking system (type and number)	Friction (one) for all three functions
Main Hoist	
Approximate capacity	See Table 9.1-5
Hook speed	See Note 1
Approximate hook travel (elevation)	To reactor vessel internals
Main hoist braking system (diverse systems)	
Control brakes (type and number)	Electric (one)
Holding brakes (type and number)	Friction (one)
Emergency drum brake (type and number)	Friction (one)
Auxiliary Hoist	
Approximate capacity	25 tons
Hook speed	See Note 1
Approximate hook travel (elevation)	To reactor coolant pump
Auxiliary hoist braking system (diverse systems)	
Control brakes (type and number)	Electric (one)
Holding brakes (type and number)	Friction (one)
Emergency drum brake (type and number)	Friction (one)

Note:
1. Bridge, trolley and hoist speeds are within the recommended ranges of ASME NOG-1.

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Table 9.1.5-4
Equipment/Maintenance Hatch Hoist Component Data

Hoist	
Approximate capacity	See Table 9.1-5.
Hoist speed	See Note 1.
Approximate load block travel (elevation)	To hatch (lowered position)
Main hoist braking system	
Control brakes (type and number)	Dynamic lowering using emergency drum brake (Refer to Emergency drum brake in this table)
Holding brakes (type and number)	Friction (two)
Emergency drum brake (type and number)	Friction (one)

Note:
1. Hoist speed is within the recommended range of ASME NOG-1.

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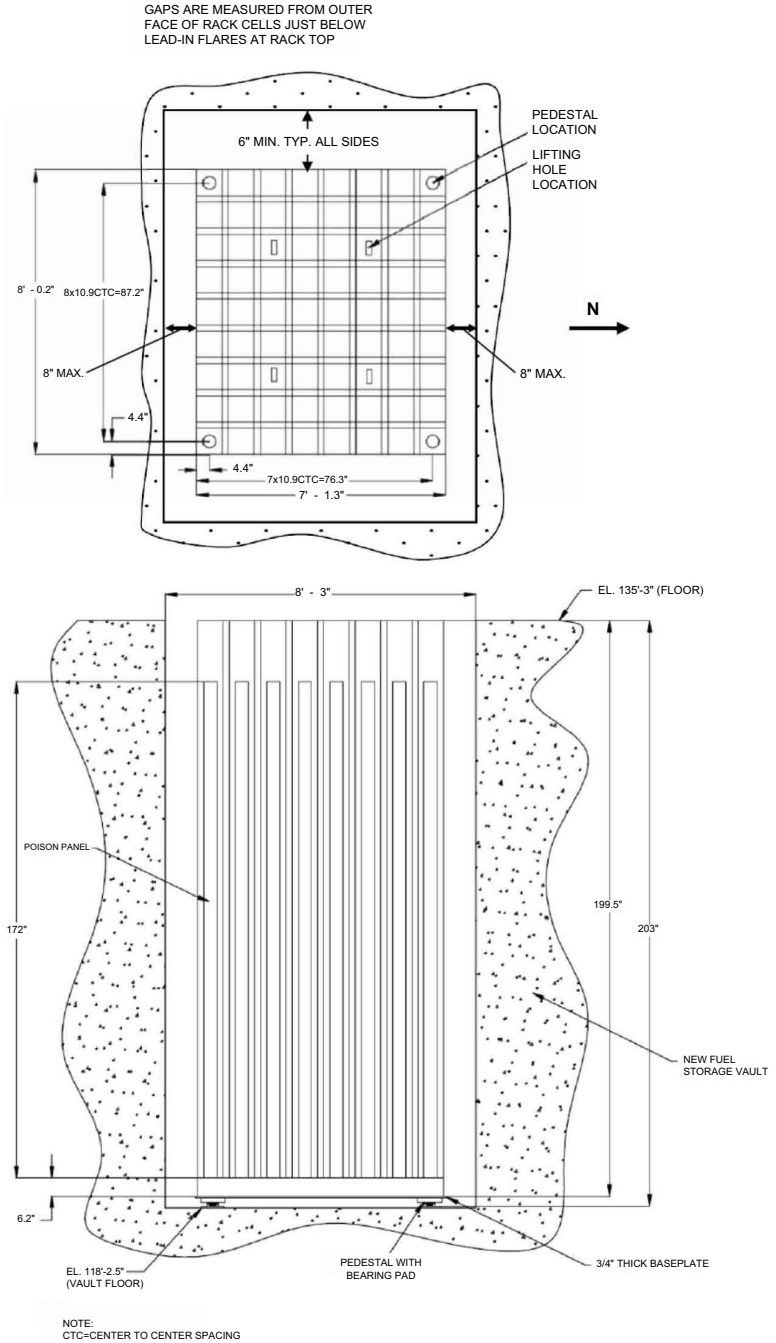

Figure 9.1-1 (Sheet 1 of 2)
New Fuel Storage Rack Layout (72 Storage Location)

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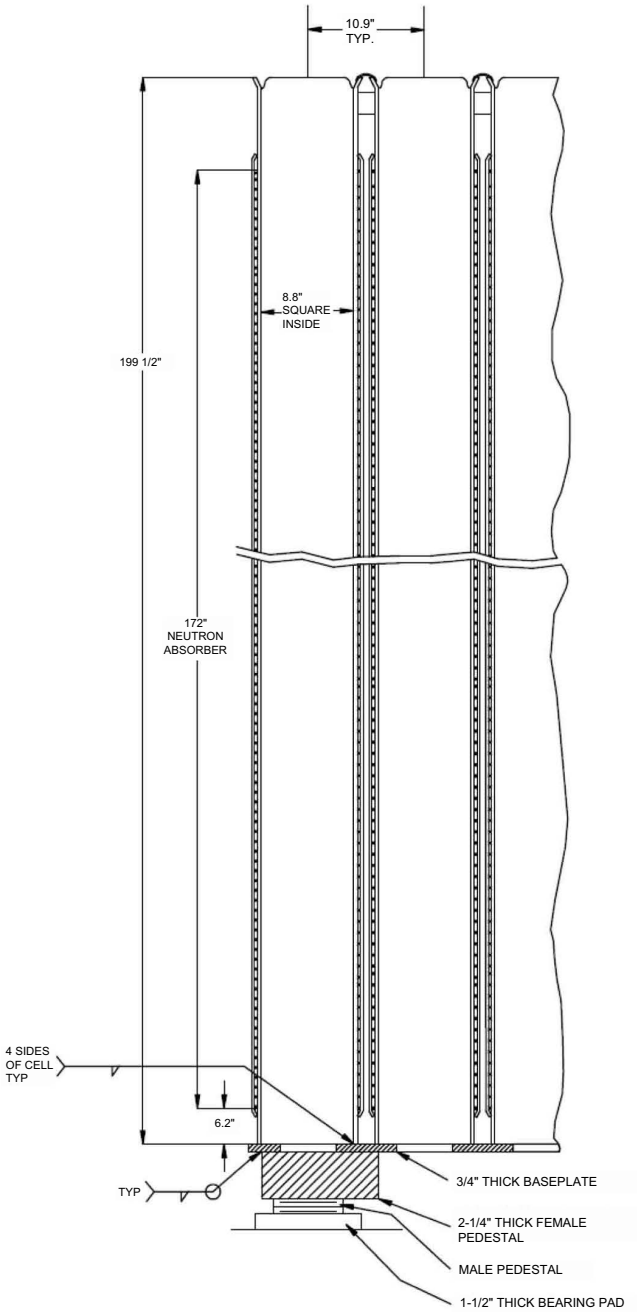

Figure 9.1-1 (Sheet 2 of 2)
New Fuel Storage Rack Cross Section

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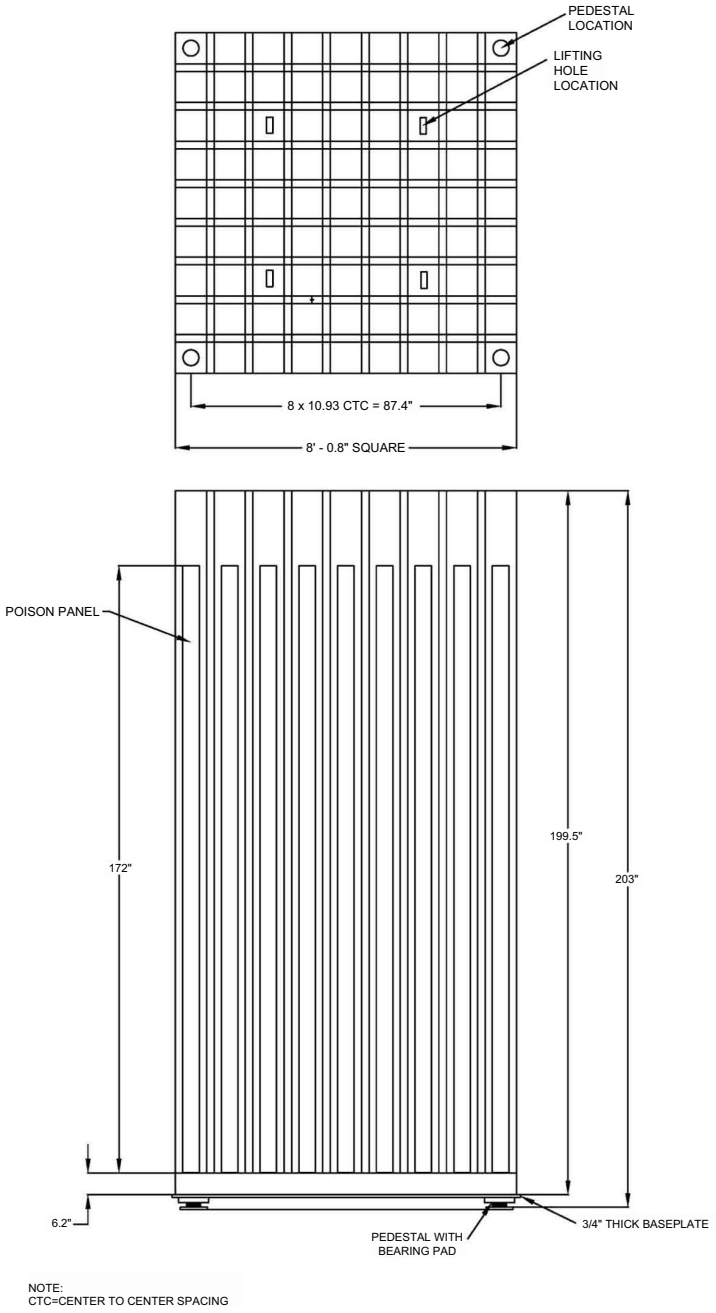

Figure 9.1-2 (Sheet 1 of 2)
Region 1 Spent Fuel Storage Rack Layout

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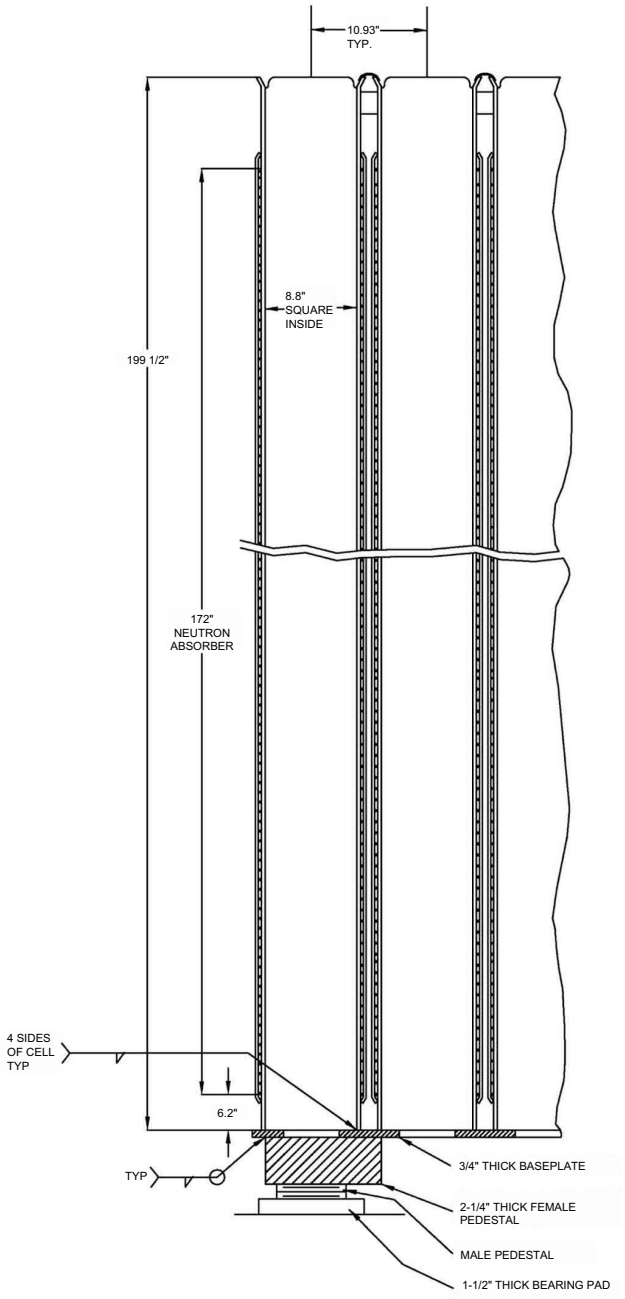

Figure 9.1-2 (Sheet 2 of 2)
Region 1 Spent Fuel Storage Rack Cross Section

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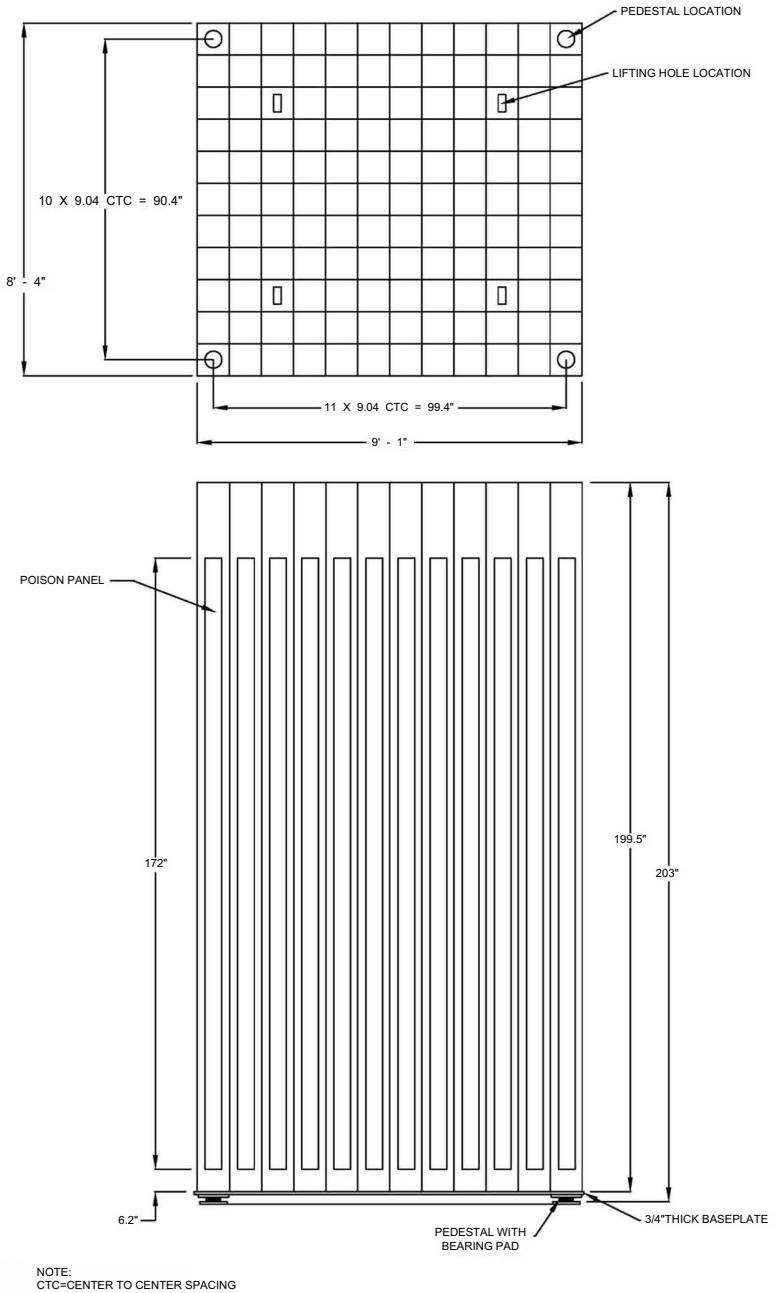

Figure 9.1-3 (Sheet 1 of 2)
Region 2 Spent Fuel Storage Rack Layout

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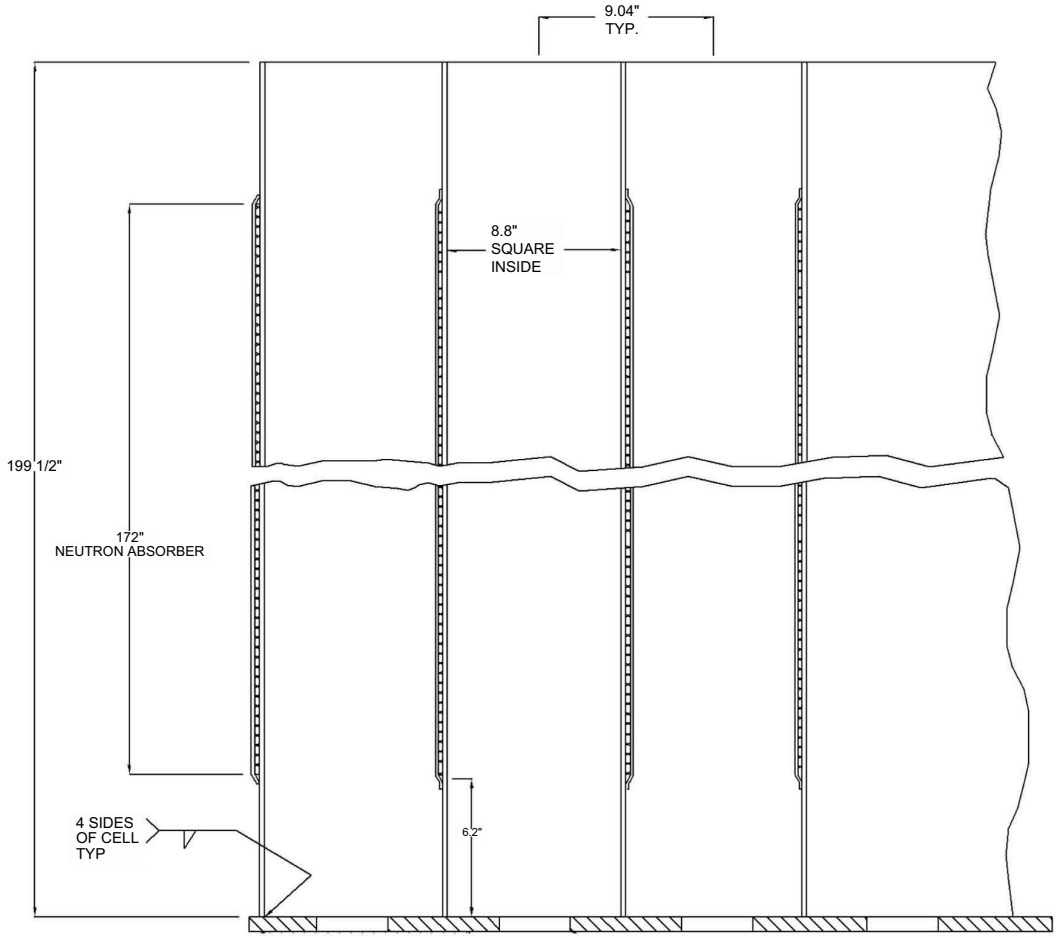

Figure 9.1-3 (Sheet 2 of 2)
Region 2 Spent Fuel Storage Rack Cross Section

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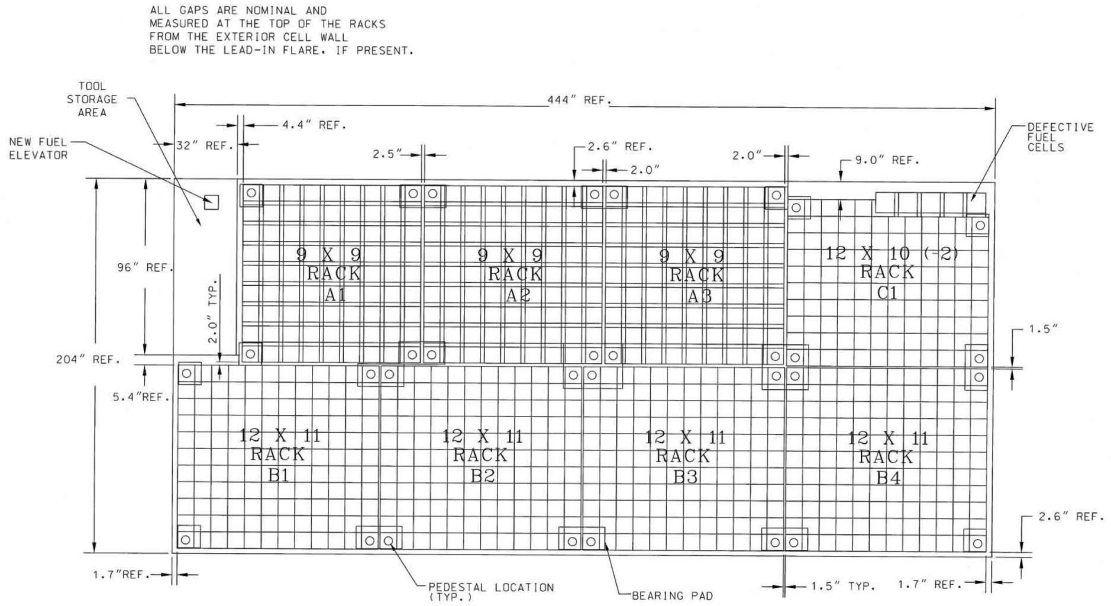

Figure 9.1-4
Spent Fuel Storage Pool Layout
(889 Storage Locations)

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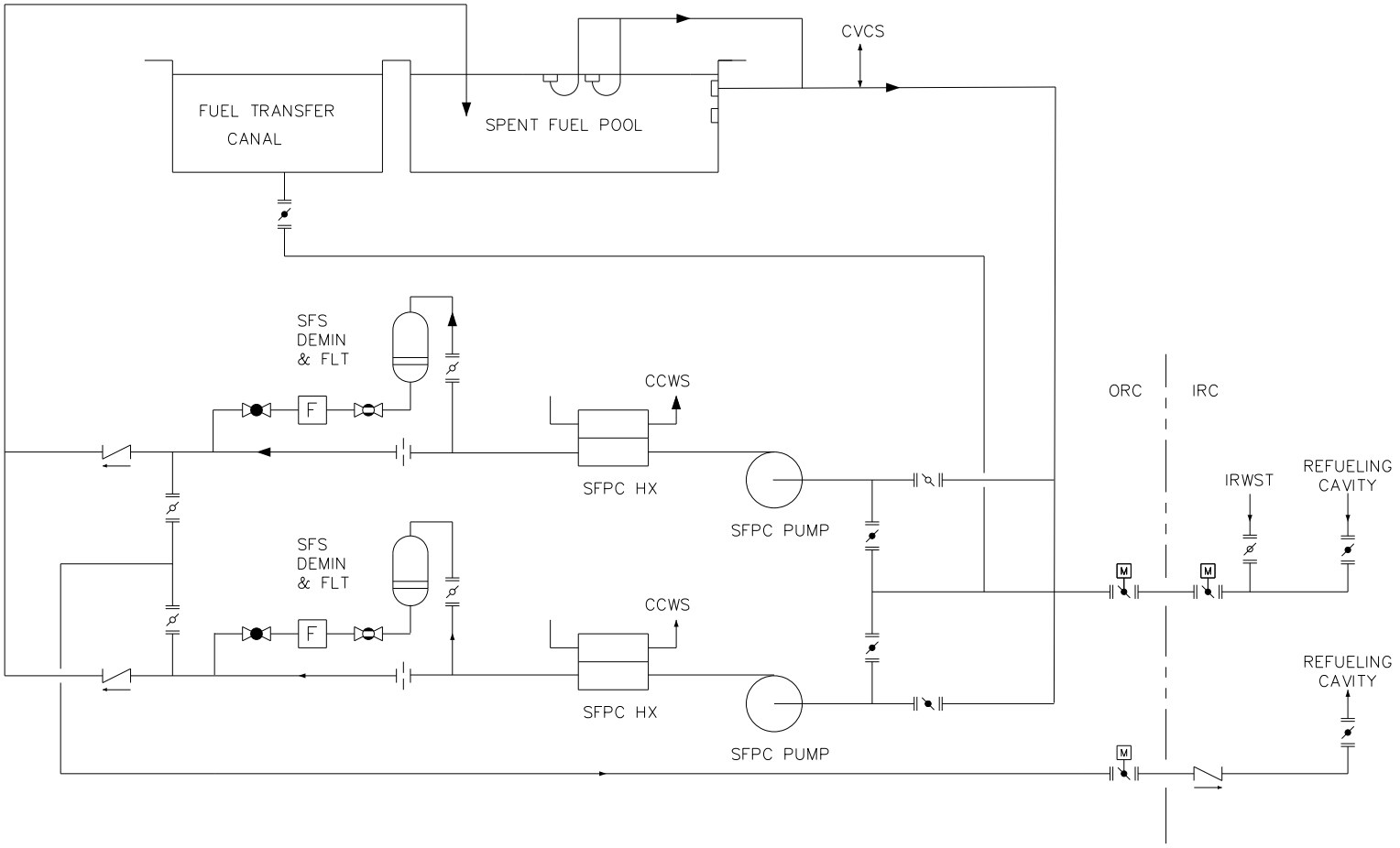

Figure 9.1-5
Spent Fuel Pool Cooling System
(Normal Operation)

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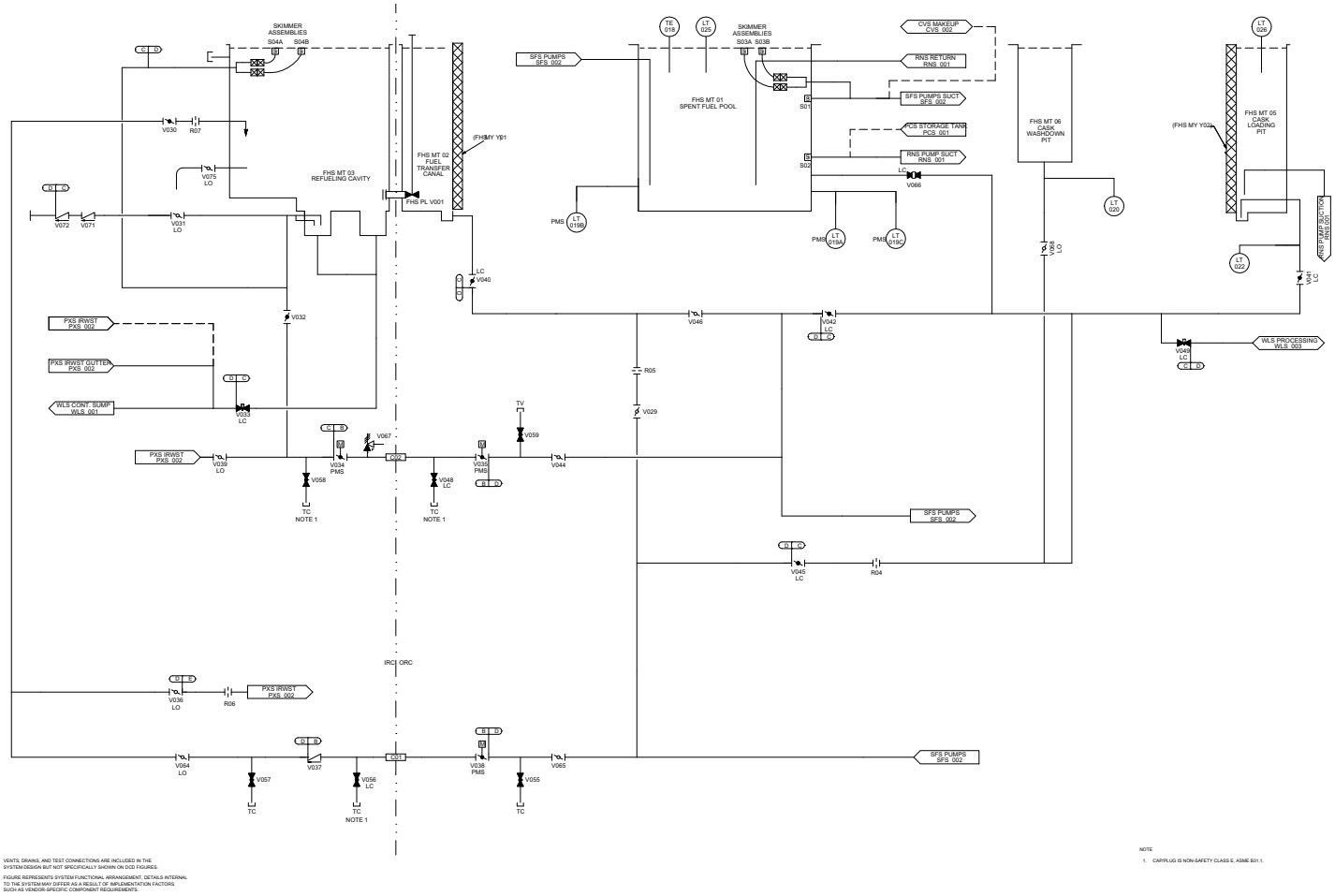

Figure 9.1-6 (Sheet 1 of 2)
Simplified Spent Fuel Pool Cooling System Piping and Instrumentation Diagram (REF) SFS 001

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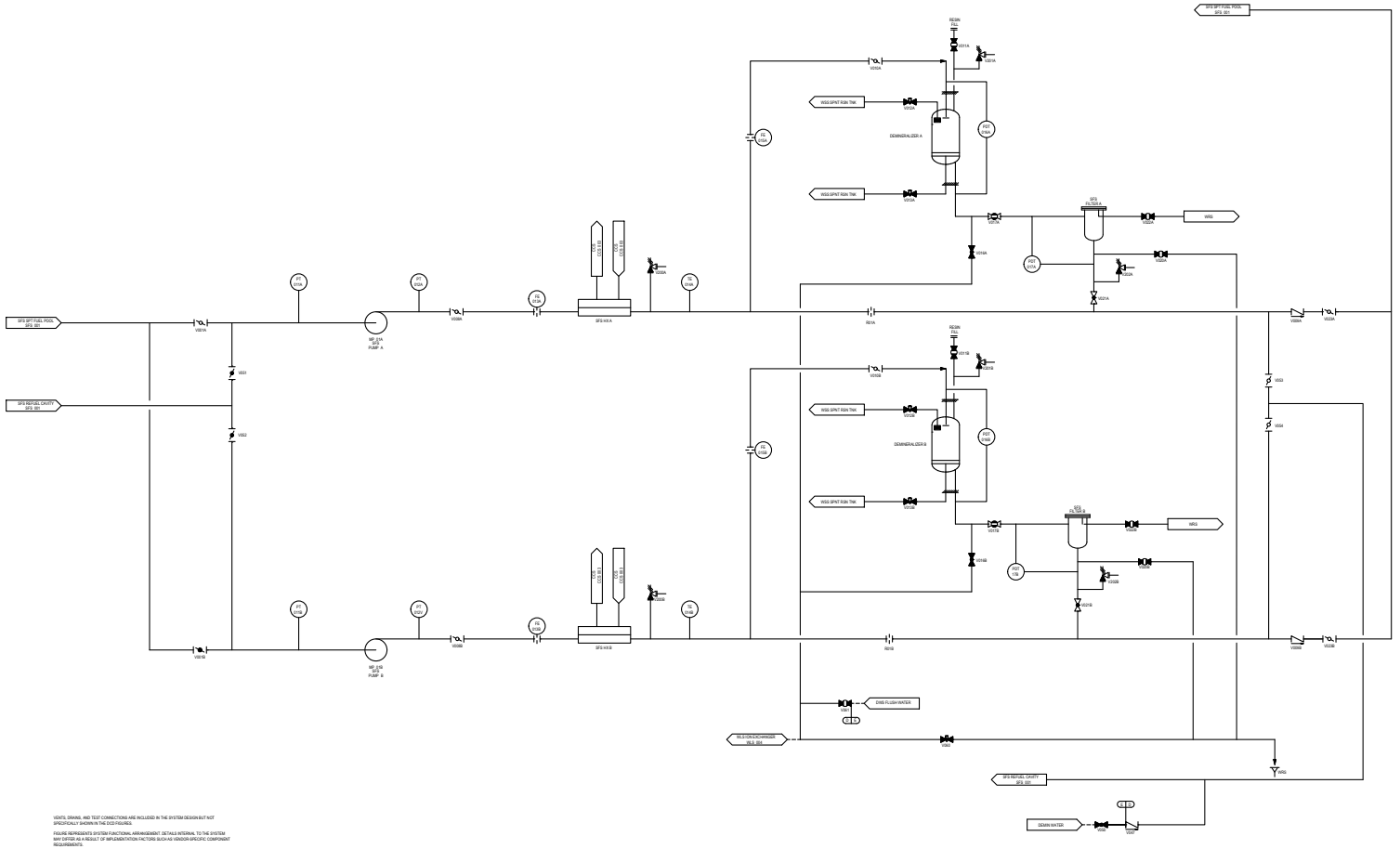

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9.2 Water Systems

9.2.1 Service Water System

The service water system (SWS) supplies cooling water to remove heat from the nonsafety-related component cooling water system (CCS) heat exchangers in the turbine building.

9.2.1.1 Design Basis

9.2.1.1.1 Safety Design Basis

The service water system serves no safety-related function and therefore has no nuclear safety design basis.

Failure of the service water system or its components will not affect the ability of safety-related systems to perform their intended function.

9.2.1.1.2 Power Generation Design Basis

The service water system provides cooling water to the component cooling water system heat exchangers located in the turbine building.

During normal power operation, the service water system supplies cooling water at a maximum temperature of 94.1°F to one component cooling water heat exchanger.

During plant cooldown and refueling, the service water system supplies cooling water to both component cooling water heat exchangers to support the cooling requirements for the component cooling water system specified in Subsection 9.2.2.1.2.

9.2.1.2 System Description

9.2.1.2.1 General Description

The service water system is shown in Figure 9.2.1-1. Classification of equipment and components is given in Section 3.2. The system consists of two 100-percent-capacity service water pumps, automatic backwash strainers, a two-cell cooling tower with a divided basin, and associated piping, valves, controls, and instrumentation.

The service water pumps, located in the turbine building, take suction from piping which connects to the basin of the service water cooling tower. Service water is pumped through strainers to the component cooling water heat exchangers for removal of heat. Heated service water from the heat exchangers then returns through piping to a mechanical draft cooling tower where the system heat is rejected to the atmosphere. Cool water, collected in the tower basin, flows through fixed screens to the pump suction piping for recirculation through the system.

A small portion of the service water flow is normally diverted to the circulating water system. This blowdown is used to control levels of solids concentration in the SWS. An alternate blowdown flow path is provided to the waste water system (WWS).

The service water system is arranged into two trains of components and piping. Each train includes one service water pump, one strainer, and one cooling tower cell. Each train provides 100-percent-capacity cooling for normal power operation. Cross-connections between the trains upstream and downstream of the component cooling water system heat exchangers allows either service water

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pump to supply either heat exchanger, and allows either heat exchanger to discharge to either cooling tower cell.

Temperatures in the system are moderate and the pressure of the service water system fluid is kept above saturation at all locations. This, along with other design features of the system arrangement and control of valves, minimizes the potential for thermodynamic or transient water hammer.

Service water system materials are compatible with the cooling water chemistry and the chemicals used for the control of long-term corrosion and organic fouling. Water chemistry is controlled by the turbine island chemical feed system (CFS).

Flooding of the turbine building resulting from a service water system failure is less severe than that for the circulating water system. Refer to Subsection 10.4.5.2.3 for a description of flooding due to the circulating water system.

9.2.1.2.2 Component Description

Service Water Chemical Injection

The turbine island chemical feed system equipment injects the required chemicals into the service water system. This injection maintains a noncorrosive, nonscale forming condition and limits biological film formation. Chemicals are injected into service water basin area.

The chemicals can be divided into six categories based upon function: biocide, nonoxidizing biocide, pH adjustor, corrosion inhibitor, scale inhibitor, and silt dispersant. Specific chemicals used within the system, other than the biocide, are determined by the site water conditions. The pH adjustor, corrosion inhibitor, scale inhibitor, and dispersant are metered into the system continuously or as required to maintain proper concentrations. A sodium hypochlorite treatment system is provided for use as the biocide and controls microorganisms that cause fouling. The biocide application frequency may vary with seasons. Nonoxidizing biocide is applied, as necessary, to control algae formation on the cooling tower. The impact of toxic material on main control room habitability is addressed in Section 6.4.

Chemical concentrations are measured through analysis of grab samples. Chlorine residual is measured to monitor the effectiveness of the biocide treatment. Addition of water treatment chemicals is performed by chemical feed system injection metering pumps and is adjusted as required.

Cooling Tower

The cooling tower is a rectilinear mechanical draft structure.

The cooling tower is a counterflow, induced draft tower and is divided into two cells. Each cell utilizes one fan, located in the top portion of the cell, to draw air upward through the fill counter to the downward flow of water. Each fan is driven by a two speed electrical motor through a gear reducer. During normal power operation, one cell is inactive and water flow to that cell is shut off by a motor operated isolation valve. One operating service water pump supplies flow to the operating cell. When the service water system is used to support plant shutdown cooling, both tower cells are normally placed in service along with both service water pumps, for increased cooling capacity.

The cooling tower cold water temperature is normally automatically controlled by operation of the tower fans. The fan in an active cell will be either on high speed, low speed or off, depending on the temperature of the heated service water returning to the cooling tower. When necessary, the water flow to each cooling tower cell can be diverted directly to the basin, bypassing the tower internals.

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This is achieved by opening a full flow bypass valve. The bypass can be used during plant startup in cold weather to maintain service water system temperature above 40°F.

After transiting through the cooling tower, cooled service water is collected in a basin located below the tower structure. The basin is partitioned into two halves, with each half collecting the segregated flow from one tower cell. An opening in the partition normally allows the two basin halves to communicate, but a stoplog can be inserted to allow one half of the basin to remain full while the other half is drained for maintenance. Raw water is automatically supplied to the basin to makeup for evaporation, drift and blowdown losses. An alternate makeup water supply is available by gravity flow from both of the fire protection storage tanks, using water that is not dedicated to fire protection purposes. With no makeup to the cooling tower basin, the storage capacity of the basin allows continued system operation for at least 12 hours under limiting conditions with a minimum usable volume of 230,000 gallons, provided that blowdown flow is isolated.

Service Water Strainers

A filter housing is provided in the service water supply piping to each component cooling water heat exchanger. The filter housing does not include strainers because the water quality does not typically require straining. If necessary, the strainer internals can be installed in the filter housing and periodic backwashing performed.

Service Water Pumps

The service water system includes two service water pumps providing cooling water in the quantities and operating conditions listed in Table 9.2.1-1.

The service water pumps are vertical, centrifugal, constant speed, electric motor-driven pumps. The pumping elements of each pump are enclosed within a suction barrel which connects to supply piping from the cooling tower basin. The suction barrel of each pump is located in the circulating water pipe trench area of the turbine building. The pumps are powered from the normal ac power system and are backed by the standby power source for occurrences of loss of normal ac power and offsite power. Each pump provides 100 percent of the normal power operation flow requirements and is therefore capable of supporting normal power operation with one pump out of service for maintenance.

The starting logic for the service water pumps requires at least one of the cooling tower valves to be open prior to pump start to provide a flow path through the cooling tower or tower bypass. The pump starting logic also interlocks with the motor operated valve at the discharge of each pump. The pump starts with the discharge valve closed and the valve then opens at a controlled rate to slowly admit water to the system while maintaining pump minimum flow. This feature results in reduced fluid velocities during system start to minimize transient effects that may occur as the system sweeps out air that may be present and obtains a water solid condition.

Piping

Service water piping is made of carbon steel and is designed, fabricated, installed, and tested in accordance with ASME B31.1, Power Piping Code. High density polyethylene piping constructed to the requirements of ASME B31.1, Appendix III is also used for the underground portions of the auxiliary makeup line from the primary and secondary fire water tanks, and for the underground portions of the SWS blowdown line to the CWS cooling tower. Cooling water supply and return piping is accessible for inspection and/or wall thickness determination. Cooling water supply and return piping that runs in the yard is either routed within trenches or may be inspected from the inside.

The service water system is designed to accommodate transient effects that may be generated by the normal starting and stopping of pumps, opening and closing of valves, or other normal operating events. The system pumps water from the basin at the cooling tower, through piping and equipment,

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to a high point located at the cooling tower riser; the cooling water is then discharged in a spray fashion above the cooling tower basin. The system arrangement is such that high points in the system piping do not lead to the formation of vapor pressure voids upon loss of system pumping. Therefore, the potential for water hammer due to vapor collapse upon pump start is minimized.

Service Water System Valves

Manual isolation valves upstream and downstream of each component cooling water system heat exchanger can be used to isolate the heat exchanger and associated strainer from the service water system. The upstream valves are also normally used during power operation to align the service water system to the component cooling water heat exchanger in use by blocking flow to the inactive heat exchanger. Manual valves in the cross-connection lines between the two service water trains are normally open during power operation to allow the standby pump or standby cooling tower cell to quickly be placed in service if needed. The cross-connection valves are closed as necessary to isolate portions of the system for maintenance, and are normally closed when the system is configured for plant shutdown cooling with both trains in operation.

A motor operated isolation valve downstream of each pump automatically closes when the associated pump stops and automatically opens when the pump starts. Motor operated isolation valves are also used at the cooling tower to isolate flow to a cell that is inactive or out of service for maintenance. The motor operated valves for each train of service water pumps and cooling tower cells are powered by the same onsite standby power source as the associated pump and cooling tower cell fan.

The service water strainers are provided with air-operated backwash valves which open during a backwash cycle. These valves fail closed upon loss of control air or electrical power.

An air operated control valve is provided in the cooling tower blowdown line. This valve allows the plant operator to set the blowdown flowrate. The valve also provides automatic isolation of blowdown flow upon loss of offsite power. The valve fails closed upon loss of control air or electrical power.

Heat Exchangers

The heat exchangers served by the service water system are part of the component cooling water system. For information concerning the component cooling water system heat exchangers refer to Subsection 9.2.2.

9.2.1.2.3 System Operation

The service water system operates during normal modes of plant operation, including startup, power operation (full and partial loads), cooldown, shutdown, and refueling. The service water system is also available during loss of normal ac power conditions.

9.2.1.2.3.1 Service Water System Startup

For initial system startup, service water piping and equipment can be filled with raw water. Thereafter, at least one train normally remains in service. An inactive train is started by starting the associated pump and realigning valves as required.

9.2.1.2.3.2 Plant Startup

During plant startup, the service water system normally provides service to both component cooling water system heat exchangers. This requires that both service water pumps, strainers and cooling tower cells be in service. At the end of this phase of operation, when one of the component cooling water system heat exchangers is removed from service, one of the service water pumps, strainers

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and cooling tower cells may also be removed from service. Refer to Subsection 9.2.2 for a description of plant startup operation and the conditions under which two component cooling water system heat exchangers may be required.

9.2.1.2.3.3 Power Operation

The service water system, during normal power operation, provides cooling water at a maximum temperature of 94.1°F to the component cooling water heat exchanger in service. One service water pump and one cooling tower cell are in service. The flow rate and heat load are shown in Table 9.2.1-1.

The standby service water pump is automatically started if the operating pump should fail, thereby providing a reliable source of cooling water. The system is designed so either pump can serve as the operating or standby pump.

9.2.1.2.3.4 Plant Cooldown/Shutdown

During the plant cooldown phase in which the normal residual heat removal system has been placed in service and is providing shutdown cooling, the service water cooling tower provides cooling water at a temperature of 88.6°F or less when operating at design heat load and at an ambient wet bulb temperature of no greater than the maximum normal wet bulb temperature as defined in Chapter 2, Table 2-1. Two service water pumps and two cooling tower cells are normally used for plant cooldown, and the cross-connection valves between trains are normally closed. The service water system heat load and flow rate are shown in Table 9.2.1-1. During these modes of operation the normal residual heat removal system and the component cooling water system remove sensible and decay heat from the reactor coolant system. The service water system cooling towers are designed with sufficient margin so that normal time-related degradation of tower performance will not prohibit their support of this heat removal function. In the event of failure of a service water system pump or cooling tower fan, the cooldown time is extended.

9.2.1.2.3.5 Refueling

During refueling, the service water system normally provides cooling water flow to both component cooling water system heat exchangers. Two service water pumps normally provide flow through the system for refueling modes.

9.2.1.2.3.6 Loss of Normal AC Power Operation and Offsite Power

In the event of loss of normal ac power and offsite power, the service water pumps and cooling tower fans, along with the associated motor operated valves, are automatically loaded onto their associated diesel bus. This includes isolation of cooling tower blowdown, which minimizes drain down of the system while both pumps are off. What drainage of system fluid that does occur is replaced by air without vapor cavities. The potential for water hammer on pump restart is minimized. Both pumps and both cooling tower cells automatically start after power from the diesel generator is available. Following automatic start, the operator may return the system to the appropriate configuration.

9.2.1.3 Safety Evaluation

The service water system has no safety-related functions and therefore requires no nuclear safety evaluation. If radioactive fluid is detected in the service water system, tower blowdown flow can be isolated by remote manual control. The tower blowdown valve fails closed upon loss of electrical power or instrument air.

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9.2.1.4 Tests and Inspections

Preoperational testing is described in Chapter 14. The performance, structural, and leaktight integrity of system components is demonstrated by operation of the system.

9.2.1.5 Instrument Applications

Pressure indication, with low and high alarms, is provided for the discharge of each service water pump. A low pressure signal automatically starts the standby pump. Flow indication, with low and high alarms, is also provided for each service water pump. Due to the system configuration, pump flow indication can also normally be used to monitor flow through the heat exchanger or heat exchangers in service.

Temperature indication is provided for the service water supply to each component cooling water heat exchanger and for the discharge from each heat exchanger to determine the temperature differential across the heat exchanger. Heat exchanger inlet temperature indication also is used for performance monitoring of the service water cooling tower. Low and high heat exchanger inlet temperature alarms are provided. A high alarm is provided for the outlet temperature from each heat exchanger. Temperature instrumentation is provided for the service water return to each cooling tower cell to automatically control the operation of the associated cell fan.

Differential pressure measurement across each service water strainer is provided and will initiate backwash of the strainer on high differential. A high-high differential pressure alarm across the strainer is provided.

Power actuated valves in the SWS are provided with valve position indication instrumentation. In addition, the tower bypass valves are provided with position indication instrumentation.

Level indication is provided for the cooling tower basin along with high and low level alarms. The basin level signal is also used to control the normal makeup water supply valve to maintain the proper level in the cooling tower basin.

A radiation monitor with a high alarm is provided to monitor the service water blowdown flow for detection of potentially radioactive leakage into the SWS from the component cooling water heat exchangers. Provisions are also available for taking local fluid samples.

9.2.2 Component Cooling Water System

The component cooling water system is a non-safety-related, closed loop cooling system that transfers heat from various plant components to the service water system during normal phases of operation. It removes heat from various components needed for plant operation and removes core decay heat and sensible heat for normal reactor shutdown and cooldown.

The AP1000 component cooling water system provides a barrier to the release of radioactivity between the plant components being cooled that handle radioactive fluid and the environment. The component cooling water system also provides a barrier against leakage of service water into primary containment and reactor systems.

9.2.2.1 Design Bases

9.2.2.1.1 Safety Design Basis

Failure of the component cooling water system or its components will not affect the ability of safety-related systems to perform their intended safety functions. The component cooling water system

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serves no safety-related function except for containment isolation and therefore has no nuclear safety design basis except for containment isolation (see Subsection 6.2.3).

9.2.2.1.2 Power Generation Basis

The component cooling water system is designed to perform its operational functions in a reliable and failure tolerant manner. This reliability is achieved with the use of reliable and redundant equipment and with a simplified system design.

9.2.2.1.2.1 Normal Operation

The component cooling water system transfers heat from various plant components needed to support normal power operation with a single active component failure. The component cooling water system is designed for normal operation in accordance with the following criteria:

- The component cooling water supply temperature to plant components is not more than 100°F assuming a 0 percent exceedance ambient design wet bulb temperature of 86.1°F for service water cooling at normal operations (maximum normal temperature per Table 2-1 for normal shutdown).
- The minimum component cooling water supply temperature to plant components is 60°F.
- The component cooling water system provides sufficient surge capacity to accept 50 gallons per minute leakage into or out of the system for 30 minutes before any operator action is required.

9.2.2.1.2.2 Normal Plant Cooldown

The first phase of plant cooldown is accomplished by transferring heat from the reactor coolant system via the steam generators to the main steam systems.

The component cooling water system, in conjunction with the normal residual heat removal system removes both residual and sensible heat from the core and the reactor coolant system and reduces the temperature of the reactor coolant system during the second phase of cooldown.

The component cooling water system reduces the temperature of the reactor coolant system from 350°F at approximately 4 hours after reactor shutdown to 125°F within 96 hours after shutdown by providing cooling to the normal residual heat removal system heat exchangers. This cooldown time is based on operation of both component cooling water system mechanical trains (one pump and one heat exchanger each), and a service water system supply temperature to the component cooling water system heat exchangers resulting from a maximum normal ambient design wet bulb temperature as defined in Table 2-1 for service water cooling. In addition to the cooldown time requirements, other system design criteria during cooldown are:

- Operation is consistent with the established reactor coolant system cooldown rates while maintaining the component cooling water supply below 110°F.
- The system design prevents boiling in the component cooling water system during plant cooldown.

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- A single failure of an active component during normal cooldown will not cause an increase in reactor coolant system temperature above 350°F. Such a single failure also will not cause the reactor coolant system to boil once the reactor vessel head has been removed and the refueling cavity flooded. The component cooling system continues to provide cooling water to the normal residual heat removal system throughout the shutdown after cooldown is complete.

9.2.2.1.2.3 Refueling

During fuel shuffling (partial core off-load) or a full core off-load, cooling water flow is provided to spent fuel pool heat exchangers to cool the spent fuel pool. For a full core off-load cooling water is also supplied to a normal residual heat removal heat exchanger as part of spent fuel pool cooling. The system design criteria during refueling are:

- System operation is with both component cooling water system mechanical trains available.

- The component cooling water system maintains the spent fuel pool water temperature below 120°F based on a maximum normal ambient design wet bulb temperature as defined in Table 2-1 for service water cooling.

9.2.2.1.3 Codes and Standards

The component cooling water system equipment applicable codes and standards are listed in Section 3.2. The containment penetrations, isolation valves, and the pipe between the isolation valves are Safety Class B. A small section of the containment supply and return lines just inside the innermost containment isolation valve is designated Safety Class C. This section of line contains the relief valves provided to protect the containment isolation valves from excess pressure buildup while being closed automatically to isolate a reactor coolant pump external heat exchanger tube leak. The remainder of the component cooling water system piping is designed to ASME B31.1.

9.2.2.2 System Description

The component cooling water system provides a reliable supply of cooling water to the various plant components listed in Table 9.2.2-2.

A simplified sketch of the component cooling water system is included as Figure 9.2.2-1. The details of the system are shown in the piping and instrumentation diagram for the component cooling water system which is included as Figure 9.2.2-2.

The component cooling water system is a closed loop cooling system that transfers heat from various plant components to the service water system cooling tower. It operates during normal phases of plant operation including power operation, normal cooldown, and refueling. The system includes two component cooling water pumps, two component cooling water heat exchangers, one component cooling water surge tank and associated valves, piping, and instrumentation.

The system components are arranged into two mechanical trains. Each train includes one component cooling water pump and one component cooling water heat exchanger. The two trains of equipment take suction from a single return header. The surge tank is connected to the return header. Each pump discharges directly to its respective heat exchanger. A bypass line around each heat exchanger containing a throttle valve prevents overcooling the component cooling water. The discharge of each heat exchanger is to the common supply header.

Component cooling water is distributed to the components by this single supply/return header. Components are grouped in branch lines according to plant arrangement, with one branch line

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cooling the components inside containment. Loads inside containment are automatically isolated in response to a safeguards actuation signal, which also trips the reactor coolant pumps, and in response to a High-2 bearing water temperature trip signal from one of the reactor coolant pumps. Individual components, except the reactor coolant pumps, can be isolated locally to permit maintenance while supplying the remaining components with cooling water.

The component cooling water surge tank accommodates thermal expansion and contraction. It also accommodates leakage into or out of the component cooling water system until the leak is isolated. Water makeup to the surge tank is provided automatically on a low surge tank level signal by the demineralized water transfer and storage system. A line routed from the pump discharge header to the surge tank includes a mixing tank to add chemicals into the system to inhibit corrosion.

9.2.2.3 Component Description

General descriptions of the component cooling water system components are provided below. The nominal equipment parameters for the component cooling water system components are contained in Table 9.2.2-1.

9.2.2.3.1 Component Cooling Water Pumps

The two component cooling water pumps are horizontal, centrifugal pumps. They have a coupled pump shaft driven by an ac powered induction motor. Each pump provides the flow required by its respective heat exchanger for removal of its heat load. The pumps are redundant for normal operation heat loads. Both pumps are required for the cooldown; however, an extended cooldown can be achieved with only one pump in operation. One pump can be out of service during normal plant operation.

These pumps are risk-significant and are included within the scope of D-RAP. See Table 17.4-1 for further information.

9.2.2.3.2 Component Cooling Water Heat Exchangers

Two component cooling water heat exchangers provide redundant cooling for normal operation heat loads. Both heat exchangers are required to achieve the design cooldown rate; however, an extended cooldown can be achieved with one heat exchanger in operation. Either heat exchanger can be aligned with either component cooling water pump allowing one heat exchanger to be out of service during normal plant operation.

The component cooling water heat exchangers are plate type heat exchangers. Component cooling water circulates through one side of the heat exchanger while service water circulates through the other side. Component cooling water in the heat exchanger is maintained at a higher pressure than the service water to prevent leakage of service water into the system.

9.2.2.3.3 Component Cooling Water Surge Tank

The component cooling water system has a single surge tank. The surge tank accommodates changes in component cooling water volume due to changes in operating temperature. During normal operation, the tank is designed to accommodate a 50 gallons per minute leakage into or out of the system for 30 minutes before any operator action is required. For abnormal operation, the surge tank vent line is sized to accommodate a double-ended tube rupture in the normal residual heat removal system (RNS) heat exchanger (i.e., 520 gpm) without exceeding the surge tank design pressure or allowing pressure to increase above CCS system design pressure at the highest pressure point in the system.

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The tank is a cylindrical, vertical unit constructed of carbon steel.

9.2.2.3.4 Component Cooling Water System Valves

Most of the valves in the component cooling water system are manual valves used to isolate cooling flow from components for which cooling is not required in a given plant operating mode.

Three motor-operated isolation valves and a check valve provide containment isolation for the supply and return component cooling water system lines that penetrate the containment barrier. The motor-operated valves are normally open; however, they are closed upon receipt of a safeguards actuation signal, or a High-2 bearing water temperature reactor trip signal. They are controlled from the main control room and fail as-is.

A relief valve is provided to prevent overpressurization of the return component cooling water line containment penetration. This relief valve prevents overpressurization that might be caused by thermal expansion of the fluid between the containment isolation valves following an event causing containment isolation. This relief valve is located inside the containment.

An air-operated isolation valve is located in the component cooling water discharge line from each reactor coolant pump. These valves, which are normally open, can be closed by an operator from the MCR, following a flow deviation alarm. The alarm is produced when there are flow deviations detected simultaneously in the reactor coolant pump CCS cooling supply and return lines that are indicative of a leak from the pump external heat exchanger to the CCS. Closing these valves prevents radioactive reactor coolant flow through the component cooling water system.

An air-operated isolation valve is located in the component cooling water discharge line from each VWS chiller. These valves are opened by the same signals that initiate the associated pump.

Relief valves are provided in the cooling water outlet line from each reactor coolant pump. These valves are sized to protect the pump motor cooling jacket (design pressure 200 psig) and the component cooling water piping in the event of a tube rupture in the reactor coolant pump external heat exchanger. A relief valve in the cooling water outlet line from the letdown heat exchanger also protects the component cooling water piping in the event of a tube rupture in the heat exchanger. Small relief valves are included in the cooling water outlet line from the other components to relieve the volumetric expansion which occurs if the cooling water lines to the component are isolated and the water temperature rises. One relief valve is also provided in each Safety Class C section of piping, just inside the innermost containment isolation valves on the CCS 10-inch supply and return lines penetrating the containment barrier, to ensure protection of the containment isolation valves from excess pressure while closing on a reactor coolant pump High-2 bearing water temperature trip signal intended to isolate a potential external heat exchanger tube leak. In addition, these valves provide protection of containment isolation valves in the event of a letdown heat exchanger tube rupture.

Relief valves in the cooling water outlet lines from each normal residual heat exchanger also protect the RNS heat exchanger shell and component cooling water piping in the event of a tube leak that occurs while the heat exchanger is isolated from the CCS. These relief valves also provide thermal relief for an isolated RNS heat exchanger.

9.2.2.3.5 Piping Requirements

Component cooling water system piping is made of carbon steel. Piping joints and connections are welded, except where flanged connections are required as indicated on the component cooling water system piping and instrumentation diagram (Figure 9.2.2-2).

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9.2.2.4 System Operation and Performance

9.2.2.4.1 Plant Startup

Plant startup is the operation that brings the reactor plant from a cold shutdown condition to no-load operating temperature and pressure, and subsequently to power operation.

Normally both component cooling water system mechanical trains are operating during this post refueling period. Both trains are aligned to provide cooling to the required components as shown in Table 9.2.2-2.

When plant heatup is initiated, the reactor coolant pumps are started, and residual heat removal from the core is discontinued by stopping the residual heat removal pumps. The letdown heat exchanger is placed on automatic temperature control to maintain a constant letdown temperature. Throughout the plant startup, cooling water flows and temperatures are monitored to verify that the values are within the required limits. Once startup activities are complete, one component cooling water pump and one heat exchanger are taken out of service.

9.2.2.4.2 Normal Operation

During normal plant operation, one component cooling water system mechanical train of equipment is operating. The standby train is aligned to automatically start in case of a failure of the operating component cooling water pump. The system is aligned to provide cooling water flows to the appropriate equipment as shown on Table 9.2.2-2.

During normal operation, leakage from the component cooling water system is replaced by automatic actuation of a valve in the makeup line on low surge tank level.

Periodically, a sample of the component cooling water is taken by the plant operator to ascertain that water chemistry specifications are met. If necessary, appropriate chemicals are added via the chemical addition tank and mixing is achieved through a recirculation line from the pump discharge header, through the surge tank to the pump suction line.

9.2.2.4.3 Plant Shutdown

Plant shutdown is the operation that brings the reactor plant from power operation to refueling conditions. During plant shutdown operations, both component cooling water system mechanical trains normally operate. The system is aligned to provide the cooling water flows to the appropriate equipment as shown on Table 9.2.2-2.

The initial phase of plant cooldown is the reactor coolant system cooldown and depressurization utilizing the steam generators and the main steam system. The second phase of plant cooldown is initiated by placing the normal residual heat removal system in service when the reactor coolant temperature and pressure have been reduced to 350°F and 400 to 450 psig, respectively (approximately 4 hours after reactor shutdown).

Prior to starting the residual heat removal pumps, the standby component cooling water pump and heat exchanger are placed in operation and component cooling water flow is initiated to the normal residual heat removal heat exchangers. Following this, the normal residual heat removal system can be placed into operation by properly aligning valves and starting a residual heat removal pump.

The component cooling water system, in conjunction with the normal residual heat removal system and service water system cools the reactor coolant system to 125°F within 96 hours after shutdown. During the cooldown period, the component cooling water inlet temperature to the various

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components does not exceed 110°F. Both component cooling water pumps and heat exchangers are required to meet the plant cooldown schedule. In the event of a failure of a component cooling water pump or heat exchanger, the cooldown time is extended.

9.2.2.4.4 Refueling

Both component cooling water system mechanical trains are in operation during refueling. The system is aligned to provide the required flow to the appropriate components as shown on Table 9.2.2-2.

For fuel shuffling (partial core off-load), cooling water flow is provided to both spent fuel pool heat exchangers to maintain the spent fuel pool water temperature below 120°F. With a full core off-load and 10 years accumulation of spent fuel in the pool, both spent fuel pool heat exchangers and one normal residual heat removal heat exchanger maintain the spent fuel pool water temperature below 120°F.

9.2.2.4.5 Abnormal Conditions

9.2.2.4.5.1 Failure of a Component Cooling Water Pump

If a component cooling water pump fails when one pump is in service, an alarm is actuated and the low header flow signal automatically initiates operation of the standby component cooling water pump. If a component cooling water pump fails during plant cooldown, the time to reach the cold shutdown condition is increased.

9.2.2.4.5.2 Leakage into the Component Cooling Water System from a High Pressure Source

Small leakage of reactor coolant into the component cooling water system is detected by a radiation monitor on the common pump suction header, by routine sampling, or by high level in the surge tank.

Flow sensors located in the cooling water inlet and outlet lines from each reactor coolant pump external heat exchanger also detect leakage from a heat exchanger tube into the component cooling water system. Simultaneous flow deviations in both the inlet and outlet lines will generate a flow deviation alarm; this alarm is indicative of leak conditions and would alert the operator to close the valve on the cooling water outlet line on each reactor coolant pump to prevent reactor coolant flow throughout the component cooling water system. Both the flow signals and the isolation valves are nonsafety-related. If the valve on the reactor coolant pump cooling water outlet line is not closed, reactor coolant leakage from the pump can be retained inside containment by closing the safety-related component cooling water containment isolation valves. These containment isolation valves close automatically if the leak rate is sufficiently large to cause a High-2 bearing water temperature reactor and pump trip signal to be generated by the protection and safety monitoring system (PMS). The containment isolation valves can also be closed manually by the operator after being alerted to a reactor coolant pump leak by alarms from component cooling water system instrumentation (surge tank level and/or radiation level in the CCS pump suction header) or from the flow instruments located on the inlet and outlet lines from the leaking reactor coolant pump external heat exchanger. Manual closure of one CCS outlet isolation valve will result in a High-2 bearing water temperature trip of the plant if the affected reactor coolant pump continues to operate.

A safeguards actuation signal results if sufficient reactor coolant system inventory is lost through the leak. This signal will trip the reactor coolant pumps and automatically close the component cooling water containment isolation valves to prevent reactor coolant leakage outside containment. Overpressure protection of the reactor coolant pump motor cooling jacket and the component cooling water piping subjected to the reactor coolant system pressure is provided by means of a relief valve

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on the cooling water outlet piping downstream of the reactor coolant pump external heat exchanger. Two additional relief valves, one on each CCS cooling water line penetrating the containment inside the innermost containment isolation valve, are provided to protect the containment isolation valves from overpressure while being closed to isolate a high-pressure leak in the CCS inside containment.

The operator is alerted to a large leak from the letdown heat exchanger by a high surge tank level or a high radiation alarm in the absence of a signal from one or both of the reactor coolant pump inlet or outlet flow sensors indicating reactor coolant leakage from a reactor coolant pump external heat exchanger to the component cooling water system. The operator can isolate the reactor coolant flow to the letdown heat exchanger from the main control room by closing the letdown flow isolation valve in the chemical and volume control system. Overpressure protection for the component cooling water system in the case of a letdown heat exchanger tube rupture is provided by a relief valve in the component cooling water system piping near the heat exchanger outlet.

During a normal plant cooldown a normal residual heat removal heat exchanger tube leak or rupture could result in reactor coolant leakage into the component cooling water system. A check of the local flow measurements in the normal residual heat removal heat exchanger cooling water outlet lines will indicate the leaking heat exchanger. High surge tank level and/or high radiation level in the CCS pump suction header would also confirm this condition. Reactor coolant flow to the faulty heat exchanger can be isolated by closing valves in the normal residual heat removal system. If the RNS heat exchanger is isolated from the remainder of the CCS, overpressure protection for the component cooling water system and RNS heat exchanger shell in the case of a normal residual heat exchanger tube rupture is provided by a relief valve in the component cooling water system piping near each heat exchanger outlet.

9.2.2.4.5.3 Leakage from the Component Cooling Water System

Excessive leakage from the component cooling water system causes the water level in the component cooling water surge tank to drop and a low level alarm to be actuated. Makeup water is added automatically to the component cooling water system as required. After the leak is identified by visual inspection or by a change in individual component cooling water flow rate, the affected cooling water circuit containing the leak is isolated from the component cooling water system.

9.2.2.4.5.4 Loss of Normal AC Power Operation and Offsite Power

The component cooling water pumps are automatically loaded on the standby diesel in the event of a loss of normal ac power and offsite power. The component cooling water system therefore continues to provide cooling of required components if normal ac power and offsite power is lost.

9.2.2.4.5.5 Fire Leading to MODE 5, Cold Shut Down

In the event of a loss of normal component cooling system function the Fire Protection System can provide the source of cooling water for a Normal Residual Heat Removal System (RNS) heat exchanger and a RNS pump. Normally closed isolation valves between the Fire Protection System and the Component Cooling Water System are manually opened. An additional valve is manually closed to prevent supply of cooling water to other heat exchangers which are not needed to provide cooling for the Reactor Coolant System. A drain valve on the component cooling water return header is opened and the Fire Protection System water is released after passing through the RNS heat exchanger. The flow rate of Fire Protection System water is controlled manually to conserve the supply.

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9.2.2.5 Evaluation

The component cooling water system penetrates the containment boundary. The containment penetration lines are designed in accordance with the containment isolation criteria system specified in Subsection 6.2.3. The containment isolation valve design evaluation and effects of failures are also presented in Subsection 6.2.3.

The component cooling water system can remove the required heat load during a loss of normal ac power.

The acceptability of the design of the component cooling water system is based on specific General Design Criteria (GDCs) and regulatory guides. The design of the component cooling water system has been compared to the criteria set forth in Subsection 9.2.2, "Reactor Auxiliary Cooling Water System," Revision 3, of the NRC's Standard Review Plan. The specific General Design Criteria identified in the Standard Review Plan section are General Design Criteria 2, 4, 5, 44, 45 and 46. Additionally, Regulatory Guide 1.29 was reviewed to determine the degree of compliance of the AP1000 design with the criteria. Branch Technical Position ASB 3-1 and IEEE 279 were also reviewed as appropriate. The compliance of the component cooling water system design with the applicable General Design Criteria and regulatory guides is discussed in Section 3.1 and Subsection 1.9.1, respectively.

9.2.2.6 Inspection and Testing Requirements

9.2.2.6.1 Preoperational Inspection and Testing

Preoperational testing of the component cooling water system is performed to verify that the system is installed in accordance with plans and specifications. The system is hydrostatically tested and is also tested to verify that proper sequence of valve positions and pump starting occurs on the appropriate signals. The pumps are tested to verify performance and the required flows to the individual components are obtained by proper orifice installation and/or valve setting.

9.2.2.6.1.1 Pump Flow Capability Testing

Each component cooling water system pump will be tested during hot functional testing. The flow paths will be aligned for shutdown cooling by one train of component cooling water system components. The flow delivered to one normal residual heat removal system heat exchanger and one spent fuel pool cooling system heat exchanger, as well as the total component cooling water system flow, will be measured by flow instruments at the normal residual heat removal system heat exchanger, spent fuel pool cooling system heat exchanger, and component cooling water system pump discharge header.

9.2.2.6.1.2 Heat Transfer Capability Analysis

An analysis will be performed on the component cooling water system heat exchangers during heat exchanger design. The analysis is to confirm that the product of the overall heat transfer coefficient and effective heat transfer area, UA, of each heat exchanger is equal to or greater than the minimum value shown in Table 9.2.2-1. This is the minimum value for the component cooling water system to meet its functional requirement of shutdown heat removal and spent fuel pool cooling.

9.2.2.6.2 Routine Testing and Inspection

During normal operation, the standby pump and heat exchanger are periodically tested for operability, or alternatively, placed in normal operation in place of the train which had been operating.

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Component cooling water system supply and return containment isolation valves are routinely tested during refueling outages. Descriptions of the testing and inspection programs for these valves are provided in Subsections 3.9.6 and 6.2.3, and Section 6.6.

9.2.2.7 Instrumentation Requirements

Instruments are provided for monitoring system parameters. Essential system parameters are monitored in the main control room. Low flow in the discharge header automatically starts the backup component cooling water pump. A radiation monitor alarms in the main control room if reactor coolant leaks into the component cooling water system.

Level instrumentation on the surge tank provides both high- and low-level alarms in the main control room. Two redundant level channels are provided to reduce the likelihood of reactor trip caused by a single downscale failure of a surge tank level channel that could cause the operating component cooling water pump(s) to trip, thereby initiating loss of cooling flow to the reactor coolant pumps and other cooled components. Also, at a low-tank level, a valve in the makeup water line is automatically actuated by one of the two level channels to provide makeup flow from the demineralized water transfer and storage system into the component cooling water system.

Flow alarms in the main control room, produced by the two flow channels located on the CCS reactor coolant pump cooling water inlet and outlet lines, will alert the operator to a leak from the reactor coolant pump external heat exchanger into the component cooling water system. Signals generated by the PMS, in the event of a High-2 bearing water temperature trip of the reactor, also close the CCS containment isolation valves to eliminate the possibility of reactor coolant from a faulted external heat exchanger tube discharging to portions of the CCS outside the containment.

Component cooling water flow instrumentation is provided in the outlet line from the remaining components as shown in Figure 9.2.2-2.

9.2.3 Demineralized Water Treatment System

The demineralized water treatment system (DTS) receives water from the raw water system (RWS), processes this water to remove ionic impurities, and provides demineralized water to the demineralized water transfer and storage system (DWS). The demineralized water transfer and storage system is described in Subsection 9.2.4.

9.2.3.1 Design Basis

9.2.3.1.1 Safety Design Basis

The demineralized water treatment system serves no safety-related function and therefore has no nuclear safety design basis.

9.2.3.1.2 Power Generation Design Basis

- The demineralized water treatment system provides makeup and fill water to the demineralized water storage tank.
- The capacity of the demineralized water treatment system is sufficient to supply the plant makeup demand during startup, shutdown, and power operation.
- The quality of the water produced by the demineralized water treatment system is in accordance with the guidelines specified in Table 9.2.3-1.

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9.2.3.2 System Description

9.2.3.2.1 General Description

Component and equipment classification for the demineralized water treatment system is given in Section 3.2. The system consists of the following major components:

- Reverse osmosis (RO) feed pumps
- Reverse osmosis units for primary demineralization
- Ultraviolet light water treatment units
- EDI units for secondary demineralization
- Mixed bed ion exchange unit
- Cartridge filters
- Reverse osmosis permeate pumps
- Clean in-place (CIP) unit
- Sample panel unit

9.2.3.2.2 Component Description

Cartridge Filter

Two cartridge-type filters arranged in a parallel configuration are provided upstream of the reverse osmosis units. These filters remove particulate such as silt or pipe scale which can foul the reverse osmosis membranes.

Reverse Osmosis Feed Pumps

The design consists of high-pressure centrifugal feed pumps. The pumps maintain the required flow and pressure through the reverse osmosis membranes.

Reverse Osmosis Unit

Each reverse osmosis unit consists of thin film composite membranes enclosed in Fiberglass Reinforced Plastic (FRP) pressure vessels. The system piping arrangement of the individual reverse osmosis units permits operational flexibility for one train to be out of service.

Manual isolation valves are furnished on the product and feed lines of each reverse osmosis unit and reject line. Sample valves are furnished on product and reject streams from each pressure vessel.

PVC piping may be used in low pressure portions of the system. Corrosion-resistant 316 stainless steel is used in higher pressure portions of the system. A pressure sensor, located on the product manifold, protects the membranes from overpressurization by alarming and shutting down the reverse osmosis unit.

Cleaning connections are provided on each of the reverse osmosis units.

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Reverse Osmosis Permeate Transfer Pumps

High-pressure, centrifugal transfer pumps are provided for the ultraviolet light treatment units, electrodeionization units, and mixed bed ion-exchange units. The pump maintains the required flow and pressure through the mixed bed polishing vessels.

Ultraviolet Light Treatment Unit

Ultraviolet light water treatment units are used to reduce total organic carbon (TOC). The ultraviolet light breaks apart organic molecules. The unit uses light bulbs that produce ultraviolet light, which breaks down the organic molecules to simpler compounds or elements.

Electrodeionization Unit

Electrodeionization (EDI) is used for secondary demineralization and the removal of dissolved carbon dioxide gas. The electrodeionization unit consists of multiple component stacks. Each stack component contains cell pairs of membranes. One cell pair consists of an ion-diluting flow (product) channel located between a cation and an anion membrane with an ion concentrating (brine) flow channel located alternately between the cell pairs. A DC potential is maintained across the electrode plates located on opposite ends of the membranes. Ion exchange resin is contained within the product flow channel, acting as an ion selective medium in the electrodeionization process. Isolation valves are provided for each stack component to allow for maintenance of a stack without removing the electrodeionization unit from service.

Mixed Bed Ion-Exhange Units

Mixed bed polishers are provided as the final stage of the effluent treatment. Bypass piping is provided to allow a mixed bed unit to be taken off line for maintenance.

Clean-In-Place Unit

A CIP unit is provided for chemically cleaning the reverse osmosis and EDI units.

Sample Panel Unit

A sample panel unit is provided for monitoring of water parameters.

9.2.3.2.3 System Operation

After receiving water from the raw water system, the water is pumped to the demineralized water treatment system. The demineralized water treatment system is a water purification system consisting of filters, pumps, reverse osmosis units, ultraviolet light treatment units, electrodeionization units, mixed bed demineralizers, and associated piping, valves, and instrumentation.

A pH adjustment chemical is added to adjust the pH of the feedwater. The pH entering the reverse osmosis is maintained within a selected operating range of the reverse osmosis to minimize scaling and corrosion.

A dilute antiscalant, which is chemically compatible with the pH adjustment chemical feed, is metered into the reverse osmosis influent water to increase the solubility of salts (that is, decrease scale formation on the membranes). Antiscalant feed rate is controlled by a signal to the metering pump based on the demineralized water flow. Note that antiscalant chemicals are considered toxic materials for industrial facilities. The impact of toxic materials on the plant main control room habitability is addressed in Section 6.4.

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A chlorine reducing agent is added before the reverse osmosis units to combine with the free chlorine in the water and produce chlorides, which are removed by the reverse osmosis units. Free chlorine must be reduced prior to entering the reverse osmosis membranes.

The pH adjustment chemical, chlorine reducing agent, and antiscalant are injected into the demineralized water treatment process from the turbine island chemical feed system. Refer to Subsection 10.4.11 for a further discussion of the chemical feed system.

The reverse osmosis influent passes through the cartridge filter which removes any particulate carried over from the raw water system and provides mixing for the upstream chemical feed systems.

Primary demineralization is achieved by a reverse osmosis system. The influent to the reverse osmosis unit is pumped from the raw water system through the cartridge filters to the suction of the reverse osmosis feed pump. The feed pump moves the water through the reverse osmosis membranes where ionic impurities are removed. A level signal from the demineralized water storage tank controls the operation of the reverse osmosis feed pumps. The pumps are started when the tank level is low and continue to run until the tank is full and the pumps are stopped.

Each reverse osmosis unit has two stages of pressure vessels; the membranes are contained within the vessels. The product water from the reverse osmosis unit is directed to the ultraviolet unit. The reject flow from the reverse osmosis unit is discharged to the waste water system. The brine flow from the second stage is recycled to the suction of the feed pump of the first unit to improve the fluid recovery rate of the reverse osmosis process. A reverse osmosis unit can be isolated for cleaning and maintenance of the membranes with the reverse osmosis treatment train placed in offline mode.

One reverse osmosis treatment train can be out of service, without affecting the demineralized water treatment effluent water quality. Operation with only one reverse osmosis treatment train results in reduced system capacity.

The product water from the reverse osmosis units flows to the reverse osmosis transfer pump in each train. It is then pumped through ultraviolet light treatment units. The ultraviolet light breaks apart and reduces the total organic carbon molecules within the flow stream. The water then flows to the electrodeionization system for secondary demineralization. Two electrodeionization units operate in parallel, each treating 50 percent of the systems flow. The electrondeionization unit removes ionic impurities and also chemically removes dissolved carbon dioxide gas. The water flows through the electrodeionization stacks where a DC voltage across the electrode plates attracts ions of opposite charge. The membranes allow the ions to penetrate the membrane only in one direction, thereby concentrating the ions in the reject flow channel. The resin serves as an ion selective medium to aid migration of the ions through the membranes. Regeneration of the resin is performed by the DC voltage potential across the stack. The brine flow is blowdown, which is sent to the waste water system. Makeup to the brine flow is provided from the influent to the electrodeionization unit.

The EDI product flow then flows through the mixed bed ion exchangers. The mixed bed units are arranged to accept the full system flow in a series. The mixed beds are piped in such a way that by closing and opening valves, either vessel can be the lead or the lag unit, or either unit can be bypassed for maintenance. The mixed bed is the final stage of demineralization using ion exchange resins.

After these water processes, demineralized water leaves the demineralized water treatment system and is supplied to the demineralized water storage tank. Refer to Subsection 9.2.4 for further discussion of the demineralized water transfer and storage system.

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9.2.3.3 Safety Evaluation

The demineralized water treatment system has no safety-related function and therefore requires no nuclear safety evaluation.

There are no potential sources of radioactive contamination within the demineralized water treatment system. Backflow prevention is addressed in the demineralized water transfer and storage system, Subsection 9.2.4.

The effects of flooding due to demineralized water treatment system component failures are described in Section 3.4.

9.2.3.4 Tests and Inspections

The demineralized water treatment system is functionally tested under anticipated operating conditions prior to initial plant startup. This verifies that system components and controls function properly. Proper system performance and integrity during normal plant operation are verified by system operation and visual inspections.

9.2.3.5 Instrumentation Applications

Pressure and flow instrumentation is provided to monitor the operation of the reverse osmosis process. The reverse osmosis feed pump discharge pressure and the effluent flow from the reverse osmosis units provide indication and control for the primary demineralization process. A pH analyzer, located upstream of the reverse osmosis units, maintains the pH level in the water to the reverse osmosis units by adjusting the stroke of the chemical feed pumps. Flow is measured downstream of the RO units and a permissive signal is sent to the chemical feed pumps. Pressure, conductivity, temperature, and flow is measured at each interval of the water treatment process.

Tank level from the demineralized water storage tank controls the operation of the system feed pumps. This level indication is described in Subsection 9.2.4.

Parameters measured such as tank level indication, pressure differentials across filters, system and pump pressures, system flow, and water conductivity outputs are displayed to the data display and processing system.

9.2.4 Demineralized Water Transfer and Storage System

The demineralized water transfer and storage system receives water from the demineralized water treatment system, and provides a reservoir of demineralized water to supply the condensate storage tank and for distribution throughout the plant. Demineralized water is processed in the demineralized water transfer and storage system to remove dissolved oxygen. In addition to supplying water for makeup of systems which require pure water, the demineralized water is used to sluice spent radioactive resins from the ion exchange vessels in the chemical and volume control system (as described in Subsection 9.3.6), the spent fuel pool cooling system (as described in Subsection 9.1.3), and the liquid radwaste system (as described in Section 11.2) to the solid radwaste system.

The demineralized water treatment system is described in Subsection 9.2.3.

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9.2.4.1 Design Basis

9.2.4.1.1 Safety Design Basis

The demineralized water transfer and storage system serves no safety-related function other than containment isolation, and therefore has no nuclear safety-related design basis except for containment isolation. See Subsection 6.2.3 for the containment isolation system.

9.2.4.1.2 Power Generation Design Basis

- The demineralized water transfer and storage system provides demineralized water through the demineralized water storage tank to fill the condensate storage tank and to meet required demands and usages of demineralized water in other plant systems.
- The demineralized water transfer pumps provide adequate capacity and head for the distribution of demineralized water.
- The demineralized water storage tank supplies a source of demineralized water to the chemical and volume control makeup pumps during startup and required boron dilution evolutions. The demineralized water transfer and storage system supplies the required amount of water to the chemical and volume control system for reactor water makeup.
- The oxygen content of water supplied to the demineralized water distribution system from the demineralized water storage tank is 100 ppb or less.
- Sufficient storage capacity is provided in the condensate storage tank to satisfy condenser makeup demand based on maximum steam generator blowdown operation during a plant startup duration.
- The condensate storage tank provides the water supply for the startup feedwater pumps during startup, hot standby, and shutdown conditions.
- The condensate storage tank provides a sufficient supply of water to the startup feedwater system to permit 8 hours of hot standby operation, followed by an orderly plant cooldown from normal operating temperature to conditions which permit operation of the normal residual heat removal system over a period of approximately 6 hours.
- The piping from the condensate storage tank to the startup feedwater pumps allows adequate net positive suction head (NPSH) at maximum tank water temperature and minimum water level.
- The condensate storage tank serves as a reservoir to supply or receive condensate as required by the condenser hotwell level control system.
- The oxygen content of water stored in the condensate storage tank is 100 ppb or less.

9.2.4.2 System Description

9.2.4.2.1 General Description

Component and equipment classification for the demineralized water transfer and storage system is given in Section 3.2.

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9.2.4.2.2 Component Description

Demineralized Water Storage Tank

The demineralized water storage tank has a capacity of 126,000 gallons. The tank is a vertical cylindrical tank constructed of stainless steel. The tank is provided with level and temperature instrumentation; level controls the operation of the demineralized water treatment system and sends a signal to the reverse osmosis feed pumps to start and stop, thus supplying water to the storage tank. Tank temperature is monitored and controls immersion-type electric heaters to keep the tank contents from freezing.

Demineralized Water Transfer Pump

Two motor-driven, centrifugal, horizontal pumps, located near the demineralized water storage tank, provide the plant demineralized water distribution system pressure and capacity. Each pump provides full flow recirculation through the catalytic oxygen reduction unit as well as providing the required system demand.

Catalytic Oxygen Reduction Units

Oxygen control of the demineralized water is performed by catalytic oxygen reduction units. Two catalytic oxygen reduction units are used in the AP1000 plant. One unit is provided for the demineralized water distribution system as water is pumped from the tank to the distribution system. The second unit is provided at the condensate storage tank to maintain a low oxygen content within the tank and is used in a recirculation path around the tank.

Each catalytic oxygen reduction unit consists of a mixing chamber, a catalytic resin vessel, and a resin trap. The mixing chamber is a stainless steel, in-line, static mixer where dissolution of the reducing agent occurs. Dissolved oxygen is removed chemically by mixing the effluent from the storage tank with hydrogen gas. Hydrogen is supplied from the plant gas system. The resin vessel is a rubber lined, carbon steel vessel containing catalytic resin. The stainless steel resin trap contains a cartridge filter to collect resin fines discharged from the resin vessel.

Condensate Storage Tank

The condensate storage tank has a volume of approximately 656,000 gallons and is a vertical cylindrical tank constructed of stainless steel. Level and temperature instrumentation are provided with the tank level controlled by the makeup valve. Freeze protection is supplied by immersion-type electric heaters.

9.2.4.3 System Operation

9.2.4.3.1 Normal Operation

The water level in the demineralized water storage tank controls the demineralized water treatment system. When the level in the demineralized water storage tank falls to a preset level, the pumps in the demineralized water treatment system start automatically. High water level in the tank stops operation of the demineralized water treatment system. This action, along with the capacitance in the tank, maintains the desired volume to supply the expected demands for demineralized water during normal plant operation.

The demineralized water transfer pumps, taking suction from the demineralized water storage tank, supply water through a catalytic oxygen reduction unit to the demineralized water distribution header. From this header, demineralized water is supplied to the condensate storage tank, is supplied as makeup to the chemical and volume control system pumps, and is distributed throughout the plant. The demineralized water distribution header pressure is maintained by the operation of one transfer

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pump. This pump recirculates water that exceeds system demand to the demineralized water storage tank. Controls are provided to automatically start the second pump upon failure of the first to maintain system pressure and demand. A low level alarm on the demineralized water storage tank signals the plant operator to isolate demands on the tank other than chemical and volume control system supply. Demineralized water is distributed to the containment, auxiliary, radwaste, annex, and turbine buildings for system usage.

The condensate storage tank level is maintained by a level control valve in the tank supply line. The valve opens when the water level in the tank drops to a specified level and closes when the level increases to a specified setpoint. When high oxygen levels exist in the condensate storage tank, an oxygen analyzer signal starts the catalytic oxygen reduction unit pump. The pump is shut off when low levels of oxygen are detected. Low oxygen demineralized water is circulated from the tank outlet connection, through the catalytic oxygen reduction unit, and is returned to the tank via the normal inlet supply line of the tank. An orifice controls the recirculation pressure and flow returning to the tank.

Changes in the condensate system inventory are controlled by the condenser hotwell level system. As level falls in the hotwell, makeup from the condensate storage tank is supplied to the hotwell by the makeup control valve. As level rises in the hotwell, condensate is rejected to the condensate storage tank via the condensate pump's discharge control valve. Subsection 10.4.1 describes the function of the condenser hotwell level system.

In the event the main feedwater system is unavailable to supply water to the steam generators during startup, hot standby, or shutdown, the startup feedwater pumps may be activated and require water from the condensate storage tank. Subsection 10.4.9 describes the startup feedwater system function and operation.

Water supplied from the condensate storage tank to the auxiliary steam supply system is described in Subsection 10.4.10.

9.2.4.4 Safety Evaluation

The demineralized water transfer and storage system has no safety-related function other than for containment isolation (see Figure 9.2.4-1), and therefore requires no nuclear safety evaluation, other than containment isolation which is described in Subsection 6.2.3.

Failure of system components has no impact on safety-related systems, structures, or components. Flooding due to demineralized water transfer and storage system component failures which may affect safe shutdown equipment are described in Section 3.4.

The condensate storage tank normally contains no significant radioactive contaminants.

A check valve or atmospheric gap, in conjunction with a block valve or control valve, is used to prevent backflow of fluids from systems that interface with the demineralized water transfer and storage system. For interfacing systems that have a higher operating pressure than the demineralized water transfer and storage system and that normally do not require a supply of demineralized water during plant operations, a check valve with a normally closed block valve is used. For interfacing systems that have a higher operating pressure than the demineralized water transfer and storage system and that normally require demineralized water during plant operations, a check valve is used to prevent backflow into the demineralized water transfer and storage system. For interfacing systems with a lower operating pressure than the demineralized water transfer and storage system, system operating pressure prevents backflow into the demineralized water transfer and storage system; when the demineralized water transfer and storage system is shut down for

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maintenance, the check valve, closed block or control valve, or atmospheric gap is relied upon to prevent backflow into the demineralized water transfer and storage system.

9.2.4.5 Tests and Inspections

Proper system performance and integrity during normal plant operation are confirmed by system operation and visual inspections.

Grab samples may be taken from the demineralized water storage tank or the condensate storage tank to verify water chemistry is maintained within acceptable limits. Grab samples are taken to the secondary sampling laboratory for analysis. Water chemistry specifications for demineralized water supplied to the demineralized water transfer and storage system are described in Subsection 9.2.3.

9.2.4.6 Instrumentation Applications

Water level is measured and automatically controlled and alarmed in the demineralized water and condensate storage tanks.

Instrumentation is provided to control the recirculation and distribution of demineralized water from the storage tank through the pumps and to the supply header and condensate storage tank. Controls are provided for automatic starting of the demineralized water transfer and storage system pumps.

An oxygen analyzer signal starts and stops the condensate storage tank catalytic oxygen reduction unit pump on low and high oxygen levels.

Monitoring of instrumentation is performed through the data display and processing system. Control functions are performed by the plant control system. Appropriate alarms and displays are available in the control room. Local indication, display and manual control are available in portable displays which may be connected to the data display and processing system. See Chapter 7.

9.2.5 Potable Water System

9.2.5.1 Design Basis

The potable water system (PWS) is designed to furnish water for domestic use and human consumption. It complies with the following standards:

- Bacteriological and chemical quality requirements as referenced in EPA "National Primary Drinking Water Standards," 40 CFR Part 141.
- The distribution of water by the system is in compliance with 29 CFR 1910, Occupational Safety and Health Standards, Part 141.

9.2.5.1.1 Safety Design Basis

Certain valves in the potable water system have the safety-related function of main control room isolation. The potable water system penetrates the main control room envelope boundary. A safety related loop seal in the PWS piping that penetrates the main control room envelope boundary prevents in-leakage into the main control room envelope during VES operation. Testing the effectiveness of the boundary isolation function of the loop seal is covered by the main control room pressure boundary testing described in Subsection 6.4.5.2, and by the inservice testing program for valves described in Subsection 3.9.6.

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9.2.5.1.2 Power Generation Design Basis

- Potable water is supplied to provide a quantity of 50 gallons/person/day for the largest number of persons expected to be at the station during a 24-hour period during normal plant power generation or outages.
- Water heaters provide a storage capacity equal to the probable hourly demand for potable hot water usage and provide hot water for the main lavatory, shower areas, and other locations where needed.
- A minimum pressure of 20 psig is maintained at the furthermost point in the distribution system.
- No interconnections exist between the potable water system and any potentially radioactive system or any system using water for purposes other than domestic water service.
- During main control room isolation, the 1175-gallon main control room potable water storage tank supplies 35.6 gallons of potable water per person for 11 people per day for 3 days.

9.2.5.2 System Description

9.2.5.2.1 General Description

Classification of components and equipment for the potable water system is given in Section 3.2.

9.2.5.2.2 Component Description

Hot Water Heaters

Electric immersion heating elements located inside the potable water hot water tank are used to produce hot water. This hot water is routed to plumbing fixtures and equipment requiring domestic hot water service.

9.2.5.3 System Operation

Filtered water is supplied from a site-specific water source for the potable water distribution system.

The onsite water supply system will maintain an appropriate pressure throughout the distribution system.

Potable water is supplied to areas that have the potential to be contaminated radioactively. Where this potential for contamination exists, the potable water system is protected by a reduced pressure zone type backflow prevention device.

No interconnections exist between the potable water system and any system using water for purposes other than domestic water service including any potentially radioactive system.

9.2.5.4 Safety Evaluation

The potable water system has no safety-related functions other than to prevent in-leakage into the main control room envelope during VES operation. A loop seal in the safety-related PWS piping that penetrates the main control room envelope boundary prevents in-leakage into the main control room envelope.

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9.2.5.5 Tests and Inspections

The potable water system is hydrostatically tested for leak-tightness in accordance with the Uniform Plumbing Code. Inspection of the system is in compliance with the Uniform Plumbing Code or governing codes of the local jurisdiction. The system is then disinfected, flushed with potable water, and placed in service. The presence of residual chlorine can be confirmed through laboratory tests of samples at sampling points as required. Tests for microbiological and bacteria presence in potable water are conducted periodically.

9.2.5.6 Instrumentation Applications

Thermostats, high-temperature limit controls, and temperature indication are installed on the potable water system hot water tank. Pressure regulators are employed in those parts of the distribution system where pressure restrictions are imposed.

9.2.6 Sanitary Drainage System

The sanitary drainage system (SDS) is designed to collect the site sanitary waste for treatment, dilution and discharge.

9.2.6.1 Design Basis

9.2.6.1.1 Safety Design Basis

The sanitary drainage system isolates the SDS vent penetration in the main control room boundary on High-2 particulate or iodine concentrations in the main control room air supply, extended loss of main control room differential pressure, or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4. The SDS vent line that penetrates the main control room envelope is safety related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a design basis accident. An additional penetration from the SDS into the main control room envelope is maintained leak tight using a loop seal in the safety-related seismic Category I piping.

9.2.6.1.2 Power Generation Design Basis

The sanitary drainage system within the scope of the plant covered by Design Certification is designed to accommodate 25 gallons/person/day for up to 500 persons during a 24-hour period.

9.2.6.2 System Description

9.2.6.2.1 General Description

The sanitary drainage system collects sanitary waste from plant restrooms and locker room facilities in the auxiliary building, and annex building, and carries this waste to the treatment plant where it is processed.

The sanitary drainage system does not service facilities in radiologically controlled areas (RCA).

Although this sanitary drainage system transports sanitary waste to the waste treatment plant, the waste treatment plant is site specific and is outside the scope of the standard AP1000 certification. This system description provides a conceptual basis for the site interface design.

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9.2.6.2.2 Component Description

Isolation Valves

The main control room pressure boundary penetration includes isolation valves, interconnecting piping, and vent and test connections. The isolation valves are classified as Safety Class C (see Subsection 3.2.2.5 and Table 3.2-3) and seismic Category I. Testing the effectiveness of their boundary isolation function is covered by the main control room pressure boundary testing described in Subsection 6.4.5.2, and by the inservice testing program for valves described in Subsection 3.9.6.

The main control room pressure boundary isolation valves have motor operators. The valves are designed to fail as is in the event of loss of electrical power. The valves are qualified to shut tight against control room pressure.

Trunk Line

The trunk line is the primary line that the sanitary drainage system piping connects into for transport of the sanitary drainage to the site treatment plant.

Branch Lines

Branch lines are the sanitary drainage lines that connect the restroom facilities to the trunk line.

Manholes

Manholes are required in the trunk line at the connection of the branch lines into the trunk line, at the change in direction of the trunk line, or at the change in slope or direction of the trunk line. Quantity and location are site specific.

Lift Stations

Lift stations are required in the trunk line when the uniform slope of the trunk line results in excessively deep and costly excavation. Quantity and location are site specific.

9.2.6.3 Safety Evaluation

The sanitary drainage system has no safety-related function other than main control room envelope isolation. Redundant safety-related isolation valves are provided in the vent line penetrating the main control room. Therefore, there are no single active failures that would prevent isolation of the main control room envelope.

There are no interconnections between this system and systems having the potential for containing radioactive material. Potentially radioactive drains are addressed in Subsection 9.3.5 dealing with the radioactive waste drain system.

9.2.6.4 Test and Inspection

The sanitary drainage system is tested by water or air and established to be watertight in accordance with the Uniform Plumbing Code Section 712 (Reference 2). System inspection is performed in compliance with the Uniform Plumbing Code or governing codes of the local jurisdiction.

9.2.6.5 Instrument Application

The instruments associated with this system are part of the waste treatment plant which is site specific. Sufficient instrumentation for operation is provided with the treatment plant.

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9.2.7 Central Chilled Water System

The plant heating, ventilation, and air conditioning (HVAC) systems require chilled water as a cooling medium to satisfy the ambient air temperature requirements for the plant. The central chilled water system (VWS) supplies chilled water to the HVAC systems and is functional during reactor full-power and shutdown operation.

9.2.7.1 Design Basis

9.2.7.1.1 Safety Design Basis

The central chilled water system serves no safety-related function other than containment isolation, and therefore has no nuclear safety design basis except for containment isolation. See Subsection 6.2.3 for the containment isolation system.

9.2.7.1.2 Power Generation Design Basis

The central chilled water system provides chilled water to the cooling coils of the supply air handling units and unit coolers of the plant HVAC systems. It also supplies chilled water to the liquid radwaste system, gaseous radwaste system, and the secondary sampling system.

9.2.7.1.3 Codes and Standards

The central chilled water system is designed to the applicable codes and standards listed in Section 3.2.

9.2.7.2 System Description

9.2.7.2.1 General Description

The system consists of two closed loop subsystems: a high cooling capacity subsystem and a low cooling capacity subsystem. The high capacity subsystem is the primary system used to provide chilled water to the majority of plant HVAC systems and other plant equipment requiring chilled water cooling. The low capacity subsystem is dedicated to the nuclear island nonradioactive ventilation system and the makeup pump and normal residual heat removal pump compartment unit coolers. The low capacity subsystem is illustrated in Figure 9.2.7-1.

The high capacity subsystem consists of two 85-percent capacity chilled water pumps, two 15-percent capacity chilled water pumps, two 85-percent capacity water-cooled chillers, two 15-percent air-cooled chillers, a chemical feed tank, an expansion tank, and associated valves, piping, and instrumentation. The subsystem is arranged in two parallel mechanical trains with common supply and return headers. Each train includes one 15-percent capacity pump, one 85-percent capacity pump, one 15-percent capacity chiller, and one 85-percent capacity chiller. A cross-connection at the discharge of each pump allows for each to feed a given chiller of matching capacity.

The low capacity subsystem consists of two 100-percent capacity chilled water loops. Each loop consists of a chilled water pump, an air-cooled chiller, an expansion tank, a chemical feed tank, and associated valves, piping, and instrumentation. The subsystem is arranged in two independent trains with separate supply and return headers. The subsystem provides a reliable source of chilled water to the main control room (MCR) and control support area (CSA) HVAC subsystem, and the Class 1E electrical equipment room HVAC subsystem. This system configuration provides 100-percent redundancy during normal plant operation and following the loss of normal ac power and offsite power. The air-cooled chillers of the low capacity subsystem are located on the auxiliary building roof.

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9.2.7.2.2 Component Description

The general descriptions and summaries of the design requirements for the central chilled water system components are provided below. The piping inside and outside containment has a design pressure of 200 psig and a design temperature of 320°F. The key equipment parameters for the central chilled water system components are contained in Table 9.2.7-1.

Pumps

Six central chilled water system pumps are provided. These pumps are single-stage, horizontal, centrifugal pumps. These pumps have an integral pump motor shaft driven by an ac-powered induction motor. The central chilled water system pumps are constructed of ductile iron or carbon steel and have flanged suction and discharge nozzles. Each pump is sized to provide the maximum water flow required by its respective chiller unit for removal of its associated design heat load.

Two pumps associated with the low capacity subsystem are risk-significant and are included with the scope of D-RAP. See Table 17.4-1 for further information.

Water-Cooled Chillers

Two water cooled liquid chillers are provided. Each chiller unit consists of a compressor, condenser, evaporator, and associated piping and controls. Environmentally safe refrigerants will be used in these chillers.

Air-Cooled Chillers

Four air-cooled liquid chillers are provided. Each chiller unit consists of a compressor, condenser, evaporator, and associated piping and controls. Environmentally safe refrigerants will be used in these chillers.

Two air-cooled chillers associated with the low capacity subsystem are risk-significant and are included with the scope of D-RAP. See Table 17.4-1 for further information.

Expansion Tank

One open and two closed expansion tanks are provided to maintain the pressure above saturation. The high capacity subsystem uses an open expansion tank which is located sufficiently above the high point of the system and connected to the suction side of the pump. The low capacity subsystem uses nitrogen charged expansion tanks on the suction side of the chilled water pumps. The expansion tanks maintain a positive suction pressure for the pumps. The tanks are sized to accommodate the volume of water expansion providing a space into which the noncompressible liquid can expand or contract as the liquid undergoes volumetric changes with changes in temperature.

Chemical Feed Tank

The chemical feed tanks and the associated piping are used to add chemicals to each chilled water subsystem stream to maintain proper water quality.

Valves

Isolation valves are provided upstream and downstream of each pump/chiller train. These valves are butterfly valves and are used to isolate a train of the subsystem for maintenance. An interlock is provided between the downstream isolation valve and the low-capacity pump/chiller controls.

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An isolation valve is provided in the line that cross-connects the pump discharge lines in the high capacity subsystem. This manual butterfly valve is normally closed and can be manually aligned to operate the standby chiller with the operating pump of either train.

An air-operated isolation valve and check valve are provided in the chilled water supply and two air-operated isolation valves are provided in the chilled water return line that penetrates containment. The air-operated containment isolation valves automatically close upon receipt of a containment isolation signal. This isolation signal can be bypassed by the MCR operator to be able to restore containment recirculation system cooling with the containment isolated.

Isolation valves are provided at the interconnection with the hot water heating system to provide hot water through the coils of the containment recirculation cooling system for heating during refueling, maintenance, and testing activities under cold weather conditions.

High capacity subsystem temperature control valves are located upstream of each cooling coil or group of coils, except for the containment recirculation cooling system coils. The containment recirculation cooling system coils are provided with temperature controlled modulating valves. These valves control chilled water flow to the containment recirculation cooling system coils, as needed, to maintain the temperature within the design conditions. The flow control valves fail open upon loss of control air or electrical power.

Low capacity subsystem three-way modulating temperature control valves are provided for each group of nuclear island nonradioactive ventilation system cooling coils. These valves bypass chilled water flow around the coils, as needed, to maintain the temperature within the design conditions.

9.2.7.2.3 Instrumentation Requirements

The chiller and pumps are operable from the plant control system. The following describes the instrumentation employed for monitoring the operation of the central chilled water system components.

- Compressor trip and malfunction alarm
- Pump trip alarm
- Flow indication and low-flow alarm
- Temperature indication and high-temperature alarm
- System low/high pressure alarm

A low pressure interlock is provided on the pump suction and a low-low flow interlock is provided on the pump discharge to protect the pumps. Level instrumentation measures expansion tank level and provides signals to low- and high-level alarms to the plant control system and to open and close the makeup supply valve.

9.2.7.2.4 System Operation

The central chilled water system operating modes are described below.

Normal Operation

The high capacity subsystem capacity is based on the maximum and minimum normal ambient design temperatures as defined in Chapter 2, Table 2-1. The high capacity subsystem operates during normal modes of plant operation, supplying chilled water to plant components at a normal temperature of 40°F. The capacity of the low capacity subsystem is based on the maximum safety ambient design temperatures as defined in Chapter 2, Table 2-1. The low capacity subsystem is designed to operate during all normal modes of operation, supplying chilled water to the nonradioactive ventilation system components at a normal temperature of 40°F. The low capacity

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system also supplies chilled water to the make-up pump and normal residual heat removal pump compartment unit coolers of the radiologically controlled area ventilation system. The low-capacity air-cooled chillers are provided with freeze protection, such as motor heaters and louvers. Heat tracing has been provided on the chiller supply and return pipe lines located on the auxiliary building roof, which are susceptible to freezing.

During normal operation of the high capacity subsystem, at least one pump and at least one chiller operate to supply chilled water to the following plant HVAC systems:

- Radiologically controlled area ventilation system (Subsection 9.4.3)
- Containment recirculation cooling system (Subsection 9.4.6)
- Containment air filtration system (Subsection 9.4.7)
- Health physics/control access area HVAC system (Subsection 9.4.11)
- Radwaste building ventilation system (Subsection 9.4.8)
- Annex/auxiliary building nonradioactive ventilation system (Subsection 9.4.2)
- In addition, they also supply chilled water to the liquid radwaste system (Section 11.2), the gaseous radwaste system (Section 11.3), the portable and mobile radwaste system (Section 11.4) components, the secondary sampling system (Subsection 9.3.4) components, the electrical switchgear room, the south bay equipment, the control cabinet rooms, and the personnel work area air handling units of the turbine building ventilation system (Subsection 9.4.9).

In the event that either the pump or chiller of the operating train becomes inoperable, the standby train would be aligned to provide chilled water service.

During normal operation of the low capacity subsystem, one pump and one chiller operate to supply chilled water to the associated cooling coils of the nuclear island nonradioactive ventilation system and the makeup pump and normal residual heat removal pump compartment unit coolers of the radiologically controlled area ventilation system. One train provides chilled water to the A and D air handling unit of the Class 1E electrical equipment room HVAC subsystem, the A air handling unit of the main control room/control support area HVAC subsystem, and the A makeup pump and the A and B normal residual heat removal pump compartment unit coolers of the radiologically controlled area ventilation system. The other train provides chilled water to the B and C air handling unit of the Class 1E electrical equipment room HVAC subsystem, the B air handling unit of the main control room/control support area HVAC subsystem, the B makeup pump and the A and B normal residual heat removal pump compartment unit coolers of the radiologically controlled area ventilation system. In the event that one train of the low capacity subsystem is inoperable, the operator can align the standby train to provide cooling to the standby nuclear island nonradioactive ventilation system air handling units and the makeup pump and the normal residual heat removal pump compartment unit coolers of the radiologically controlled area ventilation system.

During plant shutdown in cold weather conditions, the supply and return piping to the containment recirculation cooling system cooling coils may be isolated to permit manual alignment of the hot water heating system to the containment.

The central chilled water system is designed to permit use of the chilled water piping inside containment to the containment recirculation air handling units for containment heating when the plant is shutdown during cold weather. Remote manual realignment to the heating mode, utilizing the

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hot water system and the same containment recirculation air handling unit coils as the cooling mode, is performed outside containment and the procedure is administratively controlled. During this mode of operation, the high capacity subsystem is functional to meet the demand of those remaining HVAC systems and other equipment requiring chilled water.

Abnormal Operation

The high cooling capacity subsystem piping penetrates the containment to supply chilled water to the containment recirculation system fan coil units. The containment isolation valves, located on the chilled water supply and return lines, close on receipt of containment isolation signals. A bypass mode with main control room indication is provided to restore the containment recirculation cooling system cooling during containment isolation. The remainder of the chilled water system continues to operate normally following containment isolation provided that power is available.

The central chilled water system is designed to remain operable following a loss of normal ac power and offsite power by using standby onsite ac power.

The low capacity subsystem chillers, pumps, and other electrical components are connected to the plant standby diesel generator bus in accordance with the automatic electrical load sequencing. The low capacity subsystem is configured such that the operation is similar to that described above for normal operation. Following the loss of normal ac power and offsite power, one diesel generator and one train of the low capacity subsystem operate to supply chilled water to the associated cooling coils of the nuclear island nonradioactive ventilation system and the makeup pump and normal residual heat removal pump compartment unit coolers as shown in Table 9.2.7-1.

The high capacity subsystem chillers, pumps, and other electrical components are connected to the plant standby diesel generator bus in accordance with the optional electrical load sequencing and can be energized at the option of the operator for investment protection after evaluation of the diesel generator available capacity.

The high capacity subsystem can be used in conjunction with the containment recirculation cooling system to remove heat from the containment atmosphere following certain plant transients, if the systems are available.

9.2.7.3 Safety Evaluation

The central chilled water system has no safety-related function, other than containment isolation and therefore requires no nuclear safety evaluation, other than containment isolation which is described in Subsection 6.2.3.

The central chilled water system components located in safety-related areas of the plant are designed such that a failure in the system will not unacceptably impact the operation of safety-related components.

9.2.7.4 Inservice Inspection/Inservice Testing

The central chilled water piping circuits are hydrostatically tested and balanced to provide design flowrates and temperatures. Periodic inspections are performed to verify proper performance of system components. Specific test requirements and intervals are contained in the plant operating procedures.

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9.2.8 Turbine Building Closed Cooling Water System

The turbine building closed cooling water system (TCS) provides chemically treated, demineralized cooling water for the removal of heat from nonsafety-related heat exchangers in the turbine building and rejects the heat to the [[circulating water system.]]

9.2.8.1 Design Basis

9.2.8.1.1 Safety Design Basis

The turbine building closed cooling water system has no safety-related function and therefore has no nuclear safety design basis.

9.2.8.1.2 Power Generation Design Basis

The turbine building closed cooling water system provides corrosion-inhibited, demineralized cooling water to the equipment shown in Table 9.2.8-1 during normal plant operation.

During power operation, the turbine building closed cooling water system provides a continuous supply of cooling water to turbine building equipment at a temperature of 105°F or less assuming a [[circulating water]] temperature of 100°F or less.

The cooling water is treated with a corrosion inhibitor and uses demineralized water for makeup. The system is equipped with a chemical addition tank to add chemicals to the system.

The heat sink for the turbine building closed cooling water system is the [[circulating water system.]] The heat is transferred to [[circulating water]] through plate type heat exchangers which are components of the turbine building closed cooling water system.

A surge tank is sized to accommodate thermal expansion and contraction of the fluid due to temperature changes in the system.

One of the turbine building closed cooling system pumps or heat exchangers may be unavailable for operation or isolated for maintenance without impairing the function of the system.

The turbine building closed cooling water pumps are provided ac power from the 6900V switchgear bus. The pumps are not required during a loss of normal ac power.

9.2.8.2 System Description

9.2.8.2.1 General Description

Classification of equipment and components is given in Section 3.2. The system consists of two 100-percent capacity pumps, three 50-percent capacity heat exchangers (connected in parallel), one surge tank, one chemical addition tank, and associated piping, valves, controls, and instrumentation. Heat is removed from the turbine building closed cooling water system by the [[circulating water system]] via the heat exchangers.

The pumps take suction from a single return header. Either of the two pumps can operate in conjunction with any two of the three heat exchangers. Discharge flows from the heat exchangers combine into a single supply header. Branch lines then distribute the cooling water to the various coolers in the turbine building. The flow rates to the individual coolers are controlled either by flow restricting orifices or by control valves, according to the requirements of the cooled systems. Individual coolers can be locally isolated, where required, to permit maintenance of the cooler while

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supplying the remaining components with cooling water. A bypass line with a manual valve is provided around the turbine building closed cooling water system heat exchangers to help avoid overcooling of components during startup/low-load conditions or cold weather operation.

The system is kept full of demineralized water by a surge tank which is located at the highest point in the system. The surge tank connects to the system return header upstream of the pumps. The surge tank accommodates thermal expansion and contraction of cooling water resulting from temperature changes in the system. It also accommodates minor leakage into or out of the system. Water makeup to the surge tank, for initial system filling or to accommodate leakage from the system, is provided by the demineralized water transfer and storage system. The surge tank is vented to the atmosphere.

A line from the pump discharge header back to the pump suction header contains valves and a chemical addition tank to facilitate mixing chemicals into the closed loop system to inhibit corrosion in piping and components.

A turbine building closed cooling water sample is periodically taken and analyzed to verify that water quality is maintained.

9.2.8.2.2 Component Description

Surge Tank

A surge tank accommodates changes in the cooling water volume due to changes in operating temperature. The tank also temporarily accommodates leakage into or out of the system. The tank is constructed of carbon steel.

Chemical Addition Tank

The chemical addition tank is constructed of carbon steel. The tank is normally isolated from the system and is provided with a hinged closure for addition of chemicals.

Pumps

Two pumps are provided. Either pump provides the pumping capacity for circulation of cooling water throughout the system. The pumps are single stage, horizontal, centrifugal pumps, are constructed of carbon steel, and have flanged suction and discharge nozzles. Each pump is driven by an ac powered induction motor.

Heat Exchangers

Three heat exchangers are arranged in a parallel configuration. Two of the heat exchangers are in use during normal power operation and turbine building closed cooling water flow divides between them.

The heat exchangers are plate type heat exchangers. Turbine building closed cooling water circulates through one side of the heat exchanger while [[circulating water]] flows through the other side. During system operation, the turbine building closed cooling water in the heat exchanger is maintained at a higher pressure than the [[circulating water]] so leakage of [[circulating water]] into the closed cooling water system does not occur. The heat exchangers are constructed of titanium or stainless steel plates with a carbon steel frame.

Valves

Manual isolation valves are provided upstream and downstream of each pump. The pump isolation valves are normally open but may be closed to isolate the non-operating pump and allow maintenance during system operation. Manual isolation valves are provided upstream and downstream of each turbine building closed cooling water heat exchanger. One heat exchanger is

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isolated from system flow during normal power operation. A manual bypass valve can be opened to bypass flow around the turbine building closed cooling water heat exchangers when necessary to avoid low cooling water supply temperatures.

Flow control valves are provided to restrict or shut off cooling water flow to those cooled components whose function could be impaired by overcooling. The flow control valves are air operated and fail open upon loss of control air or electrical power. An air operated valve is provided to control demineralized makeup water to the surge tank for system filling and for accommodating leakage from the system. The makeup valve fails closed upon loss of control air or electrical power.

A TCS heat exchanger can be taken out of service by closing the inlet isolation valve. Water chemistry in the isolated heat exchanger train is maintained by a continuous flow of circulating water through a small bypass valve around the inlet isolation valve.

Backwashable strainers are provided upstream of each TCS heat exchanger. They are actuated by a timer and have a backup starting sequence initiated by a high differential pressure across each individual strainer. The backwash can be manually activated.

Piping

System piping is made of carbon steel. Piping joints and connections are welded, except where flanged connections are used for accessibility and maintenance of components. Nonmetallic piping also may be used.

9.2.8.2.3 System Operation

The turbine building closed cooling water system operates during normal power operation. The system does not operate with a loss of normal ac power.

Startup

The turbine building closed cooling water system is placed in operation during the plant startup sequence [[after the circulating water system is in operation but]] prior to the operation of systems that require turbine building closed cooling water flow. The system is filled by the demineralized water transfer and storage system through a fill line to the surge tank. The system is placed in operation by starting one of the pumps.

Normal Operation

During normal operation, one turbine building closed cooling water system pump and two heat exchangers provide cooling to the components listed in Table 9.2.8-1. The other pump is on standby and aligned to start automatically upon low discharge header pressure.

During normal operation, leakage from the system will be replaced by makeup from the demineralized water transfer and storage system through the automatic makeup valve. Makeup can be controlled either manually or automatically upon reaching low level in the surge tank.

Shutdown

The system is taken out of service during plant shutdown when no longer needed by the components being cooled. The standby pump is taken out of automatic control, and the operating pump is stopped.

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9.2.8.3 Safety Evaluation

The turbine building closed cooling water system has no safety-related function and therefore requires no nuclear safety evaluation.

9.2.8.4 Tests and Inspections

Pre-operational testing is described in Chapter 14. The performance, structural, and leaktight integrity of system components is demonstrated by operation of the system.

9.2.8.5 Instrument Applications

Parameters important to system operation are monitored in the main control room. Flow indication is provided for individual cooled components as well as for the total system flow.

Temperature indication is provided for locations upstream and downstream of the turbine building closed cooling water system heat exchangers. High temperature of the cooling water supply alarms in the main control room. Temperature test points are provided at locations to facilitate thermal performance testing.

Pressure indication is provided for the pump suction and discharge headers. Low pressure at the discharge header automatically starts the standby pump.

Level instrumentation on the surge tank provides level indication and both low- and high-level alarms in the main control room. On low tank level, a valve in the makeup water line automatically actuates to provide makeup flow from the demineralized water transfer and storage system.

9.2.9 Waste Water System

The waste water system collects and processes equipment and floor drains from nonradioactive building areas.

9.2.9.1 Design Basis

9.2.9.1.1 Safety Design Basis

The waste water system isolates the WWS drain line that penetrates the main control room boundary. The WWS drain lines that penetrate the main control room envelope are safety related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a design basis accident.

9.2.9.1.2 Power Generation Design Basis

The power generation design basis is:

- Remove oil and/or suspended solids from miscellaneous waste streams generated from the plant.
- Collect system flushing wastes during startup prior to treatment and discharge.
- Collect and process fluid drained from equipment or systems during maintenance or inspection activities.

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- Direct nonradioactive equipment and floor drains which may contain oily waste to the building sumps and transfer their contents for proper waste disposal. The radioactive equipment and floor drain system is described in Subsection 9.3.5.

9.2.9.2 System Description

9.2.9.2.1 General Description

The waste water system is capable of handling the anticipated flow of waste water during normal plant operation and during plant outages. The classification of components and equipment is given in Section 3.2.

Wastes from the turbine building floor and equipment drains (which include laboratory and sampling sink drains, oil storage room drains, the main steam isolation valve compartment, auxiliary building penetration area and the auxiliary building HVAC room) are collected in the turbine building sumps. Drainage from the auxiliary building sump – north (a nonradioactive sump), the annex building sump, and condensate polishing system (CPS) rinse effluent is also collected in the turbine building sumps. The main turbine building sumps provide a temporary storage capacity and a controlled source of fluid flow to the oil separator. In the event radioactivity is present in the main turbine building sumps, the waste water is diverted from the sumps to the liquid radwaste system (WLS) for processing and disposal. A radiation monitor located on the common discharge piping of the main sump pumps provides an alarm upon detection of radioactivity in the waste water. The radiation monitor also trips the main sump pumps on detection of radioactivity to isolate the contaminated waste water. Provisions are included for sampling the main sumps.

The main turbine building sump pumps route the waste water from the sumps to the oil separator for removal of oily waste. The diesel fuel oil area sump pump and the transformers area waste water collection sump pump also discharge waste water to the oil separator. A bypass line allows for the oil separator to be out of service for maintenance. The oil separator provides temporary storage prior to removal by truck for offsite disposal.

The waste water from the oil separator flows by gravity to an adjacent in-ground sump, then is pumped to the waste water retention basin for settling of suspended solids and treatment, if required, prior to discharge.

Design and routing of the condenser waterbox drains will be incorporated in the site-specific Circulating Water System (CWS) design.

9.2.9.2.2 Component Description

Isolation Valve

The main control room pressure boundary penetration includes a normally closed isolation valve and interconnecting piping. The isolation valve is classified as Safety Class C (see Subsection 3.2.2.5 and Table 3.2-3) and seismic Category I. Testing the effectiveness of the boundary isolation function is covered by the main control room pressure boundary testing described in Subsection 6.4.5.2.

Turbine Building Sumps

The sumps collect waste water from the floor and equipment drains, SWS and CWS strainer backwash flows, alternate SWS cooling tower blowdown flow, laboratory drains, sampling waste drains, plant washdowns from the turbine building, and CPS rinse effluent. Selected drains from both the annex and auxiliary buildings are also collected in these sumps.

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Turbine Building Sump Pumps

The sumps have electric submersible pumps. The main sump pumps route he waste water to the oil separator. Motor control centers in the Turbine Building provide power to the electric sump pump motors. Operation of the pumps is automatic based on sump level with controls provided for manual operation.

Oil Separator

The oil separator removes oily waste from the waste water stream. A sampling connection is provided at the discharge of the oil separator. The waste water flows by gravity to an adjacent sump served by two submersible pumps, which transfer the waste water to the waste water retention basins. A truck connection on the oil separator allows for removal of the waste oil from the oil separator for offsite disposal.

Waste Oil Storage Tank

Waste oil from the oil separator reservoir and other plant areas is stored in a waste oil storage tank. A sampling connection is provided on the tank to verify that the oil does not require handling and disposal as a hazardous material. A truck connection on the tank allows for removal of the waste oil from the tank for offsite disposal.

Waste Water Retention Basin

Waste water retention basins and associated basin transfer pumps and piping are site-specific components as addressed in subsection 9.2.11.

Waste Water Sumps

Waste water collection sumps are provided for the auxiliary building, the diesel generator building, the annex building, the transformers area, and the diesel fuel oil area. These collection sumps are drained by air operated pumps. The effluent from the auxiliary building and the annex building sumps is directed to the oil separator via the turbine building sumps for processing and release. The effluent from the diesel generator building, the diesel fuel oil area, and the transformers area is pumped directly to the oil separator.

Sump Pumps

The waste water sump pumps are pneumatic, double diaphragm pumps. The plant service air system provides the supply of air for operation of these pumps. Operation of the pumps is automatic based on sump level with controls provided for manual operation.

9.2.9.3 Safety Evaluation

The waste water system has no safety-related function other than main control room envelope isolation. A normally closed safety-related isolation valve is provided in the drain line penetrating the main control room. The drain line is safety related up to the isolation valve to ensure that the main control room habitability pressure boundary is maintained.

9.2.9.4 Tests and Inspections

System performance and integrity during normal plant operation are verified by system operation and visual inspections.

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9.2.9.5 Instrumentation Applications

Level instrumentation and associated pump controls on the turbine building sumps, the auxiliary building sump, the diesel generator building sumps, and the diesel fuel oil sump are provided to prevent overflow of these waste water collection points. High alarms indicate tank level where operator action is required.

A radiation monitor located on the turbine building sump common discharge piping initiates an alarm and trips the turbine building sump pumps when radioactivity above a preset high level point is detected in the waste stream.

9.2.10 Hot Water Heating System

The hot water heating system (VYS) supplies heated water to selected nonsafety-related air handling units and unit heaters in the plant during cold weather operation and to the containment recirculating fans coil units during cold weather plant outages.

9.2.10.1 Design Basis

9.2.10.1.1 Safety Design Basis

The hot water heating system serves no safety-related function and therefore has no nuclear safety design basis.

9.2.10.1.2 Power Generation Design Basis

- During normal plant operation, the hot water heating system maintains acceptable design ambient air temperatures in various areas throughout the AP1000.
- During plant outages in cold weather, the hot water heating system supplies hot water to the plant chilled water piping serving the containment building recirculation fan coil units to maintain acceptable ambient air temperatures inside containment.

9.2.10.2 System Description

9.2.10.2.1 General Description

Major components of the heating system include heat exchangers, pumps, a surge tank, and provisions for chemical feed. Component and equipment classification for the hot water heating system is given in Section 3.2. The hot water heating system consists of a heat transfer package for the production of hot water and a distribution system to the various HVAC systems and unit heaters. The hot water heating system is a nonsafety-related system.

During cold weather plant operation, the hot water heating system supplies hot water throughout the plant to protect equipment from freezing and for personnel comfort. During cold weather plant outages, the hot water heating system supplies hot water to the containment building recirculation fan coil units to maintain acceptable ambient air temperatures inside containment. During a loss of normal ac power and offsite power, provisions are made to power the hot water heating system from the onsite diesel generators as an investment protection load. In this mode of operation, heating steam is supplied from the auxiliary steam supply system using a temporary boiler.

The hot water heating system, using main steam or the auxiliary boiler, extracts heat energy from the steam through a heat exchanger and transfers this energy to heat water. The heated water is pumped in a closed loop system to hot water coils in the air conditioning systems. Condensate from

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the heat exchanger is level controlled and drained to the main condenser or auxiliary boiler feedwater system.

One of two system pumps supplies hot water to the heating system by taking suction from the return main and pumping water through one of two heat exchangers and into the main heating hot water system supply header. To prevent flashing of the heated water into steam, the pump in combination with the system surge tank keeps the system pressure above saturation conditions. The surge tank uses elevation to keep the minimum system pressure above saturation conditions at the pump suction. Demineralized water is supplied to the system for surge tank makeup.

During plant outages in cold weather, hot water flows to the containment building recirculation fan coil units to heat the containment atmosphere. The recirculation fan coil units, containment supply and return piping to these units, and the containment isolation valves are part of the central chilled water system as described in Subsection 9.2.7. During normal plant operation the hot water heating system is isolated from the containment recirculation fan coils.

The hot water heating system is a manually actuated system and may operate when the site ambient temperature is 73°F or below.

9.2.10.2.2 Component Description

Major component design data of the hot water heating system are listed in Table 9.2.10-1.

Heat Exchanger

Each heat exchanger is a horizontal, shell-and-tube type, with an integral drain cooler, and uses the heat of vaporization of low-pressure steam for the heating of water. The heat exchanger is located in the closed loop hot water heating system downstream of the system pumps in the turbine building. This heat exchanger provides heated water for selected air handling unit and unit heater hot water coils.

Pumps

One of two system pumps distributes hot water to the various HVAC and unit heater systems. They are motor driven centrifugal pumps.

Surge Tank

The surge tank maintains system pressure by allowing the water to expand when the water temperature increases and provides a volume to accept makeup water to the hot water heating system.

The tank is a carbon steel, welded, pressure vessel with tank recirculation and instrument connections.

Chemical Feed Tank

The chemical feed tank provides a means of chemical mixing in the system. Addition of chemicals provides control of corrosion.

The tank is a vertical cylinder of carbon steel construction with a capacity of less than 150 gallons and a top hinged opening for introducing the chemicals and side connections for transporting water through the chemical mixing tank from the pump discharge or the demineralized water transfer and storage system supply.

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9.2.10.2.3 System Operation

As the system is filled with demineralized water, samples are taken and the closed loop water chemistry adjusted with chemicals recirculated through the chemical mixing tank with the use of a single pump. A pump is started and steam is admitted to a hot water system heat exchanger and the system is gradually heated.

A condensate level is maintained in each system heat exchanger by throttling the heat exchanger discharge flow to the condenser. During a plant outage when main steam is shutdown and auxiliary steam is used from the auxiliary boiler, a manual block valve is opened to establish flow of condensate from each heat exchanger to the auxiliary steam supply system deaerator.

Hot water flowing to individual heating coils is controlled by flow balancing fixed orifices, balancing valves, modulating valves, or by temperature controlled solenoid valves, according to the requirements of the heating system. Area temperatures are controlled by cycling the fans in unit heaters, by use of integral face/bypass dampers in air handling units, modulating hot water flow to air handling units, or by thermostats controlling hot water solenoids in heating coils of HVAC ducts. Further detail of hot water heating of the individual unit heaters, air handling units, and duct heating coils is provided in Section 9.4. In the radwaste building, normally isolated hot water supply and return connections are provided for a mobile radwaste system.

9.2.10.3 Safety Evaluation

The hot water heating system has no safety-related function and therefore requires no nuclear safety evaluation.

The hot water heating system interfaces with only nonsafety-related systems. Hot water heating is used in the containment to keep piping and components from freezing during cold weather when the plant is not operating. A hot water heating system interface with the central chilled water system is outside containment and in nonsafety-related piping of the chilled water system. Piping is shared inside the containment between hot water heating and central chilled water. During normal plant operation, the hot water system is isolated from the central chilled water system and containment. Containment isolation by the central chilled water system is described in Subsection 6.2.3.

The effects of flooding on the safe shutdown capability of the plant are described in Section 3.4.

The temperature control range for areas serviced by the hot water heating system is described in Section 9.4 with the ventilation systems.

9.2.10.4 Tests and Inspections

The hot water heating piping circuits are hydrostatically tested and balanced to provide designed flowrates and temperatures. Active component performance is monitored by instrumentation on the system. System performance and integrity during normal plant operation are verified by system operation and visual inspections.

9.2.10.5 Instrument Applications

Instruments are provided for monitoring system parameters. Essential system parameters are monitored in the main control room.

Total system flow is monitored and displayed in the main control room. The system heat exchangers are level controlled with the instrument signals controlling the level control valves as well as sending level indication and low- and high-level alarms to the data system. Temperature measured

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downstream of the heat exchangers controls steam flow to, the heat exchangers and indicates the temperature of heated water being sent to the hot water heating coils. Also temperature is monitored in the system return main.

Pressure is measured in the pump suction and at the pump discharge. At low-2 pump suction pressure, a signal is sent to stop the hot water heating system pumps.

Level instrumentation on the surge tank provides both high- and low-level alarms. At tank low-level, makeup is provided from the demineralized transfer and storage system. At a high level in the tank, makeup from the demineralized transfer and storage system is discontinued.

9.2.11 Not Used

9.2.12 Combined License Information

9.2.12.1 Potable Water

The Combined License applicant will address the components of the potable water system outside of the power block, including supply source required to meet design pressure and capacity requirements, specific chemical selected for use as a biocide, and any storage requirements deemed necessary. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room habitability is addressed in Section 6.4.

9.2.12.2 Waste Water Retention Basins

The Combined License applicant will address the final design and configuration of the plant waste water retention basins and associated discharge piping, including piping design pressure, basin transfer pump size, basin size, and location of the retention basins.

9.2.13 References

1. ASME Code, Section IV, Pt. HWL, 1998.

2. Uniform Plumbing Code, Section 712, 2000.

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Table 9.2.1-1
Nominal Service Water Flows and Heat Loads
at Different Operating Modes

	CCS Pumps and Heat Exchangers	SWS Pumps and Cooling Tower Cells (Number Normally in Service)	Flow (gpm)	Heat Transferred (Btu/hr)
Normal Operation (Full Load)	1	1	10,500	96.3×10^6
Cooldown	2	2	21,000	339×10^6 (169.5×10^6 per cell)
Refueling (Full Core Offload)	2	2	21,000	61.1×10^6
Plant Startup	2	2	21,000	91.5×10^6
Minimum to Support Shutdown Cooling and Spent Fuel Cooling	1	1	10,000	170×10^6

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Table 9.2.2-1
Nominal Component Data - Component Cooling Water System

CCS Pumps (all data is per pump)	
Quantity	2
Type	Horizontal centrifugal
Minimum capacity (gpm, each) to support shutdown cooling and spent fuel pool cooling	8300
Design capacity (gpm, each)	10,030
Design total differential head (ft)	250
Minimum flow rate to support shutdown cooling (gpm)	2685
Minimum flow rate to support spent fuel cooling (gpm)	1200
CCS Heat Exchangers (all data is per exchanger)	
Quantity	2
Type	Plate
Design duty end of cooldown (MBtu/hr)	44.1
Minimum UA (MBtu/hr/°F) to support shutdown cooling and spent fuel pool cooling	14.0
Design UA (MBtu/hr/°F)	15.5
CCS side Design flow rate (gpm)	9691
Service water side Design flow rate (gpm)	10,500
Plate material	Austenitic stainless steel
Seismic design	Non-seismic

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Table 9.2.2-2
Plant Components Cooled by Component Cooling Water System

Component	System
RCP 1A	RCS
RCP 1B	RCS
RCP 2A	RCS
RCP 2B	RCS
RCP 1A Variable Frequency Drive	RCS
RCP 1B Variable Frequency Drive	RCS
RCP 2A Variable Frequency Drive	RCS
RCP 2B Variable Frequency Drive	RCS
Letdown HX	CVS
RCDT HX	WLS
RHR HX	RNS
RHR HX	RNS
RHR Pump A	RNS
RHR Pump B	RNS
SFP HX A	SFS
SFP HX B	SFS
Chiller A	VWS
Chiller B	VWS
Sample HX A	PSS
Sample HX B	PSS
Makeup Pump Miniflow HX	CVS
Makeup Pump Miniflow HX	CVS
Air Compressor A	CAS
Air Compressor B	CAS
Air Compressor C	CAS
Air Compressor D	CAS
Cond Pump A Oil Cooler	CDS
Cond Pump B Oil Cooler	CDS
Cond Pump C Oil Cooler	CDS

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Table 9.2.3-1
Guidelines for Demineralized Water
(Measured at the Outlet of the Demineralized Water Treatment System)

Parameters	Normal Value	Initiate Action
Specific conductivity, µS/cm at 77°F	≤ 0.1	≤ 0.2
Active silica, ppb	≤ 10	≤ 20
Total silica, ppb	≤ 50	
Suspended solids, ppb	≤ 50	
Aluminum, ppb	≤ 20	
Calcium, ppb	≤ 5	
Magnesium, ppb	≤ 5	
Chloride, ppb	≤ 1	
Sulfate, ppb	≤ 1	
Total organic carbon, ppb	≤ 100	

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Table 9.2.7-1
Component Data - Central Chilled Water System
(Nominal Values)

High Capacity Subsystem	
Water Cooled Chillers	
Capacity (nominal tons)	1700
Compressor type	Centrifugal
Maximum power input (kW)	1500
Entering water temperature (°F)	56
Leaving water temperature (°F)	40
Cooling water flowrate (gpm)	3500 (max)
Air-Cooled Chillers	
Capacity (nominal tons)	300
Compressor type	Reciprocating, Screw
Maximum power input (kW)	470
Entering water temperature (°F)	56
Leaving water temperature (°F)	40
Low Capacity Subsystem	
Air-Cooled Chillers	
Capacity (nominal tons)	230
Compressor type	Reciprocating, Screw
Maximum power input (kW)	450
Entering water temperature (°F)	56
Leaving water temperature (°F)	40
Coil	**Flow (gpm)**
VBS MY C01 A/B (MCR/CSA)	101
VBS MY C02 A/C (Class 1E Division A/C Electrical Room)	106
VBS MY C02 B/D (Class 1E Division B/D Electrical Room)	56
VAS MY C07 A/B (CVS Makeup Pump Room)	14
VAS MY C12 A/B (RNS Pump Room)	10
VAS MY C06 A/B (RNS Pump Room)	10

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Table 9.2.8-1
Turbine Building Closed Cooling Water System
Equipment Load List

Component
Main turbine lube oil coolers
Main feedwater pump lube oil coolers
Generator hydrogen coolers
Generator stator cooling water cooler
Isolated phase bus coolers
MSR drain pump
EH control coolers
Secondary sampling system coolers

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Table 9.2.10-1
Hot Water Heating System Design Data
(Nominal Values)

Available Steam Supply	
Main steam	
Pressure (psia)	200
Enthalpy (Btu/lbm)	1199
Temperature (°F)	382
Auxiliary steam	
Pressure (psia)	210
Enthalpy (Btu/lbm)	1199
Temperature (°F)	386
Heat Exchanger	
Quantity	2
Type	Shell and Tube – subcooled
Capacity per unit (Btu/hr)	21,090,000

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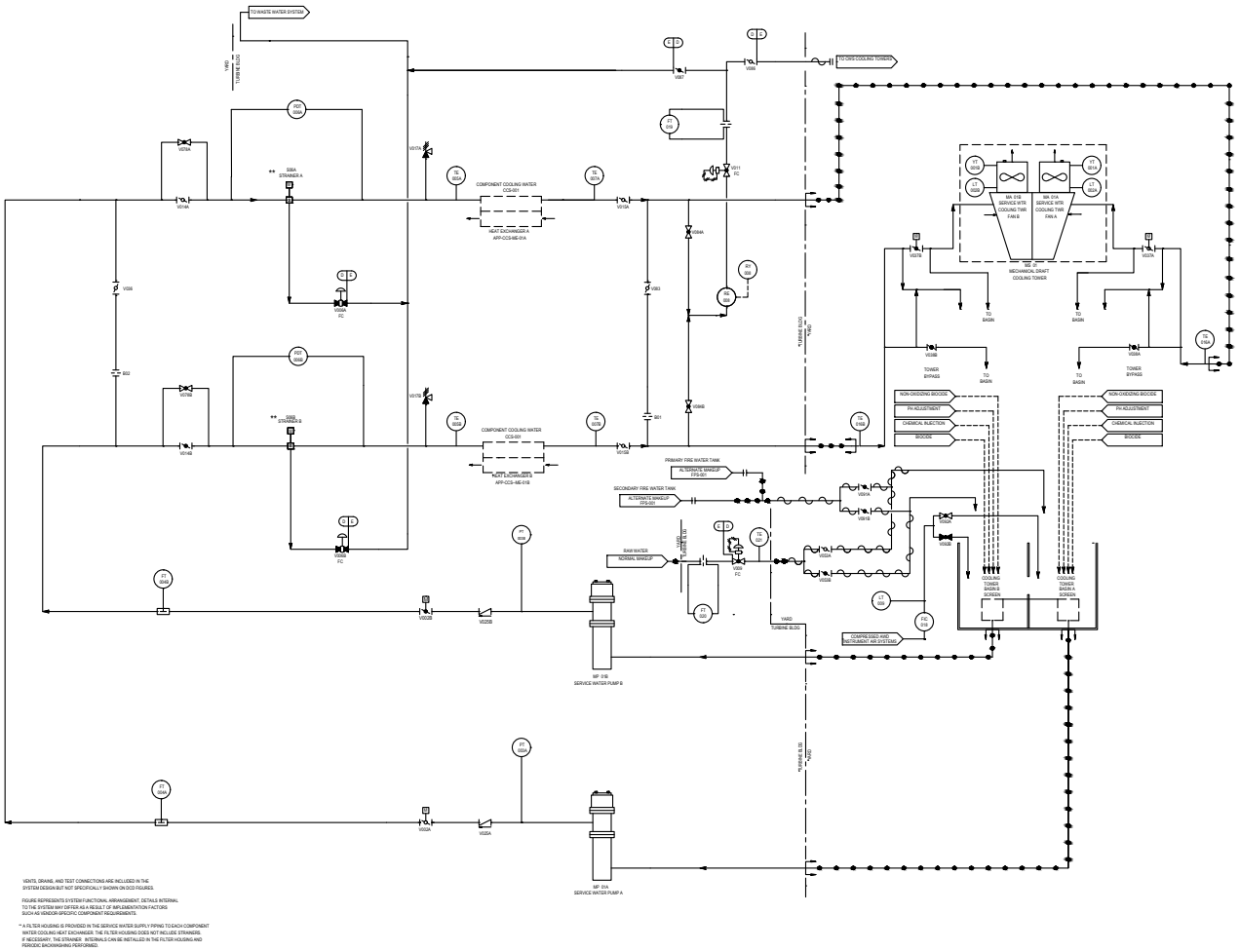

Figure 9.2.1-1
Simplified Service Water System
Piping and Instrumentation Diagram
(REF) SWS 001

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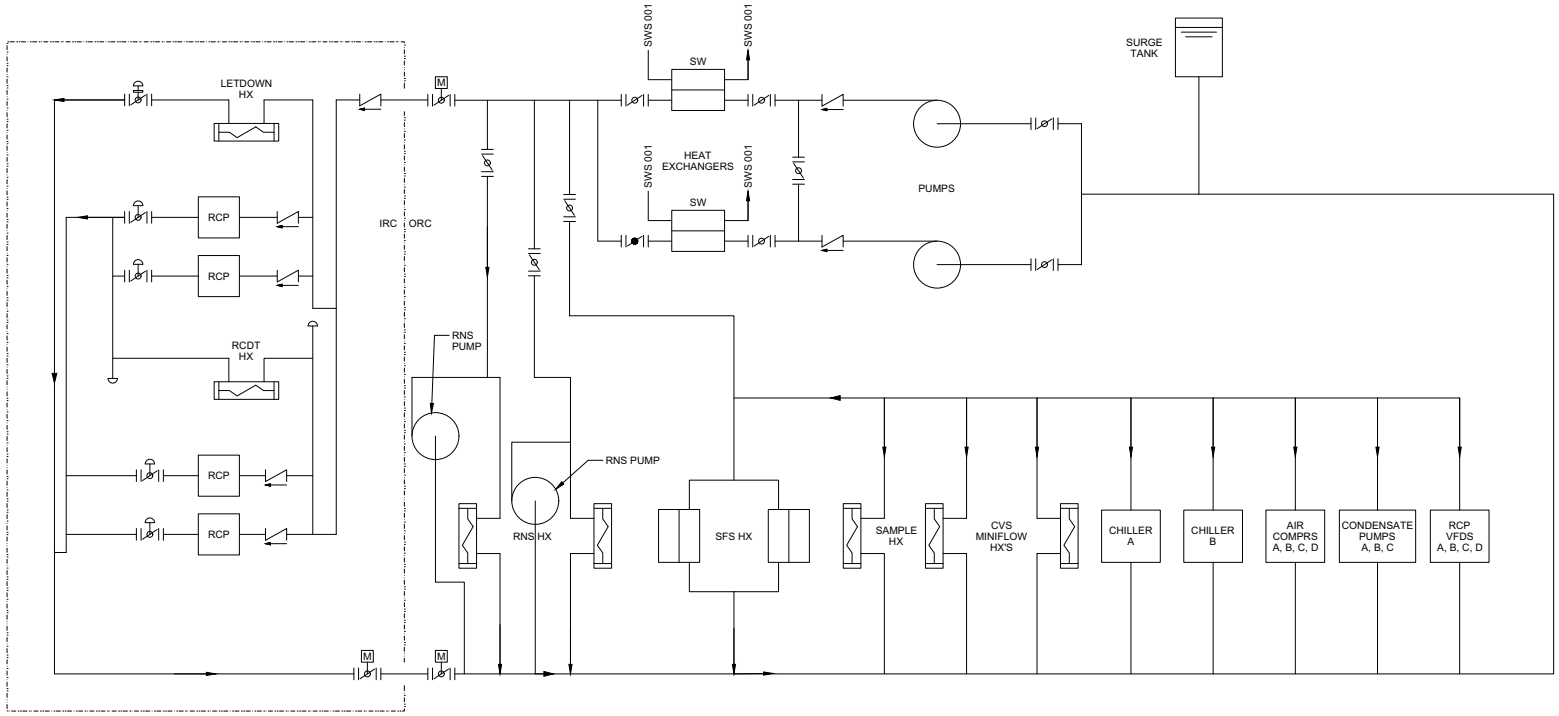

Figure 9.2.2-1
Component Cooling Water System
Simplified Flow Diagram
(REF) CCS

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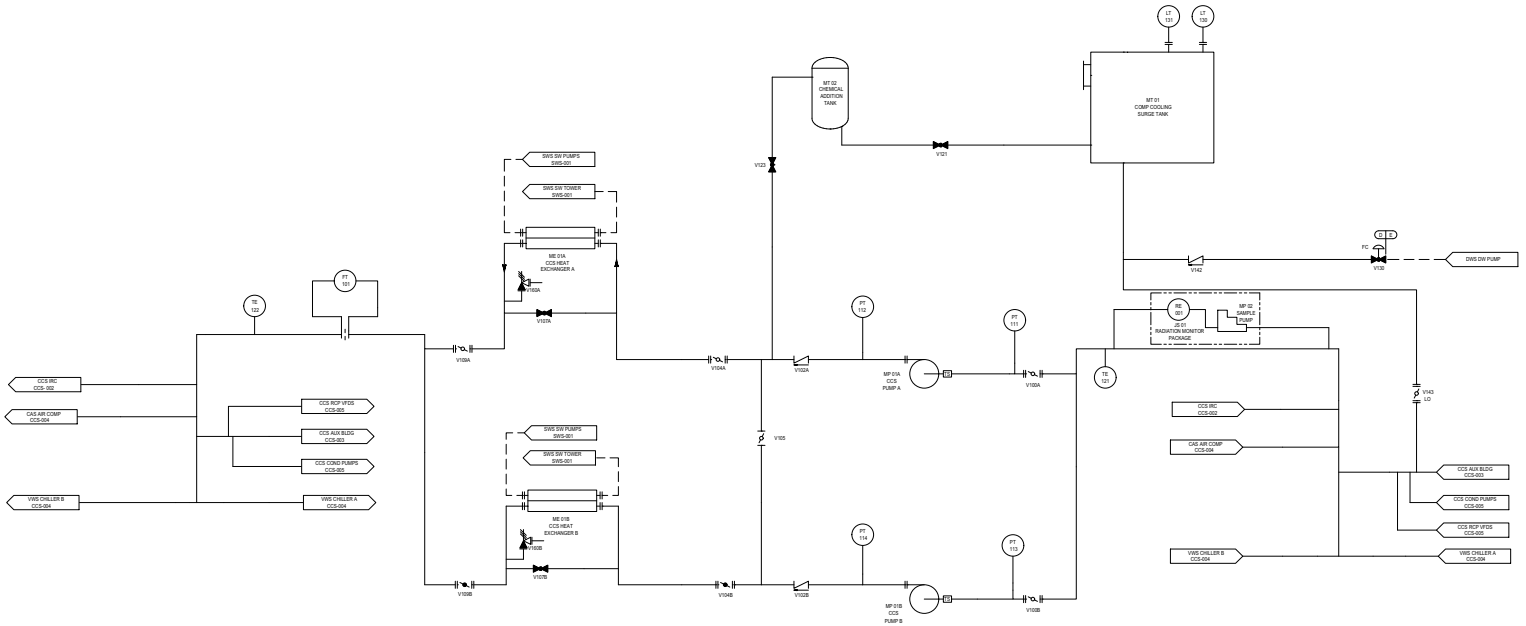

SYSTEM COMPONENTS ARE CLASS D UNLESS NOTED.

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN IN THE DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Inside Turbine Building

Figure 9.2.2-2 (Sheet 1 of 5)
Simplified Component Cooling Water System
Piping and Instrumentation Diagram
(REF) CCS 001

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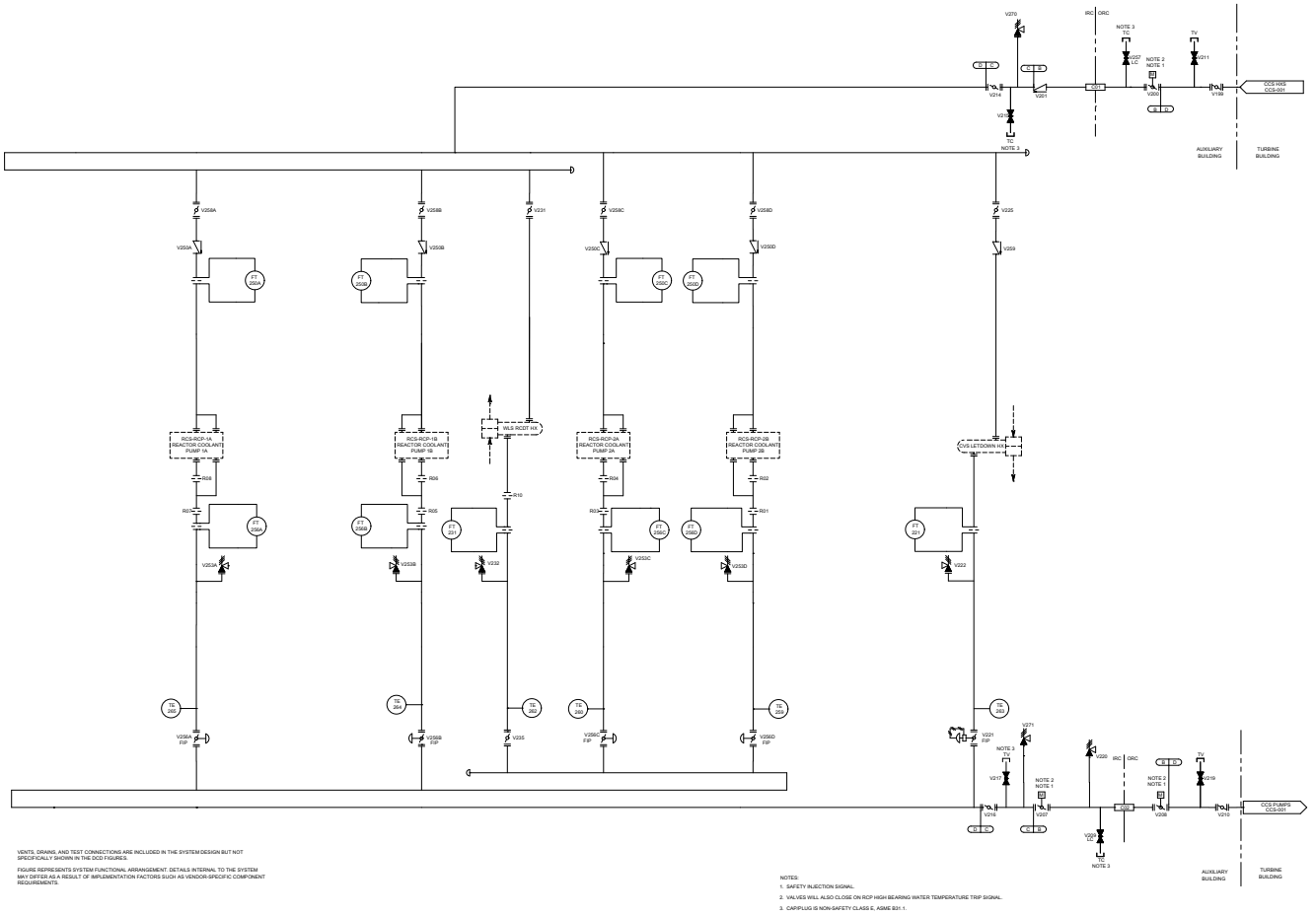

Figure 9.2.2-2 (Sheet 2 of 5)
Simplified Component Cooling Water System
Piping and Instrumentation Diagram
(REF) CCS 002

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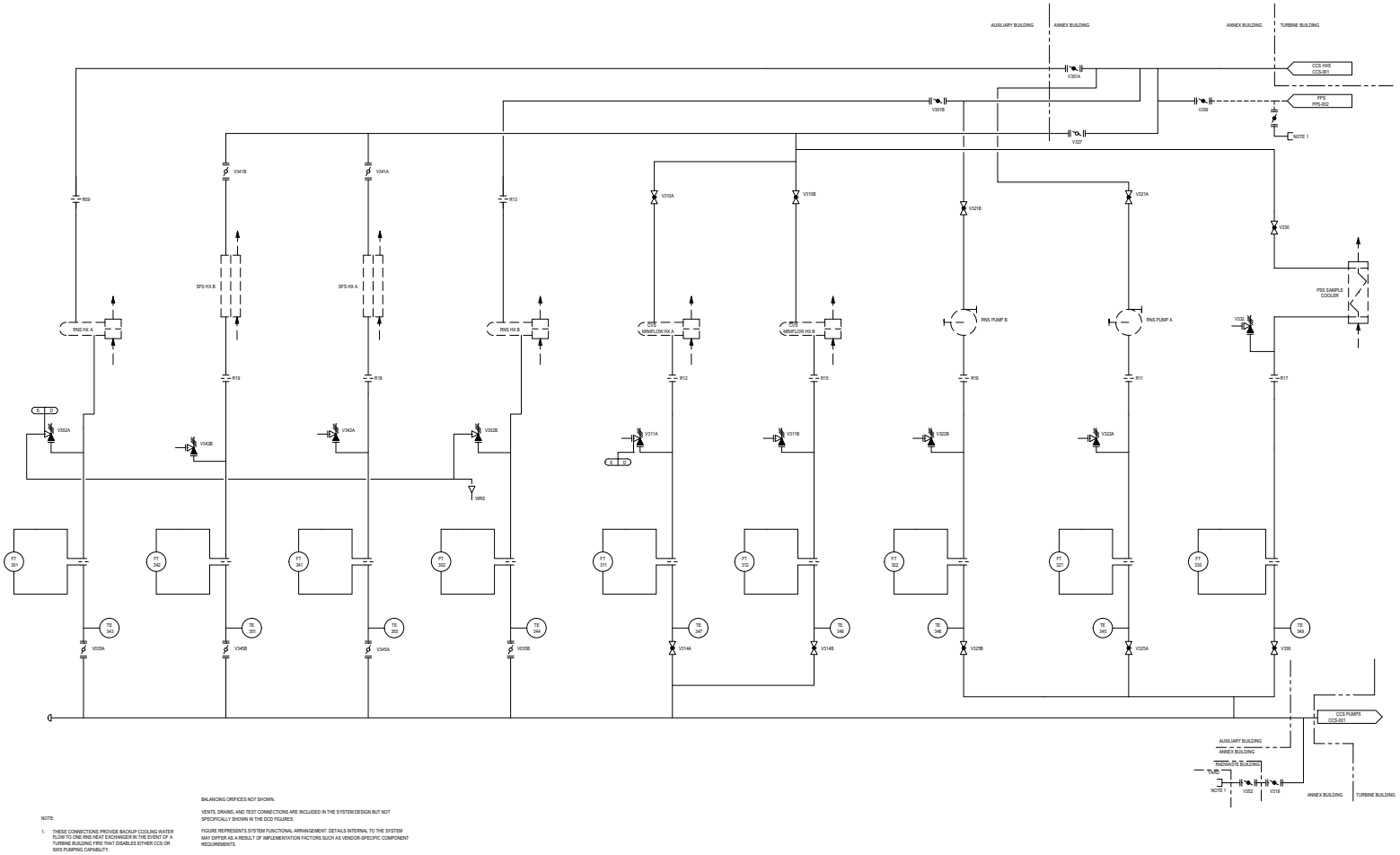

Figure 9.2.2-2 (Sheet 3 of 5)
Simplified Component Cooling Water System Piping and Instrumentation Diagram
(REF) CCS 003

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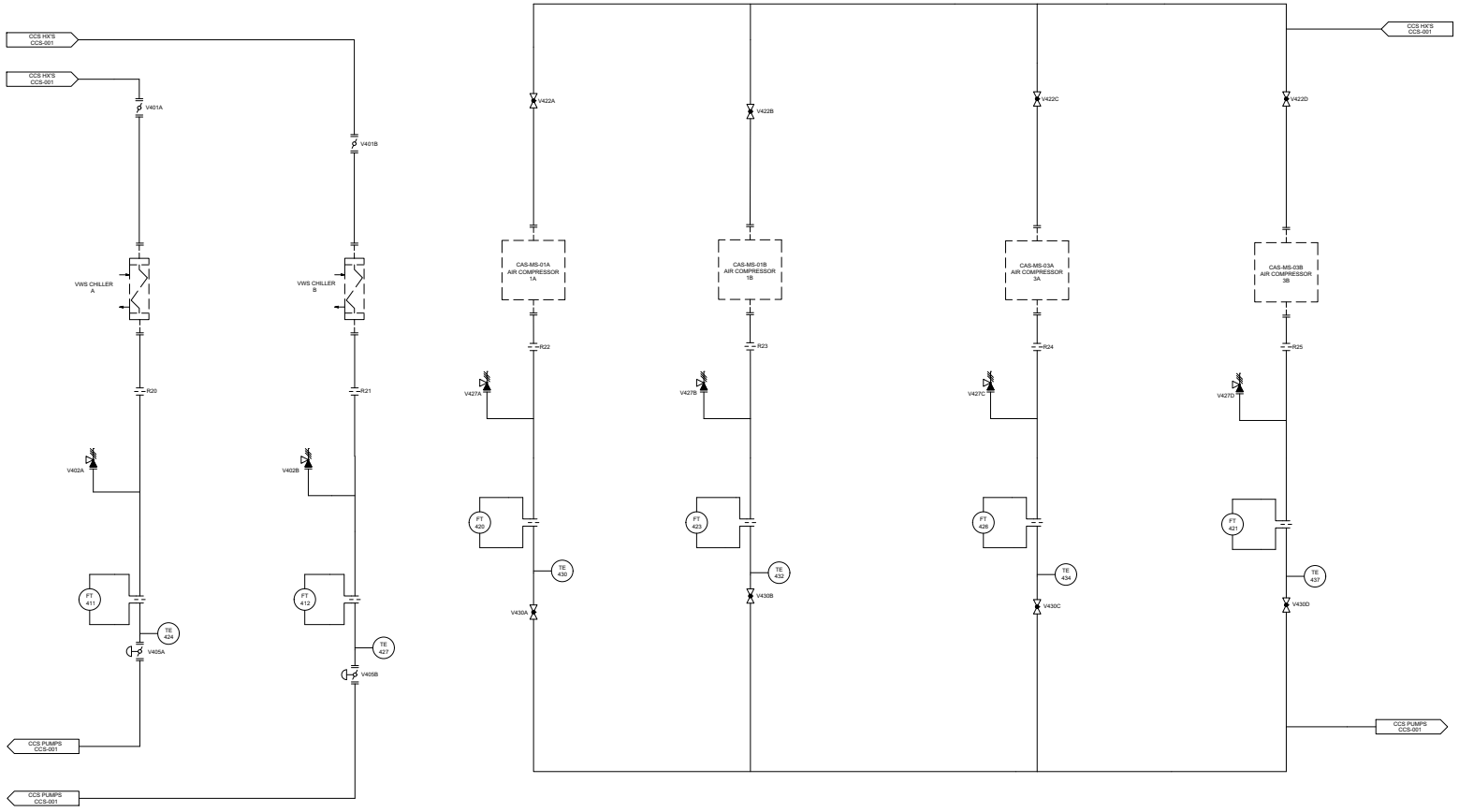

SYSTEM COMPONENTS ARE CLASS D UNLESS NOTED.

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN IN THE DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Inside Turbine Building

Figure 9.2.2-2 (Sheet 4 of 5)
Simplified Component Cooling Water System
Piping and Instrumentation Diagram
(REF) CCS 004

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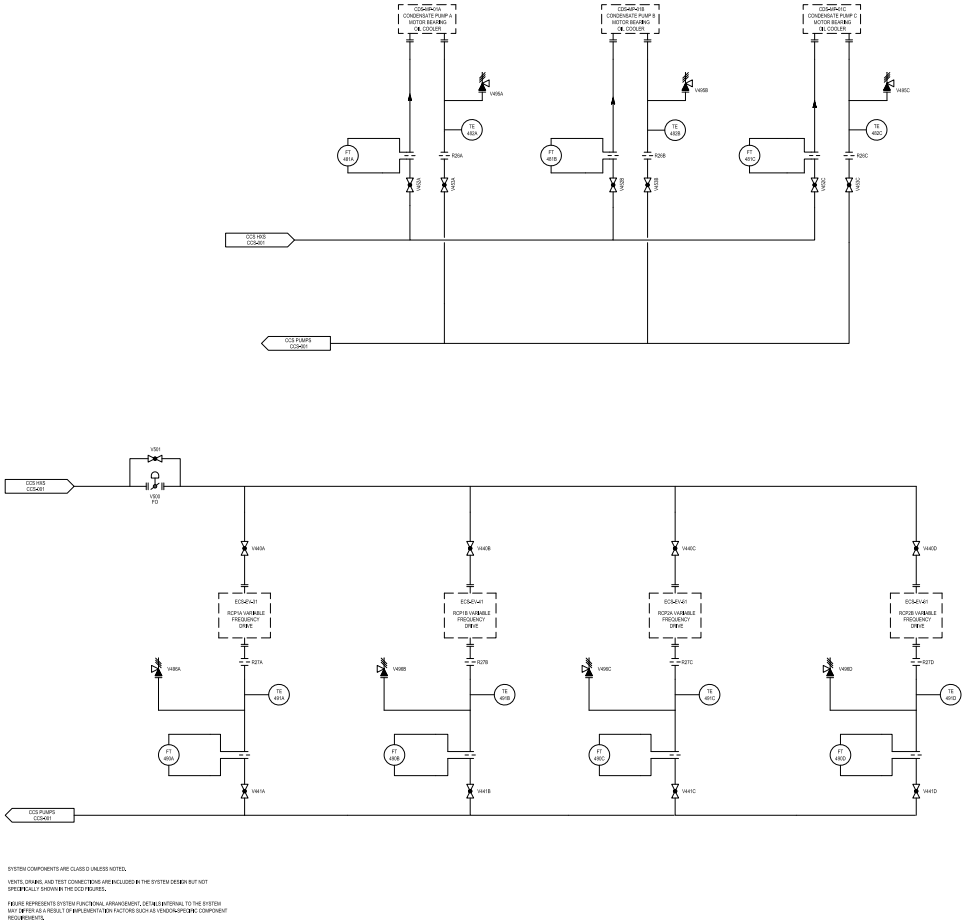

Inside Turbine Building

Figure 9.2.2-2 (Sheet 5 of 5)
Simplified Component Cooling Water System
Piping and Instrumentation Diagram
(REF) CCS 005

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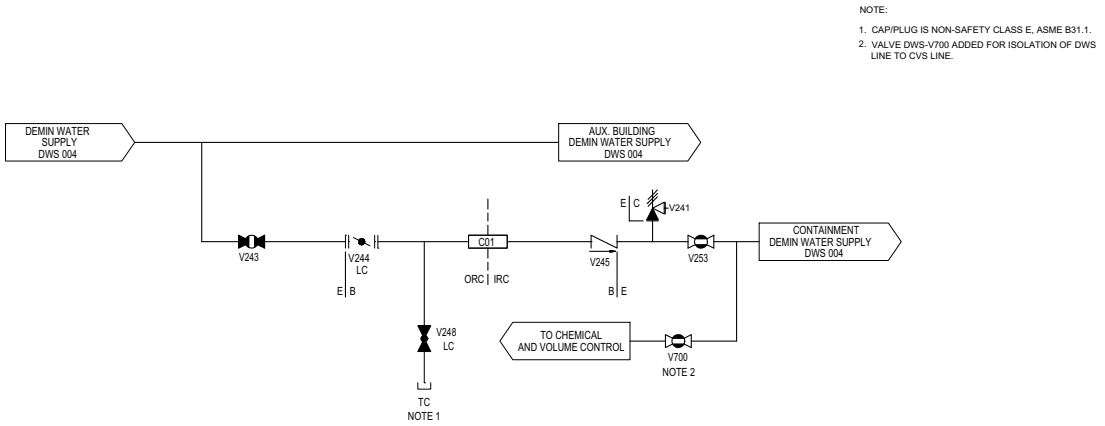

Figure 9.2.4-1
Demineralized Water Transfer and Storage System
Containment Isolation Provision
(REF) DWS 007

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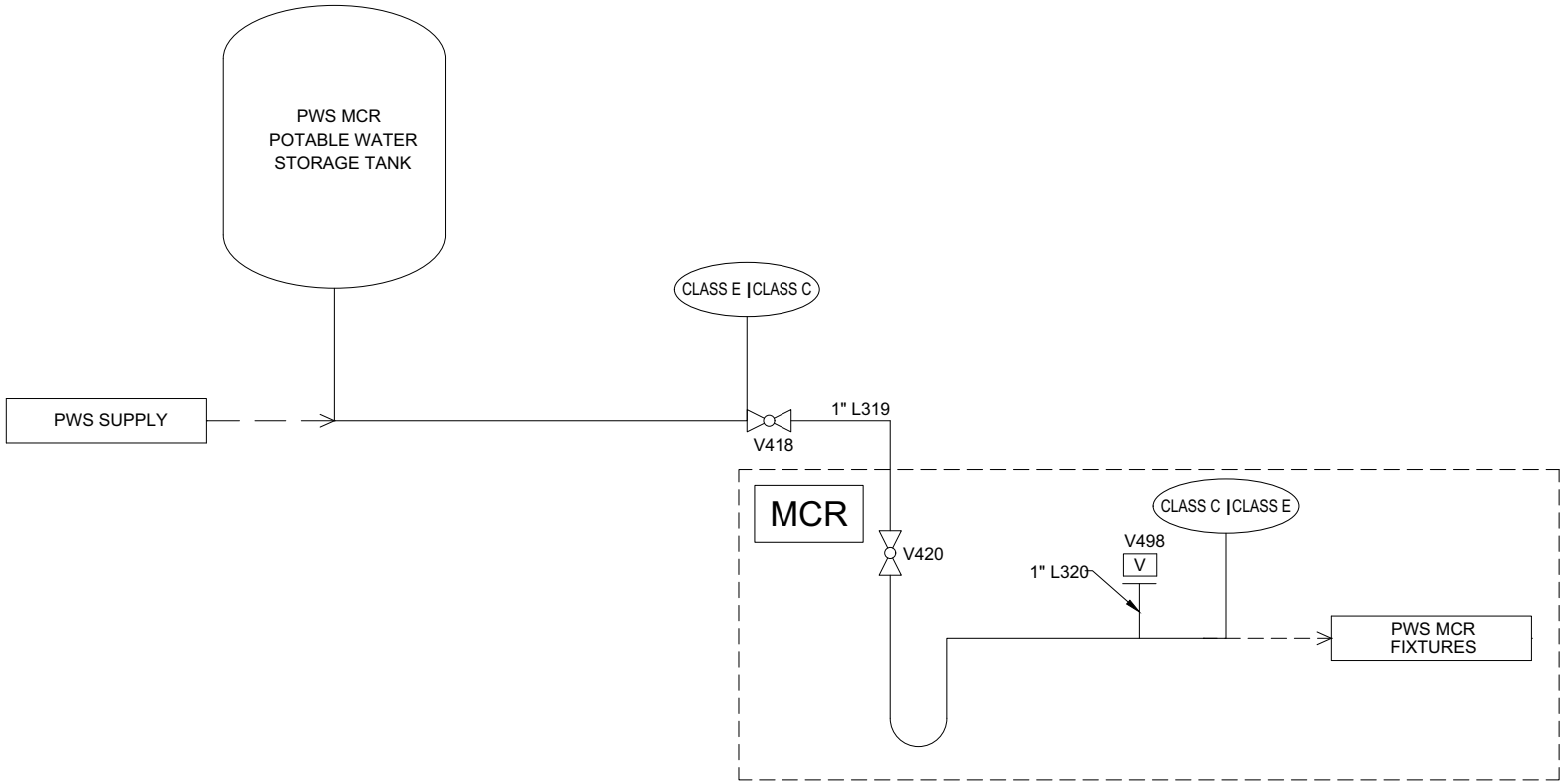

Figure 9.2.5-1
Main Control Room Potable Water System Isolation

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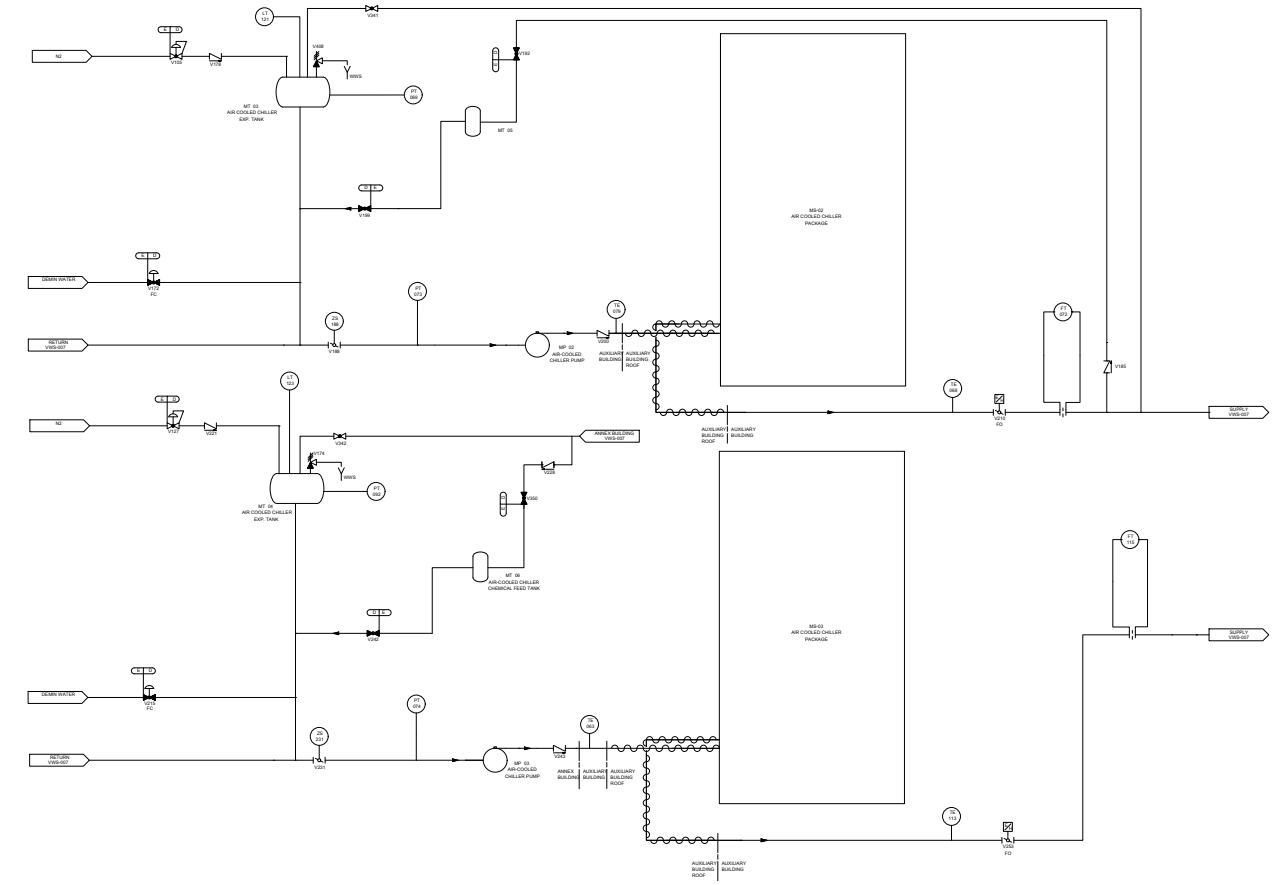

Figure 9.2.7-1 (Sheet 1 of 4)
Simplified Central Chilled Water System
Piping and Instrumentation Diagram
(REF) VWS 006

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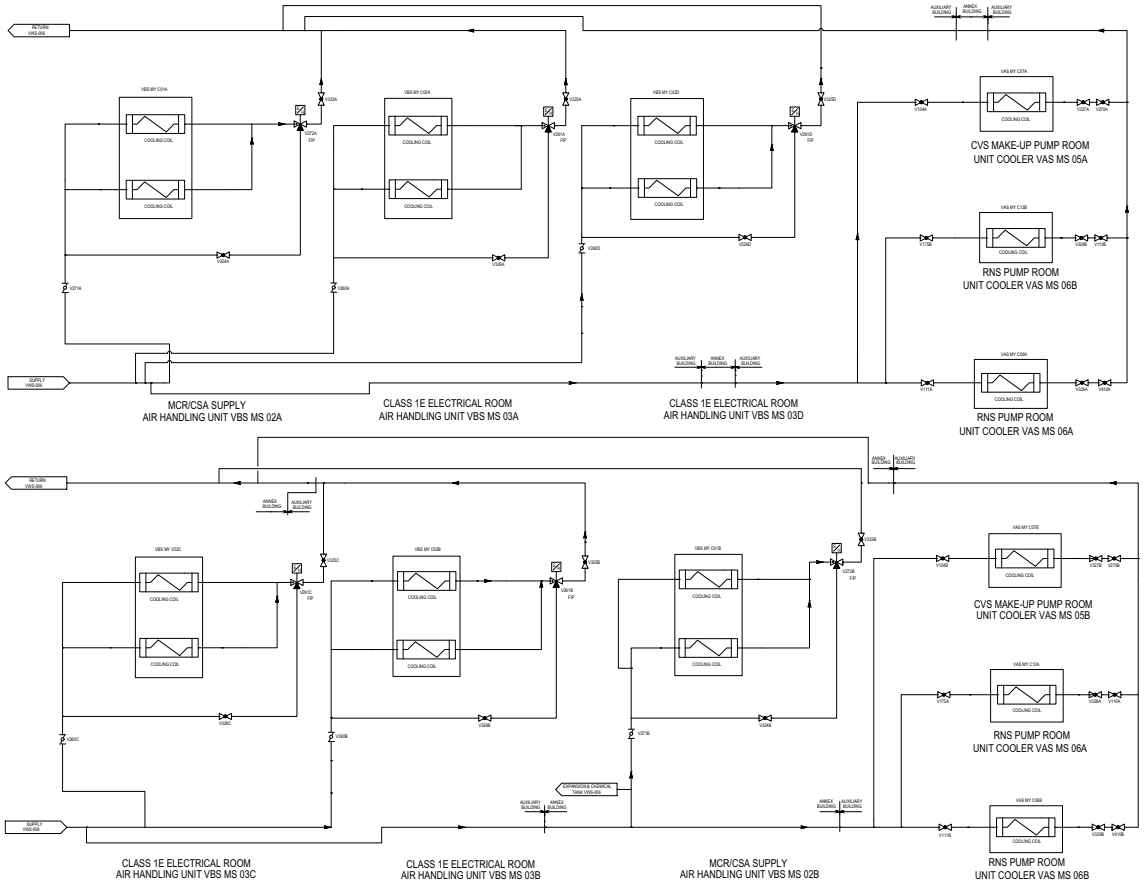

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Figure 9.2.7-1 (Sheet 2 of 4)
Simplified Central Chilled Water System Piping and Instrumentation Diagram (REF) VWS 007

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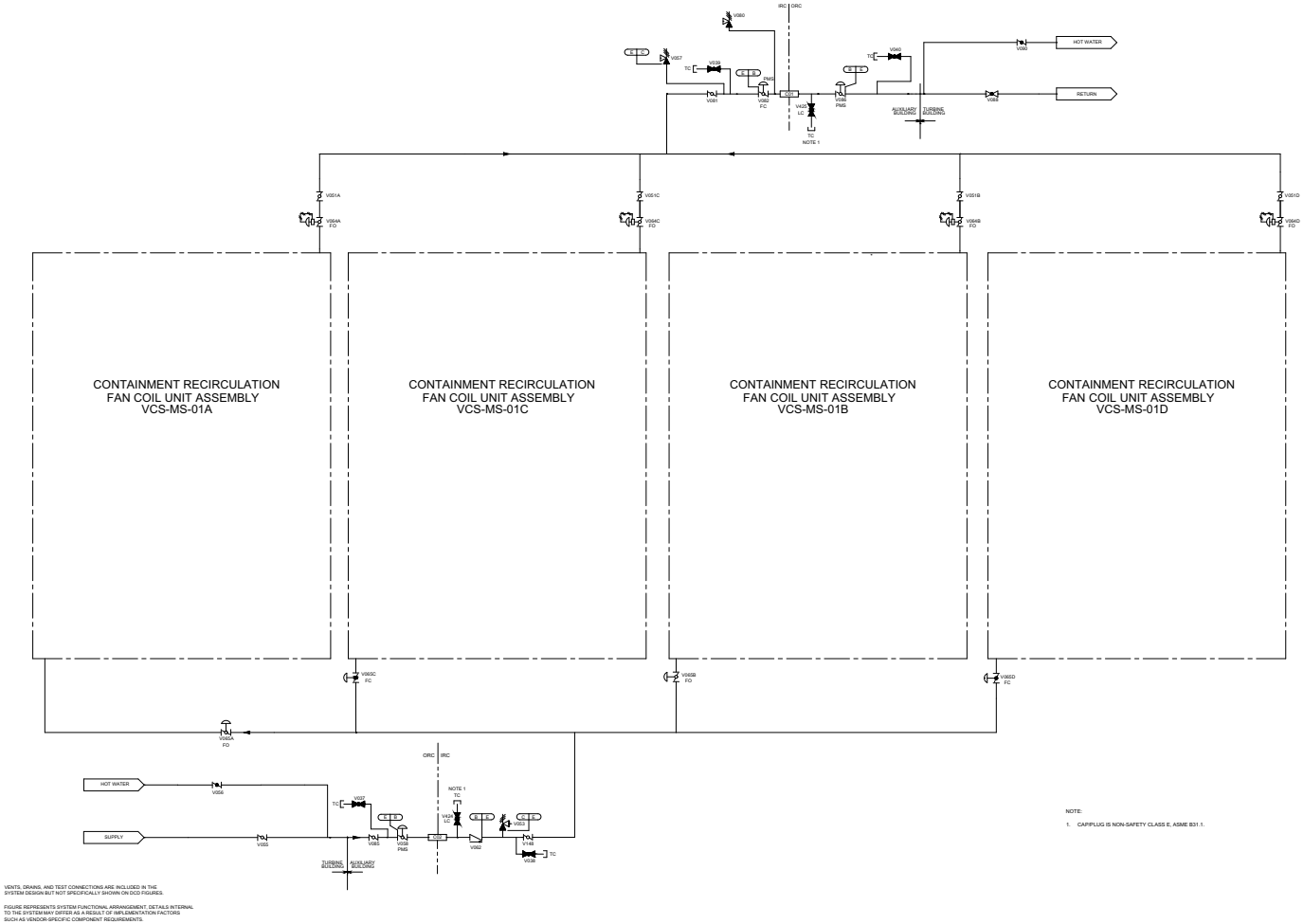

Figure 9.2.7-1 (Sheet 3 of 4)
Simplified Central Chilled Water System
Piping and Instrumentation Diagram
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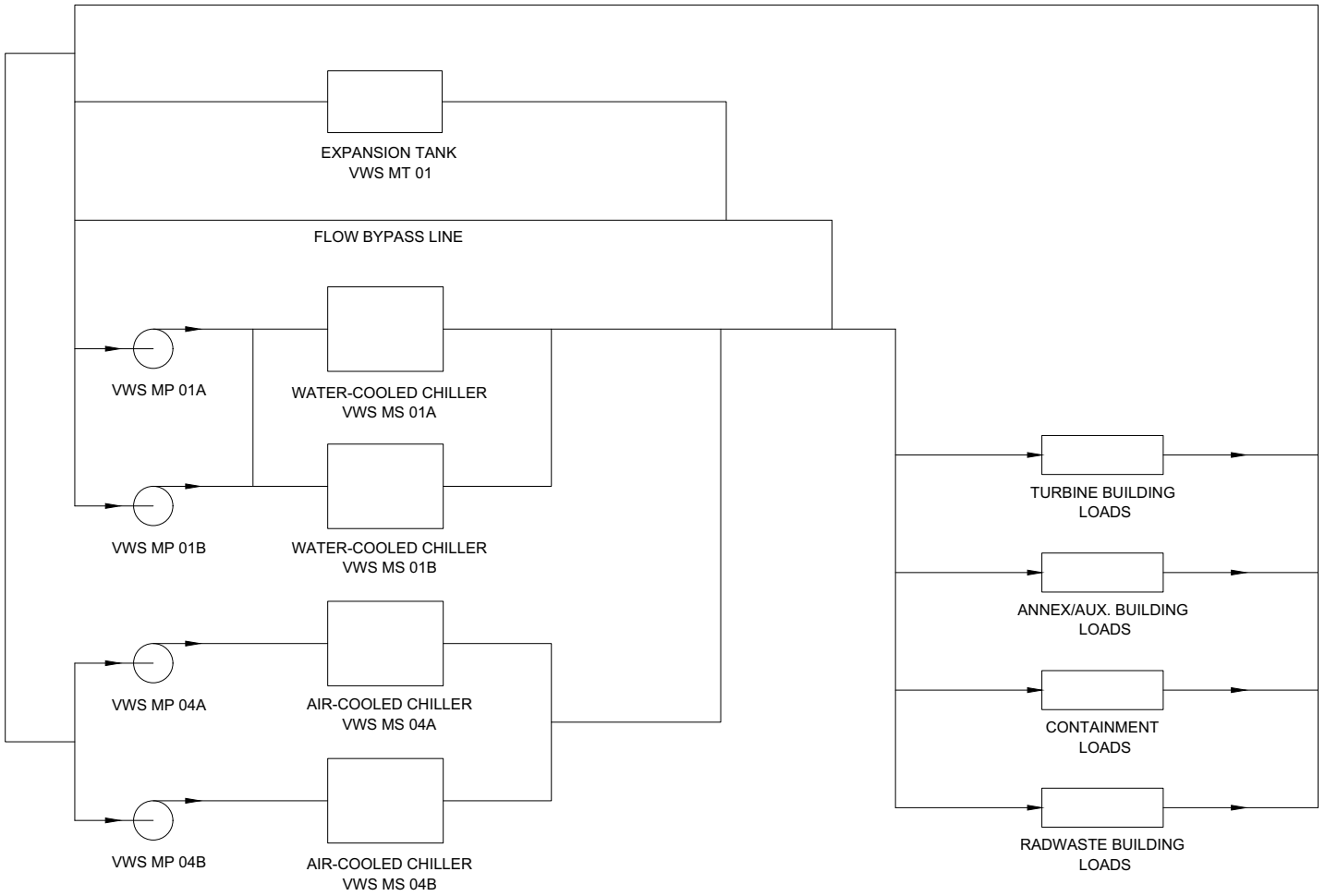

Figure 9.2.7-1 (Sheet 4 of 4)
High Capacity Subsystem Simplified Sketch

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9.3 Process Auxiliaries

9.3.1 Compressed and Instrument Air System

The compressed and instrument air system (CAS) consists of three subsystems; instrument air, service air, and high-pressure air. Instrument air supplies compressed air for air-operated valves and dampers. Service air is supplied at outlets throughout the plant to power air-operated tools and is used as a motive force for air-powered pumps. The service air subsystem is also utilized as a supply source for breathing air. Individually packaged air purification equipment is used to produce breathing quality air for protection against airborne contamination. The high-pressure air subsystem supplies air to the main control room emergency habitability system (VES) and fire fighting apparatus recharge station. The high-pressure air subsystem also provides a connection for refilling the VES storage tanks from an offsite source. Major components of the compressed and instrument air system are located in the turbine building.

9.3.1.1 Design Basis

9.3.1.1.1 Safety Design Basis

The compressed and instrument air system serves no safety-related function other than containment isolation and therefore has no nuclear safety design basis except for containment isolation. See Subsection 6.2.3 for the containment isolation system.

9.3.1.1.2 Power Generation Design Basis

The instrument air subsystem provides filtered, dried, and oil-free air for air-operated valves and dampers. The instrument air subsystem consists of two compressors and associated support equipment and controls that are powered from switchgear backed by the nonsafety-related onsite standby diesel generators as an investment protection category load. The subsystem provides high quality instrument air as specified in the ANSI/ISA S7.3 standard (Reference 9.3.8.1).

The service air subsystem provides filtered, dried, and oil-free compressed air for service outlets located throughout the plant. The service air subsystem consists of two compressors and their associated support equipment and controls. Plant breathing air requirements are satisfied by using the service air subsystem as a supply source. Individually packaged air purification equipment is used to improve the service air to Quality Verification Level D breathing air as defined in CGA G-7.1 (Reference 7).

The high-pressure air subsystem consists of one compressor, its associated air purification system and controls, and a high-pressure receiver. It provides clean, oil-free, high-pressure air to recharge the main control room emergency habitability system cylinders and refill the individual fire fighting breathing air bottles. Quality Verification Level E air as defined in CGA G-7.1 (Reference 7), with a pressure dew point of 40°F or lower at 3,400 psig or greater, is produced by this subsystem. See Section 6.4 for a description of the main control room habitability system.

9.3.1.2 System Description

9.3.1.2.1 General Description

Classifications of components and equipment in the compressed and instrument air system are given in Section 3.2. In accordance with NUREG-1275, instrument air quality meets the manufacturer's standards for pneumatic equipment supplied as a part of the plant. Intake filters for instrument air, service air, and high-pressure air compressors remove particulates 10 microns and larger.

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Instrument Air Subsystem

The instrument air subsystem consists of two 100 percent capacity parallel air supply trains discharging to a common air distribution system. An air compressor, dryer, controls, and receiver comprise one air supply train. The two compressor trains join to a single instrument air header downstream of the receivers.

Provisions are made to temporarily cross connect the instrument and service air subsystems at the distribution header.

The instrument air line to the containment is normally open; however, air flow to the containment is monitored and a high flow alarm is provided to indicate a possible instrument air line rupture inside containment. Safety-related air-operated valves supplied by the system are identified in Table 9.3.1-1. None of these valves require instrument air to perform their safety-related function. The valves with an active safety-related function fail in the safe position on loss of instrument air pressure.

One instrument air compressor train, including its air dryer and associated equipment and controls, can be connected to each of the nonsafety-related onsite standby diesel generators. The compressors are cooled by water supplied from the component cooling water system (CCS). Refer to Subsection 9.2.2 for details. The instrument air subsystem is shown schematically in Figure 9.3.1-1. Major system components are described in Table 9.3.1-2.

Service Air Subsystem

Two 100 percent capacity compressor trains are provided for the service air subsystem. These compressor trains consist of identical equipment and share a common air receiver that feeds the service air distribution system. Cooling water to the service air compressors is supplied from the component cooling water system. Refer to Subsection 9.2.2 for details.

The service air line to containment is normally closed and is opened on an as-needed basis. The service air subsystem is shown schematically in Figure 9.3.1-1 and major system components are described in Table 9.3.1-3.

High-Pressure Air Subsystem

The high-pressure air subsystem consists of a high-pressure air compressor with an integral air purification system, controls, and a receiver.

The high-pressure air subsystem is manually operated and may be loaded on an onsite standby diesel generator. This subsystem supplies air to the main control room emergency habitability system and the fire fighting apparatus recharge station. The isolation valves to these locations are normally closed and are opened on an as-needed basis to refill the specified equipment air storage reservoirs. The high-pressure air subsystem is shown schematically in Figure 9.3.1-1 and major system components are described in Table 9.3.1-4.

9.3.1.2.2 Component Description

Instrument Air Subsystem

The instrument air subsystem consists of two air compressor trains. Each compressor train consists of a multistage, low-pressure, rotary screw, air compressor package, a desiccant dryer with a prefilter and afterfilter, and an air receiver. Each compressor package includes an intake filter, rotary screw compressor elements, silencer, intercooler, aftercooler, moisture separators, bleed-off cooler, oil cooler, oil reservoir, automatic load controls, relief valves, and a discharge air check valve. Each compressor train produces oil-free air.

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Two instrument air receivers function as storage devices for compressed air. The receivers continue to supply the instrument air subsystem following a loss of the instrument air compressors until the receiver pressure drops below system requirements. Each air receiver is equipped with a pressure relief valve.

Two air dryer assemblies are provided for the instrument air subsystem. Each dryer assembly consists of a desiccant-filled, twin tower design. One tower may be used to dry air while the other tower goes through regeneration. When instrumentation senses a high dew point, the towers switch. The former operating tower then undergoes regeneration while the regenerated tower drys the instrument air.

Each dryer assembly includes a coalescing prefilter that removes oil aerosols and moisture droplets, as well as an afterfilter to remove desiccant dust. Each air dryer assembly is provided with an automatic condensate drain valve.

The instrument air subsystem supplies ANSI/ISA S-7.3 high quality instrument air. Table 9.3.1-2 provides design information for the main components associated with the instrument air subsystem.

Service Air Subsystem

The service air subsystem consists of two air compressor trains. Each compressor train consists of a multistage, low-pressure, rotary screw, air compressor package, and a desiccant dryer with a prefilter and afterfilter. A common air receiver is provided for the two trains. Each compressor package includes an intake filter, rotary screw compressor elements, silencer, intercooler, aftercooler, moisture separators, bleed-off cooler, oil cooler, oil reservoir, automatic load controls, relief valves, and a discharge air check valve. Each compressor train produces oil-free air.

The common service air receiver functions as a storage device for compressed air. This air receiver is equipped with a pressure relief valve.

Two air dryer assemblies are provided for the service air subsystem. Each dryer assembly consists of a desiccant-filled, twin tower design. One tower may be used to dry air while the other tower goes through regeneration. When instrumentation senses a high dew point, the towers switch. The former operating tower then undergoes regeneration while the regenerated tower dries the service air.

Each dryer assembly includes a coalescing prefilter that removes oil aerosols and moisture droplets, as well as an afterfilter to remove desiccant dust. Each air dryer assembly is provided with an automatic condensate drain valve.

Table 9.3.1-3 provides design information for the main components associated with the service air subsystem.

High-Pressure Air Subsystem

The high-pressure air subsystem utilizes an air-cooled, oil-lubricated, four-stage, reciprocating-air compressor with an integral air purification system to produce oil-free air for high-pressure applications. The compressor train includes an intake filter, air-cooled intercoolers, interstage oil/water separators, an air-cooled aftercooler, a final oil/water separator, relief valves, an air purification system, discharge check valves, and a high-pressure receiver.

The high-pressure air subsystem supplies CGA G-7.1 (Reference 7) Quality Verification Level E air. See Table 9.3.1-4 for the design parameters for this system.

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9.3.1.2.3 System Operation

Instrument Air Subsystem

The instrument air compressors are operated by a local pressure controller located in the instrument air distribution header, which can be programmed for various sequences of operation. Normally one compressor runs continuously loading and unloading as required to supply compressed air demand. The second compressor serves as a backup and starts automatically if the first unit fails or if demand exceeds the capacity of the operating compressor.

Air from the instrument air subsystem compressor packages discharges to the air dryers and then to the receivers where it is distributed to air-operated valves and dampers throughout the plant. Instrument air pressure is reduced by pressure regulators at the pneumatic component as required.

The onsite standby power system (diesel generators) provides an alternate source of electrical power for the instrument air compressor trains. One compressor train is supplied from each electrical load group.

Service Air Subsystem

The service air subsystem compressors are operated by a local controller that can be programmed for various sequences of operation. Normally one compressor runs continuously and loads and unloads as required to supply service air demand. The second compressor serves as a backup and starts automatically if the first compressor fails or demand exceeds the capacity of the operating compressor. Air from each service air subsystem compressor package discharges to an air dryer and then to the common receiver. Service air flows from the receiver to the various service air outlets throughout the plant.

Breathing air can be obtained from any service air subsystem outlet by attaching a portable individually packaged air purification system. The breathing air purification package consists of replaceable cartridge-type filters, a pressure regulator, carbon monoxide monitoring equipment, air supply hoses, and air supply devices. Carbon monoxide is controlled by a catalytic conversion to carbon dioxide within the package. Breathing air of a Quality Verification Level D or better is supplied to personnel from the packaged purification system in accordance with the requirements of CGA G-7.1 (Reference 7).

High-Pressure Air Subsystem

The high-pressure air subsystem is operated when a specific high-pressure source requires refilling to replace air lost to leakage or expended during plant operations. System isolation valves to the specified equipment are manually opened and the equipment storage reservoir is replenished from the high-pressure receiver. The compressor is started to replenish the air stored in the high-pressure receiver.

Breathing air of a Quality Verification Level E is supplied from the integral high-pressure air purification system in accordance with the requirements of CGA G-7.1 (Reference 7). This integral air purification system utilizes a series of replaceable cartridge-type filters to produce breathing quality air. Breathing air connections of the high-pressure air subsystem are incompatible with the breathing air connections of the service air subsystem. Carbon monoxide is controlled by a catalytic conversion to carbon dioxide within the package.

The onsite standby power system (diesel generators) provides an alternate source of electrical power for the high-pressure air compressor.

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9.3.1.3 Safety Evaluation

The compressed and instrument air system has no safety-related function other than containment isolation and therefore requires no nuclear safety evaluation. Containment isolation functions are described in Subsection 6.2.3.

The compressed and instrument air system is required for normal operation and startup of the plant. Air-operated valves that are essential for safe shutdown and accident mitigation are designed to actuate to the fail-safe position upon loss of air pressure. These air-operated valves utilize safety-related solenoid valves to control the air supply.

The instrument and service air subsystems are classified as moderate-energy systems. There are no adverse effects on safety-related components associated with a postulated failure of the instrument and service air piping.

The high-pressure air subsystem is classified as a high-energy system. The high-pressure compressor and receiver are located in the turbine building, which contains no safety-related, equipment or structures. Air piping routed in safety-related areas is 1 inch or less in diameter and the dynamic consequences of a rupture are not required to be analyzed. The high-pressure air subsystem is not required to operate following a design basis accident nor is it used for safe shutdown of the plant.

9.3.1.4 Tests and Inspections

System components, such as the air compressors and air dryers, are inspected or tested prior to installation. The installed compressed and instrument air system is inspected, tested, and operated to verify that it meets its performance requirements, including operational sequences and alarm functions.

Air compressors and associated components on standby are checked and operated periodically. Desiccant in the air dryers is changed when required.

Sample points are provided downstream of the air dryers in both the instrument and service air subsystems and downstream of the purifier in the high-pressure air subsystem. Periodic checks are made to ensure high quality instrument air and service air as specified in the ANSI/ISA S-7.3 standard. Periodic checks on the high-pressure air compressor are made on a regular basis to verify that the breathing air meets the Quality Verification Level E as indicated in CGA G-7.1 (Reference 7).

During the initial plant testing prior to reactor startup, safety systems utilizing instrument air are tested as part of the safety system test to verify fail-safe operation of air-operated valves upon complete loss of system pressure, both sudden and gradual, and to partial reduction in system pressure as described in Regulatory Guide 1.68.3. Appendix 1A summarizes conformance with Regulatory Guide 1.68.

9.3.1.5 Instrumentation Applications

An instrumentation package is included with each of the instrument and service air compressors. Each package consists of temperature and pressure transducers, indicators, and automatic protection devices. The temperature and pressure transducers support the automatic control modes of compressor operation. A manual mode of operation is also provided for each control system. Compressed and instrument air system indication and control are available in the main control room.

The high-pressure air subsystem includes pressure and carbon monoxide instrumentation, automatic protection devices, and temperature indication.

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9.3.2 Plant Gas System

The plant gas system (PGS) provides hydrogen, carbon dioxide, and nitrogen gases to the plant systems as required.

Other gases, such as oxygen, methane, acetylene, and argon, are supplied in smaller individual containers and are not supplied by the plant gas system.

9.3.2.1 Design Basis

9.3.2.1.1 Safety Design Basis

The plant gas system serves no safety-related function and therefore has no nuclear safety design basis.

9.3.2.1.2 Power Generation Design Basis

The nitrogen portion of the plant gas system supplies nitrogen for pressurizing, blanketing, and purging of various plant components.

The hydrogen gas portion of the plant gas system supplies hydrogen to the main plant electrical generator for cooling as well as to other plant auxiliary systems.

The carbon dioxide portion of the plant gas system supplies carbon dioxide to the generator for purging of hydrogen and air during layup or plant outages.

9.3.2.2 System Description

Classification of equipment and components is given in Section 3.2.

9.3.2.2.1 General Description

The nitrogen portion of the plant gas system is a packaged system consisting of a liquid nitrogen storage tank and vaporizers. Nitrogen gas is supplied in both a high-pressure and a low-pressure subsystem.The high-pressure subsystem uses a pump to pressurize the gas supplying the accumulators in the passive core cooling system. The high-pressure supply is then reduced to supply makeup to the reactor coolant drain tank for purging and blanketing. Low-pressure nitrogen is provided for component purging, layup/blanketing, and pressurization.

The main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs) use compressed nitrogen stored within the valve operators as the motive force to close the valves. The main steam isolation valves are described in Subsection 10.3.2.2.4 and the main feedwater isolation valves are described in Subsection 10.4.7.2.2. Nitrogen makeup for these valves (if needed) is provided from portable high-pressure nitrogen bottles using temporary connections on the valves.

The packaged nitrogen system is located in the gas storage area in the yard.

The low-pressure hydrogen gas portion of the plant gas system is a packaged system consisting of a liquid hydrogen storage tank and vaporizers to supply hydrogen gas to the main generator for generator cooling and the demineralized water transfer and storage system to support removal of dissolved oxygen. The packaged hydrogen system in located in the gas storage area in the yard.

A separate high-pressure hydrogen packaged system supplies hydrogen to the chemical and volume control to support removal of dissolved oxygen in the RCS. The hydrogen for this system is stored as

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a compressed gas in high-pressure tanks (gas bottles). The high-pressure hydrogen supply package system is located outdoors adjacent to the turbine building.

The gas storage area in the yard is located a sufficient distance away from safety-related structures, systems, and components that they are protected from the effects of explosion, flammable vapor cloud, and fire. The gas storage area is located a sufficient distance away from the air inlet to the control room so that the control room operators are protected from potential toxic effects and asphyxiation. The location of the gas storage area is shown in Figure 1.2-2 is an acceptable location. but that location is not part of the certified design. The location of the gas storage area is a site-specific determination.

The carbon dioxide portion of the plant gas system, which is a packaged system consisting of one liquid storage tank and a vaporizer, produces gaseous carbon dioxide to purge the main generator. This packaged system is located in the gas storage area in the yard.

Liquid gas storage tanks are built in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, 1998 Edition, 2000 Addenda (Reference 2).

9.3.2.2.2 Component Description

Liquid Nitrogen Storage Tanks

Liquid nitrogen is stored under its own vapor pressure as a saturated liquid in a dual wall tank. This tank supplies nitrogen for the high- and low-pressure nitrogen gas systems. The annular space between the inner and outer tank walls is filled with insulation and evacuated when the tank is cold.

Liquid Nitrogen Pump

A cryogenic liquid nitrogen pump is utilized to provide a supply of high-pressure nitrogen. It is a single-cylinder, positive displacement pump with the entire “cold” pumping assembly enclosed in a vacuum-jacket, which permits the pump to remain cold in the standby condition.

Nitrogen High-Pressure Ambient Air Vaporizer

Liquid nitrogen is vaporized by a high-pressure natural convection vaporizer, which vaporizes and superheats cryogenic nitrogen using heat from the ambient air.

Nitrogen Low-Pressure Ambient Air Vaporizer

The low-pressure vaporizer unit has two parallel banks. In the event of frost buildup on the active bank, flow is redirected to the opposite bank while the other bank defrosts.

Gaseous Nitrogen Storage Tubes

Gaseous nitrogen storage tubes are provided. These storage tubes provide short-term storage for high-pressure nitrogen.

Liquid Hydrogen Storage Tank

Cryogenic liquid hydrogen is stored in a dual wall tank. The annular space between the walls is insulated using a vacuum and wrapped reflective insulation to minimize heat leakage.

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Hydrogen Ambient Air Vaporizers

Two parallel banks of vaporizers are provided. In the event of frost buildup on the active bank, flow is redirected to the opposite bank while the other bank defrosts.

Gaseous Hydrogen Storage Tanks

Gaseous hydrogen storage tanks are used to provide for storage of high-pressure hydrogen.

Liquid Carbon Dioxide Storage Tank

Cryogenic liquid carbon dioxide is stored in an insulated single wall tank to minimize heat transfer.

Carbon Dioxide Electric Vaporizer

The liquid carbon dioxide is vaporized using electric resistance heating.

9.3.2.2.3 System Operation

Liquid nitrogen is stored under its own vapor pressure as a saturated liquid. An economizer circuit minimizes product loss due to vessel boil off under low-flow conditions. A pressure build circuit maintains pressure at a suitable level above line delivery pressures. For the low-pressure system, liquid is withdrawn, vaporized, and pressure regulated prior to delivery to the low-pressure nitrogen manifold. For high-pressure nitrogen, liquid is withdrawn by the pump, vaporized, and discharged into the high-pressure storage tubes. The gas is then pressure regulated and routed to the high-pressure nitrogen manifold.

Liquid hydrogen is stored in a cryogenic storage vessel complete with an economizer circuit and a pressure build circuit. Ambient air vaporizers turn the liquid into a gas, which is pressure regulated. See subsection 9.3.6 for further discussion of hydrogen use in the chemical and volume control system. The hydrogen used in the chemical and volume control system is supplied from the high-pressure gaseous hydrogen storage tanks.

Liquid carbon dioxide is distributed from a cryogenic storage vessel. An electric vaporizer turns the liquid into a gas, which is pressure regulated for the generator purge.

9.3.2.3 Safety Evaluation

The plant gas system is required for normal plant operation and startup of the plant. The plant gas system is not required for safe shutdown of the plant. Therefore, it is not designed to meet seismic Category I requirements or single failure criterion. The plant gas system serves no safety-related function and has no safety design basis.

The nitrogen, the carbon dioxide, and the hydrogen system storage is located outside of the main buildings. The storage tanks are analyzed as a potential missile source. Refer to Section 3.5. Accidents involving accidental detonations of hydrogen from the onsite storage of compressed or liquid hydrogen are evaluated for damage to safety–related structures, systems, and components. Refer to Section 2.2. For explosions, the plant gas system is designed for conformance with Regulatory Guide 1.91.

The effects of the plant gas system on main control room habitability are addressed in Section 6.4. The main control room habitability evaluation considers the flammability and asphyxiation potential for these gases.

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9.3.2.4 Tests and Inspections

9.3.2.4.1 Storage Vessel Testing

- Each storage vessel is hydrostatically tested in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, 1998 (Reference 3).
- Each vessel is examined using the magnetic particle method.

9.3.2.5 Instrumentation Requirements

Low-level indication alarms are provided in the main control room for the liquid nitrogen and the hydrogen storage tank levels.

Temperatures and pressure indications are provided at various points within the plant gas system.

9.3.3 Primary Sampling System

The AP1000 primary sampling system (PSS) performs the following functions:

- Collects in normal operation mode both liquid and gaseous samples
- Provides for local grab samples during normal operation mode

This system includes equipment to collect representative samples of the various process fluids, including reactor coolant system and containment air, in a manner that adheres to as-low-as-reasonably-achievable (ALARA) principles during normal and post-accident conditions.

The primary sampling system also includes provisions to route sample flow to a laboratory for continuous or intermittent sample analysis, as desired.

The primary sampling system provides a way to monitor the plant and various system conditions using the collected and analyzed samples.

A nonsafety-related containment hydrogen analyzer provided to monitor the containment atmosphere following a postulated loss-of-coolant accident (LOCA) is described in Subsection 6.2.4. A discussion of process radiation monitoring is provided in Section 11.5.

9.3.3.1 Design Bases

9.3.3.1.1 Safety Design Basis

The primary sampling system has no safety-related function, other than containment isolation and therefore requires no nuclear safety evaluation, other than containment isolation, which is described in Subsection 6.2.3.

The equipment and seismic classification are discussed in Section 3.2.

9.3.3.1.2 Power Generation Design Basis

9.3.3.1.2.1 Sampling During Normal Plant Operations

During normal operation, the primary sampling system collects representative samples of fluids in the reactor coolant system (RCS) and auxiliary primary systems process streams and the containment

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atmosphere for analysis, as listed in Table 9.3.3-1. Local sample points, as listed in Table 9.3.3-2, are provided at various process points of the systems.

The results of the sample analyses are used to perform the following functions:

- Monitor core reactivity
- Monitor fuel rod integrity
- Evaluate ion exchanger (demineralizer) and filter performance
- Specify chemical additions to the various systems;
- Maintain acceptable hydrogen levels in the reactor coolant system
- Detect radioactive material leakage

The measurements are used to evaluate water chemistry and to recommend corrective action by the laboratory staff.

The primary sampling system component classification is provided in Section 3.2.

9.3.3.1.2.2 Post-Accident Sampling

The primary sampling system does not include specific post-accident sampling capability. However, in accordance with Reference 5 there are contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. These plans include the procedures to analyze, during the later stages of accident response, reactor coolant for boron, containment atmosphere for hydrogen and fission products, and containment sump water for pH.

The primary means of containment atmosphere hydrogen analysis is the hydrogen analyzer described in Subsection 6.2.4, which is not part of the post-accident sampling capabilities.

9.3.3.2 System Description

The primary sampling system is a manually operated system. It collects representative samples of fluids from the reactor coolant system and various primary auxiliary system process streams for analysis by the plant operating staff. This sampling process is performed during normal plant operations.

The primary sampling system consists of two separate portions: the liquid sampling portion and the gas sampling portion.

9.3.3.2.1 Nuclear Sampling System - Liquids

The liquid sampling portion of the primary sampling system collects samples from the reactor coolant system and the auxiliary systems and transports them to a common location in a sample room in the auxiliary building. Control and instrumentation is provided for safe, reliable operation. This portion of the system uses 1/4 inch stainless steel tubing. The small tubing flow area limits flow to less than chemical and volume control system makeup capacity in the event of a leak in the sampling lines. Dissolved gases in the reactor coolant system are collected in this system also.

Sample flow is routed to a grab sampling unit. This unit is in an enclosure, which controls the spread of contamination and provides shielding. The grab sampling unit is further shielded by a concrete wall to minimize radiation exposure.

Valves inside the grab sampling unit have long handles extending outside the enclosure and are manually operated. This arrangement allows the operator to obtain a sample quickly with minimum

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radiation exposure. A schematic diagram is provided on the front of the grab sampling unit to illustrate the tube routing inside.

Since the motive force during normal operations is the system pressure, the sampling system is designed to reactor coolant system pressure. If system pressure is not available, an eductor supplies the motive force for sample collection.

A direct line from the grab sampling unit to the laboratory provides the capability for continuous liquid sampling and analysis with online monitors.

Prior to the collection of liquid samples either in the laboratory or in the grab sampling unit, the lines are purged with source liquid to provide representative samples. The purging flow returns to the effluent holdup tank of the liquid radwaste system.

Figure 9.3.3-1 is a simplified sketch of the primary sampling system.

9.3.3.2.2 Nuclear Sampling System - Gaseous

This portion of the primary sampling system collects gaseous samples from the containment atmosphere. Gaseous sampling is conducted in the sample room in the auxiliary building, and it shares with the liquid sampling portion the grab sampling unit and the control panel. Similar to the liquid sampling system, the gas sample subsystem is also manually operated with extension stems on the valves and uses stainless steel tubing. Only grab samples are collected for the gas sampling process. The lines are purged prior to sample collection to provide representative samples. The purged gas returns to the containment sump.

Provisions are also made to dilute the gas sample. The dilution process uses nitrogen from a local gas bottle.

The gas sampling system uses an ejector as the motive force for sample collection. The ejector uses nitrogen from a local gas bottle as the motive force.

9.3.3.3 Containment Isolation Valves

Containment isolation valves are classified as Safety Class B. The lines penetrating the reactor containment meet the containment isolation criteria. See Subsection 6.2.3.

Four lines penetrate the containment. Two lines carry the liquid samples from their sources to the grab sampling unit or the laboratory. The third line carries the containment air samples from their source to the sampling unit. The fourth line returns the liquid or containment air sampling flows to the containment sump. The valves fail closed.

These valves close on a containment isolation signal. In addition, the outside containment isolation valves in the liquid sampling paths close on a high sampling flow temperature, or high radiation or high pressure downstream of the sample coolers. This prevents the operator from working with high temperature fluid and minimizes the possibility of operator injury.

9.3.3.4 System Operation and Performance

The primary sampling system is manually operated. The tubing size and sampling flow rate are selected throughout the system to reduce the amount of purge flow and to provide turbulent sampling flow (to collect representative samples). A delay coil of tubing is installed inside containment to provide at least 60 seconds of transit time for the sampling fluid to exit the containment from the hot leg. This 60-second delay is needed for N-16 decay.

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9.3.3.5 Design Evaluation

The primary sampling system has no safety function, other than containment isolation and therefore requires no nuclear safety evaluation, other than containment isolation.

Subsection 6.2.3 provides the safety evaluation for the containment isolation system. Primary sampling system lines penetrating the containment are isolated at the containment boundary by valves that close upon receipt of a containment isolation signal and by manual actuation. (See Subsection 6.2.3 for a discussion of containment isolation.)

The primary sampling system connects to the reactor coolant system and the passive core cooling system (PXS) and therefore provides features consistent with ANSI standards and ASME codes to protect these system pressure boundaries.

The primary sampling system is not required for accident mitigation or post-accident sampling; but there are plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere in accordance with Reference 5.

The acceptability of the design of the primary sampling system is based on specific general design criteria and regulatory guides. The design of the primary sampling system is consistent with the criteria set forth in Subsection 9.3.2, "Process and Post-Accident Sampling Systems," of the NRC's Standard Review Plan (Reference 6) as modified by Reference 5. The specific general design criteria identified in the Standard Review Plan are General Design Criteria 1, 2, 13, 14, 26, 41, 60, 63, and 64. See Section 1.9 for a discussion of regulatory compliance.

9.3.3.6 Inspection and Testing Requirements

9.3.3.6.1 Preoperational Testing

Preoperational testing is performed after installation and prior to plant startup. Proper operation of the primary sampling system is demonstrated during preoperational testing. A sample is drawn from the reactor coolant system, containment atmosphere and other sample points via the sampling system in order to verify proper system operation.

9.3.3.6.2 Operational Testing

The proper operation and availability of the liquid and gaseous sampling subsystems are proven by continued proper sampling operations.

9.3.3.7 Instrumentation Requirements

The primary sampling system uses indicators as required to facilitate manual operation and to verify sample conditions before samples are drawn. Radiation monitoring instruments are used to monitor the incoming fluid (liquid or gas) radioactivity level.

The temperature and pressure indicators inside the grab sampling unit provide a signal to close the respective outside containment isolation valve when the sampling flow temperature or pressure exceeds pre-set limits. Likewise, the radiation monitors also provide a signal to close the outside containment isolation valves when excessive radiation levels are detected, for operator protection.

9.3.4 Secondary Sampling System

The secondary sampling system (SSS) delivers representative samples of fluids from secondary systems to sample analyzer packages. Continuous or semi-continuous online secondary chemistry

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monitoring detects impurity ingress and provides early diagnosis of system chemistry excursions in the plant. Secondary sampling monitors send control signals to the turbine island chemical feed system that automatically injects corrosion control chemicals into the condensate and feedwater systems.

9.3.4.1 Design Basis

9.3.4.1.1 Safety Design Basis

The secondary sampling system serves no safety-related function and therefore has no nuclear safety design basis.

9.3.4.1.2 Power Generation Design Basis

The secondary sampling system monitors water samples from the condensate, feedwater, main steam, heater drain, steam generator blowdown, auxiliary steam supply, and condensate polishing systems, as listed in Table 9.3.4-1 and Table 9.3.4-2. Water quality analyses are performed on these samples to determine the following:

- pH
- Conductivity levels (specific and cation)
- Dissolved oxygen level
- Residual oxygen scavenger
- Sodium content
- Sulfate content

The sample analyses are used to control water chemistry and to permit appropriate corrective action.

9.3.4.2 System Description

Classification of equipment and components for the secondary sampling system is given in Section 3.2. The sample points listed in Table 9.3.4-1 are continuously or semi-continuously monitored. The sample points listed in Table 9.3.4-2 are selectively monitored (where a single analyzer package can be used to selectively monitor multiple sample points).

Sample analysis data from the continuous and semi-continuous analyzers is recorded using computer systems that also provide trending capability of the measured process parameters. Measurements are used to automatically control condensate and feedwater system pH and dissolved oxygen levels by chemical addition. Refer to Subsection 10.4.11 for further discussion of the turbine island chemical feed system.

Samples are analyzed and the results are used for automatic or manual control of the plant secondary water chemistry. After being analyzed, pure samples are returned to the condensate system. Sample lines containing reagents and those from sink drains are collected in the waste water system and processed for disposal. Each sample line has a grab sampling capability for laboratory analysis.

Roughing coolers are provided for the samples whose temperatures exceed 125°F. Samples are cooled to approximately 77°F by chilled water supplied to trim coolers.

9.3.4.3 Safety Evaluation

The secondary sampling system has no safety-related function and therefore requires no nuclear safety evaluation.

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9.3.4.4 Tests and Inspections

Proper operation of the secondary sampling system is initially demonstrated during preoperational testing.

The system draws continuous, semi-continuous, and selective samples from the condensate, feedwater, main steam, and steam generator blowdown systems for automatic or manual water quality analysis. Calibration of the analyzers is checked periodically through laboratory analysis of a grab sample from the same process flow. The output of the continuous and semi-continuous analyzers is recorded, and abnormal values are evaluated.

9.3.4.5 Instrumentation Applications

The secondary sampling system uses pressure, temperature, and flow indicators to facilitate operation and to verify sample conditions.

9.3.5 Equipment and Floor Drainage Systems

The equipment and floor drainage systems collect liquid wastes from equipment and floor drains during normal operation, startup, shutdown, and refueling. The liquid wastes are then transferred to appropriate processing and disposal systems.

Equipment and floor drainage is segregated according to the type of waste. Liquid wastes are classified and segregated for collection as follows:

- Radioactive liquid waste
- Nonradioactive liquid waste
- Chemical and detergent liquid waste
- Oily liquid waste

9.3.5.1 Design Basis

9.3.5.1.1 Safety Design Basis

The equipment and floor drainage systems are nonsafety-related and serve no safety-related function except for the backflow preventers in drain lines from containment cavities to the containment sump. No nuclear safety design basis is required except for the backflow preventers described in Section 11.2. Single active failures do not prevent the proper function of the safety-related backflow preventers.

The floor drainage systems and equipment are designed to prevent damage to safety-related systems, structures, and equipment. Safety-related components are not damaged as a result of equipment and floor drain components failure from a seismic event. Floor drainage systems and equipment single failures will not prevent the proper function of any safety-related equipment.

9.3.5.1.2 Power Generation Design Basis

Nonradioactive liquid waste sumps and drain tanks that can be potentially radioactive during normal plant operation are provided with sampling capabilities. There are no permanent connections between the radioactive drain system and nonradioactive piping. Provisions for temporary diversion of contaminated water from normally nonradioactive drains to the liquid radwaste system are included.

Equipment drains are adequately sized to meet the flow requirements.

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Radioactive sump vents are directed to the ventilation system exhaust ducts, serving the areas where the sump is located. The containment sump vents directly to the containment.

Drainage systems are drained by gravity. The slope of the drain lines is 1/8 inch per foot as a minimum except for the embedded drain piping for areas 1 and 2 of the auxiliary building, elevation 66'-6". At this level, the slope of the drain lines is 1/16 inch per foot minimum. Exceptions are also provided for through-wall penetrations, which can remain horizontal provided that lines are sloped in the direction of drainage to and from the wall penetration. A short section of piping past each end of the wall penetration may be horizontal to accommodate piping fabrication limitations to transition from sloped and horizontal sections. Radioactive drain systems shall have no negative sloping sections except for WRS lines between the annex and auxiliary buildings provided it does not affect system capability to drain as intended, does not affect radiation zoning, and does not increase radiation exposure to plant personnel above the evaluated dose. Nonradioactive drain systems may contain short sections sloping less than 1/8 inch per foot provided it does not affect system capability to drain as intended. The drainage systems are designed not to compromise the integrity of the areas maintained under a slight negative pressure during normal plant operation. This is achieved by avoiding cross connection with adjacent areas that are not maintained under a slight negative pressure.

Radioactive drain systems are designed to avoid crud traps and to minimize drain traps.

Sump and drain tank pumps discharge at a flowrate adequate to prevent sump overflow for drain rates anticipated during normal plant operation, maintenance, decontamination, fire suppression system testing, and fire fighting activities. The containment sump pumping time between high and low level is approximately 1 minute. The auxiliary building radiologically controlled area sump has a pumping time of approximately 8 minutes between high and low level. The waste water system sumps provide a live storage capacity consistent with a quantity of fluid equal to 10 minutes generated from normal anticipated fluid flow.

Plugging of the drain headers is minimized by designing them large enough to accommodate more than the design flow and by making the flow path as straight as possible. Drain headers are at least 4 inches in diameter.

9.3.5.2 System Description

9.3.5.2.1 General Description

The drainage systems include collection piping, equipment drains, floor drains, vents, traps, cleanouts, sampling connections, valves, collection sumps, drain tanks, pumps, and discharge piping. The general arrangement of the drainage systems is shown on Figure 9.3.5-1.

Radioactive Wastes

The radioactive waste drain system is arranged to receive inputs from the radiologically controlled areas of the auxiliary, annex, and radwaste buildings based on segregation of the liquid wastes into chemical and nonchemical drains. The radioactive waste drain system collects radioactive liquid wastes at atmospheric pressure from equipment and floor drainage of the radioactive portions of the auxiliary building, annex building, and radwaste building and directs these wastes to a centrally located sump located in the auxiliary building. The contents of the sump are pumped to the liquid radwaste system tanks. Drainage lines from the negative pressure boundary areas of the auxiliary, radwaste, and annex buildings do not terminate outside the negative pressure boundary without a normally closed valve or plugged drain to maintain the integrity of the negative pressure boundary.

The liquid radwaste system collects radioactive and borated liquid wastes from equipment and floor drains in containment. Waste from the equipment drains inside containment is drained to the reactor

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coolant drain tank. The liquid waste from floor drains, fan coolers, and the containment wall gutter inside containment is drained to the containment sump. The contents of the drain tank and sump are pumped out of containment for processing by the liquid radwaste system. Refer to Section 11.2 for further details.

The sumps, pumps, and associated valves for the drain systems are located outside of high-radiation areas to the extent practical.

Nonradioactive and Potentially Radioactive Waste Drains

The waste water system collects nonradioactive waste from floor and equipment drains in auxiliary, annex, turbine, and diesel generator building sumps or tanks. Selected normally nonradioactive liquid waste sumps and tanks are monitored for radioactivity to determine whether the liquid wastes have been inadvertently contaminated. If contaminated, the wastes are diverted to the liquid radwaste system for processing and ultimate disposal. Refer to Subsection 9.2.9 for further details. Drainage lines from the positive pressure boundary areas of the auxiliary building do not terminate outside the positive pressure boundary without a closed valve, plugged drain, or water seal to maintain the integrity of the positive pressure boundary.

Chemical Waste Drains

The radioactive waste drain system collects chemical wastes from the auxiliary building chemical laboratory and decontamination solution drains from the annex building and directs these wastes to the chemical waste tank of the liquid radwaste system.

Detergent Waste Drains

The laundry and respirator cleaning functions that generate detergent wastes are performed offsite. Detergent wastes from hot sinks and showers are routed to the chemical waste tank.

Oily Waste Drains

The waste water system collects nonradioactive, oily, liquid waste in drain tanks and sumps. Drain tank and sump liquid wastes are pumped through an oil separator prior to further processing. The oil is collected in a tank for disposal.

Sampling for oil in the waste holdup tank of the liquid radwaste system is provided to detect oil contamination before the ion exchanger resins are damaged. Oily water is pumped from the tank through an oil adsorbing bag filter before further processing. The spent bag filters are transferred to drums and stored in the radwaste building as described in Section 11.4.

9.3.5.2.2 Component Description

General description and summaries of the design requirements for these components are provided below. Key equipment parameters are contained in Tables 9.3.5-1 and 11.2-2. Principal construction codes and standards and the classification applicable to the floor and equipment drainage systems are listed in Section 3.2.

Sumps and Drain Tanks

In general, the inlet drain lines to the sump or drain tank are kept submerged a minimum of 6 inches below pump shutoff level to prevent backgassing. The containment sump inlet is submerged.

Sumps are covered to keep out debris. Covers are removable, or manholes are provided for access. The total capacity of each sump includes a 10 percent freeboard allowance to permit operation of high-high level alarms and associated controls before the overflow point is reached.

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Each sump is fitted with a vent connection to exhaust potential sump gases. Nonradioactive drain tanks are vented to the atmosphere. The reactor coolant drain tank is vented to the gaseous radwaste system (Section 11.3). Where necessary for the control of airborne radioactivity, the sump vents are routed to the ventilation system exhaust duct for the room.

Radioactive sumps are stainless steel construction. Nonradioactive collection sumps are constructed of concrete with corrosion resistant coating or liner.

Sump and Drain Tank Pumps

Sumps outside containment, except for the Turbine Building sumps, are provided with air diaphragm pumps mounted on the sump cover plate. Pumps are equipped with reliable, mechanical diaphragms of demonstrated acceptable design that are easy to maintain. Pumps and associated piping connections and accessories are designed for easy replacement of pump diaphragms. The containment sump pumps are described in Section 11.2. The turbine building sump pumps are described in Subsection 9.2.9.

Valves

Solenoid valves are provided for on/off functions of air supply to the sump pump diaphragms of the radioactive waste drain system and waste water system. Swing check valves, where provided, are installed in horizontal pipe runs. Installation of swing check valves in vertical pipe runs is acceptable provided the flow is up, and the valve is designed such that the center of gravity of the disc causes the disc to close upon loss of flow. Pressure control valves are provided to control air supply pressure to the sump pump diaphragms. Manual ball valves are provided for maintenance purposes.

9.3.5.2.3 System Operation

The equipment and floor drainage systems operate during all modes of normal plant operation. Liquid wastes drain by gravity to collection tanks or sumps. Drainage flowrates vary based on the status of the plant. Sump pumps disposing of collected radioactive wastes discharge to the liquid radwaste system for further processing. Nonradioactive liquid wastes are discharged to the waste water system.

Pump operation is automatic with manual override. The pumps are automatically started and stopped by preset high, high-high, and low level instrumentation.

Where sumps are provided with two pumps, the capability is provided to allow equalizing the operational period of each pump. For the radioactive waste drain system, recirculation of the sump contents is performed through a mixing eductor.

The sump and drain tank pumps are not required to operate during design basis accidents. Sump pumps in the containment are interlocked with the associated containment isolation valves. The pumps trip and the isolation valves close on receipt of containment isolation signals (see Subsection 6.2.3).

The equipment and floor drainage systems can be operated either automatically or manually for cleanup following an accident, including fire, provided that the compressed and instrument air system and ac power are available, and the drainage systems and support systems are not disabled by the event.

9.3.5.2.4 Instrumentation Applications

Level indication is provided in the main control room for the sump in-containment to provide indication of the presence of reactor coolant from unidentified leaks (refer to Subsection 5.2.5). The

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sump and the drain tank outside containment are monitored for water level. On high sump or tank level, the solenoid-operated three-way valve for the selected pump is energized to admit air to the pump diaphragm. On high-high sump or tank level, the solenoid-operated three-way valve for the remaining pump is also energized to admit air to that pump diaphragm. On low level, both pumps are stopped by deenergizing their respective solenoid valves. Operating status of the pumps is provided to the plant control system.

9.3.5.3 Safety Evaluation

The equipment and floor drainage systems are nonsafety-related except for backflow preventers in drain lines from containment cavities to the containment sump. No nuclear safety evaluation is required other than that described for the backflow preventers in Section 11.2.

9.3.5.4 Tests and Inspections

The operability of equipment and floor drainage systems dependent upon gravity flow can be checked by normal usage. Portions of these systems dependent upon pumps to discharge to interfacing systems may be checked through instrumentation and alarms via the plant control system and trouble alarms in the main control room during operation or test.

9.3.6 Chemical and Volume Control System

The chemical and volume control system is designed to perform the following major functions:

- **Purification** - maintain reactor coolant system fluid purity and activity level within acceptable limits.
- **Reactor coolant system inventory control and makeup** - maintain the required coolant inventory in the reactor coolant system; maintain the programmed pressurizer water level during normal plant operations.
- **Chemical shim and chemical control** - maintain the reactor coolant chemistry conditions by controlling the concentration of boron in the coolant for plant startups, normal dilution to compensate for fuel depletion and shutdown boration, and provide the means for controlling the reactor coolant system pH by maintaining the proper level of lithium hydroxide.
- **Oxygen control** - provide the means for maintaining the proper level of dissolved hydrogen in the reactor coolant during power operation and for achieving the proper oxygen level prior to startup after each shutdown.
- **Filling and pressure testing the reactor coolant system** - provide the means for filling and pressure testing the reactor coolant system. The chemical and volume control system does not perform hydrostatic testing of the reactor coolant system, which is only required prior to initial startup and after major, nonroutine maintenance, but provides connections for a temporary hydrostatic test pump.
- **Borated makeup to auxiliary equipment** - provide makeup water to the primary side systems that require borated reactor grade water.
- **Pressurizer Auxiliary Spray** - provide pressurizer auxiliary spray water for depressurization.

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9.3.6.1 Design Bases

9.3.6.1.1 Safety Design Basis

The safety functions provided by the chemical and volume control system are limited to containment isolation of chemical and volume control system lines penetrating containment, termination of inadvertent reactor coolant system boron dilution, isolation of makeup on a steam generator High-3 or pressurizer High-2 level signal, and preservation of the reactor coolant system pressure boundary, including isolation of normal chemical and volume control system letdown from the reactor coolant system.

9.3.6.1.2 Power Generation Design Basis

The principal functions of the chemical and volume control system are outlined above and include controlling reactor coolant system chemistry, purity, and inventory. The system provides some functions necessary for the continued normal operation of the plant. Reliability is achieved by the use of redundant equipment (pumps, filters, and demineralizers). The equipment classification for the chemical and volume control system is contained in Section 3.2.

9.3.6.1.2.1 Purification

The chemical and volume control system removes radioactive corrosion products, ionic fission products, and fission gases from the reactor coolant system to maintain low reactor coolant system activity levels. The chemical and volume control system purification capability considers occupational radiation exposure (ORE) to support ALARA goals.

The chemical and volume control system is designed to maintain the reactor coolant system activity level at less than the technical specification limit for normal operations, with design basis fuel defects. The technical specifications allow these limits to be exceeded for a specified duration.

The purification rate is based on minimizing occupational radiation exposure and providing access to the reactor coolant system equipment. The chemical and volume control system provides a reactor coolant system purification rate of at least one reactor coolant system mass per 16 hours.

The chemical and volume control system has sufficient reactor coolant system purification and degasification capability (in conjunction with the liquid radwaste system) to allow the reactor vessel head to be removed in a timely manner during a refueling shutdown. In addition, purification during shutdowns has positive impact on the occupational radiation exposure to workers during the outage. The chemical and volume control system supports the plant ALARA goals with the shutdown purification function.

9.3.6.1.2.2 Reactor Coolant System Inventory Control and Makeup

The chemical and volume control system provides a means to add and remove mass from the reactor coolant system, as required, to maintain the programmed inventory during normal plant operations. Operations that are accommodated include startup, shutdown, step load changes, and ramp load changes.

The chemical and volume control system is capable of maintaining a constant volume in the reactor coolant system while the plant is being heated up or cooled down. During a heatup it is necessary to remove reactor coolant system mass due to expansion. The maximum rate of net expansion occurs at the end of the heatup, so the limiting case is based on controlling the pressurizer level during this phase of operation. This expansion is accommodated by the normal letdown path. During cooldown, it is necessary to add mass due to reactor coolant system shrinkage. The chemical and volume

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control system is capable of maintaining the minimum pressurizer level with makeup during cooldown from hot zero power to cold shutdown while maintaining normal purification flow. Ramp and step load changes, as well as load rejections, are accommodated by the reactor coolant system pressurizer level control system. The chemical and volume control system can function to accommodate normal pressurizer level control system makeup and letdown requirements.

The chemical and volume control system is designed to make up for leaks, including leaks up to 3/8-inch inside diameter and for anticipated steam generator tube leaks, allowing the plant to be taken to cold shutdown conditions without the use of safety-related makeup systems.

9.3.6.1.2.3 Chemical Shim and Chemical Control

The chemical and volume control system provides the means to vary the boron concentration in the reactor coolant system. The system also controls the reactor coolant system chemistry for the purpose of limiting corrosion and enhancing core heat transfer.

Chemical Shim

The concentration of boron in the reactor coolant system is changed, as required, to maintain the desired control rod position with core depletion. The chemical and volume control system has the capacity to accommodate a cold shutdown followed by a return to power at the end of core life and also (as an independent case) to borate the plant to cold shutdown immediately following return to power from refueling. The system has boration and dilution capacity to meet these requirements, as well as the capability to transfer effluents to other systems.

The chemical and volume control system boric acid solutions are stored at concentrations that do not require heat tracing or room temperatures above normal values. The 2.5 weight percent boric acid solution requires freeze protection but does not impose special ambient temperature requirements.

pH Control

Lithium hydroxide (Li7OH) is used to control the pH of the reactor coolant system. The required concentration of Li7OH is varied to minimize the formation of tritium.

9.3.6.1.2.4 Oxygen Control

The chemical and volume control system maintains the proper conditions in the reactor coolant system to minimize corrosion of the fuel and primary surfaces. During power operations, dissolved hydrogen is added to the reactor coolant system to eliminate free oxygen and to prevent ammonia formation. The chemical and volume control system is capable of maintaining the concentration of dissolved hydrogen in the reactor coolant system at a minimum of 25 cubic centimeters hydrogen, at standard temperature and pressure, per kilogram of coolant, assuming anticipated operating losses.

This concentration can be reduced to 15 cc/kg within 24 hours prior to shutdown. Prior to opening the reactor coolant system during a cold or refueling shutdown, the hydrogen concentration is reduced to approximately 5 cubic centimeters per kilogram. To prevent delays, the chemical and volume control system (in conjunction with the liquid radwaste system) is capable of making this 15 to 5 cubic centimeters per kilogram reduction within the time to achieve normal plant cooldown.

During plant startup from cold shutdown, the chemical and volume control system introduces an oxygen scavenger into the reactor coolant system. The solution is only used for oxygen control at low reactor coolant system temperatures during startup from cold shutdown conditions. At other times during plant operation, hydrogen is used for oxygen control.

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During plant startup, the chemical and volume control system will be in demineralizer bypass mode; during which time, valve CVS-PL-V062 will be “open” allowing flow to be diverted through line CVS-PL-L065 instead of through the demineralizer to avoid any negative impacts on demineralizer resin performance by contact with hydrazine.

9.3.6.1.2.5 Filling and Pressure Testing the Reactor Coolant System

The chemical and volume control system provides a means for filling and pressure testing the reactor coolant system. The chemical and volume control system also provides connections for a temporary hydrostatic test pump.

9.3.6.1.2.6 Borated Makeup

The chemical and volume control system provides makeup to the passive core cooling system accumulators, core makeup tanks, in-containment refueling water storage tank, and to the spent fuel pool at various boron concentrations.

9.3.6.2 System Description

The chemical and volume control system consists of regenerative and letdown heat exchangers, demineralizers and filters, makeup pumps, tanks, and associated valves, piping, and instrumentation. The system parameters are given in Table 9.3.6-1. The piping and instrumentation diagram for the chemical and volume control system is included as Figure 9.3.6-1.

9.3.6.2.1 Purification

9.3.6.2.1.1 Ionic Purification

The normal chemical and volume control system purification loop is inside containment and operates at reactor coolant system pressure, utilizing the developed head of the reactor coolant pumps as the motive force for the purification flow. During power operations, fluid is continuously circulated through the chemical and volume control system from the discharge of one of the reactor coolant pumps. It passes through the regenerative heat exchanger where it is cooled by the returning chemical and volume control system flow, and is further cooled by component cooling water in the letdown heat exchanger to a temperature compatible with the demineralizer resins. The purification fluid flows through a mixed bed demineralizer, optionally through a cation bed demineralizer, and through a filter. It returns to the suction of a reactor coolant pump after being heated in the regenerative heat exchanger. The purification loop operates at reactor coolant system pressure.

Since the motive force for the purification loop is the reactor coolant pump head in a closed loop with the reactor coolant system, continuous purification is provided without operating the chemical and volume control system makeup pumps.

The mixed bed demineralizers are provided in the purification loop to remove ionic corrosion products and certain ionic fission products; they also remove zinc during periods of zinc addition. The demineralizers also act as filters. One mixed bed is normally in service, with a second demineralizer acting as backup in case the normal unit should become exhausted during operation. Each demineralizer and filter is sized to provide a minimum of one fuel cycle of service without changeout.

The mixed bed demineralizer in service can be supplemented by intermittent use of the cation bed demineralizer for additional purification in the event of fuel defects. In this case, the cation resin removes mostly lithium and cesium isotopes. The cation bed demineralizer has sufficient capacity to maintain the cesium-136 concentration in the reactor coolant below 1.0 microcurie per cubic centimeter with design basis fuel defects. Each mixed bed and the cation bed demineralizer is sized

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to accept the maximum purification flow. Filters are provided downstream of the demineralizers to collect particulates and resin fines.

During plant shutdowns when the reactor coolant pumps are stopped, the normal residual heat removal system provides the motive force for the chemical and volume control system purification. Purification flow from the normal residual heat removal system heat exchanger is routed directly through the normal chemical and volume control system purification loop. Boron changes and dissolved gas control are still possible by operating the chemical and volume control system in a semiclosed loop arrangement.

9.3.6.2.1.2 Gaseous Purification

Removal of radiogases from the reactor coolant system are not normally necessary because the gases do not build up to unacceptable levels when fuel defects are within normally anticipated ranges. If radiogas removal is required because of high fuel defects, the chemical and volume control system can be operated by routing flow to the liquid radwaste system degassifier. In this configuration, the letdown fluid is depressurized by flowing through the letdown orifice. The letdown flow is routed outside of containment through the liquid radwaste system degassifier to one of the liquid radwaste system effluent holdup tanks, and then returned to the reactor coolant system with the chemical and volume control system makeup pumps. This provides efficient gas removal.

Removal of radioactive gas and hydrogen during shutdown operations is necessary to avoid extending the maintenance and refueling outages. The reactor coolant system pressure boundary cannot be opened to the containment atmosphere until the gas concentrations are reduced to low levels. The shutdown degassing process is accomplished by operating the chemical and volume control system in the open loop configuration. In addition, a line is provided to allow the letdown orifice to be manually bypassed, so gas removal can continue after the reactor coolant system has been depressurized.

9.3.6.2.2 Reactor Coolant System Inventory Control and Makeup

Changes in reactor coolant volume are accommodated by the pressurizer level program for normal power changes, including transition from hot standby to full-power operation and returning to hot standby. In addition, the pressurizer has sufficient volume, within the deadband of the level control program, to accommodate minor reactor coolant system leakage for some time. The chemical and volume control system provides inventory control to accommodate minor leakage from the reactor coolant system, expansion during heatup from cold shutdown, and contraction during cooldown. This inventory control is provided by letdown and makeup connections to the chemical and volume control system purification loop.

9.3.6.2.3 Chemical Shim and Chemical Control

The chemical and volume control system provides the following functions to support the water chemistry and chemical shim requirements of the reactor coolant system:

- Means of addition and removal of pH control chemicals for startup and normal operation.
- Means of addition and removal of soluble chemical neutron absorber (boron) and makeup water, at concentrations and rates compatible with normal plant operation.

Reactor coolant system chemistry changes are accomplished with a feed and bleed operation. The letdown and makeup paths are operated simultaneously and appropriate chemicals are provided at the suction of the reactor makeup pumps.

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9.3.6.2.3.1 Chemical Shim

Reactor coolant system boron changes are required to compensate for fuel depletion, startups, shutdowns, and refueling.

To borate the reactor coolant system, the operator sets the reactor makeup control system to automatically add a preset amount of boric acid by fully diverting the three-way valve in the pump suction line to the boric acid storage tank, with delivered flow measured at the discharge of the makeup pumps. Dilution operates in a similar fashion. In either case, if the pressurizer level exceeds its control point, the letdown path to the liquid radwaste system holdup tanks is automatically opened.

Boric acid is provided to the boric acid storage tank by mixing 2.5 weight percent boric acid solution in the boric acid batching tank. Boric acid crystals are mixed with a mixer, while the mixture is heated to an appropriate temperature to provide efficient mixing by the batching tank immersion heater. After the boric acid crystals are dissolved, the solution is drained by gravity into the boric acid storage tank. No provisions are incorporated for boric acid recycle from the liquid radwaste system.

9.3.6.2.3.2 pH Control

The chemical agent used for pH control is lithium hydroxide (Li7OH). This chemical is chosen for its compatibility with the material and water chemistry of borated water, stainless steel, and zirconium systems. In addition, lithium-7 is produced in the core region because of irradiation of the dissolved boron in the coolant. A chemical mixing tank is provided to introduce the solution to the suction of the makeup pumps as required to maintain the proper concentration of Li7OH in the reactor coolant system.

The solution is poured into the chemical mixing tank and is then flushed to the suction manifold of the makeup pumps with demineralized water. A flow orifice is provided on the demineralized water inlet pipe to allow chemicals to be flushed into the reactor coolant system at acceptable concentrations.

The concentration of lithium-7 in the reactor coolant system varies according to a pH control curve as a function of the boric acid concentration of the reactor coolant system. If the concentration exceeds the proper value, as it may during the early stages of core life when lithium-7 is produced in the core at a relatively high rate, the cation bed demineralizer is used in the letdown path in series with the mixed bed demineralizer to lower the lithium-7 concentration. Since the buildup of lithium is slow, the cation bed demineralizer is used only intermittently. When letdown is being diverted to the liquid radwaste system, the purification flow is routed through the cation bed demineralizer for removal of as much lithium-7 and cesium as possible.

9.3.6.2.3.3 Zinc Addition

A soluble zinc compound may be added to the coolant as a means to reduce radiation fields within the primary system and to reduce the potential for crud-induced power shift (CIPS). The zinc used may be either natural zinc or zinc depleted of ^{64}Zn.

9.3.6.2.4 Oxygen Control

The chemical and volume control system provides control of the reactor coolant system oxygen concentration, both during startup by introducing an oxygen scavenger and during power operations by driving toward zero the equilibrium concentration of oxygen produced by radiolysis in the core by injecting hydrogen.

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9.3.6.2.4.1 Startups

During plant startup from cold conditions, an oxygen scavenging agent is used. The oxygen scavenger solution is introduced into the reactor coolant system via the makeup flow and chemical mixing tank, in the same manner as described for lithium-7 addition. The oxygen scavenger is used for oxygen control only at startup from cold shutdown conditions.

During plant startup, the chemical and volume control system will be in demineralizer bypass mode; during which time, valve CVS-PL-V062 will be “open” allowing flow to be diverted through line CVS-PL-L065 instead of through the demineralizer to avoid any negative impacts on demineralizer resin performance by contact with hydrazine.

9.3.6.2.4.2 Power Operation

Dissolved hydrogen is employed during normal power operation to control and scavenge oxygen produced due to radiolysis of water in the core region. Hydrogen makeup is supplied to the reactor coolant system by direct injection of high-pressure gaseous hydrogen. The hydrogen comes from a bottle outside containment, through a containment penetration, and is mixed in the chemical and volume control system purification loop. Hydrogen removal from the reactor coolant system is not necessary because hydrogen is consumed in the core.

9.3.6.2.5 Reactor Coolant System Filling and Pressure Testing

Reactor coolant system filling is accomplished by using the chemical and volume control system makeup pumps to provide fluid at the proper boron concentration (refueling), taking suction from sources that are at or above the minimum shutdown margin boron concentration. If the reactor coolant system (RCS) is drained to a clean WLS effluent holdup tank, this drained volume can be returned by opening the line to the makeup pumps from that holdup tank and running one of the pumps.

The chemical and volume control system makeup pumps produce sufficient head to pressure test the reactor coolant system after maintenance and refueling.

A temporary hydrotest pump is required for initial hydrotesting, which requires higher pressures than can be achieved with the makeup pumps.

9.3.6.2.6 Borated Makeup

The makeup pumps are used to provide makeup at the proper boron concentration to the passive core cooling system accumulators, core makeup tanks, in-containment refueling water storage tank, and to the spent fuel pool. Makeup to these locations is at boric acid concentration as required, which can be varied from 0 to 4375 (nominal) parts per million (2.5 weight percent). A mixture of 2.5 weight percent boric acid and demineralized water is provided by taking suction from both the boric acid storage tank and the demineralized water tank.

9.3.6.3 Component Descriptions

The general descriptions and summaries of the chemical and volume control system components are provided below. The key equipment parameters for the chemical and volume control system components are contained in Table 9.3.6-2. Information regarding component classifications is available in Section 3.2. See Section 5.2 for additional information on analysis requirements.

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9.3.6.3.1 Chemical and Volume Control System Makeup Pumps

Two centrifugal makeup pumps are provided. These pumps are driven by ac motors, and flow is controlled by positioning a control valve in the common discharge line from the pumps. A cavitating venturi in the common discharge line limits the makeup flow and provides protection from excessive pump runout. Each pump has a recirculation loop with a heat exchanger and flow control orifice to provide adequate minimum flow for pump protection. The mini-flow heat exchanger is cooled by component cooling water.

The makeup pumps are arranged in parallel with common suction and discharge headers. Each provides full capability for normal makeup; thus, there is redundancy for normal operations. The normal makeup pump suction fluid comes from the boric acid storage tank and the demineralized water connection. A three-way valve in the suction header is positioned to provide a full range of concentrations.

One makeup pump is capable of maintaining normal reactor coolant system inventory with leaks up to a 3/8-inch inside diameter, without an actuation of the safety injection systems. The second pump can be manually started to provide additional reactor coolant makeup.

These pumps are used to pressure test the reactor coolant system.

Parts of the pump in contact with reactor coolant are constructed of austenitic stainless steel. The pump motor and lube oil are air-cooled.

9.3.6.3.2 Chemical and Volume Control System Heat Exchangers

Letdown Heat Exchanger

One single-shell pass U-tube letdown heat exchanger is provided. The heat exchanger is designed to cool the purification loop flow from the regenerative heat exchanger outlet temperature to the desired letdown temperature allowing the letdown to be processed by the demineralizers while maximizing the thermal efficiency of the chemical and volume control system.

The letdown heat exchanger outlet temperature is controlled by the operator by remotely positioning a component cooling system flow control valve.

The reactor coolant in the purification loop flows through the tubes, which are stainless steel, and component cooling water flows through the shell, which is carbon or stainless steel.

Makeup Pump Miniflow Heat Exchangers

Two miniflow heat exchangers are provided, one in each makeup pump miniflow recirculation line. Each heat exchanger is designed to cool the flow through the chemical and volume control system makeup pump minimum flow recirculation lines to the desired temperature for pump protection. The makeup water flows through the tubes, which are stainless steel, and component cooling water flows through the shell, which is carbon or stainless steel.

Regenerative Heat Exchanger

One regenerative heat exchanger is provided. This heat exchanger is used to recover heat from the purification loop flow leaving the reactor coolant system by reheating the fluid entering the reactor coolant system. This provides increased thermal efficiency and also reduces thermal stresses on the reactor coolant system.

The design basis for this heat exchanger is the last hour of plant heatup, when expansion of the reactor coolant system requires a net removal of inventory. For this case the regenerative heat

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exchanger outlet temperature must be low enough to allow the letdown heat exchanger to cool the letdown to the desired temperature with anticipated cooling water temperatures.

The reactor coolant leaving the reactor coolant system flows through the tube side of this heat exchanger, and the returning fluid flows through the shell. This arrangement places the cleaner fluid on the shell side and the lower quality fluid on the tube side, where there are fewer crevices available for crud deposition.

9.3.6.3.3 Chemical and Volume Control System Tanks

Boric Acid Storage Tank

One boric acid storage tank is provided. The tank is sized to allow for one shutdown to cold shutdown followed by a shutdown for refueling at the end of the fuel cycle.

The tank is vented to the atmosphere. Relatively little boric acid is used during power operation, since load follow is accomplished with gray rods and without changes in the reactor coolant system boron concentration. Therefore, the boric acid which is injected has a negligible effect on the free oxygen level in the reactor coolant system.

The tank is a free-standing stainless steel cylindrical design, located outside of the buildings, with only normal freeze protection required to maintain solubility of the 2.5 weight percent boric acid.

Recirculation of the tank contents can also be performed to maintain a homogenous mixture of boric acid in the tank.

Boric Acid Batching Tank

The boric acid batching tank is a cylindrical tank with an immersion heater used in the preparation of 2.5 weight percent boric acid. A mixer is included with the tank. The tank is constructed of austenitic stainless steel and is provided with fill, vent and drain connections.

Chemical Mixing Tank

The chemical mixing tank is a small vertical, cylindrical tank sized to provide sufficient capacity for injecting an oxygen scavenger solution necessary to provide a concentration of ten parts per million in the cold reactor coolant system for oxygen scavenging.

A variety of chemicals to be added to the primary system are mixed in the tank. The solution to be injected is placed into the mixing tank and then flushed to the suction of the makeup pumps with demineralized water.

The tank is constructed of austenitic stainless steel and is provided with fill, vent, and drain connections.

9.3.6.3.4 Chemical and Volume Control System Demineralizers

Cation Bed Demineralizer

One cation resin bed demineralizer is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of lithium-7 (pH control) in the reactor coolant system. The demineralizer is sized to accommodate maximum purification flow when in service, which is adequate to control the lithium-7 and/or cesium concentration in the reactor coolant.

The demineralizer vessel is designed for reactor coolant system pressure and is constructed of austenitic stainless steel, with connections for resin addition, replacement, flushing, and draining.

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The vessel incorporates a retention screen and mesh screens on the drain connections. The screens are designed to retain the resin with minimum pressure drop. A flow diverter deflects the incoming flow to preserve a smooth resin bed.

Mixed Bed Demineralizers

Two mixed bed demineralizers are provided in the purification loop to maintain reactor coolant purity. Cation or anion resin, or a mixture of these resins, is used in each demineralizer. Both forms of resin remove fission and corrosion products. Each demineralizer is sized to accept the full purification flow during normal plant operation and to have a minimum design life of one core cycle.

The construction of the mixed bed demineralizers is identical to that of the cation bed demineralizer.

9.3.6.3.5 Chemical and Volume Control System Filters

Makeup Filter

One makeup filter is provided to collect particulates in the makeup stream, such as boric acid storage tank sediment. The filter is designed to accept maximum makeup flow. The makeup filter unit is constructed of austenitic stainless steel with a disposable synthetic cartridge and designed for the makeup pump design pressure listed in Table 9.3.6-2.

Reactor Coolant Filters

Two reactor coolant filters are provided. The filters are designed to collect resin fines and particulate matter from the purification stream. Each filter is designed to accept maximum purification flow.

The units are constructed of austenitic stainless steel with disposable synthetic cartridges. The design parameters for the filter are shown in Table 9.3.6-2.

9.3.6.3.6 Chemical and Volume Control System Letdown Orifice

One letdown orifice is provided in the letdown line, where fluid leaves the high-pressure purification loop before it exits containment. The orifice limits the letdown flow to a rate compatible with the chemical and volume control system equipment and also plant heatup and dilution requirements.

The orifice consists of an assembly that provides for permanent pressure loss without recovery and is made of austenitic stainless steel.

A manual bypass line is provided around the orifice to allow shutdown purification and degassing when the reactor coolant system pressure is low.

9.3.6.3.7 Chemical and Volume Control System Valves

The chemical and volume control system valves are stainless steel for compatibility with the borated reactor coolant. Isolation valves are provided at connections to the reactor coolant system. Lines penetrating the reactor containment meet the containment isolation criteria described in Subsection 6.2.3.

Purification Stop Valves

These normally open, motor-operated valves are located inside containment and close automatically on a low pressurizer level signal from the protection and safety monitoring system to preserve reactor coolant pressure boundary and to prevent uncovering of the heater elements in the pressurizer. The valves fail "as is" on loss of power and manual control (open/auto/close) is provided in the main control room and at the remote shutdown workstation.

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Letdown Flow Inside Containment Isolation Valve

This normally closed, fail closed, air-operated globe valve is located inside containment and isolates letdown to the liquid radwaste system. This valve automatically opens and closes on a plant control system signal from the pressurizer level control or a containment isolation signal from the protection and safety monitoring system. It automatically opens on high pressurizer level and closes when the pressurizer level returns to normal. It also closes on a high-3 liquid radwaste system degassifier level or a containment isolation signal. This valve operator has a flow restricting orifice in the vent line so it closes more slowly than the letdown flow outside containment isolation valve. Manual control is also provided in the main control room and at the remote shutdown workstation.

Letdown Flow Outside Containment Isolation Valve

This normally closed, fail closed, air-operated globe valve is located outside containment and isolates letdown to the liquid radwaste system. This valve automatically opens and closes on a plant control system signal from the pressurizer level control system or automatically closes upon receipt of a containment isolation signal from the protection and safety monitoring system. This valve operates similar to the letdown flow inside containment isolation valve. The valve throttles during operations when there is letdown without makeup to provide sufficient flow to the shell side of the regenerative heat exchanger. The letdown flow outside containment isolation valve closes more quickly than inside containment letdown flow isolation valve to limit seat wear of inside containment isolation valve. This valve operator has a flow restricting orifice in the air line, so it opens more slowly than inside containment letdown flow isolation valve. In addition, during brief periods of shutdown, when the reactor coolant system is water solid, this valve throttles to maintain the reactor coolant system pressure. Manual control is also provided in the main control room and at the remote shutdown workstation.

Makeup Stop Valve

This normally open, air-operated stop check valve is located inside containment and functions to isolate the flow in the charging line to the reactor coolant system. This valve can be closed from the main control room or the remote shutdown workstation to isolate charging downstream of the regenerative heat exchanger. This valve is closed to support the auxiliary spray function. The valve fails open on loss of power or loss of instrument air so the charging line to the reactor coolant system remains available.

Auxiliary Spray Line Isolation Valve

This normally closed, air-operated globe valve is located inside containment, downstream of the regenerative heat exchanger, and functions to isolate the auxiliary spray line to the reactor coolant system pressurizer. This valve is opened to provide flow to the auxiliary spray line during heatups and cooldowns to add chemicals or to collapse the steam bubble in the pressurizer. This valve fails closed on a loss of power or loss of instrument air to accomplish the function of preserving the reactor coolant pressure boundary. This valve closes automatically on a low-1 pressurizer level signal from the protection and safety monitoring system to preserve reactor coolant pressure boundary. This valve is operated from the main control room and the remote shutdown workstation.

Makeup Line Containment Isolation Valves

These normally open, motor-operated gate valves provide containment isolation of the chemical and volume control system makeup line and automatically close on a high-2 pressurizer level, High-3 steam generator level, or high-2 containment radiation signal from the protection and safety monitoring system. The valves close on a source range flux doubling signal to terminate possible unplanned boron dilution events. The valves also close on a safeguards actuation signal coincident with high-1 pressurizer level. This allows the chemical and volume control system to continue providing reactor coolant system makeup flow, if the makeup pumps are operating following a safeguards actuation signal. These valves are also controlled by the reactor makeup control system

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and close when makeup to other systems is provided. Manual control is provided in the main control room and at the remote shutdown workstation.

Hydrogen Addition Containment Isolation Valves

The outside reactor containment valve is a normally open, fail closed, air-operated globe valve, which provides containment isolation of the hydrogen addition line. The inside reactor containment valve is a check valve that supports containment isolation but has no control function. The air-operated valve automatically closes on a containment isolation signal from the protection and safety monitoring system. The air-operated valve also automatically closes on a low-1 pressurizer level signal from the protection and safety monitoring system to preserve the reactor coolant inventory and to prevent uncovering of the heater elements in the pressurizer. Manual control is provided in the main control room and at the remote shutdown workstation.

Zinc Addition Containment Isolation Valves

The normally open, fail closed, air-operated globe valves provide containment isolation of the zinc addition line. The valves automatically close on a containment isolation signal from the protection and safety monitoring system. The valves also automatically close on a low-1 pressurizer level signal from the protection and safety monitoring system to preserve the reactor coolant inventory and to prevent uncovering of the heater elements in the pressurizer. Manual control for both valves is provided in the main control room and at the remote shutdown workstation.

Demineralized Water System Isolation Valves

These normally open, air-operated butterfly valves are located outside containment in the line from the demineralized water storage and transfer system. These valves close on a signal from the protection and safety monitoring system derived by either a reactor trip signal, a source range flux doubling signal, or low input voltage (loss of ac power) to the 1E dc and uninterruptable power supply system battery chargers, isolating the demineralized water source to prevent inadvertent boron dilution events. The protection and safety monitoring system also issues a close signal to these valves when below the P-8 setpoint when the source range flux doubling signal is blocked to prevent inadvertent boron dilution. Manual control for these valves is provided from the main control room and at the remote shutdown workstation.

Makeup Pump Suction Header Valve

This air-operated, three-way valve is automatically controlled by the reactor makeup control system to provide the desired boric acid concentration of makeup to the reactor coolant system (boric acid, demineralized water, or blend based on the desired reactor coolant system boron concentration). The valve fails with the pump suction aligned to the boric acid storage tank on a loss of instrument air. This valve will also align to the boric acid storage tank on a signal from the protection and safety monitoring system derived from a reactor trip, source range flux doubling, or low input voltage (loss of ac power) to the 1E dc and uninterruptable power supply system battery chargers. This valve also aligns the makeup pump suction to the boric acid storage tank when low pressure is detected in the demineralized water supply line to protect the pump from a loss of suction supply. Manual control for this valve is provided in the main control room and at the remote shutdown workstation.

Makeup Pump Suction Relief Valves

A relief valve is provided in the suction of each makeup pump to prevent overpressurization of the pump suction. These relief valves prevent overpressurization that might be caused by backleakage through the makeup pump discharge check valves when the pump suction valves are closed. The set pressure of these relief valves is equal to the pump suction design pressure. The relief capacity is sufficient to accommodate expected check valve back leakage rates.

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Letdown Line Relief Valve

A relief valve is provided to prevent overpressurization of the letdown line connected to the waste processing system. This relief valve prevents overpressurization that might be caused by opening the letdown line with a closed valve in the waste processing system. The set pressure of this relief valve is equal to the design pressure of the line connecting to the waste processing system. The relief capacity is sufficient to accommodate a conservatively high letdown rate assuming minimum flow resistances in the piping, valves, orifices and equipment in the letdown line.

Letdown Line Containment Isolation Thermal Relief Valve

A relief valve is provided to prevent overpressurization of the letdown line containment penetration. This relief valve prevents overpressurization that might be caused by thermal expansion of the fluid between the containment isolation valves following an event causing containment isolation. This relief valve is located inside containment.

Resin Sluice Line Relief Valve

A relief valve is provided to prevent overpressurization of the line that is used to sluice resin from the mixed bed and cation bed demineralizers to the waste processing system. The set pressure of this relief valve is equal to the design pressure of the line it is connected to which is equal to the design pressure of the CVS purification equipment inside containment. The relief capacity is sufficient to accommodate thermal expansion of the water that is trapped between the two containment isolation valves that might occur following an accident that results in heatup of the containment.

Zinc Addition Line Relief Valve

A relief valve is provided to prevent overpressurization of the zinc addition line. The set pressure of this relief valve is equal to the design pressure of the line it is connected to, which is equal to the design pressure of the chemical and volume control system purification equipment inside containment. The relief capacity is sufficient to accommodate thermal expansion of the water that is trapped between the two containment isolation valves, which might occur following an accident that results in heatup of the containment.

Demineralizer Bypass Valve

This normally closed (normal flow through demineralizers) air-operated valve is located inside the containment. The valve is used, in the open position (flow through bypass line CVS-PL-L065), during hydrazine addition at plant startup. The valve's primary function is to avoid any negative impacts on demineralizer resin performance by contact with hydrazine. This valve fails in position on loss of power and loss of air.

9.3.6.3.8 Piping Requirements

The chemical and volume control system piping that handles radioactive liquid is made of austenitic stainless steel. The piping joints and connections are welded, except where flanged connections are required for equipment removal for maintenance and hydrostatic testing.

9.3.6.4 System Operation and Performance

The operation of the chemical and volume control system for the various modes of reactor plant operation is described in the following subsections.

9.3.6.4.1 Plant Startup

Plant startup is the operation that brings the reactor plant from a cold shutdown condition to no-load operating temperature and pressure, and subsequently to power operation.

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The makeup pumps initially fill the reactor coolant system via the purification flow return line. During filling, makeup water is drawn from appropriate sources to provide makeup at the desired reactor coolant system boron concentration. The reactor coolant system is vented via the reactor vessel head and the pressurizer. A vacuum fill subsystem may be used to enhance the reactor coolant fill operation.

The auxiliary spray line may be used to fill the pressurizer and establish proper water chemistry in the pressurizer. If water solid operation is desired, reactor coolant system pressure is controlled by operation of the letdown control valve and the makeup control valve. To accomplish this, a letdown flow path is established to the liquid radwaste system with the letdown orifice bypassed. The makeup flow rate is maintained by the makeup control valve at a constant value selected by the operators. At the same time, the letdown control valve controls letdown flow to maintain reactor coolant system pressure at a constant value, also selected by the operators. These water solid operations are not required if vacuum fill is used.

During plant startup from cold shutdown, chemical treatment, using an oxygen scavenger, is performed. The oxygen scavenger is added to the reactor coolant during the initial stages of heatup to scavenge oxygen in the system. Subsequently, hydrogen makeup to the reactor coolant system is started, and the reactor coolant system hydrogen level is brought up to the normal operating point of approximately 30 cubic centimeters per kilogram.

The pressurizer heaters are used to heat up the water in the pressurizer and draw a steam bubble. As the steam bubble grows, effluent continues to be diverted to the liquid radwaste system through the chemical and volume control system letdown line. The makeup pumps are operated to supply demineralized water, so the reactor coolant system boron concentration is reduced to the level required for criticality. Following attainment of pressurizer normal water level, the letdown flow control valve and the makeup pumps are set to operate only as necessary to maintain pressurizer level or on demand from the operator.

Criticality is achieved as follows:

- The reactor coolant system boron concentration is reduced to the calculated level by dilution, routing effluent from the chemical and volume control system purification loop to the liquid radwaste system, and by providing unborated makeup with the makeup pumps taking suction from the demineralized water storage tank.

- Chemical analysis is used to measure water quality, boron concentration, and hydrogen concentration.

- Appropriate control rods are withdrawn.

- Further adjustments in boron concentration are made to establish preferred control group rod positions.

9.3.6.4.2 Normal Operation

Normal operation consists of operation at steady power (base load) level, load follow operation, and hot standby.

9.3.6.4.2.1 Base Load Operation

At a constant power level, the chemical and volume control system purification loop operates continuously as a closed loop around a reactor coolant pump. The purification flow is approximately 100 gallons per minute with one mixed bed demineralizer and one reactor coolant filter in service.

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The chemical and volume control system makeup pumps and the letdown line to the liquid radwaste system are not normally operating. The makeup pumps are normally available and are set to start automatically on Low-2 pressurizer level. The boric acid blending valve in the pump suction permits the operator to preset the blend of boric acid and demineralized water to achieve the desired makeup concentration. The letdown control valve opens automatically, if the pressurizer level reaches its high (relative to programmed level) setpoint. Reactor coolant samples are taken to check boron and H_2 concentration, water quality, pH, and activity level.

Variations in power demand are accommodated automatically by control rod and gray rod movement. The only adjustments in boron concentration necessary are those to compensate for core burnup. These adjustments are made to maintain the rod control groups within their allowable limits by setting the makeup pumps to provide the required amount of demineralized water as makeup. If necessary, effluent is automatically routed to the liquid radwaste system to maintain the required pressurizer level.

9.3.6.4.2.2 Load Follow Operation

Load follow power changes and the resulting xenon changes are accommodated by the control rods and gray rods, with no changes required to the reactor coolant system boron concentration. The chemical and volume control system does not have load follow functions.

9.3.6.4.3 Plant Shutdown

9.3.6.4.3.1 Hot Shutdown

If required for periods of maintenance or following spurious reactor trips, the reactor is maintained subcritical, with the capability to return to full power within the period of time required to withdraw the control rods. During hot standby operation, the average temperature is maintained at no-load T_{avg} by initially dumping steam to the condenser to provide residual heat removal, or at later stages by running the reactor coolant pumps to maintain system temperature.

Initially the control rods are inserted and the core is maintained at or slightly above the minimum required shutdown margin. Following shutdown, xenon buildup occurs and increases the shutdown margin. The effect of xenon buildup increases the shutdown margin to a minimum of about 3 percent $\Delta k/k$ at about 9 hours following shutdown.

If rapid recovery is required, dilution of the system may be performed to counteract this xenon buildup. A shutdown group of rods is withdrawn during dilution to provide the capability for rapid shutdown if needed, and frequent checks are made on critical rod position.

9.3.6.4.3.2 Cold Shutdown

Cold shutdown is the operation that brings the reactor plant from normal operating temperature and pressure to a cold shutdown temperature and pressure for maintenance or refueling.

The chemical and volume control system purification loop continues to operate normally in advance of a planned shutdown. In addition, in the beginning of a shutdown, the chemical and volume control system is designed so the letdown flow is routed out of containment to the liquid radwaste system, where it is stripped of gases and returned to the makeup pump suction. This gas stripping is required for approximately 48 hours to reduce reactor coolant activity level and hydrogen level sufficiently, permitting personnel access for refueling or maintenance operations.

Before cooldown and depressurization of the reactor coolant system is initiated, the reactor coolant boron concentration is increased to the cold shutdown value. The operator sets the reactor makeup

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control to "borate" and selects the volume of boric acid solution necessary to perform the boration. Correct concentration is verified by reactor coolant samples. The operator sets the reactor makeup control for makeup at the shutdown reactor coolant boron concentration.

Contraction of the coolant during cooldown of the reactor coolant system results in actuation of the pressurizer level control system to maintain normal pressurizer water level. Makeup continues to be automatic, with the makeup pumps starting and stopping as required.

During shutdowns, after the reactor coolant pumps are stopped, the normal residual heat removal system provides the motive force for chemical and volume control system purification loop. Whenever the reactor coolant system is pressurized, the chemical and volume control system can be operated to provide purification. After the normal residual heat removal system is placed in service and the reactor coolant pumps are stopped, further cooling and depressurization of the pressurizer fluids are accomplished by charging through the auxiliary spray connection.

9.3.6.4.3.2.1 Ion Exchange Media Replacement

The initial and subsequent fill of ion exchange media is made through a resin fill nozzle on the top of the ion exchange vessel. When the media is spent and ready to be transferred to the solid radwaste system (WSS), the vessel is isolated from the process flow. The flush water line is opened to the sluice piping and demineralized water is pumped into the vessel through the normal process outlet connection upward through the media retention screen. The media fluidizes in the upward, reverse flow. When the bed has been fluidized, the sluice connection is opened and the bed is sluiced to the spent resin tanks in the solid radwaste system. Demineralized water flow continues until the bed has been removed and the sluice lines are flushed clean of spent resin.

9.3.6.4.3.2.2 Filter Cartridge Replacement

Replacement of spent filter cartridges is performed as described in Subsection 11.4.2.3.2.

9.3.6.4.4 Abnormal Operation

9.3.6.4.4.1 Reactor Coolant System Leak

The chemical and volume control system is capable of making up for a small reactor coolant system leak with either makeup pump at reactor coolant system pressures above the low-pressure setpoint.

9.3.6.4.5 Accident Operation

The chemical and volume control system can provide borated makeup to the reactor coolant system following accidents such as small loss-of-coolant accidents, steam generator tube rupture events, and small steam line breaks. In addition, pressurizer auxiliary spray can reduce reactor coolant system pressure during certain events such as a steam generator tube rupture.

To protect against steam generator overfill, the makeup function is isolated by closing the makeup line containment isolation valves, if a High-3 steam generator level signal is generated. These valves also close and isolate the system on a High-2 pressurizer level signal.

Some of the valves in the chemical and volume control system are required to operate under accident conditions to effect reactor coolant system pressure boundary and containment isolation, as discussed in Subsection 9.3.6.3.7.

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9.3.6.4.5.1 Boron Dilution Events

The chemical and volume control system is designed to address a boron dilution accident by closing redundant safety-related valves, tripping the makeup pumps and/or aligning the suction of the makeup pumps to the boric acid tank.

For dilution events occurring at power (assuming the operator takes no action), a reactor trip is initiated on either an overpower trip or an overtemperature ΔT trip. Following a reactor trip signal, the line from the demineralized water system is isolated by closing two safety-related, air-operated valves. The three-way pump suction control valve aligns so the makeup pumps take suction from the boric acid tank. If the event occurs while the makeup pumps are operating, the realignment of these valves causes the makeup pumps, if they continue to operate, to borate the plant.

For dilution events during shutdown, the source range flux doubling signal is used to isolate the makeup line to the reactor coolant system by closing the two safety-related, motor-operated valves, isolate the line from the demineralized water system by closing the two safety-related, air-operated valves, and trip the makeup pumps. The source range flux doubling function can be manually blocked during shutdown conditions when below the P-8 setpoint after the operator isolates unborated water source flow paths. When blocked during shutdown conditions, an automatic close signal is also sent to the CVS demineralized water system isolation valves to prevent inadvertent boron dilution. For refueling operations, administrative controls are used to prevent boron dilutions by verifying the valves in the line from the demineralized water system are closed and secured.

9.3.6.5 Design Evaluation

The chemical and volume control system has redundant, safety-related isolation valves and piping to protect the reactor coolant system pressure boundary, and is designed in accordance with ANSI/ ANS-51.1 (Reference 4).

The chemical and volume control system lines that penetrate containment incorporate valve and piping arrangements, meeting the containment isolation criteria described in subsection 6.2.3.

Since the chemical and volume control system supplies unborated water to the reactor coolant system, the potential for inadvertent boron dilution events exists. A safety-related method of stopping an inadvertent boron dilution, which operates as described in Subsection 9.3.6.4.5.1, is incorporated into the chemical and volume control system.

The chemical and volume control system also incorporates a safety-related method of isolating the makeup to the reactor coolant system upon receipt of a High-3 steam generator level signal or a High-2 pressurizer level signal, as described in Subsection 9.3.6.4.5. Other chemical and volume control system components are not safety-related.

Chemical and volume control system components and piping are compatible with the radioactive fluids they contain or functions they perform.

The design of the chemical and volume control system is based on specific General Design Criteria and regulatory guides. The design of the chemical and volume control system is compared to the criteria set forth in Subsection 9.3.4, "Chemical and Volume Control System (PWR) (Including Boron Recovery System)," Revision 2, of the Standard Review Plan. The specific General Design Criteria identified in the Standard Review Plan section are General Design Criteria 1, 2, 3, 4, 14, 29, 30, 31, 32, 33, 53, 54, 56, 60, and 61 as discussed in Section 3.1. Additionally, Subsection 1.9.1 discusses compliance with Regulatory Guides 1.26 and 1.29.

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9.3.6.6 Inspection and Testing Requirements

The only required surveillance is for containment and reactor coolant pressure boundary isolation valves and boron dilution mitigation valves. These valves are identified as active and are tested in accordance with the in-service test provisions discussed in Subsection 3.9.6.

Other chemical and volume control system components are monitored for acceptable performance as follows:

- Mixed and cation bed demineralizer -- monitor for bed exhaustion by comparing reactor coolant system samples to samples taken at the outlet of the reactor coolant filter.

- Reactor coolant and makeup filters -- remotely monitor differential pressure with the installed gages and change the filter cartridges, or switch to the backup filter when high differential pressure is detected with the installed pressure gage.

Inspection of the various components is required in accordance with their safety class. The safety classification assignments can be found in Section 3.2.

9.3.6.6.1 Preoperational Inspection and Testing

Preoperational tests are conducted to verify proper operation of the chemical and volume control system. The preoperational tests include valve inspection and testing and flow testing.

9.3.6.6.1.1 Valve Inspection and Testing

The inspection requirements of the chemical and volume control system valves that constitute the reactor coolant pressure boundary are consistent with those identified in Subsection 5.2.4. The inspection requirements of the chemical and volume control system valves that isolate the lines penetrating containment are consistent with those identified in Section 6.6.

9.3.6.6.1.2 Flow Testing

Each chemical and volume control system pump is tested to measure the flow rate from each makeup pump to the reactor coolant system. Testing will be performed with the pump suction aligned to the boric acid storage tank and the discharge aligned to the reactor coolant system. Testing will also be performed with the pump suction aligned to the boric acid storage tank and the discharge aligned to the pressurizer auxiliary spray. Flow will be measured using instrumentation in the pump discharge line. Testing will confirm that each pump provides at least 100 gallons per minute of makeup flow at normal reactor coolant system operating pressure. This is the minimum flow rate necessary to meet the chemical and volume control system functional requirement of providing makeup and pressurizer spray to support the functions described in Subsection 9.3.6.4.4.1. Testing is performed to verify that the maximum makeup flow with both pumps operating is less than 175 gpm, as assumed in the boron dilution analyses presented in Subsection 15.4.6. Testing is performed with both pumps operating and taking suction from the demineralized water system. The chemical and volume control system is aligned to the reactor coolant system at a pressure at or near atmospheric pressure.

9.3.6.6.1.3 Boric Acid Storage Tank Inspection

Inspection of the boric acid storage tank will be performed to verify that the volume in the tank is sufficient to provide 70,000 gallons of borated makeup to the reactor coolant system. This volume of boric acid is required to meet the functional requirement of providing makeup to the reactor coolant system to support the functions described in Subsection 9.3.6.4.4.

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9.3.6.7 Instrumentation Requirements

Process control instrumentation is provided to acquire data concerning key parameters about the chemical and volume control system. The location of the instrumentation is shown on the chemical and volume control system piping and instrumentation diagram.

The instrumentation furnishes input signals for monitoring and/or alarming. Indications and/or alarms are provided in the main control room for the following parameters:

- Pressure and differential pressure
- Flow
- Temperature
- Water level

The instrumentation also supplies input signals for control purposes to maintain proper system operation and to prevent equipment damage. Some specific control functions are listed below:

- **Purification isolation** – The purification stop valves preserve the reactor coolant pressure boundary in the event of a break in the chemical and volume control system loop piping. The purification stop valves, the zinc addition line valves, and the hydrogen addition line valve outside containment close automatically on a signal from the protection and safety monitoring system generated by a low-1 pressurizer level signal. This isolation also serves as an equipment protection function to prevent uncovering of the heater elements in the pressurizer. One of these valves also closes on high temperature downstream of the letdown heat exchanger, to protect the resin in the mixed bed and cation demineralizers from being exposed to temperatures that could damage the resins.

- **Containment isolation** – To preserve the containment boundary, containment isolation valves are provided in the letdown line to the liquid radwaste system, the chemical and volume control system makeup line, and the hydrogen and zinc addition lines. These valves are opened or closed manually from the main control room and the remote shutdown workstation. Interlocks are provided to close these valves automatically upon receipt of the applicable isolation signal from the protection and safety monitoring system and require operator action to reopen.

- **Letdown isolation valves** – The letdown isolation valves are used to isolate letdown flow to the liquid radwaste system in addition to the containment isolation function described above. The plant control system provides a signal to automatically open these valves on a high-pressurizer level signal derived from the pressurizer level control system. On a containment isolation signal from the protection and safety monitoring system, a high-3 liquid radwaste system degassifier level signal (plant control system), or a low-pressurizer level signal (plant control system), these valves automatically close to provide isolation of the letdown line. The letdown isolation valves also receive a signal from the protection and safety monitoring system to automatically close and isolate letdown during midloop operations based on a Low-2 hot leg level. Manual control is provided from the main control room and at the remote shutdown workstation. The letdown flow control valve controls reactor coolant system pressure during startup, as described in Subsection 9.3.6.4.1.

- **Demineralized water system isolation valves** – To prevent inadvertent boron dilution, the demineralized water system isolation valves close on a signal from the protection and safety monitoring system derived from either a reactor trip signal, a source range flux doubling signal, or low input voltage (loss of ac power) to the 1E dc and uninterruptible power supply system battery chargers providing a safety-related method of stopping an inadvertent

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dilution. The valves are closed to prevent inadvertent boron dilution when the source range flux doubling logic is blocked below P-8. The main control room and remote shutdown workstation provide manual control for these valves.

- **Makeup isolation valves** – To isolate the makeup flow to the reactor coolant system, two valves are provided in the chemical and volume control system makeup line. These valves automatically close on a signal from the protection and safety monitoring system derived from source range flux doubling, high-2 pressurizer level, High-3 steam generator level, a safeguards signal coincident with high-1 pressurizer level, or high steam generator water level coincident with P-4 permissive (reactor trip) to protect against pressurizer or steam generator overfill. Manual control for these valves is provided in the main control room and at the remote shutdown workstation. In addition, the valves close on a high-2 containment radiation signal to protect containment integrity.

- **Makeup flow control** – To control makeup flow to the reactor coolant system, a flow controller, which operates in the makeup line, in conjunction with the reactor makeup control system is provided in the chemical and volume control system makeup pump discharge line. This flow controller controls makeup flow by modulating a flow control valve.

- **Makeup pump control** – The makeup pumps can be controlled from the main control room and at the remote shutdown workstation. On a signal from the plant control system generated by a low pressurizer level signal (relative to the programmed level), one of the chemical and volume control system makeup pumps starts automatically to provide makeup. The operating pump automatically stops when the pressurizer level increases to the correct value. During reactor coolant system boron changes (fuel depletion, startups, shutdowns, and refueling), the operator starts one of the makeup pumps after selecting the desired amount of boric acid.

The makeup pumps can be used to provide reactor coolant system makeup following an accident such as a small loss-of-coolant accident, a steam generator tube rupture, or a small steam line break. Following a safeguards actuation signal, if necessary, the operator remotely opens the makeup line isolation valves. One makeup pump automatically starts to control the pressurizer level between 10 and 20 percent. In addition, a makeup pump may be used to provide pressurizer auxiliary spray in reducing the reactor coolant system pressure for certain accident scenarios.

9.3.6.8 System Gas Accumulation Assessment and Mitigation

Subsection 6.3.6.3.1 describes the assessment method to address the potential for gas intrusion for the passive core cooling system in response to Interim Staff Guidance (ISG-019) and Generic Letter 2008-01. The same assessment methodology was used to assess the chemical and volume control system.

The chemical and volume control system locations equipped with manual vent valves will be inspected according to the system surveillance and venting procedures described in Subsection 6.3.6.3 to eliminate identified gas accumulations.

9.3.7 Combined License Information

Generic Issue 43, and the concerns of Generic Letter 88-14 and NUREG-1275 regarding degradation or malfunction of instrument air supply and safety-related valve failure, are addressed by the training and procedures for operations and maintenance of the instrument air subsystem and air-operated valves.

Plant systems, including the compressed and instrument air system, are maintained in accordance with procedures. Maintenance procedures are discussed in Subsection 13.5.2.2.6. The instrument

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air supply subsystem components are maintained and tested in accordance with manufacturers' recommendations and procedures. The safety-related air-operated valves are maintained in accordance with manufacturers' recommendations and tested in accordance with plant procedures to allow proper function on loss of air. The instrument air is periodically sampled and tested for compliance with the quality requirements of ANSI/ISA-S7.3-1981.

Operators are provided training on loss of instrument air in accordance with abnormal operating procedures. Plant systems, including the compressed and instrument air system, are operated in accordance with system operating procedures, abnormal operating procedures, and alarm response procedures which are written in accordance with Subsection 13.5.2. The training program for operations and maintenance personnel is discussed in Section 13.2.

9.3.8 References

1. Instrument Society of America Standards, "Quality Standard for Instrument Air," S7.3; 1981.

2. ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, "Pressure Vessels," 1998 Edition, 2000 Addenda.

3. ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, "Pressure Vessels," Subsection A, Part UG-99, Standard Hydrostatic Test, 1998.

4. ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

5. Safety Evaluation by the Office of Nuclear Regulation Related to WCAP-14986, “Westinghouse Owners Group Post Accident Sampling System Requirements,” Westinghouse Owners Group Project No. 694, June 14, 2000.

6. NUREG 0800, Standard Review Plan Section 9.3.2 "Process and Post-Accident Sampling Systems."

7. CGA G-7.1-1997, “Commodity Specification for Air.”

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Table 9.3.1-1 (Sheet 1 of 2)
Safety-Related Air-Operated Valves

Valve Number	Normal/Failure Position	Function
Compressed and Instrument Air System (CAS)		
CAS-PL-V014	NO/FC	Instrument Air Supply Outside Containment Isolation
Chemical and Volume Control System (CVS)		
CVS-PL-V045	NC/FC	Letdown Containment Isolation IRC
CVS-PL-V047	NC/FC	Letdown Containment Isolation ORC
CVS-PL-V084	NC/FC	Auxiliary Pressurizer Spray Line Isolation
CVS-PL-V092	NO/FC	Zinc Injection Containment Isolation ORC
CVS-PL-V094	NO/FC	Zinc Injection Containment Isolation Valve IRC
CVS-PL-V136A	NO/FC	Demineralized Water System Isolation
CVS-PL-V136B	NO/FC	Demineralized Water System Isolation
CVS-PL-V219	NO/FC	Hydrogen Injection Containment Isolation
Passive Containment Cooling System (PCS)		
PCS-PL-V001A	NC/FO	Passive Containment Cooling Water Storage Tank Isolation
PCS-PL-V001B	NC/FO	Passive Containment Cooling Water Storage Tank Isolation
Primary Sampling System (PSS)		
PSS-PL-V011A	NO/FC	Containment Isolation – Liquid Sample Line
PSS-PL-V011B	NC/FC	Containment Isolation – Liquid Sample Line
PSS-PL-V023	NO/FC	Containment Isolation – Sample Return Line
PSS-PL-V046	NO/FC	Containment Isolation – Air Sample Line
Passive Core Cooling System (PXS)		
PXS-PL-V014A	NC/FO	Core Makeup Tank A Discharge Isolation
PXS-PL-V014B	NC/FO	Core Makeup Tank B Discharge Isolation
PXS-PL-V015A	NC/FO	Core Makeup Tank A Discharge Isolation
PXS-PL-V015B	NC/FO	Core Makeup Tank B Discharge Isolation
PXS-PL-V042	NO/FC	Nitrogen Supply Containment Isolation ORC
PXS-PL-V108A	NC/FO	Passive Residual Heat Removal Heat Exchanger Control
PXS-PL-V108B	NC/FO	Passive Residual Heat Removal Heat Exchanger Control
PXS-PL-V130A	NO/FC	In-Containment Refueling Water Storage Tank Gutter Isolation
PXS-PL-V130B	NO/FC	In-Containment Refueling Water Storage Tank Gutter Isolation
PXS-PL-V230A	NC/FC	Core Makeup Tank A Fill Isolation
PXS-PL-V230B	NC/FC	Core Makeup Tank B Fill Isolation
PXS-PL-V232A	NC/FC	Accumulator A Fill/Drain Isolation
PXS-PL-V232B	NC/FC	Accumulator B Fill/Drain Isolation

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Table 9.3.1-1 (Sheet 2 of 2)
Safety-Related Air-Operated Valves

Valve Number	Normal/Failure Position	Function
Normal Residual Heat Removal System (RNS)		
RNS-PL-V006A	NO/FO	RNS HX A Outlet Flow Control
RNS-PL-V006B	NO/FO	RNS HX B Outlet Flow Control
RNS-PL-V008A	NC/FC	RNS HX A Bypass Flow Control
RNS-PL-V008B	NC/FC	RNS HX B Bypass Flow Control
RNS-PL-V029	NC/FC	RNS Discharge to CVS
RNS-PL-V057A	NO/FO	RNS Pump Miniflow Isolation
RNS-PL-V057B	NO/FO	RNS Pump Miniflow Isolation
RNS-PL-V061	NC/FC	Shutdown Purification Flow Isolation
Steam Generator System (SGS)		
SGS-PL-V036A	NO/FC	Steam Line Condensate Drain Isolation
SGS-PL-V036B	NO/FC	Steam Line Condensate Drain Isolation
SGS-PL-V074A	NO/FC	Steam Generator Blowdown Isolation
SGS-PL-V074B	NO/FC	Steam Generator Blowdown Isolation
SGS-PL-V075A	NO/FC	Steam Generator Series Blowdown Isolation
SGS-PL-V075B	NO/FC	Steam Generator Series Blowdown Isolation
SGS-PL-V086A	NC/FC	Steam Line Condensate Drain Control
SGS-PL-V086B	NC/FC	Steam Line Condensate Drain Control
SGS-PL-V233A	NC/FC	Power Operated Relief Valve
SGS-PL-V233B	NC/FC	Power Operated Relief Valve
SGS-PL-V240A	NC/FC	Main Steam Isolation Valve Bypass Isolation
SGS-PL-V240B	NC/FC	Main Steam Isolation Valve Bypass Isolation
SGS-PL-V250A	NO/FC	Main Feedwater Control
SGS-PL-V250B	NO/FC	Main Feedwater Control
SGS-PL-V255A	NC/FC	Startup Feedwater Control
SGS-PL-V255B	NC/FC	Startup Feedwater Control
Main Control Room Emergency Habitability System (VES)		
VES-PL-V022A	NC/FO	Relief Isolation Valve A
VES-PL-V022B	NC/FO	Relief Isolation Valve B
Containment Air Filtration System (VFS)		
VFS-PL-V003	NC/FC	Containment Purge Inlet Containment Isolation Valve
VFS-PL-V004	NC/FC	Containment Purge Inlet Containment Isolation Valve
VFS-PL-V009	NC/FC	Containment Purge Discharge Containment Isolation Valve
VFS-PL-V010	NC/FC	Containment Purge Discharge Containment Isolation Valve
Central Chilled Water System (VWS)		
VWS-PL-V058	NO/FC	Fan Coolers Supply Isolation Valve
VWS-PL-V082	NO/FC	Fan Coolers Return Isolation Valve
VWS-PL-V086	NO/FC	Fan Coolers Return Isolation Valve
Liquid Radwaste System (WLS)		
WLS-PL-V055	NC/FC	Sump Discharge Containment Isolation IRC
WLS-PL-V057	NC/FC	Sump Discharge Containment Isolation ORC
WLS-PL-V067	NC/FC	Reactor Coolant Drain Tank Gas Outlet Containment Isolation IRC
WLS-PL-V068	NC/FC	Reactor Coolant Drain Tank Gas Outlet Containment Isolation ORC

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Table 9.3.1-2
Nominal Component Design Data - Instrument Air Subsystem

Air Compressors	
Quantity	2
Type	Rotary
Capacity, each (scfm)	800
Design pressure (psig)	150
Air Receivers	
Quantity	2
Capacity, each (ft^3)	Minimum of 672
Design pressure (psig)	150
Prefilters	
Quantity	2
Type	Coalescing
Air Dryers	
Quantity	2
Type	Desiccant/Purge Air Regenerative
Capacity, each (scfm)	800
Operating pressure dew point, maximum (°F)	-28
Afterfilters	
Quantity	2
Type	Particulate

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Table 9.3.1-3
Nominal Component Design Data - Service Air Subsystem

Air Compressor	
Quantity	2
Type	Rotary
Capacity, each (scfm)	800
Design pressure (psig)	150
Air Receiver	
Quantity	1
Capacity (ft^3)	Minimum of 672
Design pressure (psig)	150
Prefilters	
Quantity	2
Type	Coalescing
Air Dryer	
Quantity	2
Type	Desiccant/Purge Air Regenerative
Capacity, each (scfm)	800
Design pressure dew point, maximum (°F)	-28
Afterfilters	
Quantity	2
Type	Particulate

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Table 9.3.1-4
Nominal Component Design Data - High-pressure Air Subsystem

Air Compressor	
Quantity	1
Type	Reciprocating
Capacity (scfm)	60
Design pressure (psig)	5000
Breathing Air Purifier	
Quantity	1
Type	Molecular Sieve/Activated Carbon
CO to CO_2 conversion	Catalysis
Air supply quality level	E
Air Receiver	
Quantity	1
Capacity, water volume (ft^3)	46
Design pressure (psig)	4000

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Table 9.3.3-1
Primary Sampling System Sample Points - Normal Plant Operations
(Liquid And Gaseous)

Sample Point Name	Type of Sample(a)
Liquid Sample	
1. RCS Hot Leg (before CVS demineralizer)	Grab
2. Pressurizer Liquid Space	Grab
3. CVS Demineralizer Downstream	Grab
4. PXS Accumulators	Grab
5. PXS Core Makeup Tanks (at top)	Grab
6. PXS Core Makeup Tanks (at bottom)	Grab
7. PXS Core Makeup Tanks (at middle)	Grab
8. Containment Sump (pump discharge)	Grab
Gaseous Sample	
9. Containment Air	Grab

Note:

a. This column shows methods to obtain a sample for chemical analysis. It does not specify the frequency of sampling nor does it specify actual location of sample collection. "Grab" means that a grab sample is required for the intended chemical analysis. Depending on the sampling condition, this grab sample can be obtained in the laboratory or in the grab sampling unit.

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Table 9.3.3-2 (Sheet 1 of 4)
Local Sample Points Not in the Primary Sampling System
(Normal Plant Operations)

Sample Point Name	Available Number of Points	Type of Sample[(a)]	Process Measurement
Liquid Sample			
1. CVS boric acid storage tank	1	Grab	pH, chloride, fluoride, boron, silica, suspended solids, radioisotopic liquid, dissolved oxygen, sulfate, aluminum, calcium, magnesium
2. CVS boric acid batching tank	1	Grab	Boron, chloride, fluoride, sulfate
3. Residual heat removal heat exchanger	2	Grab	Radioisotopic liquid, radioisotopic particulate, suspended solids, radioisotopic gas, gross specific activity, strontium, iron, tritium, hydrogen, I-131, conductivity, pH, dissolved oxygen, chloride, fluoride, boron, aluminum, silica, lithium radioisotopic liquid, lithium radioisotopic particulate, magnesium, sulfate, calcium, lithium, nickel
4. PXS IRWST	1	Grab	pH, dissolved oxygen, fluoride, boron, conductivity, gross specific activity, sodium, sulfate, silica, chloride, aluminum, calcium, magnesium
5. Main steam line (Outlet SG 1)	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
6. Main steam line (Outlet SG 2)	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
7. BDS steam generator blowdown	1	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
8. SFS purification (Upstream & downstream of SFS ion exchangers) (spent fuel pool treatment)	2	Grab	Conductivity, pH, chloride, silica, corrosion product metals, gross activity, corrosion product activity, fission product activity, I-131, tritium, turbidity, boron, corrosion product metals, organic impurities, fluoride, sulfate, aluminum, calcium, magnesium
9. PCS water storage tank	1	Grab	Hydrogen peroxide
10. Reactor coolant drain tank	1	Grab	Gross radioactivity and identification and concentration of principal radionuclide and emitters, dissolved gases.

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Table 9.3.3-2 (Sheet 2 of 4)
Local Sample Points Not in the Primary Sampling System
(Normal Plant Operations)

Sample Point Name	Available Number of Points	Type of Sample[(a)]	Process Measurement
11. WLS degasifier (downstream of degasifier discharge pump)	1	Grab	Dissolved gases
12. CCS component cooling surge tank	1	Grab	pH, sodium, chloride, silica, corrosion product metals, corrosion inhibitors, fluoride, sulfate
13. CCS loops (downstream of CCS pumps)	2	Grab	pH, sodium, chloride, silica, corrosion product metals, tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters, fluoride, sulfate, total organic carbon, conductivity
14. CCS hot leg (upstream of CCS pumps)	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
15. WLS discharge (liquid radwaste effluent)	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
16. WLS effluent holdup tanks MT05A, B	2	Grab	Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters, boron
17. WLS waste holdup tanks MT06A, B	2	Grab	Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
18. WLS monitor tanks MT07A, B, C, D, E, F	6	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters. State and federal environmental discharge requirements such as pH, suspended solids, oil and grease, iron, copper, sodium nitrite
19. WLS ion exchanger pre-filter (downstream)	1	Grab	Suspended solids
20. WLS ion exchanger after-filter (downstream)	1	Grab	Suspended solids

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Table 9.3.3-2 (Sheet 3 of 4)
Local Sample Points Not in the Primary Sampling System
(Normal Plant Operations)

Sample Point Name	Available Number of Points	Type of Sample[a]	Process Measurement
21. WLS chemical waste tank	1	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters, boron
22. WSS spent resin tanks (liquid)	1	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
23. SWS blowdown (service water)	1 1	Continuous Grab	Radiation monitor (See Section 11.5, Table 11.5-1) Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
24. WWS turbine building sump	2	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
25. CPS (secondary coolant) spent resin sluice line (liquid)	1	Grab	Tritium, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
26. Cask Loading pit	1	Grab[d]	Conductivity, pH, chloride, fluoride, sulfate, silica, aluminum, calcium, magnesium, corrosion-product activity, fission product activity, I-131, tritium, turbidity, boron, organic impurities
Gaseous Sample			
27. VES MCR emergency air supply headers	2	Grab	Air quality, oxygen, carbon monoxide, carbon dioxide, contaminants
28. WGS effluent discharge to environment	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
29. WGS inlet	1	Continuous	Oxygen, hydrogen
30. WGS carbon bed vault	1	Continuous	Hydrogen
31. WGS delay bed outlets MV-02A, B (waste gas holdup)	2	Grab	Noble gases, iodine, particulates, tritium
32. Condenser air removal system[b] (including hogging)	1	Grab	Iodine, noble gases, tritium

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Table 9.3.3-2 (Sheet 4 of 4)
Local Sample Points Not in the Primary Sampling System
(Normal Plant Operations)

Sample Point Name	Available Number of Points	Type of Sample[a]	Process Measurement
33. Gland seal system[b]	1	Grab	Iodine, noble gases, tritium
34. Plant vent (including containment purge, auxiliary building ventilation, fuel storage and radwaste area ventilation discharge)	1	Continuous & Grab[c]	Iodine, noble gases, particulates

Notes:

a. This column shows methods to obtain a sample for analysis. "Grab" means that a grab sample is required for the intended analysis. “Continuous” means that the required analysis is performed via a probe that monitors the sampling stream continuously.
b. Continuous monitoring of discharge for radiation provided in turbine island vent (See Section 11.5, Table 11.5-1).
c. Includes analysis for tritium.
d. It is recommended that a sampling method capable of obtaining samples from multiple depths in the CLP is used to ensure that stratification is not an issue.

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Table 9.3.4-1 (Sheet 1 of 2)
Secondary Sampling System
(Continuous and Semi-Continuous Measurements)

Continuous Sample Points	Process Measurements
Hotwell (Tube Bundle Condenser Shell A)	Specific Conductivity Cation Conductivity Sodium
Hotwell (Tube Bundle Condenser Shell B)	Specific Conductivity Cation Conductivity Sodium
Hotwell (Tube Bundle Condenser Shell C)	Specific Conductivity Cation Conductivity Sodium
Hotwell (Interconnecting Piping Between Condenser Shell A and Condenser Shell B)	Specific Conductivity Cation Conductivity Sodium
Hotwell (Interconnecting Piping Between Condenser Shell B and Condenser Shell C)	Specific Conductivity Cation Conductivity Sodium
Condensate Pump Discharge	Specific Conductivity Cation Conductivity Sodium pH Dissolved Oxygen
Deaerator Inlet (Condensate)	Specific Conductivity Cation Conductivity Sodium[(a)] pH Oxygen Scavenger Residual Dissolved Oxygen
Feedwater	Specific Conductivity Cation Conductivity Sodium Dissolved Oxygen pH Oxygen Scavenger Residual
Steam Generator Blowdown (SG 1)	Specific Conductivity Cation Conductivity Sodium pH Sulfate Dissolved Oxygen

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Table 9.3.4-1 (Sheet 2 of 2)
Secondary Sampling System
(Continuous and Semi-Continuous Measurements)

Steam Generator Blowdown (SG 2)	Specific Conductivity Cation Conductivity Sodium pH Sulfate Dissolved Oxygen
Main Steam System (SG 1)	Specific Conductivity Cation Conductivity Sodium[(a)] pH Dissolved Oxygen
Main Steam System (SG 2)	Specific Conductivity Cation Conductivity Sodium[(a)] pH Dissolved Oxygen

Note:

(a) Semi-continuous sampling. “Semi-continuous” means sampling instrumentation cycles through multiple individual sample streams that share a common sample header at pre-defined discrete intervals.

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Table 9.3.4-2
Secondary Sampling System
(Selective Measurements)

Condenser Tube Bundle B (North Side)
Condenser Tube Bundle B (South Side)
Heater Drain (LP Heater)
Heater Drain (MSR 1st Stage Tube Drain)
Heater Drain (MSR 2nd Stage Tube Drain)
Heater Drain (MSR Shell Drain)
Auxiliary Steam
Auxiliary Boiler Feedwater
Auxiliary Boiler Drum
Auxiliary Boiler Condensate
Condensate Polisher Outlet
Heater Drain (HP Heater)
Deaerator Outlet (Feedwater)
Startup Feedwater

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Table 9.3.5-1
Component Data - Radioactive Waste Drains System
(Nominal Values)

Drain Sump	
Capacity (gal)	1400
Design pressure	Atmospheric
Design temperature (°F)	150
Material	Stainless steel
Drain Sump Pumps	
Quantity per sump	2
Design flow rate (gpm)	125
Pump type	Pneumatic double diaphragm
Material	Stainless steel

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Table 9.3.6-1
Chemical and Volume Control System Parameters[(a)]

Nominal purification flow rate (gpm)	100
Nominal boration flow rate (gpm)	100
Nominal dilution flow rate (gpm)	100
Nominal temperature of reactor coolant entering chemical and volume control system (assumed) (°F)	537
Expected life of demineralizer resin	1 fuel cycle
Maximum temperature of effluent to liquid radwaste system (°F)	130
Nominal effluent flow rate to liquid radwaste system (gpm)	100

Note:
a. Volumetric flow rates are based on 130°F and 2250 psia.

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Table 9.3.6-2 (Sheet 1 of 3)
Chemical and Volume Control System
Nominal Equipment Design Parameters

Pumps		
Makeup Pumps		
Number	2	
Type	Multistage horizontal centrifugal	
Design pressure (psig)	3,100	
Design flow (gpm)	135	
Material	Stainless Steel (SS)	
Heat Exchangers		
Regenerative Heat Exchanger		
Number	1	
Type	Counterflow	
	Shell Side	**Tube Side**
Design pressure (psig)	3,100	3,100
Design temperature (°F)	600	650
Design flow (lb/hr)	54,185	54,185
Material	SS	SS
Letdown Heat Exchanger		
Number	1	
Type	U-Tube	
	Shell Side	**Tube Side**
Design pressure (psig)	200	3,100
Design temperature (°F)	200	600
Design flow (lb/hr)	224,044	54,185
Material	CS or SS	SS

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Table 9.3.6-2 (Sheet 2 of 3)
Chemical and Volume Control System
Nominal Equipment Design Parameters

Demineralizers	
Mixed Bed Demineralizer	
Number	2
Design pressure (psig)	3,100
Design temperature (°F)	200
Design flow (gpm)	250
Resin volume (ft^3)	50
Material	SS
Resin type	Mixed Bed Li7OH Form
Cation Bed Demineralizer	
Number	1
Design pressure (psig)	3,100
Design temperature (°F)	200
Design flow (gpm)	250
Resin volume (ft^3)	50
Material	SS
Resin type	Cation H+ Form

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Table 9.3.6-2 (Sheet 3 of 3)
Chemical and Volume Control System
Nominal Equipment Design Parameters

Filter	
Reactor Coolant Filter	
Number	2
Type	Disposable Cartridge
Design pressure (psig)	3,100
Design temperature (°F)	200
Design flow (gpm)	250
Maximum differential pressure at design flow (psi)	43
Tank	
Boric Acid Tank	
Number	1
Volume (gal)	74,839
Type	Cylindrical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	SS

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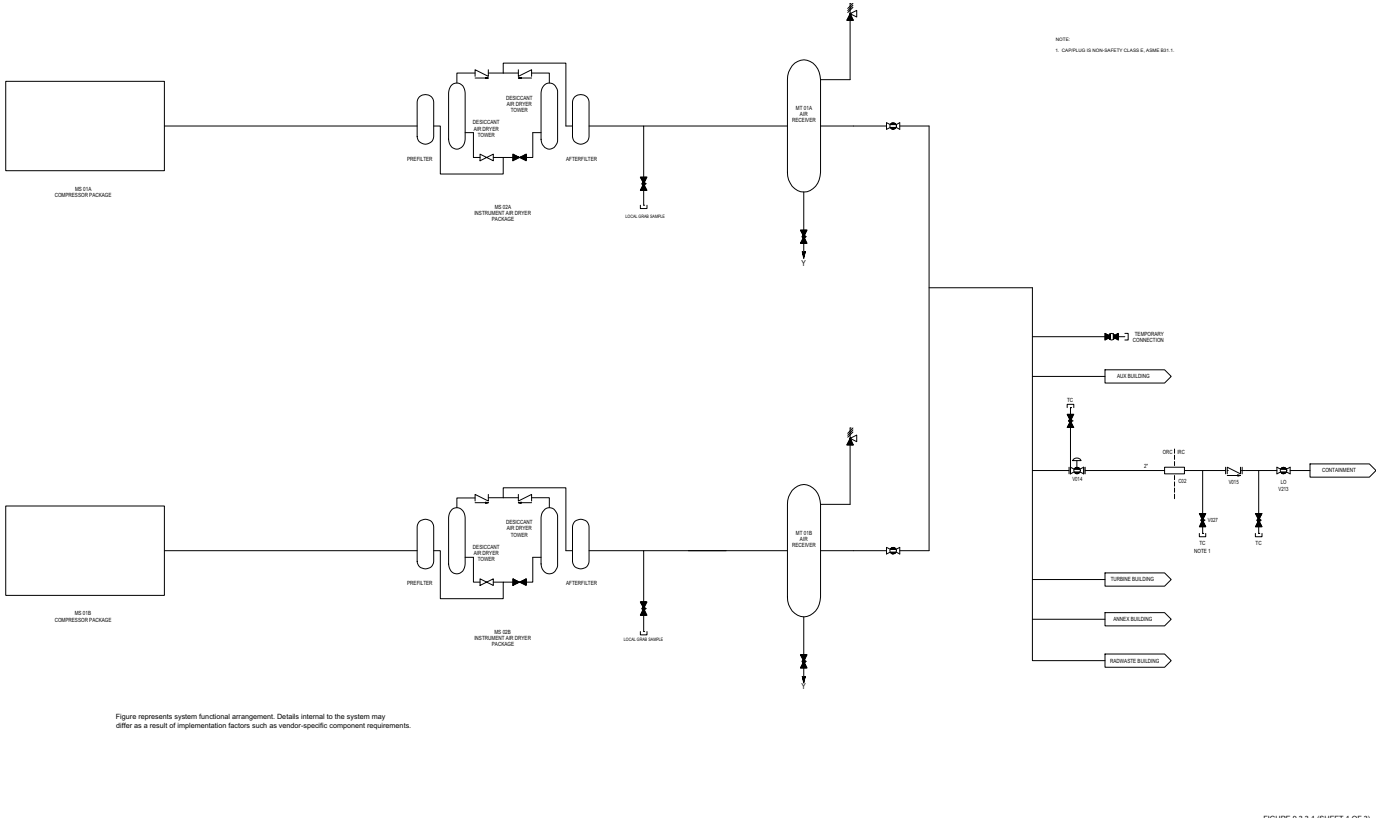

Figure 9.3.1-1 (Sheet 1 of 3)
Compressed & Instrument Air System
Piping and Instrumentation Diagram
(REF CAS 001 & 005)

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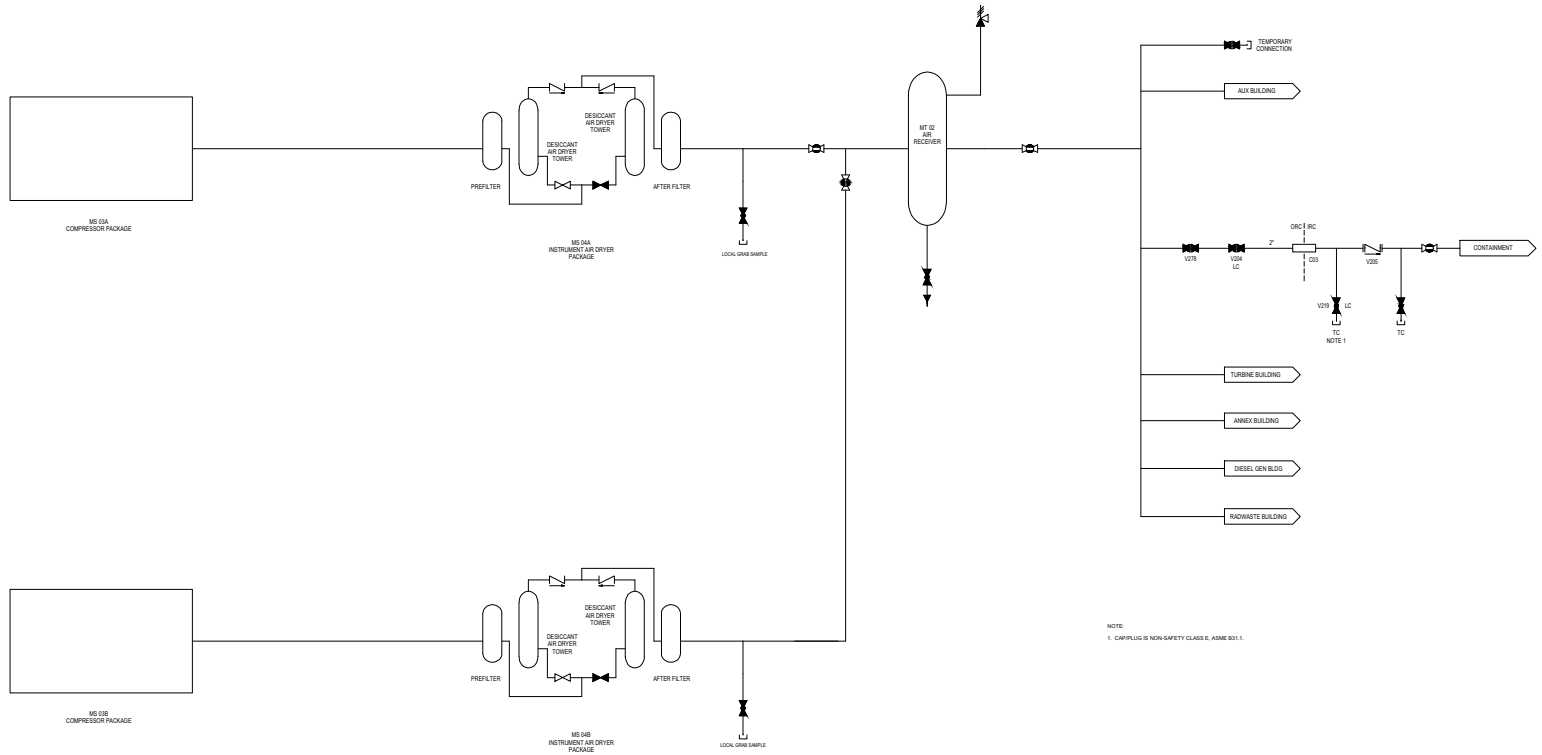

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Figure 9.3.1-1 (Sheet 2 of 3)
Compressed & Instrument Air System
Piping and Instrumentation Diagram
(REF CAS 008 & 012)

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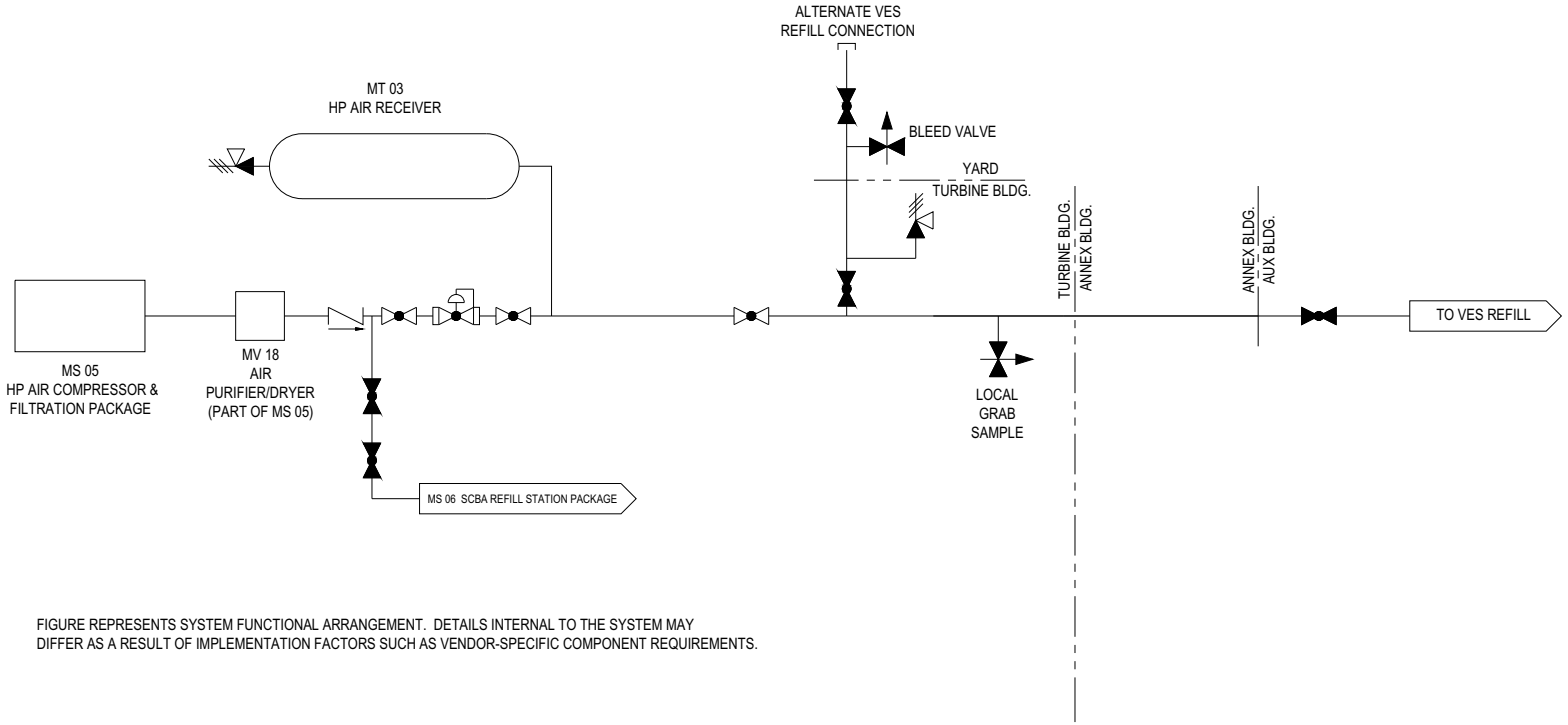

Figure 9.3.1-1 (Sheet 3 of 3)
Compressed & Instrument Air System
Piping and Instrumentation Diagram
(REF CAS 015)

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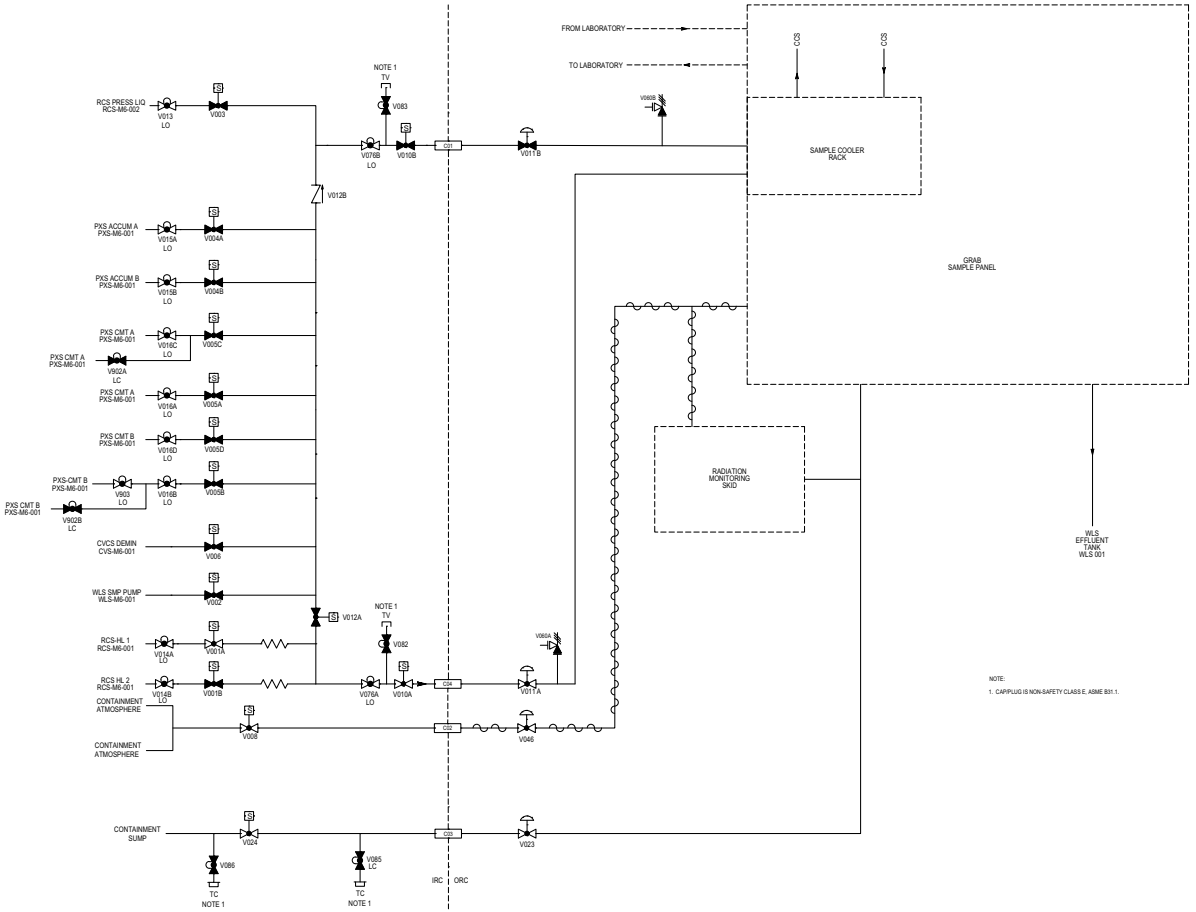

Figure 9.3.3-1
Simplified Sketch of the Primary Sampling System (REF PSS 001)

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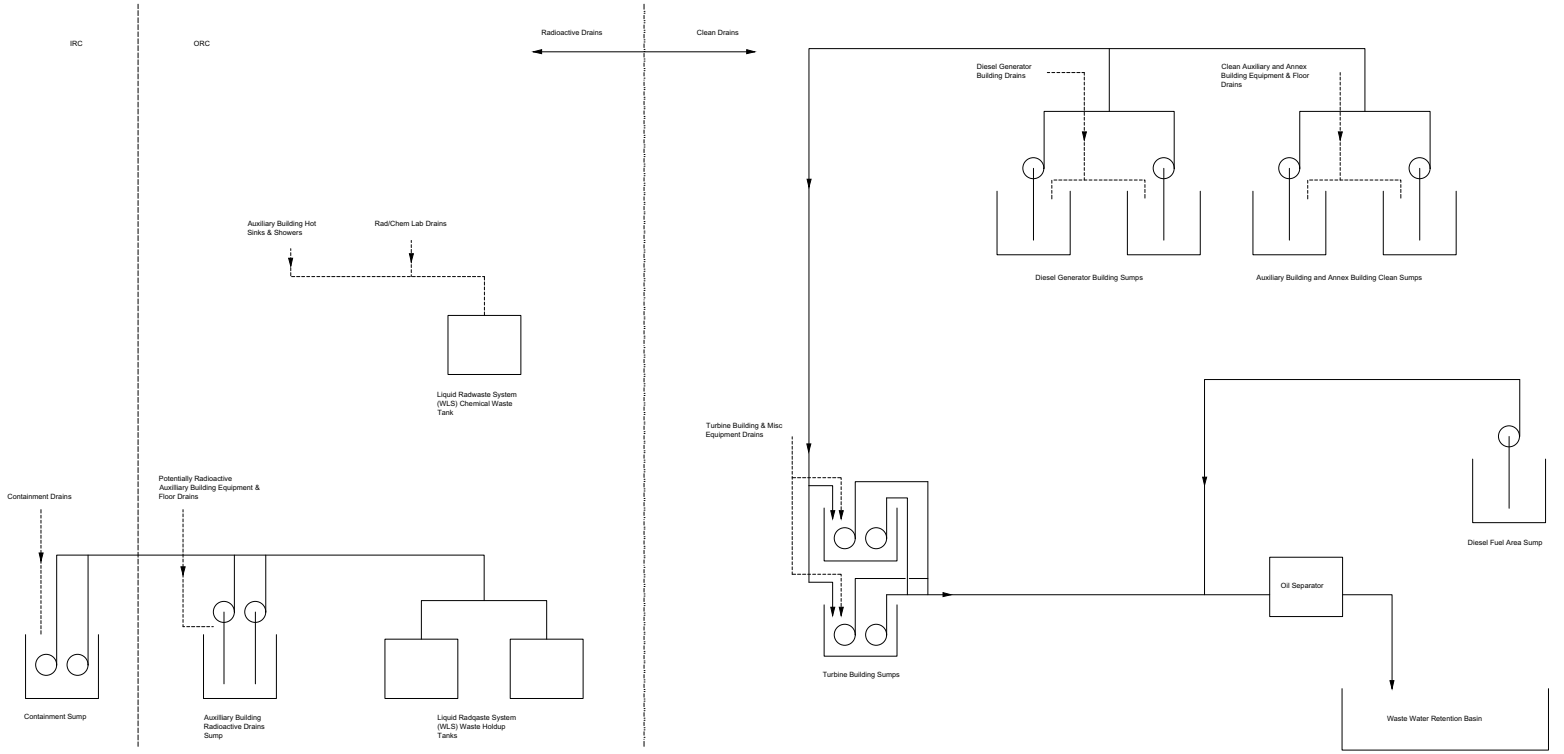

Figure 9.3.5-1
General Arrangement of Drainage Systems

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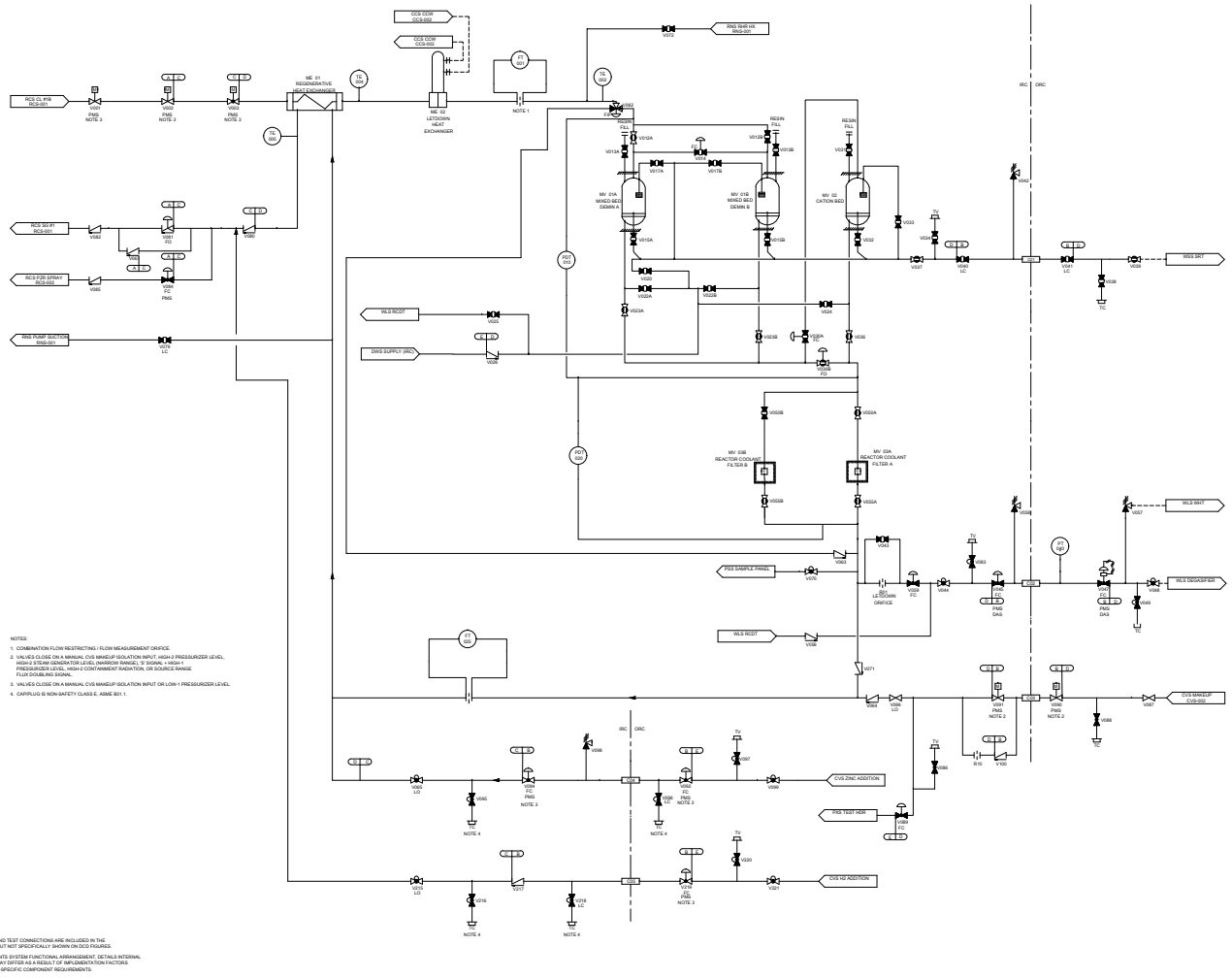

Figure 9.3.6-1 (Sheet 1 of 2)
Simplified Chemical and Volume Control System Piping and Instrumentation (REF) CVS 001, 003, 005

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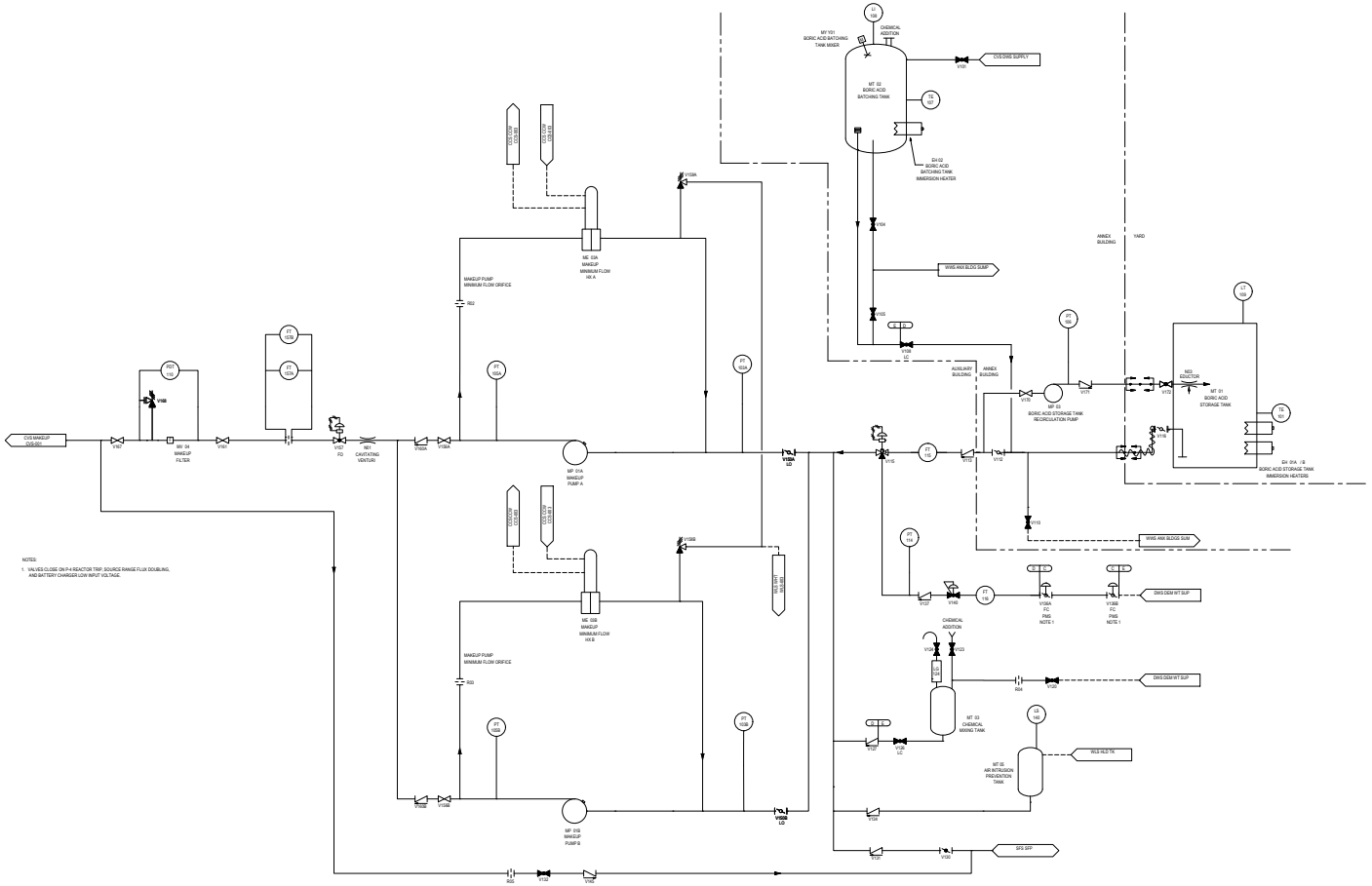

Figure 9.3.6-1 (Sheet 2 of 2)
Simplified Chemical and Volume Control System Piping and Instrumentation (REF) CVS 002, 004

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9.4 Air-Conditioning, Heating, Cooling, and Ventilation System

The air-conditioning, heating, cooling, and ventilation system is comprised of the following systems that serve the various buildings and structures of the plant:

- Nuclear island nonradioactive ventilation system (Subsection 9.4.1)
- Annex/auxiliary buildings nonradioactive HVAC system (Subsection 9.4.2)
- Radiologically controlled area ventilation system (Subsection 9.4.3)
- Containment recirculation cooling system (Subsection 9.4.6)
- Containment air filtration system (Subsection 9.4.7)
- Radwaste building HVAC system (Subsection 9.4.8)
- Turbine building ventilation system (Subsection 9.4.9)
- Diesel generator building heating and ventilation system (Subsection 9.4.10)
- Health physics and hot machine shop HVAC system (Subsection 9.4.11)

9.4.1 Nuclear Island Nonradioactive Ventilation System

The nuclear island nonradioactive ventilation system (VBS) serves the main control room (MCR), control support area (CSA), Class 1E dc equipment rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, remote shutdown room, reactor coolant pump trip switchgear rooms, adjacent corridors, and the passive containment cooling system (PCS) valve room during normal plant operation.

The main control room emergency habitability system provides main control room habitability in the event of a design basis accident (DBA) and is described in Section 6.4.

9.4.1.1 Design Basis

9.4.1.1.1 Safety Design Basis

The nuclear island nonradioactive ventilation system provides the following nuclear safety-related design basis functions:

- Monitors the main control room supply air for radioactive particulate and iodine concentrations

- Isolates the HVAC penetrations in the main control room boundary on “High-2” particulate or iodine concentrations in the main control room supply air, extended loss of main control room differential pressure, or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4

Those portions of the nuclear island nonradioactive ventilation system which penetrate the main control room envelope are safety-related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a design basis accident. Other functions of the system are nonsafety-related. HVAC equipment and ductwork whose failure could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portion of the system is nonsafety-related and nonseismic. The equipment is procured to meet the environmental qualifications used in standard building practice.

The nuclear island nonradioactive ventilation system is designed to control the radiological habitability in the main control room within the guidelines presented in Standard Review Plan (SRP) 6.4 and NUREG 0696 (Reference 1), if the system is operable and ac power is available.

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Portions of the system that provide the defense-in-depth function of filtration of main control room/ control support area air during conditions of abnormal airborne radioactivity are designed, constructed, and tested to conform with Generic Issue B-36, as described in Section 1.9 and Regulatory Guide 1.140 (Reference 30), as described in Appendix 1A, and the applicable portions of ASME AG-1 (Reference 36), ASME N509 (Reference 2), and ASME N510 (Reference 3).

Power to the ancillary fans to provide post-72-hour ventilation of the control room and I&C rooms is supplied from divisions B and C regulating transformers through two series fuses for isolation. The fuses protect the regulating transformers from failures of the non-Class 1E fan circuits. When normal ventilation is available the ancillary fan circuits are disconnected from the supply with manual normally-open switches.

The nuclear island nonradioactive ventilation system is designed to provide a reliable source of heating, ventilation, and cooling to the areas served when ac power is available. The system equipment and component functional capabilities are to minimize the potential for actuation of the main control room emergency habitability system or the potential reliance on passive equipment cooling. This is achieved through the use of redundant equipment and components that are connected to standby onsite ac power sources.

9.4.1.1.2 Power Generation Design Basis

Main Control Room/Control Support Area (CSA) Areas

The nuclear island nonradioactive ventilation system provides the following specific functions:

- Controls the main control room and control support area relative humidity between 25 to 60 percent
- Maintains the main control room and CSA areas at a slightly positive pressure with respect to the adjacent rooms and outside environment during normal operations to prevent infiltration of unmonitored air into the main control room and CSA areas
- Isolates the main control room and/or CSA area from the normal outdoor air intake and provides filtered outdoor air to pressurize the main control room and CSA areas to a positive pressure of at least 1/8 inch wg when a “High-1” radioactivity concentration (gaseous, particulate, or iodine) is detected in the main control room supply air duct
- Isolates the main control room and/or CSA area from the normal outdoor air intake and provides 100 percent recirculation air to the main control room and CSA areas when a high concentration of smoke is detected in the outside air intake
- Provides smoke removal capability for the main control room and control support area
- Maintains the main control room emergency habitability system passive cooling heat sink below its initial design ambient air temperature limit of 75°F
- Maintains the main control room/control support area carbon dioxide levels below 0.5 percent concentration and the air quality within the guidelines of Table 1 and Appendix C, Table C-1 of Reference 32.

The background noise level in the main control room does not exceed 65 dB(A) when the VBS is operating.

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The system maintains the following room temperatures based on the maximum and minimum outside air safety temperature conditions shown in Chapter 2, Table 2-1:

Area	Temperature (°F)
Main control room	67 – 75
Control support area	67 – 78

Class 1E Electrical Rooms/Remote Shutdown Room

The nuclear island nonradioactive ventilation system provides the following specific functions:

- Exhausts air from the Class 1E battery rooms to limit the concentration of hydrogen gas to less than 2 percent by volume in accordance with Regulatory Guide 1.128 (Reference 31).
- Maintains the Class 1E electrical room emergency passive cooling heat sink below its initial design ambient air temperature limit of 75°F
- Provides smoke removal capability for the Class 1E electrical equipment rooms and battery rooms

The background noise level in the remote shutdown room does not exceed 65 dB(A) when the VBS is operating.

The system maintains the following room temperatures based on the maximum and minimum outside air safety temperature conditions shown in Chapter 2, Table 2-1:

Area	Temperature (°F)
Class 1E battery rooms	67 - 73
Class 1E dc equipment rooms	67 - 73
Class 1E electrical penetration rooms	67 - 73
Class 1E instrumentation and control rooms	67 - 73
Corridors	67 - 73
Remote shutdown room	67 - 73
Reactor coolant pump trip switchgear rooms	67 - 73
HVAC equipment rooms	50 - 85

Passive Containment Cooling System Valve Room

The subsystem maintains the following room temperatures based on the maximum and minimum outside air safety temperature conditions shown in Chapter 2, Table 2-1:

Area	Temperature (°F)
Passive containment cooling system valve room	70 - 120

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Post-72-Hour Design Basis

Main Control Room

The specific function of the nuclear island nonradioactive ventilation system is to maintain the main control room below a maximum average WBGT Index of 90°F based on operation at the site maximum normal temperature.

Divisions B and C Instrumentation and Control Rooms Design Basis

The specific function of the nuclear island nonradioactive ventilation system is to maintain the I&C rooms below the qualification temperature of the I&C equipment.

9.4.1.2 System Description

The nuclear island nonradioactive ventilation system is shown in Figure 9.4.1-1. The system consists of the following independent subsystems:

- Main control room/control support area HVAC subsystem
- Class 1E electrical room HVAC subsystem
- Passive containment cooling system valve room heating and ventilation subsystem

9.4.1.2.1 General Description

9.4.1.2.1.1 Main Control Room/Control Support Area HVAC Subsystem

The main control room/control support area HVAC subsystem serves the main control room and control support area with two 100 percent capacity supply air handling units, return/exhaust air fans, supplemental air filtration units, associated dampers, instrumentation and controls, and common ductwork. The supply air handling units and return/exhaust air fans are connected to common ductwork which distributes air to the main control room, the CSA, and the toilet serving the CSA. The main control room envelope consists of the main control room, shift manager's office, operation work area, toilets, and operations break room area. The CSA consists of the main control support area operations area, conference rooms, NRC room, computer rooms, and kitchen/rest area. The main control room toilets and the toilet serving the CSA have separate exhaust fans.

Outside supply air is provided to the plant areas served by the main control room/control support area HVAC subsystem through an outside air intake duct that is protected by an intake enclosure located on the roof of the auxiliary building at elevation 153′-0″. The outside air intake duct is located more than 50 feet below and more than 100 feet laterally away from the plant vent discharge. The supply, return, and toilet exhaust are the only HVAC penetrations in the main control room envelope and include redundant safety-related seismic Category I isolation valves that are physically located within the main control room envelope. Redundant safety-related radiation monitor sample line connections are located upstream of the VBS supply air isolation valves. These monitors initiate operation of the nonsafety-related supplemental air filtration units on "High-1" radioactivity concentrations (gaseous, particulate, or iodine) and isolate the main control room from the nuclear island nonradioactive ventilation system on "High-2" particulate or iodine radioactivity concentrations. A Class R seismic Category I static air mixer is located in the VBS ductwork to provide mixed air flow for the radiation monitor sample line connections. See Section 11.5 for a description of the main control room supply air radiation monitors.

Both redundant trains of supplemental air filtration units and one train of the supply air handling unit are located in the main control room mechanical equipment room at elevation 135′-3″ in the auxiliary building. The other supply air handling unit subsystem is located in the main control room mechanical equipment room at elevation 135′-3″ in the annex building. The main control room toilet exhaust fan is located at elevation 135′-3″ in the auxiliary building. A humidifier is provided for each supply air

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handling unit. The supply air handling unit cooling coils are provided with chilled water from air-cooled chillers in the central chilled water system. See Subsection 9.2.7 for the chilled water system description.

The main control room/control support area HVAC subsystem has design features to limit migration of smoke, hot gases, and fire suppressant from one fire area to another to the extent that they could adversely affect safe shutdown capabilities, including operator actions. Fire or combination fire and smoke dampers are provided to isolate each fire area from adjacent fire areas during and following a fire in accordance with NFPA 90A (Reference 27) requirements. These combination smoke/fire dampers close in response to smoke detector signals or in response to the heat from a fire. See Appendix 9A for identification of fire areas.

9.4.1.2.1.2 Class 1E Electrical Room HVAC Subsystem

The Class 1E electrical room HVAC subsystem serves the Class 1E electrical rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, spare Class 1E battery room, remote shutdown room, and reactor coolant pump trip switchgear rooms. The A and C electrical divisions, spare battery room, and reactor coolant pump trip switchgear rooms are served by one ventilation subsystem; the B and D electrical divisions and remote shutdown room are served by a second ventilation subsystem.

Each subsystem consists of two 100 percent capacity supply air handling units, return/exhaust air fans, associated dampers, controls and instrumentation, and common ductwork. The supply air handling units and return/exhaust air fans are connected to a common ductwork which distributes air to the Class 1E electrical rooms. The outside supply air intake enclosure for the A and C subsystem is common to the main control room/control support area intake located on the roof of the auxiliary building at elevation 153′-0″. The outside supply air intake for the B and D subsystem is located separate from the main control room/control support area air intake enclosure on the auxiliary building roof at elevation 153′-0″. The exhaust ducts from the battery rooms are connected to the turbine building vent to remove hydrogen gas generated by the batteries.

The HVAC equipment which serves the A and C electrical divisions is located in the nuclear island nonradioactive ventilation system main control room/A and C equipment room at elevation 135′-3″ in the auxiliary building. The HVAC equipment which serves the B and D division of Class 1E electrical equipment is located in the upper and lower nuclear island nonradioactive ventilation system B and D equipment rooms at elevation 117′-0″ and at elevation 135′-3″.

The supply air handling unit cooling coils are provided with chilled water from the air-cooled chillers in the central chilled water system. The two air handling units for each set of electrical divisions are provided with chilled water from redundant air-cooled chillers. Refer to Subsection 9.2.7 for the chilled water system description.

Each subsystem for the Class 1E battery rooms is provided with two 100 percent capacity exhaust fans.

The Class 1E electrical room HVAC subsystem is designed so that smoke, hot gases, and fire suppressant does not migrate from one fire area to another to the extent that they could adversely affect safe shutdown capabilities, including operator actions. Separate ventilation subsystems are provided to serve the electrical division A and C equipment rooms and the electrical division B and D equipment rooms. The use of separate HVAC distribution subsystems for the redundant trains of electrical equipment prevents smoke and hot gases from migrating from one distribution division to the other through the ventilation system ducts. In addition, combination fire-smoke dampers are provided for Class 1E equipment rooms, including the remote shutdown room, to isolate each fire area and block the migration of smoke and hot gases to or from adjacent fire areas in accordance

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with NFPA 90A requirements. These combination fire/smoke dampers close in response to smoke detector signals or in response to the heat from a fire. During a fire, the pressure difference across the doors in the stairwells S01 and S02 is maintained in accordance with the guidance of NFPA 92A (Reference 33) by dedicated stairwell pressurization fans. See Appendix 9A for identification of fire areas.

9.4.1.2.1.3 Passive Containment Cooling System Valve Room Heating and Ventilation Subsystem

The passive containment cooling system valve room heating and ventilation subsystem serves the passive containment cooling system valve room.

The subsystem consists of one 100 percent ventilating fan, two 100 percent capacity electric unit heaters, associated dampers, controls and instrumentation. The passive containment cooling system valve room heating and ventilation subsystem equipment is located in the passive containment cooling system valve room in the containment dome area at elevation 286′-6″.

The exhaust fan draws outside air through an intake louver damper and directly exhausts to the environment.

9.4.1.2.2 Component Description

The nuclear island nonradioactive ventilation system is comprised of the following major components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Tables 9.4.1-1, 9.4.1-2 and 9.4.1-3 provide design parameters for major components in each subsystem.

Supply Air Handling Units

Each air handling unit consists of a mixing box section, a low efficiency filter bank, high efficiency filter bank, a hot water heating coil, a chilled water cooling coil bank, and supply and return/exhaust air fans.

Supply and Return/Exhaust Air Fans

The supply and return/exhaust fans are centrifugal type, single width single inlet (SWSI), double width double inlet (DWDI), tubular centrifugal or plenum centrifugal with backward inclined blades that produce non-overloading horsepower characteristics or radial blades that produce overloading horsepower characteristics. Fan configurations are either performance rated in accordance with ANSI/AMCA 210 (Reference 4) or certified in accordance with AMCA 211 (Reference 5). Fan configurations are sound tested in accordance with ANSI/AMCA 300 (Reference 6).

The supply and return/exhaust air fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with high efficiency wheels and backward inclined blades to produce non-overloading horsepower characteristics. The fans are designed and rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5) and ANSI/AMCA 300 (Reference 6).

Ancillary Fans

The ancillary fans are centrifugal type with non-overloading horsepower characteristics. Each can provide a minimum of 1,530 cfm. Fan configurations are performance rated in accordance with ANSI/AMCA 210 (Reference 4) and sound tested in accordance with ANSI/AMCA 300 (Reference 6).

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Supplemental Air Filtration Units

Each supplemental air filtration unit includes a fan, a high efficiency filter bank, an electric heating coil, a charcoal adsorber with upstream HEPA filter bank, and a downstream postfilter bank. The filtration unit configurations, including housing, internal components, ductwork, dampers, fans and controls, and the location of the fans on the unfiltered side of units are designed, constructed, and tested to meet the applicable performance requirements of ASME AG-1, ASME N509, and ASME N510 (References 36, 2, and 3) to satisfy the guidelines of Regulatory Guide 1.140 (Reference 30).

Low Efficiency Filters, High Efficiency Filters, and Postfilters

The low efficiency filters and high efficiency filters have a rated dust spot efficiency based on ASHRAE 52.1 and 126 (References 7 and 35). Filter minimum average dust spot efficiency is shown in Tables 9.4.1-1 and 9.4.1-2. High efficiency filter performance upstream of HEPA filter banks meet the design requirements of ASME AG-1 (Reference 36), Section FB. Postfilters downstream of the charcoal filters have a minimum DOP efficiency of 95 percent. The filters are UL listed in accordance with UL 900 (Reference 8).

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with UL-586 (Reference 9) and ASME AG-1 (Reference 36), Section FC. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol in accordance with ASME AG-1 (Reference 36), Section TA.

Charcoal Adsorbers

Each charcoal adsorber is designed, constructed, qualified, and tested in accordance with ASME AG-1 (Reference 36), Section FE; and Regulatory Guide 1.140 (Reference 30). Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with IE Bulletin 80-03 (Reference 29).

Electric Heating Coils

The electric heating coils are SCR controlled, fin tubular type. The electric heating coils are UL listed in accordance with UL 1996 (Reference 26). The electric heating coils for the supplemental air filtration subsystem are constructed, qualified, and tested in accordance with ASME AG-1 Section CA (Reference 36).

Electric Convection Heaters

The electric convection heaters are single or multi stage fin tubular type. The electric convection heaters are UL listed in accordance with UL 2021 (Reference 40) or UL 1042 (Reference 41) and meet the requirements of the National Electric Code NFPA 70 (Reference 28). The electric convection heaters are controlled by an integral temperature sensor or by a temperature sensor located in the space served by the heater.

Electric Unit Heaters

The electric unit heaters are single or multi stage, fin tubular type. The electric unit heaters are UL listed in accordance with UL 2021 (Reference 40) and meet the requirements of the National Electric Code NFPA 70 (Reference 28).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

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Hot Water Heating Coils

The hot water heating coils are counterflow, finned tubular type. Heating is controlled by modulating the hot water to the hot water heating coils. The heating coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifiers

The humidifiers are packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifiers are designed and rated in accordance with ARI 640 (Reference 13).

Isolation Dampers and Valves

Nonsafety-related isolation dampers are bubble tight, single- or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power. The isolation dampers are Class R and are designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section DA.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

The main control room pressure boundary penetrations include isolation valves, interconnecting piping, and vent and test connection with manual test valves. The isolation valves are classified as Safety Class C (see Subsection 3.2.2.5 and Table 3.2-3) and seismic Category I. Testing the effectiveness of their boundary isolation function is covered by the main control room pressure boundary testing described in Subsection 6.4.5.2, and by the inservice testing program for valves described in Subsection 3.9.6.

The main control room pressure boundary isolation valves have motor operators. The valves are designed to fail as is in the event of loss of electrical power. The valves are qualified to shut tight against control room pressure.

Tornado Protection Dampers

The tornado protection dampers are split-wing type and designed to close automatically. The tornado protection dampers are designed against the effect of 300 mph wind.

Shutoff and Balancing Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements in accordance with ANSI/AMCA 500 (Reference 14). The supplemental air filtration subsystem dampers are Class R and are designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section DA.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

Fire Damper

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL-555 (Reference 15). Fire dampers are not provided in locations where combination fire/smoke dampers are provided.

Fire dampers at duct penetrations through fire barriers, which are within ducting that is routed outside the MCR/CSA pressure boundary are Class R and are designed, constructed, and tested in

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accordance with ASME AG-1 Section DA (Reference 36). The Class R fire damper is applied to promote an ALARA HVAC design. The use of Class R fire damper mitigates the potential for infiltration of airborne radiation in ducting during operation of the fan coincident with an airborne release from spaces the ducting is routed through.

Combination Fire/Smoke Dampers

Combination fire/smoke dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The combination fire/smoke dampers are designed, constructed, and tested in accordance with UL-555S (Reference 25) and UL-555 (Reference 15).

Smoke Dampers

Smoke dampers are provided to restrict the spread of smoke in HVAC systems. Smoke dampers meet the design, leakage testing, and installation requirements of UL-555S (Reference 25).

Silencers

Silencers are provided to maintain the background noise below acceptable levels in the main control room. The silencers are steel casing packed with glass fiber absorptive fill. The principal construction code is the manufacturer's standard.

Static Air Mixer

A static air mixer is provided to support fully developed air flow to the MCR radiation monitoring packages. The principal construction code is the manufacturer's standard; however, the static air mixer is commercially dedicated to ASME AG-1 (Reference 36).

Ductwork, Duct Supports, and Accessories

Non-Seismic ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards- Metal and Flexible (Reference 17) with the exception of control support area ductwork which is noted below.

Seismic II Category ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34), SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17), and Appendix 3A, with the exception of supplemental air filtration and main control room HVAC subsystem's ductwork which is noted below.

The supplemental air filtration and main control room/control support area HVAC subsystem's ductwork, including the air filtration units and the portion of the ductwork located outside of the main control room envelope, that maintains integrity of the main control room/control support area pressure boundary during conditions of abnormal airborne radioactivity is Class R and is designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section SA, to provide low leakage components necessary to maintain main control room/control support area habitability. Ductwork is constructed of materials consistent with these standards including, but not limited to, galvanized or stainless steel. Duct supports and accessories are constructed of material consistent with these standards including, but not limited to, galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

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9.4.1.2.3 System Operation

9.4.1.2.3.1 Main Control Room/Control Support Area HVAC Subsystem

Normal Plant Operation

During normal plant operation, one of the two 100 percent capacity supply air handling units and return/exhaust air fans operates continuously. Outside makeup air supply to the supply air handling units is provided through an outside air intake duct. The outside airflow rate is automatically controlled to maintain the main control room and CSA areas at a slightly positive pressure with respect to the surrounding areas and the outside environment.

The main control room/control support area supply air handling units are sized to provide cooling air for personnel comfort, equipment cooling, and to maintain the main control room emergency habitability passive heat sink below its initial ambient air design temperature. The temperature of the air supplied by each air handling unit is controlled by temperature sensors located in the main control room return air duct and in the computer room B return air duct to maintain the ambient air design temperature within its normal design temperature range by modulating the hot water heat or chilled water cooling. Some spaces have convection heaters for temperature control.

The outside air is continuously monitored by smoke detectors located at the outside air intake plenum and the return air is monitored for smoke upstream of the supply air handling units. The supply air to the main control room is continuously monitored for airborne radioactivity while the supplemental air filtration units remain in a standby operating mode.

The standby supply air handling unit and corresponding return/exhaust fans are started automatically if one of the following conditions shuts down the operating unit:

- Airflow rate of the operating fan is below a predetermined setpoint.
- Return air temperature is above or below predetermined setpoints.
- Differential pressure between the main control room and the surrounding areas and outside environment is above or below predetermined setpoints.
- Loss of electrical and/or control power to the operating unit.

Abnormal Plant Operation

Control actions are taken at two levels of radioactivity as detected in the main control room supply air duct. The first is “High-1” radioactivity based upon radioactivity instrumentation (gaseous, particulate, or iodine). The second is “High-2” radioactivity based upon either particulate or iodine radioactivity instruments.

If “High-1” radioactivity is detected in the main control room supply air duct and the main control room/control support area HVAC subsystem is operable, both supplemental air filtration units automatically start to pressurize the main control room and CSA areas to at least 1/8 inch wg with respect to the surrounding areas and the outside environment using filtered makeup air. The normal outside air makeup duct and the exhaust duct isolation dampers for the main control room toilets and the toilet serving the control support area close. The smoke/purge exhaust isolation dampers close, if open. The main control room/control support area supply air handling unit continues to provide cooling with recirculation air to maintain the main control room passive heat sink below its initial ambient air design temperature and maintains the main control room and CSA areas within their design temperatures. The supplemental air filtration subsystem pressurizes the combined volume of the main control room and control support area concurrently with filtered outside air. A portion of the

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recirculation air from the main control room and control support area is also filtered for cleanup of airborne radioactivity. The main control room/control support area HVAC equipment and ductwork that form an extension of the main control room/control support area pressure boundary limit the overall infiltration (negative operating pressure) and exfiltration (positive operating pressure). The system is designed to maintain personnel doses within allowable General Design Criteria (GDC) 19 limits during design basis accidents in both the main control room and the control support area.

If ac power is unavailable for more than 10 minutes, or if main control room differential pressure is below the “Low” setpoint for more than 10 minutes, or if “High-2” particulate or iodine radioactivity is detected in the main control room supply air duct, which may lead to exceeding GDC 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room from the normal main control room/control support area HVAC subsystem by closing the supply, return, and toilet exhaust isolation valves. Main control room habitability is maintained by the main control room emergency habitability system, which is discussed in Section 6.4.

The main control room and CSA areas ventilation supply and return/exhaust ducts can be remotely or manually isolated from the main control room.

If a high concentration of smoke is detected in the outside air intake, an alarm is initiated in the main control room and the main control room/control support area HVAC subsystem is manually realigned to the recirculation mode by closing the outside air and toilet exhaust duct isolation valves. The exhaust fans for the main control room toilets and the toilet serving the control support area are tripped upon closure of the isolation valves. The main control room/CSA areas are not pressurized when operating in the recirculation mode. The main control room/control support area HVAC supply air subsystem continues to provide cooling, ventilation, and temperature control to maintain the emergency habitability passive heat sink below its initial ambient air design temperature and maintains the main control room and CSA areas within their design temperatures.

In the event of a fire in the main control room or control support area, in response to heat from the fire or upon receipt of a smoke signal from an area smoke detector, the main control room/control support area HVAC subsystem is automatically shut down and the combination fire/smoke dampers close automatically to isolate the fire area. The main control room/control support area HVAC subsystem can be manually realigned to the once-through ventilation mode to supply 100 percent outside air to the unaffected area. Realignment to the once-through ventilation mode minimizes the potential for migration of smoke or hot gas from the fire area to the unaffected area. Smoke and hot gases can be removed from the affected area by reopening the closed combination fire/smoke damper(s) from outside of the affected fire area during the once-through ventilation mode. In the once-through ventilation mode, the outside air intake damper to the air handling unit mixing plenum opens and the return air damper to the air handling unit closes to provide 100 percent outside air to the supply air handling unit. In this mode, the subsystem exhaust air isolation damper opens to exhaust the return air directly to the turbine building vent.

In the event of a fire with the potential to disable the main control room/control support area HVAC subsystem, operators may isolate the main control room from the normal ventilation subsystem and actuate the main control room emergency habitability system at their discretion to maintain control room habitability.

Power is supplied to the main control room/control support area HVAC subsystem by the plant ac electrical system. In the event of a loss of the normal ac power and offsite power, the main control room/control support area ventilation subsystem can be transferred to the onsite standby diesel generators. The convection heaters and duct heaters are not transferred to the onsite standby diesel generator.

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In the event normal, onsite standby and offsite ac power are lost for more than 72 hours and the outside air is acceptable radiologically and chemically, MCR habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCR. It is expected the radioactivity of the outside air will decrease to an acceptable level within 72 hours following a radiological release. See Subsection 6.4.2.2 for details. The outside air pathway to the ancillary fans is provided through the seismic Category II portion of the annex building, into the auxiliary building mechanical room at floor elevation 135′-3″, and nonradioactive ventilation system supply duct. Warm air from the MCR is vented to the annex building through the clean access corridor at elevation 117′-6″. The ancillary fan capacity and air flow rate maintain the MCR environment below a maximum average WBGT Index of 90°F based on operation at the site maximum normal temperature. The ancillary fans and ducted flow path are located within the auxiliary building which is a Seismic Category I structure.

Power supply to the ancillary fans is from the respective division B or C regulating transformers which receive power from the ancillary diesel generators. For post-72-hour power supply discussion see Subsection 8.3.1.1.1.

9.4.1.2.3.2 Class 1E Electrical Room HVAC Subsystem

The Class 1E electrical room HVAC equipment that serves electrical division A and C equipment is described in this section. The operation of the Class 1E electrical room HVAC equipment that serves electrical division B and D is similar.

Normal Plant Operation

During normal plant operation, one of the redundant supply air handling units, return fans, and battery room exhaust fans operate continuously to provide room temperature control, to maintain the Class 1E electrical room emergency passive heat sink below its initial ambient air temperature, and to purge and prevent build-up of hydrogen gas concentration in the Class 1E Battery Rooms. The temperature of air supplied for heating and cooling is controlled by a temperature sensor located in the supply air duct that provides a signal for a temperature controller. These controls maintain the room air temperature within the normal design range by modulating hot water heating or chilled water cooling. Duct heaters are controlled by temperature sensors located in the space served by the heater.

During normal plant operation, the exhaust airflow from the Class 1E battery rooms is vented directly to the turbine building vent to limit the concentration of hydrogen gas in the rooms to less than 2 percent by volume in accordance with the guidelines of Regulatory Guide 1.128.

The outside makeup air to the supply air handling units is provided through an outside air intake duct. The outside airflow rate is manually balanced during system startup to provide adequate makeup air for the battery room exhaust fans.

The standby supply air handling unit and the corresponding return/exhaust fans are started automatically if one of the following conditions occurs:

- Airflow rate of the operating fan is below a predetermined setpoint.
- Return air temperature is above or below predetermined setpoints.
- Loss of electrical and/or control power to the operating unit.

Abnormal Plant Operation

The Class 1E electrical room HVAC divisions A/C subsystem outside air intake/exhaust dampers close on the start of one or both supplemental filtration unit fans. The Class 1E electrical room HVAC divisions B/D subsystem outside air intake/exhaust dampers close on a “High-2” particulate or iodine signal from the MCR radiation package communicated through the plant control system (PLS).

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During a design basis accident (DBA), if the normal ac power and offsite power are unavailable, the Class 1E electrical room passive heat sink provides area temperature control. Refer to Section 6.4 for further details.

If a high concentration of smoke is detected in the outside air intake and an alarm is initiated in the main control room, the Class 1E electrical HVAC subsystem(s) can be manually aligned to the recirculation mode by closing the outside air intake damper to the air handling unit mixing plenum. This allows 100 percent room air to return to the supply air subsystem air handling unit. The subsystem continues to provide cooling, ventilation, and temperature control to maintain the areas served by the subsystem(s) within their design temperatures and pressures.

The Class 1E electrical room HVAC divisions A/C subsystem and the Class 1E electrical room HVAC divisions B/D subsystem are separated such that a fire in a fire area housing the equipment of one subsystem will not disable both subsystems. In the event of a fire in a Class 1E electrical room, in response to heat from the fire or upon receipt of a smoke signal from an area smoke detector, the combination fire/smoke dampers close automatically to isolate the fire area. The affected subsystem continues to provide ventilation/cooling to the remaining areas. Either or both subsystems can be manually realigned to the once-through ventilation mode to supply 100 percent outside air to the unaffected areas. Realignment to the once-through ventilation mode minimizes the potential for migration of smoke and hot gases from a non-Class 1E electrical room or a Class 1E electrical room of one division into the Class 1E electrical room of another division. Smoke and hot gases can be removed from the affected areas by reopening the closed combination fire/smoke dampers from outside of the affected fire area during the once-through ventilation mode. In the once-through ventilation mode, the outside air intake damper to the air handling unit mixing plenum opens and the return air damper to the air handling unit closes to allow 100 percent outside air to the supply air handling unit. The subsystem exhaust air isolation damper also opens to exhaust room air directly to the turbine building vent. During a fire, the pressure difference across the doors in stairwells S01 and S02 is maintained in accordance with the guidance of NFPA 92A (Reference 33) by dedicated stairwell pressurization fans.

The power supplies to the Class 1E electrical room HVAC subsystem are provided by the plant ac electrical system and the onsite standby diesel generators. In the event of a loss of the normal ac power and offsite power, the Class 1E electrical room HVAC subsystem is automatically transferred to the onsite standby diesel generators. The convection heaters and duct heaters are not transferred to the onsite standby diesel generator.

In the event normal, onsite standby and offsite ac power are lost for more than 72 hours, division B and C instrumentation and control room temperature is maintained by operating their respective ancillary fans (VBS-MA-11 and VBS-MA-12) to supply outside air to the I&C rooms. It is expected the radioactivity of the outside air will decrease to an acceptable level within 72 hours following a radiological release. The outside air pathway to the ancillary fans is through the nonradioactive seismic Category II portion of the annex building, into the auxiliary building mechanical room at floor elevation 135′-3″, stairway No. 1 doors at elevation 135′-3″ and 82′-6″, the access corridor at floor elevation 82′-6″, and the divisional battery rooms. The warm air is vented to the annex building through the clean access corridor at elevation 100′-0″. The outside air supply provides cooling and maintains room temperature below the qualification temperature of the I&C equipment. The ancillary fans and ducted flow path are located within the auxiliary building which is a Seismic Category I structure.

In the unlikely event that the nuclear island nonradioactive ventilation subsystems are disabled for more than 8 hours (for other than a loss of ac event), doors of the affected rooms may be opened to mitigate heat-up until HVAC is restored. Ancillary or portable fans also may be utilized in the event of an extended loss of HVAC in order to maintain the ambient temperature below the qualification temperature of the equipment.

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Power supply to the ancillary fans is from the respective division B or C regulating transformers which receive power from the ancillary diesel generators. For post-72-hours power supply discussion see Subsection 8.3.1.1.1.

9.4.1.2.3.3 Passive Containment Cooling System Valve Room Heating and Ventilation Subsystem

Normal Plant Operation

The passive containment cooling system valve room ventilation fan exhausts room air to the outside environment to maintain room temperature within its normal design temperature range.

When heating is required, one of the two redundant electric unit heaters provides heating to maintain the passive containment cooling system valve room temperature above its minimum design temperature. The lead electric unit heater starts or stops when the room air temperature is above or below predetermined setpoints. The standby electric unit heater starts automatically if the room air temperature drops below a predetermined setpoint.

Abnormal Plant Operation

The power supplies to the passive containment cooling system valve room unit heaters are provided by the plant ac electrical system and the onsite standby diesel generators. In the event of a loss of the normal ac power and offsite power, the passive containment cooling system valve room unit heaters can be transferred to the onsite standby diesel generators by the operator.

The power supply to the passive containment cooling system valve room ventilation fan is provided by the plant ac electrical system. The room temperature is not expected to exceed 120°F, based on maximum ambient conditions and internal heat sources.

Following a fire in the passive containment cooling system valve room, smoke and hot gases can be removed from the area using portable exhaust fans and flexible ductwork.

9.4.1.3 Safety Evaluation

The nuclear island nonradioactive ventilation system has no safety-related function other than main control room envelope isolation and main control room supply air radioactivity monitoring, and therefore requires no nuclear safety evaluation. Redundant safety-related isolation valves are provided in the supply, return, and exhaust ducts penetrating the main control room. Therefore, there are no single active failures which would prevent isolation of the main control room envelope. The safety-related redundant main control room supply air radiation monitors are provided. The nuclear island nonradioactive ventilation system is designed so that safety-related systems, structures, or components are not damaged as a result of a seismic event.

9.4.1.4 Tests and Inspection

The nuclear island nonradioactive ventilation system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of each system are accessible for periodic inspection during normal plant operation. A system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program. Airflow rates are measured and balanced within specified design flow rate ranges in accordance with the guidelines of SMACNA HVAC Systems - Testing, Adjusting and Balancing (Reference 19). The supplemental air filtration units are tested to confirm airflow capacity in accordance with the guidelines of ASME N510 (Reference 3). Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability. Air quality within the MCR/CSA environment is confirmed to be within the guidelines of Table 1 and

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Appendix C, Table C-1, of Reference 32 by analyzing air samples taken during preoperational testing.

The supplemental air filtration unit, HEPA filters, and charcoal adsorbers are tested in place in accordance with ASME N510 to verify that these components do not exceed a maximum allowable bypass leakage rate. Samples of charcoal adsorbent, used or new, are periodically tested to verify a minimum charcoal efficiency of 99 percent in accordance with Regulatory Guide 1.140 (Reference 30), except that test procedures and test frequency are conducted in accordance with ASME N510.

The ductwork for the supplemental air filtration subsystem and portions of the main control room/control support area HVAC subsystem that maintain the integrity of the main control room/control support area pressure boundary during conditions of abnormal airborne radioactivity are tested for leak tightness in accordance with ASME N510, Section 6. Testing for main control room/control support area inleakage during Main Control Room/Control Support Area HVAC Subsystem operation will be conducted in accordance with ASTM E741 (Reference 38). The remaining supply and return/exhaust ductwork, constructed in accordance with Equipment Class L requirements, is visually examined to validate ductwork construction and duct sealing methods in accordance with the applicable SMACNA standard (References 16, 17 or 34).

The main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS) is tested and inspected in accordance with ASME/ANSI AG-1-1997 and Addenda AG-1a-2000 (Reference 201), ASME N509-1989, ASME N510-1989, and Regulatory Guide 1.140.

The VBS is tested as separate components and as an integrated system. Surveillance tests are performed to monitor the condition of the system. Testing methods include:

- Visual inspection
- Housing leak tests
- Airflow capacity and distribution tests
- HEPA filter bank and adsorber bank in-place leak tests
- Air heater performance tests
- Laboratory testing of adsorbers

Testing is performed at the frequency provided in Table 1 of ASME N510-1989.

9.4.1.5 Instrumentation Applications

The nuclear island nonradioactive ventilation system is controlled by the plant control system except for the main control room isolation valves, which are controlled by the protection and safety monitoring system. Refer to Subsection 7.1.1 for a description of the plant control and plant safety and monitoring systems. The instruments discussed below satisfy Table 4.2 of ASME N509 (Reference 2).

Temperature controllers are provided in the supply and return air ducts to control the room air temperatures within the predetermined ranges. Temperature indication and alarms for the main control room return air, Class 1E electrical room return air, air handling unit supply air, supplemental

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filtration unit prefilter inlet air and charcoal adsorbers are provided to inform plant operators of abnormal temperature conditions.

Pressure differential indication and alarms are provided across each filter bank (except charcoal filters) to inform plant operators when filter changeout is necessary. Pressure differential indication and alarms are provided to control the main control room and monitor the control support area ambient room pressure differentials with respect to surrounding areas.

Radioactivity indication and alarms are provided to inform the main control room operators of gaseous, particulate, and iodine radioactivity concentrations in the main control room supply air duct. See Section 11.5 for a description of the main control room supply air duct radiation monitors and their actuation functions.

Smoke detectors are provided to detect smoke in the outside air intake duct to the main control room and the main control room and Class 1E electrical room return air ducts.

Airflow indication and alarms are provided to monitor operation of the supply and exhaust fans.

Relative humidity indication and alarms are provided to monitor the average relative humidity in the return air from the main control room/CSA areas and the inlet air to the supplemental air filtration unit charcoal filters.

Status indication is provided to monitor fans, heaters and controlled dampers.

9.4.2 Annex/Auxiliary Buildings Nonradioactive HVAC System

The annex/auxiliary buildings nonradioactive HVAC system serves the nonradioactive personnel and equipment areas, office areas, fire brigade locker room, radiation protection calibration lab, electrical equipment rooms, clean corridors, the ancillary diesel generator room and demineralized water deoxygenating room in the annex building, and the main steam isolation valve compartments, reactor trip switchgear rooms, and piping and electrical penetration areas in the auxiliary building.

9.4.2.1 Design Basis

9.4.2.1.1 Safety Design Basis

The annex/auxiliary buildings nonradioactive HVAC system serves no safety-related function and therefore has no nuclear safety design basis. System equipment and ductwork located in the nuclear island whose failure could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portion of the system is nonseismic.

9.4.2.1.2 Power Generation Design Basis

The annex/auxiliary buildings nonradioactive HVAC system provides the following specific functions:

- Provides conditioned air to maintain acceptable temperatures for equipment and personnel working in the area
- Provides suitable environmental conditions for equipment in the main steam isolation valve (MSIV) compartments
- Prevents the buildup of hydrogen in non-Class 1E battery rooms to less than 2 percent hydrogen by volume

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- Removes vitiated air from locker, toilet, shower facilities, and rest rooms

The system maintains the following room temperatures based on maximum and minimum normal outdoor air temperature conditions shown in Chapter 2, Table 2-1:

Room or Area	Temperature (°F)
Normal Operation	
Offices, office areas, conference rooms, corridors (annex building)	73-78
Locker rooms, toilet rooms (annex building)	73-78
Security rooms and areas (annex building)	65-78
Non-Class 1E battery rooms (annex building)	67-77
Switchgear and battery charger rooms (annex building)	50-105
HVAC and mechanical equipment rooms (annex building)	50-105
MSIV compartments (auxiliary building)	50-105
Non-safety electrical penetration rooms (auxiliary building)	50-85
Reactor trip SWGR rooms (auxiliary building)	50-105
Valve/piping penetration room (auxiliary building)	50-105
Ancillary diesel generator room (annex building)	50-105
Demineralized water deoxygenating room	50-105
Boric acid batching room	50-105
Fire brigade locker room (annex building)	65-78
Radiation protection calibration lab (annex building)	65-78
Corridor (annex building room 40386)	65-78
Upset Conditions (Loss of Normal ac Power and Offsite Power)	
Switchgear rooms (annex building)	122 (maximum)
Battery charger rooms (annex building)	122 (maximum)
Ancillary diesel generator room (annex building - DG sets operating)	122 (maximum)

9.4.2.2 System Description

The annex/auxiliary buildings nonradioactive HVAC system consists of the following independent subsystems:

- General area HVAC subsystem
- Switchgear room HVAC subsystem
- Equipment room HVAC subsystem
- MSIV compartment HVAC subsystem
- Mechanical equipment areas HVAC subsystem
- Valve/Piping penetration room HVAC subsystem

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The defense in depth portion of the system and selected subsystems are shown in Figure 9.4.2-1.

9.4.2.2.1 General Description

9.4.2.2.1.1 General Area HVAC Subsystem

The general area HVAC subsystem serves personnel areas in the annex building outside the security area. The general area HVAC subsystem consists of two 50-percent capacity supply air handling units of about 2,300 scfm each (referred to as the general area air handling units) and two 50-percent capacity supply air handling units of about 5,540 scfm each (referred to as the office area air handling units), a humidifier, a ducted supply and return air system, diffusers and registers, exhaust fan, automatic controls, and accessories. The general area air handling units service the ALARA briefing room and operational support center, offices, computer rooms, and shower areas. The office area air handling units service the fire brigade locker room, radiation protection calibration lab, office areas, kitchen and rest rooms. The air handling units are located on the low roof of the annex building at elevation 117'-6". The units discharge into ducted supply distribution systems which are routed through the building to provide air into the various rooms and areas served via registers. Electric heating coils are provided in select branch supply ducts for tempering the supply air.

Air from the men's and women's locker, toilet, and shower facilities in the annex building is exhausted directly to atmosphere by exhaust fans. Room air from the remaining areas served is recirculated back to the air handling unit via a ceiling return plenum and a return duct system. Outside make-up air is added to the return air stream at the air handling units to replace air exhausted from toilets and showers in the area served.

9.4.2.2.1.2 Switchgear Room HVAC Subsystem

The switchgear room HVAC subsystem serves electrical switchgear Rooms 1 and 2 in the annex building. The switchgear room HVAC system consists of two 100 percent capacity air handling units, a ducted supply and return air system, and automatic controls and accessories.

The air handling units are located in the north air handling equipment room in the annex building at elevation 135′-3″. They are connected to a common intake plenum located along the east wall adjacent to their air handling equipment room. This plenum also supplies air for the equipment room HVAC subsystem. The air handling units discharge into a common duct distribution system that is routed through the building to the rooms served. Air is returned to the air handling units from the rooms served by a return duct system.

The switchgear room HVAC subsystem is designed so that smoke can be removed after a fire by placing the system in a once-through smoke exhaust ventilation mode. See Appendix 9A for identification of fire areas.

9.4.2.2.1.3 Equipment Room HVAC Subsystem

The equipment room HVAC subsystem serves electrical and mechanical equipment rooms in the annex and auxiliary buildings. These rooms include the non-Class 1E battery charger Rooms 1 and 2, the non-Class 1E battery Rooms 1 and 2, the reactor trip switchgear Rooms I and II, the non-Class 1E penetration room on elevation 100′-0″ and the non-Class 1E penetration room on elevation 117′-6″. This subsystem also serves the rooms and areas in the annex building. These include two rest rooms, access areas, and corridors. The equipment room HVAC system consists of two 100 percent capacity air handling units, three battery room exhaust fans, a toilet exhaust fan, a ducted supply and return air system, and automatic controls and accessories.

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The air handling units are located in the north air handling equipment room in the annex building at elevation 135′-3″. They are connected to a common intake plenum located along the east wall adjacent to their air handling equipment room. This plenum also supplies air for the switchgear room HVAC subsystem. The air handling units discharge into a common duct distribution system that is routed through the buildings to the various areas served. Air is returned to the air handling units from the rooms served (except the battery rooms and rest rooms) by a return duct system. Electric reheat coils are provided in the ductwork to areas requiring close temperature control such as the security rooms and restrooms. Hot water unit heaters (VXS-MY-W01A, B, and C) are provided in the north air handling equipment room to maintain the area above 50°F.

A humidifier is provided in the branch duct to the security areas to provide a minimum space relative humidity of 35 percent.

Each non-Class 1E battery room and the additional battery room in the annex building are provided with an individual exhaust system to prevent the buildup of hydrogen gas in the room. Each exhaust system consists of an exhaust fan, an exhaust air duct and gravity back draft damper located in the fan discharge. Air supplied to the battery rooms by the air handling units is exhausted to atmosphere. Air from the rest rooms is exhausted to atmosphere by a separate exhaust fan. Battery rooms 1 and 2 in the annex building also have electric duct heaters (VXS-EH-30 and VXS-EH-31).

The portion of the equipment room HVAC subsystem servicing the auxiliary building is designed so that smoke, hot gases, and fire suppressant will not migrate from one fire area to another to the extent that they could adversely affect safe shutdown capabilities, including operator actions. Fire or combination fire and smoke dampers are provided to isolate each fire area from adjacent fire areas during and following a fire in accordance with NFPA 90A (Reference 27) requirements. These combination smoke/fire dampers close in response to smoke detector signals or in response to the heat from a fire. See Appendix 9A for identification of fire areas.

9.4.2.2.1.4 MSIV Compartment HVAC Subsystem

The main steam isolation valve compartment HVAC subsystem serves the two main steam isolation valve compartments in the auxiliary building that contain the main steam and feedwater lines routed between the containment and the turbine building. Each compartment is provided with separate heating and cooling equipment.

The main steam isolation valve compartment HVAC subsystem consists of two 100-percent-capacity supply air handling units per compartment (VXS-MS-04A, B, C, and D) of about 5,300 scfm each with only low efficiency filters, ducted supply air distribution directly to the space served, automatic controls, and accessories for each main steam isolation valve compartment.

The supply air handling units are located directly within the space served. One unit in each compartment normally operates to maintain the temperature of the compartment. The air handling units can be connected to the standby power system, for investment protection, in the event of loss of the normal ac power and offsite power.

9.4.2.2.1.5 Mechanical Equipment Areas HVAC Subsystem

The mechanical equipment areas HVAC subsystem serves the ancillary diesel generator room, demineralized water deoxygenating room, boric acid batching room, upper south air handling equipment room, and lower south air handling equipment room in the annex building.

The mechanical equipment areas HVAC subsystem consists of two 50-percent capacity air handling units (VXS-MS-07A and B) with supply fans of about 1,360 scfm each, a ducted supply and return air system, automatic controls, and accessories.

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The air handling units are located in the lower south air handling unit equipment room on elevation 135′-3″ of the annex building. They are supplied from the air intake plenum #2 located at the extreme south end of the annex building between elevation 135′-3″ and 158′. This plenum also supplies air for the health physics and hot machine shop HVAC system and the containment air filtration system. The intake is not protected from tornado missiles.

The ancillary diesel generator room is supplied air from the air handling units to maintain normal design temperatures. Air supplied to the room is exhausted direct to outdoors by means of a separate exhaust fan. Ventilation and cooling for the ancillary diesel generator room when the diesel generators operate is provided by opening the doors to allow airflow into the room. Air from the ancillary diesel generator room is exhausted to the outdoors through the diesel generator radiator(s), ducting, backdraft damper(s), and hurricane louvers. There is a manual damper in the duct downstream of each diesel radiator that discharges air into the room. This manual damper can be used to control room temperature in cold weather.

9.4.2.2.1.6 Valve/Piping Penetration Room HVAC System

The valve/piping penetration room HVAC subsystem serves the valve/piping penetration room on elevation 100′-0″ of the auxiliary building. The valve/piping penetration room HVAC subsystem consists of two 100-percent-capacity air handling units (VXS-MS-08A and B) with supply fans of about 2,070 scfm each, a return air duct system, automatic controls, and accessories.

The air handling units are located directly within the space served.

9.4.2.2.2 Component Description

The annex/auxiliary buildings HVAC system is comprised of the following major components. These components are located in buildings on the Seismic Category I Nuclear Island or in the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Tables 9.4.2-1 and 9.4.2-2 provide the design parameters for major defense-in-depth components of the system.

Air Handling Units

Air handling units with integral supply and return/exhaust fans are utilized in the equipment room HVAC subsystem and switchgear room HVAC subsystem. Each air handling unit consists of a return/exhaust fan, a return/exhaust air plenum, a low efficiency filter bank, a high efficiency filter bank, a chilled water cooling coil, and a supply air fan.

Supply Air Handling Units

Supply air handling units are utilized in the general area HVAC subsystem, main steam isolation valve compartment HVAC subsystem, the valve/piping penetration room HVAC subsystem, and the mechanical equipment area HVAC subsystem. Each air handling unit consists of a low efficiency filter bank, a hot water heating coil, a chilled water cooling coil, and a supply fan. The general area HVAC subsystem and the mechanical equipment area HVAC subsystem air handling units each include a high efficiency filter bank that has a face and bypass damper on the heating coil.

Supply and Exhaust Air Fans

The supply and exhaust fans are centrifugal type, single width single inlet (SWSI), double width double inlet (DWDI), tubular centrifugal or plenum centrifugal with backward inclined blades that produce non-overloading horsepower characteristics or radial blades and forward curved blades that produce overloading horsepower characteristics. Only air handling unit fans that have littler or no ductwork may utilize forward curved blades. Fan configurations are either performance rated in accordance with ANSI/AMCA 210 (Reference 4) or certified in accordance with AMCA 211

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(Reference 5). Fan configurations are sound tested in accordance with ANSI/AMCA 300 (Reference 6).

Low Efficiency Filters and High Efficiency Filters

The low efficiency (25 percent) filters and high efficiency (80 percent) filters have a rated dust spot efficiency based on ASHRAE 52.1 and 126 (References 7 and 35). The filters are UL listed in accordance with UL 900 (Reference 8).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Heating Coils

The hot water heating coils are counterflow, finned tubular type. The heating coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Electric Heating Coils

The electric heating coils are SCR controlled, fin tubular type. The electric heating coils are UL listed in accordance with UL 1996 (Reference 26).

Electric Unit Heaters

The electric unit heaters are single or multi stage, fin tubular type. The electric unit heaters are UL listed in accordance with UL 2021 (Reference 40) and meet the requirements of the National Electric Code NFPA 70 (Reference 28).

Humidifier

The humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the supply duct system. The humidifier is performance rated in accordance with ARI 640 (Reference 13).

Hot Water Unit Heaters

The hot water unit heaters consist of a fan section and hot water heating coil section factory assembled as a complete and integral unit. The unit heaters are either horizontal discharge or vertical downblast type. The hot water unit heaters are tested and rated using the methods and standard conditions provided within ASHRAE Handbook - HVAC Systems and Equipment, “Unit Ventilators, Unit Heaters, and Makeup Air Units” (Reference 42).

Isolation Dampers

Isolation dampers are single- or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss-of-electrical power or loss-of-air pressure. The isolation dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

Shutoff, Volume Control, Balancing, and Backdraft Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, volume control and balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through ventilators, exhaust fans and the valve/piping penetration room air handling units. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

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Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15). Fire dampers are not provided in locations where combination fire/smoke dampers are provided.

Fire dampers at duct penetrations through fire barriers, which are within ducting that is routed through the CSA Pressure Boundary are Class R and are designed, constructed, and tested in accordance with ASME AG-1 Section DA (Reference 36). The fire damper assembly (sleeve) is bubble tight at design pressure. The use of Class R fire dampers is applied to protect and maintain the main control/control support area habitability. The use of Class R fire dampers mitigates the potential for exfiltration/infiltration of air in ducting which routed through the CSA pressure boundary.

Combination Fire/Smoke Dampers

Combination fire/smoke dampers are provided at the duct penetrations through fire barriers between the annex building and the auxiliary building, at the duct penetrations through fire barriers to the ICC/ Non-1E penetration room, the Staging and Storage Area/Electrical Switchgear Rooms, the Staging and Storage Area/Annex Building (North) Air Handling Equipment Room, Annex Building (North) Air Handling Equipment Room/Battery Charger Rooms, Non-1E Penetration Room/Reactor Trip Switchgear Rooms to maintain the fire resistance ratings of the barriers. The combination fire/smoke dampers are designed, constructed, and tested in accordance with UL-555S (Reference 25) and UL-555 (Reference 15).

The combination fire/smoke damper assemblies (sleeves) at duct penetration through fire barrier to the Staging and Storage Area I Electrical Switchgear Rooms and the Staging and Storage Area I Annex Building (North) Air Handling Equipment Room are bubble tight at design pressure and Class R. The use of Class R combination fire/smoke damper assemblies mitigates the potential for exfiltration/infiltration of air in ducting, during operation of the HVAC subsystem, coincident with an airborne release from the spaces the subsystem ductwork is routed through.

Ductwork, Duct Supports, and Accessories

Non-Seismic ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17) with the exception of Class R ductwork as noted below:

Ductwork which routed through the Staging and Storage Area (RCA) or the CSA Pressure Boundary is Class R and is designed, constructed, and tested in accordance with ASME AG-1 Section SA (Reference 36).

The Class R ductwork (ASME AG-1 Section SA) is applied to promote an ALARA HVAC design and to maintain main control room/control support area habitability.

Seismic Category II ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17) and Appendix 3A.

Ductwork is constructed of materials consistent with these standards including, but not limited to, galvanized or stainless steel. Duct supports and accessories are constructed of material consistent with these standards including, but not limited to, galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

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9.4.2.2.3 System Operation

9.4.2.2.3.1 General Area HVAC Subsystem

Normal Plant Operation

During normal plant operation, all four supply air handling units and the toilet/shower and rest room exhaust fans operate continuously to maintain suitable temperatures in the areas served. Cooling is provided via the supply of VWS chilled water to the air handling units' cooling coils. Heating is provided via a supply of VYS water to the air handling units' heating coils during times when the VYS system has been manually actuated (Refer to Section 9.2.10.2). The temperature of the air supplied by the VXS-MS-01A/B air handling units is controlled by temperature controls with their sensors located in the annex building main return air plenum. For air handlers VXS-MS-03A/B, the temperature of the supply air is controlled by a sensor immediately downstream of the air handler. The temperature sensors send a signal to their temperature controller which automatically modulates the chilled water control valve and the face and bypass dampers across the supply air heating coil to maintain the area within the design range.

Supplemental heating is provided for the men's/women's change room areas by an electric reheat coil located in the supply air duct to the areas served. The reheat coil operates intermittently under the control of its temperature controller with sensor located in the women's change room, which modulates the electric heating elements to maintain the space temperature in the change room areas within the design range.

For the general area air handling units, the supply air is humidified by a common humidifier located in the ductwork downstream of the supply air handling units. A humidistat located in the annex building operates the humidifier to maintain a minimum space relative humidity of 35 percent in the areas served. For the office area air handling units, a humidifier is not provided in the system.

The differential pressure drop across each supply unit filter bank is monitored, and individual alarms are actuated when any pressure drop rises to a predetermined level indicative of the need for filter replacement. To replace the filters on a supply unit, the affected supply fan is stopped and isolated from the duct system by means of isolation dampers. The exhaust fan for the area is also stopped, which may result in a slightly positive pressure with respect to adjacent areas or the outside atmosphere. During filter replacement, the system operates at approximately 50 percent capacity.

Abnormal Plant Operation

The general area HVAC subsystem is not required to operate during any abnormal plant condition. The general area HVAC subsystem outside air intake/exhaust dampers close on a “High-2” particulate or iodine signal from the MCR radiation package communicated through the plant control system (PLS).

9.4.2.2.3.2 Switchgear Room HVAC Subsystem

Normal Plant Operation

During normal plant operation, one air handling unit operates continuously to maintain the indoor temperatures in the two switchgear rooms. Cooling is normally provided via the supply of VWS chilled water to the operating air handling unit’s cooling coil. The temperature of the air supplied by the air handling unit is maintained at setpoint temperature by the temperature controller installed in the supply side of the air handling unit. The temperature controller modulates the chilled water valve. Temperature controllers located in the switchgear rooms provide low limit temperature control. When the temperature of a switchgear room drops to the low temperature setpoint, the controller will limit the travel of the chilled water valve serving the air handling unit cooling coil to prevent the room from

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falling below room design minimum temperature. A minimum of outside air is always provided for ventilation requirements.

The differential pressure drop across each air handling unit filter bank is monitored and individual alarms are actuated when the pressure drop rises to a predetermined level indicative of the need for filter replacement. To replace the filters on an air handling unit, the unit is stopped and isolated from the duct system by means of isolation dampers. During filter replacement, the second air handling unit operates at full system capacity.

Abnormal Plant Operation

The switchgear room HVAC subsystem outside air intake/exhaust dampers close on a “High-2” particulate or iodine signal from the MCR radiation package communicated through the plant control system (PLS). In the event of a loss of the normal ac power and offsite power, the air handling unit supply and return/exhaust fans are connected to the standby power system to provide ventilation cooling to the diesel bus switchgear. This cooling permits the switchgear to perform its defense in depth functions in support of standby power system operation. In this mode of operation, the switchgear rooms are cooled using outside air. In this mode of operation, outside air is admitted via the economizer dampers (outside air dampers, return air dampers, and exhaust air dampers), which are modulated by the temperature controller installed in the supply side of the air handling unit. The switchgear rooms will be maintained between 62 and 122°F.

In the event of a fire in a non-1E electrical switchgear room, the combination fire/smoke dampers close automatically to isolate the affected fire area in response to heat from the fire or upon receipt of a smoke signal from an area smoke detector. The VXS subsystem continues to provide ventilation/ cooling to the remaining switchgear room.

9.4.2.2.3.3 Equipment Room HVAC Subsystem

Normal Plant Operation

During normal plant operation, one air handling unit and both battery room exhaust fans operate continuously to maintain the indoor temperatures in the equipment and security access areas served by the system.

The temperature of the air supplied by the air handling unit is maintained at setpoint temperature by the temperature controller installed in the supply side of the air handling unit. The temperature controller modulates the outside air dampers, return air dampers, exhaust air dampers, and chilled water valve. Cooling is normally provided via the supply of VWS chilled water to the air handling unit’s cooling coil. A minimum of outside air is always provided for ventilation requirements.

Electric reheat coils serving security (rooms 40305 and 40306) are controlled by temperature controllers with sensors located in the areas served. The temperature sensor sends a signal to a temperature controller which modulates the electric heating elements to maintain the security access areas at their design temperatures. Hot water unit heaters operate intermittently to provide supplemental heating for the north air handling equipment room to maintain the area temperature above 50°F.

A humidistat located in the security access area intermittently operates the humidifier to maintain the security office area at a minimum space relative humidity of 35 percent.

The differential pressure drop across each air handling unit filter bank is monitored, and individual alarms are actuated when the pressure drop rises to a predetermined level indicative of the need for filter replacement. To replace the filters of an air handling unit, the unit is stopped and isolated from

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the duct system by means of isolation dampers. During filter replacement, the second air handling unit operates at full system capacity.

Abnormal Plant Operation

The equipment room HVAC subsystem outside air intake/exhaust dampers close on a “High-2” particulate or iodine signal from the MCR radiation package communicated through the plant control system (PLS). In the event of a loss of the normal ac power and offsite power, the air handling unit supply, the return/exhaust fans, and the economizer dampers are connected to the standby power system to provide ventilation cooling to the dc switchgear and inverters. This cooling permits that equipment to perform its defense in depth functions. In this mode of operation, the rooms are cooled using outside air. In this mode of operation, outside air is admitted via the economizer dampers (outside air dampers, return air dampers, and exhaust air dampers), which are modulated by the temperature controller installed in the supply side of the air handling unit. In this mode of operation, the dc switchgear and inverter areas will be maintained at or below 122°F. Equipment in those areas that operate following a loss of the normal ac power and offsite power are designed for continuous operation at this temperature. To maintain the areas above freezing, the economizer dampers will modulate to maintain a supply air temperature of 52°F for outdoor temperatures below 52°F. For outdoor temperature above 52°F, the outside air, return air, and exhaust air dampers are positioned for a once-through flow.

9.4.2.2.3.4 MSIV Compartment HVAC Subsystem

Normal Plant Operation

During normal plant operation, one of the main steam isolation valve compartment air handling units in each compartment operates continuously in a recirculation mode to maintain the indoor temperature in the equipment area served by the system. Cooling is provided via the supply of VWS chilled water to the air handling units' cooling coils. Heating is provided via a supply of VYS water to the air handling units' heating coils during times when the VYS system has been manually actuated (Refer to Section 9.2.10.2). A temperature controller automatically modulates the chilled water and hot water control valves serving the operating unit to maintain the compartment temperature at or less than 105°F and above a minimum of 50°F.

The differential pressure drop across each air handling unit filter bank is monitored and individual alarms are actuated when the pressure drop rises to a predetermined level indicative of the need for filter replacement. An air handling unit may be shutdown for filter replacement or other maintenance as required, with the other air handling unit in the same compartment operating to maintain the area temperature.

Abnormal Plant Operation

The main steam isolation valve compartment HVAC subsystem is not required to operate during abnormal plant conditions.

9.4.2.2.3.5 Mechanical Equipment Areas HVAC Subsystem

During normal plant operation, the air handling units operate continuously to maintain the indoor temperatures in the areas served. Cooling is provided via the supply of VWS chilled water to the air handling units' cooling coils. Heating is provided via a supply of VYS water to the air handling units' heating coils during times when the VYS system has been manually actuated (Refer to Section 9.2.10.2). The temperature of the air supplied by each air handling unit is controlled by individual temperature controls with their sensors located in the upper south air handling equipment room. The temperature sensor sends a signal to a temperature controller which automatically modulates the face and bypass dampers across the supply air heating coil and the chilled water

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control valve to maintain the mechanical equipment areas within the design temperature range. A constant volume of outside air is used to provide ventilation.

Differential pressure drop across each air handling unit filter bank is monitored, and individual alarms are actuated when pressure drop rises to a predetermined level indicative of the need for filter replacement. During filter replacement, the system operates at approximately 50 percent capacity. To replace the filters of an air handling unit, the unit is stopped and isolated from the duct system by means of isolation dampers. To replace the filters of an air handling unit, the unit is stopped and isolated from the duct system by means of isolation dampers.

The exhaust fan for the ancillary diesel generator room operates continuously for room ventilation.

Abnormal Plant Operation

The mechanical equipment areas HVAC subsystem is not required to operate during abnormal plant conditions.

When the ancillary diesel generator sets are operated, a manual damper is opened as required and the outside door is opened to maintain acceptable temperatures.

9.4.2.2.3.6 Valve/Piping Penetration Room HVAC Subsystem

Normal Plant Operation

During normal plant operation, one air handling unit operates continuously in a recirculation mode to maintain the indoor temperature in the room. Cooling is provided via the supply of VWS chilled water to the air handling units' cooling coils. Heating is provided via a supply of VYS water to the air handling units' heating coils during times when the VYS system has been manually actuated (Refer to Section 9.2.10.2). A temperature controller automatically modulates the chilled water control valve and opens and closes the hot water control valve serving the operating unit to maintain the area temperature at or less than 105°F and above a minimum of 50°F.

The differential pressure drop across each air handling unit filter bank is monitored, and individual alarms are actuated when the pressure drop rises to a predetermined level indicative of the need for filter replacement.

Abnormal Plant Operation

The valve/piping penetration room HVAC subsystem is not required to operate during abnormal plant conditions.

9.4.2.3 Safety Evaluation

The annex/auxiliary buildings nonradioactive HVAC system has no safety-related function and therefore requires no nuclear safety evaluation.

9.4.2.4 Tests and Inspections

The annex/auxiliary buildings nonradioactive HVAC system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of each system are accessible for periodic inspection during normal plant operation. A system air balance test and adjustments to design conditions are made during the plant preoperational test program. Air flow rates are measured and balanced in accordance with the guidelines of SMACNA HVAC Systems – Testing, Adjusting, and Balancing (Reference 19). Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability.

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9.4.2.5 Instrumentation Applications

The annex/auxiliary buildings nonradioactive HVAC system operation is controlled by the plant control system (PLS). Refer to Subsection 7.1.1 for a discussion of the plant control system.

Temperature controllers and thermostats maintain the proper space temperatures. Supply air temperature is controlled by either sensing local room temperature or by sensing the supply air temperature in the air handling unit discharge duct, depending on the subsystem. Unit heaters are controlled by local thermostats. Temperature indication and alarms are accessible locally via the plant control system.

Temperature is indicated for each air handling unit supply air discharge duct, except for local recirculation units such as those in the main steam isolation valve compartment and valve/piping penetration room.

Operational status of fans is indicated in the main control room. The fans and air handling units can be placed into operation or shutdown from the main control room or locally.

Differential pressure indication is provided for each of the filters in the air handling units and an alarm for high pressure drop is provided for each air handling unit.

Airflow is indicated for the air handling unit and exhaust fan discharge ducts. Alarms are provided for low air flow rates in the fan discharge ducts.

An alarm is provided for smoke in discharge ducts from the air handling units.

Position indicating lights are provided for automatic dampers.

9.4.3 Radiologically Controlled Area Ventilation System

The radiologically controlled area ventilation system (VAS) serves the fuel handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas which are provided with a separate ventilation system (VHS).

9.4.3.1 Design Basis

9.4.3.1.1 Safety Design Basis

The radiologically controlled area ventilation system serves no safety-related function and therefore has no nuclear safety design basis. System equipment and ductwork located in the nuclear island whose failure could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portion of the system is nonseismic.

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9.4.3.1.2 Power Generation Design Basis

The radiologically controlled area ventilation system provides the following functions:

- Provides ventilation to maintain the equipment rooms within their design temperature range
- Provides ventilation to maintain airborne radioactivity in the access areas at safe levels for plant personnel
- Maintains the overall airflow direction within the areas it serves from areas of lower potential airborne contamination to areas of higher potential contamination
- Maintains each building area at a slightly negative pressure to prevent the uncontrolled release of airborne radioactivity to the atmosphere or adjacent clean plant areas
- Automatically isolates selected building areas from the outside environment by closing the supply and exhaust duct isolation dampers and starting the containment air filtration system when high airborne radioactivity in the exhaust air duct or high ambient pressure differential is detected. See Subsection 9.4.7 for a description of the containment air filtration system.

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The system maintains the following room temperatures based on the maximum and minimum normal outside air temperature conditions shown in Chapter 2, Table 2-1:

Access and Equipment Areas	Temperatures (°F)
Auxiliary/Annex Building Subsystem	
Degasifier column	50-130
RNS and CVS pump rooms (pumps not operating)	50-104
RNS and CVS pump rooms (pumps operating)	50-130
Containment purge exhaust filter rooms (fans not operating)	50-104
Containment purge exhaust filter rooms (fans operating)	50-130
Liquid radwaste tank rooms	50-130
Liquid radwaste pump rooms	50-104
HVAC equipment room	50-104
Gaseous radwaste equipment rooms	50-104
Spent fuel pool pump and heat exchanger rooms	50-104
Annex building staging and storage area	50-104
Demineralizer/filter access area and corridor 12261	50-85
VES air storage room	60-80
Other corridors and staging areas	50-104
Fuel Handling Area Ventilation Subsystem	
Rail car bay/filter storage area	50-104
Spent resin equipment rooms	50-130
Corridors and access areas	50-104
Occupied Areas	
Fuel Handling Area Ventilation Subsystem	
Fuel handling area	50-96
Auxiliary/Annex Building Ventilation Subsystem	
Radiation chemistry laboratory	73-78
Primary sample room	50-104
Security rooms	65-78

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9.4.3.2 System Description

The radiologically controlled area ventilation system consists of the following subsystems:

- Auxiliary/annex building ventilation subsystem
- Fuel handling area ventilation subsystem

The defense in depth portion of the system is shown in Figure 9.4.3-1.

9.4.3.2.1 General Description

9.4.3.2.1.1 Auxiliary/Annex Building Ventilation Subsystem

The auxiliary/annex building ventilation subsystem serves radiologically controlled equipment, piping and valve rooms and adjacent access and staging areas. See Figure 9.4.3-1, sheet 2 of 4, for a complete listing of rooms and corridors serviced by this subsystem. The auxiliary/annex building ventilation subsystem consists of two 50 percent capacity supply air handling units of about 14,000 scfm each, a ducted supply and exhaust air system, isolation dampers, diffusers and registers, exhaust fans, automatic controls and accessories. The supply air handling units are located in the south air handling equipment room of the annex building at elevation 158′-0″. They are connected to the air intake plenum #3 located in the extreme south end of the annex building. This plenum is not protected from tornado missiles. The units discharge into a ducted supply distribution system which is routed through the radiologically controlled areas of the auxiliary and annex buildings. The supply and exhaust ducts have isolation dampers that close to isolate the auxiliary and annex buildings from the outside environment when high airborne radioactivity is detected in the exhaust air duct. The supply and exhaust ducts are configured so that two building zones may be independently isolated. The annex building staging and storage area, containment air filtration exhaust rooms, containment access corridor, and adjacent auxiliary building staging, equipment areas, middle annulus, middle annulus access room, and security rooms are aligned to one zone. A radiation monitor is located in the exhaust air duct from the annex building areas in this zone, and a second radiation monitor is located in the exhaust air duct from the auxiliary building areas in this zone. The other zone includes the remaining rooms and corridors shown in Figure 9.4.3-1 sheet 2 of 4, including but not limited to the radiation chemistry laboratory, primary sample room, spent fuel pool cooling water pump and heat exchanger rooms, normal residual heat removal pump and heat exchanger rooms, CVS makeup pump room, lower annulus, and various radwaste equipment rooms, pipe chases, and access corridors. Figure 9.4.3-1, sheet 4 of 4, depicts the recirculating air cooling units for the security rooms. A radiation monitor is located in the exhaust air duct from this zone.

The two 50 percent capacity exhaust air fans sized to allow the system to maintain a negative pressure are located in the upper radiologically controlled area ventilation system equipment room at elevation 145′-9″ of the auxiliary building. The exhaust air ductwork is routed to minimize the spread of airborne contamination by directing the supply airflow from the low radiation access areas into the radioactive equipment and piping rooms with a greater potential for airborne radioactivity. Additionally, the exhaust air ductwork is connected to the radioactive waste drain system (WRS) sump to maintain the sump atmosphere at a negative air pressure to prevent the exfiltration of potentially contaminated air into the surrounding area. The sump vent line is constructed of pipe, which is routed upward from the sump to the interface with the HVAC system. The exhaust air ductwork is connected to the radwaste effluent holdup tanks to prevent the potential buildup of airborne radioactivity or hydrogen gas within these tanks. The effluent holdup tanks have an overflow line to the WRS from a nozzle on the side of the tank, and a vent line from a nozzle on the top of the tank. The vent line is constructed of pipe, and is routed upward from the tank to the interface with the HVAC system. The exhaust fans discharge the exhaust air into the plant vent for monitoring of offsite airborne radiological releases.

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The ventilation airflow dilutes potential airborne contamination to maintain the concentration at the site boundary within 10 CFR 20 (Reference 21) allowable effluent concentration limits and the internal room airborne concentrations within 10 CFR 20 occupational derived air concentration (DAC) limits during normal plant operation.

Unit coolers are located in the normal residual heat removal system (RNS) and chemical and volume control system (CVS) pump rooms because they have significant cooling loads on an intermittent basis when large equipment is operating. Each unit cooler is sized to accommodate 100 percent of its corresponding pump cooling load. The unit coolers are provided with chilled water from redundant trains of the central chilled water system (VWS) low capacity subsystem. The normal residual heat removal pump room unit coolers have two cooling coils per unit cooler so that chilled water supplied by either train A or train B alone can support concurrent operation of both normal residual heat removal system pumps. The two chemical and volume control makeup pump room unit coolers are connected to redundant trains of the chilled water system; however, operation of either the train A or train B unit cooler alone maintains the common makeup pump room temperature conditions and supports operation of either makeup pump. Condensation from these cooling coils drains to the WRS.

Heating coils are located in the supply air ducts serving plant areas that require supplemental heating during periods of cold outside air temperature conditions. The heating coils are supplied with hot water from the hot water heating system (VYS). The radiation chemistry laboratory is provided with an electric duct heater and humidifier in the supply air duct. Security Room (Room 12451) is provided with an electric duct heater and humidifier in the supply air duct and two local fan coil cooling units. Security Room (Room 40415) is provided with an electric duct heater in the supply air duct. These components are provided to maintain the environmental conditions within the areas suitable for personnel comfort. Electric unit heaters provide supplemental heating in the middle annulus.

The upper annulus is separated from the middle annulus area of the auxiliary building by a concrete floor section and flexible seals that connects the containment steel shell to the shield building. The annulus seal provides a passive barrier during normal plant operation or when the auxiliary building is isolated, preventing the exfiltration of unmonitored releases from the middle annulus to the environment.

9.4.3.2.1.2 Fuel Handling Area Ventilation Subsystem

The fuel handling area ventilation subsystem serves the fuel handling area, rail car bay/filter storage area, resin transfer pump/valve room, spent resin tank room, waste disposal container area, and WSS (spent resin) valve/piping area. The fuel handling area ventilation subsystem consists of two 50 percent capacity supply air handing units of about 4,900 scfm each, a ducted supply and exhaust air system, isolation dampers, diffusers, registers, exhaust fans, automatic controls and accessories. Hot water heating coils supplied with water from the hot water heating system (VYS), cooling coils supplied with water from the central chilled water system (VWS) and electric duct heaters are used to maintain ambient room temperatures within the normal range. The ventilation airflow capacity is designed to maintain environmental conditions that support worker efficiency during fuel handling operations based on a maximum wetbulb globe temperature of 80°F (96°F drybulb) as defined by EPRI NP-4453 (Reference 22). The supply air handling units are located in the south air handling equipment room of the annex building at elevation 158′-0″. They are connected to the air intake plenum #3 located at the south end of the annex building. This common intake plenum is described in Subsection 9.4.3.2.1.1. The units discharge into a ducted supply distribution system which is routed to the fuel handling and rail car bay/filter storage areas of the auxiliary building. The supply and exhaust ducts are provided with isolation dampers that close when high airborne radioactivity in the exhaust air or high pressure differential with respect to the outside atmosphere is detected.

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The two 50 percent capacity exhaust air fans sized to allow the system to maintain a negative pressure are located in the upper radiologically controlled area ventilation system equipment room at elevation 145′-9″ of the auxiliary building. The supply and exhaust ductwork is arranged to exhaust the spent fuel pool plume and to provide directional airflow from the rail car bay/filter storage area into the spent resin equipment rooms. The exhaust fans discharge the normally unfiltered exhaust air into the plant vent for monitoring of offsite airborne gaseous and other radiological releases.

The ventilation airflow dilutes potential airborne contamination to maintain the concentration at the site boundary within 10 CFR 20 (Reference 21) allowable effluent concentration limits and the internal room airborne concentrations within 10 CFR 20 occupational derived air concentration (DAC) limits during normal plant operation.

9.4.3.2.2 Component Description

The radiologically controlled area ventilation system is comprised of the following major components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Table 9.4.3-1 provides design parameters for major defense in depth components in the system.

Supply Air Handling Units

Each supply air handling unit consists of a low efficiency filter bank, a high efficiency filter bank, a hot water heating coil bank, a chilled water cooling coil bank, and a supply fan.

Supply and Exhaust Air Fans

The supply and exhaust air fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with backward inclined blades that produce non-overloading horsepower characteristics or radial and forward curved blades that produce overloading horsepower characteristics. Fan configurations are either performance rated in accordance with ANSI/AMCA 210 (Reference 4) or certified in accordance with AMCA 211 (Reference 5). Fan configurations are sound tested in accordance with ANSI/AMCA 300 (Reference 6).

Fan Coil Cooling Units

Each fan coil cooling unit consists of a low efficiency filter, cooling coil and supply fan.

Unit Coolers

Each unit cooler consist of a low efficiency filter bank, a chilled water cooling coil bank and a supply fan. The normal residual heat removal system pump room unit coolers have redundant cooling coil banks. The principal construction code is the manufacturer's standard.

Low and High Efficiency Filters

The low efficiency (25 percent) filters and high efficiency (80 percent) filters have a rated dust spot efficiency based on ASHRAE 52.1 and 126 (References 7 and 35). The filters minimum average dust spot efficiencies for the defense in depth filters are shown in Table 9.4.3-1. The filters are UL listed in accordance with UL 900 (Reference 8).

Electric Unit Heaters

The electric unit heaters are single or multi stage, fin tubular type. The electric unit heaters are UL listed in accordance with UL 2021 (Reference 40) and meet the requirements of the National Electric Code NFPA 70 (Reference 28).

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Hot Water Heating Coils

The hot water heating coils are finned tubular type. The outside supply air heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating the heat output. Coils are performance rated in accordance with ANSI/ARI 410 (Reference 12).

Electric Heating Coils

The electric heating coils are SCR controlled, fin tubular type. The electric heating coils are UL listed in accordance with UL 1996 (Reference 26).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifier

The humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the supply duct system. The humidifier is performance rated in accordance with ARI 640 (Reference 13).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance rating of the barriers. The fire dampers meet the design, testing and installation requirements of UL-555 (Reference 15).

Shutoff and Balancing Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow.

Shutoff dampers downstream of exhaust fans, which discharge air into the plant vent stack at a positive pressure, are Class R and are designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section DA. The Class R dampers are applied to promote an ALARA HVAC design. The use of Class R dampers mitigates the potential for exfiltration of airborne radiation in a positive pressure duct during operation of the fan coincident with an airborne release from the spaces the fans serve.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

Isolation Dampers

Isolation dampers are single- or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power or loss of air pressure.

Isolation dampers which isolate contaminated air streams are Class R. The Class R isolation dampers have bubble tight (frame/blades) and are designed, constructed, and tested in accordance with ASME AG-1 Section DA (Reference 36).

The isolation dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

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Ductwork, Duct Supports, and Accessories

Non-Seismic ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17).

Seismic II Category ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34), SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17), and Appendix 3A, with the exception of the Class R ductwork as noted below.

Ductwork downstream of exhaust fans, which discharge air into the plant vent stack at a positive pressure, is Class R and is designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section SA, and Appendix 3A. The Class R ductwork (ASME AG-1 Section SA) is applied to promote an ALARA HVAC design. The use of Class R ductwork mitigates the potential for exfiltration of airborne radiation in a positive pressure duct during operation of the fan coincident with an airborne release from the spaces the fan serves.

Ductwork is constructed of materials consistent with these standards including, but not limited to, galvanized or stainless steel. Duct supports and accessories are constructed of material consistent with these standards including, but not limited to, galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

9.4.3.2.3 System Operation

9.4.3.2.3.1 Auxiliary/Annex Building Ventilation Subsystem

Normal Plant Operation

During normal plant operation, both supply air handling units and both exhaust fans operate continuously to ventilate the areas served on a once-through basis. The inlet vanes of the supply fans are modulated automatically by flow controllers to maintain a constant air flow rate. The exhaust air flow is then balanced to maintain the areas served at a slightly negative pressure differential with respect to the outside environment. The exhaust air is unfiltered and directed to the plant vent for discharge and monitoring of offsite gaseous releases.

The temperature of the supply air is controlled by temperature sensors located in the supply air ducts. When the supply air temperature is low, the face and bypass dampers across the supply air hot water heating coil are modulated to heat the supply air. When the supply air temperature is high, the flow of chilled water is modulated to cool the supply air. A temperature sensor or local thermostat modulates electric duct heaters located in the supply ducts serving personnel comfort areas to temper air and maintain environmental conditions. Building areas that have conductive heat loss to the outside environment during periods of cold outside temperature conditions are provided with electric unit heaters or supply duct hot water heating coils. Electric unit heaters are turned on/off by local thermostats. Temperature sensors control hot water heating (VYS) isolation valves which regulate flow to the supply duct hot water heating coils. The ventilation air is continuously monitored by smoke detectors located in the ductwork downstream of the supply air handling units and upstream of the exhaust fans.

A supply air handling unit is automatically shut down if the supply air temperature is below a predetermined setpoint.

Each chemical and volume control system makeup pump and normal residual heat removal system pump unit cooler automatically starts on a high room temperature signal and stops on a low room temperature signal.

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The gaseous radwaste equipment areas have sufficient ventilation to remove hydrogen gas that may leak from the radwaste equipment into the equipment rooms to maintain the concentration of hydrogen below a safe level of about 1 percent. Instrumentation available to monitor hydrogen concentration is listed in Table 11.3-2.

Abnormal Plant Operation

If high airborne radioactivity is detected in the exhaust air from the auxiliary or annex buildings, the supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment. The containment air filtration system mitigates the exfiltration of unfiltered airborne radioactivity by maintaining the isolated zone at a slightly negative pressure with respect to the outside environment and adjacent unaffected plant areas. The auxiliary/annex building ventilation subsystem remains in operation at a reduced capacity if either the auxiliary or annex building is not isolated. A disruption in the normal ventilation airflow rate that causes a high pressure differential with respect to the outside environment causes the same automatic actuations. The containment air filtration system maintains a slightly negative pressure differential with respect to the outside environment until operation of the auxiliary/annex building ventilation subsystem is restored. Refer to Subsection 9.4.7 for a description of the containment air filtration system.

If smoke is detected in the exhaust air ducts, an alarm is initiated in the main control room. The auxiliary/annex building ventilation subsystem remains in operation unless plant operators determine that there is a need to manually shut down the subsystem. If smoke is detected in the supply air ducts, an alarm is initiated in the main control room and the supply and exhaust fans for the corresponding train are automatically shut down. In the event of a fire occurring within the auxiliary or annex buildings, local fire dampers automatically isolate the HVAC ductwork penetrating the fire area when the local air temperature exceeds predetermined setpoints.

In the event of a loss of the normal ac power and offsite power, the unit coolers serving the normal residual heat removal, and chemical and volume control pump rooms can be powered by the onsite standby diesel generators.

9.4.3.2.3.2 Fuel Handling Area Ventilation Subsystem

Normal Plant Operation

During normal plant operation, both supply air handling units and both exhaust fans operate continuously to ventilate the areas served on a once-through basis. The inlet vanes of the supply fans are modulated automatically by flow controllers to maintain a constant air flow rate. The exhaust air flow is then balanced to maintain the areas served at a slightly negative pressure differential with respect to the outside environment. The exhaust air is unfiltered and directed to the plant vent for discharge and monitoring of offsite gaseous releases.

The temperature of the supply air is controlled by temperature sensors located in the supply air ducts. When the supply air temperature is low, the face and bypass dampers across the supply air hot water heating coil are modulated to heat the supply air. A temperature sensor controls supplemental heating in the rail car bay/filter storage area by controlling a hot water (VYS) isolation valve which regulates flow to a supply duct heating coil. A temperature sensor controls supplemental heating in the Fuel Handling Area by modulating electric duct heaters located in the supply ducts. When the supply air temperature is high, the flow of chilled water is modulated to cool the supply air. The ventilation air is continuously monitored by smoke detectors located in the ductwork downstream of the supply air handling units and by a detector upstream of the exhaust fans.

A supply air handling unit is automatically shut down if the supply air temperature is below a predetermined setpoint.

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Abnormal Plant Operation

If high airborne radioactivity is detected in the exhaust air from the fuel handling area, the supply and exhaust duct isolation dampers automatically close to isolate the fuel handling area from the outside environment. The containment air filtration system mitigates exfiltration of unfiltered airborne radioactivity by maintaining the isolated zone at a slightly negative pressure differential with respect to the outside environment and adjacent unaffected plant areas. A disruption in the normal ventilation airflow rate that causes a high pressure differential with respect to the outside environment causes the same automatic actuations. The containment air filtration system maintains a slightly negative pressure differential with respect to the outside environment until operation of the fuel handling area ventilation subsystem is restored. Refer to Subsection 9.4.7 for a description of the containment air filtration system.

If smoke is detected in the exhaust air ducts, an alarm is initiated in the main control room. The fuel handling area subsystem remains in operation unless plant operators determine that there is a need to manually shut down the subsystem. If smoke is detected in the supply air ducts, an alarm is initiated in the main control room and the supply and exhaust fans for the corresponding train are automatically shut down. In the event of a fire occurring within the fuel handling area, fire dampers automatically isolate the HVAC ductwork penetrating this fire area when the local air temperature exceeds predetermined setpoints.

9.4.3.3 Safety Evaluation

The radiologically controlled area ventilation system has no safety-related function and therefore requires no nuclear safety evaluation.

The isolation dampers for the fuel handling area, auxiliary and annex buildings are provided to help keep normal plant releases below 10 CFR 20 (Reference 21) limits and 10 CFR 50 Appendix I (Reference 20) guidelines in the event of an abnormal release of airborne radioactivity.

9.4.3.4 Tests and Inspections

The radiologically controlled area ventilation system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of each system are accessible for periodic inspection during normal plant operation. A system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program. Airflow rates are measured and balanced in accordance with the guidelines of SMACNA HVAC Systems – Testing, Adjusting and Balancing (Reference 19). Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability.

9.4.3.5 Instrumentation Applications

The radiologically controlled area ventilation system is controlled by the plant control system (PLS). Refer to Subsection 7.1.1 for a discussion of the plant control system.

Temperature controllers maintain the proper air temperatures and provide indication and alarms. Main control room temperature indication is provided for the normal residual heat removal system pump rooms, and the chemical and volume control makeup pump room to allow room temperatures to be verified during pump operation without requiring personnel access to these rooms.

Operational status of fans and dampers is indicated in the main control room. Fans and air handling units can be placed into operation or shut down from the main control room.

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Differential pressure indication and high differential pressure alarms are provided for the filters in the air handling units and room coolers. Pressure differential indication and alarms are provided via instruments (VAS-030, VAS-032, VAS-033, and VAS-034) to monitor the pressure differential in the fuel handling area and the radiologically controlled areas of the auxiliary and annex buildings, and to control system operation in the event of equipment malfunction.

Radioactivity indication and alarms are provided to inform the main control room operators of gaseous radioactivity concentrations in the exhaust ducts from the fuel handling area and radiologically controlled areas of the auxiliary and annex buildings.

Flow indication and alarms are provided to alert plant operators to equipment malfunctions. Smoke alarms are provided.

9.4.4 Balance-of-Plant-Interface

Not applicable to AP1000.

9.4.5 Engineered Safety Features Ventilation System

Not applicable to AP1000.

9.4.6 Containment Recirculation Cooling System

The containment recirculation cooling system controls building air temperature and humidity to provide a suitable environment for equipment operability during normal operation and shutdown.

9.4.6.1 Design Basis

9.4.6.1.1 Safety Design Basis

The containment recirculation cooling system serves no safety-related function and therefore has no nuclear safety design basis. The containment recirculation system is not required to mitigate the consequences of a design basis accident or loss of coolant accident. System equipment and ductwork whose failure could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portion of the system is nonseismic.

9.4.6.1.2 Power Generation Design Basis

The containment recirculation cooling system provides the following functions:

- Controls the containment thermal environment to maintain an average bulk air temperature below 120°F during normal operation
- Controls the containment thermal environment to maintain an average bulk air temperature below 70°F and above 50°F for personnel accessibility and equipment operability during refueling and plant shutdown
- Maintains a homogeneous containment temperature and pressure during containment integrated leak rate testing (ILRT)
- Maintains a homogeneous containment temperature and pressure during a loss of the plant ac electrical system

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- Controls the reactor cavity area average concrete temperature to less than 150°F with a maximum local area temperature of 200°F

9.4.6.2 System Description

The containment recirculation cooling system is shown in Figure 9.4.6-1.

9.4.6.2.1 General Description

The containment recirculation cooling system is comprised of four 50 percent capacity fan coil unit assemblies which connect to a common duct ring header and distribution system. Each fan coil unit assembly contains a fan and associated cooling coil banks. The fan coil unit assemblies are located on two platforms at elevation 153'-0", with two fan coil unit assemblies occupying each platform. The platforms are located approximately 180 degrees apart to provide a proper return air and mixing pattern through the ring header. The top of the ring header is approximately at elevation 201'-6". The ring header and the fan assemblies are designed to provide uniform air and temperature distribution inside the containment, considering the possibility that one fan coil unit assembly may be out of service.

The cross-connections between the central chilled water system piping for containment cooling and hot water heating system piping for containment heating are located outside the containment. The water piping inside containment is common to both the central chilled water system and hot water heating system.

9.4.6.2.2 Component Description

The containment recirculation cooling system is comprised of the following components. These components are located in buildings on the Seismic Category I Nuclear Island. Table 9.4.6-1 provides design parameters for the major components of the system.

Containment Recirculation Fan Coil Unit Assemblies

Each 50 percent capacity fan coil unit assembly consists of a return air mixing plenum section with two cooling coils attached to each of the four sides of each plenum section. The cooling coils are counterflow finned tubular type. The cooling coils are rated and meet the performance requirements in accordance with ANSI/ARI 410 (Reference 12) and ASHRAE 33 (Reference 11).

The recirculation fans are vaneaxial upblast type, direct driven with adjustable blades and an inlet bell. The fans are mounted vertically on top of the mixing air plenum section. The fans are designed with a non-overloading two-speed motor. The high speed is used during normal operation and the low speed is used during high ambient air density operating conditions such as the integrated leak rate testing. Fan configurations are performance rated in accordance with ANSI/AMCA 210 (Reference 4) and sound tested in accordance with AHRI 575 (Reference 43).

Ductwork, Duct Supports, and Accessories

The ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction standards (References 16 and 34), SMACNA HVAC Duct Construction Standards – Metal and Flexible (Reference 17) and Appendix 3A. With the exception of nonmetallic flexible connectors, ductwork is constructed of either galvanized or stainless steel. The use of nonmetallic flexible connectors inside containment is limited to ductwork above the operating deck. Duct supports and accessories are constructed of galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

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Isolation, Balancing, and Backdraft Dampers

The isolation dampers are manually adjusted, parallel-blade type. Multiblade, balancing dampers are opposite-blade type. Backdraft dampers are provided to prevent reverse flow through the standby fan while the redundant fan is operating. The backdraft dampers also allow start up of the standby fan while the redundant fan remains in operation. The dampers are designed for the same differential pressure as the duct section in which they are located.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

Fire Damper

Fire damper is provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire damper meets the design and installation requirements of UL-555 (Reference 15).

9.4.6.2.3 System Operation

Normal Plant Operation

During normal plant operation, one of the two 50 percent capacity fan coil unit assemblies on each platform draws air from the upper levels of the operating floor and delivers cooling air through the ring duct and the secondary ductwork distribution system to the cubicles, compartments, and access areas above and below the operating floor. In addition, cooling air is delivered to the reactor cavity and reactor support areas to maintain appropriate local area and concrete temperatures.
The nominal supply air temperature provided by the fan coil unit assemblies during normal operation is 70°F in order to meet the environmental design requirements.

As the supply air absorbs the heat released from various components inside containment, return air rises through vertical passages and openings due to its lower density to the upper containment level where it is again drawn into the fan coil unit assemblies, cooled, dehumidified, and recirculated.

The standby fans will be started automatically if one of the following events occurs:

- Air discharge flow rate from the operating fans decreases to a predetermined setpoint
- Electrical and/or control power is lost

Fans are connected to 480V buses with backup power supply from the onsite standby diesel generators. Following a reactor shutdown when the containment air temperature is below a predetermined temperature, the cooling water supply to the four fan coil unit assemblies will be manually realigned by the operators from the central chilled water system to the hot water heating system. Refer to Subsection 9.2.7 for further details.

Shutdown and Refueling Operation

During reactor shutdown, the system maintains the average bulk air temperature within appropriate limits for personnel access and maintenance. In addition, a portable exhaust air filtration unit is available to protect maintenance personnel by controlling the spread of airborne contamination from the steam generator internals to the other containment areas. This is accomplished by running temporary flexible ductwork from one of the steam generator manways to a portable exhaust air filtration unit, which discharges to the containment atmosphere. This allows the steam generators to be maintained negative with respect to the steam generator compartment during testing and inspections.

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Integrated Leak Rate Testing Operation

During integrated leak rate testing, fan coil unit assembly operation is controlled by the main control room operator. When placed into operation, the vaneaxial fans are operated at low speed to prevent the fan motors from exceeding their rated horsepower. The fan coil unit assemblies may be operated during the initial phase of the testing to establish homogeneous conditions within the containment and may be operated during the test to maintain stable conditions. Operation of the vaneaxial fans after the initial containment stabilization could affect the containment integrated leak rate testing results. The recirculation fan coil unit assemblies draw air from the upper levels of the operating floor and deliver airflow through the ring header and its distribution ductwork that is connected to equipment compartments, cubicles, and access areas above and below the operating floor.

Abnormal Plant Operation

The containment recirculation system is not required to mitigate the consequences of a design basis fuel handling accident or a loss of coolant accident. If the system is available following abnormal operational transients, it can be operated at reduced speed for post-event recovery operations to lower the containment temperature and pressure.

The power supplies to the containment recirculation cooling system are provided by the plant ac electrical system and the onsite standby diesel generators. In the event of a loss of the normal ac power and offsite power, the containment recirculation components are auto sequenced on to the onsite standby diesel generators, and fans start automatically and operate at fast speed for investment protection.

9.4.6.3 Safety Evaluation

The containment recirculation cooling system has no safety function and therefore requires no nuclear safety evaluation. The containment recirculation cooling system is designed to preclude damage to safety-related systems, structures, or components as a result of a seismic event.

9.4.6.4 Tests and Inspections

The containment recirculation cooling system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of the system are accessible for periodic inspection. A system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program. Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability.

The system airflows are balanced in accordance with SMACNA HVAC Systems - Testing, Adjusting and Balancing (Reference 19).

9.4.6.5 Instrumentation Application

The containment recirculation cooling system is controlled by the plant control system. Process indication and alarm signals are locally accessible through the plant control system. Refer to Subsection 7.1.1 for a description of the plant control system.

Temperature controllers are provided in the branch ducts (steam generator compartments service) of the corresponding containment recirculation fan coil unit which provide an input signal to modulate the central chilled water system supply valves to the cooling coils when the unit is in normal power operation. When the unit is shut down for maintenance, refueling, etc., the containment volumetric average temperature is used as the input signal to modulate the central chilled water flow or hot water flow to the cooling coils. The containment volumetric average high and low temperature are

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monitored and alarmed when the temperature is out of the normal operating range. The ambient temperature in a specific equipment compartment or areas of the containment are monitored and alarmed.

The discharge flowrate from each containment recirculation fan unit is monitored and a low flow condition is alarmed to alert the operator. The low flow condition will also generate a stop signal to the operating fan, and the standby fan will be started automatically. Flow to the reactor cavity is also monitored and low flow condition is alarmed.

9.4.7 Containment Air Filtration System

The containment air filtration system (VFS) serves the containment, the fuel handling area and the other radiologically controlled areas of the auxiliary and annex buildings, except for the hot machine shop and health physics areas which are served by a separate ventilation system.

9.4.7.1 Design Basis

9.4.7.1.1 Safety Design Basis

The containment air filtration system provides the safety-related functions of containment isolation and containment vacuum relief. System equipment and ductwork whose failure could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portion of the system is non-seismic. The containment isolation function is described in Subsection 6.2.3. The containment vacuum relief function automatically adjusts the internal containment pressure as it approaches the analyzed design parameters described in Subsection 6.2.1.1.4. This adjustment in the pressure across the containment shell preserves the structural integrity of the shell by maintaining the differential pressure within the allowable limits as defined by the structural analysis described in Subsection 3.8.2. The vacuum relief function is actuated on the Low-2 containment pressure signal and manually once the low containment pressure signal is generated.

9.4.7.1.2 Power Generation Design Basis

Containment Area

The containment air filtration system provides the following functions:

- Provides intermittent flow of outdoor air to purge the containment atmosphere of airborne radioactivity during normal plant operation, and continuous flow during hot or cold plant shutdown conditions to provide an acceptable airborne radioactivity level prior to personnel access
- Provides intermittent venting of air into and out of the containment to maintain the containment pressure within its design pressure range during normal plant operation
- Directs the exhaust air from the containment atmosphere to the plant vent for monitoring, and provides filtration to limit the release of airborne radioactivity at the site boundary within acceptable levels
- Monitors gaseous, particulate and iodine concentration levels discharged to the environment through the plant vent

The system conditions and filters outside air supplied to the containment for compatibility with personnel access during maintenance and refueling operations. Based on the maximum and minimum outside air normal temperature conditions shown in Chapter 2, Table 2-1, the system

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supplies air between 50 and 70°F. The air is distributed and conditioned within the containment by the containment recirculation system (Subsection 9.4.6).

Radiologically Controlled Areas Outside Containment

The containment air filtration system provides filtration of exhaust air from the fuel handling area, auxiliary, or annex buildings to maintain these areas at a slightly negative pressure with respect to the adjacent areas when the radiologically controlled area ventilation system detects high airborne radioactivity or high pressure differential. Refer to Subsection 9.4.3 for a description of the radiologically controlled area ventilation system.

9.4.7.2 System Description

The containment air filtration system is shown in Figure 9.4.7-1.

9.4.7.2.1 General Description

The containment air filtration system consists of two 100 percent capacity supply air handling units, a ducted supply and exhaust air system with containment isolation valves and piping, registers, exhaust fans, filtration units, automatic controls and accessories. The supply air handling units are located in the south air handling equipment room of the annex building at elevation 135′-3″. The supply air handling units are connected to a common air intake plenum, located at the south end of the fan room. The common air intake plenum #2 is located at the extreme south end of the annex building between elevation 135′-3″ and 158′-0″. This plenum supplies air for the health physics and hot machine shop HVAC system, and the containment air filtration system. The intake is not protected from tornado missiles. The containment air filtration system supply air handling units discharge the supply air towards the east containment recirculation cooling system (VCS) recirculation unit to distribute the purge air within the containment. Refer to Subsection 9.4.6 for a description of the containment recirculation cooling system.

The exhaust air filtration units are located within the radiologically controlled area of the annex building at elevation 135′-3″ and 150′-3″. The filtration units are connected to a ducted system with isolation dampers to provide HEPA filtration and charcoal adsorption of exhaust air from the containment, fuel handling area, auxiliary and annex buildings. A gaseous radiation monitor is located downstream of the exhaust air filtration units in the common ductwork to provide an alarm if abnormal gaseous releases are detected. The plant vent exhaust flow is monitored for gaseous, particulate and iodine releases to the environment. During containment purge, the exhaust air filtration units satisfy 10 CFR 50 Appendix I guidelines (Reference 20) for offsite releases and meets 10 CFR 20 (Reference 21) allowable effluent concentration limits when combined with gaseous releases from other sources. During conditions of abnormal airborne radioactivity in the fuel handling area, auxiliary and/or annex buildings, the filtration units provide filtered exhaust to minimize unfiltered offsite releases.

The size of the containment air filtration system supply and exhaust air lines that penetrate the containment pressure boundary is 36 inches in diameter. The air lines on both sides of the supply and exhaust containment penetrations are reduced in size to accommodate inboard and outboard 16-inch diameter containment isolation valves that are opened when the containment air filtration system is connected to the containment. The penetrations are sized at 36 inches to allow for possible future addition of a high volume purge system. In the event of a loss-of-coolant accident (LOCA) while the containment air filtration system is aligned to containment, there will not be a significant release of radioactivity during closure of the 16 inch diameter supply and exhaust valves. The maximum time for valve closure (see Table 6.2.3-1) is consistent with the analysis assumptions for radiological consequences (see Table 15.6.5-2). The closure time is also consistent with the basis (compliance with 10 CFR Part 50.34) for Branch Technical Position CSB 6-4 to Standard Review Plan 6.2.4 (Reference 23) or described in Subsection 6.2.1.5.

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The exhaust air containment penetration also includes a containment vessel vacuum relief function to protect the containment from reaching the containment shell design external design pressure. In the event of a LOCA, while these 6-inch motor-operated vacuum relief valves are open, the releases of radioactivity during the maximum time for closure of these valves (see Table 6.2.3-1) have been evaluated. The radiological consequences are bounded by those currently presented in Subsection 15.6.5.3. The maximum time for closure of the vacuum relief valves was also evaluated to determine the impact for the calculation of the LOCA minimum backpressure. The methodology depicted in Subsection 6.2.1.5 bounds the minimum backpressure calculation.

The exhaust air containment penetrations also serve as a connection for the containment integrated leak rate test system to pressurize and depressurize the containment during integrated leak rate testing. Otherwise, the containment air filtration exhaust subsystem is not involved with the containment integrated leak rate test and is isolated from the containment during this time period.

9.4.7.2.2 Component Description

The containment air filtration system is comprised of the following components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Table 9.4.7-1 provides design parameters for the major components of the system.

Supply Air Handling Units

Each supply air handling unit consists of a low efficiency filter bank, a high efficiency filter bank, a hot water heating coil bank, a chilled water cooling coil bank and a supply fan.

Exhaust Air Filtration Units

Each exhaust air filtration unit consists of an electric heater, an upstream high efficiency filter bank, a HEPA filter bank, a charcoal adsorber with a downstream postfilter bank, and an exhaust fan. The filtration unit configurations, including housing, internal components, ductwork, dampers, fans, and controls, are designed, constructed, and tested to meet the applicable performance requirements of ASME AG-1, N509, and N510 (References 36, 2, and 3) to satisfy the guidelines of Regulatory Guide 1.140 (Reference 30) except as noted in Appendix 1A. The filtration unit housings maximum leakage rates do not exceed one percent of the design flow in accordance with ASME AG-1. Refer to Table 9.4-1 for a summary of the containment air filtration system filtration efficiencies and Appendix 1A for a comparison of the containment air filtration system exhaust air filtration units with Regulatory Guide 1.140 (Reference 30).

Isolation Dampers

Isolation dampers are bubble tight, single-blade or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power or instrument air. The isolation dampers are Class R and are designed, constructed, and tested in accordance with ASME AG-1 Section DA (Reference 36).

Pressure Differential Control Dampers

Pressure differential control dampers utilize opposed-blade type construction. Dampers are Class R and are designed, constructed, and tested in accordance with ASME AG-1 Section DA (Reference 36). The dampers maintain a slight negative pressure within the fuel handling building area, with respect to the environment and adjacent non-radiologically controlled plant areas.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

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Supply and Exhaust Fans

The supply and exhaust air fans are centrifugal type, single width single inlet (SWSI), with radial blades that produce overloading horsepower characteristics. Fan configurations are either performance rated in accordance with ANSI/AMCA 210 (Reference 4) or certified in accordance with AMCA 211 (Reference 5). Fan configurations are sound tested in accordance with ANSI/AMCA 300 (Reference 6).

Containment Penetrations

The containment penetrations include containment isolation valves, interconnecting piping, and vent and test connections with manual test valves. The containment isolation components that maintain the integrity of the containment pressure boundary after a LOCA are classified as Safety Class B and seismic Category I. Seismic Category I debris screens are mounted on Safety Class C, seismic Category I pipe to prevent entrainment of debris through the supply and exhaust openings that may prevent tight valve shutoff. The screens are designed to withstand post-LOCA pressures.

The vent and purge line containment isolation valves inside and outside the containment have air operators. The valves are designed to fail closed in the event of loss of electrical power or air pressure. These valves are controlled by the protection and plant safety monitoring system as discussed in Subsection 7.1.1. The valves shut tight against the containment pressure following a design basis accident.

The motor-operated vacuum relief valves are controlled by the protection and plant safety monitoring system as discussed in Subsection 7.1.1. The valves inside containment are self-actuated check valves. These motor-operated valves and check valves shut tight against the containment pressure following a design basis accident.

Ductwork, Duct Supports, and Accessories

The system air ductwork inside containment meets seismic Category II criteria so that it will not fall and damage any safety-related equipment following a safe shutdown earthquake.

Non-Seismic ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17) with the exception of exhaust air ductwork (per Regulatory Guide 1.140 Compliance Summary) and accessories which are Class R. Exhaust air ductwork (per Regulatory Guide 1.140 Compliance Summary) and accessories are designed, constructed, and leak tested in accordance with ASME AG-1 Section SA (Reference 36).

Seismic II Category ductwork, duct supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34), SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17), and Appendix 3A, with the exception of exhaust air ductwork (per Regulatory Guide 1.140 Compliance Summary) and accessories which are Class R. Exhaust air ductwork (per Regulatory Guide 1.140 Compliance Summary) and accessories are designed, constructed and leak tested in accordance with ASME AG-1 Section SA (Reference 36) and Appendix 3A.

With the exception of nonmetallic flexible connectors, ductwork is constructed of either galvanized or stainless steel. The use of nonmetallic flexible connectors inside containment is limited to ductwork above the operating deck. Duct supports and accessories are constructed of galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

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Shutoff and Balancing Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. The containment exhaust air dampers are Class R and are designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section DA.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

Fire Dampers

Fire dampers are provided where the ductwork penetrates a fire barrier to maintain the fire resistance rating of the fire barriers. The fire dampers meet the design and installation requirements of UL-555 (Reference 15).

The fire damper within the Class R intake duct which is upstream of the filters is Class R, and is designed, constructed, and tested in accordance with ASME AG-1 Section DA (Reference 36). The Class R fire damper is applied to promote an ALARA HVAC design.

Low Efficiency Filters, High Efficiency Filters, and Postfilters

Low and high efficiency filters are rated in accordance with ASHRAE 52.1 and 126 (References 7 and 35). The minimum average dust spot efficiencies of the filters are shown in Table 9.4.7-1. High efficiency filter performance upstream of HEPA filter banks meet the design requirements of ASME AG-1 (Reference 36), Section FB. Postfilters located downstream of the charcoal adsorbers have a minimum DOP efficiency of 95 percent. The filters are UL listed in accordance with UL 900 (Reference 8).

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with ASME AG-1 (Reference 36), Section FC. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol in accordance with ASME AG-1, Section TA.

Charcoal Adsorbers

Each charcoal adsorber is designed constructed, qualified, and tested in accordance with ASME AG-1 (Reference 36), Section FE; and Regulatory Guide 1.140 (Reference 30). Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with 1E Bulletin 80-03 (Reference 29).

Electric Heating Coils

The electric heating coils are SCR controlled, fin tubular type. The electric heating coils are UL listed in accordance with UL 1996 (Reference 26). The electric heating coils are constructed, qualified, and tested in accordance with ASME AG-1 Section CA (Reference 36).

Heating Coils

The heating coils are hot water, finned tubular type. The heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating the heat output. Coils are performance rated in accordance with ANSI/ARI 410 (Reference 12).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

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9.4.7.2.3 System Operation

Normal Plant Operation

During normal plant operation, the containment air filtration system operates on a periodic basis to purge the containment atmosphere as determined by the main control room operator to reduce airborne radioactivity or to maintain the containment pressure within its normal operating range. One supply air handling unit provides outdoor air that is filtered, cooled, or heated to the containment areas above the operating floor. The airflow rate is controlled to a constant value by modulating the supply fan inlet vanes to compensate for filter loading or changes in containment pressure. The cooling coils are supplied with chilled water from the central chilled water system (VWS) to cool and/or dehumidify the outside supply air. The heating coils are supplied with hot water by the hot water heating system (VYS). Refer to Subsections 9.2.7 and 9.2.10 for descriptions of the central chilled water and hot water heating systems.

The temperature of the air supplied by each air handling unit is controlled by temperature sensors located in the supply air duct. When the supply air temperature is low, the face and bypass dampers across the supply air heating coil are modulated to heat the supply air. When the supply air temperature is high, the flow of chilled water is modulated to cool the supply air. The supply air is continuously monitored by smoke detectors located in the ductwork downstream of the supply air handling units.

The airflow rate through the exhaust filters is controlled to a constant value when the exhaust filters are connected to the containment by modulating the exhaust fan inlet vanes to compensate for filter loading or changes in system resistance caused by single or parallel fan operation, or changes in containment pressure. The exhaust lines from the containment include a pair of isolation dampers arranged in parallel to restrict the airflow to maintain the exhaust filter plenums at a negative air pressure when the containment is positively pressurized. Based on predetermined setpoints, the operators select the appropriate damper to open. This prevents exfiltration of unfiltered air from bypassing the filters.

The filtered exhaust air from the containment is discharged to the atmosphere through the plant vent by the exhaust fan. The gaseous effluents in the plant vent are monitored for radioactivity levels before the air is discharged to the environment. Refer to Section 11.5 for a description of the plant vent radiation monitor.

During single subsystem operation, the standby supply and exhaust air units can be started manually by the operator if the operating train fails.

Prior to and during plant shutdown, one or both trains of the containment air filtration system can be operated to remove airborne radioactivity prior to personnel access. During cold ambient conditions, the supply air is heated by the hot water heating system. The exhaust filter unit electric heater controls the relative humidity of the exhaust air entering the charcoal adsorber below 70 percent.

When both trains are operated concurrently, the containment air filtration system provides a maximum airflow rate equivalent to approximately 0.21 air changes per hour.

Abnormal Plant Operation

The containment isolation valves in the supply and exhaust air lines automatically close when containment isolation signals are initiated by the protection and safety monitoring system or diverse actuation system. Refer to Subsections 6.2.3, 7.7.1.11 and 7.3 for discussions of the containment isolation system, diverse actuation system and protection and safety monitoring system.

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Main control room operators can connect the containment air filtration system to the containment for cleanup of potential airborne radioactivity while the containment remains isolated if a containment High-1 radiation signal is not present.

The containment vacuum relief valves automatically open when containment vacuum relief signals are initiated by the protection and safety monitoring system. These valves automatically close when containment isolation signals are initiated by the protection and safety monitoring system. Even though these valves are normally closed, they automatically receive a confirmatory close signal if the containment isolation valve inside the reactor containment building is open. As discussed in Subsection 7.6.2.4, this interlock verifies availability of engineered safety features for the containment vacuum relief and containment isolation functions.

If high airborne radioactivity or high pressure differential is detected in the fuel handling area, the auxiliary and/or annex buildings, the radiologically controlled area ventilation system isolates the affected area from the outside environment and starts the containment air filtration exhaust subsystem to maintain a slight negative pressure differential in the isolated zone(s). The airflow rate through the exhaust fan is maintained at a constant value by modulating the fan inlet vanes. An outside air makeup damper modulates to maintain a constant airflow rate through the HEPA and charcoal filters to maintain the isolated area(s) at a slightly negative pressure relative to the clean areas. The containment air filtration system is automatically isolated from the containment, if purging is in progress and the standby exhaust filter train does not start. If both exhaust trains are connected to the containment, one exhaust train is automatically isolated from the containment and realigned to the isolated area(s). The exhaust subsystem can be manually connected to the onsite diesel generators if there is a loss of normal ac power and offsite power.

The containment air filtration system is not required to mitigate the consequences of a design basis fuel handling accident or a loss of coolant accident. If the exhaust air filtration units are operational and ac power is available, they may be used to support post-event recovery operations. The plant vent high range radiation detectors monitor effluents discharged into the plant vent.

If smoke is detected in either supply air duct downstream of each air handling unit, an alarm is initiated and the corresponding train is shut down. The system is automatically shut down upon detection of a fire. Fire dampers are provided for HVAC ductwork that passes through a fire barrier in order to isolate each fire zone in the event of a fire.

9.4.7.3 Safety Evaluation

The containment air filtration system has the safety-related functions of containment isolation and vacuum relief. The containment isolation function is evaluated in Subsection 6.2.3. The vacuum relief system consists of redundant relief devices inside and outside containment. Two independent lines with an automatic isolation valve are provided on the positive pressure side of the system along with two independent lines with a check valve on the negative pressure side of the system penetrate containment. The lines share a common containment penetration. Therefore, there are no single active failures, which would prevent the system from performing its function.

The failure of equipment and ductwork will not reduce the functioning of safety-related systems, structures or components that are required to close to maintain containment isolation integrity after a design basis accident or those components required to function for vacuum relief. Ductwork that is located inside containment whose failure may affect any safety-related equipment is designed to seismic Category II requirements.

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9.4.7.4 Tests and Inspections

The containment air filtration system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of each system are accessible for periodic inspection during normal plant operation. The exhaust subsystem is tested to confirm airflow capacity in accordance with the guidelines of ASME N510 (Reference 3). The supply air subsystem airflow rate is measured and balanced in accordance with the guidelines of SMACNA HVAC Systems – Testing, Adjusting and Balancing (Reference 19). Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability.

The tests and inspections of the containment isolation valves associated with the containment air filtration system are discussed in Subsections 6.2.3 and 6.2.5.

HEPA filters and charcoal adsorbers are tested in place in accordance with ASME N510 to verify that these components do not exceed a maximum allowable bypass leakage. Samples of charcoal adsorbent are periodically tested to verify a minimum charcoal efficiency of 99 percent in accordance with Regulatory Guide 1.140 (Reference 30) except that test procedures and test frequency are conducted in accordance with ASME N510.

The exhaust ductwork and filter plenums are tested in place for leak tightness in accordance with ASME N510, Section 6.

The exhaust subsystem of the containment air filtration system (VFS) is tested and inspected in accordance with ASME/ANSI AG-1-1997 and Addenda AG-1a-2000 (Reference 201), ASME N509-1989, ASME N510-1989, and Regulatory Guide 1.140.

The VFS is tested as separate components and as an integrated system. Surveillance tests are performed to monitor the condition of the system. Testing methods include:

- Visual inspection
- Airflow capacity and distribution tests
- HEPA filter bank and adsorber bank in-place leak tests
- Air heater performance tests
- Laboratory testing of adsorbers

Testing is performed at the frequency provided in Table 1 of ASME N510-1989.

9.4.7.5 Instrumentation Application

The containment air filtration system operation is controlled by the plant control system (PLS) except for the containment isolation valves which are controlled by the protection and safety monitoring system (PMS) and diverse actuation system (DAS). Refer to Subsection 7.1.1 for a discussion of the plant control system, protection and safety monitoring system, and diverse actuation system. Automatic protection and safety monitoring system actuations of these valves are discussed in Section 7.3; the diverse actuation system signals are discussed in Subsection 7.7.1.11. Display and monitoring of system instrumentation is consistent with the requirements of Table 4-2 of ASME N509 (Reference 2).

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Temperature controllers maintain the proper supply air temperature. Temperature indication and alarms are provided to inform operators of abnormal temperature conditions for supply air and charcoal adsorbers.

Pressure differential indication and alarms are provided to inform plant operators when air filter changeout is necessary.

Status indication and alarms are provided to monitor operation of fans, controlled dampers and controlled valves. Fans can be placed into operation or shut down from the main control room.

Relative humidity indication and an alarm are provided to monitor the relative humidity of the air upstream of the containment air filtration exhaust air charcoal adsorbers.

Radioactivity indication and alarms are provided to inform the main control room operators of the concentration of gaseous radioactivity in the containment air filtration system exhaust duct and gaseous, particulate and iodine concentrations in the plant vent. See Section 11.5 for a description of these radiation monitors.

Flow indication and alarms are provided to alert plant operators to equipment malfunctions.

9.4.8 Radwaste Building HVAC System

The radwaste building HVAC system serves the radwaste building which includes the clean electrical/mechanical equipment room and the potentially contaminated HVAC equipment room, the waste accumulation room, and the mobile systems facility.

9.4.8.1 Design Basis

9.4.8.1.1 Safety Design Basis

The radwaste building HVAC system serves no safety-related function and therefore has no nuclear safety design basis. The system is nonseismic.

9.4.8.1.2 Power Generation Design Basis

The radwaste building HVAC system provides the following functions:

- Provide conditioned air to work areas to maintain acceptable temperatures for equipment and personnel working in the areas
- Provide confidence that air movement is from clean to potentially contaminated areas to minimize the spread of airborne contaminants
- Collect the vented discharges from potentially contaminated equipment
- Provide for radiation monitoring of exhaust air prior to release to the environment
- Maintain the radwaste building at a negative pressure with respect to ambient to prevent unmonitored releases from the radwaste building

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The system maintains the following temperature based on maximum and minimum normal outdoor air temperature conditions shown below in Chapter 2, Table 2-1:

Room or Area	Temperatures (°F)
Processing areas and storage areas	50-105
Mechanical and electrical equipment rooms	50-105
Truck staging area	50-105

9.4.8.2 System Description

The radwaste building HVAC system is shown in Figure 9.4.8-1.

9.4.8.2.1 General Description

The radwaste building HVAC system is a once-through ventilation system that consists of two integrated subsystems: the radwaste building supply air system and the radwaste building exhaust air system. The systems operate in conjunction with each other to maintain temperatures in the areas served while controlling air flow paths and building negative pressure.

The supply air system consists of two 50 percent capacity air handling units of about 5,000 scfm each with a ducted air distribution system, automatic controls, and accessories. The air handling units are located in an electrical/mechanical equipment room on elevation 100′-0″ on the southwest side of the building. Each unit draws 100 percent outdoor air through individual louvered outdoor air intakes. The two units discharge into a common duct distribution system which is routed through the building. Branch connections from the main duct supply air through registers into the various areas served.

The exhaust air system consists of two 50 percent capacity exhaust centrifugal fans sized to allow the system to maintain a negative pressure, an exhaust air duct collection system, and automatic controls and accessories. The airflow rates are balanced to maintain a constant exhaust design air flow through the fans. The exhaust fans are located in an equipment room on Elevation 100′-0″ in the northwest corner of the radwaste building.

The exhaust fans discharge to a common duct which is routed to the plant vent. A radiation monitor records activity in the discharge duct and activates an alarm in the main control room when excess activity in the effluent discharge is detected. The radiation monitoring system is described in Section 11.5.

The exhaust air collection duct inside the radwaste building exhausts air from areas and rooms where low levels of airborne contamination may be present. Exhaust connection points are provided to allow the direct exhaust of equipment located on the mobile systems. Where potential for significant airborne release exists, mobile systems include HEPA filtration. Back draft dampers are provided at each mobile system connection to prevent blowback through the equipment in the event of exhaust system trip. Criteria for mobile processing systems are included in Sections 11.2 and 11.4.

9.4.8.2.2 Component Description

The radwaste building HVAC system is comprised of the following major components. These components are located in the non-seismic radwaste building and the Seismic Category II portion of the annex building.

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Supply Air Handling Units

Each air handling unit consists of a plenum section, a low efficiency filter bank, a high efficiency filter bank, a hot water heating coil, a chilled water cooling coil bank, and a supply fan with automatic inlet vanes.

Supply and Exhaust Air Fans

The supply and exhaust fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with backward inclined blades to produce non-overloading horsepower characteristics. Fan configurations are either performance rated in accordance with ANSI/AMCA 210 (Reference 4) or certified in accordance with AMCA 211 (Reference 5). Fan configurations are sound tested in accordance with ANSI/AMCA 300 (Reference 6).

Low Efficiency Filters and High Efficiency Filters

The low efficiency (25 percent) filters and high efficiency (80 percent) filters have a rated dust spot efficiency based on ASHRAE 52.1 and 126 (References 7 and 35). The filters are UL listed in accordance with UL 900 (Reference 8).

Hot Water Unit Heaters

The hot water unit heaters consist of a fan section and hot water heating coil section factory assembled as a complete and integral unit. The unit heaters are either horizontal discharge or vertical downblast type. The hot water unit heaters are tested and rated using the methods and standard conditions provided within ASHRAE Handbook - HVAC Systems and Equipment, “Unit Ventilators, Unit Heaters, and Makeup Air Units” (Reference 42).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Heating Coils

The hot water heating coils are counterflow, finned tubular type. The heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating the heat output. The heating coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Shutoff, Balancing, and Backdraft Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through exhaust connections for mobile systems. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow.

Shutoff dampers downstream of exhaust fans, which discharge air into the plant vent stack at a positive pressure, are Class R and are designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section DA. The Class R dampers are applied to promote an ALARA HVAC design. The use of Class R dampers mitigates the potential for exfiltration of airborne radiation in a positive pressure duct during operation of the fan coincident with an airborne release from the spaces the fans serves.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

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Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Fire dampers are Class R and are designed, constructed, and tested in accordance with ASME AG-1 Section DA (Reference 36). The Class R fire dampers are applied to promote an ALARA HVAC design. The use of Class R fire dampers mitigates the potential for exfiltration of airborne radiation in a positive pressure duct during operation of the fan coincident with an airborne release from the spaces the fan serves.

Ductwork, Duct Supports, and Accessories

Ductwork, duct supports and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17) with the exception of the following:

Ductwork downstream of exhaust fans, which discharge air into the plant vent stack at a positive pressure, is Class R and is designed, constructed, and tested in accordance with ASME AG-1 Section SA (Reference 36) and Appendix 3A. The Class R ductwork and accessories are applied to promote an ALARA HVAC design. The use of Class R dampers mitigates the potential for exfiltration of airborne radiation in a positive pressure duct during operation of the fan coincident with an airborne release from the spaces the fan serves.

Ductwork is constructed of materials consistent with these standards including, but not limited to, galvanized or stainless steel. Duct supports and accessories are constructed of material consistent with these standards including, but not limited to, galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

9.4.8.2.3 System Operation

Normal Plant Operation

During normal operation, both supply air handling units and both exhaust fans operate continuously to maintain suitable temperatures in the radwaste building. The inlet vanes of the supply fans are modulated automatically by flow controllers to maintain a constant supply air flow rate. The exhaust air flow is then balanced to maintain the building at a slightly negative pressure with respect to the outside environment. The differential pressure transmitter receives input from a pressure differential sensing instrument mounted indoors in the general building area with sensing lines located inside and outside the building. The sensing line mounted outside the building is shielded from the effects of wind.

The temperature of the air supplied by the air handling unit is controlled by separate heating and cooling controllers, with sensors in the general building area. The cooling controllers modulate the control valves in the chilled water supply lines to the air handling units. The heating controllers modulate the face and bypass dampers of the hot water heating coils in the air handling units.

Differential pressure drop across the supply units filter banks is monitored, and individual alarms are actuated when any pressure drop rises to a predetermined level indicative of the need for filter replacement. To replace the filters on a supply unit, the affected supply fan and exhaust fan are stopped and isolated from the duct system by means of isolation dampers. During filter replacement, the supply and exhaust systems operate at 50 percent capacity. In this mode of operation, radwaste processing operations are adjusted to obtain acceptable temperature in the radwaste building.

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The hot water unit heaters in the mobile systems facility and truck staging area are not normally required to operate to maintain the general building temperature. These heaters operate, in response to local thermostat control, to temper air entering the building when a truck access door is opened.

The hot water unit heater in the electrical/mechanical room operates in response to local thermostat control to maintain the required minimum temperature.

Abnormal Plant Operation

The radwaste building HVAC system is not required to operate during any abnormal plant condition.

9.4.8.3 Safety Evaluation

The radwaste building HVAC system has no safety-related function and therefore requires no nuclear safety evaluation.

9.4.8.4 Tests and Inspections

The radwaste building HVAC system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of each system are accessible for periodic inspection during normal plant operation. A system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program. Air flow rates are measured and balanced in accordance with the guidelines of SMACNA HVAC systems - Testing, Adjusting and Balancing (Reference 19). Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability.

9.4.8.5 Instrumentation Applications

The radwaste building HVAC system operation is controlled by the plant control system (PLS). Refer to Subsection 7.1.1 for a discussion of the plant control system.

Temperature controllers and thermostats maintain the proper space temperatures. Supply air temperature is controlled by sensing the temperature in the mobile systems facility and the electrical/ mechanical equipment room. Unit heaters are controlled by local thermostats. Temperature indication and alarms are accessible locally via the plant control system.

Temperature is indicated for each air handling unit supply air discharge duct.

Operational status of fans is indicated in the main control room. The fans and air handling units can be placed into operation or shutdown from the main control room.

Differential pressure indication is provided for each of the filters in the air handling units and an alarm for high pressure drop is provided for each air handling unit.

Pressure differential indication and alarms are provided to monitor the radwaste building pressure with respect to the environment.

Airflow is indicated for the air handling unit and exhaust fan discharge ducts. Alarms are provided for abnormal air flow rates in the fan discharge ducts.

An alarm is provided for high radiation in the main exhaust duct to the vent stack.

An alarm is provided for smoke in the discharge ducts from the supply air handling units.

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Position indicating lights are provided for automatic dampers.

9.4.9 Turbine Building Ventilation System

The turbine building ventilation system (VTS) operates during startup, shutdown, and normal plant operations. The system maintains acceptable air temperatures in the turbine building for equipment operation and for personnel working in the building.

9.4.9.1 Design Basis

9.4.9.1.1 Safety Design Basis

The turbine building ventilation system serves no safety-related function and therefore has no nuclear safety design basis. The system is nonseismic.

9.4.9.1.2 Power Generation Design Basis

The turbine building ventilation system provides the following functions:

- Maintains acceptable temperatures for equipment operation
- Provides for removal of chemical fumes from the secondary sampling laboratory room, flammable vapors from the lube oil reservoir room and the clean and dirty lube oil storage room
- Provides conditioning air to maintain acceptable temperatures for electrical equipment rooms and personnel work areas
- Maintains the following temperatures based on the ambient outside air(1):
  - General area (operating deck, intermediate levels, and base slab).....50-105°F
  - Electrical equipment rooms (switchgear room 1, switchgear room 2, and electrical equipment room) .....50-105°F
  - Personnel work areas (Secondary sampling laboratory).....72-78°F
  - Exciter rectifier room.....50-85°F
  - Control cabinet rooms .....50-105°F
  - South bay equipment areas (various pumps and RCP variable speed drive power converter units.....50-100°F

9.4.9.2 System Description

The turbine building ventilation system consists of the following subsystems:

- General area heating, south bay equipment, and ventilation
- Electrical equipment, control cabinet rooms, and personnel work area HVAC
- Local area heating and ventilation
  - Lube oil reservoir room ventilation
  - Clean and dirty lube oil storage room ventilation

The turbine building HVAC system general area subsystem is shown in Figure 9.4.9-1.

1. Temperature conditions of 95°F DB/77°F WB (coincident), 79°F WB (non-coincident) in summer and -5°F in winter.

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9.4.9.2.1 General Description

9.4.9.2.1.1 General Area Heating and Ventilation

Most of the turbine building is supplied by the general area ventilation and heating subsystem. Air is exhausted from the turbine building to the atmosphere by roof exhaust ventilators. The roof exhaust ventilators pull in outside air through wall louvers located at elevations 100’-0", 120’-6", and 141’-3". Wall louvers are located at the operating deck to provide additional air during plant outage operations. The general area heating subsystem uses electric unit heaters to provide local heating throughout the turbine building. During heating operation, the general area ventilation system is not operated.

9.4.9.2.1.2 Electrical Equipment and Personnel Work Area HVAC

The electrical equipment, south bay equipment, control cabinet rooms, and personnel work area air conditioning subsystem serves electrical equipment areas (switchgear rooms and the electrical equipment room), the south bay equipment (CCS pumps, BDS pumps, and reactor coolant pumps variable frequency drive power converter areas), and personnel work areas (secondary sampling laboratory and exciter rectifier room). This subsystem is subdivided into four independent HVAC systems, one serving the electrical equipment areas, one serving the south bay equipment, one serving the personnel work areas, and one serving the control cabinet rooms.

The electrical equipment HVAC system consists of four air handling units with a supply fan, a ducted supply and return air system, automatic controls, and accessories. The air handling units are located on elevation 158’-7" of the turbine building. The temperature of the rooms is maintained by thermostats which control the chilled water control valves for cooling and the integral face/bypass dampers for heating.

The south bay equipment area HVAC system consists of two 50-percent capacity air handling units of about 18,800 cfm capacity each. The air handling units are located on elevation 117′-6″ of the turbine building between column lines 11 and 11.2. The temperature of the area is maintained by thermostats that control the chilled water control valves for cooling and the integral face bypass dampers for heating. Outside air is mixed with the recirculation air to maintain a positive pressure.

The personnel work area HVAC system consists of two 50 percent capacity air handling units, a ducted supply and return air system, automatic controls, and accessories. The air handling units are located on elevation 100′-0″ of the turbine building. The temperature of the rooms is maintained by thermostats which control the chilled water control valves for cooling and the integral face/bypass dampers for heating. Outside air is mixed with recirculated air to maintain a positive pressure.

9.4.9.2.1.3 Local Area Heating and Ventilation

The lube oil reservoir room, clean and dirty lube oil storage room, and secondary sampling laboratory fume hood have centrifugal exhaust fans to remove flammable vapors, odors, or chemical fumes as required.

9.4.9.2.2 Component Description

The turbine building ventilation system is comprised of the following major components. These components are located in the non-seismic turbine building.

HVAC Air Handling Units

Each air handling unit is a horizontal draw-through cabinet type consisting of a mixing box section, low efficiency (25 percent) filter, high efficiency (80 percent) filter, integral face/ bypass damper, hot

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water heating coil (except for the control cabinet rooms' air handling units), and chilled water cooling coil. The personnel area and south bay equipment area air handling units include a supply air fan.

Low Efficiency Filters and High Efficiency Filters

The efficiency (25 percent) filters and high efficiency (80 percent) filters have a rated dust spot efficiency based on ASHRAE 52.1 and 126 (References 7 and 35). The filters are UL listed in accordance with UL 900 (Reference 8).

Exhaust Ventilators

The turbine building roof exhaust ventilators are hooded, belt driven, propeller type with gravity-type backdraft damper. Ventilators in the lube oil rooms are centrifugal type. The exhaust ventilators are built to the manufacturer's standards.

Shutoff, Balancing, and Backdraft Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through shut down fans. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

Electric Unit Heaters

The electric unit heaters are single or multi stage, fin tubular type. The electric unit heaters are UL listed in accordance with UL 2021 (Reference 40) and meet the requirements of the National Electric Code NFPA 70 (Reference 28).

Electric Duct Heaters

The electric duct heaters are SCR controlled, fin tubular type. The electric duct heaters are UL listed in accordance with UL 1996 (Reference 26). The duct heaters are UL listed for zero clearance and meet requirements of NFPA 70 (Reference 28).

Humidifiers

A humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifier is designed and rated in accordance with ARI 640 (Reference 13).

Fire Dampers

Fire dampers are provided at HVAC duct penetrations through fire barriers to maintain fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL-555 (Reference 15) as applicable.

Ductwork, Duct Supports, and Accessories

Ductwork, supports and accessories meet the design and construction requirements of SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17). Ductwork is constructed of materials consistent with these standards including, but not limited to, galvanized or stainless steel. Duct supports and accessories are constructed of material consistent with these standards including, but not limited to, galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

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9.4.9.3 System Operation

9.4.9.3.1 General Area Heating and Ventilation

The general area ventilation system is manually controlled. Roof exhaust ventilators are manually started and stopped as required to satisfy space temperature conditions. Wall louvers located at the ground floor and the two intermediate levels of the turbine building are normally open during ventilation operation. The wall louvers located at the operating floor are manually opened to increase ventilation air to the area during outage operations. The operating floor louvers normally remain closed during power operation.

Electric unit heaters are controlled automatically or manually. In the automatic mode, the heater fan motors are thermostatically controlled by their respective space thermostats.

9.4.9.3.2 Electrical Equipment, Control System Cabinet Rooms, South Bay Equipment, and Personnel Work Area HVAC

During normal operation, the air handling units of the electrical equipment and south bay equipment area HVAC systems, the two air handling units of the control system cabinet rooms, and the two air handling units of the personnel work area HVAC system operate continuously. The chilled water coils are supplied from the plant central chilled water system (VWS) and the hot water coils are supplied from the plant central hot water heating system.

9.4.9.3.3 Local Area Heating and Ventilation

The ventilation operation for the lube oil reservoir room and the clean and dirty lube oil storage room is similar. Each centrifugal exhaust fan runs continuously to prevent the accumulation of chemical fumes or flammable vapors in its respective room. During normal operation, intake air comes from the turbine building general areas. When general area temperatures exceed 95°F, the intake dampers from the general areas are manually closed and the outside air intake dampers are manually opened.

9.4.9.4 Safety Evaluation

The turbine building ventilation system has no safety-related function and therefore requires no nuclear safety evaluation.

There is no safety-related equipment in the turbine building.

9.4.9.5 Tests and Inspections

The turbine building ventilation system is designed to permit periodic inspection of system components during normal plant operation. System air balance testing and adjustments for the electrical equipment and personnel work areas are conducted in accordance with SMACNA (Reference 19).

Fan configurations are performance rated in accordance with ANSI/AMCA 210 (Reference 4). Water coils are factory tested and rated in accordance with ANSI/ARI 410 (Reference 12).

Ductwork is visually examined to validate ductwork construction and duct sealing methods in accordance with the applicable SMACNA standard (References 16, 17 or 34).

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9.4.9.6 Instrumentation Applications

The turbine building ventilation system is controlled by the plant control system.

Temperature indication and controllers control the room air temperatures within a predetermined range.

Temperature indication is provided to allow surveillance of room and space temperatures in the turbine building.

Differential pressure indication is provided for the air filters in each air handling unit. Alarms are provided for high pressure drops across the air filters.

9.4.10 Diesel Generator Building Heating and Ventilation System

The diesel generator building heating and ventilation system serves the standby diesel generator rooms, electrical equipment service modules, stairwell, security room, diesel fuel oil day tank vaults in the diesel generator building and the two diesel oil transfer modules located in the yard near the fuel oil storage tanks.

9.4.10.1 Design Basis

9.4.10.1.1 Safety Design Basis

The diesel generator building heating and ventilation system serves no safety-related function and therefore has no nuclear safety design basis. The system is nonseismic.

9.4.10.1.2 Power Generation Design Basis

The diesel generator building heating and ventilation system provides the following functions:

- Provides sufficient quantities of ventilation air to maintain acceptable temperatures within the generator rooms for equipment operation and reliability during periods of diesel generator operation in order for the onsite standby power system to perform its defense in depth functions
- Provides adequate heating and ventilation for suitable environmental conditions for maintenance personnel working in the diesel generator room when the generators are not in operation
- Provides suitable environmental conditions for equipment operation in each diesel generator electrical equipment service module under the various modes of diesel generator operation
- Prevents the accumulation of combustible vapors and dissipate their concentration in the fuel oil day tank vault
- Provides adequate heating and ventilation to maintain acceptable temperature within the diesel oil transfer module enclosures
- Provides conditioned air to maintain acceptable temperatures for equipment and personnel working in the security room
- Provides adequate heating to maintain a minimum acceptable temperature within the stairwell

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The system maintains the following room temperatures based on ambient outside air temperature conditions of 95°F (summer) and -5°F (winter):

Area	Design Minimum (°F)	Temperature Maximum (°F)
Diesel Generator Area		
Diesel Generator On	None	130
Diesel Generator Off	50	105
Service Module		
Diesel Generator On	50	105
Diesel Generator Off	50	105
Diesel Oil Transfer Module Enclosure	40	115
Stairwell	50	NA
Security Room	73	78

9.4.10.2 System Description

The diesel generator building heating and ventilation system is shown in Figure 9.4.10-1.

The system consists of the following subsystems:

- Normal heating and ventilation subsystem
- Standby exhaust ventilation subsystem
- Fuel oil day tank vault exhaust subsystem
- Diesel oil transfer module enclosures ventilation and heating subsystem
- Security Room HVAC Subsystem

9.4.10.2.1 General Description

9.4.10.2.1.1 Normal Heating and Ventilation System

The normal heating and ventilation subsystem serves the diesel generator building. Each diesel generator train is provided with independent ventilation and heating equipment for the building areas serving that diesel generator train.

Each normal heating and ventilation subsystem for a diesel generator train consists of one 100 percent capacity engine room air handling unit which ventilates the diesel generator room, one 100 percent capacity service module air handling unit which ventilates the electrical equipment service module, an exhaust system for the fuel oil storage vault and electric unit heaters in the diesel generator area. Air intake louvers for these units are located as high in the diesel generator building wall as possible.

The engine room air handling units are located above the electrical equipment service module with supply and return ducts in the diesel generator room.

The service module air handling units are located above the service module with supply and return ducts into the module.

Electric unit heaters are provided in the diesel generator room to maintain the space at a minimum temperature of 50°F when the diesel generators are off.

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An electric unit heater is provided in the diesel generator stairwell to maintain the space at a minimum temperature.

9.4.10.2.1.2 Standby Exhaust Ventilation Subsystem

The standby exhaust ventilation subsystem for each diesel generator room consists of two 50 percent capacity roof mounted exhaust fans and motor operated air intake dampers mounted in the exterior walls of the room.

9.4.10.2.1.3 Fuel Oil Day Tank Vault Exhaust Subsystem

Each fuel oil day tank vault is continuously ventilated by a centrifugal exhaust fan. The exhaust fans are mounted on the roof of the vault and ducted to draw air from one foot above the vault floor and from above the oil containment dike to remove any oil fumes generated in the space. Air is drawn into the vault from the diesel generator room through an opening protected with a fire damper.

9.4.10.2.1.4 Diesel Oil Transfer Module Enclosures Ventilation and Heating Subsystem

Each diesel oil transfer module enclosure is ventilated by an exhaust fan. Outside air is drawn into the enclosure through louvered air intakes. An electric unit heater is provided in each enclosure to maintain the space at a minimum temperature of 40°F.

9.4.10.2.1.5 Security Room HVAC Subsystem

The security room HVAC subsystem serves the security room within the diesel generator building. The security room HVAC subsystem consists of a single 100 percent capacity split system direct expansion package (air handling unit and condensing unit) with automatic controls and accessories. The air handler is located in the security room of the diesel generator building while the condenser is located on the roof of the diesel generator building. The split system direct expansion package provides conditioning and ventilation of air within the security room to maintain acceptable temperatures and conditions for personnel and equipment.

9.4.10.2.2 Component Description

The diesel generator building heating and ventilation system is comprised of the following major components. These components are located in the non-seismic diesel-generator building or the diesel oil transfer enclosure in the yard. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Tables 9.4.10-1 through 9.4.10-4 provide design parameters for major components in the system.

Supply Air Handling Units

Each air handling unit consists of a mixing box section, a low efficiency filter bank, a high efficiency filter bank, and a supply fan. Electric heating coils are provided for the service module air handling units for module heating.

Supply and Exhaust Air Fans

The supply and exhaust fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with backward inclined blades that produce non-overloading horsepower characteristics or radial blades that produce overloading horsepower characteristics. The diesel oil transfer module enclosure exhaust fans are continuously operating, direct-drive ventilator fans with integral backdraft dampers. Fan configurations are either performance rated in accordance with ANSI/AMCA 210 (Reference 4) or certified in accordance with AMCA 211 (Reference 5). Fan configurations are sound tested in accordance with ANSI/AMCA 300 (Reference 6).

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Low Efficiency Filters and High Efficiency Filters

The low efficiency filters and high efficiency filters have a rated dust spot efficiency based on ASHRAE 52.1 and 126 (References 7 and 35). Filter minimum average dust spot efficiency is shown in Table 9.4.10-1. The filters are UL listed in accordance with UL 900 (Reference 8).

Electric Heating Coils

The electric heating coils are SCR controlled, fin tubular type. The electric heating coils are UL listed in accordance with UL 1996 (Reference 26).

Roof Exhaust Fans

The diesel generator building standby exhaust fans are roof mounted, belt driven upblast ventilators. The fans are equipped with gravity dampers that open when the fan operates and close when the fan is shut down.

Electric Unit Heaters

The electric unit heaters are single or multi stage, fin tubular type. The electric unit heaters are UL listed in accordance with UL 2021 (Reference 40) and meet the requirements of the National Electric Code NFPA 70 (Reference 28).

Shutoff, Volume Control, Balancing, and Backdraft Dampers

Multiblade, two-position shutoff remotely operated dampers are parallel-blade type. Multiblade, volume control and balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through shut down exhaust fans and to relieve pressure from the service module and diesel generator building.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork, Duct Supports, and Accessories

Ductwork, supports and accessories meet the design and construction requirements of SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17). Ductwork is constructed of materials consistent with these standards including, but not limited to, galvanized or stainless steel. Duct supports and accessories are constructed of material consistent with these standards including, but not limited to, galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

Split System Direct Expansion Package

The split system direct expansion package consists of an air handler unit and a condensing unit which are separated from one another. The air handling unit consists of a fan, filter, evaporator, and electric heating coil. The condensing unit utilizes refrigerant as the cooling medium and is air-cooled. The condensing unit contains a compressor, fan, condensing coil, filter, and drier.

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9.4.10.2.3 System Operation

9.4.10.2.3.1 Normal Heating and Ventilation Subsystem

Normal Plant Operation

During normal plant operation, each engine room air handling unit operates continuously. Each air handling unit has 100 percent cooling capacity for the engine room served by the unit. The unit draws outdoor air through a louvered air intake and mixes it with return air from the engine room in required proportion to satisfy a thermostat located in the space served. Excess outside air supplied to the engine room is discharged to outdoors via gravity dampers.

Each service module air handling unit operates continuously, providing 100 percent cooling and heating capacity for the service module served by the unit. The unit draws outside air through a louvered air intake and mixes it with return air from the service module in required proportion to satisfy a space thermostat located in the service module. Excess outside air supplied to the service module flows into the diesel engine area via a wall mounted gravity damper. The electric heating coil in the service module air handling unit is controlled by a separate space thermostat. The service module air handling unit operates continuously regardless of diesel generator status.

The engine room electric unit heaters operate as required to maintain the minimum room temperature when the diesel generators are not operating. No specific minimum room temperature is maintained when the diesel generators operate. Local space thermostats turn the unit heaters on and off as required for temperature control.

The stairwell electric unit heater operates as required to maintain the minimum room temperature. A local space thermostat turns the unit heater on and off as required for temperature control.

Abnormal Plant Operation

The engine room air handling units and service module air handling units operate continuously during normal operation or when the diesel generators operate during a loss of normal ac power and offsite power.

Unit heaters are not required to operate during any abnormal plant condition or when the diesel generators operate.

9.4.10.2.3.2 Standby Exhaust Ventilation Subsystem

Normal Plant Operation

During normal plant operation, the standby exhaust fans operate in conjunction with the diesel generators. Each exhaust fan has 50 percent cooling capacity for the engine room served by the fan. The fans for an engine room start when the diesel generator in that room is started. The fans start one at a time as the temperature in the engine room increases. The fans stop one at a time as the engine room temperature decreases. The fans shut down when the diesel generator is stopped and the engine room temperature satisfies the standby exhaust fan temperature controllers. One or both standby exhaust fans are required to operate to maintain the engine room temperature depending on the outdoor ambient temperature.

The motor operated air intake dampers automatically open when their associated standby exhaust fan starts and close when both their associated standby exhaust fan stops.

The standby exhaust ventilation system is not required to operate when the diesel generators are not operating.

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Abnormal Plant Operation

The standby exhaust ventilation system is required to operate to support diesel generator operation during loss of normal ac power and offsite power. System operation is identical to that for normal plant operation.

9.4.10.2.3.3 Fuel Oil Day Tank Vault Exhaust Subsystem

Normal Plant Operation

During normal plant operation, each fuel oil day tank vault exhaust fan operates continuously. The fans are manually started and shut down. Each exhaust fan has 100 percent capacity for ventilation of the day tank vault served by the fan.

Abnormal Plant Operation

The fuel oil day tank vault exhaust subsystem is required to operate to support diesel generator operation during loss of normal ac power and offsite power. System operation is identical to that for normal plant operation.

9.4.10.2.3.4 Diesel Oil Transfer Module Enclosures Ventilation and Heating Subsystem

Normal Plant Operation

During normal plant operation, each diesel oil transfer module enclosure exhaust fan operates continuously to provide ventilation and maintain the enclosure at or below the maximum design temperature. The unit heaters operate as required during the winter to maintain the minimum design enclosure temperature. Outside air is admitted through fixed intake louvers.

Abnormal Plant Operation

The diesel oil transfer module enclosure ventilation and heating subsystem is required to operate to support diesel generator operation during loss of the normal ac power and offsite power. System operation is identical to that for normal plant operation.

9.4.10.2.3.5 Security Room HVAC Subsystem

Normal Plant Operation

During normal plant operation, the split system direct expansion package operates continuously to condition and ventilate the room. The condensing unit provides refrigerant cooling to the evaporator within the air handling unit during cooling mode while the electric heating coil within the air handling unit provides the source of heating. A local thermostat automatically controls the operation of the package.

Abnormal Plant Operation

The security room HVAC subsystem is not required to operate during any abnormal plant condition. However, in the event of a loss of the normal ac power and offsite power, the security room HVAC subsystem can be connected to the plant standby diesel generator bus in accordance with the manual load sequencing after evaluation of the diesel generator available capacity.

9.4.10.3 Safety Evaluation

The diesel generator building heating and ventilation system has no safety-related function and therefore requires no nuclear safety evaluation.

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9.4.10.4 Tests and Inspection

The diesel generator building heating and ventilation system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of each system are accessible for periodic inspection during normal plant operation. A system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program. Air flow rates are measured and balanced in accordance with the guidelines of SMACNA HVAC Systems - Testing, Adjusting, and Balancing (Reference 19). Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability.

9.4.10.5 Instrumentation Applications

The diesel generator building heating and ventilation system operation is controlled by the plant control system. Refer to Subsection 7.1.1 for a discussion of the plant control system.

Temperature controllers and thermostats maintain the proper space temperatures. Temperature indication and alarms are accessible locally via the plant control system.

Operational status of fans is indicated in the main control room. All fans and air handling units can be placed into operation or shutdown from the main control room or locally.

Differential pressure indication is provided for each of the filters in the air handling units and an alarm for high pressure drop is provided for each air handling unit.

9.4.11 Health Physics and Hot Machine Shop HVAC System

The health physics and hot machine shop HVAC system serves the personnel decontamination area, frisking and monitoring facilities, containment access corridor, and health physics facilities on the 100′-0″ elevation of the annex building and the hot machine shop on the 107′-2″ elevation of the annex building.

9.4.11.1 Design Basis

9.4.11.1.1 Safety Design Basis

The health physics and hot machine shop HVAC system serves no safety-related function and therefore has no nuclear safety design basis. The system is nonseismic.

9.4.11.1.2 Power Generation Design Basis

The health physics and hot machine shop HVAC system provides the following functions:

- Provides conditioned air to work areas to maintain acceptable temperatures for equipment and personnel working in the areas
- Provides air movement from clean to potentially contaminated areas to minimize the spread of airborne contaminants
- Collects the vented discharges from potentially contaminated equipment in the area
- Provides for exhaust from welding booths, grinders and other miscellaneous equipment located in the hot machine shop

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- Provides for radiation monitoring of exhaust air prior to release to the environment
- Maintains the access control area and hot machine shop at a slight negative pressure with respect to outdoors and the clean areas of the annex building to prevent unmonitored releases of radioactive contaminants
- Provides humidification to maintain a minimum of 35 percent relative humidity

The system maintains the following temperatures based on maximum and minimum normal outside air temperature conditions shown in Chapter 2, Table 2-1:

Room or Area	Temperatures (°F)
Health physics area	73-78
Hot machine shop	65-85
Security room	73-78
Elevator shaft	120 (max.)

9.4.11.2 System Description

The health physics and hot machine shop HVAC system is shown in Figure 9.4.11-1.

9.4.11.2.1 General Description

The health physics and hot machine shop HVAC system is a once-through ventilation system consisting of two integrated subsystems: a supply air system and an exhaust air system. The systems operate in conjunction with each other to satisfy the functional requirements of maintaining temperatures in the areas served while controlling air flow paths and area negative pressure.

The supply air system consists of two 100 percent capacity air handling units of about 8,640 scfm each with a ducted air distribution system and automatic controls. The air handling units are located in the lower south air handling equipment room on elevation 135′-3″ of the annex building. Heating coils are supplied with water from the hot water heating system and cooling coils are supplied from the central chilled water system. The units draw 100 percent outdoor air through the common, louvered outdoor air intake plenum #2 as described in Subsection 9.4.2. They discharge into a duct distribution system which is routed to the health physics and machine shop areas. Humidification is controlled to maintain a minimum 35 percent relative humidity via a steam humidifier located in the main system supply duct and supplied with water from the demineralized water system.

The exhaust air system consists of two 100 percent capacity exhaust centrifugal fans sized to allow the system to maintain a negative pressure with ductwork and automatic controls, and a separate machine shop exhaust fan and high efficiency filter for exhausting from machine tools and other localized areas in the hot machine shop. The exhaust fans are located in the staging and storage area on elevation 135′-3″ of the annex building. The machine shop exhaust fan and filter are located locally in the machine shop. The air flow rates are balanced to maintain a constant exhaust design air flow through the fans.

The exhaust fans discharge to a common duct which is routed to the plant vent stack. A radiation monitor measures activity in the common discharge duct downstream of the exhaust fans and activates an alarm in the main control room when excess activity in the effluent discharge is detected. The radiation monitoring system is described in Section 11.5.

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Individual flexible exhaust duct branches are provided to machine tools. The flexible ducts are connected to a hard duct manifold which is connected to a filter and exhaust fan. The exhaust fan discharges into the main system exhaust ductwork.

Electric unit heaters are provided in the security room and stairwell to maintain the space at a minimum temperature.

9.4.11.2.2 Component Description

The health physics and hot machine shop HVAC system is comprised of the following major components. These components are located in the Seismic Category II portion of the annex building.

Supply Air Handling Units

Each air handling unit consists of a low efficiency filter bank, a high efficiency filter bank, a hot water heating coil, a chilled water cooling coil bank, and a supply fan with automatic inlet vanes.

Supply and Exhaust Air Fans

The supply and exhaust fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with backward inclined blades to produce non-overloading horsepower characteristics or radial blades to produce overloading horsepower characteristics. Fan configurations are either performance rated in accordance with ANSI/AMCA 210 (Reference 4) or certified in accordance with AMCA 211 (Reference 5). Fan configurations are sound tested in accordance with ANSI/AMCA 300 (Reference 6).

Low Efficiency Filters and High Efficiency Filters

The low efficiency (25 percent) filters and high efficiency (80 percent) filters have a rated dust spot efficiency based on ASHRAE 52.1 and 126 (References 7 and 35). The filters are UL listed in accordance with UL 900 (Reference 8).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Heating Coils

The hot water heating coils are counterflow, finned tubular type. The heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating heat output. The heating coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifier

The humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifier is designed and rated in accordance with ARI 640 (Reference 13).

Shutoff, Balancing, and Backdraft Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through ductwork when operating the machine tools exhaust fan. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow.

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Shutoff dampers downstream of exhaust fans, which discharge air into the plant vent stack at a positive pressure, are Class R and are designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section DA. The Class R dampers are applied to promote an ALARA HVAC design. The use of Class R dampers mitigates the potential for exfiltration of airborne radiation in a positive pressure duct during operation of the fan coincident with an airborne release from the spaces the fan serves.

Dampers are rated in accordance with ANSI/AMCA 500-D (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork, Duct Supports, and Accessories

Ductwork, supports, and accessories are designed and constructed in accordance with SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards – Metal and Flexible (Reference 17) with the following exception: Ductwork downstream of exhaust fans, which discharge air into the plant vent stack at a positive pressure, is Class R and is designed, constructed, and tested in accordance with ASME AG-1 (Reference 36), Section SA, and Appendix 3A. The Class R ductwork is applied to promote an ALARA HVAC design. The use of Class R ductwork mitigates the potential for exfiltration of airborne radiation in a positive pressure duct during operation of the fan coincident with an airborne release from the spaces the fan serves. Ductwork is constructed of materials consistent with these standards including, but not limited to, galvanized or stainless steel. Duct supports and accessories are constructed of material consistent with these standards including, but not limited to, galvanized, stainless or coated steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures.

9.4.11.2.3 System Operation

Normal Plant Operation

During normal operation, one supply air handling unit and one exhaust fan operate continuously to maintain suitable temperatures in the health physics and hot machine shop areas of the annex building. The inlet vanes of the supply fans are modulated automatically by flow controllers to maintain a constant air flow rate. The exhaust air flow is then balanced to maintain a slightly negative pressure differential in the areas served with respect to the outside environment and to surrounding areas which do not have their exhausts monitored for radioactivity. Differential pressure transmitters sense pressure in the general health physics area and atmospheric pressure from an outdoor location shielded from the effects of wind to monitor for negative pressure differential in the health physics area. In addition, a separate differential pressure controller with a sensor in the hot machine shop modulates a damper in the supply air duct to the hot machine shop to maintain a slightly negative pressure differential in the shop with respect to outdoors and to surrounding areas which do not have their exhausts monitored for radioactivity.

The temperature in the health physics and the hot machine shop area is maintained within the design range by a temperature sensor located in the health physics area, with which a controller modulates the control valve on the chilled water supply lines to the cooling coil and the face and bypass damper of the heating coil.

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Abnormal Plant Operation

The health physics and hot machine shop HVAC system is not required to operate during any abnormal plant condition.

9.4.11.3 Safety Evaluation

The health physics and hot machine shop HVAC system has no safety-related functions and therefore requires no nuclear safety evaluation.

9.4.11.4 Tests and Inspections

The health physics and hot machine shop HVAC system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of each system are accessible for periodic inspection during normal plant operation. A system air balance test and adjustment to design conditions is conducted during the plant preoperational test program. Air flow rates are measured and balanced in accordance with the guidelines of SMACNA HVAC Systems - Testing, Adjusting and Balancing (Reference 19). Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability.

9.4.11.5 Instrumentation Application

The health physics and hot machine shop HVAC system operation is controlled by the plant control system. Refer to Subsection 7.1.1 for a discussion of the plant control system.

Temperature controllers maintain the proper space temperature. Supply air temperature is controlled by sensing the temperature in the general health physics area.

Temperature is indicated for each air handling unit supply air discharge duct.

Operational status of fans is indicated in the main control room. The fans and air handling units can be placed into operation or shutdown from the main control room.

Differential pressure indication is provided for each of the filters in the air handling units and an alarm for high pressure drop is provided for each air handling unit.

Pressure differential indication and alarms are provided to monitor the pressure differential in the health physics area with respect to the environment. A pressure differential controller maintains a slightly negative pressure differential in the hot machine shop.

Airflow is indicated for the air handling unit and exhaust fan discharge ducts. Alarms are provided for abnormal air flow rates in the fan discharge ducts.

An alarm is provided for high radiation in the main exhaust duct to the vent stack.

An alarm is provided for smoke in the discharge ducts from the supply air handling units.

Position indicating lights are provided for automatic dampers.

9.4.12 Combined License Information

The program to maintain compliance with ASME AG-1 (Reference 36), ASME N509 (Reference 2), ASME N510 (Reference 3) and Regulatory Guide 1.140 (Reference 30) for portions of the nuclear

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island nonradioactive ventilation system and the containment air filtration system identified in Subsections 9.4.1 and 9.4.7 are addressed in Subsections 9.4.1.4 and 9.4.7.4, respectively.

The Combined License applicant will also provide a description of the MCR/CSA HVAC subsystem's recirculation mode during toxic emergencies, and how the subsystem equipment isolates and operates, as applicable, consistent with the toxic issues, including conformance with Regulatory Guide 1.78 (Reference 37), to be addressed by the Combined License applicant as discussed in DCD subsection 6.4.7.

9.4.13 References

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6. "Reverberant Room Methods for Sound Testing of Fans," ANSI/AMCA 300-2014.

7. ASHRAE 52.1, “Gravimetric and Dust Spot Procedures for Testing Air-Cleaning Devices Used in General Ventilation for Removing Particulate Matter,” 1992.

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12. "Forced-Circulation Air Cooling and Air Heating Coils," ANSI/ARI 410-91.

13. "Commercial and Industrial Humidifiers," ARI 640-96.

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20. Code of Federal Regulations, Title 10, Part 50, Appendix I.

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28. "National Electrical Code," NFPA 70, 1999.

29. "Loss of Charcoal from Adsorber Cells," IE Bulletin 80-03, 1980.

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31. "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants," Regulatory Guide 1.128, Revision 1, October 1978.

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33. NFPA 92A-2000, "Recommended Practice for Smoke Control Systems."

34. "Round Industrial Duct Construction Standards," SMACNA, 1999.

35. "Method of Testing HVAC Air Ducts," ASHRAE 126, 2000.

36. "Code on Nuclear Air and Gas Treatment," ASME/ANSI AG-1-1997.

37. "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," USNRC Regulatory Guide 1.78, Revision 1, December 2001.

38. "Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," ASTM E741, 2000.

39. Not used.

40. "Standard for Fixed and Location-Dedicated Electric Room Heaters," UL-2021, 3rd Edition.

41. "Electrical Baseboard Heating Equipment," UL 1042.

42. ASHRAE Handbook - HVAC Systems and Equipment, 2012.

43. "Method of Measuring Machinery Sound Within an Equipment Space," AHRI 575-2008.

201. ASME/ANSI AG-1a-2000, Addenda to ASME AG-1-1997 Code on Nuclear Air and Gas Treatment, Section HA, “Housings.”

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Table 9.4-1
Design Filtration Efficiencies and Nominal Airflow Rates for HVAC Systems(1)

Areas Served(1)	Design/Test Standard	Ventilation Airflow (cfm)	Recirculation Flow (cfm)	Humidity Control	HEPA Efficiency	Charcoal Efficiency(3)	Maximum Inleakage (cfm)
MCR/CSA (Supplemental Air)	RG 1.140	800	3,200	Yes	99%	99%	60(4)
Containment	RG 1.140	4,000(2)	N/A	Yes	99%	99%	N/A

Notes:
1. Ventilation cfm is shown for each train unless otherwise noted.
2. Both trains of the containment purge may be operated at the same time prior to and during cold shutdown.
3. Charcoal filters are 4-inch deep Type III adsorber cell.
4. This VBS inleakage represents the total inleakage into the combined MCR/CSA HVAC volume, which includes ingress/egress.

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Table 9.4.1-1 (Sheet 1 of 2)
Component Data – Nuclear Island
Nonradioactive Ventilation System
MCR/CSA HVAC Subsystem
(Nominal Values)

Supply Air Handling Units	
Quantity	2
System capacity per unit (%)	100
Supply Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	17,360
Fan static pressure (in. wg)	7.5
Return Air/Smoke Purge Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	15,860
Fan static pressure (in. wg)	4.4
Cooling Coil Requirements	
Type	Chilled Water
Capacity (Btu/hr)	807,140
Water flow (gpm)	See Table 9.2.7-1
Heating Coil Requirements	
Type	Hot Water
Capacity (Btu/hr)	140,810
Filter Requirements	
Low efficiency filter, minimum ASHRAE efficiency (%)	25
High efficiency filter, minimum ASHRAE efficiency (%)	80

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Table 9.4.1-1 (Sheet 2 of 2)
Component Data – Nuclear Island
Nonradioactive Ventilation System
MCR/CSA HVAC Subsystem
(Nominal Values)

Supplemental Air Filtration Subsystem	
Quantity	2
System capacity per unit (%)	100
Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	
- Single unit mode	4,000
- Dual unit mode	3,500
Fan static pressure (in. wg)	
- Single unit mode	11.3
- Dual unit mode	14.3
Heating Coil Requirements	
Type	Electric
Capacity (kw)	20
Filter Requirements	
High efficiency filter, minimum ASHRAE efficiency (%)	80
HEPA filter, DOP efficiency (%)	99.97
Post filter, DOP efficiency (%)	95
Charcoal Adsorber Requirements	
Bed depth (in.)	4.0
Decontamination efficiency (%)	99
Air residence time (sec.)	0.5

MCR Envelope and CSA Leakage Rates

Leakage	Inleakage Rate at 1/8 in. wg (scfm)	Outleakage Rate at 1/8 in. wg (scfm) (Note 3)
MCR access doors	--	Note 1
CSA access doors	--	Note 2
MCR structure	--	Note 1
CSA structure	--	Note 2
MCR/CSA HVAC equipment & ductwork (operating)	50	200

Note:
1. The total outleakage rate from the MCR access doors and the MCR structure is 35 scfm.
2. The total outleakage rate from the CSA doors and CSA structure is 120 scfm.
3. In cases where the outside air flow rate is greater than the outside air required to pressurize the MCR envelope and CSA, excess air is exhausted to the outside atmosphere.

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Table 9.4.1-2 (Sheet 1 of 3)
Component Data – Nuclear Island
Nonradioactive Ventilation System
Class 1E Electrical Room HVAC Subsystem
(Nominal Values)

Division "A & C" Supply Air Handling Units	
Quantity	2
System capacity per unit (%)	100
Supply Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	17,900
Fan static pressure (in. wg)	7.0
Return Air/Smoke Purge Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	16,280
Fan static pressure (in. wg)	4.1
Cooling Coil Requirements	
Type	Chilled Water
Capacity (Btu/hr)	851,680
Water flow (gpm)	See Table 9.2.7-1
Heating Coil Requirements	
Type	Hot Water
Capacity (Btu/hr)	131,460
Filter Requirements	
Low efficiency filter, minimum ASHRAE efficiency (%)	25
High efficiency filter, ASHRAE efficiency (%)	80

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Table 9.4.1-2 (Sheet 2 of 3)
Component Data – Nuclear Island
Nonradioactive Ventilation System
Class 1E Electrical Room HVAC Subsystem
(Nominal Values)

Division "A & C" Class 1E Battery Room Exhaust Fans	
Quantity per electrical division	2
System capacity per fan (%)	100
Type	Centrifugal
Design airflow (scfm)	1,320
Fan static pressure (in. wg)	1.5
Division "B & D" Supply Air Handling Units	
Quantity	2
System capacity per unit (%)	100
Supply Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	10,480
Fan static pressure (in. wg)	6.5
Return Air/Smoke Purge Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	9,790
Fan static pressure (in. wg)	3.0
Cooling Coil Requirements	
Type	Chilled Water
Capacity (Btu/hr)	450,320
Water flow (gpm)	See Table 9.2.7-1
Heating Coil Requirements	
Type	Hot Water
Capacity (Btu/hr)	49,820
Filter Requirements	
Low efficiency filter, minimum ASHRAE efficiency (%)	25
High efficiency filter, ASHRAE efficiency (%)	80

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Table 9.4.1-2 (Sheet 3 of 3)
Component Data – Nuclear Island
Nonradioactive Ventilation System
Class 1E Electrical Room HVAC Subsystem
(Nominal Values)

Division "B & D" Class 1E Battery Room Exhaust Fans	
Quantity per electrical division	2
System capacity per fan (%)	100
Type	Centrifugal
Design airflow (scfm)	990
Fan static pressure (in. wg)	2.1

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Table 9.4.1-3
Component Data – Nuclear Island
Nonradioactive Ventilation System
Passive Containment Cooling System
Valve Room Heating and Ventilation Subsystem
(Nominal Values)

Exhaust Fan Data	
Quantity	1
System capacity per fan (%)	100
Type	Propeller
Design airflow (scfm)	1,300
Fan static pressure (in. wg)	0.75
Electric Unit Heater	
Quantity	2
System capacity per unit heater (%)	100
Type	Horizontal
Capacity (kw)	10

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Table 9.4.2-1
Component Data –
Annex/Auxiliary Buildings Nonradioactive HVAC System
Switchgear Room HVAC Subsystem
(Nominal Values)

Air Handling Units	
Quantity	2
System capacity per unit (%)	100
Supply Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	20,400
Static pressure (in. wg)	5.0
Return/Exhaust Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	20,000
Static pressure (in. wg)	1.6

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Table 9.4.2-2
Component Data –
Annex/Auxiliary Buildings Nonradioactive HVAC System
Equipment Room HVAC System
(Nominal Values)

Supply Air Handling Units	
Quantity	2
System capacity per unit (%)	100
Supply Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	20,900
Static pressure (in. wg)	5.6
Return/Exhaust Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	18,600
Static pressure (in. wg)	2.6
Battery Rooms 40307 and 40309 Exhaust Fans	
Quantity	2
System capacity per unit (%)	100
Type	Centrifugal
Design airflow (scfm)	750
Static pressure (in. wg)	1.5
Battery Room 40412 Exhaust Fan	
Quantity	1
System capacity per unit (%)	100
Type	Centrifugal
Design airflow (scfm)	780
Static pressure (in. wg)	4.9

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Table 9.4.3-1
Component Data – Radiologically Controlled Area Ventilation System
Auxiliary/Annex Building Ventilation Subsystem
(Nominal Values)

Normal Residual Heat Removal Pump Room Unit Coolers	
Quantity	2
System capacity per unit (%)	100
Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	1,750
Fan static pressure (in. wg)	1.8
Cooling Coil Requirements	
Type	Chilled Water
Capacity per coil (Btu/hr)	75,250
Water flow (gpm)	See Table 9.2.7-1
Filter Requirements	
Low efficiency filter, minimum ASHRAE efficiency (%)	25
Chemical and Volume Control Makeup Pump Room Unit Coolers	
Quantity	2
System capacity per unit (%)	100
Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	2,625
Fan static pressure (in. wg)	1.4
Cooling Coil Requirements	
Type	Chilled Water
Capacity (Btu/hr)	113,320
Water flow (gpm)	See Table 9.2.7-1
Filter Requirements	
Low efficiency filter, minimum ASHRAE efficiency (%)	25

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Table 9.4.6-1
Component Data – Containment Recirculation Cooling System
(Nominal Design Values)

Fan Coil Unit Assemblies	
Quantity	4
System capacity per assembly (%)	50
Fan Data (per assembly)	
Quantity (fans)	1
Type	Vaneaxial
Normal design air flow (scfm @ 0 psig)	77,000
Low speed design air flow (scfm @ 59 psig)	122,700
Fan total pressure (in. wg)	11
Coil Data (per assembly)	
Quantity (coil banks)	4 (2 coils/bank)
Total cooling capacity (Btu/hr)	4,816,000
Total chilled water flow rate (gpm)	600
Total heating capacity (Btu/hr)	2,437,000
Total hot water flow rate (gpm)	63

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Table 9.4.7-1 (Sheet 1 of 2)
Component Data – Containment Air Filtration System
(Nominal Values)

Supply Air Handling Units	
Quantity	2
System capacity per assembly (%)	100
Supply Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	4,000
Fan static pressure (in. wg)	14
Cooling Coil Requirements	
Type	Chilled Water
Capacity (Btu/hr)	380,000
Water flow (gpm)	41
Heating Coil Requirements	
Type	Hot Water
Capacity (Btu/hr)	290,000
Filter Requirements	
Low efficiency filter, minimum ASHRAE efficiency (%)	25
High efficiency filter, minimum ASHRAE efficiency (%)	80

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Table 9.4.7-1 (Sheet 2 of 2)
Component Data – Containment Air Filtration System
(Nominal Values)

Exhaust Air Filtration Units	
Quantity	2
System capacity per assembly (%)	100
Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	4,000
Fan static pressure (in. wg)	24.0
Heating Coil Requirements	
Type	Electric
Capacity (kw)	20
Filter Requirements	
High efficiency filter, minimum ASHRAE efficiency (%)	80
HEPA filter, DOP efficiency (%)	99.97
Post filter, DOP efficiency (%)	95
Charcoal Adsorber Requirements	
Bed depth (in.)	4.0
Decontamination efficiency (%)	99
Air residence time (sec.)	0.5

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Table 9.4.10-1 (Sheet 1 of 2)
Component Data – Diesel Generator
Building Heating And Ventilation System
Normal Heating and Ventilation Subsystem
(Nominal Values)

Engine Room Air Handling Unit	
Quantity	2 (one per diesel generator room)
System capacity per unit (%)	100
Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	13,000
Static pressure (in. wg)	3.8
Filter Requirements	
Low efficiency filter, minimum ASHRAE efficiency (%)	25
High efficiency filter, minimum ASHRAE efficiency (%)	80
Service Module Air Handling Unit	
Quantity	2 (one per diesel generator room)
System capacity per unit (%)	100
Fan Requirements	
Type	Centrifugal
Design airflow (scfm)	4,000
Static pressure (in. wg)	4.0
Motor nameplate horsepower	5.0
Heating Coil Requirements	
Type	Electric
Capacity (kw)	3 (SCR Controlled)
Filter Requirements	
Low efficiency filter, minimum ASHRAE efficiency (%)	25
High efficiency filter, minimum ASHRAE efficiency (%)	80

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Table 9.4.10-1 (Sheet 2 of 2)
Component Data – Diesel Generator
Building Heating And Ventilation System
Normal Heating and Ventilation Subsystem
(Nominal Values)

Electric Unit Heaters	
Quantity	4 (two per diesel generator room)
System capacity per unit (%)	50
Type	Horizontal
Capacity (kw)	45

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Table 9.4.10-2
Component Data – Diesel Generator
Building Heating and Ventilation System
Standby Exhaust Ventilation Subsystem
(Nominal Values)

Standby Exhaust Fan	
Quantity	4 (two per diesel generator room)
System capacity per unit (%)	50
Type	Upblast Roof Ventilator
Design airflow (scfm)	24,200
Static pressure (in. wg)	0.25

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Table 9.4.10-3
Component Data – Diesel Generator
Building Heating and Ventilation System
Fuel Oil Day Tank Vault Exhaust Subsystem
(Nominal Values)

Fuel Oil Day Tank Vault Exhaust Fan	
Quantity	2 (one per tank vault)
System capacity per unit (%)	100
Type	Centrifugal
Design airflow (scfm)	400
Static pressure (in. wg)	0.8

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Table 9.4.10-4
Component Data – Diesel Generator
Building Heating and Ventilation System
Diesel Oil Transfer Module Enclosures
Ventilation and Heating Subsystem
(Nominal Values)

Diesel Oil Transfer Module Enclosure Exhaust Fan	
Quantity	2 (one per enclosure)
System capacity per unit (%)	100
Type	Direct Drive Ventilator
Design airflow (scfm)	160
Static pressure (in. wg)	0.1
Electric Unit Heater	
Quantity	2 (one per enclosure)
System capacity per unit (%)	100
Type	Horizontal
Capacity (kw)	4

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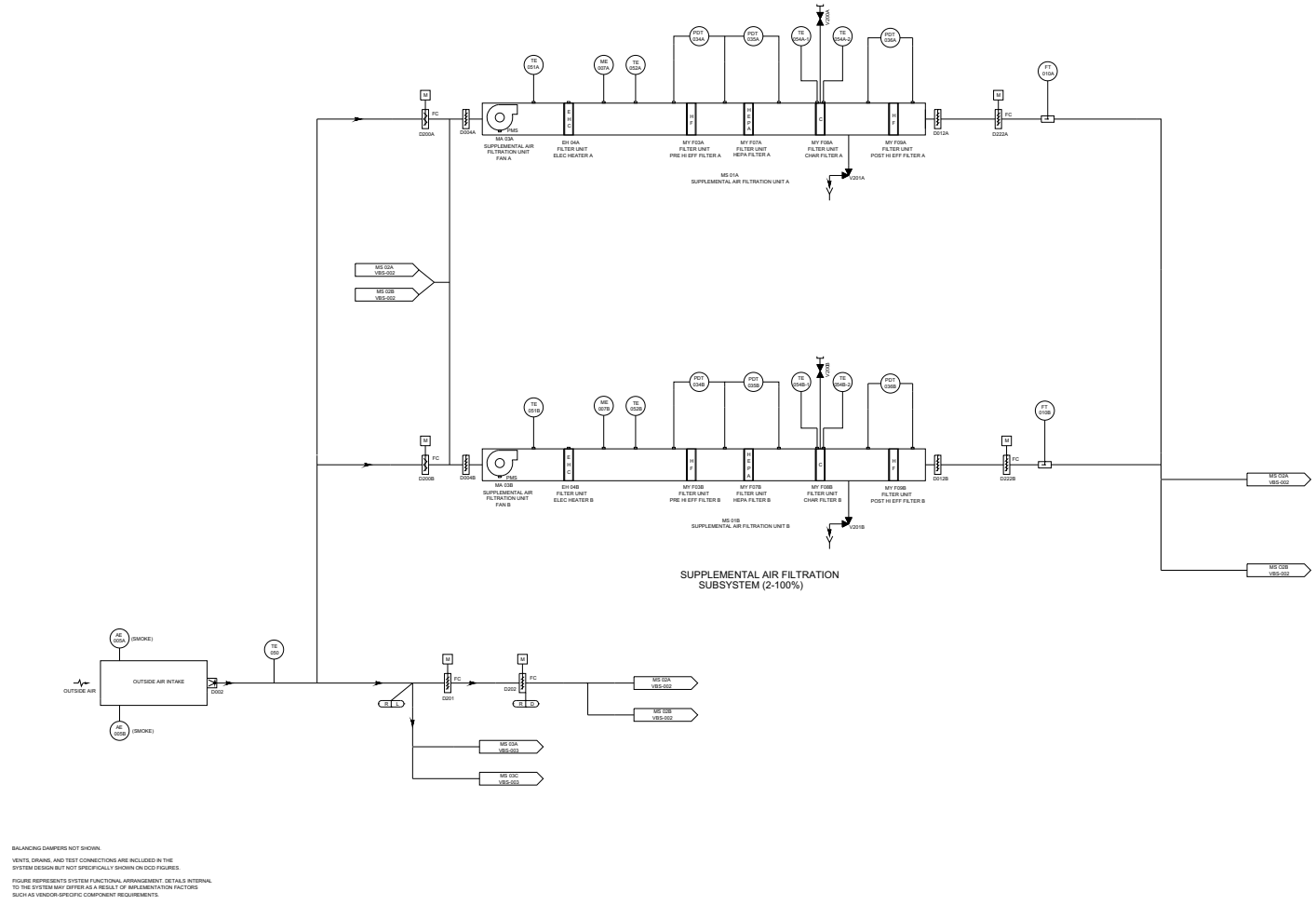

Inside Auxiliary Building

Figure 9.4.1-1 (Sheet 1 of 7)
Simplified Nuclear Island Non-Radioactive Ventilation System Piping and Instrumentation Diagram
(REF) VBS 001

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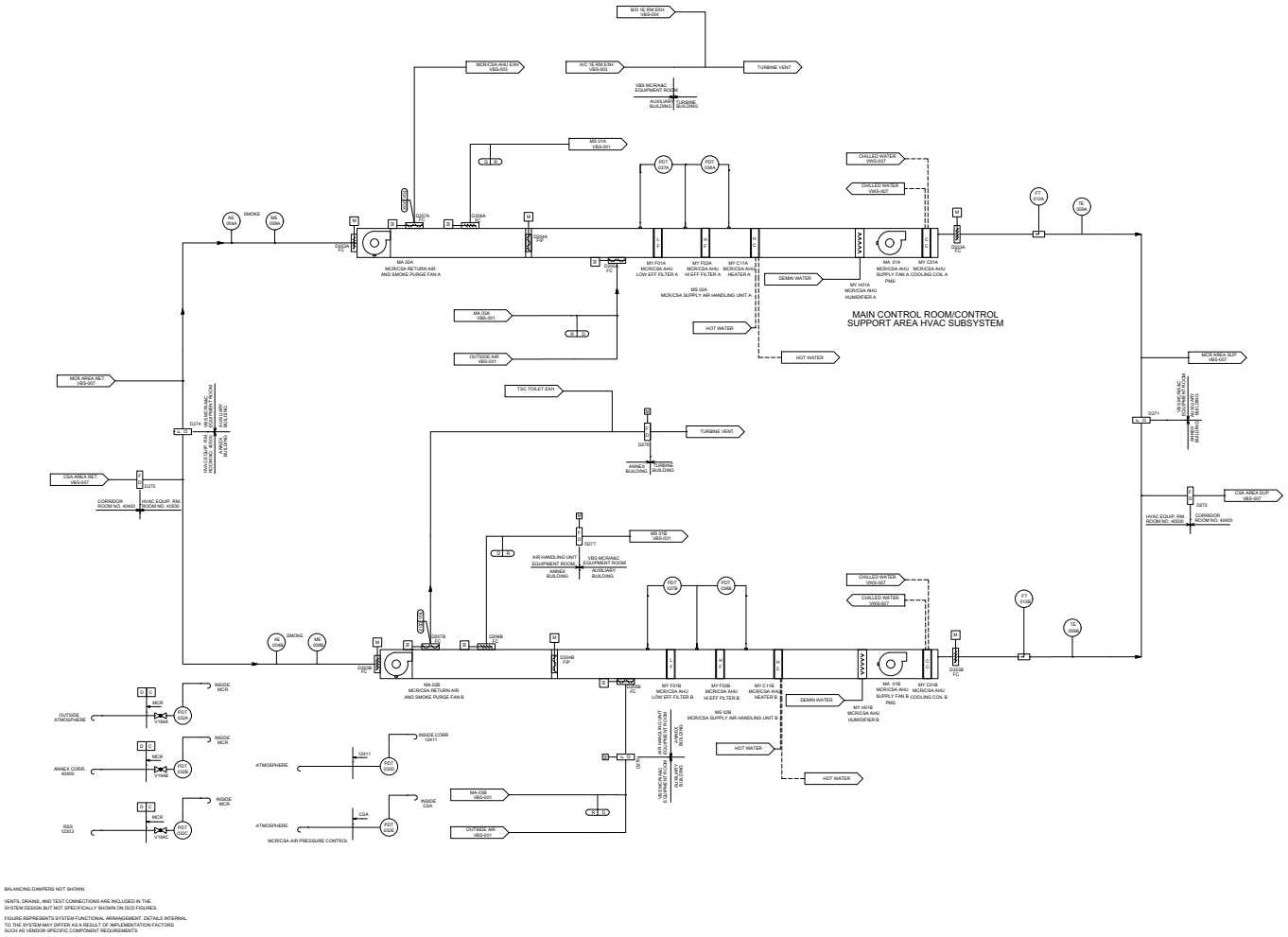

Inside Auxiliary Building

Figure 9.4.1-1 (Sheet 2 of 7)
Simplified Nuclear Island Non-Radioactive Ventilation System Piping and Instrumentation Diagram
(REF) VBS 002

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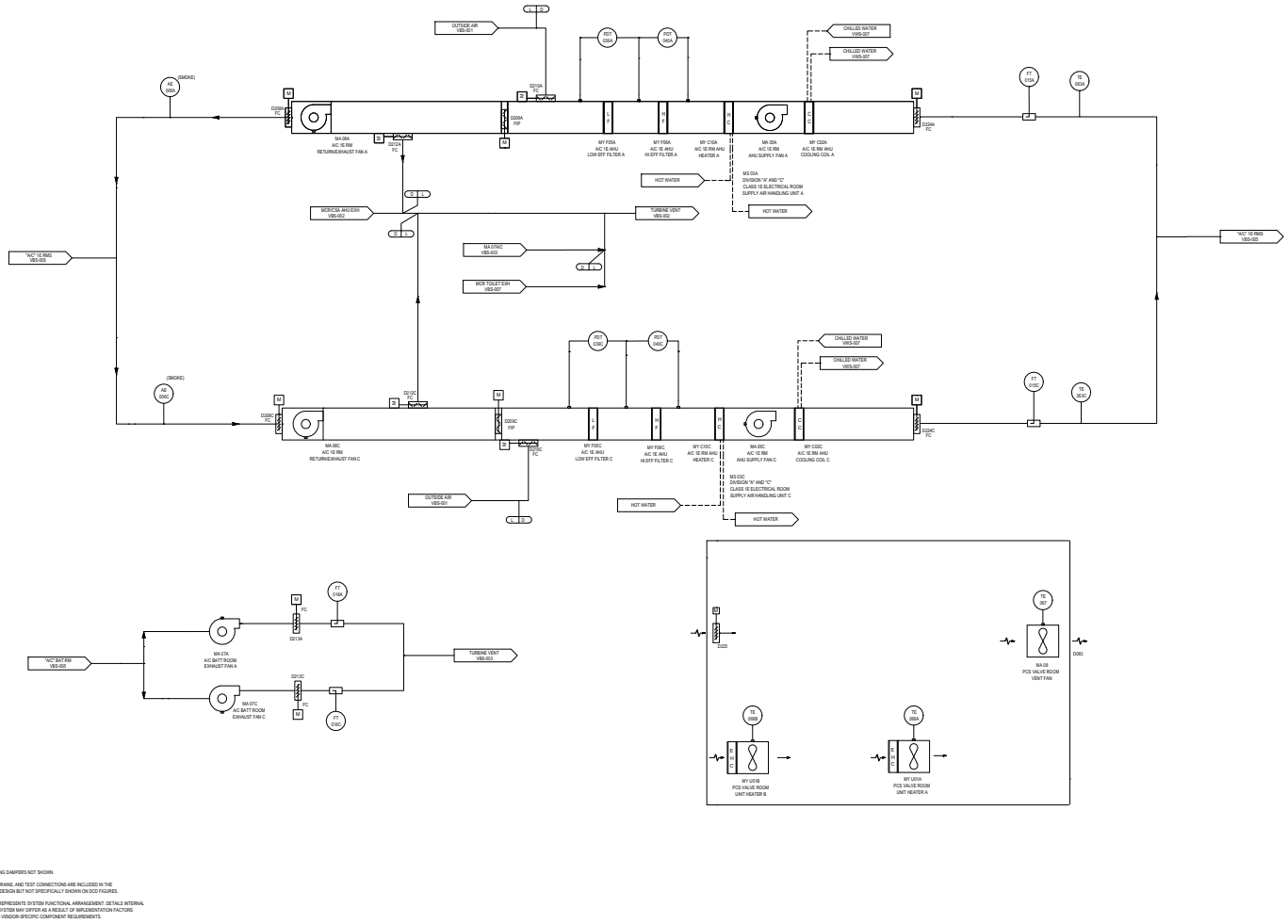

BALANCING DAMPERS NOT SHOWN.

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Inside Auxiliary Building

Figure 9.4.1-1 (Sheet 3 of 7)
Simplified Nuclear Island Non-Radioactive Ventilation System Piping and Instrumentation Diagram
(REF) VBS 003

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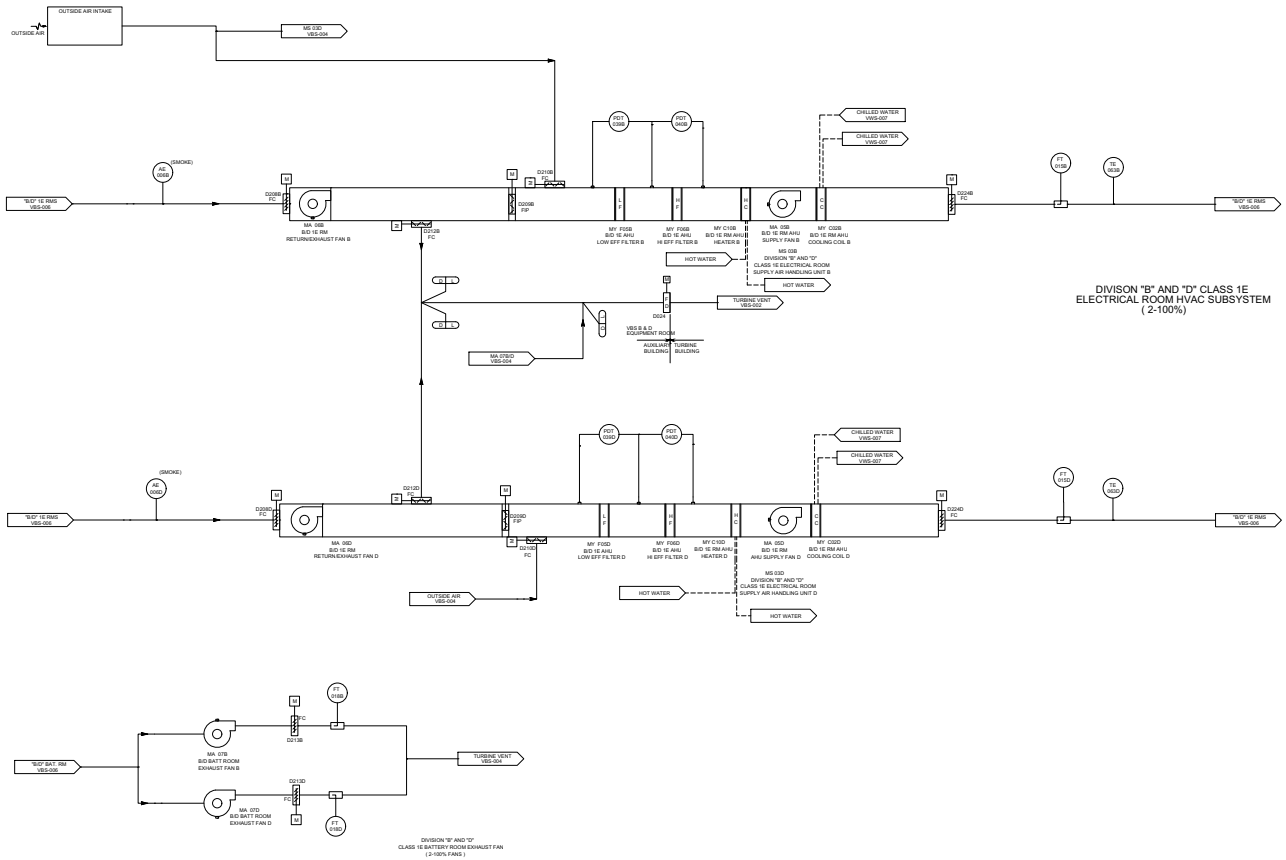

Inside Auxiliary Building

Figure 9.4.1-1 (Sheet 4 of 7)
Simplified Nuclear Island Non-Radioactive Ventilation System Piping and Instrumentation Diagram (REF) VBS 004

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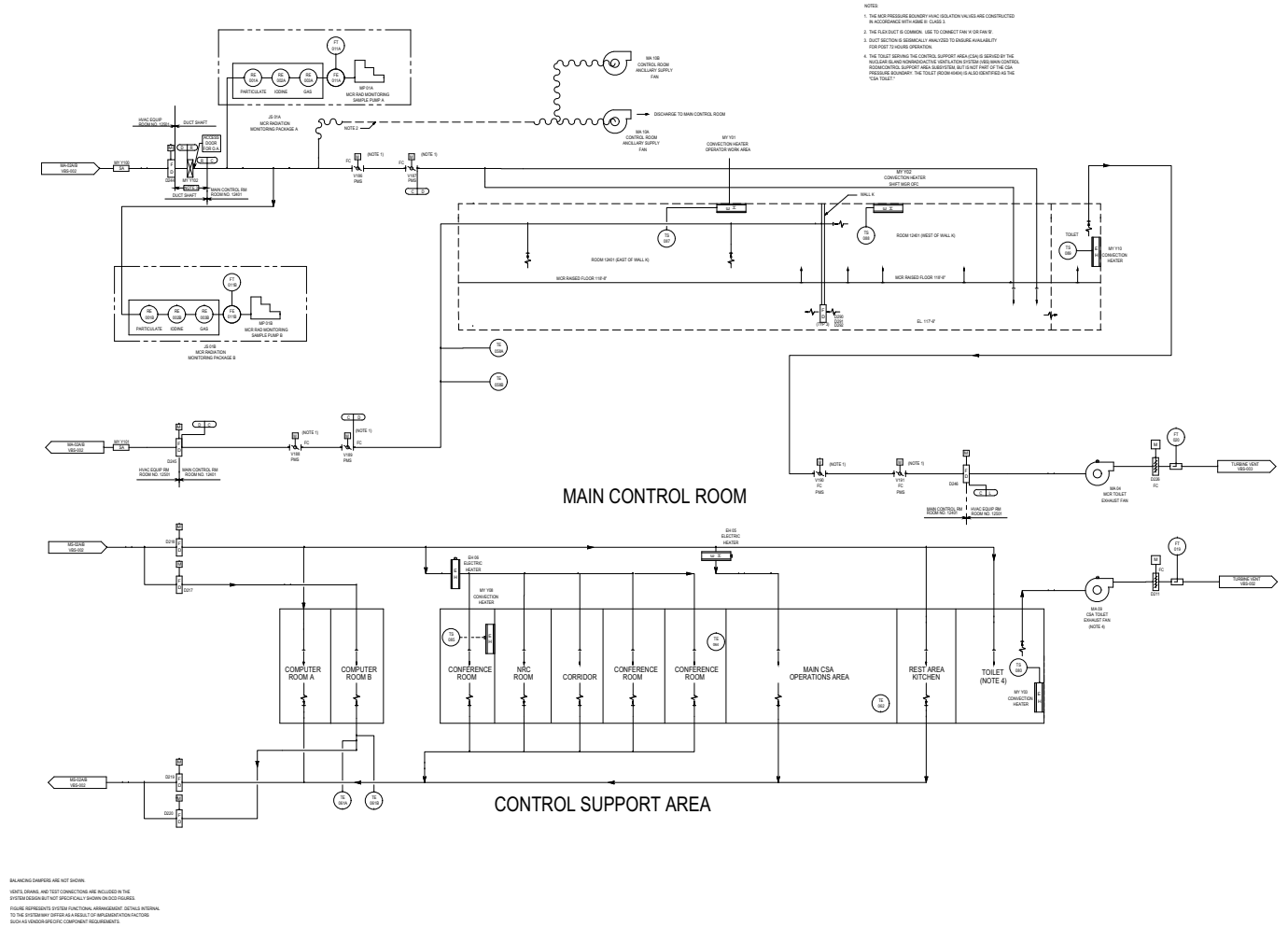

Inside Auxiliary Building

Figure 9.4.1-1 (Sheet 5 of 7)
Simplified Nuclear Island Non-Radioactive Ventilation System Piping and Instrumentation Diagram (REF) VBS 007

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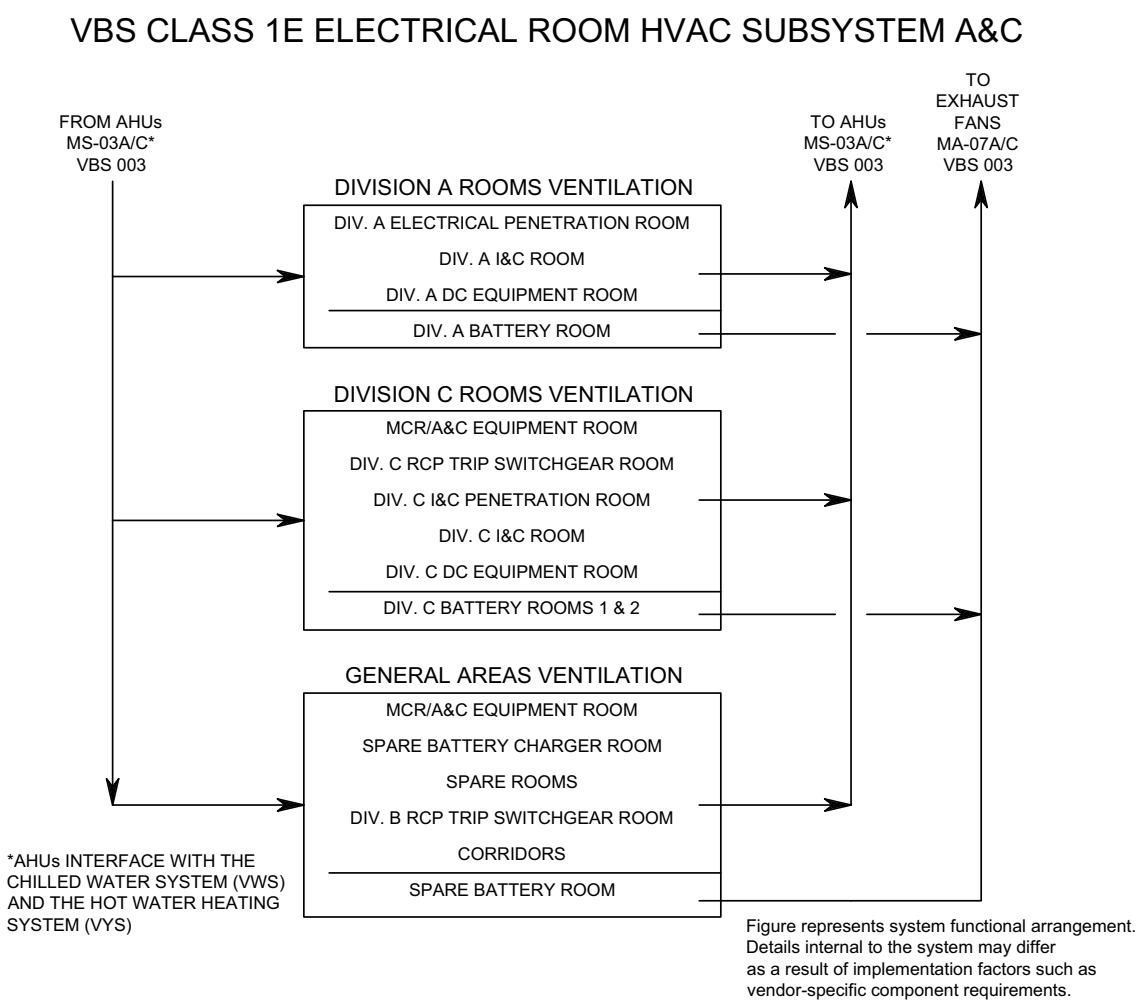

Figure 9.4.1-1 (Sheet 6 of 7)
Nuclear Island Non-Radioactive Ventilation System
Piping and Instrumentation System
(REF) VBS 005

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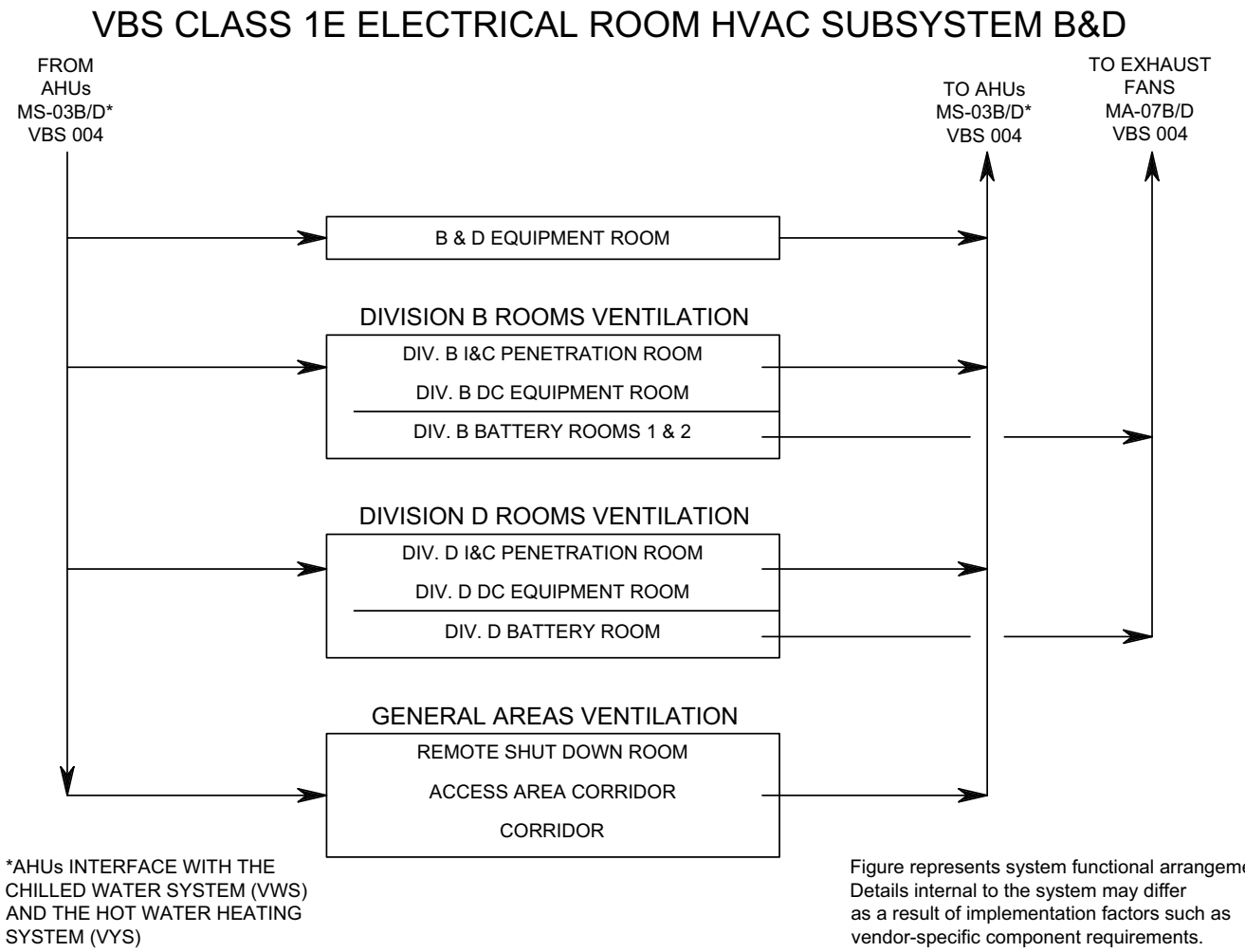

Figure 9.4.1-1 (Sheet 7 of 7)
Nuclear Island Non-Radioactive Ventilation System
Piping and Instrumentation Diagram
(REF) VBS 006

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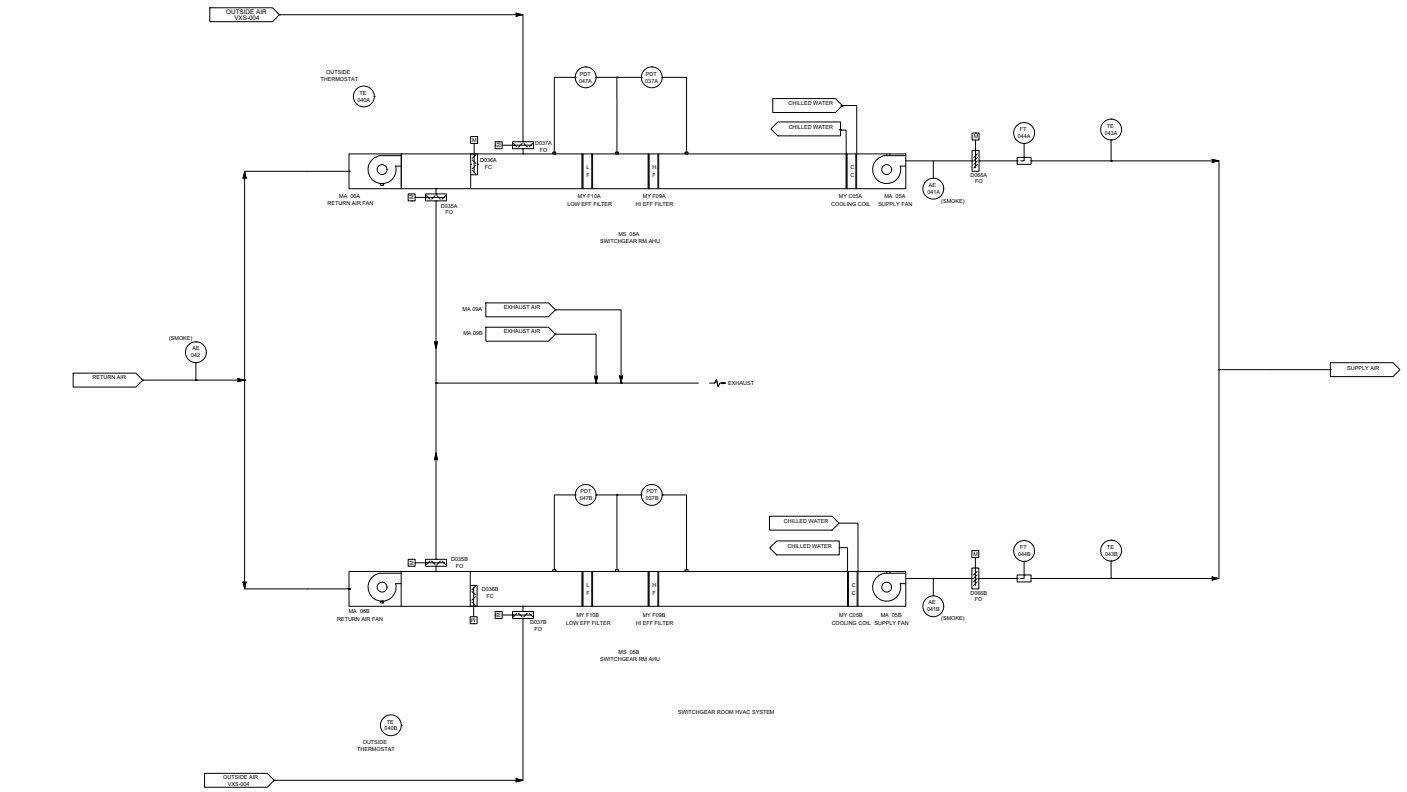

BALANCING DAMPERS ARE NOT SHOWN.

SYSTEM COMPONENTS ARE CLASS D, CLASS E OR CLASS L UNLESS NOTED.

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Inside Annex Building
Figure 9.4.2-1 (Sheet 1 of 7)
Simplified Annex/Aux Non-Radioactive Ventilation System
Piping and Instrumentation Diagram
(REF) VXS 003

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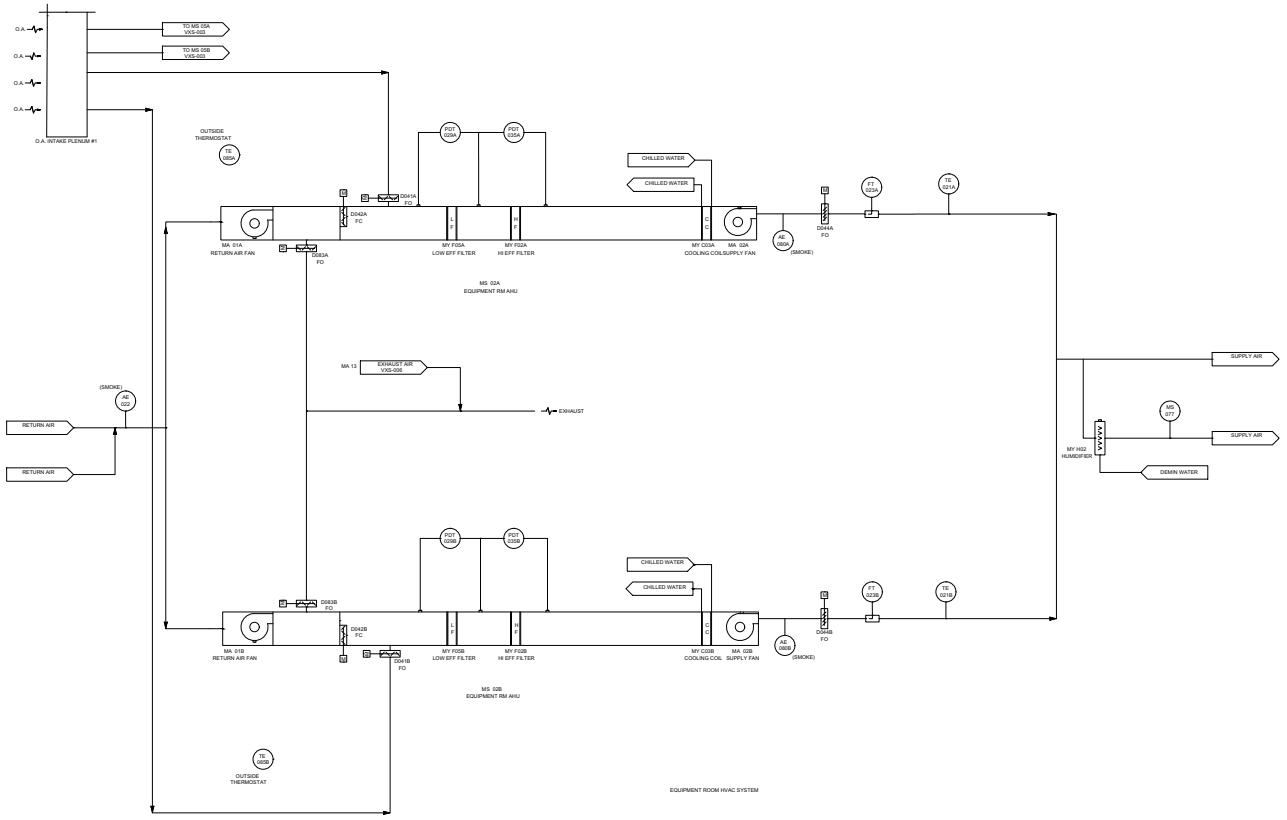

BALANCING DAMPERS ARE NOT SHOWN.

SYSTEM COMPONENTS ARE CLASS D, CLASS E OR CLASS L UNLESS NOTED.

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Inside Annex Building

Figure 9.4.2-1 (Sheet 2 of 7)
Simplified Annex/Aux Non-Radioactive Ventilation System Piping and Instrumentation Diagram (REF) VXS 004

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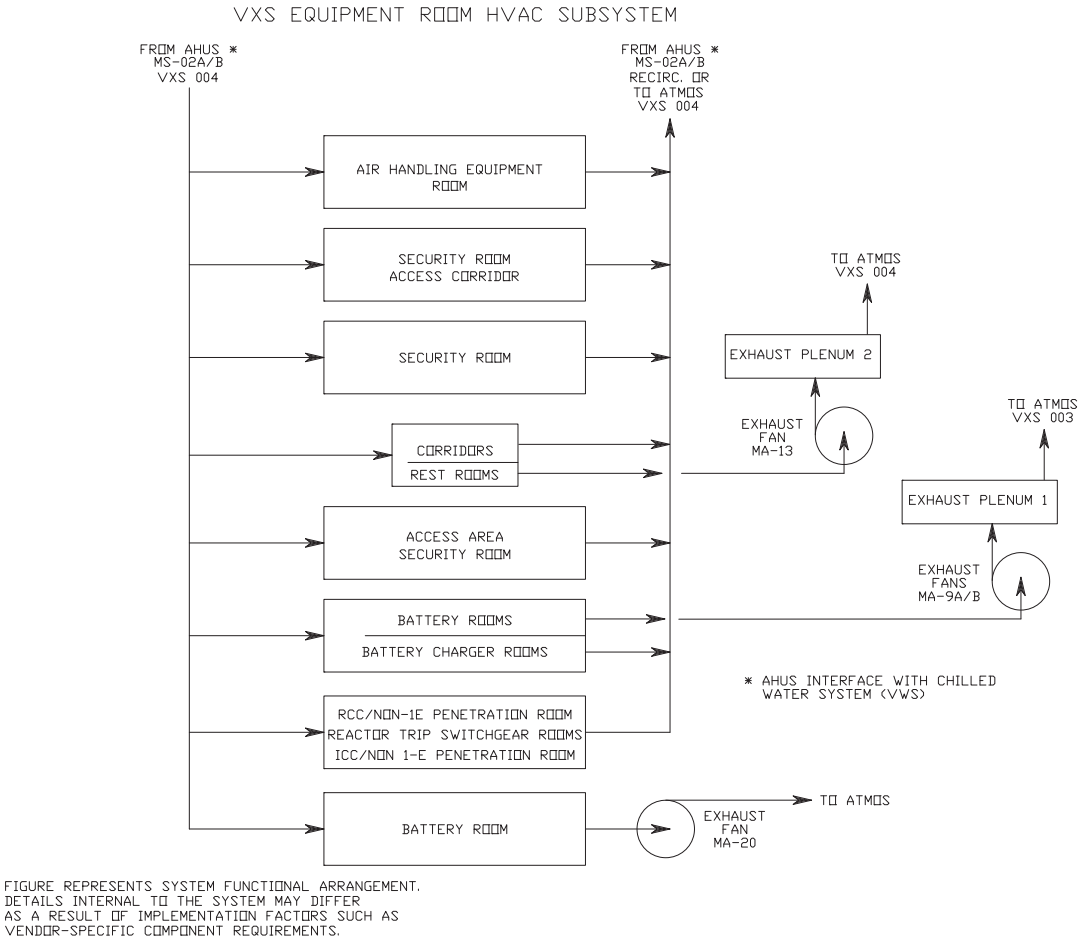

Inside Annex Building

Figure 9.4.2-1 (Sheet 3 of 7)
Annex/Aux Non-Radioactive Ventilation System
Piping and Instrumentation Diagram
(REF) VXS 005 & 006

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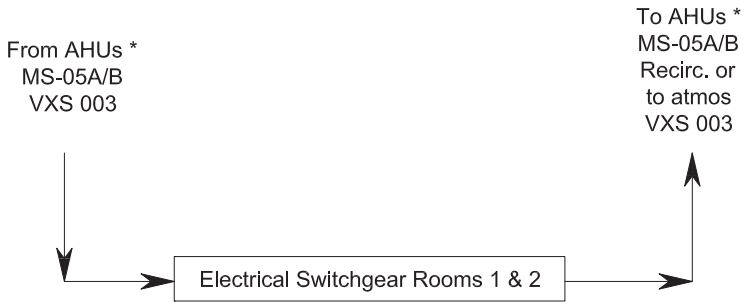
VXS SWITCHGEAR ROOM HVAC SUBSYSTEM



*AHUs interface with chilled water system (VWS)

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Inside Annex Building

Figure 9.4.2-1 (Sheet 4 of 7)
Annex/Aux Non-Radioactive Ventilation System
Piping and Instrumentation Diagram
(REF) VXS 005

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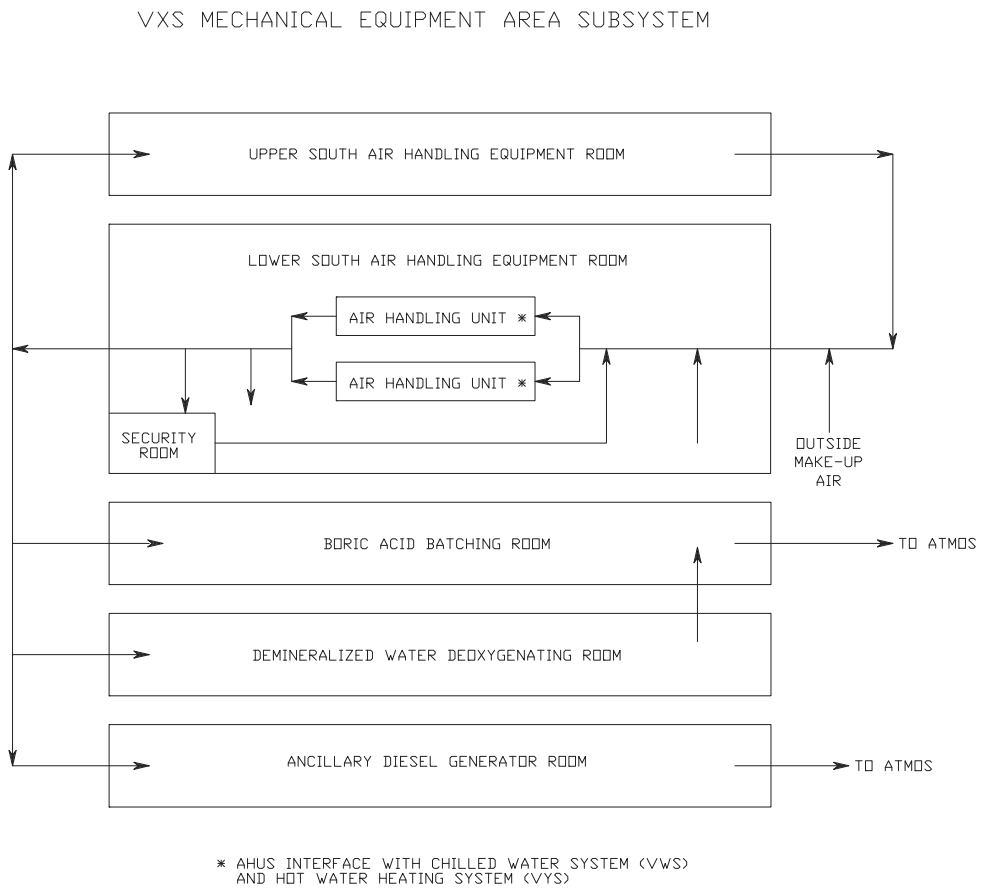

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Inside Annex Building
Figure 9.4.2-1 (Sheet 5 of 7)
Annex/Aux Non-Radioactive Ventilation System
Piping and Instrumentation Diagram
(REF) VXS 010

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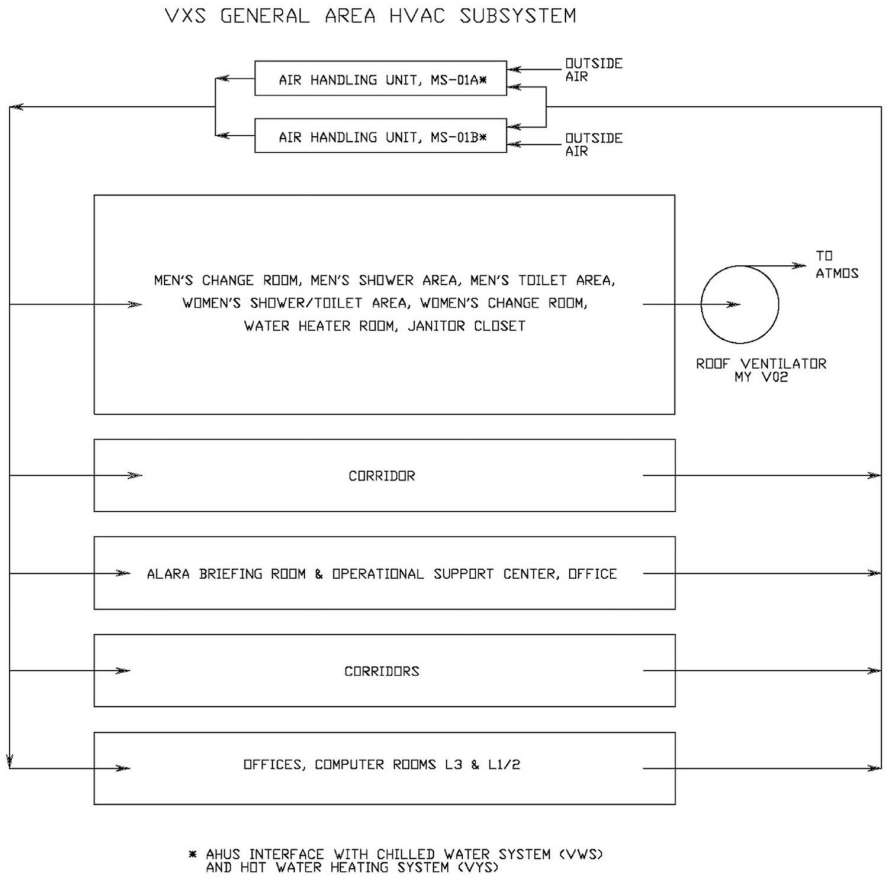

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Inside Annex Building

Figure 9.4.2-1 (Sheet 6 of 7)
Annex/Aux Non-Radioactive Ventilation System
Piping and Instrumentation Diagram
(REF) VXS 002

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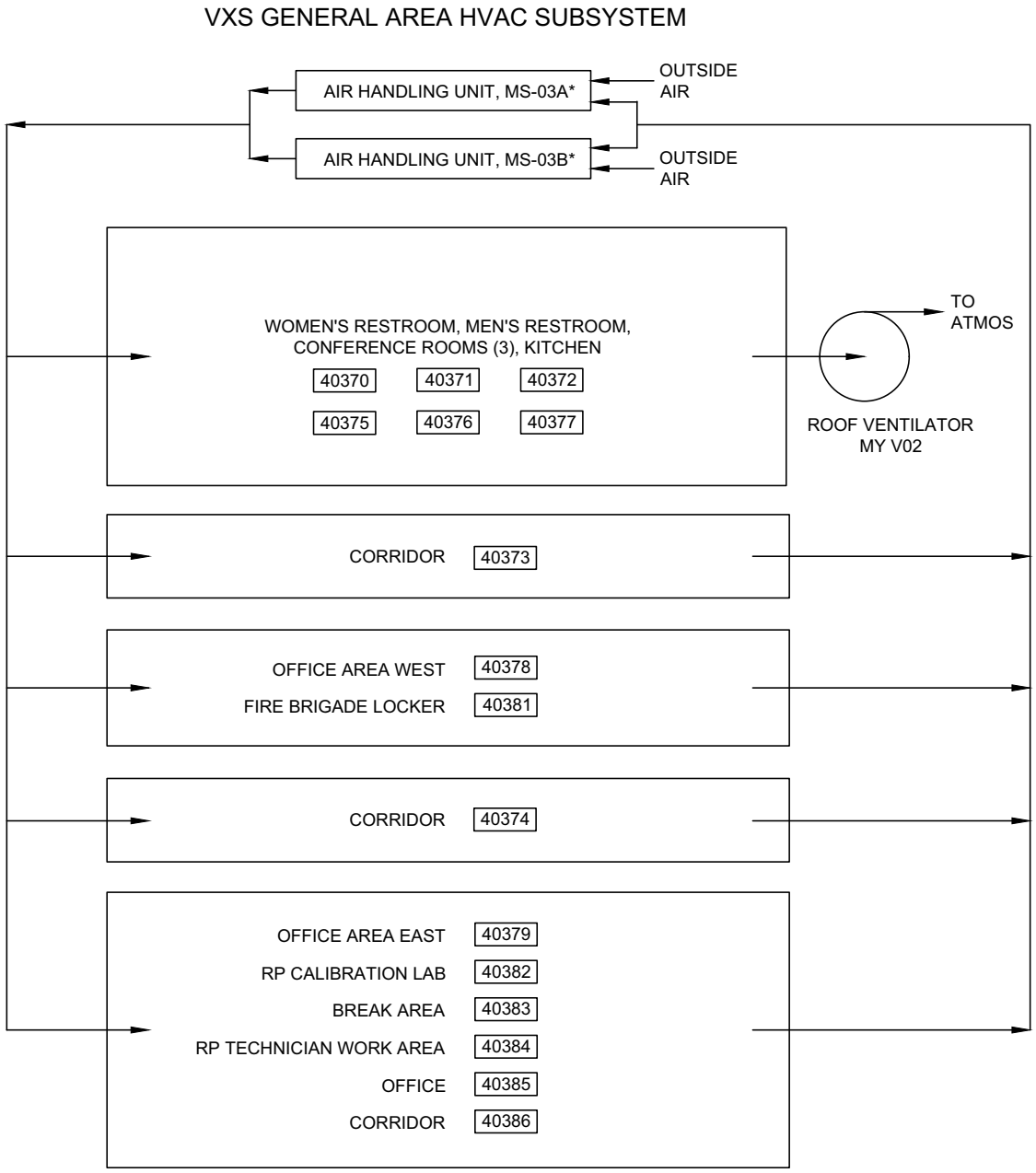

* AHUS INTERFACE WITH CHILLED WATER SYSTEM (VWS)
AND HOT WATER HEATING SYSTEM (VYS)

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Inside Annex Building

Figure 9.4.2-1 (Sheet 7 of 7)
Annex/Aux Non-Radioactive Ventilation System
Piping and Instrumentation Diagram
(REF) VXS 002

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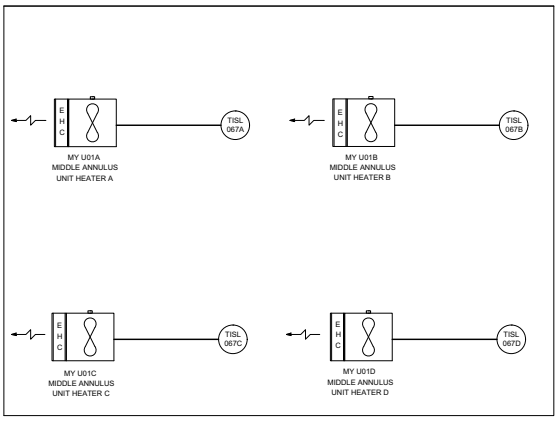

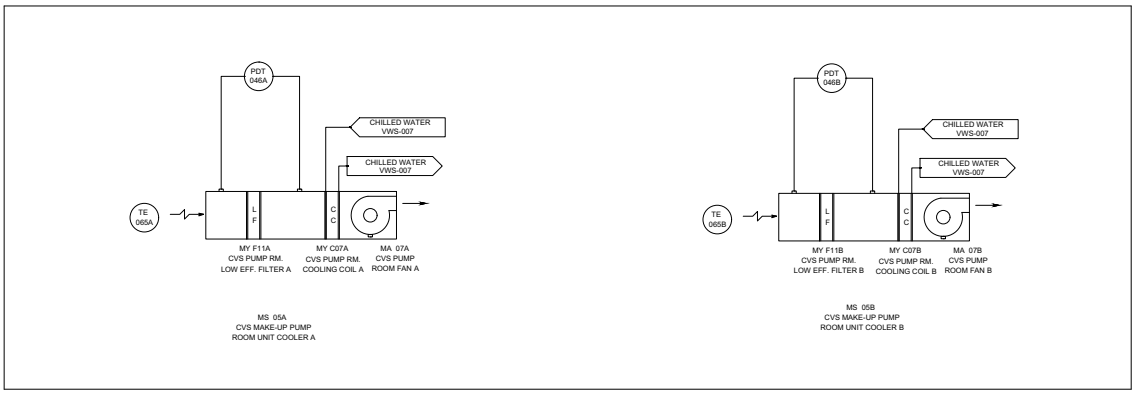

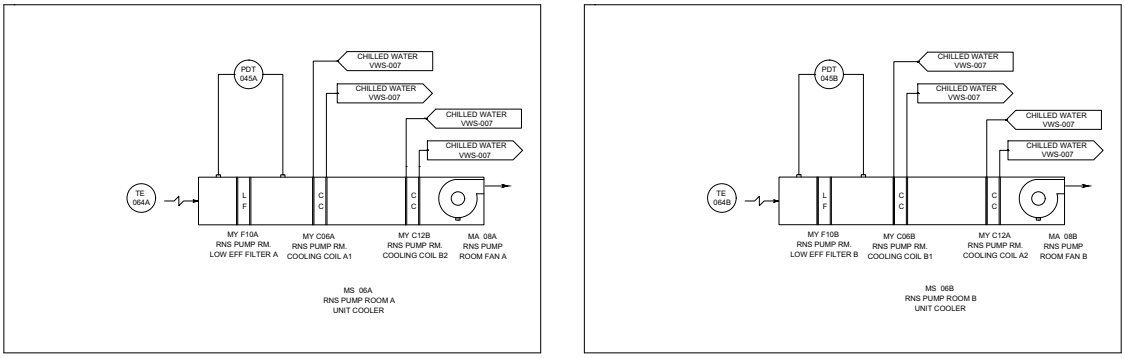

SYSTEM COMPONENTS ARE CLASS D OR CLASS E UNLESS NOTED.

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT, DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Inside Auxiliary Building

Figure 9.4.3-1 (Sheet 1 of 4)
Simplified Radiologically Controlled Ventilation System Piping and Instrumentation Diagram (REF) VAS 001

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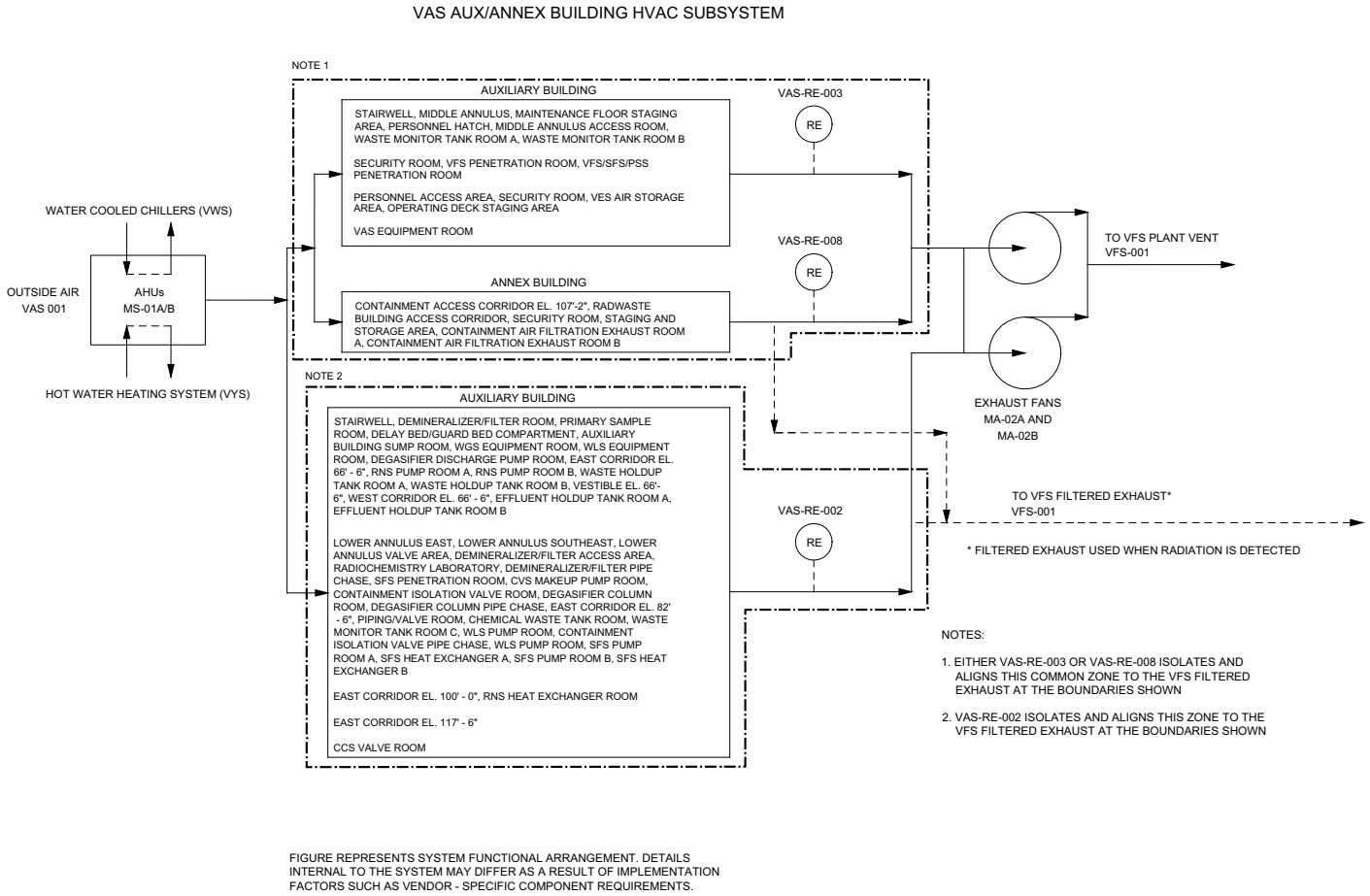

Figure 9.4.3-1 (Sheet 2 of 4)
Radiologically Controlled Area Ventilation System
Piping and Instrumentation Diagram
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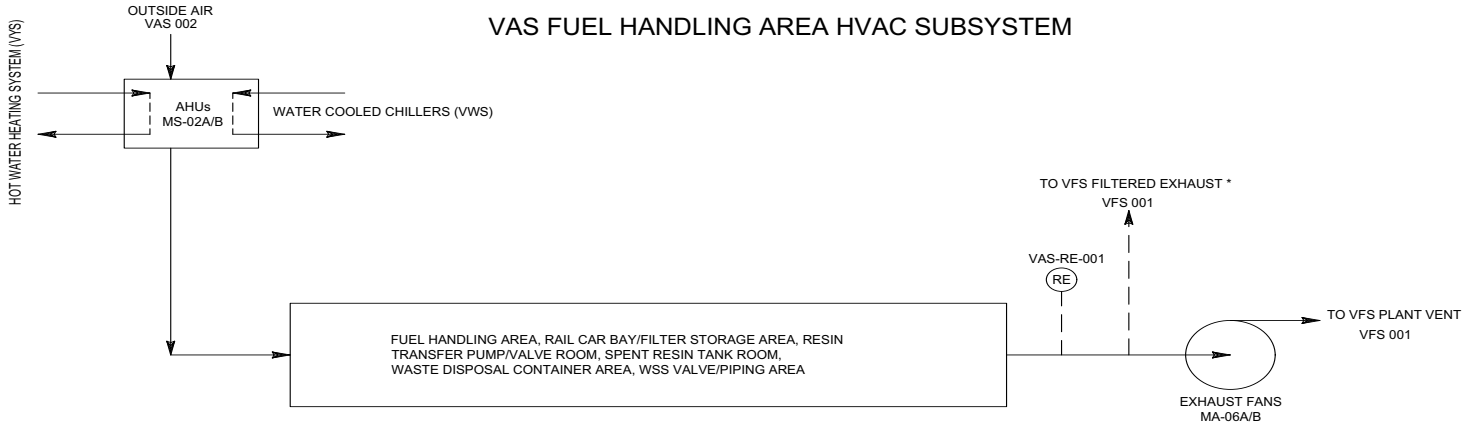

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR -SPECIFIC COMPONENT REQUIREMENTS.

Figure 9.4.3-1 (Sheet 3 of 4)
Radiologically Controlled Area Ventilation System
Piping and Instrumentation Diagram
(REF) VAS 005

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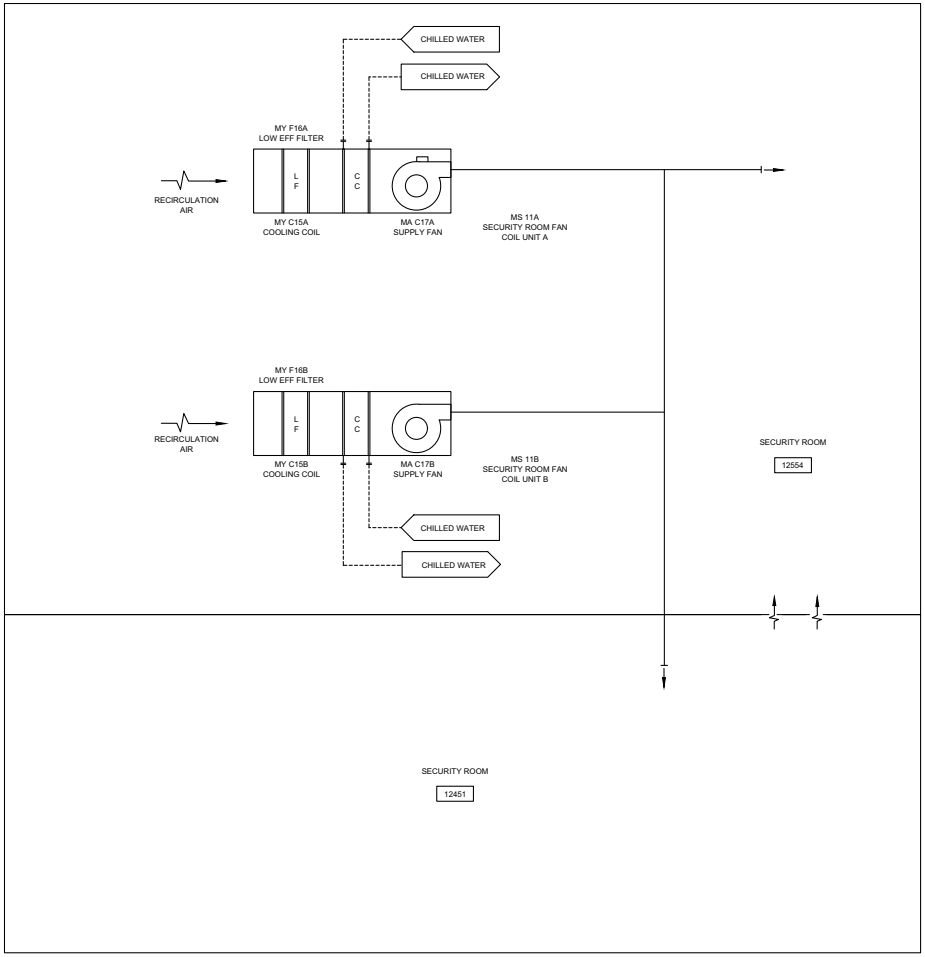

Figure 9.4.3-1 (Sheet 4 of 4)
Inside Auxiliary Building
Simplified Radiologically Controlled Ventilation System
Piping and Instrumentation Diagram
(REF) VAS 009

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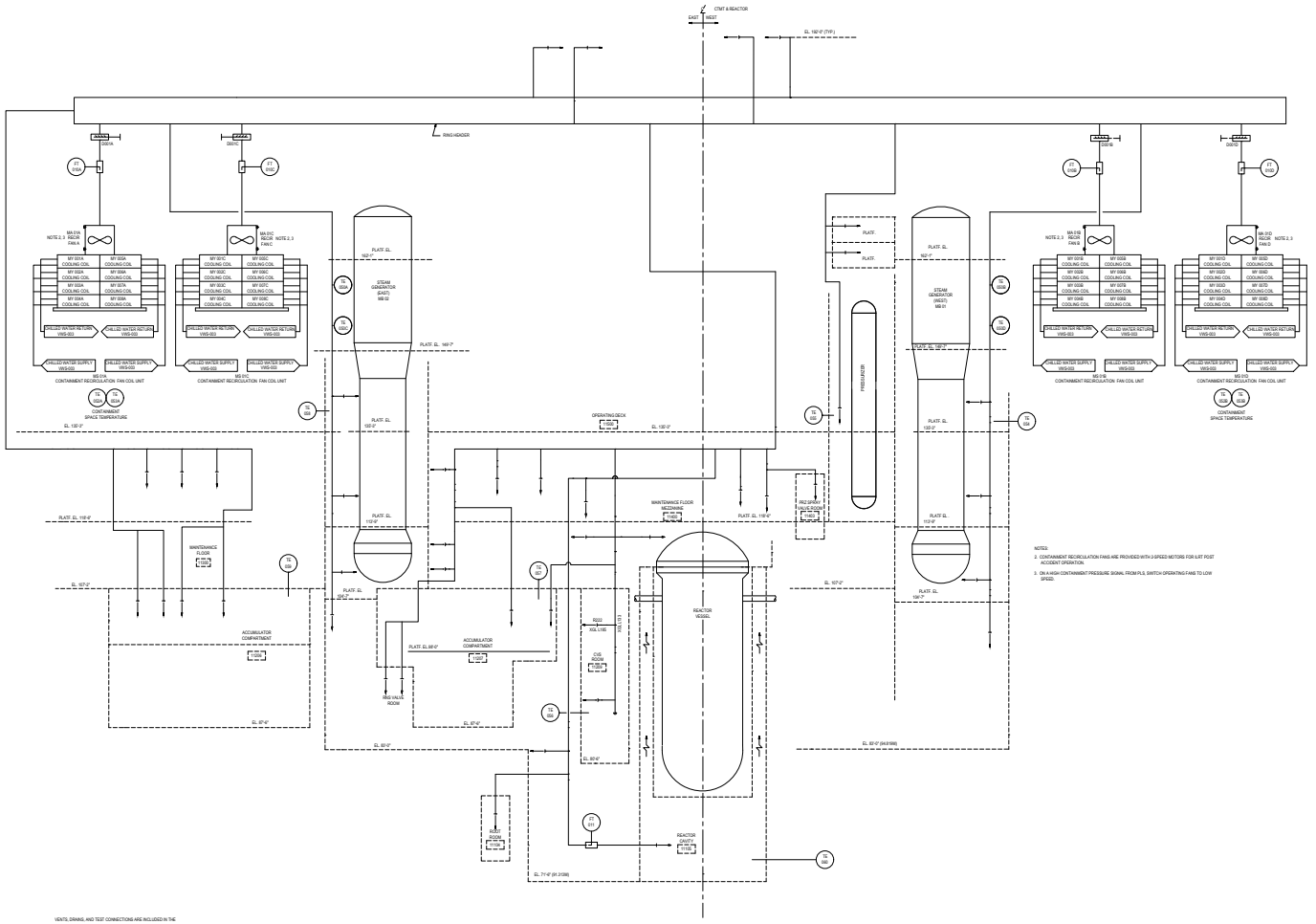

Figure 9.4.6-1
Simplified Containment Recirculation Cooling System Piping and Instrumentation Diagram (REF) VCS 001

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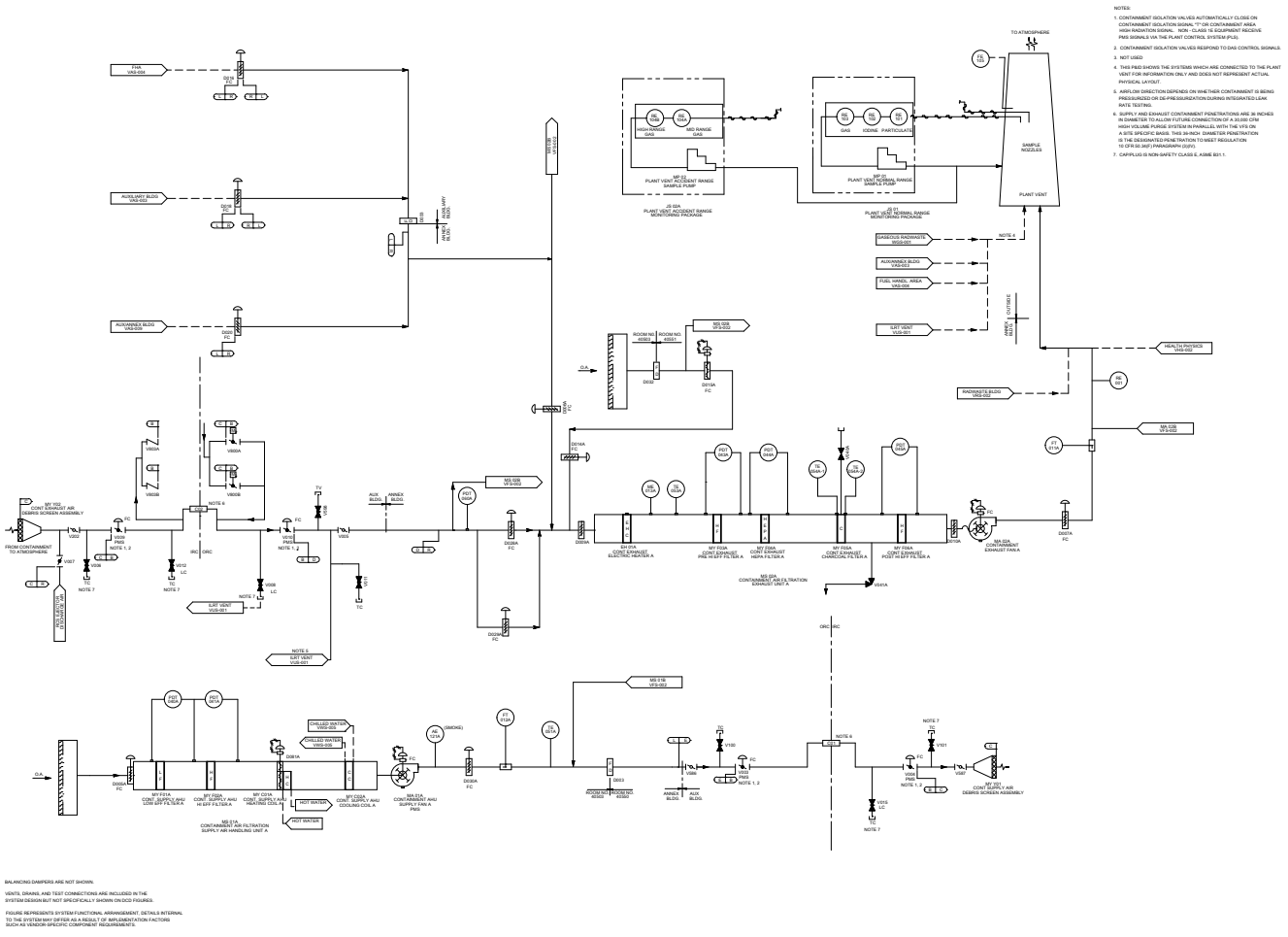

Figure 9.4.7-1 (Sheet 1 of 2)
Simplified Containment Air Filtration System Piping and Instrumentation Diagram
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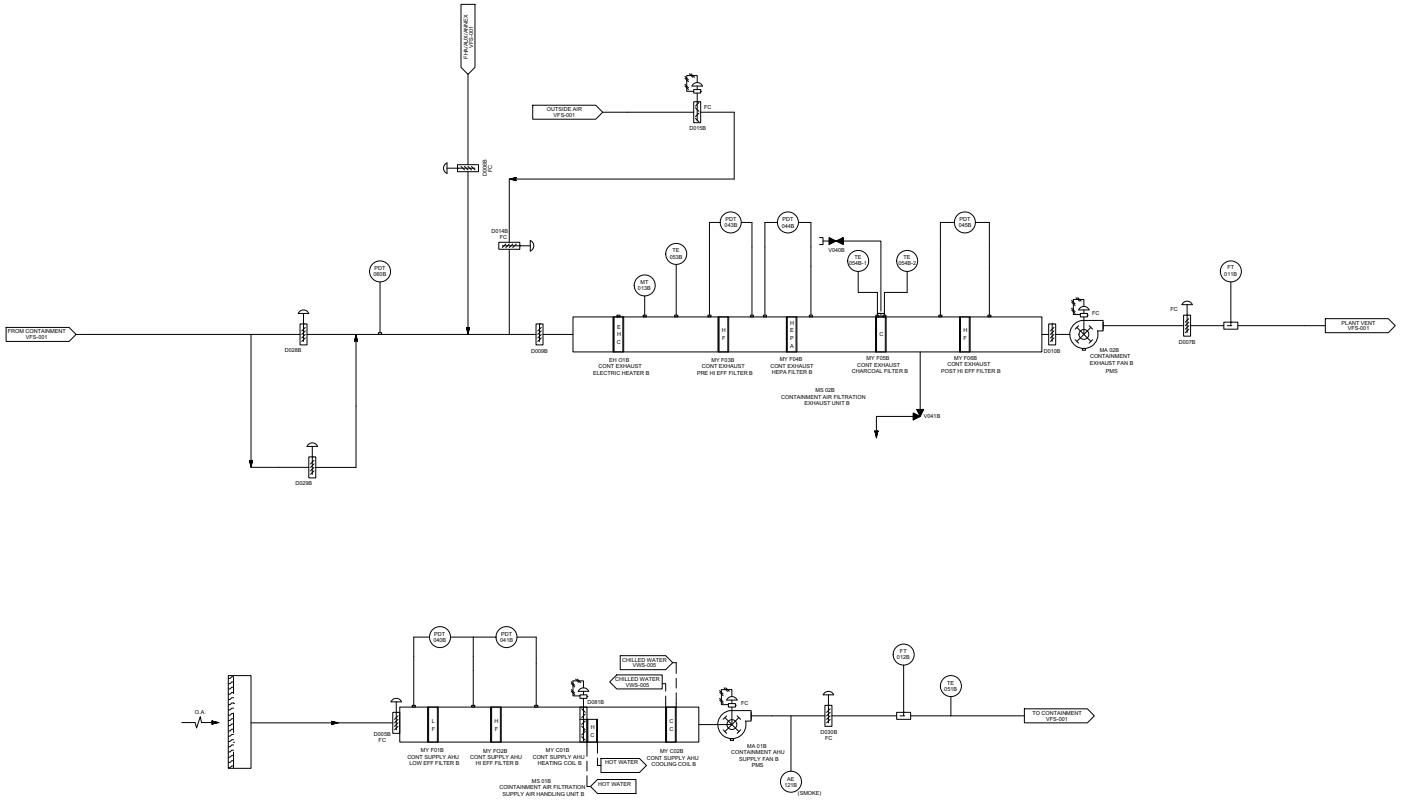

BALANCING DAMPERS ARE NOT SHOWN.

VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Figure 9.4.7-1 (Sheet 2 of 2)
Simplified Containment Air Filtration System
Piping and Instrumentation Diagram
(REF) VFS 002

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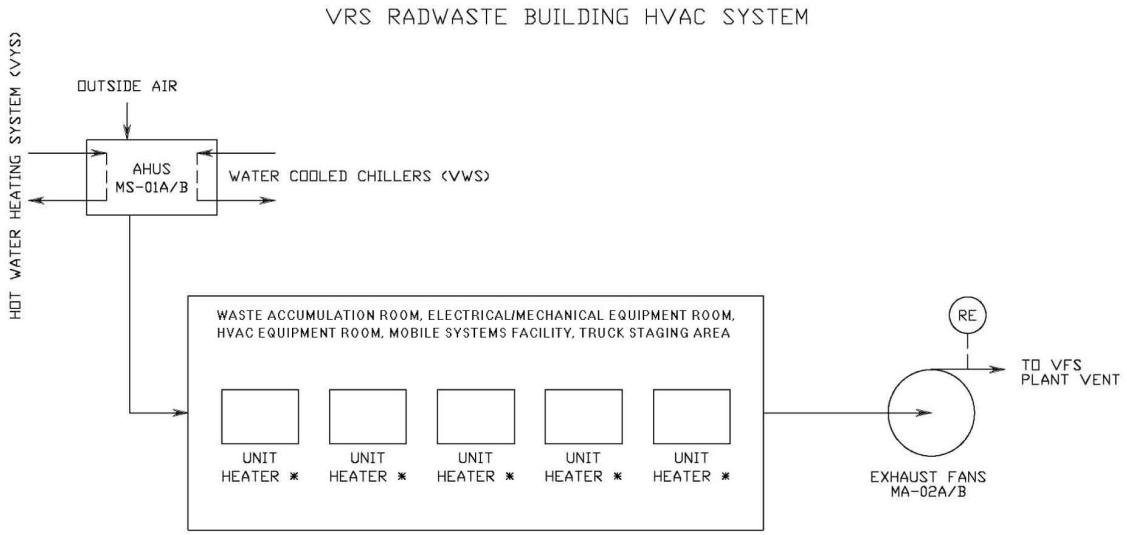

Figure 9.4.8-1
Radwaste Building HVAC System
(REF) VRS 001, 002, 003

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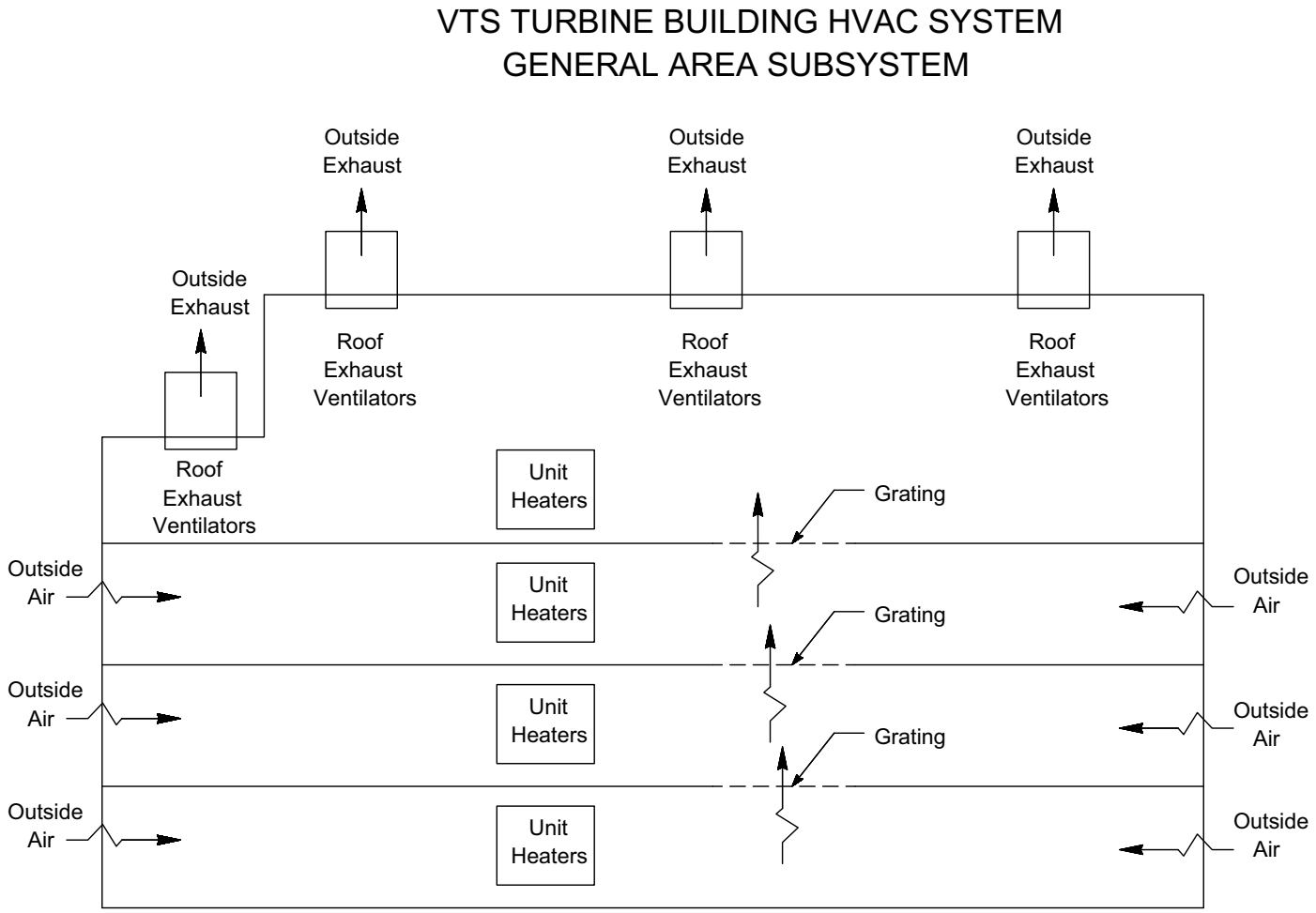

Figure 9.4.9-1
Turbine Building HVAC System
(REF) VTS 001

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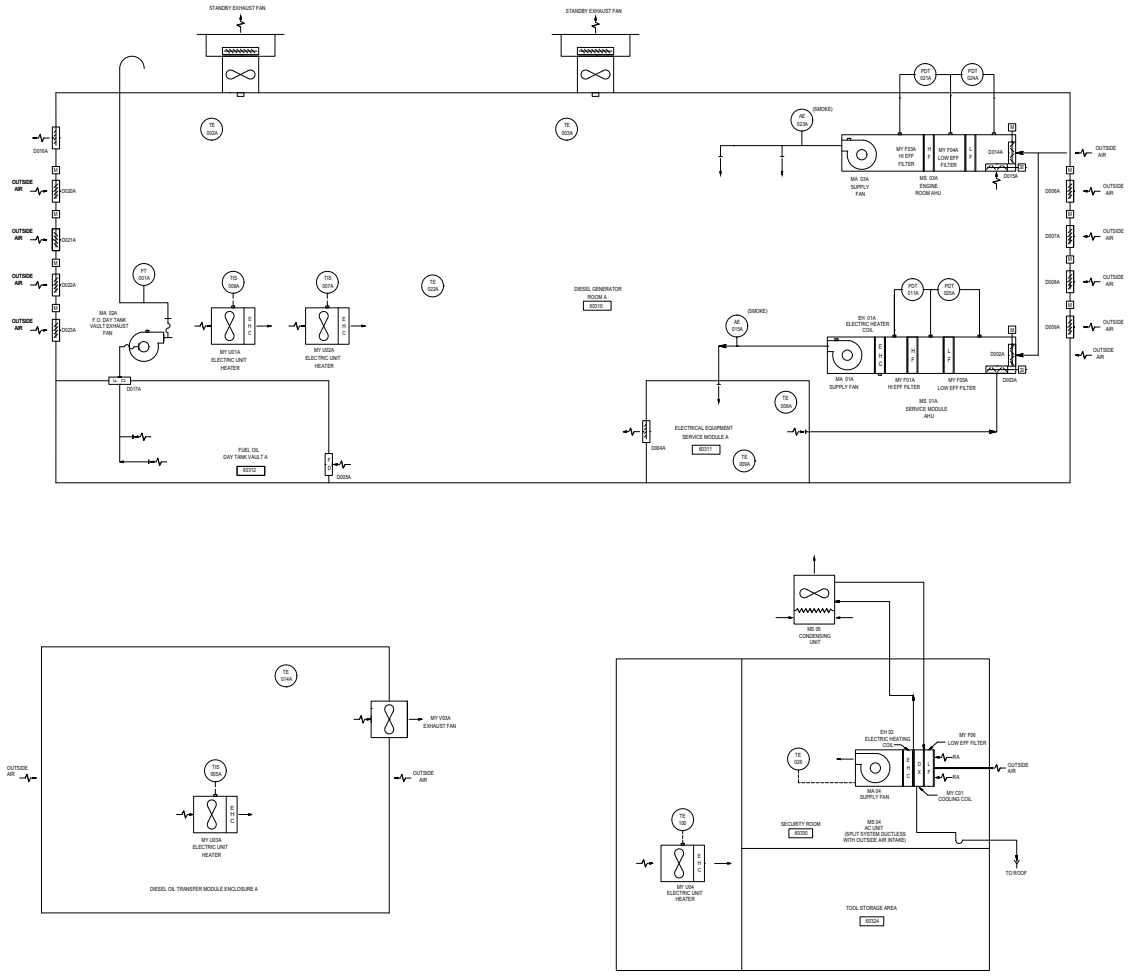

BALANCING DAMPERS ARE NOT SHOWN.
SYSTEM COMPONENTS ARE CLASS D, CLASS E OR CLASS L UNLESS NOTED.
VENTS, DRAINS, AND TEST CONNECTIONS ARE INCLUDED IN THE SYSTEM DESIGN BUT NOT SPECIFICALLY SHOWN ON DCD FIGURES.
FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Figure 9.4.10-1 (Sheet 1 of 2)
Simplified Diesel Generator Building Heating and Ventilation System Piping and Instrumentation Diagram (REF) VZS 001

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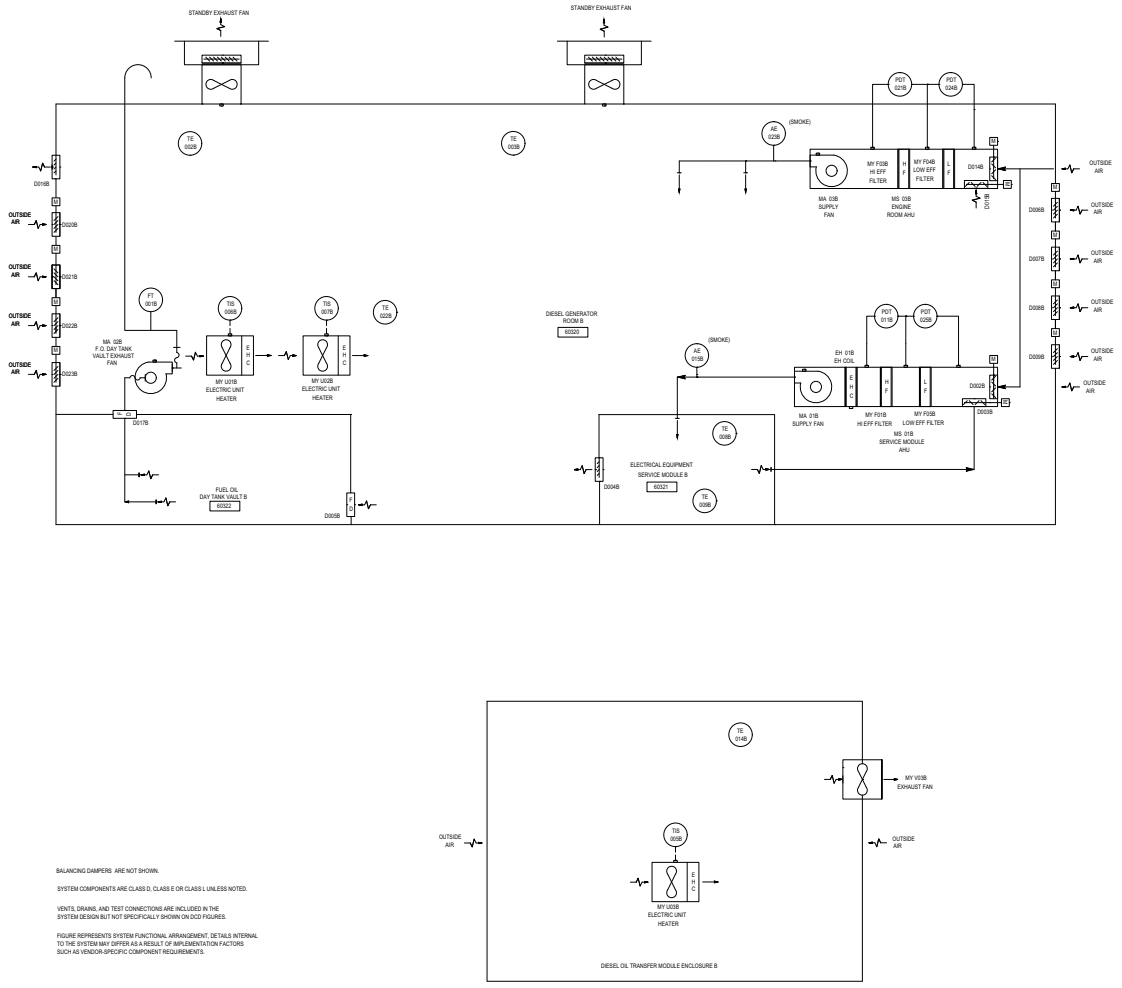

Figure 9.4.10-1 (Sheet 2 of 2)
Simplified Diesel Generator Building Heating and Ventilation System Piping and Instrumentation Diagram (REF) VZS 002

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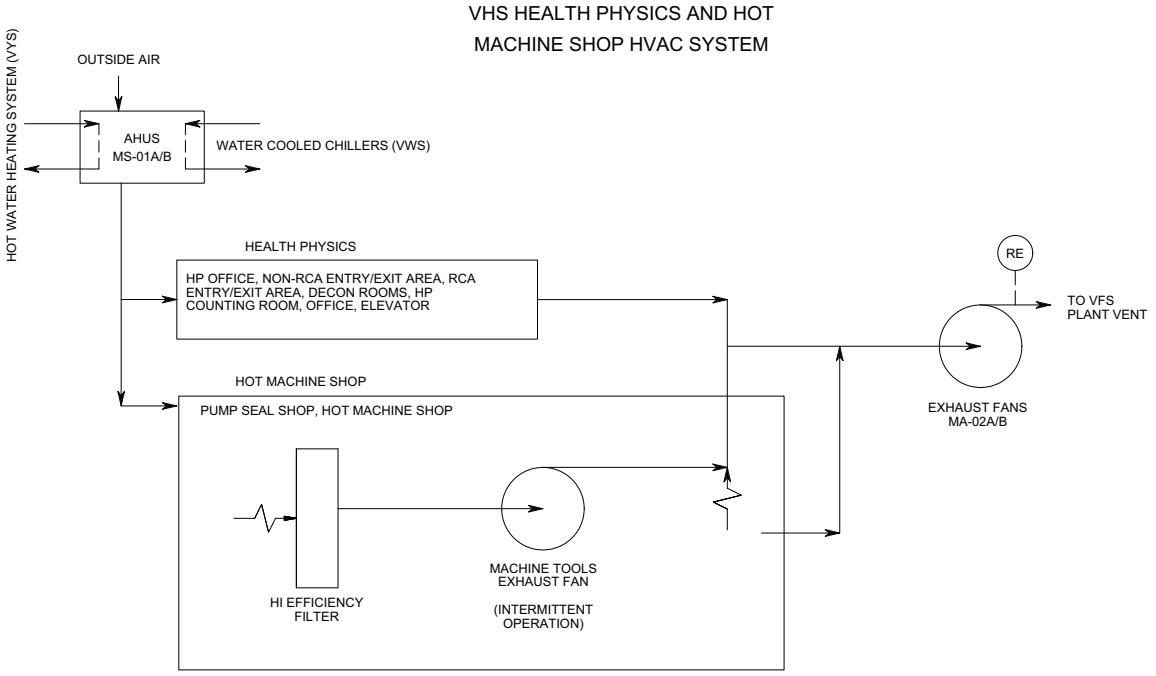

Figure 9.4.11-1
Health Physics and Hot Machine Shop HVAC System
(REF) VHS 001, 002, 003

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9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The primary objectives of the AP1000 fire protection program are to prevent fires and to minimize the consequences should a fire occur. The program provides protection so that the plant can be shut down safely following a fire. The fire protection system (FPS) detects and suppresses fires, and is an integral part of the AP1000 fire protection program. The AP1000 compliance with BTP CMEB 9.5-1 is addressed in Table 9.5.1-1.

9.5.1.1 Design Basis

9.5.1.1.1 Safety Design Basis

To achieve the required high degree of fire safety, and to satisfy fire protection objectives, the AP1000 is designed to:

- Prevent fire initiation by controlling, separating, and limiting the quantities of combustibles and sources of ignition
- Isolate combustible materials and limit the spread of fire by subdividing plant buildings into fire areas separated by fire barriers
- Separate redundant safe shutdown components and associated electrical divisions to preserve the capability to safely shut down the plant following a fire
- Provide the capability to safely shut down the plant using controls external to the main control room, should a fire require evacuation of the control room or damage the control room circuitry for safe shutdown systems
- Separate redundant trains of safety-related equipment used to mitigate the consequences of a design basis accident (but not required for safe shutdown following a fire) so that a fire within one train will not damage the redundant train
- Prevent smoke, hot gases, or fire suppressants from migrating from one fire area to another to the extent that they could adversely affect safe shutdown capabilities, including operator actions
- Provide confidence that failure or inadvertent operation of the fire protection system cannot prevent plant safety functions from being performed
- Preclude the loss of structural support, due to warping or distortion of building structural members caused by the heat from a fire, to the extent that such a failure could adversely affect safe shutdown capabilities
- Provide floor drains sized to remove expected firefighting water flow without flooding safety-related equipment
- Provide firefighting personnel access and life safety escape routes for each fire area
- Provide emergency lighting and communications to facilitate safe shutdown following a fire
- Minimize exposure to personnel and releases to the environment of radioactivity or hazardous chemicals as a result of a fire

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The fire protection system is classified as a nonsafety-related, nonseismic system. Special seismic design requirements are applied to portions of the standpipe system located in areas containing equipment required for safe shutdown following a safe shutdown earthquake, as described in Subsection 9.5.1.2.1.5. In addition, the containment isolation valves and associated piping for the fire protection system are safety-related (Safety Class 2) and seismic Category I. The fire protection system is not required to remain functional following a plant accident or the most severe natural phenomena, except as indicated below for a safe shutdown earthquake.

The fire protection system is designed to perform the following functions:

- Detect and locate fires and provide operator indication of the location
- Provide the capability to extinguish fires in any plant area, to protect site personnel, limit fire damage, and enhance safe shutdown capabilities
- Supply fire suppression water at a flow rate and pressure sufficient to satisfy the demand of any automatic sprinkler system plus 500 gpm for fire hoses, for a minimum of 2 hours
- Maintain 100 percent of fire pump design capacity, assuming failure of the largest fire pump or the loss of offsite power
- Following a safe shutdown earthquake, provide water to hose stations for manual firefighting in areas containing safe shutdown equipment
- Satisfy the requirements of the passive containment cooling system as an alternate source of water to wet the containment dome or to refill the passive containment cooling water storage tank after a loss-of-coolant accident, if the fire protection system is available
- Provide an alternate supply of cooling water to the normal residual heat removal system heat exchanger after a loss of normal component cooling water system function.
- Provide nonsafety-related containment spray capability for severe accident management.

9.5.1.1.2 Power Generation Design Basis

AP1000 fire prevention, control, detection, and suppression features provide plant and personnel safety. The fire protection analysis (see Appendix 9A) evaluates the adequacy of fire protection for systems and plant areas important to the generation of electricity.

9.5.1.1.3 NonSafety-Related Containment Spray Function

The fire protection system provides a nonsafety-related containment spray function. This function is discussed in Subsection 6.5.2.

9.5.1.2 System Description

9.5.1.2.1 General Description

The fire protection program and the design of the fire protection system conform to the applicable codes and standards listed in Section 3.2, and the following:

- 10 CFR 50.48, Fire Protection (Reference 15)
- General Design Criterion 3, Fire Protection (Reference 16)
- SECY-93-087, Section I.E., Fire Protection (Reference 17)

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Table 9.5.1-1 is a point-by-point description of the conformance of the fire protection program with the guidelines of Branch Technical Position (BTP) CMEB 9.5-1 (Reference 1). AP1000 meets the enhanced fire protection provisions of SECY-93-087 as demonstrated in the fire protection analysis (Appendix 9A). Subsection 9.5.1.8.1.1 discusses use of, exceptions to, and evaluation of the applicable fire protection codes and standards.

The plant includes features to minimize the likelihood that a fire will occur and to limit the spread of fire.

The fire protection system detects fires and provides the capability to extinguish them using fixed automatic and manual suppression systems, manual hose streams, and/or portable firefighting equipment. The fire protection system consists of a number of fire detection and suppression subsystems, referred to as systems, including:

- Detection systems for early detection and notification of a fire
- A water supply system including the fire pumps, yard main, and interior distribution piping
- Fixed automatic fire suppression systems
- Manual fire suppression systems and equipment, including hydrants, standpipes, hose stations and portable fire extinguishers

The fire detection and suppression systems are described later in this subsection.

9.5.1.2.1.1 Plant Fire Prevention and Control Features

Architectural and Structural Features

Plant buildings use noncombustible structural materials, primarily reinforced concrete, gypsum, masonry block, structural steel, steel siding, and concrete/steel composite material. Fireproofing of structural steel is not normally required, but the effects of heat generated by a fire are considered in the design. Localized structural steel fireproofing is provided as required, based on a realistic analysis of the time-temperature fire effects on the structural members. Heat transfer analyses based on the postulated fire are used to determine whether the fire will heat the structural members to a specified critical temperature. Where structural failures could adversely affect safe shutdown capabilities, this analysis of the fire resistance of structural steel members establishes the need for fireproofing.

Firefighting personnel access routes and life safety escape routes are provided for each fire area. Fire exit routes are clearly marked.

Buildings outside primary containment generally have two enclosed stairways for emergency access. Stairwells serving as escape routes, access routes for firefighting, or access routes to areas containing equipment necessary for safe shutdown of the plant are equipped with emergency lighting. Such stairwells, and elevator shafts, which penetrate fire barrier floors, are enclosed in towers. The majority of the stairwell towers in the auxiliary building contain both concrete structural walls and nonstructural walls, consisting of a concrete/steel composite material having a fire resistance rating of at least 2 hours. The fire-resistance rating is based on material testing conducted in accordance with ASTM E119 (Reference 20) and NFPA 251 (Reference 25) as addressed in Table 9.5.1-1. These auxiliary building stairwells are protected from potential missiles by other structures or by the selection of the location of the stairwell remote from potential missile sources. Openings are protected with approved automatic or self-closing doors having a rating of 1.5 hours.

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Some of the walls of the turbine building and annex building stairwell enclosures, which are exposed to the interior of the buildings, are also constructed with a material having a minimum fire rating of 2 hours. However, the turbine building and annex building stairwell enclosure walls that face the yard area are constructed with an exterior siding common to the overall siding used for the turbine and annex buildings.

The main control room is designed to permit rapid detection and location of fires in the underfloor and ceiling spaces and allow ready access for manual firefighting. Due to the need to provide passive cooling capability into the main control room ceiling, it will not be protected against fires from within the main control room. The ceiling will be a fire barrier from fires in the room above the main control room.

Plant Arrangement

The plant is subdivided into fire areas to isolate potential fires and minimize the risk of the spread of fire and the resultant consequential damage from corrosive gases, fire suppression agents, smoke, and radioactive contamination.

Some fire areas are subdivided into fire zones to permit more precise identification of the type and locations of combustible materials, fire detection, and suppression systems. The subdivision into fire zones is based on the configuration of interior walls and floor slabs, and the location of major equipment within each fire area.

Fire barriers are provided in accordance with BTP CMEB 9.5-1 as addressed in Table 9.5.1-1. Three-hour fire barriers are non-combustible and surround fire areas containing safety-related components. The resistance of fire barriers in nonsafety-related areas of the plant may be less than 3 hours, where justified by the fire protection analysis (Appendix 9A).

Three-hour fire barriers provide complete separation of redundant safe shutdown components, including equipment, electrical cables, instrumentation and controls, except where the need for physical separation conflicts with other important requirements, specifically:

- Fire barrier separation is not provided within the main control room fire area because functional requirements make such separation impractical. The risk of fires in the control room is minimized by the reduction in the quantity of electrical cables. Continuous occupancy provides confidence that fires would be quickly detected and suppressed. Should a fire require evacuation of the main control room, the plant can be safely shut down using independent controls at the remote shutdown workstation, located in a separate fire area.

- Fire barrier separation is not provided between the main control room and the room above it from fires in the main control room. There is fire barrier separation between the main control room and the room above it for fires in the room above.

- Fire barrier separation is not provided within the remote shutdown room fire area because the remote shutdown workstation is not required for safe shutdown unless a fire requires evacuation of the main control room.

- Complete fire barrier separation necessary to define a fire area is not provided throughout the primary containment fire area (including the middle and upper annulus zones of the shield building) because of the need to satisfy other design requirements, such as allowing for pressure equalization within the containment following a high-energy line break. Fire protection features and equipment arrangement which define fire zones within the containment fire area provide confidence that at least one train of safe shutdown equipment will remain undamaged following a fire in any fire zone. The quantity of combustible materials is minimized. The use of sealless reactor coolant pump motors has eliminated the need for

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an oil lubrication system. Redundant trains of safe shutdown components are separated whenever possible by existing structural walls, or by distance. Selected cables of a safety-related division which pass through a fire zone of an unrelated division are protected by fire barriers. The fire protection system provides appropriate fire detection and suppression capabilities.

Outside of the primary containment and the main control room, the arrangement of plant equipment and routing of cable are such that safe shutdown can be achieved with all components (except those protected by 3-hour fire barriers) in any one fire area rendered inoperable by fire.

Openings and penetrations through fire barriers are protected in accordance with the guidelines of BTP CMEB 9.5-1 as addressed in Table 9.5.1-1.

Penetrations through the following barriers are rated to withstand a differential pressure of 5 psi to support the conclusions in Appendix 19F:

- Wall I between elevation 117’-6" to 153’-0"
- Wall 5 between elevation 100’-0" to 153’-0"
- Penetrations in the main control room floor (12401) at elevation 117’-6" above corridor 12300
- Penetrations through Wall I into corridor 12300
- Penetrations between Wall I and Wall I.1 into corridor 12300
- Penetrations into the shield building wall from room 12351
- Penetrations into the shield building wall from room 12421
- Penetrations into the shield building wall from room 12553
- Penetrations into the shield building wall from room 12556

The fire protection analysis contains a description of plant fire areas, fire zones, fire barriers, and the protection of fire barrier openings, as well as a description of the separation between redundant safe shutdown components.

Electrical Cable Design, Routing, and Separation

Electrical cable (including fiber optic cable) and methods of raceway construction are selected in accordance with BTP CMEB 9.5-1 as addressed in Table 9.5.1-1. Metal cable trays are used. Rigid metal conduit, thin-wall conduit / tubing, or metal raceways are used for cable runs not embedded in concrete or buried underground. Flexible metallic tubing is used in short lengths for equipment connections.

The insulating and jacketing material for electrical cables are selected to meet the fire and flame test requirements of IEEE Standard 1202 (Reference 18), IEEE Standard 383 (Reference 3) or the standards for the corresponding test types as listed in Tables K-1 and K-2 in NEI 04-02, Rev 3 excluding the option to use flame source, oil, or burlap.

The design, routing, and separation of cable and raceways are further described in Section 8.3.

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Control of Combustible Materials

The plant is constructed of noncombustible materials to the extent practicable. The selection of construction materials and the control of combustible materials are generally in accordance with the design portions of BTP CMEB 9.5-1 as addressed in Table 9.5.1-1 and the design portions of Section 6.3 or Section 3.3 (respectively) of NFPA 804 (Reference 2) with exceptions to NFPA 804 identified in Table 9.5.1-3.

The storage and use of hydrogen are according to NFPA 50A and NFPA 50B (Reference 2). Hydrogen lines in safety-related areas are designed to seismic Category I requirements.

Ventilation systems are designed to maintain the hydrogen concentration in the battery rooms well below 2 percent by volume, as described in Subsections 9.4.1 and 9.4.2.

The containment recirculation cooling system (VCS) (Subsection 9.4.6) provides sufficient air flow through the reactor vessel cavity to mix any off-gassing from the neutron shield blocks with the containment volume. The mixing flow rate maintains the hydrogen concentration in containment well below 2 percent by volume. The containment air filtration system (VFS) (Subsection 9.4.7) also provides periodic flow of outdoor air to purge and filter the containment atmosphere.

The turbine lubrication oil system, located in the turbine building, is separated from areas containing safety-related equipment by 3-hour rated fire barriers.

Outdoor oil-filled transformers are separated from plant buildings according to the design portions of NFPA 804 (Reference 2) with exceptions to NFPA 804 identified in Table 9.5.1-3.

The diesel fuel oil storage tanks and the diesel fuel oil transfer pump enclosure are located in the yard area more than 50 feet from any safety-related structure. Potential oil spills from the storage tanks are confined by a diked enclosure. A diesel generator fuel day tank is located within each diesel generator room and is enclosed in a 3-hour fire rated barrier.

The diesel fuel supply for the ancillary diesel generators is in the same room as the diesel generators. The ancillary diesel generator room is separated from the rest of the annex building by a 3-hour rated fire barrier.

The diesel fuel supply for the diesel-driven fire pump is in the diesel-driven fire pump enclosure. The diesel pump enclosure is located in the yard more than 50 feet from safety-related structures. The enclosure includes a fire detector which produces an audible alarm locally with both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by operation of an automatic sprinkler system or manually, using hose streams or portable extinguishers.

Quantities and locations of other combustible materials are identified in the fire protection analysis (see Appendix 9A).

Control of Radioactive Materials

As described in the fire protection analysis, materials that collect or contain radioactivity, such as spent ion exchange resins and filters, are protected and stored in accordance with BTP CMEB 9.5-1 as addressed in Table 9.5.1-1.

9.5.1.2.1.2 Fire Detection and Alarm Systems

Fire detection and alarm systems are provided where required by the fire protection analysis, in accordance with the design portions of BTP CMEB 9.5-1 as addressed in Table 9.5.1-1 and the

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design portions of NFPA 72 (Reference 2). Fire detection and alarm systems are generally in accordance with the design portions of NFPA 804 (Reference 2) with exceptions identified in Table 9.5.1-3.

Fire detectors respond to smoke, flame, heat, or the products of combustion. The installation of fire detectors is in accordance with NFPA 72 (Reference 2) as addressed in Table 9.5.1-3 and the manufacturer's recommendations. The selection and installation of fire detectors is also based on consideration of the type of hazard, combustible loading, the type of combustion products, and detector response characteristics. The types of detectors used in each fire area are identified in the fire protection analysis.

The fire detection system provides audible and visual alarms and system trouble annunciation in the main control room and at the remote fire alarm annunciator panel. Annunciation circuits connecting zone, main, and remote annunciation panels are electrically supervised.

Each fire detection, indicating, and alarm unit is provided with reliable ac electrical power from the non-Class 1E uninterruptible power supply system. This system is described in Subsection 8.3.2.1.2.

9.5.1.2.1.3 Fire Water Supply System

The fire water supply system is generally designed in accordance with the design portions of BTP CMEB 9.5-1 as addressed in Table 9.5.1-1 and the NFPA standards referenced in NFPA 804 (Reference 2) with exceptions identified in Table 9.5.1-3.

Fire water can be supplied from either the primary or secondary fresh water storage tanks. Both fire water storage tanks contain a standpipe with inlets above the water volume dedicated for fire protection use that is used as a gravity-fed alternate makeup source for the service water system cooling water basin. The secondary fire water tank contains water for use by the fire protection system and the containment spray system.

There are two 100-percent capacity fire pumps. The lead pump is electric motor-driven and the secondary pump is diesel engine-driven. A motor-driven jockey pump is used to keep the fire water system full of water and pressurized, as required. For additional information regarding the fire water tanks and pumps, see Subsection 9.5.1.2.3.

The fire water tanks are permanently connected to the fire pumps suction piping and are arranged so that the pumps can take suction from either or both tanks. One tank is normally isolated from the system. Redundant cross-connect lines are provided to increase reliability of the pump suction lines to the tanks. Piping between the fire water sources and the fire pumps is in accordance with NFPA 20 (Reference 2). A failure in one tank or its piping will not cause both tanks to drain.

Fire protection water is distributed by an underground yard main loop, designed in accordance with NFPA 24 (Reference 2). The yard main includes a building interior header that distributes water to suppression systems within the main plant buildings. Indicator valves provide sectionalized control and permit isolation of portions of the yard main for maintenance or repair. An indicator valve also separates the individual fire pump connections to the main.

Sprinkler and standpipe systems are supplied by connections from the yard main. Where plant areas, other than the containment and outlying buildings, are protected by both sprinkler systems and standpipe systems, the connections from the yard main are arranged so that a single active failure or a crack in a moderate energy line cannot impair both systems.

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Manual valves for sectionalized control of the yard main or for shutoff of the water supply to suppression systems are electrically supervised if located above ground and administratively controlled if located underground.

Hydrants are provided on the yard main in accordance with NFPA 24 (Reference 2), at intervals of up to about 250 feet. They provide hose stream protection for every part of each building and two hose streams for every part of the interior of each building not covered by standpipe protection, excluding certain remote areas of the shield building. The lateral to each hydrant is controlled by an isolation valve.

Hose houses are in accordance with NFPA 24 (Reference 2). They are located at intervals of not more than 1000 feet along the yard main.

Outdoor fire water piping and water suppression systems located in unheated areas of the plant are protected from freezing.

A permanent connection between the fire protection system and the component cooling water system in the annex building is normally isolated by two valves in series.

A permanent connection between the fire protection system and the containment spray system in the containment is normally isolated by two valves in series.

Threads compatible with those used by the off-site fire department are provided on all hydrants, hose couplings and standpipe risers, or a sufficient number of thread adapters compatible with the off-site fire department are provided.

9.5.1.2.1.4 Automatic Fire Suppression Systems

Automatic fire suppression systems are generally in accordance with the design portions of BTP CMEB 9.5-1 as addressed in Table 9.5.1-1 and the design portions of the NFPA standards referenced in NFPA 804 (Reference 2) with exceptions identified in Table 9.5.1-3, with consideration of the unique aspects of each application, including building characteristics, materials of construction, environmental conditions, fire area contents, and adjacent structures.

Fixed automatic fire suppression systems are provided based on the results of the fire protection analysis.

The selection of automatic suppression systems for each plant area is based on the design related portions of guidance of NFPA 804 (Reference 2) with exceptions identified in Table 9.5.1-3. Water systems are preferred, but the use of automatic water suppression systems for firefighting in radiation areas is minimized because of the possible spread of contamination. Halon and carbon dioxide fixed flooding systems are not used.

The fire protection analysis describes the fire suppression systems provided for each fire area.

Automatic Water Suppression Systems

Automatic sprinkler and water spray systems are provided generally in accordance with the applicable design requirements of NFPA 13 and NFPA 15 (Reference 2) with exceptions identified in Table 9.5.1-3. Each system consists of overhead piping and components from a water supply valve to the point where water discharges from the system. Some systems have a control valve that is actuated automatically by the fire detection system. Each system has a status monitoring device for actuating an alarm when the system is in operation.

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Preaction sprinkler systems are used where the leakage or inadvertent actuation of water-filled sprinkler systems could produce undesirable consequences, such as water discharge on equipment important to continued plant operation.

Each type of automatic sprinkler and automatic water spray system used on AP1000 is briefly described below:

- **Wet Pipe** - A sprinkler system employing closed (fusible link operated) sprinklers attached to a water-filled piping network. Water discharges immediately from those sprinklers where the heat from a fire is sufficient to melt the fusible link. System operation is terminated manually by shutting the water-supply valve.

- **Dry Pipe** - A sprinkler system employing closed sprinklers attached to a piping network containing pressurized air. Heat from a fire opens one or more sprinklers, releasing the air and permitting water supply pressure to open the dry pipe valve. Water flows into the piping network and discharges from the open sprinklers. System operation is terminated manually by shutting the water-supply valve.

- **Preaction** - A sprinkler system employing closed sprinklers attached to a dry piping network, with fire detector(s) installed in the same areas as the sprinklers. Operation of the fire detection system opens a preaction valve, which permits water to flow into the sprinkler piping network and to be discharged from any sprinklers that may have been opened by the fire. System operation is terminated manually by shutting the water-supply valve.

- **Deluge Sprinkler or Water Spray System** - A system employing open sprinklers or spray nozzles attached to a dry piping network, with fire detector(s) installed in the same areas as the sprinklers. Operation of the fire detection system opens a deluge valve, which permits water to flow into the sprinkler piping network and to be discharged from all the sprinklers or spray nozzles. System operation is terminated manually by shutting the water-supply valve.

9.5.1.2.1.5 Manual Fire Suppression Systems

Manual fire suppression capability is provided in areas that do not require an automatic suppression system. Plant areas that have an automatic suppression system also have manual backup fire suppression capability.

Manual fire suppression capabilities include the yard main fire hydrants and hose stations described in Subsection 9.5.1.2.1.3.

Standpipe and Hose Systems

Standpipe systems are provided for each building generally in accordance with NFPA 14 (Reference 2) design requirements for Class III service as addressed in Table 9.5.1-1 and with exceptions identified in Table 9.5.1-3. Wet standpipe systems are used except inside containment with exceptions identified in Table 9.5.1-3. Individual standpipes are at least 4 inches in diameter for multiple hose connections and 2.5 inches in diameter for single hose connections.

Hose stations are located to facilitate access for firefighting, as described in the fire protection analysis. Areas that contain, or could present a fire exposure event to, safety-related equipment are within reach of at least one effective hose stream. Alternative hose stations are provided for an area where the fire could block access to a single hose station serving that area. To the maximum extent practical, hose stations are located outside of high-radiation areas.

Each hose station has not more than 100 feet of 1.5-inch woven-jacket lined fire hose. Nozzles are provided at each station.

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Seismic Standpipe System

The standpipe system serving areas containing equipment required for safe shutdown following a safe shutdown earthquake is designed and supported so that it can withstand the effects of a safe shutdown earthquake and remain functional. The seismically analyzed standpipe system is illustrated on Figure 9.5.1-1. This system also supplies water to automatic suppression systems in the nonradiologically controlled portion of the auxiliary building (see Appendix 9A).

The seismic standpipe system is operated in the same manner during normal plant operation or following a safe shutdown earthquake. It is supplied with water from the safety related passive containment cooling system storage tank and normally operates independently of the rest of the fire protection system. The supply line draws water from a portion of the storage tank, using water allocated for fire protection. This volume of water is sufficient to supply two hose streams, each with a flow of 75 gallons per minute, for 2 hours.

The portion of the system outside containment is a wet standpipe system that is pressurized by the static head of water in the passive containment cooling system tank. The portion of the system inside containment is a dry standpipe system. The supply valve is normally closed for containment isolation. During shutdown periods when the containment is occupied, when operation of nonsafety-related containment spray is required, or when containment access is required to fight a fire, the valve is opened to pressurize the system.

In the unlikely event that the water supply from the passive containment cooling system is unavailable or additional water is needed, the seismic standpipe system can be supplied from the fire main by opening the normally closed cross-connect valve with the plant fire main.

A passive containment cooling ancillary water storage tank is provided to supply the seismic standpipe system following a safe shutdown earthquake and after actuation of the passive containment cooling system. The tank is designed and supported so that it can withstand the effects of a safe shutdown earthquake and remain functional. A dedicated portion of the storage capacity of the tank is sufficient to supply two hose streams, each with a flow of 75 gallons per minute, for 2 hours. Normally much more water is available. (Refer to Subsection 6.2.2 for additional information.)

A failure of the seismic standpipe system does not prevent successful operation of the passive containment cooling system. A leak in the standpipe system could result in the loss of only a limited amount of water from the passive containment cooling system storage tank, even if no action were taken to isolate the leak. The volume of water allocated for fire protection is not required for passive containment cooling.

Portable Fire Extinguishers

Portable fire extinguishers are provided throughout the plant. Portable extinguishers are readily accessible for use in high radiation areas but are not located within those areas unless the fire protection analysis indicates that a specific requirement exists.

9.5.1.2.2 System Operation

The fire protection system normally operates in an active standby mode. The fire water supply piping is kept full and pressurized by operation of the jockey pump. Shutoff valves controlling fire suppression systems are normally aligned in the open position. Fire detection and alarm circuits are normally energized and monitored for trouble or loss of power as described in Subsection 9.5.1.2.1.2.

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When a fire is detected, the fire detection system produces an audible alarm locally, except in containment where the fire detection system produces a visual alarm locally, and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel.

Where the fire area is protected by an automatic suppression system, operation of the suppression system begins as described in Subsection 9.5.1.2.1.4. Where the fire area is protected by manual suppression methods, the fire brigade reacts to control and extinguish the fire.

Ventilation system fire dampers close automatically against full airflow on high temperature to control the spread of fire and combustion products. Fire dampers serving certain safety-related, smoke-sensitive areas are also closed in response to an initiation signal from the fire detection system. Smoke is removed from the fire area as described in the fire protection analysis.

When water pressure in the yard main begins to fall, due to a demand for water from automatic or manual suppression systems, the motor-driven pump starts automatically on a low-pressure signal. If the motor-driven pump fails to start, the diesel-driven pump starts upon a lower pressure signal. The pump continues to run until it is stopped manually.

Firefighting activities continue until the fire is extinguished. Suppression systems are stopped manually. Operator actions are taken to repair and restore affected detection, alarm, and suppression systems to standby status.

9.5.1.2.3 Component Description

Selected fire protection system components are described below. Table 9.5.1-2 contains additional component data for fire protection equipment.

Fire Water Storage Tanks

Two separate fresh water storage tanks are provided for fire protection in accordance with NFPA 22 (Reference 2). The storage capacity of each tank is sufficient to maintain the design fire pump flow rate for at least 2 hours. Either tank can be refilled from the raw water system within 8 hours. Freeze protection is provided as needed using electric immersion heaters.

Passive Containment Cooling Ancillary Water Storage Tank

See Subsection 6.2.2.2.3 for a description of this component.

Fire Pumps

Two 100-percent capacity fire pumps are provided in accordance with NFPA 20 (Reference 2). Each pump is rated for 2000 gpm. The lead pump is electric motor-driven and the second pump is diesel engine-driven. The pumps and their controllers are UL-listed. Fire pump status alarms are provided in the main control room.

The motor-driven fire pump is supplied with power from the turbine building 480 Vac non-Class 1E switchgear. The fuel tank for the diesel-driven pump holds enough fuel to operate the pump for at least 8 hours.

Valves

Valves used in the fire protection system are of an approved type for fire protection service. See the Fire Protection Handbook (Reference 4) for typical descriptions of these valves.

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Fire Detectors

The types of fire detectors used in specific applications are identified in the fire protection analysis. See Reference 4 for descriptions of these fire detectors and their principles of operation.

9.5.1.3 Safety Evaluation (Fire Protection Analysis)

The fire protection analysis evaluates the potential for occurrence of fires within the plant and describes how fires are detected and suppressed. It also confirms that the plant can be safely shut down following a postulated fire. The fire protection analysis is in Appendix 9A.

The fire protection analysis includes a set of fire area drawings and a discussion of the analysis methodology. It also provides the following information for each fire area in the plant:

- A description of the fire area and its fire barriers, its associated fire zones, as well as fire detection and suppression capabilities
- Identification of the type, quantity, and location of in-situ and anticipated transient combustible materials, and combustible loading
- A listing of safety-related mechanical and electrical equipment
- Fire severity category and equivalent duration
- An evaluation of fire protection system adequacy and the consequences of a fire, including a discussion of the control and removal of smoke and hot gases, and drainage system adequacy.

For fire areas containing safety-related structures, systems, and components the following information is also provided:

- An evaluation of fire protection system integrity. This includes a determination of whether the credible failure of a fire protection system component could cause inadvertent operation of an automatic fire suppression system in the fire area, and the resulting consequences. Also included is verification that no potential single impairment of the fire protection system could incapacitate both the automatic suppression system and the backup manual suppression system (generally a hose station), for fire areas where both types of suppression systems are provided.
- A safe shutdown evaluation confirming the capability to safely shut down the reactor and maintain it in a safe shutdown condition following a fire.

The safe shutdown evaluation is based upon all components in a single fire area outside containment or any fire zone inside containment being disabled by the fire. Success is based upon the plant being able to achieve safe shutdown as discussed in Section 7.4. Safe shutdown is a safe, stable condition that can be maintained indefinitely with the reactor subcritical and reactor coolant pressure at a small fraction of its design pressure. As described in Section 7.4.1.1, safety-related systems achieve this condition automatically using reliable, passive processes. The passive residual heat removal heat exchanger transfers heat to the in-containment refueling water storage tank. Steam from this tank enters the containment which is cooled by the passive containment cooling system. These systems reduce the reactor temperature and pressure to less than 420°F and 600 psia in 36 hours. See Appendix 19E for additional details about the shutdown evaluation. This is a safe and acceptable end state which is used to show compliance with BTP 9.5-1. The safe shutdown fire evaluation in Appendix 9A shows that there is sufficient safety-related equipment available after a fire which

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destroys a single fire area outside containment or any fire zone inside containment, to bring the plant to this safe shutdown condition.

It should be noted that following most fires, that nonsafety-related systems are expected to be available to bring the plant to a cold shutdown for repairs. These systems are defense in depth systems with redundant active components. These systems are expected to be available because of the use of redundant equipment and fire protection features, including separation or automatic fire suppression.

Table 9.5.1-4 lists the system capabilities that are expected to be available following a fire to bring the plant to a cold shutdown. This list does not contain the nonsafety-related support systems that are not necessary to operate following a fire. For example, chilled water cooling and non-1E instrumentation are not required following a fire. Heating and ventilation are not required except for the annex/auxiliary non-radioactive ventilation system AHUs used to ventilate the non-1E switchgear rooms. The following safety-related capabilities are used together with these nonsafety-related capabilities to achieve cold shutdown:

- Insertion of control rods to provide reactor shutdown,
- Instrumentation to monitor reactor coolant system conditions,
- Operation of one core makeup tank in a natural circulation mode to provide reactor coolant makeup and boration in case the chemical and volume control system makeup is unavailable due to a fire,
- Manual partial opening (and closing) of one first stage automatic depressurization valve to provide a controlled, limited depressurization of the reactor coolant system to allow initiation of the normal residual heat removal system in case the chemical and volume control system auxiliary spray is unavailable due to a fire.

The use of these safety-related capabilities does not result in significant plant transients. The reactor coolant system pressure boundary is maintained and containment pressure and temperature conditions are not affected by the use of these safety-related capabilities.

If a less likely, more severe fire occurs, these systems are expected to be recovered after reasonable actions are taken to utilize temporary connections or to perform repairs (see Subsections 9.2.2.4.5.5 and 9.5.1.1.1). Recovery of these systems allows the plant to be brought to a cold shutdown for plant repairs. No credit is taken in the Appendix 9A fire evaluation for nonsafety-related systems. As a result, fire separation is not required for these systems.

9.5.1.4 Testing and Inspection

The fire pumps are initially tested by the manufacturer in accordance with NFPA 20 (Reference 2) to verify pressure integrity and performance.

Preoperational testing is in accordance with the Initial Test Program (Chapter 14).

9.5.1.5 Instrumentation Applications

Pressure sensors start the fire pumps on decreasing fire main water pressure. Pressure indicators confirm adequate pressures for automatic and manual suppression systems. Valve position sensors are used to monitor the positions of water supply valves.

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Temperature instrumentation is used to monitor fire water storage tank temperature. Level instrumentation is used to monitor levels in the fire water storage tanks and the diesel-driven fire pump fuel storage tank.

9.5.1.6 Personnel Qualification and Training

Preparation and review of the fire protection analysis, and design and selection of fire protection equipment, is performed by fire protection and nuclear safety systems engineers.

The qualification requirements for individuals responsible for development of the fire protection program, training of firefighting personnel, as well as associated administrative procedures are discussed in Subsection 9.5.1.8.1.

Subsections 9.5.1.8.2 and 9.5.1.8.7 summarize the qualification and training programs that are established and implemented for the Fire Protection Program.

9.5.1.7 Quality Assurance

Quality assurance controls are applied to the activities involved in the design, procurement, installation, testing, and maintenance of fire protection systems for safety-related areas, in accordance with the programs outlined in Chapter 17.

9.5.1.8 Fire Protection Program

The fire protection program is established such that a fire does not prevent safe shutdown of the plant and does not endanger the health and safety of the public. Fire protection at the plant uses a defense-in-depth concept that includes fire prevention, detection, control and extinguishing systems and equipment, administrative controls and procedures, and trained personnel. These defense-in-depth principles are achieved by meeting the following objectives:

- Prevent fires from starting.
- Detect rapidly, control, and extinguish promptly those fires that do occur.
- Provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities does not prevent the safe shutdown of the plant.
- Minimize the potential for radiological releases.

9.5.1.8.1 Fire Protection Program Implementation

As indicated in Table 13.4-201, the required elements of the fire protection program are fully operational prior to receipt of new fuel for buildings storing new fuel and adjacent fire areas that could affect the fuel storage area in that reactor unit. Other required elements of the fire protection program described in this section are fully operational prior to initial fuel loading in that reactor unit.

Elements of the fire protection program are reviewed on a frequency established by procedures and updated as necessary.

9.5.1.8.1.1 Fire Protection Program Criteria

The fire protection program is based on the criteria of several industry and regulatory documents referenced in Subsection 9.5.5, and also based on the guidance provided in Regulatory Guide 1.189.

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Table 9.5.1-1 provides a cross-reference to information addressing compliance with BTP CMEB 9.5-1. Exceptions to the design portions of the National Fire Protection Association (NFPA) Standards Table 9.5.1-3 listed in Subsection 9.5.5 are identified in Table 9.5.1-3.

The NFPA code year identified in Subsection 9.5.5 is the code of record for the AP1000 plant. In accordance with the guidance provided in Regulatory Position C.1.8.6 of Regulatory Guide 1.189, the equivalency concept in meeting the provisions of the NFPA codes and standards may be used. Newer editions of the listed code of record and alternative codes or standards by the same or alternate agencies (e.g., NFPA, ASTM, UL) are evaluated for equivalency by a fire protection engineer prior to use. Per the guidance provided in Sections 1.8.1.3 and 1.8.6 of Regulatory Guide 1.189, equivalencies and clarifications to the NFPA Standards listed in Subsection 9.5.5 are documented and available for inspection.

9.5.1.8.1.2 Organization and Responsibilities

The site executive in charge of the fire protection program, through the engineer in charge of fire protection, is responsible for the following:

a. Programs and periodic inspections are implemented to:

    1. Minimize the amount of combustibles in safety-related areas.

    2. Determine the effectiveness of housekeeping practices.

    3. Provide for availability and acceptability of the following:

        (i) Fire protection system and components.

        (ii) Manual firefighting equipment.

        (iii) Emergency breathing apparatus.

        (iv) Emergency lighting.

        (v) Portable communication equipment.

        (vi) Fire barriers including fire rated walls, floors and ceilings, fire rated doors, dampers, etc., fire stops and wraps, and fire retardant coating. Procedures address the administrative controls in place, including fire watches, when a fire area is breached for maintenance.

    4. Confirm prompt and effective corrective actions are taken to correct conditions adverse to fire protection and preclude their recurrence.

b. Conducting periodic maintenance and testing of fire protection systems, components, and manual firefighting equipment, evaluating test results, and determining the acceptability of systems under test in accordance with established plant procedures.

c. Designing and selecting equipment related to fire protection.

d. Reviewing and evaluating proposed work activities to identify potential transient fire loads.

e. Managing the plant fire brigade, including:

    1. Developing, implementing, and administering the fire brigade training program.

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2. Scheduling and conducting fire brigade drills.

3. Critiquing fire drills to determine if training objectives are met.

4. Performing a periodic review of the fire brigade roster and initiating changes as needed.

5. Maintaining the fire training program records for members of the fire brigade and other personnel.

6. Maintaining a sufficient number of qualified fire brigade personnel to respond to fire emergencies for each shift.

f. Developing and conducting the fire extinguisher training program.

g. Implementing a program for indoctrination of personnel gaining unescorted access to the protected area in appropriate procedures which implement the fire protection program, such as fire prevention and fire reporting procedures, plant emergency alarms, including evacuation.

h. Implementing a program for instruction of personnel on the proper handling of accidental events such as leaks or spills of flammable materials.

i. Preparing procedures to meet possible fire situations in the plant and for ensuring assistance is available for fighting fires in radiological areas.

j. Implementing a program that uses a permit system that controls and documents inoperability of fire protection systems and equipment. This program initiates proper notifications and compensatory actions, such as fire watches, when inoperability of any fire protection system or component is identified.

k. Developing and implementing preventive maintenance, corrective maintenance, and surveillance test fire protection procedures.

l. Confirming that plant modifications, new procedures and revisions to procedures associated with fire protection equipment and systems that have significant impact on the fire protection program, are reviewed by an individual who possesses the qualifications of a fire protection engineer.

m. Continuing evaluation of fire hazards during construction or modification of other units on the site. Special considerations, such as fire barriers, fire protection capability, and administrative controls are provided as necessary to protect the operating unit(s) from construction or modification activities.

n. Establishing a fire prevention surveillance plan and training plant personnel on that plan.

o. Developing prefire plans and making them available to the fire brigade and control room.

9.5.1.8.2 Fire Brigade

9.5.1.8.2.1 General

To qualify as a member of the fire brigade, an individual must meet the following criteria:

a. Has attended the required training sessions for the position occupied on the fire brigade.

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b. Has passed an annual physical exam including demonstrating the ability for performing strenuous activity and the use of respiratory protection.

9.5.1.8.2.2 Fire Brigade Training

A training program is established so that the capability to fight fires is developed and documented. The program consists of classroom instruction supplemented with periodic classroom retraining, practice in firefighting, and fire drills. Classroom instruction and training is conducted by qualified individuals knowledgeable in fighting the types of fires that could occur within the plant and its environs and using onsite firefighting equipment. Individual records of training provided to each fire brigade member, including drill critiques, are maintained as part of the permanent plant files for at least three years to document that each member receives the required training.

The fire brigade leader and at least two brigade members per shift have sufficient training and knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability. The brigade leader is competent to assess the potential safety consequences of a fire and advise control room personnel. Such competence by the brigade leader may be evidenced by possession of an operator's license or equivalent knowledge of plant systems.

Personnel assigned as fire brigade members receive formal training prior to assuming brigade duties. The course subject matter is selected to satisfy the requirements of Regulatory Guide 1.189. Course material selection also includes guidance from NFPA 600 (Reference 204) and 1500 (Reference 210) as appropriate. Additional training may also include material selected from NFPA 1404 (Reference 208) and 1410 (Reference 209).

The minimum equipment provided for the fire brigade consists of personal protective equipment such as turnout coats, boots, gloves, hard hats, emergency communications equipment, portable lights, portable ventilation equipment, and portable extinguishers. Self-contained breathing apparatus (SCBA) approved by NIOSH, using full face positive pressure masks, and providing an operating life of at least 30 minutes, are provided for selected fire brigade, emergency repair, and control room personnel. At least ten masks, five per brigade locker plus spares, are provided for fire brigade personnel. Extra air bottles, with at least 60 minutes total operating life, are located onsite for each SCBA. An additional onsite 6-hour supply of reserve air is provided to permit quick and complete replenishment of exhausted supply air bottles. Subsection 6.4.2.3 discusses the portable breathing apparatus for control room personnel. At least six SCBAs are provided in the fire brigade locker for the exclusive use of the fire brigade. The fire brigade leader has ready access to keys for any locked fire doors. SCBAs are provided in the control room for use by Operations. Radiation Protection also maintains additional SCBA near the radiologically controlled area entrance for emergency usage.

The on-duty shift manager has responsibility for taking certain actions based on an assessment of the magnitude of the fire emergency. These actions include safely shutting down the plant, making recommendations for implementing the Emergency Plan, notification of emergency personnel, and requesting assistance from off-duty personnel, if necessary. Emergency Plan consideration of fire emergencies includes the guidance of Regulatory Guide 1.101.

9.5.1.8.2.2.1 Classroom Instruction

Fire brigade members receive classroom instruction in fire protection and firefighting techniques prior to qualifying as members of the fire brigade. This instruction includes:

a. Identification of the types of fire hazards along with their location within the plant and its environs.

b. Identification of the types of fires that could occur within the plant and its environs.

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c. Identification of the location of onsite fire fighting equipment and familiarization with the layout of the plant including ingress and egress routes to each area.

d. The proper use of onsite fire fighting equipment and the correct method of fighting various types of fires including at least the following:

   – fires involving radioactive materials

   – fires in energized electrical equipment

   – fires in cables and cable trays

   – fires involving hydrogen

   – fires involving flammable and combustible liquids or hazardous process chemicals

   – fires resulting from construction or modifications (welding)

   – fires involving record files.

e. Review of each individual's responsibilities under the Fire Protection Program.

f. Proper use of communication, lighting, ventilation, and emergency breathing equipment.

g. Fire brigade leader direction and coordination of firefighting activities.

h. Toxic and radiological characteristics of expected combustion products.

i. Proper methods of fighting fires inside buildings and confined spaces.

j. Detailed review of firefighting strategies, procedures and procedure changes.

k. Indoctrination of the plant firefighting plans, identification of each individual's responsibilities, and review of changes in the firefighting plans resulting from fire protection-related plant modifications.

l. Coordination between the fire brigade and offsite fire departments that have agreed to assist during a major fire onsite is provided to establish responsibilities and duties. Educating the offsite organization in operational precautions when fighting fires on nuclear power plant sites, and awareness of special hazards and the need of radiological protection of personnel.

9.5.1.8.2.2.2 Retraining

Classroom refresher training is scheduled on a biennial basis to supplement retention of the initial training. These sessions may be concurrent with the regular planned meetings.

9.5.1.8.2.2.3 Practice

Practice sessions are held for each fire brigade and for each fire brigade member on the proper method of fighting various types of fires which might occur in the plant. These sessions are scheduled on an annual basis and provide brigade members with team experience in actual fire fighting and the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting.

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9.5.1.8.2.2.4 Drills

Fire brigade drills are conducted at least once per calendar quarter for each shift. Each fire brigade member participates in at least two drills annually. Drills are either announced or unannounced. At least one unannounced drill is held annually for each shift fire brigade. At least one drill is performed annually on a “back shift” for each shift’s fire brigade. The drills provide for offsite fire department participation at least annually. Triennially, a randomly selected, unannounced drill shall be conducted and critiqued by qualified individuals independent of the plant staff. Training objectives are established prior to each drill and reviewed by plant management. Drills are critiqued on the following points:

a. Assessment of fire alarm effectiveness.

b. Assessment of time required to notify and assemble the fire brigade.

c. Assessment of the selection, placement, and use of equipment.

d. Assessment of the fire brigade leader’s effectiveness in directing the firefighting effort.

e. Assessment of each fire brigade member’s knowledge of firefighting strategy, procedures, and simulated use of equipment.

f. Assessment of the fire brigade’s performance as a team.

Performance deficiencies identified, based on these assessments, are used as the basis for additional training and repeat drills. Unsatisfactory drill performance is followed by a repeat drill within 30 days.

9.5.1.8.2.2.5 Meetings

Regular planned meetings are held at least quarterly for the fire brigade members to review changes in the Fire Protection Program and other subjects as necessary.

9.5.1.8.3 Administrative Controls

Administrative controls for the Fire Protection Program are implemented through plant administrative procedures. Applicable industry publications are used as guidance in developing those procedures.

Administrative controls include procedures to:

a. Control actions to be taken by an individual discovering a fire, such as notification of the control room, attempting to extinguish the fire, and actuation of local fire suppression systems.

b. Control actions to be taken by the control room operator, such as sounding fire alarms, and notifying the shift manager of the type, size, and location of the fire.

c. Control actions to be taken by the fire brigade after notification of a fire, including location to assemble, directions given by the fire brigade leader, the responsibilities of brigade members, such as selection of firefighting and protective equipment, and use of preplanned strategies for fighting fires in specific areas.

d. Control actions to be taken by the security force upon notification of a fire.

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e. Define the strategies established for fighting fires in safety-related areas and areas presenting a hazard to safety-related equipment, including the designation of the:

   1. Fire hazards in each plant area/zone covered by a firefighting procedure (prefire plan). Prefire plans use the guidance of NFPA 1620 (Reference 205).

   2. Fire extinguishers best suited for controlling fires with the combustible loadings of each zone and the nearest location of these extinguishers.

   3. Most favorable direction from which to attack a fire in each area in view of the ventilation direction, access hallways, stairs, and doors that are most likely to be free of fire, and the best station or elevation for fighting the fire. Access and egress routes that involve locked doors are specifically identified in the procedure with the appropriate precautions and methods for access specified.

   4. Plant systems that should be managed to reduce the damage potential during a local fire and the location of local and remote controls for such management (e.g., any hydraulic or electrical system in the zone covered by the specific firefighting procedure that could increase the hazards in the area because of overpressurization or electrical hazards).

   5. Vital heat-sensitive system components that need to be kept cool while fighting a local fire. Particularly hazardous combustibles that need cooling are designated.

   6. Potential radiological and toxic hazards in fire zones.

   7. Ventilation system operation that provides desired plant air distribution when the ventilation flow is modified for fire containment or smoke clearing operations.

   8. Operations requiring control room and shift manager coordination or authorization.

   9. Instructions for plant operators and other plant personnel during a fire.

f. Organize the fire brigade and assign special duties according to job title so that the firefighting functions are covered for each shift by personnel trained and qualified to perform these functions.These duties include command control of the brigade, transporting fire suppression, and support equipment to the fire scenes, applying the extinguishing agent to the fire, communication with the control room, and coordination with offsite fire departments.

9.5.1.8.4 Control of Combustible Materials, Hazardous Materials, and Ignition Sources

The control of combustible materials, hazardous materials, and ignition sources are defined by administrative procedures. These procedures impose the following controls:

a. Prohibit the storage of hazardous chemicals (including unused ion exchange resins) in areas that contain or expose safety-related equipment.

b. Govern the handling of and limit transient fire loads such as flammable liquids, wood, and plastic materials in buildings containing safety-related systems or equipment.

c. Assign responsibility to the appropriate supervisor for reviewing work activities to identify transient fire loads.

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d. Govern the use of ignition sources by use of a flame permit system to control welding, flame cutting, grinding, brazing and soldering operations, and temporary electrical power cables. A separate permit is issued for each area where such work is done. If work continues over more than one shift, the permit is valid for not more than 24 hours when the plant is operating or for the duration of a particular job during plant shutdown. NFPA 51B (Reference 202) and 241 (Reference 203) are used as guidance.

e. Minimize waste, debris, scrap, and oil spills or other combustibles resulting from a work activity in the safety-related area while work is in progress, and remove the same upon completion of the activity or at the end of each work shift.

f. Govern periodic inspections for accumulation of combustibles for continued compliance with these administrative controls.

g. Prohibit the storage of acetylene-oxygen and other compressed gasses in areas that contain or expose safety-related equipment or the fire protection system that serves those areas. A permit system is required to control the use of this equipment in safety-related areas of the plant.

h. Govern the use and storage of hazardous chemicals in areas that contain or expose safety-related equipment.

i. Control the use of specific combustibles in safety-related areas. Wood used in safety-related areas during maintenance, modification, or refueling operation (such as lay-down blocks or scaffolding) is treated with a flame retardant in accordance with NFPA 703 (Reference 207). Equipment or supplies shipped in untreated combustible packing containers are unpacked in safety-related areas if required for valid operating reasons. However, combustible materials are removed from the area immediately following unpacking. Transient combustible material (including packing material) is attended, stored in approved containers, or a fire watch is established. Textiles or films used are certified to conform to the fire test methods described in NFPA 701 (Reference 206).

j. Govern the control of electrical heaters in areas that contain or expose safety-related equipment.

9.5.1.8.5 Control of Radioactive Materials

The plant is designed with provisions for sampling of liquids resulting from fire emergencies that may contain radioactivity and may be released to the environment. Plant operating procedures require such liquids to be collected, sampled, and analyzed prior to discharge. Liquid discharges are required to be below activity limits prior to discharge.

9.5.1.8.6 Testing and Inspection

Testing and inspection requirements, including frequency, are imposed through administrative procedures. Maintenance or modifications to the fire protection system are subject to inspection for conformation to design requirements. Procedures governing the inspection, testing, and maintenance of fire protection alarm and detection systems, and water-based suppression and supply systems, use the guidance of NFPA 72 (Reference 9.5.5.2) and NFPA 25 (Reference 212). Frequency of fire protection equipment maintenance is based on industry guidance and adjusted based on equipment performance. Fire protection equipment is periodically reviewed by site personnel to ensure continued acceptable performance. Installation of portions of the system where performance cannot be verified through preoperational tests, such as penetration seals, fire-retardant coatings, cable routing, and fire barriers are inspected. Inspections are performed by individuals

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knowledgeable of fire protection design and installation requirements. Open flame or combustion-generated smoke is not used for leak testing or similar procedures such as air flow determination. Inspection and testing procedures address the identification of items to be tested or inspected, responsible organizations for the activity, acceptance criteria, documentation requirements and sign-off requirements.

Fire protection materials subject to degradation (such as fire stops, seals, and fire retardant coatings are visually inspected periodically for degradation or damage. Fire hoses are hydrostatically tested in accordance with NFPA 1962 (Reference 201). Hoses stored in outside hose stations are tested annually and interior standpipe hoses are tested every three years.

The fire protection system is periodically tested in accordance with plant procedures. Testing includes periodic operational tests and visual verification of damper and valve positions. Fire doors and their closing and latching mechanisms are also included in these procedures.

9.5.1.8.7 Personnel Qualification and Training

The engineer in charge of fire protection is responsible for the formulation and implementation of the fire protection program and meets the qualification requirements.

Qualification and training of other plant personnel involved in the fire protection program is governed by plant qualification procedures and is conducted by personnel qualified by training and experience in these areas. These classifications include training personnel, maintenance personnel assigned to work on the fire protection system, and operations personnel assigned to system operation and testing.

Qualification for fire watches for hot work also include hands on training with fire extinguishers. A fire watch used as a compensatory measure for an impairment is briefed on the necessary job functions prior to being assigned.

9.5.1.8.8 Fire Doors

Fire doors separating safety-related areas are self-closing or provided with closing mechanisms and are inspected semiannually to verify that the automatic hold open, release and closing mechanisms and latches are operable. Watertight and missile resistant doors are not provided with automatic closing mechanisms. Fire doors with automatic hold open and release mechanisms are inspected daily to verify that the doorways are free of obstructions.

Fire doors separating safety-related areas are normally closed and latched. Fire doors that are locked closed are inspected weekly to verify position. Fire doors that are closed and latched are inspected daily to ensure that they are in the closed position. Fire doors that are closed and electrically supervised at a continuously manned location are not inspected.

9.5.1.8.9 Emergency Planning

Emergency planning is described in Section 13.3.

9.5.1.9 Combined License Information

9.5.1.9.1 The qualification requirements for individuals responsible for development of the fire protection program, training of firefighting personnel, administrative procedures and controls governing the fire protection program during plant operation, and fire protection system maintenance are addressed as follows:

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Qualification requirements for individuals responsible for development of the Fire Protection Program are discussed in Subsections 9.5.1.6 and 9.5.1.8.7.

Training of firefighting personnel is discussed in Subsections 9.5.1.8, 9.5.1.8.2, and 9.5.1.8.7.

Administrative procedures and controls governing the Fire Protection Program during plant operation are discussed in Subsections 9.5.1.8.1.2, 9.5.1.8.3, 9.5.1.8.4, 9.5.1.8.5, and 9.5.1.8.6.

Fire protection system maintenance is discussed in Subsection 9.5.1.8.6.

9.5.1.9.2 The Combined License applicant will provide site-specific fire protection analysis information for the yard area, the administration building, and for other outlying buildings consistent with Appendix 9A.

9.5.1.9.3 The BTP CMEB 9.5-1 issues identified in Table 9.5.1-1 by the acronym "WA" are addressed in Subsections 9.5.1.8.1.1, 9.5.1.8.8, and 9.5.1.8.9 and in Table 9.5.1-1.

9.5.1.9.4 The plant-specific NFPA exceptions are addressed in Subsection 9.5.1.8.1.1.

9.5.1.9.5 The operator actions which minimize the probability of the potential for spurious ADS actuation as a result of a fire are addressed in APP-GW-GLR-027 (Reference 22), and the applicable changes are incorporated.

9.5.1.9.6 The Combined License holder will address the process for identifying deviations between the as-built installation of fire barriers and their tested configurations.

9.5.1.9.7 The procedures to minimize risk when fire areas are breached during maintenance are addressed in Subsection 9.5.1.8.1.2.

9.5.1.9.8 The fire resistance test data for the composite material selected for stairwell fire barriers is addressed in APP-GW-GLR-019 (Reference 23), and the applicable changes are incorporated.

9.5.2 Communication System

The communication system (EFS) provides effective intraplant communications and effective plant-to-offsite communications during normal, maintenance, transient, fire, and accident conditions, including loss of offsite power. The communication system consists of the following subsystems:

- Wireless telephone system
- Telephone/page system
- Private branch exchange (PBX) system
- Sound-powered system
- Emergency offsite communications
- Security communication system.
- Plant Radio System (also known as "Land Mobile Radio Service" or "LMRS")

EFS allows each guard, watchman, or armed response individual on duty to maintain continuous communication with an individual in each manned alarm station and with other agencies both onsite and offsite, as required by 10 CFR 73.55(j). This is accomplished by both the PBX system and the wireless communication system. Each system can provide these communication functions.

Communication equipment used with respiratory protection devices will be designed and selected in accordance with EPRI NP-6559 (Reference 8).

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9.5.2.1 Design Basis

EFS serves no safety-related function and therefore has no nuclear safety design basis.

EFS subsystems are independent of one another; therefore, a failure in one subsystem does not degrade performance of the other subsystems.

EFS is in accordance with applicable codes and standards minimizing electromagnetic interference and its potential effects to equipment. "Low-powered" type equipment is used, where possible, which has been demonstrated to have a limited potential for causing interference with electronic equipment (Reference 8) (EPRI NP-6559, Section 5). Communication equipment is shielded, as necessary, from the detrimental effects of electromagnetic interference.

9.5.2.2 System Description

9.5.2.2.1 Wireless Telephone System

The wireless telephone system consists of wireless portable handsets, hands-free type portable headsets, a comprehensive antenna system, and necessary electronics equipment. The wireless telephone system is the primary means of communication for plant operations and maintenance personnel. The telephone-page, PBX telephone, and sound-powered communication systems are for general plant communications and serve as a backup to the wireless system.

The wireless telephone system has the ability to dial fixed PBX telephone stations and vice versa. The wireless system has the capability to access the telephone-page system and the capability to access offsite emergency communication links.

Normal 120-V ac power supplies the wireless telephone switch. Upon loss of the normal power, the switch is powered from the non-Class 1E dc and uninterruptible power supply system and supplies power for system operation for 120 minutes.

9.5.2.2.2 Telephone/Page System

The telephone/page system consists of handsets, amplifiers, loudspeakers, and associated electronic equipment. The system supports paging and party lines without crosstalk or interference. The page party system includes the capability to establish multiple zones. Zone merging control capability is provided at the main control room, the security central alarm station, and the security secondary alarm station.

The telephone/page system has the capability to provide audible tone alarms (siren generator) and pre-recorded messages using the telephone/page system amplifiers and speakers. Alarm initiation capability is provided in the main control room. Paging zones are automatically merged during an alarm condition.

The telephone/page party system is divided into in-plant and outside area physical zones. Each zone can be disconnected from the rest of the system without interrupting the functions of the other unaffected zones or communication between the unaffected zones.

Power to the telephone/page system is provided from the non-Class 1E dc and uninterruptible power supply system sized to supply power for 120 minutes after a loss of ac power.

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9.5.2.2.3 Private Branch Exchange System

The private branch exchange (PBX) system provides communication between the system stations, with capability for transferring calls and providing conference calls.

PBX telephones are programmable. Buttons on the phone can be dedicated and color coded to specific telephone numbers.

The PBX system also interfaces with the following communication systems:

- The wireless telephone system
- Redundant connectivity to commercial telephone services to provide communication capability with the load dispatcher to support and coordinate the system grid is as described in Subsection 9.5.2.5
- Access to the telephone page system
- Direct extensions from the PBX locations exterior to the plant as dictated in Subsection 9.5.2.5

The separate, redundant connections between the PBX and commercial telephone service provides communications between the main control room and the headquarters or other facilities in case of a single fault as required in Subsection 9.5.2.5.

Commercial telephone lines are provided by the local area telephone company. The local telephone company lines terminated at the switch are programmed to reserve part of the lines for outgoing calls only. Others are programmed for incoming only so that some lines are available for calling onto and off of the site. The number of lines is defined in Subsection 9.5.2.5.

Power to the PBX is provided from the non-Class 1E dc and uninterruptible power supply system sized to supply power for 120 minutes after a loss of ac power.

9.5.2.2.4 Sound-Powered System

Two unitized systems are provided as follows:

- A loop sound-powered system for refueling
- A multiloop system throughout the plant for startup and maintenance testing

The sound-powered system does not require external power supply for operation.

9.5.2.3 System Operation Communication Stations

Table 9.5.2-1 lists the communication stations provided for operator use during transients.

The main control room and remote shutdown room are designed and instrumented to bring the plant to a safe shutdown condition without relying on communications equipment. Various communication stations are provided throughout the plant.

9.5.2.4 Inspection and Testing Requirements

Communication systems of the types described above are conventional and have a history of reliable operation. Most of these systems are in routine use, and this routine use will demonstrate their

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availability. Those systems not frequently used, but required during emergency situations, are tested at periodic intervals to demonstrate operability when required.

9.5.2.5 Combined License Information

9.5.2.5.1 Offsite Interfaces

Combined License applicants referencing the AP1000 certified design will address interfaces to required offsite locations; this will include addressing the recommendations of BL-80-15 (Reference 21) regarding loss of the emergency notification system due to a loss of offsite power.

9.5.2.5.2 Emergency Offsite Communications

The emergency offsite communication system, including the crisis management radio system, will be addressed by the Combined License applicant.

9.5.2.5.3 Security Communications

Specific details for the security communication system are as discussed in separate security documents referred to in Section 13.6.

9.5.2.5.4 Plant Radio

The plant radio system is used primarily by Security and site fire brigade. The security functions are addressed in the Physical Security Plan. Other site departments also use the radio system periodically. Separate radio talk groups are assigned to each department.

9.5.3 Plant Lighting System

The plant lighting system includes normal, emergency, panel, and security lighting. The normal lighting provides normal illumination during plant operating, maintenance, and test conditions. The emergency lighting provides illumination in areas where emergency operations are performed upon loss of normal lighting. The panel lighting in the control room is designed to provide the minimum illumination required at the safety panels. The security lighting system is described in separate security documents referred to in Section 13.6.

9.5.3.1 Design Basis

The plant lighting system serves no safety-related function and therefore has no nuclear safety design basis. Failure of the plant lighting system or its components will not affect the ability of safety-related systems to perform their intended function.

- The normal and emergency lighting in the main control room and in the remote shutdown room is non-Class 1E.
- The emergency lighting provides illumination in the main control room and the remote shutdown room for 72 hours upon loss of normal lighting. In other plant areas, the emergency lighting provides illumination for 8 hours.
- Lighting for the safety panels in the control room is provided by the panel lighting system.
- During the 72 hour period following a loss of all ac power sources, lighting in the main control room can be provided as described in Subsection 9.5.3.2.2.

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- The plant lighting system provides illumination levels for normal and emergency lighting as recommended in Illuminating Engineering Society Lighting Handbook (Reference 5).

9.5.3.2 System Description

9.5.3.2.1 Normal Lighting

Power to the normal lighting system is supplied from the non-Class 1E ac power distribution system at the following voltage levels:

- 480/277 V, three-phase, four-wire, grounded neutral system lighting panels are fed from the 480 V motor control centers; this source is for the lighting fixtures rated at 480/277 V and for the welding receptacles.

- 208/120 V, three-phase, four-wire, grounded neutral system distribution panels are fed from the 480 V motor control centers through dry-type 480-208/120 V transformers; this source is for lighting and utility receptacles.

- 208/120 V, three-phase, four-wire, grounded neutral regulated power fed from the 480 V motor control centers through the Class 1E 480 - 208/120 V voltage regulating transformers (divisions B and C); this source is for the normal and emergency lighting in the main control room and remote shutdown room and is isolated through two series fuses for isolation. The normal lighting in these plant areas is non-Class 1E.

The normal lighting system has the following features:

- The normal lighting system is powered from the diesel-backed buses and the lighting load is distributed between the two onsite standby diesel generator buses.

- The motor control centers powering the normal lighting system are energized from the 480 V load centers connected in a tie-breaker configuration.

- Lighting distribution panel branch circuit breakers are controlled by a lighting control system. Approximately 75 percent of the normal lighting is tripped off automatically upon loss of normal ac power and offsite power (except in the main control room and in the remote shutdown room) to limit the load on the onsite standby diesel generators. The lighting control system allows the operator to energize or de-energize lighting in selected areas based on the actual need and available power from the onsite standby diesel generators.

- The lighting circuits are staggered as much as practical. The staggered circuits receive power from separate buses to prevent complete loss of light in the event of a bus or a circuit failure.

- The lighting fixtures located in the vicinity of safety-related equipment are supported so that they do not adversely impact this equipment when subjected to the seismic loading of a safe shutdown earthquake.

- Mercury vapor lamps and mercury switches are not used in fuel handling areas.

- Unencapsulated, unshielded high-intensity discharge (HID) and fluorescent lamps are not used in the containment and fuel handling areas due to mercury content and the risk of being released to the area environment. Incandescent lighting or other lighting not containing restricted materials may be used in these general areas. High Pressure Sodium (HPS) underwater lighting, with appropriate barriers and protection to prevent contact with the water,

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or long life lighting not containing restricted materials, are used for underwater lighting in the fuel handling areas outside containment as they provide longer lamp life, longer service intervals and less personnel exposure.

- The control room and remote shutdown room lighting uses semi-indirect, low-glare lighting fixtures and adjustable dimming features. The normal control room lighting provides at least 50 foot candles of illumination at the safety panel and at the workstations when the dimming features are adjusted for maximum illumination. The normal remote shutdown room lighting provides at least 50 foot candles of illumination at the remote shutdown workstation when the dimming features are adjusted for maximum illumination.

9.5.3.2.2 Emergency Lighting

Emergency lighting is designed to provide the required illumination levels in the areas as described below:

- The main control room and remote shutdown room each has emergency lighting consisting of 120 V ac lighting fixtures which are continuously energized. The fixtures are powered from the Class 1E 250 V dc switchboards through the Class 1E 208Y/120 V ac inverters and are isolated through two series fuses that are coordinated for isolation. Three hour fire barrier separation is provided between redundant emergency lighting power supplies and cables outside the main control room and the remote shutdown area. The control room lighting complies with the human factor requirements by utilizing semi-indirect, low-glare lighting fixtures and adjustable dimming features. The control room emergency lighting is integrated with normal lighting that consists of identical lighting fixtures and dimming features. The emergency lighting system is designed so that, to the extent practical, alternate emergency lighting fixtures are fed from separate divisions of the Class 1E dc and uninterruptible power supply system. Both normal and emergency lighting fixtures, controllers, dimmers, and associated cables used in the main control room and remote shutdown room are nonClass 1E. The ceiling grid network, raceways and fixtures utilize seismic supports. A single fault cannot interrupt all of the lighting in the main control room and at the remote shutdown workstation simultaneously. The emergency lighting provides at least 10 foot candles of illumination at the safety panel, at the workstations in the control room, and at the remote shutdown workstation when the dimming features are adjusted for maximum illumination.

- Following the 72 hour period after a loss of all ac power sources, the lighting in the main control room is powered from two ancillary ac generators as described in Subsection 8.3.1.1.1.

- Emergency lighting in areas outside the main control room and remote shutdown room is accomplished by 8-hour, self-contained, battery pack lighting units. These units are nonClass 1E and provide illumination for safe ingress and egress of personnel following a loss of normal lighting and for those areas which could be involved in power recovery (for example, onsite standby diesel-generators and their controls). In addition, these units are provided in areas where manual actions are required for operation of equipment needed during a fire. These units are normally powered from the non-Class 1E 480/277 V ac motor control centers and they automatically switch to their internal dc source once normal ac power is lost.

9.5.3.2.3 Panel Lighting

Panel lighting is designed to provide lighting in the control room at the safety panels as described below:

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- Panel lighting consists of lighting fixtures located on or near safety panels in the control room. The panel lights are continuously energized. The fixtures are powered from the Divisions B and C Class 1E inverters through Class 1E distribution panels.

- The circuits are treated as Class 1E. The panel lighting circuits up to the lighting fixture are classified as associated and are routed in a Seismic Category I raceway system.

- The bulbs are not seismically qualified.

9.5.3.3 Safety Evaluation

The plant lighting system has no safety-related functions and therefore requires no nuclear safety evaluation. Failure of the plant lighting system or its components will not affect the ability of safety-related systems to perform their intended function.

The areas that require lighting for safe shutdown are the main control room and the remote shutdown room when the main control room is not accessible.

- Lighting fixtures in the main control room and remote shutdown room are seismic Category II.

- Emergency and panel lighting circuits up to the lighting fixture are routed in a Seismic Category I raceway system.

- Panel Lighting circuits up to the lighting fixture are treated as Class 1E and Classified as associated. This is acceptable to the Class 1E power supply because of the over current protective device coordination.

- Bulbs are not seismically qualified. However, the bulbs can only fail open and therefore do not represent a hazard to the Class 1E power sources.

- Power to normal and emergency lighting in the main control room and in the remote shutdown room is supplied from the redundant divisions of Class 1E dc and UPS system through two series fuses for isolation. The fuses protect the batteries from failures of the non1E lighting circuits. The Class 1E batteries provided in the Class 1E dc and UPS system are capable of powering the emergency lighting in these rooms for 72 hours when the normal ac sources are not available. Operation beyond 72 hours is described in Subsection 8.3.1.1.1.

9.5.3.4 Test and Inspections

The ac lighting circuits are normally energized and require no periodic testing. The 8-hour battery pack lighting is inspected and tested periodically.

9.5.3.5 Combined License Information for Plant Lighting

This section contained no requirement for additional information.

9.5.4 Standby Diesel Fuel Oil System

This subsection describes the features of the standby diesel fuel oil system. The standby diesel generators are supplied by a combined storage system of fuel oil storage tanks. Two above-ground fuel oil storage tanks for the combined system service are provided. These tanks store diesel grade fuel. The standby diesel generators are described in Subsection 8.3.1.1.2.

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9.5.4.1 Design Basis

9.5.4.1.1 Safety Design Basis

The standby diesel fuel oil system serves no safety-related function and therefore has no nuclear safety design basis.

9.5.4.1.2 Power Generation Design Basis

The standby diesel fuel oil system serves no power generation function. Its function is to store and transfer fuel oil for the onsite standby diesel generators. The system is designed to meet the following requirements:

- Provide a supply of fuel sufficient to operate each diesel generator to power automatically sequenced loads with 10% margin for load growth for 7 days
- Provide a 4-day fuel supply for the two ancillary diesel generators

9.5.4.1.3 Codes and Standards

The codes and standards that are applicable to the components of the Standby Diesel Fuel Oil System that support the standby diesel generators are listed in Section 3.2. The portions of the Standby Diesel Fuel Oil System that support the standby diesel generators follow the guidance for distillate fuel oil supply contained in Chapter 13 of the DEMA Standard Practices (Reference 19).

9.5.4.2 System Description Storage and Transfer

9.5.4.2.1 General Description

The standby diesel fuel oil system is shown in Figure 9.5.4-1. The system consists of two fuel oil storage tanks, a diesel generator fuel oil transfer system, and an ancillary diesel generator fuel oil supply system.

Two fuel oil storage tanks are provided, one for each of the standby diesel generators.

The plant finished grade elevation will be higher than the probable maximum flood level (refer to Subsection 3.4.1.1, Protection From External Flooding). Therefore the system will be safe from flooding.

The diesel generator fuel oil transfer system consists of two independent fuel storage, transfer and recirculation flow paths; that is, one path per diesel generator. Each path consists of a fuel oil storage tank, one fuel transfer pump, diesel fuel oil supply and fuel return piping, a day tank, and the associated specialties valves, fittings, and instrumentation. The supply lines from the transfer pumps to the daytanks include fuel oil heaters, filters and moisture separators. The system is protected from the effects of low temperatures by the inline electric oil heater in the transfer line.

The ancillary diesel generator fuel oil supply portion of the system consists of a single 100 percent capacity tank serving both ancillary diesel generators. The tank is located inside the annex building and is served by the annex building heating and ventilation system. The tank is insulated and provided with heaters to maintain the fuel oil above the oil cloud point. Fuel oil lines from the tank to the diesels are insulated. Fuel is transferred from the ancillary diesel generator fuel tank by an internal transfer pump within the ancillary diesel generator skid.

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Two separate prefabricated insulated, heated and ventilated weather enclosures are provided for the transfer systems. Each enclosure houses one diesel fuel oil transfer pump assembly. The enclosures are sufficiently separated to prevent a fire in either enclosure from causing an interruption in the other flow path.

Characteristics of the system components are provided in Table 9.5.4-1.

9.5.4.2.2 Component Description

9.5.4.2.2.1 Fuel Storage Tanks

The two fuel oil storage tanks are located on grade. The tanks are designed and fabricated to API-650 Standards. Fittings are provided for each tank for level instrumentation, ventilation, sampling, water removal and sounding. Flanged openings are provided as manholes for access to the tank interior and each tank is equipped with an internal sump and a drain connection. Each tank is erected on a continuous concrete slab totally contained within a concrete dike to contain spills and prevent damage to the environment and seepage into the ground water.

The design of the standby diesel fuel oil system allows replenishment of fuel without interrupting operation of the diesel generator. The tank fill connection includes an internal pipe and diffuser to limit inlet filling velocities to prevent turbulence of sediment on the bottom of the tank. In addition, the diesel fuel oil transfer connections at the fuel oil storage tanks are 12 inches above the tank bottom to reduce the potential of sediment entry into the pipe line. A moisture separator and duplex filters are provided in the diesel fuel oil piping and a duplex fuel oil filter is provided on each engine to prevent detrimental effects on diesel performance from sediment. A 55,000 gallon supply of fuel oil is sufficient to power automatically sequenced loads for 7 days with 10% margin for load growth.

9.5.4.2.2.2 Diesel Generator Fuel Oil Transfer

The diesel generator fuel oil transfer system consists of two modularized skid mounted assemblies, each consisting of suction strainers, a transfer pump, a fuel oil heater, a moisture separator, and a fuel filter with the interconnecting piping, valves and instrumentation.

The fuel oil transfer pumps are of the motor driven gear positive displacement type. Each pump capacity is approximately four times the maximum consumption rate of the associated diesel generator. The pump and pump motor are mounted on a common baseplate. A prefabricated weather enclosure protects the strainer, transfer pumps, heater, moisture separator, and duplex filters and associated piping. There is no fixed fire protection water system inside the enclosure; therefore, spurious actuation of a fire protection system cannot occur.

9.5.4.2.2.3 Standby Diesel Generator Fuel Oil Day Tanks

The diesel generator fuel oil day tanks each provide a 1,300 gallon supply of fuel, which is sufficient to power automatically sequenced loads for 4 hours with 10% margin for load growth without resupply from a fuel oil storage tank. The day tanks are located within the diesel generator building and are separated from the remainder of the diesel generator building by 3-hour rated fire barriers. The day tanks are separate from sources of ignition or high-temperature surfaces. The day tank elevation is selected to provide the necessary suction head for the diesel engine fuel oil pump. The fuel oil piping is run in a piping trench from the tank to the engine. The fuel oil piping on the engine is located away from hot surfaces. Tank fittings provide for external tank fill, water removal, recirculation, and instrumentation. The fuel oil day tank is vented to atmosphere with a line which has a ball float check valve, and flame arrestor at the end. Since venting is to the outside atmosphere, there is not a buildup of combustible fumes within the diesel generator building.

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9.5.4.2.2.4 Ancillary Diesel Generator Fuel Oil Storage Tank

The ancillary diesel generator fuel oil storage tank provides four days of operation of the ancillary diesel generators. The tank is analyzed to show that it will withstand an SSE and is located in the same room as the ancillary diesel generators in the annex building. This room is separated from the rest of the annex building by a 3-hour rated fire barrier. The ancillary diesel generator fuel oil storage tank is vented to the atmosphere with a line which has a flame arrestor at the end. Since venting is to the outside atmosphere, there is not a buildup of combustible fumes within the annex. An overfill prevention valve is installed in the tank fill nozzle to prevent an overfilled tank and overflow in the vent line.

9.5.4.2.2.5 Piping and Tank Surfaces

The exterior and interior surfaces of the fuel oil storage tanks are painted with a primer and finish coat system for corrosion protection of the tank surface. Exterior surfaces of the diesel fuel oil transfer piping are painted for corrosion protection. Buried sections are enclosed in guard pipes to prevent leakage to the environment.

The guard pipe containment system is corrosion resistant plastic, designed and fabricated for the site overburden wheel loads which result from equipment removal and replacement.

9.5.4.2.3 System Operation

The fuel oil storage tanks for the diesel generators are replenished from trucks (or other mobile suppliers) as required to maintain a seven day supply for each standby diesel generator. Each storage tank is equipped with a vent line to atmosphere at the top of the tank that ends with a flame arrester. A tank fill line runs to each tank and is extended to the truck unloading station. The fill line incorporates a normally closed valve and a filler cap at the end to preclude the entrance of water. The fill line is above grade. The fill line has a strainer located downstream of the isolation valve to prevent entrance of deleterious solid material into the tank. A water removal port is located at the tank sump.

Each diesel oil transfer pump takes suction from a fuel oil storage tank and discharges fuel oil to the diesel generator fuel oil day tank. Each pump is capable of supplying its diesel generator and, simultaneously, increasing the inventory in the fuel oil day tank. The fuel oil transfer pump is automatically started and stopped on day tank level control. The filter in the discharge line to the day tank is monitored by measuring differential pressures across the filter and by providing a high differential pressure alarm.

In the event the diesel fuel oil degrades during storage, biocides and other fuel additives are introduced to the tanked fuel oil to prevent deterioration of the oil, accumulation of sludge in the storage tanks, and the growth of algae and fungi.

Site-specific conditions determine the requirements for oil supply and emergency fuel delivery.

Provisions are included in the fuel oil storage tanks and day tanks to check and remove accumulated water.

The fuel oil storage tank for the ancillary diesel generators is replenished from trucks (or other mobile supplier) as required to maintain a 4-day supply for both ancillary diesel generators.

9.5.4.3 Safety Evaluation

The standby diesel fuel oil system serves a defense in depth function and requires no nuclear safety evaluation.

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9.5.4.4 System Evaluation

The standby diesel generator fuel oil transfer system supplies fuel oil to the diesel generators which provide defense in depth electric power for investment protection.

The fuel oil storage tanks are sized to provide sufficient capacity for seven days of operation for each standby diesel generator. Within this period, the operator can arrange for additional fuel to be delivered to the plant site. An independent fuel supply path consisting of a fuel storage tank, a day tank, strainer, transfer pump, piping, oil heater, oil filter, moisture separator and valves is provided for each diesel generator. Each pump is powered from the electrical bus on which the diesel generator it serves is connected. Failure of a pump or a diesel generator would not affect the operability of components in the other train.

Maintenance of the fuel oil temperature above the cloud point is achieved automatically on low temperatures by an electric fuel oil heater at the discharge of the transfer fuel oil pump and by burial of the transfer piping below the frostline. The fuel oil system can be maintained above the cloud point temperature with the system electric heater in service and operation in the recirculation mode (by passing the day tank) back to the fuel oil storage tank. Above grade piping and inline equipment outdoors are insulated.

Electrical power supply for the diesel fuel oil transfer pumps and electric heater is from the associated diesel generator backed 480 V bus.

The fuel oil storage tank for the ancillary diesel generators is sized to provide sufficient capacity for four days of operation for both ancillary diesel generators. The ancillary diesel generators are not needed for the first 72 hours following a loss of all ac. Therefore, the operator has seven days to arrange for additional fuel to be delivered to the plant site. Maintenance of the fuel oil temperature above the cloud point when a normal ac source is available is achieved by the normal annex building heating and ventilation system maintaining room temperature within its normal range. Maintenance of the fuel oil temperature above the cloud point during operation of the ancillary diesel generators is achieved by electric tank heaters and tank insulation.

9.5.4.5 Tests and Inspections

9.5.4.5.1 Diesel Generator Fuel Oil Supply

The standby diesel generator fuel oil storage and transfer system operability may be demonstrated during tests of the diesel generator, or testing may be performed by operation of the system in recirculation mode (bypassing the day tank) and pumping fuel through the recirculation line back to the fuel oil storage tank. Fuel reserve for testing is supplied by sizing the storage tanks to contain fuel in excess of the volume required for seven days of continuous operation of all automatically sequenced loads with 10% margin for load growth. Provisions are made to sample and analyze diesel fuel periodically to verify the fuel quality requirements.

9.5.4.5.2 Fuel Oil Quality

The diesel fuel oil testing program requires testing both new fuel oil and stored fuel oil. High fuel oil quality is provided by specifying the use of ASTM Grade 2D fuel oil with a sulfur content as specified by the engine manufacturer. The program includes sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

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1. An API gravity or an absolute specific gravity within limits, when tested in accordance with ASTM D1298;

2. A flash point and kinematic viscosity is tested to be within the limits specified in Table 1 of ASTM D975 for ASTM 2D fuel oil, and

3. A clear and bright appearance with proper color is verified per ASTM D4176 or a water and sediment content within limits of ASTM D975.

b. The remaining critical parameters per Table 1 of ASTM D975, other than those addressed above in a., are verified compliant within 30 days following sampling and addition of the new fuel oil to storage tanks.

c. Total particulate concentration of the fuel oil is ≤ 10 mg/L when tested every 92 days.

d. The fuel oil storage tanks are inspected on a quarterly basis for the presence of water. Any accumulated water is to be removed.

9.5.4.6 Instrumentation Applications

9.5.4.6.1 Standby Diesel Generator Fuel Oil Supply

The transfer pumps can be operated from the control room. Alarms and indications of tank levels and transfer pump status are displayed in the control room. A secondary means of tank level determination is provided by dipsticks or sounding ports. Day tank fuel oil transfer pumps start and stop on low and high level, respectively, and the tank level transmitter activates a day tank high or low level alarm. The diesel oil transfer pumps start automatically when the level in the day tank decreases to set capacity. The day tank low level alarm annunciates when the level decreases to a point where 2 hours of fuel remain. The diesel oil transfer pumps are automatically stopped when the day tank level has increased to a higher set level.

Low fuel oil level in the standby diesel fuel oil storage tanks is also alarmed.

9.5.4.6.2 Ancillary Diesel Generator Fuel Oil Supply

There is no control room monitoring or control associated with the ancillary diesel generator fuel oil supply system. All controls and instruments are local/manual only. Provision is made to locally monitor fuel level in the tank.

9.5.4.7 Combined License Information

9.5.4.7.1 The site-specific need for cathodic protection in accordance with NACE Standard RP-01-69 for external metal surfaces of metal tanks in contact with the ground is addressed in APP-GW-GLR-120 (Reference 24), and the applicable changes are incorporated.

9.5.4.7.2 The site-specific factors in the fuel oil storage tank installation specification to reduce the effects of sun heat input into the stored fuel, the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations, and measures to protect against fuel degradation by a program of fuel sampling and testing are partially addressed in APP-GW-GLR-120 (Reference 24), and the applicable changes are incorporated, and by the information as delineated in the following paragraph:

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The epoxyurethane paint color selected for the exterior of the standby diesel fuel oil storage tanks shall be white to minimize radiant sunlight heat transmission to the tank oil stored fuel volume.

The diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations and the measures to protect against fuel degradation by a program of fuel sampling and testing are addressed in Subsection 9.5.4.5.2.

9.5.5 References

1. NUREG-0800, U. S. Nuclear Regulatory Commission Standard Review Plan, Section 9.5.1, "Fire Protection Program," Revision 3, July 1981, including Branch Technical Position (BTP) CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," Revision 2, July 1981.

2. National Fire Protection Association Codes and Standards:

   NFPA 10, 1998: Standard for Portable Fire Extinguishers; NFPA 13, 1999: Standard for the Installation of Sprinkler Systems; NFPA 14, 2000: Standard for the Installation of Standpipe, Private Hydrants, and Hose Systems; NFPA 15, 2001: Standard for Water Spray Fixed Systems for Fire Protection; NFPA 20, 1999: Standard for the Installation of Stationary Pumps for Fire Protection; NFPA 22, 1998: Standard for Water Tanks for Private Fire Protection; NFPA 24, 1995: Standard for Installation of Private Fire Service Mains and Their Appurtenances; NFPA 30, 2000: Flammable and Combustible Liquids Code; NFPA 50A, 1999: Standard for Gaseous Hydrogen Systems at Consumer Sites; NFPA 50B, 1999: Standard for Liquefied Hydrogen Systems at Consumer Sites; NFPA 72, 1999: National Fire Alarm Code; NFPA 780, 2000: Standard for the Installation of Lightning Protection Systems; NFPA 804, 2001: Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants.

3. "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," IEEE Std 3831974.

4. Fire Protection Handbook, Edited by A. E. Cote, National Fire Protection Association, 16th edition.

5. IES - 1987 Lighting Handbook.

6. IEEE Standard 281, "IEEE Standard Service Conditions for Power System Communication Equipment," 1984.

7. Beranek, Lee L., Noise Reduction, McGraw Hill Book Co., 1960.

8. EPRI Report NP 6559, "Voice Communication Systems Compatible with Respiratory Protection."

9. NRC IE Circulator No. 80-89, "Problems with Plant Internal Communications Systems," April 18, 1990.

10. 10 CFR 50, Appendix E, IV.E.9, "ERF Communications."

11. NRC IEN 87-58, "Continuous Communication Following Emergency Notifications."

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12. NRC IEN 86-097, "Emergency Communication Systems."

13. Not Used.

14. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plan."

15. 10 CFR 50, Section 50.48 "Fire Protection."

16. 10 CFR 50, Appendix A, Criterion 3 "Fire Protection."

17. NRC Policy Issue SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," Section I.E, "Fire Protection."

18. IEEE Standard for Flame Testing of Cables for Use in Cable Tray in Industrial and Commercial Occupancies, IEEE Std. 1202 - 1991.

19. Standard Practices for Medium Speed Stationary Diesel and Gas Engines, Sixth Edition, Diesel Engine Manufacturers Association, 1972. (Note: Although this standard is obsolete, the guidance for distillate fuel systems in Chapter 13 is applicable for AP1000.)

20. ASTM E119, "Fire Test of Building Construction and Materials," 1976.

21. NRC Bulletin 80-15, "Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power," June 18, 1980.

22. APP-GW-GLR-027, "Operator Actions Minimizing Spurious ADS Actuations," Westinghouse Electric Company LLC.

23. APP-GW-GLR-019, "Fire Resistance Test Data," Westinghouse Electric Company LLC.

24. APP-GW-GLR-120, “Cathodic Protection for Metal Tanks in Contact with the Ground,” Westinghouse Electric Company LLC.

25. National Fire Protection Association, “Fire Tests, Building Construction and Materials," NFPA 251, 1975.

201. National Fire Protection Association, “Standard for Inspection, Care, and Use of Fire Hose, Couplings, and Nozzles and the Service Testing of Fire Hose,” NFPA 1962, 2003.

202. National Fire Protection Association, “Standard for Fire Prevention During Welding, Cutting, and Other Hot Work,” NFPA 51B, 2003.

203. National Fire Protection Association, “Standard for Safeguarding Construction, Alteration, and Demolition Operations,” NFPA 241, 2004.

204. National Fire Protection Association, “Standard on Industrial Fire Brigades,” NFPA 600, 2005.

205. National Fire Protection Association, “Recommended Practice for Pre-incident Planning,” NFPA 1620, 2003.

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206. National Fire Protection Association, “Standard Methods of Fire Tests for Flame Propagation of Textiles and Films,” NFPA 701, 2004.

207. National Fire Protection Association, “Standard for Fire-Retardant Treated Wood and Fire-Retardant Coatings for Building Materials,” NFPA 703, 2006.

208. National Fire Protection Association, “Standard for Fire Service Respiratory Protection Training,” NFPA 1404, 2006.

209. National Fire Protection Association, “Standard on Training for Initial Emergency Scene Operations,” NFPA 1410, 2005.

210. National Fire Protection Association, “Standard on Fire Department Occupational Safety and Health Program,” NFPA 1500, 2007.

211. Not used.

212. National Fire Protection Association, “Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems,” NFPA 25, 2008.

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Table 9.5.1-1 (Sheet 1 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
Fire Protection Program			
A. Definition: **Fire Resistance Rating** – *The time that materials or assemblies have withstood a fire exposure as established in accordance with the test procedures of "Standard Methods of Fire Tests of Building Construction and Materials" (NFPA 251).*	B.4	AC	Fire Resistance Ratings may utilize other standard methods of fire tests of building construction and materials that are recognized by a national association (i.e., ASTM, UL, NFPA) and/or the U.S. Nuclear Regulatory Commission. See also Table 9.5.1-3.
1. Direction of fire protection program; availability of personnel.	C.1.a(1)	WA C	See Note 2 Comply. Subsection 9.5.1.8.1.2 address this requirement.
2. Defense-in-depth concept; objective of fire protection program.	C.1.a(2)	WA C	See Note 2 Comply. Subsections 9.5.1.8 and 9.5.1.8.1 address this requirement.
3. Management responsibility for overall fire protection program; delegation of responsibility to staff.	C.1.a(3)	WA C	See Note 2 Comply. Subsections 9.5.1.8.1.2
4. The staff should be responsible for: a. Fire protection program requirements. b. Post-fire shutdown capability. c. Design, maintenance, surveillance, and quality assurance of fire protection features. d. Fire prevention activities. e. Fire brigade organization and training. f. Prefire planning.	C.1.a(3)	WA C	See Note 2
5. The organizational responsibilities and lines of communication pertaining to fire protection should be defined through the use of organizational charts and functional descriptions.	C.1.a(4)	WA C	See Note 2
6. Personnel qualification requirements for fire protection engineer, reporting to the position responsible for formulation and implementation of the fire protection program.	C.1.a(5)(a)	WA C	See Note 2
7. The fire brigade members' qualifications should include a physical examination for performing strenuous activity, and the training described in Position C.3.d.	C.1.a(5)(b)	WA C	See Note 2 Comply. Subsections 9.5.1.8.2.1 and 9.5.1.8.2.2 addresses this requirement.
8. The personnel responsible for the maintenance and testing of the fire protection systems should be qualified by training and experience for such work.	C.1.a(5)(c)	WA C	See Note 2 Comply. Subsection 9.5.1.8.7 addresses this requirement.

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Table 9.5.1-1 (Sheet 2 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
9. The personnel responsible for the training of the fire brigade should be qualified by training and experience for such work.	C.1.a(5)(d)	WA C	See Note 2 Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
10. The following NFPA publications should be used for guidance to develop the fire protection program: No. 4, No. 4A, No. 6, No. 7, No. 8, and No. 27.	C.1.a(6)	WA C	See Note 2 Alternate Compliance. The NFPA codes cited in BTP CMEB 9.5-1 are historical. Current NFPA codes are referenced for guidance for the fire protection program. Subsection 9.5.1.8.1.1 addresses this requirement.
11. On sites where there is an operating reactor, and construction or modification of other units is underway, the superintendent of the operating plant should have a lead responsibility for site fire protection.	C.1.a(7)	WA C	See Note 2
Fire Protection Analysis			
12. The fire protection analysis should demonstrate that the plant will maintain the capability to perform safe shutdown functions and minimize radioactive releases to the environment in the event of a fire.	C.1.b	C	
13. The fire protection analysis should be performed by fire protection and reactor systems engineers to (1) consider potential in situ and transient fire hazards; (2) determine the consequences of a fire in any location in the plant; and (3) specify measures for fire prevention, detection, suppression, and containment.	C.1.b	C	
14. Fires involving facilities shared between units should be considered.	C.1.b	WA C	Comply. The FHA demonstrates the plant's ability to perform safe shutdown functions and minimize radioactive releases to the environment. Postulated fires in shared facilities that do not contain SSCs important to safety and do not contain radioactive materials do not affect these functions.

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Table 9.5.1-1 (Sheet 3 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
15. Fires due to man-made site-related events that have a reasonable probability of occurring and affecting more than one reactor unit should be considered.	C.1.b	WA C	To be evaluated on a site-specific basis. Plant siting decisions are expected to preclude the need to consider such events. Comply. Sections 2.2 and 3.5 establish that these events are not credible.
16. Establishing three levels of fire damage limits according to safety function (hot shutdown, cold shutdown and design basis accidents).	C.1.b	C	AP1000 uses two levels of damage limits: safe shutdown and design basis accidents. Safe shutdown capability is protected from damage caused by a single fire.
17. The fire protection analysis should separately identify hazards and provide appropriate protection in locations where safety-related losses can occur.	C.1.b	C	
Fire Suppression System Design Basis			
18. Total reliance should not be placed on a single fire suppression system. Backup fire suppression capability should be provided.	C.1.c(1)	C	Automatic fire suppression systems are backed up by manual suppression systems (standpipe) and portable extinguishers.
19. A single active failure or a crack in a fire suppression system moderate energy line should not impair both the primary and backup fire suppression capabilities.	C.1.c(2)	AC	Criteria followed except for containment and outlying buildings. The piping serving the standpipe and fire hose connections located inside the containment are analyzed to remain functional following a safe shutdown earthquake. The containment isolation piping is qualified to seismic Category I criteria. These attributes reduce the potential for a failure of the system. The buildings outside the auxiliary building do not contain safety-related equipment, or present an exposure hazard to structures containing safety-related equipment. Manual fire suppression capability using hose lines connected to the outside hydrants of the yard main can be provided in the event of a failure of the interior fire suppression systems.

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Table 9.5.1-1 (Sheet 4 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
20. The fire suppression system should be capable of delivering water to manual hose stations located within hose reach of areas containing equipment required for safe shutdown following a safe shutdown earthquake (SSE).	C.1.c(3)	AC	Criteria followed except for the PCS valve room, Room 12701 (Fire Zone 1270 AF 12701). The quantity of combustible material in this fire zone is extremely low, consisting primarily of cable insulation related to the six PCS valves and related PCS instrumentation. Portable fire extinguishers are provided on both the lower level (El. 264′-6″) and the upper level (El. 286′-6″) of the PCS valve room for manual fire fighting.
Fire Suppression System Design Basis			
21. Fire protection systems should retain their design capability for natural phenomena of less severity and greater frequency than the most severe natural phenomena.	C.1.c(4)	C	Structures housing the fire protection system meet the applicable requirements of Chapter 3 to provide protection from natural phenomena.
22. Fire protection systems should retain their original design capability for potential man-made, site-related events that have a reasonable probability of occurring at a specific plant site.	C.1.c(4)	WA C	To be evaluated on a site-specific basis. Plant siting decisions are expected to preclude the need to consider such events. Comply. Sections 2.2 and 3.5 establish that these events are not credible.
23. The effects of lightning strikes should be included in the overall plant fire protection program.	C.1.c(4)	C	Lightning protection will be provided per NFPA 780.
24. The consequences of inadvertent operation or of a crack in a moderate energy line in the fire suppression system should meet the guidelines specified for moderate energy systems outside containment in SRP Section 3.6.1.	C.1.c(5)	C	

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Table 9.5.1-1 (Sheet 5 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
Alternate or Dedicated Shutdown			
25. Alternative or dedicated shutdown capability should be provided where the protection of systems whose functions are required for safe shutdown is not provided by established fire suppression methods or by Position C.5.b.	C.1.d	AC	In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G.2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and PCS are used to achieve and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP.
Fire Protection Program Implementation			
26. The fire protection program for buildings storing new reactor fuel and for adjacent fire areas that could affect the fuel storage area should be fully operational before fuel is received at the site.	C.1.e(1)	WA C	See Note 2 Comply. Subsection 9.5.1.8.1 addresses this requirement.
27. The fire protection program for an entire reactor unit should be fully operational prior to initial fuel loading in that unit.	C.1.e(2)	WA C	See Note 2 Comply. Subsection 9.5.1.8.1 addresses this requirement.
28. Special considerations for the fire protection program on reactor sites where there is an operating reactor and construction or modification of other units is under way.	C.1.e(3)	WA C	See Note 2 Comply. Subsection 9.5.1.8.1.2.m addresses this requirement.
29. Establishing administrative controls to maintain the performance of the fire protection system and personnel.	C.2	WA C	See Note 2 Comply as identified in Subsection 9.5.1.8.
Fire Brigade			
30. The guidance in Regulatory Guide 1.101 should be followed as applicable.	C.3.a	WA C	See Note 2 Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
31. Establishing site brigade: minimum number of fire brigade members on each shift; qualification of fire brigade members; competence of brigade leader.	C.3.b	WA C	See Note 2 Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
32. The minimum equipment provided for the brigade should consist of turnout coats, boots, gloves, hard hats, emergency communications equipment, portable ventilation equipment, and portable extinguishers.	C.3.c	WA C	See Note 2 Comply. Subsection 9.5.1.8.2.2 addresses this requirement.

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Table 9.5.1-1 (Sheet 6 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
33. Recommendations for breathing apparatus for fire brigade, damage control, and control room personnel.	C.3.c	C WA	Comply. Subsection 9.5.1.8.2.2 and Subsections 6.4.2.3 and 6.4.4 address these requirements. See Note 2. A breathing air compressor is provided in the compressed and instrument air system (CAS) to replenish the exhausted air supply bottles used by the fire brigade. Additionally, an equivalent 6-hour supply of reserve air (e.g., the 12 additional SCBA bottles) will be maintained in an area located outside of the turbine building. See Subsection 9.3.1 for further information.
34. Recommendations for the fire brigade training program.	C.3.d	WA C	See Note 2 Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
Quality Assurance Program			
35. Establishing quality assurance (QA) programs by applicants and contractors for the fire protection systems for safety-related areas; identification of specific criteria for quality assurance programs.	C.4	C WA	Comply. Subsection 9.5.1.7 and Chapter 17 address this requirement. Fire protection quality assurance programs are incorporated in procurement documents. Deviations are evaluated and controlled. See Items 1. through 11. of this table for on-site implementation. See Note 2.
Building Design			
36. Fire barriers with a minimum fire resistance rating of 3 hours should be provided to separate safety-related systems from any potential fires in nonsafety-related areas.	C.5.a(1)(a)	C	Structures housing safety-related systems are separated from nonsafety-related structures by 3-hour rated fire walls.
37. Fire barriers with a minimum fire resistance rating of 3 hours should be provided to separate redundant divisions of safety-related systems from each other.	C.5.a(1)(b)	C	See Subsection 9.5.1.2.1.1 for discussion of exceptions for the containment, main control room, and remote shutdown room.
38. Fire barriers with a minimum fire resistance rating of 3 hours should be provided to separate individual units on a multiple-unit site.	C.5.a(1)(c)	NA	See discussion of GDC 5 in subsection 3.1.1. The AP1000 is a single-unit plant.

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Table 9.5.1-1 (Sheet 7 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
39. Fire barriers should be provided within a single safety division to separate components or cabling that present a fire hazard to other safety-related components.	C.5.a (2)	C	
40. Openings through fire barriers for pipe, conduit, and cable trays that separate fire areas should be sealed or closed to provide a fire resistance rating equal to that required of the barrier.	C.5.a (3)	C	
41. Recommendations for internal sealing of conduits penetrating fire barriers.	C.5.a (3)	C	
42. Fire barrier penetrations that must maintain environmental isolation or pressure differentials should be qualified by test.	C.5.a (3)	C	Fire penetration seals that also perform other barrier functions are qualified by test for intended functions. The fire barrier penetration seal does not perform other barrier functions simultaneously.
43. Penetration designs should use only noncombustible materials.	C.5.a (3)	AC	Fire seal materials meet the guidance of NUREG-1552. The seal design and tests demonstrate that the seals are capable of preventing the spread of fire and perform their intended safety function.

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Table 9.5.1-1 (Sheet 8 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
44. The penetration qualification tests should use the time-temperature exposure curve specified by ASTM E119.	C.5.a (3)	C	
45. Criteria for penetration qualification tests.	C.5.a (3)	AC	Qualification tests for penetration seals meet the acceptance criteria of this paragraph or the criteria of the industry standards accepted in NUREG-1552. In lieu of BTP CMEB 9.5-1 C.5.a(3)(b), RG 1.189 Rev. 1, Section 4.2.1.5b.ii is used: "The temperature levels recorded for the unexposed side of the fire barrier are analyzed and demonstrate that the maximum temperature does not exceed 163 °C (325 °F) or 121 °C (250 °F) above the ambient temperature. Higher temperatures at through-penetrations may be permitted when justified in terms of cable insulation ignitability."
46. Penetration openings for ventilation systems should be protected by fire dampers having a rating equivalent to that required of the barrier.	C.5.a (4)	C	Penetration openings are protected in accordance with NFPA 90A. Fire dampers generally not provided for roof or exterior wall penetrations. Two discharge openings from the Main Control Room Vestibule are not provided with fire dampers. See Subsection 9A.3.1.2.1.2 and 9A.3.1.2.5.1.
47. Flexible air duct couplings in ventilation and filter systems should be noncombustible.	C.5.a (4)	C	
48. Door openings in fire barriers should be protected with equivalently rated doors, frames, and hardware that have been tested and approved by a nationally recognized lab.	C.5.a (5)	C	
49. Fire doors should be self-closing or provided with closing mechanisms.	C.5.a (5)	C	
50. Fire doors should be inspected semiannually to verify that automatic hold-open, release, and closing mechanisms and latches are operable.	C.5.a (5)	WA C	See Note 2 Comply. Subsection 9.5.1.8.8 addresses this requirement.
51. Alternative means for verifying that fire doors protect the door opening as required in case of fire.	C.5.a (5)	WA C	See Note 2 Comply. Subsection 9.5.1.8.8 addresses this requirement.

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Table 9.5.1-1 (Sheet 9 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
52. The fire brigade leader should have ready access to keys for any locked fire doors.	C.5.a (5)	WA C	See Note 2 Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
53. Areas protected by automatic total flooding gas suppression systems should have electrically supervised self-closing fire doors or should satisfy guideline (49) above.	C.5.a (5)	NA	No automatic gas suppression systems are used on AP1000.
54. Personnel access routes and escape routes should be provided for each fire area.	C.5.a (6)	C	

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Table 9.5.1-1 (Sheet 10 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
55. Stairwells serving as escape routes, access routes for firefighting, or access routes to areas containing equipment necessary for safe shutdown should be enclosed in masonry or concrete towers with a minimum fire resistance rating of 2 hours and self-closing Class B fire doors.	C.5.a(6)	AC WA	AP1000 deviates from this guideline with a design that meets applicable building codes and fire protection requirements. Auxiliary building stairwells, with exception of the exterior wall sections enclosing Stairwell S03 above the auxiliary building roof, are enclosed in towers constructed using both concrete structural walls and nonstructural walls, consisting of a concrete/steel composite material, having a fire resistance rating of at least 2 hours, and self-closing doors, having a rating of 1.5 hours. Auxiliary building Stairwell S03 exterior wall sections enclosing the stairwell and elevator shaft above the auxiliary building roof do not have a fire resistance rating of 2 hours. Openings from the stairwell into the Auxiliary Building and Stair S06 at Elevation 256'-9'' are protected with listed automatic or self-closing fire doors having a fire rating of at least 1.5 hours. For buildings outside of the Nuclear Island, exterior wall sections enclosing stairwells do not have a fire resistance rating of 2 hours, only interior stairwell wall sections are rated for the minimum 2 hour fire resistance rating. Openings in the interior stairwell wall sections are protected with listed automatic or self-closing fire doors having a fire protection rating of at least 1.5 hours.

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Table 9.5.1-1 (Sheet 11 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
		C	The two-hour fire-resistance test data for the concrete/steel composite material is in accordance with ASTM E119 and NFPA 251 acceptance criteria to ensure that the stairwell fire-related enclosure will be maintained in accordance with Regulatory Position C.5.a.6 of BTP CMEB 9.5-1. There are no missile hazards in the vicinity of such stairwells. This alternative protection provides an equivalent level of safety as that prescribed in the BTP. Comply. Subsection 9A.3.3 addresses this requirement for miscellaneous buildings located in the yard.
56. Fire exit routes should be clearly marked.	C.5.a (7)	WA C	See Note 2 Comply. Subsection 9.5.1.2.1.1 addresses this requirement.
57. Each cable spreading room should contain only one redundant safety division.	C.5.a (8)	NA	There are no cable spreading rooms in AP1000.
58. Cable spreading rooms should be separated from each other and from other areas of the plant by barriers having a minimum fire resistance of 3 hours.	C.5.a (8)	NA	There are no cable spreading rooms in AP1000.
59. Interior wall and structural components, thermal insulation materials, radiation shielding materials, and soundproofing materials should be noncombustible.	C.5.a (9)	C	
60. Interior finishes should be noncombustible.	C.5.a (9)	C	
61. Metal deck roof construction should be noncombustible and listed as "acceptable for fire" in the UL Building Materials Directory, or listed as Class I in the Factory Mutual Approval Guide.	C.5.a (10)	C	
62. Suspended ceilings and their supports should be of noncombustible construction.	C.5.a (11)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
63. Concealed spaces should be devoid of combustibles except as noted in Position C.6.b.	C.5.a(11)	AC	Underfloor or ceiling spaces, contain combustible cable insulation in the main control room, control support area and remote shutdown room. Fire detectors are provided in these areas. The insulating and jacketing materials for the cables used in the plant are qualified in accordance with the criteria specified in IEEE Standard 1202, IEEE Standard 383, or the standards for the corresponding test types as listed in Table K-1 and K-2 in NEI 04-02, Rev 3. This alternative protection provides an equivalent level of safety as that specified in the BTP.
64. Transformers installed inside fire areas containing safety-related systems should be of the dry type or insulated and cooled with noncombustible liquid.	C.5.a(12)	C	
65. Outdoor oil-filled transformers should have oil containment features or drainage away from the buildings.	C.5.a(13)	C	
66. Outdoor oil-filled transformers should be located at least 50 feet distant from the building, or building walls within 50 feet of oil-filled transformers should be without openings and have a 3-hour fire resistance rating.	C.5.a (13)	C	
67. Floor drains sized to remove expected firefighting water flow without flooding safety-related equipment should be provided in areas where fixed water fire suppression systems are installed.	C.5.a (14)	C	
68. Floor drains should be provided in areas where hand hose lines may be used if such firefighting water could cause unacceptable damage to safety-related equipment.	C.5.a (14)	C	
69. Where gas suppression systems are installed, the drains should be provided with adequate seals, or the gas suppression system should be sized to compensate for the loss of the suppression agent through the drains.	C.5.a (14)	NA	No fixed gas suppression systems are used on AP1000.
70. Drains in areas containing combustible liquids should have provisions for preventing the back flow of combustible liquids to safety-related areas through the interconnected drain systems.	C.5.a (14)	C	

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Table 9.5.1-1 (Sheet 13 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
71. Water drainage from areas that may contain radioactivity should be collected, sampled, and analyzed before discharge to the environment.	C.5.a(14)	WA C	See Note 2. Capability is provided. Comply. Capability is provided. Subsection 9.5.1.8.5 addresses this requirement.
Safe Shutdown Capability			
72. Fire damage should be limited so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the main control room or emergency control station is free of fire damage.	C.5.b(1)	C	
73. Fire damage should be limited so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.	C.5.b (1)	AC	Safe shutdown following a fire is defined for the AP1000 plant as the ability to achieve and maintain the reactor coolant system (RCS) core average temperature below 215.6°C (420°F) without uncontrolled venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. With expected RCS leakage, the AP1000 plant can maintain safe shutdown conditions for greater than 14 days. Therefore, repairs to systems necessary to reach cold shutdown need not be completed within 72 hours.
74. Separation requirements for verifying that one train of systems necessary to achieve and maintain hot shutdown is free of fire damage.	C.5.b (2)	C	

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AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
75. Provision of alternative or dedicated shutdown capability in certain fire areas.	C.5.b (3)	AC	In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G.2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and PCS are used to achieve and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP.
76. Alternative or dedicated shutdown capability.	C.5.c	AC	In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G.2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and PCS are used to achieve and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP.

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AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
			The criteria concerning cold shutdown capability deviates from the criteria applied to the evolutionary reactor designs, but is consistent with the criteria applicable to existing plants. To enhance the survivability of the normal safe shutdown and cold shutdown capability in the event of a fire, and to reduce the reliance on the infrequently utilized safety-related passive systems, automatic suppression or physical separation is provided in those fire areas outside containment where a fire could damage the normal shutdown capability. This criterion does not ensure that the normal shutdown capability will be free of fire damage, or that the equipment necessary to achieve and maintain cold shutdown can be repaired within 72 hours.
Control of Combustibles			
77. Safety-related systems should be separated from combustible materials where possible; where not possible, special protection should be provided to help prevent a fire from defeating the safety system function.	C.5.d (1)	C	Concentrations of combustible materials are located outside structures containing safety-related components. Where this is not possible, appropriate fire protection is provided (see Appendix 9A).
78. Bulk gas storage (compressed or cryogenic) should not be permitted inside structures housing safety-related equipment. Flammable gases should be stored outdoors or in separate detached buildings.	C.5.d (2)	AC	Breathing air storage tanks for the main control room habitability system are safety-related and are provided with overpressure protection. There is no other bulk gas storage in structures housing safety-related equipment.
79. High pressure gas storage containers should be located with the long axis parallel to building walls.	C.5.d (2)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
80. Use of compressed gases inside buildings should be controlled.	C.5.d (2)	WA C	See Note 2 Comply. Subsection 9.5.1.8.4.g addresses this requirement.
81. The use of plastic materials should be minimized. Halogenated plastics such as polyvinyl chloride (PVC) and neoprene should be used only when substitute noncombustible materials are not available.	C.5.d (3)	C	
82. Storage of flammable liquids should comply with NFPA 30.	C.5.d (4)	C	See Note 3
83. Hydrogen lines in safety-related areas should be either designed to seismic Category I requirements, or sleeved, or equipped with excess flow valves.	C.5.d (5)	C	Hydrogen lines in safety-related areas are designed to seismic Category I requirements.
Electrical Cable Construction, Cable Trays, and Cable Penetrations			
84. Only metal should be used for cable trays.	C.5.e (1)	C	Cable trays are of all-metal construction.
85. Only metallic tubing should be used for conduit. Thin-wall metallic tubing should not be used.	C.5.e (1)	NC	Conduit that is not buried or embedded in concrete is metallic. EMT (thin-wall metallic tubing / conduit) is used in limited designs for lightning and communications systems as well as "limited" commodity packages that are not Safety Related and do not have a "Functional Impact" to the plant design.
86. Flexible metallic tubing should only be used in short lengths to connect components to equipment.	C.5.e (1)	C	
87. Other raceways should be made of noncombustible materials.	C.5.e (1)	C	
88. Redundant safety-related cable systems outside the cable spreading room should be separated from each other and from potential fire exposure hazards in nonsafety-related areas by 3-hour rated fire barriers.	C.5.e (2)	C	
89. These cable trays should be provided with continuous line-type heat detectors.	C.5.e (2)	C	
90. Cables should be designed to allow wetting down with fire suppression water without electrical faulting.	C.5.e (2)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
91. Redundant cable trays that route safety-related cables outside the cable spreading room should be accessible for manual firefighting. Manual hose stations and portable hand extinguishers should be provided.	C.5.e (2)	C	
92. Cable trays that route safety-related cables of a single division, which are separated from redundant divisions by a 3-hour rated fire barrier and are accessible for manual firefighting, should be protected from the effects of a potential exposure fire by providing automatic water suppression, unless specific conditions are met.	C.5.e (2)	C	Automatic water suppression is not provided because there is no significant exposure fire hazards and the specified conditions are met.
93. Cable trays that route safety-related cables that are not accessible for manual firefighting should be protected by an automatic water system.	C.5.e (2)	AC	Cable trays that route safety-related cables outside containment are accessible for manual firefighting. Protection of safe shutdown components inside containment is discussed in Appendix 9A.
94. Cable trays that route safety-related cables that are not separated from redundant divisions by 3-hour rated fire barriers should be protected by automatic water suppression systems.	C.5.e (2)	AC	Protection of safe shutdown components inside containment and the main control room is discussed in Appendix 9A.
95. In areas where 3-hour fire barrier separation of redundant cable systems is precluded by overriding design considerations, the capability to achieve safe shutdown considering the effects of a fire involving fixed and transient combustibles should be evaluated with and without actuation of the automatic suppression system.	C.5.e (2)	C	
96. Electric cable construction should pass the flame test in IEEE Std 383.	C.5.e (3)	C	Use IEEE Standard 1202, IEEE Standard 383, or the standards for the corresponding test types as listed in Tables K-1 and K-2 in NEI 04-02, Rev 3 excluding the option to use the alternate flame source, oil, or burlap.
97. Cable raceways should be used only for cables.	C.5.e (4)	C	
98. Miscellaneous storage and piping for combustible liquids or gases should not create a potential exposure hazard to safety-related systems.	C.5.e (5)	C	
Ventilation			
99. Smoke and corrosive gases should be discharged directly outside to an area that will not affect safety-related plant areas.	C.5.f (1)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
100. To facilitate manual firefighting, separate smoke and heat vents should be provided in certain areas.	C.5.f (1)	C	Smoke and heat venting capability is provided as described in Appendix 9A.
101. Release of smoke and gases containing radioactive materials to the environment should be monitored.	C.5.f (2)	C	
102. Any ventilation system designed to exhaust potentially radioactive smoke or gases should be evaluated to verify that inadvertent operation or single failures will not violate the radiologically controlled areas of the plant.	C.5.f (2)	C	
103. The power supply and control for mechanical ventilation systems should be run outside the fire area served by the system.	C.5.f (3)	C	
104. Engineered safety feature filters should be protected in accordance with the guidelines of Regulatory Guide 1.52.	C.5.f (4)	C	
105. Air intakes for ventilation systems serving areas containing safety-related equipment should be located remote from the exhaust air outlets and smoke vents of other fire areas.	C.5.f (5)	C	
106. Stairwells should be designed to minimize smoke infiltration during a fire.	C.5.f (6)	C	Stair towers are provided with self-closing doors. Additional measures to minimize smoke infiltration to stair-wells are described in Appendix 9A.
107. Where total flooding gas extinguishing systems are used, ventilation dampers should be controlled in accordance with NFPA 12 and NFPA 12A.	C.5.f (7)	NA	Fixed flooding gas suppression systems are not used on AP1000.
Lighting and Communication			
108. Fixed self-contained lighting units with individual 8-hour battery power supplies should be provided in areas that must be manned for safe shutdown and for access and egress routes to and from all fire areas.	C.5.g (1)	AC	Alternate emergency lighting is provided for the main control room and the remote shutdown workstation as described in Subsection 9.5.3. Emergency lighting in other plant areas is provided by 8-hour battery-powered, fixed, self-contained units to provide safe ingress and egress of personnel and the operation of equipment following a fire, in the event of a loss of the normal lighting. Portable battery-powered lighting is provided for emergency use by plant personnel.
109. Sealed beam battery-powered portable hand lights should be provided for emergency use.	C.5.g (2)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
110. Fixed emergency communications, independent of the normal plant communication system, should be installed at preselected stations.	C.5.g (3)	C	
111. A portable radio communications system should be provided for use by the fire brigade and other operations personnel required to achieve safe plant shutdown.	C.5.g (4)	WA C	Comply. Subsections 9.5.1.8.1.2.a.3.v, 9.5.1.8.2.2, and Subsection 9.5.2 address this requirement.
Fire Detection			
112. Fire detection systems should be provided for areas that contain or present a fire exposure to safety-related equipment.	C.6.a (1)	C	
113. Fire detection systems should comply with the requirements of Class A systems as defined in NFPA 72D and Class I circuits as defined in NFPA 70.	C.6.a (2)	C	See Note 3
114. Fire detectors should be selected and installed in accordance with NFPA 72E.	C.6.a (3)	C	See Note 3
115. Testing of pulsed line-type heat detectors should demonstrate that the frequencies used will not affect the actuation of protective relays in other plant systems.	C.6.a (3)	C	
116. Fire detection systems should give audible and visual alarm and annunciation in the main control room.	C.6.a (4)	C	
117. Where zoned detection systems are used in a given fire area, local means should be provided to identify which zone has actuated.	C.6.a (4)	C	
118. Local audible alarms should sound in the fire area.	C.6.a(4)	C	
119. Fire alarms should be distinctive and unique so they will not be confused with any other plant system alarms.	C.6.a (5)	C	
120. Primary and secondary power supplies, which satisfy the provisions of Section 2220 of NFPA 72D, should be provided for the fire detection system and for electrically operated control valves for automatic suppression systems.	C.6.a (6)	C	See Note 3
Fire Protection Water Supply Systems			
121. An underground yard fire main loop should be installed to furnish anticipated water requirements.	C.6.b (1)	C	An underground yard fire main loop is provided in accordance with NFPA 24.
122. Type of pipe and water treatment should be design considerations with tuberculation as one of the parameters.	C.6.b (1)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
123. Means of inspecting and flushing the systems should be provided.	C.6.b (1)	C	Flushing of the loop can be accomplished through the use of sectional control valves to direct the flow and yard hydrants to serve as discharge points.
124. Approved visually indicating sectional control valves should be provided to isolate portions of the yard fire main loop for maintenance or repair.	C.6.b (2)	C	Indicator valves are provided for sectionalized control and isolation of portions of the yard fire main loop.
125. Valves should be installed to permit isolation of outside hydrants from the fire main for maintenance or repair without interrupting the water supply to automatic or manual fire suppression systems.	C.6.b (3)	C	A visually indicating or key-operated valve is provided in each lateral from the yard fire main loop to a fire hydrant.
126. The fire main system piping should be separate from service or sanitary water system piping.	C.6.b (4)	C	
127. A common yard fire main loop may serve multi-unit nuclear power plant sites if cross-connected between units. Sectional control valves should permit maintaining independence of the loop around each unit.	C.6.b (5)	NA	See discussion of GDC 5 in subsection 3.1.1. The AP1000 is a single-unit plant.
128. A sufficient number of pumps should be provided so that 100 percent capacity will be available assuming failure of the largest pump or loss of offsite power.	C.6.b (6)	C	Two 100 percent capacity fire pumps (one diesel-driven and one electric motor-driven) are provided.
129. Individual fire pump connections to the yard fire main loop should be separated with sectionalizing valves between connections.	C.6.b (6)	C	
130. Each pump and its driver and controls should be separated from the remaining fire pumps by a 3-hour rated fire wall.	C.6.b (6)	C	
131. The fuel for the diesel fire pump should be separated so that it does not provide a fire source exposing safety-related equipment.	C.6.b (6)	C	
132. Alarms indicating pump running, driver availability, failure to start, and low fire main pressure should be provided in the main control room.	C.6.b (6)	C	
133. The fire pump installation should conform to NFPA 20.	C.6.b (6)	C	See Note 3
134. Outside manual hose installation should be sufficient to provide an effective hose stream to any onsite location where fixed or transient combustibles could jeopardize safety-related equipment. Hydrants should be installed approximately every 250 feet on the yard main system.	C.6.b (7)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
135. Recommendations for hose houses and hose carts.	C.6.b (7)	C	
136. Threads compatible with those used by local fire departments should be provided on all hydrants, hose couplings, and standpipe risers.	C.6.b (8)	C	
137. Two separate, reliable freshwater supplies should be provided.	C.6.b (9)	C	Two water storage tanks are provided.
138. Recommendations for tanks used to supply fire protection water.	C.6.b (9)	C	See Note 3
139. Recommendations for common tank used to supply fire protection water and other system.	C.6.b (10)	C	The configuration of the water supply for the seismic standpipe system is described in Subsection 9.5.1.2.1. Both fire water supply tanks have a water volume that is dedicated for fire protection purposes. The fire pumps can be aligned through normally closed valves or through temporary connections to supply water for post-accident services. These include refilling of the passive containment cooling water supply tank or supplying the containment spray following a severe accident. This provides adequate defense in depth and will not adversely affect the performance of the fire protection water supply.
140. The fire water supply should be based on the largest expected flow rate for a period of 2 hours, but not less than 300,000 gallons.	C.6.b (11)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
141. The fire water supply should be capable of delivering the design demand over the longest route of the water supply systems.	C.6.b (11)	AC	Note that in a single loop main configuration, the longest path and the hydraulically least demanding path being out of service are the same. The NFPA 804 requirement allows for a multiple loop configuration for the fire protection system main. Since the AP1000 has a multiple loop fire water main configuration, the longest path described in BTP CMEB 9.5-1 would require multiple impairments to the fire water main loop. The AP1000 fire protection supply is designed to supply the largest demand with the hydraulically least demanding path out-of-service.
142. Recommendations for freshwater lakes or ponds used to supply fire protection water.	C.6.b (12)	NA	Lakes or ponds are not utilized for fire protection water supply.
143. Recommendations concerning use of a common water supply for fire protection and the ultimate heat sink.	C.6.b (13)	C	
144. Recommendations concerning use of other water systems as the source of fire protection water.	C.6.b (14)	NA	The fire protection system does not rely on the operation of another water system as a second water source.
Water Sprinkler and Hose Standpipe Systems			
145. Recommendations concerning connection of sprinkler systems and manual hose station standpipes to the yard fire main loop.	C.6.c (1)	AC	See remarks for Guideline 19.
146. Each sprinkler and standpipe system should be equipped with OS&Y gate valve or other approved shutoff valve and waterflow alarm.	C.6.c (1)	C	Waterflow alarms are provided for sprinkler and seismic standpipe systems only.
147. Safety-related equipment should be protected from sprinkler discharge if such discharge could result in unacceptable damage to the equipment.	C.6.c (1)	C	
148. Control and sectionalizing valves in the fire water systems should be electrically supervised (with indication in the main control room) or administratively controlled.	C.6.c (2)	C	
149. All valves in the fire protection system should be periodically checked to verify position.	C.6.c (2)	WA C	See Note 2 Comply. Subsection 9.5.1.8.6 addresses this requirement.

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
150. Fixed water extinguishing systems should conform to requirements of NFPA 13 and NFPA 15.	C.6.c (3)	AC	Automatic sprinkler systems are designed and installed in accordance with the criteria specified in NFPA 13, with the exception of providing individual fire department connections to each sprinkler system. Because the sprinkler systems are supplied by the plant's fire protection water supply, individual connections are not necessary. See Note 3 and also Table 9.5.1-3.
151. Recommendations for interior manual hose installations.	C.6.c (4)	C	
152. Individual standpipes should be at least 4 inches in diameter for multiple hose connections and 2-1/2 inches in diameter for single hose connections.	C.6.c (4)	C	
153. Standpipe and hose station installations should follow the requirements of NFPA 14.	C.6.c (4)	AC	Standpipes for each building are designed and installed in accordance with the criteria specified in NFPA 14 for Class III service except: (1) the water supply to the standpipe inside containment is manually operated, which is an exception to this standard, and (2) the containment isolation valves controlling the water supply to standpipes inside containment are not listed by independent testing laboratories for fire protection service, which is an equivalency to this standard. These deviations will not adversely affect the performance of the hose station and standpipe system. See Note 3 and also Table 9.5.1-3.
154. Hose stations should be located as dictated by the fire hazard analysis to facilitate access and use for firefighting operations.	C.6.c (4)	C	
155. Recommendations concerning seismic design of standpipes and hose connections.	C.6.c (4)	C	
156. Recommendations concerning hose nozzle selection.	C.6.c (5)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
157. The fire hose should be hydrostatically tested in accordance with NFPA 1962. Hoses stored in outside hose houses should be tested annually. The interior standpipe hose should be tested every 3 years.	C.6.c (6)	WA C	See Note 2 Comply. Subsection 9.5.1.8.6 addresses this requirement.
158. Consideration of foam suppression systems for flammable liquid fires.	C.6.c (7)	NA	Foam suppression systems are not used on AP1000.
Halon Suppression Systems			
159. Design and testing considerations for Halon fire suppression systems.	C.6.d	NA	Fixed Halon fire suppression systems are not used on AP1000.
Carbon Dioxide Suppression Systems			
160. Carbon dioxide suppression systems should comply with the requirements of NFPA 12.	C.6.e	NA	Fixed carbon dioxide suppression systems are not used on AP1000.
161. Automatic carbon dioxide systems should be equipped with a predischarge alarm system and a discharge delay to permit personnel egress.	C.6.e	NA	Fixed carbon dioxide suppression systems are not used on AP1000.
162. Provisions for locally disarming automatic carbon dioxide systems should be key locked and under administrative control. Disarming of systems should be controlled as described in Position C.2.	C.6.e	NA	Fixed carbon dioxide suppression systems are not used on AP1000.
163. Considerations for design of carbon dioxide suppression systems.	C.6.e	NA	Fixed carbon dioxide suppression systems are not used on AP1000.
Portable Extinguishers			
164. Fire extinguishers should be provided in areas that contain, or could present a fire exposure hazard to, safety-related equipment in accordance with NFPA 10.	C.6.f	C	See Note 3
165. Dry chemical extinguishers should be installed with due consideration given to possible adverse effects on safety-related equipment.	C.6.f	C	
Primary and Secondary Containment			
166. Fire protection for the primary and secondary containment areas should be provided for hazards identified by the fire protection analysis.	C.7.a (1)	C	Fires are identified and fire suppression systems are provided accordingly.
167. Because of the general inaccessibility of primary containment during normal plant operation, protection should be provided by automatic fixed systems.	C.7.a (1)	AC	No automatic suppression systems are needed due to the sealless motor reactor coolant pumps (RCPs) having no external lube oil system.
168. Operation of the fire protection systems should not compromise the integrity of the containment or other safety-related systems.	C.7.a(1)(a)	C	

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BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
169. Recommendations for protection of safety-related cables and equipment inside non-inerted containments.	C.7.a(1)(b)	AC	See Appendix 9A for a description of protection inside containment.
170. Recommendations concerning fire detection inside the primary containment.	C.7.a(1)©	C	
171. Standpipe and hose stations inside containment may be connected to a high quality water supply of sufficient quantity and pressure other than the fire main loop if plant-specific features prevent extending the fire main supply inside containment.	C.7.a(1)(d)	C	
172. Recommendations for reactor coolant pump oil collection systems in non-inerted containments.	C.7.a(1)(e)	NA	The reactor coolant pumps are sealless motor pumps and do not require an oil collection system.
173. For secondary containment areas, cable fire hazards that could affect safety should be protected as described in Position C.5.e.(2).	C.7.a (1)(f)	NA	
174. Self-contained breathing apparatus should be provided near the containment entrances for firefighting and damage control personnel. These units should be independent of any breathing apparatus provided for general plant activities.	C.7.a (2)	WA C	See Note 2 Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
Main Control Room Complex			
175. The main control room complex should be separated from other areas of the plant by 3-hour rated fire barriers.	C.7.b	C	
176. Recommendations concerning peripheral rooms in the main control room complex.	C.7.b	NC	The MCR/operator work area wall is not fire-rated based on other design criteria. Manual fire suppression is provided for peripheral rooms. If the MCR must be evacuated, the remote shutdown workstation is provided as an alternate to the MCR.See Appendix 9A.
177. Recommendations concerning the use of Halon and carbon dioxide flooding systems.	C.7.b	NA	No Halon or carbon dioxide flooding systems are used on AP1000.
178. Recommendations concerning manual firefighting capability in the main control room.	C.7.b	C	
179. Recommendations concerning fire detection in the main control room.	C.7.b	NC	Smoke detectors are not provided in cabinets and consoles. The control room is continuously occupied so that a fire is promptly detected and extinguished.

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Table 9.5.1-1 (Sheet 26 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[1]	Remarks
180. Breathing apparatus for main control room operators should be readily available.	C.7.b	WA C	See Note 2 Comply. Subsection 6.4.2.3 addresses this requirement.
181. Recommendations concerning main control room ventilation.	C.7.b	C	
182. Cables that enter the main control room should terminate in the main control room.	C.7.b	C	
183. Cables in underfloor and ceiling spaces should meet the separation criteria necessary for fire protection.	C.7.b	C	
184. Air-handling functions should be ducted separately from cable runs in such spaces.	C.7.b	NC	The underfloor space is used as a distribution plenum for ventilation of the main control room. Smoke detectors in the underfloor space cause prompt closure of combination fire/smoke dampers to shut off air flow.
185. Fully enclosed electrical raceways located in underfloor and ceiling spaces, if over 1 ft^2 in cross-sectional area, should have automatic fire suppression inside.	C.7.b	NA	AP1000 does not have enclosed raceways in the control complex with a cross-sectional area greater than 1 ft^2.
186. Recommendations concerning automatic fire suppression in underfloor and ceiling spaces.	C.7.b	NC	Manual fire suppression is to be used for underfloor and ceiling spaces. The control room is continuously occupied so that a fire is promptly detected and extinguished.
187. There should be no carpeting in the control room.	C.7.b	AC	Carpeting will be installed in the MCR for sound abatement as allowed per the guidance of Revision 1 of RG 1.189. Per the guidelines, the carpeting will be tested to standards such as ASTM D2859, “Standard Test Method for Flammability of Finished Textile Floor Covering Materials,” to establish the flammability characteristics of the material. These characteristics are addressed in the fire hazards analysis.
Cable Spreading Room			
188. Design guidelines for the cable spreading room.	C.7.c	NA	There is no cable spreading room in AP1000.

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Table 9.5.1-1 (Sheet 27 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
189. Recommendations concerning fire protection for computers performing safety-related functions.	C.7.d	C	The data display and processing system does not perform safety-related functions.
190. Nonsafety-related computers outside the control room should be separated from safety-related areas by 3-hour rated fire barriers and should be protected as needed to prevent damage to safety-related equipment.	C.7.d	C	
Switchgear Rooms			
191. Switchgear rooms containing safety-related equipment should be separated from the remainder of the plant by 3-hour rated fire barriers. Redundant switchgear safety divisions should be separated from each other by 3-hour rated fire barriers.	C.7.e	C	The electrical equipment and penetration rooms associated with each safety-related division are separated from the rooms associated with other divisions and from the remaining areas of the plant by 3-hour rated fire barriers.
192. Automatic fire detectors should alarm locally and alarm and annunciate in the main control room.	C.7.e	C	
193. Fire hose stations and portable fire extinguishers should be readily available outside the switchgear rooms.	C.7.e	C	
194. Drains should be provided to prevent water accumulation from damaging safety-related equipment.	C.7.e	C	
195. Remote manually actuated ventilation should be provided for venting smoke when manual fire suppression effort is needed.	C.7.e	C	See Subsection 9.4.1 for a description of smoke removal capability for Class 1E equipment rooms.
Remote Safety-Related Panels			
196. Recommendations concerning separation and electrical isolation of remote safety-related panels.	C.7.f	C	
197. The general area housing remote safety-related panels should be provided with automatic fire detectors that alarm locally and alarm and annunciate in the main control room. Combustible materials should be controlled and limited to those required for operation. Portable extinguishers and manual hose stations should be readily available in the general area.	C.7.f	C	
Safety-Related Battery Rooms			
198. Safety-related battery rooms should be separated from each other and other areas of the plant by 3-hour rated fire barriers.	C.7.g	C	Safety-related battery rooms are separated from associated electrical rooms of the same division by 1-hour rated fire barriers.

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Table 9.5.1-1 (Sheet 28 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
199. Dc switchgear and inverters should not be located in safety-related battery rooms.	C.7.g	C	
200. Automatic fire detection should be provided to alarm locally and annunciate in the main control room.	C.7.g	C	
201. Ventilation systems in the battery rooms should be capable of maintaining the hydrogen concentration below 2 percent.	C.7.g	C	
202. Main loss of ventilation should be alarmed in the main control room.	C.7.g	C	
203. Portable extinguishers and manual hose stations should be readily available outside the battery rooms.	C.7.g	C	
Turbine Building			
204. The turbine building should be separated from adjacent structures containing safety-related equipment by 3-hour rated fire barriers.	C.7.h	C	
205. The fire barriers should be designed to maintain structural integrity in the event of collapse of the turbine structure.	C.7.h	C	
206. Openings and penetrations in the fire barrier should be minimized and should not be located where the turbine oil system or generator hydrogen cooling system creates a fire exposure hazard to the barrier.	C.7.h	C	
Diesel Generator Areas			
207. Diesel generators should be separated from each other and from other areas of the plant by 3-hour rated fire barriers.	C.7.i	AC	The standby diesel generators are separated from each other by a 3-hour rated fire barrier and are housed in a separate structure, remote from safety-related areas. The ancillary diesel generators are separated from other areas of the plant by 3-hour rated fire barriers. The ancillary diesel generators are not separated from each other, but can be easily replaced with transportable diesel generators.
208. Automatic fire suppression should be installed to combat diesel generator or lubricating oil fires. Such systems should be designed for operation when the diesel is running without affecting the diesel.	C.7.i	C	

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Table 9.5.1-1 (Sheet 29 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
209. Automatic fire detection should be provided to alarm locally and annunciate in the main control room.	C.7.i	C	
210. Portable extinguishers and manual hose stations should be readily available outside the area.	C.7.i	C	
211. Drainage for firefighting water and means for local manual venting of smoke should be provided.	C.7.i	C	
Diesel Fuel Oil Storage Areas			
212. Day tanks with total capacity up to 1100 gallons are permitted in the diesel generator area under specified conditions.	C.7.i	NC	Each DG day tank has a total capacity of up to 2000 gallons. Separate 3-hour enclosures and automatic suppression are provided. Tanks are located more than 50 feet from buildings containing safety-related equipment.
213. The day tank should be located in a separate enclosure with a 3-hour fire rating.	C.7.i	AC	The fuel supply for the ancillary diesel generators is not separated from the diesels by a barrier. The ancillary diesels and the tank are separated from the rest of the plant by an enclosure with a 3-hour fire rating.
214. The day tank enclosure should be capable of containing the entire contents of the tank.	C.7.i	C	
215. The day tank enclosure should be protected by an automatic fire suppression system.	C.7.i	C	
216. Recommendations concerning diesel fuel oil tanks.	C.7.j	C	
217. Above-ground tanks should be protected by an automatic fire suppression system.	C.7.j	AC	The diesel fuel oil storage tanks are well-separated from each other and from safety-related structures. Automatic fire suppression systems are not provided.
Safety Related Pumps			
218. Design guidelines for safety-related pump rooms.	C.7.k	NA	There are no active ASME Section III safety-related pumps on AP1000.
New Fuel Area			
219. Recommendations for fire protection of the new fuel area.	C.7.l	C	
Cooling Towers			
Spent Fuel Pool Areas			
220. Protection should be provided by hose stations and portable extinguishers.	C.7.m	C	

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Table 9.5.1-1 (Sheet 30 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
221. Automatic fire detection should be provided to alarm locally and annunciate in the main control room.	C.7.m	C	
Radwaste and Decontamination Areas			
222. Fire barriers, automatic fire suppression and detection, and ventilation controls should be provided.	C.7.n	C	Automatic fire suppression is provided for specific areas in accordance with the fire protection analysis.
Safety-Related Water Tanks			
223. Fire protection provisions for safety-related water tanks.	C.7.o	C	
Records Storage Areas			
224. Records storage areas should be so located and protected that a fire in these areas does not expose safety-related systems or equipment.	C.7.p	C	
225. Cooling towers should be of noncombustible construction or so located and protected that a fire will not adversely affect any safety-related systems or equipment.	C.7.q	WA C	The cooling tower configuration is site-specific. See Note 2 Comply. Subsection 9A.3.3 addresses this requirement.
226. Cooling towers should be of noncombustible construction when the basins are used for the ultimate heat sink or for the fire protection water supply.	C.7.q	NA	The cooling tower basin is not used for an ultimate heat sink or as a source of fire water.
Miscellaneous Areas			
227. Location and protection of miscellaneous areas.	C.7.r	C	
Storage of Acetylene-Oxygen Fuel Gases			
228. Gas cylinder storage locations should not be in areas that contain or expose safety-related equipment or the fire protection systems that serve those safety-related areas.	C.8.a	WA C	See Note 2 Comply. Subsection 9.5.1.8.4.g addresses this requirement.
229. A permit system should be required to use this equipment in safety-related areas of the plant.	C.8.a	WA C	See Note 2 Comply. Subsection 9.5.1.8.4.g addresses this requirement.
Storage Areas for Ion Exchange Resins			
230. Unused ion exchange resins should not be stored in areas that contain or expose safety-related equipment.	C.8.b	WA C	See Note 2 Comply. Subsection 9.5.1.8.4.a addresses this requirement.
Hazardous Chemicals			
231. Hazardous chemicals should not be stored in areas that contain or expose safety-related equipment.	C.8.c	WA C	See Note 2 Comply. Subsection 9.5.1.8.4.h addresses this requirement.

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Table 9.5.1-1 (Sheet 31 of 31)
AP1000 Fire Protection Program Compliance With BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp[(1)]	Remarks
Materials Containing Radioactivity			
232. Materials that collect and contain radioactivity should be stored in closed metal tanks or containers that are located in areas free from ignition sources or combustibles.	C.8.d	C	
233. These materials should be protected from exposure to fires in adjacent areas.	C.8.d	C	
234. Consideration should be given to requirements for removal of decay heat from entrained radioactive materials.	C.8.d	C	
235. Reference Code Years	References	AC	More recent editions of the codes are acceptable upon approved technical documentation that demonstrates equivalency of the alternative standard, system, method, or device.

Notes:

1. Compliance with NUREG-0800 Subsection 9.5.1, Branch Technical Position CMEB 9.5-1 is indicated by the following codes:
   - WA - Will Address: Per Subsection 9.5.1.9.3.
   - C - Compliance: AP1000 is committed to compliance with the guideline.
   - AC - Alternate Compliance: compliance with the guideline by alternate means or intent. Alternative means or design are provided in the remarks column.
   - N/A - Not Applicable: The guideline is not applicable to AP1000.
   - NC - Not in Compliance: AP1000 is not in compliance (explanations in the remarks column.)
2. Procedures and administrative controls governing the fire protection program during plant operation, as well as responsibilities and organizational details for personnel involved in fire protection activities, are discussed in subsection 9.5.1.8.
3. It is intended to fully comply with NFPA standards referenced in Subsection 9.5.5, as they apply to AP1000. However, due to conflicting design considerations, there may be a need to take exception to specific guidance. Known exceptions to the design portions of the NFPA standards are identified in Table 9.5.1-3.

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Table 9.5.1-2
Component Data - Fire Protection System
(Nominal Values)

Fire Water Storage Tanks	
Primary Fire Water Tank	
Nominal capacity (gal)	504,000
Volume dedicated to fire protection (gal)	396,000
Design Pressure	Atmospheric
Material	Carbon steel
Secondary Fire Water Tank	
Nominal capacity (gal)	504,000
Volume dedicated to fire protection (gal)	396,000
Design Pressure	Atmospheric
Material	Carbon steel
Fire Pumps	
Motor-Driven	
Pump type	Horizontal centrifugal
Rated flow (gpm)	2000
Required head, approximate (ft)	350
Structural material	Cast iron
Diesel-Driven	
Pump type	Horizontal centrifugal
Rated flow (gpm)	2000
Required head, approximate (ft)	350
Structural material	Cast iron
Fuel tank capacity (min. gal)	385
Motor-Driven Jockey Pump	
Pump type	Centrifugal
Rated flow (gpm)	30
Required head, approximate (ft)	210
Structural material	Cast iron
Containment Spray Nozzles	
Type	Lechler (SPRACO) 1713A
Number	68
Rated flow (gpm)	15.2
Rated pressure design (psi)	40
Structural material	Stainless steel

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Table 9.5.1-3 (Sheet 1 of 4)
Exceptions to NFPA Standard Requirements

Requirement	AP1000 Exception and Justification
NFPA 13 Sections 5-14.1.1.2 and 5-15.2 require fire department connections to individual sprinkler system headers, with no intervening shutoff valves.	Individual connections are not provided. Sprinkler systems are supplied from the proprietary fire water supply system, which can be accessed by the fire department at any hydrant along the yard main. Valves between these connection points and the sprinkler systems are electrically supervised or locked open.
NFPA 14 Section 3-4.3 does not allow manual standpipes for Class III systems.	The water supply to the standpipe inside containment is manually operated. This exception was approved with the NRC certification of the AP1000 design.
NFPA 14 Section 3-5 prohibits use of dry standpipes for Class II or Class III systems, and in areas not subject to freezing.	The standpipe system inside containment is classified as a dry standpipe system because it is normally isolated by the outboard containment isolation valve as described in Subsection 9.5.1.2.1.5.
NFPA 14 Section 4-2.2 requires an isolation valve for each standpipe.	One valve is used to isolate two or more short standpipes that supply a small number of hose stations.
NFPA 14 Sections 4-3 and 5-12 require fire department connections for each standpipe system, with no intervening shutoff valves.	Individual connections are not provided. Standpipe systems are supplied from the proprietary fire water supply system, which can be accessed by the fire department at any hydrant along the yard main. Valves between these connection points and the standpipe systems are electrically supervised or locked open, except as described in Subsection 9.5.1.2.1.5.
NFPA 14 Section 5-3.2 requires Class I hose connections at each intermediate landing of exit stairways, on each side of horizontal exit openings, in each exit passageway, and on the roof or at the highest landing of stairways.	Class I hose connections are provided in exit stairways at one intermediate landing between most floors, and at other protected exit locations accessible to firefighters entering the buildings from outside. Flow testing of Class I hose connections is accomplished without providing additional connections on the roofs of buildings or at the highest stairway landings.
NFPA 14 Section 5-5 requires standpipes to be interconnected at the bottom, and when supplied by elevated tanks, also at the top.	Standpipes interconnections are constrained by layout considerations and do not always meet these requirements. Each standpipe receives an adequate water supply at an adequate pressure.
NFPA 14 Section 5-11.2 requires a separate drain connection for each standpipe.	For standpipes located outside radiologically controlled areas and supplied at an elevation above the lowest hose connection, the hose connection is used to provide a means of draining the standpipe.
NFPA 804 (Reference 2) Section 2.4 (6) requires the AP1000 fire hazards analysis to analyze the potential effects of a fire on life safety, release of contamination, impairment of operations, and property loss, assuming the operation of installed fire extinguishing equipment.	The AP1000 fire protection analysis does not analyze the potential effects of fire on life safety, impairment of operations, and property loss. This exception was approved with the NRC certification of the AP1000 design.

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Table 9.5.1-3 (Sheet 2 of 4)
Exceptions to NFPA Standard Requirements

Requirement	AP1000 Exception and Justification
NFPA 804 (Reference 2) Section 2.4 (8) requires the AP1000 fire hazards analysis to analyze the potential effects of an uncontained fire in causing other problems not related to safe shut-down, such as a release of contamination and impairment of operations.	The AP1000 fire protection analysis does not analyze the potential effects of an uncontained fire in causing other problems not related to safe shut-down. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 2.4 (9) requires the AP1000 fire hazards analysis to analyze the postfire recovery potential.	The AP1000 fire protection analysis does not provide an analysis of the postfire recovery potential. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 3.3.4.4 (2) requires the ignition of leaked or spilled liquid be minimized by using suitable electrical equipment.	The AP1000 fire protection design criteria document does not explicitly require the minimization of the ignition of leaked or spilled liquid by the use of suitable electrical equipment. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 3.3.4.4 (3) requires the ignition of leaked or spilled liquid be minimized by sealing the insulation of hot plant components to prevent liquid saturation.	The AP1000 fire protection design criteria document does not explicitly require the minimization of the ignition of leaked or spilled liquid by sealing the insulation of hot plant components to prevent liquid saturation. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 5.2.2.2 requires the AP1000 safe shutdown analysis to have a shutdown logic diagram.	The AP1000 fire protection analysis does not include a shutdown logic diagram. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 5.3.2.1 requires the preparation of a risk assessment that demonstrates the potential risk from a seismically induced fire in relationship to the plant's core damage frequency.	The AP1000 design has not prepared a seismically induced fire PRA. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 6.1.2.2 requires stairwells and elevator shafts be enclosed with barriers having a fire resistance rating of at least 2 hours.	Auxiliary building stairwells and elevator shafts, with exception of the exterior wall sections enclosing Stairwell S03 and the wall separating Stairwell S03 from the elevator shaft above the Auxiliary Building Roof, are enclosed in towers constructed using both concrete structural walls and nonstructural walls as described in Section 9A of the DCD. For buildings outside of the Nuclear Island, exterior wall sections enclosing stairwells and elevator shafts do not have a fire resistance rating of 2 hours, only interior stairwell and elevator shaft wall sections are rated for the minimum 2 hour fire resistance rating.
NFPA 804 (Reference 2) Section 6.2.1 requires NFPA 101, *Life Safety Code*, be utilized throughout for the life safety from fire in the AP1000 design and operation.	Deviations from the Life Safety Code's exit and egress requirements are identified and the degree of the life safety justified in the Means of Egress Studies.

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Table 9.5.1-3 (Sheet 3 of 4)
Exceptions to NFPA Standard Requirements

Requirement	AP1000 Exception and Justification
NFPA 804 (Reference 2) Section 6.5.3 (5) requires the drainage and any associated drainage facilities for a given area be sized to accommodate the volume of liquid based on a manual fire-fighting flow of 500 gal/min (1892.5 L/min) for a duration of 30 minutes, unless the fire hazards analysis demonstrates a different flow rate and duration.	The AP1000 drains are not designed for 500 gpm, nor does the present fire hazards analysis demonstrate a different flow rate and duration. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 6.8.6.1 Thin-wall metallic tubing shall not be used.	EMT (thin-wall metallic tubing / conduit) is used in limited designs for lighting and communications systems as well as "limited" commodity packages that are not Safety Related and do not have a "Functional Impact" to the plant design.
NFPA 804 (Reference 2) Section 8.3.5 requires smoke detectors be provided in the control room complex, the electrical cabinets, and consoles.	Smoke detectors are not provided in the cabinets and consoles. The control room is continuously occupied so that a fire is promptly detected and extinguished.
NFPA 804 (Reference 2) Section 8.3.10 requires the main control room under-floor and above-ceiling spaces to have air-handling functions ducted separately from cable runs, enclosed raceways over 1 ft^2 in cross-sectional area to have automatic fire suppression provided within them, and area automatic fire suppression for exposed cable runs.	The under-floor space of the control room complex is used as a distribution plenum for ventilation of the main control room, smoke detectors in the under-floor space cause prompt closure of combination fire/smoke dampers to shut off air flow, and automatic fire suppression is not provided in cable runs in the underfloor and ceiling spaces of the control room complex. These exceptions were approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 8.4.2.2.1 requires cable tunnels to be provided with automatic fixed suppression systems.	The AP1000 does not provide cable tunnels with automatic fixed suppression systems. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 8.4.3 requires cable tray fire breaks be installed every 20 feet in vertical cable trays that rise over 30 feet, and the provision of automatic sprinkler protection and smoke detection at the ceiling of vertical cable shafts.	Cable shafts and risers are not installed with periodic fire breaks, and automatic sprinkler and smoke detection is not provided at the ceiling of these shafts. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 8.5 requires computer and communication rooms to meet the applicable requirements of NFPA 75, *Standard for the Protection of Electronic Computer/Data Processing Equipment*.	The AP1000 is not explicitly designed to meet this requirement. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 8.8.6 requires turbine generator exciter housings to be protected either internally or externally with an automatic fire suppression system depending on the design of the housing ventilation system.	The AP1000 is not explicitly designed to provide automatic fire suppression for the turbine generator exciter housing or for the entire area it is within. This exception was approved with the NRC certification of the AP1000 design.
NFPA 804 (Reference 2) Section 8.10.2 requires aboveground diesel fuel storage tanks to be provided with automatic fire suppression systems.	Aboveground diesel fuel storage tanks are not provided with automatic fire suppression systems. This exception was approved with the NRC certification of the AP1000 design.

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Table 9.5.1-3 (Sheet 4 of 4)
Exceptions to NFPA Standard Requirements

Requirement	AP1000 Exception and Justification
NFPA 804 (Reference 2) Section 8.28 requires the intake structure to be provided with automatic sprinkler protection.	Exception/Justification is site-specific.

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Table 9.5.1-4
Capabilities Used to Achieve Cold Shutdown Following a Fire

Function	System Capability	Fire Protection
RCS Reactivity Control – Short Term – Long Term	 – Control Rods – (1)	 – separation (6) – (1)
RCS Makeup	– (1)	– (1)
RCS Pressure Control – Increase – Decrease	 – Pressurizer heaters – (2)	 – fire suppression – (2)
Decay Heat Removal (high temperature)	– SFW pumps feeding CST water to SG – SG PORV discharge to atm.	– fire suppression – separation
Decay Heat Removal (cold temperature)	– RNS pumps circulating RCS – CCS cooling RNS – SWS cooling CCS	– separation – fire suppression – fire suppression
Decay Heat Removal (cold temperature alternate)	– RNS pump circulating RCS – FPS cooling RNS (7)	– separation – fire suppression
Process Monitoring	– RCS monitoring instruments (PMS) – Non-1E Instrumentation and Control (3)	– separation – fire suppression or separation
Support Systems	– Instrument Air – Standby Diesel Generators – Non-1E AC Power and Control (3)	– fire suppression – fire suppression and separation – fire suppression or separation

Notes:

(1) CVS makeup from the BAT provides RCS makeup and boration. Automatic suppression is provided in the CVS makeup pump room and in the Non-Class 1E equipment/penetration room. If the CVS is damaged by a fire, one CMT can provide this capability.

(2) CVS auxiliary spray provides pressurizer pressure reduction. Automatic suppression is provided in the CVS makeup pump room and in the Non-Class 1E equipment/penetration room. If the CVS is damaged by a fire, one ADS stage 1 valve used in a low capacity throttled vent mode of operation can slowly depressurize the RCS without loss of RCS pressure boundary.

(3) The portions of the non-1E AC power and the non-1E instrumentation and control system required are those needed to operate cold shutdown components; local control is sufficient (switchgear/control cabinet).

(4) Portions of the non-1E heating and ventilating systems are required to ventilate the main control room, non-1E switchgear rooms, and the required portions of the non-1E instrumentation and control system (see note 3).

(5) The term "separation" means that fire barriers provide separation of redundant components, including equipment, electrical cables, instrumentation and controls, except in the main control room, remote shutdown room, and containment. See Subsection 9A.3.1.1 for discussion on containment and Subsection 9A.3.1.2.5 for discussions on the main control room and remote shutdown room.

(6) Separation is provided for the reactor trip function. The reactor trip breakers are separated.

(7) Connection is provided to allow the fire protection system to furnish water to cool a normal residual heat removal (RNS) pump and heat exchanger following a fire that disables the normal CCS cooling function.

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Table 9.5.2-1
Communication Equipment(1) and Locations

Provided for Operator Transient Response(2)
in Addition to Main Control Room and Remote Shutdown Room

Area Locator	Elevation	Primary Area/Location Served
1212	66′-6″	Divisions A, B, C, D Battery Rooms
1222	82′-6″	Divisions A, B, C, D dc Equipment Rooms
1121	96′-0″	Passive Core Cooling System Valve/Accumulator Rooms and Steam Generator
1123		Compartments
1124		
1231	100′-0″	Divisions A and C, I&C Rooms; Divisions B and D, I&C/Penetration Rooms; Valve/Piping Penetration Room
1234	107′-2″	Maintenance Floor Staging Area
1134	107′-2″	Maintenance Floor
1132		
1243	117′-6″	Non-1E Equipment/Penetration Room
1254	135′-3″	Access Corridor Serving Personnel Access Area
1151	135′-3″	Operating Deck
1152		
1251	135′-3″	MSIV Compartments
1162	159′-7″	Steam Generator Feedwater Nozzle Area

Notes:
(1) Stations have Telephone/Page, PBX, and maintenance sound-power capability.
(2) See Standard Review Plan 9.5.2.

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Table 9.5.4-1 (Sheet 1 of 2)
Nominal Component Data
Standby Diesel Fuel Oil System

Above Ground Storage Tanks	
Service	Diesel engine supply
Quantity	2
Type	Vertical, cylindrical
Normal fill level / available fuel capacity (gal) per tank	74,000
Available fuel reserved for diesel generator (gal) per tank	69,000
Excess fuel available for testing (gal) per tank	5,000
Operating pressure	Atmospheric
Operating temperature	Ambient
Diesel Oil Transfer System	
Fuel Oil Transfer Pumps	
Quantity	2
Type	Gear, positive displacement
Operating Flow (gpm)	8
Required design capacity (gpm)	35
Fuel Oil Strainer	
Quantity	2
Type	Duplex
Design Capacity (gpm)	35
Fuel Oil Heater	
Quantity	2
Type	Electric
Rating @35 gpm	90 kw
Fuel Oil Water Separator	
Quantity	2
Type	Pressurized/coalesced
Design Capacity (gpm)	35
Duplex Filters	
Quantity	2
Type	Duplex/stacked disc
Design Capacity (gpm)	35
Diesel Fuel Oil Day Tanks	
Quantity	2
Type	Horizontal, cylindrical
Normal available fuel capacity (gal)	1,560
Operating pressure	Atmospheric
Code	ASME VIII

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Table 9.5.4-1 (Sheet 2 of 2)
Nominal Component Data
Standby Diesel Fuel Oil System

Ancillary Diesel Fuel Oil Tank	
Quantity	1
Type	**Vertical, rectangular**
Design Capacity (gal)	**700**
Available capacity (gal)	**550**
Operating pressure	Atmospheric
Code	Non-Stamped ASME VIII

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Table 9.5.4-2
Indicating and Alarm Devices - Standby Diesel Fuel System

Parameter	Indication		Alarm	
	Control Room	Local	Control Room	Local
Fuel Oil Storage Tank Level - Diesel Oil (DO) Transfer	Yes	Yes	Yes	Yes
DO Day Tank Level	Yes	Yes	Yes	No
DO Transfer Pump Motor-Running Indication	Yes	Yes	No	No
DO Low Fuel Oil Pressure	Yes	Yes	Yes	No
DO Water Separator Differential Pressure	Yes	Yes	Yes	No
DO Filter Differential Pressure	Yes	Yes	Yes	No
DO Pump Suction Strainer Differential Pressure	Yes	Yes	Yes	No
DO Fuel Oil Heater in Service	Yes	Yes	Yes	Yes
DO Fuel Oil Heater Temp Out	Yes	Yes	Yes	No
Fuel Oil Tank Fill Strainer Differential Pressure	No	Yes	No	No

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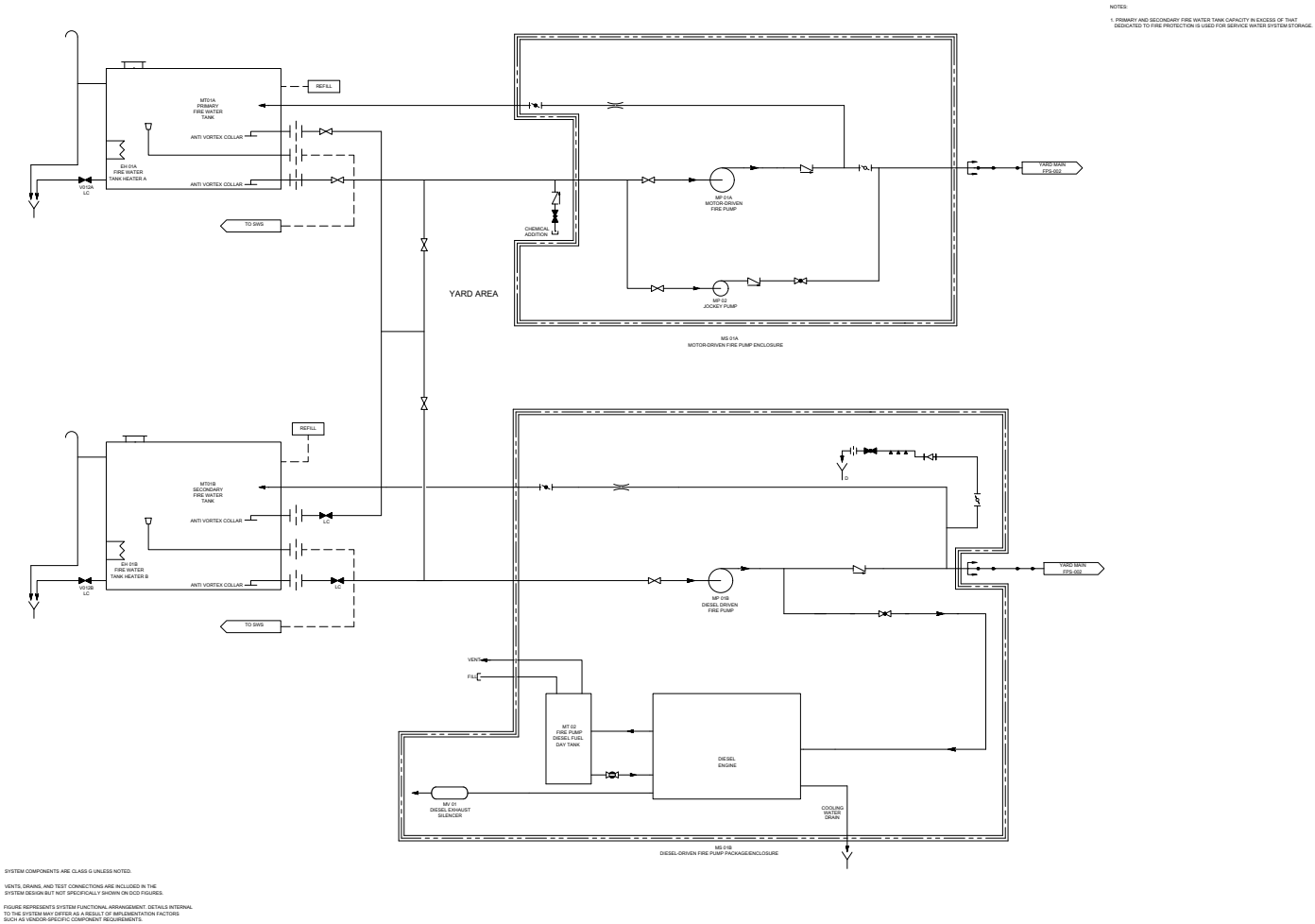

Figure 9.5.1-1 (Sheet 1 of 3)
Simplified Fire Protection System Piping and Instrumentation Diagram (REF FPS 001)

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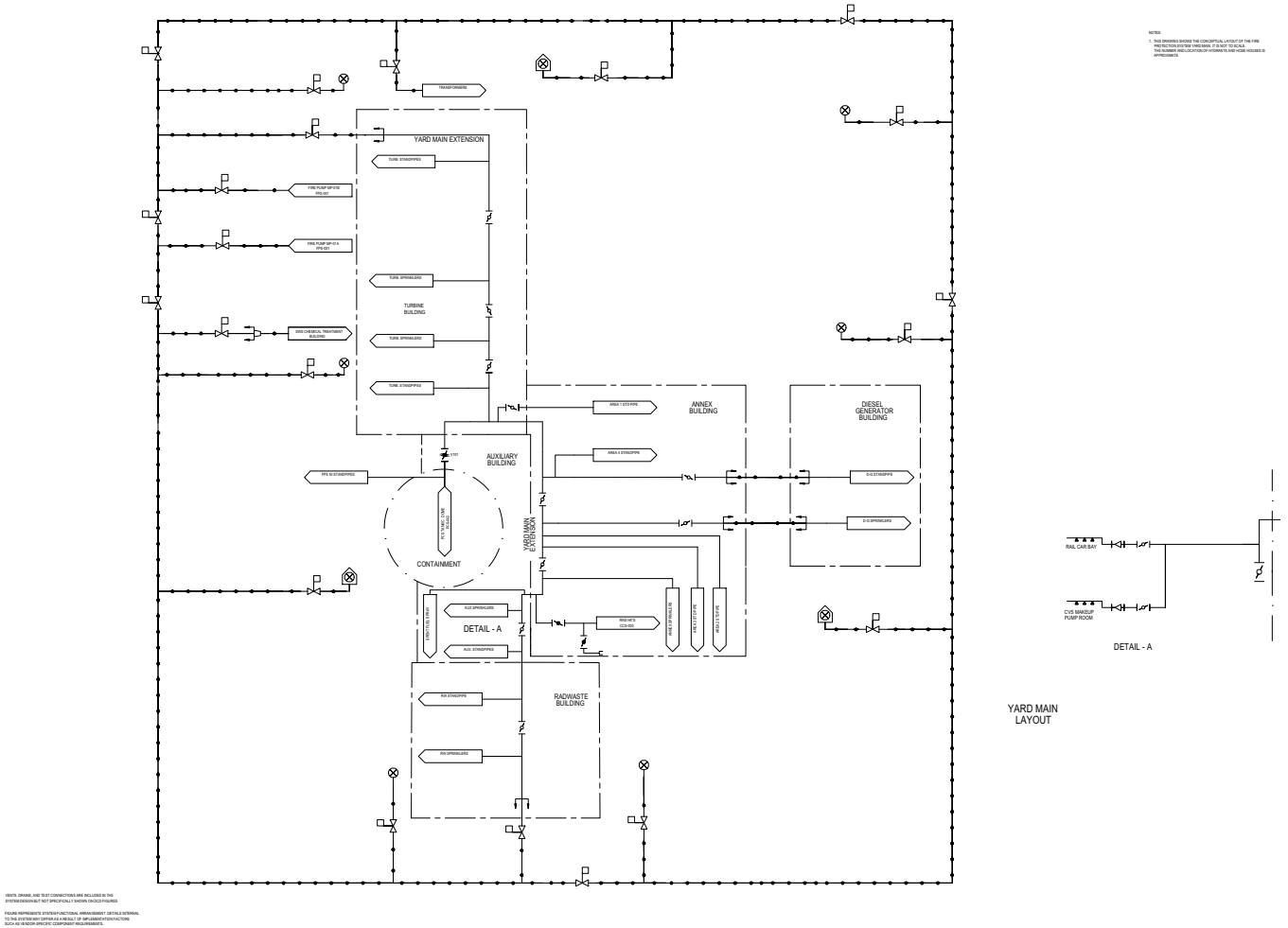

Figure 9.5.1-1 (Sheet 2 of 3)
Simplified Fire Protection System Piping and Instrumentation Diagram (REF) FPS 002

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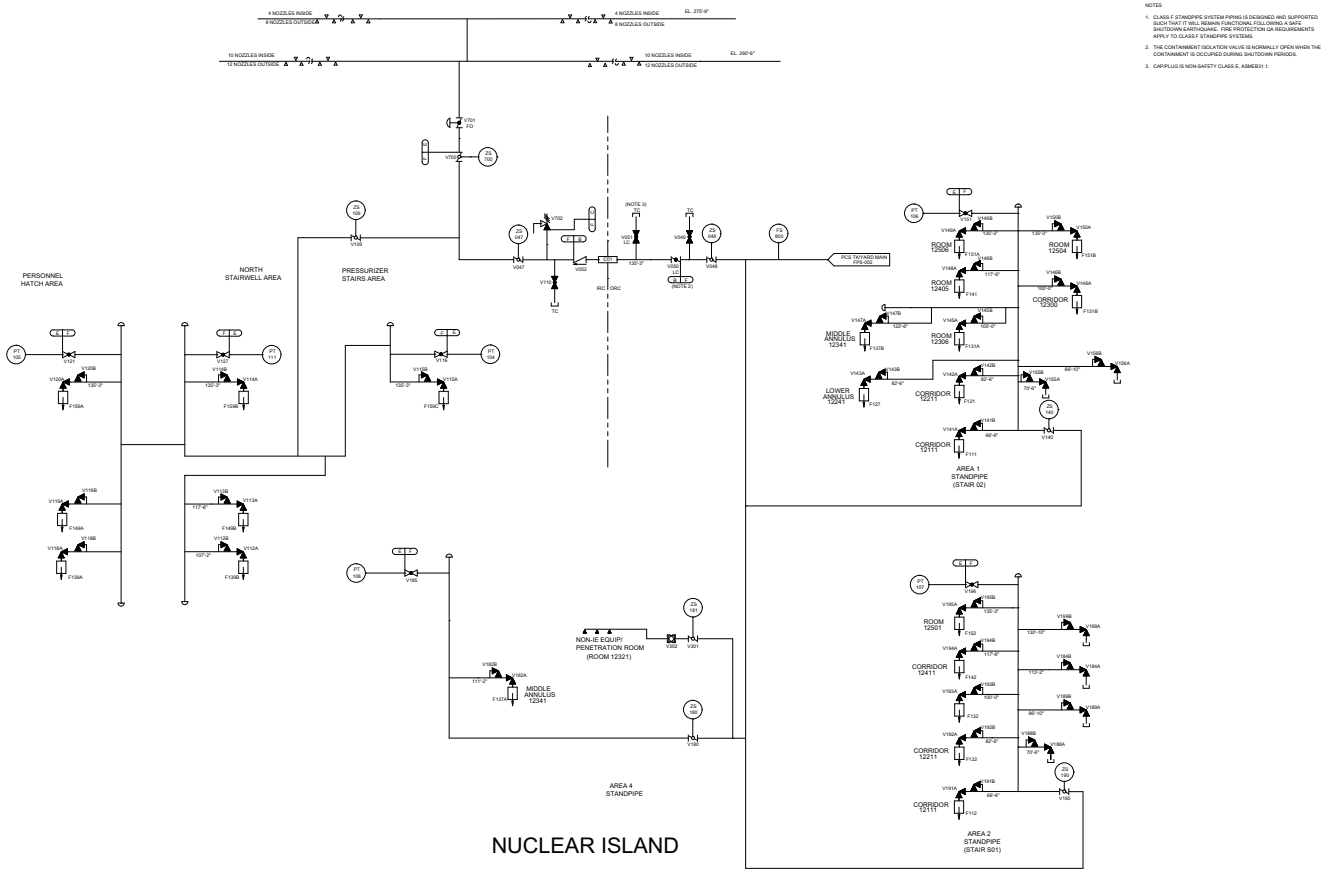

Figure 9.5.1-1 (Sheet 3 of 3)
Simplified Fire Protection System Piping and Instrumentation Diagram (REF FPS 004)

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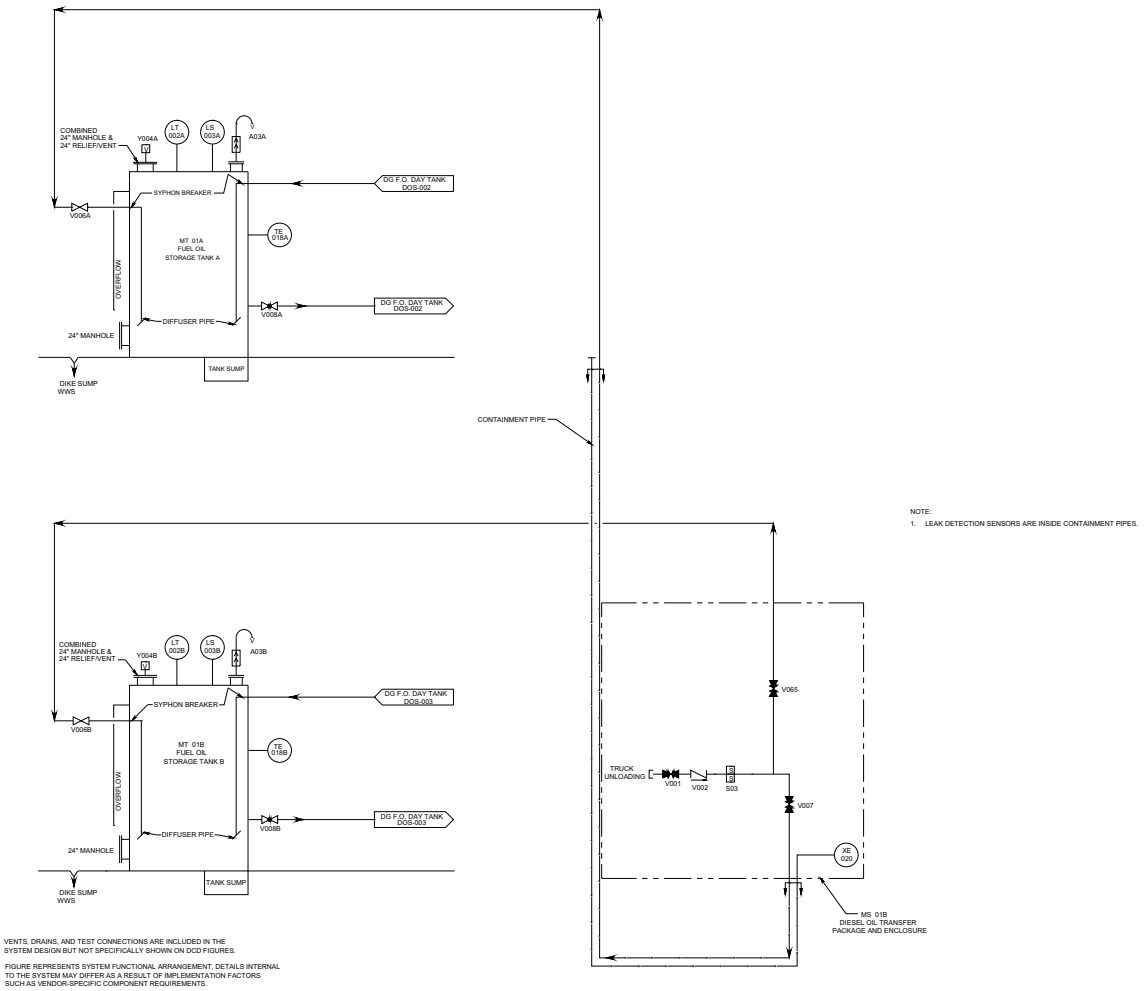

Figure 9.5.4-1 (Sheet 1 of 3)
Simplified Standby Diesel Fuel Oil System
Piping and Instrumentation Diagram
(REF) DOS 001

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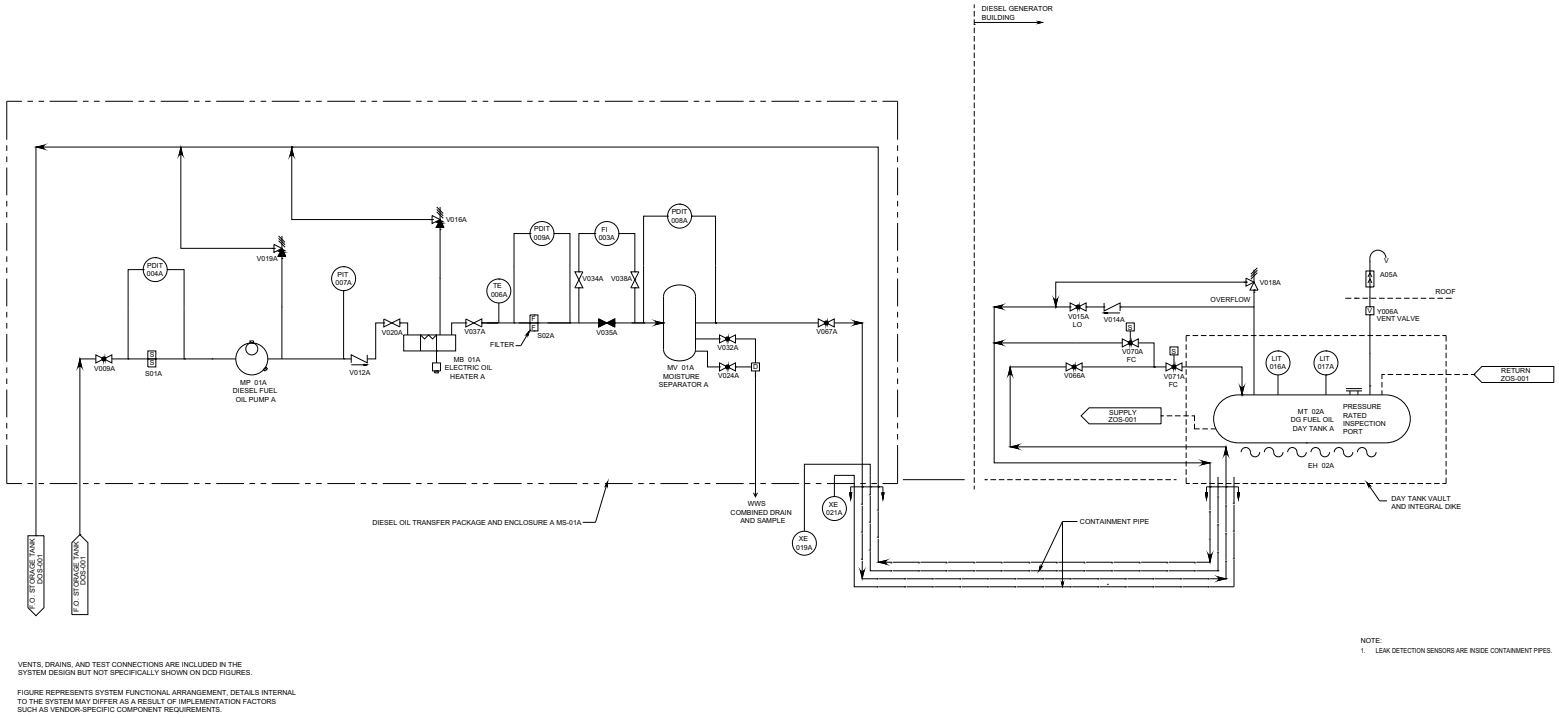

Figure 9.5.4-1 (Sheet 2 of 3)
Simplified Standby Diesel Fuel Oil System
Piping and Instrumentation Diagram
(REF) DOS-002

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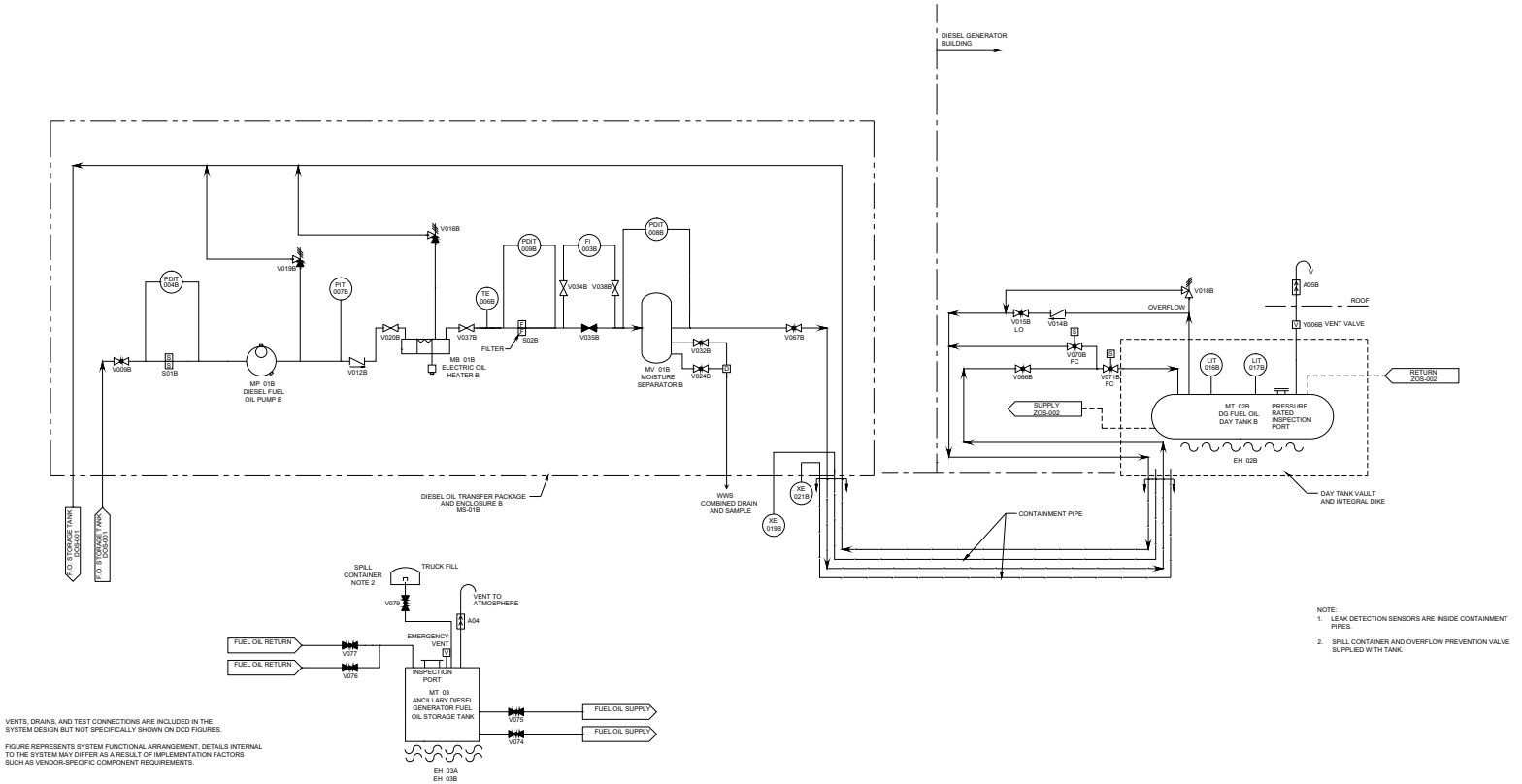

Figure 9.5.4-1 (Sheet 3 of 3)
Simplified Standby Diesel Fuel Oil System Piping and Instrumentation Diagram (REF) DOS-003 and DOS-004

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Appendix 9A Fire Protection Analysis

9A.1 Introduction

The AP1000 fire protection analysis is largely based upon the AP600 fire protection analysis that supported the AP600 Design Certification. It evaluates the potential for occurrence of fires within the plant and documents the capabilities of the fire protection system and the capability to safely shut down the plant. The fire protection analysis is an integral part of the process of selecting fire prevention, detection, and suppression methods, and provides a design basis for the fire protection system. The design of the fire protection system is described in Subsection 9.5.1.

The purpose of the fire protection analysis is as follows:

- Identify the potential for fires based on the type, quantity, and location of combustible materials
- Determine the consequences of postulated fires
- Provide a basis for decisions on how to prevent, detect, contain, and suppress fires
- Assess fire protection system adequacy
- Confirm the capability to safely shut down the plant following a fire

The fire protection analysis is performed for each fire area using the methodology described in Section 9A.2. This methodology follows the guidance of Branch Technical Position (BTP) CMEB 9.5-1 (Reference 1). The results of the analysis are provided in Section 9A.3.

The fire protection analysis is performed for areas of the plant containing safety-related components and for areas containing systems important to the generation of electricity. It is performed on an area by area basis outside containment and a zone by zone basis inside containment. This approach provides confidence that plant safety is preserved.

9A.2 Fire Protection Analysis Methodology

9A.2.1 Fire Area Description

The plant is divided into fire areas and fire zones as described in Section 9.5.1.2.1.1. These fire areas/zones and their boundaries are illustrated on Figures 9A-1 through 9A-5. Figure 9A-201 replaces Figure 9A-3 (Sheet 1), to reflect the relocation of the Operations Support Center.

The analysis for each fire area briefly describes the fire area and associated fire zones, and identifies the principal systems and safety-related components in the fire area. Fire detection and suppression features are listed and the means of smoke control is discussed. The term "smoke" is used throughout this document to imply "smoke and products of combustion".

This document also uses terminology defined in NFPA 13, such as "light hazard", "ordinary hazard", and "extra hazard". Normally, these terms apply to sprinkler installations and their water supplies only. However, as used herein, the terms apply to the quantity and combustibility of the contents of a given fire area or fire zone irrespective of whether or not sprinklers are present.

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9A.2.2 Combustible Material Survey

Each fire area and fire zone is surveyed to determine the type, quantity and distribution of in-situ combustible / flammable materials. Where the presence of transient materials (for example, materials required to support refueling activities or scheduled maintenance) or operational materials (operator provided equipment / components) is reasonably expected these materials are also accounted for within the survey.

When estimating quantities of electrical cable insulation, cable trays are normally assumed to have a cable fill of 40 percent of the usable tray depth. The final estimated percentage of cable tray fill depends on cable thickness and positioning of the cables within the cable tray. Cable enclosed in conduit or in closed metal cabinets is not included in the combustible material survey.

9A.2.3 Fire Severity Categorization

The expected fire severity for each area/zone is categorized from low to severe in accordance with Table 7-9E of the NFPA Fire Protection Handbook, 16th edition (Reference 2).

Fire severity is determined by the area / room layout, combustible / flammable material distribution, types of combustible / flammable material present, occupancy use of the space, and other room-specific factors. If there are significant concentrations of combustible / flammable materials in the area/zone, the category assigned is generally more severe than that used for a uniform distribution to reflect the limited isolated area.

9A.2.4 Combustible Loading and Equivalent Fire Duration Calculations

Fire barriers, detection and suppression methods are based on several factors, including regulatory guidance, the type of combustibles present, and investment protection considerations. Regulatory guidance takes precedent over the other considerations.

Combustible loading and equivalent fire duration calculations are performed for each fire area and each fire zone. Tables 9A-3A to 9A-3F identify the maximum equivalent fire duration for each fire area / zone, including the duration associated with anticipated transient and operational materials.The calculations provide information used in the selection of fire detection and suppression methods. Tables 9A-3A to 9A-3F identify the primary (greater than ~25%) type of combustible / flammable materials for the fire load for the design of the AP1000 Standard Plant in each fire area / zone of the primary plant structures. Fire detection and suppression features are also summarized in Tables 9A-3A to 9A-3F.

Transient and operational fire loads are managed as per the Fire Protection Program (Subsection 9.5.1.8). The Fire Protection Program maintains transient and operational fire loads within acceptable limits (within the allowable margins) of the fire area / zone.

Combustible Loading Calculations

The calculation of combustible loading provides an indication of the maximum heat that is released if all the combustibles in a given fire area/zone are consumed.

The potential heat release (expressed in British thermal units, or Btus) for each type of combustible material in the fire area/zone is a function of the quantity of each combustible and its heat of combustion. A fire load can be derated (as applicable) per the methodology in the Fire Protection Handbook, 16th Edition (Reference 2). The heat of combustion values are derived depending upon materials involved in equipment selection and from industry sources. The maximum heat release for all combustibles in the fire area/zone is found by adding the potential heat release of each combustible material.

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The combustible loading for the fire area/zone is the maximum heat release per square foot (Btus per square foot). It is determined by dividing the maximum heat release for all combustibles in the fire area/zone by the floor area of the fire area or zone.

Equivalent Fire Duration

Fire duration is estimated based on the fire severity category and the equivalent combustible loading. The appropriate fire endurance lines of Figure 7-9B of the NFPA Fire Protection Handbook, 16th edition (Reference 2), are used to estimate the fire duration in minutes.

Fire barriers are tested by exposure to a fire whose severity follows a time varying temperature curve known as the standard time-temperature curve (NFPA 251 [Reference 202], Figure 1). The estimated fire duration for each fire area is normalized based on the standard time-temperature curve to obtain an equivalent fire duration. This value is compared with the fire resistance of the fire area boundaries. This comparison is used in conjunction with other factors, including those listed above, in making a determination of the adequacy of the fire area boundaries.

9A.2.5 Fire Protection Adequacy

The adequacy of the fire protection features for a postulated fire in each fire area or fire zone is evaluated. This evaluation includes the following points:

- A review of the AP600 Design Certification and other regulatory guidance
- A review of how the fire is detected and suppressed
- Verification of the adequacy of the fire resistance of the fire area boundaries
- Verification that the ventilation system for the fire area does not contribute to the spread of fire or smoke
- Verification that a fire in a nonsafety-related area does not threaten safety-related areas of the plant
- Verification that, for a fire in an area containing radioactive materials, the capability to minimize and control a potential release of radioactivity is not adversely affected
- A determination of the need for structural steel fireproofing
- A determination of the capability of the drainage systems to handle fire protection water flow.

9A.2.6 Fire Protection System Integrity

For fire areas containing safety-related components, the potential for a credible inadvertent actuation of automatic suppression systems is determined and the consequences are evaluated.

The design of automatic and manual suppression systems is reviewed to verify that there is no potential single impairment which incapacitates both the automatic suppression system and the manual suppression system.

9A.2.7 Safe Shutdown Evaluation

This subsection describes the methodology for evaluation of the effects of postulated fires in each fire area on the ability of the operator to achieve a safe shutdown of the plant. The criteria and assumptions upon which the evaluation is based are described in Subsection 9A.2.7.1. The

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safety-related features of the plant designed to provide the safe shutdown capability are described in subsection Subsection 9A.2.7.2.

As indicated in Subsection 9.5.1, this evaluation is based upon satisfying the requirements of BTP CMEB 9.5-1. This basis includes using safe shutdown as defined in the Technical Specifications in lieu of cold shutdown wherever stated in BTP CMEB 9.5-1. The automatic depressurization system is not used as the method for achieving safe shutdown after a fire and spurious actuation of the automatic depressurization system is avoided. The passive residual heat removal heat exchanger is used to remove decay heat for safe shutdown as described in Subsection 7.4.1.3.

In addition, the plant has enhanced capability to achieve cold shutdown following a fire as discussed in Subsection 9.5.1. This capability is not relied upon in the fire evaluation contained in Appendix 9A.

The criteria concerning cold shutdown capability deviates from the criteria applied to the evolutionary reactor designs, but is consistent with the criteria applicable to existing plants. To enhance the survivability of the normal safe shutdown and cold shutdown capability in the event of a fire, and to reduce the reliance on the infrequently utilized safety-related passive systems, automatic suppression or physical separation is provided in those fire areas outside containment where a fire could damage the normal shutdown capability. This criterion is unique to the AP1000 and does not ensure that the normal shutdown capability will be free of fire damage, or that the equipment necessary to achieve and maintain cold shutdown can be repaired within 72 hours.

9A.2.7.1 Criteria and Assumptions

The criteria and assumptions described below are used in performing the safe shutdown evaluation.

Postulated Fire

Only one fire is assumed to occur within the plant at any given time. A postulated fire is assumed to occur in any area (or zone in containment), whether or not the area contains in-situ combustible materials.

Any damage which would prevent proper operation of equipment and which the fire is capable of causing is assumed to occur immediately. Except where explicitly noted, no credit is taken for proper operation of equipment or moving of valves to proper position when not protected from the effects of a postulated fire.

Fire Barriers

As described in Subsection 9.5.1.2.1.1, non-combustible fire barriers are provided in accordance with BTP CMEB 9.5-1. The minimum equivalent fire barrier ratings are shown in Figures 9A-1 through 9A-5. Fire-proofing of structural steel is determined as described in Subsections 9.5.1.2.1.1 and 9A.2.5.

Fire Areas

Fire areas are three dimensional spaces designed to contain a fire that may exist within them. They are separated by fire barriers, fire barrier penetration protection, and other devices, such as those within the heating and air conditioning ducts, that isolate a fire to within the fire area.

A postulated fire does not extend beyond the boundary of the fire area. For fire areas outside the main control room, remote shutdown room, and containment fire areas, the zone of influence is defined as the entire fire area and all equipment in any one fire area is assumed to be rendered inoperable by the fire and re-entry into the fire area for repairs and operator actions is assumed to be impossible. For investment protection, an operator may enter an adjacent room to a fire in the same fire area, that is separated by a 1-hour fire barrier, to prevent spurious actuation of the ADS Stage 4

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valves. However, no credit is taken for complete fire damage in cases in which complete damage is beneficial and partial damage is not. Chases for electrical raceway, piping or ducts that pass through the fire area but are separated from it by 3-hour fire barriers are treated as an extension of the fire area boundary that they originate from or travel to. Such chases contain internal 3-hour fire-rated barriers (e.g., fire barrier seals) to separate the connected fire areas.

Zone of Influence

A fire outside containment is assumed to affect its entire fire area. Inside the containment fire area, the zone of influence is defined as the entire fire zone containing the fire. All equipment in any one fire zone is assumed to be rendered inoperable by the fire unless the fire protection analysis demonstrates otherwise. Class 1E electrical cables that are located in or pass through the fire zone but are separated from it by a 3-hour fire barrier are outside that fire zone.

Fire Zones

Fire zones are three dimensional spaces within fire areas. Fire zones are identified uniquely to indicate that they have fire protection features or attributes different than other fire zones in a given area. In containment, fire zones are identified to establish "zones of influence."

Independence of Affected Fire Areas

Only systems, components, and circuits free of fire damage are credited for achieving safe shutdown for a given fire. Systems, components, and circuits outside the zone of influence are considered free of fire damage if the effects of the fire do not prevent them from performing their required safe shutdown functions.

Event Assumptions

Plant accidents and severe natural phenomena are not assumed to occur concurrently with a postulated fire. Furthermore, a concurrent single active component failure (independent of the fire) is not assumed.

Offsite Power

A loss of offsite power is assumed concurrent with the postulated fire only when the safe shutdown evaluation indicates the fire could initiate the loss of offsite power.

Availability of Nonsafety-Related Systems

Only safety-related components and systems are assumed to be available to perform safe shutdown functions. (This is more stringent than required by BTP CMEB 9.5-1.) For each fire area or zone, the safe shutdown evaluation is valid for the worst case fire in the area or zone and initial use of nonsafety-related equipment. Fire protection and smoke control systems are assumed to function as designed to detect and mitigate the effects of the fire.

If offsite power is available, nonsafety-related systems are assumed to continue to operate if a more conservative evaluation would result. Each safe shutdown evaluation is also valid considering the possibility that the operator may initiate safe shutdown using available nonsafety-related systems and that, should the fire later cause those systems to fail, safety-related systems may be automatically or manually actuated to continue the safe shutdown process.

Automatic Suppression Features Assessment

An assessment is performed to demonstrate the ability of the AP1000 to withstand a fire in fire areas outside containment, and achieve safe shutdown without the need for actuating the passive safety-related decay heat removal system. This evaluates the capability of the AP1000 non-safety-related systems to achieve safe shutdown.

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Fire suppression is provided outside containment in locations that would degrade the normal nonsafety-related systems used to achieve safe shutdown following a fire, such that the operation of the passive residual heat removal heat exchanger would be required to provide shutdown decay heat removal. Fire suppression minimizes the challenges to the safety-related decay heat removal systems by enhancing the survivability of nonsafety-related systems used for shutdown decay heat removal.

The safe shutdown process, using safety and nonsafety-related systems, is described in Subsection 7.4.1.2. This assessment credits the use of selected safety-related systems other than the passive residual heat removal system to facilitate the transition to cold shutdown conditions. The following safety-related features may be used:

- Insertion of the control rods to provide reactor shutdown,
- Operation of the core makeup tanks to provide boration and reactor coolant system makeup,
- Manual throttling and closing of a first stage automatic depressurization valve to reduce the reactor coolant system pressure to the operating pressure of the normal residual heat removal system,
- Instrumentation used to monitor reactor coolant system conditions.

The use of these safety-related systems do not result in significant plant transients. If the automatic depressurization system is actuated, the operators align the normal residual heat removal system to provide injection to the reactor coolant system. This action causes the core makeup tank level to remain above the fourth stage valve actuation setpoint and prevents significant steaming to and flooding of the containment.

Process Monitoring

Direct process signals are provided to monitor the shutdown process and to assist in determining proper actions for operation of the shutdown methods.

Manual Operation

One of the required manual actions to achieve plant shutdown for a postulated fire event is to scram the reactor.

Manual actions by operations personnel include manipulation of equipment located anywhere outside the affected fire area, if accessibility and staffing levels permit such actions. Entry into the fire area for repairs or operator actions is assumed to be impossible.

Although the typical shutdown sequence does not require manual actions by the operator, fire damage may not be sufficient in many cases to trip the plant. The operator may take appropriate actions to expedite an orderly shutdown. These actions are performed in the main control room. If the fire occurs in the main control room, these actions are performed at the remote shutdown workstation.

High-Low Pressure Interfaces

NRC Generic Letter 81-12 (Reference 3) requests the identification and evaluation of the interfaces between the high pressure reactor coolant system and low pressure systems such as the normal residual heat removal system. Typically, these high-low pressure interfaces contain two redundant and independent remotely-operated valves in series. These two valves and their control and power cables may be subject to a single fire. This fire may potentially cause the two valves to open, resulting in a fire-initiated loss-of-coolant accident (LOCA) through the high-low pressure system

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interface. Electrically controlled valves which provide such an interface are identified. These interface valves are considered to be required for safe shutdown.

Associated Circuits

The AP1000 was designed with separation of safety-related circuits and equipment as a primary objective. As a result of the concern for separation from the beginning of the design, the use of associated circuits has been minimized.

The safe shutdown equipment and systems for each fire area are isolated from associated circuits in the fire area so that hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment. No postulated fire involving associated circuits will prevent safe shutdown.

Associated circuits comply with Regulatory Guide 1.75 position 4 related to associated circuits and IEEE 384-1981 (Section 5.5.2).

Spurious Actuation of Equipment

Fire-caused damage is assumed to be capable of resulting in the following types of circuit faults: hot shorts, open circuits, and shorts to ground. Spurious actuation of components caused by these circuit faults are evaluated. Components are assumed to be energized or de-energized by one or more of the above circuit faults. For example, air operated and solenoid operated valves are assumed to fail open or closed; pumps are assumed to fail running or not running; electrical distribution breakers could fail open or closed. For three-phase ac circuits, the probability of getting a hot short on all three phases in the proper sequence to cause spurious operation of a motor is considered sufficiently low as to not require evaluation, except for cases involving high-low pressure interfaces. For ungrounded dc circuits, if spurious operation could only occur as a result of two ungrounded hot shorts of the proper polarity, then no further evaluation is necessary, except for any cases involving a high-low pressure interface. Therefore, spurious operation of ac or dc motor operated valves as a result of power cable hot shorts is not assumed, except for cases involving a high-low pressure interface.

It is assumed that a fire results in the loss of all automatic function (signals and logic) from the circuits located in the fire area. In addition, an assessment of multiple/simultaneous spurious actuations or signals resulting from the fire was performed (Reference 6) with the conclusion that either spurious actuations do not occur, or the consequences are such that safe shutdown can still be achieved. The spurious actuations and signals that are evaluated are those that could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

Spurious actuation of the redundant valves in any one high-low pressure interface line are postulated if the circuits for those valves are located in the fire area.

Most control room controls use soft-controls which communicate over multiplexed data channels. Fire-induced spurious actuation from these multiplexed soft controls is not assumed.

Spurious actuation from control room dedicated switches which could lead to a breach of reactor coolant system pressure boundary, loss of decay heat removal function, or loss of shutdown reactivity control is prevented by the use of dual two-pole, energize-to-actuate, ungrounded dc circuits, which require at least four simultaneous hot shorts of proper polarity for spurious actuation. In the event of a fire in the main control room, control may be transferred to the remote shutdown workstation, depending on the extent of the fire. For a small fire which can be quickly extinguished, control is maintained in the main control room, and the potential for damage or spurious signals is limited. For larger postulated fires, the main control room is evacuated and control is transferred to

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the remote shutdown workstation. Once control is transferred, the dedicated switches in the main control room are disabled by a transfer switch.

Spurious actuation of squib valves is prevented by the use of a squib valve controller circuit which requires multiple hot shorts for actuation, physical separation of potential hot short locations, and provisions for operator action to remove power from the fire zone. No postulated fire can spread to the hot short locations before the operator can remove power from the fire zone.

Automatic depressurization system stages 1, 2, and 3 consist of parallel paths, each path having two motor-operated valves in series. Spurious stage 1, 2, or 3 actuation is prevented by the use of physical separation of control circuits for the two series valves and provisions for operator action to remove power from the fire zone. No postulated fire can spread to the hot short locations before the operator can remove power from the fire zone.

Multiple High-Impedance Faults

It is postulated that fire-induced circuit faults may occur with high enough impedance to prevent tripping of the affected circuit breaker. If multiple high-impedance faults occur simultaneously, affecting branch circuits fed from a common power source, there is a potential for the sum of the currents from these multiple high-impedance faults to be high enough to trip the main circuit breaker feeding the bus. Once the main breaker trips, components powered from the bus lose their power source. Multiple high-impedance faults are considered in the evaluation of safe shutdown capability.

Plant Personnel

The plant operating staff available for manual actions to achieve safe shutdown, during and after the fire, is limited to the minimum number of posted operator positions minus those assigned to the fire brigade.

Smoke Removal

Where portable equipment is used to remove smoke after a fire, the exhausted smoke is released directly outside to an area that will not affect safety-related plant areas, unless the release is from an area potentially containing radioactive materials. Where portable equipment is used to remove smoke from areas potentially containing radioactive materials, the exhausted smoke is also monitored for radiological releases in accordance with guidance provided in BTP CMEB 9.5-1, paragraph C.5.f.(2).

Equipment Environment

The environment of the equipment required to function for shutdown should not become so severe as to prevent the equipment from functioning. If the environment does exceed the conditions for which the equipment is capable of functioning, it is assumed that the equipment no longer is capable of performing its intended function.

Emergency Lighting

In situations where the safe shutdown evaluation identifies the need for manual operator action in response to a fire, the estimate of the time required for this action considers the availability of emergency lighting in locations where these actions are performed and along the access and egress routes thereto.

Emergency Communications

The safe shutdown evaluations consider the need for and availability of emergency communications within the plant following a fire.

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9A.2.7.2 Safe Shutdown Methodology

The safe shutdown process, the systems used, and the functional requirements for safe shutdown are described in Section 7.4. As noted above, only safety-related equipment is utilized for safe shutdown. A description of this equipment is provided in the applicable sections.

Table 9A-2 lists the safety-related components used for safe shutdown and their associated electrical divisions. Each fire area is reviewed to identify the potential scope of fire damage and to verify that the capability to achieve and maintain safe shutdown is preserved.

The shutdown process uses controls located in the main control room. In the event of a fire in the main control room, controls located at the remote shutdown workstation are used.

9A.3 Fire Protection Analysis Results

The fire protection analysis is conducted for the following primary plant structures, which are shown on the site plot plan, Figure 1.2-2:

- Nuclear island
- Turbine building
- Annex building
- Radwaste building
- Diesel generator building

Openings through fire barriers for pipe, conduit, and cable trays are sealed or closed to provide a fire resistance rating at least equal to that of the fire barrier itself. Penetration designs conform to the guidelines of BTP CMEB 9.5-1. Fire barrier penetration openings for ventilation are protected by fire dampers having a rating equivalent to that of the fire barrier. For 1-hour rated fire barriers, fire dampers are not required since the duct itself is an adequate barrier. The protection of door openings conforms to the guidelines of BTP CMEB 9.5-1.

Structural steel fireproofing is provided as described in Subsection 9.5.1.2.1.1.

The fire detection and suppression capabilities for each fire area are selected based on the criteria described in Subsection 9A.2.1 and are consistent with the importance of the equipment in the fire area to plant availability. Portable fire extinguishers are accessible throughout the plant.

The presence of radioactive systems is noted in the description of each fire area. Potential releases of radioactivity as a consequence of a fire in these areas is mitigated by measures such as:

- Control and confinement of sources of radioactivity per ALARA principles
- Use of fire dampers to isolate ventilation ducts serving the fire area
- Use of fire suppression systems to quickly suppress the fire
- Provision of curbed floor areas and sizing of sumps to collect and retain fire protection water within the affected fire area or building

The safe shutdown evaluation of spurious equipment actuation as a result of a fire is addressed in Subsection 9A.3.7. The protection of accident mitigation equipment (as opposed to safe shutdown equipment) is also addressed in Subsection 9A.3.7.

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9A.3.1 Nuclear Island

Figure 9A-1 identifies fire areas and fire zones within the nuclear island and illustrates the fire resistance of the fire area boundaries. The nuclear island is comprised of the following primary areas:

- Containment/shield building
- Auxiliary building – nonradiologically controlled areas (non-RCA)
- Auxiliary building – radiologically controlled areas (RCA)

The containment/shield building comprises a single fire area for the purposes of this analysis.

The auxiliary building is divided into the radiologically controlled areas and nonradiologically controlled areas which are physically separated by structural walls and floor slabs. These structural barriers are designed to prevent fire propagation across the boundary between these areas.

The auxiliary building is further subdivided into fire areas separated by fire-rated structural barriers. These barriers provide physical separation between the four Class 1E electrical divisions and between these divisions and nonsafety-related areas.

Floor drains accommodate water flow from fire protection systems without a significant accumulation of water in a fire area. Flooding of components required for safe shutdown is also precluded by the fact that only a limited volume of fire water can be discharged from the fire protection system in fire areas containing those components. This subject is further discussed in Section 3.4.

Drain systems in the radiological controlled area of the nuclear island Annex Building and Radwaste Building drain to fire zones in the nuclear island where there are no safe shutdown components. Fires in these zones due to potential combustible liquid transport by the drains do not affect safe shutdown.

There is no drain path which could drain combustible liquids to the fire areas in the electrical portion of the nuclear island.

For mechanical equipment fire areas in the nonradioactive auxiliary building, fires caused by potential transport of combustible liquid through the drain system are included in the fire hazards analysis.

9A.3.1.1 Containment/Shield Building

This building comprises one fire area - 1000 AF 01. This fire area is separated into fire zones and includes the spaces inside containment as well as the valve room for the passive containment cooling system (PCS), the middle annulus, the upper annulus, and the operating deck staging area outside containment. This fire area excludes the middle annulus penetration room division B, C, and D (rooms 12344, 12343, and 12345) because these are treated as extensions of the adjacent auxiliary building rooms' fire areas. This fire area includes the remainder of the middle annulus including the division A containment penetration area.

The fire protection and the safe shutdown analysis for the containment identifies the location and the separation of the safe shutdown components located inside the containment. The safe shutdown components located inside the containment are primarily components of the passive core cooling system (PXS), the reactor coolant system (RCS), the steam generator system (SGS), and containment isolation.

For this evaluation, the containment shield building is divided into the following fire zones. These zones are based on the establishment of boundaries (structures or distance) that inhibit fire propagation from zone to zone. Complete fire barrier separation cannot be provided inside containment because of the need to maintain the free exchange of gases for purposes such as

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passive containment cooling. The location of safety-related equipment and the routing of Class 1E electrical cable in each fire zone enhances the separation of redundant safe shutdown components.

Fire Zone

- 1100 AF 11105 Reactor cavity
- 1100 AF 11204 Vertical access and reactor coolant drain tank room
- 1100 AF 11206 Accumulator room A
- 1100 AF 11207 Accumulator room B
- 1100 AF 11208 Normal residual heat removal valve room
- 1100 AF 11209 Chemical and volume control system room
- 1100 AF 11300A Maintenance floor (southeast quadrant access)
- 1100 AF 11300B Maintenance floor (north half)
- 1100 AF 11301 Steam generator compartment 1
- 1100 AF 11302 Steam generator compartment 2
- 1100 AF 11303 Pressurizer compartment
- 1100 AF 11303A Automatic depressurization system lower valve area
- 1100 AF 11303B Automatic depressurization system upper valve area
- 1100 AF 11500 Operating deck
- 1200 AF 12341 Middle annulus
- 1200 AF 12541 Upper annulus / PCS Air Diffuser
- 1250 AF 12555 Main control room emergency habitability system air storage/operating deck staging area
- 1270 AF 12701 Passive containment cooling system valve room

The equipment and components in this fire area contain radioactive material with the exception of passive containment cooling system and main control room emergency habitability system components.

Fire Detection and Suppression Features

- Fire detectors
  - All fire zones except fire zone 1200 AF 12341
  - Only spot heat detection (Heat) is provided in fire zone 1200 AF 12341 for PCS pressure transmitters as localized equipment protection
- Hose station(s)
- Portable fire extinguishers (during reactor shutdown for maintenance)

Smoke Control Features

Containment air filtration system (VFS) containment isolation valves, if open to the containment atmosphere, are closed by operator action to control the spread of fire and smoke. After the fire, smoke is removed from the fire area by portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by operation of a fire detector, which produces an audible alarm locally, except in containment where the fire detection system produces a visual alarm locally, and

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both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire durations in this fire area are listed in Table 9A-3A. Concentrations of combustibles are described in the evaluation of each fire zone. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

The consequences of a break in a fire protection line during normal plant operation are limited because the containment isolation valve for the fire water supply line to the containment hose stations is normally closed and are bounded by other flooding events inside containment. See Section 3.4 for further discussion of flooding events inside containment.

9A.3.1.1.1 Fire Zone 1100 AF 11105

This fire zone is comprised of the following room(s):

Room No.	
11105	Reactor vessel cavity
11205	Reactor vessel nozzle area

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. This fire zone is separated from adjacent fire zones by the thick concrete walls and floor of the reactor vessel cavity, except at the top of the fire zone, where there are penetrations associated with reactor coolant system piping and where the annular space around the reactor vessel flange is closed by the permanent cavity seal ring. There is a doorway to the reactor coolant drain tank room (fire zone 1100 AF 11204) that is closed and a ventilation duct that provides cool air from the containment recirculation cooling system.

Smoke and hot gases from a fire accumulate within this fire zone and gradually migrate via reactor coolant system piping penetrations to adjacent fire zones 1100 AF 11204, 1100 AF 11206, 1100 AF 11301, and 1100 AF 11302. The smoke and hot gases are expected to rise due to their buoyancy and be replaced by air coming from the containment recirculation cooling system. They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 identifies the safe shutdown components located in this fire zone. They are the four excore flux instrumentation channels, one for each division. Although it is unlikely that all of the components would be damaged, a fire in this fire zone is conservatively assumed to disable all of the

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above instrumentation. The source, intermediate and power range excore detectors are not required for automatic safe shutdown initiation or maintenance during or following a fire in this fire zone. These detectors are used to monitor and verify that the reactor is shut down. The redundant instrumentation used for monitoring core reactivity indirectly are the core exit thermocouples located in fire zone 1100 AF 11500. These thermocouples are mounted within the reactor and integrated head package and have exposed cable high in the integrated head package in fire zone 1100 AF 11500. The thermocouple cables will be unaffected by smoke from a fire in the reactor cavity. In addition, reactor subcriticality after shutdown is maintained by an adequate boron concentration in the reactor coolant. This concentration is establish by the automatic actions taken upon reactor trip such as, isolation of non-borated makeup sources and opening of the flow paths to sources of borated water. Boron concentrations can be checked periodically to determine if adequate levels exist.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.2 Fire Zone 1100 AF 11204

This fire zone is comprised of the following room(s):

Room No.	
11104	Reactor coolant drain tank room
11204	Vertical access area

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. Cable raceways are located against one structural concrete wall of the fire zone. The floor of this fire zone is solid concrete at the bottom of containment. Thick concrete walls separate this fire zone from adjacent fire zones, except for access passageways to and from the steam generator compartments (fire zones 1100 AF 11301/11302). Steel grating and the vertical access stairway form the boundary between this fire zone and the maintenance floor above (fire zone 1100 AF 11300B). There is a doorway between the reactor coolant drain tank room and the bottom of the reactor cavity (fire zone 1100 AF 11105) that is closed.

Smoke and hot gases from a fire in this fire zone rise through the grating at the top of the vertical access area and spread through the large maintenance floor air space (fire zones 1100 AF 11300A and B). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components contained in this fire zone. Although it is unlikely that all of the components would be damaged, a fire in this zone is conservatively assumed to disable the reactor coolant system hot leg instrumentation. The redundant reactor coolant system hot leg instrumentation located in 1100 AF 11301 and 1100 AF 11302 are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

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No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.3 Fire Zone 1100 AF 11206

This fire zone is comprised of the following room(s):

Room No.	
11206	Passive core cooling system valve/accumulator room A

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are no significant concentrations of combustible materials. This fire zone is physically separated from other fire zones by walls, floor and ceiling with minimum concrete thicknesses of one foot, except for a closed personnel hatch, an open equipment hatch, and a small CMT pipe penetration in the ceiling. The penetration is beneath core makeup tank A, located on the maintenance floor (fire zone 1100 AF 11300A).

Smoke and hot gases from a fire in this fire zone rise through the CMT pipe penetration and normally open equipment hatch and spread through the large maintenance floor air space (fire zones 1100 AF 11300A and B). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components contained in this fire zone. A fire in this fire zone is conservatively assumed to disable control of all of the valves and instrumentation in this fire zone. The passive core cooling system safe shutdown components located in fire zones 1100 AF 11207 and 1100 AF 11300B are redundant to those in this fire zone, and are sufficient to perform applicable functions to achieve and maintain safe shutdown. The spent fuel pool cooling system containment isolation valve located outside the containment fire area is redundant to the containment isolation valve inside containment in this fire zone and is sufficient to maintain containment integrity.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.4 Fire Zone 1100 AF 11207

This fire zone is comprised of the following room(s):

Room No.	
11207	Passive core cooling system valve/accumulator room B

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Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are no significant concentrations of combustible materials. This fire zone is physically separated from other fire zones by walls, floor and ceiling with concrete thicknesses of more than one foot, except for a closed personnel hatch, a closed equipment hatch, and a CMT pipe penetration in the ceiling, and a passageway to the adjacent RNS valve room (fire zone 1100 AF 11208). The large accumulator vessel stands in front of this passageway and provides a barrier to fire propagation between the two fire zones. The ceiling blockout is beneath core makeup tank B, located on the maintenance floor (fire zone 1100 AF 11300B). The physical arrangement of the small penetration and the large tank and its support results in a tortuous path for fire propagation between these two fire zones. A fire is not expected to propagate to fire zone 1100 AF 11208. If it did, however, fire zone 1100 AF 11206 provides redundant safe shutdown equipment.

Smoke and hot gases from a fire in this fire zone rise through the ceiling CMT pipe penetration and spread through the large maintenance floor air space (fire zones 1100 AF 11300A and B). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components contained in this fire zone. Although it is unlikely that more than one valve would be damaged, a fire in this fire zone is conservatively assumed to disable control of all of the valves in this fire zone. The passive core cooling system safe shutdown components located in fire zone 1100 AF 11206 and 1100 AF 11300A are redundant to those in this fire zone, and are sufficient to perform applicable functions to achieve and maintain safe shutdown.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.5 Fire Zone 1100 AF 11208

This fire zone is comprised of the following room(s):

Room No.	
11208	Normal residual heat removal valve room

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are no significant concentrations of combustible materials. This small fire zone is physically separated from other fire zones by walls, floor and ceiling with concrete thicknesses of more than one foot, except for a closed equipment hatch and a passageway to the adjacent PXS valve/accumulator room (fire zone 1100 AF 11207). The large accumulator vessel

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stands in front of this passageway and provides a barrier to fire propagation between the two fire zones. If fire were to propagate to fire zone 1100 AF 11207, however, fire zone 1100 AF 11206 provides redundant safe shutdown equipment.

Smoke and hot gases from a fire in this fire zone migrate into the adjacent PXS valve/accumulator room (fire zone 1100 AF 11207). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components located in this zone. Although it is unlikely that more than one valve would be damaged, a fire in this fire zone is conservatively assumed to disable control of all of the valves in this fire zone. During normal power operation, power to the hot leg suction isolation valves is locked out to protect the high-low pressure interface between the reactor coolant system and the normal residual heat removal such that they will be unaffected by the fire in maintaining the reactor coolant pressure boundary. The normal residual heat removal containment isolation valve, located outside the containment fire area, is redundant to the four containment isolation valves in this zone and is sufficient to maintain containment and reactor coolant pressure boundary integrity.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.6 Fire Zone 1100 AF 11209

This fire zone is comprised of the following room(s):

Room No.	
11209	Chemical and volume control system room

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire in this fire zone does not propagate to the extent that it damages safe shutdown components outside this fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are no significant concentrations of combustible materials. This fire zone is physically separated from other fire zones by walls, floor and ceiling with concrete thicknesses of more than one foot, except for an access stairway, hatches and an opening from the maintenance floor above (fire zone 1100 AF 11300B).

Smoke and hot gases from a fire in this fire zone rise through the normally open pipe chase access hatch and access staircase and spread through the large maintenance floor air space (fire zones 1100 AF 11300A and B). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

There are no safe shutdown components in this fire zone. No further evaluation is required.

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No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.7 Fire Zone 1100 AF 11300A

This fire zone is comprised of the following room(s):

Room No.	
11300	Maintenance floor (southern part)
11400	Maintenance floor mezzanine (southern part)

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are concentrations of cables at the top of the zone and at several separate locations along the walls. This fire zone is physically separated from fire zones below by the maintenance floor, with a concrete thickness of more than one foot, except for openings described in the evaluation of fire zone 1100 AF 11206. This fire zone is separated from the operating deck above (fire zone 1100 AF 11500) by a ceiling with a concrete thickness of more than one foot, except for the hatches near the containment maintenance hatch, which are covered with steel grating. The walls of this fire zone are the steel containment vessel, the steel wall of the in-containment refueling water storage tank, or walls with a concrete thickness of more than one foot, except for two designated boundaries with the adjacent portion of the maintenance floor (fire zone 1100 AF 11300B). These boundaries are approximately at the centerline of containment, one located in the narrow annular space behind the in-containment refueling water storage tank and the other near the personnel hatch. The steam generator compartments, the refueling cavity, and the in-containment refueling water storage tank provide barriers between the two large maintenance floor fire zones. Safe shutdown components fire zone 1100 AF 11300A are separated from redundant safe shutdown components in fire zone 1100 AF 11300B by these barriers or by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards. In addition, safety-related cables in both of these fire zones are routed in closed cable trays or conduit, minimizing the likelihood that a fire originating in a raceway of one division can propagate to a raceway of another division. Furthermore, passive fire stops are provided for nonsafety-related electrical cables routed in open cable trays in fire zone 1100 AF 11300B (there are no such cable trays in fire zone 1100 AF 11300A), providing additional assurance that a fire will not propagate between these fire zones.

Most of the smoke and hot gases from a fire in this fire zone rises through the large steel grating covered hatches between the containment maintenance hatch and the steam generator 2 compartment into the large air space in the upper portion of containment (fire zone 1100 AF 11500). Small quantities of smoke, especially that which has already cooled, may migrate horizontally into the adjacent portion of the maintenance floor (fire zone 1100 AF 11300B). The smoke and gases are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Temperature effects on the electrical cables routed high above the operating deck and passing over the large steel-grating covered hatches are not expected to be significant, but are not a concern as these are the same cables that continue into this fire zone and are assumed to be lost. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

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Table 9A-2 lists the safe shutdown components located in this fire zone. The passive core cooling system has two IRWST gutter isolation valves located in this zone. These valves close to divert condensate from the passive containment cooling system (on the inside of the containment shell) into the IRWST. This condensate maintains the passive residual heat removal heat exchanger heat sink for the long term. These valves are fail closed air operated valves. They are located at least 12 feet apart horizontally and at least 10 feet apart vertically. One valve is located on the south end of the refueling cavity, the other is located on the east side of the refueling cavity. In addition, a fire detector is located close to each valve. Given the low combustible materials in this fire zone, a fire will only affect one of the valves initially. The fire detector located near the valve that is initially affected will alert the operators so that they can actuate the unaffected valve before the fire can prevent operation of the second valve. These valves are qualified to operate with elevated temperatures of 340°F.

Although the consequences of a fire are expected to be very limited, a fire in this fire zone is conservatively assumed to eventually disable all of the safe shutdown components in this fire zone.

The redundant passive core cooling system and steam generator system safe shutdown components (listed in Table 9A-2), located in fire zones 1100 AF 11207 and 1100 AF 11300B, are sufficient to perform applicable functions to achieve and maintain safe shutdown.

The primary sampling system and containment air filtration system containment isolation valves, located outside the containment fire area, are redundant to the containment isolation valves in this fire zone and are sufficient to maintain containment integrity.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.8 Fire Zone 1100 AF 11300B

This fire zone is comprised of the following room(s):

Room No.	
11300	Maintenance floor (northern part)
11400	Maintenance floor mezzanine (northern part)
11403	Pressurizer Spray Valve Room

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There is a concentration of cables on the south side of the zone near the refueling cavity and concentrations of cables at the top of the zone and at several locations along the walls. This fire zone is physically separated from fire zones below by the maintenance floor, which has a concrete thickness of more than one foot, except for openings described in the evaluation of fire zones 1100 AF 11204, 1100 AF 11207, 1100 AF 11208, and 1100 AF 11209. This fire zone is separated from the operating deck above (fire zone 1100 AF 11500) by a ceiling that has a concrete thickness of more than one foot, except for several openings for an access stairway, elevator, hatches and blockouts. The walls of this fire zone are the steel containment vessel, the steel wall of the in-containment refueling water storage tank, the noncombustible enclosure for the division B and

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D penetrations and raceways (fire zone 1100 AF 11500), or walls with a concrete thickness of more than one foot, except for the designated boundaries with the adjacent portion of the maintenance floor, described in the evaluation of fire zone 1100 AF 11300A. There is a doorway between the lower pressurizer compartment (fire zone 1100 AF 11303) and the steam generator #1 lower manway platform (fire zone 1100 AF 11301).

Safety-related cables are routed in closed cable trays or conduit. For open cable trays, which represent the only significant in-situ combustibles in fire zone 1100 AF 11300B, passive fire stops are provided to prevent postulated fires in these open cable trays from propagating to an adjacent fire zone.

Smoke and hot gases from a fire in this fire zone rise through openings in the ceiling, or migrate via the large steel grating covered hatches between the containment maintenance hatch and the steam generator 2 compartment in the adjacent portions of the maintenance floor (fire zone 1100 AF 11300A), into the large air space in the upper portion of containment. They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components located in this fire zone. The division A and C electrical penetrations listed in Table 9A-2 are conservatively assumed to be disabled as a result of a fire in this fire zone. The B and D electrical penetrations and their cable trays routed from the electrical penetrations up to the operating deck are functionally part of fire zone 1100 AF 11500. These two divisions are sufficient to perform applicable functions to achieve and maintain safe shutdown as described in Subsection 9A.2.7.

These division B and D electrical penetrations and their associated raceways are protected from a fire in this fire zone by a combination of barriers, distance and passive fire stops. Noncombustible barriers of steel or steel-composite construction form vertical shaft(s) from the floor up to the operating deck, surrounding the division B and D penetrations and the associated cable trays. The significant combustible materials in this fire zone are the nonsafety-related cables routed in open cable trays. These cable trays are located at least 20 feet from the division B and D penetrations and their associated raceways, and they are provided with passive fire stops to prevent postulated fires in these open cable trays from propagating to an adjacent fire zone.

The passive core cooling system has two passive residual heat removal heat exchanger control valves which are located in this fire zone. These valves are fail-open air-operated valves. They are located within several feet of each other. The valves are separated from each other by a noncombustible barrier of steel or steel composite materials. One of the valves is located close to the IRWST wall. This valve is assigned to division A and C. The other valve is assigned to division B and D. The cables for this valve are enclosed in conduit or enclosed raceways. Separate fire detectors are provided near each valve. The only combustibles in the area are the valves themselves and their cables. A fire that would affect these valves would be expected to start at one of the valves. The barrier protects the other valve from the initial effects of the fire. The fire detectors would alert the operators and allow them to actuate the other valve before the fire could spread and damage the adjacent valve. These valves are qualified to operate with elevated temperatures of 340°F. The power cables for these air-operated valves may be disabled because of a fire in this fire zone, which will place one or both valves into their passive open position for safe shutdown.

The power cables for the PXS Core Make-up Tanks (CMT) Discharge Isolation Valves, located in fire zones 1100 AF 11206 and 1100 AF 11207 are in this fire zone and may be disabled because of a fire in this fire zone. The PXS Core Make-up Tanks (CMT) are safety-related air-operated valves with a passive open position for safe shutdown. A fire that disables one or both power cables for the valves will place the valves into their passive open position for safe shutdown.

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Reactor coolant system, and steam generator system instrumentation located in this fire zone are conservatively assumed to be disabled as a result of a fire in this fire zone. The redundant passive core cooling system instrumentation, reactor coolant system pressurizer, and steam generator system instrumentation located in fire zones 1100 AF 11206, 1100 AF 11300A, 1100 AF 11301, and 1100 AF 11500 are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

Reactor coolant system temperature instrumentation located in fire zones 1000 AF 11301 and 1000 AF 11302 are sufficient to provide the monitoring function accomplished by the passive residual heat removal heat exchanger flow instrumentation located in this fire zone.

The reactor coolant system to chemical and volume control system stop valves located in this fire zone are conservatively assumed to be disabled as a result of a fire in this fire zone. The chemical and volume control system containment isolation valves located outside of this fire zone provide backup isolation capability to maintain the reactor coolant pressure boundary.

The chemical and volume control system and the liquid radwaste system containment isolation valves located outside the containment fire area are redundant to the containment isolation valves inside containment in this fire zone and are sufficient to perform the applicable functions to maintain containment integrity.

The redundant steam line pressure instruments located in fire area 1201 AF 05 for steam generator 1 and in fire area 1201 AF 06 for steam generator 2 are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

The redundant core exit thermocouples located in fire zone 1100 AF 11500 are sufficient to provide the applicable safe shutdown monitoring function.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.9 Fire Zone 1100 AF 11301

This fire zone is comprised of the following room(s):

Room No.	
11201	Steam generator compartment 1
11301	Steam generator 1 lower manway area
11401	Steam generator 1 tubesheet area
11501	Steam generator 1 operating deck
11601	Steam generator 1 feedwater nozzle area
11701	Steam generator 1 upper manway area

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

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The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. Cables are generally located at separate locations near the perimeter of the fire zone and there are no significant cable concentrations. This fire zone is separated from other fire zones (except fire zone 1100 AF 11500) by structural barriers or partial barriers. The bottom of this fire zone is the solid concrete floor of the steam generator compartment. Up to an elevation more than 17 feet above the operating deck the fire zone is enclosed by walls with a concrete thickness of more than one foot, except for access passageways to and from the pressurizer compartment (fire zone 1100 AF 11303) and the adjoining portion of the vertical access area (fire zone 1100 AF 11204), and the floor grating interface between the vertical access area and the steam generator 1 access room (fire zone 1100 AF 11303). Above the top of these concrete walls, the fire zone is open to the large air space above the operating deck (fire zone 1100 AF 11500). A fire does not propagate beyond this fire zone to the extent that it damages redundant safe shutdown components in another fire zone.

Depending on fire location, smoke and hot gases from a fire in this fire zone rise through the annular space surrounding the steam generator or through the pressurizer compartment (fire zone 1100 AF 11303) and into the air space in the upper portion of the containment (fire zone 1100 AF 11500). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components located in this fire zone. Although the consequences of a fire are expected to be very limited, a fire in this fire zone is conservatively assumed to disable all of the safe shutdown components in this fire zone.

The redundant reactor coolant system hot leg/cold leg instrumentation located in fire zone 1100 AF 11204 and redundant steam generator system steam generator level instrumentation located in 1100 AF 11300B are sufficient to perform applicable functions to achieve and maintain safe shutdown.

The four divisions of reactor coolant system/reactor coolant pump bearing water temperature instrumentation are assumed to be disabled and would not be available to detect and provide a trip signal on a loss of component cooling water to the pump. If the fire in this fire zone does not disable the pump, the component cooling water flow to the pump will be unaffected by the fire and will continue to provide cooling water to the pump bearings until the pump is tripped by other means.

The reactor coolant system reactor coolant pump shaft speed instruments are conservatively assumed to be disabled. The redundant reactor coolant system hot leg flow instrumentation located in fire zones 1100 AF 11300A and 1100 AF 11300B is sufficient to perform applicable functions to achieve and maintain safe shutdown.

The four reactor coolant system reactor head vent valves are assumed to be disabled. If power is lost while in the closed position, the head vent valves will maintain reactor coolant pressure boundary integrity. Refer to Subsection 9A.3.7.1.1 for a discussion on spurious actuation of reactor coolant system reactor head vent valves.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

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9A.3.1.1.10 Fire Zone 1100 AF 11302

This fire zone is comprised of the following room(s):

Room No.	
11202	Steam generator compartment 2
11302	Steam generator 2 lower manway area
11402	Steam generator 2 tubesheet area
11502	Steam generator 2 operating deck
11602	Steam generator 2 feedwater nozzle area
11702	Steam generator 2 upper manway area

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. Cables are generally located at separate locations near the perimeter of the fire zone and there are no significant cable concentrations. This fire zone is separated from other fire zones (except fire zone 1100 AF 11500) by structural barriers or partial barriers. The bottom of this fire zone is the solid concrete floor of the steam generator compartment. Up to an elevation of more than 17 feet above the operating deck the fire zone is enclosed by walls with a concrete thickness of more than one foot, except for access passageways from the vertical access area (fire zone 1100 AF 11204) and the maintenance floor (fire zone 1100 AF 11300B). Above the top of these concrete walls, the fire zone is open to the large air space above the operating deck (fire zone 1100 AF 11500). A fire does not propagate beyond this fire zone to the extent that it damages redundant safe shutdown components in another fire zone.

Smoke and hot gases from a fire in this fire zone rise through the annular space surrounding the steam generator into the air space in the upper portion of the containment (fire zone 1100 AF 11500). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components located in this fire zone. Although the consequences of a fire are expected to be very limited, a fire in this fire zone is conservatively assumed to disable all of the safe shutdown components in this fire zone.

The redundant reactor coolant system hot leg/cold leg instrumentation located in fire zone 1100 AF 11204 are sufficient to perform applicable functions to achieve and maintain safe shutdown.

The four divisions of reactor coolant system/reactor coolant pump bearing water temperature instrumentation are assumed to be disabled and would not be available to detect and provide a trip signal on a loss of component cooling water to the pump. If the fire in this fire zone does not disable the pump, the component cooling water flow to the pump will be unaffected by the fire and will continue to provide cooling water to the pump bearings until the pump is tripped by other means.

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The reactor coolant system reactor coolant pump shaft speed instruments are conservatively assumed to be disabled. The redundant reactor coolant system flow instrumentation located in fire zones 1100 AF 11300A and 1100 AF 11300B are sufficient to perform applicable functions to achieve and maintain safe shutdown.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.11 Fire Zone 1100 AF 11303

This fire zone is comprised of the following room(s):

Room No.	
11303	Lower pressurizer compartment
11304	Steam generator 1 access room
11503	Upper pressurizer compartment

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are no significant cable concentrations. This fire zone is separated from other fire zones (except fire zone 1100 AF 11500) by structural barriers or partial barriers. The bottom of this fire zone is solid concrete except for floor grating in the steam generator 1 access room (above fire zone 1100 AF 11301). Up to an elevation more than 33 feet above the operating deck this fire zone is enclosed by walls with a minimum concrete thickness of more than one foot, except for access passageways to and from the steam generator 1 compartment (fire zone 1100 AF 11301) and a doorway from the maintenance floor (fire zone 1100 AF 11300B). Several feet above the top of these walls, a steel platform separates the top this fire zone from fire zone 1100 AF 11303A directly above. Between the top of the walls and this platform the sides of this fire zone are open to the large air space above the operating deck (fire zone 1100 AF 11500). A fire does not propagate beyond this fire zone to the extent that it damages redundant safe shutdown components in another fire zone.

Smoke and hot gases from a fire in this fire zone rise through the annular space surrounding the pressurizer into the air space in the upper portion of the containment (fire zone 1100 AF 11500). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components contained in this fire zone. The quantity of combustible materials is low consisting primarily of the cable insulation for the pressurizer heaters located at the bottom of the fire zone. The RCS pressurizer level reference leg temperature instruments are separated from these combustible materials by a distance of approximately 35 feet. There are no intervening combustibles or fire hazards located near to or between the RCS pressurizer level reference leg temperature instruments. Smoke and hot gases from a fire in the cable trays located approximately 35 feet below would not be expected to accumulate in the large open spaces of the fire zone because the fire zone is open to the main containment atmosphere and thus venting of the fire zone will occur. As a result, the safe shutdown components located in this fire

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zone would not be adversely impacted by a fire. The 35-foot distance will not cause the reference leg temperature, the RTD temperature or the RTD cable temperature to exceed the 700 degree Fahrenheit temperature range of the RTD, and therefore all four compensated level measurements will have good quality.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.12 Fire Zone 1100 AF 11303A

This fire zone is comprised of the following room(s):

Room No.	
11603	Lower automatic depressurization system valve area

Safe Shutdown Evaluation

There are no safe shutdown components located in this fire zone that are required to operate as a result of a fire.

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are no significant cable concentrations. This fire zone is separated from other fire zones (except fire zone 1100 AF 11500) by structural barriers. Steel plate provides separation from the pressurizer compartment (fire zone 1100 AF 11303). Steel plate provides separation from the upper automatic depressurization system valve area (fire zone 1100 AF 11303B). The sides of this fire zone are open to the large containment air space above the operating deck (fire zone 1100 AF 11500). A fire does not propagate beyond this fire zone to the extent that it damages redundant safe shutdown components in another fire zone.

Smoke and hot gases from a fire in this fire zone are deflected horizontally by the ceiling into the surrounding air space in the upper portion of the containment (fire zone 1100 AF 11500). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.13 Fire Zone 1100 AF 11303B

This fire zone is comprised of the following room(s):

Room No.	
11703	Upper automatic depressurization system valve area

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Safe Shutdown Evaluation

There are no safe shutdown components located in this fire zone that are required to operate as a result of a fire.

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are no significant cable concentrations. Steel plate provides separation from the lower automatic depressurization system valve area (fire zone 1100 AF 11303A). The top and sides of this fire zone are open to the large containment air space above the operating deck (fire zone 1100 AF 11500), which has no nearby combustibles. A fire does not propagate beyond this fire zone to the extent that it damages redundant safe shutdown components in another fire zone.

Smoke and hot gases from a fire in this fire zone rise into the large air space in the upper portion of containment (fire zone 1100 AF 11500). They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.14 Fire Zone 1100 AF 11500

This fire zone is comprised of the following room(s):

Room No.	
11500	Operating deck
11504	Refueling cavity
11306	Division B/D Penetration Room

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separates this zone from other fire zones are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in another fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. There are concentrations of cables in horizontal raceways around the circumference of this fire zone high above the operating deck, in vertical raceways at separate locations near the boundaries of this fire zone, and at the center of the fire zone in the vicinity of the reactor vessel integrated head package. This fire zone encompasses much of the containment. It is physically separated from fire zones below by the operating deck or the bottom of the refueling cavity, which have a concrete thicknesses of more than one foot, except for penetrations described in the evaluations of fire zones below this fire zone. There also is a few inch clearance annulus around the entire operating deck. The walls of this fire zone are the steel containment vessel or walls with a concrete thickness of more than one foot, with exceptions as described earlier for fire zones 1100 AF 11301, 1100 AF 11302, 1100 AF 11303, and 1100 AF 11303A & B. The boundary of this fire zone also includes the 3-hour fire barriers that protect the division B and D containment penetrations

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on elevation 107'-2" and the associated raceways from these penetrations up to the operating deck. A fire does not propagate beyond this fire zone to the extent that it damages redundant safe shutdown components in another fire zone.

Smoke and hot gases from a fire in this fire zone rise into the large air space above the operating deck. They are cooled by mixing with the air and by contact with structural surfaces and thus do not cause propagation of the fire beyond this fire zone. Safe shutdown components listed in Table 9A-2 for the adjacent fire zones are not susceptible to damage by the diluted and cooled smoke and gases from this fire zone.

Table 9A-2 lists the safe shutdown components located in this fire zone. Although the consequences of a fire are expected to be very limited, a fire in this fire zone is conservatively assumed to disable all of the safe shutdown components in this fire zone.

Control of all division B and D components in the containment is conservatively assumed to be disabled. The primary division A and C electrical cables that provide power supply to safe shutdown components in containment are located in 1100 AF 11300B and are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

The in-core instrumentation system core exit temperature instrument cable assemblies located in this fire zone are conservatively assumed to be disabled as a result of a fire in this fire zone. The reactor coolant system hot leg temperature (wide range) instrumentation located in fire zones 1100 AF 11301 and 1100 AF 11302 provide a diverse means of observing temperature conditions in the reactor vessel to support the safe shutdown process.

The reactor coolant system narrow range level instrumentation is conservatively assumed to be disabled. The redundant reactor coolant system narrow range level instrumentation located in fire zone 1100 AF 11300B is sufficient to perform the applicable functions to achieve and maintain safe shutdown.

The central chilled water system containment isolation valve located outside the containment fire area is redundant to the containment isolation valve inside containment in this fire zone and is sufficient to perform the applicable functions to maintain containment integrity.

The redundant reactor coolant system pressurizer instrumentation and redundant reactor coolant system hot leg instrumentation located in 1100 AF 11300B are sufficient to perform applicable functions to achieve and maintain safe shutdown.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.15 Fire Zone 1200 AF 12341

This fire zone is comprised of the following room(s):

Room No.	
12341	Middle annulus

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire does not propagate to or from this fire zone.

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The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. The Class 1E electrical penetration assemblies pass through this fire zone, but are separated by 3-hour rated fire barriers constructed between the containment steel shell and the shield building concrete structure. The Class 1E electrical penetration assemblies are considered extensions of the associated Class 1E divisional fire areas on the other side of the shield building wall except for the division A penetration assemblies, which are part of this fire zone. This fire zone is physically separated from other fire zones by the steel wall of containment and by the steel and concrete vessel stiffener and flexible ventilation seal above, and it is separated from adjacent fire areas by the walls and floor of the shield building, which have concrete thicknesses of more than one foot, and the 3-hour fire barriers enclosing the Class 1E electrical penetrations. The access doorway to the middle annulus fire zone is closed by a door.

The radiologically controlled area ventilation system serves this fire area on a once-through basis. Smoke and hot gases are confined in this fire zone following automatic closure of the fire dampers on high temperature, while the balance of the radiologically controlled area ventilation system continues to operate at the discretion of the operator. There is no propagation of the fire beyond this fire zone. Smoke and gases are removed from the fire zone by reopening the fire dampers after a fire. The radiologically controlled area ventilation system exhausts the smoke and gases to the atmosphere.

Table 9A-2 lists the safe shutdown components located in this fire zone. The redundant passive containment cooling system safe shutdown components located in fire area 1200 AF 01 are sufficient to perform applicable functions to achieve and maintain safe shutdown. The Class 1E electrical penetrations for division B, C, and D are separated from this fire zone by 3-hour fire barriers between the containment steel shell and the shield building concrete structure and are part of the associated divisional fire areas outside the shield building. The division A Class 1E electrical penetration assemblies that are part of this fire zone are not associated with the cables necessary to support the function of the redundant PCS components in fire area 1200 AF 01.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.16 Fire Zone 1200 AF 12541

This fire zone is comprised of the following room(s):

Room No.	
12541	Upper annulus
12702	PCS Air Diffuser

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire does not propagate to or from this fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. This fire zone is physically separated from other fire zones by the steel wall of containment and the steel and concrete vessel stiffener and flexible ventilation seal below, and it is separated from adjacent fire areas by the walls of the shield building, which have a concrete thickness of more than one foot. Access doorways are closed by doors. The physical separation between this fire zone and fire zone 1270 AF 12701 are described in the evaluation for that zone. This fire zone communicates with the environment via the passive containment cooling system air inlets at the top perimeter of the shield building and the passive containment cooling system air outlet

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at the center of the shield building roof. These openings have screens that prevent the entry of external debris.

Smoke and hot gases from a fire in this fire zone rise to the top of the fire zone and into the atmosphere and thus do not cause propagation of a fire beyond this fire zone.

The safe shutdown components in this fire zone are the Divisions A, B, and C electrical cables that serve the redundant passive containment cooling system valves and instruments located in the passive containment cooling system valve room. Although the consequences of a fire are expected to be very limited, a fire in this fire zone is conservatively assumed to disable these safe shutdown valves and instruments.

The valves for each passive containment cooling system water delivery path are arranged with a normally open motor-operated valve and normally closed/fail open air-operated valve in series. If the fire causes a loss of power to the valves, the air-operated valves will open and passive containment cooling system flow, which has no adverse impact on achieving and maintaining safe shutdown, will be initiated. Refer to Subsection 9A.3.7.1.2 for a discussion of potential spurious actuation of a passive containment cooling system water delivery valve as a result of a fire.

The passive containment cooling system water delivery flow and storage tank level instrumentation are conservatively assumed to be disabled as a result of a fire in this fire zone. The applicable function of verification of passive containment cooling system water delivery can be performed by visual observation via access to the passive containment cooling system air diffuser from the passive containment cooling system valve room.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.17 Fire Zone 1270 AF 12701

This fire zone is comprised of the following room(s):

Room No.	
12701	Passive containment cooling system valve room
S06	Stairwell

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire does not propagate to or from this fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. This fire zone is physically separated from fire zone 1200 AF 12541 by structural partitions and closed doorways.

In the unlikely event of a fire, the fire brigade would approach the PCS valve room (Room 12701, Fire Zone 1270 AF, Fire Area 1000 AF 01) by using stairwell S03 or the adjacent elevator attached to the outside wall on the shield building. Two types of portable fire extinguishers (a dry chemical and a water fire extinguisher) are provided at this lower level for manual fire fighting. The fire brigade would proceed up the inclined stairs S06 to the upper level of the PCS valve room (Room 12701) located on El. 286′-6″. Two types of portable fire extinguishers (a dry chemical and a water fire extinguisher) are also provided at this upper level for manual fire fighting.

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Smoke and hot gases from a fire in Room 12701 are exhausted by normal operation of the room exhaust fan to fire zone 1200 AF 12541, where they rise through openings in the top of that fire zone into the atmosphere. There are no combustible materials in the vicinity of the exhaust location and thus the smoke and hot gases do not cause propagation of a fire beyond this fire zone. If the exhaust fan is disabled by the fire, smoke and gases are later removed using portable fans and ductwork. The stairwell S06 is not served by a ventilation system. The quantity of combustible materials in the stairwell is negligible.

Table 9A-2 lists the safe shutdown components located in this fire zone. Although it is unlikely that all components would be damaged, a fire in this fire zone is conservatively assumed to disable all of the safe shutdown valves and instruments in this fire zone.

The valves for each passive containment cooling system water delivery path are arranged in three flow paths. Two paths have a normally open motor-operated valve and normally closed/fail open air-operated valve in series. The third path has a normally open motor-operated and a normally closed motor operated valve in series. If the fire causes a loss of power to the air-operated valves, they will open and passive containment cooling system flow, which has no adverse impact on achieving and maintaining safe shutdown, will be initiated. Refer to Subsection 9A.3.7.1.2 for a discussion of potential spurious actuation of a passive containment cooling system water delivery valve as a result of a fire.

The passive containment cooling system water delivery flow and storage tank level instrumentation are conservatively assumed to be disabled as a result of a fire in this fire zone. The applicable function of verification of passive containment cooling system water delivery can be performed by visual observation via access to the passive containment cooling system air diffuser from the passive containment cooling system valve room or from the upper annulus.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.1.18 Fire Zone 1250 AF 12555

This fire zone is comprised of the following room(s):

Room No.	
12555	Main control room emergency habitability system air storage
12556	Operating deck staging area

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire zone, and the characteristics of the barriers that separate this zone from other fire zones are such that a fire does not propagate to or from this fire zone.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3A. This fire zone is separated from adjacent fire areas by 3-hour fire barriers and it is separated from adjacent containment fire zones 1100 AF 11500 and 1200 AF 12541 by the main equipment hatch and its enclosure and the shield building wall, which has a concrete thickness of more than one foot.

Smoke and hot gases from a fire are confined in this fire zone following automatic closure of the fire dampers on high temperature and thus do not cause propagation of the fire beyond this fire zone.

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Smoke and gases are removed from the fire area by reopening the fire dampers after a fire. The radiologically controlled area ventilation system exhausts the smoke and gases to the atmosphere.

This fire zone contains no components required for safe shutdown after a fire. The pressurized main control room emergency habitability system air storage bottles are not required for safe shutdown after a fire, but are protected from fire-induced overpressure by pressure relief valves. There is no permanent storage of combustibles in room 12555.

No fire in this zone can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2 Auxiliary Building - Nonradiologically Controlled Areas

General Arrangement

The safe shutdown systems and components located in the nonradiologically controlled area are portions of the protection and safety monitoring system and the Class 1E dc system, and containment isolation.

The safe shutdown components in the protection and safety monitoring system are the instrumentation and control cabinets located in the nonradiologically controlled area on level 3 (elevation 100′-0″). The safe shutdown components in the Class 1E dc system are the Class 1E batteries on level 1 (elevation 66′-6″) and level 2 (elevation 82′-6″) and the dc electrical equipment, also on level 2.

The nonradiologically controlled areas of the auxiliary building are designed to provide separation between the mechanical and electrical equipment areas.

The piping compartments in the nonradiologically controlled area are the main steam isolation valve compartments on levels 4 and 5 (elevations 117′-6″ and 135′-3″, respectively) and the valve/piping penetration compartment on level 3 (elevation 100′-0″). The mechanical equipment rooms in the nonradiologically controlled area are the HVAC compartments on levels 4 and 5.

The nonradiologically controlled areas of the auxiliary building are also designed to provide separation between the Class 1E and the non-Class 1E electrical equipment.

Smoke Control

Table 9A-4 identifies the ventilation systems serving fire areas containing Class 1E electrical components. This section describes the approach to smoke control for fire areas in the nonradiologically controlled portion of the auxiliary building that contain the main Class 1E electrical equipment rooms served by the nuclear island nonradioactive ventilation system (VBS). Smoke control for fire areas containing other Class 1E components, such as valves, instrumentation and electrical cable, is discussed in the text for the individual fire areas.

The Class 1E electrical equipment room fire areas have been designed to prevent the migration of smoke, hot gases, and fire suppressant to the extent that they could adversely affect safe shutdown capabilities, including operator actions. These fire areas are separated from each other and from other plant areas by 3-hour fire barriers. Smoke from a fire in the turbine building or other nearby fire areas is prevented from affecting the Class 1E areas by isolation of the nuclear island nonradioactive ventilation system outdoor air intakes, as described in Subsection 9.4.1.

The nuclear island nonradioactive ventilation system is designed to control the migration of smoke and hot gases produced by a fire. As described in Subsection 9.4.1, two independent ventilation

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subsystems, located in separate fire areas, serve the Class 1E electrical equipment rooms. The division A and C Class 1E electrical room HVAC subsystem has three distribution headers. One header supplies the two division A electrical equipment room fire areas, a second header supplies the division C electrical equipment room fire area, and a third header supplies other related fire areas. The division B and D class 1E electrical room HVAC subsystem also has three distribution headers. One header supplies the division B electrical equipment room fire area, a second header supplies the division D electrical equipment room fire area, and a third header supplies other related fire areas.

A fire affecting a division A or C electrical equipment room fire area does not affect operation of the ventilation subsystem serving the division B & D electrical equipment room fire areas and vice versa. In addition, a fire affecting an electrical equipment room fire area affects the operation of only one of the three distribution headers in the subsystem. As described in Subsection 9.4.1.2.3.2, the affected subsystem continues to provide ventilation to the remaining fire areas served by the other two distribution headers.

Similarly, the ventilation subsystem serving the main control room and the ventilation subsystem serving the remote shutdown room operate independently of each other and are located in separate fire areas. A fire affecting the main control room does not affect operation of the ventilation subsystem serving the remote shutdown room and vice versa.

The migration of smoke and hot gases produced by a fire occurring in a Class 1E electrical equipment room fire area is controlled by operation of the nuclear island nonradioactive ventilation system as described in Subsection 9.4.1.2.3. Further information on smoke control is provided in the discussions for the individual fire areas.

9A.3.1.2.1 Division A Electrical Rooms

9A.3.1.2.1.1 Fire Area 1202 AF 04

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1212 AF 12101	12101	Division A battery room
● 1222 AF 12201	12201	Division A dc equipment room
● 1232 AF 12301	12301	Division A instrumentation and control room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division A & C Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas, which includes the Division C fire area (1202 AF 03). The system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may

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be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire and smoke dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Smoke dampers are provided in the ventilation system supply and return ducts to the dc equipment room and to the instrumentation and control room, to delay smoke migration between these fire zones and facilitate operator action to preclude postulated spurious actuations.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of battery cells on opposite walls of the battery room fire zone. There are concentrations of cable above the electrical cabinets located on opposite walls of the dc equipment room fire zone. There are concentrations of cable overhead and above the electrical cabinets located in the middle of the instrument and control room fire zone. Automatic suppression is not installed in this fire area as an effort is made to limit water sources in this area for internal flooding concerns. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. The three fire zones in this fire area are separated from each other by 1-hour fire barriers, which limits the spread of fire within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. These division A electrical rooms are physically separated from the other safety-related divisions by 3-hour fire barriers. In the event of a fire in one of these rooms, it is assumed that control of all division A components is lost. Because of the physical separation, the fire does not adversely affect the other safety-related electrical divisions. For this event, the division B, C, and D components identified in Table 9A-2 are sufficient to achieve and maintain safe shutdown.

Class 1E divisions B, C, and D interdivisional cables for data communication between divisions are in this fire area. In the event of a fire, it is assumed that data transmitted between division A and the other Class 1E divisions is lost. This loss of data does not disable the safe shutdown functions of the three Class 1E divisions.

Control room dedicated switches which are used to initiate engineered safety features at the system level are connected to the engineered safety features actuation cabinets using two-pole, energize-to-actuate, ungrounded dc circuits.

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Spurious actuation from control room dedicated switches which could lead to a breach of reactor coolant system pressure boundary, loss of decay heat removal function, or loss of shutdown reactivity control is prevented by the use of dual two-pole, energize-to-actuate, ungrounded dc circuits, which require at least four simultaneous hot shorts of proper polarity for spurious actuation.

Following detection of a fire in the instrumentation and control room, the operators can close the automatic depressurization system stage 4 block valve, then remove actuation power from this division using the battery transfer switch located in the dc equipment room to disconnect the battery and remote control from the control room to remove input power from the battery charger and regulating transformer. This operator action will prevent spurious actuation of motor operated valves and squib valves resulting from multiple hot shorts in the instrumentation and control room.

Following detection of a fire in the dc equipment control room, the operators can close the automatic depressurization system stage 4 block valve, then remove cabinet power from this division using the input power switches on the instrumentation and control cabinets. This operator action will prevent spurious smoke-induced actuation of motor operated valves and squib valves resulting smoke-related integrated circuit failures in the instrumentation and control room.

Power to the passive residual heat removal heat exchanger inlet isolation valve is normally locked out at power to prevent spurious closing.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.2.1.2 Fire Area 1242 AF 02

This fire area is comprised of the following room(s):

Room No.	
12412	Electrical penetration room division A

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division A & C Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas, which includes the division C electrical penetration room (in fire area 1202 AF 03). The system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

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Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of cable at the electrical penetrations and in the overhead cable trays. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

Two discharge openings penetrate the 24" thick concrete wall between room 12412 and the main control room vestibule in fire area 1242 AF 01 to provide a purge flow path for the main control room habitability system, as discussed in Subsection 6.4.2.3. The 4-inch (nominal) diameter openings are provided with downturned pipe elbows on each side terminating approximately 6 inches above the floor. A full-scale fire test conducted at a qualified testing facility in accordance with ASTM E814-13a (Reference 204) demonstrated that this design meets the 3-hour fire barrier requirements for fire rating and temperature rating.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The division A penetration room is physically separated from the other safety-related divisions and nonsafety-related equipment by 3-hour fire barriers. In the event of a fire in this room, it is assumed that control of all division A and C active components is lost. Because of the physical separation, the fire does not adversely affect the other safety-related electrical divisions. For this event, the division B and D components identified in Table 9A-2 are sufficient to achieve and maintain safe shutdown.

In addition, division C electrical cables that serve the redundant passive containment cooling system valves and instruments located in the passive containment cooling system (PCS) valve room (Room 12701) are routed through this fire area. In the event of a fire, it is also assumed that this division is disabled. The remaining division B electrical cables and components routed through and located in other fire areas are sufficient to perform the applicable functions to achieve and maintain safe shutdown. Refer to Subsection 9A.3.1.1.17, Fire Zone 1270 AF 12701, for the Safe Shutdown Evaluation of that zone.

Following detection of a fire in this fire area, the operators can close the automatic depressurization system stage 4 block valve, then remove actuation power from this division using the battery transfer switch located in the dc equipment room to disconnect the battery and remote control from the control room to remove input power from the battery charger and regulating transformer. This operator action will prevent spurious actuation of motor operated valves and squib valves resulting from multiple hot shorts in the penetration room.

Power to the passive residual heat removal heat exchanger inlet isolation valve is normally locked out at power to prevent spurious closing.

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Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.2.2 Division B Electrical Rooms

9A.3.1.2.2.1 Fire Area 1201 AF 02

This fire area contains division B electrical rooms. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1211 AF 12104	12104	Division B battery room 1
● 1221 AF 12204	12204	Division B battery room 2
● 1222 AF 12207	12207	Division B dc equipment room
● 1231 AF 12304	12304	Division B instrumentation and control/penetration room
● 1231 AF 12344	12344	Division B middle annulus penetration room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division B & D Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas, which includes the division D fire area (1201 AF 03). The system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire and smoke dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. The middle annulus penetration room division B is treated as an extension of this fire area. No ventilation system serves this fire zone, and smoke and hot gases are confined with no propagation of the fire beyond the fire zone. After the fire, smoke is removed by opening the personnel doors and using portable exhaust fans and flexible ductwork. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Smoke dampers are provided in the ventilation system supply and return ducts to the dc equipment room and to the instrumentation and control/penetration room, to delay smoke migration between these fire zones and facilitate operator action to preclude postulated spurious actuations.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector, which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

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The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of battery cells on opposite walls of each of the two battery room fire zones. There are concentrations of cable overhead and above the electrical cabinets located on opposite walls of the dc equipment room fire zone. There are concentrations of cable overhead, at the electrical penetrations, and above the electrical cabinets located in the middle of the instrumentation and control/penetration room fire zone. This is a light hazard fire area and the rate of fire growth is expected to be slow. The boundary of this fire area extends to include the electrical penetration assemblies within the containment annulus, which are enclosed by 3-hour fire barriers. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. The battery rooms are also separated from the adjacent division B electrical rooms by 1-hour fire barriers, and the dc equipment room is separated from adjacent division B electrical rooms by 1-hour fire barriers, which limit the spread of fire within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. In the auxiliary building, the division B electrical rooms are physically separated from the other safety-related divisions and nonsafety-related equipment by 3-hour fire barriers. In the middle annulus, the Division B electrical penetrations are separated from the other safety-related divisions and nonsafety-related equipment by 3-hour rated fire barriers constructed between the containment steel shell and the shield building concrete structure. In the event of a fire in a division B electrical room, it is assumed that control of all division B active components is lost. Because of the physical separation, the fire does not adversely affect the other safety-related electrical divisions. For this event, the division A, C, and D components identified in Table 9A-2 are sufficient to achieve and maintain safe shutdown.

There is a division D conduit that traverses fire area 1201 AF 02, the cables in this conduit are associated with the main control room (MCR)/remote shutdown room transfer switch function. The habitability of the MCR is not affected by a postulated fire in this zone and the affected cables cannot cause or prevent division D components from functioning.

Class 1E divisions A, C, and D interdivisional cables for data communication between divisions are in this fire area. In the event of a fire, it is assumed that data transmitted between division B and the other Class 1E divisions is lost. This loss of data does not disable the safe shutdown functions of the three Class 1E divisions.

Control room dedicated switches which are used to initiate engineered safety features at the system level are connected to the engineered safety features actuation cabinets using two-pole, energize-to-actuate, ungrounded dc circuits.

Spurious actuation from control room dedicated switches which could lead to a breach of reactor coolant system pressure boundary, loss of decay heat removal function, or loss of shutdown reactivity control is prevented by the use of dual two-pole, energize-to-actuate, ungrounded dc circuits, which require at least four simultaneous hot shorts of proper polarity for spurious actuation.

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Following detection of a fire in the instrumentation and control/penetration room, the operators can close the automatic depressurization system stage 4 block valve, then remove actuation power from this division using the battery transfer switch located in the dc equipment room to disconnect the battery and remote control from the control room to remove input power from the battery charger and regulating transformer. This operator action will prevent spurious actuation of motor operated valves and squib valves resulting from multiple hot shorts in the instrumentation and control/penetration room.

Following detection of a fire in the dc equipment control room, the operators can close the automatic depressurization system stage 4 block valve, then remove cabinet power from this division using the input power switches on the instrumentation and control cabinets. This operator action will prevent spurious smoke-induced actuation of motor operated valves and squib valves resulting smoke-related integrated circuit failures in the instrumentation and control room.

Power to the normal residual heat removal hot leg suction isolation valves is normally locked out at power to prevent spurious opening.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.2.2.2 Fire Area 1220 AF 01

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1222 AF 12212	12212	Division B reactor coolant pump trip switchgear room
● 1222 AF 12213	12213	Spare room
● 1220 AF 12211	12211	Corridor

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division A & C Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas, including the fire area for the division C trip switchgear room (1202 AF 03). Fire areas 1200 AF 03 and 1210 AF 01 are served by the same air distribution header as this fire area, but they continue to receive ventilation because they have separate supply ducts. The system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

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Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of cable overhead and above the electrical cabinets in the reactor coolant pump trip switchgear room and spare room fire zones. There are concentrations of cable overhead. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. See Section 3.4 for a discussion of the consequences of a break in a fire protection line in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. A fire in this fire area is assumed to disable the safe shutdown components in the fire area.

The reactor coolant pumps can be tripped by the redundant division C reactor coolant pump trip switchgear, located in fire area 1202 AF 03. The division B and D cable tray in the fire area includes signals to the reactor trip switchgear and inputs to other division protection logic. Inputs from divisions A and C are sufficient to trip the reactor when needed, or the reactor can be tripped manually. Loss of division A and C data input to the division B and D protection logic does not disable the safe shutdown functions of the four Class 1E divisions.

Cable trays in the fire areas include signals to B and D isolation valves outside containment. Containment isolation is provided by redundant containment isolation valves, located in another fire area inside containment. Redundant division C valves control water flow from the passive containment cooling system storage tank. The components identified in Table 9A-2 are sufficient to achieve and maintain safe shutdown.

The steam generator system steam generator 1 startup feedwater flow instrumentation is assumed to be disabled. The steam generator system steam generator 1 wide range level instrumentation located in fire zones 1100 AF 11300B and 1100 AF 11301 provides a diverse means of performing the applicable function of generating a passive residual heat removal actuation signal to achieve and maintain safe shutdown.

The steam generator system steam generator 2 startup feedwater flow instrumentation is assumed to be disabled. The steam generator system steam generator 2 wide range level instrumentation located in fire zones 1100 AF 11300A and 1100 AF 11300B provides a diverse means of performing the applicable function of generating a passive residual heat removal actuation signal to achieve and maintain safe shutdown.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

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No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.3 Division C Electrical Rooms

9A.3.1.2.3.1 Fire Area 1202 AF 03

This fire area contains division C electrical rooms. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1212 AF 12102	12102	Division C battery room 1
● 1222 AF 12202	12202	Division C battery room 2
● 1222 AF 12203	12203	Division C dc equipment room
● 1232 AF 12302	12302	Division C instrumentation and control room
● 1232 AF 12312	12312	Division C reactor coolant pump trip switchgear room
● 1232 AF 12313	12313	Instrumentation and control/division C penetration room
● 1232 AF 12343	12343	Division C middle annulus penetration room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division A & C Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas, which includes the division A fire area (1202 AF 04). The system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire and smoke dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. The middle annulus penetration room division C is treated as an extension of this fire area. No ventilation system serves this fire zone, and smoke and hot gases are confined with no propagation of the fire beyond the fire zone. After the fire, smoke is removed by opening the personnel doors and using portable exhaust fans and flexible ductwork. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Smoke dampers are provided in the ventilation system supply and return ducts to the dc equipment room, the instrumentation and control room, and the penetration room to delay smoke migration between these fire zones and facilitate operator action to preclude postulated spurious actuations.

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Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of battery cells on opposite walls of each of the two battery room fire zones. There are concentrations of cable overhead and above the electrical cabinets located on opposite walls of the dc equipment room fire zone. There are concentrations of cable overhead and above the electrical cabinets located in the middle of the instrument and control room fire zone. There are concentrations of cable overhead and above the electrical cabinets in the reactor coolant pump trip switchgear room. There are concentrations of cable overhead and at the electrical penetrations in the instrumentation and control/penetration room fire zone. This is a light hazard fire area and the rate of fire growth is expected to be slow. The boundary of this fire area extends to include the electrical penetration assemblies within the containment annulus, which are enclosed by 3-hour fire barriers. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. The battery rooms are also separated from the adjacent division C electrical rooms by 1-hour fire barriers, and the dc equipment room is separated from adjacent division C electrical rooms by 1-hour fire barriers, which limit the spread of fire within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. In the auxiliary building, the division C electrical rooms are physically separated from the other safety-related divisions and nonsafety-related equipment by 3-hour fire barriers. In the middle annulus, the Division C electrical penetrations are separated from the other safety-related divisions and nonsafety-related equipment by 3-hour rated fire barriers constructed between the containment steel shell and the shield building concrete structure. In the event of a fire in a division C electrical room, it is assumed that control of all division C components is lost. Because of the physical separation, the fire does not adversely affect the other safety-related electrical divisions. The reactor coolant pumps can be tripped by the redundant division B reactor coolant pump trip switchgear, located in fire area 1220 AF 01. For this event, the division A, B, and D components identified in Table 9A-2 are sufficient to achieve and maintain safe shutdown.

Class 1E divisions A, B, and D interdivisional cables for data communication between divisions are in this fire area. In the event of a fire, it is assumed that data transmitted between division C and the other Class 1E divisions is lost. This loss of data does not disable the safe shutdown functions of the three Class 1E divisions.

Control room dedicated switches which are used to initiate engineered safety features at the system level are connected to the engineered safety features actuation cabinets using two-pole, energize-to-actuate, ungrounded dc circuits.

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Spurious actuation from control room dedicated switches which could lead to a breach of reactor coolant system pressure boundary, loss of decay heat removal function, or loss of shutdown reactivity control is prevented by the use of dual two-pole, energize-to-actuate, ungrounded dc circuits, which require at least four simultaneous hot shorts of proper polarity for spurious actuation.

Following detection of a fire in either the instrumentation and control room or the instrumentation and control/division C penetration room, the operators can close the automatic depressurization system stage 4 block valve, then remove actuation power from this division using the battery transfer switch located in the dc equipment room to disconnect the battery and remote control from the control room to remove input power from the battery charger and regulating transformer. This operator action will prevent spurious actuation of motor operated valves and squib valves resulting from multiple hot shorts in the instrumentation and control room or the instrumentation and control/division C penetration room.

Following detection of a fire in the dc equipment control room, the operators can close the automatic depressurization system stage 4 block valve, then remove cabinet power from this division using the input power switches on the instrumentation and control cabinets. This operator action will prevent spurious smoke-induced actuation of motor operated valves and squib valves resulting smoke-related integrated circuit failures in the instrumentation and control room.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.2.4 Division D Electrical Rooms

9A.3.1.2.4.1 Fire Area 1201 AF 03

The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1211 AF 12105	12105	Division D battery room
● 1221 AF 12205	12205	Division D dc equipment room
● 1231 AF 12305	12305	Division D instrumentation and control/penetration room
● 1231 AF 12345	12345	Division D middle annulus penetration room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division B & D Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas, which includes the division B fire area (1201 AF 02). The system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire

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area by reopening the fire and smoke dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. The middle annulus penetration room division D is treated as an extension of this fire area. No ventilation system serves this fire zone, and smoke and hot gases are confined with no propagation of the fire beyond the fire zone. After the fire, smoke is removed by opening the personnel doors and using portable exhaust fans and flexible ductwork. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Smoke dampers are provided in the ventilation system supply and return ducts to the dc equipment room and to the instrumentation and control/penetration room, to delay smoke migration between these fire zones and facilitate operator action to preclude certain postulated spurious actuations.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of battery cells on opposite walls of the battery room fire zone. There are concentrations of cable overhead and above the electrical cabinets located on opposite walls of the dc equipment room fire zone. There are concentrations of cable overhead, at the electrical penetrations, and above the electrical cabinets located in the middle of the instrumentation and control/penetration room fire zone. This is a light hazard fire area and the rate of fire growth is expected to be slow. The boundary of this fire area extends to include the electrical penetration assemblies within the containment annulus, which are enclosed by 3-hour fire barriers. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. The three fire zones in this fire area are separated from each other by 1-hour fire barriers, which limits the spread of fire within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. In the auxiliary building, the division D electrical rooms are physically separated from safety-related divisions A and C by 3-hour fire barriers. In the middle annulus, the Division D electrical penetrations are separated from the other safety-related divisions and nonsafety-related equipment by 3-hour rated fire barriers constructed between the containment steel shell and the shield building concrete structure. In the event of a fire in one of these rooms, it is assumed that control of all division D components is lost. Because of the physical separation, the fire does not adversely affect the other safety-related electrical divisions with the exception of the division B components located in fire zone 1270 AF 12701. For this event, the divisions A and C components identified in Table 9A-2 are sufficient to achieve and maintain safe shutdown.

Class 1E divisions A, B, and C interdivisional cables for data communication between divisions are in this fire area. In the event of a fire, it is assumed that data transmitted between division D and the

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other Class 1E divisions is lost. This loss of data does not disable the safe shutdown functions of the three Class 1E divisions.

Control room dedicated switches which are used to initiate engineered safety features at the system level are connected to the engineered safety features actuation cabinets using two-pole, energize-to-actuate, ungrounded dc circuits.

Spurious actuation from control room dedicated switches which could lead to a breach of reactor coolant system pressure boundary, loss of decay heat removal function, or loss of shutdown reactivity control is prevented by the use of dual two-pole, energize-to-actuate, ungrounded dc circuits, which require at least four simultaneous hot shorts of proper polarity for spurious actuation.

A fire in this fire area is conservatively assumed to disable the division B PCS components located in fire zone 1270 AF 12701. See Subsection 9A.3.1.1.17 for the Safe Shutdown Evaluation for fire zone 1270 AF 12701.

Following detection of a fire in the instrumentation and control/penetration room, the operators can close the automatic depressurization system stage 4 block valve, then remove actuation power from this division using the battery transfer switch located in the dc equipment room to disconnect the battery and remote control from the control room to remove input power from the battery charger and regulating transformer. This operator action will prevent spurious actuation of motor operated valves and squib valves resulting from multiple hot shorts in the instrumentation and control/penetration room.

Following detection of a fire in the dc equipment control room, the operators can close the automatic depressurization system stage 4 block valve, then remove cabinet power from this division using the input power switches on the instrumentation and control cabinets. This operator action will prevent spurious smoke-induced actuation of motor operated valves and squib valves resulting smoke-related integrated circuit failures in the instrumentation and control room.

Power to the normal residual heat removal hot leg suction isolation valves is normally locked out at power to prevent spurious opening.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.2.5 Principal Class 1E Areas

9A.3.1.2.5.1 Fire Area 1242 AF 01

This fire area consists of the main control room. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1242 AF 12401A	12400	Control Room Vestibule
● 1242 AF 12401A	12401	Main control area/operator work area
● 1242 AF 12401B	12401	Shift manager’s office/kitchen area/operator break room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors (including detectors in the subfloor spaces)

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- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by the main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). This VBS subsystem will shut down and combination fire-smoke dampers close automatically in response to a smoke detector signal(s) or high temperature to control the spread of fire and smoke. The system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Fire Protection Adequacy Evaluation

Since the main control room is continuously manned, a fire is likely to be initially detected by an operator. Otherwise, a fire in this fire area is detected by a fire detector, which produces visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using portable extinguishers or, if necessary, using hose streams.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. The fire area is continuously manned and prompt manual fire suppression is expected.

Generally, three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. Due to the need to provide passive cooling capability into the main control room ceilings, it will not be protected against fires from within the main control room. The ceiling will be a fire barrier from fires in the room above the main control room.

The two fire zones are also separated from each other by a 1-hour fire barrier, which limits the spread of fire within the fire area. Within fire zone 1242 AF 12401A the wall that separates the main control area from the operator work area is not fire-rated because it does not extend to the ceiling. This design improves the ceiling heat-absorbing characteristics provided for post-accident main control room habitability.

Two discharge openings penetrate the 24" thick concrete wall between the main control room vestibule and room 12412 in fire area 1242 AF 02 to provide a purge flow path for the main control room habitability system, as discussed in Subsection 6.4.2.3. The 4-inch (nominal) diameter openings are provided with downturned pipe elbows on each side terminating approximately 6 inches above the floor. A full-scale fire test conducted at a qualified testing facility in accordance with ASTM E814-13a (Reference 204) demonstrated that this design meets the 3-hour fire barrier requirements for fire rating and temperature rating.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

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Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The main control room contains circuits from the four Class 1E electrical divisions. Electrical separation to and inside the control panels is maintained per industry standards. The remote shutdown workstation is provided as an alternate to the main control room. The transfer of operations to the remote shutdown workstation is controlled by a transfer switch located outside the main control room. In the event of a fire in the main control room, control may be transferred to the remote shutdown workstation, depending on the extent of the fire. For a small fire, control is maintained in the main control room, and the potential for damage or spurious signals is limited. For larger postulated fires, the main control room is evacuated and control is transferred to the remote shutdown workstation. Once control is transferred, spurious control signals potentially caused by the fire are isolated from the actuated devices by the transfer switch. In this event, the main control room is assumed to be lost for the duration of the event. Safe shutdown is controlled from the remote shutdown workstation. The extent of spurious signals is limited by the time to transfer control to the remote shutdown workstation.

Spurious actuation from control room dedicated switches which could lead to a breach of reactor coolant system pressure boundary, loss of decay heat removal function, or loss of shutdown reactivity control is prevented by the use of dual two-pole, energize-to-actuate, ungrounded dc circuits, which require at least four simultaneous hot shorts of proper polarity for spurious actuation. Following control room evacuation, the dedicated switches are disabled by the transfer switch.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.2.5.2 Fire Area 1232 AF 01

This fire area is comprised of the following room(s):

Room No.	
12303	Remote shutdown room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division B & D Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas. The system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

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Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The remote shutdown room contains circuits from the four Class 1E electrical divisions. Electrical separation to and inside the remote shutdown room is maintained per industry standards. The remote shutdown room is an alternate to the main control room. The transfer of operations to the remote shutdown workstation is controlled by a transfer switch set located in the remote shutdown workstation area. In the unlikely event that the fire damages the transfer switch set, causing transfer of control from the main control room to the remote shutdown workstation, the operator restores control to the main control room by de-energizing fire area 1202 AF 05 (stair S05). Safe shutdown is achieved using the safe shutdown components listed in Table 9A-2.

Most remote shutdown workstation controls use soft-controls which communicate over multiplexed data channels. Fire-induced spurious actuation from these multiplexed soft controls is not assumed. Fire-induced actuations from the dedicated switches in this area are prevented during normal operation by the transfer switch logic, which only enables operation from the remote shutdown workstation dedicated switches when control is transferred to the remote shutdown workstation.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.2.5.3 Fire Area 1243 AF 01

This fire area is comprised of the following room(s):

Room No.	
12423	Reactor trip switchgear 1

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

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Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. This subsystem may be restarted and manually realigned to the once-through smoke exhaust ventilation mode to minimize the potential migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. This fire area contains cable from each of the four Class 1E electrical divisions. This cable provides trip input from each of the four divisions and is separated per industry standards. The safety-related trip inputs are normally energized, so a fire in this area may result in a reactor trip. In the event the fire generates multiple hot shorts, interfering with the reactor trip signal, a reactor trip can be produced in the redundant trip cabinets located outside of this fire area in fire area 1243 AF 02. Furthermore, the reactor can be tripped with the diverse actuation system described in Section 7.7.

This fire does not affect other equipment in the Class 1E divisions. Therefore, the safe shutdown components listed in Table 9A-2 are available to achieve and maintain safe shutdown.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

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9A.3.1.2.5.4 Fire Area 1243 AF 02

This fire area is comprised of the following room(s):

Room No.	
12422	Reactor trip switchgear 2

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. This subsystem may be restarted and manually realigned to the once-through smoke exhaust ventilation mode to minimize the potential migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. This fire area contains cable from each of the four Class 1E electrical divisions. This cable provides trip input from each of the four divisions and is separated per industry standards. The safety-related trip inputs are normally energized, so a fire in this area may result in a reactor trip. In the event the fire generates multiple hot shorts, interfering with the reactor trip signal, a reactor trip can be produced in the redundant trip

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cabinets located outside of this fire area in fire area 1243 AF 01. Furthermore, the reactor can be tripped with the diverse actuation system described in Section 7.7.

This fire does not affect other equipment in the Class 1E divisions. Therefore, the safe shutdown components listed in Table 9A-2 are available to achieve and maintain safe shutdown.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.5.5 Fire Area 1210 AF 01

The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 1210 AF 12111	12111	Corridor
• 1212 AF 12103	12103	Spare battery room
• 1212 AF 12112	12112	Spare room
• 1212 AF 12113	12113	Spare battery charger room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division A & C Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas, except for fire area 1220 AF 01 which is ventilated by ducts that are isolated by a fire in this fire area. The ventilation system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

If the spare batteries are being used as a backup power source for any division, a fire in this fire area is assumed to disable that division. Smoke from a fire in this fire area does not adversely affect components associated with divisions redundant to that supplied by the spare battery. Fire area 1200 AF 03 is served by the same air distribution header as this fire area, but it continues to receive ventilation because it has a separate supply duct. Fire area 1220 AF 01 is isolated by combination fire-smoke dampers in response to a smoke detector signal. The division B reactor trip

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switchgear and the division B and D electrical cable in this fire area is not sensitive to the small quantity of smoke that may leak into this fire area.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of battery cells on opposite walls of the spare battery room fire zone. There are concentrations of cable above the electrical cabinets in the spare battery charger room fire zone. There are assumed to be concentrations of ordinary combustibles in the spare room. There are concentrations of cable overhead in the corridor fire zone. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. The battery room is also separated from the other fire zones within this fire area by a 1-hour fire barrier, which limits the spread of fire within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. The consequences of a break in a fire protection line in this fire area were considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The spare batteries may be connected as a backup power source for any one of the four Class 1E electrical divisions. The terminations of the cables to these divisions from the spare batteries are not normally energized or connected, so a fire in this area has no impact on the unconnected divisions. If the spare batteries are being used as a backup to a Class 1E division, then the consequence of a fire in this area is the same as a fire in the battery room of the division to which they are connected.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.6 Non-Class 1E Electrical Rooms

9A.3.1.2.6.1 Fire Area 1230 AF 02

The fire area is comprised of the following room:

Room No.	
12321	Non-Class 1E equipment/penetration room

There are no systems in this fire area which normally contain radioactive material.

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Fire Detection and Suppression Features

- Fire detectors
- Preaction sprinklers
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. This subsystem may be restarted and manually realigned to the once-through smoke exhaust ventilation mode to minimize the potential migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by the preaction sprinkler system or manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of cable overhead. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers, in conjunction with the automatic fire suppression system, limit the fire from propagating beyond this fire area. An automatic suppression system is provided to increase the availability of non-safety related systems required to achieve cold shutdown.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system in this fire area is considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The electrical equipment in this area is non-Class 1E; however, some division B and D cables are routed through this area. In the event of a fire, the division B and D cabling in this area can be damaged. This damage can result in loss of control of equipment serviced by these cables. Other components in divisions B and D are not affected.

This fire can also disable the division B and D inputs to the reactor trip switchgear. The signals from the remaining two divisions are sufficient to trip the reactor. Furthermore, the reactor can be tripped with the diverse actuation system described in Section 7.7.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

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9A.3.1.2.6.2 Fire Area 1240 AF 01

The fire area is comprised of the following room:

Room No.	
12421	Non-Class 1E equipment/penetration room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. This subsystem may be restarted and manually realigned to the once-through smoke exhaust ventilation mode to minimize the potential migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of cable overhead. Automatic suppression is not installed in this fire area as an effort is made to limit water sources in this area for internal flooding concerns. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The electrical equipment in this area is non-Class 1E; however, some division A and C cables are routed through this area. In the event of a fire, the division A and C cabling in this area can be damaged. This damage can result in

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loss of control of equipment serviced by these cables. Other components in divisions A and C are not affected.

This postulated fire can disable control of the division A containment isolation valves outside containment. For this event, containment isolation is provided by the redundant containment isolation valves located inside containment outside of this fire area.

Such a fire can also disable control of the divisions A and C passive containment cooling system isolation valves. The redundant division B passive containment cooling system isolation valves are not affected. Therefore, the safe shutdown capability of the passive containment cooling system is maintained.

This fire can also disable the division A and C inputs to the reactor trip switchgear. The signals from the remaining two divisions are sufficient to trip the reactor. Furthermore, the reactor can be tripped with the diverse actuation system described in Section 7.7.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.7 Mechanical/Piping Areas

9A.3.1.2.7.1 Fire Area 1201 AF 04

This fire area consists of two nuclear island nonradioactive ventilation system equipment rooms servicing divisions B and D equipment rooms. Division B and D safe shutdown equipment is located within the fire area. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1241 AF 12405	12405	Lower nuclear island nonradioactive ventilation system divisions B and D equipment room (117'-6")
● 1251 AF 12505	12505	Upper nuclear island nonradioactive ventilation system divisions B and D equipment room (135'-3")

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area houses and is served by the division B & D Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Closure of these fire dampers interrupts the operation of the division B & D Class 1E electrical room HVAC subsystem. Operation of the independent division A & C Class 1E electrical room HVAC subsystem continues unaffected. Smoke is subsequently removed from the fire area by using portable exhaust fans and flexible ductwork. Smoke from a fire in this fire area does not

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affect safe shutdown components in fire areas that are served by other ventilation systems and subsystems. The division A and C electrical cables in fire area 1230 AF 01 are unaffected by the small quantity of smoke that may leak into that fire area. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are cable concentrations overhead in each room. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. The consequences of a break in a fire protection line in this fire area were considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area.

A fire in this fire area is conservatively assumed to disable control of all division B and D active components within the fire area, as well as the compressed and instrument air system (including the division B and D components of V014), component cooling water system, passive core cooling system and central chilled water system containment isolation valves within the fire area. The remaining division A, B, C components located in other fire areas are sufficient to perform the applicable functions to achieve and maintain safe shutdown. Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas. The redundant containment isolation valves in the lines for the compressed and instrument air system, component cooling water system, passive core cooling system and central chilled water system are located inside the containment (fire area 1000 AF 01) are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

The redundant steam generator 1 steam line pressure instrumentation located in fire zone 1100 AF 11300B is sufficient to perform the applicable functions to achieve and maintain safe shutdown.

The redundant steam generator 2 steam line pressure instrumentation located in fire zone 1100 AF 11300B is sufficient to perform the applicable functions to achieve and maintain safe shutdown.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

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9A.3.1.2.7.2 Fire Area 1201 AF 05

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 1241 AF 12506	12406	Lower main steam isolation valve compartment A
	12506	Upper main steam isolation valve compartment A
• 1231 AF 12306	12306	Valve/piping penetration room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

Each of the two fire zones in this fire area is served by independent air handling units of the annex/auxiliary buildings nonradioactive HVAC system (VXS), located within the fire zones served. No ventilating system ducts penetrate the MSIV compartment fire zone. One supply duct enters the valve piping penetration room from the turbine building. A fire damper in this duct closes automatically on high temperature to control the spread of fire and smoke. After the fire, smoke is removed from the fire area by using portable exhaust fans and flexible ductwork. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems or subsystems. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of cable overhead in each room. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The recirculation type ventilation system does not contribute to the spread of the fire or smoke to other fire areas.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. The consequences of a break in a fire protection line in this fire area were considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area.

A fire in this fire area is assumed to disable the safe shutdown components in the fire area. The redundant chemical and volume control system containment isolation valves located inside

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containment outside of this fire area are sufficient to perform the applicable functions to maintain containment integrity. The steam generator, main steam line, feedwater line, and blowdown line piping located inside containment outside of this fire area is sufficient to perform the applicable functions to maintain containment integrity.

A fire in this fire area is conservatively assumed to disable the division B PCS components located in fire zone 1270 AF 12701. See Subsection 9A.3.1.1.17 for the Safe Shutdown Evaluation for fire zone 1270 AF 12701.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.7.3 Fire Area 1201 AF 06

This fire area is comprised of the following room(s):

Room No.	
12404	Lower main steam isolation valve compartment B
12504	Upper main steam isolation valve compartment B

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by independent air handling units of the annex/auxiliary buildings nonradioactive HVAC system (VXS), located within the fire zones served. No ventilating system ducts penetrate the fire area. After the fire, smoke is removed from the fire area by using portable exhaust fans and flexible ductwork. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems or subsystems. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of cable overhead in each room. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The recirculation type ventilation system does not contribute to the spread of the fire or smoke to other fire areas.

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Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. The consequences of a break in a fire protection line in this fire area were considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area.

A fire in this fire area is assumed to disable the safe shutdown components in the fire area. The steam generator, main steam line, feedwater line, and blowdown line piping located inside containment outside of this fire area is sufficient to perform the applicable functions to maintain containment integrity.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.7.4 Fire Area 1250 AF 01

This fire area is comprised of the following room(s):

Room No.	
12501	Nuclear island nonradioactive ventilation system main control room/division A and C equipment room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area houses the nuclear island nonradioactive ventilation system (VBS) main control room/control support area supplemental air filtration units, one of two VBS main control room/control support area supply air handling units, and the VBS division A & C Class 1E electrical room HVAC subsystem air handling units. The division A & C Class 1E electrical room HVAC subsystem serves this fire area. Combination fire-smoke dampers in the division A & C Class 1E electrical room HVAC subsystem close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Closure of these combination fire-smoke dampers interrupts the operation of the VBS division A & C Class 1E electrical room HVAC subsystem. Operation of the independent division B & D Class 1E electrical room HVAC subsystem continues unaffected. In the event of a fire emergency in this fire area, operators may isolate the main control room and actuate the main control room emergency habitability system at their discretion to maintain control room habitability. Smoke is removed from the fire area by using portable exhaust fans and flexible ductwork.

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Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems or subsystems. The division B reactor coolant pump trip switchgear and the divisions B and D electrical cables located in fire area 1220 AF 01 and the divisions A and C electrical cables located in fire area 1200 AF 03 are unaffected by the small quantity of smoke that may leak into these fire areas. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector, which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. A fire in the fire area is extinguished manually using hose streams or portable extinguishers.

This fire area contains two charcoal adsorbers, located in the nuclear island nonradioactive ventilation system supplemental air filtration units. The normal temperature of the air flowing through the charcoal adsorbers is well below 200°F, while the minimum charcoal ignition temperature is greater than 600°F. Two independent temperature sensors interface with the fire detection system, providing charcoal temperature indication and high and high-high temperature alarms. The filtration unit fan trips at the high temperature alarm setpoint. The setpoints of both alarms are well below the charcoal ignition temperature, allowing the operator time to investigate and take corrective action. In the unlikely event of a fire in the adsorber, the filtration unit can be manually isolated. The adsorber bed deluge piping will be connected to a nearby hose station to cool the charcoal and extinguish the fire.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. Aside from the charcoal contained within the air filtration units, these materials primarily consist of electrical cable insulation for cables supplying fans and pumps within this fire area. There are cable concentrations near the ceiling. This is a light hazard fire area and the rate of fire growth is expected to be slow. Generally, three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. Fire barrier separation is not provided between the main control room and this fire area from fires in the main control room. There are no safe shutdown components in this room. There is fire barrier separation between the main control room and this fire area for fires originating in this area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. The consequences of a break in a fire protection line in this fire area were considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

The quantity and arrangement of the combustible materials in this fire area, and the characteristics of the barriers that separate this area from other fire areas, are such that a fire in this area does not propagate to any adjacent area.

Aside from the charcoal contained within the air filtration units, these materials primarily consist of electrical cable insulation for cables supplying fans and pumps within this fire area. There are cable concentrations near the ceiling. This is a light hazard fire area and the rate of fire growth is expected to be slow. Generally, 3-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. Fire barrier separation is not provided between the main

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control room and this fire area from fires in the main control room. There is fire barrier separation between the main control room and this fire area for fires originating in this area.

The safe shutdown components routed through this fire area are division A and division C electrical cables that serve the redundant passive containment cooling system valves and instruments located in the passive containment cooling system (PCS) valve room (Room 12701). Although the consequences of a fire are expected to be very limited, a fire in this fire area is conservatively assumed to disable both the division A and division C electrical cables that serve the PCS valves and instruments used for safe shutdown. The remaining division B electrical cables and components routed through and located in other fire areas are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.8 Miscellaneous Areas

9A.3.1.2.8.1 Fire Area 1230 AF 01

This fire area is comprised of the following room(s):

Room No.	
12300	Corridor

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is served by one of the three air distribution headers of the division B & D Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Closure of these dampers interrupts ventilation of adjacent fire areas 1201 AF 02 (division B electrical rooms) and 1232 AF 01 (remote shutdown room) because their ducts are routed through this corridor. The ventilation system continues to provide ventilation to other unaffected fire areas, and may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

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Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. The consequences of a break in a fire protection line in this fire area were considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The Class 1E divisions A and C cables for data communication between divisions are in this fire area.

In the event of a fire in the division A and C corridor, it is assumed that data output from divisions B and D to the division A and C protection logic is lost. This loss of data does not disable the safe shutdown functions of the four Class 1E divisions.

The division A and C instrument cables serving safe shutdown components are assumed to be disabled. The division B and D instrument cables routed outside of this fire area that serve redundant safe shutdown components are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.8.2 Fire Area 1201 AF 01

This fire area is comprised of the following room(s):

Room No.	
S02	Stairwell

The stairwell serving the northwest portion of the auxiliary building is enclosed by fire barrier walls having a minimum rating of 2 hours. The structural walls are concrete, and the nonstructural walls are made of a concrete/steel composite material. There are no safe shutdown components and no radioactive systems in this fire area. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

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Smoke intrusion to the stairwell is limited by fire barrier walls having a minimum rating of 2 hours, and self-closing doors having a minimum rating of 1.5 hours. Additionally, during a fire, a pressure difference across the doors in the stairwell is maintained in accordance with the guidance of NFPA 92A (Reference 4), using a dedicated stairwell pressurization fan.

9A.3.1.2.8.3 Fire Area 1202 AF 01

This fire area is comprised of the following room(s):

Room No.	
S01	Stairwell

The stairwell serving the northeast portion of the auxiliary building is enclosed by fire barrier walls having a minimum rating of 2 hours. The structural walls are concrete, and the nonstructural walls are made of a concrete/steel composite material. There are no safety-related components and no radioactive systems in this fire area. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

Smoke intrusion to the stairwell is limited by fire barrier walls having a minimum rating of 2 hours, and self-closing doors having a minimum rating of 1.5 hours. Additionally, during a fire, a pressure difference across the doors in the stairwell is maintained in accordance with the guidance of NFPA 92A (Reference 4), using a dedicated stairwell pressurization fan.

9A.3.1.2.8.4 Fire Area 1202 AF 05

This fire area is comprised of the following room(s):

Room No.	
S05	Stairwell

The stairwell provides an emergency egress path from the main control room to the emergency shutdown workstation. It is enclosed by 3-hour fire barriers. These barriers are made of concrete. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B.

Smoke intrusion to the stairwell is limited by fire barrier walls having a minimum rating of 2 hours, and self-closing doors having a minimum rating of 1.5 hours. No ventilation system serves this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The stairwell contains circuits for the transfer switch set which is used to transfer control from the control room to the remote shutdown workstation in the event of a control room evacuation. Electrical separation to and inside the stairwell is maintained per industry standards. In the unlikely event that the fire damages the transfer switch set, causing transfer of control from the main control room to the remote shutdown workstation, the operator restores control to the main control room by using controls located in the instrumentation and control rooms. Safe shutdown is achieved using the safe shutdown components listed in Table 9A-2. Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

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9A.3.1.2.8.5 Fire Area 1202 AF 02

This fire area contains an elevator and elevator shaft serving five levels on the north side of the auxiliary building. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detector
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

No ventilation system serves this fire area. The nuclear island nonradioactive ventilation system provides ventilation to neighboring fire areas, and may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke and hot gases. After the fire, smoke is removed from the fire area by using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. Two-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

Safe Shutdown Evaluation

This fire area contains no components required for safe shutdown after a fire. No safe shutdown evaluation is required.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.2.8.6 Fire Area 1200 AF 03

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 1230 AF 12311	12311	Elevation 100′-0″ corridor
• 1242 AF 12411	12411	Elevation 117′-6″ corridor

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

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Smoke Control Features

This fire area is served by one of the three air distribution headers of the division A & C Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. The ventilation system continues to provide ventilation to the unaffected fire areas. Fire areas 1220 AF 01 and 1210 AF 01 are served by the same air distribution header as this fire area, but they continue to receive ventilation because they have separate supply ducts. The ventilation system may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke and hot gases. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode. Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers. Safe shutdown components in these fire areas, identified in Table 9A-4, are sufficient to achieve and maintain safe shutdown.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. The consequences of a break in a fire protection line in this fire area were considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. Class 1E divisions A and C cables are routed through these corridors.

The divisions A and C control cables to the reactor trip switchgear are assumed to be disabled. Inputs from divisions B and D, which are routed in a separate fire area, are sufficient to trip the reactor when needed. The reactor can also be tripped with the diverse actuation system described in Section 7.7.

The divisions A and C instrument and control cables serving other safe shutdown components are assumed to be disabled. The instrument and control cables routed outside of this fire area that serve redundant safe shutdown components are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

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No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.3 Auxiliary Building - Radiologically Controlled Areas

The safe shutdown components located in the radiologically controlled areas are primarily containment isolation valves, which are located near the containment vessel in the lower annulus. Containment isolation valves are also located in the pipe chases southeast of containment. These containment isolation valves are required to either close or remain closed during a safe shutdown operation.

9A.3.1.3.1 Principal Areas

9A.3.1.3.1.1 Fire Area 1200 AF 01

This fire area includes most of the radiologically controlled areas of the auxiliary building outside the fuel handling area. This fire area contains one of the two normal residual heat removal pumps and the heat exchangers, the liquid radwaste system, spent fuel pool cooling system, radiologically controlled area ventilation system, chemical and volume control system makeup pump, containment isolation valve area, and lower annulus areas. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1200 AF 12241	12241	Lower annulus east
● 1200 AF 12241	12242	Lower annulus southeast
● 1200 AF 12461	12461	Corridor
● 1205 AF 12362	12362	Normal residual heat removal heat exchanger room
● 1205 AF 12365	12365	Waste monitor tank room B
● 1210 AF 12151	12151	Demineralizer/filter room
● 1210 AF 12151	12153	Delay bed/guard bed compartment
● 1210 AF 12151	12155	Gaseous radwaste system equipment room
● 1210 AF 12151	12156	Liquid radwaste system equipment room
● 1210 AF 12151	12158	Degasifier discharge pump room
● 1210 AF 12151	12258	Degasifier column
● 1210 AF 12171	12171	Effluent holdup tank room A
● 1214 AF 12152	12152	Primary sample room
● 1214 AF 12154	12154	Auxiliary building sump room
● 1214 AF 12154	12254	Spent fuel pool cooling system penetration room
● 1214 AF 12354	12354	Mid-annulus access room
● 1215 AF 12161	12161	Corridor
● 1215 AF 12162	12162	Normal residual heat removal pump room A
● 1216 AF 12166	12166	Waste holdup tank room A
● 1216 AF 12167	12167	Waste holdup tank room B
● 1216 AF 12169	12168	Vestibule

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- 1216 AF 12169 12169 Corridor
- 1216 AF 12169 12268 Liquid radwaste system pump room
- 1216 AF 12264 12264 Chemical waste tank room
- 1216 AF 12264 12265 Waste monitor tank room C
- 1216 AF 12172 12172 Effluent holdup tank room B
- 1220 AF 12251 12251 Demineralizer/filter access area
- 1220 AF 12251 12255 Chemical and volume control system makeup pump room
- 1220 AF 12259 12259 Pipe chase
- 1220 AF 12256 12256 Containment isolation valve area
- 1220 AF 12269 12269 Pipe chase
- 1220 AF 12256 12253 Pipe chase
- 1220 AF 12272 12272 Spent fuel pool cooling system pump room A
- 1220 AF 12272 12273 Spent fuel pool cooling system heat exchanger room A
- 1220 AF 12272 12274 Spent fuel pool cooling system pump room B
- 1220 AF 12272 12275 Spent fuel pool cooling system heat exchanger room B
- 1224 AF 12252 12252 Radiation chemistry laboratory
- 1225 AF 12261 12261 Corridor
- 1225 AF 12261 12271 Liquid radwaste system pump room
- 1225 AF 12262 12262 Piping/valve room
- 1234 AF 12351 12351 Maintenance floor staging area
- 1234 AF 12352 12352 Personnel hatch
- 1235 AF 12363 12363 Waste monitor tank room A
- 1244 AF 12451 12451 Security room
- 1244 AF 12452 12452 Containment air filtration system penetration room
- 1244 AF 12454 12454 Containment air filtration system/spent fuel pool cooling
- 1235 AF 12361 12361 Corridor
- 1250 AF 12561 12561 Component cooling water system valve room
- 1254 AF 12553 12553 Personnel access area
- 1254 AF 12554 12554 Security room
- 1264 AF 12651 12651 Radiologically controlled area ventilation system equipment room

The equipment and piping in this fire area normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
  - Line type fire detection (Heat) is provided for equipment protection in the upper-most cable tray in rooms 12151, 12155, 12156, 12251, and 12255
  - Spot type detection (Heat) is provided for a dedicated pump in each of the following rooms for equipment protection only: 12162, 12271, 12272, and 12274
  - Area fire detectors are provided in the following rooms: 12152, 12161, 12252, 12256, 12261, 12262, 12269, 12351, 12352, 12361, 12451, 12452, 12461, 12553, 12554, 12561 and 12651

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- Wet pipe sprinklers (Fire Zone 1220 AF 12251, room 12255)
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The radiologically controlled area ventilation system (VAS) serves this fire area on a once-through basis. Some of the ventilation system equipment is also located within this fire area. For a fire that does not disable the ventilation system, the system continues to ventilate the fire area unless the operator decides to shut down the system, until heat from the fire is sufficient to close the fire dampers, or if smoke is detected in the supply air ducts. Fire dampers close automatically on high temperature to control the spread of fire and smoke. If the radiologically controlled area ventilation system is not affected by the fire, smoke is removed from the fire area by reopening the fire dampers after a fire and exhausting to the atmosphere. If the radiologically controlled area ventilation system is unavailable, smoke is removed from the fire area using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by fire detectors which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by the wet pipe sprinkler system or manually using hose streams or portable extinguishers.

This fire area contains three charcoal adsorbers located in fire zone 1210 AF 12151, Delay Bed/Guard Bed Compartment (Room 12153). Independent temperature monitors located at the inlet and outlet of each charcoal bed and within the delay bed compartment provide high temperature alarms to detect a charcoal bed fire. Upon indication of a fire, the operator closes the Discharge Control Valve to stop effluent, stops WLS Degasifier Vacuum Pumps, stops the WLS Degasifier Vacuum Purge Pump, closes WLS RCDT vent valves, and opens the Nitrogen Purge to System Inlet Isolation Valve to purge the system with nitrogen and extinguish the fire.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. Aside from the charcoal contained in the WGS adsorber beds, these materials consist primarily of cable insulation for cables associated with the mechanical equipment and instrumentation in this fire area. There are concentrations of cable in the overhead cable trays in many fire zones. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas containing fire safe shutdown components and the fire is contained within the fire area. An automatic suppression system is provided to increase the availability of non-safety related systems required to achieve cold shutdown.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

The consequences of inadvertent operation of an automatic suppression system in this fire area or of fire suppression systems that drains to this fire area from the radwaste building, or of a break in a fire protection line in this fire area, are considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The electrical equipment in this area is non-Class 1E; however, some division A and C cables are routed through this area. In the event of a fire, the division A and C cabling in this area can be damaged. This damage can result in

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loss of control of equipment serviced by these cables. Other components in divisions A and C are not affected.

The normal residual heat removal, primary sampling system, spent fuel pool cooling system and containment air filtration system containment isolation valves are conservatively assumed to be disabled as a result of a fire in this fire area. The redundant normal residual heat removal, primary sampling system, spent fuel pool cooling system and containment air filtration system containment isolation valves located inside containment are outside of this fire area and are sufficient to perform the applicable functions to maintain containment integrity and achieve and maintain safe shutdown. Cable trays supplying these valves and other components are not required for safe shutdown.

Spurious DAS actuation of squib valves is prevented by the use of a squib valve controller circuit, which requires multiple hot shorts for actuation, physical separation of potential hot short locations, and provisions for operator action to remove power from the fire area. No postulated fire can spread to the hot short locations before the operator can remove power from the fire area.

The redundant passive containment cooling system safe shutdown components located in fire zone 1200 AF 12341 are sufficient to perform applicable functions to achieve and maintain safe shutdown.

Following detection of a fire in the non-Class 1E equipment/penetration room, the operators can close the automatic depressurization system stage 4 block valves, then remove DAS actuation power. This operator action will prevent spurious actuation of squib valves resulting from multiple hot shorts in the non-Class 1E equipment/penetration room.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.3.1.2 Fire Area 1200 AF 02

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 1200 AF 12562	12462	Cask washdown pit
● 1200 AF 12562	12463	Cask loading pit
● 1200 AF 12562	12472	New fuel storage pit
● 1200 AF 12562	12562	Fuel handling area
● 1200 AF 12562	12563	Spent fuel storage pool
● 1200 AF 12562	12564	Fuel transfer canal
● 1230 AF 12371	12371	Rail car bay/filter storage area
● 1230 AF 12371	12374	Waste disposal container area
● 1236 AF 12372	12372	Resin transfer pump/valve room
● 1236 AF 12373	12373	Spent resin tank room
● 1246 AF 12471	12471	Solid waste system valve/piping area

The spent fuel storage pool, spent fuel handling systems and components, and the solid radwaste rooms normally contain radioactive material.

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Fire Detection and Suppression Features

- Fire detectors
  - Area fire detectors are provided in the following room: 12562
- Wet pipe sprinklers (Fire Zone 1230 AF 12371, room 12371 rail car bay only)
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The radiologically controlled area ventilation system serves this fire area on a once-through basis. In the event of a fire the system continues to ventilate the fire area unless the operator decides to shut down the system, until heat from the fire is sufficient to close the fire dampers, or if smoke is detected in the supply air ducts. Fire dampers close automatically on high temperature to control the spread of fire and smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers. The radiologically controlled area ventilation system exhausts the smoke and hot gases to the atmosphere.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by the wet pipe sprinkler system or manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. Diesel fuel may be present when there is a truck in the rail car bay. There are concentrations of cable in the overhead cable trays. This is generally a light hazard fire area and the rate of fire growth is expected to be slow. Concentrations of transient combustibles in the rail car bay may produce a rapidly growing fire, so an automatic suppression system is provided in that area. Three-hour fire barriers provide adequate separation from adjacent fire areas. The roof and building exterior walls are unrated because there are no significant exposure fire hazards in nearby outdoor areas. Automatic or manual fire suppression activities prevent the fire from propagating beyond the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system or of a break in a fire protection line in this fire area are considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

There are no safe shutdown components in this area, so a fire in this area has no impact on safe shutdown. The electrical equipment in this area is non-Class 1E; however, some division A and C cables are routed through this area. In the event of a fire, the division A and C cabling in this area can be damaged. This damage can result in loss of control of equipment serviced by these cables. Other components in divisions A and C are not affected. Safe shutdown is possible from equipment in other fire areas.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

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9A.3.1.3.1.3 Fire Area 1204 AF 01

This fire area is comprised of the following room:

Room No.	
12163	Normal residual heat removal pump room B

Some of the piping in this fire area normally contains radioactive material.

Fire Detection and Suppression Features

- Fire detectors
  - Spot type detection (Heat) is provided for the motor and pump for equipment protection only
- Hose station(s)

Smoke Control Features

The radiologically controlled area ventilation system serves this fire area on a once-through basis. The system continues to ventilate the fire area unless the operator decides to shut down the system, until heat from the fire is sufficient to close the fire dampers, or if smoke is detected in the supply air ducts. Fire dampers close automatically on high temperature to control the spread of fire and smoke. Smoke is removed from the fire area by reopening the fire damper(s) after a fire and exhausting to the atmosphere.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are concentrations of cable in overhead cable trays. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. The consequences of a break in a fire protection line in this fire area were considered in the evaluation of internal flooding in Section 3.4.

Safe Shutdown Evaluation

This fire area contains no components required for safe shutdown after a fire. No safe shutdown evaluation is required.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

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9A.3.1.3.1.4 Fire Area 1220 AF 02

This fire area is comprised of the following room(s):

Room No.	
12244	Lower annulus valve area

Some of the piping in this fire area normally contains radioactive material.

Fire Detection and Suppression Features

- Fire detector
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The radiologically controlled area ventilation system serves this fire area on a once-through basis. In the event of a fire the system continues to ventilate the fire area unless the operator decides to shut down the system, until heat from the fire is sufficient to close the fire dampers, or if smoke is detected in the supply air ducts. Fire damper(s) close automatically on high temperature to control the spread of fire and smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers. The radiologically controlled area ventilation system exhausts the smoke and hot gases to the atmosphere.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. There are no significant concentrations of combustible materials. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The chemical and volume control system and liquid radwaste system containment isolation valves are conservatively assumed to be disabled as a result of a fire in this fire area. The redundant chemical and volume control system and liquid radwaste system containment isolation valves located inside containment are outside of this fire area and are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

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Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.1.3.2 Miscellaneous Areas

9A.3.1.3.2.1 Fire Area 1204 AF 02

This fire area is comprised of the following room(s):

Room No.	
S03	Stairwell

This fire area is the stairwell serving the shield building. The portion of the stairwell below the elevation of the auxiliary building roof is enclosed by fire barrier walls partially constructed of concrete and partially constructed of a concrete/steel composite material, having a minimum rating of 3 hours. Above the auxiliary building roof, the stairwell enclosure is not fire-rated. There are no radioactive systems in this fire area. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B.

No ventilation system directly serves this fire area.

9A.3.1.3.2.2 Fire Area 1204 AF 03

This fire area contains the elevator and elevator shaft serving the south side of the shield building. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

No ventilation system directly serves this fire area. After the fire, smoke is removed from the fire area by using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers provide adequate separation from adjacent fire areas below the elevation of the auxiliary building roof. Manual fire suppression activities prevent the fire from propagating beyond the nonrated building exterior walls above the roof.

Safe Shutdown Evaluation

There are no safe shutdown components in this fire area. No safe shutdown evaluation is required.

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9A.3.1.3.2.3 Fire Area 1205 AF 01

This fire area is comprised of the following room(s):

Room No.	
S04	Stairwell

This stairwell, serving the southeast portion of the auxiliary building, is enclosed by fire barrier walls having a minimum rating of 2 hours. The structural walls are concrete, and the nonstructural walls are made of a concrete/steel composite material. The only safety-related component located in this area is a safety-related and seismically qualified louver located in the barrier from the stairwell to Room 12161 that is opened based on differential pressure to provide flood mitigation. When closed, the louver is a 2-hour fire-rated/smoke-rated component in the fire-rated barrier. There are no radioactive systems in this fire area. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

Smoke intrusion to the stairwell is limited by fire barrier walls having a minimum rating of 2 hours, and self-closing doors having a minimum rating of 1.5 hours. No ventilation system serves this fire area.

9A.3.1.3.2.4 Fire Area 1205 AF 02

This fire area contains the elevator and elevator shaft serving the radiologically controlled area of the auxiliary building. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

No ventilation system serves this fire area. For a fire in the elevator or elevator shaft, smoke is removed from the fire area by using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3B. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area.

Safe Shutdown Evaluation

There are no safe shutdown components in this fire area. No safe shutdown evaluation is required.

9A.3.2 Turbine Building

Figure 9A-2 identifies fire areas and fire zones within the turbine building and illustrates the fire resistance of the fire area boundaries.

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A fire in the turbine building fire areas does not affect safe shutdown capability. Fire areas located in the turbine building are separated from the safety-related areas of the nuclear island by a 3-hour fire barrier wall. The closing of fire dampers in the ventilation systems serving turbine building fire areas does not affect safe shutdown systems because safe shutdown systems are served by independent ventilation systems.

Neither a fire nor fire suppression activities in turbine building fire areas affect the safe shutdown capability of components located in other fire areas.

Floor drains are sized to handle water flow from fixed automatic fire suppression systems without significant accumulation of water in the fire area. Flooding of components required for safe shutdown is not a concern because there are no safe shutdown components in the turbine building.

No fire in the turbine building can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.2.1 Fire Area 2000 AF 01

This fire area contains the main condenser, lubrication equipment, auxiliary boiler, turbine-generator and auxiliaries, switchgear rooms, electrical equipment room, feedwater pumps, chemical feed equipment, chiller area, plant air compressors, digital-electrohydraulic skid, main steam piping, office area, and a sampling laboratory. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 2030 AF 20300	20300	Elevation 100'-0" (base slab) general floor area
● 2030 AF 20300	20304	Auxiliary boiler area
● 2030 AF 20300	20305	Condensate polishing area
● 2030 AF 20300	20308	Control System Cabinet Room 1
● 2030 AF 20300	20309	Circulating water pipe trench
● 2030 AF 20300	20310	Security cage
● 2131 AF 21380	21380	CCS/BDS equipment room
● 2038 AF 20300	----	Main feedwater pump area
● 2039 AF 20301	20301	Chemical storage area
● 2040 AF 20400	20400	Elevation 120'-6" general floor area
● 2040 AF 20400	20408	Control System Cabinet Room 2
● 2040 AF 20400	20409	VWS Chiller Area
● 2141 AF 21480	21480	VTS HVAC equipment room
● 2043 AF 20402	20402	Excitation Equipment Room
● 2050 AF 20500	20500	Elevation 141'-3" general floor area
● 2050 AF 20500	20503	Control System Cabinet Room 4
● 2050 AF 20500	20518	Control System Cabinet Room 3
● 2050 AF 20502	----	Digital-electrohydraulic skid
● 2151 AF 21580	21580	South bay elevation 135'-3" general area
● 2151 AF 21583	21583	VFD equipment room

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- 2052 AF 20510 20510 HVAC equipment area
- 2057 AF 20511 20511 Generator seal oil unit
- 2060 AF 20600 20600 Elevation 170'-0" general floor area
- 2063 AF 20601 20601 Tool room/storage area
- 2063 AF 20602 20602 Office area/engineering workstation at elevation 183'-1 1/2"
- 2070 AF 20700 20700 Heater bay
- 2070 AF 20701 20701 Elevation 196'-3" North West Security Room
- 2070 AF 20702 20702 Elevation 196'-3" North East Security Room
- 2070 AF 20703 20703 Elevation 196'-3" South West Security Room
- 2070 AF 20704 20704 Upper heater bay

This fire area designation consists of the entire turbine building floor areas except those special areas contained within fire rated enclosures and discussed below as separate fire areas. In zones within this area where there is a potential for spills of oil or other combustibles, such as the digital-electrohydraulic skid, condensate polishing area, chemical storage area and the generator seal oil unit, spill control curbs are provided to limit the spread of fire should one occur. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors (See Table 9A-3C for type of detector.)

- Automatic suppression for the oil spill areas around the turbine-generator, the generator seal oil unit, the main feedwater pump area, the digital-electrohydraulic skid, the chemical storage area, the heater bay, and the upper heater bay.

  Automatic suppression for the following equipment: the service water pumps, the start-up feedwater pumps and MCCs and control equipment at elevation 141'-3" (in the area defined by column 13.1 to 14 and P.1 to O).

  Automatic suppression is provided over elevation 100′-0″ down the corridor between the condenser and the main feedwater pumps defined by column 13.1 to column 18, continuing down another corridor defined by column L.2 to column K.5 and for the equipment access area in fire zone 2030-AF-20300, room 20300.

  See Table 9A-3C for identification of the specific types of automatic suppression systems for this fire area.

- Hose station(s)

- Portable fire extinguishers

Smoke Control Features

The general area heating and ventilation subsystem of the turbine building ventilation system (VTS), as described in Subsection 9.4.9, uses roof mounted exhaust ventilators to pull in air through wall louvers. The roof mounted exhaust ventilators and/or portable ventilation equipment is used to vent smoke to outside areas and prevent migration of smoke to adjacent fire areas and to mitigate excessive buildup of smoke and heat at the ceiling above the turbine building operating deck. If the roof mounted exhaust ventilators are not available, smoke is removed from the fire area using portable exhaust fans and flexible ductwork.

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Fire Protection Adequacy Evaluation

A fire in this area is detected by the wet pipe type automatic sprinkler system or a fire detector. Upon sprinkler system or fire detector actuation, visual and audible alarms are produced in the main control room and at the remote fire alarm annunciator panel, and notification alarms are produced locally by the fire alarm detection system. The fire is extinguished by operation of the automatic suppression system, if applicable, or manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. This area contains fire zones with a low quantity of combustible materials and other zones which have moderate to high quantities of combustible materials. Zones containing a low quantity of low heat release combustibles are generally protected by manual suppression. Zones containing moderate to high quantities of high heat release combustibles, such as oil spill areas, are protected by automatic suppression.

The limited amount of combustible material or automatic fire suppression activities prevent fire from propagating to other fire zones in the area and prevent fire from propagating through the exterior walls or roof of the turbine building to other fire areas. The south end of the turbine building is formed by the 3-hour fire barrier exterior walls of the annex and auxiliary building. There are no concentrations of combustibles in the turbine building adjacent to these walls. Therefore, a turbine building fire will not propagate through these exterior walls of the annex or auxiliary building.

The turbine building ventilation system (VTS) serves the turbine building only and therefore does not contribute to the spread of the fire or smoke to fire areas outside the turbine building. The turbine building ventilation system does not contribute to the spread of fire or smoke to other fire areas within the turbine building because fire dampers isolate the other fire areas.

The turbine building ventilation subsystem is shut down upon detection of a fire in fire zone 2052 AF 20504 (Room 20510).

9A.3.2.2 Fire Area 2000 AF 02

This fire area is comprised of the following room(s):

Room No.	
S02	Stairwell
S08	Stairwell

These stairwells serve the southwest portion of the turbine building. The walls of these enclosures that are exposed to the turbine building interior are constructed with material having a minimum fire rating of 2 hours. The walls of the enclosures that face the yard area would not be exposed to the turbine building interior. Therefore, these outside walls are constructed with an exterior siding common to the overall siding used for the turbine building and are not required to be constructed or maintained to a minimum fire rating. There are no safety-related components and no radioactive systems in this fire area. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. A fire protection hose riser is located in stairwell S02 with NFPA Class I hose connections at intermediate stair landings.

These stairwells are not served by a ventilation system.

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9A.3.2.3 Fire Area 2009 AF 01

This fire area is comprised of the following room(s):

Room No.	
S01	Stairwell

This stairwell serves the northeast portion of the turbine building. The walls of this enclosure that are exposed to the turbine building interior are constructed with a material having a minimum fire rating of 2 hours. The walls of the enclosures that face the yard area would not be exposed to the turbine building interior. Therefore, these outside walls are constructed with an exterior siding common to the overall siding used for the turbine building and are not required to be constructed or maintained to a minimum fire rating. There are no safety-related components or systems in this fire area which contain radioactive material. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. A fire protection hose riser is located in the stairwell with NFPA Class I hose connections at intermediate stair landings.

This stairwell is not served by a ventilation system.

9A.3.2.4 Fire Area 2009 AF 02

This fire area is comprised of the elevator serving the turbine building from elevation 100'-0" to elevation 170'-0". The elevator is enclosed by fire barrier walls having a minimum rating of 2 hours. These nonstructural walls are metal lined gypsum board. There are no radioactive systems in this fire area.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

This fire area is not served by a ventilation system. After the fire, smoke is removed from the fire area by using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. This is a light hazard fire area and the rate of fire growth is expected to be slow. Two-hour fire barriers provide adequate separation from adjacent fire areas since the fire will be contained within this fire area.

No ventilation systems penetrate the elevator enclosure.

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9A.3.2.5 Not used.

9A.3.2.6 Not used.

9A.3.2.7 Fire Area 2000 AF 03

This fire area is comprised of the following room(s):

Room No.	
S03	Stairwell

This stairwell serves the northwest portion of the turbine building from 158'-7" to 196'-3". The walls of this enclosure that are exposed to the turbine building interior are constructed with a material having a minimum fire rating of 2 hours. The walls of the enclosures that face the yard area would not be exposed to the turbine building interior; therefore, these outside walls are constructed with an exterior siding common to the overall siding used for the turbine building. There are no safety-related components or systems in this fire area that contain radioactive material. There are no systems in this fire area that contain radioactive material. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. A fire protection hose riser is located in the stairwell with NFPA Class I hose connections at intermediate stair landings.

This stairwell is not served by a ventilation system.

9A.3.2.8 Not used.

9A.3.2.9 Fire Area 2040 AF 01

This fire area is comprised of the following room(s):

Room No.	
20407	Lube oil storage room

This fire area contains the clean and dirty lube oil storage tanks. There are no systems in this fire area which contain radioactive material.

Fire Detection and Suppression Features

- Wet pipe sprinklers
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

Local heating and ventilation for the lube oil storage room is supplied by the turbine building ventilation system (VTS) as described in Subsection 9.4.9. During normal operation, an exhaust ventilator mounted on an exterior wall pulls air in from the outside environment through an exterior wall louver, or through a damper in a wall separating the lube oil storage room from the rest of the turbine building. Fire dampers close automatically on high temperature to control the spread of fire and smoke. Following a fire, the exhaust ventilator, if available, can be used to vent smoke to outside the turbine building.

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Fire Protection Adequacy Evaluation

A fire in this area is detected by the wet pipe type automatic sprinkler system. Upon sprinkler system actuation, visual and audible alarms are produced in the main control room and at the remote fire alarm annunciator panel, and notification alarms are produced locally by the fire alarm detection system. The fire is extinguished by operation of an automatic sprinkler system or manually, using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. Due to the concentration of lubricating oil, this fire area is an extra hazard fire area. This area has a high combustible loading and fires with high heat release rates could develop rapidly. Therefore, an automatic fire suppression system is provided. The 3-hour fire barriers that separate this fire area from the rest of the turbine building, in conjunction with the automatic fire suppression system, limit the fire from propagating beyond this fire area.

The lube oil storage area ventilation system portion of the turbine building ventilation system does not contribute to the spread of the fire or smoke to other fire areas because fire dampers isolate the fire area.

9A.3.2.10 Fire Area 2043 AF 01

This fire area is comprised of the following room(s):

Room No.	
20401	Secondary sampling laboratory

There are no systems in this fire area which contain radioactive material.

Fire Detection and Suppression Features

- Wet pipe sprinklers
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The turbine building ventilation system (VTS) personnel area HVAC subsystem, as described in Subsection 9.4.9, provides heating and cooling to the secondary sampling laboratory. Fire dampers close automatically on high temperature to control the spread of fire and smoke. Smoke is removed from the fire area using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this area is detected by the wet pipe type automatic sprinkler system. Upon sprinkler system actuation, visual and audible alarms are produced in the main control room and at the remote fire alarm annunciator panel, and notification alarms are produced locally by the fire alarm detection system. The fire is extinguished by the automatic suppression system or manually, using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. As volatile chemicals are likely to be present, this fire area is an extra hazard fire area. Due to high heat release rates, fires could develop rapidly. Therefore, an automatic fire suppression system is provided. There are no concentrations of combustibles that could challenge the fire area barrier. The 3-hour fire barriers that separate this fire area from the rest of the turbine building provide sufficient separation to prevent the fire from propagating beyond the fire area.

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The VTS personnel area HVAC subsystem serving the secondary sampling laboratory does not contribute to the spread of the fire or smoke to other fire areas. Fire area boundaries are equipped with fire dampers to prevent the propagation of fire between fire areas.

9A.3.2.11 Fire Area 2050 AF 01

This fire area is comprised of the following room(s):

Room No.	
20504	Turbine lube oil reservoir room

There are no systems in this fire area which contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Automatic Pre-action system
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

Local heating and ventilation for the turbine lube oil reservoir room is supplied by the turbine building ventilation system (VTS) as described in Subsection 9.4.9. During normal operation, an exhaust ventilator mounted on an exterior wall pulls air in from the outside environment through an exterior wall louver, or through a damper in a wall separating the lube oil reservoir room from the rest of the turbine building. Fire dampers close automatically on high temperature to control the spread of fire and smoke. Following a fire, the exhaust ventilator, if available, can be used to vent smoke to outside the turbine building.

Fire Protection Adequacy Evaluation

A fire in this area is detected by a fire detector which produces an audible alarm locally with both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by operation of automatic sprinkler system or manually, using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. Due to the concentration of lubricating oil, this fire area is an extra hazard fire area. This area has a high combustible loading and fires with high heat release rates could develop rapidly. Therefore, an automatic fire suppression system is provided. The 3-hour fire barriers that separate this fire area from the rest of the turbine building, in conjunction with the automatic fire suppression system, limit the fire from propagating beyond the fire area.

The turbine lube oil reservoir ventilation system portion of the turbine building ventilation system does not contribute to the spread of the fire or smoke to other fire areas because fire dampers isolate the fire area.

9A.3.2.12 Fire Area 2052 AF 01

This fire area is comprised of the following room(s):

Room No.	
20502	Switchgear room 1

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This fire area is the turbine building switchgear room 1 and contains high voltage electrical equipment. There are no systems in this fire area which contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The turbine building ventilation system (VTS) electrical equipment rooms HVAC subsystem, as described in Subsection 9.4.9, provides heating and cooling to switchgear room 1. Fire dampers close automatically on high temperature to control the spread of fire and smoke. Smoke is removed from the fire area using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this area is detected by a fire detector which produces an audible alarm locally with both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. The fire area has a low concentration of combustibles, primarily cable insulation, with moderate heat release rates. The 2-hour fire barriers that separate this fire area from the rest of the turbine building provide sufficient separation to prevent the fire from propagating beyond the fire area.

The VTS electrical equipment rooms HVAC subsystem of the turbine building ventilation system serving switchgear room 1 does not contribute to the spread of the fire or smoke to other fire areas because fire dampers isolate the fire area.

9A.3.2.13 Fire Area 2053 AF 01

This fire area is comprised of the following room(s):

Room No.	
20513	Electrical equipment room

This fire area is the electrical equipment room and contains low voltage electrical equipment. There are no systems in this fire area which contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The turbine building ventilation system (VTS) electrical equipment rooms HVAC subsystem, as described in Subsection 9.4.9, provides heating and cooling to the electrical equipment room. Fire dampers close automatically on high temperature to control the spread of fire and smoke. Smoke is removed from the fire area using portable exhaust fans and flexible ductwork.

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Fire Protection Adequacy Evaluation

A fire in this area is detected by a fire detector which produces an audible alarm locally with both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. The fire area has a low concentration of combustibles, primarily cable insulation, with moderate heat release rates. There are no concentrations of combustibles that could challenge the fire area barrier. The 2-hour fire barriers that separate this fire area from the rest of the turbine building provide sufficient separation to prevent the fire from propagating beyond the fire area.

The VTS electrical equipment rooms HVAC subsystem of the turbine building ventilation system serving the electrical equipment room does not contribute to the spread of the fire or smoke to other fire areas because fire dampers isolate the fire area.

9A.3.2.14 Fire Area 2053 AF 02

This fire area is comprised of the following room(s):

Room No.	
20501	Switchgear room 2

This fire area contains high voltage electrical equipment. There are no systems in this fire area which contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The turbine building ventilation system (VTS) electrical equipment rooms HVAC subsystem, as described in Subsection 9.4.9, provides heating and cooling to switchgear room 2. Fire dampers close automatically on high temperature to control the spread of fire and smoke. Smoke is removed from the fire area using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this area is detected by a fire detector which produces an audible alarm locally with both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. The fire area has a low concentration of combustibles, primarily cable insulation, with moderate heat release rates. There are no concentrations of combustibles that could challenge the fire area barrier. The 2-hour fire barriers that separate this fire area from the rest of the turbine building provide sufficient separation to prevent the fire from propagating beyond the fire area.

The VTS electrical equipment rooms HVAC subsystem of the turbine building ventilation system serving switchgear room 2 does not contribute to the spread of the fire or smoke to other fire areas because fire dampers isolate the fire area.

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9A.3.2.15 Fire Area 2151 AF 01

This turbine building fire area contains non-Class 1E batteries. This fire area contains one fire zone:

Fire Zone	Room No.	
2151 AF 21581	21581	Battery room

There are no systems in this fire area that normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The south bay equipment HVAC subsystem of the turbine building ventilation system (VTS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. Other VTS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally, and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. The combustible materials are relatively uniformly distributed throughout the fire area. Automatic suppression is not installed in this fire area as an effort is made to limit water sources in the vicinity of electrical equipment. This is a light hazard fire area, and the rate of fire growth is expected to be slow. Minimum 2-hour fire barriers are provided.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.2.16 Fire Area 2151 AF 02

This turbine building fire area contains non-Class 1E battery charging equipment. This fire area contains one fire zone:

Fire Zone	Room No.	
2151 AF 21582	21582	Battery charger room

There are no systems in this fire area that normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

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Smoke Control Features

The south bay equipment HVAC subsystem of the turbine building ventilation system (VTS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally, and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C. The combustible materials are relatively uniformly distributed throughout the fire area. This is a light hazard fire area, and the rate of fire growth is expected to be slow. Minimum 2-hour fire barriers are provided.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.2.17 Fire Area 2000 AF 15

Room No.	
S15	Stairwell

This stairwell serves the northwest portion of the turbine building from 158'-7" to 100'-0". The walls of this enclosure that are exposed to the turbine building interior are constructed with a concrete/steel composite material having a minimum fire rating of 2 hours. The walls of the enclosure that face the yard area would not be exposed to the turbine building interior; therefore, these outside walls are constructed with an exterior siding used for the turbine building and are not required to be constructed or maintained to a minimum fire rating. There are no safety-related components or systems in this fire area that contain radioactive material. There are no systems in this fire area that contain radioactive material. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3C.

This stairwell is not served by a ventilation system.

9A.3.3 Yard Area and Outlying Buildings

The fire protection system yard main and the location of hydrants and hose houses are described in Subsection 9.5.1. Fire protection analysis discussions for the yard area, the administration building, and other outlying buildings are site-specific and are covered by subsection 9.5.1.8.

9A.3.4 Annex Building

Figure 9A-201 identifies fire areas and fire zones within the annex building and illustrates the fire resistance of the fire area boundaries.

A fire in the annex building fire areas does not affect safe shutdown capability. A fire is confined to the fire area, and fire areas within the annex building are separated from the safety-related areas of the nuclear island by a 3-hour fire barrier wall, except for those fire areas which include portions of the auxiliary building. Closing of fire dampers in the ventilation system serving the annex building fire

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areas does not affect safe shutdown systems. Safe shutdown systems are generally served by independent ventilation systems. Fire areas which include portions of the auxiliary building are discussed later in this subsection.

Neither a fire nor fire suppression activities in annex building fire areas affect the safe shutdown capability of components located in other fire areas.

Floor drains are sized to handle water flow from fire protection systems without a significant accumulation of water in the fire area. Flooding of components required for safe shutdown is not a concern because there are no safe shutdown components in the annex building.

No fire in the annex building can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.4.1 Fire Area 4001 AF 01

This fire area is comprised of the following room(s):

Room No.	
S01	Stairwell

This stairwell is enclosed by fire barrier walls having a minimum rating of 2 hours. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

Smoke intrusion to the stairwell is limited by fire barrier walls having a minimum rating of 2 hours, and self-closing doors having a minimum rating of 1.5 hours. No ventilation system serves this fire area.

9A.3.4.2 Fire Area 4001 AF 02

This fire area is comprised of the elevator shaft and elevator.

This elevator is enclosed by fire barrier walls having a minimum rating of 2 hours.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

A damper in the annex/auxiliary buildings nonradioactive HVAC system (VXS) vents the elevator shaft in the event of a fire. After the fire, smoke is removed from the fire area by using the exhaust vent, portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

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The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. This is a light hazard fire area and the rate of fire growth is expected to be slow. Two-hour fire barriers provide adequate separation from adjacent fire areas since the fire will be contained within this fire area.

9A.3.4.3A Fire Area 4002 AF 01

This fire area is comprised of the following room(s):

Room No.	
S02	Stairwell

The west wall of this stairwell is exposed to the annex building interior, and it is constructed with a material having a minimum fire rating of 2 hours. The walls of the enclosures that face the yard area would not be exposed to the annex building interior. Therefore, these outside walls are constructed with an exterior siding common to the overall siding used for the annex building. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

Smoke intrusion to the stairwell is limited by fire barrier walls having a minimum rating of 2 hours, and self-closing doors having a minimum rating of 1.5 hours. No ventilation system serves this fire area.

9A.3.4.3B Fire Area 4002 AF 03

This fire area is comprised of the elevator shaft and the following zones:

Fire Zone	Room No.	
4002 AF 40603	40603	Storage Room
4002 AF 40604	40604	Security Room
4002 AF 40604	40605	Mechanical Room

This elevator is enclosed by fire barrier walls having a minimum rating of 2 hours.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The health physics and hot machine shop (VHS) HVAC system serves this fire area on a once-through basis. In the event of a fire in the elevator shaft, a combination fire-smoke damper on the elevator shaft closes automatically in response to a smoke detector signal controlling the spread of fire, smoke, and hot gases. The VHS continues to ventilate the remaining fire areas unless the operator decides to shut down the system. If the exhaust fire-smoke damper for the elevator shaft is operable, the damper may be reopened to further reduce the migration of smoke. Smoke is removed from the elevator shaft by reopening the fire damper after a fire and operating the ventilation system to exhaust to the atmosphere or by using portable exhaust fans and flexible ductwork.

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Fire Protection Adequacy Evaluation

A fire in this area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustible materials in this fire area are listed in Table 9A-3. Combustibles in the elevator shaft consist primarily of cable insulation associated with the elevator hoisting machinery. Combustibles in the security and storage rooms consist primarily of ordinary combustible materials such as paper and plastic. This fire area is generally a light hazard area and the rate of fire growth is expected to be slow. The storage room could contain concentrations of transient combustibles, which could produce a rapidly growing fire. Two-hour fire barriers provide adequate separation between fire zones within this fire area and from adjacent fire areas since the fire will be contained within this fire area.

9A.3.4.4 Fire Area 4002 AF 02

This fire area is comprised of the following room(s):

Room No.	
S04	Stairwell

This stairwell serves the southeast corner of the annex building. The walls of this enclosure that are exposed to the annex building interior are constructed with a material having a minimum fire rating of 2 hours. The walls of the enclosures that face the yard area would not be exposed to the annex building interior. Therefore, these outside walls are constructed with an exterior siding common to the overall siding used for the annex building. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

Smoke intrusion to the stairwell is limited by fire barrier walls having a minimum rating of 2 hours, and self-closing doors having a minimum rating of 1.5 hours. No ventilation system serves this fire area.

9A.3.4.5 Fire Area 4003 AF 01

This fire area encompasses four levels of the annex building and contains demineralized water deoxygenating equipment and air handling equipment. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 4003 AF 40340	40340	Demineralized water deoxygenating room
• 4003 AF 40442	40442	Boric acid batching room
• 4003 AF 40503	40503	Lower south air handling equipment room
• 4003 AF 40503	40504	Air intake plenum 2
• 4003 AF 40503	40505	Security room
• 4003 AF 40601	40601	Upper south air handling equipment room
• 4003 AF 40601	40602	Air intake plenum 3

There are no systems in this fire area which normally contain radioactive material.

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Fire Detection and Suppression Features

- Fire detectors (except Rooms 40504 and 40602)
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The mechanical equipment areas HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. This action controls the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustibles are relatively uniformly distributed throughout each fire zone. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers provide adequate separation from adjacent fire areas. The building exterior wall is not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.6 Fire Area 4003 AF 02

This fire area is comprised of the following room(s):

Room No.	
S03	Stairwell

This stairwell serves the east side of the annex building. The walls of this enclosure that are exposed to the annex building interior are constructed with a material having a minimum fire rating of 2 hours. The walls of the enclosures that face the yard area would not be exposed to the annex building interior. Therefore, these outside walls are constructed with an exterior siding common to the overall siding used for the annex building. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

Smoke intrusion to the stairwell is limited by fire barrier walls having a minimum rating of 2 hours, and self-closing doors having a minimum rating of 1.5 hours. No ventilation system serves this fire area.

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9A.3.4.7 Fire Area 4031 AF 01

This annex building fire area contains one train of the non-Class 1E batteries and battery charging equipment. This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 4031 AF 40307	40307	Battery room 1
• 4031 AF 40308	40308	Battery charger room 1

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally, and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. The combustible materials are relatively uniformly distributed throughout the fire area. Automatic suppression is not installed in this fire area as an effort is made to limit water sources in the vicinity of electrical equipment. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided. The battery room is also separated from the adjacent charging room by a 1-hour fire barrier, which limits the spread of fire within the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.8 Fire Area 4031 AF 02

This annex building fire area contains one train of the non-Class 1E batteries, one additional battery bank, and battery charging equipment. This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 4031 AF 40309	40309	Battery room 2
• 4031 AF 40310	40310	Battery charger room 2
• 4031 AF 40411	40411	Computer room B
• 4031 AF 40412	40412	Battery room

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There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing fire zones 4031 AF 40309, 4031 AF 40310, and 4031 AF 40412 stops upon detection of smoke in the supply duct. Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to isolate fire zones 4031 AF 40309 and 4031 AF 40310. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. This subsystem may be restarted and manually realigned to the once-through smoke exhaust ventilation mode to minimize the potential migration of smoke. If the exhaust fire-smoke dampers for these fire zones are operable, the dampers may be reopened to further reduce the migration of smoke. The exhaust fan for fire zone 4031 AF 40412 continues to operate at the discretion of the operator. In the event of a fire emergency in fire zone 4031 AF 40412, operators may isolate the main control room and actuate the main control room emergency habitability system at their discretion to maintain control room habitability. After the fire, smoke is removed from these fire zones by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

The main control room/control support area HVAC subsystem of the NI non-radioactive ventilation system (VBS) serves fire zone 4031 AF 40411. The main control room/control support area HVAC subsystem will shut down and combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Operators may actuate the main control room emergency habitability system at their discretion to maintain control room habitability in the event closure of these fire-smoke dampers interrupts the operation of the main control room/control support area HVAC subsystem. The other VBS subsystems continue to provide ventilation to the unaffected fire areas. The subsystem may be manually realigned to the once-through ventilation mode to minimize the potential for migration of smoke and hot gases. If the exhaust fire-smoke damper for fire zone 4031 AF 40411 is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from the fire zone by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode.

Smoke from a fire in this fire area does not affect safe shutdown components in fire areas that are served by other ventilation systems, subsystems, or air distribution headers.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. The combustible materials are relatively uniformly distributed throughout the fire area. There are cable concentrations in the sub-floor of the computer room. Automatic suppression is not installed in this fire area as an effort is made to limit water sources in the vicinity of electrical equipment. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided. The battery room is also separated from the adjacent charging room by a 1-hour fire barrier, which limits the spread of fire within the fire area.

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The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.9 Fire Area 4031 AF 05

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 4031 AF 40301	40301	Access corridor
● 4031 AF 40301	40305	Security room
● 4031 AF 40301	40315	Security room
● 4031 AF 40303	40303	Corridor
● 4031 AF 40303	40304	Restroom

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- 2 Hose stations in corridor 40303
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustibles are relatively uniformly distributed throughout each fire zone. There are cable concentrations overhead in the corridor. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided. Fire zones within this fire area are separated by walls as shown in Figure 9A-201. The walls for the exit corridors are rated for a minimum of 1 hour in accordance with the Uniform Building Code. The building exterior walls are not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

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9A.3.4.10 Fire Area 4031 AF 06

This fire area is comprised of the following room(s):

Room No.	
40306	Security room
40302	Security room
40300	Security room

This fire area contains computer and communication equipment. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- 2 Hose stations in Corridor 40303
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustibles are relatively uniformly distributed throughout each fire zone. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.11 Fire Area 4032 AF 01

This fire area is comprised of the following room(s):

Room No.	
40326	Non-radiologically controlled area entry/exit area
40327	Health physics office
40350	Radiologically controlled area entry/exit area
40351	Protective clothes pickup and suitup
40352	Access corridor

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40354	Health physics counting
40355	Decontamination room
40356	Corridor
40360	Decontamination room
40361	Corridor

The area is used for decontamination and monitoring of personnel leaving the radiological control area of the plant. Low levels of radioactive materials may be present within the fire area.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The health physics and hot machine shop HVAC system (VHS) servicing this fire area stops upon detection of smoke in the area or in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers and starting the exhaust fans serving this area.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustibles are relatively uniformly distributed throughout the fire area. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided and the building exterior wall is not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.12 Fire Area 4032 AF 02

This fire area is comprised of the following room(s):

Room No.	
40357	Containment access corridor elevation 107′-2″
40362	Radwaste access corridor
40415	Security Room

This fire area contains the corridors used to transport equipment and personnel through the annex building and auxiliary building to and from containment and provides access from the annex building to the radwaste building. As such, low levels of radioactive material may be present within the fire area.

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Fire Detection and Suppression Features

- Fire detectors
- Wet pipe sprinklers (Elevation 107′-2″)
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The auxiliary/annex building subsystem of the radiologically controlled area ventilation system (VAS) servicing this fire area alarms in the main control room upon detection of smoke in the area or in the exhaust duct. If smoke is detected in the supply air ducts, an alarm is initiated in the main control room and the supply and exhaust fans for the corresponding train are automatically shut down. Fire dampers close automatically on high temperature to isolate this fire area. The balance of the VAS auxiliary/annex building ventilation subsystem remains in operation unless plant operators determine that there is a need to manually shut down the subsystem. The balance of VAS remains in operation. These actions control the spread of fire and smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers. The ventilation exhaust system serving the area exhausts smoke to the atmosphere.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by the wet pipe sprinkler system or manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Diesel fuel may be present when there is a truck in the containment access corridor. Combustibles are relatively uniformly distributed throughout the fire area except when transient combustibles are present. This is an ordinary hazard fire area when transient combustibles are present and the rate of fire growth may be rapid. Minimum two-hour fire barriers are provided and the building exterior wall is not rated. An automatic suppression system is provided to address transient combustibles.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.13 Fire Area 4033 AF 01

This fire area is comprised of the following room(s):

Room No.	
40358	Hot machine shop
40359	Pump seal shop

The shop is used for decontamination and repair of equipment from the radiological control area of the plant. As such, low levels of radioactive materials may be present within the fire area.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

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Smoke Control Features

The health physics and hot machine shop HVAC system (VHS) servicing this fire area stops upon detection of smoke in the area or in the duct. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers and starting the exhaust fans serving this area.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustibles are relatively uniformly distributed throughout the fire area. This is an ordinary hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided. The building exterior wall is not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.14 Fire Area 4034 AF 01

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 4034 AF 40311	40311	Corridor
• 4034 AF 40311	40312	Corridor
• 4034 AF 40311	40373	Corridor
• 4034 AF 40311	40380	Entry
• 4034 AF 40313	40313	Office
• 4034 AF 40313	40314	Office
• 4034 AF 40316	40316	Computer Room L1/2
• 4034 AF 40317	40317	Computer Room L3
• 4034 AF 40318	40318	ALARA briefing room and HP monitoring room
• 4034 AF 40320	40320	Women's change room
• 4034 AF 40322	40321	Janitor closet
• 4034 AF 40322	40322	Men's change room
• 4034 AF 40322	40323	Water heater room
• 4034 AF 40322	40324	Drying area
• 4034 AF 40322	40325	Shower room
• 4034 AF 40370	40370	Rest room
• 4034 AF 40370	40371	Conference room
• 4034 AF 40370	40372	Conference room
• 4034 AF 40370	40374	Corridor
• 4034 AF 40370	40375	Conference room

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- 4034 AF 40370 40376 Kitchen
- 4034 AF 40370 40377 Rest room
- 4034 AF 40378 40378 Office area west
- 4034 AF 40378 40381 Fire Brigade Locker
- 4034 AF 40379 40379 Office area east
- 4034 AF 40379 40382 Radiation Protection (RP) Calibration Lab
- 4034 AF 40379 40383 Break Area
- 4034 AF 40379 40384 RP Technician Work Area
- 4034 AF 40379 40385 Office
- 4034 AF 40379 40386 Corridor

The radiation protection calibration lab includes a sealed radiation source (see Subsection 12.2.1.1.10) located inside shielded calibration equipment. There are no other systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The general area HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate portions of this fire area. These actions control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, an exhaust fan serving the change rooms, and rest rooms mounted on the roof over the fire area, is used to exhaust smoke to the atmosphere from the change rooms and rest rooms. Smoke from other areas may be exhausted using portable exhaust fans and flexible ductwork.

Fire zone 4034 AF 40311 is an exit corridor for the area and is protected with fire dampers on duct penetrations of the corridor envelope in accordance with the Uniform Building Code.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustibles are relatively uniformly distributed throughout the fire area. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided and the building exterior wall is not rated. Fire zones within this fire area are separated by walls as shown in Figure 9A-201. The walls of fire zone 4034 AF 40311 are rated for 1-hour in accordance with Uniform Building Code requirements for exit corridors.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

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9A.3.4.15 Fire Area 4035 AF 01

This fire area is comprised of the following room(s):

Room No.	
40341	Ancillary diesel generator room

This annex building fire area contains two diesel generator sets and one fuel storage tank. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Automatic dry pipe sprinkler system
- Hose station(s)
- Portable fire extinguishers
- Fire detectors

Smoke Control Features

The mechanical equipment areas HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. These actions control the spread of fire and smoke. After the fire, smoke is removed from this fire area by using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this area is detected through the operation of the automatic dry pipe sprinkler system and/or a heat detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by the automatic dry pipe sprinkler system. The sprinklers provide protection for the room and the walled pit under the floor of the room that is used to capture the sprinkler system water and the contents of the fuel storage tank. If necessary the fire can also be extinguished manually.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustible materials primarily consist of diesel fuel oil in the fuel oil storage tank. The tank is partially located in the walled spill containment pit under a grated section of the floor and extends above the floor into the room. The walled spill containment pit has no drains and no doors. The remaining combustibles are relatively uniformly distributed throughout the fire area. This is an ordinary hazard fire area and the rate of heat release is expected to be moderate to high. Minimum three-hour fire barriers are provided and the building exterior wall is not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.16 Fire Area 4041 AF 01

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 4041 AF 40403	40403	Control support area
● 4041 AF 40403	40404	Restroom

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- 4041 AF 40403 40405 Rest area, kitchen
- 4041 AF 40403 40406 Conference room
- 4041 AF 40403 40407 Conference room
- 4041 AF 40403 40408 NRC room
- 4041 AF 40403 40409 Conference room
- 4041 AF 40410 40410 Computer room A
- 4041 AF 40410 40402 Corridor

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The main control room/control support area HVAC subsystem of the VBS will shut down and combination fire-smoke dampers in this subsystem close automatically upon detection of smoke and/or on high temperature to isolate this fire area. The main control room/control support area HVAC subsystem will shut down upon detection of a fire in this fire area. This subsystem may be manually realigned to the once-through smoke exhaust ventilation mode to minimize the potential migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the once-through ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. Combustibles are relatively uniformly distributed throughout the fire area. There are cable concentrations in the sub-floor of the computer room. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided and the building exterior wall is not rated. Fire zones within this fire area are separated by walls as shown in Figure 9A-201. The corridor walls of fire zone 4041 AF 40410 are rated for one-hour in accordance with Uniform Building Code requirements for exit corridors.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.17 Fire Area 4041 AF 02

This fire area is comprised of the following room(s):

Room No.	
40400	Corridor
40401	Restroom

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There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode. In the event of a fire emergency in this fire area, operators may isolate the main control room and actuate the main control room emergency habitability system at their discretion to maintain control room habitability.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. There are cable concentrations near the ceiling. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.18 Fire Area 4042 AF 01

This fire area is comprised of the following room(s):

Room No.	
40413	Electrical switchgear room 1 including vertical electrical chase from room 40413 to room 40350 (floor slab Elevation 100′-0″)

This annex building fire area contains the non-Class 1E switchgear for one train, and two motor generator sets with their two associated control cabinets. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The switchgear room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply or return duct. Fire dampers in the system close automatically on high temperature to isolate this fire area. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is

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removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. There are cable concentrations near the ceiling and the exterior wall. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers provide adequate separation from adjacent fire areas. The building exterior wall is not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.4.19 Fire Area 4042 AF 02

This fire area is comprised of the following room(s):

Room No.	
40414	Electrical switchgear room 2

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The switchgear room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply or return duct. Fire dampers in the system close automatically on high temperature to isolate this fire area. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally, and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. There are cable concentrations near the ceiling and the exterior wall. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers provide adequate separation from adjacent fire areas. The building exterior wall is not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

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9A.3.4.20 Fire Area 4051 AF 01

This fire area is comprised of the following room(s):

Room No.	
40500	North air handling equipment room
40501	Air intake plenum 1

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors (except Room 40501)
- Hose station(s)
- Portable fire extinguishers
- Automatic wet pipe sprinklers (Room 40500 only)

Smoke Control Features

This fire area houses the annex/auxiliary building non-radioactive ventilation system (VXS) switchgear room HVAC subsystem supply air handling units and the VXS equipment room HVAC subsystem supply air handling units, as well as one of two VBS main control room/control support area HVAC subsystem air handling units. The VXS equipment room HVAC subsystem servicing this fire area stops upon detection of smoke in the supply or return duct. Fire dampers in the system close automatically on high temperature to isolate this fire area. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. After the fire, smoke is removed from this fire area by using portable exhaust fans and flexible ductwork or by reopening the fire dampers and operating the ventilation system in the smoke exhaust mode if it is still functional.

Combination fire-smoke dampers in this area that close automatically in response to heat from a fire isolate the outdoor air intake to the main control room/control support area HVAC subsystem air handling unit in this area. In the event of a fire emergency in this fire area, operators may isolate the main control room and actuate the main control room emergency habitability system at their discretion to maintain control room habitability.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by operation of an automatic sprinkler or by manually using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. There are cable concentrations along the west and south walls. This is a light hazard zone and the rate of fire growth is expected to be slow. Minimum two-hour fire barriers are provided and the exterior wall is not rated.

The ventilation systems contained in this fire area are equipped with fire dampers on their duct penetrations out of and into this fire area. The systems contained in this area are recirculating HVAC systems serving personnel and equipment areas in the annex building, and one train of HVAC equipment serving the main control room.

The ventilation system does not contribute to the spread of fire or smoke as described in the Smoke Control Features section above.

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9A.3.4.21 Fire Area 4052 AF 01

This fire area contains the staging, storing, and assembly area for the containment. It also houses the exhaust fans for the health physics area and the containment air filtration exhaust units. The containment air filtration units remove radioactive halogens and particulates from the containment air prior to discharge to the atmosphere. Low levels of radioactive material may be present within the fire area during normal plant operation. This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 4052 AF 40550	40550	Staging and storage area
● 4052 AF 40551	40551	Containment air filtration exhaust room A
● 4052 AF 40552	40552	Containment air filtration exhaust room B

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers
- Automatic wet pipe sprinklers (Room 40550 only)

Smoke Control Features

The auxiliary/annex building subsystem of the radiologically controlled area ventilation system (VAS) servicing this fire area alarms in the main control room upon detection of smoke in the area or in the exhaust duct. If smoke is detected in the supply air ducts or if a fire is detected in room 40550, an alarm is initiated in the main control room and the supply and exhaust fans for the corresponding train (or both trains for a fire detected in room 40550) are automatically shut down. The balance of the this and other VAS subsystems remain in operation unless plant operators determine that there is a need to manually shut down the subsystem. Fire dampers close automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers. The ventilation exhaust system serving the area exhausts smoke to the atmosphere.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by operation of an automatic sprinkler or manually using hose streams or portable extinguishers.

This fire area contains two charcoal adsorbers, located in the containment air filtration system exhaust units. The normal temperature of the air flowing through the charcoal adsorbers is well below 200°F, while the minimum charcoal ignition temperature is greater than 600°F. Two independent temperature sensors interface with the fire detection system, providing charcoal temperature indication, and high and high-high temperature alarms. The filtration unit fan trips at the high temperature alarm setpoint. The setpoints of both alarms are well below the charcoal ignition temperature, allowing the operator time to investigate and take corrective action. In the unlikely event of a fire in the adsorber, the filtration unit can be manually isolated. The adsorber deluge piping will be connected to a nearby hose station to cool the charcoal and extinguish the fire.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3D. The charcoal is contained within the sheet metal housings of the containment air filtration system exhaust units. This is a light hazard fire area and the rate of fire growth is

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expected to be slow. Minimum two-hour fire barriers are provided and the building exterior wall is not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.5 Radwaste Building

Figure 9A-4 identifies fire zones within the radwaste building fire area and illustrates the fire resistance of the fire area boundaries.

A fire in the radwaste building does not affect safe shutdown capability. A fire is confined to the fire area. The radwaste building fire area is separated from the safety-related areas of the nuclear island by a 3-hour fire barrier wall. The radwaste building is served by the dedicated radwaste building HVAC system. Closing of fire dampers in the ventilation system serving the radwaste building does not affect safe shutdown systems because the safe shutdown systems are served by other ventilation systems.

Neither a fire nor fire suppression activities in the radwaste building affect the safe shutdown capability of components located in other fire areas.

Floor drains are sized to handle water flow from fixed automatic fire protection systems without a significant accumulation of water in the fire area. Curbed areas within the radwaste building have sufficient capacity to retain fire protection water to prevent an unmonitored release to the environment.

No fire in the radwaste building can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

9A.3.5.1 Fire Area 5031 AF 01

The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 5031 AF 50300	50300	Electrical/mechanical equipment room
● 5031 AF 50350	50350	Mobile systems facility
● 5031 AF 50351	50351	Waste accumulation room
● 5031 AF 50353	50353	HVAC equipment room
● 5031 AF 50354	50354	Truck staging area
● 5031 AF 50355	50355	Monitor tanks room

Various radwaste processing and packaging operations are performed utilizing the mobile system facilities. Moderate quantities of radioactive materials are present in the fire area during all modes of plant operation.

Fire Detection and Suppression Features

- Fire detectors
- Preaction sprinklers
- Hose station(s)
- Portable fire extinguishers

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Smoke Control Features

The radwaste building HVAC system (VRS) stops and an alarm is sent to the main control room and to the remote fire alarm annunciator panel upon actuation of the fire suppression system in this fire area or if smoke is detected in either supply duct downstream from the air handling units. The VRS HVAC will be shut down automatically by the FPS. The fire damper to the plant vent closes automatically on high temperature to isolate this fire area. These actions control the spread of fire and smoke. After the fire, smoke is removed from the fire area and exhausted to atmosphere by reopening the fire damper and operating the ventilation system exhaust fans.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by the preaction sprinkler system or manually, using hose streams or portable extinguishers.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3E. There are concentrations of combustibles in the waste accumulation room and potentially in the mobile systems facility. Diesel fuel may periodically be present when there is a truck present in the truck staging area. Depending upon the processes being performed in the mobile systems facility, the locations of combustible concentrations may change. This is a light hazard fire area and the rate of fire growth is expected to be slow. Minimum three-hour barriers are provided and the building exterior wall is not rated. An automatic suppression system is provided due to the localized areas of high combustible loading. This system provides confidence that the fire will be promptly extinguished, thus minimizing the potential for release of radioactivity and the radiation exposure of firefighters.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.6 Diesel Generator Building

Figure 9A-5 identifies fire areas and fire zones within the diesel generator building and illustrates the fire resistance of the fire area boundaries.

A fire in the diesel generator building does not affect safe shutdown capability. The diesel generator building is not adjacent to any building or area containing safety-related equipment. The diesel generator building heating and ventilation system is dedicated to the diesel generator building and independent of other ventilation systems.

Neither a fire nor fire suppression activities in the diesel generator building fire areas affect the safe shutdown capability of components located in other fire areas.

Floor drains are sized to handle low water flow rates. Water from fixed automatic fire protection systems flows out of the building through opened doors. Flooding of components required for safe shutdown is not a concern because there are no safe shutdown components in the diesel generator building.

No fire in the diesel building can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.

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9A.3.6.1 Fire Area 6030 AF 01

This fire area contains the diesel generator and supporting equipment for one train of the onsite standby ac power system. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 6030 AF 60310	60310	Diesel generator room A
• 6030 AF 60311	60311	Service module A
• 6030 AF 60313	60313	Combustion air cleaner area A

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors in the service module, diesel generator room, and combustion air cleaner
- Dry pipe sprinklers in the diesel generator room
- Hose station(s)
- Portable fire extinguishers (including water-mist)

Smoke Control Features

The diesel generator building ventilation system (VZS) serves this fire area by means of the engine room air handling unit, the service module air handling unit, and the standby exhaust fans. The systems only serve a single room and can not spread smoke or hot gases outside of the affected fire area. If these systems are operating, they will continue to do so until manually shut down; manual action is preferred to prevent normal generator operation (engine radiant heat and smoke from exhaust) from tripping the ventilation systems unnecessarily. The standby exhaust fans, if already running, will continue to run to help remove smoke; the exhaust fans do not start automatically on a fire signal. These actions control the spread of fire and smoke. After the fire, smoke is removed from the fire area by manually turning on, if available, the standby exhaust fans mounted on the roof over the fire area, or by opening the roll-up door and personnel doors and utilizing portable exhaust fans.

Fire Protection Adequacy Evaluation

A fire in the diesel generator room is detected through the operation of the dry pipe sprinkler system and/or a heat detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. A fire in the service module is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. A fire in the diesel generator room is extinguished by the automatic fire suppression system or manually, using hose streams or portable extinguishers. A fire in the service module is extinguished manually using hose streams or portable extinguishers.

The area under the diesel generator is shielded from direct impingement of the spray from the dry pipe sprinkler system, but water accumulating on the floor will find its way into this space. The area under the diesel generator is also accessible for manual firefighting. The roll-up door permits access to this space for manual hose streams from outside the building.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3F. Combustibles concentrations occur at the diesel generator equipment and in the service module. This is an ordinary hazard fire area and the rate of heat release is expected to be moderate to high. Minimum three-hour fire barriers are provided and the building exterior walls are not rated.

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The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.6.2 Fire Area 6030 AF 02

This fire area contains the diesel generator and supporting equipment for one train of the onsite standby ac power system, tool storage area and security room. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
● 6030 AF 60320	60320	Diesel generator room B
● 6030 AF 60321	60321	Service module B
● 6030 AF 60323	60323	Combustion air cleaner area B
● 6030 AF 60324	60324	Tool storage room
● 6030 AF 60330	60330	Security room

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors in the combustion air cleaner, diesel generator room, service module, and security room
- Dry pipe sprinklers in the diesel generator room and tool storage room
- Hose station(s)
- Portable fire extinguishers (including water-mist)

Smoke Control Features

The diesel generator building ventilation system (VZS) serves this fire area by means of the engine room air handling unit, the service module air handling unit, and the standby exhaust fans. The systems only serve a single room and can not spread smoke or hot gases outside of the affected fire area. If these systems are operating, they will continue to do so until manually shut down; manual action is preferred to prevent normal generator operation (engine radiant heat and smoke from exhaust) from tripping the ventilation systems unnecessarily. The standby exhaust fans, if already running, will continue to run to help remove smoke; the exhaust fans do not start automatically on a fire signal. These actions control the spread of fire and smoke. After the fire, smoke is removed from the fire area by manually turning on, if available, the standby exhaust fans mounted on the roof over the fire area, or by opening the roll-up door and personnel doors and utilizing portable exhaust fans.

Fire Protection Adequacy Evaluation

A fire in the diesel generator room is detected through the operation of the dry pipe sprinkler system and/or a heat detector which produces an audible alarm locally, and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. A fire in the service module is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. A fire in the diesel generator room is extinguished by the automatic fire suppression system or manually, using hose streams or portable extinguishers. A fire in the service module is extinguished manually using hose streams or portable extinguishers.

The area under the diesel generator is shielded from direct impingement of the spray from the dry pipe sprinkler system, but water accumulating on the floor will find its way into this space. The area

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under the diesel generator is also accessible for manual firefighting. The roll-up door permits access to this space for manual hose streams from outside the building.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3F. Combustibles concentrations occur at the diesel generator equipment and in the service module. This is an ordinary hazard fire area and the rate of heat release is expected to be moderate to high. Minimum three-hour fire barriers are provided and the building exterior walls are not rated.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.6.3 Fire Area 6030 AF 03

This fire area is comprised of the following room(s):

Room No.	
60312	Diesel fuel day tank vault A

This fire area contains the fuel oil day tank for one train of the onsite standby ac power diesel generator. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Dry pipe sprinklers
- Hose station(s)
- Portable fire extinguishers
- Fire detectors

Smoke Control Features

Fire dampers in the diesel generator building ventilation system (VZS) close automatically on high temperature to isolate this fire area. The tank vault exhaust fan stops upon actuation of the fire suppression system in this fire area. These actions control the spread of fire and smoke. After the fire, smoke is removed from the fire area using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected through the operation of the dry pipe sprinkler system and/or a heat detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by the automatic dry pipe sprinkler system. Water from the sprinklers rapidly fills and cools the small diked area under the tank. If necessary, the fire can also be extinguished manually.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3F. Due to the small vault size and the size of the storage tank, the combustible loading is uniform throughout the fire area. This is an ordinary hazard fire area and the rate of heat release is expected to be high. Minimum three-hour fire barriers are provided.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

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9A.3.6.4 Fire Area 6030 AF 04

This fire area is comprised of the following room(s):

Room No.	
60322	Diesel fuel day tank vault B

This fire area contains the fuel oil day tank for one train of the onsite standby ac power diesel generator. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Dry pipe sprinklers
- Hose station(s)
- Portable fire extinguishers
- Fire detectors

Smoke Control Features

Fire dampers in the diesel generator building ventilation system (VZS) close automatically on high temperature to isolate this fire area. The tank vault exhaust fan stops upon actuation of the fire suppression system in this fire area. These actions control the spread of fire and smoke. After the fire, smoke is removed from the fire area using portable exhaust fans and flexible ductwork.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected through the operation of the dry pipe sprinkler system and/or a heat detector which produces an audible alarm locally and both visual and audible alarms in the main control room and at the remote fire alarm annunciator panel. The fire is extinguished by the automatic dry pipe sprinkler system. Water from the sprinklers rapidly fills and cools the small diked area under the tank. If necessary, the fire can also be extinguished manually.

The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3F. Due to the small vault size and the size of the storage tank, the combustible loading is uniform throughout the fire area. This is an ordinary hazard fire area and the rate of heat release is expected to be high. Minimum three-hour fire barriers are provided.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

9A.3.6.5 Fire Area 6001 AF 01

This fire area is comprised of the following rooms:

Room No.	
S01	Stairwell

The stairwell serves the east side of the diesel generator building. The walls of this enclosure that are exposed to the building interior are constructed with a material having a minimum fire rating of 2 hours. The exterior walls of the enclosures would not be exposed to the diesel generator building interior. Therefore, these outside walls do not require the 2-hour fire rating. The primary combustible materials, fire severity, and equivalent fire duration in this fire zone are listed in Table 9A-3F.

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Fire Detection and Suppression Features

- Dry pipe sprinklers
- Fire detectors

9A.3.7 Special Topics

9A.3.7.1 Evaluation of Spurious Actuation

The potential for spurious actuation of equipment as a result of fire damage to electrical circuits is considered for each fire area containing safety-related equipment. As discussed in Subsection 9A.2.7.1, an assessment of multiple/simultaneous spurious actuations or signals resulting from the fire was performed (Reference 6) with the conclusion that either spurious actuations do not occur, or the consequences are such that safe shutdown can still be achieved. Principal spurious actuation are discussed below. In no case does the spurious actuation of equipment prevent safe shutdown.

9A.3.7.1.1 High-Low Pressure Interfaces

NRC Generic Letter 81-12 requests the identification and evaluation of high-low pressure interfaces between the reactor coolant system and interfacing systems such as the normal residual heat removal system. Per the Generic Letter, these interfaces typically contain two redundant and independent motor-operated valves in series. On a typical pressurized water reactor plant, these two valves and their control and power cables may be subject to a single fire. Potential high-low pressure system interfaces of particular interest are discussed below.

Reactor Coolant System Valve Actuation

NRC Generic Letter 81-12 specifically addresses the reactor coolant/residual heat removal system interface on pressurized water reactors. For AP1000, the reactor coolant system to normal residual heat removal system interface is similar to the typical pressurized water reactor configuration. However, the normal residual heat removal system is not a safety-related system and is not required for safe shutdown. To preclude the spurious opening of the interface valves as a result of a fire, the power to the valves is locked out during power operations. Thus, spurious actuation of the reactor coolant system to normal residual heat removal system interface valves does not occur and the safe shutdown capability is not affected.

Automatic Depressurization System Valve Actuation

The automatic depressurization system valves are not considered to be high-low pressure interface valves when postulating spurious actuation following a fire. The safety issue related to high-low pressure interfaces is expressed in NRC Generic Letter 81-12. The concern is that the spurious opening of two or more isolation valves which form the boundary between the reactor coolant system and a low pressure system could lead to damage to the low pressure system and a loss of coolant outside the containment. Since automatic depressurization system valve actuation cannot damage a low pressure system, and since the system is entirely within containment, the automatic depressurization valves do not represent a high-low pressure interface as described in NRC Generic Letter 81-12.

Spurious actuation of automatic depressurization system stage 4 squib valves is prevented by the use of a squib valve controller circuit which requires multiple hot shorts for actuation, physical separation of potential hot short locations, and provisions for operator action to remove power from the fire zone. No postulated fire can spread to the hot short locations before the operator can remove power from the fire zone.

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Automatic depressurization system stages 1, 2, and 3 consist of parallel paths, each path having two motor-operated valves in series. Spurious stage 1, 2, or 3 actuation is prevented by the use of physical separation of control circuits for the two series valves and provisions for operator action to remove power from the fire zone. No postulated fire can spread to the hot short locations before the operator can remove power from the fire zone.

Reactor Coolant System Reactor Vessel Head Vent Valve Actuation

The reactor vessel head vent valves are connected to the reactor vessel head and discharge to the IRWST. The head vent valves are not required to operate following a fire. There are four head vent valves arranged in two flow paths with two series valves in each path. The head vent valves are fail-closed dc powered solenoid valves, and each valve is powered by a separate, safety-related power supply as shown on Table 9A-2. In the event that a spurious signal was to open a head vent valve, the flow path is blocked by the closed series head vent valve. The cables for the control of one head vent valve in each flow path is enclosed in steel conduit up to the valve to prevent a fire inside containment from spuriously actuating two head vent valves in one flow path. Therefore, a single fire is not postulated to result in an uncontrolled LOCA.

The head vent valves are controlled from switches mounted on the primary and secondary dedicated safety panels in the control room. Each safety panel contains a switch for controlling each head vent valve (4 switches per panel). Each switch is a three-position, hold-in-position switch (open-neutral-close). If both switches are in the neutral position, soft control of the valve from the primary dedicated safety panel is allowed. If both switches are in the open position the valve will open. Either switch in the close position will close the valve. If one switch is in neutral position and one in the open position, the valve will hold its previous position but soft control is defeated. During a fire, switches on one panel may be shorted but none of the head vent valves will be opened because the switches on the other panel will be deactivated before the fire shorts them.

9A.3.7.1.2 Other Spurious Actuation

Principal spurious actuation not involving high-low pressure interfaces is discussed below.

Passive Core Cooling System Passive Residual Heat Removal Heat Exchanger Inlet Valve Actuation

One normally open valve is provided to isolate the inlet line to the passive residual heat removal heat exchanger. To preclude the spurious closing of the inlet valve as a result of a fire, the power to the valve is locked out during power operations. Thus, spurious closing of the passive core cooling passive residual heat removal heat exchanger inlet valve does not occur and the safe shutdown capability is not affected.

Passive Containment Cooling System Valve Actuation

Two valves in series isolate each of the three discharge flow paths from the passive containment cooling system storage tank. For purposes of system reliability, one valve in each flow path is normally open and the other is normally closed. Electrical division assignments are shown in Table 9A-2.

Spurious actuation of one of these valves is assumed to occur where a fire affects its electrical circuitry. Such a fire can occur in the main control room, an electrical equipment fire area, in the passive containment cooling system valve room, or in fire areas or fire zones through which the applicable electrical cables are routed.

Spurious actuation of one of these valves causes a passive containment cooling system flow path to be disabled or inadvertently opened, depending on which valve is affected. If a normally closed valve spuriously opens, passive containment cooling system water delivery from that flow path will be

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initiated which does not adversely affect the capability to achieve and maintain safe shutdown. If one of the normally open valves were spuriously closed to prevent passive containment cooling system water delivery through that flow path when called upon during the safe shutdown process, the redundant passive containment cooling system water delivery flow paths would be sufficient to achieve and maintain safe shutdown.

Containment Isolation Valve Actuation

Spurious actuation of a containment isolation valve is assumed to occur where a fire affects its electrical circuitry. Each containment penetration has redundant means of containment isolation.

Reactor Trip Switchgear

The reactor trip switchgear receives signals from each of the four Class 1E electrical divisions. One signal from each division de-energizes the undervoltage trip attachments and the other signal energizes the shunt trip attachments on two reactor trip breakers in each division. Either attachment will trip related reactor trip breaker. Also, two out of four signals are required to trip. There are two redundant sets of trip switchgear in separate fire areas. There is no single spurious signal which could prevent the reactor from being tripped.

Reactor Coolant Pump Trip Switchgear

There are two redundant sets of reactor coolant pump trip switchgear in separate fire areas. One is controlled from division B; the other from division C. Thus, a spurious signal in either train will not prevent trip of the reactor coolant pumps.

9A.3.7.2 Protection of Accident Mitigation Equipment

Based on the guidance in BTP CMEB 9.5-1, redundant trains of safety-related equipment used to mitigate the consequences of a design basis accident (but not required for safe shutdown following a fire) are separated so that a fire within one train will not damage the redundant train. Both trains are permitted to be damaged by a single exposure fire.

Either diverse methods of performing the accident mitigation function or adequate separation is provided.

9A.4 References

1. NUREG-0800, U. S. Nuclear Regulatory Commission Standard Review Plan, Subsection 9.5.1, “Fire Protection Program,” Revision 3, July 1981, including Branch Technical Position (BTP) CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," Revision 2, July 1981.

2. Fire Protection Handbook, Edited by A. E. Cote, National Fire Protection Association, 16th edition.

3. NRC Generic Letter 81-12, February 20, 1981.

4. NFPA 92A-2000, “Recommended Practice for Smoke Control Systems.”

5. DELETED

6. APP-FPS-G1R-002, Revision 2, “AP1000 Fire Induced Multiple Spurious Actuation Report.”

201. DELETED

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202. National Fire Protection Association, “Standard Methods of Tests of Fire Endurance of Building Construction and Materials,” NFPA 251, 1975.

203. Not used.

204. American Society of Testing and Materials, “Standard Test Method for Fire Tests of Penetration Firestop Systems,” ASTM E814-13a.

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Table 9A-1
Not Used

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Table 9A-2 (Sheet 1 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1000 AF 01/ 1100 AF 11105	RCS	Reactor Vessel (MV-01)	Note 4	Note 4	Note 4	Note 4
	RXS	Source Range Excore Detectors	NE-001A	NE-001C	NE-001B	NE-001D
		Intermediate Range Excore Detectors	NE-002A	NE-002C	NE-002B	NE-002D
		Power Range Excore Detectors (Lower)	NE-003A	NE-003C	NE-003B	NE-003D
		Power Range Excore Detectors (Upper)	NE-004A	NE-004C	NE-004B	NE-004D
1000 AF 01/ 1100 AF 11204	RCS	Hot Leg 1 Flow	FT101A	FT101C		
		Hot Leg 2 Flow	FT102A	FT102C		
1000 AF 01/ 1100 AF 11206	PXS	Core Makeup Tank A Discharge Isolation Valve			V014A	V014A
		Core Makeup Tank A Discharge Isolation Valve			V015A	V015A
	SFS	Suction Line Containment Isolation Valve			V034	
1000 AF 01/ 1100 AF 11207	PXS	Core Makeup Tank B Discharge Isolation Valve	V014B	V014B		
		Core Makeup Tank B Discharge Isolation Valve	V015B	V015B		
1000 AF01/ 1100 AF 11208	RNS	Suction from IRWST Cont. Isolation Valve			V023	
		Return from CVS Cont. Isolation Valve			V061	

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Table 9A-2 (Sheet 2 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1000 AF 01/ 1100 AF 11300A	PSS	Containment Air Sample Cont. Isolation Valve			V008	
		Liquid Sample Line Cont. Isolation Valve			V010A	V010B
		Sample Return Line Cont. Isolation Valve			V024	
	VFS	Containment Purge Discharge Cont. Isolation Valve				V009
	VFS	Containment Purge Inlet Cont. Isolation Valve				V004
	PXS	IRWST Wide Range Level			LT-046	LT-048
		IRWST Gutter Isolation Valve			V130A	V130B
		Core Makeup Tank (MT-02A)	Note 4	Note 4	Note 4	Note 4
	SGS	Steam Generator 2 Wide Range Water Level			LT-014	LT-018
1000 AF 01/ 1100 AF 11300B	CCS	Outlet Line Cont. Isolation Valve	V207			
	CVS	Letdown Containment Isolation Valve	V045			
		Makeup Line Cont. Isolation Valve	V091			
		RCS Purification Stop Valve (RCPB)	V001	V002		
		Zinc Injection Cont. Isolation Valve	V094			

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Table 9A-2 (Sheet 3 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1000 AF 01/ 1100 AF 11300B	IDS	Class 1E Electrical Penetrations	EY-P11Z	EY-P27Z		
		Class 1E Electrical Penetrations	EY-P12Y	EY-P29Y		
		Class 1E Electrical Penetrations	EY-P13Y	EY-P28Y		
		Cable trays that route Class 1E cables	Note 1	Note 1		
	PXS	PRHR Heat Exchanger Control Valve (Note 3)	V108A	V108A		
		Heat Exchanger Control Valve (Note 3)			V108B	V108B
		IRWST Wide Range Level		LT-047		
		PRHR Flow Transmitter		FT-049A	FT-049B	
		Core Makeup Tank (MT-02B)	Note 4	Note 4	Note 4	Note 4
	RCS	Pressurizer Pressure	PT-191A	PT-191C		
		Pressurizer Level	LT-195A	LT-195C		
		Hot Leg 1 Wide Range Pressure	PT-140A	PT-140C		
	SGS	Steam Generator 1 Narrow Range Water Level	LT-001	LT-003		
		Steam Generator 2 Narrow Range Water Level	LT-005	LT-007		
		Steam Generator 2 Wide Range Water Level	LT-013	LT-017		

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Table 9A-2 (Sheet 4 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1000 AF 01/ 1100 AF 11300B	SGS	Steam Generator 1 Wide Range Water Level	LT-011	LT-015		
		SG1 Steam Line Pressure	PT-030	PT-032		
		SG2 Steam Line Pressure	PT-034	PT-036		
	WLS	Sump Discharge Cont. Isolation Valve	V055			
		RCDT Gas Outlet Cont. Isolation Valve	V067			
1000 AF 01/ 1100 AF 11301	RCS	Reactor Head Vent Valve	V150A	V150C	V150B	V150D
		RCP 1A Bearing Water Temperature	TE-211A	TE-211C	TE-211B	TE-211D
		RCP 1B Bearing Water Temperature	TE-212A	TE-212C	TE-212B	TE-212D
		Cold Leg 1A Temperature (Narrow Range)	TE-121A			TE-121D
		Cold Leg 1B Temperature (Narrow Range)		TE-121C	TE-121B	
		Cold Leg 1A Temperature (Wide Range)	TE-125A			
		Cold Leg 1B Temperature (Wide Range)		TE-125C		
		Hot Leg 1 Temperature (Narrow Range)	TE-131A	TE-131C		
		Hot Leg 1 Temperature (Narrow Range)	TE-132A	TE-132C		
		Hot Leg 1 Temperature (Narrow Range)	TE-133A	TE-133C		

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Table 9A-2 (Sheet 5 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1000 AF 01/ 1100 AF 11301	RCS	Hot Leg 1 Temperature (Wide Range)		TE-135A		
		RCP Shaft Speed	ST-281		ST-282	
		Hot Leg 1 Flow			FT-101B	FT-101D
	SGS	Steam Generator 1 Wide Range Water Level			LT-012	LT-016
1000 AF 01/ 1100 AF 11302	RCS	RCP 2A Bearing Water Temperature	TE-213A	TE-213C	TE-213B	TE-213D
		RCP 2B Bearing Water Temperature	TE-214A	TE-214C	TE-214B	TE-214D
		Cold Leg 2B Temperature (Narrow Range)	TE-122A			TE-122D
		Cold Leg 2A Temperature (Narrow Range)		TE-122C	TE-122B	
		Cold Leg 2A Temperature (Wide Range)			TE-125B	
		Cold Leg 2B Temperature (Wide Range)				TE-125D
		Hot Leg 2 Temperature (Narrow Range)			TE-131B	TE-131D
		Hot Leg 2 Temperature (Narrow Range)			TE-132B	TE-132D
		Hot Leg 2 Temperature (Narrow Range)			TE-133B	TE-133D
		Hot Leg 2 Temperature (Wide Range)			TE-135B	
		RCP Shaft Speed		ST-283		ST-284
		Hot Leg 2 Flow			FT-102B	FT-102D
1000 AF 01/ 1100 AF 11303	RCS	Reference Leg Temperature	TE-193A	TE-193C	TE-193B	TE-193D

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Table 9A-2 (Sheet 6 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1000 AF 01/ 1100 AF 11500	IDS	Cable trays that route Class 1E cables			Note 1	Note 1
		Class 1E Electrical Penetrations			EY-P30Z	EY-P14Z
		Class 1E Electrical Penetrations			EY-P31Y	EY-P15Y
		Class 1E Electrical Penetrations			EY-P32Y	EY-P16Y
	IIS	Core Exit Temperature		TE-002	TE-001	
		Core Exit Temperature		TE-003	TE-004	
		Core Exit Temperature		TE-005	TE-006	
		Core Exit Temperature		TE-007	TE-011	
		Core Exit Temperature		TE-008	TE-015	
		Core Exit Temperature		TE-010	TE-017	
		Core Exit Temperature		TE-012	TE-019	
		Core Exit Temperature		TE-014	TE-021	
		Core Exit Temperature		TE-016	TE-023	
		Core Exit Temperature		TE-018	TE-025	
		Core Exit Temperature		TE-020	TE-027	
		Core Exit Temperature		TE-022	TE-029	
		Core Exit Temperature		TE-024	TE-031	
		Core Exit Temperature		TE-026	TE-033	
		Core Exit Temperature		TE-028	TE-035	
		Core Exit Temperature		TE-032	TE-036	
		Core Exit Temperature		TE-037	TE-038	
		Core Exit Temperature		TE-039	TE-040	
		Core Exit Temperature		TE-042	TE-041	

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Table 9A-2 (Sheet 7 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1000 AF 01/ 1100 AF 11500	PXS	PRHR HX Inlet Isolation Valve (Note 3)	V101			
	SGS	Steam Generator 1 Narrow Range Water Level			LT-002	LT-004
		Steam Generator 2 Narrow Range Water Level			LT-006	LT-008
	VWS	Fan Coolers Return Cont. Isolation Valve			V082	
	RCS	Hot Leg 2 Wide Range Pressure			PT-140B	PT-140D
		Pressurizer Pressure			PT-191B	PT-191D
		Pressurizer Level			LT-195B	LT-195D
1000 AF 01/ 1200 AF 12341	PCS	Containment Pressure			PT-006	PT-008
	IDSA	Class 1E Electrical Penetration	EY-P11Z			
		Class 1E Electrical Penetration	EY-P12Y			
		Class 1E Electrical Penetration	EY-P13Y			
1000 AF 01/ 1200 AF 12541	IDS	Class 1E Cables	Note 1	Note 1	Note 1	
1000 AF 01/ 1270 AF 12701	IDS	Class 1E Cables	Note 1	Note 1	Note 1	
	PCS	PCCWST Isolation Valve	V001A	V001C	V001B	
		PCCWST Series Isolation Valve	V002A	V002C	V002B	
		PCS Water Delivery Flow		FT-001	FT-002	
		PCS Water Delivery Flow			FT-003	
		PCS Storage Tank Level		LT-010	LT-011	
1200 AF 01	IDS	Cable trays that route Class 1E cables	Note 1	Note 1		
	SFS	Suction Line Cont. Isolation Valve	V035			
	VFS	Containment Purge Inlet Cont. Isolation Valve	V003			

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Table 9A-2 (Sheet 8 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1200 AF 01	PSS	Liquid Sample Line A Cont. Isolation Valve		V011A		
		Liquid Sample Line B Cont. Isolation Valve	V011B			
		Sample Return Line Cont. Isolation Valve	V023			
		Air Sample Line Cont. Isolation Valve	V046			
	SFS	Discharge Line Cont. Isol. Valve	V038			
	VFS	Containment Purge Discharge Cont. Isolation Valve	V010			
	VFS	Vacuum Relief Containment Isolation Valve	V800A	V800B		
	RNS	Discharge Cont. Isolation Valve	V011			
		Suction Header Cont. Isolation Valve	V022			
	PCS	Containment Pressure	PT-005	PT-007		
1200 AF 03	IDS	Cable trays that route Class 1E cables	Note 1	Note 1		
1201 AF 02	IDSB	24 Hr Battery 1A			DB-1A	
		24 Hr Battery 1B			DB-1B	
		72 Hr Battery 2A			DB-2A	
		72 Hr Battery 2B			DB-2B	
		250 Vdc Distribution Panel			DD-1	
		208/120 Vac Distribution Panel			EA-1	
		208/120 Vac Distribution Panel			EA-2	
		208/120 Vac Distribution Panel			EA-3	
		250 Vdc Switchboard			DS-1	

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Table 9A-2 (Sheet 9 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1201 AF 02		250 Vdc Switchboard			DS-2	
		208/120 Vac Inverter			DU-1	
		208/120 Vac Inverter			DU-2	
		Regulating Transformer			DT-1	
		Battery Charger			DC-1	
		Battery Charger			DC-2	
		Voltage to Class 1E Battery Charger			--- 002	
		Voltage to Class 1E Battery Charger			--- 006	
	PMS	Protection and Safety Monitoring System Cabinets			Note 5	
		Class 1E Cables	Note 2	Note 2	Note 2	Note 2
	IDSB	Electrical Penetration			EY-P32Y	
		250 Vdc MCC			DK-1	
		Electrical Penetration			EY-P30Z	
		Electrical Penetration			EY-P31Y	
1201 AF 03	IDSD	Battery 1A				DB-1A
		Battery 1B				DB-1B
		250 Vdc Distribution Panel				DD-1
		208/120 Vac Distribution Panel				EA-1
		208/120 Vac Distribution Panel				EA-2
		250 Vdc Switchboard				DS-1
		208/120 Vac Inverter				DU-1
		Regulating Transformer				DT-1
		Battery Charger				DC-1
		Voltage to Class 1E Battery Charger				--- 004
		Voltage to Class 1E Battery Charger				--- 008
	PMS	Protection and Safety Monitoring System Cabinets				Note 5
		Class 1E Cables	Note 2	Note 2	Note 2	Note 2

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Table 9A-2 (Sheet 10 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1201 AF 03	IDS	250 Vdc MCC				DK-1
		Electrical Penetration				EY-P14Z
		Electrical Penetration				EY-P15Y
		Electrical Penetration				EY-P16Y
		Class 1E Cables			Note 1	Note 1
1201 AF 04	IDS	Cable trays that route Class 1E cables			Note 1	Note 1
	CAS	Instrument Air Supply Cont. Isolation Valve			V014	V014
	CCS	Inlet Line Cont. Isolation Valve				V200
	CCS	Outlet Line Cont. Isolation Valve				V208
	PXS	Nitrogen Supply Cont. Isolation Valve				V042
	VWS	Fan Coolers Supply Cont. Isolation Valve				V058
		Fan Coolers Return Cont. Isolation Valve				V086
	SGS	SG 1 Steam Line Pressure			PT-031	PT-033
		SG 2 Steam Line Pressure			PT-035	PT-037
1201 AF 05	CVS	Hydrogen Addition Cont. Isolation Valve				V219
		Zinc Injection Cont. Isolation Valve				V092
	SGS	Steam Gen. Blowdown Cont. Isolation Valve			V075A	V074A
		Steam Gen. Blowdown Cont. Isolation Valve			V074B	V075B

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Table 9A-2 (Sheet 11 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1201 AF 05	SGS	PORV and Block Valve- SG 1 Cont. Isolation Valves			V027A	V233A
		Steam Line Cond. Drain Cont. Isolation Valve				V036A
		Main Steam Line Cont. Isolation Valve			V040A	V040A
		Main Feedwater Cont. Isolation Valve			V057A	V057A
		Startup Feedwater Cont. Isolation Valve			V067A	
		Main Steam Line Cond. Drain Control Valve			V086A	
		MSIV Bypass Cont. Isolation Valve			V240A	V240A
	IDS	Class 1E Cables			Note 1	Note 1
1201 AF 06	SGS	PORV and Block Valve SG 2 Cont. Isolation Valve			V233B	V027B
		Steam Line Cond. Drain Cont. Isolation Valve				V036B
		Main Steam Line Cont. Isolation Valve			V040B	V040B
		Main Feedwater Cont. Isolation Valve			V057B	V057B
		Startup Feedwater Cont. Isolation Valve				V067B
		Steam Line Cond. Drain Control Valve			V086B	
		MSIV Bypass Cont. Isolation Valve			V240B	V240B
1202 AF 03	IDSC	24 Hr Battery 1A		DB-1A		
		24 Hr Battery 1B		DB-1B		
		72 Hr Battery 2A		DB-2A		
		72 Hr Battery 2B		DB-2B		
		250 Vdc Distribution Panel		DD-1		

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Table 9A-2 (Sheet 12 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1202 AF 03	IDSC	208/120 Vac Distribution Panel		EA-1		
		208/120 Vac Distribution Panel		EA-2		
		208/120 Vac Distribution Panel		EA-3		
		250 Vdc Switchboard		DS-1		
		250 Vdc Switchboard		DS-2		
		208/120 Vac Inverter		DU-1		
		208/120 Vac Inverter		DU-2		
		Regulating Transformer		DT-1		
		Battery Charger		DC-1		
		Battery Charger		DC-2		
		Voltage to Class 1E Battery Charger		---003		
		Voltage to Class 1E Battery Charger		---007		
	PMS	Protection and Safety Monitoring System Cabinets		Note 5		
		Class 1E Cables	Note 2	Note 2	Note 2	Note 2
	ECS	6900V RCP 1A Switchgear		ES-31		
		6900V RCP 2A Switchgear		ES-51		
		6900V RCP 1B Switchgear		ES-41		
		6900V RCP 2B Switchgear		ES-61		
	IDSC	Electrical Penetration		EY-P27Z		
		Electrical Penetration		EY-P29Y		
		Electrical Penetration		EY-P28Y		
		250 Vdc MCC		DK-1		

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Table 9A-2 (Sheet 13 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1202 AF 04	IDSA	Battery 1A	DB-1A			
		Battery 1B	DB-1B			
		250 Vdc Distribution Panel	DD-1			
		208/120 Vac Distribution Panel	EA-1			
		208/120 Vac Distribution Panel	EA-2			
		250 Vdc Switchboard	DS-1			
		208/120 Vac Inverter	DU-1			
		Regulating Transformer	DT-1			
		Battery Charger	DC-1			
		Voltage to Class 1E Battery Charger	--- 001			
		Voltage to Class 1E Battery Charger	--- 005			
	PMS	Protection and Safety Monitoring System Cabinets	Note 5			
		Class 1E Cables	Note 2	Note 2	Note 2	Note 2
1202 AF 05	PMS	Transfer Switch Set	JW-004A	JW-004C	JW-004B	JW-004D
1210 AF 01	IDS	Spare Battery (DB-1A)	Note 1	Note 1	Note 1	Note 1
		Spare Battery (DB-1B)	Note 1	Note 1	Note 1	Note 1
		Spare Battery Charger (DC-1)	Note 1	Note 1	Note 1	Note 1
		Spare Fuse Transfer Box (DF-1)	Note 1	Note 1	Note 1	Note 1
1220 AF 01	IDS	Cable trays that route Class 1E cables			Note 1	Note 1
	ECS	6900V RCP 1A Switchgear			ES-32	
		6900V RCP 2A Switchgear			ES-52	
		6900V RCP 1B Switchgear			ES-42	

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Table 9A-2 (Sheet 14 of 14)
Safe Shutdown Components

Fire Area/ Fire Zone	System	Description	Class 1E Division			
			A	C	B	D
1220 AF 01	ECS	6900V RCP 2B Switchgear			ES-62	
	SGS	SG 1 Startup Feedwater Flow			FT-055A	FT-055B
		SG 2 Startup Feedwater Flow			FT-056A	FT-056B
1220 AF 02	CVS	Letdown Containment Isolation Valve				V047
		Makeup Line Cont. Isolation Valve				V090
	WLS	Sump Discharge Cont. Isolation Valve				V057
		RCDT Gas Outlet Cont. Isolation Valve				V068
1230 AF 01	IDS	Cable trays that route Class 1E cables	Note 1	Note 1		
1230 AF 02	IDS	Cable trays that route Class 1E cables			Note 1	Note 1
1232 AF 01		Remote Shutdown Room	Note 4	Note 4	Note 4	Note 4
	IDS	Cable trays that route Class 1E cables	Note 1	Note 1	Note 1	Note 1
1240 AF 01	IDS	Cable trays that route Class 1E cables	Note 1	Note 1		
1242 AF 01		MCR Workstation	Note 4	Note 4	Note 4	Note 4
	IDS	Cable trays that route Class 1E cables	Note 1	Note 1	Note 1	Note 1
1242 AF 02	IDS	Class 1E Electrical Penetration	EY-P11Z			
		Class 1E Electrical Penetration	EY-P12Y			
		Class 1E Electrical Penetration	EY-P13Y			
		250 Vdc MCC	DK-1			
		Class 1E Cables	Note 1	Note 1		
1243 AF 01	PMS	Reactor Trip Switchgear I	JD-RTSA01	JD-RTSC01	JD-RTSB01	JD-RTSD01
	IDS	Class 1E Cables	Note 1	Note 1	Note 1	Note 1
1243 AF 02	PMS	Reactor Trip Switchgear II	JD-RTSA02	JD-RTSC02	JD-RTSB02	JD-RTSD02
	IDS	Class 1E Cables	Note 1	Note 1	Note 1	Note 1
1250 AF 01	IDS	Class 1E Cables	Note 1	Note 1		

Notes:
1. This represents equipment such as cables that have no associated tag number.
2. Class 1E interdivisional cables for data communication are addressed in the applicable Appendix 9A Fire Area Description.
3. The valve's position indication is needed as a diverse variable to confirm PRHR flow to support safe shutdown.
4. Identified as a safe shutdown component that is not associated with a specific Class 1E Division.
5. Various PMS Cabinets are located within the identified Fire Area/Fire Zone and associated with the identified division, but each PMS Cabinet is not individually identified as a component within the table.

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Table 9A-3
Not Used

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Table 9A-3A (Sheet 1 of 2)
Containment Area

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.[5]
1000 AF 01			**YES**	**30,190**			**180**	**180,000**	**SEE ZONE**	**SEE ZONE**
1100 AF 11105	REACTOR CAVITY	B		335	---	<60			SMOKE[9]	HOSE STATION
1100 AF 11204	VERTICAL ACCESS / RCDT ROOM	D		750	CABLE INS.	<90			SMOKE[9]	HOSE STATION
1100 AF 11206	PXS VALVE / ACCUMULATOR ROOM A	D		890	LUBRICANT	<60			SMOKE[9]	HOSE STATION
1100 AF 11207	PXS VALVE / ACCUMULATOR ROOM B	D		880	LUBRICANT	<60			SMOKE[9]	HOSE STATION
1100 AF 11208	NORMAL RESIDUAL HEAT REMOVAL VALVE ROOM	D		350	LUBRICANT	<60			SMOKE[9]	HOSE STATION
1100 AF 11209	CVS ROOM	D		965	PLASTIC/RUBBER	<60			SMOKE[9]	HOSE STATION
1100 AF 11300A	MAINTENANCE FLOOR SOUTHEAST	D		2045	CABLE INS.	<120			HEAT	HOSE STATION
1100 AF 11300B	MAINTENANCE FLOOR NORTH	D		4215	CABLE INS.	<180			HEAT	HOSE STATION
1100 AF 11301	SG COMPARTMENT 1	D		615	CABLE INS.	<90			SMOKE[9]	HOSE STATION
1100 AF 11302	SG COMPARTMENT 2	D		690	CABLE INS.	<90			SMOKE[9]	HOSE STATION
1100 AF 11303	PRESSURIZER COMPARTMENT	D		530	CABLE INS.	<90			SMOKE[9]	HOSE STATION
1100 AF 11303A	LOWER ADS VALVE AREA	D		275	CABLE INS.	<150			SMOKE[9]	HOSE STATION
1100 AF 11303B	UPPER ADS VALVE AREA	D		275	LUBRICANT	<60			SMOKE[9]	HOSE STATION
1100 AF 11500	OPERATING DECK AND REFUELING CAVITY	E		11420	CABLE INS.	<120			SMOKE[9]	HOSE STATION
1200 AF 12341	MIDDLE ANNULUS	B		1835	PLASTIC/RUBBER	<60			NOTE 10	NONE
1200 AF 12541	UPPER ANNULUS	B		1700	PLASTIC/RUBBER	<90			HEAT	NONE
1250 AF 12555	VES AIR STORAGE / OPERATING DECK STAGING AREA	D		1395	CABLE INS.	<90			SMOKE	HOSE STATION

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Table 9A-3A (Sheet 2 of 2)
Containment Area

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.[5]
1270 AF 12701	PCS VALVE ROOM	B		1025	PLASTIC/RUBBER	<60			SMOKE	NONE

Notes:

1. The first four digits of the fire area and fire zone numbers indicate the building, level and building area in which the fire area/zone is located. When the third or fourth digit is a zero, the fire area/zone spans more than one level or building area. The last two digits in a fire area number are a sequence number only. The last five digits in a fire zone number coincide with the room number of a prominent room in the fire zone.
2. A YES indication in the Safety Area column means that one or more safety-related components are located in the fire area.
3. Reserved.
4. The boundary fire resistance for each fire area represents the minimum resistance, in hours, for the surrounding walls, floor, and ceiling, except that:
   - A non-rated barrier capable of qualifying as a three-hour barrier is considered to have a resistance of three hours, provided that penetrations are adequately sealed.
   - Stairwells, elevator shafts and the like, which are enclosed by 2-hour (minimum) fire barrier walls, may comprise a portion of the boundary of a fire area having a three hour resistance.
   - Building exterior walls below grade (soil on the outside) are considered to have a fire resistance of at least three hours even though they are not fire-rated.

   A boundary fire resistance designation such as "180/0" indicates that part of the fire area boundary consists of nonrated building exterior walls/roof above grade, but that the minimum resistance of the fire barriers separating the fire area from adjacent fire areas is three hours. For detailed information about the fire resistance of fire area boundaries see Figures 9A-1 through 9A-5
5. The fixed suppression capability is indicated for each fire area. Sprinkler and spray systems are automatic. In most cases, charcoal bed deluge systems require operator action to connect to a water source supplied by a manual hose station. Where a hose station is indicated, one or more hose streams are available. In addition to fixed suppression capability, portable extinguishers are provided throughout the plant.
6. Reserved.
7. Reserved.
8. Reserved.
9. Containment vessel is provided with smoke detection (via HVAC duct detectors) for a general alarm.
10. See appropriate fire area discussion for identified spot detection for equipment

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Table 9A-3B (Sheet 1 of 8)
Auxiliary Building

Fire Area/Zone (1)	Fire Zone Description	Fire Severity Category	Safety Area (?)(2)	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)(4)	Max. Fire Load (BTU/ft^2)	Detect. Cap.	Fixed Suppression Cap.(5)
RADIOLOGICALLY CONTROLLED AREAS										
1200 AF 01 1200 AF 12241	LOWER ANNULUS	B	**YES**	**25156** 1865	CABLE INS.	<60	**180**	**240,000**	**SEE ZONE** NONE	**SEE ZONE** HOSE STATION
1200 AF 12461	CORRIDOR	E		525	CABLE INS.	<150			SMOKE	HOSE STATION
1205 AF 12362	RNS HX ROOM	B		310	PLASTIC/RUBBER	<60			NONE	HOSE STATION
1205 AF 12365	WASTE MONITOR TANK ROOM B	B		549	PLASTIC/RUBBER	<60			NONE	HOSE STATION
1210 AF 12151	DEMIN. / FILTER, WLS & WGS EQUIPMENT ROOMS	B		1880	CABLE INS.	<120			NOTE 10	HOSE STATION
1210 AF 12171	EFFLUENT HOLDUP TANK ROOM A	B		860	CABLE INS.	<90			NONE	HOSE STATION
1214 AF 12152	PRIMARY SAMPLE ROOM	B		505	CABLE INS.	<90			SMOKE	HOSE STATION
1214 AF 12154	AUX. BLDG SUMP AND SFS PENETRATION ROOM	B		585	CABLE INS. PLASTIC/RUBBER	<90			NONE	HOSE STATION
1214 AF 12354	MID ANNULUS ACCESS ROOM	B		235	CABLE INS.	<90			NONE	HOSE STATION
1215 AF 12161	CORRIDOR	B		615	CABLE INS.	<120			SMOKE	HOSE STATION
1215 AF 12162	RNS PUMP ROOM A	B		280	PLASTIC/RUBBER	<90			NOTE 10	HOSE STATION
1216 AF 12166	WASTE HOLDUP TANK ROOM A	B		284	PLASTIC/RUBBER	<60			NONE	HOSE STATION
1216 AF 12167	WASTE HOLDUP TANK ROOM B	B		301	PLASTIC/RUBBER	<60			NONE	HOSE STATION
1216 AF 12169	WLS PUMP ROOM / CORRIDOR	B		335	CABLE INS. PLASTIC/RUBBER	<90			NONE	HOSE STATION
1216 AF 12172	EFFLUENT HOLDUP TANK ROOM B	B		910	CABLE INS.	<90			NONE	HOSE STATION

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Table 9A-3B (Sheet 2 of 8)
Auxiliary Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap.	Fixed Suppression Cap.[5]
1216 AF 12264	WASTE MONITOR TANK ROOM C & CHEM WASTE TANK ROOM	B		875	PLASTIC/RUBBER CABLE INS.	<90			NONE	HOSE STATION
1220 AF 12251	DEMINERALIZER / FILTER ACCESS, CVS MAKEUP PUMP ROOMS	E		1865	CABLE INS.	<90			NOTE 10	WET PIPE SPRINKLER[6], HOSE STATION
1220 AF 12256	EL 92'-6" PIPE CHASE / VALVE ROOM	B		660	CABLE INS. PLASTIC/RUBBER	<60			HEAT	HOSE STATION
1220 AF 12259	EL 92'-6" PIPE CHASE	B		840	PLASTIC/RUBBER	<60			NONE	HOSE STATION
1220 AF 12269	EL 92'-6" PIPE CHASE	B		695	PLASTIC/RUBBER	<60			HEAT	HOSE STATION
1220 AF 12272	SFS EQUIPMENT ROOMS	B		1380	CABLE INS.	<90			NOTE 10	HOSE STATION
1224 AF 12252	RADIOACTIVE CHEMISTRY LABORATORY	B		545	PLASTIC/RUBBER	<90			SMOKE	HOSE STATION
1225 AF 12261	WLS PUMP ROOM AND CORRIDOR	E		985	CABLE INS.	<120			NOTE 10	HOSE STATION
1225 AF 12262	PIPING / VALVE ROOM	B		600	PLASTIC/RUBBER CABLE INS.	<90			HEAT	HOSE STATION
1234 AF 12351	MAINTENANCE FLOOR STAGING AREA	D		1145	CABLE INS.	<120			SMOKE	HOSE STATION
1234 AF 12352	ELEVATION 107'-2" PERSONNEL HATCH	D		275	PLASTIC/RUBBER	<180			SMOKE	HOSE STATION
1235 AF 12361	CORRIDOR	D		525	CABLE INS.	<90			SMOKE	HOSE STATION
1235 AF 12363	WASTE MONITOR TANK ROOM A	B		365	CABLE INS.	<90			SMOKE	HOSE STATION
1244 AF 12451	SECURITY ROOM	B		315	PLASTIC/RUBBER	<90			SMOKE	HOSE STATION

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Table 9A-3B (Sheet 3 of 8)
Auxiliary Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap.	Fixed Suppression Cap.[5]
1244 AF 12452	VFS PENETRATION ROOM	E		275	CABLE INS.	<90			SMOKE	HOSE STATION
1244 AF 12454	VFS / SFS / PENETRATION ROOM	E		210	CABLE INS.	<60			NONE	HOSE STATION
1250 AF 12561	CCS VALVE ROOM AND ACCESS	B		485	CABLE INS. PLASTIC RUBBER	<90			SMOKE	HOSE STATION
1254 AF 12553	ELEVATION 135'-3" PERSONNEL ACCESS AREA	D		1260	CABLE INS.	<120			SMOKE	HOSE STATION
1254 AF 12554	SECURITY ROOM	B		277	CABLE INS. PLASTIC RUBBER	<90			SMOKE	HOSE STATION
1264 AF 12651	VAS EQUIPMENT ROOM	B		1540	CABLE INS.	<120			SMOKE	HOSE STATION
1200 AF 02 1200 AF 12562	FUEL HANDLING AREA	D	**YES**	**6615** 5228	CABLE INS.	<90	**180/0**	**180,000**	**SEE ZONE** SMOKE	**SEE ZONE** HOSE STATION
1230 AF 12371	RAIL CAR BAY / FILTER STORAGE AREA	D		1425	CABLE INS.	<120			NONE	WET PIPE SPRINKLER[6], HOSE STATION
1236 AF 12372	RESIN TRANSFER PUMP / VALVE ROOM	B		185	PLASTIC RUBBER	<90			NONE	HOSE STATION
1236 AF 12373	SPENT RESIN TANK RM	B		125	PLASTIC/RUBBER	<90			NONE	HOSE STATION
1246 AF 12471	WSS VALVE / PIPING AREA	B		150	PLASTIC/RUBBER CABLE INS.	<90			NONE	HOSE STATION
1204 AF 01 -	RNS PUMP ROOM B	B	**NO**	**255**	CABLE INS.	<90	**180**	**120,000**	**NOTE 10**	**HOSE STATION**
1220 AF 02 -	LOWER ANNULUS VALVE AREA	B	**YES**	**225**	CABLE INS. PLASTIC RUBBER	<90	**180**	**120,000**	**HEAT**	**HOSE STATION**
1204 AF 02 -	STAIRWELL S03	A	**NO**	**1391**	PLASTIC/RUBBER	<15	**180/0**	**10,000**	**NONE**	**NONE**

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Table 9A-3B (Sheet 4 of 8)
Auxiliary Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap.	Fixed Suppression Cap.[5]
1204 AF 03 -	SHIELD BLDG ELEVATOR	A	**NO**	**45**	CABLE INS. PLASTIC/RUBBER	<45	**180/0**	**30,000**	**HEAT**	**HOSE STATION**
1205 AF 01 -	STAIRWELL S04	A	**NO**	**1053**	PLASTIC/RUBBER	<15	**120**	**10,000**	**NONE**	**NONE**
1205 AF 02 -	SOUTHEAST ELEVATOR SHAFT	A	**NO**	**68**	CABLE INS. PLASTIC/RUBBER	<45	**120**	**30,000**	**HEAT**	**HOSE STATION**
NONRADIOLOGICALLY CONTROLLED AREAS										
1200 AF 03 1230 AF 12311	CORRIDOR	E	**YES**	**570** 320	CABLE INS.	<120	**180**	**240,000**	**SMOKE**	**HOSE STATION**
1242 AF 12411	CORRIDOR	E		250	CABLE INS.	<150				
1201 AF 01 1201 AF 01	STAIRWELL S02	A	**NO**	**505**	PLASTIC/RUBBER	<15	**120**	**10,000**	**NONE**	**NONE**
1201 AF 02 1211 AF 12104	DIVISION B BATTERY ROOM 1	E	**YES**	**2260** 560	BATTERIES	<120	**180**	**240,000**	**SMOKE**	**HOSE STATION**
1221 AF 12204	DIVISION B BATTERY ROOM 2	E		595	CABLE INS. BATTERIES	<120				
1222 AF 12207	DIVISION B DC EQUIPMENT ROOM	E		435	CABLE INS.	<90				
1231 AF 12304	DIVISION B I&C / PENETRATION ROOM	E		590	CABLE INS.	<150				
1231 AF 12344	DIVISION B - MID ANNULUS PENETRATION ROOM	B		80	PLASTIC/RUBBER	<60				
1201 AF 03 1211 AF 12105	DIVISION D BATTERY ROOM	E	**YES**	**1745** 560	CABLE INS. BATTERIES	<120	**180**	**240,000**	**SMOKE**	**HOSE STATION**
1221 AF 12205	DIVISION D DC EQUIPMENT ROOM	E		565	CABLE INS.	<120				

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Table 9A-3B (Sheet 5 of 8)
Auxiliary Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap.	Fixed Suppression Cap.[5]
1231 AF 12305	DIVISION D I&C / PENETRATION ROOM	E		560	CABLE INS.	<120				
1231 AF 12345	DIVISION D - MID ANNULUS PENETRATION ROOM	B		60	PLASTIC/RUBBER	<60				
1201 AF 04 1241 AF 12405	LOWER VBS B&D EQUIPMENT ROOM	B	**YES**	**1390** 695	CABLE INS.	<120	**180**	**120,000**	**SMOKE**	**HOSE STATION**
1251 AF 12505	UPPER VBS B&D EQUIPMENT ROOM	B		695	CABLE INS.	<150				
1201 AF 05 1231 AF 12306	VALVE / PIPING PENETRATION ROOM	B	**YES**	**2055** 575	CABLE INS.	<120	**180**	**120,000**	**SMOKE**	**HOSE STATION**
1241 AF 12506	MSIV COMPARTMENT A	B		1480	CABLE INS. PLASTIC/RUBBER	<90				
1201 AF 06 -	MSIV COMPARTMENT B	B	**YES**	**1460**	CABLE INS.	<120	**180**	**120,000**	**SMOKE**	**HOSE STATION**
1202 AF 01 -	STAIRWELL S01	A	**NO**	**927**	PLASTIC/RUBBER	<15	**120**	**10,000**	**NONE**	**NONE**
1202 AF 02 -	NORTHEAST ELEVATOR SHAFT	A	**NO**	**69**	CABLE INS. PLASTIC/RUBBER	<45	**120**	**30,000**	**HEAT**	**HOSE STATION**
1202 AF 03 1212 AF 12102	DIVISION C BATTERY ROOM 1	E	**YES**	**3195** 560	CABLE INS. BATTERIES	<120	**180**	**240,000**	**SMOKE**	**HOSE STATION**
1222 AF 12202	DIVISION C BATTERY ROOM 2	E		595	CABLE INS. BATTERIES	<120				
1222 AF 12203	DIVISION C DC EQUIPMENT ROOM	E		435	CABLE INS.	<150				

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Table 9A-3B (Sheet 6 of 8)
Auxiliary Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap.	Fixed Suppression Cap.[5]
1232 AF 12302	DIVISION C I&C ROOM	E		585	CABLE INS.	<150				
1232 AF 12312	DIVISION C RCP TRIP SWITCHGEAR ROOM	E		360	CABLE INS.	<120				
1232 AF 12313	I&C / DIVISION C PENETRATION ROOM	E		580	CABLE INS.	<150				
1232 AF 12343	DIVISION C - MID ANNULUS PENETRATION ROOM	B		80	PLASTIC/RUBBER	<60				
1202 AF 04 1212 AF 12101	DIVISION A BATTERY ROOM	E	**YES**	**525**	CABLE INS. BATTERIES	<120	**180**	**240,000**	**SMOKE**	**HOSE STATION**
1222 AF 12201	DIVISION A DC EQUIPMENT ROOM	E		535	CABLE INS.	<120				
1232 AF 12301	DIVISION A I&C ROOM	E		505	CABLE INS.	<150				
1202 AF 05 -	STAIRWELL S05	A	**YES**	**243**	PLASTIC/RUBBER	<15	**180**	**10,000**	**SMOKE**	**NONE**
1210 AF 01 1210 AF 12111	CORRIDOR	B	**YES**	**3020** 1555	CABLE INS. PLASTIC/RUBBER	<90	**180**	**240,000**	**SMOKE**	**HOSE STATION**
1212 AF 12103	SPARE BATTERY ROOM	E		840	CABLE INS. BATTERIES	<120				
1212 AF 12112	SPARE ROOM	B		395	CABLE INS. PLASTIC/RUBBER	<90				
1212 AF 12113	SPARE BATTERY CHARGER ROOM	E		230	CABLE INS.	<120				
1220 AF 01 1220 AF 12211	CORRIDOR	E	**YES**	**2115** 1520	CABLE INS.	<120	**180**	**240,000**	**SMOKE**	**HOSE STATION**

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Table 9A-3B (Sheet 7 of 8)
Auxiliary Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap.	Fixed Suppression Cap.[5]
1222 AF 12212	DIVISION B RCP TRIP SWITCHGEAR ROOM	B		360	CABLE INS.	<150				
1222 AF 12213	SPARE ROOM AND MOTOR CONTROL CTRS	E		235	CABLE INS.	<90				
1230 AF 01 -	CORRIDOR	E	**YES**	**760**	CABLE INS.	<120	**180**	**240,000**	**SMOKE**	**HOSE STATION**
1230 AF 02 -	NON-1E EQUIPMENT / PENETRATION ROOM	E	**YES**	**885**	CABLE INS.	<210	**180**	**N/A**	**SMOKE**	**PREACTION SPRINKLERS, HOSE STATION**
1232 AF 01 -	REMOTE SHUTDOWN ROOM	B	**YES**	**860**[a]	PLASTIC/RUBBER	<90	**180**	**120,000**	**SMOKE**	**HOSE STATION**
1240 AF 01 -	NON-1E EQUIPMENT / PENETRATION ROOM	E	**YES**	**795**	CABLE INS.	<180	**180**	**240,000**	**SMOKE**	**HOSE STATION**
1242 AF 01 1242 AF 12401A	MCR MAIN CONTROL AREA / OPERATOR WORK AREA / VESTIBULE	B	**YES**	**4740** 2950[b]	PLASTIC/RUBBER CABLE INS.	<120	**180**	**120,000**	**SMOKE**	**HOSE STATION**
1242 AF 12401B	MCR SHIFT MANAGER'S OFFICE / KITCHEN AREA / OPERATOR BREAK ROOM	B		1790[c]	PLASTIC/RUBBER CABLE INS.	<90				
1242 AF 02 -	DIVISION A - ELECTRICAL PENETRATION ROOM	E	**YES**	**395**	CABLE INS.	<120	**180**	**240,000**	**SMOKE**	**HOSE STATION**

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Table 9A-3B (Sheet 8 of 8)
Auxiliary Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap.	Fixed Suppression Cap.[5]
1243 AF 01 -	REACTOR TRIP SWITCHGEAR 1	B	**YES**	**100**	CABLE INS. PLASTIC/RUBBER	<150	**180**	**120,000**	**SMOKE**	**HOSE STATION**
1243 AF 02 -	REACTOR TRIP SWITCHGEAR 2	B	**YES**	**100**	CABLE INS. PLASTIC/RUBBER	<120	**180**	**120,000**	**SMOKE**	**HOSE STATION**
1250 AF 01 -	VBS MCR / A&C EQUIPMENT ROOM	B	**YES**	**3540**	CABLE INS.	<120	**180**	**120,000**	**SMOKE**	**HOSE STATION**

[a] 430 sq ft is the floor area above the sub-floor. An additional 430 sq ft is present in the sub-floor space.
[b] 1505 sq ft is the floor area above the sub-floor. An additional 1445 sq ft is present in the sub-floor space.
[c] 895 sq ft is the floor area above the sub-floor. An additional 895 sq ft is present in the sub-floor space.

Notes:

1. The first four digits of the fire area and fire zone numbers indicate the building, level and building area in which the fire area/zone is located. When the third or fourth digit is a zero, the fire area/zone spans more than one level or building area. The last two digits in a fire area number are a sequence number only. The last five digits in a fire zone number coincide with the room number of a prominent room in the fire zone.
2. A YES indication in the Safety Area column means that one or more safety-related components are located in the fire area.
3. Reserved.
4. The boundary fire resistance for each fire area represents the minimum resistance, in hours, for the surrounding walls, floor, and ceiling, except that:
   - A non-rated barrier capable of qualifying as a three-hour barrier is considered to have a resistance of three hours, provided that penetrations are adequately sealed.
   - Stairwells, elevator shafts and the like, which are enclosed by 2-hour (minimum) fire barrier walls, may comprise a portion of the boundary of a fire area having a three hour resistance.
   - Building exterior walls below grade (soil on the outside) are considered to have a fire resistance of at least three hours even though they are not fire-rated.

   A boundary fire resistance designation such as "180/0" indicates that part of the fire area boundary consists of nonrated building exterior walls/roof above grade, but that the minimum resistance of the fire barriers separating the fire area from adjacent fire areas is three hours. For detailed information about the fire resistance of fire area boundaries see Figures 9A-1 through 9A-5.
5. The fixed suppression capability is indicated for each fire area. Sprinkler and spray systems are automatic. In most cases, charcoal bed deluge systems require operator action to connect to a water source supplied by a manual hose station. Where a hose station is indicated, one or more hose streams are available. In addition to fixed suppression capability, portable extinguishers are provided throughout the plant.
6. Partial suppression coverage in this zone. See Section 9A.3, “Fire Protection Analysis Results”, for details.
7. Reserved.
8. Reserved.
9. Reserved.
10. See appropriate fire area discussion for identified spot detection for equipment.

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Table 9A-3C (Sheet 1 of 3)
Turbine Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.[5]
2000 AF 02	**STAIRWELL S02 and S08**	**A**	**NO**	**2892**	**PLASTIC/RUBBER**	**<15**	**120/0**	**10,000**	**NONE**	**NONE**
2009 AF 01	**STAIRWELL S01**	**A**	**NO**	**1898**	**PLASTIC/RUBBER**	**<15**	**120**	**10,000**	**NONE**	**NONE**
2009 AF 02	**Elevator Shaft**	**A**	**NO**	**104**	**CABLE INS.**	**<45**	**120**	**30,000**	**HEAT**	**HOSE STATION**
2000 AF 03	**STAIRWELL S03**	**A**	**NO**	**554**	**PLASTIC/RUBBER**	**<15**	**120/0**	**10,000**	**NONE**	**NONE**
2000 AF 15	**STAIRWELL S015**	**A**	**NO**	**990**	**PLASTIC/RUBBER**	**<15**	**120/0**	**10,000**	**NONE**	**NONE**
2040 AF 01	**Clean & Dirty Lube Oil Storage Room**	**E**	**NO**	**842**	**PLASTIC/RUBBER VOLATILE LIQUID**	**<300**	**180**	**N/A**	**NONE**	**WET PIPE SPRINKLERS HOSE STATION**
2043 AF 01	**Secondary Sampling Laboratory**	**E**	**NO**	**1232**	**VOLATILES CABLE INS.**	**<90**	**180/0**	**240,000**	**NONE**	**WET PIPE SPRINKLERS HOSE STATION**
2050 AF 01	**Lube Oil Reservoir Room**	**E**	**NO**	**1127**	**VOLATILE LIQUID CABLE INS.**	**<360**	**180**	**N/A**	**HEAT**	**PREACTION SPRINKLERS HOSE STATION**
2052 AF 01	**Switchgear Room #1**	**E**	**NO**	**1626**	**CABLE INS.**	**<90**	**120**	**160,000**	**SMOKE**	**HOSE STATION**
2053 AF 01	**Electrical Equipment Room**	**E**	**NO**	**747**	**CABLE INS.**	**<120**	**120/0**	**160,000**	**SMOKE**	**HOSE STATION**
2053 AF 02	**Switchgear Room #2**	**E**	**NO**	**1616**	**CABLE INS.**	**<90**	**120**	**160,000**	**SMOKE**	**HOSE STATION**
2151 AF 02	**Battery Charger Room**	**B**	**NO**	**395**	**CABLE INS**	**<90**	**120**	**80,000**	**SMOKE**	**HOSE STATION**
2151 AF 01	**Battery Room**	**E**	**NO**	**1226**	**CABLE INS. BATTERIES**	**<120**	**120**	**160,000**	**SMOKE**	**HOSE STATION**
2000 AF 01			**NO**	**246,055**			**0**	**240,000**	**SEE ZONE**	**SEE ZONE**
2030 AF 20300	El. 100'-0" and El. 82'-9" General Floor Area	E		72,376	CABLE INS. VOLATILE LIQUID	<90				WET PIPE SPRINKLERS HOSE STATION
2038 AF 20300	Main Feedwater Pump Area	E		3440	PLASTIC/RUBBER CABLE INS.	<120			HEAT	PREACTION SPRINKLERS
2039 AF 20301	Chemical Storage Area	E		439	PLASTIC/RUBBER	<60				WET PIPESPRINKLERS[8] HOSE STATION

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Table 9A-3C (Sheet 2 of 3)
Turbine Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.[5]
2131 AF 21380	CCS & BDS Pump Area	E		4173	PLASTIC/RUBBER CABLE INS. VOLATILE LIQUID	<120				WET PIPE SPRINKLERS HOSE STATION
2040 AF 20400	EL. 120'-6" General Floor Area	E		50808	PLASTIC/RUBBER CABLE INS VOLATILE LIQUID	<90				WET PIPE SPRINKLERS HOSE STATION
2141 AF 21480	VTS HVAC Equipment Room	E		4140	PLASTIC/RUBBER CABLE INS.	<90				HOSE STATION
2043 AF 20402	Excitation Equipment Room	E		1067	PLASTIC/RUBBER CABLE INS.	<120			SMOKE	HOSE STATION
2050 AF 20500	Elevation 141'-3" General Floor Area	E		38308	PLASTIC/RUBBER CABLE INS VOLATILE LIQUID	<120				WET PIPE SPRINKLERS HOSE STATION
2050 AF 20502	Digital-Electrohydraulic Skid	E		400	VOLATILE LIQUID	<90			HEAT	WATER SPRAY HOSE STATION
2052 AF 20510	HVAC Equipment Area	B		3755	CABLE INS	<90			SMOKE	HOSE STATION
2057 AF 20511	Generator Seal Oil Unit	E		411	PLASTIC/RUBBER VOLATILE LIQUID	<60			HEAT	WATER SPRAY HOSE STATION
2151 AF 21580	South Bay 135'-3" General Area	E		2285	CABLE INS	<120			SMOKE	HOSE STATION
2151 AF 21583	VFD Equipment Room	E		4075	PLASTIC/RUBBER CABLE INS	<90			SMOKE	HOSE STATION
2060 AF 20600	Elevation 170'-0" General Floor Area	E		40958	PLASTIC/RUBBER CABLE INS VOLATILE LIQUID	<90				WET PIPE SPRINKLERS[7] HOSE STATION
2063 AF 20601	Tool Room / Storage Area	D		365	PLASTIC/RUBBER	<180			SMOKE	WET PIPE SPRINKLERS HOSE STATION
2063 AF 20602	Office Area / Engineering Workstation	B		455	PLASTIC/RUBBER	<60			SMOKE	HOSE STATION
2070 AF 20700	Heater Bay	E		8196	CABLE INS	<90				WET PIPE SPRINKLERS HOSE STATION
2070 AF 20701	Elevation 196'-3" North West Security Room	B		360	PLASTIC/RUBBER	<90			SMOKE	HOSE STATION
2070 AF 20702	Elevation 196'-3" North East Security Room	B		54	PLASTIC/RUBBER	<90			SMOKE	HOSE STATION

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Table 9A-3C (Sheet 3 of 3)
Turbine Building

Fire Area/Zone (1)	Fire Zone Description	Fire Severity Category	Safety Area (?)(2)	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)(4)	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.(5)
2070 AF 20703	Elevation 196'-3" South West Security Room	B		62	PLASTIC/RUBBER	<120			SMOKE	HOSE STATION
2070 AF 20704	Upper Heater Bay	E		8312	CABLE INS	<60				WET PIPE SPRINKLERS HOSE STATION

Notes:

1. The first four digits of the fire area and fire zone numbers indicate the building, level and building area in which the fire area/zone is located. When the third or fourth digit is a zero, the fire area/zone spans more than one level or building area. The last two digits in a fire area number are a sequence number only. The last five digits in a fire zone number coincide with the room number of a prominent room in the fire zone.
2. A YES indication in the Safety Area column means that one or more safety-related components are located in the fire area.
3. Reserved.
4. The boundary fire resistance for each fire area represents the minimum resistance, in hours, for the surrounding walls, floor, and ceiling, except that:
   - A non-rated barrier capable of qualifying as a three-hour barrier is considered to have a resistance of three hours, provided that penetrations are adequately sealed.
   - Stairwells, elevator shafts and the like, which are enclosed by 2-hour (minimum) fire barrier walls, may comprise a portion of the boundary of a fire area having a three hour resistance.
   - Building exterior walls below grade (soil on the outside) are considered to have a fire resistance of at least three hours even though they are not fire-rated.

   A boundary fire resistance designation such as "180/0" indicates that part of the fire area boundary consists of nonrated building exterior walls/roof above grade, but that the minimum resistance of the fire barriers separating the fire area from adjacent fire areas is three hours. For detailed information about the fire resistance of fire area boundaries see Figures 9A-1 through 9A-5.
5. The fixed suppression capability is indicated for each fire area. Sprinkler and spray systems are automatic. In most cases, charcoal bed deluge systems require operator action to connect to a water source supplied by a manual hose station. Where a hose station is indicated, one or more hose streams are available. In addition to fixed suppression capability, portable extinguishers are provided throughout the plant.
6. Reserved.
7. Wet pipe sprinklers for fire zone 2060 AF 20600 are provided for the turbine generator where oil piping is routed (for example, under lagging, at the turbine generator bearing seals, and in the exciter).
8. Wet pipe sprinklers for fire zone 2039 AF 20301 are provided by the adjacent general area wet pipe sprinkler system for fire zone 2030 AF 20300. See Section 9A.3, “Fire Protection Analysis Results”, for details.

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Table 9A-3D (Sheet 1 of 4)
Annex Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.[5]
4001 AF 01	**STAIRWELL S01**	**A**	**NO**	**640**	**PLASTIC/RUBBER**	**<15**	**120**	**10,000**	**NONE**	**NONE**
4001 AF 02	**Non-RCA Elevator**	**A**	**NO**	**60**	**PLASTIC/RUBBER CABLE INS.**	**<45**	**120**	**30,000**	**HEAT**	**HOSE STATION**
4002 AF 01	**STAIRWELL S02**	**A**	**NO**	**1628**	**PLASTIC/RUBBER**	**<15**	**120/0**	**10,000**	**NONE**	**NONE**
4002 AF 02	**STAIRWELL S04**	**A**	**NO**	**386**	**PLASTIC/RUBBER**	**<15**	**120/0**	**10,000**	**NONE**	**NONE**
4002 AF 03			**NO**	**418**			**120/0**	**120,000**	**SEE ZONE**	**SEE ZONE**
---	RCA Elevator	A		106	PLASTIC/RUBBER CABLE INS.	<45			SMOKE, HEAT	HOSE STATION
4002 AF 40603	Storage Room	D		106	PLASTIC/RUBBER	<60			SMOKE	HOSE STATION
4002 AF 40604	Security Room and Mechanical Room	B		206	PLASTIC/RUBBER	<90			SMOKE	HOSE STATION
4003 AF 01			**NO**	**7346**			**120/0**	**80,000**	**SMOKE (EXCEPT ROOMS 40504, 40602)**	**HOSE STATION**
4003 AF 40340	Demineralized water Deoxygenating Room	B		670	PLASTIC/RUBBER CABLE INS.	<90				
4003 AF 40442	Boric Acid Batching Room	B		670	PLASTIC/RUBBER	<90				
4003 AF 40503	Lower South Air Handling Equipment Room and Security Room	B		3003	PLASTIC/RUBBER CABLE INS.	<120				
4003 AF 40601	Upper South Air Handling Equipment Room	B		3003	PLASTIC/RUBBER CABLE INS.	<90				
4003 AF 02	**STAIRWELL S03**	**A**	**NO**	**990**	**PLASTIC/RUBBER**	**<15**	**120/0**	**10,000**	**NONE**	**NONE**
4031 AF 01			**NO**	**1468**			**120**	**160,000**	**SMOKE**	**HOSE STATION**
4031 AF 40307	Battery Room #1	E		734	PLASTIC/RUBBER	<90				

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Table 9A-3D (Sheet 2 of 4)
Annex Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.[5]
4031 AF 40308	Battery Charger Room #1	E		734	PLASTIC/RUBBER CABLE INS	<120				
4031 AF 02 4031 AF 40309	Battery Room #2	E	**NO**	**2855** 734	PLASTIC/RUBBER	<90	**120**	**160,000**	**SMOKE**	**HOSE**
4031 AF 40310	Battery Charger Room #2	E		734	PLASTIC/RUBBER CABLE INS	<120				
4031 AF 40411	Computer Room B	E		1128[c]	PLASTIC/RUBBER CABLE INS	<90				
4031 AF 40412	Battery Room	E		259	PLASTIC/RUBBER CABLE INS	<180				
4031 AF 05 4031 AF 40303	Corridor and Restroom	E	**NO**	**2852** 1285	PLASTIC/RUBBER CABLE INS	<120	**120/0**	**160,000**	**SMOKE**	**HOSE STATION**
4031 AF 40301	Security Rooms	B		1567	PLASTIC/RUBBER CABLE INS	<90				
4031 AF 06	**Security Area**	**B**	**NO**	**1099**	**PLASTIC/RUBBER CABLE INS**	**<120**	**120**	**80,000**	**SMOKE**	**HOSE STATION**
4032 AF 01	**Health Physics Area and Security Access Control Area**	**D**	**NO**	**6066**	**PLASTIC/RUBBER CABLE INS**	**<90**	**120/0**	**120,000**	**SMOKE**	**HOSE STATION**
4032 AF 02	**Containment Access Corridor and Radwaste Access Corridor**	**D**	**NO**	**2276**	**PLASTIC/RUBBER CABLE INS**	**<90**	**120/0**	**80,000**	**SMOKE**	**WET PIPE SPRINKLERS[6], HOSE STATION**
4033 AF 01	**Hot Machine Shop, Pump Seal Shop**	**D**	**NO**	**1847**	**PLASTIC/RUBBER CABLE INS**	**<120**	**120/0**	**120,000**	**SMOKE**	**HOSE STATION**
4034 AF 01 4034 AF 40311	Corridor and Entrance	B	**NO**	**19053** 2417	PLASTIC/RUBBER CABLE INS	<90	**120/0**	**120,000**	**SMOKE**	**HOSE STATION**

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Table 9A-3D (Sheet 3 of 4)
Annex Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.[5]
4034 AF 40313	Offices	B		344	PLASTIC/RUBBER	<90				
4034 AF 40316	Computer Room L1/2	B		514	PLASTIC/RUBBER	<90				
4034 AF 40317	Computer Room L3	B		560	PLASTIC/RUBBER	<90				
4034 AF 40318	ALARA Briefing and HP Monitoring Room	B		801	PLASTIC/RUBBER	<90				
4034 AF 40320	Women's Change Room	B		1027	PLASTIC/RUBBER	<90				
4034 AF 40322	Men's Change Room	B		2647	PLASTIC/RUBBER	<60				
4034 AF 40370	Kitchen, Conference Rooms, Restrooms, and Corridor	B		1347	PLASTIC/RUBBER	<120				
4034 AF 40378	Office Area West	D		3136	PLASTIC/RUBBER	<60				
4034 AF 40379	Office Area East	B		6260	PLASTIC/RUBBER	<60				
4035 AF 01	**Ancillary Diesel Generator Room**	**E**	**NO**	**235**	**PLASTIC/RUBBER**	**<60**	**180/0**	**240,000**	**HEAT**	**DRY PIPE SPRINKLERS, HOSE STATION**
4041 AF 01			**NO**	**5160**			**120/0**	**80,000**	**SMOKE**	**HOSE STATION**
4041 AF 40403	Control Support Area	B		3549	PLASTIC/RUBBER CABLE INS	<120				
4041 AF 40410	Computer Room A, Corridor	E		16114[d]	PLASTIC/RUBBER CABLE INS	<90				
4041 AF 02	**Corridor and Restroom**	**E**	**NO**	**1246**	**PLASTIC/RUBBER CABLE INS**	**<120**	**120**	**160,000**	**SMOKE**	**HOSE STATION**
4042 AF 01	**Electrical Switchgear Room #1**	**E**	**NO**	**3111**	**PLASTIC/RUBBER CABLE INS**	**<120**	**120/0**	**160,000**	**SMOKE**	**HOSE STATION**
4042 AF 02	**Electrical Switchgear Room #2**	**E**	**NO**	**3346**	**PLASTIC/RUBBER CABLE INS**	**<120**	**120/0**	**160,000**	**SMOKE**	**HOSE STATION**
4051 AF 01	**North Air Handling Equipment Room**	**B**	**NO**	**7304**	**PLASTIC/RUBBER CABLE INS**	**<90**	**120/0**	**80,000**	**SMOKE (EXCEPT RM 40501)**	**WET PIPE SPRINKLERS (RM 40500 ONLY), HOSE STATION**

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Table 9A-3D (Sheet 4 of 4)
Annex Building

Fire Area/Zone (1)	Fire Zone Description	Fire Severity Category	Safety Area (?)(2)	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)(4)	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.(5)
4052 AF 01 4052 AF 40550	Staging and Storage areas	D	**NO**	**9380** 8156	PLASTIC/RUBBER	<60	**120/0**	**120,000**	**SMOKE**	**SEE ZONE** WET PIPE SPRINKLERS, HOSE STATION
4052 AF 40551	Containment Air Filtration Exhaust Room A	B		610	PLASTIC/RUBBER	<60				HOSE STATION
4052 AF 40552	Containment Air Filtration Exhaust Room B			614	PLASTIC/RUBBER	<60				HOSE STATION

[c] 564 sq ft is the floor area above the sub-floor. An additional 564 sq ft is present in the sub-floor space.
[d] 1047 sq ft is the floor are above the sub-floor. An additional 564 sq ft is present in the sub-floor space.

Notes:

1. The first four digits of the fire area and fire zone numbers indicate the building, level and building area in which the fire area/zone is located. When the third or fourth digit is a zero, the fire area/zone spans more than one level or building area. The last two digits in a fire area number are a sequence number only. The last five digits in a fire zone number coincide with the room number of a prominent room in the fire zone.
2. A YES indication in the Safety Area column means that one or more safety-related components are located in the fire area.
3. Reserved.
4. The boundary fire resistance for each fire area represents the minimum resistance, in hours, for the surrounding walls, floor, and ceiling, except that:
   - A non-rated barrier capable of qualifying as a three-hour barrier is considered to have a resistance of three hours, provided that penetrations are adequately sealed.
   - Stairwells, elevator shafts and the like, which are enclosed by 2-hour (minimum) fire barrier walls, may comprise a portion of the boundary of a fire area having a three hour resistance.
   - Building exterior walls below grade (soil on the outside) are considered to have a fire resistance of at least three hours even though they are not fire-rated.

   A boundary fire resistance designation such as "180/0" indicates that part of the fire area boundary consists of nonrated building exterior walls/roof above grade, but that the minimum resistance of the fire barriers separating the fire area from adjacent fire areas is three hours. For detailed information about the fire resistance of fire area boundaries see Figures 9A-1 through 9A-5.
5. The fixed suppression capability is indicated for each fire area. Sprinkler and spray systems are automatic. In most cases, charcoal bed deluge systems require operator action to connect to a water source supplied by a manual hose station. Where a hose station is indicated, one or more hose streams are available. In addition to fixed suppression capability, portable extinguishers are provided throughout the plant.
6. Partial suppression coverage in this zone. See Section 9A.3, “Fire Protection Analysis Results”, for details.

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Table 9A-3E
Radwaste Building

Fire Area/Zone [1]	Fire Zone Description	Fire Severity Category	Safety Area (?)[2]	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)[4]	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.[5]
5031 AF 01			**NO**	**12,062**			**180/0**	**180,000**		
5031 AF 50300	ELECTRICAL / MECHANICAL EQUIPMENT ROOM	B		1015	CABLE INS.	<120			SMOKE, HEAT	PREACTION SPINKLERS, HOSE STATION
5031 AF 50350	MOBILE SYSTEMS FACILITY	E		5468	CABLE INS.	<90			SMOKE, HEAT	PREACTION SPINKLERS, HOSE STATION
5031 AF 50351	WASTE ACCUMULATION ROOM	E		2352	CABLE INS.	<90			SMOKE, HEAT	PREACTION SPINKLERS, HOSE STATION
5031 AF 50353	HVAC EQUIPMENT ROOM	B		829	PLASTIC/RUBBER	<120			SMOKE, HEAT	PREACTION SPINKLERS, HOSE STATION
5031 AF 50354	TRUCK STAGING AREA	E		861	CABLE INS.	<210			SMOKE, HEAT	PREACTION SPINKLERS, HOSE STATION
5031 AF 50355	MONITOR TANK ROOM	B		1537	CABLE INS.	<90			SMOKE, HEAT	PREACTION SPINKLERS, HOSE STATION

Notes:

1. The first four digits of the fire area and fire zone numbers indicate the building, level and building area in which the fire area/zone is located. When the third or fourth digit is a zero, the fire area/zone spans more than one level or building area. The last two digits in a fire area number are a sequence number only. The last five digits in a fire zone number coincide with the room number of a prominent room in the fire zone.
2. A YES indication in the Safety Area column means that one or more safety-related components are located in the fire area.
3. Reserved.
4. The boundary fire resistance for each fire area represents the minimum resistance, in hours, for the surrounding walls, floor, and ceiling, except that:
   - A non-rated barrier capable of qualifying as a three-hour barrier is considered to have a resistance of three hours, provided that penetrations are adequately sealed.
   - Stairwells, elevator shafts and the like, which are enclosed by 2-hour (minimum) fire barrier walls, may comprise a portion of the boundary of a fire area having a three hour resistance.
   - Building exterior walls below grade (soil on the outside) are considered to have a fire resistance of at least three hours even though they are not fire-rated.

   A boundary fire resistance designation such as "180/0" indicates that part of the fire area boundary consists of nonrated building exterior walls/roof above grade, but that the minimum resistance of the fire barriers separating the fire area from adjacent fire areas is three hours. For detailed information about the fire resistance of fire area boundaries see Figures 9A-1 through 9A-5.
5. The fixed suppression capability is indicated for each fire area. Sprinkler and spray systems are automatic. In most cases, charcoal bed deluge systems require operator action to connect to a water source supplied by a manual hose station. Where a hose station is indicated, one or more hose streams are available. In addition to fixed suppression capability, portable extinguishers are provided throughout the plant.

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Table 9A-3F (Sheet 1 of 2)
Diesel Generator

Fire Area/Zone (1)	Fire Zone Description	Fire Severity Category	Safety Area (?)(2)	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)(4)	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.(5)
6001 AF 01	**STAIRWELL**	**A**	**NO**	**224**	**PLASTIC/RUBBER**	**<15**	**120/0**	**10,000**	**SMOKE**	**DRY PIPE SPRINKLERS**
6030 AF 01			**NO**	**2356**			**180/0**	**240,000**		
6030 AF 60310	DIESEL GENERATOR ROOM A	E		1676	CABLE INS. VOLATILE LIQUID	<90			HEAT	DRY PIPE SPRINKLERS, HOSE STATION
6030 AF 60311	SERVICE MODULE A	E		300	CABLE INS.	<120			SMOKE	HOSE STATION
6030 AF 60313	COMBUSTION AIR CLEANER AREA A	E		380	VOLATILE LIQUID	<60			HEAT	HOSE STATION
6030 AF 02			**NO**	**2620**			**180/0**	**240,000**		
6030 AF 60320	DIESEL GENERATOR ROOM B	E		1676	CABLE INS. VOLATILE LIQUID	<90			HEAT	DRY PIPE SPRINKLERS, HOSE STATION
6030 AF 60321	SERVICE MODULE B	E		300	CABLE INS.	<90			SMOKE	HOSE STATION
6030 AF 60323	COMBUSTION AIR CLEANER AREA B	E		380	VOLATILE LIQUID	<60			HEAT	HOSE STATION
6030 AF 60324	TOOL STORAGE ROOM	D		132	PLASTIC/RUBBER	<60			NONE	DRY PIPE SPRINKLERS, HOSE STATION
6030 AF 60330	SECURITY ROOM	B		132	PLASTIC/RUBBER	<90			SMOKE	DRY PIPE SPRINKLERS, HOSE STATION
6030 AF 03	**DIESEL FUEL DAY TANK VAULT A**	**E**	**NO**	**183**	**VOLATILE LIQUID**	**<180**	**180**	**240,000**	**HEAT**	**DRY PIPE SPRINKLERS, HOSE STATION**

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Table 9A-3F (Sheet 2 of 2)
Diesel Generator

Fire Area/Zone (1)	Fire Zone Description	Fire Severity Category	Safety Area (?)(2)	Floor Area ft^2	AP1000 Design Primary Combustible Material	Equiv. Fire Dur (Min)	Bound. Fire Res. (Min)(4)	Max. Fire Load (BTU/ft^2)	Detect. Cap	Fixed Suppression Cap.(5)
6030 AF 04	DIESEL FUEL DAY TANK VAULT B	E	NO	180	VOLATILE LIQUID	<180	180	240,000	HEAT	DRY PIPE SPRINKLERS, HOSE STATION

Notes:

1. The first four digits of the fire area and fire zone numbers indicate the building, level and building area in which the fire area/zone is located. When the third or fourth digit is a zero, the fire area/zone spans more than one level or building area. The last two digits in a fire area number are a sequence number only. The last five digits in a fire zone number coincide with the room number of a prominent room in the fire zone.
2. A YES indication in the Safety Area column means that one or more safety-related components are located in the fire area.
3. Reserved.
4. The boundary fire resistance for each fire area represents the minimum resistance, in hours, for the surrounding walls, floor, and ceiling, except that:
   - A non-rated barrier capable of qualifying as a three-hour barrier is considered to have a resistance of three hours, provided that penetrations are adequately sealed.
   - Stairwells, elevator shafts and the like, which are enclosed by 2-hour (minimum) fire barrier walls, may comprise a portion of the boundary of a fire area having a three hour resistance.
   - Building exterior walls below grade (soil on the outside) are considered to have a fire resistance of at least three hours even though they are not fire-rated.

   A boundary fire resistance designation such as "180/0" indicates that part of the fire area boundary consists of nonrated building exterior walls/roof above grade, but that the minimum resistance of the fire barriers separating the fire area from adjacent fire areas is three hours. For detailed information about the fire resistance of fire area boundaries see Figures 9A-1 through 9A-5.
5. The fixed suppression capability is indicated for each fire area. Sprinkler and spray systems are automatic. In most cases, charcoal bed deluge systems require operator action to connect to a water source supplied by a manual hose station. Where a hose station is indicated, one or more hose streams are available. In addition to fixed suppression capability, portable extinguishers are provided throughout the plant.

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Table 9A-4
Ventilation Systems Serving Fire Areas Containing Class 1E Components

Ventilation Systems Serving Fire Areas Containing Class 1E Division A and C Components				
Fire Area	**RCA**	**Class 1E Components**	**Ventilation System - Subsystem**	**Distribution Header**
1202 AF 04	No	Division A batteries and equipment; Division A, B, C, & D interdivisional cable	VBS - A&C	AC-1
1242 AF 02	No	Division A equipment and penetrations; Division A & C cable	VBS - A&C	AC-1
1202 AF 03	No	Division C batteries, equipment & penetrations; Division A, B, C, & D interdivisional cable	VBS - A&C	AC-2
1200 AF 03	No	Division A & C cable only	VBS - A&C	AC-3
1230 AF 01	No	Division A & C cable only	VBS - B&D	BD-3
1240 AF 01	No	Division A & C cable only	VXS - Equip. room	Aux. building
1200 AF 01	Yes	Division A & C valves, equipment & cable	VAS - Aux/annex	Aux/annex
1250 AF 01	No	Division A & C cable only	VBS - A&C	AC-2
Ventilation Systems Serving Fire Areas Containing Class 1E Division B and D Components				
Fire Area	**RCA**	**Class 1E Components**	**Ventilation System - Subsystem**	**Distribution Header**
1201 AF 02	No	Division B batteries, equipment & penetrations; Division A, B, C, and D interdivisional cable	VBS - B&D	BD-1
1201 AF 03	No	Division D batteries, equipment & penetrations; Division B and D cable; Division A, B, C, and D interdivisional cable	VBS - B&D	BD-2
1201 AF 04	No	Division D valves; Division B & D equipment; Division B & D cable	VBS - B&D	BD-Common
1220 AF 01	No	Division B & D equipment & cable	VBS - A&C	AC-3
1201 AF 05	No	Division B & D valves; Division B & D cable	VXS	Self-contained
1201 AF 06	No	Division B & D valves	VXS	Self-contained
1230 AF 02	No	Division B & D cable only	VXS - Equip. Room	Aux. building
1220 AF 02	Yes	Division D valves	VAS - Aux/annex	Aux. building
Ventilation Systems Serving Principal Class 1E Fire Areas				
Fire Area	**RCA**	**Description**	**Ventilation System - Subsystem**	**Distribution Header**
1242 AF 01	No	Main control room	VBS - MCR/CSA	MCR
1232 AF 01	No	Remote shutdown room	VBS - B&D	BD-3
1210 AF 01	No	Spare battery fire area	VBS - A&C	AC-3
1243 AF 01	No	Reactor trip switchgear I	VXS - Equip. room	Aux. building
1243 AF 02	No	Reactor trip switchgear II	VXS - Equip. room	Aux. building
1202 AF 05	No	Stair S05	VBS - MCR/CSA	N/A
1000 AF 01	Yes	Containment/shield building	VFS	A and B

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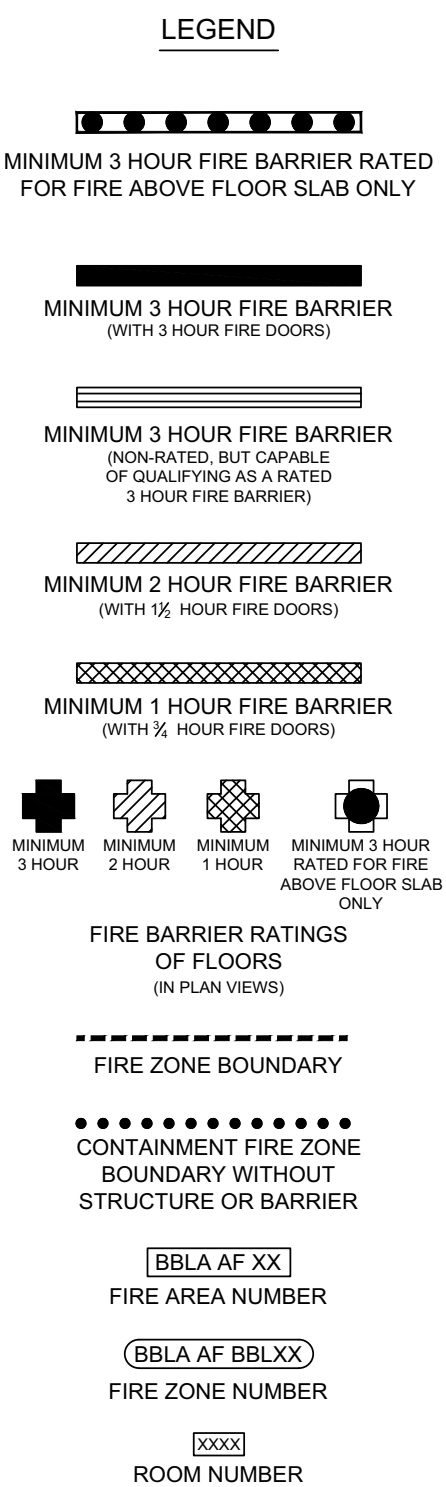

Figure 9A-1 (Sheet 1 of 16)
Fire Areas Legend

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Figure 9A-1 (Sheet 2 of 16)
Nuclear Island Fire Area
Plan at Elevation 66'-6"

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Figure 9A-1 (Sheet 3 of 16)
Nuclear Island Fire Area
Plan at Elevation 82'-6"

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Figure 9A-1 (Sheet 4 of 16)
Nuclear Island Fire Area
Plan at Elevation 96'-6″

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Figure 9A-1 (Sheet 5 of 16)
Nuclear Island Fire Areas
Plan at Elevation 100'-0″ & 107'-2″

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Figure 9A-1 (Sheet 6 of 16)
Nuclear Island Fire Area
Plan at Elevation 117'-6″

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Figure 9A-1 (Sheet 7 of 16)
Nuclear Island Fire Area
Plan at Elevation 135'-3″

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Figure 9A-1 (Sheet 8 of 16)
Nuclear Island Fire Areas
Plan at Elevation 153'-0" & 160'-6″

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Figure 9A-1 (Sheet 9 of 16)
Nuclear Island Fire Areas
Plan at Elevation 160'-6″, 180'-0″, & 329'-0"

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Figure 9A-1 (Sheet 10 of 16)
Nuclear Island Fire Area
Section A-A

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Figure 9A-1 (Sheet 11 of 16)
Nuclear Island Fire Area
Section B-B

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Figure 9A-1 (Sheet 12 of 16)
Nuclear Island Fire Areas
Section C-C & H-H

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Figure 9A-1 (Sheet 13 of 16)
Nuclear Island Fire Area
Section G-G

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Figure 9A-1 (Sheet 14 of 16)
Nuclear Island Fire Area
Section J-J

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Figure 9A-1 (Sheet 15 of 16)
Nuclear Island Fire Area
Section K-K

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Figure 9A-1 (Sheet 16 of 16)
Nuclear Island Fire Areas
Section I-I & R-R

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Figure 9A-2 (Sheet 1 of 5)
Turbine Building Fire Area
Plan at Elevation 100′-0″

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Figure 9A-2 (Sheet 2 of 5)
Turbine Building Fire Area
Plan at Elevation 120'-6"

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Figure 9A-2 (Sheet 3 of 5)
Turbine Building Fire Area
Plan at Elevation 141′-3″

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Figure 9A-2 (Sheet 4 of 5)
Turbine Building Fire Area
Plan at Elevation 170′-0″

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Figure 9A-2 (Sheet 5 of 5)
Turbine Building Fire Areas
Plan at Elevation 255′-3″

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(Note: This figure replaces Figure 9A-3 Sheet 1 of 3.)

Figure 9A-201
Annex I & II Building Fire Areas
Plan at Elevation 100'-0" & 107'-2"

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Figure 9A-3 (Sheet 2 of 3)
Annex I & II Building Fire Area
Plan at Elevation 117′-6″

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Figure 9A-3 (Sheet 3 of 3)
Annex I & II Building Fire Area
Plan at Elevation 135′-3″ through 183′-6″

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Figure 9A-4
Radwaste Building Fire Area
Plan at Elevation 100′-0″

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Figure 9A-5
Diesel Generator Building Fire Area
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Chapter 10 Steam and Power Conversion

10.1 Summary Description

The steam and power conversion system is designed to remove heat energy from the reactor coolant system via the two steam generators and to convert it to electrical power in the turbine-generator. The main condenser deaerates the condensate and transfers heat that is unusable in the cycle to the circulating water system. The regenerative turbine cycle heats the feedwater, and the main feedwater system returns it to the steam generators.

Table 10.1-1 gives the significant design and performance data for the major system components. Figure 10.1-1 shows the rated heat balance for the turbine cycle process.

10.1.1 General Description

The steam generated in the two steam generators is supplied to the high-pressure turbine by the main steam system. After expansion through the high-pressure turbine, the steam passes through the two moisture separator/reheaters (MSRs) and is then admitted to the three low-pressure turbines. A portion of the steam is extracted from the high- and low-pressure turbines for seven stages of feedwater heating.

Exhaust steam from the low-pressure turbines is condensed and deaerated in the main condenser. The heat rejected in the main condenser is removed by the circulating water system (CWS). The condensate pumps take suction from the condenser hotwell and deliver the condensate through four stages of low-pressure closed feedwater heaters to the fifth stage, open deaerating heater. Condensate then flows to the suction of the steam generator feedwater booster pump and is discharged to the suction of the main feedwater pump. The steam generator feedwater pumps discharge the feedwater through two stages of high-pressure feedwater heating to the two steam generators.

The turbine-generator has an output of approximately 1,199,500 kW for the Westinghouse nuclear steam supply system (NSSS) thermal output of 3,415 MWt. The principal turbine-generator conditions for the turbine rating are listed in Table 10.1-1. The rated system conditions for the NSSS are listed in Table 10.1-1. The systems of the turbine cycle have been designed to meet the maximum expected turbine generator conditions.

Instrumentation systems are designed for the normal operating conditions of the steam and condensate systems. The systems are designed for safe and reliable control and incorporate requirements for performance calculations and periodic heat balances. Instrumentation for the secondary cycle is also provided to meet recommendations by the turbine supplier and ANSI/ASME TDP-2-1985, "Recommended Practices for the Prevention of Water Damage to Steam Turbines Used for Electric Power Generation." Design features for prevention of water hammer in the steam generator are described in Subsection 5.4.2.2. Continuous or semi-continuous sampling instrumentation and grab sample points are provided so that water chemistry in the secondary cycle can be maintained within acceptable limits, as required by the nuclear steam system and turbine suppliers (see Subsections 9.3.4 and 10.3.5). Condenser tube/tube sheet leakage can be identified and isolated by using condenser conductivity sampling provisions.

Criteria and bases for safety-related instrumentation for main steam isolation are discussed in Section 7.3.

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10.1.2 Protective Features

Loss of External Electrical Load and/or Turbine Trip Protection

In the event of turbine trip, steam is bypassed to the condenser via the turbine bypass valves and, if required, to the atmosphere via the atmospheric relief valves. Steam relief permits energy removal from the reactor coolant system. Load rejection capability is discussed in Subsections 10.3.2.3.1 and 15.2.2.

Overpressure Protection

Spring-loaded safety valves are provided on both main steam lines, in accordance with the ASME Code, Section III. The pressure relief capacity of the safety valves is such that the energy generated at the high-flux reactor trip setting can be dissipated through this system. The design capacity of the main steam safety valves equals or exceeds 105 percent of the NSSS design steam flow at an accumulation pressure not exceeding 110 percent of the main steam system design pressure. Overpressure protection for the main steam lines is a safety-related function. The main steam safety valves are described in Subsection 10.3.2.

In addition, the shell sides of the feedwater heaters and the moisture separator/reheaters are provided with overpressure protection in accordance with ASME Code, Section VIII, Division 1, or equivalent standards.

Loss of Main Feedwater Flow Protection

The startup feedwater pumps provide feedwater to the steam generators for the removal of sensible and decay heat whenever main feedwater flow is interrupted, including loss of normal ac power and offsite power. This system is described in Subsection 10.4.9.

Turbine Overspeed Protection

During normal operations, turbine overspeed protection is provided by the action of the redundant controller of the electro-hydraulic control system. Additional protection is provided by an overspeed protection system, which continuously monitors critical turbine parameters on a three-channel basis. Each of the channels is independently testable under load with overspeed protection during testing provided by the channels not being tested. If turbine speed exceeds 110 percent of rated speed, the electronic trip system causes steam supply valves to close, tripping the unit. This system is described in Subsection 10.2.2.5.

Turbine Missile Protection

Turbine rotor integrity minimizes the probability of generating turbine missiles and is discussed in Subsection 10.2.3. Turbine missiles are addressed in Subsection 3.5.1.3. The favorable orientation of the turbine-generator directs potential missiles away from safety-related equipment and structures.

Radioactivity Protection

Under normal operating conditions, there are no significant radioactive contaminants present in the steam and power conversion system. However, it is possible for the system to become contaminated through steam generator tube leakage. In this event, radiological monitoring of the main condenser air removal system, the steam generator blowdown system, and the main steam lines will detect contamination and alarm high radioactivity concentrations. A discussion of the radiological aspects of primary-to-secondary system leakage and limiting conditions for operation is contained in Chapter 11. The steam generator blowdown system described in Subsection 10.4.8 and the condensate polishing system described in Subsection 10.4.6 serve to limit the radioactivity level in the secondary cycle.

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Erosion-Corrosion Protection

Erosion-corrosion resistant materials are used in steam and power conversion systems for components exposed to single-phase or two-phase flow where significant erosion can occur. Factors considered in the evaluation of erosion-corrosion include system piping and component configuration and geometry, water chemistry, piping and component material, fluid temperature, and fluid velocity. Carbon steel with only carbon and manganese alloying agents is not used for applications subject to significant erosion-corrosion.

In addition to material selection, pipe size and layout may also be used to minimize the potential for erosion-corrosion in systems containing water or two-phase flow. The secondary side water chemistry (see Subsection 10.3.5) uses a volatile pH adjustment chemical to maintain a noncorrosive environment. Steam and power conversion systems are designed to facilitate inspection and erosion-corrosion monitoring programs.

An industry-sponsored computer program developed for nuclear and fossil power plant applications is used to evaluate the rate of wall thinning for components and piping potentially susceptible to erosion-corrosion. The engineering models are the result of research and development in the fields of material science, water chemistry, fluid mechanics, and corrosion engineering. The program quantifies the benefits of piping material, system layout, and sizing considerations used to reduce corrosion rates.

10.1.3 Combined License Information on Erosion-Corrosion Monitoring

10.1.3.1 Erosion-Corrosion Monitoring

The flow accelerated corrosion (FAC) monitoring program analyzes, inspects, monitors and trends those nuclear power plant components that are potentially susceptible to erosion-corrosion damage such as carbon steel components that carry wet steam. In addition, the FAC monitoring program considers the information of Generic Letter 89-08, EPRI NSAC-202L-R4, and industry operating experience. The program requires a grid layout for obtaining consistent pipe thickness measurements when using Ultrasonic Test Techniques. The FAC program obtains actual thickness measurements for highly susceptible FAC locations for new lines as defined in EPRI NSAC-202L-R4 (Reference 201). At a minimum, a CHECWORKS type Pass 1 analysis is used for low and highly susceptible FAC locations and a CHECWORKS type Pass 2 analysis is used for highly susceptible FAC locations when Pass 1 analysis results warrant. To determine wear of piping and components where operating conditions are inconsistent or unknown, the guidance provided in EPRI NSAC-202L is used to determine wear rates.

10.1.3.1.1 Analysis

An industry-sponsored program is used to identify the most susceptible components and to evaluate the rate of wall thinning for components and piping potentially susceptible to FAC. Each susceptible component is tracked in a database and is inspected, based on susceptibility. Analytical methods utilize the results of plant-specific inspection data to develop plant-specific correction factors. This correction accounts for uncertainties in plant data, and for systematic discrepancies caused by plant operation. For each piping component, the analytical method predicts the wear rate, and the estimated time until it must be re-inspected, repaired, or replaced. Carbon steel piping (ASME III and ASME B31.1) that is used for single or multi-phase high temperature flow are the most susceptible to erosion-corrosion damage and receive the most critical analysis.

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10.1.3.1.2 Industry Experience

Review and incorporation of industry experience provides a valuable supplement to plant analysis. Industry experience is used to update the program by identifying susceptible components or piping features.

10.1.3.1.3 Inspections

Wall thickness measurements establish the extent of wear in a given component, provide data to help evaluate trends, and provide data to refine the predictive model. Components are inspected for wear using ultrasonic techniques (UT), radiography techniques (RT), or by visual observation. The initial inspections are used as a baseline for later inspections. Each subsequent inspection determines the wear rate for the piping and components and the need for inspection frequency adjustment for those components.

10.1.3.1.4 Training and Engineering Judgement

The FAC program is administered by both trained and experienced personnel. Task specific training is provided for plant personnel that implement the monitoring program. Specific non-destructive examination (NDE) is carried out by personnel qualified in the given NDE method. Inspection data is analyzed by engineers or other experienced personnel to determine the overall effect on the system or component.

10.1.3.1.5 Long-Term Strategy

This strategy focuses on reducing wear rates and performing inspections on the most susceptible locations.

10.1.3.2 Procedures

10.1.3.2.1 Generic Plant Procedure

The FAC monitoring program is governed by procedure. This procedure contains the following elements:

- A requirement to monitor and control FAC.
- Identification of the tasks to be performed and associated responsibilities.
- Identification of the position that has overall responsibility for the FAC monitoring program at each plant.
- Communication requirements between the coordinator and other departments that have responsibility for performing support tasks.
- Quality Assurance requirements.
- Identification of long-term goals and strategies for reducing high FAC wear rates.
- A method for evaluating plant performance against long-term goals.

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10.1.3.2.2 Implementing Procedures

The FAC implementing procedures provide guidelines for controlling the major tasks. The plant procedures for major tasks are as follows:

- Identifying susceptible systems.
- Performing FAC analysis.
- Selecting and scheduling components for initial inspection.
- Performing inspections.
- Evaluating degraded components.
- Repairing and replacing components when necessary.
- Selecting and scheduling locations for the follow-on inspections.
- Collection and storage of inspections records.

10.1.3.3 Plant Chemistry

The responsibility for system chemistry is under the purview of the plant chemistry section. The plant chemistry section specifies chemical addition in accordance with plant procedures.

10.1.4 References

201. EPRI NSAC-202L-R4, Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R4), Electric Power Research Institute (EPRI) Technical Report 3002000563, Palo Alto, CA, 2013.

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Table 10.1-1
Significant Design Features and
Performance Characteristics for Major
Steam and Power Conversion System Components

Nuclear Steam Supply System, Full Power Operation	
Rated NSSS power (MWt)	3415
Steam generator outlet pressure (psig)	821
Steam generator inlet feedwater temperature (°F)	440
Maximum steam generator secondary separator outlet steam moisture (%)	0.25
Steam generator outlet steam temperature (°F)	523
Quantity of steam generators	2
Flow rate per steam generator (lb/hr)	7.50×10^6
Turbine	
Nominal output (kW)	1,199,500
Turbine type	Tandem-compound 6-flow, 52 in. last-stage blade
Turbine elements	1 high pressure 3 low pressure
Operating speed (rpm)	1800

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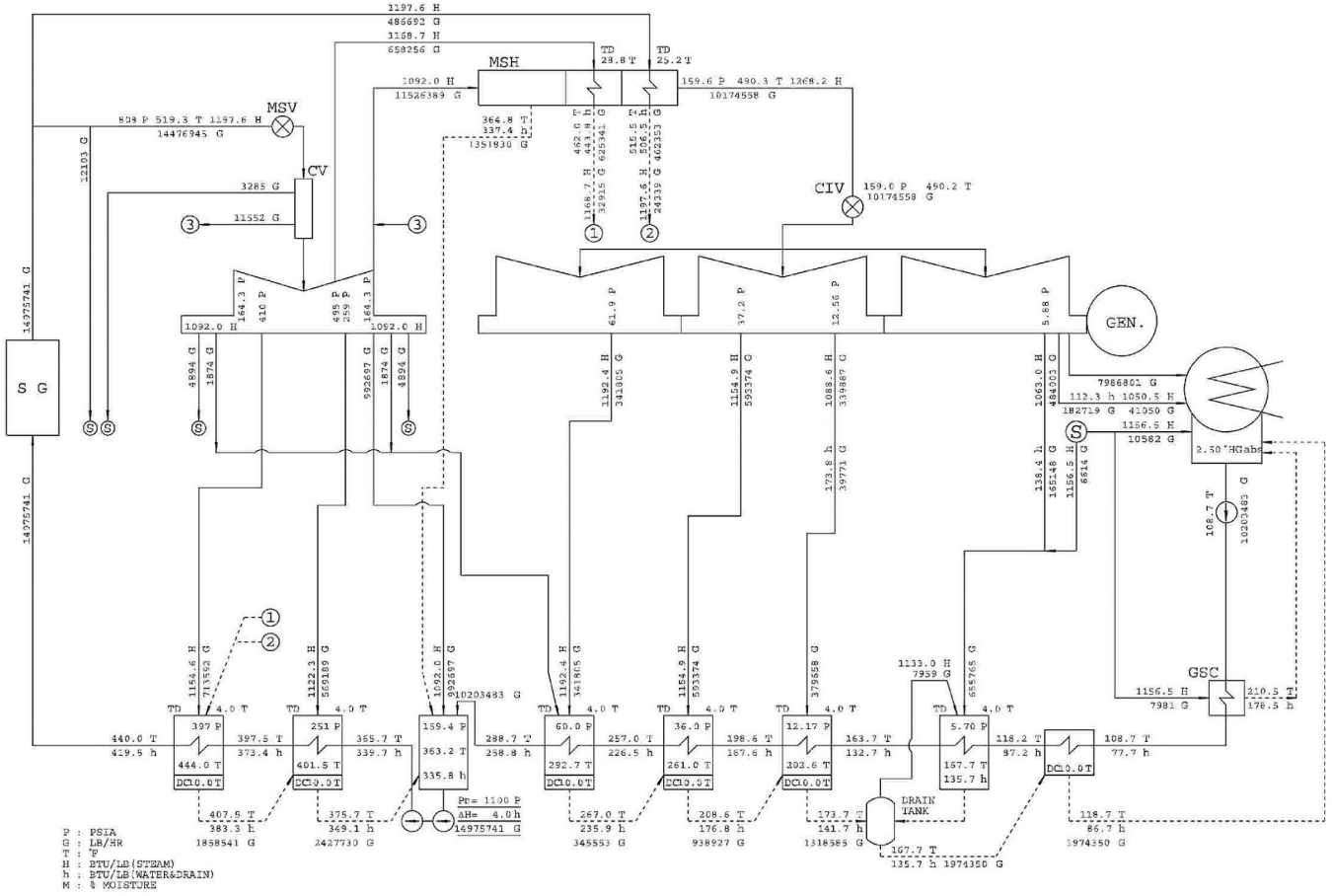

Figure 10.1-1
Rated Heat Balance

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10.2 Turbine-Generator

The function of the turbine-generator is to convert thermal energy into electric power.

10.2.1 Design Basis

10.2.1.1 Safety Design Basis

The turbine-generator serves no safety-related function and therefore has no nuclear safety design basis.

10.2.1.2 Power Generation Design Basis

The following is a list of the principal design features:

- The turbine-generator is designed for baseload operation and for load follow operation.
- The main turbine system (MTS) is designed for electric power production consistent with the capability of the reactor and the reactor coolant system.
- The turbine-generator is designed to trip automatically under abnormal conditions.
- The system is designed to provide proper drainage of related piping and components to prevent water induction into the main turbine.
- The main turbine system satisfies the recommendations of Nuclear Regulatory Commission Branch Technical Position ASB 3-1 as related to breaks in high-energy and moderate-energy piping systems outside containment. The main turbine system is considered a high-energy system.
- The system provides extraction steam for seven stages of regenerative feedwater heating.

10.2.2 System Description

The turbine-generator is designated as a TC6F 52-inch last-stage blade unit consisting of turbines, a generator, external moisture separator/reheaters, controls, and auxiliary subsystems. (See Figure 10.2-1.) The major design parameters of the turbine-generator and auxiliaries are presented in Table 10.2-1. The piping and instrumentation diagram containing the stop, control, intercept, and reheat valves is shown in Figure 10.3.2-2.

The turbine-generator and associated piping, valves, and controls are located completely within the turbine building. There are no safety-related systems or components located within the turbine building. The probability of destructive overspeed condition and missile generation, assuming the recommended inspection and test frequencies, is less than 1×10^{-5} per year. In addition, orientation of the turbine-generator is such that a high-energy missile would be directed at a 90 degree angle away from safety-related structures, systems, or components. Failure of turbine-generator equipment does not preclude safe shutdown of the reactor. The turbine-generator components and instrumentation associated with turbine-generator overspeed protection are accessible under operating conditions. Subsection 3.5.1.3 addresses the probability of generation of a turbine missile for AP1000 plants in a side-by-side configuration.

Preoperational and startup tests provide guidance to operations personnel to ensure the proper operability of the turbine generator system.

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10.2.2.1 Turbine-Generator Description

The turbine is a 1800-rpm, tandem-compound, six-flow, reheat unit with 52-inch last-stage blades (TC6F 52-inch LSB). The high-pressure turbine element includes one double-flow, high-pressure turbine. The low-pressure turbine elements include three double-flow, low-pressure turbines and two external moisture separator/reheaters (MSRs) with two stages of reheating. The single direct-driven generator is hydrogen gas and de-ionized water cooled and rated at 1375 MVA minimum at 0.90 PF lagging, 0.95 PF leading. Other related system components include a complete turbine-generator bearing lubrication oil system, a digital electrohydraulic (D-EHC) control system with supervisory instrumentation, a turbine steam sealing system (refer to Subsection 10.4.3), overspeed protective devices, turning gear, a stator cooling water system, a generator hydrogen and seal oil system, a generator CO_2 system, a rectifier section, an excitation transformer, and a voltage regulator.

The turbine-generator foundation is a spring-mounted support system. A spring-mounted turbine-generator provides a low-tuned, turbine-pedestal foundation. The springs dynamically isolate the turbine-generator deck from the remainder of the structure in the range of operating frequencies, thus allowing for an integrated structure below the turbine deck. The condenser is supported on springs and attached rigidly to the low-pressure turbine exhausts.

The foundation design consists of a reinforced concrete deck mounted on springs and supported on a structural steel frame that forms an integral part of the turbine building structural system. The lateral bracing under the turbine-generator deck also serves to brace the building frame. This "integrated" design reduces the bracing and number of columns required in the building. Additionally, the spring-mounted design allows for dynamic uncoupling of the turbine-generator foundation from the substructure. The spring mounted support system is much less site dependent than other turbine pedestal designs, since the soil structure is decoupled from turbine dynamic effects. The turbine-generator foundation consists of a concrete table top while the substructure consists of supporting beams and columns. The structure below the springs is designed independent of vibration considerations. The turbine-generator foundation and equipment anchorage are designed to the same seismic design requirement as the turbine building. See Subsection 3.7.2.8 for additional information on seismic design requirements. See Subsection 10.4.1.2 for a description of the support of the condenser.

10.2.2.2 Turbine-Generator Cycle Description

Steam from each of two steam generators enters the high-pressure turbine through four stop valves and four governing control valves. Crossties are provided upstream of the turbine stop valves to provide pressure equalization with one or more stop valves closed. Downstream of the pressure equalization header, steam from each of the main steam line flow paths A and B splits and enters the high pressure turbine through two parallel stop valves in series with two associated control valves. The bodies of the turbine stop valves in the branches of each main steam line flow path are cross-connected, and provide steam flow to both of the associated downstream control valves. After expanding through the high-pressure turbine, exhaust steam flows through two external moisture separator/reheater vessels. The external moisture separators reduce the moisture content of the high-pressure exhaust steam from approximately 10 to 13 percent at the rated load to 0.5 percent moisture or less.

The AP1000 employs a 2 stage reheater, of which the first stage reheater uses the extraction steam from the high pressure turbine and the second reheater uses a portion of the main steam supply to reheat the steam to superheated conditions. The reheated steam flows through separate reheat stop and intercept valves in each of six reheat steam lines leading to the inlets of the three low-pressure turbines. Turbine steam extraction connections are provided for seven stages of feedwater heating. Steam from the extraction points of the high-pressure turbine is supplied to high-pressure feedwater heater No. 6 and No. 7. The high-pressure turbine exhaust also supplies steam to the deaerating

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feedwater heater. The low-pressure turbine third, fourth, fifth, and sixth extraction points supply steam to the low-pressure feedwater heaters No. 4, 3, 2, and 1, respectively.

Moisture is removed at a number of locations in the blade path. The no-return drain catchers provided at the nozzle diaphragms (stationary blade rings) accumulate the water fraction of the wet steam, and the accumulated water discharges into each extract, reheat, and exhaust lines directly or through drainage holes drilled through the nozzle diaphragms. A few grooves are provided on the rotating blades near the last stage of the low-pressure turbines to capture the large water droplets of the wet steam and to enhance the moisture removal effectiveness.

The external moisture separator/reheaters use multiple vane chevron banks (shell side) for moisture removal. The moisture removed by the external moisture separator/reheaters drain to a moisture separator drain tank and is pumped to the deaerator.

Condensed steam in the reheater (tube side) is drained to the reheater drain tank, flows into the shell side of the No. 7 feedwater heater, and cascades to the No. 6 feedwater heater.

10.2.2.3 Excitation System

The excitation system is a static excitation system using the thyristor full bridge rectifier.

Excitation power used in this system is fed from the generator main lead through the excitation transformer. The excitation transformer is of outdoor use type, and it will be located adjacent to the turbine building. After stepping down the voltage at the excitation transformer, ac current from the generator main lead will be rectified by the thyristor rectifier.

The voltage control system uses the digital controller for major control function. The system has two master controller configurations (i.e., one is for normal operation and the other is stand-by). The voltage setting range of the automatic voltage regulator is +/-10% of the generator rated voltage; however, the operating range of the generator is +/-5% of the generator rated voltage. The excitation system is equipped with a power system stabilizer.

10.2.2.4 Digital Electrohydraulic System Description

The turbine-generator is equipped with a digital electrohydraulic (D-EHC) system that combines the capabilities of redundant processors and high-pressure hydraulics to regulate steam flow through the turbine. The control system provides the functions of speed control, load control, overspeed protection, and automatic turbine control (ATC), which may be used, either for control or for supervisory purposes, at the option of the plant operator.

The D-EHC system employs three electric speed inputs whose signals are processed in redundant processors. Valve opening actuation is provided by a hydraulic system that is independent of the bearing lubrication system. Valve closing actuation is provided by springs and steam forces upon reduction or relief of fluid pressure. The system is designed so that loss of fluid pressure, for any reason, leads to valve closing and consequent turbine trip.

Steam valves are provided in an in-line configuration. The stop valves are tripped by the overspeed trip system; the control valves are modulated by the control system and are also actuated by the trip system.

10.2.2.4.1 Speed Control (Normal Turbine Operation)

Three active speed sensors provide signals for the turbine rotation rate. These are the 3 probes A,B,C shown on Figure 10.2-2. The 3 signals are input to 3 separate speed detection modules each

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located on three separate I/O branches. Each of these modules has an onboard microprocessor which converts the sensor input to a turbine rpm value.

Independence of the three speed sensor signal branches is assured in that failure of the transmission of one branch of the signal does not affect the transmission of the signal in the other two branches. Each I/O branch is separately fused and fed by redundant power sources. Also, failure of a speed detection module (receiving one branch of the signal) does not affect the function of the remaining two speed detection modules from receiving their signals.

The speed control function of the turbine control and protection system's redundant controller provides speed control and acceleration functions for normal turbine operation. The speed error signal is derived by comparing the desired setpoint speed with the actual speed of the turbine. This error drives an algorithm that positions the control valves at the desired setpoint. Acceleration rates can also be entered by the operator or calculated by the control system in the auto start-up mode. A failure of one speed input generates an alarm. Failure of two or more speed inputs also generates an alarm and trips the turbine. In addition, if both of the Ovation controllers fail or power to them is lost, the turbine will trip. The design is fail safe; the system is designed to de-energize to trip.

The speed control function exists in triplicate channels, which include the load (frequency) control function if the main generator breaker is closed. If one channel fails, the average of the two remaining signals is fed into the valve positioning control function.

The control system's operator automatic (OA) controller provides the speed control function. At 101% of rated speed, the control valves and intercept valves begin to close, but do not trip the turbine.

The speed control function is designed to prevent the operator from holding the turbine speed at a bearing critical or blade resonance point.

10.2.2.4.2 Load Control

The load control function of the turbine control and protection system's OA controller develops signals that are used to regulate unit load. Signal outputs are based on a proper combination of the speed error and megawatt setpoints to generate a flow demand to the control valves.

When the first-stage pressure and megawatt control loops are out of service, steam flow is not controlled by feedback, but rather by a characterization of maximum nozzle flow (at rated pressure and temperature) per valve versus control valve position. Under this condition, the turbine operator requests a certain megawatt load target. The control system calculates the required flow demand to adjust the steam flow from the steam generators supplied to the turbine.

10.2.2.4.3 Valve Control

The flow of the main steam entering the high-pressure turbine is controlled by four stop valves and four control valves. The function of the stop valves is to shut off the steam flow to the turbine when required. The stop valves are closed by actuation of the overspeed trip system. This system is independent of the control system. Stop valves No. 1 and No. 3 are controlled by a hydraulic actuator so that the stop valve is either fully open or fully closed. The No. 2 and No. 4 stop valves each has a bypass valve, which is controlled by an electro-hydraulic servo actuator for control valve warming.

The turbine control valves are positioned by electrohydraulic servo actuators in response to signals from their respective servo modules. The servo module signal positions the control valves for wide-range speed control through the normal turbine operating range, and for load control after the turbine-generator unit is synchronized.

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The reheat stop and intercept valves, located in the hot reheat lines at the inlet to the low-pressure turbines, control steam flow to the low-pressure turbines. During normal operation of the turbine, the reheat stop valves open under hydraulics control. The intercept valve independent servo module positions valves Nos. 1, 3, and 5 during startup and normal operation. Intercept valves Nos. 2, 4, and 6 open under hydraulics control. The intercept valves close rapidly on loss of turbine load. The reheat stop and intercept valves close completely on a turbine trip.

The control, stop, reheat stop, and intercept valves have fast-acting valves connected to the hydraulic portion of their respective valve actuators. Opening a fast-acting valve causes the connected control or stop valve to rapidly close. The fast-acting valve actuators are connected to trip headers and open in response to loss of pressure in the connected trip header. The control and intercept fast-acting valves are connected to the relay trip header, and stop and reheat stop fast-acting valves are connected to the auto stop emergency trip header. Valve closure times are provided in Table 10.2-4.

10.2.2.4.4 Power/Load Unbalance (Main Breaker Closed)

A power/load unbalance circuit initiates fast closing control valve and intercept valve action under load rejection conditions that might lead to rapid rotor acceleration and consequent overspeed.

Load unbalance operates when load is equal to or greater than 40% of full load and reheat pressure exceeds the corresponding expected megawatts. Cold reheat pressure is used as a measure of power. Generator current is used as a measure of load to provide discrimination between loss of load incidents and occurrences of electric system faults. This causes the control and intercept valves to close quickly under fast-acting solenoid valve action.

When the circuitry detects a power/load unbalance condition, the turbine control and protection system initiates closure of the control and intercept valves by energizing fast-acting solenoid valves on the valve actuators. One-half of the intercept valves are “controlling” and are equipped with a servo valve. The “controlling” intercept valves allow recovery from a partial load rejection by blowing down the reheated steam. Simultaneously, the load reference signal is reduced to a specific value (approximately 60% of full turbine load). When the condition disappears, the power/load unbalance circuitry resets automatically and the load reference signal is recalculated based on the new calculation of flow demand.

10.2.2.4.5 Overspeed Protection

The turbine control and protection system has four functions to protect the turbine against overspeed. The first is the overspeed protection system (OSP), which at 101% of rated speed, begins to close the control and intercept valves as discussed in Subsection 10.2.2.4.1. The second and third are the 110% emergency electrical and 111% backup electrical overspeed trip functions also discussed in Subsection 10.2.2.5. The fourth function is the partial load unbalance discussed in Subsection 10.2.2.4.4.

Redundancy is built into the overspeed protection system. The failure of a single valve will not disable the trip functions. The overspeed protection components are designed to fail in a safe position. Loss of the hydraulic pressure in the emergency trip system causes a turbine trip. Therefore, damage to the overspeed protection components, results in the closure of the valves and the interruption of steam flow to the turbine.

Quick closure of the steam valves prevents turbine overspeed. Valve closing times are given in Table 10.2-4.

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10.2.2.4.6 Automatic Turbine Control

Automatic turbine control provides safe and proper startup and loading of the turbine generator. The applicable limits and precautions are monitored by the automatic turbine control programs even if the automatic turbine control mode has not been selected by the operator. When the operator selects automatic turbine control, the programs both monitor and control the turbine. The D-EHC controller takes advantage of the capability of the control system to scan, calculate, make decisions, and take positive action.

The automatic turbine control is capable of automatically:

- Changing speed reference
- Changing acceleration rates
- Generating speed holds
- Changing load rates
- Generating load holds

The thermal stresses in the rotor are calculated by the automatic turbine controls programs based on actual turbine steam and metal temperatures as measured by thermocouples or other temperature measuring devices. Once the thermal stress (or strain) is calculated, it is compared with the allowable value, and the difference is used as the index of the permissible first stage temperature variation. This permissible temperature variation is translated in the computer program as an allowable speed or load or rate of change of speed or load.

Values of some parameters are stored for use in the prediction of their future values or rates of change, which are used to initiate corrective measures before alarm or trip points are reached.

The rotor stress (or strain) calculations used in the program, and its decision-making counterpart are the main controlling sections. They allow the unit to roll with relatively high acceleration until the anticipated value of stress predicts that limiting values are about to be reached. Then a lower acceleration value is selected and, if the condition persists, a speed hold is generated. The same philosophy is used on load control in order to maintain positive control of the loading rates.

The automatic turbine controls programs are stored and executed in redundant distributed processing units, which contain the rotor stress programs and the automatic turbine controls logic programs. Once the turbine is reset, the automatic turbine controls programs are capable of rolling the turbine from turning gear to synchronous speed.

Once the turbine-generator reaches synchronous speed, the startup or speed control phase of automatic turbine control is completed and no further action is taken by the programs. Upon closing the main generator breaker, the D-EHC automatically picks up approximately 5 percent of rated load to prevent motoring of the generator. At this time, the D-EHC is in load control logic and automatically reverts control to the operator mode.

The operator can also select the automatic turbine control mode. The automatic turbine control selects the loading rate (based on turbine temperature) and allows load changes until an alarm condition occurs. If the operating parameters being monitored (including rotor stress) exceed their associated alarm limit, a load hold is generated in conjunction with the appropriate alarm message. The D-EHC generates the load hold by ignoring any further load increase or decrease until the alarm condition is cleared or until the operator overrides the alarm condition.

The operator may remove the turbine-generator from automatic turbine control. This action places the turbine control in operator auto mode and the automatic turbine control in a supervisory capacity.

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10.2.2.5 Turbine Protective Trips

Turbine protective trips, when initiated, cause tripping of the main stop, control, intercept, and reheat stop valves. The protective trips are:

- Low bearing oil pressure
- Low electrohydraulic fluid pressure
- High condenser back pressure
- Turbine overspeed
- Thrust bearing wear
- Remote trip that accepts external trips

A description of the trip system for turbine overspeed is provided below.

10.2.2.5.1 Overspeed Trip System

The purpose of the electrical overspeed trip system is to detect undesirable operating conditions of the turbine-generator, take appropriate trip actions, and provide information to the operator about the detected conditions and the corrective actions. In addition, means are provided for testing emergency trip equipment and circuits.

The system hydraulic manifold configuration permits on line testing with continuous protection afforded during the test sequence.

The trip system includes an online testable hydraulic manifold, speed sensors, trip relays, independent power supplies, and a test graphic. These items and the function of the overspeed trips are described in the following three subsystems.

10.2.2.5.2 Emergency Trip Block/Master Trip Device

The emergency trip supply pressure is established when the master trip solenoid valves are closed. The valves are arranged in two channels for testing purposes. Both valves in a channel will open to trip that channel. Both channels must trip before the emergency trip supply pressure collapses to close the turbine steam inlet valves. Each tripping function of the electrical emergency trip system can be individually tested from the operator/test graphic without tripping the turbine by separately testing each channel of the appropriate trip function. The solenoid valves may be individually tested.

The emergency trip system opens a drain path for the hydraulic fluid in the emergency trip supply. The loss of fluid pressure in the trip header will cause the main stop and reheat stop valves to close. Also, a relay trip valve in the connection to the emergency trip supply opens to drop the pressure in the relay emergency trip supply and cause the control and intercept valves to close. The control and intercept valves are redundant to the main stop and reheat stop valves respectively.

10.2.2.5.3 Overspeed Trip Functions and Mechanisms

The overspeed trips for the AP1000 turbine consist of a diverse 110% trip in the emergency electrical overspeed trip system and a 111% trip in the backup electrical overspeed trip system using three Ovation speed detection modules independent of the OA controller described in Subsection 10.2.4.4.1 (see also Figure 10.2-2). The overspeed trip setpoints are identified in Table 10.2-2. The overspeed protection system will function for all abnormal conditions, including a single failure of any component or subsystem.

The 110% emergency electrical overspeed trip is implemented electronically rather than mechanically as indicated in the review procedure in SRP 10.2, Part III-2-C. An independent and

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redundant backup electrical overspeed trip circuit senses the turbine speed by magnetic pickup and closes all valves associated with speed control at approximately 111% of rated speed.

The diverse 110% emergency electrical overspeed trip system has triplicated passive speed sensors separate from the triplicated active speed sensors used in the 111% backup electrical overspeed trip system. Both trip functions use solenoid valves to drain the emergency trip hydraulic supply. The hydraulic fluid in the trip and overspeed protection control headers is independent of the bearing lubrication system to minimize the potential for contamination of the fluid.

The diverse 110% emergency electrical overspeed trip system, combined with the 111% backup electrical overspeed trip system of the turbine control and protection system, provide a level of redundancy and diversity at least equivalent to the recommendations for turbine overspeed protection found in III.2 of Standard Review Plan (NUREG-0800) Section 10.2. The control signals from the two turbine-generator overspeed trip systems are isolated from, and independent of, each other. Each trip is initiated electrically in separate systems. The 110% and 111% trip systems have diverse hardware and software/firmware to eliminate common cause failures (CCFs) from rendering the trip functions inoperable. Additionally, the issues and problems with overspeed protection systems identified in NUREG-1275 (Reference 3) have been addressed.

The turbine rpm signals discussed in Subsection 10.2.2.4.1 are also used by redundant Ovation controllers for normal speed and load control functions (SRP 10.2 Part III.2.B). The rpm signals are also monitored by a setpoint on each of the speed detection modules to support the backup overspeed trip at 111% overspeed (Part III.2.D). When the rpm speed reaches the backup turbine overspeed trip setpoint, the microprocessor on each speed detection module issues a trip command to the onboard relay. When at least two out of three channels indicate a trip, a trip signal is sent to the turbine.

Independence of the electrohydraulic control system of SRP 10.2 Part III.2.B and the backup electrical overspeed trip circuit of Part III.2.D is assured in that failure of the Ovation controllers does not affect the ability of the speed detection module to trip the turbine at the backup turbine overspeed trip setpoint. Although the electrohydraulic control system and the backup electrical overspeed trip circuit are located in the same cabinet, common failure modes have been addressed to ensure the 111% backup electrical overspeed trip function in SRP 10.2 Part III.2.D cannot be rendered inoperable or affected by the failure of the Ovation controllers. The control system also has diagnostic capabilities to provide information to the operator in the main control room in case of other failures or problems in the system or its components.

10.2.2.5.4 Trip Instrumentation

Low bearing oil pressure, low electrohydraulic fluid pressure, and high condenser back pressure are each sensed by separate instrumentation. Each assembly consists of triplicate pressure transmitters with instrument valves. Each assembly is arranged into three channels.

If two of the three signals (pressure or vacuum) reach a trip setpoint, then the pressure sensors cause the master trip device to operate.

The trip function can be checked by the plant control system that simulates pressure to activate the trip outputs from the modules.

10.2.2.5.5 Thrust Bearing Trip Device

Three position pickups, which are part of the turbine supervisory instrument package, monitor movement of a disc mounted on the rotor near the thrust bearing collar. Axial movement of this collar is reflected in movement of the disc. Excessive movement of the disc is an indication of thrust

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bearing wear. Should excessive movement occur, signals from the supervisory instrument modules initiate a turbine trip by the condition that two out of the three bearing wear position pickups exceed the setpoint.

The thrust bearing trip function can be checked by the plant control system that simulates movement of the rotor to activate the trip outputs from the modules.

10.2.2.5.6 Remote Trip

The emergency trip system also has provisions to trip the turbine in response to a signal from the plant control system or plant safety and monitoring system.

10.2.2.6 Other Protective Systems

Additional protective features of the turbine and steam system are:

- Moisture separator reheater safety relief valves
- Rupture diaphragms located on each of the low-pressure turbine cylinder covers
- Turbine water induction protection systems on the extraction steam lines

10.2.2.7 Plant Loading and Load Following

The AP1000 turbine-generator control system and control strategy has the same loading and load following characteristics as the control system described in Section 7.7. In addition, the turbine-generator has the following capabilities:

- Daily load change between 100 and 50 percent of rated power
- Transition between baseload and load follow operation
- Extended weekend reduced power operation
- Rapid return to up to 90 percent of rated power

For the AP1000, this load following capability is maintained for most of cycle life.

10.2.2.8 Inspection and Testing Requirements

Major system components are readily accessible for inspection and are available for testing during normal plant operation. Turbine trip circuitry is tested prior to unit startup.

10.2.3 Turbine Rotor Integrity

Turbine rotor integrity is provided by the integrated combination of material selection, rotor design, fracture toughness requirements, tests, and inspections. This combination results in a very low probability of a condition that could result in a rotor failure.

Operations and maintenance procedures mitigate the following potential degradation mechanisms in the turbine rotor and buckets/blades: pitting, stress corrosion cracking, corrosion fatigue, low-cycle fatigue, erosion, and erosion-corrosion.

10.2.3.1 Materials Selection

Fully integral turbine rotors are made from ladle refined, vacuum deoxidized, Ni-Cr-Mo-V alloy steel by processes which maximize steel cleanliness and provide high toughness. Residual elements are controlled to the lowest practical concentrations consistent with melting practices. The chemical property limits of ASTM A470, Classes 5, 6, and 7 are the basis for the material requirements for the

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turbine rotors. This material has the lowest fracture appearance transitions temperatures (FATT) and the highest Charpy V-notch energies obtainable on a consistent basis from water-quenched Ni-Cr-Mo-V material at the sizes and strength levels used. Charpy tests and tensile tests in accordance with American Society of Testing and Materials (ASTM) specification A370 are required from the forging supplier.

The production of steel for the turbine rotors starts with the use of high-quality, low residual element scrap. An oxidizing electric furnace is used to melt and dephosphorize the steel. Ladle furnace refining is then used to remove oxygen, sulphur, and hydrogen from the rotor steel. The steel is then further degassed using a process whereby steel is poured into a mold under vacuum to produce an ingot with the desired material properties. This process minimizes the degree of chemical segregation since silicon is not used to deoxidize the steel.

10.2.3.2 Fracture Toughness

Suitable material toughness is obtained through the use of materials described in Subsection 10.2.3.1 to produce a balance of material strength and toughness to provide safety while simultaneously providing high reliability, availability, and efficiency during operation. The limits on phosphorous, sulphur, and aluminum in the rotor steel provides for the appropriate balance of material strength and toughness.

Bore stress calculations include components due to centrifugal loads and thermal gradients where applicable. Fracture toughness will be at least 220 MPa • $\sqrt{m}$ = 200 ksi • $\sqrt{in}$ and the ratio of fracture toughness to the maximum applied stress intensity factor for rotors at speeds from normal to design overspeed will be at least 2. Material fracture toughness needed to maintain this ratio is verified by mechanical property tests on material taken from the rotor.

The rotor is evaluated for fracture toughness by criteria that include the design duty cycle stresses, number of cycles, ultrasonic examination capability and growth rate of potential flaws. Conservative factors of safety are included for the size uncertainty of potential or reported ultrasonic indications, rate of flaw growth (da/dN versus dK) and the duty cycle stresses and number.

Reported rotor forging indications are adjusted for size uncertainty and interaction. A rotor forging with a reported indication that would grow to critical size in the applicable duty cycles is not accepted. The combined rotation and maximum transient thermal stresses used in the applicable duty cycles are based on the brittle fracture and rotor fatigue analyses described below.

Maximum transient thermal stresses are determined from historical maximum loading rates for nuclear service rotors.

10.2.3.2.1 Brittle Fracture Analysis

A brittle fracture analysis is performed on the turbine rotor to provide confidence that small flaws in the rotor, especially near the centerline, do not grow to a critical size with unstable growth resulting in a rotor burst. The brittle fracture analysis process includes determining the stresses in the rotor resulting from rotation, steady-state thermal loads, and transient thermal loads from startup and load change. These stresses are combined to generate the maximum stresses and locations of maximum stress for the startup and load change transients. A fracture mechanics analysis is performed at the location(s) of maximum stress to verify that an initial flaw, equal to the minimum reportable size, will not grow to critical crack size over the life of the rotor under the cumulative effects of startup and load change transients.

A fracture mechanics analysis is done at the location(s) of maximum stress to determine the critical crack size and the initial flaw area that would just grow to the critical size when subjected to the

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number of startup and load change cycles determined to represent the lifetime of the rotor. This initial flaw area is divided by a factor of safety to generate an allowable initial flaw area. The minimum reportable flaw size is multiplied by a conservative factor to correct for the imperfect nature of a flaw as an ultrasonic reflector, as compared to the calibration reflector. The resulting area is the corrected flaw area. For an acceptable design, the allowable initial flaw area must be greater than or equal to the corrected flaw area.

A flaw is assumed to be an internal elliptical crack on the centerline for rotors without bores. For rotor contour or for flaws near the rotor bore (for bored rotors), a surface connected elliptical crack is assumed. Flaw analysis is done assuming various flaw aspect ratios and the most conservative results are used. The flaw is assumed to be orientated normal to the maximum principle stress direction.

The beginning-of-life fracture appearance transition temperature for the high pressure and low pressure rotor is specified in the material specification for the specific material alloy selected. Both the high pressure and low pressure turbines operate at a temperature at which temperature embrittlement is insignificant. The beginning-of-life fracture appearance transition temperature is not expected to shift during the life of the rotor due to temperature embrittlement.

Minimum material toughness is provided in the turbine rotors by specification of maximum fracture appearance transition temperature and minimum upper shelf impact energy for the specific material alloy selected. There is not a separate material toughness (K_{IC}) requirement for AP1000 rotors.

10.2.3.2.2 Rotor Fatigue Analysis

A fatigue analysis is performed for the turbine rotors to show that the cumulative usage is acceptable for expected transient conditions including normal plant startups, load following cycling, and other load changes. The fatigue design curves are based on mean values of fatigue test data. Margin is provided by assuming a conservatively high number of turbine start and stop cycles. The Toshiba-designed turbine rotors in operating nuclear power plants were designed using this methodology and have had no history of fatigue crack initiation due to duty cycles.

In addition to the low cycle fatigue analysis for transient events, an evaluation for high cycle fatigue is performed. This analysis considers loads due to gravity bending, bearing elevation misalignment, control stage partial arc admission bearing reactions, and steady-state unbalance stress. The local alternating stress is calculated at critical rotor locations considering the bending moments due to the loads described above. The maximum alternating stress is less than the smooth bar endurance strength modified by a size factor.

The AP1000 turbine generator is supported by a spring-mounted system to isolate the dynamic behavior of the turbine-generator equipment from the foundation structure. The support system includes a reinforced concrete deck on which the turbine generator is mounted. The deck is sized to maintain the gravity load and misalignment load bending stresses within allowable limits. The evaluation of the loads includes a dynamic analysis of the combined turbine-generator and foundation structure.

10.2.3.3 High Temperature Properties

The operating temperatures of the high-pressure rotors are below the creep rupture range. Creep rupture is, therefore, not considered to be a significant factor in providing rotor integrity over the lifetime of the turbine. Basic data are obtained from laboratory creep rupture tests.

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10.2.3.4 Turbine Rotor Design

The turbine assembly is designed to withstand normal conditions and anticipated transients, including those resulting in turbine trip, without loss of structural integrity. The design of the turbine assembly meets the more restrictive of the following criteria:

- The combined stresses of a low-pressure turbine disk at design overspeed due to centrifugal forces, interference fit, and thermal gradients do not exceed 0.75 of the minimum specified yield strength of the material; or,
- The tangential stresses will not cause a flaw, which is twice the corrected ultrasonic examination reportable size, to grow to critical size in the design life of the rotor. This will result in the ratio of fracture toughness to the maximum applied stress intensity factor for the rotor at speeds from normal to design overspeed being at least 2.

The high-pressure turbine has fully integral rotors forged from a single ingot of low alloy steel. This design is inherently less likely to have a failure resulting in a turbine missile than previous designs with shrunk-on discs. A major advantage of the fully integral rotor is the elimination of disc bores and keyways. In the fully integral rotor design, the location of peak stresses is in the lower stress blade fastening areas. This difference results in a substantial reduction of the rotor peak stresses, which in turn reduces the potential for crack initiation. The reduction in peak stress also permits selection of a material with improved ductility, toughness, and resistance to stress corrosion cracking.

The non-bored design of the high-pressure turbine element provides the necessary design margin by virtue of its inherently lower centerline stress. Metallurgical processes permit fabrication of the rotors without a center borehole. The use of solid rotor forgings was qualified by evaluation of the material removed from center-bored rotors for fossil power plants. This evaluation demonstrated that the material at the center of the rotors satisfied the rotor material specification requirements. Forgings for no-bore rotors are provided by suppliers who have been qualified based on bore material performance.

The low-pressure turbine element is a fully integral rotor fabricated from a single forging. There are no keyways, which can be potential locations for stress risers and corrosive contaminate concentration, exposed to a steam environment. The integral disc profiles are carefully designed to limit the surface stress in areas vulnerable to stress corrosion. Surface stress in less than ideal steam environments is limited to less than 50 percent of the yield strength to reduce the chances of stress corrosion as far as practicable.

10.2.3.5 Preservice Tests and Inspections

Preservice inspections for turbine rotors include the following:

- Rotor forgings are rough machined with a minimum stock allowance prior to heat treatment.

- Each rotor forging is subjected to a 100-percent volumetric (ultrasonic) examination. Each finish-machined rotor is subjected to a surface magnetic particle and visual examination. Results of the above examination are evaluated by use of criteria that are more restrictive than those specified for Class 1 components in ASME Code, Section III and V. These criteria include the requirement that subsurface sonic indications are either removed or evaluated to verify that they do not grow to a size which compromises the integrity of the unit during the service life of the unit.

- Finish-machined surfaces are subjected to a magnetic particle examination. No magnetic particle flaw indications are permissible in bores (if present) or other highly stressed regions.

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- Each fully bladed turbine rotor assembly is spin tested at 20 percent overspeed, the maximum speed following a load rejection from full load.

Rotor areas which require threaded holes are not subjected to a magnetic particle examination of the threaded hole. The number of threaded holes is minimized, and threaded holes are not located in high stress areas.

10.2.3.6 Maintenance and Inspection Program Plan

The maintenance and inspection program plan for the turbine assembly and valves is based on turbine missile probability calculations, operating experience of similar equipment, and inspection results. The methodology for analysis of the probability of generation of missiles for fully integral rotors was submitted in WCAP-16650-P (Reference 1). The methodology used in the analysis of the missile generation probability was used to determine turbine valve test frequency as described in Section 5.2 through Section 8 of WCAP-16651-P (Reference 2). The maintenance and inspection program includes the activities outlined below:

- Disassembly of the turbine is conducted during plant shutdown. Inspection of parts that are normally inaccessible when the turbine is assembled for operation (couplings, coupling bolts, turbine rotors, and low-pressure turbine blades) is conducted.

  This inspection consists of visual, surface, and volumetric examinations as indicated below:

  – Each rotor and stationary and rotating blade path component is inspected visually and by magnetic particle testing on accessible surfaces. Ultrasonic inspection of the outer dovetail and the bucket pin is conducted. These inspections are conducted at intervals of about 10 years for low-pressure turbines and about 8 years for high-pressure turbines.

  – A 100 percent surface examination of couplings and coupling bolts is performed.

  – Fluorescent penetrant examination is conducted on nonmagnetic components.

- At least one main steam stop valve, one main steam control valve, one reheat stop valve, and one intercept valve are dismantled approximately every 3 years during scheduled refueling or maintenance shutdowns. A visual and surface examination of valve internals is conducted. If unacceptable flaws or excessive corrosion are found in a valve, the other valves of its type are inspected. Valve bushings are inspected and cleaned, and bore diameters are checked for proper clearance.

- Main stop valves, control valves, reheat stop and intercept valves may be tested with the turbine online. The turbine control and protection system is used to stroke or partially stroke the valves.

- Extraction nonreturn valves are tested prior to each startup.

- Turbine valve testing is performed at six-month intervals. The semi-annual testing frequency is based on nuclear industry experience that turbine-related tests are the most common cause of plant trips at power. Plant trips at power may lead to challenges of the safety-related systems. Evaluations show that the probability of turbine missile generation with a semi-annual valve test is less than the evaluation criteria.

- Extraction nonreturn valves are tested locally by stroking the valve full open with air, then equalizing air pressure, allowing the spring closure mechanism to close the valve. Closure of each valve is verified by direct observation of the valve arm movement.

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The valve inspection frequency of three years noted above is consistent with an 18-month fuel cycle for AP1000 and is based on evaluations performed to support this valve inspection interval at operating plants with 18-month fuel cycles. A monitoring program is in place at operating nuclear power plants to verify the success of longer valve inspection intervals. A recommendation for a valve inspection frequency longer than three years may be justified when a longer interval is supported by operating and inspection program experience and supported by the missile generation probability calculations.

The non-destructive examinations for the turbine assembly provide assurance that rotor flaws that lead to brittle fracture of a rotor are detected.

10.2.4 Evaluation

Components of the turbine-generator are conventional and typical of those which have been extensively used in other nuclear power plants. Instruments, controls, and protective devices are provided to confirm reliable and safe operation. Redundant, fast actuating controls are installed to prevent damage resulting from overspeed and/or full-load rejection. The control system initiates a turbine trip upon reactor trip. Automatic low-pressure exhaust hood water sprays are provided to prevent excessive hood temperatures. Exhaust casing rupture diaphragms are provided to prevent low-pressure cylinder overpressure in the event of loss of condenser vacuum. The diaphragms are flange mounted and designed to maintain atmospheric pressure within the condenser and turbine exhaust housing while passing full flow.

Since the steam generated in the steam generators is not normally radioactive, no radiation shielding is provided for the turbine-generator and associated components. Radiological considerations do not affect access to system components during normal conditions. In the event of a primary-to-secondary system leak due to a steam generator tube leak, it is possible for the steam to become contaminated. Discussions of the radiological aspects of primary-to-secondary leakage are presented in Chapters 11 and 12.

10.2.5 Instrumentation Applications

The turbine-generator is provided with turbine supervisory instrumentation including monitors for the following:

- Speed
- Stop valve position
- Control valve position
- Reheat intercept and stop valve positions
- Temperatures as required for controlled starting, including:
  - External valve chest inner surface
  - External valve chest outer surface
  - First-stage shell lower inner surface
  - Crossover pipe upstream of reheat stop valve No. 1
  - Crossover pipe upstream of reheat stop valve No. 2
  - Crossover pipe upstream of reheat stop valve No. 3
  - Crossover pipe upstream of reheat stop valve No. 4
  - Crossover pipe upstream of reheat stop valve No. 5
  - Crossover pipe upstream of reheat stop valve No. 6
- Casing and shaft differential expansion
- Vibration of each bearing
- Shaft eccentricity
- Bearing metal temperatures

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Alarms are provided for the following abnormal conditions:

- High vibration
- Turbine supervisory instruments common alarm

In addition to the turbine protective trips listed in Subsection 10.2.2.5, the following trips are provided:

- High exhaust hood temperature
- Low emergency trip system pressure
- Low shaft-driven lube oil pump discharge pressure
- High shell level in a moisture separator/reheater

Indications of the following miscellaneous parameters are provided:

- Main steam throttle pressure
- Steam seal supply header pressure
- Steam seal condenser vacuum
- Bearing oil header pressure
- Bearing oil coolers coolant temperature
- D-EHC control fluid header pressure
- D-EHC control fluid temperature
- Crossover pressure
- Moisture separator drain tank level
- First-stage pressure
- High-pressure turbine exhaust pressure
- Extraction steam pressure, each extraction point (either on the piping or on the corresponding feedwater heater shell)
- Low-pressure turbine exhaust hood spray pressure
- Exhaust hood temperature for each exhaust

Generator supervisory instruments are provided, with sensors and/or transmitters mounted on the associated equipment. These indicate or record the following:

- Multiple generator stator winding temperatures; the detectors are built into the generator, protected from the cooling medium, and distributed around the circumference in positions having the highest expected temperature

- Stator coil cooling water temperature (one detector per coil)

- Hydrogen cooler inlet gas temperature (two detectors at each point)

- Hydrogen gas pressure

- Hydrogen gas purity

- Generator ampere, voltage, and power

Alarms and trips associated with over-voltage and over-frequency protection are part of the generator protection relay system. Additional generator protective devices are listed in Table 10.2-3.

10.2.6 Combined License Information on Turbine Maintenance and Inspection

A turbine maintenance and inspection program will be submitted to the NRC staff for review prior to fuel load. The program will be consistent with the maintenance and inspection program plan activities

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and inspection intervals identified in Subsection 10.2.3.6. Plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis will be available for review after fabrication of the turbine and prior to fuel load.

10.2.7 References

1. WCAP-16650-P, Proprietary and WCAP-16650-NP, Nonproprietary, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines," Revision 0, February 2007.

2. WCAP-16651-P, Proprietary and WCAP-16651-NP, Nonproprietary, "Probabilistic Evaluation of Turbine Valve Test Frequency," Revision 1, May 2009.[1]

3. NUREG-1275, Vol. 11, Operating Experience Feedback Report - Turbine-Generator Overspeed Protection Systems, Commercial Power Reactors, H. L. Ornstein, Nuclear Regulatory Commission, April 1995.

1. The text incorporates by reference portions or all of this document.

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Table 10.2-1
Turbine-Generator and Auxiliaries Design Parameters

Manufacturer Toshiba	
Turbine	
Type	TC6F 52-in. LSB
No. of elements	1 high pressure; 3 low pressure
Last-stage blade length (in.)	52
Operating speed (rpm)	1800
Condensing pressure (in. HgA)	2.9
Generator	
Generator rated output (kW) (minimum)	1,237,500
Power factor	0.95 (leading) 0.90 (lagging)
Generator rating (kVA) (minimum)	1,375,000
Hydrogen pressure (psig)	75
Moisture separator/reheater	
Moisture separator	Chevron vanes
Reheater	U-tube
Number	2 shell
Stages of reheating	2

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Table 10.2-2
Turbine Overspeed Protection

Percent of Rated Speed (Approximate)	Event
100	Turbine is initially at valves wide open. Full load is lost. Speed begins to rise. When the breaker opens, the load drop anticipator immediately closes the control and intercept valves if the load at time of separation is greater than 30 percent.
101	Control and intercept valves begin to close.
108	Peak transient speed with normally operating speed control system. If the power/load unbalance and speed control systems had failed prior to loss of load, then:
110	A trip signal is sent by the emergency electrical overspeed trip system to actuate closure of the main stop, control, intercept, and reheat stop valves by releasing the hydraulic fluid pressure in the valve actuators using a two-out-of-three logic system.
111	The backup electrical overspeed trip system closes the main stop, control, intercept, and reheat stop valves by releasing the hydraulic fluid pressure in the valve actuators using a two-out-of-three trip logic system.

Note:
Following the above sequence of events, the turbine may approach but not exceed 120 percent of rated speed.

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Table 10.2-3 (Sheet 1 of 2)
Generator Protective Devices Furnished with the Voltage Regulator Package

Device	Action		
• Generator Minimum Excitation Limiter	Limiter	–	maintains generator reactive power output above certain level (normally steady-state stability limit level)
	Alarm	–	when limiter is limiting
• Generator Maximum Excitation Limiter	Limiter	–	maintains generator field current at or below a predetermined setpoint to prevent thermal overload
	Alarm	–	when limiter is timing
	Alarm	–	when limiter is limiting
• Generator Overexcitation Protection	Alarm	–	changes the standby controller when overexcitation protection pickup level is exceeded
		–	the system has two master controller configurations; i.e., one is for normal operations and the other is standby
Inverse Timer	Alarm	–	when timing commences
	Regulator trip	–	when timed out
Fixed Timer	Alarm	–	when timing
	Unit trip	–	when timed out
• Generator Volts/Hertz Limiter	Limiter	–	maintains machine terminal volts/Hertz ratio below certain level
	Alarm	–	when limiter is limiting
• Generator Automatic Field Ground Detection	Alarm	–	ground
• Regulator Firing Circuit - Loss of Thyristor Firing Pulse Protection	Alarm	–	loss of one firing circuit
	Unit Trip	–	loss of both firing circuits
• Thyristor Blown Fuse Detection	Alarm	–	When one or more thyristor fuses in power drawers open

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Table 10.2-3 (Sheet 2 of 2)
Generator Protective Devices Furnished with the Voltage Regulator Package

Device	Action		
• Regulator Forcing Indication	Alarm	–	online forcing
	Alarm	–	offline forcing (blocks "Raise" controls of dc regulator and ac regulator adjusters)
• Regulator Loss of Power Supply (s) Protection	Alarm	–	loss of one power supply
	Unit trip	–	loss of both power supplies
• Regulator Loss of Sensing Protection	Alarm and ac regulator trip	–	when regulator voltage transformer changes the standby controller. The system has two master controller configurations; i.e., one is for normal operation and the other is for standby
• Excitation Supply Breaker	Alarm Excitation trip		
• Power System Stabilizer (PSS) Excessive Output Protection	Alarm Power System Stabilizer trip	–	When PSS output exceeds specified level for specified time
• Power System Stabilizer Inservice Instrumentation Indication	Indicator	–	lamps and contacts

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Table 10.2-4
Turbine-Generator Valve Closure Times

Valve	Closing Time (seconds)
Main Stop Valves	0.3
Control Valves	0.3
Intercept Valves	0.3
Reheat Stop Valves	0.3
Extraction Nonreturn Valves	<1.0

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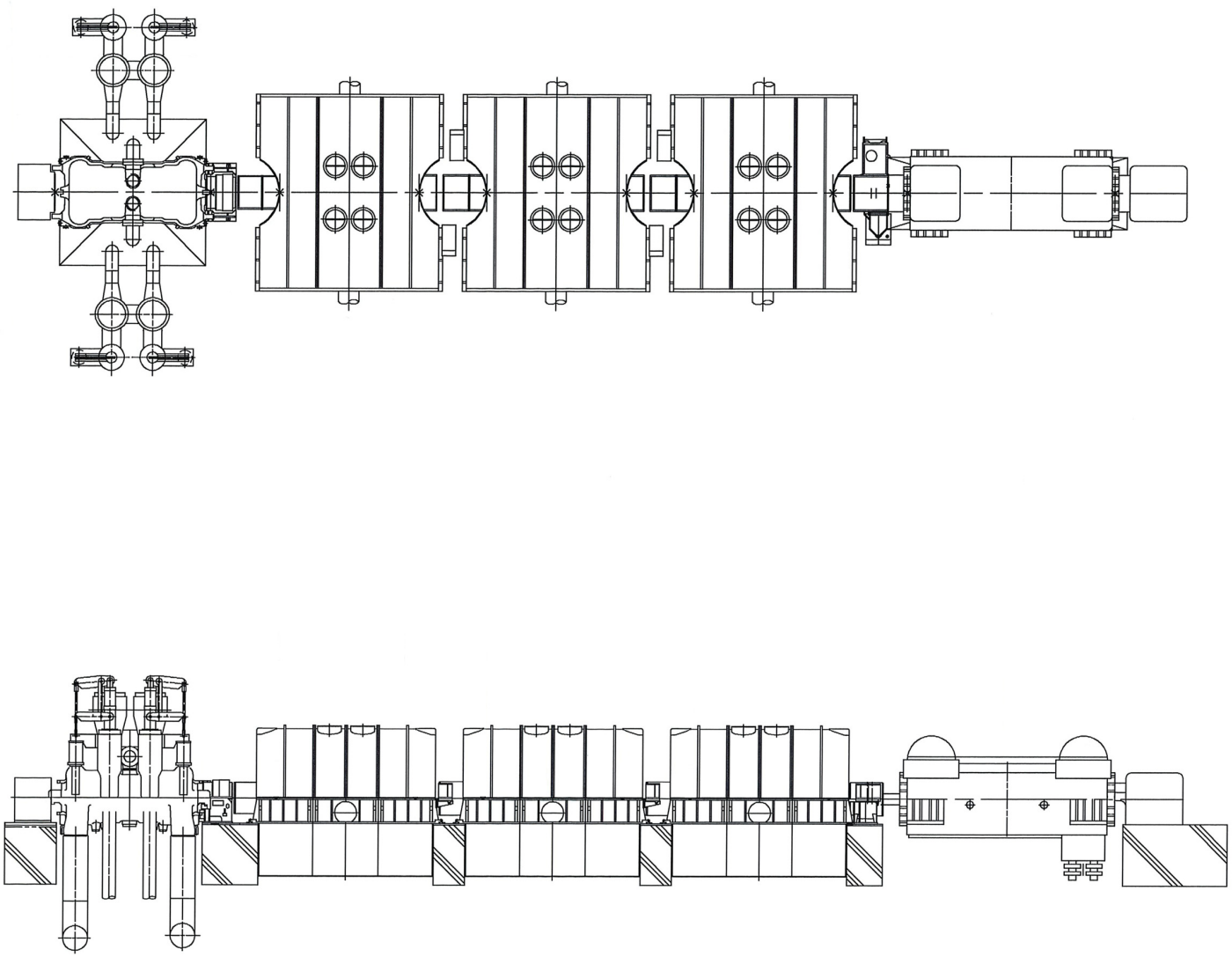

Figure 10.2-1 (Sheet 1 of 2)
Turbine Generator Outline Drawing

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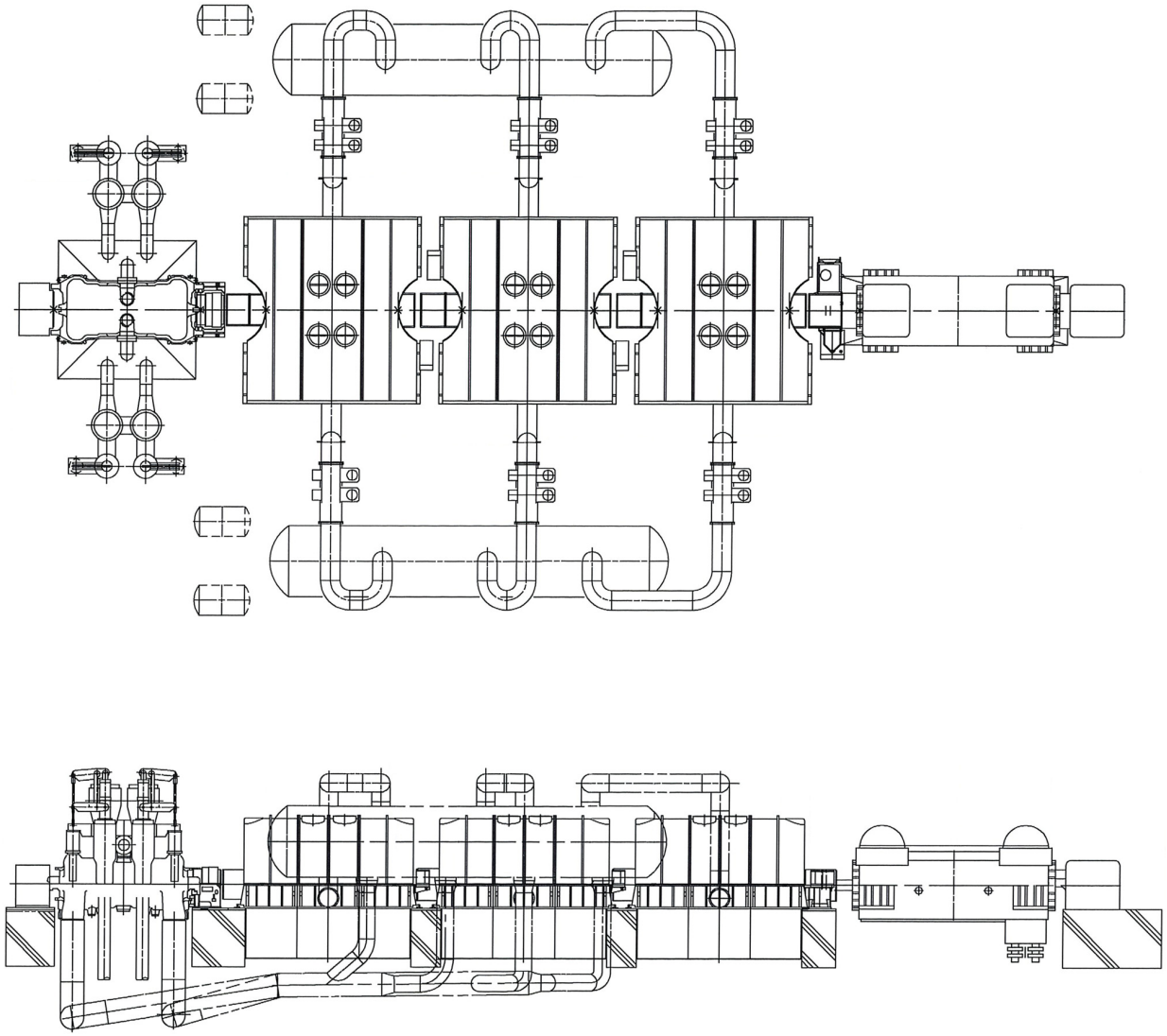

Figure 10.2-1 (Sheet 2 of 2)
Turbine Generator Outline Drawing

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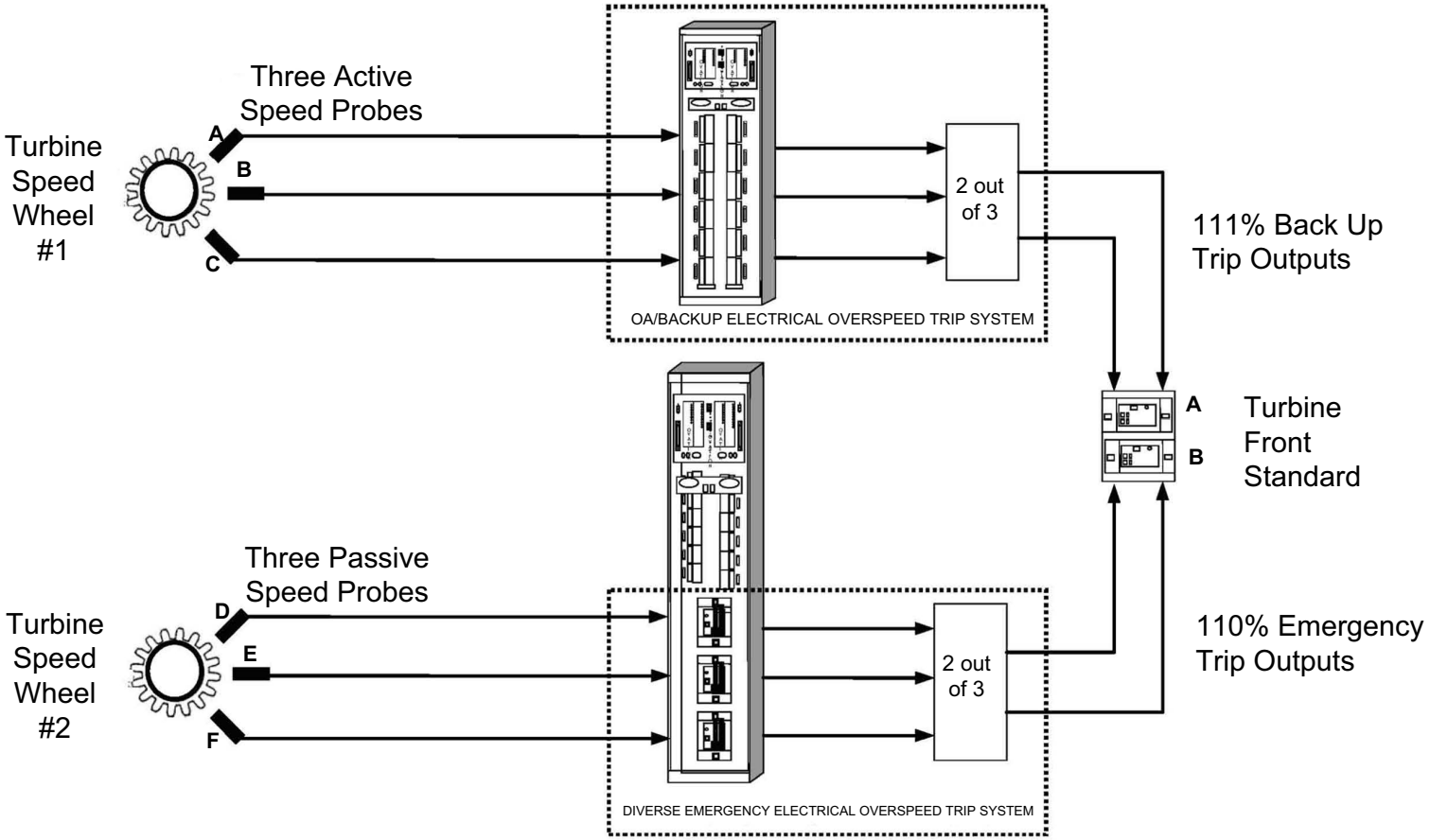

Figure 10.2-2
Turbine Overspeed Trip System Functional Diagram

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10.3 Main Steam Supply System

The main steam supply system as described in this section includes components of the AP1000 steam generator system (SGS), main steam system (MSS), and main turbine system (MTS).

The function of the main steam supply system is to supply steam from the steam generators to the high-pressure turbine over a range of flows and pressures covering the entire operating range from system warmup to maximum calculated turbine conditions.

The system provides steam to the moisture separator/reheaters and the gland seal system for the main turbine. The system dissipates heat generated by the nuclear steam supply system (NSSS) by means of steam dump valves to the condenser or to the atmosphere through power-operated atmospheric relief valves or spring-loaded main steam safety valves when either the turbine-generator or condenser is unavailable.

10.3.1 Design Basis

10.3.1.1 Safety Design Basis

The main steam supply system safety design bases are as follows:

- The system is provided with a main steam isolation valve (MSIV) and associated MSIV bypass valve on each main steam line from its respective steam generator. These valves isolate the secondary side of each of the steam generators to prevent the uncontrolled blowdown of more than one steam generator and isolate nonsafety-related portions of the system.

- Codes and standards utilized in the design of the main steam supply system are identified in Section 3.2, according to the AP1000 equipment class of the component. The main steam supply system contains class B and class C safety-related components.

  Table 3.2-3 identifies the safety-related mechanical equipment in the main steam supply system, and lists the associated ASME code class. (Since all the safety-related components of the main steam supply system are in the AP1000 steam generator system [SGS], they appear in that table with an SGS prefix. For example, the main steam isolation valves [MSIVs] are listed there as SGS-PL-V040A and B).

  The following main steam supply system components are classified as equipment class B and are safety-related:

  – The main steam line piping from the steam generator up to, and including, the main steam isolation valves

  – The main steam isolation valve bypass piping up to, and including, the main steam isolation bypass valve

  – The inlet piping from the main steam line up to, and including, the main steam safety valves

  – The inlet piping from the main steam line up to, and including, the power operated relief valve block valve

  – The instrumentation piping up to, and including, the main steam line pressure instrument root valves

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- The vent line on the main steam line up to, and including, the first isolation valve, and the nitrogen connection on the main steam line up to, and including, the first isolation valve

- The main steam drain condensate pot located upstream of the main steam isolation valves, and the drain piping up to, and including, the first isolation valve

The following main steam supply system components are classified as equipment class C and are safety-related:

- The main steam line piping from the main steam isolation valves outlet to the pipe restraint located on the wall between the auxiliary building and the turbine building

- The main steam safety valve discharge piping and vent stacks

- The piping from the outlet of the power operated relief block valve up to, and including, the power operated relief valve

- The condensate drain piping from the outlet of the class B isolation valve to the restraint on the wall between the auxiliary building and the turbine building

(The remainder of the main steam supply system is nonsafety-related. Except for the power operated relief valve discharge piping from the power operated relief valve outlet to the power operated relief valve silencer, which is class D, the remainder of the main steam supply system is class E).

- The system provides suitable overpressure protection of the steam generator secondary side and class 2 main steam piping in accordance with ASME Code, Section III.

- The safety-related portion of the system is designed to withstand the effects of a safe shutdown earthquake and to perform its intended function following postulated events.

- The safety-related portions of the system are protected from wind and tornado effects, as described in Section 3.3; flood protection is described in Section 3.4; missile protection is described in Section 3.5; protection against dynamic effects associated with the postulated rupture of piping is described in Section 3.6; seismic protection is described in Section 3.7; environmental design is described in Section 3.11; and fire protection is described in Section 9.5.

- The safety-related portion of the system is designed so that a single, active failure in the main steam supply system will not result in:

  - A loss-of-coolant accident

  - Loss of integrity of other steam lines

  - Loss of the capability of the engineered safety features system to effect a safe reactor shutdown

  - Transmission of excessive loading to the containment pressure boundary

  Component or functional redundancy is provided so that safety functions can be performed assuming a single, active failure coincident with loss of offsite power. Consistent with NUREG 0138 and Standard Review Plan Section 10.3, the nonsafety-related valves downstream of the main steam isolation valves are assumed functional to effect this capability.

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- The portion of the main steam supply system that is constructed in accordance with ASME Code, Section III, requirements is provided with access to welds and removable insulation, as required for in-service inspection in accordance with ASME Code, Section XI. (See Subsection 10.3.4.2.)

- The main steam supply system is designed to function in the normal and accident environments identified in Section 3.11.

- The main steam supply system is qualified to leak-before-break criteria as described in Section 3.6.

- The main steam supply system design complies with containment isolation criteria as discussed in Subsection 6.2.3.

- The nonsafety-related turbine stop, turbine control, turbine bypass, and moisture separator reheater 2nd stage steam isolation valves are credited in a single failure analysis to mitigate the event for those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event.

10.3.1.2 Power Generation Design Basis

The following is a list of the principal power generation design bases:

- The main steam supply system delivers steam from the steam generators to the turbine-generator for the range of flowrates, temperatures, and pressures existing from warmup to rated power conditions.

- Each main steam line is sized and routed to provide balanced steam pressures to the turbine stop valves.

- The main steam supply system provides the capacity to dump up to 40 percent of rated main steam flow to the condenser during plant step-load reductions.

- The system provides the means of dissipating residual and sensible heat generated from the nuclear steam supply system during hot shutdown and cooldown even when the main condenser is not available. Power-operated atmospheric relief valves are provided to allow controlled cooldown of the steam generator and the reactor coolant system when the condenser is not available.

- Piping system components located downstream of the auxiliary building wall anchor assemblies are designed in accordance with the Power Piping Code, ASME B31.1.

10.3.2 System Description

10.3.2.1 General Description

The main steam supply system shown in Figures 10.3.2-1 and 10.3.2-2 include the following major components:

- Main steam piping from the steam generator outlet steam nozzles to the main turbine stop valves

- One main steam isolation valve and one main steam isolation valve bypass valve per main steam line

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- Main steam safety valves
- Power-operated atmospheric relief valves and upstream isolation valves

Table 10.3.2-1 lists the design data for the major components of the main steam supply system. Table 10.3.2-2 lists the design data for the main steam safety valve.

10.3.2.2 Component Description

10.3.2.2.1 Main Steam Piping

A description of the main steam piping from the steam generators to the turbine stop valves is presented in Table 10.3.2-3.

The main steam lines deliver a steamflow from the secondary side of the two steam generators. A portion of the main steamflow is directed to the reheater and steam seals, with the turbine receiving the remaining steamflow. Table 10.3.2-1 lists the performance data for the main steam supply system. Each of the main steam lines from the steam generators is anchored at the auxiliary building wall and has sufficient flexibility to accommodate thermal expansion.

Design of seismic Category I piping and supports takes into consideration the loads discussed in Subsection 3.9.3.

The main steam lines between the steam generator outlet nozzle and the containment penetration flued head inboard weld are designed to meet the leak-before-break criteria. The portion of the main steam lines between the containment penetration flued head inboard weld and the anchor downstream of the main steam isolation valves is part of the break exclusion zone. Section 3.6 addresses the applicability of leak-before-break and break exclusion zone to the main steam line.

The layout of the steam piping provides for the collection and drainage of condensate to avoid water entrainment, by the proper sloping of lines and the use of condensate drain pots.

The sizing and layout of the main steam piping hydraulically balances the steam line pressure drops from the respective steam generator to the inlet of each turbine stop valve. Two main steam lines are cross-connected into a common header just before branching into each turbine stop valve. This arrangement equalizes flow and pressure to the inlet of the turbine stop valves. This also permits online testing of each turbine stop valve without exceeding the allowable limit on steam generator pressure differential. Each steam generator outlet nozzle contains an internal flow restrictor arrangement to limit flow in the event of a main steam line break. A further description of the flow restrictor is provided in Subsection 5.4.4.

Sampling connections are installed in the nonsafety-related portion of each main steam line, downstream of the main steam isolation valves. These nozzles are used for the sampling of steam. The sampling is monitored and analyzed through the secondary sampling system (SSS). Refer to Subsection 9.3.4 for further discussion of the secondary sampling system.

Containment penetrations are described in Subsection 6.2.3.

Turbine bypass valves are provided between the main steam isolation valves and turbine-generator stop valves, as discussed under the turbine bypass system (refer to Subsection 10.4.4).

Main steam piping is designed to consider the effects of erosion/corrosion. Piping is constructed of erosion/corrosion resistant low alloy steel. Velocities in the main steam piping to the high pressure turbine are limited to reduce the potential for pipe erosion. Low point drains are provided for

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collecting and draining moisture and to help reduce the potential for water carryover to the high and low pressure turbines. Pipe wall thickness inspections are performed as required to monitor wall erosion rates.

Branch connections are provided from the main steam system to perform various functions. Upstream of the main steam isolation valves, there are connections for the power-operated atmospheric relief valves, main steam safety valves, low point drains, high point vents, and nitrogen blanketing. Branch piping downstream of the main steam line isolation valves includes connections for the two stage reheaters, turbine bypass system, auxiliary steam/gland seal system, and low point drains. Table 10.3.2-4 further describes branch piping, 2.5 inches and larger, that is downstream of the main steam isolation valves.

Operations and maintenance procedures include precautions, when appropriate, to minimize the potential for steam and water hammer, including:

- Prevention of rapid valve motion
- Process for avoiding introduction of voids into water-filled lines and components
- Proper filling and venting of water-filled lines and components
- Process for avoiding introduction of steam or heated water that can flash into water-filled lines and components
- Cautions for introduction of water into steam-filled lines or components
- Proper warmup of steam-filled lines
- Proper drainage of steam-filled lines
- The effects of valve alignments on line conditions

10.3.2.2.2 Main Steam Safety Valves

Main steam safety valves with sufficient rated capacity are provided to prevent the steam pressure from exceeding 110 percent of the main steam system design pressure:

- Following a turbine trip without a reactor trip and with main feedwater flow maintained
- Following a turbine trip with a delayed reactor trip and with the loss of main feedwater flow

A total main steam safety valve rated capacity as indicated in Table 10.3.2-2 meets this requirement. At the same time, the individual safety valves are limited to the maximum allowable steam relief valve capacity as indicated in Table 10.3.2-2 for a system pressure equal to main steam design pressure plus 10 percent overpressure. This value sufficiently limits potential uncontrolled blowdown flow and the ensuing reactor transient should a single safety valve inadvertently fail or stick in the open position.

Six safety valves are provided per main steam line for the plant. Table 10.3.2-2 lists the performance data and set pressures for the main steam safety valves.

The main steam supply system safety valves are located in the safety-related portion of the main steam piping upstream of the main steam isolation valves and outside the containment in the auxiliary building. Adequate provision is made in the steam piping for the installation and support of

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the valves. Consideration is given to the static and dynamic loads when operating or when subjected to seismic events.

The piping and valve arrangement minimizes the loads on the attachment, and analysis confirms the design by use of guidelines in ASME Section III, Nonmandatory Appendix O, "Rules for Design of Safety Valve Installations."

Each safety valve is connected to vent stacks by an open umbrella-type transition piece schematically depicted in detail A of Figure 10.3.2-1.

The vent stacks are designed to:

- Direct the relieved steam away from adjoining structures
- Prevent backflow of relieved steam through the umbrella-type transition section
- Draw a small quantity of ambient air through the umbrella-type transition section and mix with the total steam flow which leaves the vent stack outlet
- Minimize the backpressure on the valve outlet so that it does not restrict the valve's rated capacity

The vent stacks are structurally designed to withstand safe-shutdown earthquake loads. Tornado missiles are not a design basis load on the vent stacks. Missile impact on vent stacks is described in Subsection 3.5.2.

10.3.2.2.3 Power-Operated Atmospheric Relief Valves

A power-operated atmospheric relief valve is installed on the outlet piping from each steam generator to provide for controlled removal of reactor decay heat during normal reactor cooldown when the main steam isolation valves are closed or the turbine bypass system is not available. The valves are sized to provide a flow as indicated in Table 10.3.2-1. The maximum capacity of the relief valve at design pressure is limited to reduce the magnitude of a reactor transient if one valve would inadvertently open and remain open.

Each power-operated relief valve is located outside the containment in the auxiliary building upstream of the main steam isolation valves, in the safety-related portion of the main steam line associated with each steam generator. This location permits valve operation following transient conditions, including those which could result in closure of the main steam isolation valves.

The operation of the power-operated relief valves is automatically controlled by steam line pressure during plant operations. The power-operated relief valves automatically modulate open and exhaust to atmosphere whenever the steam line pressure exceeds a predetermined setpoint. As steam line pressure decreases, the relief valves modulate closed, reseating at a pressure at least 10 psi below the opening pressure. The setpoint is selected between no-load steam pressure and the set pressure of the lowest set safety valves.

The steam generator power-operated atmospheric relief valves provide a nonsafety-related means for plant cooldown by discharging steam to the atmosphere when the turbine bypass system is not available. Under such circumstances, the relief valves (in conjunction with the startup feedwater system) allow the plant to be cooled down at a controlled cooldown rate from the pressure setpoint of the lowest set of safety valves down to the point where the normal residual heat removal (RNS) system can remove the reactor heat.

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For their use during plant cooldown, the power-operated atmospheric relief valves are automatically controlled by steam line pressure, with remote manual adjustment of the pressure setpoint from the control room or the remote shutdown workstation. To effect a plant cooldown, the operator manually adjusts the pressure setpoint downward in a step-wise fashion. The maximum cooldown rate achievable is limited by the flow-passing capability of the relief valves, the number of steam generators (and hence the number of relief valves) in service, the available startup feedwater pumping capacity and by the desire to either maintain or recover steam generator water levels during the cooldown.

The power-operated atmospheric relief valves also help to avoid actuation of the safety valves during certain transients and, following safety valve actuation, act to assist the safety valves to positively reseat by automatically reducing and regulating steam pressure to a value below the safety valve reseating pressure. The operation of each power-operated atmospheric relief valve is controlled in response to measurements of steam line pressure provided by four separate pressure taps on the associated steam line.

The valve operator is an air-operated modulating type, providing throttling capability over a range of steam pressures.

The atmospheric relief valves are controlled by nonsafety-related control systems for the modulating steam relief function. The capability for remote manual valve operation is provided in the main control room and at the remote shutdown workstation. A safety-related solenoid is provided to vent the air from the valve operator to terminate a steam line depressurization transient.

An isolation valve with remote controls is provided upstream of each power operated relief valve providing isolation of a leaking or stuck-open valve. The upstream location allows for maintenance on the power-operated relief valve operator at power. The motor-operated isolation valve employs a safety-related operator and closes automatically on Low-2 steam line pressure to terminate steam line depressurization transients. The isolation valve is a containment isolation boundary and therefore is specified as safety-related, active, ASME Code, Section III, Safety Class 2.

10.3.2.2.4 Main Steam Isolation Valves

The function of the main steam isolation is to limit blowdown to one steam generator in the event of a steam line break to:

- Limit the effect upon the reactor core to within specified fuel design limits
- Limit containment pressure to a value less than design pressure

Main steam isolation consists of one quick-acting gate valve in each main steam line and one associated globe main steam isolation bypass valve with associated actuators and instrumentation. These valves are located outside the containment, downstream of the steam generator safety valves and the atmospheric relief valve, in the auxiliary building. The isolation valves provide positive shutoff with minimum leakage during postulated line severance conditions either upstream or downstream of the valves.

The main steam isolation valves close fully upon receipt of a manual or automatic signal and remain fully closed. Upon receipt of the closing signal, the main steam isolation valves complete the closing cycle despite loss of normally required utility services for actuator and/or instrumentation. On loss of actuating hydraulic power, the valves fail to the closed position. On loss of electrical power the valves remain in their current position. Position indication and remote manual operation of the isolation valves are provided in the control room and remote shutdown workstation. Additionally, provisions are made for in-service inspection of the isolation valves.

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Closure of the main steam isolation valves and main steam isolation bypass valves is initiated by the following:

- Low-2 steam line pressure in one of two loops
- High-2 containment pressure
- High negative steam pressure rate in one of two loops
- Low-2 T_{cold} in either reactor coolant loop
- Manual actuation: There are two controls for main steam line actuation. These controls provide system level actuation: that is, isolate both steam lines.
- Manual reset: In addition to the controls for manual isolation actuation, there are two controls for manual reset of the steam line isolation signal, one for each of the logic divisions associated with steam line controls, which can be used to manually reset that division's steam line isolation signal.

Each main steam isolation valve is a bidirectional wedge type gate valve composed of a valve body that is welded into the system pipeline. The main steam isolation gate valve is provided with a hydraulic/pneumatic actuator. The valve actuator is supported by the yoke, which is attached to the top of the body. The valve actuator consists of a hydraulic cylinder with a stored energy system to provide emergency closure of the isolation valve. The energy to operate the valve is stored in the form of compressed nitrogen contained in one end of the actuator cylinder. The main steam isolation valve is maintained in a normally open position by high-pressure hydraulic fluid. For emergency closure, redundant solenoids are energized resulting in the high-pressure hydraulic fluid being dumped to a fluid reservoir.

The main steam isolation bypass valves are used to permit warming of the main steam lines prior to startup when the main steam isolation valves are closed. The bypass valves are modulating, air-operated globe valves. For emergency closure, redundant 1E solenoids are provided. Each solenoid is energized from a separate safety-related division.

10.3.2.3 System Operation

10.3.2.3.1 Normal Operation

During normal power operation, the main steam supply system supplies steam to meet the demand of the main turbine system. The main steam supply system also supplies steam as required to the auxiliary steam system, and reheating steam to the moisture separator reheater. The main steam supply system also provides steam to the turbine gland seal system.

The main steam supply system is capable of accepting a ±10-percent step change in load or a ±5-percent/min ramp change without discharging steam to the atmosphere through the main steam safety valves or to the main condenser through the turbine bypass system. For large step change load reductions, steam is bypassed (up to 40 percent of rated main steam flow) directly to the condenser via the turbine bypass system. As discussed in Subsection 10.4.4, the main steam supply system, in conjunction with the turbine bypass system, is capable of accepting a 100-percent net load rejection without reactor trip (in conjunction with a reactor rapid power reduction) and without lifting safety valves. If the turbine bypass system is not available, steam is vented to the atmosphere via the power-operated atmospheric relief valves and the main steam safety valves, as required.

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10.3.2.3.2 Emergency Operation

In the event that the plant must be shut down, the main steam isolation valves with associated main steam isolation bypass valves and other valves associated with the main steam lines can be closed. The power-operated atmospheric relief valves are then used to remove reactor decay and primary system sensible heat to cooldown to conditions at which the normal residual heat removal system can perform the remaining cooldown function. If the power-operated atmospheric relief valve for an individual main steam line is unavailable because of the loss of its control or power supply, the respective safety valves will provide overpressure protection. The remaining power-operated atmospheric relief valve is sufficient to cooldown the plant.

In the event that a design basis accident occurs, which results in a large steam line break, the main steam isolation valves with associated main steam isolation bypass valves automatically close. The closure of the main steam isolation valves and associated main steam isolation bypass valves result in no more than one steam generator supplying a postulated break.

The passive residual heat removal system (Section 6.3) provides safety-related decay heat removal capability should steam relief and feedwater be unavailable.

10.3.3 Safety Evaluation

- Each main steam line is provided with safety valves that limit the pressure in the line to limit over-pressurization and remove stored energy. Each line is provided with a power-operated atmospheric relief valve to permit reduction of the main steam line pressure and remove stored energy to achieve an orderly shutdown. The startup feedwater system, described in Subsection 10.4.9, provides makeup to the steam generators consistent with the steaming rate.

- Redundant power supplies and power divisions operate the main steam isolation valves and main steam isolation bypass valves to isolate safety and nonsafety-related portions of the system. Branch lines upstream of the main steam isolation valves contain normally closed, power-operated atmospheric relief valves which modulate open and closed on steam line pressure. In the event the atmospheric relief valves fail closed, the safety valves provide overpressure protection.

  Releases of radioactivity from the main steam system are minimized because there are no significant amounts of radioactivity in the system under normal operating conditions. Additionally, the main steam isolation system provides controls for reducing releases, as described in Chapter 15, following a steam generator tube rupture.

  Detection of radioactive leakage into the system, which is characteristic of a steam generator tube leak or rupture, is facilitated by adjacent-to-line radiation monitors on each steam line, the radiation monitor in the turbine vent and drain system which monitors condenser air removal, and the steam generator blowdown line radiation monitor.

- Section 3.2 provides the quality group classification, the required design and fabrication codes, and seismic category applicable to the safety-related portion of this system and supporting systems. The power supplies and controls necessary for safety-related functions of the main steam supply system are safety-related, as described in Chapters 7 and 8.

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- The safety-related portion of the main steam supply system is located in the containment and auxiliary building. These buildings are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other appropriate natural phenomena. Sections 3.3, 3.4, 3.5, 3.7, and 3.8 describe the bases of the structural design of these buildings.

  The safety-related portion of the main steam supply system is designed to remain functional after a safe shutdown earthquake. Sections 3.7 and 3.9 provide the design loading conditions that were considered. Sections 3.5, 3.6, and 9.5 describe the analyses to provide confidence that a safe shutdown, as outlined in Section 7.4, is achieved and maintained.

- As indicated by the failure mode and effects analysis in Table 10.3.3-1, no single failure coincident with loss of offsite power compromises the system safety functions.

- The main steam supply system is initially tested with the program given in Chapter 14. Periodic in-service functional testing is done in accordance with Subsection 10.3.4.

  Section 6.6 provides the ASME Code, Section XI requirements that are appropriate for the safety-related portions of the main steam supply system.

- The safety-related components of the main steam supply system are qualified to function in normal, test, and accident environmental conditions. The environmental qualification program is described in Section 3.11.

- A discussion of high energy pipe break locations and evaluation of effects are provided in Subsections 3.6.1 and 3.6.2.

- A discussion of the leak-before-break application and criteria is presented in Subsection 3.6.3.

10.3.4 Inspection and Testing Requirements

10.3.4.1 Preoperational Testing

10.3.4.1.1 Valve Testing and Inspection

The operability and relief setpoints of the main steam safety valves will be verified at operating temperature using steam as the pressurization fluid. The advantage of this approach is that the testing at temperature will reduce the probability of having to adjust the valve setpoints during hot functional testing heatup. The valves may be either bench tested or in-situ tested. The valves will be adjusted to lift at their set pressure defined in Table 10.3.2-2.

The sum of the rated capacities of the valves shall exceed the capacity specified in Table 10.3.2-2. The relieving capacity of the valve is certified in accordance with the ASME Code, Section III NC-7000.

The lift-point of each power-operated atmospheric relief valve is checked against pressure gauges mounted in the main steam piping.

The power operated relief valves will be verified to have a relief capacity of at least 300,000 lbs/hour at 1106 psia in order to satisfy their non-safety related function of decay heat removal.

The main steam isolation valves are tested to verify the closing time prior to initial startup.

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10.3.4.1.2 System Testing

The main steam supply system is designed to allow testing of system operation for both normal and emergency operating modes. This includes operation of applicable portions of the protection system.

The safety-related components of the system are designed and located to permit pre-service and in-service inspections.

10.3.4.1.3 Pipe Testing

The main steam lines within the containment and the auxiliary building are visually and volumetrically inspected at installation as required by ASME Code, Section XI pre-service inspection requirements.

10.3.4.2 In-service Testing

The performance and structural leaktight integrity of system components are demonstrated by operation.

Additional description of in-service inspection and in-service testing of ASME Code, Section III, Class 2 and 3 components is contained in Section 6.6 and Subsection 3.9.6. The nonsafety-related turbine stop, turbine control, and moisture separator reheater 2nd stage steam isolation valves are included in the inservice test program discussed in Subsection 3.9.6.

10.3.5 Water Chemistry

The objectives of the secondary side water chemistry program are as follows:

- Minimizing general corrosion in the steam generators, turbine, and feedwater system by maintaining proper pH control and by minimizing oxygen ingress (coupled with oxygen scavenging)
- Minimizing localized corrosion in the steam generators, turbine, and feedwater system by minimizing chemical contaminant ingress and by controlling contaminant levels through condensate polishing and steam generator blowdown.

10.3.5.1 Chemistry Control Basis

Steam Generator Owner's Group recommendations are considered in the secondary side water chemistry program.

Secondary side water chemistry control basis for AP1000 is shown below:

System Design

- Selection of secondary side materials to minimize corrosive species such as copper oxides
- Capability of deaeration in the demineralized water supply path, condenser, and deaerator
- Capability of continuous blowdown of the steam generator bulk water
- Capability of post-construction cleaning of the feedwater system followed by wet layup of the feedwater system and steam generators

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Operation Phase

- Early identification of contaminant ingress (salts, corrosion products, and oxygen)
- Capability to filter and demineralize condensate by passage of part of the condensate flow through a condensate polisher system prior to and during plant startup and shutdown and during power operation with abnormal secondary cycle chemistry.
- Chemical addition to establish and maintain an environment that minimizes system corrosion
- Identification of action levels based on chemistry conditions, as determined by high sensitivity continuous or semi-continuous monitoring or by grab sampling

10.3.5.2 Contaminant Ingress

Contaminants may be introduced into the secondary side water system through three major mechanisms: makeup water; condenser tube leaks; atmospheric leaks at the condenser or pump seals. The following methods are used to detect the ingress of contaminants in the secondary water system:

- Demineralized water is continuously monitored as it is being produced in the water treatment plant.
- Ionic contaminants are detected by monitoring (either continuous process monitors or sample analysis) the condensate pump discharge, feedwater downstream of addition of heater and moisture separator drains, and steam generator bulk flow as blowdown.
- Atmospheric ingress is detected by monitoring the condensate pump discharge for excessive dissolved oxygen and by monitoring condenser air removal rate.

10.3.5.3 Condensate Polishing

A condensate polishing system with a capacity of one third design condensate flow is provided to remove corrosion products and ionic contaminants. This polishing system will not normally be employed during all phases of plant operation.

The secondary side water system has provisions for recirculating feedwater to the condenser prior to and during startup. The polisher may be used during this phase to remove corrosion products from the feedwater and thus prevent their ingress into the steam generators. Full flow or near full flow condensate polishing is possible at the lower condensate flows that exist during startup and low-power operation. See Subsection 10.4.6 for additional information.

10.3.5.4 Chemical Addition

AP1000 employs an all-volatile treatment (AVT) method to minimize general corrosion in the feedwater system, steam generators, and main steam piping. A pH adjustment chemical and an oxygen scavenger are the two chemicals to be injected into the condensate pump discharge header, downstream of the condensate polishers.

To reduce the general corrosion rate of ferrous alloys, a volatile pH adjustment chemical is injected to maintain a noncorrosive environment. Although the pH adjustment chemical is volatile and will not concentrate in the steam generator, it will reach an equilibrium level which will help establish noncorrosive conditions.

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An oxygen scavenger is added to maintain the dissolved oxygen content in the feedwater within specified limits for each mode of operation. The oxygen scavenger also promotes the formation of a protective magnetite layer on ferrous surfaces and keeps this layer in a reduced state, further inhibiting general corrosion.

Alkaline chemistry supports maintaining iodine compounds in their nonvolatile form. When iodine is in its elemental form, it is volatile and free to react with organic compounds to create organic iodine compounds, which are not assumed to remain in solution. It is noted that no significant level of organic compounds is expected in the secondary system. The secondary water chemistry, thus, does not directly impact the radioactive iodine partition coefficients.

10.3.5.5 Action Levels for Abnormal Conditions

Appropriate responses to abnormal chemistry conditions provide for the long-term integrity of secondary cycle components. Action taken when chemistry parameters are outside normal operating ranges will, in general, be consistent with action levels described in Reference 1 in accordance with the secondary side water chemistry program.

10.3.5.6 Layup and Heatup

AP1000 anticipates no long-term steam generator layup under dry conditions. When maintenance or inspection is required on the secondary side of the steam generators, the steam generators are drained hot under nitrogen atmosphere. After cooling, the nitrogen purge is lifted and the maintenance/inspection begun.

Wet layup conditions are established for corrosion protection during outages. Guidelines are given in the secondary side water chemistry program, which is based on Reference 1.

Before heatup to full power, the bulk water in the steam generators is normally brought into power operation specifications by draining and refilling or by feeding and bleeding. Guidelines for heatup are provided in the secondary side water chemistry program, which is based on Reference 1.

10.3.5.7 Chemical Analysis Basis

Guidelines for chemical control and diagnostic parameters are provided in the secondary side water chemistry program, which is based on Reference 1. Each parameter will be addressed as indicated below.

Oxygen in the presence of moisture rapidly corrodes carbon steel. These corrosion products may be carried through the feedwater system and form sludge in the steam generator. This sludge forms an environment for localized corrosion mechanisms on steam generator tubes. Thus, concentrations of oxygen are kept as low as practical in the feedwater system, and dissolved oxygen is controlled at the condenser and deaerator to prevent oxygen transport to the feedwater system.

Residual concentration of the oxygen scavenger is also measured in the feedwater sample and is used as input for injection of the oxygen scavenger.

In the absence of significant impurities, the pH is controlled by the concentration of the volatile pH adjustment chemical and the oxygen scavenger. Maintaining the pH within the recommended band results in minimal corrosion rates of ferrous materials.

Conductivity is also a measure of the presence of ionic contamination and provision is made for monitoring conductivity in samples of condensate, feedwater, and steam generator blowdown.

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Provision is also made for specific ions, such as sodium and chloride, which could be indicative of aggressive chemistry conditions.

10.3.5.8 Sampling

The sampling points are identified in Table 9.3.4-1 (continuous and semi-continuous sample points) and Table 9.3.4-2 (selective measurements). Additionally, sample points are addressed in the secondary side water chemistry program, which is based on Reference 1.

10.3.5.9 Condenser Inspection

The secondary side water chemistry program includes an inspection program of the condenser to verify condenser integrity. This program includes a visual inspection of the condenser during outages and component inspection for air leaks during plant operation.

10.3.5.10 Conformance to Branch Technical Position 5-1

Conformance to Branch Technical Position 5-1 is discussed in Table 10.3.5-1.

10.3.6 Steam and Feedwater System Materials

10.3.6.1 Fracture Toughness

The material specifications for pressure-retaining materials in safety-related portions of the main steam and feedwater systems meet the fracture toughness requirements of ASME Code, Section III, Articles NC-2300 and ND-2300 for Quality Group B and Quality Group C components.

10.3.6.2 Material Selection and Fabrication

Pipe, flanges, fittings, valves, and other piping material conform to the referenced ASME, ASTM, ANSI, or Manufacturer Standardization Society-Standard Practice code.

No copper or copper-bearing materials are used in the steam and feedwater systems.

The following requirements apply to the nonsafety-related portion of the main steam system.

Component	**Alloy/Carbon Steel**
Pipe	ANSI/ASME B36.10M
Fittings	ANSI/ASME B16.9, B16.11
Flanges	ANSI/ASME B16.5

Material selection and fabrication requirements for ASME Code, Section III, Class 2 and 3 components in the safety-related portions of the main steam and feedwater systems are consistent with the requirements for ASME Class 2 and 3 systems and components outlined in Subsections 6.1.1.1 and 6.1.1.2. Material specifications for the main steam and feedwater systems are listed in Table 10.3.2-3.

Conformance with the applicable regulatory guides is described in Subsection 1.9.1.

Nondestructive inspection of ASME Code, Section III, Class 2 and 3 components in the safety-related portions of the main steam and feedwater systems is addressed in Subsection 6.6.5.

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Appropriate operations and maintenance procedures provide the necessary controls during operation to minimize the susceptibility of components made of stainless steel and nickel-based materials to intergranular stress-corrosion cracking by controlling chemicals that are used on system components.

10.3.7 Combined License Information

This section contained no requirement for information.

10.3.8 References

1. "PWR Secondary Water Chemistry Guidelines," EPRI.

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Table 10.3.2-1
Main Steam Supply System Design Data

Steam Flow	Maximum Calculated
Per steam generator (lb/hr)	7.50×10^6
Total (lb/hr)	14.99×10^6
Design Conditions	
Design pressure (psia)	1200
Design temperature (°F)	600°F
Operating Conditions	
Full plant load pressure (psia)	836
Full plant load temperature (°F)	523.3
No load (hot standby) pressure (psia)	1106
No load (hot standby) temperature (°F)	557
Main Steam Piping: See Table 10.3.2-3.	
Steam Generator Flow Restrictor	
Number per steam generator outlet nozzle	7
Throat size (ft^2)	0.2
Total area (ft^2)	1.4
Power-Operated Relief Valve	
Number per main steam line	1
Normal set pressure	1138 psig
Design capacity	
Minimum:	70,000 lb/hr at 100 psia steam generator pressure
Maximum:	1,020,000 lb/hr at 1200 psia steam generator pressure
Code	ASME Code, Section III, Class 3, seismic Category I
Actuator	Air-operated modulating

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Table 10.3.2-2
Design Data for Main Steam Safety Valves

Number per main steam line		6
Total number of valves required per steam line for full power operation		6
Relieving capacity per valve at 110% of design pressure		1,370,000 lb/hr
Relieving capacity per steam line at 110% of design pressure		8,240,000 lb/hr
Total relieving capacity, 2 lines at 110% of design pressure		16,480,000 lb/hr
Valve size		8 x 10 (Dual Discharge)
Design code		ASME Code, Section III, Class 2, seismic Category I
Valve Number	**Set Pressure (psig)**	**Relieving Capacity[(a)] (lb/hr)**
SGS PL V030A(B)	1185	≥ 1,320,000
SGS PL V031A(B)	1197	≥ 1,340,000
SGS PL V032A(B)	1209	≥ 1,350,000
SGS PL V033A(B)	1221	≥ 1,360,000
SGS PL V034A(B)	1232	≥ 1,370,000
SGS PL V035A(B)	1232	≥ 1,370,000
Total capacity, at 103% valve setpoint pressures, 2 lines		≥ 16,220,000

Note:

a. Based on system accumulation pressure of 3%, per Subsection NC-7512 of ASME Code, Section III, Division 1, 1998 Edition with 2000 Addenda, Subsection NC, Class 2 components.

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Table 10.3.2-3
Description of Main Steam and Main Feedwater Piping

Segment	Material Specification
Main Steam Line	
Steam generator outlet to containment penetration	SA-335 Gr. P11 seamless pipe
Containment penetration to MSIV	SA-335 Gr. P11 seamless pipe
MSIV to auxiliary/turbine building wall	SA-335 Gr. P11 seamless pipe
Auxiliary/turbine building wall to equalization header[(a)]	ASTM A-335 Gr. P11 seamless pipe
Branch lines to turbine stop valves[(a)]	ASTM A-335 Gr. P11 seamless pipe
Main Feedwater Line	
Feedwater pump outlet to individual steam generator feedwater lines[(a)]	ASTM A-335 Gr. P-11 and ASTM A-106 Gr. B
Feedwater heater bypass line[(a)]	ASTM A-335 Gr. P-11 and ASTM A-106 Gr. B
Start of individual steam generator feedwater lines to auxiliary/ turbine building wall[(a)]	ASTM A-335 Gr. P-11
Auxiliary/turbine building wall to MFIV	SA-335 Gr. P-11
MFIV to containment penetration	SA-335 Gr. P-11
Containment penetration to steam generator nozzle	SA-335 Gr. P-11

Note:
a. Piping is beyond the ASME Section III piping boundary.

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Table 10.3.2-4
Main Steam Branch Piping
(2.5-Inch and Larger)
Downstream of MSIV

Description	Maximum Steam Flow	Shutoff Valve	Valve Closure Time	Actuator	Comments
Turbine bypass lines to condenser; 6 lines total	999,333 lb/hr each line[c, d]	16-in. globe (turbine bypass valve)	5 sec or less[a] when tripped closed	Air operator; fail close	Bypass valve is tripped closed on main steam isolation signal
Reheating steam to moisture separator reheater (MSR), 2 lines total	242,000 lb/hr, each MSR[c]	10-in. globe (MSR reheat 2nd stage steam isolation valve, 2 each)	5 sec or less[a]	Air operators, fail close	Main steam flow to reheater ceases (thermodynamically) following turbine trip flow ceases following valve closure on a main steam isolation signal
Main steam supply to auxiliary steam system	<450,000 lb/hr[b]	16-in. globe (isolation valve)	10 sec or less	Air operator; fail close	Main steam flow to auxiliary steam system terminates following isolation valve closure on a main steam isolation signal
High pressure turbine steam supply lines; 4 lines total	3,744,000 lb/hr each line[c]	28-in. stop valve in each line	5 sec or less[a]	Hydraulically operated from electro-hydraulic turbine control system	Main steam flow to high pressure turbine ceases following stop valve closure on a turbine trip
Main steam supply to turbine glands	<50,000 lb/hr[b]	6-in. gate (isolation valve)	60 sec or less	Motor operator; manually operated	Main steam flow to turbine seals continues following a turbine trip; however, this steam flow is relatively small and has been considered in the steam line break analysis (Section 3.6)

Notes:

a. Specified closure times are for safety analysis purposes; other system performance requirements may dictate more rapid closure.
b. Analysis maximum — design maximum is less than value shown.
c. Design maximum.
d. Turbine Bypass System maximum flow rate is 40 percent of rated main steam flow as discussed in Subsection 10.4.4. Rated main steam flow is listed in Table 10.3.2-1.

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Table 10.3.3-1 (Sheet 1 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
1	MSIVs V040A(B) normally open, fail closed with self contained hydraulic operator	Isolates SG A(B) in the event of a MSLB to prevent blowdown of more than one SG; Isolates containment in conjunction with SG and main steam line inside containment	a. All but DBA	b. Fails closed or fails to open on command	a. Position indication on main control room & remote shutdown work-station	a. None; Plant goes to or remains in a safe shutdown condition	One MSIV is provided for each steam line. Each MSIV redundantly activated from separate safety-related power divisions. Redundant backup provided by downstream isolation valves. Redundant containment isolation provided by SG and main steam line inside containment.
			b. DBA Except SGTR	b. Fails to close upon ESF isolation signal	b. Position indica-tion on main control room & remote shutdown workstation	b. None; closure of either MSIV or downstream valves prevent blowdown of more than one SG; containment integrity is maintained by MSIV and either SG and steam line integrity inside containment or downstream valves.	
			c. DBA-SGTR	c. Fails to close on ESF isolation signal	c. Same as 1b	c. None; limiting failure is PORV failed open discharging to atmo-sphere. Termination of break flow occurs on automatic block valve closure plus PRHR actuation. Continued break flow past MSIV precluded by redundant downstream isolation valves.	Redundant isolation provided by downstream isolation valves

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Table 10.3.3-1 (Sheet 2 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
2	Main steam power operated relief valve V233A(B) normally closed fail closed, air-operated control valve	Isolates SG A(B) in the event of a MSLB to prevent blowdown from more than one SG in conjunction with block valve	a. All but DBA	a1. Fails to open upon open signal	a1. Position indication on main control room & remote shutdown workstation; steam line high pressure	a1. None; heat removal available via steam dump or PRHR; safety valves provide over-pressure protection	
				a2. Fails open or fails to close on command including spurious operation	a2. Position indication on main control room & remote shutdown workstation; low SG level or SG pressure	a2. None; maximum flow less than DBA limit; shutdown effected with 1 PORV, PRHR, or steam dump	Redundant isolation provided by PORV and block valve via separate safety-related power divisions
			b. DBA except SGTR	b. Fails to close	b. Position indication on main control room & remote shutdown workstation; steam line low pressure alarm	b. None, redundant isolation provided by PORV block valve	Dose analysis based on failed open PORV with subsequent block valve closure
			c. DBA - SGTR	c. Fail to close	c. Position indication on main control room & remote shutdown workstation; streamline low pressure alarm	c. None; automatic redundant isolation provided by PORV block valve. Releases based on signal generation and closure time delay	

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Table 10.3.3-1 (Sheet 3 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
3	Main steam line PORV block valve V027A(B), normally open fail as is motor-operated globe valve	Isolates SG A(B) in the event of a MSLB to prevent blowdown from more than 1 SG in conjunction with PORV and provides containment integrity in conjunction with SG and main steam line inside containment	a. All but DBA	a1. Same as 2.a1 a2. Same as 2.a2	a1. Same as 2.a1 a2. Same as 2.a2	a1. Same as 2.a1 a2. Same as 2.a2	Redundant steam line isolation provided by PORV and block valve via separate safety-related power divisions. Redundant containment isolation provided by SG and main steam line inside containment.
			b. DBA except SGTR	b. Same as 2b	b. Position indication on main control room & remote shutdown workstation	b. Same as 2b; containment integrity is maintained by SG and main steam line inside containment	
			c. DBA SGTR	c. Same as 2b	c. Position indication on main control room & remote shutdown workstation	c. None, automatic redundant isolation of the PORV on Low-2 steam line pressure	Dose analysis based on failed open PORV isolated by block valve. Releases equivalent.

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Table 10.3.3-1 (Sheet 4 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
4	Main steam isolation bypass valve V240A(B) normally closed, fail closed air-operated valve	Isolates SG A(B) in the event of a MSLB to prevent blowdown of more than one SG; isolates containment in conjunction with SG and main steam line inside containment	a. All but DBA	a. Fails closed or fails to open on command	a. Position indication on main control room and remote shutdown workstation	a. Plant continues operation or goes to or remains at a safe shutdown condition	One MSIV bypass is provided for each steam line. Each bypass valve redundantly activated from separate 1E power divisions. Redundant backup provided by downstream isolation valves. Redundant containment isolation provided by SG and main steam line inside containment.

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Table 10.3.3-1 (Sheet 5 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
			b. DBA except SGTR	b. Fails to close upon ESF isolation signal	b. Position indication on main control room & remote shutdown workstation	b. None, closure of either bypass valve or down-stream isolation valves prevents blowdown of more than 1 SG; containment integrity maintained by either MSIV bypass valve or SG/steam line integrity inside containment	
			c. DBA-SGTR	c. Fails to close on ESF isolation signal	c. Position indication on main control room & remote shutdown workstation	c. None, limiting failure is PORV failed open discharging to atmosphere. Termination of break flow occurs on automatic block valve closure plus passive RHR actuation. Continued break flow past MSIV bypass precluded by redundant downstream isolation valves	Redundant isolation provided by downstream isolation valves

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Table 10.3.3-1 (Sheet 6 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
5	Main steam line drain isolation valve V036A(B), normally open fail closed, air-operated valve	Isolates containment in conjunction with SG and main steam line inside containment. Isolates SG No. 1(2) in the event of a MSLB to prevent blowdown from more than one SG.	a. All but DBA	a. Fails closed or fails to open on command	a. Position indication on main control room & high level alarm in condensate drain pot	a. None; local drains provided to limit moisture carry to turbine	
			b. DBA including SGTR	b. Fails to close upon ESF isolation signal	b. Position indication on main control room	b. None; closure of either series isolation valves provides steam line isolation; containment integrity is maintained by either condensate isolation or SG and main steam line inside containment	

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Table 10.3.3-1 (Sheet 7 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
6	Main steam line drain control valve V086A(B) normally closed, fail closed, air-operated valve	Isolates SG A(B) in the event of a main steam line break to prevent blowdown to more than one SG	a. All but DBA	a. Fails closed or fails to open on command	a. Position indication on main control room and high level alarm in condensate drain pot	a. None; local drains provided to limit moisture carryover to turbine	
			b. DBA including SGTR	b. Fails to close upon ESF isolation signal	b. Position indication provided on main control room	b. None, closure of either series isolation valves provides steam line isolation	

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Table 10.3.3-1 (Sheet 8 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
7	Main steam safety valves V030A, V031A, V032A, V033A, V034A, V035A (V030B, V031B, V032B, V033B, V034B, V035B), normally closed	Protect SG A(B) and associated steam line up to MSIV from overpressurization	All	a. Fails to open when required	a. Higher pressure and/or water level in SG A(B)	a. None, 5 out of 6 safety valves for SG A(B) still available with PORV available to supplement relief capacity; also, plant trip occurs on High-3 steam generator level	
				b. Spurious opening or failure to reset after opening	b. Low steam line pressure	b. None, maximum flow from one safety valve less than DBA analysis assumptions, Shutdown effected by other SG or PRHR	

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Table 10.3.3-1 (Sheet 9 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
8	Steam Generator blowdown isolation V074A(B), normally open, fail closed air-operated valve	Isolates blowdown from SG A(B) upon PRHR actuation; isolates containment in conjunction with SG and blowdown lines inside containment	All	a. Fails closed or fails to open upon command	a. Position indication on the main control room and zero flow measured in blowdown system	a. None, blowdown is terminated but has no safety impact	
				b. Fails open or fails to close on command	b. Position indication on main control room	b. None, redundant isolation of blowdown via series isolation valve V075A(B), Containment integrity is maintained by blowdown isolation or SG and blowdown lines inside containment	Redundant isolation is provided for SG volume for PRHR operation via series valves; containment isolation via blowdown isolation or SG and blowdown lines inside containment

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Table 10.3.3-1 (Sheet 10 of 10)
Main Steam Supply System Failure Modes and Effects Analysis

Item	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
9	Steam Generator blowdown isolation V075A(B), normally open, fail closed air-operated valve	Isolates blowdown from SG A(B) upon PRHR actuation	All	a. Fails closed or fails to open upon command	a. Position indication on main control room and zero flow in blowdown system	a. None, blowdown is terminated but has no safety impact	
				b. Fails open or fails to close on command	b. Position indication on main control room	b. None, redundant isolation of blowdown via series isolation valve V074A(B)	Redundant isolation is provided for SG volume for PRHR operation via series valves

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Table 10.3.5-1
Conformance to Branch Technical Position 5-1,
Monitoring of Secondary Side Water Chemistry
in PWR Steam Generators

Branch Technical Position	Conformance
B.1 The applicant's final safety analysis report (FSAR) should describe the implementation of a secondary water chemistry monitoring and control program in accordance with the supplier's recommended procedure to inhibit steam generator corrosion and tube degradation. The applicant should address how its program meets industry guidelines (e.g., EPRI's secondary water chemistry guidelines and Nuclear Energy Institute (NEI) 97-06). In addition, this program should cover all operational modes.	Conforms. As described in Section 10.3.5, the secondary side water chemistry program is in accordance with Reference 1, as required by NEI 97-06, Section 3.7.
Each of the modes should be defined with regard to percent rated thermal power and approximate temperature range.	Conforms. The modes (rated thermal power and approximate temperature range) are specified in the secondary side water chemistry program.
B.2 The secondary water chemistry monitoring and control program should identify a sampling schedule for critical parameters during each mode of operation, as well as the acceptance control criteria for these parameters. At a minimum, the program should control pH, cation conductivity, sodium, and dissolved oxygen. However, other parameters merit consideration, such as specific conductivity, chloride, fluoride, suspended solids, silica, total iron, copper, sulfate, lead, ammonia, and residual hydrazine. Additives to each steam generator should be controlled separately.	Conforms. The secondary side water chemistry program includes the identified content, as applicable.
B.3 The reviewer will evaluate the secondary water chemistry control and monitoring program of each individual plant. Significant deviations from the industry guidelines should be noted and justified technically.	Conforms. The program has no significant deviations from the current industry guidelines.
Records should be made of the monitored item values and should be made available for audit and inspection when deemed necessary.	Conforms. Records are made of the monitored item values through use of the chemistry program procedures.
B.4 Routine changes to the secondary water chemistry control and monitoring program should be reported as part of the biannual FSAR update, as required by 10 CFR 50.71. Changes shall be evaluated in accordance with the requirements of 10 CFR 50	Conforms. Routine FSAR changes to the secondary side water chemistry program are reported as required by 10 CFR 50.71.
Changes shall be evaluated in accordance with the requirements of 10 CFR 50	Conforms. Changes to the program are screened, and evaluated as needed, in accordance with 10 CFR 50.59.

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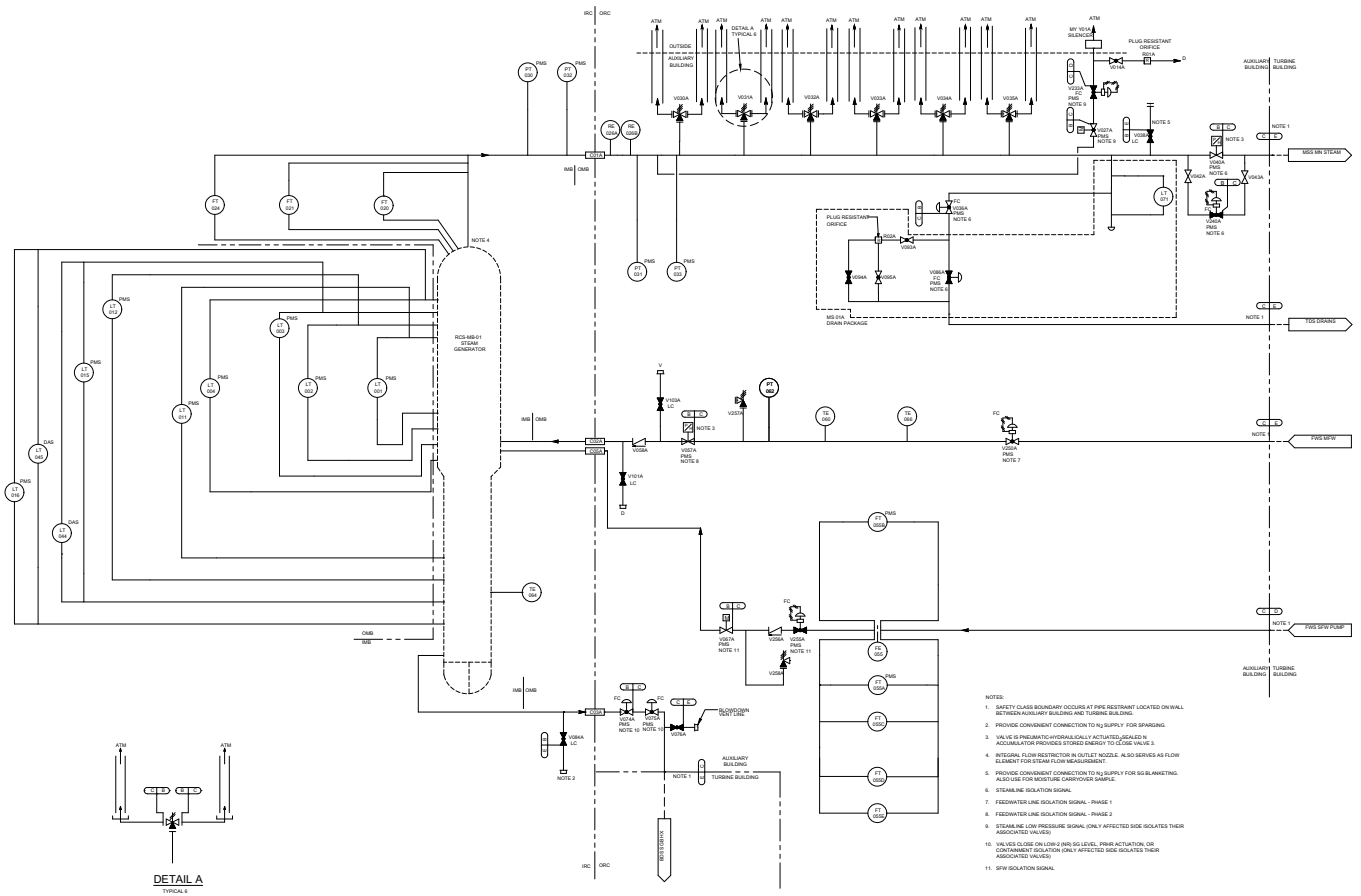

Figure 10.3.2-1 (Sheet 1 of 2)
Simplified Main Steam Piping and Instrumentation Diagram (Safety-Related System)
(REF) SGS 001

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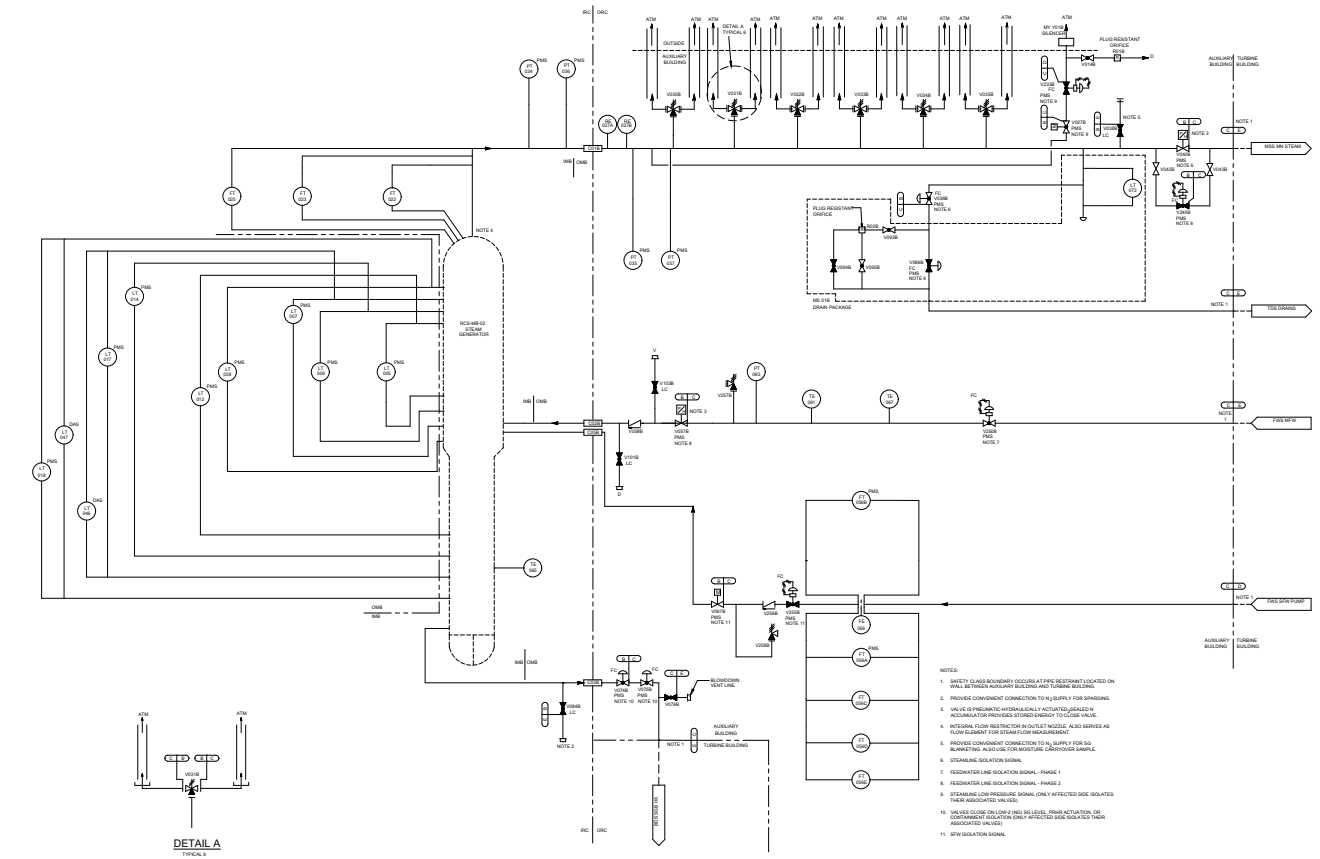

Figure 10.3.2-1 (Sheet 2 of 2)
Simplified Main Steam Piping and Instrumentation Diagram (Safety Related System)
(REF) SGS 002

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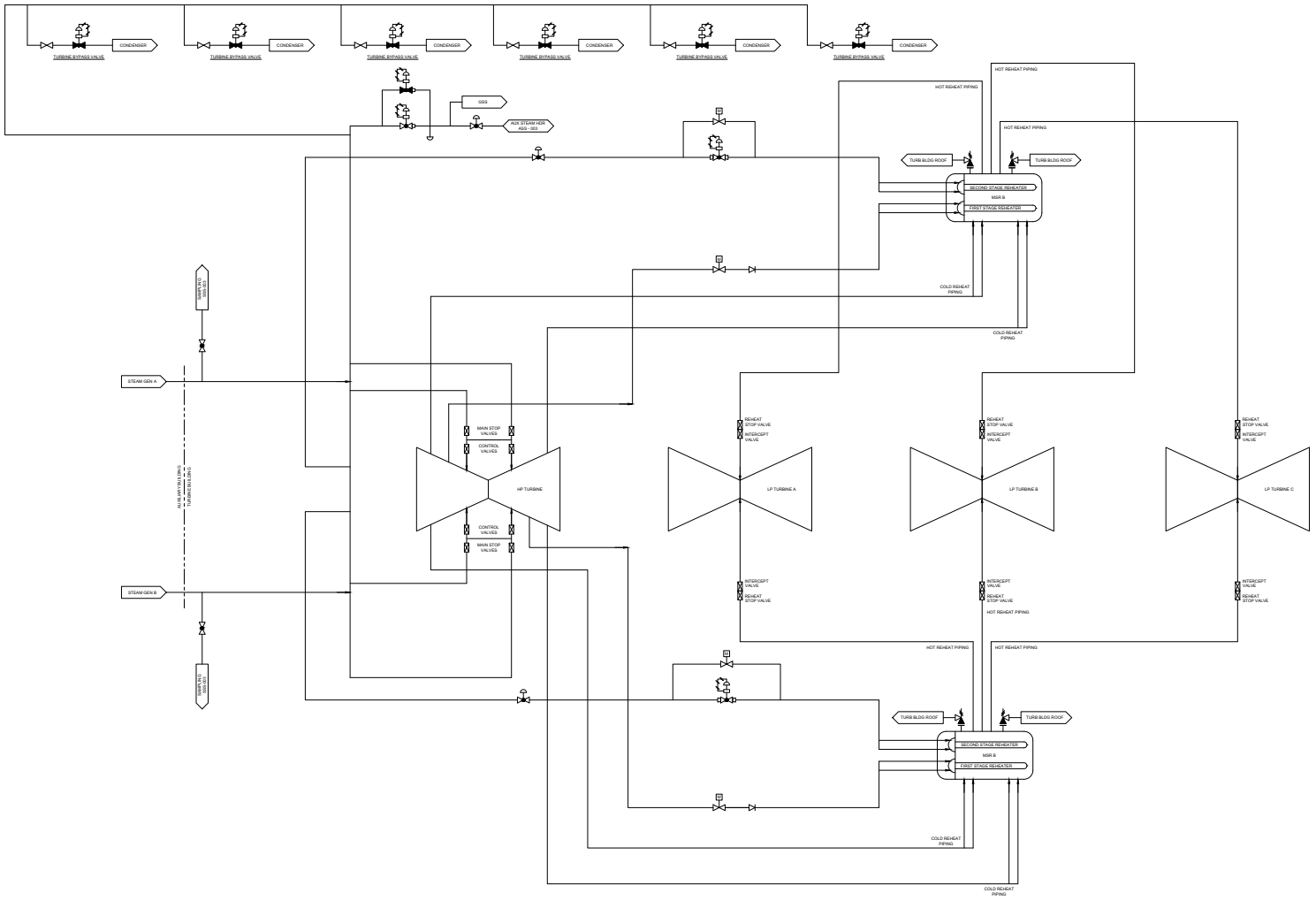

Inside Turbine Building
Figure 10.3.2-2
Main Steam System Diagram
(REF) MSS 001

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10.4 Other Features of Steam and Power Conversion System

This section provides descriptions of each of the principal design features of the steam and power conversion system not in Sections 10.2 and 10.3.

10.4.1 Main Condensers

The main condenser functions as the steam cycle heat sink, receiving and condensing exhaust steam from the main turbine and the turbine bypass system.

10.4.1.1 Design Basis

10.4.1.1.1 Safety Design Basis

The main condenser serves no safety-related function and therefore has no nuclear safety design basis.

10.4.1.1.2 Power Generation Design Basis

The main condenser is designed to receive and condense the full-load main steamflow exhausted from the main turbine and serves as a collection point for vents and drains from various components of the steam cycle system.

The main condenser is designed to receive and condense steam bypass flows up to 40 percent of rated main steam flow while condensing the remaining low-pressure turbine steam flow. This condensing action is accomplished without exceeding the maximum allowable condenser backpressure for main turbine operation.

The condenser hotwell is designed to store at the normal operating water level an amount of condensate equivalent to at least three minutes of full load condensate system operating flow.

The main condenser is designed to deaerate the condensate so that the dissolved oxygen content of the condensate remains under 10 ppb during normal full power operation.

10.4.1.2 System Description

The main condenser is part of the AP1000 condensate system (CDS). The condensate system is described in Subsection 10.4.7 and shown in Figure 10.4.7-1. Classification of equipment and components is given in Section 3.2. Table 10.4.1-1 provides main condenser design data.

The main condenser is a three-shell, single-pass, multipressure, spring-supported unit. Each shell is located beneath its respective low-pressure turbine. The condenser is equipped with titanium or stainless steel tubes. The titanium material provides good corrosion and erosion resisting properties. Freshwater cooled plants do not require the high level corrosion and erosion resistance provided by titanium; therefore, 304L, 316L, 904L, or AL-6X may be substituted if desired.

In a multipressure condenser, the condenser shells operate at slightly different pressures and temperatures. Condensate that is condensed in the low pressure condenser shell drains through internal piping to the high pressure (hottest) shell where it is slightly heated and mixed with condensate of the high pressure shell. Condensate then flows through a single outlet to the suction of the condensate pumps.

The condenser shells are located below the turbine building operating floor and are supported on a spring-mounted foundation from the turbine building basemat. A rigid connection is provided between

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each low-pressure turbine exhaust opening and the steam inlet connections of the condenser. Two low-pressure feedwater heaters are located in the neck area of each condenser shell. Piping is installed for hotwell level control and condensate sampling.

10.4.1.2.1 System Operation

During normal power operation, exhaust steam from the low-pressure turbines is directed into the main condenser shells. The condenser also receives auxiliary system flows, such as feedwater heater vents and drains and gland sealing steam spillover and drains.

The hotwell level controller provides automatic makeup or rejection of condensate to maintain a normal level in the condenser hotwells. On low level, the makeup control valves open and admit condensate by vacuum draw to the hotwell from the condensate storage tank. On high-water level the condensate reject control valves open to divert water from the condensate pump discharge to the condensate storage tank. This rejection automatically stops when the hotwell level falls to within normal operating range. Rejection to the storage tank can be manually overridden upon an indication of high-hotwell conductivity to prevent transfer of contaminants into the condensate storage tank in the event of a condenser tube failure.

Air inleakage and noncondensable gases contained in the turbine exhaust steam are collected in the condenser and removed by the main condenser air removal system. The condenser air removal system is discussed further in Subsection 10.4.2.

To protect the condenser shells and turbine exhaust hoods from overpressurization, steam relief blowout diaphragms are provided in the low-pressure turbine exhaust hoods.

The main condenser is capable of accepting up to 40 percent of rated main steam flow from the turbine bypass system. Operation of the turbine bypass system is discussed in Subsection 10.4.4. In the event of high condenser pressure or trip of the circulating water pumps, the turbine bypass valves are prohibited from opening.

Distribution headers are incorporated to protect the condenser tubes, feedwater heaters located in the condenser neck, and other condenser components from turbine bypass or high-temperature drains entering the condenser shell.

The main condenser interfaces with secondary sampling system (SSS) to permit sampling of the condensate in the condenser hotwell. Also, grab sampling capability is provided for each condenser tubesheet. Should circulating water in-leakage occur, these provisions permit determination of which tube bundle has sustained the leakage. Steps may be taken to repair or plug the leaking tubes. This is performed by isolating the circulating water system from the affected water box. Plant power is reduced as necessary. This will temporarily reduce condenser capacity by approximately 50 percent. The water box is then drained and the affected tubes are either repaired or plugged. Refer to Subsection 10.3.5.5 for a discussion regarding action levels for abnormal secondary cycle chemistry conditions.

A condenser tube cleaning system performs mechanical cleaning of the circulating water side of the tubes. This cleaning, along with chemical treatment of the circulating water, reduces fouling and helps to maintain the thermal performance of the condenser.

10.4.1.3 Safety Evaluation

The main condenser has no safety-related function and therefore requires no nuclear safety evaluation.

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During normal operation and shutdown, the main condenser has no significant inventory of radioactive contaminants. Radioactive contaminants may enter through a steam generator tube leak. A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated operating concentrations of radioactive contaminants, is included in Chapter 11. No hydrogen buildup in the main condenser is anticipated. The failure of the main condenser and any resultant flooding will not preclude operation of any essential system since no safety-related equipment is located in the turbine building and the water cannot reach safety-related equipment located in Category I plant structures.

10.4.1.4 Tests and Inspections

The condenser water boxes are hydrostatically tested at the fabrication facility, prior to field assembly. After erection, the condenser shells are tested in accordance with Reference 1. Tube joints are leak tested during construction.

10.4.1.5 Instrumentation Applications

The main condenser hotwell is equipped with level control devices for control of automatic makeup and rejection of condensate. Condensate level in the condenser hotwell is indicated in the main control room and alarms on high or low level.

Condenser pressure for each condenser shell is indicated in the main control room and alarms on high level. Also, pressure instrumentation is provided to alarm prior to reaching the maximum turbine operating backpressure limit. Pressure devices are provided to trip the main turbine on high turbine exhaust pressure.

Temperature indication for monitoring condenser performance is provided.

10.4.2 Main Condenser Evacuation System

Main condenser evacuation is performed by the condenser air removal system (CMS). The system removes noncondensable gases and air from the main condenser during plant startup, cooldown, and normal operation. This action is provided by liquid ring vacuum pumps.

10.4.2.1 Design Basis

10.4.2.1.1 Safety Design Basis

The condenser air removal system serves no safety-related function and therefore has no nuclear safety design basis.

10.4.2.1.2 Power Generation Design Basis

- The condenser air removal system removes air and noncondensable gases from the condenser during plant startup, cooldown, and normal operation from the steam side of the three main condenser shells and exhausts them into the atmosphere.

- The system establishes and maintains a vacuum in the condenser during startup and normal operation by the use of liquid ring vacuum pumps.

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10.4.2.2 System Description

10.4.2.2.1 General Description

Classification of equipment and components is given in Section 3.2.

The air removal system consists of four liquid ring vacuum pumps that remove air and noncondensable gases from the three condenser shells during normal operation and provide condenser hogging during startup. One vacuum pump is provided for each condenser shell, and one pump is provided as a standby. The noncondensable gases, together with a quantity of vapor, are drawn through the air cooler sections of condenser shells to the suction of the vacuum pumps. These noncondensables consist mainly of air, nitrogen, and ammonia. No hydrogen buildup is anticipated in the system (see Subsection 10.4.1.3). Dissolved oxygen is present in the condensate and condenser hotwell inventory. Only trace amounts of this oxygen are released in the condenser, and the amounts are negligible compared to the amount of gas and vapor being evacuated by the system. Therefore, the potential for explosive mixtures within the condenser air removal system does not exist.

The [[circulating water system (CWS)]] provides the cooling water for the vacuum pump seal water heat exchangers. The seal water is kept cooler than the saturation temperature in the condenser to maintain satisfactory vacuum pump performance.

The noncondensable gases and vapor mixture discharged to the atmosphere are not normally radioactive. However, it is possible for the mixture to become contaminated in the event of primary-to-secondary system leakage. Air inleakage and noncondensable gases removed from the condenser and discharged by the vacuum pumps are routed to the turbine island vents, drains, and relief system (TDS) and monitored for radioactivity. Upon detection of unacceptable levels of radiation, operating procedures are implemented. A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated release from the system, is included in Chapter 11.

The discharge from the condenser air removal system has a connection for taking local grab samples. Connections also allow the installation of portable, continuous sampling equipment.

Should the condenser air removal system become inoperable, a gradual increase in condenser back pressure would result from the buildup of noncondensable gases. This increase in backpressure would cause a decrease in the turbine cycle efficiency. If the condenser air removal system remains inoperable, condenser backpressure increases to the turbine trip setpoint, and a turbine trip is initiated. Loss of the main condenser vacuum causes a turbine trip but does not close the main steam isolation valves. A loss of condenser vacuum incident is described in Subsection 15.2.5.

10.4.2.2.2 Component Description

The liquid ring vacuum pumps are supplied as packaged units. Major components in each package include a vacuum pump, seal water heat exchanger, seal water pump, and air/water separator. Seal water is supplied to seal the clearances in the pump and also to condense vapor at the inlet to the pump. Seal water flows through one side of the seal water heat exchanger and [[circulating water]] flows through the other side. Seal water make up is provided by the demineralized water transfer and storage system (DWS).

Piping and valves are carbon steel. The piping is designed to ASME B31.1.

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10.4.2.2.3 System Operation

During startup operation, air is removed from the condenser by operating three liquid ring vacuum pumps. The fourth pump is on standby.

During normal plant operation, noncondensable gases are removed from the condenser by three vacuum pumps. If one pump trips, the condition is alarmed in the main control room, and the standby pump is started.

10.4.2.3 Safety Evaluation

The condenser air removal system has no safety-related function and therefore requires no nuclear safety evaluation.

10.4.2.4 Tests and Inspections

Testing and inspection of the system is performed prior to plant operation. A performance test is conducted on each pump in accordance with Reference 2. In addition, the pumps are hydrostatically tested.

10.4.2.5 Instrumentation Applications

The effectiveness of the air removal system is indicated by monitoring condenser pressure, using instrumentation described in Subsection 10.4.1.5. Vacuum pump skid status (enabled/disabled) is indicated in the main control room, and pump trips are alarmed.

Volumetric flow indication is provided to monitor the quantity of exhausted noncondensable gases.

A radiation detector monitors the discharge of the condenser vacuum pumps through the turbine island vents, drains, and relief system (TDS). The radiation detector is indicated and alarmed. For process and effluent radiological monitoring and sampling systems, refer to Section 11.5.

10.4.3 Gland Seal System

10.4.3.1 Design Basis

10.4.3.1.1 Safety Design Basis

The gland seal system (GSS) serves no safety-related function and therefore has no nuclear safety design basis.

10.4.3.1.2 Power Generation Design Basis

- The gland seal system prevents air leakage into and steam leakage out of the casings of the turbine-generator.
- The system returns condensed steam to the condenser and exhausts noncondensable gases into the atmosphere via the turbine island vents, drains, and relief system.
- The presence of radioactive contamination in the noncondensable gas exhausted from the gland seal condenser, is detected by a radiation monitor in the turbine island vents, drains, and relief system.

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10.4.3.2 System Description

10.4.3.2.1 General Description

The gland seal system consists of the following items and assemblies:

- Steam supply header
- Steam drains/noncondensable gas exhaust header
- Two motor driven gland seal condenser exhaust blowers
- Associated piping, valves, and controls
- Gland seal condenser
- Vent and drain lines

The quality group standards for the gland seal system are provided in Section 3.2. The gland seal system is shown in Figure 10.4.3-1.

10.4.3.2.2 System Operation

The annular space through which the turbine shaft penetrates the turbine casing is sealed by steam supplied to the rotor glands. Where the packing seals against positive pressure, the sealing steam connection acts as a leakoff. Where the packing seals against vacuum, the sealing steam either is drawn into the casing or leaks outward to a vent annulus maintained at a slight vacuum. The vent annulus receives air leakage from the outside. The air-steam mixture is drawn to the gland seal condenser.

Sealing steam is distributed to the turbine shaft seals through the steam-seal header. This sealing steam is supplied from either the auxiliary steam system (ASS), or from main steam (MSS), extracted ahead of the high-pressure turbine control valves. Steam flow to the header is controlled by the steam-seal feed valve which responds to maintain the steam-seal supply header pressure. The low and high pressure turbine gland steam pressures are maintained by pressure regulating valves provided in both main steam and auxiliary steam system piping. Excess steam is returned to Condenser A via the spillover control valve which automatically opens to bypass excess steam from the GSS.

During the initial startup phase of turbine-generator operation, steam is supplied to the gland seal system from the auxiliary steam header which is supplied from the auxiliary boiler. At times other than initial startup, turbine-generator sealing steam is supplied from the MSV and CV gland steam leak-off, the auxiliary steam system, or from main steam.

At the outer ends of the glands, collection piping routes the mixture of air and excess seal steam to the gland seal condenser. The gland seal condenser is a shell and tube type heat exchanger where the steam-air mixture from the turbine seals is discharged into the shell side and condensate flows through the tube side as a cooling medium. The gland seal condenser internal pressure is maintained at a slight vacuum by a motor-operated blower. There are two 100-percent blowers mounted in parallel. Condensate from the steam-air mixture drains to the main condenser while noncondensables are exhausted to the turbine island vents, drains, and relief system through a common discharge line shared by the vapor extractor blowers.

The mixture of noncondensable gases discharged from the gland seal condenser blower is not normally radioactive; however, in the event of significant primary-to-secondary system leakage due to a steam generator tube leak, it is possible to discharge radioactively contaminated gases. The headered discharge line vents to the turbine vents, drains, and relief system which contains a radiation monitor for detection of radioactivity. Upon detection of unacceptable levels of radiation,

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operating procedures are implemented. A description of the radiological aspects of primary-to-secondary system leakage is included in Chapter 11.

Failure of the gland seal system normally results in no release of radioactivity to the atmosphere.

10.4.3.3 Safety Evaluation

The gland seal system has no safety-related function and therefore requires no nuclear safety evaluation.

10.4.3.4 Tests and Inspections

The system is tested in accordance with written procedures during the initial testing and operation program. Since the gland seal system is in use and essential parameters are monitored during normal plant operation, the satisfactory operation of the system components demonstrates system operability.

10.4.3.5 Instrumentation Applications

A pressure controller is provided to maintain the steam-seal supply header pressure by providing signals to the steam-seal feed valve. Excess steam flow is handled by the gland spillover control valve which discharges to Condenser A.

The gland seal condenser is monitored for shell side pressure and internal liquid level.

Pressure indication with appropriate alarm is provided for monitoring the operation of the system. A radiation detector with an alarm is provided in the turbine island vents, drains, and relief system to detect radiation associated with primary-to-secondary side leakage in the steam generators.

10.4.4 Turbine Bypass System

The turbine bypass system provides the capability to bypass main steam from the steam generators to the main condenser in a controlled manner to dissipate heat and to minimize transient effects on the reactor coolant system during startup, hot shutdown, cooldown, and step-load reductions in generator load. The turbine bypass system is also called the steam dump system, and is part of the main steam system (MSS).

10.4.4.1 Design Basis

10.4.4.1.1 Safety Design Basis

The turbine bypass system serves no safety-related function and therefore has no nuclear safety design basis. The nonsafety-related turbine bypass valves are credited in a single failure analysis to mitigate the event for those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event.

10.4.4.1.2 Power Generation Design Basis

The turbine bypass system has the capacity to bypass up to 40 percent of rated main steam flow to the main condenser.

The turbine bypass system bypasses steam to the main condenser during plant startup and permits a manually controlled cooldown of the reactor coolant system to the point where the normal residual heat removal system can be placed in service.

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The turbine bypass system total flow capacity, in combination with bypass valve response time, reactor coolant system design, and reactor control system response, is sufficient to reduce challenges to the main steam power-operated relief valves, main steam safety valves, and pressurizer safety valves during: reactor trip from 100-percent power; and 100-percent load rejection or turbine trip from 100-percent power without reactor trip.

10.4.4.2 System Description

10.4.4.2.1 General Description

The turbine bypass system is part of the main steam system and is shown on Figure 10.3.2-2. The system consists of a manifold connected to the main steam lines upstream of the turbine stop valves and of lines from the manifold with regulating valves to each condenser shell.

The capacity of the system, along with the NSSS control systems, provides the capability to meet the design requirement bases specified in Subsection 10.4.4.1.2. For power changes less than or equal to a 10 percent change in electrical load, the turbine bypass system is not actuated; the total power change is handled by the reactor power control, pressurizer level control, pressurizer pressure control, and the steam generator level control systems described in Section 7.7. For load rejections greater than 10 percent but less than 50 percent, or a turbine trip from 50 percent power or less, the turbine bypass system operates in conjunction with the same control systems used for the 10 percent or less load change to meet the design basis requirements specified in Subsection 10.4.4.1.2. For load rejections greater than 50 percent power, the rapid power reduction system (described in Section 7.7) operates in conjunction with the previously mentioned control systems to meet the design basis requirements. The rapid power reduction system is designed to rapidly reduce the nuclear power to a value that can be handled by the turbine bypass system. Certain transient conditions or system degradations beyond those of Subsection 10.4.4.1.2 may result in a reactor trip and may result in the operation of the main steam power operated relief and safety valves.

10.4.4.2.2 Component Description

The turbine bypass valves are globe valves and are electropneumatically operated. The valves fail to a closed position upon loss of air or electric signal. A modulating positioner responds to the electric signal from the control system and provides an appropriate air pressure to the valve actuator for modulating the valve open.

Solenoid valves located in the air line to each bypass valve actuator serve as protective interlocks for bypass valve actuation and for tripping the valve open or closed. One of the solenoid valves is energized, when required, to bypass the modulating positioner and provide full air pressure to the actuator to quickly trip open the bypass valve. Other solenoid valves, when deenergized, block the air supply to the actuator and vent the actuator; this action blocks the bypass valve from opening, or closes the valve if opened.

Two of the blocking solenoid valves for each turbine bypass valve are redundant and block bypass valve actuation upon low reactor coolant system T_{avg}. This minimizes the possibility of excessive reactor coolant system cooldown. However, the low T_{avg} block can be manually bypassed for the bypass valves that are designated as cooldown valves to allow operation during plant cooldown.

Another blocking solenoid valve prevents actuation of the bypass valve when the condenser is not available. This solenoid valve also prevents unblocking the steam dump valve when the condenser is available unless one of the following signals exist:

- High negative rate of change of turbine pressure
- Reactor trip

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- Control system in the steam header pressure control mode (see Subsection 10.4.4.3)

10.4.4.3 System Operation

The turbine bypass system has two modes of operation:

- T_{avg} control mode
- Pressure control mode

The T_{avg} control mode is the normal at-power control mode. The turbine bypass system is regulated by the difference between the measured reactor coolant system average coolant temperature (T_{avg}) and a T_{avg} setpoint derived from turbine first-stage pressure. This prevents a large increase in reactor coolant temperature following a large, sudden load decrease. Turbine bypass valve control in conjunction with reactor power control results in a match between reactor power and turbine load.

The pressure control mode is used to remove decay heat during plant startup and cooldown and to control steam pressure following a reactor trip. The difference between steam header pressure and a pressure setpoint is used to control the turbine bypass flow. The pressure setpoint is manually adjustable and is based on the desired reactor coolant system temperature. The turbine bypass system is operated in the pressure control mode when the plant is at no-load and there is no turbine load reference. There are three pressure control operational modes as follows:

- Header pressure – control derived from the difference between header pressure and pressure setpoint

- Cooldown – control derived from the manually selected desired reactor coolant system cooldown rate and the target reactor coolant system temperature

- Manual – control derived from direct use of valve loading signals.

The bypass valves are divided into two banks. The banks are opened sequentially; the second bank starts to open only after a demand signal that is greater than the full-open demand of the first bank is generated.

The turbine bypass valves have two stroke control modes, modulate and trip open/close. If the demand signal is greater than the full open demand for the particular bank of valves, a trip open demand signal is generated. When the demand signal decreases below the full-open demand, the trip open demand clears and the valves return to the modulating mode. Additional description of steam dump logic is given in Section 7.7.

Chapter 15 addresses credible single failures of the turbine bypass system. If the bypass valves fail-open, additional heat load is placed on the condenser. If this load is great enough, the turbine is tripped on high condenser pressure. Ultimate overpressure protection for the condenser is provided by turbine rupture discs. If the bypass valves fail-closed, the power operated relief valves (reference Subsection 10.3.2.2.3) permit controlled cooldown of the reactor.

10.4.4.4 Safety Evaluation

There is no safety-related equipment in the vicinity of the turbine bypass system. The high-energy lines of the turbine bypass system are located in the turbine building.

The failure of a turbine bypass high-energy line will not disable the turbine speed control system. The turbine speed control system is designed in such a manner that its failure will cause a turbine trip. Additional information concerning speed control can be found in Subsection 10.2.2.3.

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10.4.4.5 Inspection and Testing Requirements

Before the system is placed in service, turbine bypass valves are tested to verify they function properly. Leaktightness of the steam lines is confirmed by an initial service test. System piping and valves are accessible for inspection. No inservice inspection and testing is required except for the turbine bypass valves which are included in the inservice program as discussed in Subsection 3.9.6.

10.4.4.6 Instrumentation Applications

Turbine bypass controls are described in Section 7.7. Controls in the main control room are provided for selection of the system operating mode. Pressure indication and valve position indication are also provided in the main control room.

10.4.5 Circulating Water System

10.4.5.1 Design Basis

10.4.5.1.1 Safety Design Basis

The circulating water system (CWS) serves no safety-related function and therefore has no nuclear safety design basis.

10.4.5.1.2 Power Generation Design Basis

[[The CWS supplies cooling water to remove heat from the main condensers. The CWS also supplies cooling water to the turbine building closed cooling water system (TCS) heat exchangers and the condenser vacuum pump seal water heat exchangers under varying conditions of power plant loading and design weather conditions.]]

10.4.5.2 System Description

10.4.5.2.1 General Description

Classification of components and equipment in the circulating water system is given in Section 3.2.

[[The CWS and the cooling tower provide a heat sink for the waste heat exhausted from the steam turbine. Additional cooling is supplied from the CWS through a tap in the main supply header for the TCS heat exchangers and the condenser vacuum pump seal water heat exchangers. CWS design parameters are provided in Table 10.4.1-1 and Table 10.4.5-1.]]

[[The CWS consists of four 25-percent-capacity circulating water pumps, one hyperbolic natural draft cooling tower (NDCT), and associated piping, valves, and instrumentation.]]

Makeup water to the CWS is provided by the raw water system (RWS). In addition, water chemistry is controlled by a local chemical feed system.

10.4.5.2.2 Component Description

Circulating Water Pumps

[[The four circulating water pumps are vertical, wet pit, single-stage, mixed-flow pumps driven by electric motors. The pumps are mounted in a pump pit, which is attached to the cooling tower basin. The four pump discharge lines connect to a]] common header which connects to the two inlet water boxes of the condenser [[and may also supply cooling water to the TCS and condenser vacuum pump seal water heat exchangers. Each pump discharge line has a motor-operated butterfly valve

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located between the pump discharge and the main header. This permits isolation of one pump for maintenance and allows for three-pump operation.]]

[[Cooling Tower]]

[[The cooling tower is site specific with this description provided as a reference design using a hyperbolic natural draft structure. Operation of the cooling tower during conditions that are more restrictive than design conditions may result in higher condenser back pressure.]]

[[The cooling tower has a basin which serves as storage for the circulating water inventory and allows bypassing of the cooling tower during cold weather operations. This basin is connected to the intake of the circulating water pumps by a canal.]]

[[Cooling Tower Makeup and Blowdown]]

The circulating water system makeup is provided by the raw water system. [[In addition, a small portion of Service Water System (SWS) flow is normally diverted to the circulating water system cooling tower basins as blowdown to control the level of solids concentration in the SWS. Makeup to and blowdown from the CWS is controlled by the makeup and blowdown control valves. The makeup control valves maintain the cooling tower basin level. These valves, along with a local chemical feed system, provide chemistry control in the circulating water in order to maintain a noncorrosive, nonscale-forming condition and limit biological growth in CWS components.]]

Piping and Valves

[[The underground portions of the CWS piping are constructed of prestressed concrete cylinder pipe (PCCP). The remainder of the piping is carbon steel and is coated internally with a corrosion-resistant compound. The primary drainage path for the condenser water boxes and tube bundles is via gravity to the cooling tower basins using the service water blowdown line (Subsection 9.2.1.2.2). As an alternate, condenser water box drain lines are provided that direct drainage to the turbine building sumps.]] Motor-operated butterfly valves are provided in each of the circulating water lines at their inlet to and exit from the condenser shell to allow isolation of portions of the condenser. [[Control valves provide regulation of cooling tower makeup and blowdown.]]

The circulating water system is designed to withstand the maximum operating discharge pressure of the circulating water pumps. [[Piping includes the expansion joints, butterfly valves, condenser water boxes, and tube bundles. Due to differences in distance and elevation from the circulating water pumps, CWS components and piping have different design working pressures. The design working pressure rating of each component exceeds the maximum operating discharge pressure developed by the circulating water pumps at the location of each component. The design working pressures for each component are listed below.]]

[[Component	Design Working
Steel Pipe	150 psi
PCCP Pipe	100 psi
Expansion Joints	90 psi
Condenser Tube Bundle	90 psi
Water Box	90 psi]]

Circulating Water Chemical Injection

Circulating water chemistry is maintained by a local chemical feed skid at the CWS cooling tower. Circulating water system chemical feed equipment injects the required chemicals into the circulating water [[at the CWS cooling tower basin area.]] This maintains a noncorrosive, nonscale-forming

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condition and limits the biological film formation that reduces the heat transfer rate in the condenser and the heat exchangers supplied by the circulating water system.

The specific chemicals used within the system are determined by the site water conditions and therefore will be provided as discussed in subsection 10.4.12. The chemicals can be divided into six categories based upon function: biocide, algaecide, pH adjuster, corrosion inhibitor, scale inhibitor, and a silt dispersant. The pH adjuster, corrosion inhibitor, scale inhibitor, and dispersant are metered into the system continuously or as required to maintain proper concentrations. The biocide application frequency may vary with seasons. [[The algaecide is applied, as necessary, to control algae formation on the cooling tower.]]

Addition of biocide and water treatment chemicals is performed by local chemical feed injection metering pumps and is adjusted as required. [[Chemical concentrations are measured through analysis of grab samples from the CWS.]] Residual chlorine is measured to monitor the effectiveness of the biocide treatment.

10.4.5.2.3 System Operation

[[The four circulating water pumps take suction from the circulating water pump pit and circulate the water through the tube side of the main condenser, with smaller flows to the TCS, the condenser vacuum pump seal water heat exchangers, and back through the piping discharge network to the NDCT. The NDCT cools the circulating water by discharging the water through nozzles in the tower distribution headers. The water is then diffused through fill material to the basin beneath the tower and, in the process, rejects heat to the atmosphere. Provisions are made during freezing weather to minimize the possibility of ice accumulation.]]

[[Isolation of portions of the cooling tower and bypass of flow directly into the basin are both accomplished via manipulation of the flume slide gates. Closing the flume slide gates takes a portion of the tower out of service and can be used to prevent freezing conditions within the tower.]]

The raw water system supplies makeup water [[to the cooling tower basin to replace water losses due to evaporation, wind drift, and blowdown. A separate connection is also provided between the RWS and CWS to fill the CWS piping.]]

A condenser tube cleaning system is installed to clean the circulating water side of the main condenser tubes. [[Blowdown from the CWS is taken from the discharge of the CWS pumps and is discharged to the plant outfall.]]

The circulating water system is used to supply cooling water to the main condenser to condense the steam exhausted from the main turbine. If the [[circulating water pumps, the cooling tower, or the circulating water piping malfunction such that]] condenser backpressure rises above the maximum allowable value, the main condenser will no longer be able to adequately support unit operation. Cooldown of the reactor may be accomplished by using the power-operated atmospheric steam relief valves or safety valves rather than the turbine bypass system when the condenser is not available.

Passage of condensate from the main condenser into the circulating water system through a condenser tube leak is not possible during power generation operation, since the circulating water system operates at a greater pressure than the condenser.

Turbine building closed cooling water in the TCS heat exchangers is maintained at a higher pressure than the [[circulating water]] to prevent leakage of the [[circulating water]] into the closed cooling water system.

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Cooling water to the condenser vacuum pump seal water heat exchangers is supplied from the [[circulating water]] system. Cooling water flow from the [[circulating water]] system is normally maintained through all four heat exchangers to facilitate placing the spare condenser vacuum pump in service. Isolation valves are provided for the condenser vacuum pump seal water heat exchanger cooling water supply lines to facilitate maintenance.

Small circulating water system leaks in the turbine building will drain into the waste water system. Large circulating water system leaks due to pipe failures will be indicated in the control room by a loss of vacuum in the condenser shell. The effects of flooding due to a circulating water system failure, such as the rupture of an expansion joint, will not result in detrimental effects on safety-related equipment since there is no safety-related equipment in the turbine building and the base slab of the turbine building is located at grade elevation. Water from a system rupture will run out of the building through a relief panel in the turbine building west wall before the level could rise high enough to cause damage. Site grading will carry the water away from safety-related buildings.

[[The cooling tower is located so that collapse of the tower has no potential to damage equipment, components, or structures required for safe shutdown of the plant.]]

10.4.5.3 Safety Evaluation

The circulating water system has no safety-related function and therefore requires no nuclear safety evaluation.

10.4.5.4 Tests and Inspections

Components of the circulating water system are accessible as required for inspection during plant power generation. [[The circulating water pumps are tested in accordance with standards of the Hydraulic Institute.]] Performance, hydrostatic, and leakage tests associated with preinstallation and preoperational testing are performed on the circulating water system. The system performance and structural and leaktight integrity of system components are demonstrated by continuous operation.

Thermal performance of the cooling towers is tested and verified using established industry test standards.

10.4.5.5 Instrumentation Applications

[[Instrumentation provided indicates the open and closed positions of motor-operated butterfly valves in the circulating water piping. Controls and interlocks are in place to ensure the safe operation of the pumps. Alarms will alert plant operators to any abnormal conditions and to take action accordingly.]]

[[Local grab samples are used to periodically test the circulating water quality to limit harmful effects to the system piping and valves due to improper water chemistry.]]

[[Pressure indication is provided on the circulating water pump discharge lines.]] A differential pressure transmitter is provided between one inlet and outlet branch to the condenser. This differential pressure transmitter is used to determine the frequency of operating the condenser tube cleaning system (CES).

Temperature indication is supplied [[on the common CWS inlet header to the TCS heat exchanger trains.]] This temperature is also representative of the inlet cooling water temperature to the main condenser.

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A flow element is provided [[on the common discharge line from the TCS heat exchangers to allow monitoring of the total flow through the TCS heat exchangers.]] Flow measurement for the raw water makeup [[to the cooling tower and for the cooling tower blowdown]] is also provided.

[[Level instrumentation provided in the cooling tower basin activates makeup flow from the RWS to the cooling tower basin. Level instrumentation also annunciates a Low-2 water level, Low level, and a High water level in the pump structure, and Low and High water levels in the cooling tower basin.]]

The circulating water chemistry is controlled [[by regulating cooling tower blowdown and by chemical addition,]] to maintain the circulating water with an acceptable Langelier Index [[and is maintained in range established by plant chemistry personnel. The system accomplishes this by regulating the blowdown valve. This regulation causes the tower basin water level to fluctuate. The fluctuation is sensed by a level controller that operates the makeup valve to the cooling tower basin.]]

The control approach is to allow the makeup water to concentrate naturally to its upper limit. Provisions are made to add chemicals for pH control.

The cycles of concentration [[at which the cooling tower is operated]] is dependent on the quality of the cooling tower makeup water.

Monitoring of the circulating water system is performed through the data display and processing system. Control functions are performed by the plant control system. Appropriate alarms and displays are available in the control room. See Chapter 7.

10.4.6 Condensate Polishing System

The condensate polishing system (CPS) can be used to remove corrosion products and ionic impurities from the condensate system during plant startup, hot standby, power operation with abnormal secondary cycle chemistry, safe shutdown, and cold shutdown operations.

10.4.6.1 Design Basis

10.4.6.1.1 Safety Design Basis

The condensate polishing system serves no safety-related function and therefore has no nuclear safety-related design basis.

10.4.6.1.2 Power Generation Design Basis

The power generation design bases are to:

- Remove corrosion products, dissolved solids and other impurities from the condensate system and maintain a noncorrosive environment within the condensate, feedwater and steam generator systems
- Provide polishing capacity for processing one-third of the maximum condensate flow in a sidestream arrangement
- Provide polishing capability during normal startup and shutdown operations of the plant
- Provide for plant operation with a “continuous” condenser tube leak of .001 gpm or a “faulted” leak of 0.1 gpm until repairs can be completed or until an orderly shutdown is achieved

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10.4.6.2 System Description

The condensate polishing system is used during operating modes of startup, hot standby, power operation with abnormal secondary cycle chemistry, safe shutdown, and cold shutdown. Classification of components in the CPS is identified in Section 3.2. The major components for the condensate polishing system are described below. The condensate polishing system is shown in Figure 10.4.6-1.

Deep Bed Mixed Resin Polishers

The polisher vessels are constructed of carbon steel with a protective rubber lining on the inside of the vessel. Level indication (site glass) is provided.

Resin Trap

The resin traps are located in the effluent piping of the vessels. Differential pressure across the trap is monitored.

Spent Resin Tank

The spent resin tank is constructed of carbon steel with an interior protective rubber lining. It is used for storage of exhausted or spent resin prior to shipping offsite for regeneration or disposal.

Resin Addition Hopper and Eductor

The resin addition hopper stores regenerated or new resin and the eductor is used to inject resin into the polisher vessels. The hopper is constructed of carbon steel. The eductor uses demineralized water to transfer the resin to the vessels.

10.4.6.3 System Operation

The condensate polishing system cleans up the condensate during startup to meet condensate and feedwater system water chemistry specifications as described in Subsection 10.3.5. The condensate system is recirculated to the hotwell during startup until the desired water quality is attained. Condensate system startup operation is described in Subsection 10.4.7. Utilization of the condensate polishing system during startup assists in minimizing the startup duration of the plant.

During power operation, the condensate polishers are used only when abnormal secondary cycle conditions exist. This allows for continued operation of the plant with a “continuous” condenser tube leak of 0.001 gpm or a “faulted” leak of 0.1 gpm until repairs can be made or until an orderly shutdown is achieved. The condensate polisher flow is controlled by the condensate polisher bypass valve.

Exhausted or spent resin is removed from the vessels and replaced with new or regenerated resin. Resin replacement requires the polisher vessel to be out of service. Spent resin is transferred directly from the polisher vessel to a truck or to the spent resin tank until it can be removed offsite. Spent condensate polishing resin will normally be nonradioactive and not require any special packaging prior to disposal. In the event of radioactive contamination of the resin in a vessel, temporary shielding is installed (if required). Radioactive resin is transferred directly from the condensate polishing vessel or from the spent resin tank to a temporary processing unit. Radiation monitors associated with the steam generator blowdown system, the steam generator system (main steam), and the turbine island vents, drains and relief system provide the means to determine if the secondary side is radioactively contaminated. Subsection 11.4.2 describes waste management of radioactively contaminated resin. A spill containment barrier is provided to contain spent resin tank or condensate polisher vessel contents in the event of a tank failure. The spill containment barrier is a

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curb surrounding the area containing the spent resin tank and condensate polisher vessels with sufficient height to contain the contents of a full tank or vessel.

The procedures for radiation protection and the handling and processing of radwaste are addressed in Chapters 11 and 12. Shielding design is described in Section 12.3.

Upon removal of the exhausted resin from the polisher vessel, the vessel is rinsed and the new resin is placed in the vessel using the resin addition hopper and eductor. After the new cation and anion resins are placed in the vessel, demineralized water is added until the water level is just above the resin bed. Compressed air from the plant service air system is injected up through the resin bed to fluidize and thoroughly mix the resins. Prior to plant startup, a new resin bed is rinsed and resin performance is verified, with flow through the vessel discharged to the waste water system. The polisher vessel is then placed in operation or on standby.

10.4.6.4 Safety Evaluations

The condensate polishing system has no safety-related function and therefore requires no nuclear safety evaluation.

10.4.6.5 Tests and Inspections

The condensate polishing system is operationally checked prior to plant startup to verify proper functioning of the polisher vessels and associated instrumentation and controls.

10.4.6.6 Instrument Applications

When the condensate polishing system is in service, polishing system differential pressure instrumentation provides a control signal to the condensate bypass valve which maintains sufficient flow through the polisher vessels for optimum performance. One or both polisher(s) is/are removed from service when: 1) a high differential pressure exists across the polisher vessel, 2) the ion exchange resin capacity becomes exhausted as evidenced by a high effluent conductivity, or 3) at the completion of a pre-determined volume through-put. The resin traps are monitored for high differential pressure and an alarm indicates the need to backwash the trap.

10.4.7 Condensate and Feedwater System

The condensate and feedwater system provides feedwater at the required temperature, pressure, and flow rate to the steam generators. Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the low-pressure feedwater heaters to the feedwater pumps, and is then pumped through the high-pressure feedwater heaters to the steam generators.

The condensate and feedwater system is composed of components from the condensate system (CDS), main and startup feedwater system (FWS), and steam generator system (SGS). The startup feedwater system is described in Subsection 10.4.9.

10.4.7.1 Design Basis

10.4.7.1.1 Safety Design Basis

The safety-related portion of the system is required to function following a design basis accident (DBA) to provide containment and feedwater isolation, as discussed below, for the main lines routed into containment.

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The portion of the feedwater system from the steam generator inlets outward through the containment up to and including the main feedwater isolation valves (MFIVs) is constructed in accordance with the requirements of ASME Code, Section III for Class 2 components and is designed to seismic Category I requirements. The portion of the feedwater system from the main feedwater isolation valve (MFIV) inlets to the piping restraints at the interface between the auxiliary building and the turbine building is constructed in accordance with the requirements of ASME Code, Section III for Class 3 components and is designed to seismic Category I requirements.

The system provides redundant isolation valves, as described below, for the main feedwater lines routed into containment. The isolation valves close after receipt of an isolation signal in sufficient time to limit the mass and energy release to containment consistent with the containment analysis presented in Chapter 6.

- The safety-related portions of the feedwater system are designed to remain functional after a safe shutdown earthquake (SSE) and to perform their intended function of isolating feedwater flow following postulated events.

- The safety-related portions of the feedwater system are protected from wind and tornado effects, as described in Section 3.3; flood protection is described in Section 3.4; missile protection is described in Section 3.5; protection against dynamic effects associated with the postulated rupture of piping is described in Section 3.6; seismic protection is described in Section 3.7; environmental design is described in Section 3.11; and fire protection is described in Section 9.5.

- The portion of the feedwater system to be constructed in accordance with ASME Code, Section III, Class 2 requirements is provided with access to welds and removable insulation for inservice inspection, in accordance with ASME Code, Section XI. The portion of the feedwater system to be constructed in accordance with ASME Code, Section III, Class 3 requirements is also designed and configured to accommodate inservice inspection in accordance with ASME Code, Section XI.

- The condensate and feedwater system classification is described in Section 3.2. The control functions and power supplies are described in Chapters 7 and 8.

- For a main feedwater or main steam line break (MSLB) inside the containment, the condensate and feedwater system is designed to limit high energy fluid to the broken loop. High energy line break for piping not qualified for leak before break (LBB) criteria is discussed in Subsection 3.6.3.

- Double valve main feedwater isolation is provided via the main feedwater control valve (MFCV) and main feedwater isolation valve (MFIV). Valves fail closed on loss of actuating fluid. Both valves are designed to close automatically on main feedwater isolation signals, an appropriate engineered safety features (ESF) isolation signal, within the time established within the Technical Specifications.

- The MFCVs provide backup isolation to their respective containment isolation valves in order to terminate feedwater flow. The MFCVs are located in the auxiliary building in piping designed to ASME Code, Section III, Class 3 seismic Category I requirements. These valves are components of the steam generator system (SGS).

- For a steam generator tube rupture event, positive and redundant isolation is provided for the main feedwater system (MFIV and MFCV) with ESF isolation signals generated by the protection and safety monitoring system.

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10.4.7.1.2 Power Generation Design Basis

- The condensate and feedwater system provides a continuous feedwater supply to the two steam generators at the required pressures and temperatures for steady-state and anticipated transient conditions.

- Plant operation is possible at 100-percent power with one condensate pump out of service, and approximately 70-percent power with one booster/main feedwater pump assembly out of service.

- Plant operation is possible at greater than 70-percent power with one feedwater heater string out of service.

- The feedwater and condensate pumps and pump control system are designed so that loss of one booster/main feedwater pump assembly or one condensate pump does not result in trip of the turbine-generator or reactor.

- The pumps and other system components are designed so that the condensate, feedwater booster and feedwater pumps are protected from running with very low net positive suction heads without tripping on short transient low levels in a hotwell or deaerator tank.

- The condenser hotwell is designed to store, at the normal operating water level, an amount of condensate equivalent to at least three minutes of full-load condensate system operating flow.

- The system is able to accommodate ten-percent step or five-percent per minute ramp load changes without significant deviation from the programmed water levels in the steam generators or major effect on the feedwater system.

- The system has the capability of accommodating the necessary changes in feedwater flow to the steam generators with the steam pressure increase resulting from a 100-percent load rejection.

- The booster/main feedwater pumps are tripped simultaneously with the feedwater isolation signal to close the main feedwater isolation valves. In addition, the same isolation signal closes the isolation valve in the cross connect line between the main feedwater pump discharge header and the startup feedwater pump discharge header.

- A check valve, which acts on reverse pressure differential, is provided in the main feedwater line to each steam generator between the MFIV and the containment penetration. The check valve is designed to withstand the forces encountered when closing after a main feedwater line rupture. The valves perform no safety-related function but will serve to prevent blowdown from more than one steam generator during feedline break while the appropriate engineered safety features signal is generated to isolate using the MFIV and MFCV. During normal or upset conditions, the function of these check valves is to prevent reverse flow from the steam generators whenever the feedwater system is not in operation.

10.4.7.2 System Description

10.4.7.2.1 General Description

The condensate and feedwater system is shown schematically in Figure 10.4.7-1, and in Figure 10.3.2-1. Classification of equipment and components is given in Section 3.2.

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The condensate and feedwater system supplies the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating. The condensate and feedwater system is composed of the condensate system, the main feedwater system, and portions of the steam generator system. The condensate system collects condensed steam from the condenser and pumps condensate forward to the deaerator. The feedwater system takes suction from the deaerator and pumps feedwater forward to the steam generator system utilizing high-pressure main feedwater pumps. The steam generator system contains the safety-related piping and valves that deliver feedwater to the steam generators. The condensate and feedwater systems are located within the turbine building, and the steam generator system is located within the auxiliary building and containment.

The main portion of the feedwater flow originates from condensate pumped from the main condenser hotwell by the condensate pumps. The main condenser hotwell receives makeup from the condensate storage tank. (Refer to Subsection 9.2.4 for a description of the condensate storage system.) The condensate passes in sequence through: the condensate polishing system or condensate polishing bypass (described in Subsection 10.4.6); the gland steam condenser; three strings of low-pressure heaters, each string consisting of a No. 1 and No. 2 low-pressure heater; two strings of low-pressure heaters No. 3 and No. 4; the No. 5 open low pressure heater (deaerator); the three parallel booster/main feedwater pumps; and two strings of high-pressure heaters, No. 6 and No. 7. Feedwater is pumped to the plant's two steam generators through each generator's respective flow element, control valve, feedwater isolation valve, and check valve. The balance of the plant's feedwater flow is provided by drains from the main steam system moisture separator reheater, drains from the No. 6 and No. 7 feedwater heaters, and steam condensed in the deaerator. These flows are collected in the deaerator and pumped forward in the feedwater cycle. A portion of the condensate flow downstream of the condensate polishers is diverted to provide cooling to the steam generator blowdown system heat exchangers before returning to the main condensate flow at the deaerator.

During plant startup, three recirculation paths facilitate system cleanup and adjustment of water quality prior to initiating feed to the steam generators. These cleanup loops are designed for approximately 33 percent of design condensate flow and include a hotwell recirculation loop, a deaerator recirculation loop, and a third recirculation loop from downstream of the No. 7 feedwater heaters. Steam is provided to the deaerating feedwater heater from the auxiliary steam supply system to preheat the feedwater as feedwater begins to flow towards the steam generators. This preheating action, along with chemical addition, minimizes formation of iron oxides in the condensate system.

The condensate polishing system is described in Subsection 10.4.6 and may be in service or bypassed. Each of the two main feedwater lines to the two steam generators contains a feedwater flow element, a main feedwater control valve, a main feedwater isolation valve, and a check valve. A thermal relief valve is included in each main feedwater line between the control valve and isolation valve to prevent overpressure due to thermal expansion when both valves are closed.

The turbine island chemical feed system (CFS) described in Subsection 10.4.11 is provided to inject an oxygen scavenging agent and a pH control agent into the condensate pump discharge downstream of the condensate polishers and an oxygen scavenging agent and pH control agent upstream of the feedwater booster pump suction. Injection points are shown in Figure 10.4.7-1. During normal power operation, the addition of an oxygen scavenging agent and pH control agent to the condensate system downstream of the condensate demineralizers is in automatic control, with manual control available. The added chemicals control pH according to the condensate and feedwater system chemistry requirements and establish an oxygen scavenging agent residual in the feedwater system. The oxygen scavenger agent is hydrazine and the pH control agent is methoxypropylamine (MPA). During shutdown conditions, carbohydrazide may be used in place of hydrazine.

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Oxygen scavenging and ammoniating agents are selected and utilized for plant secondary water chemistry optimization following the guidance of NEI-97-06, "Steam Generator Program Guidelines" (Reference 201). The EPRI Pressurized Water Reactor Secondary Water Chemistry Guidelines are followed as described in NEI 97-06.

A cross connection from the main feedwater pump discharge header to the startup feedwater header allows any booster/main feedwater pump to supply feedwater to the startup feedwater control valves. The startup feedwater system is described in Subsection 10.4.9. Thus, feedwater from the deaerator storage tank can be supplied by the booster/main feedwater pumps through the startup feedwater connections to the steam generators during hot standby, plant startup and low power operation. A check valve in the cross connection piping prevents the startup feedwater pumps from supplying the main feedwater header, and a nonsafety-related isolation valve in the cross connection piping automatically closes upon the feedwater isolation signal that trips the main feedwater pumps.

A condensate and feedwater failure analysis for safety-related components is presented in Table 10.4.7-1. Occurrences which produce an increase in feedwater flow or decrease in feedwater temperature result in increased heat removal from the reactor coolant system which is compensated for by control system action, as described in Subsection 10.4.7.5. Events which produce the opposite effect (i.e., decreased feedwater flow or increased feedwater temperature) result in reduced heat transfer in the steam generators. Normally, automatic control system action is available to adjust feedwater flow to prevent excess energy accumulation in the reactor coolant system, and the increasing reactor coolant temperature provides a negative reactivity feedback, reducing reactor power. In the absence of normal control action, either the high-outlet temperature or the high-pressure trips of the reactor protection system are available to provide reactor safety. Loss of all feedwater is examined in Section 15.3.

Refer to Subsection 5.4.2.2 for a description of steam generator design features to prevent fluid flow water hammer. The main feedwater connection on each of the steam generators is the highest point of each feedwater line downstream of the MFIV. The feedwater lines contain no high-point pockets that could trap steam and lead to water hammer. The horizontal pipe length from the main nozzle to the downward turning elbow of each steam generator is minimized.

Operations and maintenance procedures include precautions, when appropriate, to minimize the potential for steam and water hammer, including:

- Prevention of rapid valve motion
- Process for avoiding introduction of voids into water-filled lines and components
- Proper filling and venting of water-filled lines and components
- Process for avoiding introduction of steam or heated water that can flash into water-filled lines and components
- Cautions for introduction of water into steam-filled lines or components
- Proper warmup of steam-filled lines
- Proper drainage of steam-filled lines
- The effects of valve alignments on line conditions

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10.4.7.2.2 Component Description

The feedwater system is constructed in accordance with the requirements of ASME Code, Section III for Class 2 components and seismic Category I requirements from the steam generator out through the MFIVs. From upstream of the MFIV to the restraint at the interface between the auxiliary building and turbine building, the system is constructed in accordance with ASME Code, Section III for Class 3 components and seismic Category I requirements. The remaining piping of the condensate and feedwater system meets ASME B31.1 requirements. Safety-related feedwater piping materials are described in Subsection 10.3.6.

Feedwater Piping

Feedwater is supplied to each of the two steam generators by a main feedwater line during normal operation. Each of the lines is anchored at the auxiliary building/turbine building interface, and has sufficient flexibility to provide for relative movement of the steam generators resulting from thermal expansion.

The feedwater system and steam generator design minimize the potential for waterhammer and subsequent effects. Details are provided in Subsection 5.4.2.2. Feedwater piping analysis considers the following factors and events in the evaluation:

- Steam generators with top feed ring design (BTP ASB 10-2)
- Main feedwater check valves due to line breaks (BTP MEB 3-1)
- Spurious isolation or feedwater control valve trips
- Pump trips
- Deaerator regulating flow control valve trip
- Local feedwater piping, anchors, supports, and snubbers, as applicable

Feedwater Isolation Valves

One MFIV is installed in each of the two main feedwater lines outside the containment and downstream of the feedwater control valve. The MFIVs are installed to prevent uncontrolled blowdown from the steam generators in the event of a feedwater pipe rupture. The main feedwater check valve provides backup isolation. In the event of a secondary side pipe rupture inside the containment, the MFIVs limit the quantity of high energy fluid that enters the containment through the broken loop and limit cooldown. The MFCV provides backup isolation to limit cooldown and high energy fluid addition.

Each MFIV is a bidirectional wedge type gate valve composed of a valve body that is welded into the system pipeline. The MFIV gate valve is provided with a hydraulic/pneumatic actuator. The valve actuator is supported by the yoke, which is attached to the top of the body. The valve actuator consists of a hydraulic cylinder with a stored energy system to provide emergency closure of the isolation valve. The energy to operate the valve is stored in the form of compressed nitrogen contained in one end of the actuator cylinder. The MFIV is maintained in a normally open position by high-pressure hydraulic fluid. For emergency closure, redundant solenoids are energized resulting in the high-pressure hydraulic fluid being dumped to a fluid reservoir.

The feedwater isolation functional diagram is shown in Figure 7.2-1. To provide safety function actuation, the redundant actuation solenoid valves are powered from separate Class 1E power divisions. Redundant control and indication channels are provided for each of the isolation valves. Provisions are made for inservice inspection of the isolation valves.

Feedwater Control Valves

The MFCVs are air-operated control valves with the dual purpose of controlling feedwater flow rate as well as providing backup isolation of the feedwater system. The valve body is a globe design.

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Seats and trim are of an erosion resistant material. The design allows for removal and replacement of seats and other wearing parts.

The feedwater control valves (MFCVs) automatically maintain the water level in the steam generators during operational modes. Positioning of the main feedwater control valve during normal operation is the function of an automatic feedwater level control system using a refinement of a standard three element control scheme. The three-element control system maintains feedwater flow equal to the steam flow, and steam generator water level is used as an input to trim feedwater flow and maintain programmed water level. Refinements on the standard control are made by varying the flow demand of the valve based on the actual stem position.

In the event of a secondary side pipe rupture inside the containment, the main feedwater control valves provide a redundant isolation to the MFIVs to limit the quantity of high energy fluid that enters the containment through the broken loop. For emergency closure of the MFCV, solenoids are deenergized to close the valve in sufficient time to limit the mass and energy release to containment consistent with the containment analysis presented in Chapter 6.

Feedwater Check Valves

Each main feedwater line includes a check valve installed outside containment. During normal and upset conditions, the check valve prevents reverse flow from the steam generator whenever the feedwater pumps are tripped. In addition, the closure of the valves prevents more than one steam generator from blowing down in the event of feedwater pipe rupture. The check valve is designed to limit blowdown from the steam generator and to prevent slam resulting in potentially severe pressure surges due to water hammer. The valves are designed to withstand the closure forces encountered during the normal, upset and faulted conditions. Rapid closure associated with a feedline rupture does not impose unacceptable loads on the steam generator or the steam generator system. The closure of the valves provides for isolation of the steam generators in the event of a feedwater line break to prevent blowdown from both steam generators. The valves are seismic Category I, ASME Code, Section III, Class 2 valves.

Feedwater Thermal Relief Valves

A relief valve is provided to prevent overpressurization of the main feedwater lines between the MFIVs and MFCVs. The relief valve prevents overpressurization that could be caused by thermal expansion of the fluid trapped between the isolation and control valves following an event causing main feedwater isolation.

Plant Main Condenser

For a description of main condenser, refer to Subsection 10.4.1.

Condensate Pumps

The three 50-percent, vertical, multistage, centrifugal condensate pumps are motor-driven and operate in parallel. Valving allows individual pumps to be removed from service. Pump capacity meets normal, full-power requirements with two of the three pumps in operation.

Condensate Regulating Valves

The main condensate flow to the deaerator is regulated by two parallel, split-ranged, pneumatically operated control valves. Condensate is regulated to maintain the level in the deaerator storage tank. During startup and low loads, the smaller valve modulates to control flow while the larger valve remains closed. As load increases, the larger valve modulates to control flow.

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Low-Pressure Feedwater Heaters

These heaters are shell and tube heat exchangers with the heated condensate flowing through the tube side and the extraction steam condensing on the shell side. Parallel strings of low-pressure feedwater heaters No. 1 and 2 are located in each of three condenser necks. Feedwater heaters No. 3 and 4 are also parallel strings of heaters. The closed low-pressure feedwater heaters use drain coolers. The cascaded drains from the heaters are dumped to their respective condenser shell.

A drain line from the heater allows direct discharge of the heater drains to the condenser in the event the normal drain path is not available or flooding occurs in the heater.

The low-pressure feedwater heater shells are carbon steel, and the tubes are stainless steel.

Deaerator

The deaerator is a tray type, horizontal shell, direct contact heater with a horizontal storage tank. Condensate enters the deaerator from the top. The heating steam flows upward and is condensed, and raises the temperature of the condensate to near saturation, liberating dissolved gases from the condensate. Condensate drains from the deaerator into the storage tank. Noncondensables are vented from the top of the deaerator and flow through an orifice and valve assembly to the main condenser.

Auxiliary steam from the auxiliary steam supply system (see Subsection 10.4.10) is supplied to the deaerator (as pegging steam) following the recirculation and cleanup steps and maintains the pressure in the tank at saturation conditions. As main steam becomes available to supplement the auxiliary boiler steam, this pegging steam keeps the deaerator above atmospheric pressure. This pressure is maintained as condensate and feedwater flow (and also reactor power and main steam flow) increases to and beyond the point of turbine synchronization, until the normal heating steam flow from the HP turbine exhaust is sufficient to maintain deaerator pressure. Auxiliary steam is also automatically supplied to the deaerator following turbine trip to assist in maintaining deaerator pressure above atmospheric.

The shell of the deaerator storage tank is carbon steel. Most of the internals of the deaerator, including the tray assemblies, vent condenser, and spray valves, are stainless steel.

A high level dump line and control valve provide overflow protection to the deaerator storage tank. During high level conditions, water from the deaerator storage tank is drained to the main condenser.

High-Pressure Feedwater Heaters

The main feedwater pumps discharge into a parallel string of No. 6 and No. 7 high-pressure feedwater heaters. These heaters are shell and tube heat exchangers. Heated feedwater flows through the tubes and extraction steam condenses in the shell. The No. 6 and No. 7 heaters drain into low-pressure heater No. 5 (deaerator).

A drain line from each heater allows direct discharge of the heater drains to the condenser in the event the normal drain path is not available or flooding occurs in the heater.

The high-pressure feedwater heater shells are carbon steel, and the tubes are stainless steel.

Feedwater Booster Pumps

The feedwater booster pumps are horizontal, centrifugal pumps located upstream of the main feedwater pumps. Each feedwater booster pump takes suction from the deaerator storage tank and pumps forward to its associated main feedwater pump. An electric motor drives both the booster pump and the main feedwater pump. The booster pump is driven by one end of the motor shaft and

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the main pump is driven by the other end through a mechanical speed increaser. The booster pump, operating at a lower speed than the main feedwater pump, boosts the pressure of feedwater from the deaerator to meet the net positive suction head requirements of the main feedwater pump.

Main Feedwater Pumps

The three main feedwater pumps operate in parallel and take suction from the associated feedwater booster pumps. The combined discharge from the main feedwater pumps is supplied to the No. 6 high-pressure feedwater heater, the No. 7 high-pressure feedwater heater, and then to the steam generator system. Each main feedwater pump is a horizontal, centrifugal pump driven, through a mechanical speed increaser, by the motor that drives the associated feedwater booster pump.

Isolation valves allow each of the booster/main feedwater pumps to be individually removed from service while continuing power operations at reduced capacity.

Pump Recirculation Systems

Minimum flow control systems automatically protect the pumps in the condensate and feedwater system from pumping below the minimum flow rate to prevent pump damage. The condensate pumps recirculate to the main condenser. The booster/main feedwater pumps recirculate to the deaerator storage tank.

10.4.7.2.3 System Operation

10.4.7.2.3.1 Plant Startup

During plant startup, the condensate and feedwater system operates in several different configurations. These are described in Subsections 10.4.7.2.3.1.1 through 10.4.7.2.3.1.4.

10.4.7.2.3.1.1 Hotwell Recirculation

The hotwell recirculation flow path is used to recirculate flow from downstream of the gland steam condenser to the main condenser to facilitate cleanup of the condensate inventory in the main condenser hotwell. This flow path also provides a minimum flow for operation of the gland steam condenser and the condensate pumps. With a condensate pump operating, the setpoint of the recirculation valve is manually adjusted to achieve the desired flow rate for cleanup of condensate. Condensate polishing equipment is aligned and placed in service to attain the required water quality.

The hotwell recirculation valve is placed in automatic operation when minimum flow is required only for operation of the gland steam condenser and one or two condensate pumps. The recirculation valve automatically maintains the minimum flow and closes when system flow to the deaerator exceeds the required minimum. The recirculation valve remains on standby and opens, as necessary, if system flow drops below minimum.

Once the hotwell recirculation loop is placed in service and cooling is available to the gland steam condenser, sealing steam may be applied to the turbine glands. Condenser vacuum can then be drawn using condenser air removal equipment.

10.4.7.2.3.1.2 Deaerator Recirculation

The deaerator recirculation flow path is used to recirculate condensate from downstream of the deaerator storage tank to the main condenser to facilitate cleanup of condensate. Deaerator recirculation is initiated by adjusting the recirculation flow control valve from the main control room to achieve the desired flow rate. Condensate is recirculated for cleanup of water quality using the

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condensate polishing equipment. Auxiliary steam can be admitted to the deaerator to facilitate removal of noncondensable gases.

10.4.7.2.3.1.3 Third Stage Recirculation

The third stage of condensate/feedwater recirculation during the plant heatup cycle can begin when condensate and feedwater has been sufficiently cleaned and deaerated at the feedpump suction. Flow is initiated by adjusting the recirculation flow control valve from the main control room to achieve the desired flow rate. Feedwater is recirculated from downstream of the No. 7 feedwater heaters to the main condenser for cleanup and deaeration of the condensate and feedwater inventory.

10.4.7.2.3.1.4 Plant Heatup

The condenser hotwell makeup and overflow valves are enabled and function automatically during the plant heatup cycle to maintain condensate inventory. Condensate is returned to the condensate storage tank as volume expansion occurs, and makeup occurs as needed for system losses.

During heatup, the main condenser is available to accept turbine bypass steam from the main steam system, as well as various drains, vents, and condensate/feedwater recirculation flow. Noncondensable gases are removed in the air removal sections of the main condenser and through the deaerator vents. Control and monitoring of water quality and chemistry are accomplished by operation of the condensate polishing equipment, chemical feed system, and secondary sampling equipment as required.

The steam generators are filled, as required, either by the startup feedwater pumps using water from the condensate storage tank, or alternatively by a booster/main feedwater pump using water from the deaerator storage tank and supplied through cross connect piping to the startup feedwater control valves. The steam generators are drained, as required, through the steam generator blowdown system.

During the initial stages of plant heatup, one condensate pump operates as necessary to maintain level in the deaerator storage tank. Either one or both startup feedwater pumps, or one booster/main feedwater pump, is in operation when feeding water to the steam generators. The feedwater pumps in use operate on minimum flow recirculation as necessary while maintaining the water level of the steam generators. The pump heat from this recirculated minimum flow, in conjunction with auxiliary steam, is used to heat the deaerator.

Feedwater is controlled by the startup feedwater control valves (SFCVs) which are operated either manually from the control room or automatically in accordance with steam generator level demand. Condensate flow to the steam generator blowdown heat exchangers is controlled during plant heatup to obtain the necessary cooling to the blowdown stream. Any excess level in the deaerator storage tank is drained to the main condenser through the deaerator high level dump flow path.

10.4.7.2.3.2 Power Operation

One operating condensate pump supplies sufficient condensate flow to the deaerator during initial power operation and at low-power levels. As power escalates, a second condensate pump is started prior to exceeding approximately 50-percent, full-load condensate flow. The third condensate pump is in standby.

The condensate regulating valves to the deaerator automatically maintain the level of the deaerator storage tank. If condensate flow to the deaerator drops below the minimum required flow for operation of the gland steam condenser or the condensate pumps, the hotwell recirculation valve to the condenser opens to provide the minimum flow.

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Noncondensables are removed by the deaerating section of the main condenser and by the deaerator. Condensate polishing, chemical feed and condensate sampling are performed, as needed, to maintain water quality.

For normal operating conditions between 0- and 100-percent load, system operation is primarily automatic. Automatic level control systems control the water levels in the feedwater heaters and the condenser hotwell. Feedwater heater water levels are controlled by modulating flow control valves. Level control valves in the makeup line to the condenser from the condensate storage tank and in the return line to the condensate storage tank control the level in the condenser hotwell.

During reactor startup and at very low power levels, feedwater is supplied to the steam generators through the startup feedwater control valves using either the startup feedwater pumps drawing from the condensate storage tank, or a booster/main feedwater pump drawing from the deaerator storage tank. Refer to Subsection 10.4.9 for a description of the startup feedwater system. If the startup feedwater pumps are initially in use, transfer is made to a booster/main feedwater pump prior to exceeding the capacity limit of the startup pumps. As power increases, startup feedwater continues to be supplied through the startup feedwater control valves until control of feedwater is automatically transferred from the startup feedwater control valves to the main feedwater control valves. The startup feedwater control valves close, and the main feedwater control valves open to supply main feedwater to the steam generators and maintain steam generator level. Position indication is available in the main control room for the main and startup feedwater control valves. As power escalates, booster/main feedwater pump minimum flow recirculation automatically decreases as the forward flow to the steam generators increases. The second and third booster/main feedwater pumps are brought into operation as required.

Condensate flow to the steam generator blowdown heat exchangers is normally automatically controlled. In the automatic mode, condensate flow is regulated to control the steam generator blowdown outlet temperature from the blowdown heat exchangers.

Ten-percent step load and 5-percent/minute ramp changes are accommodated without major effect to the condensate and feedwater system. The system is capable of providing the necessary feedwater flow to the steam generators with the steam pressure increase resulting from a 100-percent load rejection.

10.4.7.2.3.3 Plant Shutdown

Operation during power descent is largely the reverse of power ascent. As power is decreased, one of the two operating condensate pumps may be stopped; one or two booster/main feedwater pumps may be stopped as well. At low feedwater flow, control of feedwater is automatically transferred from the main feedwater control valves to the startup feedwater control valves.

Following reactor trip or other reactor shutdown, feedwater is supplied through the startup feedwater control valves to maintain steam generator inventories. Decay heat and sensible heat are removed by steam release via the steam dump system to the condenser to cool the plant and bring it to safe shutdown. During this time, startup feedwater is supplied either by an operating booster/main feedwater pump drawing from the deaerator storage tank, or by the startup feedwater pumps drawing from the condensate storage tank.

10.4.7.2.3.4 Emergency Operation

In the event of a design basis event (with or without normal ac power supplies available), feedwater isolation signals are generated as required. The MFIVs and MFCVs automatically close on receipt of the isolation signals. The condensate and feedwater system is not required to supply feedwater under accident conditions to effect plant shutdown or to mitigate the consequences of an accident.

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However, the startup feedwater system is expected to be available as a nonsafety-related system to provide a source of feedwater for the steam generators. Also, the condenser may be available to accept turbine bypass steam for secondary side heat removal. Coordinated operation of the startup feedwater system (Refer to Subsection 10.4.9), if available, and the main steam supply system (Refer to Section 10.3) removes the primary loop sensible heat and reactor decay heat.

10.4.7.3 Safety Evaluation

- The safety-related portions of the main feedwater system are located in the containment and auxiliary buildings. These buildings are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other natural phenomena. Sections 3.3, 3.4, 3.5, 3.7, and 3.8 provide the bases for the adequacy of the structural design of these buildings.

- The safety-related portions of the main feedwater system are designed to remain functional after a design basis earthquake. Subsection 3.7.2 and Section 3.9 provide the design loading conditions that are considered. Sections 3.5, 3.6, and Subsection 9.5.1 describe the analyses to provide confidence that a safe shutdown, as outlined in Section 7.4, is achieved and maintained.

- The main feedwater system safety-related functions are accomplished by redundant means. A single, active component failure of the safety-related portion of the system does not compromise the safety function of the system. Table 10.4.7-1 provides a failure analysis of the safety-related active components of the feedwater system. Power is supplied from onsite power systems, as described in Chapter 8.

- Preoperational testing of the safety-related portion of the condensate and feedwater system is performed as described in Chapter 14. Periodic inservice functional testing is done in accordance with Subsection 10.4.7.4. Section 6.6 provides the ASME Code, Section XI requirements that are appropriate for the feedwater system.

- Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system and supporting systems. The controls and power supplies necessary for the safety-related functions of the condensate and feedwater system are Class 1E, and are described in Chapters 7 and 8.

- For a main feedwater line break inside the containment or a main steam line break, the MFIVs and the main feedwater control valves automatically close upon receipt of a feedwater isolation signal. The signals that produce a main feedwater isolation signal are identified and discussed in Subsection 7.3.1.2.6.

- The MFIVs are provided with solenoids supplied by redundant power divisions. Failure of either of the power divisions or solenoids does not prevent closure of the MFIV. Releases of radioactivity from the condensate and feedwater system, resulting from the main feedwater line break, are minimal because of the negligible amount of radioactivity in the system under normal operating conditions. Following a steam generator tube rupture, the main steam isolation system and the passive residual heat removal heat exchanger reduce accidental releases, as discussed in Section 10.3 and Chapter 15. Detection of radioactive leakage into and out of the system is facilitated by area radiation monitoring (described in Subsection 12.3.4), process radiation monitoring (described in Section 11.5), and steam generator blowdown sampling (described in Subsection 10.4.8).

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- For a steam generator tube rupture event, positive and redundant isolation is provided for the main feedwater (MFIV and MFCV) with isolation signals generated by the protection and safety monitoring system (PMS). Refer to Subsection 7.3.1.2.6.

- Prevention and mitigation of feedline-related water hammer is accomplished through operation of the feedwater delivery system as described in Subsection 5.4.2.2. The feedwater piping at the steam generators is sloped so that it does not drain into the steam generators. These features help avoid the formation of a steam pocket in the feedwater piping which, when collapsed, could create a hydraulic instability.

10.4.7.4 Tests and Inspections

10.4.7.4.1 Preoperational Valve Testing

The MFIVs and feedwater control valves are checked for closing time prior to initial startup.

10.4.7.4.2 Preoperational Pipe Testing

The main feedwater lines from the steam generator to the anchor at the interface between the turbine building and the auxiliary building are classified as ASME Code, Section III, Class 2 and 3 and seismic Category I piping. The Class 2 portions of the main feedwater system piping are tested and inspected to the requirements of ASME Code, Section III, Class 2 piping. The portion of the piping between the containment penetration and the anchor, which is considered as the break exclusion zone described in Subsection 3.6.2, is subjected to 100-percent volumetric inspection at installation.

10.4.7.4.3 Preoperational System Testing

Preoperational testing of the condensate and feedwater system is performed as described in Chapter 14. Tests described in Subsection 14.2.9.1.7, under item c) of General Test Method and Acceptance Criteria satisfy BTP (AS) 10-2. Additional testing of the feedwater system is conducted during startup testing as described in Subsection 14.2.10.4.18.

10.4.7.4.4 Inservice Inspections

The performance, and structural and leaktight integrity of the condensate and feedwater system components are demonstrated by continuous operation.

Additional description of inservice testing and inspection for the MFIV and MFCV is presented in Subsection 3.9.6 and Section 6.6.

10.4.7.5 Instrumentation Applications

The condensate and feedwater instrumentation, is designed to facilitate automatic operation, remote control, and indication of system parameters.

Positioning of the main feedwater control valve during normal operation is the function of an automatic feedwater level control system using a refinement of a standard three element control scheme. For each steam generator, the three-element control system maintains feedwater flow equal to the steam flow, and steam generator water level is used as an input to trim feedwater flow and maintain programmed water level. Refinements on the standard control are made by varying the valve flow demand based on actual stem position (accounting for varying C_v versus lift) dynamic line losses and feedwater temperature. A flow venturi is located in each feedwater line to provide signals for the three element feedwater control system. Feedwater control is further described in Subsection 7.7.1.8.

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The main feedwater pumps are tripped by manual actuation or feedwater isolation described in Section 7.3. A flow element in the discharge piping from each main feedwater pump provides a flow signal for control of the associated minimum flow recirculation valve.

Level transmitters, located at the deaerator storage tank, control deaerator level. Condensate flow to the deaerator is regulated by two split ranged control valves upstream of the deaerator. During normal power generation, the valves are regulated by a three element control system; total feedwater flow is used as a feed forward demand signal, and the control is trimmed by measured feedback of total condensate flow and deaerator storage tank level.

In the event a feedwater heater experiences a sizable tube leak or a feedwater heater water level control valve fails closed, the main turbine is protected from failure resulting from flooding on the shell side of a feedwater heater and subsequent water induction into the moving turbine blades. This is accomplished by automatic closure of the isolation valve in the steam extraction line to that heater and opening the high-level dump control valve that dumps the heater excess drains to the condenser. For heaters that do not have extraction line isolation valves, condensate isolation valves are automatically closed to isolate condensate flow to the heater tubes.

The total water volume in the condensate and feedwater system is maintained through automatic makeup and rejection of condensate to the condensate storage tank. The system makeup and rejection are controlled by the condenser hotwell level controller. Level transmitters are provided at the condenser hotwell for use by the hotwell level controller. The system water quality requirements are automatically maintained through the injection of an oxygen scavenging agent and a pH control agent into the condensate system. The pH control agent and oxygen scavenging agent injection is controlled by pH and the level of oxygen scavenging agent residual in the system which are continuously monitored by the secondary sampling system.

Instrumentation, including pressure indication, flow indication, and temperature indication, required for monitoring the system, is provided in the control room.

10.4.8 Steam Generator Blowdown System

The steam generator blowdown system (BDS) assists in maintaining acceptable secondary coolant water chemistry during normal operation and during anticipated operational occurrences of main condenser inleakage or primary to secondary steam generator tube leakage. It does this by removing impurities which are concentrated in the steam generator. The steam generator blowdown system accepts water from each steam generator and processes the water as required.

10.4.8.1 Design Basis

Appendix 1A discusses the conformance of BDS design with the criteria of Regulatory Guide 1.143. Exceptions for treatment of the BDS in comparison with the radwaste systems is consistent with the regulatory philosophy of treating structures, systems, and components commensurate with their importance to safety since the BDS does not normally contain radioactivity and only contains radioactivity in the infrequent event of primary to secondary tube leakage.

10.4.8.1.1 Safety-Related Design Basis

The safety-related portion of each blowdown line is part of the steam generator system (SGS). Effects of a blowdown system line break are discussed in Section 3.6. The safety-related design bases are as follows:

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- The system is provided with two isolation valves on each steam generator. These valves isolate the secondary side of the steam generators to preserve the steam generator inventory. This action provides a heat sink for a safe shutdown or design basis accident mitigation. It also provides isolation of nonsafety-related portions of the system.

- The steam generator blowdown system safety-related functions can be performed assuming a single, active component failure coincident with the loss-of-offsite or onsite power.

- Piping and valves from the steam generator up to and including the containment isolation valve, the first valve on the outboard side of the containment, are designed to ASME Code, Section III, Class 2, and seismic Category I requirements. The blowdown system piping and valves from the outlet of the containment isolation valve up to and including pipe anchors located at the auxiliary building wall are designed in accordance with ASME Code, Section III, Class 3, and seismic Category I requirements.

- The safety-related portion of the system is designed to withstand the effects of a safe shutdown earthquake. The safety-related portion of the system is protected from the effects of natural phenomena and is capable of performing its intended function following postulated events such as fire, internal missile, and pipe break.

- The safety-related portion of the system is designed so that a single, active failure in the blowdown system will not result in:

  – Loss-of-coolant accident
  – Loss of integrity of steam lines
  – Loss of the capability to effect a safe reactor shutdown
  – Transmission of excessive loading to the containment pressure boundary.

- The portion of the steam generator system that is constructed in accordance with ASME Code, Section III, Class 2 and 3, requirements is provided with access to welds and removable insulation, as required for inservice inspection in accordance with ASME Code, Section XI. (See Subsection 10.4.8.4.)

- The safety-related portion of the blowdown system is designed to function in the normal and accident environments identified in Subsection 3.11.1.

- The safety-related portion of the blowdown system is designed as described in Section 3.6 with regard to high-energy pipe break location and evaluation.

10.4.8.1.2 Power Generation Design Basis

The steam generator blowdown system draws secondary water from each steam generator via the blowdown line and processes this water as required to:

- Assist in controlling steam generator secondary side water chemistry during normal plant operation

- Establish and maintain steam generator wet layup conditions during plant shutdown periods

- Drain the secondary side of the steam generators for maintenance

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10.4.8.2 System Description

10.4.8.2.1 General Description

Figures 10.4.8-1 and 10.3.2-1 illustrate the steam generator blowdown system piping and instrumentation design. Classification of equipment and components for the steam generator blowdown system is given in Section 3.2. The system consists of two blowdown trains, one for each steam generator. A crosstie is provided to process blowdown from both steam generators through both heat exchangers during high capacity blowdown from one steam generator.

The blowdown water is extracted from each steam generator through holes in the top surface (secondary side) of the tube sheet. The blowdown from each steam generator is cooled by a regenerative heat exchanger, and flow is controlled and pressure reduced by blowdown flow control valves. To recover the thermal energy, the condensate system provides cooling for the heat exchangers. To recover the blowdown fluid, each blowdown train has an electrodeionization (EDI) demineralizing unit which removes impurities from the blowdown flow. Downstream of the electrodeionization units, both trains combine into a common header that contains a relief valve for providing overpressure protection for the low-pressure portion of the system. A back-pressure control valve maintains pressure in the system between the flow control valve and the back-pressure control valve.

A pump is provided to drain the secondary side of the steam generator. The pump is also used for recirculation during low-pressure steam generator wet layup operations.

System isolation from the steam generator under normal operating and transient conditions is accomplished either by the two isolation valves located in the auxiliary building or by isolation valves located downstream of the heat exchangers and sampling connections. The upstream valves in the auxiliary building close on actuation of the passive residual heat removal system or containment isolation. The isolation valves downstream of the heat exchangers close in the case of high blowdown system temperature flow, radiation, or pressure.

10.4.8.2.2 System Operation

The various modes of operation are described in the following subsections.

10.4.8.2.2.1 Plant Startup

While low-pressure conditions exist in the steam generator, the blowdown flow control valves are operated with their bypass valves fully open to discharge the blowdown flow to the condensate system (CDS) for processing and recovery.

When the steam generator pressure is sufficiently high, the bypass valves can be closed and the blowdown flow control valves modulate to control the blowdown rate. The condensate control valves, which control the supply of cooling water to the heat exchangers, are adjusted during startup. The cooling water flow to the heat exchangers controls blowdown water to a temperature that is acceptable to the blowdown system electrodeionization units.

10.4.8.2.2.2 Normal Operation

The effectiveness of the blowdown system in controlling water chemistry depends upon the blowdown rate. The normal blowdown flowrate varies from a minimum of about 0.06 percent to a maximum of about 0.6 percent of maximum steaming rate. During normal operation, when the impurities are low, the expected blowdown rate is approximately 0.1 percent of maximum steaming rate (about 30 gpm total, or 15 gpm per steam generator), which maximizes the detection sensitivity

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for condenser tube leakage. The blowdown flow is cooled by the heat exchanger, and the pressure is reduced by the flow control valves. The blowdown fluid is processed through the electrodeionization units and discharged to the condensate system (condenser hotwell) for reuse.

In the event of main condenser tube leakage, when the concentration of impurities is high, the blowdown rate is increased to a maximum of approximately 0.6 percent of the maximum steaming rate (about 186 gpm total, or 93 gpm per steam generator at standard conditions). Normal operation is to recover the blowdown flow through the condensate system. However, blowdown with high levels of impurities can be discharged to the waste water system.

The back-pressure control valve is preset to a pressure which prevents flashing of the blowdown fluid in the electrodeionization units.

The blowdown discharge flow and the electrodeionization reject (waste stream) flow are both continuously monitored for radioactivity from steam generator primary to secondary tube leakage. If such radioactivity is detected, the liquid radwaste system (WLS) is aligned to process the blowdown and electrodeionization waste effluent. If radioactivity reaches a preset high level, the blowdown flow control valves and the isolation valves downstream of the heat exchangers and sampling connections automatically close.

The system operates normally under automatic control, except for flow control adjustments or flow path changes.

10.4.8.2.2.3 Subsection Not Used

10.4.8.2.2.4 Steam Generator Wet Layup

The system can be operated to establish and maintain wet layup conditions in the steam generators during plant shutdown periods. During wet layup operation, water is circulated through the steam generators and the heat exchangers are not required. To maintain the correct pH and oxygen concentration in the secondary water, chemicals are added to the recirculation flow via the turbine island chemical feed system (CFS). (See Subsection 10.4.11 for chemical feed system details.)

10.4.8.2.2.5 Steam Generator Drain

The steam generator blowdown system can be operated to drain the steam generator using the recirculation/drain pump and bypassing the flow control valves and the electrodeionization units. Total drain flowrate is approximately 200 gpm at standard conditions. During this mode of operation, the blowdown discharge maybe sent to the waste water system, the liquid radwaste system or the condensate system.

10.4.8.2.2.6 Steam Generator Tube Sheet Flush

The system can be operated for a short time at a total flowrate of approximately 1.85 percent of the maximum steaming rate (about 280 gpm) from one steam generator. To accommodate the high flow, the blowdown from one steam generator is isolated and the flow from the other steam generator is routed through both heat exchanger trains at a rate of approximately 140 gpm per train. The blowdown flow control valves and the blowdown electrodeionization units are bypassed during this operation. The blowdown flow is controlled by throttling the flow control valve bypass isolation valves which are in series with a flow restricting orifice. The blowdown is discharged to the waste water system (WWS).

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10.4.8.2.2.7 Emergency Operation

Blowdown system isolation is actuated on Low-2 steam generator narrow range water levels. The isolation of steam generator blowdown provides for a continued availability of the steam generator as a heat sink for decay heat removal in conjunction with operation of the passive residual heat removal system and the startup feedwater system.

10.4.8.2.3 Component Description

A description of the major steam generator blowdown system components is provided in this subsection.

10.4.8.2.3.1 Blowdown Regenerative Heat Exchangers

Two regenerative heat exchangers are provided, one for each steam generator blowdown train. The heat exchangers are located in the turbine building at the base slab elevation.

10.4.8.2.3.2 Blowdown Flow Control Valves

Two blowdown flow control valves are provided, one for each steam generator blowdown train. The control valves are capable of controlling the flow and pressure over the range of normal operating conditions.

10.4.8.2.3.3 Recirculation/Drain Pump

One centrifugal pump is provided for use during operating modes when steam generator pressure is low.

10.4.8.2.3.4 Pressure Control Valve

A backpressure control valve is provided to maintain appropriate system backpressure, within the operating range of blowdown flows, and prevent flashing within the low pressure section of the system when the blowdown is discharged to the condenser hotwell.

10.4.8.2.3.5 Blowdown Isolation Valves

Two valves in series, located outside containment in the auxiliary building, are provided to automatically isolate the blowdown system in the event of abnormal conditions within the blowdown system, the reactor coolant system, or the main steam system. The valves are air-operated globe valves that fail close on loss of air or actuating power. See Section 7.3 for a description of the automatic control functions on the valves.

The first isolation valve provides a containment isolation function in addition to redundant isolation of the blowdown system. The valves close on an engineered safeguards actuation signal and provide containment integrity in conjunction with the steam generator and blowdown line inside containment. The valves are active, ASME Code, Section III, Safety Class 2, seismic Category I.

The isolation valves provide for redundant isolation of the blowdown system upon actuation of the passive residual heat removal system, Low-2 steam generator narrow range water level, or abnormal conditions in the blowdown system. Each isolation valve receives an actuation signal from the protection and safety monitoring system (PMS) upon passive residual heat removal actuation to preserve steam generator inventory. The valves also close upon receiving a Low-2 steam generator narrow range water level signal to preserve steam generator inventory. The second isolation valves are active, ASME Code, Section III, Safety Class 3, seismic Category I.

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The valves are located outside containment within the auxiliary building and are attached to seismic Category I piping.

Additionally, Class E, non-seismic, isolation valves downstream of the heat exchangers and sampling connections in the turbine building receive high radiation signal, high temperature signal, high flow signal, and high pressure signal, indicating abnormal conditions in the blowdown system and actuating automatic isolation of the system.

10.4.8.2.3.6 Electrodeionization Unit

Two trains of electrodeionization demineralizing units are provided for the steam generator blowdown system electrodeionization. The electrodeionization unit in each train is configured in a stack arrangement. The stack normally contains numerous pairs of stacked membranes. One cell pair consists of an ion-diluting flow (product) channel located between a cation and an anion membrane with an ion concentrating (reject) flow channel located alternately between the cell pairs. A DC potential is maintained across the electrode plates which are located on opposite ends of the stacked membranes. Ion exchange resin is contained within the product flow channel, acting as an ion selective media in the electrodeionization process, aiding in the transfer of ions from the product stream to the reject stream. Isolation valves are provided for each stack to allow for maintenance of a stack.

Filters upstream of the electrodeionization stack in each train remove suspended solids and particulate matter from electrodeionization influent.

The ion concentrate reject stream is directed to the waste water system (WWS) or the liquid radwaste system (WLS).

The electrodeionization stacks are located in the turbine building and in an area where provisional shielding can be put in place. Anionic and cationic resins are contained within the electrodeionization stacks. These resins are not consumed or exhausted in the electrodeionization process. Radiation monitors associated with the steam generator blowdown system, steam generator system (main steam), and the condenser air removal system provides the means to determine if the secondary side is radioactively contaminated.

After prolonged use, the electrodeionization units will be replaced. If they are not radioactively contaminated, they require no special packaging and may be disposed as clean solid waste. If they are radioactively contaminated, they will be dewatered, the nozzles blocked and packaged for transport according to DOT regulations. Packaged electrodeionization units may be stored in the Radwaste Building.

10.4.8.2.3.7 Electrodeionization Reject Pressure Regulating Valve

A backpressure regulating valve is provided to maintain appropriate system backpressure to the reject path of the EDI modules. By keeping a constant backpressure, this valve will also keep a steady flow through radiation monitor BDS-RE011. This valve will also modulate within the operating range of blowdown flows during normal operation.

10.4.8.2.4 Instrumentation Applications

Flow, pressure, temperature, and radioactivity indicators with alarms monitor system operation. If pressure, temperature, or radioactivity reach a high level setpoint, an alarm is annunciated and the blowdown flow control valves and upstream isolation valves are automatically closed.

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Flow elements and transmitters measure and control blowdown flow from the steam generators. The flow elements are located downstream of the blowdown flow control valves.

Temperature instrumentation monitors the temperature of blowdown fluid upstream and downstream of each heat exchanger. The heat exchanger outlet temperature controls heat exchanger cooling water flow as well as the blowdown flow to limit high temperature blowdown fluid to the electrodeionization unit.

Radioactivity detection instrumentation detects and monitors the presence of radioactivity in the combined blowdown stream from both trains. A radiation element is located in the common header upstream of the recovered blowdown three-way valve. This three-way valve normally directs the recovered blowdown flow to the condenser. When recovery of the blowdown fluid is not possible, the flow is diverted to the waste water system. Upon detection of significant levels of radioactivity via a radiation transmitter alarm, the steam generator blowdown flow is diverted to the liquid radwaste system for processing. A second radioactive detection instrument is located on the waste stream of the electrodeionization blowdown. Similarly, a three-way valve normally directs this electrodeionization reject to the waste water system. With detection of significant levels of radioactivity, the reject is diverted to the liquid radwaste system.

10.4.8.3 Safety Evaluation

- Each blowdown line is provided with redundant safety-related valves that isolate the secondary side of the steam generator to preserve the steam generator inventory. The inventory is maintained as a heat sink for sensible and decay heat removal from the reactor coolant system.

- The steam generator blowdown system safety-related functions are accomplished by redundant means. A single, active component failure within the safety-related portion of the system does not compromise the safety-related function of the system. Power is supplied by the Class 1E dc power system as described in Chapter 8.

- Section 3.2 delineates the quality group classification. The controls and power supplies necessary for safety-related functions of the steam generator blowdown system are Class 1E, and are described in Chapters 7 and 8.

- The safety-related portion of the steam generator blowdown system are located in the containment and auxiliary buildings. These buildings and areas are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other natural phenomena. Sections 3.3, 3.4, 3.5, 3.7, and 3.8 provide the bases for the adequacy of the structural design of these buildings and areas. The safety-related portions of the steam generator blowdown system are designed to remain functional after a safe shutdown earthquake. Sections 3.7 and 3.9 provide the design loading conditions that are considered.

- No single failure coincident with loss of offsite power compromises the safety-related functions of the system or will result in:

  – Loss-of-coolant accident
  – Loss of integrity of steam lines
  – Loss of the capability to effect a safe reactor shutdown
  – Transmission of excessive loading to the containment pressure boundary.

  Component or functional redundancy is provided so that safety-related functions can be performed, assuming a single, active failure coincident with loss of ac power.

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- The steam generator blowdown system is initially tested in accordance with the program described in Chapter 14. Periodic inservice functional testing is done in accordance with Subsection 10.4.8.4. Section 6.6 provides the ASME Code, Section XI requirements that are appropriate for the safety-related portions of the steam generator blowdown system.

- The safety-related components of the steam generator blowdown system are qualified to function in normal, test, and accident environmental conditions. The environmental qualification program is provided in Section 3.11.

- Discussions of high energy pipe break locations and evaluation of effects are provided in Subsections 3.6.1 and 3.6.2.

- Subsection 6.2.3 delineates the criteria and compliance with applicable requirements and the criteria for the containment isolation provisions.

- The failure modes and effects analysis for the steam generator blowdown system is provided in Table 10.3.3-1.

10.4.8.4 Inspection and Testing Requirements

10.4.8.4.1 Preservice Testing/Inspection

The blowdown system components are tested and inspected during plant startup as a part of the preservice test program as discussed in Chapter 14. The steam generator blowdown system's safety-related functions are designed to include the capability for testing. This includes operation of applicable portions of the protection system. The safety-related components of the system (valves and piping,) are designed and located to permit preservice and inservice inspections to the extent practical.

The steam generator blowdown lines within the containment and the auxiliary building are visually and volumetrically inspected at installation as required by ASME Code, Section XI preservice inspection requirements.

10.4.8.4.2 Inservice Testing/Inspection

The performance and structural leaktight integrity of system components are demonstrated by normal operation.

Additional discussion of inservice inspection of the blowdown containment isolation valves is contained in Section 6.6 and Subsection 3.9.6.

Instruments and controls are calibrated during startup and recalibrated, as necessary, to maintain system operation within its design specifications.

10.4.9 Startup Feedwater System

The startup feedwater system supplies feedwater to the steam generators during plant startup, hot standby and shutdown conditions, and during transients in the event of main feedwater system unavailability. The startup feedwater system is composed of components from the AP1000 main and startup feedwater system (FWS) and steam generator system (SGS).

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10.4.9.1 Design Basis

10.4.9.1.1 Safety Design Basis

The safety functions of the startup feedwater system are to provide for containment isolation, steam generator isolation and feedwater isolation following design basis events requiring these actions. Containment isolation is provided to limit radioactive releases to the environment following design basis events that result in the releases of radioactivity to the containment. Steam generator isolation is provided to limit rapid blowdown to a single steam generator following a feedwater or steam line break. Feedwater isolation limits excessive feedwater flow to the steam generators to limit mass and energy releases to containment to limit excessive RCS cooldown and to limit steam generator overfill.

The portion of the startup feedwater system from the steam generator inlets outward through the containment up to and including the startup feedwater isolation valves (SFIVs) is constructed in accordance with the requirements of ASME Code, Section III for Class 2 components and is designed to seismic Category I requirements. The portion of the startup feedwater system from the startup feedwater isolation valve inlets to the piping restraints at the interface between the auxiliary building and the turbine building is constructed in accordance with the requirements of ASME Code, Section III for Class 3 components and is designed to seismic Category I requirements.

The startup feedwater system provides redundant isolation valves, as described below, for the startup feedwater lines routed into containment. The isolation valves close after receipt of an isolation signal in sufficient time to limit the mass and energy release to containment consistent with the containment analysis presented in Section 6.2.

- The safety-related portions of the startup feedwater system are designed to remain functional after a safe shutdown earthquake (SSE) and to perform their intended function of isolating startup feedwater flow following postulated events.

- The safety-related portions of the startup feedwater system are protected from wind and tornado effects, as described in Section 3.3; flood protection is described in Section 3.4; missile protection is described in Section 3.5; protection against dynamic effects associated with the postulated rupture of piping is described in Section 3.6; seismic protection is described in Section 3.7; environmental design is described in Section 3.11; and fire protection is described in Section 9.5.

- The portion of the startup feedwater system to be constructed in accordance with ASME Code, Section III, Class 2 requirements is provided with access to welds and removable insulation for inservice inspection, in accordance with ASME Code, Section XI. The portion of the startup feedwater system to be constructed in accordance with ASME Code, Section III, Class 3 requirements is also designed and configured to accommodate inservice inspection in accordance with ASME Code, Section XI. The startup feedwater system is designed so that the active components are capable of limited testing during plant operation.

- The startup feedwater system quality group classification codes are identified in Section 3.2. The control functions and power supply are described in Chapters 7 and 8.

- Double valve startup feedwater isolation is provided by the startup feedwater control valve and the startup feedwater isolation valve. Both valves are designed to close on a startup feedwater isolation signal, an appropriate engineered safeguards features (ESF) signal as indicated on Figure 7.2-1. The startup feedwater isolation valve also serves as a containment

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isolation valve. The startup feedwater control valve fails closed on loss of air. See Section 7.3. Backflow in the startup feedwater line results in closure of the startup feedwater check valve.

- For a steam generator tube rupture event, positive and redundant isolation is provided for the startup feedwater system (startup feedwater isolation valve and startup feedwater control valve), with isolation signals generated by the protection and safety monitoring system.

10.4.9.1.2 Power Generation Design Basis

- During normal plant startup, shutdown or hot standby, feedwater can be supplied through the startup feedwater control valves to the steam generators using either a booster/main feedwater pump drawing water from the deaerator storage tank (refer to Subsection 10.4.7), or using the startup feedwater pumps drawing water from the condensate storage tank.

- In the event of loss of the main feedwater system, the startup feedwater pumps automatically supply feedwater to the steam generators for heat removal from the reactor coolant system. The heat removal function of the startup feedwater system is nonsafety-related. The startup feedwater system avoids the need for actuation of the safety-related passive core cooling system. Following the transient, the system refills the steam generators and supports reactor coolant system cooldown.

- One operating startup feedwater pump delivers sufficient flow to the steam generators to avoid actuation of the passive core cooling system following a reactor trip. The maximum flow available from two operating startup feedwater pumps does not result in overcooling the reactor coolant system, overfilling the steam generators, or inputting excessive mass/energy to containment following a main steam line break.

- The startup feedwater pumps use the condensate storage tank as a water supply source. A sufficient volume of feedwater is available from the condensate storage tank (refer to Subsection 9.2.4) to achieve cold shutdown, based on 8 hours of operation at hot standby conditions and subsequent cooldown of the reactor coolant system within 6 hours to conditions which permit operation of the normal residual heat removal system.

- The startup feedwater pumps are headered at the pump discharge, and a separate line runs from the header to each steam generator.

- For a main feedwater or main steam line break (MSLB) inside the containment, the startup feedwater lines provide a nonsafety-related path for the addition of feedwater to the remaining intact loop if ac power is available.

- For a main feedwater line break upstream of the main feedwater isolation valve (outside of the containment), the startup feedwater lines provide a nonsafety-related path for the addition of feedwater to maintain steam generator level if ac power is available.

- Two startup feedwater pumps are provided with a single pump capable of satisfying the startup feedwater system flow demand for decay heat removal. These pumps automatically start and maintain steam generator water level when the main feedwater system is unavailable.

- In the event of loss of normal ac power and offsite power, the startup feedwater pumps and associated motor operated isolation valves are powered by the onsite standby ac power supply (diesels). Each of the two startup feedwater pumps is powered by its respective standby diesel.

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- During normal plant startup, feedwater is supplied through the startup feedwater control valves to the steam generators until transition is made to the main feedwater control valves of the main feedwater system. During normal plant shutdown, feedwater is supplied through the startup feedwater control valves after transition is made from the main feedwater control valves, and until the normal residual heat removal system is placed in service.

10.4.9.2 System Description

10.4.9.2.1 General Description

The startup feedwater system is shown schematically in Figure 10.4.7-1 as part of the condensate and feedwater system piping and instrument diagram and in Figure 10.3.2-1 as part of the main steam system piping and instrument diagram. Classification of equipment and components is given in Section 3.2.

Startup feedwater is defined to be feedwater that passes through the startup feedwater control valves, and can be supplied from either of two sources. Startup feedwater can be supplied by a booster/main feedwater pump drawing from the deaerator storage tank and delivering through cross connect piping to the startup feedwater header; or, startup feedwater can be supplied by one or both startup feedwater pumps drawing from the condensate storage tank and delivering to the startup feedwater header. The startup feedwater header is defined to be the common segment of startup feedwater piping downstream of the startup feedwater pumps. The booster/main feedwater pumps are part of the condensate and feedwater system and are described in Subsection 10.4.7. As described in Subsection 10.4.7.2.1, the cross connection piping between the main feedwater pump discharge header and the startup feedwater header contains a check valve and a nonsafety-related, air-operated isolation valve. The check valve prevents the startup feedwater pumps from supplying the main feedwater header, and the isolation valve automatically closes upon a main feedwater isolation signal to isolate the main feedwater system from the startup feedwater system.

Two parallel startup feedwater pumps are provided and take suction from the condensate storage tank. Each startup feedwater pump discharges to the startup feedwater header through a venturi flow element, an automatic recirculation valve, and a remotely-operated isolation valve. The venturi flow element provides a flow measurement signal at normal flow rates, and cavitates at a flow rate near pump runout to choke the flow and avoid further flow increase. The automatic recirculation valve functions as a check valve to prevent reverse flow through the pump, and also functions as a minimum flow control valve for pump protection; during conditions of low forward flow to the system, sufficient flow from the pump is automatically recirculated back to the condensate storage tank to meet pump minimum flow requirements. The discharge isolation valve is closed when the associated pump is not operating; when in standby operation, the valve automatically opens when the associated pump starts.

The startup feedwater header branches into individual lines to the two steam generators. Each individual line contains a startup feedwater control valve, a check valve, and a startup feedwater isolation valve. Startup feedwater flow in each line is controlled by the associated startup feedwater control valve to maintain level in the associated steam generator.

A thermal relief valve is included in each startup feedwater line between the check valve and isolation valve to prevent overpressure between the control valve and isolation valve due to thermal expansion when both valves are closed.

A startup feedwater system failure analysis for safety-related components is presented in Table 10.4.9-1.

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10.4.9.2.2 Component Description

From the connections at the steam generators out through the startup feedwater isolation valves, the startup feedwater system is designed in accordance with the requirements of ASME Code, Section III for Class 2 components and seismic Category I requirements. From upstream of the startup feedwater isolation valve to the restraints at the interface between the auxiliary building and turbine building, the system is designed in accordance with ASME Code, Section III for Class 3 components and seismic Category I requirements. The remaining portion of the startup feedwater system is nonsafety-related.

Startup Feedwater Pump

Each startup feedwater pump is a multistage, centrifugal pump driven by an ac motor. Each pump can supply 100 percent of the required flow to the two steam generators to meet the decay heat removal requirements specified in Subsection 10.4.9.1.2. The pumps automatically start as described in Subsection 10.4.9.2.3.4. Isolation valves at the pump suction and discharge allow each startup feedwater pump to be individually serviced. The discharge isolation valve for each pump is powered by the same train of the onsite standby ac power supply as the associated pump.

Startup Feedwater Control Valve

The startup feedwater control valves are air-operated, modulating control valves with the dual purpose of controlling startup feedwater flow rate, as well as providing isolation of the startup feedwater system. The valve body is a globe design that provides the required range of startup feed control, as well as positive isolation. The startup feedwater control valves operator is equipped with an auxiliary air accumulator to provide independent operation of the startup feedwater control valves upon loss of normal instrument air supply.

The startup feedwater control valves automatically maintain water level in the steam generators during operation of the startup feedwater system, in response to signals generated by the plant control system.

In the event of a secondary side pipe rupture inside the containment, the startup feedwater control valve provides a secondary backup to the startup feedwater isolation valve limiting the quantity of high-energy fluid that enters the containment through the broken pipe. For emergency closure of the valve, a solenoid is deenergized, resulting in valve closure in sufficient time to limit the mass and energy release to containment consistent with the containment analysis presented in Section 6.2. The electrical solenoid is energized from a Class 1E source.

Startup Feedwater Isolation Valve

One startup feedwater isolation valve is installed in each startup feedwater line outside containment and downstream of a startup feedwater control valve and a startup feedwater check valve. The following primary functions are performed by the valve:

- The startup feedwater isolation valve is provided to prevent the uncontrolled blowdown from more than one steam generator in the event of startup feedwater line rupture. The startup feedwater control valve provides backup isolation.

- The startup feedwater isolation valve and the startup feedwater control valve provide isolation of the nonsafety-related portions of the system from the safety-related portions.

- In the event of a secondary pipe rupture inside containment, the startup feedwater isolation valve and startup feedwater control valve provide isolation to limit the quantity of high energy fluid that enters the containment.

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- In the event of a steam generator tube rupture, the startup feedwater isolation valve and startup feedwater control valve limit overfill of the steam generator by terminating startup feed flow.

The startup feedwater isolation valve is a remotely-operated gate valve designed in accordance with ASME Code, Section III Class 2 requirements. The valve operator is designed to stroke against steam generator pressure or startup feedwater pump shutoff head.

The startup feedwater isolation valve and startup feedwater control valve functional diagrams are shown in Figure 7.2-1. To provide the safety function actuation (closure) as well as reliable alignment, and redundant and independent actuation, the startup feedwater isolation valve and startup feedwater control valve are powered from separate Class 1E power sources.

Startup Feedwater Thermal Relief Valves

A relief valve is provided between the startup feedwater check valve and isolation valve to prevent overpressurization of the startup feedwater lines between the startup feedwater isolation valve and startup feedwater control valve. The relief valve prevents overpressurization that could be caused by thermal expansion of the fluid trapped between the isolation and control valves following an event causing startup feedwater isolation.

10.4.9.2.3 System Operation

The startup feedwater system supplies the steam generators with feedwater during conditions of plant startup, hot standby and shutdown, and during transients in the event of main feedwater system unavailability. The startup feedwater system also supplies feedwater during low power operation under conditions when the startup feedwater control valves regulate the feedwater flow to the steam generators.

10.4.9.2.3.1 Startup

During reactor startup and at low power levels, feedwater is supplied to the steam generators through the startup feedwater control valves using either the startup feedwater pumps drawing from the condensate storage tank, or a booster/main feedwater pump drawing from the deaerator storage tank. Refer to Subsection 10.4.7 for a description of the operation of the condensate and feedwater system and the booster/main feedwater pumps. The feedwater pumps in use operate on minimum flow recirculation as necessary while maintaining the water level of the steam generators. Feedwater is controlled by the startup feedwater control valves, which are operated either manually from the control room or automatically in accordance with steam generator level demand. If the startup feedwater pumps are initially in use, transfer is made to a booster/main feedwater pump prior to exceeding the capacity limit of the startup pumps. As power increases, feedwater continues to be supplied through the startup feedwater control valves until control of feedwater is automatically transferred from the startup feedwater control valves to the main feedwater control valves. As the main feedwater control valves open and assume responsibility for maintaining steam generator water level, the startup feedwater control valves close. Position indication is available in the main control room for the main and startup feedwater control valves.

10.4.9.2.3.2 Hot Standby

During hot standby conditions, feedwater is supplied to the steam generators through the startup feedwater control valves using either one or both startup feedwater pumps drawing from the condensate storage tank, or a booster/main feedwater pump drawing from the deaerator storage tank. The startup feedwater control valves operate to maintain the steam generator levels, and minimum flow recirculation is automatically utilized as required to protect the feedwater pumps that are in use.

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10.4.9.2.3.3 Shutdown

Operation during power descent and shutdown is generally the reverse of operation during startup and power ascent. At low feedwater flows, control of feedwater is automatically transferred from the main feedwater control valves to the startup feedwater control valves. Feedwater is supplied by an operating booster/main feedwater pump drawing from the deaerator storage tank. Feedwater can continue to be supplied by a booster/main feedwater pump during the shutdown process; alternatively, feedwater supply can be transferred to the startup feedwater pumps when flow demand has decreased to within their capacity. Feedwater continues to be supplied until the normal residual heat removal system is placed in service.

10.4.9.2.3.4 Automatic Starts

The startup feedwater pumps automatically start upon conditions resulting from insufficient main feedwater flow to the steam generators. An automatic pump start signal is generated by the plant control system (PLS). The signal is generated on low main feedwater flow coincident with low steam generator level. As a backup to this logic, it is also initiated on steam generator level alone, at a setpoint below the low steam generator level setpoint.

The amount of startup feedwater flow delivered to each steam generator is determined by the associated startup feedwater control circuit, which sends a signal to modulate the startup feedwater control valve (Figure 10.3.2-1) in response to steam generator water level control signals. The control valve is modulated as required to maintain the programmed steam generator water level setpoint.

Following a reactor trip that is not the result of a main feedwater system malfunction and in which the main feedwater system remains available, the startup feedwater pumps do not automatically start. In this case, the startup feedwater control valves take control and open to supply the steam generators using feedwater delivered from a booster/main feedwater pump through cross-connect piping. The startup feedwater pumps remain on standby for backup protection, and can be manually started if desired by the plant operator.

10.4.9.2.3.5 Emergency Operation

The startup feedwater system is not required to supply feedwater under accident conditions. However, the startup feedwater system is expected to be available as a nonsafety-related, first line of defense to provide a source of feedwater for the steam generators. Coordinated operation of the startup feedwater system (which starts automatically, as discussed in Subsection 10.4.9.2.3.4), if available, and the main steam supply system (refer to Section 10.3) are employed to remove the primary loop sensible heat and reactor decay heat. A minimum condensate storage tank volume of 325,000 gallons is required for defense-in-depth purposes. The condensate storage tank size is shown in Subsection 9.2.4.2.2.

10.4.9.3 Safety Evaluation

- The safety-related portions of the startup feedwater system are located in the containment and auxiliary buildings. These buildings are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other natural phenomena. Sections 3.3, 3.4, 3.5, 3.7, and 3.8 provide the bases for the adequacy of the structural design of these buildings.

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- The safety-related portions of the startup feedwater system are designed to remain functional after a design basis earthquake. Subsection 3.7.2 and Section 3.9 provide the design loading conditions that are considered. Sections 3.5, 3.6, and Subsection 9.5.1 provide the analyses to provide confidence that a safe shutdown, as outlined in Section 7.4, is achieved and maintained.

- The startup feedwater system safety-related functions are accomplished by redundant means. A single, active component failure of the safety-related portion of the system does not compromise the safety function of the system. Table 10.4.9-1 provides a failure analysis of the safety-related active components of the startup feedwater system. Power is supplied from onsite power systems, as described in Chapter 8.

- Preoperational testing of the safety-related portion of the condensate and feedwater system is performed as described in Chapter 14. Periodic inservice functional testing is done in accordance with Subsection 10.4.9.4. Section 6.6 provides the ASME Code, Section XI requirements that are appropriate for the startup feedwater system.

- Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system and supporting systems. The controls and power supplies necessary for the safety-related functions of the startup feedwater system are Class 1E, as described in Chapters 7 and 8.

- The startup feedwater isolation valves and the startup feedwater control valves automatically close upon receipt of a feedwater isolation signal, which occurs on a steam generator High-3 water level and other appropriate engineered safeguards signals as shown on the diagrams titled “Feedwater Isolation” and “Steam Line Isolation” in Figure 7.2-1.

- For a steam generator tube rupture event, positive and redundant isolation is provided for the startup feedwater system (startup feedwater isolation valve and startup feedwater control valve) to prevent steam generator overfill, with engineered safeguards isolation signals generated by the protection and safety monitoring system (PMS).

10.4.9.4 Tests and Inspections

10.4.9.4.1 Preoperational Valve Testing

The startup feedwater isolation valves and startup feedwater control valves are checked for closing time prior to initial startup.

10.4.9.4.2 Preoperational Pipe Testing

The Class 2 portion of the startup feedwater system piping is tested and inspected to the requirements of ASME Code, Section III, Class 2 piping. In addition, the portion of the piping between the containment penetration and the anchor, which is traditionally considered as the break exclusion zone described in Subsection 3.6.2, is subjected to 100-percent volumetric inspection at installation (that is, 100-percent volumetric examination of shop and field longitudinal and circumferential welds).

10.4.9.4.3 Preoperational System Testing

Preoperational testing of the startup feedwater system is performed as described in Chapter 14. Tests described in Subsection 14.2.9.1.7, under item c) of General Test Method and Acceptance Criteria satisfy BTP (AS) 10-2. Additional testing of the startup feedwater system is conducted during startup testing as described in Subsection 14.2.10.4.18.

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10.4.9.4.4 Inservice Inspections

The performance and structural and leaktight integrity of the startup feedwater system components are demonstrated by normal operation.

The inservice inspection program for ASME Section III Class 2 and 3 components is described in Section 6.6. The inservice testing program, including testing for the startup feedwater isolation valve and startup feedwater control valve, is described in Subsection 3.9.6.

10.4.9.5 Instrumentation Applications

The startup feedwater system instrumentation is designed to facilitate automatic operation, remote control, and continuous indication of system parameters.

The startup feedwater flow is controlled by a steam generator level demand signal modulating the startup feedwater control valve. The control valve may either be in manual or automatic control. Refer to Section 7.7. The startup feedwater flow transmitters also provide redundant indication of startup feedwater and automatic PRHR HX actuation input on Low-2 flow coincident with Low-2 steam generator narrow range water level in both SGs, coincident with High T_{cold} in either loop. See Section 7.3.

10.4.10 Auxiliary Steam System

The auxiliary steam system (ASS) provides the steam required for plant use during startup, shutdown, and normal operation. Steam is supplied from either the auxiliary boiler or the main steam system. During a loss of normal ac power, a temporary boiler can be used for investment protection.

10.4.10.1 Design Basis

10.4.10.1.1 Safety Design Basis

The auxiliary steam system serves no safety-related function and therefore has no nuclear safety design basis.

10.4.10.1.2 Power Generation Design Basis

The auxiliary steam system supplies steam required by the unit for a cold start of the main steam system and turbine-generator. Additionally, the auxiliary steam system provides steam for hot water heating. Main steam supplements the auxiliary steam header during startup and supplies the auxiliary steam header during normal operation. The auxiliary boiler provides steam to the header during plant shutdown.

10.4.10.2 System Description

10.4.10.2.1 General Description

The auxiliary boiler is located in the turbine building. The system consists of steam generation equipment and distribution headers.

Condensate from the condensate storage tank is chemically treated and pumped to the auxiliary boiler deaerator where oxygen and non-condensables are removed using auxiliary steam. The auxiliary boiler feedwater pumps deliver condensate from the auxiliary boiler deaerator to the auxiliary boiler. A feedwater control valve, located in the feedwater piping, regulates water level in the

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auxiliary boiler. Feedwater flow is proportional to auxiliary boiler steaming rate. Steam generated by the auxiliary boiler is supplied to the plant auxiliary steam distribution piping.

Boiler water quality is maintained by controlling boiler blowdown flow to an atmospheric blowdown tank and by feeding oxygen scavenging and pH control chemicals to the boiler makeup water system.

Water level in the auxiliary boiler deaerator is maintained by an automatic control valve in the condensate supply and deaerator overflow piping. Makeup water is supplied from the demineralized water transfer and storage system.

During a loss of normal ac power, the auxiliary steam supply system provides steam to the hot water heating system for investment protection. To accomplish this, provisions are made to power the auxiliary boiler makeup and feed pumps from the onsite diesel generators. Steam is supplied by a temporary boiler with diesel-backed control power

10.4.10.2.2 Component Description

Auxiliary steam system component classification is as described in Section 3.2.

Auxiliary Steam System and Boiler

The auxiliary steam boiler is an electric package boiler with a nominal net output capacity of approximately 62,000 pounds per hour of saturated steam at 195 psig. The system is protected from overpressure by safety valves located on the boiler, boiler deaerator, and auxiliary steam header.

Pumps

Two 100-percent capacity auxiliary boiler feedwater pumps are provided to feed the auxiliary steam boiler. These pumps are also used to supply the temporary boiler for investment protection.

Two 100-percent capacity auxiliary boiler makeup pumps maintain level in the boiler deaerator.

Auxiliary Boiler Deaerator

The auxiliary boiler deaerator is a 100-percent-capacity deaerator which uses steam supplied by the auxiliary steam header. The auxiliary boiler deaerator steam blanket is controlled for preheating and deaerating boiler makeup water. The auxiliary boiler deaerator removes oxygen and non-condensables from auxiliary boiler feedwater.

Chemical Treatment Components

The auxiliary boiler makeup water is treated with pH control and oxygen scavenging chemicals. Chemical injections maintain proper water chemistry during operational conditions. Batch chemicals for cleaning and layup are injected into the auxiliary boiler and auxiliary boiler deaerator when they are not in operation. Chemical feed equipment for the auxiliary steam system is part of the turbine island chemical feed system (CFS) and is described in Subsection 10.4.11.

10.4.10.2.3 System Operation

When in operation, the auxiliary steam system provides the following services:

- Steam to the plant hot water heating system (VYS) heat exchangers where water is heated and pumped to the heating system ventilation coils.

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- Steam for the condensate system deaerator prior to plant startup. This heating occurs in two stages: initially the only source of auxiliary steam is the auxiliary boiler, which provides limited heating steam to the deaerator due to the steam demand of the VYS and the gland steam system (GSS). Once main steam is available to augment the auxiliary boiler steam, the auxiliary steam system has adequate steam to maintain the deaerator at a positive pressure, at feedwater flows required for turbine synchronization.

- Sealing steam to the glands of the main turbine prior to the availability of main steam.

- Steam for maintaining pressure in the condensate system deaerator after a turbine trip when extraction steam is lost.

Operational safety features are provided within the system for the protection of plant personnel and equipment. The auxiliary steam system does not interface directly with nuclear process systems.

10.4.10.3 Safety Evaluation

The auxiliary steam system has no safety-related function and therefore requires no nuclear safety evaluation. High energy pipe rupture analysis is not required for the auxiliary steam system since none of the lines pass through areas where safety related equipment is located.

10.4.10.4 Tests and Inspections

Testing of the auxiliary steam system is performed prior to initial plant operation.

Components of the system are monitored during operation to verify satisfactory performance.

10.4.10.5 Instrumentation Applications

A boiler control system is provided with the auxiliary boiler package for automatic control of the auxiliary boiler. Features of the control system include automatic shutdown of the auxiliary boiler on an abnormal condition.

The auxiliary steam system is provided with the necessary controls and indicators for local or remote monitoring of the operation of the system.

10.4.11 Turbine Island Chemical Feed

The turbine island chemical feed system (CFS) injects required chemicals into the condensate (CDS), feedwater (FWS), auxiliary steam (ASS), service water (SWS), and demineralized water treatment (DTS). CFS components are located in the turbine building, SWS chemical treatment building, and SWS cooling tower basin area.

10.4.11.1 Design Basis

10.4.11.1.1 Safety Design Basis

The turbine island chemical feed system serves no safety-related function and therefore has no nuclear safety design basis.

10.4.11.1.2 Power Generation Design Basis

A noncorrosive condition is maintained within the systems serviced by the turbine island chemical feed system.

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The secondary sampling system (SSS), as described in Subsection 9.3.4, contains sampling requirements in accordance with water chemistry specifications that are provided in the secondary side water chemistry program, as described in Subsection 10.3.5.

10.4.11.2 System Description

Classification of equipment and components is given in Section 3.2.

10.4.11.2.1 Component Description

Condensate, Feedwater and Auxiliary Steam

An all-volatile chemical feed system (AVT) is used for condensate, feedwater and auxiliary steam water chemistry control. An oxygen scavenger is injected into the condensate system downstream of the condensate polishers to control the dissolved oxygen level. Feedwater chemistry is controlled by maintaining a residual level of oxygen scavenger. The injection point for the feedwater oxygen scavenger is located upstream of the feedwater booster pump suction. A pH adjuster is also injected into the condensate system downstream of the condensate polishers for pH control. Injection for pH control of the feedwater is located upstream of the feedwater booster pump suction. Chemical feed pumps and tanks are used to store and inject the chemicals into the piping system. Subsection 10.4.10.2.2 describes chemical feed for the auxiliary steam system.

Service Water

A biocide, pH adjuster, and dispersant/corrosion/scale inhibitor are injected into the service water system as required. A nonoxidizing biocide can be fed to the service water cooling tower basin. Subsection 9.2.1.2.2 describes chemical feed for the service water system.

Demineralized Water Treatment

A pH adjuster, chlorine reducing agent, and scale inhibitor are injected into the demineralized water treatment system. Subsection 9.2.3.2.3 describes chemical feed for the demineralized water treatment system.

10.4.11.2.2 System Operation

Condensate, Feedwater and Auxiliary Steam System Chemistry Control

An oxygen scavenger is injected upstream of the feedwater booster pump suction to maintain a residual level of oxygen scavenger and a dissolved oxygen level of not more than 5 ppb at the inlet to the steam generator.

A pH adjuster is also injected upstream of the feedwater booster pump suction to maintain the pH at the steam generator inlet within the control program for pH.

An oxygen scavenger is injected into the condensate system downstream of the condensate polishers to maintain a dissolved oxygen level of not more than 10 ppb at the inlet of the deaerator.

A pH adjuster is injected into the condensate system downstream of the condensate polishers to maintain the pH above 9.0 at the deaerator inlet within the control program for pH.

The chemical feed system may be used to place the steam generators in wet layup. This layup process is accomplished using the chemical feed system in conjunction with the steam generator blowdown system. Refer to Subsection 10.4.8.2 for details of this process.

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An oxygen scavenger and pH adjuster are injected into the auxiliary steam system downstream of the boiler makeup pumps to maintain the dissolved oxygen level and pH within the auxiliary boiler program levels. The chemical feed rates are manually adjusted.

Service Water System Chemistry Control

A biocide, pH adjuster and dispersant/corrosion/scale inhibitor are injected in the SWS cooling tower basin as required. Chemical feed rates for the biocide and dispersant/corrosion/scale inhibitor are either manually or automatically adjusted to maintain proper concentrations. The pH adjuster chemical feed rate is controlled electronically from instrumentation that measures pH.

A nonoxidizing biocide is provided to control algae formation on the service water cooling tower. The feed rate is either manually or automatically adjusted.

Demineralized Water Treatment System Chemistry Control

A pH adjuster, chlorine reducing agent, and scale inhibitor are injected into the raw water supply to the demineralized water treatment system.

10.4.11.3 Safety Evaluation

The turbine island chemical feed system has no safety-related function and therefore requires no nuclear safety evaluation.

Toxic gases, such as chlorine, are not used in the turbine island chemical feed system. The impact of toxic material on main control room habitability is addressed in Section 6.4.

10.4.11.4 Tests and Inspections

The turbine island chemical feed system is operationally checked before initial plant startup to verify proper functioning of the feed systems and chemical sensors.

10.4.11.5 Instrumentation Applications

The secondary sampling system (SSS), as described in Subsection 9.3.4, provides instrumentation which measures dissolved oxygen, oxygen scavenger residual, and pH for the condensate, feedwater, and steam generator systems. These analyzers provide an indication of water quality and inputs for either manual or automatic control of the condensate and feedwater systems oxygen scavenging and pH control chemical feed pumps. Grab samples are analyzed to provide input for manual adjustment of feed rates for the auxiliary steam system oxygen scavenging and pH control chemical feed pumps. Wet layup operations are manually performed based on the results of the grab sample analysis.

Grab samples are analyzed to provide input for either manual or automatic adjustment of feed rates for biocide, pH adjustment, and/or dispersant/corrosion/scale inhibitor chemicals for service water and demineralized water treatment.

10.4.12 Combined License Information

10.4.12.1 Circulating Water System

The Combined License applicant will address the final configuration of the plant circulating water system including piping design pressure, the cooling tower or other site-specific heat sink.

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As applicable, the Combined License applicant will address the acceptable Langelier or Stability Index range, the specific chemical selected for use in the CWS water chemistry control, pH adjuster, corrosion inhibiter, scale inhibiter, dispersant, algicide and biocide applications reflecting potential variations in site water chemistry and in micro macro biological lifeforms. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room habitability is addressed in Section 6.4. The Combined License applicant will also be responsible for the design, routing, and disposition requirements associated with the main condenser waterbox drains.

10.4.12.2 Condensate, Feedwater and Auxiliary Steam System Chemistry Control

The oxygen scavenging agent and pH adjuster selection for the turbine island chemical feed system is addressed in Subsection 10.4.7.2.1.

10.4.12.3 Potable Water

Potable water is produced on site by the Potable Water System (PWS). Source water for the PWS is from the well water subsystem to the Raw Water System. Sodium hypochlorite is used as the biocide. The PWS is discussed in Subsection 9.2.5.

10.4.13 References

1. ASME Performance Test Code 19.11, 2008.

2. Heat Exchange Institute Performance Standard for Liquid Ring Vacuum Pumps.

3. Not Used.

201. Nuclear Energy Institute, "Steam Generator Program Guidelines," NEI 97-06.

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Table 10.4.1-1
Main Condenser Design Data

Condenser Data	
Condenser type	Multipressure, Single pass
Hotwell storage capacity	3 min
Heat transfer	7,540 x 10^6 Btu/hr
Design operating pressure (average of all shells)	2.9 in.-Hg
Shell pressure (design)	0 in.-Hg absolute to 15 psig
Circulating water flow	[[600,000 gpm]]
Water box pressure (design)	90 psig
Tube-side inlet temperature	91°F
Approximate Tube-side temperature rise	25.2°F
Condenser outlet temperature	116.2°F
Waterbox material	Carbon Steel
Condenser Tube Data	
Tube material (main section)	Titanium(1)
Tube size	1″ O.D. – 23 BWG
Tube material (periphery)	Titanium(1)
Tube size	1″ O.D. – 23 BWG
Tube sheet material	Titanium or Titanium Clad Carbon Steel(2)
Support plates	Modular Design/Carbon Steel

Note:
1. For fresh water plants, an equivalent tube material such as 304L, 316L, 904L, or AL-6X may be substituted.
2. If one of the alternate tube materials is used, the tube sheet shall be carbon steel, clad with the same material as the tubes.

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Table 10.4.5-1
[[Design Parameters for Major
Circulating Water System Components (Conceptual Design)]]

Circulating Water Pump	
Quantity	Four per unit
Flow rate (gal/min)	156,200 (note 2)
Natural Draft Cooling Tower	
Quantity	One per unit
Approach temperature (°F)	13.6
Inlet temperature (°F)	116.7 (note 3)
Outlet temperature (°F)	90.9
Approximate Temperature range (°F)	25.8
Flow rate (gal/min)	622,000
Heat transfer (Btu/hr)	$7{,}628 \times 10^6$ (note 3)
Wind velocity design (mph) Seismic design criteria per Uniform Building Code	110

Notes:
1. Not used.
2. Nominal CWS pump flow rate includes blowdown (2,801 gpm) which is discharged from the pump discharge common header. This flow rate (156,200 gpm) is the minimum flow rate needed to meet system design requirements.
3. The inlet temperature (°F) is approximately 115.2°F, as the condenser outlet flow is expected to be cooled by the return flow stream from the TCS.

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Table 10.4.7-1 (Sheet 1 of 2)
Condensate and Feedwater System Component Failure Analysis

Component	Failure Effect on Train	Failure Effect on System	Failure Effect on RCS
1. SGS PL V057A (MFIV)	1a. Valve fails closed or fails to open on command. Train “A” is not available for FW flow to SG “A.”	FW Train “B” available. SFW Train “A” available.	None. Decay heat removal is maintained via PRHR actuation on ESF signal. SFW available to provide flow to SG “A.”
	1b. Valve fails open or fails to close on command. Trains “A” and “B” are available for FW flow. Isolation function of V057A is not available. Redundant power division closure of MFIV provided for reliability; backup isolation provided by V058A and V250A.	Valve V250A (MFCV) provides backup isolation to terminate feedwater flow. Check valve V058A provides redundant feedwater blowdown isolation from SG “A;” redundant containment isolation provided by SG and feedwater line inside containment.	None. RCS integrity is maintained by valve V058A/V250A available to prevent SG “A” blowdown. Decay heat removal available via PRHR and SFW actuated by ESF signal. SG overfill protection provided by backup isolation of V250A.
2. SGS PL V057B (MFIV)	2a. Same as except for SG “B.”	FW Train “A” available. SFW Train “B” available.	Same as 1a except for SG “B.”
	2b. Same as 1B except isolation function of V057B is not available.	Same as 1b except for “B” train valves and SG “B.”	Same as 1b except for “B” train valves and SG “B.”
3. SGS PL V250A (MFCV)	3a. Valve fails closed or fails to open on command. Train “A” is not available for FW flow to SG “A.”	FW Train “B” available. SFW Train “A” available.	None. Decay heat removal is maintained via PRHR actuation on ESF signal. SFW available to provide flow to SG “A.”
	3b. Valve fails open or fails to close on command. Trains “A” and “B” are available for FW flow without flow control to SG “A.” Backup isolation provided by V058A and V057A.	Valve V057A (MFIV) provides backup isolation to terminate feedwater flow. Check valve V058A and V057A provide redundant feedwater blowdown isolation from SG “A.” SFW train “A” and “B” available for decay heat removal.	None. RCS integrity is maintained by valve V057A and V058A closure to limit SG blowdown. Decay heat removal available via PRHR and SFW actuated by ESF signal. SG overfill protection provided by redundant isolation of V057A.

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Table 10.4.7-1 (Sheet 2 of 2)
Condensate and Feedwater System Component Failure Analysis

Component	Failure Effect on Train	Failure Effect on System	Failure Effect on RCS
4. SGS PL V250B (MFCV)	4a. Same as 3a, except for train “B” and SG “B.”	Same as 3a except train “B.”	Same as 3a except for SG “B.”
	4b. Same as 3b, except for SG “B” and valves V057B and V058B.	Same as 3b except valves V057B and V058B and SG “B.”	Same as 3b except for V057B and V058B.

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Table 10.4.9-1 (Sheet 1 of 2)
Startup Feedwater System Component Failure Analysis

Component	Failure Effect on Train	Failure Effect on System	Failure Effect on RCS
1. SGS PL V255A (SFCV)	1a. Valve fails closed or fails to open on command. SFW Flow is not available to SG A.	System is not available for SFW supply to SG “A.”	None. Decay heat removal is maintained via PRHR actuation on ESF signal.
	1b. Valve fails open or fails to close on command. SFW flow is uncontrolled.	Downstream isolation valve V067A trips closed on high SG level; system pumps are tripped on high SG level.	None. RCS integrity is maintained by V067A closure and main feedwater isolation. SG overfill terminated by ESF closure of V067A.
2. SGS PL V225B (SFCV)	2a. Same as 1a except flow is not available to SG B.	System is not available for SFW supply to SG “B.”	None. Same as 1a.
	2b. Same as 1b.	Same as 1b except valve V067B trips closed on high SG level.	Same as 1b except RCS integrity is maintained by V067B and main feedwater isolation, and overfill terminated by closure of V067B.
3. SGS PL V067A (SFIV)	3a. Valve fails closed. SFW flow is not available to SG A.	System is not available for SFW supply to SG “A.”	None. Same as 1a.
	3b. Valve fails open. Isolation function of V067A is not available; backup isolation provided by V0255A and V256A.	None. Valve V255A is automatically closed and SFW pumps tripped on an ESF signal. SG overfill protection provided by automatic isolation of V255A.	None. RCS integrity is maintained by V255A closure to limit cooldown; PRHR available for decay heat removal and SG overfill protection provided by redundant isolation of V255A.

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Table 10.4.9-1 (Sheet 2 of 2)
Startup Feedwater System Component Failure Analysis

Component	Failure Effect on Train	Failure Effect on System	Failure Effect on RCS
4. SGS PL V067B (SFIV)	4a. Same as 3a except flow not available to SG “B.”	Same as 3a except SFW supply not available to SG “B.”	Same as 1a.
	4b. Same as 3b.	Same as 3b except reference valve is V255B.	Same as 3b except reference valve is V255B.

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Table 10.4.9-2
Nominal Component Design Data – Startup Feedwater System

Startup Feedwater Pump	
Type	Multi-stage, centrifugal
Driver	Electric Motor
Quantity	2
Capacity	520 gpm @ 80°F
Head	3250 ft
Motor hp	800

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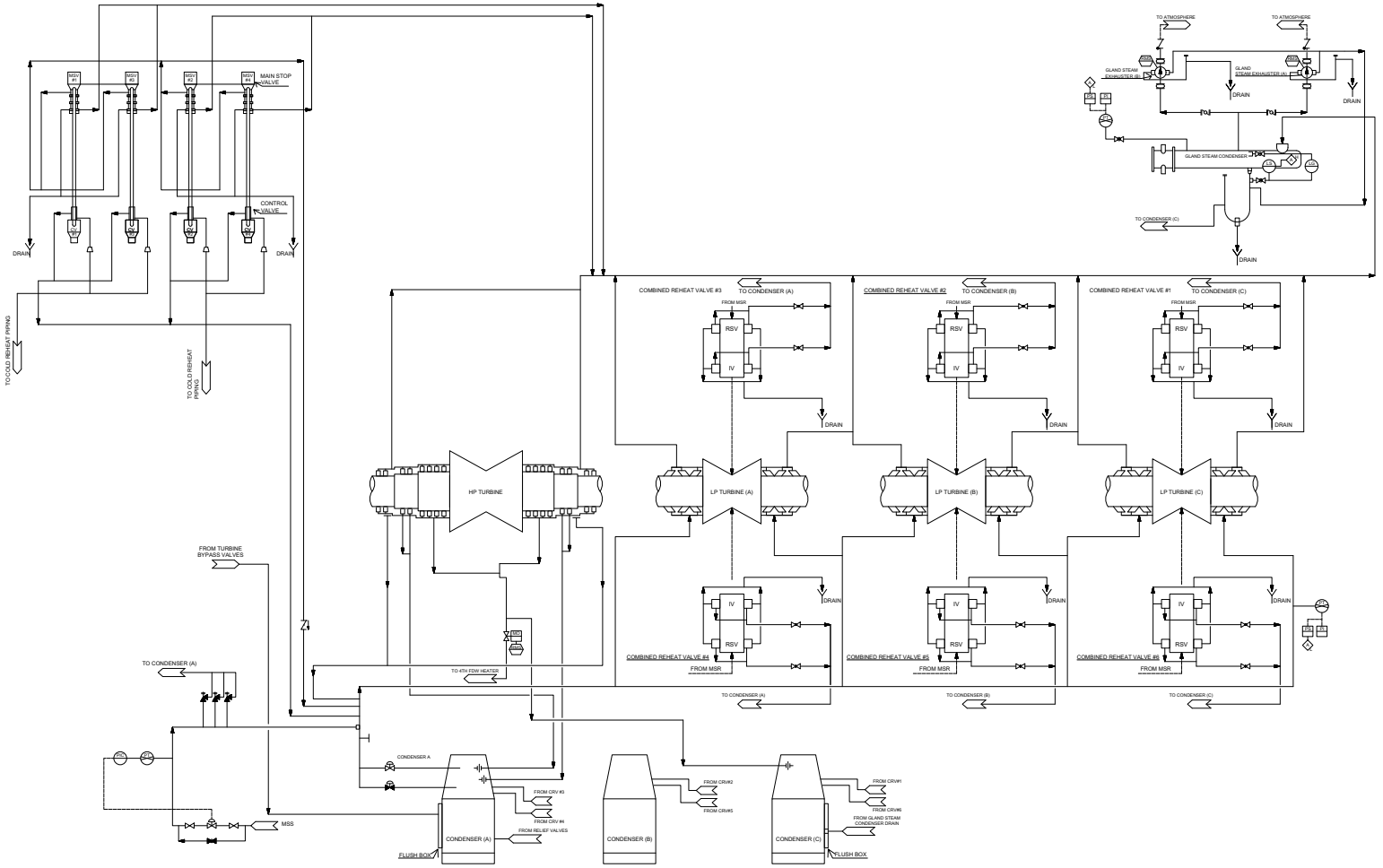

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Figure 10.4.3-1
Gland Seal System
Piping and Instrumental Diagram
(REF) GSS 001, 002

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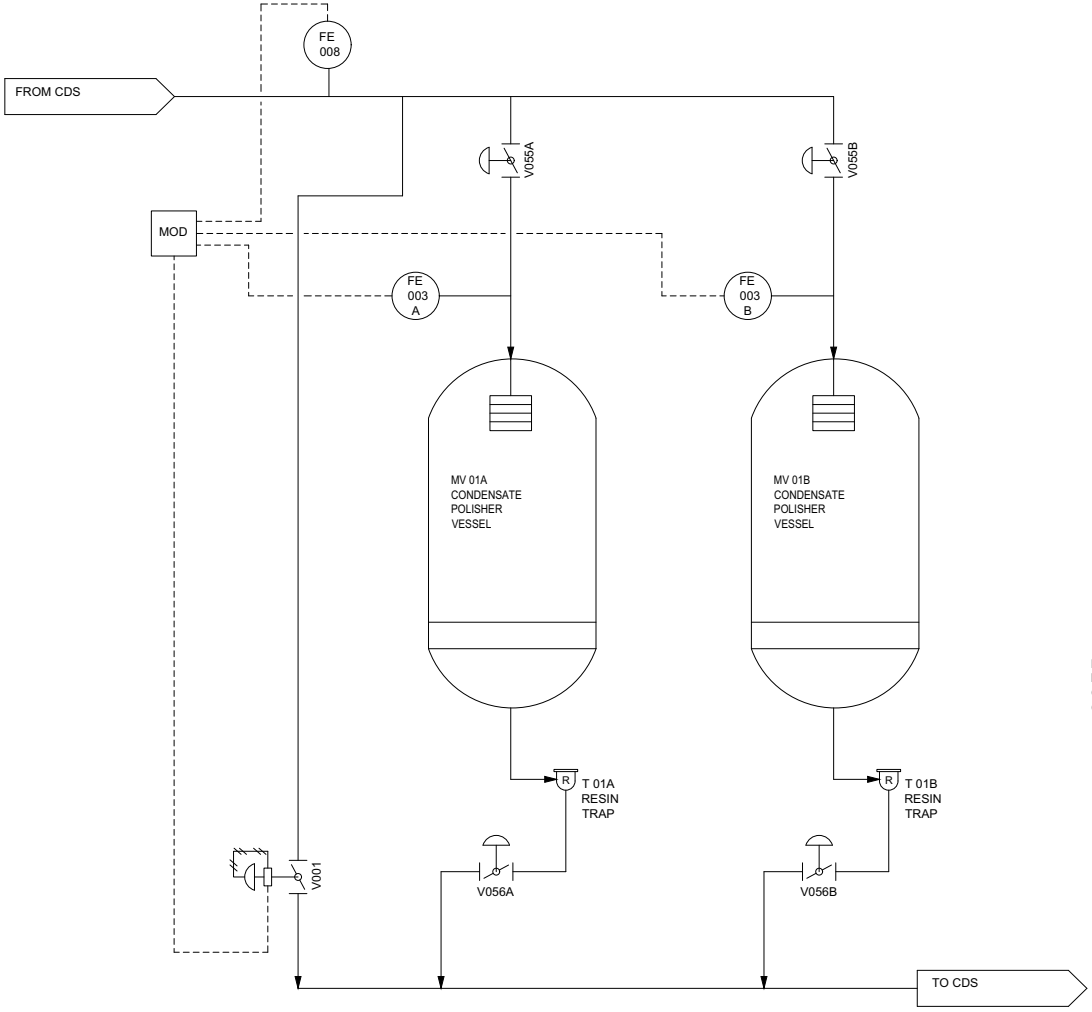

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Inside Turbine Building

Figure 10.4.6-1
Condensate Polishing System
Piping and Instrumentation Diagram (Typical)

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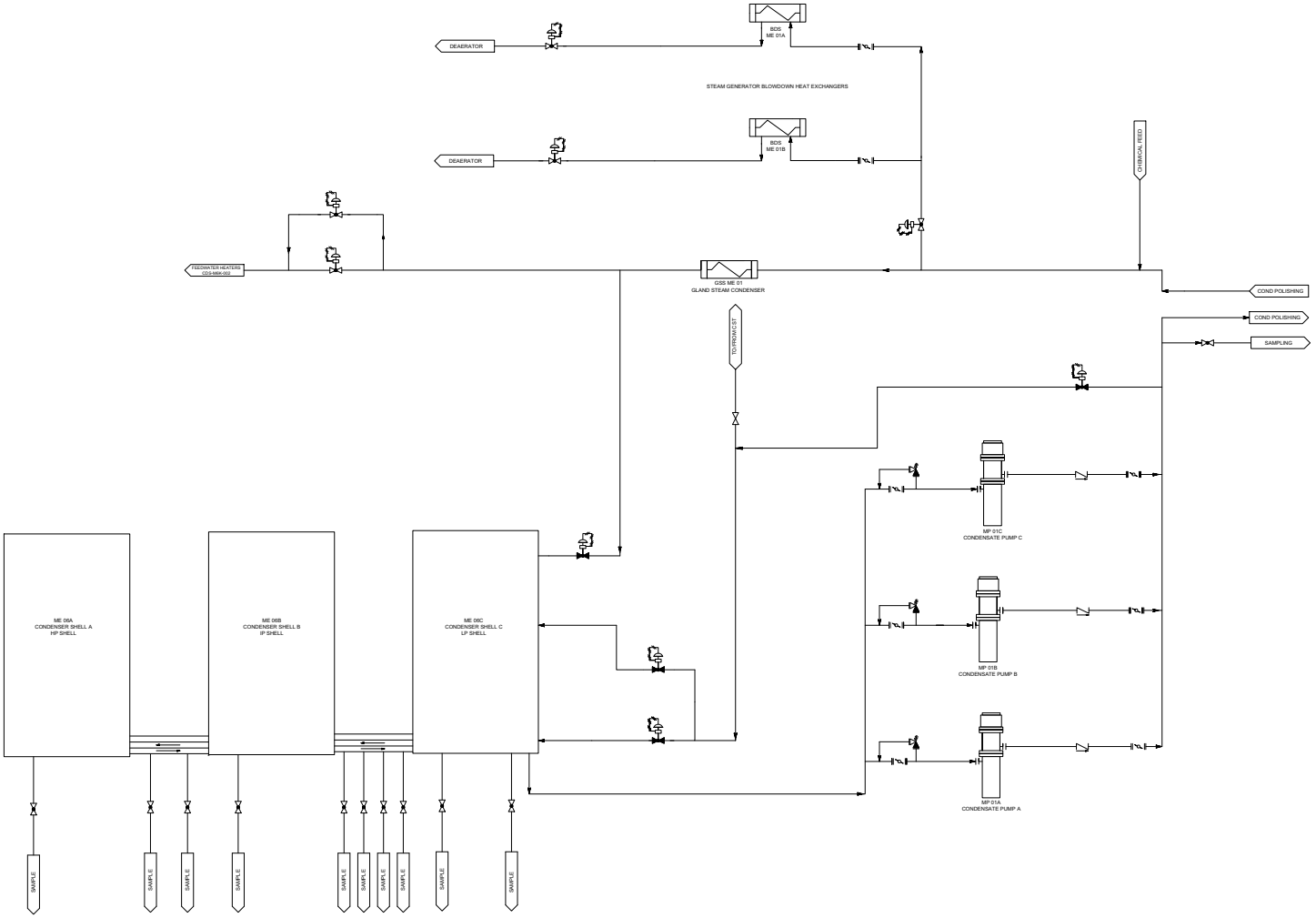

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Figure 10.4.7-1 (Sheet 1 of 4)
Condensate and Feedwater System
Piping and Instrumentation Diagram

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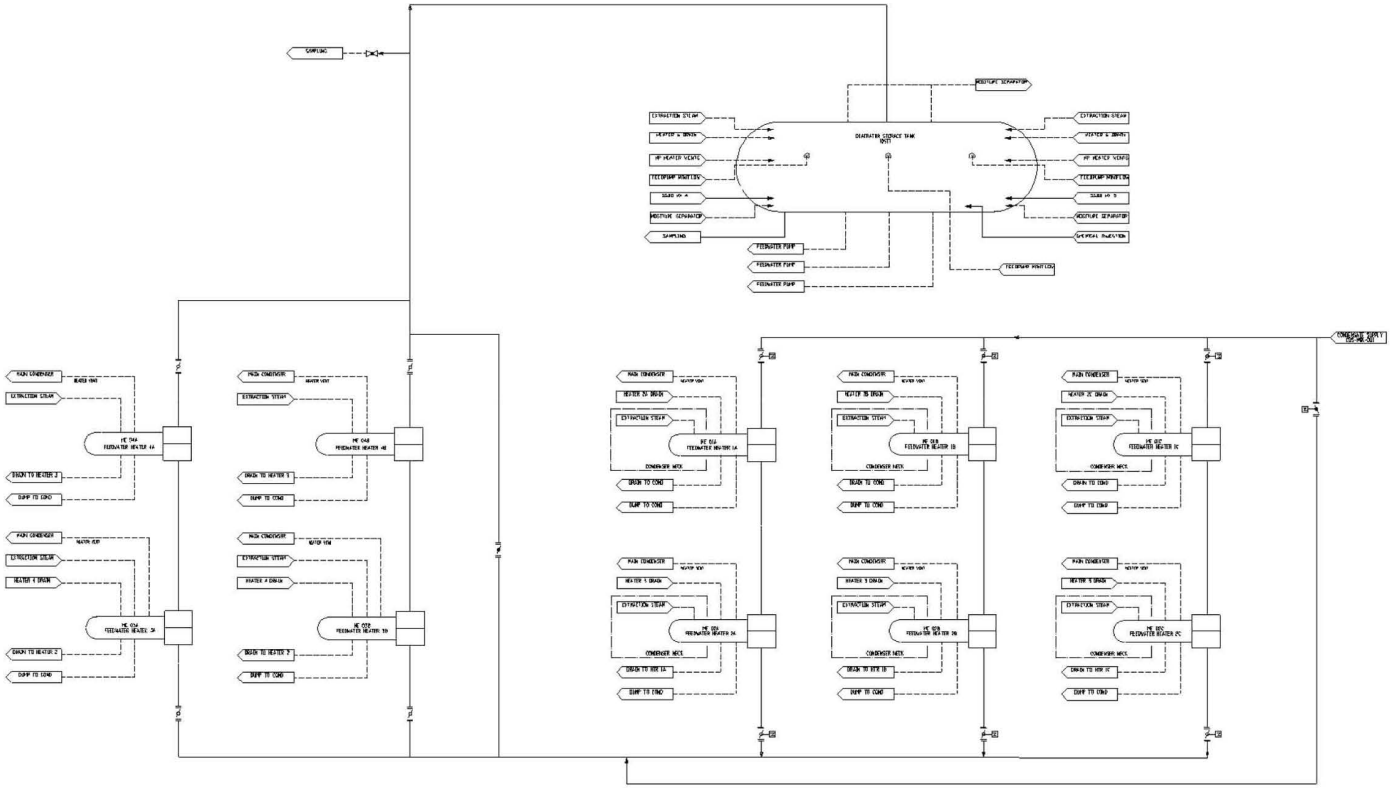

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Figure 10.4.7-1 (Sheet 2 of 4)
Condensate and Feedwater System
Piping and Instrumentation Diagram

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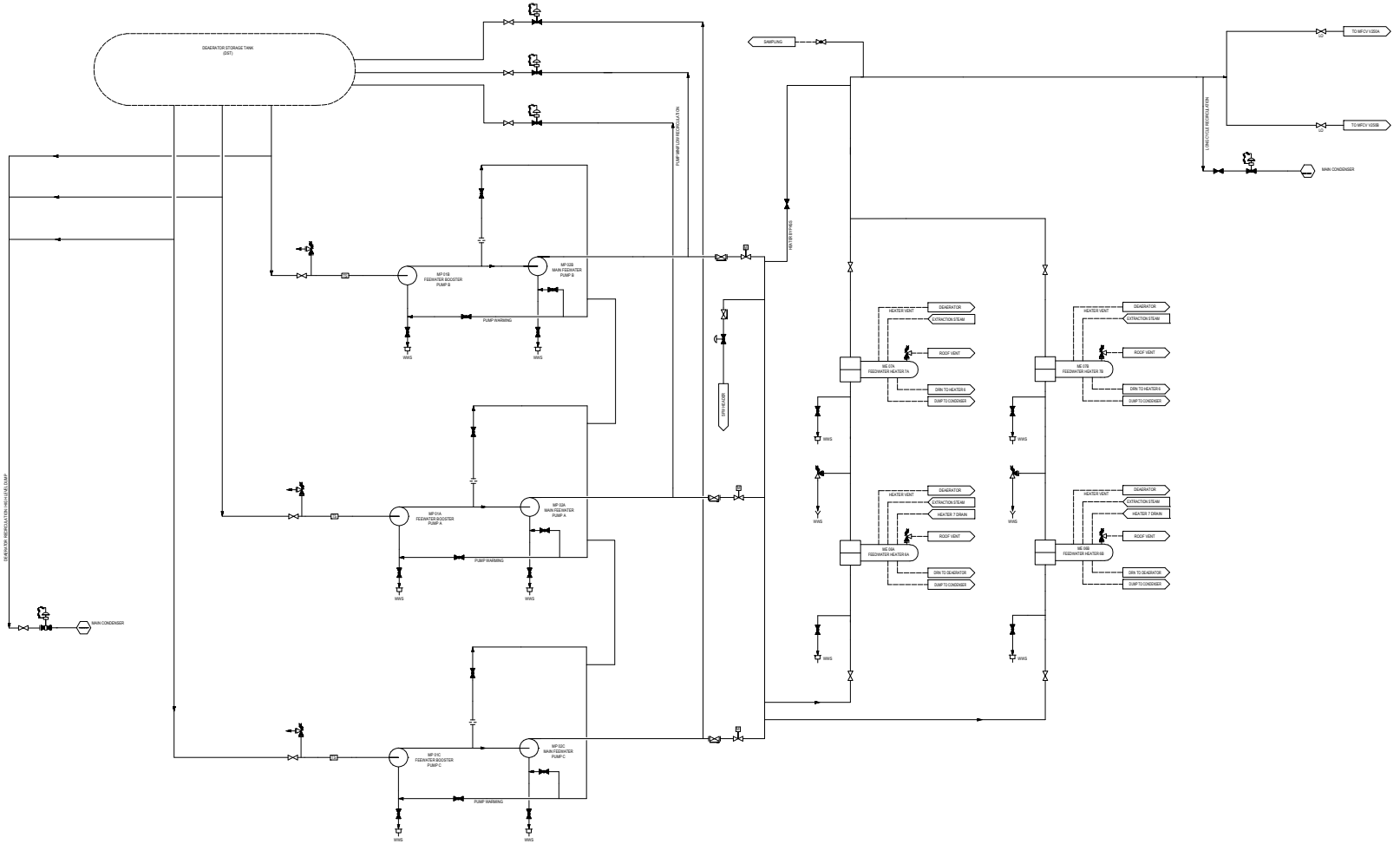

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Figure 10.4.7-1 (Sheet 3 of 4)
Condensate and Feedwater System
Piping and Instrumentation Diagram

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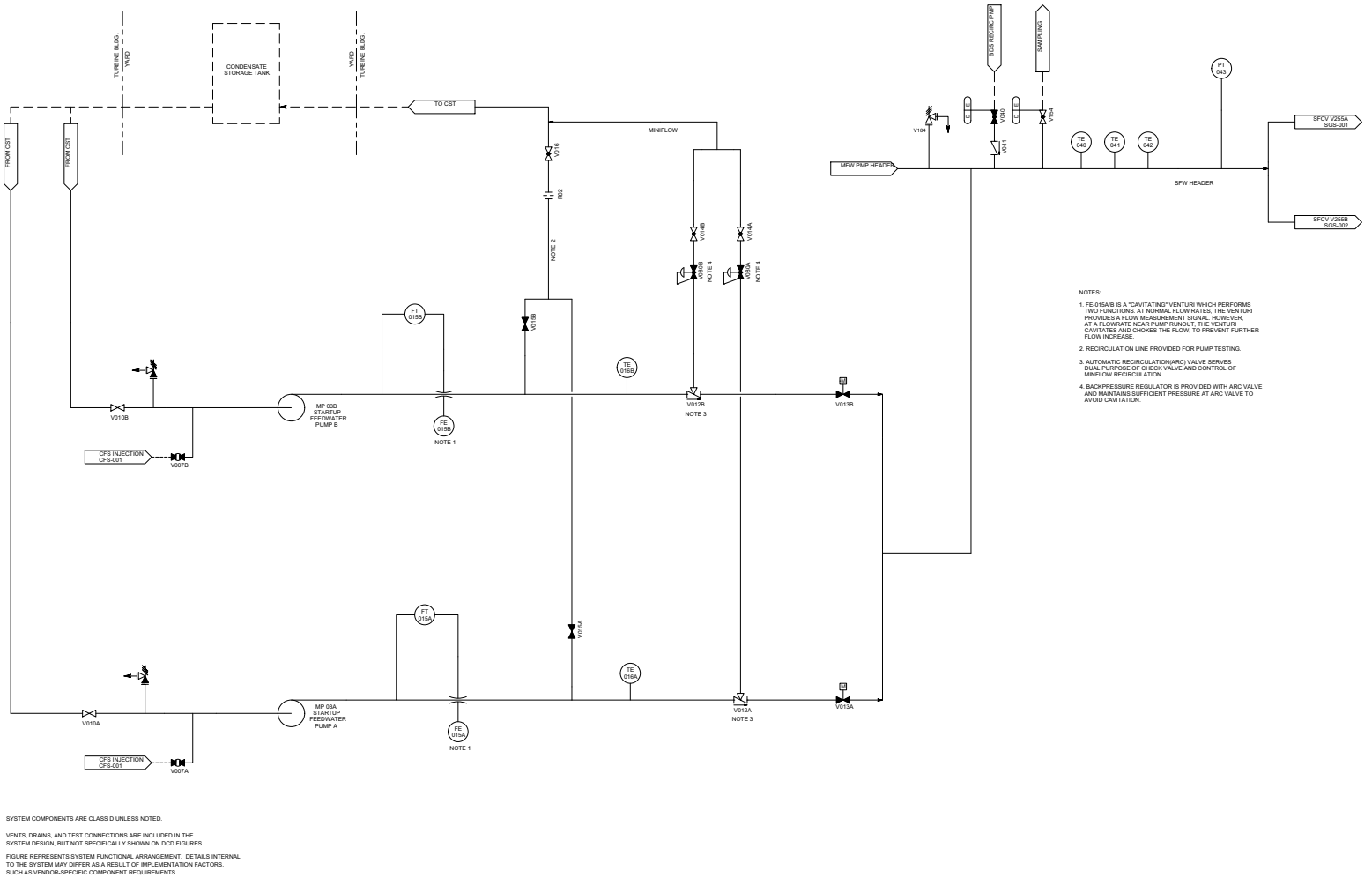

Figure 10.4.7-1 (Sheet 4 of 4)
Simplified Condensate and Feedwater System Piping and Instrumentation Diagram
(REF) FWS 002

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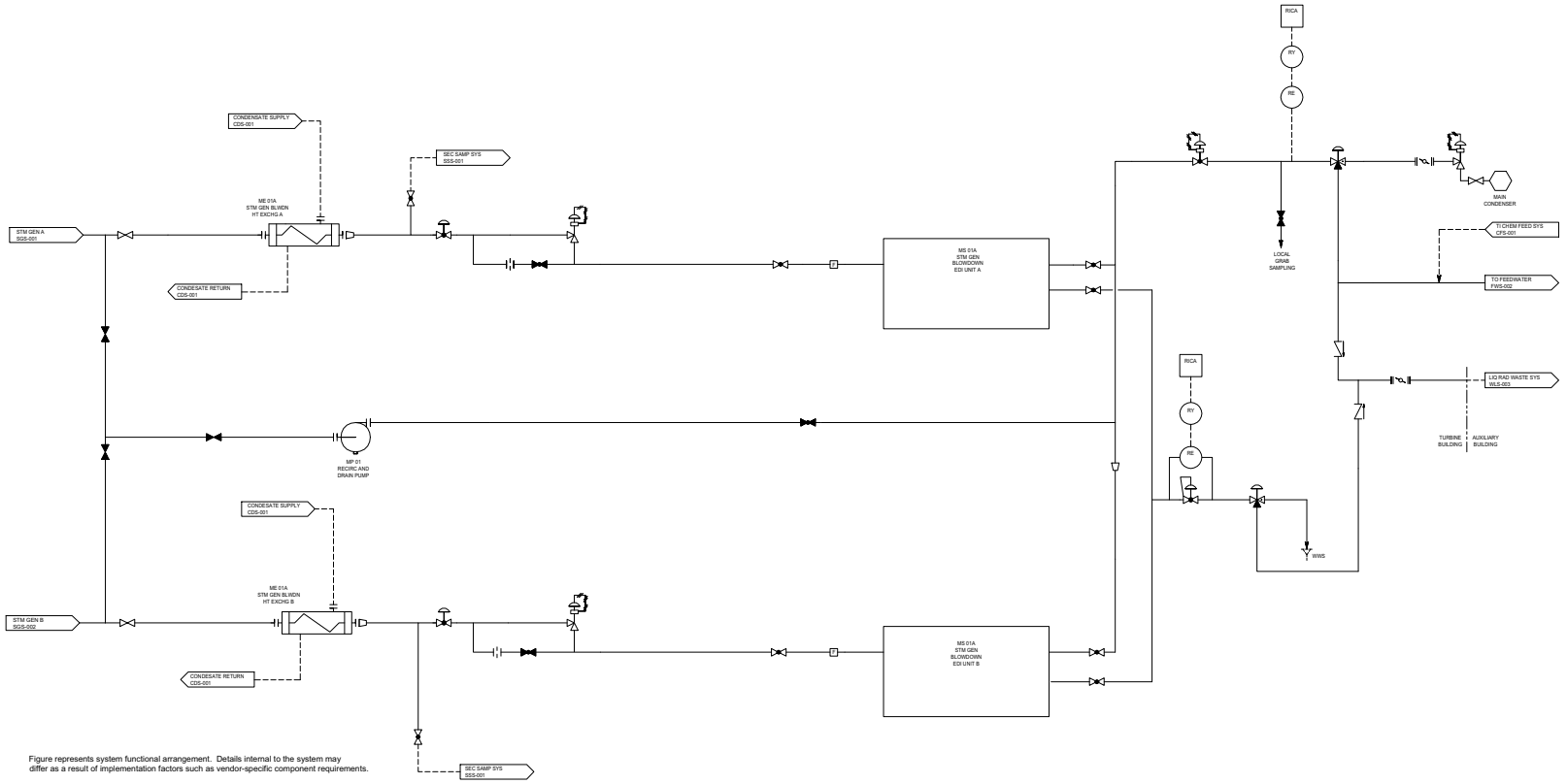

Inside Turbine Building

Figure 10.4.8-1
Steam Generator Blowdown System
Piping and Instrumentation Diagram

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Chapter 11 Radioactive Waste Management

11.1 Source Terms

This section addresses the sources of radioactivity that are treated by the liquid and gaseous radwaste systems. Radioactive materials are generated within the core (fission products) and have the potential of leaking to the reactor coolant system by way of defects in the fuel cladding. The core radiation field also results in activation of the coolant to form N-16 from oxygen and the activation of corrosion products in the reactor coolant system.

Two source terms are presented for the primary and the secondary coolant. The first is a conservative, or design basis, source term that assumes the design basis fuel defect level. This source term serves as a basis for system design and shielding requirements.

The second source term is a realistic model. This source term represents the expected average concentrations of radionuclides in the primary and the secondary coolant. These values are determined using the model in the PWR-GALE code (Reference 1) and which provides the bases for estimating typical concentrations of the principal radionuclides that are expected to occur. This source term model reflects the industry experience at a large number of operating PWR plants.

11.1.1 Design Basis Reactor Coolant Activity

11.1.1.1 Fission Products

For the design basis source term it is assumed that there is a significant fuel defect level, well above that anticipated during normal operation. It is assumed that small cladding defects are present in fuel rods producing 0.25 percent of the core power output (also stated as 0.25 percent fuel defects). The defects are assumed to be uniformly distributed throughout the core.

The parameters used in the calculation of the reactor coolant fission product concentrations, including pertinent information concerning the fission product escape rate coefficients, coolant cleanup rate, and demineralizer effectiveness, are listed in Table 11.1-1. Since the fuel defects are assumed to be uniformly distributed in the core, the fission product escape rate coefficients are based on average fuel temperature.

The determination of reactor coolant activity is based on time-dependent fission product core inventories that are calculated by the ORIGEN code (Reference 2).

The fission product activity in the reactor coolant is calculated using the following differential equations.

For parent nuclides in the coolant:

$$\frac{dN_{cp}}{dt} = \frac{FR_p\, N_{Fp}}{M_c}$$

$$-\left[\lambda_p + D_p + \frac{Q_L}{M_c}\left(\frac{DF_p - 1}{DF_p}\right)\right] N_{cp}$$

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For daughter nuclides in the coolant:

$$\frac{dN_{cd}}{dt} = \frac{FR_d N_{Fd}}{M_c} + f_p \lambda_p N_{c_p} -$$

$$\left[\lambda_d + D_d + \frac{Q_L}{M_c}\left(\frac{DF_d - 1}{DF_d}\right)\right] N_{cd}$$

where:

N_c = Concentration of nuclide in the reactor coolant (atoms/gram)

N_F = Population of nuclide in the fuel (atoms)

t = Operating time (seconds)

R = Nuclide release coefficient (1/sec)

F = Fraction of fuel rods with defective cladding

M_c = Mass of reactor coolant (grams)

λ = Nuclide decay constant (1/sec)

D = Dilution coefficient by feed and bleed (1/sec) $= \frac{\beta}{B_o - \beta t} \times \frac{1}{DF}$

B_o = Initial boron concentration (ppm)

β = Boron concentration reduction rate (ppm/sec)

DF = Nuclide demineralizer decontamination factor

Q_L = Purification or letdown mass flow rate (grams/sec)

f = Fraction of parent nuclide decay events that result in the formation of the daughter nuclide

Subscript p refers to the parent nuclide.

Subscript d refers to the daughter nuclide.

Table 11.1-2 lists the resulting reactor coolant radionuclide concentrations. The values presented are the maximum values calculated to occur during the fuel cycle from startup through the equilibrium cycle. Thus, the source term does not represent any particular time in the fuel cycle but is a conservative composite.

The design basis source term based on 0.25 percent fuel defects is used to ensure a consistent set of design values for interfaces among the radioactive waste processing systems. The Technical Specifications, which are related to fuel failure are also based upon 0.25 percent fuel defects. In

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addition, the liquid and gaseous radioactive waste processing systems have the capability to process wastes based upon 1.0 percent fuel defects.

11.1.1.2 Corrosion Products

The reactor coolant corrosion product activities are based on operating plant data and are independent of fuel defect level. The concentrations of corrosion products are included in Table 11.1-2.

11.1.1.3 Tritium

A number of tritium production processes add tritium to the reactor coolant:

- Fission product formation in the fuel (ternary fission) forms tritium which can diffuse through the fuel clad or leak through fuel clad defects
- Neutron reactions with soluble boron in the reactor coolant
- Burnable neutron absorber
- Neutron reactions with soluble lithium in the reactor coolant
- Neutron reactions with deuterium in the reactor coolant
- Neutron reactions with secondary sources

The production in the coolant through neutron reactions with soluble boron and the transfer from secondary sources through neutron reactions with lithium are the principle contributors of tritium in the reactor coolant. Table 11.1-3 contains the total conservative estimated tritium activity released into the reactor coolant each cycle.

The first two processes are the principal contributors to tritium in the reactor coolant. Table 11.1-3 lists the tritium introduced to the reactor coolant from each of the processes.

Tritium exists in the reactor coolant primarily combined with hydrogen (that is, a tritium atom replaces a hydrogen atom in a water molecule) and thus cannot be readily separated from the coolant by normal processing methods. The maximum concentration of tritium in the reactor coolant is less than 3.5 microcuries per gram as a result of losses due to leakage and the controlled release of tritiated water to the environment.

11.1.1.4 Nitrogen-16

Activation of oxygen in the coolant results in the formation of N-16 which is a strong gamma emitter. Because of its short half-life of 7.11 seconds, N-16 is not of concern outside the containment. Table 12.2-3 provides N-16 concentrations at various points in the reactor coolant system. After shutdown, N-16 is not a source of radiation inside of containment.

11.1.1.5 In-Containment Refueling Water Storage Tank

The in-containment refueling water storage tank (IRWST) liquid source term is calculated as a mass balance using the IRWST liquid as a control volume. Nuclides are produced via reactor coolant system inleakage (via passive residual heat removal heat exchanger and refueling cavity drain-down) and decay of parent radionuclides. The reactor coolant leakage into the IRWST is assumed to have radionuclide concentrations as defined in Table 11.1-2. Nuclides are removed via

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spent fuel pool cooling system cleanup, radioactive decay, draining (overflow) to the sump, evolution of gaseous radionuclides, and settling within the IRWST. Nuclides that are removed from the tank liquid via evolution into the air space or settling to the tank floor are tracked independently using similar equations.

11.1.2 Design Basis Secondary Coolant Activity

Steam generator tube defects cause the introduction of reactor coolant into the secondary cooling system. The resulting radionuclide concentrations in the secondary coolant depend upon the primary-to-secondary leak rate, the nuclide decay constant, and the steam generator blowdown rate.

The reactor coolant leakage into the secondary system is assumed to have radionuclide concentrations as defined in Table 11.1-2. The parameters used in the calculation of the secondary side activities are provided in Table 11.1-4 and the resulting radionuclide concentrations in the steam generator secondary side water and steam are presented in Tables 11.1-5 and 11.1-6.

11.1.3 Realistic Reactor Coolant and Secondary Coolant Activity

The realistic source terms for both the reactor coolant and the secondary coolant are determined using the modeling in ANSI-18.1 (Reference 3). This modeling is also incorporated into a version of the PWR-GALE code modified to accept inputs from an updated version of ANSI-18.1. The reference plant values provided in ANSI-18.1 were adjusted to be consistent with the AP1000 parameters listed in Table 11.1-7. The adjustment factors are applied to the fission products. The realistic source term is listed in Table 11.1-8.

11.1.4 Core Source Term

The core fission product inventories used to establish source terms for accident radiological consequence analyses are provided in Appendix 15A.

11.1.5 Process Leakage Sources

The systems containing radioactive liquids are potential sources for the release of radioactive material to plant buildings and then to the environment. The leakage sources and the resulting airborne concentrations are discussed in Section 12.2.

Release pathways for radioactive materials are discussed in Sections 11.2 and 11.3.

11.1.6 Combined License Information

This section contained no requirement for information.

11.1.7 References

1. “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code),” NUREG-0017, Revision 1, March 1985.

2. RSIC Computer Code Collection CCC-371, ORIGEN 2.1 Isotope Generation and Depletion Code - Matrix Exponential Method, August 1, 1991.

3. ANSI/ANS-18.1-1999, “Radioactive Source Term for Normal Operation of Light Water Reactors.”

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Table 11.1-1 (Sheet 1 of 2)
Parameters Used in the Calculation of Design Basis Fission Product Activities

Core thermal power (MWt)	3,400
Reactor coolant liquid volume (ft^3)[(a)]	9,575
Reactor coolant full-power average temperature (°F)	578.1
Purification flow rate (gal/min)[(b)]	
Maximum	100
Normal	91.3
Effective cation demineralizer flow, annual average (gal/min)[(b)]	9.1
Nuclide release coefficients (the product of the failed fuel fraction and the fission product escape rate coefficient)	
Equivalent fraction of core power produced by fuel rods containing small cladding defects (failed fuel fraction)	0.0025
Fission product escape rate coefficients during full-power operation (s^{-1}):	
Kr and Xe isotopes Br, Rb, I, and Cs isotopes Mo, Tc, and Ag isotopes Te isotopes Sr and Ba isotopes Y, Zr, Nb, Ru, Rh, La, Ce, and Pr isotopes	6.5×10^{-8} 1.3×10^{-8} 2.0×10^{-9} 1.0×10^{-9} 1.0×10^{-11} 1.6×10^{-12}
Chemical and volume control system mixed bed demineralizers	
Resin volume (ft^3)	50
Demineralizer isotopic decontamination factors:	
Kr and Xe isotopes Br and I isotopes Sr and Ba isotopes Other isotopes	1 10 10 1

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Table 11.1-1 (Sheet 2 of 2)
Parameters Used in the Calculation of Design Basis Fission Product Activities

Chemical and volume control system cation bed demineralizer	
Resin volume (ft^3)	50
Demineralizer isotopic decontamination factors:	
Kr and Xe isotopes Sr and Ba isotopes Rb-86, Cs-134, and Cs-137 Rb-88, Rb-89, Cs-136, and Cs-138 Other isotopes	1 1 10 1 1
Other isotopic removal mechanisms	See Note c.
Initial boron concentration (ppm)	1,400
Operation time (effective full-power hours)	12,492

Notes:

a. Reactor coolant mass used in defining fission product activities is based on above stated conditions before thermal expansion (conservative).

b. Flow calculated at 2250 psia and 250°F.

c. For all isotopes, except the isotopes of Kr, Xe, Br, I, Rb, Cs, Sr, and Ba, a removal decontamination factor of 10 is assumed to account for removal mechanisms other than ion exchange, such as plateout or filtration. This decontamination factor is applied to the normal purification letdown flow.

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Table 11.1-2
Design Basis Reactor Coolant Activity

Nuclide	Activity (μCi/g)	Nuclide	Activity (μCi/g)
Kr-83m	1.8×10^{-1}	Rb-88	1.5
Kr-85m	8.4×10^{-1}	Rb-89	6.9×10^{-2}
Kr-85	3.0	Sr-89	1.1×10^{-3}
Kr-87	4.7×10^{-1}	Sr-90	4.9×10^{-5}
Kr-88	1.5	Sr-91	1.7×10^{-3}
Kr-89	3.5×10^{-2}	Sr-92	4.1×10^{-4}
Xe-131m	1.3	Y-90	1.3×10^{-5}
Xe-133m	1.7	Y-91m	9.2×10^{-4}
Xe-133	1.2×10^{2}	Y-91	1.4×10^{-4}
Xe-135m	1.7×10^{-1}	Y-92	3.4×10^{-4}
Xe-135	3.5	Y-93	1.1×10^{-4}
Xe-137	6.7×10^{-2}	Zr-95	1.6×10^{-4}
Xe-138	2.5×10^{-1}	Nb-95	1.6×10^{-4}
Br-83	3.2×10^{-2}	Mo-99	2.1×10^{-1}
Br-84	1.7×10^{-2}	Tc-99m	2.0×10^{-1}
Br-85	2.0×10^{-3}	Ru-103	1.4×10^{-4}
I-129	1.5×10^{-8}	Rh-103m	1.4×10^{-4}
I-130	1.1×10^{-2}	Rh-106	4.5×10^{-5}
I-131	7.1×10^{-1}	Ag-110m	4.0×10^{-4}
I-132	9.4×10^{-1}	Te-127m	7.6×10^{-4}
I-133	1.3	Te-129m	2.6×10^{-3}
I-134	2.2×10^{-1}	Te-129	3.8×10^{-3}
I-135	7.8×10^{-1}	Te-131m	6.7×10^{-3}
Cs-134	6.9×10^{-1}	Te-131	4.3×10^{-3}
Cs-136	1.0	Te-132	7.9×10^{-2}
Cs-137	5.0×10^{-1}	Te-134	1.1×10^{-2}
Cs-138	3.7×10^{-1}	Ba-137m	4.7×10^{-1}
Cr-51	1.3×10^{-3}	Ba-140	1.0×10^{-3}
Mn-54	6.7×10^{-4}	La-140	3.1×10^{-4}
Mn-56	1.7×10^{-1}	Ce-141	1.6×10^{-4}
Fe-55	5.0×10^{-4}	Ce-143	1.4×10^{-4}
Fe-59	1.3×10^{-4}	Pr-143	1.5×10^{-4}
Co-58	1.9×10^{-3}	Ce-144	1.2×10^{-4}
Co-60	2.2×10^{-4}	Pr-144	1.2×10^{-4}

Note:
These activities are used for shielding and radwaste system interface design. For 1 percent fuel defect calculations (maximum release and liquid and gaseous radwaste system capability) multiply the activities above by 4 except for iodines, noble gases, and corrosion products (Cr-51, Mn-54, Mn-56, Fe-55, Fe-59, Co-58 and Co-60).

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Table 11.1-3
Tritium Sources

Conservative Estimated Tritium Release into Coolant (curies/cycle(1))	2672

Note:
1. Cycle length of 18 months

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Table 11.1-4
Parameters Used to Calculate Secondary Coolant Activity

Total secondary side water mass (lb/steam generator)	1.68×10^5
Steam generator steam fraction	0.058
Total steam flow rate (lb/hr)	1.5×10^7
Moisture carryover (percent)	0.1
Total makeup water feed rate (lb/hr)	700
Total blowdown rate (gpm)	186
Total primary-to-secondary leak rate (gpd)	300
Iodine partition factor (mass basis)	100

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Table 11.1-5
Design Basis Steam Generator Secondary Side Liquid Activity

Nuclide	Activity (µCi/g)	Nuclide	Activity (µCi/g)
Br-83	1.4E-05	Y-91m	1.0E-06
Br-84	2.4E-06	Y-91	1.3E-07
Br-85	3.1E-08	Y-92	2.8E-07
I-129	1.3E-11	Y-93	8.2E-08
I-130	7.9E-06	Zr-95	1.5E-07
I-131	6.3E-04	Nb-95	1.5E-07
I-132	4.2E-04	Mo-99	1.9E-04
I-133	1.0E-03	Tc-99m	1.7E-04
I-134	4.9E-05	Ru-103	1.2E-07
I-135	5.0E-04	Ru-106	4.1E-08
Rb-86	1.4E-05	Rh-103m	1.2E-07
Rb-88	1.4E-04	Rh-106	4.1E-08
Rb-89	5.6E-06	Ag-110m	4.0E-07
Cs-134	1.1E-03	Te-125m	1.5E-07
Cs-136	1.7E-03	Te-127m	7.0E-07
Cs-137	8.2E-04	Te-127	2.2E-06
Cs-138	5.9E-05	Te-129m	2.4E-06
H-3	3.8E-01	Te-129	2.1E-06
Cr-51	1.3E-06	Te-131m	5.6E-06
Mn-54	6.6E-07	Te-131	1.6E-06
Mn-56	7.8E-05	Te-132	7.0E-05
Fe-55	5.0E-07	Te-134	2.0E-06
Fe-59	1.3E-07	Ba-137m	7.7E-04
Co-58	1.9E-06	Ba-140	9.4E-07
Co-60	2.2E-07	La-140	3.3E-07
Sr-89	1.8E-06	Ce-141	1.4E-07
Sr-90	8.0E-08	Ce-143	1.2E-07
Sr-91	1.9E-06	Ce-144	1.1E-07
Sr-92	2.4E-07	Pr-143	1.4E-07
Y-90	1.4E-08	Pr-144	1.1E-07

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Table 11.1-6
Design Basis Steam Generator Secondary Side Steam Activity

Nuclide	Activity (μCi/g)
Kr-83m	1.10E-06
Kr-85m	4.30E-06
Kr-85	1.50E-05
Kr-87	2.40E-06
Kr-88	7.70E-06
Kr-89	1.80E-07
Xe-131m	6.90E-06
Xe-133m	8.70E-06
Xe-133	6.40E-04
Xe-135m	5.50E-06
Xe-135	1.90E-05
Xe-137	3.40E-07
Xe-138	1.30E-06
I-129	1.50E-13
I-130	8.70E-08
I-131	6.90E-06
I-132	4.70E-06
I-133	1.10E-05
I-134	5.40E-07
I-135	5.50E-06
H-3	3.80E-01

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Table 11.1-7
Parameters Used to Describe Realistic Sources

Parameter	Symbol	Units	AP1000 Value	Nominal Value
Thermal power	P	MWt	3400	3400
Steam flow rate	FS	lb/hr	1.5×10^7	1.5×10^7
Weight of water in reactor coolant system	WP	lb	4.3×10^5	5.5×10^5
Weight of water in all steam generators	WS	lb	3.5×10^5	4.5×10^5
Reactor coolant purification flow	FD	lb/hr	4.3×10^4	3.7×10^4
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/hr	1.5×10^2	5.0×10^2
Steam generator blowdown flow (total)	FBD	lb/hr	7.5×10^4	7.5×10^4
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system	NBD	-	0.0	1.0
Flow through the purification system cation demineralizer	FA	lb/hr	4.3×10^3	3.7×10^3
Ratio of condensate demineralizer flow rate to the total steam flow rate	NC	-	0.33	0.0
Fraction of the noble gas activity in the letdown stream which is not returned to the reactor coolant system	Y	-	0.0	0.0
Primary-to-secondary leakage	FL	lb/day	75	75

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Table 11.1-8 (Sheet 1 of 4)
Realistic Source Terms

Noble Gases		
Nuclide	**Reactor Coolant Activity (µCi/g)**	**Steam Generator Steam Activity (µCi/g)**
Kr-85m	0.021	4.40E-09
Kr-85	1.4	2.90E-07
Kr-87	0.022	4.4E-09
Kr-88	0.023	4.9E-09
Xe-131m	1.1	2.3E-07
Xe-133m	0.093	2.0E-08
Xe-133	0.041	8.4E-09
Xe-135m	0.17	3.5E-08
Xe-135	0.087	1.8E-08
Xe-137	0.044	9.2E-09
Xe-138	0.078	1.7E-08

Halogens			
Nuclide	**Reactor Coolant Activity (µCi/g)**	**Steam Generator Liquid Activity (µCi/g)**	**Steam Generator Steam Activity (µCi/g)**
Br-84	2.0E-02	1.2E-07	1.2E-09
I-131	1.8E-03	1.2E-07	1.2E-09
I-132	7.1E-02	1.5E-06	1.5E-08
I-133	2.5E-02	1.4E-06	1.4E-08
I-134	1.2E-01	1.2E-06	1.2E-08
I-135	6.0E-02	2.3E-06	2.3E-08

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Table 11.1-8 (Sheet 2 of 4)
Realistic Source Terms

Rubidium, Cesium			
Nuclide	**Reactor Coolant Activity (µCi/g)**	**Steam Generator Liquid Activity (µCi/g)**	**Steam Generator Steam Activity (µCi/g)**
Rb-88	0.24	8.9E-07	4.4E-09
Cs-134	3.1E-05	7.6E-09	4.0E-11
Cs-136	7.4E-04	1.7E-07	8.7E-10
Cs-137	4.4E-05	1.1E-08	5.4E-11
Tritium			
Nuclide	**Reactor Coolant Activity (µCi/g)**	**Steam Generator Liquid Activity (µCi/g)**	**Steam Generator Steam Activity (µCi/g)**
H-3	1	1.0E-3	1.0E-3

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Table 11.1-8 (Sheet 3 of 4)
Realistic Source Terms

Miscellaneous Nuclides			
Nuclide	**Reactor Coolant Activity (µCi/g)**	**Steam Generator Liquid Activity (µCi/g)**	**Steam Generator Steam Activity (µCi/g)**
Na-24	4.6E-02	3.6E-06	1.8E-08
Cr-51	2.6E-03	3.6E-07	1.8E-09
Mn-54	1.3E-03	1.8E-07	9.2E-10
Fe-55	1.0E-03	1.4E-07	7.0E-10
Fe-59	2.5E-04	3.3E-08	1.7E-10
Co-58	4.6E-02	1.9E-06	9.4E-09
Co-60	4.4E-04	6.1E-08	3.1E-10
Zn-65	4.3E-04	5.9E-08	2.8E-10
Sr-89	1.2E-04	1.6E-08	8.1E-11
Sr-90	1.0E-05	1.4E-09	6.7E-12
Sr-91	9.9E-04	6.4E-08	3.2E-10
Y-91m	5.7E-04	5.6E-09	2.8E-11
Y-91	4.4E-06	5.9E-10	3.1E-12
Y-93	4.3E-03	2.8E-07	1.4E-09
Zr-95	3.3E-04	4.5E-08	2.2E-10
Nb-95	2.4E-04	3.1E-08	1.6E-10
Mo-99	5.6E-03	6.7E-07	3.2E-09
Tc-99m	5.1E-03	2.4E-07	1.2E-09
Ru-103	6.3E-03	8.6E-07	4.5E-09
Ru-106	7.5E-02	1.0E-05	5.0E-08
Ag-110m	1.1E-03	1.5E-07	7.5E-10
Te-129m	1.6E-04	2.2E-08	1.1E-10

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Table 11.1-8 (Sheet 4 of 4)
Realistic Source Terms

Miscellaneous Nuclides			
Nuclide	**Reactor Coolant Activity (µCi/g)**	**Steam Generator Liquid Activity (µCi/g)**	**Steam Generator Steam Activity (µCi/g)**
Te-129	2.9E-02	3.9E-07	2.0E-09
Te-131m	1.4E-03	1.4E-07	7.0E-10
Te-131	9.7E-03	4.9E-08	2.5E-10
Te-132	1.5E-03	1.8E-07	8.9E-10
Ba-137m	4.4E-05	1.1E-08	5.4E-11
Ba-140	1.1E-02	1.4E-06	7.2E-09
La-140	2.3E-02	2.4E-06	1.2E-08
Ce-141	1.3E-04	1.7E-08	8.6E-11
Ce-143	2.6E-03	3.6E-07	1.3E-09
Ce-144	3.4E-03	4.5E-07	2.3E-09
W-187	2.3E-03	2.2E-07	1.1E-09
Np-239	2.0E-03	2.2E-07	1.1E-09

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11.2 Liquid Waste Management Systems

The liquid waste management systems include the systems that may be used to process and dispose of liquids containing radioactive material. These include the following:

- Steam generator blowdown processing system (Subsection 10.4.8);
- Radioactive waste drain system (Subsection 9.3.5);
- Liquid radwaste system (WLS) (Section 11.2).

This section primarily addresses the liquid radwaste system. The other systems are also addressed in Subsection 11.2.3, which discusses the expected releases from the liquid waste management systems.

The liquid radwaste system is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

11.2.1 Design Basis

Subsection 1.9.1 discusses the conformance of the liquid radwaste system design with the criteria of Regulatory Guide 1.143.

11.2.1.1 Safety Design Basis

The liquid radwaste system serves no safety-related functions except for:

- Containment isolation; see Subsection 6.2.3.

- Back flow prevention check valves in the drain lines from the chemical and volume control system compartment and the passive core cooling system compartments to the containment sump, which prevent cross flooding of these compartments. Each drain line has two check valves in series so that a single failure does not compromise the back flow prevention safety function. See Subsection 6.3.3.3.2 for a discussion of containment flooding.

11.2.1.2 Power Generation Design Basis

11.2.1.2.1 Capacity

The liquid radwaste system provides holdup capacity as shown in Table 11.2-2, and permanently installed processing capacity of 75 gpm through the ion exchange/filtration train. This is adequate capacity to meet the anticipated processing requirements of the plant. The projected flows of various liquid waste streams to the liquid radwaste system under normal conditions are identified in Table 11.2-1.

The liquid radwaste system design can accept equipment malfunctions without affecting the capability of the system to handle both anticipated liquid waste flows and possible surge load due to excessive leakage. Table 11.2-4 contains information on the surge capacity of individual tanks.

Portions of the liquid radwaste system may become unavailable as a result of the malfunctions listed in Subsection 11.2.1.2.2.

Ample surge capacity of the system, provisions for using mobile processing equipment and the low load factor of the processing equipment permits the system to accommodate waste until failures can

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be repaired and normal plant operation resumed. In addition, the liquid radwaste system is designed to accommodate the anticipated operational occurrences described in Subsection 11.2.1.2.3.

11.2.1.2.2 Failure Tolerance

11.2.1.2.2.1 Pump Failure

Where operation is not essential and surge capacity is available, a single pump is provided. This applies to most applications in the liquid radwaste system. Two reactor coolant drain tank pumps and two containment sump pumps are provided because the relative inaccessibility of the containment during power operation would hinder maintenance. The containment sump pumps are submersible pumps. To protect them from damage due to loss of suction, each pump is interlocked to stop on a low level condition in the sump. The reactor coolant drain tank pumps are vertical sump type pumps with motors above the reactor coolant drain tank shaft coupled to pumps submersed in the liquid within the reactor coolant drain tank. This arrangement minimizes contamination of the motors and permits removal and maintenance of the motors outside of the radiation area.

Process pumps located outside containment are air-operated, double diaphragm type. These pumps are capable of significant suction lifts, and can thus be located on or near the top of the associated waste tank, with internal suction piping. They can pump slurries with high solids fractions, run deadheaded, and run dry without damage. In addition, they can operate over a wide range of hydraulic conditions by varying the driving air input. This makes it possible to fulfill many different applications with a single pump model, thereby facilitating maintenance and reducing the inventory of spare parts.

11.2.1.2.2.2 Filter or Ion Exchanger Plugging

Instrumentation is provided to give indication of the pressure drop across filters and ion exchangers. Periodic checks of the pressure drops provide indication of equipment fouling, thus permitting corrective action to be taken before an excessive pressure drop is reached. Change of filter cartridges and ion exchange beds is expected to occur based upon radiation survey.

11.2.1.2.3 Anticipated Operational Occurrences

11.2.1.2.3.1 High Primary Coolant System Leakage Rate

The system is designed to handle an abnormal primary coolant system leak in addition to the expected leakage during normal operation. Operation of the system is the same as for normal operation, except that the load on the system is increased.

11.2.1.2.3.2 High Use of Decontamination Water

If large quantities of water are used to decontaminate areas or equipment, the load on the liquid radwaste system is increased. However, the liquid radwaste system is designed to handle a large, continuous input to the waste holdup tanks. If the water can be discharged without processing based on sampling which shows acceptably low activity, the overall liquid radwaste system capacity is increased.

To accommodate the possible use of special decontamination fluids or very large volumes of decontamination fluids, mobile equipment is used as discussed in Subsection 11.2.1.2.5.2.

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11.2.1.2.3.3 Steam Generator Tube Leakage

During normal operations, steam generator blowdown is returned to the condensate system, as described in Subsection 10.4.8. However, if excessive radioactivity is detected, the blowdown is diverted to the liquid radwaste system for processing and disposal.

The blowdown fluid is brought into the waste holdup tanks, which provide some surge capacity to hold the fluid during processing. It is then processed in the same fashion as, and combined with, other inputs.

In the event of a steam generator tube rupture, the condensate storage tank may also become contaminated. In this event, the tank is cleaned by the use of temporary equipment brought to the site for the purpose, as described in Subsection 11.2.1.2.5.2.

11.2.1.2.3.4 Refueling

The load on the liquid radwaste system is expected to increase during refueling because of the increased level of maintenance activities in the plant, but operation is the same as for normal plant operation. There is no significant effect on the performance capability of the liquid radwaste system.

11.2.1.2.4 Controlled Release of Radioactivity

The liquid radwaste system provides the capability to reduce the amounts of radioactive nuclides released in the liquid wastes through the use of demineralization and time delay for decay of short-lived nuclides.

The assumed equipment decontamination factors appear in Table 11.2-5. Estimates of the radioactive source terms and annual average flow rate that will be processed in the liquid radwaste system or discharged to the environment during normal operation appear in Table 11.2-1.

Before radioactive liquid waste is discharged, it is pumped to a monitor tank. A sample of the monitor tank contents is analyzed, and the results are recorded. In this way, a record is kept of planned releases of radioactive liquid waste.

The liquid waste is discharged from the monitor tank in a batch operation, and the discharge flow rate is restricted as necessary to maintain an acceptable concentration when diluted by the circulating water discharge flow. These provisions preclude uncontrolled releases of radioactivity.

In addition, the discharge line contains a radiation monitor with diverse methods of stopping the discharge. The first method closes an isolation valve in the discharge line, which prevents any further discharge from the liquid radwaste system. The valve automatically closes and an alarm is actuated if the activity in the discharge stream reaches the monitor setpoint. The second method stops the monitor tank pumps.

To minimize leakage from the liquid radwaste system, the portion of the system normally expected to contain radioactive material is of welded construction except where flanged connections are required to facilitate component maintenance or to allow connection of temporary or mobile equipment. Air-operated diaphragm pumps, sealless pumps, or pumps having mechanical seals are used. These pumps minimize system leakage thereby minimizing the release of radioactive gas that might be entrained in the leaking fluid to the building atmosphere.

Provisions are made to control spills of radioactive liquids due to tank overflows. Table 11.2-3 lists the provisions for tank level indication, alarms, and overflow disposition for liquid radwaste system tanks outside containment. In addition, the radioactive waste collection tanks (i.e., the effluent holdup

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tanks, waste holdup tanks, and chemical tank) are located within the auxiliary building, which is well sealed and equipped with an extensive floor drain system. The radwaste monitor tanks are located in the auxiliary building and in the radwaste building, which has a well sealed, contiguous basemat with integral curbing and a floor drain system. Routing of both of the auxiliary building and radwaste building floor drain systems are to the liquid radwaste system. This eliminates the potential for undetected tank leakage to the environment and supports compliance with 10 CFR 20.1406 (Reference 5).

The liquid radwaste system is designed so that the annual average concentration limits established by 10 CFR 20 (Appendix B, table 2, column 2) (Reference 1) for liquid releases are not exceeded during plant operation. Subsection 11.2.3 describes the calculated releases of radioactive materials from the liquid radwaste system and other portions of the liquid waste management systems resulting from normal operation.

The monitored radwaste discharge pipeline is engineered to preclude leakage to the environment. This pipe is routed from the auxiliary building to the radwaste building (the short section of pipe between the two buildings is fully available for visual inspection as noted above) and then out of the radwaste building to the licensed release point for dilution and discharge. The discharge radiation monitor and isolation valve are located inside the radiologically controlled area. The exterior piping is designed to preclude inadvertent or unidentified releases to the environment; it is either enclosed within a guard pipe and monitored for leakage, or accessible for visual inspection. No valves or vacuum breakers are incorporated outside of monitored structures. This greatly reduces the potential for undetected leakage from this discharge to the environment at a non-licensed release point, and supports compliance with 10 CFR 20.1406 (Reference 5).

11.2.1.2.4.1 Abnormal Operation

Subsections 11.2.1.2.2 and 11.2.1.2.3 describe the capability of the liquid radwaste system to accommodate abnormal conditions for various equipment and other anticipated operational occurrences. During these anticipated occurrences, the effectiveness of the liquid radwaste system in controlling releases of radioactivity remains unaffected, so releases are limited as during normal operation.

Subsection 11.2.3 discusses the calculated releases of radioactive materials from the liquid radwaste system for abnormal situations.

The liquid radwaste system also provides indication of floodup within the radiologically controlled area of the auxiliary building through the use of level sensors. See Subsection 3.4.1.2.2.2 for a discussion of floodup for the radiologically controlled area of the auxiliary building.

11.2.1.2.5 Equipment Design

11.2.1.2.5.1 Permanently Installed Equipment

The liquid radwaste system equipment design parameters are provided in Table 11.2-2.

The seismic design classification and safety classification for the liquid radwaste system components and structures are listed in Section 3.2. The components listed are located in the Seismic Category I Nuclear Island and in the radwaste building.

The monitor tanks in the non-seismic radwaste building are used to store processed water. The radioactivity content of processed water in each tank will be less than the A_1 and A_2 levels of 10 CFR 71 Appendix A, Table A-1.

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11.2.1.2.5.2 Use of Mobile and Temporary Equipment

The liquid radwaste system is designed to handle most liquid effluents and other anticipated events using installed equipment. However, for events occurring at a very low frequency or producing effluents not compatible with the installed equipment, temporary equipment may be brought into the radwaste building mobile treatment facility truck bays.

Connections are provided to and from various locations in the liquid radwaste system to these mobile equipment connections. This allows the mobile equipment to be used in series with installed equipment, as an alternate to it with the treated liquids returned to the liquid radwaste system, or as an ultimate disposal point for liquids that are to be removed from the plant site for disposal elsewhere.

The use of temporary equipment is common practice in operating plants. The radwaste building truck bays and laydown space for mobile equipment, in addition to the flexibility of numerous piping connections to the liquid radwaste system, allow the plant operator to incorporate mobile equipment in an integrated fashion.

Temporary equipment is also used to clean up the condensate storage tank if it becomes contaminated following steam generator tube leakage. This use of temporary equipment is similar to that just described, except that the equipment is used in the yard rather than in the radwaste building truck bays.

When mobile or temporary equipment is selected to process liquid effluents, the equipment design and testing meets the applicable requirements of Regulatory Guide 1.143. When confirmed through sampling that the radioactive waste contents do not exceed the A_2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71, the liquid effluent may be processed with mobile or temporary equipment in the Radwaste Building. When the A_2 quantities are exceeded, liquid effluent is processed in the Seismic Category I auxiliary building.

Operating procedures also discussed in Section 13.5 include administrative controls to limit the total cumulative radioactive inventory of unpackaged wastes located in the radwaste building so that the Regulatory Guide 1.143 unmitigated radiological release criteria of 5 rem to site personnel, and the 10 CFR Part 20.1301 dose limits of 100 millirem at the protected area boundary for members of the public, are not exceeded. These unpackaged wastes include liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste. These administrative controls limit the radionuclide inventory to less than the A2 limit specified in Appendix A to 10 CFR Part 71 in each of the three (3) radwaste monitor tanks, in each of up to three (3) mobile radwaste processing systems, and in any additional equipment located in the radwaste building. Transfer or packaging of spent media from a mobile radwaste processing system located in the radwaste building is procedurally controlled such that either spent media packaging for off-site shipment in process in the radwaste building is considered in the inventory with the operation of the mobile radwaste processing system, or the spent media is transferred to the seismic Category I auxiliary building for packaging. Once the packaging in the radwaste building is complete, the activity of the packaged spent media is no longer added to the applicable mobile radwaste processing system activity for comparison with the applicable A2 quantity limit. This results in preventing exposures from any unmitigated radiological release in the radwaste building, occurring over a two hour exposure period, from exceeding the Regulatory Guide 1.143 unmitigated radiological release criteria and 10 CFR Part 20.1301 dose limits to site personnel and members of the public at the protected area boundary, respectively. The unmitigated, unshielded worker dose is calculated at 11 feet from the source. Unlimited worker occupancy workstations and low dose rate waiting areas are located no closer than 11 feet from a mobile radwaste processing system or a radwaste monitor tank.

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Mobile and temporary equipment are designed in accordance with the applicable mobile and temporary radwaste treatment systems guidance provided in Regulatory Guide 1.143, including the codes and standards listed in Table 1 of the Regulatory Guide.

Mobile and temporary equipment has the following features:

- Level indication and alarms (high-level) on tanks.
- Screwed connections are permitted only for instrument connections beyond the first isolation valve.
- Remote operated valves are used where operations personnel would be required to frequently manipulate a valve.
- Local control panels are located away from the equipment, in low dose areas.
- Instrumentation readings are accessible from the local control panels (i.e., temperature, flow, pressure, liquid level, etc.).
- Wetted parts are 300 series stainless steel, except flexible hose and gaskets.
- Flexible hose is used only for mobile equipment within the designated “black box” locations between mobile components and at the interface with the permanent plant piping.
- The contents of tanks are capable of being mixed, either through recirculation or with a mixer.
- Grab sample points are located in tanks and upstream and downstream of the process equipment.

Inspection and testing of mobile or temporary equipment is in accordance with the codes and standards listed in Table 1 of Regulatory Guide 1.143 with the following additions:

- After placement in the station, the mobile or temporary equipment is hydrostatically, or pneumatically, tested prior to tie-in to permanent plant piping.
- A functional test, using demineralized water, is performed. Remote operated valves are stroked (open-closed-open or closed-open-closed) under full flow conditions. The proper function of the instrumentation, including alarms, is verified. The operating procedures are verified correct during the functional test.
- Tank overflows are routed to floor drains.
- Floor drains are confirmed to be functional prior to placing mobile or temporary equipment into operation.

11.2.1.3 Compliance with 10 CFR 20.1406

In accordance with the requirements of 10 CFR 20.1406 (Reference 5), the liquid radwaste system is designed to minimize, to the extent practicable, contamination of the facility and the environment, facilitate decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. This is done through appropriate selection of design technology for the system, and incorporation of the ability to update the system to use the best available technology throughout the life of the plant.

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11.2.2 System Description

The liquid radwaste system, shown in Figure 11.2-1, includes tanks, pumps, ion exchangers, and filters. The liquid radwaste system is designed to process, or store for processing by mobile equipment, radioactively contaminated wastes in four major categories:

- Borated, reactor-grade, waste water – this input is collected from the reactor coolant system (RCS) effluents received through the chemical and volume control system (CVS), primary sampling system sink drains and equipment leakoffs and drains.
- Floor drains and other wastes with a potentially high suspended solids content – this input is collected from various building floor drains and sumps.
- Detergent wastes – this input comes from the plant hot sinks and showers, and some cleanup and decontamination processes. It generally has low concentrations of radioactivity.
- Chemical waste – this input comes from the laboratory and other relatively small volume sources. It may be mixed hazardous and radioactive wastes or other radioactive wastes with a high dissolved-solids content.

Nonradioactive secondary-system waste is not processed by the liquid radwaste system. Secondary-system effluent is normally handled by the steam generator blowdown processing system and the waste water system as described in Subsection 10.4.8.

Radioactivity can enter the secondary systems from steam generator tube leakage. If significant radioactivity is detected in secondary-side systems, blowdown is diverted to the liquid radwaste system for processing and disposal.

11.2.2.1 Waste Input Streams

11.2.2.1.1 Reactor Coolant System Effluents

The effluent subsystem receives borated and hydrogen-bearing liquid from two sources: the reactor coolant drain tank and the chemical and volume control system. The reactor coolant drain tank collects leakage and drainage from various primary systems and components inside containment. Effluent from the chemical and volume control system is produced mainly as a result of reactor coolant system heatup, boron concentration changes and RCS level reduction for refueling.

Input collected by the effluent subsystem normally contains hydrogen and dissolved radiogases. Therefore, it is routed through the liquid radwaste system vacuum degasifier before being stored in the effluent holdup tanks.

The liquid radwaste system degasifier can also be used to degas the reactor coolant system before shutdown by operating the chemical and volume control system in an open loop configuration. This is done by taking one of the effluent holdup tanks out of normal waste service and draining it. Then normal chemical and volume control system letdown is directed through the degasifier to the dedicated effluent holdup tank. From there, it is pumped back to the suction of the chemical and volume control system makeup pumps with the effluent holdup tank pump. The makeup pumps return the fluid to the reactor coolant system in the normal fashion. This process is continued as necessary for degassing the reactor coolant system as described in Subsection 9.3.6.

The input to the reactor coolant drain tank is potentially at high temperature. Therefore, provisions are made for recirculation through a heat exchanger for cooling. The tank is inerted with nitrogen and

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is vented to the gaseous radwaste system. Transfer of water from the reactor coolant drain tank is controlled to maintain an essentially fixed tank level to minimize tank pressure variation.

Reactor coolant system effluents from the chemical and volume control system letdown line or the reactor coolant drain subsystem pass through the vacuum degasifier, where dissolved hydrogen and fission gases are removed. These gaseous components are sent via a water separator to the gaseous radwaste system. A degasifier discharge pump then transfers the liquid to the currently selected effluent holdup tank. If flows from the letdown line and the reactor coolant drain tank are routed to the degasifier concurrently, the letdown flow has priority and the drain tank input is automatically suspended.

In the event of abnormally high degasifier water level, inputs are automatically stopped by closing the letdown control and containment isolation valves.

The effluent holdup tanks vent to the radiologically controlled area ventilation system and, in abnormal conditions, may be purged with air to maintain a low hydrogen gas concentration in the tanks' atmosphere. Hydrogen monitors are included in the tanks vent lines to alert the operator of elevated hydrogen levels.

The contents of the effluent holdup tanks may be recirculated and sampled, recycled through the degasifier for further gas stripping, returned to the reactor coolant system via the chemical and volume control system makeup pumps, discharged to the mobile treatment facility, processed through the ion exchangers, or directed to the monitor tanks for discharge without treatment.

Processing through the ion exchangers is the normal mode.

The AP1000 liquid radwaste system processes waste with an upstream filter followed by four ion exchange resin vessels in series. Any of these vessels can be manually bypassed and the order of the last two can be interchanged, so as to provide complete usage of the ion exchange resin.

The top of the first vessel is normally charged with activated carbon, to act as a deep-bed filter and remove oil from floor drain wastes. Moderate amounts of other wastes can also be routed through this vessel. It can be bypassed for processing of relatively clean waste streams. This vessel is somewhat larger than the other three, with an extra sluice connection to allow the top bed of activated carbon to be removed. This feature is associated with the deep bed filter function of the vessel; the top layer of activated carbon collects particulates, and the ability to remove it without disturbing the underlying zeolite bed minimizes solid-waste production.

The second, third and fourth beds are in identical ion exchange vessels, which are selectively loaded with resin, depending on prevailing plant conditions.

After deionization, the water passes through an after-filter where radioactive particulates and resin fines are removed. The processed water then enters one of the monitor tanks. When one of the monitor tanks is full, the system is automatically realigned to route processed water to another tank.

The contents of the monitor tank are recirculated and sampled. In the unlikely event of radioactivity in excess of operational targets, the tank contents are returned to a waste holdup tank for additional processing.

Normally, however, the radioactivity will be well below the discharge limits, and the dilute boric acid is discharged for dilution to the circulating water blowdown. The discharge flow rate is set to limit the boric acid concentration in the circulating water blowdown stream to an acceptable concentration for local requirements. Detection of high radiation in the discharge stream stops the discharge flow and operator action is required to re-establish discharge. The raw water system which provides makeup

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for the circulating water system is used as a backup source for dilution water when cooling tower blowdown is not available for the discharge path.

11.2.2.1.2 Floor Drains and Other Wastes with Potentially High Suspended Solid Contents

Potentially contaminated floor drain sumps and other sources that tend to be high in particulate loading are collected in the waste holdup tank. Additives may be introduced to the tank to improve filtration and ion exchange processes. Tank contents may be recirculated for mixing and sampling. The tanks have sufficient holdup capability to allow time for realignment and maintenance of the process equipment.

The waste water is processed through the waste pre-filter to remove the bulk of the particulate loading. Next it passes through the ion exchangers and the waste after-filter before entering a monitor tank. The monitor tank contents are sampled and, if necessary, returned to a waste holdup tank and processed through the filters and ion exchangers.

Waste water meeting the discharge limits is discharged to the circulating water blowdown through a radiation detector that stops the discharge if high radiation is detected.

11.2.2.1.3 Detergent Wastes

The detergent wastes from the plant hot sinks and showers contain soaps and detergents. These wastes are generally not compatible with the ion exchange resins described in Subsections 11.2.2.1.1 and 11.2.2.1.2. The detergent wastes are not processed and are collected in the chemical waste tank. If the detergent wastes activity is low enough, the wastes can be discharged without processing.

When sufficient detergent wastes are produced and processing is necessary, installed equipment or mobile processing equipment (brought into one of the radwaste building mobile systems facility truck bays provided for this purpose) may be used to process the waste prior to discharge or waste water may be shipped offsite for processing.

11.2.2.1.4 Chemical Wastes

Inputs to the chemical waste tank normally are generated at a low rate. Chemicals can be added to the tank for pH or other adjustment. Since the volume of these wastes is low, they can be treated by installed equipment or the use of mobile equipment or by shipment offsite.

11.2.2.1.5 Steam Generator Blowdown

Steam generator blowdown is normally accommodated within the steam generator blowdown system, which is described in Subsection 10.4.8.

If steam generator tube leakage results in significant levels of radioactivity in the steam generator blowdown stream, this stream is redirected to the liquid radwaste system for treatment before release. In this event, one of the waste holdup tanks is drained to prepare it for blowdown processing. The blowdown stream is brought into that holdup tank, and continuously or in batches pumped through the waste ion exchangers. The number of ion exchangers in service is determined by the operator to provide adequate purification without excessive resin usage. The blowdown is then collected in a monitor tank, sampled, and discharged in a monitored fashion.

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11.2.2.2 Other Operations

11.2.2.2.1 Sampling

Grab sampling taps are provided where required to monitor influent boron and radioactivity concentrations; to monitor performance of various components; to determine tank water characteristics before transfer, processing or discharge; to verify performance of the on-line analyzers; and to collect samples of discharges to the environs for analysis and documentation. Samples are taken in low radiation areas.

11.2.2.2.2 Tank Cleaning

Extraordinary measures for tank cleaning are not normally required because the pumps take suction from the low point of the tank, and the tank bottoms are sloped so that the tank can be fully drained. Recirculation connections are provided to allow the tanks to be effectively mixed. Also, the air-operated double-diaphragm pumps used can pump air, water or slurries without damage, and can run dry to clear the bottoms of the tanks.

Provisions are made for tank cleaning using a portable tank cleaning rig. Suction is taken from the tank bottom via a temporary hose. The pump discharge passes through a filter and the hose to a tank cleaning lance, which is manually inserted through a manway on the tank. The operator can direct the high-velocity water throughout the inside of the tank.

11.2.2.3 Component Description

The general descriptions and summaries of the design basis requirements for the liquid radwaste system components follow. Table 11.2-2 contains the operating parameters for the liquid radwaste system components.

Additional information regarding the applicable codes and classifications is also available in Section 3.2.

11.2.2.3.1 Liquid Radwaste System Pumps

Reactor Coolant Drain Tank Pumps

Two full-capacity, stainless steel, reactor coolant drain tank pumps recirculate the reactor coolant drain tank contents for cooling and to discharge the reactor coolant drain tank contents to the degasifier or to an effluent holdup tank. These vertical sump pumps have permanently lubricated bearings and mechanical seals. The pumps start and stop on high and low level.

Containment Sump Pumps

Two full-capacity containment sump pumps are provided. These submersible pumps discharge the containment sump contents to the waste holdup tank. The pumps start and stop on high and low level.

Degasifier Vacuum Pumps

Two stainless steel, full-capacity, liquid ring type, degasifier vacuum pumps maintain the degasifier at a low pressure for efficient gas stripping.

These liquid ring pumps use water as the compressant. The water is recycled to minimize consumption. Excess water from vapor condensation is discharged to an effluent holdup tank.

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Degasifier Vacuum Purge Pump

One vacuum pump provides continuous removal of air in-leakage and continuous removal of nitrogen purge.

Degasifier Separator Pump

Two full capacity centrifugal pumps are provided to discharge recovered compressor water from the degasifier separator back to the degasifier vacuum pumps. The pump also serves to discharge any excess compressor water accumulation in the separator to an effluent holdup tank. The pumps start and stop to share the duty. The pump is constructed of stainless steel and has a mechanical seal.

Other Pumps

The following air-operated double-diaphragm pumps are mounted near the associated tanks with internal suction piping. Construction is of stainless steel, with elastomeric diaphragms.

- Degasifier discharge pumps (2)
- Effluent holdup tank pumps (2)
- Waste holdup tank pumps (2)
- Monitor tank pumps (6)
- Chemical waste tank pump (1)

11.2.2.3.2 Liquid Radwaste System Heat Exchangers

Reactor Coolant Drain Tank Heat Exchanger

One horizontal U-tube heat exchanger is provided. The heat exchanger shell is removable to allow for inspection and cleaning of the tube bundles.

The heat exchanger is designed to prevent the reactor coolant drain tank contents from boiling with hot leakage influent as shown in Table 11.2-4.

The reactor coolant drain tank contents flow through stainless steel tubes, and component cooling water flows inside a carbon or stainless steel shell.

Vapor Condenser

One horizontal U-tube heat exchanger assists in drying the gases drawn out of the liquid waste by the vacuum pump, before they are sent to the gaseous radwaste system. As the gas bearing water cascades down through the packing in the degasifier vessel, it boils in the low pressure. To minimize the size of the vacuum pumps, a vapor condenser is provided between the degasifier vessel and the vacuum pumps. In the vapor condenser, most of the water vapor is condensed out of the gas stream before it enters the vacuum pump. The vapor condenser is cooled by chilled water. Chilled water flows through the tubes, which are stainless steel. Water vapor condenses on the tubes and drains through a subcooling section in the stainless steel shell. The non-condensible gases and condensate are recombined in a common pipe leading to the suction of the liquid ring type vacuum pumps.

11.2.2.3.3 Liquid Radwaste System Tanks

Reactor Coolant Drain Tank

One reactor coolant drain tank is provided. The tank is sized to accommodate two vertical sump type pumps and to have a volume above the normal operating water level sufficient to accept the influent rate shown in Table 11.2-4.

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The reactor coolant drain tank is a stainless steel, horizontal, cylindrical tank with dished heads. It is provided with a vacuum breaker to prevent excess external pressure during containment leak testing. It is protected from excess internal pressure by a relief valve which vents to the containment sump.

Effluent Holdup Tanks

These stainless steel tanks contain effluent waste prior to processing. They are horizontal cylinders with internal pump suction piping at the low point of the tank, and with side manways for maintenance.

Waste Holdup Tanks

These stainless steel tanks contain floor and equipment drain waste before processing. They are vertical cylinders with internal pump suction piping at the low points of the tanks and with side manways for maintenance.

Monitor Tanks

These stainless steel tanks contain processed waste before discharge. They are vertical cylinders with internal pump suction piping at the low points of the tanks and with side manways for maintenance.

Chemical Waste Tank

This stainless steel tank contains chemical waste along with detergent wastes from hot sinks, hot showers, and equipment decontamination before processing. The configuration is a vertical cylinder with internal pump suction piping at the low point of the tank and with a side manway for maintenance.

11.2.2.3.4 Liquid Radwaste System Sump

Containment Sump

The containment sump is a stainless steel, rectangular sump tank designed for embedment in concrete. The containment sump is sized as shown in Table 11.2-4.

11.2.2.3.5 Liquid Radwaste System Vessels

Degasifier Column

A one-stage, stainless steel degasifier column is provided. The degasifier column is designed to meet the performance parameters shown in Table 11.2-5.

Agitation and surface exposure are accomplished by spraying the influent onto the top of a column of packing which breaks up the flow and spreads it into thin films as it cascades downward. The low pressure causes the inlet water to boil. The flashed vapor accompanies the gas bearing water downward through the packing. Exposure to low pressure draws out the non-condensible gases consistent with Henry's Law and they pass out the vacuum connection. The vacuum connection is located near the last point of contact with the degassed water where the vacuum is greatest and conditions are least conducive to reabsorption. A stainless steel mesh demister is provided at the vessel vacuum connection to remove water droplets which are entrained in the gas/vapor mixture as it is exiting to the vapor condenser.

Degasifier Separator

One stainless steel separator is provided. It is designed to remove compressor water from the vacuum pump discharge flow for reuse. It also serves as a silencer.

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11.2.2.3.6 Liquid Radwaste System Ion Exchangers

Four ion exchange vessels are provided, with resin volumes as shown in Table 11.2-2. The media will be selected by the plant operator to optimize system performance. The ion exchange vessels are stainless steel, vertical, cylindrical pressure vessels with inlet and outlet process nozzles plus connections for resin addition, sluicing, and draining. The process outlet and flush water outlet connections are equipped with resin retention screens designed to minimize pressure drop.

11.2.2.3.7 Liquid Radwaste System Filters

Waste Pre-Filter

This filter is provided to collect particulate matter in the process stream before ion exchange. The unit is constructed of stainless steel and has disposable filter bags.

Waste After-Filter

This filter is provided downstream of the ion exchangers to collect particulate matter, such as resin fines. The unit is constructed of stainless steel and has disposable filter cartridges.

11.2.2.4 Instrumentation Design

Instrumentation readout is available in the main control room and on display and control panels.

Alarms are provided to the data display system including a radwaste system annunciator in the main control room.

Pressure indicators provide pressure drops across demineralizers, filters, and strainers.

Releases to the environment are monitored for radioactivity. Section 11.5 describes this instrumentation.

Each tank is provided with level instrumentation that actuates an alarm on high liquid level in the tank, thus warning of potential tank overflow. High level in redundant tank pairs also diverts the flow to the standby tank. Table 11.2-3 provides a summary of the tank level alarms.

11.2.2.5 System Operation and Performance

11.2.2.5.1 Reactor Coolant System Effluent Processing

11.2.2.5.1.1 Reactor Coolant Systems Effluent: Letdown Line

Chemical and volume control system letdown is directed to the degasifier. This letdown flow automatically takes priority by causing isolation of influent to the degasifier from the reactor coolant drain tank pumps to prevent the design capacity of the degasifier from being exceeded.

When the degasifier and waste gas system are placed in operation one of the degasifier vacuum pumps operates to maintain a vacuum in the degasifier column. The degasifier separator pump operates to return compressor water to the vacuum pump. The degasifier separator vents to the gaseous radwaste system. Its level is automatically controlled by discharging excess water (due to condensation of vapor carryover from the degasifier column) to an effluent holdup tank. In the event of abnormally high level, chemical and volume control system letdown flow is automatically stopped.

Two effluent holdup tanks are provided. One is aligned to receive inputs. When it fills to the appropriate level, an alarm alerts the operator that the tank is full and ready for processing. The inlet

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diversion valve automatically realigns the system to route input to the other tank upon high-high alarm.

11.2.2.5.1.2 Reactor Coolant System Effluent: Reactor Coolant Drain Tank

The reactor coolant drain tank receives input from the reactor coolant system and other drains inside containment that have the potential to contain radioactive gas or hydrogen.

Initially and after servicing, the reactor coolant drain tank is filled with demineralized water and then purged with nitrogen to dilute and displace oxygen. The tank vent to the gaseous radwaste system normally remains closed. One of the reactor coolant drain tank pumps and the discharge valve are automatically controlled to maintain reactor coolant drain tank water level within a narrow band to minimize tank pressure variation. An alarm alerts the operator if the reactor coolant drain tank reaches a temperature consistent with the design leak of saturated RCS coolant. The system automatically realigns valves and recirculates the tank contents through the reactor coolant drain tank heat exchanger.

The cumulative quantity discharged from the reactor coolant drain tank is totalized and indicated for use in reactor coolant leakage evaluations.

The discharge may have a relatively high dissolved hydrogen concentration and is therefore aligned to the degasifier. However, during reactor coolant system loop drain operations the hydrogen and radioactive gas concentrations should be low and discharge may be directly aligned to an effluent holdup tank.

11.2.2.5.1.3 Processing of the Reactor Coolant System Effluents

Each effluent holdup tank vent includes a hydrogen detector to monitor the hydrogen concentration in the tank atmosphere. In the event of high alarm, the operator initiates air purge through the tank to dilute the hydrogen gas and maintain it below the flammable limits. The tanks vent to the radiologically controlled area ventilation system.

An effluent holdup tank high level alarm alerts the operator that the tank is full and ready for processing. The inlet diversion valve automatically directs the influent to the other tank upon high-high alarm.

To process the contents of the filled tank, the effluent holdup tank pump is started to recirculate and sample the tank contents. If additional gas stripping is required, the tank contents may be recirculated through the degasifier. The degasifier functions automatically as described in Subsection 11.2.2.5.1.1.

The discharge of either effluent holdup tank pump can be aligned to the suction of the chemical and volume control system makeup pumps. This mode of operation is used during reactor coolant system degassing operations. Reactor coolant from the chemical and volume control system letdown is degassed in the degasifier, collected in one of the effluent holdup tanks, and continuously pumped back to the chemical and volume control system makeup pumps. The pump returns the degassed water to the reactor coolant system.

Reactor coolant collected in an effluent holdup tank during reactor coolant system loop drain operations may also be pumped to the chemical and volume control system makeup pumps for refill of the reactor coolant system. Before beginning this process, the operator fully drains the effluent holdup tank receiving the reactor coolant so that the boron concentration of the reactor coolant system is not significantly affected.

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The effluent may be transferred to the mobile treatment facility for concentration or solidification. This disposal method is used only during unusual conditions that restrict the normal processed waste discharge mode described in the following paragraphs.

The normal mode of operation is to process the effluent by ion exchange and filtration to remove the radioactive materials. The ion exchangers operate in series as described in Subsection 11.2.2.1.1.

The last bed provides a polishing function and also prevents radioactivity breakthrough to the monitor tank when the upstream unit becomes exhausted. This allows the full capacity of the upstream resin beds to be used, reducing the amount of spent resin that is generated.

When the analysis of samples taken periodically downstream of the ion exchange processing indicates an increase in radioactivity above prescribed limits, the operator isolates the expended unit(s) for resin replacement. Flow continues through the other units until a fresh resin bed is ready. When one of the last two ion exchangers has been replenished, the fresh unit is then brought online as the downstream unit.

The after-filter removes resin fines and other particulate matter that may pass through the ion exchangers. A high differential pressure alarm alerts the operator to the need for filter element replacement. Normally, filter element replacement is initiated on high radioactivity determined by periodic survey.

Process discharge is normally aligned to one of the monitor tanks. When one of the tanks is full, an alarm alerts the operator that the tank is full and ready to be discharged. The inlet diversion valve automatically realigns the system to route processed waste to another tank upon high-high level.

The operator then starts the monitor tank pump to recirculate the tank contents and samples the processed waste. Since the ion exchangers operate in the borated saturated mode, the water contains boric acid. The radioactivity and chemistry of the processed waste is determined by sample analysis. In the unlikely event that radioactivity exceeds discharge limitations, the tank contents are returned to a waste holdup tank for reprocessing.

Once it is confirmed that the waste water is within radioactivity discharge limitations, the operator prepares the system for discharge. The operator initiates discharge by starting the monitor tank pump and opening the remotely operated discharge valve. During controlled discharge, grab samples are taken for laboratory analysis and documentation of discharge.

If the radiation monitor in the discharge line detects high radiation, the valve automatically closes. The operator is alerted to this condition by a high radiation alarm, and is required to take corrective action. A manual drain valve is opened to flush the radiation monitor and confirm low radiation before re-establishing discharge to the circulating water blowdown. Low monitor tank level automatically stops the monitor tank pump.

11.2.2.5.2 Floor Drain and Equipment Drain Waste Processing

Miscellaneous liquid wastes normally include influent from the radioactive floor drains, equipment drains and auxiliary building sump and excess water from the solid radwaste system. These wastes collect in one of two waste holdup tanks.

A high level alarm in the tank alerts the operator that the tank is full and ready to be processed. The inlet diversion valve automatically directs influents to the second waste holdup tank upon high-high level. The waste holdup tank pump is started to recirculate and sample the tank contents. Additives may be introduced to the waste holdup tank to optimize filtration and ion exchange processes.

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Floor drain wastes are also brought into the waste holdup tanks from the containment sump. High sump level automatically opens the containment isolation valves and starts a pump to transfer the sump contents. Low level automatically stops the pump and closes the isolation valves. An alarm is provided to alert the operator to abnormally high containment sump level and the standby pump is automatically started. Cumulative flow is totalized and indicated to support reactor coolant leakage analysis.

The normal mode of operation is to process the waste water through the pre-filter, ion exchangers, and after-filter to the monitoring tank as described for the reactor coolant system effluent processing. Under abnormal conditions, the waste may also be transferred directly to a mobile treatment facility.

11.2.2.5.3 Detergent Waste Processing

The detergent wastes from the plant hot sinks and showers are routed to the chemical waste tank. Normally, these wastes are sampled and confirmed suitable for discharge without processing. If processing prior to discharge is necessary, three courses of action are available. The waste water may be transferred to a waste holdup tank and processed in the same manner as other radioactively contaminated waste water. If the onsite processing capabilities are not suitable for the composition of the detergent waste, processing can be performed using mobile equipment brought into one of the truck bays of the radwaste building or the waste water can be shipped offsite for processing. After processing by mobile equipment the water may be transferred to a waste holdup tank for further processing by the onsite equipment or transferred to a monitor tank for sampling and discharge.

11.2.2.5.4 Chemical Waste Processing

Radioactively contaminated chemical wastes are collected in the chemical waste tank. Chemicals may be added to the tank for pH or other adjustment. The volume of these wastes is expected to be low. The design includes alternatives for processing or discharge of chemical wastes. They may be processed onsite using mobile equipment, but are expected to be suitable for discharge without treatment or for processing by installed equipment before discharge. When not suitable for onsite processing, they can be shipped offsite for processing. After processing by mobile equipment the water may be transferred to a waste holdup tank for further processing by the onsite equipment or transferred to a monitor tank for sampling and discharge.

11.2.2.5.5 Steam Generator Blowdown Processing

Normal steam generator blowdown processing is accommodated by the steam generator blowdown system, which is described in Subsection 10.4.8.

If steam generator tube leakage results in levels of radioactivity in the blowdown stream above what can be accommodated by the secondary-side systems, this stream is directed to the liquid radwaste system. For this function, the operator aligns the steam generator blowdown system to the inlet of the waste holdup tank. The blowdown waste is then processed in the same way as other wastes.

11.2.2.5.6 Ion Exchange Media Replacement

The initial and subsequent fill of ion exchange media is made through a resin fill nozzle on the top of the ion exchange vessel. When the media are spent and ready to be transferred to the solid radwaste system, the vessel is isolated from the process flow. The flush water line is opened to the sluice piping and demineralized water is pumped into the vessel through the normal process outlet connection upward through the media retention screen. The media fluidize in the upward, reverse flow. When the bed has been fluidized, the sluice connection is opened and the bed is sluiced to the spent resin tanks in the solid radwaste system (WSS). Demineralized water flow continues until the bed has been removed and the sluice lines are flushed clean of spent resin.

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11.2.3 Radioactive Releases

Liquid waste is produced both on the primary side (primarily from adjustment of reactor coolant boron concentration and from reactor coolant leakage) and the secondary side (primarily from steam generator blowdown processing and from secondary side leakage). Primary and secondary coolant activity levels are provided in Section 11.1 for both the design case and the anticipated case, which is based on operating plant experience.

Except for reactor coolant system degasification in anticipation of shutdown, the AP1000 does not recycle primary side effluents for reuse. Primary effluents are discharged to the environment after processing. Fluid recycling is provided for the steam generator blowdown fluid which is normally returned to the condensate system.

11.2.3.1 Discharge Requirements

The release of radioactive liquid effluents from the plant may not exceed the concentration limits specified in Reference 1 nor may the releases result in the annual offsite dose limits specified in 10 CFR 50, Appendix I (Reference 2) being exceeded.

11.2.3.2 Estimated Annual Releases

The annual average release of radionuclides from the plant is determined using the PWR-GALE code (Reference 3). The PWR-GALE code models releases which use source terms derived from data obtained from the experience of operating PWRs. The code input parameters used in the analysis to model the AP1000 plant are listed in Table 11.2-6. The annual releases for a single-unit site are presented in Table 11.2-7.

In agreement with Reference 3, the total releases include an adjustment factor of 0.16 curies per year to account for anticipated operational occurrences. The adjustment uses the same distribution of nuclides as the calculated releases.

11.2.3.3 Dilution Factor

The dilution factor provided for the activity released is site dependent; the value of 6000 gpm used herein is based on cooling tower blowdown requirements and is expected to be conservatively low. The plant operator will select dilution flow rates to ensure that the effluent concentration limits of 10 CFR Part 20, the annual offsite dose limits in 10 CFR 50 Appendix I, and any local requirements are continuously met. If the available dilution is low, the discharge rate can be reduced to maintain acceptable concentrations.

The required dilution flow is dependent on the liquid waste discharge rate and, while the monitor tank pumps have a design flow rate of 100 gpm, the discharge flow is controlled to be compatible with the available dilution flow. With a typical liquid waste release of 1925 gallons per day, the nominal circulating water blowdown flow of 6000 gpm provides sufficient dilution flow to maintain the annual average discharge concentrations well below the effluent concentration limits. Actual plant operation is dependent on the waste liquid activity level and the available dilution flow.

11.2.3.4 Release Concentrations

The annual release data provided in Table 11.2-7 represent expected releases from the plant. To demonstrate compliance with the Reference 1 effluent concentration limits, the discharge concentrations have been evaluated for the release of a typical daily liquid waste volume of 1925 gallons per day and using the nominal circulating water blowdown flow of 6000 gpm. Table 11.2-8 lists the annual average nuclide release concentrations and the fraction of the effluent

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concentration limits using base GALE code assumptions. As shown in Table 11.2-8, the overall fraction of the effluent concentration limit is well below the allowable value of 1.0.

The annual releases from the plant have also been evaluated based on operation with the maximum defined fuel defect level. The maximum defined fuel defect level is conservatively established at 1.0 percent fuel defects, corresponding to four times the Technical Specification limit on coolant activity which is based on 0.25 percent fuel defects, as described in Table 11.1-2. Table 11.2-9 lists the annual average nuclide release concentrations and the fractions of the effluent concentration limits for the maximum defined fuel defects. As shown in Table 11.2-9, the overall fraction of the effluent concentration limit for the maximum defined fuel defect level is well below the allowable value of 1.0.

11.2.3.5 Estimated Doses

Estimated doses are site specific and are discussed in subsection 11.2.5.

11.2.3.5.1 Not Used.

11.2.3.5.2 Not Used.

11.2.3.5.3 Liquid Radwaste Cost Benefit Analysis Methodology

The application of the methodology of Regulatory Guide 1.110 was used to satisfy the cost benefit analysis requirements of 10 CFR Part 50. Appendix I, Section II.D. The parameters used in calculating the Total Annual Cost (TAC) are fixed and are given for each radwaste treatment system augment listed in Regulatory Guide 1.110, including the Annual Operating Cost (AOC) (Table A-2), Annual Maintenance Cost (AMC) (Table A-3), Direct Cost of Equipment and Materials (DCEM) (Table A-1), and Direct Labor Cost (DLC) (Table A-1). The following variable parameters were used:

- Capital Recovery Factor (CRF) -This factor is taken from Table A-6 of Regulatory Guide 1.110 and reflects the cost of money for capital expenditures. A cost-of-money value of 7% per year is assumed in this analysis, consistent with the "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (NUREG/BR-0058). A CRF of 0.0806 was obtained from Table A-6.

- Indirect Cost Factor (ICF) -This factor takes into account whether the radwaste system is unitized or shared (in the case of a multi-unit site) and is taken from Table A-5 of Regulatory Guide 1.110. It is assumed that the radwaste system for this analysis is a unitized system at a 2-unit site, which equals an ICF of 1.625.

- Labor Cost Correction Factor (LCCF) -This factor takes into account the differences in relative labor costs between geographical regions and is taken from Table A-4 of Regulatory Guide 1.110. A LCCF of 1.0 (the lowest value) is assumed in this analysis.

Appendix I to 10 CFR Part 50 prescribes a $1,000 per person-rem criterion for determining the cost benefit of actions to reduce radiation exposure.

The analysis used a conservative assumption that the respective radwaste treatment system augment is a "perfect" system that reduces the effluent and dose by 100 percent. The liquid radwaste treatment system augments annual costs were determined and the lowest annual cost considered a threshold value. The lowest-cost option for liquid radwaste treatment system augments is a 20 gpm Cartridge Filter at $11,140 per year, which yields a threshold value of 11.14 person-rem total body or thyroid dose from liquid effluents.

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For AP1000 sites with population dose estimates less than 11.14 person-rem total body or thyroid dose from liquid effluents. no further cost-benefit analysis is needed to demonstrate compliance with 10 CFR 50, Appendix I Section II.D.

11.2.3.5.4 Not Used.

11.2.3.6 Quality Assurance

The quality assurance program for design, fabrication, procurement, and installation of the liquid radwaste system is in accordance with the overall quality assurance program described in Chapter 17.

Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation and testing provisions of the liquid radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.2.4 Preoperational Testing

11.2.4.1 Sump Level Instrument Testing

One of the diverse methods of detecting small reactor coolant pressure boundary leaks is monitoring the containment sump level. (See Subsection 5.2.5 for a full discussion.) A sump capacity calibration test is performed so the containment sump level instruments can provide a display that is correlated to the contained volume of water in the sump.

In addition to a normal level accuracy calibration of the containment sump level instruments, WLS-LIT-034, WLS-LIT-035, and WLS-LIT-036, their displays will be correlated to the volume of water during preoperational testing. A known volume of water will be added to the containment sump. The change in sump level will be measured by using a local measurement method. The change in the display of the sump level instruments will be compared to the level change measured by a local measurement methodology. A sump level change corresponding to a volume of water which is smaller than that released in an hour by 0.5 gpm reactor coolant system leak can be detected.

11.2.4.2 Discharge Control/Isolation Valve Testing

The AP1000 effluent discharge line includes a radiation monitor, WLS-RE-229, as described in Subsection 11.5.2.3.3. A concentration of radioactivity in the effluent, which exceeds the radiation monitor setpoint, causes a high radiation signal to automatically close the discharge control/isolation valve.

A test will be performed on the liquid radwaste system discharge control/isolation valve, WLS-PL-V223, during preoperational testing. A simulated WLS-RE-229 high radiation signal will be sent to the plant control system and the discharge control/isolation valve will be observed to close.

11.2.4.3 Preoperational Inspection

The performance of the liquid radwaste system has been evaluated based upon using a predetermined quantity and type of ion-exchange media. An inspection will confirm that the proper volume of media, as listed in Table 11.2-2, “Component Data – Liquid Radwaste System,” has been installed into the appropriate liquid radwaste system components, MV03 and MV04A/B/C.

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11.2.5 Combined License Information

11.2.5.1 Liquid Radwaste Processing by Mobile Equipment

The mobile or temporary equipment used for storing or processing liquid radwaste is addressed in Subsection 11.2.1.2.5.2.

11.2.5.2 Cost Benefit Analysis of Population Doses

The analysis performed to determine offsite dose due to liquid effluents is based upon the AP1000 generic site parameters included in Chapter 1 and Tables 11.2-5 and 11.2-6. The Combined License applicant will provide a site specific cost-benefit analysis to address the requirements of 10 CFR 50, Appendix I, regarding population doses due to liquid effluents.

The Combined License applicant will also comply with individual dose limits to members of the public in 10 CFR 50 Appendix I and 10 CFR 20.1301.

11.2.5.3 Identification of Ion Exchange and Adsorbent Media

The types of liquid waste ion exchange and absorbent media to be used in the liquid radwaste system (WLS) are addressed in APP-GW-GLR-008 (Reference 6), and the applicable changes are incorporated.

11.2.5.4 Dilution and Control of Boric Acid Discharge

The planned discharge flow rate for borated wastes and controls for limiting the boric acid concentration in the circulating water system blowdown is addressed in APP-GW-GLR-014 (Reference 7), and the applicable changes are incorporated.

11.2.6 References

1. "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," 10 CFR Part 20, Appendix B, Issued by 58 FR 67657, April 28, 1995.

2. "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," 10 CFR Part 50, Appendix I.

3. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," NUREG-0017, Revision 1, March 1985.

4. Not used.

5. "Minimization of Contamination," 10 CFR 20.1406.

6. APP-GW-GLR-008, "Identification of Ion Exchange and Absorbent Media, Completing COL Items 11.2-3 and 11.3-2," Westinghouse Electric Company LLC.

7. APP-GW-GLR-014, "Closure of COL Items in DCD Chapter 11, Dilution and Control of Boric Acid Discharge," Westinghouse Electric Company LLC.

8. Not used.

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Table 11.2-1 (Sheet 1 of 2)
Liquid Inputs and Disposition

Collection Tank and Sources	Expected Input Rate	Activity	Basis	Disposition
1. Effluent holdup tanks				Filtered, demineralized, and discharged
Chemical and volume control system letdown	159,000 gpy	100% of reactor coolant	AP1000-specific calculations[(a)]	
Leakage inside containment (to reactor coolant drain tank)	10 gpd	167% of reactor coolant	ANSI/ANS-55.6	
Leakage outside containment (to effluent holdup tanks)	80 gpd	100% of reactor coolant	ANSI/ANS-55.6	
Sampling drains	200 gpd	5% of reactor coolant	ANSI/ ANS-55.6	
2. Waste holdup tank				Filtered, demineralized and discharged
Reactor containment cooling	500 gpd	0.1% of reactor coolant	ANSI/ANS-55.6	
Spent fuel pool liner leakage	25 gpd	0.1% of reactor coolant	ANSI/ANS-55.6	
Misc. drains	675 gpd	0.1% of reactor coolant	ANSI/ANS-55.6	

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Table 11.2-1 (Sheet 2 of 2)
Liquid Inputs and Disposition

Collection Tank and Sources	Expected Input Rate	Activity	Basis	Disposition
3. Detergent waste				Filtered, demineralized, monitored, and discharged. If necessary, processed with mobile equipment or shipped offsite.
Hot shower	0 gpd	10^{-7} µCi/g	ANSI/ANS-55.6	
Hand wash	200 gpd	10^{-7} µCi/g	ANSI/ANS-55.6	
Equipment and area decontamination	40 gpd	0.1% of reactor coolant	ANSI/ANS-55.6	
Laundry			Offsite laundry	
4. Chemical wastes	2 gpd	≤ reactor coolant	Estimate	Filtered, demineralized, and discharged. If necessary, processed with mobile equipment or shipped offsite.

Notes:
a. Average letdown for all normal reactor fuel cycle operations; initial heatup, dilutions and borations.

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Table 11.2-2 (Sheet 1 of 7)
Component Data – Liquid Radwaste System

Pumps	
Containment sump pumps	
Number	2
Type	Submersible centrifugal
Design pressure (psig)	32.5 external
Design temperature (°F)	250
Design flow (gpm)	100
Material	Stainless steel
Reactor coolant drain tank pumps	
Number	2
Type	Vertical sump type, centrifugal
Design pressure (psig)	20 external
Design temperature (°F)	260
Design flow (gpm)	100
Material	Stainless steel
Degasifier separator pump (part of vacuum degasifier)	
Number	2
Type	Centrifugal
Design pressure (psig)	125
Design temperature (°F)	200
Design flow (gpm)	7
Material	Stainless steel

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Table 11.2-2 (Sheet 2 of 7)
Component Data – Liquid Radwaste System

Pumps	
Standard waste processing pump Standard waste processing pump used for: Number Application 2 Degasifier discharge pumps 2 Effluent holdup tank pumps 2 Waste holdup tank pumps 6 Monitor tank pumps 1 Chemical waste tank pump	
Type	Air-operated, double-diaphragm
Design pressure (psig)	125
Design temperature (°F)	150
Design flow (gpm)	100 (can be varied by varying air supply flow)
Material	Stainless steel body, Elastomeric diaphragm
Degasifier vacuum pumps (part of vacuum degasifier package)	
Number	2
Type	Liquid ring
Design pressure (psig)	75
Design temperature (°F)	150
Design flow (scfm) (total hydrogen and water vapor flow)	641
Material	Stainless steel

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Table 11.2-2 (Sheet 3 of 7)
Component Data – Liquid Radwaste System

Filters	
Waste pre-filter	
Number	1
Type	Disposable bag
Design pressure (psig)	150
Design temperature (°F)	150
Design flow (gpm)	75
Particle size (micron, 98% retention)	Up to 40 microns
Materials	
Housing	Stainless steel
Filter	Polypropylene/pleated paper
Waste after-filter	
Number	1
Type	Disposable bag or cartridge
Design pressure (psig)	150
Design temperature (°F)	150
Design flow (gpm)	75
Particle size (micron, 98% retention)	Up to 20 microns
Materials	
Housing	Stainless steel
Filter medium	Polypropylene/pleated paper

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Table 11.2-2 (Sheet 4 of 7)
Component Data – Liquid Radwaste System

Heat Exchangers	
Reactor Coolant drain tank heat exchanger	
Number	1
Type	Horizontal U-tube
Design pressure (psig)	200 tubeside, 200 shellside
Design temperature (°F)	250 tubeside, 200 shellside
Design flow (lb/hr)	48,700 tubeside, 62,200 shellside
Heat Transfer Design Case	
Temperature inlet (°F)	175 tubeside, 95 shellside
Temperature outlet (°F)	143 tubeside, 120 shellside
Material	SS tubeside, CS or SS shellside
Vapor condenser	
Number	1
Type	Horizontal U-tube
Design pressure (psig)	150
Design temperature (°F)	150
Design flow (lb/hr)	125,000 tubeside, 1800 shellside
Heat Transfer Design Case	
Temperature inlet (°F)	40 tubeside, 87.1 shellside
Temperature outlet (°F)	55.7 tubeside, 60 shellside
Material	SS

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Table 11.2-2 (Sheet 5 of 7)
Component Data – Liquid Radwaste System

Ion Exchangers	
Deep bed filter	
Number	1
Design pressure (psig)	150
Design temperature (°F)	150
Design flow (gpm)	75
Nominal resin volume (ft^3)	50
Material	Stainless steel
Resin type	Layered: Activated charcoal on zeolite resin (Adjustable for plant conditions)
Process decontamination factors	See Table 11.2-5
Waste ion exchangers	
Number	3
Design pressure (psig)	150
Design temperature (°F)	150
Design flow (gpm)	75
Nominal resin volume (ft^3)	30
Materials	Stainless steel
Resin type	One cation, Two mixed (Adjustable for plant conditions)
Process decontamination factors	See Table 11.2-5

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Table 11.2-2 (Sheet 6 of 7)
Component Data – Liquid Radwaste System

Tanks	
Reactor coolant drain tank	
Number	1
Nominal volume (gal)	900
Type	Horizontal
Design pressure (psig)	Internal — 20 for the tank, valves and attached instruments external pressure range — -0.2 to 1.0
Material	Stainless steel
Containment sump	
Number	1
Nominal volume (gal)	220
Type	Rectangular
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Effluent holdup tanks	
Number	2
Nominal volume (gal)	28,000
Type	Horizontal
Design pressure (psig)	Atmospheric
Design temperature (°F)	150
Material	Stainless steel
Waste holdup tanks	
Number	2
Nominal volume (gal)	15,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	150
Material	Stainless steel

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Table 11.2-2 (Sheet 7 of 7)
Component Data – Liquid Radwaste System

Monitor tanks	
Number	6
Nominal volume (gal)	15,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	150
Material	Stainless steel
Chemical waste tank	
Number	1
Nominal volume (gal)	7,700
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	150
Material	Stainless steel
Degasifier separator (part of vacuum degasifier package)	
Number	1
Nominal volume (gal)	45
Type	Vertical
Design pressure (psig)	150
Design temperature (°F)	200
Material	Stainless steel
Degasifier column (part of vacuum degasifier package)	
Number	1
Nominal volume (gal)	900
Type	Vertical
Design pressure (psig)	150 internal 15 external
Design temperature (°F)	150
Material	Stainless steel

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Table 11.2-3
Summary of Tank Level Indication, Level Annunciators, and Overflows

Tank	Level Indication Location (Note 3)	Alarm Location	Alarm	Overflow To
Effluent holdup	MCR	MCR	High	Room drains to auxiliary building sump which is pumped to waste holdup tank (Note 2)
Waste holdup	MCR	MCR	High	Room (Note 4)
Chemical waste	MCR	MCR	High	Room (Note 2)
Monitor	MCR	MCR	High	Room (Note 5)

Notes:

1. MCR = main control room
2. Room is piped to a floor drain within the auxiliary building, which is seismic Category I and water-tight with curbs or walls of sufficient height to contain the entire contents of the contained tank.
3. Monitoring of the liquid radwaste system is performed through the data display and processing system. Control functions are performed by the plant control system. Appropriate alarms and displays are available in the control room. Local indication and control are available on displays which may be connected to the data display and processing system. See Chapter 7.
4. Room is within the auxiliary building, which is seismic Category I and water-tight with curbs or walls of sufficient height to contain the entire contents of the contained tank.
5. Room is piped to a floor drain within the auxiliary building, which is seismic Category I and water-tight with curbs or walls of sufficient height to contain the entire contents of the contained tank, or to a floor drain within the radwaste building, which is water tight with curbs or walls of sufficient height to contain the entire contents of the contained tank.

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Table 11.2-4
Tank Surge Capacity

Reactor Coolant Drain Tank • Sized to accept 10 gpm of saturated reactor coolant for 30 minutes without discharge or overflow. • Reactor coolant drain tank heat exchanger designed to limit the temperature to less than 175°F with this input assumed to be at 580°F.
Containment Sump • Sized to allow collection of 60 gallons of water between pumping cycles.
Effluent Holdup Tanks • Sized to allow (together) a back-to-back plant shutdown and restart without delay at any time during the first 85 percent of core life. This operation requires nominal processing of the effluent monitor tanks and normal discharge with temporary storage of waste fluid in the cask washdown pit. • Sized to allow (together) a single plant shutdown and restart without delay at any time during the first 80 percent of core life. This operation requires nominal processing to the monitor tanks, but no discharge from the plant.
Other Tanks • Sized based on accommodating maximum input without operator intervention for reasonable lengths of time.

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Table 11.2-5
Decontamination Factors

Decontamination factors assumed per NUREG-0017, Revision 1 (PWR-GALE code input) to be as follows:

Resin Type/Component	Iodine	Cs/Rb	Other
Zeolite/deep bed filter (Note 1)	1	100	1
Cation/waste ion exchanger 1	1	10	10
Mixed/waste ion exchanger 2	100	2 (Note 2)	100
Mixed/waste ion exchanger 3	10	10 (Note 2)	10 (Note 2)

Other components not directly involved in discharge from the plant:

Degasifier Column

Reduce hydrogen by a factor of 40

Assuming inlet flow of 100 gpm at 130°F.

Notes:

1. This component is not included in NUREG-0017. DFs based upon "Reduction of Cesium and Cobalt Activity in Liquid Radwaste Processing Using Clinoptilolite Zeolite at Duke Power Company," by O.E. Ekechokwu, et al., Proc. Waste Management '92, Tucson, Arizona, March 1992, University of Arizona, Tucson.
2. Credit for this decontamination factor not taken in determination of anticipated annual releases.

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Table 11.2-6 (Sheet 1 of 3)
Input Parameters for the GALE Computer Code

Thermal power level (MWt)	3400
Mass of primary coolant (lb)	4.35×10^5
Primary system letdown rate (gpm)	100
Letdown cation demineralizer flow rate, annual average (gpm)	10
Number of steam generators	2
Total steam flow (lb/hr)	14.97×10^6
Mass of liquid in each steam generator (lb)	1.75×10^5
Total blowdown rate (lb/hr)	4.2×10^4
Blowdown treatment method	0(1)
Condensate demineralizer regeneration time	N/A
Condensate demineralizer flow fraction	0.33
Primary coolant bleed for boron control	
Bleed flow rate (gpd)	435
Decontamination factor for I(3)	10^3
Decontamination factor for Cs and Rb(3)	10^3
Decontamination factor for others(3)	10^3
Collection time (day)	30
Process and discharge time (day)	0
Fraction discharged	1.0
Equipment Drains and Clean Waste	
Equipment drains flow rate (gpd)	290
Fraction of reactor coolant activity	0.368
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^3
Decontamination factor for others	10^3
Collection time (day)	30
Process and discharge time (day)	0
Fraction discharged	1.0

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Table 11.2-6 (Sheet 2 of 3)
Input Parameters for the GALE Computer Code

Dirty Waste	
Dirty waste input flow rate (gpd)	1200
Fraction of reactor coolant activity	0.001
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^3
Decontamination factor for others	10^3
Collection time (day)	10
Process and discharge time (day)	0
Fraction discharged	1.0
Blowdown Waste	
Blowdown fraction processed	1
Decontamination factor for I	100
Decontamination factor for Cs and Rb	10
Decontamination factor for others	100
Collection time	N/A
Process and discharge time	N/A
Fraction discharged	0
Regenerant Waste	N/A

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Table 11.2-6 (Sheet 3 of 3)
Input Parameters for the GALE Computer Code

Gaseous Waste System	
Continuous gas stripping of full letdown purification flow	None
Holdup time for xenon, (days)	38
Holdup time for krypton, (days)	2
Fill time of decay tanks for gas stripper	N/A
Gas waste system: HEPA filter	None
Auxiliary building: Charcoal filter	None
Auxiliary building: HEPA filter	None
Containment volume (ft^3)	2.1×10^6
Containment atmosphere internal cleanup rate (ft^3/min)	N/A
Containment high volume purge:	
Number of purges per year (in addition to two shutdown purges)	0
Charcoal filter efficiency (%)	90
HEPA filter efficiency (%)	99
Containment normal continuous purge rate (ft^3/min) (based on 20 hrs/week at 4000 ft^3/min)	500
Charcoal filter efficiency (%)	90
HEPA filter efficiency (%)	99
Fraction of iodine released from blowdown tank vent	N/A
Fraction of iodine removed from main condenser air ejector release	0.0
Detergent Waste Decontamination Factor	0.0(2)

Notes:
1. A "0" is input to indicate that the blowdown is recycled to the condensate system after treatment in the blowdown system.
2. A "0.0" is input to indicate that the plant does not have an onsite laundry.
3. Input decontamination factors result in overall decontamination factors of 1000 for iodine, 1000 for cesium and rubidium, and 1000 for other elements.

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Table 11.2-7 (Sheet 1 of 2)
Releases to Discharge Canal Calculated by GALE Code

Nuclide	Shim Bleed +Equip. Drains	Miscellaneous Wastes	Turbine Building	Total Release
	Activity Release[(1)], Ci/year	Activity Release[(1)], Ci/year	Activity Release[(1)], Ci/year	Activity Release[(1)], Ci/year
Corrosion and Activation Products				
C-14	1.9E-06	negl.	negl.	1.9E-06
Na-24	9.4E-04	6.3E-06	7.6E-05	1.0E-03
Cl-36	negl.	negl.	negl.	negl.
Cr-51	1.2E-03	3.4E-06	7.5E-06	1.2E-03
Mn-54	8.6E-04	1.9E-06	3.8E-06	8.6E-04
Fe-55	1.3E-02	2.9E-05	5.7E-05	1.3E-02
Fe-59	1.3E-04	negl.	negl.	1.3E-04
Co-58	1.1E-02	2.7E-05	5.5E-05	1.1E-02
Co-60	6.1E-03	1.4E-05	2.5E-05	6.1E-03
Ni-63	1.4E-02	3.2E-05	5.8E-05	1.5E-02
Zn-65	2.7E-04	negl.	1.2E-06	2.7E-04
Nb-94	negl.	negl.	negl.	negl.
W-187	7.6E-05	negl.	4.7E-06	8.1E-05
U-234	negl.	negl.	negl.	negl.
U-235	negl.	negl.	negl.	negl.
U-238	negl.	negl.	negl.	negl.
Np-237	negl.	negl.	negl.	negl.
Pu-238	negl.	negl.	negl.	negl.
Pu-239	negl.	negl.	negl.	negl.
Pu-240	negl.	negl.	negl.	negl.
Pu-241	2.2E-06	negl.	negl.	2.2E-06
Pu-242	negl.	negl.	negl.	negl.
Am-241	negl.	negl.	negl.	negl.
Am-243	negl.	negl.	negl.	negl.
Cm-242	negl.	negl.	negl.	negl.
Cm-244	negl.	negl.	negl.	negl.
Fission Products				
As-76	negl.	negl.	negl.	negl.
Br-82	negl.	negl.	negl.	negl.
Rb-86	negl.	negl.	negl.	negl.
Rb-88	1.1E-05	negl.	negl.	1.1E-05
Sr-89	6.4E-05	negl.	negl.	6.5E-05
Sr-90	6.6E-06	negl.	negl.	6.7E-06
Y-91	2.4E-06	negl.	negl.	2.5E-06
Zr-95	1.9E-04	negl.	negl.	1.9E-04
Nb-95	1.7E-04	negl.	negl.	1.7E-04

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Table 11.2-7 (Sheet 2 of 2)
Releases to Discharge Canal Calculated by GALE Code

Nuclide	Shim Bleed +Equip. Drains	Miscellaneous Wastes	Turbine Building	Total Release
	Activity Release[(1)], Ci/year	Activity Release[(1)], Ci/year	Activity Release[(1)], Ci/year	Activity Release[(1)], Ci/year
Mo-99	5.0E-04	3.1E-06	1.4E-05	5.2E-04
Tc-99m	4.8E-04	3.0E-06	1.0E-05	4.9E-04
Tc-99	negl.	negl.	negl.	negl.
Ru-103	3.2E-03	8.5E-06	1.8E-05	3.3E-03
Ru-106	negl.	negl.	negl.	negl.
Ag-110m	6.9E-04	1.6E-06	3.1E-06	7.0E-04
Sn-117m	negl.	negl.	negl.	negl.
Sb-122	negl.	negl.	negl.	negl.
Sb-124	negl.	negl.	negl.	negl.
Sb-125	negl.	negl.	negl.	negl.
I-129	negl.	negl.	negl.	negl.
I-131	4.1E-04	1.7E-06	6.7E-06	4.2E-04
I-132	5.1E-04	2.5E-06	2.3E-05	5.4E-04
I-133	7.1E-04	4.7E-06	7.2E-05	7.8E-04
I-134	1.6E-04	1.1E-06	negl.	1.6E-04
Cs-134	2.0E-04	negl.	negl.	2.1E-04
I-135	5.5E-04	3.6E-06	8.7E-05	6.4E-04
Cs-136	2.5E-04	negl.	2.3E-06	2.5E-04
Cs-137	6.1E-04	1.4E-06	2.9E-06	6.2E-04
Ba-140	3.6E-04	1.3E-06	3.0E-06	3.7E-04
La-140	4.8E-04	1.8E-06	5.5E-06	4.9E-04
Ce-144	2.1E-03	4.8E-06	9.2E-06	2.2E-03
Pr-144	2.1E-03	4.8E-06	9.2E-06	2.2E-03
All Others	negl.	negl.	negl.	negl.
Total	**6.6E-02**	**1.7E-04**	**5.8E-04**	**6.6E-02**

Tritium Release in Liquid Effluents[(2)] = 902 Ci/Yr

Notes:
(1) Values less than 1 microcurie are considered to be negligible
(2) Tritium release based on Westinghouse TRICAL computer code

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Table 11.2-8 (Sheet 1 of 2)
Comparison of Annual Average Liquid Release Concentrations with 10 CFR 20 for Expected Releases Effluent Concentration Limits

Nuclide	Discharge Concentration(1) (μCi/ml)	Effluent Concentration Limit(2) (μCi/ml)	Fraction of Concentration Limit
Na-24	3.8E-10	5.0E-05	7.5E-06
Cr-51	4.5E-10	5.0E-04	9.1E-07
Mn-54	3.2E-10	3.0E-05	1.1E-05
Fe-55	4.9E-09	1.0E-04	4.9E-05
Fe-59	4.9E-11	1.0E-05	4.9E-06
Co-58	4.1E-09	2.0E-05	2.0E-04
Co-60	2.2E-09	3.0E-06	7.5E-04
Zn-65	1.0E-10	5.0E-06	2.0E-05
W-187	3.0E-11	3.0E-05	1.0E-06
Np-239	1.5E-16	2.0E-05	7.4E-12
Br-84	0.0E+00	4.0E-04	0.0E+00
Rb-88	3.9E-12	4.0E-04	9.8E-09
Sr-89	2.4E-11	8.0E-06	3.0E-06
Sr-91	0.0E+00	2.0E-05	0.0E+00
Y-91m	0.0E+00	2.0E-03	0.0E+00
Y-93	0.0E+00	2.0E-05	0.0E+00
Zr-95	6.9E-11	2.0E-05	3.4E-06
Nb-95	6.1E-11	3.0E-05	2.0E-06
Mo-99	1.9E-10	2.0E-05	9.6E-06
Tc-99m	1.8E-10	1.0E-03	1.8E-07
Ru-103	1.2E-09	3.0E-05	4.0E-05
Rh-103m	1.2E-09	6.0E-03	2.0E-07
Ru-106	1.0E-13	3.0E-06	3.3E-08
Rh-106	1.0E-13	6.0E-03	1.7E-11
Ag-110m	2.6E-10	6.0E-06	4.3E-05
Ag-110	3.3E-11	0.0E+00	---
Te-129m	0.0E+00	7.0E-06	0.0E+00
Te-129	0.0E+00	4.0E-04	0.0E+00
Te-131m	0.0E+00	8.0E-06	0.0E+00
Te-131	0.0E+00	8.0E-05	0.0E+00
I-131	1.5E-10	1.0E-06	1.5E-04
Te-132	5.9E-11	9.0E-06	6.5E-06
I-132	2.0E-10	1.0E-04	2.0E-06

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Table 11.2-8 (Sheet 2 of 2)
Comparison of Annual Average Liquid Release Concentrations with 10 CFR 20 for Expected Releases Effluent Concentration Limits

Nuclide	Discharge Concentration(1) (μCi/ml)	Effluent Concentration Limit(2) (μCi/ml)	Fraction of Concentration Limit
I-133	2.9E-10	7.0E-06	4.1E-05
I-134	5.9E-11	4.0E-04	1.5E-07
Cs-134	7.6E-11	9.0E-07	8.4E-05
I-135	2.4E-10	3.0E-05	7.9E-06
Cs-136	9.3E-11	6.0E-06	1.5E-05
Cs-137	2.3E-10	1.0E-06	2.3E-04
Ba-137m	4.3E-09	0.0E+00	---
Ba-140	1.3E-10	8.0E-06	1.7E-05
La-140	1.8E-10	9.0E-06	2.0E-05
Ce-141	2.3E-11	3.0E-05	7.6E-07
Ce-143	0.0E+00	2.0E-05	0.0E+00
Pr-143	0.0E+00	2.0E-05	0.0E+00
Ce-144	7.9E-10	3.0E-06	2.6E-04
Pr-144	7.9E-10	6.0E-04	1.3E-06
Tritium (H-3)	9.4E-05	1.0E-03	9.4E-02
		Total =	0.096

Notes:
1. Annual average discharge concentration based on release of average daily discharge for 292 days per year with 6000 gpm dilution flow.
2. Effluent concentration limits are from Reference 1.

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Table 11.2-9 (Sheet 1 of 2)
Comparison of Annual Average Liquid Release Concentrations with 10 CFR 20 Effluent Concentration Limits for Releases with Maximum Defined Fuel Defects

Nuclide	Discharge Concentration(1) (μCi/ml)	Effluent Concentration Limit(2) (μCi/ml)	Fraction of Concentration Limit
Na-24	3.8E-10	5.0E-05	7.5E-06
Cr-51	7.1E-10	5.0E-04	1.4E-06
Mn-54	5.0E-10	3.0E-05	1.7E-05
Fe-55	4.9E-09	1.0E-04	4.9E-05
Fe-59	7.8E-11	1.0E-05	7.8E-06
Co-58	4.1E-09	2.0E-05	2.0E-04
Co-60	2.2E-09	3.0E-06	7.5E-04
Zn-65	1.0E-10	5.0E-06	2.0E-05
W-187	3.0E-11	3.0E-05	1.0E-06
Np-239	1.5E-16	2.0E-05	7.4E-12
Br-84	0.0E+00	4.0E-04	0.0E+00
Rb-88	3.1E-11	4.0E-04	7.6E-08
Sr-89	2.6E-10	8.0E-06	3.3E-05
Sr-90	1.5E-11	5.0E-07	3.1E-05
Sr-91	0.0E+00	2.0E-05	0.0E+00
Y-91m	0.0E+00	2.0E-03	0.0E+00
Y-93	0.0E+00	2.0E-05	0.0E+00
Zr-95	8.7E-11	2.0E-05	4.3E-06
Nb-95	9.0E-11	3.0E-05	3.0E-06
Mo-99	8.3E-09	2.0E-05	4.2E-04
Tc-99m	8.2E-09	1.0E-03	8.2E-06
Ru-103	1.2E-09	3.0E-05	4.0E-05
Rh-103m	2.2E-09	6.0E-03	3.7E-07
Ru-106	1.9E-13	3.0E-06	6.2E-08
Rh-106	N/A	6.0E-03	0.0E+00
Ag-110m	1.1E-09	6.0E-06	1.8E-04
Ag-110	N/A	0.0E+00	0.0E+00
Te-129m	0.0E+00	7.0E-06	0.0E+00
Te-129	0.0E+00	4.0E-04	0.0E+00
Te-131m	0.0E+00	8.0E-06	0.0E+00
Te-131	0.0E+00	8.0E-05	0.0E+00
I-131	1.7E-08	1.0E-06	1.7E-02
Te-132	4.0E-09	9.0E-06	4.5E-04
I-132	8.8E-10	1.0E-04	8.8E-06
I-133	4.4E-09	7.0E-06	6.3E-04

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Table 11.2-9 (Sheet 2 of 2)
Comparison of Annual Average Liquid Release Concentrations with 10 CFR 20 Effluent Concentration Limits for Releases with Maximum Defined Fuel Defects

Nuclide	Discharge Concentration[1] (μCi/ml)	Effluent Concentration Limit[2] (μCi/ml)	Fraction of Concentration Limit
I-134	7.3E-11	4.0E-04	1.8E-07
Cs-134	1.9E-07	9.0E-07	2.1E-01
I-135	1.0E-09	3.0E-05	3.5E-05
Cs-136	1.4E-07	6.0E-06	2.4E-02
Cs-137	1.5E-07	1.0E-06	1.5E-01
Ba-137m	N/A	0.0E+00	0.0E+00
Ba-140	2.2E-09	8.0E-06	2.8E-04
La-140	1.8E-10	9.0E-06	2.0E-05
Ce-141	5.4E-11	3.0E-05	1.8E-06
Ce-143	0.0E+00	2.0E-05	0.0E+00
Pr-143	0.0E+00	2.0E-05	0.0E+00
Ce-144	1.5E-09	3.0E-06	4.9E-04
Pr-144	7.9E-10	6.0E-04	1.3E-06
(Tritium) H-3	9.4E-05	1.0E-03	9.4E-02
		Total =	0.496

Notes:
1. Annual average discharge concentration based on release of average daily discharge for 292 days per year with 6000 gpm dilution flow.
2. Effluent concentrations limits are from Reference 1.

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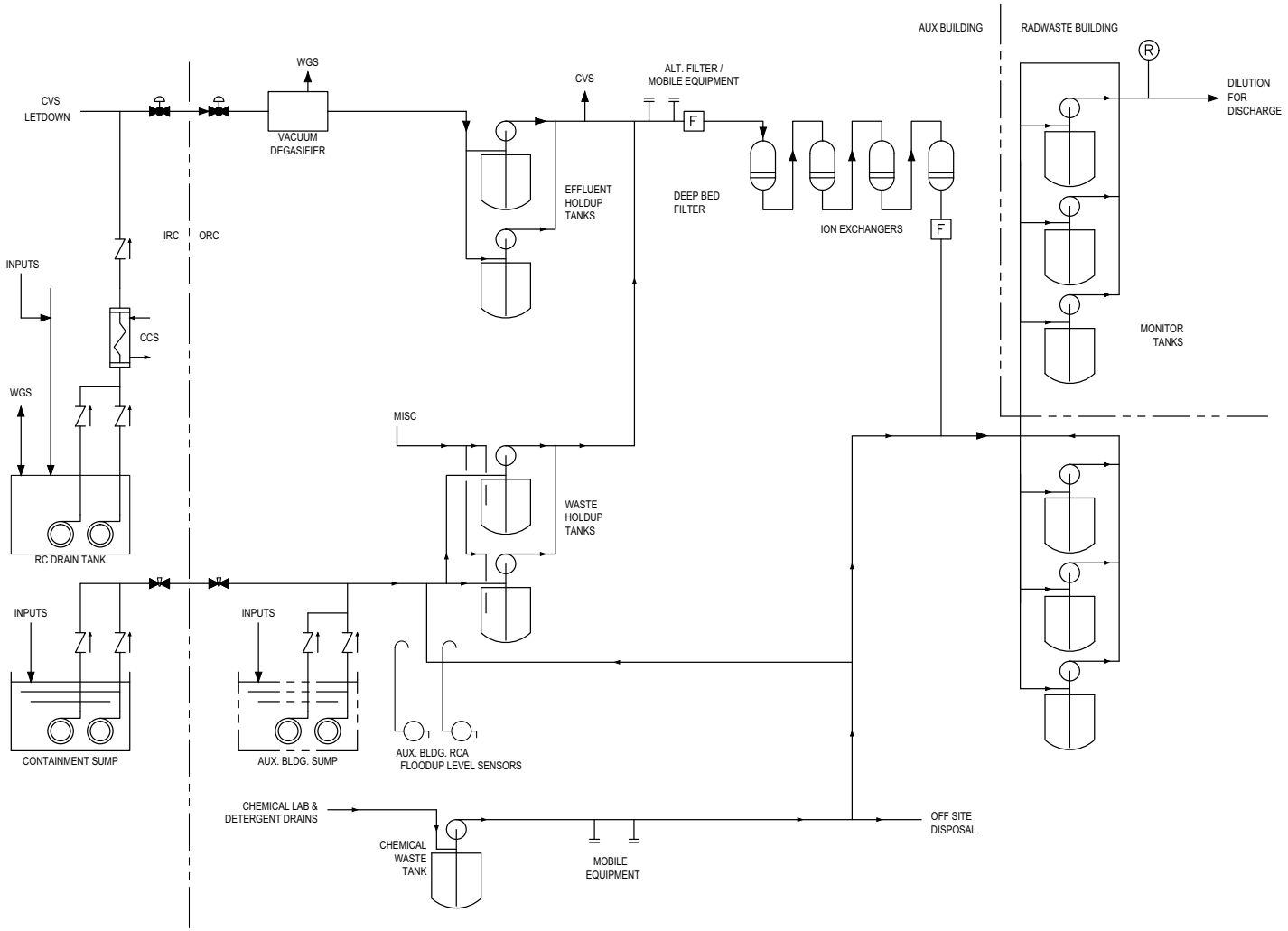

Figure 11.2-1
Liquid Radwaste System
Simplified Piping and Instrumentation Diagram
(REF) WLS

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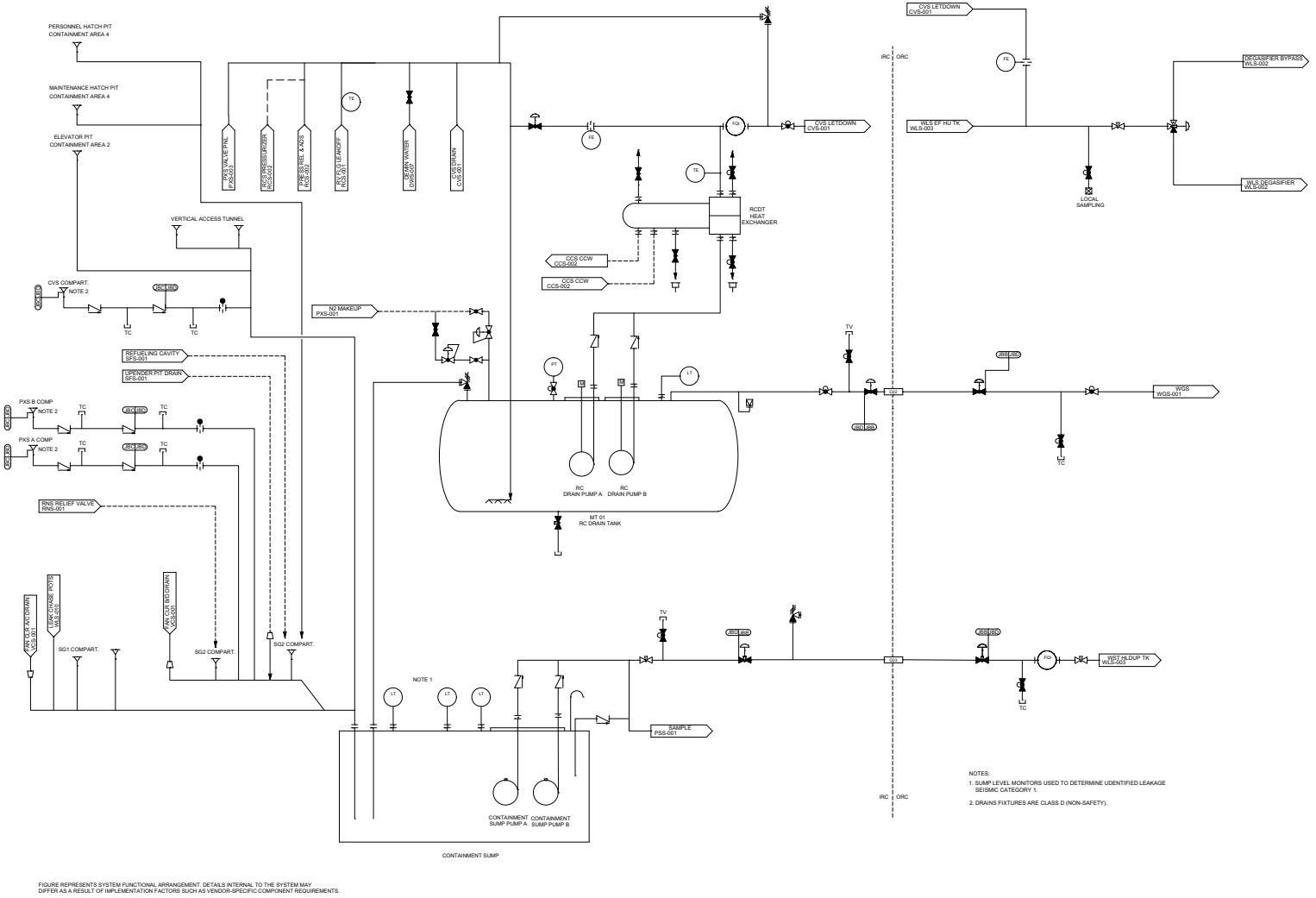

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Figure 11.2-2 (Sheet 1 of 8)
Liquid Radwaste System
Piping and Instrumentation Diagram
(REF) WLS 001

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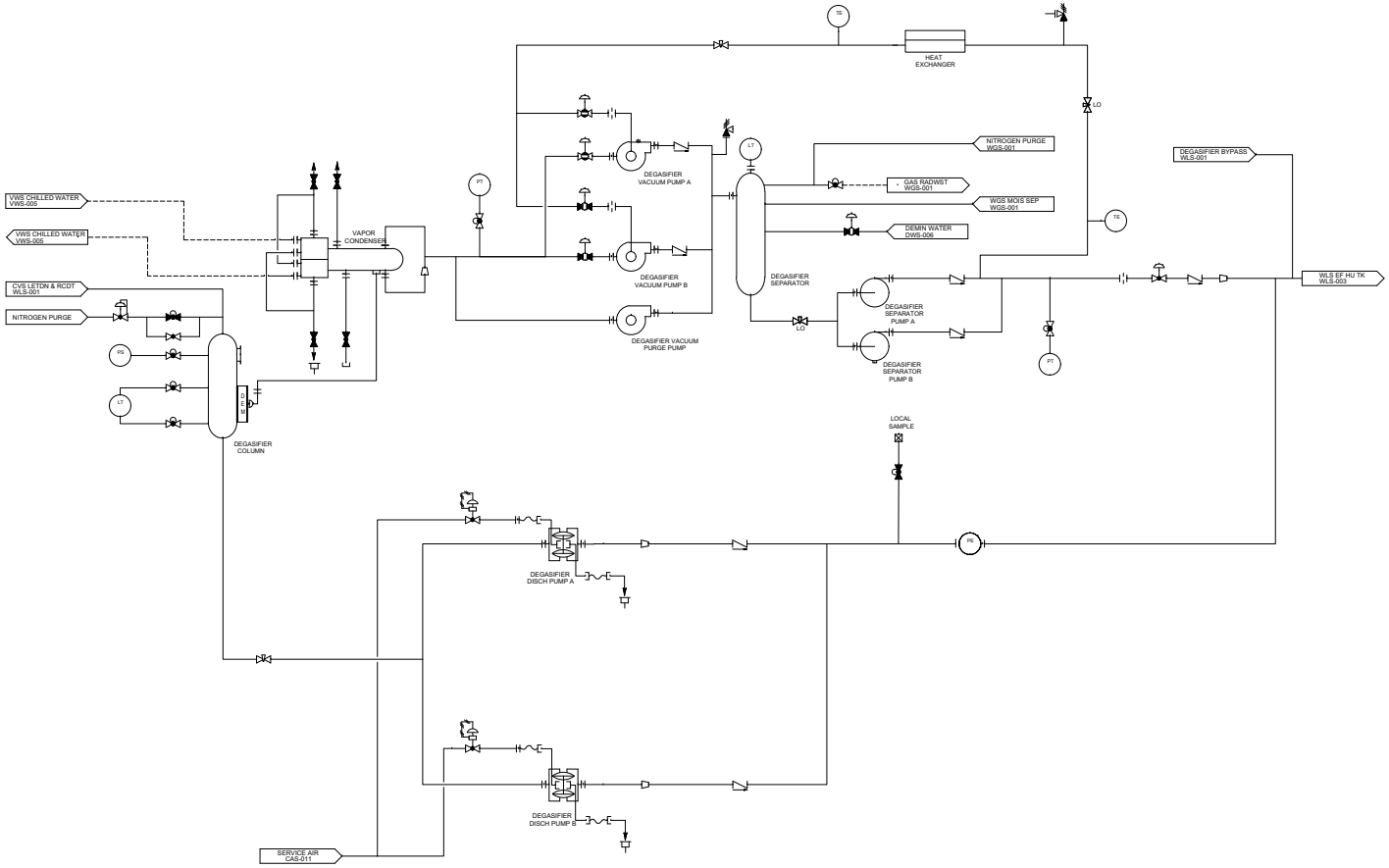

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Inside Auxiliary Building
Figure 11.2-2 (Sheet 2 of 8)
Liquid Radwaste System
Piping and Instrumentation Diagram
(REF) WLS 002

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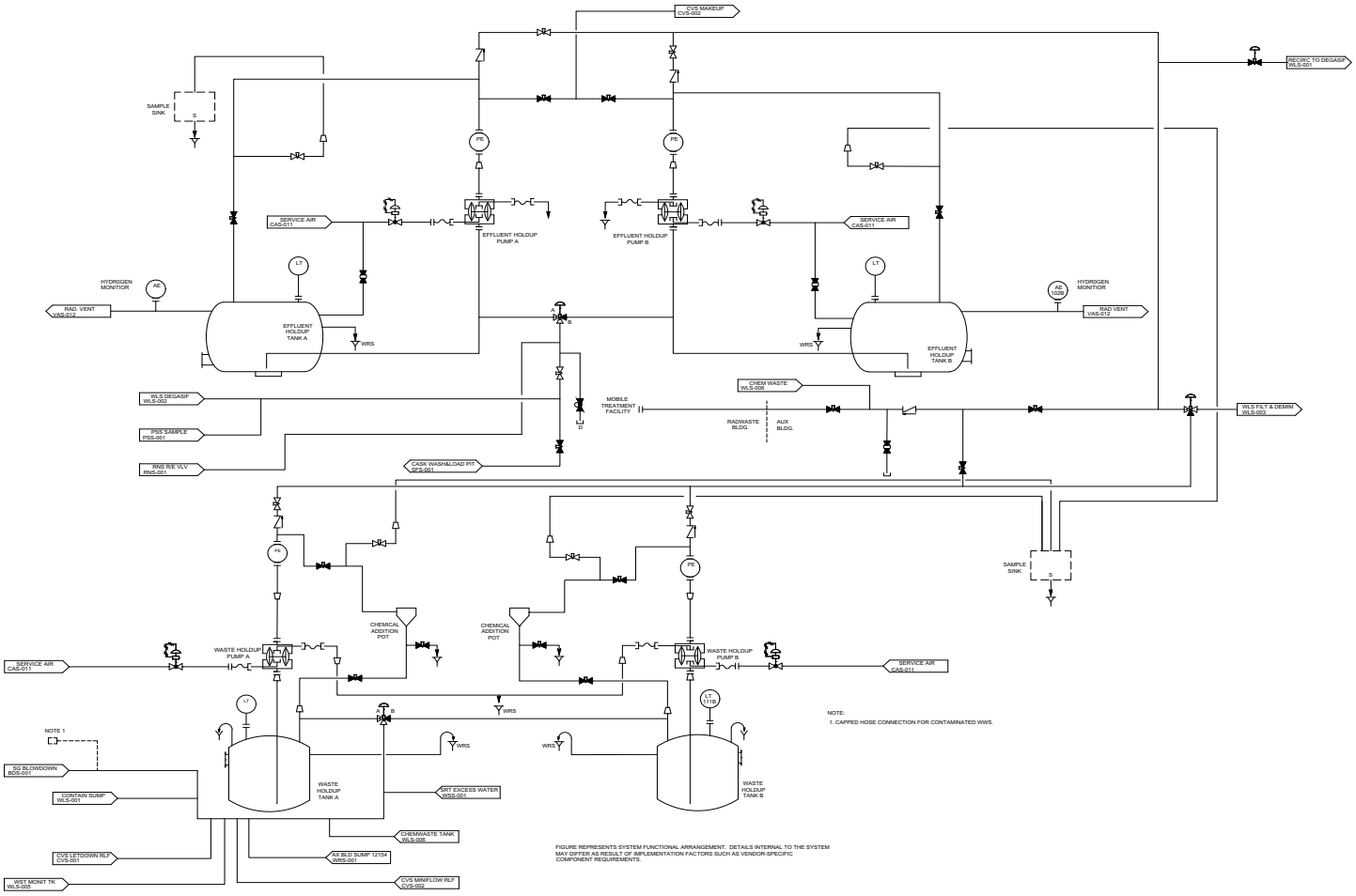

Figure 11.2-2 (Sheet 3 of 8)
Liquid Radwaste System
Piping and Instrumentation Diagram
(REF WLS 003)

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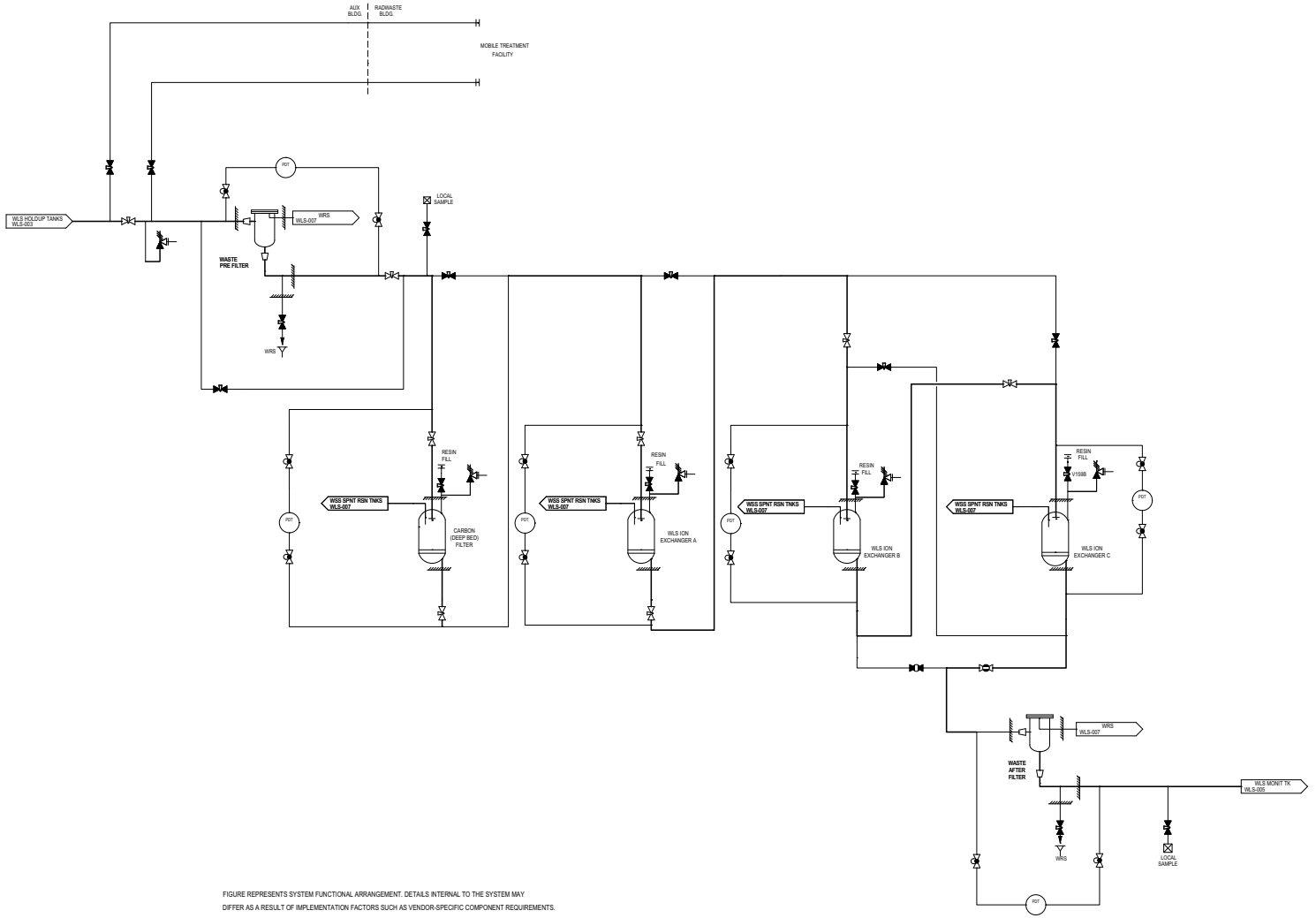

Figure 11.2-2 (Sheet 4 of 8)
Liquid Radwaste System
Piping and Instrumentation Diagram
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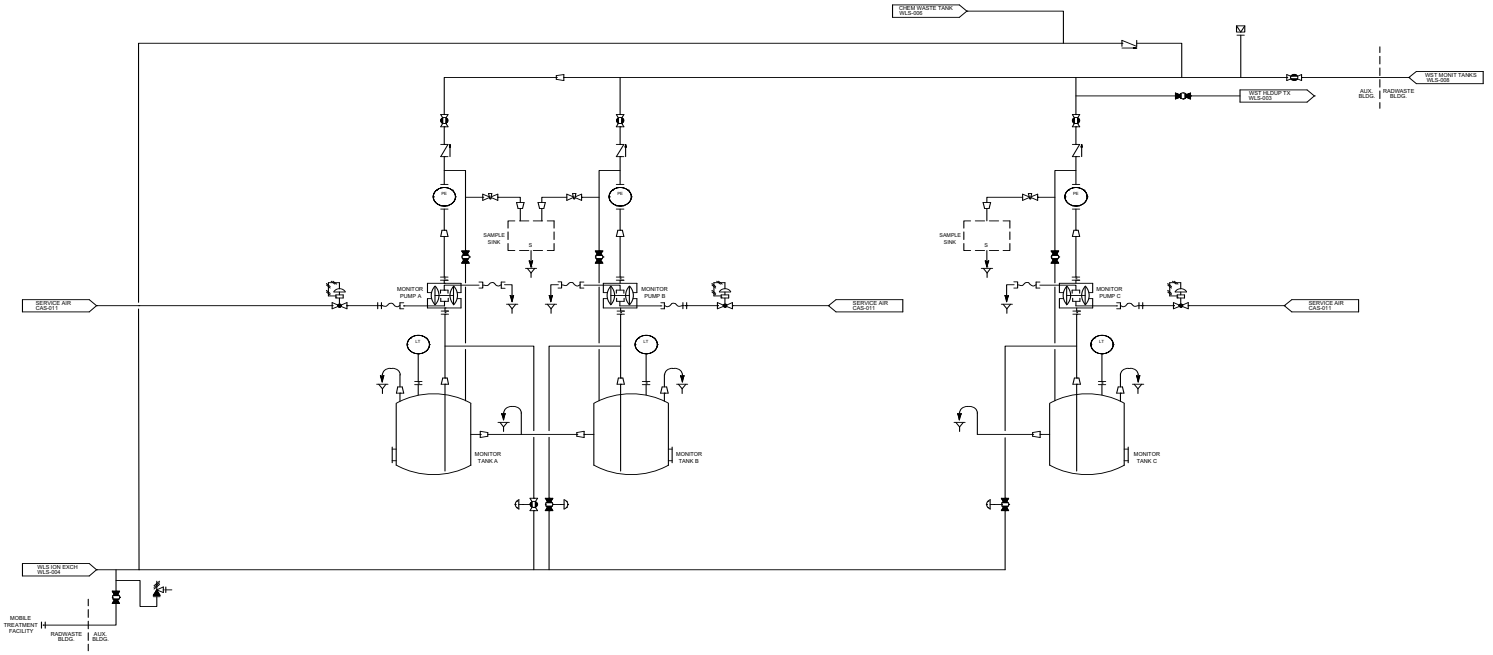

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Figure 11.2-2 (Sheet 5 of 8)
Liquid Radwaste System
Piping and Instrumentation Diagram
(REF) WLS 005

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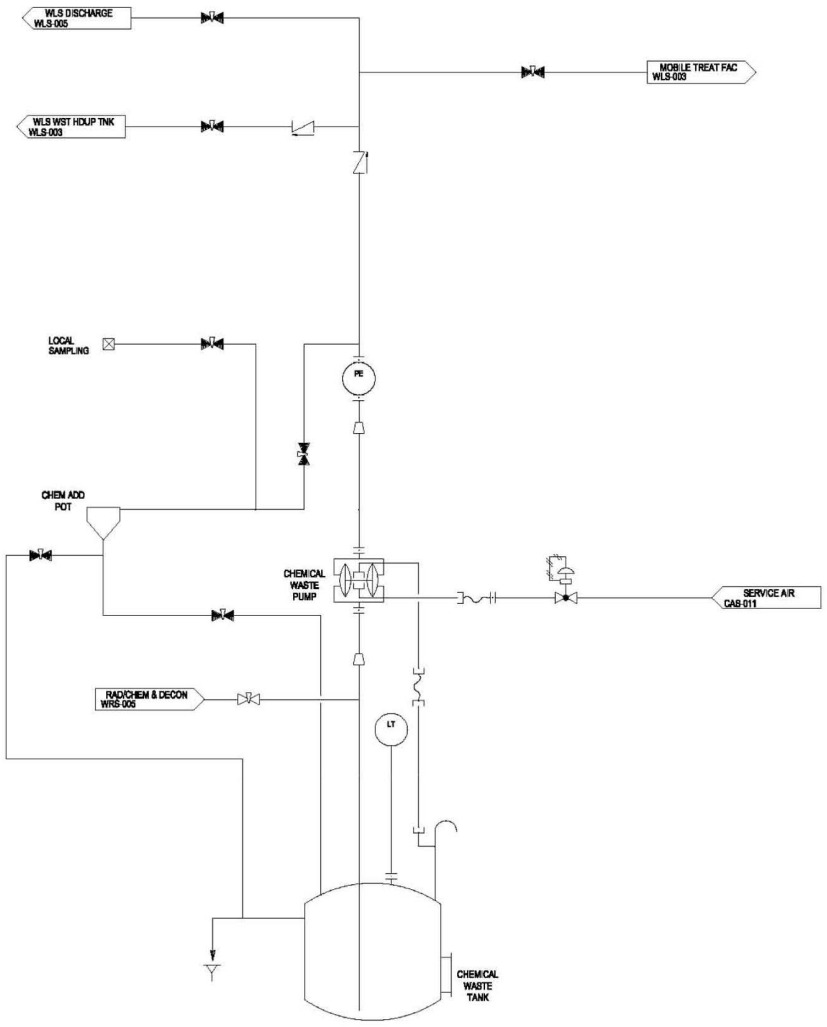

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Inside Auxiliary Building
Figure 11.2-2 (Sheet 6 of 8)
Liquid Radwaste System
Piping and Instrumentation Diagram
(REF) WLS 006

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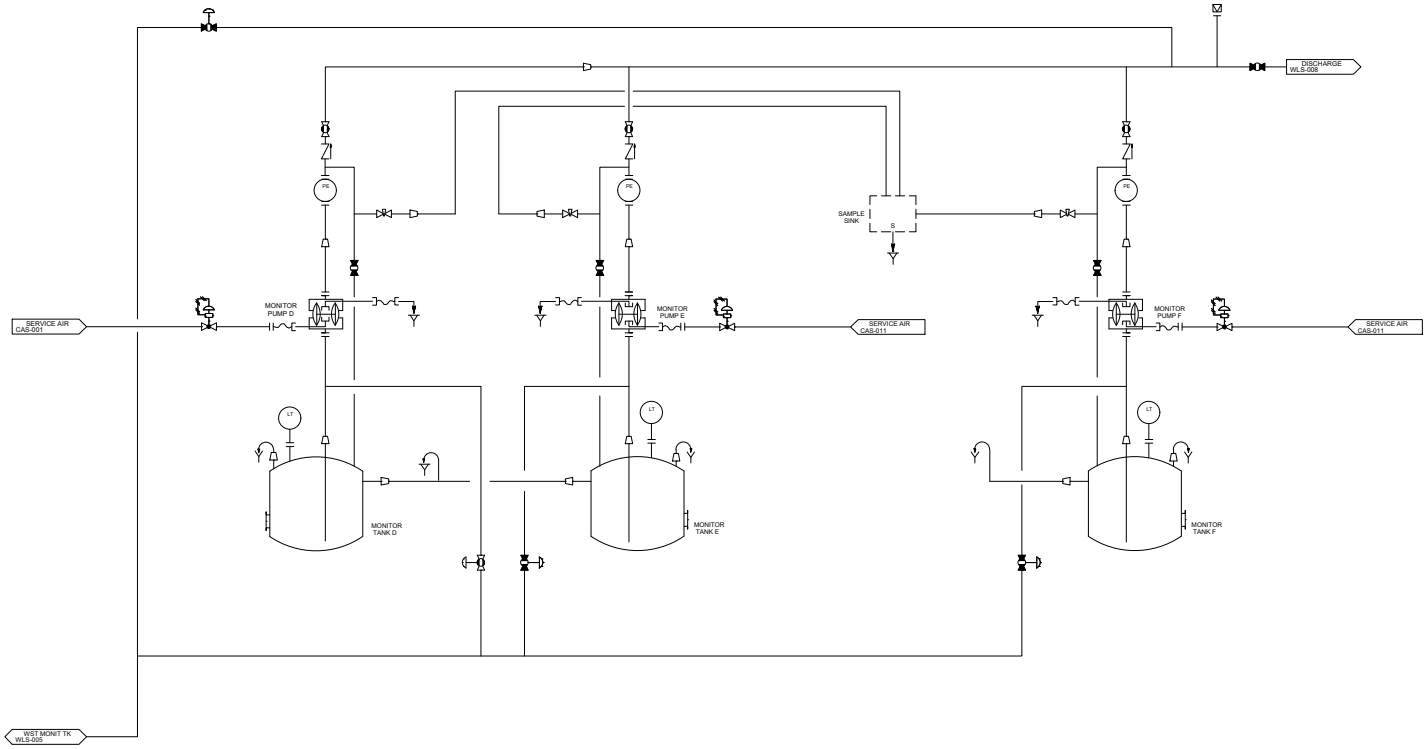

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Inside Radwaste Building
Figure 11.2-2 (Sheet 7 of 8)
Liquid Radwaste System
Piping and Instrumentation Diagram
(REF) WLS 008

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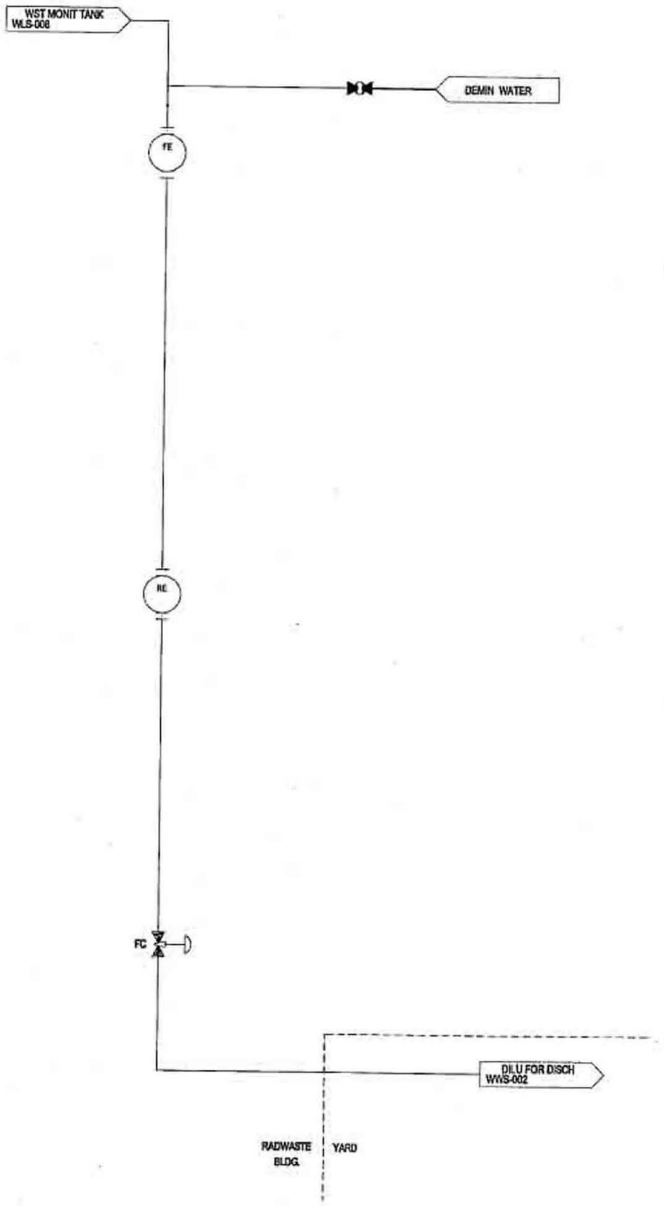

Figure represents system functional arrangement. Details internal to the system may differ as a result of implementation factors such as vendor-specific component requirements.

Inside Radwaste Building
Figure 11.2-2 (Sheet 8 of 8)
Liquid Radwaste System
Piping and Instrumentation Diagram
(REF) WLS 009

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11.3 Gaseous Waste Management System

During reactor operation, radioactive isotopes of xenon, krypton, and iodine are created as fission products. A portion of these radionuclides is released to the reactor coolant because of a small number of fuel cladding defects. Leakage of reactor coolant thus results in a release to the containment atmosphere of the noble gases. Airborne releases can be limited both by restricting reactor coolant leakage and by limiting the concentrations of radioactive noble gases and iodine in the reactor coolant system.

Iodine is removed by ion exchange in the chemical and volume control system (CVS). Removal of the noble gases from the reactor coolant system (RCS) is not normally necessary because the gases will not build up to unacceptable levels when fuel defects are within normally anticipated ranges. If noble gas removal is required because of high reactor coolant system concentration, the chemical and volume control system can be operated in conjunction with the liquid radwaste system degasifier, to remove the gases. See Subsection 9.3.6 for a description of these operations.

The AP1000 gaseous radwaste system (WGS) is designed to perform the following major functions:

- Collect gaseous wastes that are radioactive or hydrogen bearing
- Process and discharge the waste gas, keeping off-site releases of radioactivity within acceptable limits.

In addition to the gaseous radwaste system release pathway, release of radioactive material to the environment occurs through the various building ventilation systems. These systems are described in Section 9.4 with a summary of system air flow rates and filter efficiencies provided in Table 9.4-1. The estimated annual release reported in Subsection 11.3.3 includes contributions from the major building ventilation pathways.

11.3.1 Design Basis

Subsection 1.9.1 discusses the conformance of the gaseous radwaste system design with the criteria of Regulatory Guide 1.143.

11.3.1.1 Safety Design Basis

The gaseous radwaste system serves no safety-related functions and therefore has no nuclear safety design basis.

11.3.1.2 Power Generation Design Basis

11.3.1.2.1 Capacity

11.3.1.2.1.1 Gaseous Waste Collection

The gaseous radwaste system is designed to receive hydrogen bearing and radioactive gases generated during process operation. The radioactive gas flowing into the gaseous radwaste system enters as trace contamination in a stream of hydrogen and nitrogen.

The design basis period of operation is the last 45 days of a fuel cycle. During this time, reactor coolant system dilution and subsequent letdown from the chemical and volume control system into the liquid radwaste system is at a maximum. Gaseous radwaste system inputs are as follows:

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- Letdown diversion for dilution, reactor coolant system with maximum hydrogen concentration. This input is 0.5 standard cubic feet per minute (scfm) on an intermittent basis carrying a very small volume of radiogas, yielding 550 scf total hydrogen.

- Letdown diversion for degasification of the entire reactor coolant system in anticipation of shutdown, assumed to remove gases from the reactor coolant system to a level of 5 cc/kg beginning with the reactor coolant system at the maximum hydrogen concentration of 45 cc/kg. At its maximum this input is 0.58 scfm hydrogen carrying a very small volume of radiogas yielding approximately 281 scf total hydrogen.

- Reactor coolant drain tank liquid transfer to maintain proper reactor coolant drain tank level, assuming 0.25 gallons per minute liquid input from the reactor coolant system, intermittently yielding 0.5 scfm hydrogen and nitrogen carrying a very small volume of radiogas, yielding about 80 scf hydrogen and nitrogen total.

- Reactor coolant drain tank gas venting, conservatively estimated at 1 scf per day during normal operation, yielding 45 scf total nitrogen and hydrogen. Prior to an outage, the pressurizer may be vented to the reactor coolant drain tank as the primary means of reactor coolant system degassing. In this case, the majority of the hydrogen accumulated in the pressurizer gas space is processed through the reactor coolant drain tank to the gaseous radwaste system. The flow rate from the reactor coolant drain tank is 1.6 scfm, with approximately 2330 scf total hydrogen removed prior to an outage.

11.3.1.2.1.2 Waste Gas Processing

The gaseous radwaste system is designed to reduce the controlled activity releases in support of the overall AP1000 release goals.

Given the various inputs to the gaseous radwaste system, with licensing basis assumptions for analysis and with normally operating gaseous radwaste system equipment available, the combined plant releases must be within the limits outlined in 10 CFR 20 and 10 CFR 50 Appendix I (References 1 and 2, respectively).

11.3.1.2.2 Failure Tolerance

11.3.1.2.2.1 System Leakage

The gaseous radwaste system operates at low pressures, slightly above atmospheric pressure, thus limiting the potential for leakage. Manual valves are the type which eliminate the potential for stem leakage. The system is of welded construction to further limit leakage.

11.3.1.2.2.2 Water Incursion

A number of features prevent wetting the activated carbon delay beds. These features include controls and alarms in the liquid radwaste system to prevent high degasifier separator water level, the gas cooler, moisture separator, drain traps, and automatic isolation of the guard bed inlet on high moisture separator level in the gaseous radwaste system. Additional protection is provided by the activated carbon guard bed, which removes residual moisture from the gas stream.

If moisture enters the first activated carbon delay bed, the operator bypasses that bed and either dries it with a nitrogen purge or replaces the activated carbon.

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11.3.1.2.3 Anticipated Operational Occurrences

11.3.1.2.3.1 Prevention of Hydrogen Ignition

Since the carrier gas for the radiogas inputs to the gaseous radwaste system includes hydrogen, the gaseous radwaste system is designed to prevent hydrogen ignition both within its own boundaries and in connected systems (the liquid radwaste system and the nuclear island radioactive ventilation system).

The gaseous radwaste system is operated at a slightly positive pressure to prevent air ingress. The room containing gaseous radwaste system components incorporates a hydrogen monitor to detect leakage out of the system before combustible levels are reached. In addition, continuous oxygen analysis, using independent, redundant monitors, is provided within the gaseous radwaste system. Upon high oxygen level in the system, an alarm alerts the operator. At an oxygen concentration of 3 percent or less, the liquid radwaste system vacuum pumps automatically stop to isolate potentially oxygenated inputs to the gaseous radwaste system, and a valve automatically opens to initiate a nitrogen purge. The discharge isolation valve of the gaseous radwaste system is continuously pressurized with nitrogen to prevent ingress of air into the system from the discharge path.

The gaseous radwaste system also eliminates sources of hydrogen ignition. The system incorporates electrical grounding, and a nitrogen purge. In addition, the system design and control logic minimizes automatic actuation of valves when an oxygen concentration of greater than 3 percent exists in the system. Discharge to the heating, ventilating and air-conditioning duct is downstream of the exhaust fans to provide additional protection against hydrogen ignition.

11.3.1.2.4 Controlled Release of Radioactivity

11.3.1.2.4.1 Expected Releases

The AP1000 design prevents the annual average concentration limits established by 10 CFR 20 (Appendix B, table 2, column 1) (Reference 1) for gaseous releases from being exceeded due to the releases resulting during plant operation. Subsection 11.3.3 describes the calculated releases of radioactive materials from the gaseous radwaste system and other pathways during normal operation.

Subsection 11.3.3 also contains an evaluation which demonstrates that the doses to individuals, at or beyond the site boundary, resulting from the expected releases from the gaseous waste management systems are within numerical design objectives of Appendix I of 10 CFR 50 (Reference 2).

11.3.1.2.4.2 Monitoring Releases

Releases from the gaseous radwaste system are continuously monitored by a radiation detector in the discharge line. In addition, the system includes provisions for taking grab samples of the discharge flow stream for analysis. In this manner, the requirements of General Design Criterion 64 are met as described in Section 3.1. Section 11.5 discusses radiation monitoring.

11.3.1.2.4.3 Operator Error or Equipment Malfunction

To prevent the release of radioactive gases resulting from equipment failure or operator error, a radiation monitor is located in the discharge line. This instrument provides an alarm signal at a high level setpoint to alert operators of rising radiation levels. The monitor is also interlocked with an isolation valve in the discharge line; the valve closes on the same high radiation signal.

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Few operator actions are required during gaseous radwaste system operation since, once aligned for operation, the system operates automatically in response to the control signals from the instrumentation.

11.3.1.3 Compliance with 10 CFR 20.1406

In accordance with the requirements of 10 CFR 20.1406 (Reference 4), the gaseous radwaste system is designed to minimize, to the extent practicable, contamination of the facility and the environment, facilitate decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. This is done through appropriate selection of design technology for the system.

11.3.2 System Description

11.3.2.1 General Description

The AP1000 gaseous radwaste system, as shown on Figure 11.3-1 is a once-through, ambient-temperature, activated carbon delay system. The system includes a gas cooler, a moisture separator, an activated carbon-filled guard bed, and two activated carbon-filled delay beds. Also included in the system are an oxygen analyzer subsystem and a gas sampling subsystem.

The radioactive fission gases entering the system are carried by hydrogen and nitrogen gas. The primary influent source is the liquid radwaste system degasifier. The degasifier extracts both hydrogen and fission gases from the chemical and volume control system letdown flow which is diverted to the liquid radwaste system or from the reactor coolant drain tank discharge.

Reactor coolant degassing is not required during power operation with fuel defects at or below the design basis level of 0.25 percent. However, the gaseous radwaste system periodically receives influent when chemical and volume control system letdown is processed through the liquid radwaste system degasifier during reactor coolant system dilution and volume control operations. Since the degasifier is a vacuum type and requires no purge gas, the maximum gas influent rate to the gaseous radwaste system from the degasifier equals the rate that hydrogen enters the degasifier (dissolved in liquid).

The other major source of input to the gaseous radwaste system is the reactor coolant drain tank. Hydrogen dissolved in the influent to the reactor coolant drain tank enters the gaseous radwaste system either via the tank vent or the liquid radwaste system degasifier discharge.

The tank vent is normally closed, but is periodically opened on high pressure to vent the gas that has come out of solution. The reactor coolant drain tank liquid is normally discharged to the liquid radwaste system via the degasifier, where the remaining hydrogen is removed.

The reactor coolant drain tank is purged with nitrogen gas to discharge nitrogen and fission gases to the gaseous radwaste system before operations requiring tank access. The reactor coolant drain tank is also purged with nitrogen gas to dilute and discharge oxygen after tank servicing or inspection operations which allow air to enter the tank.

Influents to the gaseous radwaste system first pass through the gas cooler where they are cooled to about 40°F by the chilled water system. Moisture formed due to gas cooling is removed in the moisture separator.

After leaving the moisture separator, the gas flows through a guard bed that protects the delay beds from abnormal moisture carryover or chemical contaminants. The gas then flows through two delay beds in series where the fission gases undergo dynamic adsorption by the activated carbon and are thereby delayed relative to the hydrogen or nitrogen carrier gas flow. Radioactive decay of the fission

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gases during the delay period significantly reduces the radioactivity of the gas flow leaving the system.

The effluent from the delay bed passes through a radiation monitor and discharges to the ventilation exhaust duct. The radiation monitor is interlocked to close the gaseous radwaste system discharge isolation valve on high radiation. The discharge isolation valve also closes on low ventilation system exhaust flow rate to prevent the accumulation of hydrogen in the aerated vent.

11.3.2.2 System Operation

11.3.2.2.1 Normal Operation

The gaseous radwaste system is used intermittently. Most of the time during normal operation of the AP1000, the gaseous radwaste system is inactive. When there is no waste gas inflow to the system, the discharge isolation valve closes, which maintains the gaseous radwaste system at a positive pressure, preventing the ingress of air during the periods of low waste gas flow.

When the gaseous radwaste system is in use, its operation is passive, using the pressure provided by the influent sources to drive the waste gas through the system.

The largest input to the gaseous radwaste system during steady state power operations is from the liquid radwaste system degasifier, which processes the chemical and volume control system letdown flow when diverted to the liquid radwaste system and the liquid effluent from the liquid radwaste system reactor coolant drain tank.

The chemical and volume control system letdown flow is diverted to the liquid radwaste system during dilutions, borations, and reactor coolant system degassing in anticipation of shutdown. The design basis influent rate from the liquid radwaste system degasifier is the full diversion of the chemical and volume control system letdown flow, when the reactor coolant system is operating with maximum allowable hydrogen concentration. Since the liquid radwaste system degasifier is a vacuum type that operates without a purge gas, this input rate is very small, about 0.6 scfm.

The liquid radwaste system degasifier is also used to degas liquid pumped out of the reactor coolant drain tank. The amount of fluid pumped out, and therefore the gas sent to the gaseous radwaste system, is dependent upon the input into the reactor coolant drain tank. This is smaller than the input from the chemical and volume control system letdown line.

The final input to the gaseous radwaste system is from the reactor coolant drain tank vent. A nitrogen cover gas is maintained in the reactor coolant drain tank. This input consists of nitrogen, hydrogen, and radioactive gases. The tank operates at nearly constant level, with its vent line normally closed, so this input is often minimal. Venting is required only after enough gas has evolved from the input fluid to increase the reactor coolant drain tank pressure. Input from the tank to the gaseous radwaste system increases temporarily during reactor coolant system degassing prior to shutdown.

The influent first passes through a gas cooler. Chilled water flows through the gas cooler at a fixed rate to cool the waste gas to about 40°F regardless of waste gas flow rate. Moisture formed due to gas cooling is removed in the moisture separator, and collected water is periodically discharged automatically. To reduce the potential for waste gas bypass of the gas cooler in the event of valve leakage, a float-operated drain trap is provided which automatically closes on low water level.

The gas leaving the gas cooler is monitored for temperature, and a high alarm alerts the operator to an abnormal condition requiring attention. Oxygen concentration is also monitored. On a high oxygen alarm, a nitrogen purge is automatically injected into the influent line.

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The waste gas then flows through the guard bed, where chemical (oxidizing) contaminants are removed. The guard bed also removes any remaining excessive moisture from the waste gas.

The waste gas then flows through the two delay beds where xenon and krypton are delayed by a dynamic adsorption process. The discharge line is equipped with a valve that automatically closes on either high radioactivity in the gaseous radwaste system discharge line or low ventilation exhaust duct flow.

The adsorption of radioactive gases in the delay bed occurs without reliance on active components or operator action. Operator error or active component failure does not result in an uncontrolled release of radioactivity to the environment. Failure to remove moisture prior to the delay beds (due to loss of chilled water or other causes) results in a gradual reduction in gaseous radwaste system performance. Reduced performance is indicated by high temperature and discharge radiation alarms. High-high radiation automatically terminates discharge.

11.3.2.2.2 Purge Operations

The gaseous radwaste system is purged with nitrogen gas to expel residual oxygen gas after servicing operations. The system is purged until the effluent from the outlet indicates a low oxygen concentration. The gaseous radwaste system oxygen analyzer is temporarily aligned to monitor the flow in the discharge line. Nitrogen connections are also provided to the sample system and to the system discharge line for purge before and after maintenance operations.

11.3.2.3 Component Description

The general descriptions and summaries of the design basis requirements for the gaseous radwaste system components follow. Table 11.3-2 lists the key design parameters for the gaseous radwaste system components.

The seismic design classification and safety classification for the gaseous radwaste system components are listed in Section 3.2. The components listed are located in the Seismic Category I Nuclear Island.

11.3.2.3.1 Sample Pumps

Two sample pumps are provided. One sample pump normally operates continuously to provide flow through the oxygen analyzers. The other sample pump is periodically used to provide flow from various sample points through a sample cylinder. It is used as a backup to provide flow through the oxygen analyzers.

11.3.2.3.2 Gas Cooler

The gas cooler heat exchanger is designed to cool the gas flow to near the temperature of the chilled water supply (40°F) for efficient moisture removal. The pressure of the gas flow through the gas cooler is less than the chilled water pressure to minimize the potential for contaminating the chilled water system.

11.3.2.3.3 Gaseous Radwaste System Tanks

Moisture Separator

The moisture separator is sized for the design basis purge gas flow rate and is oversized for the lower normal flow rate. The unit includes connections for water level instrumentation for detecting high and low water levels.

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Guard Bed

The activated carbon guard bed protects the delay beds from abnormal moisture or chemical contaminants. Under normal operating conditions, the guard bed provides increased delay time for xenon and krypton.

The flow through the activated carbon bed is downward. A retention screen on the outlet of the guard bed prevents the loss of activated carbon from the unit. Activated carbon can be added to or vacuumed from the unit via a blind flange port.

Delay Beds

Two activated carbon delay beds in series are provided. Together, the beds provide 100 percent of the stated system capacity under design basis conditions. During normal operation a single bed provides adequate performance. This provides operational flexibility to permit continued operation of the gaseous radwaste system in the event of operational upsets in the system that requires isolation of one bed.

The waste gas flows vertically through columns of activated carbon. The activated carbon mass is given in Table 11.3-1.

No retention screens are required on the delay beds since the flow enters and leaves each delay bed at its top.

The guard bed and the delay beds, including supports, in the gaseous radwaste system are designed for seismic loads in conformance with Regulatory Guide 1.143. These are the only AP1000 components used to store or delay the release of gaseous radioactive waste. The beds are located in the seismic Category I auxiliary building at elevation 66′6″.

11.3.2.3.4 Remotely Operated Valves

Moisture Separator Level Control Valve

This normally closed, fail-open globe valve is located in the liquid drain line from the moisture separator outlet line. It maintains the level in the moisture separator by regulating the flow from the moisture separator to the liquid radwaste system. The valve receives a signal to automatically open on a high level in the moisture separator and to close on low level. The valve can also be manually controlled from the gaseous waste panel.

A float-operated drain trap serves as a backup to this valve. This drain trap automatically closes on a low water level in the moisture separator to stop drain flow to the liquid radwaste system in the event of a valve or instrument failure. This prevents waste gas bypass around the gas cooler due to level control valve failure.

Gaseous Radwaste System Discharge Isolation Valve

This normally closed, fail-closed globe valve is at the outlet of the system as shown in Figures 11.3-1 and 11.3-2. The valve opens during operation of the system. The valve is interlocked to close on a high radiation signal in the gaseous radwaste system discharge line to prevent the release of radioactivity in the event of a gaseous radwaste system failure. The valve also receives a signal to automatically close in the event of a low ventilation system exhaust flow rate which prevents accumulation of a flammable or explosive concentration of hydrogen in the aerated vent line.

Manual control is provided on the gaseous radwaste panel.

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Nitrogen Purge Pressure Control Valve

This is a self-contained pressure regulating valve in the nitrogen purge line. It is set to maintain a small positive pressure in the gaseous radwaste system to prevent ingress of air during periods of low flow.

11.3.3 Radioactive Releases

Releases of radioactive effluent by way of the atmospheric pathway occur due to:

- Venting of the containment which contains activity as a result of leakage of reactor coolant and as a result of activation of naturally occurring Ar-40 in the atmosphere to form radioactive Ar-41
- Ventilation discharges from the auxiliary building which contains activity as a result of leakage from process streams
- Condenser air removal system and gland seal system, which connect together as one vent line discharging from the turbine island vent (gaseous activity entering the secondary coolant as a result of primary to secondary leakage is released via this pathway)
- Gaseous radwaste system discharges.

These releases are on-going throughout normal plant operations. There is no gaseous waste holdup capability in the gaseous waste management system and thus no criteria are required for determining the timing of releases or the release rates to be used.

There are no gaseous effluent site interface parameters outside of the Westinghouse scope.

11.3.3.1 Discharge Requirements

The release of radioactive gaseous and particulate effluents to the atmosphere may not exceed the concentration limits specified in Reference 1 nor may the releases result in the annual offsite dose limits specified in 10 CFR 50, Appendix I (Reference 2) being exceeded.

11.3.3.2 Estimated Annual Releases

The annual average airborne releases of radionuclides from the plant are determined using the PWR-GALE code (Reference 3). The GALE code models releases using realistic source terms derived from data obtained from the experience of many operating pressurized water reactors. The code input parameters used in the analysis to model the AP1000 plant are provided in Table 11.2-6. The expected annual releases for a single unit site are presented in Table 11.3-3.

To demonstrate compliance with the effluent concentration limits in Reference 1, the expected releases from Table 11.3-3 are used to determine the annual average concentration at the site boundary, and the results are compared with the Reference 1 concentration limits for unrestricted areas in Table 11.3-4. As shown in Table 11.3-4, the overall fraction of the effluent concentration limit for the expected releases is 0.030, which is significantly below the allowable value of 1.0.

11.3.3.3 Release Points

Airborne effluents are normally released through the plant vent or the turbine island vent. The plant vent provides the release path for containment venting releases, auxiliary building ventilation releases, annex building releases, radwaste building releases, and gaseous radwaste system

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discharge. The turbine island vents, drains, and relief system (TDS) provides the release path for the condenser air removal system and gland seal condenser exhaust.

11.3.3.4 Estimated Doses

With the annual airborne releases listed in Table 11.3-3, the air doses at ground level at the site boundary are 2.1 mrad for gamma radiation and 10.1 mrad for beta radiation. These doses are based on the annual average atmospheric dispersion factor from Section 2.3 (2.0×10^{-5} seconds per cubic meter). These doses are below the 10 CFR 50, Appendix I, design objectives of 10 mrad per year for gamma radiation or 20 mrad per year for beta radiation.

The radiological consequences due to a single failure of an active component in the gaseous radwaste system are evaluated assuming a 1- hour bypass of the delay beds and 30 minutes of decay before release to the environs. This analysis assumes a pre-existing condition of operation with reactor coolant activity corresponding to 1 percent fuel defects as described in the Note for Table 11.1-2. Using the site boundary (0 to 2 hr) atmospheric dispersion factor from Table 2-1, the site boundary whole body dose is 0.1 rem.

11.3.3.4.1 Not Used.

11.3.3.4.2 Not Used.

11.3.3.4.3 Gaseous Radwaste Cost-Benefit Analysis Methodology

The guidance for performing cost-benefit analysis for the gaseous radwaste system is similar to that used and described for the liquid radwaste system in Section 11.2. The gaseous radwaste treatment system augments annual costs were determined and the lowest annual cost considered a threshold value. The lowest-cost option for gaseous radwaste treatment system augments is the Steam Generator Flash Tank Vent to Main Condenser at $6,320 per year, which yields a threshold value of 6.32 person-rem total body or thyroid from gaseous effluents.

For AP1000 sites with population dose estimates less than 6.32 person-rem total body or thyroid dose from gaseous effluents, no further cost-benefit analysis is needed to demonstrate compliance with 10 CFR 50, Appendix I, Section II.D.

11.3.3.4.4 Not Used.

11.3.3.5 Maximum Release Concentrations

The annual releases of radioactive gases and iodine provided in Table 11.3-3 represent expected releases from the plant and reflect an expected level of fuel cladding defects. If the plant operates with the maximum defined fuel defect level, the releases would be substantially greater. The maximum defined fuel defect level is conservatively established at 1.0 percent fuel defects, corresponding to four times the Technical Specification limit on coolant activity which is based on 0.25 percent fuel defects, as described in Table 11.1-2. To demonstrate compliance with the effluent concentration limits of Reference 1, the releases from Table 11.3-3 have been adjusted to reflect operation with the maximum defined fuel defect level, and the resulting airborne radionuclide concentrations at the site boundary are compared in Table 11.3-4 with the Reference 1 limits for concentrations in unrestricted areas. As shown in Table 11.3-4, the overall fraction of the effluent concentration limit for operation with the maximum defined fuel defect level is 0.33, which is well below the allowable value of 1.0.

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11.3.3.6 Quality Assurance

The quality assurance program for design, fabrication, procurement, and installation of the gaseous radwaste system is in accordance with the overall quality assurance program described in Chapter 17.

Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation, and testing provisions of the gaseous radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.3.4 Inspection and Testing Requirements

11.3.4.1 Preoperational Testing

Preoperational tests are performed to verify the proper operation of the WGS. The operational tests include automatic closure of the discharge control/isolation valve, WGS-PL-V051, upon receipt of a simulated high radiation signal. The discharge line of the gaseous radwaste system includes a radiation monitor, WGS-RE017, which detects a high radiation condition and generates an alarm that automatically closes the discharge control/isolation valve. By imposing a simulated high radiation alarm signal, proper operation of the discharge control/isolation valve is confirmed by its closure.

11.3.4.2 Preoperational Inspection

The proper performance of the gaseous radwaste system depends upon delay of gaseous radionuclides by chemical adsorption on activated carbon. As the radionuclides are delayed, they decay and are no longer available for release to the environment. The rate of release and site boundary dose rates have been evaluated based upon the quantity of activated carbon in a delay bed being at least 80 cubic feet. An inspection of the gaseous radwaste system activated carbon delay beds, WGS-MV-02A and WGS-MV-02B, will confirm that the contained volume of each delay bed is at least 80 cubic feet.

11.3.5 Combined License Information

11.3.5.1 Cost Benefit Analysis of Population Doses

The analysis performed to determine offsite dose due to gaseous effluents is based upon the AP1000 generic site parameters included in Chapter 1 and Tables 11.3-1, 11.3-2, and 11.3-4. The Combined License applicant will provide a site specific cost-benefit analysis to demonstrate compliance with 10 CFR 50, Appendix I, regarding population doses due to gaseous effluents.

The Combined License applicant will also provide an analysis to demonstrate compliance with individual dose limits to members of the public in accordance with 10 CFR 50, Appendix I Sections II.B and II.C.

11.3.5.2 Identification of Adsorbent Media

The types of adsorbent media to be used in the gaseous radwaste system is addressed in APP-GW-GLR-008 (Reference 5), and the applicable changes are incorporated.

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11.3.6 References

1. "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," 10 CFR Part 20, Appendix B, Issued by 58 FR 67657, April 28, 1995.

2. "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion >As-Low-As-Is-Reasonably-Achievable= for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," 10 CFR Part 50, Appendix I.

3. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," NUREG-0017, Revision 1, March 1985.

4. "Minimization of Contamination," 10 CFR 20.1406.

5. APP-GW-GLR-008, "Request for Closure of COL Items in DCD Chapter 11, Identification for Adsorbent Media," Westinghouse Electric Company LLC.

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Table 11.3-1
Gaseous Radwaste System Parameters

Operating influent pressure (psig)	2[1]
Design influent flow rate (scfm)	2.28
Activated carbon bed design operating dew point (°F)	45
Minimum Activated carbon in delay beds (pounds combined total)	4495
Maximum Activated carbon in delay beds (pounds combined total)	6593

Notes:
1. During pressurizer venting operations, operating influent pressure is 15 psig

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Table 11.3-2 (Sheet 1 of 2)
Component Data (Nominal) — Gaseous Radwaste System

Mechanical Components		
Pumps		
Sample Pumps		
Number	2	
Type	Diaphragm	
Heat Exchangers		
Gas Cooler		
Number	1	
Type	Dual tube coil	
	Process Side	**Cooling Side**
Design pressure (psig)	150	150
Design temperature (°F)	200	200
Design flow	2.28 scfm	0.15 gpm
Temperature inlet (°F)	125	40
Temperature outlet (°F)	40.1	42
Material	Stainless steel	Stainless steel
Tanks		
Guard Bed		
Number	1	
Nominal volume (ft^3)	8	
Type	Vertical pipe	
Design pressure (psig)	150	
Design temperature (°F)	200	
Material	Stainless steel	
Delay Bed		
Number	2	
Nominal volume (ft^3)	80	
Type	Vertical serpentine	
Design pressure (psig)	150	
Design temperature (°F)	200	
Material	Carbon steel	
Moisture Separator		
Number	1	
Nominal volume (gal)	3	
Type	Vertical	
Design pressure (psig)	150	
Design temperature (°F)	200	
Material	Stainless steel	

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Table 11.3-2 (Sheet 2 of 2)
Component Data (Nominal) — Gaseous Radwaste System
Summary of Instrument Indication and Alarms

Instrumentation	Indicate (Note 4)	Alarm
Gas Cooler		
Gas inlet temperature	X	
Cooling water outlet temperature	X	
Gas inlet pressure	X	X – Hi
Gas outlet temperature	X	X – Hi
Carbon Delay Beds		
Gas inlet temperature	X	X – Hi
Gas outlet temperature 2 channels	X	X – Hi
Gas outlet flow	X	
Gas outlet radiation (Note 3)	X	X – Hi
Gas outlet pressure	X	X – Hi, Lo-2
Carbon Bed Vault		
Vault hydrogen (Note 2)	X	X – Hi
Vault temperature (Note 1)	X	X – Hi
Moisture Separator		
Water level	X	X – Hi-2
Sampling Subsystem		
Hydrogen concentration	X	X
Oxygen concentration 2 channels	X	X – Hi
Gas flow	X	X – Lo

Notes:
1. Vault temperature monitor common for guard bed and delay bed.
2. Vault hydrogen monitor common for guard bed and delay bed.
3. High outlet radiation alarm closes gas outlet isolation valve.
4. Monitoring of the gaseous radwaste system is performed through the data display and processing system. Control functions are performed by the plant control system. Appropriate alarms and displays are available in the control room. Local indication and control are available on portable displays which may be connected to the data display and processing system. See Chapter 7.

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Table 11.3-3 (Sheet 1 of 3)
Expected Annual Release of Airborne Radionuclides as Determined by the Revised GALE Code

Release Rates in Ci/year						
		Building/Area Ventilation			Condenser Air Removal System	
Nuclide	Waste Gas System	Containment Building	Auxiliary Building	Turbine Building		Total Release
Kr-85m	1.2E-02	3.8E-03	4.3E-01	2.3E-05	2.1E-01	6.5E-01
Kr-85	8.1E+01	3.0E-01	1.4E+00	7.9E-05	6.9E-01	8.4E+01
Kr-87	negl.	1.2E-03	4.6E-01	6.9E-06	6.0E-02	5.2E-01
Kr-88	1.8E-04	2.7E-03	4.8E-01	2.6E-05	2.3E-01	7.2E-01
Xe-131m	3.0E+01	8.3E-01	4.8E+00	2.5E-04	2.2E+00	3.7E+01
Xe-133m	9.6E-04	1.8E-01	2.0E+00	1.1E-04	9.4E-01	3.1E+00
Xe-133	6.6E+00	2.4E+00	1.7E+01	9.0E-04	7.9E+00	3.4E+01
Xe-135m	negl.	1.8E-03	3.5E+00	1.9E-04	1.6E+00	5.1E+00
Xe-135	negl.	8.3E-02	4.5E+00	7.9E-04	6.9E+00	1.2E+01
Xe-137	negl.	negl.	9.1E-01	4.9E-05	4.3E-01	1.3E+00
Xe-138	negl.	7.8E-04	1.6E+00	8.9E-05	7.8E-01	2.4E+00
					Total Noble Gas	1.8E+02

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Table 11.3-3 (Sheet 2 of 3)
Expected Annual Release of Airborne Radionuclides as Determined by the Revised GALE Code

Release Rates in Ci/year (GBq/year)						
		Building/Area Ventilation			Condenser Air Removal System	
Nuclide	Fuel Handling Area	Containment Building	Auxiliary Building	Turbine Building		Total Release
I-131	2.0E-04	15.1E-04	4.8E-03	6.5E-05	2.6E-05	5.6E-03
I-133	2.9E-04	2.0E-03	7.1E-03	2.0E-05	8.0E-05	9.5E-03
					Total Airborne Radioiodine	1.5E-02

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Table 11.3-3 (Sheet 3 of 3)
Expected Annual Release of Airborne Radionuclides as Determined by the Revised GALE Code

Release Rates in Ci/year (GBq/year)					
		Building/Area Ventilation			
Nuclide	Waste Gas System	Containment Building	Auxiliary Building	Fuel Handling Area	Total Release
Cr-51	negl.	negl.	3.2E-06	1.8E-06	6.1E-06
Mn-54	negl.	negl.	negl.	3.0E-06	4.3E-06
Co-57	negl.	negl.	negl.	negl.	negl.
Co-58	negl.	2.5E-06	1.9E-05	2.1E-04	2.3E-04
Co-60	negl.	negl.	5.1E-06	8.2E-05	8.7E-05
Fe-59	negl.	negl.	negl.	negl.	negl.
Sr-89	negl.	1.3E-06	7.5E-06	2.1E-05	3.0E-05
Sr-90	negl.	negl.	2.9E-06	8.0E-06	1.2E-05
Zr-95	negl.	negl.	1.0E-05	negl.	1.0E-05
Nb-95	negl.	negl.	negl.	2.4E-05	2.5E-05
Ru-103	negl.	negl.	negl.	negl.	negl.
Ru-106	negl.	negl.	negl.	negl.	negl.
Sb-125	negl.	negl.	negl.	negl.	negl.
Cs-134	negl.	negl.	5.4E-06	1.7E-05	2.3E-05
Cs-136	negl.	negl.	negl.	negl.	negl.
Cs-137	negl.	negl.	7.2E-06	2.7E-05	3.6E-05
Ba-140	negl.	negl.	4.0E-06	negl.	4.2E-06
Ce-141	negl.	negl.	negl.	negl.	negl.
				Total Particulates	4.7E-04

Tritium Release via Gaseous Pathway(2) (Ci/Yr) = 48
C-14 Released via Gaseous Pathway (Ci/Yr) = 7.3
Ar-41 Released via Gaseous Pathway (Ci/Yr) = 34

Notes:

(1) Values less than 1 microcurie are considered to be negligible.
(2) Tritium release based on Westinghouse TRICAL computer code.
(3) The fuel handling area is within the auxiliary building but is considered separately.

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Table 11.3-4 (Sheet 1 of 2)
Comparison of Calculated Offsite Airborne Concentrations with 10 CFR 20 Limits

Radionuclide	Effluent Concentration Limit[a] µCi/ml	Expected Site Boundary[b] Concentration µCi/ml	Fraction of Concentration Limit[b] (expected)	Maximum Site Boundary Concentration[c] µCi/ml	Fraction of Concentration Limit[c] (maximum)
Kr-85m	1.00E-07	5.15E-13	5.15E-06	2.05E-11	2.05E-04
Kr-85	7.00E-07	6.66E-11	9.51E-05	1.42E-10	2.03E-04
Kr-87	2.00E-08	4.12E-13	2.06E-05	8.88E-12	4.44E-04
Kr-88	9.00E-09	5.71E-13	6.34E-05	3.70E-11	4.11E-03
Xe-131m	2.00E-06	2.93E-11	1.47E-05	3.56E-11	1.78E-05
Xe-133m	6.00E-07	2.46E-12	4.10E-06	4.44E-11	7.40E-05
Xe-133	5.00E-07	2.70E-11	5.39E-05	8.15E-08	1.63E-01
Xe-135m	4.00E-08	4.04E-12	1.01E-04	4.14E-12	1.03E-04
Xe-135	7.00E-08	9.51E-12	1.36E-04	3.83E-10	5.47E-03
Xe-138	2.00E-08	1.90E-12	9.51E-05	5.98E-12	2.99E-04
I-131	2.00E-10	4.44E-15	2.22E-05	1.74E-12	8.71E-03
I-133	1.00E-09	7.53E-15	7.53E-06	3.86E-13	3.86E-04
H-3	1.00E-07	3.81E-11	3.81E-04	3.81E-11	3.81E-04
C-14	3.00E-09	1.30E-11	4.33E-03	1.30E-11	4.33E-03
Ar-41	1.00E-08	2.70E-11	2.70E-03	2.70E-11	2.70E-03
Cr-51	3.00E-08	4.84E-18	1.61E-10	4.84E-18	1.61E-10
Mn-54	1.00E-09	3.41E-18	3.41E-09	3.41E-18	3.41E-09
Co-57	9.00E-10	6.50E-20	7.22E-11	6.50E-20	7.22E-11
Co-58	1.00E-09	1.82E-16	1.82E-07	1.82E-16	1.82E-07
Co-60	5.00E-11	6.90E-17	1.38E-06	6.90E-17	1.38E-06
Fe-59	5.00E-10	6.26E-19	1.25E-09	6.26E-19	1.25E-09
Sr-89	2.00E-10	2.38E-17	1.19E-07	8.72E-16	4.36E-06
Sr-90	6.00E-12	9.51E-18	1.59E-06	1.86E-16	3.11E-05
Zr-95	4.00E-10	7.93E-18	1.98E-08	1.54E-17	3.84E-08
Nb-95	2.0E-09	2.0E-17	9.9E-09	5.28E-17	2.64E-08
Ru-103	9.00E-10	6.34E-19	7.05E-10	6.34E-19	7.05E-10

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Table 11.3-4 (Sheet 2 of 2)
Comparison of Calculated Offsite Airborne Concentrations with 10 CFR 20 Limits

Radionuclide	Effluent Concentration Limit[a] µCi/ml	Expected Site Boundary[b] Concentration µCi/ml	Fraction of Concentration Limit[b] (expected)	Maximum Site Boundary Concentration[c] µCi/ml	Fraction of Concentration Limit[c] (maximum)
Ru-106	2.0E-11	6.2E-19	3.1E-08	6.18E-19	3.09E-08
Sb-125	7.0E-10	4.8E-19	6.9E-10	4.84E-18	6.91E-09
Cs-134	2.0E-10	1.8E-17	9.1E-08	1.62E-12	8.12E-03
Cs-136	9.0E-10	6.7E-19	7.5E-10	3.64E-15	4.05E-06
Cs-137	2.0E-10	2.9E-17	1.4E-07	1.30E-12	6.49E-03
Ba-140	2.0E-09	3.3E-18	1.7E-09	3.33E-18	1.66E-09
Ce-141	8.0E-10	3.3E-19	4.2E-10	6.53E-19	8.16E-10
Totals			9.06E-03		2.05E-01

Notes:
(a) Effluent concentration limit is from Reference 1.
(b) Expected site boundary concentration based on annual releases predicted by the PWR-GALE code (Table 11.3-3) and an annual average X/Q of 2.0×10^{-5} seconds per cubic meter.
(c) Maximum site boundary concentration based on adjusting the releases predicted by the PWR-GALE code (Table 11.3-3) to reflect operation with maximum defined fuel defect level and an annual average X/Q of 2.0×10^{-5} seconds per cubic meter.

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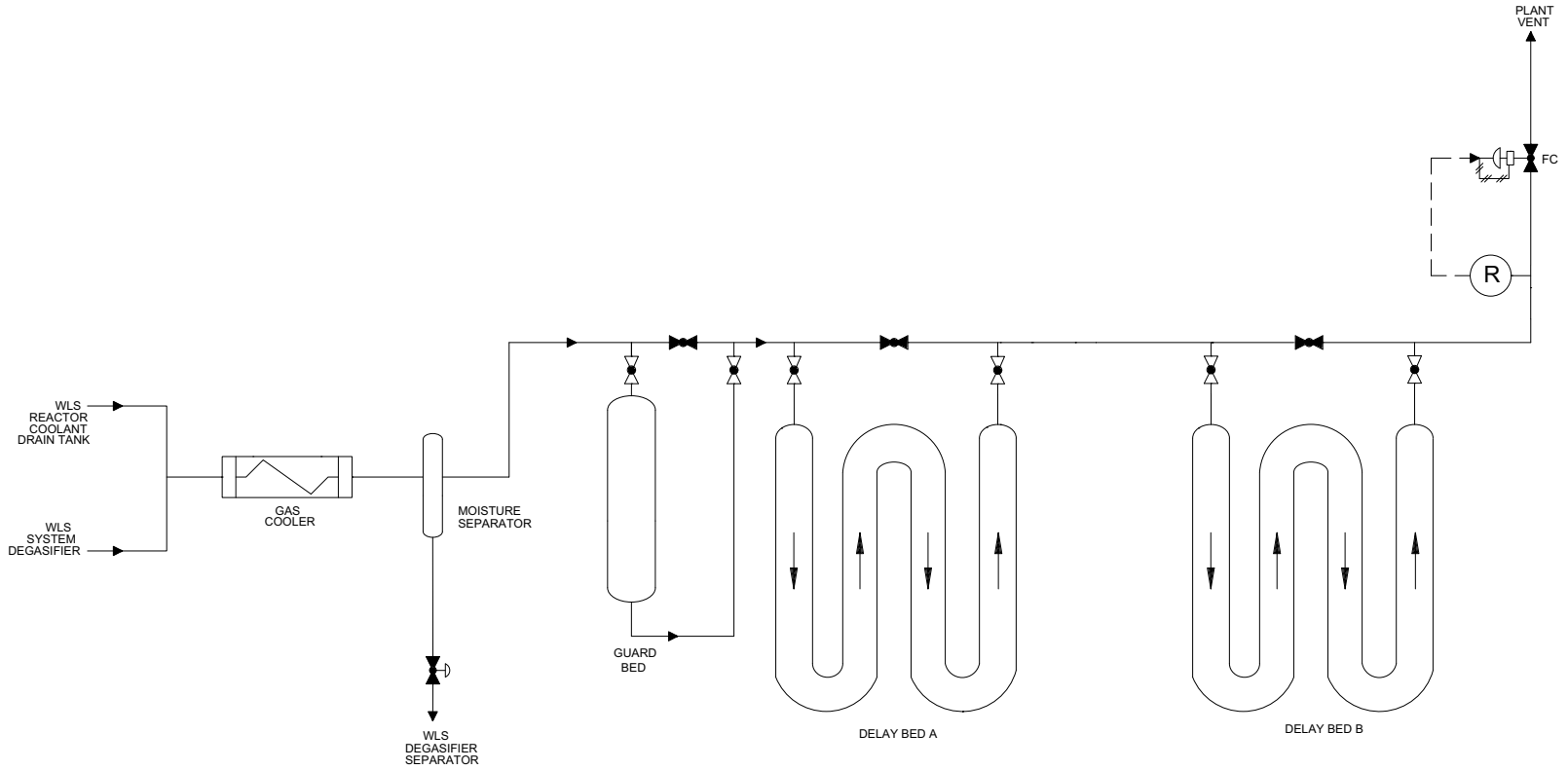

FIGURE REPRESENTS SYSTEM FUNCTIONAL ARRANGEMENT. DETAILS INTERNAL TO THE SYSTEM MAY DIFFER AS A RESULT OF IMPLEMENTATION FACTORS SUCH AS VENDOR-SPECIFIC COMPONENT REQUIREMENTS.

Figure 11.3-1
Gaseous Radwaste System
Simplified Sketch

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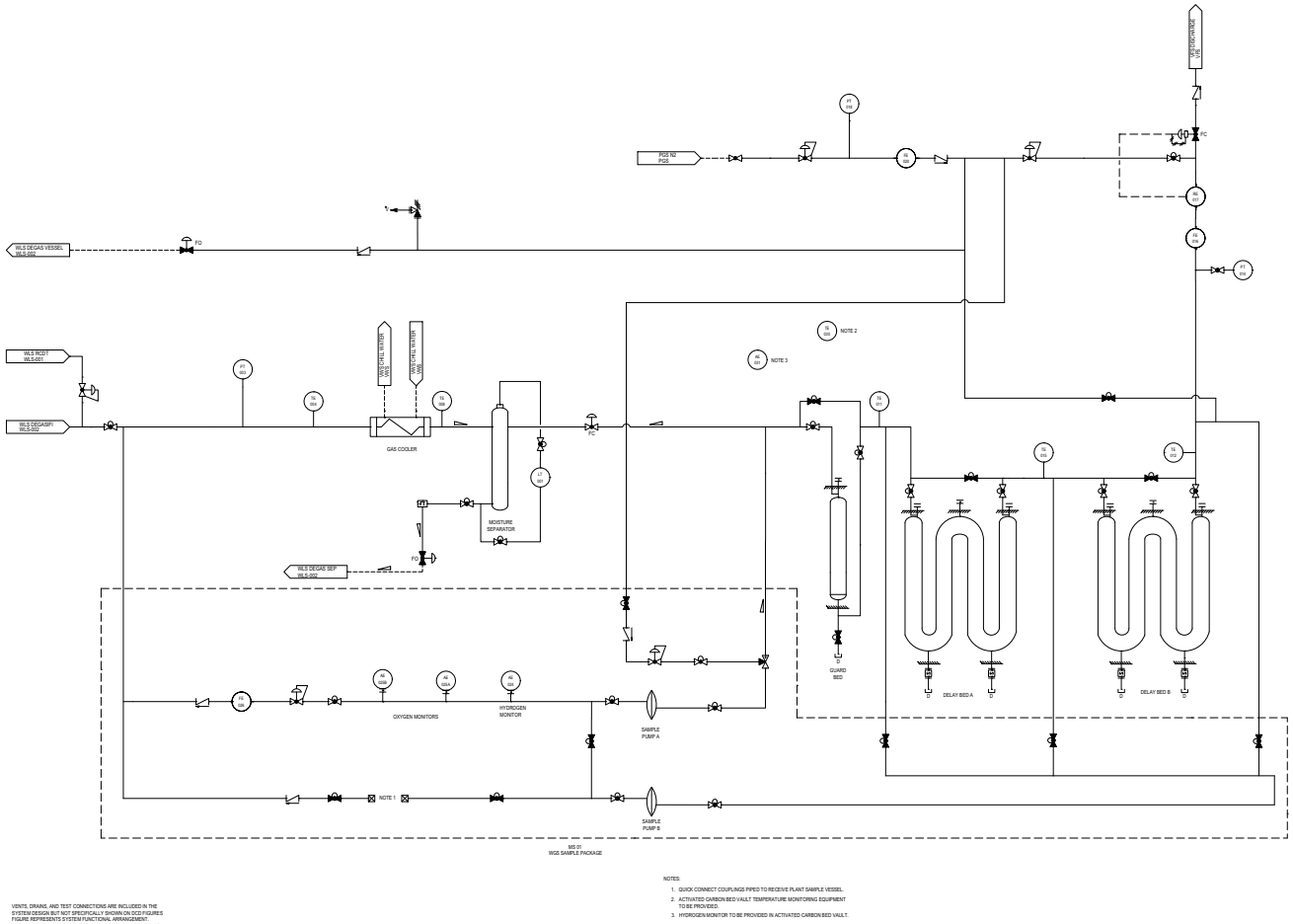

Figure 11.3-2
Simplified Gaseous Radwaste System
Piping and Instrumentation Diagram
(REF) WGS 001

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11.4 Solid Waste Management

The solid waste management system (WSS) is designed to collect and accumulate spent ion exchange resins and deep bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes are by mobile systems in the auxiliary building rail car bay and in the mobile systems facility part of the radwaste building. The packaged waste is stored in the auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

The use of mobile systems for the processing functions permits the use of the latest technology and avoids the equipment obsolescence problems experienced with installed radwaste processing equipment. The most appropriate and efficient systems may be used as they become available.

This system does not handle large, radioactive waste materials such as core components or radioactive process wastes from the plant's secondary cycle. However, the volumes and activities of the secondary cycle wastes are provided in this section.

11.4.1 Design Basis

11.4.1.1 Safety Design Basis

The solid waste management system performs no function related to the safe shutdown of the plant. The system's failure does not adversely affect any safety-related system or component; therefore, the system has no nuclear safety design basis.

There are no safety related systems located near heavy lifts associated with the solid waste management system. Therefore, a heavy loads analysis is not required.

11.4.1.2 Power Generation Design Basis

The solid waste management system provides temporary onsite storage for wastes prior to processing and for the packaged wastes. The system has a 60-year design objective and is designed for maximum reliability, minimum maintenance, and minimum radiation exposure to operating and maintenance personnel. The system has sufficient temporary waste accumulation capacity based on maximum waste generation rates so that maintenance, repair, or replacement of the solid waste management system equipment does not impact power generation.

11.4.1.3 Functional Design Basis

The solid waste management system is designed to meet the following objectives:

- Provide for the transfer and retention of spent radioactive ion exchange resins and deep bed filtration media from the various ion exchangers and filters in the liquid waste processing, chemical and volume control, and spent fuel cooling systems

- Provide the means to mix, sample, and transfer spent resins and filtration media to high integrity containers or liners for dewatering or solidification as required

- Provide the means to change out, transport, sample, and accumulate filter cartridges from liquid systems in a manner that minimizes radiation exposure of personnel and spread of contamination

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- Provide the means to accumulate spent filters from the plant heating, ventilation, and air-conditioning systems
- Provide the means to segregate solid wastes (trash) by radioactivity level and to temporarily store the wastes
- Provide the means to accumulate radioactive hazardous (mixed) wastes
- Provide the means to segregate clean wastes originating in the radiologically controlled area (RCA)
- Provide the means to store packaged wastes for at least 6 months in the event of delay or disruption of offsite shipping
- Provide the space and support services required for mobile processing systems that will reduce the volume of and package radioactive solid wastes for offsite shipment and disposal according to applicable regulations, including Department of Transportation regulation 49 CFR 173 (Reference 1) and NRC regulation 10 CFR 71 (Reference 2)
- Provide the means to return liquid radwaste to the liquid radwaste system (WLS) for subsequent processing and monitored discharge

The solid waste management system is designed according to NRC Regulatory Guide 1.143 to meet the requirements of General Design Criterion (GDC) 60 as discussed in Sections 1.9 and 3.1. The seismic design classifications of the radwaste building and system components are provided in Section 3.2.

Provisions are made in the auxiliary and radwaste buildings to use mobile radwaste processing systems for processing and packaging each waste stream including concentration and solidification of chemical wastes from the liquid waste management system, spent resin dewatering, spent filter cartridge encapsulation and dry active waste sorting and compaction.

The radioactivities of influents to the solid waste management system are based on estimated radionuclide concentrations and volumes. These estimates are based on operating plant experience, adjusted for the size and design differences of AP1000. The influent source terms are consistent with Section 11.1.

The solid waste management system airborne effluents are released through the monitored plant vent as described as part of the 10 CFR 50 (Reference 3), Appendix I, analysis presented in Subsection 11.3.3.

The solid waste management system collects and stores radioactive wastes within shielding to maintain radiation exposure to plant operation and maintenance personnel as low as is reasonably achievable (ALARA) according to General Design Criteria 60 as discussed in Section 3.1 and Regulatory Guide 8.8. Personnel exposures will be maintained well below the limits of 10 CFR 20 (Reference 4). Design features incorporated to maintain exposures ALARA include remote and semi-remote operations, automatic resin transport line flushing, and shielding of components, piping and containers holding radioactive materials. Access to the solid waste storage areas is controlled, to minimize inadvertent personnel exposure, by suitable barriers such as heavy storage cask covers and locked or key-card-operated doors or gates (see Section 12.1).

The solid waste management system conforms to the design criteria of NRC Branch Technical Position ETSB 11-3. Suitable fire protection systems are provided as described in Subsection 9.5.1.

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Waste disposal containers are to be selected from available designs that meet the requirements of the DOT and NRC. The solid waste management system does not require source-specific waste containers. Waste containers must meet the regulatory requirements for radioactive waste transportation in 49 CFR 173 and for radioactive waste disposal in 10 CFR 61 (Reference 5) as well as specific disposal facility requirements.

11.4.1.4 Compliance with 10 CFR 20.1406

In accordance with the requirements of 10 CFR 20.1406 (Reference 11), the solid radwaste system is designed to minimize, to the extent practicable, contamination of the facility and the environment, facilitate decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. This is done through appropriate selection of design technology for the system, plus incorporating the ability to update the system to use the best available technology throughout the life of the plant.

11.4.2 System Description

11.4.2.1 General Description

The solid waste management system includes the spent resin system. The flows of wastes through the solid waste management system are shown on Figure 11.4-1. The radioactivity of influents to the system are dependent on reactor coolant activities and the decontamination factors of the processes in the chemical and volume control system, spent fuel cooling system, and the liquid waste processing system.

The parameters used to calculate the estimated activity of the influents to the solid waste management system are listed in Table 11.4-1. The estimated expected isotopic curie content of the primary spent resin and filter cartridge wastes to be processed on an annual basis is listed on Table 11.4-2. Table 11.4-3 provides the same information for the estimated maximum annual activities. The AP1000 has sufficient radwaste storage capacity to accommodate the maximum generation rate.

The radioactivity of the dry active waste is expected to normally range from 0.1 curies per year to 8 curies per year with a maximum of about 16 curies per year. This waste includes spent HVAC filters, compressible trash, non-compressible components, mixed wastes and solidified chemical wastes. These activities are produced by relatively long lived radionuclides (such as Cr-51, Fe-55, Co-58, Co-60, Nb-95, Cs-134 and Cs-137), and therefore, radioactivity decay during processing and storage is minimal. These activities thus apply to the waste as generated and to the waste as shipped.

The estimated expected and maximum annual quantities of waste influents by source and form are listed in Table 11.4-1 with disposal volumes. The annual radwaste influent rates are derived by multiplying the average influent rate (e.g. volume per month, volume per refueling cycle) by one year of time. The annual disposal rate is determined by applying the radwaste packaging efficiency to the annual influent rate. The influent volumes are conservatively based on an 18-month refueling cycle. Annual quantities based on a 24-month refueling cycle are less than those for an 18-month cycle. The estimated expected isotopic curie content of the primary spent resin and filter cartridge wastes to be shipped offsite are presented in Table 11.4-4 based on 90 days of decay before shipment. The same information is presented in Table 11.4-5 for the estimated maximum activities based on 30 days of decay before shipment.

Section 11.1 provides the bases for determination of liquid source terms used to calculate several of the solid waste management system influent source terms. The influent data presented in Tables 11.4-2 and 11.4-3 are conservatively based on Section 11.1 design basis (Technical Specification) values.

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All radwaste which is packaged and stored by AP1000 will be shipped for disposal. The AP1000 has no provisions for permanent storage of radwaste. Radwaste is stored ready for shipment. Shipped volumes of radwaste for disposal are estimated in Table 11.4-1 from the estimated expected or maximum influent volumes by making adjustments for volume reduction processing by mobile systems and the expected container filling efficiencies. For drum compaction, the overall volume reduction factor, including packaging efficiency, is 3.6. For box compaction, the overall volume reduction factor is 5.4. These adjustments result in a packaged internal waste volume for each waste source, and the number of containers required to hold this volume is based on the container's internal volume. The disposal volume is based on the number of containers and the external (disposal) volume of the containers.

The expected disposal volumes of wet and dry wastes are approximately 547 and 1417 cubic feet per year, respectively as shown in Table 11.4-1. The wet wastes shipping volumes include 510 cubic feet per year of spent ion exchange resins and deep bed filter activated carbon, 20 cubic feet of volume reduced liquid chemical wastes and 17 cubic feet of mixed liquid wastes. The spent resins and activated carbon are initially stored in the spent resin storage tanks located in the rail car bay of the auxiliary building. When a sufficient quantity has accumulated, the resin is sluiced into two 158 cubic feet high-integrity containers in anticipation of transport for offsite disposal. Liquid chemical wastes are reduced in volume and packaged into three 55-gallon drums per year (about 20 cubic feet) and are stored in the waste accumulation room of the radwaste building. The mixed liquid wastes fill less than three drums per year (about 17 cubic feet per year) and are stored on containment pallets in the waste accumulation room of the radwaste building until shipped offsite for processing.

The two spent resin storage tanks (250 cubic feet usable, each) and one high integrity container in the spent resin waste container fill station at the west end of the rail car bay of the auxiliary building provide more than a year of spent resin storage at the expected rate, and several months of storage at the maximum generation rate. The expected radwaste generation rate is based upon the following:

- All ion exchange resin beds are disposed and replaced every refueling cycle.
- The WGS activated carbon guard bed is replaced every refueling cycle.
- The WGS delay beds are replaced every ten years.
- All wet filters are replaced every refueling cycle.
- Rates of compactible and non-compactible radwaste, chemical waste, and mixed wastes are estimated using historical operating plant data.

The maximum radwaste generation rate is based upon the following:

- The ion exchange resin beds are disposed based upon operation with 0.25% fuel defects.
- The WGS activated carbon guard bed is replaced twice every refueling cycle.
- The WGS delay beds are replaced every five years.
- All wet filters are replaced based upon operation with 0.25% fuel defects.
- The expected rates of compactible and non-compactible radwaste, chemical waste, and mixed wastes are increased by about 50%.

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- Primary to secondary system leakage contaminates the condensate polishing system and blowdown system resins and membranes which are replaced.

The dry solid radwaste includes 1383 cubic feet per year of compactible and non-compactible waste packed into about 14 boxes (90 cubic feet each) and ten drums per year. Drums are used for higher activity compactible and non-compactible wastes. Compactible waste includes HVAC exhaust filter, ground sheets, boot covers, hair nets, etc. Non-compactible waste includes about 60 cubic feet per year of dry activated carbon and other solids such as broken tools and wood. Solid mixed wastes will occupy 7.5 cubic feet per year (one drum). The low activity spent filter cartridges may be compacted to fill about 0.40 drums per year (3 ft^3/year) and are stored in the waste accumulation room. Compaction is performed by mobile equipment or is performed offsite. High activity filter cartridges fill three drums per year (22.5 cubic feet per year) and are stored in the filter storage area in the auxiliary building.

The total volume of packaged radwaste to be stored in the radwaste building waste accumulation room is 1417 cubic feet per year at the expected rate and 2544 cubic feet per year at the maximum rate. The compactible and non-compactible dry wastes, packaged in drums or steel boxes, are stored with the mixed liquid and mixed solid, volume reduced liquid chemical wastes, and the lower activity filter cartridges. The quantities of packaged liquid radwaste stored in the waste accumulation room of the radwaste building consist of 20 cubic feet of chemical waste and 17 cubic feet of mixed liquid waste. The available minimum useful storage volume for packaged waste in the waste accumulation room is 3900 cubic feet (10 feet deep, 30 feet long, and 13 feet high), which accommodates more than one full offsite waste shipment using a tractor-trailer truck. The waste accumulation room provides storage for more than two years at the expected rate of generation and more than a year at the maximum rate of generation. One four-drum containment pallet provides more than 8 months of storage capacity for the liquid mixed wastes and the volume reduced liquid chemical wastes at the expected rate of generation and more than 4 months at the maximum rate.

A conservative estimate of solid wet waste includes blowdown material based on continuous operation of the steam generator blowdown purification system, with leakage from the primary to secondary system. The volume of radioactively contaminated material from this source is estimated to be 540 cubic feet per year. Provisions for processing and disposal of radioactive steam generator blowdown resins and membranes are described in Subsection 10.4.8. Note that, although included here for conservatism, this volume of contaminated resin will be removed from the plant within the contaminated electrodeionization unit and not stored as wet waste.

The condensate polishing system includes mixed bed ion exchanger vessels for purification of the condensate as described in Subsection 10.4.6. Should the resins become radioactive, the resins are transferred from the condensate polishing vessel directly to a temporary processing unit or to the temporary processing unit via the spent resin tank. The processing unit, located outside of the turbine building, dewaters and processes the resins as required for offsite disposal. Radioactive condensate polishing resin will have very low activity. It will be disposed in containers as permitted by DOT regulations. After packaging, the resins may be stored in the radwaste building. Based on a typical condensate polishing system operation of 30 days per refueling cycle with leakage from the primary system to the secondary system, the volume of radioactively contaminated resin is estimated to be 331 cubic feet per year (two 248 cubic foot beds per refueling cycle). Normal disposal of nonradioactive condensate polishing system resins is described in Subsection 10.4.6.

The parameters used to calculate the activities of the steam generator blowdown solid waste and condensate polishing resins are given in Table 11.4-1. Based on the above volumes, the disposal volume is estimated to be 1095 cubic feet per year. The expected and maximum activities of the resins as generated are given in Tables 11.4-6 and 11.4-7, respectively. The expected and maximum activities of resins as shipped, based on 90 days decay prior to shipment, are given in Tables 11.4-8 and 11.4-9, respectively.

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11.4.2.2 Component Description

The seismic design classification and safety classification for the solid waste management system components are listed in Section 3.2. The components listed are located in the seismic Category I Nuclear Island. Table 11.4-10 lists the solid waste management system equipment design parameters. The following subsections provide a functional description of the major system components.

11.4.2.2.1 Spent Resin Tanks

The spent resin tanks provide holdup capacity for spent resin and filter bed media decay before processing. High- and low-activity resins may be mixed to limit the radioactivity concentration in the waste containers to 10 Ci/ft^3 in accordance with the USNRC Technical Position on Waste Form (Reference 6).

Resin mixing capability is provided by mixing eductors in each tank, and resin dewatering, air sparging and complete draining capabilities are also provided. The ultrasonic level sensors and dewatering screens are arranged for remote removal. The vent and overflow connections have screens to prevent the inadvertent discharge of spent resin, and they are routed to the radioactive waste drain system (WRS).

11.4.2.2.2 Resin Mixing Pump

The resin mixing pump provides the motive force to fluidize and mix the resins in the spent resin tanks, to transfer water between spent resin tanks, to discharge excess water from the spent resin tanks to the liquid waste processing system, and to flush the resin transfer lines.

11.4.2.2.3 Resin Fines Filter

The resin fines filter minimizes the spread of high-activity resin fines and dislodged crud particles by filtering the water used for line flushing or discharged from the spent resin tanks to the liquid waste processing system.

11.4.2.2.4 Resin Transfer Pump

The resin transfer pump provides the motive force for recirculation of spent resins via either one of the spent resin tanks for mixing and sampling, for transferring spent resin between tanks, and for blending high- and low-activity resins to meet the specific activity limit for disposal. The resin transfer pump is also used to transfer spent resins to a waste container in the fill station or in its shipping cask located in the auxiliary building rail car bay.

11.4.2.2.5 Resin Sampling Device

The resin sampling device collects a representative sample of the spent resin either during spent resin recirculation or during spent resin waste container filling operations. A portable shielded cask is provided for sample jar transfer.

11.4.2.2.6 Filter Transfer Cask

The filter transfer cask permits remote changing of filter cartridges, dripless transport to the storage area in the auxiliary building, transfer of the filter cartridges into and out of the filter storage, and loading of the filter cartridges into disposal containers.

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11.4.2.3 System Operation

11.4.2.3.1 Spent Resin Handling Operations

Demineralized water is used to transfer spent resins from the various ion exchangers to the spent resin tanks. A demineralized water transfer pump provides the pressurized water flow to transfer the spent resins as described in Subsection 9.2.4. Before the transfer operation, it is verified that the selected spent resin tank is aligned as a receiver and has the capacity to accept the bed. It is also verified that the resin mixing pump is aligned to discharge excess transfer water through the resin fines filter to the liquid waste processing system.

During the transfer operation the tank level is monitored and the resin mixing pump is operated, if required, to limit tank water level. The operator stops the transfer when the CCTV camera viewing the sight flow glass indicates on a control panel monitor that the sluice water is clear and the transfer line is, therefore, flushed of resins.

After the bed transfer, the tank solids level can be checked by operating the resin mixing pump to lower the water level below the solids level. The solids level can be determined by the ultrasonic surface detector.

Between bed transfer operations the water level in the spent resin tanks is maintained above the solids level. Demineralized water is supplied for water level adjustment as well as a backup water source for flushing resin handling lines after resin recirculation and waste disposal container filling operations.

The solids bed can be agitated and mixed at any time by using compressed air or by operating the resin mixing pump in the resin mixing mode. In the resin mixing mode, water is drawn from the spent resin tank via resin retention screens. The water is returned via tank mixing eductors that generate a resin slurry recirculation within the tank equivalent to about four times the flow rate generated by the resin mixing pump. The solids bed is locally fluidized during this operation.

The resin mixing mode is established to fluidize and mix the solids bed in the spent resin tank before waste disposal container filling. The resin transfer pump is then started in the recirculation mode. A resin slurry is drawn from the spent resin tank and returned to the same tank. A representative resin sample may be obtained during recirculation or container filling modes by operating the sampling device.

The portable system's container fill valve is opened to initiate the filling operation. The resin dewatering pump of the portable dewatering system is started to dewater the resin as it accumulates in the container. The resin dewatering pump discharges the water to the recirculation line. The water flows back to the spent resin tank, thereby preserving the water inventory in the system and retaining any resin fines or dislodged crud within the system.

The resin mixing pump can be stopped at any time during the filling operation. When the solids level nears the top of the container, as detected by level sensors and observed by a television camera, the fill valve is closed and cycled to top off the container. Excessive water or solids level automatically closes the fill valve.

When the filling operation is complete, the line flushing sequence controller is manually initiated to automatically operate the pumps and valves to flush the resin transfer lines back to the spent resin tank. The container fill valve is opened for a short time period to flush the remaining resin to the waste container. The resin mixing pump supplies filtered flush water from the spent resin tank. The portable dewatering system's dewatering pump is operated periodically until no further dewatering flow is detected by the pump discharge pressure indicator and/or audible indications from the pump.

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11.4.2.3.2 Spent Filter Processing Operations

A filter transfer cask is used to change the higher-activity filters of the chemical and volume control system and spent fuel cooling system. The filter vessel is drained, and the filter cover is opened remotely. The shield plug of the port over the filter is removed and the transfer cask, without its bottom shield cover, is lifted and positioned on the port directly over the cartridge in the filter vessel.

A grapple inside the transfer cask is remotely lowered and connected to the filter cartridge. The cartridge is lifted into the transfer cask, and the cask is transferred over plastic sheeting to the bottom shield cover. The dose rate of the cartridge is measured with a long probe, and the cask is lowered onto and connected to the bottom shield cover. The transfer cask is then moved to the auxiliary building rail car bay.

If recent applicable sample analysis results are available, the filter cartridge can be loaded directly into a disposal container as described in the following paragraph. If analysis is required, a sample of the filter media may be obtained through a port in the transfer cask or by other sampling methods. The filter cartridge is placed in the waste disposal container area until sample analysis results are available. The transfer cask bottom cover is disconnected, the transfer cask is lifted by the crane and transferred to a position over a waste container in the waste disposal container area, and the spent filter cartridge is lowered into the container. After moving the transfer cask away, the crane is used to replace the lid over the waste disposal container area. Any water draining from the filter during storage drains to a floor drain for subsequent transfer to the liquid radwaste system.

When sample analysis is complete and packaging requirements are established, the transfer cask is used to retrieve the spent cartridges from storage and deposit them into a waste container via a port in the top of a portable processing and storage cask. Plastic coverings are removed and the container is capped, smear-surveyed, and decontaminated as required, using reach rod tools through a cask port. The dose rate survey is also made through a cask port. Transfer of the filled waste container to the shipping cask, including cask cover handling, is then performed using the rail car bay crane under remote control.

Filters with dose rates less than 15 R/hr on contact may be changed from outside of filter vessel shielding by using reach rod tools. The filter vessel is drained, and the cover is removed. Then the spent filter cartridge is grappled and lifted out and into a filter transfer cask.

At the radwaste building, low and moderate activity filter cartridges are deposited into disposal or storage drums. The drums are stored within portable shield casks in the shielded accumulation room, which is serviced by the mobile systems facility crane. Depending on dose rates and analysis results, stabilization may or may not be required. Cartridges not requiring stabilization are loaded into standard, 55 gallon shipping drums with absorbent and may be compacted using a mobile system. When stabilization is required, the cartridges may be loaded into either high integrity containers or standard drums. If standard drums are used, mobile equipment is used to encapsulate the contents of the drums.

The drum covers are manually installed, and the drums are smear surveyed, decontaminated by wiping, if required, weighed, stacked on pallets, and placed in the waste accumulation room.

When a truck-load quantity of waste containers accumulates, shipment to a low-level waste disposal facility is initiated by loading pallets of drums and other low-level waste containers into a closed van using the scissor lift or onto a flat-bed trailer using the crane. If the activity level is too high for unshielded shipment, the drums are loaded onto a cask pallet and into a shielded shipping cask using the mobile systems facility crane.

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Radioactive filters from ventilation exhaust filtration units are bagged and transported to the radwaste building, where they are temporarily stored. The filters are compacted along with other dry active wastes by a mobile system as described in the following subsection.

11.4.2.3.3 Dry Waste Processing Operations

Dry wastes are segregated by measuring the contact dose rate of the wastes to determine the appropriate processing method. The contact dose rates for initial waste segregation are as follows:

Low activity	<5 mR/hr
Moderate activity	5 mR/hr to 100 mR/hr
High activity	>100 mR/hr

These activity levels may be adjusted by the operator to minimize exposures while maximizing processing efficiency.

Wastes from surface contamination areas in the radiologically controlled area are placed in bags or containers and tagged at the point of origin with information on radiation levels, waste type, and destination. The bags or containers are transported to the radwaste building, where they are placed into low-, moderate-, or high-activity storage, segregated by portable shielding as appropriate.

The high-activity wastes (greater than 100 mR/hr) are normally expected to be compacted in drums using a mobile compactor system in the same manner as lower-activity filter cartridges.

Moderate-activity wastes (5 mR/hr to 100 mR/hr) are expected to be sorted in a mobile system to remove reusable items such as protective clothing articles and tools, hazardous wastes, and larger noncompressible items. The remaining wastes are normally compacted by mobile equipment. The packaged wastes may be loaded directly onto a truck for shipment or may be stored in the waste accumulation room until a truck load quantity accumulates.

Low-activity, dry active waste (less than 5 mR/hr) generally contains a large amount of nonradioactive material. It is expected that these wastes normally will be processed through a mobile radiation monitoring and sorting system to remove non-radioactive items for reuse or local disposal. A radiation survey allows identification and removal of potentially clean items for the clean waste verification. The remaining radioactive wastes are normally compacted or packaged for disposal as appropriate.

Materials that enter the radiologically controlled area are verified as nonradioactive before being released for reuse or disposal. Tools and equipment belonging to personnel and contractors are surveyed at the radiologically controlled area exit in the annex building. If these items cannot be released or decontaminated, they become plant inventory or dry active waste and are handled as described previously.

Other wastes generated in the radiologically controlled area but outside of surface contamination areas are collected in bags or containers and are delivered to the temporary storage location in the radwaste building. These wastes normally are processed through a mobile radiation monitoring system to verify that they are nonradioactive and suitable for disposal in a local waste landfill.

11.4.2.3.4 Mixed Waste Processing Operations

Mixed wastes from the radiologically controlled area are collected in suitable containers and brought to the radwaste building, where separate containment pallets and accumulation drums are provided

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for solid and liquid mixed wastes. Mixed wastes are normally sent to an offsite facility having mixed-waste processing and disposal capabilities.

11.4.2.4 Waste Processing and Disposal Alternatives

11.4.2.4.1 Portable and Mobile Radwaste Systems Capabilities

Portable or mobile processing and packaging systems can be located in the auxiliary building rail car bay or the radwaste building mobile systems facility. Chemical wastes may be processed in the radwaste building by a mobile concentration and/or solidification system when a batch accumulates in the chemical waste tank. Mobile systems may also be used to encapsulate high-activity filters, to sort, decontaminate and compact dry active wastes, and to verify nonradioactive wastes.

The spent resin system includes connections in the fill station and rail car bay to allow spent resins to be delivered to a disposal container in either location for dewatering using portable equipment.

Branch Technical Position ETSB 11-3 provides guidance for portable solid waste systems in Section IV. Compliance with the four guidance items is achieved as follows:

IV.I The spent resin tanks are the only tanks that contain a significant volume of wet wastes, and these tanks are permanently installed. Concentrates that may be produced by mobile evaporation systems will be produced and stored by the mobile systems only in small batches prior to being solidified by the mobile systems. As described in Subsection 1.2.7, the radwaste building is designed to retain spillage from mobile or portable systems.

IV.2 Permanently installed piping for transport of radioactive wastes to mobile or portable systems is routed close to the mobile or portable systems thereby minimizing the use of flexible interfacing hose. The hydrostatic test requirements of Regulatory Guide 1.143 will be applied to the flexible interfacing hose.

IV.3 Portable or mobile systems will be located in either the rail car bay of the auxiliary building or in the mobile systems facility in the radwaste building. The spent resin waste container fill station or the shipping cask in the auxiliary building collects spillage of spent resin during waste container filling operations. The radwaste and auxiliary buildings contain and drain spillage to the liquid radwaste system via the radioactive waste drain system as described in Subsection 1.2.7 and Section 11.2. Portable or mobile systems will, when required, have their own HEPA filtered exhaust ventilation system. HEPA filtered exhaust is required when airborne radioactivity would exceed 10 CFR 20 derived air concentration limits for radiation workers. The mobile systems facility has connections on the exhaust ventilation ducts for connecting exhaust duct from mobile or portable processing systems to the building's exhaust ventilation system.

IV.4 Although the seismic criteria of Regulatory Guide 1.143 are not applicable to structures housing mobile or portable solid radwaste systems, the portable equipment used for spent resin container filling and dewatering and high-activity filter cartridge packaging will be housed within the Seismic Category I auxiliary building. The radwaste building, which provides shelter for mobile or portable radwaste systems, is non-seismic in accordance with Branch Technical Position ETSB 11-3.

11.4.2.4.2 Central Radwaste Processing Facility

As an alternative to the mobile or portable processes for lower-activity wastes, the wastes may be sent to a licensed central radwaste processing facility for processing and disposal. This option requires minimal onsite processing to remove radioactive materials from the waste streams. The

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wastes are loaded into a cargo container. The mobile systems facility includes a designated laydown area, and the mobile systems facility crane may be used to handle a cargo container.

11.4.2.4.3 Not Used.

11.4.2.5 Facilities

11.4.2.5.1 Auxiliary Building

Resin and filtration media transfer lines from the various ion exchangers are routed to the spent resin tanks on elevation 100′ - 0″ in the southwest corner of the auxiliary building. The spent resin system pumps, valves, and piping are located in shielded rooms near the spent resin tanks.

Liquid radwaste system transfer lines to and from the radwaste building are routed to the south wall of the auxiliary building where they penetrate and enter into a shielded pipe pit in the base mat of the radwaste building.

Accessways in the auxiliary building are used to move the filter transfer casks. This includes filter transfer cask handling from the containment, where the chemical and volume control filters are located, to the auxiliary building rail car bay, where the filter cartridges are stored and subsequently packaged using mobile equipment. These accessways are also used to move dry active waste from various collection locations to the radwaste building. Enclosed access is provided between the auxiliary building and the radwaste building on elevation 100′-0″ (grade level).

11.4.2.5.2 Radwaste Building

The radwaste building, described in Section 1.2, houses the mobile systems facility and the waste accumulation room. These rooms are serviced by the mobile systems facility crane.

In the mobile systems facility, three truck bays provide for mobile or portable processing systems and for waste disposal container shipping and receiving. A shielded pipe trench to each of the truck bays is used to route liquid radwaste supply and return lines from the connections in the shielded pipe pit at the auxiliary building wall. Separate areas are reserved for empty (new) waste disposal container storage, container laydown, and forklift charging. An area is available near the door to the annex building for protective clothing dropoff and frisking.

The waste accumulation room is divided as needed, using partitions and portable shielding to adjust the storage areas for different waste categories as needed to complement the radioactivity levels and volumes of generated wastes. The accumulation room also contains three 1000 cubic feet (10 feet x 10 feet x 10 feet) bunkers with removable steel shield bunker roof plates and removable steel shield bunker door plates. The design functions of this removable shielding are described in Subsection 12.3.2.2.5. The accumulation room has lockable doors to minimize unauthorized entry and inadvertent exposure.

The bunkers have been evaluated for hydrogen gas generation resulting from the temporary storage of resin, specifically condensate polishing system (CPS) resin and steam generator blowdown system electrodeionization (EDI) unit resin. Hydrogen was selected as the most credible source of explosive gas generation. Based on table 3.1 of NUREG/CR 6673 other flammable gases formed by radiolysis in resin are insignificant in comparison to hydrogen. Methane was also considered, but was found not to be a credible source because the AP1000 CPS and EDI systems utilize resin beads which do not have the cellulose component that could support the growth of bacteria and methane production. Generation of flammable gases by processes such as biodegradation, decomposition and waste material interaction due to chemistry, was also determined to not be credible based on the

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waste forms intended for storage in the AP1000 radwaste bunkers; i.e., secondary side spent resin, and miscellaneous contaminated / activated components or tools.

The evaluation assumes two storage scenarios of the bounding amounts of CPS and EDI resins expected to be generated over one cycle of operation and then stored in a single, unventilated bunker for one year. The scenarios result in a maximum hydrogen concentration of less than 4 volume percent hydrogen in air, and conclude that there is no risk of radiolytic hydrogen gas generation that could result in the creation of a hazard within the stated assumptions. Although the evaluation is focused on storing a bounding amount of resin into only one bunker, all three bunkers are available for use.

The existing administrative controls in Subsections 11.2.1.2.5.2 and 13.5.2.2.5 limit the total cumulative radioactive inventory of unpackaged wastes allowed in the radwaste building and bunkers, and the existing radiation zoning and access requirements in Subsection 12.3.1.2 further restrict the amount and activities of resins that are allowed to be stored in the bunker. Therefore, a new evaluation for hydrogen gas generation resulting from the temporary storage of resin would only be needed in the unlikely event that storage of resins with total volume or activity higher than that of the evaluated resins, or for a longer storage period, is desired. The new evaluation would be needed to confirm the risk associated with potential hydrogen gas generation by demonstrating that the hydrogen concentration in the bunker air space will not exceed 4 volume percent hydrogen in air.

The heating and ventilating system for the radwaste building is described in Subsection 9.4.8.

11.4.3 System Safety Evaluation

The solid waste management system has no safety-related function and therefore requires no nuclear safety evaluation.

11.4.4 Tests and Inspections

Preoperational tests are conducted as described in Subsection 14.2.9. Tests are performed to demonstrate the capability to transfer ion exchange resins and deep bed filtration media from the ion exchangers and filters to the spent resin tanks or directly to a waste disposal container. Preoperational tests of the solid waste management system components are performed to prepare the system for operation.

After plant operations begin, the operability and functional performance of the solid waste management system is periodically evaluated according to Regulatory Guide 1.143 by monitoring for abnormal or deteriorating performance during routine operations. Instruments and setpoints are also calibrated on a scheduled basis. The preventive maintenance program includes periodic inspection and maintenance of active components.

11.4.5 Quality Assurance

The quality assurance program for design, installation, procurement, and fabrication issues of the solid waste management system is in accordance with the overall quality assurance program described in Chapter 17.

Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation and testing provisions of the solid radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

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11.4.6 Combined License Information for Solid Waste Management System Process Control Program

A Process Control Program (PCP) is developed and implemented in accordance with the recommendations and guidance of NEI 07-10A (Reference 201). The PCP describes the administrative and operational controls used for the solidification of liquid or wet solid waste and the dewatering of wet solid waste. Its purpose is to provide the necessary controls such that the final disposal waste product meets applicable federal regulations (10 CFR Parts 20, 50, 61, 71, and 49 CFR Part 173), state regulations, and disposal site waste form requirements for burial at a low level waste (LLW) disposal site that is licensed in accordance with 10 CFR Part 61.

Waste processing (solidification or dewatering) equipment and services may be provided by the plant or by third-party vendors. Each process used meets the applicable requirements of the PCP.

No additional onsite radwaste storage is required beyond that described in the DCD.

Table 13.4-201 provides milestones for PCP implementation.

11.4.6.1 Procedures

Operating procedures specify the processes to be followed to ship waste that complies with the waste acceptance criteria (WAC) of the disposal site, 10 CFR 61.55 and 61.56, and the requirements of third party waste processors.

Each waste stream process is controlled by procedures that specify the process for packaging, shipment, material properties, destination (for disposal or further processing), testing to verify compliance, the process to address non-conforming materials, and required documentation.

Where materials are to be disposed of as non-radioactive waste (as described in Subsection 11.4.2.3.3), final measurements of each package are performed to verify there has not been an accumulation of licensed material resulting from a buildup of multiple, non-detectable quantities. These measurements are obtained using sensitive scintillation detectors, or instruments of equal sensitivity, in a low-background area.

Procedures document maintenance activities, spill abatement, upset condition recovery, and training.

Procedures document the periodic review and revision, as necessary, of the PCP based on changes to the disposal site, WAC regulations, and third party PCPs.

11.4.6.2 Third Party Vendors

Third party equipment suppliers and/or waste processors are required to supply approved PCPs. Third party vendor PCPs describe compliance with Regulatory Guide 1.143 (Reference 7), Generic Letter 80-09 (Reference 8), and Generic Letter 81-39 (Reference 9). Third party vendor PCPs are referenced appropriately in the plant PCP before commencement of waste processing.

11.4.7 References

1. "Shippers-General Requirements for Shipments and Packagings," 49 CFR 173.

2. "Packaging and Transportation of Radioactive Material," 10 CFR 71.

3. "Domestic Licensing of Production and Utilization Facilities," 10 CFR 50.

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4. "Standards for Protection Against Radiation," 10 CFR 20.

5. "Licensing Requirements for Land Disposal of Radioactive Waste," 10 CFR 61.

6. "USNRC Technical Position on Waste Form," Rev. 1, January 1991.

7. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

8. USNRC Generic Letter GL-80-009, "Low Level Radioactive Waste Disposal," dated January 29, 1980.

9. USNRC Generic Letter GL-81-039, "NRC Volume Reduction Policy (Generic Letter No. 81-39)," dated November 30, 1981.

10. Not used.

11. USNRC, "Minimization of Contamination," 10 CFR 20.1406.

201. NEI 07-10A, “Generic FSAR Template Guidance for Process Control Program (PCP),” Revision 0, March 2009 (ML091460627).

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Table 11.4-1
Estimated Solid Radwaste Volumes

Source	Expected Generation (ft^3/yr)	Expected Shipped Solid (ft^3/yr)	Maximum Generation (ft^3/yr)	Maximum Shipped Solid (ft^3/yr)
Wet Wastes				
Primary Resins (includes spent resins and wet activated carbon)	400(2)	510	1700(4)	2160
Chemical	350	20	700	40
Mixed Liquid	15	17	30	34
Condensate Polishing Resin(1)	0	0	331(5)	415
Steam Generator Blowdown(1)(6) Material (Resin and Membrane)	0	0	540(5)	680
Wet Waste Subtotals	**765**	**547**	**3301**	**3329**
Dry Wastes				
Compactible Dry Waste	4750	1010	7260	1550
Non-Compactible Solid Waste	234	373	567	910
Mixed Solid	5	7.5	10	15
Primary Filters (includes high activity and low activity cartridges)	5.2(3)	26	9.4(3)	69
Dry Waste Subtotals	**4994**	**1417**	**7846**	**2544**
TOTAL WET & DRY WASTES	**5759**	**1964**	**11147**	**5873**

Notes:

1. Radioactive secondary resins and membranes result from primary to secondary systems leakage (e.g., SG tube leak).
2. Estimated activity basis is ANSI 18.1 source terms in reactor coolant.
3. Estimated activity basis is breakdown and transfer of 10% of resin from upstream ion exchangers.
4. Reactor coolant source terms corresponding to 0.25% fuel defects.
5. Estimated activity basis from Tables 11.1-5, 11.1-7 and 11.1-8 and a typical 30 day process run time, once per refueling cycle.
6. Estimated volume and activity used for conservatism. Resin and membrane will be removed with the electrodeionization units and not stored as wet waste. See Subsection 10.4.8.

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Table 11.4-2 (Sheet 1 of 2)
Expected Annual Curie Content of Primary Influents

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Br-83	—	—
Br-84	1.98E-01	1.98E-02
Br-85	—	—
I-129	—	—
I-130	—	—
I-131	1.42E+02	1.42E+01
I-132	1.04E+01	1.04E+00
I-133	5.29E+01	5.29E+00
I-134	6.89E+00	6.89E-01
I-135	3.49E+01	3.49E+00
Rb-86	—	—
Rb-88	9.72E-01	9.72E-02
Rb-89	—	—
Cs-134	3.06E+02	3.06E+01
Cs-136	3.16E+00	3.16E-01
Cs-137	4.64E+02	4.64E+01
Cs-138	—	—
Ba-137m	4.44E+02	4.44E+01
Cr-51	3.21E+01	3.21E+00
Mn-54	1.04E+02	1.04E+01
Mn-56	—	—
Fe-55	1.04E+02	1.04E+01
Fe-59	5.00E+00	5.00E-01
Co-58	2.05E+02	2.05E+01
Co-60	9.59E+01	9.59E+00
Zn-65	3.02E+01	3.02E+00
Sr-89	2.67E+00	2.67E-01
Sr-90	1.13E+00	1.13E-01
Sr-91	1.72E-01	1.72E-02
Sr-92	—	—
Ba-140	6.29E+01	6.29E+00
Y-90	—	—
Y-91m	—	—
Y-91	3.74E-06	3.74E-07

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Table 11.4-2 (Sheet 2 of 2)
Expected Annual Curie Content of Primary Influents

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Y-92	—	—
Y-93	—	—
La-140	—	—
Zr-95	2.80E-04	2.80E-05
Nb-95	—	—
Mo-99	—	—
Tc-99m	—	—
Ru-103	5.35E-03	5.35E-04
Ru-106	6.37E-02	6.37E-03
Rh-103m	—	—
Rh-106	—	—
Te-132	—	—
Te-125m	—	—
Te-127m	—	—
Te-127	—	—
Te-129m	1.36E-04	1.36E-05
Te-129	—	—
Te-131m	—	—
Total:	2.11E+03	2.11E+02

Note:
Values shown as "—" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

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Table 11.4-3 (Sheet 1 of 2)
Maximum Annual Curie Content of Primary Influents

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Br-83	7.03E+00	7.03E-01
Br-84	3.42E-01	3.42E-02
Br-85	3.74E-03	3.74E-04
I-129	3.44E-03	3.44E-04
I-130	9.00E+00	9.00E-01
I-131	5.45E+03	5.45E+02
I-132	1.97E+02	1.97E+01
I-133	1.66E+03	1.66E+02
I-134	7.31E+00	7.31E-01
I-135	3.81E+02	3.81E+01
Rb-86	2.97E+01	2.97E+00
Rb-88	2.52E+01	2.52E+00
Rb-89	9.83E-01	9.83E-02
Cs-134	9.57E+03	9.57E+02
Cs-136	1.72E+03	1.72E+02
Cs-137	9.14E+03	9.14E+02
Cs-138	1.06E+01	1.06E+00
Ba-137m	8.66E+03	8.66E+02
Cr-51	3.95E+01	3.95E+00
Mn-54	1.18E+02	1.18E+01
Mn-56	4.75E+01	4.75E+00
Fe-55	1.14E+02	1.14E+01
Fe-59	5.84E+00	5.84E-01
Co-58	3.03E+02	3.03E+01
Co-60	2.45E+02	2.45E+01
Zn-65	—	—
Sr-89	4.56E+01	4.56E+00
Sr-90	1.09E+01	1.09E+00
Sr-91	1.16E+00	1.16E-01
Sr-92	9.96E-02	9.96E-03
Ba-140	1.19E+01	1.19E+00
Y-90	1.07E+01	1.07E+00
Y-91m	3.48E-01	3.48E-02
Y-91	5.48E-01	5.48E-02

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Table 11.4-3 (Sheet 2 of 2)
Maximum Annual Curie Content of Primary Influents

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Y-92	4.19E-02	4.19E-03
Y-93	9.07E-05	9.07E-06
La-140	1.07E+01	1.07E+00
Zr-95	—	—
Nb-95	—	—
Mo-99	—	—
Tc-99m	—	—
Ru-103	—	—
Ru-106	—	—
Rh-103m	—	—
Rh-106	—	—
Te-132	—	—
Te-125m	—	—
Te-127m	—	—
Te-127	—	—
Te-129m	—	—
Te-129	—	—
Te-131m	—	—
Total:	3.78E+04	3.78E+03

Note:
Values shown as "—" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

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Table 11.4-4 (Sheet 1 of 2)
Expected Annual Curie Content of Shipped Primary Wastes

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Br-83	—	—
Br-84	—	—
Br-85	—	—
I-129	—	—
I-130	—	—
I-131	6.04E-02	6.04E-03
I-132	—	—
I-133	—	—
I-134	—	—
I-135	—	—
Rb-86	—	—
Rb-88	—	—
Rb-89	—	—
Cs-134	2.81E+02	2.81E+01
Cs-136	2.61E-02	2.61E-03
Cs-137	4.61E+02	4.61E+01
Cs-138	—	—
Ba-137m	4.61E+02	4.61E+01
Cr-51	3.37E+00	3.37E-01
Mn-54	8.50E+01	8.50E+00
Mn-56	—	—
Fe-55	9.75E+01	9.75E+00
Fe-59	1.23E+00	1.23E-01
Co-58	8.51E+01	8.51E+00
Co-60	9.29E+01	9.29E+00
Zn-65	2.34E+01	2.34E+00
Sr-89	8.05E-01	8.05E-02
Sr-90	1.13E+00	1.13E-01
Sr-91	—	—
Sr-92	—	—
Ba-140	4.80E-01	4.80E-02
Y-90	1.13E+00	1.13E-01
Y-91m	—	—
Y-91	4.03E-04	4.03E-05

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Table 11.4-4 (Sheet 2 of 2)
Expected Annual Curie Content of Shipped Primary Wastes

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Y-92	—	—
Y-93	—	—
La-140	5.52E-01	5.52E-02
Zr-95	1.09E-04	1.09E-05
Nb-95	1.31E-04	1.31E-05
Mo-99	—	—
Tc-99m	—	—
Ru-103	1.10E-03	1.10E-04
Ru-106	5.38E-02	5.38E-03
Rh-103m	1.11E-03	1.11E-04
Rh-106	5.38E-02	5.38E-03
Te-132	—	—
Te-125m	—	—
Te-127m	—	—
Te-127	—	—
Te-129m	2.10E-05	2.10E-06
Te-129	1.37E-05	1.37E-06
Te-131m	—	—
Total:	1.60E+03	1.60E+02

Note:
Values shown as "—" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

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Table 11.4-5 (Sheet 1 of 2)
Maximum Annual Curie Content of Shipped Primary Wastes

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Br-83	—	—
Br-84	—	—
Br-85	—	—
I-129	3.44E-03	3.44E-04
I-130	—	—
I-131	4.10E+02	4.10E+01
I-132	—	—
I-133	6.27E-08	6.27E-09
I-134	—	—
I-135	—	—
Rb-86	9.76E+00	9.76E-01
Rb-88	—	—
Rb-89	—	—
Cs-134	9.31E+03	9.31E+02
Cs-136	3.47E+02	3.47E+01
Cs-137	9.13E+03	9.13E+02
Cs-138	—	—
Ba-137m	9.13E+03	9.13E+02
Cr-51	1.86E+01	1.86E+00
Mn-54	1.10E+02	1.10E+01
Mn-56	—	—
Fe-55	1.12E+02	1.12E+01
Fe-59	3.66E+00	3.66E-01
Co-58	2.26E+02	2.26E+01
Co-60	2.42E+02	2.42E+01
Zn-65	—	—
Sr-89	3.06E+01	3.06E+00
Sr-90	1.09E+01	1.09E+00
Sr-91	—	—
Sr-92	—	—
Ba-140	2.35E+00	2.35E-01
Y-90	1.09E+01	1.09E+00
Y-91m	—	—
Y-91	3.90E-01	3.90E-02

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Table 11.4-5 (Sheet 2 of 2)
Maximum Annual Curie Content of Shipped Primary Wastes

Isotope	Primary Resin Total Ci/yr	Primary Filter Total Ci/yr
Y-92	—	—
Y-93	—	—
La-140	2.70E+00	2.70E-01
Zr-95	—	—
Nb-95	—	—
Mo-99	—	—
Tc-99m	—	—
Ru-103	—	—
Ru-106	—	—
Rh-103m	—	—
Rh-106	—	—
Te-132	—	—
Te-125m	—	—
Te-127m	—	—
Te-127	—	—
Te-129m	—	—
Te-129	—	—
Te-131m	—	—
Total:	2.91E+04	2.91E+03

Note:
Values shown as "—" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

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Table 11.4-6 (Sheet 1 of 2)
Expected Annual Curie Content of Secondary Waste as Generated

Isotope	Secondary Resin Total Ci/yr
Na-24	1.83E-02
Cr-51	4.29E-02
Mn-54	2.95E-02
Fe-55	2.35E-02
Fe-59	4.49E-03
Co-58	7.78E-02
Co-60	1.03E-02
Zn-65	9.56E-03
Br-84	2.22E-05
Rb-88	8.99E-05
Sr-89	2.24E-03
Sr-90	2.37E-04
Sr-91	2.11E-04
Y-90	2.06E-04
Y-91	2.53E-04
Y-91m	1.82E-04
Y-93	9.80E-04
Zr-95	6.53E-03
Nb-95	5.19E-03
Nb-95m	4.74E-03
Mo-99	1.52E-02
Tc-99m	1.41E-02
Ru-103	1.13E-01
Ru-106	1.65E+00
Rh-103m	1.39E-01
Rh-106	2.11E+00
Ag-110	2.12E-02
Ag-110m	2.45E-02
Te-129	2.29E-03
Te-129m	2.79E-03
Te-131	1.14E-03
Te-131m	1.42E-03
Te-132	4.74E-04

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Table 11.4-6 (Sheet 2 of 2)
Expected Annual Curie Content of Secondary Waste as Generated

Isotope	Secondary Resin Total Ci/yr
I-131	1.70E-01
I-132	7.93E-03
I-133	5.23E-02
I-134	1.18E-03
I-135	2.56E-02
Xe-131m	—
Xe-133	—
Xe-135	—
Cs-134	2.50E-01
Cs-135	4.70E-10
Cs-136	1.48E-02
Cs-137	3.39E-01
Ba-136m	1.39E-02
Ba-137m	3.42E-01
Ba-140	1.17E-01
La-140	1.47E-01
Ce-141	2.13E-03
Ce-143	2.91E-03
Ce-144	7.35E-02
Pr-143	2.04E-03
Pr-144	6.37E-02
Total:	5.96E+00

Note:
Values shown as "—" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

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Table 11.4-7 (Sheet 1 of 2)
Maximum Annual Curie Content of Secondary Waste as Generated

Isotope	Secondary Resin Total Ci/yr
Na-24	4.62E-04
Cr-51	5.17E-01
Mn-54	3.55E-01
Mn-56	2.24E-01
Fe-55	2.78E-01
Fe-59	5.88E-02
Co-58	9.25E-01
Co-60	1.23E-01
Br-83	3.73E-02
Br-84	1.41E-03
Br-85	1.64E-06
Kr-83m	—
Kr-85	—
Kr-85m	—
Rb-88	4.56E-02
Rb-89	1.53E-03
Sr-89	9.10E-01
Sr-90	5.00E-02
Sr-91	2.13E-02
Sr-92	7.25E-04
Y-90	4.60E-02
Y-91	4.34E-02
Y-91m	2.11E-02
Y-92	2.66E-03
Y-93	1.04E-03
Zr-95	7.74E-02
Nb-95	8.25E-02
Nb-95m	5.52E-02
Mo-99	1.52E+01
Tc-99m	1.68E+01
Ru-103	6.28E-02
Ru-103m	3.87E-02
Rh-103m	6.29E-02
Rh-106	5.95E-02

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Table 11.4-7 (Sheet 2 of 2)
Maximum Annual Curie Content of Secondary Waste as Generated

Isotope	Secondary Resin Total Ci/yr
Ag-110	1.34E-02
Ag-110m	2.24E-01
Te-129	1.19E+00
Te-129m	1.10E+00
Te-131	2.35E+00
Te-131m	2.01E-01
Te-132	6.75E+00
Te-134	1.49E-03
I-130	1.19E-01
I-131	1.37E+02
I-132	6.77E+00
I-133	2.51E+01
I-134	4.99E-02
I-135	3.99E+00
Xe-131m	—
Xe-133	—
Xe-135	—
Cs-134	6.90E+02
Cs-135	6.16E-08
Cs-136	5.15E+02
Cs-137	5.00E+02
Cs-138	3.41E-02
Ba-136m	6.35E+02
Ba-137m	5.14E+02
Ba-140	2.83E-01
La-140	3.31E-01
Ce-141	6.42E-02
Ce-143	4.94E-03
Ce-144	6.33E-02
Pr-143	4.63E-02
Pr-144	6.33E-02
Total:	3.08E+03

Note:
Values shown as "—" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

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Table 11.4-8 (Sheet 1 of 2)
Expected Annual Curie Content of Shipped Secondary Wastes

Isotope	Secondary Resin Total Ci/yr
Na-24	—
Cr-51	4.55E-03
Mn-54	2.40E-02
Fe-55	2.19E-02
Fe-59	1.14E-03
Co-58	3.25E-02
Co-60	9.95E-03
Zn-65	7.42E-03
Br-84	—
Rb-88	—
Sr-89	6.86E-04
Sr-90	2.36E-04
Sr-91	—
Y-90	2.31E-04
Y-91	6.71E-09
Y-91m	—
Y-93	—
Zr-95	2.52E-03
Nb-95	4.06E-03
Nb-95m	2.32E-03
Mo-99	—
Tc-99m	—
Ru-103	2.34E-02
Ru-106	1.38E+00
Rh-103m	2.87E-02
Rh-106	1.77E+00
Ag-110	1.66E-02
Ag-110m	1.92E-02
Te-129	3.44E-04
Te-129m	4.48E-04
Te-131	—
Te-131m	—

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Table 11.4-8 (Sheet 2 of 2)
Expected Annual Curie Content of Shipped Secondary Wastes

Isotope	Secondary Resin Total Ci/yr
Te-132	—
I-131	7.32E-05
I-132	—
I-133	—
I-134	—
I-135	—
Xe-131m	—
Xe-133	—
Xe-135	—
Cs-134	2.31E-01
Cs-135	4.86E-10
Cs-136	1.56E-04
Cs-137	3.36E-01
Ba-136m	1.47E-04
Ba-137m	3.40E-01
Ba-140	8.97E-04
La-140	1.05E-03
Ce-141	3.13E-04
Ce-143	—
Ce-144	5.91E-02
Pr-143	2.38E-05
Pr-144	5.12E-02
Total:	4.38E+00

Note:
Values shown as "—" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

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Table 11.4-9 (Sheet 1 of 2)
Maximum Annual Curie Content of Shipped Secondary Wastes

Isotope	Secondary Resin Total Ci/yr
Na-24	—
Cr-51	5.47E-02
Mn-54	2.89E-01
Mn-56	—
Fe-55	2.60E-01
Fe-59	1.50E-02
Co-58	3.87E-01
Co-60	1.19E-01
Br-83	—
Br-84	—
Br-85	—
Kr-83m	—
Kr-85	—
Kr-85m	—
Rb-88	—
Rb-89	—
Sr-89	2.79E-01
Sr-90	4.96E-02
Sr-91	—
Sr-92	—
Y-90	5.12E-02
Y-91	1.12E-06
Y-91m	—
Y-92	—
Y-93	—
Zr-95	2.98E-02
Nb-95	5.19E-02
Nb-95m	2.70E-02
Mo-99	2.72E-09
Tc-99m	3.04E-09
Ru-103	1.30E-02
Ru103m	3.27E-02

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Table 11.4-9 (Sheet 2 of 2)
Maximum Annual Curie Content of Shipped Secondary Wastes

Isotope	Secondary Resin Total Ci/yr
Rh-103m	1.30E-02
Rh-106	5.03E-02
Ag-110	1.05E-02
Ag-110m	1.76E-01
Te-129	1.92E-01
Te-129m	1.77E-01
Te-131	—
Te-131m	—
Te-132	2.90E-08
Te-134	—
I-130	—
I-131	5.94E-02
I-132	2.36E-08
I-133	—
I-134	—
I-135	—
Xe-131m	—
Xe-133	—
Xe-135	—
Cs-134	6.35E+02
Cs-135	6.36E-08
Cs-136	5.42E+00
Cs-137	4.98E+02
Cs-138	—
Ba-136m	6.69E+00
Ba-137m	5.11E+02
Ba-140	2.18E-03
La-140	2.87E-03
Ce-141	9.41E-03
Ce-143	—
Ce-144	5.08E-02
Pr-143	4.75E-04
Pr-144	5.08E-02
Total:	1.66E+03

Note:
Values shown as "—" Ci/yr are those calculated to be lower than 1.0E-10 Ci/yr, and thus considered to have insignificant contributions to total.

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Table 11.4-10 (Sheet 1 of 2)
Component Data — Solid Waste Management System (Nominal)

Tanks	
Spent resin tank	
Number	2
Total volume (ft^3)	300
Type	Vertical, conical bottom, dished top
Design pressure (psig)	15
Design temperature (°F)	150
Material	Stainless steel
Pumps	
Resin mixing pump	
Number	1
Type	Pneumatic diaphragm
Design pressure (psig)	125
Design temperature (°F)	150
Design flow rate (gpm)	100
Air supply pressure (psig)	100
Air consumption (scfm)	130
Material	Stainless steel housing, Buna N diaphragms
Resin transfer pump	
Number	1
Type	Material handling positive displacement
Design pressure (psig)	125
Design temperature (°F)	150
Design flow rate (gpm)	100
Material	Stainless steel housing, Buna N flexible parts

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Table 11.4-10 (Sheet 2 of 2)
Component Data — Solid Waste Management System (Nominal)

Filters	
Resin fines filter	
Number	1
Type	Filter cartridge for inside to outside flow
Design pressure (psig)	150
Design temperature (°F)	150
Design flowrate (gpm)	120
Filtration rating	Up to 20 microns
Material	Stainless steel housing and pleated polypropylene cartridge with stainless steel screen outer jacket
Sampler	
Resin sampling device	
Number	1
Type	Inline sampler, positive displacement sample collection and portable pig for sample jar
Material	Stainless steel and EPDM wetted parts

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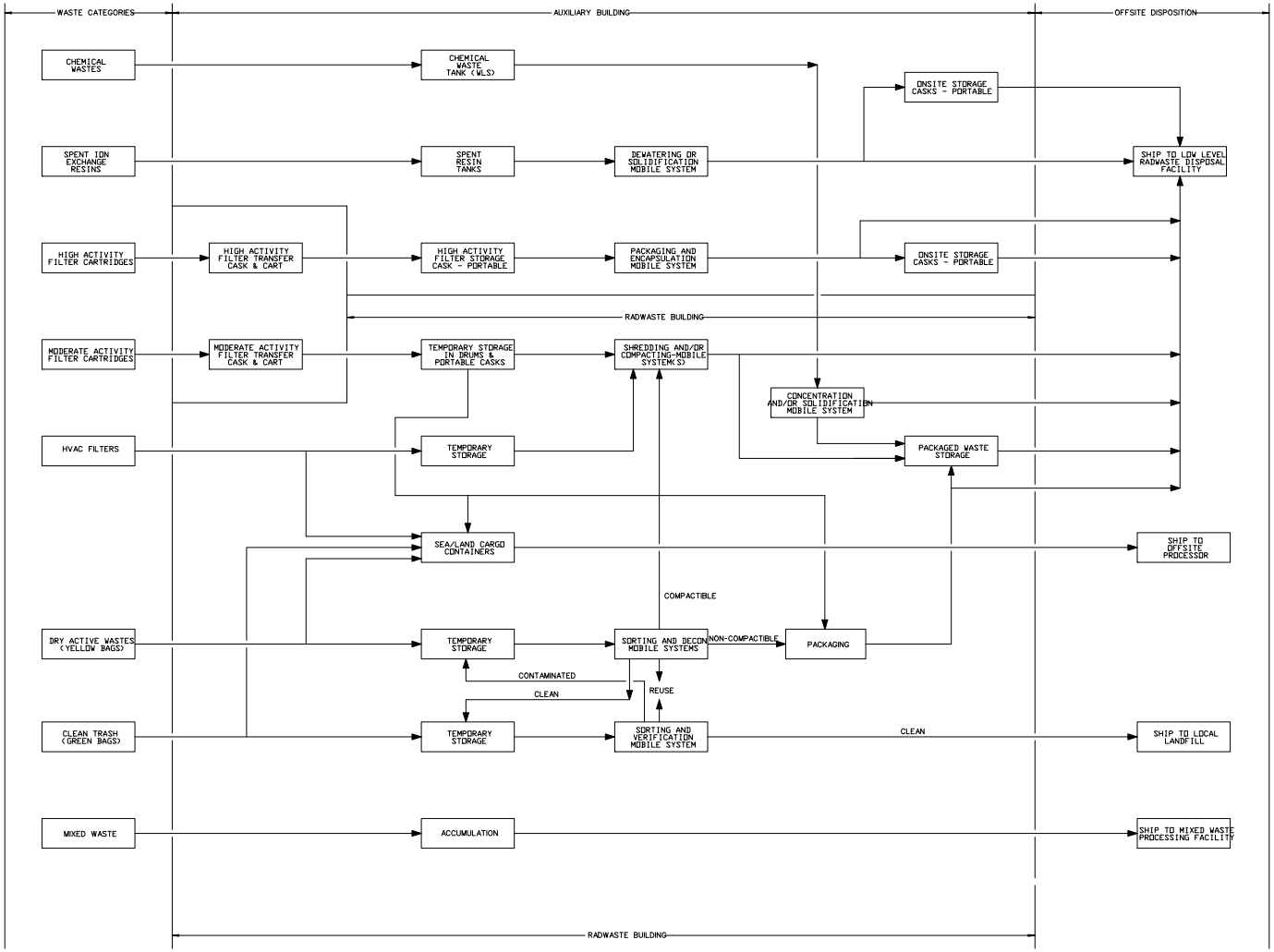

Figure 11.4-1
Waste Processing System
Flow Diagram

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11.5 Radiation Monitoring

The radiation monitoring system (RMS) provides plant effluent monitoring, process fluid monitoring, airborne monitoring, and continuous indication of the radiation environment in plant areas where such information is needed. Radiation monitors that have a safety-related function are qualified environmentally, seismically, or both. Class 1E radiation monitors conform to the separation criteria described in Subsection 8.3.2 and to the fire protection criteria described in Subsection 9.5.1. Equipment qualification requirements, including seismic qualification requirements, and general location information for radiation monitors are listed in Section 3.11. Seismic Categories for the buildings housing radiation monitors are listed in Section 3.2.

The radiation monitoring system is installed permanently and operates in conjunction with regular and special radiation survey programs to assist in meeting applicable regulatory requirements. The radiation monitoring system for the VFS plant is designed in accordance with guidelines in ANSI N13.1-1969. The use of alternate sampling systems such as single point shrouded nozzles is also an acceptable method for obtaining representative samples. The liquid effluent monitors are designed in accordance with ANSI-N42.18-1980.

The radiation monitoring system is divided functionally into two subsystems:

- Process, airborne, and effluent radiological monitoring and sampling
- Area radiation monitoring

11.5.1 Design Basis

11.5.1.1 Safety Design Basis

While the radiation monitoring system is primarily a surveillance system, certain detector channels perform safety-related functions. The components used in these channels meet the equipment qualification requirements for safety-related equipment as described in Subsection 7.1.4.

Channel and equipment redundancy is provided for safety-related monitors to maintain the safety-related function in case of a single failure.

The design objectives of the radiation monitoring system during postulated accidents are:

- Initiate containment air filtration isolation in the event of abnormally high radiation inside the containment (High-1)
- Initiate normal residual heat removal system suction line containment isolation in the event of abnormally high radiation inside the containment (High-2)
- Initiate main control room supplemental filtration in the event of abnormally high particulate, iodine, or gaseous radioactivity in the main control room supply air (High-1)
- Initiate main control room ventilation isolation and actuate the main control room emergency habitability system in the event of abnormally high particulate or iodine radioactivity in the main control room supply air (High-2)
- Provide long-term post-accident monitoring (using both safety-related and nonsafety-related monitors)

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The scope of the radiation monitoring system for post-accident monitoring is set forth in General Design Criterion 64 and in the provisions of Regulatory Guide 1.97.

11.5.1.2 Power Generation Design Basis

The radiation monitoring system is designed to support the requirements of 10 CFR 20 and to provide:

- Equipment to meet the applicable regulatory requirements for both normal operation and transient events
- Data to aid plant health physics personnel in limiting release of radioactivity to the environment and limiting exposure of operation and maintenance personnel to meet ALARA (as-low-as-reasonably-achievable) guidance
- Early indication of a system or equipment malfunction that could result in excessive radiation dose to plant personnel or lead to plant damage
- Data collection and data storage to support compliance reporting for the applicable NRC requirements and guidelines, such as General Design Criterion 64 and Regulatory Guide 1.21 and Regulatory Guide 4.15, Revision 1.
- Exhausts to the environment from the personnel areas in the annex building, electrical and mechanical equipment rooms in the annex and auxiliary buildings, and the diesel generator rooms will not be radioactive because they contain no radioactive materials. These ventilation exhausts are not monitored.

11.5.2 System Description

11.5.2.1 Radiation Monitoring System

The radiation monitoring system uses distributed radiation monitors, where each radiation monitor consists of one or more radiation detectors and a dedicated radiation processor.

Each radiation processor receives, averages and stores radiation data and transmits data to the data display and processing system (which, in turn, transmits data to the plant control system) or to the protection and safety monitoring system (for safety-related monitors) for control (as required), display and recording. These alarms include: low (fail), alert, and high. Storage of radiation readings is provided.

Each radiation detector, except the in-duct radiation detectors and the containment high range ion chambers, has a check source that is actuated from the associated local radiation processor. The check source is used to verify detector and monitor operation. The check source is shielded to meet ALARA requirements, and returns to its fully retracted/shielded position upon loss of actuator power. The in-duct radiation detector operation may be checked using an internal LED to simulate light pulses emitted in response to radiation. The containment high range monitors have an internal source that provides a minimum reading; loss of signal from the detector indicates detector inoperability.

Radiation monitoring data, including alarm status, are provided to AP1000 operators via the plant control system (and the protection and safety monitoring system for Class 1E monitors). The information is available in either counts per minute (count rate), microCuries/cc (activity concentration), or R/hr (radiation dose rate).

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Safety-related channels are environmentally qualified and are powered from the Class 1E dc and uninterruptible power supply system. Nonsafety-related Regulatory Guide 1.97 channels are powered by non-Class 1E dc and uninterruptible power supply system. Other nonsafety-related channels are powered by non-Class 1E ac power.

11.5.2.2 Monitor Functional Description

The process and effluent radiological monitoring and sampling subsystem provides radiation monitoring for the four functional classifications listed below. Individual monitors may provide functionality in more than one of these classifications.

- Fluid process monitors determine concentrations of radioactive material in plant fluid systems
- Airborne monitors provide operators with information on concentrations of radioactivity at various points in the ventilation system, providing information on airborne concentrations in the plant
- Liquid and gaseous effluent monitors measure radioactive materials discharged to the environs
- Post-accident monitors monitor potential pathways for release of radioactive materials during accident conditions

The area radiation monitoring subsystem provides plant personnel information on radiation at fixed locations in AP1000. Post-accident monitoring functions are also performed by certain area monitors.

11.5.2.3 Monitor Descriptions

For offline gaseous monitors, the radiation monitor includes a low pressure drop flow sensor suitable for measuring the sample flow. The radiation processor receives an analog signal input from this flow sensor. This signal is used by the radiation processor to control sample flow. The analog signal is transmitted to the plant control system (protection and safety monitoring system for safety-related monitors). For offline liquid monitors, a flow indicator is provided for manual adjustment of the flow.

Those airborne radiation monitors which monitor plant areas which may be occupied by plant personnel will be capable of detecting 10 DAC-hours. The specific radiation monitors which are included in this category are identified in Table 11.5-1.

11.5.2.3.1 Fluid Process Monitors

Steam Generator Blowdown Radiation Monitors

The steam generator blowdown radiation monitors (BDS-JE-RE010, RE011) measure the concentration of radioactive material in the blowdown from the steam generators. One measures radiation in the purification process before it is returned to the condensate system. The other measures radioactivity in the blowdown system electrodeionization reject (waste stream) before it is discharged to the waste water system. The presence of radioactive material in the steam generator blowdown indicates a leak between the primary side and the secondary side of the steam generator. Refer to Subsection 5.2.5 for details of leakage monitoring and to Subsections 10.4.8 and 11.2 for process system details. The steam generator blowdown radiation monitors meet the guidelines of Regulatory Guide 1.97 as discussed in Appendix 1A and Section 7.5.

AP1000 has two steam generators, each of which has a blowdown line. Each blowdown line has a heat exchanger upstream of the blowdown flow control valve. The steam generator blowdown

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radiation detectors are located in the lines downstream of these heat exchangers. Therefore, the radiation monitors do not require a sample cooler.

When its predetermined setpoint is exceeded, each steam generator blowdown radiation monitor initiates an alarm in the main control room, initiates closure of the blowdown system train isolation valves and the steam generator blowdown flow control valves, and diverts flow to the liquid radwaste system.

The steam generator blowdown radiation monitors use inline gamma-sensitive, thallium-activated, sodium iodide scintillation detectors. The steam generator blowdown radiation monitor detector range and principal isotopes are listed in Table 11.5-1.

The arrangement for the steam generator blowdown radiation monitor is shown in Figure 11.5-1.

Component Cooling Water System Radiation Monitor

The component cooling water system radiation monitor (CCS-JE-RE001) measures the concentration of radioactive material in the component cooling water system. Radioactive material in the component cooling water system provides indication of leakage. Refer to Subsection 5.2.5 for details of leakage monitoring and to Subsection 9.2.2 for process system details.

If the concentration of radioactive materials exceeds a predetermined setpoint, the component cooling water system radiation monitor initiates an alarm in the main control room.

The component cooling water system radiation monitor is an offline monitor that uses a gamma-sensitive, thallium-activated, sodium iodide scintillation detector. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the component cooling water system radiation monitor is shown in Figure 11.5-7.

Main Steam Line Radiation Monitors

The main steam line radiation monitors (SGS-JE-RE026A/B and SGS-JE-RE027A/B) measure the concentration of radioactive materials in the two main steam lines. Additionally, the main steam line radioisotope concentration data are used to calculate releases to the environment if the steam generator safety relief or power operated relief valves release steam to the atmosphere. The main steam line radiation monitors (SGS-JE-RE026A and SGS-JE-RE027A) meet the guidelines of Regulatory Guide 1.97 as discussed in Appendix 1A and Section 7.5. If the concentration of radioactive materials exceeds a predetermined setpoint, the main steam line radiation monitors initiate alarms in the main control room.

The main steam line radiation monitors are positioned adjacent to the steam lines. Each monitor detector shield is arranged so that the detector sensitive volume is exposed to the radiation originating inside the steam line on which it is located, and is shielded from radiation originating in the other steam line. Radioactive material in the main steam line provides early indication of leakage in the form of a steam generator tube leak. Refer to Subsection 5.2.5 for details of leakage monitoring and to Section 10.3 for process system details.

The main steam line radiation monitor detectors use gamma-sensitive detectors.

Each main steam line radiation monitor range and principal isotopes are listed in Table 11.5-1.

The arrangement for a main steam line radiation monitor is shown in Figure 11.5-8.

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Service Water Blowdown Radiation Monitor

The service water blowdown radiation monitor (SWS-JE-RE008) measures the concentration of radioactive materials in the blowdown flow from the service water system. Upstream of the radiation monitor, local grab sampling is available.

The service water blowdown radiation monitor initiates an alarm in the main control room if the concentration of radioactive materials exceeds a predetermined setpoint. Following the alarm, the operator can manually isolate the blowdown flow. Refer to Subsection 9.2.1 for system details.

The service water blowdown monitor is an inline monitor using a gamma-sensitive, thallium-activated, sodium iodide scintillation detector. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the service water blowdown radiation monitor is shown in Figure 11.5-1.

Primary Sampling System Liquid Sample Radiation Monitor

The primary sampling system (PSS) liquid sample radiation monitor (PSS-JE-RE050) measures and indicates the concentration of radioactive materials in the samples from the reactor coolant system. The liquid sample radiation monitor's primary function is to indicate elevated sample radiation levels following a design basis or severe accident. High radiation levels show the need for sample dilution to limit operator exposure during sampling and sample transport for analysis. The monitor may also be used to provide early indication of a significant increase in the radioactivity of the reactor coolant indicating a possible fuel cladding breach. When a predetermined setpoint is exceeded, the primary sampling system liquid sample radiation monitor isolates the sample flow by closing the outside containment isolation valve and initiates an alarm in the main control room and locally to alert the operator. Refer to Subsection 9.3.3 for system details.

The primary sampling system liquid sample radiation monitor utilizes a gamma-sensitive radiation detector that is adjacent to the sampling line immediately downstream of the sample cooler. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the primary sampling system liquid sample radiation monitor is shown in Figure 11.5-8.

Primary Sampling System Gaseous Sample Radiation Monitor

The primary sampling system gaseous sample radiation monitor (PSS-JE-RE052) measures the radiation from gaseous samples taken from containment atmosphere. The gaseous sample radiation monitor is used to provide indication of significant radioactivity in the gaseous sample being taken and the need for dilution of the sample to limit operator exposure during sampling and transport for analysis. When a predetermined setpoint is exceeded, the primary sampling system gaseous sample radiation monitor initiates an alarm locally and in the main control room to alert the operator. Refer to Subsection 9.3.3 for system details.

The primary sampling system gaseous sample radiation monitor utilizes a gamma-sensitive radiation detector that is adjacent to the sampling line immediately upstream of the sample bottle. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the primary sampling system gaseous sample radiation monitor is shown in Figure 11.5-8.

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Main Control Room Supply Air Duct Radiation Monitors

The main control room supply air duct radiation monitors (particulate detectors VBS-JE-RE001A and VBS-JE-RE001B, iodine detectors VBS-JE-RE002A and VBS-JE-RE002B, and noble gas detectors VBS-JE-RE003A and VBS-JE-RE003B) are offline monitors that continuously measure the concentration of radioactive materials in the air that is supplied to the main control room by the nuclear island nonradioactive ventilation system air handling units. The control support area ventilation is also part of this air supply system. The air supply is partially outside air. Refer to Subsection 9.4.1 for system details. The main control room supply air duct radiation monitors receive safety-related power. When predetermined setpoints are exceeded, the monitors provide signals to initiate the supplemental air filtration system on a High-1 gaseous, particulate, or iodine concentration, and to isolate the main control room air intake and exhaust ducts and activate the main control room emergency habitability system on High-2 particulate or iodine concentrations. Alarms are also provided in the main control room for these high concentrations.

The main control room supply air duct radiation monitor components are qualified environmentally and seismically in accordance with the guidelines of Regulatory Guides 1.89 and 1.100, respectively. Each monitor meets the guidelines of Regulatory Guide 1.97 as discussed in Appendix 1A and Section 7.5.

The particulate detectors are beta-sensitive scintillation detectors that view a fixed filter. The iodine detectors are gamma-sensitive, thallium-activated, sodium iodide scintillation detectors that view a fixed charcoal filter. The gas detectors are beta-sensitive scintillation detectors. The range and principal radioisotopes are listed in Table 11.5-1.

The arrangement for a main control room supply air duct radiation monitor is shown in Figure 11.5-6.

Containment Air Filtration Exhaust Radiation Monitor

The containment air filtration exhaust radiation monitor (VFS-JE-RE001) measures the concentration of radioactive materials in the containment purge exhaust air.

The monitor provides an alarm in the main control room when the concentration of radioactive gases in the exhaust exceeds a predetermined setpoint. Refer to Subsection 9.4.7 for system details.

The containment air filtration exhaust radiation monitor is an inline monitor that uses a beta-sensitive scintillation detector. It is located downstream of the containment air filtration units with its sensitive volume inside the duct. The detector range and principal radioisotopes are listed in Table 11.5-1.

The arrangement of the containment air filtration exhaust radiation monitor is shown in Figure 11.5-5.

Gaseous Radwaste Discharge Radiation Monitor

The gaseous radwaste discharge radiation monitor (WGS-JE-RE017) measures the concentration of radioactive materials in the releases from the gaseous radwaste system to the plant vent. The measurement is made before the discharge reaches the plant vent or is diluted by any other flows.

The gaseous radwaste discharge radiation monitor provides an alarm in the main control room and terminates the release of radioactive gas to the plant vent by closing the discharge isolation valve when a predetermined setpoint is exceeded. Refer to Section 11.3 for system details.

The monitor is an inline monitor using a gamma sensitive, thallium-activated, sodium iodine scintillation detector with its sensitive volume inside the piping. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the gaseous radwaste discharge radiation monitor is shown in Figure 11.5-1.

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Containment Atmosphere Radiation Monitor

The containment atmosphere radiation monitor measures the radioactive gaseous (PSS-JE-RE026) and F18 particulate (PSS-JE-RE027) concentrations in the containment atmosphere. The containment atmosphere radiation monitor is a part of the reactor coolant pressure boundary leak detection system described in Subsection 5.2.5. The presence of gaseous or F18 radioactivity in the containment atmosphere is an indication of reactor coolant pressure boundary leakage. Refer to Subsection 5.2.5 for further details. Conformance with Regulatory Guide 1.45 is discussed in Appendix 1A.

The containment atmosphere radiation monitor has the capability to compensate for the influence of pressure differentials between the sampling system and the containment atmosphere.

The radiogas detector is a beta-sensitive scintillation detector. The F18 particulate detector is also a beta-sensitive scintillation detector. The ranges and principal isotopes are listed in Table 11.5-1.

The arrangement for the containment atmosphere radiation monitor is shown in Figure 11.5-3.

11.5.2.3.2 Airborne Monitors

Fuel Handling Area Exhaust Radiation Monitor

The fuel handling area exhaust radiation monitor (VAS-JE-RE001) measures the concentration of radioactive materials in the exhaust air from the fuel handling area. This radiation monitor is located upstream of the exhaust air isolation damper.

When a predetermined setpoint is exceeded, the fuel handling area exhaust radiation monitor provides signals to alarm in the main control room, to initiate closure of the fuel handling area supply and exhaust air isolation dampers, to open the fuel handling area exhaust air isolation damper to the containment air filtration exhaust units, and to start a containment air filtration exhaust unit. These actions provide a filtered air path from the fuel handling area to the plant vent. Refer to Subsection 9.4.3 for system details.

The fuel handling area exhaust radiation monitor is an inline monitor that uses a beta-sensitive scintillation detector. It is located with the sensitive volume inside the exhaust duct. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the fuel handling area exhaust radiation monitor is shown in Figure 11.5-5.

Auxiliary Building Exhaust Radiation Monitor

The auxiliary building exhaust radiation monitors (VAS-JE-RE002 and VAS-JE-RE003) measure the concentration of radioactive materials in the radiologically controlled area ventilation system exhaust air from the auxiliary building. The auxiliary building radiation monitor detectors are upstream of the exhaust air isolation damper.

When a predetermined setpoint is exceeded, indicating abnormal airborne radiation, the auxiliary building exhaust radiation monitors provide signals to alarm in the main control room, to initiate automatic closure of the affected radiologically controlled area ventilation system zone supply and exhaust air isolation dampers, to automatically open the radiologically controlled area ventilation system zone exhaust air isolation damper to the containment air filtration exhaust units, and to automatically start a containment air filtration exhaust unit. These actions provide a filtered air path from the affected radiologically controlled area ventilation system zone to the plant vent. Once automatically closed, the zone supply and exhaust air isolation dampers do not automatically reopen and can only be opened manually when airborne radioactivity levels decrease to below the predetermined setpoint of the affected radiation monitor. The operators only restore normal zone

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supply and exhaust operation after determining and correcting the source of abnormal airborne radioactivity in the affected zone. Refer to Subsection 9.4.3 for system details.

The auxiliary building exhaust radiation monitors are inline monitors that use a beta-sensitive scintillation detector. The detectors are located with the sensitive volume inside the exhaust duct. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the auxiliary building exhaust radiation monitors are shown in Figure 11.5-5.

Annex Building Exhaust Radiation Monitor

The annex building exhaust radiation monitor (VAS-JE-RE008) measures the concentration of radioactive materials in the radiologically controlled area ventilation system exhaust air from the annex building. The annex building exhaust radiation monitor is located upstream of the annex building exhaust air isolation damper.

When a predetermined setpoint is exceeded, indicating abnormal airborne radiation, the annex building exhaust radiation monitor provides signals to alarm in the main control room, to initiate automatic closure of the affected radiologically controlled area ventilation system zone supply and exhaust air isolation dampers, to automatically open the radiologically controlled area ventilation system zone exhaust air isolation damper to the containment air filtration units, and to automatically start a containment air filtration exhaust unit. These actions provide a filtered air path from the affected radiologically controlled area ventilation system zone to the plant vent. Once automatically closed, the zone supply and exhaust air isolation dampers do not automatically reopen and can only be opened manually when airborne radioactivity levels decrease to below the predetermined setpoint of the affected radiation monitor. The operators only restore normal zone supply and exhaust operation after determining and correcting the source of abnormal airborne radioactivity in the affected zone. Refer to Subsection 9.4.3 for system details.

The annex building monitor is an inline monitor that uses a beta-sensitive scintillation detector. It is located with the sensitive volume inside the exhaust duct. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the annex building exhaust radiation monitor is shown in Figure 11.5-5.

Health Physics and Hot Machine Shop Exhaust Radiation Monitor

The health physics and hot machine shop exhaust radiation monitor (detector VHS-JE-RE001) measures the concentration of radioactive materials in the exhaust air from the health physics area and the hot machine shop. The monitor provides an alarm in the main control room when the concentration of radioactive gases in the exhaust exceeds a predetermined setpoint. Refer to Subsection 9.4.11 for system details.

The monitor is an offline monitor, located downstream of the exhaust fans, that uses a beta-sensitive scintillation detector viewing a fixed particulate filter. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the health physics and hot machine shop exhaust radiation monitor is shown in Figure 11.5-9.

Radwaste Building Exhaust Radiation Monitor

The radwaste building exhaust radiation monitor (VRS-JE-RE023) measures the concentration of radioactive materials in the exhaust air from the radwaste building. The monitor provides an alarm in the main control room when radioactive material concentrations in the exhaust duct exceed a predetermined setpoint. Refer to Subsection 9.4.8 for system details.

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The monitor is an offline monitor, located downstream of the exhaust fans, that uses a beta-sensitive scintillation detector viewing a fixed particulate filter. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the radwaste building exhaust radiation monitor is shown in Figure 11.5-9.

11.5.2.3.3 Liquid and Gaseous Effluent Monitors

Plant Vent Radiation Monitor

The plant vent radiation monitor measures the concentration of radioactive airborne contamination being released through the plant vent, which is the only design pathway for the release of radioactive materials to the atmosphere. The plant vent radiation monitor includes sample nozzle assemblies and a plant vent effluent flow sensor. Heat tracing is provided for the sample line. The monitor also provides particulate, iodine, and gaseous grab sampling capability.

The plant vent is sampled continuously for the full range of concentrations between normal conditions and those postulated in Regulatory Guide 1.97. The plant vent radiation monitor is a post-accident monitor and meets the guidelines of Regulatory Guide 1.97 and NUREG-0737 as discussed in Appendix 1A and Section 7.5. Alarms are provided in the main control room if radioactivity concentrations exceed predetermined setpoints. The plant vent radiation monitor also provides data for plant effluent release reports identified in Regulatory Guide 1.21. For further process details, refer to Subsection 11.3.3.

The normal range particulate detector, VFS-JE-RE101, uses a beta-sensitive scintillation detector that views a fixed filter. The accident range particulate filter is fixed and identical to the normal range filter. The accident range particulate filter is analyzed in an onsite laboratory.

The normal range iodine detector, VFS-JE-RE102, is a gamma-sensitive, thallium-activated, sodium iodide, scintillation detector that views a fixed charcoal filter. The accident range iodine filter is a fixed silver zeolite filter. The accident range iodine filter is analyzed in an onsite laboratory.

The three radiogas channels measure the entire specified range, with overlap in the detector ranges. The normal range radiogas detector, VFS-JE-RE103, is a beta-sensitive scintillation detector. The accident range radiogas detectors, VFS-JE-RE104A (mid-range) and VFS-JE-RE104B (high-range), are beta/gamma-sensitive detectors with small sensitive volumes compared to the normal range radiogas detector.

The plant vent radiation monitor detector ranges and principal radioisotopes are listed in Table 11.5-1. The arrangement for the plant vent radiation monitor is shown in Figure 11.5-4.

The plant vent radiation monitor accepts analog signal inputs from the plant vent effluent and sample flow sensor. These signals are used to calculate concentrations, releases and flow rates at standard conditions. These analog signals are also used to calculate total process flow, total sample flow, and total discharge for an operator-selected period.

The normal range particulate, iodine, and radiogas detectors are deactivated automatically when the gas channel concentration exceeds the normal range. The sample flow bypasses the normal range detectors and a small portion is extracted for the accident range particulate and iodine sample filters and radiogas detectors. This prevents normal range detector damage and allows these detectors to be used to measure the concentrations after they decrease again to within the normal range detector ranges.

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The following design criteria for particulate and iodine collection are applied to the design of the plant vent and vent sampling system:

- The sample extraction point is located at a sufficient distance downstream of perturbations or flow entry points to provide fully developed flow in the turbulent regime.
- The sample extraction point is located between the discharge plane of a fan and the stack exit plane, and is not located close the to the stack exit plane where wind effects significantly influence the velocity profile at the sampling location.
- The sample nozzles provide high efficiency transmission ratios (80 to 130%) and an aspiration ratio of 0.80 to 1.50 over the expected normal and off-normal flow range for 10 micron aerodynamic diameter (AD) particles.
- The sample line layout includes features to provide particle transport efficiency, including the following:
  - Non-reactive materials are used in the construction of sample lines.
  - Sample line deposition analyses are performed.
  - The distance between the sampling nozzles and the sample collection stations is minimized, within the requirements of the overall layout requirements.
  - Long horizontal runs are avoided.
  - Long radius bends are used.
  - Heat tracing is included if needed to avoid condensation of water or iodine.

Turbine Island Vent Discharge Radiation Monitor

The turbine island vent discharge radiation monitor (TDS-JE-RE001A/B) measures the concentration of radioactive gases in the steam and non-condensable gases that are discharged by the condenser vacuum pumps and the gland seal steam condenser. This measurement provides early indication of leakage between the primary and secondary sides of the steam generators. The monitor provides an alarm in the main control room if concentrations exceed a predetermined setpoint. Refer to Subsection 5.2.5 for leakage monitoring details and to Subsections 10.4.2 and 10.4.3 for process system details. The turbine island vent discharge radiation monitor meets the guidelines of Regulatory Guide 1.97 as discussed in Appendix 1A and Section 7.5.

The turbine island vent discharge radiation monitor provides data for reports of gaseous releases of radioactive materials in accordance with Regulatory Guide 1.21. The monitor is an inline monitor that uses two beta/gamma-sensitive Geiger-Mueller tubes with overlap in the detector ranges. The range and principal isotopes are listed in Table 11.5-1.

The arrangement for the turbine island vent discharge radiation monitor is shown in Figure 11.5-1.

Liquid Radwaste Discharge Radiation Monitor

The liquid radwaste discharge radiation monitor (WLS-JE-RE229) measures the concentration of radioactive materials in liquids released to the environment. The liquid releases are made in batches that are mixed thoroughly and sampled. The samples are analyzed on site before discharge to determine that the discharge is within allowable concentration limits and within allowable totals.

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The liquid radwaste discharge radiation monitor provides data for reports of liquid releases of radioactive materials in accordance with Regulatory Guide 1.21.

The liquid radwaste discharge radiation monitor is an inline monitor that provides signals to isolate the discharge of liquid radwaste, stop the liquid radwaste system discharge pumps and alarms in the main control room if the concentrations exceed a predetermined setpoint. For process system details refer to Section 11.2.

The range and principal isotopes are listed in Table 11.5-1. The detector is a gamma-sensitive, thallium-activated, sodium iodide scintillation detector.

The arrangement for the liquid radwaste discharge radiation monitoring channel is shown in Figure 11.5-1.

Waste Water Discharge Radiation Monitor

The waste water discharge radiation monitor (WWS-JE-RE021) measures the concentration of radioactive materials in the discharge from the waste water system. The waste water discharge radiation monitor provides data for reports of liquid releases of radioactive materials in accordance with Regulatory Guide 1.21.

The waste water discharge radiation monitor is an inline monitor. It stops the turbine building sump pumps and initiates an alarm in the main control room if the concentration of radioactive materials exceeds a predetermined setpoint. Following an alarm, the operator can manually realign the discharge to the liquid radwaste system for processing. For process system details refer to Subsection 9.2.9.

The range and principal isotopes are listed in Table 11.5-1. The detector is a gamma-sensitive, thallium-activated, sodium iodide scintillation detector.

The arrangement for the waste water discharge radiation monitor is shown in Figure 11.5-1.

11.5.2.4 Inservice Inspection, Calibration, and Maintenance

The operability of each radiation monitoring system channel is checked periodically.

Test and inspection requirements for safety-related channels and certain nonsafety-related channels are provided in the Technical Specifications.

Daily checks of effluent monitoring system operability are made by observing channel behavior. Detector response is routinely observed with a remotely-positioned check source in accordance with plant procedures. Instrument background count rate is also observed to determine proper functioning of the monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely-positioned check source can have its response checked with a portable check source. A record is maintained showing the background radiation level and the detector response.

Calibration of the continuous radiation monitors is done with commercial radionuclide standards that have been standardized using a measurement system traceable to the National Institute of Standards and Technology.

11.5.3 Effluent Monitoring and Sampling

The primary means of quantitatively evaluating the isotopic activities in effluent paths is a program of sampling and onsite laboratory measurements. Gross activity measurements provided by the

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radiation monitors described in Subsection 11.5.2.3 are used to determine the activities released in effluent paths by calibrating the monitors against normalized laboratory results.

Sample points are located on the gaseous effluent radiation monitor skids.

The requirements of General Design Criterion 64 are satisfied by the sampling program and the effluent radiation monitors described in Subsection 11.5.2.3.

11.5.4 Process and Airborne Monitoring and Sampling

Radiation monitors are used to initiate automatic closure of isolation valves and dampers in liquid and gaseous process systems as described in Subsection 11.5.2.3. These radiation monitors address the requirement of General Design Criterion 60 to suitably control the release of radioactive materials in gaseous and liquid effluents. The sampling program for liquid and gaseous effluents will conform to Regulatory Guide 4.15, Revision 1 (See Appendix 1A).

Radiation monitors are used in the radioactive waste processing systems as described in Subsection 11.5.2.3. These radiation monitors address the requirement of General Design Criterion 63 to monitor radiation levels in radioactive waste systems.

Radiation monitors are used in the ventilation systems as described in Subsection 11.5.2.3 to ensure that airborne concentrations within the plant are within the limits of 10 CFR 20.

11.5.4.1 Effluent Sampling

Effluent sampling of potential radioactive liquid and gaseous effluent paths is conducted on a periodic basis to verify effluent processing meets the discharge limits to offsite areas. The effluent sampling program provides the information for the effluent measuring and reporting required by 10 CFR 50.36a and 10 CFR Part 20 and implemented through the Offsite Dose Calculation Manual (ODCM) and plant procedures. The frequency of the periodic sampling and analyses described herein are nominal and may be increased as permitted by procedure. Tables 11.5-201 and 11.5-202 summarize the sample and analysis schedules and sensitivities, respectively. The information contained in Tables 11.5-201 and 11.5-202 are derived from Regulatory Guide 1.21.

Laboratory isotopic analyses are performed on continuous and batch effluent releases in accordance with the ODCM. Results of these analyses are compiled and appropriate portions are utilized to produce the Radioactive Effluent Release Report.

11.5.4.2 Representative Sampling

Representative samples are obtained from well-mixed streams or volumes of effluent liquid through the use of proper sampling equipment, proper location of sampling points, and the development and use of sampling procedures. The recommendations of ANSI N 42.18 (Reference 203) are considered for the selection of instrumentation specific to the continuous monitoring of radioactivity in liquid effluents.

Sampling of effluent liquids is consistent with guidance in Regulatory Guide 1.21. When practical, effluent releases are batch-controlled, and prior to sampling, large volumes of liquid waste are mixed, in as short a time span as practicable, so that solid particulates are uniformly distributed in the liquid volume. Sampling and analysis is performed, and release conditions set, before release. Sample points are located to minimize flow disturbance due to fittings and other characteristics of equipment and components. Sample lines are flushed consistent with plant procedures to remove sediment deposits.

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Representative sampling of process effluents is attained through sample and monitor locations and methods and criteria detailed in plant procedures.

Composite sampling is employed to analyze for hard to measure radionuclides and to monitor effluent streams that normally are not expected to contain significant amounts of radioactive contamination. Composite liquid samples are collected in proportion to the volume of each batch of effluent release. The composite is thoroughly mixed prior to analysis. Collection periods for composites are as short as practicable and periodic checks are performed to identify changes in composite samples. When grab samples are collected instead of composite samples, the time of the sample, location, and frequency are considered to provide a representative sample of the radioactive materials.

The pressure head of the fluid, if available, is used for taking samples. If sufficient pressure head is not available to take samples, then sample pumps are used to draw the sample from the process fluid to the detector panels and back to the process.

Testing and obtaining representative samples using the radiation monitors described in Section 11.5 will be performed in accordance with ANSI N13.1 (Reference 201).

For obtaining representative samples in unfiltered ducts, sample probe configuration is tested and used in accordance with ANSI N13.1 (Reference 201).

Analytical Procedures

Typically, samples of process and effluent gases and liquids are analyzed in the station laboratory or by an outside laboratory via the following techniques:

- Gross alpha/beta counting
- Gamma spectrometry
- Liquid scintillation counting

"Available" instrumentation and counting techniques change as other instruments and techniques become available. For this reason, the frequency of sampling and the analysis of samples are generalized in this subsection.

Gross alpha/beta analysis may be performed directly on unprocessed samples (e.g., air filters) or on processed samples (e.g., evaporated liquid samples). Sample volume, counting geometry, and counting time are chosen to match measurement capability with sample activity. Correction factors for sample-detector geometry, self-absorption and counter resolving time are applied to provide the required accuracy.

Liquid effluent samples are prepared for alpha/beta counting by evaporation onto steel planchets. Gamma analysis may be done on any type of sample (gas, solid or liquid) in a gamma spectrometer.

Tritiated water vapor samples are collected by condensation or adsorption, and the resultant liquid is analyzed by liquid scintillation counting techniques.

Radiochemical separations are used for the routine analysis of Sr-89 and Sr-90.

Liquid samples are collected in polyethylene bottles to minimize absorption of nuclides onto container walls.

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11.5.5 Post-Accident Radiation Monitoring

The radiation monitors listed below meet the guidelines of Regulatory Guide 1.97 and are described in Subsections 11.5.2.3 and 11.5.6.2. For further Regulatory Guide 1.97 information refer to Appendix 1A and Section 7.5.

- Main steam line radiation monitors
- Steam generator blowdown radiation monitor
- Main control room supply air duct radiation monitors
- Plant vent radiation monitor
- Turbine island vent discharge radiation monitor
- Containment high range radiation monitors
- Primary sampling room area monitor
- CSA area monitor

The post-accident sampling system is described in Subsection 9.3.3 and is used to obtain samples for onsite laboratory analysis, including radioisotopic analysis, after a postulated accident.

11.5.6 Area Radiation Monitors

The area radiation monitors are provided to supplement the personnel and area radiation survey provisions of the AP1000 health physics program described in Section 12.5 and to comply with the personnel radiation protection guidelines of 10 CFR 20 and 10 CFR 50 and Regulatory Guides 1.97, 8.2, and 8.8.

During refueling operations in containment and the fuel handling area, radiation monitoring functions, as stated in 10 CFR 50.68(b)(6), are performed by the area radiation monitors in combination with portable bridge monitors.

11.5.6.1 Design Objectives

The design objectives of the area radiation monitors during normal operating plant conditions and anticipated operational occurrences are to:

- Measure the radiation intensities in specific areas of AP1000
- Warn of uncontrolled or inadvertent movement of radioactive material in AP1000
- Provide local and remote indication of ambient gamma radiation and local and remote alarms at key points where substantial changes in radiation flux might be of immediate importance to personnel
- Annunciate and warn of possible equipment malfunctions and leaks in specific areas of AP1000
- Furnish information for radiation surveys

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- Minimize the time, effort, and radiation received by operating personnel during routine maintenance and calibration
- Incorporate modular design concepts throughout, to provide easy maintenance

By meeting the above objectives, the radiation monitoring system aids health physics personnel in keeping radiation exposures as-low-as-reasonably-achievable (ALARA).

Locations of area monitor detectors are based on the following criteria:

- Area monitors are located in areas that are normally accessible and where changes in normal plant operating conditions can cause significant increases in exposure rates above those expected for the areas.
- Area monitors are located in areas that are normally or occasionally accessible where significant increases in exposure rates might occur because of operational transients or maintenance activities.
- Area monitors are located to best measure the increase in exposure rates within a specific area and to avoid shielding of the detector by equipment or structural materials.
- In the selection of area monitors, consideration is given to the environmental conditions under which the monitor operates.
- Area monitors are located to provide access so that minimal maintenance equipment is required and to provide an uncluttered area near the detector and local processing electronics to allow for field alignment and calibration.

The area radiation monitors are listed in Table 11.5-2.

11.5.6.2 Post-Accident Area Monitors

The following area monitors are provided to meet Regulatory Guide 1.97 guidelines as discussed in Appendix 1A and Section 7.5.

Containment High Range Radiation Monitor

The containment high range radiation monitors (PXS-JE-RE160, PXS-JE-RE161, PXS-JE-RE162, and PXS-JE-RE163) measure the radiation from the radioactive gases in the containment atmosphere. The monitors receive safety-related power. The detectors are ion chambers, designed to measure the radiation from the radioactive gases inside the containment in accordance with Regulatory Guide 1.97 and NUREG-0737. The monitors are qualified environmentally and seismically in accordance with the guidelines of Regulatory Guides 1.89 and 1.100, respectively.

The containment high range radiation data are displayed in the main control room. When predetermined setpoints are exceeded, the containment high range radiation monitors provide main control room alarms and signals to the protection and safety monitoring system for containment air filtration isolation and normal residual heat removal system valve closure (refer to Section 7.3 for further details). The containment high range radiation monitors provide data for maintaining a record of the gamma radiation intensities after a postulated accident as a function of time, so that the inventory of radioactive materials in the containment volume can be estimated.

The monitor range and type are listed in Table 11.5-2.

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The high range radiation detectors are mounted inside the containment on the containment wall in widely separated locations. The locations allow the detectors to be exposed to a significant volume of containment atmosphere without obstruction so that the readouts are representative of the containment atmosphere. The arrangement for a containment high range monitor is shown in Figure 11.5-2.

Primary Sampling Room Area Monitor

The primary sampling station is the location where samples are collected and/or analyzed after a postulated accident. The primary sampling room area radiation monitor (RMS-JE-RE008) is located so that its readout is representative of the radiation to which the operating personnel are exposed. A local readout, an audible alarm, and visual alarms are provided in the primary sampling room to alert operating personnel to increasing exposure rates. Indication and alarms are also provided in the main control room.

The monitor is an extended range monitor that uses a gamma-sensitive ion chamber and Geiger-Mueller tube. The monitor range and type are listed in Table 11.5-2.

Control Support Area (CSA) Area Monitor

The control support area is the location from which engineering support will be provided to the operators following a postulated accident. The CSA area radiation monitor (RMS-JE-RE016) is located so that its readout is representative of the radiation to which the support personnel are exposed. A local readout, an audible alarm, and visual alarms are provided locally to alert personnel to increasing exposure rates.

The monitor is a normal range monitor that uses a gamma-sensitive Geiger-Mueller tube. The monitor range and type are listed in Table 11.5-2.

11.5.6.3 Normal Range Area Monitors

Normal range area radiation monitors are located in accordance with the location criteria given in Subsection 11.5.6.1. A local readout, an audible alarm, and visual alarms are provided in each monitored area to alert operating personnel to increasing exposure rates. Indication and alarms are also provided in the main control room.

The monitor detectors are gamma-sensitive Geiger-Mueller tubes. The monitors and their ranges are listed in Table 11.5-2.

11.5.6.4 Fuel Handling Area Radiation Monitors

Radiation monitoring of the fuel handling and storage areas is performed in accordance with 10 CFR 50.68(b)(6) by radiation monitors RMS-JE-RE012 and RMS-JE-RE020. The area radiation monitoring is augmented during fuel handling operations by a portable radiation monitor on the machine handling fuel. The fuel handling area radiation monitor parameters are provided in Table 11.5-2. For additional information regarding radiation monitoring during initial fuel receipt and storage, see Section 12AA.5.

11.5.6.5 Quality Assurance

The quality assurance program for design, fabrication, procurement, and installation of the radiation monitoring system and radiation monitors from other systems is in accordance with the overall quality assurance program described in Chapter 17.

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The sampling program and the associated monitors conform to Regulatory Guide 4.15, Revision 1 (See Appendix 1A).

11.5.7 Preoperational Testing

Except as noted below, confirmation testing on the plant vent will be performed during plant startup to qualify the sample extraction location.

Velocity profile mapping at the sample extraction point will confirm that the velocity profile, including cyclonic flow, does not substantially affect flow mixing or sample nozzle performance, and is acceptable for obtaining a representative sample.

Performance testing with tracer gas and particulates will be performed over normal and selected off-normal flow conditions. Tracer gas and particulates testing will confirm an acceptably representative sample is obtained. The tracer gas and particulates testing can be performed on a scale model of the plant vent prior to preoperational testing.

The quantitative test acceptance criteria are dependent on the final design of the sampling system. The acceptance criteria will be established prior to testing and will be defined in the test procedures.

This set of confirmation tests will be performed for the first plant. For subsequent units, either these tests may be performed, or documentation may be used to justify that the plant vent geometry and the effluent flow conditions are the same or similar, and that these test results remain applicable.

11.5.8 Combined License Information

An Offsite Dose Calculation Manual (ODCM) is developed and implemented in accordance with the recommendations and guidance of NEI 07-09A (Reference 202). The ODCM contains the methodology and parameters used for calculating doses resulting from liquid and gaseous effluents. The ODCM addresses operational setpoints, including planned discharge rates, for radiation monitors and monitoring programs (process and effluent monitoring and environmental monitoring) for the control and assessment of the release of radioactive material to the environment. The ODCM provides the limitations on operation of the radwaste systems, including functional capability of monitoring instruments, concentrations of effluents, sampling, analysis, 10 CFR Part 50, Appendix I dose and dose commitments, and reporting. The ODCM will be finalized prior to fuel load with site-specific information.

Table 13.4-201 provides milestones for ODCM implementation.

The process and effluent monitoring and sampling per ANSI N13.1 and Regulatory Guides 1.21 and 4.15 is addressed in Subsections 11.5.1.2, 11.5.2.4, 11.5.3, 11.5.4, 11.5.4.1, 11.5.4.2, and 11.5.6.5.

The Combined License applicant is responsible for addressing the 10 CFR 50, Appendix I guidelines for maximally exposed offsite individual doses and population doses via liquid and gaseous effluents.

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11.5.9 References

201. ANSI N13.1-1969, “Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.”

202. NEI 07-09A, “Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description,” Revision 0, March 2009 (ML091050234).

203. ANSI N42.18-2004, “Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents.”

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Table 11.5-1 (Sheet 1 of 2)
Radiation Monitor Detector Parameters

Detector	Type	Service	Isotopes	Nominal Range
BDS-JE-RE010	γ	Steam Generator Blowdown Discharge Radiation (Note 6)	Cs-137	1.0E-7 to 1.0E-2 µCi/cc
BDS-JE-RE011	γ	Steam Generator Blowdown Electrodeionization Reject (Note 6)	Cs-137	1.0E-7 to 1.0E-2 µCi/cc
CCS-JE-RE001	γ	Component Cooling Water System	Cs-137	1.0E-7 to 1.0E-2 µCi/cc
VFS-JE-RE101	β	Plant Vent Particulate	Sr-90 Cs-137	1.0E-12 to 1.0E-7 µCi/cc
VFS-JE-RE102	γ	Plant Vent Iodine	I-131	1.0E-11 to 1.0E-6 µCi/cc
VFS-JE-RE103	β	Plant Vent Gas (Normal Range)	Kr-85 Xe-133	1.0E-7 to 1.0E-2 µCi/cc
VFS-JE-RE104A	β/γ	P.V. Extended Range Gas (Accident Mid Range) (Note 6)	Kr-85 Xe-133	1.0E-4 to 1.0E+2 µCi/cc
VFS-JE-RE104B	β/γ	P.V. Extended Range Gas (Accident High Range) (Note 6)	Kr-85 Xe-133	1.0E-1 to 1.0E+5 µCi/cc
PSS-JE-RE026	β	Containment Atmosphere Gas (Note 2)	Kr-85 Xe-133 Ar-41 N-13	1.0E-6 to 1.0E-1 µCi/cc
PSS-JE-RE027	β	Containment Atmosphere beta-sensitive scintillation detector (Note 2)	F18	1.0E-10 to 1.0E-5 µCi/cc
PSS-JE-RE050	γ	Primary Sampling Liquid	I-131 Cs-137	1.0E-1 to 1.0E+4 µCi/cc
PSS-JE-RE052	γ	Primary Sampling Gaseous	Kr-85 Xe-133	1.0E-1 to 1.0E+4 mR/hr
SGS-JE-RE026A	γ	Main Steam Line (Note 6)	Kr, Xe, I	1.0E-1 to 1.0E+3 µCi/cc
SGS-JE-RE026B	γ	Main Steam Line	N-16	30 to 200 gallons per day
SGS-JE-RE027A	γ	Main Steam Line (Note 6)	Kr, Xe, I	1.0E-1 to 1.0E+3 µCi/cc
SGS-JE-RE027B	γ	Main Steam Line	N-16	30 to 200 gallons per day
SWS-JE-RE008	γ	Service Water Blowdown	Cs-137	1.0E-7 to 1.0E-2 µCi/cc
TDS-JE-RE001A/B	ß/γ	Turbine Island Vent Discharge (Note 3) (Note 6)	Kr-85 Xe-133	1.0E-6 to 1.0E+5 µCi/cc (Note 4)
VAS-JE-RE001	β	Fuel Handling Area Exhaust (Note 5)	Kr-85 Xe-133	1.0E-6 to 3.0E-2 µCi/cc
VAS-JE-RE002	β	Auxiliary Building Exhaust (Note 5)	Kr-85 Xe-133	1.0E-6 to 3.0E-2 µCi/cc
VAS-JE-RE003	β	Auxiliary Building Exhaust (Note 5)	Kr-85 Xe-133	1.0E-6 to 3.0E-2 µCi/cc

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Table 11.5-1 (Sheet 2 of 2)
Radiation Monitor Detector Parameters

Detector	Type	Service	Isotopes	Nominal Range
VAS-JE-RE008	β	Annex Building Exhaust (Note 5)	Kr-85 Xe-133	1.0E-6 to 3.0E-2 µCi/cc
VBS-JE-RE001A	β	Main Control Room Supply Air Duct (Particulate) (Note 1) (Note 5) (Note 6)	Sr-90 Cs-137	1.0E-12 to 1.0E-7 µCi/cc
VBS-JE-RE001B	β	Main Control Room Supply Air Duct (Particulate) (Note 1) (Note 5) (Note 6)	Sr-90 Cs-137	1.0E-12 to 1.0E-7 µCi/cc
VBS-JE-RE002A	γ	MCR Supply Air Duct (Iodine) (Note 1) (Note 5) (Note 6)	I-131	1.0E-11 to 1.0E-5 µCi/cc
VBS-JE-RE002B	γ	MCR Supply Air Duct (Iodine) (Note 1) (Note 5) (Note 6)	I-131	1.0E-11 to 1.0E-5 µCi/cc
VBS-JE-RE003A	β	MCR Supply Air Duct (Gas) (Note 1) (Note 5) (Note 6)	Kr-85 Xe-133	1.0E-7 to 1.0E-1 µCi/cc
VBS-JE-RE003B	β	MCR Supply Air Duct (Gas) (Note 1) (Note 5) (Note 6)	Kr-85 Xe-133	1.0E-7 to 1.0E-1 µCi/cc
VFS-JE-RE001	β	Containment Air Filtration Exhaust (Note 5)	Kr-85 Xe-133	5.0E-7 to 3.0E-2 µCi/cc
VHS-JE-RE001	β	H.P. & Hot Machine Shop Exhaust (Note 5)	Sr-90 Cs-137	1.0E-12 to 1.0E-7 µCi/cc
VRS-JE-RE023	β	Radwaste Building Exhaust (Note 5)	Sr-90 Cs-137	1.0E-12 to 1.0E-7 µCi/cc
WGS-JE-RE017	γ	Gaseous Radwaste Discharge	Kr-85 Xe-133	1.0E-4 to 1.0E+2 µCi/cc
WLS-JE-RE229	γ	Liquid Radwaste Discharge	Cs-137	1.0E-7 to 1.0E-1 µCi/cc
WWS-JE-RE021	γ	Waste Water Discharge	Cs-137	1.0E-7 to 1.0E-2 µCi/cc

Notes:
1. Safety-related
2. Seismic Category I
3. The condenser air removal system (CMS) and the gland seal system (GSS) discharge into the turbine island vents, drains and relief system (TDS). The exhaust from the TDS into the turbine island vent is continuously monitored for radiation.
4. Turbine island vent radiation monitor includes two G-M tubes with nominal ranges of 1.0E-6 to 1.0E+0 µci/cc and 1.0E-1 to 1.0E+5 µci/cc.
5. Monitor is sensitive enough to detect 10 Derived Air Concentration (DAC)-hours.
6. These detectors are for post-accident monitoring; refer to Table 7.5-1.

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Table 11.5-2
Area Radiation Monitor Detector Parameters

Detector	Type	Service	Nominal Range
PXS-JE-RE160	γ	Containment High Range (Note 3)	1.0E-0 to 1.0E+7 R/hr
PXS-JE-RE161	γ	Containment High Range (Note 3)	1.0E-0 to 1.0E+7 R/hr
PXS-JE-RE162	γ	Containment High Range (Note 3)	1.0E-0 to 1.0E+7 R/hr
PXS-JE-RE163	γ	Containment High Range (Note 3)	1.0E-0 to 1.0E+7 R/hr
RMS-JE-RE008	γ	Primary Sampling Room	1.0E-1 to 1.0E+7 mR/hr
RMS-JE-RE009	γ	Containment Area Upper Personnel Hatch – Operating Deck	1.0E-1 to 1.0E+4 mR/hr (Note 1)
RMS-JE-RE010	γ	Main Control Room	1.0E-1 to 1.0E+4 mR/hr
RMS-JE-RE011	γ	Chemistry Laboratory Area	1.0E-1 to 1.0E+4 mR/hr
RMS-JE-RE012	γ	Fuel Handling Area	1.0E-1 to 1.0E+4 mR/hr (Note 2)
RMS-JE-RE013	γ	Rail Car Bay/Filter Storage Area (Note 4)	1.0E-1 to 1.0E+4 mR/hr
RMS-JE-RE014	γ	Liquid and Gaseous Radwaste Area	1.0E-1 to 1.0E+4 mR/hr
RMS-JE-RE016	γ	CSA Area	1.0E-1 to 1.0E+4 mR/hr
RMS-JE-RE017	γ	Radwaste Bldg. Mobile Systems Facility (Note 4)	1.0E-1 to 1.0E+4 mR/hr
RMS-JE-RE018	γ	Hot Machine Shop	1.0E-1 to 1.0E+4 mR/hr
RMS-JE-RE019	γ	Annex Staging & Storage Area	1.0E-1 to 1.0E+4 mR.hr
RMS-JE-RE020	γ	Fuel Handling Area	1.0E-1 to 1.0E+4 mR/hr (Note 2)
RMS-JE-RE021	γ	Containment Area Lower Personnel Hatch – Maintenance Level	1.0E-1 to 1.0E+04 mR/hr (Note 1)

Notes:

1. Radiation levels are monitored by the permanent containment area radiation monitor and by a portable bridge monitor during refueling operations. The containment area radiation monitor is located to best measure the increase in exposure rates for this area and to provide an alarm locally and in the main control room.
2. Radiation levels are monitored by the permanent fuel handling area radiation monitors and by a portable bridge monitor during fuel handling operations. The fuel handling area radiation monitors are located to best measure the increase in exposure rates for this area and to provide an alarm locally and in the main control room.
3. Safety-related
4. Monitors areas used for storage of wet wastes (including processed and packaged spent resins) and dry wastes.

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Table 11.5-201 (Sheet 1 of 2)
Minimum Sampling Frequency

Stream	Sampled Medium	Frequency
Gaseous	Continuous Release	A sample is taken within one month of initial criticality, and at least weekly thereafter to determine the identity and quantity for principal nuclides being released. A similar analysis of samples is performed following each refueling, process change, or other occurrence that could alter the mixture of radionuclides. When continuous monitoring shows an unexplained variance from an established norm. Monthly for tritium.
	Batch Release	Prior to release to determine the identity and quantity of the principal radionuclides (including tritium).
	Filters (particulates)	Weekly. Quarterly for Sr-89 and Sr-90. Monthly for gross alpha.

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Table 11.5-201 (Sheet 2 of 2)
Minimum Sampling Frequency

Stream	Sampled Medium	Frequency
Liquid	Continuous Releases	Weekly for principal gamma-emitting radionuclides.
		Monthly, a composite sample for tritium and gross alpha.
		Monthly, a representative sample for dissolved and entrained fission and activation gases.
		Quarterly, a composite sample for Sr-89, Sr-90, and Fe-55.
	Batch Releases	Prior to release for principal gamma-emitting radionuclides.
		Monthly, a composite sample for tritium and gross alpha.
		Monthly, a representative sample from at least one representative batch for dissolved and entrained fission and activation gases.
		Quarterly, a composite sample for Sr-89, Sr-90 and Fe-55.

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Table 11.5-202
Minimum Sensitivities

Stream	Nuclide	Sensitivity
Gaseous	Fission & Activation Gases	1.0E-04 μCi/cc
	Tritium	1.0E-06 μCi/cc
	Iodines & Particulates	Sufficient to permit measurement of a small fraction of the activity that would result in annual exposures of 15 mrem to thyroid for iodines, and 15 mrem to any organ for particulates, to an individual in an unrestricted area.
	Gross Radioactivity	Sufficient to permit measurement of a small fraction of the activity that would result in annual air dose of 1) 10 mrad due to gamma, and 2) 20 mrad of beta at any location near ground level at or beyond the site boundary.
Liquid	Gross Radioactivity	1.0E-07 μCi/ml
	Gamma-emitters	5.0E-07 μCi/ml
	Dissolved & Entrained Gases	1.0E-05 μCi/ml
	Gross Alpha	1.0E-07 μCi/ml
	Tritium	1.0E-05 μCi/ml
	Sr-89 & Sr-90	5.0E-08 μCi/ml
	Fe-55	1.0E-06 μCi/ml

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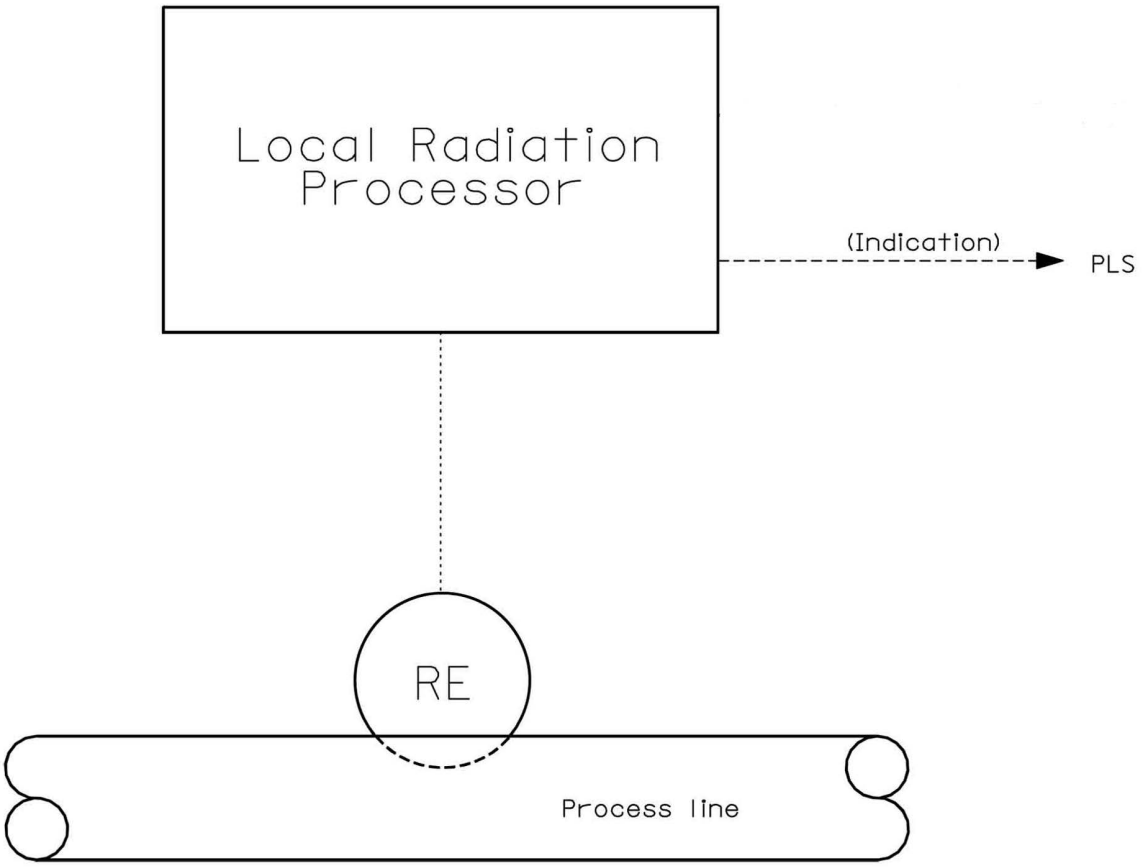

Figure 11.5-1
Process In-Line Radiation Monitor

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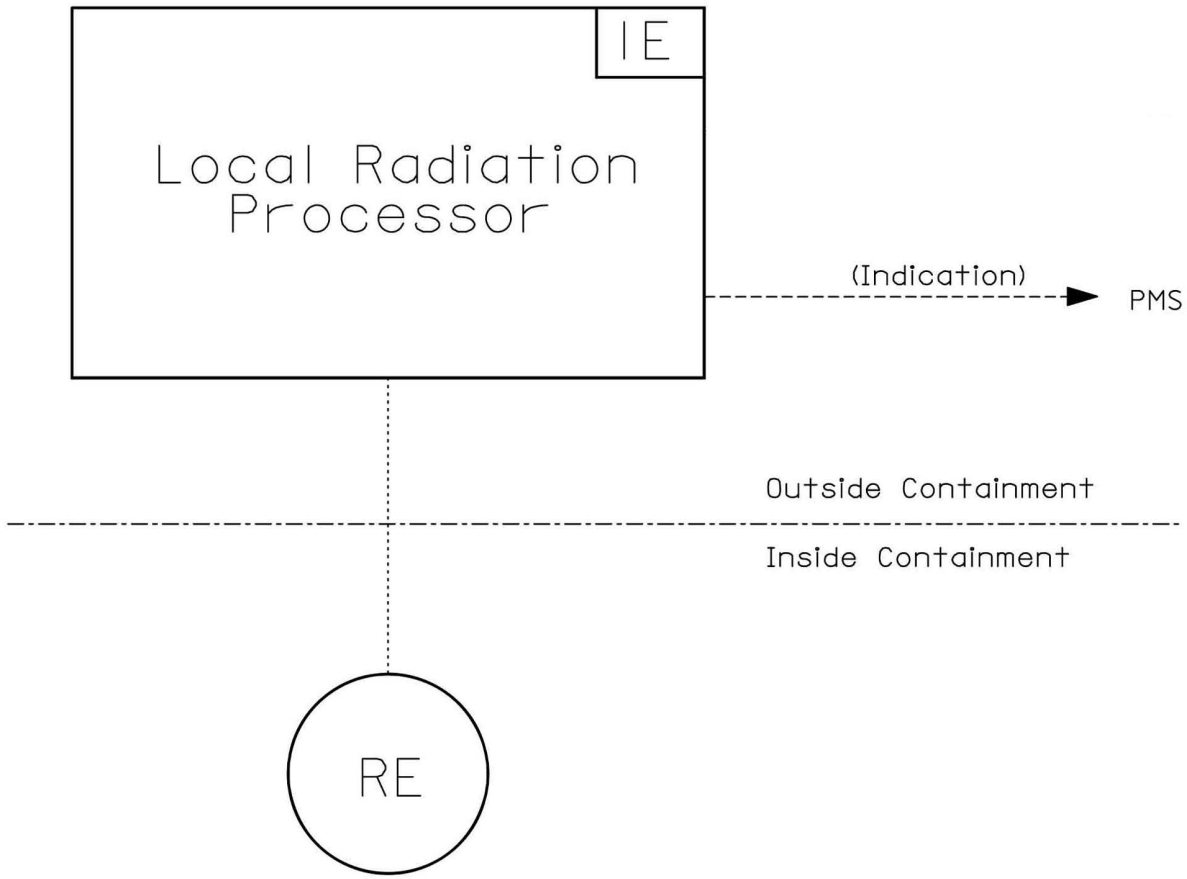

Figure 11.5-2
Safety-Related Containment High Range Radiation Monitor

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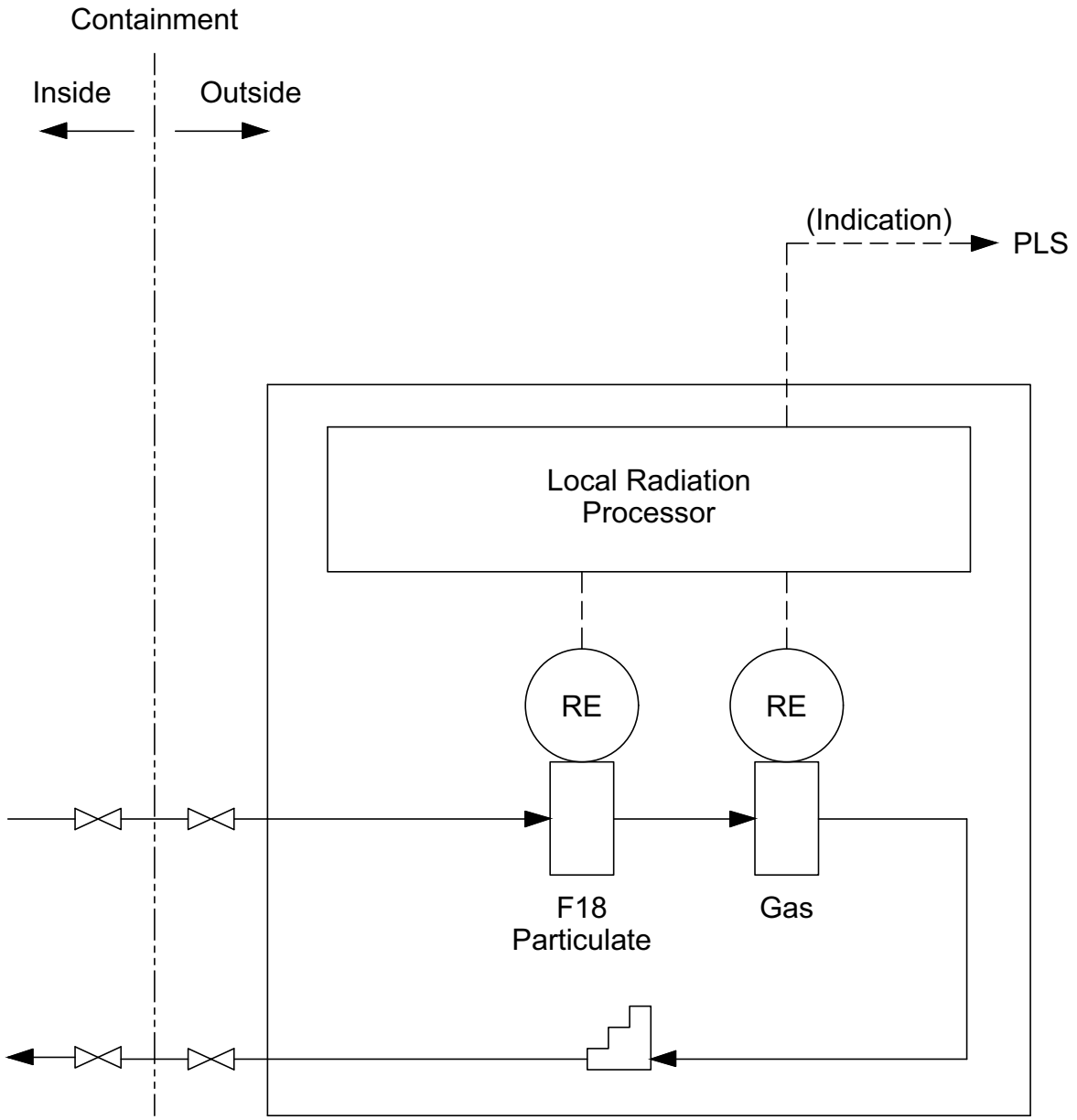

Figure 11.5-3
Containment Atmosphere Radiation Monitor

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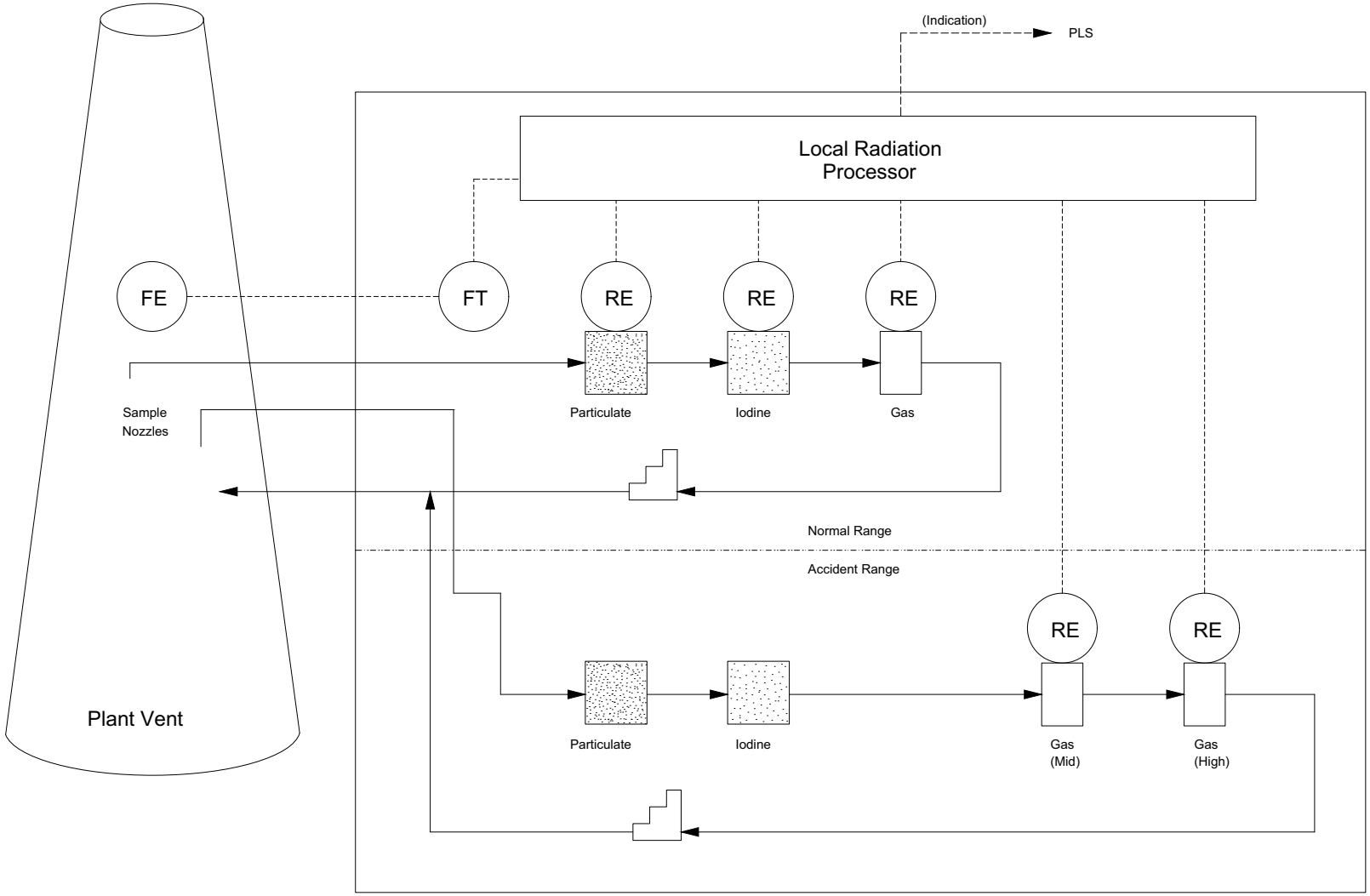

Figure 11.5-4
Plant Vent Radiation Monitor

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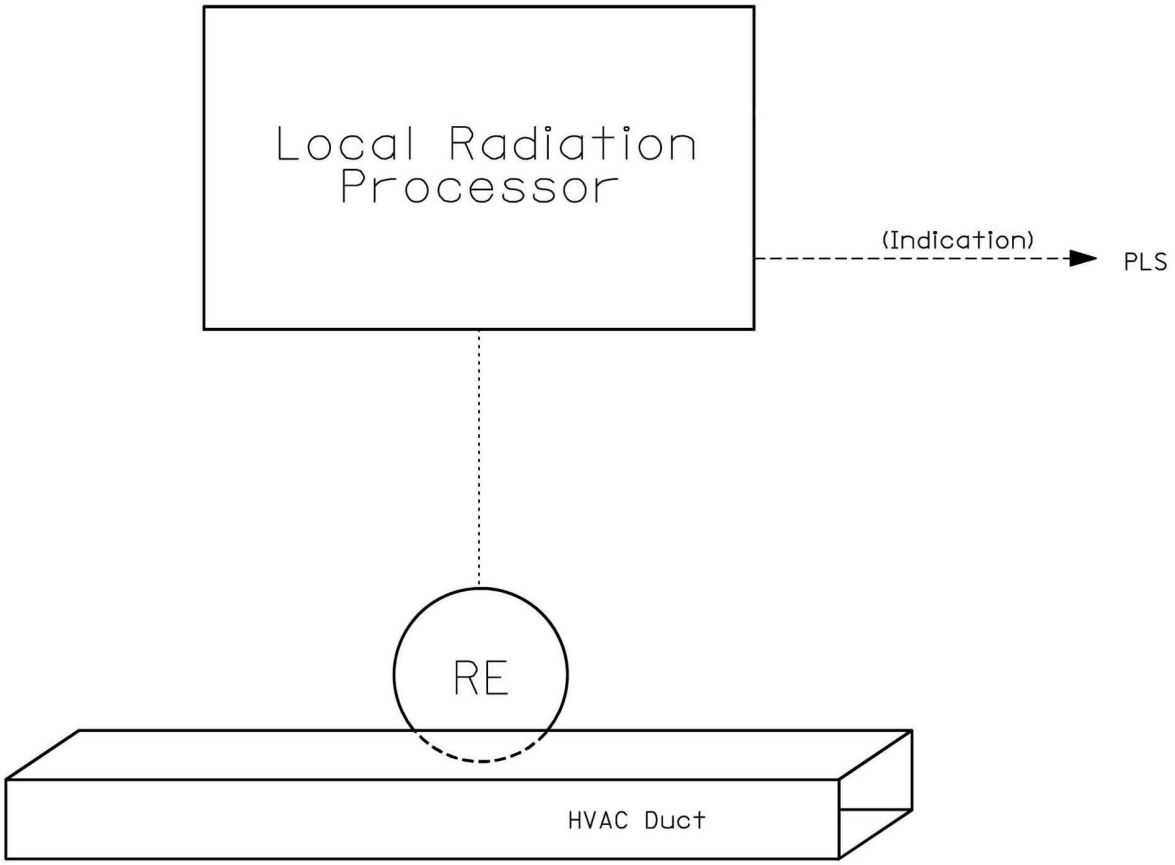

Figure 11.5-5
In-Line HVAC Duct Radiation Monitor

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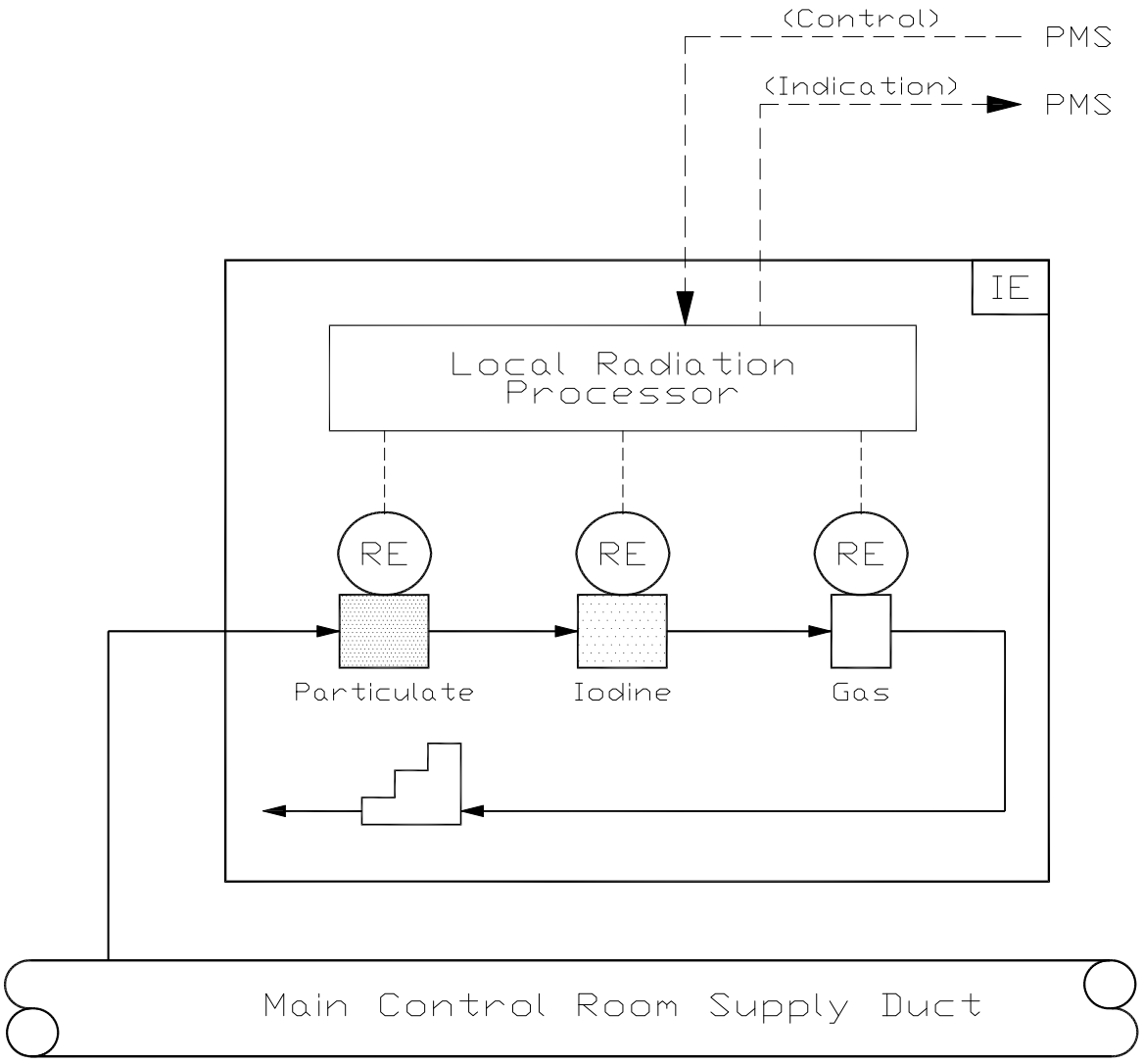

Figure 11.5-6
Safety-Related Main Control Room Supply Duct Radiation Monitor

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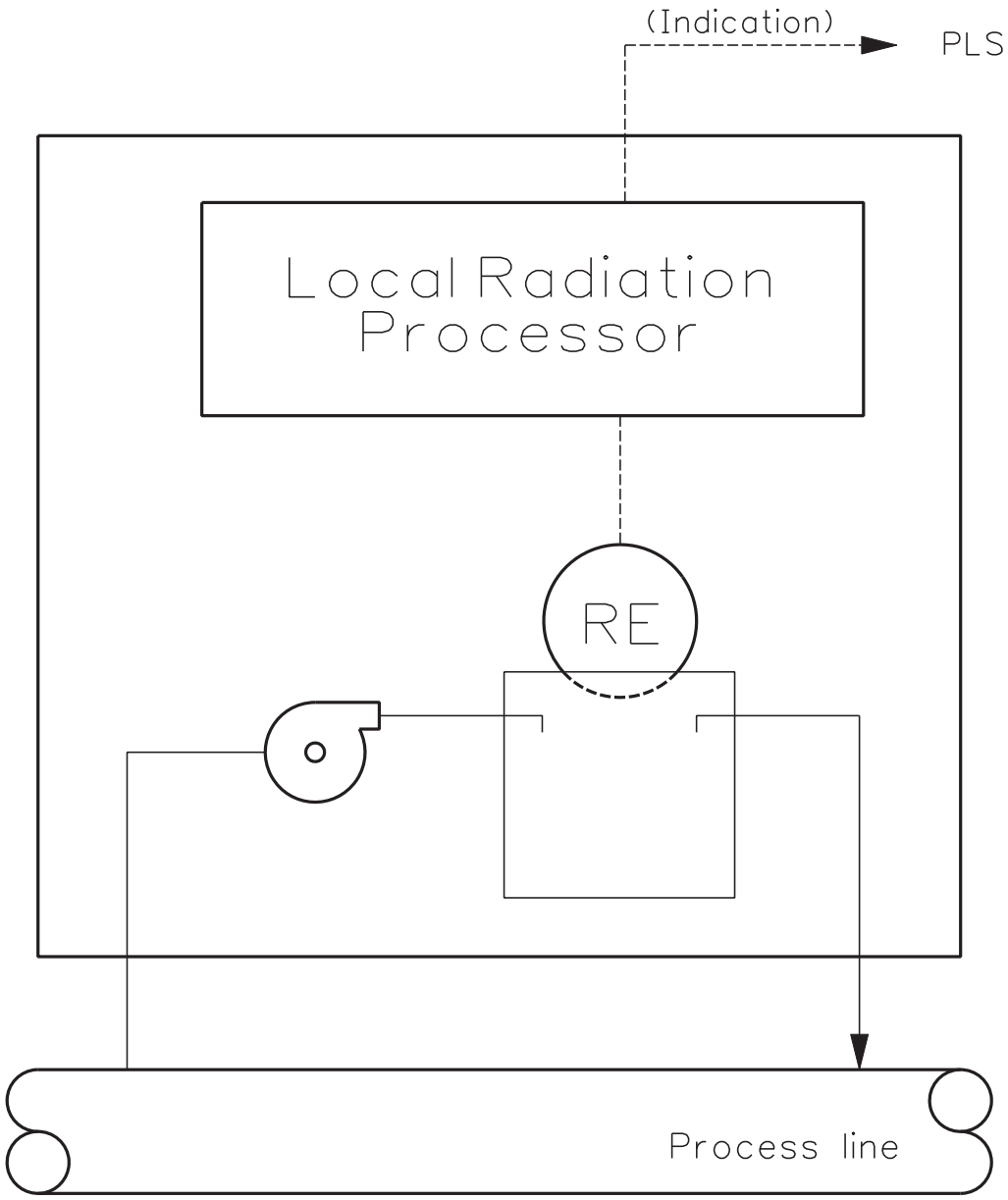

Figure 11.5-7
Liquid Offline Radiation Monitor

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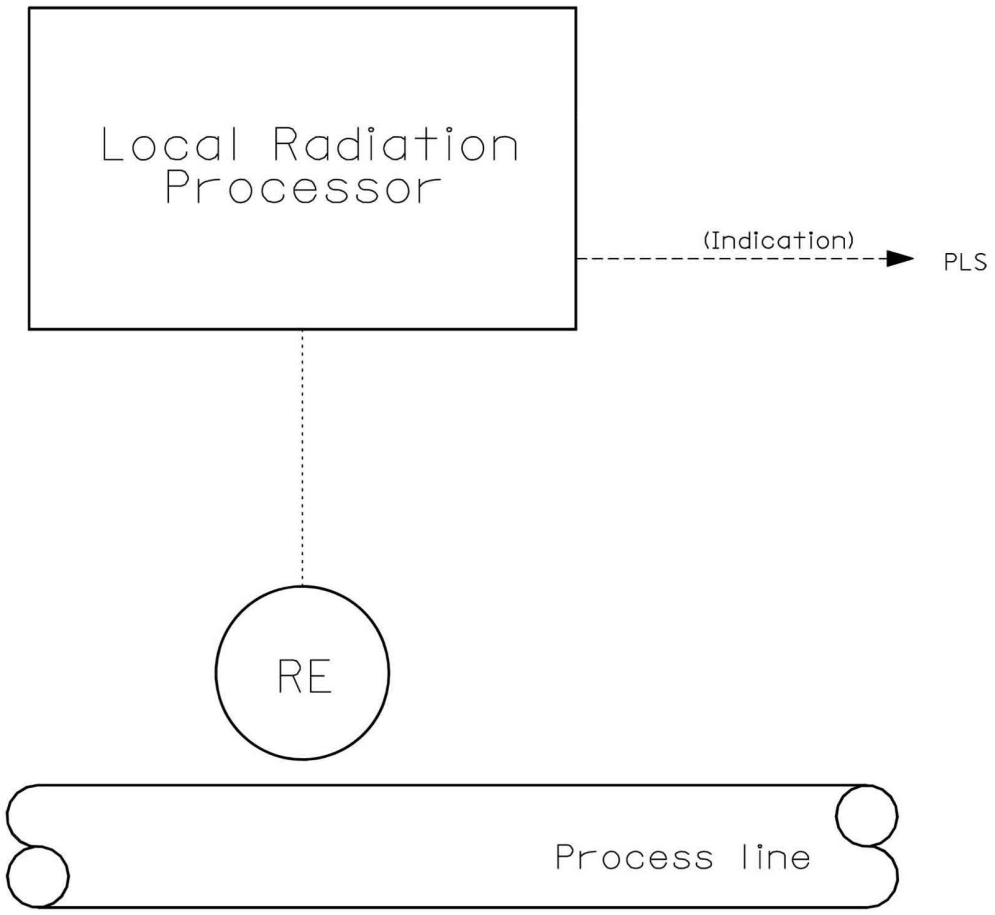

Figure 11.5-8
Adjacent to Line Radiation Monitor

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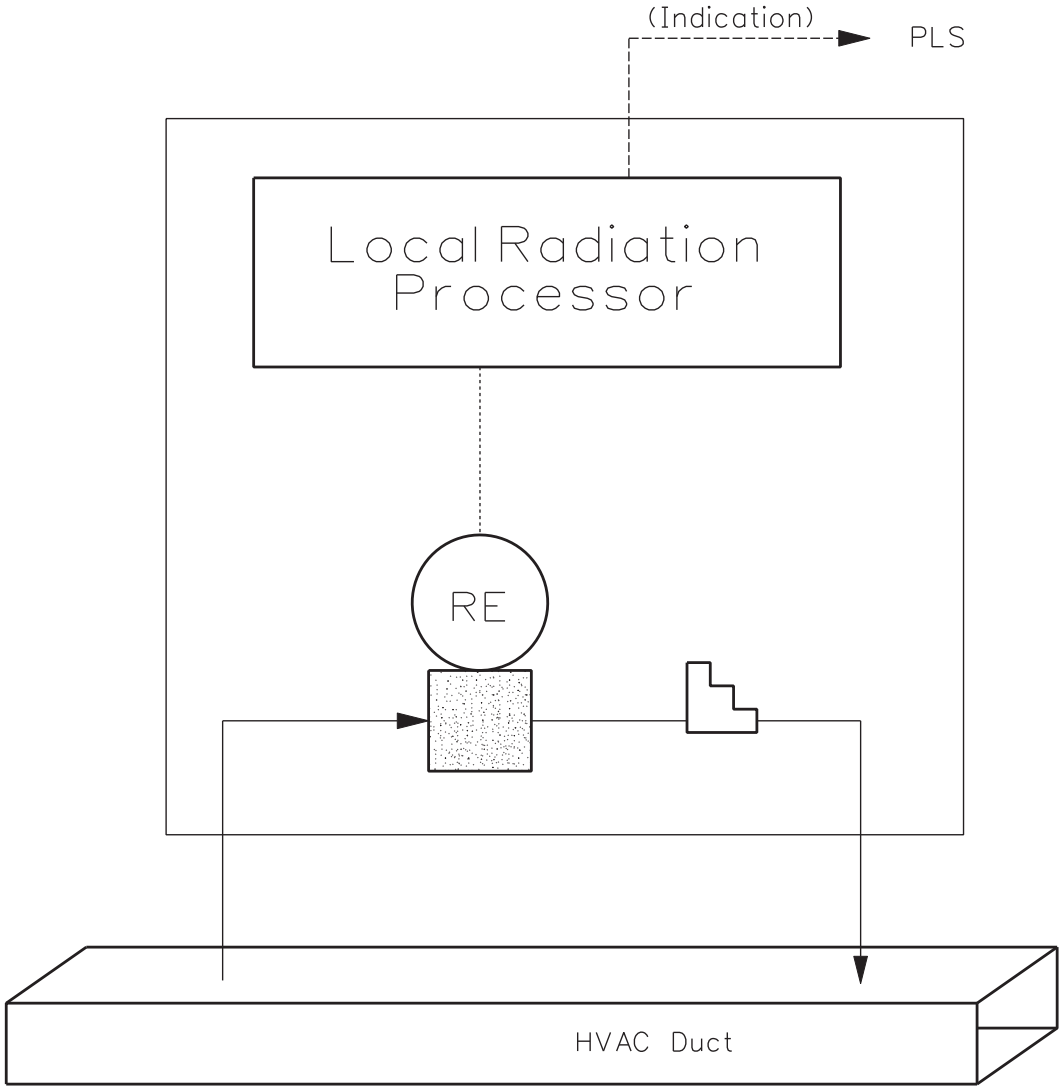

Figure 11.5-9
HVAC Duct Particulate Radiation Monitor

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Chapter 12 Radiation Protection

12.1 Assuring that Occupational Radiation Exposures Are As-Low-As-Reasonably Achievable (ALARA)

This section incorporates NEI 07-08A, Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA). An exception is taken to the annual ALARA retraining frequency defined in NEI 07-08A Subsection 12.1.1. ALARA retraining frequency is defined in Subsection 12.1.1.2. ALARA practices are developed in a phased milestone approach as part of the procedures necessary to support the Radiation Protection Program. Table 13.4-201 describes the major milestones for ALARA procedures development and implementation.

12.1.1 Policy Considerations

The AP1000 plant is designed with administrative programs and procedures to maximize the incorporation of good engineering practices and lessons learned to accomplish ALARA objectives.

12.1.1.1 Design and Construction Policies

The ALARA policy is applied during the design of AP1000. The design is reviewed for ALARA considerations and updated and modified as experience from operating plants is applied. ALARA reviews include the plant design and integrated layout, considering shielding, ventilation, and monitoring instrument designs as they relate to traffic control, security, access control and health physics.

Similarly, routing of pipe containing radioactive fluids is reviewed as part of the design effort. This confirms that lines expected to contain significant radiation sources are adequately shielded and properly routed to minimize exposure of personnel.

Many of the engineers and supervisors assigned to the AP1000 design have performed similar design work or service work on other nuclear power plants. Through this experience, they have acquired knowledge of the radiation protection aspects which are applied to AP1000. Nuclear plant operating experience is incorporated through Nuclear Regulatory Commission (NRC) inspection and enforcement bulletins, information notices, and other documents. Independent reviews are conducted by the Electric Power Research Institute (EPRI) and Utility Steering Committee and its subcommittees. Knowledge of radiation protection and ALARA is applied to AP1000 design. This allows integration of experience and ALARA considerations from plant operators and plant designers and promotes incorporation of recent operating and service experience and lessons learned.

12.1.1.2 Operation Policies

Company and station policies are to keep all radiation exposure of personnel within limits defined by 10 CFR 20, Standards for Protection Against Radiation (Reference 12.1-1). Administrative procedures and practices related to maintaining radiation exposure of personnel as low as is reasonable achievable (ALARA) are described in this section.

The ALARA policy is consistent with the guidelines of Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As is Reasonably Achievable* (Reference 12.1-7), and Regulatory Guide 8.10, *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable* (Reference 12.1-9) in establishing, organizing, and operating an effective ALARA program.

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To varying extents, all station personnel are responsible for ALARA. Each supervisor is responsible for enforcing the ALARA requirements as described in 10CFR 20.1101. Individual workers are responsible for complying with ALARA requirements, which are presented during initial plant training and reinforced through retraining in accordance with the requirements contained in 10 CFR 19.12. Retraining frequency is established by a Systematic Approach to Training analysis at a frequency of every 48 months nominal not to exceed 51 months. The extent of ALARA training provided for each person is at least commensurate with the worker's job responsibilities and plant areas frequented. The radiation protection training program is maintained and implemented by the training department.

To ensure that personnel comply with established radiological policies, procedures and practices, radiation protection management personnel are charged with the responsibility to promptly advise higher management of any radiologically unsafe practices which exceed their authority to correct. They have the authority to halt any operation which, in their judgment, is radiologically unsafe. Radiation protection technicians are responsible for notifying the operations shift supervision or radiation protection management immediately in order to stop work on any operation deemed to be radiologically unsafe.

12.1.1.3 Compliance with 10 CFR 20 and Regulatory Guides 1.8, 8.8, and 8.10

Compliance of the design with 10 CFR 20 is confirmed by compliance of the design and operation of the facility within the guidelines of Regulatory Guides 1.8, 8.8, and 8.10. Compliance with Regulatory Guides 1.8, 8.8, and 8.10 is addressed as discussed in Appendix 1A and below.

The design of AP1000 meets the guidelines of Regulatory Guide 8.8, Sections C.2 and C.4, which address facility, equipment and instrumentation design features. Features of the plant that are examples of compliance with Regulatory Guide 8.8 are delineated in Section 12.3.

As further discussed in Section 12.5, the station's ALARA policies and practices are consistent with the applicable regulations in 10 CFR 20 (Reference 201). Compliance with the guidance found in Regulatory Guides 1.8, 1.206, 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38 is discussed in Appendix 1A and Section 12AA.5. Compliance with the applicable portions of NUREG-1736 (References 202 and 204 through 219) is discussed in Section 12AA.5, Radiation Protection Program. Compliance with Regulatory Guide 1.8 is further discussed in Section 13.2, Training. ALARA procedures are established, implemented, maintained and reviewed consistent with 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description, which is discussed in Section 17.5.

12.1.2 Design Considerations

Provisions and designs for maintaining personnel exposures ALARA are presented in the following paragraphs. The basic management philosophy guiding the AP1000 design effort so that radiation exposures are ALARA can be expressed as:

- Design structures, systems and components for reliability and maintainability, thereby effectively reducing the maintenance requirements on radioactive components.

- Design structures, systems and components to reduce the radiation fields, thereby allowing operation, maintenance and inspection activities to be performed in the minimum design radiation field.

- Design structures, systems and components to reduce access, repair and removal times, thereby effectively reducing the time spent in radiation fields during operation, maintenance, and inspection.

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- Design structures, systems and components to accommodate remote and semi-remote operation, maintenance and inspection, thereby effectively reducing the time spent in radiation fields.

12.1.2.1 General Design Considerations for ALARA Exposures

General design considerations and methods to maintain in-plant radiation exposures ALARA consistent with the recommendations of Regulatory Guide 8.8 have two objectives:

- Minimizing the necessity for access to and personnel time spent in radiation areas
- Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention

Equipment and facility layouts and designs are considered for maintaining exposures ALARA during plant operations, including:

- Normal operation
- Maintenance and repairs
- Refueling operations and fuel storage
- Inservice inspection and calibrations
- Radioactive waste handling and disposal
- Other anticipated operational occurrences
- Decommissioning

The actual design features are described in Section 12.3. Examples of features that assist in maintaining exposures ALARA include:

- Provision of features to allow maintenance of state-of-the-art reactor coolant chemistry conditions, such that corrosion and consequential source terms are minimized: these include pH control capability sufficient to meet current and evolving industry standards and the ability to add zinc to the primary coolant
- Provision of features to allow draining, flushing, and decontaminating equipment and piping
- Design of equipment to minimize the creation and buildup of radioactive material and to ease flushing of crud traps
- Provision of shielding for personnel protection during maintenance or repairs and during decommissioning
- Provision of means and adequate space for the use of movable shielding
- Separation of more highly radioactive equipment from less radioactive equipment and provision of separate shielded compartments for adjacent items of radioactive equipment
- Provision of shielded access hatches for installation and removal of plant components
- Provision of design features, such as the chemical and volume control system, to minimize crud buildup
- Provision for means and adequate space for the use of remote and robotic maintenance and inspection equipment

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- Simplifying the plant design compared to previous pressurized water reactors with design approaches such as:

  – Elimination of boron recycle
  – Elimination of evaporators
  – Use of an extended fuel cycle
  – Reduction in components containing radioactive fluids
  – Clearly and deliberately separating clean areas from potentially radioactive ones

12.1.2.2 Equipment General Design Considerations for ALARA

Equipment design considerations to minimize the necessity for, and amount of, time spent in a radiation area generally include:

- Reliability, durability, constructibility, and design features of equipment, components, and materials to reduce or eliminate the need for repair or preventive maintenance.

- Servicing convenience for anticipated maintenance or potential repair, including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair (For example, the passive residual heat removal heat exchanger is designed with extra tubes to allow for plugging of some tubes. Heat exchangers have drains to allow draining of the shell side water.)

- Provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment.

- Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high and when there is no feasible method available to reduce radiation levels.

- Provisions for equipment to be operated from, and have its instrumentation and control in, accessible areas both during normal and abnormal operating conditions.

- Provisions for remote operation, draining and flushing of systems such as the chemical and volume control system.

- Past experience and lessons learned from servicing currently operating nuclear power plants.

Equipment design considerations directed toward minimizing radiation levels near equipment or components requiring personnel attention include:

- Selection of materials that minimize the creation of radioactive contamination.

- Provision of equipment and piping designs that minimize the accumulation of radioactive materials (for example, the use of seamless piping and minimizing the number of fittings reduces radiation accumulation at the seams and welds).

- Provisions for draining, flushing, or if necessary, remote cleaning or decontamination of equipment containing radioactive materials.

- Provision in the design for limiting leaks or controlling the fluid that does leak. This includes the use of high quality valves and valve packings, and the direction of leakage via drip pans and piping to sumps and floor drains.

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- Provisions for isolating equipment from radioactive process fluids.

- Provisions for the chemical and volume control system; the spent fuel pool cleanup system; and the liquid radwaste cleanup system to limit radioactive isotope levels in the process water.

12.1.2.3 Facility Layout General Design Considerations for ALARA

Facility design considerations to minimize the amount of personnel time spent in a radiation area include the following:

- Locating equipment, instruments, and sampling stations that require routine maintenance, calibration, operation, or inspection, in a manner that promotes ease of access and minimum of required occupancy time in radiation areas

- Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment

- Providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area

Facility design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention generally include the following:

- Separating radiation sources and occupied areas, where practicable (for example, pipes or ducts containing potentially highly radioactive fluids do not pass through occupied areas). Redundant components requiring periodic maintenance that are a source of radiation are located in separate compartments to allow maintenance of one component while the other component is in operation.

- Providing shielding to separate equipment such as demineralizers and filters from nonradioactive equipment to provide unrestricted maintenance on the nonradioactive equipment.

- Providing shielding between radiation sources and access and service areas.

- Providing labyrinth entrances to radioactive pump, equipment, and valve rooms. Adequate space is provided in labyrinth entrances for easy access. Highly radioactive passive components with minimal maintenance requirements are located in completely enclosed compartments and are provided with access via a shielded hatch or removable blocks.

- Separating equipment or components in service areas with permanent shielding, where appropriate.

- Providing means and adequate space for using movable shielding for sources within the service area, when required.

- Incorporating, within the plant layout, restrictions and control of access to the various radiation zones. Access to a given radiation zone generally does not require passing through a higher radiation zone. In the case of an abnormal occurrence or accident, the zone restrictions may change due to increased dose rates. Special access controls would be implemented at that time as discussed in Appendix 12AA.

- Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone.

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- Providing control panels to permit remote operation of essential instrumentation and controls from the lowest radiation zone practicable.
- Providing means to control contamination or facilitate decontamination of potentially contaminated areas.
- Providing means for decontamination of service areas.
- Maintaining ventilation air flow patterns from areas of lower radioactivity to areas of higher radioactivity.
- Provide adequate lighting and support services (electrical power, compressed air, demineralized water, ventilation, and communications) at workstations.

12.1.2.4 Equipment and Facility Layout General Design Considerations for 10 CFR 20.1406

General equipment and facility layout design considerations to prevent the spread of contamination and to facilitate eventual decommissioning in accordance with 10 CFR 20.1406 include the features discussed in the following subsections.

12.1.2.4.1 Piping

The use of embedded pipes is minimized to the extent possible, consistent with maintaining radiation doses ALARA.

To the extent possible, radioactive piping is located inside the auxiliary building and the containment vessel. This minimizes the potential for leakage to the groundwater from piping or fittings. The few exceptions are as follows:

- Process piping to and from the radwaste building (which can be fully visually inspected from the radwaste building pipe trench to the auxiliary building wall).
- Drain lines from the radwaste building and annex building back to the auxiliary building. These lines are not normally water filled, and can also be fully visually inspected from the annex or radwaste building pipe trench to the auxiliary building wall.
- Piping associated with the waste monitor tanks in the radwaste building. These tanks contain processed water, and they are located within the curbed radwaste building, which drains to the liquid radwaste system.
- Monitored radwaste discharge pipeline as discussed below.

The monitored radwaste discharge pipeline is engineered to preclude leakage to the environment. This pipe is routed from the auxiliary building to the radwaste building (the short section of pipe between the two buildings is fully available for visual inspection as noted above) and then out of the radwaste building to the appropriate point for dilution and discharge. The exterior piping either incorporates a guard pipe, or is available for visual inspection. No valves or vacuum breakers are incorporated outside of monitored structures.

As discussed in Subsections 11.1.2 and 11.1.3, operation with primary to secondary leakage would lead to limited radiological contamination of some secondary side systems. Liquid leakage from piping or components located in the turbine building will be collected by the waste water system and routed to a turbine building sump.

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Outdoor piping between the condenser hotwell and the condensate storage tank is located above ground where liquid leakage can be detected.

12.1.2.4.2 Fuel Pool Design

The spent fuel pool and connected pools are designed to eliminate undetected leakage to the groundwater as follows:

- The walls of these pools are constructed using modular construction techniques. This allows higher quality than traditional construction. The advanced welding techniques used minimize the potential for weld failures during operation, and allow for inspection to verify weld quality.
- The walls are made of 1/2-inch stainless steel plates, joined to one another with full penetration welds.
- The thickness of the wall plate and the use of full penetration welds prevents wall or weld damage from fuel handling operations, including tool manipulation and storage.
- The pools are equipped with leak chases or alternate means of leak detection at each weld located below normal operating water level range upper value. For a limited number of pool structural welds where the use of leak chase channels behind the weld are not practical, alternate means that establish a leak tight inspectable barrier between the pool water and these structural welds are provided. The leak chase collection system uses piping that is adequately sized to allow testing and to minimize the potential for blockage by encrustation of precipitates (boric acid), and facilitates removal of any such blockage.
- The pool leak detection system will be zoned to allow identification of the area of the pool liner which is leaking, even for very small leaks.
- To the extent possible, these pools are located entirely inside the auxiliary and containment building, so that any theoretical leakage from the tanks is into the building rather than having the potential for release to the environment. Specifically, for pools other than a portion of the fuel transfer canal, the concrete support structure of the pools may be inspected from rooms adjacent to or below (i.e., outside) the pool.

12.1.2.4.3 Equipment Layout

A video record of the equipment layout in areas where radiation fields are expected to be high following operations may be used to assist in ALARA planning and to facilitate decommissioning.

12.1.3 Operational Considerations

The ALARA program is based on mature programs in use at other operating commercial nuclear facilities. As such, it inherently incorporates lessons-learned from decades of operating experience. Industry operating experience is regularly reviewed, and applicable lessons-learned are incorporated into plans, procedures, and policies as warranted.

Functional Structure

The functional structure for the ALARA program is described in Section 12.5.

Organizational Structure

Organizational structure is discussed in Section 13.1 and/or Section 17.5, the Quality Assurance Program Description.

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Radiation Protection Program

The station has a radiation protection program and an ALARA program which contain the operational ALARA philosophy. These programs, made available to plant personnel, define management's commitment to ALARA and designate those individuals who have the responsibility and authority to implement the ALARA program. Section 12.5 provides a complete description of the radiation protection program.

Training

ALARA training is described in Sections 12.5 and 13.2.

Procedures

Procedures are developed and maintained in accordance with Sections 13.5 and 17.5. During initial preparation, radiation protection personnel review the procedures with potential radiological impact for operations, maintenance, refueling, inservice inspections, and operation of the radwaste system for compliance with ALARA guidelines outlined in the radiation protection program.

ALARA Program Review and Improvement

The ALARA Committee and plant management perform periodic reviews of the radiation program content and implementation. In addition to evaluating the implementation of the radiation protection program, these reviews are used to monitor workgroup trends as a means of controlling and reducing personnel exposure. All employees are encouraged to submit suggestions on methods of reducing personnel exposure and improving the ALARA program. Operating procedures are revised, as necessary, to incorporate ALARA lessons-learned from these reviews and suggestions.

Plant Modifications

Modifications to plant equipment and facilities are made where they will substantially reduce exposures at a reasonable cost. Specifications for replacement equipment reflect modifications based on experience gained from using the original equipment. Written procedures direct that all proposed plant modifications are screened for potential adverse radiological impacts. The initial screening review of these proposed modifications is typically performed by engineering personnel. Radiological protection and management personnel perform further review as warranted by level of potential radiological impact.

Work Practices

Radiation protection training, the radiation protection plan, the RWP system, and procedure reviews all help to ensure that radiation exposure of personnel is maintained ALARA. The following examples illustrate the incorporation of ALARA work practices:

- Personnel required to be monitored for radiation exposure in accordance with 10 CFR 20.1502 are assigned appropriate dosimetry to establish exposure history.
- Workers are provided with direct-reading dosimeters on jobs, so that the worker can determine accumulated exposure at any time during a job.
- Dose rate meters are used as needed to identify elevated dose rates.
- Pre-job briefs are used to review radiological surveys and to plan work before personnel enter a radiation area. Written procedures provide guidelines regarding the amount of detail to be included in the pre-job briefings.

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- Post-job debriefs are used to ascertain lessons learned. Incorporation of these lessons may result in lower personnel exposure on future jobs. The requirement for post-job debriefings is specified by procedures.

- For work involving high radiation areas, high collective doses, high levels of removable contamination relative to site posting criteria for contamination areas, or known or suspected airborne radioactivity areas:

  1. Work is preplanned to minimize personnel exposure as defined in ALARA program procedures.

  2. Radiation protection personnel provide coverage as required by radiation protection procedures.

- On complex jobs in high radiation areas, dry-run training may be utilized. In some cases, mockups are used to familiarize workers with the operations that they are to perform. These techniques are beneficial to improving worker efficiency and minimizing the amount of time spent in the radiation field.

- On jobs where general area radiation levels are greater than 1.5 rem/hour or when individual exposure greater than 500 mrem per entry is expected, stay times are considered as further protection against unnecessary exposure.

- As practical, work area entry and exit points are established in areas with low radiation levels. This is done to minimize dose accumulated while changing protective clothing and respiratory equipment. Control points are also established to minimize the spread of removable contamination from the job site.

- As much as practicable, jobs and activities such as reading instruction manuals or maintenance procedures, adjusting tools or jigs, repairing valve internals, and prefabricating components are performed outside radiation areas.

- Individuals working in radiologically controlled areas are trained to be aware of the varying intensities of radiation fields within the general vicinity of their job locations, and are instructed to remain in the areas of lower radiation levels as much as possible, consistent with performing their assigned tasks.

- For high radiation area jobs, maps, postings, and/or detailed instructions are provided to clearly delineate the source of radiation or to alert personnel concerning the location of elevated dose rates. Provided with this information, workers will be cognizant of their immediate radiological environment, and will minimize their stay times in areas of elevated dose rates, thus maintaining exposures ALARA.

- Protective clothing and respiratory equipment prescribed by radiation protection personnel are commensurate with the radiological hazards involved. These requirements cannot be modified without the permission of radiation protection personnel. Consideration is given to the discomfort of workers to minimize the effect of protective efforts on efficiency and the time spent in a radiation area.

- Contamination containments (e.g., glove bags, plastic bottles, tents) and special ventilation systems (e.g., HEPA units) are used where practicable when personnel are working on highly contaminated equipment.

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- Special tools or jigs are used on jobs when their use permits the job to be performed more efficiently or prevents errors, thus reducing the time spent in a radiation area.
- Where applicable, special tools are used to increase the distance from the source to the worker, thereby reducing the exposure received.
- Consideration is given to the use of remote monitoring of personnel with various combinations of audio, visual and dose information to reduce exposure of personnel. Direct communications (e.g., radios) may be used to further enhance radiation protection.
- Some systems and components which are subject to buildup of activated corrosion products are equipped with flush connections to reduce hot-spot buildup. Prior to performing maintenance work on these systems or components, consideration is given to flushing and/or chemically decontaminating the system or piece of equipment in order to reduce the crud levels, thereby reducing dose rates which may result in lower personnel exposure.
- Permanent shielding is used, where practicable, to reduce radiation exposure at the work site and in designated "waiting areas" for personnel during periods when they are not actively involved in the work.
- On some jobs, temporary shielding such as lead sheets draped or strapped over a pipe or concrete blocks stacked around a piece of equipment is used. Temporary shielding is used only if the estimated total exposure, which includes exposure received during installation and removal, is reduced. Experience with such operations is used in developing guidelines in this area.

12.1.4 Combined License Information

Operational considerations of ALARA, as well as operational policies and continued compliance with 10 CFR 20 and associated Regulatory Guides, are addressed in NEI 07-08A and Appendix 12AA.

12.1.5 References

201. 10 CFR 20, Standards for Protection Against Radiation.

202. USNRC, “Qualification and Training of Personnel for Nuclear Power Plants,” Regulatory Guide 1.8, Revision 3, May 2000.

203. Not used.

204. USNRC, “Combined License Applications for Nuclear Power Plants (LWR Edition)”, Regulatory Guide 1.206, June 2007.

205. USNRC, “Guide for Administrative Practices in Radiation Monitoring”, Regulatory Guide 8.2, February 1973.

206. USNRC, “Instructions for Recording and Reporting Occupational Radiation Dose Data”, Regulatory Guide 8.7, Revision 2, November 2005.

207. USNRC, “Information Relevant to Ensuring that Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable”, Regulatory Guide 8.8, Revision 3, June 1978.

208. USNRC, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassy Program”, Regulatory Guide 8.9, Revision 1, July 1993.

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209. USNRC, “Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable”, Regulatory Guide 8.10, Revision 1-R, September 1975.

210. USNRC, “Instructions Concerning Prenatal Radiation Exposure”, Regulatory Guide 8.13, Revision 3, June 1999.

211. USNRC, “Acceptable Programs for Respiratory Protection”, Regulatory Guide 8.15, Revision 1, October 1999.

212. USNRC, “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants”, Regulatory Guide 8.27, March 1981.

213. USNRC, “Audible-Alarm Dosimeters”, Regulatory Guide 8.28, August 1981.

214. USNRC, “Instructions Concerning Risks from Occupational Radiation Exposure”, Regulatory Guide 8.29, Revision 1, February 1996.

215. USNRC, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses”, Regulatory Guide 8.34, July 1992.

216. USNRC, “Planned Special Exposures”, Regulatory Guide 8.35, June 1992.

217. USNRC, “Radiation Dose to the Embryo/Fetus”, Regulatory Guide 8.36, July 1992.

218. USNRC, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants”, Regulatory Guide 8.38, Revision 1, May 2006.

219. USNRC, “Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation”, NUREG-1736, October 2001.

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12.2 Radiation Sources

This section describes the sources of radiation that form the basis for shielding design calculations and the sources of airborne radioactivity used for the design of personnel protection measures and dose assessment.

12.2.1 Contained Sources

The shielding design source terms are based on the three plant conditions of normal full-power operation, shutdown, and design basis accident events.

12.2.1.1 Sources for Full-Power Operation

The primary sources of radioactivity during normal full-power operation are direct core radiation, coolant activation processes, leakage of fission products from pinhole defects in fuel rod cladding, and activation of reactor coolant corrosion products. The design basis for fission product activities is operation with cladding defects in fuel rods producing 0.25 percent of the core thermal power. The design basis for activation and corrosion product activities is derived from measurements at operating plants and is independent of the fuel defect level.

12.2.1.1.1 Reactor Core

The neutron and gamma flux from the reactor core is reduced by the reactor internals and by the reactor vessel. Table 12.2-1 lists the neutron and gamma energy flux spectra in the reactor cavity outside the reactor vessel for several energy groups. The values are maximum values on the inside surface of the primary shield concrete at the core midplane.

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the reactor coolant system are fission products released from fuel and activation of the coolant and of corrosion products that are circulated in the reactor coolant. These sources and their bases are described in Section 11.1.

The activation product, nitrogen-16 (N-16), is the predominant contributor to the activity in the reactor coolant pumps, steam generators, and reactor coolant piping during operation. The N-16 activity in each of the components depends on the total transit time to the component and the average residence time in the component. Table 12.2-3 presents the reactor coolant N-16 activity as a function of transport time in a reactor coolant loop. The N-16 activity for the pressurizer is tabulated in Table 12.2-4.

Fission and corrosion product activities circulating in the reactor coolant system and out-of-core crud deposits comprise the remaining significant radiation sources during full-power operation. The fission and corrosion product activities circulating in the reactor coolant are given in Section 11.1. The fission and corrosion product source strengths and specific activities in the pressurizer liquid and vapor phases are given in Table 12.2-5.

The isotopic composition and specific activity of typical out-of-core crud deposits are given in Table 12.2-6. Typically, one milligram of deposited crud material is found on one square centimeter of a relatively smooth surface. This may be as much as 50 times higher in crud trap areas. Crud trap areas are generally locations of high turbulence, areas of high momentum change, gravitational sedimentation areas, high affinity material areas, and possibly thin boundary layer regions.

The N-16 activity is not a factor in the radiation sources for systems and components located outside containment. This is due to its short half-life (7.11 seconds) and the greater than one minute transport

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time before flow exits the containment. The normal letdown flow path is entirely inside containment. Primary coolant is directed outside containment only when it is diverted to the liquid radwaste system (e.g., due to boron dilution operations or for degassing prior to shutdown).

12.2.1.1.3 Chemical and Volume Control System

Radiation sources in the chemical and volume control system consist of radionuclides carried in the reactor coolant. The chemical and volume control system components in the purification path are located inside containment. The chemical and volume control system carries radioactive fluid out of the containment only when reactor coolant is directed to the liquid radwaste system.

The shielding design of the chemical and volume control system components is based on processing reactor coolant having the design basis source term presented in Section 11.1. The regenerative and letdown heat exchanger sources include contributions from N-16. Owing to its short half-life, the concentration of N-16 is highly sensitive to the location of these heat exchangers with respect to the reactor coolant loop piping. The concentration of N-16 at the heat exchangers is assumed to be the value in the reactor coolant when it exits the steam generator (see Table 12.2-3). The radiation sources for the other components in the purification loop do not include a contribution from N-16. The N-16 contribution to the shielding source term for the filter and demineralizers is determined based on the additional decay afforded by the time delay resulting from the system layout. The chemical and volume control system component sources are provided in Table 12.2-7.

12.2.1.1.4 Service Water System and Component Cooling Water System

These systems are normally nonradioactive or, if there is inleakage of radioactive material into the systems, of very low level activity. For shielding and dose assessment purposes, these systems are assumed to be nonradioactive.

12.2.1.1.5 Spent Fuel Pool Cooling System

One of the functions of the spent fuel pool cooling system is to provide cleanup of the water in the spent fuel pool, the refueling cavity, and the in-containment refueling water storage tank. The equipment considered in designing shielding are the spent fuel pool cooling system demineralizers and filters which accumulate activity, primarily Co-58 and Co-60 from radioactive crud that is resuspended in the water during the course of fuel handling. The source terms for this equipment are provided in Table 12.2-8. Based on operating experience, the remainder of the spent fuel pool cooling system may contain a significant amount of crud and thus requires shielding. The composition of crud is provided in Table 12.2-6.

12.2.1.1.6 Main Steam Supply System

Potential radioactivity in the main steam supply system is a result of steam generator tube leaks and is sufficiently low so that radiation shielding is needed around secondary components, which are capable of concentrating the system's activity, including the steam generator blowdown system (BDS) filter, the electrodeionization (EDI) unit, and the condensate polisher(s).

12.2.1.1.7 Liquid Radwaste System

Radioactive inputs include fission and activation product radionuclides produced in the core and reactor coolant. Shielding for each component of the liquid radwaste system is based on the sources listed on Table 12.2-9. Radiation sources for the various pumps in the liquid radwaste system are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

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12.2.1.1.8 Gaseous Radwaste System

Radioactive gases and hydrogen removed from the reactor coolant when coolant is discharged to the liquid radwaste system comprise the bulk of the gas processed by the gaseous radwaste system. There is no gas stripping performed in the reactor coolant purification loop of the chemical and volume control system. The result is that the volume of gases processed by the gaseous radwaste system is small. Table 12.2-10 lists the shielding sources for the components in the gaseous radwaste system.

12.2.1.1.9 Solid Radwaste System

The solid radwaste system handles various radioactive waste products ranging from relatively low activity materials to high activity spent resins and filter cartridges. Solid wastes are packaged for shipment to a burial or long-term storage facility.

Prior to packaging, the spent resin is stored in a spent resin storage tank. Two spent resin storage tanks are provided, one for high activity resins and the other for low activity resins. The initial gamma source strength in the high activity spent resin storage tank is assumed to be the same as that in the chemical and volume control system mixed bed demineralizer. After a 30-day decay period, only the cesium and cobalt isotopes are significant contributors to the radiation field. Table 12.2-11 lists the source strengths and specific activities both initially and after 30 days of decay.

Spent filter cartridge sources are as listed in Tables 12.2-7, 12.2-8, and 12.2-9.

12.2.1.1.10 Miscellaneous Sources

There are additional contained sources used for instrument calibration or for radiography. These sources will be identified as discussed in Subsection 12.2.3.

Licensed sources containing byproduct, source, and special nuclear material that warrant shielding design consideration meet the applicable requirements of 10 CFR Parts 20, 30, 31, 32, 33, 34, 40, 50, and 70.

There are byproduct and source materials with known isotopes and activity manufactured for the purpose of measuring, checking, calibrating, or controlling processes quantitatively or qualitatively.

These sources include but are not limited to:

- Sources in field monitoring equipment.
- Sources in radiation monitors to maintain a threshold sensitivity.
- Sources used for radiographic operations.
- Depleted uranium slabs used to determine beta response and correction factors for portable monitoring instrumentation.
- Sources used to calibrate and response check field monitoring equipment (portable and fixed).
- Liquid standards and liquids or gases used to calibrate and verify calibration of laboratory counting and analyzing equipment.
- Radioactive waste generated by the use of radioactive sources.

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Specific details of these sources are maintained in a database on-site following procurement. This database, at a minimum, contains the following information:

- Isotopic composition
- Location in the plant
- Source strength
- Source geometry

Written procedures are established and implemented that address procurement, receipt, inventory, labeling, leak testing, surveillance, control, transfer, disposal, storage, issuance and use of these radioactive sources. These procedures are developed in accordance with the radiation protection program to comply with 10 CFR Parts 19 and 20.

Sources maintained on-site for instrument calibration purposes are shielded while in storage to keep personnel exposure ALARA. Sources used to service or calibrate plant instrumentation are also routinely brought on-site by contractors. Radiography is performed by the licensed utility group or licensed contractors. These sources are maintained and used in accordance with the provisions of the utility group's or contractor's license. Additional requirements and restrictions may apply depending on the type of source, use, and intended location of use. If the utility group or contractor source must be stored on-site, designated plant personnel must approve the storage location, and identify appropriate measures for maintaining security and personnel protection.

The radiation protection (RP) calibration lab equipment containing a sealed source (up to 725 curies Cs-137) is to be located so that the adjacent rooms are maintained at Zone I levels.

During the period prior to the implementation of the Emergency Plan (in preparation for the initial fuel loading following the 52.103(g) finding), no specific materials related emergency plan will be necessary because:

a) No byproduct material will be received, possessed, or used in a physical form that is "in unsealed form, on foils or plated sources, or sealed in glass," that exceeds the quantities in Schedule C in 10 CFR 30.72, and

b) No 10 CFR Part 40 specifically licensed source material, including natural uranium, depleted uranium and uranium hexafluoride will be received, possessed, or used during this period.

The following radioactive sources will be used for the Radiation Monitoring System and laboratory/ portable monitoring instrumentation:[1]

Radioactive Licensee Material (Element and Mass Number)[1]	Chemical and/or Physical Form[1]	Maximum Quantity That Licensee May Possess at Any One Time[1]
• Any byproduct material with atomic numbers 1 through 93 inclusive	Sealed Sources[2]	No single source to exceed 100 millicuries 5 Curies total
• Americium-241	Sealed Sources[2]	No single source to exceed 300 millicuries 500 millicuries total
• Cesium-137 stationary source located in RP calibration lab calibration equipment	Sealed Source	725 curies

Notes:

1. This information remains in effect between the issuance of the COL and the Commission's 52.103(g) finding for each unit, and will be designated historical information after that time.
2. Includes calibration and reference sources.

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12.2.1.1.11 Passive Core Cooling System

The passive core cooling system is comprised, in part, of the in-containment refueling water storage tank (IRWST) and the passive residual heat removal heat exchanger (PRHR HX). Radioactivity is assumed to enter the IRWST via reactor coolant system inleakage from the PRHR HX and refueling cavity drain-down. Nuclides are removed from the liquid via the spent fuel pool cooling system (SFS) cleanup, radioactive decay, draining (overflow) to the sump, evolution of gaseous radionuclides, and plateout within the IRWST.

IRWST design basis fission and corrosion product source strengths and specific activities are listed in Table 12.2-30. IRWST expected fission and corrosion product source strengths and specific activities are listed in Table 12.2-31.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only additional significant sources requiring permanent shielding consideration are the spent reactor fuel and the residual heat removal system. Individual components may require shielding during shutdown due to deposited crud material. Estimates of accumulated crud in the reactor coolant system are given in Subsection 12.2.1.1. The radiation sources in the reactor coolant system and other systems addressed in Subsection 12.2.1.1 are bounded by the sources given for full power operation with the exception of a short time period (less than 24 hours) following shutdown, during which crud bursts can result in increased radiation sources. Crud bursts are the resuspension of a portion of the accumulated deposited corrosion products into the reactor coolant system during shutdown operation. Activity increases also occur during planned coolant oxygenation procedures prior to refueling activities.

12.2.1.2.1 Normal Residual Heat Removal System

The maximum gamma ray source strengths in the normal residual heat removal system for four and eight hours after reactor shutdown are given in Table 12.2-12 along with the listing of contributing nuclides. The system may be placed in operation at the maximum flow rate at approximately four hours following a shutdown. The system removes decay heat from the reactor for the duration of the shutdown. The sources given are maximum values taking into account activity increases due to coolant oxygenation measures.

12.2.1.2.2 Reactor Core

The core average gamma ray and neutron source strengths are used in the evaluation of radiation levels within and around the shutdown reactor.

The basis for the core average source strengths is an equilibrium cycle core at end-of-life. Feed enrichment of 4.9 (68 assemblies) weight-percent U-235 was assumed. The regions operate at a specific power of 40.7 megawatts (thermal) per metric ton of uranium for 520, 1040, and 1561 effective full-power days, respectively (this is for an 18 month fuel cycle with a 95 percent capacity factor).

Core average gamma ray source strengths are presented in Table 12.2-2 for various times after shutdown. These source strengths may be put on a per-unit volume of homogenized core basis by multiplying them by the core power density (109.7 watts/cc).

Neutrons are produced in the shutdown reactor by spontaneous fission of the transplutonium isotopes and by (α, n) reactions of alpha particles with 0-17 and 0-18 in the uranium dioxide fuel.

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Core average neutron source strengths are given in Table 12.2-13 for various times after shutdown. The neutron source strengths may be put on a per-unit volume of homogenized core basis by multiplying them by the power density.

12.2.1.2.3 Spent Fuel

Spent fuel gamma ray and neutron source strengths are used in the evaluation of radiation levels for fuels handling, spent fuel storage, and shipping.

The basis for the spent fuel data presented here is the discharge region of an equilibrium cycle core at end of life. A feed enrichment of 4.9 weight-percent U-235 is assumed. The discharge region was operated at a specific power of 40.7 megawatts (thermal) per metric ton of uranium for 1561 effective full-power days.

Spent fuel gamma ray source strengths are presented in Table 12.2-14 for various times after shutdown. These source strengths may be put on a per-unit volume of homogenized core basis by multiplying by the power density (109.7 watts/cc).

Spent fuel neutron source strengths are given in Table 12.2-13 for various times after shutdown. The neutron source strengths may be put on a per-unit volume of homogenized core basis by multiplying them by the power density.

12.2.1.2.4 Irradiated Control Rods, Gray Rods, and Secondary Source Rods

The gamma ray source strengths of the irradiated control rods, gray rods, and secondary source rods are used in establishing radiation shielding requirements during refueling operations and during shipping of irradiated rods.

The absorber material used in the control rods is silver-indium-cadmium (Ag-In-Cd). The gray rods contain tungsten in a nickel alloy 718 sleeve. The gamma ray source strengths associated with the irradiated Ag-In-Cd control rod absorber and gray rod materials are listed in Table 12.2-15 for various times after shutdown.

The photoneutron source material used in the secondary source rods is an equal volume mixture of antimony and beryllium (Sb-Be). The gamma ray source strengths associated with the secondary source rods are listed in Table 12.2-16 for various times after shutdown and Table 12.2-17 lists the neutron source strengths. The source values are per cubic centimeter of source material for an irradiation period of 400 days.

The material used for the control rod cladding, gray rod cladding and/or pellets and secondary source rod cladding is Type 304 stainless steel with a maximum cobalt content of 0.05 weight percent. The gamma ray source strengths associated with the irradiated stainless steel are listed in Table 12.2-18 for various times after shutdown.

12.2.1.2.5 Incore Flux Thimbles

Irradiated incore flux thimble gamma ray source strengths are given in Table 12.2-19.

12.2.1.3 Sources for the Core Melt Accident

The AP1000 is designed to provide adequate core cooling in the event of a postulated loss of coolant accident (LOCA) so that there is no significant core damage. Following a LOCA, the normal residual heat removal system could be used, if available, to provide post-accident cooling. Use of the normal

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residual heat removal system is acceptable only if the source term is close to the design basis source term (see Table 12.2-12).

For the evaluation of the radiological consequences of the LOCA, it is assumed that major degradation of the core takes place, including melting of the core. The source term used for the LOCA dose analysis assumes no core release for 10 minutes, then there is a gap release from a small number of fuel rods before the onset of core degradation. The first half hour of core release is restricted to releases from the fuel cladding gap; this gap release phase is followed by the in-vessel core melt phase that has a duration of 1.3 hours. After the in-vessel core melt phase, there is assumed to be no further release of activity from the core. This core activity release model is based on the source term model from NUREG-1465 (Reference 1). The source term is described in detail in Subsection 15.6.5.3.

12.2.1.3.1 Containment

If there is core degradation, core cooling would be provided by the passive core cooling system which is totally inside the containment such that no high activity sump solution would be recirculated outside the containment. The shielding provided for the containment addresses this post-LOCA source term. The source strengths as a function of time are provided in Table 12.2-20 and the integrated source strengths are provided in Table 12.2-21.

12.2.1.3.2 Main Control Room HVAC Filter

During operation of the nuclear island nonradioactive ventilation system (VBS) supplemental filtration or the main control room emergency habitability system (VES), filters in the control room HVAC work to remove particulate and iodine from the air. As radioactivity accumulates within the filters, this becomes a potential source of dose. These source strengths as a function of time are provided in Table 12.2-28 and the integrated source strengths are provided in Table 12.2-29.

12.2.2 Airborne Radioactive Material Sources

This subsection deals with the models, parameters, and sources required to evaluate airborne concentration of radionuclides during plant operations in various plant radiation areas where personnel occupancy is expected.

12.2.2.1 Containment Atmosphere

The main sources of airborne activity in the containment is leakage of primary coolant and activation of naturally occurring argon in the atmosphere. During normal power operation, excessive activity buildup in the containment atmosphere is prevented by periodic purging of the containment (approximately 20 hours per week). When the plant is shut down for refueling or maintenance, additional purging of the containment atmosphere is performed to further reduce the activity levels consistent with the increased level of worker presence in the containment. The assumptions and parameters used to determine the airborne activity levels in the containment are listed in Table 12.2-22. The airborne concentrations are provided in Table 12.2-23. Three situations are considered: normal power operation without purge, normal power operation with 20 hours of purge operation per week, and shutdown operation.

12.2.2.2 Fuel-Handling Area Atmosphere

The source of airborne activity in the fuel-handling area is leakage from stored spent fuel assemblies and the evaporation losses from the spent fuel pool. The maximum airborne concentration in the fuel-handling area is calculated using the assumptions and parameters in Table 12.2-24. The resulting airborne isotopic concentrations are provided in Table 12.2-25.

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12.2.2.3 Auxiliary Building Atmosphere

The source of airborne activity in the auxiliary building atmosphere is primarily equipment leakage. The ventilation system constantly removes activity and discharges it to the plant vent. The maximum airborne concentration in the auxiliary building is calculated using the assumptions and parameters in Table 12.2-26. The resulting airborne isotopic concentrations are provided in Table 12.2-27.

12.2.2.4 Airborne Activity Model

The airborne concentration of each nuclide in the atmosphere is calculated by:

$$C_i(t) = \frac{(LR)_i \, A_i (PF)_i (1 - e^{-\lambda_{Ti} t})}{(V)(\lambda_{Ti})}$$

where:

$(LR)_i$ = leak or evaporation rate of the I^{th} radioisotope in the applicable region (g/s)

A_i = activity concentration of the I^{th} leaking or evaporating radioisotope (μCi/g)

$(PF)_i$ = partition factor or the fraction of the leaking activity that is airborne for the I^{th} radioisotope

λ_{Ti} = total removal rate constant for the I^{th} radioisotope from the applicable region (s^{-1})

= $\lambda_{di} + \lambda_e$, the removal rate constants in s^{-1} due to radioactive decay for the I^{th} radioisotope and the exhaust from the applicable region, respectively

t = time elapsed from the start of the leak and the time at which the concentration is evaluated (s)

V = free volume of the region in which the leak occurs (cm^3)

$C_i(t)$ = airborne concentration of the Ith radioisotope at time t in the applicable region ($\mu Ci/cm^3$).

From the above equation, the peak or equilibrium concentration, C_{Egi}, of the I^{th} radioisotope in the applicable volume is given by the following expression:

$$C_{Egi} = (LR)_I \, A_I \, (PF)_I / V\lambda_{Ti}$$

With high exhaust rates, this peak concentration is reached within a few hours.

12.2.3 Combined License Information

The additional contained radiation sources not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography, are addressed in Subsection 12.2.1.1.10.

12.2.4 References

1. L. Soffer, et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

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Table 12.2-1
Radiation Flux at the Primary Shield Concrete

Neutron Flux	
Energy	**Flux (n/cm^2-sec)**
≥ 1 Mev	2.4E+09
0.1 Mev to 1 Mev	1.8E+10
0.414 ev to 0.1 Mev	2.7E+10
< 0.414 ev	6.4E+9
Gamma Energy Flux	
Energy	**Flux (Mev/cm^2-sec)**
≥ 6.0 Mev	1.2E+10
3.0 Mev to 6.0 Mev	8.3×10^9
1.0 Mev to 3.0 Mev	5.9×10^9
< 1.0 Mev	3.9×10^9

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Table 12.2-2 (Sheet 1 of 2)
Core Average Gamma Ray Source Strengths
at Various Times After Shutdown

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)				
	12 Hours	24 Hours	100 Hours	1 Week	1 Month
0.000-0.020	1.8E+08	1.5E+08	6.5E+07	3.1E+07	1.4E+06
0.020-0.030	4.2E+07	3.7E+07	2.2E+07	1.6E+07	3.9E+06
0.030-0.045	1.1E+08	1.0E+08	6.1E+07	4.5E+07	1.5E+07
0.045-0.070	6.5E+07	5.9E+07	3.2E+07	2.0E+07	1.9E+06
0.070-0.100	3.6E+08	3.2E+08	1.6E+08	9.4E+07	8.6E+06
0.100-0.150	1.5E+09	1.3E+09	6.3E+08	3.6E+08	8.7E+07
0.150-0.300	2.2E+09	1.8E+09	7.1E+08	3.5E+08	1.4E+07
0.300-0.450	9.3E+08	8.2E+08	5.2E+08	3.9E+08	7.6E+07
0.450-0.700	5.3E+09	4.3E+09	2.4E+09	1.9E+09	9.5E+08
0.700-1.000	6.8E+09	5.9E+09	4.3E+09	3.8E+09	2.6E+09
1.000-1.500	2.0E+09	1.2E+09	4.9E+08	3.3E+08	9.2E+07
1.500-2.000	3.4E+09	3.1E+09	2.6E+09	2.3E+09	6.6E+08
2.000-2.500	3.0E+08	2.0E+08	1.4E+08	1.2E+08	4.9E+07
2.500-3.000	1.9E+08	1.8E+08	1.5E+08	1.3E+08	3.9E+07
3.000-4.000	5.2E+06	1.9E+06	1.6E+06	1.4E+06	4.4E+05
4.000-6.000	3.5E+05	1.9E+04	4.6E+00	4.6E+00	4.4E+00
6.000-8.000	7.4E-01	7.4E-01	7.4E-01	7.4E-01	7.1E-01
8.000-11.000	1.2E-01	1.2E-01	1.2E-01	1.1E-01	1.1E-01

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Table 12.2-2 (Sheet 2 of 2)
Core Average Gamma Ray Source Strengths
at Various Times After Shutdown

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)			
	3 Months	6 Months	1 Year	5 Years
0.000-0.020	3.0E+05	1.7E+05	1.0E+05	2.9E+04
0.020-0.030	9.6E+05	3.8E+05	2.2E+05	7.4E+04
0.030-0.045	6.7E+06	4.3E+06	2.8E+06	4.5E+05
0.045-0.070	2.5E+05	1.6E+05	1.1E+05	5.6E+04
0.070-0.100	2.5E+06	1.9E+06	1.3E+06	1.4E+05
0.100-0.150	3.4E+07	1.6E+07	8.7E+06	7.0E+05
0.150-0.300	1.5E+06	6.4E+05	3.8E+05	2.0E+05
0.300-0.450	5.6E+06	2.5E+06	1.9E+06	6.4E+05
0.450-0.700	4.8E+08	3.1E+08	2.3E+08	9.4E+07
0.700-1.000	1.5E+09	7.2E+08	2.0E+08	3.3E+07
1.000-1.500	4.3E+07	3.5E+07	2.7E+07	8.4E+06
1.500-2.000	2.9E+07	2.8E+06	1.8E+06	2.9E+05
2.000-2.500	2.1E+07	1.6E+07	1.0E+07	3.1E+05
2.500-3.000	1.7E+06	1.6E+05	1.0E+05	6.6E+03
3.000-4.000	4.8E+04	2.8E+04	1.9E+04	1.2E+03
4.000-6.000	4.1E+00	3.8E+00	3.4E+00	2.7E+00
6.000-8.000	6.7E-01	6.2E-01	5.5E-01	4.4E-01
8.000-11.000	1.0E-01	9.6E-02	8.6E-02	6.8E-02

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Table 12.2-3
Reactor Coolant Nitrogen-16 Activity

Position in Loop	Loop Transit Time (sec)	Nitrogen-16 Activity (μCi/g)
Leaving core	0.0	280
Leaving reactor vessel	0.9	256
Entering steam generator	1.2	249
Leaving steam generator	6.8	144
Entering reactor vessel	8.0	128
Entering core	9.5	122
Leaving core	10.3	280

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Table 12.2-4
Pressurizer Nitrogen-16 Source Strengths

Discrete Energy (Mev/gamma)	Energy Group (Mev/gamma)	Source Strength	
		Liquid Phase[(a)] (Mev/gram-sec)	Steam Phase[(b)] (Mev/cm^3-sec)
1.75	1.35 - 1.80	5.4E-02	1.4E-01
2.74	2.6 - 3.0	5.0E-01	1.3E+00
6.13	6.0 - 7.0	1.0E+02	2.6E+02
7.12	7.0 - 7.5	8.5E+00	2.2E+01

Notes:
a. Based on an insurge to the pressurizer following a ten percent step load power increase.
b. Based on a boron equalization spray rate of 80 gpm.

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Table 12.2-5 (Sheet 1 of 4)
Pressurizer Liquid and Steam Phase
Source Strengths and Specific Activity

1000 Cubic Foot Liquid Phase Source Strengths	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	1.4E+03
0.02-0.03	1.6E+03
0.03-0.045	6.9E+04
0.045-0.07	3.2E+02
0.07-0.1	1.4E+05
0.1-0.15	1.1E+03
0.15-0.3	4.3E+04
0.3-0.45	2.3E+04
0.45-0.7	1.0E+05
0.7-1.0	1.3E+05
1.0-1.5	1.2E+05
1.5-2.0	5.9E+04
2.0-2.5	9.3E+04
2.5-3.0	1.5E+04
3.0-4.0	2.6E+03
4.0-6.0	5.0E+02
6.0-8.0	-
8.0-11.00	-
Total	7.9E+05

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Table 12.2-5 (Sheet 2 of 4)
Pressurizer Liquid and Steam Phase
Source Strengths and Specific Activity

1000 Cubic Foot Liquid Phase Specific Activity	
Nuclide	**Activity (μCi/gram)**
Kr-87	4.7E-01
Kr-88	1.5E+00
Kr-89	3.5E-02
Xe-131m	1.3E+00
Xe-133m	1.7E+00
Xe-133	1.2E+02
Xe-135	3.5E+00
Br-84	1.7E-02
I-131	7.1E-01
I-132	9.3E-01
I-133	1.3E+00
I-134	2.2E-01
I-135	7.8E-01
Rb-88	1.5E+00
Cs-134	6.9E-01
Cs-136	1.0E+00
Cs-138	3.7E-01
Tc-99m	2.0E-01

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Table 12.2-5 (Sheet 3 of 4)
Pressurizer Liquid and Steam Phase
Source Strengths and Specific Activity

1100 Cubic Foot Steam Phase Source Strengths (Mev/cm^3-sec)	
Energy Group (Mev/gamma)	**Normal 2-gpm Spray**
0-0.02	2.3E+03
0.02-0.03	2.1E+03
0.03-0.045	1.3E+05
0.045-0.07	2.0E-04
0.07-0.1	2.7E+05
0.1-0.15	1.9E+00
0.15-0.3	6.7E+03
0.3-0.45	5.0E+02
0.45-0.7	1.7E+03
0.7-1.0	8.4E+02
1.0-1.5	6.3E+02
1.5-2.0	6.3E+02
2.0-2.5	2.9E+03
2.5-3.0	1.3E+02
3.0-4.0	4.3E+00
4.0-6.0	2.2E-05
6.0-8.0	-
8.0-11.00	-
Total	4.1E+05

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Table 12.2-5 (Sheet 4 of 4)
Pressurizer Liquid and Steam Phase
Source Strengths and Specific Activity

1100 Cubic Foot Steam Phase Specific Activity ($\mu Ci/cm^3$)	
Nuclide	**Normal 2-gpm Spray**
Kr-85m	5.2E-02
Kr-85	9.0E+00
Kr-87	6.7E-03
Kr-88	5.5E-02
Kr-89	1.7E-08
Xe-131m	2.9E+00
Xe-133m	1.4E+00
Xe-133	2.4E+02
Xe-135	4.6E-01
Xe-138	1.7E-04
I-131	7.1E-03
I-132	9.3E-03
I-133	1.3E-02
I-134	2.2E-03
I-135	7.8E-03

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Table 12.2-6
Isotopic Composition and Specific Activity of
Typical Out-of-Core Crud Deposits[(a)]

Composition (Nuclide)	Activity (μCi/mg) of Deposited Crud for Effective Full Power Years of Plant Operation			
	1 Year	2 Years	5 Years	10 Years
Mn-54	1.0	1.1	1.3	1.4
Fe-59	0.5	0.5	0.5	0.5
Co-58	12.0	12.0	12.0	12.0
Co-60	1.5	2.3	4.0	6.0

Note:

a. In addition to corrosion products, about 1.0 μg of mixed actinides and fission products may be present for each 1 g of deposited crud.

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Table 12.2-7 (Sheet 1 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

A. Regenerative heat exchanger		
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**	
	Tube Side	**Shell Side**
0-0.02	1.4E+03	1.4E+03
0.02-0.03	1.6E+03	1.6E+03
0.03-0.045	6.9E+04	6.9E+04
0.045-0.07	3.2E+02	3.2E+02
0.07-0.1	1.4E+05	1.4E+05
0.1-0.15	1.1E+03	1.1E+03
0.15-0.3	4.3E+04	4.2E+04
0.3-0.45	2.3E+04	1.5E+04
0.45-0.7	1.0E+05	4.7E+04
0.7-1.0	1.3E+05	7.5E+04
1.0-1.5	1.2E+05	7.7E+04
1.5-2.0	7.1E+04	4.2E+04
2.0-2.5	9.3E+04	8.9E+04
2.5-3.0	1.3E+05	1.5E+04
3.0-4.0	2.6E+03	2.2E+03
4.0-6.0	5.0E+02	5.0E+02
6.0-8.0	2.5E+07	-
8.0-11.00	-	-
Total	2.5E+07	6.2E+05

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Table 12.2-7 (Sheet 2 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

A. Regenerative heat exchanger		
Nuclide	**Activity (µCi/gram)**	
	Tube Side	**Shell Side**
Kr-87	4.7E-01	4.7E-01
Kr-88	1.5E+00	1.5E+00
Kr-89	3.5E-02	3.5E-02
Xe-131m	1.3E+00	1.3E+00
Xe-133m	1.7E+00	1.7E+00
Xe-133	1.2E+02	1.2E+02
Xe-135	3.5E+00	3.5E+00
Br-84	1.7E-02	--
I-131	7.1E-01	--
I-132	9.3E-01	--
I-133	1.3E+00	--
I-134	2.2E-01	--
I-135	7.8E-01	--
Rb-88	1.5E+00	1.5E+00
Cs-134	6.9E-01	6.9E-01
Cs-136	1.0E+00	1.0E+00
Cs-138	3.7E-01	3.7E-01
Tc-99m	2.0E-01	2.0E-01
Ba-137m	--	4.7E-01
N-16	1.4E+02	--

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Table 12.2-7 (Sheet 3 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

B. Letdown heat exchanger	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	1.4E+03
0.02-0.03	1.6E+03
0.03-0.045	6.9E+04
0.045-0.07	3.2E+02
0.07-0.1	1.4E+05
0.1-0.15	1.1E+03
0.15-0.3	4.3E+04
0.3-0.45	2.3E+04
0.45-0.7	1.0E+05
0.7-1.0	1.3E+05
1.0-1.5	1.2E+05
1.5-2.0	7.1E+04
2.0-2.5	9.3E+04
2.5-3.0	1.3E+05
3.0-4.0	2.6E+03
4.0-6.0	5.0E+02
6.0-8.0	2.5E+07
8.0-11.00	-
Total	2.5E+07

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Table 12.2-7 (Sheet 4 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

B. Letdown heat exchanger	
Nuclide	**Activity (µCi/gram)**
Kr-87	4.7E-01
Kr-88	1.5E+00
Kr-89	3.5E-02
Xe-131m	1.3E+00
Xe-133m	1.7E+00
Xe-133	1.2E+02
Xe-135	3.5E+00
Br-84	1.7E-02
I-131	7.1E-01
I-132	9.3E-01
I-133	1.3E+00
I-134	2.2E-01
I-135	7.8E-01
Rb-88	1.5E+00
Cs-134	6.9E-01
Cs-136	1.0E+00
Cs-138	3.7E-01
Tc-99m	2.0E-01
N-16	1.4E+02

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Table 12.2-7 (Sheet 5 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

C. Mixed bed demineralizer (50 cubic feet of resin)	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	1.7E+04
0.02-0.03	1.1E+05
0.03-0.045	3.5E+05
0.045-0.07	4.8E+04
0.07-0.1	2.2E+05
0.1-0.15	4.8E+03
0.15-0.3	2.2E+06
0.3-0.45	2.8E+07
0.45-0.7	1.8E+08
0.7-1.0	9.9E+07
1.0-1.5	2.8E+07
1.5-2.0	4.2E+06
2.0-2.5	8.7E+05
2.5-3.0	7.4E+05
3.0-4.0	9.9E+04
4.0-6.0	2.7E+03
6.0-8.0	-
8.0-11.00	-
Total	3.4E+08

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Table 12.2-7 (Sheet 6 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

C. Mixed bed demineralizer (50 cubic feet of resin)	
Nuclide	**Activity (µCi/gram)**
Mn-54	6.2E+01
Mn-56	1.3E+01
Co-58	7.2E+01
Co-60	7.8E+01
I-131	2.4E+03
I-132	4.3E+01
I-134	3.9E+00
I-135	9.9E+01
Rb-88	8.9E+00
Cs-134	3.0E+03
Cs-136	1.6E+02
Cs-137	3.0E+03
Ba-137m	2.8E+03

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Table 12.2-7 (Sheet 7 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

D. Cation bed demineralizer (50 cubic feet of resin)	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	6.3E+03
0.02-0.03	-
0.03-0.045	3.3E+05
0.045-0.07	4.8E+04
0.07-0.1	3.1E+04
0.1-0.15	2.9E+03
0.15-0.3	5.1E+05
0.3-0.45	9.9E+05
0.45-0.7	1.6E+08
0.7-1.0	9.0E+07
1.0-1.5	1.6E+07
1.5-2.0	1.9E+06
2.0-2.5	4.4E+05
2.5-3.0	6.9E+05
3.0-4.0	9.3E+04
4.0-6.0	2.7E+03
6.0-8.0	-
8.0-11.00	-
Total	2.7E+08

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Table 12.2-7 (Sheet 8 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

D. Cation bed demineralizer (50 cubic feet of resin)	
Nuclide	**Activity (µCi/gram)**
Rb-88	8.9E+00
Cs-134	3.0E+03
Cs-136	1.6E+02
Cs-137	3.0E+03
Cs-138	3.7E+00
Ba-137m	2.8E+03
E. Reactor coolant filter	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.4 - 0.9	5.7E+07
0.9 - 1.35	1.5E+07

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Table 12.2-8
Spent Fuel Pool Cooling System Component
Source Strengths and Specific Activity

A. Demineralizer (75 cubic feet of resin)	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.4 - 0.9	3.0E+06
0.9 - 1.35	4.7E+06
Nuclide	**Activity ($\mu Ci/cm^3$)**
Co-58	5.8E+01
Co-60	3.4E+01
B. Filters	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.4 - 0.9	1.1E+07
0.9 - 1.35	3.0E+06
Source Dimensions (inches)	**Source Composition (volume percent)**
Radius = 3.375	Air – 67
Length = 19	Water – 33

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Table 12.2-9 (Sheet 1 of 7)
Liquid Radwaste System
Component Source Terms

A. Reactor coolant drain tank		
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**	
	Liquid Phase (450 gallons)	**Gas Space (60 cubic feet)**
0.000-0.020	1.4E+03	1.4E+02
0.020-0.030	1.6E+03	2.0E+02
0.030-0.045	6.9E+04	7.6E+03
0.045-0.070	3.2E+02	6.1E-05
0.070-0.100	1.4E+05	1.5E+04
0.100-0.150	1.1E+03	8.4E-02
0.150-0.300	4.3E+04	4.2E+02
0.300-0.450	2.3E+04	2.6E+01
0.450-0.700	1.0E+05	4.1E+03
0.700-1.000	1.3E+05	2.3E+01
1.000-1.500	1.2E+05	1.4E+01
1.500-2.000	5.9E+04	3.3E+01
2.000-2.500	9.3E+04	2.0E+02
2.500-3.000	1.5E+04	1.0E+01
3.000-4.000	2.6E+03	3.7E-01
4.000-6.000	5.0E+02	2.2E-03
6.000-8.000	-	-
8.000-11.000	-	-
Total	7.9E+05	2.8E+04

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Table 12.2-9 (Sheet 2 of 7)
Liquid Radwaste System
Component Source Terms

A. Reactor coolant drain tank		
Nuclide	**Activity ($\mu Ci/cm^3$)**	
	Liquid Phase	**Gas Space**
Mn-56	1.7E-01	-
I-132	9.3E-01	-
I-134	2.2E-01	-
Kr-87	4.7E-01	-
Kr-88	1.5E+00	-
I-131	7.1E-01	-
I-133	1.3E+00	-
Xe-133	1.2E+02	-
I-135	7.8E-01	-
Xe-135	3.5E+00	-
Xe-138	2.4E-01	-
Rb-88	1.5E+00	-
Cs-134	6.9E-01	-
Cs-136	1.0E+00	-
Ba-137m	4.7E-01	-
Cs-138	3.7E-01	-
Kr-85	-	5.0E+01
Kr-87	-	5.4E-04
Kr-88	-	3.8E-03
Xe-131m	-	3.4E-01
Xe-133m	-	8.0E-02
Xe-133	-	1.4E+01
Xe-135	-	2.8E-02

Notes:
The liquid activities listed are 99% of the total source strength.
The vapor activities listed are essentially 100% of the source strength.

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Table 12.2-9 (Sheet 3 of 7)
Liquid Radwaste System
Component Source Terms

B. Effluent tank (28,000 gal) and waste holdup tank (15,000 gal)			
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**		
	Effluent Tank		**Holdup Tank Liquid**
	Liquid Phase	**Vapor Phase**	
0.000-0.020	1.9E+01	3.4E+03	1.9E+01
0.020-0.030	1.6E+01	4.0E+03	1.2E+01
0.030-0.045	2.7E+02	1.8E+05	2.7E+02
0.045-0.070	3.2E+02	4.5E-05	3.2E+02
0.070-0.100	2.2E+02	3.7E+05	2.1E+02
0.100-0.150	1.0E+03	3.6E+01	9.8E+02
0.150-0.300	3.4E+03	7.6E+04	3.3E+03
0.300-0.450	7.5E+03	5.4E+03	7.0E+03
0.450-0.700	4.1E+04	6.1E+03	3.8E+04
0.700-1.000	6.5E+04	1.0E+04	5.6E+04
1.000-1.500	7.1E+04	6.5E+03	4.7E+04
1.500-2.000	2.5E+04	1.5E+04	2.1E+03
2.000-2.500	6.5E+03	9.1E+04	5.7E+02
2.500-3.000	7.3E+03	4.0E+03	1.4E+02
3.000-4.000	1.5E+03	1.3E+02	1.7E+01
4.000-6.000	4.5E+02	9.8E-18	-
6.000-8.000	-	-	-
8.000-11.000	-	-	-
Total	2.3E+05	7.7E+05	1.6E+05

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Table 12.2-9 (Sheet 4 of 7)
Liquid Radwaste System
Component Source Terms

B. Effluent tank (28,000 gal) and waste holdup tank (15,000 gal)			
	Activity ($\mu Ci/cm^3$)		
	Effluent Tank		
Nuclides	**Liquid Phase**	**Vapor Phase**	**Holdup Tank Liquid**
Mn-56	1.7E-02	–	1.7E-02
I-131	-	–	7.1E-02
I-132	9.3E-02	–	9.3E-02
I-133	1.3E-01	–	1.3E-01
I-134	2.2E-02	–	–
I-135	7.8E-02	–	7.8E-02
Rb-88	1.5E+00	–	–
Rb-89	6.9E-02	–	–
Cs-134	6.9E-01	–	6.9E-01
Cs-136	1.0E+00	–	1.0E+00
Cs-137	5.0E-01	–	5.0E-01
Ba-137m	5.0E-01	–	5.0E-01
Sr-89	–	–	1.1E-04
Cs-138	3.7E-01	–	–
Mo-99	2.1E-01	–	2.1E-01
Kr-85m	–	1.3E+00	–
Kr-85	–	8.0E+00	–
Kr-87	–	2.1E-01	–
Kr-88	–	1.8E+00	–
Xe-131m	–	3.6E+00	–
Xe-133m	–	4.3E+00	–
Xe-133	–	3.3E+02	–
Xe-135	–	7.3E+00	–

Notes:
The liquid activities listed are 99% of the total source strength.
The vapor activities listed are essentially 100% of the source strength.

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Table 12.2-9 (Sheet 5 of 7)
Liquid Radwaste System
Component Source Terms

C. Chemical waste tank (7,700 gal)	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	6.3E-01
0.02-0.03	1.2E-01
0.03-0.045	2.5E+01
0.045-0.07	3.1E-04
0.07-0.1	5.3E-03
0.1-0.15	3.2E-02
0.15-0.3	3.8E+00
0.3-0.45	5.6E+00
0.45-0.7	1.6E+04
0.7-1.0	9.7E+03
1.0-1.5	1.0E+03
1.5-2.0	1.5E+02
2.0-2.5	3.2E+01
2.5-3.0	5.3E+01
3.0-4.0	6.9E+00
4.0-6.0	–
6.0-8.0	–
8.0-11.00	–
Total	2.7E+04
Nuclide	**Activity (µCi/gram)**
Co-58	1.9E-03
Cs-134	3.4E-01
Cs-137	2.5E-01
Te-127m	3.8E-04
Ba-137m	2.3E-01
Ce-144	5.8E-05

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Table 12.2-9 (Sheet 6 of 7)
Liquid Radwaste System
Component Source Terms

D. Waste ion exchanger and charcoal deep bed filter vessel[a]	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0-0.02	1.2E+03
0.02-0.03	5.5E+03
0.03-0.045	5.1E+04
0.045-0.07	5.8E+04
0.07-0.1	4.8E+04
0.1-0.15	3.4E+03
0.15-0.3	6.7E+05
0.3-0.45	2.5E+06
0.45-0.7	9.1E+06
0.7-1.0	1.1E+07
1.0-1.5	9.3E+06
1.5-2.0	6.0E+05
2.0-2.5	1.4E+05
2.5-3.0	2.9E+04
3.0-4.0	3.5E+03
4.0-6.0	–
6.0-8.0	–
8.0-11.00	–
Total	3.3E+07

Note:
a. Source term for the charcoal deep bed filter vessel is based on operation charged with resin instead of charcoal since this is the most conservative mode of operation for source terms.

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Table 12.2-9 (Sheet 7 of 7)
Liquid Radwaste System
Component Source Terms

D. Waste ion exchanger and charcoal deep bed filter vessel[(a)]	
Nuclide	**Activity (μCi/cm^3)**
Mn-56	3.4E+00
I-131	1.2E+02
I-132	1.7E+01
I-133	1.1E+02
I-135	2.8E+01
Cs-134	1.4E+02
Cs-136	1.9E+02
Cs-137	9.9E+01
Ba-137m	9.4E+01
E. Waste prefilter and waste after filter	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.4 - 0.9	1.1E+07
0.9 - 1.35	3.0E+06
Total	1.4E+07
Source Dimensions (inches)	**Source Composition (volume percent)**
Radius = 3.375	Air - 67
Length = 19	Water - 33

Note:

a. Source term for the charcoal deep bed filter vessel is based on operation charged with resin instead of charcoal since this is the most conservative mode of operation for source terms.

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Table 12.2-10 (Sheet 1 of 4)
Gaseous Radwaste System
Component Source Terms

A. Gas cooler and moisture separator	
Energy Group (MeV/gamma)	**Source Strength (MeV/cm^3-sec)**
0.000 - 0.020	3.2E+04
0.020 - 0.030	3.8E+04
0.030 - 0.045	1.7E+06
0.045 - 0.070	7.1E+00
0.070 - 0.100	3.4E+06
0.100 - 0.150	6.6E+02
0.150 - 0.300	9.4E+05
0.300 - 0.450	1.7E+05
0.450 - 0.700	1.6E+05
0.700 - 1.000	2.4E+05
1.000 - 1.500	1.6E+05
1.500 - 2.000	4.1E+05
2.000 - 2.500	2.0E+06
2.500 - 3.000	1.8E+05
3.000 - 4.000	1.8E+04
4.000 - 6.000	1.1E+03
6.000 - 8.000	0.0E+00
8.000 - 11.000	0.0E+00
Total	9.4E+06

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Table 12.2-10 (Sheet 2 of 4)
Gaseous Radwaste System
Component Source Terms

A. Gas cooler and moisture separator	
Nuclide	**Activity ($\mu Ci/cm^3$)**
Kr-83m	4.4E+00
Kr-85m	2.0E+01
Kr-85	7.2E+01
Kr-87	1.2E+01
Kr-88	3.6E+01
Kr-89	8.4E-01
Xe-131m	3.2E+01
Xe-133m	4.1E+01
Xe-133	3.0E+03
Xe-135m	4.2E+00
Xe-135	8.5E+01
Xe-137	1.6E+00
Xe-138	5.9E+00

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Table 12.2-10 (Sheet 3 of 4)
Gaseous Radwaste System
Component Source Terms

B. Charcoal guard and delay beds	
Energy Group (MeV/gamma)	**Source Strength (MeV/cm^3-sec)**
0.000 - 0.020	3.2E+04
0.020 - 0.030	3.8E+04
0.030 - 0.045	1.7E+06
0.045 - 0.070	7.1E+00
0.070 - 0.100	3.4E+06
0.100 - 0.150	6.6E+02
0.150 - 0.300	9.4E+05
0.300 - 0.450	1.7E+05
0.450 - 0.700	1.6E+05
0.700 - 1.000	2.4E+05
1.000 - 1.500	1.6E+05
1.500 - 2.000	4.1E+05
2.000 - 2.500	2.0E+06
2.500 - 3.000	1.8E+05
3.000 - 4.000	1.8E+04
4.000 - 6.000	1.1E+03
6.000 - 8.000	0.0E+00
8.000 - 11.000	0.0E+00
Total	9.4E+06

Note:
Due to the long hold-up time in the guard and delay beds, the source strength in the beds vary throughout the length of the beds as the source decays. The source strength listed represents the source strength entering the guard bed and delay beds.

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Table 12.2-10 (Sheet 4 of 4)
Gaseous Radwaste System
Component Source Terms

B. Charcoal guard and delay beds	
Nuclide	**Activity ($\mu Ci/cm^3$)**
Kr-85m	2.0E+01
Kr-85	7.2E+01
Kr-87	1.2E+01
Kr-88	3.6E+01
Xe-131m	3.2E+01
Xe-133m	4.1E+01
Xe-133	3.0E+03
Xe-135m	4.2E+00
Xe-135	8.5E+01
Xe-138	5.9E+00

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Table 12.2-11 (Sheet 1 of 2)
Spent Demineralizer Resin
Source Strengths and Specific Activities

Energy Group (Mev/gamma)	Spent Resin Source Strength (Mev/cm^3-sec)	
	Initial	After 30 Days
0-0.02	1.7E+04	1.3E+04
0.02-0.03	1.1E+05	8.1E+03
0.03-0.045	3.5E+05	3.0E+05
0.045-0.07	4.8E+04	9.7E+03
0.07-0.1	2.2E+05	2.1E+04
0.1-0.15	4.8E+03	5.7E+02
0.15-0.3	2.2E+06	2.6E+05
0.3-0.45	2.8E+07	2.3E+06
0.45-0.7	1.8E+08	1.6E+08
0.7-1.0	9.9E+07	8.8E+07
1.0-1.5	2.8E+07	1.7E+07
1.5-2.0	4.2E+06	1.9E+06
2.0-2.5	8.7E+05	3.8E+05
2.5-3.0	7.4E+05	6.4E+05
3.0-4.0	9.9E+04	8.3E+04
4.0-6.0	2.7E+03	–
6.0-8.0	–	–
8.0-11.00	–	–
Total	3.4E+08	2.7E+08

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Table 12.2-11 (Sheet 2 of 2)
Spent Demineralizer Resin
Source Strengths and Specific Activities

Nuclide	Spent Resin Activity (μCi/cm^3)	
	Initial	After 30 Days
Mn-54	6.2E+01	5.8E+01
Mn-56	1.3E+01	-
Co-58	7.2E+01	5.4E+01
Co-60	7.8E+01	7.7E+01
I-131	2.4E+03	1.8E+02
I-132	4.3E+01	–
I-134	3.9E+00	–
I-135	9.9E+01	–
Rb-88	8.9E+00	–
Cs-134	3.0E+03	3.0E+03
Cs-136	1.6E+02	3.1E+01
Cs-137	3.0E+03	3.0E+03
Ba-137m	2.8E+03	2.8E+03

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Table 12.2-12 (Sheet 1 of 2)
Normal Residual Heat Removal System
Source Strengths and Specific Activities

Energy Group (Mev/gamma)	Source Strength (Mev/gram-sec)	
	4 Hours After Shutdown	8 Hours After Shutdown
0.000 -0.020	1.3E+03	1.2E+03
0.020 -0.030	1.5E+03	1.4E+03
0.030 -0.045	6.7E+04	6.6E+04
0.045 -0.070	2.1E+02	1.4E+02
0.070 -0.100	1.4E+05	1.3E+05
0.100 -0.150	9.6E+02	9.2E+02
0.150 -0.300	3.0E+04	2.2E+04
0.300 -0.450	1.1E+04	7.2E+03
0.450 -0.700	4.6E+04	2.7E+04
0.700 -1.000	3.5E+05	3.3E+05
1.000 -1.500	5.6E+04	3.5E+04
1.500 -2.000	2.0E+04	7.8E+03
2.000 -2.500	3.0E+04	1.1E+04
2.500 -3.000	2.6E+03	7.9E+02
3.000 -4.000	4.7E+02	1.7E+02
4.000 -6.000	1.7E+02	6.2E+01
6.000 -8.000	–	–
8.000-11.000	–	–
Total	7.6E+05	6.4E+05

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Table 12.2-12 (Sheet 2 of 2)
Normal Residual Heat Removal System
Source Strengths and Specific Activities

Nuclide	Activity (μCi/gram)	
	4 Hours After Shutdown	8 Hours After Shutdown
Kr-87	5.3E-02	6.0E-03
Kr-88	5.5E-01	2.1E-01
Xe-131m	1.3E+00	1.3E+00
Xe-133m	1.6E+00	1.5E+00
Xe-133	1.2E+02	1.2E+02
Xe-135	2.7E+00	2.0E+00
I-131	4.7E-01	3.1E-01
I-132	1.9E-01	4.0E-02
I-133	7.5E-01	4.4E-01
I-135	3.4E-01	1.5E-01
Rb-88	5.9E-01	2.2E-01
Cs-134	4.6E-01	3.1E-01
Cs-136	6.8E-01	4.5E-01
Tc-99m	1.9E-01	1.8E-01
Ba-137m	3.1E-01	2.1E-01
Co-58	1.0E+01	1.0E+01
Co-60	1.0E-01	1.0E-01

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Table 12.2-13
Core Average and Spent Fuel Neutron Source Strengths at Various Times After Shutdown

Time After Shutdown	Core Average (n/watt-sec)	Spent Fuel (n/watt-sec)
12 hours	22	77
24 hours	22	77
100 hours	22	76
1 week	22	76
1 month	21	75
3 months	20	71
6 months	18	68
1 year	16	63
5 years	13	51

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Table 12.2-14 (Sheet 1 of 2)
Spent Fuel Gamma Ray Source Strengths

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)				
	12 Hours	24 Hours	100 Hours	1 Week	1 Month
0-0.02	2.3E+08	2.0E+08	8.4E+07	4.1E+07	1.7E+06
0.02-0.03	4.3E+07	3.7E+07	2.2E+07	1.6E+07	4.1E+06
0.03-0.045	1.1E+08	1.0E+08	6.0E+07	4.4E+07	1.5E+07
0.045-0.07	8.5E+07	7.6E+07	4.3E+07	2.9E+07	2.9E+06
0.07-0.1	4.4E+08	3.9E+08	1.9E+08	1.1E+08	9.6E+06
0.1-0.15	1.8E+09	1.6E+09	7.4E+08	4.0E+08	8.2E+07
0.15-0.3	2.5E+09	2.1E+09	8.6E+08	4.2E+08	1.6E+07
0.3-0.45	1.0E+09	8.9E+08	5.4E+08	3.9E+08	7.5E+07
0.45-0.7	5.4E+09	4.5E+09	2.7E+09	2.2E+09	1.2E+09
0.7-1.0	6.7E+09	5.9E+09	4.2E+09	3.6E+09	2.5E+09
1.0-1.5	2.3E+09	1.5E+09	7.6E+08	5.6E+08	1.8E+08
1.5-2.0	3.3E+09	3.0E+09	2.4E+09	2.1E+09	6.1E+08
2.0-2.5	3.9E+08	3.1E+08	2.4E+08	2.0E+08	7.9E+07
2.5-3.0	1.8E+08	1.7E+08	1.4E+08	1.2E+08	3.5E+07
3.0-4.0	4.6E+06	1.8E+06	1.5E+06	1.3E+06	4.2E+05
4.0-6.0	2.5E+05	1.3E+04	1.6E+01	1.6E+01	1.6E+01
6.0-8.0	2.6E+00	2.6E+00	2.6E+00	2.6E+00	2.6E+00
8.0-11.00	4.1E-01	4.1E-01	4.1E-01	4.1E-01	4.0E-01

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Table 12.2-14 (Sheet 2 of 2)
Spent Fuel Gamma Ray Source Strengths

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)			
	3 Months	6 Months	1 Year	5 Years
0-0.02	3.8E+05	2.4E+05	1.6E+05	6.4E+04
0.02-0.03	1.2E+06	5.2E+05	3.2E+05	1.1E+05
0.03-0.045	6.8E+06	4.6E+06	3.1E+06	6.9E+05
0.045-0.07	3.4E+05	2.2E+05	1.7E+05	9.6E+04
0.07-0.1	2.7E+06	2.1E+06	1.4E+06	2.5E+05
0.1-0.15	3.3E+07	1.6E+07	9.1E+06	1.2E+06
0.15-0.3	1.9E+06	9.6E+05	6.6E+05	3.7E+05
0.3-0.45	6.8E+06	3.7E+06	2.9E+06	9.7E+05
0.45-0.7	7.3E+08	5.3E+08	4.1E+08	1.6E+08
0.7-1.0	1.5E+09	7.9E+08	3.2E+08	6.8E+07
1.0-1.5	7.7E+07	6.2E+07	5.0E+07	1.7E+07
1.5-2.0	3.1E+07	5.2E+06	3.3E+06	5.6E+05
2.0-2.5	2.3E+07	1.6E+07	1.0E+07	3.2E+05
2.5-3.0	1.6E+06	2.4E+05	1.6E+05	1.0E+04
3.0-4.0	6.6E+04	4.4E+04	3.1E+04	2.0E+03
4.0-6.0	1.5E+01	1.4E+01	1.3E+01	1.1E+01
6.0-8.0	2.5E+00	2.3E+00	2.2E+00	1.8E+00
8.0-11.00	3.8E-01	3.6E-01	3.4E-01	2.8E-01

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Table 12.2-15
Irradiated Control Rod and Gray Rod Source Strengths

Energy Group (Mev/gamma)	Silver-Indium-Cadmium Control Rod Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	**1 Week**	**1 Month**
0.20 - 0.40	2.3E+08	2.3E+08	2.2E+08
0.40 - 0.90	1.1E+12	1.1E+12	1.0E+12
0.90 - 1.35	2.0E+11	1.9E+11	1.8E+11
1.35 - 1.80	3.7E+11	3.7E+11	3.4E+11
(Mev/gamma)	**6 Months**	**1 Year**	**5 Years**
0.20 - 0.40	1.4E+08	8.5E+07	1.5E+06
0.40 - 0.90	6.6E+11	4.0E+11	7.1E+09
0.90 - 1.35	1.2E+11	7.2E+10	1.3E+09
1.35 - 1.80	2.3E+11	1.4E+11	2.5E+09

Note: The absorber cross-sectional area is 0.589 and 0.521 square centimeters for the upper and lower portions of the rods, respectively, and the absorber material density is 10.2 grams per cubic centimeter.

Energy Group (Mev/gamma)	Tungsten Gray Rod Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	**1 Week**	**1 Month**
0.00 - 0.25	6.5E+11	2.7E+11	8.3E+10
0.25 - 0.40	1.2E+11	5.2E+10	4.1E+10
0.40 - 0.90	5.1E+11	1.3E+11	1.1E+11
0.90 - 1.35	3.7E+10	7.9E+09	7.4E+09
1.35 - 1.80	9.2E+09	4.5E+07	8.7E+06
1.80 - 2.20	2.0E+09	7.2E+06	4.7E+05
(Mev/gamma)	**6 Months**	**1 Year**	**5 Years**
0.00 - 0.25	8.1E+09	1.7E+09	1.9E+06
0.25 - 0.40	2.1E+10	9.8E+09	2.6E+07
0.40 - 0.90	5.5E+10	2.5E+10	6.8E+07
0.90 - 1.35	5.5E+09	4.3E+09	2.1E+09
1.35 - 1.80	3.2E+06	1.0E+06	2.6E+03
1.80 - 2.20	1.3E+05	5.0E+04	2.1E+04

Note: The absorber cross-sectional area is 0.197 square centimeters and the absorber material density is 19.2 grams per cubic centimeter.

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Table 12.2-15 (continued)
Irradiated Control Rod and Gray Rod Source Strengths

Energy Group (Mev/gamma)	Gray Rod Sleeve (Nickel Alloy 718) Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	1 Week	1 Month
0.00 - 0.25	8.2E+09	2.9E+09	1.3E+09
0.25 - 0.40	1.3E+10	1.1E+10	5.9E+09
0.40 - 0.90	4.1E+11	3.6E+11	2.4E+11
0.90 - 1.35	2.1E+12	2.1E+12	2.1E+12
1.35 - 1.80	8.0E+08	5.6E+08	4.4E+08
1.80 - 2.20	1.9E+08	2.0E+07	2.0E+07
2.20 - 2.60	7.9E+06	4.2E+04	4.1E+04
2.60 - 3.00	1.0E+08	1.0E+05	-

(Mev/gamma)	6 Months	1 Year	5 Years
0.00 - 0.25	9.4E+08	8.4E+08	4.9E+08
0.25 - 0.40	1.8E+08	3.1E+07	1.6E+07
0.40 - 0.90	2.5E+10	5.1E+09	4.5E+08
0.90 - 1.35	2.0E+12	1.8E+12	1.1E+12
1.35 - 1.80	1.0E+08	1.6E+07	4.4E+02
1.80 - 2.20	1.9E+07	1.8E+07	1.0E+07
2.20 - 2.60	3.9E+04	3.7E+04	2.2E+04
2.60 - 3.00	-	-	-

Note: The absorber cross-sectional area is 0.29 square centimeters and the absorber material density is 8.2 grams per cubic centimeter.

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Table 12.2-16
Irradiated SB-BE Secondary Source Rod
Gamma Ray Source Strengths

Energy Group (MeV/gamma)	Source Strength at Time After Shutdown (MeV/cm^3-sec)					
	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.020 - 0.035	5.9E+10	2.4E+10	1.1E+10	2.5E+09	5.7E+08	5.5E+07
0.035 - 0.05	4.2E+10	1.7E+10	7.9E+09	1.4E+09	1.8E+08	6.0E+06
0.05 - 0.075	5.7E+10	2.3E+10	1.1E+10	1.9E+09	2.2E+08	6.0E+05
0.075 - 0.125	1.0E+11	4.0E+10	1.9E+10	3.3E+09	4.0E+08	1.7E+06
0.125 - 0.175	7.3E+10	3.8E+10	2.4E+10	7.8E+09	2.3E+09	1.7E+07
0.175 - 0.250	6.6E+10	2.6E+10	1.2E+10	2.2E+09	3.1E+08	2.0E+07
0.250 - 0.40	1.4E+11	5.8E+10	2.9E+10	5.2E+09	6.6E+08	2.0E+07
0.40 - 0.90	1.3E+13	7.1E+12	4.4E+12	7.8E+11	9.5E+10	1.0E+09
0.90 - 1.35	8.6E+11	5.9E+11	4.0E+11	7.2E+10	8.5E+09	4.3E+02
1.35 - 1.80	7.3E+12	6.8E+12	5.2E+12	9.2E+11	1.1E+11	5.5E+03
1.80 - 2.20	9.8E+11	9.2E+11	7.0E+11	1.3E+11	1.5E+10	7.4E+02
2.20 - 2.60	8.1E+09	7.6E+09	5.8E+09	1.0E+09	1.2E+08	6.1E+00
2.60 - 3.00	9.8E+08	9.2E+08	7.0E+08	1.3E+08	1.5E+07	7.4E-01

Notes:
The Sb-Be material density is 3.56 g/cm^3.
The secondary source rod cross-sectional area is 0.432 cm^2 per rod.
The average neutron energy is 30 kev.

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Table 12.2-17
Irradiated SB-BE Secondary Source Rod
Neutron Source Strengths

Time After Shutdown	Sb-124 Concentration (curies/cm^3)	Neutron Source (N/cm^3-sec)
1 day	2.2E+02	4.4E+08
1 week	2.0E+02	4.1E+08
1 month	1.6E+02	3.1E+08
6 months	2.8E+01	5.6E+07
1 year	3.3E+00	6.6E+06
5 years	1.6E-07	3.3E-01

Notes:
The Sb-Be material density is 3.56 g/cm^3.
The secondary source rod cross-sectional area is 0.432 cm^2 per rod.
The average neutron energy is 30 kev.

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Table 12.2-18
Irradiated Stainless Steel Source Strengths
(0.05 Weight Percent Cobalt)

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	**1 Week**	**1 Month**
0.00 - 0.25	4.7E+08	4.2E+08	3.2E+08
0.25 - 0.40	1.1E+10	9.6E+09	5.4E+09
0.40 - 0.90	3.7E+10	3.5E+10	3.2E+10
0.90 - 1.35	2.3E+11	2.2E+11	2.1E+11
1.35 - 1.80	3.6E+08	1.3E+08	1.0E+08
1.80 - 2.20			
2.20 - 2.60			
(Mev/gamma)	**6 Months**	**1 Year**	**5 Years**
0.00 - 0.25	1.1E+08	7.8E+07	4.4E+07
0.25 - 0.40	1.3E+08	4.2E+06	1.5E+06
0.40 - 0.90	1.8E+10	1.1E+10	4.3E+08
0.90 - 1.35	1.8E+11	1.6E+11	9.7E+10
1.35 - 1.80	2.1E+07	3.2E+06	4.0E+01
1.80 - 2.20	1.7E+06	1.6E+06	9.3E+05
2.20 - 2.60	3.5E+10	3.3E+03	1.9E+03

Notes:

The various cross-section areas per rod are as follows:

– Ag-In-Cd control rod cladding - 0.136 cm^2
– Sb-Be secondary source rod cladding - 0.136 cm^2 (outer) and 0.118 cm^2 (inner)
– Gray rod cladding - 0.184 cm^2

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Table 12.2-19
Irradiated Flux Thimble
Source Strengths

Energy Group (MeV/gamma)	Source Strength at Time After Shutdown (MeV/cm^3-sec)			
	12 Hours	**1 Day**	**70 Hours**	**1 Week**
<0.15	4.0E+09	3.2E+09	2.2E+09	1.2E+09
0.15 - 0.45	7.9E+09	7.2E+09	6.4E+09	5.4E+09
0.45 - 1.0	4.2E+10	2.4E+10	2.2E+10	2.0E+10
1.0 - 1.5	1.2E+11	1.2E+11	1.2E+11	1.2E+11
1.5 - 2.0	1.0E+10	4.8E+08	7.5E+07	7.2E+07
2.0 - 2.5	6.3E+09	2.5E+08	9.9E+05	9.8E+05
2.5 - 3.0	1.1E+09	4.3E+07	1.2E+03	1.0E+03
3.0 - 4.0	1.2E+08	4.7E+06	2.0E+01	–
(MeV/gamma)	**1 Month**	**6 Months**	**1 Year**	**5 Years**
<0.15	5.5E+08	3.6E+08	3.1E+08	1.2E+08
0.15 - 0.45	2.9E+09	7.5E+07	5.1E+06	2.1E+06
0.45 - 1.0	1.7E+10	9.3E+09	5.5E+09	2.1E+08
1.0 - 1.5	1.1E+11	9.7E+10	9.0E+10	5.3E+10
1.5 - 2.0	5.7E+07	1.3E+07	2.2E+06	1.7E+00
2.0 - 2.5	9.8E+05	9.2E+05	8.6E+05	5.1E+05
2.5 - 3.0	1.0E+03	9.6E+02	9.0E+02	5.3E+02
3.0 - 4.0	–	–	–	–

Note:
The flux thimble cross-sectional area is 0.682 cm^2.

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Table 12.2-20
Core Melt Accident Source Strengths
in Containment Atmosphere as a Function of Time

Energy Group (Mev/gamma)	Source Strength at Time After Release (Mev/watt-sec)				
	10 Min.	**40 Min**	**1.97 Hours**	**3.97 Hours**	**5.97 Hours**
<0.175	2.0E+07	2.0E+07	2.4E+08	1.9E+08	1.6E+08
0.175 - 0.4	4.0E+07	3.9E+07	3.9E+08	2.5E+08	1.7E+08
0.4 - 0.9	3.5E+08	3.1E+08	1.6E+09	6.2E+08	2.7E+08
0.9 - 1.35	1.8E+08	1.5E+08	7.5E+08	3.1E+08	1.4E+08
1.35 - 1.8	1.6E+08	1.4E+08	6.0E+08	2.3E+08	1.1E+08
1.8 - 2.2	6.3E+07	5.3E+07	5.0E+08	2.9E+08	1.7E+08
2.2 - 2.6	7.8E+07	7.7E+07	7.1E+08	3.9E+08	2.3E+08
2.6 - 3.5	2.8E+07	1.8E+07	7.4E+07	3.3E+07	1.9E+07
3.5 - 5.0	7.1E+06	1.9E+06	8.5E+06	4.5E+06	2.7E+06
5.0 - 10.0	2.1E+05	2.1E+03	3.0E+04	1.9E+04	1.2E+04
Beta	4.3E+08	3.6E+08	2.9E+09	1.7E+09	1.1E+09
(Mev/gamma)	**11.97 Hours**	**1 Day**	**1 Week**	**1 Month**	**1 Year**
<0.175	1.3E+08	1.1E+08	4.9E+07	2.5E+06	7.7E+04
0.175 - 0.4	8.2E+07	3.9E+07	7.0E+06	9.5E+05	2.4E+04
0.4 - 0.9	5.4E+07	2.8E+07	4.0E+06	1.9E+06	1.3E+06
0.9 - 1.35	2.6E+07	7.3E+06	3.2E+05	8.4E+04	2.0E+04
1.35 - 1.8	2.2E+07	3.8E+06	7.1E+05	2.3E+05	2.6E+04
1.8 - 2.2	3.8E+07	2.5E+06	1.6E+04	6.1E+02	1.1E+02
2.2 - 2.6	5.1E+07	2.9E+06	4.7E+04	1.4E+04	2.7E+01
2.6 - 3.5	4.2E+06	2.2E+05	1.2E+03	3.8E+02	5.7E+00
3.5 - 5.0	6.2E+05	3.3E+04	1.0E-03	9.7E-04	5.2E-04
5.0 - 10.0	2.7E+03	1.5E+02	6.3E-06	6.1E-06	4.6E-06
Beta	5.6E+08	3.7E+08	1.5E+08	1.2E+07	3.6E+06

Note:
No release from core until 10 minutes after incident (see Subsection 15.6.5.3).

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Table 12.2-21
Core Melt Accident Integrated Source Strengths in Containment Atmosphere

Energy Group (Mev/gamma)	Source Strength at Time After Release (Mev/watt)			
	10 Min.	**40 Min**	**1.97 Hours**	**3.97 Hours**
<0.175	2.0E+07	3.5E+10	6.6E+11	2.2E+12
0.175 - 0.4	4.0E+07	7.0E+10	1.2E+12	3.4E+12
0.4 - 0.9	3.5E+08	5.9E+11	6.3E+12	1.4E+13
0.9 - 1.35	1.8E+08	2.9E+11	2.8E+12	6.3E+12
1.35 - 1.8	1.6E+08	2.8E+11	2.4E+12	5.1E+12
1.8 - 2.2	6.3E+07	1.0E+11	1.5E+12	4.2E+12
2.2 - 2.6	7.8E+07	1.4E+11	2.2E+12	6.0E+12
2.6 - 3.5	2.8E+07	3.9E+10	2.9E+11	6.4E+11
3.5 - 5.0	7.1E+06	5.7E+09	3.3E+10	7.7E+10
5.0 - 10.0	2.1E+05	5.2E+07	1.2E+08	3.0E+08
Beta	4.3E+08	6.9E+11	9.0E+12	2.5E+13

(Mev/gamma)	11.97 Hours	1 Day	1 Week	1 Month	1 Year
<0.175	6.4E+12	1.1E+13	5.0E+13	8.1E+13	8.6E+13
0.175 - 0.4	7.5E+12	1.0E+13	1.7E+13	2.3E+13	2.4E+13
0.4 - 0.9	1.9E+13	2.1E+13	2.6E+13	3.1E+13	7.4E+13
0.9 - 1.35	9.2E+12	9.9E+12	1.1E+13	1.1E+13	1.2E+13
1.35 - 1.8	7.4E+12	7.9E+12	8.4E+12	9.3E+12	1.1E+13
1.8 - 2.2	7.7E+12	8.2E+12	8.4E+12	8.4E+12	8.4E+12
2.2 - 2.6	1.1E+13	1.1E+13	1.2E+13	1.2E+13	1.2E+13
2.6 - 3.5	1.0E+12	1.1E+12	1.1E+12	1.1E+12	1.1E+12
3.5 - 5.0	1.3E+11	1.4E+11	1.4E+11	1.4E+11	1.4E+11
5.0 - 10.0	5.4E+08	5.8E+08	5.8E+08	5.8E+08	5.8E+08
Beta	5.2E+13	7.2E+13	1.9E+14	3.0E+14	4.2E+14

Note:
No release from core until 10 minutes after incident (see Subsection 15.6.5.3)

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Table 12.2-22
Parameters and Assumptions Used for Calculating
Containment Airborne Radioactivity Concentrations

Parameter/Assumption	Value
Reactor coolant leakage rate	30 lb/day
Time used to estimate equilibrium concentration	100 days
Containment free air volume	2.06E6 cu. ft
Flashing fraction	0.40
Fuel defects	0.250%
Reactor coolant tritium concentration	3.5 μCi/g
Normal operation purge flow rate	4,000 cfm
Normal operation purge duration	20 hrs/week
Shutdown purge flow rate	8,000 cfm

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Table 12.2-23 (Sheet 1 of 3)
Containment Airborne Radioactivity Concentrations
($\mu Ci/cm^3$)

Isotope	Equilibrium Activity (no purge)	Maximum Activity (with normal purge)	Shutdown Activity (shutdown purge for 24 hrs)
Cr-51	1.0E-11	9.1E-12	2.1E-13
Mn-54	5.1E-12	4.6E-12	1.1E-13
Mn-56	0.0E+00	0.0E+00	0.0E+00
Fe-55	3.9E-12	3.5E-12	8.2E-14
Fe-59	9.7E-13	8.8E-13	2.0E-14
Co-58	1.5E-11	1.4E-11	3.2E-13
Co-60	1.7E-12	1.6E-12	3.6E-14
Br-83	0.0E+00	0.0E+00	0.0E+00
Br-84	3.4E-11	3.2E-11	2.5E-26
Br-85	0.0E+00	0.0E+00	0.0E+00
Kr-83m	0.0E+00	0.0E+00	0.0E+00
Kr-85m	1.3E-08	7.9E-09	2.2E-10
Kr-85	3.3E-05	4.1E-07	6.5E-08
Kr-87	3.4E-09	2.9E-09	1.5E-14
Kr-88	1.4E-08	1.0E-08	4.4E-11
Kr-89	0.0E+00	0.0E+00	0.0E+00
Rb-88	2.8E-10	2.7E-10	4.1E-36
Rb-89	0.0E+00	0.0E+00	0.0E+00
Sr-89	4.7E-13	4.2E-13	9.7E-15
Sr-90	3.9E-14	3.5E-14	8.2E-16
Sr-91	3.6E-12	3.2E-12	1.4E-14
Sr-92	0.0E+00	0.0E+00	0.0E+00
Y-90	4.6E-15	4.2E-15	7.6E-17
Y-91m	1.2E-12	1.1E-12	6.8E-23
Y-91	1.7E-14	1.6E-14	3.6E-16
Y-92	0.0E+00	0.0E+00	0.0E+00
Y-93	1.6E-11	1.4E-11	6.9E-14

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Table 12.2-23 (Sheet 2 of 3)
Containment Airborne Radioactivity Concentrations
($\mu Ci/cm^3$)

Isotope	Equilibrium Activity (no purge)	Maximum Activity (with normal purge)	Shutdown Activity (shutdown purge for 24 hrs)
Zr-95	1.3E-12	1.2E-12	2.7E-14
Nb-95	9.3E-13	8.4E-13	1.9E-14
Mo-99	2.2E-11	2.0E-11	3.6E-13
Tc-99m	1.8E-11	1.6E-11	2.5E-14
Ru-103	2.5E-11	2.2E-11	5.1E-13
Ag-110m	4.3E-12	3.9E-12	9.0E-14
Te-127m	0.0E+00	0.0E+00	0.0E+00
Te-129m	6.2E-13	5.6E-13	1.3E-14
Te-129	7.1E-11	6.6E-11	1.1E-18
Te-131m	5.3E-12	4.8E-12	6.5E-14
Te-131	1.4E-11	1.4E-11	2.1E-30
Te-132	5.8E-12	5.2E-12	9.9E-14
TE-134	0.0E+00	0.0E+00	0.0E+00
I-129	0.0E+00	0.0E+00	0.0E+00
I-130	0.0E+00	0.0E+00	0.0E+00
I-131	2.9E-09	1.4E-10	8.8E-12
I-132	6.5E-09	6.9E-10	3.7E-14
I-133	8.9E-09	4.8E-10	1.5E-11
I-134	4.1E-09	8.6E-10	4.2E-19
I-135	1.4E-08	9.0E-10	5.2E-12
Xe-131m	4.4E-06	2.7E-07	4.8E-08
Xe-133m	6.9E-08	1.3E-08	3.1E-09
Xe-133	6.4E-06	7.1E-07	1.5E-07
Xe-135m	6.2E-10	6.0E-10	5.8E-37
Xe-135	1.4E-07	6.1E-08	8.0E-09
Xe-137	3.9E-11	3.9E-11	0.0E+00
Xe-138	5.0E-10	4.8E-10	0.0E+00

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Table 12.2-23 (Sheet 3 of 3)
Containment Airborne Radioactivity Concentrations
($\mu Ci/cm^3$)

Isotope	Equilibrium Activity (no purge)	Maximum Activity (with normal purge)	Shutdown Activity (shutdown purge for 24 hrs)
Cs-134	2.3E-11	2.1E-11	4.8E-13
Cs-136	2.9E-12	2.6E-12	5.8E-14
Cs-137	3.1E-11	2.8E-11	6.5E-13
Cs-138	0.0E+00	0.0E+00	0.0E+00
Ba-137m	1.7E-12	1.7E-12	0.0E+00
Ba-140	4.3E-11	3.9E-11	8.5E-13
La-140	8.8E-11	8.0E-11	1.2E-12
Ce-141	5.1E-13	4.6E-13	1.0E-14
Ce-143	9.9E-12	9.0E-12	1.3E-13
Pr-143	1.2E-11	1.1E-11	2.3E-13
Ce-144	1.3E-11	1.2E-11	2.8E-13
Pr-144	3.9E-12	3.8E-12	0.0E+00
H-3	9.4E-06	1.2E-07	1.7E-09
Ar-41	3.8E-07	2.9E-07	2.2E-13
Na-24	1.7E-10	1.6E-10	1.2E-12
Zn-65	1.7E-12	1.5E-12	3.5E-14
Ru-106	2.9E-10	2.6E-10	6.1E-12
Rh-103m	1.4E-11	1.3E-11	7.3E-21
W-187	8.7E-12	7.9E-12	9.3E-14
Np-239	7.7E-12	7.0E-12	1.2E-13
Total	5.4E-05	1.9E-06	2.7E-07
Iodines	3.6E-08	3.1E-09	2.9E-11
Particulates	1.3E-09	1.2E-09	1.4E-11
Noble Gases	4.4E-05	1.8E-06	2.7E-07

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Table 12.2-24
Parameters and Assumptions Used for Calculating
Fuel Handling Area Airborne Radioactivity Concentrations

Parameter/Assumption	Value
Assumed fuel load	Full core offload
Ventilation flow through fuel handling area(1)	9,500 cfm(2)
Fuel handling area free air volume	284,000 ft^3
Fuel defects	0.25%
Time from shutdown to reactor vessel head removal	48 hours
Refueling time	10 days
Spent fuel pool purification flow rate	150 gpm
Decontamination factors of mixed-bed demineralizer for spent fuel pool purification system:	
Iodines	100
Cs and Rb	2
Others	50
Spent fuel pool temperature	120°F
Partition coefficients:	
Noble gases	Freely released
Tritium	1
Halogens	100
Others	1,000
Evaporation rate of spent fuel pool water	430 lbs/hr
Reactor coolant tritium concentration	3.5 µCi/g

Notes:
1. This flow rate is defined as the sum of the fuel handling area exhaust flow from room 12562 (fuel handling area) and exhaust flow from room 12562 to room 12471 (solid radwaste system valve/piping area).
2. This is the nominal expected ventilation flow rate. For conservatism, the calculated airborne radioactivity concentrations are based on a 10% lower flow rate.

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Table 12.2-25 (Sheet 1 of 2)
Fuel Handling Area Airborne Radioactivity Concentrations(1)
($\mu Ci/cm^3$)

Isotope	Activity(2)
Cr-51	2.2E-12
Mn-54	1.2E-12
Fe-55	8.7E-13
Fe-59	2.2E-13
Co-58	3.3E-12
Co-60	3.8E-13
Br-83	1.9E-16
Kr-85m	4.0E-12
Kr-85	4.6E-10
Kr-87	1.6E-20
Kr-88	1.2E-13
Rb-88	1.2E-13
Sr-89	1.9E-12
Sr-90	8.6E-12
Sr-91	6.9E-14
Sr-92	8.1E-19
Y-90	8.7E-14
Y-91	2.4E-13
Y-92	2.5E-17
Y-93	5.8E-15
Zr-95	2.7E-13
Nb-95	2.7E-13
Mo-99	2.1E-10
Tc-99m	1.9E-10
Ru-103	2.4E-13
Ag-110m	7.0E-13
Te-127m	1.3E-12
Te-129m	4.3E-12
Te-131m	3.9E-12
Te-131	8.7E-13
Te-132	8.7E-11
I-129	2.6E-16
I-130	1.1E-11
I-131	1.0E-08
I-132	2.7E-15
I-133	4.1E-09
I-135	5.9E-11
Xe-131m	4.1E-10
Xe-133m	1.2E-09
Xe-133	6.0E-08
Xe-135	4.1E-10

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Table 12.2-25 (Sheet 2 of 2)
Fuel Handling Area Airborne Radioactivity Concentrations(1)
(μCi/cm^3)

Isotope	Activity(2)
Cs-134	1.2E-09
Cs-136	1.6E-09
Cs-137	8.9E-10
Ba-137m	8.4E-10
Ba-140	1.6E-12
La-140	2.2E-13
Ce-141	2.7E-13
Ce-143	8.3E-14
Pr-143	2.4E-13
Ce-144	2.1E-13
Pr-144	2.1E-13
H-3	6.4E-06
Rb-86	4.6E-18
Rh-103m	2.3E-13
Ru-106	1.3E-19
Rh-106	1.3E-19
Np-239	2.7E-18
Sb-127	1.1E-17
Sb-129	1.5E-20
Rh-105	8.6E-20
Zr-97	5.0E-20
Nd-147	1.5E-19
Pu-241	2.9E-20
Total (excluding tritium)	8.2E-08
Iodines	1.4E-08
Particulates	5.1E-09
Noble Gases	6.3E-08

Notes:

1. The maximum activity concentration is calculated to occur 2 hours after removal of the head, or 50 hours after shutdown in this case.
2. The following nuclides are expected to exist in the FHA at the time of maximum airborne concentrations with individual nuclide activity concentrations less than 1.0E-20 μCi/cm^3:
^{91m}Y, ^{129}Te, ^{134}Te, ^{134}I, ^{139}Ba, ^{105}Ru, ^{141}La, ^{142}La, ^{241}Am, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Cm, and ^{244}Cm.

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Table 12.2-26
Parameters and Assumptions Used for Calculating Auxiliary Building Airborne Radioactivity Concentrations

Parameter/Assumption	Value
Ventilation exhaust flow(1)	25,320 cfm(2)
Free air volume	411,200 ft^3
Primary coolant leakage to auxiliary building(3)	296 lb/day
Flashing fraction	
Noble gases and Tritium	1
Other Radionuclides	0.1
Primary coolant source term	See Table 11.1-2.
Fuel defects	0.25%

Notes:

1. This flow rate is defined as the sum of the auxiliary/annex building exhaust fan flow minus the annex building exhaust flows from room 40357 (containment access corridor), room 40551 (containment air filtration exhaust room A), and 40552 (containment air filtration exhaust room B).
2. This is the nominal expected ventilation flow rate. For conservatism, the calculated airborne radioactivity concentrations are based on a 10% lower flow rate.
3. This value represents the total leakage of primary coolant to the auxiliary building from equipment leakage (128 lb/day) and from primary sampling activities (168 lb/day). As indicated in ANSI/ANS 55.6-1993, only 5% of the 168 lb/day leakage from primary sampling activities can be assumed to contain design basis RCS source term activity. This decreases the total primary coolant leakage to the auxiliary building with design basis RCS source term activity used for calculating auxiliary building radioactivity concentrations to 136 lb/day.

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Table 12.2-27 (Sheet 1 of 2)
Auxiliary Building Airborne Radioactivity Concentrations
($\mu Ci/cm^3$)

Isotope	Activity
Cr-51	8.6E-12
Mn-54	4.5E-12
Mn-56	1.1E-09
Fe-55	3.3E-12
Fe-59	8.6E-13
Co-58	1.3E-11
Co-60	1.5E-12
Br-83	2.0E-10
Br-84	8.1E-11
Br-85	2.5E-12
Kr-83m	1.1E-08
Kr-85m	5.3E-08
Kr-85	2.0E-07
Kr-87	2.7E-08
Kr-88	9.3E-08
Kr-89	4.7E-10
Rb-88	5.8E-09
Rb-89	2.5E-10
Sr-89	7.3E-12
Sr-90	3.3E-13
Sr-91	1.1E-11
Sr-92	2.5E-12
Y-90	8.6E-14
Y-91m	4.9E-12
Y-91	9.3E-13
Y-92	2.1E-12
Y-93	7.2E-13
Zr-95	1.1E-12
Nb-95	1.1E-12
Mo-99	1.4E-09
Tc-99m	1.3E-09
Ru-103	9.3E-13
Rh-103m	7.6E-13
Ag-110m	2.7E-12
Te-127m	5.1E-12
Te-129m	1.7E-11
Te-129	2.1E-11
Te-131m	4.4E-11

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Table 12.2-27 (Sheet 2 of 2)
Auxiliary Building Airborne Radioactivity Concentrations
($\mu Ci/cm^3$)

Isotope	Activity
Te-131	1.9E-11
Te-132	5.2E-10
Te-134	5.6E-11
I-129	1.0E-16
I-130	7.2E-11
I-131	4.7E-09
I-132	5.7E-09
I-133	8.6E-09
I-134	1.2E-09
I-135	5.0E-09
Xe-131m	8.6E-08
Xe-133m	1.1E-07
Xe-133	8.0E-06
Xe-135m	6.2E-09
Xe-135	2.3E-07
Xe-137	1.0E-09
Xe-138	8.8E-09
Cs-134	4.6E-09
Cs-136	6.6E-09
Cs-137	3.3E-09
Cs-138	1.8E-09
Ba-137m	5.3E-10
Ba-140	6.6E-12
La-140	2.1E-12
Ce-141	1.1E-12
Ce-143	9.3E-13
Pr-143	1.0E-12
Ce-144	8.0E-13
Pr-144	4.6E-13
H-3	2.3E-07
Total	9.1E-06
Iodines	2.5E-08
Particulates	2.8E-08
Noble Gases	8.8E-06

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Table 12.2-28 (Sheet 1 of 2)
Core Melt Accident Source Strengths from MCR HVAC Filters as a Function of Time

VES Filter[1] Source Strengths after a Loss of Coolant Accident				
Energy Group (Mev/gamma)	Source Strength (Mev/sec)			
	2 hours	8 hours	24 hours	30 days
0.01 - 0.02	1.19E+06	3.11E+06	1.81E+06	1.97E+05
0.02 - 0.03	1.47E+06	5.26E+06	3.89E+06	2.65E+05
0.03 - 0.06	2.87E+06	8.30E+06	5.46E+06	6.47E+05
0.06 - 0.1	3.03E+06	8.13E+06	5.22E+06	5.41E+05
0.1 - 0.2	5.76E+06	1.41E+07	8.76E+06	9.02E+05
0.2 - 0.4	6.14E+07	2.61E+08	2.46E+08	1.87E+07
0.4 - 0.6	1.86E+08	6.02E+08	3.60E+08	1.83E+07
0.6 - 0.7	1.47E+08	2.33E+08	1.47E+08	1.03E+08
0.7 - 0.8	1.09E+08	1.80E+08	1.05E+08	7.30E+07
0.8 - 1.0	1.85E+08	1.67E+08	6.99E+07	7.13E+06
1.0 - 1.5	3.36E+08	6.99E+08	1.85E+08	1.22E+07
1.5 - 2.0	1.21E+08	2.55E+08	4.97E+07	2.69E+04
2.0 - 3.0	3.13E+07	3.87E+07	7.28E+06	9.07E+03
3.0 - 4.0	3.68E+05	5.98E+03	5.56E+02	1.41E+02
4.0 - 5.0	1.42E+04	3.16E+01	8.55E-04	7.80E-04
5.0 - 6.0	3.31E-05	3.12E-04	3.35E-04	3.21E-04
6.0 - 7.0	1.32E-05	1.24E-04	1.33E-04	1.28E-04
7.0 - 8.0	5.11E-06	4.82E-05	5.17E-05	4.96E-05
8.0 - 10.0	2.68E-06	2.53E-05	2.71E-05	2.60E-05
10.0 - 14.0	1.69E-07	1.60E-06	1.71E-06	1.64E-06
Total	1.19E+09	2.47E+09	1.19E+09	2.35E+08

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Table 12.2-28 (Sheet 2 of 2)
Core Melt Accident Source Strengths from MCR HVAC Filters as a Function of Time

VBS Filter[2] Source Strengths after a Loss of Coolant Accident				
Energy Group (Mev/gamma)	Source Strength (Mev/sec)			
	2 hours	8 hours	24 hours	30 days
0.01 - 0.02	6.86E+08	1.00E+09	5.75E+08	6.21E+07
0.02 - 0.03	9.55E+08	1.76E+09	1.27E+09	8.46E+07
0.03 - 0.06	1.71E+09	2.71E+09	1.75E+09	2.10E+08
0.06 - 0.1	1.72E+09	2.60E+09	1.63E+09	1.70E+08
0.1 - 0.2	3.49E+09	4.61E+09	2.81E+09	2.91E+08
0.2 - 0.4	3.54E+10	8.45E+10	7.59E+10	5.76E+09
0.4 - 0.6	1.03E+11	1.91E+11	1.10E+11	5.61E+09
0.6 - 0.7	7.99E+10	7.20E+10	4.39E+10	3.04E+10
0.7 - 0.8	5.97E+10	5.62E+10	3.17E+10	2.16E+10
0.8 - 1.0	1.03E+11	5.23E+10	2.13E+10	2.11E+09
1.0 - 1.5	1.86E+11	2.20E+11	5.64E+10	3.62E+09
1.5 - 2.0	6.71E+10	8.03E+10	1.53E+10	8.78E+06
2.0 - 3.0	1.66E+10	1.22E+10	2.24E+09	3.09E+06
3.0 - 4.0	1.82E+08	1.93E+06	1.89E+05	4.81E+04
4.0 - 5.0	6.86E+06	7.65E+03	2.91E-01	2.65E-01
5.0 - 6.0	3.74E-02	1.12E-01	1.14E-01	1.09E-01
6.0 - 7.0	1.49E-02	4.47E-02	4.54E-02	4.35E-02
7.0 - 8.0	5.78E-03	1.74E-02	1.76E-02	1.69E-02
8.0 - 10.0	3.03E-03	9.11E-03	9.24E-03	8.86E-03
10.0 - 14.0	1.92E-04	5.75E-04	5.84E-04	5.60E-04
Total	6.59E+11	7.82E+11	3.65E+11	7.00E+10

Notes:
1. Based upon a particulate filter density of 0.212 g/cc and charcoal filter density of 0.440 g/cc.
2. Based upon a particulate filter density of 0.230 g/cc and charcoal filter density of 0.632 g/cc.

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Table 12.2-29
Core Melt Accident Integrated Source Strengths from MCR HVAC Filters

Energy Group (Mev/gamma)	30-Day Integrated Source Strength (Mev)	
	VES[1]	VBS[2]
0.01 - 0.02	1.75E+08	5.65E+10
0.02 - 0.03	3.81E+08	1.26E+11
0.03 - 0.06	5.89E+08	1.90E+11
0.06 - 0.1	5.77E+08	1.84E+11
0.1 - 0.2	9.03E+08	2.95E+11
0.2 - 0.4	3.34E+10	1.05E+13
0.4 - 0.6	2.36E+10	7.44E+12
0.6 - 0.7	3.81E+10	1.15E+13
0.7 - 0.8	2.63E+10	7.92E+12
0.8 - 1.0	7.57E+09	2.39E+12
1.0 - 1.5	1.77E+10	5.67E+12
1.5 - 2.0	4.03E+09	1.34E+12
2.0 - 3.0	6.47E+08	2.18E+11
3.0 - 4.0	1.20E+06	4.46E+08
4.0 - 5.0	4.17E+04	1.52E+07
5.0 - 6.0	1.03E-01	3.51E+01
6.0 - 7.0	4.08E-02	1.40E+01
7.0 - 8.0	1.59E-02	5.42E+00
8.0 - 10.0	8.32E-03	2.84E+00
10.0 - 14.0	5.25E-04	1.80E-01
Total	1.54E+11	4.79E+13

Notes:
1. Based upon a particulate filter density of 0.212 g/cc and charcoal filter density of 0.440 g/cc.
2. Based upon a particulate filter density of 0.230 g/cc and charcoal filter density of 0.632 g/cc.

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Table 12.2-30 (Sheet 1 of 6)
IRWST Design Basis Source Strengths and Specific Activity

Vapor Specific Activity	
Nuclide	**Concentration (μCi/cm^3)**
Kr-83m	3.0E-04
Kr-85m	3.5E-03
Kr-85	2.4E+01
Kr-87	5.5E-04
Kr-88	3.9E-03
Kr-89	1.7E-06
Xe-131m	3.4E-01
Xe-133m	8.1E-02
Xe-133	1.4E+01
Xe-135m	4.1E-05
Xe-135	2.9E-02
Xe-137	3.9E-06
Xe-138	5.4E-05

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Table 12.2-30 (Sheet 2 of 6)
IRWST Design Basis Source Strengths and Specific Activity

Plateout Specific Activity	
Nuclide	**Concentration ($\mu Ci/cm^2$)**
Zr-95	2.4E-01
Ru-103	2.4E-01
Ce-141	1.5E-02
Ce-144	1.3E-02
Cr-51	8.6E-03
Mn-54	7.6E-01
Mn-56	3.2E-05
Fe-55	2.3E-01
Fe-59	2.4E-01
Co-58	5.4E+00
Co-60	3.1E+00

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Table 12.2-30 (Sheet 3 of 6)
IRWST Design Basis Source Strengths and Specific Activity

Liquid Specific Activity			
Nuclide	**Concentration ($\mu Ci/cm^3$)**	**Nuclide**	**Concentration ($\mu Ci/cm^3$)**
Br-83	4.1E-06	Te-134	4.2E-07
Br-84	5.0E-07	I-129	5.8E-11
Br-85	5.3E-09	I-130	6.4E-06
Rb-88	2.5E-05	I-131	2.1E-03
Rb-89	9.6E-07	I-132	1.2E-04
Sr-89	4.0E-06	I-133	1.2E-03
Sr-90	1.9E-07	I-134	1.1E-05
Sr-91	7.9E-07	I-135	2.6E-04
Sr-92	5.9E-08	Cs-134	4.8E-03
Y-90	1.9E-07	Cs-136	5.0E-03
Y-91m	4.2E-08	Cs-137	3.5E-03
Y-91	5.2E-07	Cs-138	1.1E-05
Y-92	6.4E-08	Ba-137m	3.5E-03
Y-93	5.5E-08	Ba-140	3.2E-06
Zr-95	1.8E-07	La-140	4.4E-07
Nb-95	5.7E-07	Ce-141	1.8E-07
Mo-99	4.0E-04	Ce-143	1.8E-07
Tc-99m	6.1E-05	Ce-144	1.4E-07
Ru-103	1.6E-07	Pr-143	4.8E-07
Rh-103m	7.2E-09	Pr-144	1.9E-09
Rh-106	2.1E-11	N-16	2.8E-05
Ag-110m	1.6E-06	Cr-51	8.5E-06
Sb-124	3.3E-06	Mn-54	4.4E-06
Te-127m	2.9E-06	Mn-56	2.2E-05
Te-129m	9.2E-06	Fe-55	3.4E-06
Te-129	2.4E-07	Fe-59	8.2E-07
Te-131m	7.8E-06	Co-58	1.6E-04
Te-131	9.8E-08	Co-60	1.5E-06
Te-132	1.7E-04	-	-

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Table 12.2-30 (Sheet 4 of 6)
IRWST Design Basis Source Strengths and Specific Activity

Vapor Source Strength	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.01 - 0.02	8.4E+02
0.02 - 0.03	2.0E+03
0.03 - 0.06	8.6E+03
0.06 - 0.1	1.8E+04
0.1 - 0.2	2.0E+03
0.2 - 0.4	1.4E+03
0.4 - 0.6	2.1E+03
0.6 - 0.7	2.4E+01
0.7 - 0.8	1.8E+00
0.8 - 1.0	2.2E+01
1.0 - 1.5	1.5E+01
1.5 - 2.0	4.3E+01
2.0 - 3.0	2.1E+02
3.0 - 4.0	3.7E-01
4.0 - 5.0	2.8E-03
5.0 - 6.0	4.0E-08
6.0 - 7.0	0.0E+00
7.0 - 8.0	0.0E+00
8.0 - 10.0	0.0E+00
10.0 - 14.0	0.0E+00
Total	3.6E+04

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Table 12.2-30 (Sheet 5 of 6)
IRWST Design Basis Source Strengths and Specific Activity

Liquid Source Strength	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.01 - 0.02	3.8E-01
0.02 - 0.03	4.8E-01
0.03 - 0.06	1.7E+00
0.06 - 0.1	2.3E+00
0.1 - 0.2	6.8E+00
0.2 - 0.4	5.9E+01
0.4 - 0.6	5.0E+01
0.6 - 0.7	1.9E+02
0.7 - 0.8	1.4E+02
0.8 - 1.0	1.5E+02
1.0 - 1.5	2.1E+02
1.5 - 2.0	5.7E+00
2.0 - 3.0	1.5E+00
3.0 - 4.0	6.8E-02
4.0 - 5.0	2.1E-02
5.0 - 6.0	9.1E-03
6.0 - 7.0	4.2E+00
7.0 - 8.0	3.6E-01
8.0 - 10.0	7.4E-03
10.0 - 14.0	4.8E-06
Total	8.3E+02

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Table 12.2-30 (Sheet 6 of 6)
IRWST Design Basis Source Strengths and Specific Activity

Plateout Source Strength	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^2-sec)**
0.01 - 0.02	1.0E+02
0.02 - 0.03	8.7E+01
0.03 - 0.06	2.0E+02
0.06 - 0.1	2.0E+02
0.1 - 0.2	3.9E+02
0.2 - 0.4	2.9E+02
0.4 - 0.6	3.5E+04
0.6 - 0.7	3.4E+02
0.7 - 0.8	6.5E+03
0.8 - 1.0	1.9E+05
1.0 - 1.5	3.0E+05
1.5 - 2.0	1.8E+03
2.0 - 3.0	3.2E+00
3.0 - 4.0	6.7E-03
4.0 - 5.0	0.0E+00
5.0 - 6.0	0.0E+00
6.0 - 7.0	0.0E+00
7.0 - 8.0	0.0E+00
8.0 - 10.0	0.0E+00
10.0 - 14.0	0.0E+00
Total	5.4E+05

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Table 12.2-31 (Sheet 1 of 6)
IRWST Expected Source Strengths and Specific Activity

Vapor Specific Activity	
Nuclide	**Concentration (μCi/cm^3)**
Kr-85m	7.7E-06
Kr-85	9.9E-02
Kr-87	2.0E-06
Kr-88	8.4E-06
Xe-131m	2.6E-03
Xe-133m	4.0E-05
Xe-133	3.8E-03
Xe-135m	3.7E-07
Xe-135	8.2E-05
Xe-137	2.3E-08
Xe-138	2.9E-07

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Table 12.2-31 (Sheet 2 of 6)
IRWST Expected Source Strengths and Specific Activity

Plateout Specific Activity	
Nuclide	**Concentration ($\mu Ci/cm^2$)**
Zr-95	6.0E-03
Ru-103	6.1E-03
Ce-141	3.6E-04
Ce-144	1.2E-03
Cr-51	3.4E-05
Mn-54	1.7E-02
Fe-55	9.1E-04
Fe-59	6.0E-03
Co-58	1.3E-01
Co-60	7.4E-02
Zn-65	1.2E-01

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Table 12.2-31 (Sheet 3 of 6)
IRWST Expected Source Strengths and Specific Activity

Liquid Specific Activity			
Nuclide	**Concentration (μCi/cm^3)**	**Nuclide**	**Concentration (μCi/cm^3)**
Br-84	5.3E-09	I-133	1.4E-06
Rb-88	3.6E-08	I-134	1.9E-07
Sr-89	2.5E-08	I-135	9.1E-07
Sr-90	3.2E-09	Cs-134	1.9E-06
Sr-91	4.6E-09	Cs-136	7.8E-08
Y-90	3.2E-09	Cs-137	2.6E-06
Y-91m	2.4E-10	Ba-137m	2.6E-06
Y-91	9.6E-10	Ba-140	1.2E-06
Y-93	2.2E-08	La-140	4.4E-07
Zr-95	4.4E-09	Ce-141	1.7E-09
Nb-95	4.4E-08	Ce-143	4.1E-08
Mo-99	1.7E-07	Ce-144	4.6E-08
Tc-99m	1.6E-08	Pr-143	3.3E-07
Ru-103	8.3E-08	Pr-144	4.9E-10
Ru-106	3.1E-10	W-187	5.1E-07
Rh-103m	3.0E-09	Np-239	5.4E-07
Rh-106	3.1E-10	N-16	4.0E-08
Ag-110m	3.2E-07	Na-24	8.5E-06
Sb-124	6.6E-08	Cr-51	3.4E-08
Te-129m	2.9E-08	Mn-54	1.8E-08
Te-129	1.7E-08	Fe-55	1.4E-08
Te-131m	2.0E-08	Fe-59	3.3E-09
Te-131	2.0E-09	Co-58	6.1E-07
Te-132	5.2E-08	Co-60	6.0E-09
I-131	3.0E-06	Zn-65	5.8E-09
I-132	2.9E-07	-	-

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Table 12.2-31 (Sheet 4 of 6)
IRWST Expected Source Strengths and Specific Activity

Vapor Source Strength	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.01 - 0.02	3.0E+00
0.02 - 0.03	3.7E+00
0.03 - 0.06	8.3E+00
0.06 - 0.1	9.9E+00
0.1 - 0.2	7.6E+00
0.2 - 0.4	4.9E+00
0.4 - 0.6	8.6E+00
0.6 - 0.7	7.0E-02
0.7 - 0.8	4.3E-03
0.8 - 1.0	5.1E-02
1.0 - 1.5	3.4E-02
1.5 - 2.0	9.7E-02
2.0 - 3.0	4.6E-01
3.0 - 4.0	1.3E-03
4.0 - 5.0	1.2E-08
5.0 - 6.0	0.0E+00
6.0 - 7.0	0.0E+00
7.0 - 8.0	0.0E+00
8.0 - 10.0	0.0E+00
10.0 - 14.0	0.0E+00
Total	4.7E+01

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Table 12.2-31 (Sheet 5 of 6)
IRWST Expected Source Strengths and Specific Activity

Liquid Source Strength	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.01 - 0.02	1.3E-03
0.02 - 0.03	1.2E-03
0.03 - 0.06	2.7E-03
0.06 - 0.1	2.9E-03
0.1 - 0.2	6.2E-03
0.2 - 0.4	4.3E-02
0.4 - 0.6	5.9E-02
0.6 - 0.7	1.3E-01
0.7 - 0.8	6.7E-02
0.8 - 1.0	6.1E-02
1.0 - 1.5	4.9E-01
1.5 - 2.0	4.5E-02
2.0 - 3.0	8.7E-01
3.0 - 4.0	8.1E-04
4.0 - 5.0	3.1E-05
5.0 - 6.0	1.4E-05
6.0 - 7.0	6.0E-03
7.0 - 8.0	5.1E-04
8.0 - 10.0	1.1E-05
10.0 - 14.0	6.9E-09
Total	1.8E+00

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Table 12.2-31 (Sheet 6 of 6)
IRWST Expected Source Strengths and Specific Activity

Plateout Source Strength	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^2-sec)**
0.01 - 0.02	2.5E+00
0.02 - 0.03	2.2E+00
0.03 - 0.06	5.0E+00
0.06 - 0.1	4.8E+00
0.1 - 0.2	1.0E+01
0.2 - 0.4	6.8E+00
0.4 - 0.6	8.6E+02
0.6 - 0.7	8.7E+00
0.7 - 0.8	1.7E+02
0.8 - 1.0	4.2E+03
1.0 - 1.5	9.5E+03
1.5 - 2.0	3.9E+01
2.0 - 3.0	6.5E-02
3.0 - 4.0	0.0E+00
4.0 - 5.0	0.0E+00
5.0 - 6.0	0.0E+00
6.0 - 7.0	0.0E+00
7.0 - 8.0	0.0E+00
8.0 - 10.0	0.0E+00
10.0 - 14.0	0.0E+00
Total	1.5E+04

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12.3 Radiation Protection Design Features

12.3.1 Facility Design Features

Specific design features for maintaining personnel exposure as low as reasonably achievable (ALARA) are presented in this subsection. The design feature recommendations given in Regulatory Guide 8.8 are utilized to minimize exposures to personnel.

12.3.1.1 Plant Design Features for ALARA

The equipment and plant design features employed to maintain radiation exposures ALARA are based upon the design considerations of Subsection 12.1.2 and are outlined in this subsection.

12.3.1.1.1 Common Equipment and Component Designs for ALARA

This subsection describes the design features utilized for several general classes of equipment or components. These classes of equipment are common to many of the plant systems; thus, the features employed for each system to maintain minimum exposures are similar and are presented by equipment class in the following paragraphs.

Reactor Vessel

The reactor vessel design includes an integrated head package which combines the head lifting rig, control and gray rod drive mechanism (CRDM/GRDM), lift columns, control rod drive mechanism cooling system and power and instrumentation cabling into an effective, one-package reactor vessel head design. Mounted directly on the reactor vessel head assembly, the system helps to minimize the time, manpower, and radiation exposure associated with head removal and replacement during refueling. Integral in the design is permanent shielding for reducing work area dose rates from the control rod drive mechanism drive shafts.

The combination thermocouple/incore detector system is not kept with head assembly during refueling, but instead remains with the upper internals. This allows the thermocouple/incore detector system to be shielded underwater in the refueling cavity during a majority of refueling operations, reducing dose rates around the head assembly.

The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors. The nozzle area is tapered along the reinforced areas to provide a smooth transition, and pipe branch locations are selected to avoid interference from one branch to the next. Weld-to-pipe interfaces require a smooth, high quality finish.

Reactor Vessel Upper and Lower Blocks

The reactor vessel is equipped with upper and lower shielding blocks (shown in Figure 5.3-7) to minimize neutron and gamma radiation streaming within the containment in order to reduce worker radiation doses during operations and outage maintenance.

Reactor Coolant Pumps

The sealless high-inertia reactor coolant pumps are designed to require infrequent maintenance and inspection. When maintenance or replacement is required, the pump can be removed and moved to a low radiation background work area using a specially provided pump removal cart.

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Reactor Vessel Insulation

Insulation in the area of the reactor vessel nozzle welds is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate removal of the insulation. Permanent identification markings of the sections of insulation are provided to accommodate rapid reinstallation.

Steam Generators

The steam generator incorporates many design features to facilitate maintenance and inspection in reduced radiation fields. The tube ends are designed to be flush with the tube sheet in the steam generator channel head to eliminate a potential crud trap. The steam generator manways (entrance to channel head) are sized for easy entrance and exit of workers with protective clothing, and to facilitate the installation and removal of tooling.

The specification of low cobalt tubing material for the AP1000 steam generator design is an important feature of the design; not only in terms of reduced exposure relative to the steam generator, but to the total plant radiation source term. The cobalt content has been substantially reduced to 0.015 weight percent for the AP1000 steam generator tubing.

The steam generator design includes a sludge control system/mud drum which is designed to reduce the need for sludge lancing, and reduces tube and tube support degradation. Steam generator tube support plates design and full depth tubesheet expansion of tubes reduce corrosion and occupational exposure.

Reactor Coolant Pipe Connections

To minimize crud buildup in branch lines, piping connections to the reactor coolant loops are located on or above the horizontal centerline of the pipe wherever practicable.

Filters

Cartridges and filter bags that accumulate radioactivity are removed with semi-remote tools. Adequate space is provided to allow removing, and transporting the cartridge to storage and packaging areas as described in Section 11.4.

Liquid systems containing radioactive filters are provided with remote or semi-remote filter handling systems for the removal of spent radioactive filter elements from their housings and for their transfer to temporary storage or for packaging and shipment from the site for burial. The process is accomplished in such a manner that exposure to personnel and the possibility of inadvertent radioactive release to the environment is minimized. The filter handling is designed to be simple, with a minimum of components susceptible to malfunction.

Demineralizers

Demineralizers for radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin tanks prior to processing and so that fresh resin can be loaded into the demineralizer remotely. The demineralizers and piping include provisions for being flushed with demineralized water. The system design prevents inadvertent flushing of the resin into the purification loop through the demineralizer inlet.

Pumps

Air operated diaphragm, sealless pumps or pumps with mechanical seals are used in radioactive systems to reduce leakage and seal servicing time. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal if necessary. Large pumps are selected with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping.

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Pumps in radioactive waste systems are provided with flanged or threaded connections for ease of removal.

Tanks and Sumps

Tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system to control contamination within plant structures. Tanks containing radioactivity are fabricated from stainless steel, and sumps which can contain radioactive liquid are lined with stainless steel to facilitate decontamination.

Heat Exchangers

Vertical heat exchangers are designed so that the shell-to-tube sheet joint need not be broken for inspection. The shell and tube assembly can be lifted intact above the channel head to expose the tube ends for inspection and testing for leaks.

Heat exchangers are provided with corrosion-resistant tubes of stainless steel to reduce leakage. Impingement plates are provided and as necessary and tube side and shell side velocities are limited to minimize erosive effects. Wherever practicable, the radioactive fluid passes through the tube side of the heat exchanger.

Instruments

Instrument devices are located in low radiation zones away from radiation sources whenever practicable. Primary instrument devices, which for functional reasons are located in high radiation zones, are designed for easy removal to a lower radiation zone for calibration. Transmitters and readout devices are located in low radiation zones, such as corridors for servicing. Non-contact type instruments or self cleaning instruments are used whenever possible.

Some instruments in high radiation zones, such as thermocouples, are provided in duplicate to reduce access and service time required. In-containment instruments are located outside the secondary shield (area of lower radiation at power and shutdown) whenever practicable.

Integral radiation check sources for response verification for airborne radiation monitors and area radiation monitors are provided.

Chemical seals are provided on the instrument sensing lines on process piping, which may contain highly radioactive solids, to reduce the servicing time required to keep the lines free of solids. Instrument and sensing line connections are located slightly above the pipe midplane wherever practicable to minimize radioactive crud buildup.

Valves

To minimize personnel exposures from valve operations, motor-operated, air-operated, or other remotely actuated valves are used where justified by the activity levels and frequency of use. Valves are located in valve galleries so that they are shielded separately from the major components. Long runs of exposed piping are minimized in valve galleries. In areas where manual valves are used on frequently operated process lines, either valve stem extenders or shielding is provided such that personnel need not enter a high radiation area for valve operation.

Wherever testing is required, valves of the bolted body-to-bonnet forging type are used to permit the use of ultrasonic testing in place of radiography. This facilitates inspection and maintenance time. Valves under 2 inches in diameter located in the piping carrying radioactive fluids in containment or carrying highly radioactive fluids outside containment are hermetically sealed valves to preclude radioactive releases to the environment, except for some passive core cooling system test header valves, where the valve stems are not normally pressurized, and hermetically sealed valves are

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impractical due to design constraints and availability limitations. The design of large-bore valves may include live-loaded packing configuration and other improved design features such as graphite packing to reduce the potential for stem leakage.

When equipment in high radiation areas is operated infrequently, those valves associated with normal processing are provided with remote-manual operators or reach rods. Other valve operations are performed with equipment in the shutdown mode.

For valves located in radiation areas, provisions are made to drain adjacent radioactive components when maintenance is required. To the extent practicable, valves are not located at piping low points.

Manually operated valves in the filter and demineralizer modules required for normal operation and shutdown are equipped with reach rods extending through the shield plates. Personnel do not enter the module during spent resin or cartridge transfer operations. The modules are designed to reduce personnel exposure during maintenance of components within or adjacent to the modules and to protect personnel who operate the valves.

Piping

The piping in pipe chases is designed for 60 year design objective with consideration for corrosion and operating environment. Pipe bends are used instead of elbows where practicable to reduce potential crud traps. Welds are made smooth to prevent crud traps from forming. Butt welds are used to the extent practicable. When radioactive piping is routed through areas where routine maintenance is required, pipe chases or distance separation are provided to reduce the radiation contribution from these pipes to levels appropriate for the inspection or maintenance requirements. Piping containing radioactive material is routed to minimize radiation exposure to plant personnel.

Floor and Equipment Drains

Floor drains and sloped floors are provided, where practicable, for rooms or cubicles containing serviceable components which contain radioactive liquids on plant elevations above the basemat. Floor drains are also provided on the Nuclear Island basemat in potentially contaminated areas, but sloping of floors around the drains is not required. If floor sloping is not incorporated on top of the Nuclear Island basemat, housekeeping activities shall be used to convey radioactive leakage to the available floor drains. When practicable, shielded pipe chases are used for radioactive pipes. Floor coatings are specified which simplify cleanup of spills. If a radioactive drain line must pass through a plant area requiring personnel access, shielding or distance separation is provided as necessary to maintain radiation levels consistent with the required access.

Lighting

Wherever practicable, multiple electric lights are provided for rooms containing highly radioactive components so that the burnout of a single lamp does not require entry and immediate replacement of the defective lamp since sufficient illumination is still available. Incandescent lights are provided inside containment. They require less time for servicing and, hence, the personnel exposure is reduced. The fluorescent lights which are used outside containment do not require frequent service due to the increased life of the tubes. High Pressure Sodium (HPS) underwater lighting, with appropriate barriers and protection to prevent contact with the water, or long life lighting not containing restricted materials, are used for underwater lighting in the fuel handling areas outside containment as they provide longer lamp life, longer service intervals and less personnel exposure.

Heating, Ventilation, and Air-Conditioning

The heating, ventilation, and air-conditioning (HVAC) system design facilitates replacement of the filter elements. Ventilation airflow is routed from areas of lower potential airborne contamination to areas of potentially higher contamination. In the radiologically-controlled area ventilation system

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(VAS) high airborne activity causes the exhaust air to be rerouted through HEPA and charcoal filters in the containment air filtration system (VFS).

Sample Stations

Proper shielding and ventilation are provided at the primary sample room to minimize personnel exposure during sampling. The counting room and laboratory facilities are described in Section 12.5. The fly ash used in the concrete for these areas is screened before use to confirm the radioactivity is less than the design limit.

Clean Services

Whenever practicable, clean services and equipment such as compressed air piping, clean water piping, ventilation ducts, and cable trays are not routed through radioactive pipeways.

Materials

Equipment specifications for components exposed to high temperature reactor coolant as part of normal plant operations contain limitations on the cobalt content of the base metal as given in Table 12.3-1. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations. Nickel-based alloys in the reactor coolant system (Co-58 is produced from activation of Ni-58) are similarly used only where component reliability may be compromised by the use of other materials. The major use of nickel-based alloys in the reactor coolant system is the inconel steam generator tubes.

General prohibitions on antimony and other low melting point metals are contained in Subsection 6.1.1. In addition, the reactor coolant pump mechanical design criteria prohibits antimony completely from the reactor coolant pump and its bearings.

Improved Head Closure System

The head closure system is designed to minimize the reactor head stud tensioning time.

12.3.1.1.2 Common Facility and Layout Designs for ALARA

This subsection describes the design features utilized for standard plant process and layout situations. These features are employed in conjunction with the general equipment described in Subsection 12.3.1.1.1 and include the features described in the following paragraphs.

Valve Modules

Selected valve modules are provided with shielded entrances for personnel protection. Floor drains are provided to control radioactive leakage. To facilitate decontamination, concrete surfaces are covered with a smooth surface coating which allows decontamination.

Piping

Pipes carrying radioactive materials are routed through controlled access areas properly zoned for that level of activity. Radioactive piping runs are analyzed to determine the potential radioactivity level and the dose rate 1 foot from the surface, in accordance with 10 CFR 20. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, shielded pipeways or distance separation are provided. Whenever practicable, valves and instruments are not placed in radioactive pipeways. Equipment compartments are used as pipeways for those pipes associated with equipment in the compartment.

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When practicable, radioactive and nonradioactive piping are separated to minimize personnel exposure. Should maintenance be required, provision is made to isolate and drain radioactive piping and associated equipment.

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. In radioactive systems, the use of nonremovable backing rings in the piping joints is prohibited. Whenever practicable, branch lines having little or no flow during normal operation are connected above the horizontal midplane of the main pipe.

Piping which carries resin slurries is run vertically and horizontal runs carrying spent resin are sloped toward the spent resin tanks, as much as practicable. Large radius bends are utilized instead of elbows. Where sloped lines or large radius bends are impractical, adequate flush and drain capability is provided to prevent flow blockage and minimize crud traps.

The use of embedded pipes is minimized to the extent possible, consistent with maintaining radiation doses ALARA. To the extent possible, pipes are routed in accessible areas, such as dedicated pipe routing tunnels or pipe trenches, which provide good conditions for decommissioning.

Wall Penetrations

To minimize radiation streaming through wall penetrations, as many wall penetrations as practicable are located with offsets between the radioactive source and the normally accessible areas. If offsets are not practicable, penetrations are located as far as practicable above the floor elevation to reduce radiation exposure to personnel. If these two methods are not used, alternate means are employed, such as baffle shield walls or grouting the penetration annulus.

Contamination Control

Access control and traffic patterns are considered in the plant layout to reduce the spread of contamination. Routinely used equipment vents and drains from highly radioactive systems are piped directly to the collection system to minimize airborne and floor contamination. Vents and drains from highly radioactive systems that are used less than once per refueling cycle are either piped directly to the collection system or are isolated with a valve and pipe end cap if direct piping is impractical. Discharge from infrequently used vents and drains that use a valve and pipe end cap is routed to the appropriate collection system with temporary flexible hose during the infrequent use. Welded piping systems are employed on radioactive systems to the maximum extent practicable to reduce system leakage and crud buildup at joints.

The number of passageways (doors) between the radiologically controlled area and the environment has been minimized. When such doors are incorporated, systems of drains and floor and exterior concrete sloping are used to prevent (potentially radioactive) fluid from the interior of the buildings from exiting the buildings, and also to prevent surface water from entering the buildings.

Decontamination of potentially contaminated areas and equipment within the plant is facilitated by the application of epoxy paints and suitable smooth-surface coatings to the concrete floors and walls. Sloping floors with floor drains are provided, where practicable, in potentially contaminated areas of the plant on plant elevations above the basemat. Floor drains are also provided on the Nuclear Island basemat in potentially contaminated areas, but sloping of floors around the drains is not required. If floor sloping is not incorporated on top of the Nuclear Island basemat, housekeeping activities shall be used to convey radioactive leakage to the available floor drains. In addition, radioactive and potentially radioactive drains are separated from nonradioactive drains.

In radiologically controlled areas where contamination is expected, radiation monitoring equipment is provided (Section 11.5). Those systems that become highly radioactive, such as the spent resin lines in the radwaste system, are provided with flush and drain connections.

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Because of the potential for adsorption of contaminated fluids, the use of concrete block walls in the radiologically controlled areas of the plant is minimized. Where such walls are used, they are fully sealed at the ceiling or top of the block in order to prevent liquid incursion.

The role of the ventilation systems in minimizing the spread of airborne contamination is described in Subsection 12.3.3.

Equipment Layout

In those systems where process equipment is a major radiation source; pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of these items in reduced radiation zones. Control panels are located in low radiation zones.

Major components such as tanks, demineralizers, and filters in radioactive systems are located in shielded compartments insofar as practical. Labyrinth shields or shielding doors are provided for compartments where radiation could stream or scatter to access areas and exceed the radiation zone dose limits for those areas. For potentially high radiation components (such as ion exchangers, filters and spent resin tanks), shielded compartments with hatch openings or removable shield walls are used. Equipment in nonradioactive systems that requires lubrication is located in low radiation zones. Wherever practicable, lubrication of equipment in high radiation areas is achieved with the use of tube-type extensions to reduce exposure during maintenance.

Exposure from routine in-plant inspection is controlled by locating, whenever practicable, inspection points in low-background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, space for portable shielding is provided.

Field Run Piping

Field run radioactive piping is minimized in the plant design. Radioactive process piping is routed dimensionally on orthographic drawings. Fabrication isometrics of radioactive process piping are reviewed to provide adequate shielding.

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yard area is regulated and controlled by posting of radiation signs, control of personnel, and use of alarms and locks (Section 12.5). During plant operation, access to the radiologically controlled area (RCA) is through the access control area in the annex building.

Plant areas are categorized into radiation zones according to design basis radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Rooms, corridors, and pipeways are evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Each radiation zone defines the radiation level range expected in the zone. The radiation zone categories employed and zoning for each plant area under normal conditions is shown in Figure 12.3-1. The zoning for each plant area under accident conditions is shown in Figure 12.3-2. Radiation zones shown in the figures are based upon conservative design data. Actual in-plant zones and control of personnel access are based upon surveys conducted by the Combined License holder. Access control provisions for each plant area under normal expected conditions are shown in Figure 12.3-3. These provisions implement the

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requirements of 10 CFR 20 and utilize the alternative access control methods outlined in Regulatory Guide 8.38.

Figure 12.3-201, Figure 12.3-202, and Figure 12.3-203 replace Figure 12.3-1 (sheet 11), Figure 12.3-2 (sheet 11), and Figure 12.3-3 (sheet 11), respectively, to reflect the relocation of the Operations Support Center.

Based on actual operating plant data, ingress or egress of plant operating personnel to radiologically restricted areas is controlled and monitored as discussed in Subsection 12.3.5 such that radiation levels and exposures are within the limits prescribed in 10 CFR 20.

Posting of radiation signs, control of personnel access, and use of alarms and locks are discussed in Subsection 12.3.5.

12.3.2 Shielding

The bases for the nuclear radiation shielding and the shielding configurations are discussed in this subsection.

12.3.2.1 Design Objectives

The objective of the plant radiation shielding is to minimize personnel and population exposures, while maintaining a program of controlled personnel access to and occupancy of radiation areas. Radiation levels are within the requirements of 10 CFR 50 during design basis accidents and ALARA within the requirements of 10 CFR 20 during normal operation. Shielding and equipment layout and design are considered in providing confidence that exposures are kept ALARA during anticipated personnel activities in areas of the plant containing radioactive materials. Design recommendations given in Regulatory Guide 8.8 are utilized where practicable.

The nuclear radiation shielding is designed to provide personnel protection and is based on the following operating states:

- Normal, full-power operation
- Shutdown operation
- Spent resin transfer
- Emergency operations (for required access to safety-related equipment)

The shielding design objectives for the plant during these operating states are:

- Radiation exposure to plant operating personnel, contractors, administrators, visitors, and site boundary occupants is ALARA and within the limits of 10 CFR 20.

- Sufficient personnel access and occupancy time is provided to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.

- Reduce potential equipment neutron activation and mitigate the effects of radiation on materials.

- Provide sufficient shielding for the control room so that for design basis accidents (DBAs) the direct dose plus the inhalation dose (calculated in Chapter 15) does not exceed the limits of 10 CFR 50, Appendix A, General Design Criterion 19.

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12.3.2.2 General Shielding Design

Systems containing radioactivity and other sources of radiation are identified for four plant conditions defined in Subsection 12.3.2.1. Shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area shown in Figure 12.3-1. Design criteria for shield penetrations are consistent with the recommendations of Regulatory Guide 8.8 and are described in Subsection 12.3.1.1.2.

Materials used in shielding typically include lead, steel, water, and concrete. The material used for most of the plant shielding is ordinary concrete with a bulk density of approximately 140 lb/ft^3. Whenever poured-in-place concrete has been replaced by concrete blocks, an equivalent shielding basis as determined by the density of the concrete block is selected. Steel is used as shielding in the chemical and volume control system and other modules, as well as around the reactor vessel flange at the floor of the refueling cavity. Water is used as the primary shield material for areas above the spent fuel storage area and refueling cavity during refueling operations.

12.3.2.2.1 Containment Shielding Design

During reactor operation, the shield building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete shield building wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the shield building to less than 0.25 mrem/hr from sources inside containment. The shield building completely surrounds the reactor coolant system components.

For design basis accidents, the shield building and the main control room shielding reduce the plant radiation intensities from fission products inside the containment to acceptable levels, as defined by 10 CFR 50, Appendix A, General Design Criterion 19, for the main control room. (See Subsection 12.3.2.2.7.)

Where personnel locks and equipment hatches or penetrations pass through the shield building wall, additional shielding is provided to attenuate radiation to the level defined by the outside radiation zone during normal operation and shutdown, and to acceptable levels during design basis accidents as defined by General Design Criterion 19.

12.3.2.2.2 Containment Interior Shielding Design

During reactor operation, many areas inside the containment are Zone V or greater and are normally inaccessible. Shielding is provided to reduce dose rates to approximately 100 mrem/hr or less in areas of the containment that potentially require access at power. These are the Zone IV or lower areas shown in Figure 12.3-1.

The main sources of radiation are the reactor vessel and the primary loop components, consisting of the steam generators, pressurizer, reactor coolant pumps, and associated piping. The reactor vessel is shielded by the concrete primary shield and by the concrete secondary shield which also surrounds other primary loop components. Air cooling is provided to prevent overheating, dehydration, and degradation of the shielding and structural properties of the primary shield.

The primary shield is a large mass of reinforced concrete surrounding the reactor vessel. The primary shield meets the following objectives:

- In conjunction with the secondary shield, reduce the radiation level from sources within the reactor vessel and reactor coolant system to allow limited access to the containment during normal, full-power operation.

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- After shutdown, limit the radiation level from sources within the reactor vessel, permit limited access to the reactor vessel and the reactor coolant system equipment.
- Limit neutron activation of component and structural materials.

The secondary shield is a structural module filled with concrete surrounding the reactor coolant system equipment, including piping, pumps, and steam generators. This shield protects personnel from the direct gamma radiation resulting from reactor coolant activation products and fission products carried away from the core by the reactor coolant. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full-power operation.

The reactor cavity has been designed so that the dose rates on the operating deck due to neutron streaming are less than 100 mrem/hr.

Components of the purification portion of the chemical and volume control system (CVS) in the containment are located in a shielded compartment. Shielding is provided for equipment in the purification system consistent with its postulated maximum activity (Subsection 12.2.1) and with the access and zoning requirements of adjacent areas. This equipment includes the regenerative heat exchanger, the letdown heat exchanger, chemical and volume control system filters and demineralizers, and the letdown lines.

After shutdown, the containment is accessible for limited periods of time and access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range from 0.5 to 1000 mrem/hr, depending on the location inside the containment (excluding reactor cavity). These dose rates result from residual fission products and neutron activation products (components and corrosion products) in the reactor coolant system.

Spent fuel is the primary source of radiation during refueling. Because of the high activity of the fission products contained in the spent fuel elements, extensive shielding is provided for areas surrounding the refueling cavity and the fuel transfer canal to limit the radiation levels to below zone levels specified for adjacent areas. Water provides the shielding over the spent fuel assemblies during fuel handling.

12.3.2.2.3 Auxiliary Building Shielding

During normal operations, the major components in the auxiliary building with potentially high radioactivity are those in liquid radwaste, gaseous radwaste, and spent resin handling systems. Shielding is provided consistent with the postulated maximum activity (See Sections 11.1, 11.2, 11.3, and 12.2) and with the access and zoning requirements of adjacent areas. (See Figure 12.3-1.)

Depending on the equipment in the compartments, the radiation zones vary. Corridors are generally shielded to allow Zone II access, and operator areas for valve modules are generally Zone II or III for access.

Concrete plugs are utilized to provide necessary access for equipment maintenance and spent filter cartridge replacement. Where necessary, labyrinth entrances with provisions for adequate ingress and egress for equipment maintenance and inspection are provided and are designed to be consistent with the access and zoning requirements of adjacent areas.

Following reactor shutdown, the normal residual heat removal (RNS) system pumps and heat exchangers are in operation to remove heat from the reactor coolant system. The radiation levels in the vicinity of this equipment temporarily reach Zone V or higher levels due to corrosion and fission

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products in the reactor coolant water. Shielding is provided to attenuate radiation from normal residual heat removal equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas.

12.3.2.2.4 Fuel Handling Area Shielding Design

The concrete shield walls surrounding the spent fuel cask loading and decontamination areas, and the shield walls surrounding the fuel transfer and storage areas are sufficiently thick to limit radiation levels outside the shield walls in accessible areas to Zone II. The building external walls are sufficient to shield external plant areas which are not controlled to Zone I.

Spent fuel removal and transfer operations are performed under borated water to provide radiation protection and maintain subcriticality. Minimum allowable water depths above active fuel in a fuel assembly during fuel handling are 8.75 feet in the reactor cavity and 8.75 feet in the fuel transfer canal and spent fuel pool. This limits the dose to personnel on the spent fuel pool handling machine to less than 2.5 mrem/hr for an assembly in a vertical position. Minimum water depth above the stored assemblies is about 26 feet, and for this depth the dose rate at the pool surface is insignificant. The concrete walls of the fuel transfer canal and spent fuel pool walls supplement the water shielding and limit the maximum radiation dose levels in working areas to less than 2.5 mrem/hr.

The spent fuel pool cooling system (SFS) shielding (Section 9.1) is based on the activity discussed in Subsection 12.2.1 and the access and zoning requirements of adjacent areas. Equipment in the spent fuel pool cooling system to be shielded includes the spent fuel cooling system heat exchangers, pumps, piping, filters and demineralizers which may be contaminated with radioactive crud.

12.3.2.2.5 Radwaste Building Shielding Design

Shielding is provided as necessary for the waste storage areas in the radwaste building to meet the radiation zone and access requirements. Depending on the equipment in the compartments, the radiation zoning varies from Zone I through IV as shown on the radiation zone drawing of Figure 12.3-1. Temporary partitions and shield walls will be provided, as required, to supplement the permanent shield walls surrounding the waste accumulation room inside the radwaste building. The three bunkers described in Subsection 11.4.2.5.2 include removable steel shield bunker roof plates that provide vertical shielding during storage of moderate and high activity waste to maintain radiation levels as Zone II to Zone III on the normally unoccupied and access controlled radwaste building roof, and removable steel shield bunker door plates, that may be installed for ALARA considerations. The removable steel shield bunker door plates are not required to be installed to maintain radiation levels in the worker occupied areas in the radwaste building and in the adjacent plant yard areas as Zone I to Zone IV as defined in Figure 12.3-1 (Sheet 14).

12.3.2.2.6 Turbine Building Shielding Design

The steam generator blowdown demineralizers and the resin columns associated with the secondary sampling system (SSS) steam generator blowdown sample panel are shielded when required to meet the radiation zone and access requirements. Radiation shielding is not required for other process equipment located in the turbine building. Space has been provided so that shielding may be added around the steam generator blowdown system (BDS) filter, the electrodeionization (EDI) unit, and the condensate polishing demineralizers if they become radioactive.

12.3.2.2.7 Control Room Shielding Design

The design basis loss-of-coolant accident dictates the shielding requirements for the control room. The rod ejection accident dictates the shielding requirements for the main control room emergency

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habitability (VES) filter in the operator break room. Consideration is given to shielding provided by the shield building structure. Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated loss-of-coolant accident, so that radiation doses are limited to five rem whole body from contributing modes of exposure for the duration of the accident, in accordance with General Design Criterion 19.

Shielding of the VES filtration unit is accomplished by safety-related metal shielding. This shielding is composed of either tungsten that is 0.25 inches thick or stainless steel shown to provide an equivalent amount of shielding. The length and width of the shielding are designed to match the length and width of the filtration unit being shielded.

12.3.2.2.8 Miscellaneous Plant Areas and Plant Yard Areas

With limited exceptions, sufficient shielding is provided for plant buildings containing radiation sources so that radiation levels at the outside surfaces of the buildings are maintained below Zone I levels. The yard area west of the turbine building first bay could experience Zone II levels under design basis conditions if the steam generator blowdown system demineralizers and resin columns are not shielded as described in Subsection 12.3.2.2.6. The yard area outside the auxiliary building fuel handling area could experience Zone III levels at elevations more than 20 feet above the grade elevation during resin transfer operations under design basis conditions. Plant yard areas that are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. Tanks containing radioactive materials are not located in the yard.

12.3.2.2.9 Spent Fuel Transfer Canal and Tube Shielding

The spent fuel transfer tube is shielded to within adjacent area radiation zone limits. This is primarily achieved through the use of concrete and water. The only removable shielding consists of concrete or steel hatches which reduce radiation in accessible areas to within those levels prescribed in the normal operation radiation zone maps (Figure 12.3-1).

The spent fuel transfer tube is completely enclosed in concrete and there is no unshielded portion of the spent fuel transfer tube during the refueling operation. The only potential radiation streaming path associated with the tube shielding configuration is the 2 inch (5.08 cm) seismic gap between the fuel transfer tube shielding and the steel containment wall. Shielding of this gap is provided by a water-filled bladder. This "expansion gap" radiation shield provides effective reduction of the radiation fields during fuel transfer and accommodates relative movement between the containment and the concrete transfer tube shielding with no loss in shield integrity. A removable hatch in the shield configuration provides access for inspection of the fuel transfer tube welds. The opening of this hatch is administratively controlled and is treated as an entrance to a very high radiation area under 10 CFR 20. This hatch is in place during the spent fuel transfer operation.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided for compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions described in Chapter 11 and Section 12.2. The thickness of each shield wall surrounding radioactive equipment is determined by approximating as closely as practicable the actual geometry and physical condition of the source or sources. The isotopic concentrations are converted to energy group sources using data from standard references (References 1 through 6).

The geometric model assumed for shielding evaluation of most tanks, heat exchangers, filters, ion exchangers, and the containment is a finite cylindrical volume source. For shielding evaluation of piping, the geometric model is a finite shielded cylinder. In cases where radioactive materials are deposited on surfaces such as pipe, the latter is treated as an annular cylindrical surface source.

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Computer codes based on point kernel and Monte Carlo methods are used to calculate gamma dose rates. Most dose rates for non-complex geometries are calculated with a point kernel code MicroShield 6.20 (Reference 22), which is a PC shielding code with a menu-guided user-interface. For complex geometries, Monte Carlo or discrete ordinate methods were used for radiation analysis. Some simplifications are made in the modeling, concerning non-active components connected to the sources, and shielding. As a rule, these simplifications result in conservative dose rate estimates, but do not significantly affect the overall evaluation of the radiological conditions in the containment. Non-homogenous sources, such as fuel assemblies, ion exchange resin beds are homogenized, where this does not underestimate the dose rates.

Complex geometries are modeled in MCNP code (Reference 21). Due to the need of larger computer and work resources MCNP is used only in those cases that cannot be calculated by methods based on line-of-sight attenuation such as point kernel method. Such cases may involve labyrinth structures, penetrations, dominance of scattered radiation etc.

For very simple geometries also analytical formulas using gamma energy yields of radioactive isotopes are used.

The source activity (Ci) and gamma ray source strengths (MeV/sec) are calculated using one of the following computer codes: ORIGEN (Reference 17), SOURCE2/ACCUM (Reference 12), or RADGAS3 (Reference 13). ACCUM (Reference 12) is an option within SOURCE2 that computes isotope accumulation for several time periods from a given flow of isotopes in curies per second. This accumulated activity may then be decayed for any number of decay times at which gamma energy spectra and isotope Curie activity are computed. The generation of daughter products is included during the accumulation and decay periods. FIPCO, CORA, and RADGAS3 compute isotopic activity in radioactive liquid and gaseous systems. The total activity in system lines or equipment is computed from the initial isotope flow, equipment accumulating (operating) time, and parameters which describe the physical accounts for instantaneous mixing or uniform flow and plateout of particulate daughter products. Isotope data is based on the Table of Isotopes (Reference 5) and ORIGEN library data (Reference 6).

The shielding thicknesses of walls and slabs are selected to reduce the aggregate computed radiation level from the contributing sources below the upper limit of the radiation zone specified for each plant area. The labyrinths are constructed so that the scattered dose rate, plus the transmitted dose rate through the shield wall from all contributing sources, is below the upper limit of the radiation zone specified for each plant area. Shielding requirements in each plant area are evaluated at the point of maximum radiation dose through any wall. In addition, for shielding design purposes the concrete density of 140 lb/ft^3 was assumed. Therefore, the actual anticipated radiation level in each plant area is less than this maximum dose and consequently less than the radiation zone upper limit.

Neutron radiation is calculated either with MCNP code or hand calculation methods combined to literature data on neutron attenuation (References 7 and 8).

12.3.3 Ventilation

The plant heating, ventilating, and air-conditioning systems are designed to provide a suitable environment for personnel and equipment during normal operation.

12.3.3.1 Design Objectives

The plant heating, ventilating, and air-conditioning systems for normal operation are designed to meet the requirements of 10 CFR 20 and 10 CFR 50.

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12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following:

- During normal operation the average and maximum airborne radioactivity levels to which plant personnel are exposed in restricted areas of the plant are ALARA and within the limits specified in 10 CFR 20. The average and maximum airborne radioactivity levels in unrestricted areas of the plant during normal operation, are ALARA and within the limits of 10 CFR 20.

- During normal operations the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary is ALARA and within the limits specified in 10 CFR 20 and 10 CFR 50, Appendix I.

12.3.3.3 Design Features

To accomplish the design objectives and to conform to the design criteria, the following design features are incorporated wherever practicable.

12.3.3.3.1 Design Features to Minimize Airborne Radioactivity

- Access control and traffic patterns are considered in the plant layout to minimize the spread of contamination.

- Routinely used equipment vents and drains from highly radioactive systems are piped directly to a collection device connected to the collection system. This is to minimize airborne contamination and to prevent contaminated fluid from flowing across the floor to a floor drain. Vents and drains from highly radioactive systems that are used less than once per refueling cycle are either piped directly to the collection system or are isolated with a valve and end cap if direct piping is impractical. Discharge from infrequently used vents and drains that use a valve and pipe end cap is routed to the appropriate collection system with temporary flexible hose during the infrequent use.

- Welded piping systems are employed on systems containing radioactive fluids to the maximum extent practicable. If welded piping systems are not employed, drip trays are provided at the points of potential leakage. Drains from drip trays are piped directly to the collection system.

- Suitable coatings are applied to the concrete floors and walls of potentially contaminated areas to facilitate decontamination.

- Design of equipment incorporates features that minimize the spread of radioactivity during maintenance operations. These features include flush and drain connections on pump casings for draining and flushing the pump prior to maintenance and flush connections on piping systems that could become highly radioactive.

12.3.3.3.2 Design Features to Control Airborne Radioactivity

- The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination.

- In building compartments with a potential for contamination, the exhaust is designed for greater volumetric flow than is supplied to that area. This minimizes the amount of uncontrolled exfiltration from the area.

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- Consideration is given to the potential disruption of normal airflow patterns by maintenance operations, and provisions are made in the design to the extent practicable to prevent adverse airflow direction. Administrative controls, such as those described in Subsection 12AA.5.4.8, are implemented during maintenance operations where necessary to prevent the spread of contamination.

- The ventilation system design for radiologically controlled areas is discussed in Subsections 9.4.3, 9.4.7, 9.4.8, and 9.4.11. The exhaust air from these areas is normally unfiltered except for the containment atmosphere which is filtered by the containment air filtration system exhaust filters. A description of these filter units is given in Subsection 12.3.3.5.

- Air discharged from the containment is passed through high efficiency particulate air filters and charcoal adsorbers to remove particulates and halogens. Air exhausted from the auxiliary building, fuel handling area of the auxiliary building, and the annex building is monitored for high airborne activity. Means are provided to shut off supply air and divert exhaust air through high efficiency particulate air filters and charcoal adsorbers upon detection of high airborne activity. Alarms are provided in the main control room for these discharge flows and for flows from the radwaste building and the health physics/hot machine shop area. These alarms alert the operator of high radioactivity concentrations in the air. This minimizes the discharge of contaminants to the environment and in-plant exposures.

- Atmospheric tanks which contain radioactive materials are vented to the respective building ventilation system for release to the monitored plant vent.

12.3.3.3.3 Design Features to Minimize Personnel Exposure from HVAC Equipment

- The guidelines of Regulatory Guide 8.8 have been utilized, as practicable, in the design of the plant ventilation systems.

- Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure. The HVAC system is designed to allow rapid replacement of components.

- Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts.

- Ventilating air for radiologically controlled areas of the plant is a once-through design.

- Access to ventilation systems in potentially radioactive areas can result in operator exposure during maintenance, inspection, and testing. Equipment locations are selected to minimize personnel exposures. The outside air supply units and building exhaust system components are located in ventilation equipment rooms. These equipment rooms are accessible to the operators. Work space is provided around each unit for anticipated maintenance, testing, and inspection.

12.3.3.4 Design Description

The ventilation systems serving the following structures are considered to be potentially radioactive and are discussed in detail in Section 9.4.

- Containment building (See Subsection 9.4.7)
- Auxiliary building (See Subsection 9.4.3)
- Fuel handling area of the auxiliary building (See Subsection 9.4.3)

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- Annex building (See Subsection 9.4.3)
- Radwaste building (See Subsection 9.4.8)
- Health physics and hot machine shop (See Subsection 9.4.11)

The main control room is considered to be a nonradioactive area. The associated ventilation system design is described in Section 6.4 and Subsection 9.4.1.

Other structures contain insignificant sources of airborne radioactivity and are not addressed in this chapter.

12.3.3.5 Air Filtration Units

The guidance and recommendations of Regulatory Guide 1.140 concerning maintenance and inplace testing provisions for atmospheric cleanup systems, air filtration, and adsorption units are used in the design of the various ventilation systems. The extent to which Regulatory Guide 1.140 has been incorporated is discussed in Subsection 1.9.1.

Provisions specifically included to minimize personnel exposures and to facilitate maintenance or inplace testing operations are as follows.

A. The loading of the filters and adsorbers with radioactive material during normal plant operation is a slow process. Therefore, in addition to monitoring for pressure drop, the filters are checked for radioactivity on a scheduled maintenance basis with portable equipment. The filter elements are replaced before the radioactivity level is of sufficient magnitude to create a personnel hazard. No shielding is provided since it is not required for the level of radioactivity accumulation during normal operation. In case of excessive radioactivity caused by a postulated accident, the filter is replaced before normal personnel access is resumed. It is not necessary for workers to handle filter units immediately after a design basis accident, so exposures can be minimized by allowing the short-lived isotopes to decay before changing the filter.

B. Active components of the atmospheric cleanup systems are designed for ease of removal.

C. Access to active components is direct from working platforms to simplify element handling. Ample space is provided on the platforms for accommodating safe personnel movement during replacement of components, including the use of necessary material handling equipment and inplace testing devices.

D. No filter bank is more than three filter cells high, where each filter cell is 2 feet by 2 feet. The access to the level or platform at which the filter is serviced is by stairs.

E. The clear space for access to filter banks and active components is a minimum of 20 inches by 50 inches.

F. The HEPA filter banks are designed with replaceable cells that are clamped in place against compression seals. The charcoal adsorbers are designed to be replaced with bulk charcoal using a vacuum transfer system. The filter housing is designed and tested to be airtight with bulkhead type doors that are closed against compression seals.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

For a description of the radiation monitoring system (RMS), refer to Section 11.5.

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Procedures detail the criteria and methods for obtaining representative measurement of radiological conditions, including in-plant airborne radioactivity concentrations in accordance with applicable portions of 10 CFR Part 20 and consistent with the guidance in Regulatory Guides 1.21-Appendix A, 8.2, 8.8, and 8.10. Additional discussion of radiological surveillance practices is included in the radiation protection program description provided in Appendix 12AA.

Surveillance requirements are determined by the functional manager in charge of radiation protection based on actual or potential radiological conditions encountered by personnel and the need to identify and control radiation, contamination, and airborne radioactivity. These requirements are consistent with the operational philosophy in Regulatory Guide 8.10. Frequency of scheduled surveillance may be altered by permission of the functional manager in charge of radiation protection or their designee. Radiation Protection periodically provides cognizant personnel with survey data that identifies radiation exposure gradients in area resulting from identified components. This data includes recent reports, with survey data, location and component information.

The following are typical criteria for frequencies and types of surveys:

Job Coverage Surveys

- Radiation, contamination, and/or airborne surveys are performed and documented to support job coverage.
- Radiation surveys are sufficient in detail for Radiation Protection to assess the radiological hazards associated with the work area and the intended/specified work scope.
- Surveys are performed commensurate with radiological hazard, nature and location of work being conducted.
- Job coverage activities may require surveys to be conducted on a daily basis where conditions are likely to change.

Radiation Surveys

- Radiation surveys are performed at least monthly in any radiological controlled area (RCA) where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.
- Radiation surveys are performed prior to or during entry into known or suspected high radiation areas for which up to date survey data does not exist.
- Radiation surveys are performed prior to work involving highly contaminated or activated materials or equipment.
- Radiation surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.
- Radiation surveys are performed to support movement of highly radioactive material.
- Neutron radiation surveys are performed when personnel may be exposed to neutron emitting sources.

Contamination Surveys

- Contamination surveys are performed at least monthly in any RCA where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.

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- Contamination surveys are performed during initial entry into known or suspected contamination area(s) for which up to date survey data does not exist.

- Contamination surveys are performed at least daily at access points, change areas, and high traffic walkways in RCAs that contain contaminated areas. Area access points to a High Radiation Area or Very High Radiation Area are surveyed prior to or upon access by plant personnel or if access has occurred.

- Contamination surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.

- A routine surveillance is conducted in areas designated by the functional manager in charge of radiation protection or their designee likely to indicate alpha radioactivity. If alpha contamination is identified, frequency and scope of the routine surveillance is increased.

Airborne Radioactivity Surveys

- Airborne radioactivity surveys are performed during any work or operation in the RCA known or suspected to cause airborne radioactivity (e.g., grinding, welding, burning, cutting, hydrolazing, vacuuming, sweeping, use of compressed air, using volatiles on contaminated material, waste processing, or insulation).

- Airborne radioactivity surveys are performed during a breach of a radioactive system, which contains or is suspected of containing significant levels of contamination.

- Airborne radioactivity surveys are performed during initial entry (and periodically thereafter) into any known or suspected airborne radioactivity area.

- Airborne radioactivity surveys are performed immediately following the discovery of a significant radioactive spill or spread of radioactive contamination, as determined by the functional manager in charge of radiation protection.

- Airborne radioactivity surveys are performed daily in occupied radiological controlled areas where the potential for airborne radioactivity exists, including containment.

- Airborne radioactivity surveys are performed any time respiratory protection devices, alternative tracking methods such as derived air concentration-hour (DAC-hr), and/or engineering controls are used to control internal exposure.

- Airborne radioactivity surveys are performed using continuous air monitors (CAMs) for situations in which airborne radioactivity levels can fluctuate and early detection of airborne radioactivity could prevent or minimize inhalations of radioactivity by workers. Determination of air flow patterns are considered for locating air samplers.

- Airborne radioactivity surveys are performed prior to use and monthly during use on plant service air systems used to supply air for respiratory protection to verify the air is free of radioactivity.

- Tritium sampling is performed near the spent fuel pool when irradiated fuel is in the pit and other areas of the plant where primary system leaks occur and tritium is suspected.

Appropriate counting equipment is used based on the sample type and the suspected identity of the radionuclides for which the sample is being done. Survey results are documented, retrievable, and processed per site document control and records requirements consistent with Regulatory Guide 8.2.

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Completion of survey documentation includes the update of room/area posting maps and revising area or room postings and barricades as needed.

Air samples indicating activity levels greater than a procedure specified percentage of DAC are forwarded to the radiochemistry laboratory for isotopic analysis. Samples which cannot be analyzed on-site are forwarded to an offsite laboratory or a contractor for analysis; or, the DAC percentage may be hand calculated using appropriate values from 10 CFR Part 20, Appendix B.

The responsible radiation protection personnel review survey documentation to evaluate if surveys are appropriate and obtained when required, records are complete and accurate, and adverse trends are identified and addressed.

An in-plant radiation monitoring program maintains the capability to accurately determine the airborne iodine concentration in areas within the facility where personnel may be present under accident conditions. This program includes the training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment consistent with Regulatory Guides 1.21 (Appendix A) and 8.8. Training and personnel qualifications are discussed in Appendix 12AA.

Emergency operating procedures include provisions for use of a portable monitoring system, consistent with the criteria in NUREG-0737, Item III.D.3.3, to sample and analyze for radioiodine in areas of the plant during and following an accident. Procedures include methods for taking and analyzing samples in the field, as well as for analyzing samples in the count room facility, accounting for techniques to reduce counting system saturation from a high-activity sample. Accident monitoring instrumentation complies with applicable parts of 10 CFR Part 50, Appendix A.

Sampling cartridges can be removed to a low background area for further analysis. These cartridge samples can be purged of any entrapped noble gases, when necessary, prior to being analyzed.

12.3.5 Combined License Information

12.3.5.1 Administrative Controls for Radiological Protection

The administrative controls for use of the design features provided to control access to radiologically restricted areas, including potentially very high radiation areas, are addressed in Subsection 12.5.4 and Appendix 12AA.

12.3.5.2 Criteria and Methods for Radiological Protection

The criteria and methods for obtaining representative measurement of radiological conditions, including airborne radioactivity concentrations in work areas, and the use of portable instruments, and the associated training and procedures, are addressed in Subsection 12.3.4.

12.3.5.3 Groundwater Monitoring Program

In accordance with Reference 23, the groundwater monitoring program is addressed in Appendix 12AA.

12.3.5.4 Record of Operational Events of Interest for Decommissioning

In accordance with Reference 23, the program to provide documentation of operational events deemed to be of interest for decommissioning, including remediation of any leaks that have the potential to contaminate groundwater, is addressed in Appendix 12AA.

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23. USNRC, "Minimization of Contamination," 10 CFR 20.1406.

24. Electric Power Research Institute, “Advanced Light Water Reactor Utility Requirements Document,” Vol. II. ALWR Passive Plant, Chapter 1, Revision 6, December 1993.

25. EPRI NP-2681, Project 1784-3, “Evaluation of Cobalt Sources in Westinghouse-Designed Three- and Four-Loop Plants,” Final Report, October 1982.

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Table 12.3-1
Equipment Specification Limits for
Cobalt Impurity Levels(1)

<table>
<tr><th colspan="3">Region, Component, or Application
(Specific inclusions and examples are noted parenthetically.)</th><th>Maximum Weight Percent of Cobalt</th></tr>
<tr><td rowspan="2">Primary Components</td><td>Steam generator tubing</td><td>(Inconel tubing)</td><td>0.015</td></tr>
<tr><td>Primary components other than steam generators</td><td>(Primary weld cladding surfaces core shroud, lower and upper core plates, lower core barrel and neutron panels or thermal shields)</td><td rowspan="4">0.05</td></tr>
<tr><td colspan="2">Fuel Assembly Components</td><td>Inconel and stainless steel components in the fuel assembly</td></tr>
<tr><td colspan="3">Auxiliary Heat Exchangers</td></tr>
<tr><td colspan="3">Other Components in Regions of High Neutron Flux</td></tr>
<tr><td colspan="3">Steam Generator Surfaces Other than Tubing</td><td>0.10</td></tr>
<tr><td>Small Components</td><td colspan="2">In regions of high neutron flux</td><td>0.20</td></tr>
<tr><td>Bolting Materials or Fasteners</td><td>Outside of regions of high neutron flux</td><td>(Bolting materials in primary and auxiliary components)</td><td rowspan="4">Not limited</td></tr>
<tr><td colspan="3">Weld Material Other than Cladding(2)</td></tr>
<tr><td colspan="3">Bearing and Hardfacing Materials(3)</td></tr>
<tr><td colspan="2">Other Auxiliary Components(2)</td><td>(Valves, piping, instrumentation, and tanks)</td></tr>
</table>

Notes:

1. Parts or components identified to exceed the cobalt limits are evaluated for acceptability considering actual cobalt content, the wetted surface area of the part, the location of the part within the primary system, the estimated plant cobalt input, and the estimates for personnel dose presented in Section 12.4. The evaluation considers industry ALARA guidance in Reference 24, along with the methods of Reference 25 to evaluate cobalt input to the RCS, and in conjunction with ALARA principles from Regulatory Guide 8.8, the estimated dose impact from each exceedance is estimated. The dose impact for the part or component is then used to determine whether re-fabrication, replacement, or use of the part as-is will be warranted to satisfy ALARA design principles. The evaluation methods maintain part and plant design conformance with ALARA design principles as described in this section of the licensing basis and Regulatory Guide 8.8.
2. Stainless steels manufactured without cobalt limitations generally contain 0.15 to 0.20 weight percent of cobalt.
3. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations.

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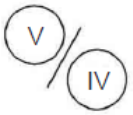
LEGEND:

A. PLANT RADIATION ZONES:

DESIGNATION	MAXIMUM DESIGN DOSE RATE	DESCRIPTION
0	≤ 0.05 mRem/hr	NO RADIATION SOURCES; UNLIMITED GENERAL OCCUPANCY; OUTSIDE "CONTROLLED AREA"
I	≤ 0.25 mRem/hr	VERY LOW OR NO RADIATION SOURCES; INSIDE "CONTROLLED AREA" AND OUTSIDE "RESTRICTED AREA"
"RESTRICTED AREA" ZONES		
II	≤ 2.5 mRem/hr	LOW RADIATION SOURCES; UNLIMITED WORKER OCCUPANCY
III	≤ 15.0 mRem/hr	LOW-TO-MODERATE RADIATION SOURCES; LIMITED WORKER OCCUPANCY
IV	≤ 100 mRem/hr	MODERATE RADIATION SOURCES; LIMITED WORKER OCCUPANCY
V	≤ 1 Rem/hr	HIGH RADIATION SOURCES; LIMITED WORKER OCCUPANCY
VI	≤ 10 Rem/hr	SAME AS ZONE V ABOVE
VII	≤ 100 Rem/hr	SAME AS ZONE V ABOVE
VIII	≤ 500 Rad/hr	SAME AS ZONE V ABOVE
IX	> 500 Rad/hr	VERY HIGH RADIATION SOURCES; VERY LIMITED WORKER ACCESS

B. DRAWING SYMBOLS:



- UPPER RADIATION ZONE NUMERAL FOR FULL POWER OPERATION/ LOWER NUMERAL FOR 24 HOURS AFTER PLANT SHUTDOWN (IF DIFFERENT)

- RADIATION ZONE BOUNDARY

C. GENERAL DRAWING NOTES

1. ACCESS CONTROL REQUIREMENTS AND TRAFFIC PATTERNS ARE SHOWN IN SERIES 201 DRAWINGS.
2. DOSE RATES INSIDE CONTAINMENT DURING POWER OPERATION ARE SUBJECT TO SIGNIFICANT VARIABILITY OWING TO LOCALIZED NEUTRON STREAMING/ SCATTERING EFFECTS. ACTUAL RADIATION FIELDS WILL BE DETERMINED FROM RADIATION SURVEYS AND ACCESS TO THE CONTAINMENT DURING POWER OPERATION WILL BE STRICTLY CONTROLLED.

Figure 12.3-1 (Sheet 1 of 16)
Radiation Zones, Normal Operation/Shutdown Legend

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Figure 12.3-1 (Sheet 2 of 16)
Site Radiation Zones, Normal Operations/Shutdown

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Figure 12.3-1 (Sheet 3 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 66'-6"

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Figure 12.3-1 (Sheet 4 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 82'-6"

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Figure 12.3-1 (Sheet 5 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 96'-6"

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Figure 12.3-1 (Sheet 6 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 100'-0" & 107'-2"

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Figure 12.3-1 (Sheet 7 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 117'-6"

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Figure 12.3-1 (Sheet 8 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 135'-3"

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Figure 12.3-1 (Sheet 9 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 153'-0" & 160'-6"

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Figure 12.3-1 (Sheet 10 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 160'-6", 180'-0", & 329'-0"

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(Note: This figure replaces Figure 12.3-1 Sheet 11 of 16.)

Figure 12.3-201
Radiation Zones, Normal Operations/Shutdown
Annex Building, Elevation 100'-0" & 107'-2"

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Figure 12.3-1 (Sheet 12 of 16)
Radiation Zones, Normal Operations/Shutdown
Annex Building, Elevation 117'-6" & 126'-3"

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Figure 12.3-1 (Sheet 13 of 16)
Radiation Zones, Normal Operations/Shutdown
Annex Building, Elevation 135'-3" through 183'-6"

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Figure 12.3-1 (Sheet 14 of 16)
Radiation Zones, Normal Operations/Shutdown
Radwaste Building, Elevation 100'-0"

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Figure 12.3-1 (Sheet 15 of 16)
Radiation Zones, Normal Operations/Shutdown
Turbine Building, Elevation 100'-0"

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Figure 12.3-1 (Sheet 16 of 16)
Radiation Zones, Normal Operations/Shutdown
Turbine Building, Elevation 120'-6"

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LEGEND:

A. POST-ACCIDENT RADIATION ZONES:

DESIGNATION	MAXIMUM DESIGN DOSE RATE	DESCRIPTION
0	≤ 0.05 mRem/hr	NO RADIATION SOURCES
I	≤ 0.25 mRem/hr	VERY LOW OR NO RADIATION SOURCES
II	≤ 2.5 mRem/hr	LOW RADIATION SOURCES
III	≤ 15.0 mRem/hr	LOW-TO-MODERATE RADIATION SOURCES
IV	≤ 100 mRem/hr	MODERATE RADIATION SOURCES
V	≤ 1 Rem/hr	HIGH RADIATION SOURCES
VI	≤ 10 Rem/hr	SAME AS ZONE V ABOVE
VII	≤ 100 Rem/hr	SAME AS ZONE V ABOVE
VIII	≤ 500 Rad/hr	SAME AS ZONE V ABOVE
IX	> 500 Rad/hr	VERY HIGH RADIATION SOURCES

B. DRAWING SYMBOLS:

VI / ECS – RADIATION ZONE NUMERAL AT POST-ACCIDENT PEAK / DOMINANT POST-ACCIDENT RADIATION SOURCE(S)

– NON-RADIOACTIVE AREA BOUNDARY

– RADIOACTIVE AREA BOUNDARY

– ANNEX AREA BOUNDARY

– RADIATION ZONE BOUNDARY

– POST-ACCIDENT ACCESS ROUTE

C. POST-ACCIDENT SOURCES:

SYMBOL	POST-ACCIDENT RADIATION SOURCE
ECS	EXTERNAL CLOUD SHINE
NRA	NON-RADIOACTIVE AUXILIARY BUILDING AREA CLOUD
RAC	RADIOACTIVE AUXILIARY BUILDING AREA CLOUD
SCC	SHIELDED CONTAINMENT CLOUD
UCC	UNSHIELDED CONTAINMENT CLOUD
CPS	CONTAINMENT AND PENETRATION RADIATION STREAMING
AXC	ANNEX BUILDING AREA CLOUD
PAS	POST-ACCIDENT SAMPLE PIPING

D. GENERAL DRAWING NOTES:

1. ZONING IS BASED ON PEAK POST-ACCIDENT DOSE RATES IN THE DESIGNATED AREA.
2. INCLUDES CONTRIBUTIONS FROM POST-ACCIDENT CONTAINED AND AIRBRONE CLOUD SOURCES.

Figure 12.3-2 (Sheet 1 of 15)
Radiation Zones, Post-Accident Legend

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Figure 12.3-2 (Sheet 2 of 15)
Site Radiation Zones, Post-Accident

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**Figure 12.3-2 (Sheet 3 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 66'-6"**

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Figure 12.3-2 (Sheet 4 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 82'-6"

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**Figure 12.3-2 (Sheet 5 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 96'-6"**

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Figure 12.3-2 (Sheet 6 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 100'-0" & 107'-2"

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**Figure 12.3-2 (Sheet 7 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 117'-6"**

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Figure 12.3-2 (Sheet 8 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 135'-3"

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Figure 12.3-2 (Sheet 9 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 153'-0" & 160'-6"

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Figure 12.3-2 (Sheet 10 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 160'-6", 180'-0", & 329'-0"

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(Note: This figure replaces Figure 12.3-2 Sheet 11 of 15.)

Figure 12.3-202
Radiation Zones, Post-Accident
Annex Building, Elevation 100'-0" & 107'-2"

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Figure 12.3-2 (Sheet 12 of 15)
Radiation Zones, Post-Accident
Annex Building, Elevation 117'-6" & 126'-3"

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Figure 12.3-2 (Sheet 13 of 15)
Radiation Zones, Post-Accident
Annex Building, Elevation 135'-3" through 183'-6"

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Figure 12.3-2 (Sheet 14 of 15)
Radiation Zones, Post-Accident
Radwaste Building, Elevation 100'-0"

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Figure 12.3-2 (Sheet 15 of 15)
Radiation Zones, Post-Accident
Turbine Building, Elevation 100'-0"

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LEGEND:

A. PLANT ACCESS CONTROL PROVISIONS:

AREA TYPE	DOSE RATE	SINGLE AREA	MULTIPLE AREAS
RADIATION AREA	> 5 mRem/hr		
HIGH RADIATION AREA	> 100 mRem/hr	BARRICADED OR ALARMED	BARRICADED OR ALARMED
HIGH RADIATION AREA	> 1 Rem/hr	LOCKED OR (IF OPEN AREA) BARRICADED WITH LOCAL ALARM	LOCKED OR BARRICADED WITH LOCKED COMMON ENTRY AND LOCAL CONTROL POINT
VERY HIGH RADIATION AREA	> 500 Rad/hr	LOCKED	LOCKED OR BARRICADED WITH LOCKED COMMON ENTRY, LOCAL CONTROL POINT AND SURVEILLANCE

B. DRAWING SYMBOLS:

⇨ - PERSONNEL TRAFFIC PATTERN

▲ - ENTRANCE BARRICADE (e.g. ROPE, CHAIN, ETC.)

● - LOCKED ENTRANCE

(C) - LOCAL ACCESS CONTROL POINT

(A) - ALARM LOCATION

(S) - SURVEILLANCE POINT

– – – – – – – – ACCESS CONTROL BARRIER (e.g. CHAIN LINK FENCE, ETC.)

———————— "RESTRICTED" AREA BOUNDARY

– · – · – · – · – · – "CONTROLLED" AREA BOUNDARY

C. GENERAL DRAWING NOTES:

1. ACCESS CONTROL PROVISIONS ARE BASED ON NORMAL EXPECTED RADIATION SOURCES.

Figure 12.3-3 (Sheet 1 of 16)
Radiological Access Controls Legend

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Figure 12.3-3 (Sheet 2 of 16)
Site Radiation Access Controls, Normal Operations/Shutdown

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Figure 12.3-3 (Sheet 3 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 66'-6"

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Figure 12.3-3 (Sheet 4 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 82'-6"

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Figure 12.3-3 (Sheet 5 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 96'-6"

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Figure 12.3-3 (Sheet 6 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 100'-0" & 107'-2"

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Figure 12.3-3 (Sheet 7 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 117'-6"

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Figure 12.3-3 (Sheet 8 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 135'-3"

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Figure 12.3-3 (Sheet 9 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 153'-0" & 160'-6"

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Figure 12.3-3 (Sheet 10 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 160'-6", 180'-0", & 329'-0"

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(Note: This figure replaces Figure 12.3-3 Sheet 11 of 16.)

Figure 12.3-203
Radiological Access Controls, Normal Operations/Shutdown
Annex Building, Elevation 100'-0" & 107'-2"

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Figure 12.3-3 (Sheet 12 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Annex Building, Elevation 117'-6" & 126'-3"

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Figure 12.3-3 (Sheet 13 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Annex Building Elevation 135'-3" through 183'-6"

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Figure 12.3-3 (Sheet 14 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Radwaste Building, Elevation 100'-0"

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Figure 12.3-3 (Sheet 15 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Turbine Building, Elevation 100'-0"

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Figure 12.3-3 (Sheet 16 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Turbine Building, Elevation 120'-6"

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12.4 Dose Assessment

Radiation exposures in the AP1000 are primarily due to direct radiation from components and equipment containing radioactive material. In addition, in some areas of the plant there can be radiation exposure to personnel due to the presence of airborne radionuclides. This section addresses the anticipated occupational radiation exposure (ORE) due to normal operation and anticipated inspection and maintenance.

12.4.1 Occupational Radiation Exposure

Radiation exposures to operating personnel are restricted to be within the limits of 10 CFR 20. The health physics program in Section 12.5 and the radiation protection features described in Section 12.3 together maintain occupational radiation exposures as low as reasonably achievable (ALARA).

In the analysis of occupational radiation exposure data from operating plants (domestic plants having Westinghouse designed nuclear steam supply systems), the best operating plant performance is 0.1 man-rem per MWe-year of electricity produced. Major factors contributing to this level of occupational radiation exposure include low plant radiation fields, good layout and access provisions, and operational practices and procedures that minimize time spent in radiation fields.

As discussed in Section 12.3, the AP1000 design incorporates features to reduce occupational radiation exposure that go beyond the designs provided for plants currently in operation.

The estimated annual occupational radiation exposures are developed within the following categories:

- Reactor operations and maintenance
- Routine maintenance
- Inservice inspection
- Special maintenance
- Waste processing
- Fuel handling operations

Exposure data obtained from operating plants have been reviewed to obtain a breakdown of the doses incurred within each category. For several routinely performed operations, this information has been used to develop detailed dose predictive models. These models identify the various steps that are included in the operation, the radiation zones, the required number of workers, and the time to perform each step. This information has been used to develop dose estimates for each of the preceding categories.

There is no separate determination of doses due to airborne activity. Past experience demonstrates that the dose from airborne activity is not a significant contributor to the total doses.

12.4.1.1 Reactor Operations and Surveillance

To support plant operations, the performance of various systems and components is monitored. Also, operation of some manual valves requires personnel to enter radiation fields. Examples of activities in this category are:

- On-Line Exposure
- Routine patrols and inspections
- Tagging
- Testing

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- Health physics
- Shielding
- Calibration of instrumentation

When the plant is at power, the containment radiation fields are significantly higher than at plant shutdown. The frequency and duration of at-power containment entries is dependent on the plant operator. Based on review of current plant operations and on the AP1000 design changes and reliability improvements, it is assumed that 100 worker-hours per year spent in the containment during power operations.

Table 12.4-1 provides a breakdown of the collective doses for reactor operations and surveillance.

12.4.1.2 Routine Inspection and Maintenance

Routine inspection and maintenance are required for mechanical and electrical components. Table 12.4-2 provides a breakdown of the collective doses for routine inspection and maintenance. These estimates are based on having good access to equipment (a characteristic of the AP1000 layout).

Table 12.4-4 itemizes the doses estimated to be incurred from steam generator sludge lancing operations and Table 12.4-5 lists the doses resulting from the visual examination of the secondary side of the steam generators.

12.4.1.3 Inservice Inspection

ASME Code, Section XI requires periodic inservice inspection (ISI) on plant safety-related components. The Code defines the inservice inspection interval as a 10-year period and sets requirements for each one-third interval (each 40 months). In general, at least 25 percent (with credit for no more than 33-1/3 percent) of the specified inspections must be performed in each 40-month testing interval. The amount of inspection required for an area varies according to the category but is explicitly defined in the Code. Table 12.4-6 provides the doses for inservice inspection activities.

Detailed listings of the doses associated with certain major inservice inspection activities appear in Table 12.4-7 (eddy current inspection of 33 1/3 percent of the steam generator tubes and plugging of three tubes) and Table 12.4-8 (steam generator exterior). The dose estimates in Table 12.4-7 reflect the dose-reducing features of the AP1000 design, such as:

- Permanent work platforms
- Manway cover handling device
- Improved manway insert fasteners (tapered-end type)
- Trailer-mounted data collection station
- Use of robotics to perform eddy current inspection and tube plugging

12.4.1.4 Special Maintenance

Maintenance that goes beyond the routine scheduled maintenance is considered to be special maintenance. This category includes both the modification of equipment to upgrade the plant and repairs to failed components. Dose estimates assume no significant equipment upgrade efforts. The occupational radiation exposure resulting from unscheduled repairs on valves, pumps, and other components will be lower for the AP1000 than for current plant designs because of the reduced radiation fields, increased equipment reliability, and the reduced number of components relative to currently operating plants.

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In the past, special maintenance of the steam generators has resulted in significant personnel doses. The AP1000 benefits both from design changes and from improved primary and secondary water chemistry. The plugging of three tubes per steam generator each time eddy current examination is performed is included in the inservice inspection category.

No special maintenance activities are forecast for the sealless motor reactor coolant pumps.

Table 12.4-9 provides the estimated doses due to special maintenance operations.

12.4.1.5 Waste Processing

The AP1000 radwaste system designs incorporate an uncomplicated approach to waste processing. The AP1000 design does not include waste or boron recycle evaporators and it does not include a catalytic hydrogen recombiner in the gaseous radwaste system. Elimination of high maintenance components contributes significantly to lower anticipated doses due to waste processing activities.

Estimated annual doses from waste processing operations appear in Table 12.4-10.

12.4.1.6 Fuel Handling

Radiation monitoring of the new fuel handling and storage areas is performed in accordance with 10 CFR 50.68(b)(6). Details of the fuel handling area monitoring are provided in Subsections 11.5.6 and 11.5.6.4.

The refueling process is labor intensive. Detailed planning and coordination of effort are essential in order to maintain personnel doses as-low-as-reasonably achievable. Incorporation of advanced technology into the refueling process also reduces doses. Table 12.4-11 lists some of the AP1000 features that reduce doses during refueling operations.

Table 12.4-12 provides dose estimates for the various refueling activities.

12.4.1.7 Overall Plant Doses

The estimated annual personnel doses associated with the six activity categories discussed above are summarized below:

Category	Percent of Total	Estimated Annual (man-rem)
Reactor operations and surveillance	12.0	3.7
Routine inspection and maintenance	37.4	11.7
Inservice inspections	22.6	7.1
Special maintenance	6.1	1.9
Waste processing	10.5	3.3
Refueling	11.5	3.6
Total	100	31.2

These dose estimates are based on operation with an 18-month fuel cycle and are bounding for operation with a 24-month fuel cycle.

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12.4.1.8 Post-Accident Actions

Requirements of 10 CFR 52.79(b) relative to plant area access and post-accident sampling (10 CFR 50.34 (f) (2)(viii) are included in Subsection 1.9.3. If procedures are followed, the design prevents radiation exposures to any individual from exceeding 5 rem to the whole body or 50 rem to the extremities. Figure 12.3-2 in Section 12.3 contains radiation zone maps for plant areas including those areas requiring post-accident access. This figure shows projected radiation zones in areas requiring access and access routes or ingress, egress and performance of actions at these locations. The radiation zone maps reflect maximum radiation fields over the course of an accident. The analyses that confirm that the individual personnel exposure limits following an accident are not exceeded reflect the time-dependency of the area dose rates and the required post-accident access times. The analyses include the assumption that the appropriate respiratory protection equipment is used to maintain radiation exposure within the exposure limits. The areas that require post-accident accessibility are:

- Main control room
- Class 1E regulating transformer areas
- Ventilation control area for MCR and I & C rooms with PAMS equipment
- Valve area to align spent fuel pool makeup
- Ancillary diesel generator room
- Passive containment water inventory makeup area
- Auxiliary building RCA for long-term containment water inventory makeup

12.4.1.9 Radiation Exposure to Construction Workers

12.4.1.9.1 Operating Unit Radiological Surveys

The operating unit conducts radiological surveys in the unrestricted and controlled area and radiological surveys for radioactive materials in effluents discharged to unrestricted and controlled areas in implementing 10 CFR 20.1302. These surveys demonstrate compliance with the dose limits of 10 CFR 20.1301 for construction workers.

12.4.2 Radiation Exposure at the Site Boundary

12.4.2.1 Direct Radiation

The direct radiation from the containment and other plant buildings is negligible. The AP1000 design also provides storage of refueling water inside the containment instead of in an outside storage tank that eliminates it as a radiation source.

12.4.2.2 Doses due to Airborne Radioactivity

Subsection 11.3.3 discusses doses at the site boundary due to activity released as a result of normal operations.

12.4.3 Combined License Information

This section contained no requirement for additional information.

12.4.4 Not Used

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Table 12.4-1
Dose Estimate for Reactor Operations and Surveillance

Work Description	Cycle Dose (man-rem)
Operation Supervision	
On-Line Exposure	2.39
Routine Patrols and Inspections	0.387
Tagging	0.072
Testing	0.177
Health Physics	1.168
Shielding	0.356
Calibration of Instrumentation	0.545
Total Collective Dose (annual)	3.397
Total Collective Dose (per fuel cycle)	5.095

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Table 12.4-2
Dose Estimate for Routine Inspection and Maintenance

Work Description	Annual Dose (man-Rem)
Valve Adjustment / Repacking	2.117
Pressurizer Spray Valve Maintenance	0.379
SG Sludge Lance	2.543
Scaffolding Work (erection, removal)	0.253
Snubber Inspections	0.308
Insulation-related Tasks	0.287
Miscellaneous Work	1.775
SG Secondary Side Inspection	0.963
Ex-core Detector Work (removal, maintenance, replacement)	1.880
IRWST / Annulus Maintenance	0.103
Total Collective Dose (annual)	10.608
Total Collective Dose (per fuel cycle)	15.912

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Table 12.4-3
Not Used

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Table 12.4-4
Dose Estimate for Sludge Lancing of Steam Generators

Activity	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Move equipment into containment[c]	2.5	10	4	0.1
Remove insulation and handhole cover[b]	15	2	3	0.09
Initial sludge lance containment equipment setup[c]	6.25[c]	5	12	0.375
Install lance on handholes[b]	15	2	4	0.12
Operate water lance[b]	15	2	12	0.36
Cleanliness Inspection[b]	15	2	2	0.06
Foreign object search and retrieval[b]	15	2	8	0.24
Sludge lance equipment move across containment (SG1 to SG2)[c]	2.5	5	12	0.15
Final tear down and stage sludge lance equipment[c]	15	5	12	0.90
Remove equipment[c]	6.25[d]	10	4	0.25
Install handhole cover and insulation[b]	15	2	5	0.15
Total ORE for both SG = 3.815 man-Rem / 18 month Annual total ORE for two SGs = 2.543 man-Rem/year[a]				

Notes:
a. The dose calculated based on an 18-month fuel cycle bounds plant operation with a 24-month fuel cycle.
b. These activities are performed twice per outage; once for each steam generator.
c. These activities are performed once per outage.
d. Only about 30% of the time for setup and demobilization is spent at the handholes, while the rest of the time is spent outisde of the SG compartment shield.

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Table 12.4-5
Dose Estimate for Visual Examination of
Steam Generator Secondary Side

Activity[d]	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Remove insulation and two manway covers[a]	0.4	2	7	0.0056
Inspection equipment pack-in and set-up / check-out	4	2	8	0.064
Inspect separator orifices and feedwater ring[a]	4	2	8	0.064
Install two manway covers and insulation[a]	0.4	2	9	0.0072
Remove handhole cover and insulation	15	2	1	0.030
Photograph support plates	15	2	12	0.360
Install handhole cover and insulation	15	2	5	0.150
Teardown / move and set-up inspection equipment (SG A to SG B)	6.875[c]	2	6	0.083
Total ORE for two SGs = 1.444 man-Rem / 18 months Annual total ORE for two SGs = 0.963 man-Rem/year[b]				

Notes:
a. Secondary side water level at the lower deck plate.
b. The dose calculated based on an 18-month fuel cycle bounds plant operation with a 24-month fuel cycle.
c. Data from existing plants show that 34-35% of the time will be spent at the SG, while the rest of the time will be spent moving the equipment through containment.
d. All of the activities are completed twice per outage, except for the “Teardown / move and set-up inspection equipment (SG A to SG B)” activity, which is performed once per outage.

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Table 12.4-6
Dose Estimate for Inservice Inspection

Component	Cycle Dose (man-Rem)
Valve Bodies and Boltings	1.05
Reactor Coolant Loop Piping and Supports	0.15
Other Piping	0.30
Heat Exchanger Shells	0.20
Pressurizer Shell	0.18
Pumps	0.06
Tank Shells and Supports	0.03
Filter Housing and Supports	0.02
SG Primary Side In-service Inspection	2.971
SG Eddy Current Testing	3.862
Reactor Head and Reactor Vessel	0.523
PRHR Inspection in IRWST	0.077
Total Dose (per Fuel Cycle)	9.423
Total Dose (Annual)	6.282

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Table 12.4-7 (Sheet 1 of 2)
Dose Estimate for Steam Generator Eddy Current Tube Inspection and Tube Plugging

Activity	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Move equipment into containment	0.8	8	4	0.0256
Install ventilation equipment	15	4	2	0.120
Remove insulation on both manway covers	15	2	1	0.030
Remove both manway covers with handling fixture	20	3	4	0.240
Remove both manway inserts	60	2	0.5	0.060
Insert nozzel dams cold & hot legs (manual entries)	2,000/20[(a)]	3	1.5	0.1567
Install robotic manipulator hot leg, set up eddy current (EC)	20	2	4	0.160
Perform EC exam of 3340 tubes (33 1/3%)	0.4	1	8	0.0032
Remove EC end effector and replace with mechanical plugging end effector	20	1	1	0.020
Insert plugs in three tubes	20	1	0.75	0.015
Transfer robotic arm to cold leg	20	2	2	0.080
Cold leg eddy current	0.4	1	2	0.0008
Remove EC end effector and replace with mechanical plugging end effector	20	1	1	0.020

Note:
a. Nozzle dam installation involves a limited amount of manual access to the SG, requiring an individual to spend about two minutes in a 2,000 mrem/hr field.

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Table 12.4-7 (Sheet 2 of 2)
Dose Estimate for Steam Generator Eddy Current Tube Inspection and Tube Plugging

Activity	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Insert plugs in 3 tubes	20	1	0.75	0.015
Remove robotic manipulator, equipment teardown	20	2	3	0.120
Remove nozzle dams hot & cold legs	2,000/20[b]	3	2	0.1867
Install manway inserts	60	2	1	0.120
Install both manway covers with handling fixture	20	3	6	0.360
Replace insulation both manway covers	20	2	2	0.080
Remove ventilation equipment	20	2	2	0.080
Move equipment out of containment	0.8	8	6	0.0384
Total SG special maintenance ORE = 1.931 man-Rem / 18 months Total SG special maintenance ORE for two SGs = 3.862 man-Rem / 18 months Annual total ORE for two SGs = 2.575 man-Rem/year[c]				

Note:

b. This task involves a limited amount of manual access to the SG, requiring an individual to spend about two minutes in a 2,000 mrem/hr field.

c. The dose calculated based on an 18-month fuel cycle bounds plant operation a 24-month fuel cycle.

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Table 12.4-8
Dose Estimate for Steam Generator
Inservice Inspection (10-Year Interval)

Activity	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Ultrasonic Inspection				
Tubesheet-to-channel head weld	40	2	48	3.84
Inlet nozzle-to-safe end butt weld	40	2	12	0.960
Outlet nozzle-to-RCP pump casing butt weld	2000/0.04[(a)]	2	120	0.343
PRHR nozzle-to-safe end butt weld	40	2	12	0.960
Elliptical head knuckle-to-upper shell weld	0.4	2	48	0.0384
Tubesheet-to-lower shell weld	40	2	48	3.84
Start-up feedwater nozzle-to-shell weld	4	2	12	0.096
Main feedwater nozzle-to-shell weld	4	2	12	0.096
Main feedwater nozzle inside radius section	4	2	12	0.096
Dye Penetrant Inspection				
Inlet nozzle-to-safe end butt weld	40	2	4	0.320
Outlet nozzle-to-RCP pump casing butt weld	40	2	8	0.640
PRHR nozzle-to-safe end butt weld	40	2	4	0.320
CVS nozzle-to-channel head weld	40	1	1	0.040
Visual Inspection				
Inlet nozzle-to-safe end butt weld	60	1	4	0.240
Outlet nozzle inside radius section	60	1	8	0.480
PRHR nozzle inside radius section	60	1	1	0.060
Studs and nuts less than 2" diameter	20	2	4	0.160
Pedestal steam generator support	15	1	2	0.030
Magnetic Particle Inspection				
Start-up feedwater nozzle-to-shell weld	4	2	4	0.032
Main feedwater nozzle-to-shell weld	4	2	4	0.032
Welded attachments; intermediate and upper lateral support trunnions and support pads	15	2	16	0.480
Total ORE from first 10-year inspection interval = 26.2068 man-Rem. Total ORE for remaining five 10-year inspection intervals = 18.527 man-Rem. Outage average over 60-year (40 18-month outages) life of plant = 2.971 man-Rem/outage. Annual average over 60-year life of plant = 1.980 man-Rem/year				

Note:
a. 10 minutes of this task requires a worker inside of the channel head to set up equipment.

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Table 12.4-9
Dose Estimate for Special Maintenance Operations

Work Description	Annual Dose (man-rem)
Emergent Work	2.135
Modifications	0.461
Total Collective Dose (per Fuel Cycle)	2.597
Total Collective Dose (Annual)	1.731

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Table 12.4-10
Dose Estimate for Waste Processing

Work Description	Annual Dose (man-rem)
Routine rad waste handling and decontamination	0.740
Dry active waste processing	0.586
Waste liquid processing	0.237
Trash, laundry, and decon activities	1.206
Filter change-outs / Resin sluicing	1.478
Tri-nuclear filters	0.209
Total Collective Dose (per Fuel Cycle)	4.456
Total Collective Dose (Annual)	2.971

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Table 12.4-11
Design Improvements That Reduce Refueling Doses

Improved Design/Method	Reference Design/Method
Integrated RV Head Package	Conventional RV head package
RV Head Insulation with Suitcase-Type Fasteners and Permanent ID Markings	Insulation fastened with screws (no markings)
Combination Thermocouples and Flux Detectors	Top-mounted thermocouples and bottom-mounted flux detectors
Fuel Transfer Tube Hatch/Blind Flange	Bolted cover
Quick-Acting Stud Tensioner	Threaded-on stud tensioner
Pass and One-Half Stud Tensioning Procedure	Three-pass stud tensioning procedure
Electrical-Driven Stud Spin-Out Tool	Air-driven, spin-out tool
Permanent Cavity Seal Ring	Bolted or inflatable seal ring
Expandable Stud Hole Plugs	Threaded stud hole plugs
Shielded RV Head Storage Stand	Nonshielded stand
Smooth-Finish Reactor Cavity Liner (#1 Finish)	Rough-finish reactor cavity liner

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Table 12.4-12
Dose Estimate for Refueling Activities

Refueling Operations Work Description	Dose (man-rem)
Preparation	0.207
Reactor Disassembly	1.859
Fuel Shuffle	0.571
Reactor Reassembly	2.171
Clean-Up	0.088
Total Refueling Dose:	5.385[(b)]
Average Annual Dose:	3.590

Note:

a. Based on an 18-month fuel cycle. The stated dose bounds operation with a 24-month fuel cycle.

b. The sum of refueling activity dose estimates is increased by 10% to consider small changes in work activity time durations, the number of personnel exposed, and the addition of unplanned outage tasks.

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12.5 Health Physics Facilities Design

12.5.1 Objectives

The health physics (HP) facilities are designed with the objectives of:

- Providing capability for administrative control of the activities of plant personnel to limit personnel exposure to radiation and radioactive materials as low as reasonably achievable (ALARA) and within the guidelines of 10 CFR 20.
- Providing capability for administrative control of effluent releases from the plant to maintain the releases ALARA and within the limits of 10 CFR 20 and the plant Technical Specifications.
- Providing capability for administrative control of waste shipments from the plant to meet applicable requirements for the shipment and receipt of the material at the storage or burial site.

12.5.2 Equipment, Instrumentation, and Facilities

The health physics (HP) facilities are located at elevation 100′-0″ in the annex building. See Figure 1.2-201 for a plan view of elevation 100′-0″ of the annex building.

12.5.2.1 Access and Exit of Radiologically Controlled Areas

Access to the radiologically controlled area (RCA) encompassing the containment and potentially contaminated areas of the annex, auxiliary, and radwaste buildings is normally through the entry/exit area of the health physics section of the annex building. Exit from the RCA is at the same location.

12.5.2.2 Facilities

The ALARA briefing room is located off the main corridor immediately beyond the main entry to the annex building. Near this room are several offices that may be used for other health physics functions.

Changes rooms are provided where radiation workers remove street clothes and put on modesty garments. These rooms are provided with lockers, wash sinks, showers and toilet facilities.

Radiation workers don anti-contamination clothing in the protective clothing pickup and suitup room. Workers then proceed to the central health physics booth.

Personnel access to and from the RCA is controlled at the health physics booth at the entry/exit points of the health physics area. Logging into the Radiation Permit System and issuance of dosimetry is also handled at this location. The health physics booth is equipped with computer terminals, desks, filing cabinets, and shelves, and other facilities needed for effective control and monitoring of radiation workers in the RCA. Workers are logged into a radiation exposure tracking system. The health physics and security log-in functions are integrated. Facilities and equipment are provided at the health physics booth for the following functions:

- Issuing respirators, as needed
- Issuing radiation dosimetry, as required

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- Updating radiation work permits as needed based on information provided by health physics at local control points and at the work locations

The booth has a counter such that the health physics personnel can easily monitor the flow of workers. It is located adjacent to and visible from the health physics pickup and suitup room.

As radiation workers exit the work areas they go through personnel contamination monitors, shower for decontamination if needed, and receive radiologically controlled first-aid if needed. The health physics area contains the personnel contamination monitoring equipment, decontamination shower facilities, and first-aid equipment.

The hot machine shop is located at elevation 107′-2″ in the south end of the annex building. Contaminated equipment can be decontaminated at the facility and maintenance and repair operations can be performed in a low radiation background area with in the RCA and with appropriate radiation protection and contamination control measures in place.

12.5.2.3 Whole Body Counting Instrumentation

The whole body counter(s) is located in a low background radiation area of the plant. The whole body counting equipment is capable of detecting fractional body burdens of gamma emitting radionuclides.

12.5.2.4 Portable Survey Instrumentation

Portable radiation survey instrumentation is stored at the access control health physics booth and at in-plant control points. This instrumentation allows plant personnel to perform radiation, contamination, and neutron surveys, as needed, as well as collect samples for airborne analysis. Adequate facilities are provided in accordance with Subsection 12AA.5.3 for radioactivity analysis laboratory facilities and for calibration of survey instruments.

12.5.2.5 Other Health Physics Instrumentation

The area radiation monitoring system is installed in areas where it is desirable to have constant dose rate information. Monitors indicate dose rate in the control room and provide appropriate alarms upon reaching a preset dose rate. Fixed continuous airborne radioactivity monitors are also provided at strategic locations, where personnel exposure to airborne radionuclides is likely. More information on these fixed instruments is given in Sections 12.3 and 11.5.

12.5.3 Other Design Features

12.5.3.1 Radiation Protection Design Features

Specific design features for maintaining personnel exposure ALARA and plant shielding provisions are incorporated into the plant design. These features are described in Section 12.3.

12.5.3.2 Job Planning Facilities

Areas are provided where personnel may study, as appropriate: blueprints, drawings, photographs, videotapes, previous inspection reports, previous radiation and contamination surveys, or previous RWPs appropriate to the particular job prior to entry into radiation areas to perform inspections. Work rooms are provided where equipment is checked or calibrated to verify it is operating properly prior to entry into the radiation area. The ALARA briefing and operational support room in the annex building is an example of such a facility where job planning and ALARA briefing and debriefing activities can take place.

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12.5.3.3 Radwaste Handling

The handling of radwaste has been minimized by plant design. Some of the activities involving radwaste or radioactively contaminated materials are performed offsite or using mobile equipment brought onsite. Cleaning of protective clothing and respiratory protective equipment are activities that are performed offsite or in mobile equipment.

The radwaste system is shielded and incorporates remotely operated liquid and solid radwaste systems. The systems are designed to minimize operator exposure in waste processing and handling operations. The liquid radwaste system and solid waste handling system are described in Chapter 11.

12.5.3.4 Spent Fuel Cask Loading and Shipping

Spent fuel handling and loading of a shipping cask is designed to be performed underwater, using the fuel handling cranes and/or manual extension tools.

Some of the design features included to maintain exposure ALARA are:

- Maintenance of at least 8.75 feet of water above the active fuel in a fuel assembly to minimize direct radiation.
- Purification of fuel pool water to minimize exposure due to water activity.
- Cooling of the spent fuel pool water.
- Providing continuous air sampling while moving fuel to evaluate airborne activity.

12.5.3.5 Normal Operation

The plant is designed so that significant radiation sources are minimized, locally shielded, and/or located in shield cubicles. Much of the instrumentation required for normal operation reads out remotely in the control room or in other low radiation areas. Instrumentation that cannot be placed remotely or that is read infrequently is situated, where possible, so that it can be read from the entrance to the cubicle or from a low radiation area within the cubicle.

Area radiation monitoring equipment, which is included as part of the radiological monitoring system, is available and provides indication of radiation levels and local alarms. The ventilation system is designed to minimize spread of airborne contamination.

12.5.3.6 Sampling

Provisions are made for sampling of radioactive systems in the sampling room. Protective clothing and gloves are available when sampling radioactive systems to prevent contamination of personnel.

12.5.3.7 Surface Coatings

Special coatings are applied to walls and floors of areas containing radioactive fluids, which aid in decontaminating these areas.

12.5.4 Controlling Access and Stay Time

Areas in the plant are classified as non-radiation areas and restricted radiologically controlled areas for radiation protection purposes. Restricted areas are further categorized as radiation areas, high

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radiation areas, airborne radioactivity areas, contamination areas, and radioactive materials areas, to comply with 10 CFR 20 and plant procedures and instructions.

The radiation protection calibration lab in the annex building is a Zone II restricted area and access control is consistent with Subsection 12.3.1.2.

Entrance to the RCA area is normally through the access control area at the health physics area entry/exit location in the annex building, see Subsection 12.5.2.

High and very high radiation areas are segregated and identified in accordance with 10 CFR 20. The entrances to high and very high radiation areas are locked or barricaded and equipped with audible and/or visible alarms, as required.

A closed circuit television system may be installed in high radiation areas to allow remote monitoring of individuals entering high radiation areas by personnel qualified in radiation protection procedures.

12.5.5 Combined License Information

The organization and procedures used for radiological protection, and to provide methods so that personnel radiation exposures are maintained ALARA, are addressed in Appendix 12AA.

Radiation protection program information is included in Appendix 12AA.

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Appendix 12AA Radiation Protection Program Description

This appendix incorporates NEI 07-03A, Generic FSAR Template Guidance for Radiation Protection Program Description. The numbering of NEI 07-03A is revised from 12.5# to 12AA.5#. Table 13.4-201 provides milestones for radiation protection program implementation.

12AA.5 Radiation Protection Program

A radiation protection program is developed, documented, and implemented through plant procedures that address quality requirements commensurate with the scope and extent of licensed activities, sufficient to ensure compliance with the provisions of 10 CFR Parts 19, 20, 50, 52, and 71 and consistent with the guidance in Regulatory Guides 1.8, 1.206, 8.2, 8.4, 8.6, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, 8.38, and the consolidated guidance in NUREG-1736.

In accordance with 10 CFR 20, Subpart B, the purpose of the radiation protection program is to maintain occupational and public doses below regulatory limits and as low as is reasonably achievable (ALARA). To achieve this, the program will include:

I. A documented management commitment to keep exposures ALARA;

II. A trained and qualified organization with sufficient authority and well-defined responsibilities; and

III. Adequate facilities, equipment, and procedures to effectively implement the program.

The operational radiation protection program is implemented in stages consistent with the following milestones:

1. Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18), and thereafter, when such radioactive materials are possessed under this license, the following radiation protection program elements will be in place:

    a. Organization – A radiation protection supervisor and at least one (1) radiation protection technician, each selected, trained and qualified consistent with the guidance in Regulatory Guide 1.8.

    b. Facilities – A facility or facilities to support the receipt, storage and control of non-exempt radioactive sources in accordance with 10 CFR 20.1801, 20.1802, and 20.1906.

    c. Instrumentation and Equipment – Adequate types and quantities of instrumentation and equipment will be selected, maintained, and used to provide for the appropriate detection capabilities, ranges, sensitivities, and accuracies to conduct radiation surveys and monitoring (in accordance with 10 CFR 20.1501 and 20.1502) for the types and levels of radiation anticipated for the non-exempt sources possessed under this license.

    d. Procedures – Procedures will be established, implemented and maintained sufficient to maintain adequate control over the receipt, storage, and use of radioactive materials possessed under this license and as necessary to assure compliance with 10 CFR 19.11 and 19.12 and the applicable portions of 10 CFR Part 20, commensurate with the types and quantities of radioactive materials received and possessed under this license.

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e. Training – Initial and periodic training will be provided to individuals responsible for the receipt, control or use of non-exempt radioactive sources possessed under this license in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 1.8, 8.13, 8.27, and 8.29.

2. Prior to receiving reactor fuel under this license, and thereafter, when reactor fuel is possessed under this license, plant procedures on criticality accident requirements will be established, implemented and maintained and radiation monitoring will be provided in accordance with 10 CFR 50.68, in addition to the radiation protection program elements specified under item 1, above. During initial fuel receipt and storage, portable radiation monitors meeting the functional requirements described in Subsection 11.5.6 may be used in lieu of permanently installed fuel handling and storage area radiation monitors.

3. Prior to initial loading of fuel in the reactor, all of the radiation program functional areas described in Appendix 12AA will be fully implemented, with the exception of the organization, facilities, equipment, instrumentation, and procedures necessary for transferring, transporting or disposing of radioactive materials in accordance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71. In addition, the position of radiation protection manager (as described in Section 13.1) will be filled and at least one (1) radiation protection technician for each operating shift, selected, trained, and qualified consistent with the guidance in Regulatory Guide 1.8, will be onsite and on duty when fuel is initially loaded in the reactor, and thereafter, whenever fuel is in the reactor.

4. Prior to initial transfer, transport or disposal of radioactive materials, the organization, facilities, equipment, instrumentation, and procedures will be in place as necessary to assure compliance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71.

The radiation protection program content and effectiveness of implementation are reviewed periodically (at least annually) pursuant to plant procedures.

12AA.5.1 Management Policy

Plant management will establish written policy on radiation protection that is consistent with the guidance in Regulatory Guides 8.8 and 8.10, including management's commitment to:

I. Assure that the plant is designed, constructed, and operated such that occupational and public radiation exposures and releases of licensed radioactive materials are ALARA;

II. Comply with regulatory radiation requirements, dose limits, and limits on release of radioactive materials;

III. Implement and maintain a radiation protection program to keep radiation doses below regulatory limits and ALARA;

IV. Assure that each manager and supervisor in the plant organization understands and is held accountable for implementing his or her responsibility to integrate appropriate radiation protection controls into work activities;

V. Assure that each individual working at the facility understands and accepts the responsibility to follow radiation protection procedures and instructions provided by radiation protection staff and to maintain his or her dose ALARA;

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VI. Provide the radiation protection manager the delegable authority to stop work or order an area evacuated (in accordance with approved procedures) when, in his or her judgment, the radiation conditions warrant such an action and such actions are consistent with plant safety;

VII. Establish a direct reporting chain of the Radiation Protection Manager to the Plant Manager that is at the same reporting level as, but independent of, the reporting chains for Operations and Maintenance.

VIII. Establish an ALARA Committee with delegated authority from the Plant Manager that includes, at a minimum, the managers of Operations, Maintenance, Work Control, Engineering and Radiation Protection to help assure effective implementation of line organization responsibilities for maintaining worker doses ALARA.

12AA.5.2 Organization

Qualification and training criteria for site personnel are consistent with the guidance in Regulatory Guide 1.8 and clarified in Appendix 1A and are described in Chapter 13. Specific radiation protection responsibilities for key positions within the plant organization are described in Section 13.1.

Subsections 12.5.2.1 through 12.5.2.5 of NEI 07-03A are not incorporated into Appendix 12AA.

12AA.5.3 Facilities, Instrumentation and Equipment

Adequate facilities, instrumentation and equipment are provided to support implementation of the radiation protection program during routine operations, refueling and other outages, abnormal occurrences, and accident conditions. The types and characteristics of facilities, instrumentation, and equipment provided are consistent with the guidance in Regulatory Guides 1.97 (and guidance provided in Branch Technical Position 7-10, Revision 5 to NUREG 0800), 8.2, 8.4, 8.6, 8.8, 8.9, 8.10, 8.15, and 8.28 and the criteria in NUREG-0737, Items II.B.3 and III.D.3.3.

Subsection 12.5.3.1 of NEI 07-03A is not incorporated into Appendix 12AA. Facilities are described in Subsection 12.5.2.2.

12AA.5.3.1 Monitoring Instrumentation and Equipment

Radiation monitoring instrumentation and equipment are selected, maintained and used to provide the appropriate detection capabilities, ranges, sensitivities and accuracies required for the types and levels of radiation anticipated at the plant and in the environs during routine operations, major outages, abnormal occurrences, and postulated accident conditions. The quantities of instrumentation and equipment are sufficient to meet the anticipated needs of the plant during all anticipated conditions –taking into account the amount of instrumentation and equipment that may be unavailable at any one time due to periodic testing and calibration, maintenance, and repair.

The types and nominal characteristics of the instrumentation are as follows:

Laboratory and Fixed Instrumentation

- Multi-channel gamma analysis system to identify and measure gamma emitting radionuclides in solid, liquid and gaseous samples. Some of the sample types analyzed include primary reactor coolant, liquid and gaseous waste and airborne contaminants.
- Counters to measure gross beta and gamma activity.
- A low background counter to detect and measure gross alpha activity.

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- A liquid scintillation counter to measure tritium in liquid and gaseous samples.
- A whole-body counter to detect and quantify personnel intakes of radioactivity.
- Fixed instrumentation, such as small article monitors, hand and foot monitors, and portal monitors, to monitor for contamination on personnel, materials, and equipment.

Portable Monitoring Instrumentation and Equipment (nominal ranges are given in parentheses for illustrative purposes only)

- Beta-gamma count rate survey meters (0-50,000 cpm) to detect radioactive contamination on surfaces and for low level exposure rate measurements.
- Low-range (0-50 mR/hr) and high range (0-1,000 R/hr) beta-gamma survey meters and ion chamber survey meters (0-50 R/hr) are used to measure the full range of dose rates necessary for radiation protection purposes during routine operations (including underwater operations), abnormal occurrences and accidents.
- Beta-gamma survey meters (0-10,000 R/hr) to monitor the plant and environs during and following an accident.
- Count rate meters (0-500,000 cpm) to monitor directly for alpha activity.
- Neutron survey instruments (0-5 rem/hr) to measure neutron dose rates for radiation protection purposes (including underwater operations).
- High and low volume air samplers equipped with appropriate filter media are used to take grab samples that are analyzed to assess airborne radioactivity concentrations, estimate actual or potential exposure, and to determine respiratory protection measures.
- Continuous air monitors (CAMs) provide a means to observe trends in airborne radioactivity concentrations. CAMs equipped with local alarm capability are used in occupied areas where needed to alert personnel to sudden changes in airborne radioactivity concentrations.
- Hand-held friskers to detect radioactive contamination.
- Portable air sampling and analysis system to determine airborne radioiodine concentrations during and following an accident consistent with the criteria in NUREG-0737, Item III.D.3.3.
- Portable sampling and onsite analysis capability to assess airborne radiohalogens and particulates released during and following an accident consistent with the criteria in Regulatory Guide 1.97 (and guidance provided in Branch Technical Position 7-10, Revision 5 to NUREG 0800).

Personnel Monitoring Instrumentation and Equipment

- Individual personnel dosimeters to measure gamma, beta and neutron radiation dose with sensitivities and ranges appropriate to measure the expected levels and types of radiation.
- Direct-reading dosimeters to provide real-time gamma dose information with sensitivities and ranges appropriate to measure the expected levels and types of radiation.
- Special dosimeters to monitor extremity dose with sensitivities and ranges appropriate to measure the expected levels and types of radiation.

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- Personnel air samplers to monitor individual exposure to airborne radioactivity.
- Remote and local reading alarm dosimeters (which may be coupled with direct or electronic surveillance equipment, as necessary).

12AA.5.3.2 Personnel Protective Clothing and Equipment

A sufficient inventory of serviceable personnel protective clothing and equipment is maintained for use during plant operations, refueling and other outages, abnormal conditions, and accidents. Only respirators that are tested and certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA), or otherwise approved by the NRC, are used.

If circumstances arise in which NIOSH tested and certified respiratory equipment is not used, compliance with 10 CFR 20.1703(b) and 20.1705 is maintained.

Personnel protective clothing and equipment includes the following:

- Anti-contamination clothing for both dry and wet work conditions, including heat stress reduction accessories
- Head covers, shoe covers, gloves, and safety-related items
- Full facemask respirators with high-efficiency particulate and charcoal filters
- Pressure demand full facemask air line respirators
- Pressure demand full facemask self-contained breathing apparatus

12AA.5.3.3 Other Protective Equipment

- Portable ventilation systems with HEPA filters
- Temporary containments, tents, and enclosures
- Heat-stress reduction equipment
- Vacuums with HEPA filters
- Portable liquid filtration equipment
- Temporary shielding such as lead and/or tungsten shield bricks, blankets, and curtains.

12AA.5.4 Procedures

Radiation protection procedures are established, implemented and maintained sufficient to provide adequate control over the receipt, possession, use, transfer, and disposal of byproduct, source, and special nuclear material and assure compliance with applicable requirements in 10 CFR Parts 19, 20, 50, 70, and 71. Procedures for radiation protection that include quality assurance requirements are prepared consistent with the guidance in Regulatory Guides 1.8, 8.2, 8.7, 8.8 and 8.10 and the consolidated guidance referenced in NUREG-1736 that is applicable to power reactors. The procedures are implemented by Radiation Protection staff trained and qualified in accordance with the requirements in 10 CFR 50.120 and consistent with the guidance in Regulatory Guide 1.8. Additionally, some procedures are implemented by plant staff trained in accordance with the

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requirements of 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 8.13, 8.27, and 8.29.

12AA.5.4.1 Radiological Surveillance

Radiological surveillance procedures comply with 10 CFR 20.1501 and are consistent with the guidance in Regulatory Guides 8.2, 8.8, and 8.10.

Trained and qualified radiation protection staff will routinely survey accessible areas in the plant and environs to assess the presence and levels of radiation, radioactive contamination, and airborne radioactivity. The instrumentation and techniques used for these surveys are selected based upon the purpose of the survey and the anticipated types and levels of radiation and radioactivity involved. Surveys are performed using effective practices to minimize personnel exposure and avoid the spread of contamination.

The frequency and extent of the surveys will depend upon several factors, such as location, actual or potential radiation levels, plant operational status and work in progress, and accessibility/occupancy. The frequency of surveys may be weekly, monthly, quarterly, semiannually, annually, or as directed by the Radiation Protection Manager. Surveys are performed more frequently in accessible areas subject to changes in radiological conditions. Site specific procedures will define the survey frequencies and extent.

Survey results are recorded and maintained in accordance with the requirements in 10 CFR Part 20. Survey results for accessible areas are posted or otherwise made available to provide adequate notice to workers of radiological conditions.

Radiation surveys are routinely performed for detection of beta and gamma radiation. Surveys for neutron radiation are performed in accessible areas where such radiation may be present.

Area contamination surveys are routinely performed for the detection of removable and fixed beta-gamma contamination. Surveys for alpha contamination are performed where alpha contamination is anticipated. Alpha contamination surveys will also be performed periodically as a check to verify that alpha contamination is not present. Personnel will monitor themselves for contamination after exiting from contaminated areas and at exit points from the RCA or other Restricted Areas with a potential for contamination. Materials and equipment are monitored for contamination after removal from contaminated areas and prior to being released from the RCA or other Restricted Areas with a potential for contamination.

Surveys to assess airborne radioactivity levels are performed with continuous air monitors (CAMs) and by taking grab samples (using portable low or high volume air samplers) with appropriate media for collecting particulate, iodine, gas, or tritium samples. In order to warn personnel of changing airborne conditions, CAM alarm set points are set at a fraction of the concentration values given in 10 CFR Part 20, Appendix B, Table 1, Column 3, for radionuclides expected to be encountered. Air monitoring and sampling are sufficient to identify the potential hazard(s), determine the need for and verify the effectiveness of process and engineering controls, permit proper selection of respiratory protection equipment, and estimate doses from intakes.

Emergency operating procedures include provisions for use of a portable monitoring system, consistent with the criteria in NUREG-0737, Item III.D.3.3, to sample and analyze for radioiodine in areas of the plant during and following an accident. Procedures include methods for taking and analyzing samples in the field, as well as for analyzing samples in the count room facility, accounting for techniques to reduce counting system saturation from a high-activity sample.

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Instrumentation and equipment used to perform surveys are calibrated prior to initial use, after performance of maintenance or repairs that might affect the calibration, and at least annually. Operational checks to test function or response are made daily for continuously operating instrumentation and equipment (e.g., friskers, portal monitors, and continuous air monitors) and prior to use or daily, whichever is less frequent, for other instrumentation and equipment. Operational checks are performed for emergency and special use instrumentation and equipment on a regular schedule as specified in written procedures.

Survey records and records of calibration and maintenance of instrumentation and equipment used for surveys are documented and maintained in accordance with applicable requirements in 10 CFR 20.2101-20.2110.

12AA.5.4.2 Methods to Maintain Exposures ALARA

Methods to maintain exposures ALARA in accordance with Regulatory Guides 8.8 and 8.10 are included in radiation protection procedures, as well as applicable operating and maintenance procedures. Key ALARA operational policies and considerations are described in Section 12.1. Some examples of the types of methods that will be used to maintain exposures ALARA are discussed below for the following operational categories.

Refueling

After the reactor coolant system is depressurized, it is degassed as needed and sampled to verify that the gaseous radioactivity is low, prior to removing the reactor head. The Radiation Work Permit (RWP) system is used to maintain positive radiological control over work in progress. Prior to and during refueling, the refueling pool water is continually purified in order to maintain exposures from activity in the water ALARA. During refueling operations, irradiated fuel assemblies are maintained underwater at all times. By following these procedures, exposures from refueling operations are maintained ALARA.

Inservice Inspection

Prior to entry into radiation areas to perform inspections, personnel should study, as appropriate: blueprints, drawings, photographs, videotapes, previous inspection reports, previous radiation and contamination surveys, and/or previous RWPs appropriate to the particular inspection/job to be performed. This will acquaint personnel with the inspection location, room layout and equipment configuration, the work to be done, and radiation and contamination levels previously experienced at the location. Surveys are performed to the extent required to determine current contamination and/or radiation levels. From this data, previous data, and past work experience of personnel for similar jobs/inspections performed, an RWP (paragraph 12.5.4.5) is issued. Equipment is checked and/or calibrated to verify it is operating properly prior to entry into the radiation area. Temporary shielding will be used, where practicable, to reduce personnel radiation exposures.

Routine Maintenance

Routine maintenance is comprised of the categories of preventive maintenance (planned and scheduled maintenance such as lubrication, adjustments, and tests) and corrective maintenance (unscheduled maintenance such as valve packing, pump seal replacement, and stopping leaks). Procedures are usually written for preventive maintenance jobs and for some recurring corrective maintenance jobs. These procedures specify the precautions to be taken to minimize personnel exposure while performing the maintenance. The procedures list the required lubricants, special tools and equipment, and the acceptance standards. This serves to minimize the time spent in the radiation area and thereby minimize personnel dose.

In addition, the preventive maintenance procedure normally states whether an RWP is required. When the RWP is issued, the radiation and/or contamination levels are listed, shielding is specified, if

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appropriate, and additional specific instructions are given to personnel. For corrective maintenance jobs in radiation areas, a similar approach is used.

Extension tools are used when practical to minimize personnel dose when working on radioactive components/equipment. Detailed surveys are performed and the RWP is issued (if required) with specific instructions. The individuals performing the work may be required to read procedure manuals or may be shown pictures or sketches of the work area to aid in understanding what is to be accomplished, how it is to be accomplished as safely and quickly as possible, and what the acceptance criteria are for completing the job. At the discretion of health physics personnel, additional requirements may be imposed to reduce personnel exposures.

After the job is completed, debriefings may be conducted to obtain input from personnel actually performing the work, as well as from supervisory and support personnel. This will assist in revising procedures for ALARA considerations.

Calibration

Calibration of most ranges of the portable gamma detection instruments is performed inside a shielded calibrator, thereby eliminating a large portion of the exposure received from calibration of portable instruments. Portable sources used to calibrate fixed instruments (such as the area radiation monitoring system) are transported in shielded containers to minimize personnel exposure.

Where possible, fixed instruments requiring routine calibration are situated so that the necessary test signals needed for calibration can be inserted from a low radiation area with the instruments in place.

12AA.5.4.3 Posting and Labeling

Procedures for posting and labeling will assure compliance with 10 CFR 20.1901, 20.1902, 20.1903, 20.1904, and 20.1905.

Based on current survey results, Radiation Areas, High Radiation Areas, Very High Radiation Areas, Airborne Radioactivity Areas, and Radioactive Materials Areas are posted in accordance with the requirements in 10 CFR 20.1901, 20.1902, and 20.1903. Containers of licensed radioactive materials are labeled in accordance with 10 CFR 20.1904 and 20.1905.

Criteria and procedures are established for posting areas and marking items (e.g., tools and equipment) to indicate the presence of fixed or removable surface contamination. Areas posted to indicate the presence of removable contamination, are referred to hereafter as “Contamination Areas.”

“Posted areas”, as used in Section 12.5, refers to Radiation Areas, High Radiation Areas, Very High Radiation Areas, Airborne Radioactivity Areas, Contamination Areas, and Radioactive Materials Areas.

12AA.5.4.4 Access Control

Procedures for access control will assure compliance with 10 CFR 20.1902, 20.1903, 20.1601, and 20.1602 and are consistent with the guidance in Regulatory Guide 8.38.

Access to posted areas is restricted and controlled, at a minimum, through the use of instructions to workers, radiation work permits, caution signs, and barriers. Access to High and Very High Radiation Areas is controlled consistent with the guidance in Regulatory Guide 8.38, including the use of alternative methods for access control as described in the regulatory guide and specified in plant technical specifications.

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Table 12AA-201 identifies plant areas designated as Very High Radiation Areas (VHRAs), lists corresponding plant layout drawings showing the VHRA in Section 12.3, specifies the condition under which the area is designated VHRA, identifies the primary source of the VHRA, and summarizes the frequency of access and reason for access. VHRAs are listed as Radiation Zone IX, which corresponds to a dose rate greater than 500 rad/hr.

In each of the VHRAs, with the exception of the Reactor Vessel Cavity and Delay-Bed / Guard-Bed Compartment, the primary radioactive source is transient (such as fuel passing through the transfer tube), removable (such as resin in the demineralizers), or can be relocated. When the primary source is removed, the dose rate in each of these areas will be less than Zone IX and, in effect, the area will no longer be a VHRA. With planning, the need for human entrance to a VHRA when the primary source is present can be largely or entirely avoided.

In addition to the access control requirements for high radiation areas, the following control measures are implemented to control access to very high radiation areas in which radiation levels could be encountered at 500 rads or more in one hour at one meter from a radiation source or any surface through which the radiation penetrates:

- Sign(s) conspicuously posted stating GRAVE DANGER, VERY HIGH RADIATION AREA.

- Area is locked. Each lock shall have a unique core. The keys shall be administratively controlled by the functional manager in charge of radiation protection as described in Section 13.1.

- Plant Manager's (or designee) approval required for entry.

- Radiation Protection personnel shall accompany person(s) making the entry. Radiation Protection personnel shall assess the radiation exposure conditions at the time of the entry.

A verification walk down will be performed with the purpose of verifying barriers to the Very High Radiation Areas in the final design of the facility are consistent with Regulatory Guide 8.38 guidance as part of the implementation of the Radiation Protection and ALARA programs on the schedule identified in Table 13.4-201.

Unescorted access to Radiation Areas or Radioactive Materials Areas will require, at a minimum, authorization by Radiation Protection, the use of an RWP, and instruction of individuals gaining unescorted access in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guide 8.13. In addition to the foregoing, unescorted access to Contamination, High Radiation, Very High Radiation, or Airborne Radioactivity Areas will require, at a minimum, training of individuals gaining unescorted access consistent with Regulatory Guides 8.27 and 8.29.

Posted areas will generally be contained within the plant Security Area, i.e., an area to which access is controlled in accordance with 10 CFR Part 73. Unescorted access to the plant Security Area will require instruction of individuals gaining unescorted access in accordance with 10 CFR 19.12.

Areas where significant doses could be received (e.g., High Radiation, Very High Radiation, and Airborne Radioactivity Areas), are generally contained within the plant building complex. A Radiological Controlled Area (RCA) is established to encompass the plant building complex to enhance control over access to such areas. Access to the RCA is through a primary access control point or alternate access control points as established by Radiation Protection. Unescorted access to the RCA will require authorization by Radiation Protection, the use of an RWP and instruction and

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training of individuals gaining unrestricted access in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 8.13, 8.27, and 8.29.

Radiation Protection may authorize access to the Security Area, RCA, or a Radiation or Radioactive Materials Area for individuals without instruction or training where such individuals are continuously under the control of a designated escort. The designated escort shall be instructed and trained in accordance with the requirements of 10 CFR 19.12 and the guidance in Regulatory Guides 8.13, 8.27, and 8.29, and shall be instructed on the duties and responsibilities associated with being an escort.

Access by a worker who is a minor (i.e., under the age of 18 years) or a declared pregnant worker to posted areas with a potential for significant exposure, e.g., High Radiation, Very High Radiation, and Airborne Radioactivity Areas is restricted unless otherwise authorized by Radiation Protection.

12AA.5.4.5 Radiation Work Permits

Procedures covering the use of a radiation work permit (RWP) are consistent with the guidance in Regulatory Guide 8.8.

RWPs are issued by Radiation Protection to help ensure adequate protection of personnel for access to and work within areas with a potential for significant exposure. Access to any posted area will require an RWP. An RWP may control access to multiple areas or to a set of related jobs or tasks.

At a minimum, each RWP will include the following information:

- Description of the area(s) to be accessed and work to be performed;
- Designation of personnel or groups covered by the RWP;
- Radiological conditions existing within the area(s) to be accessed, based on current radiological surveys, and anticipated radiological conditions for the time span over which the work is performed (including location of hot spots, radiation gradients, and low dose “waiting areas”);
- Requirements for use of personnel monitoring devices, protective clothing, and respiratory protection equipment;
- Special instructions and a description of special tools, shielding, other equipment utilized to perform work, and any process and engineering controls being employed to minimize exposures; and
- Extent and type of radiation protection monitoring and surveillance to be provided.

As described in Section 12.1, for access to and work within High Radiation and Very High Radiation Areas, the applicable RWP will specify a limitation on staytime or a means for limiting dose received while in the area (e.g., via an alarm set point for an electronic dosimeter).

12AA.5.4.6 Personnel Monitoring

Personnel monitoring procedures are sufficient to assure compliance with 10 CFR Parts 19 and 20 and are consistent with the guidance in Regulatory Guides 8.2, 8.7, 8.9, 8.13, 8.34, 8.35, and 8.36.

Each individual accessing the RCA or a posted area on an unescorted basis, or for whom occupational dose monitoring of external dose is required in accordance with 10 CFR 20, is monitored using an individual monitoring device that is appropriate for monitoring the types of

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external radiation to which the individual is exposed. For individuals who are required to be monitored in accordance with 10 CFR Part 20, if the individual monitoring device does not provide real-time dose information (i.e., the capability for the individual to track his or her own dose as it occurs), then an additional means of monitoring is provided for the individual that fulfills that function.

Individuals accessing the RCA or a posted area on an escorted basis, for whom occupational dose monitoring of external dose is not required in accordance with 10 CFR Part 20, are monitored either with an individual monitoring device worn by the individual or via an individual monitoring device worn by the escort.

Individual monitoring devices that require processing, except for those devices excluded by 10 CFR 20.1501(c), are processed and evaluated by a NVLAP-accredited processor, as appropriate, for the type(s) and ranges of radiation being monitored with the device. Each individual whose internal dose is required to be monitored in accordance with 10 CFR Part 20, or who wears a respirator for radiation protection purposes, or who accesses an Airborne Radioactivity Area, is monitored by means sufficient to identify and quantify intakes in order to be able to estimate his or her committed effective dose equivalent (CEDE) and, as applicable, his or her committed dose equivalent (CDE).

Situations that may result in a person receiving an abnormal or inadvertent intake are evaluated on a case-by-case basis to determine the need for monitoring by means sufficient to identify and quantify intakes in order to be able to estimate the CEDE or CDE, as applicable.

Individuals suspected of having received an intake are evaluated to quantify the intake, if any, in order to estimate the CEDE or CDE, as applicable. In demonstrating compliance with regulatory requirements, effective dose equivalent may be used in lieu of deep dose equivalent consistent with the guidance in Regulatory Issue Summary (RIS) 2003-04 and other related guidance.

Individual monitoring results are reported annually to the individual, and at the request of an individual who is terminating employment or who is requesting this information from a previous employer, in accordance with the requirements in 10 CFR 19.13.

Personnel monitoring records, as well as records associated with testing, calibration, processing, and maintaining instrumentation and equipment used for personnel monitoring, are documented and maintained in accordance with applicable requirements in 10 CFR 20-2101-20.2110.

12AA.5.4.7 Dose Control

Compliance is maintained with the requirements in 10 CFR 20.1201, 20.1202, 20.1203, and 20.1204, as they relate to demonstrating compliance with internal and external occupational dose limits contained in 10 CFR 20, Subpart C. Doses to adult workers are kept below the occupational dose limits in 10 CFR 20.1201. Doses to workers who are minors and declared pregnant workers are kept below the respective occupational dose limits in 10 CFR 20.1207 and 10 CFR 20.1208. Doses to members of the public are kept below public dose limits in 10 CFR 20.1301, which is demonstrated by complying with the requirements of 10 CFR 20.1302.

To the extent practical, procedures and engineered controls based on sound radiation protection principles are used to keep occupational doses and doses to members of the public as low as is reasonably achievable (ALARA). A description of facility design features and engineered controls intended to maintain occupational exposures ALARA is included in Sections 12.3-12.4. A description of systems and facility design features intended to maintain public exposures ALARA is included in Chapter 11.

As described in Sections 12.1, Appendix 12AA and 13.1, management policy is established, and organizational responsibilities and authorities are assigned to implement an effective program for

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maintaining occupational radiation exposures ALARA. Procedures are established and implemented that are in accordance with 10 CFR 20.1101 and consistent with the guidance in Regulatory Guides 8.8 and 8.10. Examples of such procedures include the following:

I. During the construction, pre-operational and operational phases, Radiation Protection will assure that new or modified designs and the selection of equipment are reviewed to assure that measures are considered to minimize occupational and public radiation exposures during operation, refueling, and decommissioning of the plant.

II. Radiation Protection will assure that procedures and methods for operation, maintenance, repair, surveillance, refueling, and other activities that may involve significant exposures are reviewed prior to initial use and periodically thereafter that assure measures are considered to minimize occupational and public radiation exposures. For example, “significant exposures” may include activities that are estimated to involve greater than 1 person-rem of collective dose.

III. For activities involving significant exposures, pre-job briefings are conducted for personnel who will receive the exposures. The briefings are intended to assure that personnel understand the radiological conditions expected to be present and the measures being employed to control and minimize dose. Post-job reviews are performed to evaluate the effectiveness of measures employed to control and minimize dose and to identify and implement improvements to minimize occupational and/or public radiation exposures for future similar activities.

Planned special exposures, as described in 10 CFR 20.1206, if used, will be conducted in accordance with the requirements in 10 CFR 20.2104 and consistent with the guidance in Regulatory Guide 8.35.

12AA.5.4.8 Contamination Control

Contamination control procedures are established to help assure compliance with 10 CFR Parts 20.1406 and 20.1701 and to prevent the unauthorized release of radioactive materials to unrestricted areas.

Areas, items, and personnel are routinely surveyed and monitored for contamination to protect personnel, ensure that contamination control methods are effective and to prevent licensed materials from being released from an RCA or Controlled Area in an unauthorized manner. Areas and items with fixed or removable contamination are posted, labeled, or marked in a conspicuous manner to indicate the presence of contamination.

Personnel accessing Contamination or Airborne Radioactivity Areas are required to use protective clothing and equipment appropriate to the circumstances to prevent personal contamination.

Personnel found with external contamination are decontaminated promptly. Contaminated items are decontaminated or disposed of as radioactive waste or are marked and controlled. Areas that become contaminated are decontaminated as soon and as thoroughly as practical, taking into account factors such as the nature of operations in the area and the potential for exposure associated with the decontamination. The number of accessible contaminated areas within the plant are kept to a minimum.

Facility design and operational procedures are reviewed to identify nonradioactive systems that could possibly become radioactive through interfaces with radioactive systems. Routine sampling and monitoring of these systems is described in the plant radiation monitoring program, and overall guidance is consistent with Bulletin 80-10.

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Practical measures are implemented to prevent the spread of contamination, including:

- Air pressure gradients and airflows are maintained from areas of low potential contamination to areas of higher potential contamination and then to installed filters and/or ventilation systems;
- Leaks and spills are contained promptly and repaired or cleaned up as soon as practical;
- Potentially contaminated systems, equipment, and components are surveyed for the presence of contamination when opened or prior to removal;
- Containments, caches and enclosures are used during maintenance, repairs, and testing, when practical, to contain spills or releases;
- Engineering controls, such as portable ventilation or filtration units to reduce concentrations of radioactivity in air or fluids, are used where practical;
- Criteria for selecting tools, materials, and equipment for use in contaminated areas will include minimizing the use of porous or other materials that are difficult to decontaminate;
- The use of disposable materials that are likely to become contaminated and necessitate disposal as radioactive waste are minimized;
- Areas, surfaces, and tools that are prone to contamination are designed and coated (e.g., using agents to “fix” contamination, such as strippable coatings), as practical, to facilitate decontamination;
- Contaminated tools and equipment are segregated from clean tools and equipment.

This subsection adopts NEI 08-08A (Reference 201), for a description of the operational and programmatic elements and controls that minimize contamination of the facility, site, and the environment, to meet the requirements of 10 CFR 20.1406.

12AA.5.4.9 Respiratory Protection

Respiratory protection procedures will assure compliance with 10 CFR Part 20, Subpart H, and are consistent with the guidance in Regulatory Guide 8.15.

A written policy statement established by the plant management covers the use of process and engineering controls in lieu of respirator use to limit intakes and to limit the routine, non-routine, and emergency use of respirators.

Written procedures are established and implemented that cover the following:

- Monitoring, including air sampling and bioassays;
- Supervision and training of respirator users;
- Fit-testing;
- Respirator selection;
- Breathing air quality;

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- Inventory, control, storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment;
- Recordkeeping; and
- Limitations on periods of use and relief from respirator use.

An assessment is performed to assure that the total effective dose equivalent (TEDE) is maintained ALARA, when respiratory protection equipment is used to limit intakes of radioactive materials.

Airborne radioactivity is minimized by the design and configuration of the plant's heating, ventilation and air conditioning systems (HVAC), the use of enclosures and containments, and good housekeeping practices. Portable air movers and vacuums equipped with HEPA filters to minimize concentrations of radioactivity in air or on surfaces are vented to monitored, filtered discharge pathways.

When it is not practical to apply process and engineering controls to control the concentrations of radioactive materials in the air and maintain the TEDE ALARA, intakes are limited by controlling access to and limiting stay times in Airborne Radioactivity Areas and by using respiratory protection equipment or other controls.

The Radiation Protection Manager will assign to a single individual, knowledgeable in the area of respiratory protection consistent with the guidance in Regulatory Guide 8.15, the overall responsibility to establish and maintain a respiratory protection program and procedures that include:

- air sampling and monitoring sufficient to identify hazards, select proper equipment, and determine doses from intakes;
- conducting surveys and bioassays as necessary to evaluate actual intakes; and
- testing respirators for operability immediately prior to each use.

Only respiratory protection equipment that is tested and certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) is used, unless otherwise authorized by the NRC.

Prior to being fit-tested for a face sealing respirator, and before the first field use of a non-face sealing respirator, individuals are certified as medically fit by a qualified medical practitioner. Recertification of medical fitness is made every twelve months or at a frequency specified by the medical practitioner.

Each respirator user is advised that he or she may leave the area at any time for relief from any conditions (such as equipment malfunction, physical or psychological distress, or communications failure) that might require such relief.

In selecting and using respiratory protection equipment, provisions are made for vision correction, adequate communications, extreme temperature conditions, and concurrent use of other safety or radiological protection equipment.

For circumstances when respiratory protection equipment is used from which an unaided individual would have difficulty extricating himself or herself, and therefore might be exposed to a potentially life-threatening situation, a standby rescue person is required. The standby rescue person shall be

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equipped with respiratory protection equipment appropriate for the potential hazards and shall be immediately available to provide assistance.

12AA.5.4.10 Radioactive Material Control

Procedures are established, implemented and maintained that assure compliance with the requirements of 10 CFR 20.1801, 20.1802, 20.1902, 20.1904, 20.1905, 20.1906, 20.2001, 20.2005, 20.2006, 20.2007, 20.2201, and 10 CFR 71.5 to assure positive control over licensed radioactive material so that unnecessary or inadvertent exposures do not occur and such material is not released into uncontrolled areas in a manner that is not authorized by regulation or the license.

12AA.5.4.11 Radiation Protection Training

Procedures are developed, implemented, and maintained that assure that selection, qualification, training, and periodic retraining of radiation protection staff and radiation workers are conducted in accordance with the requirements in 10 CFR Parts 19, 20, and 10 CFR 50.120 and consistent with the guidance in Regulatory Guides 1.8, 8.13, 8.15, 8.27, and 8.29.

12AA.5.4.12 Quality Assurance

The radiation protection program and procedures are established, implemented, maintained and reviewed consistent with the 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description described in Section 17.5.

Consistent with the requirements in 10 CFR 71.101(f), quality assurance requirements apply to the program, procedures and activities involving the transportation of radioactive material.

12AA.5.4.13 Reports

Procedures are established, implemented, and maintained to assure that reports and notifications are made in accordance with 10 CFR 20, Subpart M.

12AA.5.4.14 Groundwater Monitoring Program

A groundwater monitoring program beyond the normal radioactive effluent monitoring program is developed. If necessary to support this groundwater monitoring program, design features will be installed during the plant construction process. Areas of the site to be specifically considered in this groundwater monitoring program are (all directions based on plant standard):

- West of the auxiliary building in the area of the fuel transfer canal.
- West and south of the radwaste building.
- East of the auxiliary building rail bay and the radwaste building truck doors.

This subsection adopts NEI 08-08A (Reference 201), for the Groundwater Monitoring Program description.

12AA.5.4.15 Record of Operational Events of Interest for Decommissioning

This subsection adopts NEI 08-08A (Reference 201), for discussion of record keeping practices important to decomissioning.

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12AA.5.5 References

1. 10 CFR Part 19, “Notices Instructions, and Reports to Workers: Inspections and Investigations.”

2. 10 CFR Part 20, “Standards for Protection Against Radiation.”

3. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

4. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

5. 10 CFR Part 71, “Packaging and Transportation of Radioactive Material.”

6. 10 CFR Part 73, “Physical Protection of Plants and Materials”

7. Regulatory Guide 1.8, Revision 3, “Qualification and Training of Personnel for Nuclear Power Plants.”

8. Regulatory Guide 1.97, Revision 3, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.”

9. Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

10. Regulatory Guide 8.2, “Guide for Administrative Practices in Radiation Monitoring.”

11. Regulatory Guide 8.4, “Direct-Reading and Indirect-Reading Pocket Dosimeters.”

12. Regulatory Guide 8.6, “Standard Test Procedures for G-M Counters.”

13. Regulatory Guide 8.7, Revision 2, “Instructions for Recording and Reporting Occupational Radiation Exposure Data.”

14. Regulatory Guide 8.8, Revision 3, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable.”

15. Regulatory Guide 8.9, Revision 1, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.”

16. Regulatory Guide 8.10, Revision 1R, “Operational Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable.”

17. Regulatory Guide 8.13, Revision 3, “Instruction Concerning Prenatal Radiation Exposure.”

18. Regulatory Guide 8.15, Revision 1, “Acceptable Programs for Respiratory Protection.”

19. Regulatory Guide 8.27, “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants.”

20. Regulatory Guide 8.28, “Audible Alarm Dosimeters.”

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21. Regulatory Guide 8.29, Revision 1, “Instruction Concerning Risks from Occupational Radiation Exposure.”

22. Regulatory Guide 8.34, “Monitoring Criteria and Methods To Calculate Occupational Radiation Doses.”

23. Regulatory Guide 8.35, “Planned Special Exposures.”

24. Regulatory Guide 8.36, “Radiation Doses to Embryo/Fetus.”

25. Regulatory Guide 8.38, Revision 1, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants.”

26. NUREG-0737, “Clarification of TMI Action Plan Requirements.”

27. NUREG-1736, “Consolidated Guidance: 10 CFR Part 20 Standards For Protection Against Radiation.”

28. Regulatory Issue Summary 2003-04, “Use of the Effective Dose Equivalent In Place of the Deep Dose Equivalent in Dose Assessments.”

29. SRP Branch Technical Position (BTP) 7-10, “Guidance on Application of Regulatory Guide 1.97,” NUREG-0800.

201. NEI 08-08A, Generic FSAR Template Guidance for Life Cycle Minimization of Contamination, Revision 0, October 2009 (ML093220445).

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Table 12AA-201 (Sheet 1 of 2)
Very High Radiation Areas (VHRA)

Room Number	VHRA Location	Figure 12.3-1, Sheet No.	Primary Source(s)	VHRA Conditional Notes	Frequency of Access to VHRA Areas While VHRA Conditions Exist
11105	Reactor Vessel Cavity	3, 4, 5	Neutron activation of the material in and around the cavity during reactor operations, such as the concrete shield walls and the reactor insulation	Note 1	None Required
12151	Spent Fuel Pool Cooling System / Liquid Radwaste System Demineralizer/ Filter room (Inside Wall)	3	Resin in vessels	Notes 6, 8	None Required
12153	Delay-Bed/ Guard-Bed Compartment	3	Activated carbon holding radioactive gases	Note 10	None Required
12371	Filter-Storage Area	6, 7	Spent filter cartridges	Notes 4, 6, 7	None required
12372	Resin Transfer Pump/Valve Room	6	Spent resin in lines	Note 6	None required
12373	Spent-Resin Tank Room	6	Spent resin in tanks	Note 6	None Required
12374	Waste Disposal Container Area	6	Spent resin in vault	Note 6	None Required
12463	Cask Loading Pit	6	Spent fuel	Notes 2, 6	None Required
12563	Spent Fuel Pool	5, 6	Spent fuel	Note 6	None Required
Fuel Transfer Areas					
12564	Fuel Transfer Tube	6	Fuel in transit	Notes 2, 5, 9	None Required
11205	Reactor Vessel Nozzle Area	5	Fuel in transit	Notes 2, 3, 9	None Required
11504	Refueling Cavity	6	Fuel in transit	Notes 2, 3, 9	None Required

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Table 12AA-201 (Sheet 2 of 2)
Very High Radiation Areas (VHRA)

Notes

1. VHRA during full power operation; less than 10 Rem/hr 24 hours after plant shutdown.
2. During underwater spent fuel transfer operations, this area can be as high as VHRA.
3. During underwater reactor internals transfers/ storage, this area can be as high as VHRA.
4. During spent resin waste disposal container transfer or loading, this area can be as high as VHRA. The contact dose rate of spent resin containers can be greater than 1000 Rem/hr.
5. Discussion about the Spent Fuel Transfer Canal and Tube Shielding is provided in Subsection 12.3.2.2.9.
6. Source is transient, removable, or can be relocated.
7. VHRA when hatch is removed during spent resin container handling operation.
8. In the event that the room does need to be accessed for maintenance or other reasons, temporary shielding is put in place and the resin is removed from the vessels. These measures reduce exposure rates in the room, such that this room is no longer a VHRA. Remote handling is used for any tasks that require the opening of the access hatch in the ceiling of this room when media is present.
9. These areas have no planned reasons for entry and are only classified as VHRAs during periods of fuel movement. In the event that these rooms do need to be accessed to repair the Fuel-Transfer System, Fuel Transfer Tube Gate Valve, or other components, it is done during a non-fuel movement time. This keeps the dose received by the worker as low as reasonably achievable.
10. Inspection of the equipment in this room, when required, is done using remote viewing equipment. Two plugs between Room 12153 and 12155 contain instruments and the plugs are expected to be removed every 12 to 18 months for performance of maintenance. Administrative procedures are implemented to protect workers pursuant to Regulatory Guide 8.38.

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Chapter 13 Conduct of Operation

This chapter provides information relating to the preparations and plans for operation of the AP1000 plant. Its purpose is to provide reasonable assurance that the plant will establish and maintain a staff of sufficient size and technical competence and that operating plans provide reasonable assurance of adequate protection of the public health and safety. The requirements for the Initial Test Program personnel and activities are covered under Chapter 14, Initial Test Program.

13.1 Organizational Structure

The organizational structure must be consistent with the human system interface design assumptions. See Section 1.8 and Chapter 18 for interface requirements pertaining to organizational structure.

13.1.1 Combined License Information Item

Combined License applicants referencing the AP1000 certified design will address adequacy of the organizational structure.

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13.2 Training

This section incorporates NEI 06-13A, Template for an Industry Training Program Description with the following exceptions:

- The annual General Employee Requalification training frequency defined in Subsection 1.3.3. The General Employee Requalification training frequency is defined in Subsection 13.2.3.3.

- The Engineering Personnel Initial Training defined in Subsection 1.2.9, Fundamentals, which is satisfied by prerequisite entry level education and experience.

Training programs incorporate instructional requirements to qualify personnel to operate and maintain the facility in a safe manner in all modes of operation (References 202). The programs are developed and maintained in compliance with the facility license and applicable regulations. The training programs are periodically evaluated and revised to reflect industry experience and to incorporate changes to the facility, procedures, regulations, and quality assurance requirements, and are periodically reviewed by management for effectiveness. These training programs are described in site and/or corporate procedures, as appropriate. Sufficient records are maintained and kept available for NRC inspection to verify adequacy of the programs.

The Training Department provides the required training based on individual employee experience, the intended position, and previous training and education. Training Department personnel may be supplemented by other personnel such as subject matter experts, contract staff, and vendor representatives. Formal instruction may be presented through a combination of classroom lectures, e-learning, assigned reading, simulator training and evaluations, and other delivery techniques.

For reactor operators, senior reactor operators, fuel handlers, fire protection personnel, and positions specified in 10 CFR 50.120 (References 13.2-4), programs are developed, established, implemented and maintained using a systems (or systematic) approach to training (SAT) as defined by 10 CFR 55.4 (References 13.2-8) and ANSI/ANS-3.1-1993 (References 13.2-14), as endorsed by Regulatory Guide 1.8 (References 13.2-16) and clarified in Appendix 1A.

Initial and continuing training programs accredited by the National Academy for Nuclear Training (NANT) provide personnel with the skills and knowledge to perform assigned tasks. Accredited training programs include the following:

- Non-licensed operator
- Reactor operator
- Senior reactor operator
- Shift manager
- Shift technical advisor
- Continuing training for licensed personnel
- Instrument and control technician
- Electrical maintenance personnel
- Mechanical maintenance personnel
- Chemistry technician
- Radiological protection technician
- Engineering personnel

The results of reviews of operating experience are incorporated into training and retraining programs in accordance with the provisions of TMI Action Item I.C.5, Appendix 1A. Training programs encompass all phases of plant operation including preoperational testing and low-power operation in accordance with the provisions of TMI Action Item I.G.1 (Reference 13.2-19). Supplemental operator

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training requirements during startup testing are described in Subsection 14.2.6.1. Before initial fuel loading, sufficient plant staff will be trained to provide for safe plant operations. Table 13.4-201 provides milestones for training implementation.

Operators involved in the Human Factors Engineering Verification and Validation (V&V) Program receive additional training specific to the task of performing V&V. A systematic approach to training is incorporated in developing this training program along with input from WCAP-14655, Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel (Reference 201).

Chapter 18, Section 18.10 references WCAP 14655. This document describes input from the designer on the training of the operations personnel who participate as subjects in the human factors engineering (HFE) verification and validation. The WCAP also describes how training insights are passed from the designer.

13.2.1 Licensed Operator Training

The Reactor Operator (RO) and Senior Reactor Operator (SRO) training programs, including initial and requalification training, provide the means to train individuals in the knowledge, skills, and abilities needed to perform licensed operator duties. The licensed operator training program includes the requalification program as required by 10 CFR 55.59 (Reference 13.2-13). Collectively, ROs and SROs are referred to as Licensed Operators. Before initial fuel loading, the number of persons trained in preparation for RO and SRO licensing examinations will be sufficient to meet regulatory requirements, with allowances for examination contingencies and without the need for planned overtime.

The site employs a simulator in accordance with 10 CFR 55.46 (Reference 13.2-12) and guidance of Regulatory Guide 1.149 (Reference 13.2-17). This simulator is used for training licensed personnel, and for the administration of the operating test.

13.2.1.1 Licensed Operator Initial Training Program

The Licensed Operator Initial Training Program prepares RO and SRO candidates for the NRC license exam. This program is implemented in accordance with administrative procedures.

13.2.1.1.1 Reactor Operator

Reactor Operator candidates receive training in the topics listed in 10 CFR 55.41 (Reference 13.2-9). RO candidates receive plant simulator training to demonstrate understanding and the ability to perform the actions listed in 10 CFR 55.45 (Reference 13.2-11).

13.2.1.1.2 Senior Reactor Operator

In addition to the Reactor Operator topics listed in 10 CFR 55.41 (Reference 13.2-9), candidates for the Senior Reactor Operator license receive training in the topics listed in 10 CFR 55.43 (Reference 13.2-10). SRO candidates receive plant simulator training to demonstrate understanding and the ability to perform the actions listed in 10 CFR 55.45 (Reference 13.2-11).

13.2.1.2 Continuing Training for Licensed Personnel

Continuing training for licensed personnel consists of regularly scheduled formal instruction, evaluation, and on-the-job training. Training material is developed using the SAT process, and includes Operational Experience (OE). Licensed operators participate in continuing training.

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Program content, course schedules and examination schedules comply with 10 CFR 55.59 (Reference 13.2-13). Continuing training for licensed personnel is conducted in accordance with administrative procedures.

13.2.2 Training for Positions Listed in 10 CFR 50.120[1]

This section addresses training programs for the positions listed in 10 CFR 50.120 (Reference 13.2-4). The systematic approach to training (SAT) process is used to establish and maintain training programs. Course duration and content are determined by the SAT process and by administrative procedure. This program will commence no later than eighteen months prior to initial fuel loading.

13.2.2.1 Non-Licensed Operator (NLO) Initial Training

Personnel employed as NLOs receive instruction on operation of plant equipment and components under normal and emergency conditions. This program is a combination of formal instruction and on-the-job training. Training is given in:

- Fundamentals of mechanical and electrical components
- Operation of equipment and systems
- Operating procedures
- Surveillance requirements
- Operation of systems important to plant safety

In-plant training includes system walk downs, which emphasize the use of procedures, the proper operation of equipment, and safe operating practices.

13.2.2.2 Shift Manager Initial Training

Shift managers have been trained as Senior Reactor Operators and receive additional training that addresses higher-level management skills and behaviors, and provides a broader perspective of plant operations. Initial training includes such topics as:

- Application of Operating Experience
- Problem-solving skills
- Planning and managing evolutions
- Maintaining a broad view of plant operations
- Application of observation skills
- Operating philosophy
- Shift team management
- Application of design bases to plant operations
- Emergency Plan
- Transient and Accident Analysis
- Systematic Approach to Training
- Work controls

13.2.2.3 Shift Technical Advisor Initial Training Program

Shift technical advisors provide engineering expertise on-shift. Training provides them with the skill and knowledge to monitor equipment and system operation, and assess plant conditions during abnormal and emergency events. Initial training for individuals who fill the position of shift technical advisor includes instruction in the following areas:

1. 10 CFR 52.78 (Reference 13.2-6) requires that Combined License applicants demonstrate compliance with 10 CFR 50.120.

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- Responses to accidents and analyses of plant transients
- Application of engineering principles to protection of the core
- Mitigation of plant accidents
- Basis of plant and systems design
- Reactor theory, thermodynamics, heat transfer, and fluid flow
- General Operating Procedures, Technical Specifications, and Administrative Controls
- Operational transient and accident analysis
- Simulator training, including exercises in the following situations:
  - Plant or reactor startups to include a range such that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established
  - Plant shutdown
  - Manual control of feedwater during startup or shutdown.
  - Significant (10 percent) power changes due to manual changes in control rod position.
- Accident response training

13.2.2.4 Instrumentation and Control (I&C) Technician Initial Training

Initial training for I&C technicians includes instruction in the following areas:

- Fundamentals of instrumentation and control
- Pneumatic systems and equipment
- Electronics
- Fundamental systems training
- I&C and other job related procedures
- Surveillance requirements
- Mitigating core damage training commensurate with their responsibilities during accidents that involve severe core damage
- On-the-job training

On-the-job training allows I&C technicians to practice the skills learned in the classroom under the guidance of experienced and qualified I&C personnel.

13.2.2.5 Electrical Maintenance Initial Training Program

Initial training for electrical maintenance technicians includes instruction in the following areas:

- Print reading
- Use of electrical tools and test equipment
- Fundamental systems training
- Electrical components and equipment
- Electrical maintenance practices
- Maintenance procedures
- On-the-job training

On-the-job training allows Electricians to practice the skills learned in the classroom under the guidance of experienced and qualified electrical maintenance personnel.

13.2.2.6 Mechanical Maintenance Initial Training Program

Initial training for mechanical maintenance technicians includes instruction in the following areas:

- Print reading
- Use of hand tools, power tools, and measurement devices
- Fundamental systems training

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- Mechanical components and equipment
- Mechanical maintenance practices
- Maintenance procedures
- On-the-job training

On-the-job training allows Mechanics to practice the skills learned in the classroom under the guidance of experienced and qualified mechanical maintenance personnel.

13.2.2.7 Radiological Protection Technician Initial Training

Initial training for radiological protection technicians includes instruction in the following areas:

- Principles of radiation
- Radiation protection and safety
- Use of survey instruments
- Use of analytical equipment
- Radiation Protection procedures
- Emergency Plan procedures
- ALARA practices and procedures
- Fundamental systems training
- Mitigating core damage training commensurate with their responsibilities during accidents that involve severe core damage

On-the-job training provides the trainee opportunities to practice actual operation of radiation protection equipment and use of procedures under the guidance of experienced technicians. Further information on training for radiological protection technicians can be found in Appendix 12AA.5.

13.2.2.8 Chemistry Technician Initial Training

Initial training for chemistry technicians includes instruction in the following areas:

- Chemistry procedures
- Laboratory practices
- Conduct of analytical tests
- Operation of laboratory equipment
- Fundamental systems training
- On-the-job training to include actual operation of analytical equipment and the use of procedures
- Mitigating core damage training commensurate with their responsibilities during accidents that involve severe core damage
- Power plant chemistry

On-the-job training provides the trainee opportunities to practice actual operation of analytical equipment and use of procedures under the guidance of experienced technicians.

13.2.2.9 Engineering Personnel Initial Training

Engineering personnel complete orientation training on topics such as those listed below. The topics are chosen to familiarize engineering support personnel with various aspects of nuclear technology in an operating plant environment. Training topics include:

- Records management and document control
- Applicable industrial and nuclear regulations, codes, and standards
- Procedures and drawings

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- Applicable programs such as corrective action, configuration management, work control, and the QA program
- Technical Specifications
- Plant systems, instrumentation, and components
- Plant operations
- Introductory review of accidents
- Design processes

13.2.2.10 Continuing Training for Personnel Listed in 10 CFR 50.120

Non-licensed plant personnel specified in Subsection 13.2.2 [i.e., personnel listed in 10 CFR 50.120 (Reference 13.2-4)] receive continuing training to maintain qualifications and enhance proficiency. Continuing training reinforces initial training by reiterating selected portions of the material. Continuing training also addresses new and modified procedures and plant design changes.

Operating Experience (OE) is included in continuing training, providing personnel with actual examples of good practices and lessons learned. OE topics are selected from Licensee Event Reports, corrective action databases, industry groups, and other sources.

Continuing training material is developed in accordance with the systematic approach to training and is conducted in accordance with administrative procedures.

STA qualifications are maintained by participation in continuing training for licensed personnel.

13.2.3 General Employee Training (GET) Program

13.2.3.1 Plant Access Training

As part of the GET program, members of the station staff, contractor workers, and unescorted visitors participate in Plant Access Training, which consists of the following topics, prior to being granted unescorted access to the plant:

- Station organization
- Station facilities and layout
- Station administration
- Nuclear plant overview
- Industrial safety
- Fire protection
- Quality assurance and quality control
- Plant security
- Emergency planning
- Radiological orientation
- Appropriate portions of 10 CFR 26 (Reference 13.2-2)
- Appropriate portions of 10 CFR 19 (Reference 13.2-1)

13.2.3.2 Radiation Worker Training Program

Personnel whose job duties require them to have unescorted access to radiologically controlled areas of the plant receive instruction in the applicable aspects of radiation protection. Topics include the following:

- Sources of radiation
- Types and measurement of radiation
- Biological effects

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- Limits and guidelines, including Reg. Guide 8.13 (Reference 13.2-18)
- Concept of As Low As Reasonably Achievable (ALARA)
- Radiation dosimetry
- Contamination
- Internal exposure
- Radiation work permits
- Radiological postings
- Radiological alarms
- Radioactive waste
- Rights and responsibilities
- Protective clothing

13.2.3.3 General Employee Requalification Training

Personnel with unescorted access to the plant participate in requalification training at a frequency of every 48 months nominal, not to exceed 51 months. Requalification training includes those topics in 13.2.3.1 and 13.2.3.2, as applicable to access requirements. Emphasis is placed on significant changes to the plant, plant procedures, government regulations regarding the operation of the plant, and quality assurance requirements. As applicable, training is conducted on industry operating experiences, Licensee Event Reports, and personnel errors.

13.2.4 Selected Other Training Programs

This subsection addresses training for positions not specified by 10 CFR 55 (Reference 13.2-7) or 10 CFR 50.120 (Reference 13.2-4).

13.2.4.1 Fire Protection Training

Initial fire protection training is completed prior to receipt of fuel at the site. Personnel assigned as fire brigade members receive formal training prior to assuming brigade duties, and regularly scheduled retraining. Fire brigade training complies with NFPA Standard 600 (Reference 13.2-15).

Training appropriate to the assigned work is also provided for the fire protection staff, fire watch personnel, and the general employee. Subsection 9.5.1 includes additional information regarding fire protection training.

13.2.4.2 Emergency Plan Training Program

Emergency Plan training meets the requirements of 10 CFR 50 Appendix E Section IV.F (Reference 13.2-5) and the standards of 10 CFR 50.47(b)(15) (Reference 13.2-3). Further details of the Emergency Plan training program can be found in the Emergency Plan, which is a separate document.

13.2.4.3 Physical Security Training Program

Training of security personnel is discussed in Section 13.6 and in the Physical Security Plan, which is a separate document.

13.2.4.4 Station Management Training Program

Station supervisors receive Fitness for Duty (FFD) supervisory training in accordance with 10 CFR 26.29. The remaining definitions and recommendations in this subsection are taken from ANSI/ANS-3.1-1993 (Reference 13.2-14) as endorsed by Regulatory Guide 1.8 (Reference 13.2-16).

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The qualification requirements for managers and middle managers include training or experience in supervision or management. Training for supervisors develops their skills in the following areas:

- Leadership
- Interpersonal communications
- Management responsibilities and limits
- Motivation of personnel
- Problem analysis and decision making
- Administrative policies and procedures
- Observation skills
- Coaching

13.2.5 Training Effectiveness Evaluation Program

The program to evaluate the effectiveness of training programs is based on three independent inputs or perspectives: the supervisor of the trainee, the trainee, and an educational content evaluation. Each of these reviews is discussed below.

13.2.5.1 Supervisory Review for Training Effectiveness

The purpose of this review is to monitor the content and effectiveness of training programs as related to the duties and job responsibilities of the trainees. Reviews may be performed by supervisors of employees meeting with appropriate Training personnel, by designated oversight personnel, or by observing subsequent job performance. Observations are discussed to determine topics that may require additional training or subjects that may be removed from the training program.

13.2.5.2 Trainee Review of Training Effectiveness

Following selected courses, or training cycles, trainees have the opportunity to provide comments regarding the effectiveness of the instructional methods and content relevancy to their jobs. These comments are used in the evaluation of both instruction and content of the training program.

13.2.5.3 Review for Effectiveness of Instructional Techniques and Materials

Training material and instructional aids are assessed for clarity and applicability. Observations of instructors in the teaching environment are conducted by this qualified individual to monitor classroom performance. Full time instructors receive basic indoctrination in instructional techniques as soon as practicable after assuming instructional duties. The educational specialist conducts periodic seminars in instructional techniques, discussing areas where group performance could be improved and recommends innovative techniques observed at this or other power stations.

13.2.6 Combined License Information Item

The training programs for plant personnel, including the training program for the operations personnel who participate as subjects in the human factors engineering verification and validation and the scope of licensing examinations, as well as new training requirements, are addressed in Section 13.2.

13.2.7 References

13.2-1. 10 CFR 19, "Notices, Instructions, and Reports to Workers; Inspection and Investigations"

13.2-2. 10 CFR 26, "Fitness for Duty"

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13.2-3. 10 CFR 50.47, "Emergency Plans"

13.2-4. 10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel"

13.2-5. 10 CFR 50 Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities"

13.2-6. 10 CFR 52.78, "Contents of Applications; Training and Qualification of Nuclear Power Plant Personnel"

13.2-7. 10 CFR 55, "Operator's Licenses"

13.2-8. 10 CFR 55.4, "Definitions"

13.2-9. 10 CFR 55.41, "Written Examinations: Operators"

13.2-10. 10 CFR 55.43, "Written Examinations, Senior Operators"

13.2-11. 10 CFR 55.45, "Operating Tests"

13.2-12. 10 CFR 55.46(c), "Plant-Referenced Simulators"

13.2-13. 10 CFR 55.59, "Requalification"

13.2-14. American National Standards Institute, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," ANSI/ANS-3.1-1993

13.2-15. National Fire Protection Association, "Standard on Industrial Fire Brigade," NFPA Standard 600, 2005 Edition

13.2-16. U.S. Nuclear Regulatory Commission, "Qualification and Training of Personnel for Nuclear Power Plants," Regulatory Guide 1.8, Revision 3, May 2000

13.2-17. U.S. Nuclear Regulatory Commission, "Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements," Regulatory Guide 1.149, Revision 4, April 2011

13.2-18. U.S. Nuclear Regulatory Commission, "Instruction Concerning Prenatal Radiation Exposure," Regulatory Guide 8.13, November, 1980.

13.2-19. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements", NUREG-0737, Revision 3, June 1999

13.2-20. Not used.

201. Westinghouse, "Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel," WCAP-14655, Revision 1, August 1996.

202. NEI 06-13A, Template for an Industry Training Program Description, Nuclear Energy Institute, Revision 2, March 2009.

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13.2A — Cold License Training Plan

LICENSED OPERATOR TRAINING PROGRAM PRIOR TO COMPLETION OF THE FIRST REFUELING OUTAGE

Prior to operation, plant experience requirements specified in Regulatory Guide 1.8 (Revision 3) and ANSI/ANS 3.1-1993 cannot be met. Additionally, other standard guidance for operator selection, training, and qualification cannot be met.

Cold licensing of operators provides the method for operations personnel to acquire the knowledge and experience required for licensed operator duties during the unique conditions of new plant construction and initial operation.

Persons eligible for the cold license process shall meet the following requirements:

- Candidates for a Reactor Operator license shall have a High School Diploma or equivalent as required by R.G. 1.8 Revision 3.
- Candidates for a Senior Reactor Operator license shall have at least one of the following qualifications:
  - Previously held a Senior Reactor Operator license for an operating nuclear power plant.
  - Previously held a Reactor Operator license for an operating nuclear power plant.
  - Bachelor's Degree in engineering or science as defined by R.G. 1.8 Revision 3.
  - Experience as a licensed operator training instructor with an SRO certification. This experience will be evaluated and approved on a case by case basis by the NRC.
  - Two years military experience in a position equivalent to a reactor operator.

The provisions in this section are applicable to each unit of a multiple unit site separately.

The cold licensing process for the selection, training and licensing of Operations personnel for the new nuclear plants adheres to current industry guidance for operating plants with exemptions and alternatives in the following areas.

13.2A.1 Licensed Operator Experience Requirements Prior To Commercial Operation

Licensed operator candidates need not satisfy the experience requirements prior to entering a licensed operator training program. Experience and plant evolution requirements that have not been met at the time the licensed operator examination is administered shall be met prior to issuing the individual's NRC operator license. In such a case, the Licensee will notify the NRC when the candidate meets the experience and plant evolution requirements.

The methods listed below provide the licensed operator candidate with meaningful experience on the reactor for which the license is sought. Methods for gaining meaningful experience include completing systematically designed training courses, and participating in practical work assignments such as preoperational testing, procedure development and validation, human factors engineering activities, task analysis verification, or conducting licensed operator classroom or simulator training. Additionally, for these activities to be considered meaningful, they must be associated with safety significant, defense-in-depth, or other major plant components or systems. All cold licensed operator candidates will:

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- Complete a systematically designed site layout course.
- Complete a site-specific non-licensed operator on-the-job training program on selected non-licensed operator tasks. The selected non-licensed operator tasks are those tasks that are important to plant operation with regard to nuclear safety, defense-in-depth, or that are risk significant.
- Participate in practical work assignments for a minimum of six months that includes preoperational testing, and one or more of the following:
  - Procedure development and validation
  - Human factors engineering activities
  - Task analysis verification
  - Licensed operator classroom presentations or simulator training implementation

Senior reactor operator cold license candidates will complete a site-specific reactor operator and senior reactor operator training course.

Senior reactor operator cold license candidates without "hot" plant experience will complete a plant operational excellence course that is conducted in a plant simulator or they will observe control room activities at an operating nuclear plant for at least six weeks. The course and the observation activity are designed to familiarize the candidate with the operational interfaces encountered by decision makers in a nuclear plant control room.

Hot plant experience is defined as performance of senior reactor operator duties for at least six months including:

- At least 6 weeks of operation above 20 percent power
- A startup from subcritical to 20 percent power
- A shutdown from above 20 percent power to cold (less than 212°F) and subcritical
- Startup preparations following a fueling or refueling outage

The startup, shutdown, and startup preparations may have been performed at an operating plant or a plant simulator.

Table 13.2A-201, Comparison of Hot and Cold License Guidance, shows the current experience requirement and the associated cold license experience method. Table 13.2A-202, Illustration of Cold Licensing Plan by Candidate Type, shows education and experience methods for each licensed operator candidate type.

13.2A.2 Crew Experience Requirements during First Year of Operation

Each operating crew's cumulative nuclear power plant experience shall be > 6 years; and the crew's cumulative power plant experience shall be > 13 years.

The crew's cumulative nuclear power plant experience is gained by working at nuclear power plants and military nuclear propulsion plants, conducting licensed operator training, participating in new nuclear plant construction and testing, and completing academic degree requirements. The

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cumulative crew nuclear power plant experience is the sum of each individual's experience after applying weighting factors and maximum credit limits in Table 13.2A-201, Cumulative Nuclear Power Plant Operating Crew Experience Equivalencies.

When determining cumulative nuclear power plant experience, all 6 years shall not be attributed from one crew member.

The crew's cumulative power plant experience is the sum of each individual's power plant experience. Power plant experience, for example, is experience gained by working at nuclear power plants, conventional power plants, and military propulsion plants. Cumulative power plant experience does not involve weighting factors or maximum credit limits.

In addition to the experience requirement mentioned above, each operating crew shall be staffed with a senior reactor operator with hot plant experience (previously defined in 1.1, Licensed Operator Experience Requirements Prior to Commercial Operation). If a senior reactor operator with hot plant experience is not available, then a shift advisor may be substituted. The shift advisor will have at least one year of on-shift licensed senior reactor operator experience at a similar type (PWR/BWR) operating plant, and will have completed a training program on the design for which they are advising. While observing crew performance, the shift advisor will make recommendations to the shift manager only, and will not interfere with the licensed responsibilities of the operating crew. The shift advisor will have direct access to plant senior management to resolve issues. Shift advisor duties include, but are not limited to the following:

- Monitor procedure adherence
- Observe the conduct of prejob briefs shift turnover, plant evolutions, non- licensed operator rounds, plant tours, and post job debriefs
- Monitor overall station risk

Weighting factors and maximum credit limits for determining cumulative nuclear power plant operating crew experience are shown in Table 13.2A-203, Cumulative Nuclear Power Plant Operating Crew Experience Equivalencies.

13.2A.3 Conduct of On-the-Job Training (OJT)

Until plant construction is completed, acceptable methods for the conduct of on-the-job training include discussion, simulation, and use of mockup equipment and virtual reality technology.

13.2A.4 Use of Part-Task/Limited Scope Simulators

Part-task or limited scope simulators may be used during licensed operator training.

13.2A.5 Licensed Operator Continuing Training

Licensed operator continuing training begins within 90 days following the issuance of the first operator license. Continuing training content is systematically determined to maintain operator knowledge of plant operation.

13.2A.6 Cold Licensing Process Applicability and Termination

The cold licensing process described in this document may be applied to each unit of a multi-unit site.

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Cold license guidance items 1 through 9 on Table 13.2A-201 will apply to any licensed operator training class started prior to initial fuel load.

Cold license guidance items 3 through 9 on Table 13.2A-201 will apply to any licensed operator training class started after initial fuel load and before completion of the first refueling outage. Items 1 and 2 cold license guidance are no longer allowable after initial fuel load.

The cold licensing process will terminate after completion of the first refueling outage.

As plant systems, components, and structures are completed, and as integrated plant operations begin, the systematic approach to training process will be used to adjust cold license class training methods and settings used to implement the guidance in Table 13.2A-201 items 1 through 9. The purpose is to optimize student learning using actual in-plant training and experience opportunities as they become available.

13.2A.7 Initial Licensed Operator Examination Schedule

Administration of licensed operator examinations begins approximately 18 months prior to fuel load.

13.2A.8 References

13.2-201. Nuclear Energy Institute (NEI), "Technical Report on a Template for an Industry Training Program Description," NEI 06-13A

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Table 13.2A-201
Comparison of Hot and Cold License Guidance

Current Hot License Guidance	Applicable Position	References	Cold License Guidance
1. Six months on-site at reactor for which license is sought.	All	ANSI 3.1-1993; 4.4.1 4.4.2 4.5.1. Regulatory Guide 1.8 Rev 3: 2.8 2.10. NUREG 1021 Rev 9 ES-202	Six months practical work assignments and Complete a site layout course
2. One year on-site at the reactor for which the license is sought with six months as a nonlicensed operator.	Reactor operator	ANSI 3.1-1993: 4.5.1. Regulatory Guide 1.8 Rev 3: 2.10. NUREG 1021 Rev 9 ES-202	Six months practical work assignments and Complete a site layout course and Complete a site-specific non-licensed operator training program for selected nonlicensed operator tasks

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Table 13.2A-201
Comparison of Hot and Cold License Guidance

Current Hot License Guidance	Applicable Position	References	Cold License Guidance
3. Pre-requisite experience requirements must be met to enter training program.	All	NUREG 1021 Rev 9 ES-202 Section D.	Applicable experience requirements shall be met prior to NRC license issuance.
4. Three years power plant experience at least one of which should have been at the plant for which the license is sought.	Reactor operator	ANSI 3.1-1993: 4.5.1 Regulatory Guide 1.8 Rev 3: 2.10 NUREG 1021 Rev 9 ES-202	Six months practical work assignments and Cumulative operating crew experience requirements apply
5. Reactor operator license actively involved in the performance of licensed duties for at least one year.	Senior reactor operator (Non-degreed)	Regulatory Guide 1.8 Rev 3: 2.8 NUREG 1021 Rev 9 ES-202	Complete a site layout course and Complete a site-specific non-licensed operator training program for selected nonlicensed operator tasks and Complete a reactor operator and senior reactor operator training course

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Table 13.2A-201
Comparison of Hot and Cold License Guidance

Current Hot License Guidance	Applicable Position	References	Cold License Guidance
6. At least three years of responsible nuclear power plant experience.	Senior reactor operator (Degreed)	Regulatory Guide 1.8 Rev 3: 1.3 2.8	Complete a site layout course and Complete a site-specific non-licensed operator training program for selected nonlicensed operator tasks and Complete a reactor operator and senior reactor operator training course
7. At least six weeks of operation above 20% power, and startup from subcritical to 20% power, and shutdown from above 20% power to cold (less than 212°F) and subcritical, and startup preparations following a fueling or refueling outage.	Shift Supervisor (Shift Manager)	ANSI 3.1-1993: 4.4.1	Cumulative Operating Crew Experience requirements apply and Complete a Plant Operational Excellence Course or plant observation activity

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Table 13.2A-201
Comparison of Hot and Cold License Guidance

Current Hot License Guidance	Applicable Position	References	Cold License Guidance
8. At least six weeks of operation above 20% power.	Senior reactor operator	ANSI 3.1-1993: 4.4.2	Cumulative Operating Crew Experience requirements apply and Complete a Plant Operational Excellence Course or plant observation activity
9. Three years power plant experience and three years nuclear power plant experience	Senior reactor operator	ANSI 3.1-1993 4.4.1 4.4.2 Regulatory Guide 1.8 Rev 3: 2.8 NUREG 1021 Rev 9 ES-202	Six months practical work assignments and Cumulative Operating Crew Experience requirements apply

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Table 13.2A-202
Illustration of Cold Licensing Plan by Candidate Type

License Candidate	Education	Site Layout Course	NLO Task Training	RO Training	SRO Training	Plant Operational Excellence Course or Observation Activity	Six Months Practical Work Assignments (1)
Reactor operator	High school diploma	Yes	Yes	Yes	N/A	N/A	Yes
Senior reactor operator – degreed manager or degreed nonlicensed operator or technical staff	Bachelor of Science or equivalent in engineering, engineering technology, or physical science	Yes	Yes	Yes	Yes	Yes	Yes
Senior reactor operator – previous license or military equivalent	High school diploma	Yes	Yes	Yes	Yes	Yes (2)	Yes
Senior reactor operator – certified instructor	High school diploma	Yes	Yes	Yes	Yes	Yes	Yes

(1): practical work assignments includes activities such as participating in preoperational testing, procedure development and validation, human factors engineering activities, and task analysis verification, or conducting licensed operator classroom or simulator training

(2): No, if candidate has hot license experience

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Table 13.2A-203
Cumulative Nuclear Power Plant Operating Crew Experience Equivalencies (1)

Type of Experience	Weighting Factor	Max Credit	Justification
1. Commercial Nuclear Plant RO/SRO on same type plant (PWR/BWR)	1.00	No Limit	Task Analysis for same type plant are essentially the same
2. Commercial Nuclear Plant RO/SRO from different type plant (PWR/BWR)	0.75	No Limit	Task Analysis demonstrates that 75% of PWR/BWR tasks are similar
3. Military Nuclear Propulsion Plant Experience (Propulsion Plant Watch Officer, Engineering Watch Supervisor, Reactor Operator, Engineering Officer of the Watch, Propulsion Plant Watch Supervisor)	0.5	36 months	For these military nuclear propulsion plant watch qualifications, approximately 50% of the job tasks are similar
4. Military Nuclear Propulsion Plant Experience (Other than watch qualifications in 3 above such as Machinist Mate, Electricians Mate, Engineering Laboratory Technician, or Electronics Technician)	0.25	36 months	For these (other) watch qualifications, a range of similarities between job tasks (25-75%) exists, so a conservative value of 25% is credited

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Table 13.2A-203
Cumulative Nuclear Power Plant Operating Crew Experience Equivalencies (1)

Type of Experience	Weighting Factor	Max Credit	Justification
5. Reference Plant Simulator	5.00	12 months	Industry analysis demonstrated that activities completed in a simulator, compare to an operating Control Room, occur in a ratio of approx. 400/1
6. Limited Scope Simulator	3.00	9 months	Similar to Reference Plant
7. Actual nuclear plant experience during construction	0.25	12 months	Approximately 25% of the tasks during construction testing in preparation for system turnover to operations is similar to an operating facility
8. Actual nuclear plant experience during pre-operational testing	0.75	12 months	75% of tasks during pre- operational testing are similar to an operating facility
9. Actual nuclear plant experience during fuel load and startup testing	1.00	12 months	Tasks during initial startup are similar to operating facility

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Table 13.2A-203
Cumulative Nuclear Power Plant Operating Crew Experience Equivalencies (1)

Type of Experience	Weighting Factor	Max Credit	Justification
10. License Classroom training	0.25	9 months	Theory of ops and specific plant design knowledge is critical to an operator's success
11. Participation in operator duties at another commercial nuclear facility. This includes nonlicensed operator duties	0.75	12 months	Task similarities
12. Other Nuclear Plant experience	0.25	12 months	Procedure writing, facility operation (water plant and other support facilities, etc)
13. Licensed operator instructor	0.50	12 months	Instructors will have participated in a train-the- trainer program that includes simulator, classroom (systems, theory).
14. Bachelors Degree in an Engineering, Science or Technical field	n/a	24 months	College work (in these fields) gives student an understanding of the fundamentals of plant operations

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Table 13.2A-203
Cumulative Nuclear Power Plant Operating Crew Experience Equivalencies (1)

Type of Experience	Weighting Factor	Max Credit	Justification
15. Associates Degree (technical)	n/a	6 months	Student gains knowledge of fundamentals

(1): Weighting factors and max credit values based on those in “Industry Evaluation of Operating Shift Experience Requirements” By: J.H. Miller Jr. 2/24/1984, and endorsed by Generic Letter number 84-16, Adequacy of On-Shift Operating Experience For Near Term Operating License Applicants, except for shaded rows which are added experience types based on new technology or additional analysis.

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13.3 Emergency Planning

See the Emergency Plan for the locations of the Technical Support Center and the Operations Support Center. See Subsection 1.2.5 for the location of the decontamination facilities. See Section 9.4 for a description of the HVAC systems for the main control room/control support area and the annex building. See Section 18.8 for the high level requirements for the technical support center and the operations support center. See Section 7.5 for identification of plant variables that are provided for interface to the emergency planning areas.

Communication interfaces among the main control room, the technical support center and the emergency planning centers are discussed below and in the Emergency Plan.

Staffing of the emergency operations facility occurs consistent with current operating practice and with revision 1 of NUREG-0654/FEMA-REP-1.

The emergency planning information is submitted to the Nuclear Regulatory Commission as a separate licensing document.

Post-72 hour support actions, as discussed in Subsections 1.9.5.4 and 6.3.4, are addressed in Subsections 6.2.2, 8.3, and 9.1.3. Provisions for establishing post-72 hour ventilation for the main control room, instrumentation and control rooms, and dc equipment rooms are established in operating procedures.

The emergency plan describes the plans for coping with emergency situations, including communications interfaces and staffing of the emergency operations facility.

Table 13.4-201 provides milestones for emergency planning implementation.

13.3.1 Combined License Information Item

Emergency planning including post-72 hour actions and its communication interface are addressed in Section 13.3.

Activation of the emergency operations facility consistent with current operating practice and NUREG-0654/FEMA-REP-1 is addressed in Section 13.3 and in the Emergency Plan.

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13.4 Operational Programs

Operational programs are specific programs that are required by regulations. Table 13.4-201 lists each operational program, the regulatory source for the program, the section of the FSAR in which the operational program is described, and the associated implementation milestone(s).

13.4.1 Combined License Information Item

Combined License applicants referencing the AP1000 certified design will address each operational program.

13.4.2 References

201. ASME Boiler and Pressure Vessel Code (B&PVC), “Section XI - Rules for Inservice Inspection of Nuclear Power Plant Components.”

202. ASME “OM Code for the Operation and Maintenance of Nuclear Power Plants.”

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Table 13.4-201 (Sheet 1 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
1.	Inservice Inspection Program	10 CFR 50.55a(g)	5.2.4, 5.4.2.5, 6.6	To be addressed by the COL applicant	10 CFR 50.55a(g), ASME XI IWA-2430(b) (Reference 201)
2.	Inservice Testing Program	10 CFR 50.55a(f); 10 CFR Part 50, Appendix A	3.9.6, 5.2.4	To be addressed by the COL applicant	10 CFR 50.55a(f), ASME OM Code (Reference 202)
3.	Environmental Qualification Program	10 CFR 50.49(a)	3.11	To be addressed by the COL applicant	License Condition
4.	Preservice Inspection Program	10 CFR 50.55a(g)	5.2.4, 5.4.2.5, 6.6	To be addressed by the COL applicant	10 CFR 50.55a(g); ASME XI IWB-2200(a) (Reference 201)
5.	Reactor Vessel Material Surveillance Program	10 CFR 50.60; 10 CFR 50.61; 10 CFR Part 50, Appendix H	5.3.2.6	To be addressed by the COL applicant	License Condition
6.	Preservice Testing Program	10 CFR 50.55a(f)	3.9.6	To be addressed by the COL applicant	License Condition

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Table 13.4-201 (Sheet 2 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
7.	Containment Leakage Rate Testing Program	10 CFR 50.54(o); 10 CFR 50, Appendix A (GDC 52); 10 CFR 50, Appendix J	6.2.5.1	To be addressed by the COL applicant	License Condition
8.	Fire Protection Program	10 CFR 50.48	9.5.1.8	To be addressed by the COL applicant	License Condition
	(portions applicable to radioactive material)	10 CFR 30.32 10 CFR 40.31 10 CFR 70.22		To be addressed by the COL applicant	10 CFR 30.32(a) 10 CFR 40.31(a) 10 CFR 70.22(a)
9.	Process and Effluent Monitoring and Sampling Program:				
	Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls	10 CFR 20.1301 and 20.1302; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, Appendix I, Section II and IV	11.5	To be addressed by the COL applicant	License Condition
	Offsite Dose Calculation Manual	Same as above	11.5	To be addressed by the COL applicant	License Condition

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Table 13.4-201 (Sheet 3 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
	Radiological Environmental Monitoring Program	Same as above	11.5	To be addressed by the COL applicant	License Condition
	Process Control Program	Same as above	11.4	To be addressed by the COL applicant	License Condition
10.	Radiation Protection Program (including ALARA principle) • Radioactive Source Control (assignment of RP Supervisor) • Assignment of RP Supervisor • Minimization of Contamination • Personnel Dosimetry • Radiation Monitoring and Surveys • Radiation Work Permits • Assignment of RP Manager • Respiratory Protection • Bioassay	10 CFR 20.1101 10 CFR 20.1406 10 CFR Part 37	12.1 12.5 11.4	To be addressed by the COL applicant	License Condition

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Table 13.4-201 (Sheet 4 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
	• Effluents and Environmental Monitoring and Assessment • Job Coverage • Radioactive Waste Shipping				
11.	Non Licensed Plant Staff Training Program	10 CFR 50.120	13.2	To be addressed by the COL applicant	10 CFR 50.120(b)
	(portions applicable to radioactive material)	10 CFR 30.32 10 CFR 40.31 10 CFR 70.22		To be addressed by the COL applicant	10 CFR 30.32(a) 10 CFR 40.31(a) 10 CFR 70.22(a)
12.	Reactor Operator Training Program	10 CFR 55.13; 10 CFR 55.31; 10 CFR 55.41; 10 CFR 55.43; 10 CFR 55.45	13.2	To be addressed by the COL applicant	License Condition
13.	Reactor Operator Requalification Program	10 CFR 50.34(b); 10 CFR 50.54(i); 10 CFR 55.59	13.2	To be addressed by the COL applicant	10 CFR 50.54 (i-1)

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Table 13.4-201 (Sheet 5 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
14.	Emergency Planning	10 CFR 50.47; 10 CFR 50, Appendix E	13.3	To be addressed by the COL applicant.	10 CFR Part 50, Appendix E, Section IV.F.2.a(ii)
				To be addressed by the COL applicant	10 CFR Part 50, Appendix E, Section IV.F.2.a(ii)
				To be addressed by the COL applicant	10 CFR Part 50, Appendix E, Section V
15.	Security Program:				
	Physical Protection Program (applicable to protection of special nuclear material prior to the protected area being declared operational)	10 CFR 73.1 10 CFR 73.67	13.5.2.2.8 13.6	To be addressed by the COL applicant	10 CFR 73.1(a) 10 CFR 73.67
	Physical Security Program	10 CFR 73.55(b); 10 CFR 73.55(c)(3); 10 CFR 73.56; 10 CFR 73.57;	13.6	To be addressed by the COL applicant	10 CFR 73.55(a)(4)

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Table 13.4-201 (Sheet 6 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
	Safeguards Contingency Program	10 CFR 73.55(c)(5); 10 CFR 73.55(k); 10 CFR Part 73, Appendix C	13.6	To be addressed by the COL applicant	10 CFR 73.55(a)(4)
	Training and Qualification Program	10 CFR 73.55(c)(4); 10 CFR 73.55(d)(3); 10 CFR Part 73, Appendix B	13.6	To be addressed by the COL applicant	10 CFR 73.55(a)(4)
16.	Quality Assurance Program – Operation	10 CFR 50.54(a); 10 CFR Part 50, Appendix A (GDC 1); 10 CFR Part 50, Appendix B	17.5	To be addressed by the COL applicant	10 CFR 50.54(a)(1)
17.	Maintenance Rule	10 CFR 50.65	17.6	To be addressed by the COL applicant	10 CFR 50.65(a)(1)
18.	Motor-Operated Valve Testing	10 CFR 50.55a(b)(3)(ii)	3.9.6	To be addressed by the COL applicant	License Condition

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Table 13.4-201 (Sheet 7 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
19.	Initial Test Program	10 CFR 50.34; 10 CFR 52.79(a)(28)	14.2	To be addressed by the COL applicant	License Condition
20.	Fitness for Duty (FFD) Program for Construction (workers and first-line supervisors)	10 CFR 26.4(f)	13.7	To be addressed by the COL applicant	10 CFR Part 26, Subpart K
	FFD Program for Construction (management and oversight personnel)	10 CFR 26.4(e)	13.7	To be addressed by the COL applicant	10 CFR Part 26, Subparts A – H, N, and O

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Table 13.4-201 (Sheet 8 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
	FFD Program for Security Personnel	10 CFR 26.4(e)(1)	13.7	To be addressed by the COL applicant	10 CFR Part 26, Subparts A – H, N, and O
		10 CFR 26.4(a)(5) or 26.4(e)(1)			10 CFR Part 26, Subparts A – I, N, and O
	FFD Program for FFD Program personnel	10 CFR 26.4(g)	13.7	To be addressed by the COL applicant	10 CFR Part 26, Subparts A, B, D – H, N, O, and C per licensee's discretion
	FFD Program for persons required to physically report to the Technical Support Center (TSC) or Emergency Operations Facility (EOF)	10 CFR 26.4(c)	13.7	To be addressed by the COL applicant	10 CFR Part 26, Subparts A – I, N, and O, except for §§ 26.205 – 209

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Table 13.4-201 (Sheet 9 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation	
				Milestone	Requirement
	FFD Program for Operation	10 CFR 26.4(a) and (b)	13.7	To be addressed by the COL applicant	10 CFR Part 26, Subparts A – I, N, and O, except for individuals listed in § 26.4(b), who are not subject to §§ 26.205 – 209
21.	Cyber Security Program	10 CFR 73.54(b); 10 CFR 73.55(b)(8); 10 CFR 73.55(c)(6)	13.6	To be addressed by the COL applicant	10 CFR 73.55(a)(4)
22.	SNM Material Control and Accounting Program	10 CFR 74, Subpart B (§§ 74.11 – 74.19, excl. § 74.17)	13.5.2.2.9	To be addressed by the COL applicant	License Condition
23.	10 CFR Part 37 Physical Protection Program	10 CFR 37 Subparts A, B, C, D, F	13.5.2.2.8, 13.5.2.2.10 13.6, 11.4	To be addressed by the COL applicant	10 CFR Part 37

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13.5 Plant Procedures

This section describes the administrative and other procedures that the operating organization (plant staff) uses to conduct the routine operating, abnormal, and emergency activities in a safe manner.

The Quality Assurance Program Description (QAPD), as discussed in Section 17.5, describes procedural document control, record retention, adherence, assignment of responsibilities, and changes.

Procedures are identified in this section by topic, type, or classification in lieu of the specific title and represent general areas of procedural coverage.

Procedures are issued prior to fuel load to allow sufficient time for plant staff familiarization and to develop operator licensing examinations.

The format and content of procedures are controlled by the applicable AP1000 Writer's Guideline. Upon site acceptance of these procedures, the format and content will be controlled per plant specific administrative procedures.

Each procedure is sufficiently detailed for an individual to perform the required function without direct supervision, but does not provide a complete description of the system or plant process. The level of detail contained in the procedure is commensurate with the qualifications of the individual normally performing the function.

Procedures are developed consistent with guidance described in Section 18.9, "Procedure Development" and with input from the human factors engineering process and evaluations.

References to applicable combined license information are included in Section 1.8. This includes, for example, reference to guidelines on inservice inspection in Chapters 3 and 8, and initial testing in Chapter 14. Operational experience and the resolution of generic issues to be considered in the preparation of plant procedures are outlined in Section 1.9. All portions of the safety-related logic circuitry will be adequately covered in the surveillance procedures as described in Generic Letter 96-01 (Reference 8).

The acceptability of the computerized procedure system, and its backup, for application to the AP1000 design will be determined as outlined in Section 18.8.

The development of plant specific refueling plans (Appendix 19E provides input for refueling plans) is as discussed in Subsection 13.5.2.1.

Outage plans are discussed in Subsection 13.5.2.1 and should as a minimum address the following elements:

- An outage philosophy, which includes safety as a primary consideration in outage planning and implementation,
- Separate organizations responsible for scheduling and overseeing the outage; provisions for an independent safety review team that would be assigned to perform final review and grant approval for outage activities,
- Control procedures, which address both the initial outage plan and all safety-significant changes to schedule,
- Provisions to ensure that all activities receive adequate resources,

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- Provisions to ensure defense-in-depth during shutdown and ensure that margins are not reduced; an alternate or backup system must be available if a safety system or a defense-in-depth system is removed from service, and

- Provisions to ensure that all personnel involved in outage activities are adequately trained; this should include operator simulator training to the extent practicable; other plant personnel, including temporary personnel, should receive training commensurate with the outage tasks they will be performing.

If freeze seals are to be used, plant-specific guidelines will be developed to reduce the potential for loss of RCS boundary and inventory when they are in use.

13.5.1 Administrative Procedures

This section describes administrative procedures that provide administrative control over activities that are important to safety for the operation of the facility.

Procedures outline the essential elements of the administrative programs and controls as described in ANSI/ANS 3.2-1988 (Reference 201) and in Section 17.5. These procedures are organized such that the program elements are prescribed in documents normally referred to as administrative procedures. Regulatory and industry guidance for the appropriate format, content and typical activities delineated in written procedures is implemented as appropriate.

Administrative procedures contain adequate programmatic controls to provide effective interface between organizational elements. This includes contractor and owner organizations providing support to the station operating organization.

A Writer's Guideline promotes the standardization and application of human factors engineering principles to procedures. The Writer's Guideline establishes the process for developing procedures that are complete, accurate, consistent, and easy to understand and follow. The Writer's Guideline provides objective criteria so that procedures are consistent in organization, style, and content. The Writer's Guideline includes criteria for procedure content and format including the writing of action steps and the specification of acceptable acronym lists and acceptable terms to be used.

Procedure maintenance and control of procedure updates are performed in accordance with the QAPD, as discussed in Section 17.5.

The administrative programs and associated procedures developed in the pre-COL phase are described in Table 13.5-201 (for future designation as historical information).

The plant administrative procedures provide procedural instructions for the following:

- Procedures review and approval.

- Equipment control procedures - These procedures provide for control of equipment, as necessary, to maintain personnel and reactor safety, and to avoid unauthorized operation of equipment.

- Control of maintenance and modifications.

- Crane Operation Procedures - Crane operators who operate cranes over fuel pools are qualified and conduct themselves in accordance with ANSI B30.2 (Chapter 2-3), "Overhead and Gantry Cranes" (Reference 202).

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- Temporary changes to procedures.
- Temporary procedure issuance and control.
- Special orders of a temporary or self-canceling nature.
- Standing orders to shift personnel including the authority and responsibility of the shift manager, licensed senior reactor operator in the control room, control room operator and shift technical advisor.
- Manipulation of controls and assignment of shift personnel to duty stations per the requirements of 10 CFR 50.54 (i), (j), (k), (l), and (m) including delineation of the space designated for the “At the Controls” area of the control room.
- Shift relief and turnover procedures.
- Fitness for Duty.
- Control Room access.
- Working hour limitations.
- Feedback of design, construction, and applicable important industry and operating experience.
- Shift Manager administrative duties.
- Verification of correct performance of operational activities.
- A vendor interface program that provides vendor information for safety related components is incorporated into plant documentation.
- Fire protection program implementation.
- A process for implementing the safety/security interface requirements of 10 CFR 73.58.

13.5.2 Operating and Maintenance Procedures

13.5.2.1 Operating and Emergency Operating Procedures

The process to manage the development, review, and approval of AP1000 Normal Operating, Abnormal Operating, Emergency Operating, Refueling and Outage Planning, Alarm Response, Administrative, Maintenance, Inspection, Test, and Surveillance Procedures, as well as the procedures which address the operation of post-72 hour equipment, is delineated in APP-GW-GLR-040 (Reference 10). In addition, APP-GW-GLR-040 provided to the NRC the Writer’s Guidelines for Normal Operating and Two-Column Format Procedures, APP-GW-GJP-100 and APP-GW-GJP-200, respectively.

Upon site review and acceptance of the procedures delineated above, these procedures become site specific. The development, review, and approval of these site specific procedures will be controlled per plant specific processes/procedures.

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13.5.2.2 Maintenance and Other Operating Procedures

The QAPD, as described in Section 17.5, provides guidance for procedural adherence. Regulatory and industry guidance for the appropriate format, content, and typical activities delineated in written procedures is implemented as appropriate.

13.5.2.2.1 Plant Radiation Protection Procedures

The plant radiation protection program is contained in procedures. Procedures are developed and implemented for such things as: maintaining personnel exposures, plant contamination levels, and plant effluents ALARA; monitoring both external and internal exposures of workers, considering industry-accepted techniques; routine radiation surveys; environmental monitoring in the vicinity of the plant; radiation monitoring of maintenance and special work activities; evaluation of radiation protection implications of proposed modifications; establishing quality assurance requirements applicable to the radiation protection program; and maintaining radiation exposure records of workers and others.

13.5.2.2.2 Emergency Preparedness Procedures

A discussion of emergency preparedness procedures can be found in the Emergency Plan.

13.5.2.2.3 Instrument Calibration and Test Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for instrumentation calibration and testing.

13.5.2.2.4 Chemistry Procedures

Procedures provided for chemical and radiochemical control activities include the nature and frequency of sampling and analyses; instructions for maintaining fluid quality within prescribed limits; the use of control and diagnostic parameters; and limitations on concentrations of agents that could cause corrosive attack, foul heat transfer surfaces or become sources of radiation hazards due to activation.

Procedures are also provided for the control, treatment, and management of radioactive wastes and control of radioactive calibration sources.

13.5.2.2.5 Radioactive Waste Management Procedures

Procedures for the operation of the radwaste processing systems provide for the control, treatment, and management of on-site radioactive wastes. Procedural controls are in place for radiological releases.

Operating procedures to limit the total cumulative radioactive inventory of unpackaged wastes located in the radwaste building will be developed, implemented, and maintained prior to initial fuel load. These operating procedures are based on monitoring and controlling the radioactive inventory of unpackaged wastes so that the Regulatory Guide 1.143 unmitigated radiological release criteria of 5 rem to site personnel, and the 10 CFR Part 20.1301 dose limits of 100 millirem at the protected area boundary for members of the public, are not exceeded. These administrative controls limit the radionuclide inventory to less than the A2 limit specified in Appendix A to 10 CFR Part 71 in the radwaste building. Further details regarding administrative controls of unpackaged radioactive waste located in the radwaste building are described in Section 11.2.

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13.5.2.2.6 Maintenance, Inspection, Surveillance, and Modification Procedures

13.5.2.2.6.1 Maintenance Procedures

Maintenance procedures describe maintenance planning and preparation activities. Maintenance procedures are developed considering the potential impact on the safety of the plant, license limits, availability of equipment required to be operable, and possible safety consequences of concurrent or sequential maintenance, testing or operating activities.

Maintenance procedures contain sufficient detail to permit the maintenance work to be performed correctly and safely. Procedures include provisions for conducting and recording results of required tests and inspections, if not performed and documented under separate test and inspection procedures. References are made to vendor manuals, plant procedures, drawings, and other sources as applicable.

Instructions are included, or referenced, for returning the equipment to its normal operating status. Testing is commensurate with the maintenance that has been performed. Testing may be included in the maintenance procedure or be covered in a separate procedure.

The preventive maintenance program, including preventive and predictive procedures, as appropriate for structures, systems and components, prescribes the frequency and type of maintenance to be performed. An initial program based on service conditions, experience with comparable equipment and vendor recommendations is developed prior to fuel loading. The program is revised and updated as experience is gained with the equipment. To facilitate this, equipment history files are created and kept current. The files are organized to provide complete and easily retrievable equipment history.

13.5.2.2.6.2 Inspection Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for inspections.

13.5.2.2.6.3 Modification Procedures

Plant modifications and changes to setpoints are developed in accordance with approved procedures. These procedures control necessary activities associated with the modifications such that they are carried out in a planned, controlled, and orderly manner. For each modification, design documents such as drawings, equipment and material specifications, and appropriate design analyses are developed or the as-built design documents are utilized. Separate reviews are conducted by individuals knowledgeable in both technical and QA requirements to verify the adequacy of the design effort.

Proposed modification(s) which involve a license amendment or a change to Technical Specifications are processed as proposed license amendment request(s).

Plant procedures impacted by modifications are changed prior to declaring the system operable to reflect revised plant conditions; and cognizant personnel who are responsible for operating and maintaining the modified equipment are adequately trained.

13.5.2.2.6.4 Surveillance Procedures

Procedures to perform rod control system surveillance tests specified below (from Reference 7, Section 4.8.2), at the beginning of each fuel cycle are provided as discussed in Subsection 13.5.1.

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The recommended additional surveillance tests to meet the detectability requirement is as follows:

- Once per cycle, verify that correct current orders are being generated for each group of control and shutdown rods. It is recommended that the surveillance be performed during the rod drop timing test. Essentially, the same equipment and personnel are required for both tests. With proper coordination, performing the two together will not result in any additional plant down time.

- It is recommended that the test be performed by monitoring the lift, movable and stationary currents, in the Power Cabinet, for one rod in each group, for both insertion and withdrawal. Performing the monitoring in the Power Cabinet rather than in the Logic Cabinet is recommended because this will detect any Power Cabinet failures that result in current regulation failures that also can corrupt the current profiles and affect individual rod motions within a group.

- While a bank of rods is being withdrawn for the rod drop test, record a few steps of the lift, movable and stationary withdrawal current profiles of one rod in each group. Before the rods are fully withdrawn, insert them a few steps and record the lift, movable and stationary current profiles. When the recordings have been made, continue to withdraw the rods to complete the rod drop test for that group. Verify that the current profiles are in accordance with the system design. Repeat for each shutdown and control bank.

The test being recommended verifies that correct current orders are being generated, for insertion and withdrawal, and therefore that no failures are present in the system at each startup following refueling. The justification for only performing the test once per cycle is that the failures that are required to be detected are only those that do not affect rod motion and therefore are not detectable by means of the existing rod motion surveillance requirement. In order to affect rod motion, a second failure is required. It has further been determined that all failures that affect rod motion of a single group of rods, are bounded by existing analyses. Performing the additional surveillance once per cycle, with the reactor shut down, is therefore considered appropriate.

13.5.2.2.7 Material Control Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for material control.

13.5.2.2.8 Security Procedures

A discussion of security procedures is provided in the Security Plan.

The Special Nuclear Material (SNM) Physical Protection Program describes the 10 CFR Part 70 required protection program in effect for the period of time during which new fuel as SNM is received and stored in a controlled access area (CAA), in accordance with the requirements of 10 CFR 73.67.

Procedures address the applicable 10 CFR 73.67 requirements in the event that unirradiated new fuel assemblies or components are returned to the supplying fuel manufacturer(s) facility.

A 10 CFR Part 37 Physical Protection Program addresses the applicable 10 CFR Part 37 requirements to provide physical protection for an aggregated category 1 or category 2 quantity of radioactive material.

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13.5.2.2.9 Special Nuclear Material (SNM) Material Control and Accounting Procedures

A material control and accounting system consisting of special nuclear material accounting procedures is utilized to delineate the requirements, responsibilities, and methods of special nuclear material control from the time special nuclear material is received until it is shipped from the plant. These procedures provide detailed steps for SNM shipping and receiving, inventory, accounting, and preparing records and reports.

13.5.2.2.10 10 CFR Part 37 Physical Protection Program Procedures

A 10 CFR Part 37 Physical Protection Program consisting of procedures is utilized to delineate the requirements, responsibilities, and methods associated with the physical protection of an aggregated category 1 or category 2 quantity of radioactive material. These procedures provide the detailed steps for receipt, use, storage, transport and preparation of records and reports.

13.5.3 Combined License Information Item

The plant procedures are addressed in APP-GW-GLR-040 (Reference 10), and the applicable changes are incorporated in Section 13.5.

13.5.4 References

1. Not used.

2. Not used.

3. Not used.

4. Not used.

5. Not used.

6. Not used.

7. WCAP-13864, “Rod Control System Evaluation Program,” Revision 1-A, November 1994.

8. USNRC Generic Letter 96-01, “Testing of Safety-Related Logic Circuits,” January 10, 1996.

9. Not used.

10. APP-GW-GLR-040, “Plant Operations Maintenance and Surveillance Procedures,” Westinghouse Electric Company LLC.

201. ANSI/ANS 3.2-1988, “Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants.”

202. ANSI B30.2 (Chapter 2-3), “Overhead and Gantry Cranes.”

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Table 13.5-201
Pre-COL Phase Administrative Programs and Procedures

(This table is included for future designation as historical information.)

- Design/Construction Quality Assurance Program
- Reporting of Defects and Noncompliance, 10 CFR Part 21 Program
- Design Reliability Assurance Program

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13.6 Security

The Security Plan consists of the "AP1000 Physical Security Plan," Training and Qualification Plan, and Safeguards Contingency Plan. The Security Plan will be submitted to the Nuclear Regulatory Commission as a separate licensing document in order to fulfill the requirements for 10 CFR 52.79(a)(35) and 10 CFR 52.79(a)(36). The Security Plan will meet the requirements of 10 CFR 52.98(c). The plan is classified as Security Safeguards Information and is withheld from public disclosure pursuant to 10 CFR 73.21. Additionally, the "AP1000 Interim Compensatory Measures Report" (Reference 2), the "AP1000 Enhancement Report" (Reference 3), and the "AP1000 Safeguards Assessment Report" (Reference 4) are submitted to the Nuclear Regulatory Commission as separate licensing documents to establish the design of the AP1000 Security Systems. Each document is classified as Security Safeguards information and is withheld from public disclosure pursuant to 10 CFR 73.21.

The explosive cartridge assembly used in the squib valves (see Subsection 6.3.2.2.8.9) has been determined by the Bureau of Alcohol, Tobacco, Firearms and Explosives (ATF) to be exempt (Reference 5) from the requirements of 27 CFR 555 as special explosive devices pursuant to 27 CFR 555.32 for possession and use for their intended purpose in the AP1000 nuclear reactor passive safety system. However, the explosive materials used in their manufacture are subject to the provisions of 27 CFR, Part 555. Thus, any alterations to the explosive cartridge assembly will render this exemption void and any explosive materials removed from the explosive cartridge assembly are subject to the regulations of 27 CFR 555.

13.6.1 Combined License Information Item

Combined License applicants referencing the AP1000 certified design will address site-specific information related to the security, contingency, and guards training plans. The Combined License applicant will develop the Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan. Combined License applicants will address site-specific security ITAACs as applicable.

The Combined License holder will develop and implement a Cyber Security Program.

13.6.2 References

1. Not used.

2. APP-GW-GLR-067, "AP1000 Interim Compensatory Measures Report," Westinghouse Electric Company LLC.

3. APP-GW-GLR-062, "AP1000 Enhancement Report," Westinghouse Electric Company LLC.

4. APP-GW-GLR-066, "AP1000 Safeguards Assessment Report," Westinghouse Electric Company LLC.

5. U.S. Department of Justice, Bureau of Alcohol, Tobacco, Firearms and Explosives (ATF) letter to Westinghouse Electric Company, dated October 6, 2015, and associated Special Explosive Device Determination and exemption (ATF Reference No. 15-801205).

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13.7 Fitness for Duty

The Fitness for Duty Program (FFD) is implemented and maintained in multiple and progressive phases dependent on the activities, duties, or access afforded to certain individuals at the construction site. In general, two different FFD programs will be implemented: a construction FFD program and an operations FFD program. The construction and operations phase programs are illustrated in Table 13.4-201.

The construction FFD program is consistent with NEI 06-06 (Reference 201), as endorsed by NRC Regulatory Guide 5.84, Revision 0, and the pre-access provisions for reinstatement described in 10 CFR 26.65. NEI 06-06 applies to persons constructing or directing the construction of safety- and security-related structures, systems, or components performed onsite where the new reactor will be installed and operated. Management and oversight personnel, as further described in NEI 06-06, and security personnel prior to the receipt of special nuclear material in the form of fuel assemblies (with certain exceptions) will be subject to the operations FFD program that meets the requirements of 10 CFR Part 26, Subparts A through H, N, and O. At the establishment of a protected area, all persons who are granted unescorted access will meet the requirements of an operations FFD program. The NRC endorsed NEI 06-06 (Reference 201) and the FFD program subject to Subpart K was revised to reflect the changes.

The operations phase FFD program is consistent with the applicable subparts of 10 CFR Part 26 (elements Subpart A – I, N, and O, except for individuals listed in §26.4(b), who are not subject to §§ 26.205 – 209.

13.7.1 References

201. Nuclear Energy Institute “Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites,” NEI 06-06, Revision 6, April 2013 (ML13093A342).

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Chapter 14 Initial Test Program

14.1 Specific Information to be Included in Preliminary/Final Safety Analysis Reports

Not applicable to the AP1000.

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14.2 Specific Information to be Included in Standard Safety Analysis Reports

14.2.1 Summary of Initial Test Program and Objectives

The purpose of this section is to describe the initial test program that is performed for the AP1000 plant.

The overall objective of the initial test program is to demonstrate that the plant has been constructed as designed, that the systems perform consistent with the plant design, and that activities culminating in operation at full licensed power including initial fuel load, initial criticality, and power ascension are performed in a controlled and safe manner.

A graded approach, based on the criteria provided in Regulatory Guide 1.68, is used to develop and implement the Initial Test Program (ITP). Preoperational and/or startup testing is performed on those systems that are:

a) Relied upon for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period;

b) Relied upon for safe shutdown and cooldown of the reactor under transient and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions;

c) Relied upon for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications;

d) Classified as engineered safety features actuation systems (ESFAS) or are relied upon to support or ensure operation of engineered safety features actuation systems within design limits;

e) Assumed to function or for which credit is taken in the accident analysis of the AP1000 as described in this document.

f) Used to process, store, control, or limit the release of radioactive materials.

g) Other systems identified in Regulatory Guide 1.68, Revision 2, Appendix A that are in the AP1000 and are not captured by criteria a) through f).

The inspections, tests, analyses and acceptance criteria of 10 CFR 52.47 (b)(1) relating to the AP1000 design are found in the AP1000 Certified Design Material (see Section 14.3).

The ITP is implemented in two phases, categorized as preoperational and startup testing. The individual programs for these two phases, which make up the overall ITP, are discussed in Subsections 14.2.1.1 and 14.2.1.2. For clarity, it is noted that construction and installation tests, including component tests, are performed to verify that the construction, installation, and assembly processes have been properly performed. Construction and installation tests, including component tests, are not considered part of the ITP.

- Preoperational tests are performed after associated prerequisites for the required test, such as a completion of associated construction activities, component and/or preliminary tests, and inspections, but prior to initial fuel loading to demonstrate the capability of plant systems to meet performance requirements. These prerequisites should be completed before beginning preoperational tests (to the degree that outstanding items could not be expected to affect the validity of test results).

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- Startup tests begin with the initial fuel loading and are performed to demonstrate the capability of individual systems, as well as the integrated plant, to meet performance requirements.

Section 14.2 provides the requirements to be included in the ITP Administrative Manual (Procedures), as discussed in Subsection 14.4.3. The information referenced in this section meets the Initial Test Program (ITP) criteria of NUREG-0800 and is formatted to follow Regulatory Guide 1.206, Part I, Section C.I.14.2.

The ITP is applied to structures, systems, and components that perform the functions described in the Regulatory Guide 1.68 evaluation in Section 1.9. The ITP is also applied to other structures, systems and components. The ITP Administrative Manual will reference the AP1000 structures, systems and components to which the ITP is applied.

14.2.1.1 Preoperational Test Program Objectives

Following completion of associated prerequisites, preoperational tests are performed to demonstrate that equipment and systems perform in accordance with design criteria so that initial fuel loading, initial criticality, and subsequent power operation can be safely undertaken. Preoperational tests at elevated pressure and temperature are referred to as hot functional tests.

The general objectives of the preoperational test program are the following:

- Demonstrate that essential plant components and systems, including alarms and indications, meet appropriate criteria based on the design
- Provide documentation of the performance and condition of equipment and systems
- Provide baseline test and operating data on equipment and systems for future use and reference
- Operate equipment for a sufficient period to demonstrate performance
- Demonstrate that plant systems operate on an integrated basis

A graded approach, in accordance with Regulatory Guide 1.68, is used to develop and implement the preoperational test program. Acceptance criteria, consisting of performance objectives and functional requirements, are established for each safety-related or defense-in-depth function the SSCs perform. Either qualitative or quantitative in nature, acceptance criteria are used to determine success or failure of these preoperational tests. Functional criteria are established for functions the SSCs perform which are neither safety-related nor defense-in-depth. Test procedures will differentiate between acceptance criteria and functional criteria for preoperational tests.

- Abstracts for preoperational tests for portions of systems/components that perform safety-related functions and/or defense-in-depth functions are provided in Subsections 14.2.9.1 and 14.2.9.2, respectively.
- Abstracts for preoperational tests for portions of systems/components that contain, transport, or isolate radioactive material and for other applicable systems that are specified in Regulatory Guide 1.68, Appendix A, Revision 2, are provided in Subsections 14.2.9.3 and 14.2.9.4, respectively.

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Plant operating, emergency, and surveillance procedures are incorporated into the initial test program procedures to the extent practical. These procedures are verified through use, to the extent practicable, during the preoperational test program and revised if necessary, prior to fuel loading.

Plant equipment used in the performance of preoperational tests is operated, to the extent practical, in accordance with appropriate operating procedures, thereby giving the plant operating staff an opportunity to gain experience in using these procedures and demonstrating their adequacy prior to plant initial criticality.

14.2.1.2 Startup Test Program Objectives

The startup test program begins with initial fuel loading after the preoperational testing has been successfully completed.

Startup tests can be grouped into four broad categories:

- Tests related to initial fuel loading
- Tests performed after initial fuel loading but prior to initial criticality
- Tests related to initial criticality and those performed at low power (less than 5 percent)
- Tests performed at power levels greater than 5 percent

During performance of the startup test program, the plant operating staff has the opportunity to obtain practical experience in the use of normal and abnormal operating procedures while the plant progresses through heatup, criticality, and power operations.

The general objectives of the startup test program are:

- Install the nuclear fuel in the reactor vessel in a controlled and safe manner.
- Verify that the reactor core and components, equipment, and systems required for control and shutdown have been assembled according to design and meet specified performance requirements.
- Achieve initial criticality and operation at power in a controlled and safe manner.
- Verify that the operating characteristics of the reactor core and associated control and protection equipment are consistent with design requirements and accident analysis assumptions.
- Obtain the required data and calibrate equipment used to control and protect the plant.
- Verify that the plant is operating within the limits imposed by the Technical Specifications.

Abstracts of the startup tests are provided in Subsection 14.2.10.

14.2.1.3 Not Used

14.2.1.4 Not Used

14.2.2 Organization, Staffing, and Responsibilities

The Licensee has the overall responsibility to manage and conduct preoperational and startup testing.

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The ITP Organization is described in Subsection 14.2.2.1. The organization for operating and maintaining the AP1000 plant is described in Section 13.1.

Table 13.4-201 provides milestones for initial test program implementation.

14.2.2.1 ITP Organization

The ITP organization is functionally comprised of testing groups and organizational support groups. The groups are responsible for supporting and conducting preoperational and startup testing and may be divided by functional area. The ITP Organization structure (organizational chart) is included in the ITP Administrative Manual.

14.2.2.1.1 Senior Managers Responsible for ITP

The Senior Managers Responsible for ITP are responsible for:

- Staffing the ITP Organization.
- Developing and reviewing the administrative and technical procedures associated with preoperational and startup testing.
- Managing the ITP Organization.
- Implementing the preoperational and startup testing schedule.
- Managing contracts associated with preoperational and startup testing.

14.2.2.1.2 Functional Managers In Charge of ITP Organization Support

The functional managers in charge of ITP Organization support report to the Senior Managers Responsible for ITP and are responsible for providing the necessary supporting structure and processes required for the effective implementation of the ITP. Specifically, these managers are responsible for:

- Planning and scheduling procedure development to support preoperational and startup testing
- Managing turnover from construction to the ITP organization and the transition from the ITP organization to operations
- Implementing work controls during the ITP including planning of test packages
- Managing the training and qualification of ITP personnel
- Managing the processing of ITP records and documents for storage
- Managing JTWG activities including the review and approval of ITP administrative procedures
- Scheduling and tracking ITP activities to support the project schedule

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14.2.2.1.3 Functional ITP Managers in Charge of Testing

The functional managers in charge of testing report to the Senior Managers Responsible for ITP and are responsible for:

- Managing testing per the test schedule.
- Managing initial energization.
- Managing component checkout, calibration and digital controls interface verification for plant components.
- Ensuring that the JTWG reviews and approves preoperational test procedures for SSCs identified in Subsections 14.2.9.1 and 14.2.9.2.
- Reviewing and approving preoperational test procedures for SSCs identified in Subsections 14.2.9.3 and 14.2.9.4.
- Ensuring that the JTWG reviews and approves startup test procedures.
- Coordinating vendor participation in testing activities, as required.
- Supervising and directing test personnel.
- Supporting administrative controls to address system and equipment configuration control.
- Verifying the adequacy of test results.
- Supporting the transition from construction to the preoperational testing phase.
- Preparing and/or reviewing preoperational test procedures.
- Involving operations personnel in testing activities. Utilizing operations personnel, to the extent practical, as test witnesses or test performers to provide the operations personnel with experience and knowledge.
- Preparing a detailed startup testing schedule.
- Maintaining the startup schedule.

14.2.2.1.4 Test Engineers

Test engineers report directly to one of the functional managers in charge of testing. The test engineers are responsible for:

- Complying with administrative controls.
- Developing, reviewing, and performing test procedures.
- Identifying any special or temporary equipment or services needed to support testing.
- Coordinating testing with involved groups.
- Reviewing and evaluating test results.

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14.2.2.2 ITP Organization Personnel Qualifications and Training

Procedures are prepared to identify the appropriate training, qualification, and certification for test personnel. Records are kept to document training, qualification and certification, as appropriate. Test personnel are qualified as described below:

- Acceptable qualifications of non-supervisory test engineers follow the guidance provided in Regulatory Guide 1.28 as discussed in Appendix 1A, i.e., ASME NQA-1-1994, Appendix 2A-1, Nonmandatory Guidance on the Qualification of Inspection and Test Personnel.

- Acceptable qualifications of supervisory test engineers follow the guidance in Regulatory Guide 1.28, i.e., ANSI/ANS-3.1-1993, Selection Qualification, and Training of Personnel for Nuclear Power Plants.

The training program/procedures shall include:

- The education, training, experience, and qualification requirements for ITP Organization supervisory and test personnel responsible for managing, developing, or conducting each test phase, or development of testing procedures. Refer to Section 13.2 for training and qualification information for the organization responsible for operating and maintaining the plant.

The ITP Administrative Manual (Procedure) shall include:

- The implementation of measures to verify that personnel formulating and conducting test activities are not the same personnel who designed or are responsible for satisfactory performance of the system(s) or design features(s) being tested. This provision does not preclude members of the design organization from participating in test activities. This description also includes considerations of staffing effects that could result from overlapping initial test programs at multi-unit sites.

14.2.2.3 Joint Test Working Group

The Joint Test Working Group (JTWG) consists of an organizational group of authorized representative personnel from the Plant's operations and support group functions, Responsible Design Organization and other test support groups as identified below.

The Licensee has the overall responsibility for conduct of the ITP. The Senior Managers Responsible for ITP have been assigned overall responsibility and authority for technical direction of the preoperational and startup testing and may act as the JTWG Chairman.

The JTWG Chairman reports to the Senior Managers Responsible for ITP or designee(s), for matters of preoperational or startup testing authority and acceptance.

The JTWG provides the following administrative oversight activities associated with preoperational and startup testing:

- Review and approve ITP Administrative Manual.

- Review and approve preoperational test procedures, for tests described in Subsections 14.2.9.1 and 14.2.9.2, and startup test procedures.

- Oversee the implementation of preoperational and startup testing, including planning, scheduling, and performance.

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- Review and approve preoperational test results, for tests described in Subsections 14.2.9.1 and 14.2.9.2, and startup test results.

The JTWG is composed of qualified representatives provided from the following organizations:

- Operations Group
- Maintenance Group
- ITP Test Groups
- Engineering Group
- Responsible Design Organization

The following are additional generic details of the key responsibilities, authorities and interfaces of the organizations delineated above:

- Operations Group

  The Operations Group has the overall responsibility for Plant Operations, including administrative control and tag-outs subsequent to system turnover. Their primary interfaces are with the Engineering and Technical Support organizations as well as the responsible design organization and ITP Testing Teams.

- Maintenance Group

  The Maintenance Group has the overall responsibility for the Maintenance of Plant systems and components subsequent to System Turnover. They are key participants and maintainers of system maintenance control and tag-outs. Their primary interfaces are with the Operations Group and Technical Support organizations as well as the responsible design organization and ITP Testing Teams.

- ITP Test Groups

  The ITP Test Groups have the primary responsibility for the development, maintenance and performance of preoperational and startup procedures at the site. The primary interfaces for these groups are the Operations Group and Technical support organizations as well as the responsible design organization and the Construction Services Group.

- Engineering Group

  This group has the primary responsibility for engineering and design oversight of the plant components and systems, as well as interfacing with the vendor engineering organization. This organization primarily interfaces with the Operations Group as well as the responsible design organization and ITP Testing Teams.

- Responsible Design Organization

  The responsible design organization will vary and may be Westinghouse Electric Company (WEC), and/or the Licensee depending on the particular area or SSC in question. This organization has the primary responsibility for the approval of the document indicating that the acceptance and/or functional criteria are in accordance with the current approved design. Resolution of any questions from the review and use of engineering documents to support test

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procedures is the responsible design organization's responsibility through interface between the site and the vendor's home offices. The responsible design organization will resolve any deficiencies that occur during testing of structures, systems, and components that cannot be resolved within the ITP Organization.

The responsible design organization will interface with the ITP Organization. As necessary, the responsible design organization may also interface with the Licensee Operations, Maintenance and Engineering groups.

14.2.2.4 ITP Test Groups

The ITP Test Groups perform the following functions and scope of work as necessary to support the Initial Test Program:

- Support acceptance of construction turnover to the ITP organization.
- Support/perform initial energization and operation of plant equipment as well as functional performance tests at the component or sub-system level.
- Perform component checkout, calibration, and digital controls interface verification for plant components.
- Coordinate tagging and maintenance to support testing.
- Manage and oversee the testing of plant systems to support completion of hot functional testing.
- Resolve open items and exceptions identified during implementation of the Preoperational Test Program.
- Present preoperational test results, for tests described in Subsections 14.2.9.1 and 14.2.9.2, to the JTWG for review and approval.
- Coordinate other support tasks required during Preoperational Testing activities with responsible groups.
- Assist in planning, scoping, and scheduling plant systems, structures and components for testing, to support startup testing.
- Support initial fuel loading and precritical testing, initial criticality, low power, and power ascension testing.
- Resolve open items and exceptions identified during implementation of the Startup Test Program.
- Coordinate other support tasks required during Startup Testing activities with responsible groups.

14.2.2.5 ITP Organizational Support Groups

The ITP Organizational Support Groups perform the following functions and scope of work as necessary to support the Initial Test Program:

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- Support acceptance of construction to the ITP organization and transition from the ITP organization to operations
- Implement work controls during the ITP
- Process ITP records and documents for storage
- Train and qualify ITP personnel
- Plan, scope and schedule plant systems for testing

14.2.3 Test Procedures

Preoperational and startup tests are performed using test procedures.

For each test, the test procedure specifies the following, as applicable:

- Objectives for performing the test
- Prerequisites that must be completed before the test can be performed
- Initial conditions under which the test is started
- Special precautions required for the safety of personnel or equipment
- Instructions delineating how the test is to be performed
- Identification of the required data to be obtained and the methods for documentation
- Data reduction analysis methods as appropriate
- Test acceptance and/or functional criteria

Test procedures are developed and reviewed by personnel with appropriate technical backgrounds and experience. This includes the participation of responsible design organizations in the establishment of test performance requirements and acceptance and/or functional criteria.

Available information on operating or testing experiences of operating reactors is factored into the test procedures as appropriate.

Copies of the test procedures for the startup tests are available to NRC inspection personnel not less than 60 days prior to the scheduled fuel loading date.

Copies of the test procedures are available to NRC inspection personnel approximately 60 days prior to the scheduled performance of the following preoperational tests:

- Tests of systems/components that perform safety-related functions
- Tests of systems/components that are nonsafety-related but perform defense in-depth functions.

Test procedures for preoperational tests described in Subsections 14.2.9.3 and 14.2.9.4 of the plant systems/components which perform no safety-related or defense-in-depth functions are available to NRC inspection personnel prior to the scheduled performance of these tests.

Preoperational and startup tests are performed with the quality assurance requirements as specified in Section 17.5.

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The ITP Administrative Manual shall include the following controls:

- Controls to provide for turnover of plant systems and components from construction to the preoperational testing group. These administrative controls also ensure that general prerequisites are satisfied prior to preoperational and/or startup tests of individual systems or components.
- For preoperational and startup tests, controls to provide test procedures that include appropriate prerequisites, objectives, special precautions, initial test conditions, methods to direct and control test performance, and acceptance and/or functional criteria by which the test is evaluated.
- For preoperational and startup tests, controls for the format of individual test procedures to provide consistency with the guidance contained in Regulatory Guide 1.68, as applicable.
- For preoperational and startup tests, controls to provide for participation of the responsible design organizations in establishing test objectives, test acceptance and/or functional criteria, and related performance requirements during the development of detailed test procedures. Each test procedure should include acceptance criteria that account for the uncertainties used in transient and accident analyses. The responsible design organization will solicit support from design agents (i.e., major contractors, subcontractors, and vendors) as needed.
- Controls to provide for personnel with appropriate technical backgrounds and experience to develop and review preoperational and startup test procedures and/or written work instructions. Persons filling designated management positions should perform final procedure review and approval.
- Controls to make the approved preoperational and startup test procedures for satisfying FSAR testing commitments available to NRC inspectors approximately 60 days prior to their intended use and prior to fuel load, respectively.

14.2.3.1 Conduct of Initial Test Program

Administrative procedures and requirements that govern the activities of the conduct of the initial test program include the following:

- Format and content of test procedures
- Process for both initial issue and subsequent revisions of test procedures
- Review process for test results
- Process for resolution of failures to meet performance criteria and of other operational problems or design deficiencies
- Various phases of the initial test program and the requirements for progressing from one phase to the next, as well as requirements for moving beyond selected hold points or milestones within a given phase
- Controls to monitor the as-tested status of each system and modifications including retest requirements deemed necessary for systems undergoing or already having completed testing
- Qualifications and responsibilities of test personnel within the ITP Organization

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The ITP Administrative Manual (procedures) supplements normal plant administrative procedures by addressing those administrative issues that are unique to the ITP.

The ITP Administrative Manual (procedure) governs the initial testing and is issued no later than 60 days prior to the beginning of the preoperational test phase. Testing during all phases of the test program is conducted using approved test procedures.

14.2.3.1.1 Preoperational and Startup Test Procedure Verification

Since procedures may be approved for implementation weeks or months in advance of the scheduled test date, a review of the approved test procedure is required before commencement of testing. The test engineer is responsible for verifying:

- Drawing and document revision numbers listed in the reference section of the test procedure agree with the latest revisions.
- The procedure text reflects any design and licensing (i.e., FSAR and Technical Specifications) changes made since the procedure was originally approved for implementation.
- Any new (since preparation of the procedure) Operating Experience lessons learned are incorporated into individual test procedures.

Procedures require signoff verification for prerequisites and instruction steps. This signoff includes identification of the person doing the signoff and the date of completion.

Test engineers maintain logs of test status to facilitate turnover and aid in maintaining configuration control. These logs become part of the test documentation.

There is a documented test turnover process to make known the test status and equipment configuration when personnel transfer responsibilities, such as during a shift change.

Test briefings are conducted for each test in accordance with administrative procedures. When a shift change occurs before test completion, another briefing occurs before resumption or continuation of the test.

Data collected is marked or identified with test, date, and person collecting data. This data becomes part of the test documentation.

Test issues (collectively including deficiencies, discrepancies, exceptions, non-conformances, delays, etc.) will be documented and addressed in accordance with the requirements stipulated in the ITP Administrative Manual. Those determined by the ITP Administrative Manual criteria and the test results that do not meet test acceptance or functional criteria are entered into the applicable corrective action program.

Plant management approves proceeding from one test phase to the next during the ITP.

Administrative procedures detail the test documentation review and approval process.

14.2.3.1.2 Work Control

The group having jurisdictional control is responsible for initiating work requests when assistance is required from the Construction or Licensee’s organization. Work requests are issued in accordance with site specific procedures governing the work management process.

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After turnover to the Licensee, activities requiring Construction organization work efforts are performed under plant equipment clearance procedures. Equipment clearance requests are governed by a site-specific procedure for protection of personnel and equipment.

The group having jurisdictional control is responsible for supervising minor repairs and modifications, changing equipment settings, and disconnecting and reconnecting electrical terminations as stipulated in a test or administrative procedure. Test Engineers may perform independent verification of changes made in accordance with approved procedures.

14.2.3.1.3 System Turnover

During the construction phase, systems, subsystems, and equipment are completed and turned over in an orderly and well-coordinated manner. Guidelines are established to define the boundary and interface between related system/subsystem and are used to generate boundary scope documents; for example, marked-up piping and instrument diagrams (P&IDs) and electrical schematic diagrams are provided for scheduling and subsequent development of component and system turnover packages. The system turnover process includes requirements for the following:

- Documenting inspections performed by the construction organization (e.g., highlighted drawings showing areas inspected).
- Documenting results of construction testing.
- Determining the construction-related inspections and tests, including component testing, that need to be completed before preoperational testing begins.
- Evaluating any open items for acceptability before commencing preoperational or startup testing.
- Developing and implementing plans for correcting adverse conditions and open items, and means for tracking such conditions and items.
- Verifying completeness of construction and documentation of incomplete items.

14.2.3.1.4 Conduct of Modifications and Configuration Changes During the Initial Test Program

Temporary configuration changes may be required to conduct certain tests. These configuration changes are documented in the test procedures, and/or written work instructions, or in accordance with the ITP Administrative Manual. The test procedures, and/or written work instructions, contain restoration steps and retesting necessary to confirm satisfactory restoration to the required configuration. Modifications may be performed by the Construction organization or the plant staff processes prior to NRC issuance of the 10 CFR 52.103(g) finding. If the modification invalidates a previously completed ITAAC, then that ITAAC is re-performed. Each modification is reviewed to determine the scope of post-modification testing that is to be performed. Testing is conducted and documented to maintain the validity of preoperational testing and ITAAC. Modifications and configuration changes made following NRC issuance of the 10 CFR 52.103(g) finding are in accordance with plant processes and meet license conditions. Modifications that require changes to ITAAC require NRC approval of the ITAAC change.

14.2.3.1.5 Conduct of Maintenance During the Initial Test Program

Corrective or preventive maintenance activities are reviewed to determine the scope of postmaintenance testing to be performed. Prior to NRC issuance of the 10 CFR 52.103(g) finding,

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post-maintenance testing is conducted and documented to maintain validity of associated preoperational testing and ITAAC remain valid. Maintenance performed following NRC issuance of the 10 CFR 52.103(g) finding is in accordance with plant staff processes and meets license conditions.

14.2.3.2 Review of Test Results

Final review of the individual tests is discussed in Section 14.4.

14.2.3.2.1 Review and Approval Responsibilities

Upon completion of a test, the test engineer is responsible for:

- Reviewing the test data.
- Evaluating the test results.
- Verifying that the acceptance and/or functional criteria are met.
- Verifying that the test results that do not meet acceptance and/or functional criteria are entered into the applicable corrective action program.
- Verifying that the results of retesting do not invalidate ITAAC acceptance criteria.

Preoperational test results, for tests described in Subsections 14.2.9.1 and 14.2.9.2, and startup test results are reviewed and approved by the JTWG. Preoperational test results, for tests described in Subsections 14.2.9.3 and 14.2.9.4, are reviewed and approved by Functional ITP Manager in Charge of Testing. Test deficiencies which do not meet acceptance and/or functional criteria are identified to the affected and responsible design organizations, as appropriate, and entered into the applicable corrective action program. Implementation of corrective actions and retests are performed as required.

Prior to initial fuel load, the results of the preoperational test phase are comprehensively reviewed by the ITP Organization and the JTWG to verify the results indicate that the required plant structures, systems, and components are capable of supporting the initial fuel load and subsequent startup testing. The plant manager approves fuel loading.

Completed startup tests are reviewed and evaluated by the ITP Organization and the JTWG. The test results at each power ascension testing power plateau are reviewed and evaluated by the ITP Organization and the JTWG and approved by the plant manager before proceeding to the next plateau. Startup test reports are prepared in accordance with the guidance in position C.9 of Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

14.2.3.2.2 Technical Evaluation

Each completed test is reviewed by technically qualified personnel to confirm satisfactory demonstration of plant, system or component performance and compliance with design and license criteria.

14.2.3.3 Test Records

Retention periods for test records are based on considerations of their usefulness in documenting initial plant performance characteristics, and are retained in accordance with Regulatory Guide 1.28.

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14.2.3.3.1 Startup Test Reports

Startup test reports are generated describing and summarizing the completion of tests performed during the ITP. A startup report is completed at the earliest of:

1) 9 months following initial criticality,

2) 90 days after completion of the ITP, or

3) 90 days after start of commercial operations. If one report does not cover all three events, then supplemental reports are completed every three months until all three events are completed. These reports:

- Address each ITP test described in the FSAR.
- Provide a general description of measured values of operating conditions or characteristics obtained from the ITP as compared to design or specification values.
- Describe any corrective actions that were required to achieve satisfactory operation.
- Include any other information required by license conditions.

14.2.4 Compliance of Initial Test Program with Regulatory Guides

Subsection 1.9.1 and Table 1.9-1 discuss compliance with the applicable NRC regulatory guides.

14.2.5 Utilization of Reactor Operating and Testing Experience in the Development of Initial Test Program

The design, testing, startup, and operating experience from previous pressurized water reactor plants is utilized in the development of the preoperational and startup test program for the AP1000 plant. Other sources of experience reported and described in documents such as NRC reports, including Inspection and Enforcement bulletins and Institute of Nuclear Power Operations (INPO) reports, including Significant Operating Event Reports (SOER), are also utilized in the AP1000 preoperational and startup test program.

Utilization of Operating Experience

Administrative procedures provide methodologies for evaluating and initiating action for operating experience information (OE). This subsection describes the general use of operating experience in the development of the test program.

14.2.5.1 Use of OE During Test Procedure Preparation

Administrative procedures require review of recent internal and external operating experience when preparing test procedures.

14.2.5.2 Sources and Types of Information Reviewed for ITP Development

Multiple sources of operating experience were reviewed to develop this description of the ITP administration program. These included INPO Reports, INPO Guidelines, INPO Significant Event Reports, INPO Significant Operating Experience Reports and NRC Regulatory Guide 1.68.

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14.2.5.3 Conclusions from Review

The following conclusions are a result of the OE review conducted to develop this ITP administration program description:

- The test procedures should provide guidance as to the expected plant response and instructions concerning what conditions warrant aborting the test. Errors and problems with the procedures should be anticipated. A means for prompt but controlled approval of changes to test procedures is needed. Critical test procedures should provide specific criteria for test termination and specific steps to conduct termination are conducted in a safe and orderly manner. Providing procedural guidance for aborting the test could prevent delays in plant restoration. Conservative guidance for actions to be taken should be included in the procedures.
- Plant simulators may prove useful in preparing for special tests and verifying procedures.
- Appropriate component/system operability should be verified prior to critical tests.
- The need to perform physics tests that can produce severe power tilts should be evaluated, particularly if tests at other similar reactors have provided sufficient data to verify the adequacy of the nuclear physics analysis.
- Compensatory measures should be implemented in accordance with guidance for infrequently performed tests or evolutions, where appropriate.

14.2.5.4 Summary of Test Program Features Influenced by the Review

The conclusions from the preceding section were incorporated in Section 14.2.

14.2.5.5 Use of OE during Conduct of ITP

Administrative procedures require discussion of operating experience when performing pre-job briefs immediately prior to the conduct of a test.

14.2.6 Use of Plant Operating and Emergency Procedures

As appropriate and to the extent practicable, plant normal, abnormal, and emergency operating procedures are used when performing preoperational tests.

The use of these procedures is intended:

- To demonstrate the adequacy of the specific procedure or to identify changes that may be required
- To increase the level of knowledge of plant personnel on the systems being tested

A test procedure using a normal, abnormal, or emergency operating procedure references the procedure directly or extracts a series of steps from the procedure in the way that accomplishes the operator training goals while safely and efficiently performing the specified testing.

These procedures are used extensively in the Human-Machine Interface Testing, which is integrated as a part of the Control Room Design finalization. Additionally, the AP1000 plant operating and emergency procedures are developed to support the following design finalization activities:

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- Human Factors Engineering
- Operational Task Analysis
- Training Simulator Development
- Verification and Validation of the Procedures and the Training Material

The AP1000 emergency, abnormal and some normal operating procedures, along with some Alarm Response Procedures and surveillance procedures, are exercised and verified in the processes delineated above and in the Control Room design finalization process.

14.2.6.1 Operator Training and Participation during Certain Initial Tests (TMI Action Plan Item I.G.1, NUREG-0737)

The objective of operator participation is to increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and offnormal events is conducted.

The ITP may result in the discovery of an acceptable plant or system response that differs from the expected response. Test results are reviewed to identify these differences and the training for operators is changed to reflect them. Training is conducted as soon as is practicable in accordance with training procedures. Just in Time training is conducted for selected startup tests.

14.2.7 Initial Fuel Loading and Initial Criticality

Initial fuel loading and subsequent initial criticality and power ascension to full licensed power are performed during the startup test program. Prior to the initiation of these operations, the systems and conditions necessary to bring the plant into compliance with the Technical Specifications must be operable and satisfied. These operations are performed in a controlled and safe manner by using test procedures that specify:

- Required prerequisite testing
- Operational status of required systems
- Step-by-step instructions
- Precautions which must be observed
- Actions to be taken in the event of unanticipated or abnormal response

14.2.7.1 Initial Fuel Loading

The minimum conditions for initial core loading include:

- The composition, duties, and emergency procedure responsibilities of the fuel handling crew are established.
- Radiation monitors, nuclear instrumentation, manual initiation controls, and other devices to actuate alarms and ventilation controls are tested and verified to be operable.
- The status of systems required for fuel loading is established and verified.
- The status of protection systems, interlocks, alarms, and radiation protection equipment is established and verified for fuel loading.
- Inspections of fuel and control rods have been made.

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- Containment integrity has been established to the extent required by the Technical Specifications.
- The reactor vessel status has been established for fuel loading. Components are verified to be in place or out of the vessel as required for fuel loading.
- Required fuel handling tools are available, operational, and calibrated to include indexing of the manipulator crane with a dummy fuel element. The fuel handling tools have been successfully tested.
- Reactor coolant water quality requirements are established and the reactor coolant water quality is verified.
- The reactor vessel is filled with water to a level at or above the center of the vessel outlet nozzles. The reactor coolant water is circulating at a rate which provides uniform mixing.
- The boron concentration in the reactor coolant is verified to be equal to or greater than required by the plant Technical Specifications for refueling and is being maintained under a surveillance program.
- Sources of unborated water to the reactor coolant system have been isolated and are under a surveillance program.
- At least two neutron detectors are calibrated, operable, and located in such a way that changes in core reactivity can be detected and recorded. One detector is connected to an audible count rate indicator and a containment alarm.
- A response check of nuclear instruments to a neutron source is required within 8 hours prior to loading (or resumption of loading if delayed for 8 hours or more).

Fuel assemblies together with inserted components (control rods, burnable poison assemblies, primary and secondary neutron sources) are placed in the reactor vessel, according to an established and approved sequence.

During and following the insertion of each fuel assembly, until the last fuel assembly has been loaded, the response of the neutron detectors is observed and compared with previous fuel loading data or calculations to verify that the observed changes in core reactivity are as expected. Specific instructions are provided if unexpected changes in reactivity are observed.

Because of the unique conditions that exist during initial fuel loading, temporary neutron detectors may be used in the reactor vessel to provide additional reactivity monitoring. Credit for the use of temporary detectors may be taken in meeting Technical Specifications requirements on the number of operable source range channels.

14.2.7.2 Initial Criticality

Following initial fuel loading, the reactor upper internals and the pressure vessel head are installed. Additional mechanical and electrical tests are performed in preparation for critical and power operations. The following conditions exist prior to initial criticality:

- The reactor coolant system is filled and vented.

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- Tests are completed on the control rod drive system that demonstrate that the control rods have been latched, that the control and position indication systems are functioning properly, and that the rod drop time under hot full flow conditions is less than or equal to the Technical Specifications limit.

- Tests are completed that demonstrate that plant control and protection systems are operable and that the reactor trip breakers respond as designed to appropriate trip signals.

- The reactor coolant system is at hot no-load temperature and pressure. The reactor coolant boron concentration is such that the shutdown margin requirements of the Technical Specifications are satisfied for the safe shutdown condition.

Initial criticality is achieved in an orderly, controlled fashion by the combination of shutdown and control bank withdrawal and reactor coolant system boron concentration reduction.

During the approach to initial criticality, the response of the source range nuclear instruments is used as an indication of the rate of reactivity addition and the proximity to a critical condition so that criticality is achieved in a controlled, predictable fashion.

Rates for rod withdrawal and boron reduction are specified in such a way that the startup rate is less than one decade per minute.

Following criticality and prior to operation at power levels greater than 5 percent of rated power, physics tests are performed to verify that the operating characteristics of the reactor core are consistent with design predictions. During these tests, values are obtained for the reactivity worth of control and shutdown rod banks, isothermal temperature coefficient, and critical boron concentration for selected rod bank configurations.

Other tests at low power include verification of the response of the nuclear instrumentation system and radiation surveys.

14.2.7.3 Power Ascension

After the operating characteristics of the reactor have been verified by low-power testing, a power ascension program brings the unit to its full rated power level in successive stages. At each successive stage, hold points are provided to evaluate and approve test results prior to proceeding to the next stage. The minimum test requirements for each successive stage of power ascension are specified in the applicable startup test procedures.

During the power ascension program, tests are performed at various power levels as follows:

- Statepoint data, including secondary system heat balance measurements, are obtained at various power levels up to full licensed power. This information is used to project plant performance during power escalation, provide calibration data for the various plant control and protection systems, and provide the bases for plant trip setpoints.

- At prescribed power levels, the dynamic response characteristics of the primary and secondary systems are evaluated. System response characteristics are measured for design step load changes, rapid load reductions, and plant trips.

- Adequacy of the radiation shielding is verified by gamma and neutron radiation surveys. Periodic sampling is performed to verify the chemical and radiochemical analysis of the reactor coolant.

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- Using the incore instrumentation as appropriate, the power distribution of the reactor core is measured to verify consistency with design predictions and Technical Specifications limits on peaking factors.

14.2.8 Initial Test Program Schedule

The schedule for the initial fuel load and for each major phase of the initial test program includes the timetable for generation, review, and approval of procedures as well as the actual testing and analysis of results.

Preoperational testing is performed as system and equipment availability allows. The interdependence of systems is also considered.

Sequencing of the startup tests depends on specified power and flow conditions and intersystem prerequisites. The initial test schedule establishes that, prior to core load, the preoperational test requirements identified in Subsection 14.2.9 are met for those plant structures, systems, and components that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. Testing is sequenced so that the safety of the plant is not dependent on untested systems, components, or features.

A site-specific initial test program schedule will be provided to the NRC after issuance of the COL. This schedule will address each major phase of the test program (including tests that are required to be completed before fuel load), as well as the organizational impact of any overlap of first unit initial testing with initial testing of the second unit, if applicable.

The sequential schedule for individual startup tests should establish that testing is completed in accordance with plant technical specification requirements for structures, systems and components (SSC) operability before changing plant modes. Additionally, the schedule establishes that the safety of the plant is not dependent on the performance of untested SSCs. Guidance provided in Regulatory Guide 1.68 is used for development of the schedule.

The ITP Administrative Manual shall include the following controls:

- Test Procedure Development Schedule:
  - Controls to establish a schedule for the development of detailed testing procedures. Startup procedures, to the extent practical, are trial-tested and corrected during the initial test program prior to fuel loading in order to establish their adequacy.
  - Controls to confirm that approved preoperational test procedures are in a form suitable for review by NRC inspectors in accordance with Subsection 14.2.3, or at least 60 days prior to fuel loading for fuel loading and startup test procedures.
  - Controls to provide timely notification to the NRC of changes in approved preoperational and startup test procedures previously available for NRC review.
- Initial Test Program Schedule:
  - Controls to establish a schedule to conduct the major phases of the initial test program, relative to the expected fuel loading date. This is covered in License Conditions in Part 10 of the COL Application.
  - Controls to allow at least 9 months for conducting preoperational testing.

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– Controls to allow at least 3 months for conducting startup testing, including fuel loading, low-power tests, and power-ascension tests.

– Controls to overlap test program schedules (for multi-unit sites) such that they do not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.

– Controls to sequence the schedule for individual startup tests, insofar as is practicable, such that testing is completed prior to exceeding 25 percent power for the plant SSCs that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. The schedule should establish that, insofar as is practicable, testing is accomplished as early in the test program as is feasible and that the safety of the plant is not dependent on the performance of untested SSCs.

The milestone schedule for developing plant operating procedures is presented in Table 13.4-201. The operating and emergency procedures are available prior to start of licensed operator training and, therefore, are available for use during the ITP. Required or desired procedure changes may be identified during their use. Administrative procedures describe the process for revising plant operating procedures.

14.2.9 Preoperational Test Descriptions

During preoperational testing, it may be necessary to return system control to Construction organization to repair or modify the system or to correct new problems. Administrative procedures include direction for:

- Means of releasing control of systems and or components to construction.
- Methods used for documenting actual work performed and determining impact on testing.
- Identification of required testing to restore the system to operability/functionality/availability status, and to identify tests to be re-performed based on the impact of the work performed.
- Verifying retests stay in compliance with ITAAC.

Test abstracts are provided for testing of systems/components in accordance with Regulatory Guide 1.68, Revision 2, Appendix A. Test abstracts for preoperational tests contain a section which identifies the test method, acceptance criteria, and functional criteria. Certain criteria in this section may be required to be completed prior to testing; these criteria are considered prerequisites in accordance with Regulatory Guide 1.68. Acceptance criteria are those criteria used to evaluate whether SSCs can successfully perform their safety-related or defense-in-depth functions. All other criteria, listed in this section, are considered functional criteria.

Test procedures, which contain a greater level of detail than test abstracts, will distinguish between acceptance criteria and functional criteria as well as clearly identify the prerequisites required for completion prior to performance of each test.

A limited number of these testing abstracts establish performance parameters of AP1000 design features that will not change from plant to plant.

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14.2.9.1 Preoperational Tests of Systems with Safety-Related Functions

14.2.9.1.1 Reactor Coolant System Testing

Purpose

The purpose of the reactor coolant system testing is to verify that the as-installed reactor coolant system properly performs the following safety-related functions:

- Provide reactor coolant system pressure boundary integrity as described in Section 5.2
- Provide core cooling and boration in conjunction with the passive core cooling system as described in Sections 5.1 and 6.3
- Measure process parameters required for safety-related actuations and safe shutdown as described in Sections 7.2, 7.3 and 7.4
- Measure selected process parameters required for post-accident monitoring as described in Section 7.5
- Vent the reactor vessel head as discussed in Subsection 5.4.12

Testing is also performed to verify that the system properly performs the following defense-in-depth functions described in Section 5.2:

- Provide forced circulation cooling of the reactor core in conjunction with heat removal by the steam generator(s) as described in Section 5.1
- In conjunction with the steam generator(s) and normal residual heat removal system, provide the capability to remove core decay heat and cool the reactor coolant to permit the reactor to be refueled and started up in a controlled manner
- Provide pressurizer pressure control during normal operation
- Provide pressurizer level control in conjunction with the chemical and volume control system
- Provide pressurizer spray

Prerequisites

The component testing of the reactor coolant system has been successfully completed. The pre-operational testing of the component cooling water system, service water system, chemical and volume control system, main ac power electrical power system, and required interfacing systems is completed to the extent sufficient to support the specified testing. The reactor coolant system is filled, vented, and pressurized above the minimum required pressure for reactor coolant pump operation, and component cooling water flow to the reactor coolant pumps is initiated prior to starting the pumps.

In preparation for the hydrostatic test of the reactor coolant system, the reactor vessel lower and upper internals and the closure head are installed. The closure head studs are properly tensioned for the hydrostatic test pressure. The pressurizer safety valves and instrumentation within the test boundary are either removed, recalibrated or verified to be able to withstand the hydrostatic test pressure. Welds within the test boundaries are verified as ready for hydrostatic testing. A hydrostatic test pump is available for the pressure boundary integrity testing.

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General Test Method and Acceptance and Functional Criteria

Reactor coolant system performance is observed and recorded during a series of individual component and system tests. The following testing demonstrates that the reactor coolant system can perform the functions described above and in appropriate design specifications:

a) The integrity and leaktightness of the reactor coolant system and the high-pressure portions of associated systems is verified by performing a cold hydrostatic pressure test in conformance with Section III of the American Society of Mechanical Engineers (ASME) Code. The reactor coolant system is pressurized in stages by operation of the temporary hydrostatic test pump, while monitoring system welds, piping, and components for leaks at each stage. The hydrostatic test verifies that there are no leaks at welds or piping within the test boundaries during the final inspection. Any identified pressure boundary leaks (i.e. piping walls, vessel walls, welds, valve bodies, etc.) are repaired and the hydrostatic test repeated.

   Leakage through valve seats, valve packing, flanges, and threaded or mechanical fittings is acceptable during the hydrostatic test as long as the hydrostatic test pump can maintain the proper test pressure. Leakage through these items may, as necessary and practical, be isolated, repaired, and retested at a later date.

b) Proper operation of the safety-related reactor coolant system and reactor coolant pressure boundary valves is verified by the performance of baseline in-service tests as described in Subsection 3.9.6.

c) The operability of the pressurizer safety valves is demonstrated by a bench test at temperature and pressure with steam as the pressurizing fluid or with a suitable in-situ test. This testing verifies that each pressurizer safety valve actuates at the required set pressure, with appropriate tolerance as specified in the Technical Specifications. The safety valve rated capacity, as recorded on the valve vendor code plates, is verified to be greater than or equal to that described in Section 5.4.

d) During hot functional testing, reactor coolant system leakage is verified to be within the limits specified in the Technical Specifications. Proper calibration and operation of instrumentation, controls, actuations, and interlocks related to reactor coolant system leak detection are verified. The pressurizer water level is set to the no-load level, the chemical and volume control system makeup pumps and letdown line do not operate, and no primary system samples are taken. During this test, the identified and unidentified reactor coolant system leakage rates are determined by monitoring the reactor coolant system water inventory, reactor coolant drain tank level, containment sump level, and other leak detection instrumentation as described in Subsection 5.2.5 over a specified period of time.

e) The leakage across individual valves between high pressure and low pressure systems, as specified in the Technical Specifications, is verified to be less than design requirements.

f) The as-installed safety valve discharge chamber rupture disks are inspected to verify the manufacturer's stamped set pressure is within the limits specified in the appropriate design specifications.

g) Proper calibration and operation of safety-related instrumentation, controls, actuation signals and interlocks are verified. This testing includes the following:

   - Hot leg and cold leg resistance temperature detectors
   - Flow instrumentation at selected locations in the reactor coolant loop
   - Reactor coolant system wide range pressure transmitters
   - Hot leg level instruments

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- Pressurizer pressure and level instruments
- Reactor coolant pump bearing water temperature detectors
- Reactor coolant pump speed sensor instruments
- Reactor vessel head vent valve controls

This testing includes demonstration of proper actuation of safety-related functions from the main control room.

h) Automatic trip of the reactor coolant pumps following appropriate safety-related actuation signals is demonstrated.

i) Proper operation of the reactor vessel head vent valves is verified with the reactor coolant system pressurized.

The following testing demonstrates that the system properly performs the defense-in-depth functions described above and in appropriate design specifications:

j) The pressurizer spray valves are verified to operate properly over the range of reactor coolant system operating temperatures and with the reactor coolant pumps operating.

k) Proper calibration and operation of defense-in-depth related instrumentation, controls, actuation signals and interlocks are verified. This testing includes actuation of the pressurizer spray valves on receipt of appropriate signals, as well as actuation from the main control room.

l) Reactor coolant pump and motor performance and operating characteristics are initially verified with the reactor coolant system at cold conditions. This testing includes verification of the proper flow through the reactor coolant system when all four reactor coolant pumps are operated in various combinations and speeds as specified in the appropriate design specifications and operating procedures. In addition, the proper operation of the pump motor instrumentation, alarms, and interlocks is verified including:

- Motor current
- Motor power
- Pump vibration
- Motor Stator temperature
- Proper transfer from variable speed startup operation

m) The reactor coolant system is heated from cold conditions to hot standby conditions by operating the reactor coolant pumps and the pressurizer heaters. The reactor coolant system is operated at conditions representative of full flow for at least 240 hours prior to core loading. The reactor coolant temperature is maintained at or above 515°F for at least one-half of this operating time. In addition to facilitating the reactor coolant system tests that are required to be performed hot and pressurized, these hot functional testing conditions allow the plant operators to control the plant using the plant operating procedures for the reactor coolant system, secondary side systems, and auxiliary systems.

Other preoperational tests that require these hot and/or dynamic conditions are conducted during this hot functional testing period.

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n) During hot functional testing, the reactor coolant pump and motor operating characteristics are measured and recorded at various temperature plateaus during reactor coolant system heatup to verify proper operation over their operating temperature range. This testing includes verification of the proper pump flow; proper motor current, power, and stator temperature; and pump vibration level.

o) The pressurizer spray continuous flow rate is established, and the proper spray line temperature is verified for each pressurizer spray line.

p) The proper operation of the pressurizer heaters, pressurizer spray, and pressure control functions and alarms is verified during the heatup, operation at hot functional test conditions, and cooldown of the reactor coolant system.

q) The proper operation of the pressurizer level control functions and alarms is verified during the heatup, operation at hot functional test conditions, and cooldown of the reactor coolant system.

r) The pressure drops across the major components of the reactor coolant system are measured and recorded using temporary instrumentation during flow testing, and verified to be in accordance with appropriate design specifications.

Tests associated with the automatic depressurization functions of reactor coolant system components are described in Subsection 14.2.9.1.3.

14.2.9.1.2 Steam Generator System Testing

Purpose

The purpose of the steam generator system testing is to verify that the as-installed components properly perform the following safety-related functions as described in Sections 5.4, 10.3 and 10.4:

- Provide steam generator isolation, including isolation of the main steam lines, feedwater lines, and blowdown lines
- Remove heat from the reactor coolant system and provide secondary side overpressure protection
- Measure process parameters required for safety-related actuations as described in Sections 7.2, 7.3, and 7.4
- Measure process parameters required for post-accident monitoring as described in Section 7.5

This testing also verifies that the as-installed components properly perform the following defense-in-depth functions as described in Section 10.4:

- Provide heat removal from the reactor coolant system
- Provide overpressure protection for the steam generators to minimize required actuations of the spring-loaded safety valves
- Measure process parameters and provide actuation signals for the diverse actuation system

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Prerequisites

The component tests of the as-installed system have been completed. The reactor coolant system as well as other systems used in power generation are functional since portions of the steam generator system testing is performed during the plant hot functional tests. Prerequisite testing of required interfacing systems are completed to the extent sufficient to support the specified testing and the appropriate system configuration. Component testing of the portion of the special monitoring system identified as the digital metal impact monitoring system has been completed to the extent necessary to support preoperational testing. Required electrical power supplies are energized and operational.

General Test Method and Acceptance and Functional Criteria

The performance of the steam generator system is observed and recorded during a series of individual component and integrated system testing that characterizes its modes of operation. The following testing demonstrates that the steam generator system operates as specified in Sections 10.3 and 10.4, and appropriate design specifications:

a) Proper operation of the steam generator system safety-related valves is verified by the performance of baseline in-service tests as described in Subsection 3.9.6. In addition, the ability of these valves to perform their safety related functions is verified during hot functional testing with the steam generators at normal operating pressure and temperature. The following valves are tested:

   - Steam line condensate drain control and isolation valves
   - Main steam line isolation valves
   - Main and startup feedwater isolation valves
   - Steam generator blowdown isolation valves
   - Steam generator power-operated relief valves
   - Main steam isolation valve bypass isolation valves
   - Main and startup feedwater control valves

   This testing includes verification of the capability of the steam generator power operated relief valves to provide the required heat removal rate from steam generators/reactor coolant system.

b) Proper operation of safety-related and defense-in-depth instrumentation, controls, actuation signals, and interlocks is verified. This testing includes actuation of equipment from the main control room.

c) The proper operation of the steam generator safety valves is demonstrated in a bench test at temperature and pressure with steam as the pressurizing fluid or with suitable in-situ testing. The safety valve rated capacity recorded on the valve vendor code plates is verified to be greater than or equal to the required relief capacity.

Heat transfer performance of the steam generator system is verified by startup testing of the reactor coolant system described in other sections.

14.2.9.1.3 Passive Core Cooling System Testing

Purpose

The purpose of the passive core cooling system testing is to verify that the as-installed components and their associated piping and valves properly perform the following safety functions, described in Section 6.3:

- Emergency core decay heat removal
- Reactor coolant system emergency makeup and boration

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- Safety injection
- Containment pH control

Prerequisites

The component testing of the passive core cooling system, or of a specific portion of the system to be tested, is successfully completed. The preoperational testing of the reactor coolant system, normal residual heat removal system, chemical and volume control system, the refueling cavity, the Class 1E dc and uninterruptible power supply, the ac electrical power and distribution systems, and other interfacing systems required for operation of the above systems is completed as needed to support the specified testing and system configurations. A source of water, of a quality acceptable for filling the passive core cooling system components and the reactor coolant system, is available.

General Test Method and Acceptance and Functional Criteria

The performance of the passive core cooling system is observed and recorded during a series of individual component testing and testing with the reactor coolant system. The following testing demonstrates that the passive core cooling system operates as described in Section 6.3 and appropriate design specifications.

a) Proper operation of safety-related valves is verified by the performance of baseline in-service tests as described in Subsection 3.9.6. Also, the proper operation of non-safety-related valves is verified including manual valve locking devices. This testing does not include actuation of the squib valves, which is discussed in Item t, below.

b) Proper calibration and operation of safety-related instrumentation, controls, actuation signals, and safety related interlocks as specified in Section 7.6, is verified. This testing includes the following:

   - Passive residual heat removal heat exchanger flow
   - Core makeup tank level
   - In-containment refueling water storage tank level
   - Containment floodup level
   - Core makeup tank inlet/outlet valve controls
   - Passive residual heat removal heat exchanger inlet/outlet valve controls
   - In-containment refueling water storage tank outlet valve controls
   - Containment recirculation valve controls
   - Automatic depressurization valve controls
   - In-containment refueling water storage tank gutter isolation valve controls

   This testing includes demonstration of proper actuation of safety-related functions from the main control room.

c) Proper calibration and operation of instrumentation, controls, and interlocks required to demonstrate readiness of a safety-related component is verified. This testing includes the following:

   - Accumulator pressure and level and alarms
   - Passive residual heat removal heat exchanger temperatures
   - Passive residual heat removal heat exchanger high point vent level
   - Core makeup tank inlet line temperatures
   - Core makeup tank inlet line high point levels
   - Direct vessel injection line temperatures
   - In-containment refueling water storage tank level and temperatures

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d) Proper calibration and operation of temporary instrumentation and data recording devices used in this testing is verified. This testing includes the following:

- CMT level
- CMT flow and balance line temperatures
- PRHR supply line temperatures
- Accumulator wide range level
- In-containment refueling water storage tank and sump-recirculation flow
- ADS piping differential pressure

The passive core cooling system emergency core decay heat removal function is verified by the following testing of the passive residual heat removal heat exchanger.

e) During hot functional testing of the reactor coolant system, the heat exchanger supply and return line piping water temperatures are recorded to verify that natural circulation flow can initiate. The measured supply line temperature is > 400°F and the measured return line temperature is < 220°F.

f) Not used.

g) The proper operation of the passive residual heat removal heat exchanger and its heat transfer capability with forced flow is verified by initiating and operating the heat exchanger with all four reactor coolant pumps running. This testing is performed during hot functional testing with the reactor coolant system at an elevated initial temperature ≥ 350°F. The heat exchanger heat transfer is determined by measuring the heat exchanger flow rate and its inlet and outlet temperatures while the reactor coolant system is cooled to ≤ 250°F. The acceptance criteria for the PRHR HX heat transfer under forced circulation conditions are listed in Table 3.9-17. The heat transfer rate and flow resistance measured in the test should be adjusted to account for differences in the hot leg and IRWST temperatures and number of tubes plugged. The line resistance of the PRHR subsystem is measured. The flow rate and the differential pressure across the PRHR subsystem are used to determine the resistance of the PRHR subsystem. The acceptance criterion for the resistance of the line is ≤ 3.085E-06 ft/gpm^2.

h) Not used.

The passive core cooling system emergency makeup and boration function is verified by the following testing of the core makeup tanks.

i) The resistance of the core makeup tank cold leg balance lines is determined by filling the core makeup tanks with flow from the cold legs. This testing is performed by filling the cold, depressurized reactor coolant system using a constant, measured discharge flow from the normal residual heat removal pumps. The reactor coolant system is maintained at a constant level above the top of the cold leg balance line(s). The normal residual heat removal system flow rate and the differential pressure across the cold leg balance lines are used to determine the resistance of the balance lines. The acceptance criterion for the resistance of these lines is ≤ 7.21 x 10^{-6} ft/gpm^2.

j) During hot functional testing of the reactor coolant system, the core makeup tank cold leg balance line piping water temperature at various locations is recorded to verify that the water in this line is sufficiently heated to initiate recirculation flow through the CMTs.

k) Not used.

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The passive core cooling system safety injection function is verified by the following testing of the core makeup tanks, accumulators, in-containment refueling water storage tank, containment sump, automatic depressurization, and their associated piping and valves.

l) Proper flow resistance of each of the core makeup tank injection lines is verified by gravity draining each tank filled with cold water through the direct vessel injection flow path, while measuring the CMT level (driving head) and discharge flow rate. Air enters the top of the draining tank from the reactor coolant system cold leg via the cold leg balance line. If necessary, the flow limiting orifice in the core makeup tank discharge line is to be resized, and the core makeup tank retested to obtain the required line resistance. The acceptance criteria for the resistance of these lines are $\leq 2.25 \times 10^{-5}$ ft/gpm^2 and $\geq 1.81 \times 10^{-5}$ ft/gpm^2 with all valves open.

m) The proper flow resistance of each of the accumulator injection lines is verified by performing a blowdown from a partially pressurized accumulator through the direct vessel injection flow path, while measuring the change in accumulator level and pressure. If necessary, the flow orifice in the accumulator discharge line is to be resized and the accumulator retested to obtain the required discharge line resistance. The acceptance criteria for the resistance of these lines are $\leq 1.83 \times 10^{-5}$ ft/gpm^2 and $\geq 1.47 \times 10^{-5}$ ft/gpm^2.

n) The proper flow resistance of each of the in-containment refueling water storage tank injection lines is verified by gravity draining water from the tank through the direct vessel injection flow path, while measuring the water level (driving head) and discharge flow rate using temporary instrumentation. A test fixture with prototypical resistance may be used to simulate the squib valves in the flow paths tested. The acceptance criteria for the resistance of these lines are $\leq 9.09 \times 10^{-6}$ ft/gpm^2 and $\geq 5.35 \times 10^{-6}$ ft/gpm^2 for line A and $\leq 1.05 \times 10^{-5}$ ft/gpm^2 and $\geq 6.15 \times 10^{-6}$ ft/gpm^2 for line B with all valves open.

o) The flow resistance of each of the flow paths from the in-containment refueling water storage tank to each containment sump, and from each containment sump to the reactor is verified by a series of tests. These tests gravity drain water from the in-containment refueling water storage tank to the containment sump, and from the sump through the direct vessel injection flow path, while measuring the storage tank water level (driving head) and injection flow rate using temporary instrumentation. This testing is performed using temporary piping to prevent flooding of the containment. A test fixture with prototypical resistance may be used to simulate the squib valves in the flow paths tested. The acceptance criteria for the resistance of the lines between each containment sump and the reactor are $\leq 1.33 \times 10^{-5}$ ft/gpm^2 for line A and $\leq 1.21 \times 10^{-5}$ ft/gpm^2 for line B with all valves open. The acceptance criterion for the resistance of the lines between the IRWST and each containment sump is $\leq 4.44 \times 10^{-6}$ ft/gpm^2.

p) The resistance of each automatic depressurization stage 1, 2, and 3 flow path and flow path combination is verified by pumping cold water from the in-containment refueling water storage tank into the cold, depressurized, water-filled reactor coolant system; and back to the in-containment refueling water storage tank using the normal residual heat removal pump(s). The resistances are determined by measuring the residual heat removal pump flow rate and the pressure drop across the flow paths tested using temporary instrumentation. The acceptance criteria for the resistance of these lines is $\leq 2.91 \times 10^{-6}$ ft/gpm^2 for each ADS stage 1, 2, 3 group with all valves open.

q) The resistance of each automatic depressurization stage 4 flow path is verified by pumping cold water from the in-containment refueling water storage tank into the cold, depressurized, water-filled reactor coolant system using the normal residual heat removal pump(s). The resistances are determined by measuring the residual heat removal pump flow rate and the

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pressure drop across the flow paths tested using temporary instrumentation. A test fixture with prototypical resistance may be used to simulate the squib valves in the flow paths tested. The acceptance criteria for the resistance of these lines are:

- Loop 1, sub-loop A: ≤ 5.91×10^{-7} ft/gpm^2
- Loop 1, sub-loop C: ≤ 6.21×10^{-7} ft/gpm^2
- Loop 2, sub-loop B: ≤ 4.65×10^{-7} ft/gpm^2
- Loop 2, sub-loop D: ≤ 6.20×10^{-7} ft/gpm^2

r) The proper operation of the vacuum breakers in the automatic depressurization discharge lines is verified.

s) Not used.

t) The proper operation of at least one of each squib valve size and type including a containment recirculation, in-containment refueling water storage tank injection, and a stage 4 automatic depressurization squib valve is demonstrated. Squib valve functionality is validated via squib valve lot acceptance testing. Flow through the actuated squib valves is validated via the squib valve qualification program. This test does not have to be performed in the plant.

u) The proper operation of the containment sump instrumentation is demonstrated by simulating the containment flood-up water levels.

v) The proper operation of the CMT level instrumentation is demonstrated during the draindown testing of the CMTs, specified in Item l) above.

w) Not used.

14.2.9.1.4 Passive Containment Cooling System Testing

Purpose

The purpose of the passive containment cooling system testing is to verify that the as-installed components perform properly to accomplish their safety-related functions to transfer heat from inside the containment to the environment, as described in Subsection 6.2.2. The passive containment cooling water storage tank also provides a safety-related source of makeup water for the spent fuel pool, and provides a seismically qualified source of water for the fire protection system. Testing of these functions are discussed in Subsections 14.2.9.2.7 Spent Fuel Pool Cooling System Testing, and 14.2.9.2.8 Fire Protection System Testing.

Prerequisites

The component testing of the passive containment cooling system is successfully completed. The preoperational testing of the Class 1E dc electrical power and uninterruptible power supply systems, the non-Class 1E electrical power supply system, the compressed and instrument air system, and other interfacing systems required for operation of the above systems is available as needed to support the specified testing and system configurations. Additionally, a sufficient quantity of acceptable quality water for filling the passive containment cooling water storage tank and draining onto the containment is available, and a means of filling the tank is available.

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General Test Method and Acceptance and Functional Criteria

Passive containment cooling system performance is observed and recorded during a series of individual component testing that characterizes passive containment cooling system operation. The following testing demonstrates that the passive containment cooling system operates as described in Section 6.2 and appropriate design specifications:

a) Proper operation of safety-related valves is verified by the performance of baseline in-service tests as described in Subsection 3.9.6.

b) Proper calibration and operation of safety-related, defense-in-depth, and system readiness instrumentation, controls, actuation signals and interlocks as discussed in Sections 7.3 and 7.5 are verified. This testing includes the following:

- Normal range containment pressure
- High range containment pressure
- Passive containment cooling water flow rate
- Passive containment cooling water storage tank level
- Passive containment cooling water isolation valve instrumentation and controls
- Diverse actuation system passive containment cooling initiation
- Passive containment cooling water storage tank water temperature
- Air inlet and shield plate freeze protection heater controls

This testing includes demonstration of proper actuation of these functions from the main control room.

c) Flow testing is performed to demonstrate proper system flow rates by draining the passive containment cooling system water storage tank. This testing demonstrates the proper resistance of the four passive containment cooling water storage tank delivery flow paths. This testing also demonstrates that water is supplied at the specified flow rates and times for 72 hours consistent with the design basis analyses presented in Subsection 6.2.1.

d) The proper operation of the passive containment cooling water distribution bucket and weirs is verified and proper wetting of the containment is observed and recorded during draindown testing in Item c, above. Water delivery and coverage is verified at the initial minimum water level and as each of the first two standpipes is uncovered. Water coverage is measured at any elevation between elevation 266 ft. and the spring line and the base of the upper annulus as described in Subsection 6.2.2.4.2.

e) The proper operation of the drains in the upper containment/shield building annulus to drain the containment cooling water from the annulus floor is verified.

f) Not used.

g) Sample coupons from the containment shell with and without an appropriate coating of paint are laboratory tested to determine their conductivity.

h) The proper operation of each of the passive containment cooling water storage tank recirculation/makeup pumps to makeup sufficient water to the passive containment cooling water storage tank from the passive containment cooling ancillary water storage tank is verified.

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14.2.9.1.5 Chemical and Volume Control System Isolation Testing

Purpose

The purpose of the chemical and volume control system isolation testing is to verify that the as-installed components properly perform the following safety-related isolation functions, described in Section 9.3:

- Termination of inadvertent dilution of the reactor coolant boron concentration
- Isolation of unborated water sources for reactor makeup
- Reactor coolant system pressure boundary isolation
- Isolation/termination of excessive makeup to the reactor

Prerequisites

The component testing of the chemical and volume control system has been successfully completed. The required preoperational testing of appropriate support and interfacing systems is completed. Data collection is available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Performance of the chemical and volume control system isolation functions is observed and recorded during a series of individual component and integrated system testing that characterizes the system isolation modes of operation. The following testing demonstrates that the chemical and volume control system properly performs the safety-related isolations as specified in Section 9.3 and appropriate design specifications:

a) Proper operation of the safety-related valves is verified by the performance of baseline in-service tests as described in Subsection 3.9.6, including:

   - Purification loop isolation valves
   - Letdown isolation valves
   - Demineralized water isolation valves
   - Makeup isolation valves
   - Auxiliary spray isolation valve

b) Proper calibration and operation of safety-related instrumentation, controls, actuation signals and interlocks is verified. This testing includes the following:

   - Purification isolation valve controls
   - Letdown isolation valve controls
   - Demineralized water isolation controls
   - Makeup isolation valve controls

   This testing includes demonstration of proper actuation of safety-related functions from the main control room.

14.2.9.1.6 Main Control Room Emergency Habitability System Testing

Purpose

The purpose of the main control room emergency habitability system testing is to verify that the as-installed components properly perform the safety-related functions described in Section 6.4, including the following:

- Provide sufficient breathable quality air to the main control room

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- Maintain the main control room at positive pressure
- Provide passive cooling of designated equipment

In addition, the following safety-related functions performed by the nuclear island nonradioactive ventilation system described in Subsection 9.4.1 are tested:

- Provide isolation of the main control room from the surrounding areas and outside environment during a design basis accident if the nuclear island nonradioactive ventilation system becomes inoperable.

- Monitor the radioactivity in the main control room normal air supply and provide signals to isolate the incoming air and actuate the main control room emergency habitability system.

In addition, the following safety-related functions performed by the potable water system, described in Subsection 9.2.5; the sanitary drainage system, described in Subsection 9.2.6; and the waste water system, described in Subsection 9.2.9, are tested:

- Provide isolation of the main control room from the surrounding areas and outside environment during a design basis accident.

Prerequisites

The component testing of the main control room habitability system has been successfully completed. The required preoperational testing of the compressed and instrument air system, Class 1E electrical power and uninterruptible power supply systems, normal control room ventilation system, and other interfacing systems required for operation of the above systems is available as needed to support the specified testing and system configurations. The main control room air supply tanks are filled with air acceptable for breathing. The main control room construction is complete and its leak-tight barriers are in place.

General Test Method and Acceptance and Functional Criteria

Performance of the main control room habitability system is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the habitability system operates as specified in Section 6.4 and as specified in the appropriate design specifications:

a) Proper operation of safety-related valves is verified by the performance of baseline in-service tests as described in Subsection 3.9.6.

b) Proper calibration and operation of safety-related and system readiness instrumentation, controls, actuation signals and interlocks is verified. This testing includes the following:

   - Air storage tank pressure
   - Refill line connection pressure
   - Main control room differential pressure
   - Air supply line flow rate
   - Controls for the main control room pressure relief valves
   - Controls for the air supply isolation valves
   - Controls for the main control room air inlet isolation valves
   - Air intake radiation
   - Passive filtration line flow rate
   - Filter performance
   - Sanitary drainage system vent isolation valves

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c) The proper flow rate of emergency air to the main control room is verified, demonstrating proper sizing of each air flow limiting orifice, proper operation of each air supply pressure regulator, and the ability to maintain proper control room air quality. The MCR passive filtration system flow rate and filter performance will also be verified at this time to ensure a filtration flow rate of at least 600 cfm greater than the main control room emergency habitability system makeup design flow rate. This testing demonstrates the control room pollutant concentrations during the first 6 hours of operation. To determine the control room air quality at 72 hours, the CO_2 concentrations can be predicted based on calculations. The other pollutants described in Table 1 and Appendix C, Table 1 of ASHRAE Standard 62-1989 can be predicted by extrapolating their concentrations for the entire 72-hour period.

d) The ability of the emergency air supply to maintain the main control room at the proper positive pressure is demonstrated, verifying proper operation of the main control room pressure relief dampers.

e) The ability of the emergency air supply to limit air inleakage to the main control room is verified by inleakage testing as specified in Subsection 6.4.5.4.

f) The ability to maintain the main control room environment within specified limits for 72 hours (Reference Subsection 6.4.3.2) is verified with a test simulating a loss of the nuclear island nonradioactive ventilation system. This testing demonstrates the control room heatup from 0 to 6 hours with the actual heat loads from the battery powered equipment and personnel specified for this time period (for the MCR [room 12401], there is automatic de-energization of specific nonsafety-related MCR heat loads). The control room temperature versus time versus heat load data are used to verify the analysis basis used to assure that the control room conditions remain within specified limits for the 72 hour time period. Periodic grab samples will be taken of the control room air environment to support analyses to confirm that specified limits would not be exceeded for 72 hours.

g) The ability to maintain temperatures in the protection and safety monitoring system cabinet and emergency switchgear rooms within specified limits for 72 hours (Reference Subsection 6.4.3.2) is verified with a test simulating a loss of the nuclear island nonradioactive ventilation system. This testing demonstrates the room heatup from 0 to 6 hours with the actual heat loads from battery powered equipment. The room temperature versus time versus heat load data are used to verify the analysis basis used to assure that the room temperature will not exceed the specified limit for the 72 hour time period.

14.2.9.1.7 Expansion, Vibration and Dynamic Effects Testing

Purpose

The purpose of the expansion, vibration and dynamic effects testing is to verify that the safety-related, high energy piping and components are properly installed and supported such that expected movement due to thermal expansion during normal heatup and cooldown, and as a result of transients; thermal stratification and thermal cycling; as well as vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems and equipment, as described in Section 3.9.

Prerequisites

The component testing and preoperational testing of the reactor coolant system at cold conditions has been successfully completed. Required portions of the chemical and volume control system, passive core cooling system, normal residual heat removal system, main feedwater system, startup feedwater system, steam generator system, and steam generator blowdown system are operational. Piping and components within the reactor coolant system and steam generator system pressure

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boundaries and their associated supports and restraints have been inspected and determined to be installed as designed. Permanently installed support devices have been verified to be in their expected cold, static positions and temporary restraining devices such as hanger locking pins have been removed. The instrumentation required for this testing is installed.

General Test Method and Acceptance and Functional Criteria

During hot functional testing, verifications that ASME Code Class 1, 2, and 3 high-energy piping system components, piping, support, and restraint deflections are unobstructed and within design basis functional requirements. The systems to be monitored during preoperational vibration and dynamic effects tests include:

- ASME Code, Class 1, 2, and 3 piping
- High-energy piping systems inside seismic Category I structures
- High-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level
- Seismic Category I portions of moderate-energy piping systems located outside the containment

The high-temperature portions of the following systems are considered for inclusion in this test:

- Reactor coolant system
- Chemical and volume control system
- Passive core cooling system
- Steam generator system (including the safety-related portions of main steam system, main and startup feedwater systems, and steam generator blowdown system)
- Normal residual heat removal system

a) Thermal expansion testing during the preoperational testing phase consists of displacement measurements on the above systems during heatup and cooldown of the reactor coolant system and associated systems (including heatup and cooldown of the passive core cooling system). The testing is performed in accordance with Subsection 3.9.2.1.2 and consists of a combination of visual inspections and local and remote displacement measurements. This testing includes the inspection and measurement of deflection data associated with support thermal movements to verify support swing clearance at specified heatup and cooldown intervals; that there is no evidence of blocking of the thermal expansion of any piping or components, other than by installed supports, restraints, and hangers; that spring hanger movements remain within the hot and cold setpoints; that moveable supports do not become fully retracted or extended; and that piping and components return to their approximate baseline cold positions.

b) Vibration testing is performed on safety-related and high-energy system piping and components during both cold and hot conditions to demonstrate that steady-state vibrations are within acceptable limits. See Subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to steady-state vibration. This testing includes visual observation and local and remote monitoring in critical steady-state operating modes. Results are acceptable when visual observations show no signs of excessive vibration and when measured vibration amplitudes are within acceptable levels.

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c) Testing for significant vibrations caused by dynamic effects is conducted during hot functional testing and may be performed as part of other specified preoperational tests. This testing is conducted to verify that stress analyses of safety-related and high-energy system piping under transient conditions are acceptable. See Subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to dynamic effects vibration. These tests are performed to verify that the dynamic effects caused by transients such as pump starts and stops, valve stroking, and significant process flow changes are within expected values. These tests include anticipated normal operating evolutions with system differential temperatures, such as startup, which could induce dynamic effects. Suitable instrumentation is used to monitor for the occurrence of water hammer noise and vibration. Visual inspections are performed to confirm the integrity of system piping and supports.

   Deflection measurements during various plant transients are recorded and compared to acceptance limits and it is confirmed that no effects due to water hammer are detected.

d) Not used.

The main control room habitability system is classified as a high energy system based on the pressure criteria, not temperature. Tests that measure thermal movements are not required. Vibration testing of the high pressure portion of the main control room habitability system is performed during testing of the air delivery rate provided to the control room. See Subsection 14.2.9.1.6 for information on the testing of the main control room habitability system.

14.2.9.1.8 Control Rod Drive System

Purpose

The purpose of the control rod drive system testing is to verify the proper operation of the control rod drive mechanisms, motor-generator sets and system components as described in Subsection 3.9.4 and Section 4.6, and in appropriate design specifications.

Prerequisites

The component tests of the control rod system have been completed. Required interfacing systems, as needed, are completed to the extent sufficient to support the specified testing and the appropriate system configuration. Required electrical power supplies are energized and operational.

For the control rod drive mechanism cooling test, the plant is at or near normal operating temperature and pressure, and hot functional testing is in progress. The integrated head and control rod drive mechanism cooling system are in their normal operational alignment.

For the control rod drive mechanism motor-generator sets tests, a three-phase load bank is available for motor generator set testing under loaded conditions.

General Test Method and Acceptance and Functional Criteria

Performance is observed and recorded during a series of individual component and integrated system tests. The following tests verify that the control rod drive system operates properly:

a) Tests are conducted to verify the current command sequence, timing, and rod speed signals by initiating control rod drive mechanism withdrawal and insertion. Proper operation of the bank overlap unit to control rod bank sequence is verified.

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b) Tests are conducted to verify the adequacy of the integrated head and control rod drive mechanism cooling system for maintaining control rod drive mechanism temperature. This test is conducted by measuring control rod drive mechanism coil resistances and calculating the coil temperatures.

c) Tests are conducted to verify control rod drive mechanism motor-generator set and system component control circuits, including interlock and alarm functions.

d) Tests are conducted to verify generator phasing for parallel generator operation. Operation of the control rod drive mechanism motor generator sets and control system during starting, running, and parallel operations is verified.

14.2.9.1.9 Reactor Vessel Internals Vibration Testing

Purpose

The AP1000 reactor internals testing is part of a comprehensive vibration assessment program performed in accordance with Regulatory Guide 1.20 as discussed in Subsection 3.9.2.4. This verifies the structural integrity of the AP1000 reactor internals with regard to flow-induced vibrations, as part of an internals vibration assessment program. This program includes visual examination of the reactor internals before and after hot functional testing.

The reactor internals are visually inspected before and after the hot functional test to confirm the structural integrity of the reactor internals with regard to flow-induced vibrations. The major features of the reactor internals outlined in Subsection 3.9.2.4 are visually inspected for signs of abnormal wear and structural changes.

Prerequisites

The component testing of the reactor coolant system has been completed. The internals pre-test visual inspection has been completed. The internals are installed in the reactor vessel. The reactor vessel head is installed in preparation for the cold hydrostatic test of the reactor coolant system.

General Test Method and Acceptance and Functional Criteria

Pre-test and post-test visual inspections of the internals are performed to confirm that the stresses and wear on the AP1000 internals, due to flow induced vibration during plant operation, are acceptably low. The criteria for evaluating testing results are established in the AP1000 reactor internals flow-induced vibration assessment program (see Section 7 of WCAP-17894), and appropriate design specifications.

Visual inspections are performed before and after the hot functional test. When no indications of harmful vibrations or signs of abnormal wear are detected and no structural damage or changes are apparent, the core support structures are considered to be structurally adequate and sound for operation. If such indications are detected, further evaluation is required.

14.2.9.1.10 Containment Isolation and Leak Rate Testing

Purpose

The purpose of the containment isolation and leak rate testing is to demonstrate that the as-installed containment isolation valves, piping and electrical containment penetrations, and hatches, and the containment vessel properly perform the following safety functions as described in Section 6.2:

- Automatic isolation of the piping penetrating containment required to assure containment integrity

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- The containment vessel, penetration, and isolation valve leakage is less than the design basis leakage at or near the containment design pressure consistent with 10 CFR 50, Appendix J pressure test requirements.

Prerequisites

The component testing of the containment, containment hatches/airlocks and containment penetrations including the containment pressure test as specified in Subsection 3.8.2.7 has been completed. The component testing of the piping and isolation valves or electrical wiring through the penetrations, has been completed. The instrumentation to be used in performing the Type A, B, and C testing is calibrated and available, including their associated data processing equipment. The required preoperational testing of the protection and safety monitoring system, plant control system, the Class 1E electrical power uninterruptible power supply, and other interfacing systems required for operation of the containment isolation devices and data collection is available.

General Test Method and Acceptance and Functional Criteria

Containment isolation functions, leak rate, and structural integrity performance are observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the containment functions as described in Section 6.2 and the appropriate design specifications are achieved. The testing is in accordance with the Containment System Leakage Testing Program and is discussed in Subsection 6.2.6, which meets the requirements of ANSI/ANS-56.8-1994, as appropriate.

a) Proper operation of safety-related containment isolation valves, listed in Table 6.2.3-1, is verified by the performance of baseline in-service tests as specified in Subsection 3.9.6.

b) Proper calibration and operation of safety-related containment isolation instrumentation, controls, actuation signals and interlocks is verified. This testing includes actuation of the containment isolation valves from the main control room, and upon receipt of a containment isolation signal.

c) The appropriate Type C leakage testing is performed for each piping path penetrating the containment boundary, verifying the leakage for each containment isolation valve (listed in Table 6.2.3-1) or set of isolation valves. This testing for individual isolation valves may be performed in conjunction with the associated system test.

d) The appropriate Type B leakage testing is performed for each containment penetration whose design incorporates seals, gaskets, sealants, or bellows. This testing includes door or hatch operating mechanisms and seals.

e) A general inspection of the accessible interior and exterior surfaces of the containment structure and components is performed as specified in Subsection 6.2.5.2.2.

f) A Type A integrated leak rate test is performed to verify that the actual containment leak rate does not exceed the design basis leak rate specified in the Technical Specifications.

14.2.9.1.11 Containment Hydrogen Control System Testing

Purpose

The purpose of the containment hydrogen control system testing is to verify that the system properly performs the following non-safety defense-in-depth functions described in Section 6.2:

- Prevent the concentration of hydrogen in containment from reaching the flammability limit.
- Prevent the concentration of hydrogen in containment from reaching the detonation limit.

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- Monitor the containment hydrogen concentration as required by Regulatory Guide 1.97.

Prerequisites

The component testing of the containment hydrogen control system is completed. The non-Class 1E dc and uninterruptible power supply system, main ac power system, and other interfacing systems required for operation of the above systems and calibrated data collection instrumentation are available as needed to support the specified testing.

General Test Method and Acceptance and Functional Criteria

Performance of the containment hydrogen control system is observed and recorded during a series of individual component testing. The following testing verifies that the system operates as described in Subsection 6.2.4 and as specified in the appropriate design specifications:

a) Proper operation of the non-Class 1E containment hydrogen concentration instrumentation and alarms is verified.

b) The ability of the passive autocatalytic recombiners to properly respond to a known inlet hydrogen/air mixture is verified by removing and testing one plate or cartridge from each manufacturing lot of catalyst material, contained in each recombiner unit. This verification is performed in accordance with the guidance provided in Subsection 6.2.4.5.1 using a manufacturer's standard test device and test procedure. Plate performance is verified to be consistent with the response obtained in manufacturer's tests.

c) Manual actuation and operation of the hydrogen igniters confirm that the igniters are supplied by two power groups from two subsystems of the non-Class 1E dc and UPS system. Operability of the igniters is confirmed by verification that the igniter surface temperature meets or exceeds the temperature specified in Subsection 6.2.4.

14.2.9.1.12 Protection and Safety Monitoring System Testing

Purpose

The purpose of the protection and safety monitoring system preoperational testing is to verify that the as-installed components properly perform the following safety-related functions, described in Section 7.1:

- Receive and analyze sensor inputs required for reactor trip and automatically initiate reactor trip signals when plant conditions reach the appropriate setpoints
- Provide actuation signals to the engineered safety features to limit the consequences of design basis accidents
- Provide instrumentation and display systems to monitor the safety-related functions of the plant during and following the occurrence of design basis accidents in accordance with Regulatory Guide 1.97

Preoperational testing is also performed to verify proper operation of the following defense-in-depth functions, described in Section 7.1:

- Provide data from the safety-related sensors to the plant control system
- Provide information to the data display and processing system
- Provide data to the monitor bus for use by other systems within the plant

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Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and the initial software diagnostics tests have been completed. Required electrical power supplies and control circuits are energized and operational. Plant systems or components which are to be operated during testing are specifically identified in the preoperational test procedures, are properly aligned, and have proper support systems operating prior to actuation of the particular system or component. Equipment or components which can not be actuated without damage or upsetting the plant are isolated using the test switches provided by the Protection and Safety Monitoring System to block device actuation. Continuity of wiring up to the actuation equipment is verified.

General Test Method and Acceptance and Functional Criteria

Performance of the protection and safety monitoring system is observed and recorded during a series of individual component and integrated tests designed to verify operation of the system components. The following testing verifies that the system operates as described in Section 7.1 and appropriate design specifications:

a) Processing of the analog and digital signals is verified by injecting reference signals and verifying the outputs at the associated division safety display and non-Class1E workstation display.

b) Capability to process sensor data and main control room manual inputs resulting in the initiation of appropriate reactor trip signals is demonstrated by simulating inputs for each of the trip functions. Response times are verified by demonstrating that the applicable trip or actuate signal reaches the actuated equipment within the maximum allowable period following a defined step change in the applicable simulated input, above or below the trip or actuate setpoint. Operation of the protection cabinet trip/normal/bypass switches and indicators for each of the reactor trip functions is demonstrated by verifying appropriate outputs. Verification that the reactor trip bypass logic satisfies the single failure criteria is demonstrated by operating the bypass switches while simulating channel failures. Proper operation of the reactor trip reset function which is a nonsafety function of the Plant Control System (PLS) will be verified.

c) Operation of the reactor trip breakers, including breaker interlock, alarm, and tripping functions and verification that reactor trip response times are less than the specified maximum allowable response times is performed by initiating a manual reactor trip from the main control room. The capability of the undervoltage coil and the shunt trip coil functions to independently trip the reactor trip breakers is verified during this test using the test capabilities provided by the reactor trip switchgear interface.

d) The capability to trip the reactor from the remote shutdown workstation is demonstrated by verifying actuation of the reactor trip breaker undervoltage and shunt trip attachments upon initiation of a reactor trip at the remote shutdown workstation location.

e) The capability of the protection and safety monitoring system to process sensor data and manual inputs, resulting in appropriate engineered safety features actuation at design setpoints, is demonstrated by verifying that injection of simulated inputs for each of the engineered safety features actuation functions results in the proper output as indicated by contact operation, component actuation, or electrical test. Response times associated with the engineered safety features actuation functions are evaluated during these tests to provide verification that the applicable trip or actuate signal reaches the actuated equipment within the maximum allowable period following a defined step change in the applicable simulated

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input above or below the trip or actuate setpoint. Operation of the manual actuation/bypass switches and indicators for each of the engineered safety features functions is verified by demonstrating appropriate system outputs. Verification that the engineered safety features bypass logic satisfies the single failure criteria is demonstrated by operating the bypass switches while simulating channel failures. Correct input processing and calculational accuracy of the redundant actuation equipment and operator interface features is verified for each defined engineered safety features actuation function using simulated inputs. Proper operation of the engineered safety features reset functions will be verified.

f) Correct processing of inputs by redundant equipment and operation of the processing, permissive, interlock, display and operator interface features is verified by demonstrating that simulated command inputs result in correct output or actuation functions as indicated by contact operation, component actuation, or electrical test.

g) Accurate processing of component-level manual actuation commands from the main control room to the protection logic cabinets is verified by issuing main control room commands. Processing of component status information is demonstrated by monitoring protection logic cabinet outputs at the associated division safety display and on a non-Class 1E workstation display.

h) Processing of component-level actuation commands from the remote shutdown workstation to the protection logic cabinets is verified by actuating remote shutdown workstation commands. Processing of component status information is verified by monitoring protection logic cabinet outputs on the non-Class 1E remote shutdown workstation display.

i) Operation of the automatic testing features provided in the protection and safety monitoring system is verified by observing the automatic test functions while simulating component failures and utilizing man-machine interface capabilities to evaluate system performance.

j) The capability of the protection and safety monitoring system to provide the plant operator with correct equipment status, component position indication, component control modes and abnormal operating conditions is verified by evaluating system response to simulated inputs representing feedback from actuation devices and position indicators. Communication of information via the plant monitor bus/data display and processing system, such as channel input quality, neutron flux detector high voltage, partial trip/actuation, permissive, interlock, block, reset, bypass, automatic test, reactor trip switchgear and system level actuation status, from the protection and safety monitoring system to external systems is verified by evaluating system response to injected reference signals and operating applicable block and bypass controls.

k) Operation of the qualified data processing equipment is verified by monitoring outputs and qualified display indications generated in response to simulated inputs representing data from the integrated protection cabinets and sensor inputs to the qualified data processing I/O cabinets.

l) Operation of the isolated data links and data highways used for communication between the engineered safety features actuation cabinets and protection logic cabinets is verified.

m) Preoperational testing of plant sensors used to provide data related to plant equipment monitored by the protection and safety monitoring system is performed in conjunction with testing of the respective systems in which these sensors are located.

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n) The capability of the protection and safety monitoring system to provide data from the safety-related sensors to the plant control system is verified by injecting reference signals into the integrated protection cabinets and monitoring the plant control system display.

14.2.9.1.13 Incore Instrumentation System Testing

Purpose

The purpose of the incore instrumentation system preoperational testing is to verify that the as-installed components properly perform the following safety-related functions, described in Section 7.1:

- Provide reactor coolant system pressure boundary integrity for the incore instrumentation thimble assemblies which penetrate the upper head of the reactor vessel
- Provide the protection and safety monitoring system with the core exit temperature signals required for post-accident monitoring

Testing is also performed to verify the following nonsafety-related defense-in-depth functions, described in Subsection 4.4.6:

- Provide core exit temperature signals to the diverse actuation system dedicated display in the main control room

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Required electrical power supplies are energized and operational.

General Test Method and Acceptance and Functional Criteria

Performance of the incore instrumentation system is observed and recorded during a series of individual component and integrated tests designed to confirm operation of the system components outside the reactor vessel. The following testing verifies that the system operates as described in Section 7.1 and the appropriate design specifications:

a) Reactor coolant system pressure boundary integrity at the incore instrumentation reactor vessel head penetrations is verified during hydrostatic testing of the reactor coolant system.

b) Processing of the incore thermocouple signals is verified by thermocouple signals at the incore instrumentation thimble assembly connectors and verifying the thermocouple signal paths.

14.2.9.1.14 Class 1E DC Power and Uninterruptible Power Supply Testing

Purpose

The purpose of the Class 1E dc power and uninterruptible power supply testing is to verify that the as-installed components properly perform the following safety-related functions described in Section 8.3:

- Provide the electrical power required for the operation of the plant safety-related equipment, equipment controls, and instrumentation

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- Provide the required safety-related electrical power for at least 72 hours following a design basis event, independent of both offsite and onsite ac electrical power supplies

- Provide separation and independence of Class 1E power divisions from other Class 1E divisions and non-Class 1E systems

Testing is also performed to verify proper operation of the following defense-in-depth functions described in Subsection 8.3.2:

- The capability to recharge the batteries from the onsite or offsite ac electrical sources is verified so that safety-related functions can be supported for an indefinite time

Prerequisites

The component testing of the Class 1E dc power and uninterruptible power supply components has been completed. The necessary permanently installed and test instrumentation is calibrated and operational. The 480V ac electrical power system is in operation and supplying power to the battery chargers and regulating transformers. A test load is available for the performance of battery capacity tests.

General Test Method and Acceptance and Functional Criteria

Performance of the Class 1E dc power and uninterruptible power supply is observed and recorded during a series of individual component and integrated system tests that characterize the operation of the system. The following testing verifies that this system operates as described in Section 8.3 and appropriate design specifications:

a) The capability of each of the seven Class 1E batteries to provide the required momentary and continuous load is verified by a battery service test performed in accordance with IEEE Standard 450. Following this discharge testing, the voltage of each cell is verified to be greater than or equal to the specified minimum cell voltage.

b) The capacity of each of the seven Class 1E batteries is verified to meet or exceed the required ampere-hour rating by a battery performance test performed in accordance with IEEE Standard 450. Following this discharge testing, the voltage of each cell is verified to be greater than or equal to the specified minimum cell voltage.

c) The capability of each of the seven battery chargers to charge its associated battery at the required rate is verified. This testing includes verification that the individual voltage of each cell is within the specified limits for a charged battery.

d) The capability of each of the six inverters to provide the required output current, frequency, and voltage is verified.

e) The capability of each of the four regulating transformers to provide the proper ac current to the Class 1E ac distribution panels is verified.

f) The capability of each of the static transfer switches to automatically transfer the electrical loads supplied by each inverter to its associated regulating transformer is verified.

g) The separation and independence of each redundant division of the Class 1E dc power and uninterruptible power supply is verified by successively powering only one division at a time and verifying power to the proper loads and the absence of voltage at the bus and loads not under test.

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h) The proper calibration and operation of instrumentation and alarms, electrical ground detection, and permissive and prohibitive interlocks is verified.

Testing the Class 1E DC and uninterruptible power supply system battery monitors is not required during the initial test program.

14.2.9.1.15 Fuel Handling and Reactor Component Servicing Equipment Test

Purpose

To verify proper operation of the fuel-handling and reactor component servicing equipment as described in Section 9.1. This includes the refueling machine, fuel handling machine, fuel transfer system, and refueling tools used to lift, transport, or otherwise manipulate fuel, control rods and other incore instruments.

Prerequisites

The component tests have been completed. Prerequisites of the required interfacing systems are completed to the extent sufficient to support the specified testing. Required electrical power supplies are energized and operational. Compressed air, as required for tool operation, is available. The reactor vessel head has been removed, the reactor vessel and refueling cavity are drained, and the area in which the refueling machine moves is free of structures or components that could interfere with fuel handling operations.

The spent fuel pool and fuel transfer canal are drained, and the area in which the fuel handling machine moves is free of any structures or components that interfere with design fuel handling operations.

The fuel transfer system is operable and capable of transporting a dummy fuel assembly from the spent fuel pool to containment. A dummy fuel assembly, resembling an actual fuel assembly in weight, envelope, and mating hardware, is available for use. The fuel transfer system and new fuel elevator are operable as required to permit testing of fuel handling machine functions.

General Test Method and Acceptance and Functional Criteria

The following tests are performed to verify the refueling machine operation:

a) The refueling machine is operated to simulate actual refueling operations, using a dummy fuel assembly. This testing includes manual and automatic modes of operation, displays, interlocks, and limits. These tests verify:

- The ability to move a fuel assembly from the fuel transfer system to the reactor vessel and back
- The consistency of measured trolley, bridge, and hoist speeds with each mode of operation
- The operability of interlocks limiting motion, speed, and weight, including interlocks with other plant equipment
- The operability of displays indicating position, mode, alarm status, and load
- The adequacy of indexing (by placing the dummy fuel assembly in selected core locations)

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b) A known weight or a calibrated spring scale is used to calibrate and set the load limits for the refueling machine load cells. A static load test or the manufacturer's test results are used to verify the ability of the refueling machine hoists to support 125 percent of their rated loads.

The following tests are performed to verify the operation of the fuel handling machine:

c) The fuel handling machine is operated to simulate actual refueling operations, using a dummy fuel assembly. These tests verify:

- The ability to transfer fuel assemblies between the new fuel elevator, fuel transfer system, fuel storage racks, and other areas of the pool where fuel is serviced or stored
- The consistency of measured trolley, bridge, and hoist speeds with each mode of operation
- The operability of interlocks limiting motion, speed, and weight, including interlocks with other plant equipment
- The operability of displays indicating position, mode, alarm status, and load

d) The fuel handling machine is operated to verify its capability to transfer fuel between the new fuel elevator, fuel transfer system, fuel storage racks, and other areas of the pool where fuel is serviced or stored.

e) A known weight or a calibrated spring scale is used to calibrate and set the load limits for the fuel handling machine load cells. A static load test or the manufacturer's test results are used to verify the ability of the fuel handling machine hoists to support 125 percent of their rated loads.

The following tests are performed to verify the proper operation of the fuel transfer system and refueling tools:

f) Using appropriate plant operating procedures, the operability of the new fuel elevator is verified. Testing is performed to demonstrate the proper operation of controls, displays, and limit switches, including operation of the interlock that prevents raising the elevator when it contains a fuel assembly.

g) Using appropriate plant operating procedures, the fuel transfer system is operated to simulate actual refueling operations, using a dummy fuel assembly. During these operations, the following items are verified:

- The ability to move fuel assemblies between the fuel building and containment, including proper operation of upenders in both locations
- The operability and setpoints of limit switches and of interlocks between stations and with other plant equipment
- The operability of displays indicating mode of operation and status

h) Tests are performed to verify that the refueling tools operate properly. Included are tools for handling new fuel assemblies, fuel assembly inserts, irradiation specimens, control rod drive shafts, as well as tools for such operations as control rod drive shaft latching and reactor vessel stud tensioning. As applicable, power is applied to each tool to verify proper operation

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of controls, limit switches, actuators, and indicators. Stud tensioning equipment is checked when assembling the reactor for hot functional testing. The new fuel handling tool is tested with the dummy fuel assembly during the test of the new fuel elevator.

14.2.9.1.16 Long-Term Safety-Related System Support Testing

Purpose

The purpose of this testing is verify the capability to perform the following functions for maintaining the extended operation of the safety-related systems and components as described in Section 1.9:

- Supply makeup water to the passive containment cooling system.
- Supply makeup water to the spent fuel pool.
- Provide electrical power for post-accident instrumentation, control room lighting and ventilation, division B and C I&C room ventilation, passive containment cooling system pumps, ancillary generator room lights, ancillary generator tank heaters.
- Provide ventilation cooling to the main control room.
- Provide ventilation cooling to the Class 1E cabinets for post-accident instrumentation.

Prerequisites

The component tests of the safety-related systems and/or components designed for long-term actions have been successfully completed. The preoperational testing of these systems and/or components, including instrument calibrations, has been completed as required for the specified testing, system configurations, and operations. Equipment required for data collection is available and operable. Water used in this testing should be of a quality suitable for filling the specified components. Equipment used to provide the required long-term actions is available.

General Test Method and Acceptance and Functional Criteria

The ability to perform the required long-term actions is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the long-term actions can be performed as discussed in Section 1.9 and as specified in appropriate design specifications:

a) The ability to provide makeup water to the passive containment cooling water storage tank as described in Subsection 6.2.2 is verified.

b) The ability to provide electrical power to the post-accident monitoring instrumentation, control room lighting and ventilation, division B and C I&C room ventilation, passive containment cooling system pumps, ancillary generator room lights, ancillary generator tank heaters, using the ancillary diesel generators as described in Section 8.3 is verified.

c) The ability to provide main control room ventilation cooling using ancillary fans as described in Subsection 9.4.1 is verified.

d) The ability to provide ventilation cooling to post-accident monitoring instrumentation equipment rooms using ancillary fans as described in Subsection 9.4.1 is verified.

e) The ability to provide makeup water to the spent fuel pool via the safety-related makeup connection from the passive containment cooling system water storage tank, as described in Subsection 9.1.3, is verified.

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14.2.9.2 Preoperational Testing of Defense-in-Depth Systems

14.2.9.2.1 Main Steam System Testing

Purpose

The purpose of the main steam system testing is to verify that the as-installed system properly performs the following functions, as described in Sections 10.2, 10.3 and 10.4 and appropriate design specifications:

- Provide backup isolation of the steam lines as credited in the safety analysis by closure of the turbine bypass valves, the main steam moisture separator reheater 2nd stage steam isolation valves, and the turbine steam stop and control valves to prevent blowdown of steam from the steam generators following an event where steam line isolation is required.(1)

- Provide backup isolation of the steam lines not credited in the safety analysis by closure of the auxiliary steam supply isolation valve.(2)

- Provide protection of the turbine from water induction and/or overspeed by closure of the extraction steam isolation and non-return valves.(2)

Notes:

1. This is not a defense-in-depth function but is included in this section of the test descriptions since these backup isolation valves are credited in the safety analysis and included in inservice testing which is not typical of Non-Safety Class E structures, systems and components.

2. These are not defense-in-depth functions but are included in this section since the testing of these valves is performed in the same test procedure as the testing of the backup isolation valves credited in the safety analysis.

Prerequisites

The component tests of the as-installed main steam system have been completed. Prerequisites of the required interfacing systems are completed to the extent sufficient to support the specified testing and the appropriate system configuration.

General Test Method and Acceptance and Functional Criteria

Main steam system performance is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the system operates as described in Sections 10.2, 10.3 and 10.4 and appropriate design specifications:

Proper operation of the following system valves is verified.

- Turbine steam stop and control valves
- Turbine bypass valves
- Auxiliary steam system supply header isolation valve
- Main steam moisture separator reheater 2nd stage steam isolation valves
- Extraction steam isolation and non-return valves

This testing includes actuation of these valves from the main control room.

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14.2.9.2.2 Main and Startup Feedwater System

Purpose

The purpose of the main and startup feedwater system testing is to verify that the as-installed system properly performs the following nonsafety-related defense-in-depth function, as described in Subsections 10.4.7 and 10.4.9:

- Provide startup feedwater to the steam generators to remove heat from the reactor coolant system following the loss of normal feedwater

Prerequisites

The component testing of the main and startup feedwater system components and instruments, or specific portion to be tested has been completed. Required interfacing systems are available.

General Test Method and Acceptance and Functional Criteria

The main and startup feedwater system performance is observed and recorded during a series of individual component and integrated system testing. The following defense-in-depth testing demonstrates that the system operates as described in Subsections 10.4.7 and 10.4.9 and appropriate design specifications:

a) Proper operation of defense-in-depth instrumentation, controls, actuation signals and interlocks is verified. This testing includes actuation of startup feedwater pumps and remotely-operated valves from the main control room including isolation of the main feedwater system.

b) The capability of the startup feedwater pumps to operate properly when performing their defense-in-depth function and main feedwater pumps are verified with the steam generator at normal operating pressure. Testing of one Main Feedwater Pump can be performed any time prior to initial criticality during the startup test phase.

c) The capability of the startup feedwater pumps to operate properly with miniflow to the condensate storage tank is verified.

d) The capability to restore normal steam generator water level from the low narrow range water level, without causing unacceptable feedwater or steam generator water hammer, is demonstrated (refer to Subsections 14.2.9.1.7 and 14.2.10.4.18).

14.2.9.2.3 Chemical and Volume Control System Testing

Purpose

The purpose of the chemical and volume control system testing is to verify that the as-installed system properly performs the following defense-in-depth functions described in Subsection 9.3.6 and appropriate design specifications:

- Provide makeup water to the reactor coolant system
- Provide boration of the reactor coolant system
- Provide auxiliary pressurizer spray

Prerequisites

The component testing of the as-installed chemical and volume control system is completed. The following interfacing and support systems are available as necessary to support testing: component cooling water system; service water system; reactor coolant system; electrical power and distribution

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systems. Data collection is available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Chemical and volume control system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies the system properly performs the defense-in-depth functions described in Subsection 9.3.6 and appropriate design specifications:

a) Operation of pumps and valves which perform defense-in-depth functions is verified, including:

- Makeup pumps
- Boric acid mixing control valve
- Makeup flow control valve

b) Calibration and operation of defense-in-depth related instrumentation, controls, actuation signals and interlocks is verified, including:

- Automatic makeup pump actuation and shutoff
- Automatic alignment of the boric acid tank
- Pressurizer auxiliary spray initiation and termination
- Letdown/purification isolation

This testing includes actuation of defense-in-depth pumps and remotely-operated valves from the main control room. Pressurizer level control testing is described in Subsection 14.2.9.1.1.

c) The capability of the makeup pumps to operate when performing their normal makeup and pressurizer spray functions is verified with the reactor coolant system at normal operating pressure.

d) The capability of the makeup pumps to operate at miniflow and the operation of the makeup pump miniflow heat exchanger is verified.

e) The proper purification loop flowrate through the demineralizers and filters is verified.

14.2.9.2.4 Normal Residual Heat Removal System Testing

Purpose

The purpose of the normal residual heat removal system testing is to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the following defense-in-depth functions, as discussed in Section 5.4:

- Remove reactor core decay heat and cool the reactor coolant system during shutdown operations at low pressure and temperature
- Remove reactor core decay heat from the reactor coolant system during reduced reactor coolant inventory operations in Modes 5 and 6
- Following actuation of the automatic depressurization system, provide makeup to the reactor coolant system at low pressure

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- Circulate and cool water from the containment after draindown of the in-containment water storage tank
- Provide low temperature overpressure protection for the reactor coolant system
- Remove reactor core decay heat and cool the spent fuel pool during refueling operations when the core is off-loaded from the reactor vessel to the spent fuel pool.

Prerequisites

The component testing of the normal residual heat removal system is completed. The required preoperational testing of the in-containment refueling water storage tank, reactor coolant system, passive core cooling system, component cooling water system, service water system, ac electrical power and distribution systems, and other interfacing systems required for operation of the above systems and data collection is available as needed to support the specified testing and system configurations. The reactor coolant system and the in-containment refueling water storage tank have an adequate water inventory to support testing.

General Test Method and Acceptance and Functional Criteria

Normal residual heat removal system performance is observed and recorded during a series of individual component and system testing, that characterizes system operation. The following testing verifies that the normal residual heat removal system performs its defense-in-depth functions as described in Subsection 5.4.7.6.1 and appropriate design specifications:

a) Operation of valves to open, to close, or to control flow as required to perform the above defense-in-depth functions is verified.

b) Operation of system controls, alarms, instrumentation, and interlocks associated with performing the above defense-in-depth functions is verified. In addition, the proper operation of the normal residual heat removal system/reactor coolant system isolation valve interlocks specified in Section 7.6 is verified.

c) The normal residual heat removal system pumps testing includes verification that the pump flow rate corresponds to the expected system alignment, proper pump miniflow operation, and verification that adequate net positive suction head is available for the configurations tested. The following system configurations are tested with each pump operating individually and with two pumps operating:

   - Recirculation from and to the reactor coolant system with the reactor coolant system at mid-loop hot leg water level and atmospheric pressure
   - Makeup to the reactor from the in-containment refueling water storage tank with approximately 4 feet of water in the tank
   - Makeup to the reactor from the cask loading pit with water in the pit at a sufficient level to support pump operation
   - Recirculation from and to the spent fuel pool with the pool at normal minimum level.

d) During the verifications of normal residual heat removal system flow to the reactor coolant system, verify that the pumped flow provides sufficient back pressure to maintain a water level in the CMT.

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e) The capability of the normal residual heat removal heat exchangers to provide the required heat removal rate from the reactor coolant system is verified by testing performed with flow from and to the heated reactor coolant system, with each normal residual heat removal pump/heat exchanger operating individually.

f) The capability of the normal residual heat removal heat exchangers to provide the required heat removal rate from the spent fuel pool is verified. Since the spent fuel pool is not heated during pre-operational testing, this verification can be made based on the flowrate from Item c and heat removal capability from Item e, above.

g) Operation of the normal residual heat removal system relief valves which provide low temperature overpressure protection for the reactor coolant system is verified by the performance of baseline in-service testing, as specified in Subsection 3.9.6. The acceptance criteria are based on the valve performance criteria specified in Subsection 5.4.9.

h) Operation of the system to facilitate draining the reactor coolant system water level to near the centerline of the hot leg for reduced inventory operations is verified. This test is performed in conjunction with the chemical and volume control system, and is used to demonstrate the performance of the reactor coolant system hot leg level instruments as discussed in Subsection 14.2.9.1.1.

14.2.9.2.5 Component Cooling Water System Testing

Purpose

The purpose of the component cooling water system testing is to verify that the as-installed system properly performs the following defense-in-depth functions as described in Subsection 9.2.2:

- Provide cooling water to defense-in-depth components and transfer heat to the service water system. In addition, this system provides cooling water to other nonsafety-related components for heat removal.

Prerequisites

The component testing of the component cooling water system is completed. Preoperational testing of the cooled components has been completed as necessary to support testing of the component cooling water system. Required support systems are available, including applicable portions of the service water system and electrical power and distribution systems. Data collection is available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Component cooling water system performance is observed and recorded during a series of individual component and integrated system testing that characterizes the various modes of system operation. The following testing demonstrates that the system operates as described in Subsection 9.2.2 and in appropriate design specifications:

a) Proper operation of the component cooling water pumps is verified.

b) Proper operation of defense-in-depth related instrumentation, controls, actuation signals and interlocks is verified, including:

- Automatic pump actuation if an operating pump stops
- Pump flow rate
- Pump discharge pressure
- Surge tank water level and control

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- Water flow rate to defense-in-depth components

This testing includes actuation of the system pumps and remotely-operated valves from the main control room as appropriate.

c) The capability to provide the expected cooling water flow rates to and from the required components with both pumps operating, and with either individual pump and heat exchanger operating as specified in the appropriate design specifications is verified.

d) In conjunction with Item c above, the pump(s) runout flow rate is verified to be properly limited, and adequate net positive suction head is verified to be available during its operating modes.

e) The capability of the heat exchanger(s) to transfer heat properly to the service water system is verified under simulated plant conditions during plant hot functional testing. Testing conditions assume both pumps/heat exchangers in operation and with either one of the pumps/heat exchangers operating.

14.2.9.2.6 Service Water System Testing

Purpose

The purpose of the service water system testing is to verify the capability of the as-installed system to perform the following defense-in-depth function as described in Subsection 9.2.1:

- Transfer heat from the component cooling water heat exchangers to the environment

Prerequisites

The component testing of the service water system is completed. Preoperational testing of the component cooling water heat exchangers so that they can receive service water has been completed, as well as the electrical power and distribution systems, and other interfacing systems required for operation of the service water system. Data collection is available as needed to support the specified testing and system configurations. The component cooling water system and components it cools are functional and hot preoperational testing of the reactor coolant system is in progress in order to confirm the service water system heat removal and heat rejection capability.

General Test Method and Acceptance and Functional Criteria

Service water system performance is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the service water system properly performs its defense-in-depth functions, as described in Subsection 9.2.1 and appropriate design specifications:

a) Proper operation of the service water pumps, valves, strainers, cooling tower fans, and freeze protection provisions are verified.

b) Proper operation of the defense-in-depth related instrumentation, controls, actuation signals and interlocks is verified, including:

- Automatic pump actuation if an operating pump stops
- Pump flow rate
- Pump discharge pressure
- Cooling tower water level and control
- Cooling tower basin water temperature and control
- Water supply and return temperature

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- Cooling tower fan control

This testing includes actuation of defense-in-depth pumps and remotely-operated valves from the main control room as appropriate.

c) The capability of the pumps to provide the expected cooling flow rates to and from the component cooling water heat exchangers is verified. Testing conditions include both pumps operating and either individual pump operating.

d) In conjunction with Item c above, the pump(s) runout flow rate is verified to be properly limited, and adequate net positive suction head is verified to be available during appropriate operating modes.

e) The heat removal and heat rejection capability of the service water system during the conditions of the plant hot functional testing is verified. Testing conditions include both pumps/cooling towers cells in operation and with either one of the pumps/cooling tower cells operating.

14.2.9.2.7 Spent Fuel Pool Cooling System Testing

Purpose

The purpose of the spent fuel pool cooling system testing is to verify that the system properly performs the following defense-in-depth function described in Subsection 9.1.3:

- Remove heat from the spent fuel stored in the spent fuel pool.
- Prevention of back flow through refueling canal drain lines when other in-containment compartments have been flooded is also verified.

Prerequisites

The component testing of the spent fuel pool cooling system has been completed. The spent fuel pool is filled with water of acceptable quality and chemistry. The ac electrical power and distribution systems and other interfacing systems required for operation of the pumps and for data collection are available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Spent fuel pool cooling system performance is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the system properly performs its defense-in-depth function as described in subsection 9.1.3 and appropriate design specifications:

a) Proper operation of the spent fuel pool cooling pumps, valves, and strainers is verified.

b) Proper operation of the instrumentation, controls, actuation signals, and interlocks is verified, including:

- Pump flow rate
- Pump discharge pressure
- Spent fuel pool water level and control
- Spent fuel pool water temperature
- Water return temperature

This testing includes operation of the system pumps from the main control room.

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c) The capability of the pumps to provide the expected cooling flow rates to and from the pool is verified; with both pumps operating, with either individual pump operating, and with either heat exchanger operating.

d) In conjunction with Item c above, the pump(s) runout flow rate is verified to be properly limited, and adequate net positive suction head is verified to be available during the appropriate operating modes.

e) The proper operation of the spent fuel pool siphon breakers is verified.

f) The proper operation of the spent fuel pool post-72 hour gravity drain flowpaths from the cask washdown pit and the passive containment cooling water storage tank is verified.

g) The gates, drains, bellows, and gaskets in the refueling canal and fuel storage pool are checked for unacceptable leakage.

14.2.9.2.8 Fire Protection System Testing

Purpose

The purpose of the fire protection system testing is to verify the system properly performs the following defense-in-depth function as described in Subsection 9.5.1:

- Provide equipment for manual fire fighting in areas containing safe shutdown equipment
- Provide automatic fire suppression in areas containing selected non-safety-related equipment.
- Provide a nonsafety-related containment spray to reduce offsite dose following a severe accident

Prerequisites

The component tests of the fire protection system have been completed. Required preoperational testing of the ac power and distribution systems and other interfacing systems required for operation of the fire protection system. Data collection is available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Fire protection system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth function. The following testing demonstrates that the system performs its defense-in-depth functions specified in Subsection 9.5.1 and as specified in appropriate design specifications:

a) The capability of the seismic standpipes to supply the required fire water quantity and adequate water pressure for effective hose streams as the required flow rate is verified.

b) The operability of the fire detection equipment is verified to be able to properly detect fires and alert personnel.

c) The proper installation and operation of fire barriers, fire walls, and portions of HVAC systems used for smoke control and exhaust is verified.

d) The proper operation of the fire pumps, fire water storage tank, and fire water supply piping, valves, and instrumentation to provide the as-designed fire water supply is verified.

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e) The proper installation and operation of automatic fire suppression equipment is verified.

f) The proper installation and operation of electrical isolation devices for non-safety related equipment in opposite divisional fire areas is verified.

g) Operation of the containment spray remotely operated valve and the continuity of a flow path through the containment spray piping is verified.

14.2.9.2.9 Central Chilled Water System Testing

Purpose

The purpose of the central chilled water system testing is to verify that the as-installed low capacity portion of this system properly performs the following defense-in-depth function, as described in Subsection 9.2.7:

- Provide chilled water to cool air used to cool safety-related or defense-in-depth equipment rooms

The proper function of the high capacity portion of this system is also verified.

Prerequisites

The component testing of the low capacity subsystem of the central chilled water system has been completed. The required preoperational testing of the component cooling and service water systems, ac electrical power and distribution systems, and other interfacing systems required for operation of the central chilled water system has been completed. Data collection is available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Central chilled water system performance is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the central chilled water system performs its defense-in-depth functions described in Subsection 9.2.7 and appropriate design specifications:

a) Proper operation of the low capacity portion of the central chilled water system equipment is verified, including chillers, pumps, and valves.

b) Proper calibration and operation of defense-in-depth related instrumentation, controls, actuation signals and interlocks are verified, including:

   - Temperature control of the chilled water
   - Chiller and chilled water pump actuation
   - Chilled water pump flow
   - Chilled water flow control to air handling units

   This testing includes actuation of the defense-in-depth pumps and remotely operated valves from the main control room.

c) The proper chilled water flow rate to each of the nuclear island nonradioactive ventilation system air handling units is established, and the capability of each pump to provide this chilled water flow rate is verified.

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d) In conjunction with Item c above, the pump(s) runout flow rate is verified to be properly limited, and adequate net positive suction head is verified to be available during the appropriate operating modes.

e) The heat removal capability of the air-cooled chillers is verified when the component areas cooled by the nuclear island nonradioactive ventilation system air handling units are operating.

In addition, the operability of the high capacity portion of the central chilled water system described in Subsection 9.2.7 and appropriate design specifications, is verified.

14.2.9.2.10 Nuclear Island Nonradioactive Ventilation System Testing

Purpose

The purpose of the nuclear island nonradioactive ventilation system testing is to verify that the as-installed system properly performs the following defense-in-depth functions, as described in Subsection 9.4.1:

- Protect the main control room and control support area from smoke infiltration
- Provide the capability to remove smoke from the main control room, control support area, and Class 1E electrical equipment rooms
- Provide heating, ventilation, and cooling for the main control room, control support area, and Class 1E electrical equipment rooms
- Provide air filtration to limit radioactivity in the main control room and control support area
- Maintain passive heat sinks at acceptably low initial temperatures
- Maintain the main control room and control support area at positive pressure

The safety-related functions associated with this system are tested as part of the main control room emergency habitability testing described in Subsection 14.2.9.1.6.

Prerequisites

The component testing of the nuclear island nonradioactive ventilation system has been completed. The required preoperational testing of central chilled water system, the hot water heating system, the ac electrical power and distribution systems, and other interfacing systems required for operation of the above systems has been completed. Data collection is available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Nuclear island nonradioactive ventilation system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth functions. The following testing demonstrates that the system performs its defense-in-depth functions as described in Subsection 9.4.1 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

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- Smoke detectors and alarms
- Air handling unit and fan flows, controls, and alarms
- Differential air pressures and alarms
- Air and air filtration unit charcoal temperatures, controls, and alarms
- Air relative humidity measurements, controls, and alarms
- Isolation/shutoff damper controls
- Fire/smoke damper controls

This testing includes operation from the main control room.

c) The proper air flows from and through each air handling unit, as well as to and from the main control room, control support area, and other equipment rooms is established for each mode of operation.

d) The main control room and control support area are verified to be maintained at the proper positive pressure.

e) The main control room, control support area, class 1E equipment rooms, and passive heat sink areas are verified to be maintained at their proper temperature during hot functional testing.

f) Air inleakage into the main control room and control support area is measured using a tracer gas.

14.2.9.2.11 Radiologically Controlled Area Ventilation System

Purpose

The purpose of the radiologically controlled area ventilation system testing is to verify that the as-installed system properly performs the following defense-in-depth function, as described in Subsection 9.4.3:

- In conjunction with the low capacity portion of the central chilled water system, maintain the normal residual heat removal system and chemical and volume control system pump rooms at proper temperature during pump operation

Prerequisites

The component testing of the radiologically controlled area ventilation system has been completed. The required preoperational testing of the central chilled water system, the ac electrical power and distribution systems, and other interfacing systems required for operation of the radiologically controlled area ventilation system has been completed. Data collection is available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Radiologically controlled area ventilation system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth function as described in Subsection 9.4.3 and appropriate design specifications:

a) Proper function of the defense-in-depth fans, filters, heaters, and coolers is verified.

b) Proper operation of defense-in-depth instrumentation, controls, actuation signals, alarms, and interlocks is verified. This testing includes operation of the normal residual heat removal and chemical and volume control pump room cooler/fans from the main control room.

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c) The proper air flow and cooling capability of the normal residual heat removal and chemical and volume control pump room cooler/fans is verified.

d) The proper actuation of the normal residual heat removal and chemical and volume control pump room cooler fans in response to high and low room temperature is verified.

14.2.9.2.12 Plant Control System Testing

Purpose

The purpose of the plant control system testing is to verify that the as-installed components perform the following nonsafety-related defense-in-depth functions, described in Section 7.1:

- Provide control and coordination of the plant during startup, ascent to power, power operation and shutdown conditions by integrating the automatic and manual control of the reactor, reactor coolant and reactor support processes required for normal and off-normal conditions. This includes rod control, pressurizer pressure and level control, steam generator water level control, steam dump (turbine bypass) control and rapid power reduction.

- Provide control of other defense-in-depth systems and components.

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and the initial software diagnostics tests have been completed. Required electrical power supplies and control circuits are energized and operational. Required plant control system field wiring is electrically isolated to prevent operation of components controlled by the plant control system. Equipment or components that cannot be operated without damage or upsetting the plant are isolated, either by using test switches provided by the Plant Control System or by racking out power circuit breakers, to block device operation. Continuity of wiring up to the equipment is verified.

General Test Method and Acceptance and Functional Criteria

Performance of the plant control system hardware and software is observed and recorded during a series of individual component and integrated tests designed to verify operation of defense-in-depth functions. The following testing demonstrates that the system operates as described in Section 7.1 and applicable design specifications:

a) Processing of analog and digital signals is verified by injecting reference signals and monitoring the outputs of the plant control system.

b) Interfaces with other applicable plant equipment and systems such as reactor power control, feedwater control and turbine control are verified by demonstrating that injection of simulated inputs for each of the control functions provided in the main control room results in the proper output as indicated by contact operation, component actuation, or electrical test.

c) Interfaces with applicable plant equipment and systems are verified by demonstrating that injection of simulated inputs for selected control functions provided at the remote shutdown workstation results in the proper output as indicated by contact operation, component actuation, or electrical test.

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d) Proper operation of defense-in-depth processing, signal selector processing, monitoring, display and operator interface features provided by the plant control system is demonstrated by monitoring system outputs in response to simulated inputs, including simulated device or data highway failures, and utilization of provided self-test functions.

e) Proper functioning of the rod control system is verified by evaluating response to simulated demands from the plant control system and protection and safety monitoring system. The reactor vessel integrated head package is installed for this test and tests f) and g). Control rod drive mechanism operation is verified with all control rod drive mechanism cables connected, and the integrated head and control rod drive mechanism cooling system in operation. Control rod drive mechanisms are operated and current traces are taken to verify proper current, rod speed, and coil timing.

f) Proper operation of the digital rod position indication system is demonstrated by evaluating system response to simulated control rod position inputs, utilizing applicable displays, annunciators and alarms.

g) Proper operation of logic and controls for the pressurizer level and pressure control functions, including interlocks and equipment protective devices, is demonstrated by injecting simulated input signals representing anticipated pressurizer level and pressure transients.

14.2.9.2.13 Data Display and Processing System Testing

Purpose

The purpose of the data display and processing system testing is to verify that the as-installed components properly perform the following nonsafety-related defense-in-depth functions, described in Section 7.1:

- Display plant parameters for normal and emergency operations
- Provide plant alarm functions for normal and emergency plant operations
- Provide operational support for plant personnel, including computerized, interactive plant procedures
- Provide analysis, logging and historical storage and retrieval of plant data
- Provide a redundant communications network for transmission of plant parameters, plant status, displays, alarms and data files

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and the initial software diagnostics tests have been determined acceptable. Required electrical power supplies are energized and operational. Required system interfaces are connected and available or simulated as necessary to support the specified test configurations.

General Test Method and Acceptance and Functional Criteria

Performance of the data display and processing system hardware and software is observed and recorded during a series of individual component and integrated tests designed to verify that the data display and processing system equipment operates as described in Section 7.1 and the applicable design specifications:

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a) Initial operation of installed devices is verified by completing the diagnostics tests provided for the components and equipment.

b) Proper operation of the data display and processing system software and hardware is demonstrated by utilizing the data display and processing system to provide the processing, monitoring, display and operator interface features required during preoperational testing of associated plant instrumentation and control systems.

c) Verification that the time periods associated with accessing displays, displaying data after it has been made available on the plant monitor bus and display refresh or update rates are within the maximum allowable times is demonstrated. This verification is performed while utilizing the data display and processing system to provide the processing, monitoring, display and operator interface features required during preoperational testing of associated plant instrumentation and control systems.

14.2.9.2.14 Diverse Actuation System Testing

Purpose

The purpose of the diverse actuation system preoperational testing is to verify that the as-installed components properly perform the following nonsafety-related defense-in-depth functions, described in Section 7.7:

- Provide diverse (from the safety-related protection and safety monitoring system) automatic actuation of the following:
  - Reactor/turbine trip
  - Passive residual heat removal heat exchanger
  - Core makeup tanks/reactor coolant pump trip
  - Passive containment cooling
  - Isolation of selected containment penetrations
- Provide a diverse, alternate means for manual actuation of reactor trip and engineered safety features functions
- Provide a diverse system for monitoring selected plant parameters used to provide guidance for manual operation and confirmation of reactor trip and selected engineered safety features actuation

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and initial system diagnostics tests have been determined acceptable. Required electrical power supplies and control circuits are energized and operational. Required field wiring is electrically isolated to prevent operation of components controlled by the diverse actuation system. Exceptions are specifically identified in the preoperational test procedures if plant systems or components are to be operated during testing and these systems or components are to be properly aligned and have proper support systems operating prior to actuation of the particular system or component. Equipment or components that cannot be actuated without damage or upsetting the plant are isolated using the test switches provided by the Diverse Actuation System to block device actuation. Continuity of wiring up to the actuation equipment is verified.

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General Test Method and Acceptance and Functional Criteria

Performance of the diverse actuation system is observed and recorded during a series of individual component and integrated tests designed to verify operation of the system components. The following testing demonstrates that the system operates as described in Section 7.7 and applicable design specifications:

a) Processing of the analog and digital signals is verified by injecting reference signals and verifying the outputs at various locations in the system.

b) Correct outputs or actuation functions, for the automatic actuation logic mode, are verified by demonstrating that injection of simulated inputs for each of the specified actuation functions results in the proper output as indicated by contact operation, component actuation, or electrical test.

c) Correct outputs or actuation functions, for the manual actuation logic mode, are verified by demonstrating that each manual actuation function results in the proper output as indicated by contact operation, component actuation, or electrical test.

d) Proper operation of indications and alarms for the specified inputs, including those which provide reactor trip or engineered safety features actuation status, are verified by injecting simulated input signals.

14.2.9.2.15 Main AC Power System Testing

Purpose

The purpose of the main ac power system testing is to verify that the as-installed components properly perform the following nonsafety-related function:

- Provide ac electrical power to plant nonsafety-related loads as described in Subsection 8.3.1; and the following nonsafety-related function:

- Provide onsite power for post-72 hour electrical requirements.

Prerequisites

The component tests for the individual components associated with the main ac power system have been completed. The required test instrumentation is properly calibrated and operational. Additionally, the plant offsite grid connection is complete and available.

General Test Method and Acceptance and Functional Criteria

The capability of the main ac power system to provide power to plant loads under various plant operating conditions is verified. The system components to be tested include the ancillary diesel generator, the medium and low voltage power system, load centers, motor control centers, and instrumentation and controls. The following tests verify that the main ac power system provides its functions as specified in Subsection 8.3.1 and appropriate design specifications:

a) Verify the operability of medium-voltage supply breakers.

b) Energize the diesel-backed buses from their associated onsite standby diesel-generator supplies. Verify the bus voltages are within design limits. This test can be performed in conjunction with the testing of the standby diesel generator.

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c) Energize the medium voltage buses from their associated unit auxiliary transformer. Verify the bus voltages are within design limits.

d) Energize each medium voltage bus from its applicable reserve auxiliary transformer. Verify the bus voltages are within design limits.

e) Operate the automatic and maintenance bus transfer schemes. Verify successful transfer and return operation.

f) Verify correct operation of the manual controls, annunciation, and instrumentation for the 480 V load centers and their 6900 V feeder breakers.

g) Simulate fault conditions at the 480 V load centers and verify alarms and operation of trip devices and protective relays.

h) Energize the 480 V load centers. Verify the bus voltages are within design limits.

i) Verify the operability of motor control center supply breakers.

j) Simulate fault conditions at the motor control centers and verify alarms and operation of trip devices and protective relays.

k) Energize the motor control centers. Verify the bus voltages are within design limits.

l) Start ancillary diesel generators, energize voltage regulating transformers. Verify the input voltages to the regulating transformers are within design limits.

14.2.9.2.16 Non-Class 1E dc and Uninterruptible Power Supply System Testing

Purpose

The purpose of the non-Class 1E dc and uninterruptible power supply system testing is to verify the ability to provide continuous, reliable power for the non-Class 1E control and instrumentation defense-in-depth loads.

Prerequisites

The component tests for the individual components associated with the non-Class 1E dc and uninterruptible power supply system have been completed. Permanently installed and test instrumentation are properly calibrated and operational. The 480 V ac system is in operation to supply power to the battery chargers. Additionally, a test load is available for the performance of battery capacity tests.

General Test Method and Acceptance and Functional Criteria

The non-Class 1E dc and uninterruptible power supply system consists of electrical equipment including batteries, battery chargers, inverters, static transfer switches, and associated instrumentation and alarms that is used to supply power for the non-Class 1E control and instrumentation loads. Performance is observed and recorded during a series of individual component and integrated system tests. These tests verify that the non-Class 1E dc and uninterruptible power supply system operates as specified in Subsection 8.3.2 and appropriate design specifications:

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a) The capability of each non-Class 1E battery serving defense-in-depth loads is verified to meet or exceed the required ampere-hour rating by a battery performance test in accordance with IEEE 450. Following this discharge, the voltage of each cell is verified to be greater than or equal to the specified minimum cell voltage.

b) The capability of each charger serving defense-in-depth loads to meet the rating specified by Table 8.3.2-6 is verified. This testing includes a verification that the charger output voltage is within design limits.

c) The capability of each inverter to meet the rating specified by Table 8.3.2-6 is verified. This testing includes a verification that the output frequency and voltage to be within the limits specified in Table 8.3.2-6.

d) The proper operation and calibration of instrumentation and alarms, electrical ground detection, and permissive and prohibitive interlocks is verified.

Testing the non-class 1E dc and uninterruptible power supply system battery monitors is not required during the initial test program.

14.2.9.2.17 Standby Diesel Generator Testing

Purpose

The purpose of the standby diesel generator testing is to verify the capability to provide electrical power to plant nonsafety-related loads that enhance an orderly plant shutdown if off-site ac power is not available.

Prerequisites

The component tests have been completed. The necessary permanently installed instrumentation is properly calibrated and operational. Appropriate electrical power sources and diesel generator building heating and ventilation system are available for use. The plant control system is available for operation as applicable to the diesel generators. Sufficient diesel fuel is available, on site or readily accessible, to perform the tests.

General Test Method and Acceptance and Functional Criteria

Performance is observed and recorded during a series of individual component and integrated tests. These tests verify that the diesel generators operate properly as specified in Sections 8.3 and 9.5 through the following testing:

a) Verify the operability of generator protection features described in Subsection 8.3.1.1.2.2.

b) Simulate the loss of ac voltage and verify proper operation of undervoltage relay. Verify sequencer control logic support the description in Tables 8.3.1-1 and 8.3.1-2.

c) Verify the diesel generators fuel transfer pumps start and stop automatically in response to simulated day tank low level and high level signals.

d) Transfer fuel oil from the fuel oil storage tank to the diesel fuel oil day tanks by means of the transfer pumps. Verify flow parameters are within design limits.

e) Verify proper operation of diesel generators building heating and ventilation system fans and dampers, manual and automatic controls, alarms, and indicating instruments, as described in Subsection 9.4.10.

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f) Verify the air flow in the diesel generator building heating and ventilation system is acceptable.

g) Verify the diesel generator lockout features (turning gear engaged, emergency stop).

h) Verify that the diesel generator air starting system has sufficient capacity for cranking the engine for prescribed number of automatic or manual starts without recharging.

i) Start the diesel generators. Verify voltage and frequency control.

j) Verify the full load-carrying capability for a period of not less than 24 hours, of which 2 hours at are at a load equivalent to the 2-hour (Standby) rating of the diesel generators and 22 hours at a load equivalent to the continuous rating of the diesel generators. Verify the voltage and frequency requirements are maintained. Verify that the diesel generator cooling system functions within design limits.

k) Following the full-load capability test, simulate loss of ac voltage and verify proper automatic startup, sequencing, and operation of the diesel generators. Verify diesel generators bus de-energization and load shedding. Verify diesel generators attain frequency and voltage within design limits within the time described in Subsection 8.3.1.1.2.3. Verify sequencer control logic meets the description in Tables 8.3.1-1 and 8.3.1-2. Verify that the diesel generators continuous rating is not exceeded. Verify voltage and frequency requirements are maintained.

l) Verify that the rate of fuel consumption and the operation of the fuel transfer pumps and associated components, while providing power to the load equivalent to those specified in Table 8.3.1-1 or 8.3.1-2, are such that the design capacity of the fuel oil storage tanks meets the Subsection 9.5.4 requirement for 7-day storage inventory.

m) With each diesel generator bus supplied only by the diesel generator and supplying loads up to its continuous rating, trip a load equivalent to the largest single load in Table 8.3.1-1 or 8.3.1-2. Verify that the voltage and frequency values are maintained within design limits.

n) With each diesel generator supplying loads up to its continuous rating, trip the generator breaker that supplies power to the diesel generator bus. Verify that the diesel engine continues to run and does not trip on overspeed.

14.2.9.2.18 Radiation Monitoring System Testing

Purpose

The purpose of the radiation monitoring system testing is to verify that the as-installed radiation monitors perform their defense-in-depth function as described in Section 11.5.

Prerequisites

The component testing of the radiation monitoring system has been completed. The radiation monitors have been calibrated and the monitor check sources are installed, as appropriate. The required preoperational testing of the protection and safety monitoring system, plant control system, the electrical power and distribution systems, and other interfacing systems required for operation and data collection is available as needed to support the specified testing.

General Test Method and Acceptance and Functional Criteria

Radiation monitoring system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth

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functions. The following testing demonstrates that the system operates as specified in Section 11.5 and as specified in appropriate design specifications:

a) The proper calibration and operation of each radiation detector assembly and associated equipment using a standard radiation source or portable calibration unit are verified.

b) Proper operation of the monitoring equipment and controls required for manually initiated operation of the monitor check sources is verified.

c) Proper operation of the local processors that process and transmit radiation monitoring data to the protection and safety monitoring system or plant control system, as appropriate, is verified.

d) Proper actuation of alarms and signals for actuation of equipment responses following receipt of a high radiation signal is verified.

With the potential exception of the tracer gas and particulates testing, the preoperational testing discussed in Subsection 11.5.7 is performed following successful completion of the testing described above.

14.2.9.2.19 Plant Lighting System Testing

Purpose

The purpose of plant lighting system testing is to verify that the system can perform its defense-in-depth function of providing emergency lighting in the main control room and remote shutdown workstation area to illuminate these areas for emergency operations upon loss of normal lighting, as described in Subsection 9.5.3. In addition, the operability of lighting for emergency ingress and egress is verified.

Prerequisites

The component testing of the plant lighting system is completed. The required preoperational testing of the interfacing and support systems required for testing the emergency lighting function is available as needed to support the specified testing and system configurations including the Class 1E dc and uninterruptible power supply system, and the main ac power system.

General Test Method and Acceptance and Functional Criteria

Plant lighting system performance is observed during a series of individual component and integrated system testing to verify the system capability to perform its defense-in-depth functions. The following testing verifies that the system operates as described in Subsection 9.5.3 and in appropriate design specifications:

a) The proper operation of the plant lighting system emergency lighting is verified when powered from the Class 1E dc and uninterruptible power supply system.

b) Self-contained emergency lighting units are verified to be operable and installed into the proper ingress and egress paths, standby diesel generator rooms, switchgear rooms (annex and turbine buildings), fire pump enclosures, access route between the main control room and remote shutdown workstation, and appropriate connecting corridors and stairwells.

Testing of self contained emergency lighting units can be performed any time prior to use of the secondary system for heat removal during the startup test phase.

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14.2.9.2.20 Primary Sampling System Testing

Purpose

The purpose of the primary sampling system testing is to verify that the as installed components properly perform the following nonsafety-related defense-in-depth functions described in Subsection 9.3.3:

- Provide the capability to obtain samples of the reactor coolant, passive core cooling system, containment sump water, and containment atmosphere
- Provide the capability to analyze and measure samples.

Prerequisites

Component testing of the primary sampling system has been completed. Component cooling water is being provided to the sample cooler when samples are taken from the reactor coolant system when it is at elevated temperature. The systems/components to be sampled are filled and at their normal pressure and temperature. The liquid radwaste system is available to receive discharged sample fluid. Electrical power is available for operation of the system components and a source of compressed gas is available for operation of the gas sample eductor.

General Test Method and Acceptance and Functional Criteria

The performance of the primary sampling system is observed and recorded during a series of individual component tests and testing in conjunction with the reactor coolant system and passive core cooling system operation. The following testing demonstrates that the primary sampling system performs its defense-in-depth functions as described in Subsection 9.3.3 and appropriate design specifications.

a) Proper operation of the system's remotely-operated valves and eductor supply pump is verified.

b) Proper calibration and operation of instrumentation, controls, actuation signals, and interlocks are verified.

c) Verify the capability to obtain samples from the reactor coolant system, core makeup tanks, accumulators, containment sump, and containment atmosphere.

d) Verify the ability to return the sample stream fluid to the containment sump or liquid radwaste system, as appropriate.

e) Verify the capability to route sample streams to the laboratory.

f) Verify the operability of the test laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations.

14.2.9.2.21 Annex/Auxiliary Building Nonradioactive HVAC System

Purpose

The purpose of the annex/auxiliary non-radioactive HVAC system testing is to verify that the as installed system properly performs the defense-in-depth function, as described in Subsection 9.4.2, to provide conditioned air to maintain the diesel bus switchgear rooms and battery charger rooms (containing DC switchgear) within their design temperature range during operation of the onsite standby power system.

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Prerequisites

The component testing of the annex/auxiliary building HVAC system has been successfully completed. The required preoperational testing of the interfacing systems required for the operation of the above system is completed and these systems are available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

The annex/auxiliary building non-radioactive HVAC system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 9.4.2 and appropriate design specifications:

a) Proper function of the fans, filters, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

- Air handling unit and fan flows, controls, and alarms
- Air temperatures, alarms, and controls
- Damper open, close and modulate control in response to monitored parameters

This testing includes operation from the main control room.

c) The switchgear and equipment room subsystem air handling unit supply and return fans are verified to be automatically connected to the onsite standby power supplies on a loss of power to the buses powered by the standby diesels.

14.2.9.3 Preoperational Testing of Nonsafety-Related Radioactive Systems

14.2.9.3.1 Liquid Radwaste System Testing

Purpose

The purpose of the liquid radwaste system testing is to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the following safety-related function described in Subsection 11.2.1.1:

- Prevent back flow through the drain lines from the containment sump to the chemical and volume control system compartment and the passive core cooling system compartments, in order to prevent cross flooding of these compartments

The liquid radwaste system testing is performed to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the nonsafety-related functions described in Subsection 11.2.1.2, including receiving and processing reactor coolant system effluents, radioactive equipment and floor drains, and other radioactive liquid wastes from the plant.

Prerequisites

The component testing of the liquid radwaste system is completed. The required preoperational testing of the interfacing and support systems required for testing has been completed. Data collection is available as needed to support the specified testing and system configurations.

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General Test Method and Acceptance and Functional Criteria

Liquid radwaste system performance is observed and recorded during a series of individual component and system testing that characterizes system operation. This testing verifies that the system operates as specified in Section 11.2 and appropriate design specifications.

a) The drain lines from the passive core cooling system compartments and the refueling cavity are verified to provide a flow path to the reactor compartment.

b) Proper operation of the backflow prevention check valves is verified by the performance of baseline in-service tests, as specified in Subsection 3.9.6.

c) Proper operation of the system pumps and valves is verified, including:

- Effluent holdup tank pumps
- Waste holdup tank pumps
- Degasifier separator pumps
- Chemical waste tank pump
- Monitor tank pumps
- Reactor coolant drain tank pumps

d) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified, including:

- Pump controls and alarms
- Tank level control and alarms
- Valve and pump responses to safeguards signals
- Valve and pump responses to high radiation isolation signals

e) In conjunction with the gaseous radwaste system testing in Subsection 14.2.9.3.2, the proper operation of the degasifier is verified.

f) The proper operation of the liquid radwaste filters and ion exchangers is verified.

14.2.9.3.2 Gaseous Radwaste System Testing

Purpose

The purpose of the gaseous radwaste system testing is to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the following nonsafety-related functions described in Section 11.3.

- Collect waste gases that contain radioactivity or hydrogen
- Provide holdup for radioactive waste gases as appropriate

Prerequisites

The component testing of the gaseous radwaste system is completed. The required preoperational testing of the interfacing and support systems required for testing is completed, and data collection is available as needed to support the specified testing and system configurations. In addition, a source of hydrogen and calibration gases is available.

General Test Method and Acceptance and Functional Criteria

The performance of the gaseous radwaste system is observed and recorded during a series of individual component and system tests that characterizes the various modes of system operation.

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This testing verifies that the gaseous radwaste system operates as described in Section 11.3 and appropriate design specifications:

a) System and component control circuits, including response to normal control, interlock, and alarm signals are verified. The gaseous radwaste system instrumentation, controls, valves, and interlocks are verified to respond to various inputs and provide proper isolation and alarm signals. Appropriate automatic control functions are verified in response to abnormal conditions inputs.

b) Nitrogen, hydrogen, and calibration gases are routed through the system. Performance characteristics of the instrumentation and control systems are verified, and the delay bed operation is verified.

c) Moist test gas is routed through the system to verify proper moisture removal and detection.

d) The degasifier vacuum pump is verified to operate properly. Manual override of the automatic control functions of the drainpot and moisture separator drain and isolation valves is verified.

e) Sample pumps are operated and the sample flow meter indication is observed.

f) The proper operation of the degasifier moisture separator is demonstrated.

14.2.9.3.3 Solid Radwaste System Testing

Purpose

The purpose of the solid radwaste system testing is to verify that the as-installed components and associated piping, valves, and instrumentation operate properly to prepare waste generated during the normal operation of the plant for processing, packaging, and shipment as described in Subsection 11.4.1.2.

Prerequisites

The component testing of the solid radwaste system is completed. The interfacing and support systems required for testing and data collection are available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

The performance of the solid radwaste system is observed and recorded during a series of individual component and system tests that characterizes the various modes of system operation. This testing verifies that the solid radwaste system operates as described in Section 11.4 and in appropriate design specifications:

a) Tests are performed to verify that manual and automatic system controls, alarms, and instruments are functional; the system instrumentation, controls, valves, and interlocks respond properly to various inputs and provide proper isolation and alarm signals; and appropriate automatic control functions occur in response to abnormal condition inputs.

b) Tests are performed to verify proper system process rates as described in Section 11.4, and that no free liquids are present in packaged waste.

c) The capability to properly transfer and retain spent resins is verified.

d) The capability to properly handle filter cartridges in a manner that minimizes personnel radiation exposure is demonstrated.

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Any portion of the Solid Radwaste System testing can be performed any time prior to initial criticality during the startup test phase.

14.2.9.3.4 Radioactive Waste Drain System Testing

Purpose

The purpose of the radioactive waste drain system testing is to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the following functions, described in Section 11.2 and Subsection 9.3.5:

- Drain floor and equipment compartments
- Collect drainage and transfer drainage to the liquid radwaste system

Prerequisites

The component testing of the radioactive waste drain system is completed. The interfacing and support systems required for testing and data collection are available as needed to support the specified testing and system configurations, including the liquid radwaste system and compressed air supply.

General Test Method and Acceptance and Functional Criteria

The performance of the radioactive drain system is observed and recorded during a series of individual component and system tests that characterizes the various modes of system operation. This testing verifies that the system operates as described in Section 11.2 and Subsection 9.3.5, and in appropriate design specifications:

a) Proper operation of system instrumentation, controls, alarms, and interlocks is verified.

b) Proper operation of the system pumps and valves is verified.

c) Proper system and component flow paths and flowrates, including pump capacities and sump tank volumes, is verified.

d) Flow water in each drain path to verify that the drains discharge to their designated destination and that proper drain path segregation is maintained.

14.2.9.3.5 Steam Generator Blowdown System Testing

Purpose

The purpose of the steam generator blowdown system testing is to verify that the as-installed components and associated piping, valves, and instrumentation operate properly to provide an isolatable flow path for the controlled removal of water from the secondary side of the steam generators as described in Section 10.4.

Prerequisites

The component testing of the steam generator blowdown system is completed. The interfacing and support systems required for testing and data collection are available as needed to support the specified testing and system configurations. A portion of this testing is performed during the hot functional testing of the plant, when the steam generators are at or near normal operating pressure and temperature.

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General Test Method and Acceptance and Functional Criteria

The performance of the steam generator blowdown system is observed and recorded during a series of individual component and system tests that characterize the various modes of system operation. This testing demonstrates that the system operates as described in Section 10.4 and in appropriate design specifications:

a) Proper operation of system instrumentation, controls, alarms, and interlocks is verified.

b) Proper operation of the system pump and valves is verified.

c) The proper operation of the electrodeionization units is verified.

d) The heat transfer capability of each blowdown heat exchanger is verified.

e) The automatic isolation of steam generator blowdown on Low-2 steam generator narrow range water level is verified.

Any portion of the Steam Generator Blowdown System testing (except automatic isolation described in e) can be performed any time prior to initial criticality during the startup test phase.

Furthermore, Section 14.2.9.3.5 testing requirements c) and d) can be performed any time prior to exceeding 30% power during the startup test phase, so long as the flowpath through the blowdown heat exchanger can be placed in service and the blowdown system is available to support sampling and SGS chemistry control.

14.2.9.3.6 Waste Water System Testing

Purpose

The purpose of the waste water system testing is to verify that the as-installed components and associated piping, valves, and instrumentation operate properly to collect and perform appropriate processing of normally non-radioactive drains, as described in Section 11.2 and Subsection 9.2.9.

Prerequisites

The component testing of the waste water system is completed. The interfacing and support systems required for testing and data collection are available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

Waste water system performance is observed and recorded during a series of individual component and system testing that characterizes system operation. This testing verifies that the system operates as described in Section 11.2 and Subsection 9.2.9 and appropriate design specifications.

a) Proper operation of the system pumps and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks is verified.

c) Proper system and component flow paths and flowrates, including pump capacities and sump tank volumes is verified.

d) Verify the ability of the waste water system radiation alarm to stop the turbine building sump pumps.

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14.2.9.4 Preoperational Tests of Additional Nonsafety-Related Systems

14.2.9.4.1 Condensate System Testing

Purpose

The purpose of the condensate system testing is to verify that the as-installed components properly perform the system functions, described in Subsection 10.4.7, of delivering the required flow of heated water from the condenser hotwell to the feedwater system.

Prerequisites

The component testing of the condensate system has been completed. The component testing of the condenser is completed and a source of water of appropriate quality is available for filling the condenser hotwell. The steam generator feedwater system is available to receive flow from the condensate system. Required electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Condensate system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the condensate system can perform its functions as described in Subsection 10.4.7 and appropriate design specifications:

a) Proper operation of the condensate pumps and system valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) Proper operation of the heater drains is verified.

d) During the plant hot functional testing, the integrated operation of the condensate system in conjunction with the feedwater system is verified with the condenser and circulating water system in operation.

Any portion of Condensate System testing can be performed any time prior to use of the secondary system for heat removal during the startup test phase.

14.2.9.4.2 Condenser Air Removal System Testing

Purpose

The purpose of the condenser air removal system testing is to verify that the as-installed components properly perform the system functions to establish and maintain the required vacuum in the main condenser, as described in Subsection 10.4.2.

Prerequisites

The component testing of the condenser air removal system has been completed. The component testing of the condenser has been completed and a source of water of appropriate quality is available for filling the condenser hotwell. The turbine gland sealing system and exhaust blower are in operation. A source of steam such as the auxiliary boiler is available. Required support systems, electrical power supplies and control circuits are operational.

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General Test Method and Acceptance and Functional Criteria

Condenser air removal system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the condensate system can perform its functions as described in Subsection 10.4.2 and appropriate design specifications:

a) Proper operation of the vacuum pumps and system valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) The capability of the vacuum pumps to establish the required vacuum in the main condenser is verified.

14.2.9.4.3 Main Turbine System and Auxiliaries Testing

Purpose

The purpose of the main turbine system testing is to verify that the as-installed main turbine and its auxiliary components properly perform their functions, described in Sections 10.2 and 10.4. This testing includes testing of the turbine gland sealing system, lube oil system, turning gear, turbine controls and protective functions, and moisture separator reheater.

Prerequisites

The component testing of the main turbine and its auxiliaries has been completed. The component testing of the condenser is completed and a source of water of appropriate quality is available for filling the condenser hotwell. The main turbine is on turning gear and the condenser air removal system is operable. A source of steam such as the auxiliary boiler is available. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Because this testing is performed using a temporary steam source, the extent to which the turbine can be tested in preoperational testing is limited. However, the proper function of the turbine auxiliaries is verified to assure the turbine will operate properly when a greater amount of steam is provided.

Main turbine system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the turbine and its auxiliaries function as described in Sections 10.2 and 10.4 and in appropriate design specifications:

a) Proper operation of the turbine lube oil pump and turning gear motor, gland seal exhaust blower, and moisture separator and gland seal valves is verified.

b) Proper operation of system valves including the turbine control and intercept valves is verified.

c) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

d) Proper turbine operation during the turning gear testing is verified. The turning gear engagement and disengagement functions are verified to operate properly.

e) Proper performance of the turbine trip functions is verified.

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Any portion of Main Turbine System and Auxiliaries testing can be performed any time prior to placing the main generator on-line during the startup test phase.

14.2.9.4.4 Main Generator System and Auxiliaries Testing

Purpose

The purpose of the main generator system testing is to verify that the as-installed main generator and its auxiliary components properly perform their functions, described in Sections 8.2 and 10.2. This testing includes testing of the generator cooling systems, lube oil system, controls, and protective functions.

Prerequisites

The component testing of the main generator and its auxiliaries has been completed. The component testing of the condenser is completed. The turbine cooling water system is operable, and required support systems, electrical power supplies, and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Performance is observed and recorded during a series of individual component and integrated tests. These tests verify that the generator operated as specified in Sections 8.2 and 10.2 through the following testing:

a) Verify the operability of the generator protection features.

b) Verify proper cooling of the generator stator and rotor.

c) Verify MW, MVAR, and frequency control.

Any portion of Main Generator System and Auxiliaries testing can be performed any time prior to the load swing test described in subsection 14.2.10.4.20 during the startup test phase.

14.2.9.4.5 Turbine Building Closed Cooling Water System Testing

Purpose

The purpose of the turbine building closed cooling water system testing is to verify that the as-installed components properly perform their functions of supplying adequate cooling water to the designated turbine building components, as described in Subsection 9.2.8.

Prerequisites

The component testing of the turbine building closed cooling water system has been completed. The cooled components are operational and operating to the extent possible, especially for verifying the heat exchanger capability. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Turbine building closed cooling water system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 9.2.8 and appropriate design specifications:

a) Proper operation of the system pumps and valves is verified.

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b) Proper operation of the system instrumentation, controls, actuation signals, and interlocks is verified.

14.2.9.4.6 Circulating Water System Testing

Purpose

The purpose of the circulating water system testing is to verify that the as-installed components properly perform the functions of cooling and circulating adequate cooling water to the main condenser and turbine building closed cooling water system heat exchangers as described in Subsection 10.4.5.

Prerequisites

The component testing of the circulating water system has been completed. The main condenser and turbine building closed cooling water heat exchangers are operational. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Since there will be little, if any, heat rejected to the circulating water system, verification of the heat removal capability of the cooling tower(s) is performed during the startup testing of the plant when the reactor is producing power.

Circulating water system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 10.4.5 and appropriate design specifications:

a) Proper operation of the system pumps and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

   The proper operation of the system freeze protection equipment is verified, as applicable.

Any portion of the Circulating Water System testing can be performed any time prior to use of the secondary system for heat removal during the startup test phase.

14.2.9.4.7 Turbine Island Chemical Feed System Testing

Purpose

The purpose of the turbine island chemical feed system testing is to verify that the as-installed components properly perform the functions of adding appropriate chemicals to the condensate, service water, and auxiliary boiler in a controlled manner, as described in Subsection 10.4.11.

Prerequisites

The component testing of the chemical feed system has been completed. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Turbine island chemical feed system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 10.4.11 and appropriate design specifications:

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a) Proper operation of the system pumps and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

Any portion of Turbine Island Chemical Feed System testing can be performed any time prior to use of the secondary system for heat removal during the startup test phase.

14.2.9.4.8 Condensate Polishing System Testing

Purpose

The purpose of the condensate polishing system testing is to verify that the as-installed components properly perform the functions of removing corrosion products, dissolved solids, and other impurities from the condensate system, as described in Subsection 10.4.6.

Prerequisites

The component testing of the condensate polishing system has been completed and a source of water of appropriate quality is available for filling the condenser hotwell. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Condensate polishing system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 10.4.6 and appropriate design specifications:

a) Proper operation of the system valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

14.2.9.4.9 Demineralized Water Transfer and Storage System Testing

Purpose

The purpose of the demineralized water transfer and storage system testing is to verify that the as-installed components properly perform the function of providing reservoirs of demineralized water and deliver deoxygenated, demineralized water to various plant users, as described in Subsection 9.2.4.

Prerequisites

The component testing of the demineralized water transfer and storage system has been completed. The demineralized water treatment system is operational and the equipment which uses demineralized water is able to accept water. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Demineralized water transfer and storage system performance is observed and recorded during a series of individual component and integrated system testing. The following defense-in-depth testing verifies that the system functions as described in Subsection 9.2.4 and appropriate design specifications:

a) Proper operation of the system pumps, valves, blower, and heaters is verified.

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b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

14.2.9.4.10 Compressed and Instrument Air System Testing

Purpose

The purpose of the compressed and instrument air system testing is to verify that the as-installed components properly perform the functions of providing compressed air at the required pressures to various plant users, as described in the Compressed and Instrument Air portion of Section 9.3.

Prerequisites

The component testing of the compressed and instrument air system has been completed. The component cooling water system is operational and providing cooling for the compressor units. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Compressed and instrument air system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system and its plant users, where applicable, function as described in Subsection 9.3.1.4 and appropriate design specifications:

a) Proper operation of the system compressors, receivers, prefilters, air dryers, afterfilters, purifiers, and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) Integral testing is performed to verify that the instrument air subsystem can provide sufficient air pressure to accommodate the maximum number of air-operated valves expected to operate simultaneously.

d) Testing is performed to verify the fail-safe positioning of safety-related air-operated valves for complete loss of system pressure, both sudden and gradual, and to partial reduction in system pressure as described in Subsection 9.3.1.4.

e) Proper calibration is verified for system relief valves that protect the system from overpressure conditions.

14.2.9.4.11 Containment Recirculation Cooling System Testing

Purpose

The purpose of the containment recirculation cooling system testing is to verify that the as-installed components properly perform the functions of maintaining the proper containment air temperature during normal plant operation and during refueling and maintenance operations, as described in Subsection 9.4.6.

Prerequisites

The component testing of the containment recirculation cooling system has been completed. The central chilled water system and hot water heating system are operational. Required support systems, electrical power supplies and control circuits are operational.

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General Test Method and Acceptance and Functional Criteria

Containment recirculation cooling system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 9.4.6 and appropriate design specifications:

a) Proper operation of the system fans and dampers is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

14.2.9.4.12 Containment Air Filtration System Testing

Purpose

The purpose of the containment air filtration system testing is to verify that the as-installed components properly perform the functions of supplying and exhausting air to maintain the proper containment air pressure, and filter exhaust air to minimize radiation release, as described in Subsection 9.4.7.

Prerequisites

The component testing of the containment air filtration system has been completed. The portions of the radiologically controlled area ventilation system connected to the air filtration system are operational. The hot water heating and chilled water systems are required for verification of the air filtration heating and cooling functions. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Containment air filtration system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 9.4.7 and appropriate design specifications:

a) Proper operation of the system fans and dampers is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) Proper operation of the containment air filtration filters is verified.

14.2.9.4.13 Plant Communications System Testing

Purpose

The purpose of the plant communications system testing is to verify that the as-installed components properly perform the functions of verifying the proper operation and adequacy of the plant communication systems used during normal and abnormal operations, as described in Section 9.5.

Prerequisites

The component testing of the communication system has been completed. Required support systems, electrical power supplies and control circuits are operational.

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General Test Method and Acceptance and Functional Criteria

Plant communications system performance is observed and recorded during a series of individual component and integrated system testing. The inplant communications system includes the following subsystems:

- Wireless telephone system
- Telephone/page system
- Private Branch Exchange (PBX) System
- Sound Powered Phone System
- Emergency Offsite Communication System
- Security Communication System
- Plant Radio System

The following testing verifies that the system functions as described in Section 9.5 and appropriate design specifications:

a) Transmitters and receivers are verified to operate without excessive interference.

b) Proper operation of controls, switches, and interfaces is verified.

c) Proper operation of the public address, including the plant emergency alarms, is verified.

d) The proper operation of equipment expected to function under abnormal conditions such as a loss of electrical power, shutdown from outside the control room, or execution of the plant emergency plan is verified. This functional testing will be performed under conditions that simulate the maximum plant noise levels being generated during the various operating conditions, including fire and accident conditions, to demonstrate system capabilities.

14.2.9.4.14 Mechanical Handling System Crane Testing

Purpose

The purpose of the mechanical handling system crane testing is to verify that the as-installed components properly perform their functions. The test ensures operation and adequacy of the containment polar crane, which is used to lift and relocate components providing access to the reactor fuel, vessel internals, and reactor components during refueling and servicing operations.

In addition, the following load handling systems described in Subsection 9.1.5 are tested; the equipment hatch hoist, the maintenance hatch hoist, and the cask handling crane.

Prerequisites

The component testing of the heavy lift cranes has been completed. Required support systems, electrical power supplies and control circuits are operational. The heavy load analysis, defining the load paths, has been completed.

General Test Method and Acceptance and Functional Criteria

Heavy load crane performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the crane systems function as described in Subsection 9.1.5 and in appropriate design specifications:

a) Proper operation and assembly of the various cables, grapples, and hoists including brakes, limit switches, load cells, and other equipment protective devices are verified.

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b) Proper operation of control, instrumentation, interlocks, and alarms is verified.

c) Dynamic and static load testing of cranes and hoists, and associated lifting and rigging equipment are performed including a static load test at 125 percent of rated load and full operational test at 100 percent of rated load.

14.2.9.4.15 Seismic Monitoring System Testing

The seismic monitoring system testing described in this section also applies to site-specific seismic sensors.

Purpose

The purpose of the seismic monitoring system testing is to verify that the as-installed components properly perform the functions of verifying proper operation in response to a seismic event, as described in Section 3.7.

Prerequisites

The component testing of the seismic monitoring system has been completed. Required support systems, electrical power supplies and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

Seismic monitoring system instrumentation performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Section 3.7 and appropriate design specifications:

a) Proper calibration and response of seismic instrumentation are verified, including verification of alarm and initiation setpoints.

b) Proper operation of internal calibration and test features are verified.

c) Proper integrated system response, including actuations, alarms, and annunciations, is verified.

d) Verify the proper operation of the recording and analysis functions on a loss of AC power sourced.

14.2.9.4.16 Special Monitoring System Testing

Purpose

The purpose of the special monitoring system testing is to verify that the as-installed components of the portion of the special monitoring system that is identified as the digital metal impact monitoring system properly perform the following nonsafety-related functions, described in Subsection 4.4.6:

- Detect the presence of metallic debris in the reactor coolant system
- Obtain baseline data for metal impact monitoring prior to power operations

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and initial system diagnostics tests have been determined acceptable. Required electrical power supplies are energized and operational.

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General Test Method and Acceptance and Functional Criteria

Performance of the special monitoring system is observed and recorded during a series of individual component and integrated tests designed to verify system operation in response to specified input conditions. The following testing demonstrates that the system operates as described in Subsection 4.4.6 and the applicable design specifications:

a) Proper calibration and response of digital metal impact monitoring instrumentation are verified.

b) Proper operation of the digital metal impact monitoring system is verified by evaluating system response to simulated input signals representing the anticipated signal range.

c) Baseline response data is obtained for the digital metal impact monitoring system to serve as a reference for monitoring degradation of sensor response.

14.2.9.4.17 Secondary Sampling System Testing

Purpose

The purpose of the secondary sampling system testing is to verify that the as-installed components properly perform the following nonsafety-related functions, described in Subsection 9.3.4:

- Provide the capability to continuously or semi-continuously monitor selected secondary water and steam process streams in order to establish and maintain proper water chemistry during plant operation
- Provide the capability to manually analyze additional secondary water and steam process streams

Prerequisites

Component testing of the secondary sampling system has been completed. Cooling water is being provided to the sample coolers when samples are taken from sample points with fluid temperatures exceeding 125°F. The systems/components to be sampled are filled and operating at their preoperational pressure and temperature. Electrical power is available for operation of the on-line chemistry analyzers.

General Test Method and Acceptance and Functional Criteria

The performance of the secondary sampling system is observed and recorded during a series of individual component tests and testing in conjunction with the plant in operation at preoperational pressure and temperature. The following testing verifies that the secondary sampling system operates as described in Subsection 9.3.4 and appropriate design specifications.

a) Proper calibration and operation of on-line continuous and semi-continuous analyzers, data collection and display, controls, and actuation signals to the turbine island chemical feed system are verified.

b) Proper calibration and operation of the portable analyzer are verified.

c) Proper operation of the sample coolers is verified.

d) Capability to obtain grab samples from the sample points is verified.

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Any portion of Secondary Sampling System testing can be performed any time prior to placing the main generator on-line during the startup test phase (except for the heater drain system portion of the system, which will be tested as extraction steam increases with the main generator online prior to completion of the startup test phase of the initial test program).

14.2.9.4.18 Turbine Building Ventilation System

Purpose

The purpose of the turbine building ventilation system testing is to verify that the as installed system properly performs the normal air conditioning and ventilation functions, as described in Subsection 9.4.9.

Prerequisites

The component testing of the turbine building ventilation system has been successfully completed. The required preoperational testing of the central chilled water and hot water heating systems, and other interfacing systems required for the operation of the above systems and data collection is completed and these systems are available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

The turbine building ventilation system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 9.4.9 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

   - Air handling unit and fan flows, controls, and alarms
   - Damper open, close and modulate control

   This testing includes operation from the main control room.

Any portion of Turbine Building Ventilation System testing can be performed any time prior to placing the main generator on-line during the startup test phase.

14.2.9.4.19 Health Physics and Hot Machine Shop HVAC System

Purpose

The purpose of the health physics and hot machine shop HVAC system testing is to verify that the as installed system properly performs the normal air conditioning and ventilation functions, as described in Subsection 9.4.11.

Prerequisites

The component testing of the health physics and hot machine shop HVAC system has been successfully completed. The required preoperational testing of the central chilled water and hot water heating systems, and other interfacing systems required for the operation of the above systems is completed and these systems are available as needed to support the specified testing and system configurations.

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General Test Method and Acceptance and Functional Criteria

The health physics and hot machine shop HVAC system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 9.4.11 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

- Radiation detectors and alarms
- Air handling unit and fan flows, controls, and alarms
- Air temperatures, alarms, and controls
- Differential air pressure and alarms
- Damper open, close and modulate control

This testing includes operation from the main control room.

c) The health physics and hot machine shop HVAC system is verified to maintain the access control area and hot machine shop at a slightly negative pressure with respect to outdoors and clean areas of the annex building to prevent unmonitored releases of radioactive contaminants.

14.2.9.4.20 Radwaste Building HVAC System

Purpose

The purpose of the radwaste building HVAC system testing is to verify that the as installed system properly performs the normal air conditioning and ventilation functions, as described in Subsection 9.4.8, as required for personnel and equipment in serviced areas; and provides the proper filtration of air from potentially contaminated areas.

Prerequisites

The component testing of the radwaste building HVAC system has been successfully completed. The required preoperational testing of the central chilled water and hot water heating systems, the ac electrical power and distribution systems, and other interfacing systems required for the operation of the above systems is completed and these systems are available as needed to support the specified testing and system configurations.

General Test Method and Acceptance and Functional Criteria

The radwaste building HVAC system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Subsection 9.4.8 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

- Radiation detectors and alarms
- Air handling unit and fan flows, controls, and alarms
- Air temperatures, alarms, and controls
- Differential air pressures and alarms

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- Damper open, close and modulate control in response to monitored parameters

This testing includes operation from the main control room.

c) The radwaste building is verified to be maintained at a slightly negative pressure with respect to outdoors to prevent unmonitored releases of radioactive contaminants.

14.2.9.4.21 Main, Unit Auxiliary and Reserve Auxiliary Transformer Test

Purpose

The purpose of the main, unit auxiliary and reserve auxiliary transformer testing is to demonstrate the energization of the transformers and the proper operation of associated protective relaying, alarms, and control devices.

Prerequisites

The component tests for the individual components associated with the main, unit auxiliary and reserve auxiliary transformers have been completed. The required test instrumentation is properly calibrated and operational. Additionally, the plant offsite grid connection is complete and available.

General Test Method and Acceptance and Functional Criteria

The following tests demonstrate proper energization of the main, unit auxiliary and reserve auxiliary transformers and proper operation of protective relaying, alarms, and control devices associated with the transformers:

a) Energize the unit auxiliary transformers. Verify phase rotation. Verify phase voltages are within design limits.

b) Energize the reserve auxiliary transformers. Verify phase rotation. Verify phase voltages are within design limits.

c) Simulate fault conditions and verify alarms and operation of protective relaying circuits.

14.2.9.4.22 Not Used

14.2.9.4.23 Off-site AC Power Systems

Purpose

Off-site alternating current (ac) power system testing demonstrates the energization and proper operation of the as-installed switchyard components, as described in Section 8.2.

Prerequisites

Component testing of plant off-site ac power systems, supporting systems, and components is completed. The components are operational and the switchyard equipment is ready to be energized. The required test instrumentation is properly calibrated and operational. The off-site grid connection is complete and available.

General Test Method and Acceptance and Functional Criteria

The plant off-site ac power system components undergo a series of individual component and integrated system tests to verify that the off-site ac power system performs in accordance with the associated component design specifications. The individual component and integrated tests include:

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a. Availability of ac and direct current (dc) power to the switchyard equipment is verified.

b. Operation of high voltage (HV) circuit breakers is verified.

c. Operation of HV disconnect switches and ground switches is verified.

d. Operation of substation transformers is verified.

e. Operation of current transformers, voltage transformers, and protective relays is verified.

f. Operation of switchyard equipment controls, metering, interlocks, and alarms that affect plant off-site ac power system performance is verified.

g. Design limits of switchyard voltages and stability are verified.

h. Under simulated fault conditions, proper function of alarms and protective relaying circuits is verified.

i. Operation of instrumentation and control alarms used to monitor switchyard equipment status.

j. Proper operation and load carrying capability of breakers, switchgear, transformers, and cables, and verification of these items by a non-testing means such as a QC nameplate check of as built equipment where testing would not be practical or feasible.

k. Verification of proper operation of the automatic transfer capability of the preferred power supply to the maintenance power supply through the reserve auxiliary transformer.

l. Switchyard interface agreement and protocols are verified.

The test results confirm that the off-site ac power systems meet the technical and operational requirements described in Section 8.2.

14.2.9.4.24 Raw Water System

Purpose

Raw water system testing verifies that the as-installed components supply raw water to the circulating water cooling tower basin, service water system cooling tower basin, fire protection water storage tanks, and other systems, as described in Subsection 9.2.11.

Prerequisites

Component testing of the raw water system is completed. The components are operational and the storage tanks and cooling tower basins are able to accept water. Required support systems, electrical power supplies, and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

The raw water system component and integrated system performance is observed to verify that the system functions, as described in Subsection 9.2.11 and in appropriate design specifications. The individual component and integrated system tests include:

a. Operation of the system pumps, traveling screens, automatic strainers, and valves is verified.

b. Operation of the system instrumentation, controls, actuation signals, alarms, and interlocks is verified.

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c. Operation of heat tracing on system piping is verified.

14.2.9.4.25 Sanitary Drainage System

Purpose

Sanitary drainage system testing verifies that the as-installed components properly collect and discharge sanitary waste, as described in Subsection 9.2.6.

Prerequisites

Component testing of the sanitary drainage system is completed. Required support systems, electrical power supplies, and control circuits are operational.

General Test Method and Acceptance and Functional Criteria

The sanitary drainage system component and integrated system performance is observed to verify that the system functions, as described in Subsection 9.2.6.2.1 and in appropriate design specifications. The individual component and integrated system tests include:

a. Operation of lift stations and valves is verified.

b. Operation of the system instrumentation, controls, actuation signals, and interlocks is verified.

14.2.9.4.26 Fire Brigade Support Equipment

Purpose

Fire brigade support equipment testing verifies that the equipment operates and is available when needed to perform the fire brigade functions, as described in Section 9.5.

Prerequisites

Equipment is ready and available for testing.

General Test Method and Acceptance and Functional Criteria

The fire brigade support equipment undergoes a series of inspections to verify availability and operability. Equipment is available for selection and use, based on the hazard. Fire brigade support equipment tests include:

a. Location of portable extinguishers is verified; portable extinguishers are verified fully charged.

b. Operation of portable ventilation equipment is verified.

c. Operation of portable communication equipment is verified.

d. Operation of portable lighting is verified.

e. Operation of self-contained breathing apparatus and face masks is verified.

f. Operation of keys to open locked fire area doors is verified.

g. Turnout gear functionality and availability is verified.

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h. Compatibility of threads for hydrants, hose couplings, and standpipe risers with the local fire department equipment is verified, or alternatively, an adequate supply of readily available hose thread adaptors is verified.

14.2.9.4.27 Portable Personnel Monitors and Radiation Survey Instruments

Purpose

Portable personnel monitors and radiation survey instruments testing verifies that the devices operate in accordance with their intended function in support of the radiation protection program, as described in Chapter 12.

Prerequisites

Portable personnel monitors, radiation survey instruments, and appropriate certified test sources are on site.

General Test Method and Acceptance and Functional Criteria

The portable personnel monitors and radiation survey instruments are source checked, tested, maintained, and calibrated in accordance with the manufacturers' recommendations. The portable monitors and instruments tests include:

a. Proper function of the monitors and instruments to respond to radiation is verified, as required.

b. Proper operation of instrumentation controls, battery, and alarms, if applicable.

14.2.10 Startup Test Procedures

The startup testing program is based on increasing power in discrete steps. Major testing is performed at discrete power levels as described in Subsection 14.2.7. The first tests during Power Ascension Testing that verify movements and expansion of equipment are in accordance with design, and are conducted at a power level as low as practical (approximately 5 percent).

The governing Power Ascension Test Plan requires the following operations to be performed at appropriate steps in the power-ascension test phase:

- Conduct any tests that are scheduled at the test condition or power plateau.
- Confirm core performance parameters (core power distribution) are within expectations.
- Determine reactor power by heat balance, calibrate nuclear instruments accordingly, and confirm the existence of adequate instrumentation overlap between the startup range and power range detectors.
- Reset high-flux trips just prior to ascending to the next level to a value no greater than 20 percent beyond the power of the next level unless Technical Specification limits are more restrictive.
- Perform general surveys of plant systems and equipment to confirm that they are operating within expected values.
- Check for unexpected radioactivity in process systems and effluents.
- Perform reactor coolant leak checks.

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- Review the completed testing program at each plateau; perform preliminary evaluations, including extrapolation core performance parameters for the next power level; and obtain the required management approvals before ascending to the next power level or test condition.

Upon completion of a given test, a preliminary evaluation is performed that confirms acceptability for continued testing. Smaller transient changes are performed initially, gradually increasing to larger transient changes. Test results at lower powers are extrapolated to higher power levels to determine acceptability of performing the test at higher powers. This extrapolation is included in the analysis section of the lower power procedure.

Surveillance test procedures may be used to document portions of tests, and ITP tests or portions of tests may be used to satisfy Technical Specifications surveillance requirements in accordance with administrative procedures. At Startup Test Program completion, a plant capacity warranty test is performed to satisfy the contract warranty and to confirm safe and stable plant operation.

Those tests comprising the startup test phase are discussed in this subsection. For each test a general description is provided for test objective, test prerequisites, test description, and test performance criteria, where applicable. In describing a test, the operating and safety-related characteristics of the plant to be tested and evaluated are identified. The startup test phase also includes any testing from Section 14.2.9 that allows deferral of the applicable testing scope to be completed prior to the specified startup test conditions.

Where applicable, the relevant performance criteria for the test are discussed. Some of the criteria relate to the value of process variables assigned in the design or analysis of the plant, component systems, and associated equipment. Other criteria may be associated with expectations relating to the performance of systems.

The specifics of the startup tests relating to test methodology, plant prerequisites, initial conditions, performance criteria, and analysis techniques are discussed in Section 14.4 in the form of plant, system and component performance and testing procedures.

14.2.10.1 Initial Fuel Loading and Precritical Tests

Tests performed after preoperational testing is complete but prior to initial criticality are described in this section. These tests include those performed prior to core load to verify the plant is ready for core loading, the loading of the core and the tests performed under hot conditions after the core has been loaded but prior to initial criticality.

Tests to be performed prior to and during initial core loading are described in Subsections 14.2.10.1.1 through 14.2.10.1.5. These tests verify the systems necessary to monitor the fuel loading process are operational and that the core loading is conducted properly.

After core load, tests are performed at hot conditions to bring the plant to a final state of readiness prior to initial criticality.

14.2.10.1.1 Fuel Loading Prerequisites and Periodic Checks

Objectives

- Specify the prerequisites for initial fuel loading, including the status of required systems, plant conditions, and special equipment

- Provide a checklist for periodic verification that the conditions required for fuel loading are being maintained

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Prerequisites

- Plant systems required for initial fuel loading have been satisfactorily tested and turned over to the plant operating staff, and are in the status specified
- Plant conditions required for initial core loading are as specified
- Special equipment required for initial fuel loading is available and operable

Test Method

- Prior to the beginning of fuel loading, verify and document the required status of test prerequisites
- Throughout fuel loading, verify through periodic checks that conditions required for safe fuel loading are being maintained

Performance Criterion

The required status of prerequisites for initial fuel loading is verified and documented prior to fuel loading and maintained throughout the loading process.

14.2.10.1.2 Reactor Systems Sampling for Fuel Loading

Objective

- Verify that the dissolved boron concentration in the reactor coolant system and directly connected portions of associated auxiliary systems is uniform and equals or exceeds the value required by the plant Technical Specifications for fuel loading.

Prerequisites

- Plant Technical Specifications for fuel loading are complete and verified.
- The boric acid storage tank, chemical and volume control system makeup pumps, and associated piping and equipment are filled and operable.
- The reactor vessel is filled with borated water to a level at or above the centerline of the outlet nozzles.
- The water in the reactor vessel and reactor coolant system piping, including all directly connected auxiliary systems, is borated to a value that equals or exceeds the value specified in the plant Technical Specifications for fuel loading, and that water is circulating through the normal residual heat removal system at a rate that provides reasonable assurance of a uniform concentration.

Test Method

- Operate the normal residual heat removal system until a uniform boron concentration is established, and then obtain boron concentration samples from the reactor vessel and operating residual heat removal loop to verify the concentrations are within a specified range of each other.
- Obtain and analyze samples from at least one representative point in each auxiliary system and at four equidistant depths in the reactor vessel for boron concentration
- Periodically repeat sampling until the performance criteria are met

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Performance Criterion

- The minimum boron concentration of all samples equals or exceeds the value specified in the plant Technical Specifications for fuel loading. If the minimum boron concentration criteria is not met, the chemical and volume control system is used to increase the boron concentration to above the specified limit.

14.2.10.1.3 Fuel Loading Instrumentation and Neutron Source Requirements

Objectives

- Verify alignment, calibration, and neutron response of the temporary core loading instrumentation (if temporary instrumentation is implemented to satisfy the test requirements) prior to the start of fuel loading

- Verify the neutron response of the nuclear instrumentation system source range channels prior to the start of fuel loading

- Verify the neutron response of the temporary (if temporary instrumentation is implemented to satisfy the test requirements) and nuclear instrumentation system source range instrumentation prior to resumption of fuel loading following any delay of 8 hours or more

Prerequisites

- The following special equipment is available:

  – The temporary core loading package (if temporary instrumentation is implemented to satisfy the test requirements) consisting of three complete counting channels, including preshipment alignment and calibration data

  – A neutron source, either a portable source or a fuel assembly containing a primary source, with sufficient strength to verify detector response

- Preoperational testing of the nuclear instrumentation system source range channels is completed

Test Method

- Prior to the start of fuel loading, verify the response of temporary (if temporary instrumentation is implemented to satisfy the test requirements) and nuclear instrumentation system source range channels to neutrons by using a neutron source

- Verify proper alignment and calibration of the temporary channels (if temporary instrumentation is implemented to satisfy the test requirements) by comparing the neutron response data to the data obtained during preshipment testing

- Prior to resumption of fuel loading following a delay of 8 hours or more, verify proper operation of the temporary (if temporary instrumentation is implemented to satisfy the test requirements) and nuclear instrumentation system source range channels by performing a neutron response check (using the neutron source) or by statistical analysis of the count rate data

- Verify count rates attributable to core neutrons are available on at least two of the pulse-type nuclear channels (either temporary channels or nuclear instrumentation system source range channels) at all times following the installation of the initial nucleus of nuclear fuel assemblies

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(approximately eight fuel assemblies, one of which contains a neutron source) and during subsequent fuel loadings, which permits meaningful Inverse Count Rate Ratio (ICRR) monitoring

Performance Criterion

Equipment used for neutron monitoring during fuel loading is operating correctly and is responsive to changes in neutron flux levels. At least two pulse-type nuclear channels are available and detecting neutrons during fuel loading, following installation of the initial nucleus of fuel assemblies (approximately eight fuel assemblies, one of which contains a neutron source).

14.2.10.1.4 Inverse Count Rate Ratio Monitoring for Fuel Loading

Objective

Verify the neutron monitoring data obtained during initial fuel loading is consistent with calculations showing the predicted response and, for plants subsequent to the first plant, with data obtained during a previous similar fuel loading.

Prerequisites

- Temporary (if temporary instrumentation is implemented to satisfy the test requirements) and plant source range nuclear instrumentation has been operational for a minimum of 60 minutes to allow the equipment to attain stable operating conditions
- The plant is prepared for initial fuel loading
- Neutron monitoring data from a previous similar initial fuel loading or calculations showing the predicted response of monitoring channels are available for evaluating monitoring data

Test Method

- Prior to inserting the first fuel assembly into the reactor vessel, obtain background count rates for each temporary (if temporary instrumentation is implemented to satisfy the test requirements) and plant source range channel
- During the insertion of each fuel assembly, continuously observe the response of at least one channel for unexpected changes in count rate
- Construct a plot of Inverse Count Rate Ratio, versus fuel loading step number, from monitoring data obtained after each fuel assembly is loaded to aid in the assessment of whether fuel loading may continue
- Compare Inverse Count Rate Ratio data to predicted response using data obtained from calculations or obtained during a previous similar fuel loading for plants subsequent to the first plant

Performance Criteria

The following criteria are used to suspend core alterations until evaluated by engineering:

- An unanticipated increase in the neutron count rate by a factor of two occurs in all responding nuclear channels during any single loading step after the initial nucleus of fuel assemblies is loaded (approximately eight fuel assemblies, one of which contains a neutron source)
- An unanticipated increase in the count rate by a factor of five occurs on any individual responding nuclear channel during any single loading step after the initial nucleus of fuel assemblies is loaded (approximately eight fuel assemblies, one of which contains a neutron source)

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14.2.10.1.5 Initial Fuel Loading

Objectives

- Establish the conditions under which the initial fuel loading is to be accomplished
- Accomplish initial fuel loading in a safe manner

Prerequisites

- The nuclear design of the initial reactor core specifying the final core configuration of fuel assemblies and inserts is completed.
- Preoperational testing is completed on systems specified as required for initial fuel loading.
- Preoperational testing is completed on required fuel handling tools. Tools are available, operational, and calibrated, including indexing of the manipulator crane with a dummy fuel element.
- Containment integrity is established.
- The reactor vessel is filled with water to a level at or above the center of the vessel outlet nozzles. The water is being circulated at a rate to provide uniform mixing.
- The boron concentration in the reactor coolant equals or exceeds the concentration required by the plant Technical Specifications for refueling. Core moderator chemistry conditions (particularly boron concentration) are prescribed in the core loading procedure document and are verified periodically by chemical analysis of moderator samples taken prior to and periodically during core loading operations.
- Sources of unborated water to the reactor coolant are isolated.
- Temporary (if temporary instrumentation is implemented to satisfy the test requirements) and plant source range channels are operable as required to monitor changes in core reactivity.
- A surveillance program verifies that the conditions for fuel loading are maintained throughout the fuel loading program.
- Auxiliary system status is in accordance with Technical Specification requirements.
- The overall process of initial fuel loading will be supervised by a licensed senior reactor operator with no other concurrent duties.

Test Method

- Place fuel assemblies, together with inserted components (control rods, burnable poison elements, primary and secondary neutron sources), in the reactor vessel one at a time according to an established and approved sequence
- During and following the insertion of each fuel assembly and until the last fuel assembly has been loaded, the response of the neutron detectors is observed and compared to previous fuel loading data, or calculations, to verify that the observed changes in response are as expected
- Check sheets are completed at prescribed intervals verifying that the conditions required for initial fuel loading are being maintained

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- Fuel assemblies, together with inserted components (control rod assemblies, burnable poison inserts, source spider, or thimble plugging devices) are placed in the reactor vessel one at a time according to a previously established and approved sequence, which was developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The core loading procedure documents include detailed tabular check sheets that prescribe and verify the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position and orientation in the core. Multiple checks are made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components, and fuel assembly status boards are maintained throughout the core loading operation. The results of each loading step will be reviewed and evaluated before the next prescribed step is started.

- Immediately stop loading operations if a decrease in boron concentration greater than 20 ppm is determined from two successive samples of reactor coolant system water until the decrease is explained.

Performance Criteria

All fuel assemblies have been loaded into the vessel in the correct location and orientation consistent with the prespecified configuration for the initial reactor core. All fuel loading steps are documented, including the final core configuration.

14.2.10.1.6 Post-Fuel Loading Precritical Test Sequence

Objective

Specify the sequence of events constituting the precritical test program.

Prerequisites

- Plant system conditions are established as required by the individual test instructions within the precritical test sequence, as described in Subsections 14.2.10.1.7 through 14.2.10.1.20

- The systems, structures, and components required by Technical Specifications shall be operable as required for the specified plant operational mode prior to initiation of precritical testing. Preoperational and precritical tests shall be completed to confirm the operability of required plant safety systems to support precritical testing prior to the initiation of the precritical tests.

Test Method

The instructions establish the sequence for required testing after core loading, until the plant has completed precritical testing.

Performance Criteria

Performance criteria are contained in the various individual tests conducted during this time (Subsections 14.2.10.1.7 through 14.2.10.1.20).

14.2.10.1.7 Incore Instrumentation System Precritical Verification

Objectives

- Verify that the incore instrumentation thimbles have been installed correctly following initial fuel loading

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- Verify proper operation of the incore thermocouples prior to plant heatup

Prerequisites

- Initial fuel loading has been completed, all incore instrumentation thimble assemblies have been installed, and all mechanical and electrical connections have been completed, with the exception of the connections to the instruments at the incore instrumentation thimble assembly refueling disconnect.
- The plant is at ambient temperature and pressure prior to heatup for initial criticality.
- Incore instrumentation system signal processing software has been installed and is operational.

Test Method

- With the plant at ambient conditions following initial fuel loading and prior to heatup for initial criticality, make electrical continuity checks at the incore instrumentation system panel to verify proper installation and connection of the incore sensor strings.
- Obtain incore thermocouple data and compare with the measured reactor coolant system temperature to verify proper operation of the incore thermocouples and signal processing.

Performance Criterion

- Prior to plant heatup, proper connections to the incore instrumentation thimbles are verified and outputs from the incore thermocouple system are consistent with existing plant conditions.

14.2.10.1.8 Resistance Temperature Detectors-Incore Thermocouple Cross Calibration

Objectives

- Verify calibration coefficients for the resistance temperature detectors installed in the reactor coolant system.
- Determine calibration coefficients for resistance temperature detectors replaced in the reactor coolant system following hot functional testing as required.
- Verify the incore instrumentation system incore thermocouples are operating within design requirements by comparing the temperature indications to the average of the derived resistance temperature detector temperature indications.

Prerequisites

- Initial fuel loading has been completed and the reactor coolant system is filled and vented prior to heatup for initial criticality.
- Reactor coolant system resistance temperature detectors that were replaced as a result of preoperational testing are operational, and an initial alignment has been completed according to the manufacturer’s calibration data.
- The incore instrumentation system, including signal processing software, has been installed and is operational, and the preoperational testing has been completed.
- Instrumentation and data collection equipment is operational and available for logging plant data.

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Test Method

- With the reactor coolant system at ambient temperature, and at isothermal conditions at specified temperature plateaus during heatup to normal operating temperature, collect data from each resistance temperature detector installed in the reactor coolant system and each installed incore thermocouple, along with supplemental plant data.

- Using the calibration coefficients determined during hot functional testing and the manufacturer's resistance versus temperature calibration data for the replaced resistance temperature detectors, determine the best-estimate temperature of each temperature plateau from the average of the median value for each resistance temperature detector's temperature.

- On an iterative basis, recompute the best-estimate plateau temperature after removing from the average calculation the data from resistance temperature detectors whose temperature differs from the average by a predetermined amount.

- Verify or recompute calibration coefficients for each resistance temperature detector, as required, based on the final plateau average temperatures.

Performance Criteria

- For each resistance temperature detector, the adequacy of the final calibration coefficients is verified when the temperature derived from the resistance temperature detector resistance agrees with the plateau average temperatures within predetermined limits.

- The adequacy of each incore thermocouple is verified when the temperature derived from the thermocouple output agrees with the plateau average temperatures within predetermined limits.

14.2.10.1.9 Nuclear Instrumentation System Precritical Verification

Objective

Establish and determine voltage settings, trip settings, operational settings, and alarm settings on source range instrumentation prior to initial criticality.

Prerequisite

The nuclear instrumentation system is aligned according to the design requirements.

Test Method

- Calibrate, test, and verify functions using permanently installed controls and adjustment mechanisms.

- Set operational modes of the source range channels for their proper functions, in accordance with the test instructions.

Performance Criteria

- The source range channels are operable.

- Neutron count rate is greater than or equal to 0.5 cps in Mode 3 for the source range channels in proximity of the primary neutron sources.

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- Signal to noise ratio is greater than or equal to 2 for the source range channels in proximity of the primary neutron sources.

14.2.10.1.10 Setpoint Precritical Verification

Objectives

- Prior to fuel load, verify that initial values of instrumentation setpoints assumed in the design, operation, and safety analysis of the nuclear steam supply system have been installed correctly, and identify which of these are expected to be readjusted based on the results of startup testing and initial operations.

- Prior to fuel load, document final values of instrumentation setpoints assumed in the design, operation, and safety analysis of the plant and as modified by preoperational testing or reanalysis to serve as a basis for future plant operations.

Prerequisites

- Initial alignment and calibration of plant instrumentation has been completed, and initial set points are installed per applicable design documentation.

- Preoperational testing of affected plant instrumentation has been completed, and test results are documented.

Test Method

- Review applicable design documentation and generate a list of the instrumentation setpoints assumed in the design, operation, and safety analysis of the plant. Identify setpoints expected to be modified based on the results of initial startup tests and operations.

- Prior to fuel load, the results of preoperational tests, as applicable, are reviewed to verify that initial setpoints have been installed correctly. Document the results of this review for future use.

- Prior to fuel load, summarize and document the setpoint values for future plant operations.

Performance Criterion

Prior to fuel load, installed setpoint values are verified to be consistent with Technical Specifications.

14.2.10.1.11 Rod Control System

Objective

Demonstrate and document that the rod control system performs the required control and indication functions just prior to initial criticality.

Prerequisites

- The reactor coolant system is at no-load operating temperature and pressure
- The nuclear instrumentation system source range channels are aligned and operable

Test Method

- With the reactor at no-load temperature and pressure, just prior to initial criticality, verify the operation of the rod control system in various modes including tests of control rod block and inhibit functions.

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- Verify the operation of status lights, alarms, and indicators

Performance Criteria

- The rod control system performs in accordance with its design requirements
- The rod control system withdraws and inserts each rod bank
- The rod position and indication system tracks each rod bank as it is being moved
- The control banks overlap system starts and stops rod movement at the designated bank positions

14.2.10.1.12 Rod Position Indication System

Objective

Verify that the rod position indication system satisfactorily performs required indication and alarm functions for each individual rod and that each rod operates satisfactorily over its entire range of travel.

Prerequisites

- The reactor coolant system is at no-load operating temperature and pressure
- At least one reactor coolant pump is in service, with reactor coolant boron concentration not less than specified in the Technical Specifications for refueling shutdown

Test Method

Individually withdraw rod banks from the core and reinsert them, according to the test procedure. Record rod position sensor output signals, and rod position readouts and group step counters in the main control room.

Performance Criterion

The rod position indication system performs the required indication and alarm functions as discussed in Subsection 7.7.1.3, and each rod operates over its entire range of travel.

14.2.10.1.13 Control Rod Drive Mechanisms

Objectives

- Demonstrate operation of each control rod drive mechanism under hot standby conditions
- Provide verification of coil stack assembly current profile

Prerequisites

- The reactor coolant system is in hot standby
- Rods are fully inserted
- Nuclear instrumentation channels are available
- A fast-speed oscillograph, or equivalent, to monitor test parameters is available

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Test Method

- With the reactor core installed and the reactor in hot standby condition, confirm that the rod control system power cabinets supply current profiles to the control rod drive mechanism stepping magnet coils.

- Verify operation of all control rod drive mechanisms under hot standby conditions. Record the control rod drive mechanism magnet coil currents.

Performance Criterion

The control rod drive mechanisms conform to the requirements for proper mechanism operation and timing including control rod withdrawal and insertion speeds as described in the applicable design specifications.

14.2.10.1.14 Rod Drop Time Measurement

Objectives

- Determine the rod drop time of each rod cluster control assembly under hot full-flow conditions, with the reactor at normal operating temperature and pressure.

- Verify the operability of the control rod deceleration device.

Prerequisites

- Initial core loading is completed
- Source range channels are in operation
- Rods are fully inserted
- Reactor coolant pumps are operational

Test Method

- Withdraw each rod cluster control assembly

- Interrupt the electrical power to the associated control rod drive mechanism

- Measure and record the rod drop time, and verify control rod deceleration

- Perform a minimum of three additional drops for each control rod whose drop time falls outside the two-sigma limit, as determined from the drop times obtained for each test condition

Performance Criteria

- Measured rod drop times are consistent with the design basis functional requirements and the applicable plant Technical Specifications

- The control rod is slowed by the control rod deceleration device during rod drop testing

14.2.10.1.15 Rapid Power Reduction System

Objective

Verify proper operation of the rapid power reduction system prior to power operations.

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Prerequisites

- The following systems are operable to the extent necessary to support the test: rod control system, rod position indication system, reactor trip breakers, and reactor protection system.
- The reactor is shut down, the reactor coolant system boron concentration is such that Technical Specifications requirements for shutdown margin will be met with required rod withdrawal, and all control banks are near their fully inserted positions.

Test Method

- Input signals simulating operation at the full power condition to the reactor control and protection system. Close the reactor trip breakers.
- Input signals simulating a rapid loss of load exceeding 50 percent power are input to the rapid power reduction system. Verify the response of the system.
- Demonstrate procedures for returning the plant to power following a partial trip.

Performance Criteria

- Performance of the rapid power reduction system is in accordance with Subsection 7.7.1.10.
- In response to the simulated loss of load, gripper power is interrupted to a selected grouping of control rods, so that rods drop freely into the core.
- Gripper power to only those control rods selected for drop is interrupted.
- Procedures for returning the plant to power operations without a reactor trip are verified.

14.2.10.1.16 Process Instrumentation Alignment

Objective

Align ΔT process instrumentation under isothermal conditions prior to initial criticality.

Prerequisites

- Reactor coolant pumps are operating
- The reactor coolant system average temperature is at the hot no-load average temperature

Test Method

- Align ΔT according to test instructions at isothermal conditions prior to criticality

Performance Criterion

The indicated values for reactor coolant system ΔT under isothermal conditions are within the limits of the applicable design requirements.

14.2.10.1.17 Reactor Coolant System Flow Measurement

Objectives

- Prior to initial criticality, verify that the reactor coolant system flow rate is sufficient to permit operation at power.

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Prerequisites

- The core is installed and the plant is at normal operating temperature and pressure.
- Special instrumentation is installed and calibrated for obtaining reactor coolant flow data.

Test Method

- Prior to initial criticality, measure the reactor coolant flow measurement parameters with all four coolant pumps in operation. Estimate the reactor coolant flow rate using these data.

Performance Criterion

The estimated reactor coolant flow rate from data taken prior to initial criticality equals or exceeds 90 percent of the minimum value required by the plant Technical Specifications for full power operation.

14.2.10.1.18 Reactor Coolant System Flow Coastdown

Objectives

- Measure the rate at which reactor coolant loop flow and pump speed changes, subsequent to tripping all reactor coolant pumps.

Prerequisites

- Required component testing and instrument calibration are complete
- Required electrical power supplies and control circuits are operational
- The reactor core is installed, and the plant is at normal operating temperature and pressure with all reactor coolant pumps running

Test Method

- Record loop flow, pump speeds following the trip of all reactor coolant pumps

Performance Criterion

The loop flows and pump speed data are obtained for verification of the complete loss of forced reactor coolant flow in Subsection 15.3.2.

14.2.10.1.19 Pressurizer Spray Capability and Continuous Spray Flow Verification

Objectives

- Establish the optimum continuous spray flow rate
- Determine the effectiveness of the normal control spray

Prerequisites

- The reactor coolant system is at no-load operating temperature and pressure.
- All reactor coolant pumps are operating.

Test Method

- While maintaining constant pressurizer level, adjust spray bypass valves until a minimum flow is achieved that maintains the temperature difference between the spray line and the pressurizer within acceptable limits.

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- With the pressurizer heaters de-energized, fully open both spray valves, and record the time to lower the pressurizer pressure a specified amount.

Performance Criteria

- The spray bypass valves are throttled so that the minimum flow necessary to keep the spray line warm is achieved.

- The pressurizer pressure response to the opening of the pressurizer spray valves is within design basis functional limits as specified in the applicable pressure control system design specification documentation.

14.2.10.1.20 Feedwater Valve Stroke Test

Objective

Verify proper operation of the main and startup feedwater control valves prior to the start of power operations.

Prerequisites

- Preoperational testing of the feedwater control systems has been completed
- Main and startup feedwater pumps are off
- Initial fuel loading has been completed prior to initial criticality.

Test Method

For each main and startup feedwater flow control valve, the following tests are performed:

- Using simulated signals for several valve demand positions covering the range from fully closed to fully open, verify the actual valve position to be within the prescribed tolerances of the demand signal.

- For selected valve position changes, measure the time required from the initiation of the demand signal until the valve reaches the final position. Typical demands changes are the following: fully closed to fully open, fully open to fully closed, 25 percent open to 75 percent open, and 75 percent open to 25 percent open.

- Use simulated signals to vary the valve position over the full range of travel and verify the valve overshoot/undershoot is less than prescribed tolerances. Typical demand changes are 3% to 10% over the full range of valve travel.

Performance Criterion

The main and startup feedwater valves operate as described in Subsection 7.7.1.8 and appropriate design specifications including:

- The time between the initiation of the demand signal and the final valve position for each of the demand changes is within specified ranges as discussed in applicable design specifications.

14.2.10.2 Initial Criticality Tests

Initial criticality testing is described in this section. Following completion of the core loading and precriticality testing, the plant is brought to initial criticality, according to the test procedures in Subsection 14.2.10.2.1.

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14.2.10.2.1 Initial Criticality Test Sequence

Objective

Define the sequence of tests and operations to bring the core to initial criticality.

Prerequisite

Plant system conditions are established as required by the individual test instructions within this sequence.

Test Method

An individual test instruction will establish the plant conditions required for initial criticality.

Performance Criteria

Relevant performance criteria are provided in each of the test procedure abstracts.

14.2.10.2.2 Initial Criticality

Objective

Achieve initial criticality in a controlled manner.

Prerequisites

- The nuclear instrumentation is verified to be operating properly (See Subsection 14.2.10.2.3)
- The reactor coolant system temperature and pressure are stable at the normal hot no-load values
- Control rod banks are inserted, and shutdown rod banks are withdrawn
- The reactor coolant system boron concentration is sufficiently high so the reactor is shut down by at least 1000 pcm with all banks withdrawn

Test Method

- Accomplish initial criticality by the controlled withdrawal of the rods using the same rod withdrawal sequence used for normal plant startup, followed by the dilution of the reactor coolant system boron concentration.
- At preselected points during rod withdrawal and/or boron dilution, gather data to plot the inverse count rate ratio to monitor the approach to critical evolution for reactivity monitoring.
- As criticality is approached, slow or stop dilution rate to allow criticality to occur during mixing or by withdrawal of rods that have been slightly inserted for control.

Performance Criterion

The reactor is critical.

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14.2.10.2.3 Nuclear Instrumentation System Verification

Objective

Establish and determine voltage settings, trip settings, operational settings, alarm settings, and overlap of channels on source and intermediate range instrumentation, from prior to initial criticality and during initial criticality.

Prerequisite

The nuclear instrumentation system is aligned according to the design requirements.

Test Method

- Calibrate, test, and verify functions using permanently installed controls and adjustment mechanisms.
- Set operational modes of the source and intermediate range channels for their proper functions, in accordance with the test instructions.

Performance Criteria

- The nuclear instrumentation system operates in accordance with the design basis functional requirements.
- The nuclear instrumentation system demonstrates an overlap of indication between the source and intermediate range instrumentation.
- The nuclear instrumentation minimum neutron count rate and noise to signal ratio are within appropriate design specifications.

14.2.10.2.4 Post-Critical Reactivity Computer Checkout

Objective

Demonstrate proper operation of the reactivity computer through a dynamic test using neutron flux signals.

Prerequisites

- The reactor is critical with the neutron flux level within the range for low-power physics testing
- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- The reactivity computer is installed, checked out, and operational, and input flux signals are representative of the core average neutron flux level
- The controlling rod bank is positioned in such a way that the required reactivity insertion can be made by rod motion alone
- The systems, structures, and components required by Technical Specifications shall be operable as required for the specified plant operational mode prior to initiation of precritical, low power physics, and power ascension testing. Verification of proper operation of

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source-range and intermediate-range excore nuclear instrumentation and associated alarms and protective functions in Startup Test Subsection 14.2.10.2.3 shall be completed prior to initiation of this startup test.

Test Method

- By control rod motion, add positive reactivity to the core as directed in the test procedure.
- During the resultant increase in flux level, make two independent measurements of core reactivity; one using the reactivity computer, and one using an analysis of the rate of change of flux level (for example, reactor period or doubling time).

Performance Criterion

Each measurement deviation between the two independent sources of reactivity is within design tolerances. Adjustment and recalibration or repair of the reactivity computer may be required if the deviation between the two independent sources of reactivity is not within design tolerances.

14.2.10.3 Low Power Tests

Following successful completion of the initial criticality tests, low power tests are conducted, typically at power levels less than 5 percent, to measure physics characteristics of the reactor system and to verify the operability of the plant systems at low power levels.

14.2.10.3.1 Low-Power Test Sequence

Objective

Define the sequence of tests and operations that constitutes the low-power testing program.

Prerequisite

Plant system conditions are established as required by the individual test instructions within this sequence.

Test Method

Individual test instruction will establish the plant conditions required for and during the low-power testing program following initial criticality.

Performance Criteria

Relevant performance criteria are provided in each of the test procedure abstracts.

14.2.10.3.2 Determination of Physics Testing Range

Objectives

- Determine the reactor power level at which the effects from fuel heating are detectable
- Establish the range of neutron flux in which zero power reactivity measurements are to be performed

Prerequisites

- The reactor is critical, and the neutron flux level is below the expected level of nuclear heating

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- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- The reactivity computer is installed, checked out, and operational, and input flux signals are representative of the core average neutron flux level
- The controlling rod bank is positioned in such a way that the required reactivity insertion can be made by rod motion alone

Test Method

- Withdraw the control rod bank and allow the neutron flux level to increase until nuclear heating effects are indicated by the reactivity computer
- Record the reactivity flux level and the corresponding intermediate range channel currents at which nuclear heating occurs
- Multiply the measured reactivity flux level by 0.3 to determine the maximum value for the zero power testing range

Performance Criterion

The zero power testing range is determined.

14.2.10.3.3 Boron Endpoint Determination

Objective

Determine the critical reactor coolant system boron concentration appropriate to an endpoint rod configuration.

Prerequisites

- The reactor is critical, and the neutron flux level is within the range for low-power physics testing
- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- Instrumentation and equipment used to measure and compute reactivity is installed, checked out, and operational, with input flux signals representative of the core average neutron flux level
- The controlling rod bank is positioned in such a way that limited reactivity insertion will be required to achieve the endpoint condition

Test Method

- Move the rods to the desired endpoint configuration without boron concentration adjustment
- Directly measure the just-critical boron concentration by chemical analysis

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- Measure and convert the change in reactivity and the reactor coolant temperature difference from program to an equivalent change in boron concentration
- Add the changes to the just-critical boron concentration to yield the endpoint for the given rod configuration

Performance Criterion

The measured value for the boron endpoint is consistent with the design value within design limits as specified in the Technical Specifications.

14.2.10.3.4 Isothermal Temperature Coefficient Measurement

Objectives

- Determine the isothermal temperature coefficient
- Calculate the moderator temperature coefficient

Prerequisites

- The reactor is critical, and the neutron flux level is within the range for low-power physics testing
- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- Instrumentation and equipment used to measure and compute reactivity is installed, checked out, and operational, with input flux signals representative of the core average neutron flux level
- The controlling rod bank is positioned near fully withdrawn

Test Method

- Vary reactor coolant system temperature (heatup/cooldown) while maintaining rods and boron concentration constant
- Monitor reactivity results and determine the isothermal temperature coefficient
- Calculate the moderator temperature coefficient using the isothermal temperature coefficient and design values

Performance Criterion

- The measured value for the moderator temperature coefficient is more negative than the Technical Specification limit

14.2.10.3.5 Bank Worth Measurement

Objective

Validate design calculations of the reactivity worth of the rod cluster control banks.

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Prerequisites

- The reactor is critical and the neutron flux level is within the range for low-power physics testing
- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- Instrumentation and equipment used to measure and compute reactivity is installed and operational, with input flux signals representative of the core average neutron flux level

Test Method

- One of the following methods will be used to measure the worth of all of the individual control rod banks:
  - A bank is stepwise inserted into the core from fully withdrawn and the worth is measured using the reactivity computer
  - Exchange bank with another bank measured as above, with the worth determined from the critical positions and the worth of the reference bank

Performance Criteria

- The measured value for the individual bank worth is consistent with the design value within specified limits.
- The sum of the measured bank worth is consistent with the design value within the assumed uncertainty used in the shutdown margin calculation

14.2.10.4 Power Ascension Tests

After low power testing is completed, testing is performed at specified elevated power levels to demonstrate the facility operates in accordance with design during normal steady-state operations, and to the extent practical, during and following anticipated transients. During power ascension, tests are performed to obtain operational data and to demonstrate the operational capabilities of the plant.

14.2.10.4.1 Test Sequence

Objective

Define the sequence of operations, beginning at approximately 5 percent rated thermal power, that constitutes the power ascension testing program.

Prerequisite

Plant system conditions are established, as required, by the individual test instruction within this sequence.

Test Method

Present the sequence of operations and tests, along with instructions, specific plant conditions, and test procedures.

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Performance Criteria

Relevant performance criteria are provided in each of the test procedures.

14.2.10.4.2 Incore Instrumentation System

Objectives

- Obtain data for incore thermocouple and flux maps at various power levels during ascension to full power determine flux distributions and verify proper core loading and fuel enrichments.

Prerequisites

- Incore instrumentation system signal processing software is installed and operational
- For incore thermocouple and flux mapping, the plant is at various power levels greater than approximately 20 percent of rated thermal power

Test Method

- With the plant at approximate power levels of 25, 50, 75 and 100 percent of rated thermal power, obtain data from the incore instrumentation system and process to produce incore thermocouple and flux maps. (Actual power levels will be specified in the power ascension program test sequence.)
- Use data from the incore maps to verify that core power distribution is consistent with design predictions and the limits imposed by the plant Technical Specifications, including detection of potential fuel loading errors, and to calibrate other plant instrumentation. Refer to Technical Specifications Section 3.2, Power Distribution Limits.

Performance Criteria

- Core power peaking factors derived from the incore data are consistent with design predictions and the limitations of the plant Technical Specifications
- Data required for calibration of other plant instrumentation are obtained

14.2.10.4.3 Nuclear Instrumentation System

Objective

Establish and determine voltage settings, trip settings, operational settings, alarm settings, and overlap of channels on intermediate range and power range instrumentation from zero power to at or near full rated thermal power.

Prerequisite

The nuclear instrumentation system is aligned according to the design requirements.

Test Method

- Calibrate, test, and verify functions using permanently installed controls and adjustment mechanisms
- Set operational modes of the intermediate range and power range channels for their proper functions, in accordance with the test instructions

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Performance Criteria

- The nuclear instrumentation system operates in accordance with the design basis functional requirements.
- The nuclear instrumentation system demonstrates an overlap of indication between the intermediate and power range instrumentation.

14.2.10.4.4 Setpoint Verification

Objective

During power ascension, document final values of instrumentation setpoints as modified by initial startup testing, operations, or reanalysis to serve as a basis for future plant operations.

Prerequisites

- Initial alignment and calibration of plant instrumentation have been completed, and initial set points are installed per applicable design documentation
- Preoperational and startup testing of affected plant instrumentation has been completed, and test results are documented
- The results of the precritical verification of the instrument setpoints are completed and documented

Test Method

- Identify setpoints modified based on the results of initial startup tests and operations
- During power ascension testing, readjust specific setpoints noted for readjustment on the data sheets if required. Record final setpoint values.

Performance Criterion

Setpoint changes based on initial startup testing and operations are documented for future reference.

14.2.10.4.5 Startup Adjustments of Reactor Control Systems

Objectives

- Determine the adequacy of the reactor coolant system programmed T_{avg}
- Obtain plant data during power ascension which would provide the basis for any required changes to the turbine pressure versus turbine power program

Prerequisites

- The reactor coolant system is at no-load operating temperature and pressure
- The reactor coolant system temperature is being controlled by the steam dump valves with the turbine offline

Test Method

- Verify the reactor coolant temperature is controlled at an average temperature that is consistent with the T_{avg} program calculated based on the calorimetric power when the turbine is online.

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- Obtain system temperature and steam pressure data at steady-state conditions for zero rated thermal power and at hold points during power escalations

- Evaluate the turbine pressure versus turbine power curve against plant data at intermediate and full powers and perform calibration based on calorimetric power. Final calibration is performed at the full power testing plateau.

- Compare the full power steam pressure to the full power design value to determine adequacy of the programmed T_{avg}.

Performance Criterion

The full power T_{avg} program does not exceed the design T_{avg} limit to obtain the design steam pressure.

14.2.10.4.6 Not Used

14.2.10.4.7 Axial Flux Difference Instrumentation Calibration

Objectives

- Calibrate the power range nuclear instrumentation signals used as axial flux difference (delta flux) input to the reactor protection system

- Calibrate instrumentation used to display and monitor axial flux difference

Prerequisites

- The reactor is at a power level greater than 50 percent of rated thermal power
- The incore instrumentation system is available for obtaining incore power distribution data
- A preliminary calibration of the axial flux difference indication instrumentation is completed

Test Method

- Using control rod movement, xenon redistribution, or a combination of both, vary the axial power distribution of the core over a specified range of interest. At selected values of indicated axial flux difference, obtain reactor thermal power data along with the outputs from the nuclear instrumentation power range channels and the incore instrumentation system. (For the first plant, a minimum of three data sets will be taken; subsequent cores may require less.)

- Calibrate signals from the nuclear instrumentation power range channels based on incore power distribution and thermal power data.

Performance Criterion

Axial flux difference signals, derived from the nuclear instrumentation power range detectors and input to the reactor protection system, display, and monitoring instrumentation, reflect actual incore power distribution within specified limits.

14.2.10.4.8 Primary and Secondary Chemistry

Objective

Verify proper water quality in the reactor coolant system and secondary coolant system.

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Prerequisite

The plant is at the steady-state condition at approximately 0, 25, 50, 75, and 100 percent rated thermal power.

Test Method

Analyze samples to determine the chemical and radiochemical concentrations.

Performance Criterion

The chemical and radiochemical control systems maintain the water chemistry within the applicable guidelines as discussed in Subsections 5.2.3.2 and 10.3.5.

14.2.10.4.9 Process Measurement Accuracy Verification

Objectives

- Measure the temperature variation in the reactor coolant loops resulting from non-uniform flow effects such as streaming
- Measure the sensitivity of the excore detectors to variations in control bank position and reactor coolant loop cold leg temperature

Prerequisites

- For the reactor coolant loop temperature measurements:
  - Special temperature measuring equipment, including recording and indicating instrumentation, is installed, as required, on the reactor coolant loops hot and cold leg piping
  - The reactor is at a stable power level of approximately 0, 50, 75 and 100 percent of rated thermal power
- For the excore detector measurements:
  - The reactor is at a stable power level of approximately 25, 50 and 100 percent of rated thermal power

Test Method

- For the reactor coolant loop temperature measurements, at each power level:
  - Measure reactor power level, using calorimetric data
  - Simultaneously, measure the hot and cold leg temperatures, using normal plant instrumentation and any other required instrumentation
- For the excore detector tests, with the reactor at constant power level:
  - Measure the response of the excore detectors as selected control banks are moved over prescribed ranges of travel
  - Measure excore detector response as the reactor coolant cold leg temperature is varied over a prescribed range

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– Simultaneously, for each of the preceding measurements, obtain calorimetric data to verify reactor power level

Performance Criteria

- Uncertainties in reactor coolant loop temperature measurements resulting from non-uniform flow effects such as streaming are consistent with allowances used in the plant safety analyses.

- Uncertainties in excore detector response resulting from control rod motion and reactor coolant loop cold leg temperature changes are consistent with allowances used in the plant safety analyses.

14.2.10.4.10 Process Instrumentation Alignment at Power Conditions

Objective

Align ΔT and T_{avg} process instrumentation at power conditions.

Prerequisites

- Reactor coolant pumps are operating.
- The reactor system is operating at the required power level.

Test Method

- Align ΔT and T_{avg} according to test instructions at approximately 75 percent rated thermal power. Extrapolate the 75 percent data to determine ΔT and T_{avg} values for the 100 percent plateau.

- At or near 100 percent rated thermal power, check the alignment of the ΔT and T_{avg} channels for agreement with the results of the thermal power measurement.

Performance Criterion

The indicated values for reactor coolant system T_{hot}, T_{cold}, T_{avg}, and ΔT at or near full thermal power are within the limits of the applicable design requirements, as discussed in Section 5.1.

14.2.10.4.11 Reactor Coolant System Flow Measurement at Power Conditions

Objective

At power, verify that the reactor coolant flow equals or exceeds the minimum value required by the plant Technical Specifications.

Prerequisites

- The reactor is at power levels greater than 75 percent and up to and including 100 percent of rated thermal power

- Special instrumentation required for measuring reactor thermal power and reactor coolant inlet and outlet temperatures is installed and calibrated

Test Method

With the reactor at steady-state power greater than 75 percent and up to and including 100 percent of rated thermal power, measure the reactor thermal power and coolant inlet and outlet temperatures.

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Determine the reactor coolant flow rate using the data in conjunction with hydraulic analysis of differential pressures at different locations in the reactor coolant system.

Performance Criterion

The reactor coolant system flow determined from the measurements at approximately 100 percent rated thermal power equals or exceeds the minimum value required by the plant Technical Specifications.

14.2.10.4.12 Steam Dump Control System

Objective

Verify automatic operation of the T_{avg} steam dump control system, demonstrate controller setpoint adequacy, and obtain final settings from steam pressure control of the condenser dump valves.

Prerequisites

- Steam dump control system is aligned and calibrated to initial settings
- Plant is at no-load temperature and pressure
- Condenser vacuum is established
- Reactor is critical

Test Method

- Increase reactor power to less than 10 percent rated thermal power by rod withdrawal and steam dump to condenser to demonstrate setpoint adequacy
- Adjust the pressure controller setpoint prior to switching to T_{avg} control mode, which modulates the steam condenser dump valves to obtain the new pressure.
- Enable the steam dump arming logic and transfer to the T_{avg} control mode to demonstrate the closed loop response of the load rejection controller.

Performance Criteria

- After placing the steam dump controller system in the T_{avg} control mode, the load rejection controller demonstrates closed loop control wherein the steam dump valves respond properly to changes in T_{avg}.
- The steam header pressure controller responds to maintain a stable pressure.

14.2.10.4.13 Steam Generator Level Control System

Objective

Verify the stability of the automatic steam generator level control system by introducing simulated transients at various power levels during escalation to full power.

Prerequisites

- The reactor is critical and stable at various power levels during the power escalation test program. (Typical power levels are 30, 75 and 90 percent of full rated thermal power)
- The steam generator level control system is checked and calibrated
- Steam generator alarm setpoints are set for each generator

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Test Method

- At each power level, with the steam generator control system in manual mode, simulate level transients by changing the level setpoint. Verify the steam generator level control response when the control system is returned to automatic control.

- During the recovery from a simulated steam generator level transient, verify steam generator level control response is within design requirements for overshoot or undershoot to the new level, time required to achieve the new level, and error between the actual level and the control setpoint.

- Verify the main feedwater control valves open and stabilize at various power plateaus in accordance with design requirements.

Performance Criterion

- No manual intervention of steam generator level control system is required to prevent steam generator water level from reaching the high or low steam generator level protection setpoints.

14.2.10.4.14 Radiation and Effluent Monitoring System

Objectives

- For monitors that:
  - Are used for establishing conformance within the safety limits or limiting conditions for operation that are included in the Technical Specifications, or
  - Are classified as engineered safety features, or are relied on to support operation of the engineered safety features within design limits, or
  - Are assumed to function or for which credit is taken in the accident analysis of the facility, and
  - Are used to process, store, control, or limit the release of radioactive materials
- The objectives are:
  - Verify the calibration of the process and effluent radiation monitor against an acceptable standard
  - Establish baseline activity and background levels
  - Demonstrate that process and effluent radiation monitoring systems respond correctly by performing independent analyses

Prerequisites

- The plant is stable at the desired power level
- The sampling systems for the process and effluent radiation monitoring systems are operable

Test Method

- Perform calibrations with the use of radioactive sources to verify proper operation of the monitors and detectors

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- Collect and analyze samples with laboratory instruments, and compare the results from the process and effluent monitor to verify proper monitor operation
- Establish background levels at low power (less than 5 percent rated thermal power)
- Establish background levels and baseline activity levels determined by sampling at 100 percent rated thermal power to monitor the buildup of activity

Performance Criteria

- Radiation monitors are calibrated against radioactive standards
- Baseline activities are established

14.2.10.4.15 Ventilation Capability

Objective

Verify that heating, ventilation, and air conditioning systems for the containment and areas housing engineered safety features continue to maintain design temperatures.

Prerequisite

The plant is operating at or near the desired power (0, 50, and 100 percent of rated power).

Test Method

- Record temperature readings in specified areas while operating with normal ventilation lineups
- Record temperature readings in specified areas while operating the designed minimum number of heating ventilation and air conditioning components consistent with existing plant conditions
- Record surface concrete temperatures adjacent to the high temperature piping penetrations and at selected locations on the concrete shielding (at 100 percent rated thermal power only)

Performance Criterion

The heating, ventilation and air conditioning systems for the containment and areas housing engineered safeguards features perform as designed.

14.2.10.4.16 Biological Shield Survey

Objectives

- Document the radiation levels in accessible locations of the plant outside of the biological shield while at power
- Obtain baseline radiation levels for comparison with future measurements of level buildup with operation

Prerequisites

- Radiation survey instruments are calibrated
- Background radiation levels are measured in designated locations prior to initial criticality
- The plant is stable at the applicable power level

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Test Method

Measure gamma and neutron radiation dose rates at designated locations at approximately 25, 50, 75, and 100 percent rated thermal power.

Performance Criterion

Radiation levels are acceptable for full-power operation and consistent with design expectations.

14.2.10.4.17 Thermal Power Measurement and Statepoint Data Collection

Objective

Obtain thermal power measurement and statepoint data at selected power levels during the power ascension testing program, typically at 25, 50, 75, and 100 percent of rated thermal power.

Prerequisites

- The following equipment is installed and is operational: sensors for measuring steam generator feedwater temperature, differential pressure measuring devices for determining feedwater flow to each steam generator, and pressure gauges to measure steam pressure at steam generator outlets.
- The pressurizer pressure and level control system, and the steam generator level control system are in automatic mode.
- Instrumentation and data collection equipment is available for logging supplemental plant data.
- Reactor power is stable at the required level.

Test Method

The required data are obtained using installed plant equipment, special test equipment, and the plant data processing equipment. These data are subsequently used to determine reactor thermal power and assess the performance of the plant.

Performance Criterion

Reactor thermal power is stable at each power level and at the rated level at full power conditions. Operability of the pressurizer pressure and level control systems not previously verified as part of reactor coolant system preoperational testing (Subsection 14.2.9.1.1) is demonstrated.

14.2.10.4.18 Dynamic Response

Objectives

Demonstrate during power range testing that the stress analysis for selected systems and components, under transient conditions is within design functional requirements. Portions of systems that meet the selection criteria for Subsection 14.2.9.1.7 for dynamic effects testing, but were not tested because system conditions during hot functional testing are not conducive to prototypical systems conditions, are tested.

Prerequisites

- Temporary instrumentation is installed, as required, to monitor the deflections of components under test and the occurrence of water hammer noise and vibration.

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- Points are monitored and baseline data are established.

Test Method

- Record deflection measurements during various plant transients.
- Monitor for the occurrence of water hammer noise and vibration.

Performance Criteria

- The movements due to flow-induced loads do not exceed the stress analysis of the monitored points. See Subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to vibration.
- Flow-induced movements and loads do not cause malfunctions of plant equipment or instrumentation.
- No effects due to water hammer are detected.

14.2.10.4.19 Reactor Power Control System

Objective

Demonstrate the capability of the reactor power control system to respond to input signals.

Prerequisites

- The reactor is at equilibrium at the power level specified by the startup test program reference document.
- Setpoints and controls for the pressurizer, steam generator steam dump, and feedwater pump are checked and are set to proper values.

Test Method

- Vary T_{avg} from the T_{ref} setpoint to verify the transient recovery capabilities of the automatic reactor power control system.
- Vary the axial flux difference setpoint to verify the capabilities of the automatic axial offset control and the interaction with the automatic reactor temperature control.

Performance Criteria

- T_{avg} returns to the T_{ref} setpoint, within pre-specified limits and without manual intervention.
- Axial flux difference returns to the setpoint within pre-specified limits without manual intervention.

14.2.10.4.20 Load Swing Test

Objective

Verify nuclear plant transient response, including automatic control system performance, when 10 percent step-load changes are introduced to the turbine-generator at 30, 75, and 100 percent rated thermal power levels.

Prerequisite

The plant is operating in a steady-state condition at the desired thermal power level.

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Test Method

Change the turbine-generator output as rapidly as possible to achieve a step 10 percent load increase or decrease. Monitor and record plant parameters of reactor power, reactor coolant system temperature, pressurizer pressure and level, and steam generator pressure and level during the load transients. Core power should not exceed 100-percent power as indicated by the excore nuclear instrumentation.

Performance Criterion

The primary and secondary control systems, with no manual intervention, maintain reactor power, reactor coolant system temperatures, pressurizer pressure and level, and steam generator levels and pressures within acceptable ranges during and following the transient. Control system response is reviewed and compared to the control system setpoint and performance analysis, and adjustments to the control systems are made, if necessary, prior to proceeding to the next power plateau.

14.2.10.4.21 100 Percent Load Rejection

Objective

Demonstrate the ability of the AP1000 plant to accept a 100 percent load rejection from full power.

Prerequisites

- The plant is operating at a stable power level of approximately 100 percent rated thermal power. Reactor and turbine control systems are in the automatic mode of operation. Plant temperatures, pressures, levels, and flow rates are within their normal range for full-power operation.
- Startup testing of the reactor and turbine control and protection systems is completed, and final setpoints are installed according to applicable plant technical manuals.
- The incore instrumentation system, including signal processing software, is operational, and all preoperational and startup testing is completed.
- Instrumentation and data collection equipment is operational and available for logging plant data.
- Special test instrumentation is installed and operational as required to augment normal data logging ability.

Test Method

- With the plant at nominal full-power steady-state conditions, to effect a rejection of 100 percent load, manually place the main step-up transformer high side breaker in the trip position.
- Prior to the load rejection, and until the plant stabilizes at the lower power level, record key plant parameters using the data display and processing system and special test instrumentation. The key plant parameters include plant temperatures, pressures, levels and flow rates for the primary and secondary systems.

Performance Criteria

- The plant is capable of accepting a 100 percent load rejection from full rated thermal power without reactor trip or operation of the steam generator relief valves or pressurizer safety valves.

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- The turbine speed does not exceed 108% of rated speed.
- The turbine is capable of continued stable operation at the minimum house loads.

14.2.10.4.22 Not Used

14.2.10.4.23 Hot Full Power Boron Endpoint

Objective

Measure the reactor coolant system critical boron concentration at beginning of cycle life for the all rods out, hot full power, xenon equilibrium condition.

Prerequisites

- The reactor is operating at approximately 100 percent of full licensed power and has been at that power for a sufficient time to reach xenon equilibrium.
- The reactor power level and reactor coolant system boron concentration and temperature are stable, and control and shutdown rod banks are in the near fully withdrawn position.
- Current core burnup data are available.

Test Method

- During the power ascension test program, and, as soon as practicable after achieving xenon equilibrium at full licensed power, obtain and analyze samples of reactor coolant for dissolved boron content.
- Using plant calorimetric and statepoint data obtained at the same time as coolant sampling, correct the measured boron concentration, as required, for control rod insertion, xenon nonequilibrium, and any difference between T_{avg} and T_{ref}.
- The resultant boron value, corresponding to the measured critical boron concentration for all rods out, hot full power, and xenon equilibrium, is compared with design predictions for the current accumulated core burnup (Figure 4.3-3).
- As permitted by the plant Technical Specifications, use the corrected measured boron concentration to renormalize the predicted curve of boron concentration as a function of core burnup.

Performance Criterion

The reactivity equivalent of the difference between measured and predicted boron concentrations is less than the design limit.

14.2.10.4.24 Manual Reactor Trip Test from 100 Percent Power and Main Generator Trip Test from 100 Percent Power

Objectives

- Verify the ability of the plant automatic control systems to sustain a main generator trip from 100 percent rated thermal power and bring the plant to stable conditions following the transient.
- Assess the dynamic response of the plant for the event that subjects the turbine to its maximum credible overspeed condition.

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- Verify the ability of the plant automatic control systems to sustain a manual reactor trip from 100 percent thermal power and bring the plant to stable conditions following the transient and determine the overall response time of the hot leg resistance temperature detectors (RTDs).

- Optimize the control systems setpoints, if necessary.

Prerequisite

The plant is operating in a steady-state condition at full rated thermal power.

Test Method

Main Generator Output Breaker Trip Test

- Trip the plant by opening the main generator breaker.
- Monitor and record selected plant parameters.
- If necessary, adjust the control systems setpoints to obtain optimal response.

Manual Reactor Trip Test

- Manually trip the reactor.
- Monitor plant conditions and verify the plant is stabilized and maintained at the no-load conditions.
- Calculate the overall response time of the hot leg temperature measurement.
- If necessary, adjust the control system setpoints to obtain optimal response.

Performance Criteria

Main Generator Output Breaker Trip Test

- Following the opening of the main generator breaker while at 100 percent rated thermal power:

  – The primary and secondary control systems and operator actions can stabilize and maintain reactor coolant system temperature, pressurizer pressure and level, and steam generator levels at the reduced power conditions.
  – The primary and secondary safety valves are not challenged.
  – The turbine speed does not exceed 108 percent of rated speed.

Manual Reactor Trip Test

- Following the manual reactor trip while at 100 percent rated thermal power:

  – The primary and secondary control systems can stabilize and maintain reactor coolant system temperature, pressurizer pressure and level, and steam generator levels at the no-load operating temperature and pressure.
  – The primary and secondary safety valves are not challenged.
  – The overall hot leg response times are verified to be less than or equal to the values determined by the safety analysis method.

14.2.10.4.25 Thermal Expansion

Objective

Demonstrate that essential nuclear steam supply system and balance-of-plant components can expand without obstruction and that the expansion is in accordance with design. Also, during cooldown, the components return to their approximate baseline cold position. Testing is conducted to resolve discrepancies from hot functional testing as in Subsection 14.2.9.1.1, and to test

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modifications made since hot functional testing was completed. Portions of systems that meet the selection criteria for Subsection 14.2.9.1.7 for thermal dynamic testing, but were not tested because system conditions during hot functional testing are not conducive to prototypical system conditions are tested.

Prerequisite

Temporary instrumentation is installed, as required, to monitor the deflections for the components under test.

Test Method

For the components tested, the following apply:

- During plant heatup and cooldown, record deflection data.
- Verify support movements by recording hot and cold positions.

Performance Criteria

Thermal expansion testing is performed in accordance with Subsection 3.9.2.1.2. For the components tested, the following apply:

- There is no evidence of blocking of the thermal expansion of piping or component, other than by installed supports, restraints, and hangers.
- Spring hanger movements must remain within the hot and cold setpoints and supports must not become fully retracted or extended.
- Piping and components return to their approximate baseline cold position.

14.2.10.4.26 Loss of Offsite Power

Objective

Demonstrate plant response following a plant trip with no offsite power available.

Prerequisites

- The plant is at minimum power level supplying normal house loads through the unit auxiliary transformers.
- The unit is disconnected from the electrical grid.

Test Method

- The turbine is tripped and the generator output breaker opens, removing ac power from the unit auxiliary transformers.

Performance Criteria

- The reactor trips.
- Both standby diesel generators start and pick up the required loads.
- Class 1E dc and non-1E dc loads are uninterrupted and are provided by the battery subsystems.
- The primary plant is placed in a stable condition.

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14.2.10.4.27 Feedwater Heater Loss and Out of Service Test

Objective

Demonstrate the plant response to the loss of one of the feedwater heaters during power operation due to single failure or operator error. Demonstrate the plant response to a pair of feedwater heaters taken out of service during power operation. Verify the ability of operators to manually reduce steam flow and place a pair of feedwater heaters out of service while maintaining reactor power operation.

Prerequisites

The plant is operating in a steady-state condition at the rated thermal powers described.

Test Method

LOSS OF FEEDWATER HEATER

- With the plant operating at 50% power, isolate the extraction steam supply to one of the main feedwater heaters.
- With the plant operating at 90% power, isolate the extraction steam supply to one of the main feedwater heaters.

FEEDWATER OUT OF SERVICE TEST

- The operators calculate the appropriate steam flow reduction which will maintain the plant at the desired thermal load after the heaters have been taken out of service.
- Reduce steam flow by the appropriate amount and allow plant conditions to reach a new steady-state (approximately 10 minutes).
- Take a pair of feedwater heaters out of service.

Performance Criteria

The plant control systems properly respond to the loss of a main feedwater heater, without reactor or turbine trip.

The operator successfully removes a pair of feedwater heaters from service without causing a reactor trip.

14.2.10.4.28 Remote Shutdown Workstation

Objective

Demonstrate the ability of the operators to conduct a remote shutdown of the plant during a simulated main control room evacuation.

Prerequisites

Approved operation procedures for performing a remote shutdown is available. Communication exists between the control room and the remote shutdown room. Procedures for transferring control back to the main control room are available if an emergency or unsafe condition develops during the testing that cannot be managed by the shutdown crew.

The plant is operating in a steady-state condition at 10-20 percent of power.

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Test Method

- Using the appropriate operating procedures, the operators transfer control of the plant from the main control room to the remote shutdown workstation.
- From the remote shutdown workstation, the operators bring the plant to hot standby, and maintain hot standby conditions for at least 30 minutes.
- From the remote shutdown workstation, the operators lower the reactor coolant system pressure and temperature to the appropriate conditions, and place the normal residual heat removal system into service. The normal residual heat removal system, in conjunction with the component cooling water system and service water system are used to cool the plant at least 50°F without exceeding prescribed cooldown limits.

Performance Criteria

The operators successfully demonstrate the ability transfer control of the plant to the remote shutdown workstation, shut down the reactor, maintain hot standby, and then demonstrate the ability to transition to cold shutdown conditions, while performing these operations from the remote shutdown workstation.

14.2.10.4.29 Not Used

14.2.10.4.30 Cooling Tower(s)

Objectives

- Verify proper cooling tower(s) function. Provide thermal acceptance testing of the cooling tower's heat removal capabilities.

Prerequisites

- The cooling tower(s) is structurally complete and in good operating condition.
- Circulating water system testing is complete.
- Required support systems, electrical power supplies, and control circuits are operational.

Test Method

Verify cooling tower operation at 100% power when the cooling tower maintains condenser pressure below the alarm setpoint, the turbine building closed cooling water system (TCS) heat exchangers maintain cooling water below the alarm setpoints, and the condenser air removal system (CMS) maintains adequate cooling of the condenser vacuum pumps.

Performance Criteria

The cooling tower(s) perform as described in Subsection 10.4.5.2.3 and in appropriate design specifications.

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14.3 Certified Design Material

This section provides the selection criteria and processes used to develop the AP1000 Certified Design Material (CDM). This document provides the principal design bases and design characteristics that are certified by the 10 CFR Part 52 rulemaking process and included in the design certification rule.

The top-level design information in the Certified Design Material is extracted directly from the AP1000 design information. Limiting the certified design contents to top-level information reflects the tiered approach to design certification endorsed by the U.S. Nuclear Regulatory Commission (see References 1 through 5).

The objective of this section is to define the bases and methods that were used to develop the Certified Design Material for the AP1000. This section contains no new technical information regarding the AP1000 design.

The AP1000 Certified Design Material consists of the following:

- An introduction section which defines terms used in the Certified Design Material and lists general provisions that are applicable to all Certified Design Material entries. Also included is a list of acronyms and legends used in the Certified Design Material. (Because this material is self-explanatory, it is not discussed in this section.)

- Design descriptions for selected systems that are within the scope of the AP1000 design certification, and the applicable portions of those selected systems that are only partially within the scope of the AP1000 design certification. The Certified Design Material design descriptions delineate the principal design bases and principal design characteristics that are referenced in the design certification rule. The design descriptions are accompanied by the inspections, tests, analyses, and acceptance criteria (ITAAC) required by 10 CFR 52.47(a)(26) to be part of the design certification application. The ITAAC define verification activities that are to be performed for a facility with the objective of confirming that the plant is built and will operate in accordance with the design certification. Completion of these certified design ITAAC, together with the Combined License applicant's ITAAC for the site-specific portions of the plant, will be the basis for NRC authorization to load fuel per the provisions of 10 CFR Part 52.103.

- Design descriptions and their associated ITAAC for design and construction activities that are applicable to more than one system. Design-related processes have been included in the Certified Design Material for:

  – Aspects of the AP1000 design likely to undergo rapid, beneficial technological developments in the lifetime of the design certification. Certifying the design processes associated with these areas of the design, rather than specific design details, permits future license applicants referencing the AP1000 design certification to take advantage of the best technology available at the time of combined license application and facility construction.

  – Aspects of the design dependant upon characteristics of as-procured, as-installed systems, structures, and components. These characteristics are not available at the time of certification and, therefore, cannot be used to develop and certify design details.

  – Aspects of the seismic, structural and piping design for which detailed design has not been developed. These details are not available at the time of certification and, therefore, cannot be used to certify design details. Certifying the design processes associated with

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these design details provides the basis for future license applicants referencing the AP1000 design certification to establish and implement seismic, structural and piping design details as part of the COL application process.

- Interface requirements as defined by 10 CFR Part 52.47(a)(25). Interface requirements are defined as those which must be met by the site-specific portions of the complete nuclear power plant that are not within the scope of the certified design. These requirements define characteristics of the site-specific features that must be provided for the certified design to comply with certification commitments. AP1000 has no interfaces meeting this definition. The Certified Design Material does not include ITAAC or a requirement for COL developed ITAAC for interface requirements.

- Site parameters used as the basis for AP1000 design presented in the Tier 2 Material. These parameters represent a bounding envelope of site conditions for any license application referencing the AP1000 design certification. No ITAAC are necessary for the site parameters entries because compliance with site parameters will be verified as part of issuance of a license for a plant that references the AP1000 design certification.

The following is a description of the criteria and methods used to select specific technical entries for the Certified Design Material. The structure of the description is based on the Certified Design Material report structure.

The criteria and methods discussed in the following sections are guidelines only. For some matters, the contents of the Certified Design Material may not directly correspond to these guidelines because special considerations related to the matters may warrant a different approach. For such matters, a case-by-case determination is made regarding how or whether the matters should be addressed in the Certified Design Material. These determinations are based upon the principles inherent in Part 52.

14.3.1 CDM Section 1.0, Introduction

This section provides definitions, general provisions, a figure legend, and a list of acronyms used in the AP1000 Certified Design Material.

Selection Criteria – Section 1.1 is used to define terms used throughout the Certified Design Material. Selection of entries is based on a judgment that a particular word/phrase merits definition – with particular emphasis on terms associated with implementation of the ITAAC. Section 1.2 contains a mixture of provisions that is selected on the basis that the provision is necessary to either define technical requirements applicable to multiple systems in the Certified Design Material or to provide clarification and guidance for future users of the Certified Design Material.

Selection Methodology – Entries in the Definition section are made on the basis of a self-evident need for a term to be defined. These terms are accumulated during the preparation and review of the Certified Design Material. Entries in the General Provisions section also are developed as part of the Certified Design Material selection and review process. Each entry has a unique background, but the overall intent is to state the broad guidelines and interpretations that are used to prepare Certified Design Material for the AP1000.

14.3.2 CDM Section 2.0, System Based Design Descriptions and ITAAC

This section of the Certified Design Material has the design description and ITAAC material for the selected AP1000 systems. The intent of this list of AP1000 systems is to define at the Certified Design Material level the full scope of the certified design.

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14.3.2.1 Design Descriptions

The certified design descriptions for selected AP1000 systems address the top-level design features and performance standards that pertain to the safety of the plant and include descriptive text and supporting figures. The intent of the Certified Design Material design descriptions is to define the AP1000 design characteristics referenced in the design certification rule as a result of the certification provisions of 10 CFR Part 52.

Selection Criteria – The following criteria are considered in determining the information included in the certified design descriptions:

- The information in the certified design descriptions is selected from the technical information presented in the Tier 2 Material. This reflects the approach that the Certified Design Material contains top-level design information and is based on the NRC directive in Reference 2 that there "be less detail in a certification than in an application for certification." In this context, the certification is the Certified Design Material and the application for certification includes the Tier 2 Material.

- The certified design descriptions contain only the information from the Tier 2 Material that is most significant to safety. The Tier 2 Material contains a wide spectrum of information on various aspects of the AP1000 design. Not all of this information is included in the certified design descriptions. This selection criterion reflects the NRC directive in Reference 2 that the certified design should "encompass roughly the same design features that Section 50.59 prohibits changing without prior NRC approval." In determining those structures, systems, or components for which certified design descriptions and ITAAC must be prepared, the following questions are considered for each structure, system, or component:

  – Are there any features or functions classified as Class A, B, or C?

  – Are there any defense–in–depth features or functions provided?

  – For nonsafety–related systems, are there any features or functions credited for mitigation of design basis events?

  – For nonsafety–related systems, are there any features or functions that have been identified in Section 16.3 as candidates for additional regulatory oversight?

  If the answer to the first question is yes, then a certified design description and ITAAC are prepared using the safety function stated in the Tier 2 Material and the parameters from the safety analysis.

  If the answer to either of the next two questions is yes, then a certified design description and ITAAC are prepared using the functions stated in the Tier 2 Material and the parameters from the system design calculations.

  If the answer to the last question is yes and the feature or function is not a programmatic requirement related to operations, maintenance or other programs, then a certified design description and ITAAC are prepared using the functions stated in the Tier 2 Material and the parameters from system design calculations.

  In addition, the following questions were considered for each structure, system, or component not already selected for ITAAC using the above selection criteria:

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– Are any features or functions necessary to satisfy the NRC's regulations in Parts 20, 50, 52, 73 and 100?

– Are there any features or functions that represent an important assumption for probabilistic risk assessment?

– Are any features or functions important in preventing or mitigating severe accidents?

– Are there any features or functions that have a significant impact on the safety and operation of the plant?

– Are any features or functions the subject of a provision in the Technical Specifications?

If the answer to any of the above questions is yes, then a design description and ITAAC are prepared using the appropriate functions stated in the Tier 2 material and the parameters from the system design calculations.

A summary of the AP1000 structures, systems, or components considered for selection is given in Table 14.3-1.

- In general, safety-related and defense-in-depth features and functions of structures, systems, and components are discussed in the certified design descriptions. Structures, systems, and components that are not classified as safety-related or defense-in-depth are discussed in the certified design descriptions to the extent that they have features or functions that mitigate a design basis event.

- The certified design descriptions for structures, systems, and components are limited to a discussion of design features and functions. The design bases of structures, systems, and components, and explanations of their importance to safety, are provided in the Tier 2 Material and are not included in the certified design descriptions. The Certified Design Material design descriptions define the certified design. Justification that the design meets regulatory requirements is presented in the Tier 2 Material.

- The certified design descriptions focus on the physical characteristics of the facility. The certified design descriptions do not contain programmatic requirements related to operating conditions or to operations, maintenance, or other programs. These matters are controlled by other means such as the technical specifications.

- The certified design descriptions in Section 2.0 of the Certified Design Material discuss the functional arrangement and performance characteristics that the structures, systems, and components should have after construction is completed. In general, the certified design descriptions do not address the processes that will be used for designing and constructing a plant that references the AP1000 design certification. This is acceptable because the safety-function of a structure, system, or component is dependent upon its final as-built condition and not the processes used to achieve that condition. Exceptions to this criterion are the selected design and qualification processes defined in the instrumentation and control portions and piping portions of Section 2 and the piping, seismic, structural and human factors portion of Section 3.

  The programmatic aspects of the design and construction processes (training, qualification of welders, and the like) are part of the licensee's programs and are subject to commitments made at the time of combined license issuance. Consequently, these issues are not addressed in the AP1000 Certified Design Material.

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- The certified design descriptions address fixed design features expected to be in place for the lifetime of the facility. Portable equipment and replaceable items are controlled through operational related programs.

- The certified AP1000 design descriptions do not discuss component types (for example, valve and instrument types), component internals, or component manufacturers. This approach is based on the premise that the safety function of a particular design element can be performed by a variety of component types from different manufacturers.

- The certified design descriptions do not contain proprietary information.

- For the applicant or licensee of a plant that references the AP1000 design certification to take advantage of improvements in technology, the certified design descriptions in general do not prescribe design features that are the subject of rapidly evolving technology.

- The Certified Design Material design description is intended to be self-contained and does not make direct reference to the Tier 2 Material, industrial standards, regulatory requirements, or other documents. (There are some exceptions involving the ASME Code and the Code of Federal Regulations.) If these sources contain technical information of sufficient safety significance to warrant Certified Design Material treatment, the information is extracted from the source and included directly in the appropriate system design description.

  This approach is appropriate because it is unambiguous and it avoids potential questions regarding how much of a referenced document is encompassed in, and becomes part of, the Certified Design Material.

- Selection of the technical terminology to be used in the Certified Design Material is guided by the principle that the terminology should be as consistent as possible with that used in the Tier 2 Material and the body of regulatory requirements and industrial standards applicable to the nuclear industry. This approach is intended to minimize problems in interpreting Certified Design Material commitments.

A review of those sections of the AP1000 Tier 2 Material that document plant safety evaluations was conducted. Specifically, reviews were conducted of the following chapters of the AP1000 Tier 2 Material; the flooding analysis in Chapter 5, the analysis of overpressure protection in Chapter 5, containment analysis in Chapter 6, the core cooling analysis in Chapters 6 and 15, the analysis of fire protection in Chapter 9, the safety analysis of transients in Chapter 15, the analysis of anticipated transients without scram (ATWS) in Chapters 7 and 15, the radiological analysis in Chapter 15, the resolution of unresolved or generic safety issues and Three Mile Island issues in Chapter 1, and the PRA and severe accident information in Chapter 19. These reviews were important in identifying safety-related system design information warranting consideration in the design descriptions and the accompanying design commitments.

Selection Methodology – The Certified Design Material uses a system report structure. The certified design description entry for any system is based on review of the multiple sources having technical information related to that system. Using the selection criteria listed, design description material is developed for each system by reviewing the Tier 2 Material, safety analysis, test programs, and design documents relating to that system.

Application of the criteria listed results in a graded treatment of the systems. This leads to variation in the scope of the design description entries. The following lists the types of AP1000 systems and is a summary of this graded treatment:

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System Type	Scope of Design Description
Systems with safety-related functions that contribute to plant performance during design basis accidents	Major safety-related features and performance characteristics
Systems with defense-in-depth functions that contribute to plant performance during design basis accidents	Major defense-in-depth features and performance characteristics
Nonsafety-related systems potentially impacting safety	Brief discussion of design features that prevent or mitigate the potential safety concern
Nonsafety-related systems with no relationship to safety	No discussion

For safety-related systems, application of this criteria results in design description entries that include the following information, as applicable:

- System name and scope
- System purpose
- Summary of the system's safety-significant components (usually shown by a figure)
- Equipment seismic and ASME classifications
- Piping ASME classification and Leak-Before-Break criteria
- Type of electrical power provided for the system
- System's important instruments, controls, and alarms to the extent located in the main control room or remote shutdown workstation
- Equipment to be qualified for harsh environments
- Motor-operated valves within the system that have an active safety-related function
- Other features or functions that are significant to safety

The certified design descriptions for nonsafety-related systems include the information listed to the extent that the information is relevant to the system and is significant to safety. Since much of this information is not relevant to nonsafety-related systems, the certified design descriptions for nonsafety-related systems are less extensive than the descriptions for safety-related systems.

14.3.2.2 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

A table of ITAAC entries is provided for each system that has design description entries. The intent of these ITAAC is to define activities that will be undertaken to verify the as-built system conforms with the design features and characteristics defined in the design description. ITAAC are provided in tables with the following three-column format:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

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Each design commitment in the left-hand column of the ITAAC tables has an associated inspections, tests, or analyses (ITA) requirement specified in the middle column. The acceptance criteria for the ITA are defined in the right-hand column.

Design Acceptance Criteria (DAC)/ITAAC closure is outlined in Appendix 14A.

Selection Criteria – The following are considered when determining what information is included in the Certified Design Material ITAAC entries:

- The scope and content of the ITAAC correspond to the scope and content of the certified design descriptions. There are no ITAAC for aspects of the design not addressed in the design description. This is appropriate because the objective of the ITAAC design certification entries is to verify that the as-built facility has the design features and performance characteristics defined in the Certified Design Material descriptions.

  Each AP1000 system with a design description has an ITAAC table. This reflects the assessment that a design feature meriting a Certified Design Material description also merits an ITAAC entry to verify that the feature has been included in the as-built facility.

- One inspection, test, or analysis may verify one or more provisions in the certified design description. An ITAAC that specifies a system functional test or an inspection may verify a number of provisions in the design description. There is not necessarily a one-to-one correspondence between the ITAAC and the design descriptions.

- As required by 10 CFR 52.103, the inspections, tests, and analyses must be completed (and the acceptance criteria satisfied) prior to fuel loading. Therefore, the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load.

- Because the design descriptions are limited to fixed design features expected to be in place for the lifetime of the facility, the ITAAC are limited to a verification of fixtures in the plant. There are no ITAAC for nuclear fuel, fuel channels, and control rods because they are changed by a licensee.

- The ITAAC verify the as-built configuration and performance characteristics of structures, systems, and components as identified in the Certified Design Material design descriptions.

Selection Methodology – Using the selection criteria, ITAAC table entries are developed for each selected system. This is achieved by evaluating the design features and performance characteristics defined in the Certified Design Material design description and preparing an ITAAC table entry for the design description criteria that satisfied the selection criteria. There is a close correlation between the left-hand column of the ITAAC table and the corresponding design description entries.

The ITAAC table is completed by selecting the method to be used for verification (either a test, an inspection, or an analysis [ITA]) and the acceptance criteria for the as-built feature.

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The selection of the ITAs is guided by the following:

ITA Approach	Application
Inspection	To be used when verification can be accomplished by visual observations, physical examinations, review of records based on visual observations, or physical examinations that compare the as-built structure, system, or component condition to one or more design description commitments.
Test	To be used when verification can be accomplished by the actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of the as-built structures, systems, or components. The type of tests identified in the ITAAC tables includes activities such as factory testing, special test facility programs, and laboratory testing.
Analysis	To be used when verification can be accomplished by calculation, mathematical computation, or engineering or technical evaluations of the as-built structures, systems, or components.

The proposed verification activity is identified in the middle column of the ITAAC table. Where appropriate, the Tier 2 Material provides details regarding implementation of the verification activity. This Tier 2 Material is not referenced in the Certified Design Material and is not part of the Certified Design Material; Tier 2 Material is considered as providing one of potentially several acceptable methods for completing the ITA.

Selection of acceptance criteria is dependent upon the design characteristic being verified by the ITAAC table entry: in most cases, the appropriate acceptance criteria is self-evident and is based upon the Certified Design Material design description. For many of the AP1000 ITAAC, the acceptance criteria is a statement that the as-built facility has the design feature or performance characteristic identified in the design description. A guiding principle for acceptance criteria preparation is the recognition that the criteria should be objective and unambiguous. The use of objective and unambiguous terms for the acceptance criteria will minimize opportunities for multiple, subjective (and potentially conflicting) interpretations as to whether an acceptance criteria has, or has not, been met. In some cases, the ITAAC acceptance criteria contain numerical parameters from the Tier 2 Material that are not specifically identified in the Certified Design Material design description or the design commitment column of the ITAAC table. This is acceptable because the design description defines the important design feature/performance that merits Certified Design Material treatment. The acceptance criterion defines a measurement standard for determining if the as-built facility is in compliance with the Certified Design Material design description commitment. Where appropriate, the Tier 2 Material identifies criteria applicable to the same design feature or function that is the subject of more general acceptance criteria in the ITAAC table.

For numerical acceptance criteria, ranges and/or tolerances are included. This is necessary and acceptable because of the following:

- Specification of a single-value acceptance criteria is impractical because trivial deviations will represent unnecessary noncompliances.
- Tolerances recognize that legitimate site variations can occur in complex construction projects.
- Minor variations in plant parameters within the tolerance bounds have no impact on plant safety.

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14.3.2.3 Site-Specific ITAAC (SS-ITAAC)

A table of inspections, tests, analyses, and acceptance criteria (ITAAC) entries is provided for each site-specific system described in this document that meets the selection criteria, and that is not included in the certified design. The intent of these ITAAC is to define activities that are undertaken to verify the as-built system conforms with the design features and characteristics defined in the system design description. ITAAC are provided in tables with the following three-column format:

Design Commitment	Inspection, Tests, Analyses	Acceptance Criteria

Each design commitment in the left-hand column of the ITAAC tables has associated inspections, tests, or analyses (ITA) requirements specified in the middle column. The acceptance criteria for the ITA are defined in the right-hand column.

SS-ITAAC do not address ancillary buildings and structures on the site, such as administrative buildings, parking lots, warehouses, training facilities, etc.

Selection Criteria

- In determining those structures, systems, or components for which ITAAC must be prepared, the following questions are considered for each structure, system, or component:
  - Are any features or functions classified as Class A, B, or C?
  - Are any defense-in-depth features or functions provided?
  - For nonsafety-related systems, are any features or functions credited for mitigation of design basis events?
  - For nonsafety-related systems, are there any features or functions that have been identified in Section 16.3 as candidates for additional regulatory oversight?

  If the answer to any of the above questions is yes, then ITAAC are prepared.
- The scope and content of the ITAAC correspond to the scope and content of the site-specific system design description.
- One inspection, test, or analysis may verify one or more provisions in the system design description. An ITAAC that specifies a system functional test or an inspection may verify a number of provisions in the system design description. There is not necessarily a one-to-one correspondence between the ITAAC and the system design descriptions.
- As required by 10 CFR 52.103, the inspections, tests, and analyses are completed (and the acceptance criteria satisfied) prior to initial fuel loading.
- The ITAAC verify the as-built configuration and performance characteristics of structures, systems, and components as identified in the system design descriptions.

Selection Methodology – Using the selection criteria, ITAAC table entries are developed for each selected system. This is achieved by evaluating the design features and performance characteristics defined in the system design descriptions and preparing an ITAAC table entry for each design description criterion that satisfies the selection criteria. A close correlation exists between the left-hand column of the ITAAC and the corresponding design description entries.

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The ITAAC table is completed by selecting the method to be used for verification (either a test, an inspection, or an analysis) and the acceptance criteria for the as-built feature.

The approach used to perform the tests, inspections, or analyses is similar to that described in Subsection 14.3.2.2.

14.3.2.3.1 Emergency Planning ITAAC (EP-ITAAC)

EP-ITAAC were developed to address implementation of elements of the Emergency Plan. Site-specific EP-ITAAC are based on the generic ITAAC provided in Table 13.3-1 of SECY-05-0197. These ITAAC have been tailored to the specific reactor design and emergency planning program requirements.

14.3.2.3.2 Physical Security ITAAC (PS-ITAAC)

Generic PS-ITAAC have been developed in a coordinated effort between the NRC and the Nuclear Energy Institute (NEI). These generic ITAAC have been tailored to the AP1000 design and site-specific security requirements.

14.3.2.3.3 Other Site-Specific Systems

One additional site-specific system has been determined to meet the ITAAC selection criteria, and ITAAC have been included for the Transmission Switchyard and Offsite Power System (ZBS) as indicated in Table 14.3-1. Systems not meeting the selection criteria are subject to the normal functional testing to verify that newly designed and installed systems, structures, or components perform as designed.

A summary of the AP1000 structures, systems, or components considered for selection is given in Table 14.3-1.

14.3.3 CDM Section 3.0, Non-System Based Design Descriptions and ITAAC

Entries in this section of the Certified Design Material have the same structure as the system material discussed in Subsection 14.3.2; that is, design description text and figures and a table of ITAAC entries. The objective of this Certified Design Material is to address selected design and construction activities which are applicable to more than one system. There are six entries in Section 3.0 of the Certified Design Material: nuclear island buildings, initial test program, emergency response facilities, human factors engineering, Design Reliability Assurance Program, and radiation protection.

14.3.3.1 Not Used

14.3.3.2 Pipe Rupture Hazard Analysis ITAAC

A pipe rupture hazard analysis is part of the piping design. The analyses will document that structures, systems, and components (SSCs) which are required to be functional during and following a design basis event have adequate high-energy and moderate-energy pipe break mitigation features. The locations of postulated ruptures and essential targets will be established and required pipe whip restraint and jet shield designs will be included. The as-designed pipe rupture hazards analysis will be based on the as-designed piping analysis and will be in accordance with the criteria outlined in Subsections 3.6.1.3.2 and 3.6.2.5. The evaluation will address environmental and flooding effects of cracks in high and moderate energy piping. The report of the pipe rupture hazard analysis shall conclude that, for each postulated piping failure, the systems, structures, and components that are required to be functional during and following a design basis event are protected.

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The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in Subsections 3.6.1.3.2 and 3.6.2.5 are covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built pipe rupture hazards mitigation features reflect the design, as reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed.

The as-designed pipe rupture hazard analysis completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure.

14.3.3.3 Piping Design ITAAC

The piping design ITAAC consists of the piping analysis for safety-related ASME Code piping. The piping design is completed on a package-by-package basis for applicable systems. In order to support closure of the piping design ITAAC, information consisting of the as-designed piping analysis for piping lines chosen to demonstrate all aspects of the piping design will be made available for NRC review, inspection, and/or audit. This information will consist of a design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class I piping. The piping packages to be analyzed are identified in Table 3.9-20.

The ASME Code prescribes certain procedures and requirements that are to be followed for completing the piping design. The piping design ITAAC includes a verification of the ASME Code design report to ensure that the appropriate code design requirements for each system's safety class have been implemented.

A reconciliation of the applicable safety-related as-built piping systems is covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built piping reflects the design, as reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed.

The as-designed piping analysis completed for the Reference Plant is applied to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure.

14.3.4 Certified Design Material Section 4.0, Interface Requirements

AP1000 is a plant design incorporating the nuclear island, the annex building and associated equipment, the diesel generator building and associated equipment, the turbine building, the turbine/generator equipment, and the radwaste facilities. As a result, no interfaces need to be identified between or among these portions of the plant. There are no safety-related interfaces between the AP1000 certified design and other portions of a facility with a combined license under 10 CFR Part 52.

Initial testing of interfacing non-safety systems in portions of the plant outside the scope of design certification is as discussed in Section 14.4. Section 1.8, Table 1.8-1, lists the interfacing systems and structures. Those systems that meet the requirements of 10 CFR 52.47(a)(25) are tabulated in Subsection 14.4.5.

14.3.5 CDM Section 5.0, Site Parameters

This section of the Certified Design Material defines the site parameters used as a basis for the design defined in the AP1000 certification application. These entries respond to the 10 CFR 52.47(a)(25) requirement that the design certification documentation include site parameter information. It is intended that applicants referencing the AP1000 design certification demonstrate

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that these parameters for the selected site are within the certification envelope or provide additional analysis to show acceptability of deviations from the interface envelope.

Site-specific external events that relate to the acceptability of the design (and not to the acceptability of the site) are not considered site parameters and are addressed as interface requirements in the appropriate system entry in Section 4 of the Certified Design Material.

Section 5.0 of the Certified Design Material does not include any ITAAC and is limited to defining the AP1000 site parameters. This is an appropriate approach because compliance of the site with these parameters is demonstrated by a license applicant prior to issuance of the license.

Selection Criteria – Section 2.0, Table 2.0-1, provides the envelope of site design parameters used for the AP1000 design. The corresponding Certified Design Material Section 5.0 is based on using Table 2.0-1. Section 5.0 is limited to a tabular entry; no supporting text material is required.

14.3.6 Initial Test Program

The AP1000 Initial Test Program defines testing activities that will be conducted following completion of construction and construction-related inspections and tests. The Initial Test Program extends through the start of commercial operation of the facility. This program is discussed in Chapter 14.

A summary of the Initial Test Program is included in Certified Design Material Section 3.4. This summary includes an overview of the Initial Test Program structure. This information is included in the Certified Design Material because of the importance of the Initial Test Program defining pre- and post-fuel load testing for the as-built facility. Key pre-fuel load Initial Test Program testing for individual systems is defined in the system ITAAC in Certified Design Material Sections 2 and 3.

No ITAAC entries have been included in the Certified Design Material for the Initial Test Program. This is acceptable because of the following:

- The Initial Test Program activities involve testing with the reactor at various power levels and thus cannot be completed prior to fuel load (Part 52 requires ITAAC to be completed prior to fuel load).

- Testing activities specified as part of the ITAAC in Certified Design Material Sections 2 and 3 must be performed prior to fuel load. Because these ITAAC testing activities address the design features and characteristics of safety significance, additional ITAAC for the Initial Test Program are not necessary to ensure that the as-built plant conforms with the certified design.

14.3.7 Elements of AP1000 Design Material Incorporated into the Certified Design Material

Tables 14.3-2 through 14.3-8 summarize the design material that has been incorporated into the CDM in the areas of 1) Design Basis Accident Analysis, 2) Anticipated Transients Without Scram (ATWS), 3) Fire Protection, 4) Flood Protection, 5) Probabilistic Risk Assessment, 6) Radiological Analysis, and 7) Severe Accident Analysis. PRA assumptions incorporated into these tables encompass elements of the system design and assumptions that were expressly included in Tier 1 due to their importance. Both types of PRA assumptions were included for completeness, but are not distinguished in the tables. CDM falling outside of the seven subject areas are intentionally not incorporated in these tables. However, the referenced AP1000 DCD sections may contain more information than encompassed by these seven subject areas. Each table may also include design information (certified or non-certified) that is not directly related to the particular subject area. Further,

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these tables are not intended to include all system-specific CDM information that is provided in the AP1000 Tier 2 system descriptions.

14.3.8 Summary

An element of the design certification processes deriving from 10 CFR Part 52 is the selection and documentation of the technical information to be included in the design certification rule as the certified design. The certified design material is a subset of the design information presented in the Tier 2 Material. It includes the following:

- Key, important safety-significant aspects of the design described in the certification application
- Inspections, tests, analyses, and acceptance criteria (ITAAC) that will be used to verify that the as-built facility conforms with the certified design
- Interface requirements and site parameters

The information presented in the AP1000 Certified Design Material is prepared using the selection criteria and methodology described in this section and is intended to satisfy the above Part 52 requirements for design certification. The ITAAC entries in Sections 2.0 and 3.0 confirm that key design performance characteristics and design features are implemented in the as-built facility.

14.3.9 References

1. SECY-90-377, “Requirements for Design Certification under 10 CFR Part 52,” February 15, 1991.

2. 10 CFR, Part 52, “Statements of Consideration,” (54 Federal Register 15372 [1989]).

3. SECY-90-241, “Level of Detail Required for Design Certification under Part 52,” August 31, 1990.

4. SECY-90-377, “Requirements for Design Certification Under 10 CFR Part 52,” November 8, 1990.

5. SECY-91-178, “Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses,” June 12, 1991.

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Table 14.3-1 (Sheet 1 of 4)
ITAAC Screening Summary

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
ADS	Automatic Depressurization System	X
ASS	Auxiliary Steam Supply System	X
BDS	Steam Generator Blowdown System	X
CAS	Compressed and Instrument Air System	X
CCS	Component Cooling Water System	X
CDS	Condensate System	X
CES	Condenser Tube Cleaning System	X
CFS	Turbine Island Chemical Feed System	X
CMS	Condenser Air Removal System	X
CNS	Containment System	X
CPS	Condensate Polishing System	X
CVS	Chemical and Volume Control System	X
CWS	Circulating Water System	X
DAS	Diverse Actuation System	X
DDS	Data Display Processing System	X
DOS	Standby Diesel Fuel Oil System	X
DRS	Storm Drain System	XX
DTS	Demineralized Water Treatment System	X
DWS	Demineralized Water Transfer and Storage System	X
ECS	Main AC Power System	X
EDS	Non Class 1E DC and UPS System	X
EFS	Communication System	X
EGS	Grounding and Lightning Protection System	X

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Table 14.3-1 (Sheet 2 of 4)
ITAAC Screening Summary

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
EHS	Special Process Heat Tracing System	X
ELS	Plant Lighting System	X
EQS	Cathodic Protection System	X
FHS	Fuel Handling System	X
FPS	Fire Protection System	X
FWS	Main and Startup Feedwater System	X
GSS	Gland Seal System	X
HCS	Generator Hydrogen and CO_2 Systems	X
HDS	Heater Drain System	X
HSS	Hydrogen Seal Oil System	X
IDS	Class 1E DC and UPS System	X
IIS	Incore Instrumentation System	X
LOS	Main Turbine and Generator Lube Oil System	X
MES	Meteorological and Environmental Monitoring System	XX
MHS	Mechanical Handling System	X
MSS	Main Steam System	X
MTS	Main Turbine System	X
OCS	Operations and Control Centers	X
PCS	Passive Containment Cooling System	X
PGS	Plant Gas System	X
PLS	Plant Control System	X
PMS	Protection and Safety Monitoring System	X
PSS	Primary Sampling System	X

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Table 14.3-1 (Sheet 3 of 4)
ITAAC Screening Summary

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
PWS	Potable Water System	X
PXS	Passive Core Cooling System	X
RCS	Reactor Coolant System	X
RDS	Gravity and Roof Drain Collection System	X
RMS	Radiation Monitoring System	X
RNS	Normal Residual Heat Removal System	X
RWS	Raw Water System	XX
RXS	Reactor System	X
SDS	Sanitary Drainage System	X
SES	Plant Security System	X
SFS	Spent Fuel Cooling System	X
SGS	Steam Generator System	X
SJS	Seismic Monitoring System	X
SMS	Special Monitoring System	X
SSS	Secondary Sampling System	X
SWS	Service Water System	X
TCS	Turbine Building Closed Cooling Water System	X
TDS	Turbine Island Vents, Drains and Relief Systems	X
TOS	Main Turbine Control and Diagnostics System	X
TVS	Closed Circuit TV System	XX
VAS	Radiologically Controlled Area Ventilation System	X
VBS	Nuclear Island Nonradioactive Ventilation System	X
VCS	Containment Recirculation Cooling System	X
VES	Main Control Room Emergency Habitability System	X

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Table 14.3-1 (Sheet 4 of 4)
ITAAC Screening Summary

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
VFS	Containment Air Filtration System	X
VHS	Health Physics and Hot Machine Shop HVAC System	X
VLS	Containment Hydrogen Control System	X
VPS	Pump House Building Ventilation System	NA
VRS	Radwaste Building HVAC System	X
VTS	Turbine Island Building Ventilation System	X
VUS	Containment Leak Rate Test System	X
VWS	Central Chilled Water System	X
VXS	Annex/Auxiliary Nonradioactive Ventilation System	X
VYS	Hot Water Heating System	X
VZS	Diesel Generator Building Ventilation System	X
WGS	Gaseous Radwaste System	X
WLS	Liquid Radwaste System	X
WRS	Radioactive Waste Drain System	X
WSS	Solid Radwaste System	X
WWS	Waste Water System	X
YFS	Yard Fire Water System	XX
ZAS	Main Generator System	X
ZBS	Transmission Switchyard and Offsite Power System	XX
ZOS	Onsite Standby Power System	X
ZRS	Offsite Retail Power System	XX
ZVS	Excitation and Voltage Regulation System	X

Legend: X = Selected for ITAAC
X = Selected for ITAAC - title only, no entry for Design Certification

XX = Site-specific system selected for ITAAC – title only, no entry for COLA
XX = Selected for ITAAC
NA = System is not part of the design

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Table 14.3-2 (Sheet 1 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 3.10	The protection and safety monitoring system equipment is seismically qualified to meet safe shutdown earthquake levels.	
Section 3.11.3	The design of the protection and safety monitoring system equipment has margin to accommodate a loss of the normal HVAC.	
Section 5.1.2	Safety valves are installed above and connected to the pressurizer to provide overpressure protection for the reactor coolant system.	
Section 5.1.2	The RCS has two hot legs and four cold legs.	
Section 5.1.2	The RCS has two steam generators and four reactor coolant pumps.	
Section 5.1.2	The RCS contains a pressurizer and a surge line connected to one hot leg.	
Section 5.1.3.3	Rotating inertia needed for flow coast-down, is provided.	
Table 5.1-3	Minimum measured flow rate with 10% tube plugging (gpm/ loop)	150,835
Table 5.1-3	Initial rated reactor core thermal power (MWt)	3400
Section 5.2.2	Reactor coolant system and steam system overpressure protection during power operation are provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the PMS.	
Section 5.2.2.1	Safety valve capacity exists to prevent exceeding 110 percent of system design pressure for the following events: – Loss of electrical load and/or turbine trip – Uncontrolled rod withdrawal at power – Loss of reactor coolant flow – Loss of normal feedwater – Loss of offsite power to the station auxiliaries	
Section 5.2.2.1	Overpressure protection for the steam system is provided by steam generator safety valves	
Section 5.3.2.3	Non-destructive examination (NDE) of the reactor vessel and its appurtenances is conducted in accordance with ASME Code Section III requirements.	

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Table 14.3-2 (Sheet 2 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 5.3.2.5	The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline base metal transverse direction and welds are 75 foot-pounds, as required by Appendix G of 10 CFR 50.	
Section 5.4.1.2.1	Resistance temperature detectors (RTDs) monitor motor cooling circuit water temperature. These detectors provide indication of anomalous bearing or motor operation. They also provide a system for automatic shutdown in the event of a prolonged loss of component cooling water.	
Section 5.4.1.3.4	It is important to reactor protection that the reactor coolant continues to flow for a time after reactor trip and loss of electrical power. To provide this flow, each reactor coolant pump has a high-inertia rotor.	
Section 5.4.1.3.4	A reactor coolant pump trip occurs on High-2 bearing water temperature.	
Section 5.4.5.2.3	Power to the pressurizer heaters is blocked when the core makeup tanks are actuated.	
Section 5.4.6	Automatic depressurization system stage 1, 2 and 3 valves are connected to the pressurizer and discharge via the spargers to the in-containment refueling water storage tank.	
Section 5.4.6	Automatic depressurization system stage 4 valves are connected to each hot leg.	
Section 5.4.9.3	In the analysis of overpressure events, the pressurizer safety valves are assumed to actuate at 2500 psia. The safety valve flowrate assumed is based on full flow at 2575 psia, assuming 3 percent accumulation.	
Section 5.4.9.3	The pressurizer safety valves prevent reactor coolant system pressure from exceeding 110% of system design pressure.	
Section 5.4.12	The reactor head vent valves can be operated from the main control room to provide an emergency letdown path.	
Table 5.4-1	Minimum reactor coolant motor/pump moment of inertia sufficient to provide flow coastdown as given in Figure 15.3.2-1.	
Table 5.4-11	Reactor Coolant System Design Pressure Settings: – Safety valves begin to open (psig)	2485

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Table 14.3-2 (Sheet 3 of 17)
Design Basis Accident Analysis

Reference		Design Feature	Value
Table	5.4-17	Pressurizer Safety Valves - Design Parameters: – Number – Minimum required relieving capacity per valve (lbm/hr) – Set pressure (psig)	 2 ≥ 750,000 2485 ± 25
Section	6.1.1.4	The aluminum surfaces of the excore (source range, intermediate range, and power range) detectors are encased in a watertight stainless steel or titanium housing.	
Table	6.1-2	The exterior of the containment vessel (above plant elevation 135′ 3″) and the interior of the containment vessel is coated with an inorganic zinc coating.	
Section	6.1.2.1.5	The nonsafety-related coatings used inside containment on walls, floors, ceilings, structural steel which is part of the building structure, and on the polar crane have a minimum dry film density (lb/ft^3).	≥ 100
Figure	6.2.2-1	The passive containment cooling system consists of a water storage tank, cooling water flow discharge path to the containment shell, a water distribution system for the containment shell, and a cooling air flow path.	
Table	6.2.2-1	The minimum duration the PCS cooling water flow is provided from the PCCWST (hours).	≥ 72
Table	6.2.2-1	The water coverage of the containment shell exceeds the amount used in the safety analysis.	
Table	6.2.2-1	The minimum drain flow rate capacity of the upper annulus drain (gpm).	≥ 525
Table	6.2.2-1	The minimum makeup flow rate capability from an external source to the PCS water storage tank (gpm).	≥ 100
Table	6.2.2-1	The minimum makeup flow rate capability from the PCS water storage tank to the spent fuel pool (gpm).	≥ 118
Table	6.2.2-1	The minimum PCS water storage tank volume for makeup to the spent fuel pool (non-coincident with PCS operation) (gallons).	≥ 756,700
Table	6.2.2-1	The minimum long term makeup capability from the PCCAWST to the PCCWST (days).	≥ 4

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Table 14.3-2 (Sheet 4 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Table 6.2.2-1	The minimum long term makeup flow capability from the PCCAWST to the PCCWST (gpm).	≥ 100
Table 6.2.2-1	The minimum long term makeup flow capability from the PCCAWST to the spent fuel pool (gpm).	> 35
Table 6.2.2-2	The first (i.e., tallest) standpipe's elevation above the tank floor (feet).	24.1 ± 0.2
Table 6.2.2-2	The second tallest standpipe's elevation above the tank floor (feet).	20.3 ± 0.2
Table 6.2.2-2	The third tallest standpipe's elevation above the tank floor (feet).	16.8 ± 0.2
Table 6.2.2-1	The passive containment cooling water flow rate at 72 hours (gpm).	≥ 100.7
Table 6.2.2-1	The passive containment cooling water flow rate when the PCCWST water level uncovers the third tallest standpipe (gpm).	≥ 144.2
Table 6.2.2-1	The passive containment cooling water flow rate when the PCCWST water level uncovers the second tallest standpipe (gpm).	≥ 176.3
Table 6.2.2-1	The passive containment cooling water flow rate when the PCCWST water level uncovers the first (i.e., tallest) standpipe (gpm).	≥ 226.6
Table 6.2.2-1	The passive containment cooling water flow rate with water inventory at a level of 27.4 ft + 0.2, -0.0 ft above the tank floor (gpm).	≥ 469.1
Section 6.3	The passive core cooling system provides core decay heat removal during design basis events.	
Section 6.3	The passive core cooling system provides RCS makeup, boration, and safety injection during design basis events.	
Section 6.3	The passive core cooling system provides pH adjustment of water flooding the containment following design bases events.	

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Table 14.3-2 (Sheet 5 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 6.3.1.1	The passive core cooling system is designed to provide emergency core cooling during events involving increases and decreases in secondary side heat removal and decreases in reactor coolant system inventory.	
Section 6.3.2.1.1	The heat exchanger consists of a bank of C-tubes, connected to a tubesheet and channel heat arrangement at the top (inlet) and bottom (outlet). The passive exchanger connects to the reactor coolant system through an inlet line from one reactor coolant system hot leg and an outlet line to the associated steam generator cold leg plenum (reactor coolant pump suction).	
Section 6.3.2.1.1	For the passive residual heat removal heat exchanger, the normal water temperature in the inlet line will be hotter than the discharge line.	
Section 6.3.2.1.2	The actuation of the core makeup tanks following a steam line break provides injection of borated water via water recirculation to mitigate the reactivity transient and provide the required shutdown margin.	
Section 6.3.2.2.1	The CMT inlet diffuser has a minimum flow area (in^2).	≥ 165
Section 6.3.2.2.3	The in-containment refueling water storage tank contains one passive residual heat removal heat exchanger.	
Section 6.3.2.2.6	The connection of the sparger branch arms to the sparger hub are submerged below the in-containment refueling water storage tank overflow level (ft).	≤ 11.5
Section 6.3.2.2.6	Automatic depressurization system stage 1, 2 and 3 valves are connected to the pressurizer and discharge via the spargers to the in-containment refueling water storage tank.	
Section 6.3.2.2.7.1	The containment recirculation screens have a protective plate that is located no more than 1 foot, 3 inches above the top of the face of the screens, which extends at least 8 feet, 3 inches perpendicular to the front and at least 7 feet to the side of the face of the screens, to prevent coating debris from reaching the screens.	
Section 6.3.2.2.7.1	The type of insulation used on ASME Class 1 lines inside containment and on the reactor vessel, reactor coolant pumps, pressurizer and steam generators is a metal reflective or suitable equivalent insulation.	

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Table 14.3-2 (Sheet 6 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 6.3.2.2.7.3	The surface materials used in the vicinity of the containment recirculation screens are stainless steel. In the vicinity of the containment recirculation screens includes surfaces located above the bottom of the recirculation screens up to and including the bottom surface of the plate discussed in subsection 6.3.2.2.7.1, and the surfaces extending at least 8 feet, 3 inches perpendicular to the front and at least 7 feet to the side of the face of the screens.	
Section 6.3.2.2.7.3	The bottom of the containment recirculation screens are located above the loop compartment floor (ft).	≥ 2
Section 6.3.3.2.1	For a loss of main feedwater event, the passive residual heat removal heat exchanger is actuated. If the core makeup tanks are not initially actuated, they actuate later when passive residual heat exchanger cooling sufficiently reduces pressurizer level.	
Section 6.3.3.2.2	For a feedwater system pipe failure event, the passive residual heat removal heat exchanger and the core makeup tanks are actuated.	
Section 6.3.3.3.1	For a steam generator tube rupture event, the nonsafety-related makeup pumps are automatically actuated when reactor coolant system inventory decreases and a reactor trip occurs, followed by actuation of the startup feedwater pumps. Makeup pumps automatically function to maintain the programmed pressurizer level. The core makeup tanks subsequently actuate on Low-2 pressurizer level, if they are not already actuated. Actuation of the core makeup tanks automatically actuates the passive residual heat removal system heat exchanger.	
Section 6.3.6.1	The piping resistances connecting the following PXS components and the RCS are bounded by the resistances assumed in the Chapter 15 safety analysis: – Core makeup tanks – Accumulators – In-containment refueling water storage tank injection – Containment recirculation – Automatic depressurization system valves	
Section 6.3.6.1.3	The bottom of the core makeup tanks are located above the reactor vessel direct vessel injection nozzle centerline (ft).	≥ 7.5

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Table 14.3-2 (Sheet 7 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 6.3.6.1.3	The bottom of the in-containment refueling water storage tank is located above the direct vessel injection nozzle centerline (ft).	≥ 3.4
Section 6.3.6.1.3	The pH baskets are located below plant elevation 107′ 2″.	
Figure 6.3-1	The passive core cooling system has two direct vessel injection lines.	
Table 6.3-2	The passive core cooling system has two core makeup tanks, each with a minimum required volume (ft^3).	2487
Table 6.3-2	The passive core cooling system has two accumulators, each with a minimum required volume (ft^3)	2,000
Table 6.3-2	The passive core cooling system has an in-containment refueling water storage tank with a minimum required water volume (ft^3)	73,100
Section 6.3.2.2.3	The containment floodup volume for a LOCA in PXS room B has a maximum volume (ft^3) (excluding the IRWST) below a containment elevation of 107.68 feet.	71,960
Table 6.3-2	Each sparger has a minimum discharge flow area (in^2).	≥ 274
Table 6.3-2	The passive core cooling system has four pH adjustment baskets with a total minimum required volume (ft^3).	560
Section 14.2.9.1.3f	The passive residual heat removal heat exchanger minimum natural circulation heat transfer rate (Btu/hr) – With 520°F hot leg and 80°F IRWST – With 420°F hot leg and 80°F IRWST	 ≥ 1.78 E+08 ≥ 1.11 E+08
Section 6.3.6.1.3	The centerline of the HX's upper channel head is located above the HL centerline (ft).	≥ 26.3
Section 6.3.7.4.1	The CMT level sensors (PXS-11A/B/C/D, -12A/B/C/D, -13A/B/C/D, and -14A/B/C/D) upper level tap line has a downward slope of ≥ 2.4 degrees from the centerline of the connection to the CMT to the centerline of the connection to the standpipe.	≥ 2.4 degrees
Figure 6.3-1	The CMT inlet lines (cold leg to high point) have no downward sloping sections.	
Figure 6.3-1	The maximum elevation of the CMT injection lines between the connection to the CMT and the reactor vessel is the connection to the CMTs.	
Figure 6.3-1	The PRHR inlet line (hot leg to high point) has no downward sloping sections.	

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Table 14.3-2 (Sheet 8 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Figure 6.3-1	The maximum elevation of the IRWST injection lines (from the connection to the IRWST to the reactor vessel) and the containment recirculation lines (from the containment to the IRWST injection lines) is less than the bottom inside surface of the IRWST.	
Figure 6.3-1	The maximum elevation of the PRHR outlet line (from the PRHR to the SG) is less than the PRHR lower channel head top inside surface.	
Section 7.1.2.10	Isolation devices are used to maintain the electrical independence of divisions and to see that no interaction occurs between nonsafety-related systems and the safety-related system. Isolation devices serve to prevent credible faults in circuit from propagating to another circuit.	
Section 7.1.4.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire, flooding, explosions, missiles, electrical faults and pipe whip.	
Section 7.1.2	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
Section 7.2.2.2.1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
Section 7.2.2.2.8	The reactor is tripped by actuating one of two manual reactor trip controls from the main control room.	
Section 7.3.1.2.2	The in-containment refueling water storage tank is aligned for injection upon actuation of the fourth stage automatic depressurization system via the protection and safety monitoring system.	
Section 7.3.1.2.3	The core makeup tanks are aligned for operation on a safeguards actuation signal or on a low-2 pressurizer level signal via the protection and safety monitoring system.	
Section 7.3.1.2.4	The fourth stage valves of the automatic depressurization system receive a signal to open upon the coincidence of a Low-6 core makeup tank water level in either core makeup tank and low reactor coolant system pressure following a preset time delay after the third stage depressurization valves receive a signal to open via the protection and safety monitoring system.	

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Table 14.3-2 (Sheet 9 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 7.3.1.2.4	The first stage valves of the automatic depressurization system open upon receipt of a signal generated from a core makeup tank injection alignment signal coincident with core makeup tank water level less than the Low-3 setpoint in either core makeup tank via the protection and safety monitoring system.	
Section 7.3.1.2.4	The second and third stage valves open on time delays following generation of the first stage actuation signal via the protection and safety monitoring system.	
Section 7.3.1.2.5	The reactor coolant pumps are tripped upon generation of a safeguards actuation signal or upon generation of a low-2 pressurizer water level signal.	
Section 7.3.1.2.7	The passive residual heat removal heat exchanger control valves are opened on Low-2 steam generator wide range water level or on a CMT actuation signal via the protection and safety monitoring system.	
Section 7.3.1.2.9	The containment recirculation isolation valves are opened on a safeguards actuation signal in coincidence with low-3 in-containment refueling water storage tank water level via the protection and safety monitoring system.	
Section 7.3.1.2.14	The demineralized water system isolation valves close on a signal from the protection and safety monitoring system derived from either a reactor trip signal, a source range flux doubling signal, or low input voltage to the 1E dc uninterruptible power supply battery chargers.	
Section 7.3.1.2.15	The chemical and volume control system makeup line isolation valves automatically close on a signal from the protection and monitoring system derived from a source range flux doubling, high-2 pressurizer level, High-3 steam generator level signal, a safeguards signal coincident with high-1 pressurizer level, high steam generator water level coincident with P-4 permissive (reactor trip) or high-2 containment radioactivity.	
Section 7.3.2.2.1	The protection and monitoring system automatically generate an actuation signal for an engineered safety feature whenever a monitored condition reaches a preset level.	
Section 7.3.2.2.9	Manual initiation at the system-level exists for the engineered safety features actuation.	

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Table 14.3-2 (Sheet 10 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 7.4.3.1	If temporary evacuation of the main control room is required because of some abnormal main control room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the main control room through the use of controls and monitoring located at the remote shutdown workstation.	
Section 7.4.3.1.1	The remote shutdown workstation equipment is similar to the operator workstations in the main control room and is designed to the same standards. One remote shutdown workstation is provided.	
Section 7.4.3.1.3	The remote shutdown workstation achieves and maintains safe shutdown conditions from full power conditions and maintains safe shutdown conditions thereafter.	
Section 7.5.4	The protection and safety monitoring system provides signal conditioning, communications, and display functions for Category 1 variables and for Category 2 variables that are energized from the Class 1E uninterruptible power supply system.	
Section 7.6.1.1	An interlock is provided for the normally closed motor-operated normal residual heat removal system inner and outer suction isolation valves. Each valve is interlocked so that it cannot be opened unless the reactor coolant system pressure is below a preset pressure.	
Section 8.2.2	Following a turbine trip during power operation, the reverse-power relay will be blocked for a minimum time period (sec).	≥ 15
Section 8.3.2.1.2	The non-Class 1E dc and UPS system (EDS) consists of the electric power supply and distribution equipment that provides dc and uninterruptible ac power to nonsafety-related loads.	

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Table 14.3-2 (Sheet 11 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 9.1.1.2.1.C	During normal fuel handling operations, a single failure-proof hoist, designed to meet the requirements of NUREG-0554, is the only hoist capable of moving new fuel above the operating floor. Per the design criteria contained in NUREG-0554, drops from a single failure-proof hoist are deemed unlikely and do not require further analysis. The consequences of such a drop are minimal since no safety-related equipment would be impacted and there are no radiological releases with new unirradiated fuel. Because the likelihood of a new fuel assembly being dropped into the new fuel pit and onto the new fuel racks is minimal, it is unnecessary to evaluate drop scenarios for the new fuel storage rack.	
Section 9.1.3.5	The spent fuel pool is designed such that a water level is maintained above the spent fuel assemblies for at least 7 days following a loss of the spent fuel cooling system using only on-site makeup water sources (See Table 9.1-4).	
Section 9.1.3.5	The spent fuel pool cooling system includes safety-related connections to establish safety-related makeup to the spent fuel pool following a design basis event including a seismic event.	
Section 9.1.4.1.1	In the event of a safe shutdown earthquake (SSE), handling equipment cannot fail in such a manner as to prevent required function of seismic Category 1 equipment.	
Section 9.3.6.3.7	The chemical and volume control system contains two redundant safety-related valves to isolate the demineralized water system from the makeup pump suction.	
Section 9.3.6.3.7	The chemical and volume control system contains two safety-related valves to isolate the makeup flow to the reactor coolant system.	
Section 9.3.6.4.5	The chemical and volume control system contains two safety-related valves to isolate the makeup flow to the reactor coolant system.	
Section 9.3.6.4.5.1	The chemical and volume control system contains two redundant safety-related valves to isolate the demineralized water system from the makeup pump suction.	

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Table 14.3-2 (Sheet 12 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 9.3.6.7	The demineralized water system isolation valves close on a signal from the protection and safety monitoring system derived from either a reactor trip signal, a source range flux doubling signal, or low input voltage to the 1E dc and uninterruptible power supply battery chargers.	
Section 9.3.6.7	The chemical and volume control system makeup line isolation valves automatically close on a signal from the protection and safety monitoring system derived from a source range flux doubling, high-2 pressurizer level, High-3 steam generator level signal, high steam generator water level coincident with P-4 permissive (reactor trip), high-2 containment radioactivity, or a safeguards signal coincident with high-1 pressurizer level.	
Section 10.1.2	Safety valves are provided on both main steam lines.	
Section 10.2.2.4.3	The flow of the main steam entering the high-pressure turbine is controlled by four stop valves and four governing control valves. The stop valves are closed by actuation of the emergency trip system devices.	
Section 10.3.1.1	The main steam supply system is provided with a main steam isolation valve and associated MSIV bypass valve on each main steam line from its respective steam generator.	
Section 10.3.1.1	A main steam isolation valve (MSIV) on each main steam line prevents the uncontrolled blowdown of more than one steam generator and isolates nonsafety-related portions of the system.	
Section 10.3.1.2	Power-operated atmospheric relief valves are provided to allow controlled cooldown of the steam generator and the reactor coolant system when the condenser is not available.	
Section 10.3.2.1	The main steam supply system includes: – One main steam isolation valve and one main steam isolation valve bypass valve per main steam line. – Main steam safety valves. – Power-operated atmospheric relief valves and upstream isolation valves.	
Section 10.3.2.3.2	In the event that a design basis accident occurs, which results in a large steam line break, the main steam isolation valves with associated main steam isolation bypass valves automatically close.	

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Table 14.3-2 (Sheet 13 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Figure 10.3.2-1	The steam generator system consists of two main steam, two main feedwater, and two startup feedwater lines.	
Table 10.3.2-2	Design data for main steam supply safety system valves: – Number per main steam line – Minimum relieving capacity per valve at 110% of design pressure (lb/hr)	 6 1,370,000
Table 10.3.2-2	The flow capacity of the steam generator safety valves (lbm/hr) at 110% of design pressure.	≥ 8,240,000
Table 10.3.2-2	The maximum set pressure of the steam generator safety valves (psig).	≤ 1,242
Section 10.4.8.3	The safety-related portions of the steam generator blowdown system are located in the containment and auxiliary buildings and are designed to remain functional after a safe shutdown earthquake.	
Section 10.4.7.1.1	Double valve main feedwater isolation is provided via the main feedwater control valve and main feedwater isolation valve. Both valves close automatically on main feedwater isolation signals, an appropriate engineered safety features isolation signal, within the time established with the Technical Specifications. The startup feedwater control valve also serves as a containment isolation valve.	
Section 10.4.7.1.1	The condensate and feedwater system provides redundant isolation valves for the main feedwater lines routed into containment.	
Section 10.4.7.1.1	For a main feedwater or main steam line break (MSLB) inside the containment, the condensate and feedwater system is designed to limit high energy fluid to the broken loop.	
Section 10.4.7.1.2	The booster/main feedwater pumps are tripped simultaneously with the feedwater isolation signal to close the main feedwater isolation valves.	
Section 10.4.7.2.1	The main feedwater pumps and booster pumps are tripped with the feedwater isolation signal that closes the main feedwater isolation valves. The same isolation signal closes the isolation valve in the cross connect line between the main feedwater pump discharge header and the startup feedwater pump discharge header.	

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Table 14.3-2 (Sheet 14 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 10.4.7.2.2	One MFIV is installed in each of the two main feedwater lines outside the containment and downstream of the feedwater control valve. The MFIVs are installed to prevent uncontrolled blowdown from the steam generators in the event of a feedwater pipe rupture. The main feedwater check valve provides backup isolation. In the event of a secondary side pipe rupture inside the containment, the MFIVs limit the quantity of high energy fluid that enters the containment through the broken loop and limit cooldown. The MFCV provides backup isolation to limit cooldown and high energy fluid addition.	
Section 10.4.7.2.2	In the event of a secondary side pipe rupture inside the containment, the main feedwater control valves provide a redundant isolation to the MFIVs to limit the quantity of high energy fluid that enters the containment through the broken loop.	
Section 10.4.7.3	For a main feedwater line break inside the containment or a main steam line break, the MFIVs and the main feedwater control valves automatically close upon receipt of a feedwater isolation signal.	
Section 10.4.7.3	For a steam generator tube rupture event, positive and redundant isolation is provided for the main feedwater (MFIV and MFCV) with isolation signals generated by the protection and safety monitoring system (PMS).	
Section 10.4.8.2.2.7	Blowdown system isolation is actuated on Low-2 steam generator narrow range water levels. The isolation of steam generator blowdown provides for a continued availability of the steam generator as a heat sink for decay heat removal in conjunction with operation of the passive residual heat removal system and the startup feedwater system.	
Section 10.4.8.3	The safety-related portions of the steam generator blowdown system located in the containment and auxiliary buildings are designed to remain functional after a safe shutdown earthquake.	
Section 10.4.9.1.1	Double valve startup feedwater isolation is provided by the startup feedwater control valve and the startup feedwater isolation valve. Both valves close on a startup feedwater isolation signal, an appropriate engineered safeguards features signal, within the time established within the Technical Specifications.	

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Table 14.3-2 (Sheet 15 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 10.4.9.1.1	For a steam generator tube rupture event, positive and redundant isolation is provided for the startup feedwater system (startup feedwater isolation valve and startup feedwater control valve), with isolation signals generated by the protection and safety monitoring system.	
Section 10.4.9.2.2	In the event of a steam generator tube rupture, the startup feedwater isolation valve and startup feedwater control valve limit overfill of the steam generator by terminating startup feed flow.	
Section 10.4.9.2.2	In the event of a secondary pipe rupture inside containment, the startup feedwater isolation valve and startup feedwater control valve provide isolation to limit the quantity of high energy fluid that enters the containment.	
Section 10.4.9.2.2	The startup feedwater isolation valve is provided to prevent the uncontrolled blowdown from more than one steam generator in the event of startup feedwater line rupture. The startup feedwater isolation valve provides backup isolation.	
Table 15.0-1	Initial core thermal power (MWt).	3400
Table 15.0-3	Nominal values of pertinent plant parameters used in accident analysis with 10% steam generator tube plugging – Reactor coolant flow (gpm)	296,000
Section 15.1.2.1	Continuous addition of excessive feedwater is prevented by the steam generator High-3 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps and reactor.	
Section 15.1.4.1	For an inadvertent opening of a steam generator relief or safety valve, core makeup tank actuation occurs from one of four sources: – Two out of four Low-3 pressurizer pressure signals – Two out of four low-2 pressurizer level signals – Two out of four Low-2 T_{cold} signals in any one loop – Two out of four Low-2 steam line pressure signals in any one loop	
Section 15.1.4.1	After an inadvertent opening of a steam generator relief or safety valve, redundant isolation of the main feedwater lines closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.	

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Table 14.3-2 (Sheet 16 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 15.1.5.1	Following a steam line rupture, core makeup tank actuation occurs from one of five sources: – Two out of four Low-3 pressurizer pressure signals – Two out of four high-2 containment pressure signals – Two out of four Low-2 steam line pressure signals in any loop – Two out of four Low-2 T_{cold} signals in any one loop – Two out of four low-2 pressurizer level signals	
Section 15.1.5.1	After a steam line rupture, redundant isolation of the main feedwater lines closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.	
Section 15.1.5.2.1	Core makeup tanks and the accumulators are the portions of the passive core cooling system used in mitigating a steam line rupture.	
Section 15.1.6.1	The heat sink for the PRHR heat exchanger is provided by the IRWST, in which the PRHR heat exchanger is submerged.	
Section 15.2.6.2.1	Following a loss of ac power, the PRHR heat exchanger is actuated by Low-2 steam generator wide range water level.	
Section 15.2.8.2.1	Receipt of a Low-2 steam line pressure signal in at least one steam line initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives a safeguards signal that initiates flow of cold borated water from the core makeup tanks to the reactor coolant system.	
Section 15.3.3.2.2	The pressurizer safety valves are fully open at 2575 psia. Their capacity for steam relief is described in Section 5.4.	
Section 15.4.6.2.2	A safety signal from the protection and safety monitoring system automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution.	
Section 15.5.1.1	Following inadvertent operation of the core makeup tanks during power operation, the high-3 pressurizer level signal actuates the PRHR heat exchanger and blocks the pressurizer heaters.	

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Table 14.3-2 (Sheet 17 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 15.5.2.1	The pressurizer heaters are blocked, and the main feedwater lines, steam lines, and chemical and volume control system are isolated.	
Table 15.6.5-10	ADS Valve Minimum Effective Flow Areas (in^2) – ADS Stage 1 Control Valve – ADS Stage 2 Control Valve – ADS Stage 3 Control Valve – ADS Stage 4A Valve – ADS Stage 4B Valve	 ≥ 4.6 ≥ 16 ≥ 16 ≥ 66 ≥ 66
Table 15.6.5-10	ADS Valve Opening Times (sec) – ADS Stage 1 Control Valve – ADS Stage 1 Isolation Valve – ADS Stage 2 Control Valve – ADS Stage 2 Isolation Valve – ADS Stage 3 Control Valve – ADS Stage 3 Isolation Valve	 ≤ 48 ≤ 30 ≤ 120 ≤ 60 ≤ 120 ≤ 60
Section 18.8.3.2	The main control area includes the reactor operator workstations, the supervisor's workstation, the dedicated safety panel and the wall panel information system.	
Section 18.8.3.2	The human system interface resources available at each workstation are the plant information system displays, the control displays (soft controls), the alarm system displays, computerized procedure system, and the screen and component selector.	

Note:

The ADS Stage 4A Valve and ADS Stage 4B Valve minimum effective flow areas of ≥ 66 in^2 are assumed in the small-break LOCA analyses to account for potential deformation during actuation that may reduce the effective flow area to less than the required as-manufactured flow area of ≥ 67 in^2.

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Table 14.3-3
Anticipated Transient Without Scram

Reference	Design Feature	Value
Section 7.7.1.11	The diverse actuation system is a nonsafety-related system that provides a diverse backup to the protection and safety monitoring system.	
Section 7.7.1.11	The diverse actuation system trips the reactor control rods and the turbine on low wide range steam generator water level, or on low pressurizer water level, or on high hot leg temperature.	
Section 7.7.1.11	The diverse actuation system initiates passive residual heat removal on low wide range steam generator water level or high hot leg temperature; actuates core makeup tanks and trips the reactor coolant pumps on low pressurizer water level; and isolates selected containment penetrations and starts passive containment cooling on high containment temperature.	
Section 7.7.1.11	The manual actuation function of the diverse actuation system is implemented by wiring the controls located in the main control room directly to the final loads in a way that bypasses the normal path through the protection and safety monitoring system cabinets and the diverse actuation system logic.	
Section 7.7.1.11	The diverse actuation system uses microprocessor or special purpose logic processor boards different from those used in the protection and safety monitoring system.	
Section 7.7.1.11	The diverse actuation system hardware implementation is different from that of the protection and safety monitoring system.	
Section 7.7.1.11	The operating system and programming language of the diverse actuation system is different from that of the protection and safety monitoring system.	

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Table 14.3-4 (Sheet 1 of 2)
Fire Protection

Reference	Design Feature	Value
Section 9A.3.1	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section 3.4.1.1.2	The AP1000 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section 3.8.4.1.1	The conical roof supports the passive containment cooling system tank, which is constructed with a stainless steel liner on reinforced concrete walls.	
Section 7.1.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	
Section 7.4.3.1	If temporary evacuation of the main control room is required because of some abnormal main control room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the main control room through the use of controls and monitoring located at the remote shutdown workstation.	
Section 7.4.3.1.1	The remote shutdown workstation equipment is similar to the operator workstations in the main control room and is designed to the same standards. One remote shutdown workstation is provided.	
Section 7.4.3.1.3	The remote shutdown workstation achieves and maintains safe shutdown conditions from full power conditions and maintains safe shutdown conditions thereafter.	
Section 8.3.2.2	The four divisions of Class 1E battery chargers and Class 1E voltage regulating transformers are independent, located in separate rooms, cannot be interconnected, and their circuits are routed in dedicated, physically separated raceways.	
Section 8.3.2.3	Each safety-related circuit and the raceway that routes them is given a unique identification number to distinguish between circuits and raceways of different voltage level or separation groups.	

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Table 14.3-4 (Sheet 2 of 2)
Fire Protection

Reference	Design Feature	Value
Section 8.3.2.4.2	Cables of one separation group are run in separate raceway and physically separated from cables of other separation groups. Group N raceways are separated from safety-related groups A, B, C, and D. Non-class 1E circuits are electrically isolated by isolation devices, shielding and wiring techniques, physical separation, or an appropriate combination thereof.	
Section 9.5.1.2.1.1	Separation is maintained between redundant safe shutdown components, including equipment, electrical cables, and instrumentation controls, in accordance with the fire areas.	
Section 9.5.1.2.1.5	The standpipe system is supplied with water from the safety-related passive containment cooling system storage tank and normally operates independently of the rest of the fire protection system. The supply line draws water from a portion of the storage tank, using water allocated for fire protection.	
Section 9.5.1.2.1.5	The standpipe system serving areas containing equipment required for safe shutdown following a safe shutdown earthquake is designed and supported so that it can withstand the effects of a safe shutdown earthquake and remain functional.	
Section 9.5.1.2.1.5	The volume of the water in the PCS tank is sufficient to supply two hose streams, each with a flow of 75 gallons per minute, for two hours (gal).	≥ 18,000
Table 9.5.1-2	Each fire pump is rated: – Flow rate (gpm) – Total head (ft)	 ≥ 2000 ≥ 350
Section 18.8.3.2	The human system interface resources available at each workstation are the plant information system displays, the control displays (soft controls), the alarm system displays, computerized procedure system, and the screen and component selector.	
Section 18.8.3.4	The mission of the remote shutdown workstation is to provide the resources to bring the plant to a safe shutdown condition after an evacuation of the main control room.	
Section 18.12.3	The controls, displays, and alarms listed in Table 18.12.2-1 are retrievable from the remote shutdown workstation.	

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Table 14.3-5 (Sheet 1 of 2)
Flood Protection

Reference		Design Feature	Value
Section	Appendix 1-A RG 1.143 Section C.1.1.3 Clarification	The lowest level of the auxiliary building, elevation 66′-6″, contains the components of the radwaste system within a common flood zone with watertight floors and walls. This volume of this enclosed flood zone is sufficient to contain the contents of the radwaste system.	
Table	2-1	Plant elevation for maximum flood level (ft).	≤ 100
Section	3.4.1.1.1	The seismic Category I structures below grade are protected against flooding by a waterproofing system.	
Section	3.4.1.1.2	The boundaries between mechanical equipment rooms and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3.4.1.2.2	The boundaries between mechanical equipment rooms inside containment and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3.4.1.2.2	Boundaries exist to prevent flooding between the following rooms which contain safety-related equipment: PXS valve/ accumulator room A, PXS valve/accumulator room B, and chemical and volume control room.	
Section	3.4.1.2.2	The AP1000 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section	3.4.1.2.2	The safety-related components available for safety shutdown are located in the auxiliary building and inside containment. No credit is taken for operation of sump pumps to mitigate the consequences of flooding.	
Section	3.4.1.2.2.1	The PXS-A compartment, PXS-B compartment and the chemical and volume control system compartment are physically separated and isolated from each other by structural walls such that flooding in any one of these compartments cannot cause flooding in any of the other compartments at elevations up to the top of these compartments.	

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Table 14.3-5 (Sheet 2 of 2)
Flood Protection

Reference	Design Feature	Value
Section 3.6	In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided so that essential structures, systems, or components are not impacted by the adverse effects of postulated pipe failure.	
Section 7.1.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	

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Table 14.3-6 (Sheet 1 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Table 3.2-2	The Nuclear Island structures include the containment and the shield and auxiliary buildings. These structures are seismic Category I.	
Table 3.2-3	The components identified under Reactor Systems in Table 3.2-3, as ASME Code Section III are designed and constructed in accordance with ASME Code Section III Requirements.	
Table 3.2-3	The Nuclear Island structures include the containment and the Shield and Auxiliary Buildings. These structures are seismic Category I.	
Section 3.4.1.1.2	The boundaries between mechanical equipment rooms and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section 3.4.1.1.2	The AP1000 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section 9A.3.1	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section 3.4.1.2.2	Boundaries exist to prevent flooding between the following rooms which contain safety-related equipment: PXS valve/ accumulator room A, PXS valve/accumulator room B, and chemical and volume control room.	
Section 3.4.1.2.2	The boundaries between mechanical equipment rooms inside containment and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section 3.4.1.2.2	The safety-related components available for safety shutdown are located in the auxiliary building and inside containment. No credit is taken for operation of sump pumps to mitigate the consequences of flooding.	

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Table 14.3-6 (Sheet 2 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 3.4.1.2.2.1	The PXS-A compartment, PXS-B compartment and the chemical and volume control system compartment are physically separated and isolated from each other by structural walls such that flooding in any one of these compartments or in the reactor coolant system compartment cannot cause flooding in any of the other compartments.	
Section 3.11.3	The design of the protection and safety monitoring system equipment has margin to accommodate a loss of the normal HVAC.	
Section 3D.6	RXS equipment in Appendix 3D is seismically qualified.	
Section 5.1.3.7	ADS has four stages. Each stage is arranged into two separate groups of valves and lines. – Stages 1, 2, and 3 discharge from the top of the pressurizer to the IRWST. – Each stage 4 discharges from a hot leg to the RCS loop compartment.	
Section 5.3.1.1	The reactor vessel provides a high integrity pressure boundary to contain the reactor coolant, heat generating reactor core, and fuel fission products. The reactor vessel is the primary boundary for the reactor coolant and the secondary barrier against the release of radioactive fission products.	
Section 5.4.6	ADS has four stages. Each stage is arranged into two separate groups of valves and lines. – Stages 1, 2, and 3 discharge from the top of the pressurizer to the IRWST. – Each stage 4 discharges from a hot leg to the RCS loop compartment.	
Section 5.4.6.2	Each ADS stage 1, 2, and 3 line contains two normally closed motor-operated valves (MOVs).	
Section 5.4.6.2	Each ADS stage 4 line contains a normally open MOV valve and a normally closed squib valve.	
Section 5.4.7	The RNS removes heat from the core and reactor coolant system at reduced RCS pressure and temperature conditions after shutdown.	

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Table 14.3-6 (Sheet 3 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 5.4.7	The normal residual heat removal system (RNS) provides a safety-related means of performing the following functions: – Containment isolation for the RNS lines that penetrate the containment – Long-term, post-accident makeup water to the RCS	
Section 5.4.7.1.1	The RNS containment isolation and pressure boundary valves are safety-related. The motor-operated valves are powered by Class 1E dc power.	
Section 5.4.7.1.2.1	The component cooling water system (CCS) provides cooling to the RNS heat exchanger.	
Section 6.2.4	The containment hydrogen control system provides nonsafety-related hydrogen igniters for control of the containment hydrogen concentration for beyond design basis accidents.	
Section 6.2.4.2.3	At least 66 hydrogen igniters are provided.	
Section 6.3.1.1.3	The automatic depressurization system provides a safety-related means of depressurizing the RCS.	
Section 6.3	The in-containment refueling water storage tank subsystem provides a safety-related means of performing the following functions: – Low-pressure safety injection – Core decay heat sink during design basis events – Flooding of the lower containment, the reactor cavity and the loop compartment by draining the IRWST into the containment. – Borated water	
Section 6.3.1	The core makeup tanks provide safety-related means of safety injection of borated water to the RCS.	
Section 6.3.1	Passive residual heat removal (PRHR) provides a safety-related means of removing core decay heat during design basis events.	
Section 6.3.2	The ADS valves are powered from Class 1E dc power.	

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Table 14.3-6 (Sheet 4 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 6.3.2	There are two CMTs, each with an injection line to the reactor vessel/DVI nozzle. – Each CMT has a pressure balance line from an RCS cold leg. – Each injection line is isolated with a parallel set of air-operated valves (AOVs). – These AOVs open on loss of air. – The injection line for each CMT also has two check valves in series.	
Section 6.3.2	The IRWST subsystem has the following flow paths: – Two (redundant) injection lines from the IRWST to the reactor vessel/DVI nozzle. Each line is isolated with a parallel set of valves; each set with a check valve in series with a squib valve. – Two (redundant) recirculation lines from the containment to the IRWST injection line. Each recirculation line has two paths: one path contains a squib valve and an MOV, the other path contains a squib valve and a check valve. – The two MOV/squib valve lines also provide the capability to flood the reactor cavity.	
Section 6.3.2	There are screens for each IRWST injection line and recirculation line.	
Section 6.3.2	PRHR is actuated by opening redundant, parallel air-operated valves. These air-operated valves open on loss of air.	
Section 6.3.2.2	The passive core cooling system (PXS) is composed of the following: – Accumulator subsystem – Core makeup tank (CMT) subsystem – In-containment refueling water storage tank (IRWST) subsystem – Passive residual heat removal (PRHR) subsystem. – The automatic depressurization system (ADS), which is a subsystem of the reactor coolant system (RCS), also supports passive core cooling functions.	
Section 6.3.2.2.2	There are two accumulators, each with an injection line to the reactor vessel/direct vessel injection (DVI) nozzle. Each injection line has two check valves in series.	

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Table 14.3-6 (Sheet 5 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 6.3.2.2.2	The accumulators provide a safety-related means of safety injection of borated water to the RCS.	
Section 6.3.2.2.8.7	The accumulator discharge check valves are of a different type than the CMT discharge check valves.	
Section 6.3.3	IRWST squib valves and MOVs are powered by Class 1E dc power.	
Section 6.3.3	The CMT AOVs are automatically and manually actuated from PMS and DAS.	
Section 6.3.3	The PRHR air-operated valves are automatically actuated and manually actuated from the control room by either PMS or DAS.	
Section 6.3.3	The squib valves and MOVs for injection and recirculation are automatically and manually actuated via PMS, and the squib valves are manually actuated via DAS.	
Section 6.3.3	The squib valves and MOVs for reactor cavity flooding are manually actuated via PMS from the control room. The squib valves are manually actuated via DAS from the control room.	
Section 6.3.7	The positions of the containment recirculation isolation MOVs are indicated in the control room.	
Section 6.3.7	The position of the inlet PRHR valve is indicated in the control room.	
Section 6.3.7.6.1	The ADS first-, second-, and third-stage valve positions are indicated in the control room.	
Section 7.1.1	The diverse actuation system provides a nonsafety-related means of performing the following functions: – Initiates automatic and manual reactor trip – Automatic and manual actuation of selected engineered safety features – Main control room display of selected plant parameters.	
Section 7.1.1	The protection and safety monitoring system provides a safety-related means of performing the following functions: – Automatic and manual reactor trip – Automatic and manual actuation of engineered safety features (ESF).	

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Table 14.3-6 (Sheet 6 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 7.1.1	PMS provides for the minimum inventory of fixed position controls and displays in the control room.	
Section 7.1.2	Each PMS division is powered from its respective Class 1E dc division.	
Section 7.1.2	PMS has four divisions of reactor trip and ESF actuation.	
Section 7.1.2.5	PMS has two divisions of safety-related post-accident parameter data processing equipment.	
Section 7.1.2.9	PMS automatically blocks an attempt to bypass more than one channel of a function that uses 2-out-of-4 logic.	
Section 7.1.2.14	The PMS hardware and software are developed using a planned design process which provides for specific design documentation and reviews during the design requirement, system definition, development, test and installation phases.	
Section 7.1.4.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	
Section 7.1.2	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
Section 7.2.2.2.1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
Section 7.4.3 18.12.2	The PMS allows for the transfer of control capability from the main control room to the remote shutdown workstation. The minimum inventory of displays and controls in the remote shutdown workstation is provided.	
Section 7.3.1 8.3.2.1.1	The ADS valves are powered from Class 1E dc power.	
Section 7.7.1.11 7.3.1.2.4	The ADS valves are automatically and manually actuated via the protection and safety monitoring system (PMS), and manually actuated via the diverse actuation system (DAS).	
Section 7.3.1.2.3 7.7.1.11	The CMT AOVs are automatically and manually actuated from PMS and DAS.	
Section 7.3.1.2.2 7.3.1.2.9 7.7.1.11	The squib valves and MOVs for injection and recirculation are automatically and manually actuated via PMS, and the squib valves are manually actuated via DAS.	

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Table 14.3-6 (Sheet 7 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Figure 7.2-1 (Sheets 16 and 21)	The squib valves and MOVs for reactor cavity flooding are manually actuated via PMS from the control room. The squib valves are manually actuated via DAS from the control room.	
Section 7.3.1.2.7 7.7.1.11	The PRHR air-operated valves are automatically actuated and manually actuated from the control room by either PMS or DAS.	
Section 7.3.1.2.20	The RNS containment isolation MOVs are actuated via PMS.	
Section 7.5.4	PMS has four divisions of safety-related post-accident parameter display.	
Section 7.6.1.1	An interlock is provided for the normally closed motor-operated normal residual heat removal system inner and outer suction isolation valves. Each valve is interlocked so that it cannot be opened unless the reactor coolant system pressure is below a preset pressure.	
Section 7.7.1.11	The diverse actuation system is a nonsafety-related system that provides a diverse backup to the protection and safety monitoring system.	
Section 7.7.1.11	The diverse actuation system trips the reactor control rods and the turbine on low wide range steam generator water level, or on low pressurizer water level, or on high hot leg temperature.	
Section 7.7.1.11	DAS manual initiation functions are implemented in a manner that bypasses the signal processing equipment of the DAS.	
Section 7.7.1.11	The DAS automatic actuation signals are generated in a functionally diverse manner from the PMS signals. Diversity between DAS and PMS is achieved by the use of different architectures, different hardware implementations, and different software, if any. Software diversity between the DAS and PMS will be achieved through the use of different algorithms, logic, program architecture, executable operating system, and executable software/logic.	

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Table 14.3-6 (Sheet 8 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 8.3.1.1.1	On loss of power to a 6900V diesel-backed bus, the associated diesel generator automatically starts and produces ac power. The source circuit breakers and bus load circuit breakers are opened, and the generator is connected to the bus. Each generator has an automatic load sequencer to enable controlled loading on the associated buses.	
Section 8.3.1.1.2.1	Two onsite standby diesel generator units provide power to the selected nonsafety-related ac loads.	
Section 8.3.1.1.4	The main ac power system distributes non-Class 1E power from onsite sources to selected nonsafety-related loads.	
Section 8.3.2.1	The Class 1E dc and uninterruptible power supply (UPS) system (IDS) provides dc and uninterruptible ac power for the safety-related equipment.	
Section 8.3.2.1.1.1	There are four independent, Class 1E 250 Vdc divisions. Divisions A and D are each composed of one battery bank, one switchboard, and one battery charger. Divisions B and C are each composed of two battery banks, two switchboards, and two battery chargers. The first battery bank in the four divisions is designated as the 24-hour battery bank. The second battery bank in Divisions B and C is designated as the 72-hour battery bank.	
Section 8.3.2.1.1.1	Battery chargers are connected to dc switchboard buses. The input ac power for the Class 1E dc battery chargers is supplied from onsite diesel-generator-backed low-voltage ac power supplies.	
Section 8.3.2.1.1.1	The 24-hour battery banks provide power to the loads for a period of 24 hours without recharging. The 72-hour battery banks supply a dc switchboard bus load for a period of 72 hours without recharging.	
Section 8.3.2.1.2	The non-Class 1E dc and UPS system (EDS) consists of the electric power supply and distribution equipment that provides dc and uninterruptible ac power to nonsafety-related loads.	
Section 8.3.2.1.2	EDS load groups 1, 2, 3, and 4 provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E uninterruptible power supply ac system.	

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Table 14.3-6 (Sheet 9 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 8.3.2.1.2	Battery chargers are connected to dc switchboard buses. The input ac power for the non-Class 1E dc battery chargers is supplied from onsite diesel-generator-backed low-voltage ac power supplies.	
Section 8.3.2.1.2	The onsite standby diesel-generator-backed low-voltage ac power supply provides the normal ac power to the battery chargers.	
Section 8.3.2.4.2	Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cables.	
Section 9.2.1	The service water system is a nonsafety-related system that transfers heat from the component cooling water heat exchangers to the atmosphere.	
Section 9.2.1.2.1	The SWS is arranged into two trains. Each train includes one pump and one cooling tower cell.	
Section 9.2.2	The component cooling water system is a nonsafety-related system that removes heat from various components and transfers the heat to the service water system (SWS).	
Section 9.2.2.2	The CCS is arranged into two trains. Each train includes one pump and one heat exchanger.	
Section 9.3.6	The CVS provides a nonsafety-related means to perform the following functions: – Makeup water to the RCS during normal plant operation – Boration following a failure of reactor trip – Coolant to the pressurizer auxiliary spray line.	
Section 9.3.6.1.1	The chemical and volume control system (CVS) provides a safety-related means to terminate inadvertent RCS boron dilution.	
Section 9.4.1	The main control room has its own ventilation system and is pressurized. The ventilation system for the remote shutdown room is independent of the ventilation system for the main control room.	
Section 9.5.1.2.1.1	The PMS allows for the transfer of control capability from the main control room to the remote shutdown workstation. The minimum inventory of displays and controls at the remote shutdown workstation is provided.	

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Table 14.3-6 (Sheet 10 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 9.5.1.2.1.1	Class 1E divisional cables are routed in their respective divisional raceways.	
Section 9.5.1.2.1.1	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section 17.4.1	D-RAP provides reasonable assurance that the design of risk-significant SSCs is consistent with their PRA assumptions.	
Section 18.8.3.2	The main control area includes the reactor operator workstations, the supervisor's workstation, the dedicated safety panel and the wall panel information system.	
Section 18.12.2	The minimum inventory of instrumentation includes those displays, controls, and alarms that are used to monitor the status of the critical safety functions and to manually actuate the safety-related systems that achieve the critical safety functions. The minimum inventory resulting from the implementation of the selection criteria is provided in Table 18.12.2-1.	

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Table 14.3-7 (Sheet 1 of 3)
Radiological Analysis

Reference		Design Feature	Value
Table	2-1	Plant elevation for maximum flood level (ft)	≤ 100
Section	2.3.4	Atmospheric dispersion factors - X/Q (sec/m^3) – Site Boundary X/Q 0 - 2 hour time interval – Low Population Zone Boundary X/Q 0 - 8 hours 8 - 24 hours 24 - 96 hours 96 - 720 hours	 ≤ 5.1×10^{-4} ≤ 2.2×10^{-4} ≤ 1.6×10^{-4} ≤ 1.0×10^{-4} ≤ 8.0×10^{-5}
Table	6.2.3-1	Containment penetration isolation features are configured as in Table 6.2.3-1	
Table	6.2.3-1	Maximum closure time for remotely operated containment purge valves (seconds)	≤ 10
Table	6.2.3-1	Maximum closure time for all other remotely operated containment isolation valves (seconds)	≤ 60
Section	6.4.2.3	The minimum storage capacity of all storage tanks in the VES (scf)	≥ 327,574
Section	6.4.4	The maximum temperature in the instrumentation and control rooms and dc equipment rooms following a loss of the nuclear island nonradioactive ventilation system remains over a 72-hour period (°F).	≤ 120
Section	6.4.4	The main control emergency habitability system nominally provides 65 scfm of ventilation air to the main control room from the compressed air storage tanks.	65 ± 5
Section	6.4.4	Sixty-five ± five scfm of ventilation flow is sufficient to pressurize the control room to $1/8^{th}$ inch water gauge differential pressure (WIC).	$1/8^{th}$
Section	6.4.5.1	The maximum temperature in the main control room pressure boundary following a loss of the nuclear island nonradioactive ventilation system over a 72-hour period (°F) (dry bulb temperature).	95
Figure	6.4-2	The main control room emergency habitability system consists of four sets of emergency air storage tanks and an air delivery system to the main control room.	
Section	6.5.3	The passive heat removal process and the limited leakage from the containment result in offsite doses less than the regulatory guideline limits.	

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Table 14.3-7 (Sheet 2 of 3)
Radiological Analysis

Reference	Design Feature	Value
Section 8.3.1.1.6	Electrical penetrations through the containment can withstand the maximum short-circuit currents available either continuously without exceeding their thermal limit, or at least longer than the field cables of the circuits so that the fault or overload currents are interrupted by the protective devices prior to a potential failure of a penetration.	
Section 9.4.1.1.1	The VBS isolates the HVAC ductwork that penetrates the main control room boundary on High-2 particulate or iodine concentration in the main control room supply air, extended Low main control room differential pressure, or on extended loss of ac power to support operation of the main control room emergency habitability system.	
Section 12.3.2.2.1	During reactor operation, the shield building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete shield building wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the shield building to less than 0.25 mrem/hr from sources inside containment. The shield building completely surrounds the reactor components.	
Section 12.3.2.2.2	The reactor vessel is shielded by the concrete primary shield and by the concrete secondary shield which also surrounds other primary loop components. The secondary shield is a structural module filled with concrete surrounding the reactor coolant system equipment, including piping, pumps and steam generators. Extensive shielding is provided for areas surrounding the refueling cavity and the fuel transfer canal to limit the radiation levels.	
Section 12.3.2.2.3	Shielding is provided for the liquid radwaste, gaseous radwaste and spent resin handling systems consistent with the maximum postulated activity. Corridors are generally shielded to allow Zone II access, and operator areas for valve modules are generally Zone II or III for access. Shielding is provided to attenuate radiation from normal residual heat removal equipment during shutdown cooling operations to levels consistent with radiation zoning requirements of adjacent areas.	

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Table 14.3-7 (Sheet 3 of 3)
Radiological Analysis

Reference	Design Feature	Value
Section 12.3.2.2.4	The concrete shield walls surrounding the spent fuel cask loading and decontamination areas, and the shield walls surrounding the fuel transfer and storage are sufficiently thick to limit radiation levels outside the shield walls in accessible areas to Zone II. The building walls are sufficient to shield external plant areas which are not controlled to Zone I.	
Section 12.3.2.2.5	Shielding is provided as necessary for the waste storage areas in the radwaste building to meet the radiation zone and access requirements.	
Section 12.3.2.2.7	Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated loss-of-coolant accident, so that radiation doses are limited to five rem whole body from contributing modes of exposure for the duration of the accident, in accordance with General Design Criteria 19.	
Section 12.3.2.2.9	The spent fuel transfer tube is shielded to within adjacent area radiation limits, is completely enclosed in concrete, and there is no unshielded portion of the spent fuel transfer tube during the refueling operation.	

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Table 14.3-8
Severe Accident Analysis

Reference	Design Feature	Value
Section 1.2	The discharge from the IRWST vents located in the roof of the IRWST next to the containment vessel are oriented away from the containment vessel.	
Section 5.3.1.2	There are no penetrations in the reactor vessel below the core.	
Section 5.3.5	The reflective (or suitable equivalent) reactor vessel insulation provides an engineered flow path to allow the ingression of water and venting of steam for externally cooling the vessel. – A flow path exists from the loop compartment to the reactor vessel cavity (ft^2). – A flow path area to vent steam exists between the vessel insulation and the reactor vessel (ft^2).	 ≥ 6 ≥ 12
Section 6.2.4.2.3	The hydrogen ignition subsystem consists of 66 hydrogen igniters strategically distributed throughout the containment.	
Table 6.2.4-3	The minimum surface temperature of the hydrogen igniters (°F).	≥ 1700
Section 6.3	The ADS provides a safety-related means of depressurizing the RCS.	
Section 6.3	The PXS provides a safety-related means of flooding the reactor cavity by draining the IRWST into the containment.	
Section 7.3.1.2.9	Signals to align the IRWST containment recirculation isolation valves are generated by manual initiation.	
Section 7.7.1.11	Initiation of containment recirculation is a diverse manual function.	

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14.4 Combined License Applicant Responsibilities

14.4.1 Organization and Staffing

The staff, staff responsibilities, authorities, and personnel qualifications for performing the AP1000 initial test program are addressed in Section 14.2. This test organization is responsible for the planning, executing, and documenting of the plant initial testing and related activities that occur between the completion of plant/system/component construction and commencement of plant commercial operation. Transfer and retention of experience and knowledge gained during initial testing for the subsequent commercial operation of the plant is an objective of the test program.

14.4.2 Test Procedures

Preoperational and startup test procedures are available to the NRC in accordance with the requirements of Subsection 14.2.3. The controls for development of test procedures are also described in Subsection 14.2.3.

A cross reference list is provided between ITAACs and test procedures and/or sections of test procedures.

14.4.3 Conduct of Initial Test Program

A site-specific ITP Administrative Manual (procedures), which contains the administrative procedures and requirements that govern the activities associated with the plant initial test program, as described in Section 14.2 is provided.

14.4.4 Review and Evaluation of Test Results

Review and evaluation of individual test results, as well as final review of overall test results and selected milestones or hold points are addressed in Subsection 14.2.3.2. Test exceptions or results that do not meet acceptance and/or functional criteria are identified to the affected and responsible design organizations, and corrective actions and retests, as required, are performed.

14.4.5 Interface Requirements

The test abstracts required of structures and systems which are outside the scope of the design certification are addressed in Subsections 14.2.9.4.15, 14.2.9.4.23 through 14.2.9.4.27, 14.2.10.4.30, and in the Physical Security Plan.

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Appendix 14A Design Acceptance Criteria/ITAAC Closure Process

Design Acceptance Criteria (DAC) are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification. DAC is to be objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as a part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design (SECY 92-053).

There are three process options for DAC/ITAAC resolution:

- Resolve through amendment to design certification
- Resolve as part of COL review
- Resolve after COL is issued

In the first two options, the applicant will submit the design information and the NRC will document its review in a safety evaluation. In the third option, the COL holder notifies the NRC of availability of design information and the staff will document its review in an inspection report.

Should the third option be implemented for the first standard AP1000 plant, subsequent COL applicants may reference the first standard plant closure documentation and close the DAC/ITAAC under the concept of “one issue, one review, one position,” identified in NRC guidance.

Additionally, Westinghouse may submit licensing topical reports for NRC review of the material supporting the DAC/ITAAC closure and request that the NRC issue a safety evaluation in conjunction with a closure letter or inspection report concluding that the acceptance criteria of the DAC/ITAAC have been met. Subsequent COL applicants may reference these reports and NRC closure documents in an effort to close the DAC/ITAAC.

For technical areas where DAC/ITAAC applies in the design certification rule, COL applicants will provide an ITAAC and associated closure schedule indicating the approach to be applied. For subsequent COL applicants following the first standard AP1000 plant, the indication could be to reference the existing DAC/ITAAC closure documentation for the first standard plant.

NRC guidance for DAC/ITAAC is provided in Regulatory Guide 1.206, Section C.III.5. Further information on the staff’s position of DAC/ITAAC being used as part of the 10 CFR Part 52 review process is provided in SECY-92-053.

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Chapter 15 Accident Analyses

15.0.1 Classification of Plant Conditions

The ANSI 18.2 (Reference 1) classification divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

- Condition I: Normal operation and operational transients
- Condition II: Faults of moderate frequency
- Condition III: Infrequent faults
- Condition IV: Limiting faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk, and those extreme situations having the potential for the greatest risk should be those least likely to occur. Where applicable, reactor trip and engineered safeguards functioning are assumed to the extent allowed by considerations such as the single failure criterion in fulfilling this principle.

15.0.1.1 Condition I: Normal Operation and Operational Transients

Condition I occurrences are those that are expected to occur frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between a plant parameter and the value of that parameter requiring either automatic or manual protective action.

Because Condition I events occur frequently, they must be considered from the point of view of their effect on the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions that can occur during Condition I operation.

A typical list of Condition I events follows.

Steady-State and Shutdown Operations

See Table 1.1-1 of the Technical Specifications.

Operation with Permissible Deviations

Various deviations that occur during continued operation as permitted by the plant Technical Specifications are considered in conjunction with other operational modes. These deviations include the following:

- Operation with components or systems out of service (such as an inoperable rod cluster control assembly [RCCA])
- Leakage from fuel with limited cladding defects
- Excessive radioactivity in the reactor coolant:
  - Fission products
  - Corrosion products
  - Tritium
- Operation with steam generator tube leaks

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- Testing

Operational Transients

- Plant heatup and cooldown
- Step load changes (up to ±10 percent)
- Ramp load changes (up to 5 percent/minute)
- Load rejection up to and including design full-load rejection transient

15.0.1.2 Condition II: Faults of Moderate Frequency

These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault (Condition III or IV events). In addition, Condition II events are not expected to result in fuel rod failures, reactor coolant system failures, or secondary system overpressurization. The following faults are included in this category:

- Feedwater system malfunctions that result in a decrease in feedwater temperature (see Subsection 15.1.1)
- Feedwater system malfunctions that result in an increase in feedwater flow (see Subsection 15.1.2)
- Excessive increase in secondary steam flow (see Subsection 15.1.3)
- Inadvertent opening of a steam generator relief or safety valve (see Subsection 15.1.4)
- Inadvertent operation of the passive residual heat removal heat exchanger (see Subsection 15.1.6)
- Loss of external electrical load (see Subsection 15.2.2)
- Turbine trip (see Subsection 15.2.3)
- Inadvertent closure of main steam isolation valves (see Subsection 15.2.4)
- Loss of condenser vacuum and other events resulting in turbine trip (see Subsection 15.2.5)
- Loss of ac power to the station auxiliaries (see Subsection 15.2.6)
- Loss of normal feedwater flow (see Subsection 15.2.7)
- Partial loss of forced reactor coolant flow (see Subsection 15.3.1)
- Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition (see Subsection 15.4.1)
- Uncontrolled RCCA bank withdrawal at power (see Subsection 15.4.2)
- RCCA misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly) (see Subsection 15.4.3)
- Startup of an inactive reactor coolant pump at an incorrect temperature (see Subsection 15.4.4)

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- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (see Subsection 15.4.6)
- Inadvertent operation of the core makeup tanks during power operation (see Subsection 15.5.1)
- Chemical and volume control system malfunction that increases reactor coolant inventory (see Subsection 15.5.2)
- Inadvertent opening of a pressurizer safety valve (see Subsection 15.6.1)
- Failure of small lines carrying primary coolant outside containment (see Subsection 15.6.2)

15.0.1.3 Condition III: Infrequent Faults

Condition III events are faults that may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with the guidelines of 10 CFR 50.34. By definition, a Condition III event alone does not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

- Steam system piping failure (minor) (see Subsection 15.1.5)
- Complete loss of forced reactor coolant flow (see Subsection 15.3.2)
- RCCA misalignment (single RCCA withdrawal at full power) (see Subsection 15.4.3)
- Inadvertent loading and operation of a fuel assembly in an improper position (see Subsection 15.4.7)
- Inadvertent operation of automatic depressurization system (see Subsection 15.6.1)
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (see Subsection 15.6.5)
- Gas waste management system leak or failure (see Subsection 15.7.1)
- Liquid waste management system leak or failure (atmospheric release) (see Subsection 15.7.2)
- Release of radioactivity to the environment due to a liquid tank failure (see Subsection 15.7.3)
- Spent fuel cask drop accidents (see Subsection 15.7.5)

15.0.1.4 Condition IV: Limiting Faults

Condition IV events are faults that are not expected to take place, but are postulated because their consequences include the potential of the release of significant amounts of radioactive material. They are the faults that must be designed against, and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of the guideline values of 10 CFR 50.34. A single Condition IV event is not to cause a consequential loss of

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required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults are classified in this category:

- Steam system piping failure (major) (see Subsection 15.1.5)
- Feedwater system pipe break (see Subsection 15.2.8)
- Reactor coolant pump shaft seizure (locked rotor) (see Subsection 15.3.3)
- Reactor coolant pump shaft break (see Subsection 15.3.4)
- Spectrum of RCCA ejection accidents (see Subsection 15.4.8)
- Steam generator tube rupture (see Subsection 15.6.3)
- LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (see Subsection 15.6.5)
- Fuel handling accidents (see Subsection 15.7.4)

15.0.2 Optimization of Control Systems

A control system setpoint study is performed prior to plant operation to simulate performance of the primary plant control systems and overall plant performance. In this study, emphasis is placed on the development of the overall plant control systems that automatically maintain conditions in the plant within the allowed operating window and with optimum control system response and stability over the entire range of anticipated plant operating conditions. The control system setpoints are developed using the nominal protection and safety monitoring system setpoints implemented in the plant. Where appropriate (such as in margin to reactor trip analyses), instrumentation errors are considered and are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and plant control system setpoint study in combination show that the plant can be operated and meet both safety and operability requirements throughout the core life and for various levels of power operation.

The plant control system setpoint study is comprised of analyses of the following control systems: plant control, axial offset control, rapid power reduction, steam dump (turbine bypass), steam generator level, pressurizer pressure, and pressurizer level.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values assumed in the analyses performed. The thermal power output includes the effective thermal power generated by the reactor coolant pumps. Selected AP1000 loop layout elevations are shown in Figure 15.0.3-2 to aid in interpreting plots shown in other Chapter 15 subsections.

The values of other pertinent plant parameters used in the accident analyses are given in Table 15.0-3.

15.0.3.2 Initial Conditions

For most accidents that are departure from nucleate boiling (DNB) limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a

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statistical basis and are included in the departure from nucleate boiling ratio (DNBR) design limit values (see Section 4.4), as described in WCAP-11397-P-A (Reference 2). This procedure is known as the Revised Thermal Design Procedure (RTDP) and is discussed more fully in Section 4.4.

For most accidents that are not DNB limited, or for which the revised thermal design procedure is not used, the initial conditions are obtained by adding the maximum steady-state errors to rated values. The following conservative steady-state errors are assumed in the analysis:

Core power	± 1 percent allowance for calorimetric error.
Average reactor coolant system temperature	± 8.0°F allowance for controller deadband and measurement errors
Pressurizer pressure	± 50 psi allowance for steady-state fluctuations and measurement errors

Initial values for core power, average reactor coolant system temperature, and pressurizer pressure are selected to minimize the initial DNBR unless otherwise stated in the sections describing the specific accidents. Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

The plant operating instrumentation selected for feedwater flow measurement is a Caldon [Cameron] LEFM CheckPlus System (Reference 201), which will be calibrated (in a certified laboratory using a piping configuration representative of the plant piping design) prior to installation and will be tested after installation in the plant in accordance with the LEFM CheckPlus commissioning procedure. This selected plant operating instrumentation has documented instrumentation uncertainties to calculate a power calorimetric uncertainty that confirms the 1% uncertainty assumed for the initial reactor power in the safety analysis bounds the calculated calorimetric power uncertainty values. The calculated calorimetric is done in accordance with a previously accepted Westinghouse methodology (Reference 202). Administrative controls implement maintenance and contingency activities related to the power calorimetric instrumentation.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and the total peaking factor (F_q). Unless specifically noted otherwise, the peaking factors used in the accident analyses are those presented in Chapter 4.

For transients that may be DNB limited, the radial peaking factor is important. The radial peaking factor increases with decreasing power level due to control rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.0.3-1. Transients that may be departure from nucleate boiling limited are assumed to begin with an $F_{\Delta H}$, consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is a chopped cosine, as discussed in Section 4.4, for transients analyzed at full power and the most limiting power shape calculated or allowed for accidents initiated at nonfull power or asymmetric RCCA conditions.

The radial and axial power distributions just described are input to the VIPRE-01 code as described in Section 4.4.

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For transients that may be overpower-limited, the total peaking factor (F_q) is important. Transients that may be overpower-limited are assumed to begin with plant conditions, including power distributions, which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant (for example, the chemical and volume control system malfunction that results in a slow decrease in the boron concentration in the reactor coolant system as well as an excessive increase in secondary steam flow) and that may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Section 4.4.

For overpower transients that are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled RCCA bank withdrawal from subcritical or lower power startup and RCCA ejection incident, both of which result in a large power rise over a few seconds), a detailed fuel transient heat transfer calculation is performed.

15.0.4 Reactivity Coefficients Assumed in the Accident Analysis

The transient response of the reactor system is dependent on reactivity feedback effects, in particular, the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients are discussed in Subsection 4.3.2.3.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values; while for other events, the use of small reactivity coefficient values is conservative. The values used are given in Figure 15.0.4-1, which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life, although these combinations may not represent possible realistic situations.

15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs as a function of time and the variation in rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. In analyses where all of the reactor coolant pumps are coasting down prior to, or simultaneous, with RCCA insertion, a time of 2.3 seconds is used for insertion time to dashpot entry.

In Figure 15.0.5-1, the curve labeled “complete loss of flow transients” shows the RCCA position versus time normalized to 2.3 seconds assumed in accident analyses where all reactor coolant pumps are coasting down. In analyses where some or all of the reactor coolant pumps are running, the RCCA insertion time to dashpot is conservatively taken as 2.7 seconds. The RCCA position versus time normalized to 2.7 seconds is also shown in Figure 15.0.5-1.

The use of such a long insertion time provides conservative results for accidents and is intended to apply to all types of RCCAs, which may be used throughout plant life. Drop time testing requirements are specified in the Technical Specifications.

Figure 15.0.5-2 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, which is input to the point kinetics core models used in transient analyses. The bottom-skewed power distribution itself is not an input into the point kinetics core model.

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There is inherent conservatism in the use of Figure 15.0.5-2 in that it is based on a skewed flux distribution, which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significantly more negative reactivity is inserted than that shown in the curve, due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in Figure 15.0.5-3. The curves shown in this figure were obtained from Figures 15.0.5-1 and 15.0.5-2. A total negative reactivity insertion following a trip of 4 percent Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3-3.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0.5-3) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.0.5-1 is used as code input.

15.0.6 Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breaker sets connected in series, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4a. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure 15.0.3-1. As described in Section 7.2 and in Reference 16, the overpower ΔT trip protects the core from exceeding the design overpower limit, and the overtemperature ΔT trip protects the core from exceeding the DNB design limit. The overtemperature ΔT setpoint plus the error allowances bounds the core DNB design limits and includes an upper limit on allowable inlet temperature as shown on Figure 15.0.3-1.

Table 15.0-4a also summarizes the setpoints and the instrumentation delay for engineered safety features (ESF) functions used in accident analyses. Time delays associated with equipment actuated (such as valve stroke times) by ESF functions are summarized in Table 15.0-4b.

The difference between the limiting setpoint assumed for the analysis and the nominal setpoint represents an allowance for instrumentation channel error and setpoint error. Nominal setpoints are maintained by the setpoint program, which is described in the plant Technical Specifications. During plant startup tests, it is demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, the protection system is calibrated and surveillances are performed in accordance with the plant Technical Specifications.

15.0.7 Instrumentation Drift and Calorimetric Errors

The calorimetric uncertainty is the uncertainty assumed in the determination of core thermal power as obtained from secondary plant measurements. On a daily basis, these secondary plant measurements are compared with the ΔT power signal (Reference 16) and with the total ion chamber current (sum of the top and bottom currents) and those signals are adjusted if necessary for acceptable conformance with the calorimetric power measurement.

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The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. Installed plant instrumentation is used for these measurements.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

The plant is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features that minimize the probability and effects of fires and explosions.

Chapter 17 discusses the quality assurance program that is implemented to provide confidence that the plant systems satisfactorily perform their assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, provides confidence that the normally operating systems and components listed in Table 15.0-6 are available for mitigation of the events discussed in Chapter 15.

In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI N18.2-1973 (Reference 1) is used. The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-2000 and Regulatory Guide 1.53 in the application of the single-failure criterion. Conformance to Regulatory Guide 1.53 is summarized in Subsection 1.9.1.

Table 15.0-8 summarizes the nonsafety-related systems assumed in the analyses to mitigate the consequences of events. Except for the cases listed in Table 15.0-8, control system action is not used for mitigation of accidents.

15.0.9 Fission Product Inventories

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the reactor core if the accident involves fuel damage. The radiological consequences analyses use the conservative design basis source terms identified in Appendix 15A.

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the LOCA according to the requirements of 10 CFR 50.46, as described in WCAP-10054-P-A and WCAP-12945-P-A and WCAP-16009-P-A (References 3, 4, and 15). The large-break LOCA methodology considers uncertainty in the decay power level. The small-break LOCA events and post-LOCA long-term cooling analyses use 10 CFR 50, Appendix K, decay heat, which assumes infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used, except that fission product decay energy is based on core average exposure at the end of an equilibrium cycle.

15.0.10.2 Distribution of Decay Heat Following a Loss-of-Coolant Accident

During a LOCA, the core is rapidly shut down by void formation, RCCA insertion, or both, and a large fraction of the heat generation considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady-state fission power. Local peaking effects, which are important for the neutron-dependent part of the heat generation, do not apply to the gamma ray contribution. The steady-state factor, which represents the fraction of heat generated within the cladding and pellet, drops to 95 percent or less for the hot rod in a LOCA.

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For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; one-half second after the rupture, about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect on the hot rod is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total heat. Because the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods; the remaining 2 percent is absorbed by water, thimbles, sleeves, and grids. Combining the 3 percent total heat reduction from gamma redistribution with this 2 percent absorption produce as the net effect a factor of 0.95, which exceeds the actual heat production in the hot rod. The actual hot rod heat generation is computed during the AP1000 large-break LOCA transient as a function of core fluid conditions.

15.0.11 Computer Codes Used

Summaries of some of the principal computer codes used in transient analyses are given as follows. Other codes – in particular, specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (see Subsection 15.6.5) – are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.0-2. WCAP-15644 (Reference 11) provides the basis for use of analysis codes.

15.0.11.1 FACTRAN Computer Code

FACTRAN (Reference 5) calculates the transient temperature distribution in a cross section of a metal-clad UO_2 fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

- A sufficiently large number of radial space increments to handle fast transients
- Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation
- The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials

FACTRAN is further discussed in WCAP-7908-A (Reference 5).

15.0.11.2 LOFTRAN Computer Code

The LOFTRAN (Reference 6) program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides), and pressurizer. The pressurizer heaters, spray, and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The protection and safety monitoring system is simulated to include reactor trips on high neutron flux, power range high positive flux rate, overtemperature ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated, including rod control, steam dump, feedwater control, and pressurizer level and pressure control. The emergency core cooling system, including the accumulators, is also modeled.

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LOFTRAN is a versatile program suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated in Figure 15.0.3-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

The LOFTRAN code is modified to allow the simulation of the passive residual heat removal (PRHR) heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. A discussion of these models and additional validation is presented in WCAP-14234 (Reference 10).

LOFTTR2 (Reference 8) is a modified version of LOFTRAN with a more realistic break flow model, a two-region steam generator secondary side, and an improved capability to simulate operator actions during a steam generator tube rupture (SGTR) event.

The LOFTTR2 code is modified to allow the simulation of the PRHR heat exchanger, core makeup tanks, and associated protection system actuation logic. The modifications are identical to those made to the LOFTRAN code. A discussion of these models is presented in WCAP-14234 (Reference 10).

15.0.11.3 TWINKLE Computer Code

The TWINKLE (Reference 7) program is a multidimensional spatial neutron kinetics code, which is patterned after steady-state codes currently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions, such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided (for example, channelwise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures).

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

15.0.11.4 VIPRE-01 Computer Code

The VIPRE-01 code is described in Subsection 4.4.4.5.2.

15.0.11.5 COAST Computer Program

The COAST computer program is used to calculate the reactor coolant flow coastdown transient for any combination of active and inactive pumps and forward or reverse flow in the hot or cold legs. The program is described in Reference 13 and was referenced in Reference 12. The program was approved in Reference 14.

The equations of conservation of momentum are written for each of the flow paths of the COAST model assuming unsteady one-dimensional flow of an incompressible fluid. The equation of conservation of mass is written for the appropriate nodal points. Pressure losses due to friction, and geometric losses are assumed proportional to the flow velocity squared. Pump dynamics are modeled using a head-flow curve for a pump at full speed and using four-quadrant curves, which are

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parametric diagrams of pump head and torque on coordinates of speed versus flow, for a pump at other than full speed.

15.0.11.6 ANC Computer Code

The ANC computer code is used to solve the two-group neutron diffusion equation in three spatial dimensions. ANC can also solve the three-dimensional kinetics equations for six delayed neutron groups. The ANC code is described in Subsection 4.3.3.3.

15.0.12 Component Failures

15.0.12.1 Active Failures

SECY-77-439 (Reference 9) provides a description of active failures. An active failure results in the inability of a component to perform its intended function.

An active failure is defined differently for different components. For valves, an active failure is the failure of a component to mechanically complete the movement required to perform its function. This includes the failure of a remotely operated valve to change position on demand. The spurious, unintended movement of the valve is also considered as an active failure. Failure of a manual valve to change position under local operator action is included.

Spring-loaded safety or relief valves that are designed for and operate under single-phase fluid conditions are not considered for active failures to close when pressure is reduced below the valve set point. However, when valves designed for single-phase flow are challenged with two-phase flow, such as a steam generator or pressurizer safety valve, the failure to reseat is considered as an active failure.

For other active equipment – such as pumps, fans, and rotating mechanical components – an active failure is the failure of the component to start or to remain operating.

For electrical equipment, the loss of power, such as the loss of offsite power or the loss of a diesel generator, is considered as a single failure. In addition, the failure to generate an actuation signal, either for a single component actuation or for a system-level actuation, is also considered as an active failure.

Spurious actuation of an active component is considered as an active failure for active components in safety-related passive systems. An exception is made for active components if specific design features or operating restrictions are provided that can preclude such failures (such as power lockout, confirmatory open signals, or continuous position alarms).

A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure; the error is limited to manipulation of safety-related equipment and does not include thought-process errors or similar errors that could potentially lead to common cause or multiple errors.

15.0.12.2 Passive Failures

SECY-77-439 also provides a description of passive failures. A passive failure is the structural failure of a static component that limits the effectiveness of the component in carrying out its design function. A passive failure is applied to fluid systems and consists of a breach in the fluid system boundary. Examples include cracking of pipes, sprung flanges, or valve packing leaks.

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Passive failures are not assumed to occur until 24 hours after the start of the event. Consequential effects of a pipe leak – such as flooding, jet impingement, and failure of a valve with a packing leak – must be considered.

Where piping is significantly overdesigned or installed in a system where the pressure and temperature conditions are relatively low, passive leakage is not considered a credible failure mechanism. Line blockage is also not considered as a passive failure mechanism.

15.0.12.3 Limiting Single Failures

The most limiting single active failure (where one exists), as described in Section 3.1, of safety-related equipment, is identified in each analysis description. The consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure that could adversely affect the consequences of the transient is identified. The failure assumed in each analysis is listed in Table 15.0-7.

15.0.13 Operator Actions

There are several events analyzed in the following sections which require operator action to terminate or mitigate the event. The loss of normal feedwater (Subsection 15.2.7), the inadvertent actuation of a core makeup tank (Subsection 15.5.1), and the chemical and volume control system malfunction (Subsection 15.5.2) assume operator action, after the high-2 pressurizer water level setpoint is reached, to open the safety grade reactor vessel head vent. This action prevents filling the pressurizer and allowing water to escape through the pressurizer safety valves. The analysis of the boron dilution for Mode 1 operation with automatic rod control (Subsection 15.4.6) relies on the operator to terminate the dilution source, after the rod insertion limit alarm, before the required shutdown margin is lost. The small line break outside containment event (Subsection 15.6.2) assumes the operator will isolate the break. In all cases where operator actions are credited, no operator actions are required within the first 30 minutes of the transient. For these events, before operator action is required numerous alarms and indications would be available to the operator to diagnose the transient and ensure that the proper action is taken.

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a safe, stable condition. Where a stabilized condition is reached automatically following a reactor trip, it is expected that the operator may, following event recognition, take manual control and proceed with orderly shutdown of the reactor in accordance with the normal, abnormal, or emergency operating procedures. The exact actions taken and the time at which these actions occur depend on what systems are available and the plans for further plant operation.

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition for at least 72 hours. Operator actions typical of normal operation are credited for the inadvertent actuations of equipment in response to a Condition II event.

15.0.14 Loss of Offsite ac Power

As required in GDC 17 of 10 CFR Part 50, Appendix A, anticipated operational occurrences and postulated accidents are analyzed assuming a loss of offsite ac power. The loss of offsite power is not considered as a single failure, and the analysis is performed without changing the event category. In the analyses, the loss of offsite ac power is considered to be a potential consequence of the event.

A loss of offsite ac power will be considered a consequence of an event due to disruption of the grid following a turbine trip during the event. Event analyses that do not result in a possible consequential disruption of offsite ac power do not assume offsite power is lost.

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For those events where offsite ac power is lost, an appropriate time delay between turbine trip and the postulated loss of offsite ac power is assumed in the analyses. A time delay of 3 seconds is used. This time delay is based on the inherent stability of the offsite power grid as discussed in Section 8.2. Following the time delay, the effect of the loss of offsite ac power on plant auxiliary equipment – such as reactor coolant pumps, main feedwater pumps, condenser, startup feedwater pumps, and RCCAs – is considered in the analyses. Turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay is part of the AP1000 reactor trip system.

Design basis LOCA analyses are governed by the GDC-17 requirement to consider the loss of offsite power. For the AP1000 design, in which all the safety-related systems are passive, the availability of offsite power is significant only regarding reactor coolant pump operation for LOCA events. A sensitivity study for AP1000 has shown that for large-break LOCAs, assuming the loss of offsite power coincident with the inception of the LOCA event is nonlimiting relative to assuming continued reactor coolant pump operation until the automatic reactor coolant pump trip occurs following an “S” signal less than 10 seconds into the transient. For small-break LOCA events, the AP1000 automatic reactor coolant pump trip feature prevents continued operation of the reactor coolant pumps from mixing the liquid and vapor present within a two-phase reactor coolant system inventory to increase the liquid break flow and deplete the reactor coolant system mass inventory rapidly. The automatic reactor coolant pump trip occurs early enough during AP1000 small-break LOCA transients that emergency core cooling system performance is not affected by the loss of offsite power assumption because the total break flow is approximately equivalent for reactor coolant pump trip occurring either at time zero or as a result of the “S” signal. Whether a loss of offsite power is postulated at the inception of the LOCA event or occurs automatically later on is unimportant in the Subsection 15.6.5.4C long-term cooling analyses because with either assumption, the reactor coolant pumps are tripped long before the long-term cooling timeframe.

The AP1000 protection and safety monitoring system and passive safeguards systems are not dependent on offsite power or on any backup diesel generators. Following a loss of ac power, the protection and safety monitoring system and passive safeguards are able to perform the safety functions and there are no additional time delays for these functions to be completed.

15.0.15 Combined License Information

15.0.15.1 Following selection of the actual plant operating instrumentation, calculation of the primary power calorimetric uncertainty is addressed in Subsection 15.0.3.2.

15.0.16 References

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Table 15.0-1
Nuclear Steam Supply System Power Ratings

Thermal power output (MWt)	3415
Effective thermal power generated by the reactor coolant pumps (MWt)	15
Core thermal power (MWt)	3400

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Table 15.0-2 (Sheet 1 of 4)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	
15.1	Increase in heat removal from the primary system					
	Feedwater system malfunctions causing a reduction in feedwater temperature	Bounded by excessive increase in secondary steam flow	–	–	–	–
	Feedwater system malfunctions that result in an increase in feedwater flow	LOFTRAN	0.470	–	Upper curve of Figure 15.0.4-1	0 and 3415
	Excessive increase in secondary steam flow	LOFTRAN	0.0 and 0.470	–	Upper and lower curves of Figure 15.0.4-1	3415
	Inadvertent opening of a steam generator relief or safety valve	LOFTRAN, VIPRE-01	Function of moderator density (see Figure 15.1.4-1)	–	See Subsection 15.1.4.	0 (subcritical)
	Steam system piping failure	LOFTRAN, VIPRE-01	Function of moderator density (see Figure 15.1.4-1) for zero power case 0.470 for full power case	–	See Subsection 15.1.5 for zero power case Upper curve of Figure 15.0.4-1 for full power case	0 (subcritical) and 3415
	Inadvertent operation of the PRHR heat exchanger	N/A	N/A	–	N/A	3415

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Table 15.0-2 (Sheet 2 of 4)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	
15.2	Decrease in heat removal by the secondary system					
	Loss of external electrical load and/or turbine trip	LOFTRAN, FACTRAN, VIPRE-01	0.470 and function of moderator density	–	Lower and upper curves of Figure 15.0.4-1	3415 and 3449.15 (a)
	Inadvertent closure of main steam isolation valves	Bounded by turbine trip event	–	–	–	–
	Loss of condenser vacuum and other events resulting in turbine trip	Bounded by turbine trip event	–	–	–	–
	Loss of nonemergency ac power to the plant auxiliaries	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
	Loss of normal feedwater flow	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
	Feedwater system pipe break	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
15.3	Decrease in reactor coolant system flow rate					

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Table 15.0-2 (Sheet 3 of 4)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	
15.3	Partial and complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, COAST, VIPRE-01	0.0 and function of moderator density	–	Lower curve of Figure 15.0.4-1	3415
	Reactor coolant pump shaft seizure (locked rotor) and reactor coolant pump shaft break	LOFTRAN, FACTRAN, COAST, VIPRE-01	0.0 and function of moderator density	–	Lower curve of Figure 15.0.4-1	3415 and 3449.15 (a)
15.4	Reactivity and power distribution anomalies					
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, VIPRE-01	–	0.0	Coefficient is consistent with a Doppler defect of -0.90%Δk	0
	Uncontrolled RCCA bank withdrawal at power	LOFTRAN	0.0 and 0.470	–	Upper and lower curves of Figure 15.0.4-1	10%, 60%, and 100% of 3415
	RCCA misalignment	LOFTRAN, VIPRE-01	NA	–	NA	3415
	Startup of an inactive reactor coolant pump at an incorrect temperature	NA	NA	–	NA	NA
15.4	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	–	NA	0 and 3415
	Inadvertent loading and operation of a fuel assembly in an improper position	ANC	NA	–	NA	3415
	Spectrum of RCCA ejection accidents	ANC, VIPRE	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8

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Table 15.0-2 (Sheet 4 of 4)
Summary of Initial Conditions and Computer Codes Used

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	
15.5	Increase in reactor coolant inventory					
	Inadvertent operation of the core makeup tanks during power operation	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3449.15 (a)
	Chemical and volume control system malfunction that increases reactor coolant inventory	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3449.15 (a)
15.6	Decrease in reactor coolant inventory					
	Inadvertent opening of a pressurizer safety valve and inadvertent operation of ADS	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3415
	Steam generator tube failure	LOFTTR2	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
	A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate containment	NA	NA	–	NA	NA
	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	NOTRUMP WCOBRA/ TRAC	See Subsection 15.6.5 references	–	See Subsection 15.6.5 references	3434.0 (a) (b)

Notes:

a. The Non LOCA analyses assume an initial power of 101% of the NSSS Power (NSSS Power = rated thermal power (RTP) plus 15 MWt for pump heat) and the LOCA analyses assume an initial power of 101% of RTP.

b. Subsection 15.6.5.4A describes the large-break LOCA analysis methodology, which includes treatment of the initial thermal power output uncertainty.

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Table 15.0-3
Nominal Values of Pertinent Plant
Parameters Used in Accident Analyses

	RTDP With 10% Steam Generator Tube Plugging	Without RTDP[a]	
		Without Steam Generator Tube Plugging	With 10% Steam Generator Tube Plugging
Thermal output of NSSS (MWt)	3415	3415	3415
Core inlet temperature (°F)	535.8	535.5	535.0
Vessel average temperature (°F)	573.6	573.6	573.6
Reactor coolant system pressure (psia)	2250.0	2250.0	2250.0
Reactor coolant flow per loop (gpm)	15.08 E+04	14.99 E+04	14.8 E+04
Steam pressure at steam generator outlet (psia)	802.2	814.0	796.0
Assumed feedwater temperature at steam generator inlet (°F)	440.0	440.0	440.0
Average core heat flux (Btu/-hr-ft^2)	1.99 E+05	1.99 E+05	1.99 E+05

Note:

a. Steady-state errors discussed in Subsection 15.0.3 are added to these values to obtain initial conditions for most transient analyses.

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Table 15.0-4a (Sheet 1 of 2)
Protection and Safety Monitoring System
Setpoints and Time Delay Assumed in Accident Analyses

Function	Limiting Setpoint Assumed in Analyses	Time Delays (seconds)
Reactor trip on power range high positive flux rate	15% with 60-second time constant	0.9
Reactor trip on power range high neutron flux, low setting	35%	0.9
Overtemperature ΔT	Variable (see Figure 15.0.3-1)	2.0(1)
Overpower ΔT	Variable (see Figure 15.0.3-1)	1.0(1)
Reactor trip on High-2 pressurizer pressure	2460 psia	2.0
Reactor trip on Low-2 pressurizer pressure	1800 psia	2.0
Reactor trip on Low-2 reactor coolant flow in either hot leg	87% loop flow	1.45
Reactor trip on Low-2 reactor coolant pump speed	90%	0.65
Reactor trip on Low-2 steam generator narrow range water level	0% of span	2.0
High steam generator narrow range level coincident with reactor trip (P-4)	85% of narrow range level span	2.0 (startup feedwater isolation) 2.0 (chemical and volume control system makeup isolation)
High-3 steam generator level	100% of narrow range level span	2.0 (reactor trip) 7.0 (turbine trip, includes 5.0-second timer delay) 2.0 (feedwater isolation, plus valve closure time, see Table 15.0-4b) 2.0 (CVS isolation)
Reactor trip on high-3 pressurizer water level	76% of span	2.0
Reactor trip on “S” signal	–	2.0(1)
PRHR actuation on Low-2 steam generator wide range water level	13.3% of span	2.0
“S” signal, startup feedwater isolation, and steam line isolation on Low-2 T_{cold}	500°F lower bound 510°F upper bound	2.0(1)

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Table 15.0-4a (Sheet 2 of 2)
Protection and Safety Monitoring System
Setpoints and Time Delay Assumed in Accident Analyses

Function	Limiting Setpoint Assumed in Analyses	Time Delays (seconds)
"S" signal, power operated relief valve isolation, and steam line isolation on Low-2 steam line pressure	405 psia (with an adverse environment assumed) 535 psia (without an adverse environment assumed)	2.0
"S" signal on Low-3 pressurizer pressure	1700 psia	2.0
Reactor trip on PRHR discharge valves not closed	Valve not closed	1.25
"S" signal and steamline isolation on high-2 containment pressure	6.7 psig	2.0
Reactor coolant pump trip following "S"	–	5.0 5.3 (LBLOCA)
PRHR actuation on high-3 pressurizer water level	76% of span	2.0 (plus 15.0-second timer delay)
Chemical and volume control system isolation on high-2 pressurizer water level	63.5% of span	2.0
Chemical and volume control system isolation on high-1 pressurizer water level coincident with "S" signal	33% of span	2.0(1)
Boron dilution block on source range flux doubling	3 over 50 minutes	80.0
ADS Stage 1 actuation on core makeup tank Low-3 level signal	67.5% of tank volume	32.0 seconds for control valve to begin to open)
ADS Stage 4 actuation on core makeup tank Low-6 level signal	20% of tank volume	2.0 seconds for squib valve to begin to open)
CMT actuation on pressurizer low-2 water level	0% of span	2.0
PRHR heat exchanger actuation on Low-2 SG narrow range water level coincident with Low-2 startup feedwater flow coincident with T_{cold} -High	0% of span (Low-2 SG narrow range water level) 100 gpm (Low-2 startup feedwater flow)	2.0
DWS isolation on reactor trip	–	2.0
Main control room isolation, air supply initiation, and electrical load de-energization on High-2 control room radioactivity	Iodine: 2.00E-7 Ci/m^3 Particulate: 2.00E-08 Ci/m^3	200.0

Notes:
1. In addition to the response time given, a 4.0 second lag time is used in the accident analysis for this function to account for the associated RTDs. For automatic safeguards actuation, the 4.0 second lag time is only used for the temperature inputs.

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Table 15.0-4b
Limiting Delay Times for
Equipment Assumed in Accident Analyses

Component	Time Delays (seconds)
Feedwater isolation valve closure, feedwater control valve closure, or feedwater pump trip	10 (maximum value for non-LOCA) 5 (maximum value for mass/energy)
Steam line isolation valve closure	5
Core makeup tank discharge valve opening time	15 (maximum) 10 (nominal value for best-estimate LOCA)
Chemical and volume control system isolation valve closure	30
PRHR discharge valve opening time	15 (maximum) 10 (nominal value for best-estimate LOCA) 2 (small-break LOCA signal processing delay time added to the maximum opening time of 15 seconds)
Demineralized water transfer and storage system isolation valve closure time	20
Steam generator power-operated relief valve block valve closure	44
Automatic depressurization system (ADS) valve opening times	See Table 15.6.5-10.

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Table 15.0-5
Not Used

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Table 15.0-6 (Sheet 1 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.1			
Increase in heat removal from the primary system			
Feedwater system malfunctions that result in an increase in feedwater flow	High-2 steam generator level, power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, manual	High-2 steam generators level produced feedwater isolation and turbine trip	Feedwater isolation valve
Excessive increase in secondary steam flow	Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, manual	–	–
Inadvertent opening of a steam generator safety valve	Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, Low-2 pressurizer pressure, “S”, manual	Low-3 pressurizer pressure, Low-2 compensated steam line pressure, Low-2 T_{cold}, low-2 pressurizer level	Core makeup tank, feedwater isolation valves, main steam isolation valves (MSIVs), startup feedwater isolation, accumulators
Steam system piping failure	Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, Low-2 pressurizer pressure, “S”, manual	Low-3 pressurizer pressure, Low-2 compensated steam line pressure, high-2 containment pressure, Low-2 T_{cold}, manual	Core makeup tank, feedwater isolation valves, main steam line isolation valves (MSIVs), accumulators, startup feedwater isolation
Inadvertent operation of the PRHR	Power range high positive flux rate and high neutron flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, “S”, manual PRHR discharge valve position	Low-3 pressurizer pressure, Low-2 T_{cold}, low-2 pressurizer level	Core makeup tank

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Table 15.0-6 (Sheet 2 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.2			
Decrease in heat removal by the secondary system			
Loss of external load/turbine trip	High-2 pressurizer pressure, High-3 pressurizer water level, overtemperature ΔT, overpower ΔT, Low-2 steam generator narrow range water level, Low-2 RCP speed, manual	–	Pressurizer safety valves, steam generator safety valves
Loss of nonemergency ac power to the station auxiliaries	Low-2 steam generator narrow range water level, High-2 pressurizer pressure, High-3 pressurizer level, Low-2 RCP speed, manual	Low-2 steam generator narrow range water level coincident with Low-2 startup water flow, Low-2 steam generator wide range water level	PRHR, steam generator safety valves, pressurizer safety valves
Loss of normal feedwater flow	Low-2 steam generator narrow range water level, High-2 pressurizer pressure, High-3 pressurizer level, manual	Low-2 steam generator narrow range water level coincident with Low-2 startup water flow, Low-2 steam generator wide range water level	PRHR, steam generator safety valves, pressurizer safety valves, reactor vessel head vent
Feedwater system pipe break	Low-2 steam generator narrow range water level, High-2 pressurizer pressure, High-3 pressurizer level, overtemperature ΔT, “S”, manual	Low-2 steam generator narrow range water level coincident with Low-2 startup feedwater flow, Low-2 steam generator wide range water level, Low-2 steam line pressure, high-2 containment pressure	PRHR, core makeup tank, MSIVs, feedline isolation, pressurizer safety valves, steam generator safety valves

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Table 15.0-6 (Sheet 3 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.3			
Decrease in reactor coolant system flow rate			
Partial and complete loss of forced reactor coolant flow	Low-2 flow, Low-2 RCP speed, manual	–	Steam generator safety valves, pressurizer safety valves
Reactor coolant pump shaft seizure (locked rotor)	Low-2 flow, High-2 pressurizer pressure, manual	–	Pressurizer safety valves, steam generator safety valves
Section 15.4			
Reactivity and power distribution anomalies			
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Source range high neutron flux, intermediate range high neutron flux, power range high neutron flux (low setting), power range high neutron flux (high setting), power range high positive flux rate, manual	–	–
Uncontrolled RCCA bank withdrawal at power	Power range high neutron flux, power range high positive flux rate, overtemperature ΔT, over-power ΔT, High-2 pressurizer pressure, High-3 pressurizer water level, manual	–	Pressurizer safety valves, steam generator safety valves
RCCA misalignment	Overtemperature ΔT, Low-2 pressurizer pressure, manual	–	–
Startup of an inactive reactor coolant pump at an incorrect temperature	Power range high neutron flux, power range high positive flux rate, low flow (P-10 interlock), manual	–	–

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Table 15.0-6 (Sheet 4 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.4 (Cont.)			
Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high neutron flux, power range high neutron flux, power range high positive flux rate, overtemperature ΔT, overpower ΔT, manual	Source range flux doubling	CVS to RCS isolation valves, makeup pump suction isolation valves, from the demineralized water transfer and storage system
Spectrum of RCCA ejection accidents	Power range high neutron flux, power range high positive flux rate, manual	–	Pressurizer safety valves
Section 15.5			
Increase in reactor coolant inventory			
Inadvertent operation of the CMT during power operation	High-2 pressurizer pressure, manual, “safeguards” trip, High-3 pressurizer level	High-3 pressurizer level, Low-2 T_{cold}	Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR, steam generator safety valves, reactor vessel head vent
Chemical and volume control system malfunction that increases reactor coolant inventory	High-2 pressurizer pressure, “safeguards” trip, High-3 pressurizer level, manual	High-3 pressurizer level, Low-2 T_{cold}, Low-2 steam line pressure	Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR, reactor vessel head vent
Section 15.6			
Decrease in reactor coolant inventory			
Inadvertent opening of a pressurizer safety valve or ADS path	Low-2 pressurizer pressure, overtemperature ΔT, manual	Low-3 pressurizer pressure	Core makeup tank, accumulator

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Table 15.0-6 (Sheet 5 of 5)
Plant Systems and Equipment
Available for Transient and Accident Conditions

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.6 (Cont.)			
Failure of small lines carrying primary coolant outside containment	–	Manual isolation of the Sample System or CVS discharge lines	Sample System isolation valves, Chemical and volume control system discharge line isolation valves
Steam generator tube rupture	Low-2 pressurizer pressure, overtemperature ΔT, safeguards (“S”), manual	Low-3 pressurizer pressure, High-3 steam generator water level, high steam generator level coincident with reactor trip (P-4), Low-2 steam line pressure, Low-2 pressurizer level	Core makeup tank, PRHR, steam generator safety and/or relief valves, MSIVs, radiation monitors (air removal, steam line, and steam generator blowdown), startup feedwater isolation, chemical and volume control system pump isolation, pressurizer heater isolation, steam generator power-operated relief valve isolation
LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low-2 pressurizer pressure, safeguards (“S”), manual	High-2 containment pressure, Low-3 pressurizer pressure	Core makeup tank, accumulator, ADS, steam generator safety and/or relief valves, PRHR, in-containment water storage tank (IRWST)

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Table 15.0-7 (Sheet 1 of 2)
Single Failures Assumed in Accident Analyses

Event Description	Failure
Feedwater temperature reduction[a]	–
Excessive feedwater flow	One protection division
Excessive steam flow[a]	
Inadvertent secondary depressurization	One core makeup tank discharge valve
Steam system piping failure	One core makeup tank discharge valve (zero power case) One protection division (full power case)
Inadvertent operation of the PRHR	One protection division
Steam pressure regulator malfunction[b]	–
Loss of external load	One protection division
Turbine trip	One protection division
Inadvertent closure of main steam isolation valve	One protection division
Loss of condenser vacuum	One protection division
Loss of ac power	One PRHR discharge valve
Loss of normal feedwater	One PRHR discharge valve
Feedwater system pipe break	One PRHR discharge valve
Partial loss of forced reactor coolant flow	One protection division
Complete loss of forced reactor coolant flow	One protection division
Reactor coolant pump locked rotor	One protection division
Reactor coolant pump shaft break	One protection division
RCCA bank withdrawal from subcritical	One protection division
RCCA bank withdrawal at power	One protection division
Dropped RCCA, dropped RCCA bank	One protection division
Statically misaligned RCCA[c]	–
Single RCCA withdrawal	One protection division

Notes:
a. No protection action required
b. Not applicable to AP1000
c. No transient analysis

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Table 15.0-7 (Sheet 2 of 2)
Single Failures Assumed in Accident Analyses

Event Description	Failure
Flow controller malfunction[b]	–
Uncontrolled boron dilution	One protection division
Improper fuel loading[c]	–
RCCA ejection	One protection division
Inadvertent CMT operation at power	One PRHR discharge valve
Increase in reactor coolant system inventory	One PRHR discharge valve
Inadvertent reactor coolant system depressurization	One protection division
Failure of small lines carrying primary coolant outside containment[c]	–
Steam generator tube rupture	Ruptured steam generator power-operated relief valve fails open
Spectrum of LOCA Small breaks Large breaks	 One ADS Stage 4 valve One CMT valve
Long-term cooling	One ADS Stage 4 valve

Notes:
a. No protection action required
b. Not applicable to AP1000
c. No transient analysis

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Table 15.0-8
Nonsafety-Related System and
Equipment Used for Mitigation of Accidents

Event		Nonsafety-related System and Equipment
15.1.2	Feedwater system malfunctions that result in an increase in feedwater flow	Main feedwater pump trip
15.1.4	Inadvertent opening of a steam generator relief or safety valve	MSIV backup valves[1] Main steam branch isolation valves
15.1.5	Steam system piping failure	MSIV backup valves[1] Main steam branch isolation valves
15.2.7	Loss of normal feedwater	Pressurizer heater block
15.5.1	Inadvertent operation of the core makeup tanks during power operation	Pressurizer heater block
15.5.2	Chemical and volume control system malfunction that increases reactor coolant inventory	Pressurizer heater block
15.6.2	Failure of small lines carrying primary coolant outside containment	Sample line isolation valves
15.6.3	Steam generator tube rupture	Pressurizer heater block MSIV backup valves(1) Main steam branch isolation valves
15.6.5	Small-break LOCA	Pressurizer heater block

Note:
1. These include the turbine stop or control valves, the turbine bypass valves, and the moisture separator reheater 2nd stage steam isolation valves.

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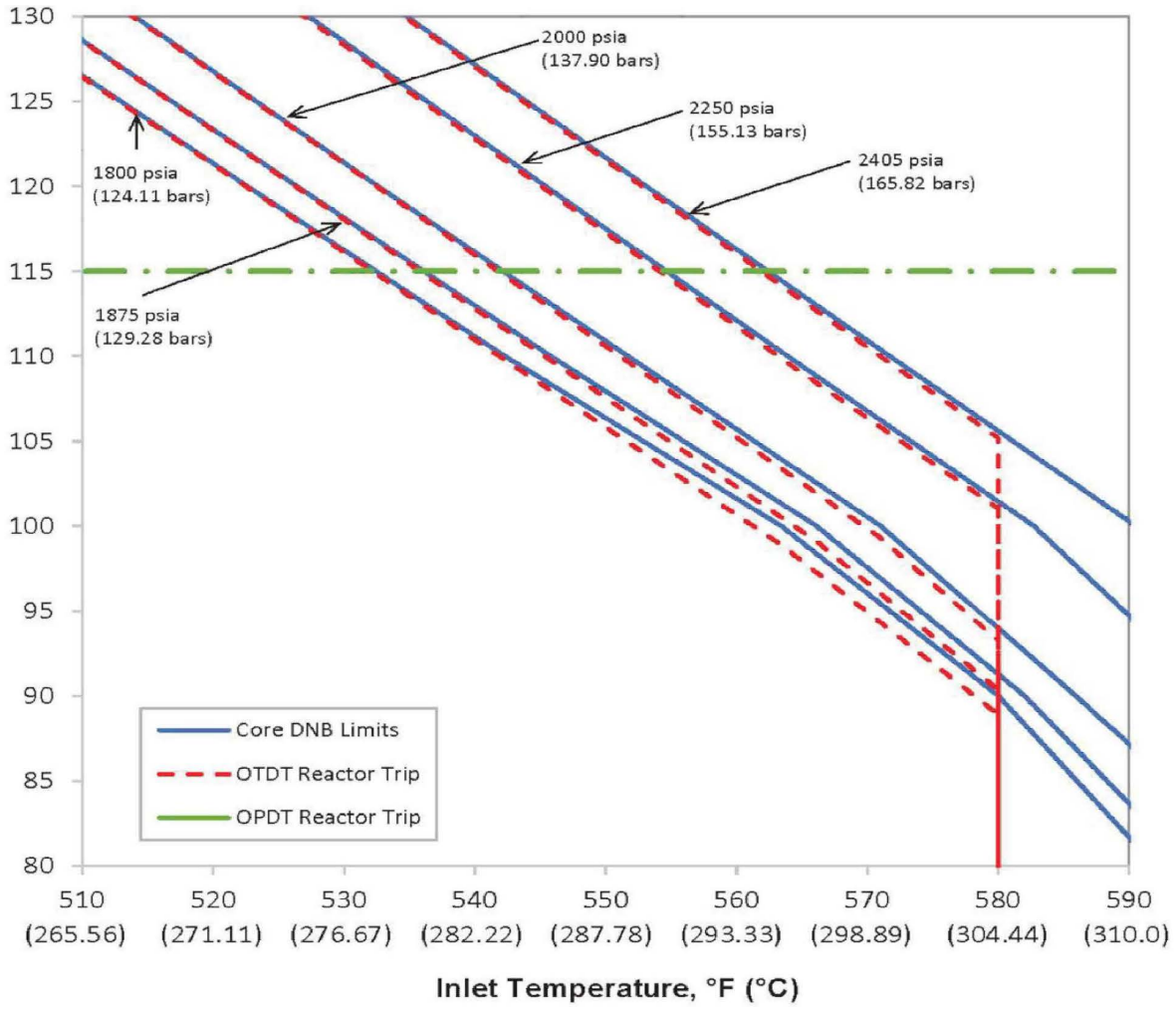

Figure 15.0.3-1
Overpower and Overtemperature ΔT Protection

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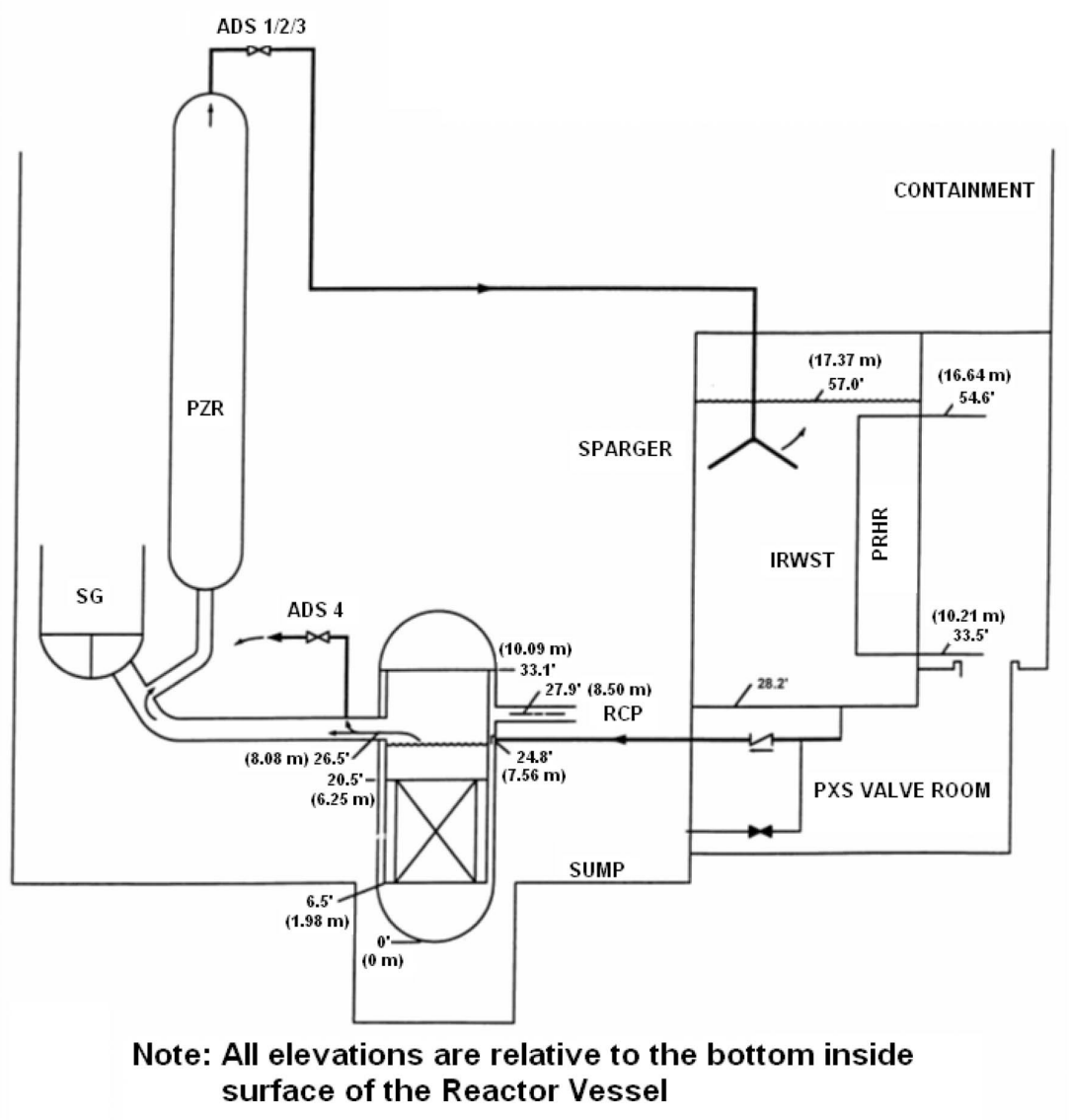

Figure 15.0.3-2
AP1000 Loop Layout

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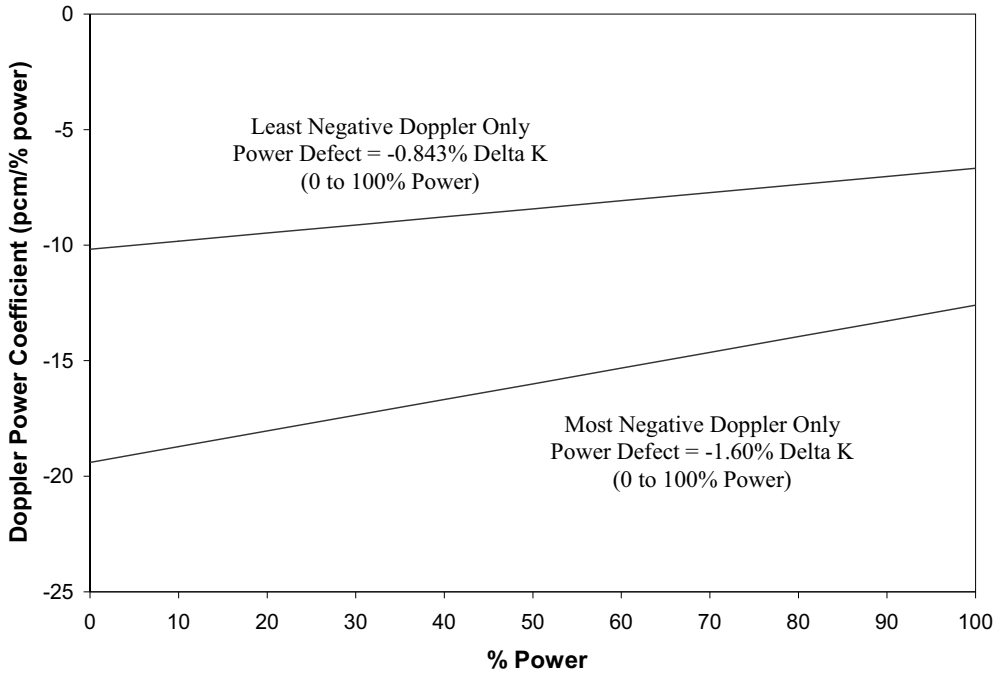

Figure 15.0.4-1
Doppler Power Coefficient used in Accident Analysis

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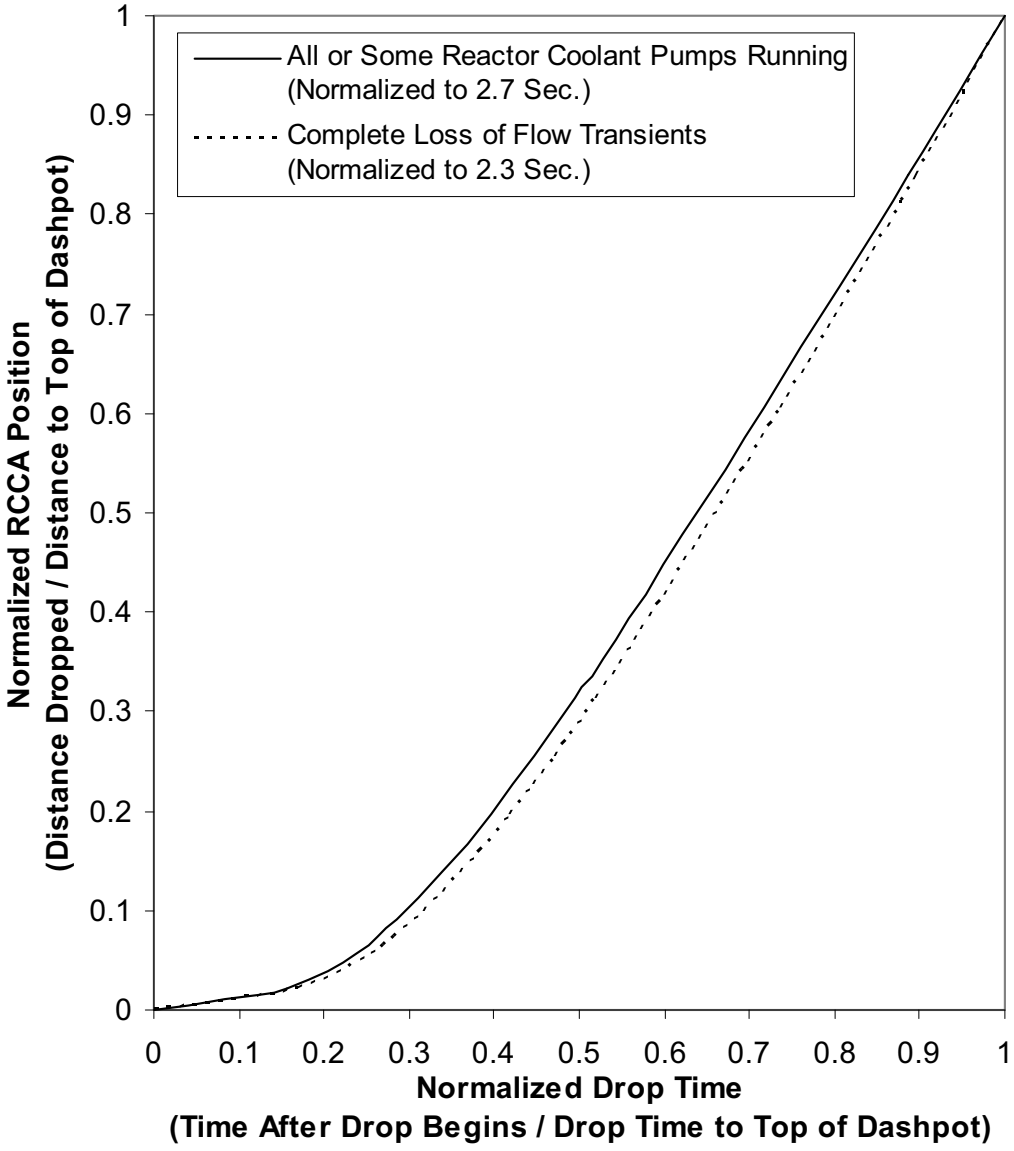

Figure 15.0.5-1
RCCA Position Versus Time to Dashpot

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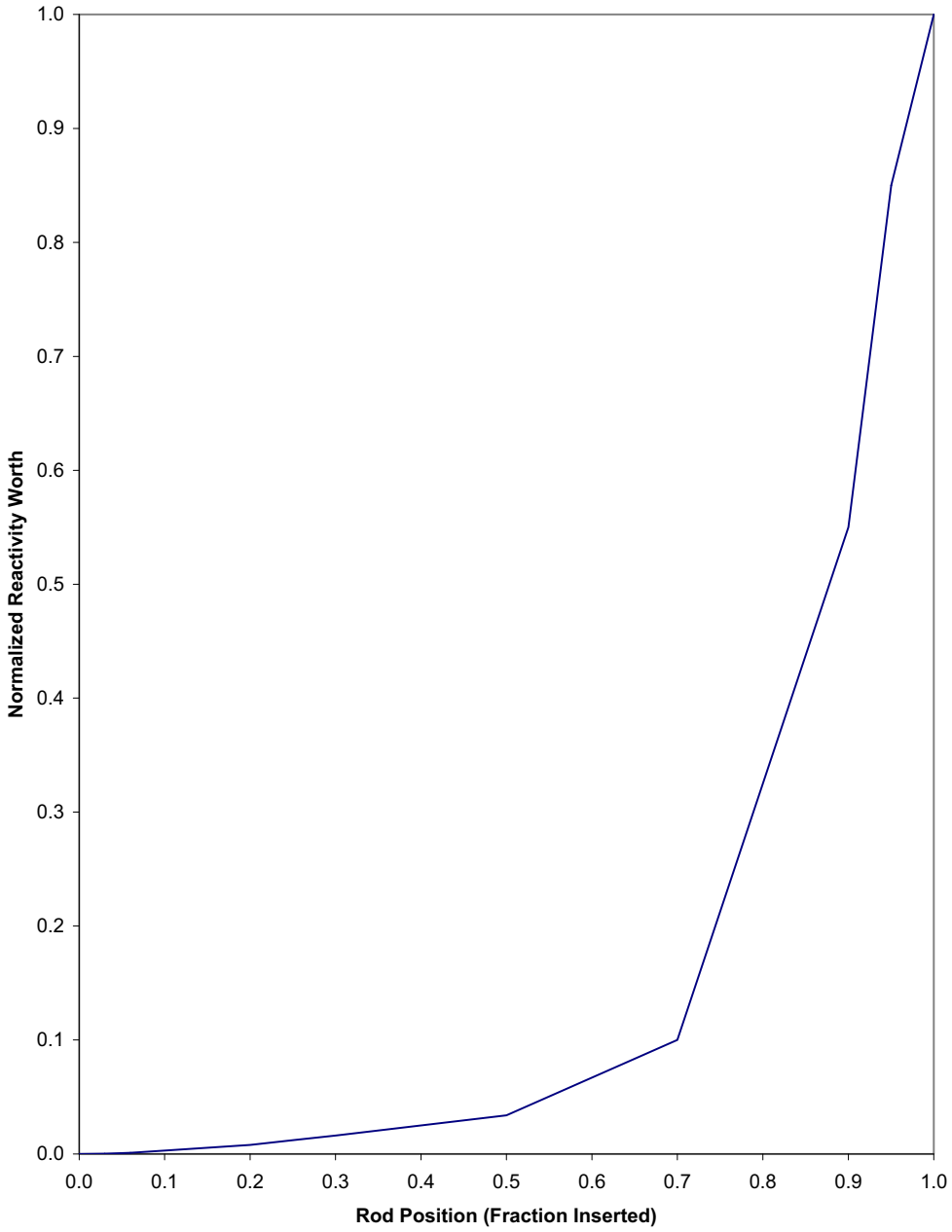

Figure 15.0.5-2
Normalized Rod Worth Versus Position

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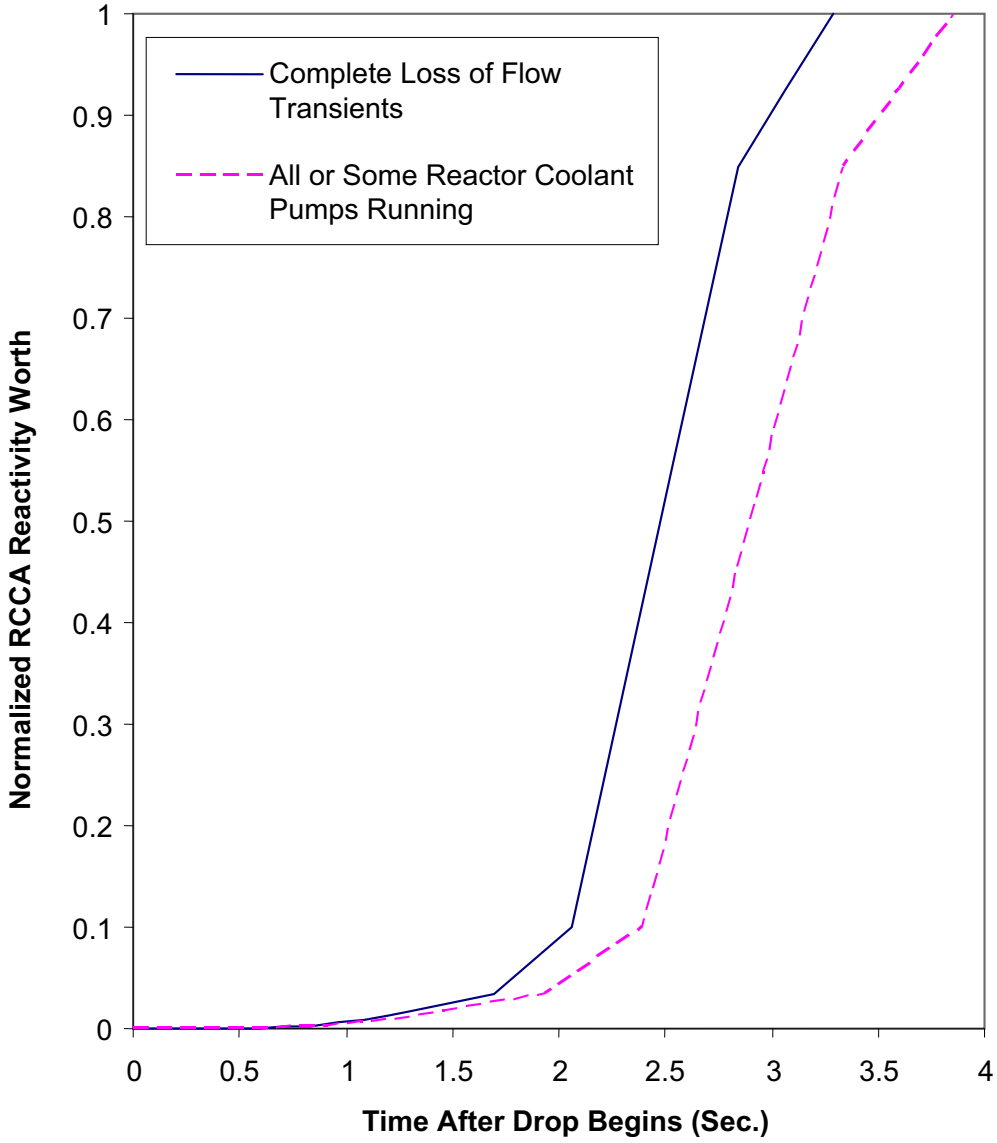

Figure 15.0.5-3
Normalized RCCA Bank
Reactivity Worth Versus Drop Time

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15.1 Increase in Heat Removal From the Primary System

A number of events that could result in an increase in heat removal from the reactor coolant system are postulated. Detailed analyses are presented for the events that have been identified as limiting cases.

Discussions of the following reactor coolant system cooldown events are presented in this section:

- Feedwater system malfunctions causing a reduction in feedwater temperature
- Feedwater system malfunctions causing an increase in feedwater flow
- Excessive increase in secondary steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure
- Inadvertent operation of the passive residual heat removal (PRHR) heat exchanger

The preceding events are Condition II events, with the exception of small steam system piping failures, which are considered to be Condition III, and large steam system piping failure Condition IV events. Subsection 15.0.1 contains a discussion of classifications and applicable criteria.

The accidents in this section are analyzed. The most severe radiological consequences result from the main steam line break accident discussed in Subsection 15.1.5. The radiological consequences are reported only for that limiting case.

15.1.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system. The overpower/overtemperature protection (power range high neutron flux, power range high positive flux rate, overtemperature ΔT, and overpower ΔT trips) prevents a power increase that could lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit value.

A reduction in feedwater temperature may be caused by a low-pressure heater train or a high-pressure heater train out of service or bypassed. At power, this increased subcooling creates an increased load demand on the reactor coolant system.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in reactor coolant system temperature and a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flows decrease, so the no-load transient is less severe than the full-power case. The net effect on the reactor coolant system due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow; that is, the reactor reaches a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A decrease in normal feedwater temperature is classified as a Condition II event, an incident of moderate frequency.

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in Subsection 15.0.8 and listed in Table 15.0-6.

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15.1.1.2 Analysis of Effects and Consequences

15.1.1.2.1 Method of Analysis

This transient is analyzed by calculating conditions at the feedwater pump inlet following the removal of a low-pressure feedwater heater train from service. These feedwater conditions are then used to recalculate a heat balance through the high-pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

- Initial plant power level corresponding to 100-percent nuclear steam supply system thermal output.
- The worst single failure in the pre-heating section of the Main Feedwater System, resulting in the maximum reduction in feedwater temperature, occurs.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

15.1.1.2.2 Results

A fault in the feedwater heaters section of the Feedwater System causes a reduction in feedwater temperature that increases the thermal load on the primary system. The maximum reduction in feedwater enthalpy, due to a single failure in the feedwater system, is 49.98 Btu/lbm. This value is bounded by the enthalpy reduction associated with the Excessive Increase in Secondary Steam Flow event described in Subsection 15.1.3.

15.1.1.3 Conclusions

The decrease in feedwater temperature transient is bounded by the Excessive Increase in Secondary Steam Flow event. Based on the results presented in Subsection 15.1.3, the applicable Standard Review Plan Subsection 15.1.1 evaluation criteria for the decrease in feedwater temperature event are met.

15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

Addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and the reactor coolant system. The overpower/overtemperature protection (power range high neutron flux, power range high positive flux rate, overtemperature ΔT, and overpower ΔT trips) prevents a power increase that could lead to a DNBR less than the safety analysis limit value.

An example of excessive feedwater flow is a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes an increased load demand on the reactor coolant system due to increased subcooling in the steam generator.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in reactor coolant system temperature and a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

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Continuous addition of excessive feedwater is prevented by the steam generator High-3 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps, and reactor.

An increase in normal feedwater flow is classified as a Condition II event, fault of moderate frequency.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power would be assumed to occur as a consequence of the turbine trip for the excessive feedwater flow case initiated from full-power conditions. However, a consequential loss of ac power would not adversely impact the analysis results, and thus has not been explicitly modeled in the analysis. This conclusion is based on a review of the time sequence of events associated with a consequential loss of ac power in comparison to the reactor shutdown time for the event. The primary effect of the loss of ac power is the coastdown of the reactor coolant pumps (RCPs). The protection and safety monitoring system (PMS) includes a 5-second minimum delay between the reactor trip and the turbine trip. In addition, a 3-second delay between the turbine trip and the loss of offsite ac power is assumed. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of ac power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of ac power does not adversely impact this analysis, because the plant will be shut down well before the RCPs begin to coast down. As discussed in Subsection 15.0.14, an excessive feedwater flow transient initiated with the plant at no-load conditions need not consider a consequential loss of offsite power. With the plant initially at zero-load, the turbine would not have been connected to the grid, so any subsequent reactor or turbine trip would not disrupt the grid and produce a consequential loss of offsite ac power.

15.1.2.2 Analysis of Effects and Consequences

15.1.2.2.1 Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the LOFTRAN computer code (Reference 1). LOFTRAN simulates a multiloop system, neutron kinetics, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

The transient is analyzed to demonstrate plant behavior if excessive feedwater addition occurs because of system malfunction or operator error that allows a feedwater control valve to open fully. The following four cases are analyzed assuming a conservatively large negative moderator temperature coefficient:

- Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions.
- Accidental opening of both feedwater control valves with the reactor just critical at zero load conditions.
- Accidental opening of one feedwater control valve with the reactor in manual and automatic rod control at full power.

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- Accidental opening of both feedwater control valves with the reactor in manual and automatic rod control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

- For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 120 percent of nominal feedwater flow to one steam generator.
- For the full power condition, feedwater temperature is conservatively assumed to be reduced by 34°F from the initial value as a result of the increased flow through the feedwater heaters.
- For the feedwater control valve accident at zero-load condition, a feedwater control valve malfunction occurs, which results in a step increase in flow to one steam generator from 0 to 120 percent of the nominal full-load value for one steam generator.
- For the zero-load condition, feedwater temperature remains constant at a conservatively low value of 179°F.
- No credit is taken for the heat capacity of the reactor coolant system and steam generator thick metal in attenuating the resulting plant cooldown.
- The feedwater flow resulting from a fully open control valve is terminated by a steam generator High-3 level trip signal, which closes feedwater control and isolation valves and trips the main feedwater pumps, the turbine, and the reactor.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Normal reactor control systems are not required to function. The protection and safety monitoring system may function to trip the reactor because of overpower or high-3 steam generator water level conditions. No single active failure prevents operation of the protection and safety monitoring system. A discussion of anticipated transients without trip considerations is presented in Section 15.8.

15.1.2.2.2 Results

In the case of an accidental full opening of one or both feedwater control valves with the reactor at zero power and the preceding assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Subsection 15.4.1 for an uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition. Therefore, the results of the analysis are not presented here. If the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25-percent nominal full power, or by the power range high positive flux rate trip.

The full-power case (maximum reactivity feedback coefficients, automatic rod control, multi-loop malfunction) results in the greatest power increase. Assuming the rod control system to be in the manual control mode results in a slightly less severe transient.

When the steam generator water level in the faulted loop reaches the High-3 level setpoint, the feedwater control valves and feedwater isolation valves are automatically closed and the main feedwater pumps are tripped. This prevents continuous addition of the feedwater. In addition, a turbine trip and a reactor trip are initiated.

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Transient results show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor (see Figures 15.1.2-1 and 15.1.2-2). A new equilibrium condition is reached and all the plant parameters, except for the SG water level, remain almost constant. The minimum DNBR is predicted to occur before the reactor trip (see Figure 15.1.2-3). The minimum DNBR predicted remains above the safety analysis DNBR limit value; therefore, the DNBR design basis defined in Section 4.4 is met. Following the reactor trip, the plant approaches a stabilized and safe condition; standard plant shutdown procedures may then be followed to further cool down the plant.

Because the power level increases above its nominal value during the excessive feedwater flow incident, the fuel temperature also increases until after reactor trip occurs. The core heat flux lags behind the neutron flux response because of the fuel rod thermal time constant. However, since the peak core heat value does not exceed 118 percent of its nominal value, the peak fuel temperature remains well below the fuel melting temperature.

The transient results show that departure from nucleate boiling (DNB) is not predicted to occur at any time during the excessive feedwater flow incident. Thus, the capability of the primary coolant to remove heat from the fuel rods is not reduced and the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown in Table 15.1.2-1.

15.1.2.3 Conclusions

The results of the analysis show that the minimum DNBR encountered for an excessive feedwater addition at power is above the design limit value. The DNBR design basis is described in Section 4.4.

Additionally, the reactivity insertion rate that occurs at no-load conditions following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from subcritical condition analysis (see Subsection 15.4.1).

15.1.3 Excessive Increase in Secondary Steam Flow

15.1.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) results in a power mismatch between the reactor core power and the steam generator load demand. The plant control system is designed to accommodate a 10-percent step load increase or a 5-percent-per-minute ramp load increase in the range of 25- to 100-percent full power. Any loading rate in excess of these values may cause a reactor trip actuated by the protection and safety monitoring system. Steam flow increases greater than 10 percent are analyzed in Subsections 15.1.4 and 15.1.5.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, turbine bypass to the condenser is controlled by reactor coolant condition signals. A high reactor coolant temperature indicates a need for turbine bypass. A single controller malfunction does not cause turbine bypass. An interlock blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following protection and safety monitoring system signals:

- Overpower ΔT
- Overtemperature ΔT

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- Power range high positive flux rate
- Power range high neutron flux

The possible consequence of this accident (assuming no protective functions) is a departure from nucleate boiling (DNB) with subsequent fuel damage. Note that the accident is typically characterized by an approach of parameter values to the protection setpoints without the setpoints actually being reached. However, the reactor trip setpoints (power range high positive flux rate, power range high neutron flux, overpower ΔT, and overtemperature ΔT) could be reached during the analysis of the excessive load increase event. These protection functions are defeated in the analysis to preclude reactor trip, ensure the most severe DNB condition is reached, and demonstrate that the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

An excessive load increase incident is considered to be a Condition II event, as described in Subsection 15.0.1.

The requirements of GDC 17 of 10 CFR Part 50, Appendix A, which require determination of the effects produced by a possible consequential loss of offsite power during the excessive load increase event are not applicable. As discussed in Subsection 15.0.14, the loss of offsite power need be considered only as a direct consequence of a turbine trip occurring while the plant is operating at power. For the four excessive load increase cases presented, reactor and turbine trips are not predicted to occur. However, even if a reactor trip were to occur, a consequential loss of ac power would not adversely impact the analysis results. This conclusion is based on a review of the time sequence of events associated with a consequential loss of ac power in comparison to the reactor shutdown time for the event. The primary effect of the loss of ac power is the coastdown of the Reactor Coolant Pumps (RCPs). The Protection & Safety Monitoring System (PMS) includes a five second minimum delay between the reactor trip and the turbine trip. In addition, a three second delay between the turbine trip and the loss of offsite ac power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of ac power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of ac power does not adversely impact this analysis because the plant will be shut down well before the RCPs begin to coast down.

15.1.3.2 Analysis of Effects and Consequences

15.1.3.2.1 Method of Analysis

This accident is primarily analyzed using the LOFTRAN computer code (Reference 1). LOFTRAN simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate plant behavior following a 10-percent step load increase from rated load. These cases are as follows:

- Reactor control in manual with minimum moderator reactivity feedback
- Reactor control in manual with maximum moderator reactivity feedback
- Reactor control in automatic with minimum moderator reactivity feedback
- Reactor control in automatic with maximum moderator reactivity feedback

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity; therefore, reductions in coolant temperature have the least impact on core power. For the maximum moderator feedback cases, the moderator temperature coefficient of

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reactivity has its highest absolute value. This results in the largest amount of reactivity feedback due to changes in coolant temperature. For all the cases analyzed both with and without automatic rod control, no credit is taken for ΔT trips on overtemperature or overpower in order to demonstrate the inherent transient capability of the plant. Under actual operating conditions, such a trip may occur, after which the plant quickly stabilizes.

A 10-percent step increase in steam demand is assumed, and each case is analyzed without credit being taken for pressurizer heaters. At initial reactor power, reactor coolant system pressure and temperature are assumed to be at their full power values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A (Reference 2). Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Normal reactor control systems and engineered safety systems are not required to function.

15.1.3.2.2 Results

Figures 15.1.3-1 through 15.1.3-10 show the transient with the reactor in the manual control mode and no reactor trip signals occur. At the beginning of the minimum moderator feedback case, there is a slight power increase and the average core temperature shows a large decrease. This results in a DNBR that increases above its initial value. At the beginning of the maximum moderator feedback manually controlled case, there is a much faster increase in reactor power due to the moderator feedback. A reduction in the DNBR occurs, but the DNBR remains above the design limit (see Section 4.4).

Figures 15.1.3-11 through 15.1.3-20 show the transient assuming the reactor is in the automatic control mode. At the beginning of the maximum moderator feedback case, the core power increases and the coolant average temperature and pressurizer pressure decrease slowly. For this case, no reactor trip signal is generated. For the minimum moderator feedback case, a reactor trip signal setpoint is reached, but conservatively, reactor trip is not credited. At the beginning of the minimum moderator feedback case, the core power increases but the coolant average temperature and pressurizer pressure decrease rapidly. For this case, the transient oscillate and eventually stabilize. For both these cases, the minimum DNBR remains above the design limit (see Section 4.4).

The excessive load increase incident is an overpower transient for which the fuel temperature rises. Reactor trip is not credited in any of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Because DNB does not occur during the excessive load increase transients, the capability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for the excessive load increase cases with no reactor trip are shown in Table 15.1.2-1.

15.1.3.3 Conclusions

The analysis presented in this subsection demonstrates that for a 10-percent step load increase, the DNBR remains above the design limit. The design basis for DNB is described in Section 4.4. The plant rapidly reaches a stabilized condition following the load increase.

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15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.

The steam release, as a consequence of this accident, results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following Standard Review Plan Subsection 15.1.4 evaluation criterion is satisfied.

Assuming the most reactive stuck RCCA, with offsite power available, and assuming a single failure in the engineered safety features system, there will be no consequential damage to the fuel or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve. This criterion is met by showing the DNB design basis is not exceeded.

Accidental depressurization of the secondary system is classified as a Condition II event as described in Subsection 15.0.1.2.

The following systems provide the necessary protection against an accidental depressurization of the main steam system (see Subsection 7.2.1.1.2):

- Core makeup tank actuation from one of the following signals:
  - Safeguards (“S”) signal from:
    - Two out of four Low-3 pressurizer pressure signals
    - Two out of four high-2 containment pressure signals
    - Two out of four Low-2 T_{cold} signals in any one loop or
    - Two out of four Low-2 steam line pressure signals in any one loop
  - Two out of four low-2 pressurizer level signals

- The overpower reactor trips (power range high positive flux rate, neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the “S” signal

- Redundant isolation of the main feedwater lines

  Sustained high feedwater flow causes additional cooldown. Therefore, in addition to the normal control action that closes the main feedwater control valves following reactor trip, an “S” signal rapidly closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.

- Redundant isolation of the startup feedwater system

  Sustained high startup feedwater flow causes additional cooldown. Therefore, the Low-2 T_{cold} signal closes the startup feedwater control and isolation valves.

- Trip of the fast-acting main steam line isolation valves (assumed to close in less than 10 seconds) on one of the following signals:

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- Two out of four Low-2 steam line pressure signals in any one loop (above permissive P-11)
- Two out of four high negative steam pressure rates in any one loop (below permissive P-11)
- Two out of four Low-2 T_{cold} signals in any one loop, or
- Two out of four high-2 containment pressure signals

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.1.4.2 Analysis of Effects and Consequences

15.1.4.2.1 Method of Analysis

The analysis of a secondary system steam release is performed to determine the following:

- The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown, due to the steam release. The LOFTRAN code (References 1 and 6) is used to model the system transient.

- The thermal-hydraulic behavior of the core due to the steam release. A detailed thermal-hydraulic digital computer code, VIPRE-01 (Reference 7), is used to determine if DNB occurs for the core transient conditions computed by the LOFTRAN code.

The following conditions are assumed to exist at the time of a secondary system steam release:

- End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive RCCA stuck in its fully withdrawn position. Operation of RCCA mechanical shim and axial offset banks during core burnup is restricted by the insertion limits so that shutdown margin requirements are satisfied.

- A most negative moderator temperature coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature is included. The k_{eff} (considering moderator temperature and density effects) versus temperature corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The core power is calculated as a function of core mass flow, core boron concentration, and core inlet temperature.

- Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the passive core cooling system. There are no single failures that prevent core makeup tank injection, however, the analysis models the failure of one core makeup tank discharge valve. Low-concentration boric acid must be swept from the core makeup tank lines downstream of isolation valves before delivery of boric acid (3400 ppm) to the reactor coolant loops. This effect has been accounted for in the analysis.

- The case analyzed models a flow area of 0.2 ft^2, which is based on a steam flow of 520 pounds per second at 1200 psia with offsite power available. This conservatively bounds the maximum capacity of any single steam dump, relief, or safety valve.

- Initial hot shutdown conditions at time zero are assumed because this represents the most conservative initial conditions. Should the reactor be just critical or operating at power at the time of a steam release, the reactor is tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the reactor coolant system contains more stored energy than at no-load. This is because the average coolant

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temperature is higher than at no-load, and there is appreciable energy stored in the fuel. The additional stored energy is removed via the cooldown caused by the steam release before the no-load conditions of the reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy is removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis that assumes no-load condition at time zero. However, because the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the reactor coolant system cooldown are less for a steam line release occurring at power:

- In computing the steam flow, the Moody Curve (Reference 3) for f(L/D) = 0 is used.
- Perfect moisture separation occurs in the steam generator.
- Offsite power is available, because this maximizes the cooldown.
- Maximum cold startup feedwater flow is assumed.
- Four reactor coolant pumps are initially operating.
- Manual actuation of the PRHR system at time zero is conservatively assumed to maximize the cooldown.

15.1.4.2.2 Results

The calculated sequence of events for the analyzed case is shown in Table 15.1.2-1. The results presented conservatively indicate the events that would occur assuming a secondary system steam release because it is postulated that the conditions described in Subsection 15.1.4.2.1 exist simultaneously.

Figures 15.1.4-2 through 15.1.4-12 show the transient results for the event. The steam release accounted for in the analysis is bounding compared to the capacity of any single steam dump, relief, or safety valve.

Core makeup tank injection and the associated tripping of the reactor coolant pumps are initiated automatically by the Low-2 T_{cold} “S” signal. Boron solution at 3400 ppm enters the reactor coolant system, providing enough negative reactivity to prevent a significant return to power and core damage. Later in the transient, as the reactor coolant pressure continues to fall, the accumulators actuate and inject boron solution at 2600 ppm.

The transient is conservative with respect to cooldown, because no credit is taken for the energy stored in the system metal other than that of the fuel elements and steam generator tubes, and the PRHR system is assumed to be actuated at time zero. Because the limiting portion of the transient occurs over a period of about 5 minutes, the neglected stored energy would have a significant effect in slowing the cooldown.

The calculated time sequence of events for this accident is listed in Table 15.1.2-1.

15.1.4.3 Margin to Critical Heat Flux

The analysis demonstrates that the DNB design basis, as described in Section 4.4, is met for the inadvertent opening of a steam generator relief or safety valve. As shown in Figure 15.1.4-2, no significant return to power occurs and, therefore, DNB does not occur. The minimum DNBR is conservatively calculated and is above the 95/95 limit.

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15.1.4.4 Conclusions

The analysis shows that the criterion stated in this subsection is satisfied. For an inadvertent opening of any single steam dump or a steam generator relief or safety valve, the DNB design basis is met.

15.1.5 Steam System Piping Failure

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the existing high-power peaking factors, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the passive core cooling system.

The analysis of a main steam line rupture is performed to demonstrate that the following Standard Review Plan Subsection 15.1.5 evaluation criterion is satisfied.

Assuming the most reactive stuck RCCA with or without offsite power and assuming a single failure in the engineered safety features system, the core cooling capability is maintained. As shown in Subsection 15.1.5.4, radiation doses are within the guidelines.

DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable. The following analysis shows that the DNB design basis is not exceeded for any steamline rupture, assuming the most reactive RCCA is stuck in its fully withdrawn position.

A major steam line rupture is classified as a Condition IV event.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Subsection 15.0.1.3.

The major rupture of a steam line is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat retards the cooldown and thereby reduces the likelihood that the reactor returns to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented here. Certain assumptions used in this analysis are discussed in WCAP-9226-P-A (Reference 4). WCAP-9226-P-A also contains a discussion of the spectrum of break sizes and power levels analyzed.

The steam line rupture at full power conditions is explicitly analyzed and discussed in Subsection 15.1.5.5.

The following functions provide the protection for a steam line rupture (see Subsection 7.2.1.1.2):

- Core makeup tank actuation from one of the following:
  - Safeguards (“S”) signal from:
    - Two out of four Low-3 pressurizer pressure signals
    - Two out of four high-2 containment pressure signals

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    - Two out of four Low-2 T_{cold} signals in one loop, or
    - Two out of four Low-2 steam line pressure signals one loop
  - Two out of four low-2 pressurizer level signals

- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the “S” signal

- Redundant isolation of the main feedwater lines

  Sustained high feedwater flow causes additional cooldown. Therefore, in addition to the normal control action that closes the main feedwater control valves following reactor trip, an “S” signal rapidly closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.

- Redundant isolation of the startup feedwater system

  Sustained high startup feedwater flow causes additional cooldown. Therefore, the Low-2 T_{cold} signal closes the startup feedwater control and isolation valves.

- Trip of the fast-acting main steam line isolation valves (assumed to close in less than 10 seconds) on one of the following signals:

  – Two out of four high-2 containment pressure

  – Two out of four Low-2 steam line pressure signals in any one loop (above permissive P-11)

  – Two out of four high negative steam pressure rates in any one loop (below permissive P-11)

  – Two out of four Low-2 T_{cold} signals in any one loop, or

  – Two out of four high-2 containment pressure signals.

A fast-acting main steam isolation valve is provided in each steam line. These valves are assumed to fully close within 10 seconds of actuation following a large break in the steam line. For breaks downstream of the main steam line isolation valves, closure of the isolation valves will terminate the blowdown. For any break in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the main steam line isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

Flow restrictors are installed in the steam generator outlet nozzle, as an integral part of the steam generator. The effective throat area of the nozzles is 1.4 ft^2, which is considerably less than the main steam pipe area; thus, the flow restrictors serve to limit the maximum steam flow for a break at any location.

Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

15.1.5.2.1 Method of Analysis

The analysis of the steam pipe rupture is performed to determine the following:

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- The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code (References 1 and 6) is used to model the system transient.

- The thermal-hydraulic behavior of the core following a steam line break. A detailed thermal-hydraulic digital computer code, VIPRE-01 (Reference 7), is used to determine if DNB occurs for the core transient conditions computed by the LOFTRAN code.

The following conditions are assumed to exist at the time of a main steam line break accident:

- End-of-cycle shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of RCCA mechanical shim and axial offset banks during core burnup is restricted by the insertion limits so that shutdown margin requirements are satisfied.

- A most negative moderator temperature coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature is included. The k_{eff} (considering moderator temperature and density effects) versus temperature corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The core power is calculated as a function of core mass flow, core boron concentration, and core inlet temperature.

The moderator properties used in the LOFTRAN code for feedback calculations are generated by combining those in the sector nearest the affected steam generator with those associated with the remaining sector. The resultant properties reflect a combination process that accounts for inlet plenum fluid mixing and a conservative weighing of the fluid properties from the coldest core sector.

In verifying the conservatism of this method, the power predictions of the LOFTRAN modeling are confirmed by comparison with detailed core analysis for the limiting conditions of the cases considered. This core analysis conservatively models the hypothetical core configuration (that is, stuck RCCA, non-uniform inlet temperatures, pressure, flow, and boron concentration) and directly evaluates the total reactivity feedback including power, boron, and density redistribution in an integral fashion. The effect of void formation is also included.

Comparison of the results from the detailed core analysis with the LOFTRAN predictions verifies the overall conservatism of the methodology. That is, the specific power, temperature, and flow conditions used to perform the DNB analysis are conservative.

- Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the passive core cooling system. The core makeup tanks and the accumulators are the portions of the passive core cooling system used in mitigating a steam line rupture. There are no single failures that prevent core makeup tank injection however, the analysis models the failure of one core makeup tank discharge valve. Low-concentration boric acid must be swept from the core makeup tank lines downstream of isolation valves before delivery of boric acid (3400 ppm) to the reactor coolant loops. This effect has been accounted for in the analysis.

- The maximum overall fuel-to-coolant heat transfer coefficient is used to maximize the rate of cooldown.

- Because the steam generators are provided with integral flow restrictors with a 1.4-ft^2 throat area, any rupture in a steam line with a break area greater than 1.4 ft^2, regardless of location, has the same effect on the primary plant as the 1.4-ft^2 double-ended rupture. The limiting case considered in determining the core power and reactor coolant system transient is the

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complete severance of a pipe, with the plant initially at no-load conditions and full reactor coolant flow with offsite power available. The results of this case bound the loss of offsite power case for the following reasons:

– Loss of offsite power results in an immediate reactor coolant pump coastdown at the initiation of the transient. This reduces the severity of the reactor coolant system cooldown by reducing primary-to-secondary heat transfer. The lessening of the cooldown, in turn, reduces the magnitude of the return to power.

– Following its actuation, the core makeup tank provides borated water that injects into the reactor coolant system. Flow from the core makeup tank increases if the reactor coolant pumps have coasted down. Therefore, the analysis performed with offsite power and continued reactor coolant pump operation reduces the rate of boron injection into the core and is conservative.

– The protection system automatically provides a safety-related signal that initiates the coastdown of the reactor coolant pumps in parallel with core makeup tank actuation. Because this reactor coolant pump trip function is actuated early during the steam line break event (right after core makeup tank actuation), there is very little difference in the predicted DNBR between cases with and without offsite power.

– Because of the passive nature of the safety injection system, the loss of offsite power does not delay the actuation of the safety injection system.

- Power peaking factors corresponding to one stuck RCCA are determined at the end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck RCCA during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck RCCA. The power peaking factors depend upon the core power, temperature, pressure, and flow and, therefore, may differ for each case studied.

- The analysis assumes initial hot standby conditions at time zero in order to present a representative case which will yield limiting post-trip DNBR results for this transient. If the reactor is just critical or operating at power at the time of a steam line break, the reactor is tripped by the overpower protection system when power level reaches a trip point.

  Following a trip at power, the reactor coolant system contains more stored energy than at no-load because the average coolant temperature is higher than at no-load, and there is energy stored in the fuel. The additional stored energy reduces the cooldown caused by the steam line break before the no-load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis that assumes a no-load condition at time zero. However, because the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the reactor coolant system cooldown are less for a steam line break occurring at power.

- In computing the steam flow during a steam line break, the Moody Curve (Reference 3) for f(L/D) = 0 is used.

- Perfect moisture separation occurs in the steam generator.

- Maximum cold startup feedwater flow plus nominal 100 percent main feedwater flow is assumed.

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- Four reactor coolant pumps are initially operating.
- Manual actuation of the PRHR system at time zero is conservatively assumed in order to maximize the cooldown.

15.1.5.2.2 Results

The calculated sequence of events for the analyzed case is shown in Table 15.1.2-1. The results presented conservatively indicate the events that would occur assuming a steam line rupture because it is postulated that the conditions described in Subsection 15.1.5.2.1 exist simultaneously.

15.1.5.2.3 Core Power and Reactor Coolant System Transient

Figures 15.1.5-1 through 15.1.3-13 show the transient results following a main steam line rupture (complete severance of a pipe) at initial no-load condition.

Offsite power is assumed available so that, initially, full reactor coolant flow exists. During the course of the event, the reactor protection system initiates a trip of the reactor coolant pumps in conjunction with actuation of the core makeup tanks. The transient shown assumes an uncontrolled steam release from only one steam generator. Steam release from more than one steam generator is prevented by automatic trip of the main steam isolation valves in the steam lines by Low-2 steam line pressure signals. Even with the failure of one valve, release is limited to approximately 10 seconds for the other steam generator while the one generator blows down. The main steam isolation valves fully close in less than 10 seconds from receipt of a closure signal.

As shown in Figure 15.1.5-1, the core attains criticality with the RCCAs inserted (with the design shutdown assuming the most reactive RCCA stuck) before boron solution at 3400 ppm (from core makeup tanks) or 2600 ppm (from accumulators) enters the reactor coolant system. A peak core power significantly lower than the nominal full-power value is attained.

The calculation assumes that the boric acid is mixed with and diluted by the water flowing in the reactor coolant system before entering the reactor core. The concentration after mixing depends upon the relative flow rates in the reactor coolant system and from the core makeup tanks or accumulators (or both). The variation of mass flow rate in the reactor coolant system due to water density changes is included in the calculation. The variation of flow rate from the core makeup tanks or accumulators (or both) due to changes in the reactor coolant system pressure and temperature and the pressurizer level is also included. The reactor coolant system and passive injection flow calculations include line losses.

At no time during the analyzed steam line break event does the core makeup tank level approach the setpoint for actuation of the automatic depressurization system. During non-LOCA events, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tank is offset by an equal volume of recirculation flow that enters the core makeup tanks via the reactor coolant system cold leg balance lines.

The PRHR system provides a passive, long-term means of removing the core decay and stored heat by transferring the energy via the PRHR heat exchanger to the in-containment refueling water storage tank (IRWST). The PRHR heat exchanger is normally actuated automatically when the steam generator level falls below the Low-2 steam generator wide range water level. For the main steam line rupture case analyzed, the PRHR exchanger is conservatively actuated at time zero to maximize the cooldown.

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15.1.5.2.4 Margin to Critical Heat Flux

The case analyzed conservatively models the expected behavior of the plant during a steam system piping failure. This includes the tripping of the reactor coolant pumps coincident with core makeup tank actuation. A DNB analysis was performed using limiting assumptions that bound those of Subsection 15.1.5.2.1.

Under the low flow (natural circulation) conditions present in the transient, the return to power is severely limited by the large negative feedback due to flow and power. The minimum DNBR is conservatively calculated and remains above the 95/95 limit.

15.1.5.3 Conclusions

DNB and possible cladding perforation are not unacceptable consequences following a steam pipe rupture based on the applicable acceptance criteria. Nevertheless, the preceding analysis shows that no DNB, and therefore no cladding perforation, occurs for the main steam line rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

15.1.5.4 Radiological Consequences

The evaluation of the radiological consequences of a postulated main steam line break outside containment assumes that the reactor has been operating with a limited number of fuel rods containing cladding defects) and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant. See Subsection 15.1.5.4.1 and Table 15.1.5-1.

Following the rupture, startup feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Any radioiodines carried from the primary coolant into the faulted steam generator via leaking tubes are assumed to be released directly to the environment. It is conservatively assumed that the reactor is cooled by steaming from the intact loop.

15.1.5.4.1 Source Term

The only significant radionuclide releases due to the main steam line break are the iodines and alkali metals that become airborne and are released to the environment as a result of the accident. Noble gases are also released to the environment. Their impact is secondary because any noble gases entering the secondary side during normal operation are rapidly released to the environment.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs prior to the accident and that the maximum resulting reactor coolant iodine concentration exists at the time the accident occurs.

The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The reactor coolant alkali metal concentrations are based on those associated with the design basis fuel defect level.

The secondary coolant is assumed to have an iodine source term of 0.01 μCi/g dose equivalent I-131. This is 1 percent of the maximum primary coolant activity at equilibrium operating conditions. The secondary coolant alkali metal concentration is also assumed to be 1 percent of the primary concentration.

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15.1.5.4.2 Release Pathways

There are three components to the accident releases:

- The secondary coolant in the steam generator of the faulted loop is assumed to be released out the break as steam. Any iodine and alkali metal activity contained in the coolant is assumed to be released.
- The reactor coolant leaking into the steam generator of the faulted loop is assumed to be released to the environment without any credit for partitioning or plateout onto the interior of the steam generator.
- The reactor coolant leaking into the steam generator of the intact loop would mix with the secondary coolant and thus raise the activity concentrations in the secondary water. While the steam release from the intact loop would have partitioning of non-gaseous activity, this analysis conservatively assumes that any activity entering the secondary side is released.

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.1.5.4.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.1.5.4.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.1.5-1.

15.1.5.4.5 Identification of Conservatisms

The assumptions and parameters used in the analysis contain a number of significant conservatisms:

- The reactor coolant activities are based on conservative assumptions (see Table 15.1.5-1). The activities based on the expected fuel defect level are far less than this (see Section 11.1).
- The assumed leakage of 150 gallons of reactor coolant per day into each steam generator is conservative. The leakage is expected to be a small fraction of this during normal operation.
- The conservatively selected meteorological conditions are present only rarely.

15.1.5.4.6 Doses

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than 0.6 rem at the site boundary for the limiting 2-hour interval (4.8 to 6.8 hours) and 1.1 rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A “small fraction” is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be less than 0.5 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.4 rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

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At the time the main steam line break occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. The 30-day contribution to the dose at the site boundary and low population zone boundary is less than 0.01 rem TEDE. When this is added to the dose calculated for the main steam line break, the resulting total dose remains less than the values reported above.

15.1.5.5 Steam System Piping Failure at Full Power

15.1.5.5.1 Identification of Causes and Accident Description

A rupture in the main steam system piping from an at-power condition creates an increased steam load, which extracts an increased amount of heat from the reactor coolant system via the steam generators. This results in a reduction in reactor coolant system temperature and pressure. In the presence of a strong negative moderator temperature coefficient, typical of end-of-life conditions, the colder core inlet coolant temperature causes the core power to increase from its initial level due to the positive reactivity insertion. The power approaches a level equal to the total steam flow.

Depending upon the break size, the reactor may be tripped on any of the following trip signals to provide the necessary protection against the rupture of a main steam line.

- Overpower ΔT
- Low-2 pressurizer pressure
- Safeguards (“S”) actuation signal
  - Low-2 steam line pressure
  - Low-2 cold leg temperature

The steam system piping failure accident analysis described in Subsection 15.1.5 is performed assuming a hot zero power initial condition with the control rods inserted in the core, except for the most reactive rod in the fully withdrawn position, out of the core. That condition could occur while the reactor is at hot shutdown at the minimum required shutdown margin or after the plant has been tripped manually or by the reactor protection system following a steam line break from an at-power condition. For an at-power break, the analysis of Subsection 15.1.5 represents the limiting condition with respect to core protection for the time period following reactor trip. The purpose of this section is to describe the analysis of a steam system piping failure occurring from an at-power initial condition, to demonstrate that core protection is maintained prior to and immediately following reactor trip. The analysis initiated from hot full power does not extend into the portion of the transient where the PRHR or CMTs are actuated.

Depending on the size of the break, this event is classified as either an ANS Condition III or IV event.

15.1.5.5.2 Analysis of Effects and Consequences

15.1.5.5.2.1 Method of Analysis

The analysis of the steam line rupture is performed in the following stages:

1. The LOFTRAN code (References 1 and 6) is used to calculate the nuclear power, core heat flux, and reactor coolant system temperature and pressure transients resulting from the cooldown following the steam line break.

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2. The core radial and axial peaking factors are determined using the thermal hydraulic conditions from LOFTRAN as input to the nuclear core models. A detailed thermal-hydraulic code, VIPRE-01 (Reference 7), is then used to calculate the DNBR for the limiting time during the transient.

This accident is analyzed with the Revised Thermal Design Procedure (RTDP) as described in WCAP-11397-P-A (Reference 2).

The following assumptions are made in the transient analysis:

1. Initial Conditions — RTDP DNB methodology was used, therefore the uncertainties in the initial conditions are included in the DNBR limits; thus, nominal full power values are used in LOFTRAN. The RCS Minimum Measured Flow is used.

2. Break Size — A spectrum of break sizes was analyzed. Small breaks do not result in a reactor trip. Intermediate breaks result in a reactor trip on overpower ΔT. Larger break sizes result in a reactor trip on Low-2 steam line pressure safeguards actuation.

3. Break flow — In computing the steam flow during a steam line break, the Moody curve (Reference 3) for fL/D = 0 is used.

4. Reactivity Coefficients — The analysis assumes maximum moderator reactivity feedback and minimum Doppler power feedback to maximize the power increase following the break.

5. Protection System — The protection system features that mitigate the effects of a steam line beak are described in Subsection 15.1.5. This analysis only considers the initial phase of the transient initiated from an at-power condition. Protection in this phase of the transient is provided by reactor trip, if necessary (specifically overpower ΔT, and Low-2 steam line pressure safeguards actuation).

6. Control Systems — Control systems are not credited in the accident analysis unless their function would result in more severe consequences. The only control system that is assumed to function during the hot full power steam line break event is the main feedwater system. For this event, the feedwater flow is assumed to match the steam flow.

As required in GDC 17 of 10 CFR Part 50, Appendix A, anticipated operational occurrences and postulated accidents are analyzed assuming a loss of offsite ac power. The loss of offsite power is not considered as a single failure, and the analysis is performed without changing the event category. In the analyses, the loss of offsite ac power is considered to be a potential consequence of an event due to disruption of the grid following a turbine trip during the event.

For those events where offsite ac power is lost, an appropriate time delay between turbine trip and the postulated loss of offsite ac power is assumed in the analyses. A time delay of 3 seconds is used. This time delay is based on the inherent stability of the offsite power grid. Following the time delay, the effect of the loss of offsite ac power on plant auxiliary equipment — such as reactor coolant pumps, main feedwater pumps, condenser, startup feedwater pumps, and RCCAs — is considered in the analyses. Turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay is part of the reactor trip system and was chosen to allow the reactor to be tripped and have the rods inserted to the bottom of the core before a turbine trip signal. As a result, RCP coastdown would be delayed an additional 5 seconds, the control rods would be fully inserted and there would be no adverse DNB impact from the resulting core flow reduction. Thus, there is no need for an explicit analysis of this event with loss of offsite ac power.

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15.1.5.5.3 Results

A spectrum of steam line break sizes was analyzed from 0.1 ft^2 to 1.4 ft^2. The results show that for small break sizes up to and including 0.35 ft^2, a reactor trip is not generated. In this case, the event is similar to an excessive load increase event; the core reaches a new equilibrium condition at a higher power equivalent to the increased steam release. For break sizes from 0.36 ft^2 up to and including 0.87 ft^2, the reactor trips on overpower ΔT. For break sizes from 0.88 ft^2 to 1.4 ft^2 the reactor trips on the Low-2 steam line pressure safeguards actuation signal.

The limiting case for demonstrating DNB and kW/ft protection is the 0.87 ft^2 break, the largest break size that results in a trip on overpower ΔT. The time sequence of events for this case is shown on Table 15.1.5.5-1. Figures 15.1.5.5-1 through 15.1.5.5-7 show the transient response.

15.1.5.5.4 Conclusions

The analysis shows that the DNB and fuel centerline melt (kW/ft) design bases are met for the limiting case. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily precluded by the criteria, the above analysis, in fact, shows that the minimum DNBR remains above the limit value for any rupture occurring from an at-power condition prior to and immediately following a reactor trip.

15.1.6 Inadvertent Operation of the PRHR Heat Exchanger

15.1.6.1 Identification of Causes and Accident Description

The inadvertent actuation of the PRHR heat exchanger causes an injection of relatively cold water into the reactor coolant system. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. To prevent this reactivity increase from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve is not closed.

The inadvertent actuation of the PRHR heat exchanger could be caused by operator error or a false actuation signal, or by malfunction of a discharge valve. Actuation of the PRHR heat exchanger involves opening one of the isolation valves, which establishes a flow path from one reactor coolant system hot leg, through the PRHR heat exchanger, and back into its associated steam generator cold leg plenum.

The PRHR heat exchanger is located above the core to promote natural circulation flow when the reactor coolant pumps are not operating. With the reactor coolant pumps in operation, flow through the PRHR heat exchanger is enhanced. The heat sink for the PRHR heat exchanger is provided by the IRWST, in which the PRHR heat exchanger is submerged. Because the fluid in the heat exchanger is in thermal equilibrium with water in the tank, the initial flow out of the PRHR heat exchanger is significantly colder than the reactor coolant system fluid. Following this initial insurge, the reduction in cold leg temperature is limited by the cooling capability of the PRHR heat exchanger. Because the PRHR heat exchanger is connected to only one reactor coolant system loop, the cooldown resulting from its actuation is asymmetric with respect to the core.

The response of the plant to an inadvertent PRHR heat exchanger actuation with the plant at no-load conditions is bounded by the analyses performed for the inadvertent opening of a steam generator relief or safety valve event (Subsection 15.1.4) and the steam system piping failure event (Subsection 15.1.5). Both of these events are conservatively analyzed assuming PRHR heat exchanger actuation coincident with the steam line depressurization. Therefore, only the response of the plant to an inadvertent PRHR initiation with the core at power is considered.

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In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of ac power during an inadvertent PRHR heat exchanger actuation event have been evaluated to not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of ac power in comparison to the reactor shutdown time for an inadvertent PRHR heat exchanger actuation event. The primary effect of the loss of ac power is to cause the Reactor Coolant Pumps (RCPs) to coast down. The PMS system includes a 5-second minimum delay between the reactor trip and the turbine trip. In addition, a 3-second delay between the turbine trip and the loss of offsite ac power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of ac power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of ac power does not adversely impact this inadvertent PRHR heat exchanger actuation analysis because the plant will be shut down well before the RCPs begin to coast down.

The inadvertent actuation of the PRHR heat exchanger event is a Condition II event, a fault of moderate frequency. Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. The following reactor protection system functions are available to provide protection in the event of an inadvertent PRHR heat exchanger actuation:

- PRHR discharge valve not closed
- Overpower/overtemperature reactor trips (neutron flux and ΔT)
- Two out of four Low-3 pressurizer pressure signals

  Due to the potential consequences as a result of the reactivity excursion, a reactor trip has been designed so that upon an inadvertent PRHR actuation, a reactor trip will occur. This reactor trip is generated when either of the discharge valves is not closed. This ensures that the reactor will be tripped prior to a power increase due to the cold water injection.

15.1.6.2 Analysis of Effects and Consequences

Since a reactor trip is initiated as soon as the PRHR discharge valves are not fully closed, this event is essentially a reactor trip from the initial condition and requires no separate transient analysis. Table 15.1.2-1 shows the sequence of events for the inadvertent PRHR heat exchanger actuation.

15.1.6.3 Conclusions

Inadvertent actuation of the PRHR does not result in violation of the core thermal design limits (DNB and linear power generation) or RCS overpressure.

15.1.7 Combined License Information

This section contained no requirement for additional information.

15.1.8 References

1. Burnett, T. W. T., et al., “LOFTRAN Code Description,” WCAP-7907-P-A (Proprietary), and WCAP-7907-A (Nonproprietary), April 1984.

2. Friedland, A. J., and Ray, S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

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3. Moody, F. S., “Transactions of the ASME, Journal of Heat Transfer,” Figure 3, page 134, February 1965.

4. Wood, D. C. and Hollingsworth, S. D., “Reactor Core Response to Excessive Secondary Steam Releases,” WCAP-9226-P-A, Revision 1 (Proprietary) and WCAP-9227 Revision 1 (Nonproprietary), Approved February 1998.

5. Not used.

6. “AP1000 Code Applicability Report,” WCAP-15644-P Revision 2 (Proprietary) and WCAP-15644-NP, (Nonproprietary), March 2004.

7. Sung, Y. X., Schueren, P., and Meliksetian, A., “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), October 1999.

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Table 15.1.2-1 (Sheet 1 of 2)
Time Sequence of Events for Incidents That
Result in an Increase in Heat Removal From the Primary System

Accident	Event	Time (seconds)
Excessive increase in secondary steam flow		
– Manual reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	200.0
– Manual reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	170.0
– Automatic reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	400.0*
– Automatic reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	70.0
Feedwater system malfunctions that result in an increase in feedwater flow (full power)	Both main feedwater control valves fail fully open	0.0
	Minimum DNBR occurs	139.4
	Turbine trip/feedwater isolation and reactor trip on High-3 steam generator level setpoint reached	243.8
	Rod motion begins	245.8
	Turbine trip occurs	250.8
	Feedwater isolation occurs	255.8
Inadvertent operation of the PRHR	PRHR discharge valves go fully open	0.0
	Reactor trip setpoint reached	0.0
	Rod motion begins	1.25
	Rods fully inserted	3.95

* Although oscillation in the transients occurs, the nuclear power and DNBR stabilize after 400 seconds.

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Table 15.1.2-1 (Sheet 2 of 2)
Time Sequence of Events for Incidents That
Result in an Increase in Heat Removal From the Primary System

Accident	Event	Time (seconds)
Inadvertent opening of a steam generator relief or safety valve	Inadvertent opening of one main steam safety or relief valve	0.0
	"S" actuation signal on safeguards Low-2 T_{cold}	119.0
	Core makeup tank actuation	136.0
	Boron reaches core	156.2
Steam system piping failure	Steam line ruptures	0.0
	"S" actuation signal on safeguards Low-2 steam line pressure	1.4
	Boron reaches core	30.6
	Pressurizer and surge line empty	50.4
	Criticality attained	55.2

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Table 15.1.5-1
Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break

Reactor coolant iodine activity	
– Accident-initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Appendix 15A). Duration of spike is 5 hours.
– Pre-accident spike	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Based on design basis activity (see Table 15A-9)
Secondary coolant initial iodine and alkali metal activity	1% of reactor coolant concentrations at maximum equilibrium conditions
Duration of accident (hr)	72
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Steam generator in faulted loop	
– Initial water mass (lb)	3.32 E+05
– Primary to secondary leak rate (lb/hr)	52.25[(a)]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.321 E+05 3.66 E+03
Steam generator in intact loop	
– Primary to secondary leak rate (lb/hr)	52.25[(a)]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.321 E+05 3.66 E+03
Nuclide data	See Table 15A-4

Note:
a. Equivalent to 150 gpd cooled liquid at 62.4 lb/ft^3.

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Table 15.1.5.5-1
Time Sequence of Events for Steam System Piping Failure at Full Power — 0.87ft^2 Break Size

Event	Time (seconds)
Steam line rupture	0.0
OPΔT reactor trip setpoint reached	12.9
Rods begin to drop	13.9
Minimum DNBR occurs	14.9
Maximum core heat flux occurs	14.9

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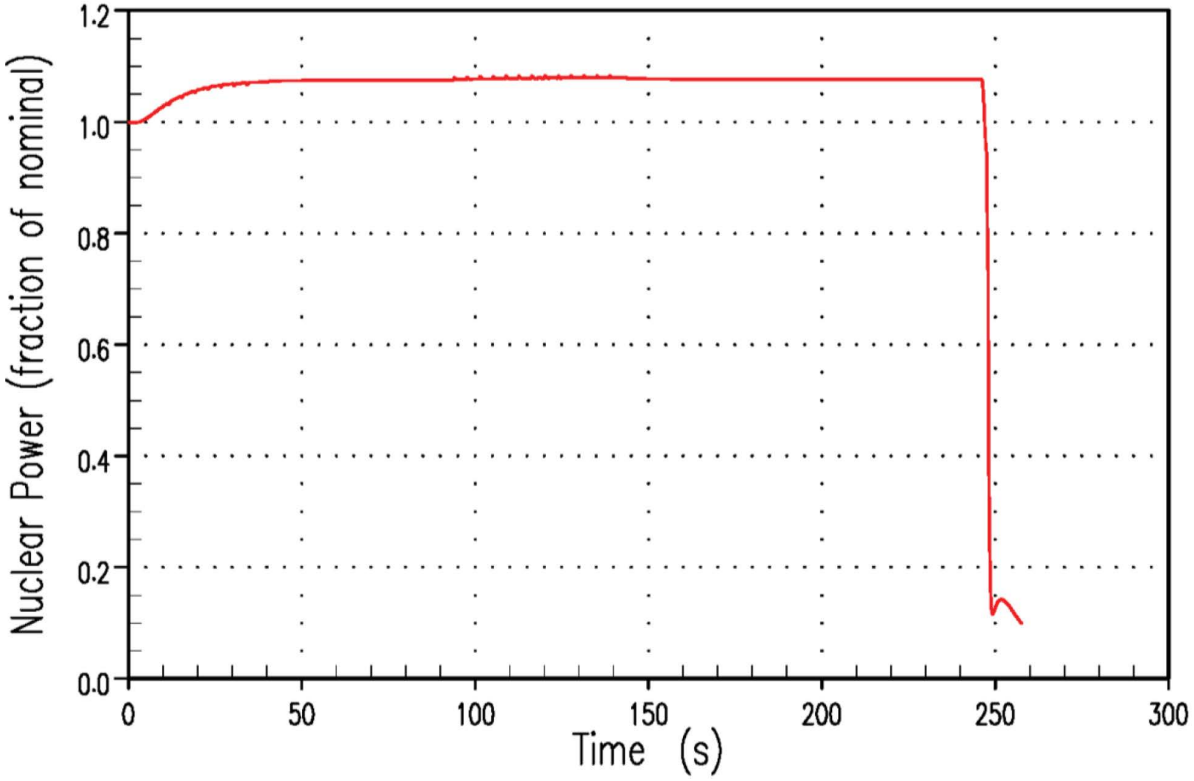

Figure 15.1.2-1
Feedwater Control Valve Malfunction Nuclear Power

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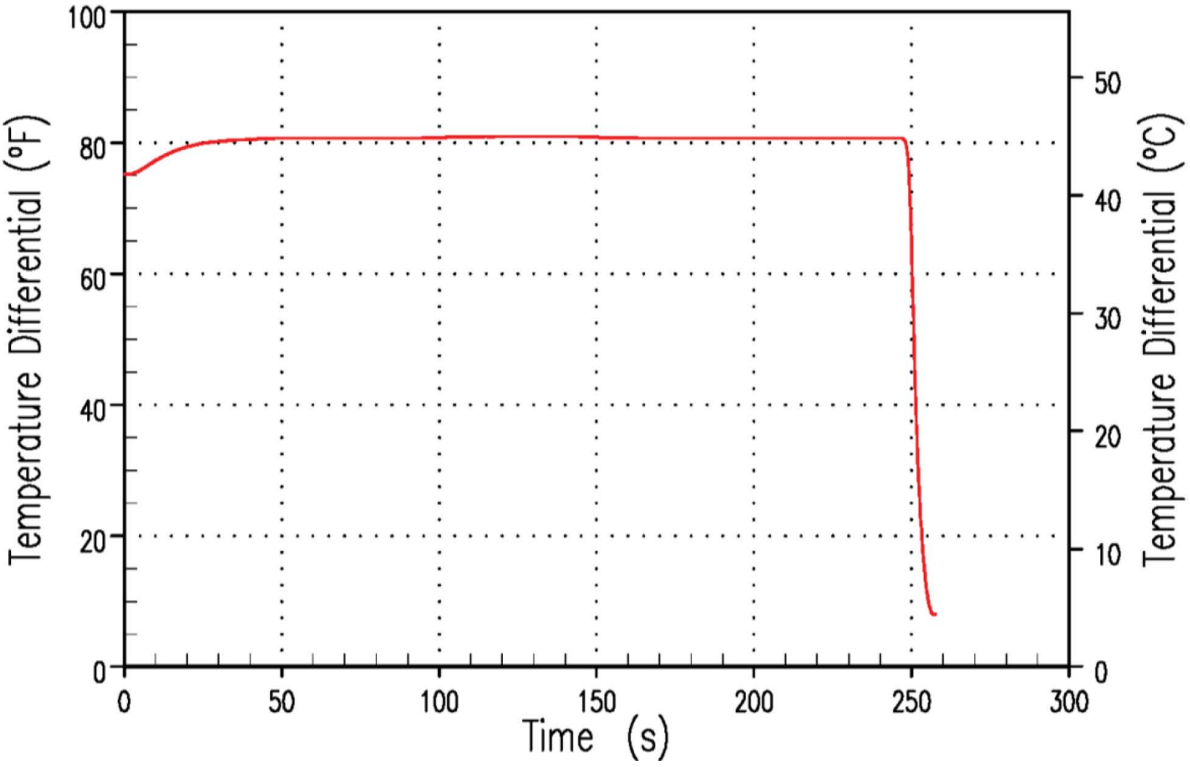

Figure 15.1.2-2
Feedwater Control Valve Malfunction Loop ΔT

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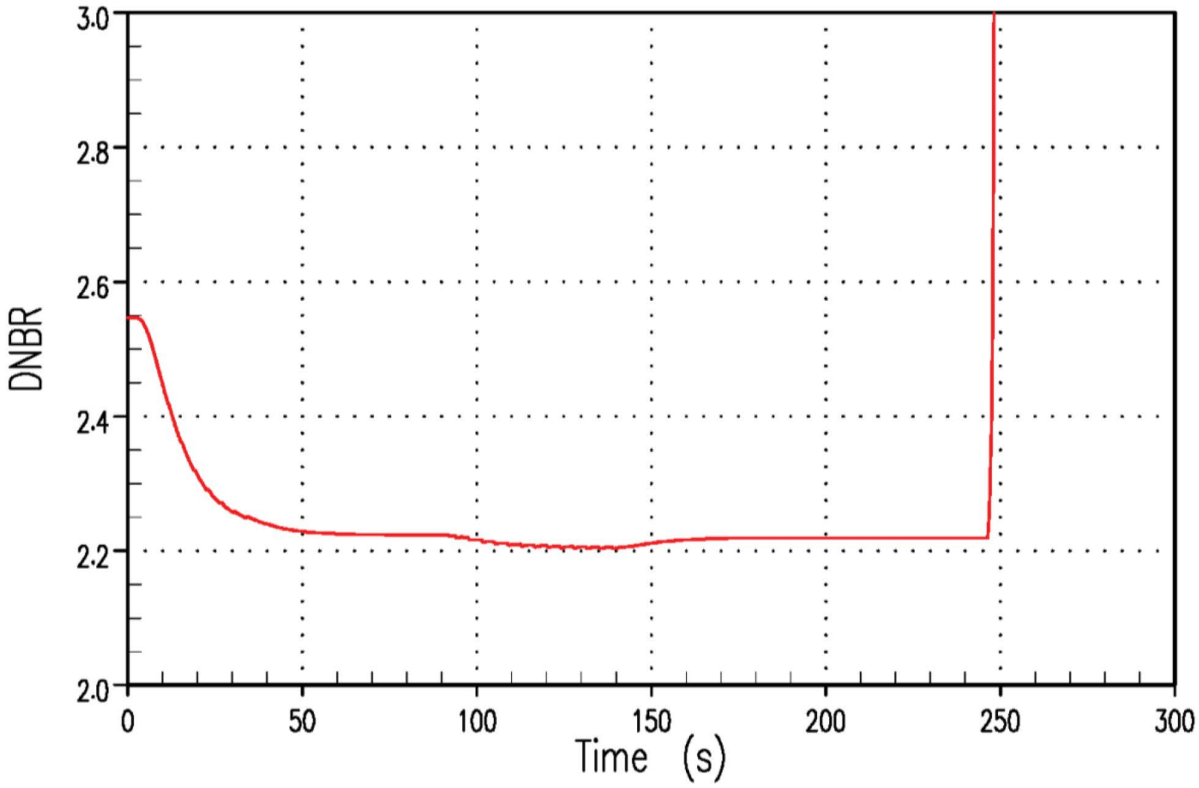

Figure 15.1.2-3
Feedwater Control Valve Malfunction DNBR

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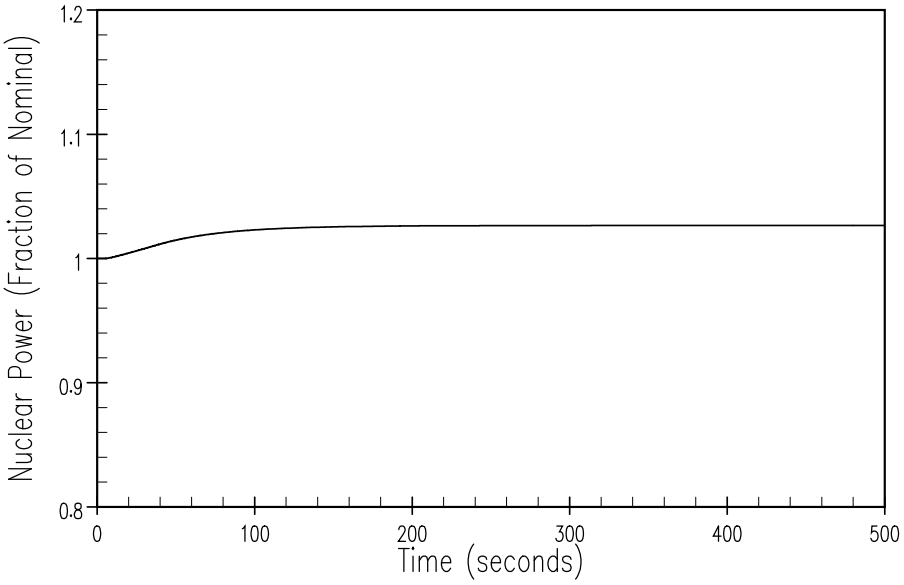

Figure 15.1.3-1
Nuclear Power Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

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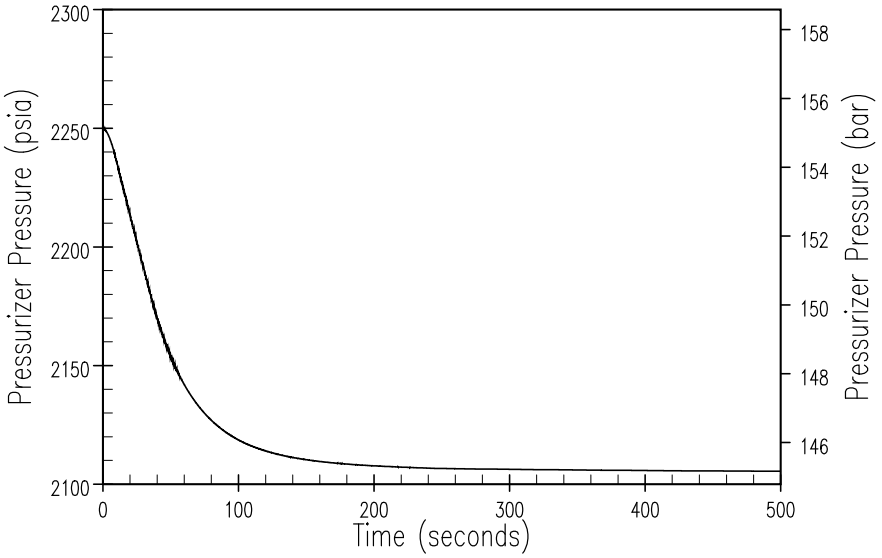

Figure 15.1.3-2
Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

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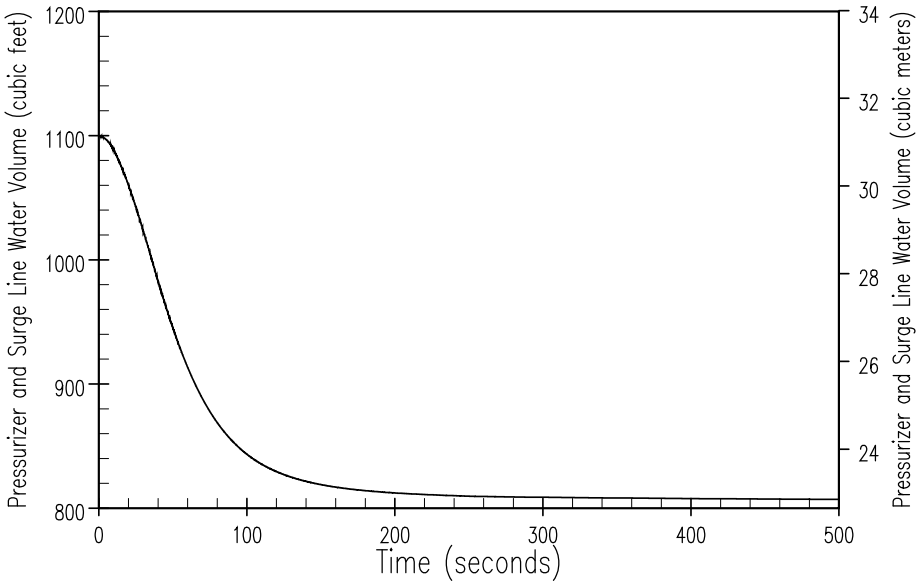

Figure 15.1.3-3
Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

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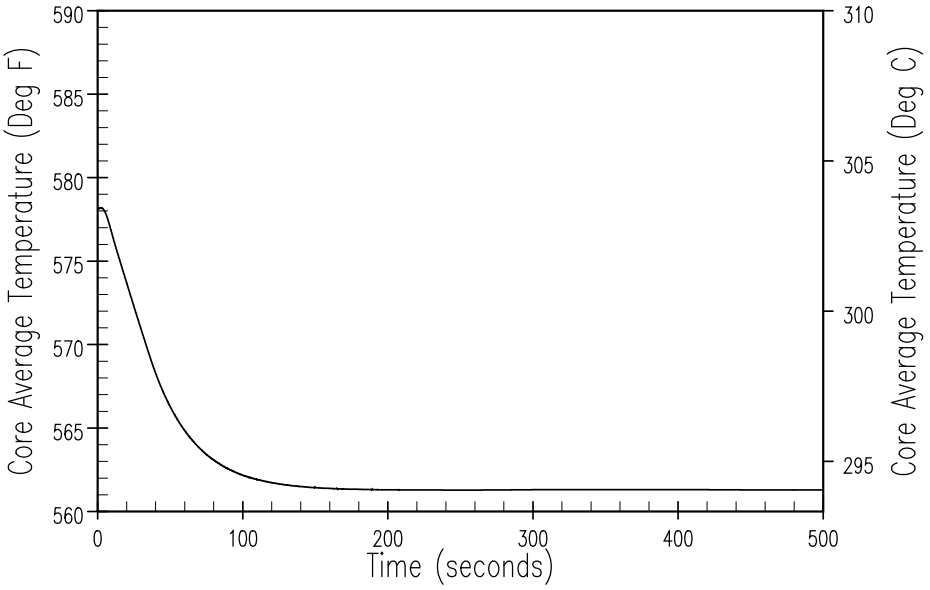

Figure 15.1.3-4
Core Average Temperature Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

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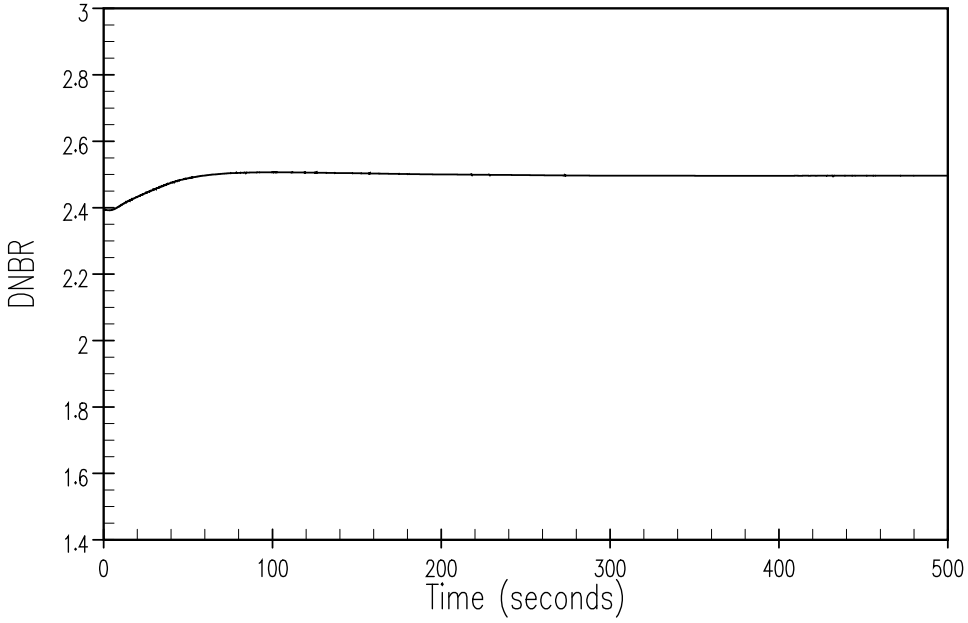

Figure 15.1.3-5
DNBR Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

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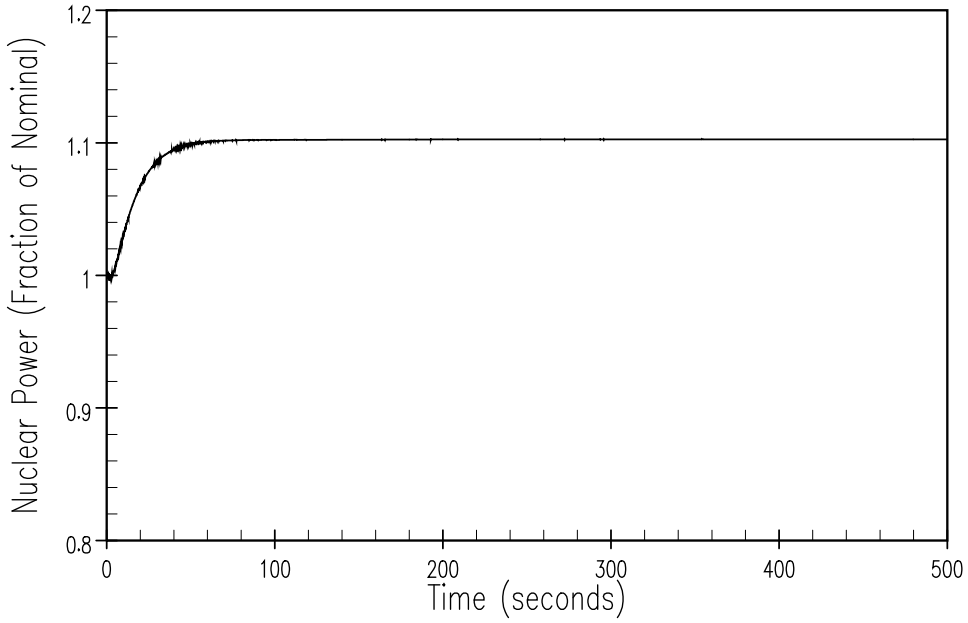

Figure 15.1.3-6
Nuclear Power Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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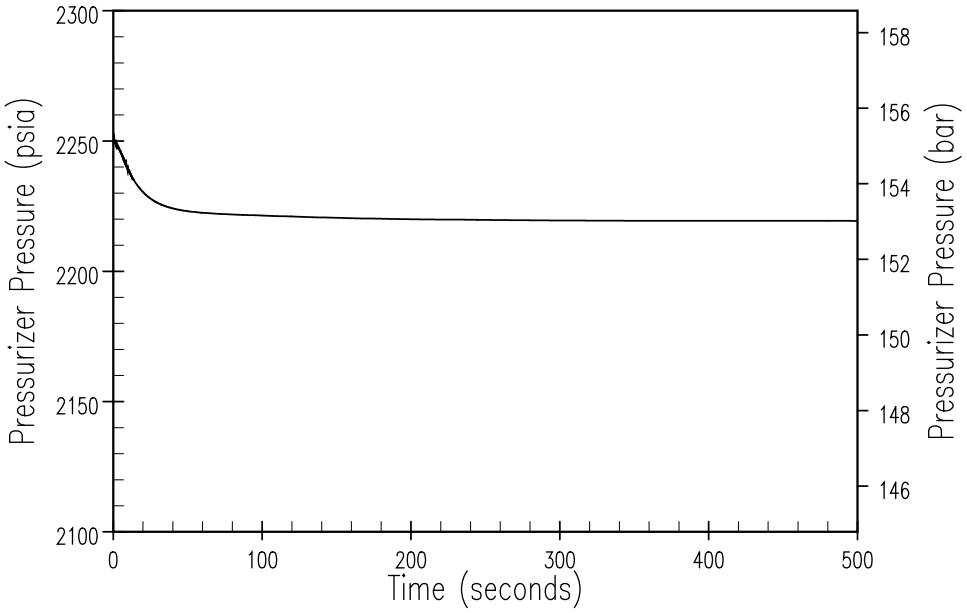

Figure 15.1.3-7
Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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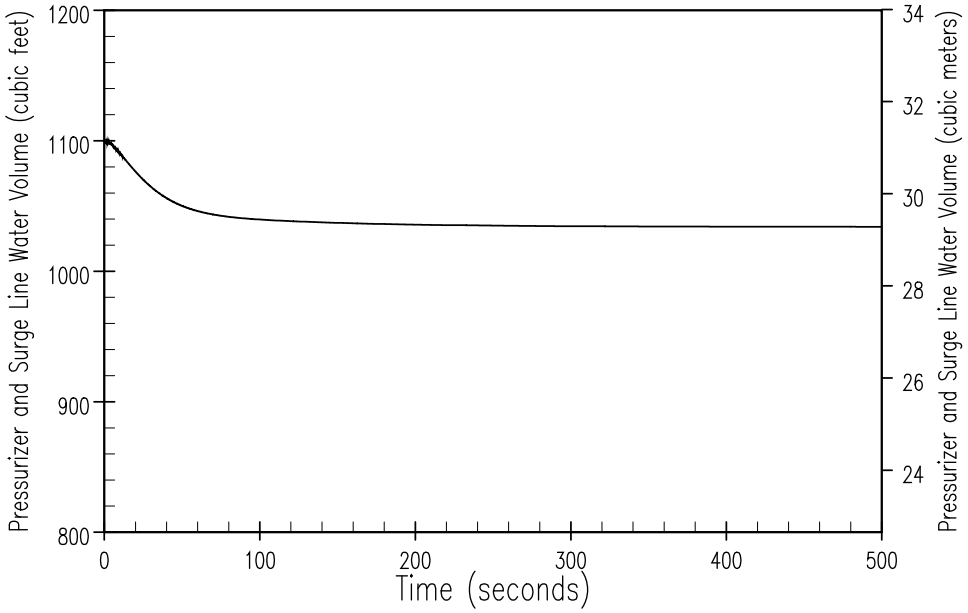

Figure 15.1.3-8
Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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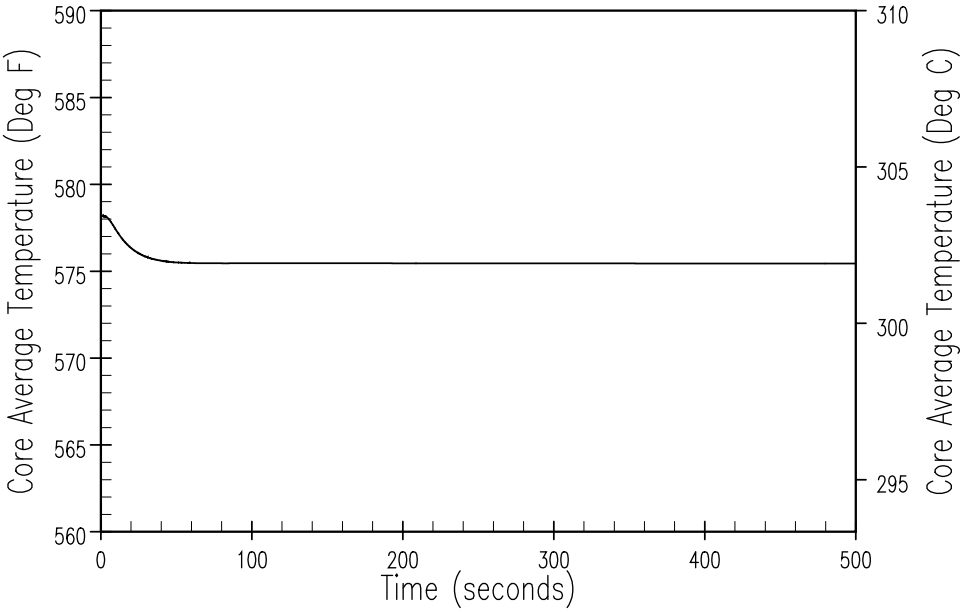

Figure 15.1.3-9
Core Average Temperature Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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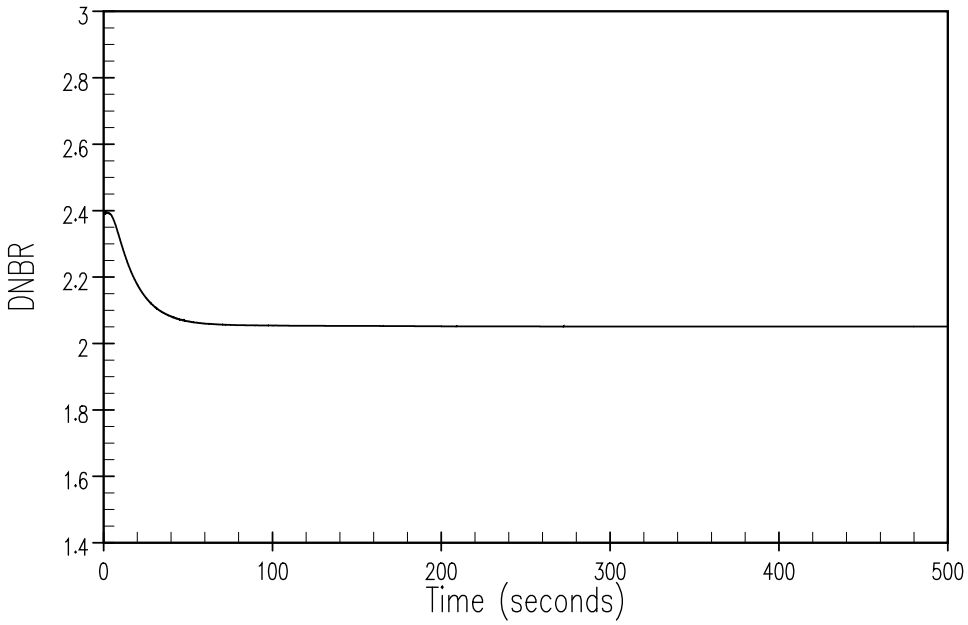

Figure 15.1.3-10
DNBR Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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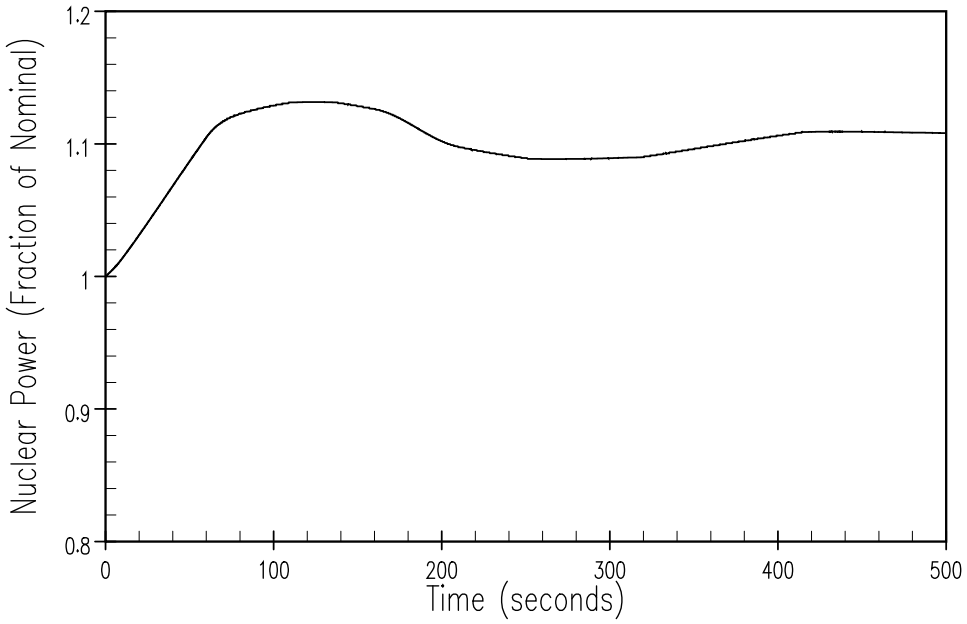

Figure 15.1.3-11
Nuclear Power Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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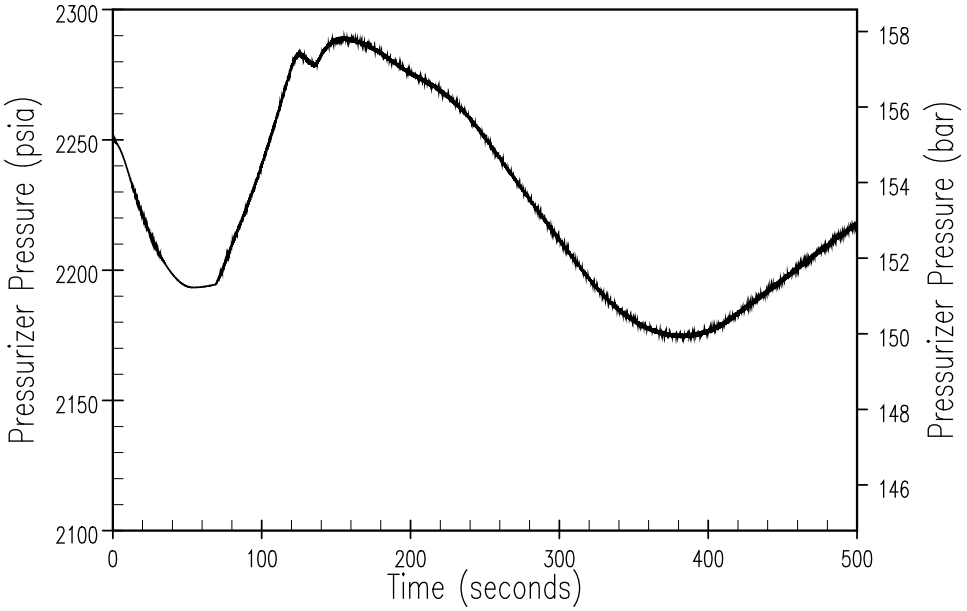

Figure 15.1.3-12
Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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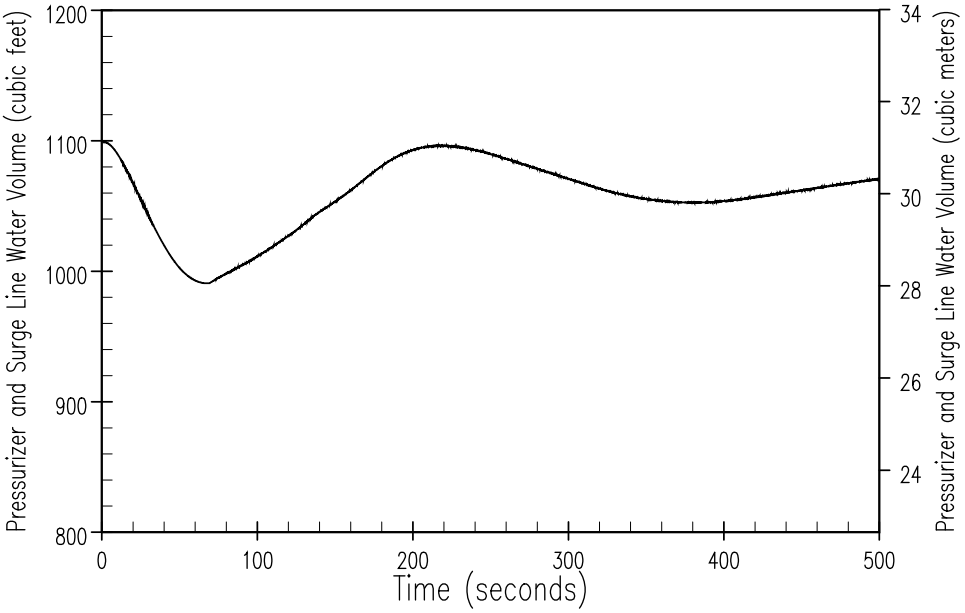

Figure 15.1.3-13
Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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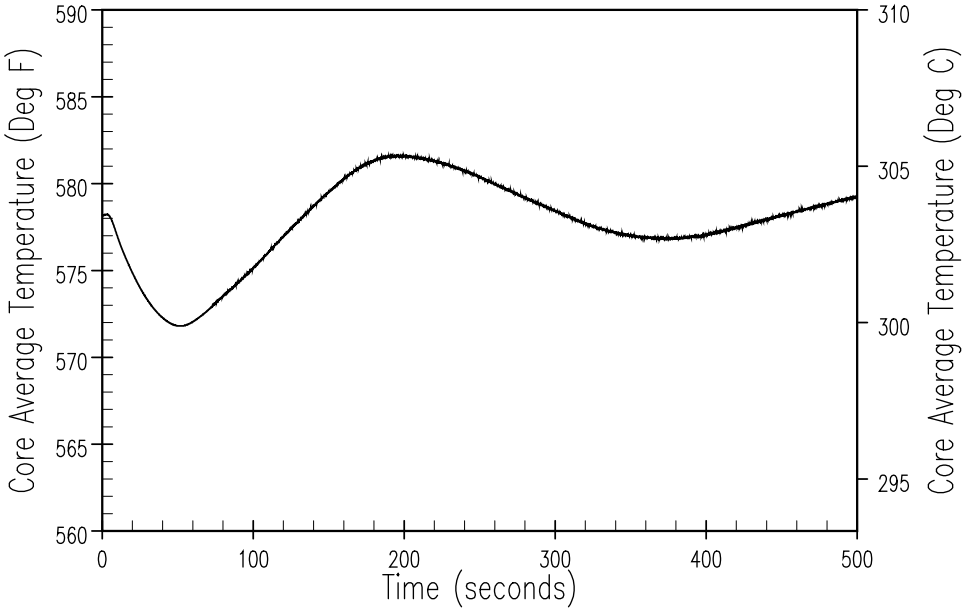

Figure 15.1.3-14
Core Average Temperature Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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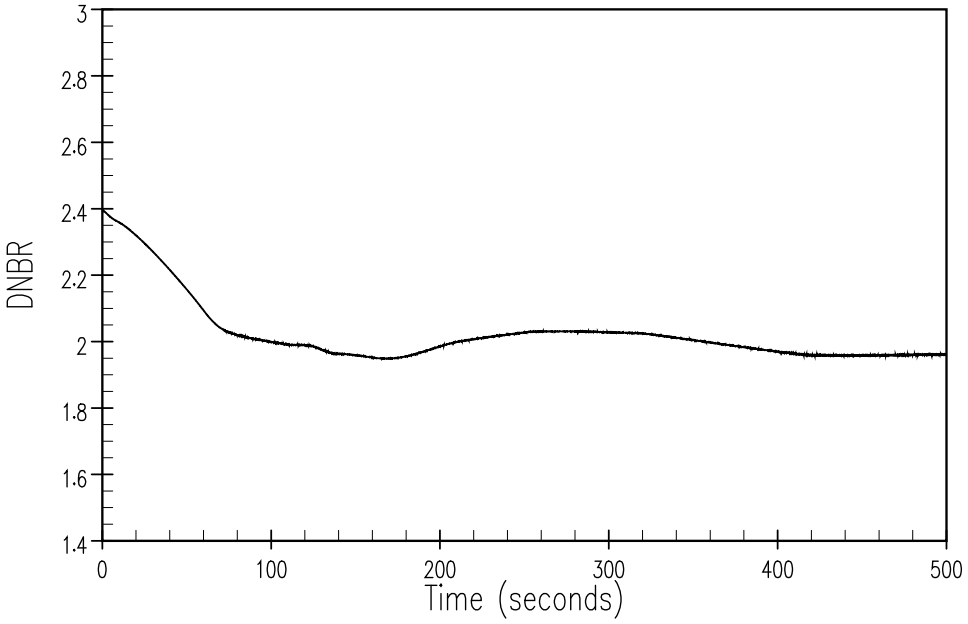

Figure 15.1.3-15
DNBR Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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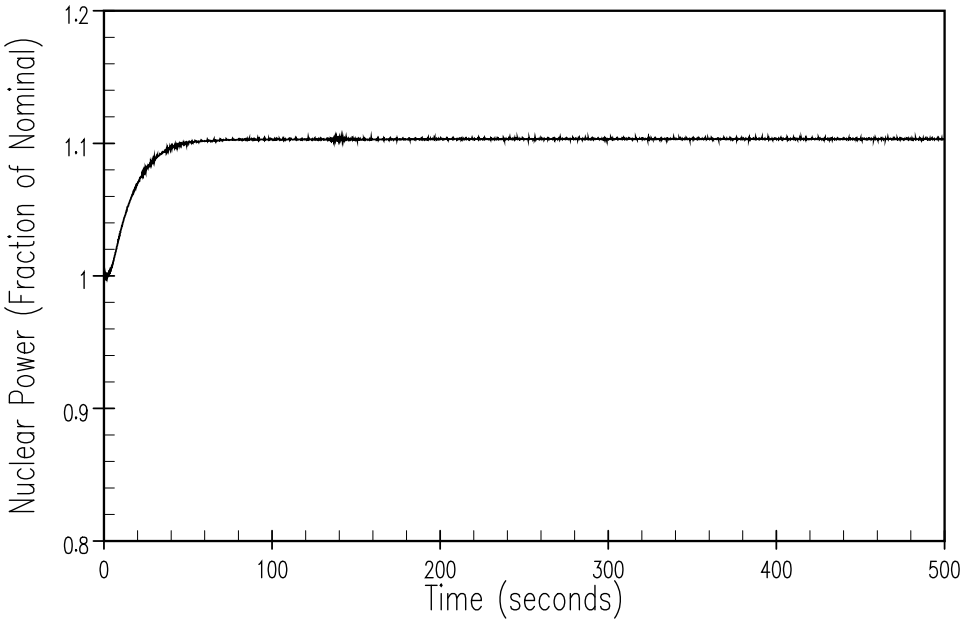

Figure 15.1.3-16
Nuclear Power Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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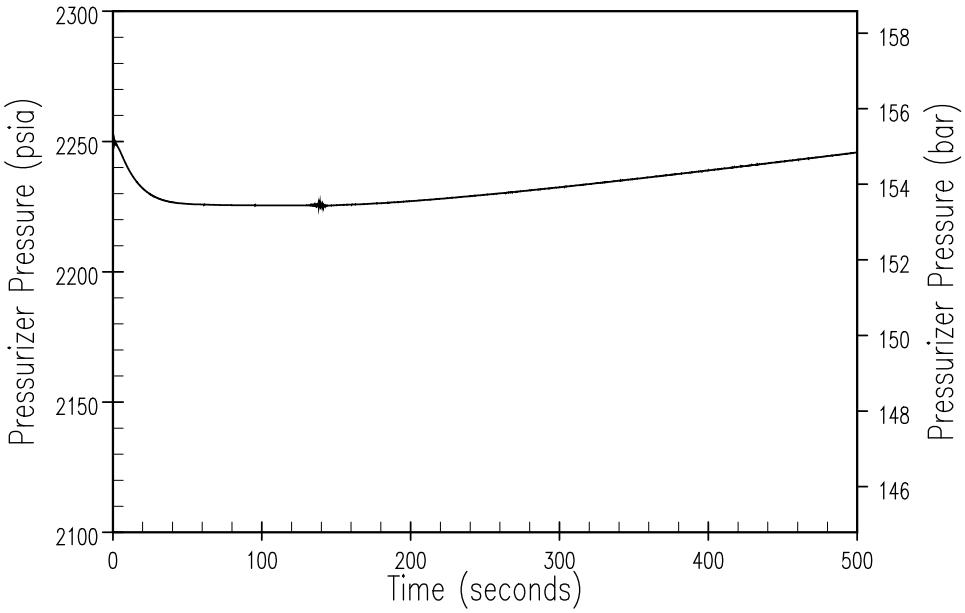

Figure 15.1.3-17
Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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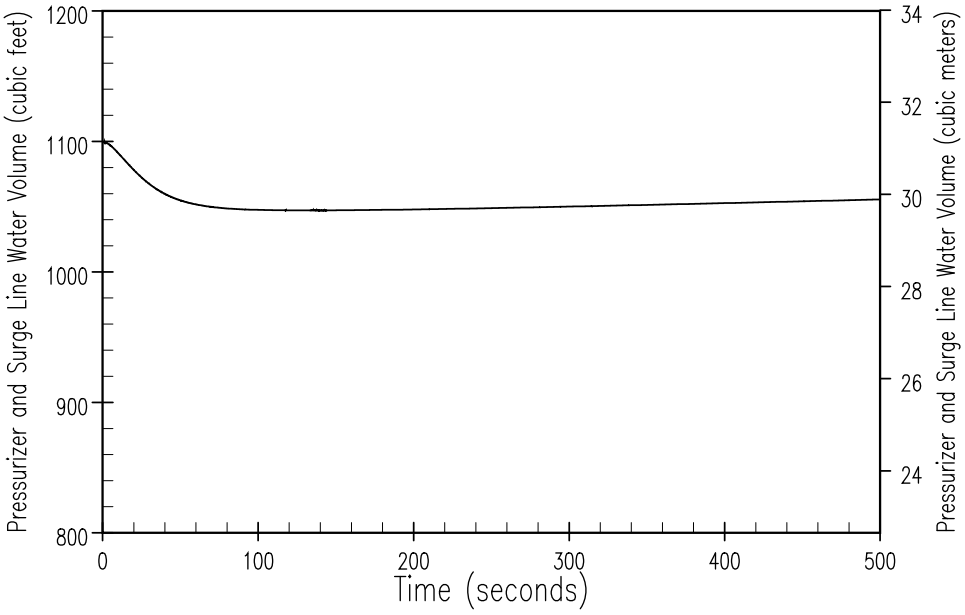

Figure 15.1.3-18
Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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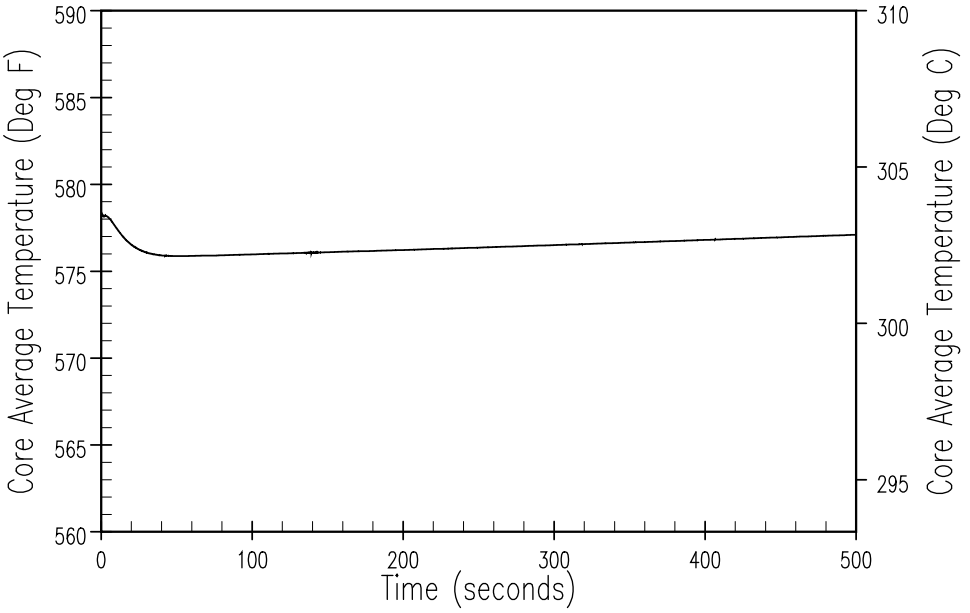

Figure 15.1.3-19
Core Average Temperature Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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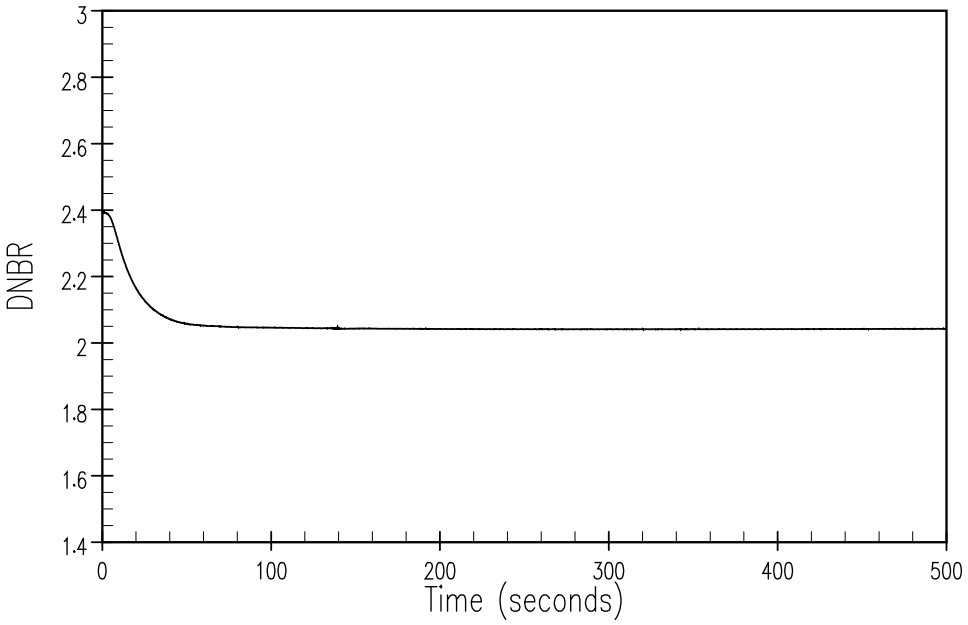

Figure 15.1.3-20
DNBR Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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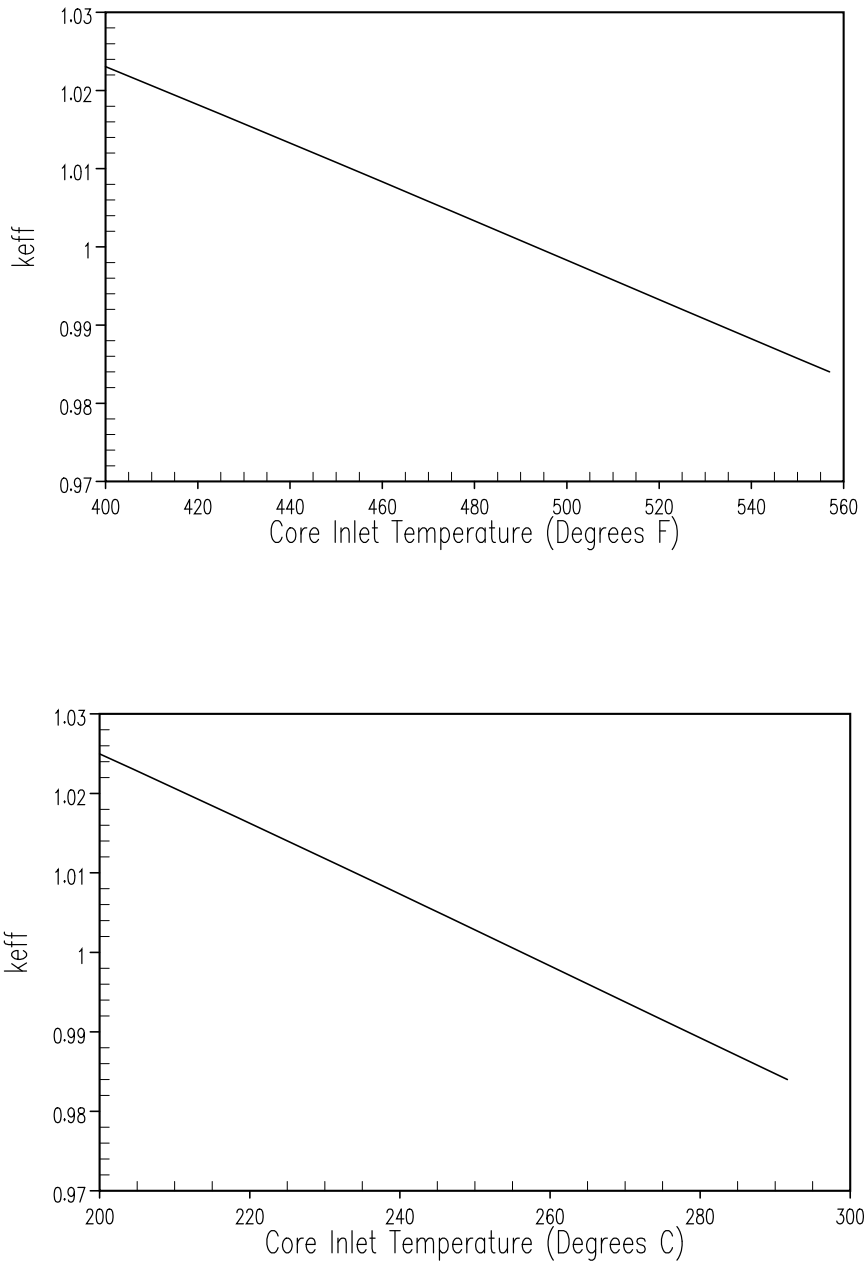

Figure 15.1.4-1
K_{eff} Versus Core Inlet Temperature
Steam Line Break Events

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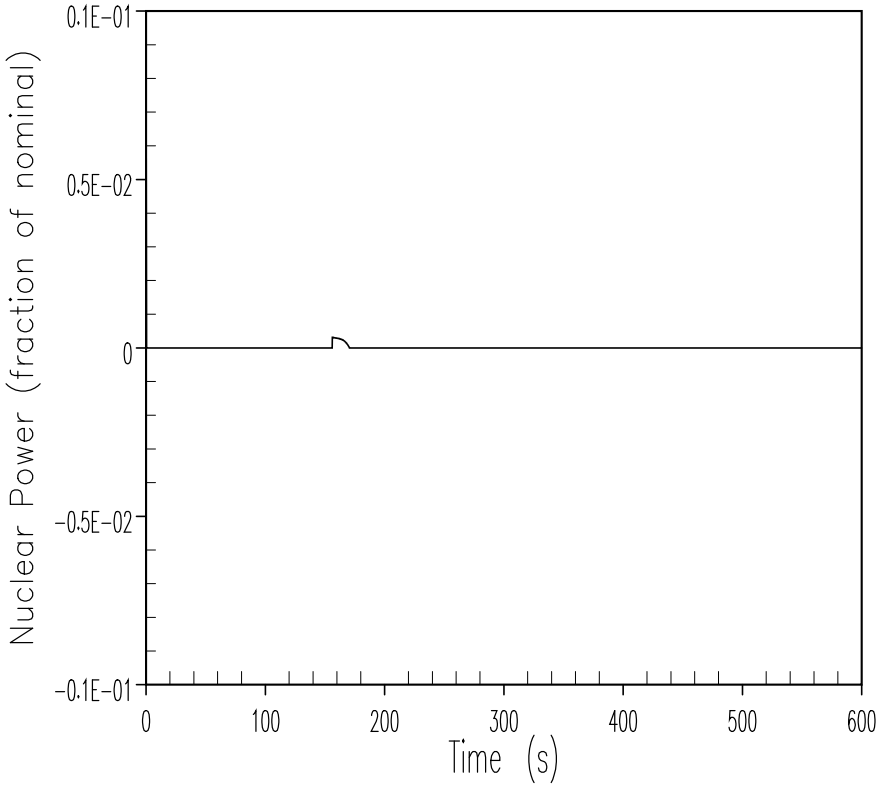

Figure 15.1.4-2
Nuclear Power Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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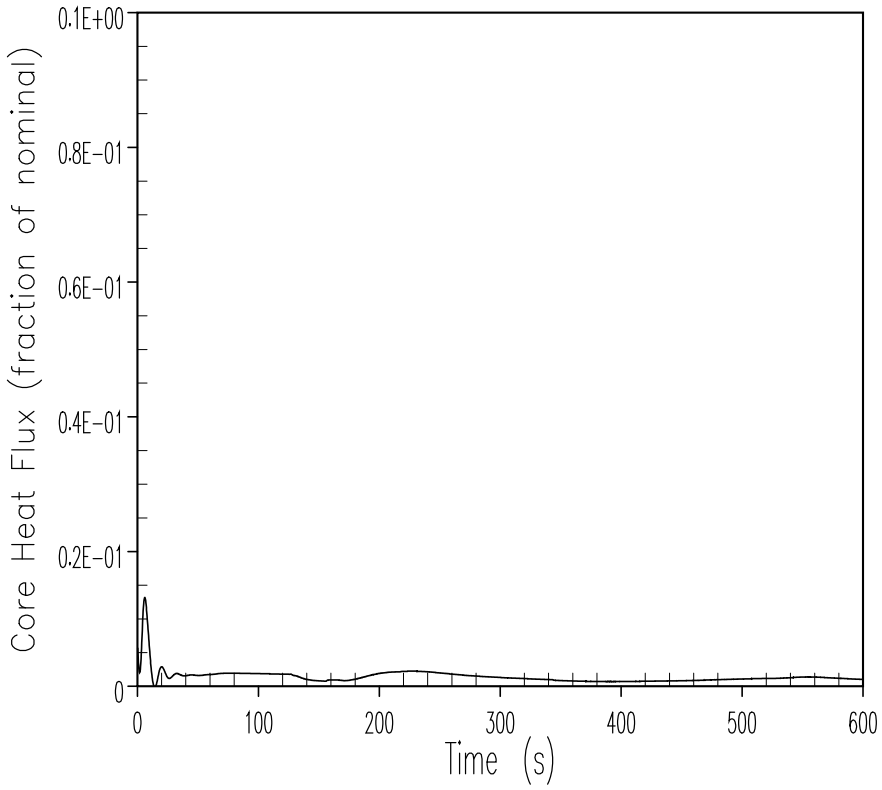

Figure 15.1.4-3
Core Heat Flux Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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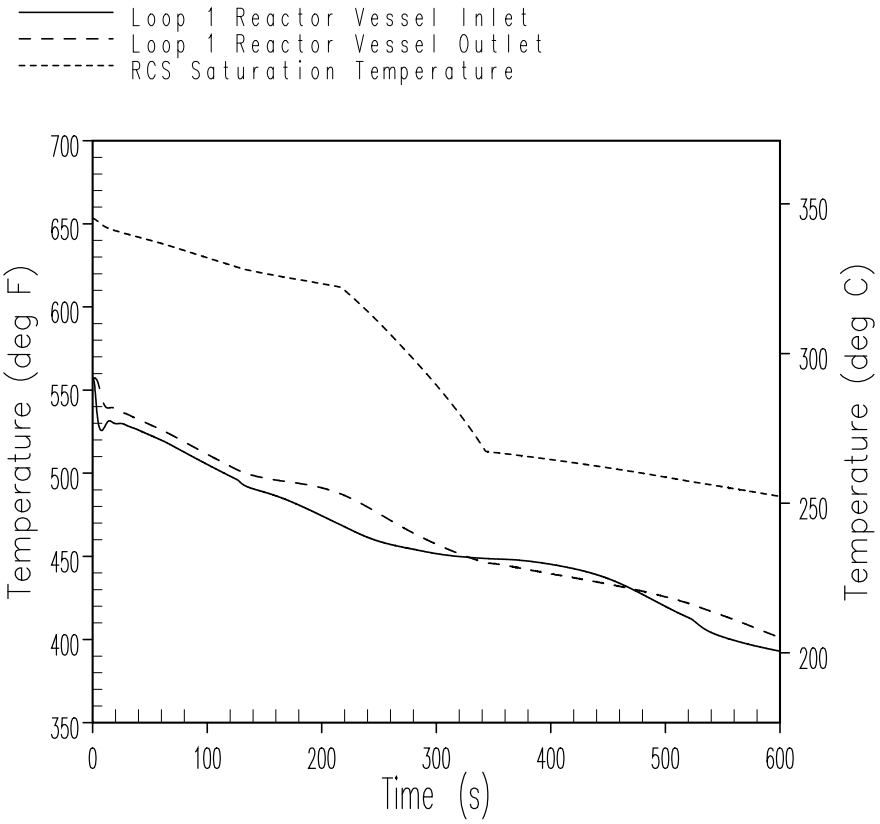

Figure 15.1.4-4
Loop 1 Reactor Coolant Temperatures
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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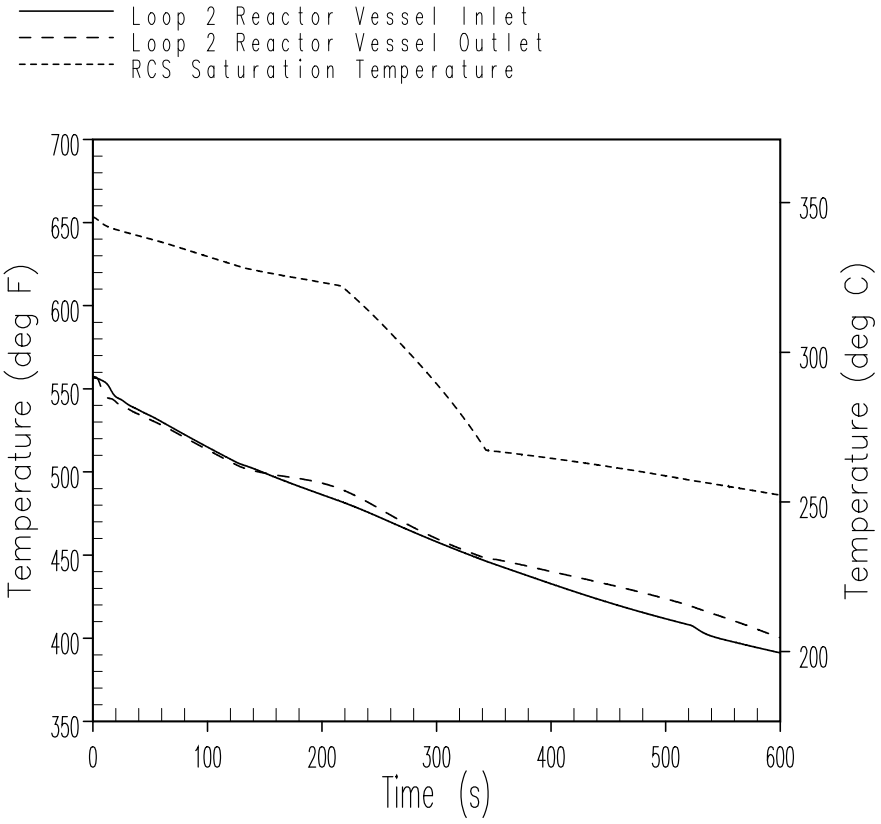

Figure 15.1.4-5
Loop 2 (Faulted Loop) Reactor Coolant Temperatures
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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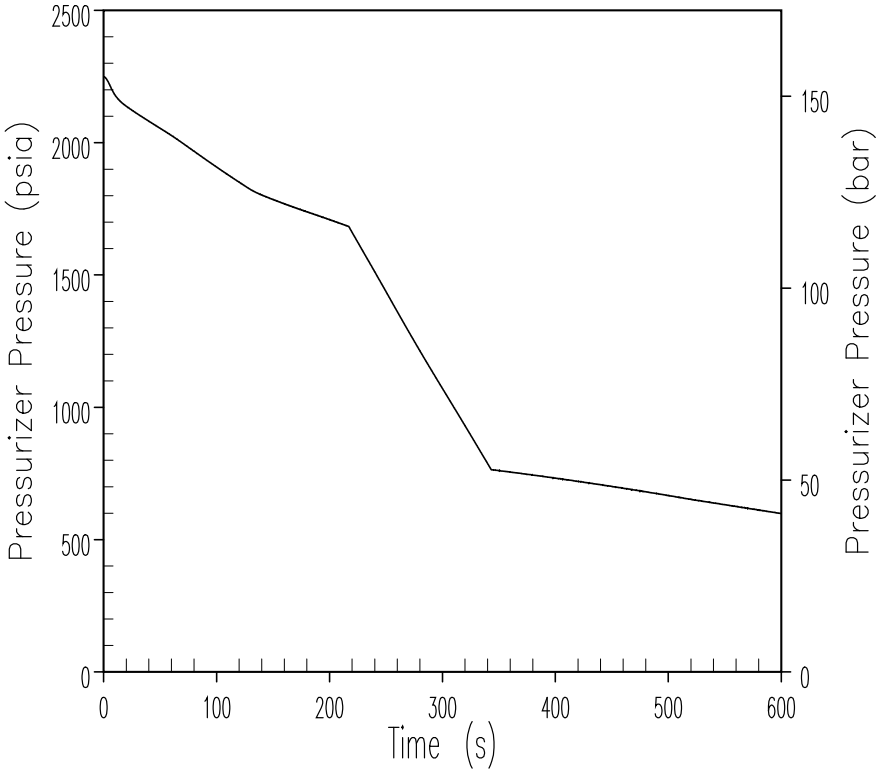

Figure 15.1.4-6
Pressurizer Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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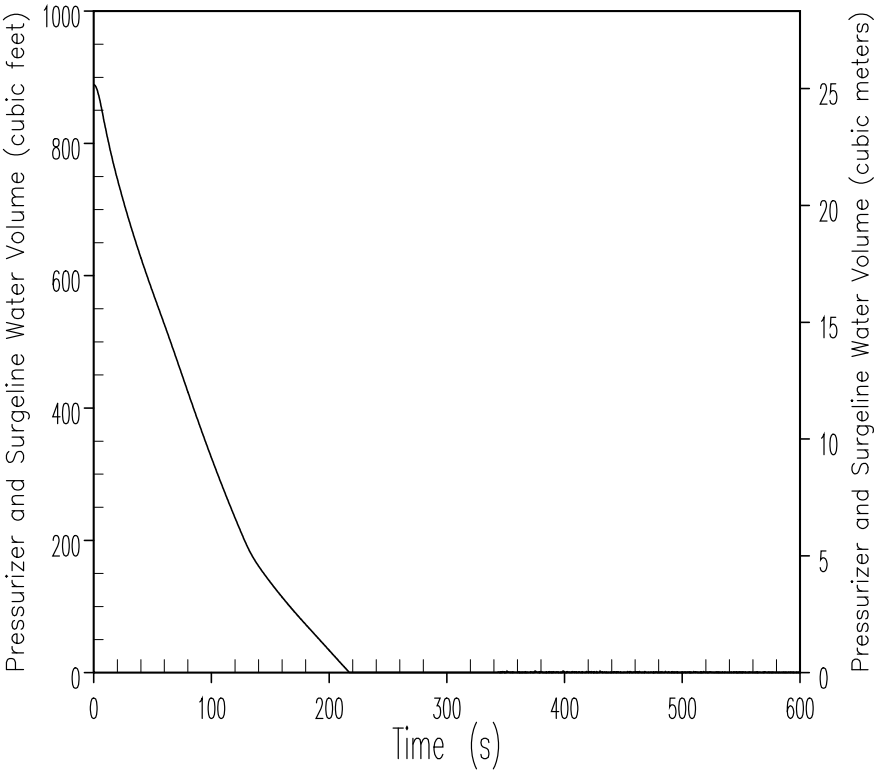

Figure 15.1.4-7
Pressurizer and Surge Line Water Volume Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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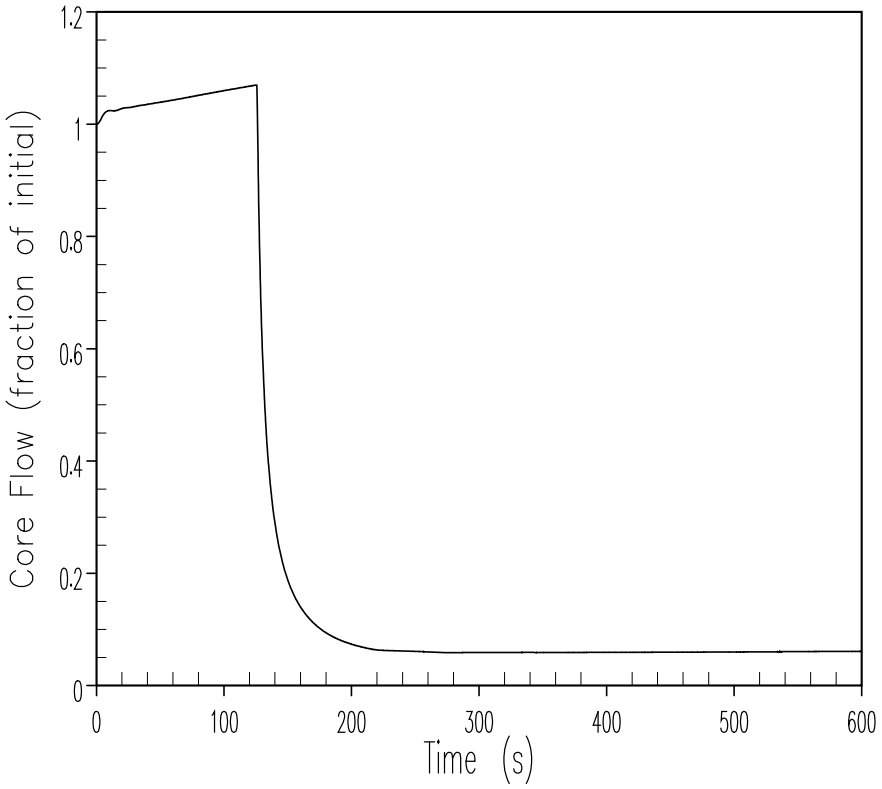

Figure 15.1.4-8
Core Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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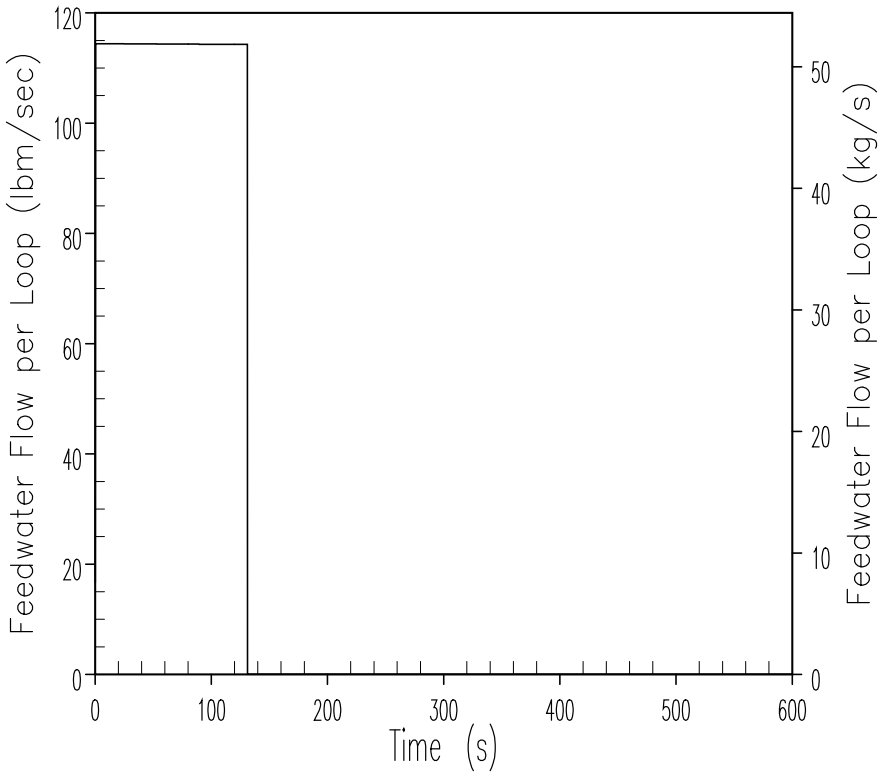

Figure 15.1.4-9
Feedwater Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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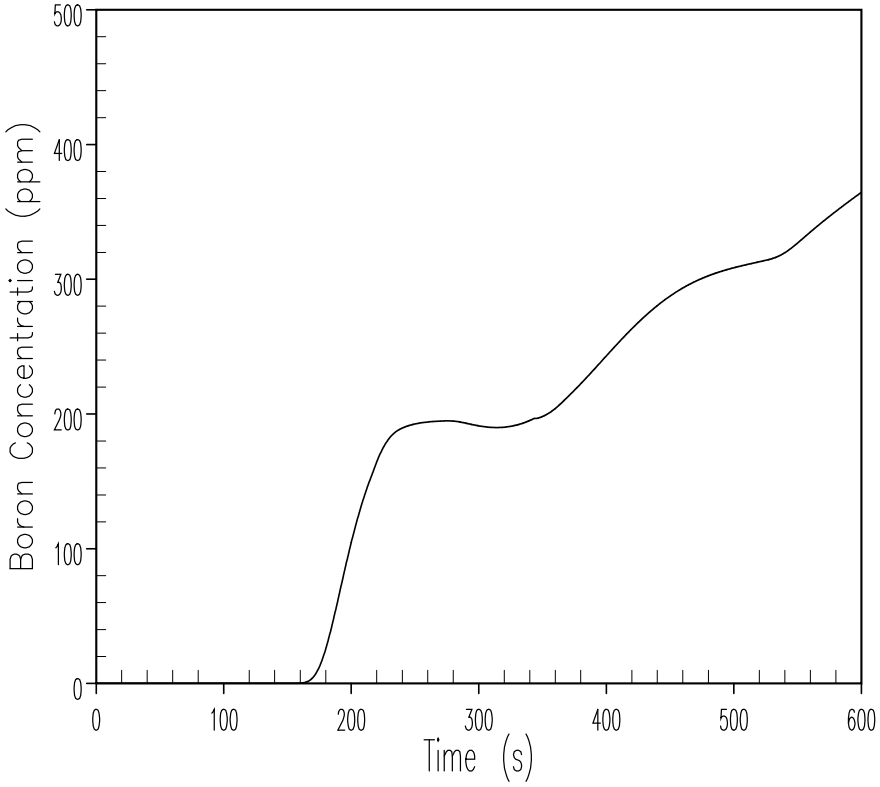

Figure 15.1.4-10
Core Boron Concentration Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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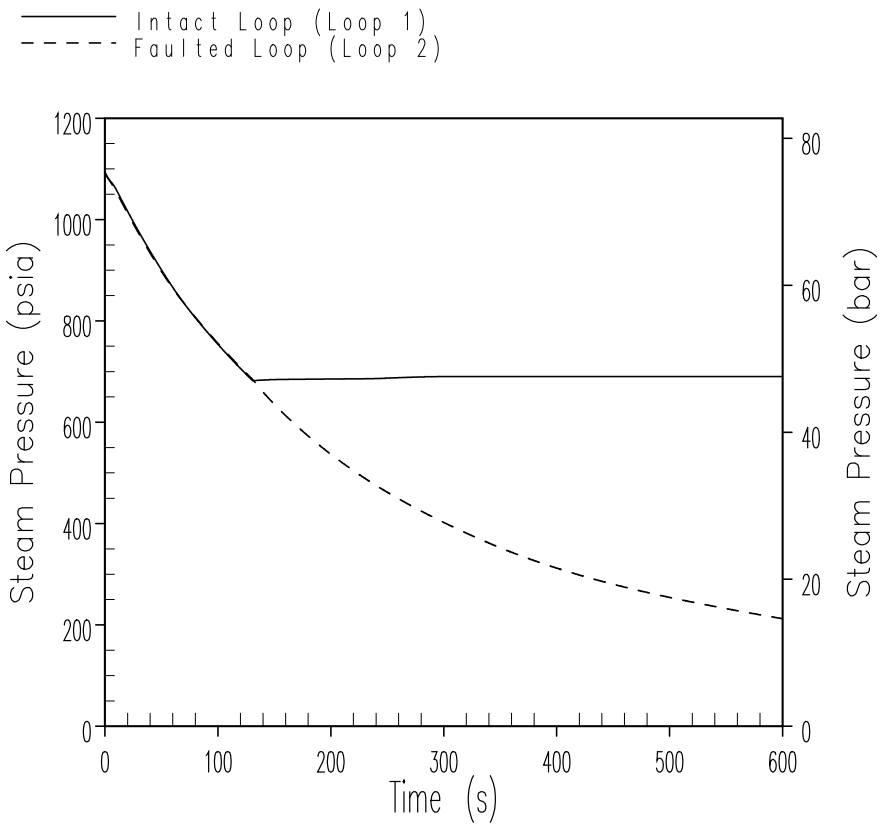

Figure 15.1.4-11
Steam Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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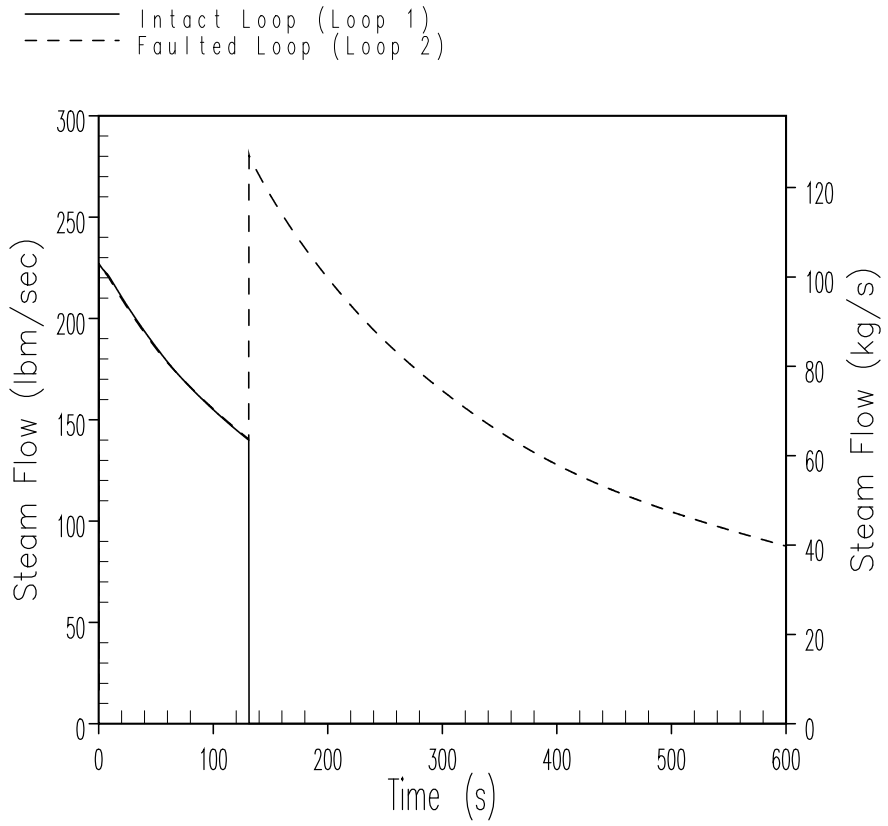

Figure 15.1.4-12
Steam Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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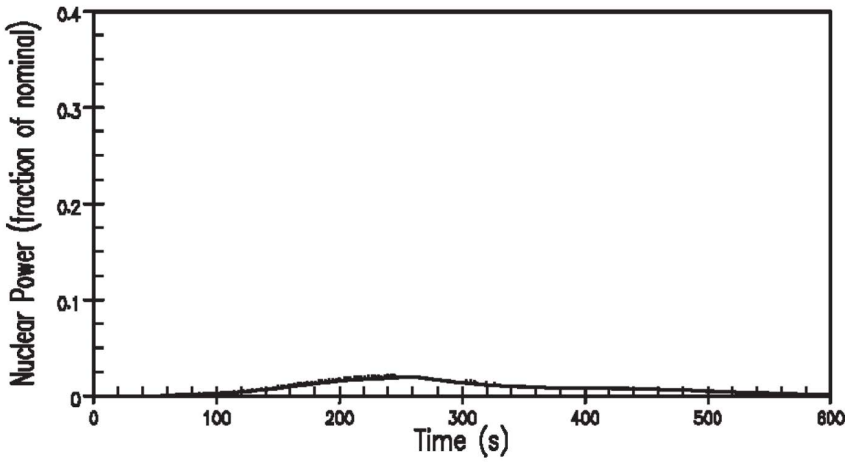

Figure 15.1.5-1
Nuclear Power Transient Steam System Piping Failure

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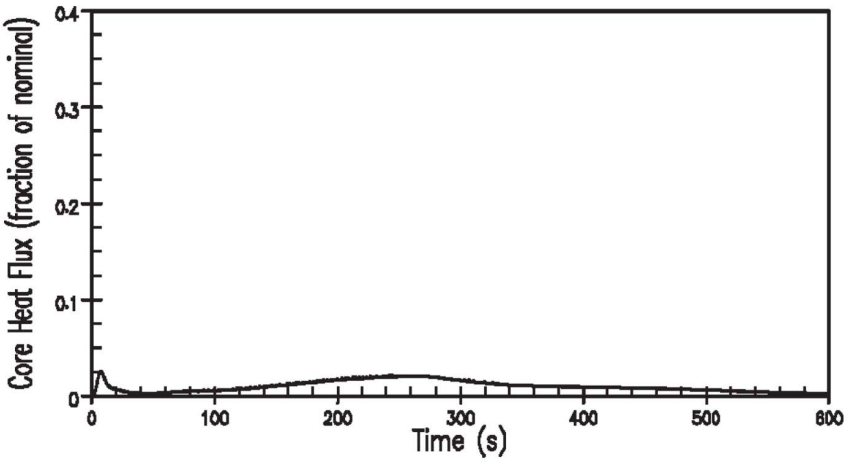

Figure 15.1.5-2
Core Heat Flux Transient Steam System Piping Failure

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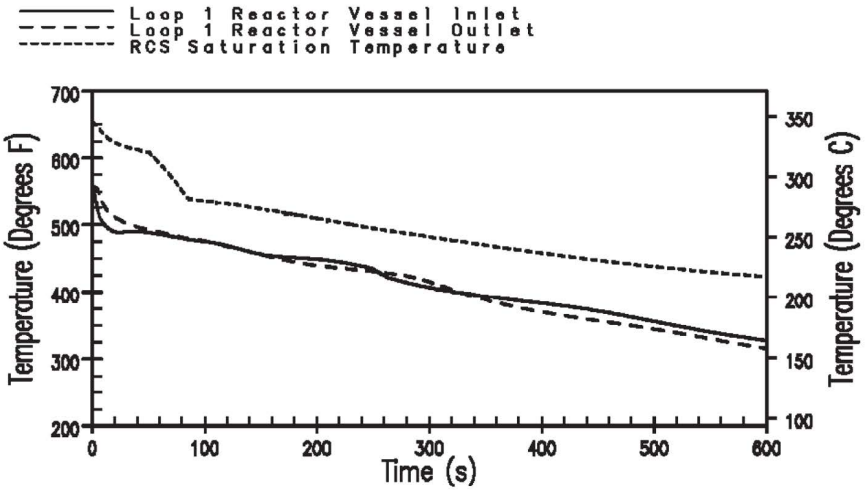

Figure 15.1.5-3
Loop 1 Reactor Coolant Temperatures
Steam System Piping Failure

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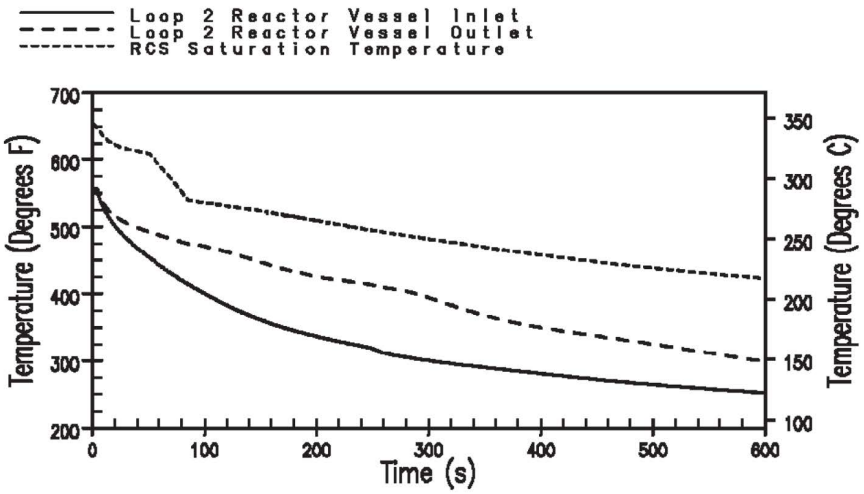

Figure 15.1.5-4
Loop 2 Reactor Coolant Temperatures
Steam System Piping Failure

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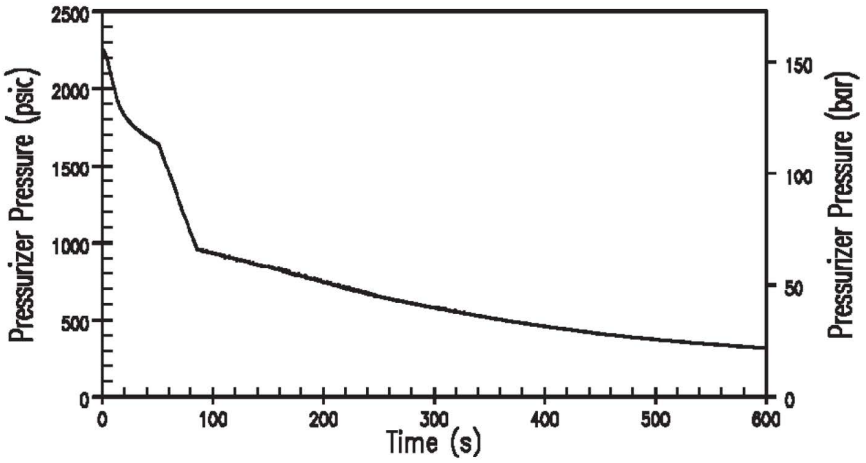

Figure 15.1.5-5
Pressurizer Pressure Transient
Steam System Piping Failure

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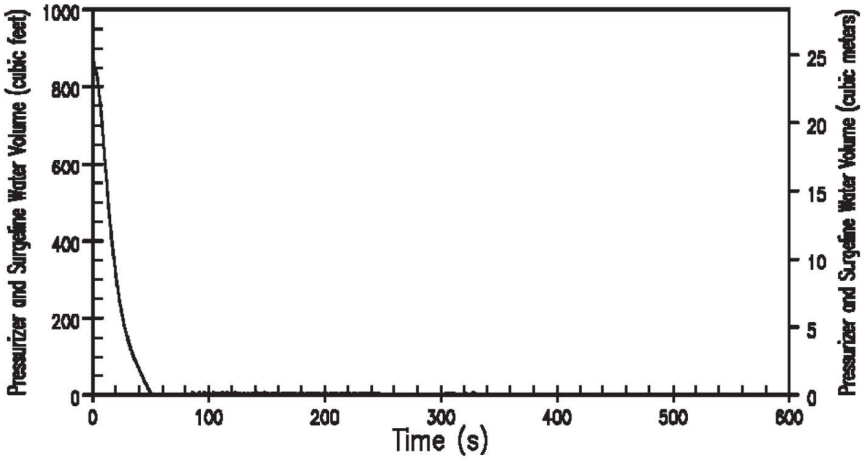

Figure 15.1.5-6
Pressurizer and Surge Line Water Volume Transient
Steam System Piping Failure

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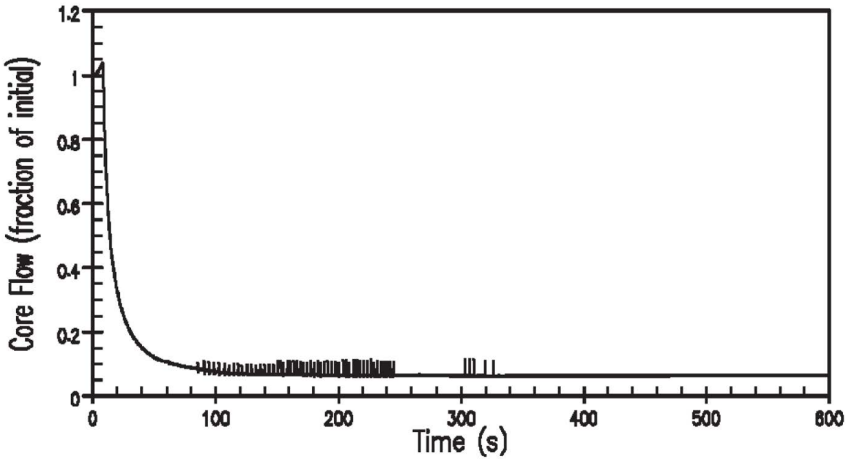

Figure 15.1.5-7
Core Flow Transient Steam System Piping Failure

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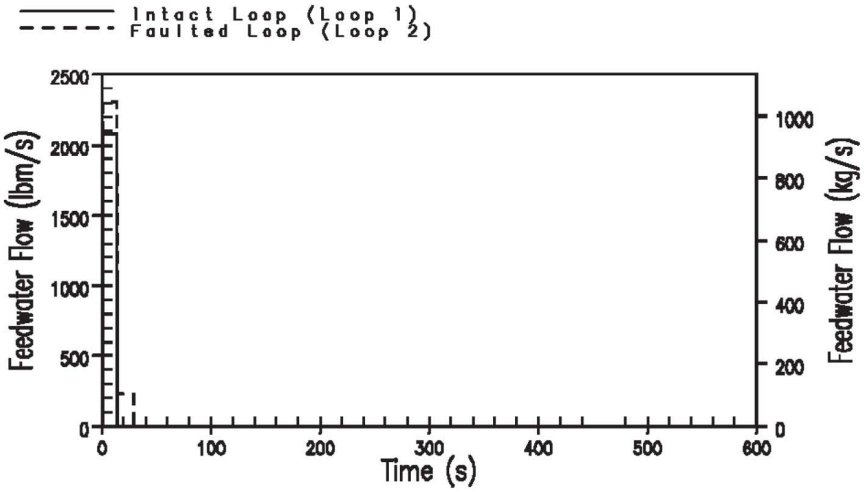

Figure 15.1.5-8
Feedwater Flow Transient Steam System Piping Failure

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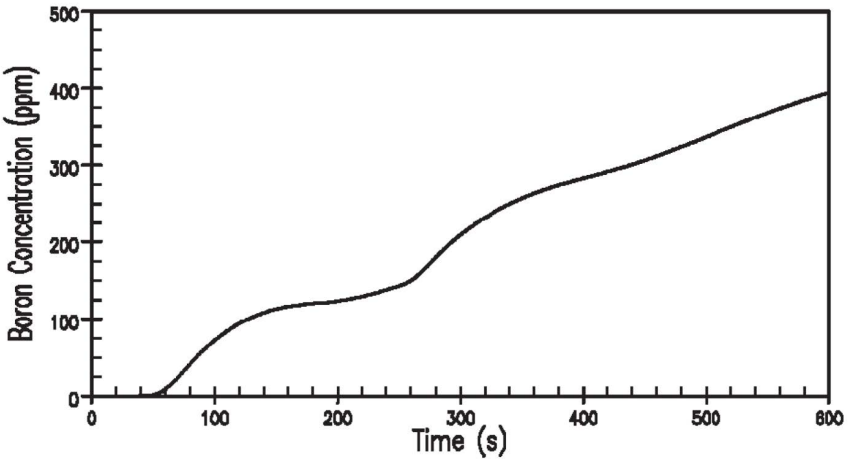

Figure 15.1.5-9
Core Boron Concentration Transient Steam System Piping Failure

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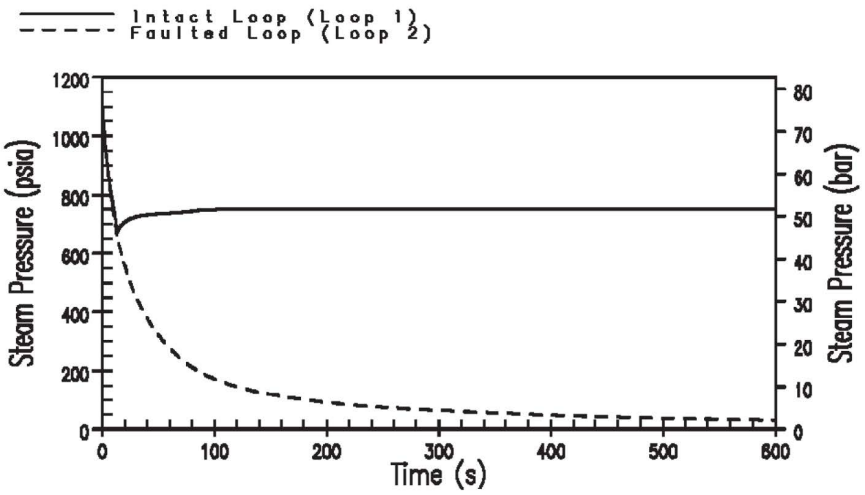

Figure 15.1.5-10
Steam Pressure Transient Steam System Piping Failure

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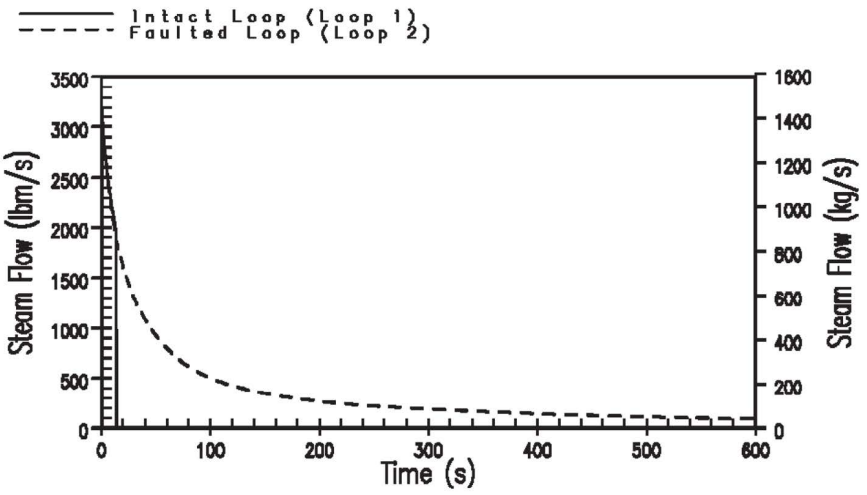

Figure 15.1.5-11
Steam Flow Transient Steam System Piping Failure

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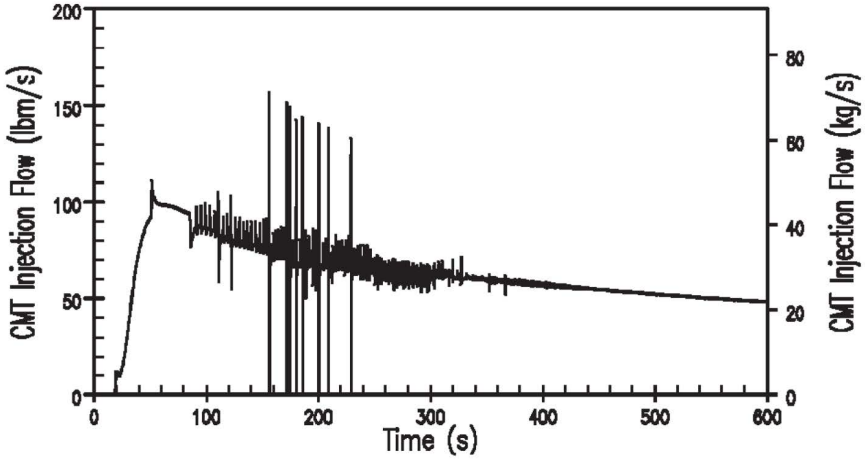

Figure 15.1.5-12
Core Makeup Tank Injection Flow
Steam System Piping Failure

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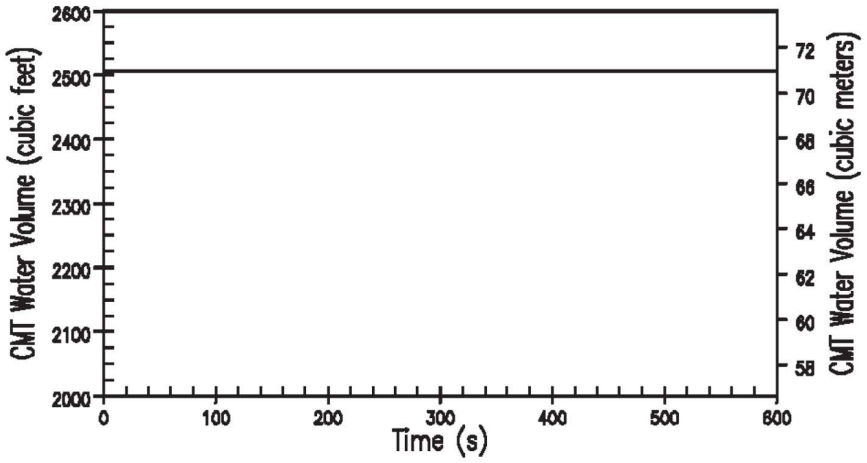

Figure 15.1.5-13
Core Makeup Tank Water Volume Steam System Piping Failure

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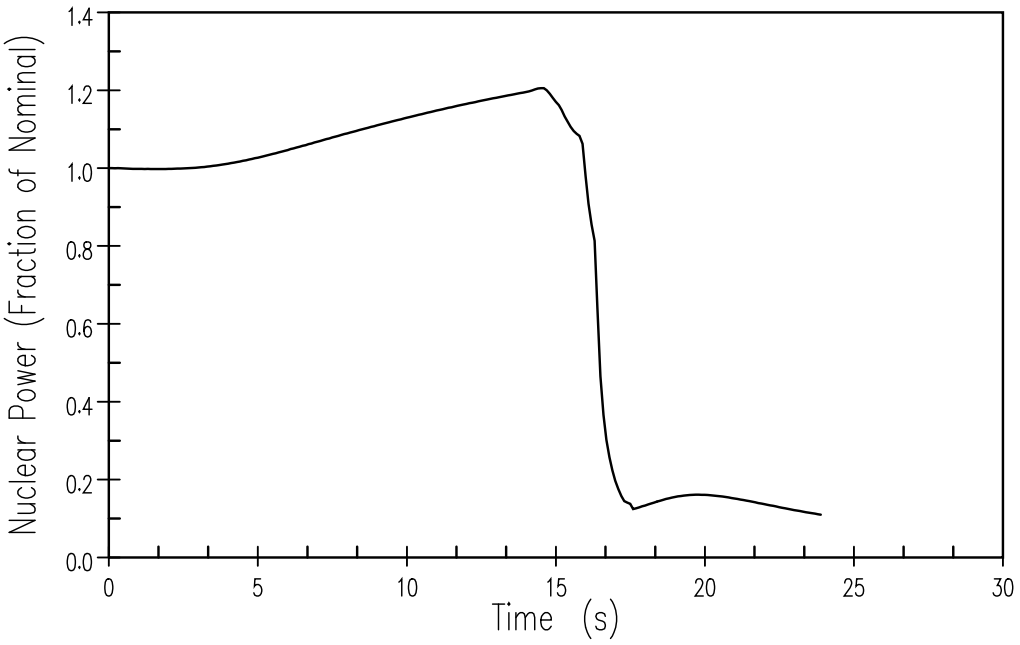

Figure 15.1.5.5-1
Nuclear Power Transient
Steam System Piping Failure at Full Power — 0.87 ft^2 Break Size

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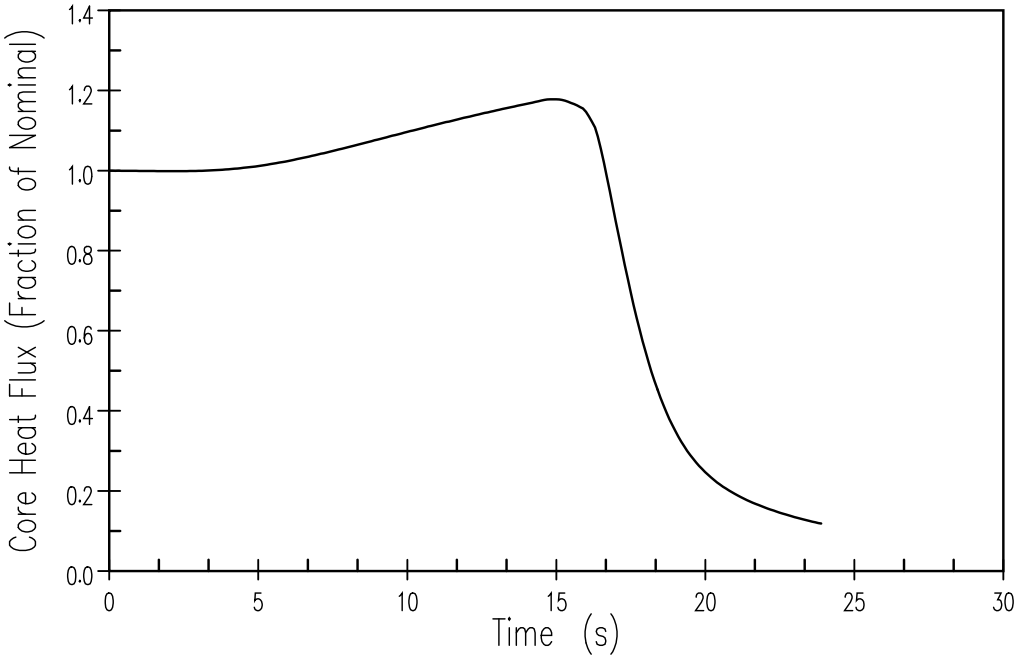

Figure 15.1.5.5-2
Core Heat Flux Transient
Steam System Piping Failure at Full Power — 0.87 ft^2 Break Size

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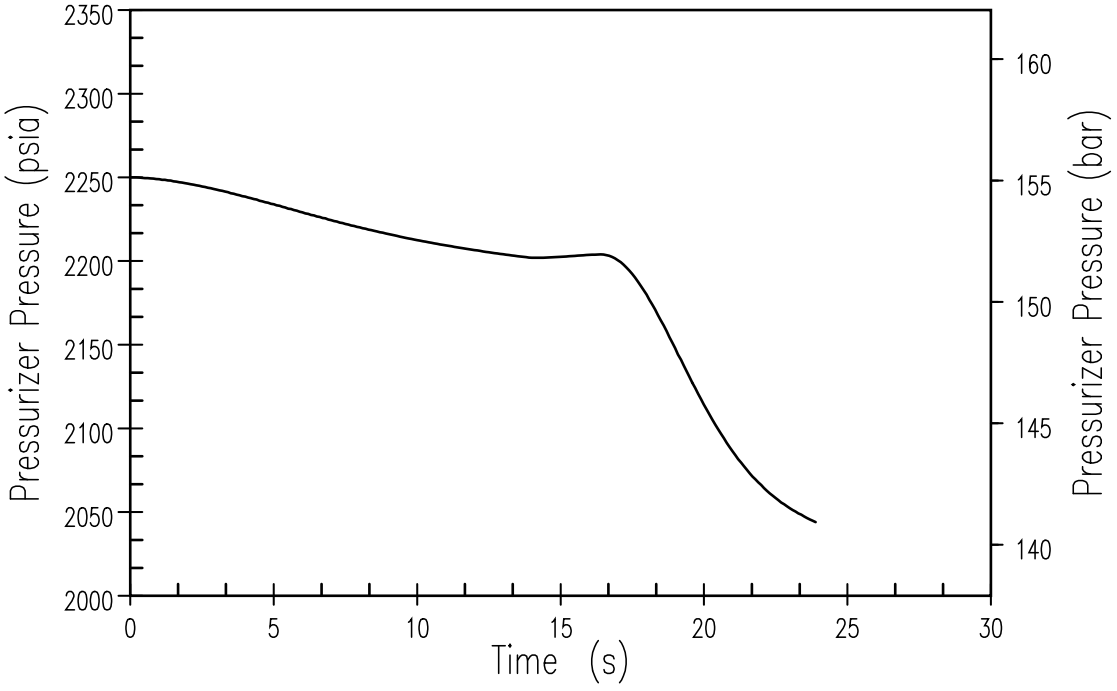

Figure 15.1.5.5-3
Pressurizer Pressure Transient
Steam System Piping Failure at Full Power — 0.87 ft^2 Break Size

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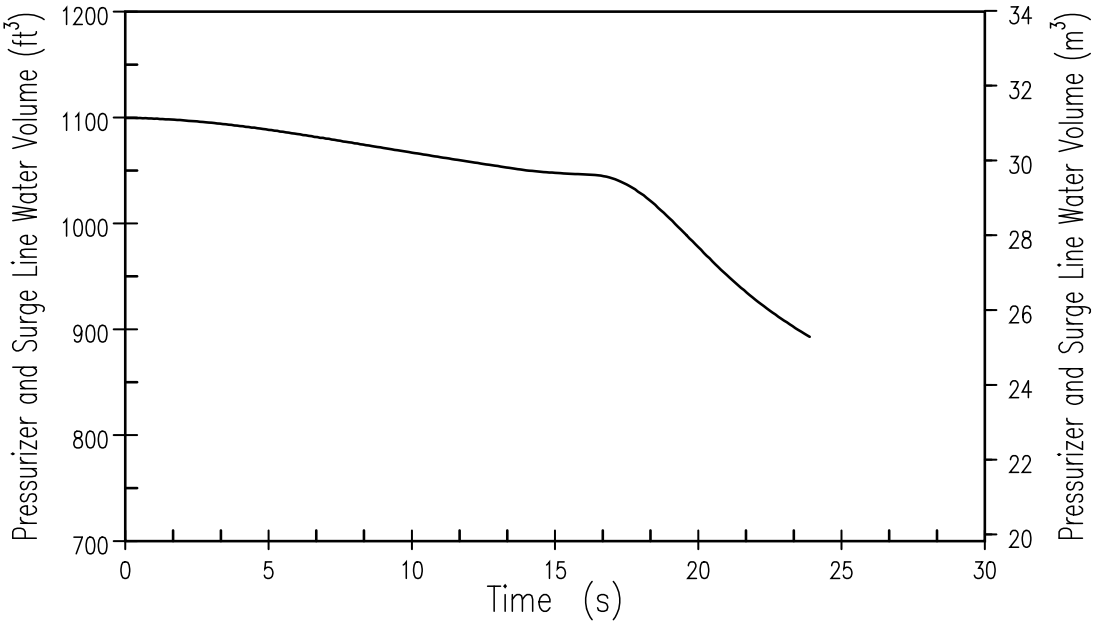

Figure 15.1.5.5-4
Pressurizer Water Volume Transient
Steam System Piping Failure at Full Power — 0.87 ft^2 Break Size

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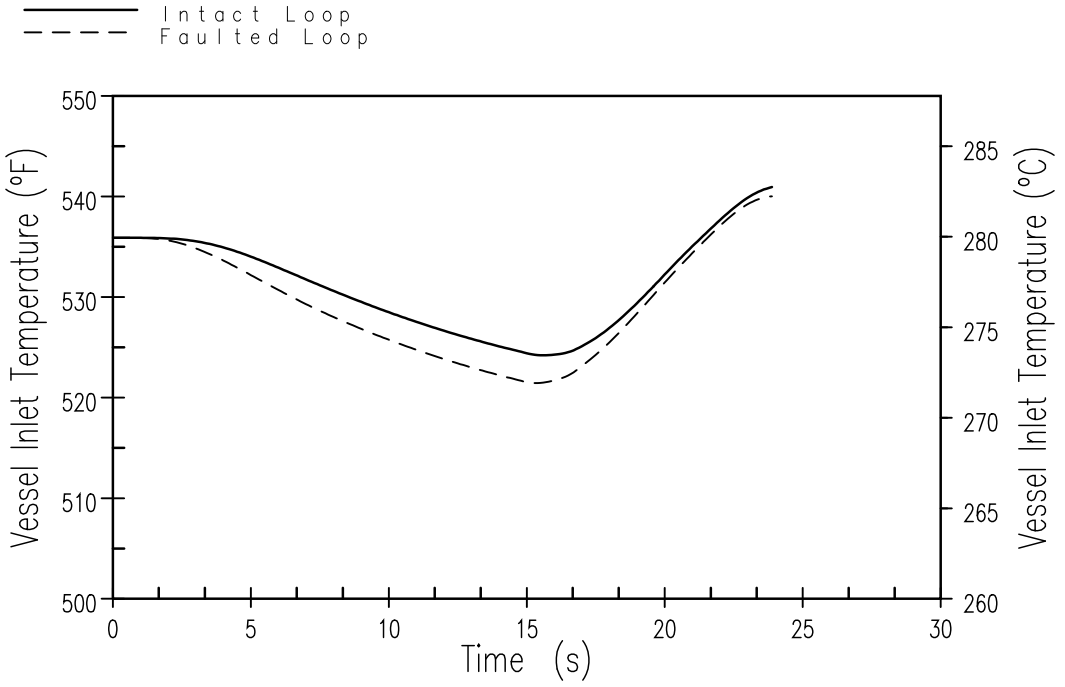

Figure 15.1.5.5-5
Vessel Inlet Temperature Transient
(Intact and Faulted Loops)
Steam System Piping Failure at Full Power — 0.87 ft^2 Break Size

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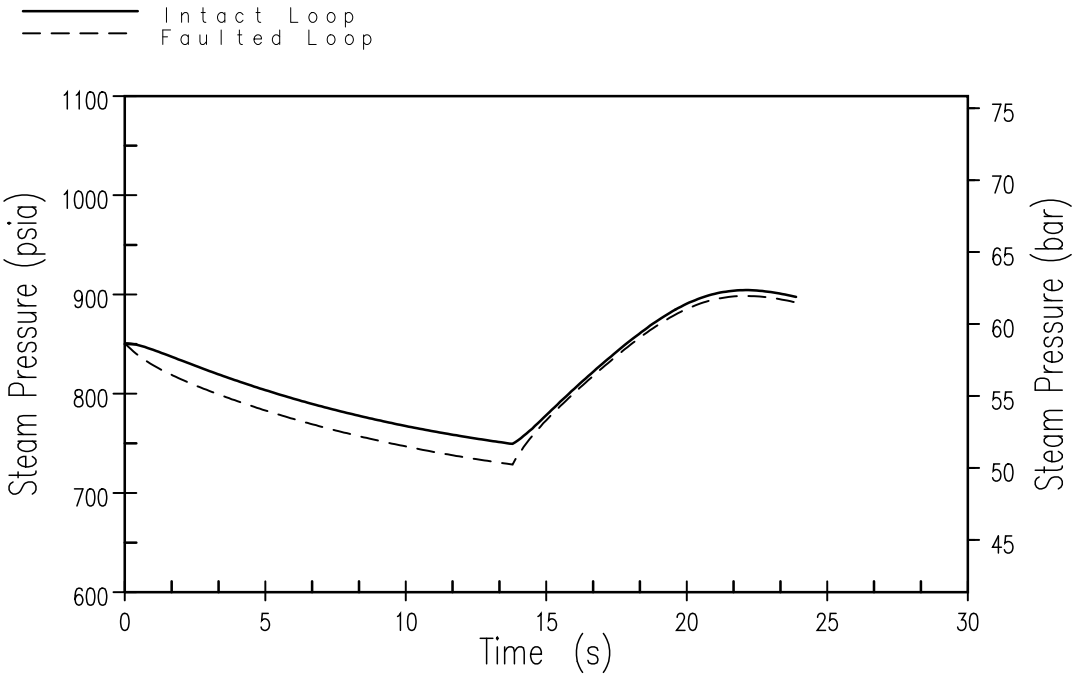

Figure 15.1.5.5-6
Steam Generator Pressure Transient
(Intact and Faulted Loops)
Steam System Piping Failure at Full Power — 0.87 ft^2 Break Size

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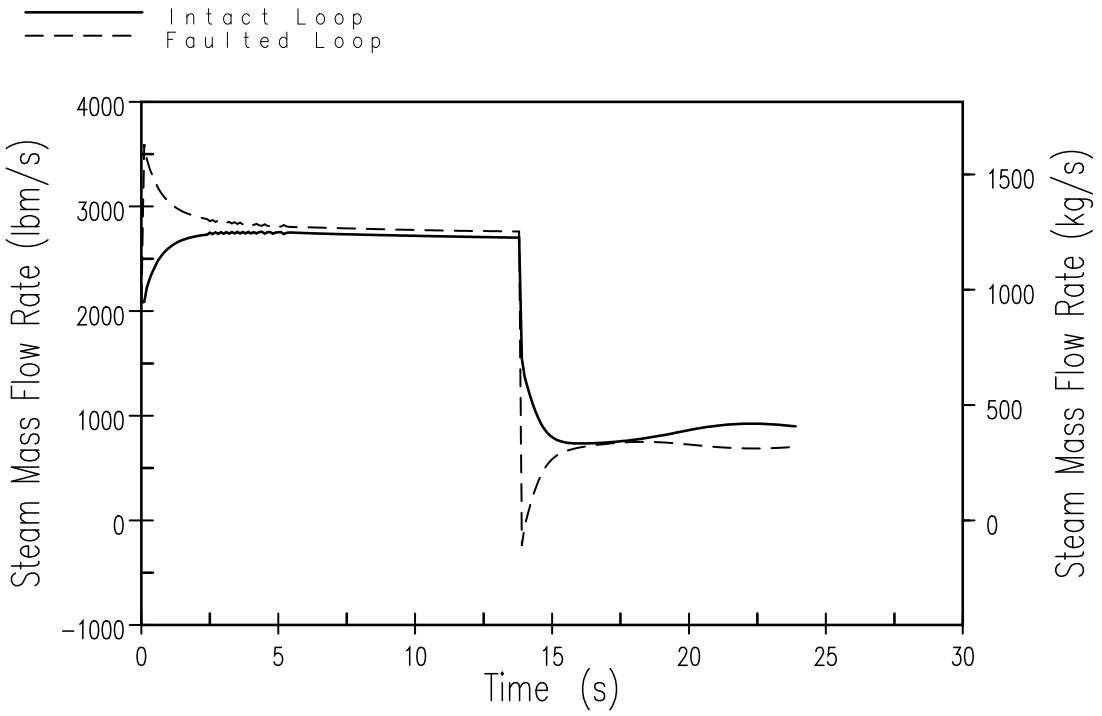

Figure 15.1.5.5-7
Steam Flow Transient (Intact and Faulted Loops)
Steam System Piping Failure at Full Power — 0.87 ft^2 Break Size

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Figures 15.1.6-1–15.1.6-8 Not Used

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15.2 Decrease in Heat Removal by the Secondary System

A number of transients and accidents that could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system are postulated. Analyses are presented in this section for the following events that are identified as more limiting than the others:

- Steam pressure regulator malfunction or failure that results in decreasing steam flow
- Loss of external electrical load
- Turbine trip
- Inadvertent closure of main steam isolation valves
- Loss of condenser vacuum and other events resulting in turbine trip
- Loss of ac power to the station auxiliaries
- Loss of normal feedwater flow
- Feedwater system pipe break

The above items are considered to be Condition II events, with the exception of a feedwater system pipe break, which is considered to be a Condition IV event.

For events in this section where PRHR heat exchanger actuation occurs, transients are presented until the PRHR heat exchanger heat removal matches decay heat generation. After that point in time, PRHR heat exchanger performance is driven by the performance of the passive containment cooling systems to control containment pressure and the ability of the condensate collection features to return condensate to the in-containment refueling water storage tank. The performance of these systems, for extended decay heat removal, is described in Subsection 6.3.1.1.1.

The radiological consequences of the accidents in this section are bounded by the radiological consequences of a main steam line break (see Subsection 15.1.5).

15.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

There are no steam pressure regulators in the AP1000 whose failure or malfunction causes a steam flow transient.

15.2.2 Loss of External Electrical Load

15.2.2.1 Identification of Causes and Accident Description

A major load loss on the plant can result from a loss of electrical load due to an electrical system disturbance. The ac power remains available to operate plant components such as the reactor coolant pumps; as a result, the standby onsite diesel generators do not function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves occurs. The automatic turbine bypass system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system function properly. If the condenser is not available, the excess steam generation is relieved to the atmosphere. Additionally, main feedwater flow is lost if the condenser is not available. For this transient, feedwater flow is maintained by the startup feedwater system.

For a loss of electrical load without subsequent turbine trip, no direct reactor trip signal is generated. The plant trips from the protection and safety monitoring system if a safety limit is approached. A continued steam load of approximately 5 percent exists after total loss of external electrical load because of the steam demand of plant auxiliaries.

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If a safety limit is approached, protection is provided by High-2 pressurizer pressure, High-3 pressurizer water level, and overtemperature ΔT trips. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of external electrical load, the maximum turbine overspeed is not expected to affect the voltage and frequency sensors. Any increased frequency to the reactor coolant pump motors results in a slightly increased flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine-generator overspeed, an overfrequency condition is not seen by the protection and safety monitoring system equipment or other safety-related loads. Safety-related loads and the protection and safety monitoring system equipment are supplied from the 120-Vac instrument power supply system, which in turn is supplied from the inverters. The inverters are supplied from a dc bus energized from batteries or by a regulated ac voltage.

If the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the High-2 pressurizer pressure signal, the High-3 pressurizer water level signal, or the overtemperature ΔT signal. This would cause steam generator shell side pressure and reactor coolant temperature to increase rapidly. However, the pressurizer safety valves and steam generator safety valves are sized to protect the reactor coolant system and steam generator against overpressure for load losses, without assuming the operation of the turbine bypass system, pressurizer spray, or automatic rod cluster control assembly control.

The steam generator safety valve capacity is sized to remove the steam flow at the nuclear steam supply system thermal rating from the steam generator, without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized to accommodate a complete loss of heat sink, with the plant initially operating at the maximum turbine load. The pressurizer safety valves can then relieve sufficient steam to maintain the reactor coolant system pressure within 110 percent of the reactor coolant system design pressure.

A discussion of overpressure protection can be found in WCAP-7769, Revision 1 (Reference 1) and WCAP-16779 (Reference 9).

A loss-of-external-load event is classified as a Condition II event, fault of moderate frequency.

A loss-of-external-load event results in a plant transient that is bounded by the turbine trip event analyzed in Subsection 15.2.3. Therefore, a detailed transient analysis is not presented for the loss-of-external-load event.

The primary side transient is caused by a decrease in heat transfer capability, from primary to secondary, due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow (should feedwater flow not be reduced, a larger heat sink is available and the transient is less severe). Reduction of steam flow to the turbine following a loss-of-external load event occurs due to automatic fast closure of the turbine control valves. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure, which occurs in approximately 0.15 seconds. The transient in primary pressure, temperature, and water volume is less severe for the loss-of-external-load event than for the turbine trip due to a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss-of-external-load event is the same as that for a turbine trip, as listed in Table 15.0-6.

15.2.2.2 Analysis of Effects and Consequences

Refer to Subsection 15.2.3.2 for the method used to analyze the limiting transient (turbine trip) in this grouping of events. The results of the turbine trip event analysis bound those expected for the loss-of-external-load event, as discussed in Subsection 15.2.2.1.

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Plant systems and equipment that may be required to function in order to mitigate the effects of a complete loss of load are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

The protection and safety monitoring system may be required to terminate core heat input and to prevent departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may open to maintain system pressures below allowable limits. No single active failure prevents operation of any system required to function. Normal plant control systems and engineered safety systems are not required to function. The passive residual heat removal (PRHR) system may be automatically actuated following a loss of main feedwater, further mitigating the effects of the transient.

15.2.2.3 Conclusions

Based on results obtained for the turbine trip event and considerations described in Subsection 15.2.2.1, the applicable Standard Review Plan, Subsection 15.2.1, evaluation criteria for a loss-of-external-load event, are met (see Subsection 15.2.3).

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

The turbine stop valves close rapidly (about 0.15 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

- Generator trip
- Low condenser vacuum
- Loss of lubricating oil
- Turbine thrust bearing failure
- Turbine overspeed
- Manual trip
- Reactor trip

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. The decrease in turbine first-stage impulse pressure when in T_{avg} mode, or an increase in steam header pressure when in pressure control mode, causes the steam dump controller to initiate turbine bypass. The loss of steam flow results in a rapid increase in secondary system temperature and pressure, with a resultant primary system transient, described in Subsection 15.2.2.1, for the loss-of-external-load event. A slightly more severe transient occurs for the turbine trip event due to the rapid loss of steam flow caused by the abrupt valve closure.

The automatic turbine bypass system accommodates up to 40 percent of rated main steam flow. Reactor coolant temperatures and pressure do not increase significantly if the turbine bypass system and pressurizer pressure control system are functioning properly. If the condenser is not available, the excess steam generation is relieved to the atmosphere and main feedwater flow is lost. For this situation, feedwater flow is maintained by the startup feedwater system to provide adequate residual and decay heat removal capability. Should the turbine bypass system fail to operate, the steam generator safety valves may lift to provide pressure control. See Subsection 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as a Condition II event, fault of moderate frequency.

A turbine trip is a more limiting than a loss-of-external-load event, loss of condenser vacuum, and other events which result in a turbine trip. As such, this event is analyzed and presented in Subsection 15.2.3.2.

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15.2.3.2 Analysis of Effects and Consequences

15.2.3.2.1 Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 100 percent of full power, without rapid power reduction, primarily to show the adequacy of the pressure-relieving devices, and to demonstrate core protection margins. The turbine is assumed to trip without actuating the rapid power reduction system. This assumption delays reactor trip until conditions in the reactor coolant system result in a trip due to other signals. Thus, the analysis assumes a bounding transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for startup feedwater or the PRHR heat exchanger (except for long-term recovery) to mitigate the consequences of the transient.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, analyses are performed to evaluate the effects produced by a possible consequential loss of offsite power during a complete loss of steam load. As discussed in Subsection 15.0.14, the loss of offsite power is considered as a direct consequence of a turbine trip occurring while the plant is operating at power. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

The turbine trip transients are analyzed by using a modified version of the LOFTRAN code (Reference 2), as described in Reference 6. The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables, including temperatures, pressures, and power level.

In the turbine trip analyses, which include a primary coolant flow coastdown caused by a consequential loss of offsite power, a combination of three computer codes is used to perform the departure from nucleate boiling ratio (DNBR) analyses. First, the LOFTRAN code (References 2 and 6) is used to calculate the plant system transient. The FACTRAN code (Reference 7) or the VIPRE-01 fuel rod model (Reference 8), which is equivalent to FACTRAN, is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN (or VIPRE-01 fuel rod model) and flow from LOFTRAN.

The major assumptions used in the analysis are summarized below.

Initial Operating Conditions

Two sets of initial operating conditions are used. Cases performed to evaluate the minimum DNBR obtained are analyzed using the revised thermal design procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 5). Instrument bias on the RCS temperature signal is also considered to ensure it is reflected in either the modeled initial conditions or in the safety analysis DNBR limit value.

Cases performed to evaluate the maximum calculated RCS pressure include uncertainties on the initial conditions. Initial core power, reactor coolant temperature, and pressure are assumed to be at the nominal full-power values plus or minus uncertainties. The direction of the uncertainties is chosen to maximize the RCS pressure.

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Reactivity Coefficients

Two cases are analyzed:

- Minimum reactivity feedback – A least-negative moderator temperature coefficient and a least-negative Doppler-only power coefficient are assumed (see Figure 15.0.4-1).

- Maximum reactivity feedback – A conservatively large negative moderator temperature coefficient and a most-negative Doppler-only power coefficient are assumed (see Figure 15.0.4-1).

Rod Control

From the standpoint of the maximum RCS pressure and minimum DNBR attained, it is conservative to assume that the reactor is in manual rod control. If the reactor is in automatic rod control, the control rod banks move prior to trip and reduce the severity of the transient.

Steam Release

No credit is taken for the operation of the turbine bypass system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

Pressurizer Spray

Two cases for both the minimum and maximum reactivity feedback cases are analyzed:

- Full credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are also available. These cases are analyzed primarily to address DNBR concerns.

- No credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are operable. These cases are analyzed to address RCS overpressure concerns.

Feedwater Flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for startup feedwater flow or the PRHR heat exchanger, because a stabilized plant condition is reached before initiation of the startup feedwater or the PRHR heat exchanger is normally assumed to occur. The startup feedwater flow or PRHR heat exchanger removes core decay heat following plant stabilization.

Reactor Trip

Reactor trip is actuated by the first reactor trip setpoint reached, with no credit taken for the rapid power reduction on the turbine trip. Trip signals are expected due to High-2 pressurizer pressure, overtemperature ΔT, Low-2 RCP speed, High-3 pressurizer water level, or Low-2 steam generator narrow range water level.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3. Plant systems and equipment that may be required to function in order to mitigate the effects of a turbine trip event are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

The protection and safety monitoring system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure prevents operation of systems

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required to function. Cases are analyzed, both with and without the operation of pressurizer spray, to determine the worst case for presentation.

Availability of Offsite Power

Each case is analyzed with and without offsite power available. As discussed in Subsection 15.0.14, the loss of offsite power is considered to be a consequence of an event due to disruption of the electrical grid following a turbine trip during the event. The grid is assumed to remain stable for 3 seconds following the turbine trip. In the analysis for the complete loss of steam load, the event is initiated by a turbine trip. Therefore, offsite power is assumed to be lost 3 seconds after the start of the event. For the loss of steam load analysis, the primary impact of the loss of offsite power is a coastdown of the reactor coolant pumps.

Main Steam System Pressure

Additional cases are performed to evaluate the maximum Main Steam System (MSS) pressure, with initial condition uncertainties chosen to maximize MSS pressure. The additional cases include cases with and without offsite power available for minimum and maximum reactivity feedback.

15.2.3.2.2 Results

The transient responses for a turbine trip from 100 percent of full-power operation are shown for eight cases. The eight analysis cases are performed assuming minimum and maximum reactivity feedback, with and without credit for pressurizer spray, and with and without offsite power available. The results of the analyses are shown in Figures 15.2.3-1 through 15.2.3-26. The calculated sequence of events for the accident is shown in Table 15.2-1.

Minimum Reactivity Feedback, With Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-1 through 15.2.3-7 show the transient responses for two cases analyzed for DNBR concerns, with and without offsite power available. In the case with offsite power available, the reactor is tripped by the overtemperature ΔT trip function. The transient DNBR is shown in Figure 15.2.3-6; the minimum DNBR remains above the safety analysis DNBR limit value at all times. Based on this, the DNB design basis defined in Section 4.4 is met.

The case without offsite power is tripped by the Low-2 reactor coolant pump speed trip function. The minimum DNBR remains above the safety analysis DNBR limit value at all times, as shown in Figure 15.2.3-6; therefore, the DNBR design basis defined in Section 4.4 is met. This case is the limiting case with respect to the DNBR margin of the turbine trip cases.

Maximum Reactivity Feedback, With Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-8 through 15.2.3-14 show the transient responses for the other two cases analyzed for DNBR concerns, with and without offsite power available. In the case with offsite power available, the reactor is tripped by the overtemperature ΔT trip function. The transient DNBR is shown in Figure 15.2.3-13; the minimum DNBR remains above the safety analysis DNBR limit value at all times. Based on this, the DNBR design basis defined in Section 4.4 is met for this case.

The case without offsite power is tripped by the Low-2 reactor coolant pump speed trip function. The DNBR transient is similar to, and bounded by, the minimum feedback case with pressurizer spray and without offsite power discussed above. The minimum DNBR remains above the safety analysis DNBR limit value at all times, as shown in Figure 15.2.3-13; therefore, the DNBR design basis defined in Section 4.4 is met.

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Minimum Reactivity Feedback, Without Pressurizer Spray, With and Without Offsite Power Available

The results for these cases analyzed to address RCS pressure concerns are shown in Figures 15.2.3-15 through 15.2.3-20. In the case with offsite power available, the reactor is tripped by the High-2 pressurizer pressure trip function. The pressurizer safety valves are actuated in this case and maintain the reactor coolant system pressure below 110 percent of the design value.

If offsite power is lost, the reactor is tripped by the Low-2 reactor coolant pump speed reactor trip function. Offsite power is assumed to be lost 3 seconds after turbine trip. This causes a reduction in the reactor coolant system flow, which is illustrated in Figure 15.2.3-20.

The pressurizer safety valves actuate in both of these cases and maintain the reactor coolant system pressure below 110 percent of the design value. RCS pressure for these cases is shown in Figure 15.2.3-16. Note that the with and without power cases have different assumptions regarding initial pressure. The initial pressure assumptions were based upon sensitivities that were run. With respect to maximum reactor coolant system pressure, the case with offsite power available is the most limiting for turbine trip cases.

Maximum Reactivity Feedback, Without Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-21 through 15.2.3-26 show the transient responses for the two other cases analyzed to address RCS pressure concerns, with and without offsite power available. In the case with offsite power available, the reactor is tripped by the High-2 pressurizer pressure function.

The case without offsite power is tripped by the Low-2 reactor coolant pump speed trip function. RCS pressure for both cases is shown in Figure 15.2.3-22. The pressure within the reactor coolant system is maintained below 110 percent of the design value. Note that the with and without power cases have different assumptions regarding initial pressure. The initial pressure assumptions were based upon sensitivities that were run.

The additional cases performed to address maximum MSS pressure concerns confirm that the steam generator safety valves provide sufficient pressure relief to prevent overpressurization of the MSS.

15.2.3.3 Conclusions

Results of the analyses show that a turbine trip presents no challenge to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The analyses show that the predicted DNBR is greater than the safety analysis DNBR limit value at any time during the transient. Thus, the departure from nucleate boiling design basis, as described in Section 4.4, is met.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves results in a turbine trip with no credit taken for the turbine bypass system. Turbine trips are discussed in Subsection 15.2.3.

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Subsection 15.2.3. A loss of condenser vacuum prevents the use of steam dump to the condenser. Because steam dump is assumed to be unavailable in the turbine trip

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analysis, no additional adverse effects result if the turbine trip is caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Subsection 15.2.3 apply to the loss of the condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, listed in Subsection 15.2.3.1, are covered by Subsection 15.2.3. Possible overfrequency effects, due to a turbine overspeed condition, are discussed in Subsection 15.2.2.1 and are not a concern for this type of event.

15.2.6 Loss of ac Power to the Plant Auxiliaries

15.2.6.1 Identification of Causes and Accident Description

The loss of power to the plant auxiliaries is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The onsite standby ac power system remains available but is not credited to mitigate the accident.

From the decay heat removal point of view, in the long term this transient is more severe than the turbine trip event analyzed in Subsection 15.2.3 because, for this case, the decrease in heat removal by the secondary system is accompanied by a reactor coolant flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip:

- Upon reaching one of the trip setpoints in the primary or secondary systems as a result of the flow coastdown and decrease in secondary heat removal.
- Due to the loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the sequence described below occurs:

- Plant vital instruments are supplied from the Class 1E and uninterruptable power supply.
- As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for turbine bypass. If the steam flow rate through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.
- The onsite standby power system, if available, supplies ac power to the selected plant non-safety loads.
- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition if the startup feedwater is available to supply water to the steam generators.
- If startup feedwater is not available, the PRHR heat exchanger is actuated.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system if available, which is started automatically when low levels occur in either steam generator. If that system is not available, emergency core decay heat removal is provided by the PRHR heat exchanger. The PRHR heat exchanger is a C-tube heat exchanger connected, through inlet and outlet headers, to the reactor coolant system. The inlet to the heat exchanger is from the reactor coolant system hot leg, and the return is to the steam generator outlet plenum. The heat exchanger is located above the core to provide natural circulation flow when the reactor coolant pumps are not operating. The IRWST provides the heat sink for the heat exchanger. The PRHR heat exchanger, in

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conjunction with the passive containment cooling system, provides core cooling and maintains reactor coolant system conditions to satisfy the evaluation criteria. After the IRWST water reaches saturation, steam starts to vent to the containment atmosphere. The condensation that collects on the containment steel shell (cooled by the passive containment cooling system) returns to the IRWST, maintaining fluid level for the PRHR heat exchanger heat sink. The analysis shows that the natural circulation flow in the reactor coolant system following a loss of ac power event is sufficient to remove residual heat from the core.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant and PRHR loops.

A loss of ac power to the plant auxiliaries is a Condition II event, a fault of moderate frequency. This event is more limiting with respect to long-term heat removal than the turbine trip initiated decrease in secondary heat removal without loss of ac power, which is discussed in Subsection 15.2.3. A loss of offsite power to the plant auxiliaries will also result in a loss of normal feedwater.

The plant systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.2.6.2 Analysis of Effects and Consequences

15.2.6.2.1 Method of Analysis

The analysis is performed to demonstrate the adequacy of the protection and safety monitoring system, the PRHR heat exchanger, and the reactor coolant system natural circulation capability in removing long-term (approximately 36,000 seconds) decay heat. This analysis also demonstrates the adequacy of these systems in preventing excessive heatup of the reactor coolant system with possible reactor coolant system overpressurization or loss of reactor coolant system water.

A modified version of the LOFTRAN code (Reference 2), described in WCAP-15644 (Reference 6), is used to simulate the system transient following a plant loss of offsite power. The simulation describes the plant neutron kinetics and reactor coolant system, including the natural circulation, pressurizer, and steam generator system responses. The digital program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The assumptions used in this analysis minimize the energy removal capability of the PRHR heat exchanger and maximize the coolant system expansion.

The assumptions used in the analysis are as follows:

- The plant is initially operating at 101 percent of the design power rating with initial reactor coolant temperature 8°F below the nominal value and the pressurizer pressure 50 psi below the nominal value.
- Core residual heat generation is based on ANSI 5.1 (Reference 3). ANSI 5.1 is a conservative representation of the decay energy release rates.
- Reactor trip occurs on RCP Low-2 speed.
- A heat transfer coefficient is assumed in the steam generator associated with reactor coolant system natural circulation flow conditions following the reactor coolant pump coastdown.

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- The PRHR heat exchanger is actuated by the Low-2 steam generator narrow range water level coincident with Low-2 start up feed water flow coincident with two-out-of-four T_{cold} - High in either loop.

- For the loss of ac power to the station auxiliaries and following reactor trip, the main safety function required is core decay heat removal. That is accomplished by the secondary steam relief through the steam generator safety valves and the PRHR heat exchanger. One of two parallel valves in the PRHR outlet line is assumed to fail to open. This is the worst single failure.

- The pressurizer safety valves are assumed to function.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Plant systems and equipment necessary to mitigate the effects of a loss of ac power to the station auxiliaries are discussed in Subsection 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The protection and safety monitoring system is required to function following a loss of ac power. The PRHR heat exchanger is required to function with an overall minimum capability to extract heat from the reactor coolant system. No single active failure prevents operation of any system required to function.

Parameters used in the analysis are selected to maximize the pressurizer water volume. Input parameters are not selected to maximize the transient primary side and secondary side pressure. Transient primary side and secondary side pressures during a loss of ac power to station auxiliaries are bounded by those calculated for the turbine trip analyses presented in Subsection 15.2.3.

With respect to DNB concerns, the loss of ac power to station auxiliaries event is bounded by the loss of ac power case analyzed for the turbine trip event presented in Subsection 15.2.3. The consequences of a loss of ac power to station auxiliaries that results in CMT actuation are bounded by the analysis of the Inadvertent Operation of the Core Makeup Tanks During Power Operation event described in Subsection 15.5.1.

15.2.6.2.2 Results

The transient response of the reactor coolant system following a loss of ac power to the plant auxiliaries is shown in Figures 15.2.6-1 through 15.2.6-12. The calculated sequence of events for this event is listed in Table 15.2-1.

The loss of ac power event results in a pressurizer water volume increase until the actuation of the steam generator safety valves. Actuation of the steam generator safety valves attenuates the pressurizer water volume until actuation of the PRHR which turns around the pressurizer water volume increase. PRHR heat extraction and steam generator safety valve relief results in a consequential decrease in the water volume until the safety valve relief stops. After the steam generator safety valve flow stops the pressurizer water volume begins a slight increase until the PRHR heat extraction matches and then exceeds the decay heat addition resulting in a reduction in the pressurizer water volume.

15.2.6.3 Conclusions

Results of the analysis show that for the loss of ac power to plant auxiliaries event, all safety criteria are met. The heat extraction provided by the steam relief capacity of the steam generator safety valves and the operation of the PRHR is sufficient to prevent water relief through the pressurizer safety valves.

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The analysis demonstrates that sufficient long-term reactor coolant system heat removal capability exists, via the steam generator safety valves, natural circulation and the PRHR heat exchanger, following reactor coolant pump coastdown to prevent fuel or cladding damage and reactor coolant system overpressure.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of ac power sources) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If startup feedwater is not available, the safety-related PRHR heat exchanger is automatically aligned by the protection and safety monitoring system to remove decay heat.

A small secondary system break can affect normal feedwater flow control, causing low steam generator levels prior to protective actions for the break. This scenario is addressed by the assumptions made for the feedwater system pipe break (see Subsection 15.2.8).

The following occurs upon loss of normal feedwater (assuming main feedwater pump fails or valve malfunctions):

- The steam generator water inventory decreases as a consequence of the continuous steam supply to the turbine. The mismatch between the steam flow to the turbine and the feedwater flow leads to the reactor trip on a Low-2 steam generator narrow range water level signal. The same signal also actuates the startup feedwater system (see Subsection 15.2.6.1).

- As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. The condenser is assumed to be unavailable for turbine bypass. If the steam flow path through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the decay heat and to maintain the plant at the hot shutdown condition, if the startup feedwater is used to supply water to the steam generator.

- If startup feedwater is not available, the PRHR heat exchanger is actuated on either a Low-2 steam generator narrow range water level, coincident with a Low-2 startup feedwater flow rate and coincident with T_{cold} - High signal or a Low-2 steam generator wide range water level signal.

- The PRHR heat exchanger extracts heat from the reactor coolant system and can lead to an “S” signal on a Low-2 T_{cold} signal. This would actuate the core makeup tanks. Note that in order to confirm the capability of the PRHR heat exchanger to remove decay heat, CMT actuation is not modeled. CMT actuation results in a significant primary side cooldown, and transitions the event from a decrease in heat removal event to an inventory increase event. The consequences of a loss of normal feedwater flow event that results in CMT actuation are bounded by the analysis of the Inadvertent Operation of the Core Makeup Tanks During Power Operation event described in Subsection 15.5.1.

A loss-of-normal-feedwater event is classified as a Condition II event, a fault of moderate frequency.

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15.2.7.2 Analysis of Effects and Consequences

The analysis is performed to demonstrate the adequacy of the protection and safety monitoring system and the capability of the PRHR heat exchanger in removing long-term (approximately 36,000 seconds) decay heat following a loss of normal feedwater. This analysis also demonstrates the adequacy of these systems in preventing excessive heatup of the reactor coolant system with possible reactor coolant system overpressurization or loss of reactor coolant system water.

15.2.7.2.1 Method of Analysis

An analysis using a modified version of the LOFTRAN code (Reference 2), described in WCAP-15644 (Reference 6), is performed to obtain the plant transient following a loss of normal feedwater. The simulation describes the neutron kinetics, reactor coolant system (including the natural circulation), pressurizer, and steam generators. The program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Two cases are analyzed. One case assumes a consequential loss of ac power to the plant auxiliaries resulting from the turbine trip after reactor trip. The loss of ac power results in a coast down of the reactor coolant pumps. A second case does not assume the consequential loss of ac power, which maintains the reactor coolant pumps at normal speed until automatically tripped when an “S” signal is generated.

The assumptions used in the analysis are as follows:

- The plant is initially operating at 101 percent of the design power rating.
- Reactor trip occurs on Low-2 steam generator narrow range water level.
- The principle safety function required after reactor trip is the core decay heat removal. That function is carried out by the PRHR heat exchanger. The worst single failure is assumed to occur in the PRHR heat exchanger. The actuation of the PRHR heat exchanger requires the opening of one of the two fail-open valves arranged in parallel at the PRHR heat exchanger discharge. Because no single failure can be assumed that impairs the opening of both valves, the failure of a single valve is assumed.

  The PRHR heat exchanger is actuated by the Low-2 steam generator narrow range water level signal, coincident with Low-2 start up feedwater flow and coincident with High T_{cold} or by the Low-2 steam generator wide range water level signal.
- Secondary system steam relief is achieved through the steam generator safety valves.
- The initial reactor coolant average temperature is 8°F lower than the nominal value, and initial pressurizer pressure is 50 psi lower than nominal for the case with ac power available. The initial reactor coolant average temperature is 8°F higher than the nominal value, and initial pressurizer pressure is 50 psi higher than nominal for the case without ac power available.

The loss of normal feedwater analyses are performed to demonstrate the adequacy of the protection and safety monitoring system and the PRHR heat exchanger in removing long-term decay heat. Such decay heat removal prevents excessive heatup of the reactor coolant system with possible resultant reactor coolant system overpressurization or loss of reactor coolant system water. The assumptions used in this analysis minimize the energy removal capability of the system, and maximize the coolant system expansion.

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With respect to the overpressure evaluation, the loss of normal feedwater transient with and without ac power available events are bounded by the turbine trip event.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Plant systems and equipment necessary to mitigate the effects of a loss of normal feedwater accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The protection and safety monitoring system is required to function following a loss of normal feedwater. The PRHR heat exchanger is required to function with an overall minimum capability to extract heat from the reactor coolant system. No single active failure prevents operation of any system to perform its required function.

15.2.7.2.2 Results

Figures 15.2.7-1 through 15.2.7-13 show the significant plant parameters following a loss of normal feedwater with ac power available.

The loss of main feedwater with ac power available results in an increase in the pressurizer water volume until reactor trip on Low-2 steam generator narrow range water level. The pressurizer water volume then decreases briefly due to the reactor trip. However, due to the loss of secondary heat removal capacity the pressurizer water volume begins to increase. The pressurizer water volume then increases until the PRHR actuates.

The capacity of the PRHR heat exchanger, when the reactor coolant pumps are operating, is much larger than the decay heat, and in the first part of the transient, the reactor coolant system is cooled down. The cool down continues until the reactor coolant temperature reaches the Low-2 T_{cold} setpoint. When the Low-2 T_{cold} setpoint is reached, the reactor coolant pumps are tripped and the core makeup tanks would be actuated. However, in order to confirm the capability of the PRHR heat exchanger to remove long term decay heat, CMT actuation is not modeled.

As a result of the RCP trip, the PRHR heat exchanger heat removal no longer exceeds decay heat, and a second heatup phase occurs, causing pressurizer water volume to begin to increase. This increase continues until the PRHR heat extraction again exceeds the magnitude of the decay heat addition.

The DNBR transient for the loss of normal feedwater event with ac power available is shown in Figure 15.2.7-12.

The calculated sequence of events for this accident is listed in Table 15.2-1.

Figures 15.2.7-14 through 15.2.7-26 show the significant plant parameters following a loss of normal feedwater with a consequential loss of ac power to plant auxiliaries.

The first increase in pressurizer water volume is turned around by the reactor trip on low-2 steam generator narrow range water level, with subsequent increase attenuated by the heat extraction provided by the steam generator safety valves. Due to the steam generator safety valve relief, the pressurizer water volume remains relatively constant until the heat extraction provided by the steam generator safety valves relief stops once steam generator dryout occurs. With no steam generator safety valve relief, the pressurizer water volume begins to increase until the PRHR heat extraction matches and then exceeds the decay heat addition.

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15.2.7.3 Conclusions

Results of the analyses show that a loss of normal feedwater or a loss of normal feedwater with a consequential loss of ac power to the plant auxiliaries do not adversely affect the core, the reactor coolant system, or the steam system. The heat removal capacity of the PRHR heat exchanger and the steam generator safety valves is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values. If CMT actuation were to occur, an operator action may be required to mitigate the event, consistent with the operator action presented in Subsection 15.5.1.

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture is a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators in order to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. (A break upstream of the feedwater line check valve would affect the plant only as a loss of feedwater. This case is covered by the evaluation in Subsections 15.2.6 and 15.2.7.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in Subsection 15.1.5. Therefore, only the reactor coolant system heatup effects are evaluated for a feedwater line rupture in this subsection.

The feedwater line rupture reduces the ability to remove heat generated by the core from the reactor coolant system for the following reasons:

- Feedwater flow to the steam generators is reduced. Because feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.
- Fluid in the steam generator may be discharged through the break and would not be available for decay heat removal after trip.
- The break may be large enough to prevent the addition of main feedwater after trip.

A major feedwater line rupture is classified as a Condition IV event.

The severity of the feedwater line rupture transient depends on a number of system parameters, including the break size, initial reactor power, and the functioning of various control and safety-related systems. Sensitivity studies presented in WCAP-9230 (Reference 4) illustrate that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. At the beginning of the transient, the main feedwater control system is assumed to malfunction due to an adverse environment. Interactions between the break and the main feedwater control system result in no feedwater flow being injected or lost through the steam generator feedwater nozzles. This assumption causes the water levels in both steam generators to decrease equally until the Low-2 steam generator narrow range water level reactor trip setpoint is reached. After reactor trip, a full double-ended rupture of the feedwater line is assumed such that the faulted steam generator blows down through the break and no main feedwater is delivered to the intact steam generator. These assumptions conservatively bound the most limiting feedwater line rupture that can occur. Analysis is performed at full power assuming the loss of offsite power at the time of the reactor trip. This is more

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conservative than the case where power is lost at the initiation of the event. The case with offsite power available is not explicitly examined because, due to the fast generation of an “S” signal (generated by the Low-2 steam line pressure), the reactor coolant pumps would be tripped by the protection and safety monitoring system shortly after the reactor trip. The only difference between the cases with and without offsite power available would be a small difference in when the reactor coolant pumps are tripped.

The following provides the protection for a main feedwater line rupture:

- A reactor trip on any of the following five conditions:
  - High-2 pressurizer pressure
  - Overtemperature ΔT
  - High-3 pressurizer water level
  - Low-2 steam generator narrow range water level in either steam generator
  - “S” signals from either of the following:
    - Two out of four Low-2 steam line pressure in either steam generator
    - Two out of four high containment pressure (high-2)

Refer to Sections 7.1 and 7.2 for a description of the actuation system.

The PRHR heat exchanger functions to:

- Provide a passive method for decay heat removal. The heat exchanger is a C-tube type, located inside the IRWST. The heat exchanger is above the reactor coolant system to provide natural circulation of the reactor coolant. Operation of the PRHR heat exchanger is initiated by the opening of one of the two parallel power-operated valves at the PRHR heat exchanger cold leg.

- Prevent substantial overpressurization of the reactor coolant system (less than 110 percent of design pressures).

- Maintain sufficient liquid in the reactor coolant system so that the core remains in place, and geometrically intact, with no loss of core cooling capability.

Refer to Subsection 6.3.2.2.5 for a description of the PRHR heat exchanger.

15.2.8.2 Analysis of Effects and Consequences

15.2.8.2.1 Method of Analysis

An analysis using a modified version, described in WCAP-15644 (Reference 6), of the LOFTRAN code (Reference 2) is performed to determine the plant transient following a feedwater line rupture. The code describes the reactor thermal kinetics, reactor coolant system (including natural circulation), pressurizer, steam generators, and feedwater system responses and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The case analyzed assumes a double-ended rupture of the largest feedwater pipe at full power. Major assumptions used in the analysis are as follows:

- The plant is initially operating at 101 percent of the design plant rating. The main feedwater flow measurement supports a 1-percent power uncertainty.

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- Initial reactor coolant average temperature is 8.0°F above the nominal value, and the initial pressurizer pressure is 50 psi below its nominal value.
- The pressurizer spray is modeled to function as designed. It is noted that following loss of offsite power and RCP coastdown, the pressurizer spray is not functional (when plant is in natural circulation mode).
- Initial pressurizer level is at a conservative maximum value and a conservative initial steam generator water level is assumed in both steam generators.
- At the start of the transient, interaction between the break in the feedline and the main feedwater control system is assumed to result in a complete loss of feedwater flow to both steam generators. No feedwater flow is delivered to or lost through the steam generator nozzles.
- Reactor trip is assumed to be initiated by the Low-2 steam generator narrow range water level signal on the ruptured steam generator. A two-second delay is assumed following the low level setpoint being reached to allow for the system response times.
- After reactor trip, the faulted steam generator blows down through a double-ended break area of 1.117 ft^2. A saturated liquid discharge is assumed until all the water inventory is discharged from the faulted steam generator. This minimizes the heat removal capability of the faulted steam generator and maximizes the resultant heatup of the reactor coolant. No feedwater flow is assumed to be delivered to the intact steam generator.
- The PRHR heat exchanger is assumed to be actuated by the Low-2 steam generator wide range water level signal. A 17-second delay is assumed following the low level setpoint being reached to allow for the system response times and the valve stroke time.
- Credit is taken for heat energy deposited in reactor coolant system metal during the reactor coolant system heatup.
- No credit is taken for charging or letdown.
- Pressurizer safety valve setpoint is assumed to be at its minimum value.
- Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases. The heat transfer remains approximately 100 percent in the faulted steam generator until the liquid mass reaches about 11 percent. The heat transfer is then reduced to 0 percent with the liquid inventory.
- Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip (Reference 3).
- No credit is taken for the following four protection and safety monitoring system reactor trip signals to mitigate the consequences of the accident:
  - High-2 pressurizer pressure
  - Overtemperature ΔT
  - High-3 pressurizer water level
  - High-2 containment pressure

The PRHR heat exchanger is initiated once the steam generator water level drops to the Low-2 steam generator wide range water level. Similarly, receipt of a Low-2 steam line pressure signal in at

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least one steam line initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives an “S” signal that initiates flow of cold borated water from the core makeup tanks to the reactor coolant system.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

No credit is taken for the plant control system to mitigate the consequences of the event. The protection and safety monitoring system is required to function following a feedwater line rupture as analyzed here. No single active failure prevents operation of this system.

The engineered safety features assumed to function are the PRHR heat exchanger, core makeup tank, and steam line isolation valves. The single failure assumed is the failure of one of the two parallel discharge valves in the PRHR outlet line (see Table 15.0-7).

A description and analysis of the core makeup tank is provided in Subsection 6.3.2.2.1. The PRHR heat exchanger is described in Subsection 6.3.2.2.5.

15.2.8.2.2 Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2.8-1 through 15.2.8-10. The calculated sequence of events for the case analyzed is listed in Table 15.2-1.

The results presented in Figures 15.2.8-5 and 15.2.8-7 show that pressures in the reactor coolant system and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure decreases after reactor trip on the Low-2 steam generator narrow range water level due to the loss of heat input.

In the first part of the transient, due to the conservative analysis assumptions, the system response following the feedwater line rupture is similar to the loss of ac power to the station auxiliaries (Subsection 15.2.6). Accordingly, like the loss of ac power event documented in Subsection 15.2.6, the feedwater line rupture event is bounded by the turbine trip event presented in Subsection 15.2.3 with respect to DNB concerns.

After the trip, the core makeup tanks are actuated on Low-2 steam line pressure in the ruptured loop while the PRHR heat exchanger is actuated on a Low-2 steam generator wide range water level.

The addition of the PRHR heat exchanger and flow from the core makeup tanks help to cool down the primary system and to provide sufficient fluid to keep the core covered with water.

Pressurizer safety valves open due to the mismatch between decay heat and the heat transfer capability of the PRHR heat exchanger. In the first part of the transient, there is a cooling effect due to the core makeup tanks that inject cold water into the reactor coolant system and receive hot water from the cold leg. This effect decreases due to the heatup of the core makeup tanks from recirculation flow. Also, the injection driving head is lowered as the core makeup tanks heat up.

Reactor coolant system temperatures are low (with the hot leg temperature of the faulted loop at approximately 510°F at about 2,200 seconds) and, in this condition, the PRHR heat exchanger cannot remove the entire decay heat load. Reactor coolant system temperatures then increase until approximately 20,000 seconds reflecting the heat removal limitations of the PRHR heat exchanger. At approximately 23,400 seconds, the heat transfer capability of the PRHR exceeds the decay heat power and the reactor coolant system pressure begins to decrease. Since subcooling is maintained throughout the transient and the reactor coolant system inventory increases (i.e., net core makeup tank injection exceeds net pressurizer safety valve relief), core cooling capability is maintained.

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15.2.8.3 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the capacity of the PRHR heat exchanger is adequate to remove decay heat, to prevent overpressurization of the reactor coolant system, and to maintain the core cooling capability. Radioactivity doses from postulated ruptures of the feedwater lines are less than those presented for the postulated main steam line break. The Standard Review Plan, Subsection 15.2.8, evaluation criteria are therefore met.

15.2.9 Combined License Information

This section contained no requirement for additional information.

15.2.10 References

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Table 15.2-1 (Sheet 1 of 8)
Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
I. Turbine trip		
A.1. With pressurizer spray, minimum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater	0.0
	Initiation of steam release from steam generator safety valves	10.6
	Minimum DNBR occurs	12.2
	Overtemperature ΔT reactor trip setpoint reached	18.9
	Rods begin to drop	20.9
A.2. With pressurizer spray, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low-2 reactor coolant pump speed reactor trip setpoint reached	3.5
	Rods begin to drop	4.1
	Minimum DNBR occurs	6.0
	Initiation of steam release from steam generator safety valves	14.9

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Table 15.2-1 (Sheet 2 of 8)
Time Sequence of Events for Incidents Which
Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
B.1. With pressurizer spray, maximum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	Minimum DNBR occurs	0.0
	Initiation of steam release from steam generator safety valves	10.7
	Overtemperature ΔT reactor trip setpoint reached	20.9
	Rod motion begins	22.9
B.2. With pressurizer spray, maximum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low-2 reactor coolant pump speed reactor trip setpoint reached	3.5
	Rods begin to drop	4.1
	Minimum DNBR occurs	4.9
	Initiation of steam release from steam generator safety valves	16.3

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Table 15.2-1 (Sheet 3 of 8)
Time Sequence of Events for Incidents Which
Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
C.1. Without pressurizer spray, minimum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	High-2 pressurizer pressure reactor trip point reached	5.1
	Rods begin to drop	7.1
	Initiation of steam release from steam generator safety valves	8.2
	Peak RCS pressure occurs	8.5
C.2. Without pressurizer spray, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low-2 reactor coolant pump speed reactor trip setpoint reached	3.5
	Rods begin to drop	4.1
	Peak RCS pressure occurs	6.1
	Initiation of steam release from steam generator safety valves	9.9

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Table 15.2-1 (Sheet 4 of 8)
Time Sequence of Events for Incidents Which
Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
D.1. Without pressurizer spray, maximum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	High-2 pressurizer pressure reactor trip	5.0
	Rods begin to drop	7.0
	Peak RCS pressure occurs	7.9
	Initiation of steam release from steam generator safety valves	8.2
D.2. Without pressurizer spray, maximum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low-2 reactor coolant pump speed reactor trip setpoint reached	3.5
	Rods begin to drop	4.1
	Peak RCS pressure occurs	5.8
	Initiation of steam release from steam generator safety valves	10.0

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Table 15.2-1 (Sheet 5 of 8)
Time Sequence of Events for Incidents Which
Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
II.A. Loss of ac power to the plant auxiliaries	Offsite ac power is lost, feedwater is lost, RCPs begin to coast down, turbine trip occurs (transient initiation)	10.0
	RCP Low-2 speed reactor trip set point is reached	10.5
	Rods begin to drop	11.2
	Pressurizer safety valves first open	~80.0
	Steam generator safety valves first open	99.2
	Low steam generator narrow range water level reactor trip setpoint reached	454.8
	PRHR heat exchanger actuation on Low-2 steam generator narrow range water level coincident with Low-2 start up flow rate coincident with T_{cold} - High	637.0
	Maximum long term pressurizer water volume reached	17,100
	PRHR heat exchanger heat removal matches decay heat	~17,700

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Table 15.2-1 (Sheet 6 of 8)
Time Sequence of Events for Incidents Which
Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
IIIA. Loss of normal feedwater flow	Feedwater is lost (transient initiation)	10.0
	Low-2 steam generator narrow range water level reactor trip reached	61.1
	Rods begin to drop	63.1
	Minimum DNBR is reached	64.0
	Steam generator safety valves first open	94.5
	PRHR heat exchanger actuation on Low-2 steam generator narrow range water level coincident with Low-2 start up feeedwater flow rate coincident with T_{cold} - High	243.1
	Cold leg temperature reaches Low-2 T_{cold} setpoint	1455.6
	Reactor coolant pump trip on Low-2 T_{cold} “S” signal	1455.6
	Steam line isolation on Low-2 T_{cold} “S” signal	1467.6
	Core makeup tank actuation on Low-2 T_{cold} “S” signal	1587.9
	Pressurizer safety valves first open	~2500
	PRHR heat exchanger heat removal matches decay heat addition	~15,500
	Maximum long term pressurizer water volume reached	16,500

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Table 15.2-1 (Sheet 7 of 8)
Time Sequence of Events for Incidents Which
Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
III.B. Loss of normal feedwater flow with a consequential loss of ac power	Feedwater is lost (transient initiation)	10.0
	Pressurizer safety valves first open	~40
	Low-2 steam generator narrow range water level reactor trip setpoint is reached	60.8
	Rods begin to drop	62.8
	Minimum DNBR is reached	63.5
	RCP trip due to loss of ac power	70.2
	Steam generator safety valves first open	70.8
	Low steam generator wide range water level reactor trip setpoint is reached	194.4
	PRHR heat exchanger actuation on Low-2 steam generator wide range water level	211.4
	PRHR heat exchanger heat removal matches decay heat addition	~1,900
	Maximum long term pressurizer water volume reached	2026

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Table 15.2-1 (Sheet 8 of 8)
Time Sequence of Events for Incidents Which
Result in a Decrease in Heat Removal By the Secondary System

Accident	Event	Time (seconds)
IV. Feedwater system pipe break	Main feedwater flow to both steam generators stops due to interaction between the break and the main feedwater control system (transient initiation)	10.0
	Low-2 steam generator narrow range water level setpoint reached	60.3
	Rods begin to drop	62.3
	Reverse flow from the faulted steam generator through a full double-ended rupture starts	62.3
	Loss of offsite power	70.3
	Low-2 steam line pressure setpoint is reached	76.6
	Core makeup tank valves fully opened	76.7
	Low-2 steam generator wide range water level setpoint reached	81.6
	All steam isolation valves close	88.6
	PRHR heat exchanger actuation on Low-2 steam generator wide range water level	98.6
	Faulted steam generator empties	122.0
	Intact steam generator safety valves open for the first time	239.6
	Pressurizer safety valves open for the first time	1,998
	PRHR heat exchanger extracted heat matches decay heat	23,400

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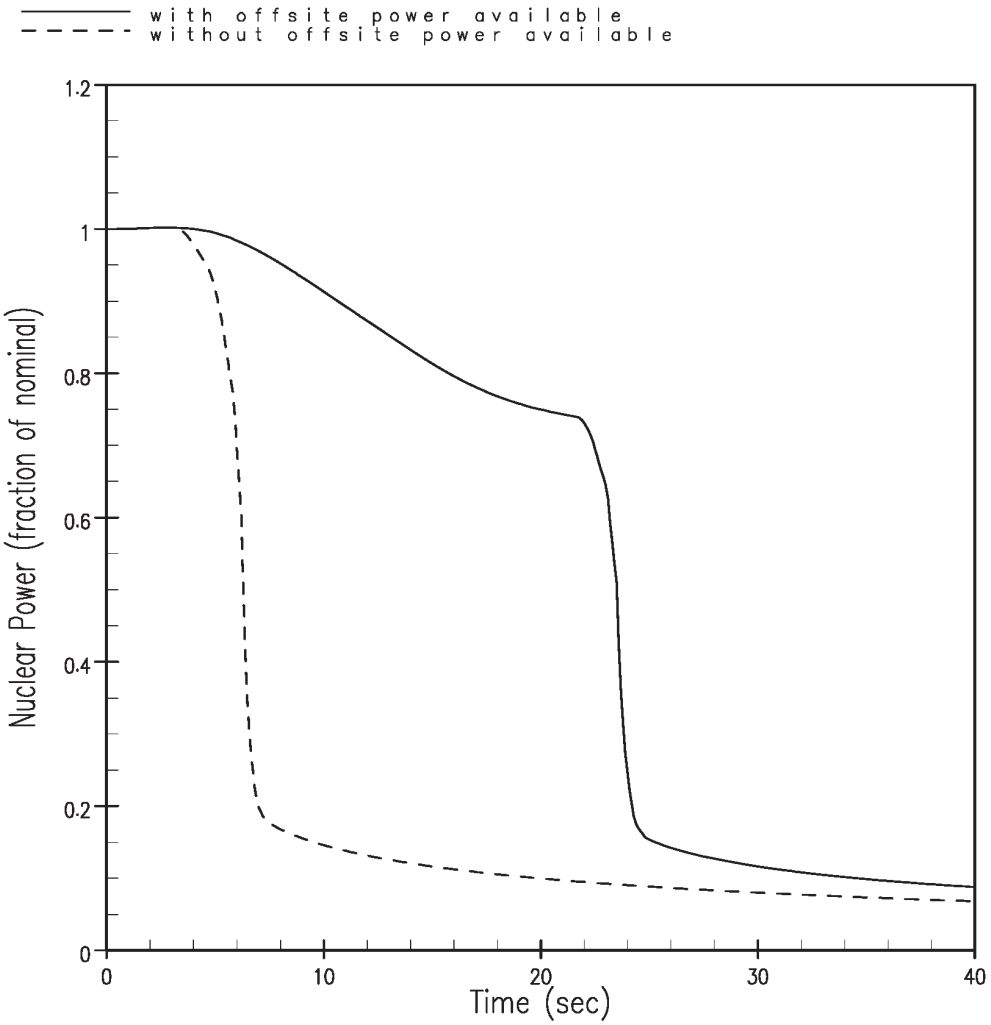

Figure 15.2.3-1
Nuclear Power versus Time for Turbine
Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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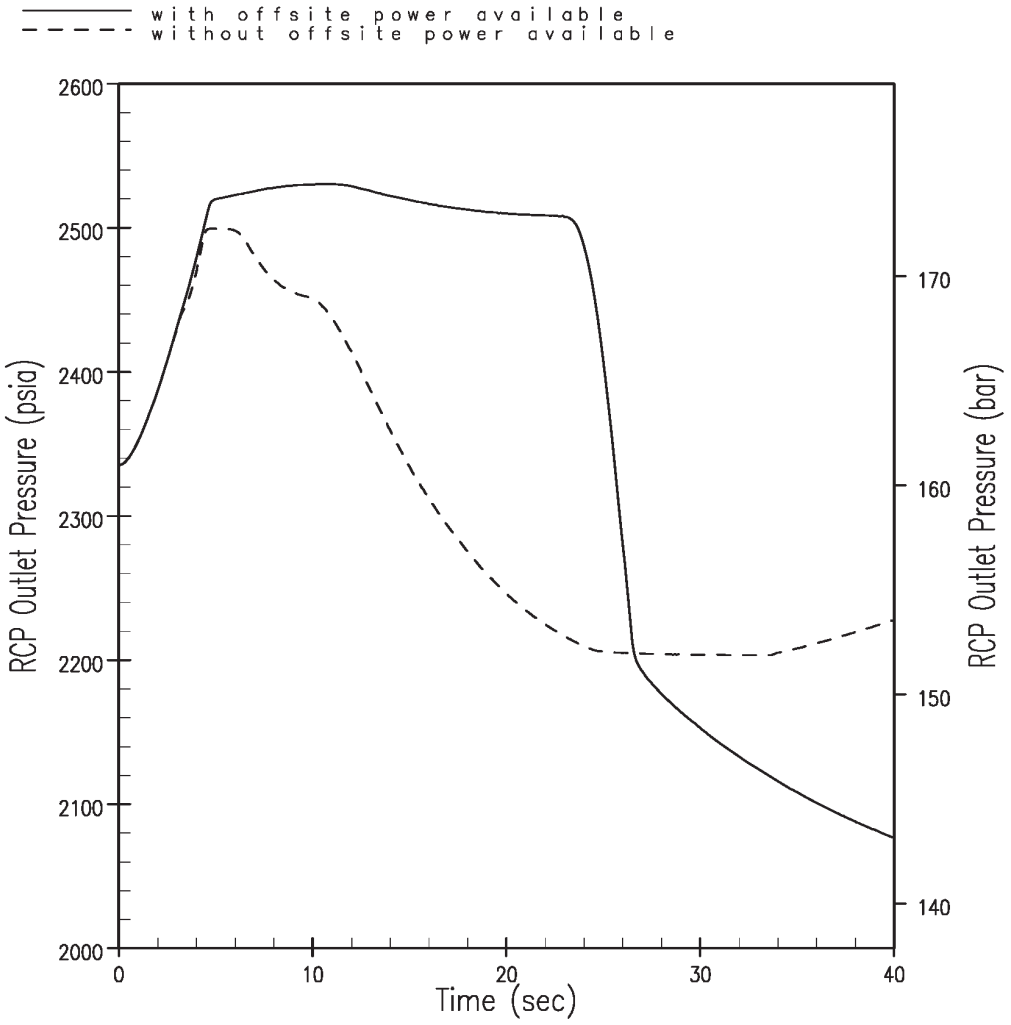

Figure 15.2.3-2
RCP Outlet Pressure versus Time for Turbine Trip
Accident with Pressurizer Spray and Minimum Moderator Feedback

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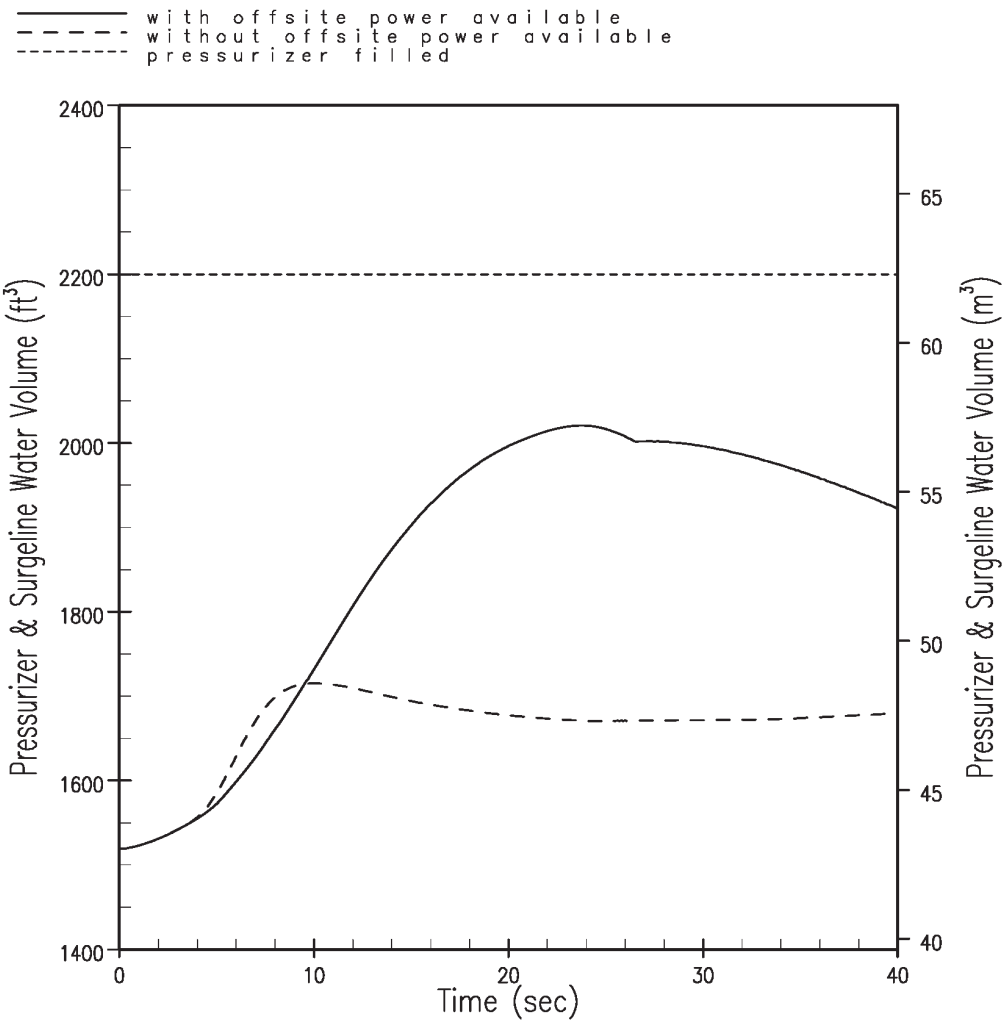

Figure 15.2.3-3
Pressurizer & Surge Line Water Volume versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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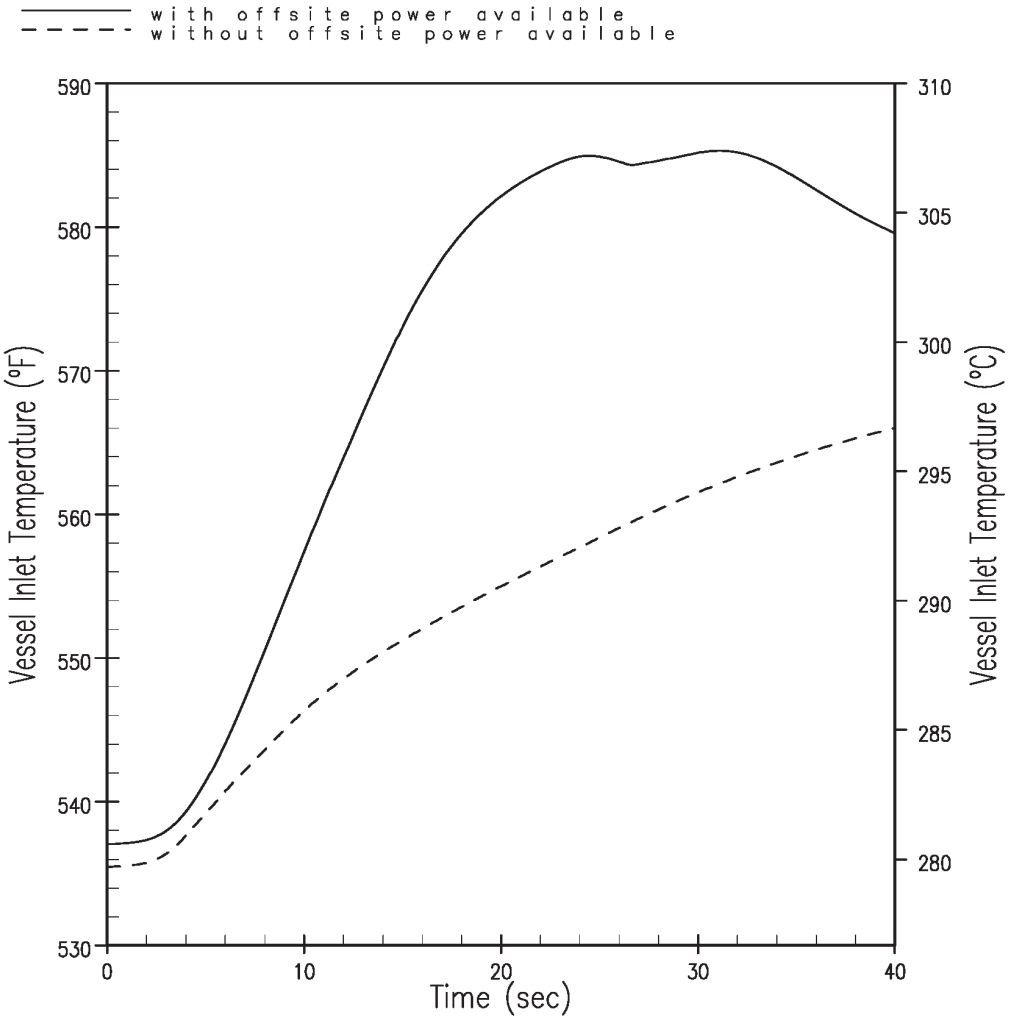

Figure 15.2.3-4
Vessel Inlet Temperature versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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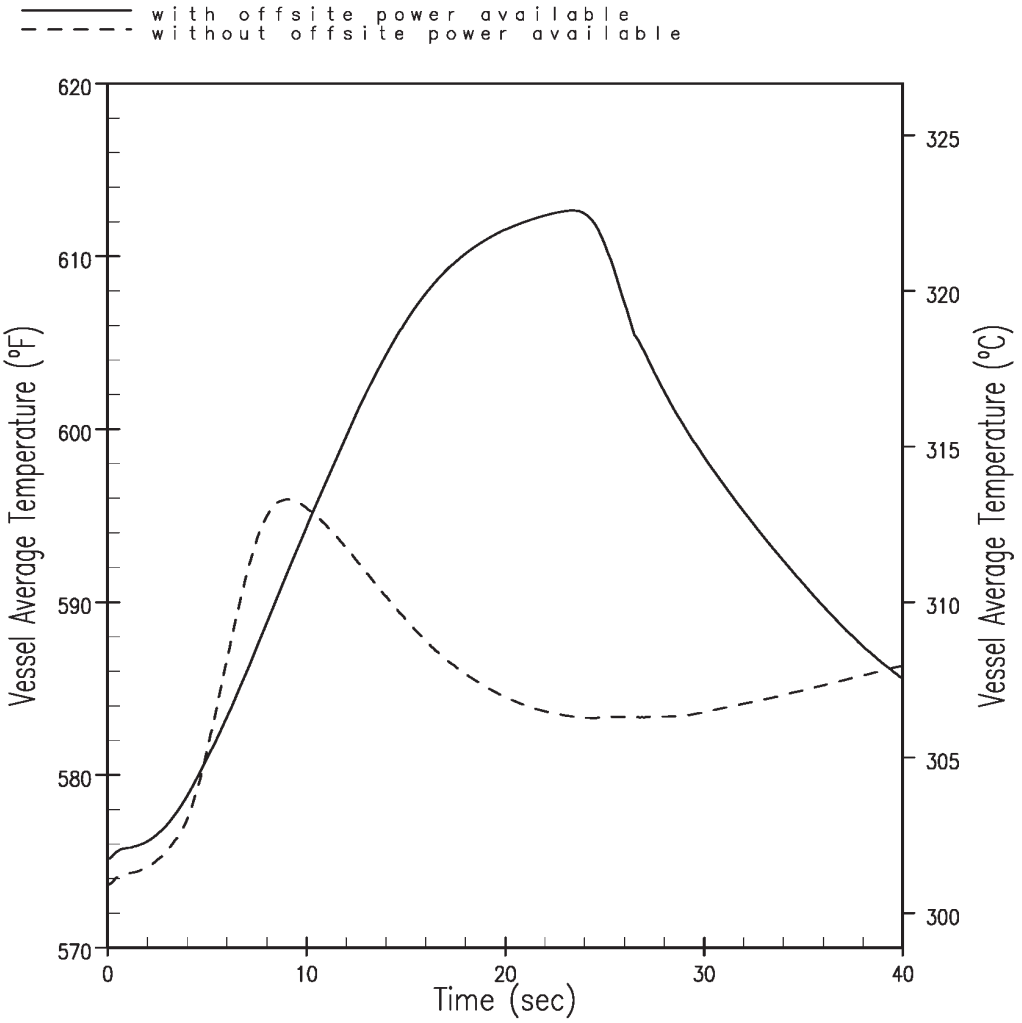

Figure 15.2.3-5
Vessel Average Temperature versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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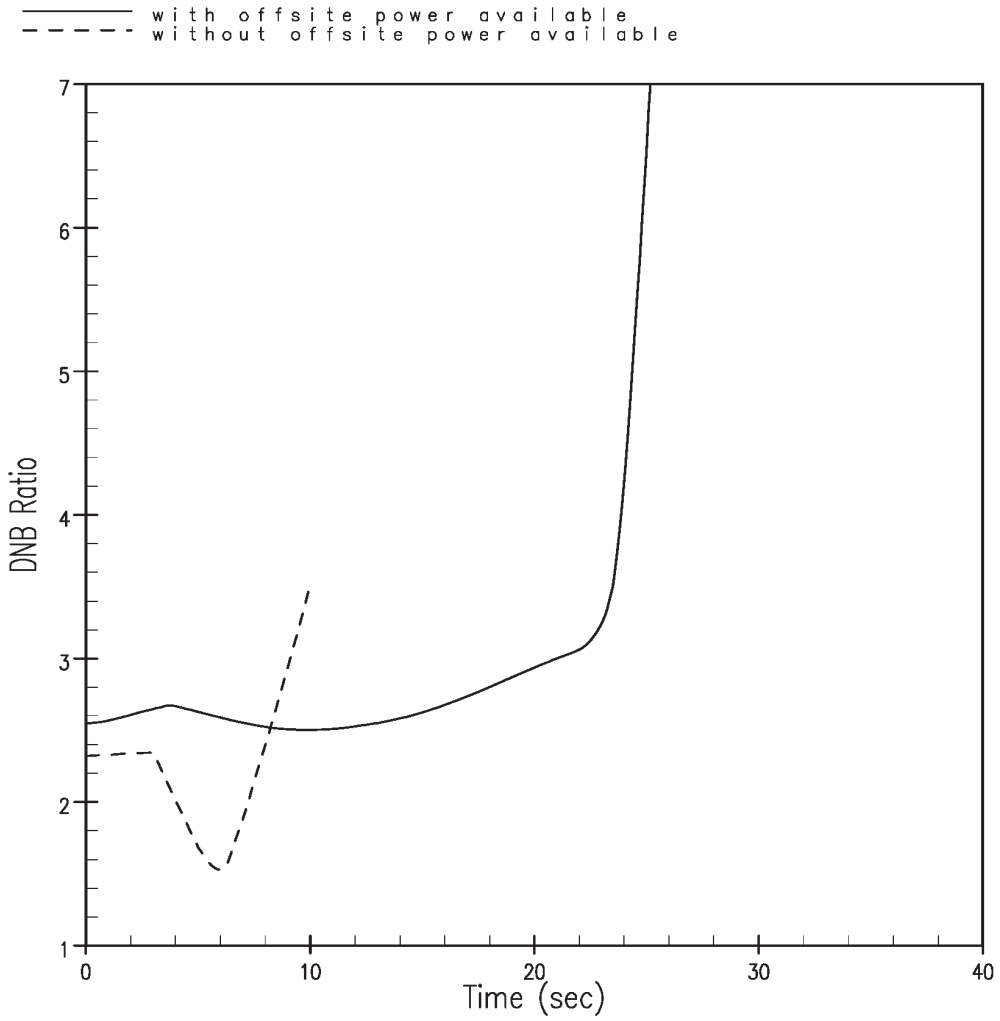

Figure 15.2.3-6
DNBR versus Time for Turbine Trip Accident
with Pressurizer Spray and Minimum Moderator Feedback

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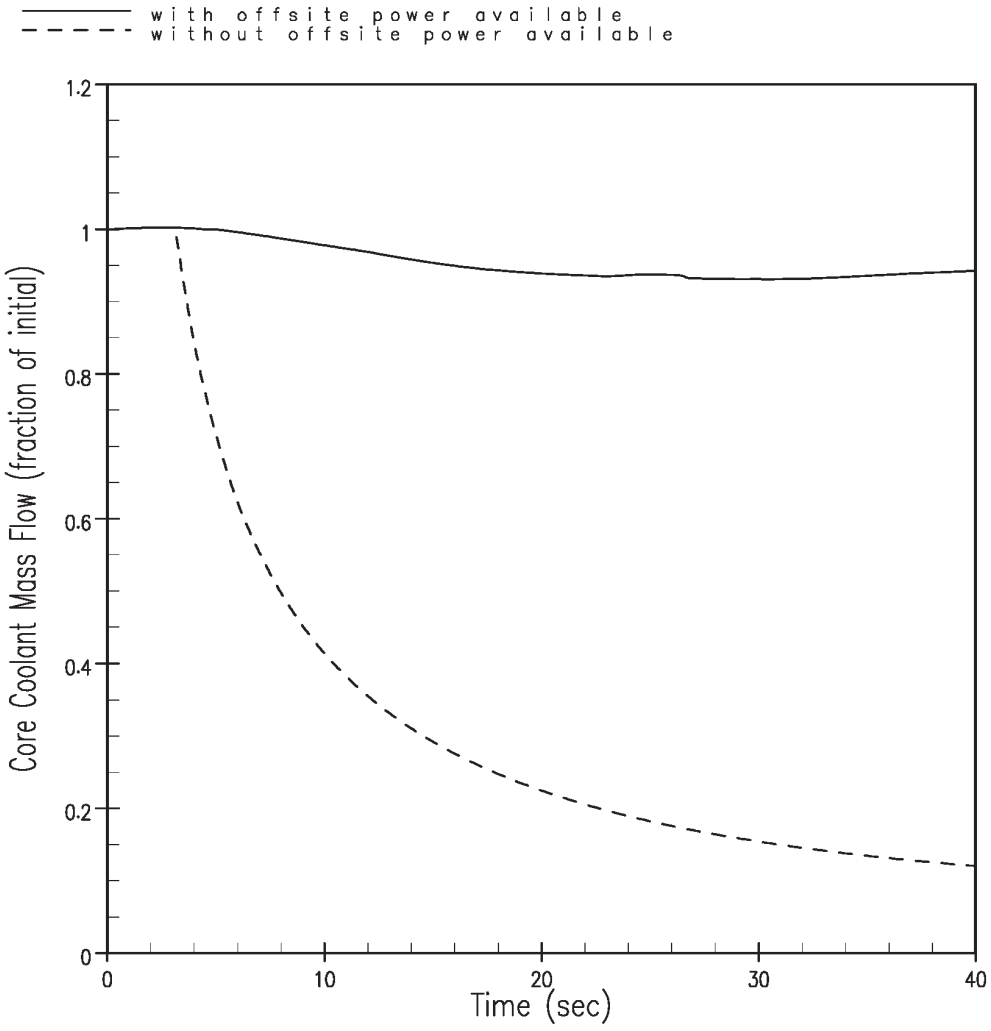

Figure 15.2.3-7
Core Coolant Mass Flow Rate versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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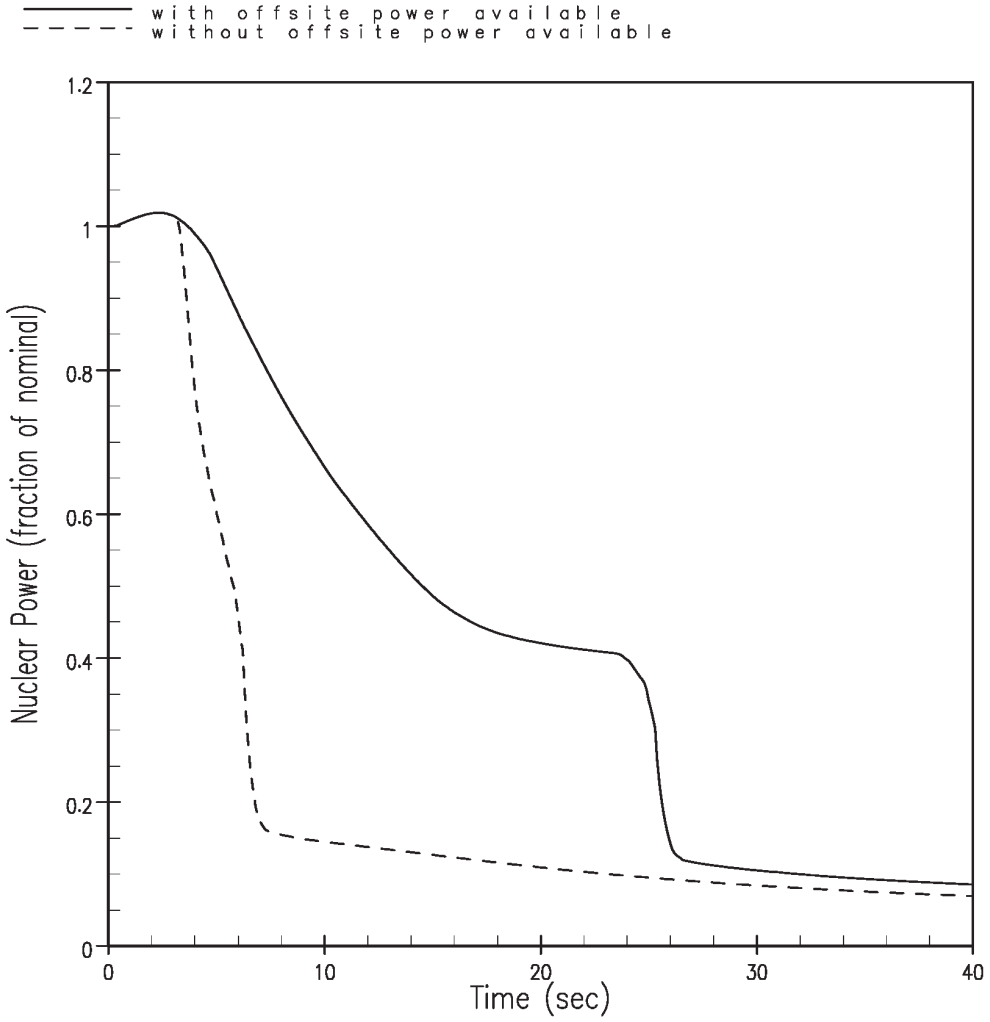

Figure 15.2.3-8
Nuclear Power versus Time for Turbine
Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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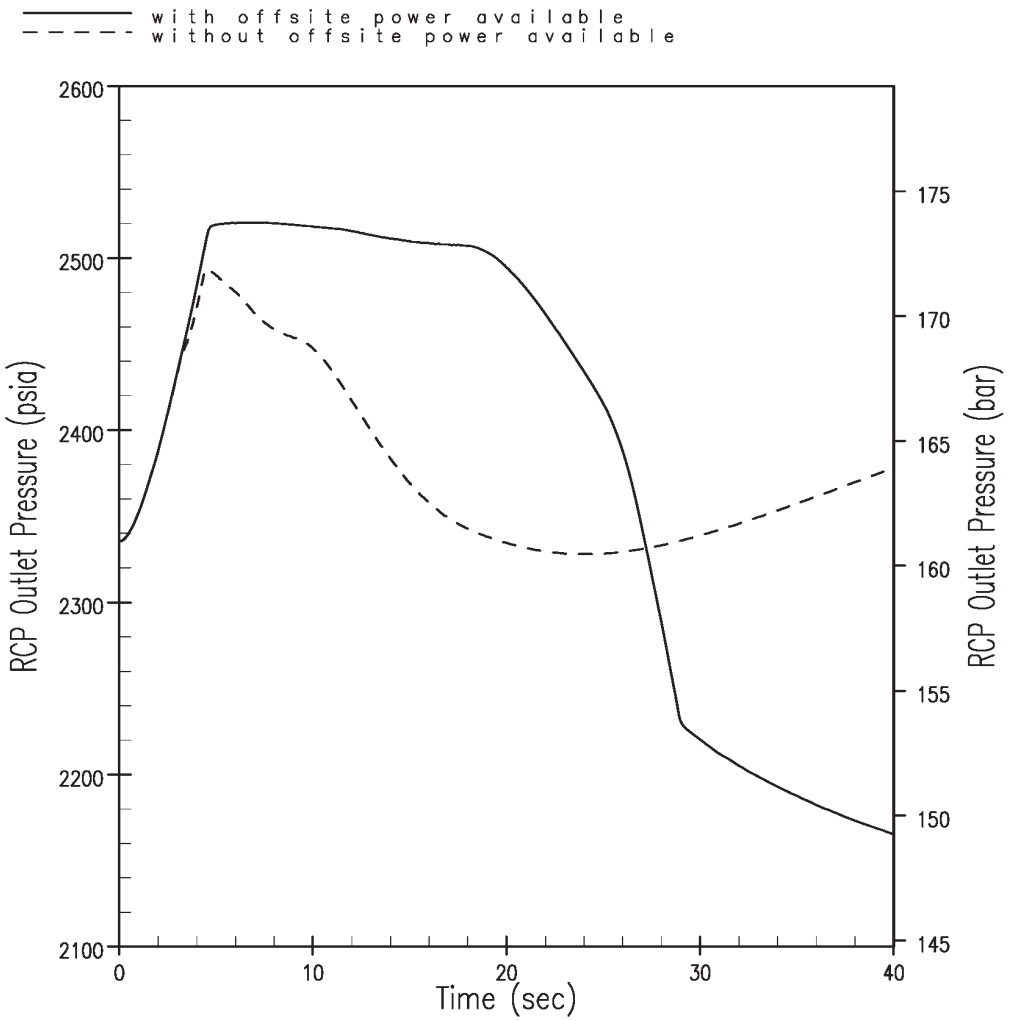

Figure 15.2.3-9
RCP Outlet Pressure versus Time for Turbine Trip
Accident with Pressurizer Spray and Maximum Moderator Feedback

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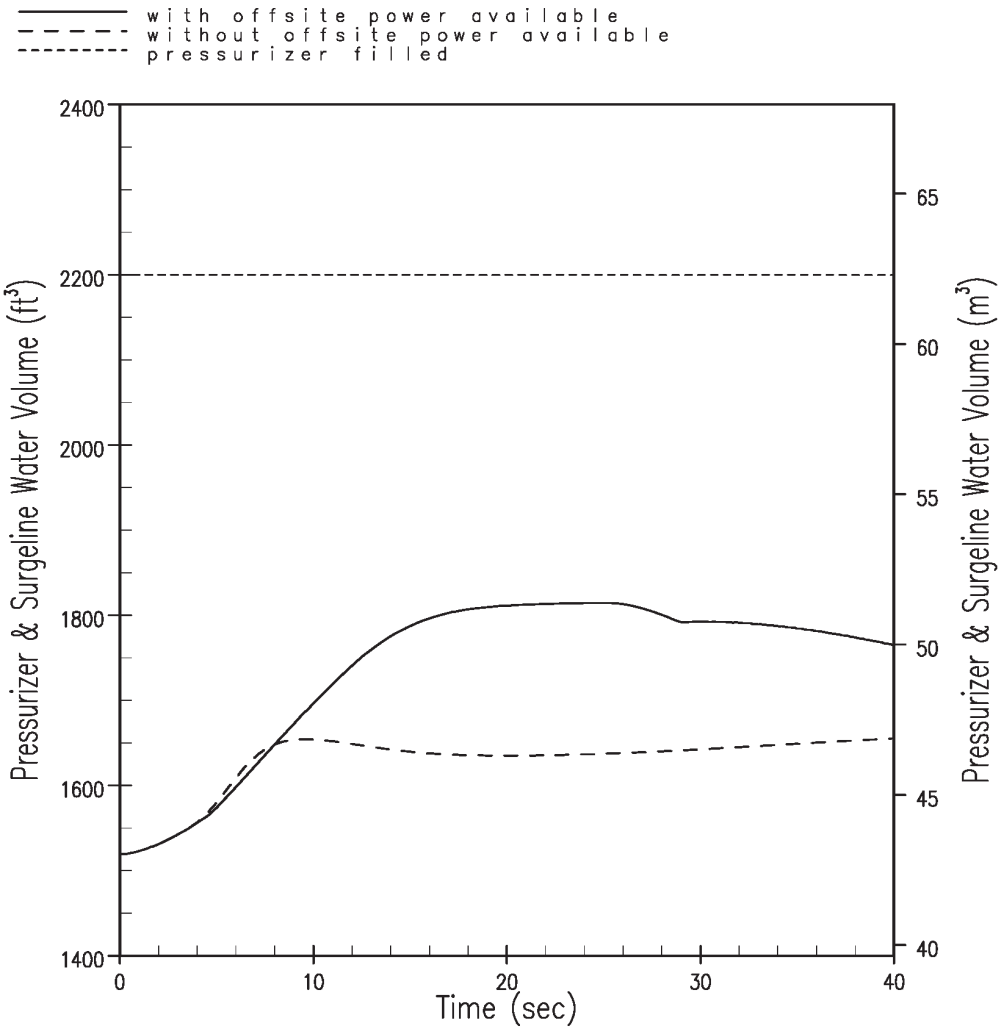

Figure 15.2.3-10
Pressurizer & Surge Line Water Volume versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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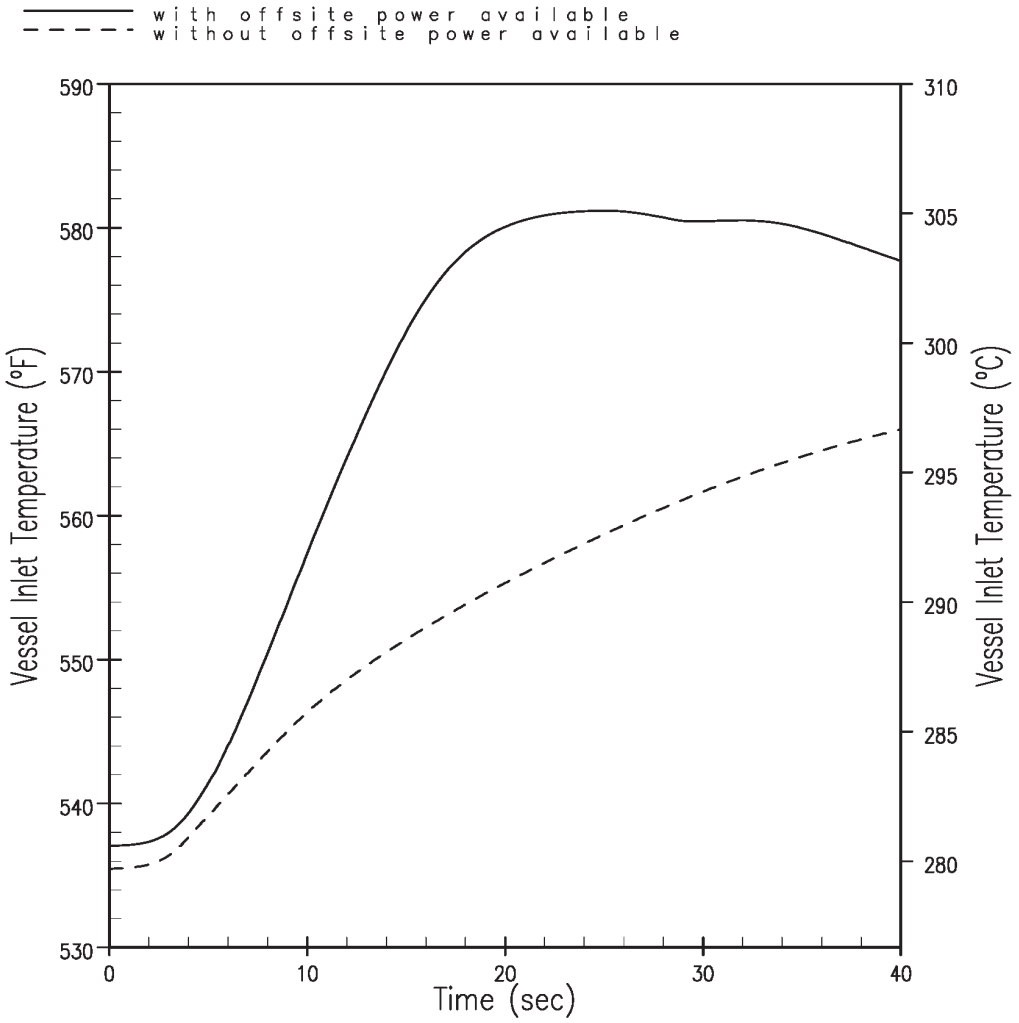

Figure 15.2.3-11
Vessel Inlet Temperature versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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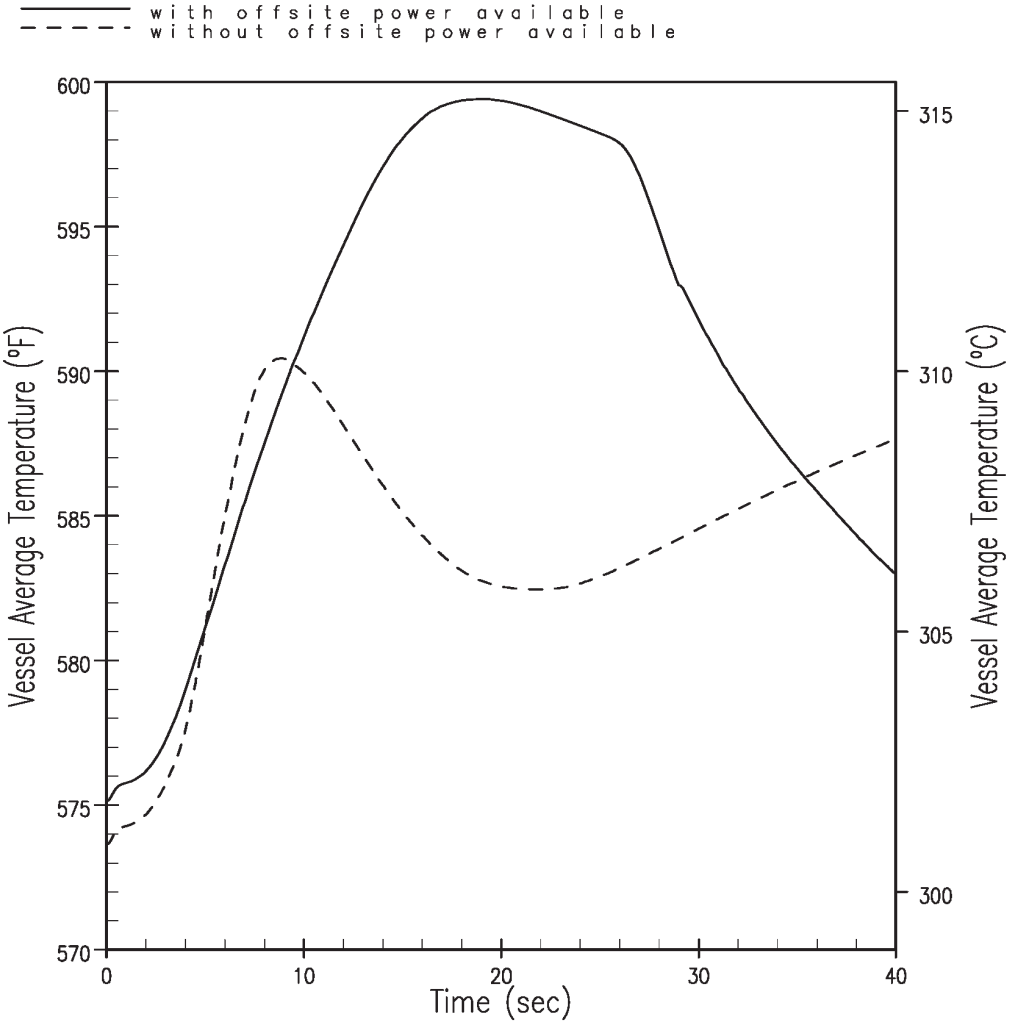

Figure 15.2.3-12
Vessel Average Temperature versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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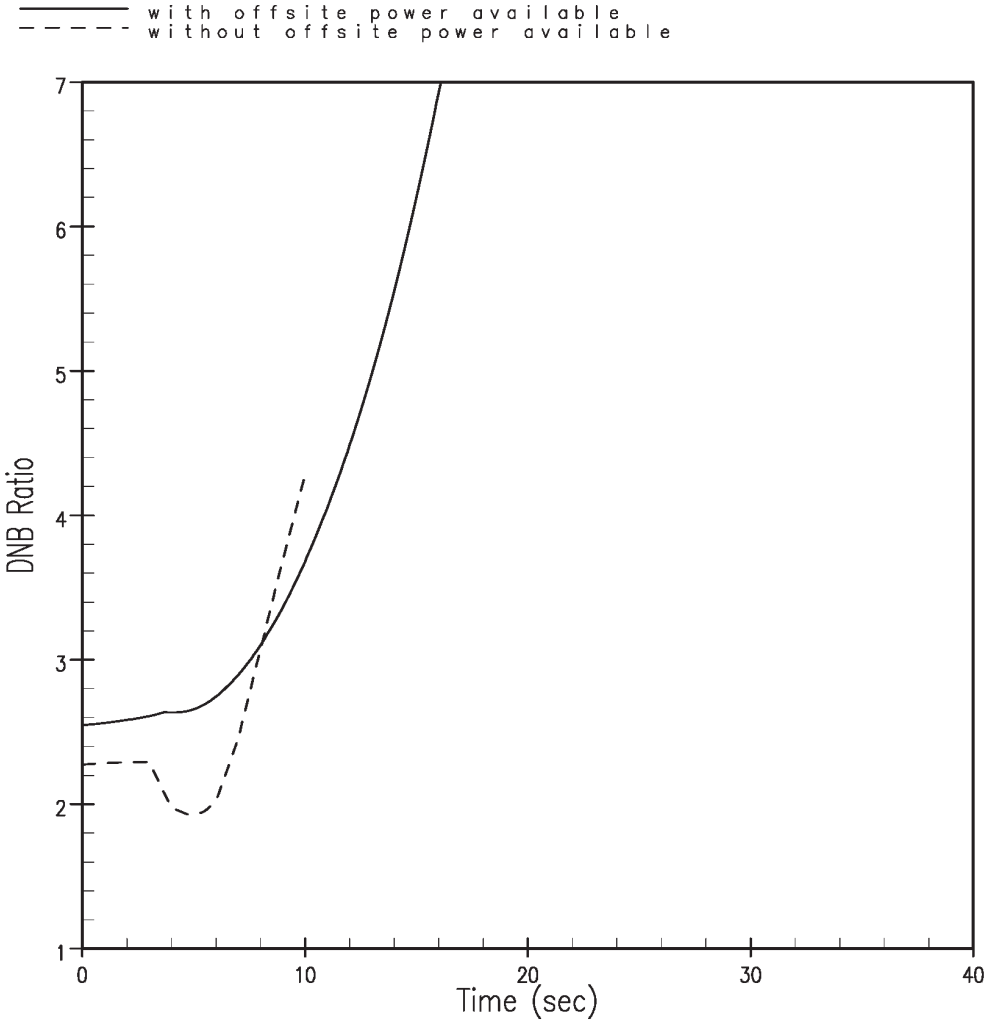

Figure 15.2.3-13
DNBR versus Time for Turbine Trip Accident
with Pressurizer Spray and Maximum Moderator Feedback

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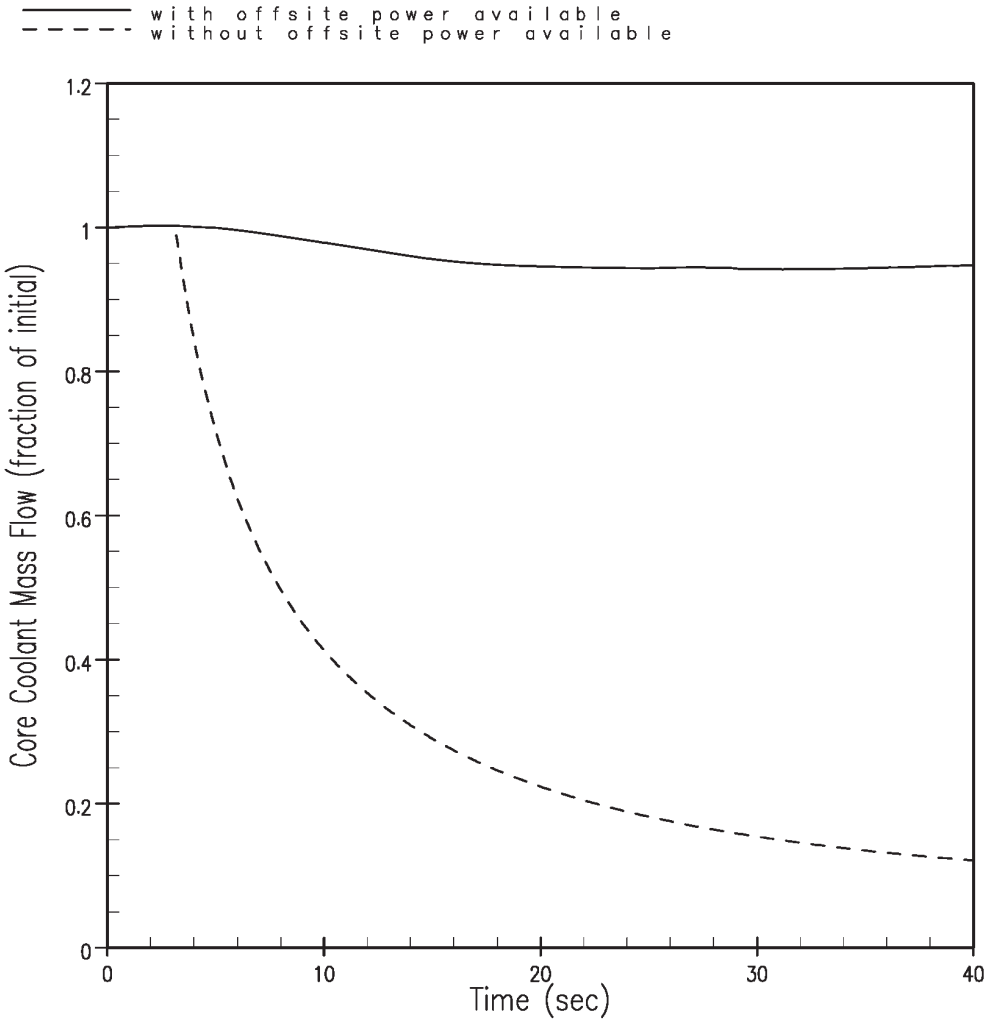

Figure 15.2.3-14
Core Coolant Mass Flow Rate versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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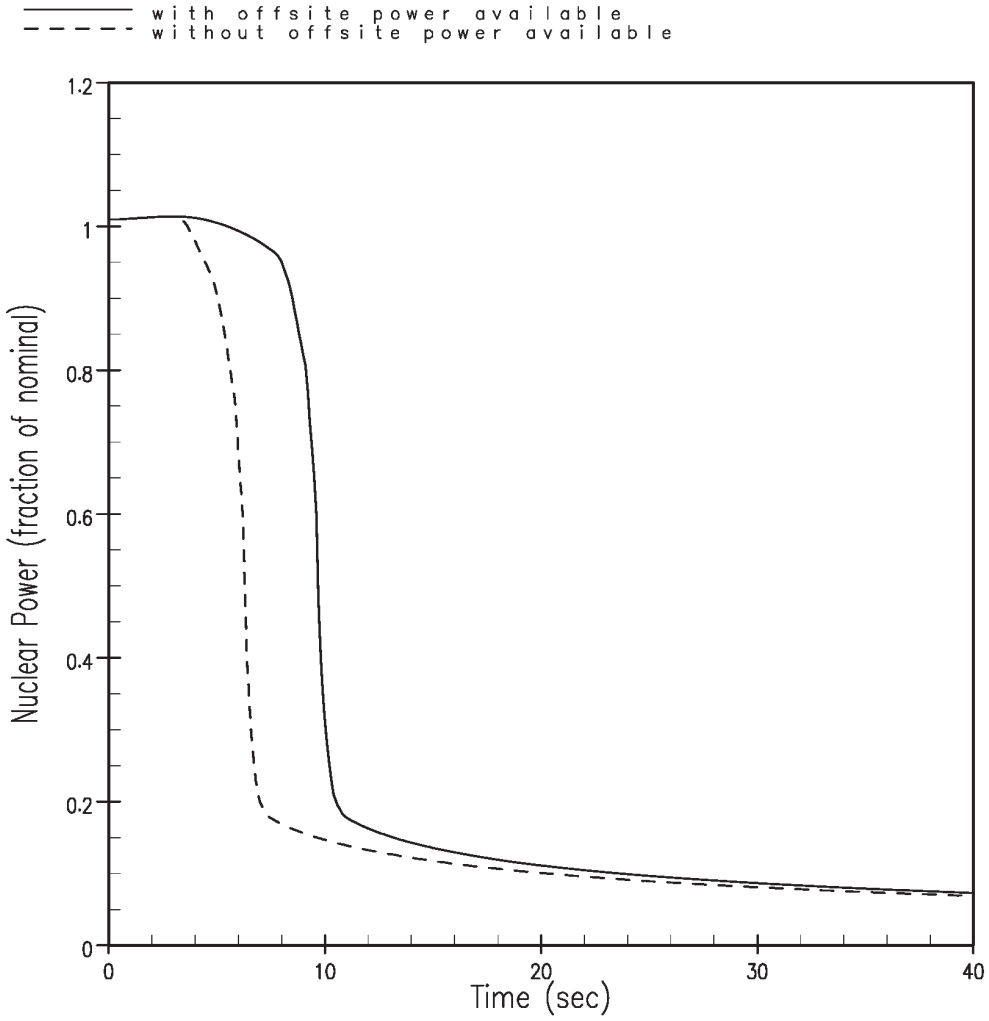

Figure 15.2.3-15
Nuclear Power versus Time for Turbine
Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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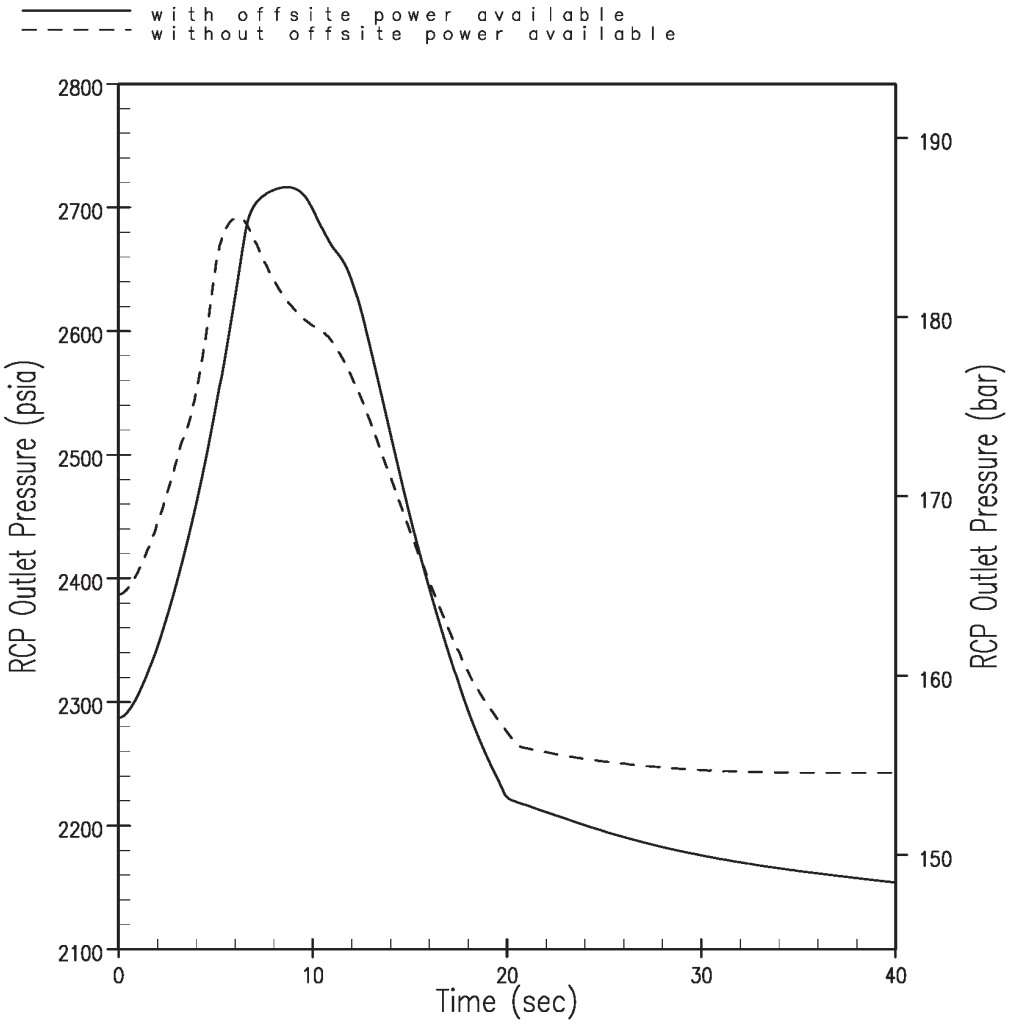

Figure 15.2.3-16
RCP Outlet Pressure versus Time for Turbine Trip
Accident Without Pressurizer Spray and Minimum Moderator Feedback

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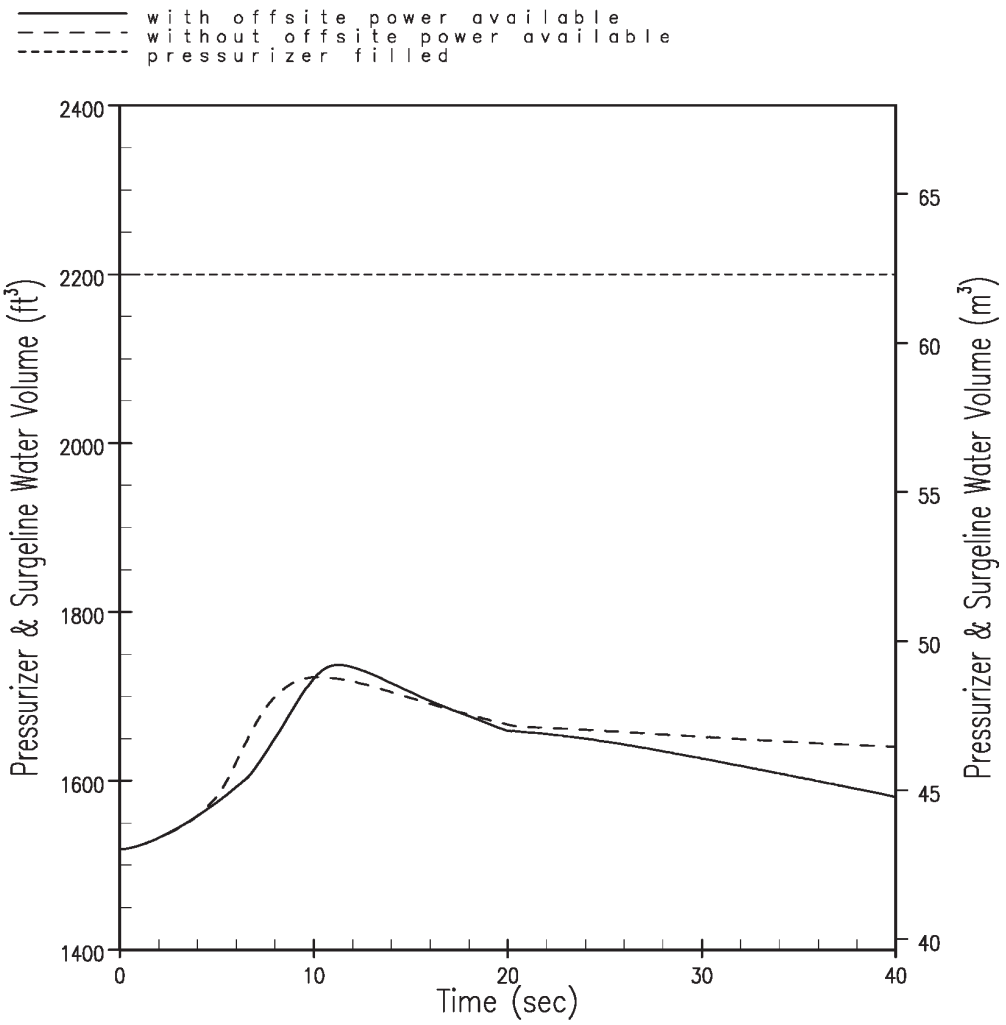

Figure 15.2.3-17
Pressurizer & Surge Line Water Volume versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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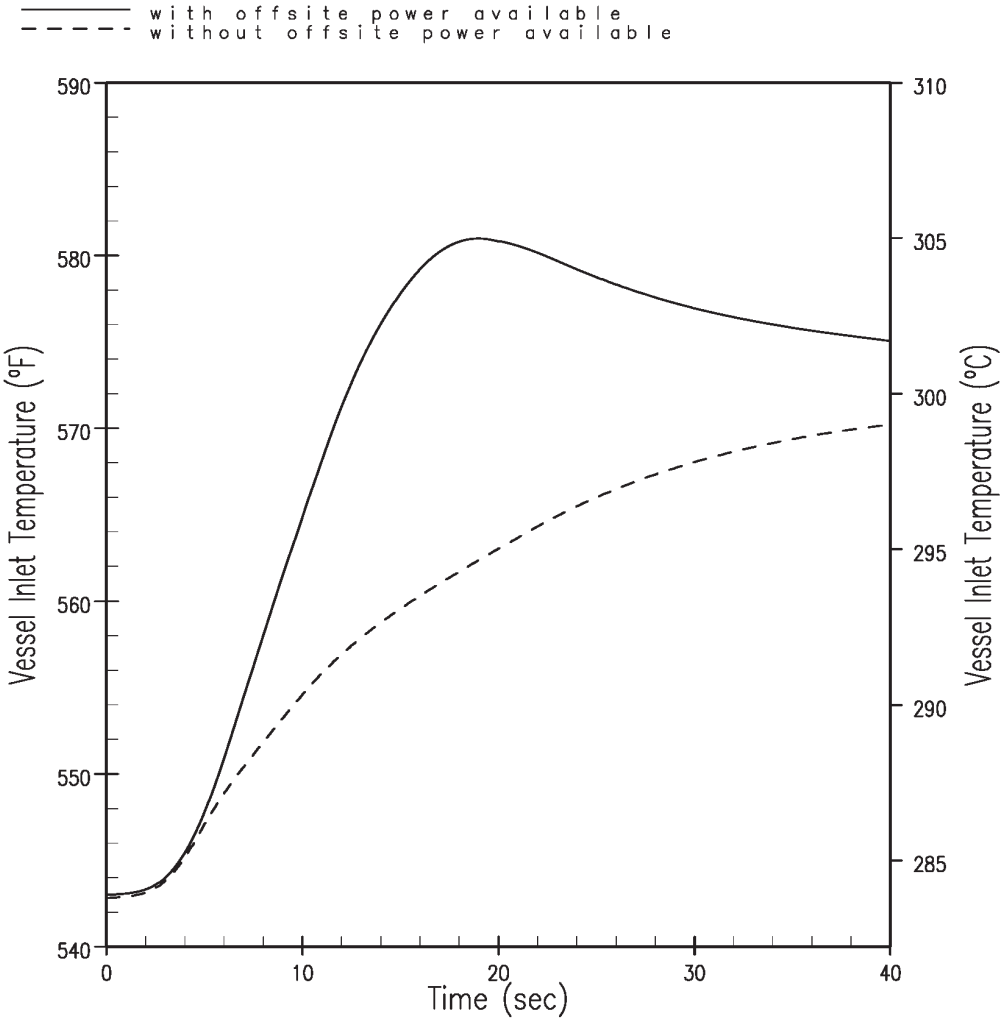

Figure 15.2.3-18
Vessel Inlet Temperature versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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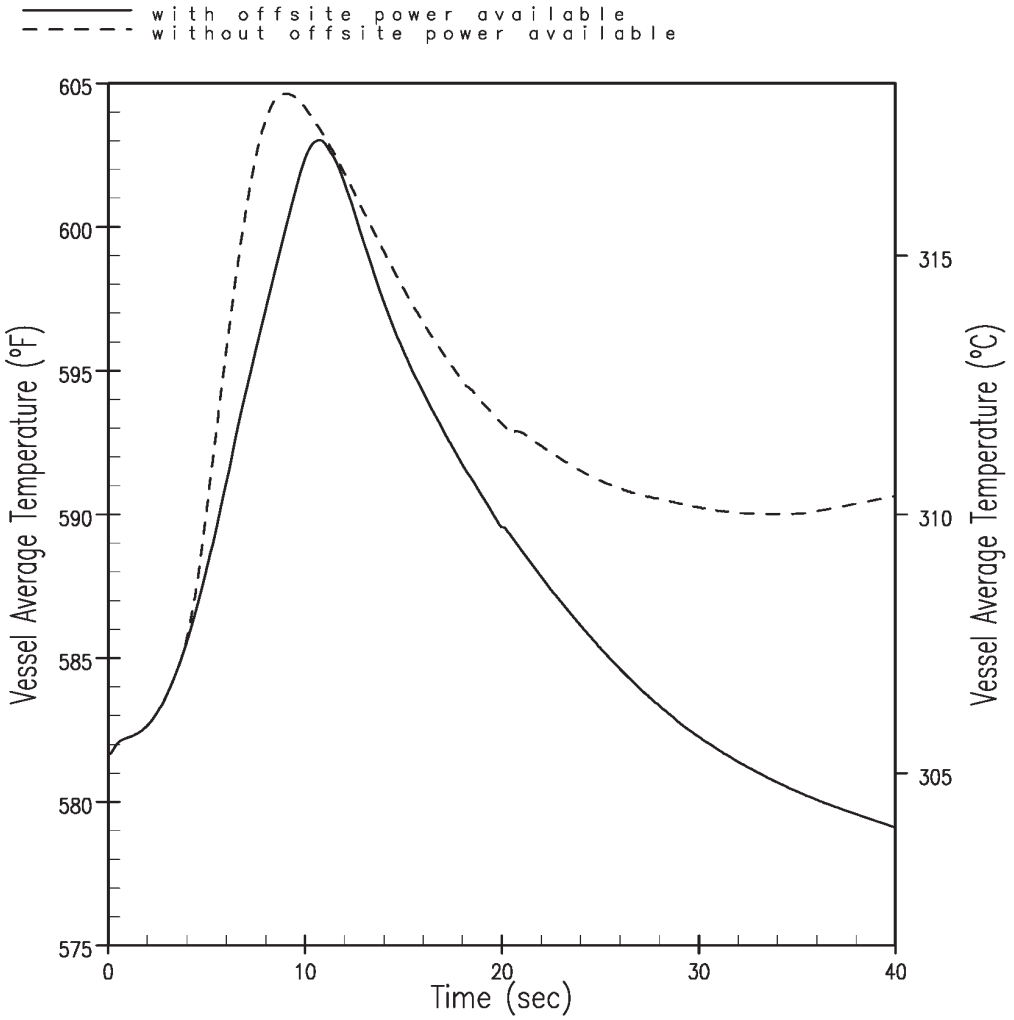

Figure 15.2.3-19
Vessel Average Temperature versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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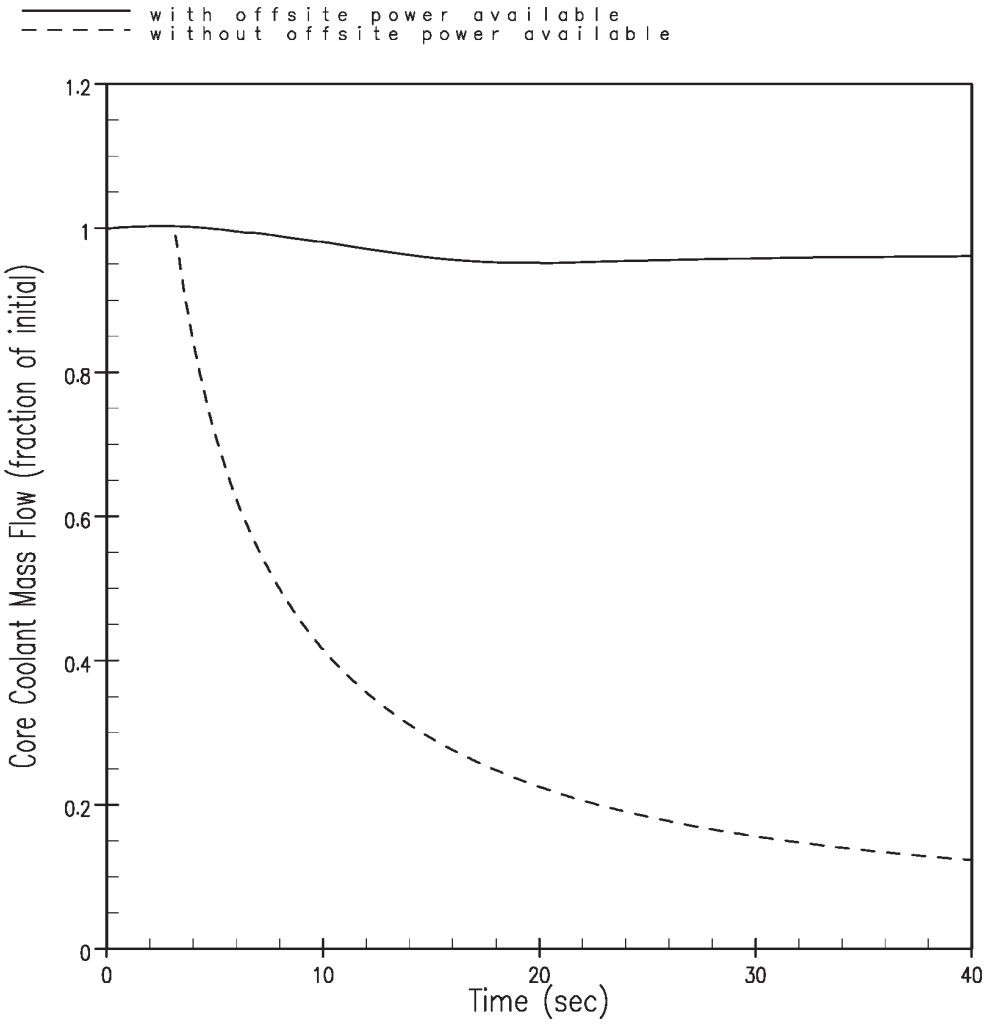

Figure 15.2.3-20
Core Coolant Mass Flow Rate versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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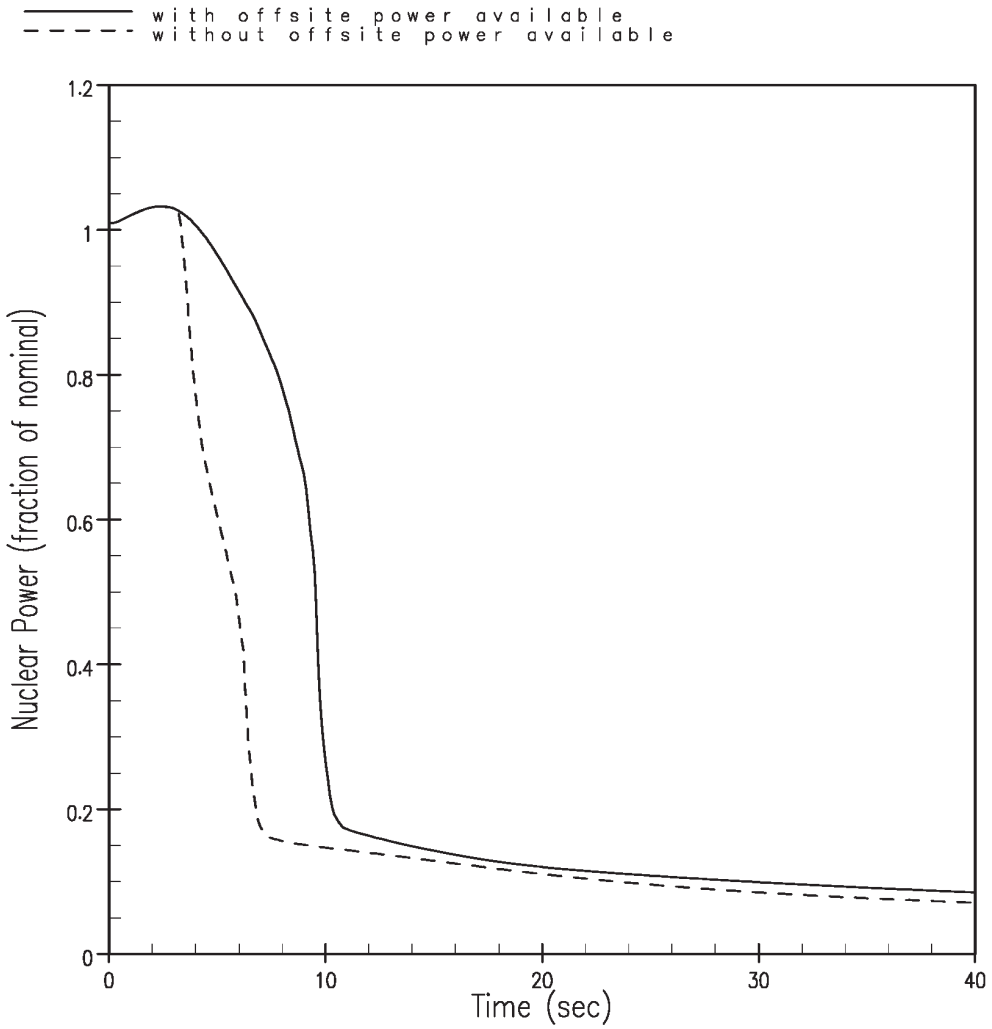

Figure 15.2.3-21
Nuclear Power versus Time for Turbine Trip
Accident Without Pressurizer Spray and Maximum Moderator Feedback

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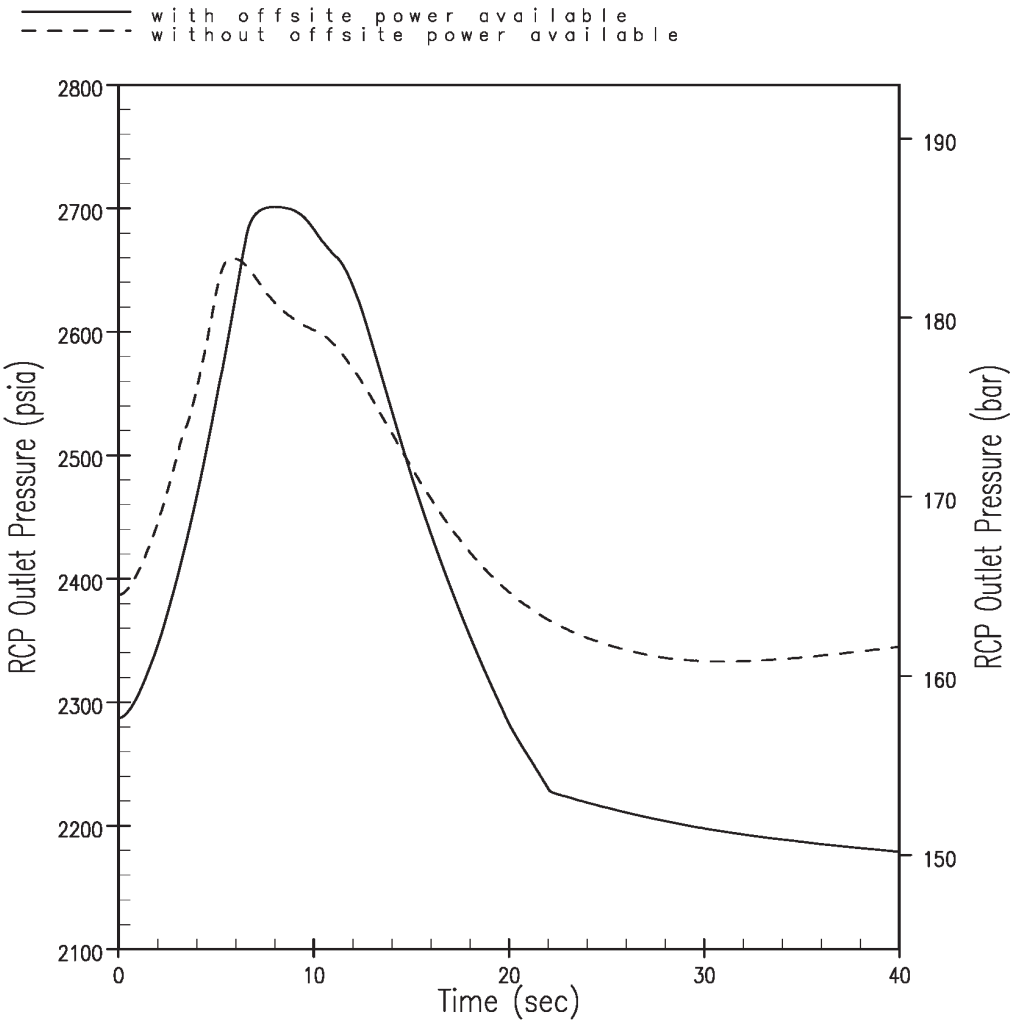

Figure 15.2.3-22
RCP Outlet Pressure versus Time for Turbine Trip
Accident Without Pressurizer Spray and Maximum Moderator Feedback

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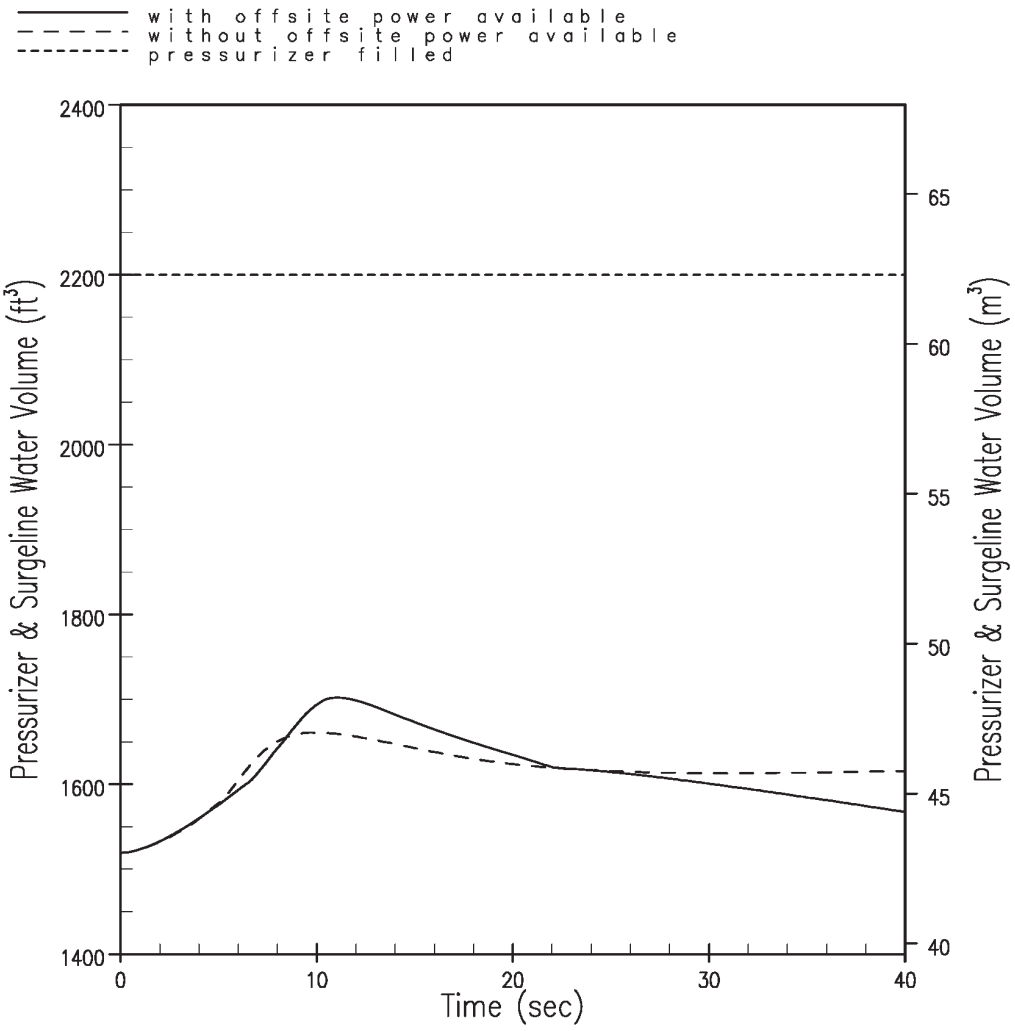

Figure 15.2.3-23
Pressurizer & Surge Line Water Volume versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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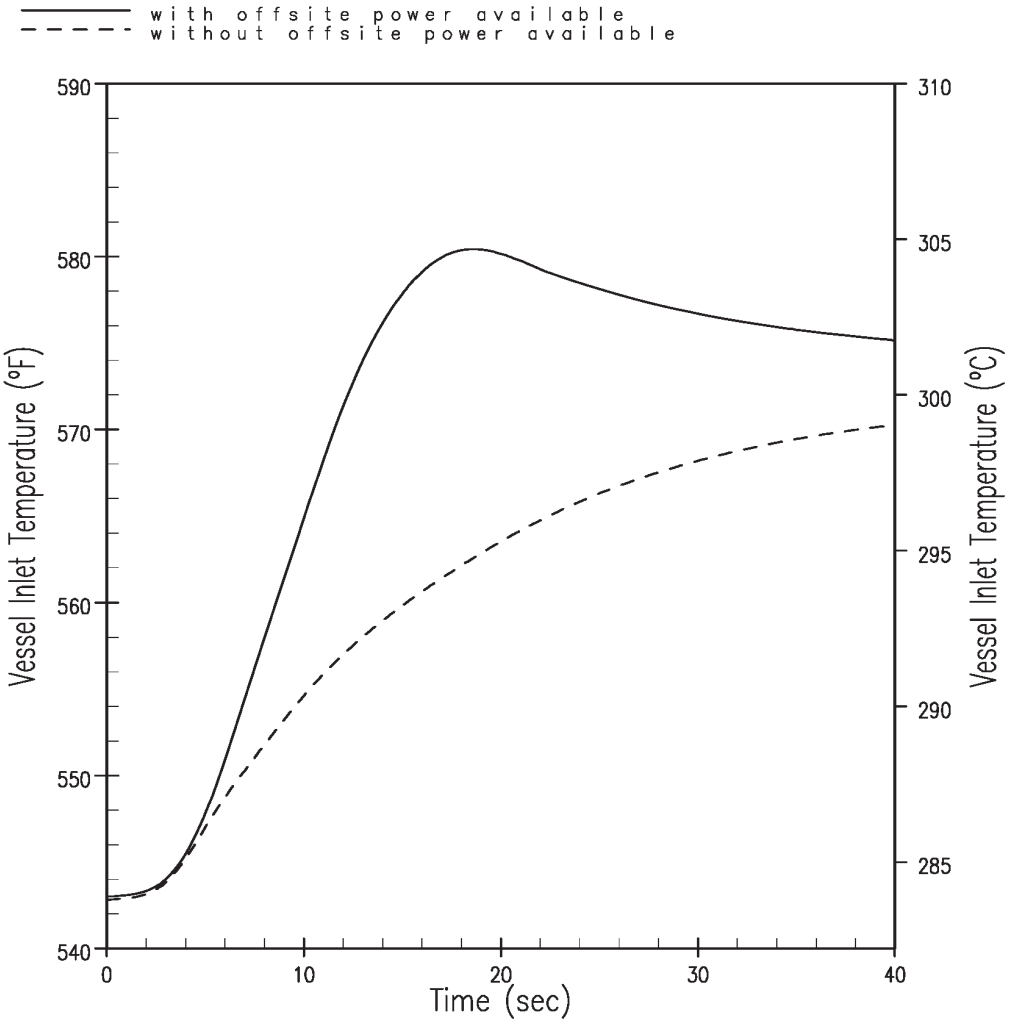

Figure 15.2.3-24
Vessel Inlet Temperature versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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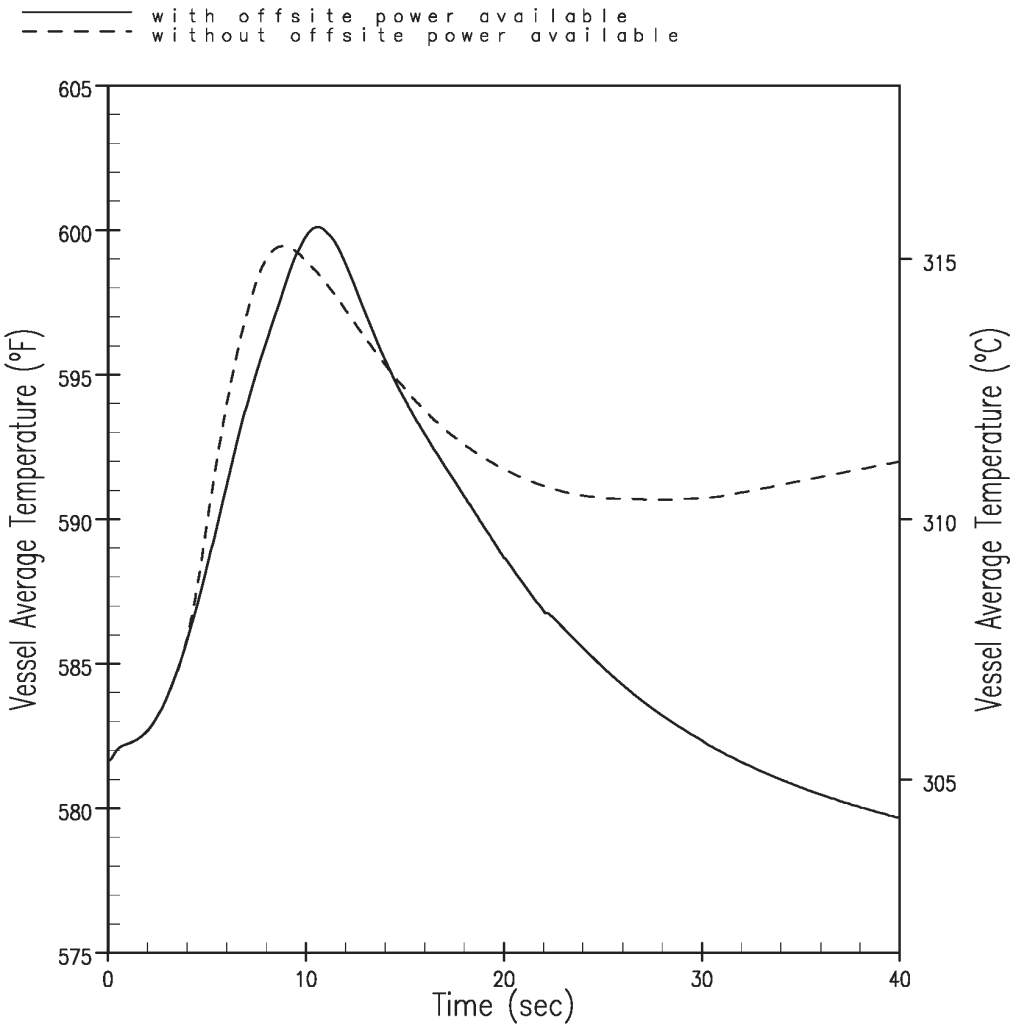

Figure 15.2.3-25
Vessel Average Temperature versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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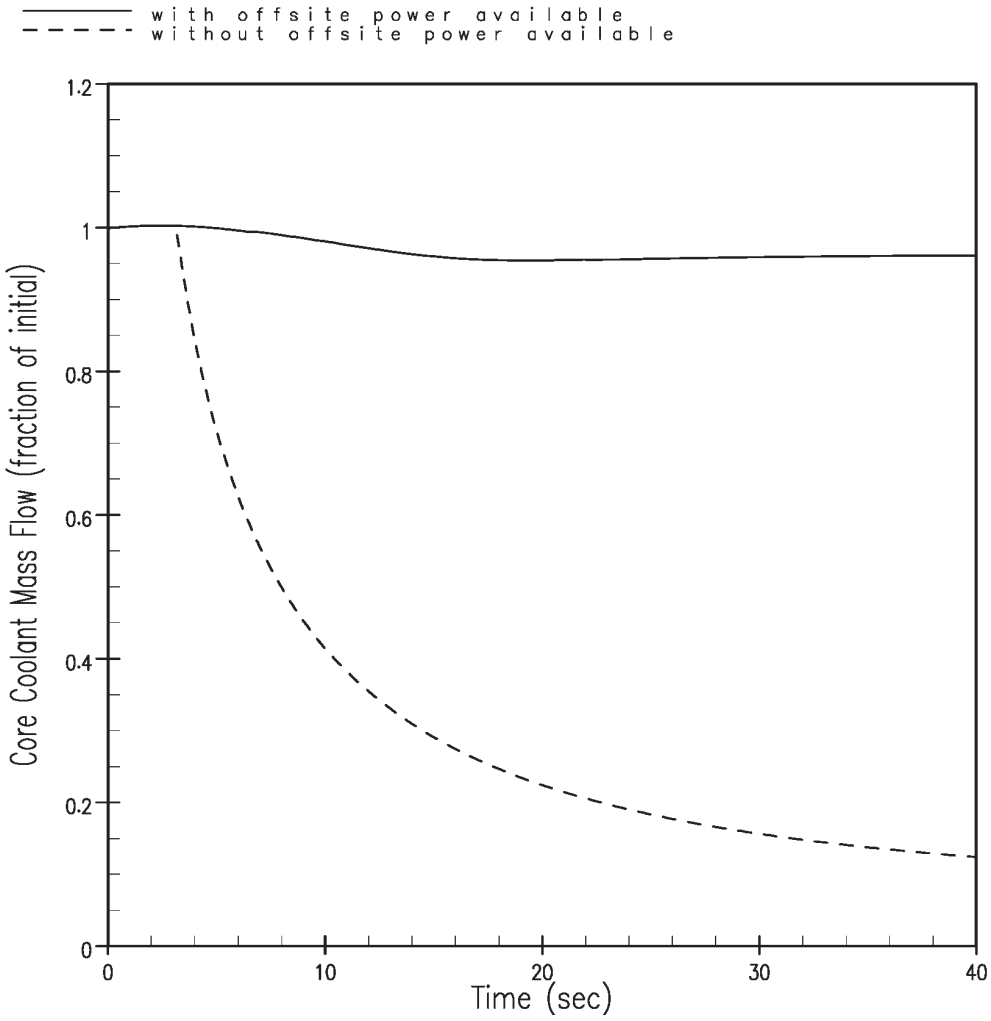

Figure 15.2.3-26
Core Coolant Mass Flow Rate versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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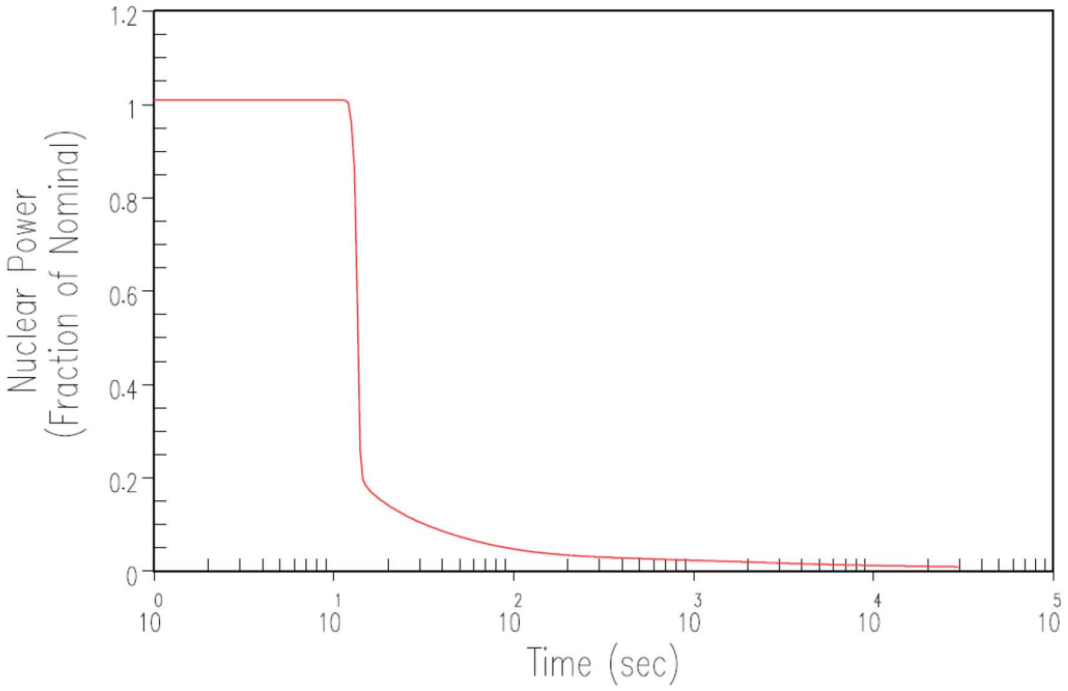

Figure 15.2.6-1
Nuclear Power Transient for Loss of ac Power to the Plant Auxiliaries

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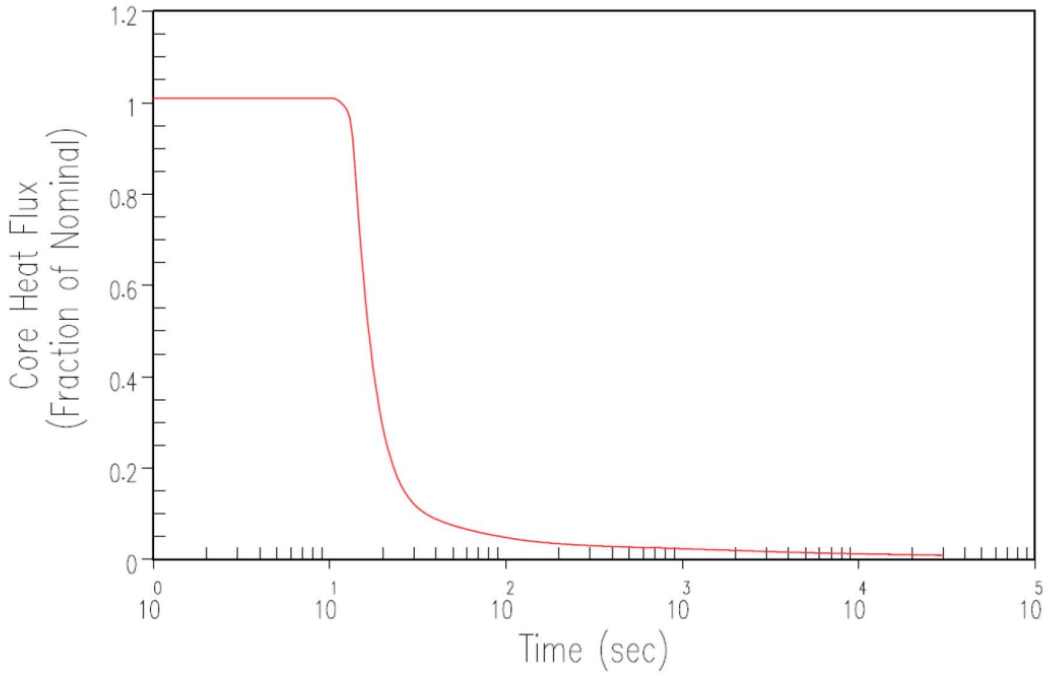

Figure 15.2.6-2
Core Heat Flux Transient for Loss of ac Power to the Plant Auxiliaries

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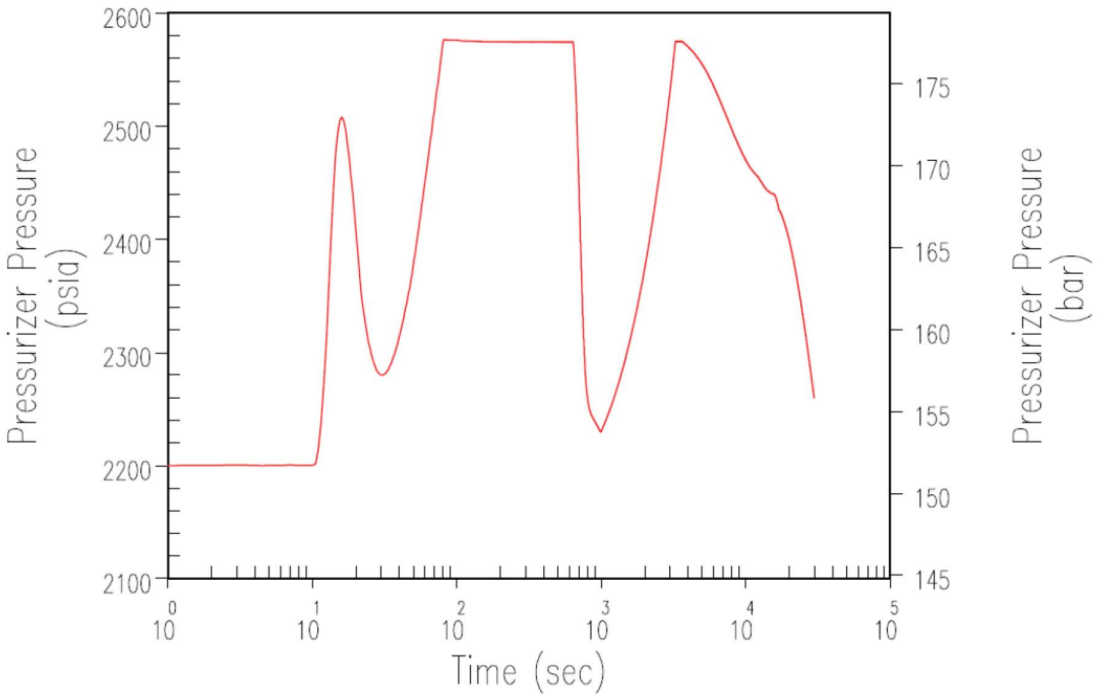

Figure 15.2.6-3
Pressurizer Pressure Transient for Loss of ac Power to the Plant Auxiliaries

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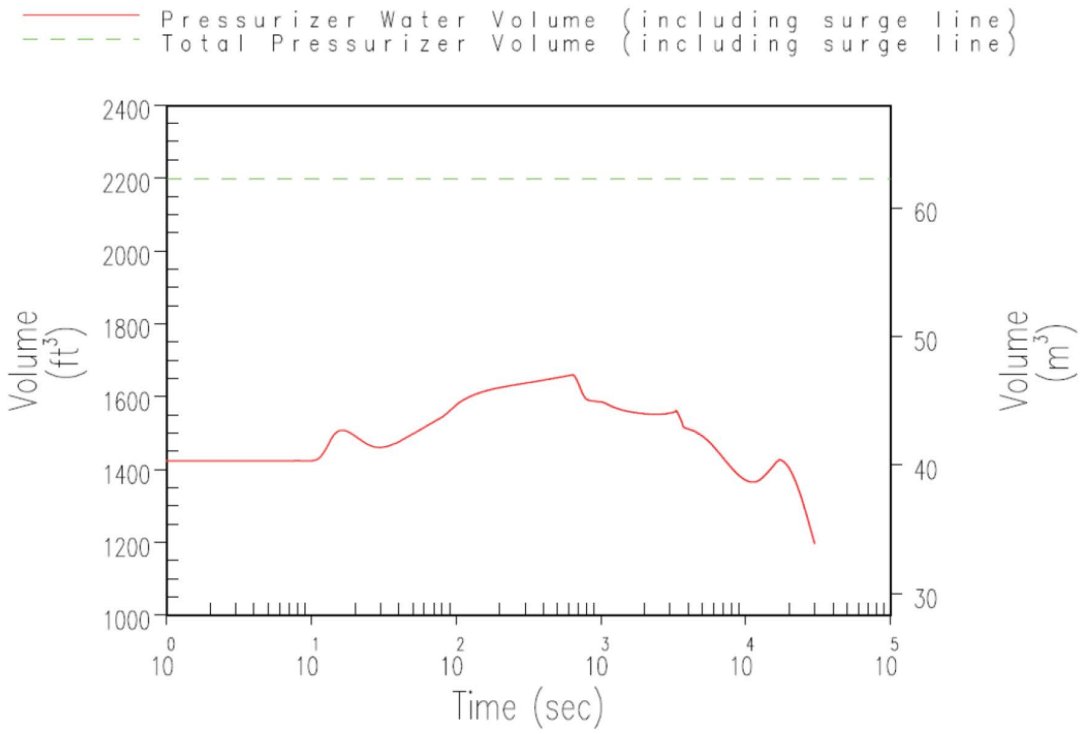

Figure 15.2.6-4
Pressurizer Water Volume Transient for Loss of ac Power to the Plant Auxiliaries

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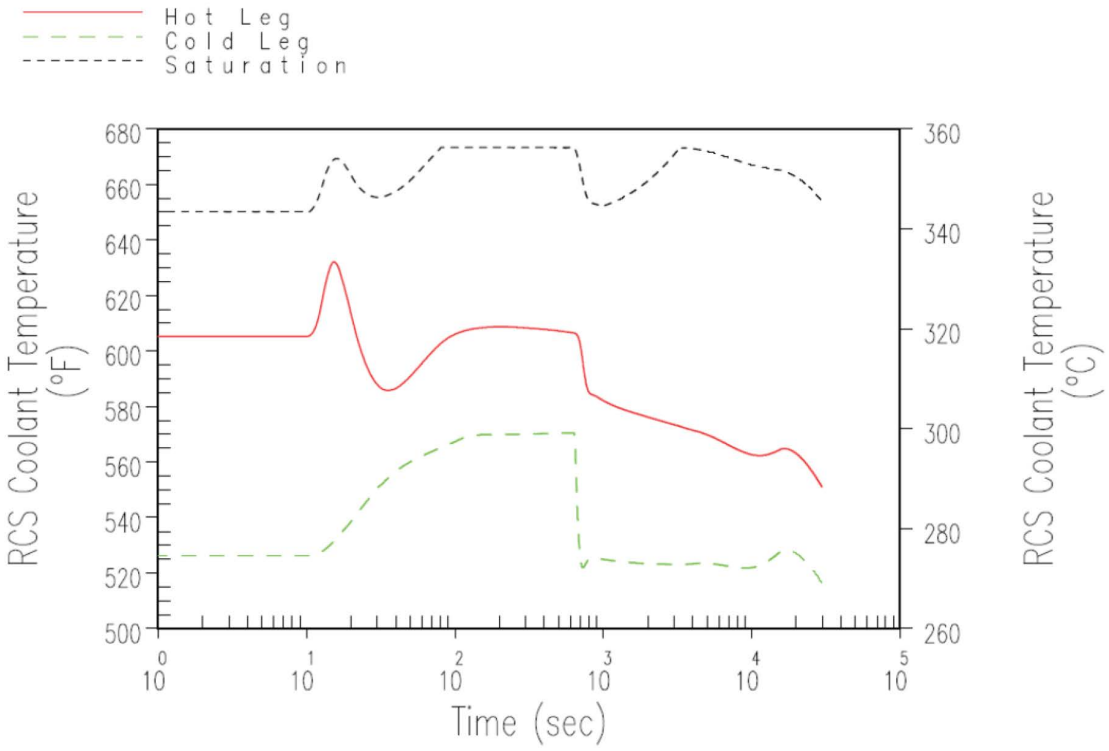

Figure 15.2.6-5
Reactor Coolant System Temperature Transients in Loop Containing the PRHR for Loss of ac Power to the Plant Auxiliaries

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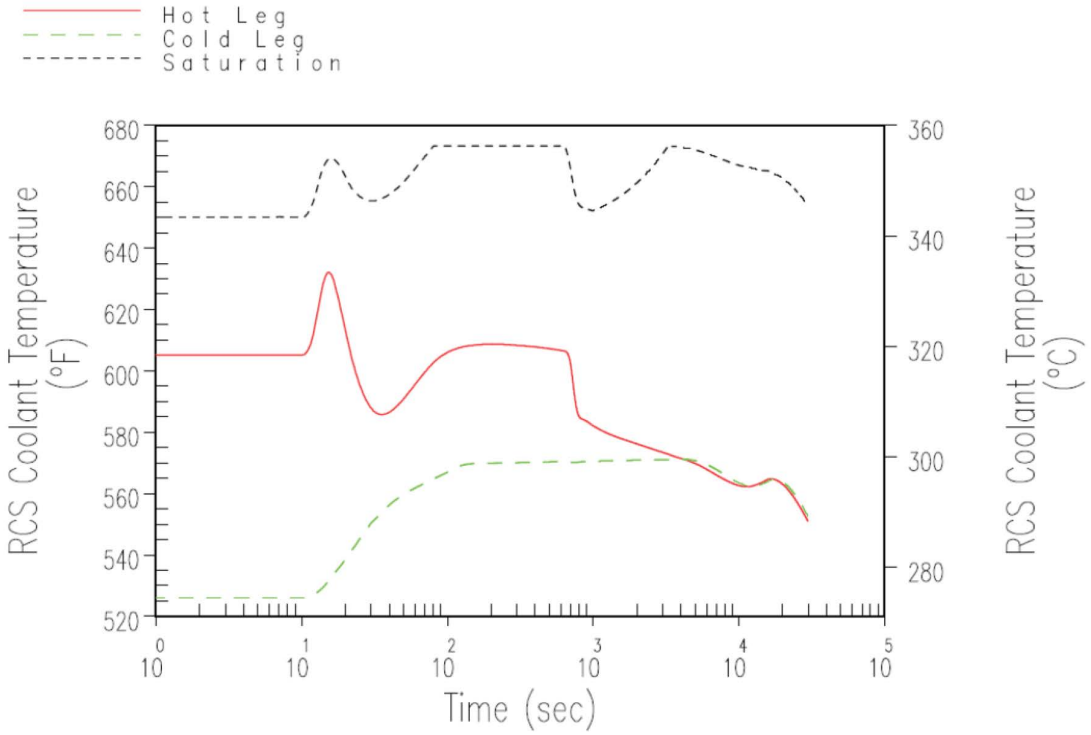

Figure 15.2.6-6
Reactor Coolant System Temperature Transients in Loop Not Containing the PRHR for Loss of ac Power to the Plant Auxiliaries

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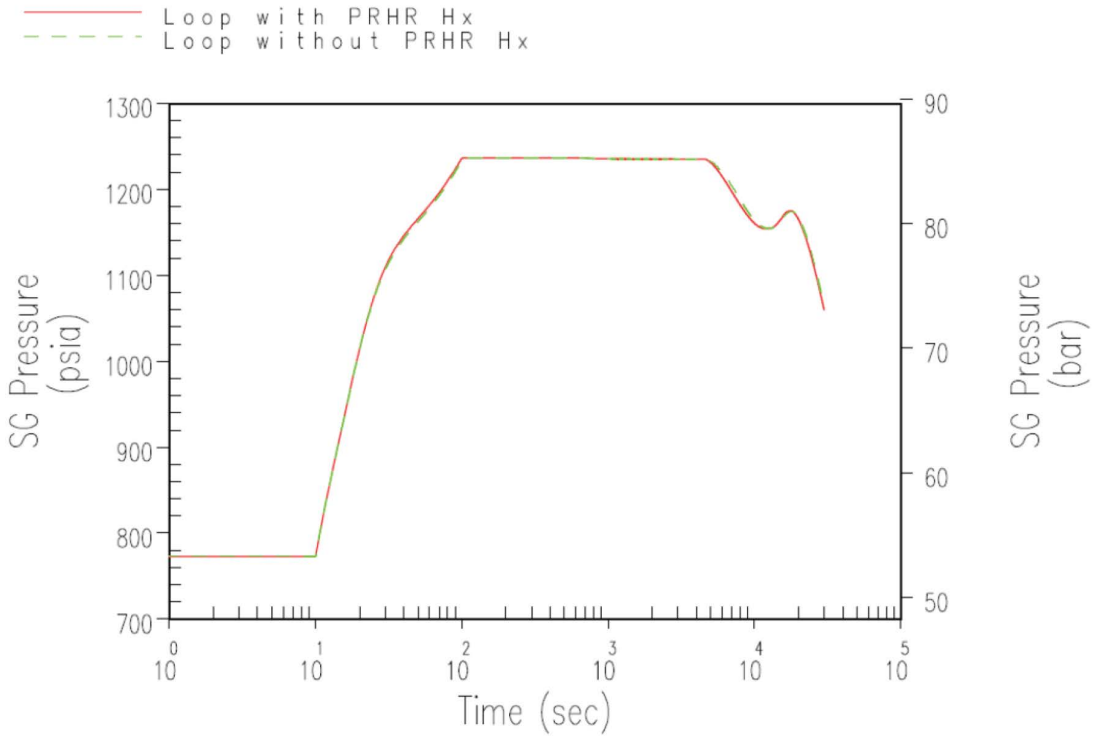

Figure 15.2.6-7
Steam Generator Pressure Transient
for Loss of ac Power to the Plant Auxiliaries

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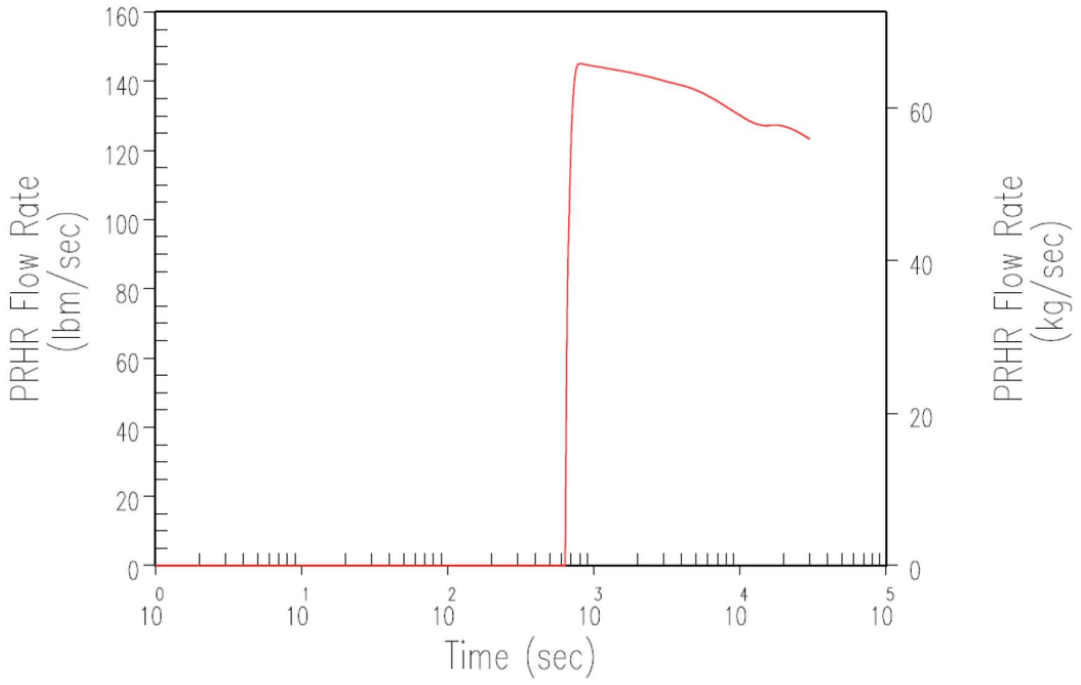

Figure 15.2.6-8
PRHR Flow Rate Transient
for Loss of ac Power to the Plant Auxiliaries

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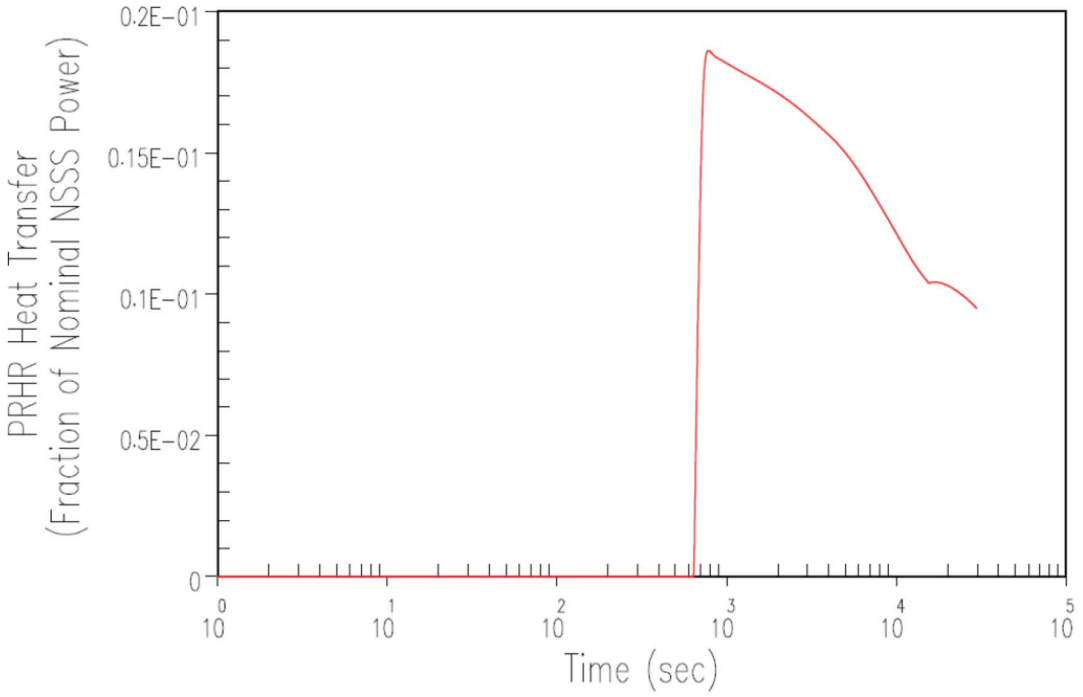

Figure 15.2.6-9
PRHR Heat Transfer Transient
for Loss of ac Power to the Plant Auxiliaries

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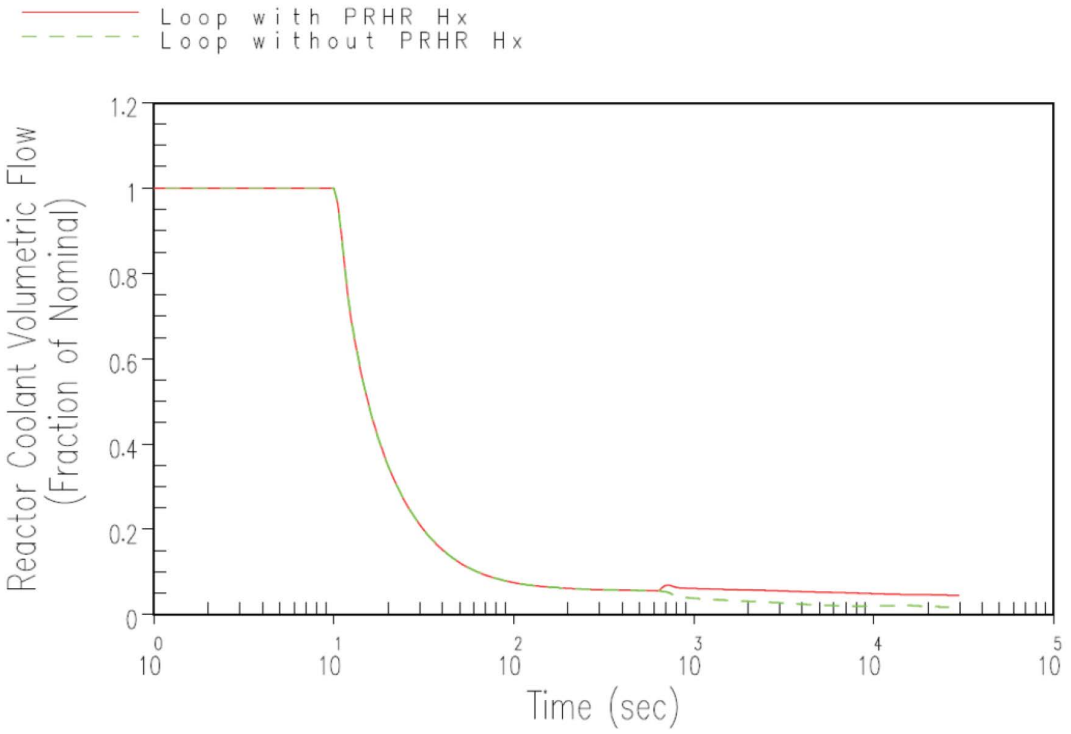

Figure 15.2.6-10
Reactor Coolant Volumetric Flow Rate
Transient for Loss of ac Power to the Plant Auxiliaries

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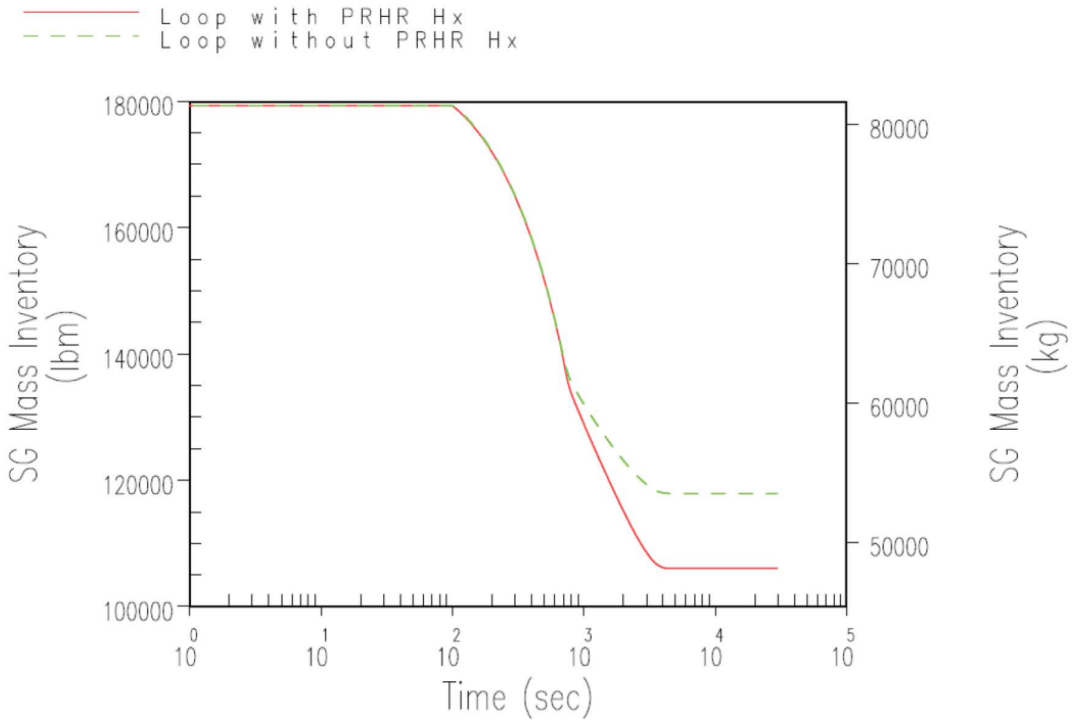

Figure 15.2.6-11
Steam Generator Inventory Transient
for Loss of ac Power to the Plant Auxiliaries

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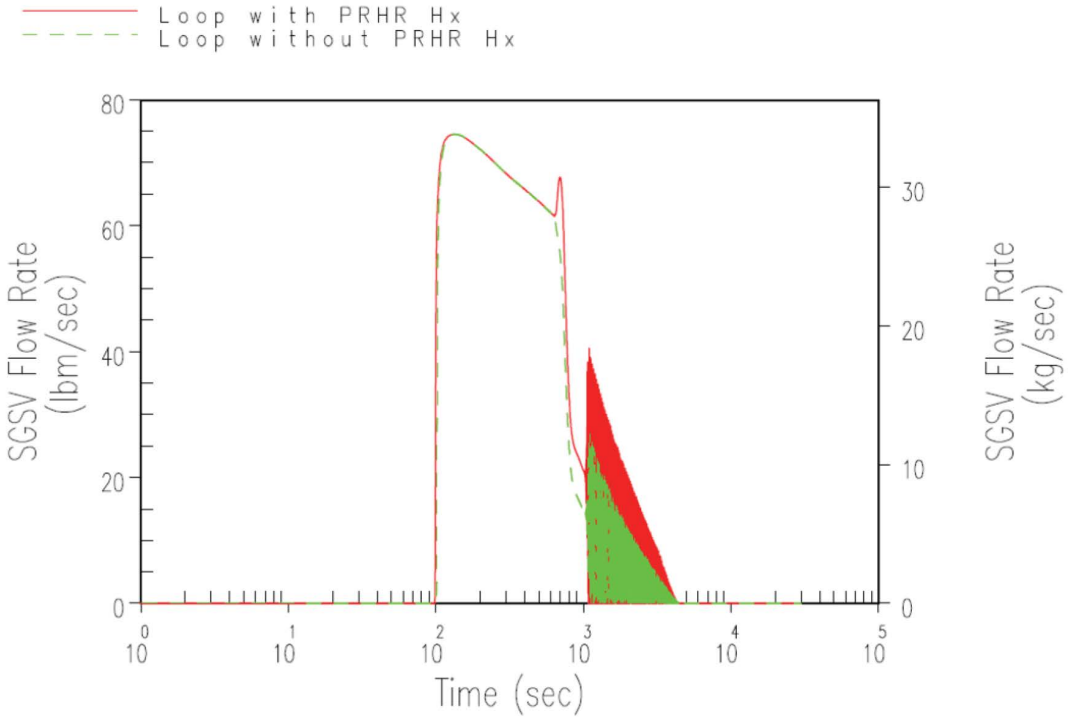

Figure 15.2.6-12
Steam Generator Safety Valve Relief
for Loss of ac Power to the Plant Auxiliaries

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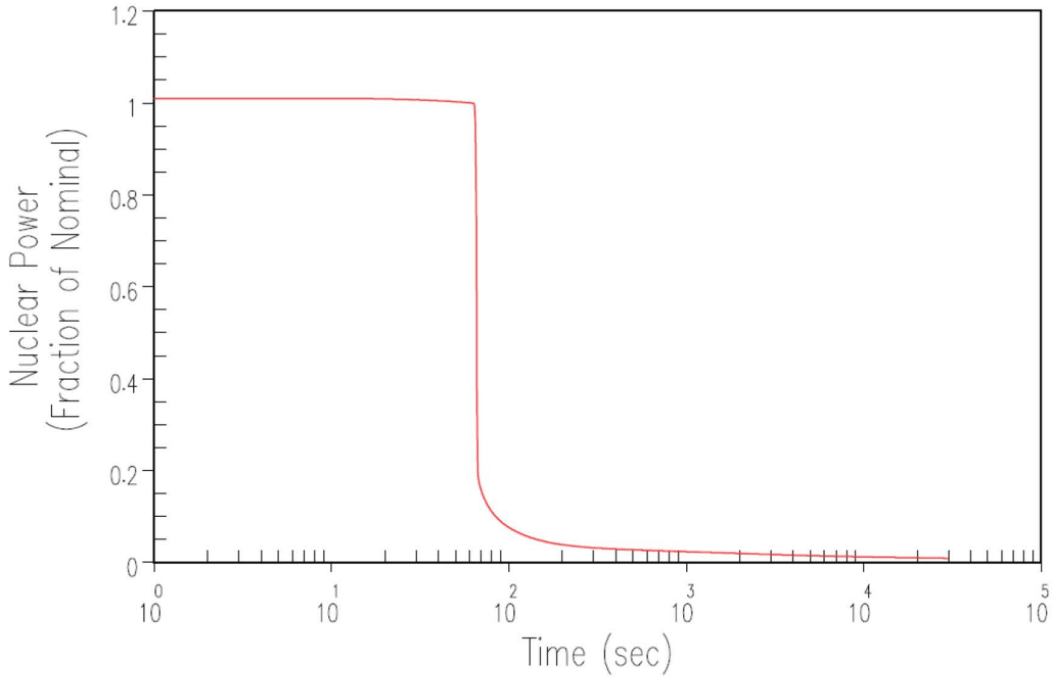

Figure 15.2.7-1
Nuclear Power Transient for Loss of Normal Feedwater

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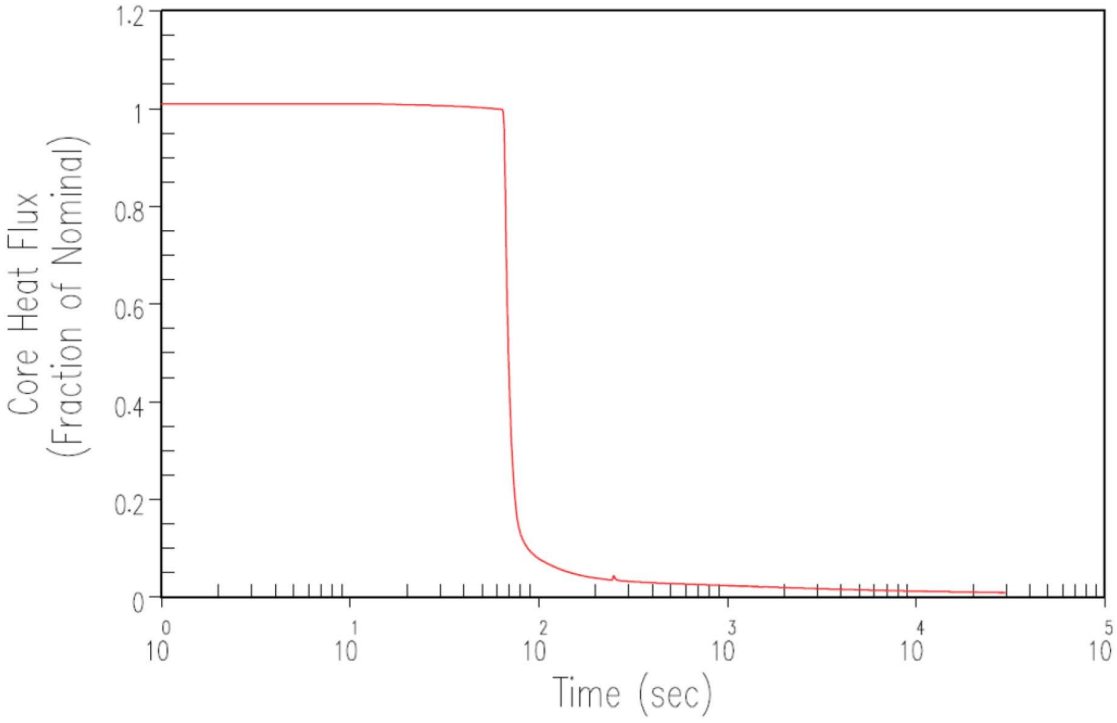

Figure 15.2.7-2
Core Heat Flux Transient for
Loss of Normal Feedwater

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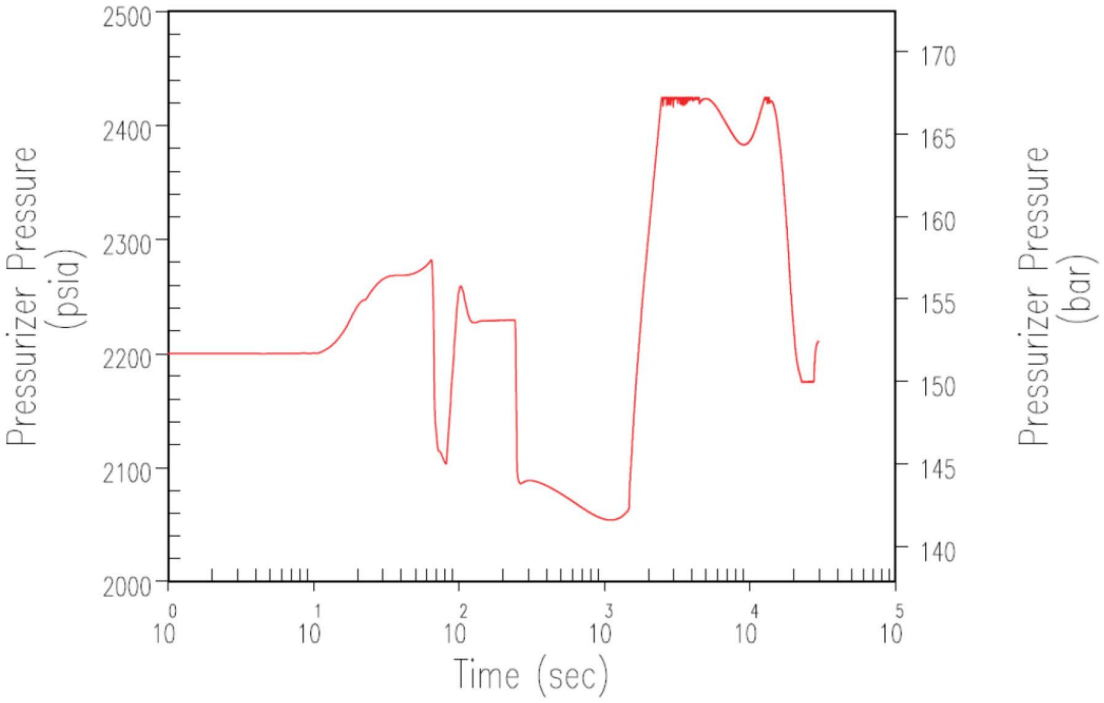

Figure 15.2.7-3
Pressurizer Pressure Transient for Loss of Normal Feedwater

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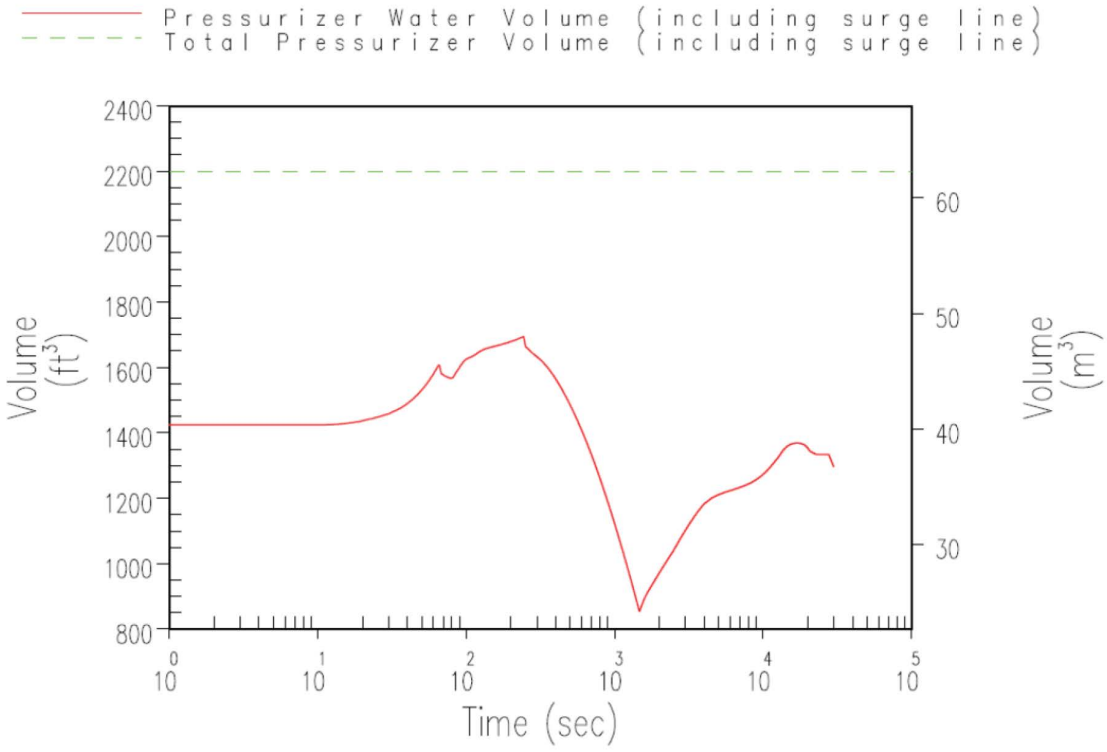

Figure 15.2.7-4
Pressurizer Water Volume Transient
for Loss of Normal Feedwater

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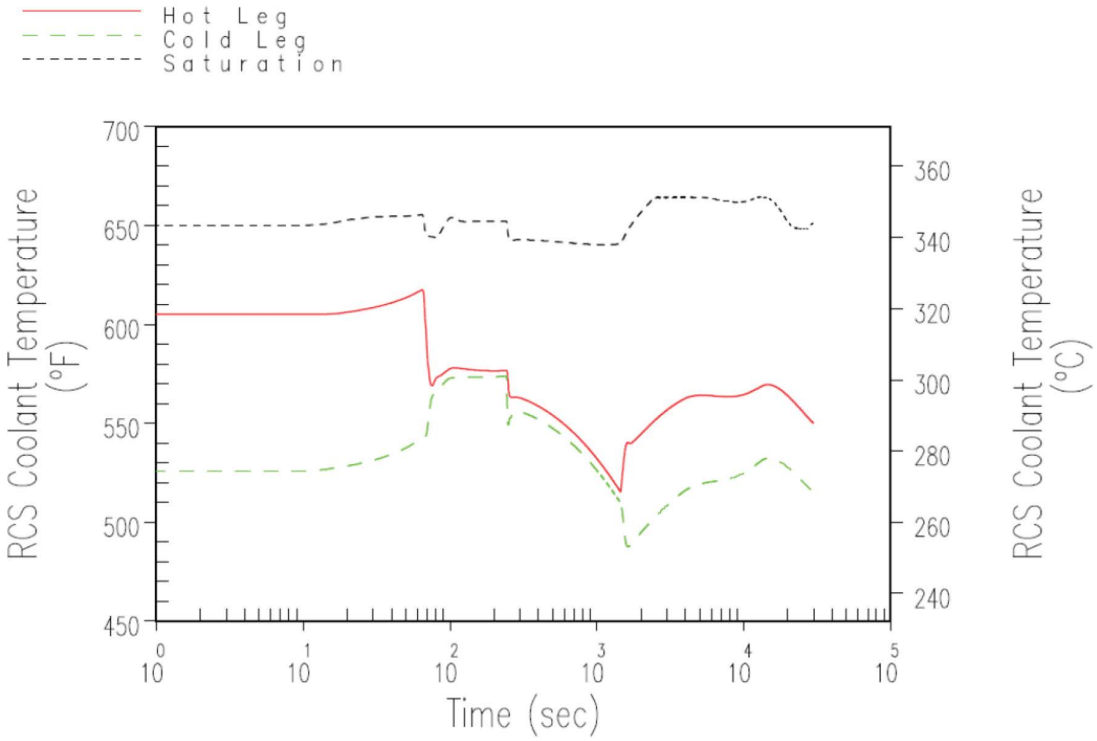

Figure 15.2.7-5
Reactor Coolant System Temperature Transients in Loop Containing the PRHR for Loss of Normal Feedwater Flow

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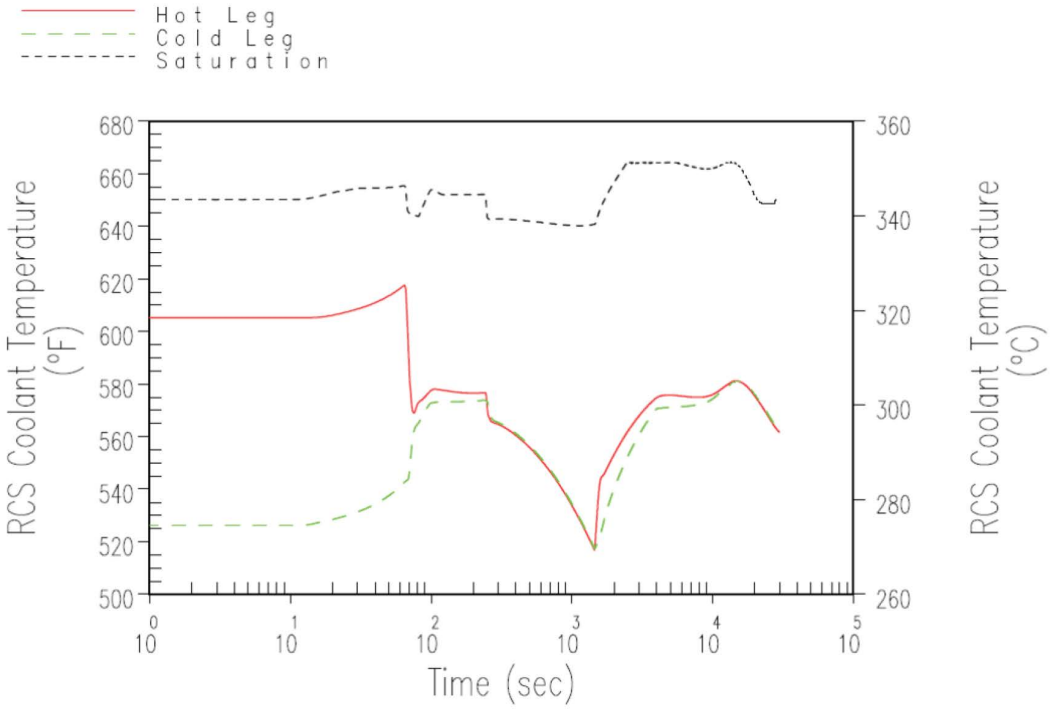

Figure 15.2.7-6
Reactor Coolant System Temperature Transient in Loop Not Containing the PRHR for Loss of Normal Feedwater Flow

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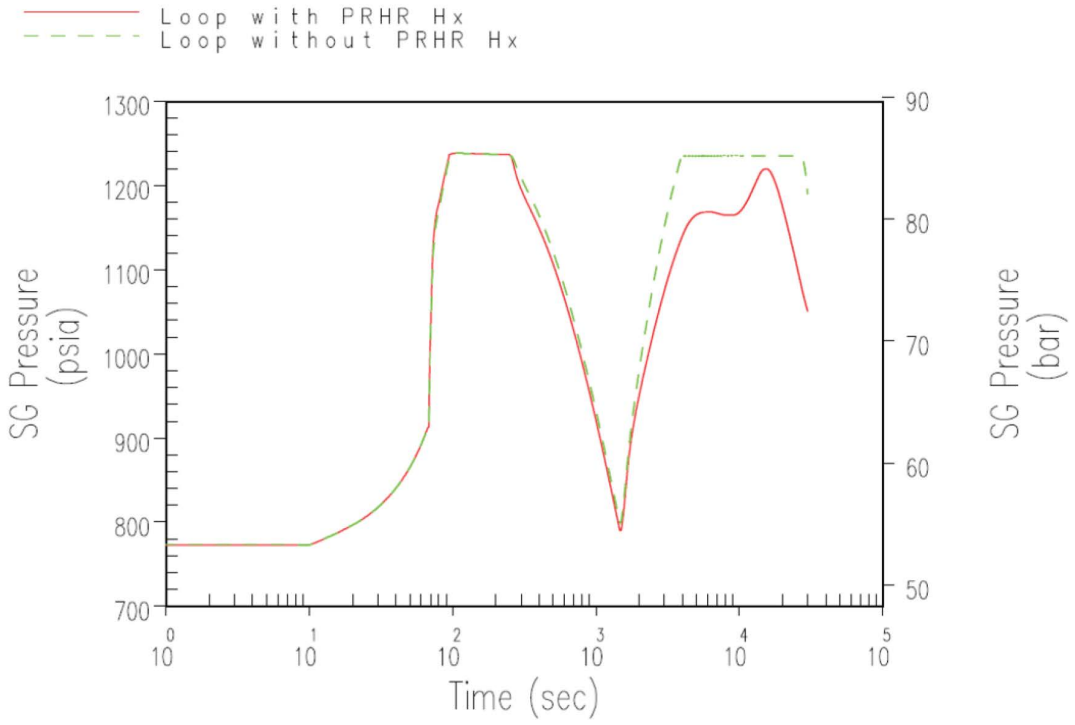

Figure 15.2.7-7
Steam Generator Pressure Transient for Loss of Normal Feedwater

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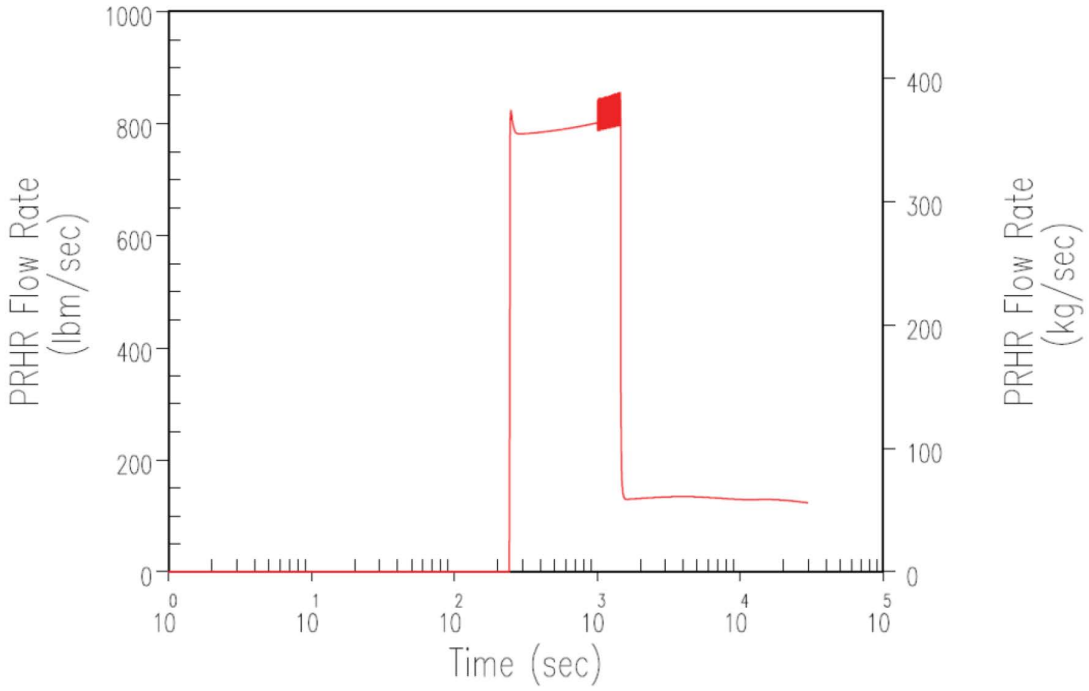

Figure 15.2.7-8
PRHR Flow Rate Transient
for Loss of Normal Feedwater

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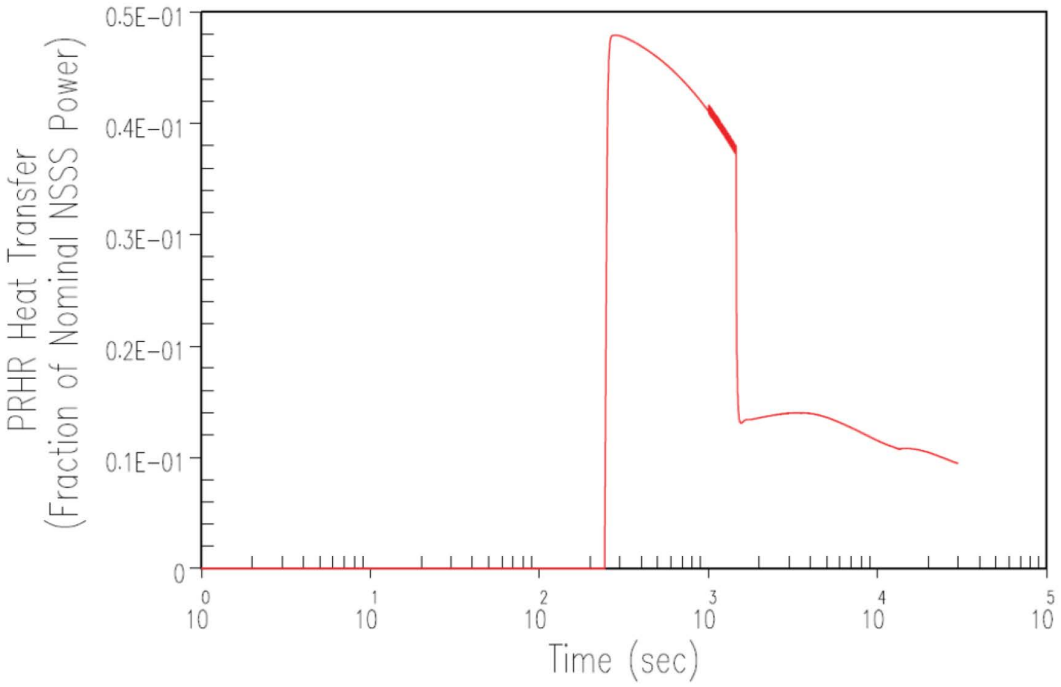

Figure 15.2.7-9
PRHR Heat Transfer Transient
for Loss of Normal Feedwater

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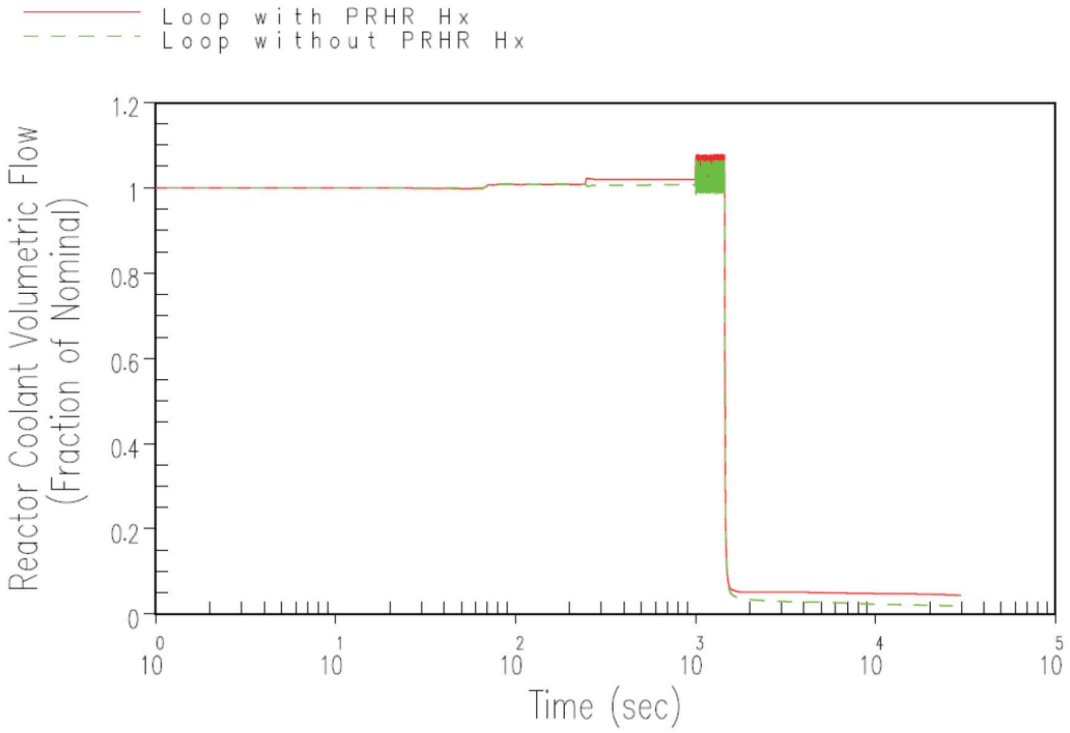

Figure 15.2.7-10
Reactor Coolant Volumetric Flow
Transient for Loss of Normal Feedwater

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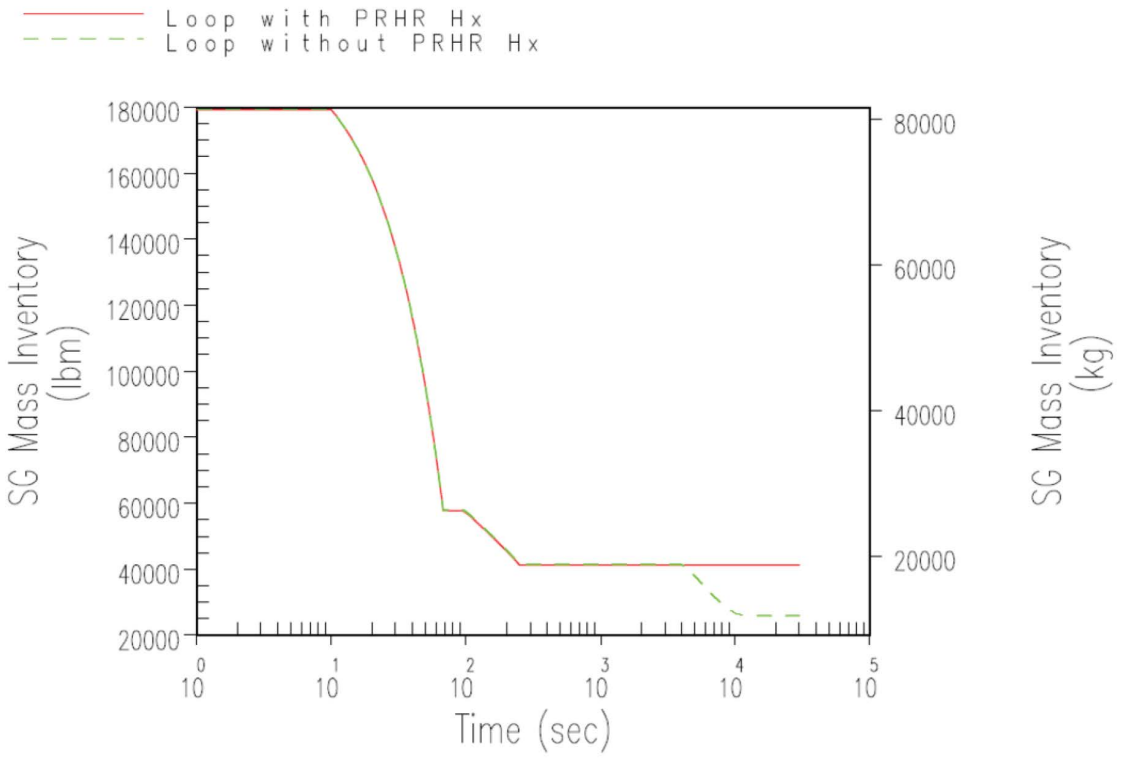

Figure 15.2.7-11
Steam Generator Inventory Transient for Loss of Normal Feedwater

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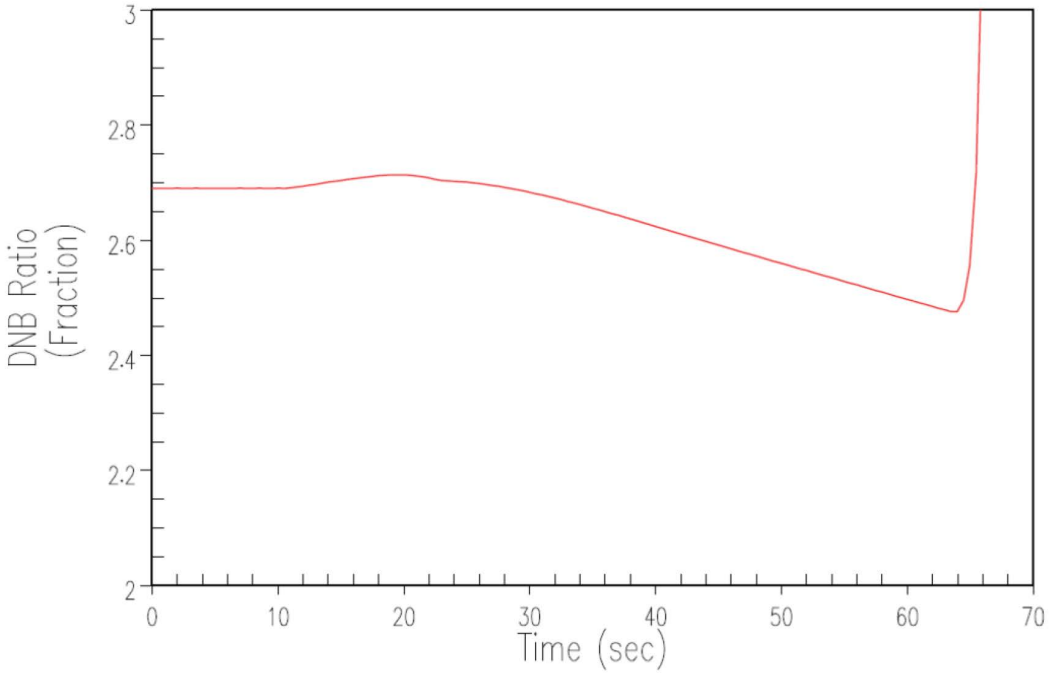

Figure 15.2.7-12
DNB Ratio Transient
for Loss of Normal Feedwater

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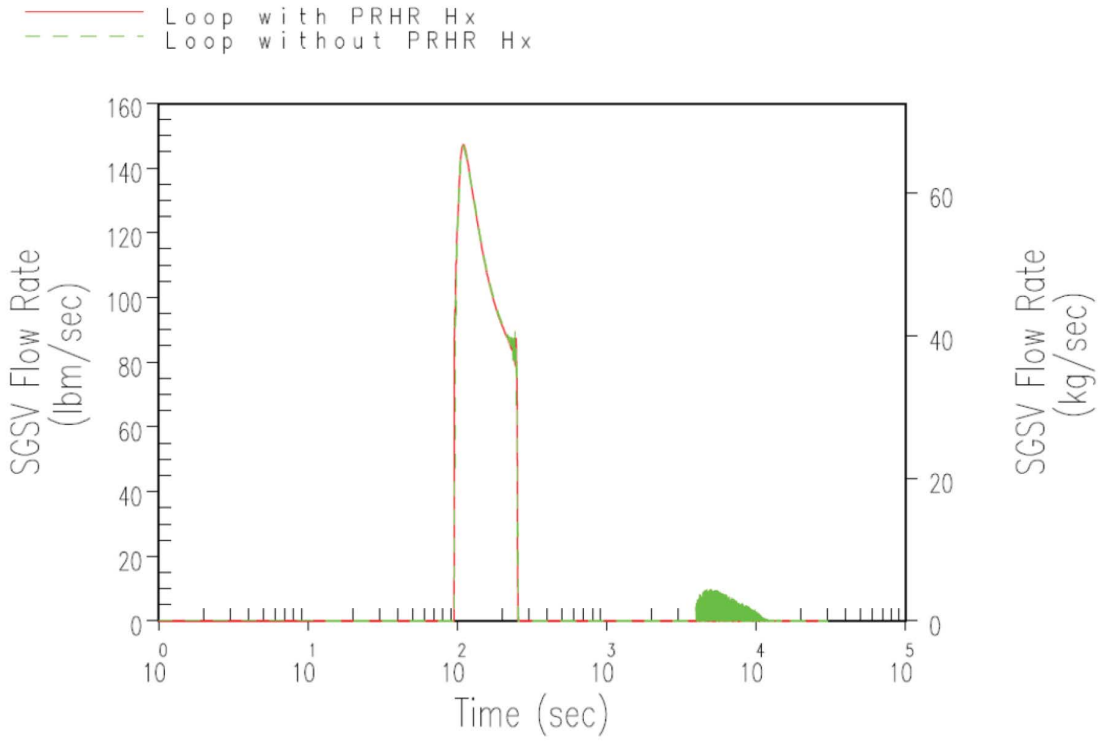

Figure 15.2.7-13
Steam Generator Safety Valve Relief Transient for Loss of Normal Feedwater

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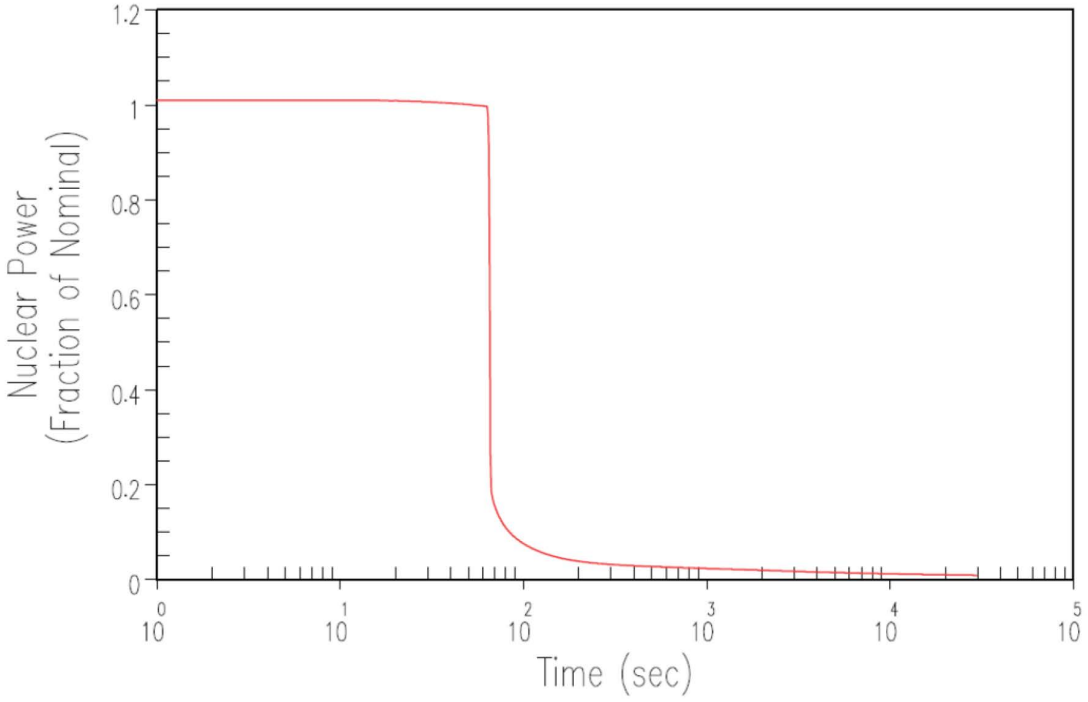

Figure 15.2.7-14
Nuclear Power Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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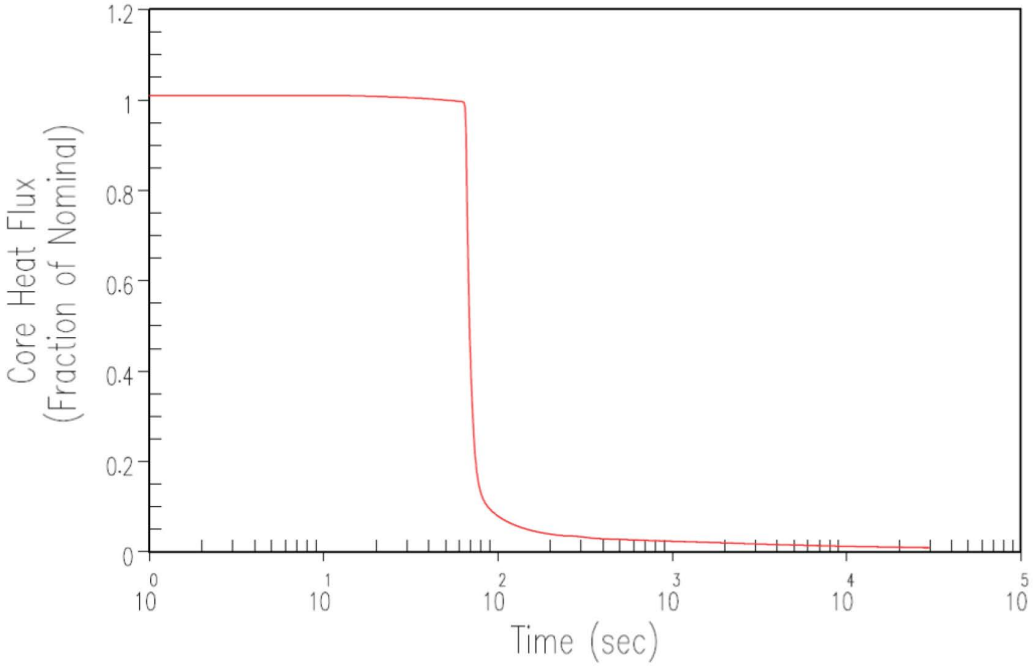

Figure 15.2.7-15
Core Heat Flux Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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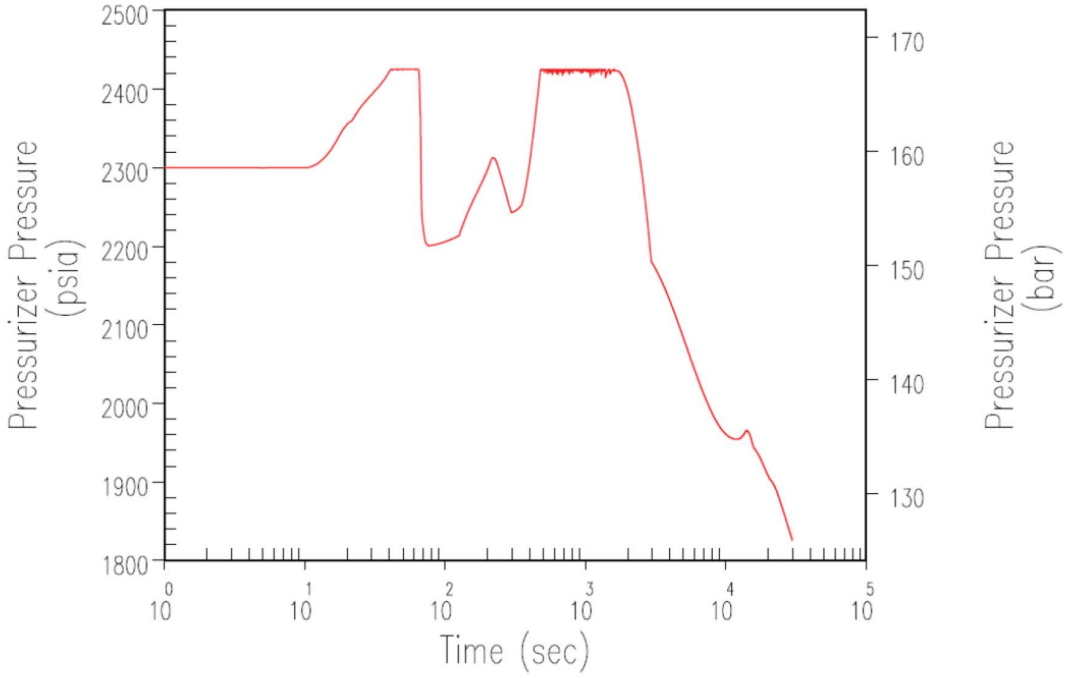

Figure 15.2.7-16
Pressurizer Pressure Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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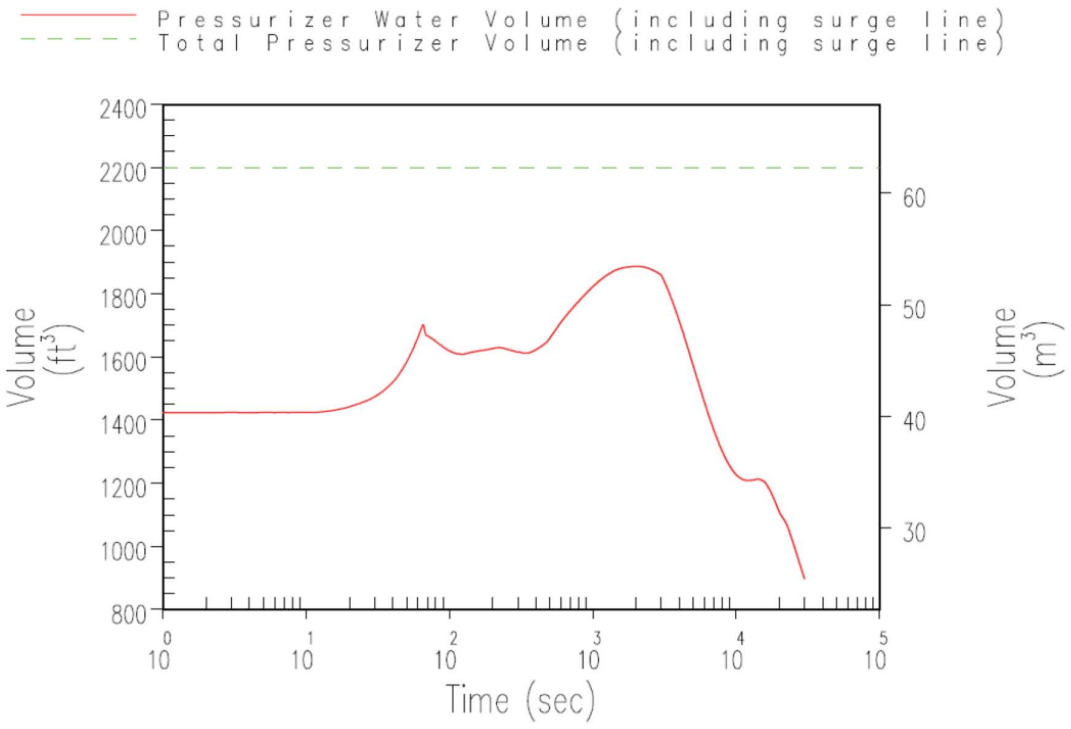

Figure 15.2.7-17
Pressurizer Water Volume Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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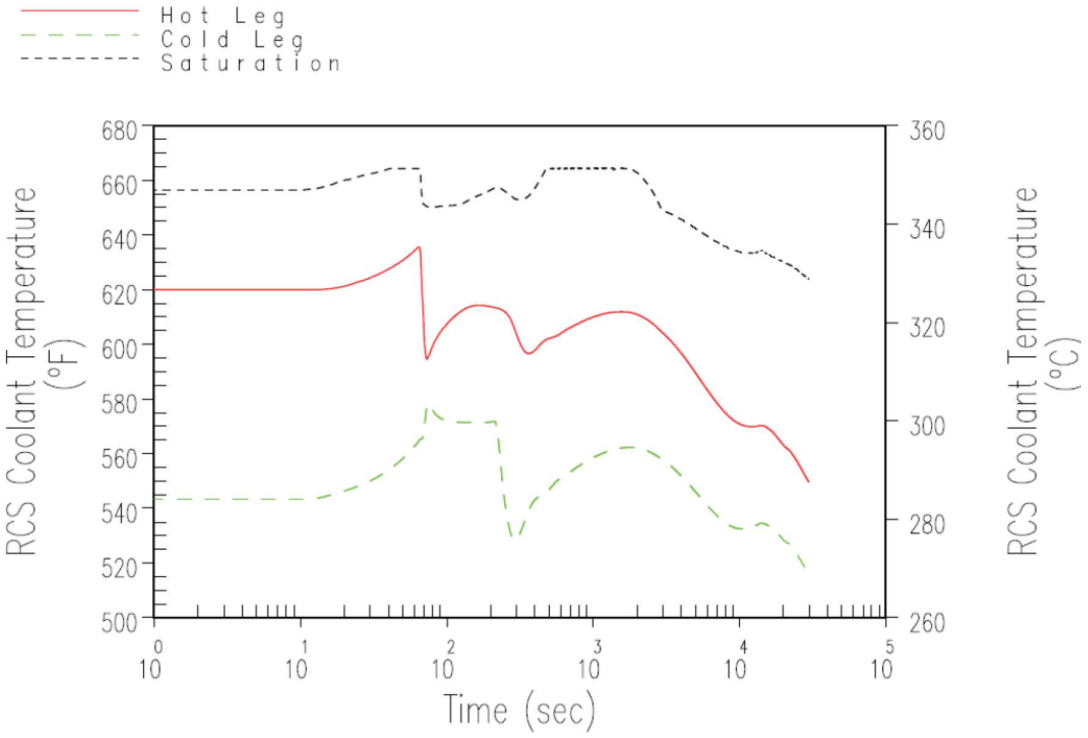

Figure 15.2.7-18
Reactor Coolant System Temperature Transients in Loop Containing the PRHR for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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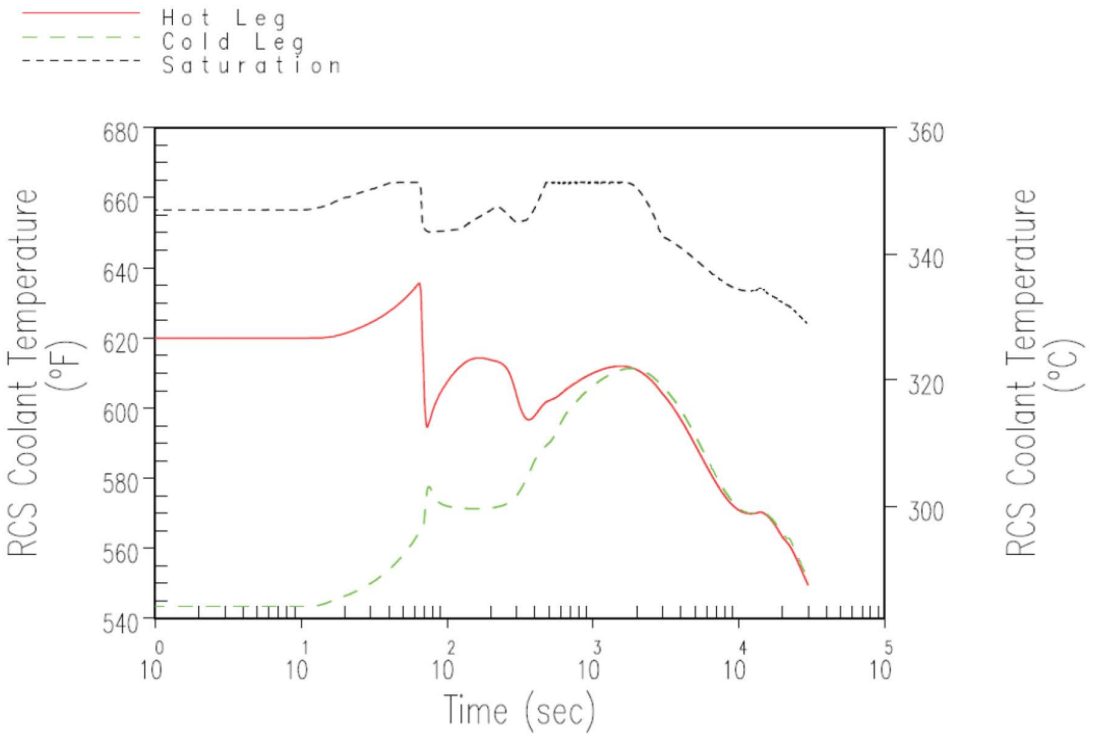

Figure 15.2.7-19
Reactor Coolant System Temperature Transients in Loop Not Containing the PRHR for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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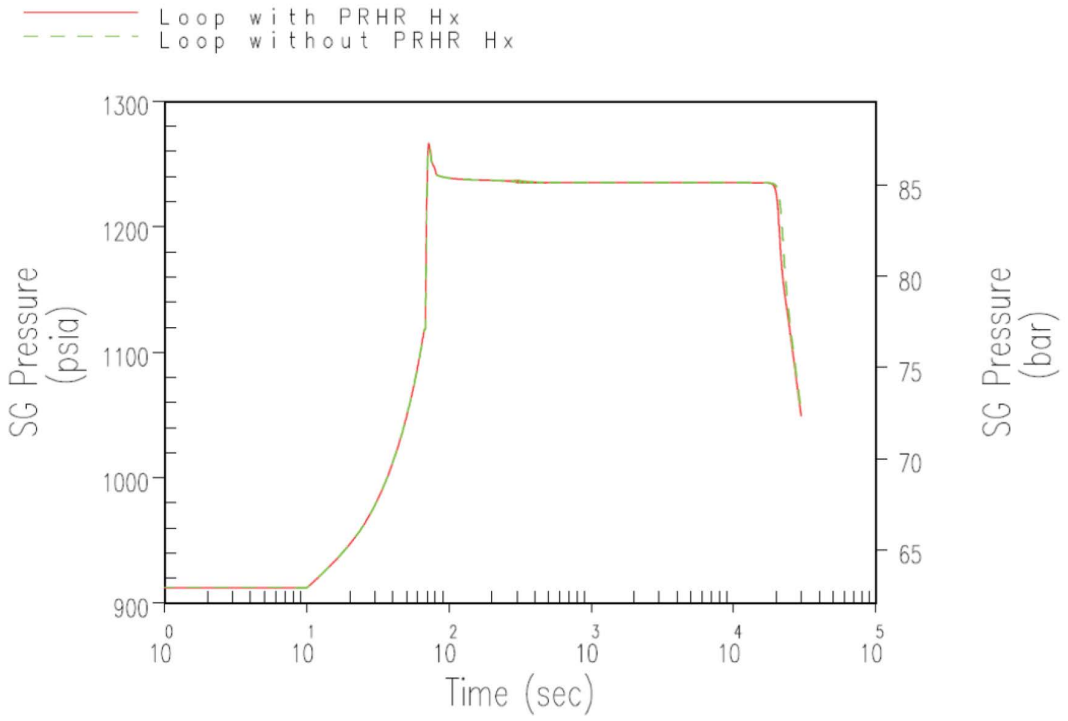

Figure 15.2.7-20
Steam Generator Pressure Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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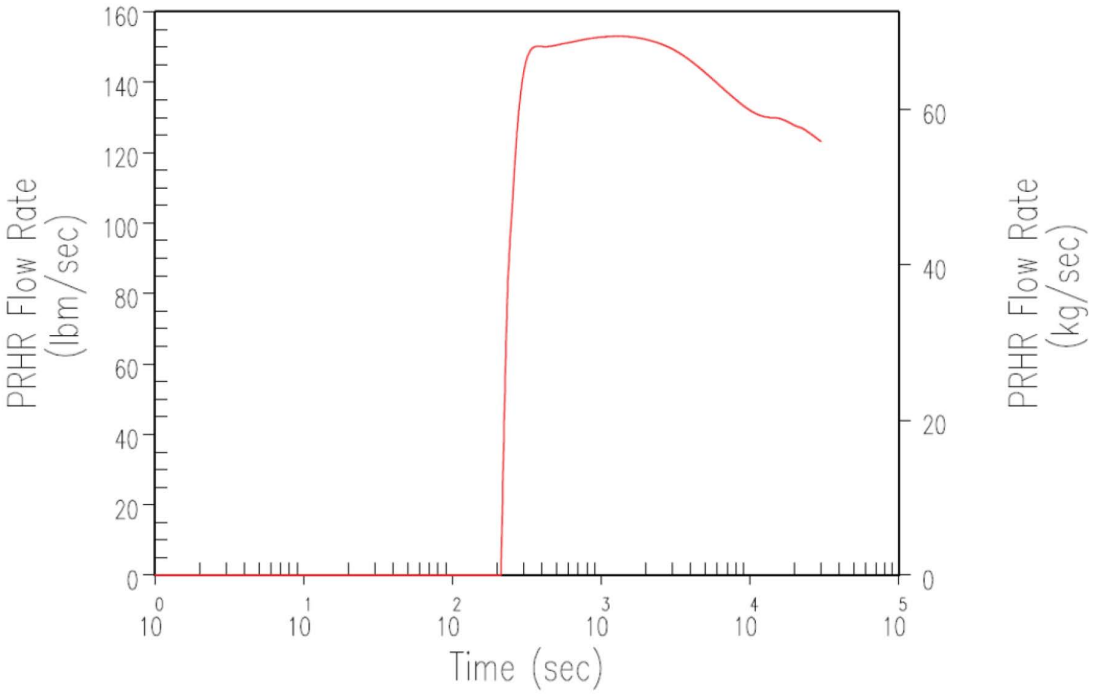

Figure 15.2.7-21
PRHR Flow Rate Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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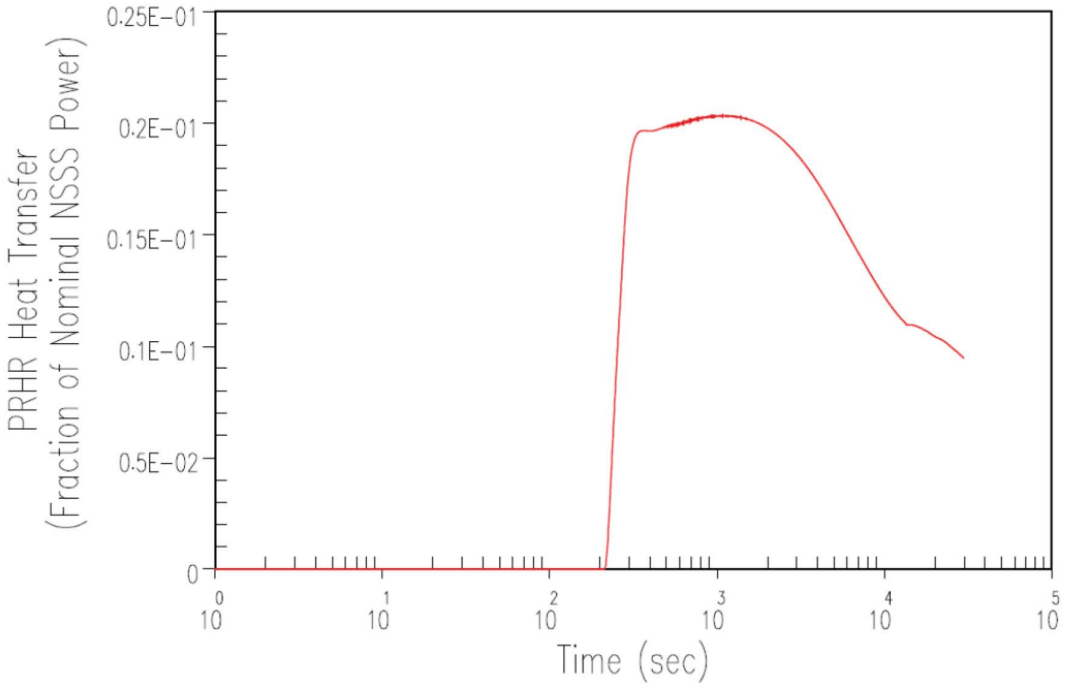

Figure 15.2.7-22
PRHR Heat Transfer Transient
for Loss of Normal Feedwater with a
Consequential Loss of ac Power to the Plant Auxiliaries

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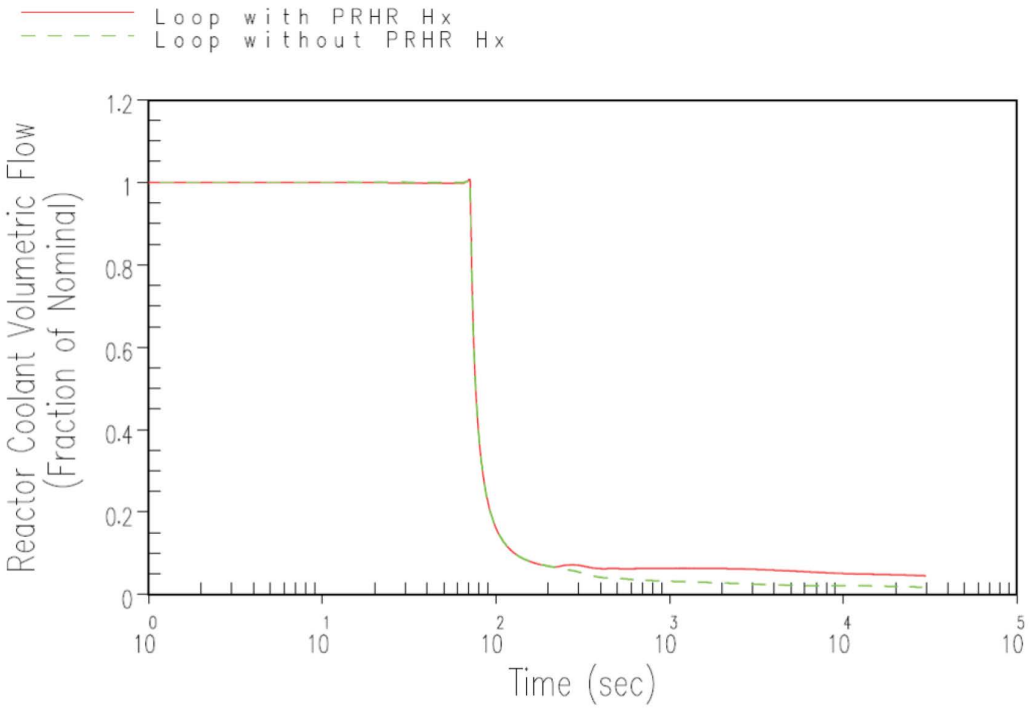

Figure 15.2.7-23
Reactor Coolant Volumetric Flow Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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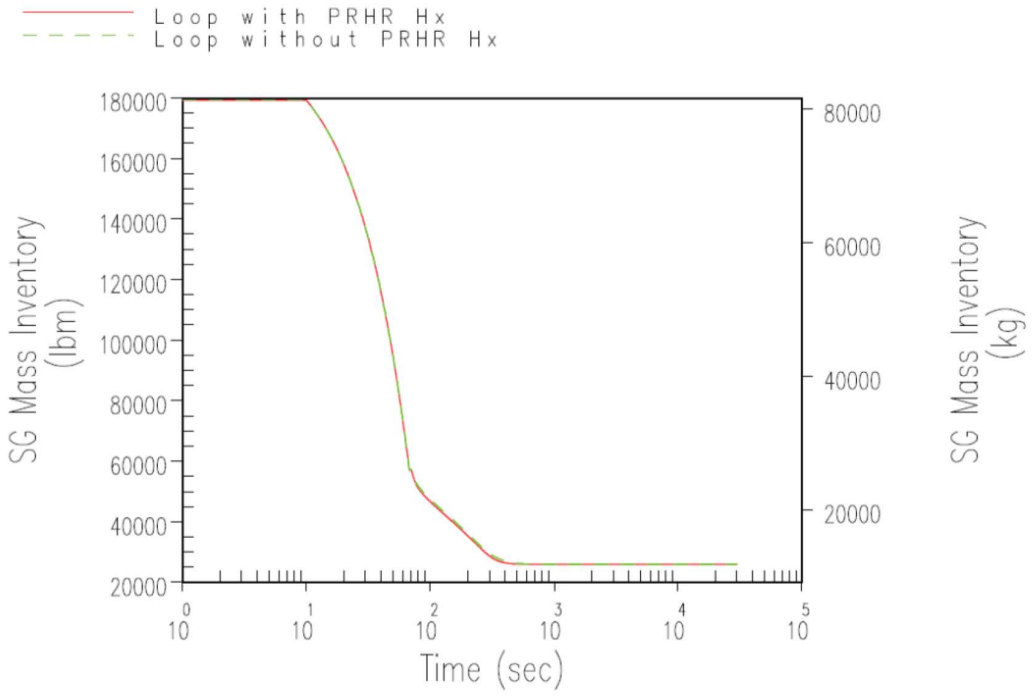

Figure 15.2.7-24
Steam Generator Inventory Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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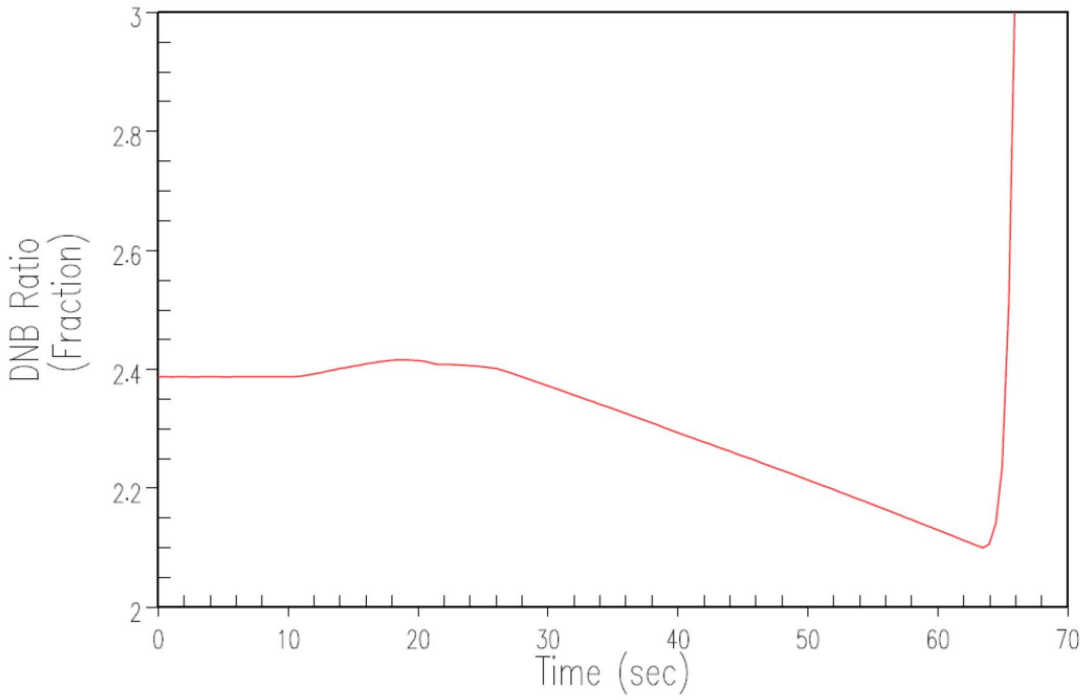

Figure 15.2.7-25
DNB Ratio Transient
for Loss of Normal Feedwater with a
Consequential Loss of ac Power to the Plant Auxiliaries

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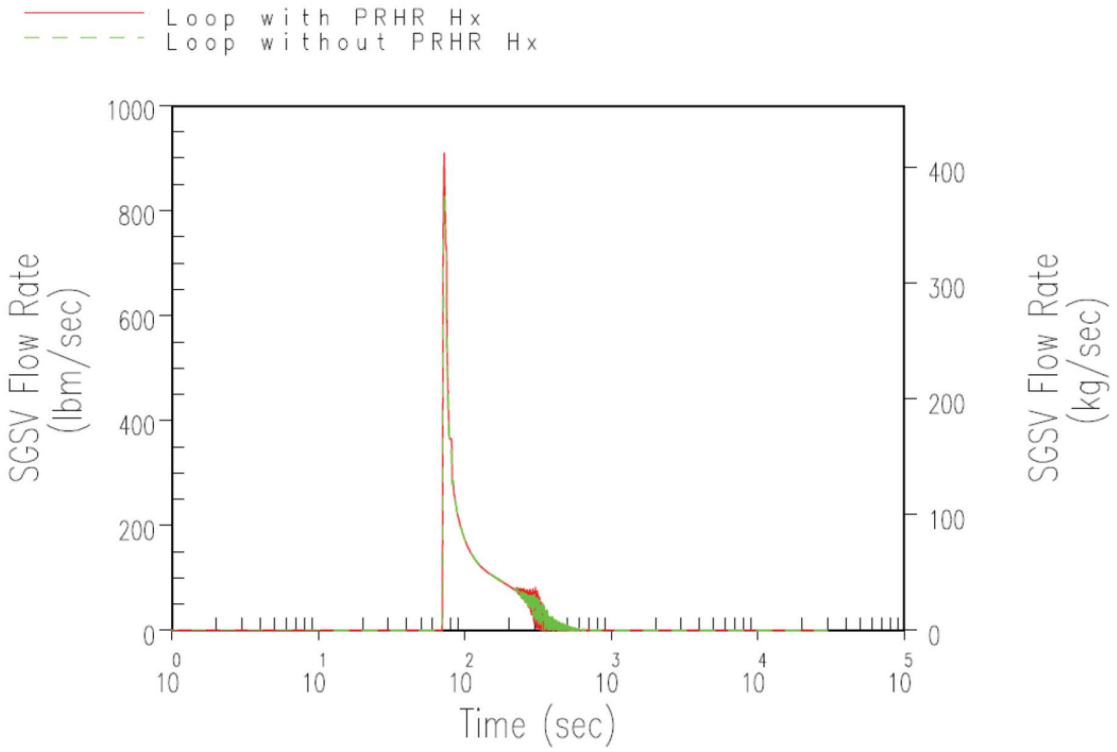

Figure 15.2.7-26
Steam Generator Safety Valve Relief Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

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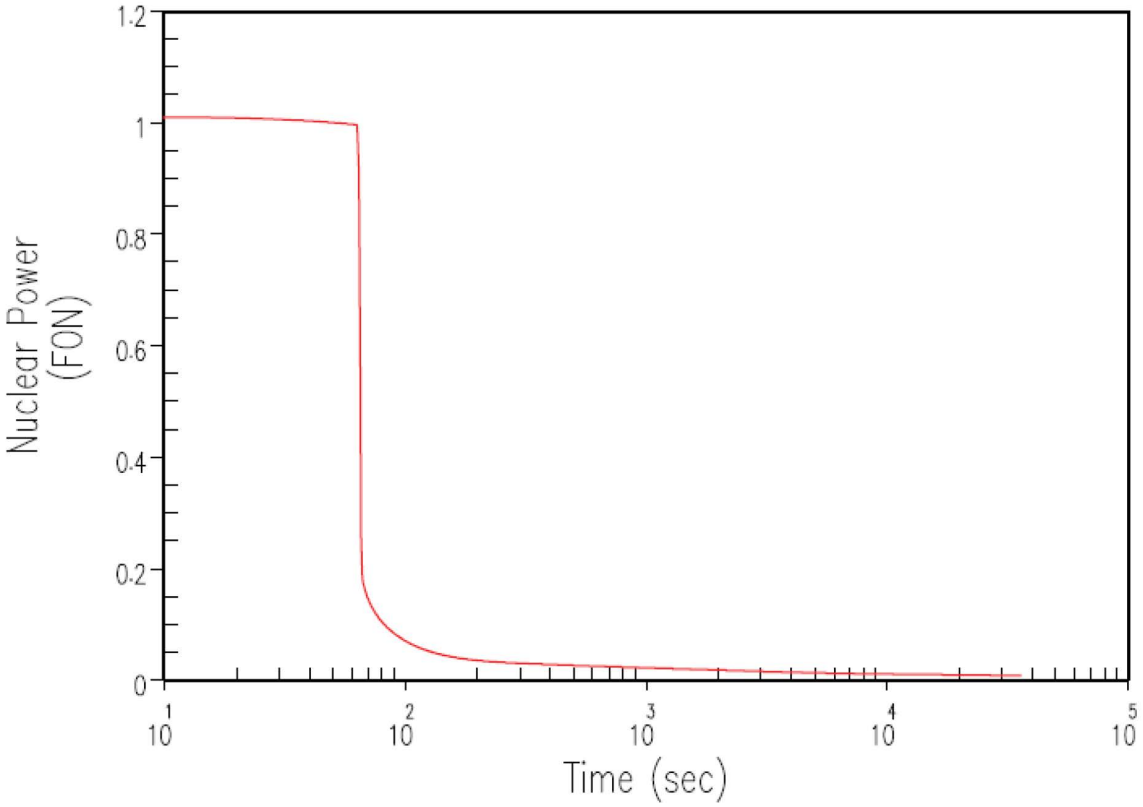

Figure 15.2.8-1
Nuclear Power Transient for
Main Feedwater Line Rupture

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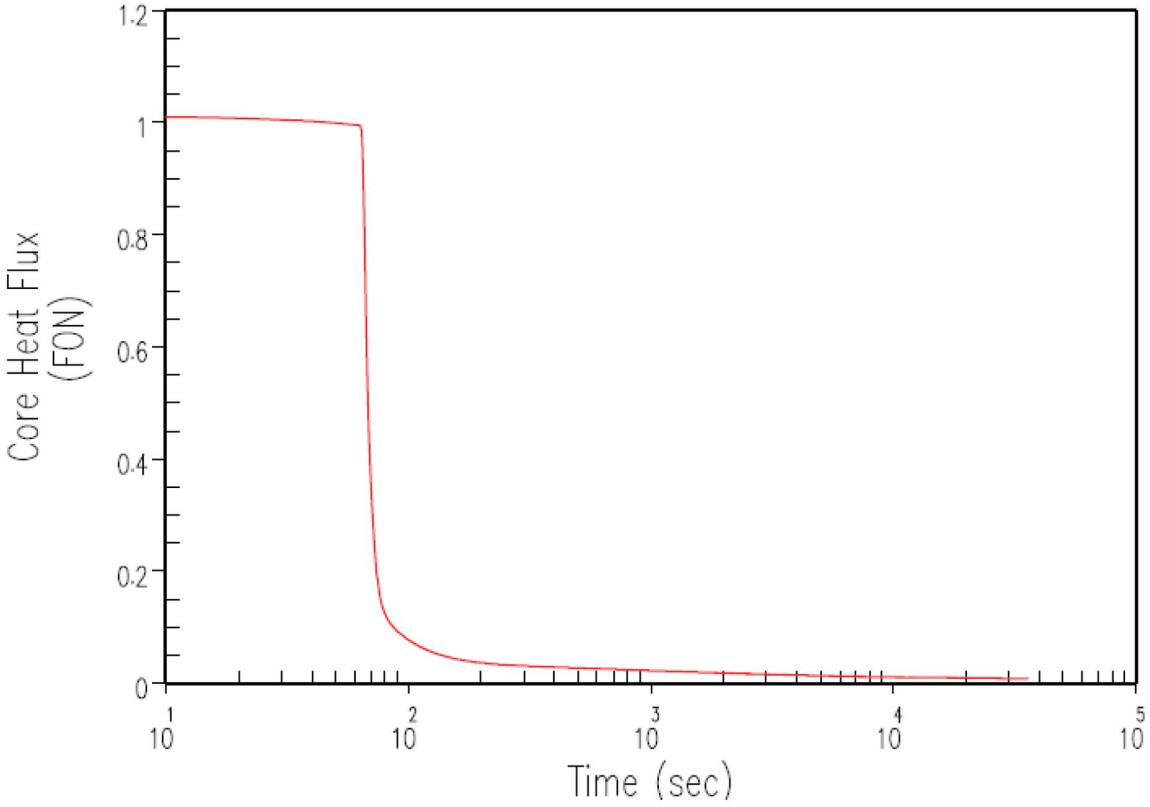

Figure 15.2.8-2
Core Heat Flux Transient for
Main Feedwater Line Rupture

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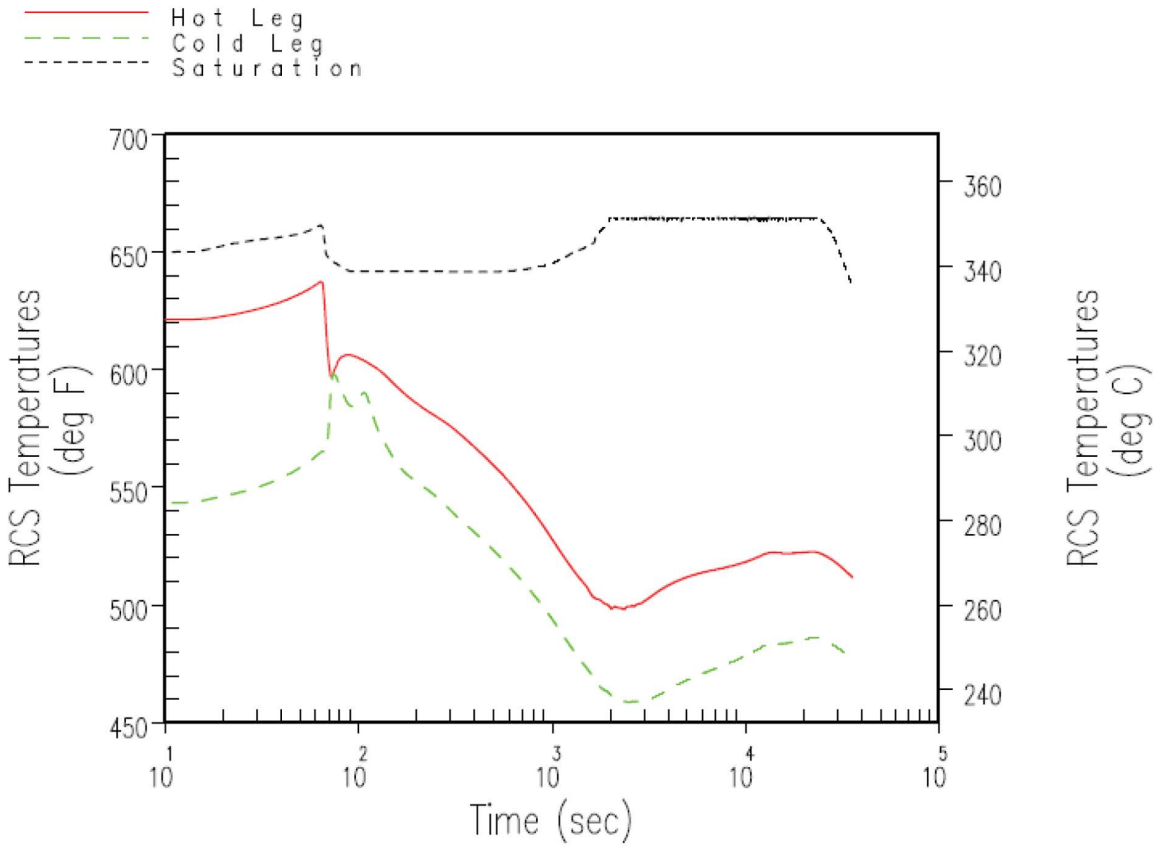

Figure 15.2.8-3
Faulted Loop Reactor Coolant System
Temperature Transients for Main Feedwater Line Rupture

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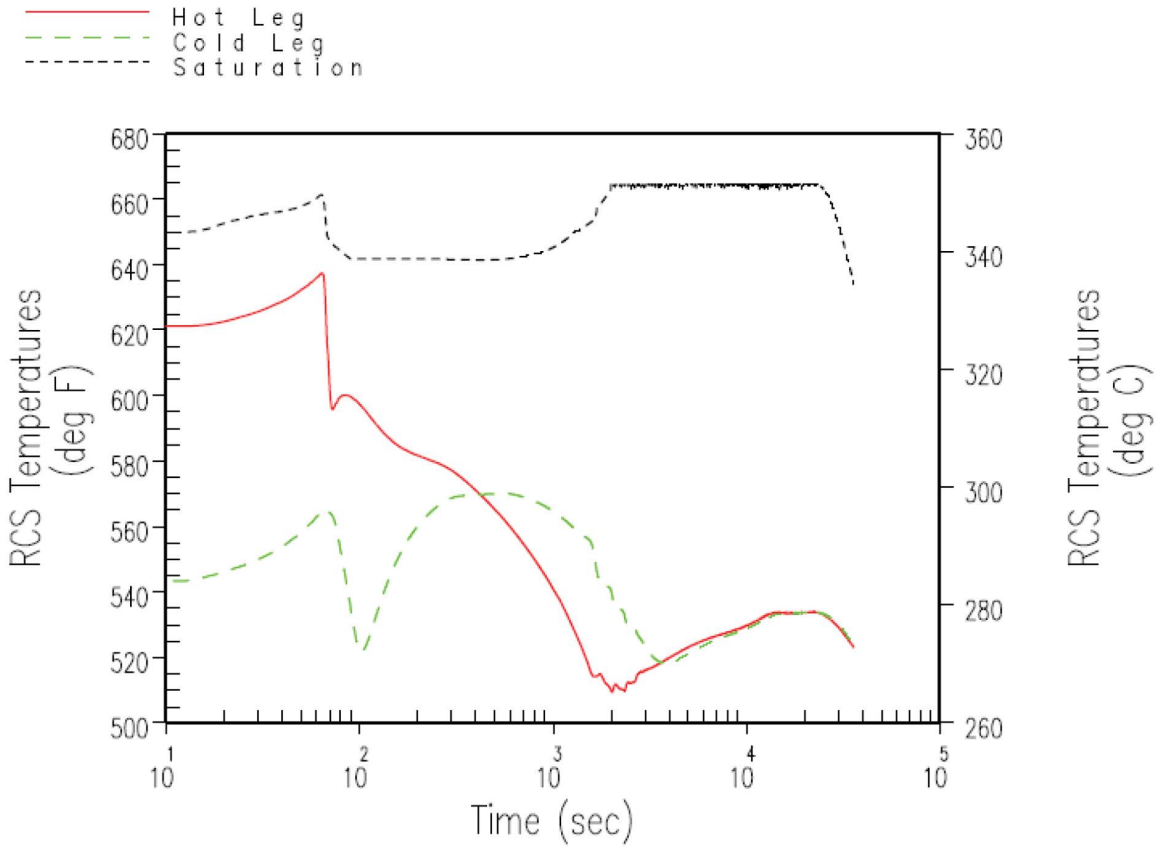

Figure 15.2.8-4
Intact Loop Reactor Coolant System
Temperature Transients for Main Feedwater Line Rupture

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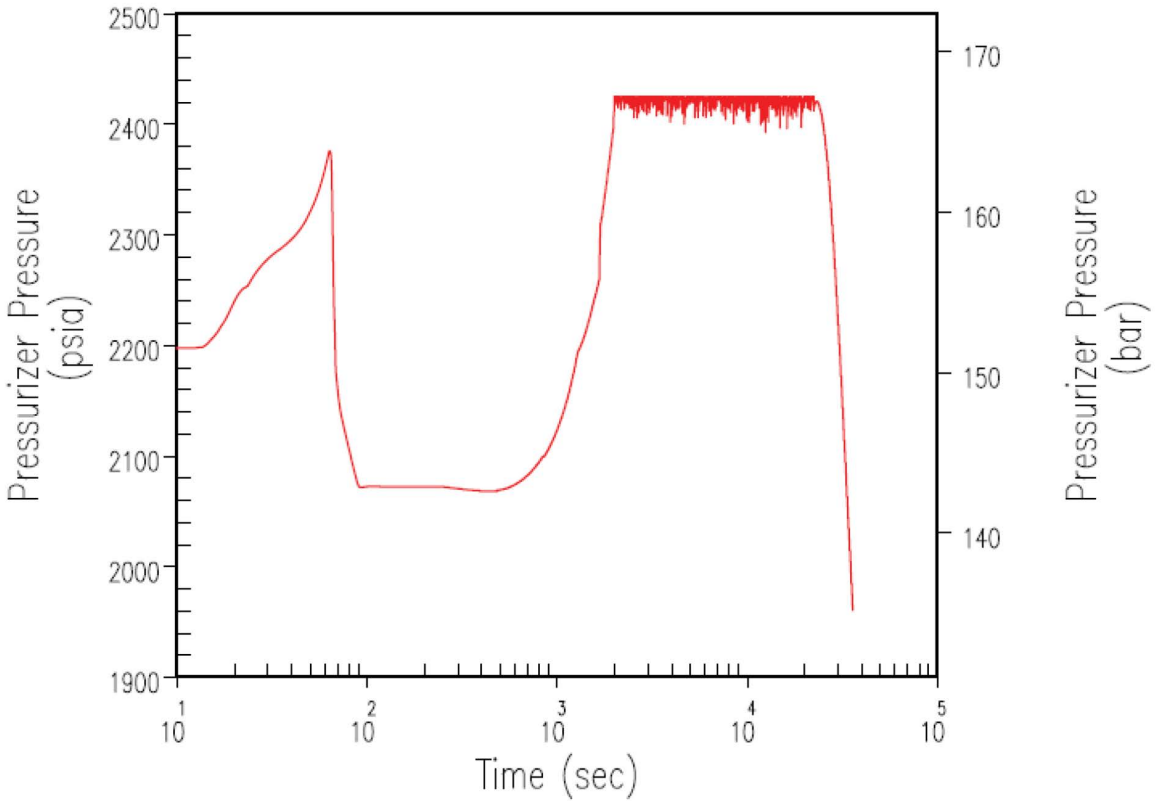

Figure 15.2.8-5
Pressurizer Pressure Transient for Main Feedwater Line Rupture

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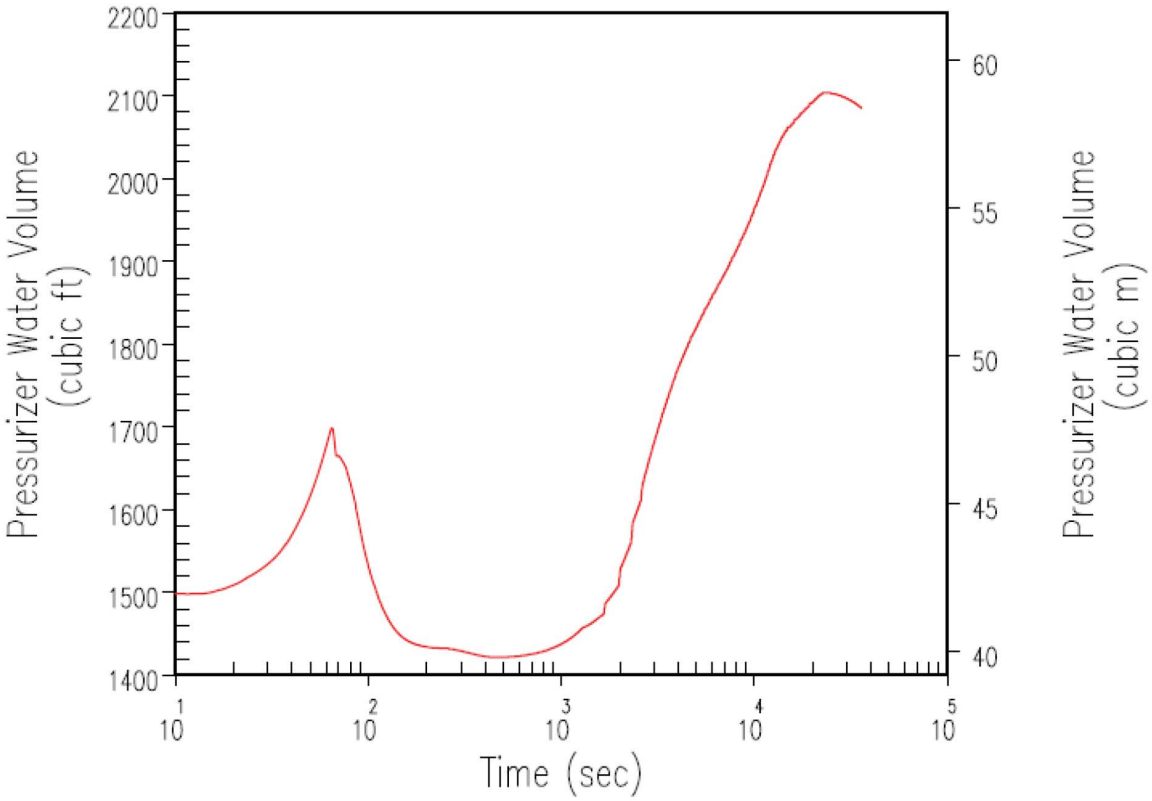

Figure 15.2.8-6
Pressurizer Water Volume Transient for
Main Feedwater Line Rupture

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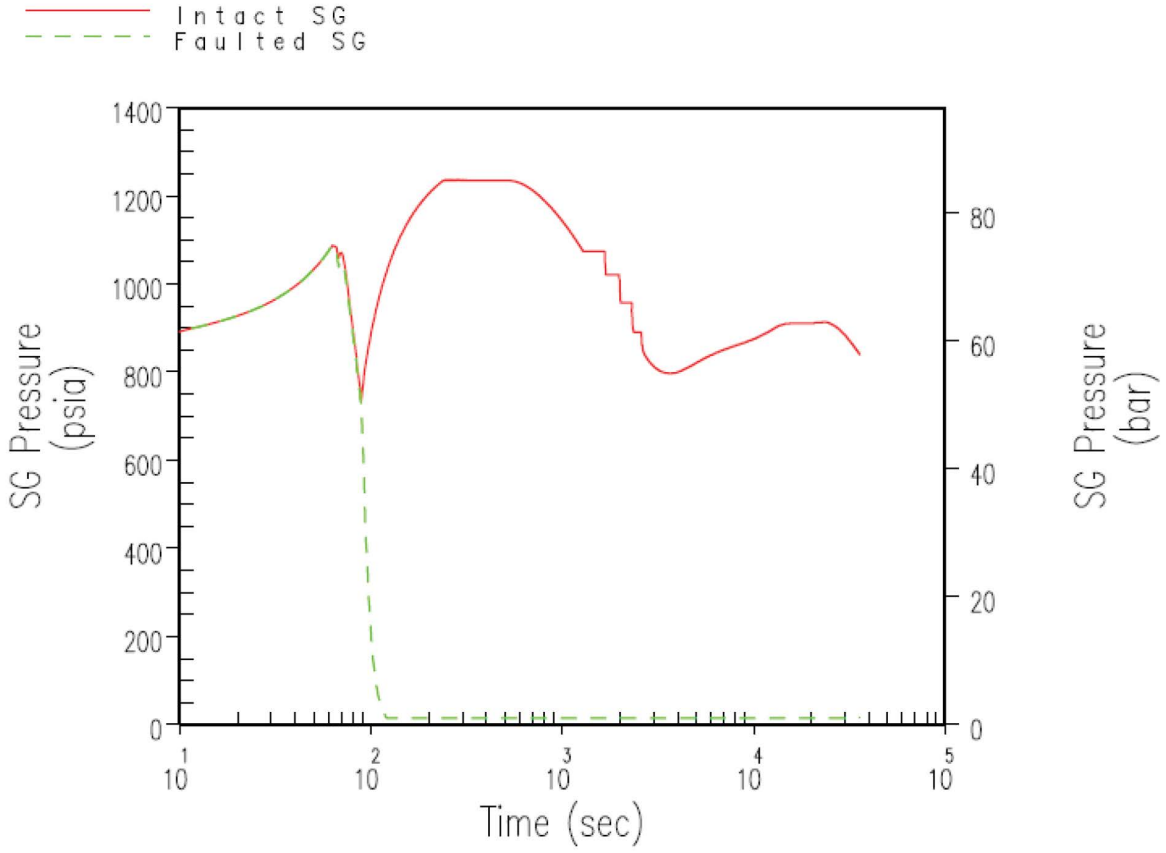

Figure 15.2.8-7
Steam Generator Pressure Transient for Main Feedwater Line Rupture

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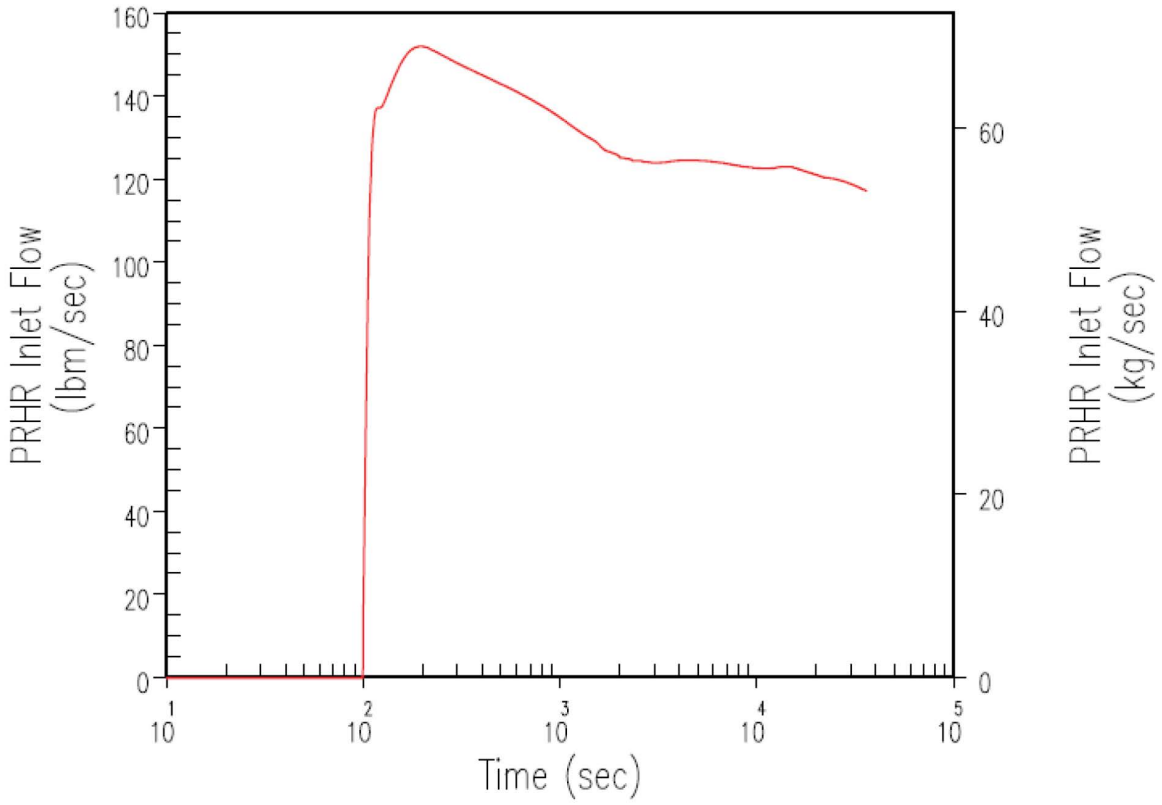

Figure 15.2.8-8
PRHR Flow Rate Transient for
Main Feedwater Line Rupture

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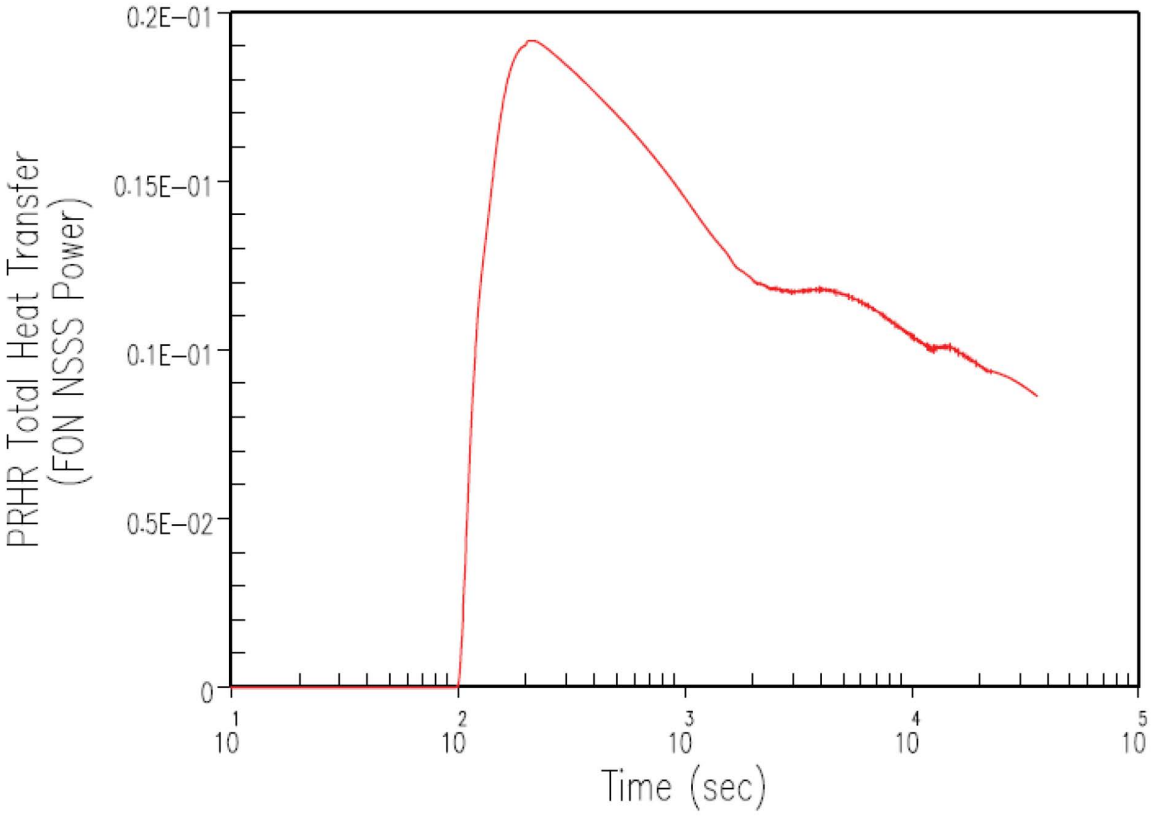

Figure 15.2.8-9
PRHR Heat Flux Transient for
Main Feedwater Line Rupture

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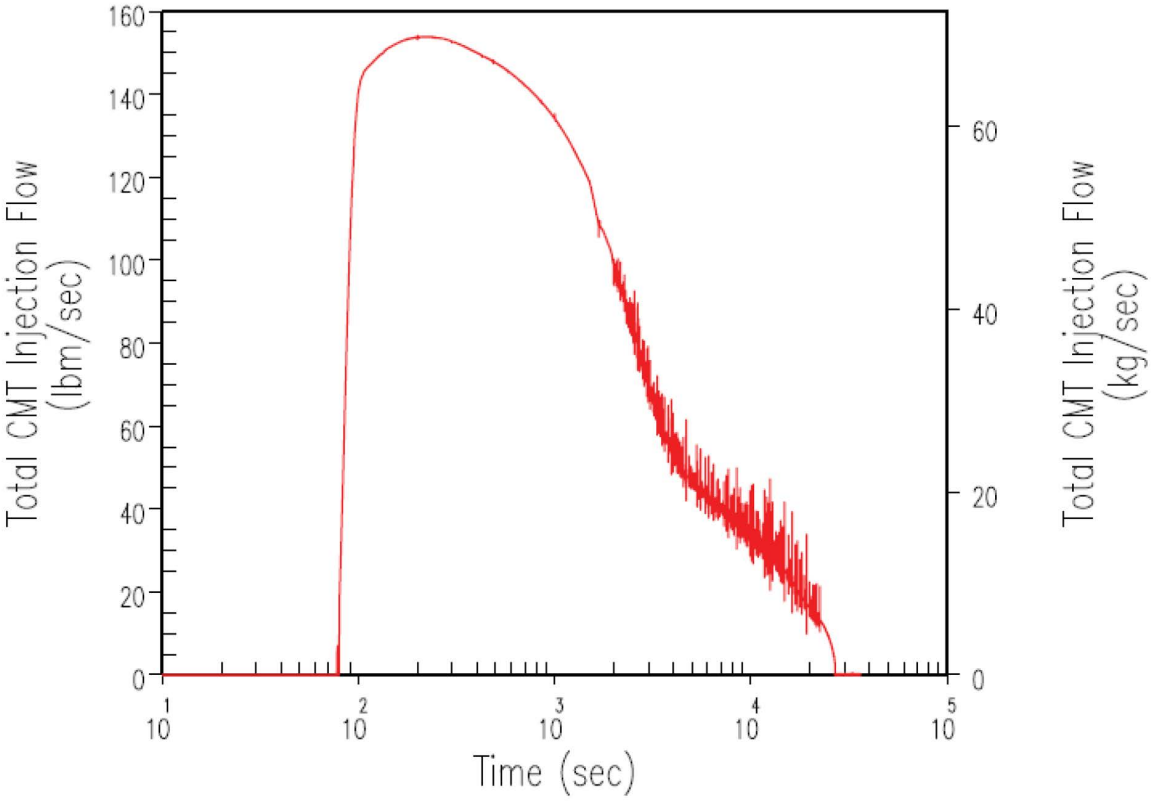

Figure 15.2.8-10
CMT Injection Flow Rate Transient for Main Feedwater Line Rupture

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15.3 Decrease in Reactor Coolant System Flow Rate

A number of faults that could result in a decrease in the reactor coolant system flow rate are postulated. These events are discussed in this section. Detailed analyses are presented for the most limiting of the following reactor coolant system flow decrease events:

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump shaft seizure (locked rotor)
- Reactor coolant pump shaft break

The first event is a Condition II event, the second is a Condition III event, and the last two are Condition IV events.

The four limiting flow rate decrease events described above are analyzed in this section. The most severe radiological consequences result from the reactor coolant pump shaft seizure accident discussed in Subsection 15.3.3. Doses are reported only for that case.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or an electrical failure of a reactor coolant pump or from a fault in the power supply to the pump or pumps. If the reactor is at power at the time of the event, the immediate effect of the loss of coolant flow is a rapid increase in the coolant temperature. For the AP1000 plant design, there are two potential partial loss of flow scenarios. These scenarios include the coast down of one reactor coolant pump and the coast down of two reactor coolant pumps in diametrically opposite loops. Although both scenarios are analyzed, the loss of two reactor coolant pumps bounds the loss of one pump, because it results in a more severe flow coast down. Thus, the two pump partial loss of flow is used as the basis for the discussion within this section.

Normal power for the pumps is supplied through four buses connected to the generator. When a generator trip occurs, the buses are supplied from offsite power and the pumps continue to operate.

A partial loss of coolant flow is classified as a Condition II incident (a fault of moderate frequency), as defined in Subsection 15.0.1.

Protection against this event is provided by the Low-2 primary coolant flow reactor trip signal, which is actuated by two-out-of-four Low-2 flow signals. Above permissive P10, Low-2 flow in either hot leg actuates a reactor trip (see Section 7.2).

As specified in GDC 17 of 10 CFR Part 50, Appendix A, the effects of a loss of offsite power are considered in evaluating partial loss of forced reactor coolant flow transients. As discussed in Subsection 15.0.14, the loss of offsite power is considered to be a potential consequence of the event due to disruption of the electrical grid following a turbine trip during the event. A delay of 3 seconds is assumed between the turbine trip and the loss of offsite power. In addition, turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system. The primary effect of the loss of offsite power is to cause the remaining operating reactor coolant pumps to coast down. However, since the loss of offsite power would occur no earlier than 8 seconds into the event, it is well beyond the critical time frame of interest for the partial loss of flow events (i.e., time of rod insertion). Thus, it is not explicitly modeled in the case runs.

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15.3.1.2 Analysis of Effects and Consequences

15.3.1.2.1 Method of Analysis

This transient is analyzed using three computer codes. First, the LOFTRAN code (References 1 and 8) is used to calculate the core flow during the transient based on the input loop flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code (Reference 2) or the VIPRE-01 fuel rod model (Reference 7), which is equivalent to FACTRAN, is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The calculated DNBR transient represents the minimum of the typical cell or the thimble cell. Note that an explicit DNBR calculation is only performed for the more limiting complete loss of flow event described in Subsection 15.3.2.

15.3.1.2.2 Initial Conditions

Initial reactor power, pressurizer pressure, and reactor coolant system temperature are assumed to be at their nominal values. Uncertainties in initial conditions are statistically accounted for in the DNBR limit, as described in WCAP-11397-P-A (Reference 5).

Plant characteristics and initial conditions assumed in this analysis are further discussed in Subsection 15.0.3.

15.3.1.2.3 Reactivity Coefficients

The reactivity feedback parameters are chosen so as to maximize the energy transferred to the primary coolant during the flow coastdown. A most-negative Doppler-only power coefficient (see Figure 15.0.4-1) is applied to maximize the positive reactivity addition during the reactor trip and rod motion, which acts to slow the rate of power reduction; the equivalent total integrated Doppler reactivity from 0 to 100 percent power of 0.016 Δk. As there is an initial heatup due to the reduction in RCS flow, a least-negative (minimum feedback) moderator temperature coefficient is most conservative. Therefore, a constant moderator density coefficient of 0.0 Δk/g/cc is modeled. Finally, a curve of trip reactivity versus time based on a 2.7-second rod cluster control assembly insertion time to the dashpot is applied (see Subsection 15.0.5).

15.3.1.2.4 Flow Coastdowns

Conservative flow coastdowns are used to simulate the transient. The flow coastdowns are calculated externally to the LOFTRAN code using the COAST computer code which is described in Subsection 15.0.11.

15.3.1.2.5 Protection Systems

Plant systems and equipment necessary to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.1.2.6 Results

Figures 15.3.1-1 through 15.3.1-5 show the transient response for the loss of two reactor coolant pumps with offsite power available. The partial loss of flow event is evaluated to be bounded by the complete loss of flow event described in Subsection 15.3.2, for which the analysis demonstrates that

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the DNBR is always greater than the safety analysis limit value. This demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4.

The affected reactor coolant pumps coast down and the core flow reaches a new equilibrium value. The plant is tripped by the Low-2 flow trip rapidly enough so that the capability of the reactor coolant to remove heat from the fuel rods is not greatly reduced. The average fuel and cladding temperatures do not increase significantly above their initial values. With the reactor tripped, a stable plant condition is attained and plant shutdown may then proceed.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1.

In the event that a loss of offsite power occurs as a consequence of a turbine trip during a partial loss of reactor coolant flow, the DNB design basis continues to be met as discussed in Subsection 15.3.1.1.

15.3.1.3 Conclusions

The analysis shows that, for the partial loss of reactor coolant flow, the DNBR does not decrease below the safety analysis limit value at any time during the transient, which demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4. The applicable Standard Review Plan, Subsection 15.3.1 (Reference 4), evaluation criteria are met.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

A complete loss of flow accident may result from a simultaneous loss of electrical supplies to the reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature. Electric power for the reactor coolant pumps is normally supplied through buses, connected to the generator through the unit auxiliary transformers. When a generator trip occurs, the buses receive power from external power lines and the pumps continue to supply coolant flow to the core.

A complete loss of flow accident is a Condition III event (an infrequent fault), as defined in Subsection 15.0.1. The following signals provide protection against this event:

1. Low-2 Reactor coolant pump speed
2. Low primary coolant loop flow

The reactor trip on Low-2 reactor coolant pump speed protects against conditions that can cause a loss of voltage to two-out-of-four reactor coolant pumps. This function is blocked below approximately 10-percent power (permissive P10). The reactor trip on Low-2 reactor coolant pump speed also protects against an underfrequency condition resulting from frequency disturbances on the power grid, so long as the maximum grid frequency decay rate is less than approximately 5 hertz per second. WCAP-8424, Revision 1 (Reference 3), provides analyses of grid frequency disturbances and the resulting protection requirements that are applicable to the AP1000 plant design.

15.3.2.2 Analysis of Effects and Consequences

15.3.2.2.1 Method of Analysis

The complete loss of flow transient is analyzed for a loss of power to four reactor coolant pumps.

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For the scenario of a complete loss of voltage, which results in all the reactor coolant pumps coasting down, the method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Subsection 15.3.1, with three exceptions. Following the loss of power supply to all pumps at power, a reactor trip is actuated by the Low-2 reactor coolant pump speed trip instead of the Low-2 primary coolant flow trip. Also, rather than the bounding value of 0.0 Δk/g/cc, a less limiting, yet still conservative, moderator density coefficient (MDC) curve (MDC as a function of coolant density) is modeled. Finally, a curve of trip reactivity versus time based on a 2.3-second rod cluster control assembly insertion time to the dashpot is applied (see Subsection 15.0.5).

A complete loss of forced primary coolant flow can result from a reduction in the reactor coolant pump motor supply frequency. However, the results of the complete loss of voltage scenario (i.e., free spinning pump coastdown) bound the results of the complete loss of flow initiated by a frequency decay of up to 5 hertz per second. This is due to the reactor coolant pump design, which initially (during the critical time frame of the transient) has a more rapid coastdown as a free spinning pump than for an electrical frequency decay. Therefore, only the results of the complete loss of voltage case scenario are presented in Subsection 15.3.2.2.2.

15.3.2.2.2 Results

Figures 15.3.2-1 through 15.3.2-6 show the transient response for the complete loss of voltage to all four reactor coolant pumps. The reactor is tripped on the Low-2 reactor coolant pump speed signal. Figure 15.3.2-6 demonstrates that the DNBR is always greater than the safety analysis limit value which demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. With respect to DNB concerns, the event is essentially over shortly after reactor trip. However, if the event was extended beyond the time frame analyzed for DNB, the reactor coolant pumps would continue to coast down, and natural circulation flow would be established, as demonstrated in Subsection 15.2.6. With the reactor tripped, a stable plant condition is attained and plant shutdown may then proceed.

15.3.2.3 Conclusions

The analysis demonstrates that, for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit value at any time during the transient, which demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4. The applicable Standard Review Plan, Subsection 15.3.1 (Reference 4), evaluation criteria are met.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a Low-2 flow signal.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant temperature to increase and expand. At the same time, heat transfer to the shell side of the steam generator in the faulted loop is reduced because: 1) the reduced flow results in a decreased tube-side film coefficient, and 2) the reactor coolant in the tubes cools down while the shell-side temperature increases. (Consistent with the AP1000 design, the peak pressure and fuel rod thermal analyses assume a 5 second delay in turbine trip following reactor trip.) The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam

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generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

This event is classified as a Condition IV incident (a limiting fault), as defined in Subsection 15.0.1.

15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN code (Reference 1) calculates the resulting core flow transient following the pump seizure and the nuclear power following reactor trip. This code is also used to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated by using the FACTRAN code (Reference 2) or the VIPRE-01 fuel rod model (Reference 7) which is equivalent to FACTRAN. This fuel thermal calculation uses the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes a film-boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (at the time the shaft in one of the reactor coolant pumps is assumed to seize), the plant is assumed to be in operation under the most adverse steady-state operating conditions, that is, maximum steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature. Plant characteristics and initial conditions are further discussed in Subsection 15.0.3. A curve of trip reactivity versus time based on a 2.3-second rod cluster control assembly insertion time to the dashpot is applied (see Subsection 15.0.5). The accident is evaluated for both cases with and without offsite power available. For the case without offsite power available, power is lost to the unaffected pumps at 3.0 seconds following turbine/generator trip. Turbine trip occurs 5.0 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 50 psi above nominal pressure (2250 psia), which allows for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure.

15.3.3.2.2 Evaluation of the Pressure Transient and Fuel Rod Thermal Design Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1.45 seconds after the flow in the affected loop reaches the reactor trip setpoint. No credit is taken for the pressure-reducing effect of the pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to result in a lower peak reactor coolant system pressure, an additional conservatism is provided by ignoring their effect.

The pressurizer safety valves are fully open at 2575 psia. Their capacity for steam relief is described in Section 5.4.

For this accident, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this “hot spot” condition represent the upper limit with respect to cladding temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be 3 times the average rod power (that is, $F_Q = 3.0$) at the initial core power level.

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15.3.3.2.3 Evaluation of Departure from Nucleate Boiling in the Core During the Accident

An analysis is performed to determine the percentage of fuel rods that experience DNB. The percentage is determined to be less than the limit value used for the fraction of fuel rods that are predicted to experience a DNB in the radiological consequences calculations reported in Subsection 15.3.3.3.

15.3.3.2.4 Film-Boiling Coefficient

The film-boiling coefficient is calculated in the FACTRAN code (Reference 2) using the Bishop-Sandberg-Tong film-boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient because they are the most conservative with respect to cladding temperature response. For conservatism, DNB is assumed to start at the beginning of the accident.

15.3.3.2.5 Fuel Cladding Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and the cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient is assumed to increase from a steady-state value consistent with initial fuel temperature to 10,000 Btu/h-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value of the gap coefficient is released to the cladding at the initiation of the transient.

15.3.3.2.6 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above a cladding temperature of 1800°F. The Baker-Just parabolic rate equation is used to define the rate of the zirconium-steam reaction:

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45{,}500}{1.986\,T}\right)$$

where:

w = amount reacted (mg/cm^2)

t = time (s)

T = temperature (Kelvin)

The reaction heat is 1510 cal/g.

The effect of the zirconium-steam reaction is included in the calculation of the hot spot cladding temperature transient.

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Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.3.2.7 Results

Figures 15.3.3-1 through 15.3.3-7 show the transient results for one locked rotor with four reactor coolant pumps in operation. The without-offsite-power case bounds the results for the case with offsite power. The results of these calculations are also summarized in Table 15.3-2. The peak reactor coolant system pressure reached during the transient is less than that which causes stresses to exceed the faulted condition stress limits of the ASME Code, Section III. Also, the peak cladding surface temperature is considerably less than 2700°F. The cladding temperature is conservatively calculated, assuming that DNB occurs at the initiation of the transient. These results represent the most limiting conditions with respect to the locked rotor event or the pump shaft break.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. With the reactor tripped, a stable plant condition is eventually attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated locked reactor coolant pump rotor accident assumes that the reactor has been operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant. Refer to Subsection 15.3.3.3.1 and Table 15.3-3.

As a result of the accident, it is determined that no fuel rods are damaged such that the activity contained in the fuel-cladding gap is released to the reactor coolant. However, a conservative analysis has been performed assuming 10 percent of the rods are damaged. Activity carried over to the secondary side because of primary-to-secondary leakage is available for release to the environment via the steam line safety valves or the power-operated relief valves.

15.3.3.3.1 Source Term

The significant radionuclide releases due to the locked rotor accident are the iodines, alkali metals (cesiums, rubidiums) and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant alkali metal concentrations are based on those associated with the design basis fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of the gap inventory of fission products from the portion of the core assumed to fail because of the accident.

Based on NUREG-1465 (Reference 6), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fraction is increased to 8 percent of the inventory for I-131, 10 percent for Kr-85, 5 percent for other iodines and noble gases and 12 percent for alkali metals. Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor.

The initial secondary coolant activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

15.3.3.3.2 Release Pathways

There are two components to the accident releases:

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- The activity initially in the secondary coolant is available for release as long as steam releases continue.

- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant. As steam is released, a portion of the iodine and alkali metal activity in the coolant is released. The fraction of activity released is defined by the assumed flashing fraction and the partition coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.

Credit is taken for the decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.3.3.3.3 Dose Calculation Models

The models used to calculate offsite doses are provided in Appendix 15A.

15.3.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.3-3.

Two separate accident scenarios are addressed. In the first scenario, it is assumed that the non-safety grade startup feedwater system is not available to provide feedwater to the steam generators. In this event, the water level in the steam generators drops, resulting in tube uncovery and there is flashing of a portion of the primary coolant assumed to be leaking into the secondary side of the steam generators. Also, the period of steaming is terminated at 1.5 hours when the capacity of the passive residual heat removal system exceeds the decay heat generation rate.

In the second scenario, it is assumed that the startup feedwater system is available to maintain water level in the steam generators such that the tubes remain covered. In this scenario, direct release of flashed primary coolant is not considered. Also, the passive residual heat removal system does not actuate, resulting in a longer period of steaming releases.

15.3.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.

- The reactor coolant activities are based on conservative assumptions (Refer to Table 15.3-3); whereas, the expected activities based on the fuel defect level are far less (see Section 11.1).

- The leakage of reactor coolant into the secondary system, at 300 gallons per day, is conservative. The leakage is normally a small fraction of this.

- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

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15.3.3.3.6 Doses

Using the assumptions from Table 15.3-3, the calculated total effective dose equivalent (TEDE) doses are determined to be less than 0.5 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and less than 0.2 rem at the low population zone outer boundary for the scenario in which there is no feedwater available to maintain water level in the steam generators. The doses for the scenario in which it is assumed that water level in the steam generators is maintained are 0.4 rem at the exclusion area boundary for the limiting 2-hour interval of 6 to 8 hours and 0.4 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A “small fraction” is identified as 10 percent or less consistent with the Standard Review Plan (Reference 4).

At the time the locked reactor coolant pump rotor event occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the locked rotor event, the resulting total dose remains less than the value reported above.

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip occurs on a Low-2 flow signal in the affected loop.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator in the faulted loop is reduced because: 1) the reduced flow results in a decreased tube-side film coefficient, and 2) the reactor coolant in the tubes cools down while the shell-side temperature increases. The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

This event is classified as a Condition IV incident (limiting fault), as defined in Subsection 15.0.1.

15.3.4.2 Conclusion

With a failed shaft, the impeller could be free to spin in a reverse direction as opposed to being fixed in position as is the case when a locked rotor occurs. This results in a decrease in the end point (steady-state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor analysis conservatively assumes that DNB occurs at the beginning of the transient. The calculated results presented for the locked rotor analysis bound the reactor coolant pump shaft break event.

15.3.5 Combined License Information

This section contained no requirement for additional information.

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15.3.6 References

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Table 15.3-1
Time Sequence of Events for Incidents
That Result in a Decrease In Reactor Coolant System Flow Rate

Accident	Event	Time (seconds)
Partial loss of forced reactor coolant flow		
– Loss of two pumps with four pumps running	Two pumps lose power and begin coasting down Low-2 flow reactor trip setpoint reached Rods begin to drop Minimum DNBR occurs	0.00 1.70 3.15 5.20
Complete loss of forced reactor coolant		
– Loss of four pumps with four pumps running	All pumps lose power and begin coasting down Low-2 Reactor coolant pump speed trip setpoint reached Rods begin to drop Minimum DNBR occurs	0.00 0.49 1.14 3.00
Reactor coolant pump shaft seizure (locked rotor)		
– One locked rotor with four pumps running without offsite power available	Rotor on one pump locks Low-2 flow trip point reached Rods begin to drop Maximum reactor coolant system pressure occurs Maximum cladding temperature occurs	0.00 0.10 1.55 3.40 4.10

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Table 15.3-2
Summary of Results for Locked Rotor Transients
(Four Reactor Coolant Pumps Operating Initially)

	Without Offsite Power Available
Maximum reactor coolant system pressure (psia)	2716.30
Maximum cladding average temperature, core hot spot (°F)	2013
Zr-H_2O reaction, core hot spot (percentage by weight)	0.57

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Table 15.3-3 (Sheet 1 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Locked Rotor Accident

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/gm of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/gm dose equivalent Xe-133
Reactor coolant alkali metal activity	Based on design basis activity (see Table 15A-9)
Secondary coolant initial iodine and alkali metal activity	1% of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.10
Core activity	See Table 15A-3
Radial peaking factor (for determination of activity in failed fuel rods)	1.75
Fission product gap fractions I-131 Kr-85 Other iodines and noble gases Alkali metals	 0.08 0.10 0.05 0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	6.04 E+05
Condenser	Not available
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	104.5[b]
Partition coefficient in steam generators iodine alkali metals	 0.01 0.0035
Accident scenario in which startup feedwater is not available Duration of accident (hr) Steam released (lb) 0-1.5 hours[c] Leak flashing fraction[d] 0-60 minutes > 60 minutes	 1.5 hr 6.48 E+05 0.04 0

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Table 15.3-3 (Sheet 2 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Locked Rotor Accident

Accident scenario in which startup feedwater is available	
Duration of accident (hr)	8.0 hr
Steam release rate (lb/sec)	60
Leak flashing fraction	Not applicable

Notes:

a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity released to the coolant from the assumed fuel failures, it is not significant.

b. Equivalent to 300 gpd cooled liquid at 62.4 lb/ft^3.

c. Heat removal is achieved by steaming and by passive core cooling system operation in the limiting case where the startup feedwater system is not available. When heat removal by the passive core cooling system exceeds the decay heat load, steam releases are terminated.

d. No credit for iodine partitioning is taken for flashed leakage. Credit is taken for a partition coefficient of 0.10 for alkali metals. Flashing is terminated by the passive core cooling system operation reducing the RCS below the saturation temperature of the secondary.

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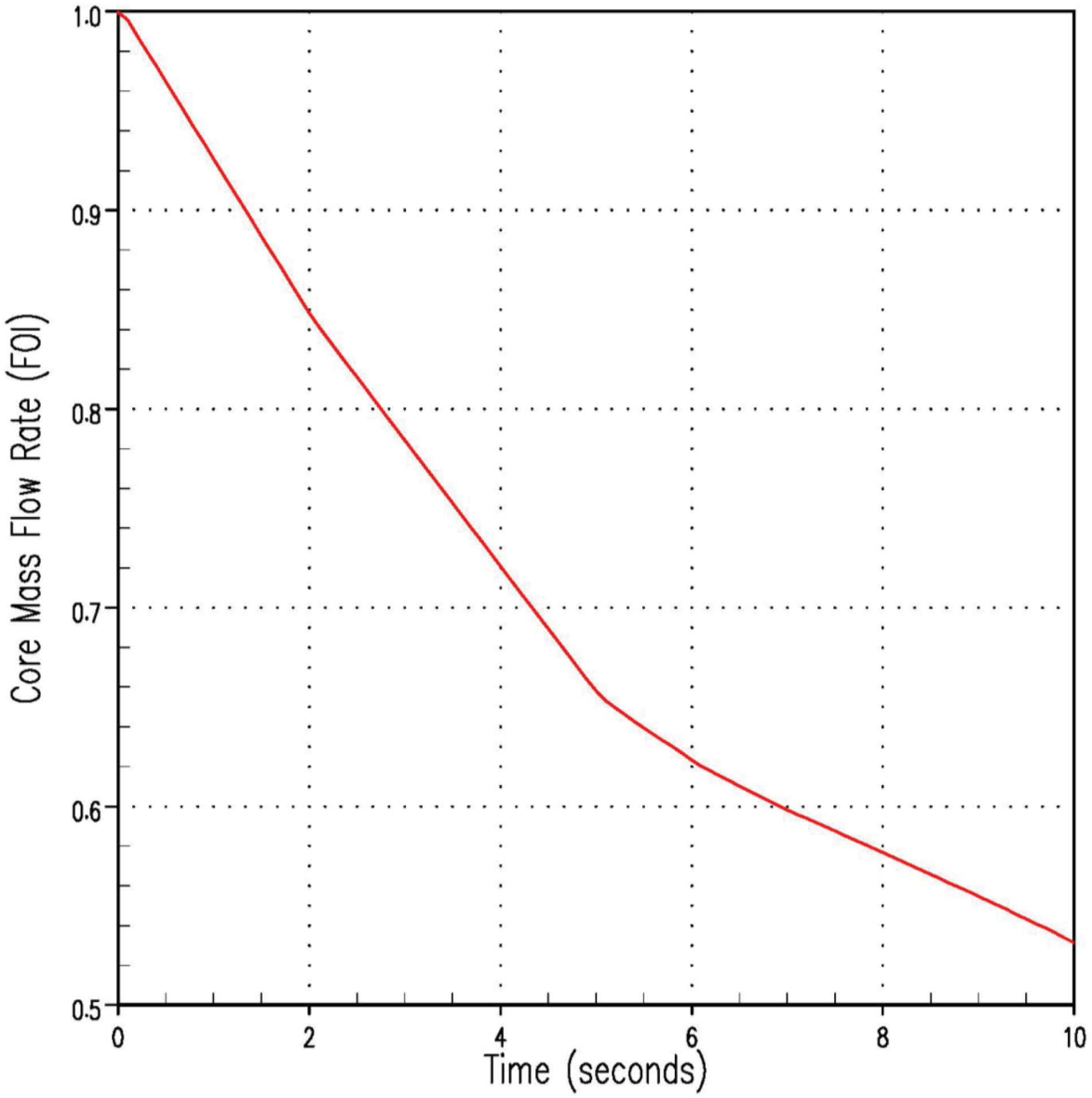

Figure 15.3.1-1
Core Mass Flow Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

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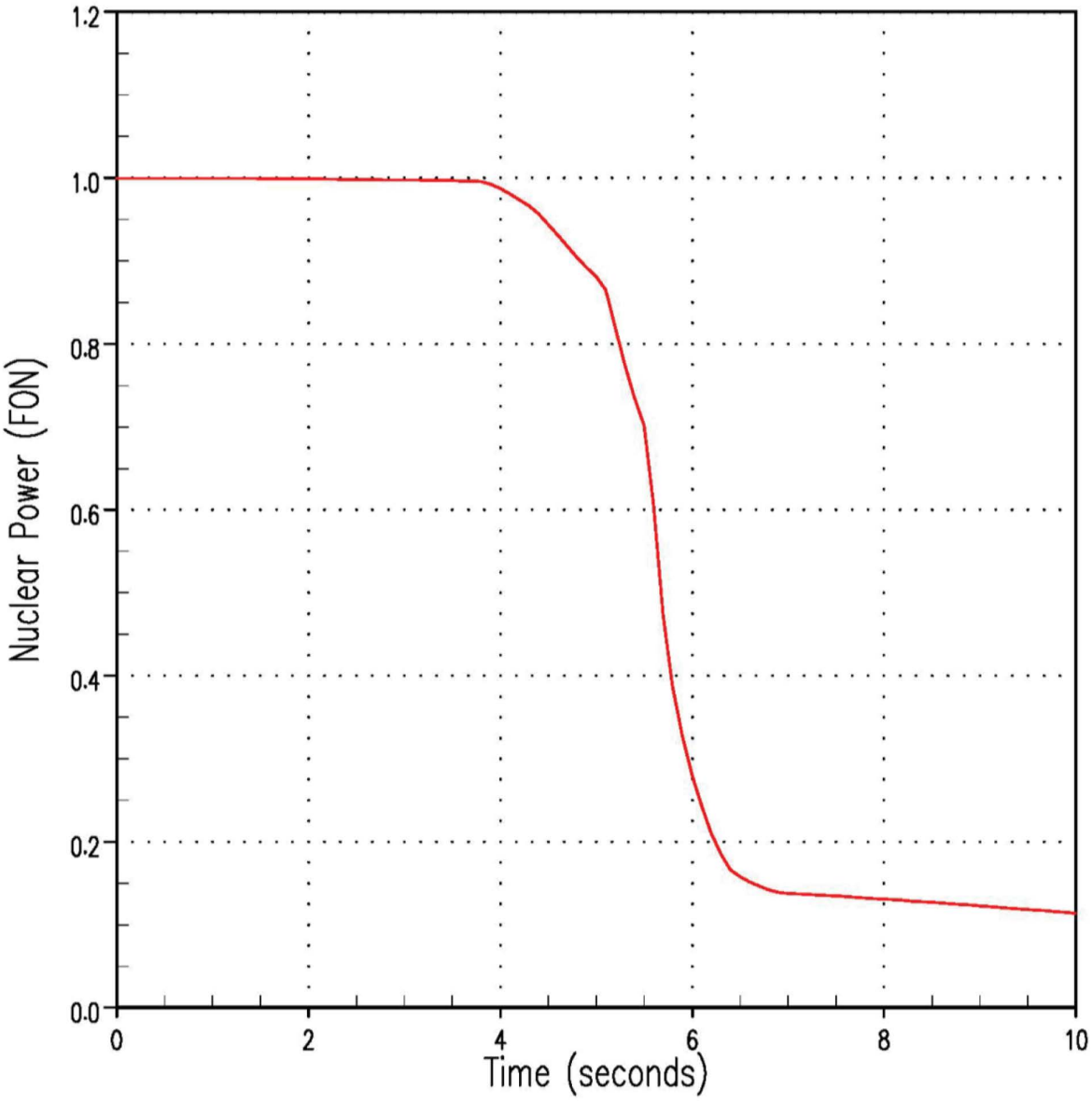

Figure 15.3.1-2
Nuclear Power Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

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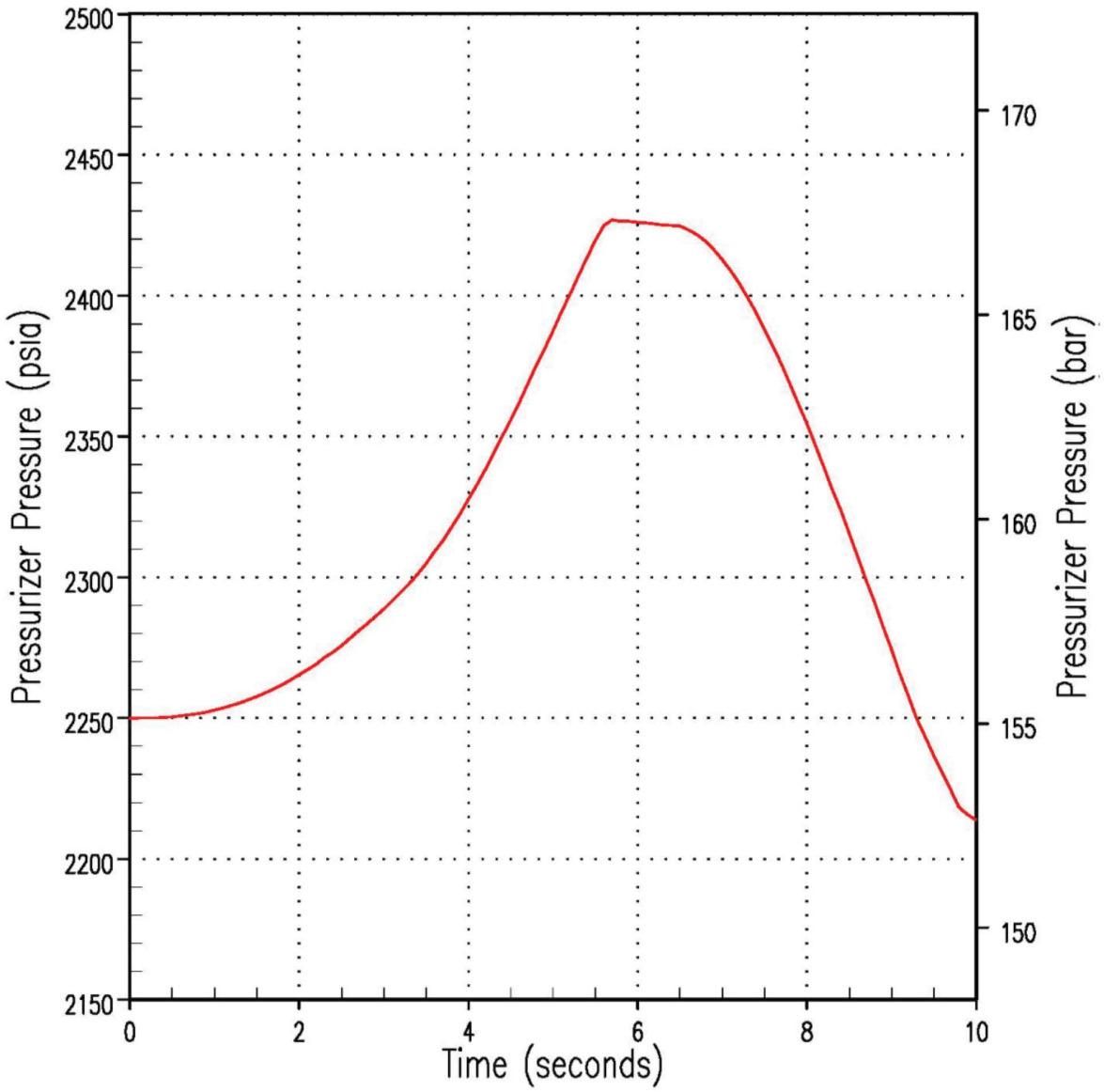

Figure 15.3.1-3
Pressurizer Pressure Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

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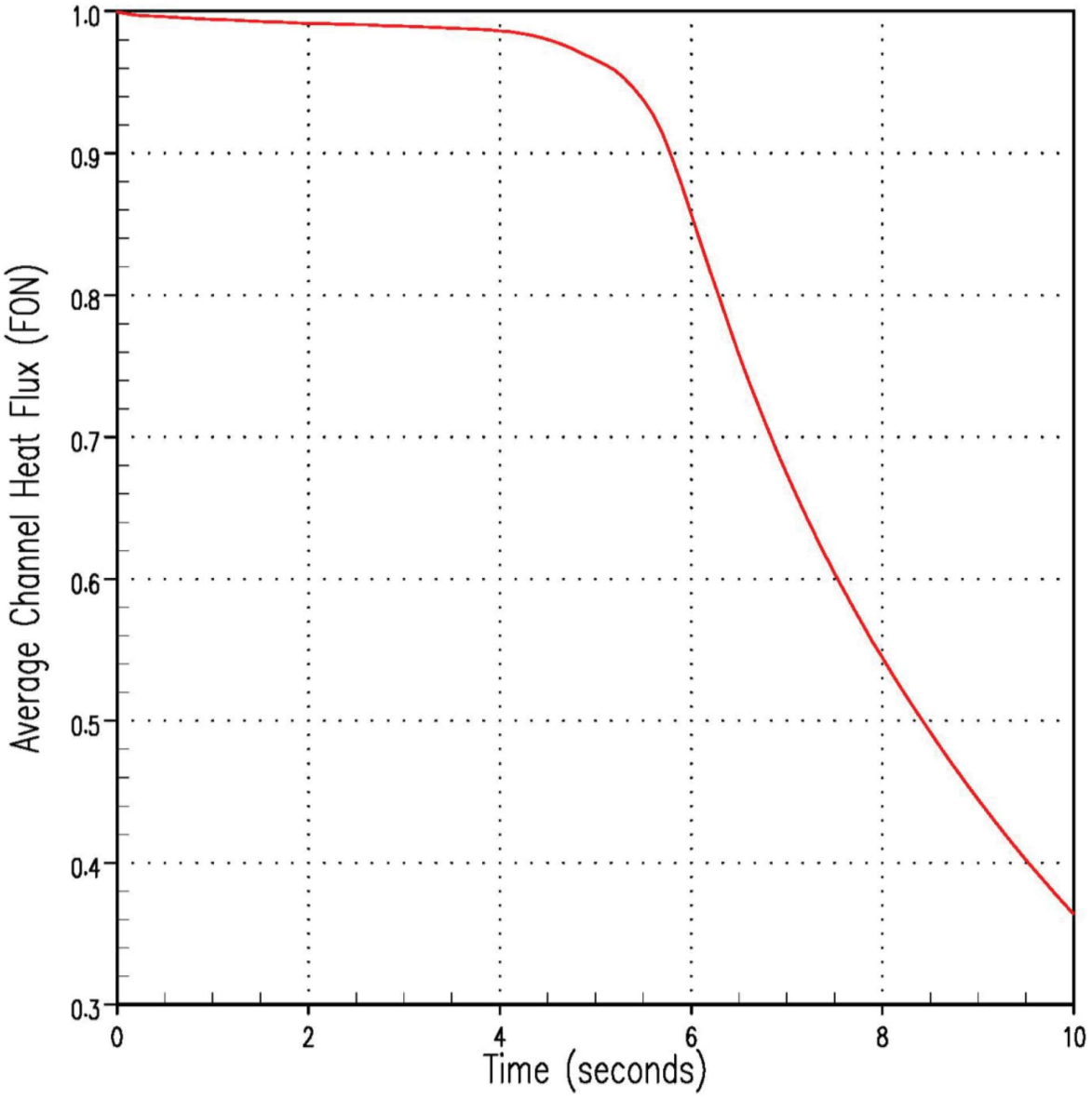

Figure 15.3.1-4
Average Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

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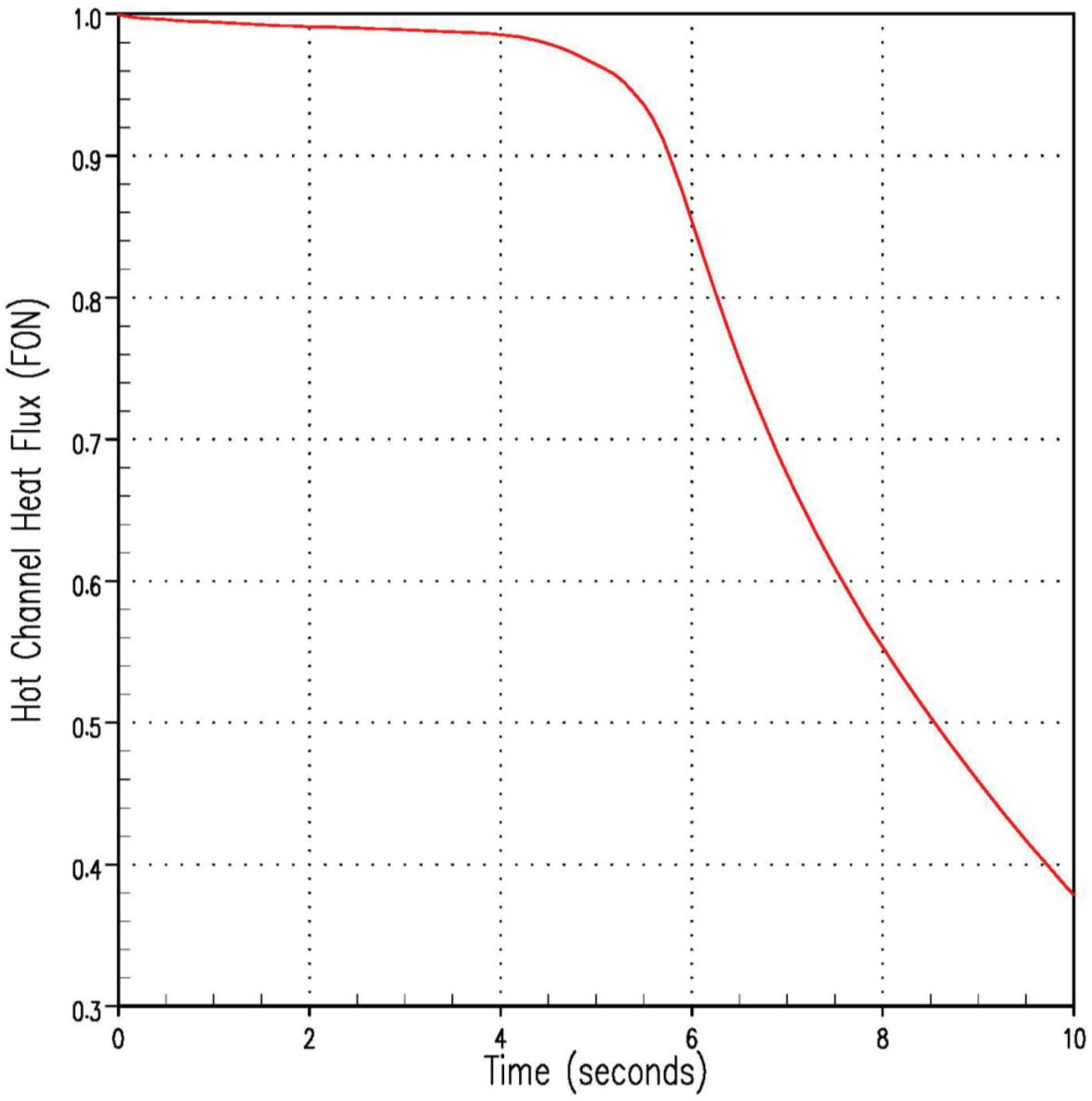

Figure 15.3.1-5
Hot Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

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Figure 15.3.1-6 Not Used

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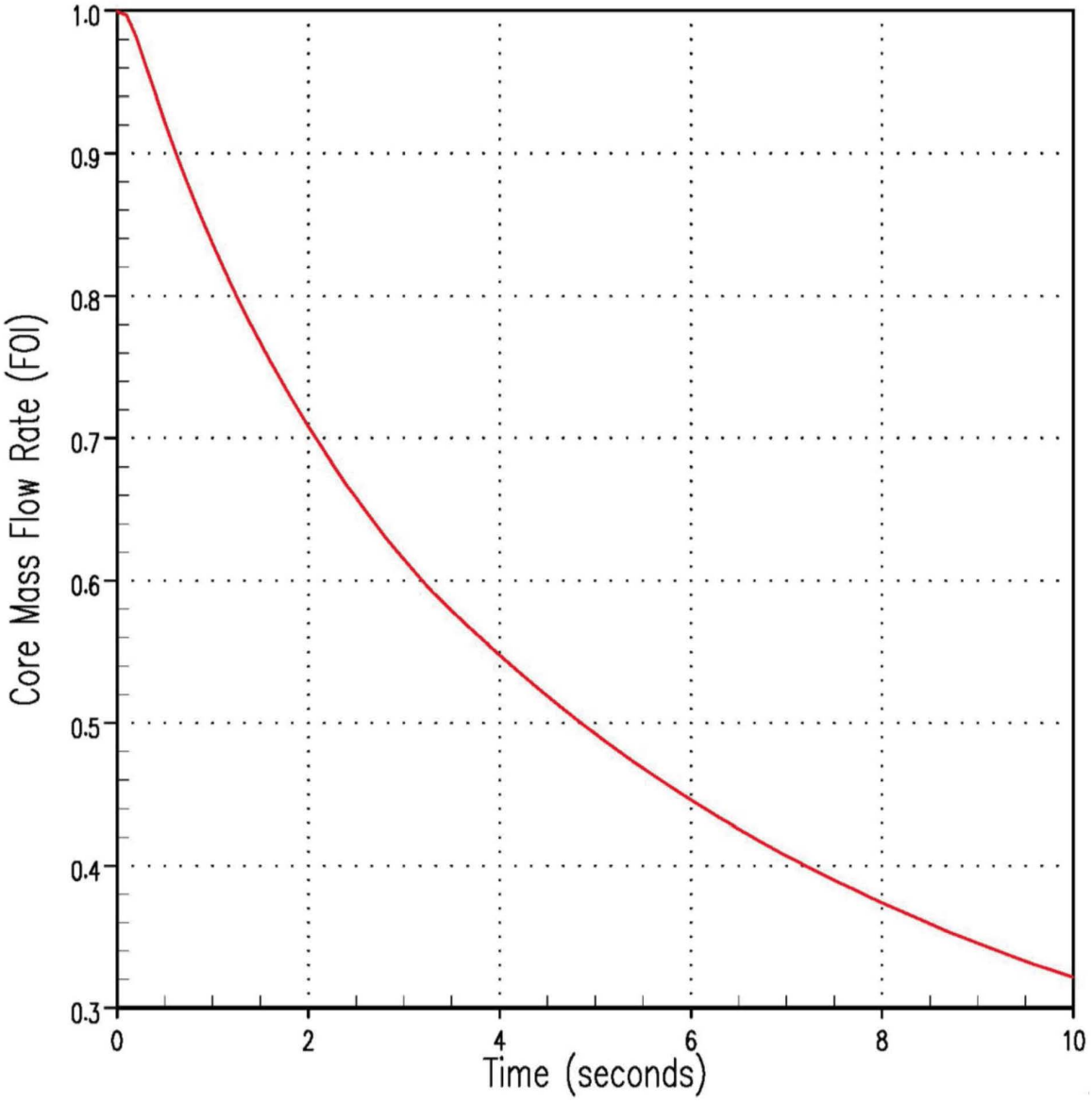

Figure 15.3.2-1
Core Mass Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

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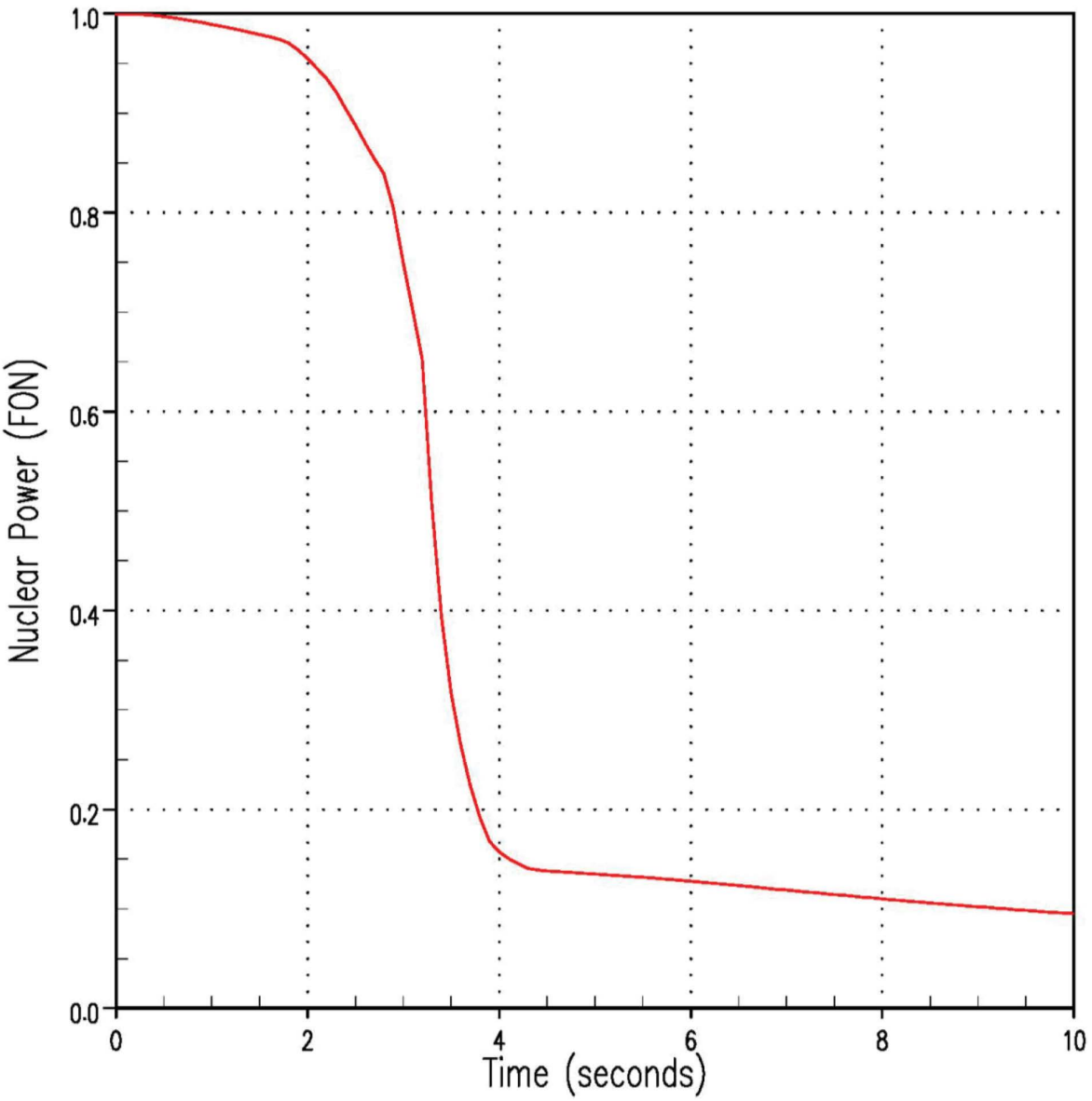

Figure 15.3.2-2
Nuclear Power Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

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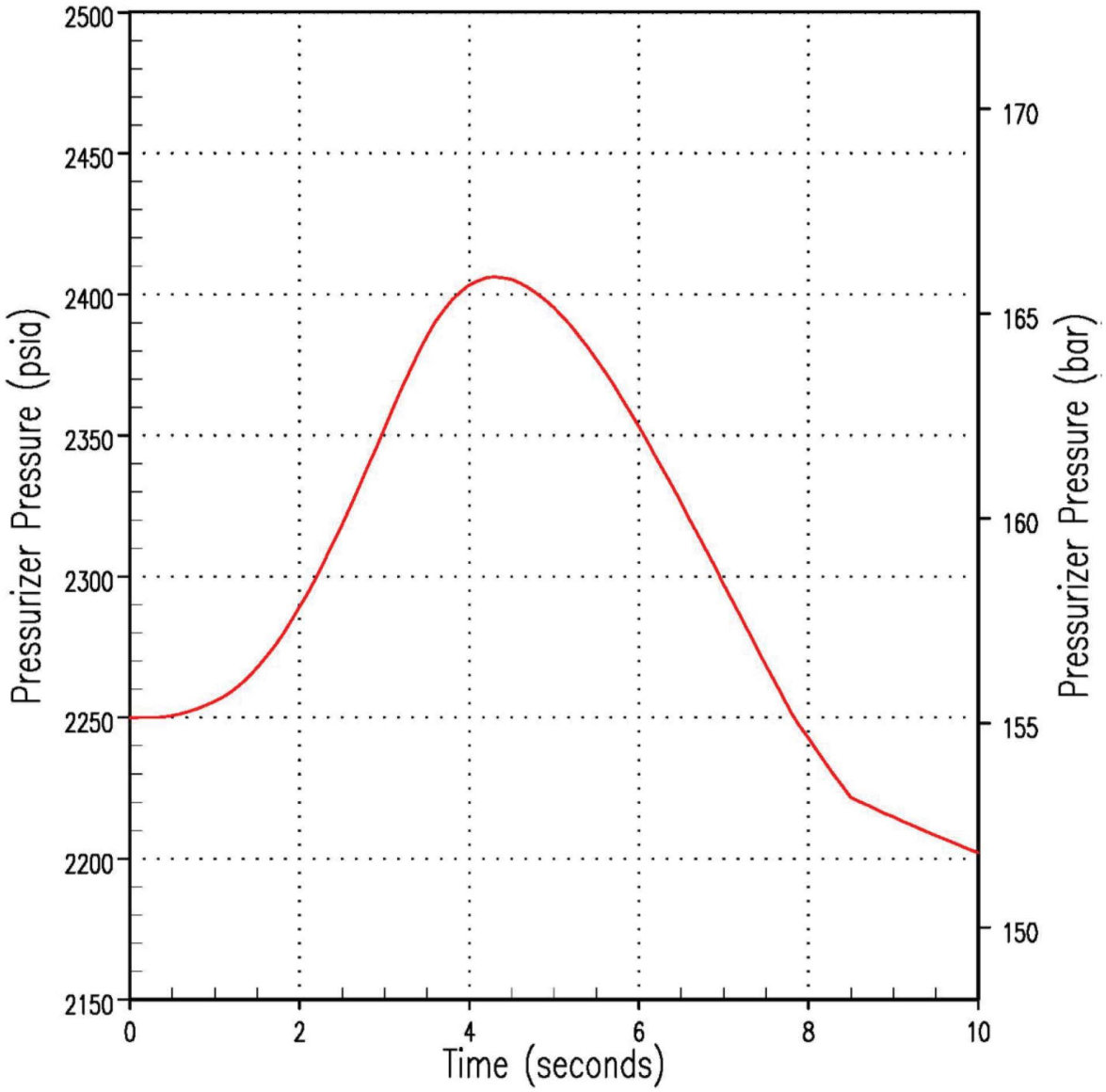

Figure 15.3.2-3
Pressurizer Pressure Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

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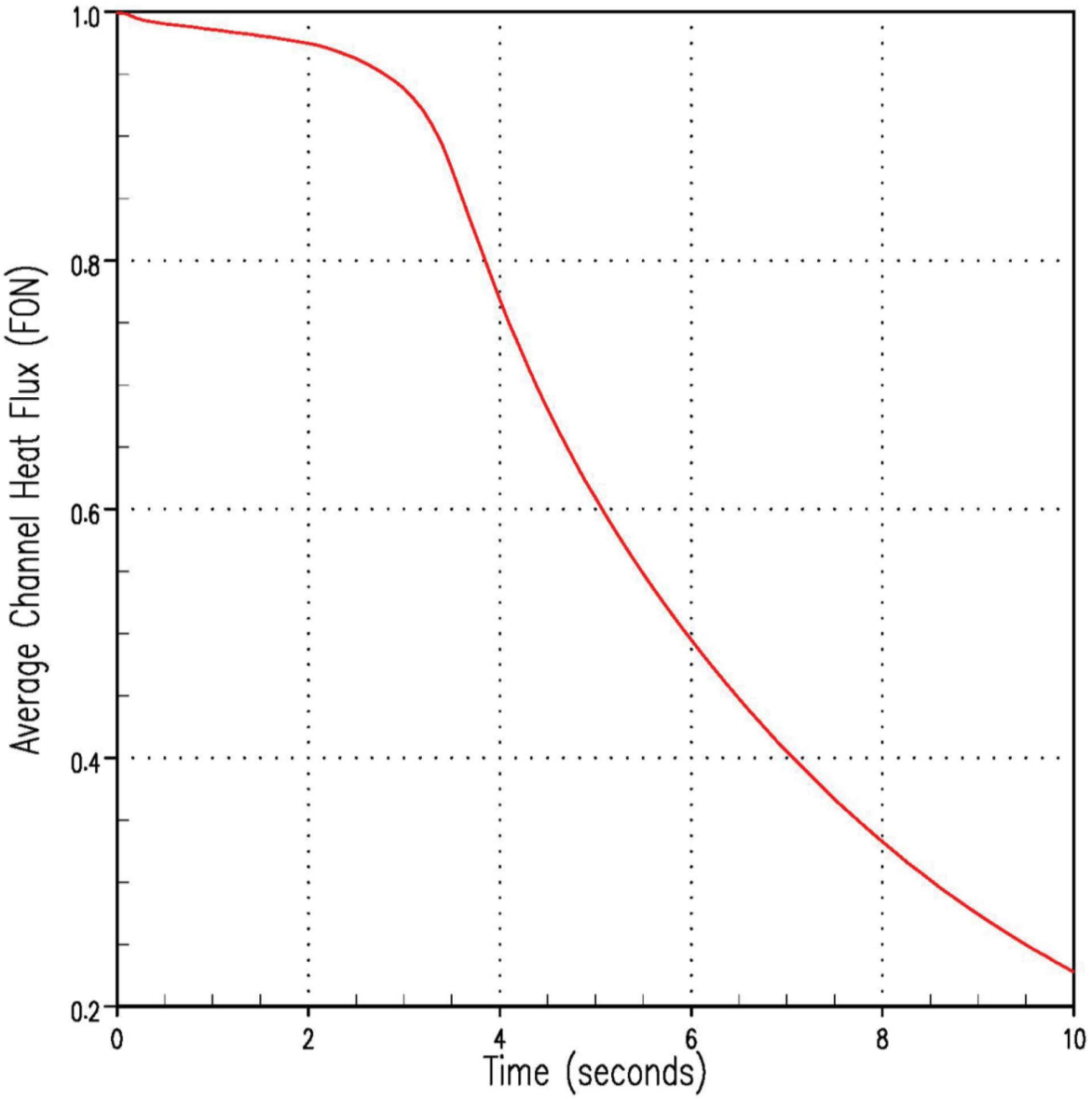

Figure 15.3.2-4
Average Channel Heat Flux Transient for
Four Cold Legs in Operation, Four Pumps Coasting Down

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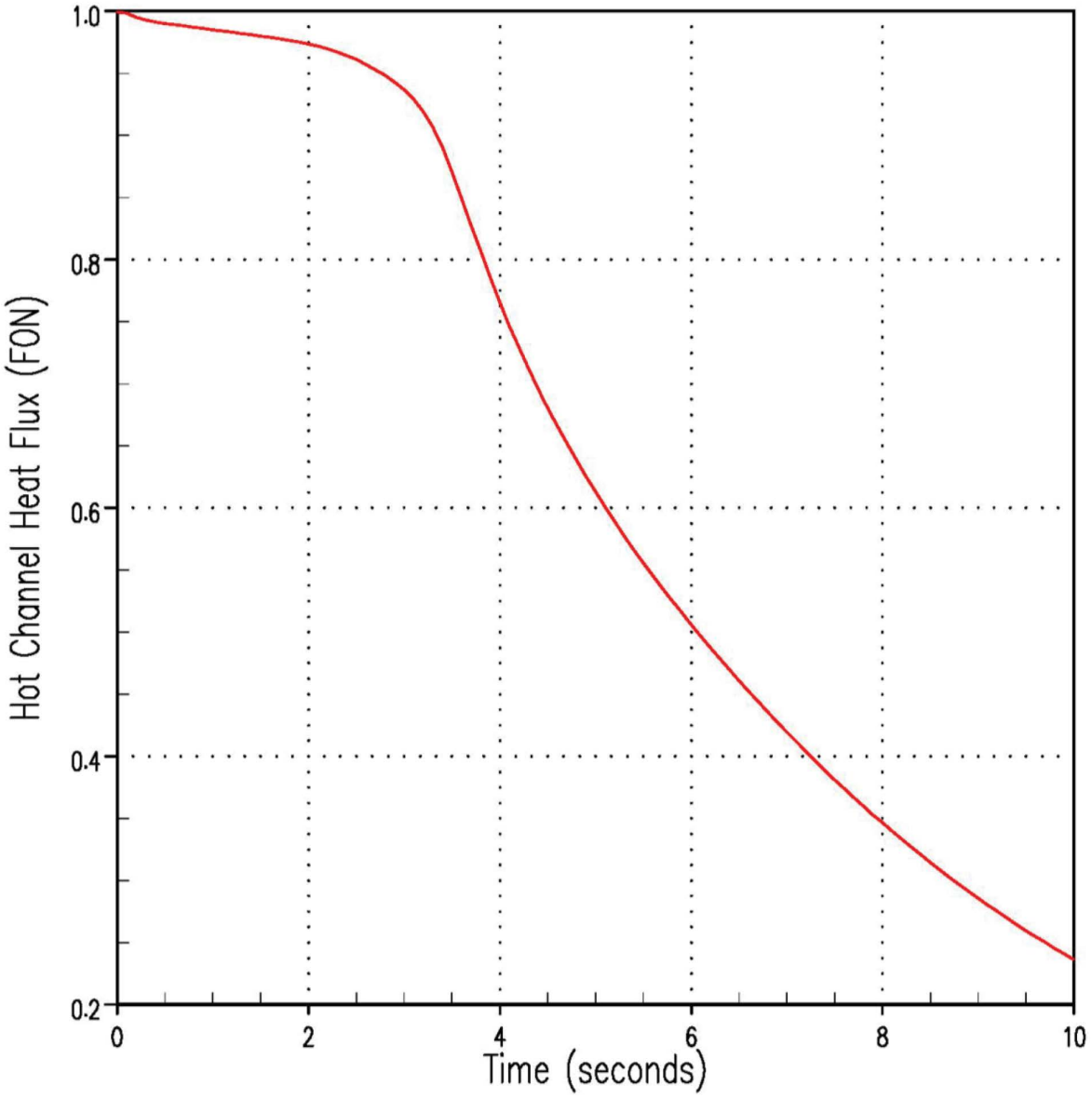

Figure 15.3.2-5
Hot Channel Heat Flux Transient for
Four Cold Legs in Operation, Four Pumps Coasting Down

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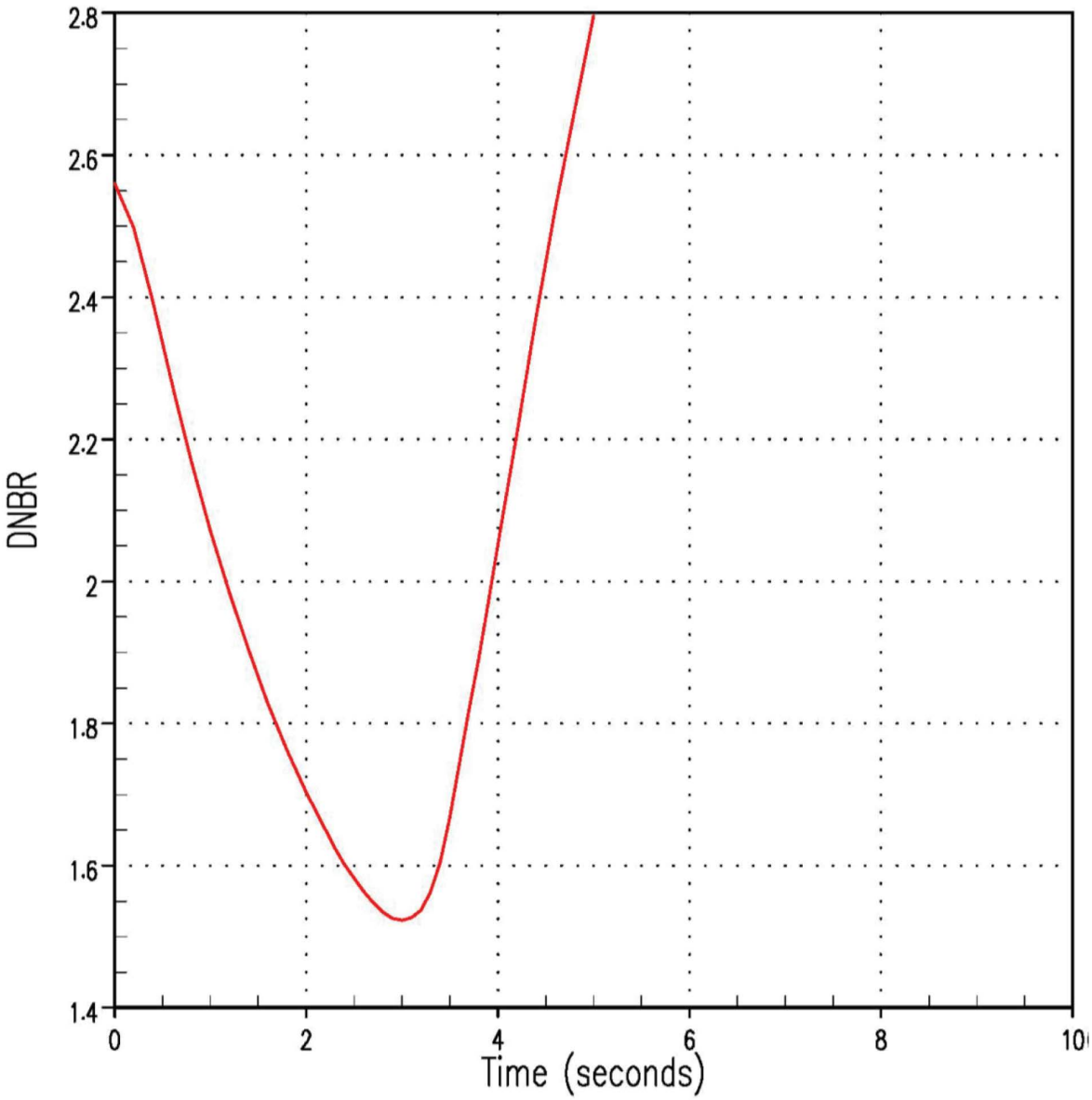

Figure 15.3.2-6
DNBR Transient for Four Cold Legs
in Operation, Four Pumps Coasting Down

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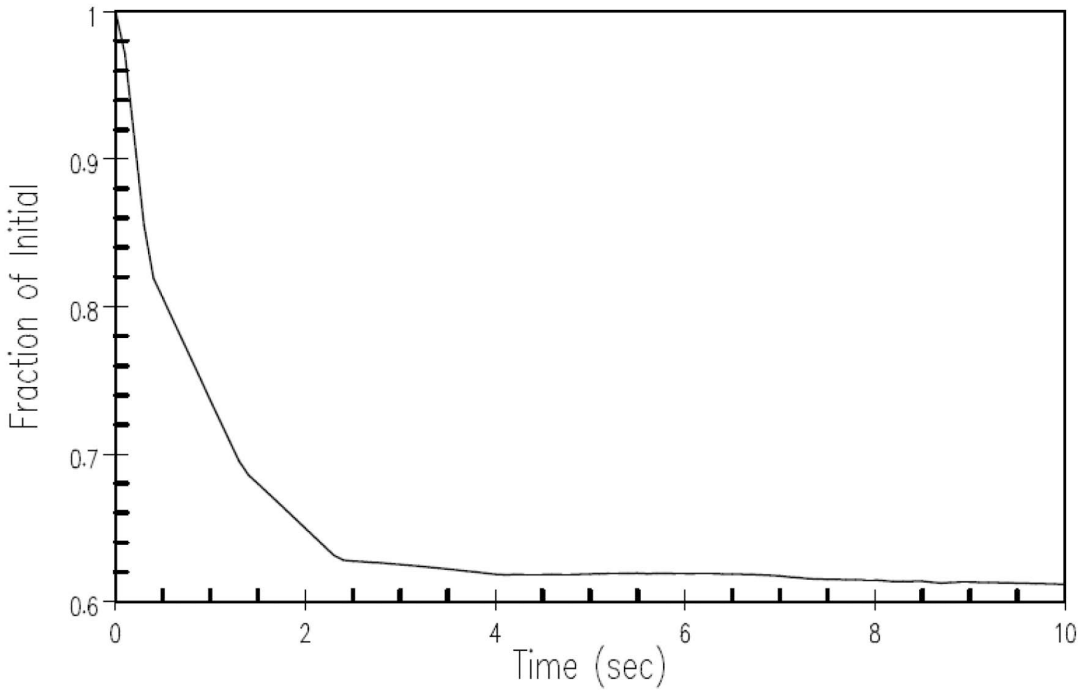

Figure 15.3.3-1
Core Mass Flow Transient for
Four Cold Legs in Operation, One Locked Rotor

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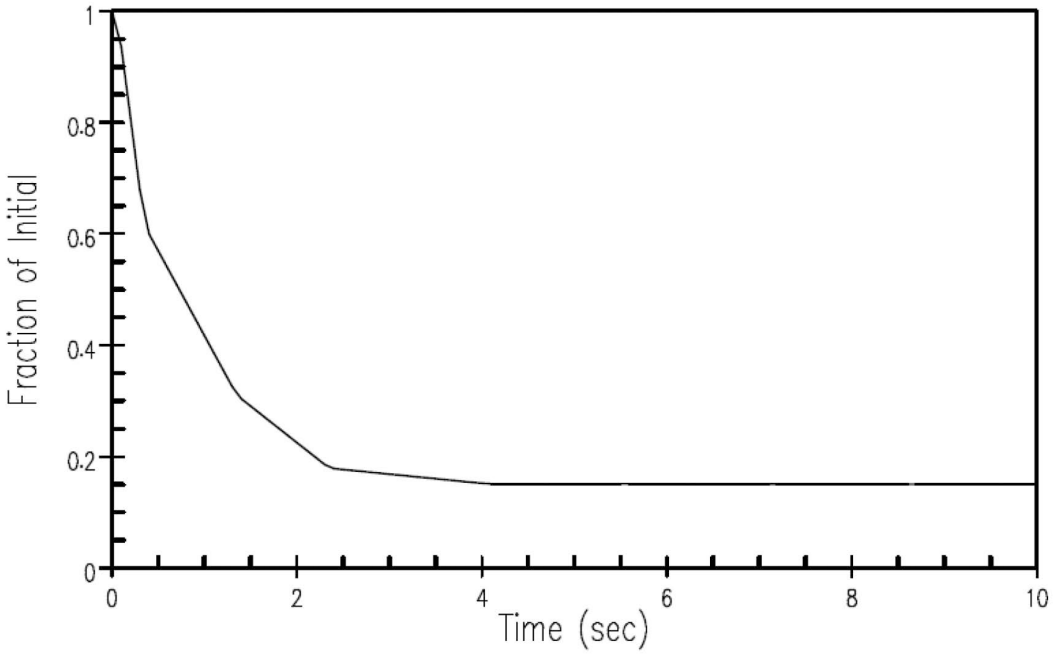

Figure 15.3.3-2
Faulted Loop Volumetric Flow Transient for
Four Cold Legs in Operation, One Locked Rotor

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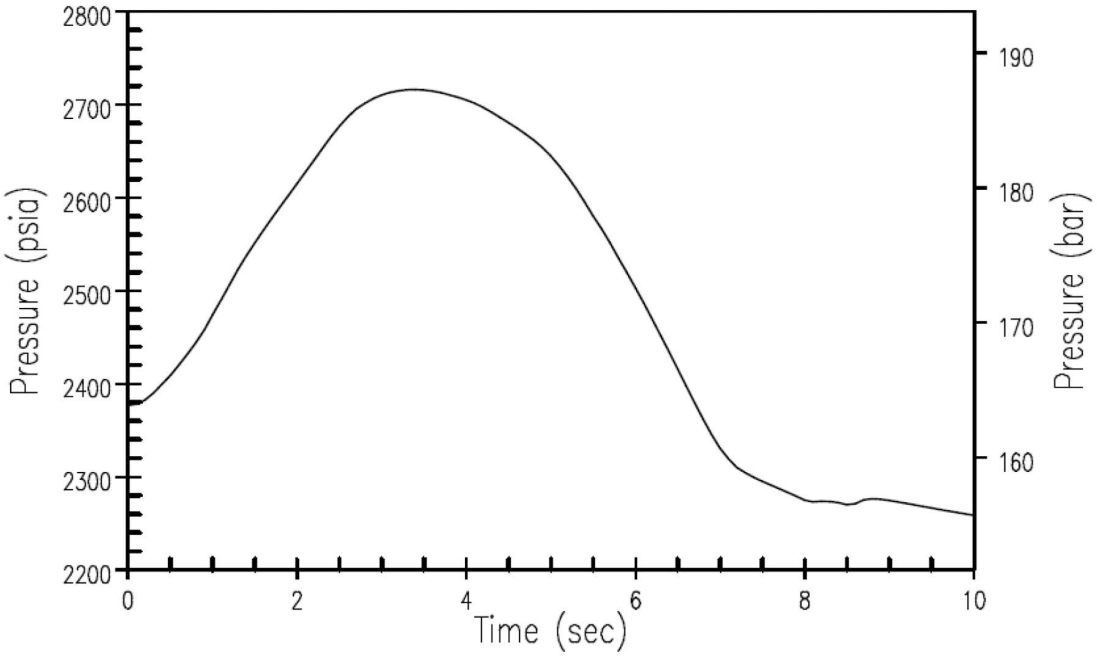

Figure 15.3.3-3
Peak Reactor Coolant Pressure for
Four Cold Legs in Operation, One Locked Rotor

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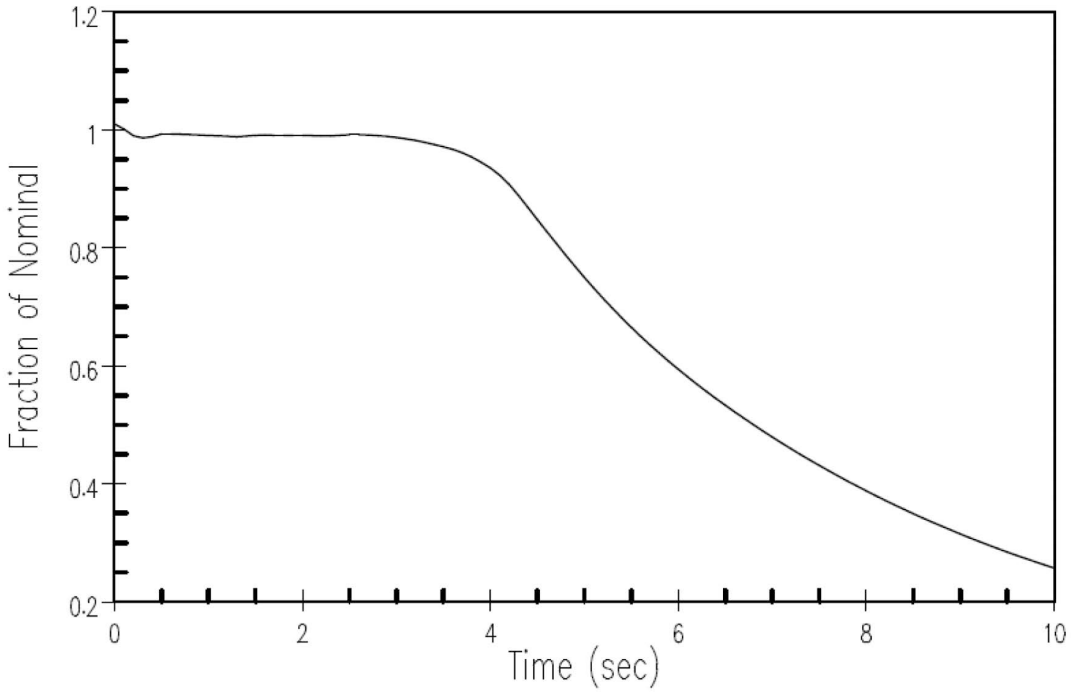

Figure 15.3.3-4
Average Channel Heat Flux Transient for
Four Cold Legs in Operation, One Locked Rotor

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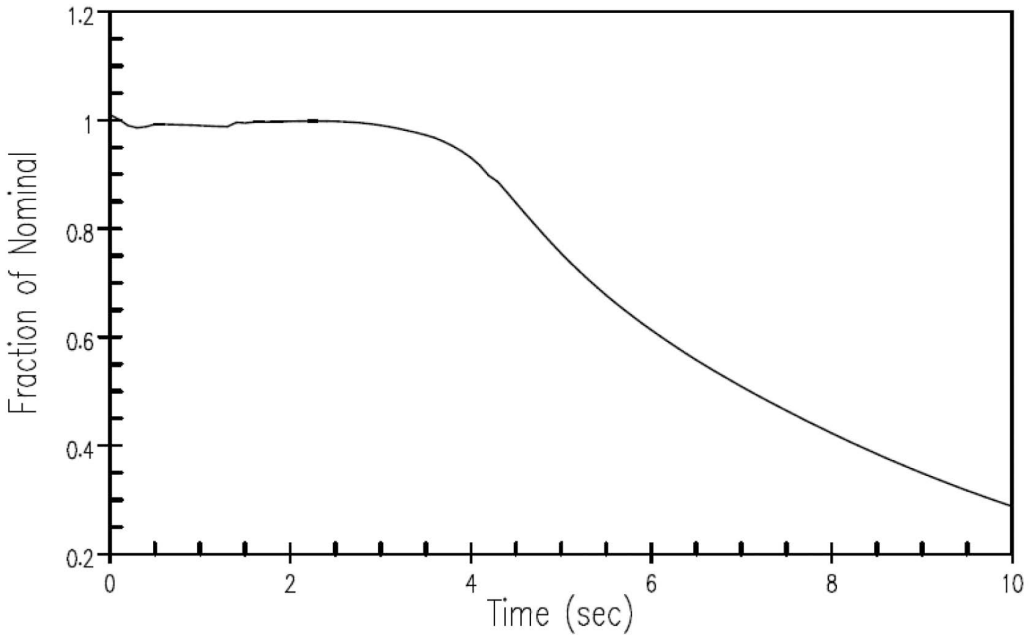

Figure 15.3.3-5
Hot Channel Heat Flux Transient for
Four Cold Legs in Operation, One Locked Rotor

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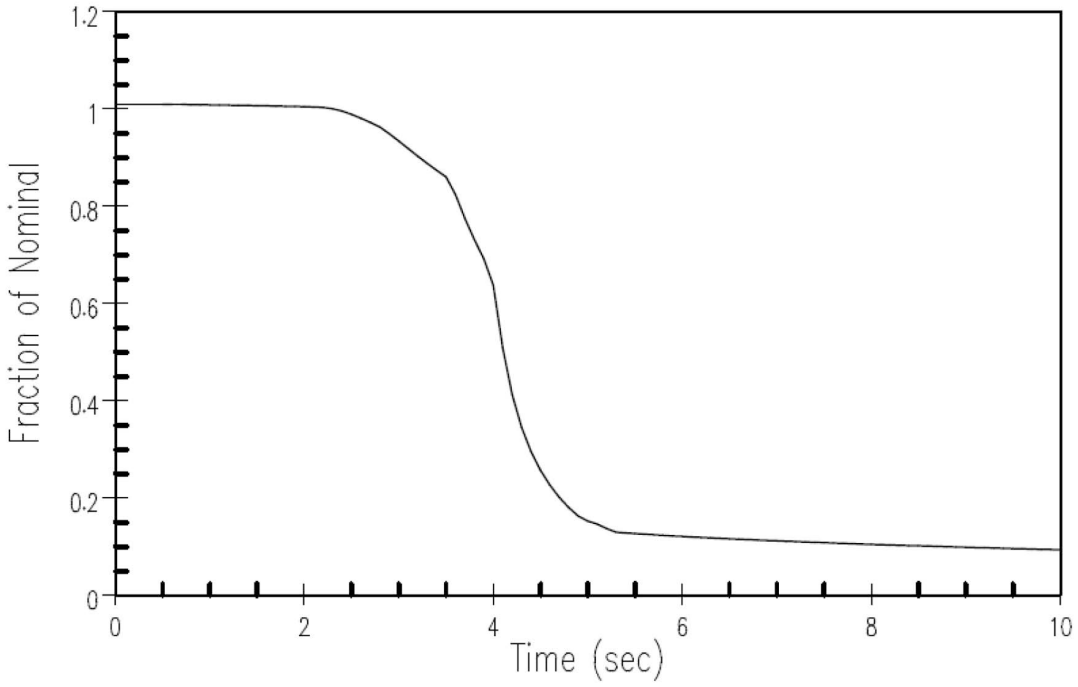

Figure 15.3.3-6
Nuclear Power Transient for
Four Cold Legs in Operation, One Locked Rotor

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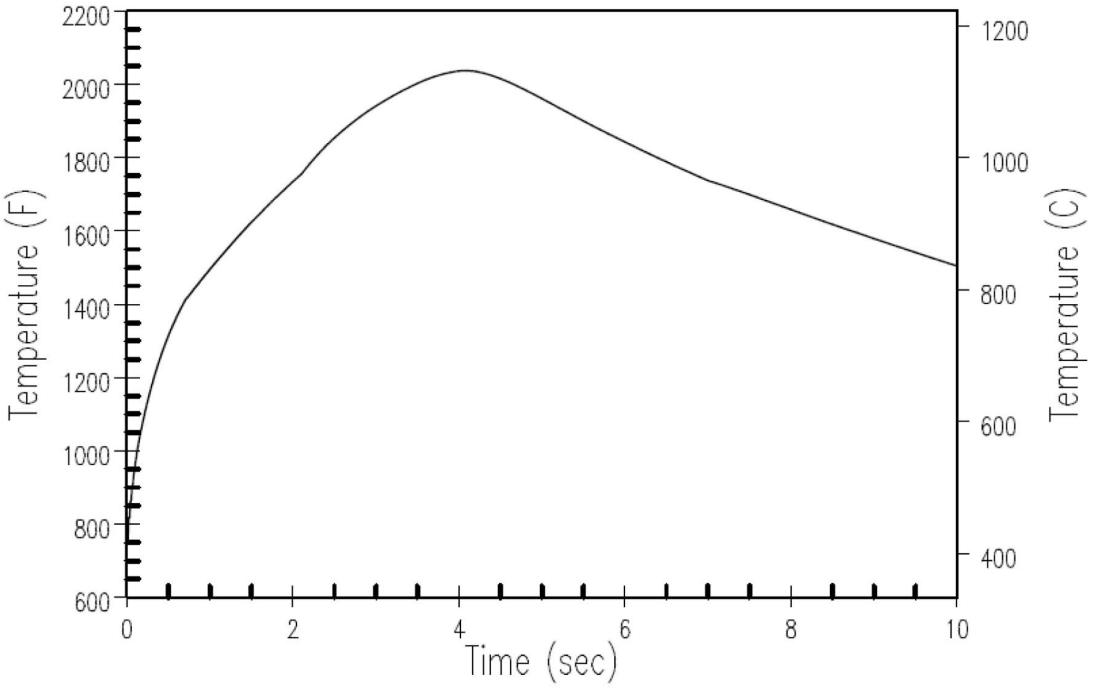

Figure 15.3.3-7
Cladding Inside Temperature Transient for
Four Cold Legs in Operation, One Locked Rotor

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15.4 Reactivity and Power Distribution Anomalies

A number of faults are postulated that result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system. Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Analyses are presented for the most limiting of these events.

The following incidents are discussed in this section:

A. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition

B. Uncontrolled RCCA bank withdrawal at power

C. RCCA misalignment

D. Startup of an inactive reactor coolant pump at an incorrect temperature

E. A malfunction or failure of the flow controller in a boiling water reactor recirculation loop that results in an increased reactor coolant flow rate (not applicable to AP1000)

F. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

G. Inadvertent loading and operation of a fuel assembly in an improper position

H. Spectrum of RCCA ejection accidents

Items A, B, D, and F above are Condition II events, item G is a Condition III event, and item H is a Condition IV event. Item C includes both Conditions II and III events.

The applicable transients in this section have been analyzed. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing as discussed in Subsection 15.4.8.

Radiological consequences are reported only for the limiting case.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition

15.4.1.1 Identification of Causes and Accident Description

An RCCA withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCAs which results in a power excursion. Such a transient can be caused by a malfunction of the reactor control or rod control systems. This can occur with the reactor subcritical, at hot zero power, or at power. The at-power case is discussed in Subsection 15.4.2.

The reactor may be brought to a critical condition by either RCCA withdrawal or boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Subsection 15.4.6).

The RCCA drive mechanisms are grouped into preselected bank configurations. These groups prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power

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supplied to the banks is controlled such that no more than two banks are withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed is that occurring with the simultaneous withdrawal of the combination of two sequential RCCA banks having the maximum combined worth at maximum speed.

This event is a Condition II event (a fault of moderate frequency) as defined in Subsection 15.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion limits the power during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient is terminated by the following automatic features of the protection and safety monitoring system:

- Source range high neutron flux reactor trip

  This trip function is actuated when two out of four independent source range channels indicate a neutron flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when the coincident two out of four intermediate range channels indicate a flux level below a specified level.

- Intermediate range high neutron flux reactor trip

  This trip function is actuated when two out of four independent, intermediate range channels indicate a flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after two out of four power range channels are reading above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (low setting)

  This trip function is actuated when two out of four power range channels indicate a power level above approximately 25 percent of full power. It may be manually bypassed when two out of four power range channels indicate a power level above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (high setting)

  This trip function is actuated when two out of four power range channels indicate a power level above a preset setpoint. It is always active.

- High nuclear flux rate reactor trip

  This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above a preset setpoint.

  In addition, control rod stops on high intermediate range flux level (one out of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

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15.4.1.2 Analysis of Effects and Consequences

15.4.1.2.1 Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation; then, an average core heat transfer calculation; and finally, the departure from nucleate boiling ratio (DNBR) calculation. In the first stage, the average core nuclear calculation is performed using spatial neutron kinetics methods, using the code TWINKLE (Reference 1), to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity).

In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). In the final stage, the average heat flux is used in VIPRE-01 (described in Section 4.4) for the transient DNBR calculation.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. The following assumptions are made to give conservative results for a startup accident:

- Because the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values, as a function of power, are used (see Table 15.0-2).

- Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. After the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux (see Table 15.0-2).

- The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect and thereby increase the neutron flux peak. The initial effective multiplication factor (k_{eff}) is assumed to be 1.0 because this results in the worst nuclear power transient.

- Reactor trip is assumed to be initiated by the power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10-percent uncertainty increase is assumed for the power range flux trip setpoint, raising it to 35 percent from the nominal value of 25 percent.

  Because the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Subsection 15.0.5 for RCCA insertion characteristics.

- The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential RCCA banks having the greatest combined worth at maximum speed (45 inches per minute). Control rod drive mechanism design is discussed in Section 4.6.

- The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high-worth position, are assumed in the departure from nucleate boiling (DNB) analysis.

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- The initial power level is assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.
- Four reactor coolant pumps are assumed to be in operation.
- Pressurizer pressure is assumed to be 50 psi below nominal for steady-state fluctuations and measurement uncertainties.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or components adversely affects the consequences of the accident. A loss of offsite power as a consequence of a turbine trip disrupting the grid is not considered because the accident is initiated from a subcritical condition where the plant is not providing power to the grid.

15.4.1.2.2 Results

Figures 15.4.1-1 through 15.4.1-4 show the transient behavior for the uncontrolled RCCA bank withdrawal from subcritical incident. The accident is terminated by reactor trip at 35 percent of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest-worth sequential rod cluster control banks, both assumed to be in their highest incremental worth region.

Figure 15.4.1-1 shows the average neutron flux transient. The energy release and the fuel temperature increases are relatively small. The heat flux response (of interest for DNB considerations) is also shown in Figure 15.4.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full-power nominal value. There is margin to DNB during the transient because the rod surface heat flux remains below the critical heat flux value, and there is a high degree of subcooling at all times in the core. Figures 15.4.1-3 and 15.4.1-4 shows the response of the average fuel temperature and the inner clad temperature, respectively. The minimum DNBR at all times remains above the design limit value (see Section 4.4).

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. Subsequently, the plant may be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected because the combination of thermal power and the coolant temperature results in a DNBR greater than the safety analysis limit value. Thus, no fuel or cladding damage is predicted as a result of DNB.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

An uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Because the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in a violatation of the DNB design basis or excessive linear power. Therefore, to avert damage to the fuel cladding, the protection and safety

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monitoring system (PMS) is designed to terminate any such transient before the DNBR falls below the design limit (see Section 4.4) or the overpower limit is exceeded.

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

The automatic features of the PMS that prevent core damage following the postulated accident include the following:

- Power range neutron flux instrumentation actuates a reactor trip if two out of four divisions exceed setpoint. In particular, the power range neutron flux instrumentation provides the following reactor trip functions:

  1. Reactor trip on power range high positive flux rate

  2. Reactor trip on power range high neutron flux (high setpoint)

  The first trip protects the core when a sudden abnormal increase in power is detected in the power range neutron flux channel in two out of four PMS divisions. It provides protection against rapid reactivity insertion accidents and is always active. The second trip serves as a backup to the overpower ΔT reactor trip and is always active.

- Reactor trip is actuated if any two out of four ΔT power divisions exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressurizer pressure to protect against violating the DNB design basis. The overtemperature ΔT reactor trip function initiates a reactor trip to prevent the plant from exceeding the core thermal limits. With the overtemperature ΔT reactor trip function, setpoints are selected to bound the non-linear characteristics of the core thermal limits. Dynamic compensation is included to account for transport times from the hot and cold legs to the core and to provide protection in a timely fashion such that the core thermal limits are not exceeded.

- Reactor trip is actuated if any two out of four ΔT power divisions exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to prevent the allowable linear heat generation rate (kW/ft) from being exceeded.

- A High-2 pressurizer pressure reactor trip is actuated from any two out of four pressure divisions when a set pressure is exceeded. This set pressure is less than the set pressure for the pressurizer safety valves.

- A High-3 pressurizer water level reactor trip is actuated from any two out of four level divisions that exceed the setpoint when the reactor power is above approximately 10 percent (permissive-P10).

In addition to the preceding reactor trips, there are the following RCCA withdrawal blocks:

- High neutron flux (two out of four power range)
- Overpower ΔT (two out of four)
- Overtemperature ΔT (two out of four)

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips:

- Power range high positive flux rate (fixed setpoint)
- High-2 pressurizer pressure (fixed setpoint)

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- Low-2 pressurizer pressure (fixed setpoint)
- Overpower and overtemperature ΔT (variable setpoints)

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of ac power during an uncontrolled RCCA bank withdrawal at power event have been evaluated; and did not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of ac power in comparison to the reactor shutdown time for an uncontrolled RCCA bank withdrawal at-power event. The primary effect of the loss of ac power is to cause the reactor coolant pumps (RCPs) to coast down. The PMS includes a five second minimum delay between the reactor trip and the turbine trip. In addition, a three second delay between the turbine trip and the loss of offsite ac power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coast down due to the loss of ac power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of ac power does not adversely impact this uncontrolled RCCA bank withdrawal at-power analysis because the plant will be shut down well before the RCPs begin to coast down.

15.4.2.2 Analysis of Effects and Consequences

15.4.2.2.1 Method of Analysis

This transient is analyzed using the LOFTRAN (References 3 and 11) code. This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.0.3-1 are used to define the inputs to LOFTRAN that determine the minimum DNBR during the transient.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. In performing a conservative analysis for an uncontrolled RCCA bank withdrawal at-power accident, the following assumptions are made:

- The nominal initial conditions are assumed in accordance with the revised thermal design procedure. Uncertainties in the initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 9).

- Two sets of reactivity coefficients are considered:

  Minimum reactivity feedback — A least-negative moderator temperature coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed (see Figure 15.0.4-1).

  Maximum reactivity feedback — A conservatively large positive moderator density coefficient corresponding to the end of core life and a large (in absolute magnitude) negative Doppler power coefficient are assumed (see Figure 15.0.4-1).

- The power range high positive flux rate trip is assumed to be actuated when the power range neutron flux changes at a rate higher than 15% per second with a 60 second rate-lag time constant. The overpower ΔT and overtemperature ΔT trips include adverse instrumentation and setpoint uncertainties. The delays for trip actuation assumed are given in Table 15.0-4a.

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- The RCCA trip insertion characteristic is based on the assumption that the highest-worth assembly is stuck in its fully withdrawn position.

- A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks, having the maximum combined worth at maximum speed.

If RCCA movement causes an adverse effect on the axial core power distribution, then the overtemperature ΔT trip setpoint is decreased as necessary to maintain margin to the DNBR design limit, and the overpower ΔT trip setpoint is decreased as necessary to maintain margin to the overpower limit.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in these systems or equipment adversely affects the consequences of the accident.

15.4.2.2.2 Results

Three reactor trip functions were credited in the analyses to provide protection over the entire range of reactivity insertion rates. These are the power range high positive flux rate, overtemperature ΔT, and overpower ΔT trips.

Figures 15.4.2-1 through 15.4.2-6 show the transient response for a representative rapid (80 pcm/s) RCCA withdrawal incident starting from full power. Reactor trip on power range high positive flux rate occurs shortly after the start of the transient. Because this is rapid with respect to the thermal time constants of the fuel, small changes in temperature and pressure result, and the DNB design basis described in Section 4.4 is met.

The transient response for a representative intermediate (34 pcm/s) RCCA withdrawal from full power is shown in Figures 15.4.2-7 through 15.4.2-12. Reactor trip on overpower ΔT occurs preventing the peak heat flux from exceeding 118%. The DNB design basis described in Section 4.4 is met.

The transient response for a representative slow (5 pcm/s) RCCA withdrawal from full power is shown in Figures 15.4.2-13 through 15.4.2-18. Reactor trip on overtemperature ΔT occurs after a longer period. The rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. The DNB design basis described in Section 4.4 is met.

Figure 15.4.2-19 shows the minimum DNBR as a function of reactivity insertion rate from initial full-power operation for minimum and maximum reactivity feedback. Minimum DNBR occurs immediately after rod motion. Three reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the power range high positive flux rate, overpower ΔT, and overtemperature ΔT trip functions. The minimum DNBR is greater than the design limit value described in Section 4.4.

Figures 15.4.2-20 and 15.4.2-21 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents for minimum and maximum reactivity feedback, starting at 60-percent and 10-percent power, respectively. Minimum DNBR, occurs immediately after rod motion. The results are similar to the 100-percent power case, except the transient is terminated by either the power range high positive flux rate or overtemperature ΔT reactor trip. The minimum DNBR is greater than the design limit value described in Section 4.4.

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The shape of the curves of minimum DNBR versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to PMS action in initiating a reactor trip.

Referring to Figure 15.4.2-19, it is noted that for transients initiated from full power, three reactor trip functions provide the DNB and overpower protection over the range of reactivity insertion rates analyzed. The overtemperature ΔT trip provides DNB protection except for rapid power excursions. The overpower ΔT trip provides protection for the slow to moderate power excursions. The power range high positive flux rate trip prevents both overpower and low DNBR for rapid power excursions.

Steam generator safety valves never open before the reactor trip for transients initiated at full power.

Because the RCCA bank withdrawal at-power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For rapid power excursions, the overpower transient is fast with respect to the fuel rod thermal time constant and the core heat flux lags behind the neutron flux response. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature still remains below the fuel melting temperature.

For slow to moderate power excursions, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by either the overpower ΔT or overtemperature ΔT reactor trip before the DNB design basis is violated. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak centerline temperature remains below the fuel melting temperature.

The reactor is tripped during the RCCA bank withdrawal at-power transient such that the ability of the primary coolant to remove heat from the fuel rods is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may be cooled down further by following normal plant shutdown procedures.

15.4.2.2.3 Overpressure Evaluation Results

In addition to the DNB cases discussed above, several cases are analyzed to ensure that the maximum reactor coolant system pressure does not exceed 110% of the design pressure. The cases cover a range of reactivity insertion rates from less than 1 pcm/s to 110 pcm/s and power levels from 10% to 100% power. Initial condition uncertainties on power, pressure and average temperature are conservatively included and the thermal design flow rate is assumed. The most limiting case was for a reactivity insertion rate of 10 pcm/s and an initial power level of 65% power. The peak pressure calculated is well below the limit.

15.4.2.3 Conclusions

The overpower ΔT, overtemperature ΔT, and power range high positive flux rate trip functions provide adequate protection over the entire range of possible reactivity insertion rates. The DNB design basis, as defined in Section 4.4, is met for all cases. The maximum reactor coolant system pressure remains below 110% of design.

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15.4.3 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

RCCA misoperation accidents include:

- One or more dropped RCCAs within the same group
- Statically misaligned RCCA
- Withdrawal of a single RCCA

Each RCCA has a position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod-at-bottom signal, which actuates a local alarm and a main control room annunciator. Group demand position is also indicated.

RCCAs are moved in preselected banks, and the banks are moved in a preselected sequence. Each bank of RCCAs is divided into one or two groups of four or five RCCAs each. The rods comprising a group operate in parallel. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Because the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which causes rod withdrawal affects the entire group. A single electrical or mechanical failure in the plant control system could, at most, result in dropping one or more RCCAs within the same group. Mechanical failures can cause either RCCA insertion or immobility, but not RCCA withdrawal.

The dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA events are Condition II incidents (incidents of moderate frequency) as defined in Subsection 15.0.1. The single RCCA withdrawal event is a Condition III incident, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full-power operation. The operator could withdraw a single RCCA in the control bank because this feature is necessary to retrieve an assembly should one be accidentally dropped. The event analyzed results from multiple wiring failures or multiple significant operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered low such that the limiting consequences may include slight fuel damage.

The event is classified as a Condition III incident consistent with the philosophy and format of American National Standards Institute, ANSI N18.2. By definition, "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged . . ." (Reference 10).

This selection of criterion is in accordance with General Design Criterion 25, which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods." (Emphases have been added.) It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that criterion established for the single rod withdrawal at power is appropriate and in accordance with General Design Criterion 25.

A dropped RCCA or RCCA bank may be detected by one or more of the following:

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- Sudden drop in the core power level as seen by the nuclear instrumentation system
- Asymmetric power distribution as seen by the incore or excore neutron detectors or core exit thermocouples, through online core monitoring
- Rod at bottom signal
- Rod deviation alarm
- Rod position indication

Misaligned RCCAs are detected by one or more of the following:

- Asymmetric power distribution as seen by the incore or excore neutron detectors or core exit thermocouples, through online core monitoring
- Rod deviation alarm
- Rod position indicators

The resolution of the rod position indicator channel is ±5 percent span (±7.5 inches). A deviation of any RCCA from its group by twice this distance (10 percent of span or 15 inches) does not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span.

If one or more of the rod position indicator channels is out of service, operating instructions are followed to verify the alignment of the nonindicated RCCAs. The operator also takes action as required by the Technical Specifications.

In the extremely unlikely event of multiple electrical failures that result in single RCCA withdrawal, rod deviation and rod control urgent failure are both displayed to the operator, and the rod position indicators indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, results in activation of the same alarm and the same visual indication. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip. The Condition III Standard Review Plan Section 15.4.3 evaluation criteria are met; however, due to the increase in local power density, the limits in Figure 15.0.3-1 may be exceeded.

Plant systems and equipment available to mitigate the effects of the various control rod misoperations are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

15.4.3.2.1.1 Method of Analysis

- One or more dropped RCCAs from the same group

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A drop of one or more RCCAs from the same group results in an initial reduction in the core power and a perturbation in the core radial power distribution. Depending on the worth and position of the dropped rods, this may cause the allowable design power peaking factors to be exceeded. Following the drop, the reduced core power and continued steam demand to the turbine causes the reactor coolant temperature to decrease. In the manual control mode, the plant will establish a new equilibrium condition. The new equilibrium condition is reached through reactivity feedback. In the presence of a negative moderator temperature coefficient, the reactor power rises monotonically back to the initial power level at a reduced inlet temperature with no power overshoot. The absence of any power overshoot establishes the automatic operating mode as a limiting case. If the reactor coolant system temperature reduction is very large, the turbine power may not be able to be maintained due to the reduction in the secondary-side steam pressure and the volumetric flow limit of the turbine system. In this case, the equilibrium power level is less than the initial power. In the automatic control mode, the plant control system detects the drop in core power and initiates withdrawal of a control bank. Power overshoot may occur, after which the control system will insert the control bank and return the plant to the initial power level. The magnitude of the power overshoot is a function of the plant control system characteristics, core reactivity coefficients, the dropped rod worth, and the available control bank worth.

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code (References 3 and 11). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures and power level.

Steady-state nuclear models using the computer codes described in Table 4.1-2 are used to obtain a hot channel factor consistent with the primary system transient conditions and reactor power. By combining the transient primary conditions with the hot channel factor from the nuclear analysis, the departure from nucleate boiling design basis is shown to be met using the VIPRE-01 code.

- Statically misaligned RCCA

  Steady-state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the VIPRE-01 code to calculate the DNBR.

15.4.3.2.1.2 Results

- One or more dropped RCCAs

  Figures 15.4.3-1 through 15.4.3-4 show the transient response of the reactor to a dropped rod (or rods) in automatic control. The nuclear power and heat flux drop to a minimum value and recover under the influence of both rod withdrawal and thermal feedback. The prompt decrease in power is governed by the dropped rod worth because the plant control system does not respond during the short rod drop time period. The plant control system detects the reduction in core power and initiates control bank withdrawal to restore the primary side power. Power overshoot occurs after which the core power is restored to the initial power level.

  The primary system conditions are combined with the hot channel factors from the nuclear analysis for the DNB evaluation. Uncertainties in the initial conditions are included in the DNB evaluation as discussed in Subsection 15.0.3.2. The calculated minimum DNBR for any single or multiple rod drop from the same group is greater than the design limit value described in Section 4.4. The sequence of events for a representative case is shown in Table 15.4-1.

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The analysis described previously includes consideration of drops of the RCCA groups which can be selected for insertion as part of the rapid power reduction system. This system is provided to allow the reactor to ride out a complete loss of load from full power without a reactor trip and is described in Subsection 7.7.1.10. If these RCCAs are inadvertently dropped (in the absence of a loss-of-load signal), the transient behavior is the same as for the RCCA drop described. The evaluation showed that the DNBR remains above the design limit value as a result of the inadvertent actuation of the rapid power reduction system.

The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the dropped RCCA event. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR occurs before the reactor coolant pumps begin to coast down.

- Statically misaligned RCCA

  The most severe misalignment situations with respect to DNBR arise from cases in which one RCCA is fully inserted, or where the mechanical shim or axial offset rod banks are inserted up to their insertion limit with one RCCA fully withdrawn while the reactor is at full power. Multiple independent alarms, including a bank insertion limit or rod deviation alarm, alert the operator well before the postulated conditions are approached.

  For RCCA misalignments in which the mechanical shim or axial offset banks are inserted to their respective insertion limits, with any one RCCA fully withdrawn, the DNBR remains above the safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and reactor coolant system temperature are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included in the DNB evaluation as described in Subsection 15.0.3.2.

  DNB does not occur for the RCCA misalignment incident, and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature is that corresponding to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which causes fuel melting.

  Following the identification of an RCCA group misalignment condition by the operator, the operator takes action as required by the plant Technical Specifications and operating instructions.

15.4.3.2.2 Single Rod Cluster Control Assembly Withdrawal

15.4.3.2.2.1 Method of Analysis

Power distributions within the core are calculated using the computer codes described in Table 4.1-2. The peaking factors are then used by VIPRE-01 to calculate the DNBR for the event. The case of the worst rod withdrawn from the mechanical shim or axial offset bank inserted at the insertion limit, with the reactor initially at full power, is analyzed. This incident is assumed to occur at beginning of life because this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

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15.4.3.2.2.2 Results

For the single rod withdrawal event, two cases are considered as follows:

A. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature and an increase in the local hot channel factor in the area of the withdrawing RCCA. In the overall system response, this case is similar to those presented in Subsection 15.4.2. The increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum DNBR from falling below the safety analysis limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip is expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the safety analysis limit value is 5 percent.

B. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA result in the immobility of the other RCCAs in the controlling bank. The transient then proceeds in the same manner as case A.

   For such cases, a reactor trip ultimately occurs although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the safety analysis limit value. Following reactor trip, normal shutdown procedures are followed.

   The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the single RCCA withdrawal event. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR, for rods where the DNBR did not fall below the design limit value (see Section 4.4) in the cases described, occurs before the reactor coolant pumps begin to coast down.

15.4.3.3 Conclusions

For cases of dropped RCCAs or dropped banks, including inadvertent drops of the RCCAs in those groups selected to be inserted as part of the rapid power reduction system, it is shown that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For cases of any one RCCA fully inserted, or the mechanical shim or axial offset banks inserted to their rod insertion limits with any single RCCA in one of those banks fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value (see Section 4.4).

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with the mechanical shim or axial offset banks at their insertion limits, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

The Technical Specifications (3.4.4) require all RCPs to be operating while in Modes 1 and 2. The maximum initial core power level for the startup of an inactive loop transient is approximately zero MWt. Furthermore, the reactor will initially be subcritical by the Technical Specification requirement. There will be no increase in core power, and no automatic or manual protective action is required.

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15.4.5 A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate

This subsection is not applicable to the AP1000.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

15.4.6.1 Identification of Causes and Accident Description

Other than control rod withdrawal, the principal means of positive reactivity insertion to the core is the addition of unborated, primary-grade water from the demineralized water transfer and storage system into the reactor coolant system through the reactor makeup portion of the chemical and volume control system. Normal boron dilution with these systems is manually initiated under strict administrative controls requiring close operator surveillance. Procedures limit the rate and duration of the dilution. A boric acid blend system is available to allow the operator to match the makeup water boron concentration to that of the reactor coolant system during normal charging.

An inadvertent boron dilution is caused by the failure of the demineralized water transfer and storage system or chemical and volume control system, either by controller, operator or mechanical failure. The chemical and volume control system and demineralized water transfer and storage system are designed to limit, even under various postulated failure modes, the potential rate of dilution to values that, with indication by alarms and instrumentation, allowing sufficient time for automatic or operator response to terminate the dilution.

An inadvertent dilution from the demineralized water transfer and storage system through the chemical and volume control system may be terminated by isolating the makeup flow to the reactor coolant system, by isolating the makeup pump suction line to the demineralized water transfer and storage system storage tank, or by tripping the makeup pumps. Lost shutdown margin may be regained by adding borated water to the reactor coolant system from the boric acid tank.

Generally, to dilute, the operator would need to perform two actions:

- Switch control of the makeup from the automatic makeup mode to the dilute mode.
- Start the chemical and volume control system makeup pumps.

Failure to carry out either of those actions prevents initiation of dilution. Because the AP1000 chemical and volume control system makeup pumps do not run continuously (they are expected to be operated once per day to make up for reactor coolant system leakage), a makeup pump is started when the volume control system is placed into dilute mode.

The status of the reactor coolant system makeup is available to the operator by the following:

- Indication of the boric acid and blended flow rates
- Chemical and volume control system makeup pumps status
- Deviation alarms, if the boric acid or blended flow rates deviate by more than the specified tolerance from the preset values
- When reactor is subcritical:
  - High flux at shutdown alarm
  - Indicated source range neutron flux count rate

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- Audible source range neutron flux count rate
- Source range neutron flux-multiplication alarm

- When the reactor is critical:

  - Axial flux difference alarm (reactor power ≥ 50 percent rated thermal power)
  - Control rod insertion limit low and low-low alarms
  - Overtemperature ΔT alarm (at power)
  - Overtemperature ΔT reactor trip
  - Overpower ΔT alarm
  - Overpower ΔT reactor trip
  - Power range neutron flux-high, both high and low setpoint reactor trips

This event is a Condition II incident (a fault of moderate frequency), as defined in Subsection 15.0.1.

The Technical Specifications contain exceptions which allow all RCPs to be removed from operation in Modes 3, 4, and 5 for a limited duration while performing required tests or while an unborated water source flow path through the chemical mixing tank to the pressurizer is unisolated under administrative controls. This is acceptable provided no operations are permitted that would add coolant to the RCS that has an insufficient boron concentration to meet the required shutdown margin, and the core outlet temperature remains sufficiently subcooled. With respect to chemical additions, the system design includes a flow orifice which limits flow from the chemical mixing tank to 2 gpm and allows combining the chemical addition solution with flow from the boric acid storage tank to the suction of the CVS makeup pumps. Administrative controls include dedicated control room and local operators to secure the unborated water source flow path after completion of the chemical addition or if flow rates indicate the potential for inadequately borated blended flow. Procedural controls also require the chemical addition flow path to be directed to the pressurizer for the short duration allowed for this exception. This limits the flow rate and duration of an unisolated unborated water flow path during this evolution.

15.4.6.2 Analysis of Effects and Consequences

Boron dilutions during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation are considered in this analysis. Conservative values for critical/key parameters are used (high reactor coolant system critical boron concentrations, high boron worths, minimum shutdown margins, and lower-than-actual reactor coolant system volumes). These assumptions (see Table 15.4-2) result in conservative determinations of the time available for operator or automatic system response after detection of a dilution transient in progress.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is considered for the boron dilution case initiated from the power mode of operation (Mode 1) with the reactor in manual control. This is the analyzed Mode 1 boron dilution case that produces a reactor and turbine trip (Subsection 15.4.6.2.6). The loss of offsite power is assumed to occur as a direct result of a turbine trip that would disrupt the grid and produce a consequential loss of offsite ac power. As discussed in Subsection 15.0.14, that scenario can occur only with the plant at power and connected to the grid. Therefore, only a boron dilution case initiated from full power will be addressed with respect to the consequential loss of offsite power.

15.4.6.2.1 Dilution During Refueling (Mode 6)

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls, which isolate the reactor coolant system from the potential source of unborated water by closing, and locking or otherwise securing specified valves in the chemical and volume control system, and Demineralized Water Transfer and Storage System,

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during refueling operations. These valves block the flow paths that allow unborated makeup water to reach the reactor coolant system. Makeup which is required during refueling uses water supplied from the boric acid tank (which contains borated water).

15.4.6.2.2 Dilution During Cold Shutdown (Mode 5)

The following conditions are assumed for inadvertent boron dilution while in this operating mode:

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 200°F and 14.7 psia.

- The reactor coolant system volume is 7605.9 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system with the reactor coolant system filled and vented and one reactor coolant pump running. The assumed active volume does not include the volume of the reactor vessel upper head region.

- Control rods are fully inserted, which is the normal condition in cold shutdown and a critical boron concentration is 1483 ppm. This is a conservative boron concentration with control rods inserted and accounts for the most reactive rod stuck in the fully withdrawn position.

- The shutdown margin is equal to 1.6-percent Δk/k, the minimum value identified by the core operating limits report (COLR) for the cold shutdown mode. Combined with the critical boron concentration identified above, this gives an initial boron concentration of 1675 ppm.

- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that, when in Mode 5, at least one RCP shall be operating with a flow of at least 3000 gpm. This provides sufficient flow through the system to maintain the system well-mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed (subject to the limited exception discussed in Subsection 15.4.6.1) and an uncontrolled boron dilution transient cannot occur, as discussed in Subsection 15.4.6.2.1.

- A Boron Dilution Protection System (BDPS) safety analysis limit (SAL) flux multiplier setpoint of 3.0 is assumed.

In the event of an inadvertent boron dilution transient during cold shutdown, the source range nuclear instrumentation detects an increase in the neutron flux by comparing the current source range flux to the data recorded over the previous 50 minutes. Upon detecting a sufficiently large flux increase, an alarm is sounded for the operator, and valves are actuated to terminate the dilution automatically.

Upon the actuation of a source range flux multiplier signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. In addition, the makeup pumps are tripped for equipment protection purposes.

No operator action is required to terminate this transient. The analysis demonstrates that the flux multiplier SAL will be reached 30.75 minutes after the dilution transient begins and that there is sufficient time at this point for the automatic protective features to terminate the dilution prior to losing all shutdown margin. After the automatic protection functions take place, the operator may take action to restore the Technical Specification shutdown margin.

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15.4.6.2.3 Dilution During Safe Shutdown (Mode 4)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 420°F and 401 psia.
- The reactor coolant system volume is 7605.9 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system with the reactor coolant system filled and vented and one reactor coolant pump running. The assumed active volume does not include the volume of the reactor vessel upper head region.
- All control rods are fully inserted, except the most reactive rod which is assumed stuck in the fully withdrawn position. The critical boron concentration is 1449 ppm.
- The shutdown margin is equal to 1.6-percent Δk/k, the minimum value required by the core operating limits report (COLR) for the hot shutdown mode. Combined with the critical boron concentration given above, this gives an initial boron concentration of 1649 ppm.
- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that at least one reactor coolant pump shall be operating with a flow of at least 3000 gpm when in Mode 4. This provides sufficient flow through the system to maintain the system well-mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed (subject to the limited exception discussed in Subsection 15.4.6.1) and an uncontrolled boron dilution transient cannot occur, as discussed in Subsection 15.4.6.2.1.
- A Boron Dilution Protection System (BDPS) Safety Analysis Limit (SAL) setpoint 3.0 is assumed.

In the event of an inadvertent boron dilution transient during safe shutdown, the source range nuclear instrumentation detects a sufficiently large increase in the neutron flux by comparing the current source range flux to that of about 50 minutes earlier, automatically initiates valve movement to terminate the dilution, and sounds an alarm.

Upon the actuation of a source range flux multiplier signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. Also, the makeup pumps are tripped for equipment protection purposes.

No operator action is required to terminate this transient. The analysis demonstrates that the flux multiplier SAL will be reached 28.83 minutes after the dilution transient begins and that there is sufficient time at this point for the automatic protective features to terminate the dilution prior to losing all shutdown margin. After the automatic protection functions take place, the operator may take action to restore the Technical Specification shutdown margin.

15.4.6.2.4 Dilution During Hot Standby (Mode 3)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 557°F and 2250 psia.

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- The reactor coolant system volume is 7605.9 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system with the reactor coolant system filled and vented and one reactor coolant pump running. The assumed active volume does not include the volume of the reactor vessel upper head region.

- Critical boron concentration is 1281 ppm. This is a conservative boron concentration assuming control rods are fully inserted minus the most reactive rod, which is assumed stuck in the fully withdrawn position.

- The shutdown margin is equal to 1.6-percent Δk/k, the minimum value required by the core operating limits report (COLR) for the hot standby mode. Combined with the critical boron concentration given above, this gives an initial boron concentration of 1509 ppm.

- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that at least one reactor coolant pump shall be operating with a flow of at least 3000 gpm when in Mode 3. This provides sufficient flow through the system to maintain the system well mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed (subject to the limited exception discussed in Subsection 15.4.6.1) and an uncontrolled boron dilution transient cannot occur, as discussed in Subsection 15.4.6.2.1.

In the event of an inadvertent boron dilution transient in hot standby, the source range nuclear instrumentation detects a sufficiently large increase in the neutron flux by comparing the current source range flux to that of about 50 minutes earlier, automatically initiates valve movement to terminate the dilution, and sounds an alarm. Upon the actuation of a source range flux multiplier signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. Also, the makeup pumps are tripped for equipment protection purposes.

The source range flux doubling function can be manually blocked during shutdown conditions when below the P-8 setpoint. Prior to manually blocking the source range flux doubling function, the operator isolates unborated water source flow paths. When blocked below the P-8 setpoint, an automatic close signal provides backup to the operator manual isolation of the CVS demineralized water system isolation valves to prevent boron dilution.

No operator action is required to terminate this transient. The analysis demonstrates that the flux multiplier SAL will be reached 32.07 minutes after the dilution transient begins and that there is sufficient time at this point for the automatic protective features to terminate the dilution prior to losing all shutdown margin. After the automatic protection functions take place, the operator may take action to restore the Technical Specification shutdown margin.

15.4.6.2.5 Dilution During Startup (Mode 2)

The plant is in the startup mode at the beginning of each cycle. During this mode of operation, rod control is in manual. Normal actions taken to change power level, either up or down, require operator actuation. The Technical Specifications require an available shutdown margin of 1.6-percent Δk/k and four reactor coolant pumps operating. Other conditions assumed are the following:

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 565.83°F (5% power) and 2250 psia.

- Minimum reactor coolant system water volume is 8425.5 ft^3. This is a very conservative estimate of the active reactor coolant system volume, minus the pressurizer volume.

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- The initial maximum boron concentration, corresponding to the rods inserted to the insertion limits, is 2031 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 1097 ppm, which gives a critical boron concentration of 934 ppm.

This mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control. For a normal approach to criticality, the operator manually withdraws control rods and dilutes the reactor coolant with unborated water at controlled rates until criticality is achieved. Once critical, the power escalation is slow enough to allow the operator to manually block the source range reactor trip after receiving the P-6 permissive signal from the intermediate range detectors (nominally at 10^5 cps). Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

Upon any reactor trip signal, or low input voltage to the Class 1E dc and uninterruptible power supply system battery chargers, a safety-related function automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Additionally, the suction lines for the chemical and volume control system pumps are automatically realigned to draw borated water from the chemical and volume control system boric acid tank.

If, however, P-6 is reached in an orderly fashion and the source range trip is manually blocked and an uncontrolled dilution then occurs, the likely result is a reactor trip on the power range high neutron flux - low function, as reflected in Table 15.4-1. Regardless of the trip signal, the reactor trip results in termination of the dilution and also indicates to the operators that a dilution was underway. The Table 15.4-1 time to loss of shutdown margin (SDM) ignores that the dilution would be terminated by the trip, giving an indication of the additional safety margin for operator action to terminate the dilution manually. As indicated in Table 15.4-1, after reactor trip, the dilution would have to continue for approximately 205 minutes to overcome the available shutdown margin.

15.4.6.2.6 Dilution During Full Power Operation (Mode 1)

The plant may be operated at power two ways: automatic T_{avg}/rod control and under operator control. The COLR and Technical Specifications require an available shutdown margin of 1.6-percent Δk/k and four reactor coolant pumps operating. With the plant at power and the reactor coolant system at pressure, the dilution rate is limited by the capacity of the chemical and volume control system makeup pumps. The analysis is performed assuming two chemical and volume control system pumps are in operation, even though normal operation is with one pump. Conditions assumed for a dilution in this mode are the following:

- A dilution flow of 175 gpm of unborated water exists.The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 581.6°F (full power) and 2250 psia.

- Minimum reactor coolant system water volume is 8425.5 ft^3. This is a very conservative estimate of the active reactor coolant system volume, minus the pressurizer volume.

- An initial maximum boron concentration, corresponding to the rods inserted to the insertion limits, is 1811 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 877 ppm, which gives a critical boron concentration of 934 ppm. Full rod insertion, minus the most reactive stuck rod, occurs due to reactor trip.

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With the reactor in automatic rod control, the pressurizer level controller limits the dilution flow rate to the maximum letdown rate. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller throttles charging flow down to match the letdown rate. For the safety analysis, a conservative dilution flow rate of 175 gpm is assumed. With the reactor in automatic rod control, a boron dilution results in a power and temperature increase in such a way that the rod controller attempts to compensate by slow insertion of the control rods. This action by the controller results in at least three alarms to the operator:

A. Rod insertion limit- low level alarm

B. Rod insertion limit- low-low level alarm if insertion continues

C. Axial flux difference alarm (ΔI outside of the target band)

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. The operator has at least 170.6 minutes from the rod insertion limit low-low alarm until shutdown margin is lost at the beginning of the cycle, which is more than 15 minutes and is not required before 30 minutes. The time is significantly longer at the end of the cycle because of the lower initial and critical boron concentrations.

Because the analysis for the boron dilution event with the reactor in automatic rod control does not predict a reactor and turbine trip, considering the consequential loss of offsite power for this case is not needed.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature would rise and cause the reactor to reach the overtemperature or overpower ΔT trip setpoint resulting in a reactor trip. Upon any reactor trip signal, a safety-related function automatically isolates the unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Additionally, the suction lines for the chemical and volume control system pumps are automatically realigned to draw borated water from the chemical and volume control system boric acid tank.

The boron dilution transient in this case is essentially equivalent to an uncontrolled rod withdrawal at power (see Subsection 15.4.2). The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be approximately 0.5 pcm/s and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. Before reaching the overtemperature or overpower ΔT reactor trip, the operator receives an alarm overtemperature or overpower ΔT and an overtemperature or overpower ΔT turbine runback.

Should a consequential loss of offsite power occur after reactor and turbine trip, it does not alter the fact that the dilution event has been terminated by automatic protection features. As indicated previously, the reactor trip signal that occurs in parallel with the turbine trip will actuate a safety-related function that automatically isolates the unborated water from the demineralized water system and thereby terminates the dilution. A subsequent loss of offsite power will cause the chemical and volume control system pumps to shut down.

After reactor trip, the automatic termination of the dilution flow from the demineralized water transfer and storage system precludes a post-trip return to criticality.

15.4.6.3 Conclusions

Inadvertent boron dilution events are administratively prevented by the Technical Specifications during refueling (Mode 6) and automatically terminated during cold shutdown (Mode 5), safe shutdown (Mode 4), and hot standby (Mode 3) modes. Inadvertent boron dilution events during

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startup (Mode 2) or power operation (Mode 1), if not detected and terminated by the operators, result in an automatic reactor trip. Following reactor trip, automatic termination of the dilution occurs and post-trip return to criticality is prevented.

The preceding results demonstrate that in all modes of operation, an inadvertent boron dilution is prevented or responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration, the reactor is in a stable condition.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors can inadvertently occur, such as those arising from the inadvertent loading of one or more fuel assemblies into improper positions, having a fuel rod with one or more pellets of the wrong enrichment, or having a full fuel assembly with pellets of the wrong enrichment. This leads to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core-loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

An error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes more peaked than those calculated with the correct enrichments. A 5-percent uncertainty margin is included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The online core monitoring system is used to verify power shapes at the start of life and is capable of revealing fuel assembly enrichment errors or loading errors that cause power shapes to be peaked in excess of the design value. Power-distribution-related measurements are incorporated into the evaluation of calculated power distribution information using the incore instrumentation processing algorithms contained within the online monitoring system. The processing algorithms contained within the online monitoring system are functionally identical to those historically used for the evaluation of power distributions measurements in Westinghouse pressurized water reactors.

Each fuel assembly is marked with an identification number and loaded in accordance with a core-loading diagram to reduce the probability of core loading errors. During core loading, the identification number is checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed.

The power distortion due to a combination of misplaced fuel assemblies could significantly increase peaking factors and is readily observable with the online core monitoring system. The fixed incore instrumentation within the instrumented fuel assembly locations is augmented with core exit thermocouples. There is a high probability that these thermocouples would also indicate any abnormally high coolant temperature rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

This event is a Condition III incident (an infrequent fault) as defined in Subsection 15.0.1.

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15.4.7.2 Analysis of Effects and Consequences

15.4.7.2.1 Method of Analysis

Steady-state power distributions in the x-y plane of the core are calculated at 30-percent rated thermal power using the three-dimensional nodal code ANC (Reference 7). Representative power distributions in the x-y plane for a correctly loaded core are described in Chapter 4.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown in the incore detector locations. (See Figures 15.4.7-1 through 15.4.7-4.)

15.4.7.2.2 Results

The following core loading error cases are analyzed:

Case A:

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered is the interchange of two assemblies near the periphery of the core (see Figure 15.4.7-1).

Case B:

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. For the particular case considered, the interchange is assumed to take place close to the core center and with burnable poison rods located in the correct Region 2 position, but in a Region 1 assembly mistakenly loaded in the Region 2 position (see Figure 15.4.7-2).

Case C:

Enrichment error – Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.4.7-3).

Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.4.7-4).

15.4.7.3 Conclusions

Fuel assembly enrichment errors are prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and cladding temperatures are limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects are either readily detected by the online core monitoring system or cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

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15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.8.1.1 Design Precautions and Protection

15.4.8.1.1.1 Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to prevent the possibility of an RCCA drive mechanism housing failure are listed below:

- Each control rod drive mechanism housing is completely assembled and shop tested at a minimum of 125% of system design pressure.
- The mechanism housings are hydrotested after they are attached to the reactor vessel head. The housings are included as part of the hydrotest of the completed reactor coolant system.
- The allowable stress levels in the mechanism are not exceeded due to system thermal transients at power or by the thermal movement of the coolant loops. Moments induced by the safe shutdown earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.
- The latch mechanism housing and rod travel housing are each a single length of forged stainless steel. This material exhibits excellent notch toughness at temperatures that are encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional confidence that gross failure of the housing does not occur. The joint between the latch housing and latch housing nozzle is a full penetration bi-metallic weld, and the joint between the latch housing and rod travel housing is a threaded joint with a canopy seal weld.

15.4.8.1.1.2 Nuclear Design

If a rupture of an RCCA drive mechanism housing is postulated, the operation using chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the power control (or mechanical shim) RCCAs inserted only far enough to permit load follow. The axial offset RCCAs are positioned so that the targeted axial offset can be met throughout core life. Reactivity changes caused by core depletion and xenon transients are normally compensated for by boron changes and the mechanical shim banks, respectively. Further, the location and grouping of the power control and axial offset RCCAs are selected with consideration for an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full-power operation, a less severe reactivity excursion than analyzed is expected.

It may occasionally be desirable to operate with larger than normal insertions. For this reason, a power control and axial offset rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit provides adequate shutdown capability and an acceptable power distribution. The position of the RCCAs is continuously indicated in the main control room. An alarm

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occurs if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require rapid boration at the low-low level alarm. If the CVS is unavailable or insufficient then the CMTs can be used for boration.

15.4.8.1.1.3 Reactor Protection

The reactor protection in the event of a rod ejection accident is described in WCAP-15806-P-A, (Reference 4). The protection for this accident is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are described in Section 7.2.

15.4.8.1.1.4 Effects on Adjacent Housings

Failures of an RCCA mechanism housing, due to either longitudinal or circumferential cracking, does not cause damage to adjacent housings. The control rod drive mechanism is described in Subsection 3.9.4.1.1.

15.4.8.1.1.5 Not Used

15.4.8.1.1.6 Not Used

15.4.8.1.1.7 Consequences

The probability of damage to an adjacent housing is considered remote. If damage is postulated, it is not expected to lead to a more severe transient because RCCAs are inserted in the core in symmetric patterns and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal. This is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

15.4.8.1.1.8 Summary

Failure of a control rod housing does not cause damage to adjacent housings that increase the severity of the initial accident.

15.4.8.1.2 Limiting Criteria

This event is a Condition IV incident (ANSI N18.2). See Subsection 15.0.1 for a discussion of ANS classification. Because of the extremely low probability of an RCCA ejection accident, some fuel damage is considered an acceptable consequence.

NUREG-0800 Standard Review Plan (SRP) 4.2 Revision 3 (Reference 24) interim criteria applicable to new plant design certification are applied to provide confidence that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are the following:

- The pellet clad mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 of SRP 4.2 Revision 3 Appendix B.

- The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure.

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- For intermediate (greater than 5% rated thermal power) and full power conditions, fuel cladding is presumed to fail if local heat flux exceeds thermal design limits (e.g. DNBR).
- For core coolability, it is conservatively assumed that the average fuel pellet enthalpy at the hot spot remains below 200 cal/g (360 btu/lb) for irradiated fuel. This bounds non-irradiated fuel, which has a slightly higher enthalpy limit.
- For core coolability, the peak fuel temperature must remain below incipient fuel melting conditions.
- Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.
- Peak reactor coolant system pressure is less than that which could cause stresses to exceed the “Service Limit C” as defined in the ASME code.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transients is performed in two stages: first, an average core calculation and then, a hot rod calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity). Enthalpy, fuel temperature and DNB transients are then determined by performing a conservative fuel rod transient heat transfer calculation.

A discussion of the method of analysis appears in WCAP-15806-P-A (Reference 4).

Average Core Analysis

The three-dimensional nodal code ANC (References 14, 15, 16, 17, 21, 22, and 27) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in 3 spatial dimensions (rectangular coordinates) for 6 delayed neutron groups. The core moderator and fuel temperature feedbacks are based on the NRC approved Westinghouse version of the VIPRE-01 code and methods (References 18 and 19).

Hot Rod Analysis

The hot fuel rod models are based on the Westinghouse VIPRE models described in WCAP-15806-P-A (Reference 4). The hot rod model represents the hottest fuel rod from any channel in the core. VIPRE performs the hot rod transients for fuel enthalpy, temperature and DNBR using as input the time-dependent nuclear core power and power distribution from the core average analysis. A description of the VIPRE code is provided in Reference 18.

System Overpressure Analysis

If the fuel coolability limits are not exceeded, the fuel dispersal into the coolant or a sudden pressure increase from thermal to kinetic energy conversion is not needed to be considered in the overpressure analysis. Therefore, the overpressure condition may be calculated on the basis of conventional fuel rod to coolant heat transfer and the prompt heat generation in the coolant. The system overpressure analysis is conducted by first performing the core power response analysis to

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obtain the nuclear power transient (versus time) data. The nuclear power data is then used as input to a plant transient computer code to calculate the peak reactor coolant system pressure. This code calculates the pressure transient, taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. For conservatism, no credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.2.1 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected as described in Reference 4.

15.4.8.2.1.1 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using three-dimensional methods. Standard nuclear design codes are used in the analysis. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate safety analysis allowances are added to the ejected rod worth and hot channel factors to account for calculational uncertainties, including an allowance for nuclear peaking due to densification as discussed in Reference 4.

15.4.8.2.1.2 Not Used

15.4.8.2.1.3 Moderator and Doppler Coefficients

The critical boron concentration is adjusted in the nuclear code to obtain a moderator temperature coefficient that is conservative compared to actual design conditions for the plant consistent with Reference 4. The fuel temperature feedback in the neutronics code is reduced consistent with Reference 4 requirements.

15.4.8.2.1.4 Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.50 percent at the end of cycle. The accident is sensitive to β_{eff} if the ejected rod worth is equal to or greater than β_{eff}. To allow for future cycles, a pessimistic estimate of β_{eff} of 0.44 percent is used in the analysis.

15.4.8.2.1.5 Trip Reactivity Insertion

The trip reactivity insertion accounts for the effect of the ejected rod and one adjacent stuck rod. The trip reactivity is simulated by dropping a limited set of rods of the required worth into the core. The start of rod motion occurs 0.9 second after either the power range high positive flux rate or the high or low setting of the high neutron flux setpoint is reached. A curve of trip rod insertion versus time is used, which assumes that insertion to the dashpot does not occur until 2.7 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This conservatism is important for the hot full power accidents.

The minimum design shutdown margin available at hot zero power may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Calculations show that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1-percent Δk. Therefore, following a reactor trip

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resulting from an RCCA ejection accident, the reactor is subcritical when the core returns to hot zero power.

15.4.8.2.1.6 Reactor Protection

As discussed in Subsection 15.4.8.1.1.3, reactor protection for a rod ejection is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are part of the protection and safety monitoring system. No single failure of the protection and safety monitoring system negates the protection functions required for the rod ejection accident or adversely affects the consequences of the accident.

15.4.8.2.1.7 Results

For all cases, the core is preconditioned by assuming a fuel cycle depletion with control rod insertion that is conservative relative to expected baseload operation. All cases assume that the mechanical shim and axial offset control RCCAs are inserted to their insertion limits before the event and xenon is skewed to yield a conservative initial axial power shape. The limiting RCCA ejection cases for a typical cycle are summarized following the criteria outlined in Subsection 15.4.8.1.2.

- Pellet-Clad Mechanical Interaction (PCMI) and High Clad Temperature (Hot Zero Power)

  The resulting maximum fuel average enthalpy rise and maximum fuel average enthalpy are less than the criteria given in Subsection 15.4.8.1.2.

- High Clad Temperature (≥5% Rated Thermal Power)

  The fraction of the core calculated to have a DNBR less than the safety analysis limit is less than the amount of failed fuel assumed in the dose analysis described in Subsection 15.4.8.3.

- Core Coolability

  The resulting maximum fuel average enthalpy is less than the criterion given in Subsection 15.4.8.1.2. Fuel melting is not predicted to occur at the hot spot.

There are no fuel failures due to the fuel enthalpy deposition, i.e., both fuel and cladding enthalpy limits were met. Additionally, the coolability criteria for peak fuel enthalpy and the fuel melting criteria were met. Therefore, the fuel dispersal into the coolant, a sudden pressure increase from thermal to kinetic energy conversion, gross lattice distortion, or severe shock waves are precluded.

The nuclear power transients for the limiting cases are presented in Figures 15.4.8-1 through 15.4.8-3.

The calculated sequence of events for the limiting cases are presented in Table 15.4-1. Reactor trip occurs early in the transients, after which the nuclear power excursion is terminated.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents (LOCAs) are discussed in Subsection 15.6.5. Following the RCCA ejection, the plant response is the same as a LOCA.

The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the enthalpy and temperature transients resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak fuel and cladding temperatures occur before the reactor coolant pumps begin to coast down.

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15.4.8.2.1.8 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In the cases considered, less than 10 percent of the rods are assumed to enter DNB based on a detailed three-dimensional kinetics and hot rod analysis. The maximum fuel average enthalpy rise of rods predicted to enter DNB will be less than 60 cal/g. Fuel melting does not occur at the hot spot.

The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the calculation of the number of rods assumed to enter DNB for the RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR, for rods where the DNBR did not fall below the design limit (see Section 4.4) in the cases described, occurs before the reactor coolant pumps begin to coast down.

15.4.8.2.1.9 Peak RCS Pressure

Calculations of the peak reactor coolant system pressure demonstrate that the peak pressure does not exceed that which would cause the stress to exceed the Service Level C Limit as described in the ASME Code, Section III. Therefore, the accident for this plant does not result in an excessive pressure rise or further damage to the reactor coolant system.

The consequential loss of offsite power described in Subsection 15.0.14 is not limiting for the pressure surge transient resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak system pressure occurs before the reactor coolant pumps begin to coast down.

15.4.8.2.1.10 Lattice Deformations

A large temperature gradient exists in the region of the hot spot. Because the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion, tending to bow the midpoint of the rods toward the hotter side of the rod.

Calculations indicate that this bowing results in a negative reactivity effect at the hot spot because the core is undermoderated, and bowing tends to increase the undermoderation at the hot spot. In practice, no significant bowing is anticipated because the structural rigidity of the core is sufficient to withstand the forces produced.

Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that crossflow is sufficient to produce lattice deformation. Even if massive and rapid boiling, sufficient to distort the lattices, is hypothetically postulated, the large void fraction in the hot spot region produces a reduction in the total core moderator to fuel ratio and a large reduction in this ratio at the hot spot. The net effect is therefore a negative feedback.

In conclusion, no credible mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes result in a buildup of activity in the secondary coolant. Refer to Subsection 15.4.8.3.1 and Table 15.4-4.

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As a result of the accident, 10 percent of the fuel rods are assumed to be damaged (see Subsection 15.4.8.2.1.8) such that the activity contained in the fuel-cladding gap is released to the reactor coolant. No fuel melt is calculated to occur as a result of the rod ejection (see Subsection 15.4.8.2.1.8).

Activity released to the containment via the spill from the reactor vessel head is assumed to be available for release to the environment because of containment leakage. Activity carried over to the secondary side due to primary-to-secondary leakage is available for release to the environment through the steam line safety or power-operated relief valves.

15.4.8.3.1 Source Term

The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals, and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant alkali metal concentrations are based on those associated with the design basis fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.

Based on NUREG-1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fractions are modified following the guidance of Draft Guide 1199 (Reference 25), which incorporates the effects of enthalpy rise in the fuel following the reactivity insertion, consistent with Appendix B of SRP 4.2, Revision 3 (Reference 24). Draft Guide 1199 included expanded guidance for determining nuclide gap fractions available for release following a rod ejection. Reference 26 was issued as a clarification to the gap fraction guidance in Draft Guide 1199. An enthalpy rise of 60 cal/gm is used to calculate the gap fractions (see Subsection 15.4.8.2.1.8). Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor. No fuel melt is calculated to occur as a result of the rod ejection (see Subsection 15.4.8.2.1.8).

The initial secondary coolant activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

15.4.8.3.2 Release Pathways

There are three components to the accident releases:

- The activity initially in the secondary coolant is available for release as long as steam releases continue.

- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant and, as steam is released, a portion of the iodine and alkali metal in the coolant is released. The fraction of activity released is defined by the assumed flashing fraction and the partition coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.

- The activity from the reactor coolant system and the core is released to the containment atmosphere and is available for leakage to the environment through the assumed design basis containment leakage.

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Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.4.8.3.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.4.8.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.4-4.

15.4.8.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.
- The reactor coolant activities are based on conservative assumptions (refer to Table 15.4-4); whereas the activities based on the expected fuel defect level are far less (see Section 11.1).
- The leakage of reactor coolant into the secondary system, at 300 gallons per day, is conservative. The leakage is normally a small fraction of this.
- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.
- The leakage from containment is assumed to continue for a full 30 days. It is expected that containment pressure is reduced to the point that leakage is negligible before this time.

15.4.8.3.6 Doses

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be 4.0 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 5.9 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem total effective dose equivalent identified in 10 CFR Part 50.34. The phrase “well within” is taken as being 25 percent or less.

At the time the rod ejection accident occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the rod ejection accident, the resulting total dose remains less than the value reported above.

15.4.9 Combined License Information

This section contained no requirement for additional information.

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15.4.10 References

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Table 15.4-1 (Sheet 1 of 3)
Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time (seconds)
Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux (low setting) setpoint reached	10.4
	Peak nuclear power occurs	10.6
	Rods begin to fall into core	11.3
	Peak heat flux occurs	12.9
	Minimum DNBR occurs	12.9
	Peak average clad temperature occurs	13.5
	Peak average fuel temperature occurs	13.7
One or more dropped RCCAs	Rods drop	0.0
	Control system initiates control bank withdrawal	0.4
	Peak nuclear power occurs	21.7
	Peak core heat flux occurs	24.2
Uncontrolled RCCA bank withdrawal at power		
1. Case A - Full power with maximum reactivity feedback	Initiation of uncontrolled RCCA withdrawal at a fast reactivity insertion rate (80 pcm/s)	0.0
	Power range high positive flux rate trip point reached	5.2
	Rods begin to fall into core	6.1
	Minimum DNBR occurs	6.4
2. Case B - Full power with maximum reactivity feedback	Initiation of uncontrolled RCCA withdrawal at an intermediate reactivity insertion rate (34 pcm/s)	0.0
	Overpower ΔT setpoint reached	18.0
	Rods begin to fall into core	19.9
	Minimum DNBR occurs	20.1
3. Case C - Full power with maximum reactivity feedback	Initiation of uncontrolled RCCA withdrawal at a slow reactivity insertion rate (5 pcm/s)	0.0
	Overtemperature ΔT setpoint reached	568.3
	Rods begin to fall into core	570.3
	Minimum DNBR occurs	570.4

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Table 15.4-1 (Sheet 2 of 3)
Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time (minutes)
Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant		
Dilution during power operation (Mode 1)		
Automatic reactor control	Operator receives low-low rod insertion limit alarm due to dilution	0.0
	Shutdown margin lost	170.6
Manual reactor control	Dilution initiated	0.0
	Reactor trip on overtemperature ΔT due to dilution	3.3
	Dilution automatically terminated by demineralized water transfer and storage system isolation	3.8
Dilution during startup (Mode 2)	Power range high neutron flux-low setpoint reactor trip due to dilution	0.0
	Shutdown margin lost	205.3
Dilution during hot standby (Mode 3)	Dilution initiated	0.0
	Boron dilution protection system setpoint reached, which initiates isolation of the dilution source	32.1
	Shutdown margin lost	39.6
Dilution during safe shutdown (Mode 4)	Dilution initiated	0.0
	Boron dilution protection system setpoint reached, which initiates isolation of the dilution source	28.8
	Shutdown margin lost	35.6
Dilution during cold shutdown (Mode 5)	Dilution initiated	0.0
	Boron dilution protection system setpoint reached, which initiates isolation of the dilution source	30.8
	Shutdown margin lost	38.1
Dilution during refueling (Mode 6)	Prevented by administrative control	N/A

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Table 15.4-1 (Sheet 3 of 3)
Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies

Accident	Event	Time (seconds)
RCCA ejection accident		
1. PCMI Limiting Event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.14
	Reactor trip setpoint reached	< 0.30
	Peak cladding temperature occurs	0.36
	Peak enthalpy deposition occurs	0.44
	Rods begin to fall into core	1.20
2. Peak Clad Temperature Limiting Event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.08
	Minimum DNBR occurs	0.11
	Peak cladding temperature occurs	0.11
	Reactor trip setpoint reached	< 0.30
	Rods begin to fall into core	1.20
3. Peak enthalpy / Peak Fuel Centerline Temperature Event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.06
	Reactor trip setpoint reached	< 0.30
	Rods begin to fall into core	1.20
	Peak fuel center temperature occurs	2.50
	Peak cladding temperature occurs	2.80

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Table 15.4-2
Key Input Parameters for Boron Dilution

Dilution Flow Rates		
Mode[1]	**Flow Rate (gal/min)**	**Flow Rate (m^3/hr)**
1 through 5	175	39.75
Active RCS Volume		
Mode[1]	**Volume (ft^3)**	**Volume (m^3)**
1 and 2	8425.5	238.584
3, 4 and 5	7605.98	215.375
Boron Concentration		
Mode[1]	**Initial Concentration (ppm)**	**Critical Concentration (ppm)**
1	1811	934
2	2031	934
3	1509	1281
4	1649	1449
5	1675	1483

Notes:

1. Uncontrolled boron dilution transient in Mode 6 is prevented by administrative control.

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Table 15.4-3
Not Used

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Table 15.4-4 (Sheet 1 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Rod Ejection Accident

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g (2.22E+06 Bq/g) of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g (1.036E+07 Bq/g) dose equivalent Xe-133
Reactor coolant alkali metal activity	Based on design basis activity (see Table 15A-9)
Secondary coolant initial iodine and alkali metal activity	1% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in failed/melted fuel)	1.75
Fuel cladding failure – Fraction of fuel rods assumed to fail – Fuel Enthalpy Increase (cal/gm) – Fission product gap fractions Iodine 131 Iodine 132 Krypton 85 Other Nobles Gases Other Halogens Alkali Metals	 0.1 60 0.1238 0.1338 0.5120 0.1238 0.0938 0.6860
Iodine chemical form (%) – Elemental – Organic – Particulate	 4.85 0.15 95.0
Core activity	See Table 15A-3 in Appendix 15A
Nuclide data	See Table 15A-4 in Appendix 15A
Reactor coolant mass (lb)	3.7 E+05 (1.68E+05 kg)

Note:
a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity assumed to be released from damaged fuel, it is not significant.

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Table 15.4-4 (Sheet 2 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Rod Ejection Accident

Condenser	Not available
Duration of accident (days)	30
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Secondary system release path	
– Primary to secondary leak rate (lb/hr)	104.5[(a)] (47.4 kg/hr)
– Leak flashing fraction	0.04[(b)]
– Secondary coolant mass (lb)	6.06 E+05 (2.75E+05 kg)
– Duration of steam release from secondary system (sec)	1800
– Steam released from secondary system (lb)	1.08 E+05 (4.90E+04 kg)
– Partition coefficient in steam generators	
• Iodine • Alkali metals	0.01 0.0035
Containment leakage release path	
– Containment leak rate (% per day)	
• 0-24 hr • >24 hr	0.10 0.05
– Airborne activity removal coefficients (hr^{-1})	
• Elemental iodine • Organic iodine • Particulate iodine or alkali metals	1.9[(c)] 0 0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	2.78

Notes:
a. Equivalent to 300 gpd (1.14 m^3/day) cooled liquid at 62.4 lb/ft^3 (999.6 kg/m^3).
b. No credit for iodine partitioning is taken for flashed leakage.
c. From Appendix 15B.

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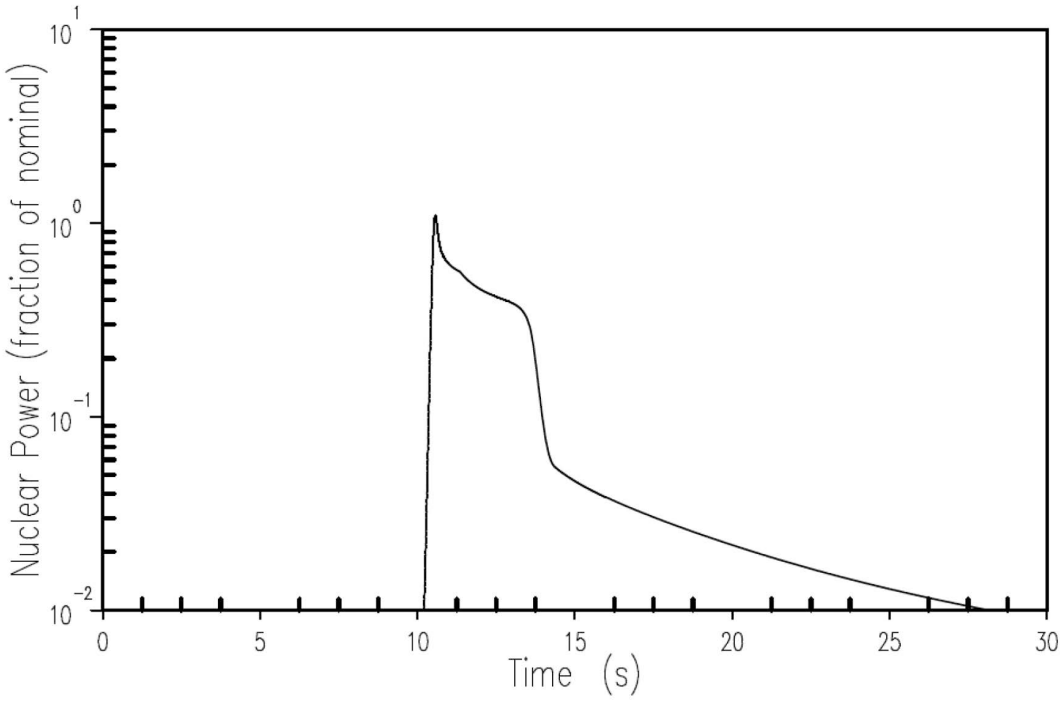

Figure 15.4.1-1
RCCA Withdrawal from Subcritical Nuclear Power

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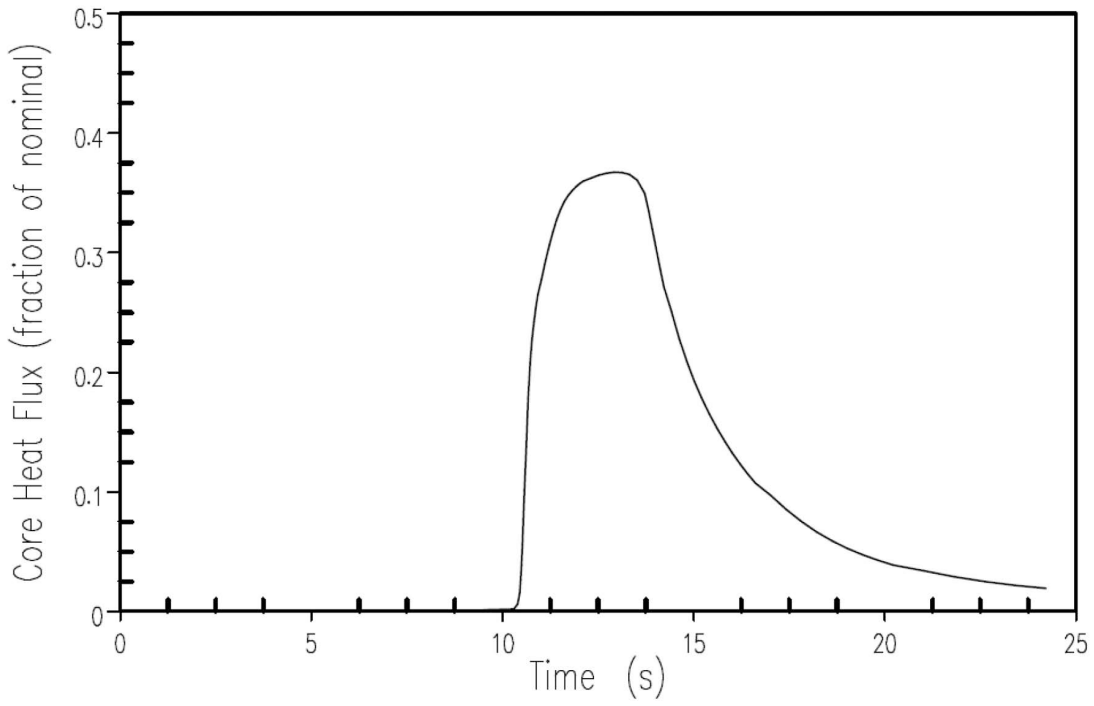

Figure 15.4.1-2
RCCA Withdrawal from Subcritical
Average Channel Core Heat Flux

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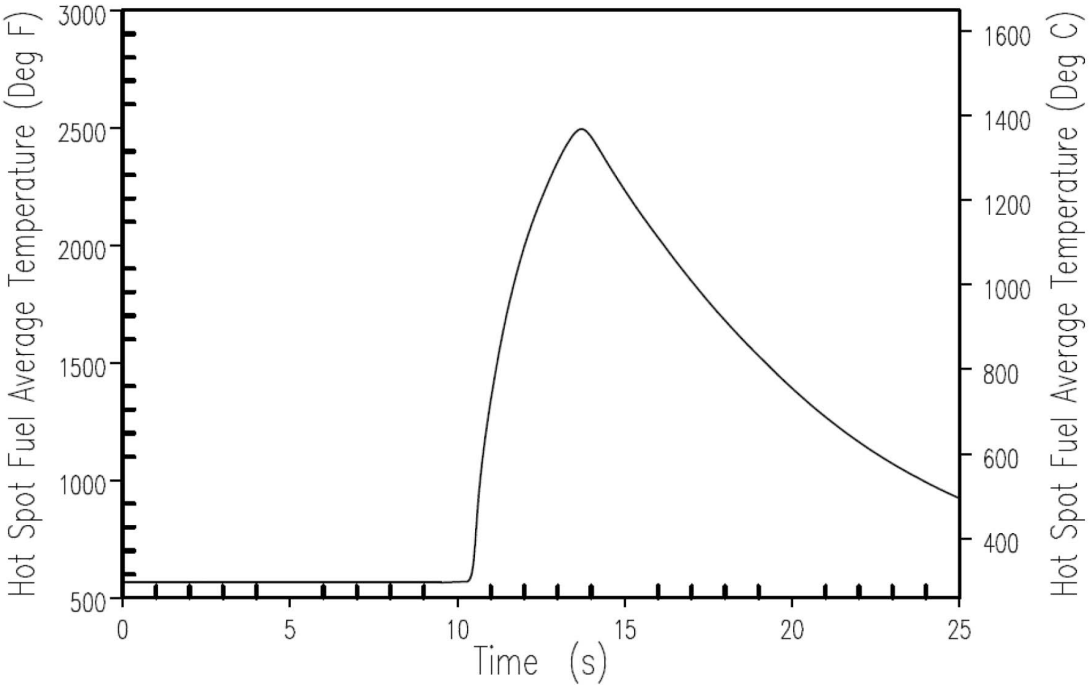

Figure 15.4.1-3
RCCA Withdrawal from Subcritical
Hot Spot Fuel Average Temperature

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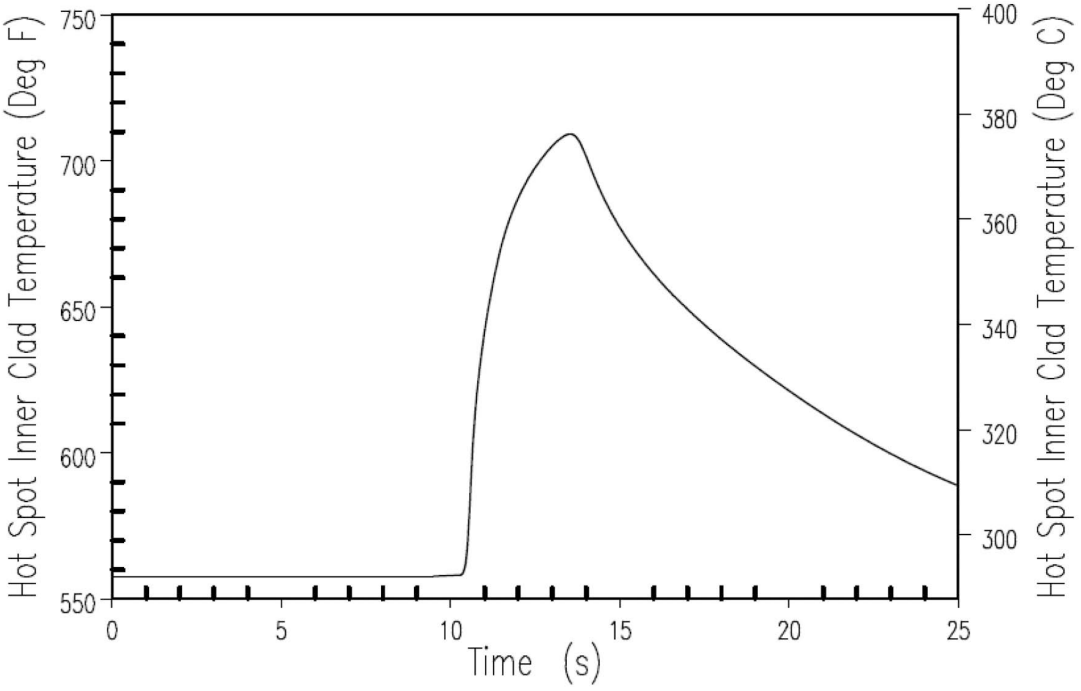

Figure 15.4.1-4
RCCA Withdrawal from Subcritical
Hot Spot Cladding Inner Temperature

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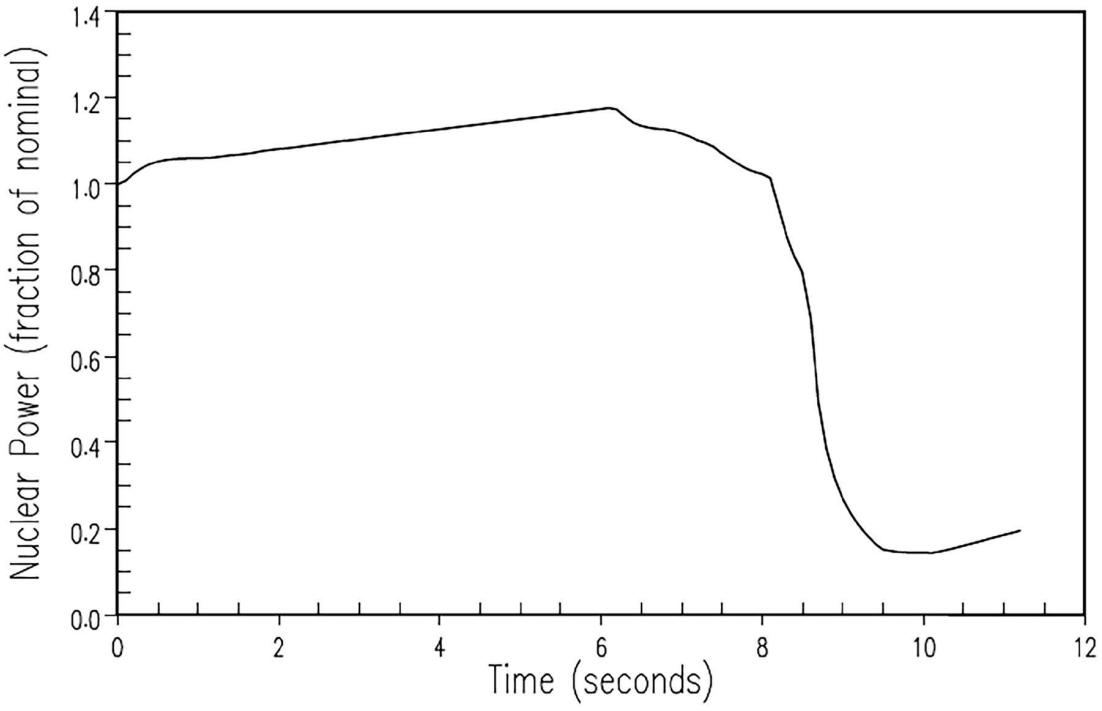

Figure 15.4.2-1
Nuclear Power Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (80 pcm/s)

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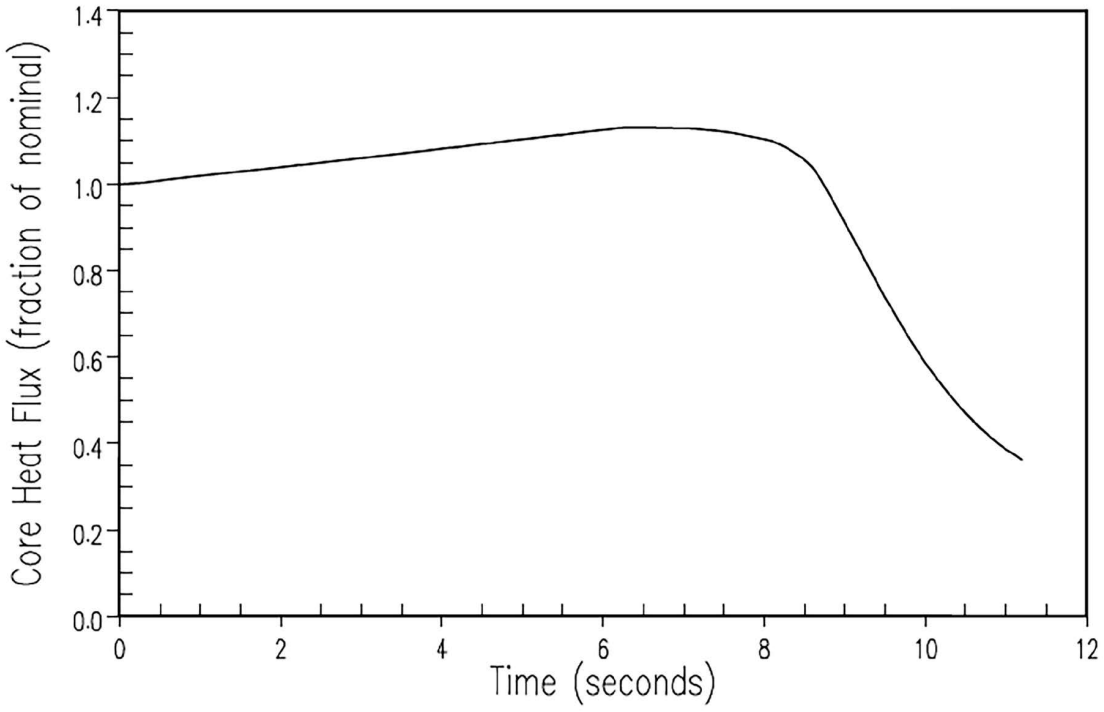

Figure 15.4.2-2
Core Heat Flux Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (80 pcm/s)

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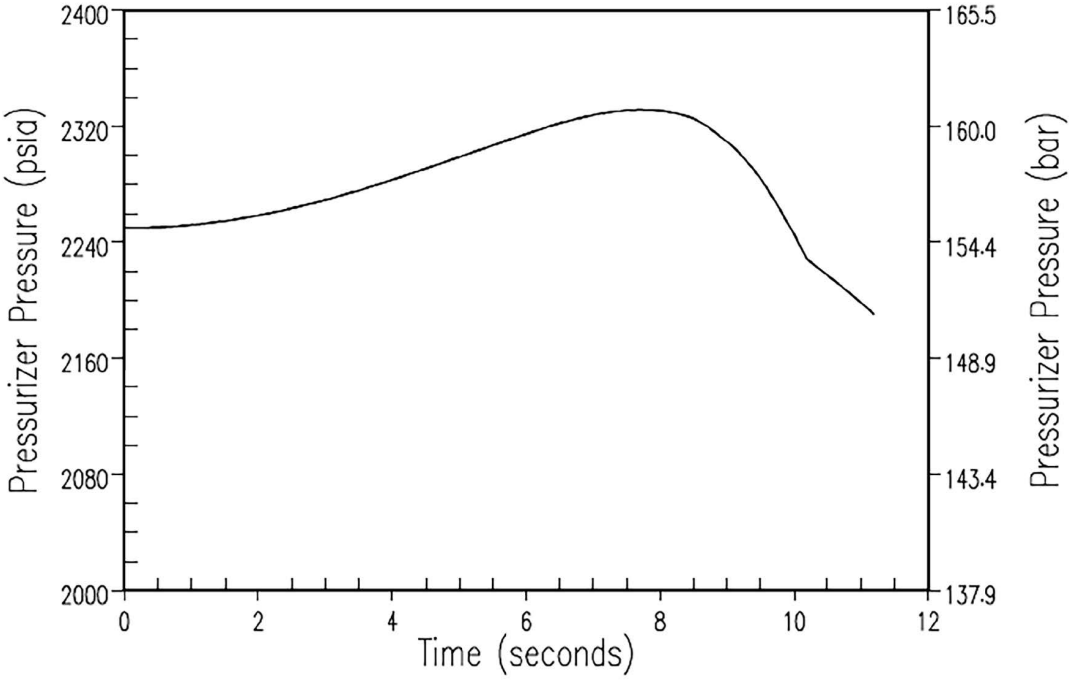

Figure 15.4.2-3
Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s)

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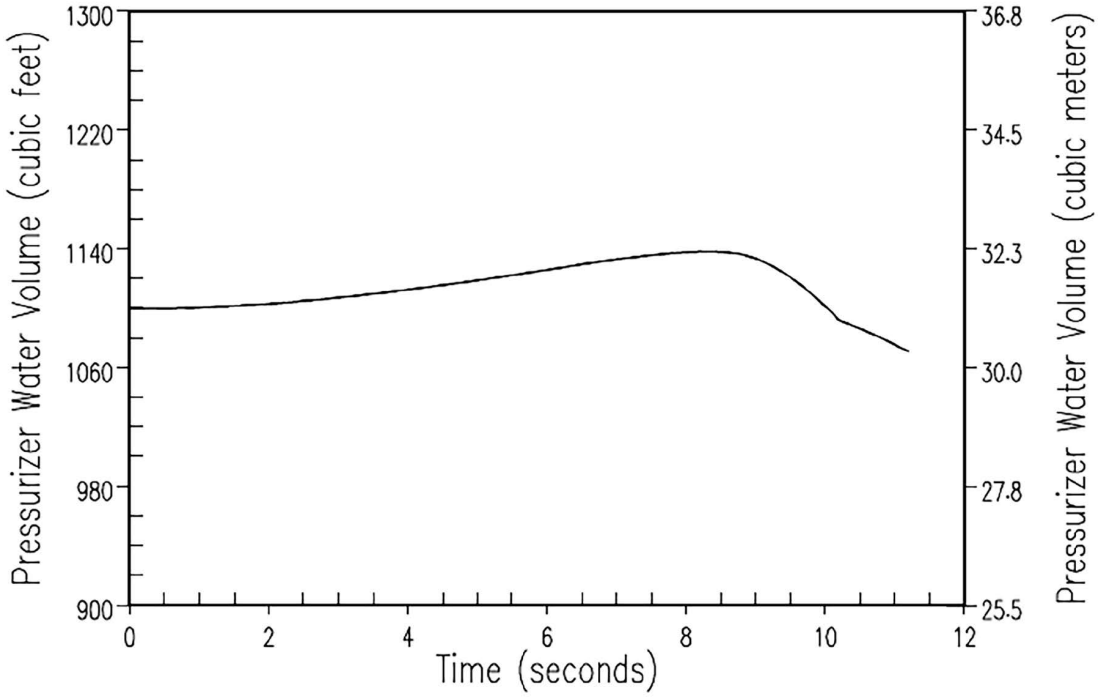

Figure 15.4.2-4
Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s)

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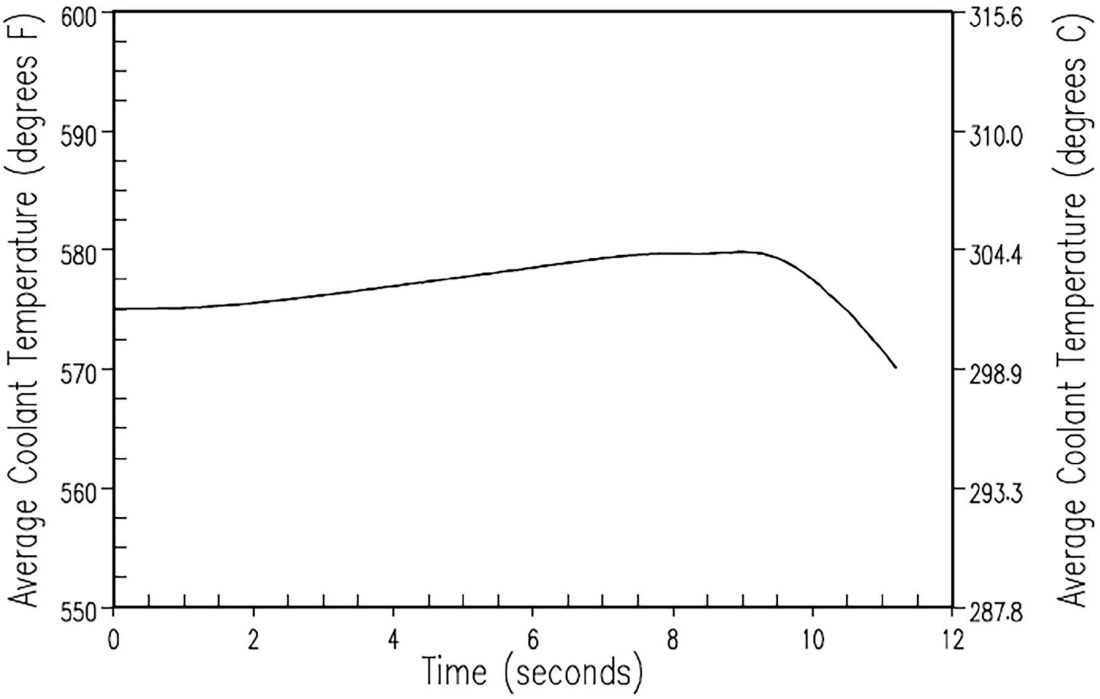

Figure 15.4.2-5
Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s)

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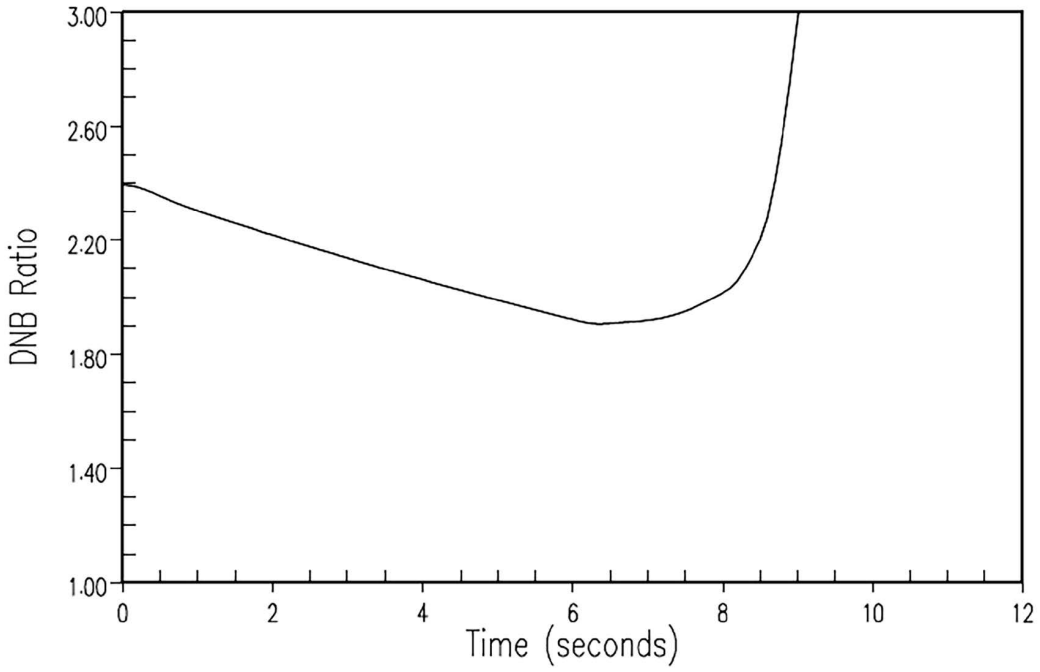

Figure 15.4.2-6
DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (80 pcm/s)

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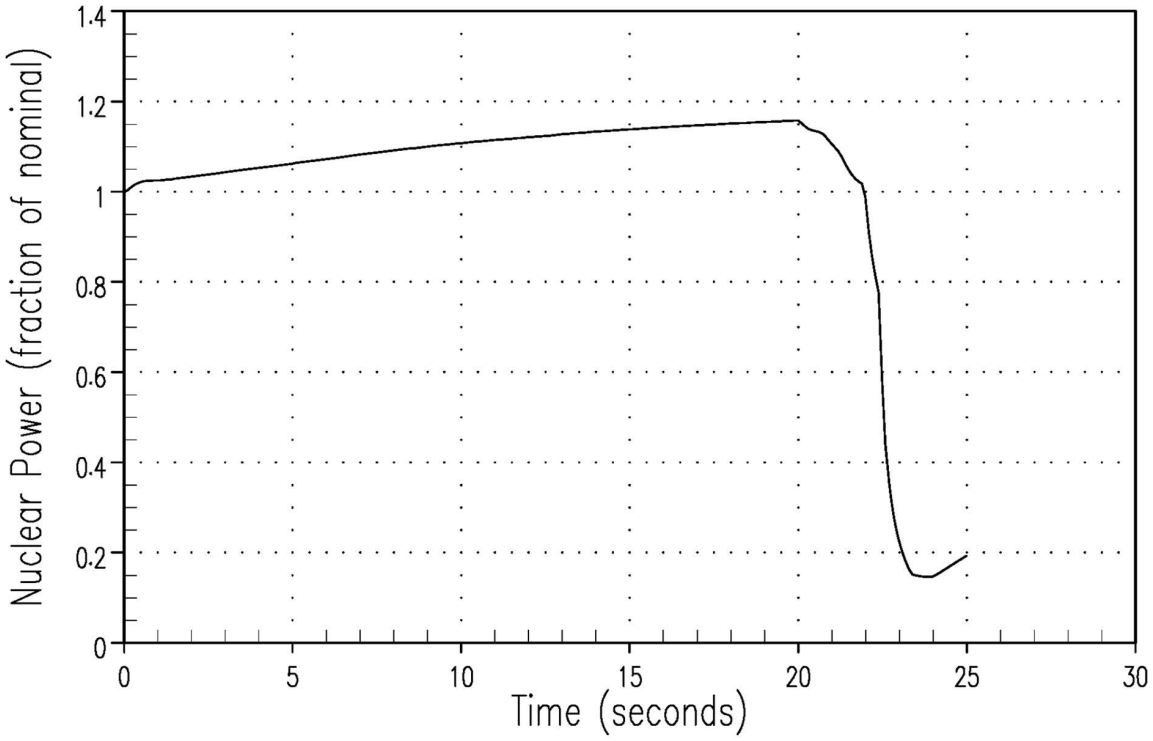

Figure 15.4.2-7
Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

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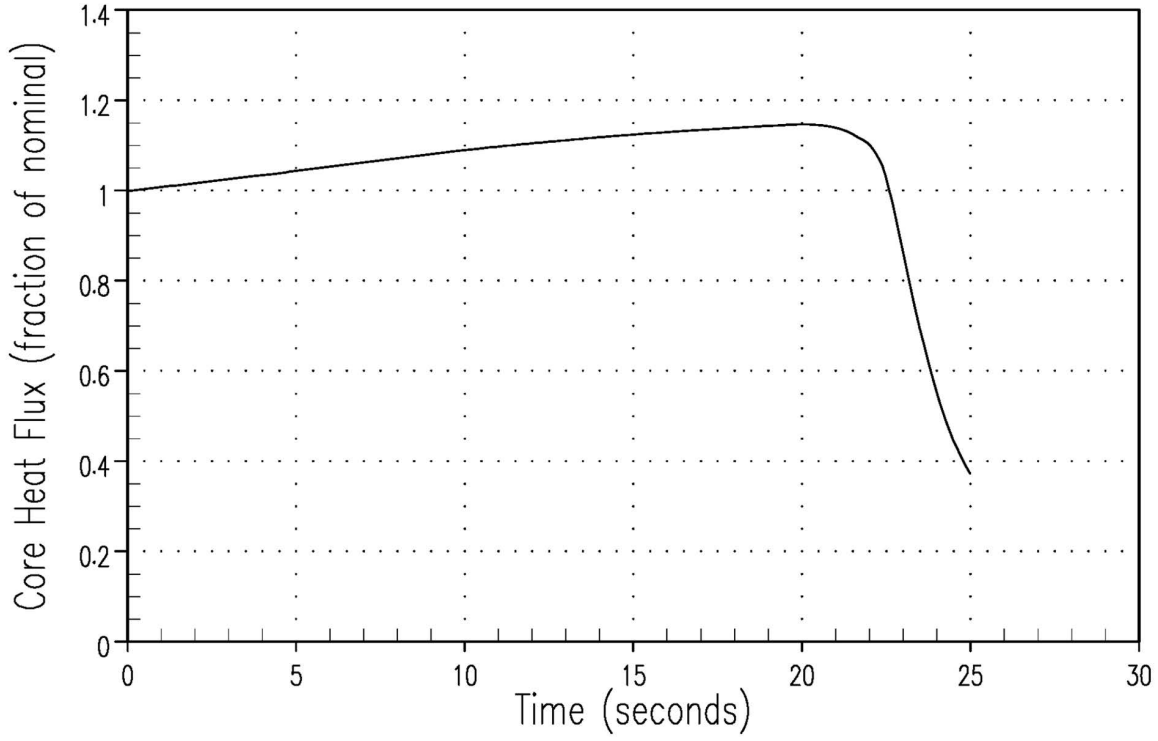

Figure 15.4.2-8
Core Heat Flux Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (34 pcm/s)

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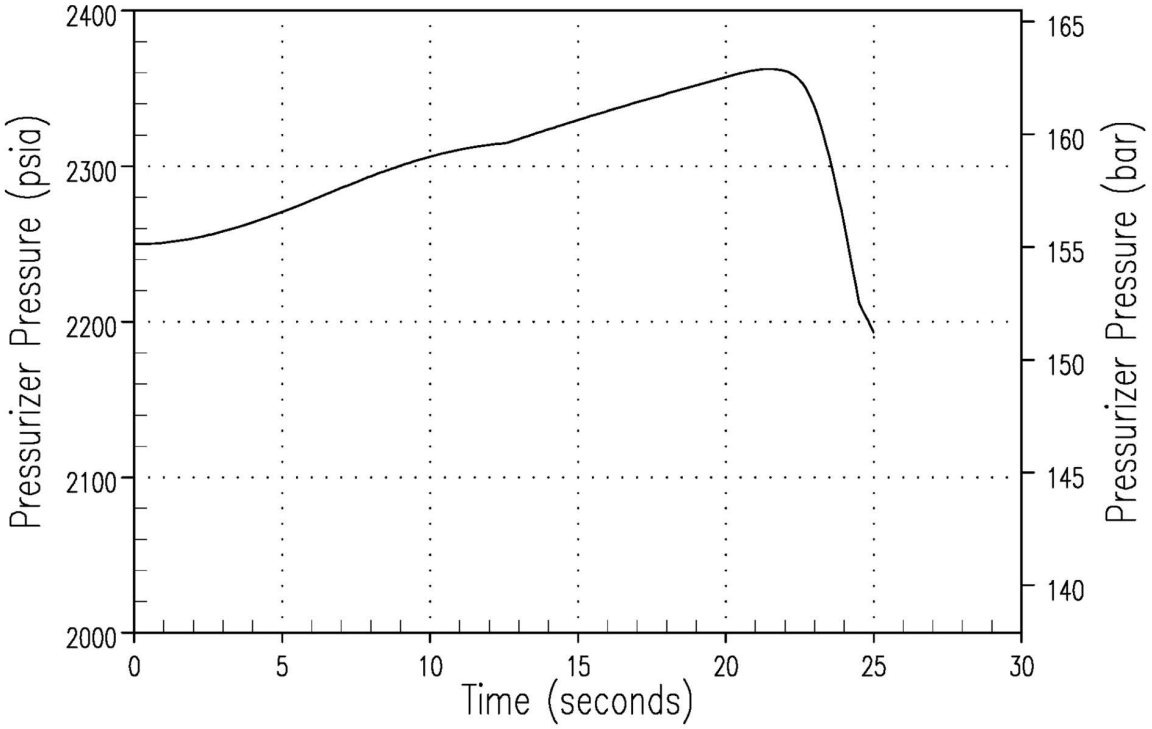

Figure 15.4.2-9
Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

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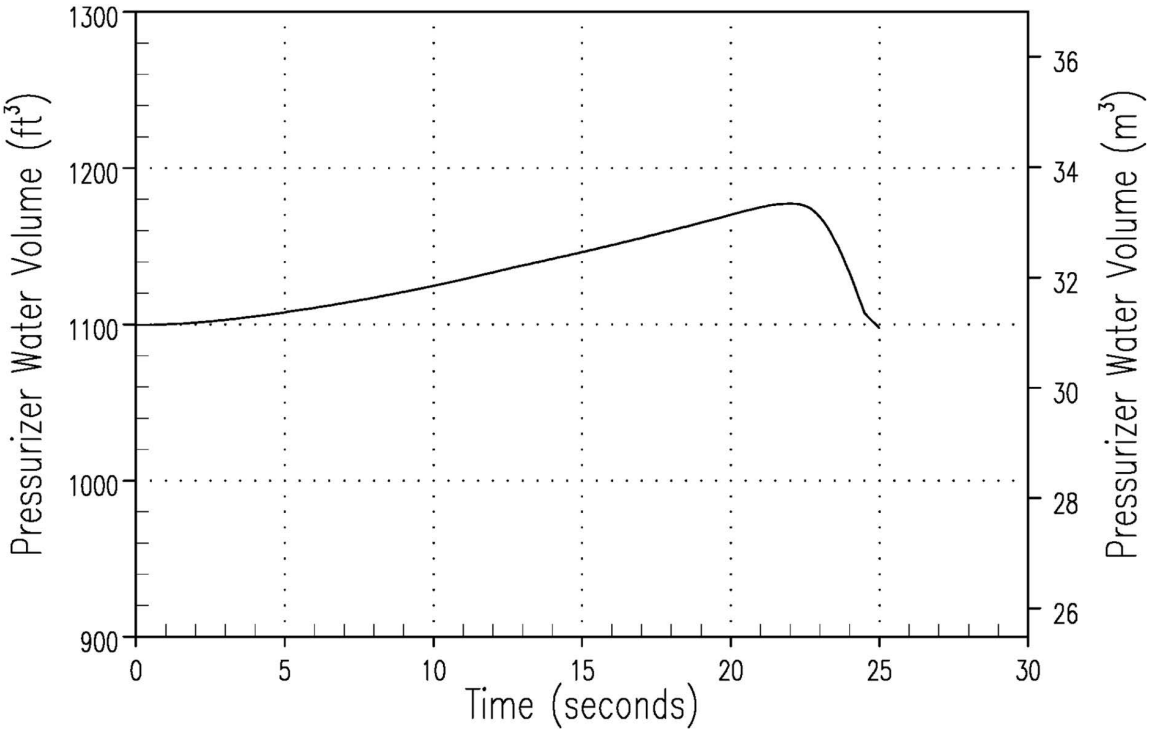

Figure 15.4.2-10
Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

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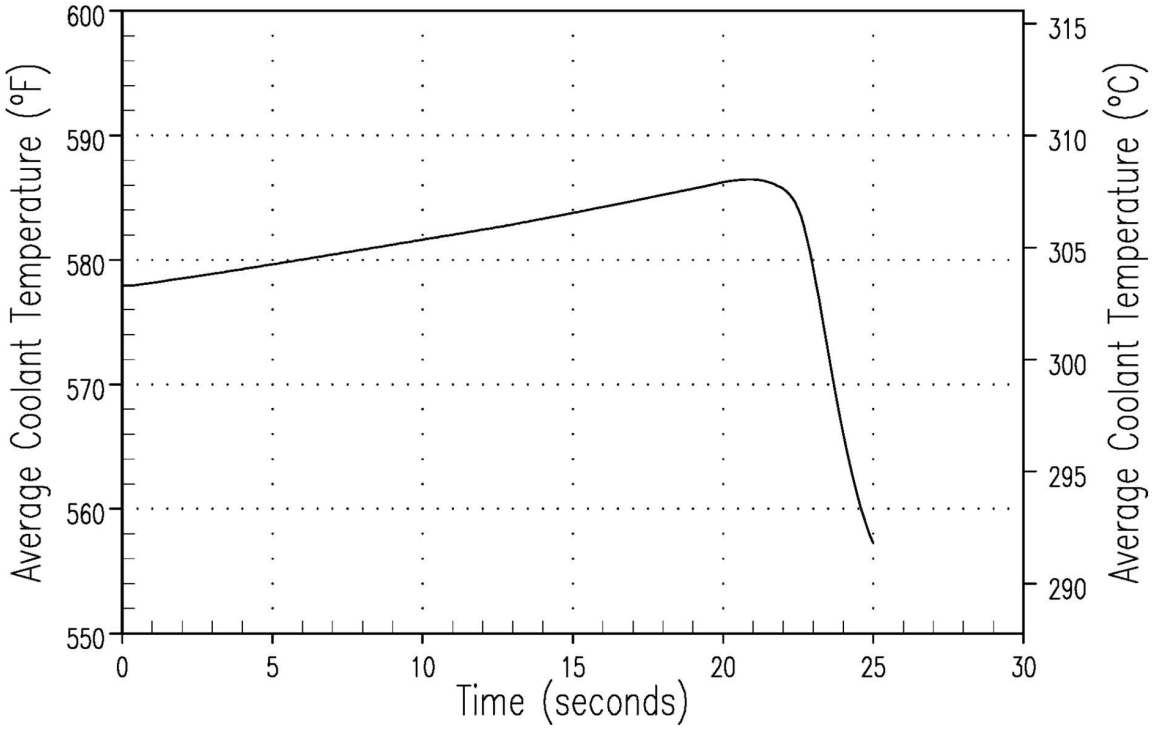

Figure 15.4.2-11
Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (34 pcm/s)

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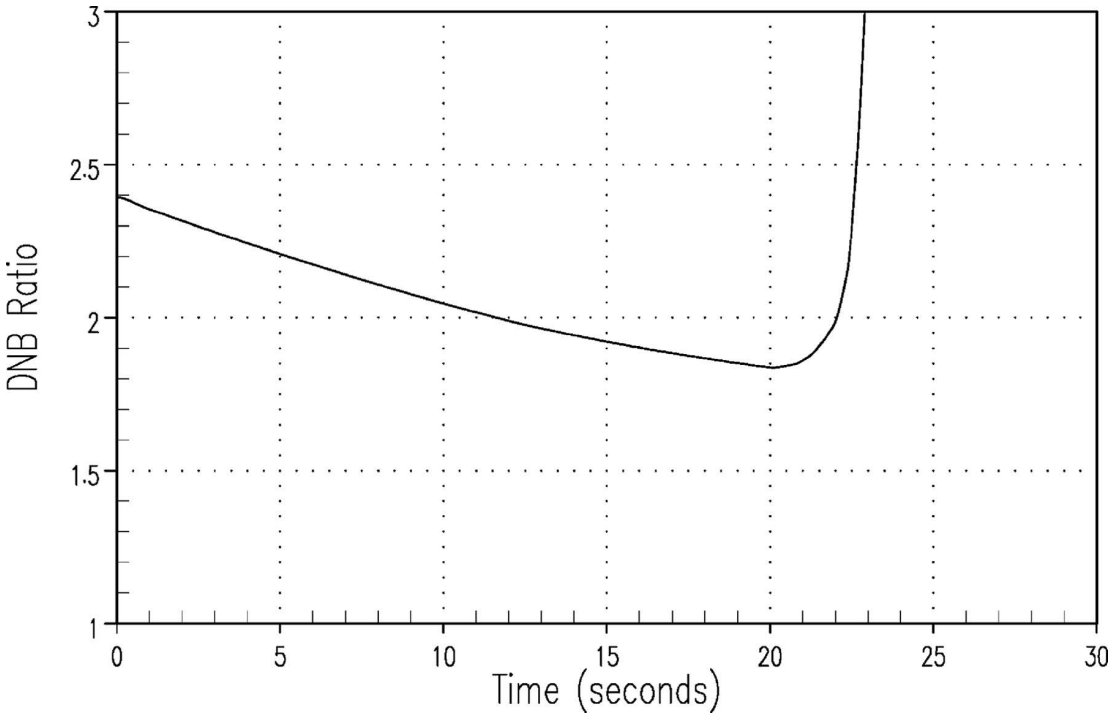

Figure 15.4.2-12
DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (34 pcm/s)

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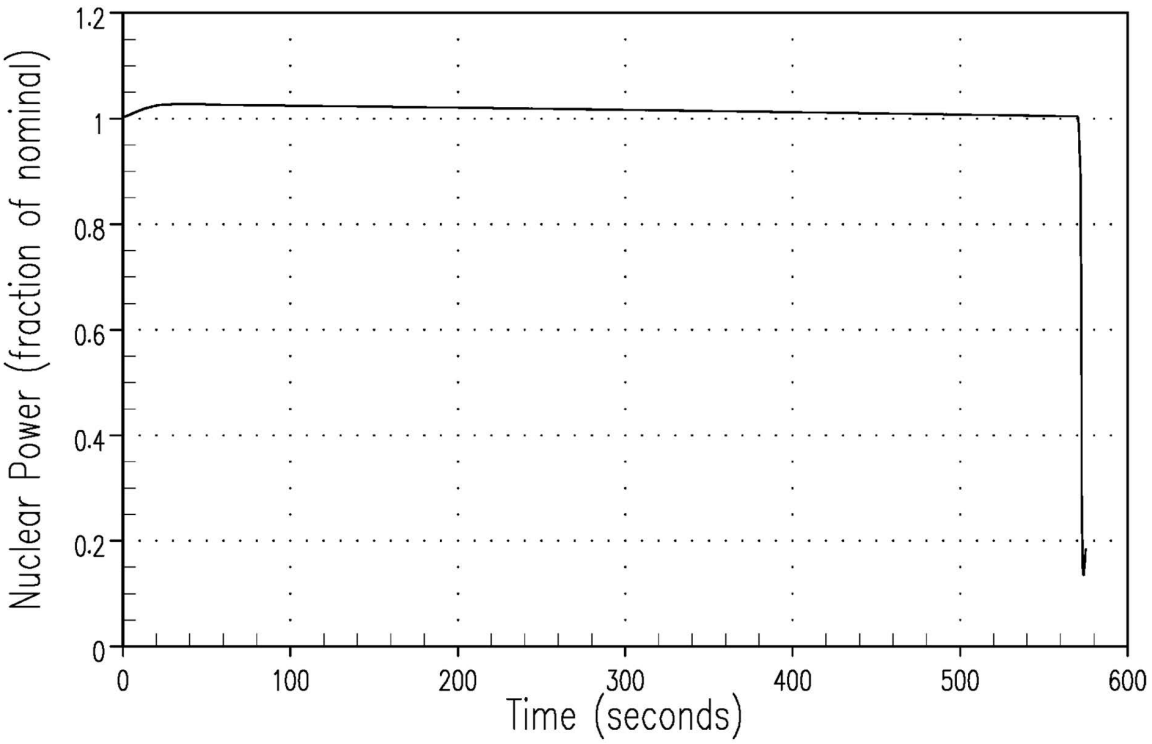

Figure 15.4.2-13
Nuclear Power Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (5 pcm/s)

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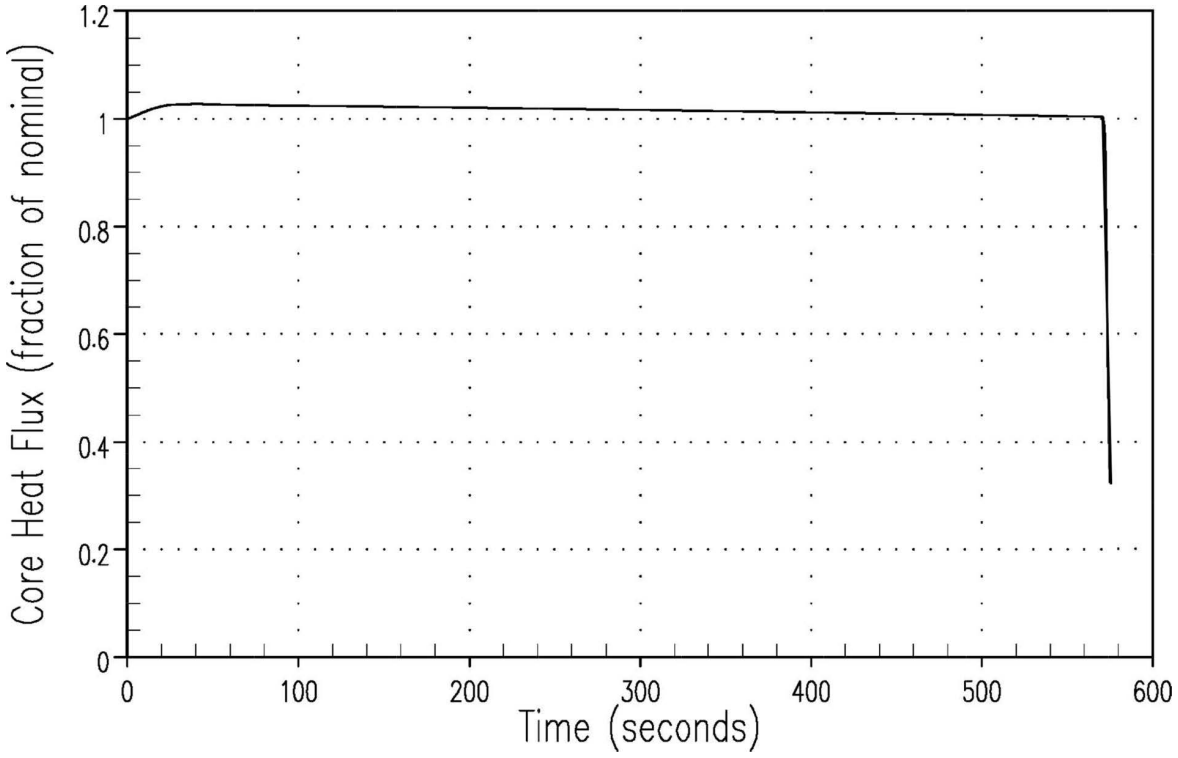

Figure 15.4.2-14
Core Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

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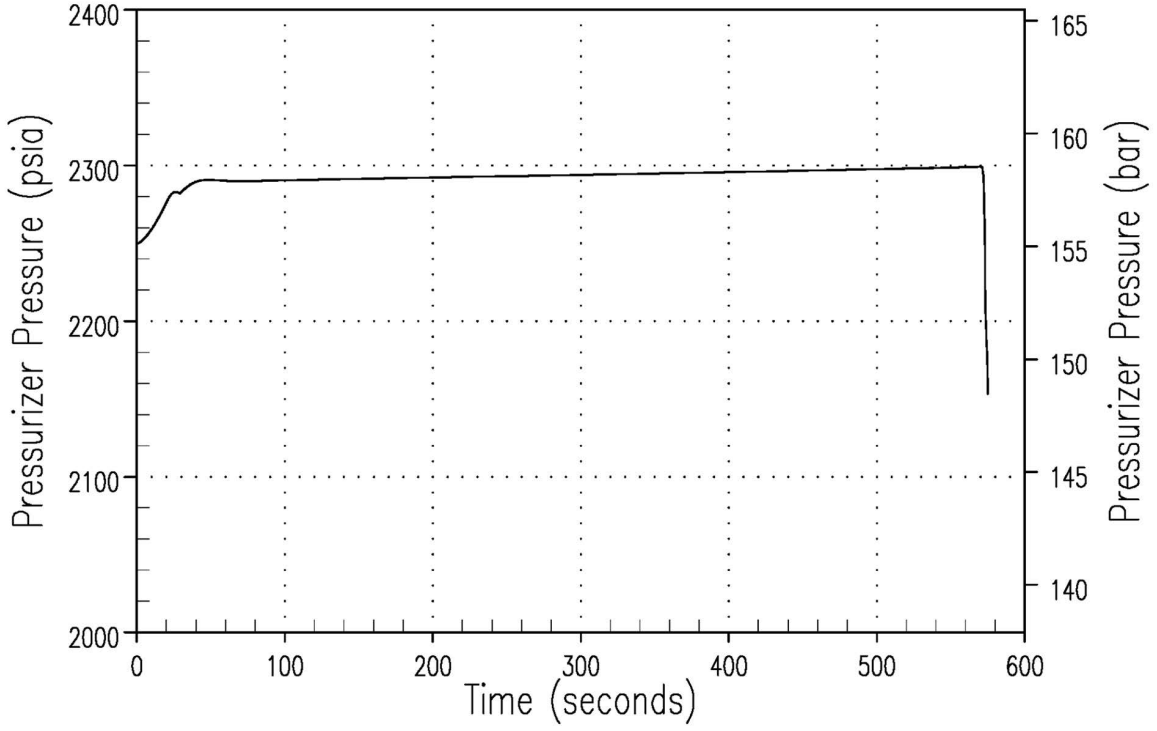

Figure 15.4.2-15
Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

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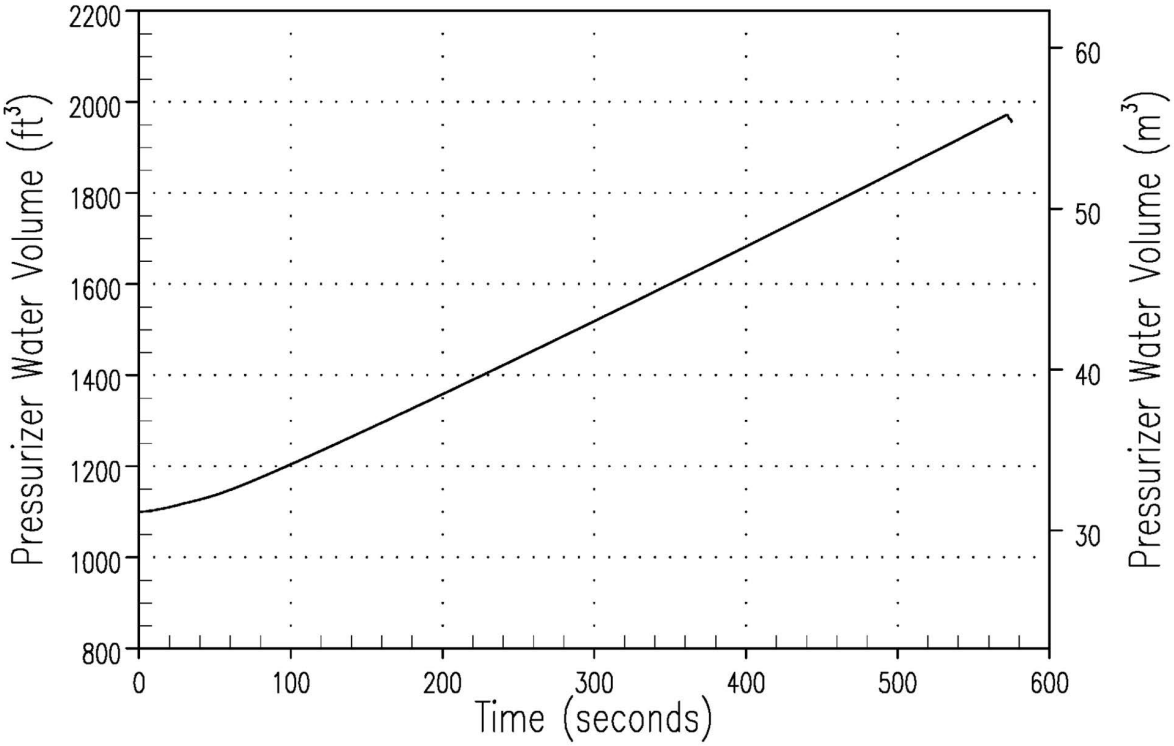

Figure 15.4.2-16
Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

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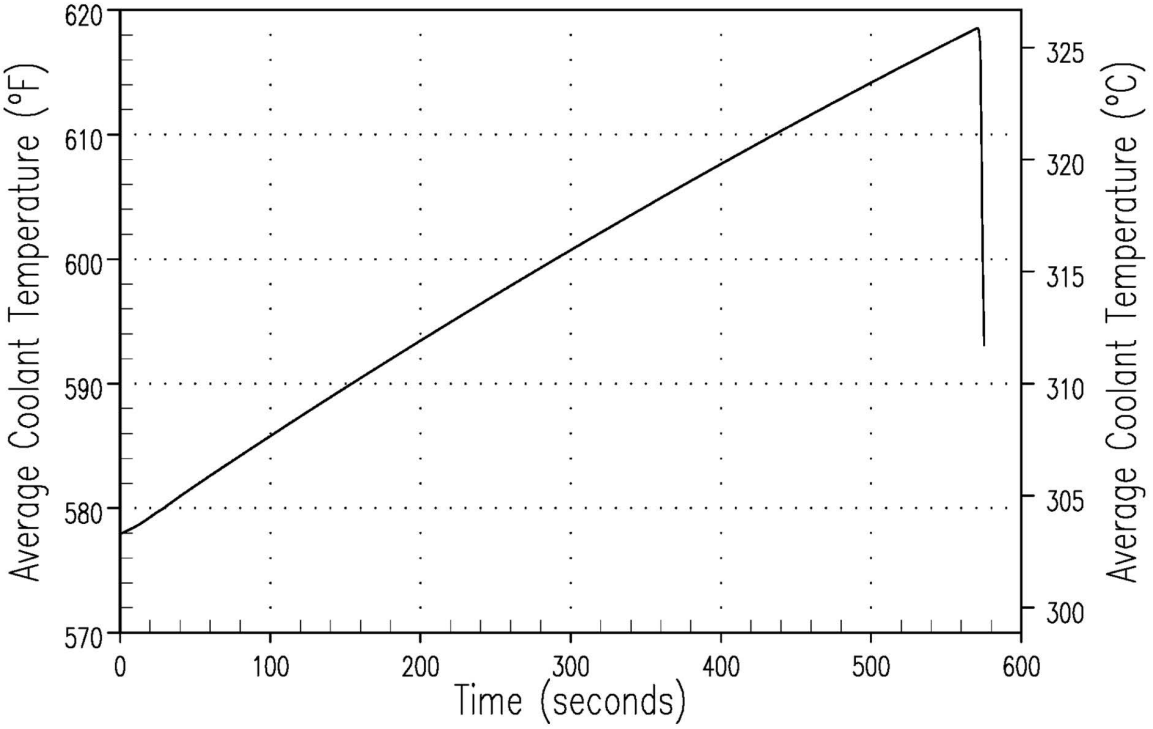

Figure 15.4.2-17
Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

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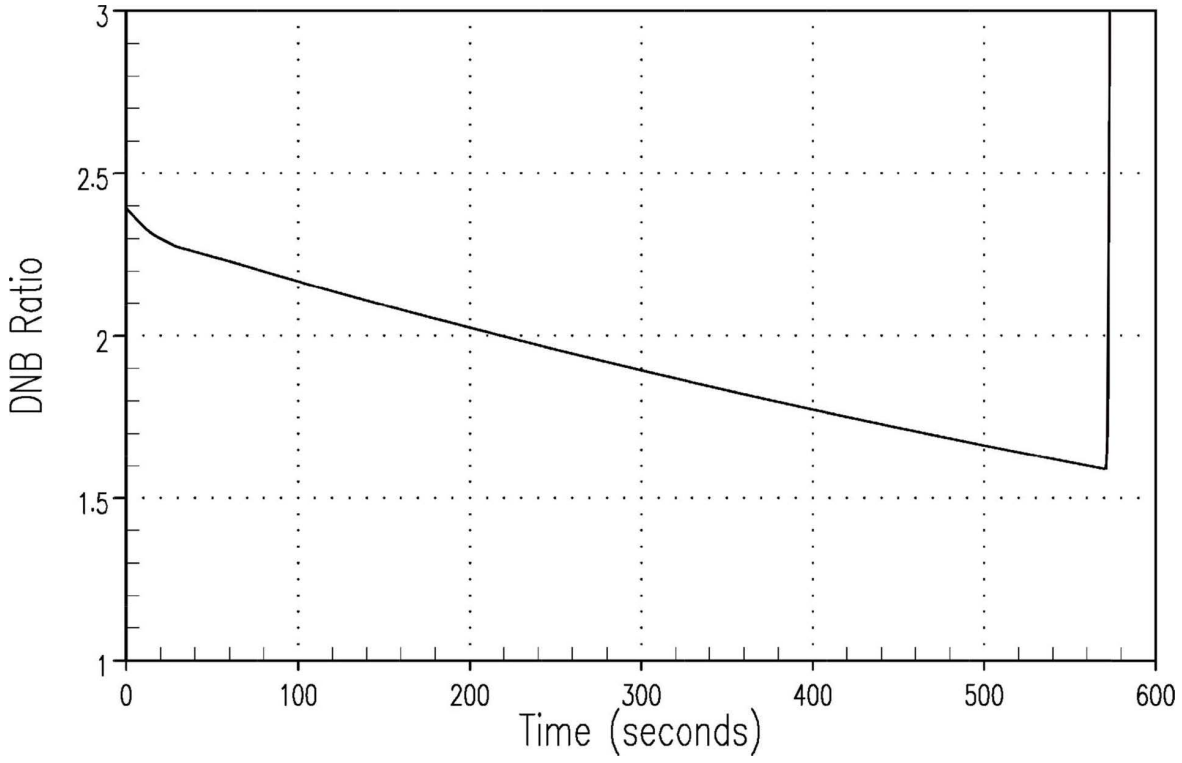

Figure 15.4.2-18
DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
With Maximum Reactivity Feedback (5 pcm/s)

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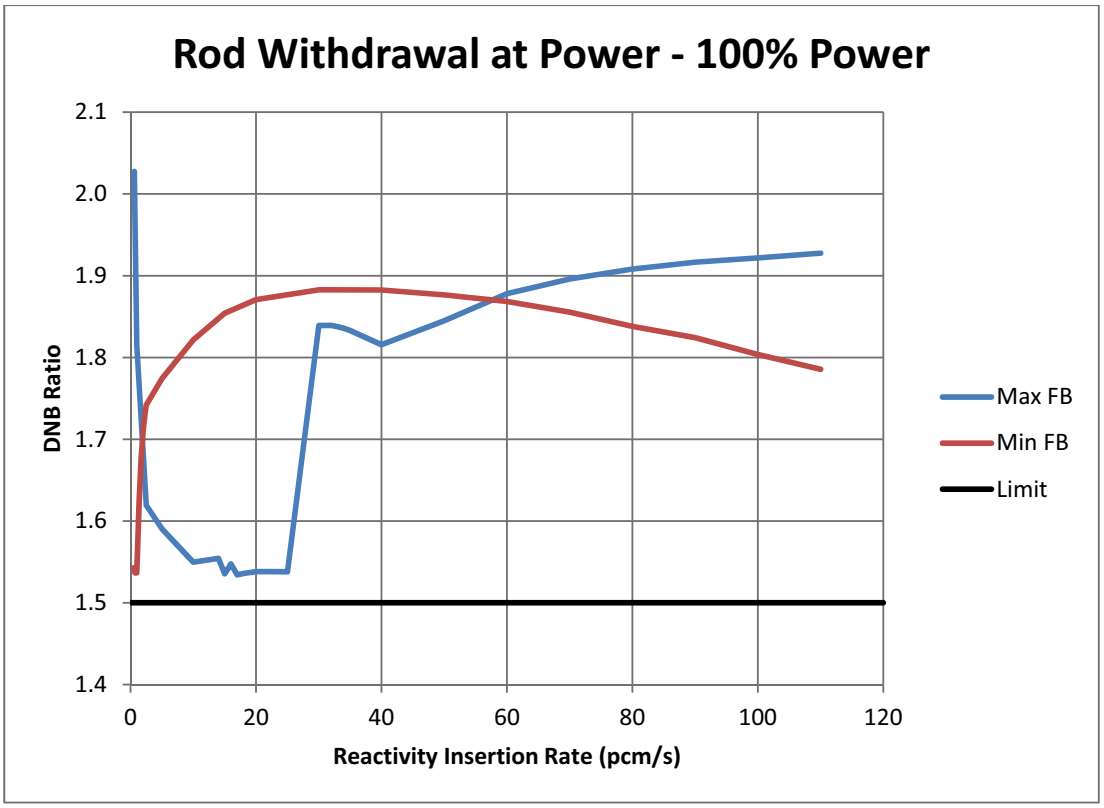

Figure 15.4.2-19
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 100-percent Power

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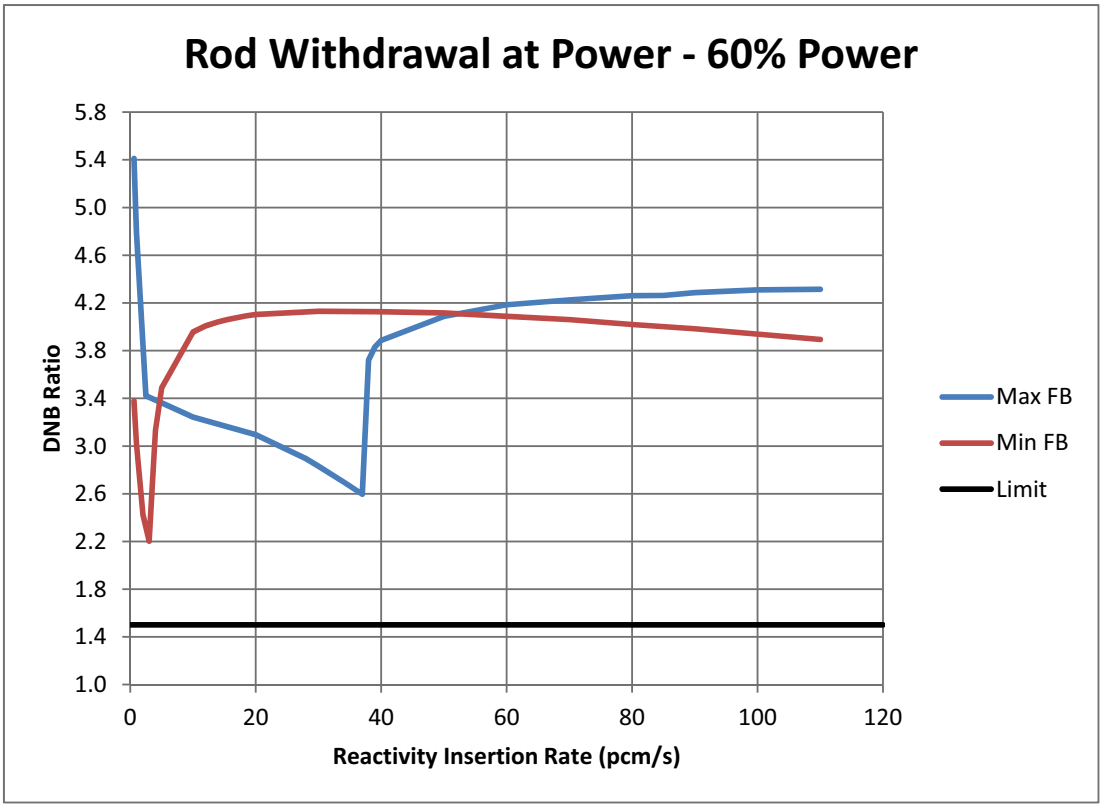

Figure 15.4.2-20
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 60-percent Power

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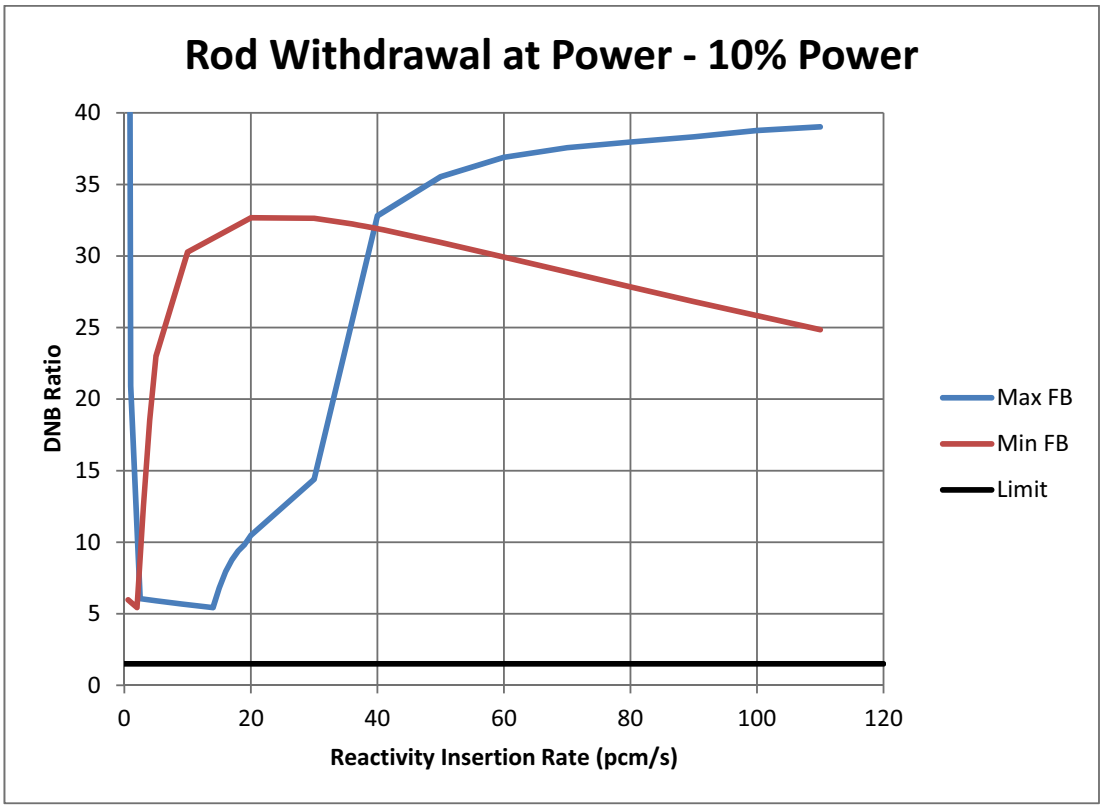

Figure 15.4.2-21
Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 10-percent Power

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Figures 15.4.2-22–15.4.2-23 Not Used

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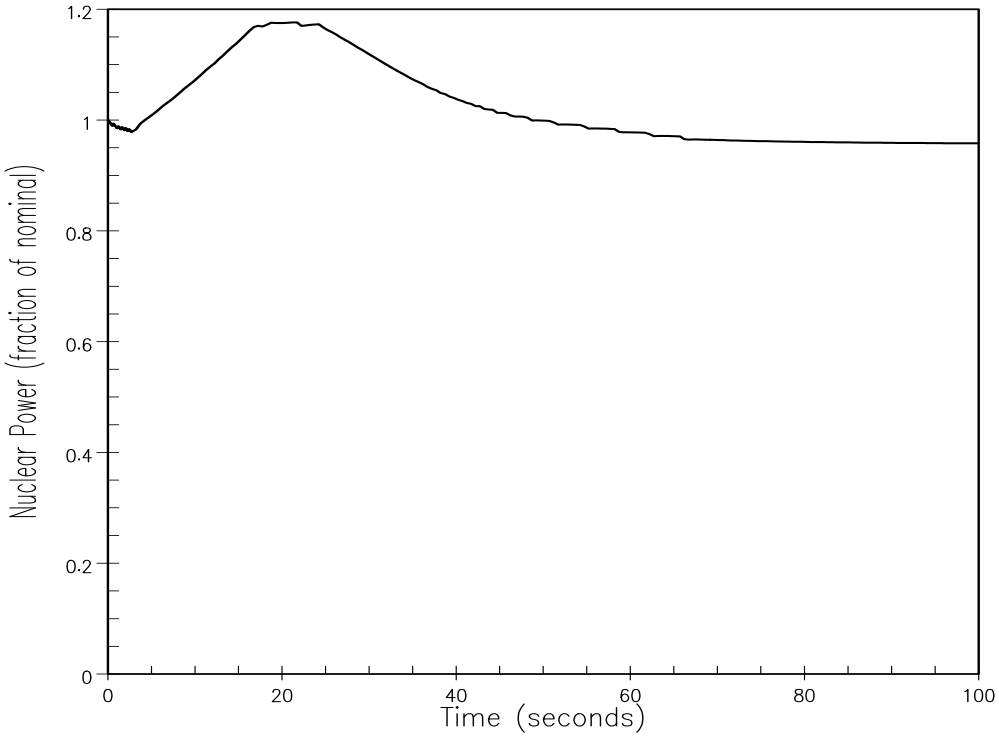

Figure 15.4.3-1
Nuclear Power Transient for Dropped RCCA

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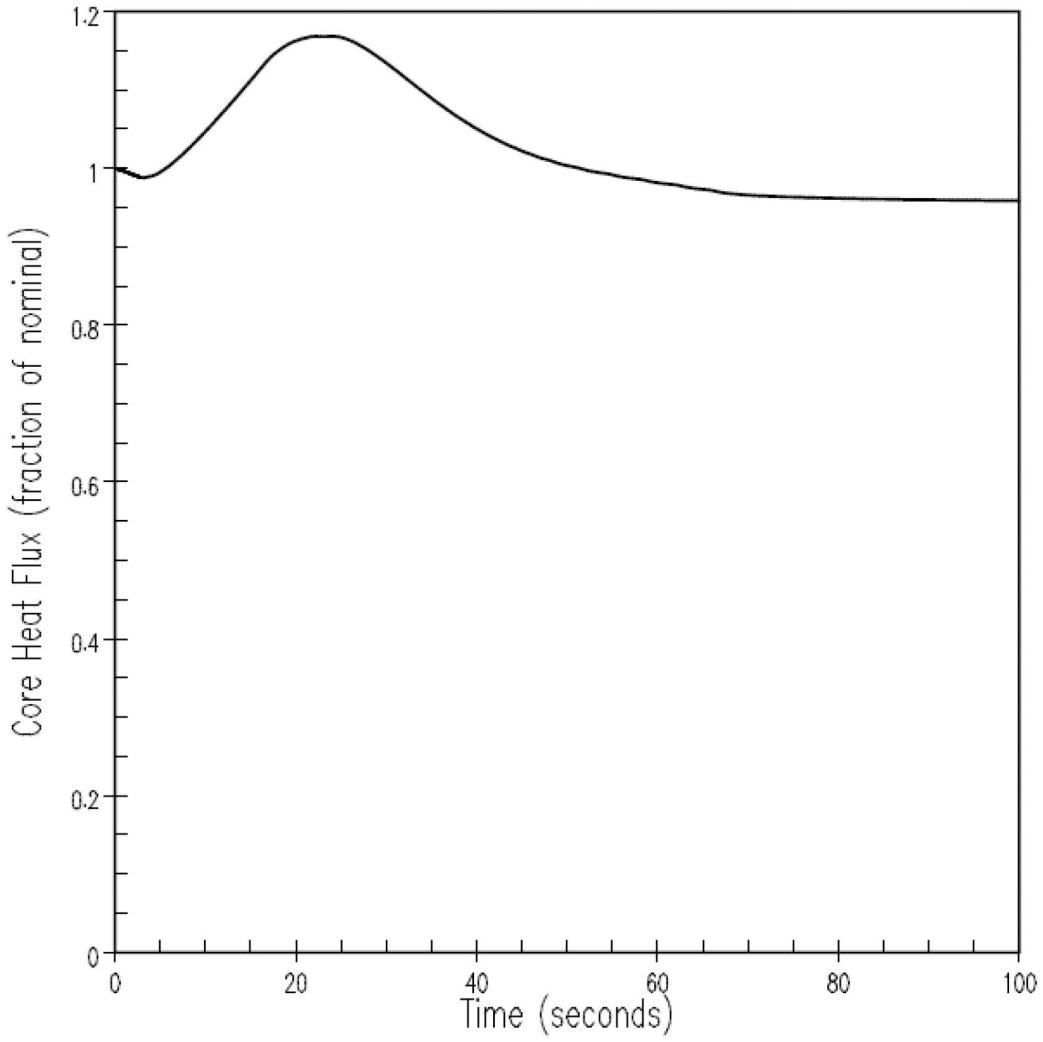

Figure 15.4.3-2
Core Heat Flux Transient for Dropped RCCA

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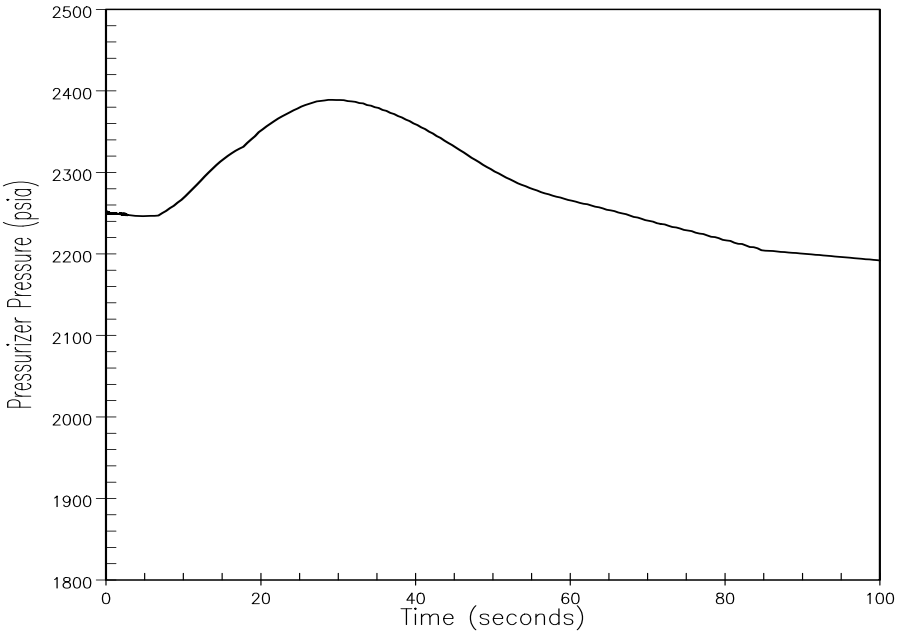

Figure 15.4.3-3
Pressurizer Pressure Transient for Dropped RCCA

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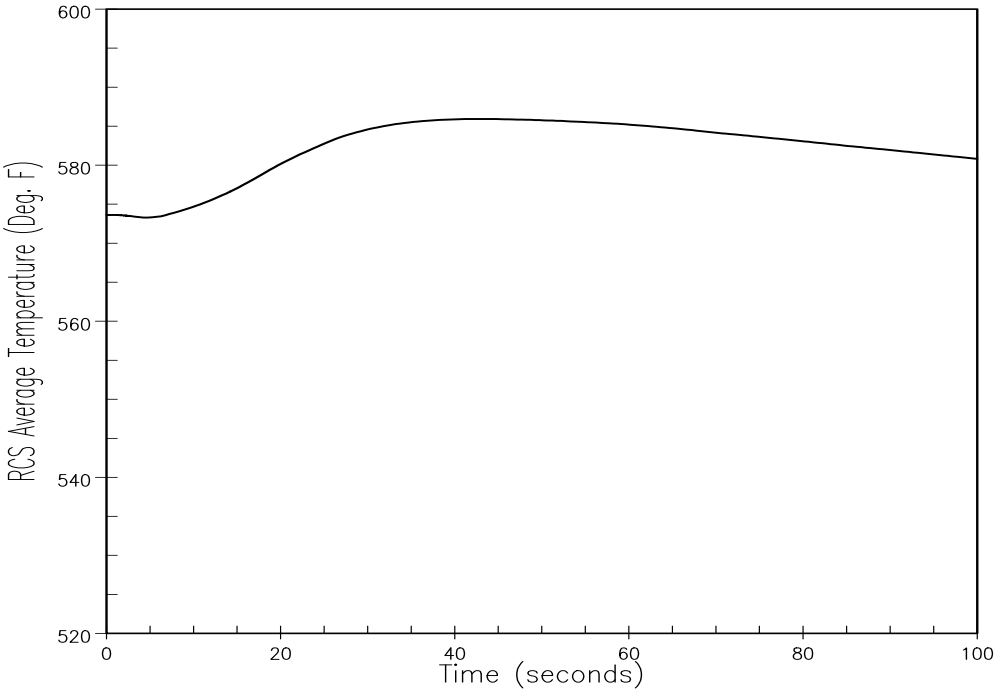

Figure 15.4.3-4
RCS Average Temperature Transient for Dropped RCCA

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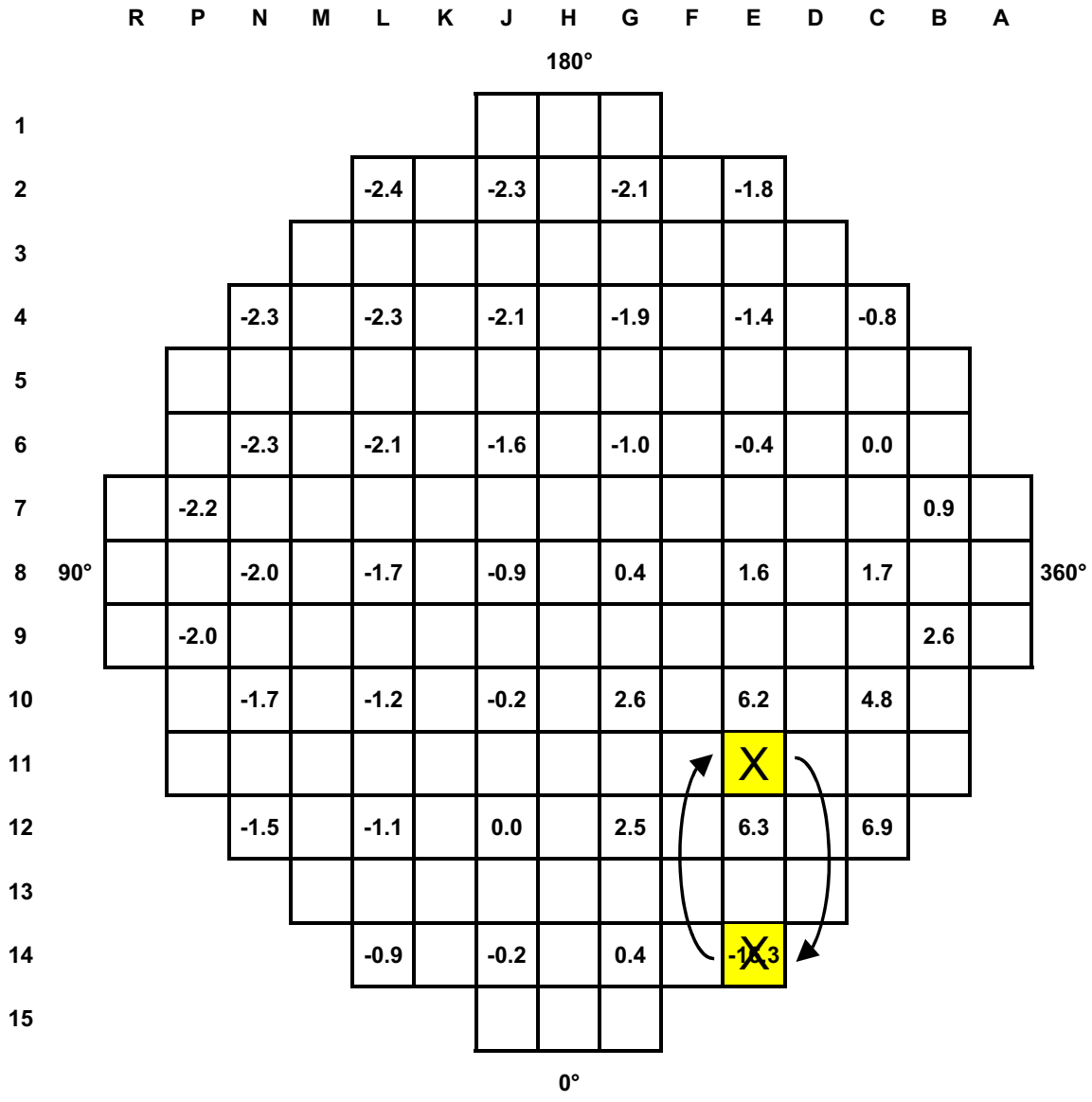

Figure 15.4.7-1
Representative Percent Change in Local Assembly Average Power for Interchange Between Region 1 and Region 3 Assembly

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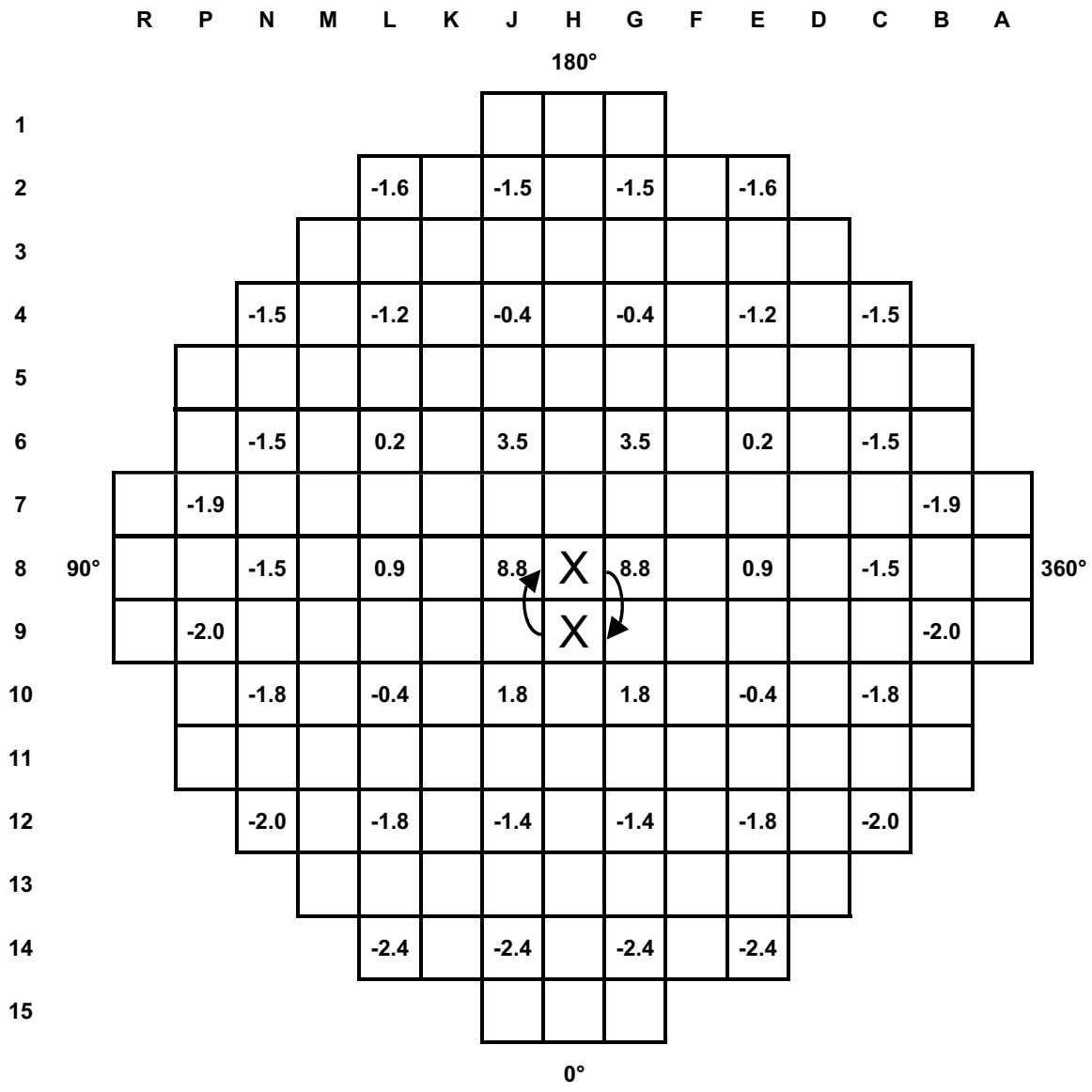

Figure 15.4.7-2
Representative Percent Change in Local Assembly Average Power for Interchange Between Region 1 and Region 2 Assembly with the BP Rods Transferred to Region 1 Assembly

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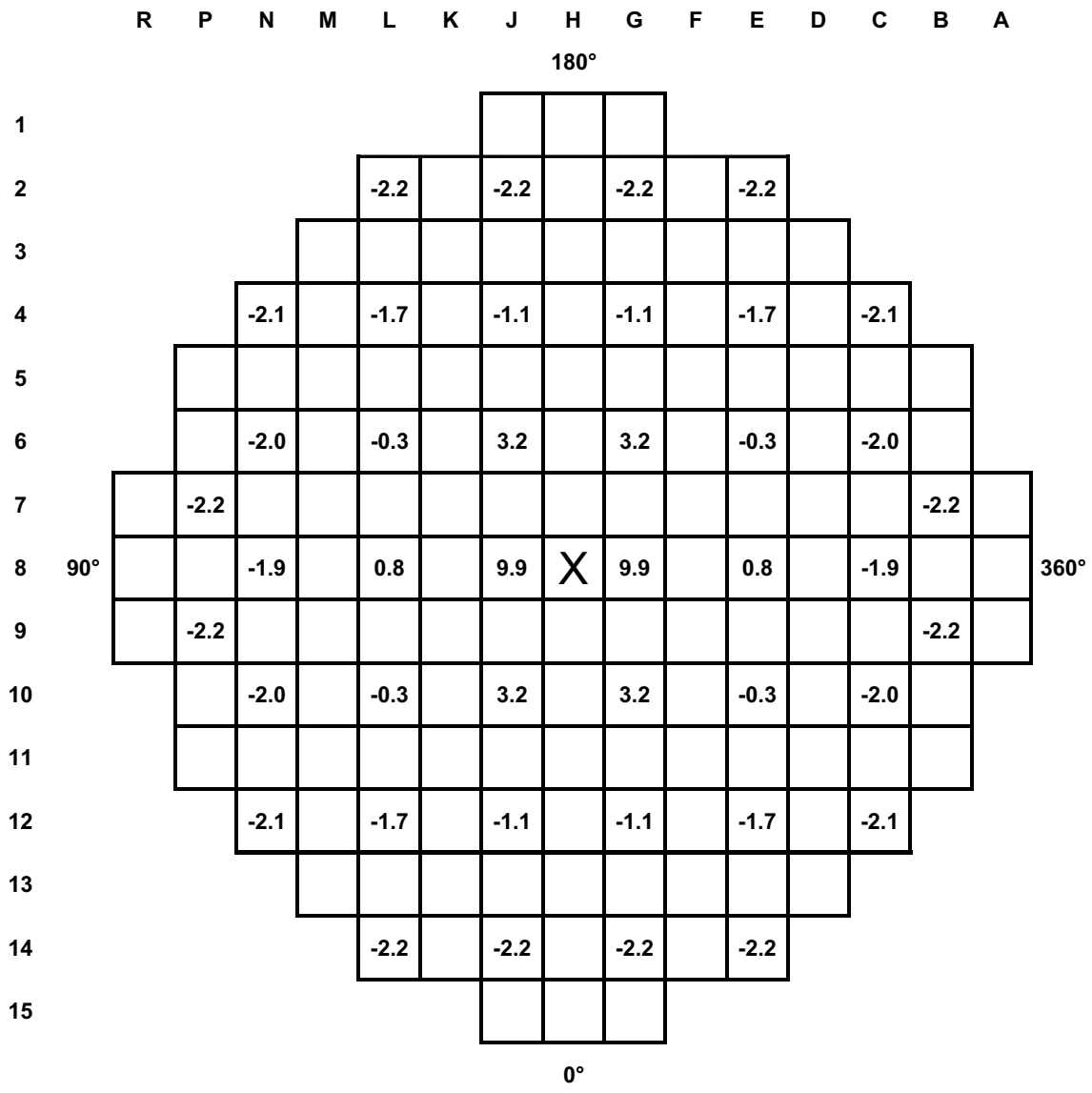

Figure 15.4.7-3
Representative Percent Change in Local Assembly Average Power for Enrichment Error (Region 2 Assembly Loaded into Core Central Position)

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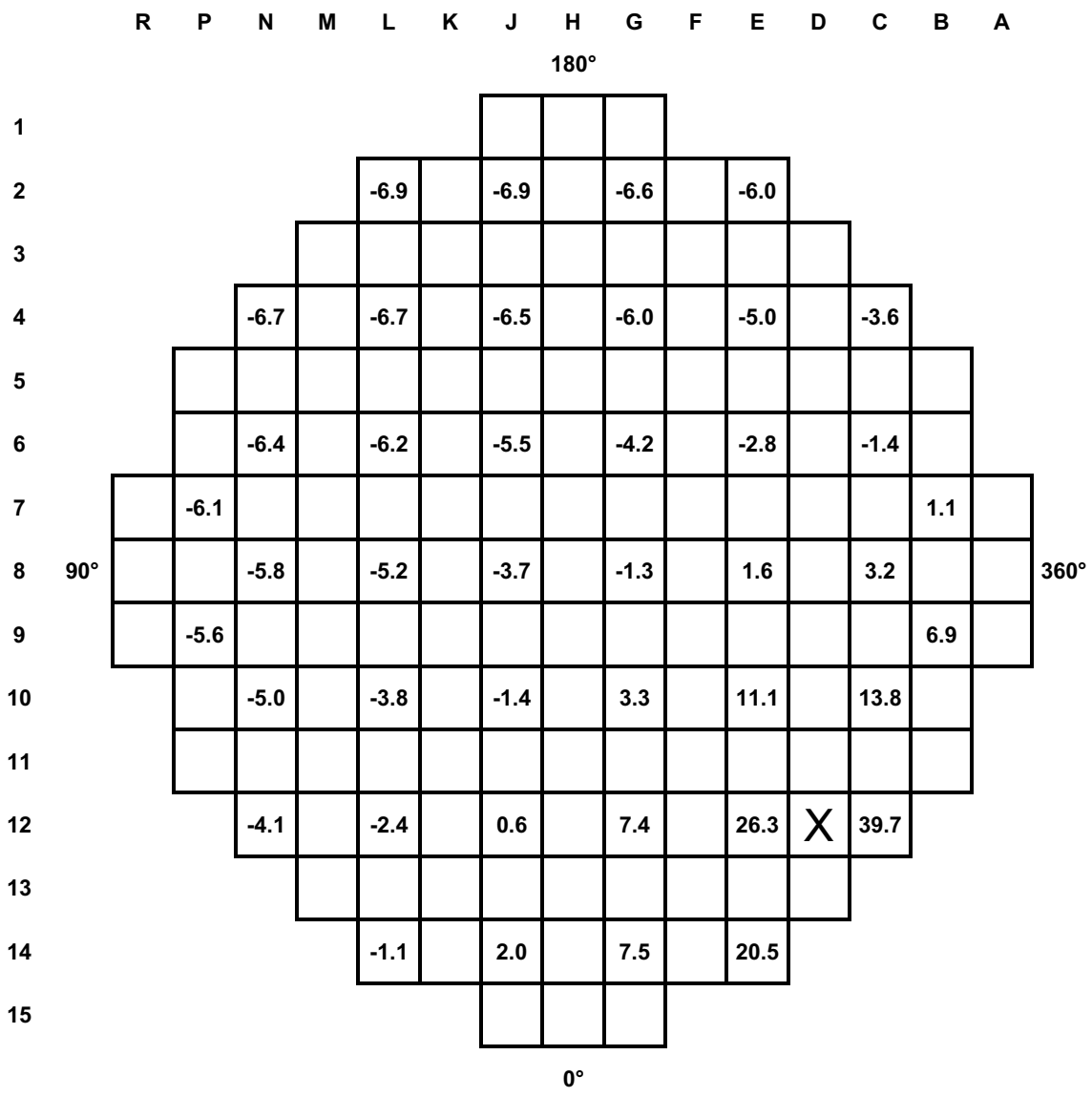

Figure 15.4.7-4
Representative Percent Change in Local Assembly Average Power for Loading Region 2 Assembly into Region 1 Position Near Core Periphery

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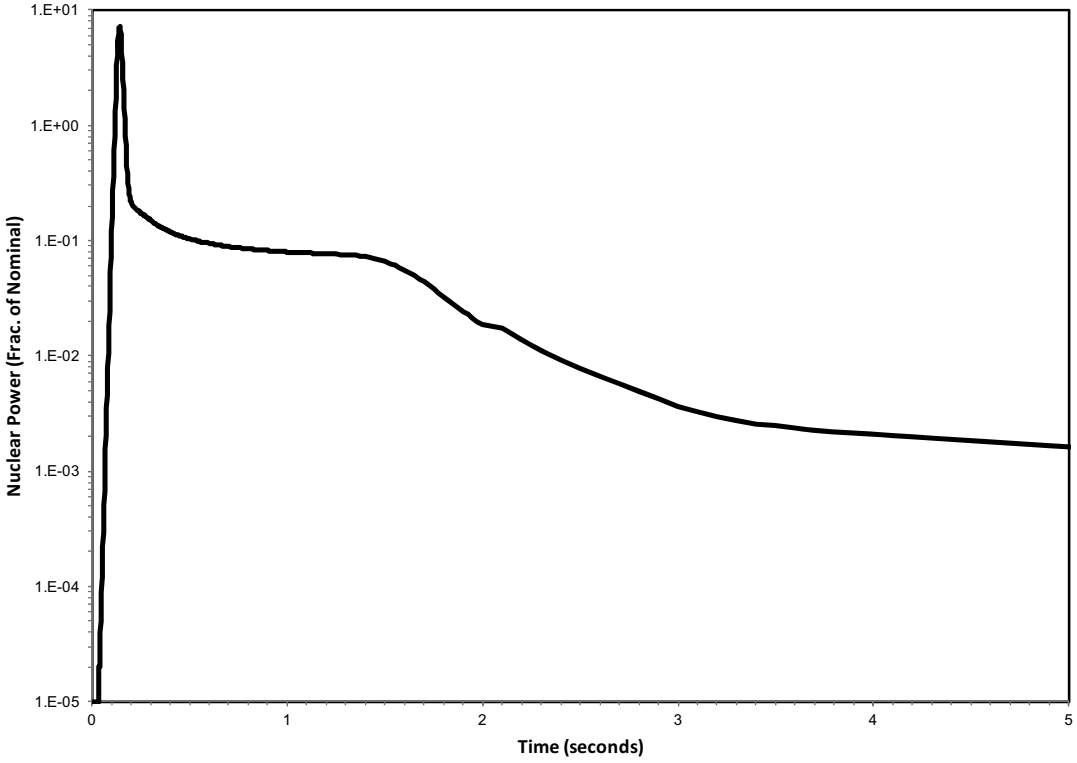

Figure 15.4.8-1
Nuclear Power Transient Versus Time
for the PCMI Rod Ejection Accident

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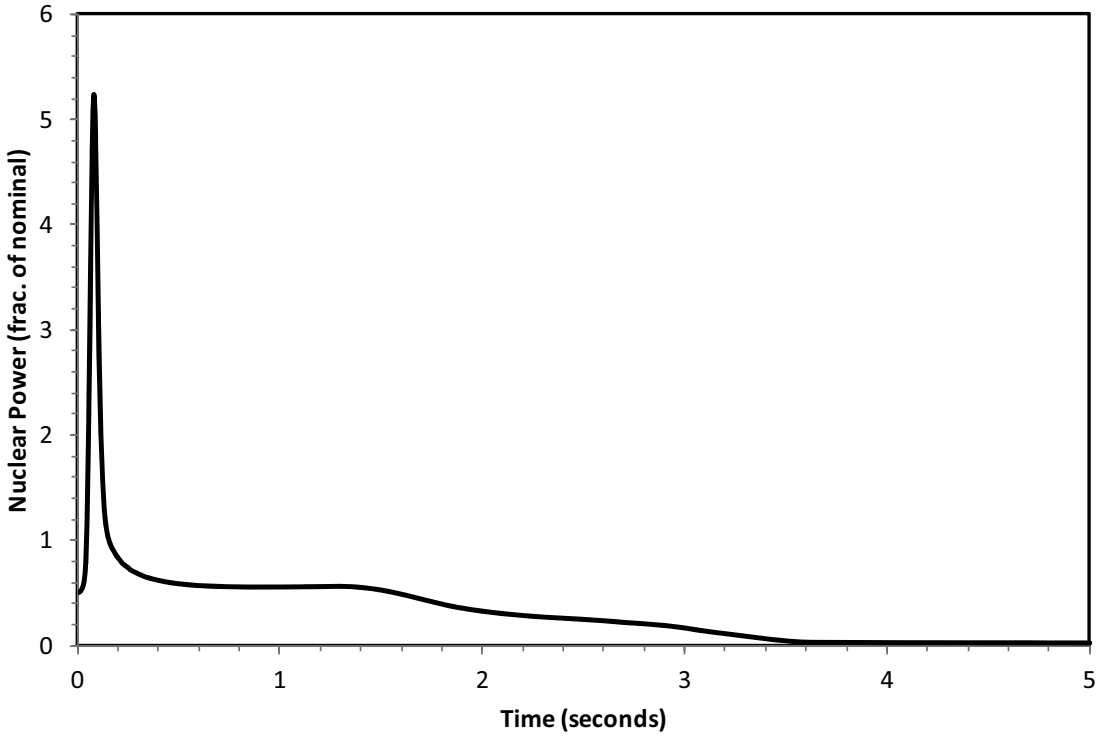

Figure 15.4.8-2
Nuclear Power Transient Versus Time
for the High Clad Temperature Rod Ejection Accident

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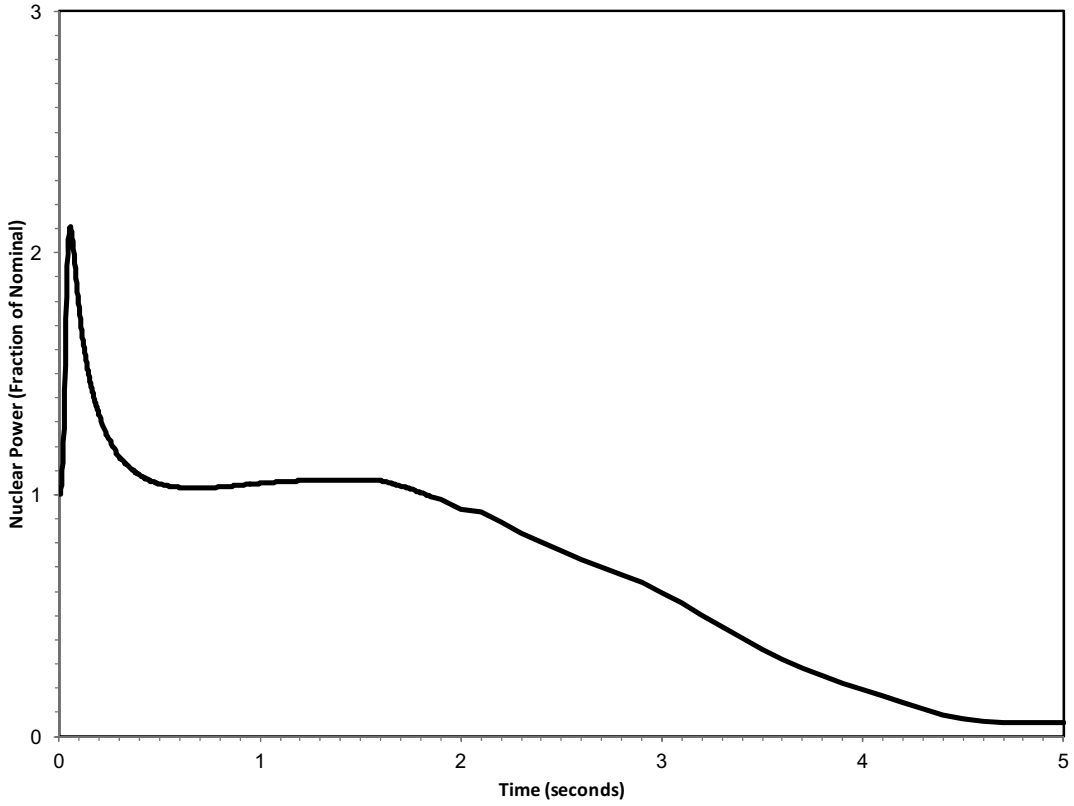

Figure 15.4.8-3
Nuclear Power Transient Versus Time
for the Peak Enthalpy and Fuel Centerline Temperature Rod Ejection Accident

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Figure 15.4.8-4 Not Used

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15.5 Increase in Reactor Coolant Inventory

This section presents a discussion and analysis of the following events:

- Inadvertent operation of the core makeup tanks during power operation
- Chemical and volume control system malfunction that increases reactor coolant inventory

These Condition II events cause an increase in reactor coolant inventory.

15.5.1 Inadvertent Operation of the Core Makeup Tanks During Power Operation

15.5.1.1 Identification of the Causes and Accident Description

Spurious core makeup tank operation at power could be caused by an operator error, a false electrical actuation signal, or a valve malfunction. A spurious signal may originate from any of the safeguards (“S”) actuation channels as described in Section 7.3. The AP1000 protection logic is such that a single failure cannot actuate both core makeup tanks without also actuating the passive residual heat removal (PRHR) heat exchanger. A scenario such as this is the spurious “S” signal event. However, if one core makeup tank is inadvertently actuated by a single failure, the event may progress with the plant at power until a reactor trip is reached. For the plant under automatic rod control, a reactor trip on high-3 pressurizer water level reactor trip is expected to occur followed by the PRHR actuation and eventually by an “S” signal, which would then actuate the second core makeup tank. When a consequential loss of offsite power is assumed, this event is more limiting than the spurious “S” signal event.

The inadvertent opening of the core makeup tank discharge valves, due to operator error or valve failure, results in significant core makeup tank injection flow leading to a boration similar to that resulting from a chemical and volume control system malfunction event. If the automatic rod control system is operable, it will begin to withdraw rods from the core to counteract the reactivity effects of the boration. As a result, the core makeup tank will continue injection and slowly increase the pressurizer level until the high-2 pressurizer level setpoint is reached and continues until the high-3 pressurizer level reactor trip setpoint is reached. In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. The core makeup tank injection will increase as the steam generator outlet temperature increases resulting in a lower density in the CMT balance line. This event will then proceed similar to a spurious “S” signal or chemical and volume control system malfunction event. However, this event is more limiting primarily due to the higher pressurizer level at the time of reactor trip and to the significant heat up of the injected fluid during the pre-trip phase of the accident. Thus, the inadvertent core makeup tank actuation event with a consequential loss of offsite power is analyzed here.

Upon receipt of the high-3 pressurizer level reactor trip signal, the reactor is tripped; then the turbine is tripped after a 5-second delay and a consequential loss of offsite power is assumed 3 seconds after turbine trip. The basis for the 3-second delay is described in Subsection 15.0.14. The high-3 pressurizer level signal also actuates the PRHR heat exchanger and blocks the pressurizer heaters, but a 15-second timer delay prevents unnecessary actuation of the PRHR heat exchanger if offsite power is maintained. Because of the reactor trip the pressurizer level drops and the high-3 signal is cleared before the timer delay is completed, and thus the PRHR heat exchanger is not actuated.

Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. However, due to the assumed loss of offsite power, the reactor coolant cold leg temperature in the loop without PRHR increases and the core makeup tank begins injecting cold water into the reactor coolant system at a much higher rate. The primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer

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water volume increases because of the heatup of the cold injected fluid by the decay heat. The high-3 pressurizer level setpoint is once again reached, and after the 15-second timer delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters.

The PRHR heat exchanger extracts heat from the reactor coolant system leading to an “S” signal on a Low-2 T_{cold} signal. The PRHR heat exchanger may inject asymmetrically into the steam generator outlet plenum such that a higher percentage of the PRHR flow goes to one of the two cold legs coming from the steam generator on the PRHR loop. To account for this, the analysis conservatively assumes that the Low-2 T_{cold} setpoint is reached coincident with PRHR heat exchanger actuation. This actuates the second core makeup tank sooner in the transient, which is more limiting with respect to filling the pressurizer.

Both core makeup tanks inject into the reactor coolant system and the pressurizer level continues to increase until the operators act to end the pressurizer level increase transient. The operators are assumed to be alerted to a potential filling event on the high-2 pressurizer level signal, which occurs well before the reactor trip on the first of two high-3 pressurizer level signals. The operator action assumed in the analysis is to open the reactor vessel head vent following receipt of the second high-3 pressurizer level signal; this action occurs at least 30 minutes (45 minutes is conservatively assumed in the analysis) after the operator has been alerted by the high-2 pressurizer level signal. When the head vent is opened, the pressurizer level increase slows and ultimately the level begins to decrease.

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

15.5.1.2 Analysis of Effects and Consequences

The plant response to an inadvertent core makeup tank actuation is analyzed by using a modified version of the computer program LOFTRAN (Reference 1) described in Subsection 15.0.11.2. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, PRHR heat exchanger, and core makeup tanks. The program computes pertinent plant variables, including temperatures, pressures, and power level.

Reactor power and average temperature drop immediately following the trip, and the operating conditions never approach the core limits. The analysis demonstrates that no reactor coolant system overpressurization occurs.

Core makeup tank and PRHR system performance is conservatively simulated. Core makeup tank enthalpies have been maximized. This is conservative because it minimizes the cooling provided by the core makeup tanks as flow recirculates and thereby increases the peak pressurizer water volume during the transient. The core makeup tank injection and balance lines pressure drops are minimized. This maximizes the core makeup tank flow injected in the primary system. During this event, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tanks is offset by an equal volume of recirculation flow that enters the core makeup tanks via the balance lines. PRHR heat transfer capability has been minimized.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

- Initial operating conditions

  The initial reactor power is assumed to be 101 percent of nominal. The initial pressurizer pressure is assumed to be 50 psi below nominal. The initial reactor coolant system average temperature is assumed to be 8°F below nominal.

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- Control systems

  The pressurizer spray, the pressurizer heaters, and the automatic rod control system are conservatively assumed to operate. The pressurizer heaters are automatically blocked on a high-3 pressurizer level signal following a 15-second timer delay, but add heat to the system during the initial part of the transient. Other control systems are conservatively not assumed to function during the transient.

- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a low (absolute value) Doppler power coefficient, and a minimum boron worth are assumed. With these minimum feedback parameters and the operability of the pressurizer spray system and automatic rod control system assumed, the reactivity effects of the boron injection from the core makeup tanks is counteracted. As a result, the High-3 pressurizer water level signal is the first reactor trip signal generated during the transient.

- Boron injection

  The transient is initiated by an inadvertent opening of the discharge valves of one of the two core makeup tanks, resulting in the injection of 3400 ppm borated water.

- Protection and safety monitoring system actuations

  The operators are assumed to be alerted to the pressurizer level increase transient by the high-2 pressurizer level signal. Reactor trip is initiated by the first of two high-3 pressurizer level signals. The second high-3 pressurizer level signal triggers the operators to open the reactor vessel head vent; this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal.

  The core decay heat is removed by the PRHR heat exchanger. The worst single failure is assumed to occur in the outlet line of the PRHR heat exchanger. One of the two parallel isolation valves is assumed to fail to open.

Plant systems and equipment available to mitigate the effect of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.5.1.3 Results

Figures 15.5.1-1 through 15.5.1-8 show the transient response to the inadvertent operation of one of the two core makeup tanks during power operation. The inadvertent opening of the core makeup tank discharge valves occurs at 10 seconds. As the core makeup tank adds inventory to the primary system, the pressurizer level begins to increase until the high-2 pressurizer level setpoint is reached (783.2 seconds) and continues until the high-3 pressurizer level reactor trip setpoint is reached at about 2,402.0 seconds. After a 2-second delay, the neutron flux starts decreasing due to the reactor trip, which is followed by turbine trip after a 5-second turbine trip delay. Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. Due to the assumed loss of offsite power, the reactor coolant pumps trip at about 2,412.0 seconds. The cold leg temperature increases and the core makeup tank starts injecting cold water into the reactor coolant system at a higher rate due to the increased driving head resulting from the density decrease in the balance line and due to the reduced pressure drop between the cold leg and the injection line connection on the reactor vessel following the trip of the reactor coolant pumps. The post-trip primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer water volume starts to increase because of the heatup of

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the cold injected fluid by the decay heat. The high-3 pressurizer level setpoint is once again reached at 2,554.8 seconds, and after a 15-second timer delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters. Following a conservative 17-second delay, the valves are assumed to open to actuate the PRHR heat exchanger at 2,586.8 seconds.

If the PRHR heat exchanger coolant asymmetrically injects into the steam generator outlet plenum, then one cold leg may reach the Low-2 T_{cold} "S" setpoint more quickly than if the flow were split evenly. To conservatively account for this effect, the Low-2 T_{cold} "S" signal is modeled to actuate simultaneously with the actuation of the PRHR heat exchanger (2,586.8 seconds). The Low-2 T_{cold} "S" signal activates the second core makeup tank, which then begins injecting additional mass into the reactor coolant system. An earlier actuation of the second core makeup tank results in a more limiting pressurizer fill transient.

As the second core makeup tank begins injecting, the pressurizer level continues to increase. The operators are assumed to be alerted by the high-2 pressurizer level signal (783.2 seconds) that a pressurizer level increase transient is underway, and it is assumed that the operators are ready to take corrective action no sooner than 30 minutes later. In this analysis, since pressurizer level continues to increase, the high-3 pressurizer level reactor trip setpoint is reached within this time. The operator action assumed in this case is to open the reactor vessel head vent to preclude overfill following receipt of the second high-3 pressurizer level signal (3,483.2 seconds); this action is at least 30 minutes (45 minutes is conservatively assumed in the analysis) after the operator has been alerted by the high-2 pressurizer level signal.

The safety related reactor vessel head vent is opened by the operators and the pressurizer water level increase slows and eventually the level begins to decrease. This demonstrates that the capacity of the reactor vessel head vent is sufficient to preclude pressurizer overfill as a result of an inadvertent actuation of a core makeup tank.

During the event, the departure from nucleate boiling ratio (DNBR) would not drop significantly below the initial value due to the addition of highly borated water from the core makeup tanks to the reactor coolant system. At the time of reactor trip core power and heat flux drop rapidly, which would cause the DNBR to increase well above the design limit value defined in Section 4.4.

The calculated sequence of events is shown in Table 15.5-1.

As noted above, the limiting case presented here models explicit operator action 45 minutes after receipt of the high-2 pressurizer level signal. For pressurizer level increase events, the operator would take action to reduce the increase in coolant inventory. As the pressurizer water level would increase above the High-2 pressurizer water level that normally isolates chemical and volume control system makeup, the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed into service, the operator could use the safety related reactor vessel head vent valves to reduce the increase in coolant inventory (this is explicitly modeled in the case presented here). For these events, there is sufficient time for the operator to mitigate the consequences by following the procedures outlined in the Emergency Response Guidelines AFR-I.1.

15.5.1.4 Conclusions

The results of this analysis show that inadvertent operation of the core makeup tanks during power operation does not adversely affect the core, the reactor coolant system, or the steam system. Water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

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15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

15.5.2.1 Identification of Causes and Accident Description

An increase of reactor coolant inventory from the addition of cold unborated water to the reactor coolant system, is analyzed in Subsection 15.4.6.

In this Subsection 15.5.2, the increase of reactor coolant system inventory due to the addition of borated water is analyzed.

The increase of reactor coolant system coolant inventory may be due to the spurious operation of one or both of the chemical and volume control system pumps or by the closure of the letdown path. If the chemical and volume control system is injecting highly borated water into the reactor coolant system, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power and subsequent coolant shrinkage. The load decreases due to the effect of reduced steam pressure after the turbine control valve fully opens.

At high chemical and volume control system boron concentration, low reactivity feedback conditions, and reactor in manual rod control, an “S” signal will be generated by either the Low-2 T_{cold} or Low-2 steam line pressure setpoints before the chemical and volume control system can inject a significant amount of water into the reactor coolant system. In this case, the chemical and volume control system malfunction event proceeds similar to, and is only slightly more limiting than, a spurious “S” signal event. If automatic rod control is modeled and the pressurizer spray functions properly to prevent a High-2 pressure reactor trip signal, no “S” signals are generated and this specific event is terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

Under typical operating conditions for the AP1000 plant, the boron concentration of the injected chemical and volume control system water is equal to that of the reactor coolant system. If the chemical and volume control system is functioning in this manner and the pressurizer spray system functions properly to prevent a High-2 pressure reactor trip signal, no “S” signals are generated and this specific event is also terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

While these scenarios are the most probable outcomes of a chemical and volume control system malfunction, several combinations of boron concentration, feedback conditions, and plant system interactions have been identified that could result in more limiting scenarios with respect to pressurizer overfill. The key factors that make this event more limiting than a spurious “S” signal event are that the reactor coolant system is at a lower average temperature, higher pressure, and a higher pressurizer level at the time an “S” signal is generated. These factors produce a greater volume of higher density water and, thus, a larger reactor coolant system mass at the time of the “S” signal. In addition, at lower reactor coolant system average temperature, the PRHR is less effective in removing decay heat, which results in greater expansion of the cold water injected by the core makeup tanks.

The limiting analysis scenario minimizes reactor coolant system average temperature, maximizes reactor coolant system mass, and maximizes pressurizer water volume at the time of an “S” signal. This scenario is as follows:

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- Both of the chemical and volume control system pumps spuriously begin delivering flow at a boron concentration slightly higher than that of the reactor coolant system. (Assuming that a chemical and volume control system malfunction results in both chemical and volume control system pumps delivering flow is a conservative assumption. One chemical and volume control system pump is automatically controlled and one is manually controlled.)

- The non-safety-related pressurizer spray is assumed to be available, so that a High-2 pressurizer pressure reactor trip is prevented.

Due to the boron addition in the core, the plant cools down until an “S” signal is generated on Low-2 T_{cold}. On the “S” signal, the reactor is tripped, the core makeup tank discharge valves are opened, the reactor coolant pumps are tripped, the pressurizer heaters are blocked (on core makeup tank actuation), and the main feedwater lines, steam lines, and chemical and volume control system are isolated. After a conservative 17-second delay, the PRHR heat exchanger is actuated.

Normally, the reactor coolant pumps would be tripped 15 seconds after the receipt of the “S” signal. However, to meet the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. Following reactor trip and a 5-second timer delay, the turbine is tripped; 3 seconds after the turbine trip a consequential loss of offsite power is assumed. The basis for the 3-second delay is described in Subsection 15.0.14. As a result, the reactor coolant pumps are conservatively assumed to trip about 10 seconds before they would otherwise trip due to the “S” signal.

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

15.5.2.2 Analysis of Effects and Consequences

The malfunction of the chemical and volume control system is analyzed by using a modified version of the computer program LOFTRAN (Reference 1) described in Subsection 15.0.11.2. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, PRHR heat exchanger, and core makeup tanks. The program computes pertinent plant variables including temperatures, pressures, and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. The analysis demonstrates that no reactor coolant system overpressurization or loss of reactor coolant system water occurs.

The assumptions are as follows:

- Initial operating conditions

  The initial reactor power is assumed to be 101 percent of nominal. The initial pressurizer pressure is assumed to be 50 psi above nominal. The initial reactor coolant system average temperature is assumed to be 8°F above nominal.

- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a low (absolute value) Doppler power coefficient, and a minimum boron worth are assumed. For a different set of reactivity feedback parameters, a different chemical and volume control system boron concentration can result in an identical transient.

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- Reactor control

  Rod control is not modeled.

- Pressurizer heaters

  The pressurizer heaters are automatically blocked on the core makeup tank actuation resulting from the “S” signal on Low-2 T_{cold}, but add heat to the system during the initial part of the transient. Thus, the pressurizer heaters are assumed to be operable during this event.

- Pressurizer spray

  The spray system controls the pressurizer pressure so that a High-2 pressurizer pressure reactor trip is prevented.

- Boron injection

  After 10 seconds at steady state, the chemical and volume control system pumps start injecting borated water, which is slightly above the reactor coolant system boron concentration. Upon receipt of an “S” signal, the core makeup tanks begin injecting 3400 ppm borated water. The chemical and volume control system pumps are isolated on high-2 pressurizer level. In this analysis the boron concentration of the chemical and volume control system is determined by iteration such that the high-2 pressurizer level and the Low-2 T_{cold} “S” setpoint are reached at the same time. This begins core makeup tank injection when the chemical and volume control system pumps are isolated, which is conservative with respect to filling the pressurizer.

- Turbine load

  The turbine load is assumed constant until the turbine D-EHC drives the control valve wide open, after which the turbine load drops as steam pressure drops.

- Protection and safety monitoring system actuations

  If the automatic rod control system is modeled and the pressurizer spray system functions properly, no reactor trip signal is expected to occur. Instead, the event is terminated by automatic isolation of the chemical and volume control system on the safety grade high-2 pressurizer level setpoint. If the automatic rod control system is not active and the pressurizer spray system is assumed to be available, reactor trip may be initiated on either Low-2 T_{cold} “S” or a Low-2 steam line pressure “S” signal.

  The core decay heat is removed by the PRHR heat exchanger. The worst single failure is assumed to occur in the outlet line of the PRHR heat exchanger. One of the two parallel isolation valves is assumed to fail to open.

  Plant systems and equipment available to mitigate the effect of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.5.2.3 Results

Figures 15.5.2-1 through 15.5.2-9 show the transient response to a chemical and volume control system malfunction that results in an increase of reactor coolant system inventory.

As the chemical and volume control system injection flow increases reactor coolant system inventory, pressurizer water volume begins increasing while the primary system is cooling down. At

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2,009.6 seconds, the Low-2 T_{cold} setpoint is reached, the reactor trips on the resulting “S” signal, the control rods start moving into the core, the core makeup tank discharge valves are opened, and the pressurizer heaters are blocked. At approximately the same time, the high-2 pressurizer level setpoint is reached and after a conservative delay, the chemical and volume control system injection is isolated.

The turbine is tripped as a result of the reactor trip following a 5-second turbine trip timer delay. After a 3-second delay following a turbine trip, a consequential loss of offsite power is assumed and the reactor coolant pumps trip. The basis for the 3-second delay is described in Subsection 15.0.14. Soon after reactor trip, the main feedwater lines, steam lines, and chemical and volume control system are isolated on the “S” signal. After a conservative 17-second delay, the PRHR heat exchanger is actuated. The core makeup tanks work in recirculation mode, meaning they are always filled with water because cold borated water injected through the injection lines is replaced by hot water coming from the cold leg balance lines.

The operation of the PRHR heat exchanger and the core makeup tanks cools down the plant. Due to the expansion of the core makeup tank water, the pressurizer level continues to increase. The operators are assumed to be alerted by the high-2 pressurizer level signal (2,008.3 seconds) that a pressurizer level increase transient is underway, and it is assumed that the operators are ready to take corrective action at least 30 minutes later. The specific operator action as analyzed in this case is to open the reactor vessel head vent to preclude pressurizer overfill following the high-3 pressurizer level signal at least 30 minutes after the high-2 pressurizer level signal is reached (3,803.3 seconds).

The safety related reactor vessel head vent is opened by the operators and the pressurizer water level increase slows and eventually the level begins to decrease. This demonstrates that the capacity of the reactor vessel head vent is sufficient to preclude pressurizer overfill as a result of a chemical and volume control system malfunction that causes an increase in reactor coolant inventory.

During the event, the DNBR would not drop significantly below the initial value since both the chemical and volume control system and the core makeup tanks add borated water to the reactor coolant system. At the time of reactor trip, core power and heat flux drop rapidly, which would cause the DNBR to increase well above the design limit value defined in Section 4.4.

The calculated sequence of events is shown in Table 15.5-1.

The limiting case presented here models operator action to open the reactor vessel head vent following receipt of the high-3 pressurizer level signal; this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal. For pressurizer level increase events, the operator could take other actions to reduce the increase in coolant inventory. As the pressurizer water level would increase above the High-2 pressurizer water level that normally isolates chemical and volume control system makeup, the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed into service, the operator would use the safety-related reactor vessel head vent valves to reduce the increase in coolant inventory. For these events, following operations procedures, there is sufficient time for the operator to mitigate the consequences of this event.

15.5.2.4 Conclusions

The results of this analysis show that a chemical and volume control system malfunction does not adversely affect the core, the reactor coolant system, or the steam system. Water is not relieved from the pressurizer safety valves. DNBR remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

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If the automatic rod control system and the pressurizer spray systems are assumed to function, no reactor trip signal is expected to occur. Instead, the event would be terminated by automatic isolation of the chemical and volume control system on the safety grade High-2 pressurizer level setpoint. If manual rod control is assumed and the pressurizer spray system is assumed to be unavailable, reactor trip may be initiated on either a High-2 pressurizer pressure, Low-2 T_{cold} "S", or a Low-2 steamline pressure "S" signal.

15.5.3 Boiling Water Reactor Transients

This subsection is not applicable to the AP1000 design.

15.5.4 Combined License Information

This subsection contained no requirement for additional information.

15.5.5 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

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Table 15.5-1 (Sheet 1 of 2)
Time Sequence of Events For Incidents Which Result in an Increase in Reactor Coolant Inventory

Accident	Event	Time (seconds)
Inadvertent operation of the core makeup tanks during power operation	Core makeup tank discharge valves open	10.0
	High-2 pressurizer level setpoint reached	783.2
	High-3 pressurizer level setpoint reached	2402.0
	Rod motion begins	2404.0
	Loss of offsite power	2412.0
	Reactor coolant pumps trip	2412.0
	High-3 pressurizer level setpoint reached	2554.8
	Pressurizer heaters blocked on high-3 level	2569.8
	PRHR heat exchanger actuated	2586.8
	Low-2 T_{cold} “S” setpoint is reached	2586.8
	Second CMT starts recirculating	2586.8
	Main steam and feed lines are isolated	2598.8
	Operators open the reactor vessel head vent after the high-3 pressurizer level signal is reached (at least 30 minutes after high-2 pressurizer level setpoint is reached; 45 min as analyzed)	3483.2
	Peak pressurizer water volume occurs	5312.5

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Table 15.5-1 (Sheet 2 of 2)
Time Sequence of Events For Incidents Which Result in an Increase in Reactor Coolant Inventory

Accident	Event	Time (seconds)
Chemical and volume control system malfunction that increases reactor coolant inventory	Chemical and volume control system charging pumps start	10.0
	High-2 pressurizer level signal is reached	2008.3
	Low-2 T_{cold} "S" signal is reached (approximately coincident with high-2 pressurizer level)	2009.6
	Core makeup tank discharge valves open and pressurizer heaters are blocked	2009.7
	Rod motion begins	2011.6
	Loss of offsite power	2019.6
	Reactor coolant pumps trip	2019.6
	Main steam and feed lines are isolated	2021.6
	PRHR heat exchanger actuated	2026.7
	Chemical and volume control system charging pumps are isolated	2044.3
	Operators open the reactor vessel head vent after the high-3 pressurizer level signal is reached (at least 30 minutes after high-2 pressurizer level setpoint is reached; 30 min as analyzed)	3803.3
	Peak pressurizer water volume occurs	5413.0

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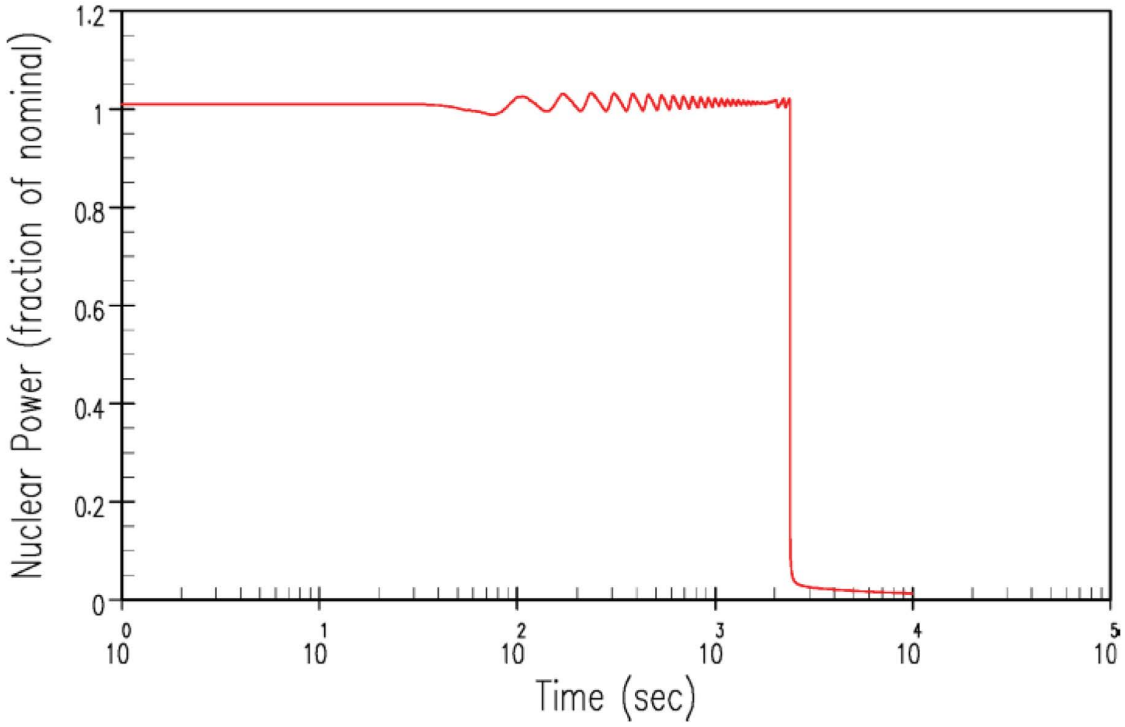

Figure 15.5.1-1
Core Nuclear Power Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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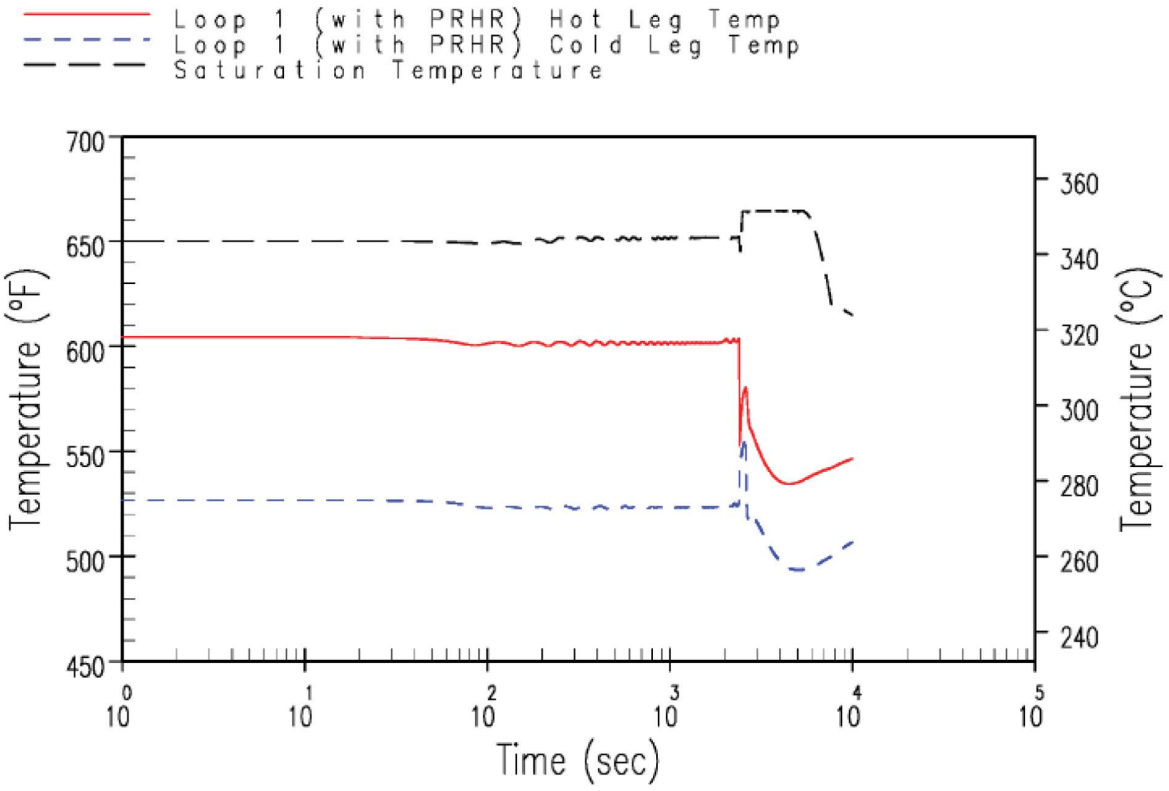

Figure 15.5.1-2
RCS Temperature Transient in Loop Containing the PRHR
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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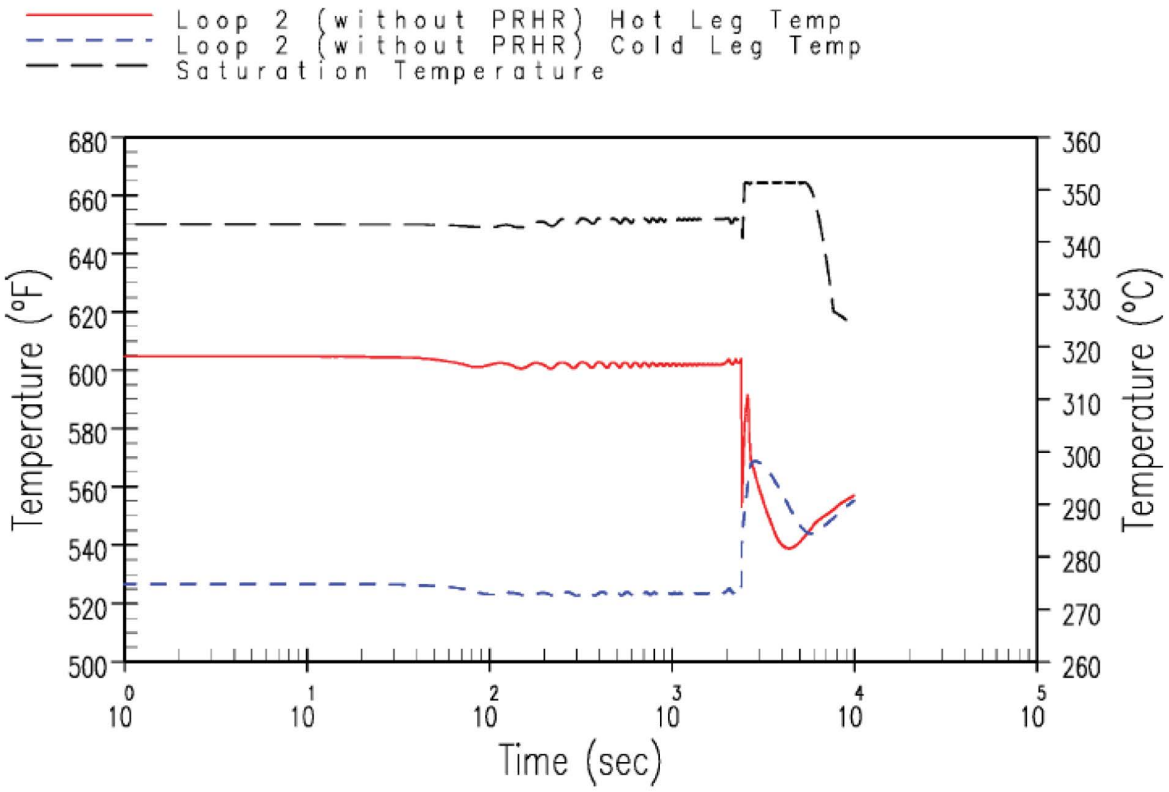

Figure 15.5.1-3
RCS Temperature Transient in Loop Not Containing the PRHR for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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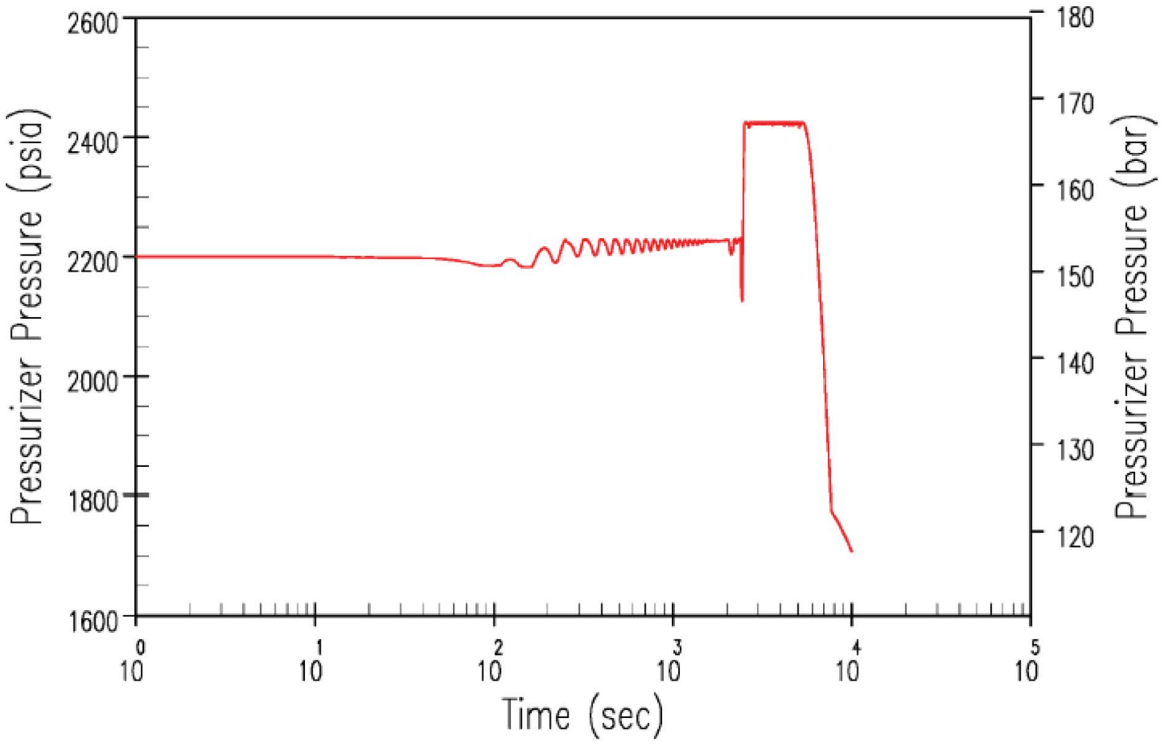

Figure 15.5.1-4
Pressurizer Pressure Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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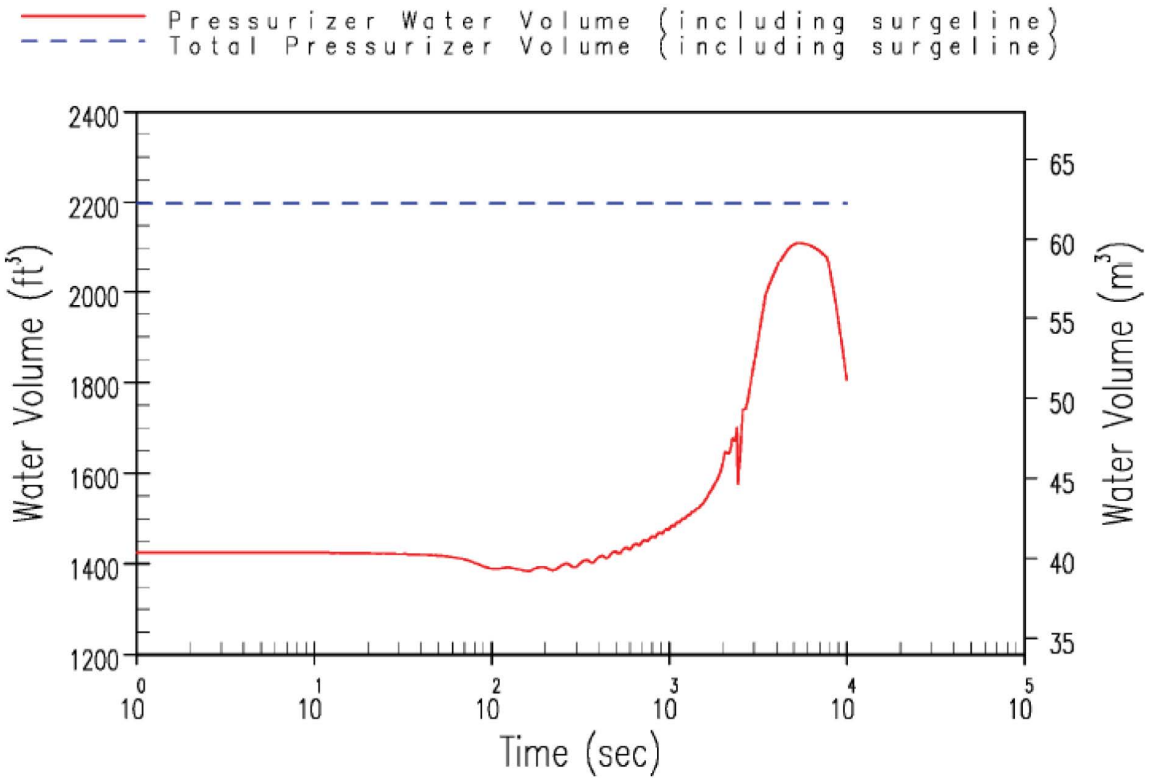

Figure 15.5.1-5
Pressurizer Water Volume Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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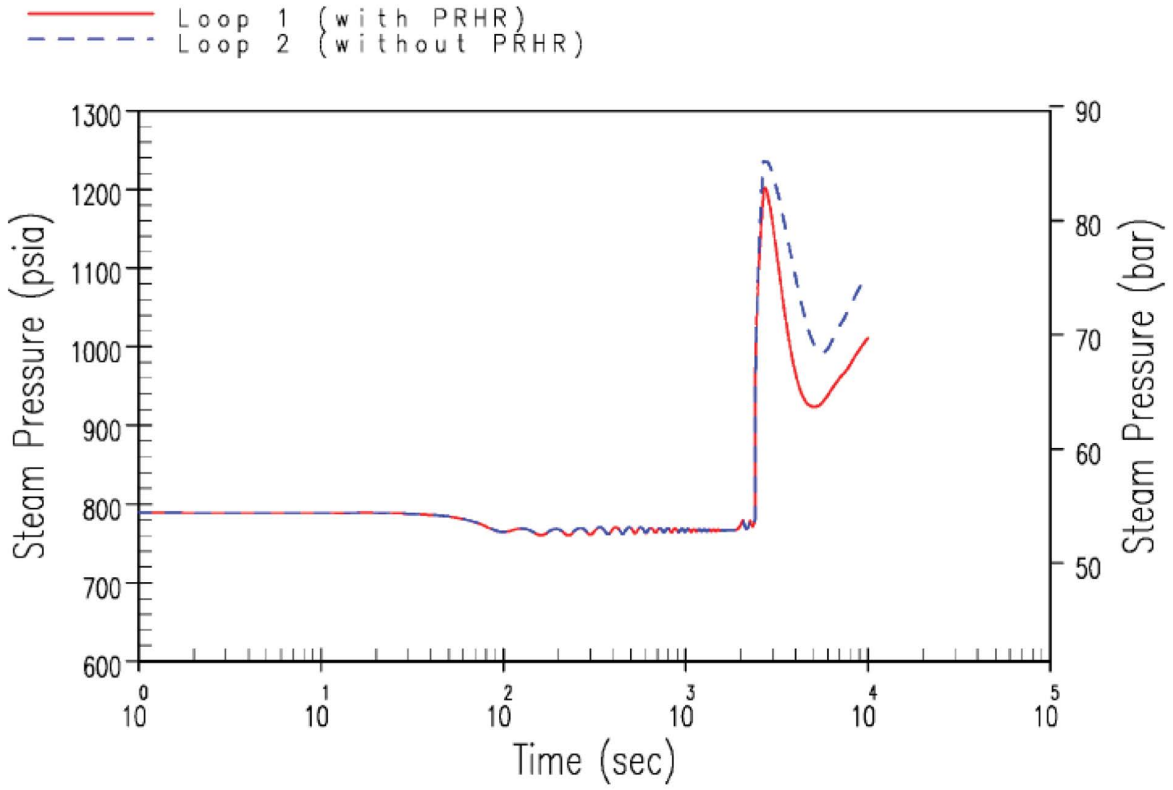

Figure 15.5.1-6
Steam Generator Pressure Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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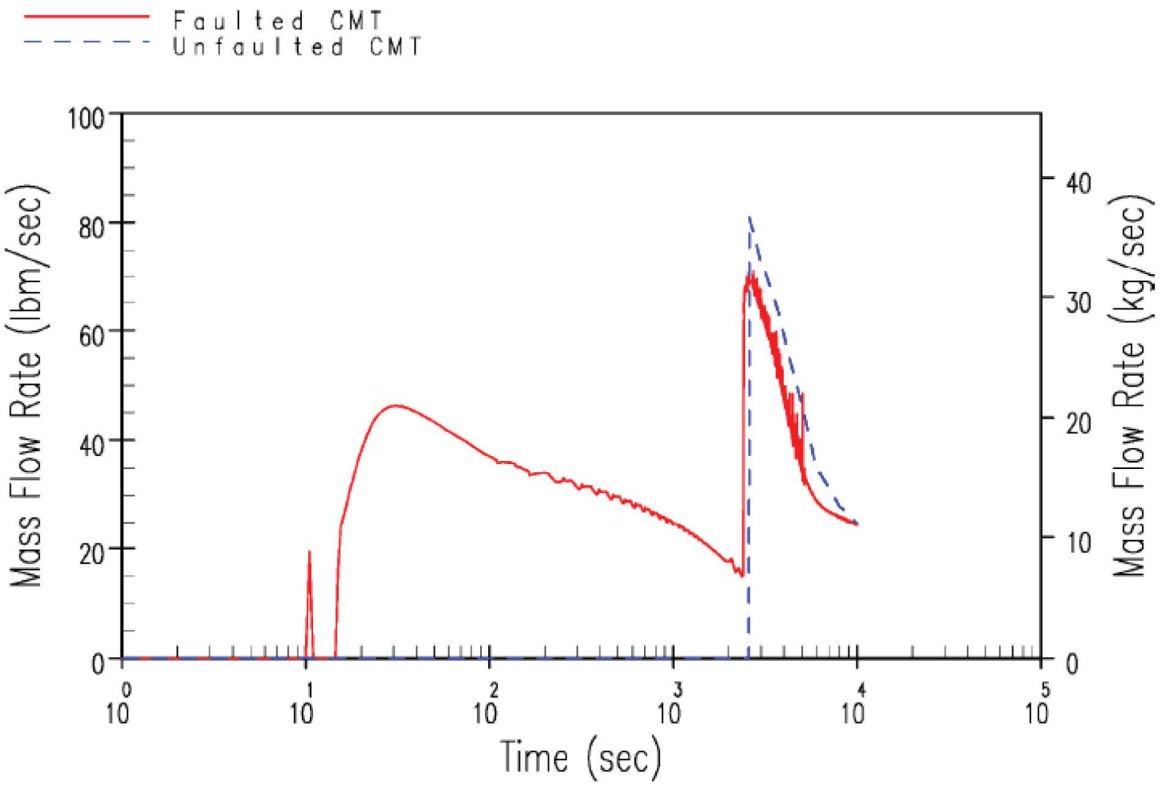

Figure 15.5.1-7
CMT Flow Rate Transient
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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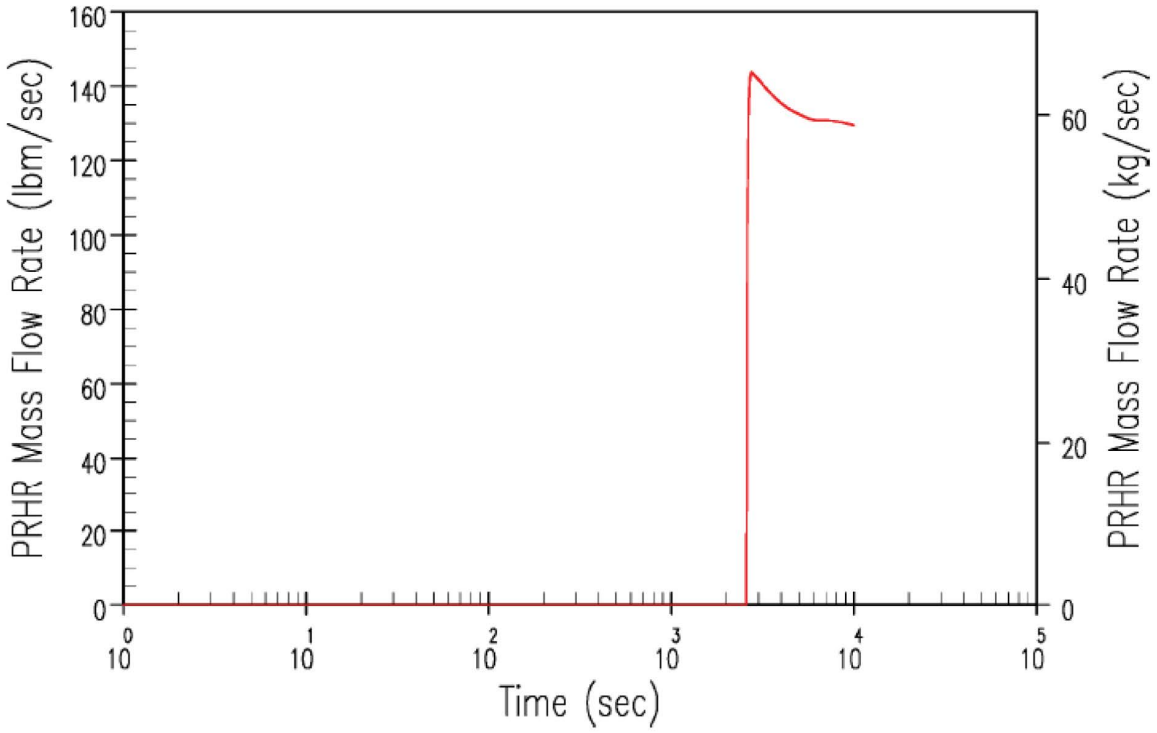

Figure 15.5.1-8
PRHR Flow Rate Transient
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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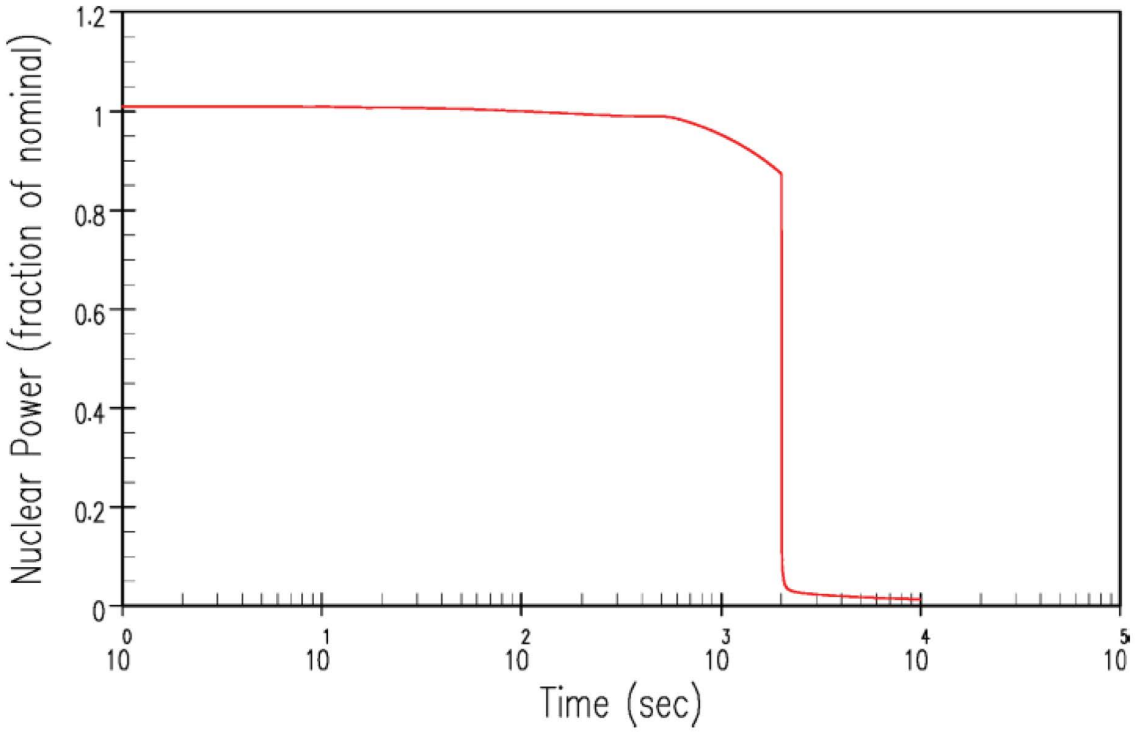

Figure 15.5.2-1
Core Nuclear Power Transient for Chemical and Volume Control System Malfunction

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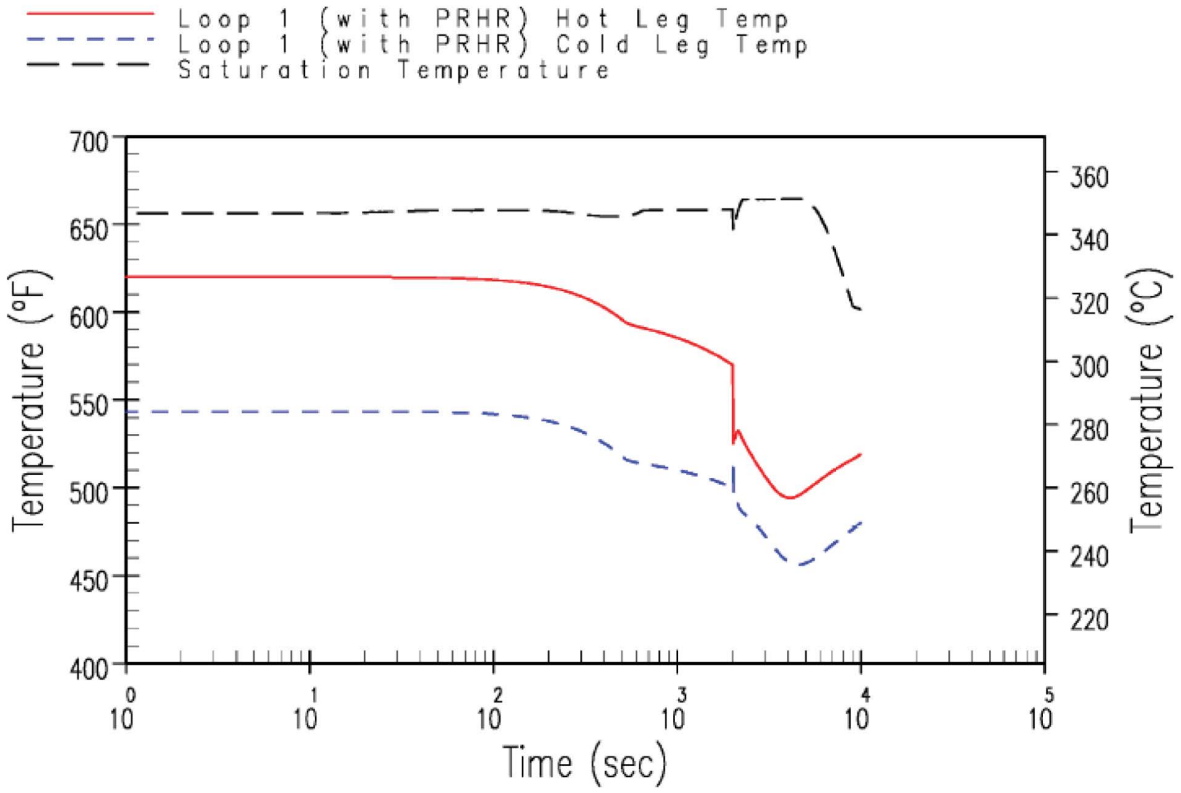

Figure 15.5.2-2
RCS Temperature Transient in Loop Containing the PRHR for Chemical and Volume Control System Malfunction

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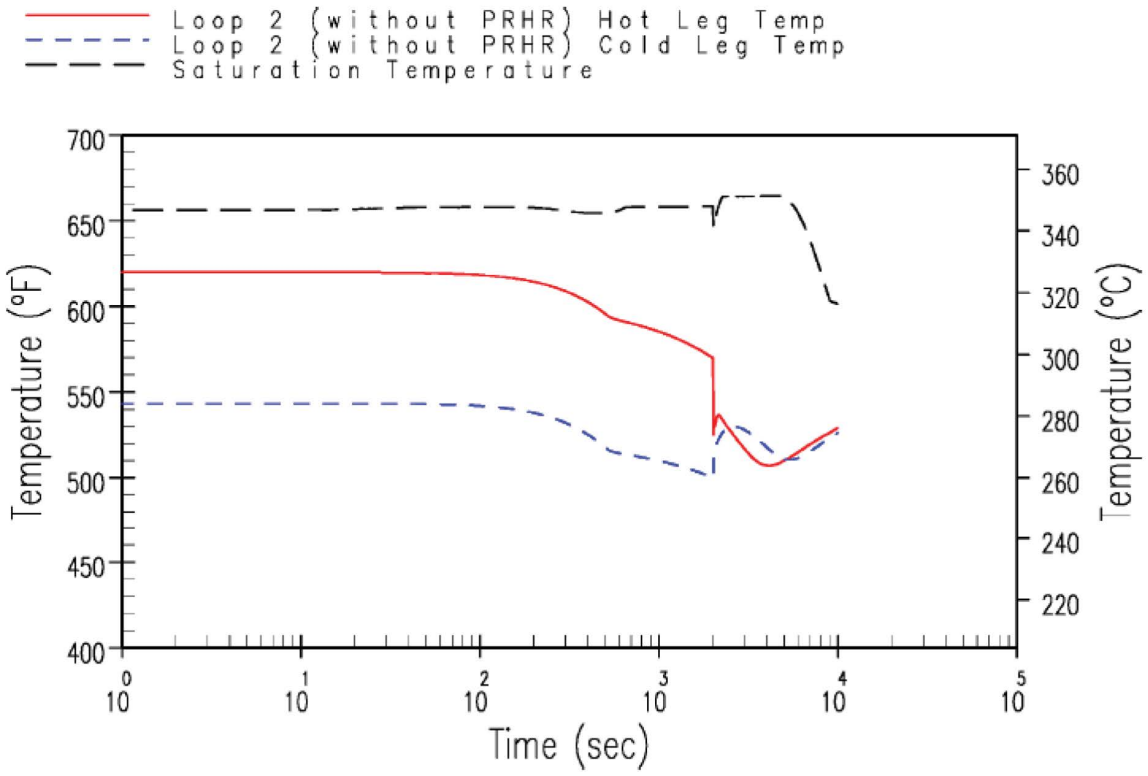

Figure 15.5.2-3
RCS Temperature Transient in Loop Not Containing the PRHR for Chemical and Volume Control System Malfunction

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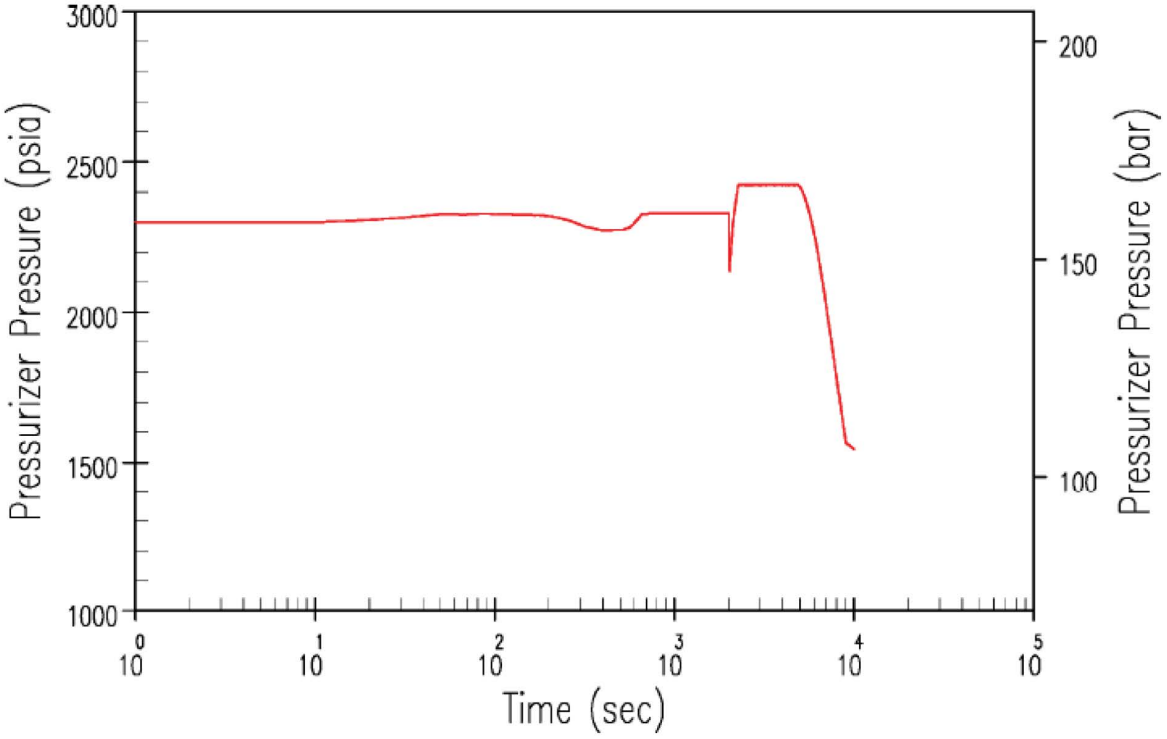

Figure 15.5.2-4
Pressurizer Pressure Transient
for Chemical and Volume Control System Malfunction

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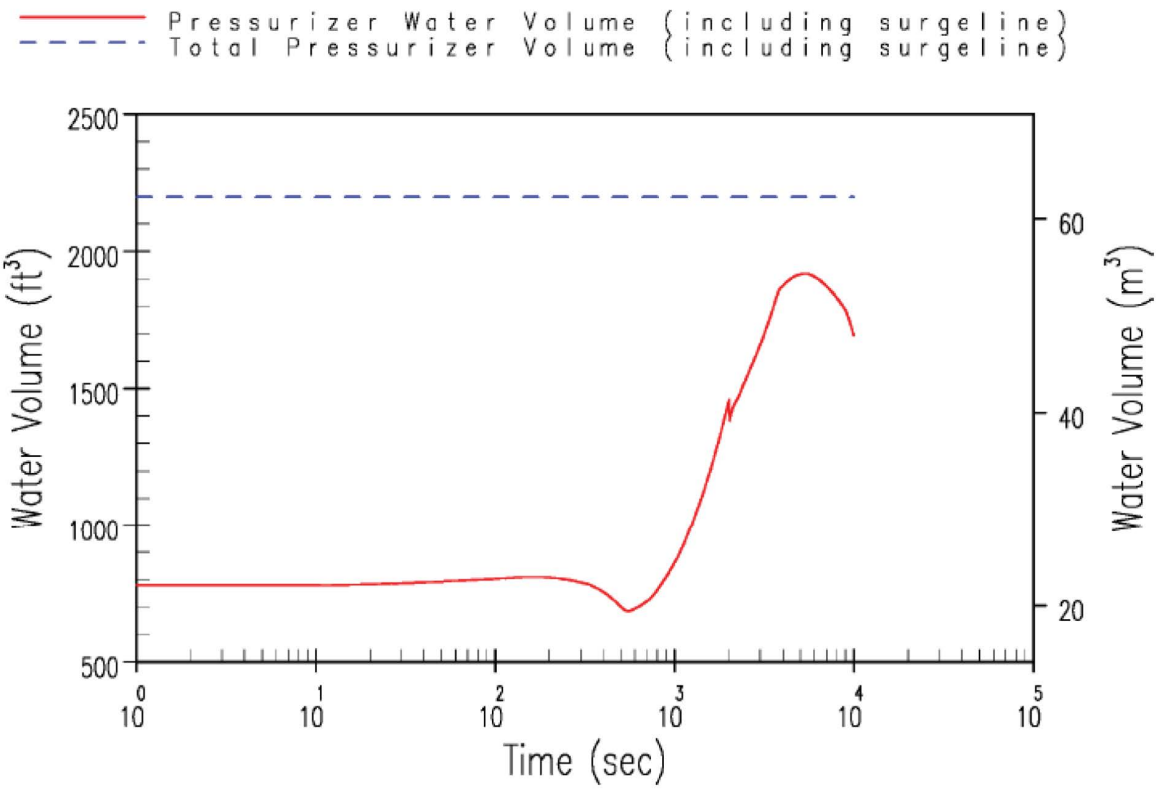

Figure 15.5.2-5
Pressurizer Water Volume Transient
for Chemical and Volume Control System Malfunction

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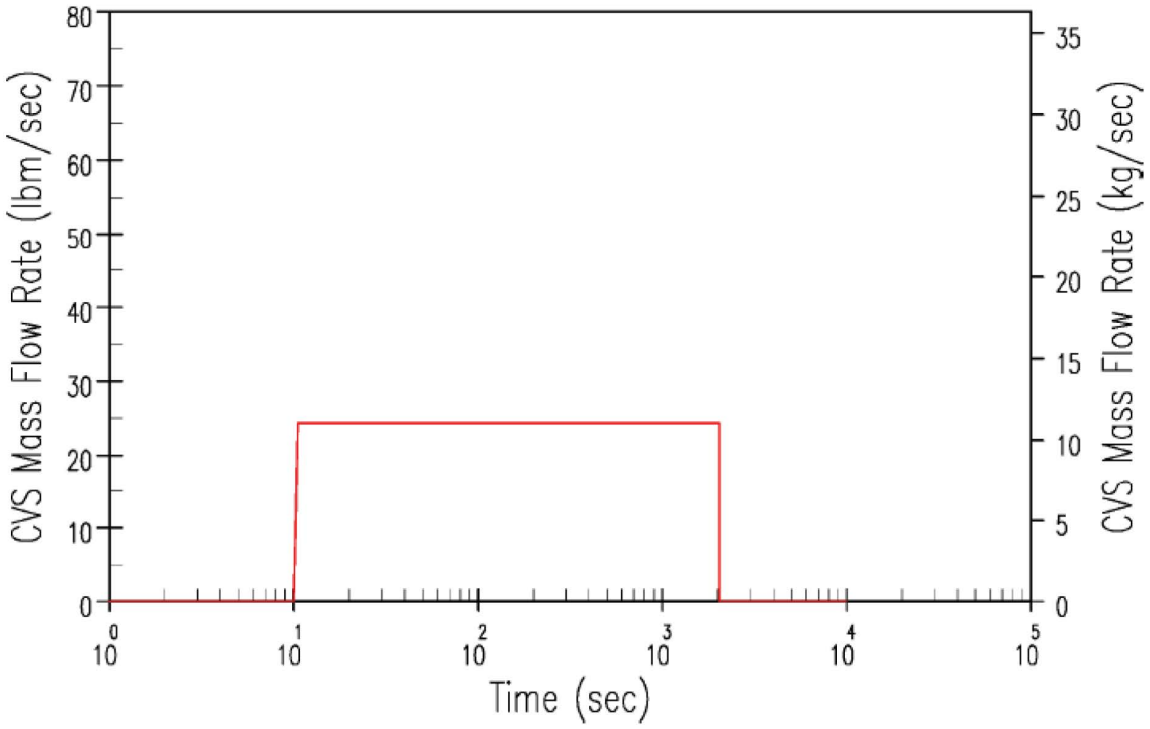

Figure 15.5.2-6
CVS Flow Rate Transient
for Chemical and Volume Control System Malfunction

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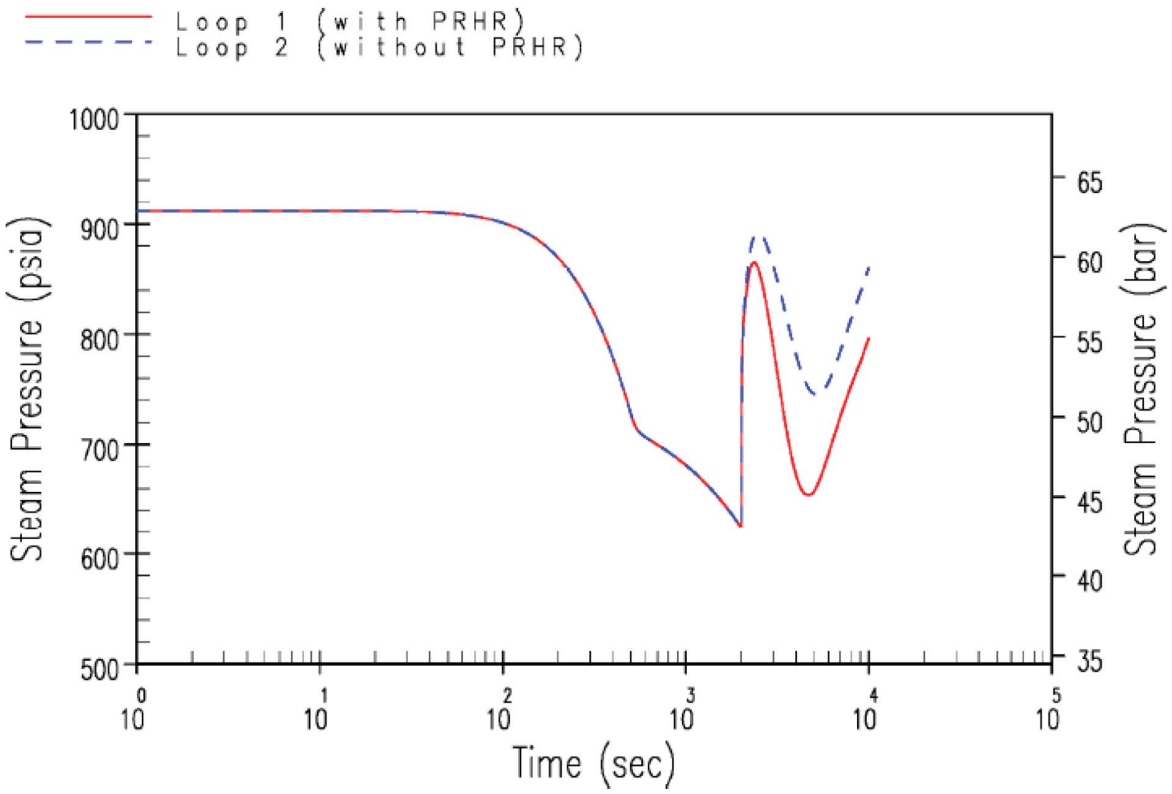

Figure 15.5.2-7
Steam Generator Pressure Transient
for Chemical and Volume Control System Malfunction

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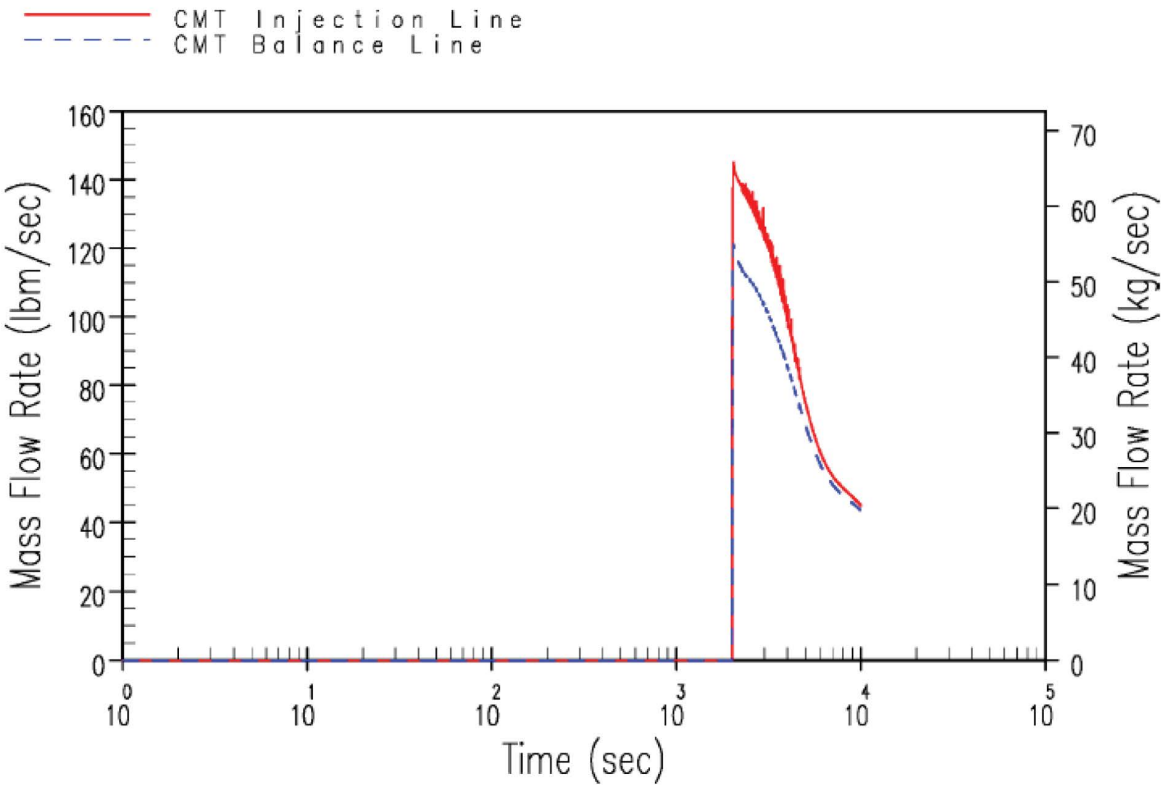

Figure 15.5.2-8
CMT Injection Line and Balance Line Flow Transient for Chemical and Volume Control System Malfunction

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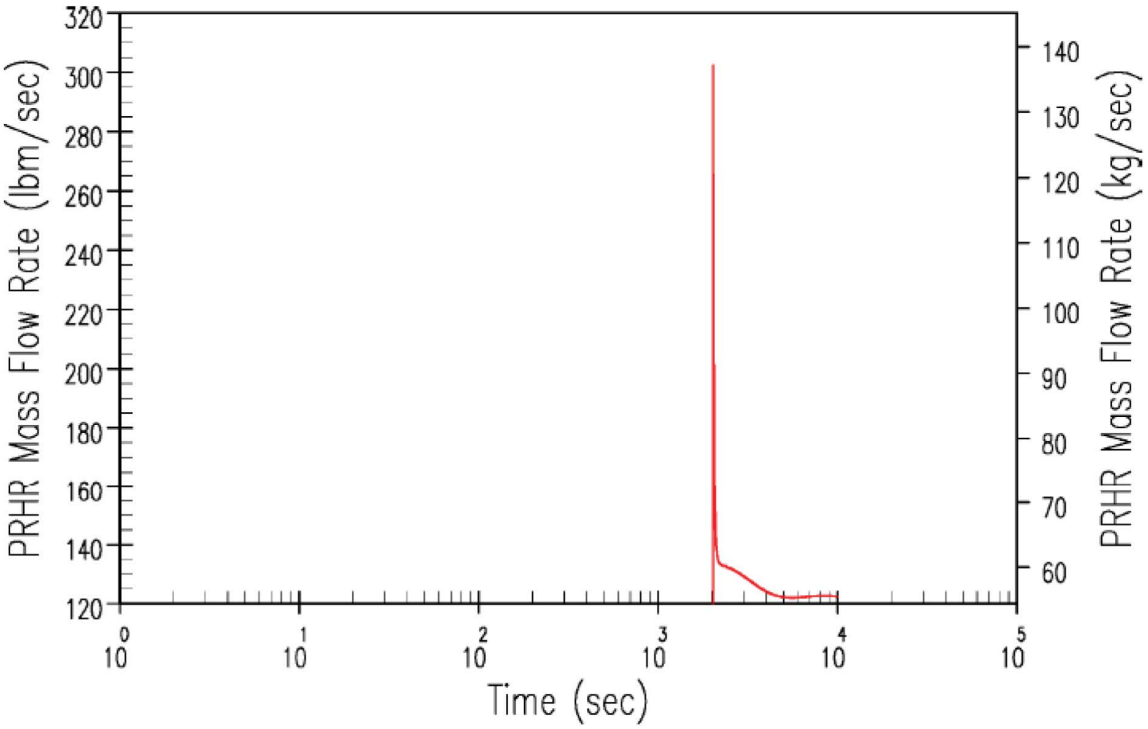

Figure 15.5.2-9
PRHR Flow Rate Transient
for Chemical and Volume Control System Malfunction

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15.6 Decrease in Reactor Coolant Inventory

This section discusses the following events that result in a decrease in reactor coolant inventory:

- An inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system (ADS)
- A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment
- A steam generator tube failure
- A loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

The applicable accidents in this category have been analyzed. It has been determined that the most severe radiological consequences result from the major LOCA described in Subsection 15.6.5. The LOCA, chemical and volume control system letdown line break outside the containment and the steam generator tube rupture (SGTR) accidents are analyzed for radiological consequences. Other accidents described in this section are bounded by these accidents.

15.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS

15.6.1.1 Identification of Causes and Accident Description

Two types of inadvertent depressurization are discussed in this section. One covers the inadvertent operation of automatic depressurization system (ADS) valves. The other covers inadvertent opening of a pressurizer safety valve.

An inadvertent depressurization of the reactor coolant system can occur as a result of an inadvertent opening of a pressurizer safety valve or ADS valves. Initially, the event results in a rapidly decreasing reactor coolant system pressure. The pressure decrease causes a decrease in power via the moderator density feedback. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system signals:

- Overtemperature ΔT
- Pressurizer Low-2 pressure

The ADS is designed such that inadvertent operation of the ADS is classified as a Condition III event, an infrequent fault.

An inadvertent opening of a pressurizer safety valve is a Condition II event, a fault of moderate frequency.

The ADS system consists of four stages of depressurization valves. The ADS stages are interlocked. For example, Stage 1 is initiated first and subsequent stages are not actuated until previous stages have completed actuation. Each stage includes two redundant parallel valve paths with two valves in series in each path such that no single failure prevents operation of the ADS stage when it is called upon to actuate and the spurious opening of a single ADS valve does not initiate ADS flow. Since each ADS path includes two valves in series, no mechanical failure could result in an inadvertent operation of an ADS stage. The ADS Stage 4 squib valves cannot be opened while the reactor

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coolant system is at nominal operating pressure. To actuate the ADS manually from the main control room, the operators actuate two separate controls positioned at some distance apart on the main control board. Therefore, one unintended operator action does not cause ADS actuation.

ADS Stage 1 has a minimum opening time of 20 seconds and a maximum effective flow area of 7 in^2 (maximum). ADS Stages 2 and 3 have a minimum opening time of 30 seconds and a maximum effective flow area of 19.4 in^2.

For this analysis, multiple failures and or errors are assumed which actuate both Stage 1 ADS paths. Although ADS Stages 2 and 3 have larger depressurization valves, the opening time of the Stage 1 depressurization valves is faster. This results in a more severe reactor coolant system depressurization due to ADS operation with the reactor at power.

Inadvertent opening of a pressurizer safety valve can only be postulated due to a mechanical failure. Although a pressurizer safety valve is smaller than the combined two Stage 1 ADS valves, the pressurizer safety valve is postulated to open in a short time.

Analyses are presented in this section for the inadvertent opening of a pressurizer safety valve and the inadvertent opening of a path of Stage 2 or 3 of the ADS. These analyses are performed to demonstrate that the departure from nucleate boiling ratio (DNBR) does not decrease below the design limit values (see Section 4.4) while the reactor is at power.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of AC power during an RCS Depressurization event have been evaluated to not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of AC power in comparison to the reactor shutdown time for an RCS Depressurization event. The primary effect of the loss of AC power is to cause the Reactor Coolant Pumps (RCPs) to coast down. The Protection & Safety Monitoring System (PMS) includes a five second minimum delay between the reactor trip and the turbine trip. In addition, a three second delay between the turbine trip and the loss of offsite AC power is assumed, consistent with the discussion of Subsection 15.0.14. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of AC power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of AC power does not adversely impact this analysis because the plant will be shut down well before the RCPs begin to coast down.

15.6.1.2 Analysis of Effects and Consequences

15.6.1.2.1 Method of Analysis

The accidental depressurization transient is analyzed by using the computer code LOFTRAN (References 14 and 15). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, main steam isolation valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. The following assumptions are made to give conservative results in calculating the DNBR during the transient:

- Initial conditions are discussed in Subsection 15.0.3. Uncertainties in initial conditions are included in the DNBR limit as discussed in WCAP-11397-P-A (Reference 16).

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- A least negative moderator temperature coefficient is assumed. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

- A large (absolute value) Doppler coefficient of reactivity is used such that the resulting amount of positive feedback is conservatively high to retard any power decrease.

Plant systems and equipment necessary to mitigate the effects of reactor coolant system depressurization are discussed in Subsection 15.0.8 and are listed in Table 15.0-6.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode to maintain the core at full power until the reactor trip protection function is reached. This is a worst case assumption. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure prevents the reactor protection system from functioning properly.

15.6.1.2.2 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 15.6.1-1 through 15.6.1-4. The calculated sequence of events for the inadvertent opening of a pressurizer safety valve scenario is shown in Table 15.6.1-1.

A pressurizer safety valve is assumed to step open at the start of the event. The reactor coolant system then depressurizes until the Low-2 pressurizer pressure reactor trip setpoint is reached. Figure 15.6.1-3 shows the pressurizer pressure transient.

Prior to tripping of the reactor, the core power remains relatively constant (Figure 15.6.1-1). The minimum DNBR during the event occurs shortly after the rods begin to be inserted into the core (Figure 15.6.1-2). The DNBR remains above the design limit values as discussed in Section 4.4 throughout the transient.

The system response for inadvertent operation of the ADS is shown in Figures 15.6.1-5 through 15.6.1-8. The sequence of events is provided in Table 15.6.1-1. The system response for inadvertent operation of the ADS is very similar to that obtained for inadvertent opening of a pressurizer safety valve.

15.6.1.3 Conclusion

The results of the analysis show that the Low-2 pressurizer pressure reactor protection system signal provides adequate protection against the reactor coolant system depressurization events. The calculated DNBR remains above the design limit defined in Section 4.4. The long-term plant responses due to a stuck-open ADS valve or pressurizer safety valve, which cannot be isolated, are bounded by the small-break LOCA analysis.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

The small lines carrying primary coolant outside containment are the reactor coolant system sample line and the discharge line from the chemical and volume control system to the liquid radwaste system. These lines are used only periodically. No instrument lines carry primary coolant outside the containment.

When excess primary coolant is generated because of boron dilution operations, the chemical and volume control system purification flow is diverted out of containment to the liquid radwaste system. Before passing outside containment, the flow stream passes through the chemical and volume

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control system heat exchangers and mixed bed demineralizer. The flow leaving the containment is at a temperature of less than 140°F and has been cleaned by the demineralizer. The flow out a postulated break in this line is limited to the chemical and volume control system purification flow rate of 100 gpm. Considering the low temperature of the flow and the reduced iodine activity because of demineralization, this event is not analyzed. The postulated sample line break is more limiting.

A continuous sample of the RCS hot legs flows through the normally open isolation valves inside and outside containment. The failure of the sample line is postulated to occur between the isolation valve outside the containment and the sample panel. The loss of sample flow provides indication of the break to plant personnel. In addition, a break in a sample line results in activity release and a resulting actuation of area and air radiation monitors. The loss of coolant tends to reduce the pressurizer level and creates a demand for makeup to the reactor coolant system providing additional indication. Upon indication of a sample line break, the operator would take action to isolate the break.

The sample line includes a flow restrictor at the point of sample to limit the break flow to less than 130 gpm. The liquid sampling lines are 1/4 inch tubing which further restricts the break flow of a sampling line outside containment. Offsite doses are based on a conservative break flow of 130 gpm with isolation after 30 minutes.

15.6.2.1 Source Term

The only significant radionuclide releases are the iodines and the noble gases. The analysis assumes that the reactor coolant iodine is at the maximum Technical Specification level for continuous operation. In addition, it is assumed that an iodine spike occurs at the time of the accident. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity.

15.6.2.2 Release Pathway

The reactor coolant that is spilled from the break is assumed to be at high temperature and pressure. A large portion of the flow flashes to steam, and the iodine in the flashed liquid is assumed to become airborne.

The iodine and noble gases are assumed to be released directly to the environment with no credit for depletion, although a large fraction of the airborne iodine is expected to deposit on building surfaces. No credit is assumed for radioactive decay after release.

15.6.2.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.6.2.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.2-1.

15.6.2.5 Identification of Conservatisms

The assumptions used contain the following significant conservatisms:

- The reactor coolant activities are based on conservative assumptions (see Table 15.6.2-1); whereas, the expected activities based on the fuel defect level are far less (see Section 11.1).

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- It is unlikely that the conservatively selected meteorological conditions would be present at the time of the accident.

15.6.2.6 Doses

Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be 1.3 rem at the exclusion area boundary and 0.6 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase “a small fraction” is taken as being ten percent or less.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the small line break outside containment, the resulting total dose remains less than the value reported above.

15.6.3 Steam Generator Tube Rupture

15.6.3.1 Identification of Cause and Accident Description

15.6.3.1.1 Introduction

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods within the allowance of the Technical Specifications. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or a failure of the condenser steam dump, discharge of radioactivity to the atmosphere takes place via the steam generator power-operated relief valves or the safety valves.

The assumption of a complete tube severance is conservative because the steam generator tube material (Alloy 690) is a corrosion-resistant and ductile material. The more probable mode of tube failure is one or more smaller leaks of undetermined origin. Activity in the secondary side is subject to continual surveillance, and an accumulation of such leaks, which exceeds the limits established in the Technical Specifications, is not permitted during operation.

The AP1000 design provides automatic protective actions to mitigate the consequences of an SGTR. The automatic actions include reactor trip, actuation of the passive residual heat removal (PRHR) heat exchanger, initiation of core makeup tank flow, termination of pressurizer heater operation, and isolation of chemical and volume control system flow and startup feedwater flow on High-3 steam generator level or high steam generator level coincident with reactor trip (P-4). These protective actions result in automatic cooldown and depressurization of the reactor coolant system, termination of the break flow and release of steam to the atmosphere, and long-term maintenance of stable conditions in the reactor coolant system. These protection systems serve to prevent steam generator overfill (see discussion in Subsections 15.6.3.1.2 and 15.6.3.1.3) and to maintain offsite radiation doses within the allowable guideline values for a design basis SGTR. The operator may take actions that would provide a more rapid mitigation of the consequences of an SGTR.

Because of the series of alarms described next, the operator can readily determine when an SGTR occurs, identify and isolate the ruptured steam generator, and complete the required recovery actions

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to stabilize the plant and terminate the primary-to-secondary break flow. The recovery procedures are completed on a time scale that terminates break flow to the secondary system before steam generator overfill occurs and limits the offsite doses to acceptable levels without actuation of the ADS. Indications and controls are provided to enable the operator to carry out these functions.

15.6.3.1.2 Sequence of Events for a Steam Generator Tube Rupture

The following sequence of events occur following an SGTR:

- Pressurizer low pressure and low level alarms are actuated and chemical and volume control system makeup flow and pressurizer heater heat addition starts or increases in an attempt to maintain pressurizer level and pressure. On the secondary side, main feedwater flow to the affected steam generator is reduced because the primary-to-secondary break flow increases steam generator level.

- The condenser air removal discharge radiation monitor, steam generator blowdown radiation monitor, and/or main steam line radiation monitor alarm indicate an increase in radioactivity in the secondary system.

- Continued loss of reactor coolant inventory leads to a reactor trip generated by a Low-2 pressurizer pressure or over-temperature ΔT signal. Following reactor trip, the SGTR leads to a decrease in reactor coolant pressure and pressurizer level, counteracted by chemical and volume control system flow and pressurizer heater operation. Either the Low-2 pressurizer level signal or a safeguards (“S”) signal from Low-3 pressurizer pressure, actuates the core makeup tanks. A Low-2 pressurizer level signal also trips the reactor coolant pumps. The “S” signal automatically terminates the normal feedwater supply and trips the reactor coolant pumps. The core makeup tank actuation signal will actuate the PRHR heat exchanger and trip pressurizer heaters. Startup feedwater flow is initiated on a low steam generator narrow range level signal and controls the steam generator levels to the programmed level.

- The reactor trip automatically trips the turbine, and if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of a loss of offsite power or loss of the condenser, the steam dump valves automatically close to protect the condenser. The steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the steam generator power-operated relief valves and/or the safety valves.

- Following reactor trip and core makeup tank and PRHR actuation, the PRHR heat exchanger operation – combined with startup feedwater flow, borated core makeup tank flow, and chemical and volume control system flow – provides a heat sink that absorbs the decay heat. This reduces the amount of steam generated in the steam generators and steam bypass to the condenser. In the case of loss of offsite power, this reduces steam relief to the atmosphere.

- Injection of the chemical and volume control system and core makeup tank flow stabilizes reactor coolant system pressure and pressurizer water level, and the reactor coolant system pressure trends toward an equilibrium value, where the total injected flow rate equals the break flow rate.

15.6.3.1.3 Steam Generator Tube Rupture Automatic Recovery Actions

The AP1000 incorporates several protection system and passive design features that automatically terminate a steam generator tube leak and stabilize the reactor coolant system, in the highly unlikely event that the operators do not perform recovery actions. Following an SGTR, the injecting chemical

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and volume control system flow (and pressurizer heater heat addition if the pressure control system is operating) maintains the primary-to-secondary break flow and the ruptured steam generator secondary level increases as break flow accumulates in the steam generator. Eventually, the ruptured steam generator secondary level reaches the high and High-3 steam generator narrow range level setpoint, which is near the top of the narrow range level span.

The AP1000 protection system automatically provides several safety-related actions to cool down and depressurize the reactor coolant system, terminate the break flow and steam release to the atmosphere, and stabilize the reactor coolant system in a safe condition. The safety-related actions include initiation of the PRHR system heat exchanger, isolation of the chemical and volume control system pumps and pressurizer heaters, and isolation of the startup feedwater pumps. In addition, the protection and safety monitoring system provides a safety-related signal to trip the redundant, nonsafety related pressurizer heater breakers.

Actuating the PRHR heat exchanger transfers core decay heat to the in-containment reactor water storage tank (IRWST) and initiates a cooldown (and a consequential depressurization) of the reactor coolant system.

Isolation of the chemical and volume control system pumps and pressurizer heaters minimizes the repressurization of the primary system. This allows primary pressure to equilibrate with the secondary pressure, which effectively terminates the primary-to-secondary break flow. Because the core makeup tank continues to inject when needed to provide boration following isolation of the chemical and volume control system pumps, isolating the chemical and volume control system pumps does not present a safety concern.

Isolation of the startup feedwater provides protection against a failure of the startup feedwater control system, which could potentially result in the ruptured steam generator being overfilled.

With decay heat removal by the PRHR heat exchanger, steam generator steaming through the power-operated relief valves ceases and steam generator secondary level is maintained.

15.6.3.1.4 Steam Generator Tube Rupture Assuming Operator Recovery Actions

In the event of an SGTR, the operators can diagnose the accident and perform recovery actions to stabilize the plant, terminate the primary-to-secondary leakage, and proceed with orderly shutdown of the reactor before actuation of the automatic protection systems. The operator actions for SGTR recovery are provided in the plant emergency operating procedures. The major operator actions include the following:

- Identify the ruptured steam generator – The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level or a high radiation indication from any main steam line monitor, steam generator blowdown line monitor, or steam generator sample.

- Isolate the ruptured steam generator – Once the steam generator with the ruptured tube is identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator.

- Cooldown of the reactor coolant system using the intact steam generator or the PRHR system – After isolation of the ruptured steam generator, the reactor coolant system is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure. This provides adequate subcooling in the reactor coolant system after depressurization of the reactor coolant system to the ruptured steam generator pressure in subsequent actions.

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- Depressurize the reactor coolant system to restore reactor coolant inventory – When the cooldown is completed, the chemical and volume control system and core makeup tank injection flow increases the reactor coolant system pressure until break flow matches the total injection flow. Consequently, these flows must be terminated or controlled to stop primary-to-secondary leakage. However, adequate reactor coolant inventory must first be provided. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after the injection flow is stopped.

Because leakage from the primary side continues after the injection flow is stopped, until reactor coolant system and ruptured steam generator pressures equalize, the reactor coolant system is depressurized to provide sufficient inventory to verify that the pressurizer level remains on span after the pressures equalize.

- Termination of the injection flow to stop primary to secondary leakage – The previous actions establish adequate reactor coolant system subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to verify that injection flow is no longer needed. When these actions are completed, core makeup tank and chemical and volume control system flow is stopped to terminate primary-to-secondary leakage. Primary-to-secondary leakage continues after the injection flow is stopped until the reactor coolant system and ruptured steam generator pressures equalize. Chemical and volume control system makeup flow, letdown, pressurizer heaters, and decay heat removal via the intact steam generator or the PRHR heat exchanger are then controlled to prevent repressurization of the reactor coolant system and reinitiation of leakage into the ruptured steam generator.

Following the injection flow termination, the plant conditions stabilize and the primary-to-secondary break flow terminates. At this time, a series of operator actions is performed to prepare the plant for cooldown to cold shutdown conditions. The actions taken depend on the available plant systems and the plan for further plant repair and operation.

15.6.3.2 Analysis of Effects and Consequences

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. An analysis is performed to demonstrate that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines.

One of the concerns for an SGTR is the possibility of steam generator overfill because this can potentially result in a significant increase in the offsite radiological consequences. Automatic protection and passive design features are incorporated into the AP1000 design to automatically terminate the break flow to prevent overfill during an SGTR. These features include actuation of the PRHR system, isolation of chemical and volume control system flow, and isolation of startup feedwater.

An analysis is performed, without modeling expected operator actions to isolate the ruptured steam generator and cool down and depressurize the reactor coolant system, to demonstrate the role that the AP1000 design features have in preventing steam generator overfill. The limiting single failure for the overfill analysis is assumed to be the failure of the startup feedwater control valve to throttle flow when nominal steam generator level is reached. Other conservative assumptions that maximize steam generator secondary volume (such as high initial steam generator level, minimum initial reactor coolant system pressure, loss of offsite power, maximum chemical and volume control system injection flow, maximum pressurizer heater addition, maximum startup feedwater flow, and minimum startup feedwater delay time) are also assumed.

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The results of this analysis demonstrate the effectiveness of the AP1000 protection system and passive system design features and support the conclusion that an SGTR event would not result in steam generator overfill.

For determining the offsite radiological consequences, an SGTR analysis is performed assuming the limiting single failure and limiting initial conditions relative to offsite doses. Because steam generator overfill is prevented for the AP1000, the results of this analysis represent the limiting radiological consequences for an SGTR.

A thermal-hydraulic analysis is performed to determine the plant response for a design basis SGTR, the integrated primary-to-secondary break flow, and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere. This information is then used to calculate the radioactivity release to the environment and the resulting radiological consequences.

15.6.3.2.1 Method of Analysis

15.6.3.2.1.1 Computer Program

The plant response following an SGTR until the primary-to-secondary break flow is terminated is analyzed with the LOFTTR2 program (Reference 21). The LOFTTR2 program is modified to model the PRHR system, core makeup tanks, and protection system actions appropriate for the AP1000. These modifications to LOFTTR2 are described in WCAP-14234, Revision 1 (Reference 14).

15.6.3.2.1.2 Analysis Assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The location of the break on the cold leg side of the steam generator results in higher initial primary-to-secondary leakage than a break on the hot side of the steam generator.

The reactor is assumed to be operating at full power at the time of the accident, and the initial secondary mass is assumed to correspond to operation at nominal steam generator mass minus an allowance for uncertainties. Offsite power is assumed to be lost and the rods are assumed to be inserted at the start of the event because continued operation of the reactor coolant pumps has been determined to reduce flashing of primary-to-secondary break flow and, consequently, lower offsite radiological doses. Maximum chemical and volume control system flows and pressurizer heater heat addition are assumed immediately (even though offsite power is not available) to conservatively maximize primary-to-secondary leakage. The steam dump system is assumed to be inoperable, consistent with the loss of offsite power assumption, because this results in steam release from the steam generator power-operated relief valves to the atmosphere following reactor trip. The chemical and volume control system and pressurizer heater modeling is conservatively chosen to delay the Low-3 pressurizer pressure “S” and the low-2 pressurizer level signal and associated protection system actions.

The limiting single failure is assumed to be the failure of the ruptured steam generator power-operated relief valve. Failure of this valve in the open position causes an uncontrolled depressurization of the ruptured steam generator, which increases primary-to-secondary leakage and the mass release to the atmosphere.

It is assumed that the ruptured steam generator power-operated relief valve fails open when the low-2 pressurizer level signal is generated. This results in the maximum integrated flashed primary-to-secondary break flow.

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The valve is subsequently isolated when the associated block valve is automatically closed on a Low-2 steam line pressure protection system signal.

No operator actions are modeled in this limiting analysis, and the plant protection system provides the protection for the plant. Not modeling operator actions is conservative because the operators are expected to have sufficient time to recover from the accident and supplement the automatic protection system. In particular, the operator would take action to reduce the primary pressure before the high steam generator level coincident with reactor trip (P-4) chemical and volume control and startup feedwater system shutoff signals are generated. It is also expected that the operator can close the block valve to the ruptured steam generator power-operated relief valve in much shorter time than the automatic protection signal. The operators can quickly diagnose a power-operated relief valve failure based on the rapid depressurization of the steam generator and increase in steam flow. They can then close the block valve from the control panel.

Consistent with the assumed loss of offsite power, the main feedwater pumps coast down and no startup feedwater is assumed to conservatively minimize steam generator secondary inventory and thus maximize secondary activity concentration and steam release.

15.6.3.2.1.3 Results

The sequence of events for this transient is presented in Table 15.6.3-1. The system responses to the SGTR accident are shown in Figures 15.6.3-1 to 15.6.3-10.

Offsite power is lost concurrent with the rupture of the tube. The reactor trips due to the loss of offsite power. The main feedwater pumps are assumed to coast down following reactor trip. The startup feedwater pumps are conservatively assumed not to start. Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator. In response to this loss of reactor coolant, pressurizer level and reactor coolant system pressure decreases as shown in Figures 15.6.3-1 and 15.6.3-2. As a result of the decreasing pressurizer level and pressure, two chemical and volume control system pumps are automatically initiated to provide makeup flow and the pressurizer heaters turn on.

After reactor trip, core power rapidly decreases to decay heat levels and the core inlet to outlet temperature differential decreases. The turbine stop valves close, and steam flow to the turbine is terminated. The steam dump system is conservatively assumed to be inoperable. The secondary side pressure increases rapidly after reactor trip until the steam generator power-operated relief valves (and safety valves, if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.6.3-3.

Maximum heat addition to the pressurizer from the pressurizer heaters increases the primary pressure.

As the leak flow continues to deplete primary inventory, Low-2 pressurizer level, core makeup tank actuation, and PRHR actuation signals area reached. Power to the pressurizer heaters is shut off so that they will not provide additional heat to the primary should the pressurizer level return. The ruptured steam generator power-operated relief valve is assumed to fail open at this time.

The failure causes the intact and ruptured steam generators to rapidly depressurize (Figure 15.6.3-3). This results in an initial increase in primary-to-secondary leakage and a decrease in the reactor coolant system temperatures. Both the intact and ruptured steam generators depressurize because the steam generators communicate through the open steam line isolation valves.

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The decrease in the reactor coolant system temperature results in a decrease in the pressurizer level and reactor coolant system pressure (Figures 15.6.3-1 and 15.6.3-2). Depressurization of the primary and secondary systems continues until the Low-2 steam line pressure setpoint is reached. As a result, the steam line isolation valves and intact and ruptured steam generator power-operated relief block valves are closed.

Following closure of the block valves, the primary and secondary pressures and the ruptured steam generator secondary water volume and mass increase as break flow accumulates. This increase continues until the steam generator secondary level reaches the high narrow range level when the chemical and volume control and startup feedwater systems are isolated.

With continued reactor coolant system cooldown, depressurization provided by the PRHR heat exchanger, and with the chemical and volume control system isolated, primary system pressure eventually falls to match the secondary pressure. The break flow terminates as shown in Figure 15.6.3-5, and the system is stabilized in a safe condition. As shown in Figure 15.6.3-8, steam release through the intact loop, unfaulted power-operated relief valve does not occur following PRHR initiation because the PRHR is capable of removing the core decay heat.

As shown in Figure 15.6.3-9, the core makeup tank flow trends toward zero because the gravity head diminishes as the core makeup tank temperature approaches the reactor coolant system temperature due to the continued balance line flow. The core makeup tank remains full, and ADS actuation does not occur.

The ruptured steam generator water volume is shown in Figure 15.6.3-6. The water volume in the ruptured steam generator when the break flow is terminated is significantly less than the total steam generator volume of greater than 9000 ft^3.

The design basis SGTR event does not result in fuel failures. In the event of an SGTR, the reactor coolant system depressurizes due to the primary-to-secondary leakage through the ruptured steam generator tube. This depressurization reduces the calculated DNBR. The depressurization prior to reactor trip for the SGTR has been compared to the depressurization for the reactor coolant system depressurization accidents analyzed in Subsection 15.6.1. The rate of depressurization is much slower for the SGTR than for the reactor coolant system depressurization accidents. Following reactor trip, the DNBR increases rapidly. Thus, the conclusion of Subsection 15.6.1, that the calculated DNBR remains above the limit, is extended to the SGTR analysis, justifying the assumption of no failed fuel.

15.6.3.2.1.4 Mass Releases

The mass release of an SGTR event is determined for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators and the primary-to-secondary leakage into the ruptured steam generator are determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following reactor trip, the releases to the atmosphere are through the steam generator power-operated relief valves (and steam generator safety valves for a short period). Steam relief through the power-operated relief valves continues until RNS conditions are met. The mass releases for the SGTR event are presented in Table 15.6.3-2.

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15.6.3.3 Radiological Consequences

The evaluation of the radiological consequences of the postulated SGTR assumes that the reactor is operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes result in a buildup of activity in the secondary coolant.

Following the rupture, any noble gases carried from the primary coolant into the ruptured steam generator via the break flow are released directly to the environment. The iodine and alkali metal activity entering the secondary side is also available for release, with the amount of release dependent on the flashing fraction of the reactor coolant and on the partition coefficient in the steam generator. In addition to the activity released through the ruptured loop, there is also a small amount of activity released through the intact loop.

15.6.3.3.1 Source Term

The significant radionuclide releases from the SGTR are the noble gases, alkali metals and the iodines that become airborne and are released to the environment as a result of the accident.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with the equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs before the accident and that the maximum reactor coolant iodine concentration exists at the time the accident occurs. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The reactor coolant alkali metal concentrations are based on those associated with the design basis fuel defect level.

The secondary coolant iodine and alkali metal activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity.

15.6.3.3.2 Release Pathways

The noble gas activity contained in the reactor coolant that leaks into the intact steam generator and enters the ruptured steam generator through the break is assumed to be released immediately as long as a pathway to the environment exists. There are three components to the modeling of iodine and alkali metal releases:

- Intact loop steaming, with credit for partitioning of iodines and alkali metals (includes continued primary-to-secondary leakage at the maximum rate allowable by the Technical Specifications)

- Ruptured loop steaming, with credit for partitioning of iodines and alkali metals (includes modeling of increasing activity in the secondary coolant due to the break flow)

- Release of flashed reactor coolant through the ruptured loop, with no credit for scrubbing (this conservatively assumes that break location is at the top of the tube bundle)

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion of iodines by ground deposition during transport offsite.

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15.6.3.3.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.6.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.3-3.

15.6.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms, such as:

- The reactor coolant activities are based on conservative assumptions whereas, the activities based on the expected fuel defect level are far less (see Section 11.1).
- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

15.6.3.3.6 Doses

Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be 0.7 rem at the exclusion area boundary for the limiting 2-hour interval (0-2 hours) 0.5 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A “small fraction” is defined, consistent with the Standard Review Plan, as being ten percent or less.

For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be 1.4 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and 0.7 rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour exclusion area boundary dose because pool boiling would not occur until after 2.0 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the doses calculated for the steam generator tube rupture, the resulting total doses remain as reported above.

15.6.3.4 Conclusions

The results of the SGTR analysis show that the overfill protection logic and the passive system design features provide protection to prevent steam generator overfill. Following an SGTR accident, the operators can identify and isolate the ruptured steam generator and complete the required actions to terminate the primary-to-secondary break flow before steam generator overfill or ADS actuation occurs.

Even when no operator actions are assumed, the AP1000 protection system and passive design features initiate automatic actions that can terminate a steam generator tube leak and stabilize the reactor coolant system in a safe condition while preventing steam generator overfill and ADS actuation.

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The resulting offsite radiological doses for the limiting case analyzed are within the dose acceptance limits.

15.6.4 Spectrum of Boiling Water Reactor Steam System Piping Failures Outside of Containment

This section is not applicable to the AP1000.

15.6.5 Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the reactor coolant system pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered a Condition IV event (a limiting fault) because it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see Subsection 15.0.1).

A minor pipe break (small break), as considered in this subsection, is defined as a rupture of the reactor coolant pressure boundary (Section 5.2) with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event because it is an infrequent fault that may occur during the life of the plant.

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) and are summarized as follows:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- Localized cladding oxidation shall not exceed 17 percent of the total cladding thickness before oxidation.
- The amount of hydrogen generated from fuel element cladding reacting chemically with water or steam shall not exceed 1 percent of the total amount if all metal cladding were to react.
- The core remains amenable to cooling for any calculated change in core geometry.
- The core temperature is maintained at a low value, and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are established to provide significant margin in emergency core cooling system performance following a LOCA.

For the AP1000 plant, the small breaks (less than 1.0 ft^2) yield results with more margin than large breaks.

15.6.5.2 Basis and Methodology for LOCA Analyses

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer Low-2-pressure trip setpoint is reached. A safeguards actuation (“S”) signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

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- Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. Insertion of control rods to shut down the reactor is neglected in the large-break analysis.

- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

The acceptability of the computer codes approved for AP600 LOCA analyses for the AP1000 application is documented in Reference 24. The acceptability of additional computer codes for the AP1000 Best-Estimate Large-Break LOCA analysis is documented in Reference 34.

15.6.5.2.1 Description of Large-Break LOCA Transient

Before the break occurs, the unit is in an equilibrium condition in which the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay stored energy in the fuel, hot internals, and vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid, which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break, the core heat transfer is based upon local fluid conditions. Transition boiling and dispersed flow film boiling are the major heat transfer mechanisms.

The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending upon the relative temperatures. In the case of continued heat addition to the secondary system, secondary system pressure increases and the main steam safety valves may lift to limit the pressure. The safeguards actuation signal actuates a feedwater isolation signal, which isolates normal feedwater flow by closing the main feedwater isolation valves.

The reactor coolant pumps trip automatically during the accident following an “S” signal. The effects of pump coastdown are included in the blowdown. The blowdown phase of the transient ends when the reactor coolant system pressure (initially assumed at 2250 psia) falls to a value approaching that of the containment atmosphere.

When the “S” signal occurs, the core makeup tank isolation valves are opened. The core makeup tank begins to inject subcooled borated water into the reactor vessel through the direct vessel injection lines.

Subsection 15.6.5.4C presents calculations that show the effective post-LOCA long-term cooling of the AP1000 by passive means.

15.6.5.2.2 Description of Small-Break LOCA Transient

The detailed description of the small-break LOCA transients for the AP1000 plant design is provided in Subsection 15.6.5.4B.1. Calculations described in Subsection 15.6.5.4B indicate that acceptable core cooling is provided for the small-break LOCA transients. Subsection 15.6.5.4C calculations show that effective post-LOCA core cooling is provided in the long term by passive means.

15.6.5.3 Radiological Consequences

Although the analysis of the core response during a LOCA (see Subsection 15.6.5.4) shows that core integrity is maintained, for the evaluation of the radiological consequences of the accident, it is assumed that major core degradation and melting occur.

The dose calculations take into account the release of activity by way of the containment purge line prior to its isolation near the beginning of the accident and the release of activity resulting from

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containment leakage. Purge of the containment for hydrogen control is not an intended mode of operation and is not considered in the dose analysis. While the normal residual heat removal system is capable of post-LOCA cooling, it is not a safety-related system and may not be available following the accident. If it is operable, it would be used only if the source term is not far above the normal shutdown primary coolant source term. It is assumed that core cooling is accomplished by the passive core cooling system, which does not pass coolant outside of containment. Thus, there is no recirculation leakage release path to be modeled.

15.6.5.3.1 Source Term

The release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity.

15.6.5.3.1.1 Primary Coolant Release

The reactor coolant is assumed to have activity levels consistent with operation at the Technical Specification limits of 280 μCi/gm dose equivalent Xe-133 and 1.0 μCi/gm dose equivalent I-131.

Based on NUREG-1465 (Reference 19), for a plant using leak-before-break methodology, the release of coolant into the containment can be assumed to last for 10 minutes. The AP1000 is a leak-before-break plant, and the water in the reactor coolant system is assumed to blow down into the containment over a period of 10 minutes. The flow rate is assumed to be constant over the 10-minute period. As the reactor coolant enters the containment, the noble gases and half of the iodine activity are assumed to be released into the containment atmosphere.

15.6.5.3.1.2 Core Release

The release of activity from the fuel takes place in two stages as summarized in Table 15.6.5-1. First is the gap release which is assumed to occur at the end of the primary coolant release phase (i.e., at ten minutes into the accident) and continue over a period of half an hour. The second stage is that of the in-vessel core melt in which the bulk of the activity releases associated with the accident occur. The source term model is based on NUREG-1465 and Regulatory Guide 1.183 (Reference 20).

The core fission product inventory at the time of the accident is based on operation near the end of a fuel cycle at 101-percent power and is provided in Table 15A-3 of Appendix 15A. The main feedwater flow measurement supports a 1-percent power uncertainty. Consistent with NUREG-1465, there are three groups of nuclides considered in the gap activity releases: noble gases, iodines, and alkali metals (cesium and rubidium). For the core melt phase, there are five additional nuclide groups for a total of eight. The five additional nuclide groups are the tellurium group, the noble metals group, the cerium group, the lanthanide group, and barium and strontium. The specific nuclides included in the source term are as shown in Table 15A-3.

Gap Activity Release

Consistent with NUREG-1465 guidance for a plant using leak-before-break methodology, the gap release phase begins after the primary coolant release phase ends at ten minutes and has a duration of 0.5 hour.

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In-vessel Core Release

After the gap activity release phase, there is an in-vessel release phase which lasts for 1.3 hours and which releases activity to the containment due to core melting. The fractions of the core activity released to the containment atmosphere during this phase are from NUREG-1465:

Noble gases	0.95
Iodines	0.35
Alkali metals	0.25
Tellurium group	0.05
Noble metals	0.0025
Ba and Sr	0.02
Cerium group	0.0005
Lanthanide group	0.0002

Consistent with NUREG-1465, the releases are assumed to occur at a constant rate over the 1.3-hour phase duration.

15.6.5.3.1.3 Iodine Form

The iodine form is consistent with the NUREG-1465 model. The model shows the iodine to be predominantly in the form of nonvolatile cesium iodide with a small fraction existing as elemental iodine. Additionally, the model assumes that a portion of the elemental iodine reacts with organic materials in the containment to form organic iodine compounds. The resulting iodine species split is as follows:

- Particulate 0.95
- Elemental 0.0485
- Organic 0.0015

If the post-LOCA cooling solution has a pH of less than 6.0, part of the cesium iodide may be converted to the elemental iodine form. The passive core cooling system provides sufficient trisodium phosphate to the post-LOCA cooling solution to maintain the solution pH at 7.0 or greater following a LOCA (see Subsection 6.3.2.1.4).

15.6.5.3.2 In-containment Activity Removal Processes

The AP1000 does not include active systems for the removal of activity from the containment atmosphere. The containment atmosphere is depleted of elemental iodine and of particulates as a result of natural processes within the containment.

Elemental iodine is removed by deposition onto surfaces. Particulates are removed by sedimentation, diffusiophoresis (deposition driven by steam condensation), and thermophoresis (deposition driven by heat transfer). No removal of organic iodine is assumed. Appendix 15B provides a discussion of the models and assumptions used in calculating the removal coefficients.

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Particulates removed from the containment atmosphere to the containment shell are assumed to be washed off the shell by the flow of water resulting from condensing steam (i.e., condensate flow). The particulates may be either washed into the sump, which is controlled to a pH ≥7 post-accident or into the IRWST, which is not pH controlled post-accident. Due to the conditions in the IRWST, a portion of the particulate iodine washed into the IRWST may chemically convert to an elemental form and re-evolve, subject to partitioning, as airborne. A water-steam partition factor of 10 for elemental iodine is applied. This value bounds the time-dependent partition factors calculated using the NUREG/CR-5950 (Reference 36) models and the calculated IRWST water temperature and pH as a function of time.

The IRWST is a closed tank with weighted louvers, and without boiling, there would be no motive force for the release of re-evolved gaseous iodine from the IRWST gas space to the containment. Thus, the assumption of boiling in the IRWST liquid is imposed to force the release of the re-evolved iodine to the containment atmosphere. A portion (3%) of the re-evolved elemental iodine is assumed to convert to an organic form upon its release to containment.

15.6.5.3.3 Release Pathways

The release pathways are the containment purge line and containment leakage. The activity releases are assumed to be ground level releases.

During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. No credit is taken for the filters in the purge exhaust line.

The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate for the first 24 hours and at half that rate for the remainder of the analysis period.

15.6.5.3.4 Offsite Dose Calculation Models

The offsite dose calculation models are provided in Appendix 15A. The models address the determination of the TEDE doses from the combined acute doses and the committed effective dose equivalent doses.

The exclusion area boundary dose is calculated for the 2-hour period over which the highest doses would be accrued by an individual located at the exclusion area boundary. Because of the delays associated with the core damage for this accident, the first 2 hours of the accident are not the worst 2-hour interval for accumulating a dose.

The low population zone boundary dose is calculated for the nominal 30-day duration of the accident.

For both the exclusion area boundary and low population zone dose determinations, the calculated doses are compared to the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

15.6.5.3.5 Main Control Room Dose Model

There are two approaches used for modeling the activity entering the main control room. If power is available, the normal heating, ventilation, and air-conditioning (HVAC) system will switch over to a supplemental filtration mode (Section 9.4). The normal HVAC system is not a safety-class system but provides defense in depth.

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control room, the

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emergency habitability system (Section 6.4) would be actuated when High-2 iodine or particulate activity is detected. The emergency habitability system provides passive pressurization of the main control room from a bottled air supply to prevent inleakage of contaminated air to the main control room. The bottled air also induces flow through the passive air filtration system which filters contaminated air in the main control room. There is a 72-hour supply of air in the emergency habitability system. After this time, the main control room is assumed to be opened and unfiltered air is drawn into the main control room by way of an ancillary fan. After 7 days, offsite support is assumed to be available to reestablish operability of the control room habitability system by replenishing the compressed air supply. As a defense-in-depth measure, the nonsafety-related normal control room HVAC would be brought back into operation with the supplemental filtration train if power is available.

The main control room is accessed by a vestibule entrance, which restricts the volume of contaminated air that can enter the main control room from ingress and egress. The design of the emergency habitability system (VES) provides 65 scfm ±5 scfm to the control room and maintains it in a pressurized state. The path for the purge flow out of the main control room is through the vestibule entrance and this should result in a dilution of the activity in the vestibule and a reduction in the amount of activity that might enter the main control room. However, no additional credit is taken for dilution of the vestibule via the purge. The projected inleakage into the main control room through ingress/egress is 5 cfm. An additional 10 cfm of unfiltered inleakage is conservatively assumed from other sources.

Activity entering the main control room is assumed to be uniformly dispersed. With the VES in operation, airborne activity is removed from the main control room atmosphere via the passive recirculation filtration portion of the VES.

The main control room dose calculation models are provided in Appendix 15A for the determination of doses resulting from activity which enters the main control room envelope.

15.6.5.3.6 Analytical Assumptions and Parameters

The analytical assumptions and parameters used in the radiological consequences analysis are listed in Table 15.6.5-2.

15.6.5.3.7 Identification of Conservatisms

The LOCA radiological consequences analysis assumptions include a number of conservatisms. Some of these conservatisms are discussed in the following subsections.

15.6.5.3.7.1 Primary Coolant Source Term

The source term is based on conservative assumptions whereas, the activities based on the expected fuel defect level are far less.

15.6.5.3.7.2 Core Release Source Term

The assumed core melt is a major conservatism associated with the analysis. In the event of a postulated LOCA, no major core damage is expected. Release of activity from the core is limited to a fraction of the core gap activity.

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15.6.5.3.7.3 Atmospheric Dispersion Factors

The atmospheric dispersion factors assumed to be present during the course of the accident are conservatively selected. Actual meteorological conditions are expected to result in significantly higher dispersion of the released activity.

15.6.5.3.8 LOCA Doses

15.6.5.3.8.1 Offsite Doses

The doses calculated for the exclusion area boundary and the low population zone boundary are listed in Table 15.6.5-3. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The reported exclusion area boundary doses are for the time period of 1.3 to 3.3 hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

At the time the LOCA occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after the limiting 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.3-3. Also listed on Table 15.6.3-3 are the doses due to direct shine from the activity in the adjacent buildings, shine from radioactivity accumulated on the VES or VBS filters, and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of these dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and is reported in Table 15.6.3-3.

15.6.5.4 Core and System Performance

Subsection 15.6.5.4A describes the large-break LOCA analysis methodology and results.
Subsection 15.6.5.4B describes the small-break LOCA analysis methodology and results.

15.6.5.4A Large-Break LOCA Analysis Methodology and Results

Westinghouse applies the WCOBRA/TRAC computer code to perform best-estimate large-break LOCA analyses in compliance with 10 CFR 50 (Reference 5). WCOBRA/TRAC is a thermal-hydraulic computer code that calculates realistic fluid conditions in a PWR during the blowdown and reflood of a postulated large-break LOCA. The methodology used for the AP1000 analysis is documented in WCAP-12945-P-A, WCAP-14171, Revision 2, WCAP-16009-P-A (References 10, 11, 32), and Reference 31.

The NRC staff has reviewed and approved the ASTRUM best-estimate LOCA methodology (ASTRUM methodology), as documented in the SER attached in front of Reference 32 for estimating

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the 95th percentile PCT for two-loop, three-loop and four-loop Westinghouse pressurized water reactors (PWRs) and the AP600. Application of the ASTRUM methodology for the AP1000 plant was submitted to the NRC staff per Reference 34. The NRC staff has reviewed and approved the ASTRUM methodology for estimating the 95th percentile PCT for the AP1000 plant, as documented in Reference 35. In the ASTRUM methodology, the WCOBRA/TRAC code is used to calculate the effects of initial conditions, power distributions, and global models, and the HOTSPOT code is used to calculate the effects of local models.

In the ASTRUM uncertainty methodology (Reference 32), as used in the AP1000 LB LOCA analysis, global models and initial-condition, power-distribution, and local uncertainties are sampled independently for each of 124 runs over the same ranges of uncertainty and distributions as in References 10, 32, and 33, as described in References 34 and 31. The sampled global models, initial conditions, and power-distribution uncertainties become inputs to each of the WCOBRA/ TRAC calculations. The thermal-hydraulic boundary conditions for the hot rod are input to the local uncertainties calculation performed by the HOTSPOT code.

Results from the calculations are ranked by PCT from highest to lowest. A similar procedure is repeated for maximum local oxidation (MLO) and core wide oxidation (CWO). In order statistics as applied in the ASTRUM methodology, the limiting case for a parameter, such as peak cladding temperature (PCT), is a conservative estimate of the 95th percentile with 95 percent confidence. The limiting PCT, limiting MLO, and CWO may come from the same case or as many as three different cases because each parameter is assumed to be independent of the other two. The assumption of independence of the calculated licensing parameters is a conservative assumption because there is a dependence of MLO and CWO on cladding temperature.

For the AP1000 large-break LOCA analysis, a plant-specific adaptation of the ASTRUM methodology is applied as described in Reference 31. The plant-specific adaptation explicitly models the effects of thermal conductivity degradation and peaking factor burndown. The best-estimate large-break LOCA analysis complies with the stipulated applicability limits in the Reference 3, Reference 32, and Reference 35 approvals. The post-LOCA long-term core cooling and core boron concentration analyses discussed in Subsection 15.6.5.4C are applicable to the large-break LOCA transient.

15.6.5.4A.1 General Description of WCOBRA/TRAC Modeling

WCOBRA/TRAC is the best-estimate thermal-hydraulic computer code used to calculate realistic fluid conditions in the PWR during blowdown and reflood of a postulated large-break LOCA.

The WCOBRA/TRAC Code Qualification Document (Reference 10) contains a complete description of the code models and justifies their applicability to PWR large-break LOCA analysis.

Table 15.6.5-4 lists AP1000-specific parameters identified for use in the large-break LOCA analysis. WCOBRA/TRAC studies were performed for AP1000 to establish sensitivities to parameter variations. These studies included effects of ranging steam generator tube plugging, ranging the relative power in the low-power assemblies, loss of offsite power coincident with the break initiation, and break location. The calculated results were used to identify bounding conditions, which are then used in the AP1000 uncertainty calculations.

The WCOBRA/TRAC vessel nodalization is developed from plant design drawings to divide the vessel into 10 vertical sections. The bottom of section 1 is the inside vessel bottom, and the top of section 10 is the inside top of the vessel upper head. In addition to the major downcomer and core flow paths, the modeled bypass flow paths are the upper head cooling spray, guide thimbles, and core bypass. After defining the elevations for each section, a noding scheme is defined for the WCOBRA/TRAC model as shown in Reference 34. WCOBRA/TRAC assumes a vertical flow path for vertically stacked channels, unless specified otherwise in the input. Positive flow for the

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vertically connected channels (and cells) is upward. Several of the 10 sections are divided vertically into 2 or more levels; these levels are referred to as cells within a channel.

The WCOBRA/TRAC loop model represents the major primary, secondary, and passive safety systems components. Both loops are explicitly modeled, including the hot leg, the steam generator, and the two cold legs and associated pumps. The loop designated “1” has the pressurizer and the PRHR system connections, and loop “2” cold legs have the core makeup tank pressure balance line connections. The reactor coolant pump models contain the AP1000 homologous curves together with appropriate two-phase head and torque multipliers and degradation data. AP1000 values for pump coastdown characteristics are also applied. The passive safety features are modeled using design data for elevations, liquid volumes, and line losses. Because the ADS is not actuated until long after the time of PCT in large-break LOCA events, it is not modeled in detail.

15.6.5.4A.2 Steady-State Calculation

A WCOBRA/TRAC LOCA calculation is initiated from a point at which the flows, temperatures, powers, and pressures are at their approximate steady-state values before the postulated break occurs. Steady-state WCOBRA/TRAC calculations are run for a brief time period to verify that the calculated conditions are steady and that the desired reactor conditions are achieved.

The values used to set the steady-state plant conditions reflect the AP1000 parameters for reactor coolant pump flows, core power, and steam generator tube plugging levels. The fuel parameters provide the steady-state fuel temperatures, pressures, and gap conductances as a function of fuel burnup and linear power, accounting for the effects of thermal conductivity degradation as described in Reference 31. The calculated fuel temperatures from WCOBRA/TRAC are adjusted to match the specified fuel data by adjusting the gap heat transfer coefficient between the pellet and the cladding. Once the vessel fluid temperatures, flows, pressures, loop pressure drop, and core parameters are in agreement with the desired values and are steady, a suitable initial condition is achieved.

15.6.5.4A.3 Signal Logic for Large-Break LOCA

The reactor trip signal occurs due to compensated pressurizer pressure within the first seconds of the large-break transient, however control rod insertion is not modeled in WCOBRA/TRAC and no effects of control rod insertion on reactivity ensue. A safeguards “S” signal occurs due to containment High-2 pressure of 6.7 psig at 2.2 seconds of large-break LOCA transients.

As a consequence of this signal, after appropriate delays, the PRHR and core makeup tank isolation valves open, containment isolation occurs, and the reactor coolant pump automatic trip timer begins. The rapid depressurization of the primary system during a large-break LOCA leads to the initiation of accumulator injection early in the large-break transient. The accumulator flow diminishes core makeup tank delivery to such an extent that the core makeup tank level does not approach the ADS Stage 1 valve actuation point until after the accumulator tank is empty. The accumulator empties long after the blowdown portion of the large-break LOCA transient is complete. Actuation of the ADS on CMT water level does not occur until long after the AP1000 PCT is calculated to occur.

15.6.5.4A.4 Transient Calculation

Once the steady-state calculation is found to be acceptable, the transient calculation is initiated. The semi-implicit pipe break model is added to the desired break location. Cold-leg breaks are analyzed because the hot-leg break location is nonlimiting in the large-break LOCA best-estimate methodology. The break size and type are sampled consistent with the WCAP-16009-P-A (Reference 32) methodology. The containment backpressure is specified consistent with WCAP-16009-P-A (Reference 32) methodology. The steady-state calculation is restarted with the above changes to begin the transient.

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Table 15.6.5-5 shows a general sequence of events following a large cold-leg break LOCA and the relationship of these events to the blowdown and reflood portion of the transient.

15.6.5.4A.5 Large-Break LOCA Analysis Results

For the AP1000 large-break LOCA analysis, a plant-specific adaptation of the ASTRUM best-estimate LOCA analysis methodology is applied, as described in Reference 31. The AP1000 large-break LOCA analysis complies with the restrictions in Reference 3, Reference 32, and Reference 35. AP1000 sensitivity calculations evaluated the sensitivity to the modeling of the CMT and PRHR relative to the reference transient configuration. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the reference transient configuration. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was 2°F higher than the PCT of the reference transient configuration. The ASTRUM methodology samples the parameters ranged in the global model matrix of calculations, and the final 95 percent uncertainty calculations have been performed for AP1000. Further, local and core-wide cladding oxidation values have been determined using the plant-specific adaption of the approved Reference 32 methodology as described in Reference 31.

In the AP1000 ASTRUM analysis, the limiting PCT and limiting MLO results were from two different uncertainty calculations. Both the limiting PCT case and the limiting MLO case were double ended guillotine breaks. Figures 15.6.5.4A-1 through 15.6.5.4A-12 present the parameters of principal interest for the limiting PCT case. Values of the following parameters are presented:

- Highest calculated cladding temperature at any elevation for the five fuel rods modeled
- Hot rod cladding temperature transient at the limiting elevation for PCT
- Core fluid mass flows at the top of the core for the fuel assemblies modeled in WCOBRA/TRAC
- Pressurizer pressure
- Break flow rates
- Core and downcomer collapsed liquid levels
- Accumulator water flow rates
- Core makeup tank flow rates

15.6.5.4A.6 Description of AP1000 Large-Break LOCA Transient

A description of the limiting PCT case from the AP1000 ASTRUM analysis follows. The limiting PCT case is a double-ended guillotine break. The sequence of events is presented in Table 15.6.5-6. The break was modeled to occur in one of the cold legs in the loop containing the core makeup tanks. After the break opens, the vessel rapidly depressurizes and the core flow quickly reverses. The hot assembly fuel rods dry out and begin to heat up (Figures 15.6.5.4A-1 and 15.6.5.4A-2) after the initial flow reversal (Figure 15.6.5.4A-3).

In Figure 15.6.5.4A-1, “Hot Rod” refers to the hot fuel rod at the maximum linear heat rate for the run, “Hot Assembly” refers to the average fuel rod in the hot assembly that contains the hot rod, “Support Column/Open Hole” refers to the fuel rod in average assemblies under support columns or open holes, “Guide Tubes” refers to the fuel rod in average assemblies under guide tubes, and “Low Power” refers to the fuel rod in the low power peripheral fuel assemblies.

The steam generator secondaries are assumed to be isolated immediately at the inception of the break, which maximizes their stored energy. The massive size of the break causes an immediate, rapid pressurization of the containment. At 2.2 seconds, an “S” signal is generated due to High-2 containment pressure. Applying the pertinent signal processing delay means that the valves isolating the core makeup tanks from the direct vessel injection line and the PRHR begin to open at 4.2 seconds into the transient. The reactor coolant pumps automatically trip after a 5.3 second delay

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from the actuation of the core makeup tank isolation valves, which is 9.5 seconds into the transient. Core shutdown occurs due to voiding; no credit is taken for the control rod insertion effect.

The system depressurizes rapidly (Figure 15.6.5.4A-4) as the initial mass inventory is depleted due to break flow. The pressurizer drains completely approximately 30 seconds into the transient, and accumulator injection commences 13 seconds into the transient (Figure 15.6.5.4A-5). Accumulator actuation shuts off core makeup tank flow (Figure 15.6.5.4A-6), which has been occurring since the isolation valve opened. The CMT liquid level remains well above the ADS Stage 1 actuation setpoint throughout the AP1000 LBLOCA cladding temperature excursion, even though CMT injection begins again around 200 seconds.

The dynamics of the 95th percentile estimator PCT case are shown in terms of the flow rates of liquid, vapor, and entrained liquid at the top of the core (Figures 15.6.5.4A-7 through 15.6.5.4A-9) for the peripheral, open hole/support column average power interior, and guide tube average power interior assemblies (the corresponding figure for the hot assembly is Figure 15.6.5.4A-3).

Figure 15.6.5.4A-7 demonstrates that liquid downflow exists through the top of the peripheral core assemblies from approximately 1 to 3 seconds and again from 9 to 20 seconds in the 95th percentile estimator PCT case. The power of the fuel in this region is significantly lower than that of the fuel in the open hole/support column and guide tube locations (Table 15.6.5-4), so liquid downflow occurs earlier on the periphery than in the average power assemblies. Once the upper head begins to flash, liquid drains directly down the guide tubes and that fraction that is able to penetrate into the core does so, at a maximum flow rate exceeding 1000 lbm/sec of total liquid flow between 5-23 seconds (Figure 15.6.5.4A-8).

Figure 15.6.5.4A-9 presents the open hole/support column assembly top of core flow behavior. In this case, liquid downflow into the support column/open hole assemblies is delayed relative to downflow into the guide tubes; there is continuous liquid flow from approximately 10 seconds until 22 seconds; the entrained liquid flow continues to be significant until 28 seconds as fluid drains through the upper core plate holes into the upper plenum.

The timing of the initial downflow into the hot assembly is similar to that of the downflow into the open hole/support column average assemblies. Around 10 seconds into the transient, liquid that has built up in the global region above the hot assembly begins to flow into the hot assembly (Figure 15.6.5.4A-3). Significant flow of continuous liquid into the hot assembly exists between 10 to 20 seconds. The liquid flow is not enough to quench the hot rod and hot assembly rod or the average rods at all elevations (Figure 15.6.5.4A-1) although some cooling is achieved.

After 13 seconds into the transient, the accumulator begins to inject water into the upper downcomer region, most of which is initially bypassed to the break. The break flow rate diminishes as the transient progresses (Figure 15.6.5.4A-10). At 27.5 seconds, the accumulator injection begins to refill the lower plenum. At approximately 40.0 seconds, the lower plenum fills to the point that water begins to reflood the core from below (Figure 15.6.5.4A-11). The void fraction at the core bottom begins to decrease, and as time passes, core cooling increases substantially. Figure 15.6.5.4A-11 presents the collapsed liquid levels in the core; Figure 15.6.5.4A-12 presents the collapsed liquid levels in the downcomer. The cladding temperature begins to decrease once the core water level has risen high enough in the core.

15.6.5.4A.7 Global Model Sensitivity Studies and Uncertainty Evaluation

Subsection 15.6.5.4A discusses the treatment of the global model parameters and the uncertainty evaluation in the ASTRUM methodology.

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15.6.5.4A.8 Large-Break LOCA Conclusions

In accordance with 10 CFR 50.46, the conclusions of the best-estimate large-break LOCA analysis are that there is a high level probability that the following criteria are met.

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature (PCT)) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

   Note that criterion 4 has historically been satisfied by adherence to criteria 1 and 2, and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. Criteria 1 and 2 are satisfied for best-estimate large-break LOCA applications. The approved methodology specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low power channel as defined in the WCOBRA/TRAC model. This situation has not been calculated to occur for the AP1000. Therefore, acceptance criterion 4 is satisfied.

5. After successful initial operation of the emergency core cooling system (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

   Criterion 5 is satisfied if a coolable core geometry is maintained and the core is cooled continuously following the LOCA. The AP1000 passive core cooling system provides effective core cooling following a large-break LOCA event, even assuming the limiting single failure of a core makeup tank delivery line isolation valve. The large-break LOCA transient has been extended beyond fuel rod quench to the time at which the CMT liquid level has decreased to the setpoint that actuates the fourth-stage ADS valves and IRWST injection. A significant increase in safety injection flow rate occurs when the IRWST becomes active. The analysis performed demonstrates that CMT injection is sufficient to maintain the mass inventory in the core and downcomer, from the period of fuel rod quench until IRWST injection. The AP1000 passive core cooling system provides effective post-LOCA long-term core cooling (Subsection 15.6.5.4C).

Table 15.6.5-8 presents the calculated 95th percentile PCT, maximum cladding oxidation, maximum hydrogen generation, and core cooling results.

Based on the analysis, the Westinghouse Best-Estimate Large-Break LOCA methodology has shown that the acceptance criteria of 10 CFR 50.46 are satisfied for AP1000 when the burnup-related effects of thermal conductivity degradation and peaking factor burndown are considered.

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15.6.5.4B Small-Break LOCA Analyses

Should a small-break LOCA occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal occurs when the pressurizer Low-2 pressure trip setpoint is reached. An “S” signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip leads to a rapid reduction of power to a residual level corresponding to fission product decay heat by the insertion of control rods to shut down the reactor.
- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

15.6.5.4B.1 Description of Small-Break LOCA Transient

The AP1000 plant design includes passive safety features to prevent or minimize core uncovery during small-break LOCAs. The passive safety design approach is to depressurize the reactor coolant system if the break or leak is greater than the capability of the makeup system or if the non-safety makeup system fails to perform. By depressurizing the reactor coolant system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core. The small-break analyses demonstrate that, with a single failure of one of the ADS Stage 4 valves located off the non-pressurizer loop, the passive systems are capable of depressurizing the reactor coolant system while maintaining acceptable core conditions and establishing stable delivery of cooling water from the IRWST.

During a small-break LOCA, the reactor coolant system depressurizes to the pressurizer Low-2 pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an “S” signal when the pressurizer Low-3 pressure setpoint is reached. The passive core cooling system includes two core makeup tanks, two accumulators, a large IRWST, and the PRHR heat exchanger.

The core makeup tanks operate at reactor coolant system pressure. They provide high-pressure safety injection in the event of a small-break LOCA and are filled with borated water to provide core shutdown margin. The core makeup tanks share a common discharge line with the accumulators and IRWST. Gravity head of the colder water in the core makeup tanks provides the injection of the core makeup tanks. The core makeup tanks are located above the reactor coolant loops, and each is equipped with a pressure balancing line from a cold leg to the top of the tank.

The pressurized accumulators provide additional borated water to the reactor coolant system in the event of a LOCA. Nominally, these 2020 ft^3 tanks are filled with 1700 ft^3 of water and nitrogen cover gas at an initial pressure of 700 psig. Once sufficient reactor coolant system depressurization occurs, either as a result of a LOCA or the actuation of the ADS, accumulator injection begins.

The IRWST provides an additional source of water for core cooling. To attain injection from the IRWST, the reactor coolant system pressure must be lowered to approximately 13 psi above containment pressure. For this pressure to be achieved during a small-break LOCA, the actuation of the ADS valves is required.

The ADS consists of a series of valves, connected to the pressurizer and hot legs, which provide a phased depressurization of the reactor coolant system. As the reactor coolant system loses inventory through the break, the core makeup tanks provide flow to the reactor vessel. When the inventory in one core makeup tank drops to the 67.5 percent volume, the ADS valves begin opening to accelerate the reactor coolant system depressurization rate. The ADS Stage 1 valves open at the 67.5 percent volume setpoint; the Stage 2 and the Stage 3 valves open in a timed sequence thereafter. The flow

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from the first three stages of the ADS is discharged into the IRWST through a sparger system. The fourth stages of the ADS are connected to the reactor coolant system hot legs and discharge to containment atmosphere. The ADS Stage 4 valves are activated when one core makeup tank reaches the 20 percent volume and a delay has elapsed after the ADS Stage 3 actuation.

As the reactor system depressurizes and mass is lost out of the break, mass is added to the reactor vessel from the core makeup tanks and the accumulators. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state. Calculations described in this section indicate that acceptable core cooling is provided for the small-break LOCA transients.

15.6.5.4B.2 Small-Break LOCA Analysis Methodology

Small-break LOCA response is evaluated for the AP1000 plant with an evaluation model that conforms to 10 CFR 50 Appendix K (Reference 1). The elements of the AP1000 small-break LOCA evaluation model are the following:

- NOTRUMP computer code
- NOTRUMP homogeneous sensitivity model
- Critical heat flux assessment during accumulator injection
- SBLOCTA computer code

15.6.5.4B.2.1 NOTRUMP Computer Code

The NOTRUMP computer code is used in the analysis of LOCAs due to small-breaks in the reactor coolant system. The NOTRUMP computer code is a one-dimensional, general network code, which includes a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The version of NOTRUMP used in AP1000 small-break LOCA calculations has been validated against applicable passive plant test data (References 22 and 24). The code has limited capability in modeling upper plenum (UP) and hot leg entrainment and did not predict the core collapsed level during the accumulator injection phase adequately. The NOTRUMP homogeneous sensitivity model (discussed in Subsection 15.6.5.4B.2.2) and the critical heat flux assessment during the accumulator injection phase (discussed in Subsection 15.6.5.4B.2.3) supplement the base NOTRUMP analysis to demonstrate the adequacy of the design.

In NOTRUMP, the reactor coolant system is nodalized into volumes interconnected by flow paths. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A description of NOTRUMP is given in References 12 and 13. The applicability of NOTRUMP to AP600 is documented in Reference 26. The AP600 modeling approach, described in Reference 17, is also used to develop the AP1000 model; the applicability of the NOTRUMP code to the AP1000 plant is documented in Reference 24.

The use of NOTRUMP in the analysis involves the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables an explicit and detailed spatial representation of various system components. Table 15.6.5-9 lists important input parameters and initial conditions of the analysis.

A steady-state input deck was set up to comply, where appropriate, with the standard small-break LOCA Evaluation Model methodology. Major features of the modeling of the AP1000 plant follow:

- Accumulators are modeled at an initial pressure of 715 psia.

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- The flow through the ADS links is modeled using the Henry-Fauske and the homogeneous equilibrium (HEM) critical flow models. The Henry-Fauske subcooled correlation is used for subcooled flow, and the HEM correlation is used for superheated flow. For saturated donor conditions above the transition static quality of 10 percent, the HEM correlation is used. Below the transition quality, the flow is transitioned between the HEM correlation and the Henry-Fauske saturated correlation.

- Isolation and check valves used in the passive safety systems are modeled.

- The IRWST is modeled as two connected fluid nodes. The lower node is connected to the direct vessel injection line and is the source of injection water to the direct vessel injection (DVI) lines driven by gravity head. The upper node acts as a sink for the ADS flow from the pressurizer and as a heat sink for the PRHR heat exchanger. These nodes are modeled as having an initial temperature of 120°F, a pressure reflective of transient-specific containment conditions (described in a subsequent item), and the minimum full-power operation level of 28.03 feet. Therefore, the minimum head for IRWST injection is assumed.

- The PRHR system is modeled in accordance with the guidance provided in References 22 and 24. The PRHR isolation valve is modeled as opening with the maximum delay after the generation of an “S” signal to conservatively deny the cooling capability of the heat exchanger to the reactor coolant system for an extended period.

- The core power is initially set to 101 percent of the nominal core power. The reactor trip signal occurs when the pressurizer pressure falls below 1800 psia. A conservative delay time is modeled between the reactor trip signal and reactor trip. Decay heat is modeled according to the ANS-1971 (Reference 2) standard, with 20 percent uncertainty added consistent with Reference 1.

- The “S” signal is generated when the pressurizer pressure falls below 1700 psia. The isolation valves on the core makeup tank injection lines begin to open after the signal setpoint is reached; the valves are then assumed to open linearly. The main feedwater isolation valves are ramped closed between 2 and 7 seconds after the “S” signal. The reactor coolant pumps are tripped 7 seconds after the “S” signal.

- The ADS actuation signals are generated on low core makeup tank levels and the ADS timer delays. A list of the ADS parameters is given in Table 15.6.5-10. Note that the small-break LOCA transients model the minimum valve flow area given in Table 15.6.5-10. The ADS Stages 1, 2, and 3 are modeled as discharging through spargers submerged in the IRWST at the appropriate depth.

- The inadvertent ADS actuation and 2-inch cold leg break NOTRUMP simulations utilize a time-dependent containment pressure in the boundary node modeling of the containment. The containment conditions were generated by providing mass and energy releases from these breaks to the AP1000 WGOTHIC minimum containment pressure model (described in Section 13.8 of Reference 6). The WGOTHIC code then calculates the containment pressure response. The inadvertent ADS actuation and 2-inch cold leg break NOTRUMP simulations then utilized the time-dependent pressure history curves as generated by WGOTHIC. For the double-ended DVI (DEDVI) line break simulations, a conservatively low 20 psia containment pressure was used based on containment pressurization calculations performed with the WGOTHIC containment model. The 10-inch cold leg break case models a constant containment pressure of 14.7 psia.

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- The steam generator secondary is isolated immediately after the reactor trip signal, due to closure of the turbine stop valves. The main steam safety valves actuate and remove energy from the steam generator secondary when pressure reaches approximately 1210 psia.

Active single failures of the passive safeguards systems are considered. The limiting failure is judged to be one out of four ADS Stage 4 valves failing to open on demand, which is the failure that most severely impacts depressurization capability. The safety design approach of the AP1000 plant is to depressurize the reactor coolant system to the containment pressure in an orderly fashion such that the large reservoir of water stored in the IRWST is available for core cooling. The mass inventory plots provided for the breaks show the minimum inventory condition generally occurs at the start of IRWST injection. Penalizing the depressurization is the most conservative approach in postulating the single failure for such breaks.

The analyzed small-break LOCA break spectrum includes breaks that exhibit a minimum reactor vessel inventory early in the transient, before the ADS Stage 4 depressurization phase: the DEDVI line break at 20 psia containment backpressure and 10-inch cold leg break transients. In these transients, the early mass inventory decrease is terminated by injection flow from the accumulators. For consistency and because the transition to IRWST injection is still a tenuous period of the transients, the conservative failure of one of the ADS Stage 4 valves located off of the non-pressurizer loop is assumed in all cases.

15.6.5.4B.2.1.1 AP1000 Model-Detailed Noding

The AP1000 NOTRUMP model was developed in the same fashion as the AP600 NOTRUMP model (details of the AP600 NOTRUMP model are provided in Reference 17). Modifications of the AP600 model to create the AP1000 model include the following:

- The addition of two core nodes one foot each in length to reflect the added active fuel length of this design.

- The ADS Stage 4 (ADS-4) flow path resistances were increased to accommodate shortcomings in NOTRUMP identified during the integral test facility simulations; namely, the lack of a detailed momentum flux model in the ADS-4 discharge paths. A detailed calculation of the energy and momentum equations is performed for the ADS-4 piping over a range of flow and pressure conditions to provide a benchmark for the NOTRUMP ADS-4 flow path resistance. The methodology used to determine the resistance increase is described in Reference 24. By increasing the ADS-4 resistances, the onset of IRWST injection is more appropriately calculated. This methodology directly addresses the effect of momentum flux in ADS-4. The ADS-4 resistance increase utilized is computed for the NOTRUMP analyses in this section to be a 78 percent ADS-4 flow path resistance increase.

15.6.5.4B.2.1.2 Plant Initial Conditions/Steady-State

Steady-state calculations are performed prior to initiating the transient portion of the calculations.

Table 15.6.5-9 contains the most important initial conditions for the transient calculations. The behaviors of the primary pressure and pressurizer level, steam generator pressures, and the core flow rate are stable at the end of the 100-second steady-state calculation.

15.6.5.4B.2.2 NOTRUMP Homogeneous Sensitivity Model

In order to address the uncertainties associated with entrainment in the upper plenum and hot leg following ADS-4 operation, a sensitivity study is performed with the limiting break with respect to these phenomena, effectively maximizing the amount of entrainment downstream of the core. This

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methodology is described and the original results are presented for the double-ended direct vessel injection (DEDVI) line break in detail in Reference 24.

[*In order to maximize the entrainment downstream of the core for the limiting break with respect to entrainment, NOTRUMP is run with the regions of the upper plenum, hot leg, and ADS-4 lines in a homogeneous fluid condition, with slip = 1, to demonstrate that even with maximum entrainment, the 10 CFR 50.46 criteria are met.*]*

15.6.5.4B.2.3 Critical Heat Flux Assessment During Accumulator Injection

[*An assessment is performed of the peak core heat flux with respect to the critical heat flux during the later ADS depressurization time period for a double-ended rupture of the direct vessel injection line. This time period corresponds to the accumulator injection phase of the transient. The predicted average mass flux at the core inlet and the reactor pressure from the NOTRUMP computer code base model analysis are used as input parameters to critical heat flux correlation as described in Reference 30. The requirements of 10 CFR 50.46 are met provided the maximum heat flux is less than the critical heat flux calculated by the correlation.*]* NOTRUMP has been shown (Reference 24) to adequately predict mass flux and pressure for integral systems tests.

The predicted mass flux at the core inlet is on the average constant and corresponds to 7.2 lbm ft^{-2} s^{-1} (~35 kg m^{-2} s^{-1}). The key thermal-hydraulic parameters at different times during the ADS depressurization time period are summarized in following table.

Time (sec)	UP Pressure (kPa)	UP Pressure (psia)	Mass Flux (kg/m^2s)	Hot Channel Heat Flux (kW/m^2)
400.5	1300	188.6	35	48.6
450.6	1018	147.6	35	47.2
500.7	643	93.2	35	46.0
569.9	343	49.8	35	44.5

For the critical heat flux assessment, the peak core heat flux is applied to simulate the hot channel condition in a conservative manner. No credit is taken for increased flow in the hot channel that is known to occur in rod bundles.

The correlation applied for this assessment is from vertical tube data (Reference 30) and recognizes two regimes depending on the mass flux. The main difference between the two is the mass flux dependence. The equations for the two regimes are as follows:

$$q^*_{CL} = q^*_{CF} + 0.01351\left(D^*\right)^{-0.473}\left(L/D\right)^{-0.533}\left|G^*\right|^{1.45} \quad \text{for low G*}$$

and,

$$q^*_{CH} = q^*_{CF} + 0.05664\left(D^*\right)^{-0.247}\left(L/D\right)^{-0.501}\left|G^*\right|^{0.77} \quad \text{for high G*}$$

The first term of above correlations is,

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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$$q^*_{CF} = 1.61\left(\frac{A}{Ah}\right)\frac{\left(D^*\right)^{0.5}}{\left[1+\left(\frac{\rho_g}{\rho_l}\right)^{0.25}\right]^2}$$

where A is the flow area and Ah is the heated area.

The dimensionless CHF is calculated as,

$$q^*_{CHF} = \min(q^*_{CL}, q^*_{CH})$$

Dimensionless CHF, G, and D are defined as,

$$q^*_{CHF} = \frac{q''_{CHF}}{h_{fg}\sqrt{\lambda\rho_g g\Delta\rho}}$$

$$G^* = \frac{G}{\sqrt{\lambda\rho_g g\Delta\rho}}$$

$$D^* = \frac{D}{\lambda}$$

where λ is the length scale of the Taylor instability:

$$\lambda = \sqrt{\frac{\sigma}{g\Delta\rho}}$$

Conservative application of this correlation with the AP1000 parameters indicates that the peak AP1000 heat flux during this period is approximately 30 percent or more below the predicted critical heat flux.

This CHF assessment addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 are met during this period since this CHF assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

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15.6.5.4B.2.4 SBLOCTA Computer Code

The LOCTA-IV computer code (Reference 4) was modified as described in Reference 13 to form SBLOCTA, a small-break LOCA specific version of the LOCTA-IV code. The SBLOCTA code calculates the cladding temperature and oxidation transients for the hot rod and hot assembly average rod, which represent the highest power rod and the average of the highest power fuel assembly in the core. Peak cladding temperature (PCT) calculations are performed with the SBLOCTA code using boundary conditions from the NOTRUMP calculation. In addition to PCT, SBLOCTA also calculates the maximum local and axial average zirconium-water oxidation reaction based on the Baker-Just oxidation model. In the event that the NOTRUMP code predicts core uncovery in the core average channel, the NOTRUMP boundary conditions will be transferred to the SBLOCTA code to perform fuel rod heat-up calculations.

15.6.5.4B.3 Small-Break LOCA Analysis Results

Several small-break LOCA transients are analyzed using NOTRUMP, and the results of these calculations are presented. The transients documented herein analyze a single failure of one ADS Stage 4 valve on the non-pressurizer side. The results demonstrate that the minimum reactor vessel mixture mass inventory condition occurs for the relatively small system pipe breaks. Larger breaks exhibit a greater margin-to-core uncovery.

15.6.5.4B.3.1 Introduction

The small-break LOCA safety design approach for the AP1000 plant is to provide for a controlled depressurization of the primary system if the break cannot be terminated, or if the nonsafety-related charging system is postulated to be lost or cannot maintain acceptable plant conditions. Nonsafety-related systems are not modeled in this design basis analysis; the testing conducted in the SPES-2 facility has indicated that the mass inventory condition during small LOCAs is significantly improved when these nonsafety-related systems operate. The core makeup tank level activates primary system depressurization through the actuation of the automatic depressurization system. The core makeup tank provides makeup to help compensate for the postulated break in the reactor coolant system. As the core makeup tank level drops, Stages 1 through 4 of the ADS valves are ramped open in sequence. The ADS valve parameters are presented in Table 15.6.5-10. Note that the small-break LOCA transients model the minimum valve flow area given in Table 15.6.5-10. The reactor coolant system depressurizes due to the break and the ADS valves, while subcooled water from the core makeup tanks and accumulators enters the reactor vessel downcomer to maintain system inventory. Design basis maximum values of passive core cooling system resistances are applied to obtain a conservative prediction of system behavior during the small LOCA events.

During controlled depressurization via the ADS, the accumulators and core makeup tanks maintain system inventory for small-break LOCAs. Once the reactor coolant system depressurizes, injection from the IRWST maintains core cooling. For continued injection from the IRWST, the reactor coolant system must remain depressurized. To conservatively model this condition, design maximum resistance values are specified for the IRWST delivery lines.

A series of small-break LOCA calculations are performed to assess the passive core cooling safety system design performance. In these calculations, the decay heat used is the ANS-1971 standard (Reference 2) plus 20 percent for uncertainty as specified in 10 CFR 50, Appendix K (Reference 1). This maximizes the core steam generation to be vented. The breaks analyzed in this document include the following:

Inadvertent ADS Actuation

A “no-break” small-break LOCA calculation is analyzed that assumes an inadvertent actuation of the ADS. The transient begins with the opening of the Stage 1 valves, which is a situation that minimizes

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the venting capability of the reactor coolant system. Only the ADS valve vent area is available; no additional vent area exists due to a break. This case examines whether sufficient vent area is available to completely depressurize the reactor coolant system and achieve injection from the IRWST to prevent/minimize core uncovery. The worst single failure for this situation is a failure of one of two ADS Stage 4 valves connected to the non-pressurizer side hot leg. The ADS Stage 4 valve is the largest ADS valve, and it vents directly to the containment with no additional backpressure from the spargers being submerged in the IRWST. The containment pressure is a conservative, time-dependent containment pressure response. This pressure response is based on iterative execution of the NOTRUMP and WGOTHIC codes. The NOTRUMP code generates the mass and energy releases from the inadvertent ADS actuation simulation which are then input into the WGOTHIC minimum containment pressure model, which calculates the containment pressure response.

2-inch Break in a Cold Leg

A 2-inch equivalent diameter break is analyzed as a representative break, not specific to a particular pipe connection. The small size of the break leads to a long period of recirculatory flow from the cold legs into the core makeup tanks. This delays the formation of a vapor space in the core makeup tank and therefore the actuation of the ADS. The containment pressure is a conservative, time-dependent containment pressure response. This pressure response is based on iterative execution of the NOTRUMP and WGOTHIC codes. The NOTRUMP code generates the mass and energy releases from the 2-inch cold leg break simulation which are then input into the WGOTHIC minimum containment pressure model, which calculates the containment pressure response.

Double-Ended Rupture of the Direct Vessel Injection Line

The DEDVI line break evaluates the ability of the plant to recover from a moderately sized break with only half of the total passive core cooling system capacity available. The vessel side of the break of the DEDVI line break is 4 inches in equivalent diameter. The double-ended nature of this break means that there are effectively two breaks modeled:

- Downcomer to containment. The direct vessel injection nozzle includes a venturi, which limits the available break area.

- Direct vessel injection line into containment from the cold leg balance line and the broken line core makeup tank.

The containment pressure is conservatively assumed to be 20 psia based on iterative execution of the NOTRUMP and WGOTHIC codes. The NOTRUMP code generates the mass and energy releases from the DEDVI line break which are input into the WGOTHIC minimum containment pressure model, which calculates the containment pressure response used to confirm the NOTRUMP containment pressure assumption remains conservative.

The peak core heat flux during the accumulator injection period is assessed relative to the predicted critical heat flux as discussed in Subsection 15.6.5.4B.2.3.

Double-Ended Rupture of the Direct Vessel Injection Line Entrainment Sensitivity

The sensitivity case is performed to assess the effect of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling.

10-inch Cold Leg Break

The 10-inch equivalent diameter break models a break size that approaches the upper limit size for small-break LOCAs.

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15.6.5.4B.3.2 Transient Results

The transient results are presented in tables and figures for the key AP1000 parameters of interest in the following sections.

15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System

An ADS actuation signal is spuriously generated and the ADS Stage 1 valves open. The plant, which is operating at 101-percent power, is depressurized via the ADS alone. Only safety-related systems are assumed to operate in this and other small-break LOCA cases. The ADS Stage 2 and 3 valves open in sequence after the ADS Stage 1 valves open (Table 15.6.5-10). At the 20 percent core makeup tank volume, the operating ADS Stage 4A valve, which is connected to the PRHR inlet pipe, receives a signal to open. After a 60 second delay, both Stage 4B valves (one connected to the non-pressurizer side hot leg and the other connected to the PRHR inlet pipe) open. The path that fails to open as the assumed single active failure is the Stage 4A valve connected to the non-pressurizer side hot leg. The assumed reactor steady-state initial conditions can be found in Table 15.6.5-9. The sequence of events for the transient is given in Table 15.6.5-11.

This case uses a containment backpressure based on the containment pressure history that occurs as a result of the inadvertent ADS actuation. It represents a conservatively low estimate of the expected containment pressure response during the transient. The containment pressurizes for an inadvertent ADS actuation as a result of the ADS Stage 4 discharge paths that vent directly to the containment atmosphere. The time-dependent containment pressure curve (Figure 15.6.5.4B-1(c)) was calculated using the mass and energy releases from the NOTRUMP small-break LOCA code, which were used as inputs in the WGOTHIC containment model.

Transient results are shown in Figures 15.6.5.4B-1(a) through 15.6.5.4B-16(b). The transient is initiated by the opening of the two ADS Stage 1 paths. Reactor trip, reactor coolant pump trip, and safeguards actuation signals are generated via pressurizer low-pressure signals with appropriate delays. After generation of the reactor trip signal, the turbine stop valves close. The main feedwater isolation valves begin to close 2 seconds after the “S” signal pressure setpoint is reached. The reactor coolant pumps begin to coast down due to an automatic trip signal following a 7 second delay from the “S” signal. The opening of the ADS valves and the reduction in core power due to reactor trip causes the primary pressure to fall rapidly (Figure 15.6.5.4B-1(a)). Flow of fluid toward the open ADS paths causes the pressurizer to fill rapidly (Figure 15.6.5.4B-2), and the ADS flow becomes two-phase (Figures 15.6.5.4B-3 and 15.6.5.4B-4(a)). The safeguards actuation signal opens the valves isolating the core makeup tanks and circulation of cold water begins (Figures 15.6.5.4B-5 and 15.6.5.4B-6). The core makeup tanks begin to drain (Figures 15.6.5.4B-7 and 15.6.5.4B-8) during the accumulator injection period (Figures 15.6.5.4B-10 and 15.6.5.4B-11).

Continued mass flow through the ADS Stage 1, 2, and 3 valves drains the upper parts of the reactor coolant system (Figure 15.6.5.4B-4(b)). The cold leg side of the steam generator tubes start to drain, followed by the drop in mixture levels in the hot leg sides. As the ADS Stage 2 and 3 paths begin to open, increased ADS flow causes the primary pressure to fall rapidly (Figures 15.6.5.4B-1(a) and 15.6.5.4B-1(b)). As the cold leg sides of the steam generator tubes empty, the cold legs drain and a mixture level forms in the downcomer (Figure 15.6.5.4B-9).

The primary pressure falls below the pressure in the accumulators thus causing the accumulator check valves to open and accumulator delivery to begin (Figures 15.6.5.4B-10 and 15.6.5.4B-11). The accumulators, and then the core makeup tanks inject until they empty. The ADS flow falls off (Figures 15.6.5.4B-3, 15.6.5.4B-4(a) and 15.6.5.4B-4(b)) as the primary pressure decreases. The flow from the accumulators raise the mixture levels in the upper plenum and downcomer (Figures 15.6.5.4B-16(a) and 15.6.5.4B-9).

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As the levels in the core makeup tanks reach the ADS Stage 4 actuation setpoint, the ADS Stage 4 valves open as described earlier. Activating the Stage 4 paths (Figures 15.6.5.4B-12(a), 15.6.5.4B-12(b) and 15.6.5.4B-12(c)) leads to reduced flow through ADS Stages 1, 2, and 3 (Figures 15.6.5.4B-3, 15.6.5.4B-4(a), and 15.6.5.4B-4(b)). The reduced flow allows the pressurizer mixture level to fall (Figure 15.6.5.4B-2), and these stages begin to discharge only steam. After the core makeup tanks are empty (Figures 15.6.5.4B-7 and 15.6.5.4B-8), IRWST injection (Figures 15.6.5.4B-13 and 15.6.5.4B-14) does not begin until the pressure in the DVI line drops below the IRWST injection pressure, creating an injection gap (Table 15.6.5-11 and Figures 15.6.5.4B-5, 15.6.5.4B-6, 15.6.5.4B-13 and 15.6.5.4B-14). The overall decrease in reactor vessel mixture inventory (Figure 15.6.5.4B-15(b)) is large enough to result in a core uncovery (Figure 15.6.5.4B-16(a)). At 4000 seconds, the calculation is considered complete; IRWST delivery exceeds the ADS flows (which are removing the decay heat), and the reactor coolant system inventory and reactor vessel mixture inventory are slowly rising (Figures 15.6.5.4B-15(a) and 15.6.5.4B-15(b)).

The inadvertent opening of the ADS Stage 1 transient confirms the capability of the minimum venting area to depressurize the reactor coolant system to the IRWST pressure. The analysis indicates that the ADS sizing is sufficient to depressurize the reactor coolant system assuming the worst single failure as the failure of a Stage 4 ADS path to open and the 10 CFR 50 Appendix K (Reference 1) required decay heat (based on the ANS-1971 Standard (Reference 2) plus 20 percent), which lead to an over estimation of the core steam generation rate. Even under these limiting conditions, IRWST injection is obtained, and the core mixture level recovers, which terminates the cladding heatup transient (Figure 15.6.5.4B-16(b)).

15.6.5.4B.3.4 2-inch Cold Leg Break

This case models a 2-inch (50.8 mm) break occurring in a cold leg on the pressurizer side of the RCS. The reactor steady-state initial conditions assumed for this transient can be found in Table 15.6.5-9. The event times for this transient are given in Table 15.6.5-12.

This case uses a containment backpressure based on the containment pressure history that occurs as a result of the 2-inch cold leg break. It represents a conservatively low estimate of the expected containment pressure response during the transient. The containment pressurizes for a 2-inch cold leg break as a result of the break and the ADS Stage 4 discharge paths that vent directly to the containment atmosphere. The time-dependent containment pressure curve (Figure 15.6.5.4B-17(c)) was calculated using the mass and energy releases from the NOTRUMP small-break LOCA code, which were used as inputs in the WGOTHIC containment model.

Transient results are shown in Figures 15.6.5.4B-17(a) through 15.6.5.4B-35. The break opens at time zero, and the pressurizer pressure begins to fall as shown in Figure 15.6.5.4B-17(a), as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 15.6.5.4B-18. The liquid and vapor flow out of the break is shown in Figures 15.6.5.4B-32 and 15.6.5.4B-33. The pressurizer pressure falls below the reactor trip set point, causing the reactor to trip (after the appropriate time delay) and causing isolation of the steam generator main steam lines. The core makeup tank discharge isolation valves for both core makeup tanks and the PRHR delivery line isolation valves open after an “S” signal occurs (with appropriate delays); the reactor coolant pumps trip after an “S” signal with a 7 second delay. The reactor coolant system is cooled by natural circulation with decay heat being removed by the steam generators through their safety valves, through the break and via the PRHR. The PRHR heat removal and integrated heat removal are shown in Figure 15.6.5.4B-34 and Figure 15.6.5.4B-35. Once the core makeup tank isolation valves open, the core makeup tanks begin to inject borated water into the reactor coolant system as shown in Figures 15.6.5.4B-22 and 15.6.5.4B-23.

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As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 15.6.5.4B-21. The core makeup tank reaches the 67.5 percent volume, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 15.6.5.4B-18) because an opening has been created at the top of the pressurizer. After these valves open, a rapid depressurization occurs as seen in Figure 15.6.5.4B-17(a); the accumulator gas cover pressure is reached, and the accumulators begin to inject. The injection flow from the core makeup tanks are shown in Figures 15.6.5.4B-22 and 15.6.5.4B-23, and from the accumulators, in Figures 15.6.5.4B-24 and 15.6.5.4B-25.

As Figures 15.6.5.4B-22 and 15.6.5.4B-23 indicate, when the accumulators begin to inject, the flow from both core makeup tanks is reduced, and the flow is temporarily reversed due to the pressurization of the DVI line and core makeup tanks injection lines by the accumulators.

The ADS Stage 2 valves open, maintaining the depressurization rate as shown in Figure 15.6.5.4B-17(a). ADS Stage 3 valves open, thereby increasing the system venting capability. Figures 15.6.5.4B-31(a), 15.6.5.4B-31(b) and 15.6.5.4B-31(c) indicate the instantaneous liquid, instantaneous vapor and integrated total mass discharged from the ADS Stage 1-3 valves. The ADS Stage 4 valves open when the core makeup tank water volume is reduced to 20 percent. Figures 15.6.5.4B-28(a), 15.6.5.4B-28(b) and 15.6.5.4B-28(c) indicate the instantaneous liquid, instantaneous vapor and integrated total mass discharged from the ADS Stage 4 valves. After the ADS Stage 4 paths open, the pressurizer begins to drain mixture into the hot legs as seen in Figure 15.6.5.4B-18. After the CMTs are empty, IRWST injection does not begin until the pressure in the DVI line drops below the IRWST injection pressure, creating an injection gap (Table 15.6.5-12 and Figures 15.6.5.4B-22, 15.6.5.4B-23, 15.6.5.4B-26, and 15.6.5.4B-27). The mass inventory shown in Figure 15.6.5.4B-29(a) considers the primary inventory to be the reactor coolant system proper, including the pressurizer; the mass present in the passive safety system components is not included. The mass inventory shown in Figure 15.6.5.4B-29(b) considers the reactor vessel mixture inventory, including the downcomer, lower plenum, core fluid channel, upper plenum and upper head, and clearly shows the decrease in the inventory during the injection gap period. Once the pressures in the DVI lines drop below the IRWST injection pressure, flow enters the reactor vessel from the IRWST. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 15.6.5.4B-30(a) for the majority of the transient; however, the upper plenum mixture level drops during the injection gap period and the core uncovers as the mixture level drops below the top of the active fuel. The 2-inch break cases exhibits a cladding heatup transient as a result of the core uncovery as shown in Figure 15.6.5.4B-30(b).

15.6.5.4B.3.5 Direct Vessel Injection Line Break

This case models the double-ended rupture of the DVI line at the nozzle into the downcomer. The broken loop injection system (consisting of an accumulator, a core makeup tank, and an IRWST delivery line) is modeled to spill completely out of the DVI line side of the break into containment. The steady-state reactor coolant system conditions for this transient are shown in Table 15.6.5-9. Minimum resistances are applied to the broken loop IRWST injection line to maximize the spill to containment, thus minimizing the reactor coolant system mass inventory. The containment pressurization is calculated using the mass and energy releases from the NOTRUMP small-break LOCA code in the WGOTHIC minimum containment pressure model and demonstrates that the 20 psia containment backpressure assumption is conservative. This case uses a containment backpressure defined to be a constant 20 psia. While not exactly reflecting the containment pressure history that occurs as a result of the DVI line break, it represents a conservatively low estimate of the expected containment pressure response during a DEDVI transient. The containment pressurizes for a DEDVI line break as a result of the break mass and energy releases in addition to the ADS-4 discharge paths that vent directly to the containment atmosphere.

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The event times for this transient are shown in Table 15.6.5-13. Transient results are shown in Figures 15.6.5.4B-36 through 15.6.5.4B-55. The break is assumed to open instantaneously at 0 seconds. The accumulator on the broken loop starts to discharge via the DVI line to the containment. Figure 15.6.5.4B-36 shows the subcooled discharge from the downcomer nozzle, which causes a rapid reactor coolant system (RCS) depressurization (Figure 15.6.5.4B-38(a)). A reactor trip signal is generated, followed by generation of the “S” signal. Following a delay, the isolation valves on the core makeup tank and PRHR delivery lines begin to open. The PRHR heat removal and integrated heat removal are shown in Figure 15.6.5.4B-54 and Figure 15.6.5.4B-55. The “S” signal also causes closure of the main feedwater isolation valves after a 2 second delay and trips the reactor coolant pumps after a 7 second delay. The opening of the core makeup tank isolation valves allows the broken loop core makeup tank to discharge directly to the containment (Figure 15.6.5.4B-39), and a small circulatory flow develops through the intact loop core makeup tank (Figure 15.6.5.4B-40).

As the pressure falls, the reactor coolant system fluid saturates, and a mixture level forms in the upper plenum and then falls to the hot leg elevation (Figure 15.6.5.4B-41). The upper parts of the reactor coolant system start to drain, and a mixture level forms in the downcomer (Figure 15.6.5.4B-42) and falls below the elevation of the break. Two-phase discharge, then vapor flow occurs from the downcomer side of the break (Figure 15.6.5.4B-37).

In the core makeup tank connected to the broken loop, a level forms and starts to fall. The ADS Stage 1 setpoint is reached, and the ADS Stage 1 valves open after the signal delay time elapses. The ensuing steam discharge from the top of the pressurizer (through the ADS valves; Figures 15.6.5.4B-43(a), 15.6.5.4B-43(b), and 15.6.5.4B-43(c)) increases the reactor coolant system depressurization rate. The depressurization rate is also increased due to the steam discharge from the downcomer to the containment (Figure 15.6.5.4B-37) as the downcomer mixture level falls below the DVI nozzle (Figure 15.6.5.4B-42).

During the initial portion of the DEDVI line break, only liquid flows out the top of the core (Figure 15.6.5.4B-45). Soon, steam flow follows (Figure 15.6.5.4B-46) correlating with the void fraction increase in the core (Figure 15.6.5.4B-44). The break in the downcomer stalls fluid flow into the bottom of the core (Figure 15.6.5.4B-47) leaving insufficient liquid in the upper plenum. The upper plenum/core mixture level therefore starts to decrease (Figure 15.6.5.4B-41). The mixture level falls early in the transient and then recovers slightly, as flow slowly re-enters the core from the downcomer (Figure 15.6.5.4B-41 compared to 15.6.5.4B-47).

The ADS Stage 2 valves open after the appropriate time delay following the actuation of ADS Stage 1. The intact loop accumulator starts to inject into the downcomer (Figure 15.6.5.4B-50) causing the mixture level in the downcomer to slowly rise (Figure 15.6.5.4B-42). The mixture level also increases slightly within the upper plenum.

The ADS Stage 3 valves open upon completion of the time delay of 120 seconds following the actuation of ADS Stage 2. The broken loop core makeup tank level reaches the ADS Stage 4 setpoint, but the ADS Stage 4 valves do not open until the minimum time delay between the actuation of ADS Stages 3 and 4 occurs. Two-phase discharge ensues through three of the four Stage 4 paths (Figures 15.6.5.4B-48(a), 15.6.5.4B-48(b), and 15.6.5.4B-49). During the same timeframe, the broken loop core makeup tank and accumulator empty rapidly.

The fluid level at the top of the intact loop core makeup tank starts to decrease slowly (Figure 15.6.5.4B-52) because injection from the tank has begun (Figure 15.6.5.4B-40). The intact loop core makeup tank injection is temporarily reduced and then reversed as the intact accumulator injects (Figure 15.6.5.4B-50). Following the accumulator injection, the core makeup tank injects continuously until empty.

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During the period of accumulator injection, the downcomer mixture level rises slowly (Figure 15.6.5.4B-42). Figure 15.6.5.4B-53(a) presents the RCS mass inventory, and Figure 15.6.5.4B-53(b) presents the reactor vessel mixture inventory which includes the downcomer, lower plenum, core fluid channel, upper plenum and upper head. Both figures also show the increase in inventory due to the accumulator injection. With injection available only from the intact loop core makeup tank, the downcomer level remains fairly constant. However, reactor vessel mixture inventory depletion continues until sufficient IRWST injection flow can be introduced. The level in the upper plenum is maintained near the hot leg elevation (Figure 15.6.5.4B-41) throughout the remainder of the transient.

Once the pressure in the broken DVI line falls below that in the IRWST discharge piping, the water from the tank begins spilling to containment.

Stable, but decreasing, injection continues from the intact loop core makeup tank as the inventory slowly depletes; the reactor coolant system pressure declines slowly. The reactor coolant system pressure continues to fall until it drops below that of the IRWST and injection begins (Figure 15.6.5.4B-51). With the reduced initial RCS inventory recovery from a single accumulator, core makeup tank, and IRWST injection path available for the DEDVI line break, the minimum inventory occurs after the initiation of continuous IRWST injection flow. After injection flow greater than the sum of the break and ADS flows exists, a slow rise in the reactor vessel mixture inventory (Figure 15.6.5.4B-53(b)) occurs. Since no core uncovery is predicted for this scenario, no cladding heatup occurs.

The critical heat flux assessment described in Subsection 15.6.5.4B.2.3 addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 (Reference 1) are met during this period since this CHF assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

15.6.5.4B.3.6 10-inch Cold Leg Break

This case models a 10-inch break occurring in the cold leg connected to the balance line of CMT-1. The reactor steady-state initial conditions assumed for this transient are found in Table 15.6.5-9. The event times for this transient are given in Table 15.6.5-14.

Transient results are shown in Figures 15.6.5.4B-56(a) through 15.6.5.4B-78. The break opens at time zero, and the pressurizer pressure begins to fall, as shown in Figure 15.6.5.4B-56(a), as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 15.6.5.4B-57. The break fluid flow is shown in Figures 15.6.5.4B-75 and 15.6.5.4B-76 for the liquid and vapor components respectively. The pressurizer pressure falls below the reactor trip set point which causes the reactor to trip (after the appropriate time delay) and isolation of the steam generator main steam lines. The core makeup tank discharge isolation valves for both core makeup tanks and the PRHR delivery line isolation valves open after an “S” signal occurs (with appropriate delays); the reactor coolant pumps trip after an “S” signal with a 7 second delay. The reactor coolant system is cooled by natural circulation with energy being removed by the steam generator safety valves, the core makeup tanks, and the PRHR heat exchanger. The PRHR heat removal rate and integrated heat removal are shown in Figure 15.6.5.4B-77 and Figure 15.6.5.4B-78. Once the core makeup tank isolation valves open, the core makeup tanks begin to inject borated water into the reactor coolant system as shown in Figures 15.6.5.4B-61 and 15.6.5.4B-62.

As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 15.6.5.4B-60, and the core remains completely covered with the exception of a few short oscillatory time intervals in which the mixture level drops below the active fuel

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(Figure 15.6.5.4B-69). Due to the size and location of the break involved, the accumulator setpoint is reached prior to the core makeup tanks transitioning from recirculation to injection mode. The flows from the core makeup tanks are shown in Figures 15.6.5.4B-61 and 15.6.5.4B-62, and from the accumulators, in Figures 15.6.5.4B-63 and 15.6.5.4B-64. Once the first core makeup tank reaches the 67.5 percent volume setpoint, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 15.6.5.4B-57) because an opening has been created at the top of the pressurizer. After these valves open, a more rapid depressurization occurs as seen in Figure 15.6.5.4B-56(a).

During the initial portion of the 10-inch break, both liquid and steam flow out the top of the core (Figures 15.6.5.4B-71 and 15.6.5.4B-72) as the void fraction in the core increases (Figure 15.6.5.4B-73). The break in the cold leg draws fluid from the bottom of the core leaving insufficient liquid in the upper plenum. The mixture level, therefore, starts to decrease (Figure 15.6.5.4B-69). The mixture level falls until accumulator flows enter the downcomer (Figures 15.6.5.4B-63 and 15.6.5.4B-64).

As Figures 15.6.5.4B-61 and 15.6.5.4B-62 indicate, when the accumulators begin to inject, the flow from both core makeup tanks is reduced and briefly reversed, due to the pressurization of the core makeup tank injection lines by the accumulators. The opening of ADS Stage 2 valves maintains the depressurization rate as shown in Figure 15.6.5.4B-56(a). ADS Stage 3 valves subsequently open. This increases the system venting capability. Figures 15.6.5.4B-70(a), 15.6.5.4B-70(b) and 15.6.5.4B-70(c) indicate the instantaneous liquid, instantaneous vapor and integrated total mass discharged from the ADS Stage 1-3 valves, respectively. The ADS Stage 4 valves open when the core makeup tank water volume is reduced to 20 percent. Figures 15.6.5.4B-67(a), 15.6.5.4B-67(b) and 15.6.5.4B-74 indicate the instantaneous liquid, instantaneous vapor and integrated total mass discharged from the ADS Stage 4 valves. After the ADS Stage 4 path opens, the pressurizer begins to drain mixture into the hot legs as seen in Figure 15.6.5.4B-57. The Figure 15.6.5.4B-68(a) mass inventory plot considers the primary inventory to be the reactor coolant system proper, including the pressurizer; the mass present in the passive safety system components is not included. The Figure 15.6.5.4B-68(b) mass inventory plot considers the reactor vessel mixture inventory, including the downcomer, lower plenum, core fluid channels, upper plenum and upper head. Once the downcomer pressure drops below the IRWST injection pressure, flow enters the reactor vessel from the IRWST. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 15.6.5.4B-69 throughout this transient; core uncovery does not occur for any prolonged period of time and may be deemed negligible. As such, the 10-inch break case exhibits large margins to the 10 CFR 50.46 (Reference 1) peak cladding temperature limit of 2200°F (1204.44°C).

15.6.5.4B.3.7 Direct Vessel Injection Line Break (Entrainment Sensitivity)

In order to assess the potential impact of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling, an AP1000 plant sensitivity run was performed. The simulation utilizes the same initial conditions as the base DEDVI line break case presented in Subsection 15.6.5.4B.3.5.

The transient response is essentially identical until ADS-4 actuation, at which time the bounding entrainment conditions are included in the analysis by assuming homogenous conditions in the regions downstream of the core (upper plenum, hot leg, and PRHR inlet). In addition, since homogenous treatment of these regions will eliminate the pressure drop effect of the accumulated mass stored in the upper plenum, the NOTRUMP model was conservatively adjusted to account for this effect following the transition of the ADS-4 flow paths to non-critical conditions.

The event times for this transient are shown in Table 15.6.5-15. Transient results are shown in Figures 15.6.5.4B-79(a) through 15.6.5.4B-90. Figures 15.6.5.4B-79(a) and 15.6.5.4B-79(b) present comparisons of the pressure in the upper portion of the downcomer between the base and sensitivity

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cases. The sensitivity case results in higher pressure in the upper portion of the downcomer and subsequently results in delayed IRWST injection (Figure 15.6.5.4B-80). This can also be observed in the intact DVI line flow, which comprises all intact injection flow components (accumulator, CMT, and IRWST) per Figure 15.6.5.4B-81, and the pressurizer mixture level response (Figure 15.6.5.4B-90), which follows the change in pressure response. As expected, the initial ADS-4 liquid discharge is much higher (Figure 15.6.5.4B-82) until the upper plenum and hot leg region inventory is depleted (Figure 15.6.5.4B-83). The net effect is a decrease in the ADS-4 vapor discharge rate (Figure 15.6.5.4B-84) and subsequently higher RCS pressures.

Due to the elimination of the inventory stored in the upper plenum, the downcomer mass is also reduced (Figure 15.6.5.4B-85). Since the static head that existed in the upper plenum is eliminated when the model is made homogenous, the downcomer mixture is subsequently driven into the core as the static heads equilibrate. This results in the core region mass increasing initially due to the introduction of cold downcomer fluid to the core region (Figures 15.6.5.4B-86(a) and 15.6.5.4B-86(b)). The net effect of the sensitivity case is that the vessel inventory is substantially decreased over the base model simulation (Figure 15.6.5.4B-87); however, this inventory is sufficient to provide adequate core cooling because the ADS-4 continually draws liquid flow through the core (Figure 15.6.5.4B-82). Even though there is no liquid storage in the upper plenum for the homogenous case (Figures 15.6.5.4B-88(a) and 15.6.5.4B-88(b)), the core collapsed liquid level (Figures 15.6.5.4B-89(a) and 15.6.5.4B-89(b)) is not impacted significantly.

The impact on fuel rod heatup due to the periods of oscillatory core uncovery observed in Figures 15.6.5.4B-88(a) and 15.6.5.4B-88(b) was assessed and it was demonstrated that the inadvertent ADS actuation transient presented in Subsection 15.6.5.4B.3.3 remains limiting compared to the entrainment study.

This sensitivity demonstrates that the AP1000 plant response is relatively insensitive to upper plenum and hot leg entrainment for the DEDVI line break, which is the limiting break with respect to entrainment. Overall, the sensitivity shows that even with the assumption of homogenous fluid nodes above the core (i.e., maximum entrainment), adequate core cooling is demonstrated.

15.6.5.4B.4 Conclusions

The small-break LOCA analyses performed show that the passive safeguards systems are sufficient to mitigate small-break LOCAs. Specifically, it is concluded that:

- The primary side can be depressurized by the ADS to allow stable injection into the core.
- Injection from the core makeup tanks, accumulators, and IRWST prevents excessive cladding heatup for small-break LOCAs analyzed, including double-ended ruptures in the passive safeguards system lines. The peak heat flux during the DEDVI line break accumulator injection period is below the predicted critical heat flux.
- The effect of increasing upper plenum/hot leg entrainment does not significantly affect plant safety margins.

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The analyses performed demonstrate that the 10 CFR 50.46 Acceptance Criteria are met. Summarizing the small-break LOCA spectrum:

Break Location/Diameter	Minimum Reactor Vessel Mixture Inventory (lbm)	Peak Cladding Temperature (°F)
Inadvertent ADS	54,095	1099
2-inch cold leg break	54,855	874
10-inch cold leg break	78,826	(1)
DEDVI (20.0 psia)	74,575	(1)
DEDVI (Entrainment Study)	54,527	(2)

(1) There is no core heatup as a result of this transient. PCT occurs at transient initiation.
(2) There is minimal core heatup as a result of this transient and the PCT remains significantly less than the limiting inadvertent ADS actuation transient.

The inadvertent ADS actuation transient exhibits the limiting minimum reactor vessel mixture inventory conditions and has the limiting peak cladding temperature. The AP1000 design is such that the minimum reactor vessel mixture inventory occurs around the time of IRWST injection for the limiting breaks. All breaks simulated in the break spectrum produce results that demonstrate significant margin to the 10 CFR 50.46 regulatory limits.

15.6.5.4C Post-LOCA Long-Term Cooling

15.6.5.4C.1 Long-Term Cooling Analysis Methodology

The AP1000 safety-related systems are designed to provide adequate cooling of the reactor indefinitely. Initially, this is achieved by discharging water from the IRWST into the vessel. When the low-3 level setpoint is reached in the IRWST, the containment recirculation subsystem isolation valves open and water from the containment reactor coolant system (RCS) compartment can flow into the vessel through the PXS piping. The water in containment rises in temperature toward the saturation temperature. Long-term heat removal from the reactor and containment is by heat transfer through the containment shell to atmosphere.

The purpose of the long-term cooling analysis is to demonstrate that the passive systems provide adequate emergency core cooling system performance during the IRWST injection/containment recirculation time scale. The long-term cooling analysis is performed using the WCOBRA/TRAC computer code to verify that the passive injection system is providing sufficient flow to the reactor vessel to cool the core and to preclude boron precipitation.

The AP1000 long-term cooling analysis is supported by the series of tests at the Oregon State University AP600 APEX Test Facility. This test facility is designed to represent the AP600 reactor safety-related systems and nonsafety-related systems at quarter-scale during long-term cooling. The data obtained during testing at this facility has been shown to apply to the AP1000 (Reference 25). These tests were modeled using WCOBRA/TRAC (Reference 23) with an equivalent noding scheme to that used for AP600 (Reference 17) in order to validate the code for long-term cooling analysis.

Reference 24 provides details of the AP1000 WCOBRA/TRAC modeling. The coarse reactor vessel modeling used for AP600 has been replaced with a detailed noding like that applied in the large-break LOCA analyses described in Subsection 15.6.5.4A. The reactor vessel noding used in the AP1000 long-term cooling analyses in core and upper plenum regions is equivalent to that used in full-scale test simulations (see Reference 24).

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A DEDVI line break is analyzed because it is the most limiting long-term cooling case in the relationship between decay power and available liquid driving head. Because the IRWST spills directly onto the containment floor in a DEDVI break, this event has the highest core decay power when the transfer to sump injection is initiated. In postulated DEDVI break cases, the compartment water level exceeds the elevation at which the DVI line enters the reactor vessel, so water can flow from the containment into the reactor vessel through the broken DVI line; this in-flow of water through the broken DVI line assists in the heat removal from the core. The steam produced by boiling in the core vents to the containment through the ADS valves and condenses on the inner surface of the steel containment vessel. The condensate is collected and drains to the IRWST to become available for injection into the reactor coolant system. The WCOBRA/TRAC analysis presented analyzes the DEDVI small-break LOCA event from a time (3000 seconds) at which IRWST injection is fully established to beyond the time of containment recirculation. During this time, the head of water to drive the flow into the vessel for IRWST injection decreases from the initial level to its lowest value at the containment recirculation switchover time. PXS Room B is the location of the break in the DVI line. At this break location, liquid level in containment at the time of recirculation is a minimum.

A continuous analysis of the post-LOCA long term cooling is provided from the time of stable IRWST injection through the time of sump recirculation for the DEDVI break. Maximum design resistances are applied in WCOBRA/TRAC for both the ADS Stage 4 flow paths and the IRWST injection and containment recirculation flow paths.

The break modeled is a double-ended guillotine rupture of one of the direct vessel injection lines. The long-term cooling phase begins after the simultaneous opening of the isolation valves in the IRWST DVI lines and the opening of ADS Stage 4 squib valves, when flow injection from the IRWST has been fully established. Initial conditions consider the NOTRUMP DEDVI case at 20 psia containment pressure reported in Subsection 15.6.5.4B.

15.6.5.4C.2 DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case

This subsection presents the results of a DEDVI line break analysis during IRWST injection phase continuing into sump recirculation. Initial conditions at the start of the case are prescribed based on the NOTRUMP DEDVI break results to allow a calculation to begin shortly after IRWST injection begins in the small break long-term cooling transient. The WCOBRA/TRAC calculation is then allowed to proceed until a quasi-steady-state is achieved. At this time, the predicted results are independent of the assumed initial conditions. This calculation uses boundary conditions taken from a WGOTHIC analysis of this event. During the calculation, which is carried out for 10,000 seconds until a quasi-steady-state sump recirculation condition has been established, the IRWST water level is decreased continuously until the sump recirculation setpoint is reached.

In the analysis, one of the two ADS Stage 4 valves in the PRHR loop is assumed to have failed. The initial reactor coolant system liquid inventory and temperatures are determined from the NOTRUMP calculation. The core makeup tanks do not contribute to the DVI injection during this phase of the transient. Steam generator secondary side conditions are taken from the NOTRUMP calculation (at the beginning of long-term cooling). The reactor coolant pumps are tripped and not rotating.

The temperatures of the liquid in the containment sump and the containment pressure are based on WGOTHIC calculations of the conservative minimum pressure during this long-term cooling transient, including operation of the containment fan coolers. Small changes in the RCS compartment level do not have a major effect on the predicted core collapsed liquid level or on the predicted flow rate through the core. The minimum compartment floodup level for this break scenario is 107.6 feet or greater.

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In this transient, the IRWST provides a hydraulic head sufficient to drive water into the downcomer through the intact DVI nozzle. Also, water flows into the downcomer from the broken DVI line once the liquid level in the compartment with the broken line is adequate to support flow. The water flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core and liquid flow out of the reactor coolant system via the ADS Stage 4 valves. There is little flow out of ADS Stages 1, 2, and 3 even when the IRWST liquid level falls below the sparger elevation, so they are not modeled in this calculation. The venting provided by the ADS-4 paths enables the liquid flow through the core to maintain core cooling.

Approximately 500 seconds of WCOBRA/TRAC calculation are required to establish the quasi-steady-state condition associated with IRWST injection at the start of long-term cooling and so are ignored in the following discussion. The hot leg levels are such that during the IRWST injection phase the quality of the ADS Stage 4 mass flows varies as water is carried out of the hot legs. This periodically increases the pressure drop across the ADS Stage 4 valves and the upper plenum pressure. The higher pressure in the upper plenum reduces the injection flow. This cycle of pressure variations due to changing void fractions in the flow through ADS Stage 4 is consistent with test observations and is expected to recur often during long-term cooling.

The head of water in the IRWST causes a flow of subcooled water into the downcomer at an approximate rate of 140 lbm/s through the intact DVI nozzle at the start of long-term cooling. The downcomer level at the end of the code initiation (the start of long-term cooling) is about 16.0 feet (Figure 15.6.5.4C-1). Note that the time scale of this and other figures in Subsection 15.6.5.4C.2 is offset by 2500 seconds; that is, a time of 500 seconds on the Figure 15.6.5.4C-1 axis equals 3000 seconds transient time for the DEDVI break. All of the injection water flows down the downcomer and up through the core. The accumulators have been fully discharged before the start of the time window and do not contribute to the DVI flow.

Boiling in the core produces steam and a two-phase mixture, which flows into the upper plenum. The core is 14 feet high, and the core average collapsed liquid level (Figure 15.6.5.4C-2) is shown from the start of long-term cooling. The boiling process causes a variable rate of steam production and resulting pressure changes, which in turn causes oscillations in the liquid flow rate at the bottom of the core and also variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC noding, the core is divided both axially and radially as described in Reference 24. The void fractions in the top two cells of the hot assembly are shown as Figures 15.6.5.4C-3 and 15.6.5.4C-4. The average void fraction of these upper core cells is approximately 0.9 during long-term cooling, during IRWST injection, and into the containment recirculation period. There is a continuous flow of two-phase fluid into the hot legs, and mainly vapor flow toward the ADS Stage 4 valve occurs at the top of the pipe. The collapsed liquid level in the hot leg averages around 1.3 feet (Figure 15.6.5.4C-5). The hot legs on average are approximately 50-percent full. Vapor and liquid flows at the top of the core are shown in Figures 15.6.5.4C-6 and 15.6.5.4C-7, the upper plenum collapsed liquid level in Figure 15.6.5.4C-8. Figures 15.6.5.4C-9 and 15.6.5.4C-10 are ADS stage 4 mass flowrates.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-11. The upper plenum pressure fluctuation that occurs is due to the ADS Stage 4 water discharge. The PCT of the hot rod follows saturation temperature (Figure 15.6.5.4C-12), which demonstrates that no uncovery and no cladding temperature excursion occurs. A small pressure drop is calculated across the reactor vessel, and injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-13 and 15.6.5.4C-14. Figure 15.6.5.4C-14 shows the broken DVI line flow during the start of the long-term cooling period increases to about 75 lbm/s after the compartment water level has increased above the nozzle elevation to permit liquid injection into the reactor vessel. In contrast, the intact DVI line flow falls from 140 lbm/s with a full IRWST to about 60 lbm/s flow from the containment at the end of the calculation. The recirculation core liquid throughput is more than adequate to preclude any boron buildup on the fuel.

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15.6.5.4C.3 DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation

This subsection presents a DEDVI line break analysis with wall-to-wall flooding due to leakage between compartments, using the window mode methodology. All containment free volume beneath the level of the liquid is assumed filled in this calculation to generate the minimum water level condition during containment recirculation. The time identified for this calculation is around 14 days into the event, and the core power is calculated accordingly. The initial conditions at the start of the window are consistent with the analysis described in Subsection 15.6.5.4C.2. Containment recirculation is simulated during the time window. The calculation is carried out over a time period long enough to establish a quasi-steady-state solution; after 500 seconds of problem time, the flow dynamics are quasi-steady-state and the predicted results are independent of the assumed initial conditions. The liquid level is simulated constant at 28.2 feet above the bottom inside surface of the reactor vessel (refer to Figure 15.0.3-2 for AP1000 reference plant elevations) during the time window, and the liquid temperatures in the containment sump and the PXS “B” room are both modeled at 212°F. The containment pressure is conservatively assumed to be 14.7 psia. The single failure of an ADS Stage 4 flow path is assumed as in the Subsection 15.6.5.4C.2 case.

Focusing on the post 400-second time interval of this case, the containment liquid provides a hydraulic head sufficient to drive water into the downcomer through the DVI nozzles. The water introduced into the downcomer flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core entrains liquid and flows out of the reactor coolant system via the ADS Stage 4 valves. The DVI flow and the venting provided by the ADS paths provide a liquid flow through the core that enables the core to remain cool.

The downcomer collapsed liquid level (Figure 15.6.5.4C-15) varies between 23 and 24 feet during the analysis. Pressure spikes produced by boiling in the core can cause the mass flow of the DVI flow rates shown in Figures 15.6.5.4C-27 and 15.6.5.4C-28 into the vessel to fluctuate upward and downward.

Boiling in the core produces steam and a two-phase mixture, which flows out of the core into the upper plenum. The core is 14 feet high, and the core collapsed liquid level (Figure 15.6.5.4C-16) maintains a mean level close to the top of the core. The boiling process causes pressure variations, which in turn, cause variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC analysis, the core is nodalized as described in Reference 24. The void fraction in the top cell is shown in Figure 15.6.5.4C-17 for the core hot assembly, and Figure 15.6.5.4C-18 shows the void fraction that exists one cell further down in the hot assembly. The PCT does not rise appreciably above the saturation temperature (Figure 15.6.5.4C.3-26). The flow through the core and out of the reactor coolant system is more than sufficient to provide adequate flushing to preclude concentration of the boric acid solution. Liquid collects above the upper core plate in the upper plenum, where the average collapsed liquid level is about 3.1 feet (Figure 15.6.5.4C-22). There is no significant flow through the cold legs into either the broken or the intact loops, and there is no significant quantity of liquid residing in any of the cold legs.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-25. The upper plenum pressurization, which occurs periodically, is due to the ADS Stage 4 water discharge. The collapsed liquid level in the hot leg of the pressurizer loop increases to >1.5 feet by the end of the simulation (Figure 15.6.5.4C-19). Injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-27 and 15.6.5.4C-28.

15.6.5.4C.4 Post Accident Core Boron Concentration

An evaluation has been performed of the potential for the boron concentration to build up in the core following a cold leg LOCA. The evaluation methodology is discussed and demonstrated via simplified calculations and results in Subsection 2.3.4 and Appendix H of Reference 24. The detailed

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calculations and results are described in Reference 27. This evaluation considers both short-term operations, before ADS is actuated, and long-term operations, after ADS is actuated. These evaluations and their results are discussed in the follow paragraphs.

Short-term – Prior to ADS actuation, it is not likely for boron to build up significantly in the core. Normally, water circulation mixes boron in the RCS and prevents buildup in the core. In order for boron to start to build up in the core region, water circulation through the steam generators and PRHR HX has to stop. In addition, significant injection of borated water is needed from the CMTs and the CVS. For this situation to happen, the hot legs need to void sufficiently to allow the steam generator tubes to drain. Once the steam generator tubes void, the cold legs will also void since they are located higher than the hot legs. When the top of the cold legs void, the CMTs will begin to drain. When the CMTs drain to the ADS stage 1 setpoint, ADS is actuated.

Short-term Results – As shown in Subsection 15.6.5.4B.3.4, a 2-inch LOCA requires less than 16 minutes from the time that the hot legs void significantly until ADS is actuated. For larger LOCAs, this time difference is shorter, as seen for the 10-inch cold leg LOCA (Subsection 15.6.5.4B.3.6). The core boron concentration will not build up significantly in this short time. If the break is smaller than 2 inches, voiding of the hot legs will occur at a later time. With maximum operation of CVS makeup, it takes more than 8 hours for the core boron concentration to build up significantly. In addition, the volume of the boric acid storage tank limits the maximum buildup of boron in the core.

Following a small LOCA where ADS is not actuated, the operators are guided to sample the RCS boron concentration and to initiate a post-LOCA cooldown and depressurization. The cooldown and depressurization of the RCS reduces the leak rate and facilitates recovery of the pressurizer level. Recovery of the pressurizer level allows for re-establishment of water flow through the RCS loops, which mixes the boron. The operators are guided to take an RCS boron sample within 8 hours of the accident and several more during the plant cooldown. The purpose of the boron samples is to assess that there is adequate shutdown margin and that the RCS boron concentration has not built up to excessive levels. The maximum calculated core boron concentration 8 hours after a LOCA without ADS actuation is less than 40,000 ppm. Operator action within 8 hours maintains the maximum core boron concentration well below the boron solubility limit for the core inlet temperatures during the cooldown.

Long-term – Once ADS is actuated, water carryover out the ADS Stage 4 lines limits the potential core boron concentration buildup following a cold leg LOCA. The design of the AP1000 facilitates water discharge from the hot legs as follows:

- PXS recirculation flow capability tends to fill the hot legs and bring the water level up to the ADS Stage 4 inlet.
- ADS Stage 4 lines discharge at an elevation 3 to 4 feet above the containment water level.

With water carried out ADS Stage 4, the core boron concentration increases until the boron added to the core in the safety injection flow equals the boron removed in the water leaving the RCS through the ADS Stage 4 flow. The lower the ADS Stage 4 vent quality, the lower the core boron concentration buildup.

Long-term Results – Analyses have been performed to bound the maximum core boron concentration buildup. These analyses demonstrate that highest ADS Stage 4 vent qualities result from the following:

- Highest decay heat levels

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- Lowest PXS injection/ADS 4 vent flows, including high line resistances and low containment water levels

The long-term cooling analysis discussed in Subsection 15.6.5.4C.2 is consistent with these assumptions. The ADS Stage 4 vent quality resulting from this analysis is approximately 62 percent at the beginning of IRWST injection and the quality decreases to less than 40 percent at the initiation of recirculation at around 2 hours. The vent quality from ADS Stage 4 continues to decrease to approximately 10 percent at 14 days.

With the maximum ADS Stage 4 vent qualities, the maximum core boron concentration peaks at a value of about 12,000 ppm at approximately 0.5 hours after the break. After this time, the core boron concentration decreases as the ADS Stage 4 vent quality decreases, reaching 5400 ppm about 8 hours after the accident. The core boron solubility temperature reaches a maximum of 90°F (at 12,000 ppm) and quickly drops to 40°F (at 5400 ppm). With these low core boron solubility temperatures, there is no concern with cold PXS injection water causing boron precipitation in the core. With the IRWST located inside containment, its water temperature is normally expected to be above these solubility temperatures. The minimum core inlet temperature is greater than the solubility temperature considering heatup of the injection by steam condensation in the downcomer and pickup of sensible heat from the reactor vessel, core barrel, and lower support plate.

The boron concentration of water in the containment is about 2900 ppm at the start of recirculation. In the long term, the containment boron concentration increases as the reactor vessel boron concentration decreases, reaching a value of approximately 3100 ppm after the boric acid storage tank empties. The solubility temperature of the containment water at its maximum boron concentration is 32°F.

With high decay heat values, the ADS Stage 4 vent flows and velocities are high. These high vent velocities result in flow regimes that are annular for more than 30 days. The annular flow regime moves water up and out the ADS Stage 4 lines. This flow regime is based on the Taitel-Dukler vertical flow regime map. Lower decay heat levels can be postulated later in time or just after a refueling outage. Significantly lower decay heat levels result in lower ADS Stage 4 vent qualities. They also result in ADS Stage 4 vent flows/velocities that are lower. Even with low ADS Stage 4 vent flow velocities, the AP1000 plant will move water out the ADS Stage 4 operating as a manometer. Small amounts of steam generated in the core reduce the density of the steam/water mixture in the core, upper plenum, and ADS Stage 4 line as it bubbles up through the water. As a result, the injection head is sufficient to push the less dense, bubbly steam/water mix out the ADS Stage 4 line.

At the time recirculation begins, the containment level will be about 109.3 feet (for a non-DVI LOCA) and will be about 108.0 feet (for a DVI LOCA). Over a period of weeks after a LOCA, water may slowly leak from the flooded areas in containment to other areas inside containment that did not initially flood. As a result, the minimum containment water could decrease to 103.5 feet. During recirculation operation following a LOCA and ADS actuation, the operators are guided to maintain the containment water level above the 107-foot elevation by adding borated water to the containment. In addition, if the plant continues to operate in the recirculation mode, the operators are guided to increase the level to 109 feet within 30 days of the accident. These actions provide additional margin in water flow through the ADS Stage 4 line. The operators are also guided to sample the hot leg boron concentration prior to initiating recovery actions that might introduce low temperature water to the reactor.

15.6.5.4C.5 Conclusions

Calculations of AP1000 long-term cooling performance have been performed using the WCOBRA/TRAC model developed for AP1000 and described in Reference 24. The DEDVI case was chosen because it reaches sump recirculation at the earliest time (and highest decay heat). A window mode

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case at the minimum containment water level postulated to occur around 2 weeks into long-term cooling was also performed.

The DEDVI small-break LOCA exhibits no core uncovery due to its adequate reactor coolant system mass inventory condition during the long-term cooling phase from initiation into containment recirculation. Adequate flow through the core is provided to maintain a low cladding temperature and to prevent any buildup of boric acid on the fuel rods. The wall-to-wall floodup case using the window mode technique demonstrates that effective core cooling is also provided at the minimum containment water level. The results of these cases demonstrate the capability of the AP1000 passive systems to provide long-term cooling for a limiting LOCA event.

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35. “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design,” NUREG-1793, Supplement 2, August 2011.

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Table 15.6.1-1
Time Sequence of Events for Incidents That Cause a Decrease in Reactor Coolant Inventory

Accident	Event	Time (seconds)
Inadvertent opening of a pressurizer safety valve	Pressurizer safety valve opens fully	0.00
	Low-2 pressurizer pressure reactor trip setpoint reached	19.50
	Rods begin to drop	21.50
	Minimum DNBR occurs	22.30
Inadvertent opening of an ADS Stage 2 or 3 train	ADS valves begin to open	0.00
	Low-2 pressurizer pressure reactor trip setpoint reached	22.49
	Rods begin to drop	24.49
	Minimum DNBR occurs	25.40
	ADS valves fully open	30.00

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Table 15.6.2-1
Parameters Used in Evaluating the Radiological Consequences of a Small Line Break Outside Containment

Reactor coolant iodine activity	Initial activity equal to the design basis reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Table 15A-2 in Appendix 15A)[(a)]
Reactor coolant noble gas activity	280 μCi/g dose equivalent Xe-133
Break flow rate (gpm)	130[(b)]
Fraction of reactor coolant flashing	0.47
Duration of accident (hr)	0.5
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Nuclide data	See Table 15A-4

Notes:
a. Use of accident-initiated iodine spike is consistent with the guidance in the Standard Review Plan.
b. At density of 62.4 lb/ft^3.

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Table 15.6.3-1
Steam Generator Tube Rupture Sequence of Events

Events	Time (seconds)
Double-ended steam generator tube rupture	0
Loss of offsite power	0
Reactor trip	0
Reactor coolant pumps and main feedwater pumps assumed to trip and begin to coastdown	0
Two chemical and volume control pumps actuated and pressurizer heaters turned on	0
Low-2 pressurizer level signal generated	2,591
Ruptured steam generator power-operated relief valve fails open	2,591
Core makeup tank injection and PRHR operation begins (following maximum delay)	2,608
Ruptured steam generator power-operated relief valve block valve closes on Low-2 steam line pressure signal	3,177
Chemical and volume control system isolated on High steam generator narrow range level coincident with reactor trip (P-4) permissive	15,057
Break flow terminated	34,185

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Table 15.6.3-2
Steam Generator Tube Rupture Mass Release Results

Total Mass Flow from Initiation of Event to Cooldown to RNS[1] Conditions

	Start of Event to Break Flow Termination (Pounds Mass)	Break Flow Termination to Cut-in of RNS (Pounds Mass)
Ruptured steam generator – Atmosphere	267,190	83,800
Intact steam generator – Atmosphere	197,000	1,019,000
Break flow	452,761	0

Note:
1. RNS = normal residual heat removal

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Table 15.6.3-3
Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture

Reactor coolant iodine activity	
– Accident initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 335 (see Appendix 15A). Duration of spike is 8.0 hours.
– Pre-accident spike	An assumed iodine spike that results in an increase in the reactor coolant activity to 60 μCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	280 μCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Based on design basis activity (see Table 15A-9)
Secondary coolant initial iodine and alkali metal	1% of reactor coolant concentrations at maximum equilibrium conditions
Reactor coolant mass (lb)	3.7 E+05
Offsite power	Lost on reactor trip
Condenser	Lost on reactor trip
Time of reactor trip	Beginning of the accident
Duration of steam releases (hr)	16
Atmospheric dispersion factors	See Appendix 15A
Nuclide data	See Appendix 15A
Steam generator in ruptured loop	
– Initial secondary coolant mass (lb)	1.17 E+05
– Primary-to-secondary break flow	See Figure 15.6.3-5
– Integrated flashed break flow (lb)	See Figure 15.6.3-10
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02[a]
– Alkali metals partition coefficient	3.5 E-03[a]
Steam generator in intact loop	
– Initial secondary coolant mass (lb)	2.30 E+04
– Primary-to-secondary leak rate (lb/hr)	52.16[b]
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02[a]
– Alkali metals partition coefficient	3.5 E-03[a]

Notes:
a. Iodine partition coefficient does not apply to flashed break flow.
b. Equivalent to 150 gpd at psia cooled liquid at 62.4 lb/ft^3.

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Table 15.6.5-1
Core Activity Releases to the Containment Atmosphere

Nuclide	Gap Release Released over 0.5 hr. (0.167 - 0.667 hr)(1)	Core Melt In-vessel Release (0.667 - 1.967 hr)(1)
Noble gases	0.05	0.95
Iodines	0.05	0.35
Alkali metals	0.05	0.25
Tellurium group	–	0.05
Strontium and barium	–	0.02
Noble metals group	–	0.0025
Cerium group	–	0.0005
Lanthanide group	–	0.0002

Notes:
1. Releases are stated as fractions of the original core fission product inventory.
2. Dash (–) indicates not applicable.

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Table 15.6.5-2 (Sheet 1 of 3)
Assumptions and Parameters Used in Calculating
Radiological Consequences of a Loss-of-Coolant Accident

Primary coolant source data	
– Noble gas concentration	280 μCi/g dose equivalent Xe-133
– Iodine concentration	1.0 μCi/g dose equivalent I-131
– Primary coolant mass (lb)	4.39 E+05
Containment purge release data	
– Containment purge flow rate (cfm)	16,000
– Time to isolate purge line (seconds)	30
– Time to blow down the primary coolant system (minutes)	10
– Fraction of primary coolant iodine that becomes airborne	1.0
Core source data	
– Core activity at shutdown	See Table 15A-3
– Release of core activity to containment atmosphere (timing and fractions)	See Table 15.6.5-1
– Iodine species distribution (%)	
• Elemental	4.85
• Organic	0.15
• Particulate	95
Containment leakage release data	
– Containment volume (ft^3)	2.06 E+06
– Containment leak rate, 0-24 hr (% per day)	0.10
– Containment leak rate, > 24 hr (% per day)	0.05
– Elemental iodine deposition removal coefficient (hr^{-1})	1.9
– Decontamination factor limit for elemental iodine removal	200
– Removal coefficient for particulates (hr^{-1})	See Appendix 15B
Main control room model	
– Main control room gross volume (ft^3)	3.89 E+04
– Control support area gross volume (ft^3)	7.56 E+04
– Normal HVAC operation (prior to switchover to an emergency mode)	
• Air intake flow (cfm)	1650
• Filter efficiency	Not applicable
– Atmospheric dispersion factors (sec/m^3)	See Table 15A-6

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Table 15.6.5-2 (Sheet 2 of 3)
Assumptions and Parameters Used in Calculating
Radiological Consequences of a Loss-of-Coolant Accident

Main control room model (cont.)	
– Occupancy	
• 0 –24 hr	1.0
• 24–96 hr	0.6
• 96–720 hr	0.4
– Breathing rate (m^3/sec)	3.5 E-04
Control room with emergency habitability system credited (VES Credited)	
– Main control room activity level at which the emergency habitability system actuation is actuated (Ci/m^3 of dose equivalent I-131)	2.0 E-07
– Response time to actuate VES based on radiation monitor response time and VBS isolation (sec)	200
– Interval with operation of the emergency habitability system	
• Flow from compressed air bottles of the emergency habitability system (cfm)	60
• Unfiltered inleakage via ingress/egress (scfm)	5
• Unfiltered inleakage from other sources (scfm)	10
• Recirculation flow through filters (scfm)	600
• Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99
– Time at which the compressed air supply of the emergency habitability system is depleted (hr)	72
– After depletion of emergency habitability system bottled air supply (>72 hr)	
• Air intake flow (cfm)	1900
• Intake flow filter efficiency (%)	Not applicable
• Recirculation flow (cfm)	Not applicable
– Time at which the compressed air supply is restored and emergency habitability system returns to operation (hr)	168

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Table 15.6.5-2 (Sheet 3 of 3)
Assumptions and Parameters Used in Calculating
Radiological Consequences of a Loss-of-Coolant Accident

Control room/CSA with credit for continued operation of HVAC (VBS Supplemental Filtration Mode Credited)	
– Time delay to switch from normal operation to the supplemental air filtration mode (sec)	265
– Unfiltered inleakage via ingress/egress (cfm)	10
– Unfiltered inleakage from other sources (cfm)	50
– Filtered air intake flow (cfm)	800
– Filtered air recirculation flow (cfm)	3200
– Filter efficiency (%)	
• Elemental iodine	99
• Organic iodine	99
• Particulates	99
Miscellaneous assumptions and parameters	
– Offsite power	Not applicable
– Atmospheric dispersion factors (offsite)	See Table 15A-5
– Nuclide dose conversion factors	See Table 15A-4
– Nuclide decay constants	See Table 15A-4
– Offsite breathing rate (m^3/sec)	
0–8 hr	3.5 E-04
8–24 hr	1.8 E-04
24–720 hr	2.3 E-04

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Table 15.6.5-3
Radiological Consequences of a
Loss-of-Coolant Accident With Core Melt

	TEDE Dose (rem)
Exclusion zone boundary dose (1.3 - 3.3 hr)(1)	23.5
Low population zone boundary dose (0 - 30 days)	22.2
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	3.70
– Direct radiation from adjacent structures, including sky shine	0.30
– Filter shine	0.32
– Spent fuel pool boiling	0.01
– Total	4.33
Main control room dose (normal HVAC operating in the supplemental filtration mode)	
– Airborne activity entering the main control room	4.50
– Direct radiation from adjacent structures, including sky shine	0.30
– Filter shine	0.03
– Spent fuel pool boiling	0.01
– Total	4.84

Note:
1. This is the 2-hour period having the highest dose.

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Table 15.6.5-4
Major Plant Parameter Assumptions
Used in the Best-Estimate Large-Break LOCA Analysis

Parameter	Value
Plant Physical Configuration	
Steam generator tube plugging level	≤ 10% (10% tube plugging bounds 0%)
Hot assembly location	Under support column (Bounds under open hole or guide tube)
Pressurizer location	In intact loop (Bounds location in broken loop)
Initial Operating Conditions	
Reactor power	Core power < 1.01*3400 MWt
Peak linear heat rate	See Table 15.6.5-7
Hot rod assembly power	See Table 15.6.5-7
Hot assembly power	$P_{HA} \leq 1.654$
Axial power distribution(1)	See Figure 15.6.5.4A-13
Peripheral assembly power	$0.2 \leq P_{LOW} \leq 0.8$
Fluid Conditions	
Reactor coolant system average temperature	$573.6 - 8.0°F \leq T_{AVG} \leq 573.6 + 8.0°F$
Pressurizer pressure	2250 ± 50 psia
Pressurizer level (water volume)	1000 ft^3 (nominal)
Accumulator temperature	$50°F \leq T_{ACC} \leq 120°F$
Accumulator pressure	$652 \text{ psia} \leq P_{ACC} \leq 784 \text{ psia}$
Accumulator water volume	$1666.8 \text{ ft}^3 \leq V_{ACC} \leq 1732.3 \text{ ft}^3$
Reactor Coolant System Boundary Conditions	
Single failure assumption	Failure of one CMT isolation valve to open
Offsite power availability	Available (Bounds loss of offsite power at time zero)
Reactor coolant pump automatic trip delay time after receiving S-signal	5.3 s
Containment pressure	Bounded (minimum)

Note:
1. Treatment of axial power distribution consistent with WCAP-16009-P-A (Reference 32) methodology.

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Table 15.6.5-5
AP1000 LOCA Chronology

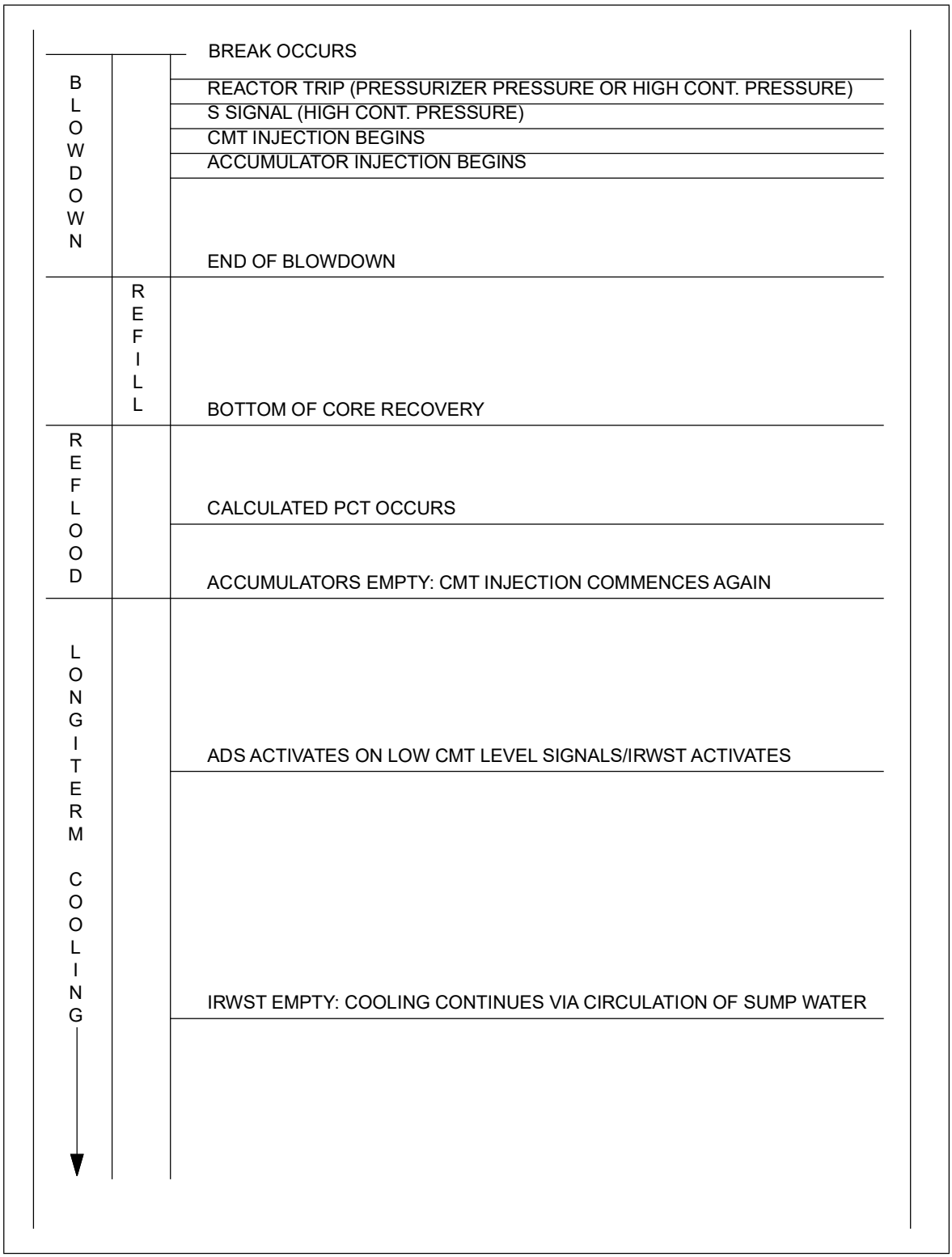

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Table 15.6.5-6
Best-Estimate Large-Break Sequence of Events for the Limiting PCT Case

Event	Time (seconds)
Break initiation	0.0
Safeguards signal	2.2
CMT isolation valves begin to open	4.2
Reactor coolant pumps trip	9.5
Accumulator injection begins	~13
End of blowdown	27.5
Bottom of core recovery	39.5
Calculated PCT occurs	~58
Core quench occurs	~240
CMT injection resumes	~200
End of transient	265

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Table 15.6.5-7
Summary of Peaking Factor Burndown Supported by AP1000 Plant Best Estimate Large Break LOCA CRR Updated Analysis Considering TCD

Hot Rod Burnup (GWd/MTU)	Fdh (includes uncertainties)(1)	FQ Transient (Max FQ, includes uncertainties)	FQ SS Baseload (without uncertainties)
0	1.72	2.60	2.10
30	1.72	2.60	2.10
49	1.55	2.30	1.85
55	1.55	2.30	1.85
62	1.40	1.90	1.45

Note:
1. Hot assembly power follows the same burndown, since it is a function of FdH.

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Table 15.6.5-8
Best-Estimate Large-Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
Calculated 95th percentile PCT (°F)	1936*	≤ 2200
Maximum local cladding oxidation (%)	4.2	≤ 17
Maximum core-wide cladding oxidation (%)	0.30	≤ 1
Coolable geometry	Core remains coolable	Core remains coolable
Long-term cooling	Core remains cool in long term	Core remains cool in long term

*Value contains 2°F bias for PCT sensitivity to PRHR isolation

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Table 15.6.5-9
Initial Conditions for AP1000 Small-Break LOCA Analysis

Condition	Calculation	Nominal Steady-state
Pressurizer pressure (psia)	2300.1	2300
Vessel inlet temperature (°F)	534.03	534.59
Vessel outlet temperature (°F)	612.16	612.61
Vessel flow rate (lbm/sec)	31118	31118
Steam generator pressure (psia)	794.76	794.59

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Table 15.6.5-10
ADS Parameters Used in Small-Break LOCA Analyses

ADS Valve Actuation Signal [percentage of core makeup tank volume]		Actuation Time (seconds)	Minimum Valve Flow Area (for each path, in^2)	Number of Paths	Valve Opening Time (seconds)
Stage 1 — Control CMT Level Low-3	67.5	32 after CMT Level Low-3	4.6(3)	2 out of 2	≤ 48
Stage 2 — Control		48 after Stage 1	16(3)	2 out of 2	≤ 120
Stage 3 — Control		120 after Stage 2	16(3)	2 out of 2	≤ 120
Stage 4A — CMT Level Low-6	20	128 after Stage 3(1)	66	1 out of 2	≤ 4(2)
Stage 4B		60 after Stage 4A	66	2 out of 2	≤ 4(2)

Notes:
1. The interlock requires coincidence of CMT Level Low-6 as well as 128 seconds after the Stage 3 actuation signal is generated.
2. This includes “arm-fire” processing delay and the assumed valve opening time.
3. The areas listed for the ADS Stage 1/2/3 valves are an effective flow area.

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Table 15.6.5-11
Inadvertent ADS Depressurization Sequence of Events

Event	AP1000 Time (seconds)
Inadvertent opening of ADS valves	0.0
ADS Stage 2	48.0
Reactor trip signal	50.8
Steam turbine stop valves close	50.8
“S” signal	59.7
Main feed isolation valves close	66.7
Reactor coolant pumps start to coast down	66.7
ADS Stage 3	168.0
Accumulator injection starts	288
Accumulator tank empties (1 / 2)	753.1 / 751.8
ADS Stage 4	1664.5
Core makeup tank empties (1 / 2)	2020 / 2027
Core uncovery begins	2567
IRWST injection starts*	2854
Core uncovery ends	2905

Note:
*Continuous injection period

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Table 15.6.5-12
2-Inch Cold Leg Break Sequence of Events

Event	Time (seconds)
Break opens	0.0
Reactor trip signal	53.7
Steam turbine stop valves close	53.7
“S” signal	65.3
Main feed isolation valves close	72.3
Reactor coolant pumps start to coast down	72.3
ADS Stage 1	1202.8
ADS Stage 2	1250.8
Accumulator injection starts	1315
ADS Stage 3	1370.8
Accumulator tank empties (1 / 2)	1799.8 / 1798.2
ADS Stage 4	2228.6
Core makeup tank empty (1 / 2)	2577 / 2594
Core uncovery begins	2910
IRWST injection starts*	3147
Core uncovery ends	3215

Note:
*Continuous injection period

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Table 15.6.5-13
Double-Ended Direct Vessel Injection Line Break Sequence of Events – 20 psia Containment Backpressure

Event	Time (seconds)
Break opens	0.0
Reactor trip signal	13.4
Steam turbine stop valves close	13.4
“S” signal	19.8
Main feed isolation valves close	26.8
Reactor coolant pumps start to coast down	26.8
ADS Stage 1	181.4
ADS Stage 2	229.4
Accumulator injection starts	246.2
ADS Stage 3	349.4
ADS Stage 4	477.4
Accumulator empties (1 / 2)	453.9 / 590.6
Core makeup tank empties (1 / 2)	225.1 / 1861.2
Intact DVI line IRWST injection starts*	1868.3

Note:
* Continuous injection period

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Table 15.6.5-13A
Not Used

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Table 15.6.5-14
10-Inch Cold Leg Break Sequence of Events

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	5.5
Steam turbine stop valves close	5.5
“S” signal	6.7
Main feed isolation valves close	13.7
Reactor coolant pumps start to coast down	13.7
Accumulator injection starts	84
Accumulator tank empties (1 / 2)	477.4 / 488.6
ADS Stage 1	761.1
ADS Stage 2	809.9
ADS Stage 3	929.1
ADS Stage 4	1383.6
IRWST injection starts**	2746
Core makeup tank empties (1 / 2)	* / 2063

Note:
* The CMT does not empty.
** Continuous injection period.

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Table 15.6.5-15
Double-Ended Direct Vessel Injection Line Break Sequence of Events
(Entrainment Sensitivity)

Event	Time (seconds)
Break opens	0.0
Reactor trip signal	13.4
Steam turbine stop valves close	13.4
“S” signal	19.9
Main feed isolation valves close	26.8
Reactor coolant pumps start to coast down	26.8
ADS Stage 1	180.5
ADS Stage 2	228.5
Intact accumulator injection starts	245
ADS Stage 3	348.5
ADS Stage 4	476.5
Accumulator tank empties (1 / 2)	451.6 / 597.1
Intact loop IRWST injection starts*	2508
Core makeup tank empties (1 / 2)	223 / 1889

Note:
* Continuous injection period

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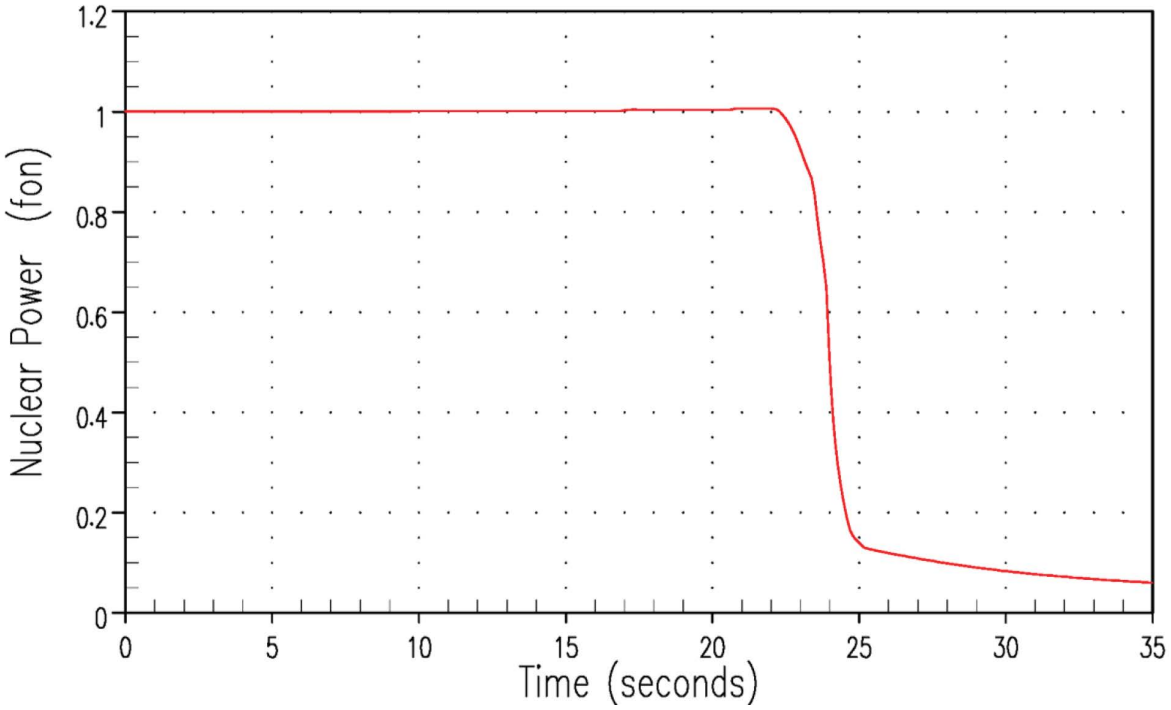

Figure 15.6.1-1
Nuclear Power Transient
Inadvertent Opening of a Pressurizer Safety Valve

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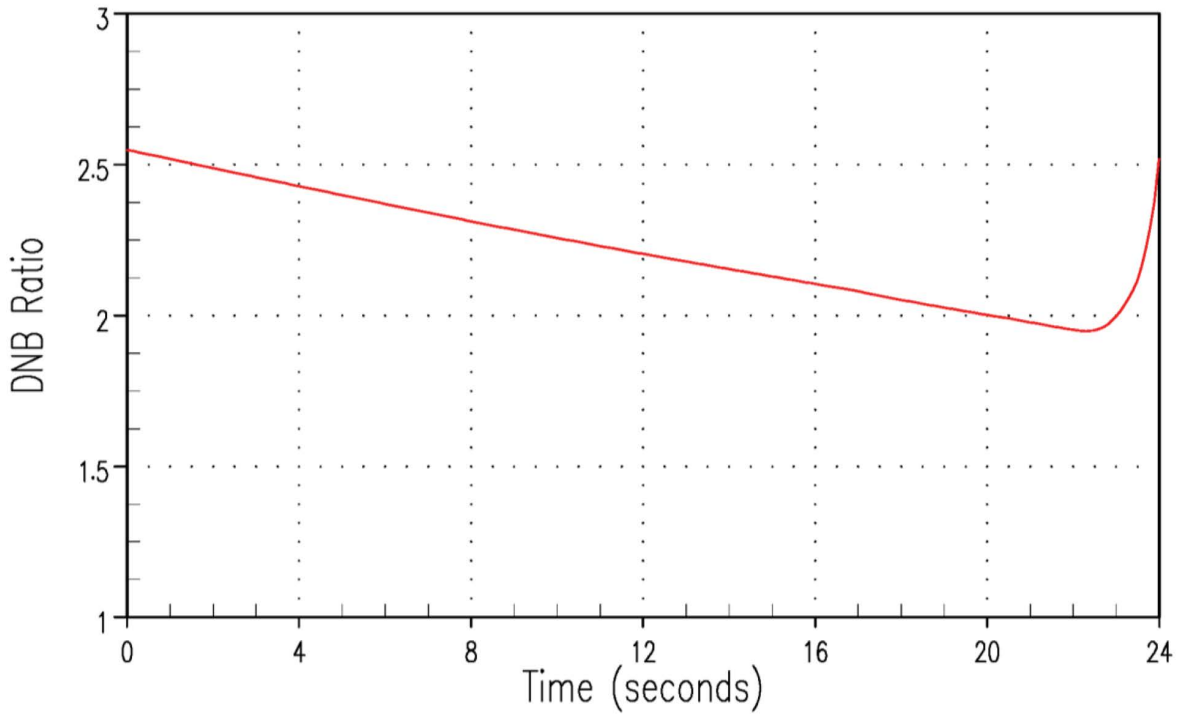

Figure 15.6.1-2
DNBR Transient
Inadvertent Opening of a Pressurizer Safety Valve

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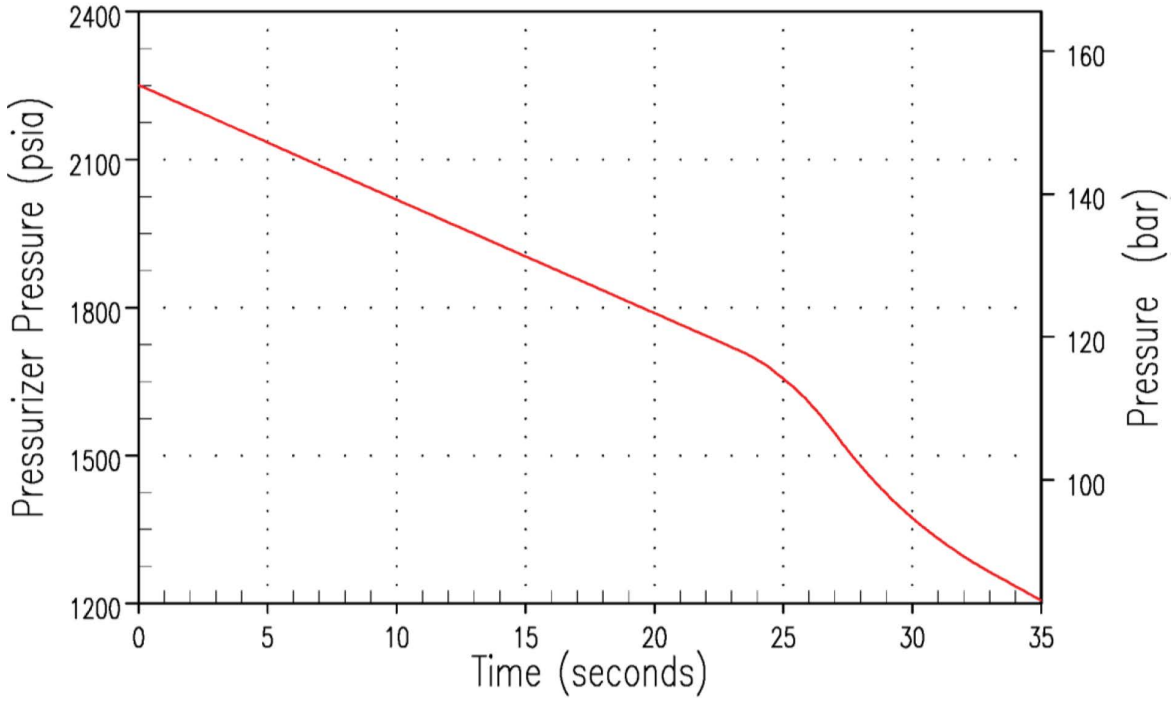

Figure 15.6.1-3
Pressurizer Pressure Transient
Inadvertent Opening of a Pressurizer Safety Valve

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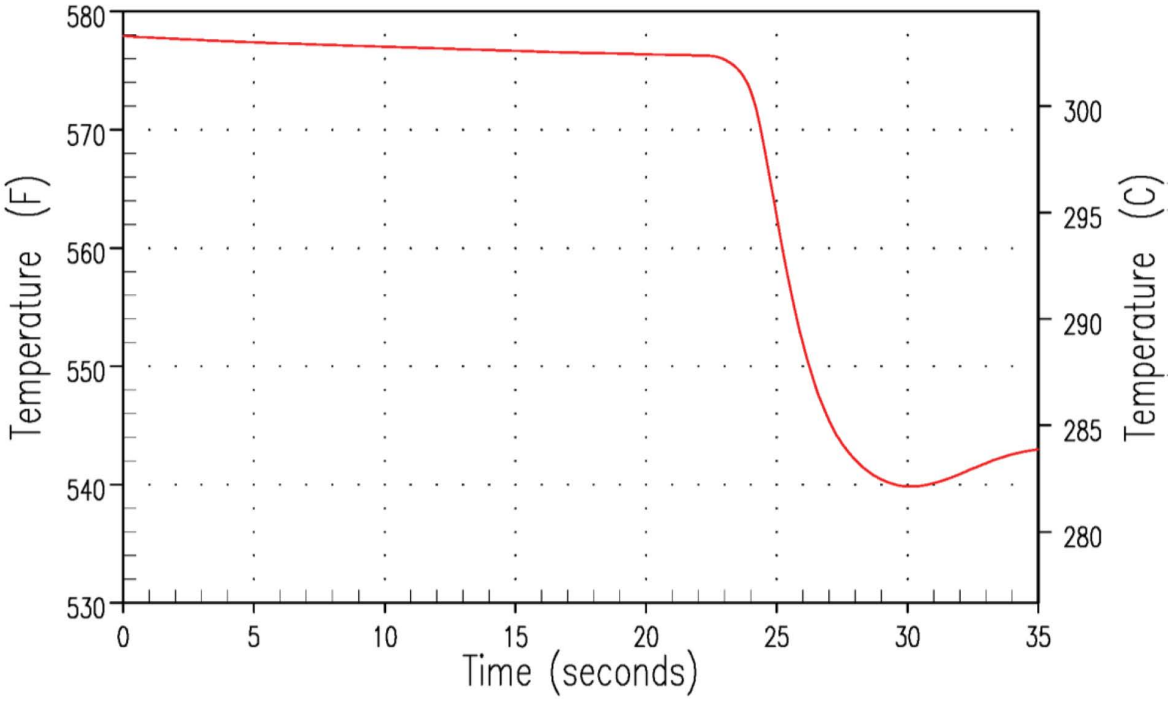

Figure 15.6.1-4
Core Average Temperature Transient
Inadvertent Opening of a Pressurizer Safety Valve

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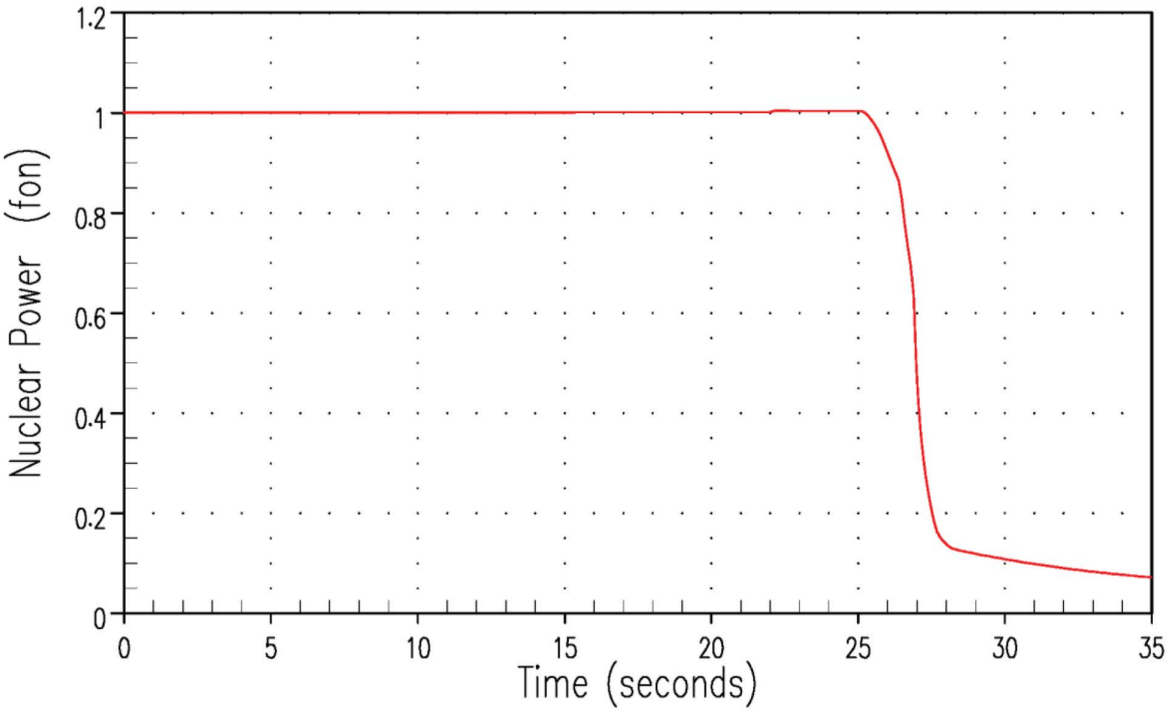

Figure 15.6.1-5
Nuclear Power Transient
Inadvertent Opening of an ADS Stage 2 or 3 Train

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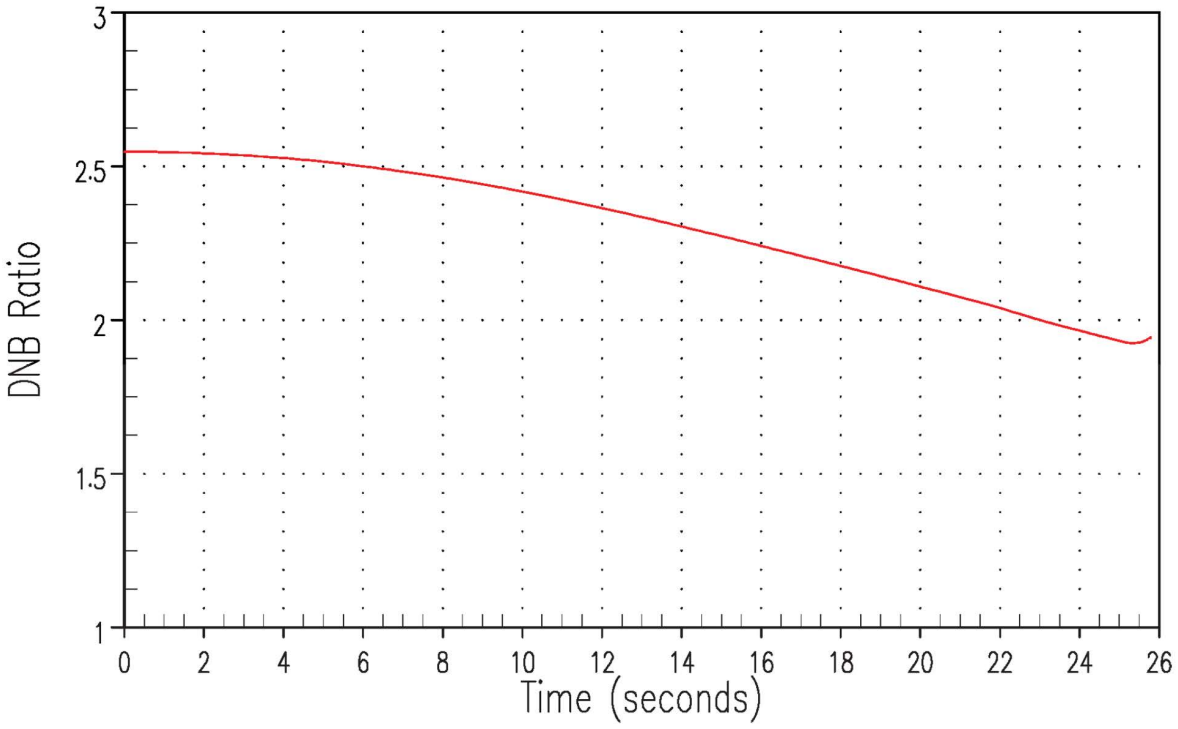

Figure 15.6.1-6
DNBR Transient
Inadvertent Opening of an ADS Stage 2 or 3 Train

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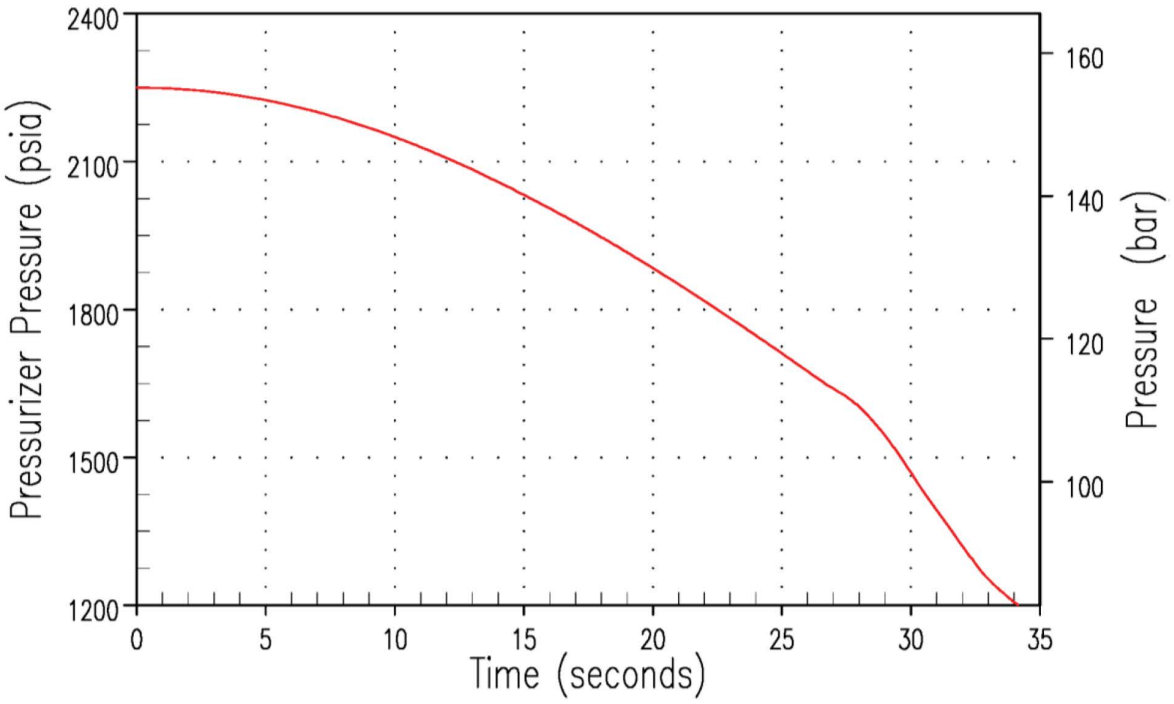

Figure 15.6.1-7
Pressurizer Pressure Transient
Inadvertent Opening of an ADS Stage 2 or 3 Train

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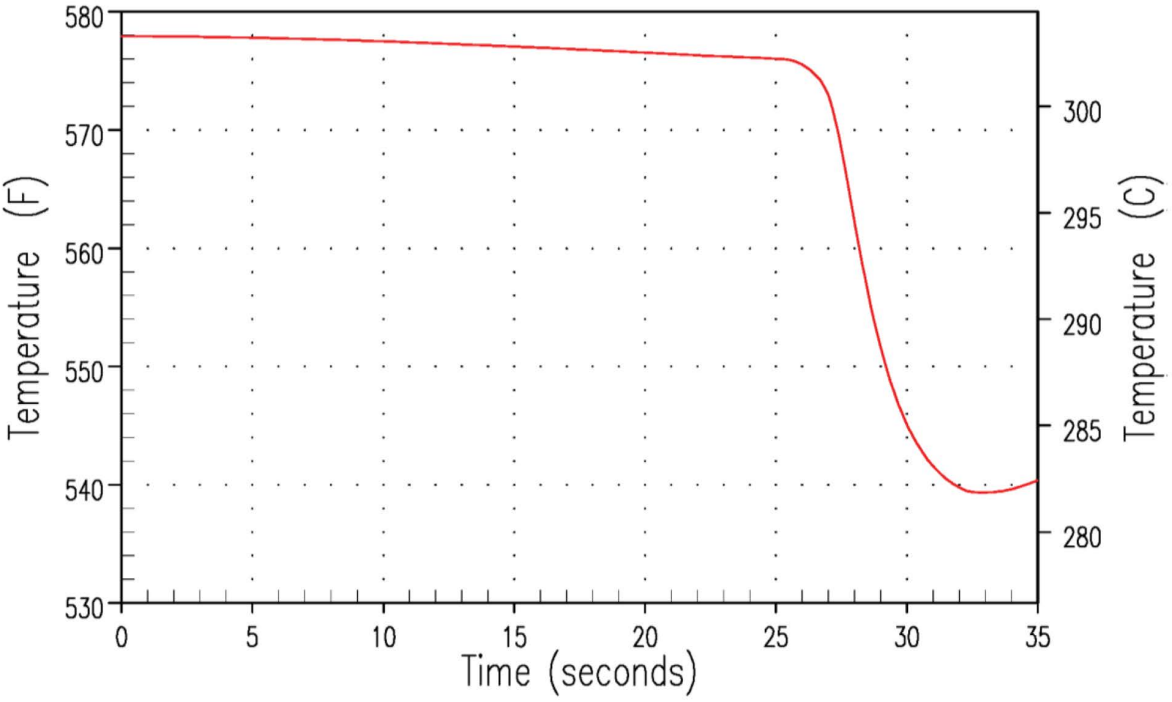

Figure 15.6.1-8
Core Average Temperature Transient
Inadvertent Opening of an ADS Stage 2 or 3 Train

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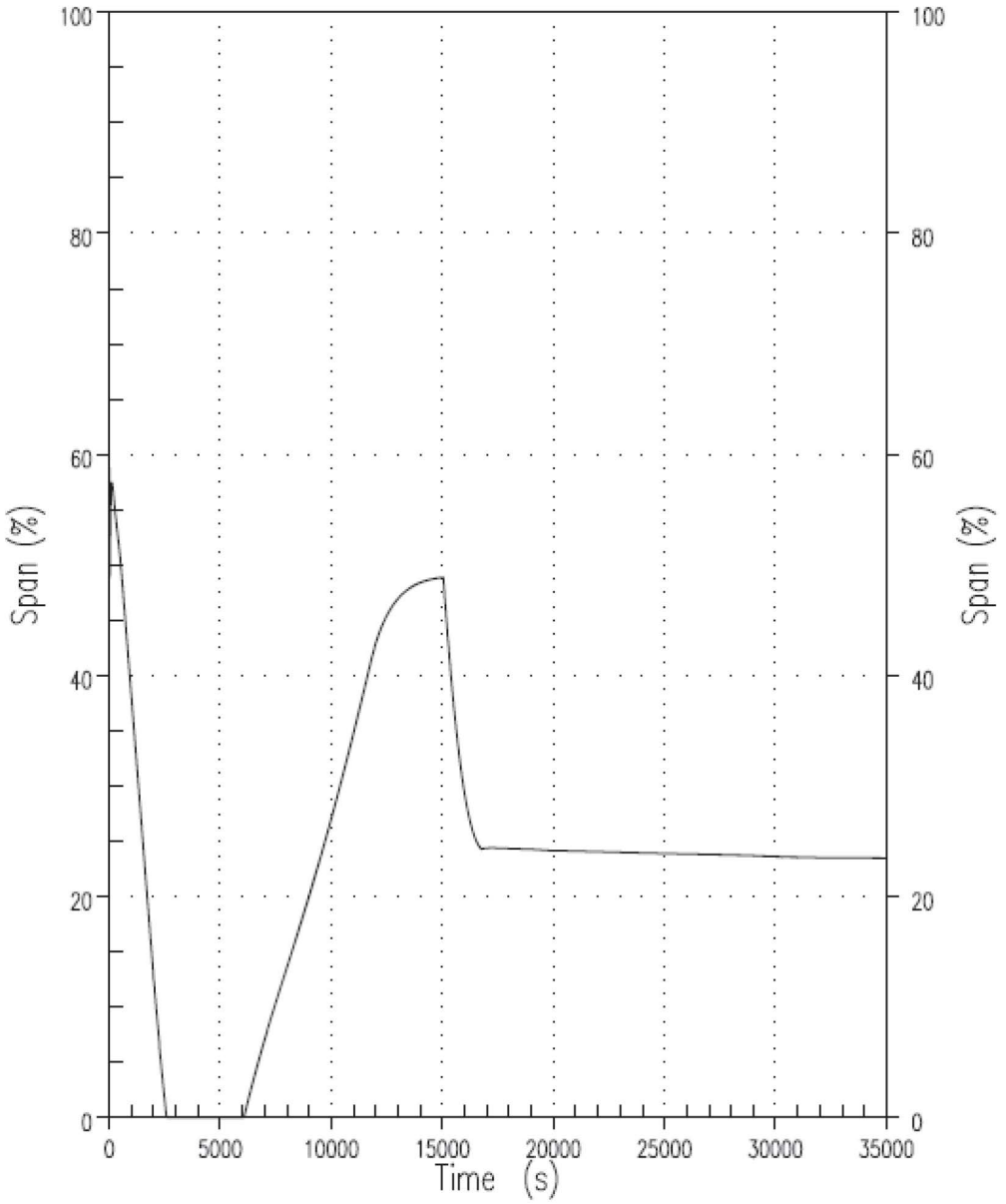

Figure 15.6.3-1
Pressurizer Level for SGTR

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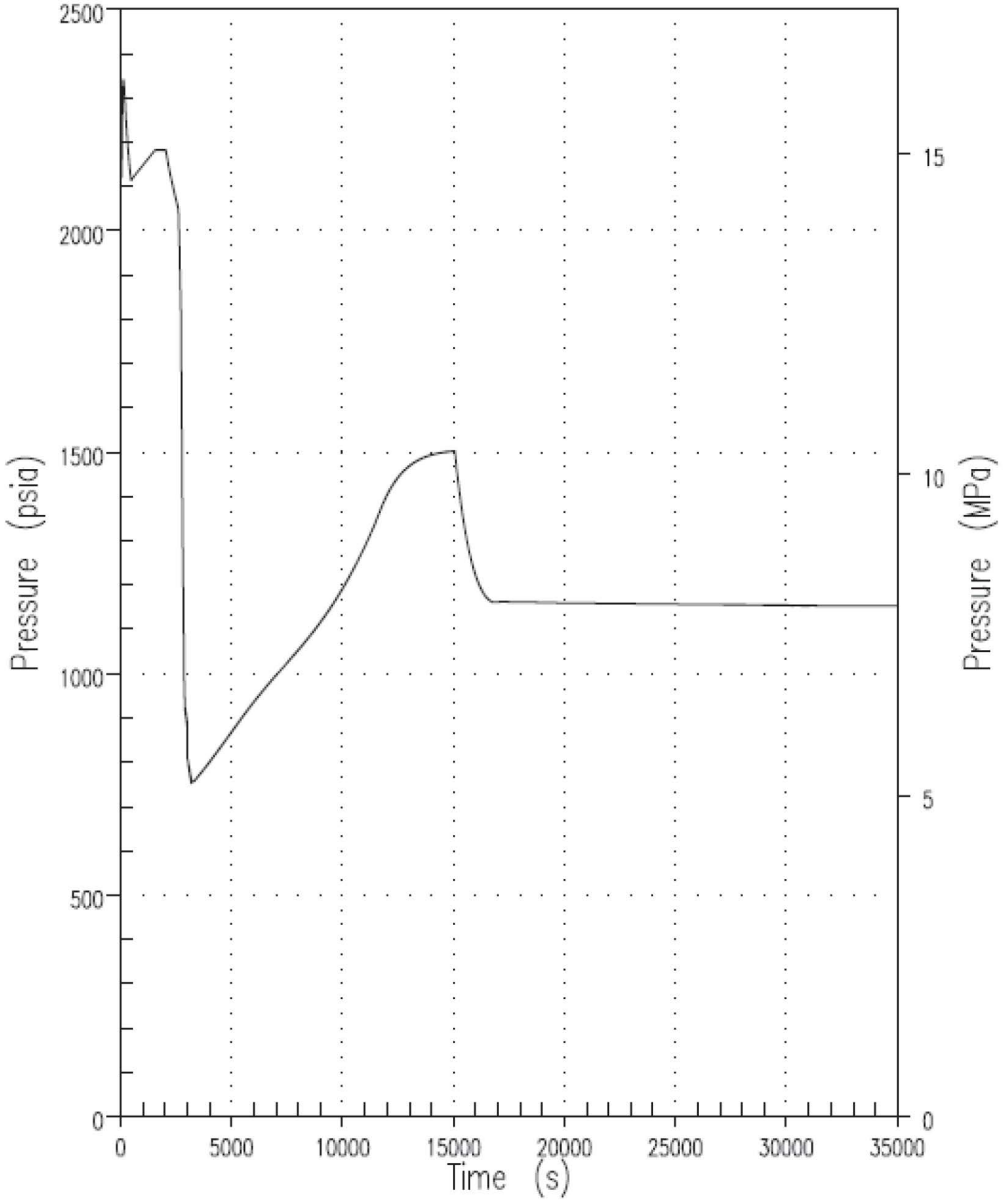

Figure 15.6.3-2
Reactor Coolant System Pressure for SGTR

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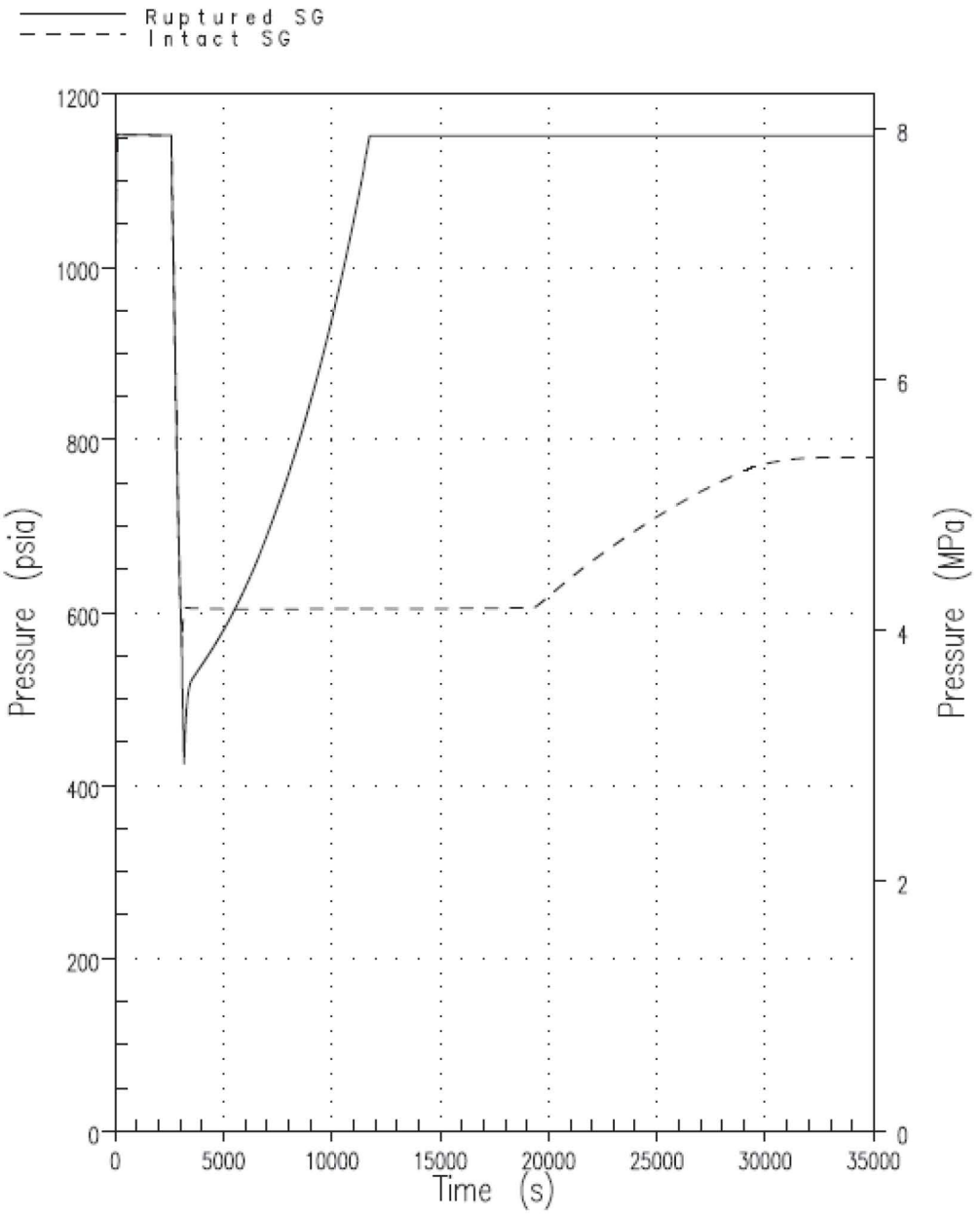

Figure 15.6.3-3
Secondary Pressure for SGTR

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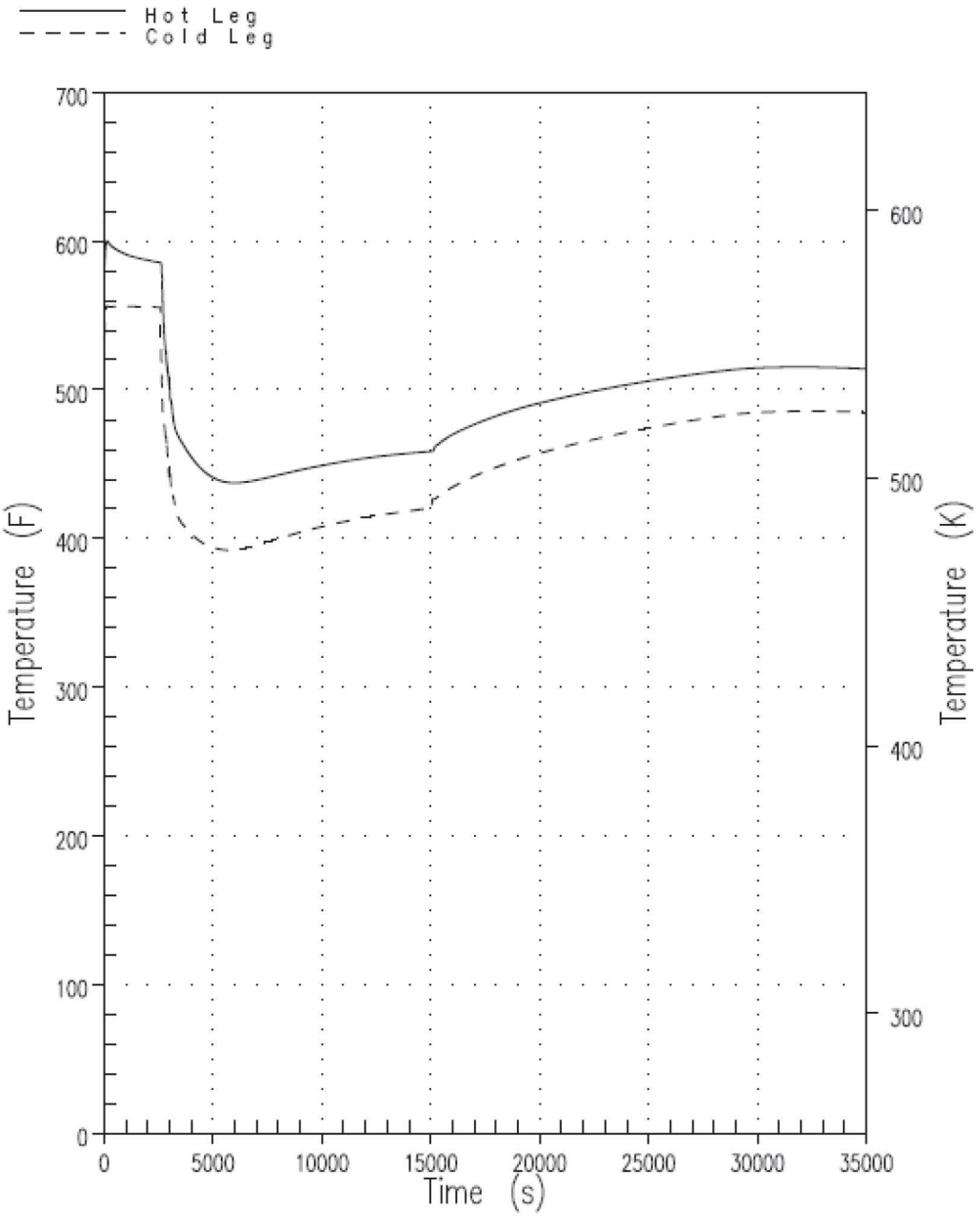

Figure 15.6.3-4
Intact Loop Hot and Cold Leg
Reactor Coolant System Temperature for SGTR

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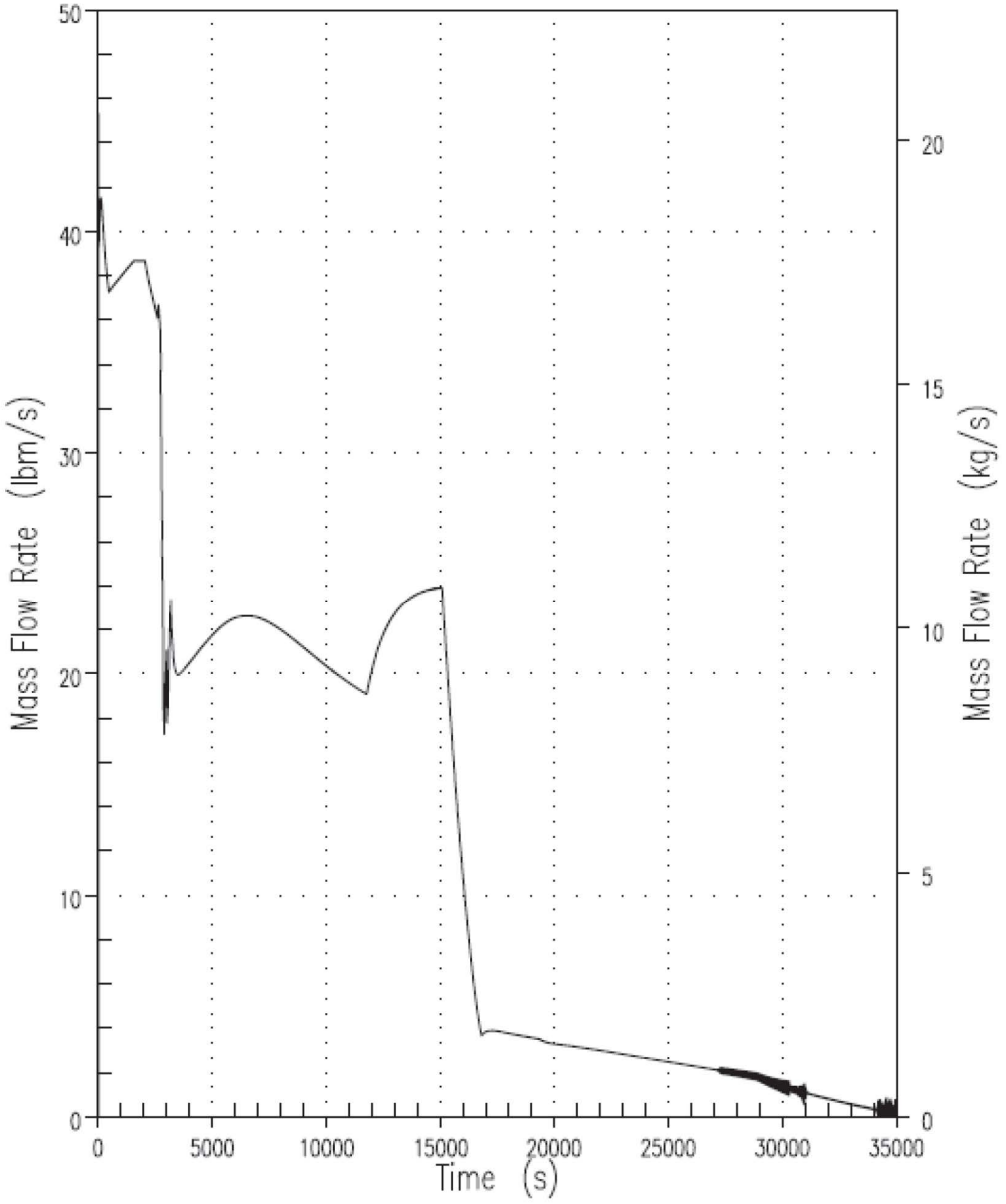

Figure 15.6.3-5
Primary-to-Secondary Break Flow Rate for SGTR

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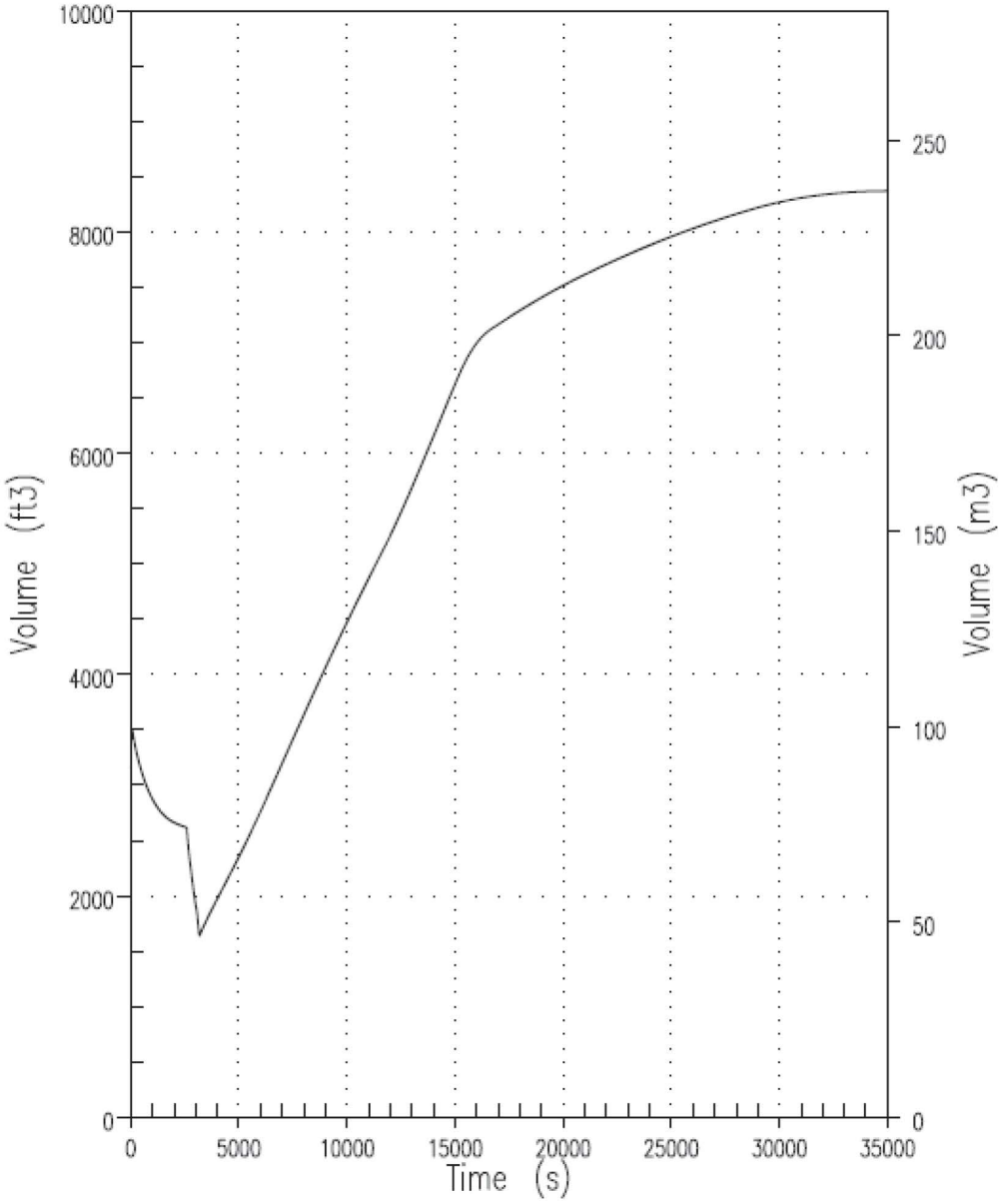

Figure 15.6.3-6
Ruptured Steam Generator Water Volume for SGTR

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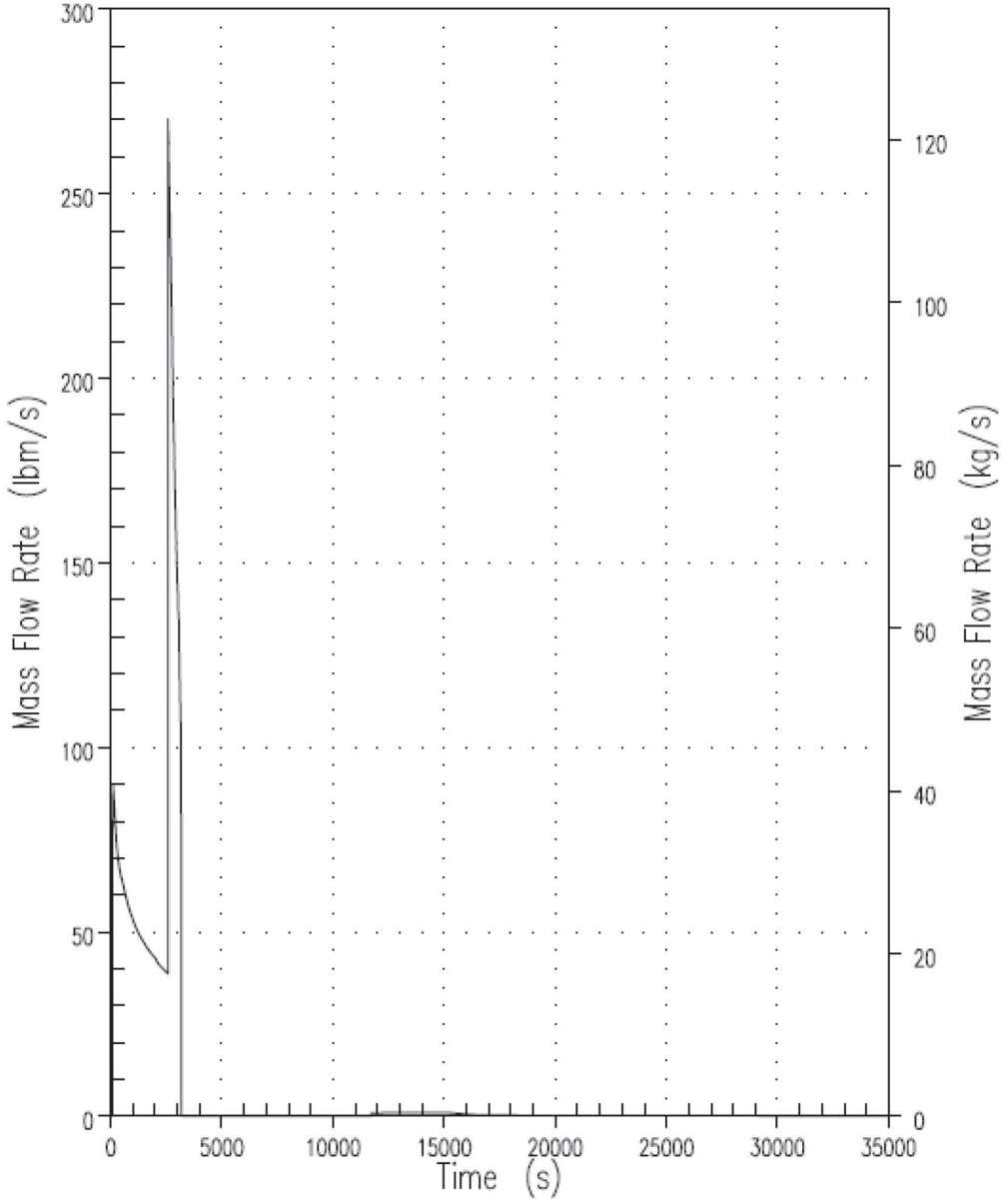

Figure 15.6.3-7
Ruptured Steam Generator Mass
Release Rate to the Atmosphere for SGTR

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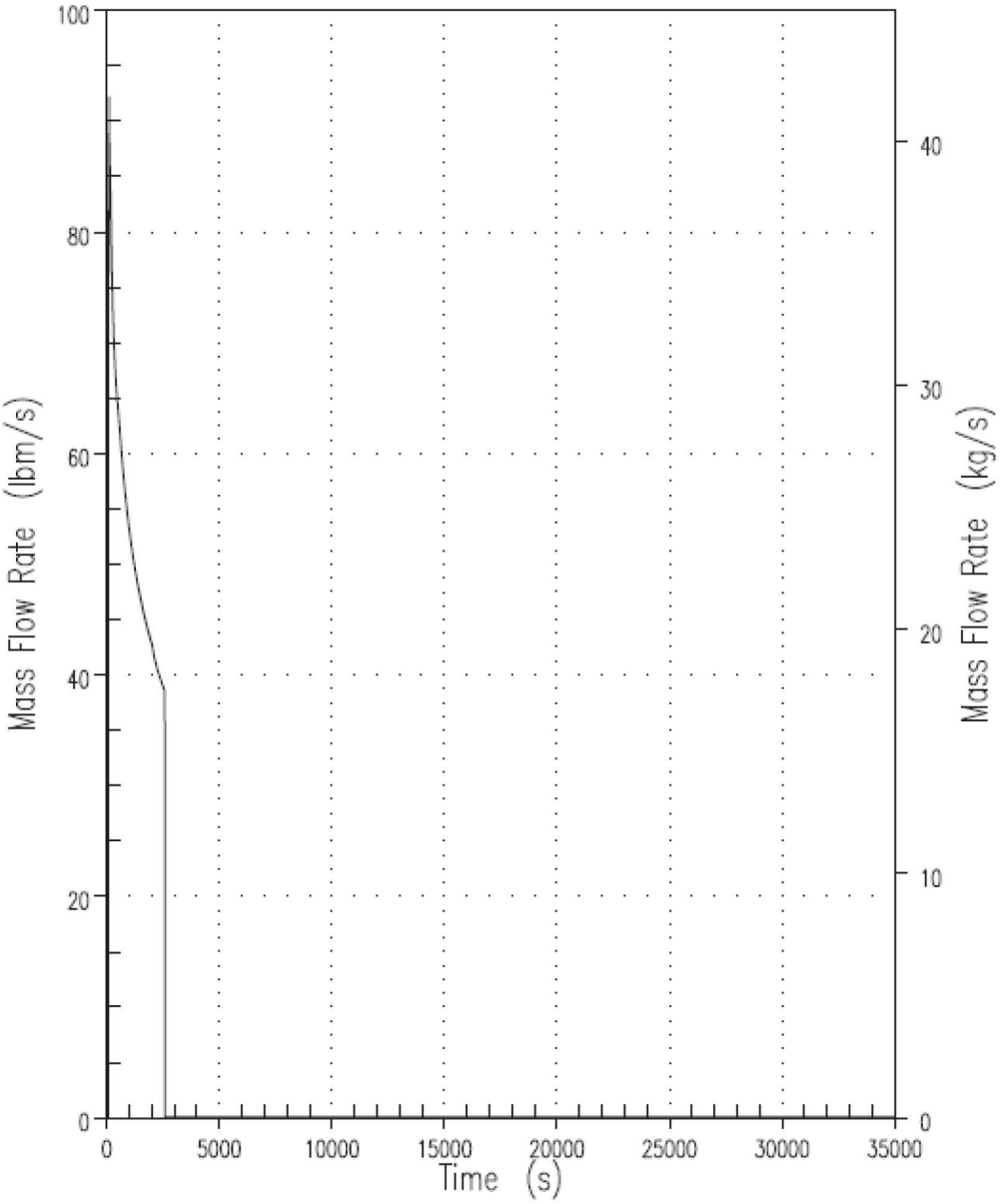

Figure 15.6.3-8
Intact Steam Generator Mass
Release Rate to the Atmosphere for SGTR

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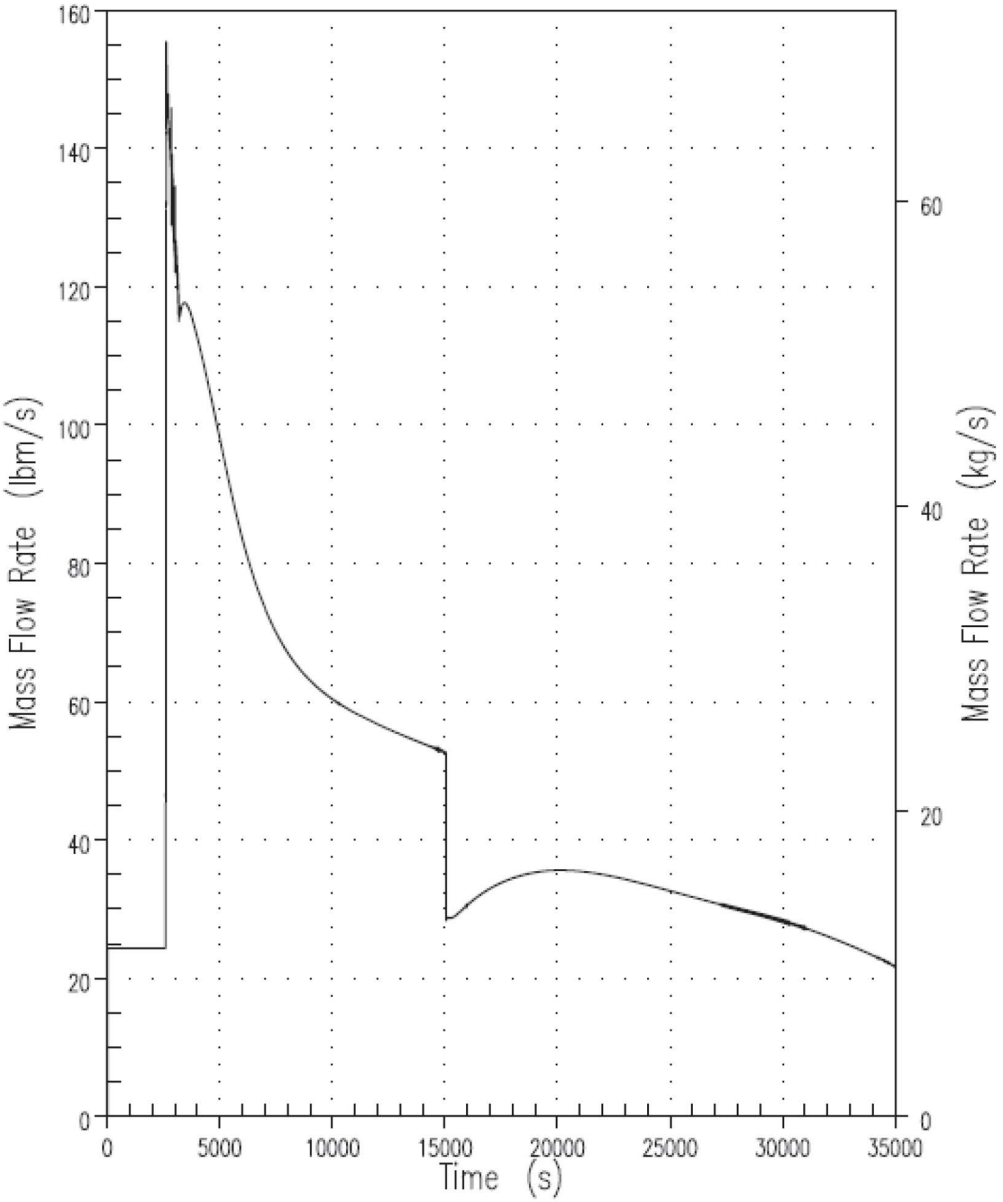

Figure 15.6.3-9
Plant Total Chemical and Volume Control System and Core Makeup Tank Injection Flow for SGTR

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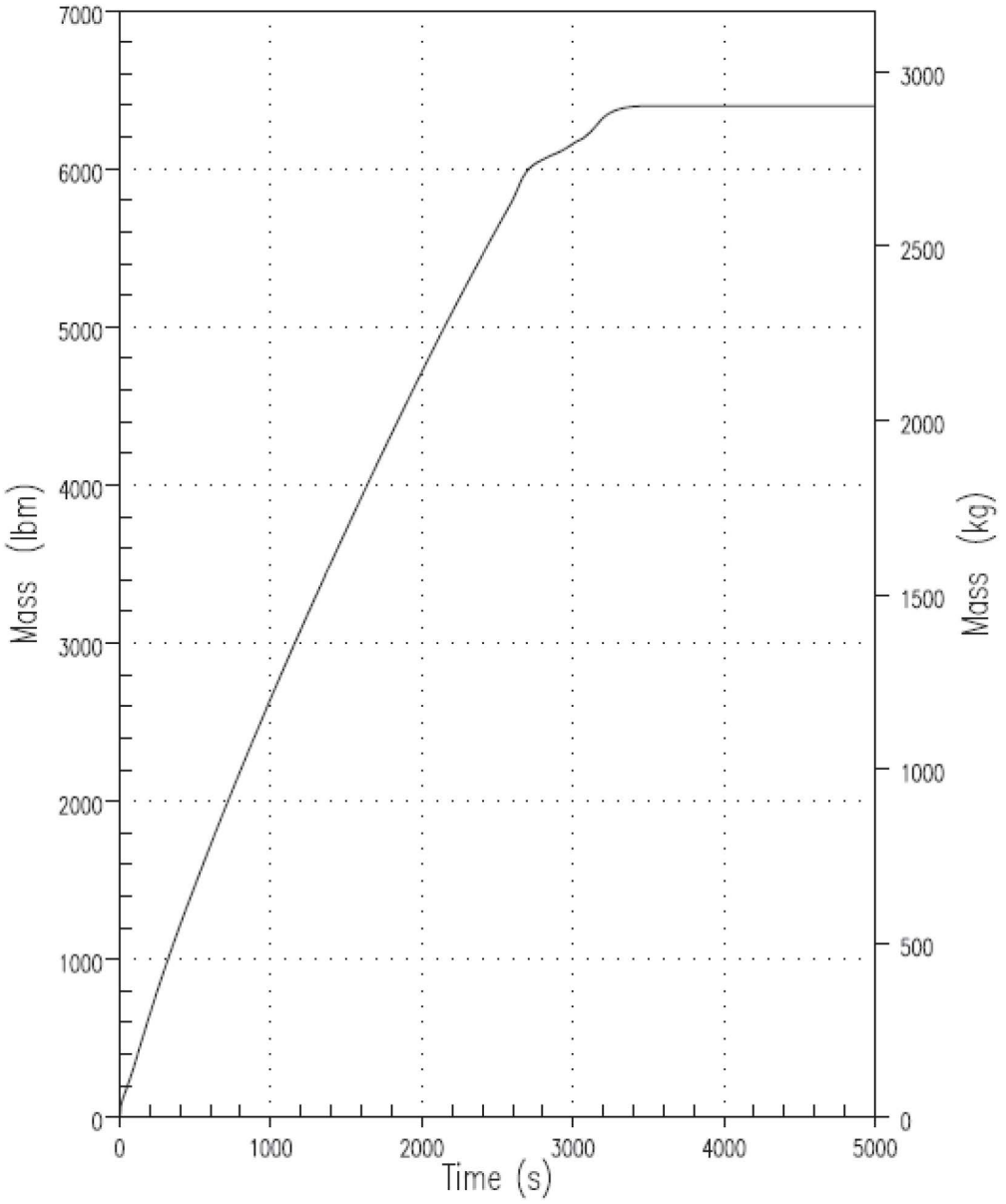

Figure 15.6.3-10
Integrated Flashed Break Flow for SGTR

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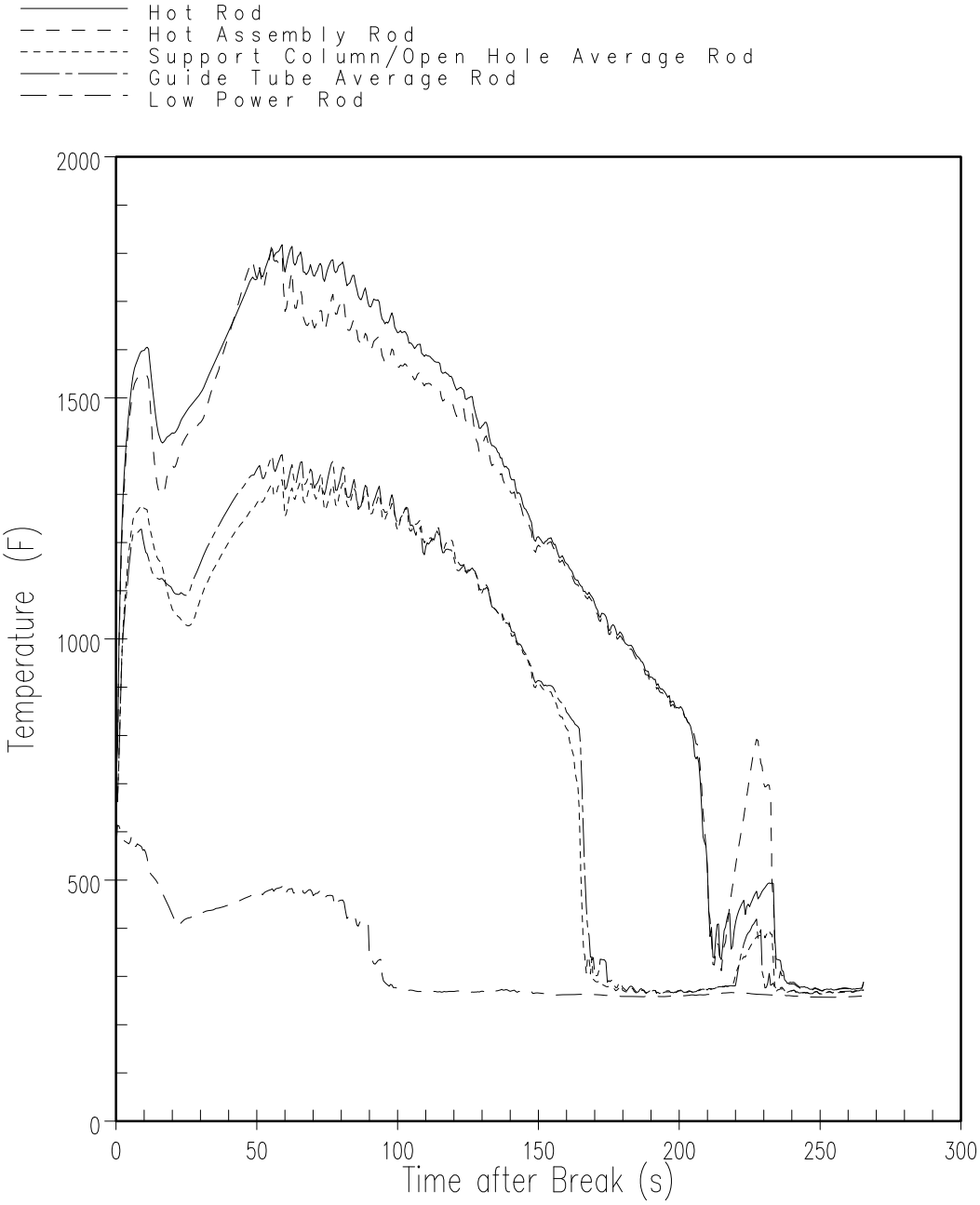

Figure 15.6.5.4A-1
WCOBRA/TRAC Peak Cladding Temperature for All Five Rod Groups for 95th Percentile Estimator PCT Case

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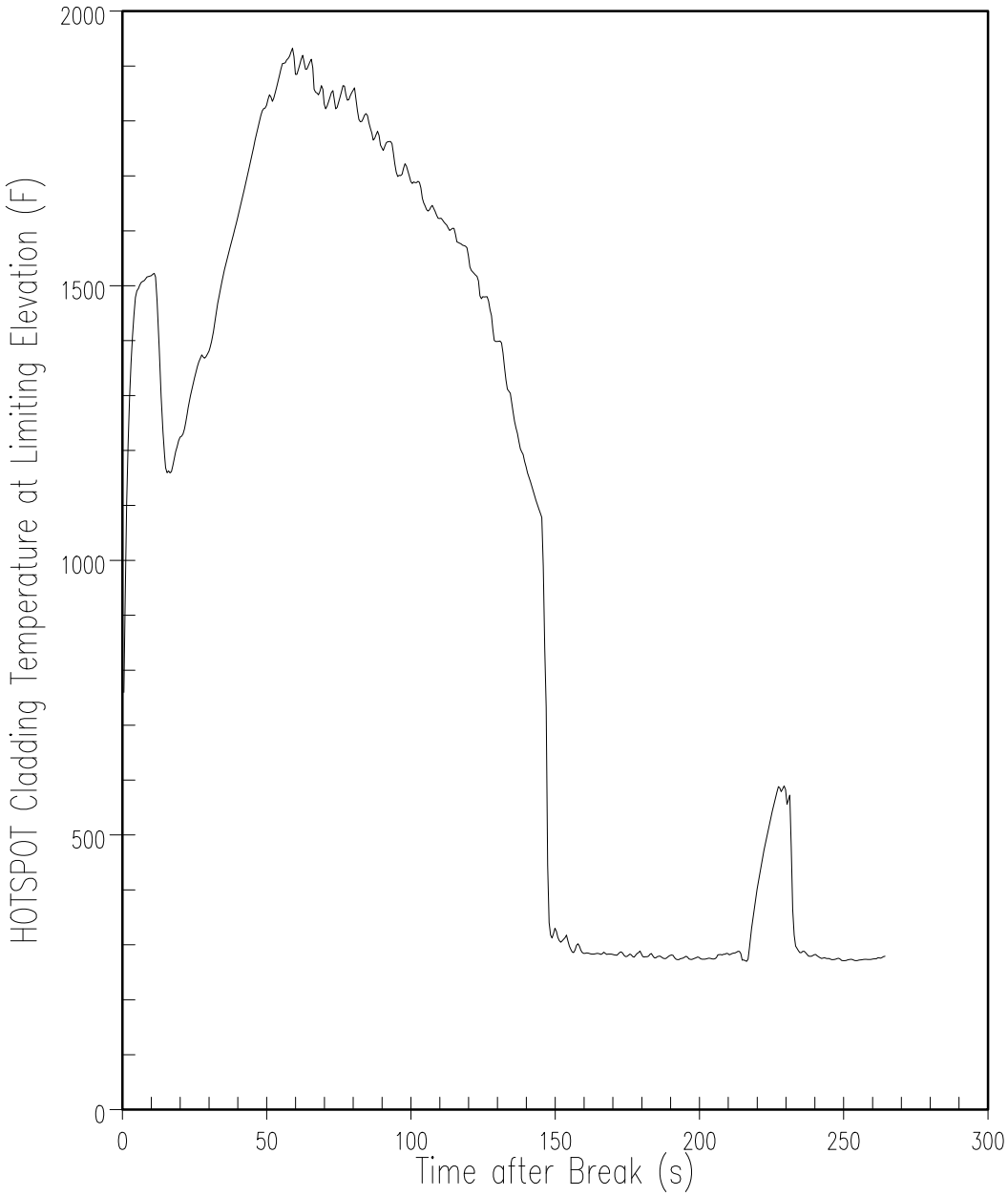

Figure 15.6.5.4A-2
HOTSPOT Cladding Temperature Transient at Limiting Elevation for 95th Percentile Estimator PCT Case

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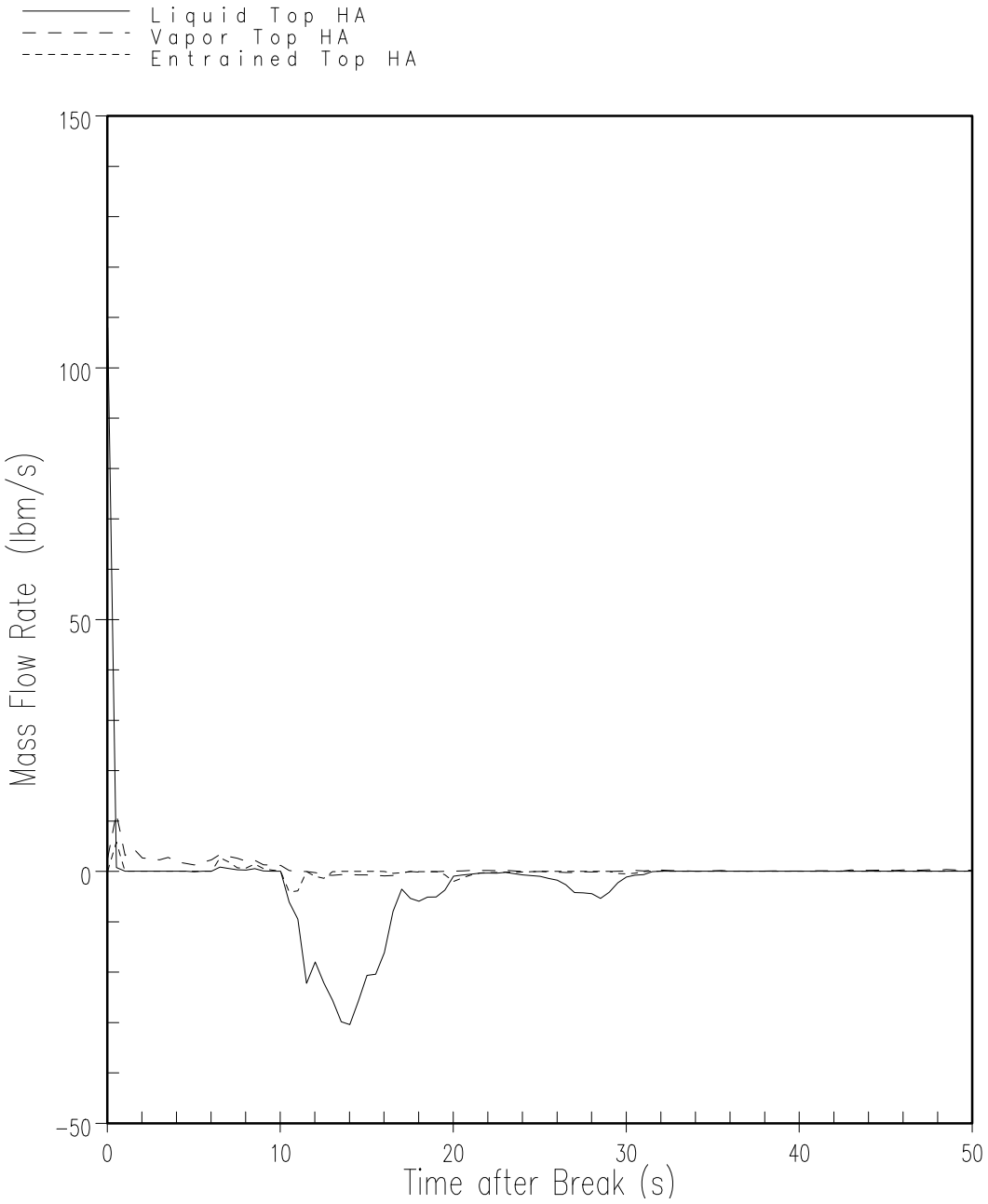

Figure 15.6.5.4A-3
Mass Flow at Top of Hot Assembly Channel
for 95th Percentile Estimator PCT Case

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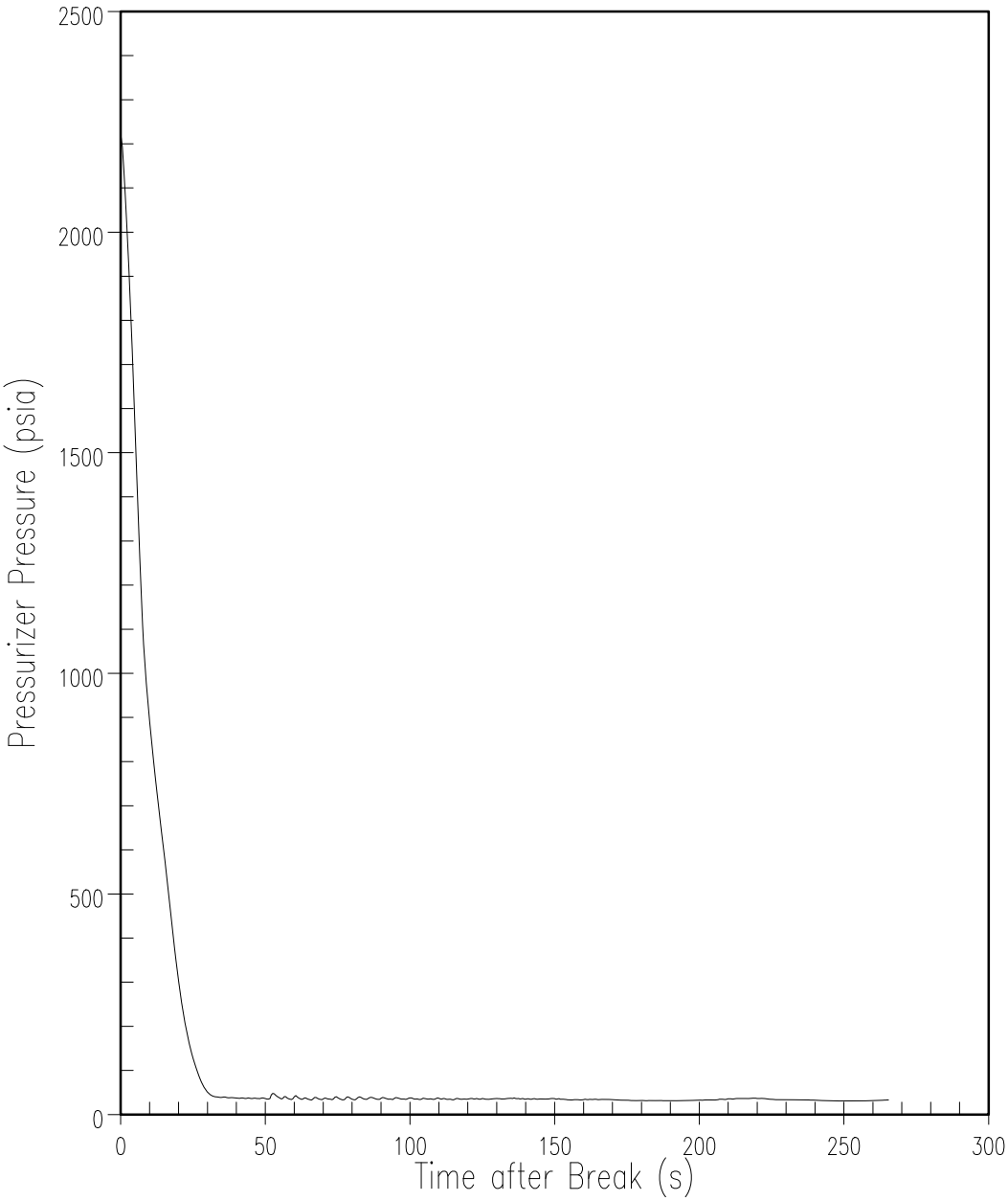

Figure 15.6.5.4A-4
Pressurizer Pressure for
95th Percentile Estimator PCT Case

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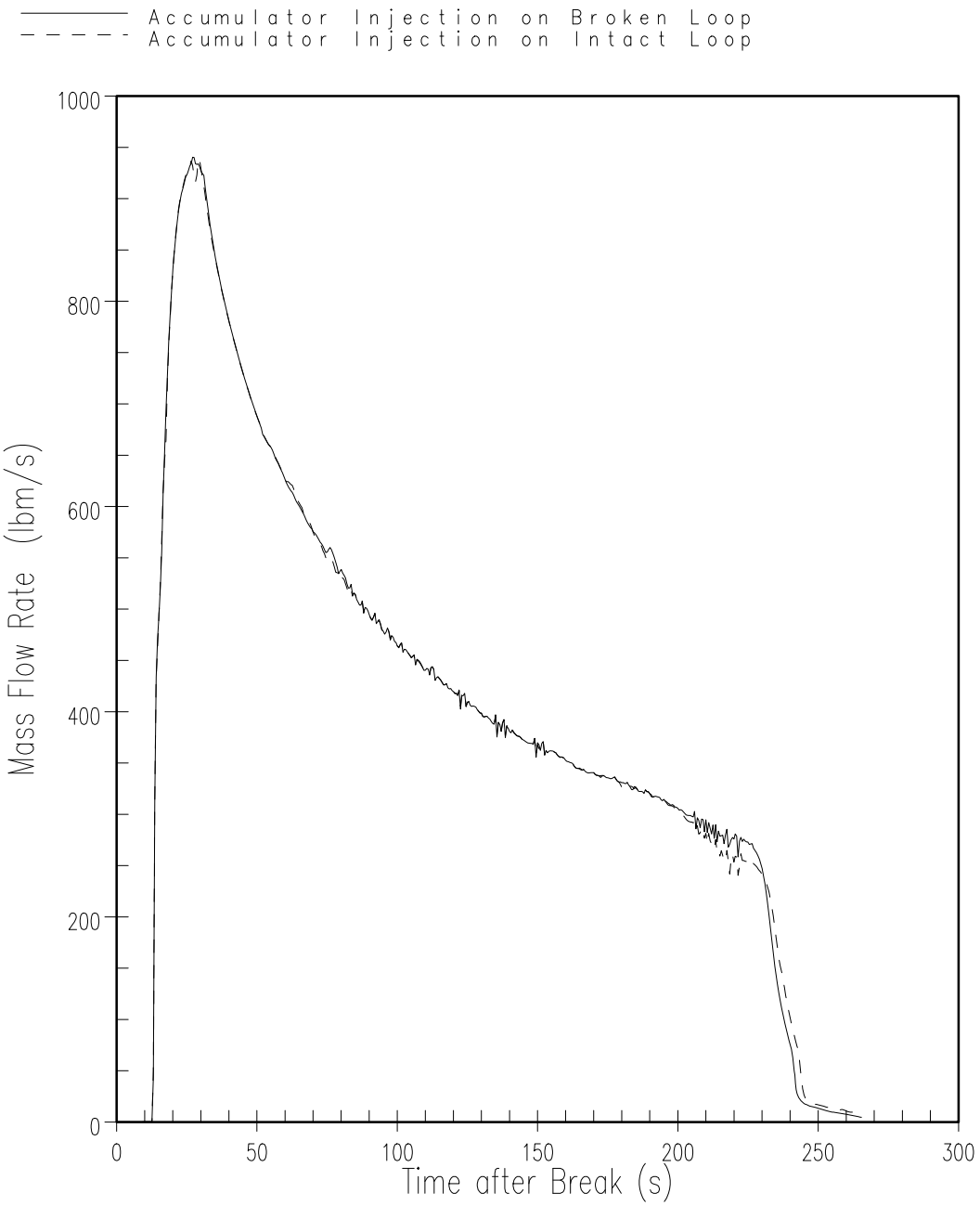

Figure 15.6.5.4A-5
Accumulator Injection Flow for
95th Percentile Estimator PCT Case

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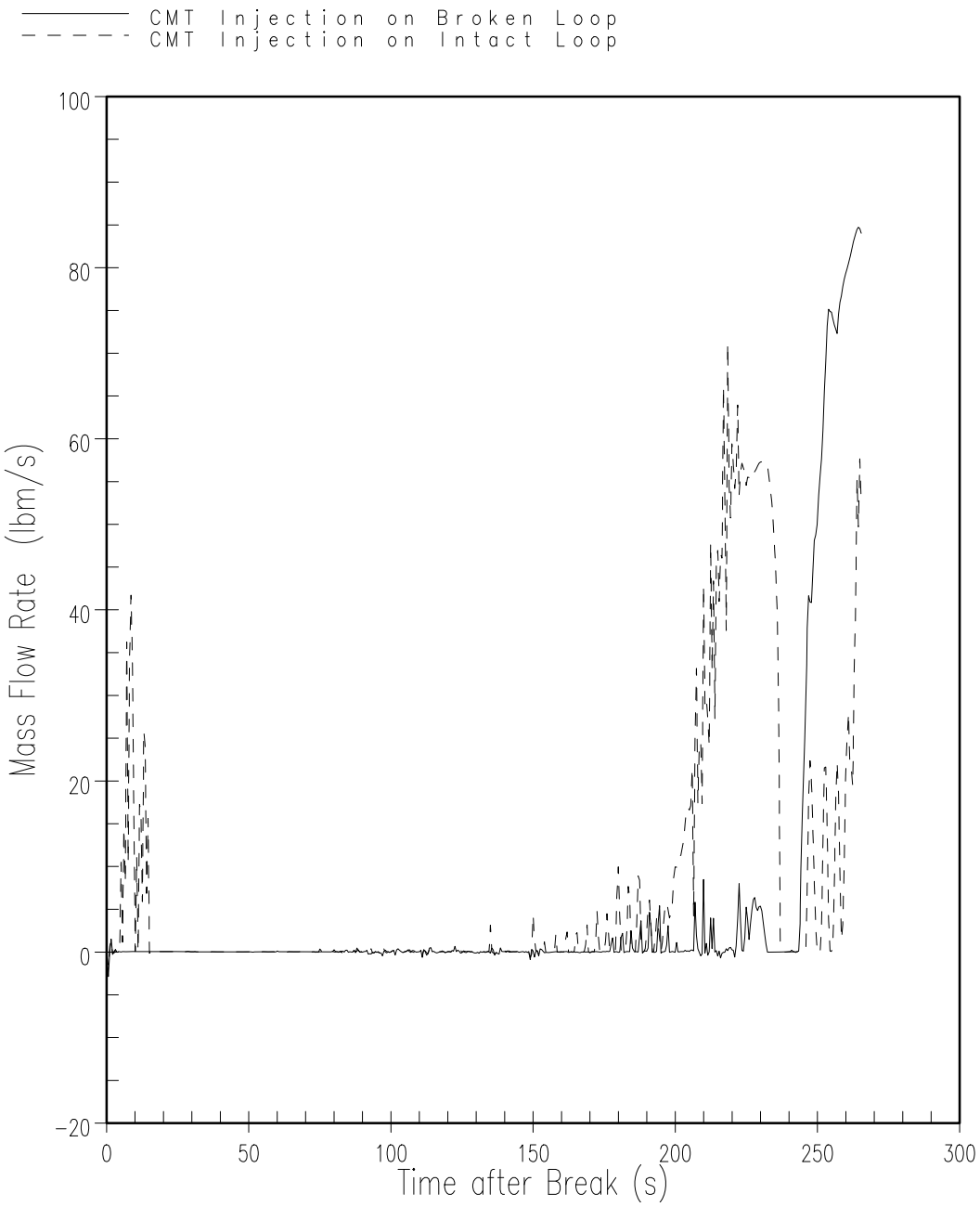

Figure 15.6.5.4A-6
Core Makeup Tank Injection Flow
for 95th Percentile Estimator PCT Case

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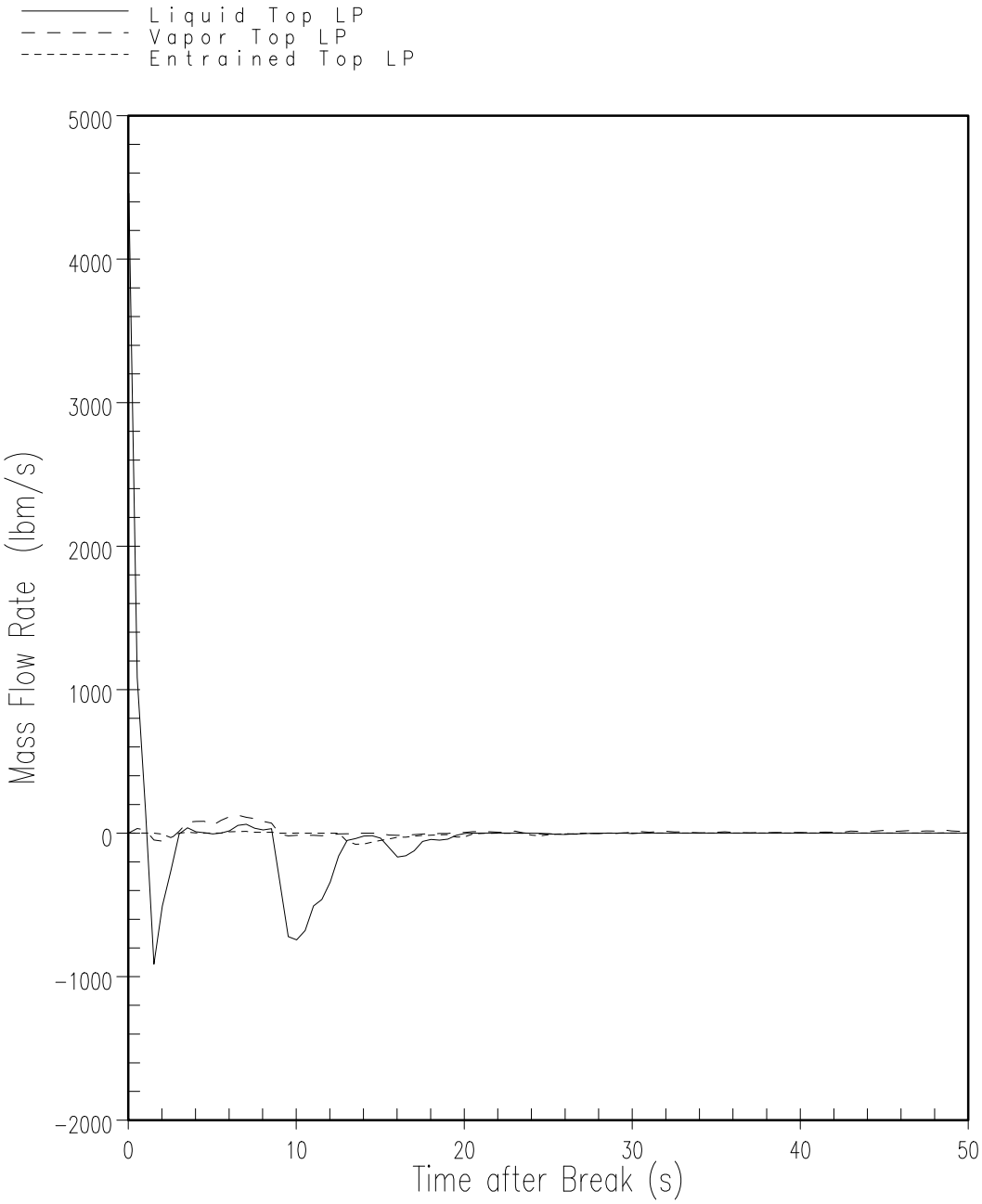

Figure 15.6.5.4A-7
Mass Flow at Top of Peripheral Assemblies Channel for 95th Percentile Estimator PCT Case

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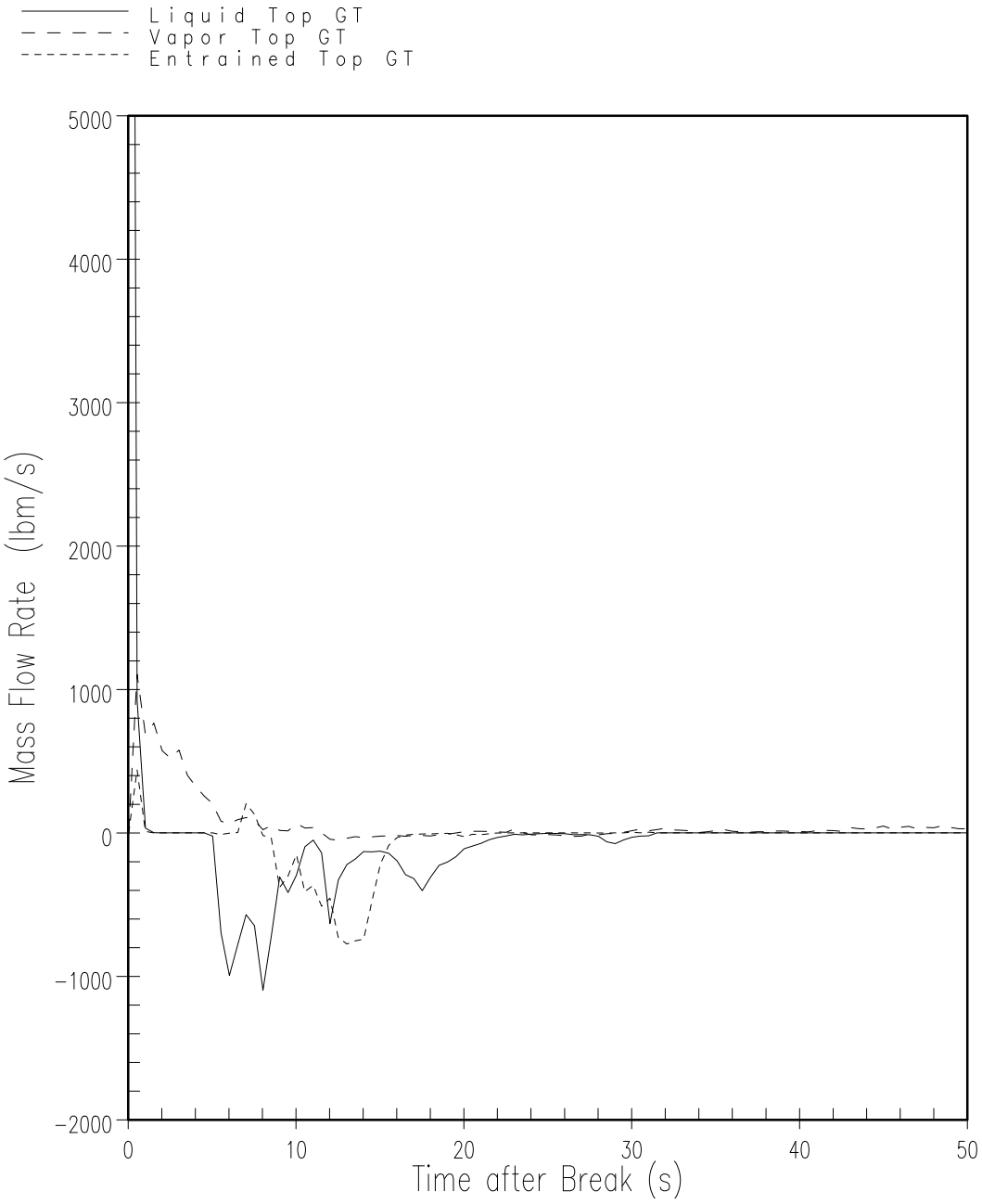

Figure 15.6.5.4A-8
Mass Flow at Top of Guide Tube Assemblies Channel for 95th Percentile Estimator PCT Case

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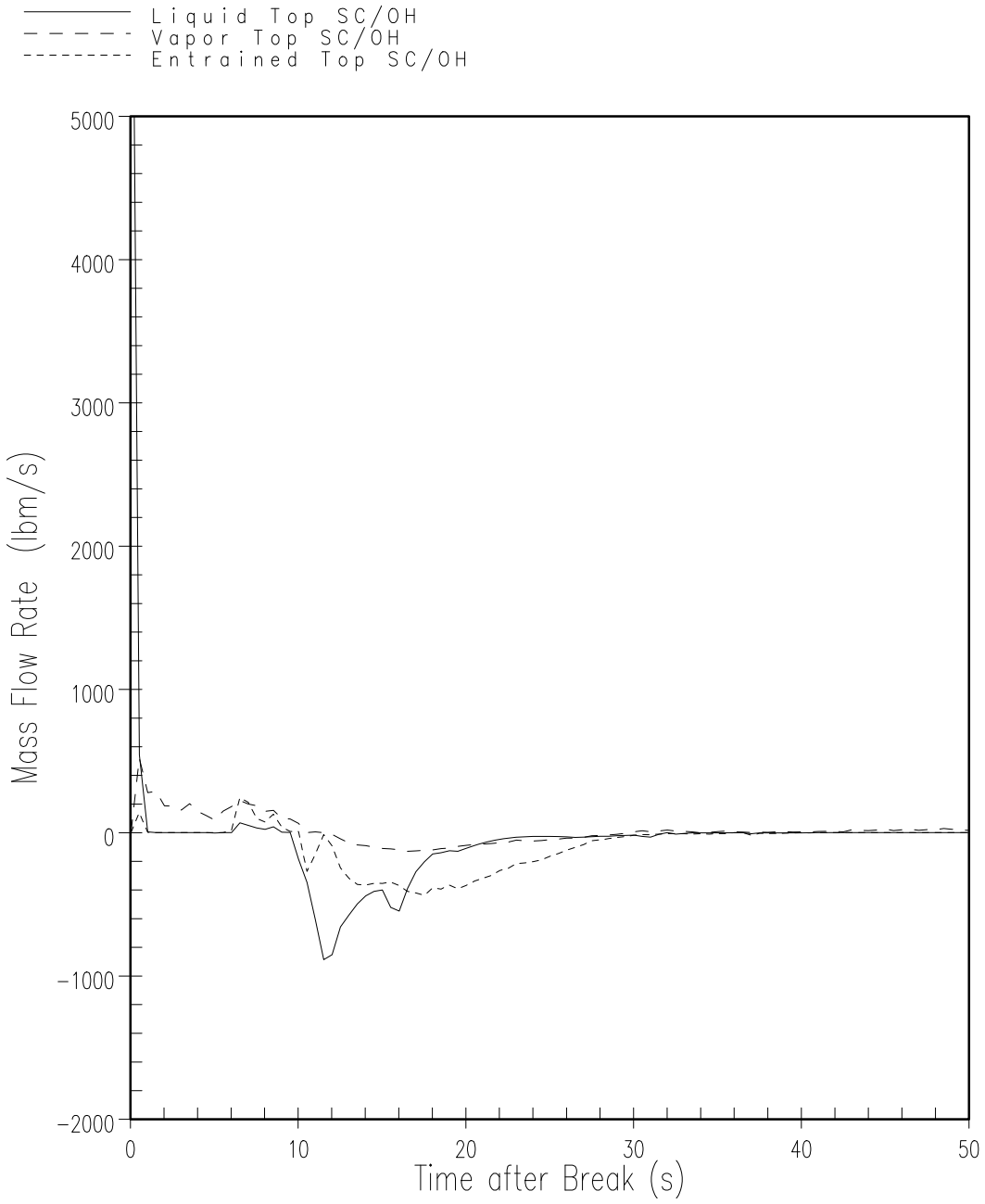

Figure 15.6.5.4A-9
Mass Flow at Top of Support Column/Open Hole Assemblies Channel for 95th Percentile Estimator PCT Case

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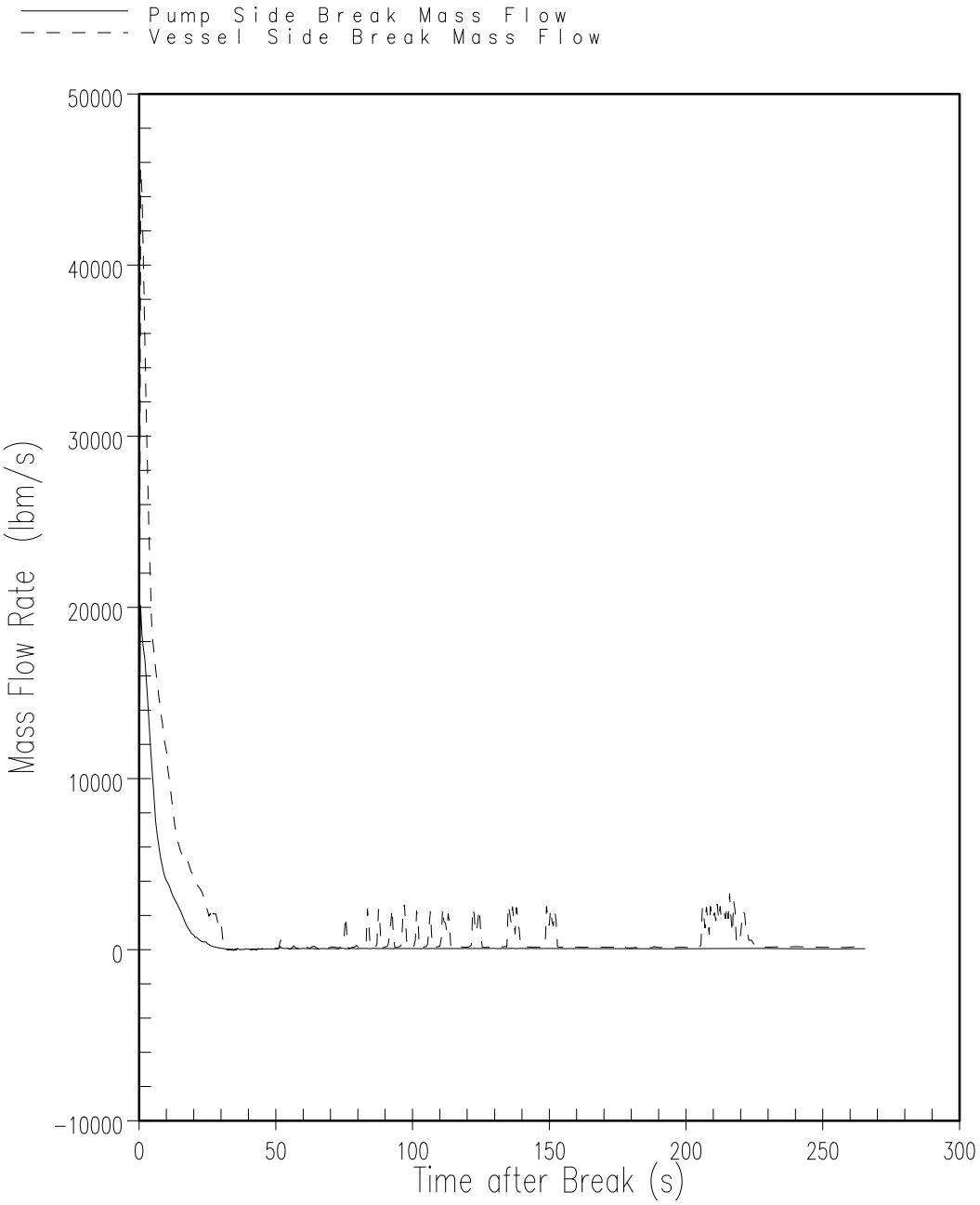

Figure 15.6.5.4A-10
Break Mass Flow for
95th Percentile Estimator PCT Case

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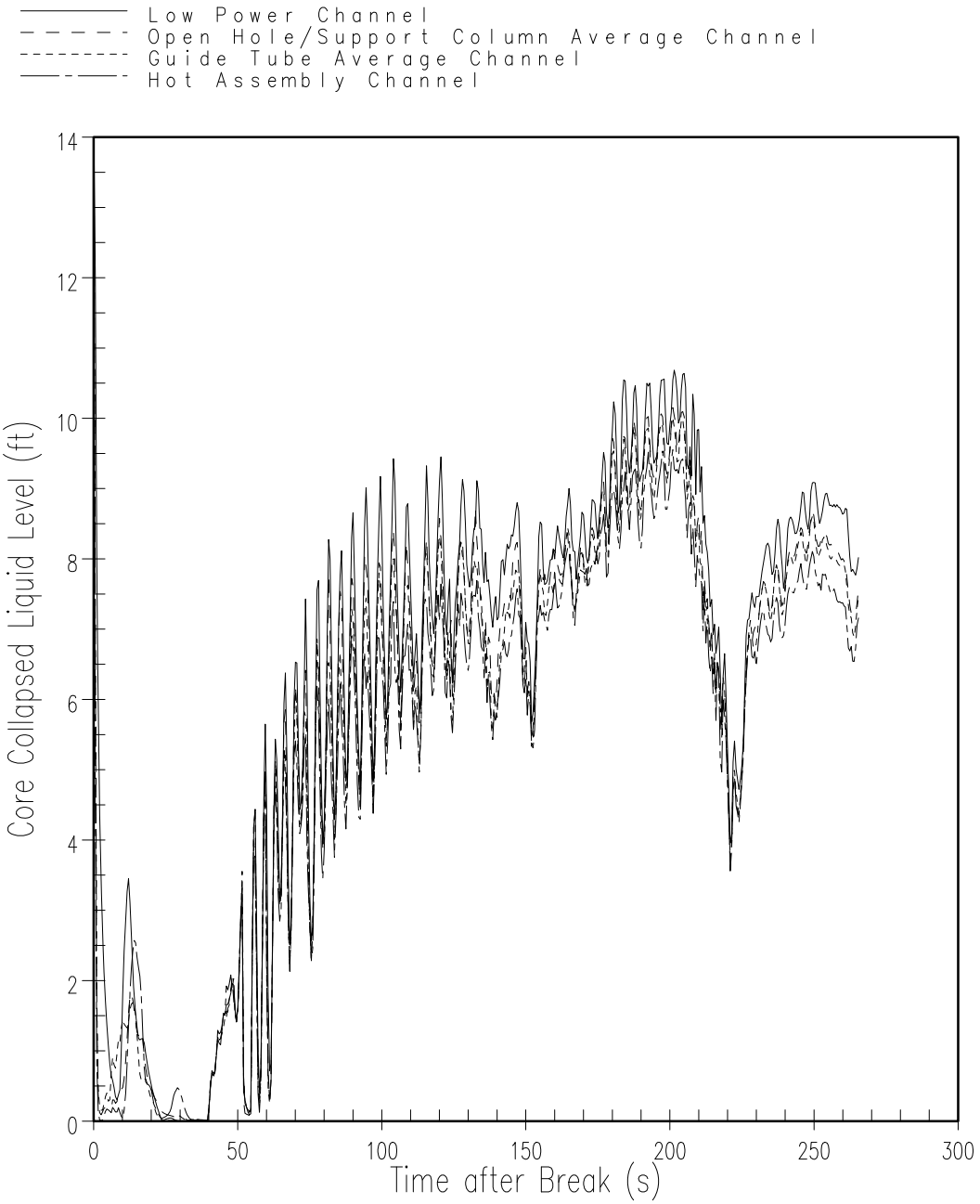

Figure 15.6.5.4A-11
Core Channel Collapsed Liquid Levels
for 95th Percentile Estimator PCT Case
(Reference Point: Bottom of Active Fuel)

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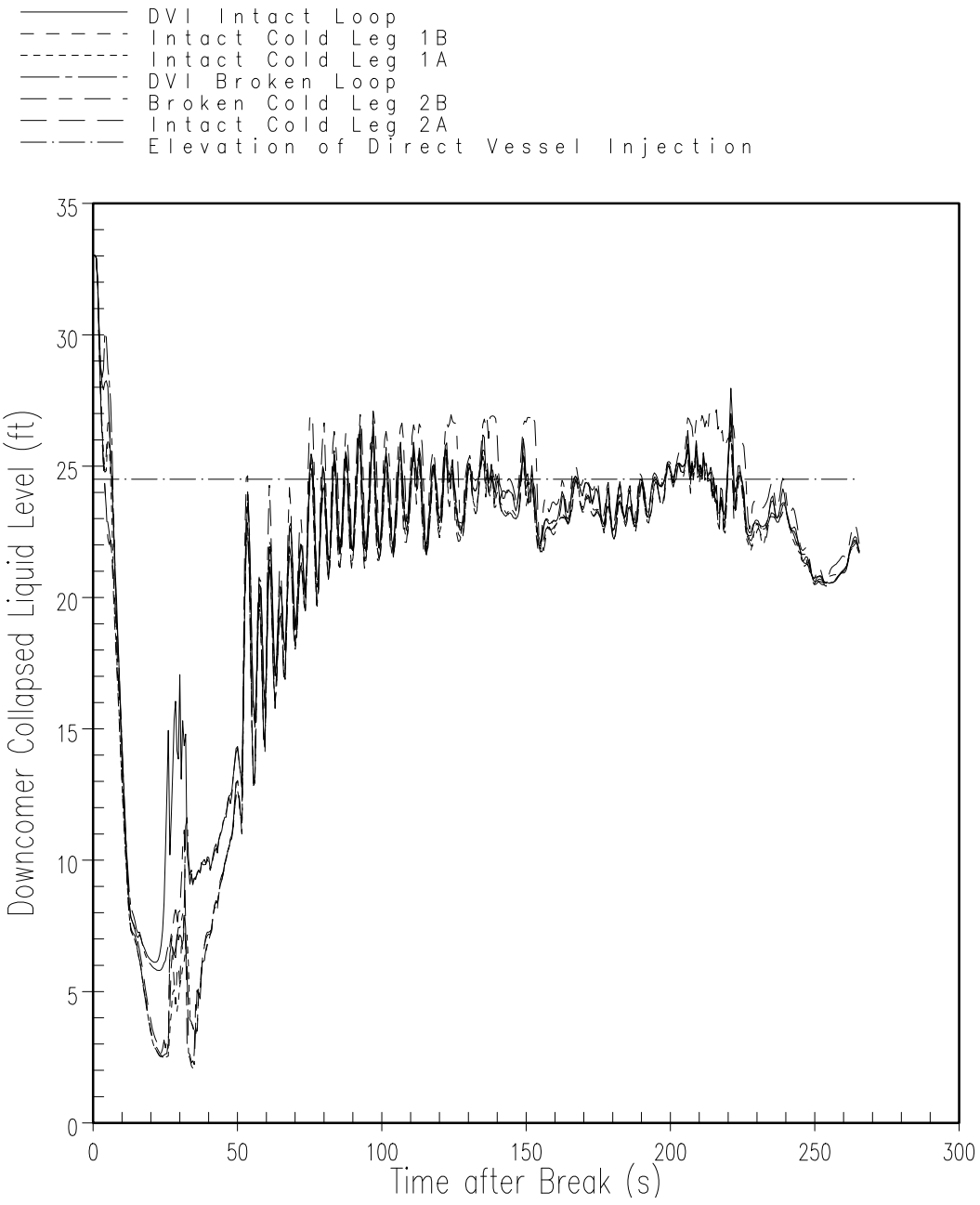

Figure 15.6.5.4A-12
Downcomer Channel Collapsed Liquid Levels
for 95th Percentile Estimator PCT Case
(Reference Point: Inside Bottom of Vessel)

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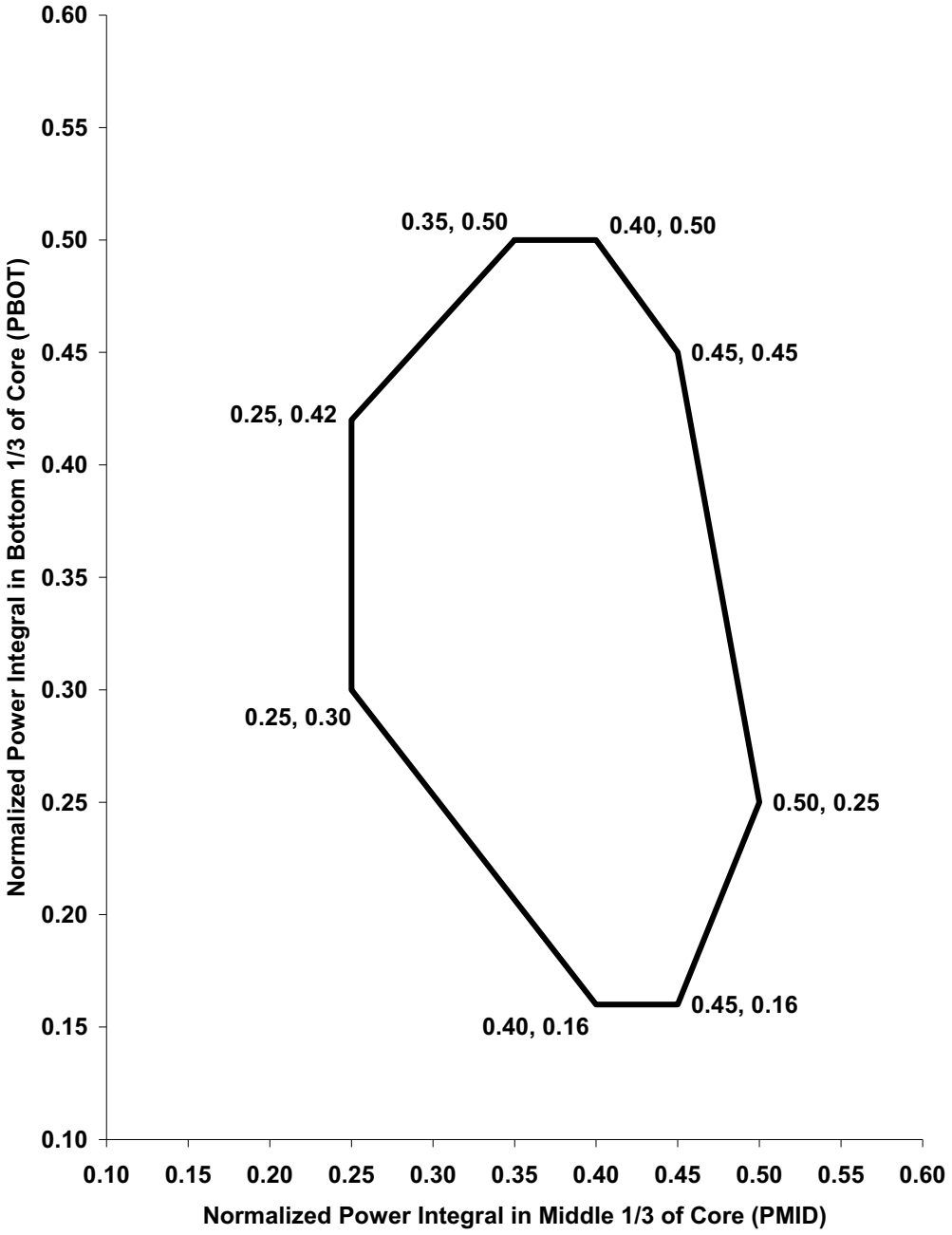

Figure 15.6.5.4A-13
PBOT/PMID Box Supported by
AP1000 ASTRUM Analysis

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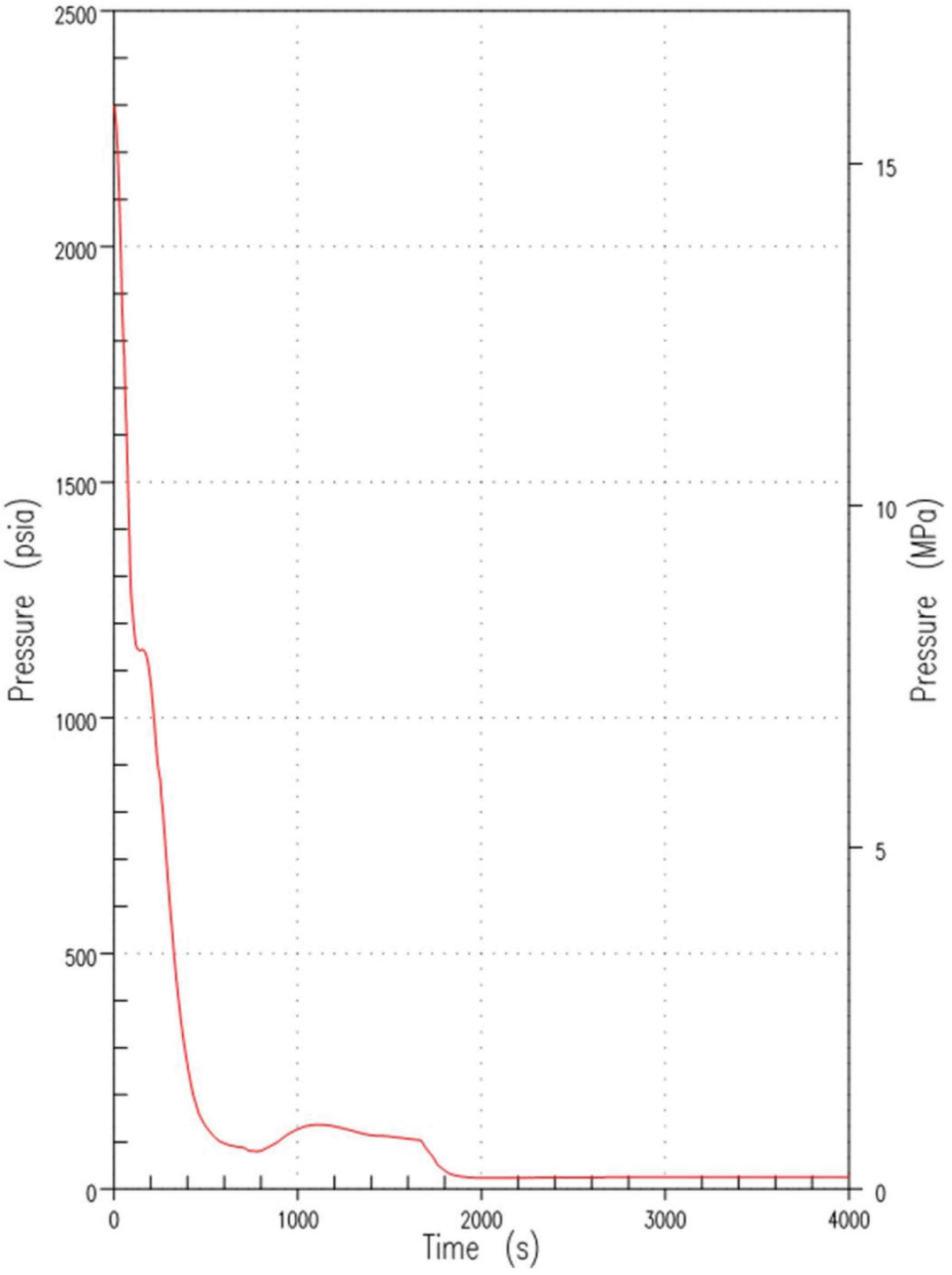

Figure 15.6.5.4B-1(a)
Inadvertent ADS – RCS Pressure

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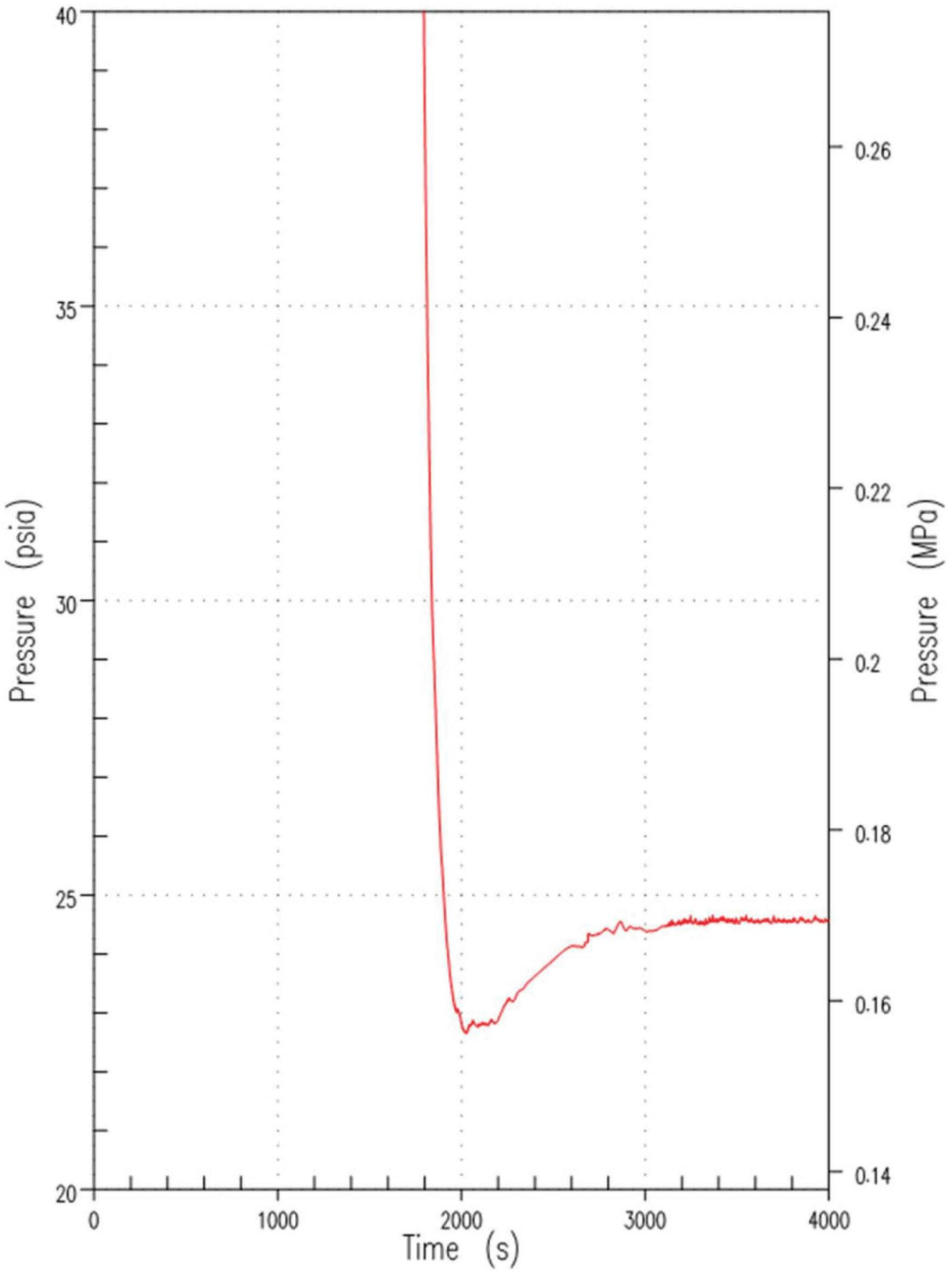

Figure 15.6.5.4B-1(b)
Inadvertent ADS – RCS Pressure (Zoomed)

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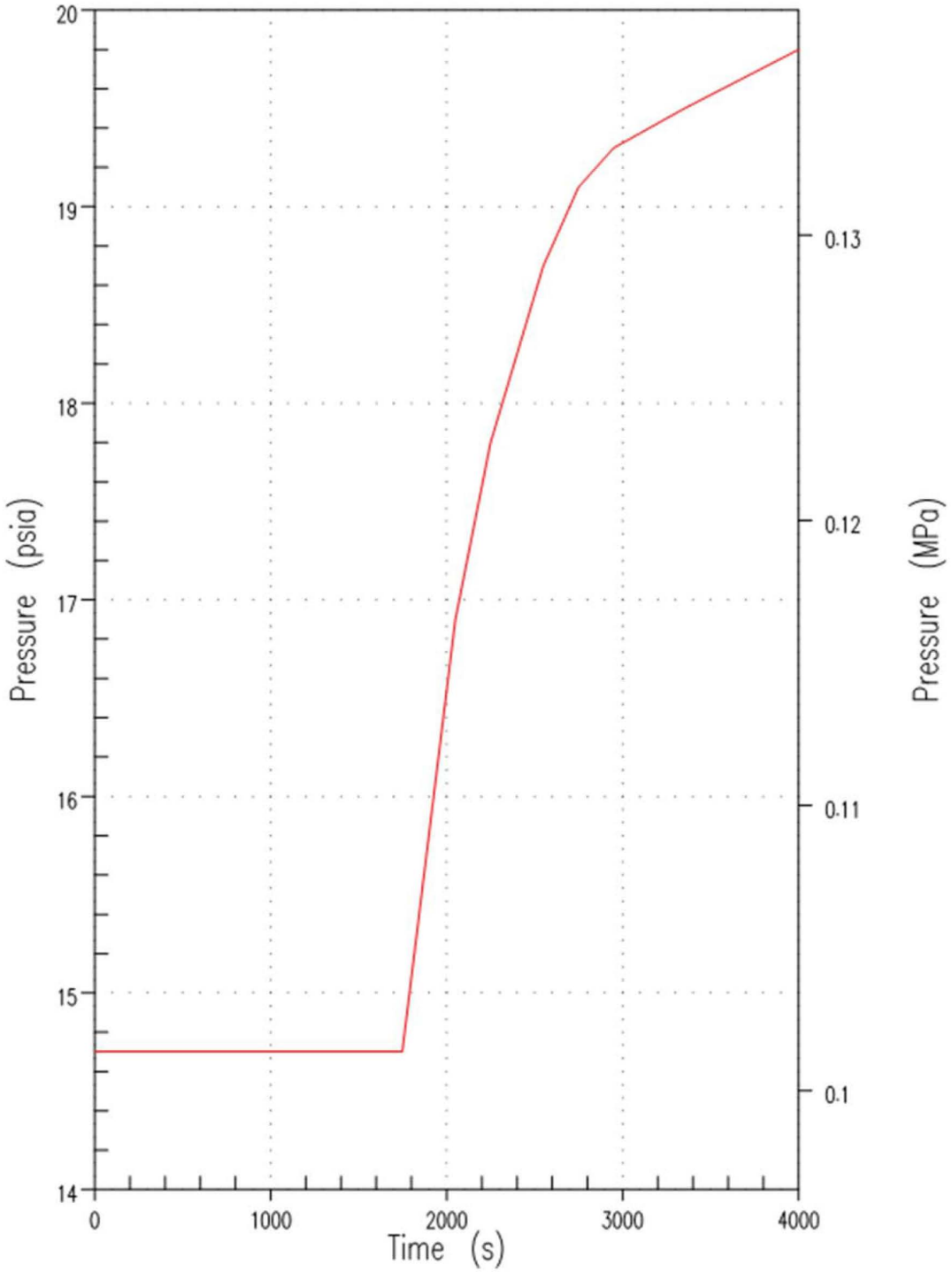

Figure 15.6.5.4B-1(c)
Inadvertent ADS – Containment Pressure

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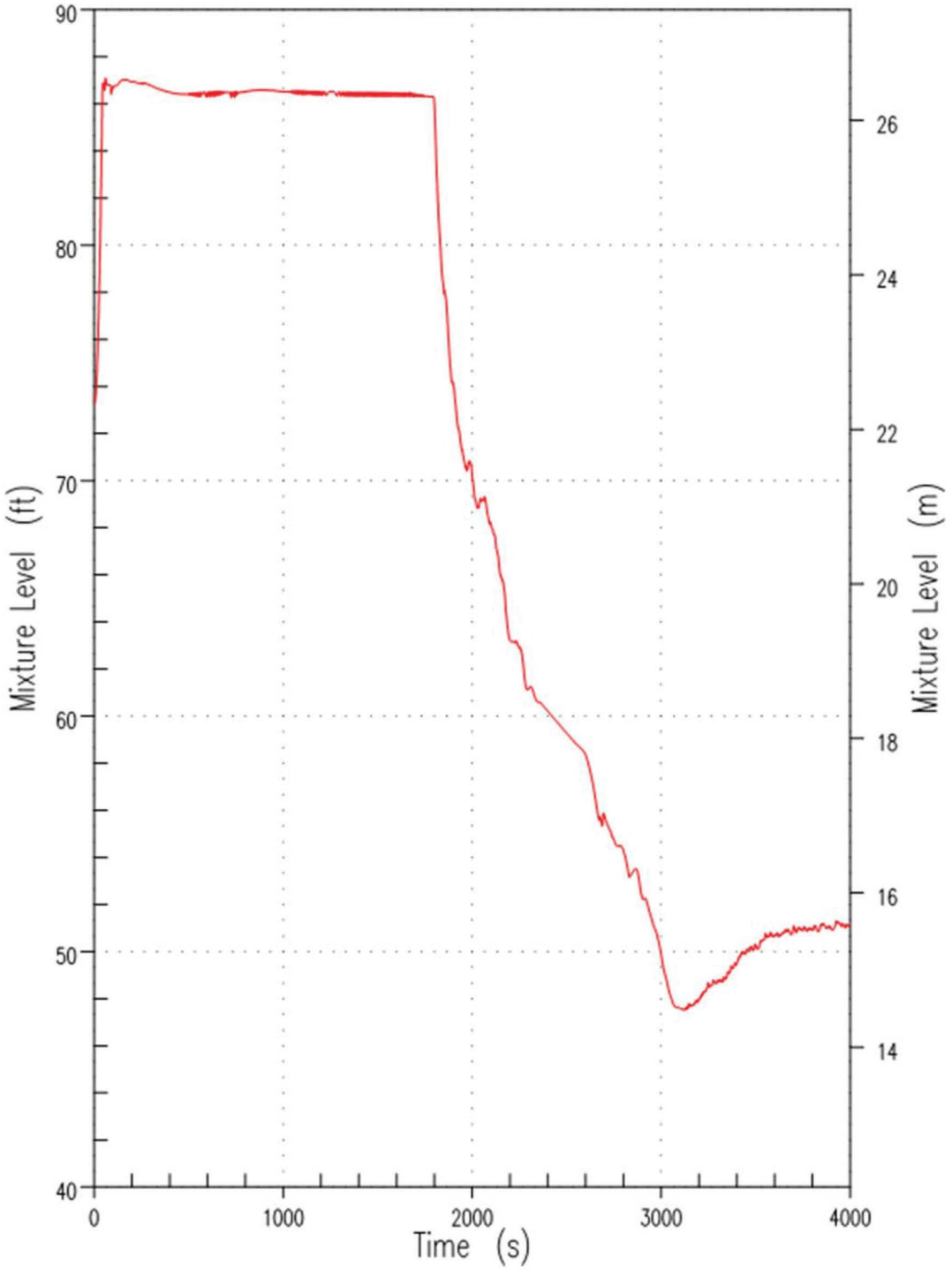

Figure 15.6.5.4B-2
Inadvertent ADS – Pressurizer Mixture Level

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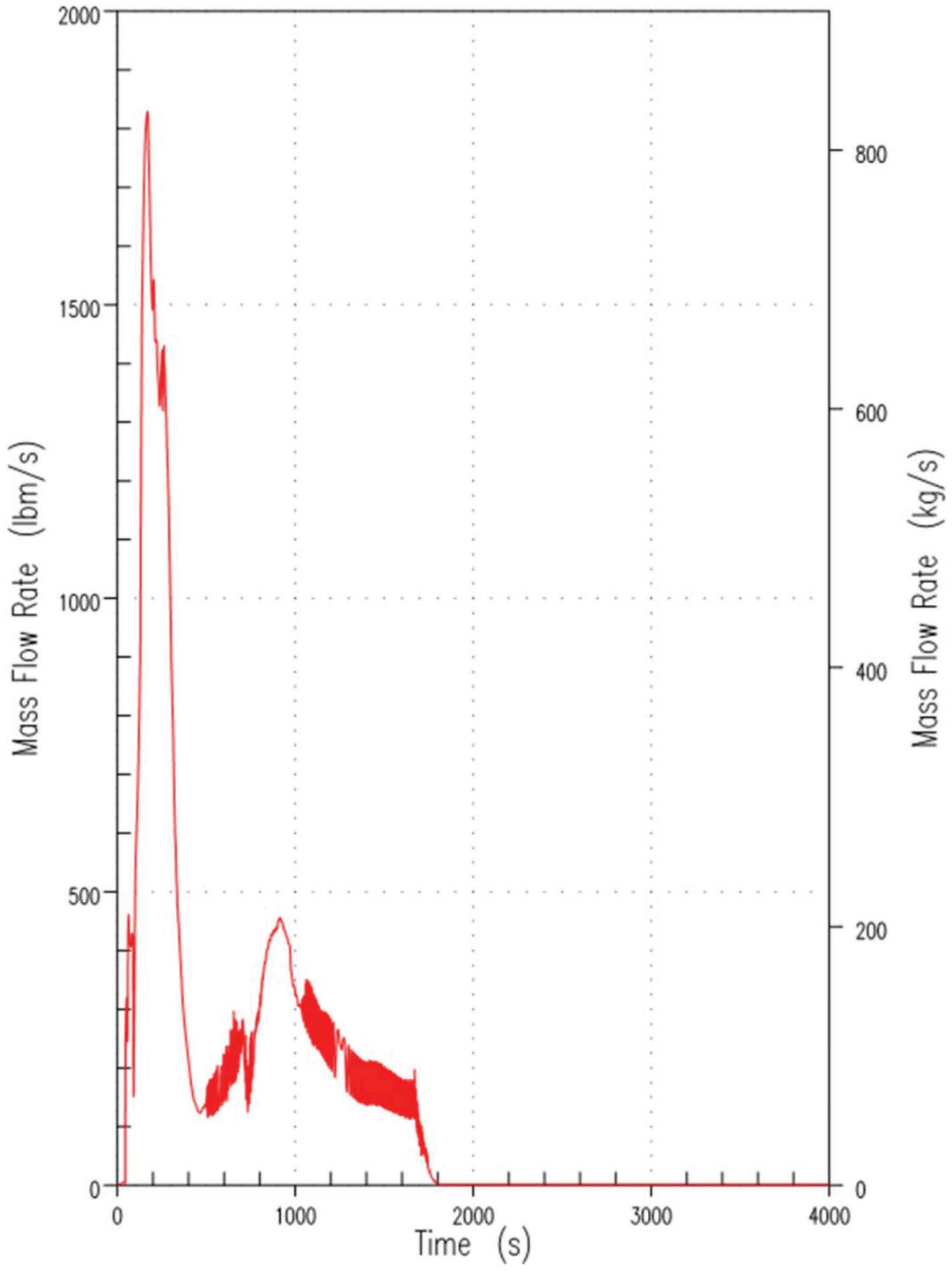

Figure 15.6.5.4B-3
Inadvertent ADS – ADS 1-3 Liquid Discharge

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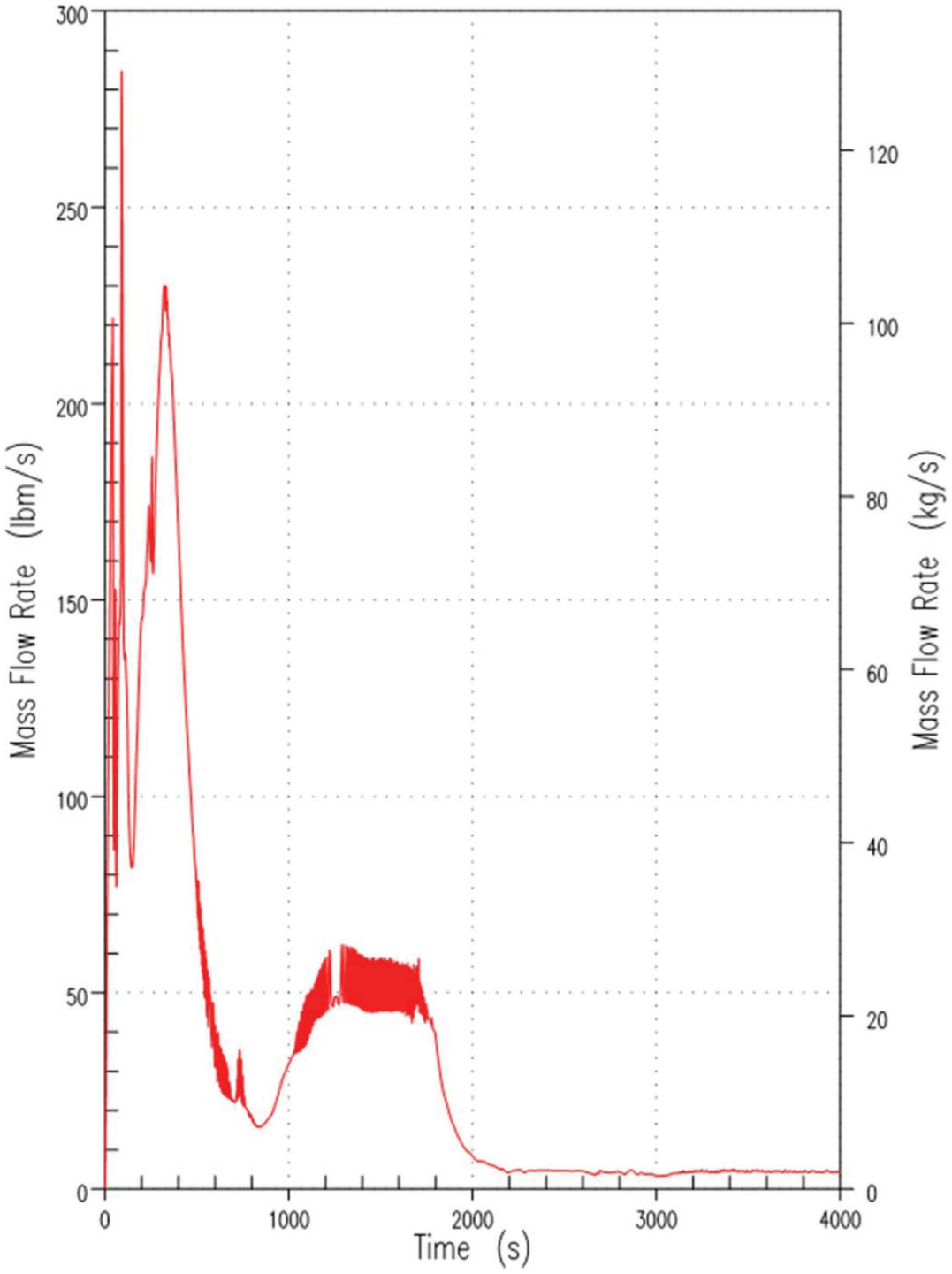

Figure 15.6.5.4B-4(a)
Inadvertent ADS – ADS 1-3 Vapor Discharge

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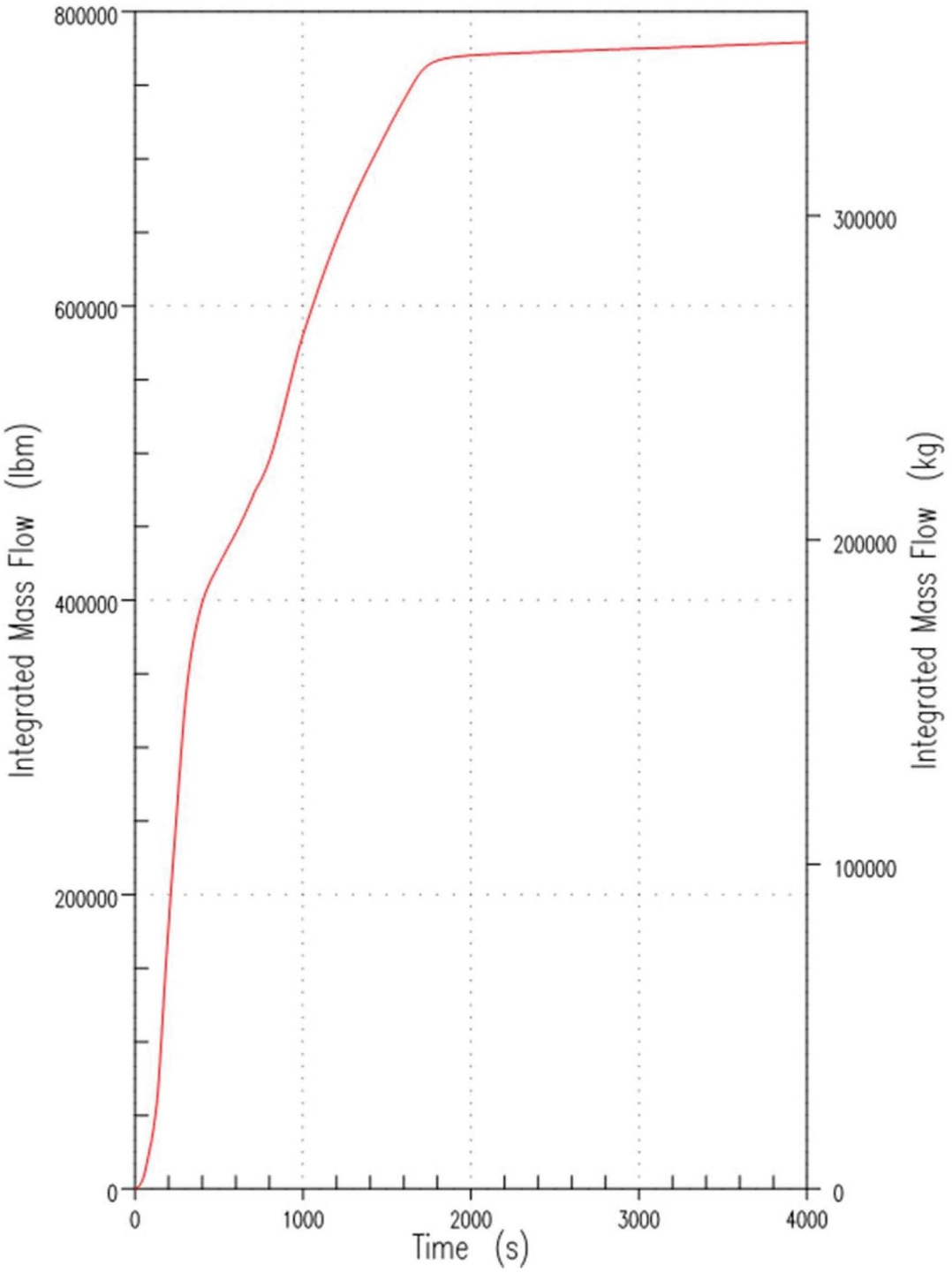

Figure 15.6.5.4B-4(b)
Inadvertent ADS – ADS 1-3 Integrated Discharge

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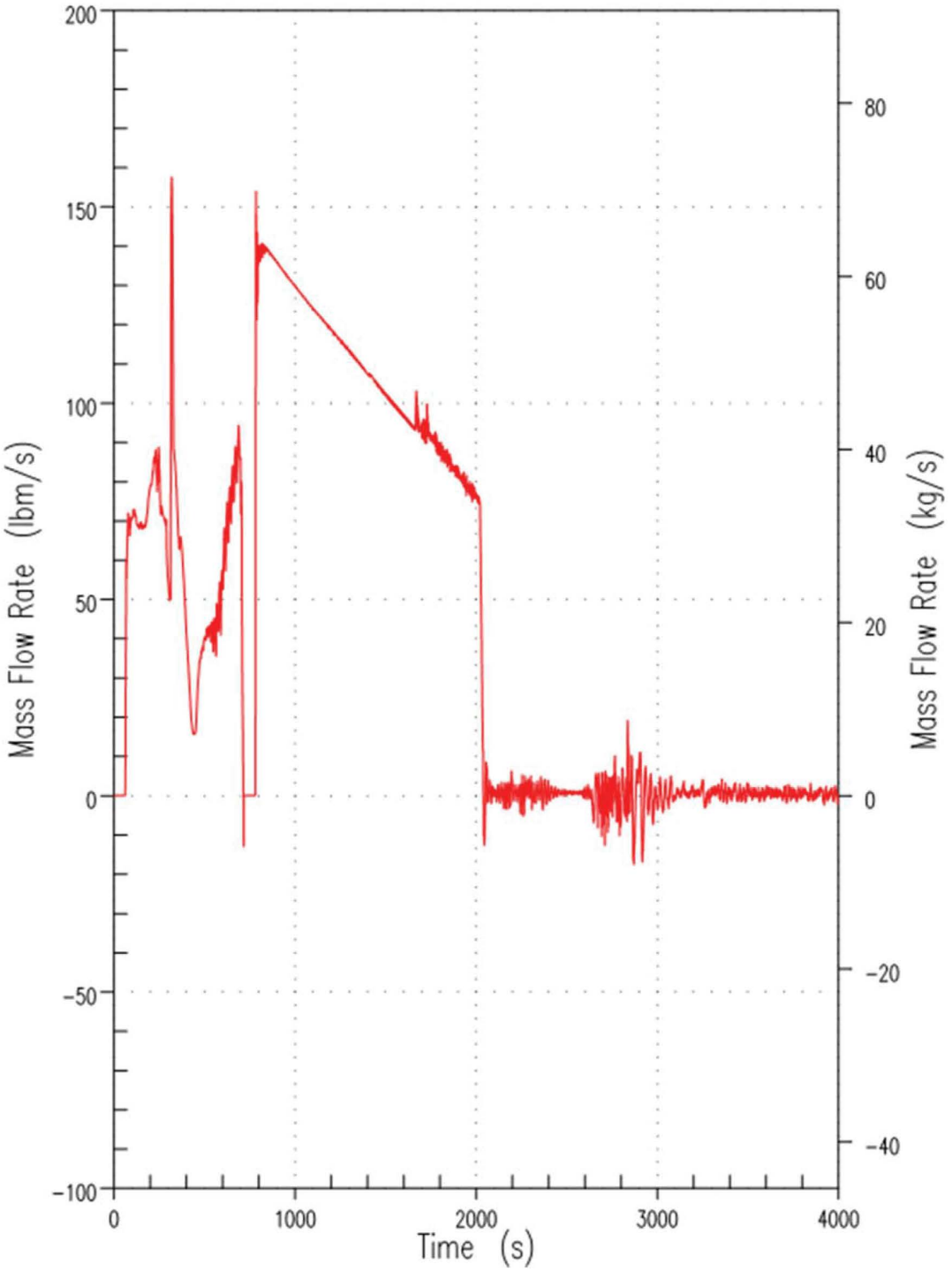

Figure 15.6.5.4B-5
Inadvertent ADS – CMT-1 Injection Rate

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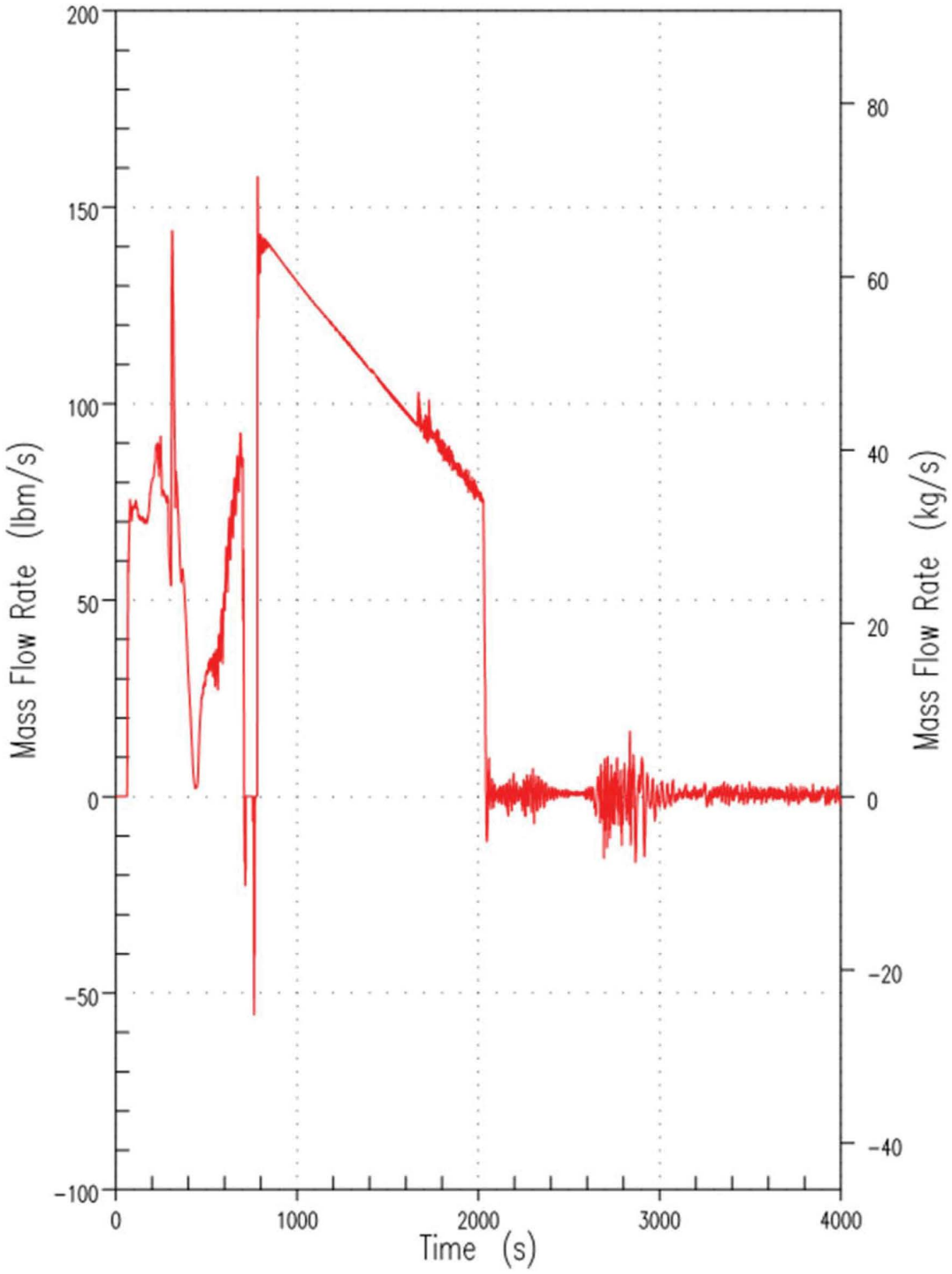

Figure 15.6.5.4B-6
Inadvertent ADS – CMT-2 Injection Rate

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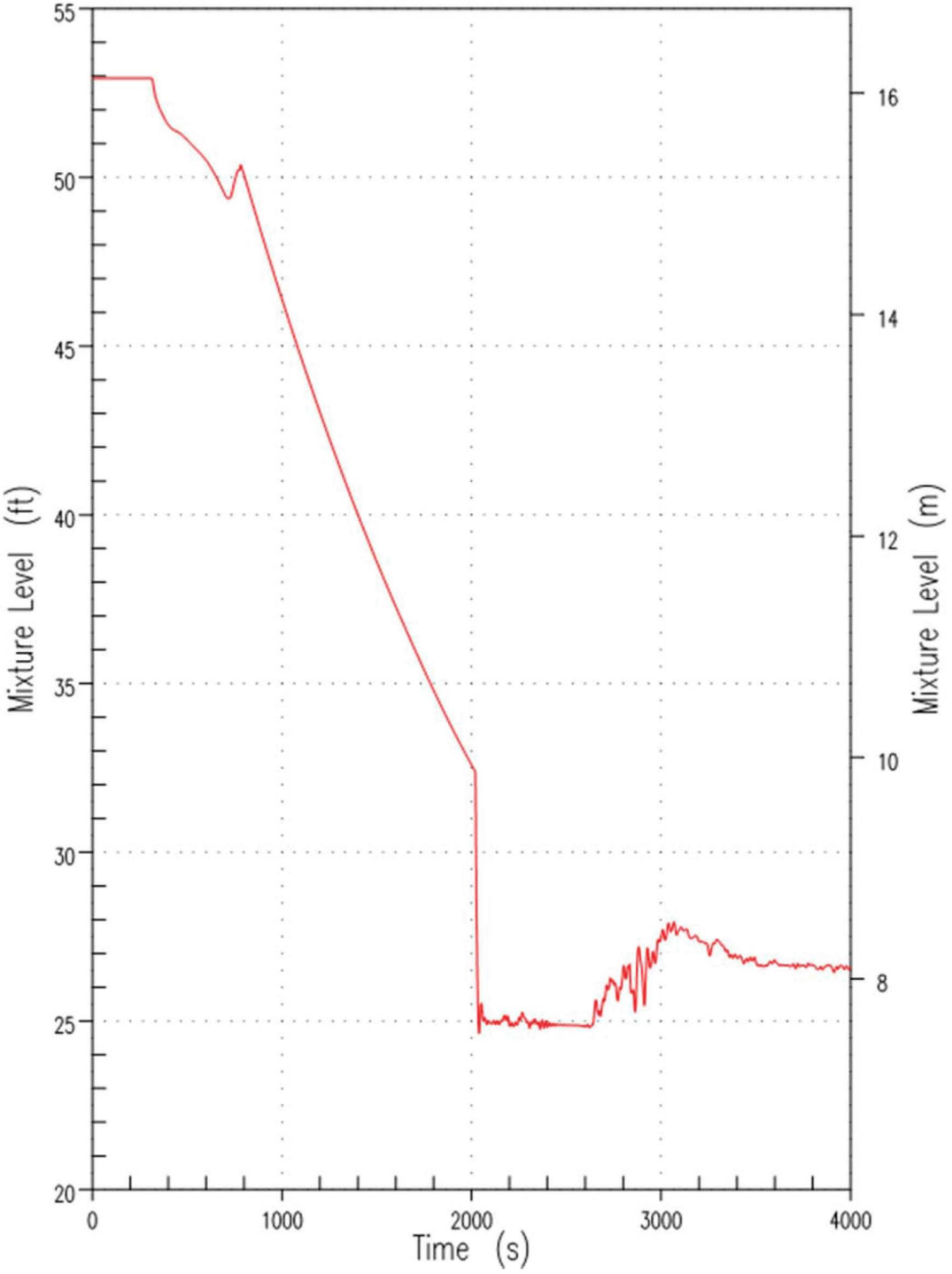

Figure 15.6.5.4B-7
Inadvertent ADS – CMT-1 Mixture Level

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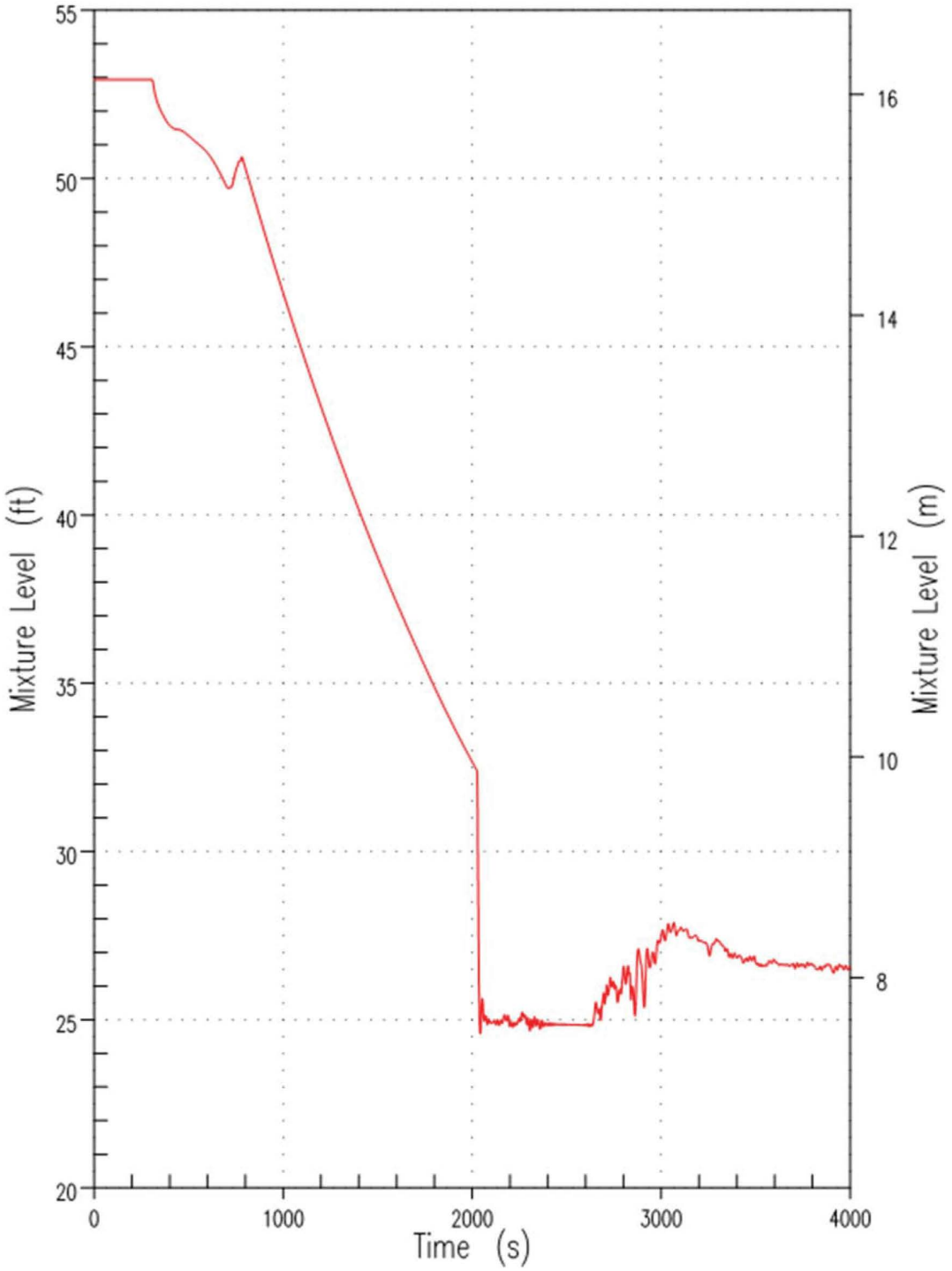

Figure 15.6.5.4B-8
Inadvertent ADS – CMT-2 Mixture Level

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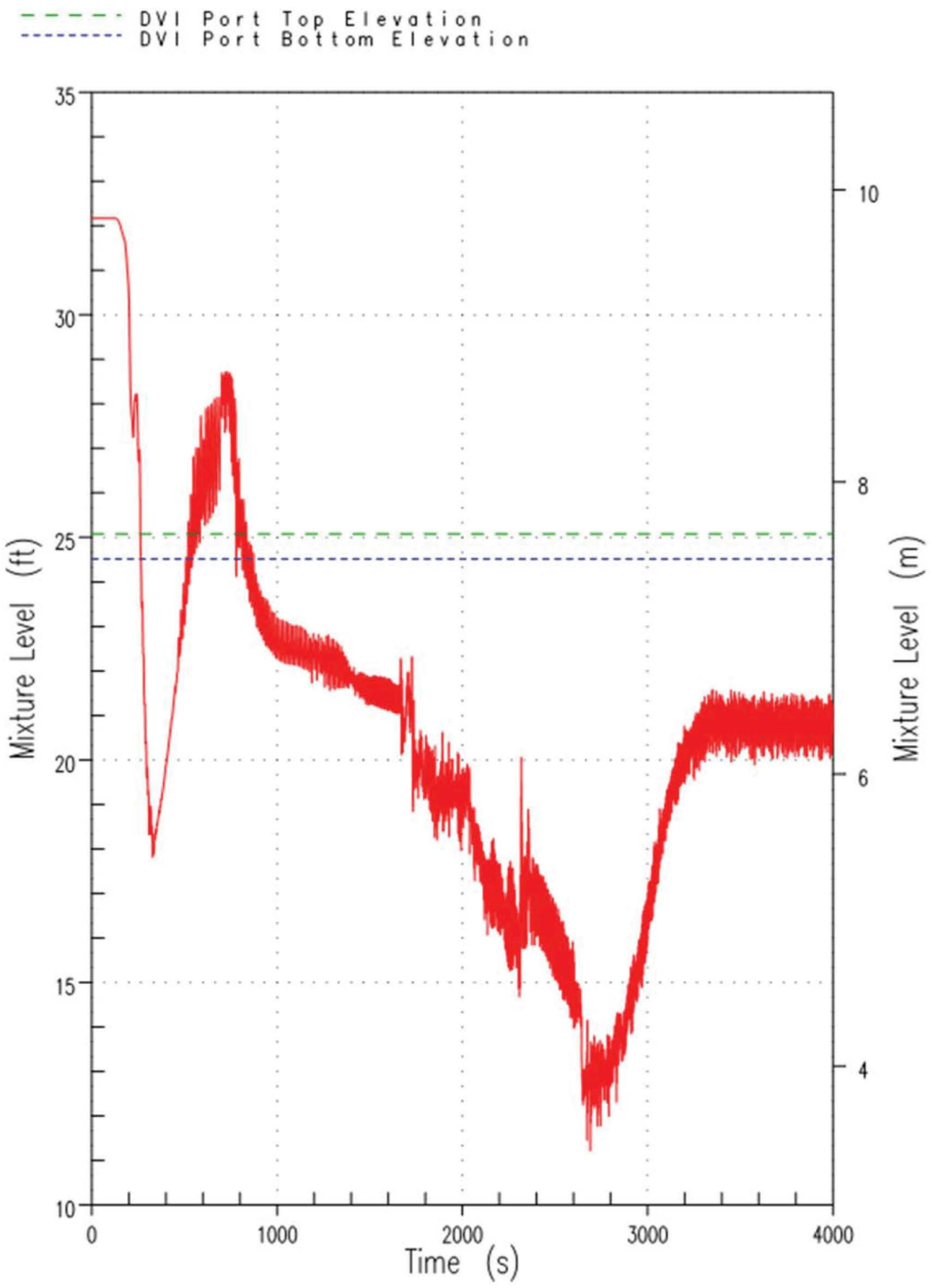

Figure 15.6.5.4B-9
Inadvertent ADS – Downcomer Mixture Level

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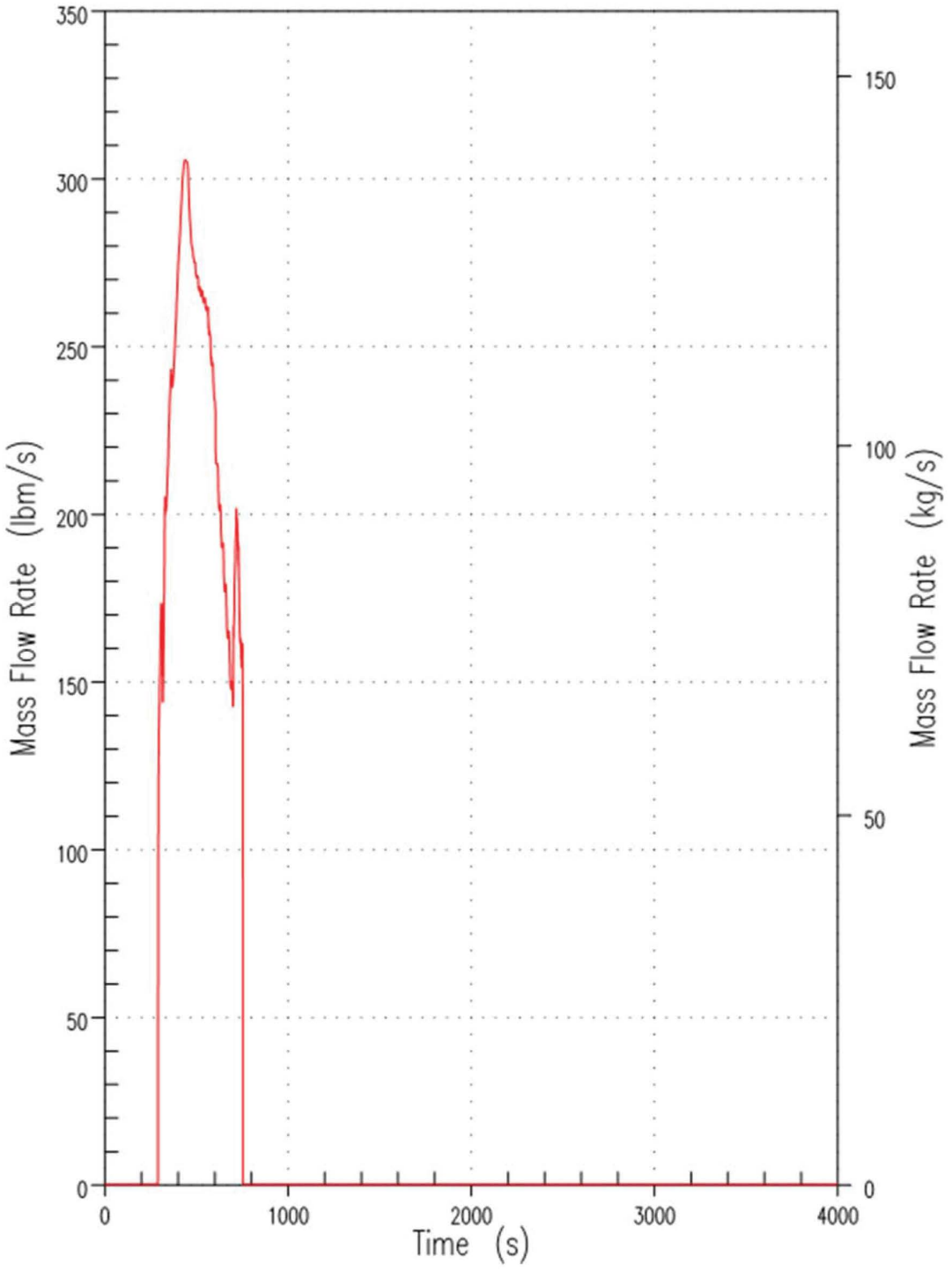

Figure 15.6.5.4B-10
Inadvertent ADS – Accumulator-1 Injection Rate

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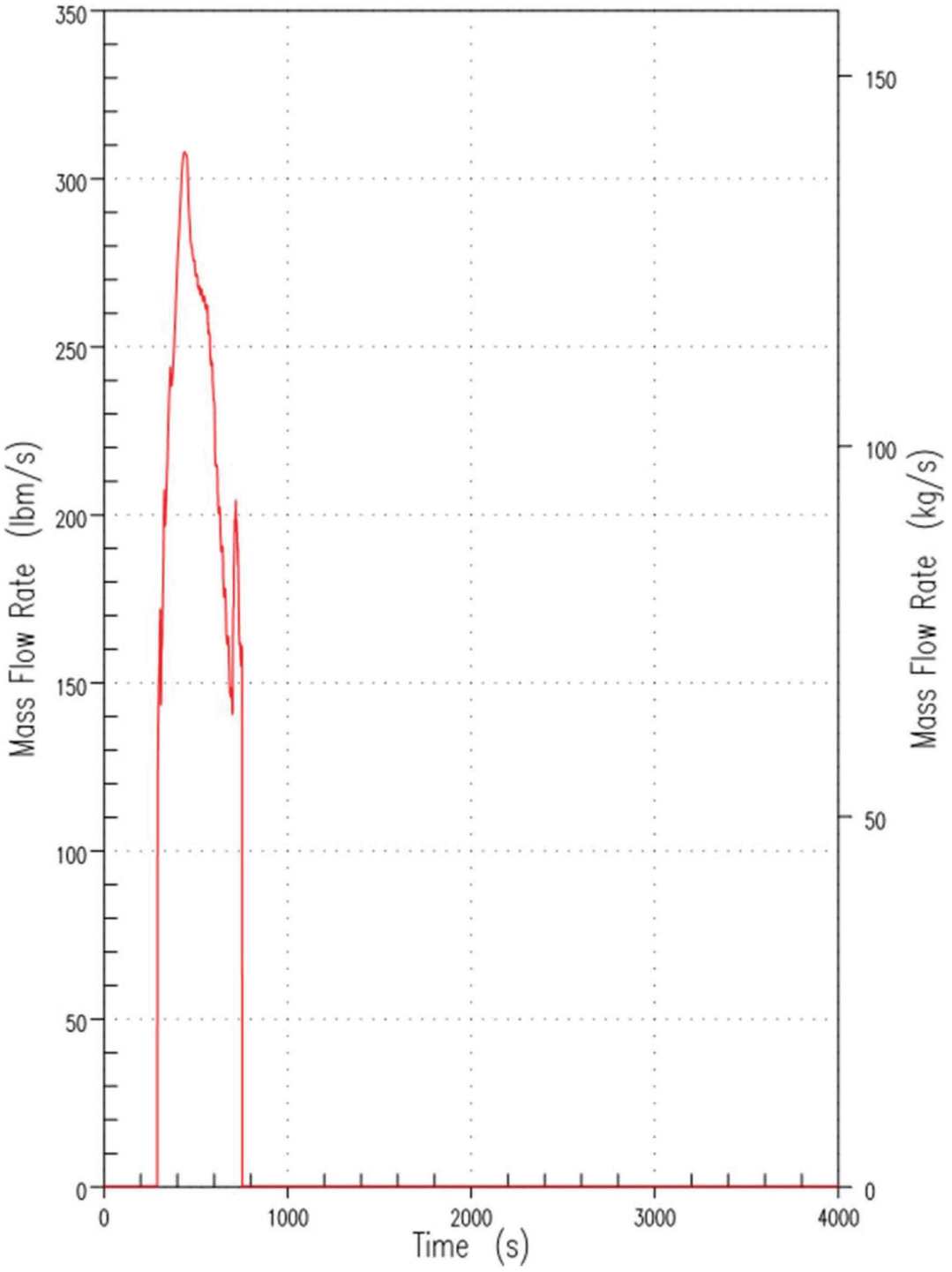

Figure 15.6.5.4B-11
Inadvertent ADS – Accumulator-2 Injection Rate

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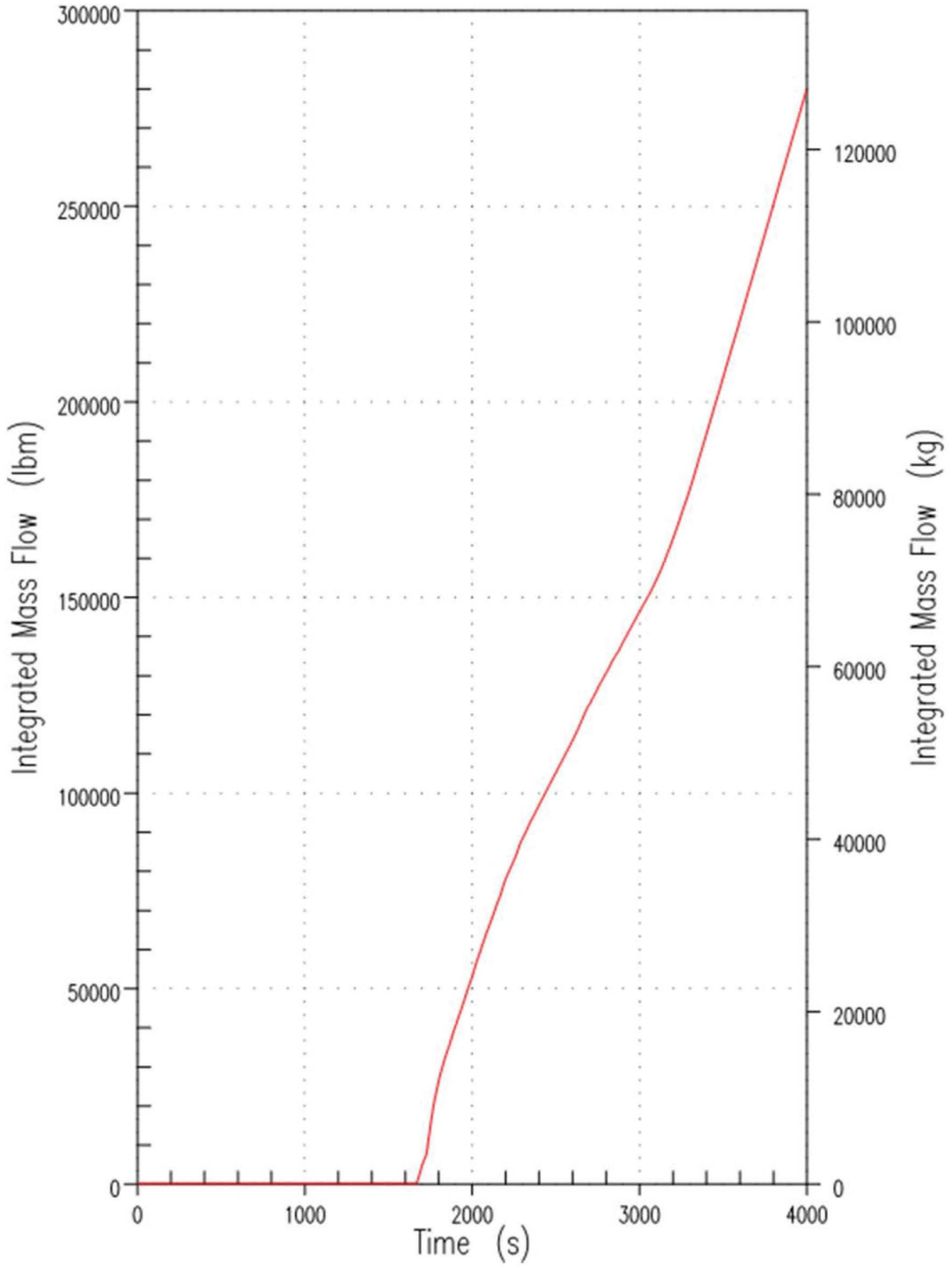

Figure 15.6.5.4B-12(a)
Inadvertent ADS – ADS-4 Integrated Discharge

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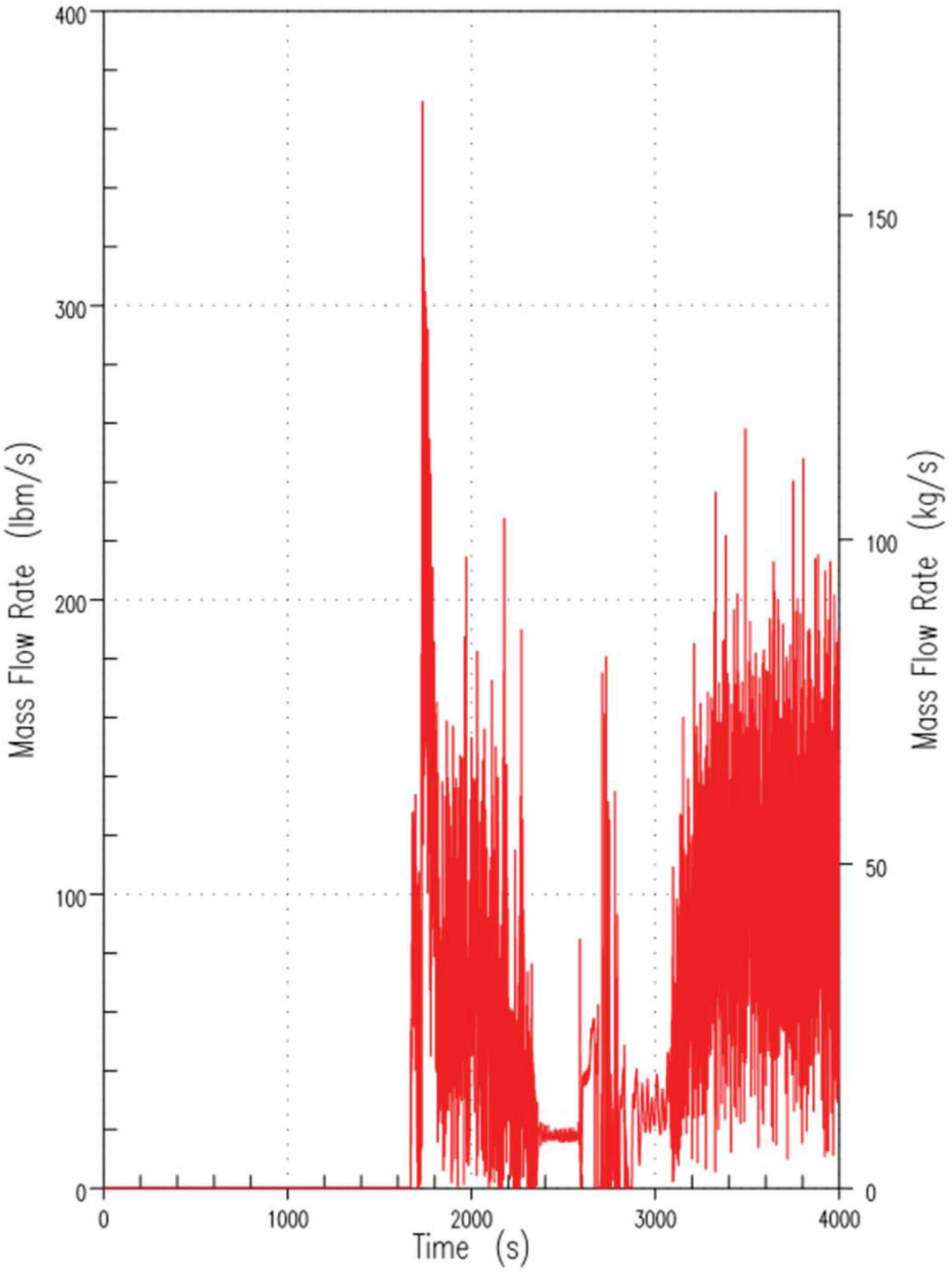

Figure 15.6.5.4B-12(b)
Inadvertent ADS – ADS-4 Liquid Discharge

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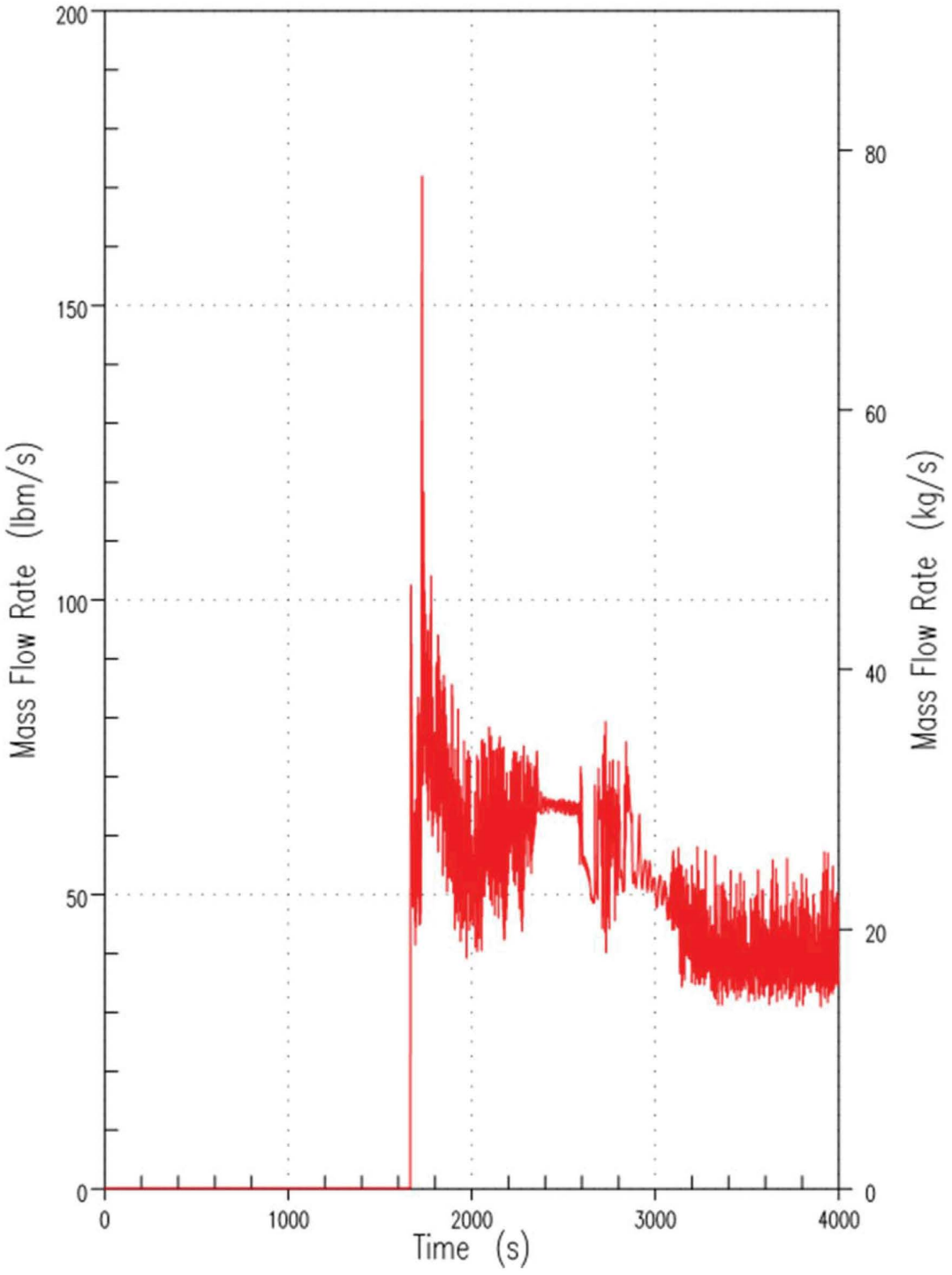

Figure 15.6.5.4B-12(c)
Inadvertent ADS – ADS-4 Vapor Discharge

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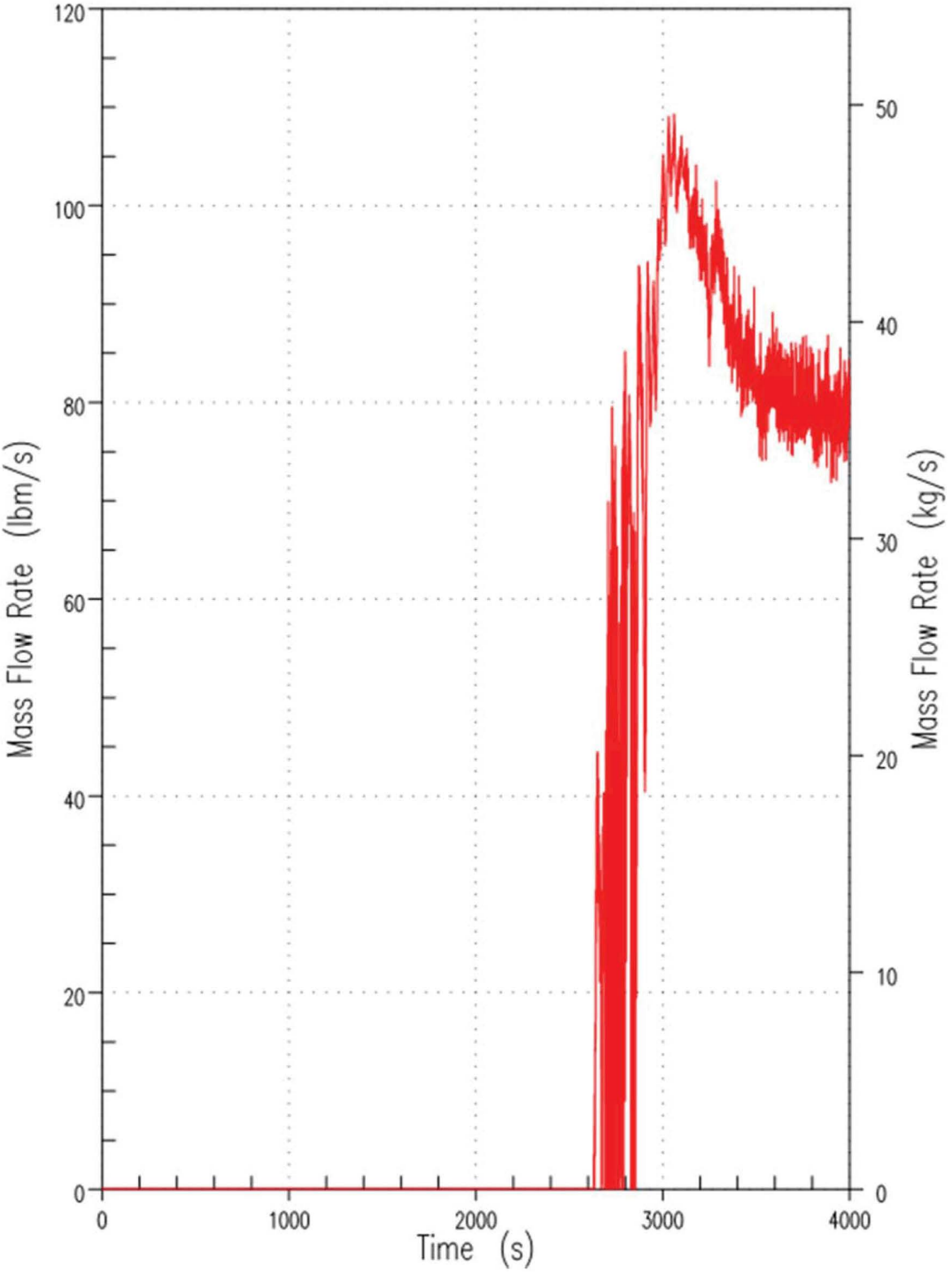

Figure 15.6.5.4B-13
Inadvertent ADS – IRWST-1 Injection Rate

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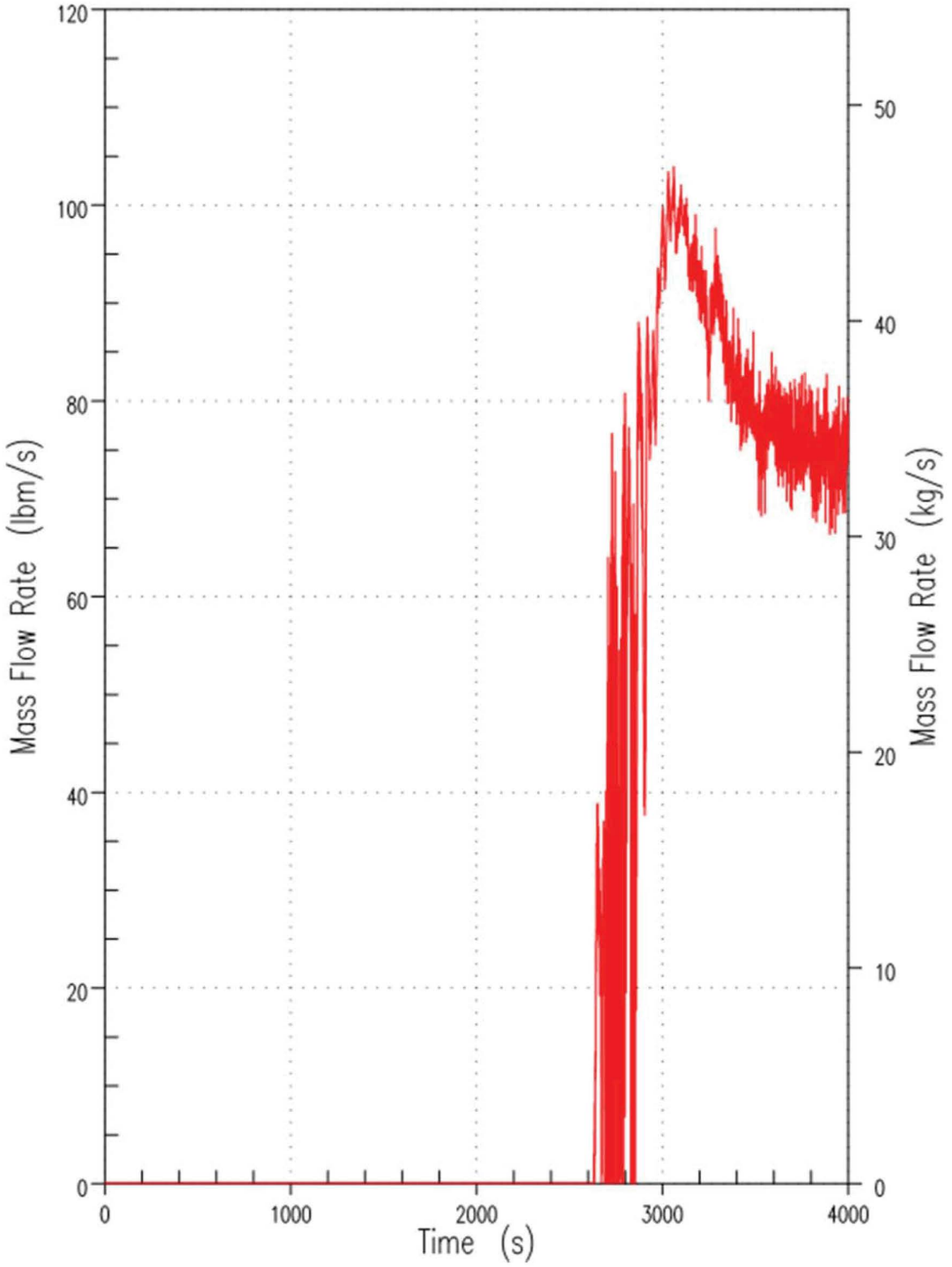

Figure 15.6.5.4B-14
Inadvertent ADS – IRWST-2 Injection Rate

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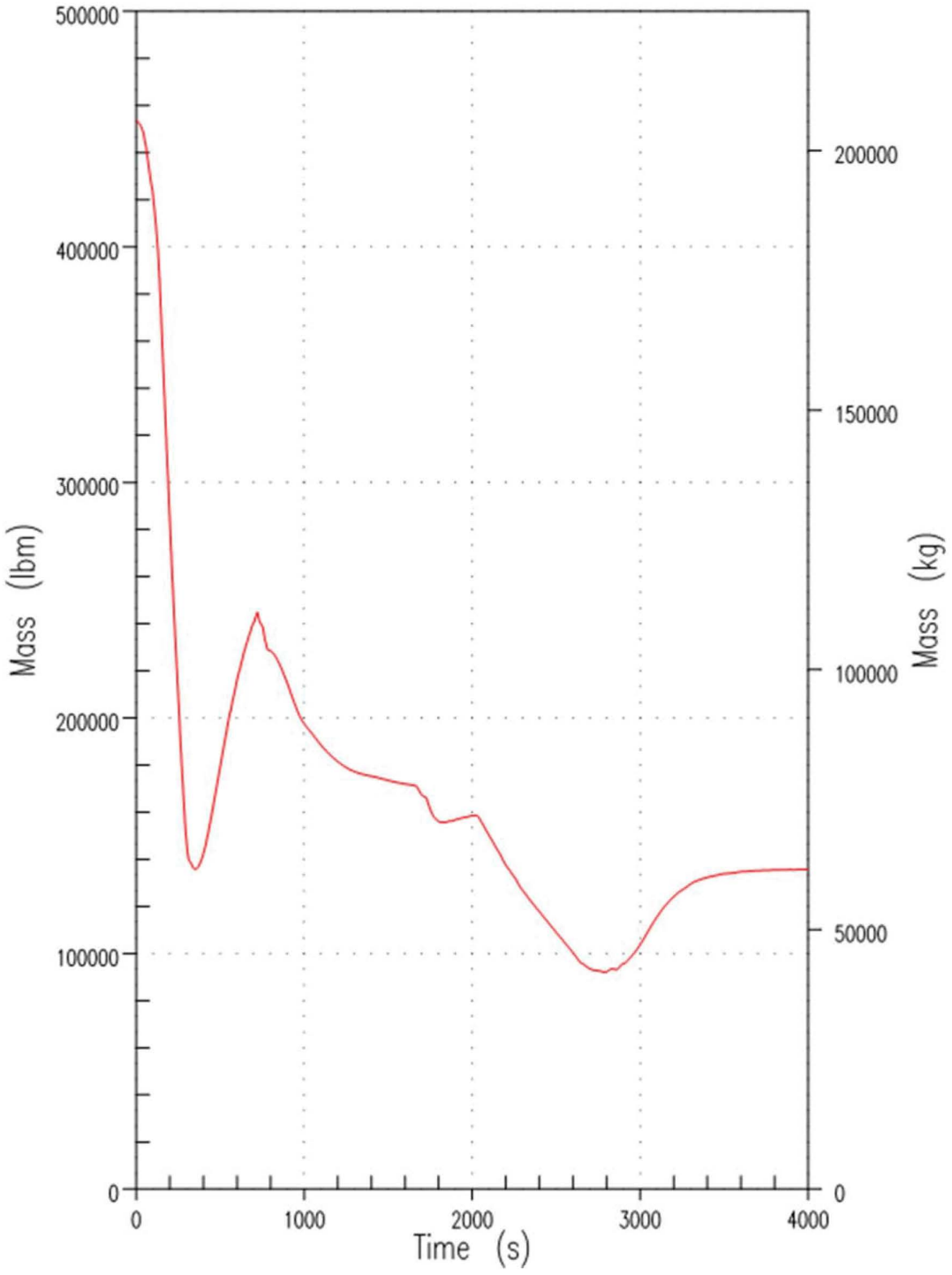

Figure 15.6.5.4B-15(a)
Inadvertent ADS – RCS System Inventory

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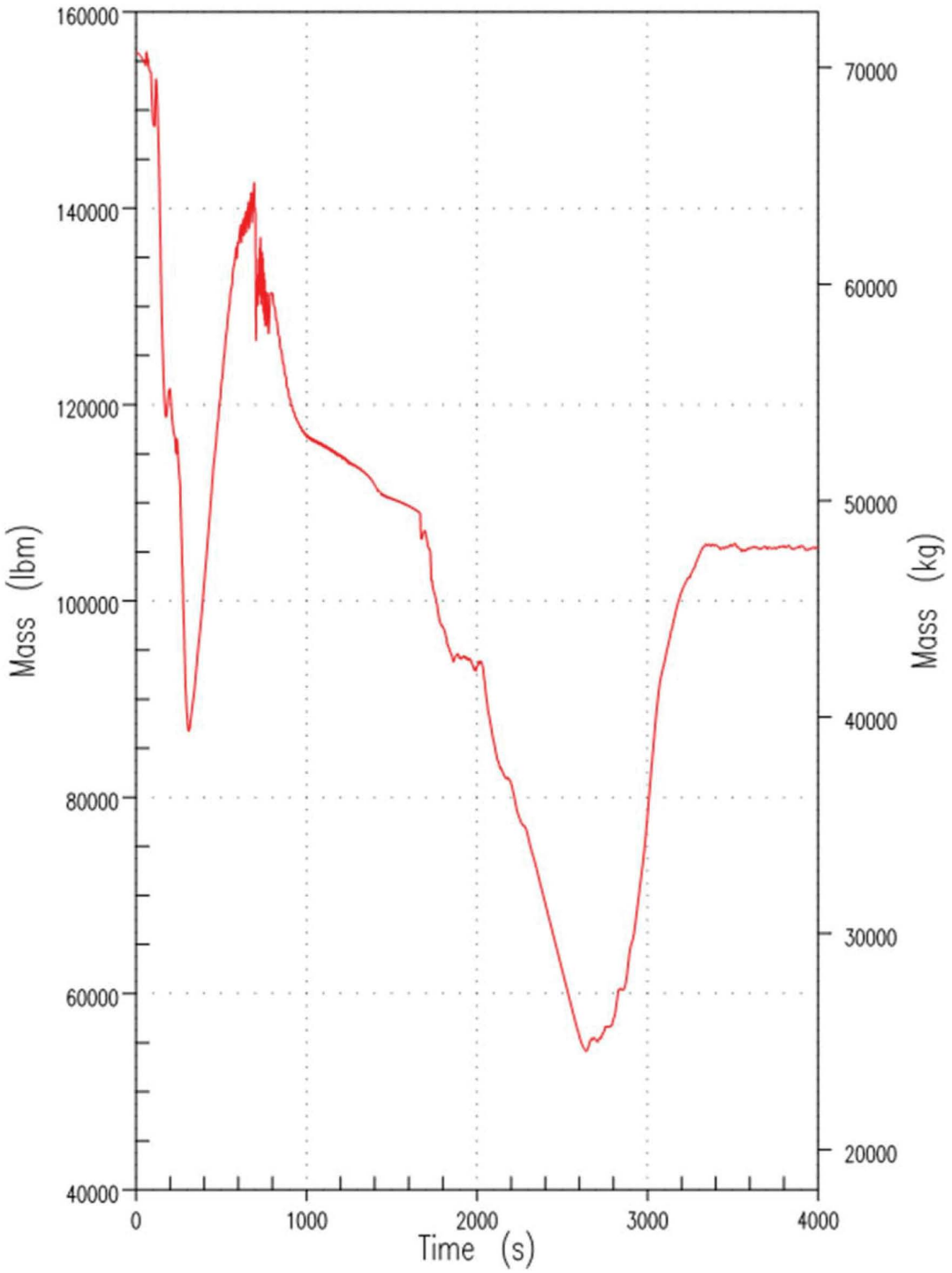

Figure 15.6.5.4B-15(b)
Inadvertent ADS – Reactor Vessel Mixture Inventory

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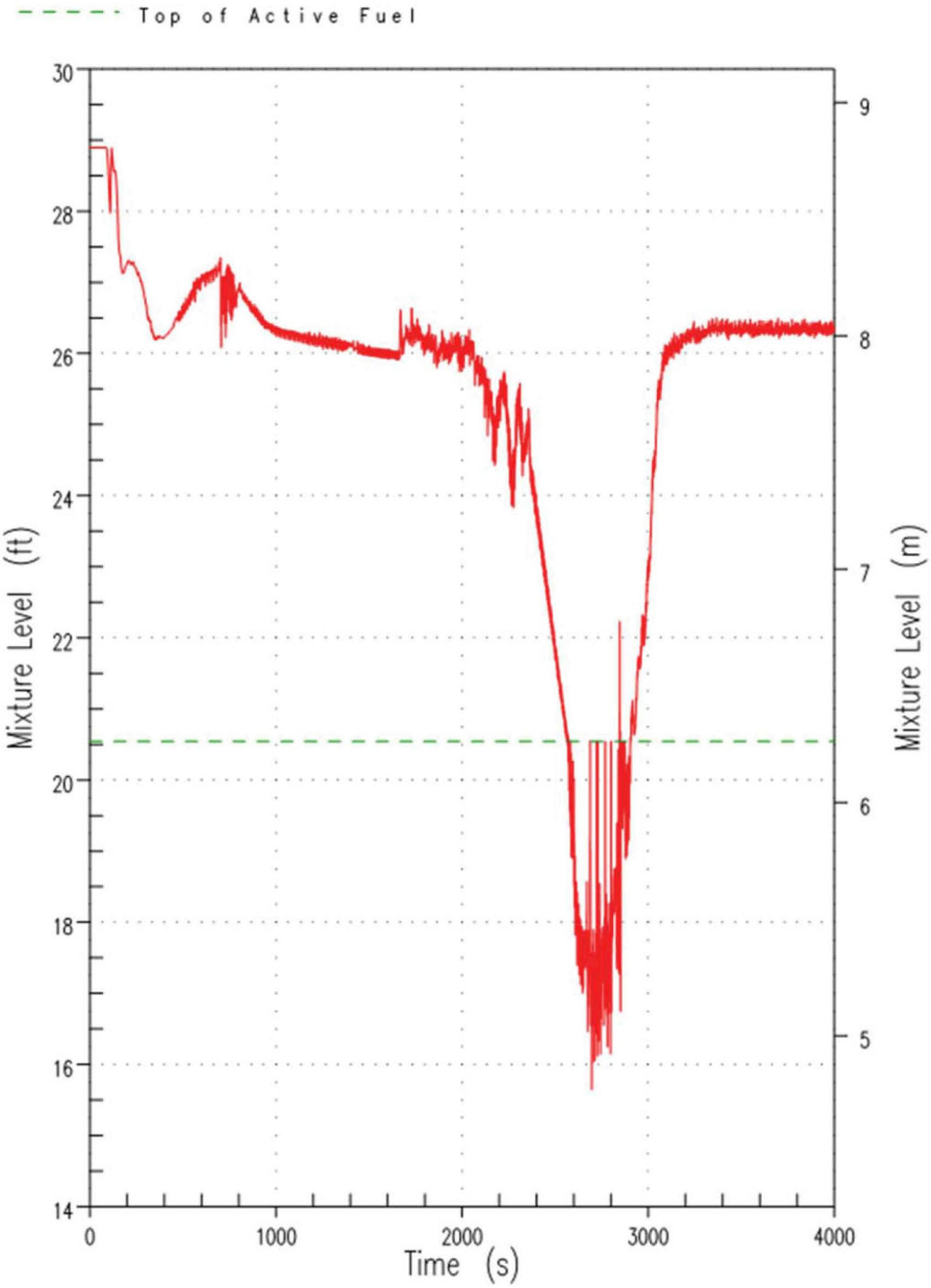

Figure 15.6.5.4B-16(a)
Inadvertent ADS – Core/Upper Plenum Mixture Level

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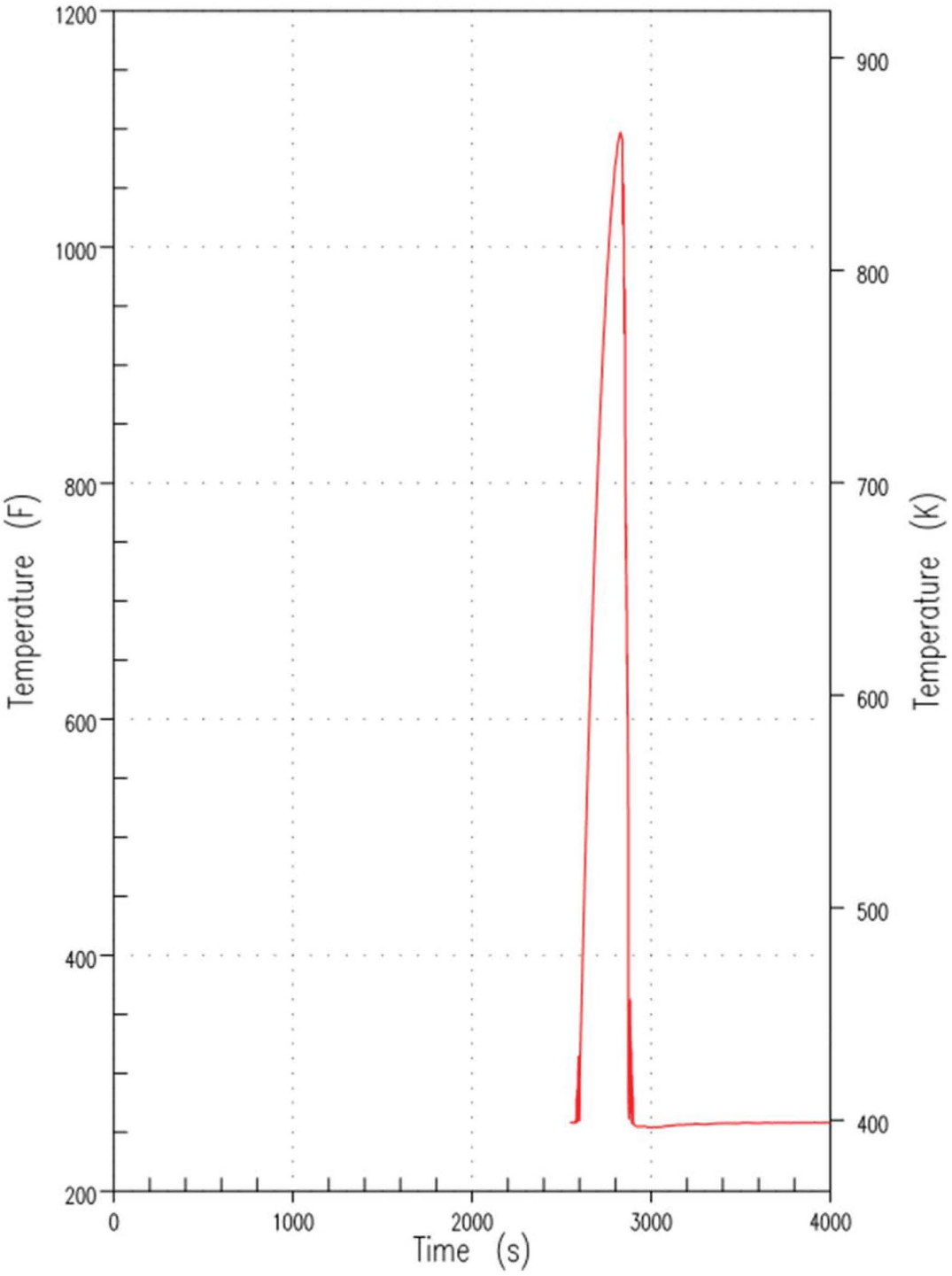

Figure 15.6.5.4B-16(b)
Inadvertent ADS – Peak Cladding Temperature

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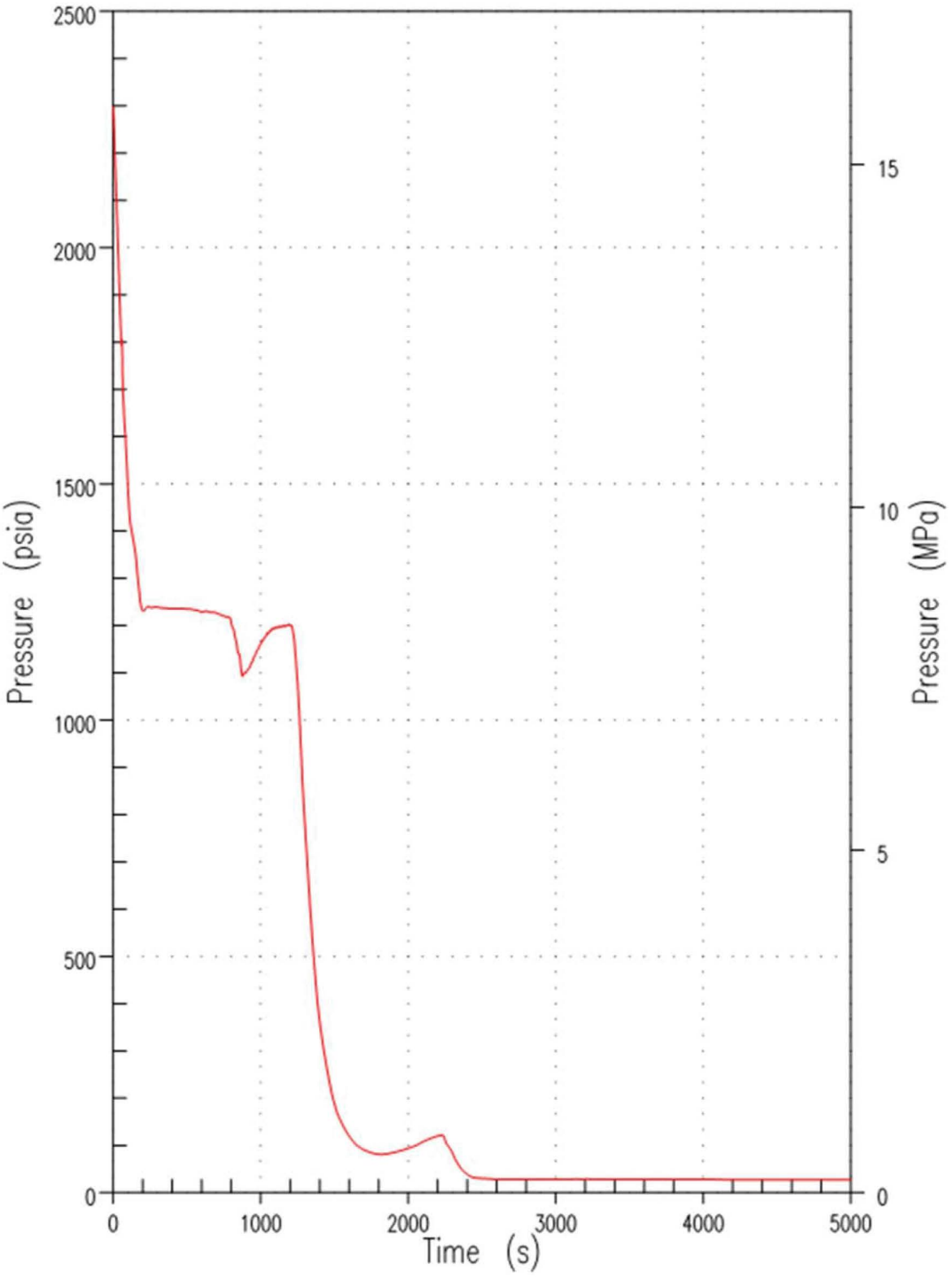

Figure 15.6.5.4B-17(a)
2-inch Cold Leg Break – RCS Pressure

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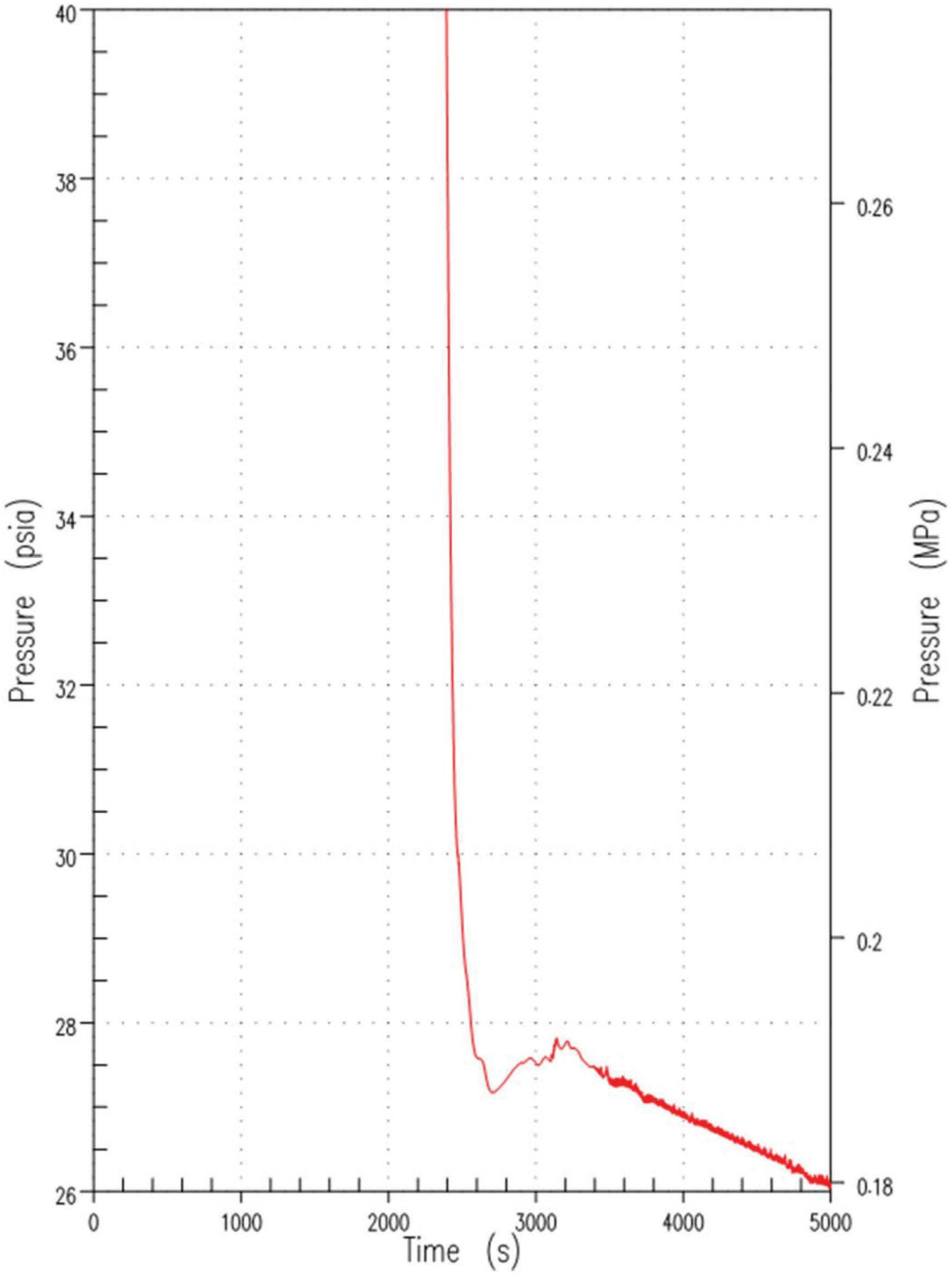

Figure 15.6.5.4B-17(b)
2-Inch Cold Leg Break – RCS Pressure (Zoomed)

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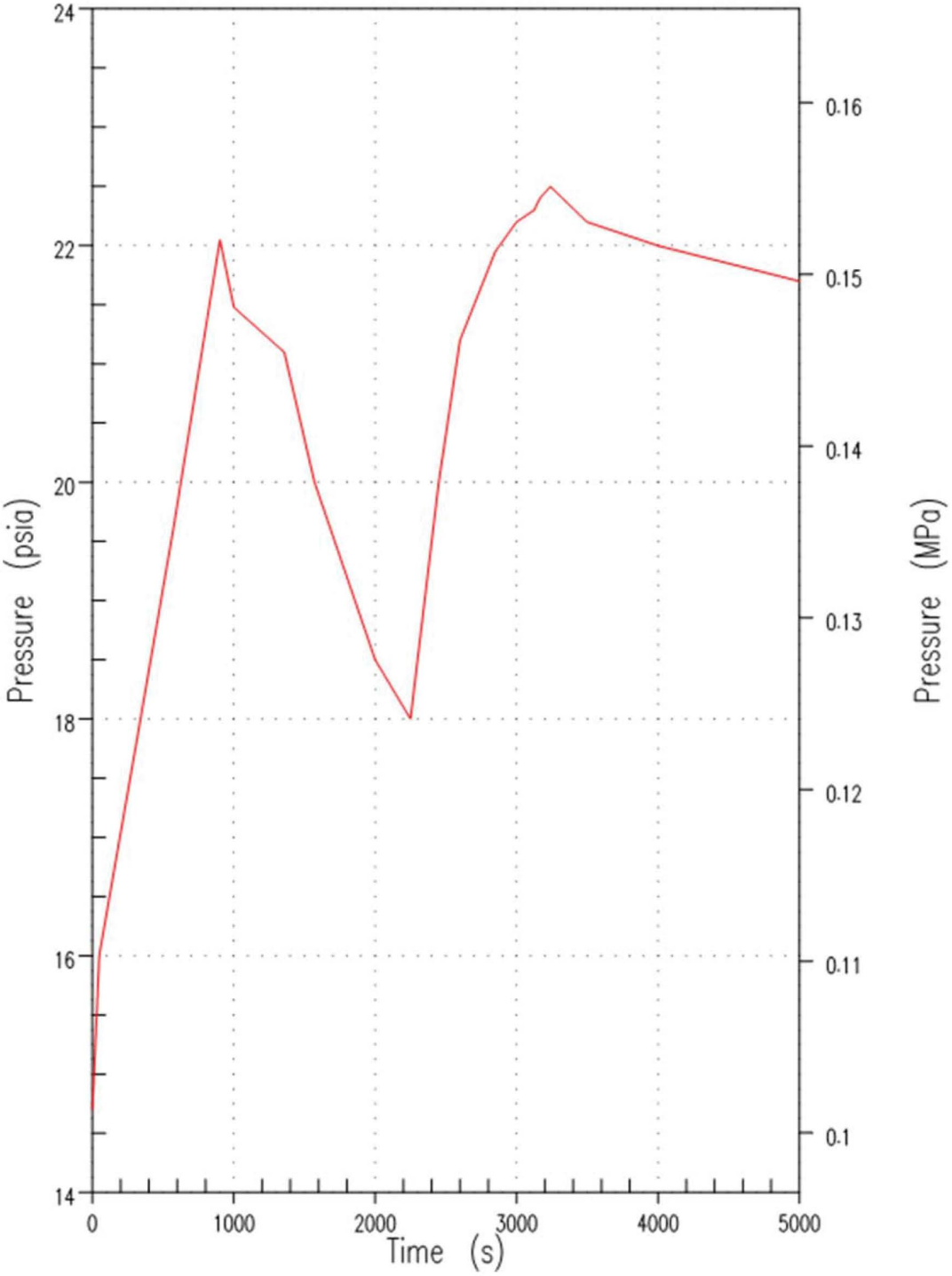

Figure 15.6.5.4B-17(c)
2-Inch Cold Leg Break – Containment Pressure

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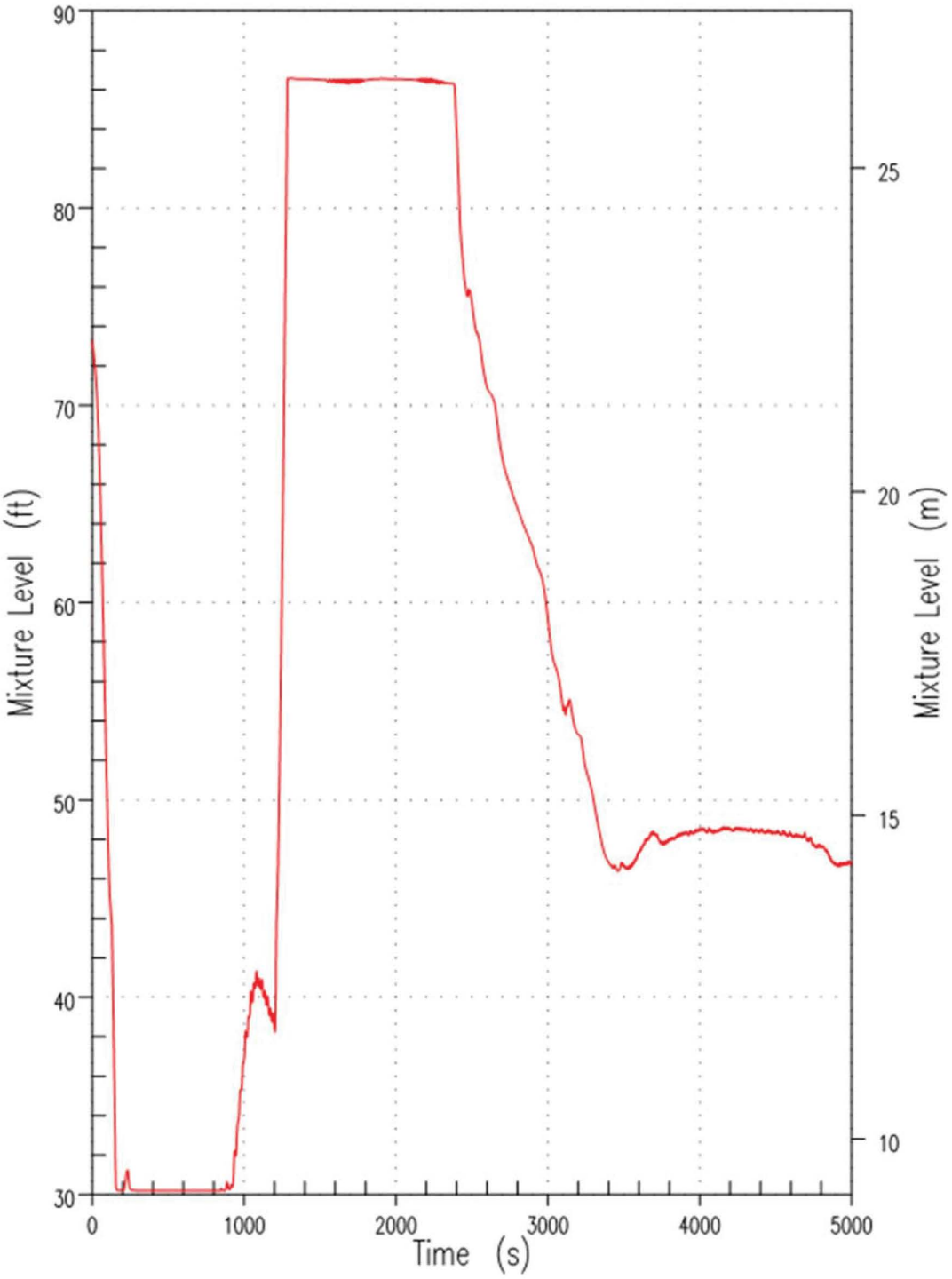

Figure 15.6.5.4B-18
2-inch Cold Leg Break – Pressurizer Mixture Level

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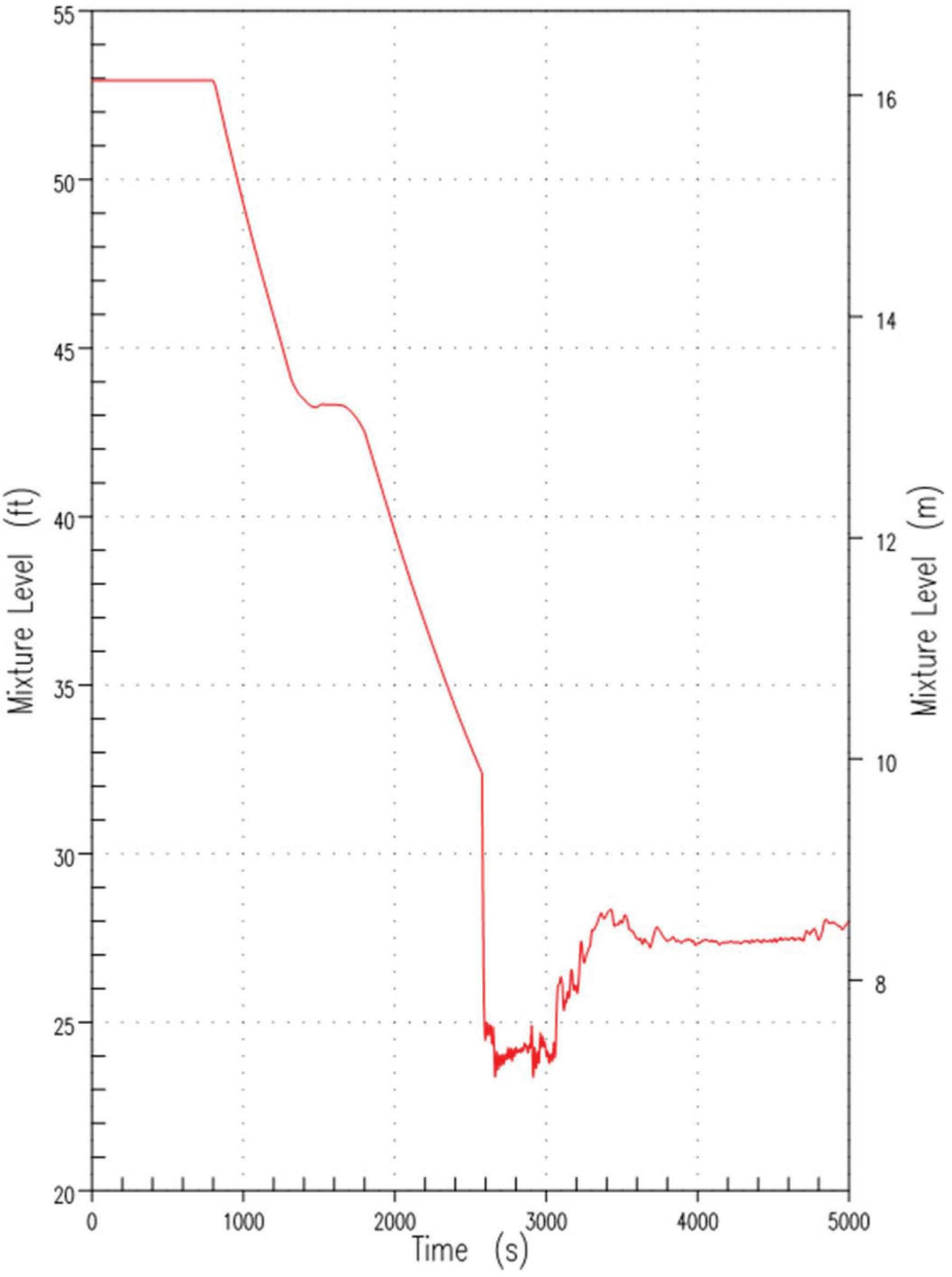

Figure 15.6.5.4B-19
2-inch Cold Leg Break – CMT-1 Mixture Level

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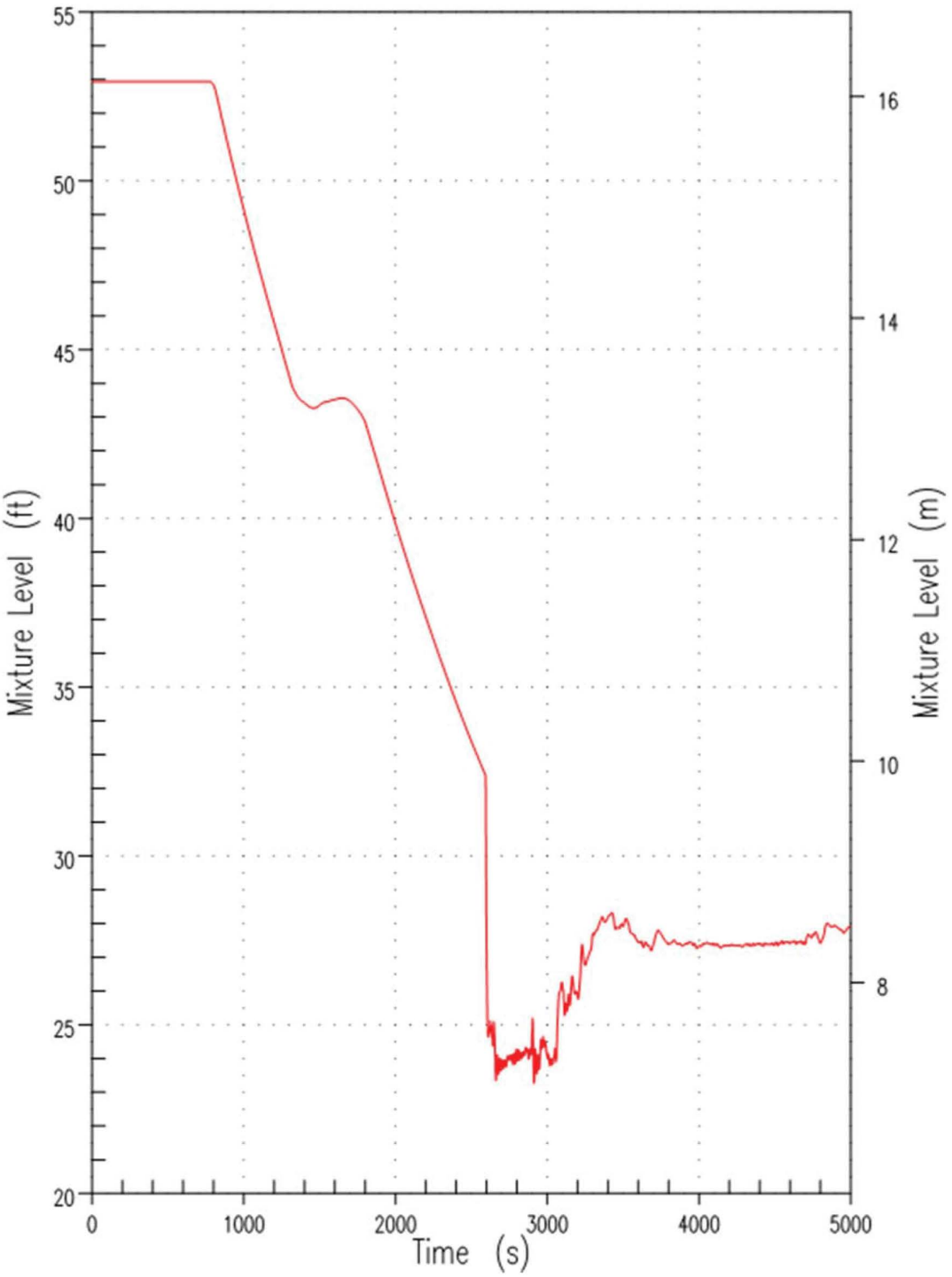

Figure 15.6.5.4B-20
2-inch Cold Leg Break – CMT-2 Mixture Level

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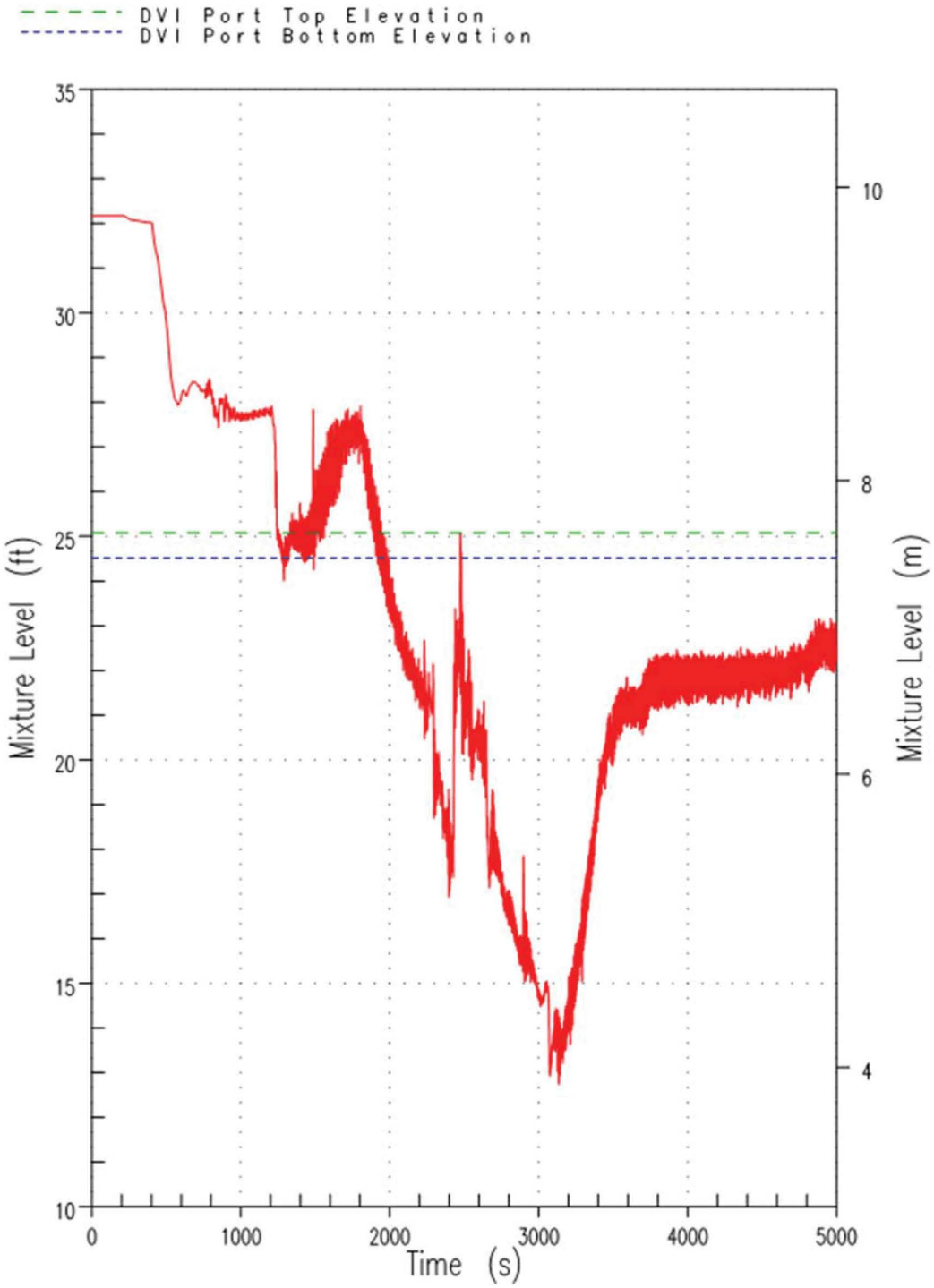

Figure 15.6.5.4B-21
2-inch Cold Leg Break – Downcomer Mixture Level

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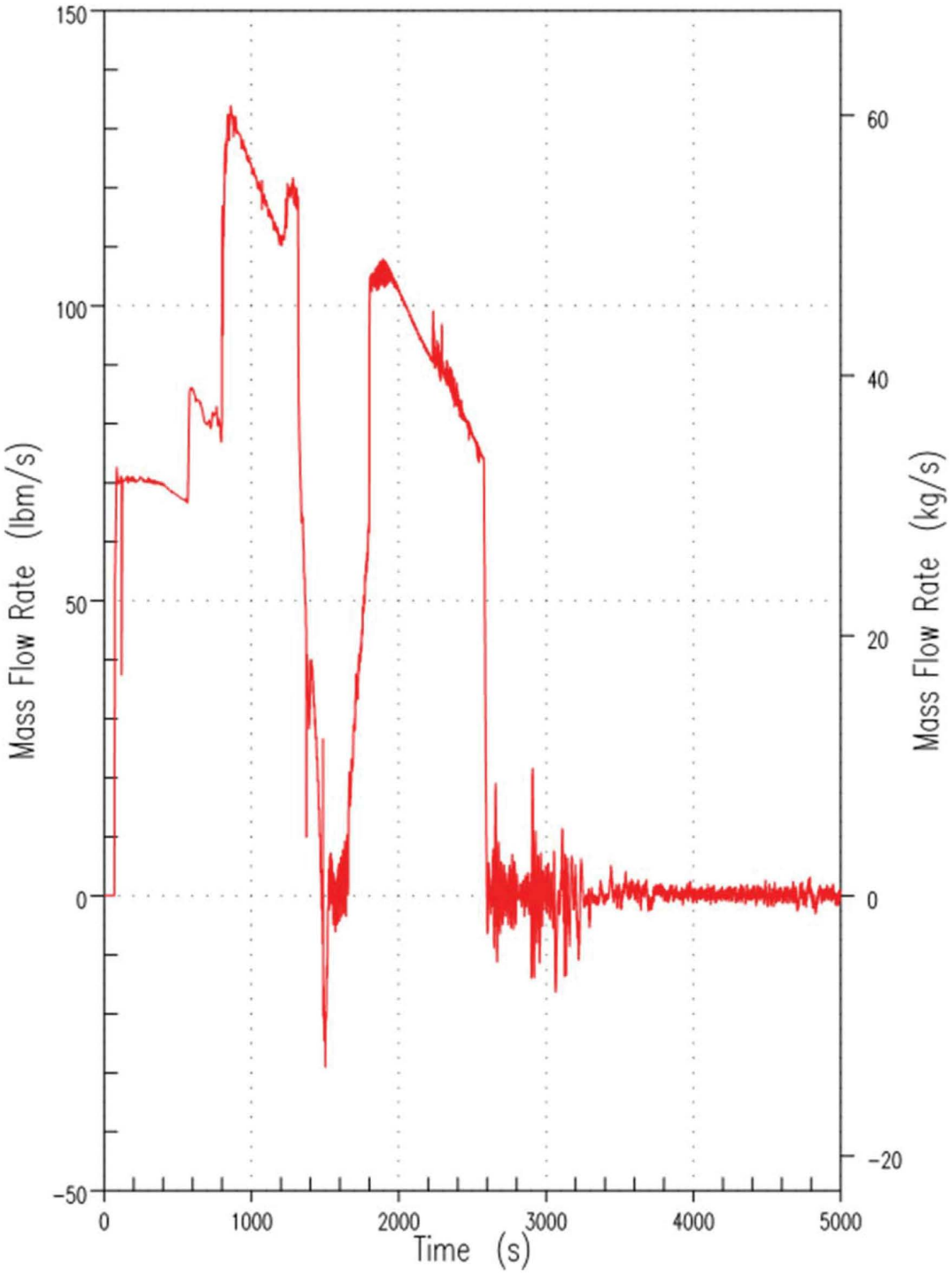

Figure 15.6.5.4B-22
2-inch Cold Leg Break – CMT-1 Injection Rate

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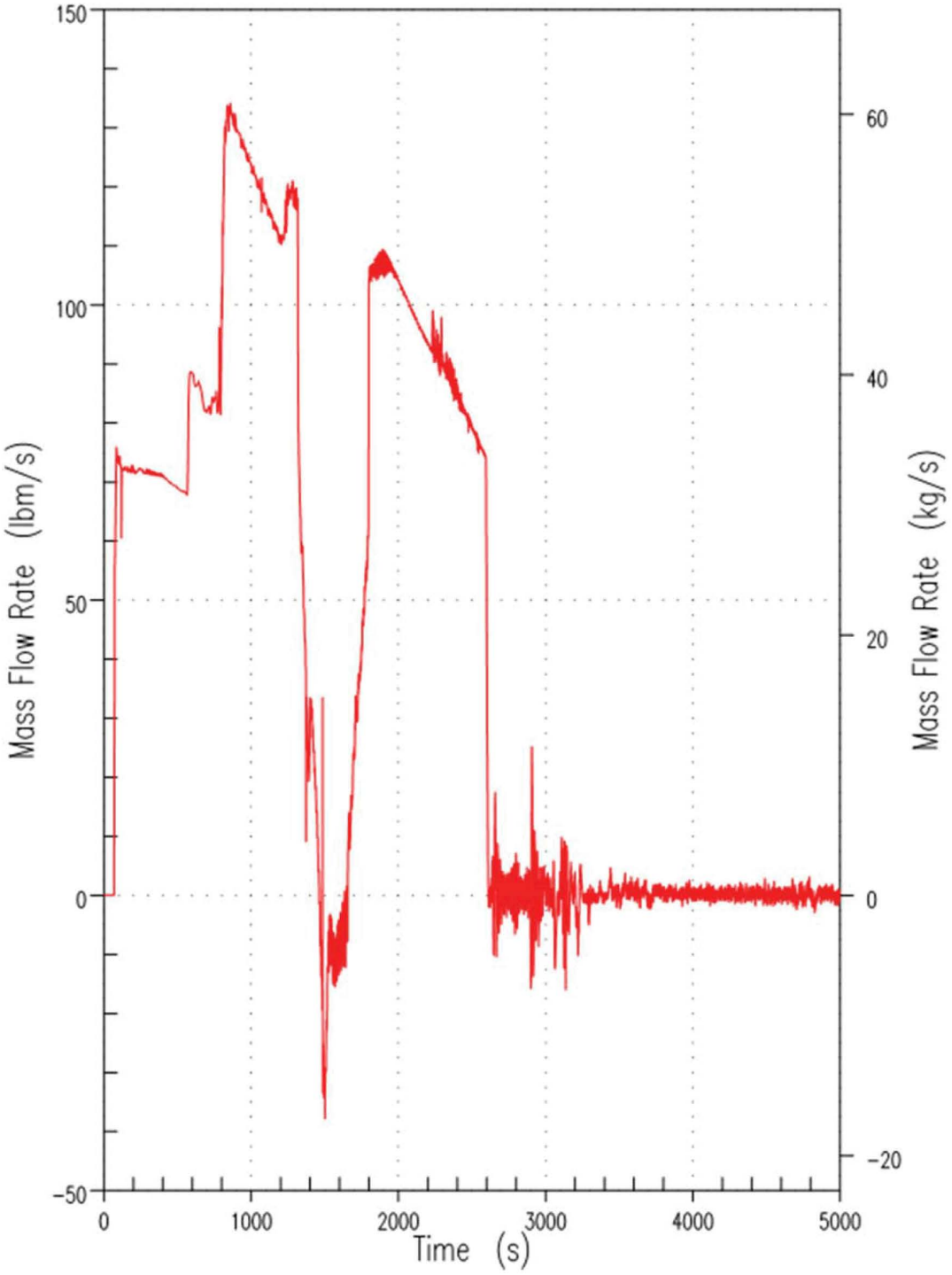

Figure 15.6.5.4B-23
2-inch Cold Leg Break – CMT-2 Injection Rate

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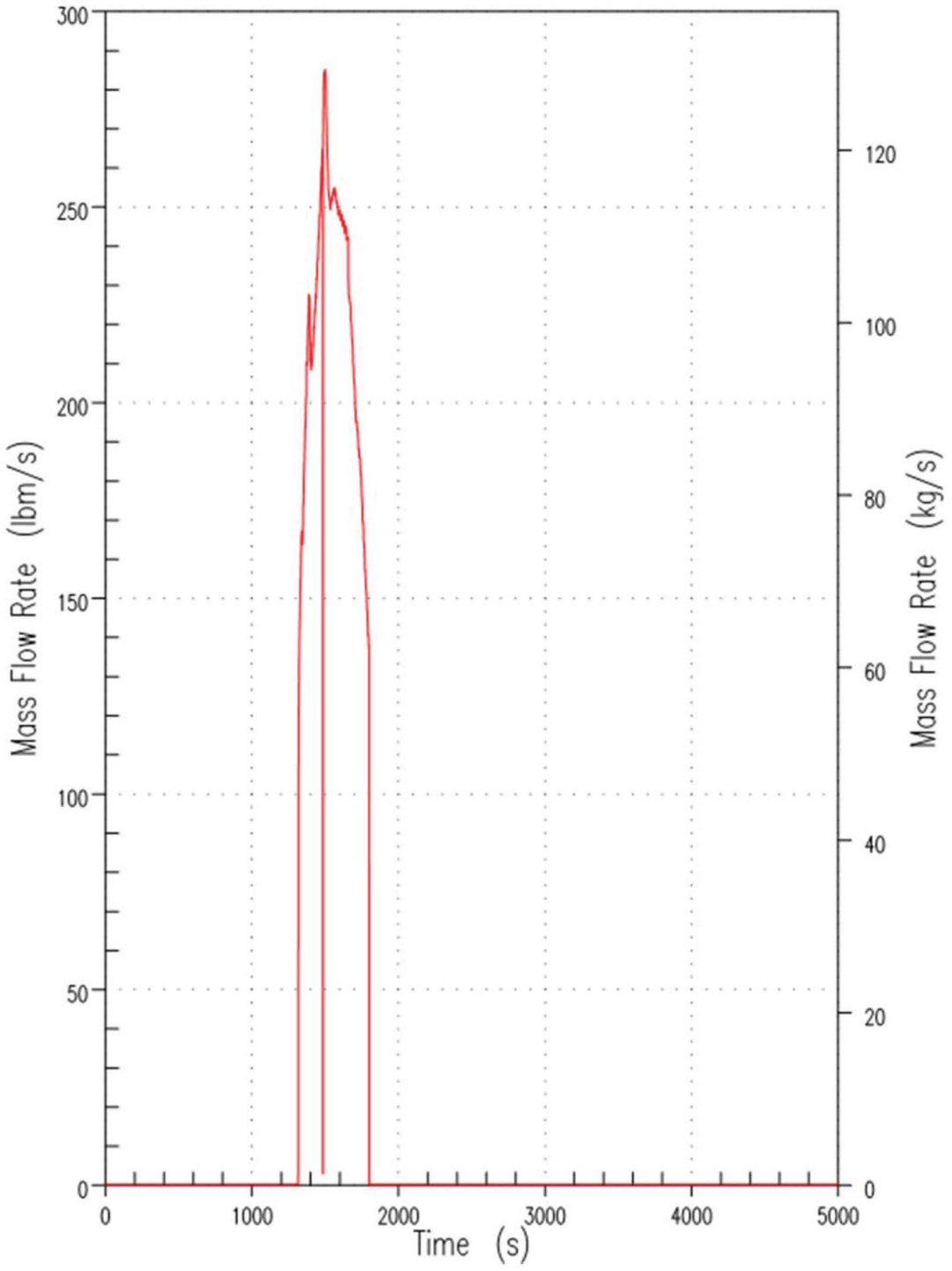

Figure 15.6.5.4B-24
2-inch Cold Leg Break – Accumulator-1 Injection Rate

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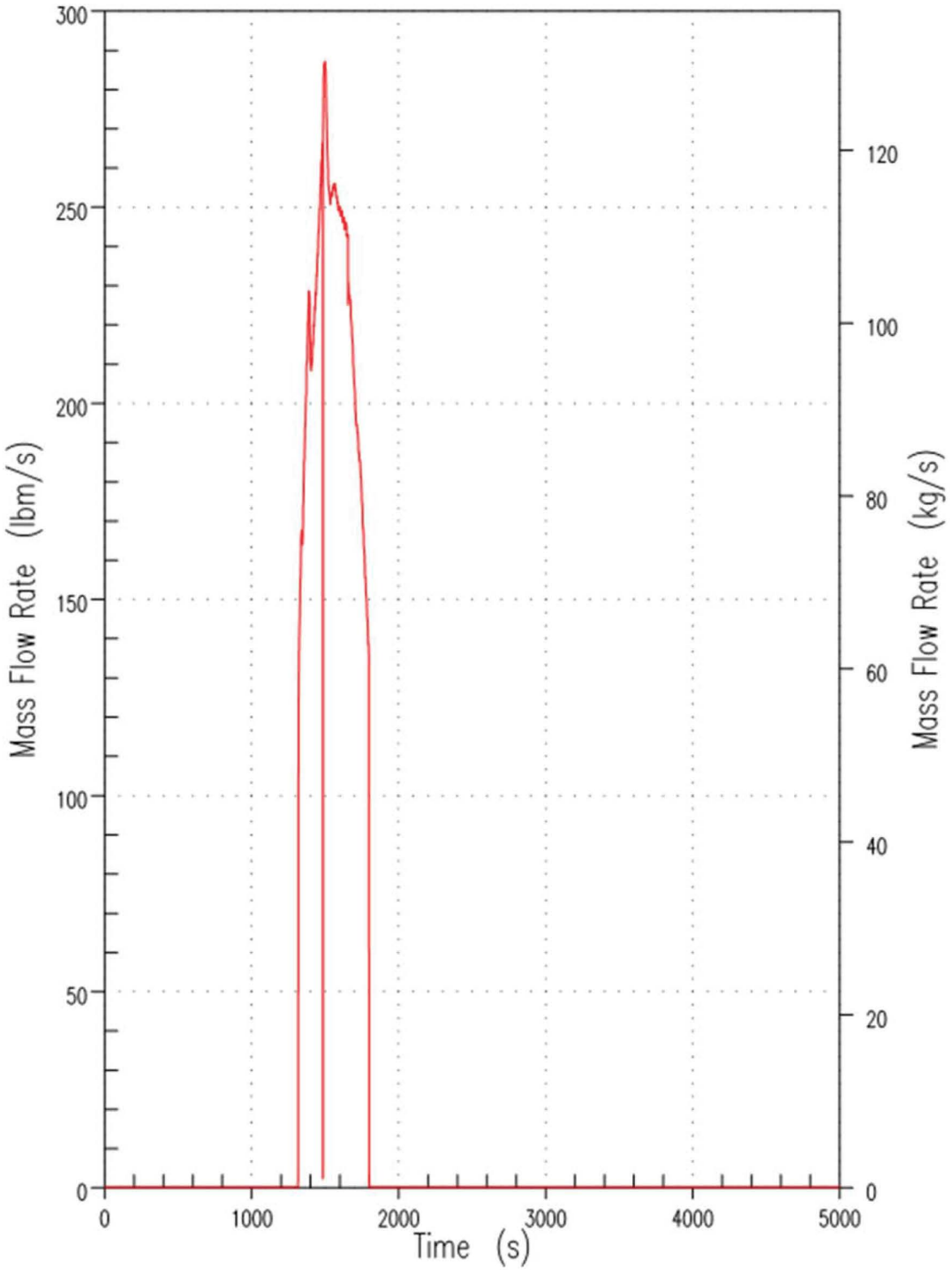

Figure 15.6.5.4B-25
2-inch Cold Leg Break – Accumulator-2 Injection Rate

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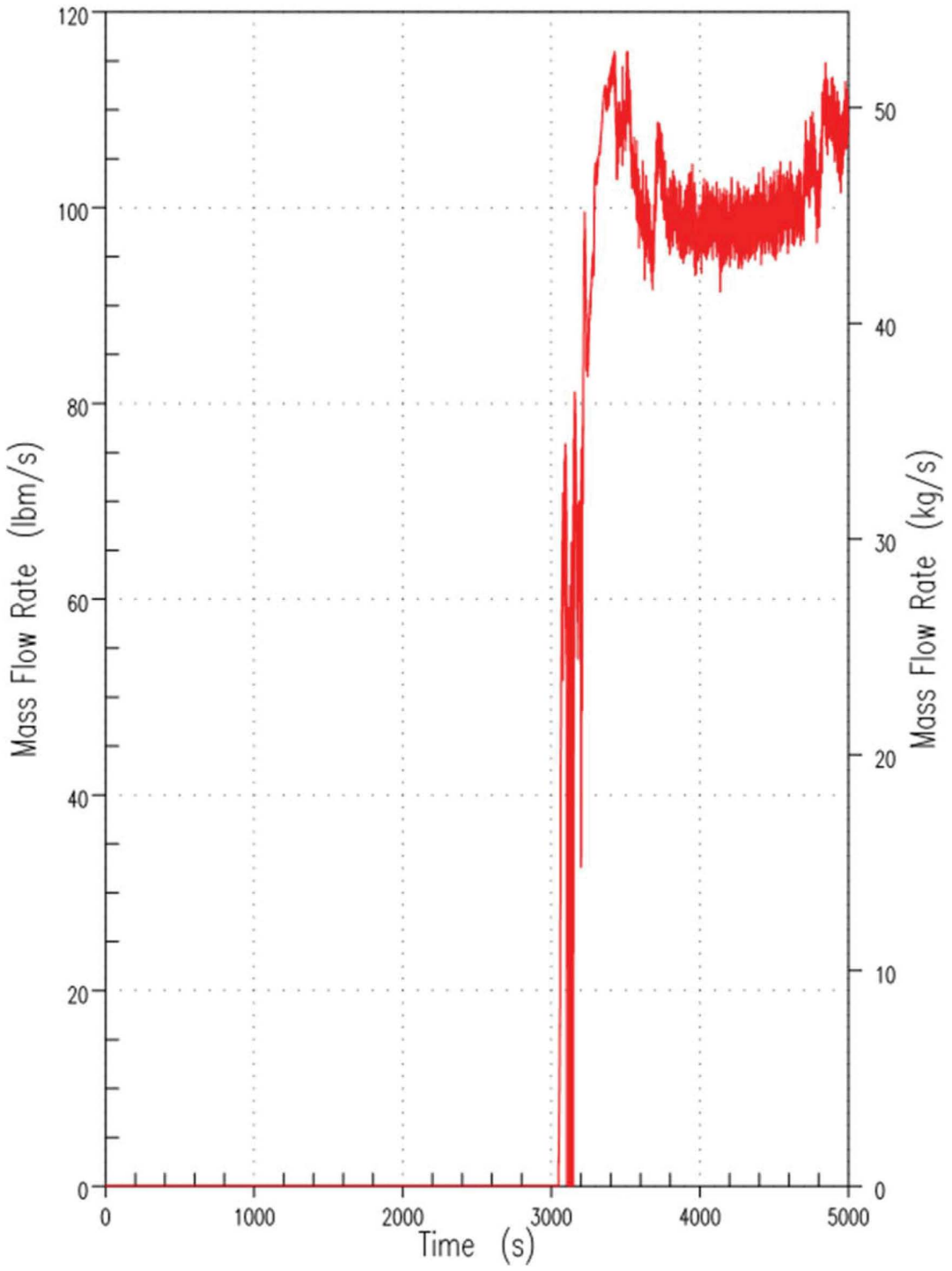

Figure 15.6.5.4B-26
2-inch Cold Leg Break – IRWST-1 Injection Rate

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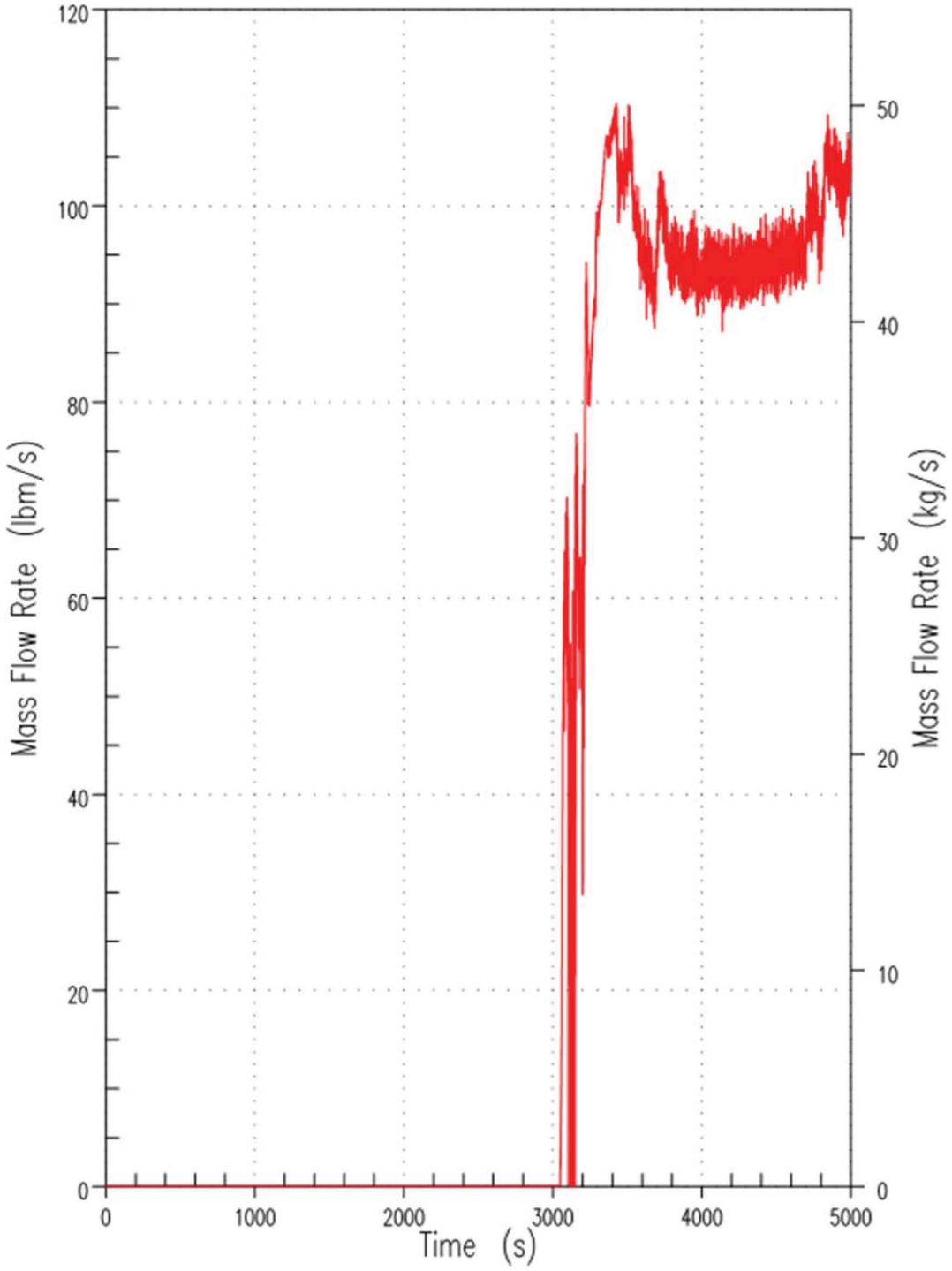

Figure 15.6.5.4B-27
2-inch Cold Leg Break – IRWST-2 Injection Rate

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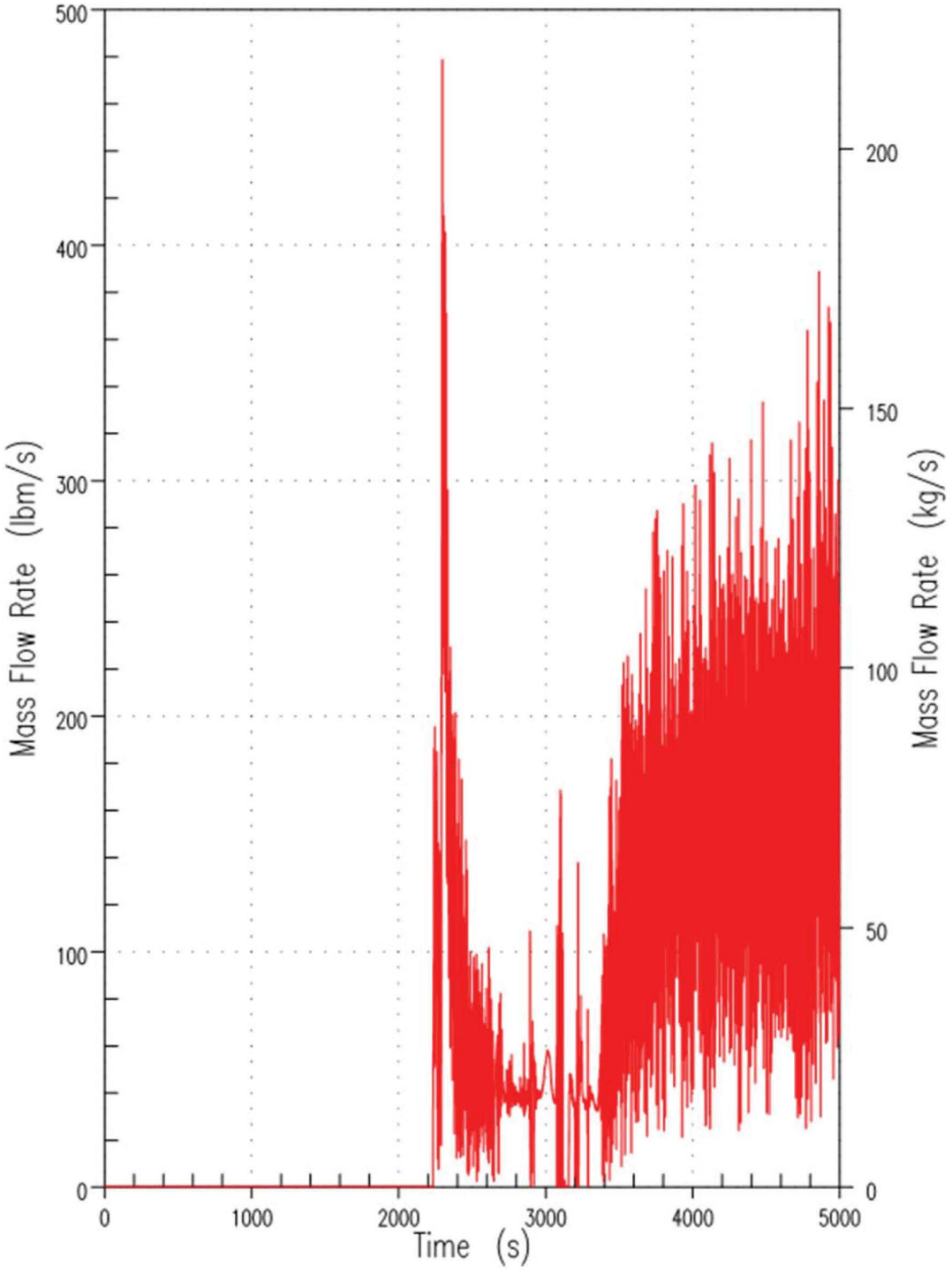

Figure 15.6.5.4B-28(a)
2-inch Cold Leg Break – ADS-4 Liquid Discharge

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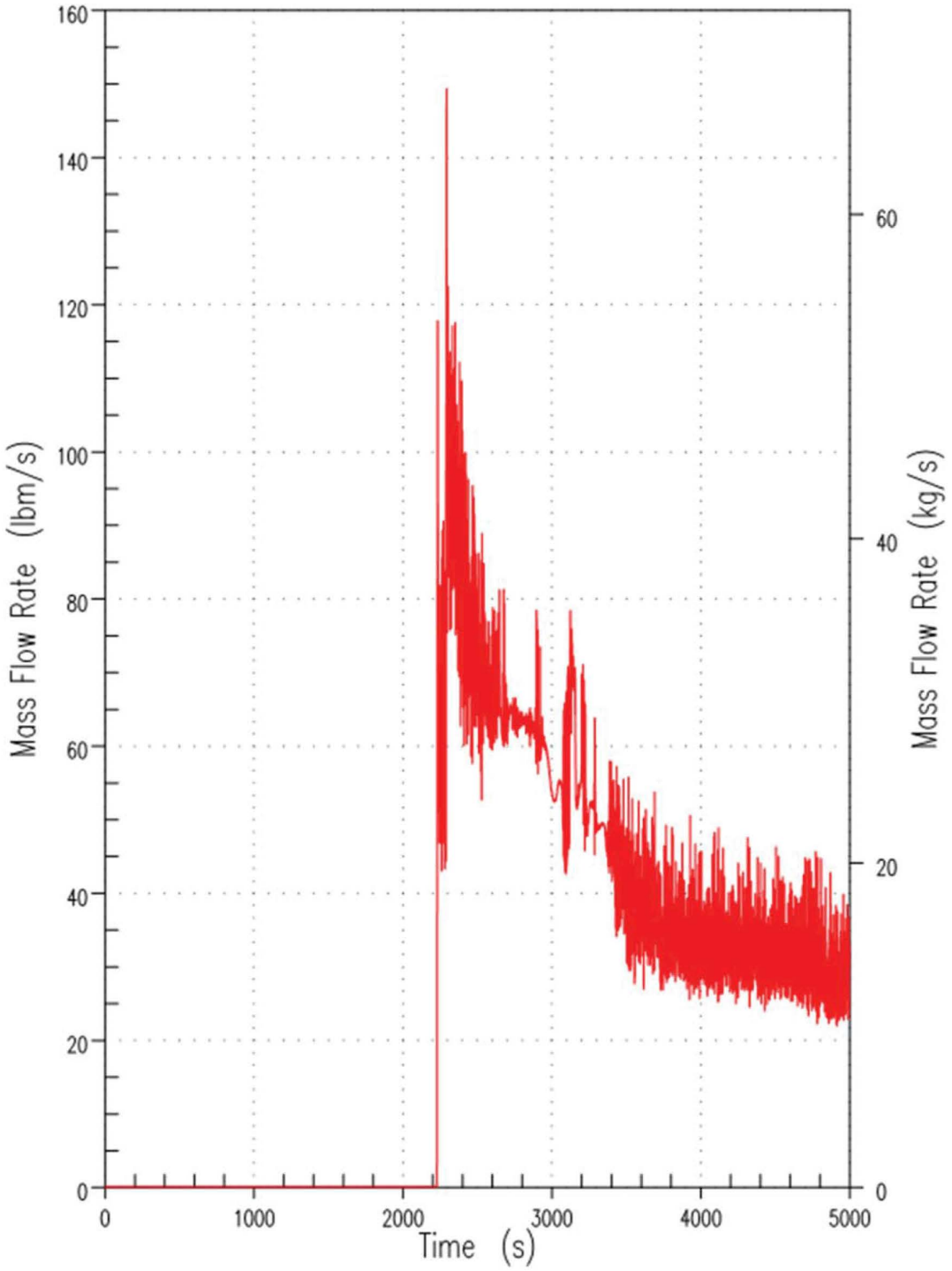

Figure 15.6.5.4B-28(b)
2-Inch Cold Leg Break – ADS-4 Vapor Discharge

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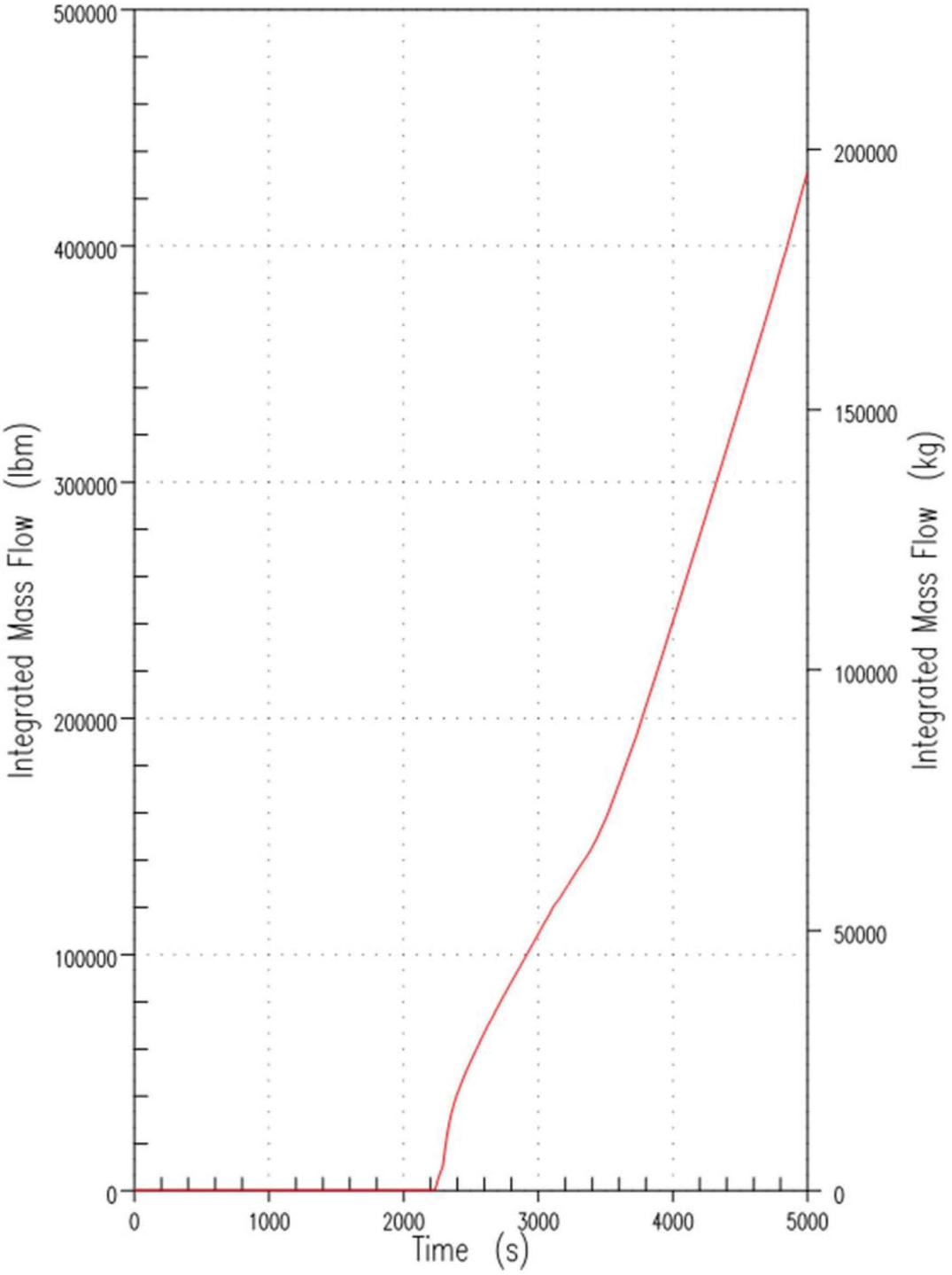

Figure 15.6.5.4B-28(c)
2-Inch Cold Leg Break – ADS-4 Integrated Discharge

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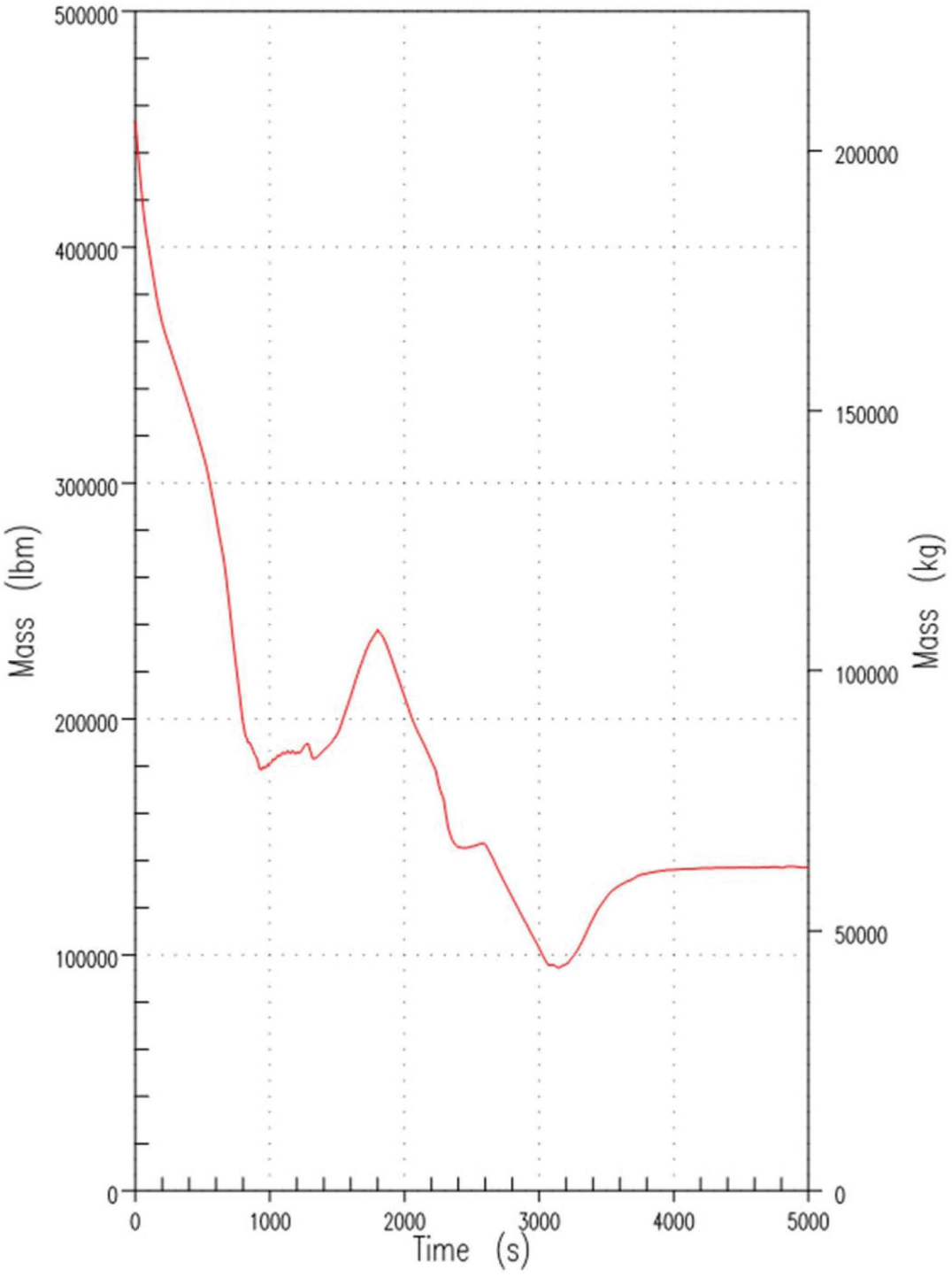

Figure 15.6.5.4B-29(a)
2-inch Cold Leg Break – RCS System Inventory

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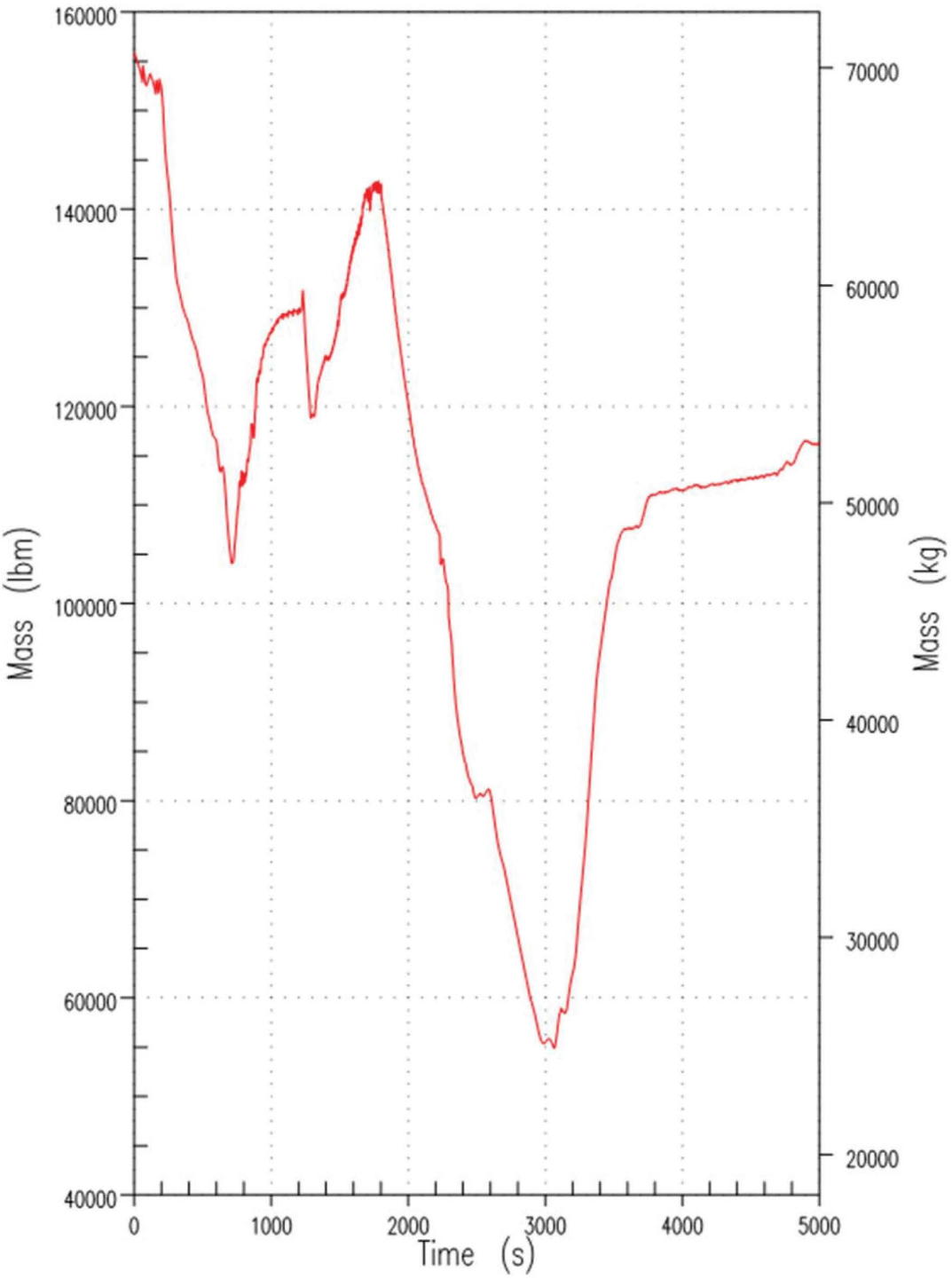

Figure 15.6.5.4B-29(b)
2-Inch Cold Leg Break – Reactor Vessel Mixture Inventory

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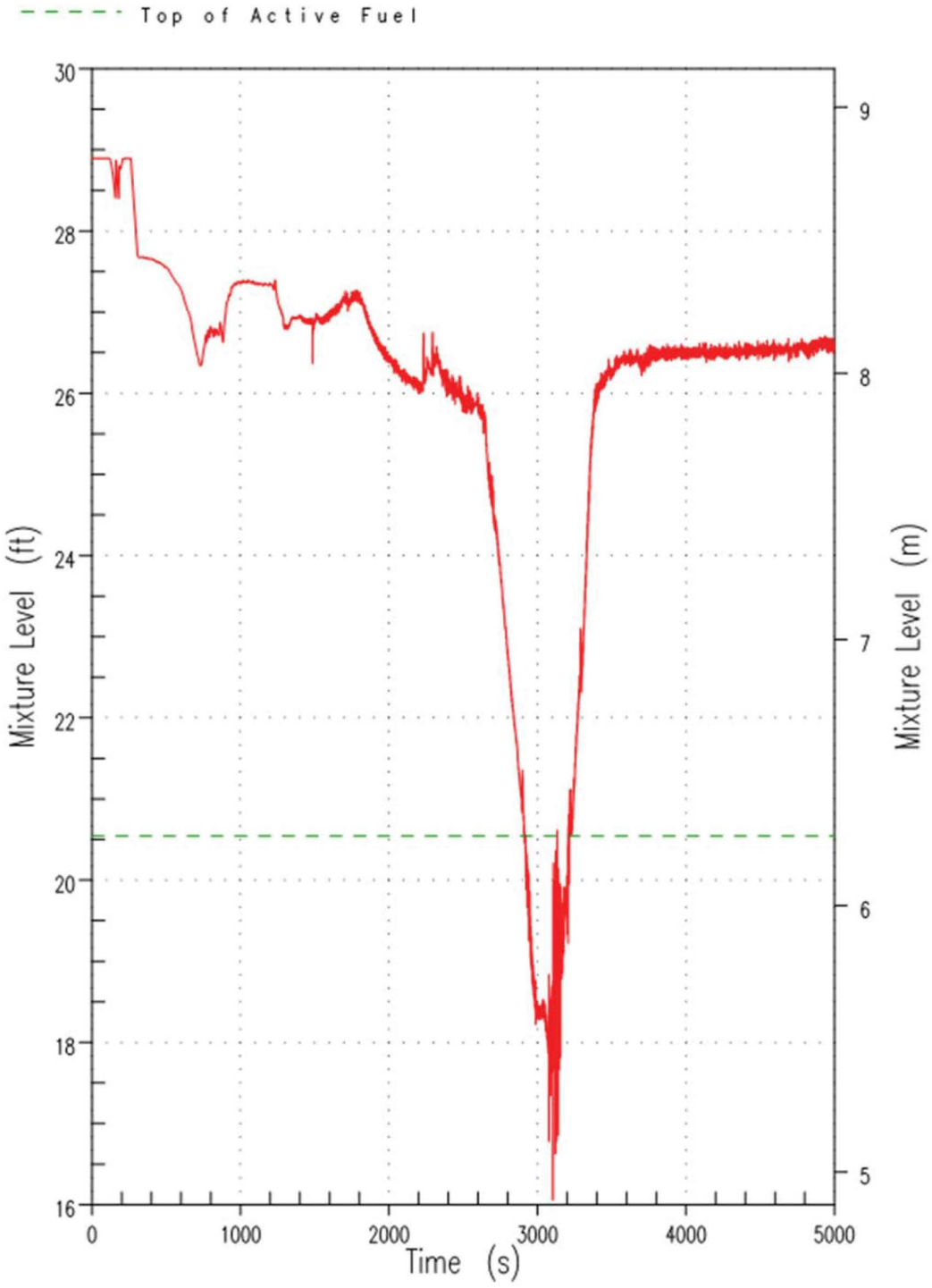

Figure 15.6.5.4B-30(a)
2-inch Cold Leg Break – Core/Upper Plenum Mixture Level

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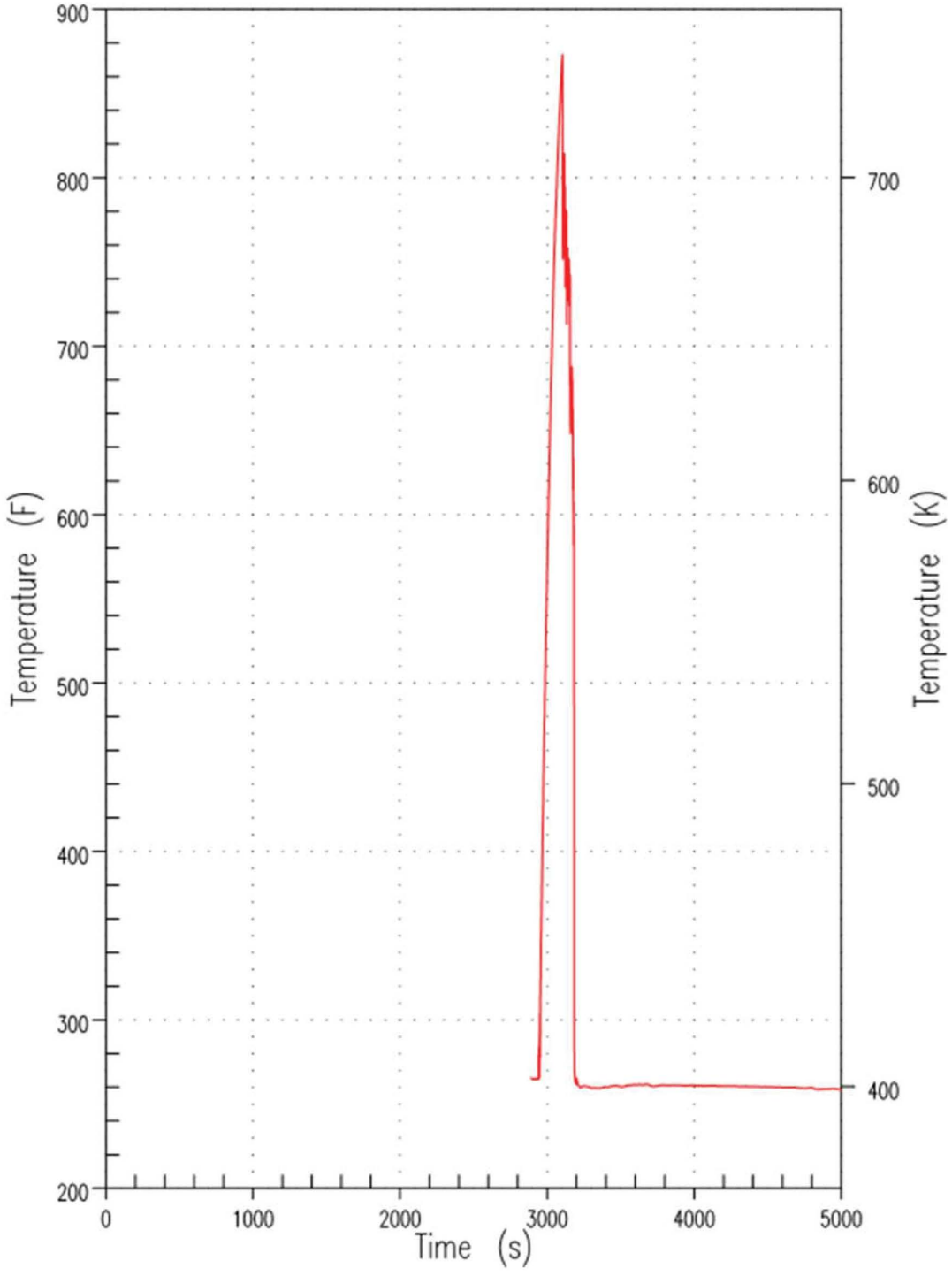

Figure 15.6.5.4B-30(b)
2-Inch Cold Leg Break – Peak Cladding Temperature

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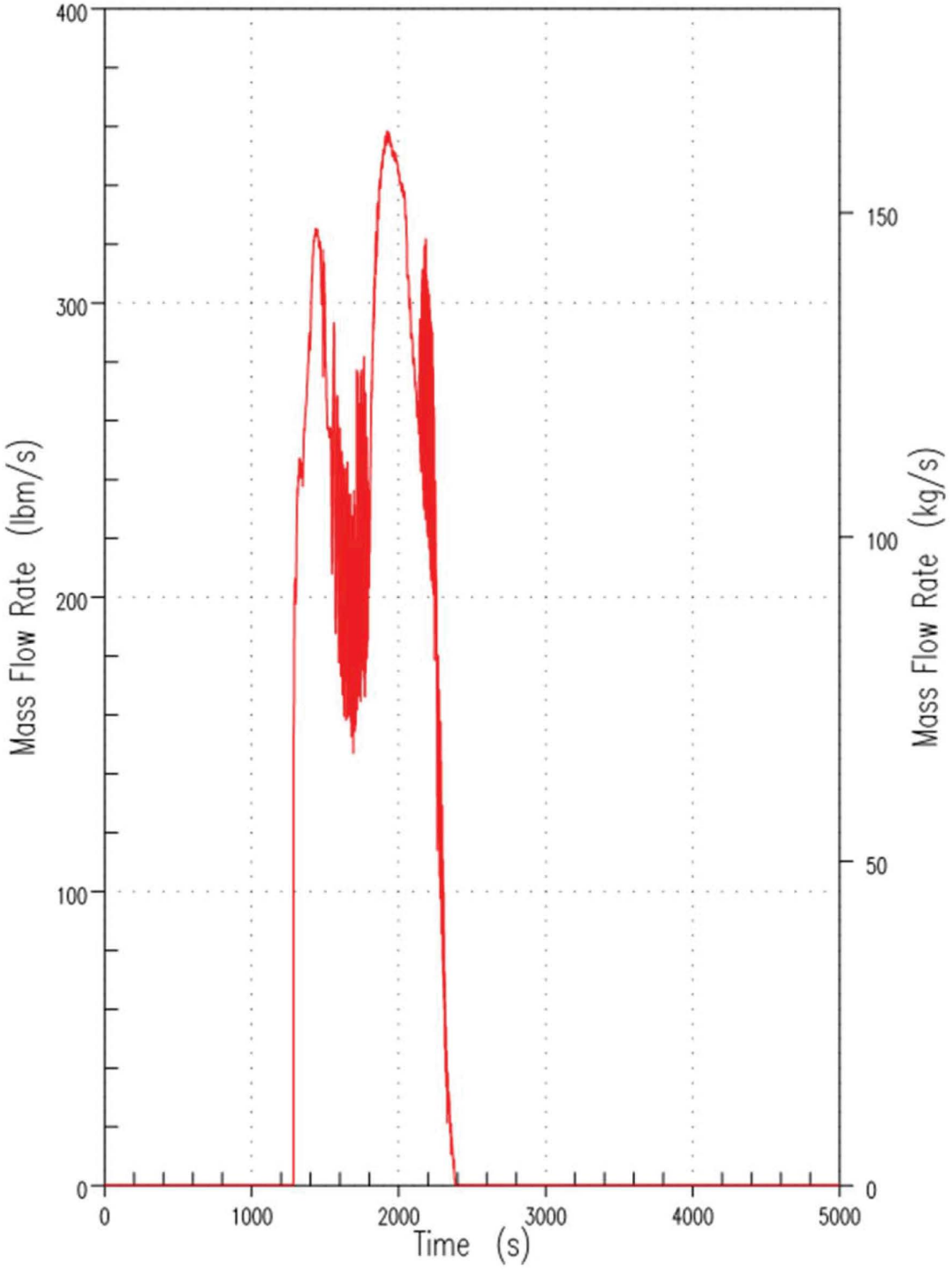

Figure 15.6.5.4B-31(a)
2-inch Cold Leg Break – ADS 1-3 Liquid Discharge

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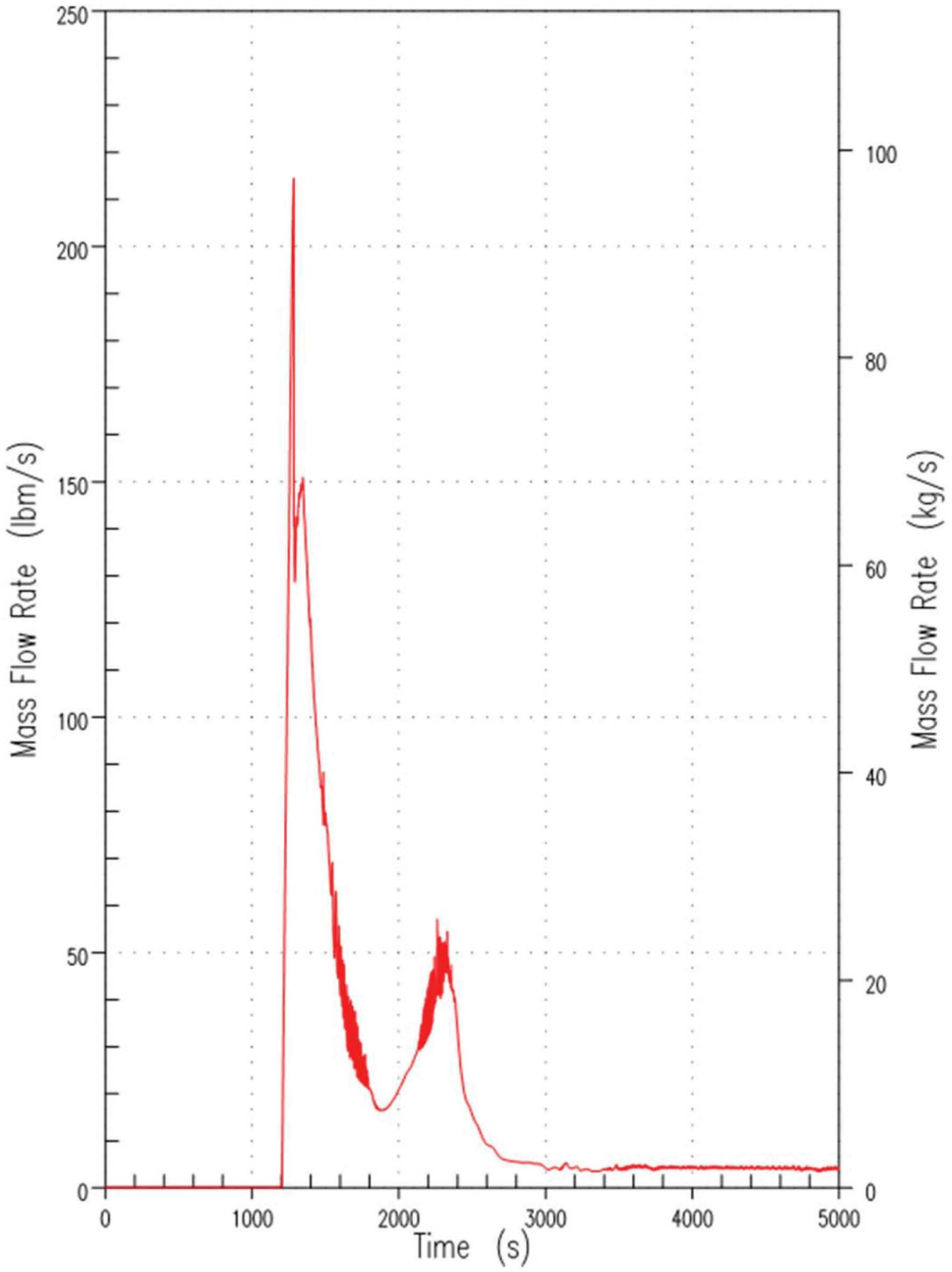

Figure 15.6.5.4B-31(b)
2-Inch Cold Leg Break – ADS 1-3 Vapor Discharge

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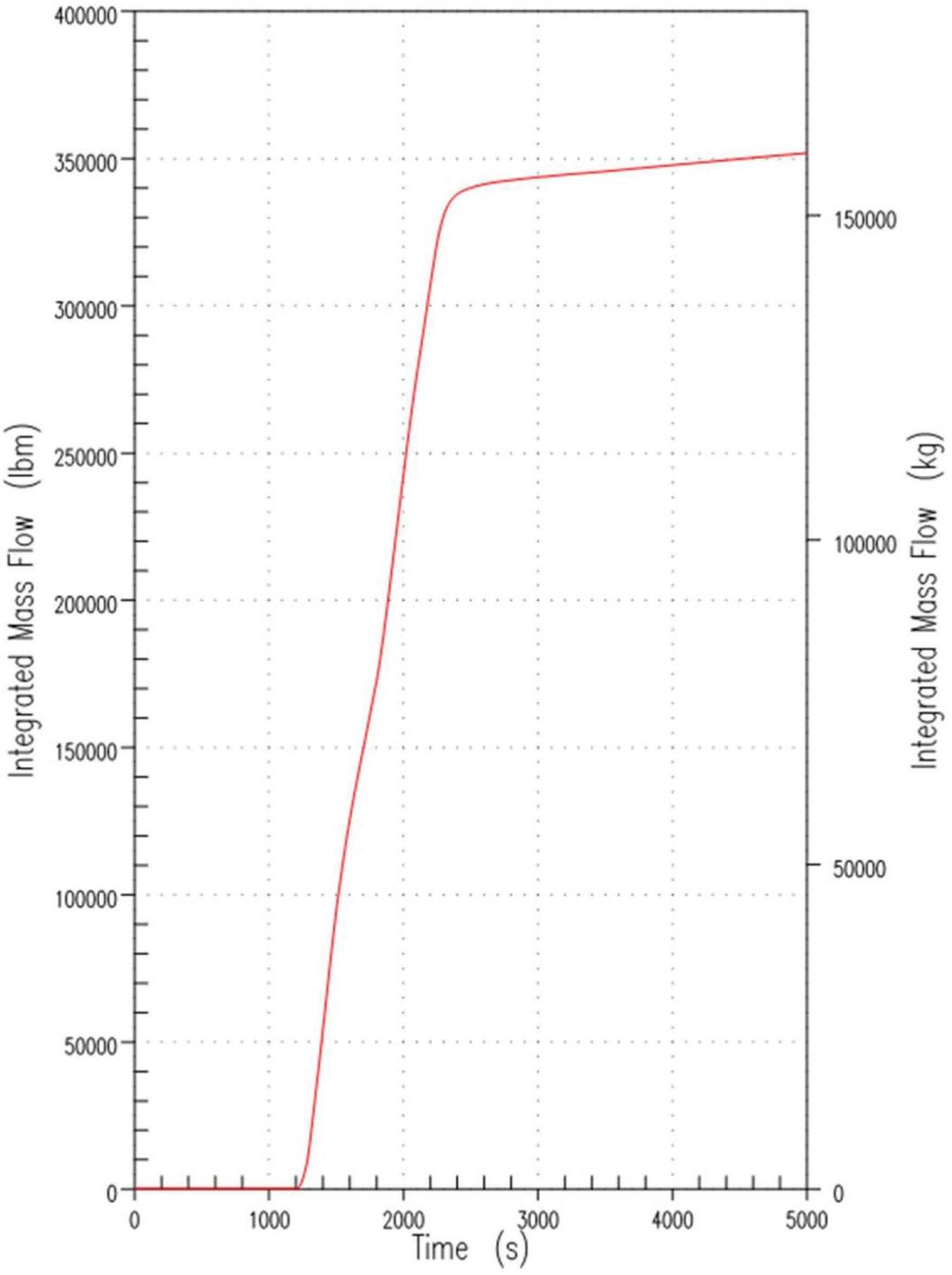

Figure 15.6.5.4B-31(c)
2-Inch Cold Leg Break – ADS 1-3 Integrated Discharge

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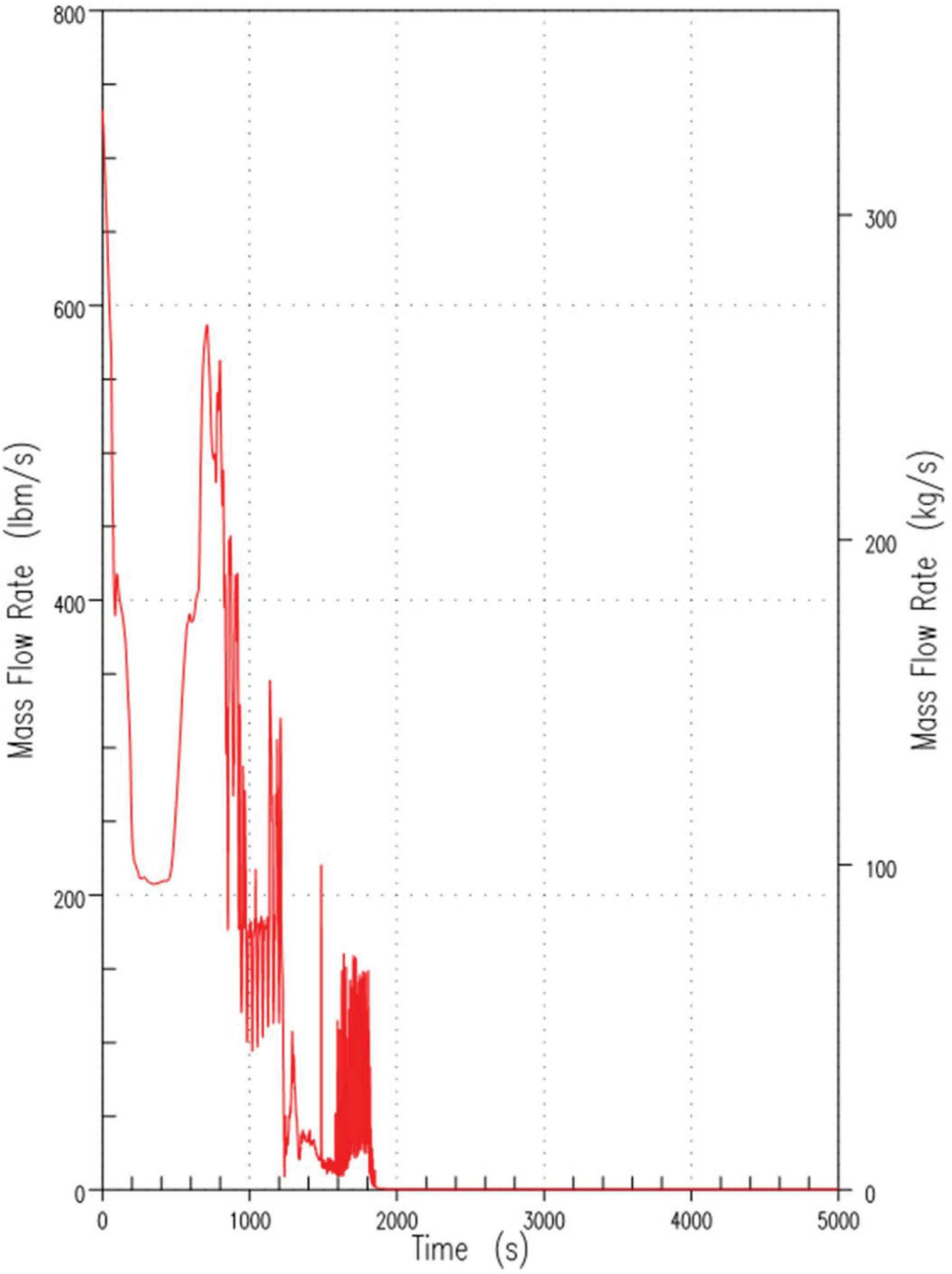

Figure 15.6.5.4B-32
2-inch Cold Leg Break – Liquid Break Discharge

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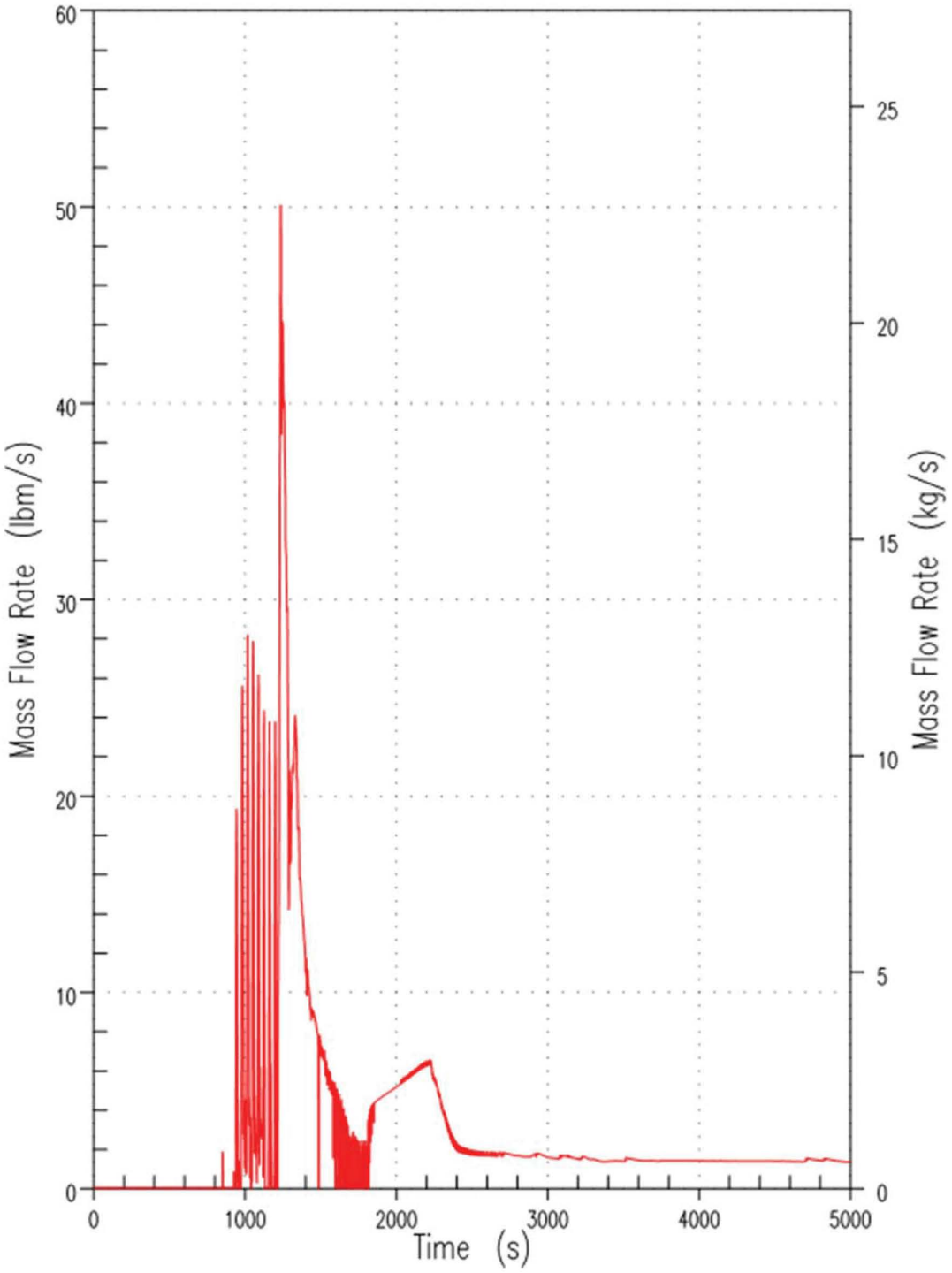

Figure 15.6.5.4B-33
2-inch Cold Leg Break – Vapor Break Discharge

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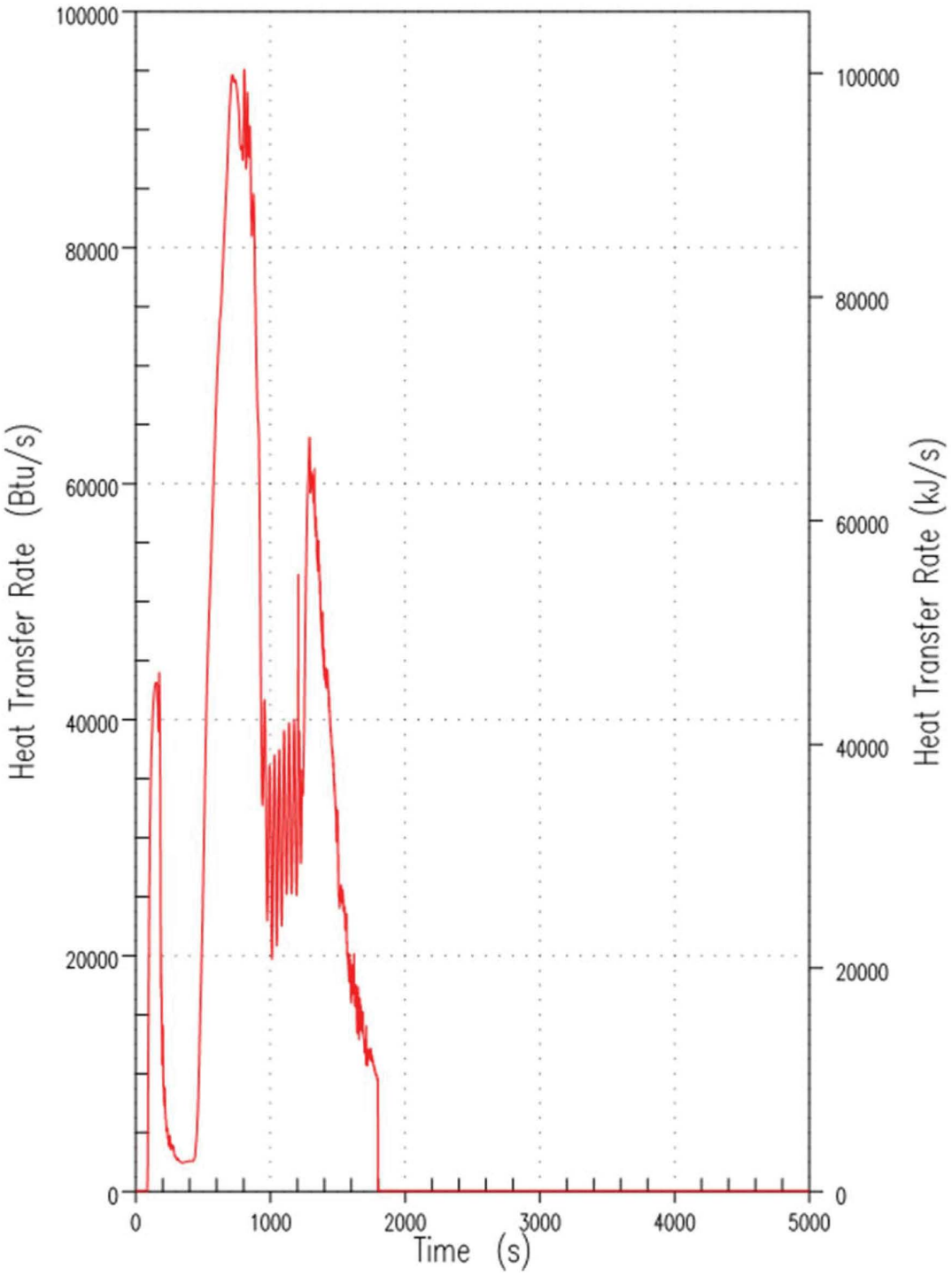

Figure 15.6.5.4B-34
2-inch Cold Leg Break – PRHR Heat Removal Rate

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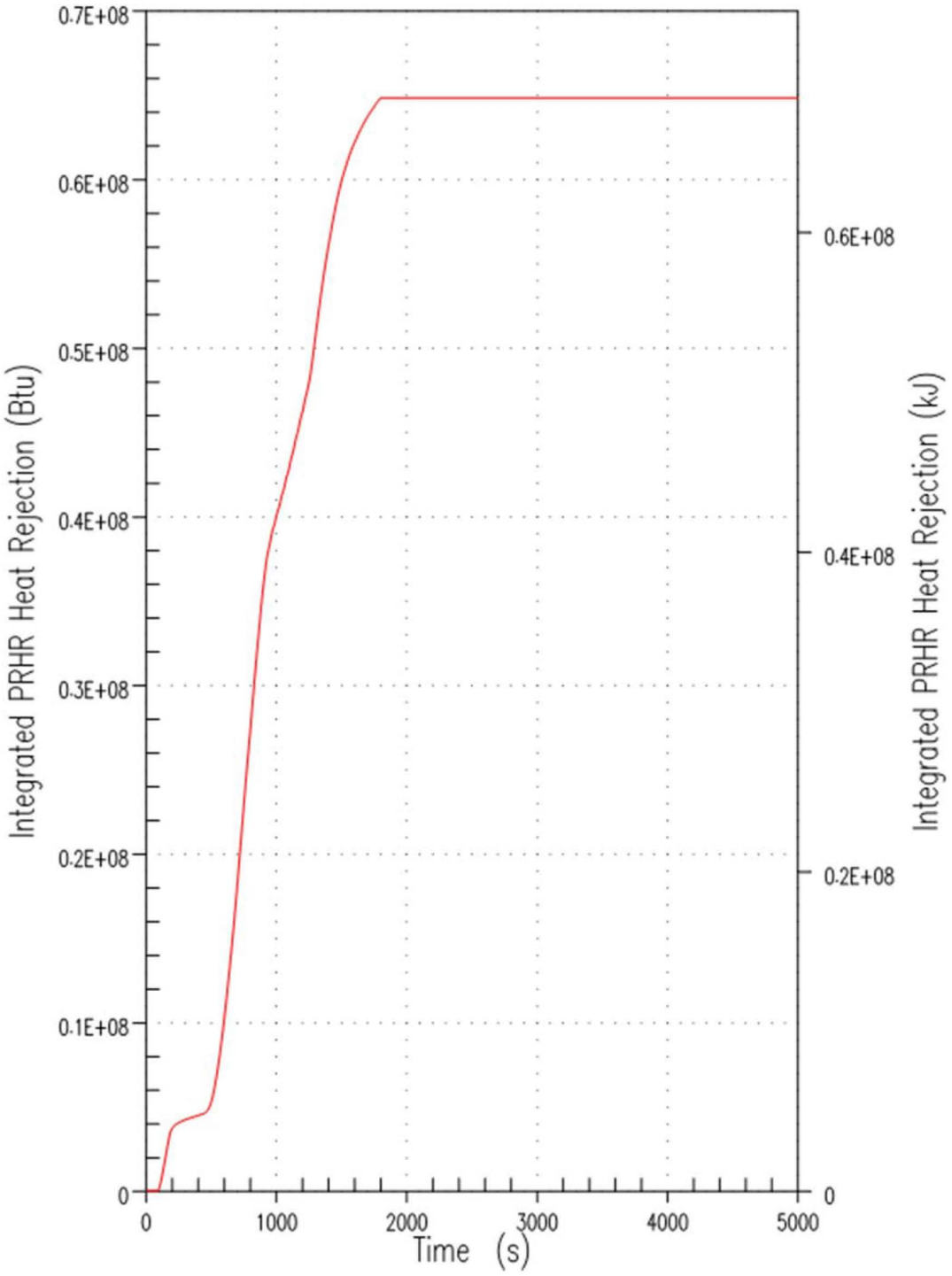

Figure 15.6.5.4B-35
2-inch Cold Leg Break – Integrated PRHR Heat Removal

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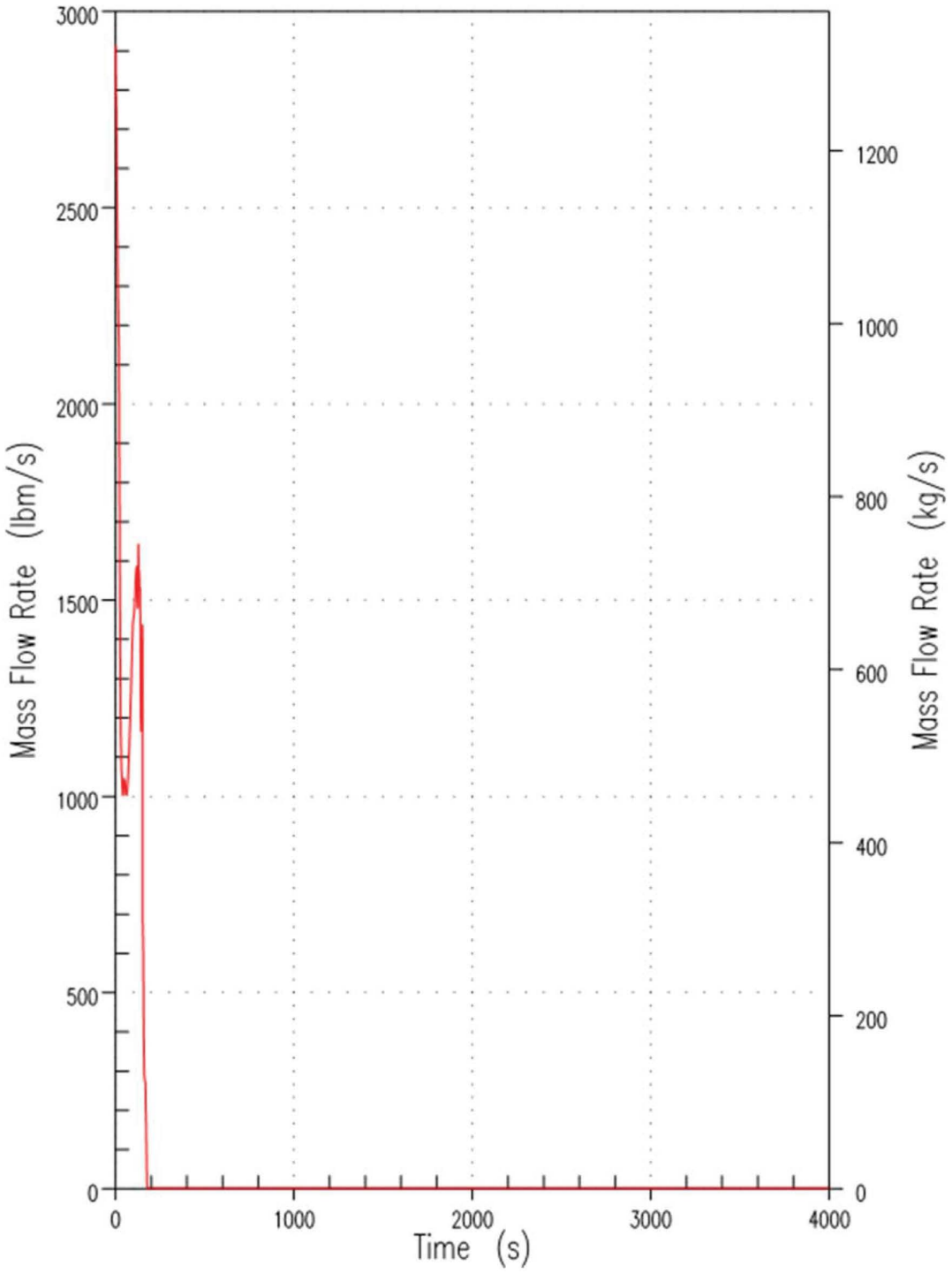

Figure 15.6.5.4B-36
20 psia DEDVI Line Break – Vessel Side Liquid Break Discharge

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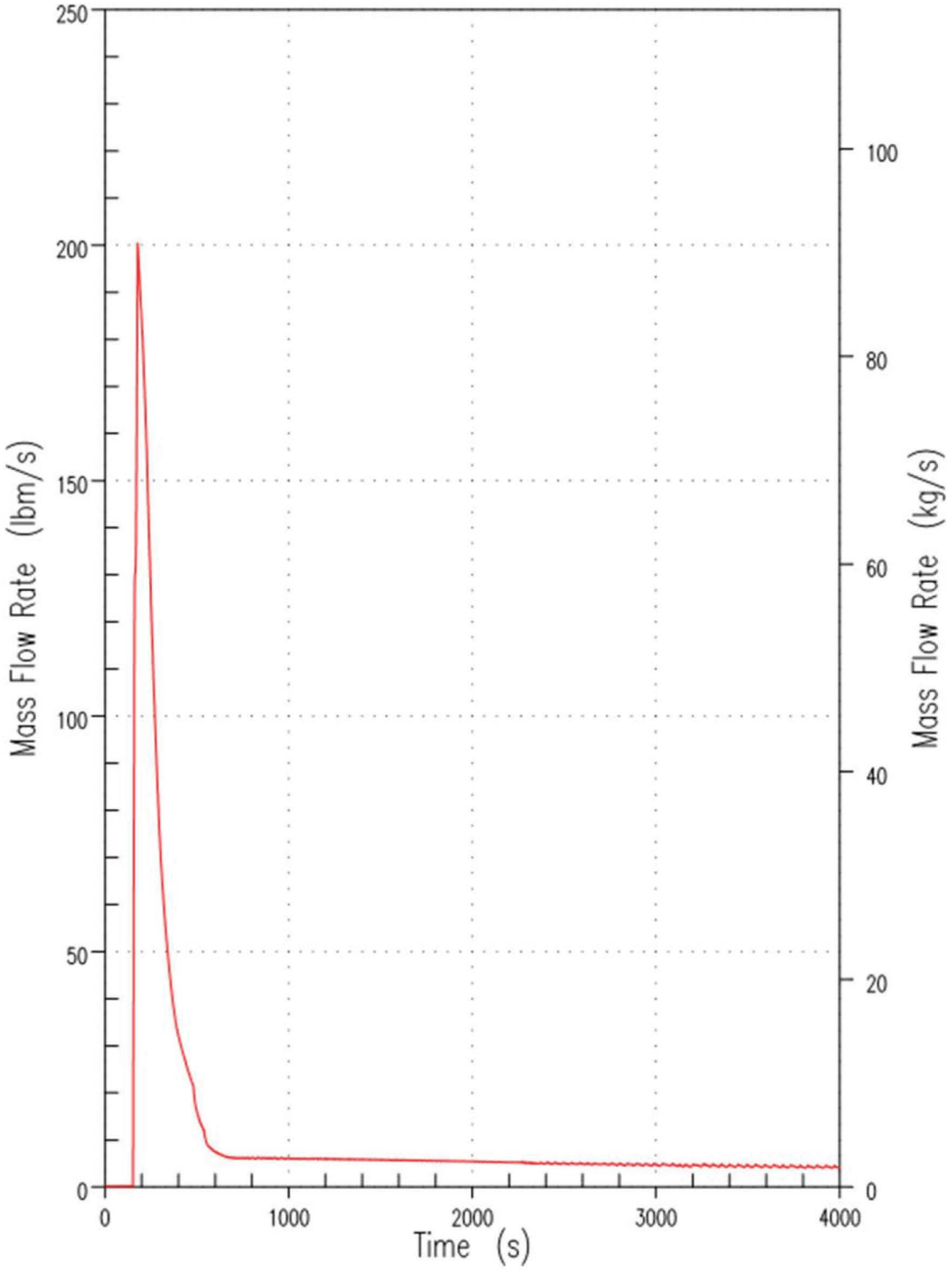

Figure 15.6.5.4B-37
20 psia DEDVI Line Break – Vessel Side Vapor Break Discharge

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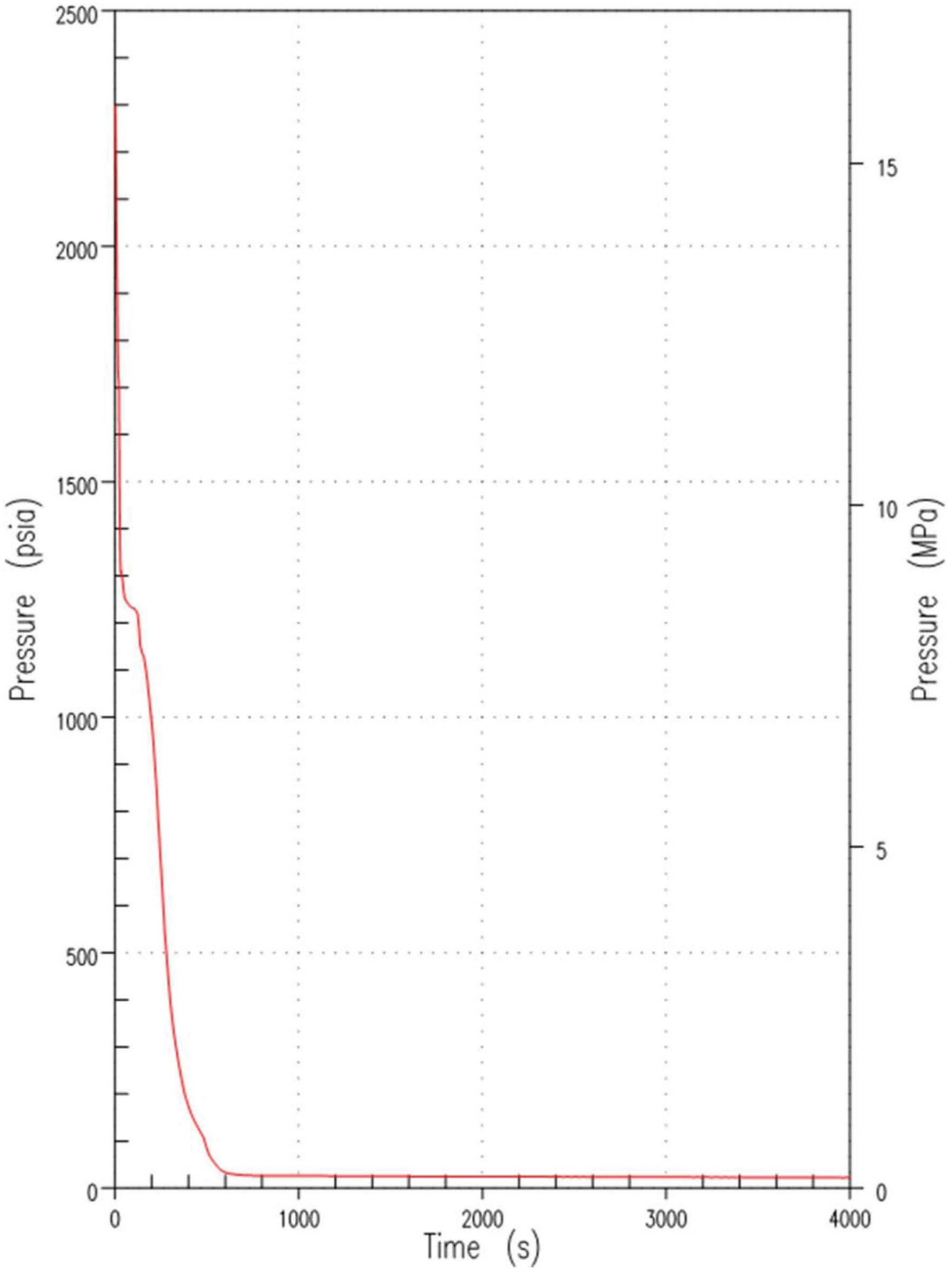

Figure 15.6.5.4B-38(a)
20 psia DEDVI Line Break – RCS Pressure

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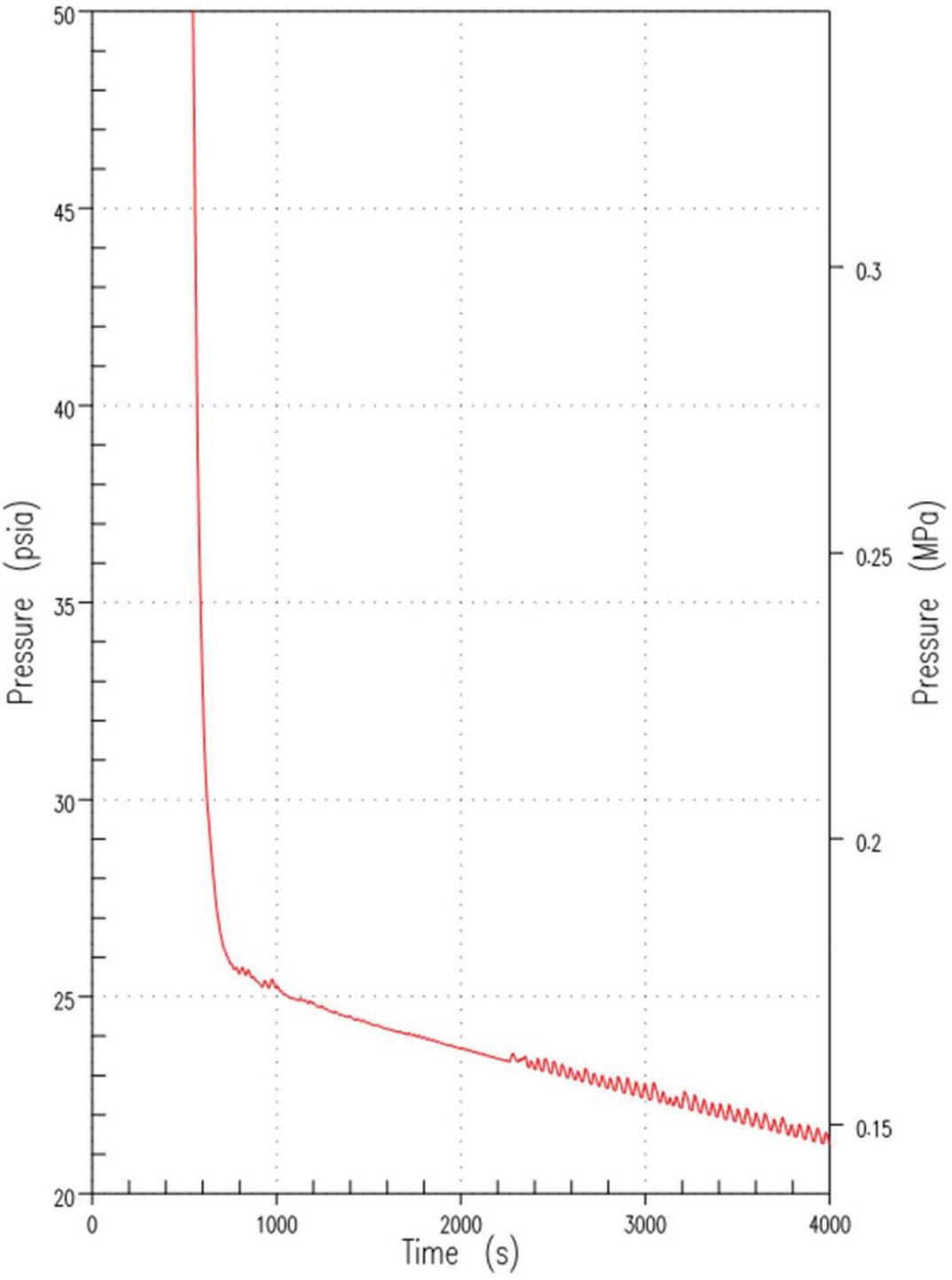

Figure 15.6.5.4B-38(b)
20 psia DEDVI Line Break – RCS Pressure (Zoomed)

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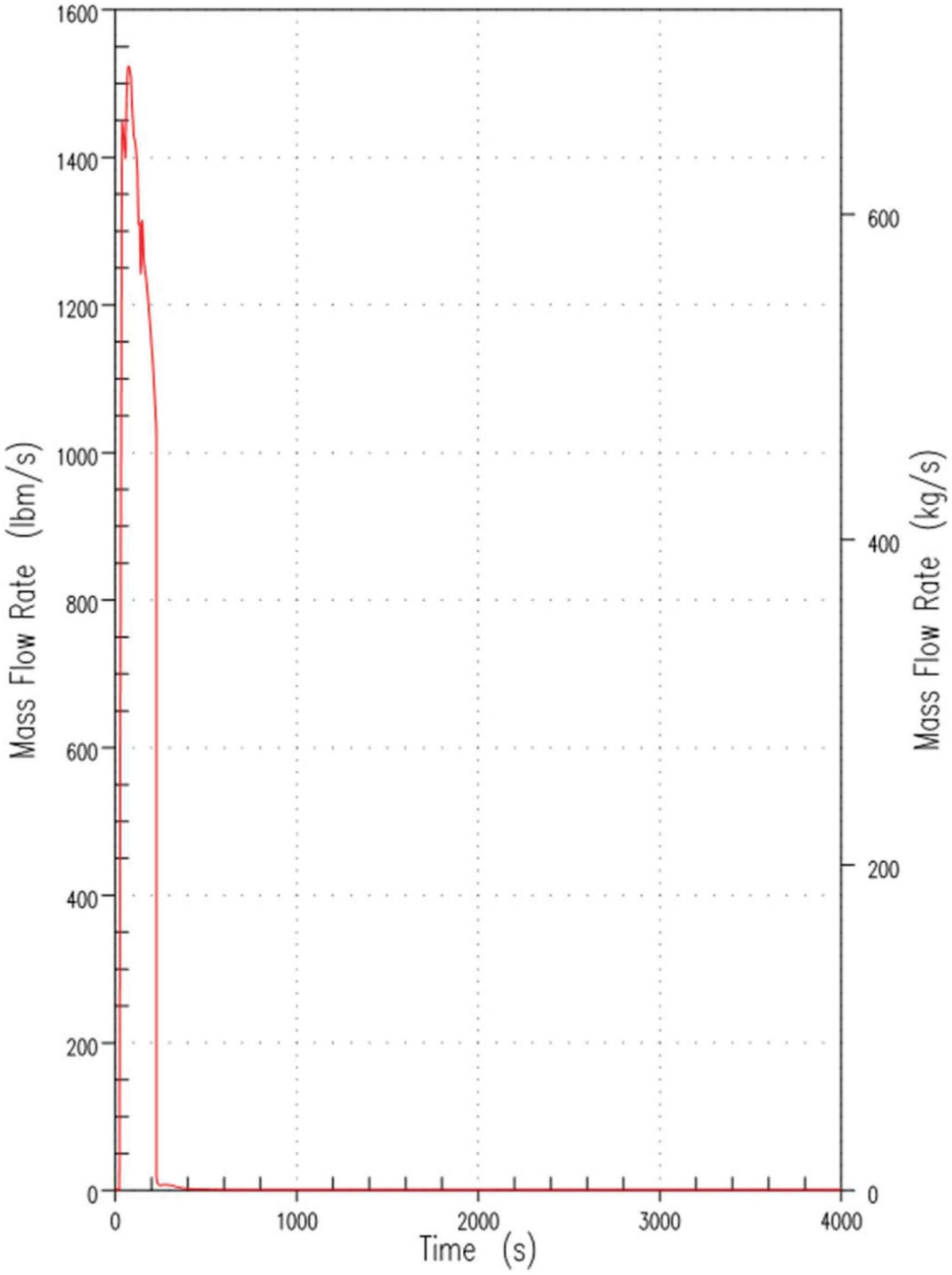

Figure 15.6.5.4B-39
20 psia DEDVI Line Break – Broken CMT Injection Rate

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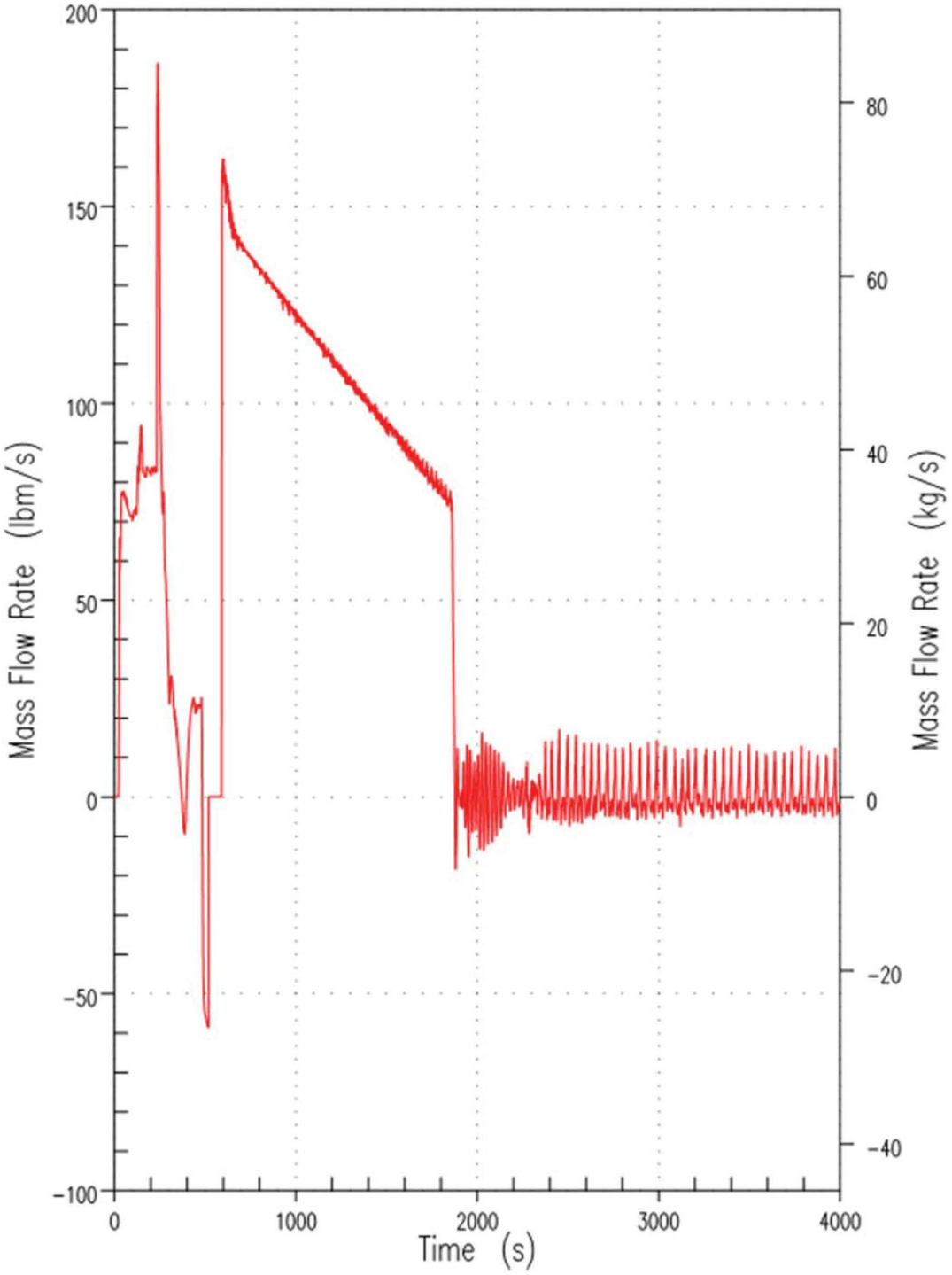

Figure 15.6.5.4B-40
20 psia DEDVI Line Break – Intact CMT Injection Rate

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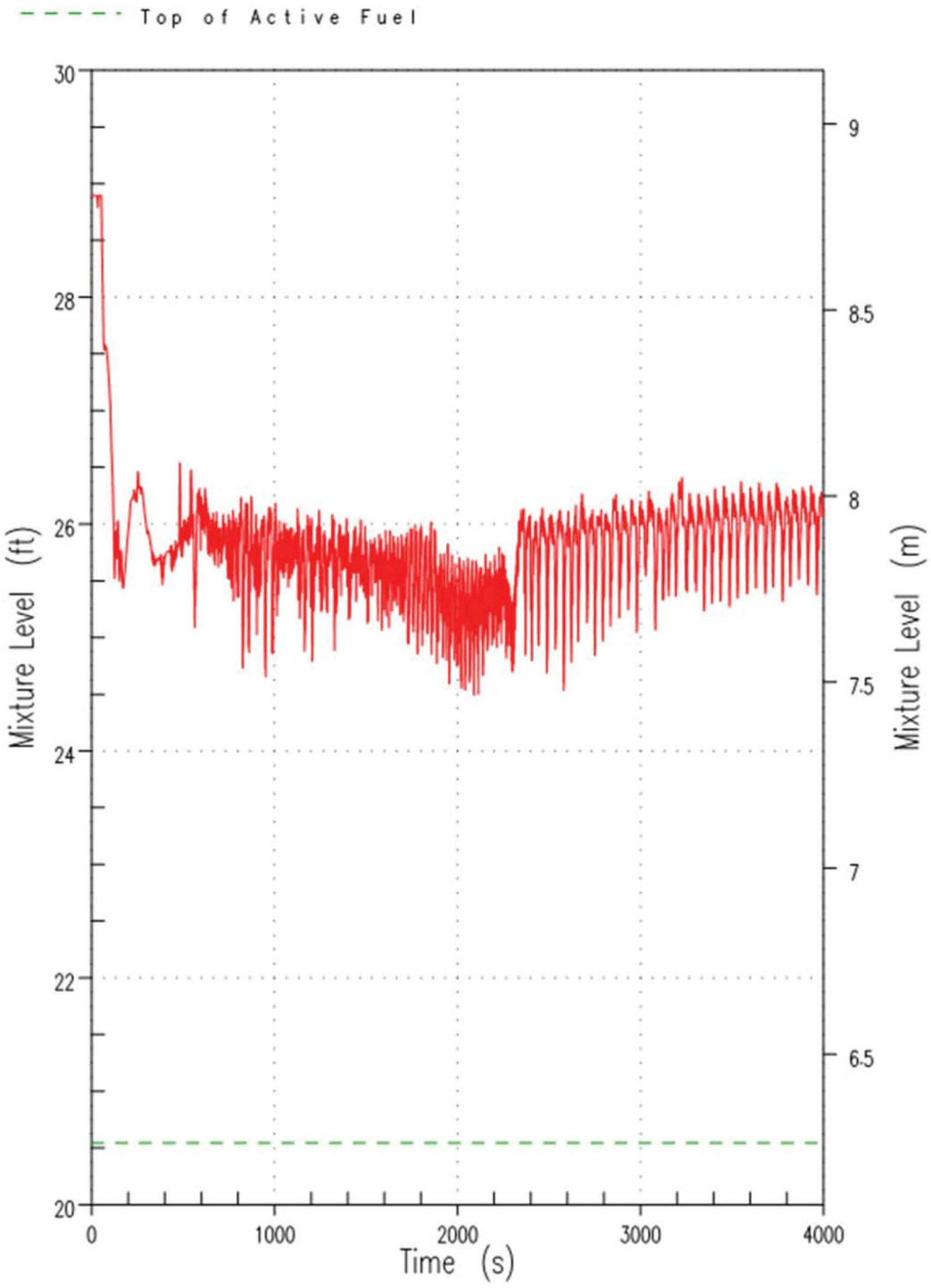

Figure 15.6.5.4B-41
20 psia DEDVI Line Break – Core/Upper Plenum Mixture Level

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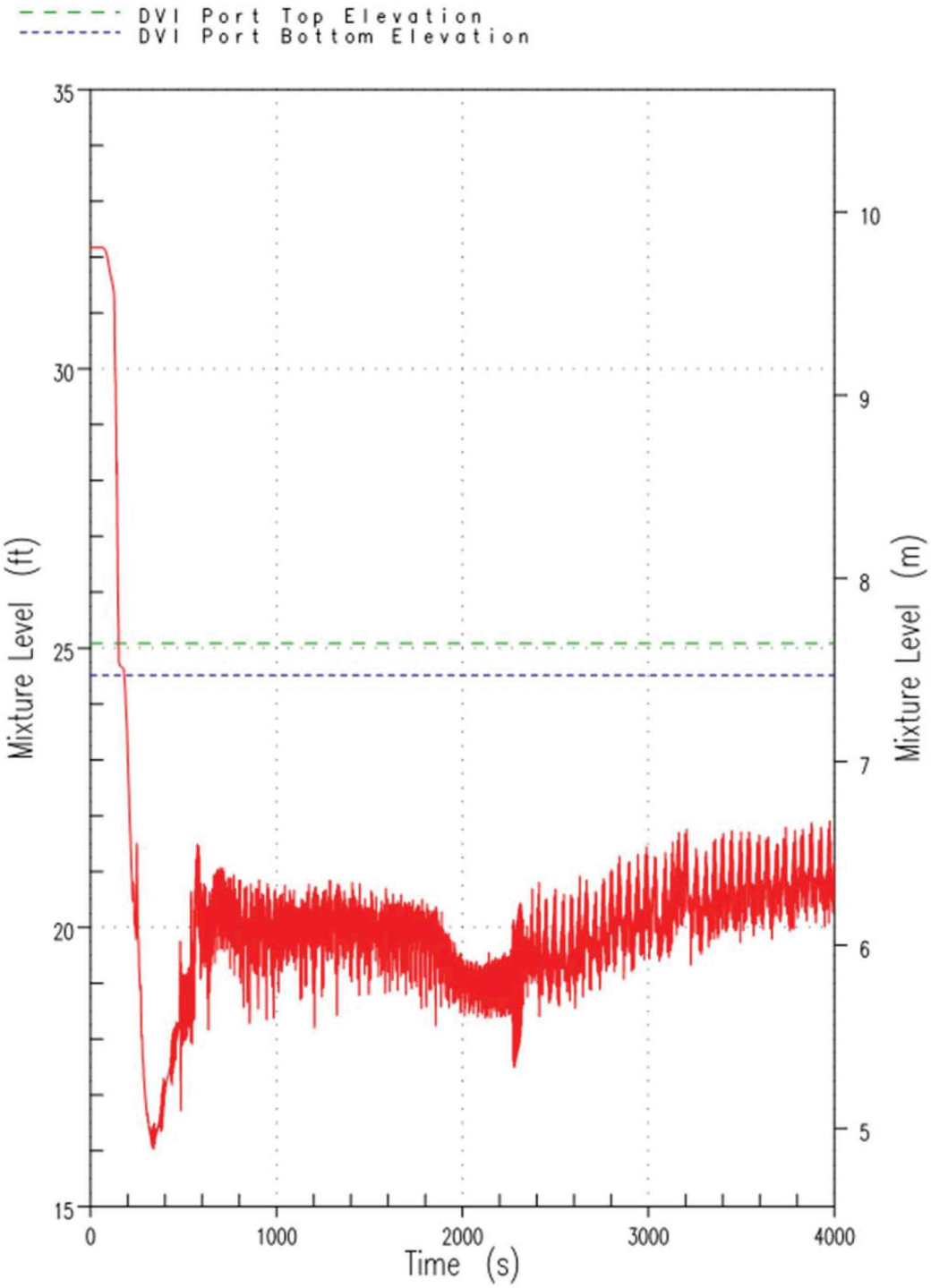

Figure 15.6.5.4B-42
20 psia DEDVI Line Break – Downcomer Mixture Level

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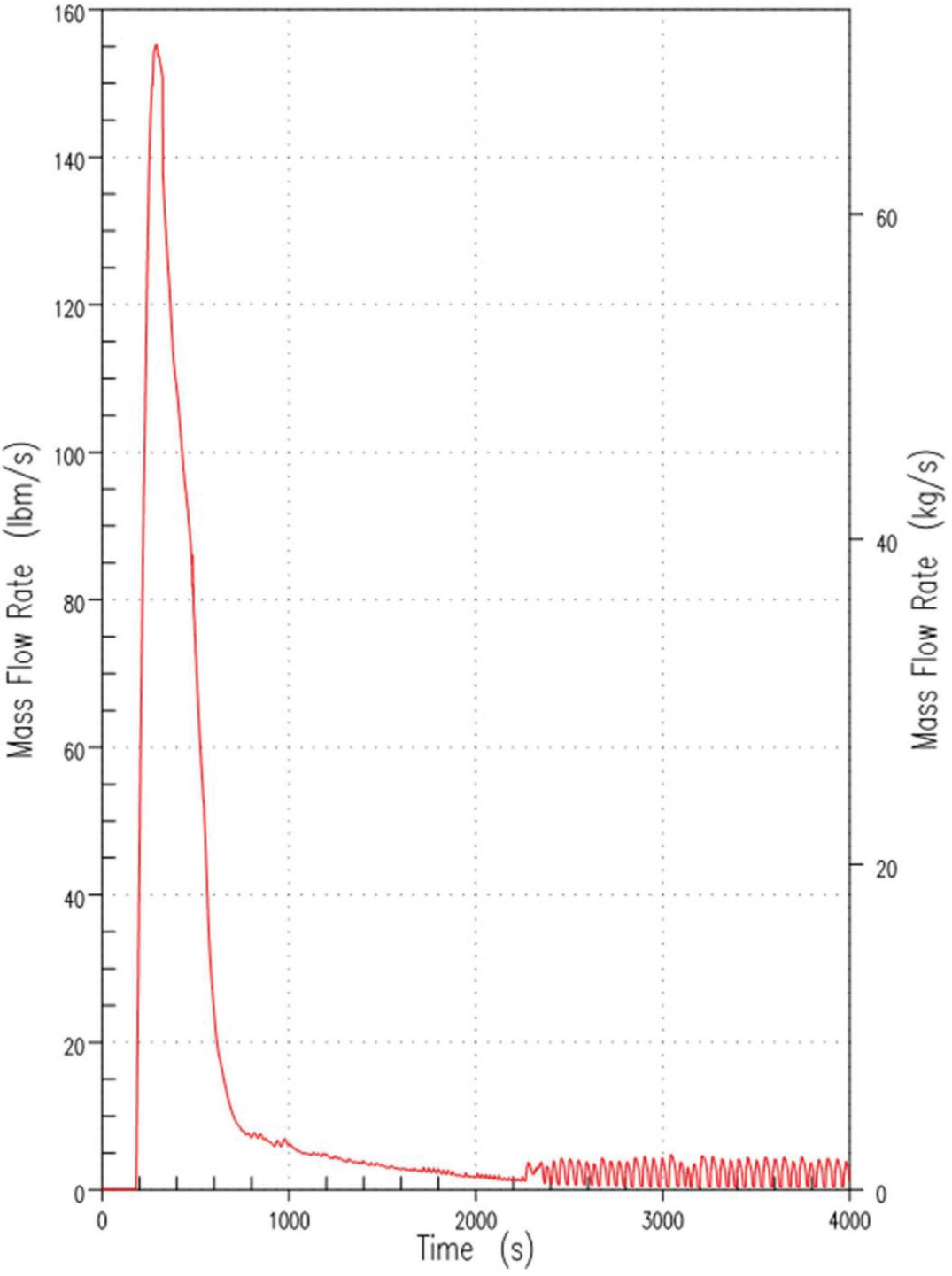

Figure 15.6.5.4B-43(a)
20 psia DEDVI Line Break – ADS 1-3 Vapor Discharge

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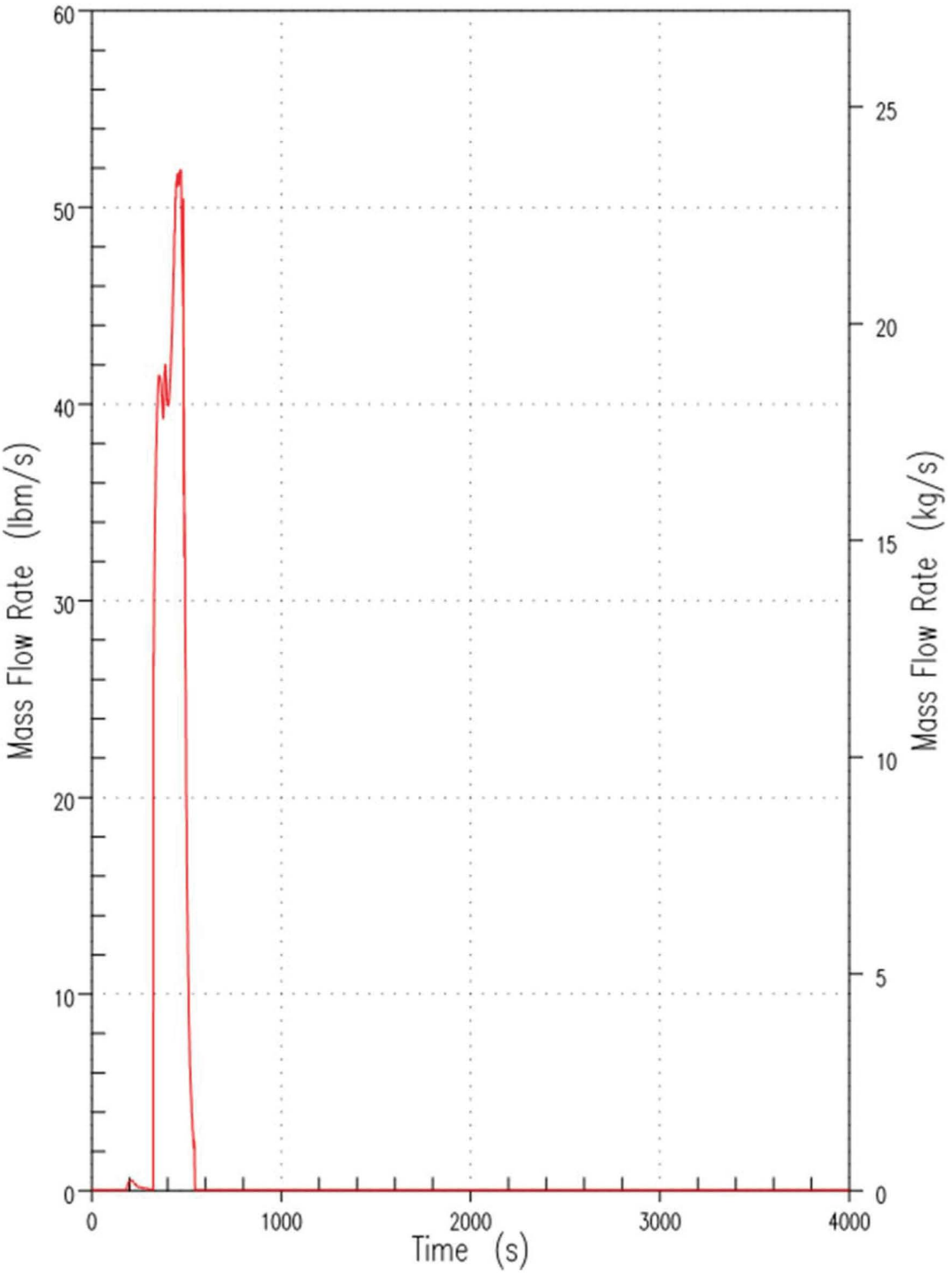

Figure 15.6.5.4B-43(b)
20 psia DEDVI Line Break – ADS 1-3 Liquid Discharge

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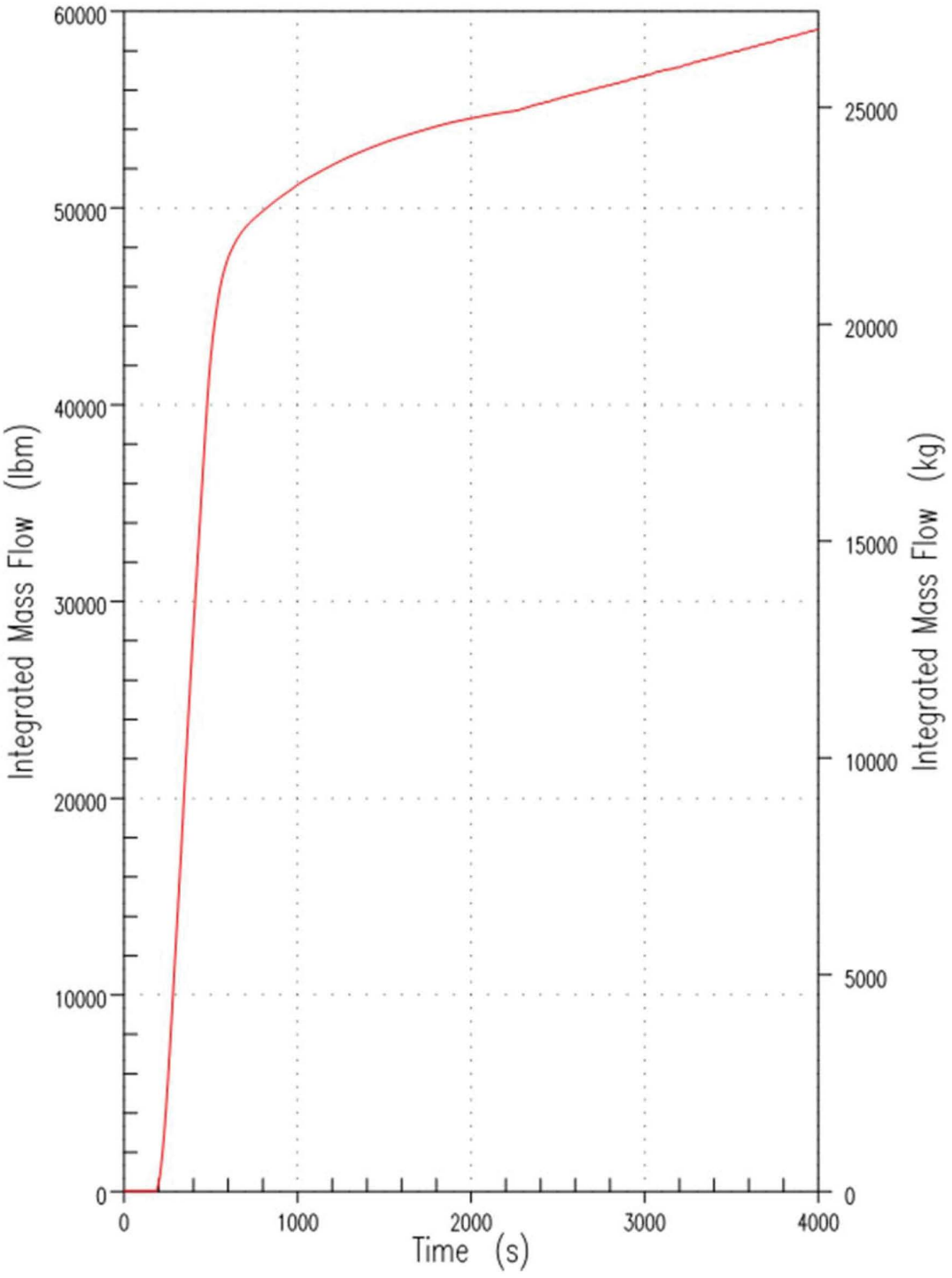

Figure 15.6.5.4B-43(c)
20 psia DEDVI Line Break – ADS 1-3 Integrated Discharge

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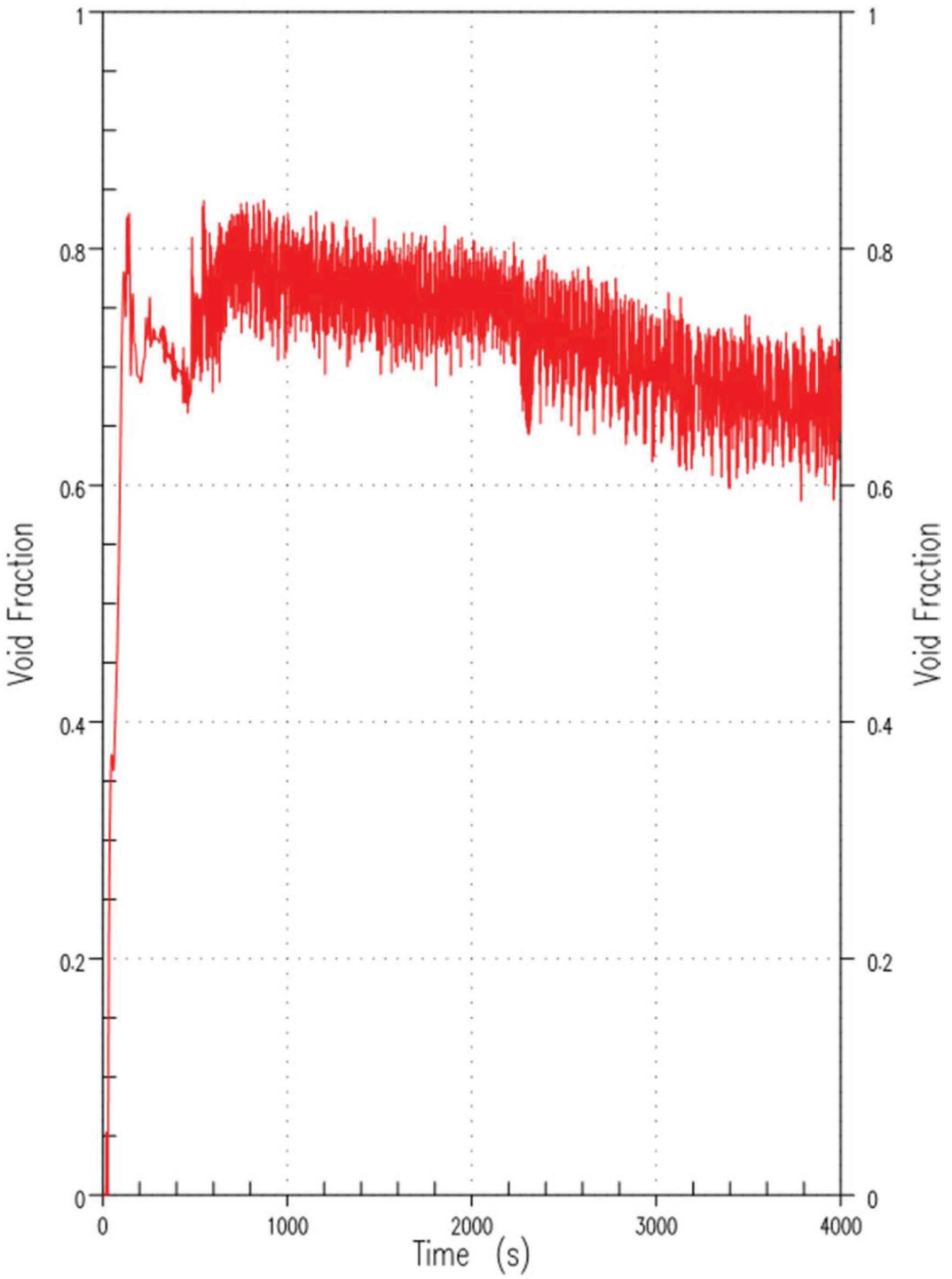

Figure 15.6.5.4B-44
20 psia DEDVI Line Break – Core Exit Void Fraction

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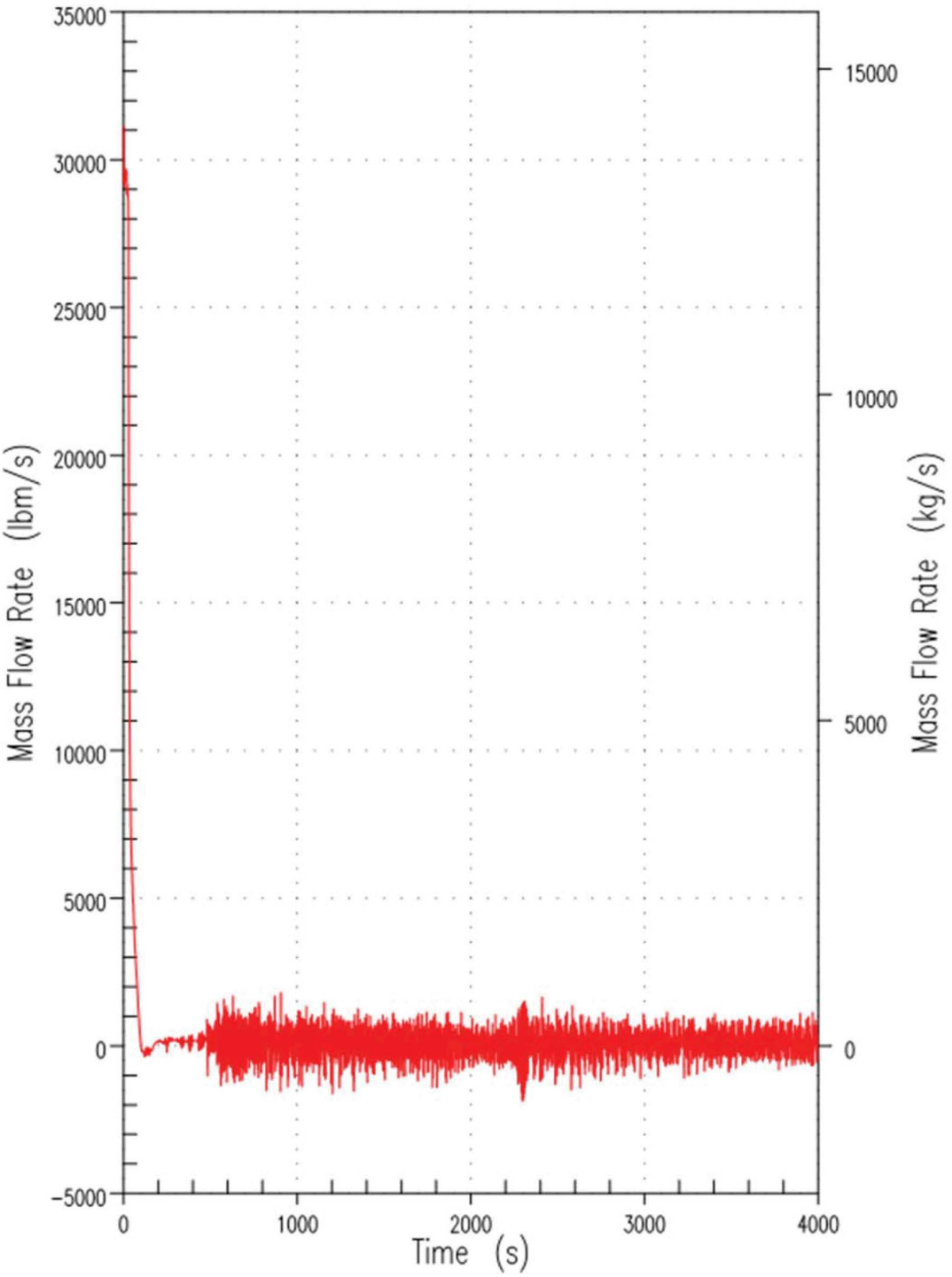

Figure 15.6.5.4B-45
20 psia DEDVI Line Break – Core Exit Liquid Flow Rate

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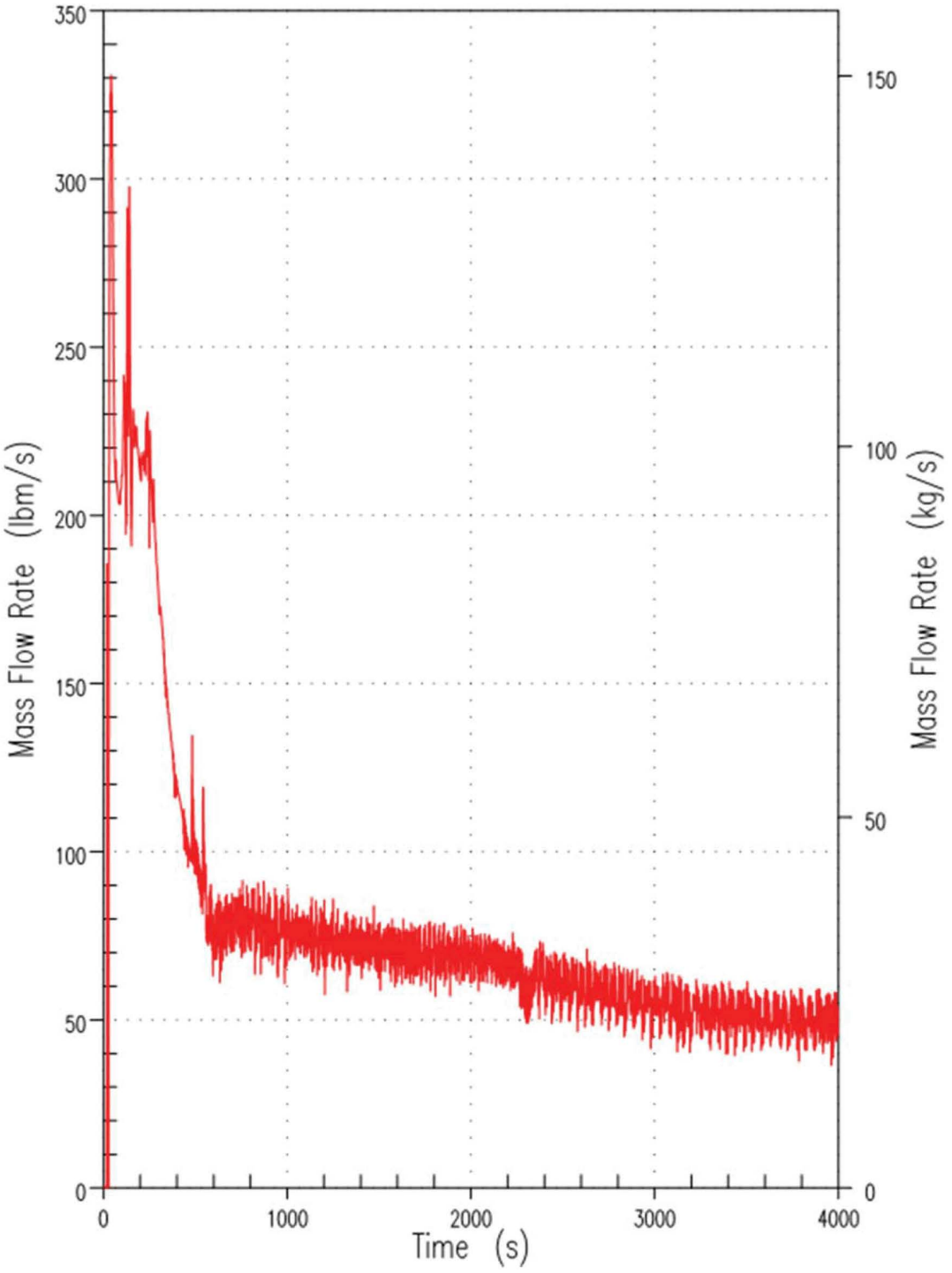

Figure 15.6.5.4B-46
20 psia DEDVI Line Break – Core Exit Vapor Flow Rate

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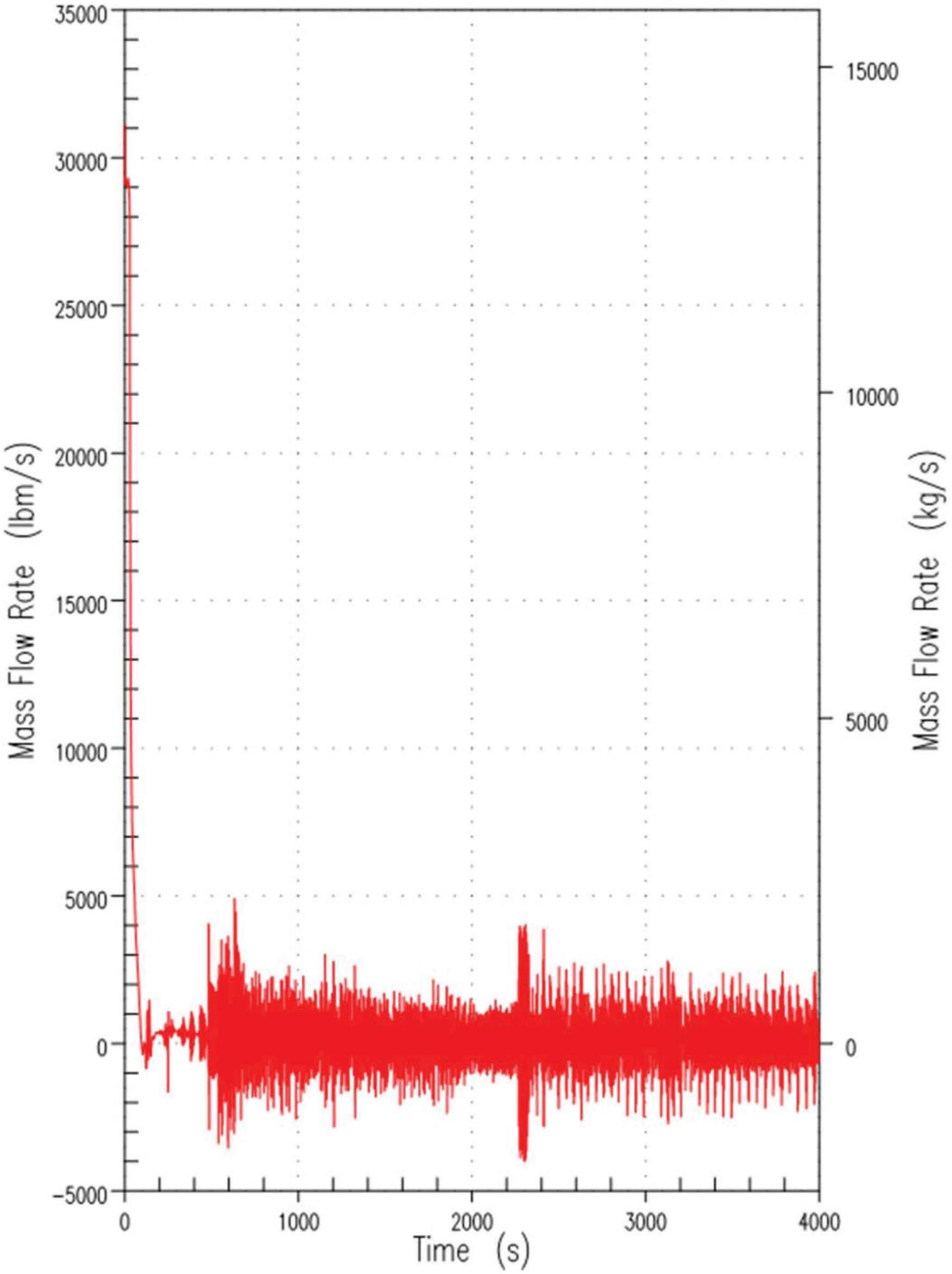

Figure 15.6.5.4B-47
20 psia DEDVI Line Break – Lower Plenum to Core Flow Rate

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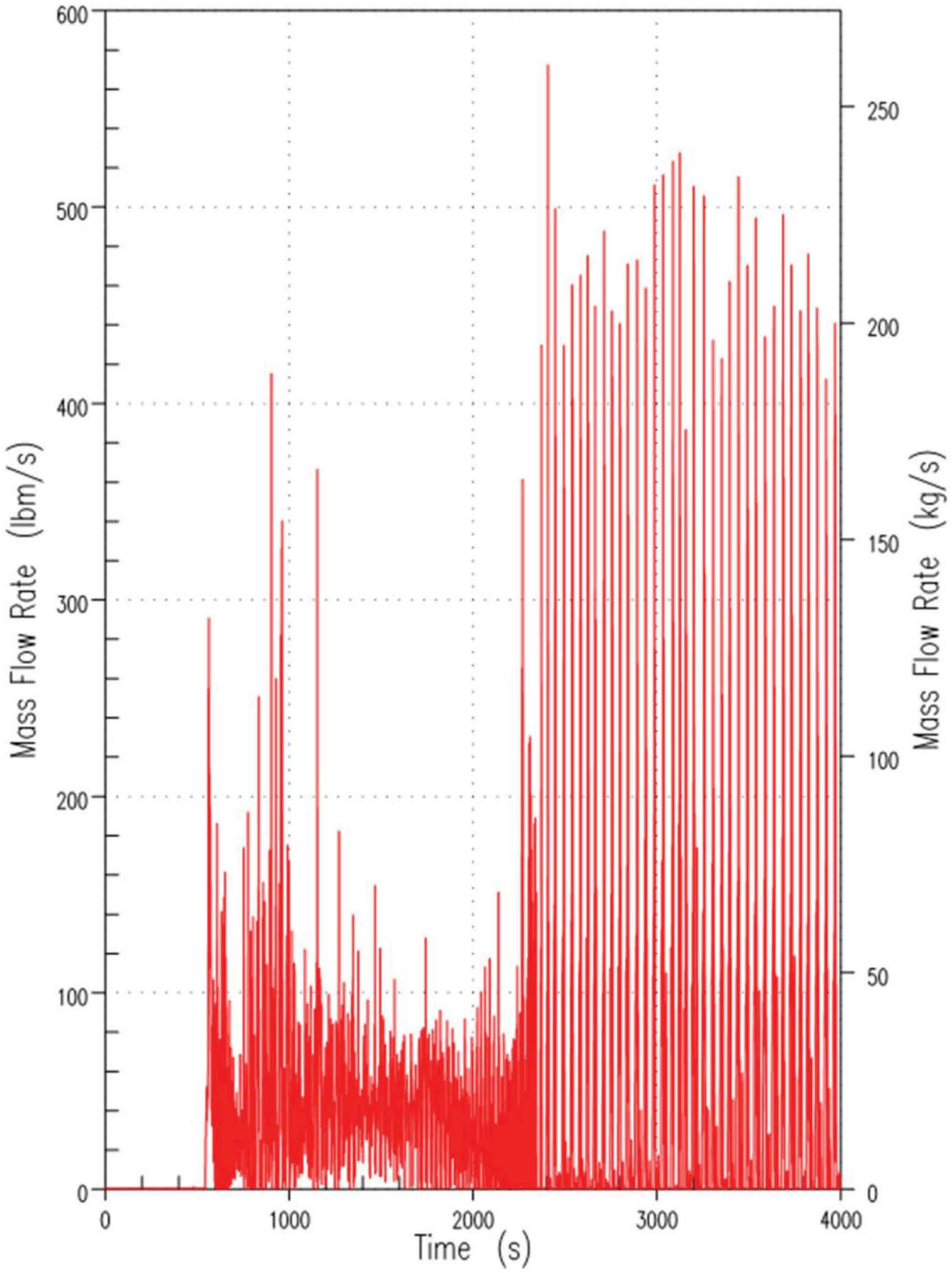

Figure 15.6.5.4B-48(a)
20 psia DEDVI Line Break – ADS-4 Liquid Discharge

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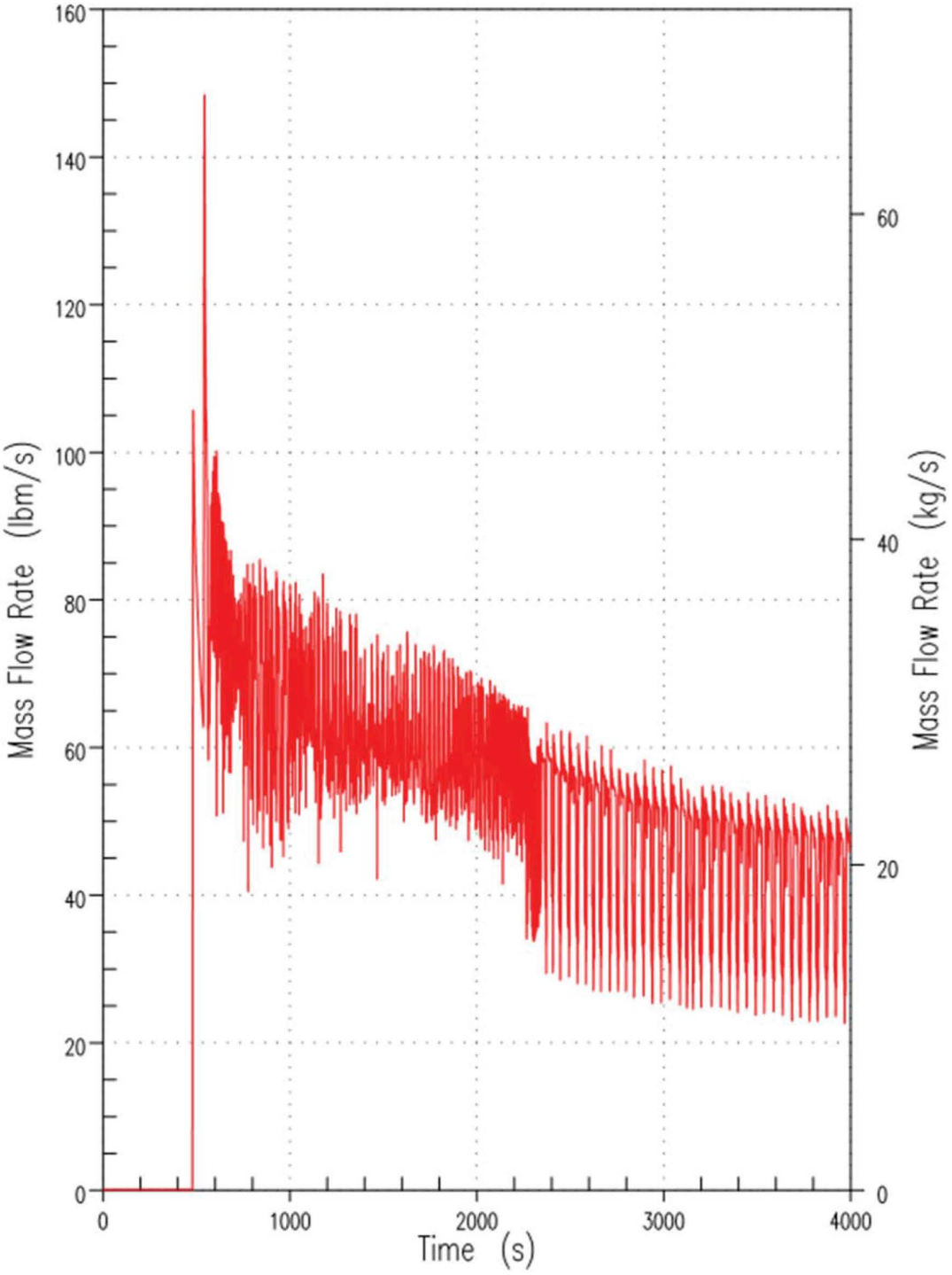

Figure 15.6.5.4B-48(b)
20 psia DEDVI Line Break – ADS-4 Vapor Discharge

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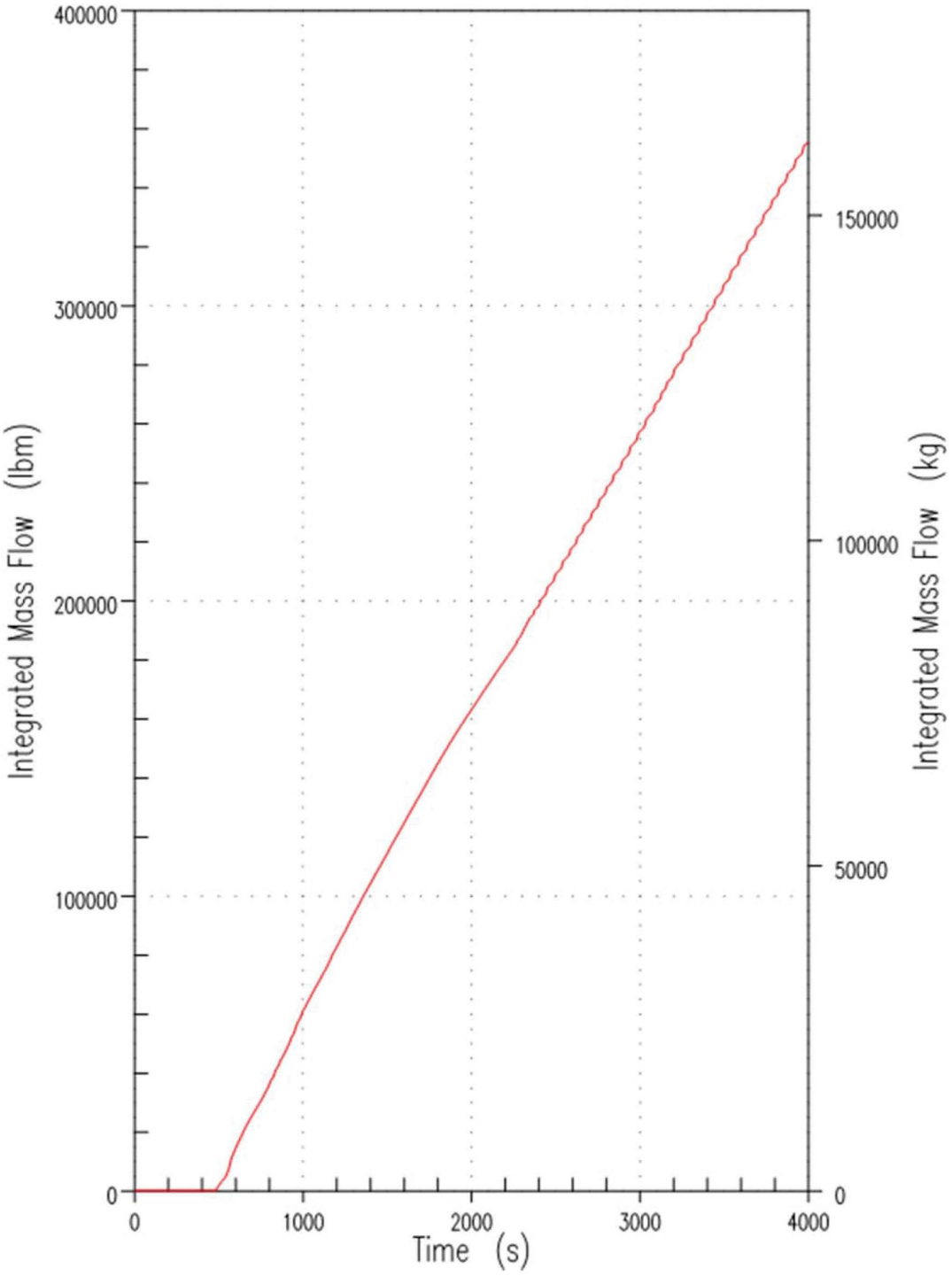

Figure 15.6.5.4B-49
20 psia DEDVI Line Break – ADS-4 Integrated Discharge

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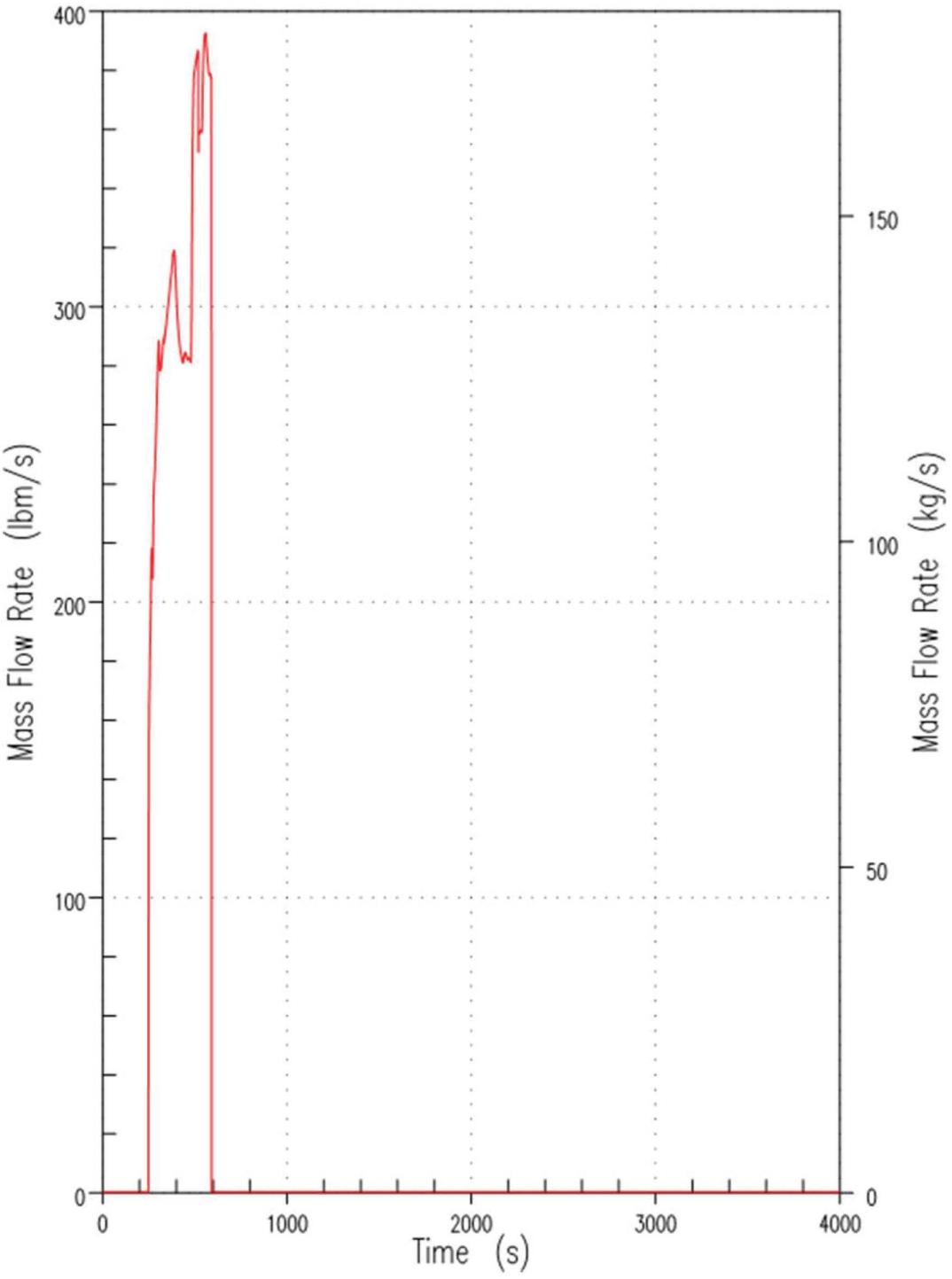

Figure 15.6.5.4B-50
20 psia DEDVI Line Break – Intact Accumulator Flow Rate

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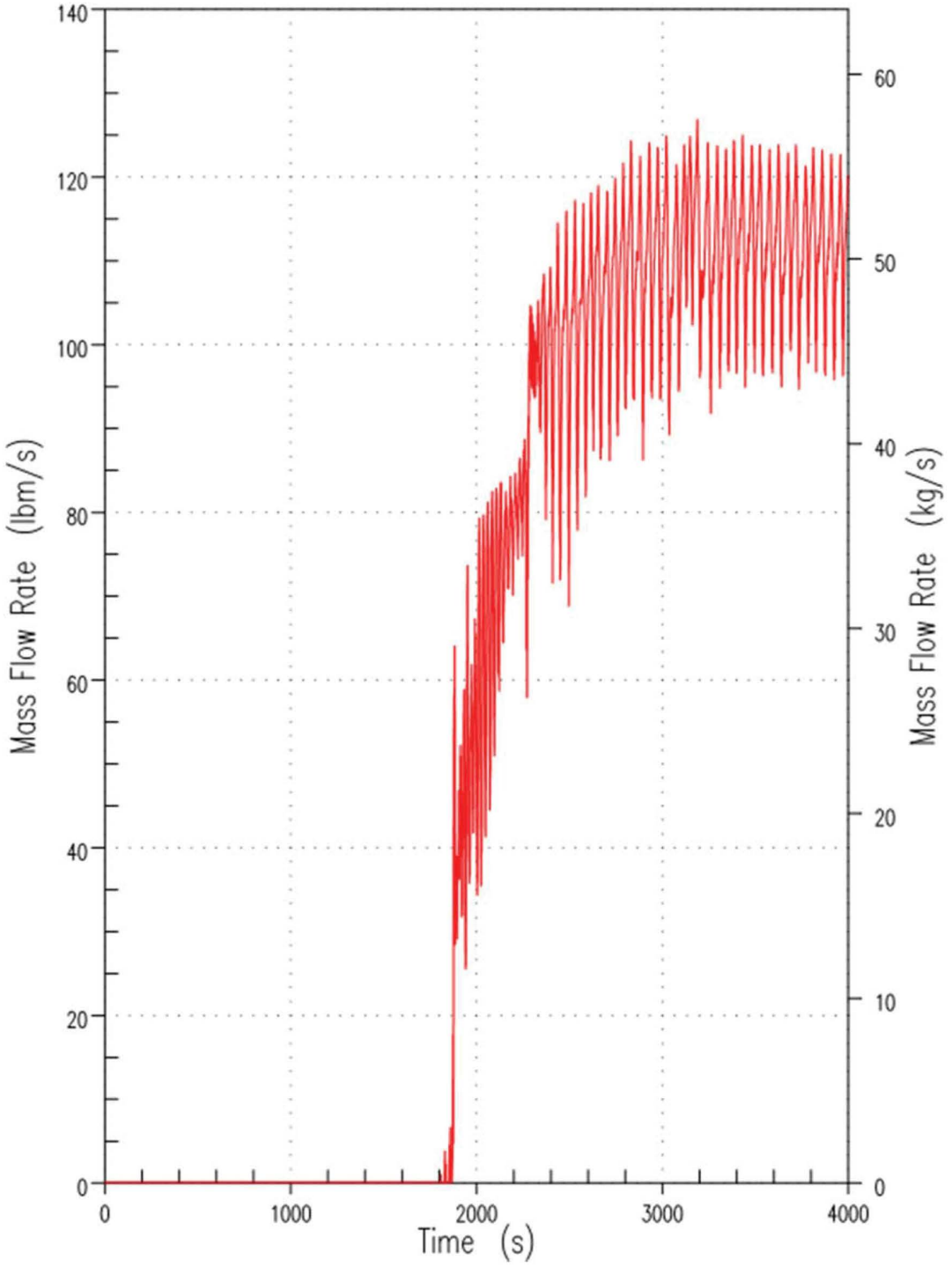

Figure 15.6.5.4B-51
20 psia DEDVI Line Break – Intact IRWST Injection Rate

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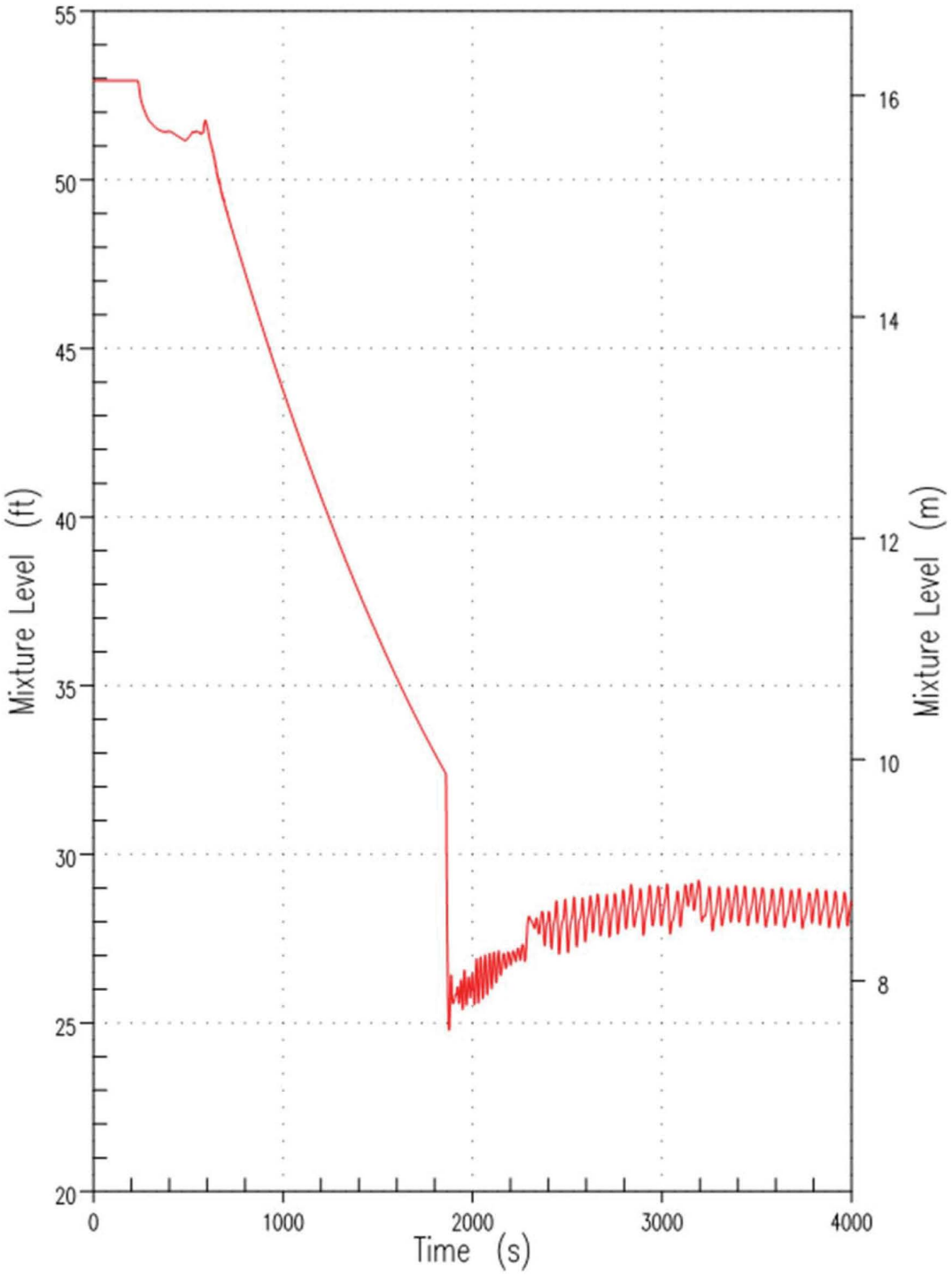

Figure 15.6.5.4B-52
20 psia DEDVI Line Break – Intact CMT Mixture Level

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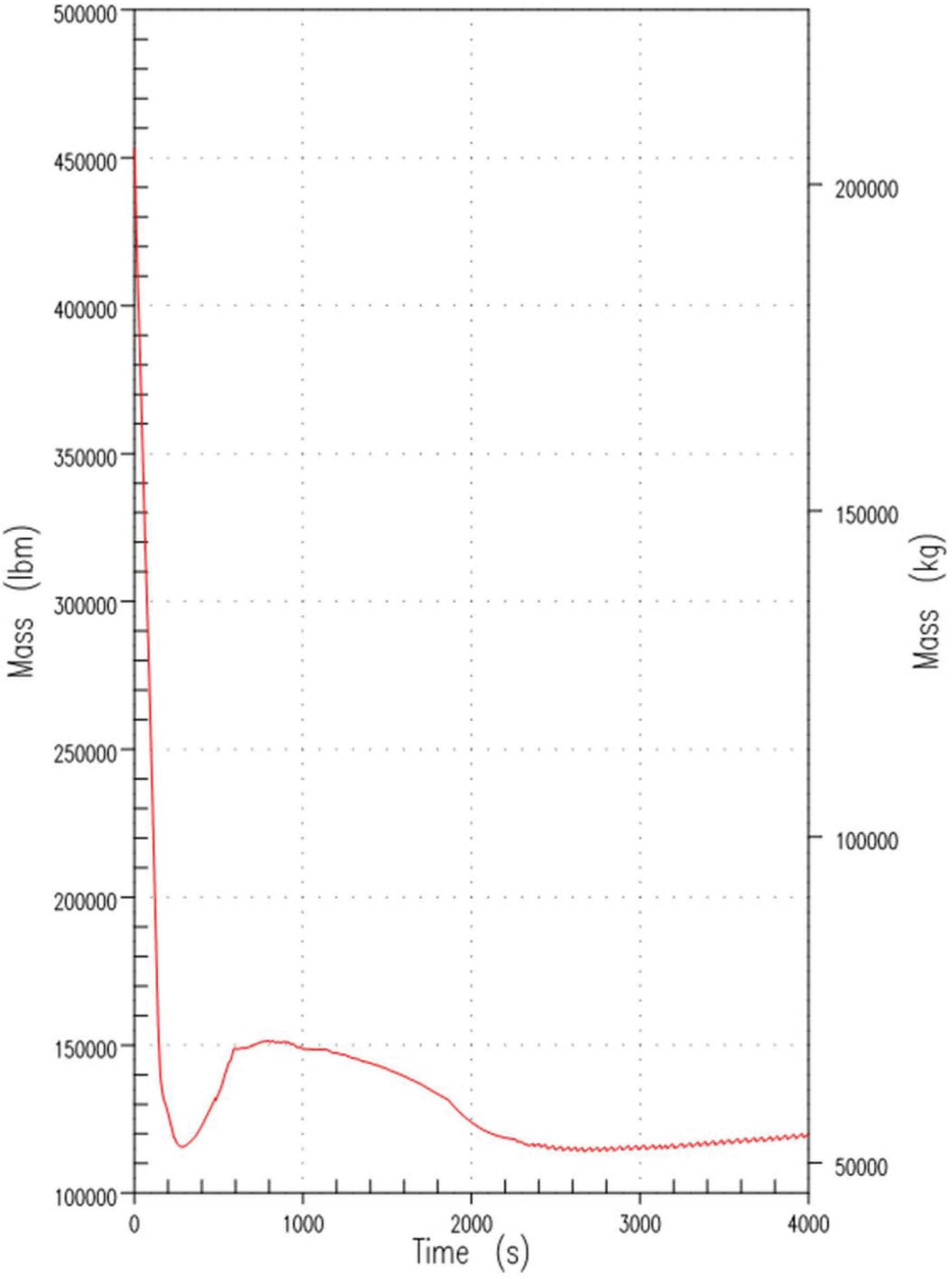

Figure 15.6.5.4B-53(a)
20 psia DEDVI Line Break – RCS System Inventory

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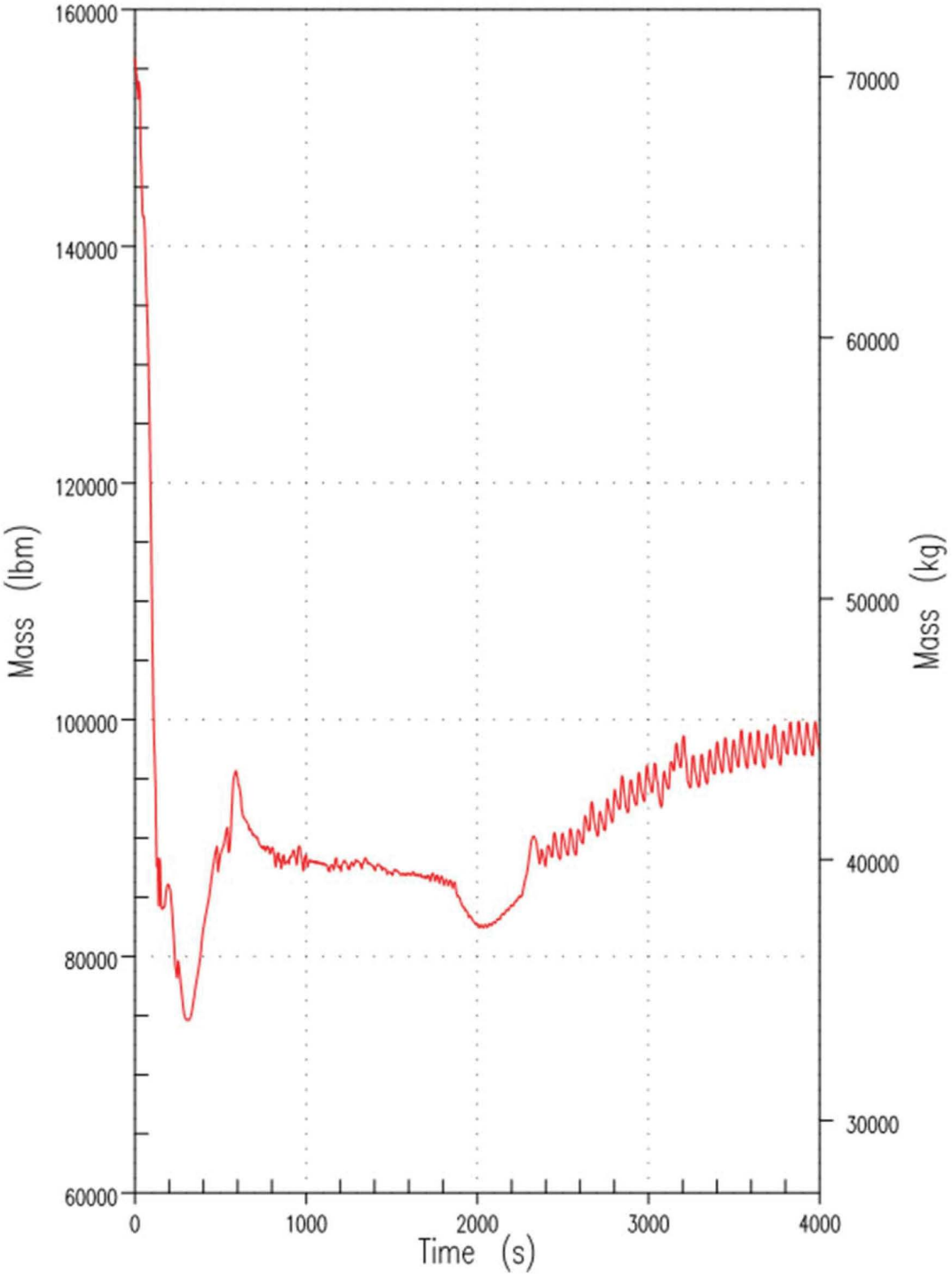

Figure 15.6.5.4B-53(b)
20 psia DEDVI Line Break – Reactor Vessel Mixture Inventory

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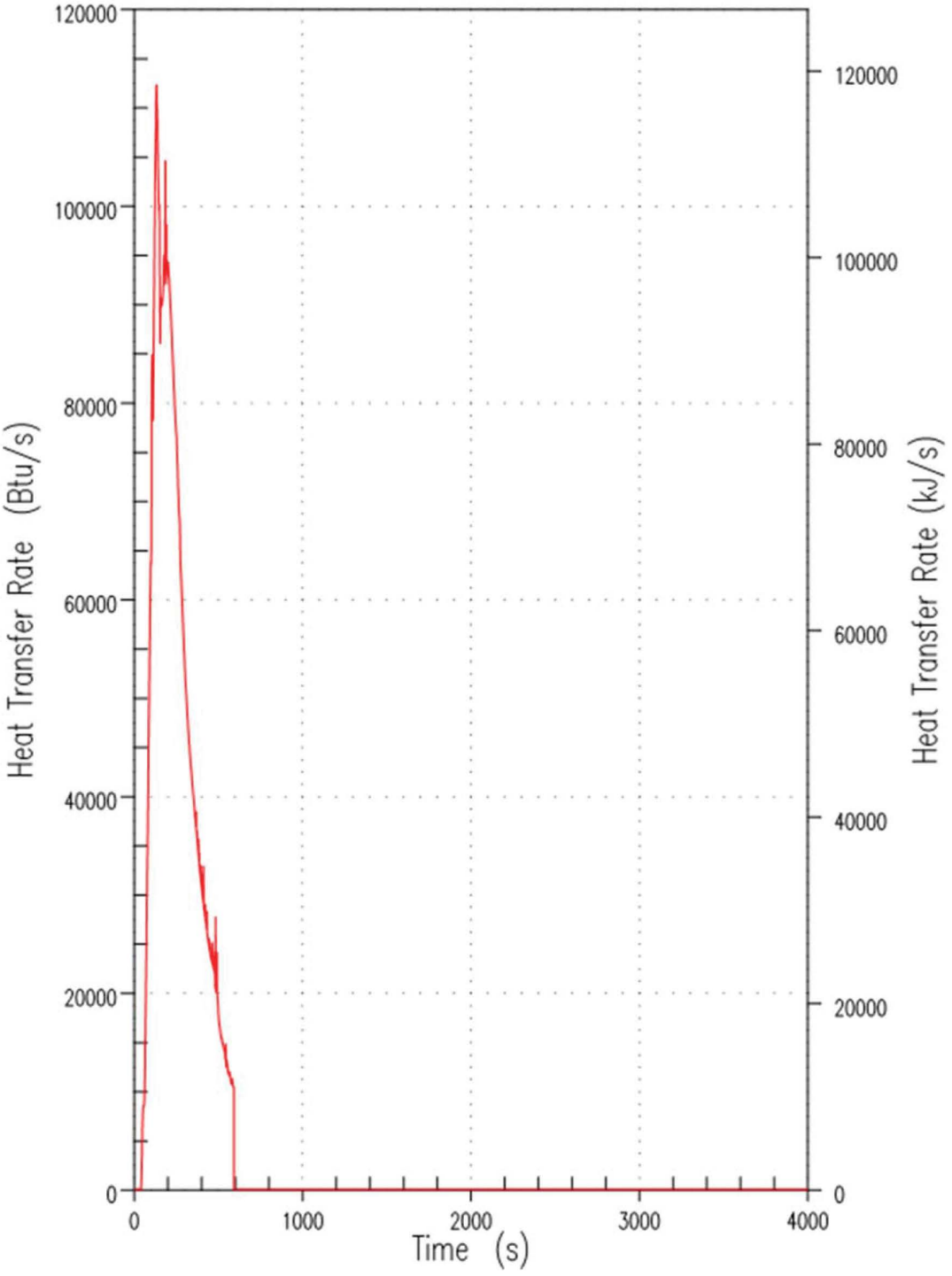

Figure 15.6.5.4B-54
20 psia DEDVI Line Break – PRHR Heat Removal Rate

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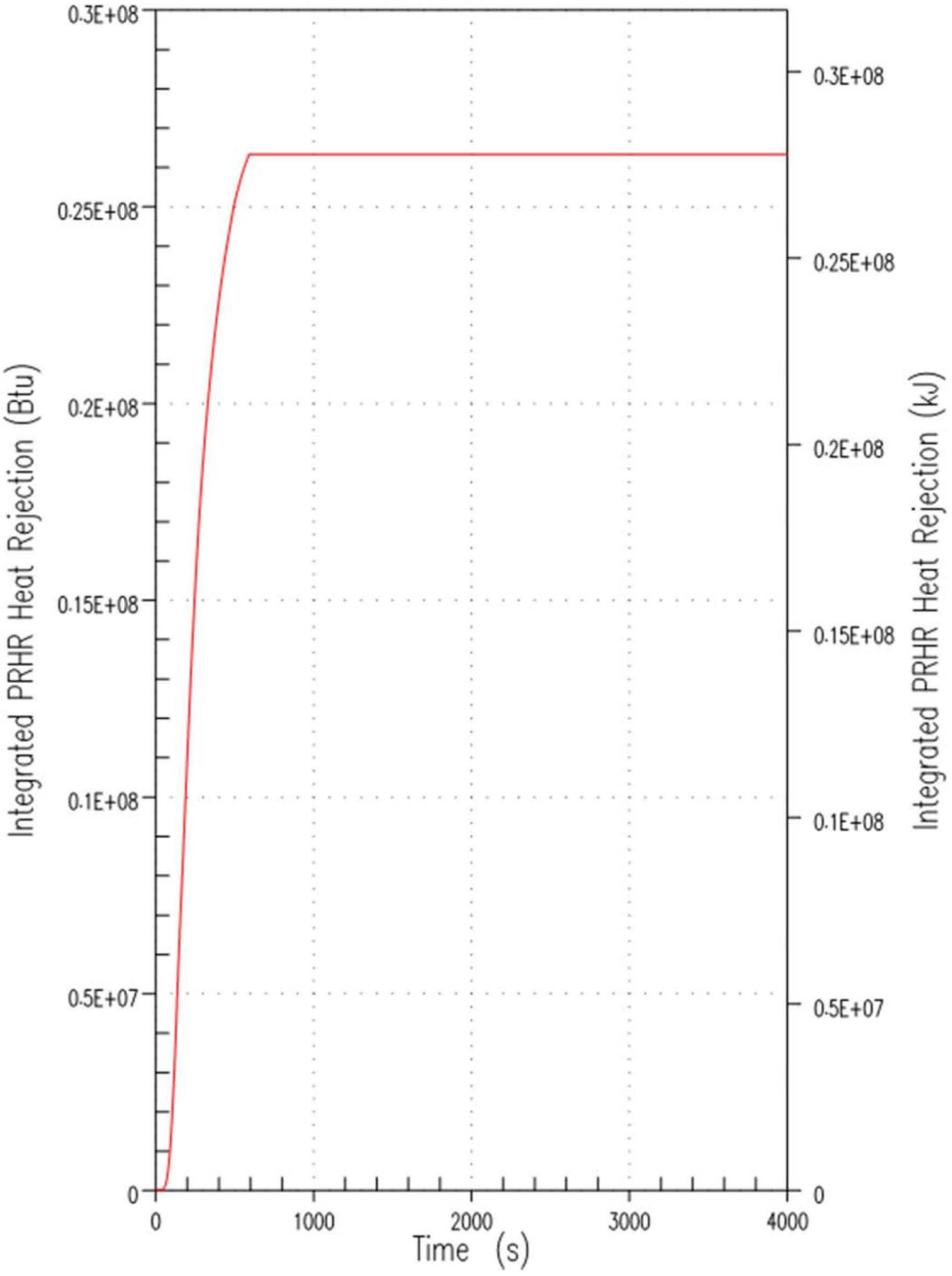

Figure 15.6.5.4B-55
20 psia DEDVI Line Break – Integrated PRHR Heat Removal

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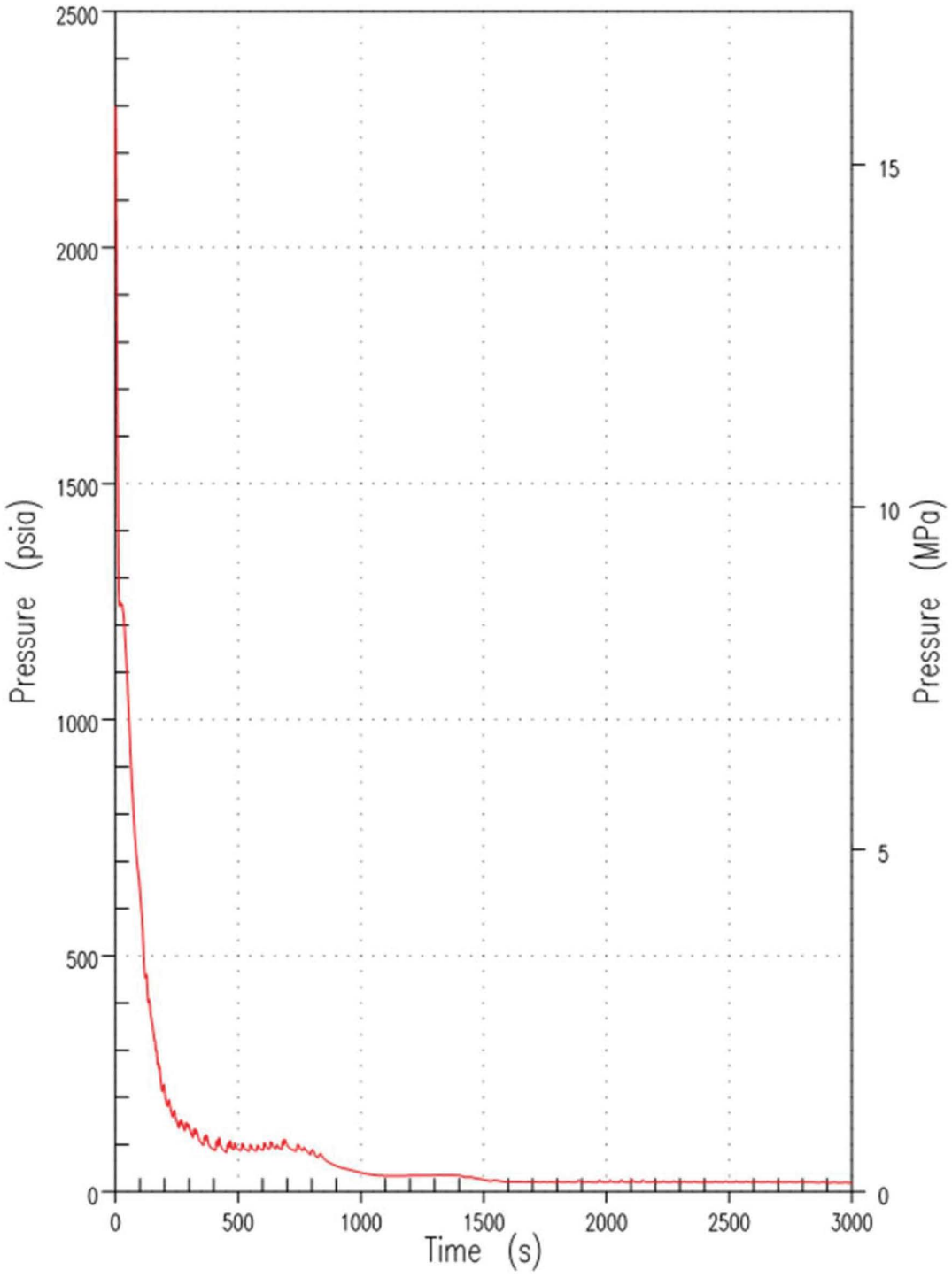

Figure 15.6.5.4B-56(a)
10-inch Cold Leg Break – RCS Pressure

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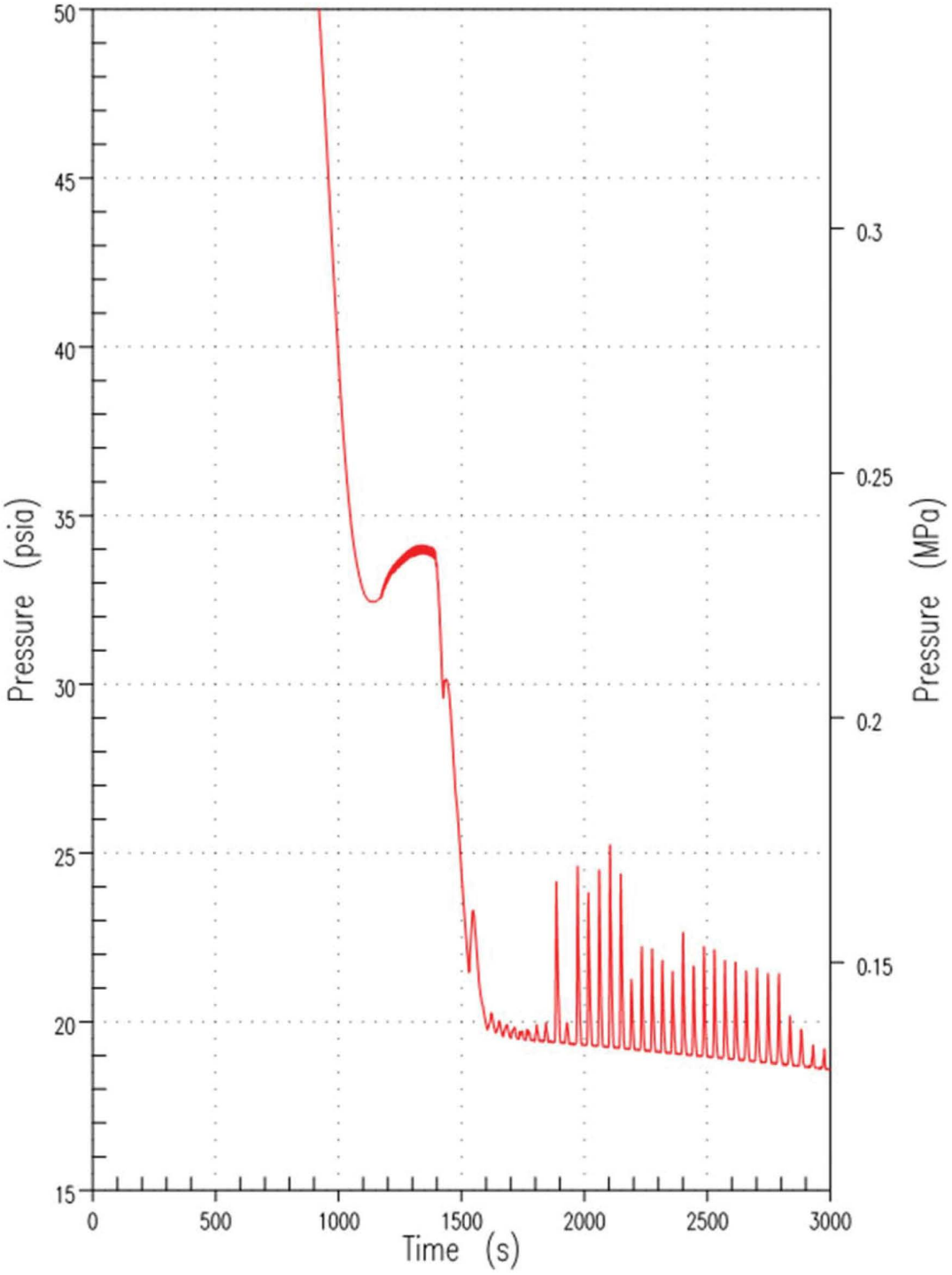

Figure 15.6.5.4B-56(b)
10-Inch Cold Leg Break – RCS Pressure (Zoomed)

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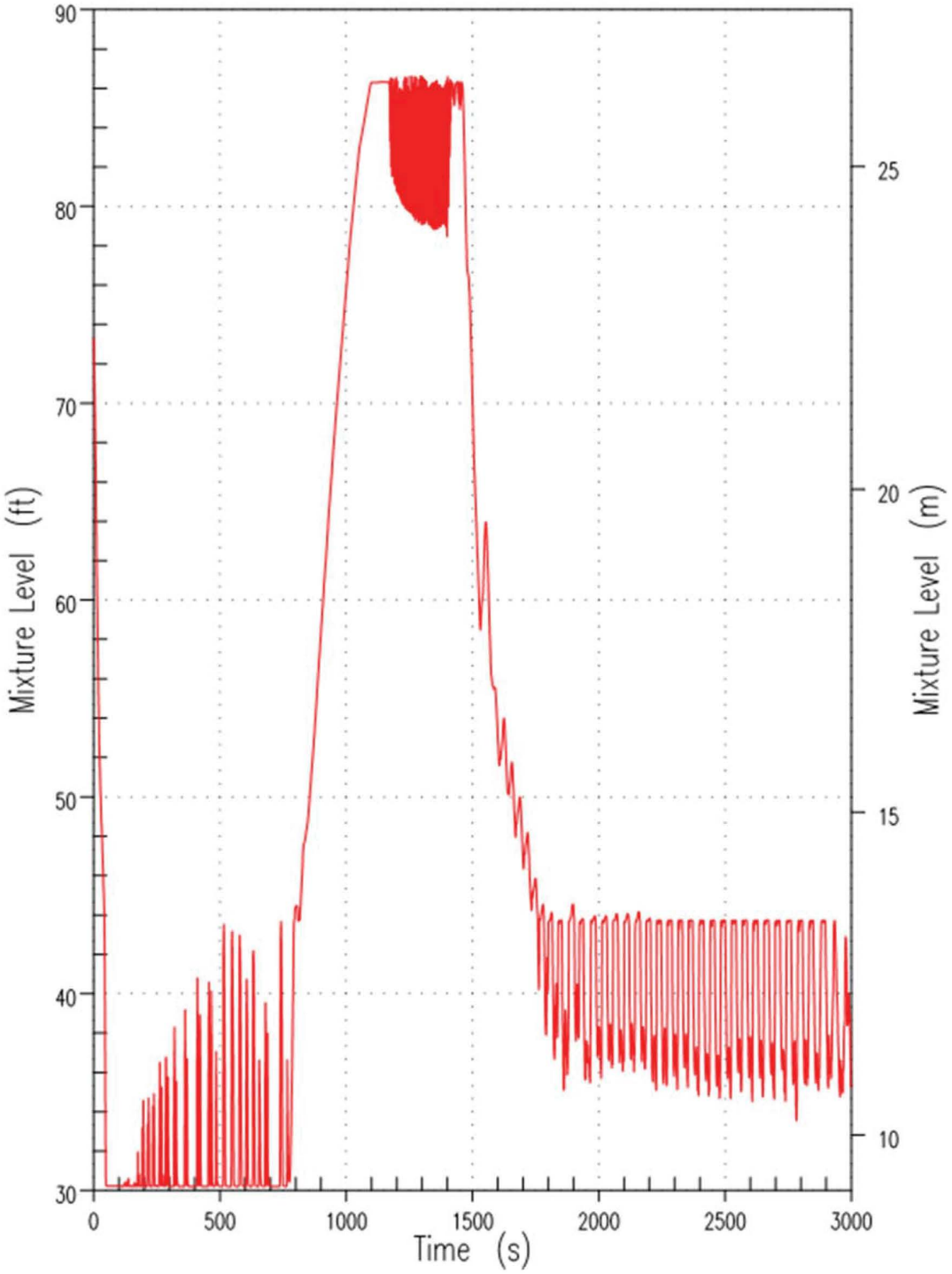

Figure 15.6.5.4B-57
10-inch Cold Leg Break – Pressurizer Mixture Level

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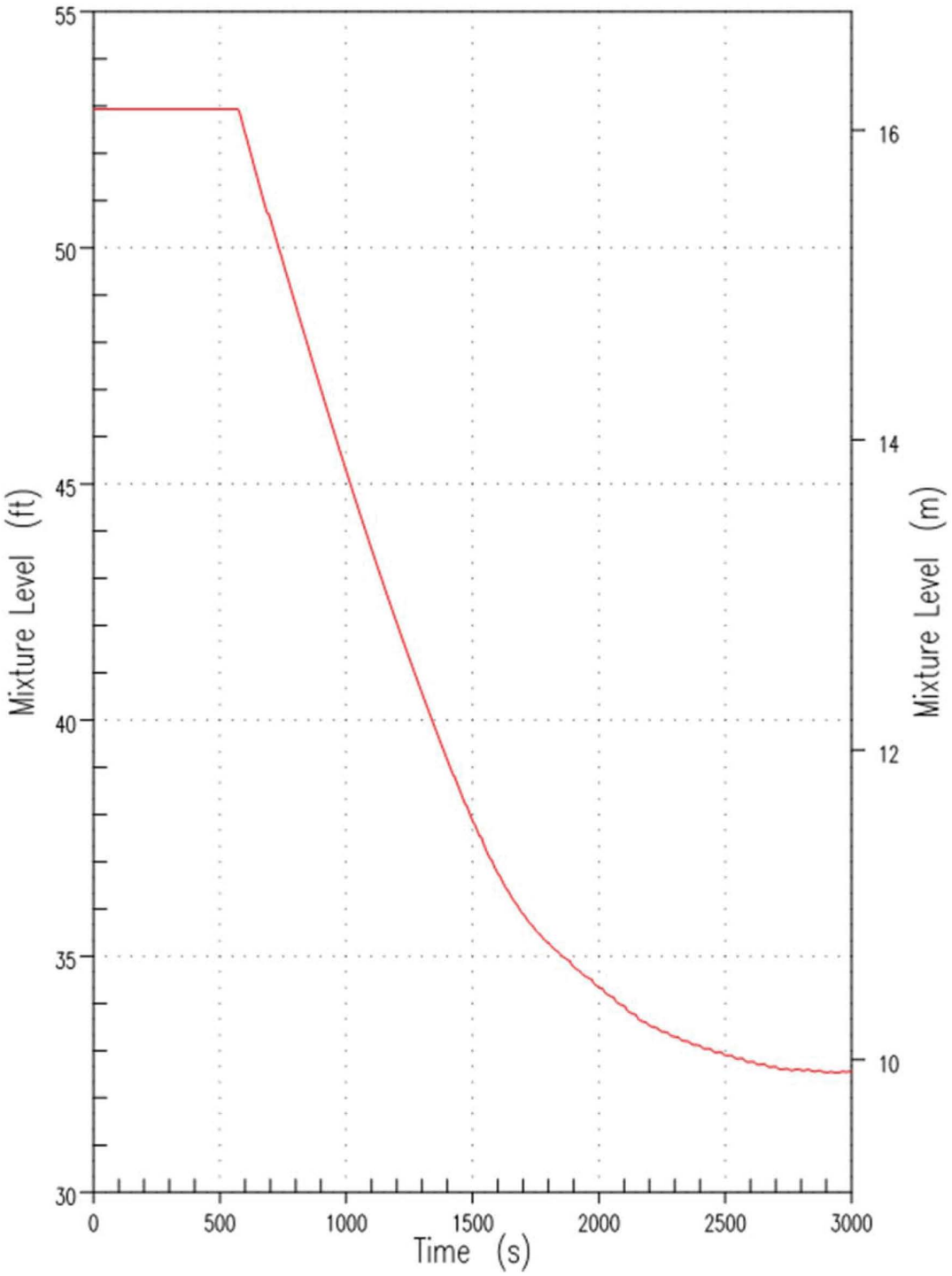

Figure 15.6.5.4B-58
10-inch Cold Leg Break – CMT-1 Mixture Level

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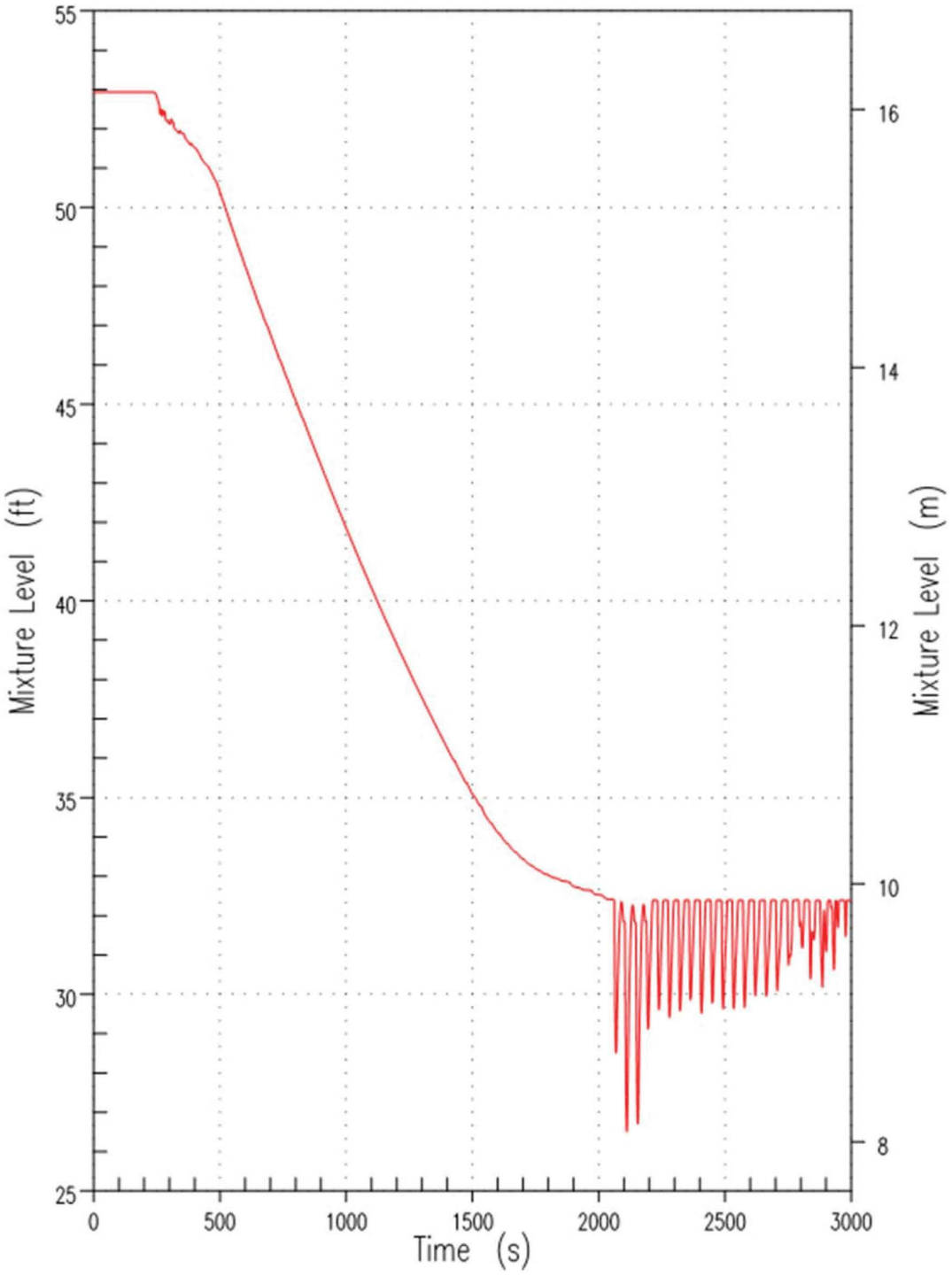

Figure 15.6.5.4B-59
10-inch Cold Leg Break – CMT-2 Mixture Level

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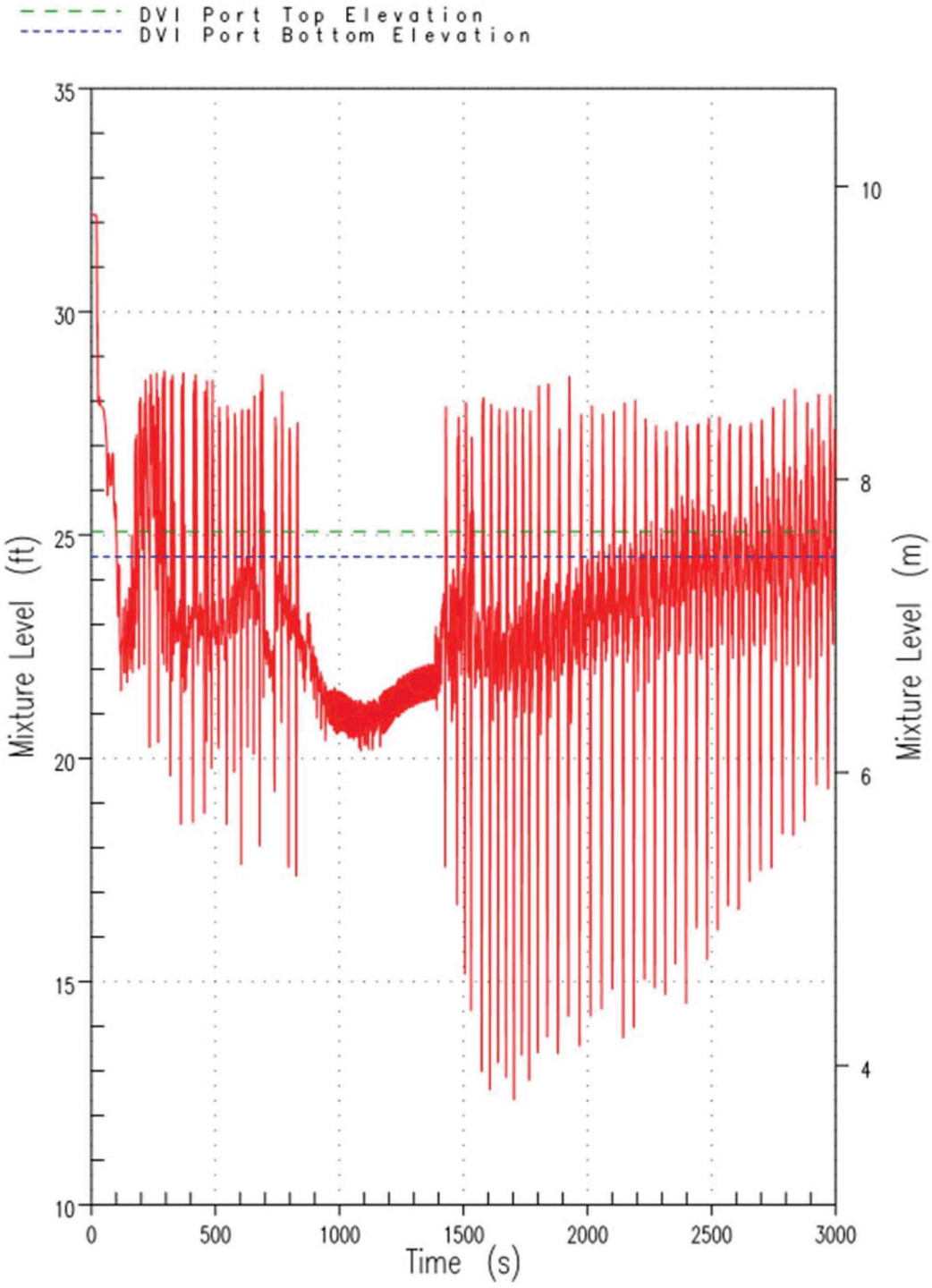

Figure 15.6.5.4B-60
10-inch Cold Leg Break – Downcomer Mixture Level

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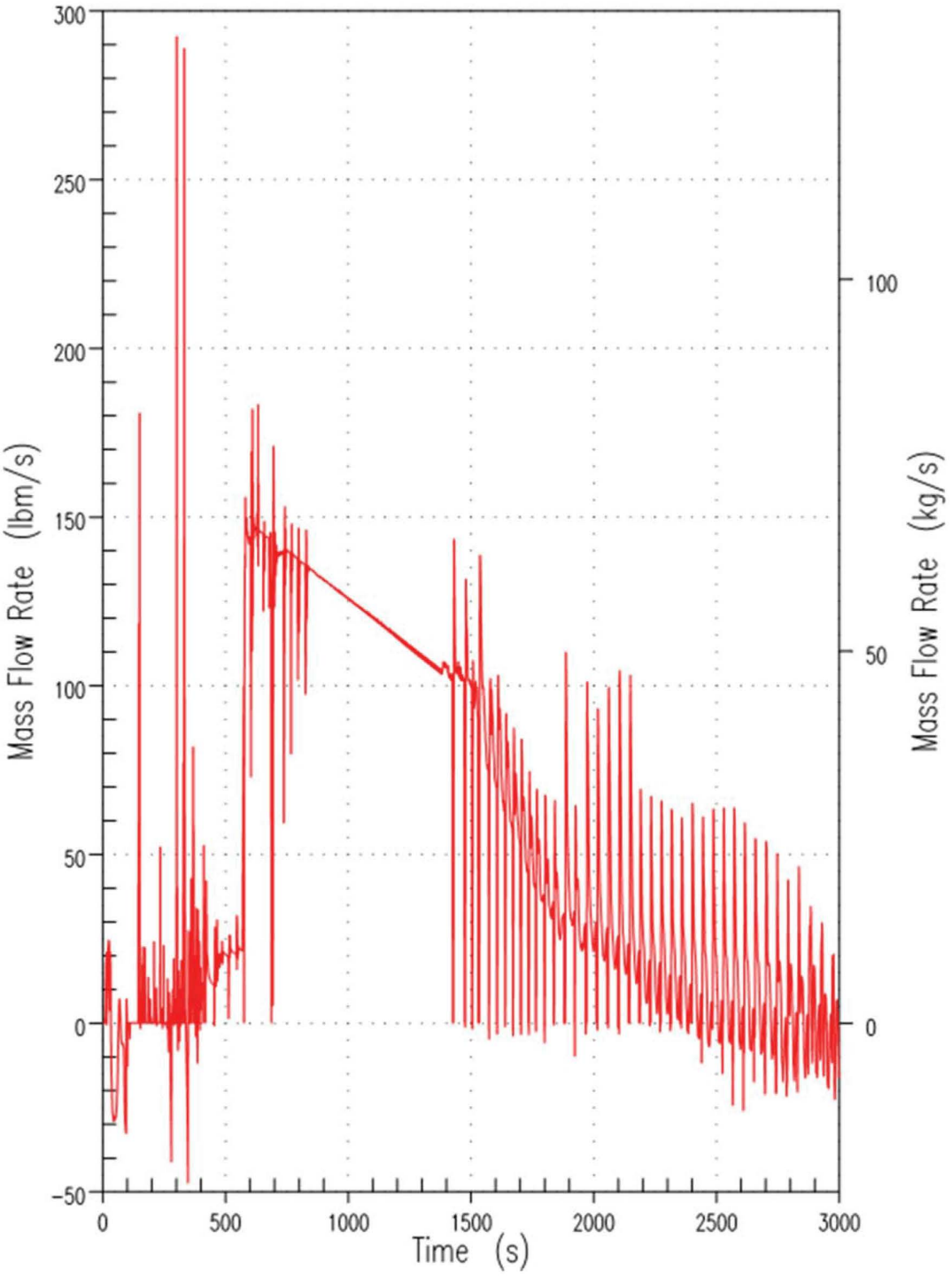

Figure 15.6.5.4B-61
10-inch Cold Leg Break – CMT-1 Injection Rate

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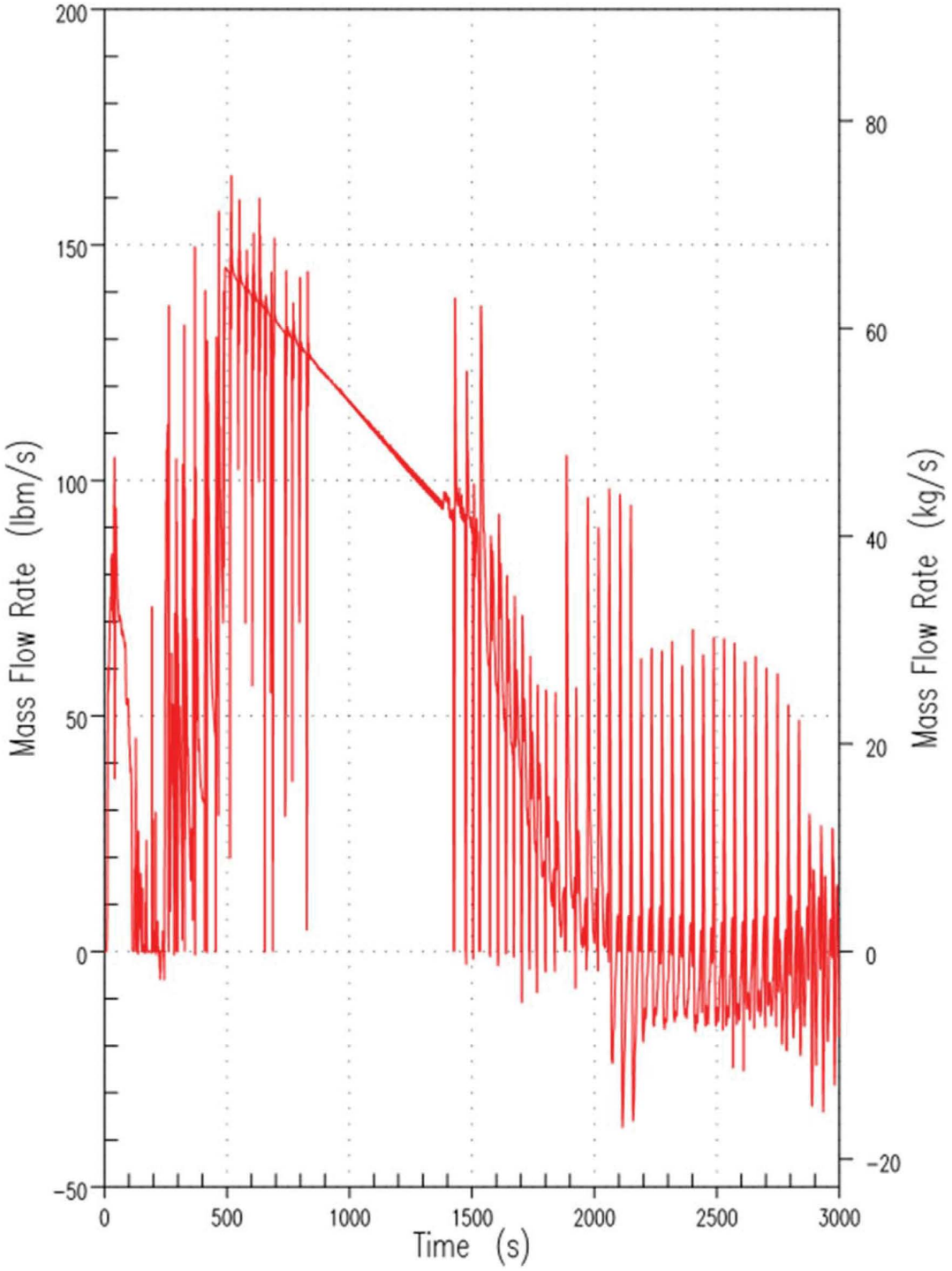

Figure 15.6.5.4B-62
10-inch Cold Leg Break – CMT-2 Injection Rate

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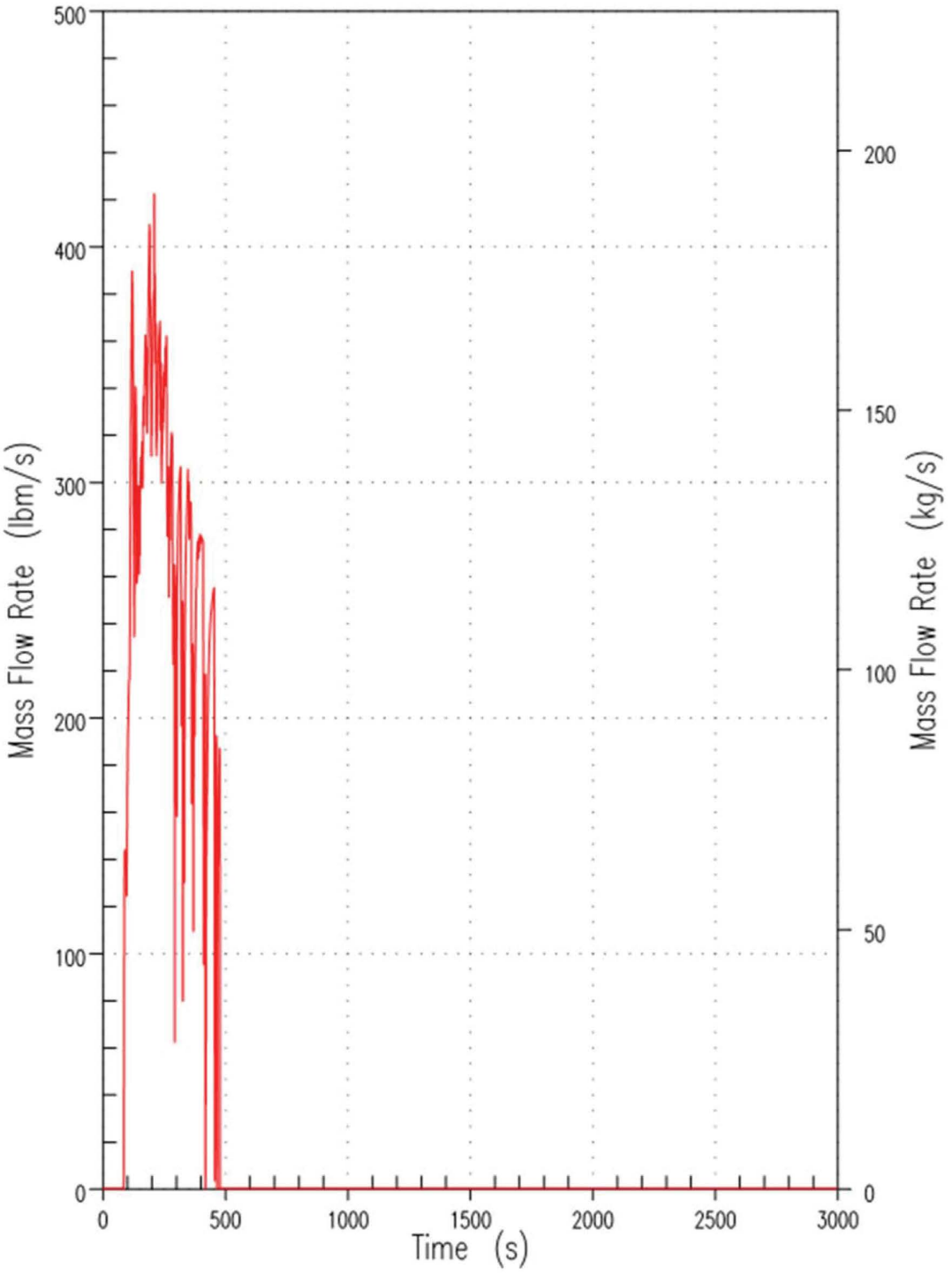

Figure 15.6.5.4B-63
10-inch Cold Leg Break – Accumulator-1 Injection Rate

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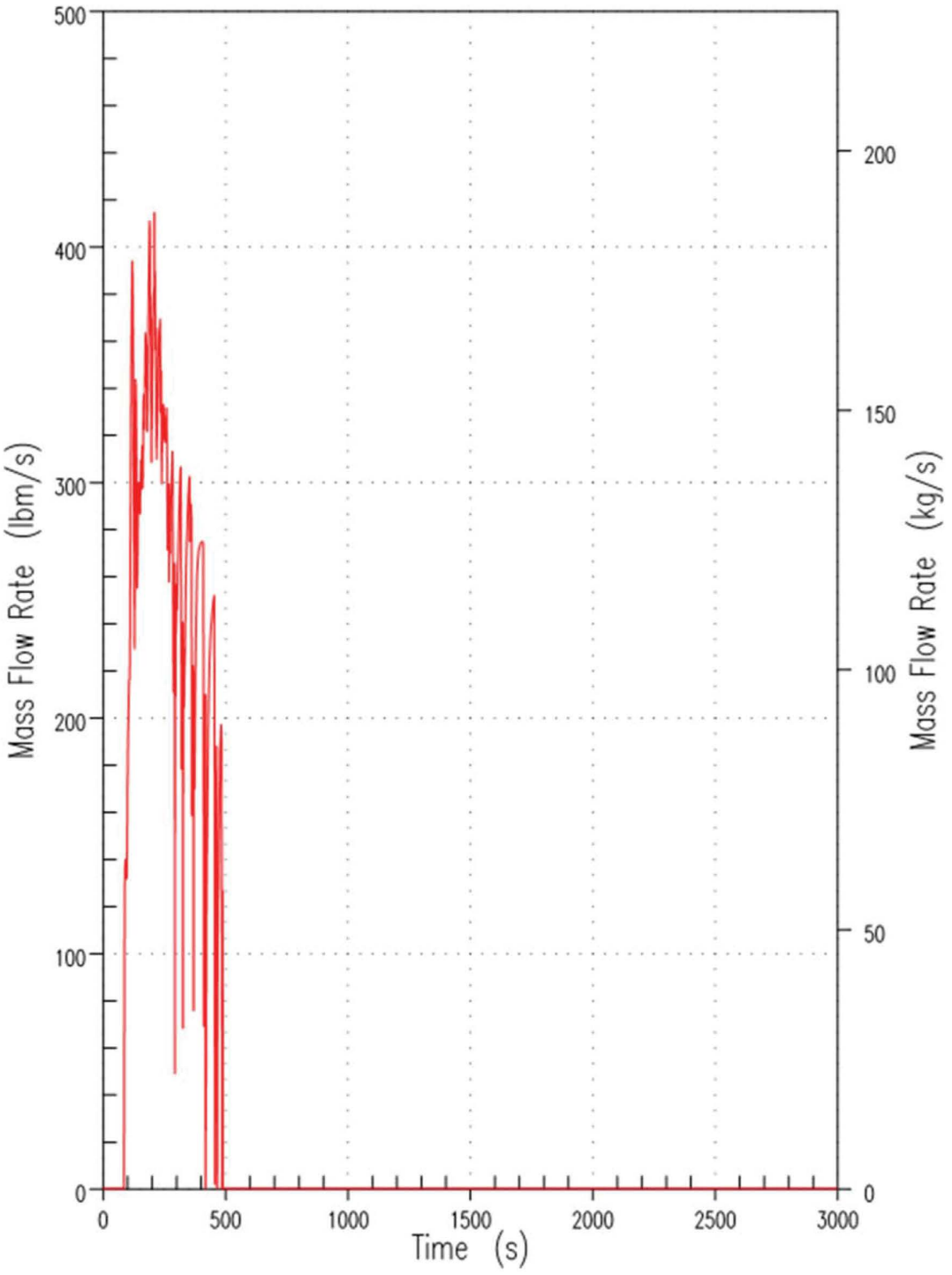

Figure 15.6.5.4B-64
10-inch Cold Leg Break – Accumulator-2 Injection Rate

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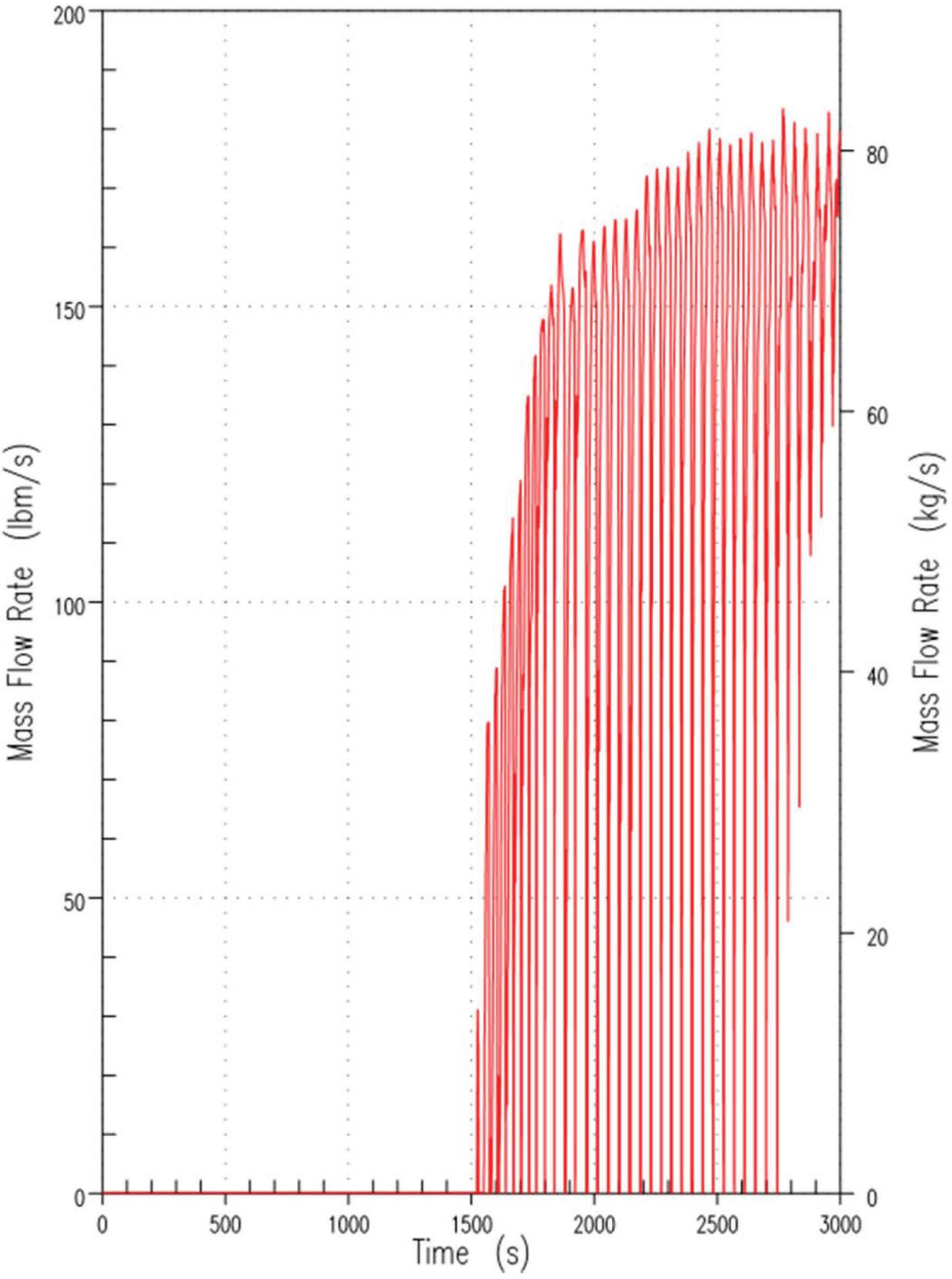

Figure 15.6.5.4B-65
10-inch Cold Leg Break – IRWST-1 Injection Rate

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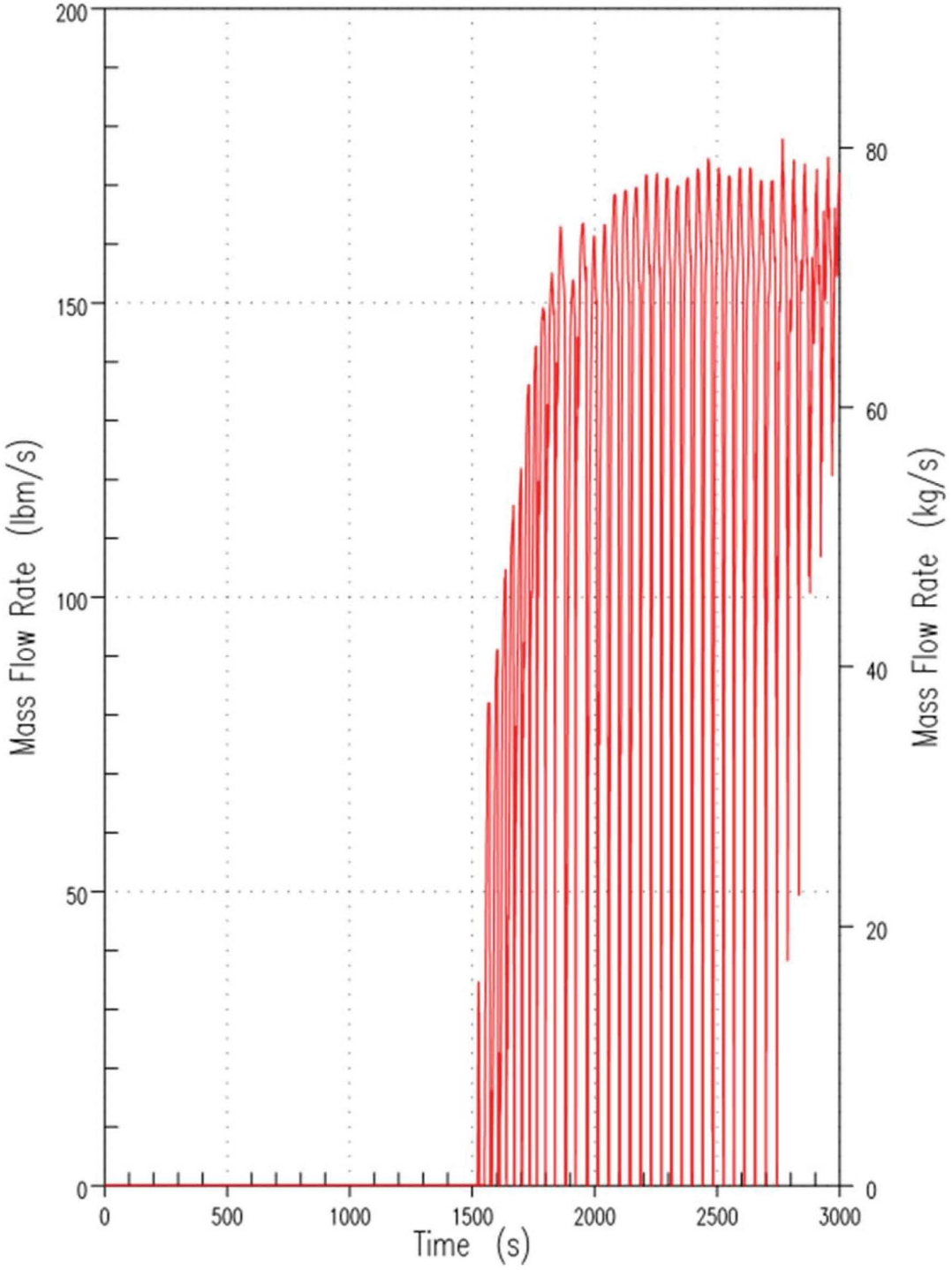

Figure 15.6.5.4B-66
10-inch Cold Leg Break – IRWST-2 Injection Rate

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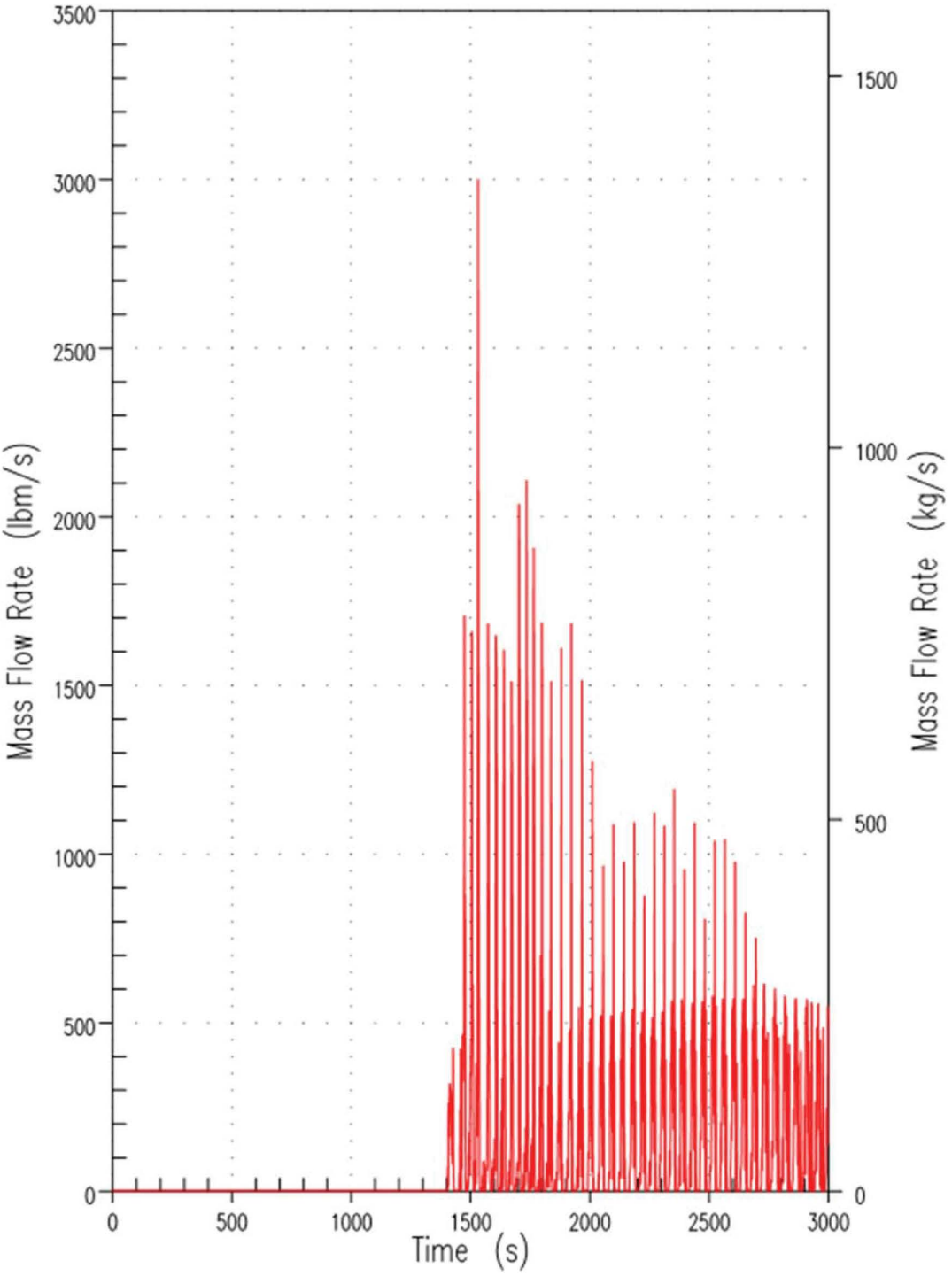

Figure 15.6.5.4B-67(a)
10-inch Cold Leg Break – ADS-4 Liquid Discharge

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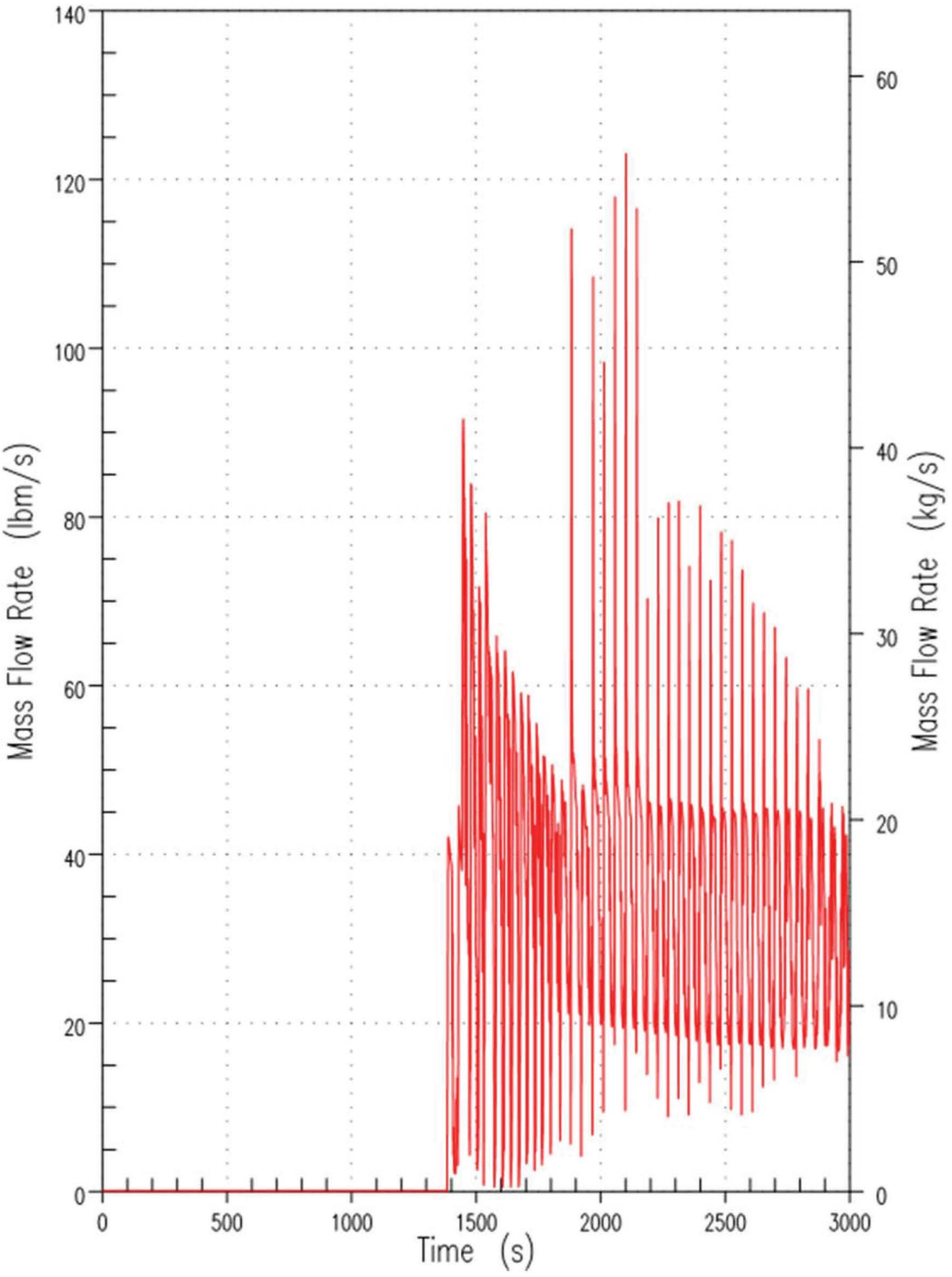

Figure 15.6.5.4B-67(b)
10-Inch Cold Leg Break – ADS-4 Vapor Discharge

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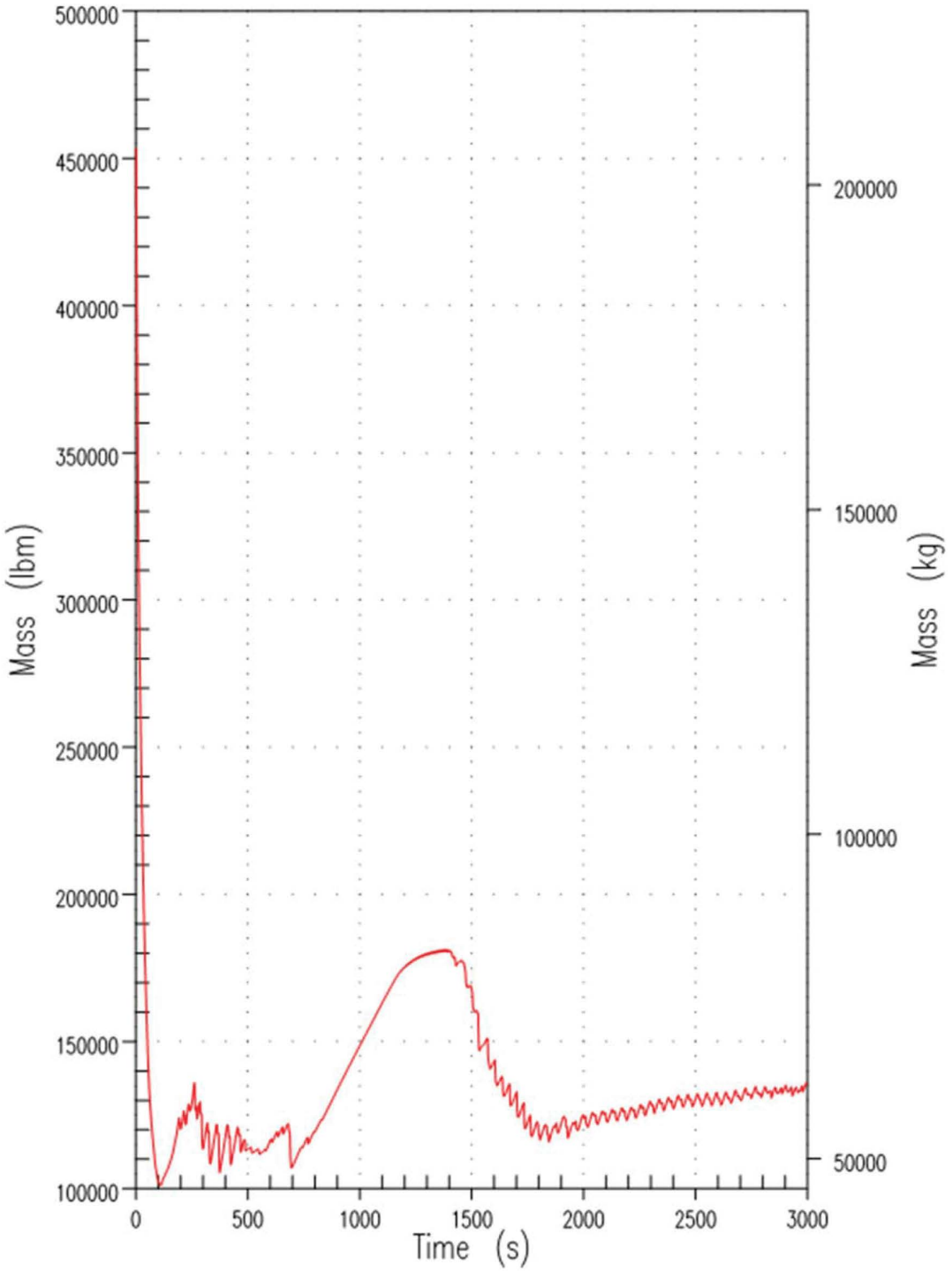

Figure 15.6.5.4B-68(a)
10-inch Cold Leg Break – RCS System Inventory

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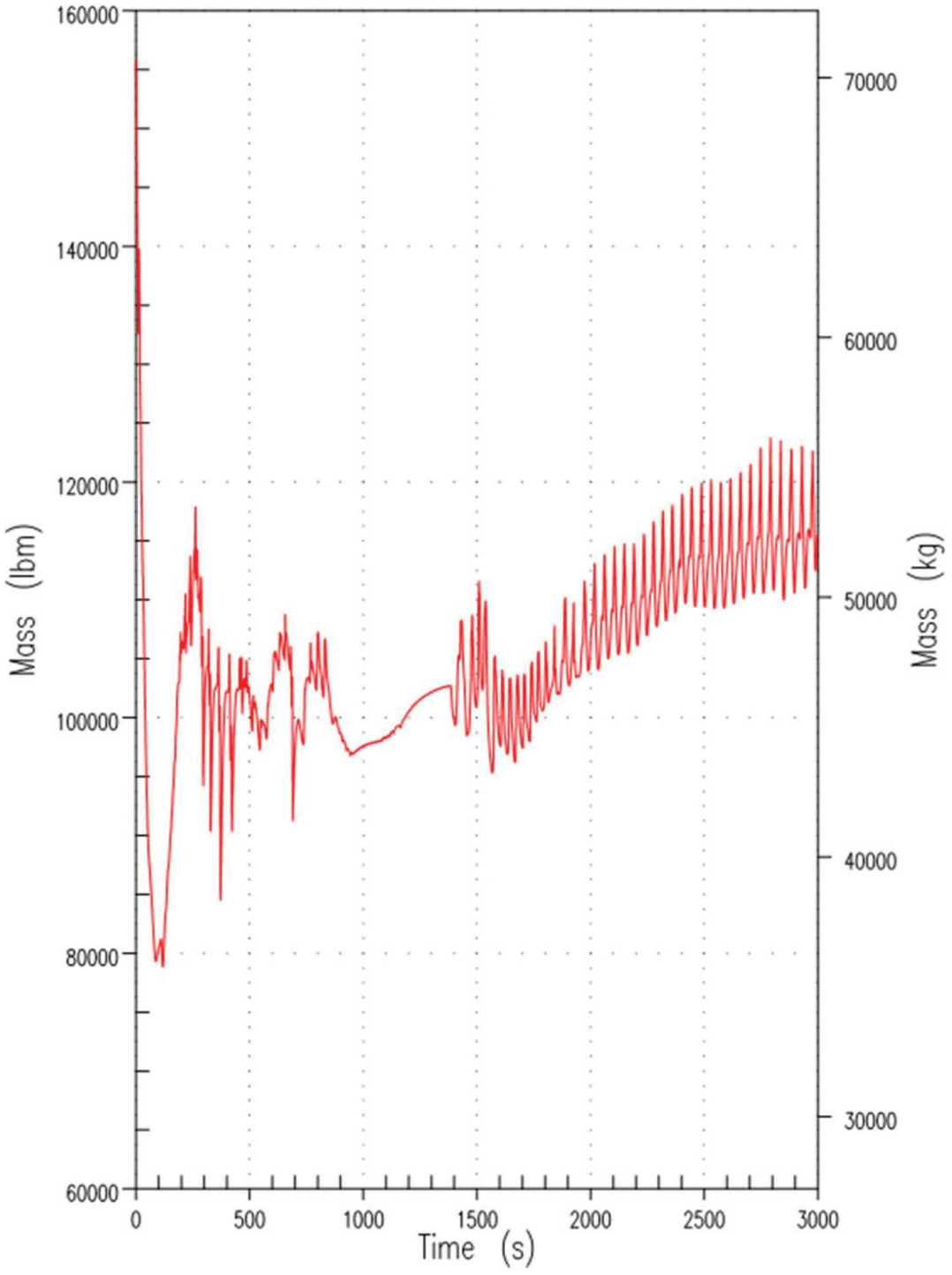

Figure 15.6.5.4B-68(b)
10-Inch Cold Leg Break – Reactor Vessel Mixture Inventory

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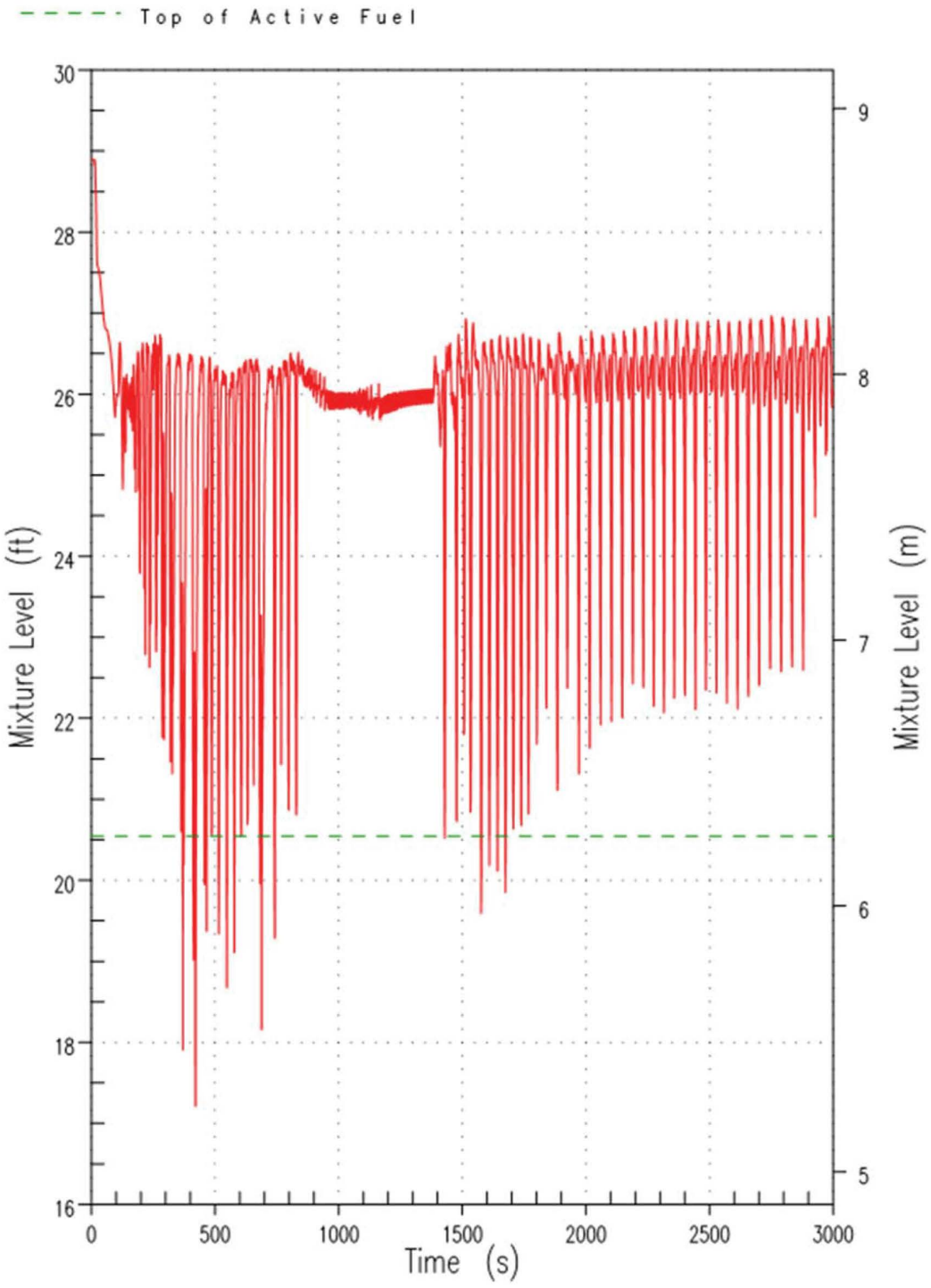

Figure 15.6.5.4B-69
10-inch Cold Leg Break – Core/Upper Plenum Mixture Level

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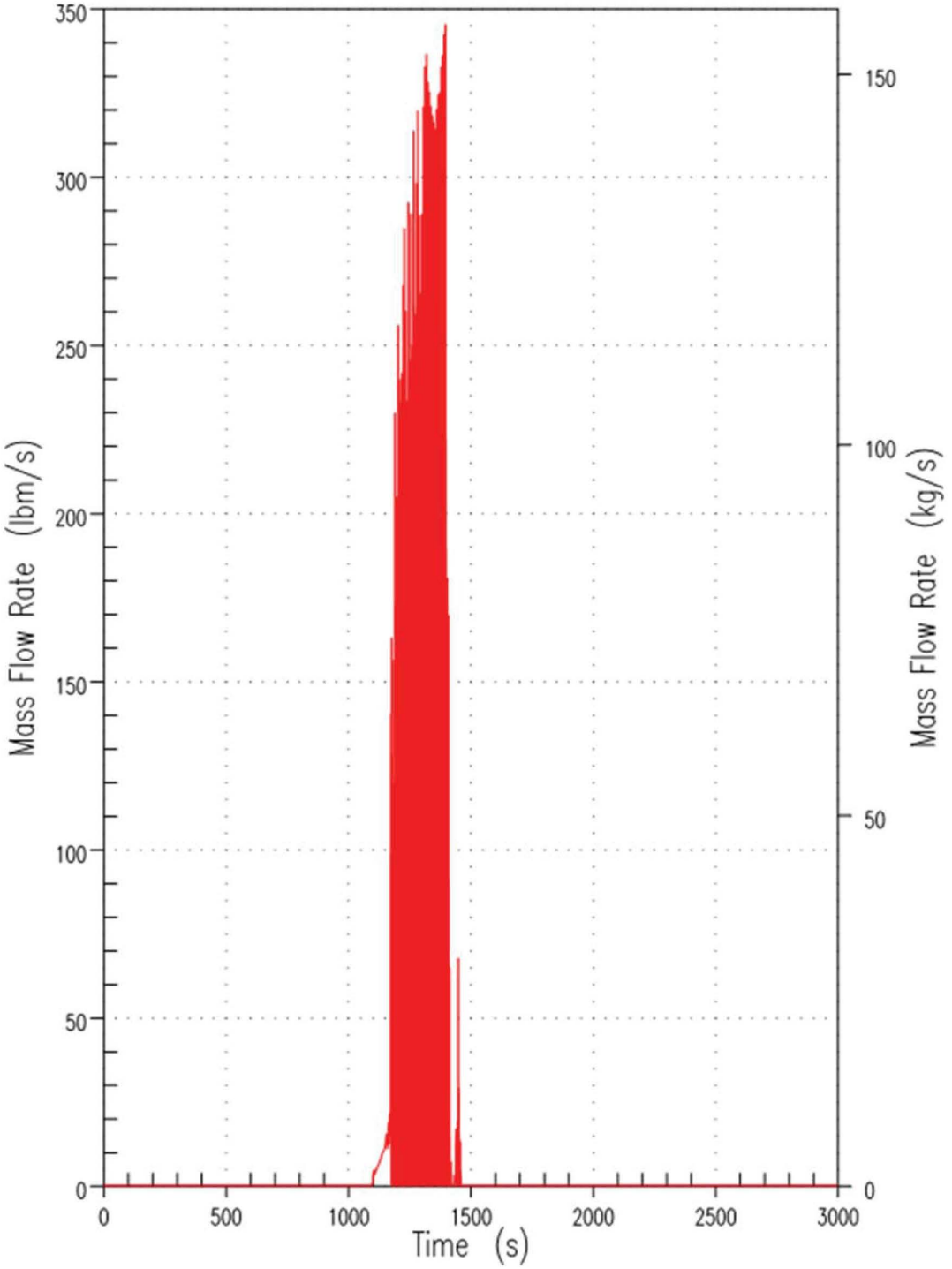

Figure 15.6.5.4B-70(a)
10-inch Cold Leg Break – ADS 1-3 Liquid Discharge

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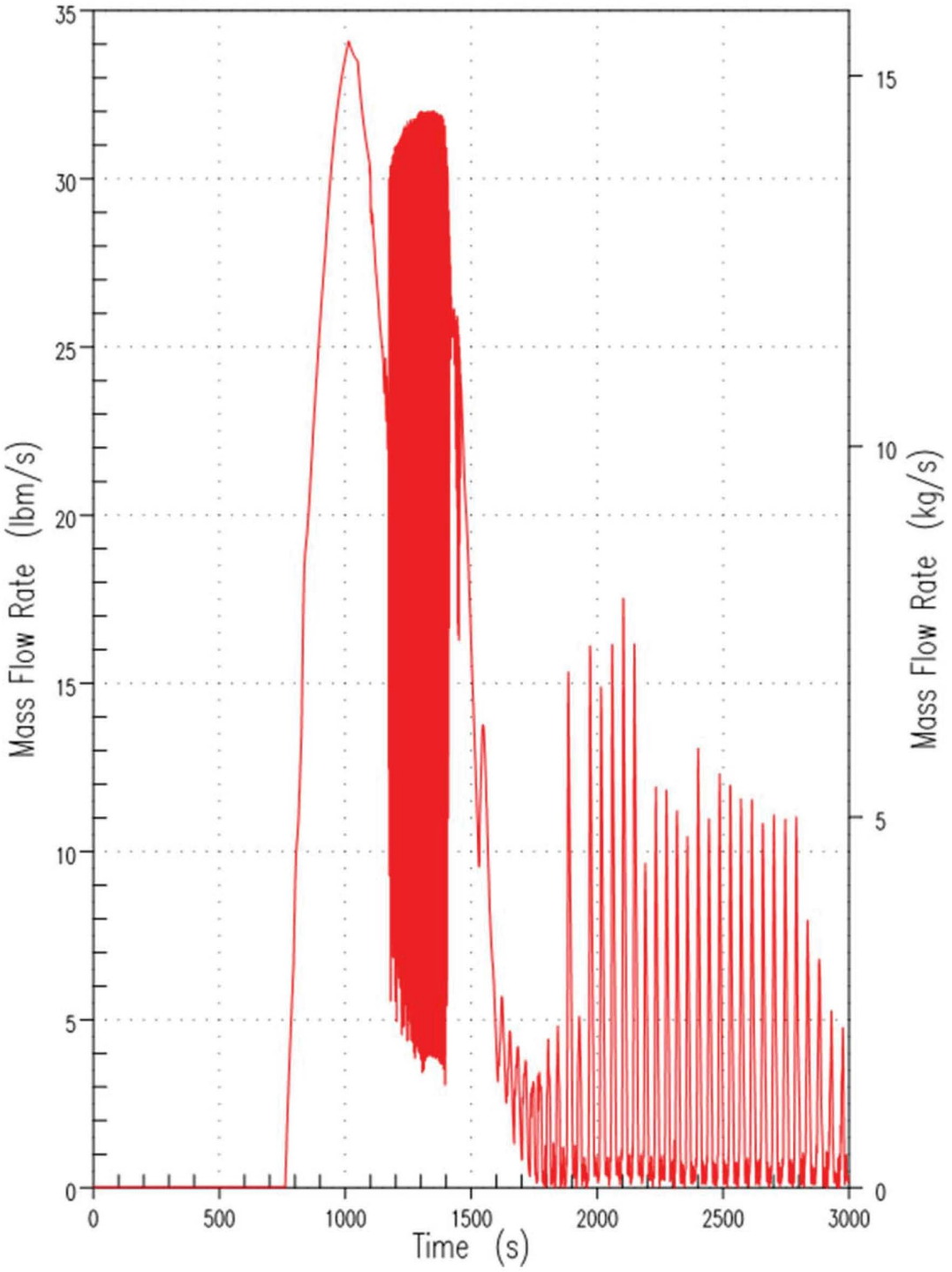

Figure 15.6.5.4B-70(b)
10-Inch Cold Leg Break – ADS 1-3 Vapor Discharge

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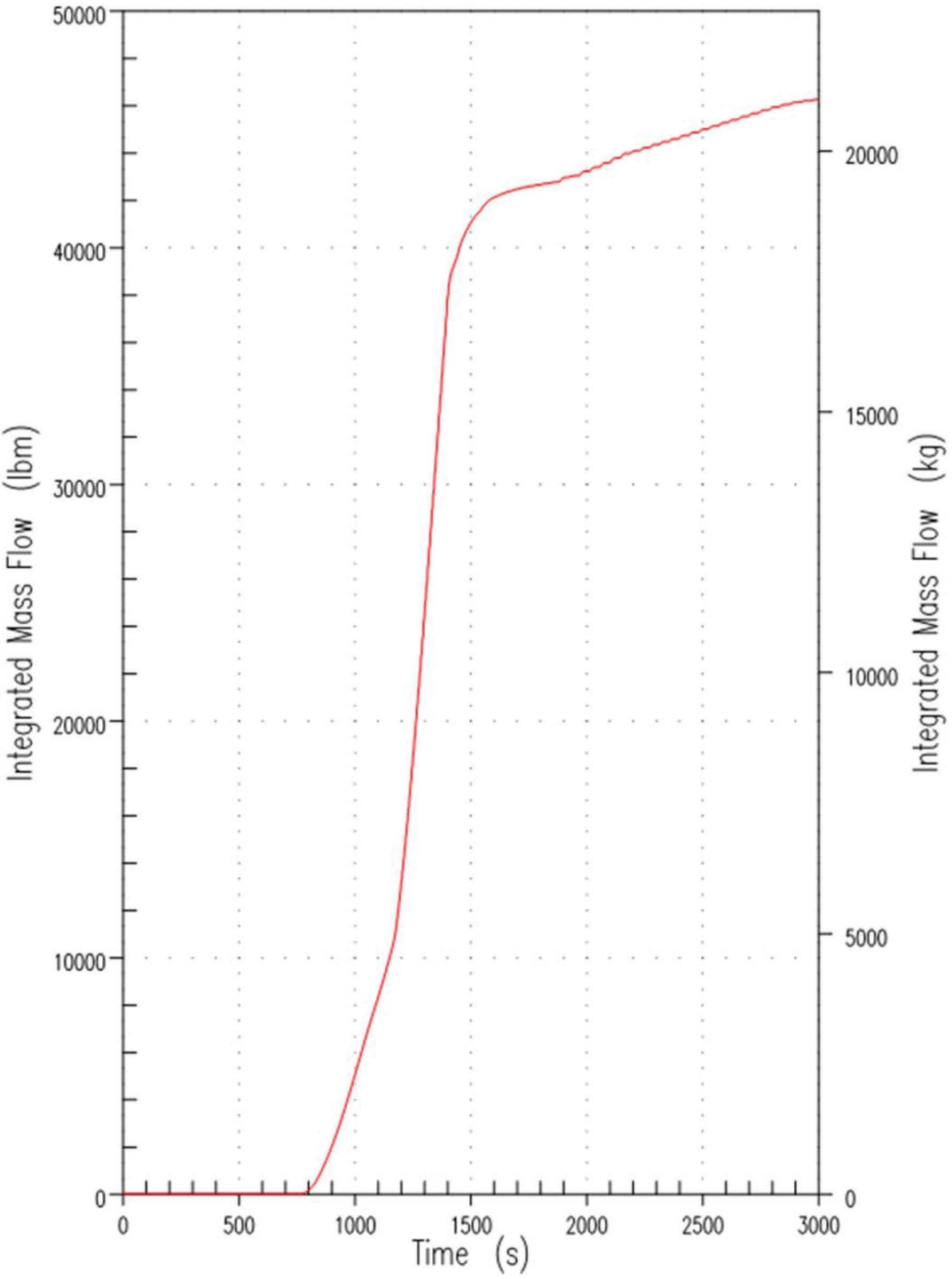

Figure 15.6.5.4B-70(c)
10-Inch Cold Leg Break – ADS 1-3 Integrated Discharge

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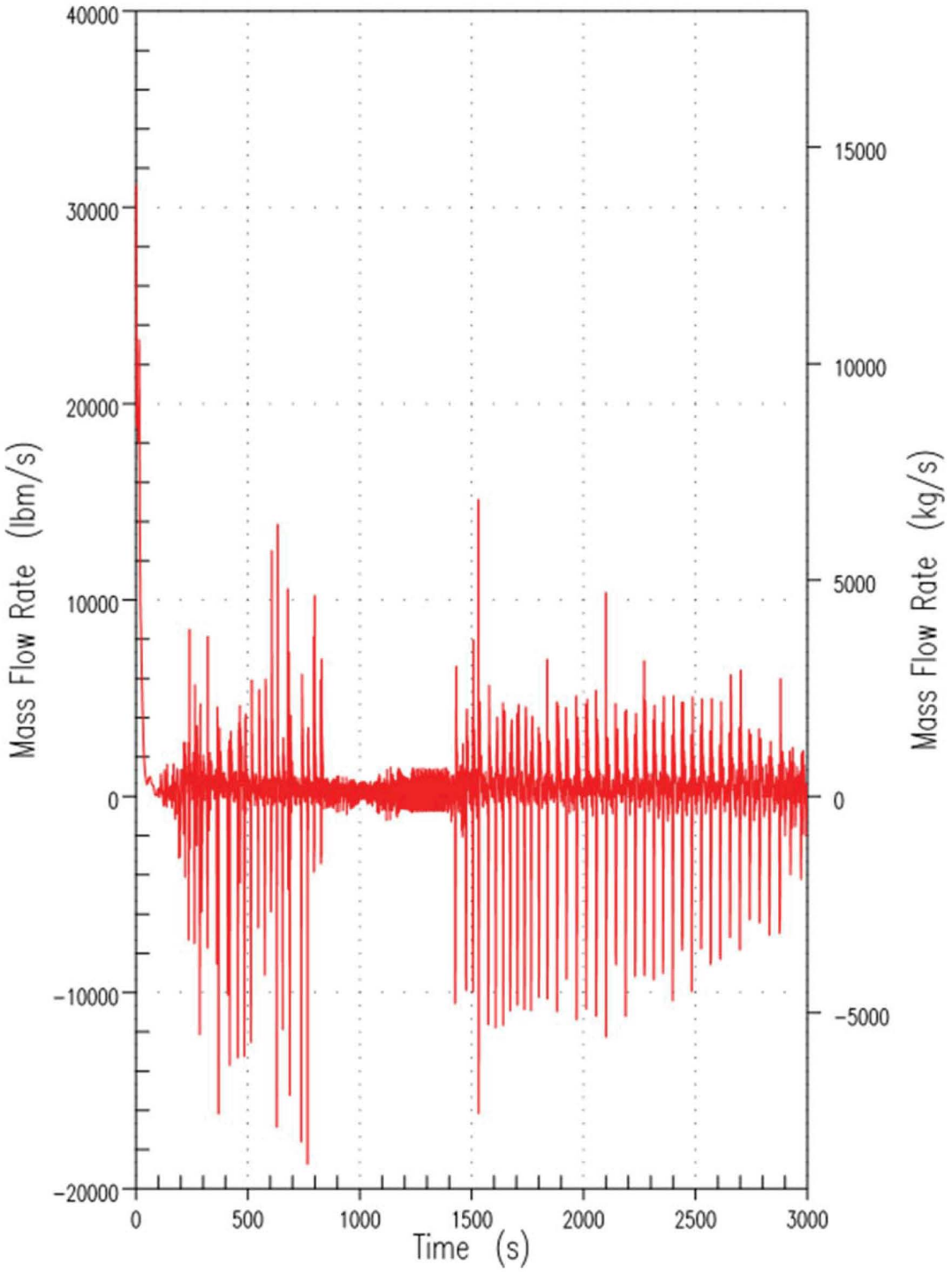

Figure 15.6.5.4B-71
10-inch Cold Leg Break – Core Exit Liquid Flow

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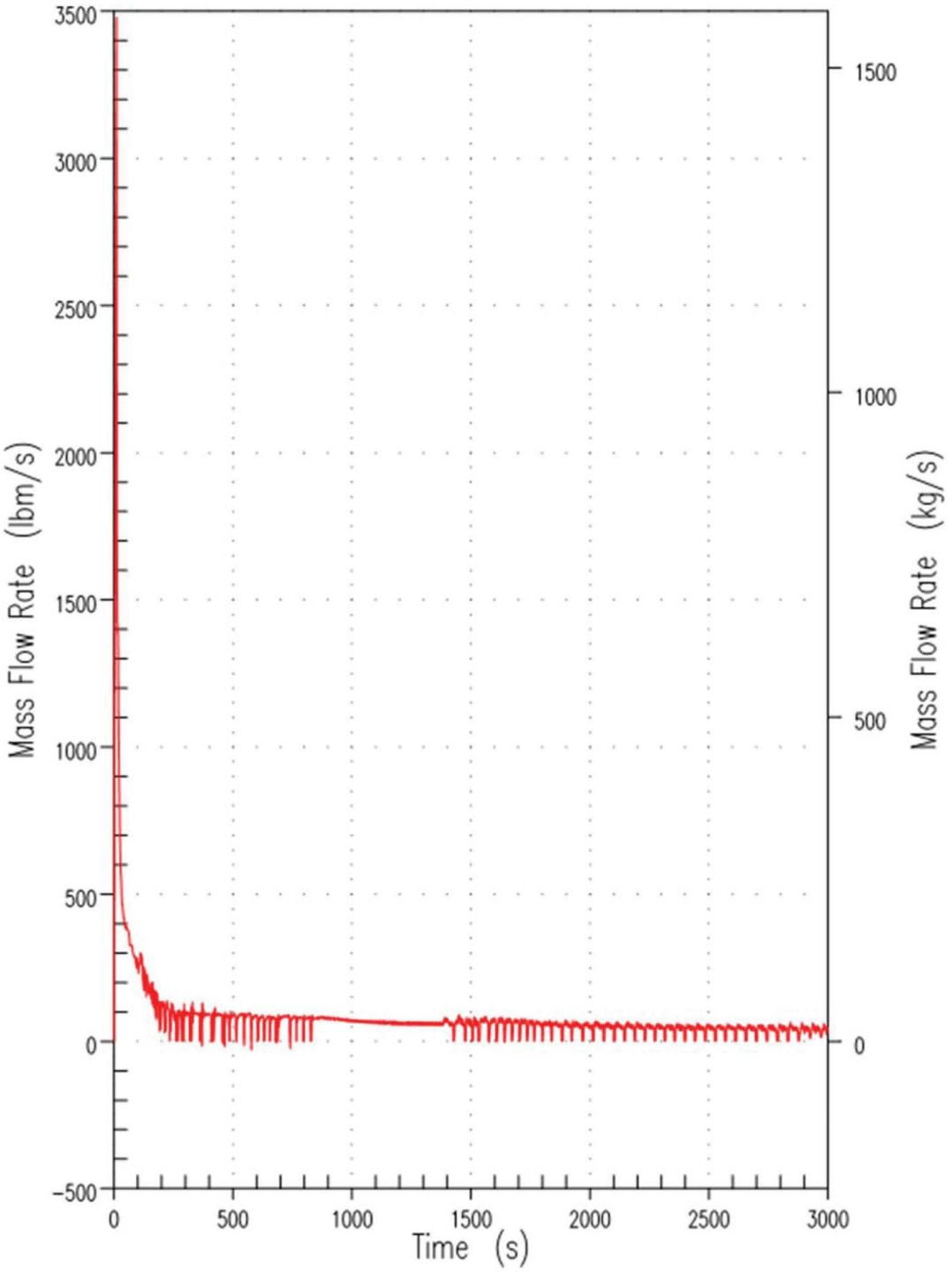

Figure 15.6.5.4B-72
10-inch Cold Leg Break – Core Exit Vapor Flow

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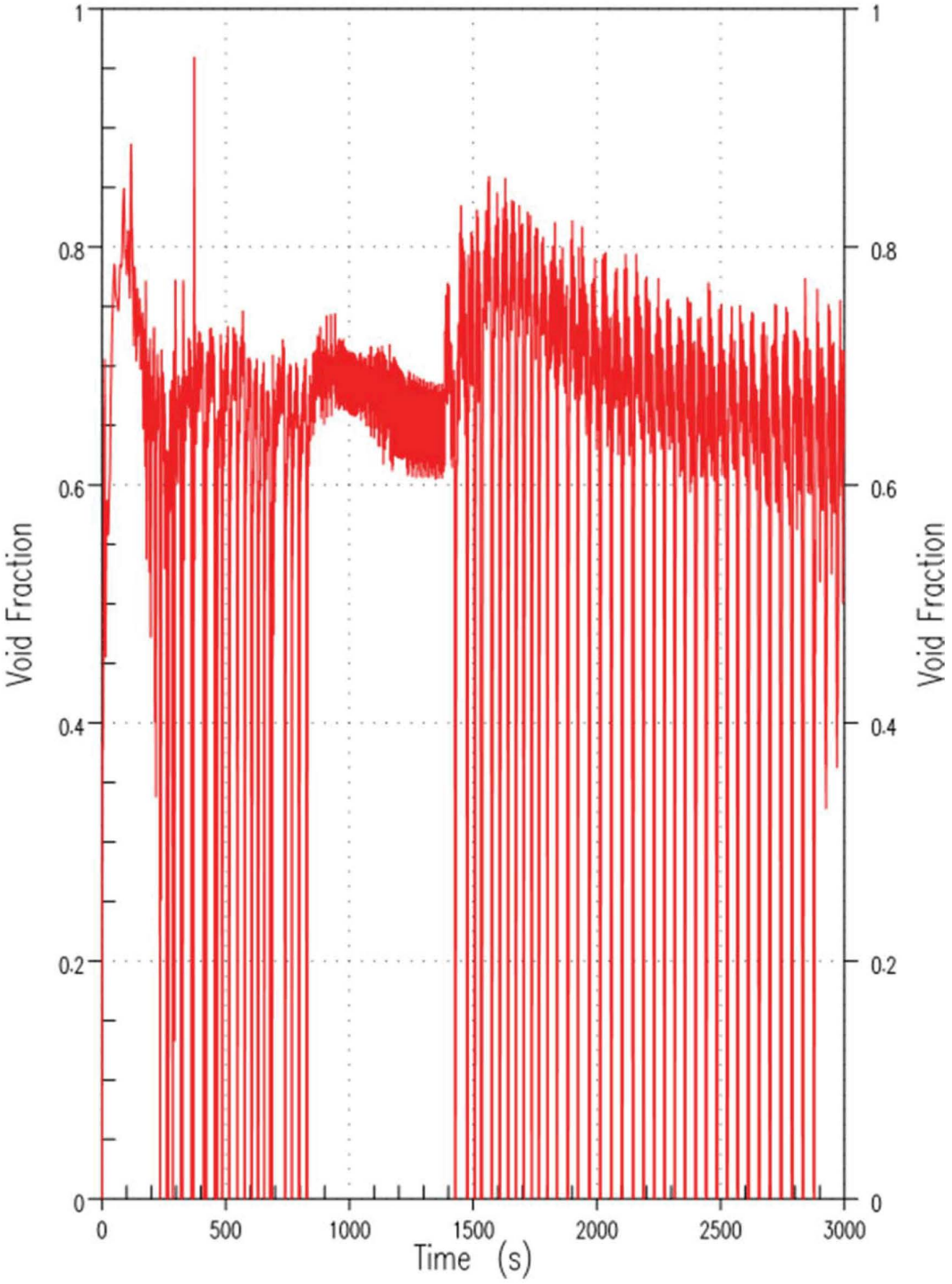

Figure 15.6.5.4B-73
10-inch Cold Leg Break – Core Exit Void Fraction

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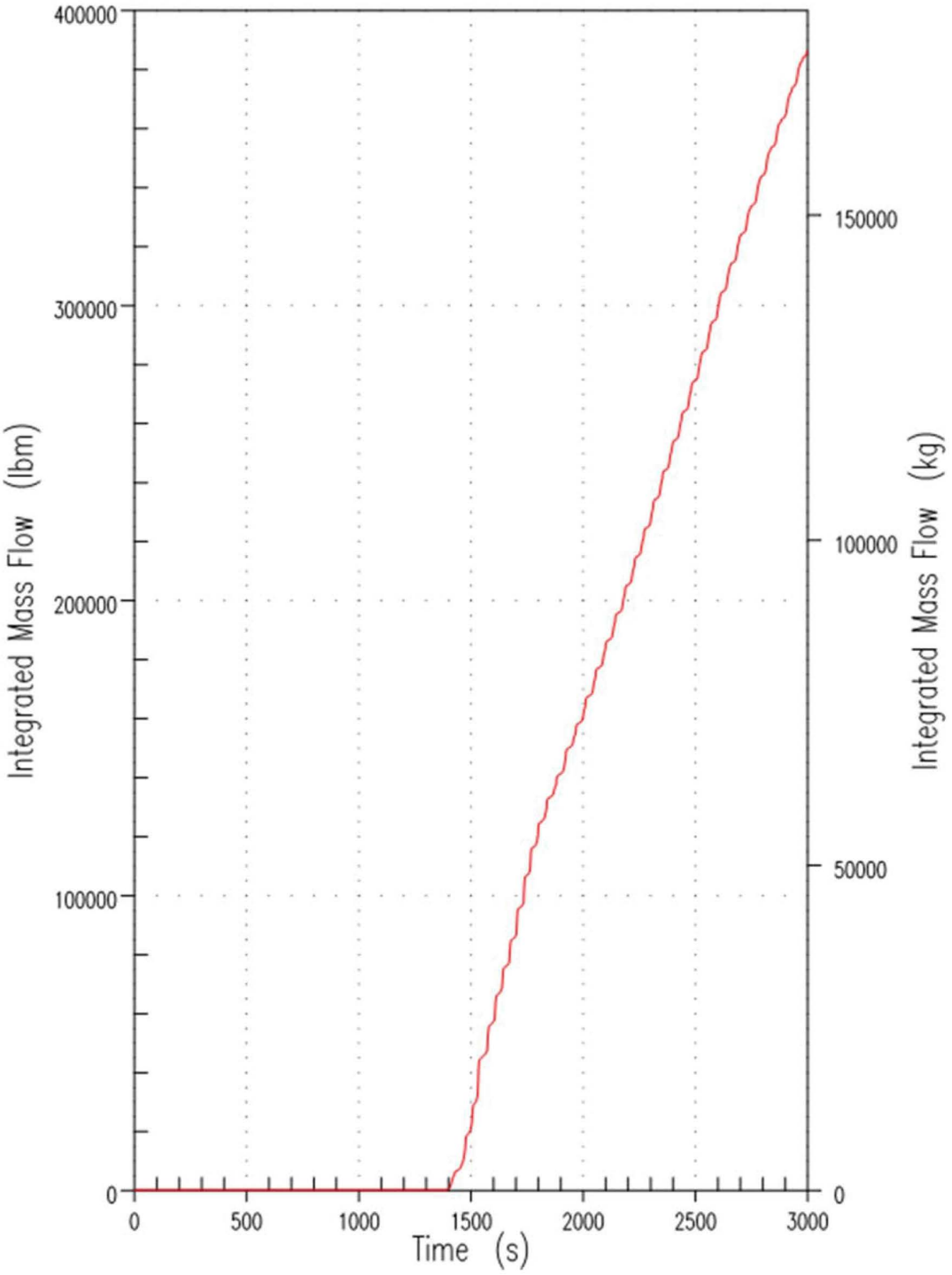

Figure 15.6.5.4B-74
10-inch Cold Leg Break – ADS-4 Integrated Discharge

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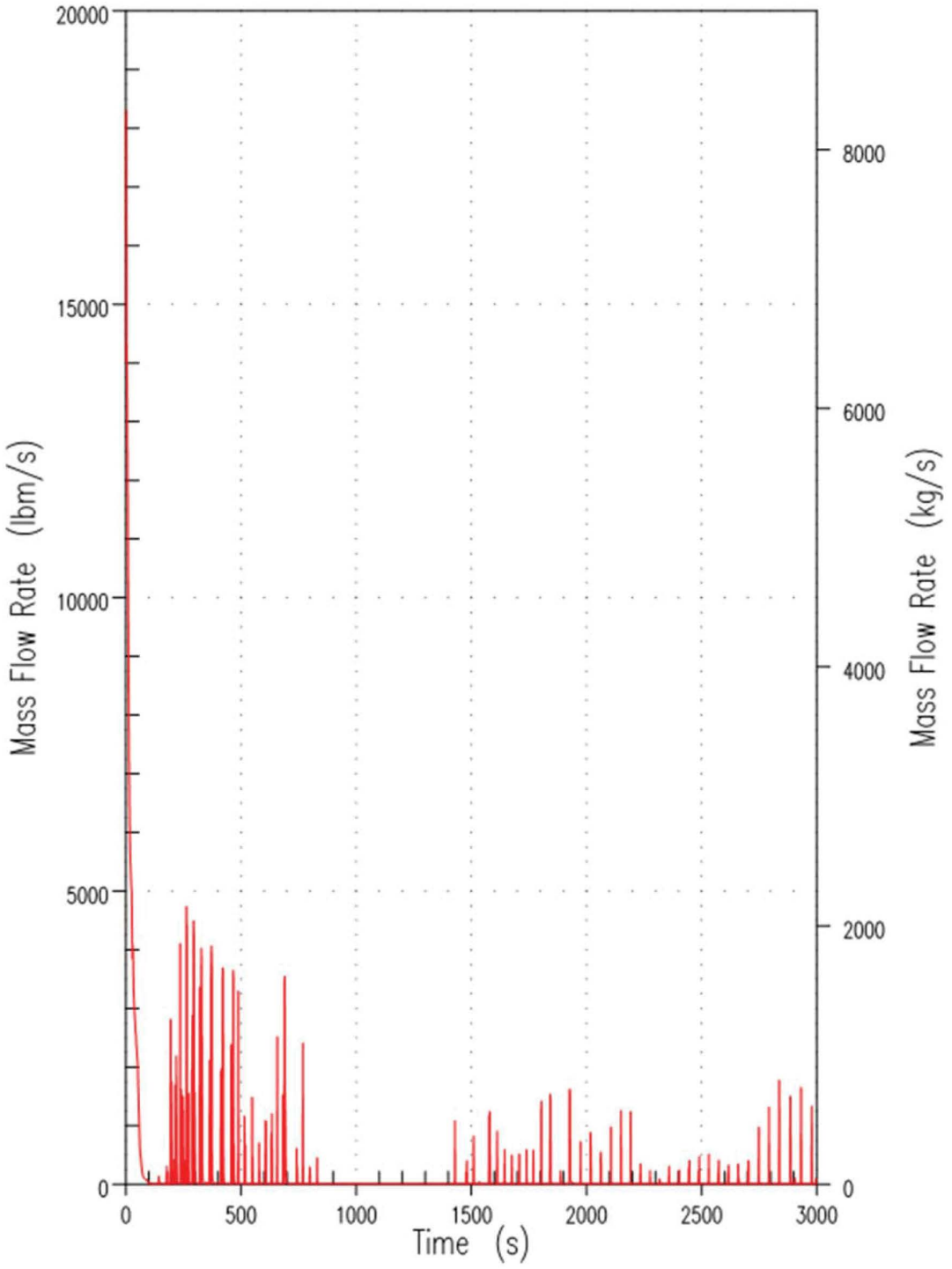

Figure 15.6.5.4B-75
10-inch Cold Leg Break – Liquid Break Discharge

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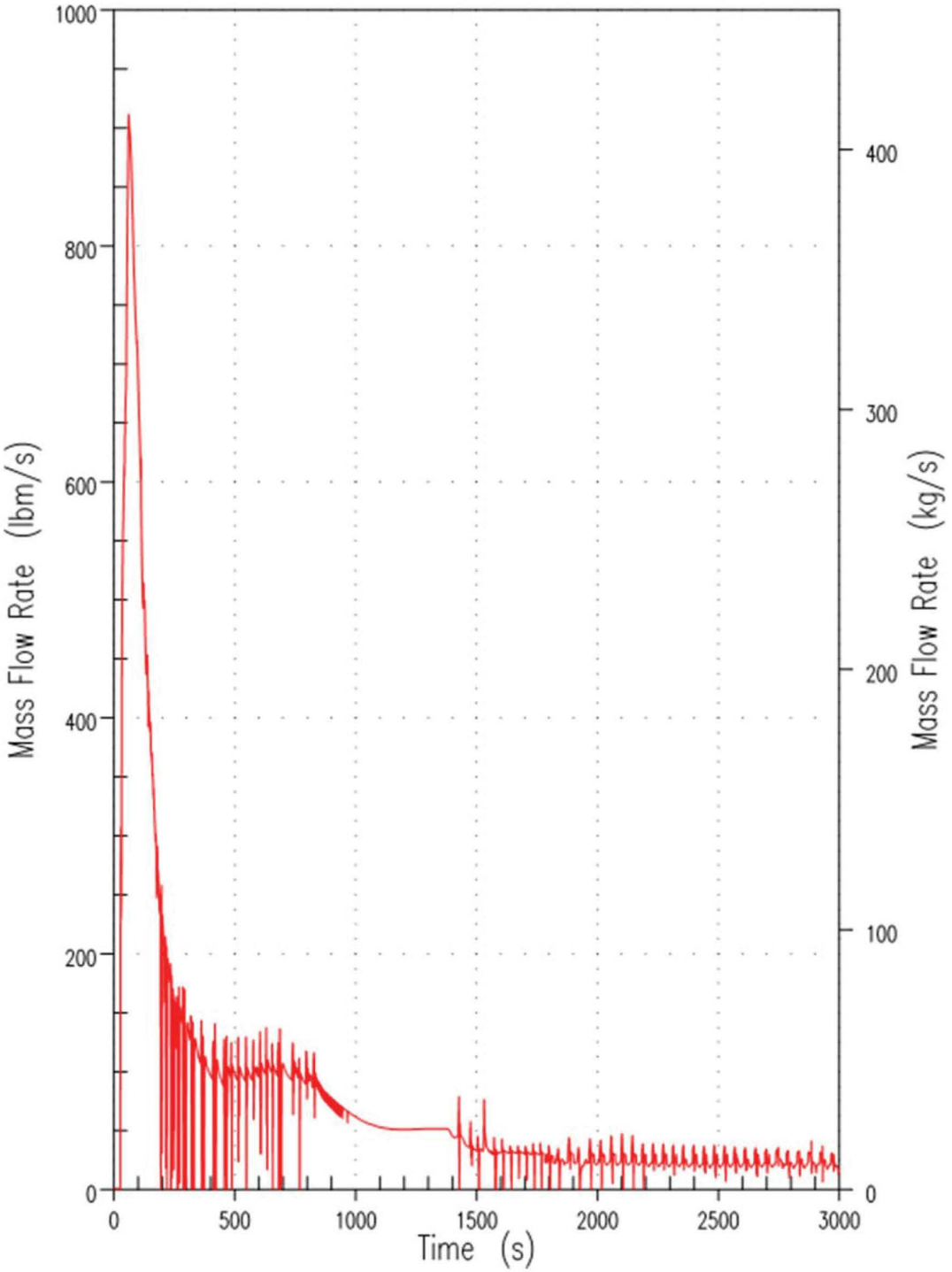

Figure 15.6.5.4B-76
10-inch Cold Leg Break – Vapor Break Discharge

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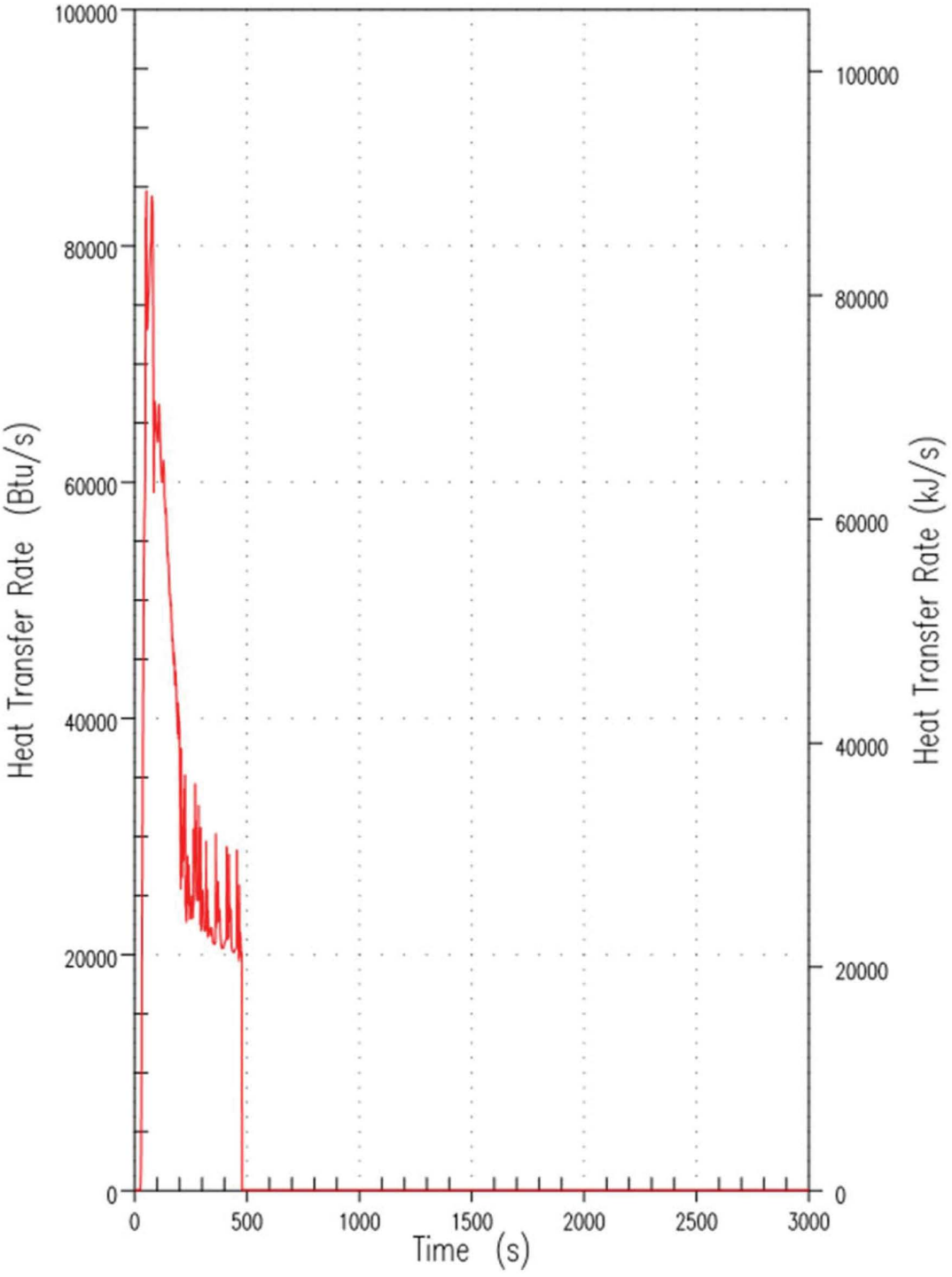

Figure 15.6.5.4B-77
10-inch Cold Leg Break – PRHR Heat Removal Rate

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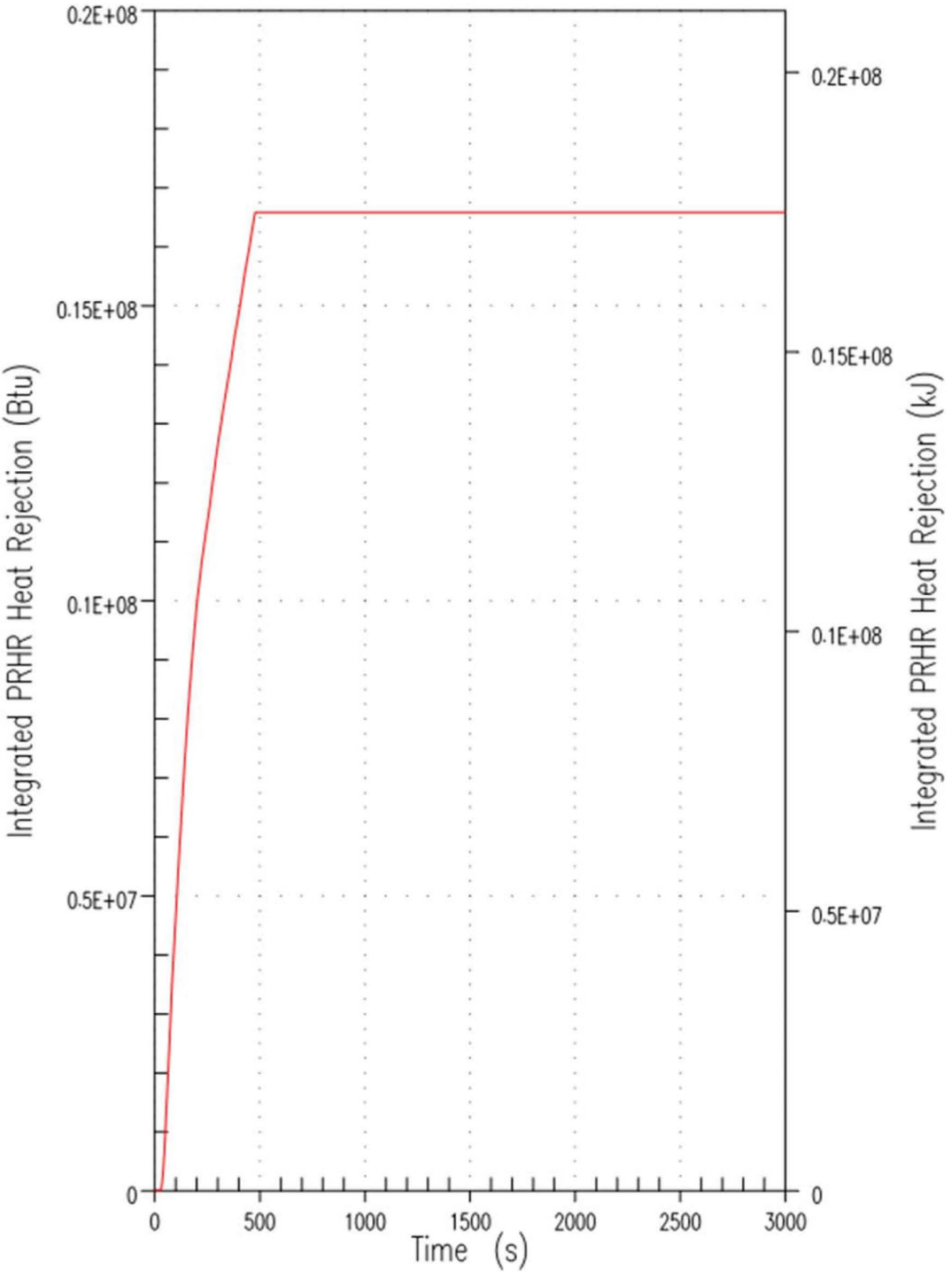

Figure 15.6.5.4B-78
10-inch Cold Leg Break – Integrated PRHR Heat Removal

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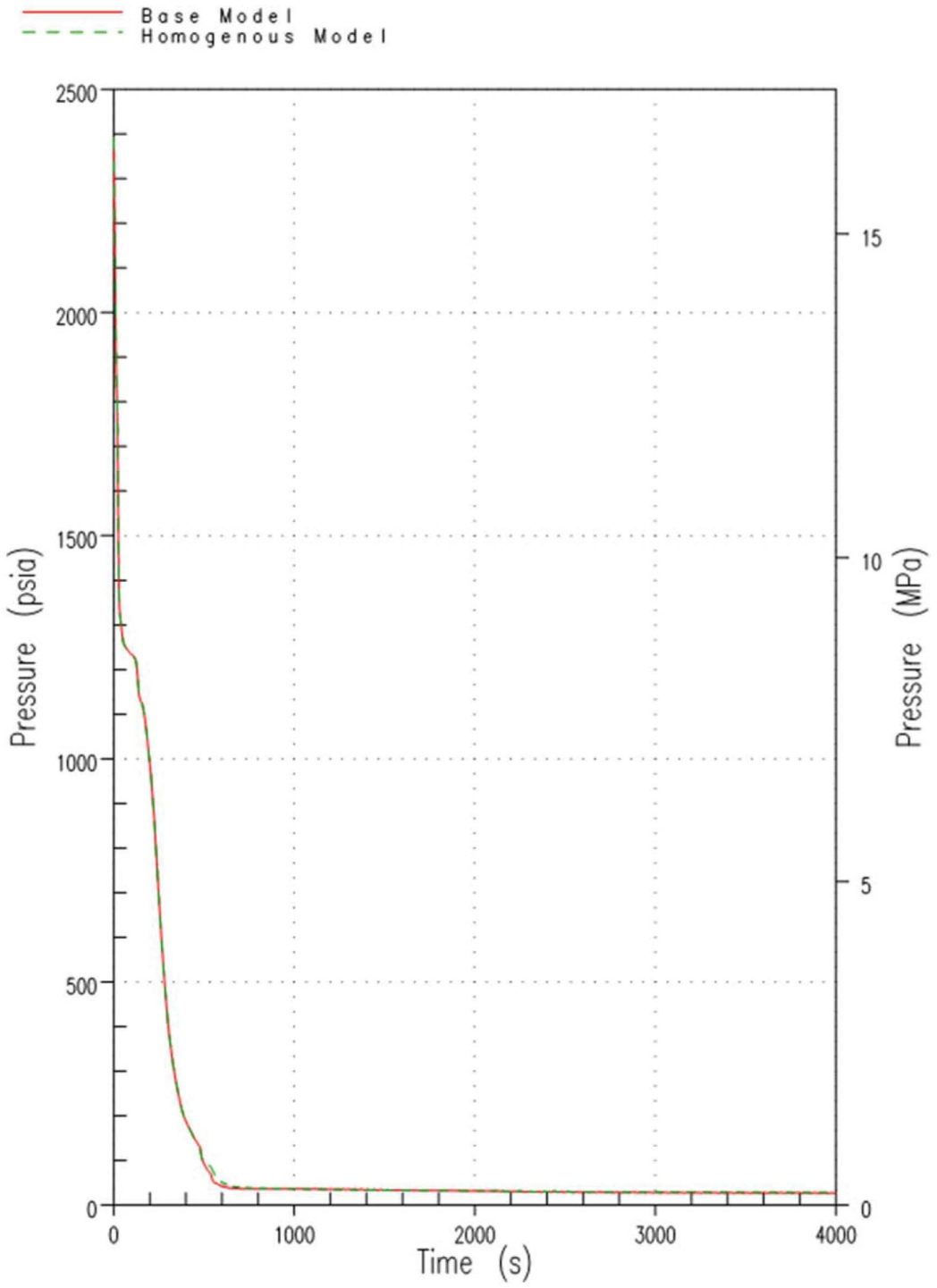

Figure 15.6.5.4B-79(a)
DEDVI Entrainment – Downcomer Pressure Comparison

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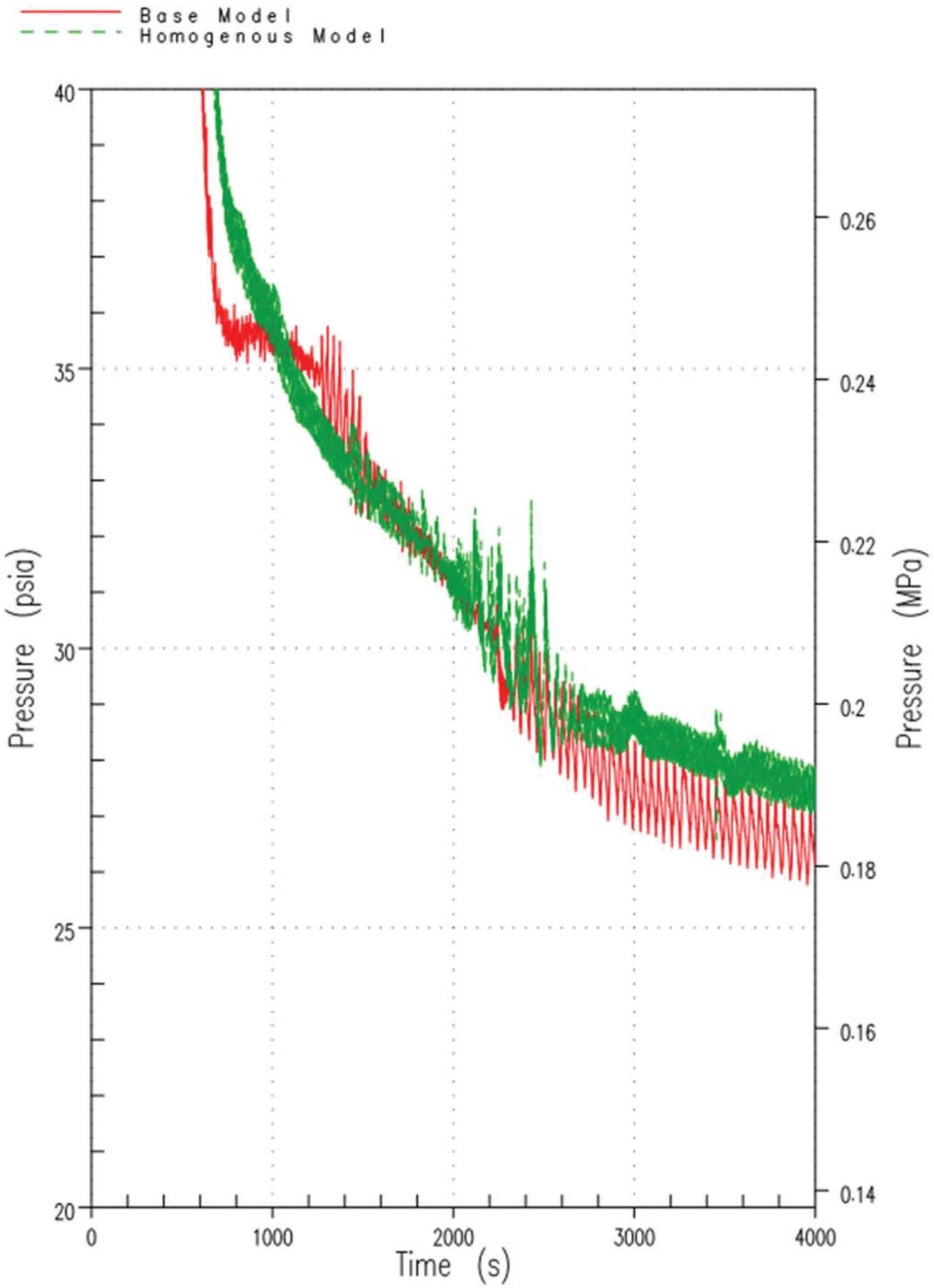

Figure 15.6.5.4B-79(b)
DEDVI Entrainment – Downcomer Pressure Comparison (Zoomed)

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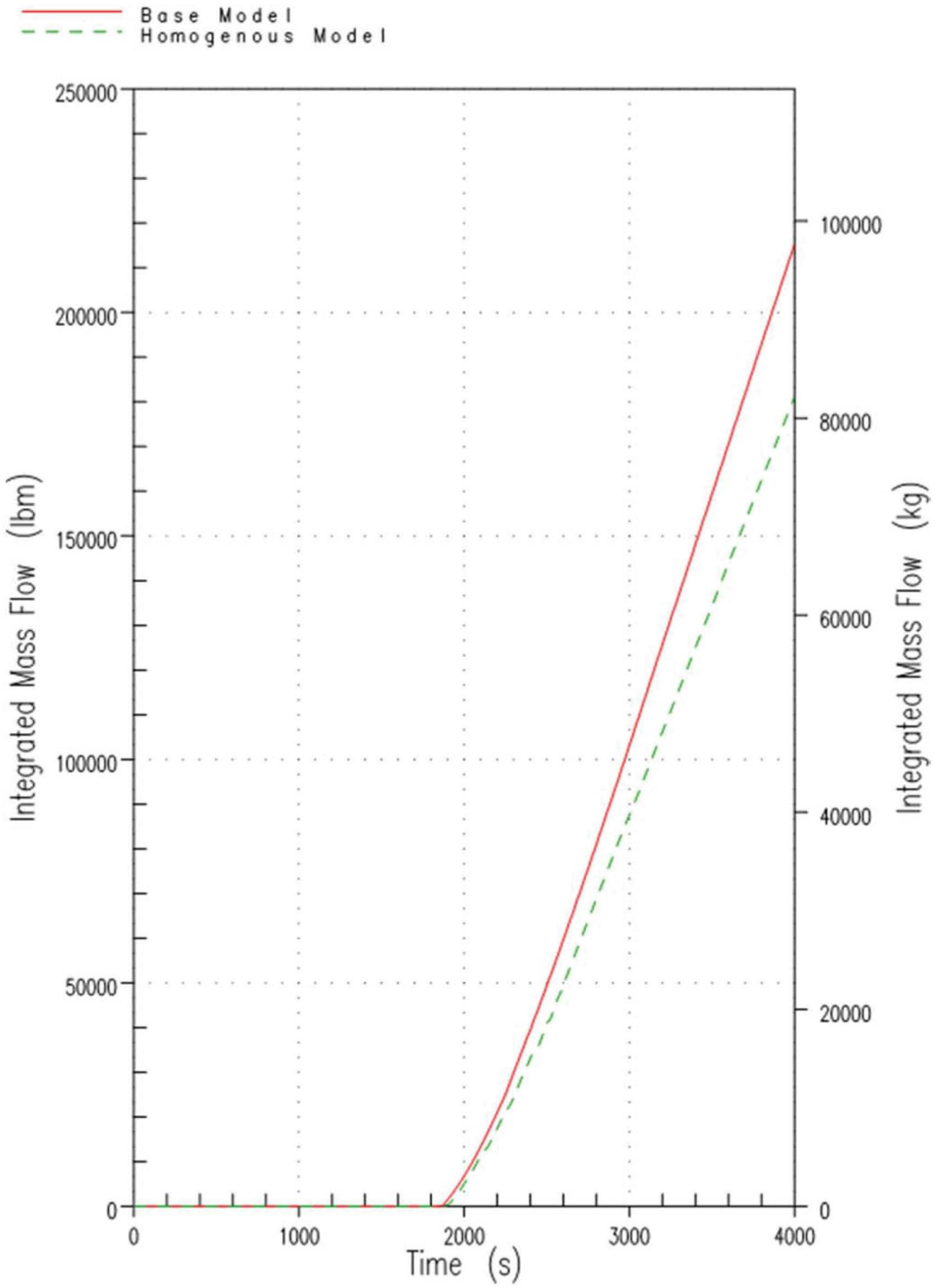

Figure 15.6.5.4B-80
DEDVI Entrainment – Intact IRWST Integrated Injection Flow

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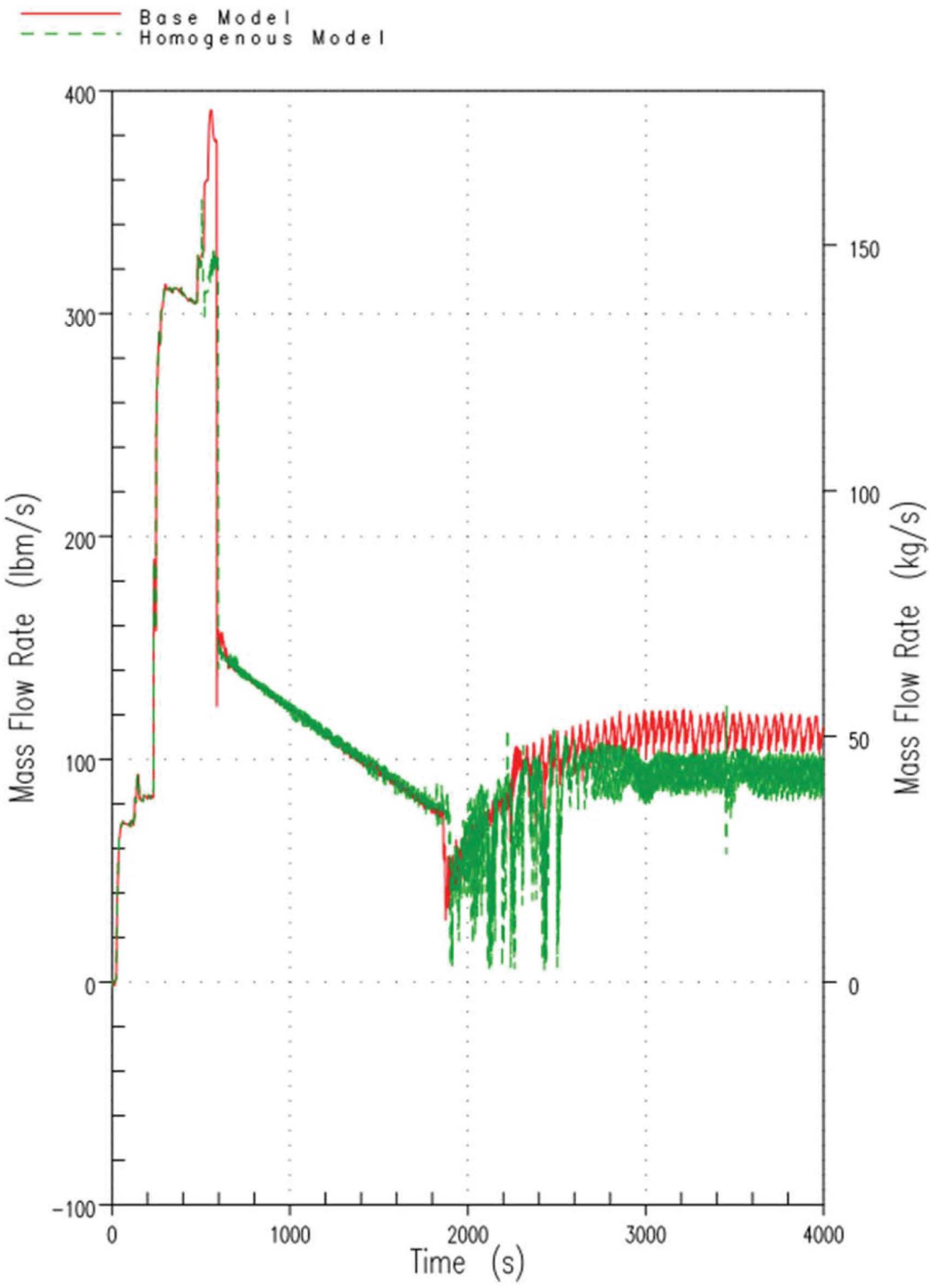

Figure 15.6.5.4B-81
DEDVI Entrainment – Intact DVI Line Injection Flow

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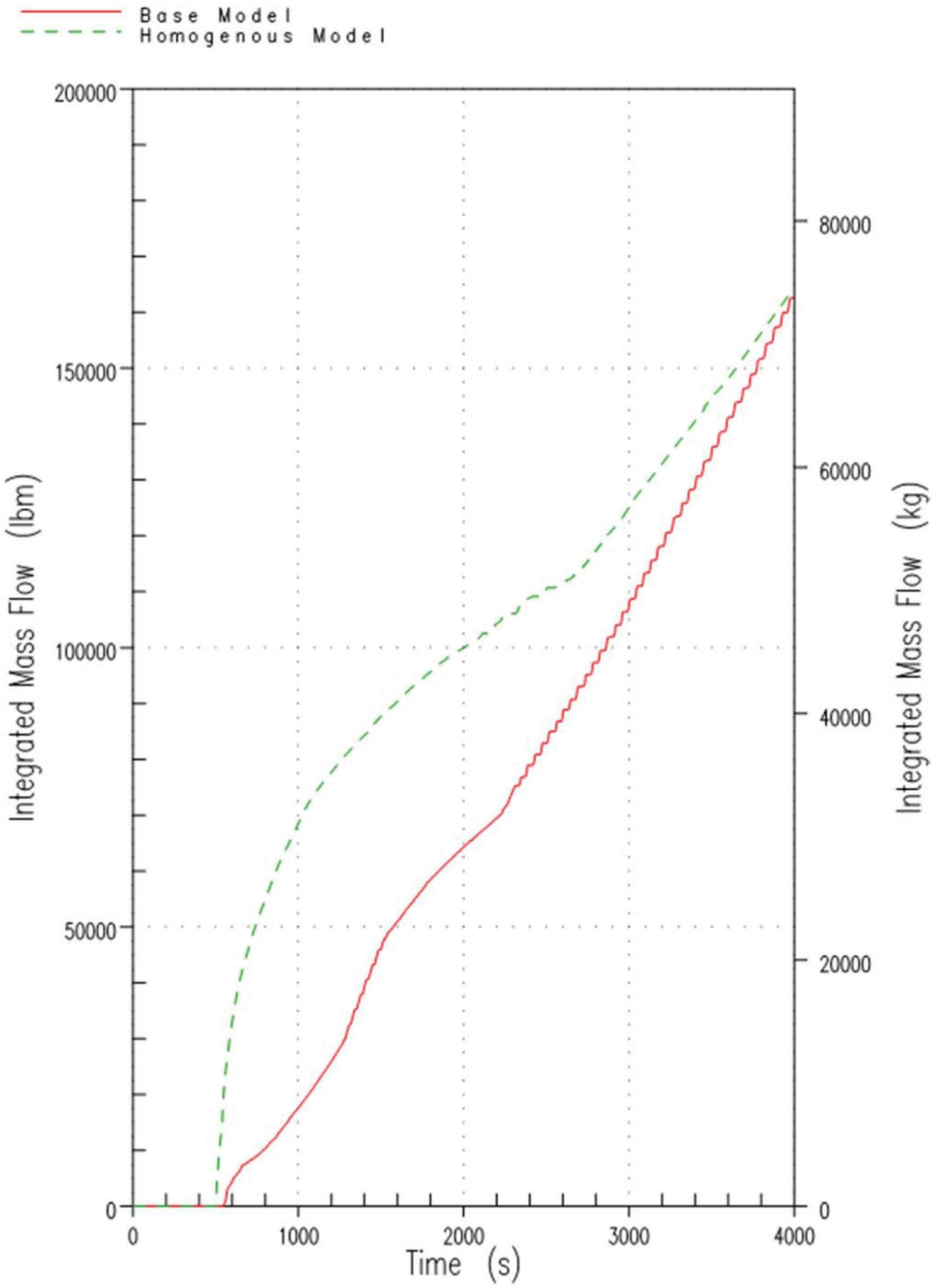

Figure 15.6.5.4B-82
DEDVI Entrainment – ADS-4 Integrated Liquid Discharge Comparison

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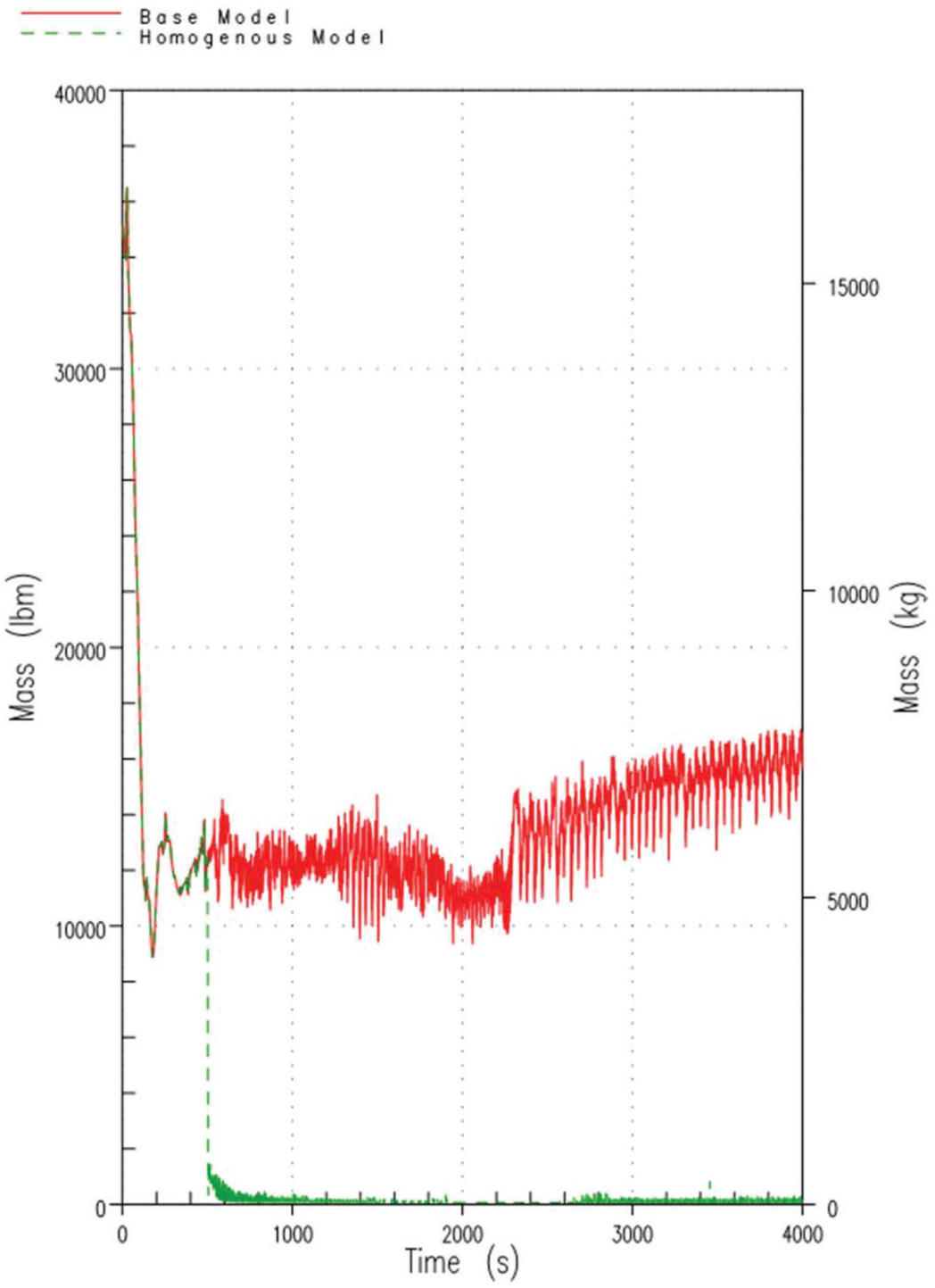

Figure 15.6.5.4B-83
DEDVI Entrainment – Upper Plenum Mixture Mass Comparison

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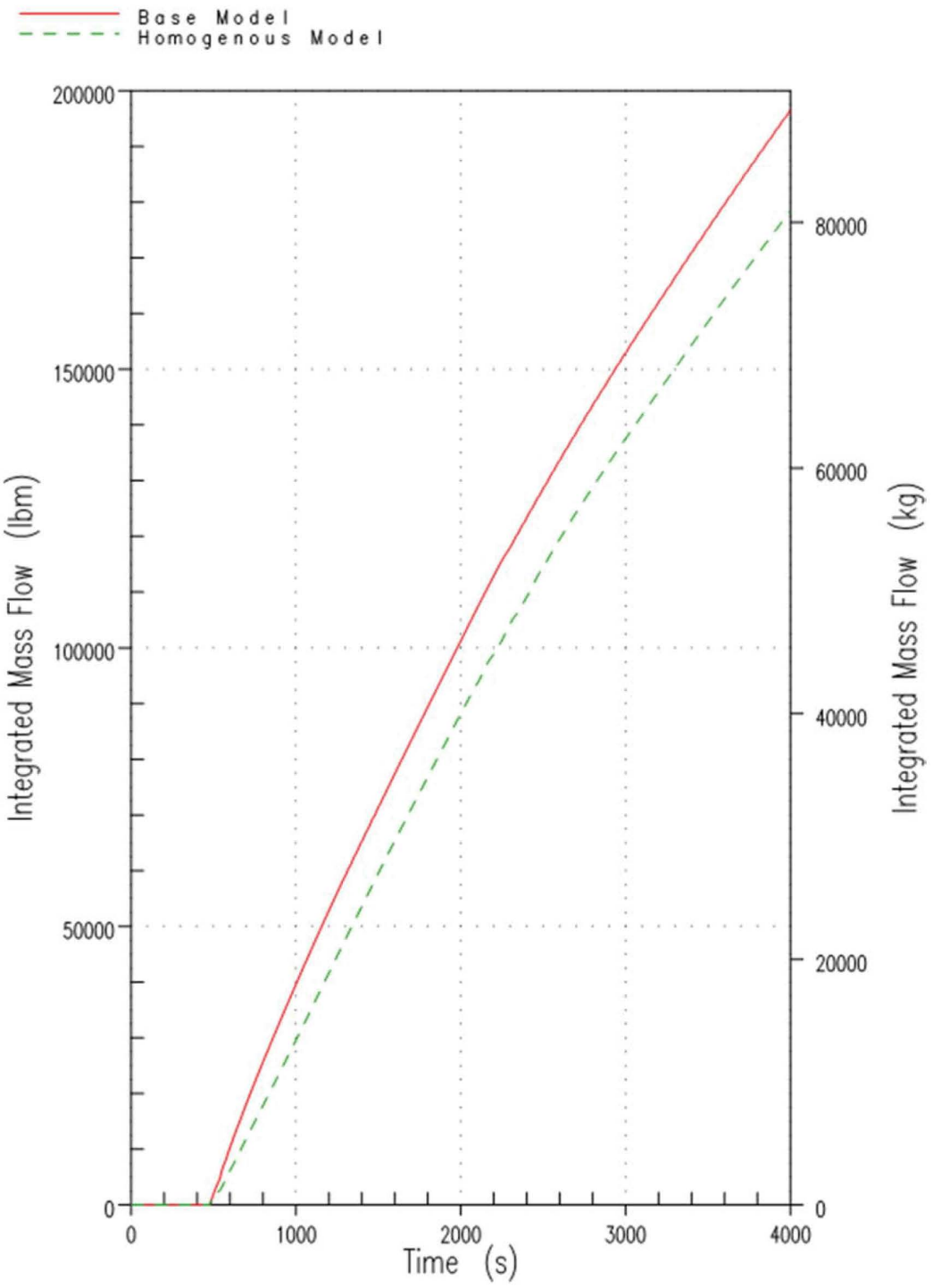

Figure 15.6.5.4B-84
DEDVI Entrainment – ADS-4 Integrated Vapor Discharge Comparison

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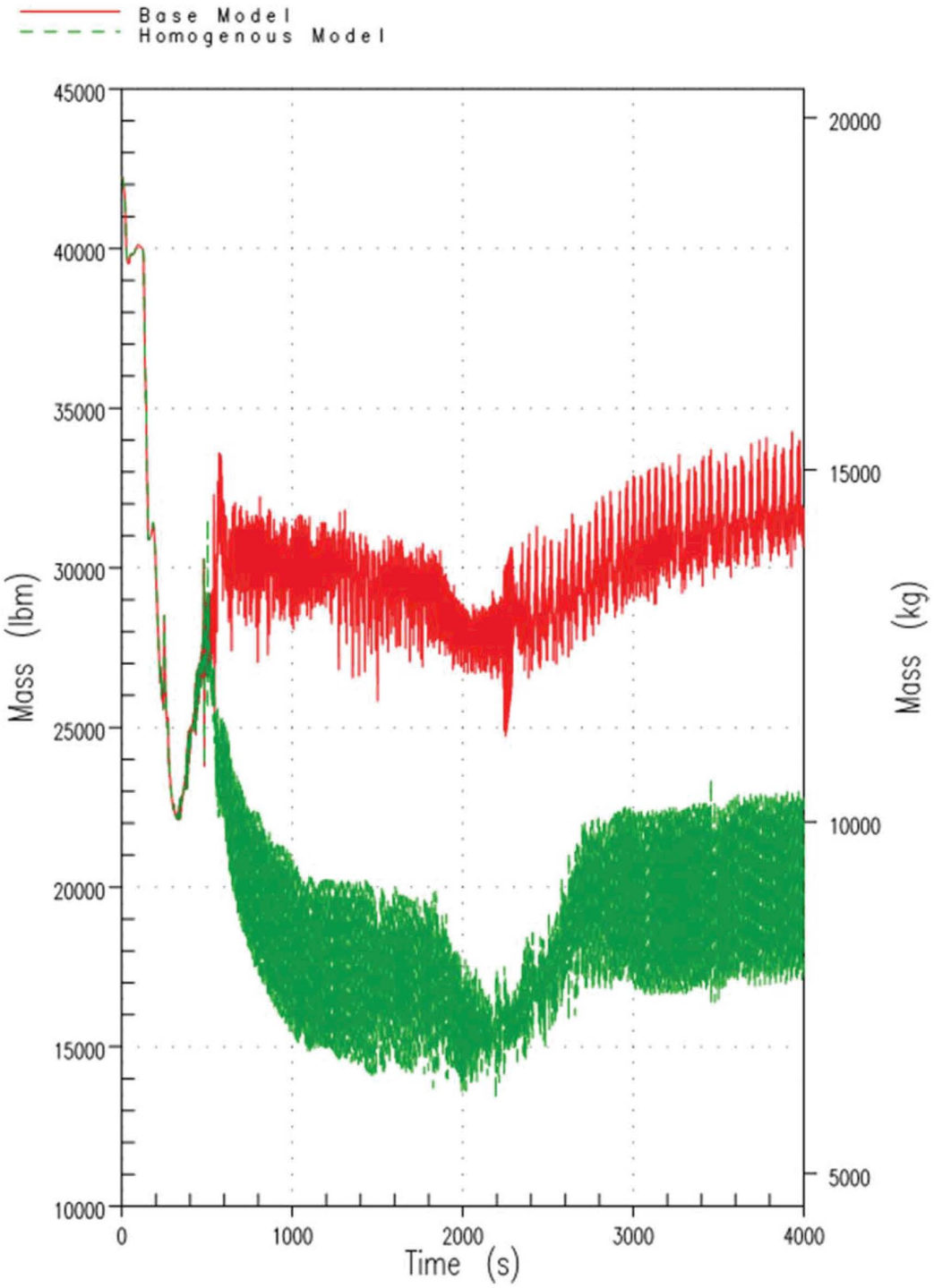

Figure 15.6.5.4B-85
DEDVI Entrainment – Downcomer Region Mass Comparison

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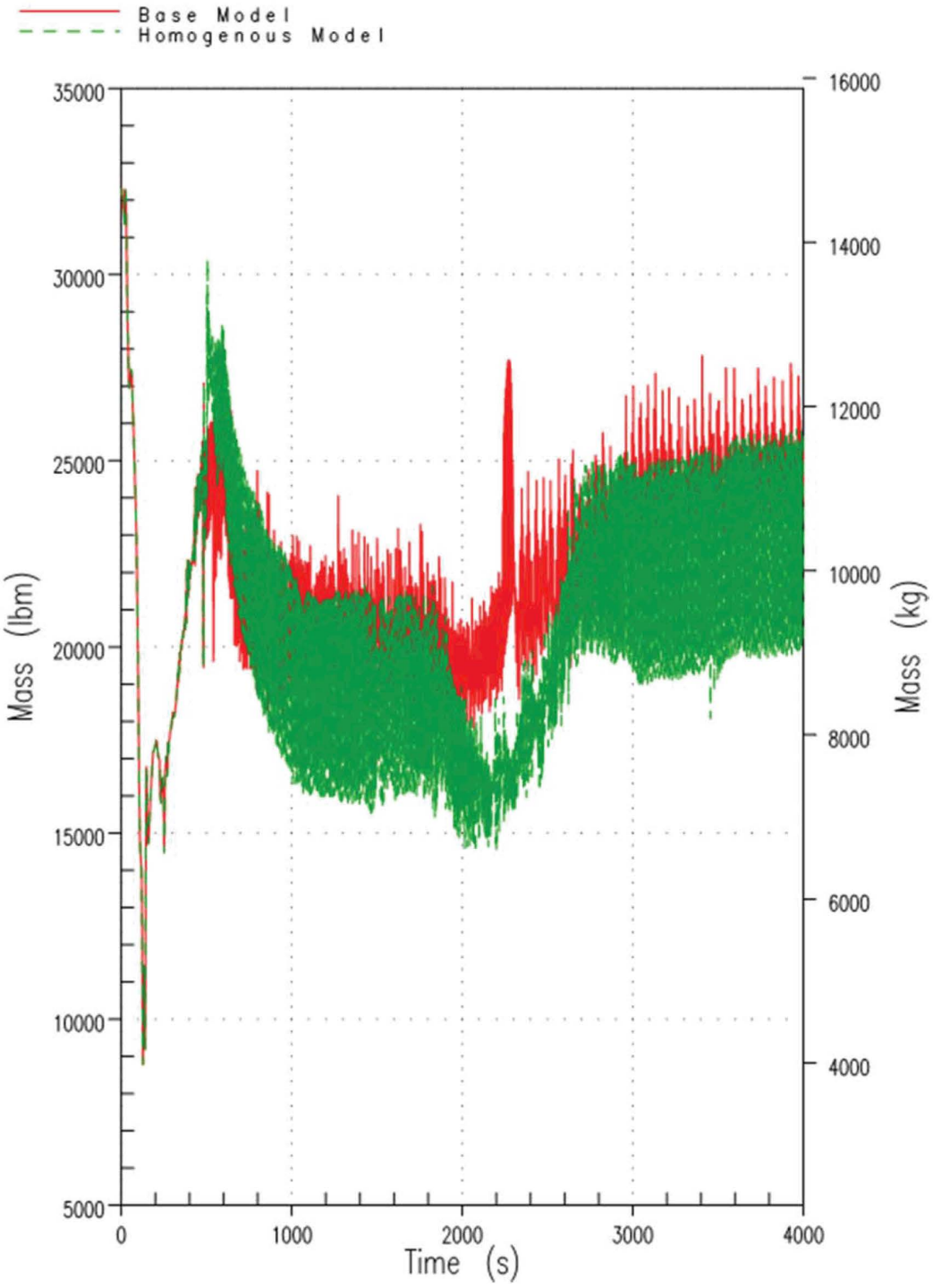

Figure 15.6.5.4B-86(a)
DEDVI Entrainment – Core Region Mass Comparison

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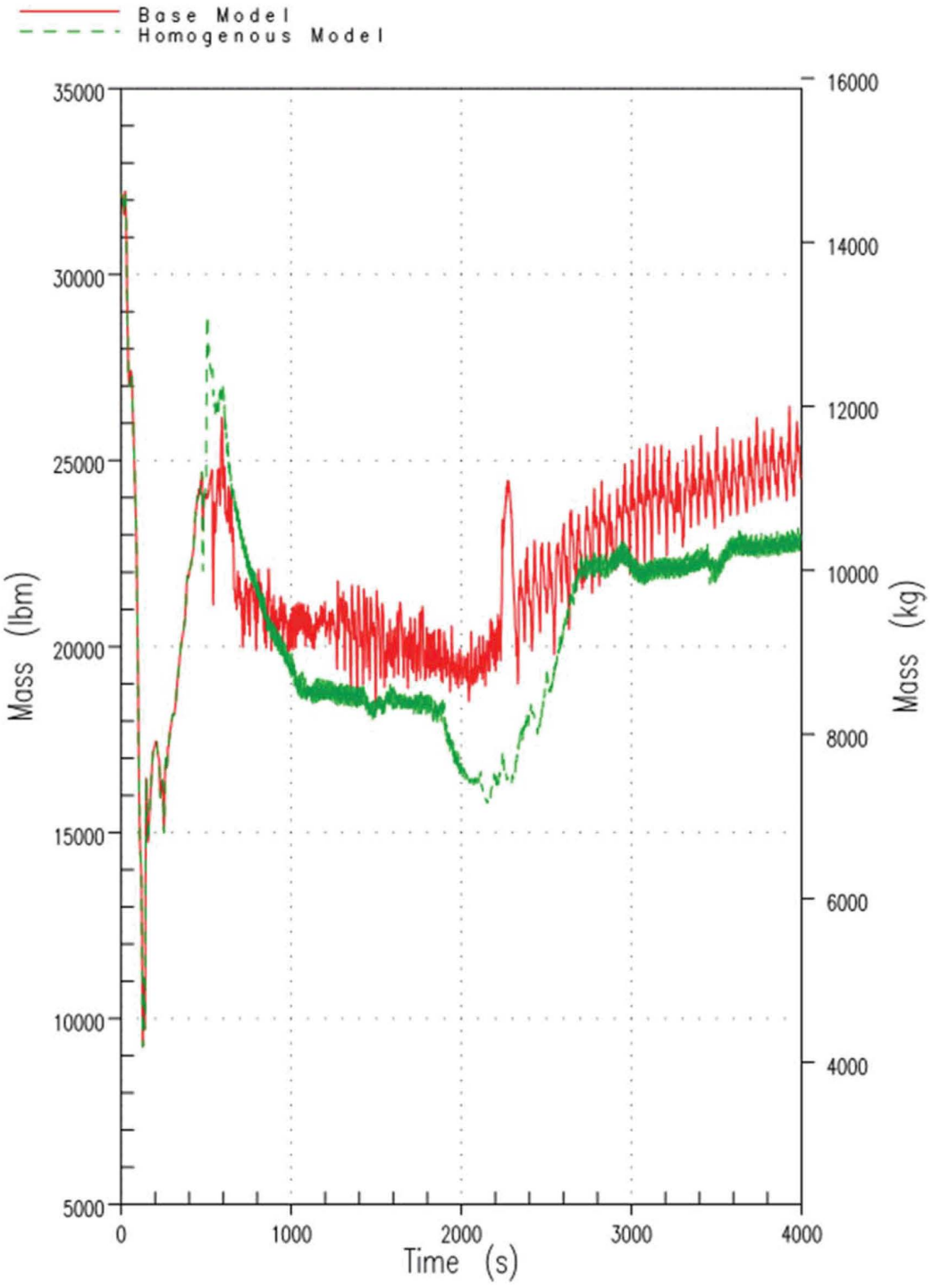

Figure 15.6.5.4B-86(b)
DEDVI Entrainment – Core Region Mass Comparison (Smoothed)

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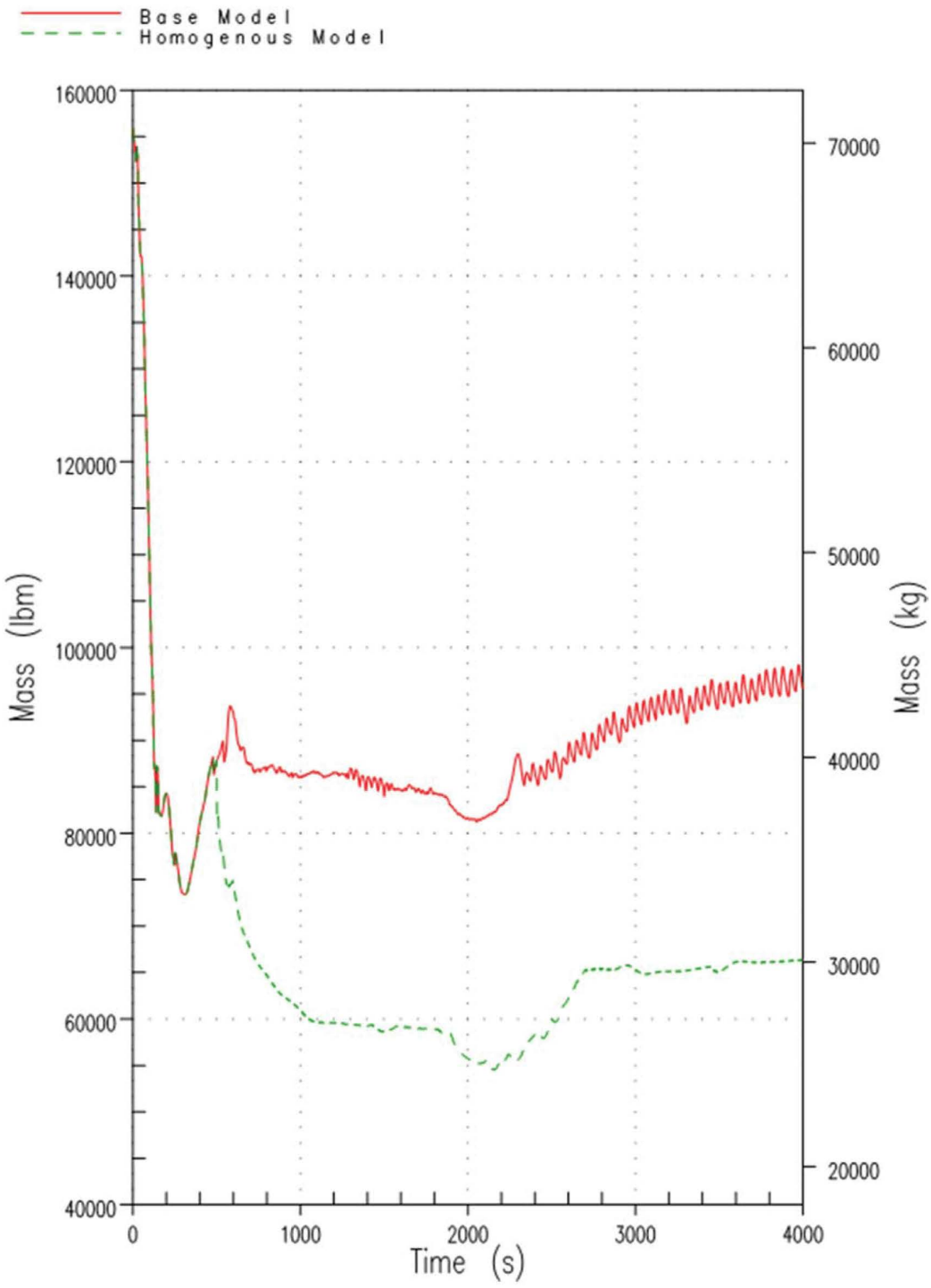

Figure 15.6.5.4B-87
DEDVI Entrainment – Vessel Mixture Mass Comparison

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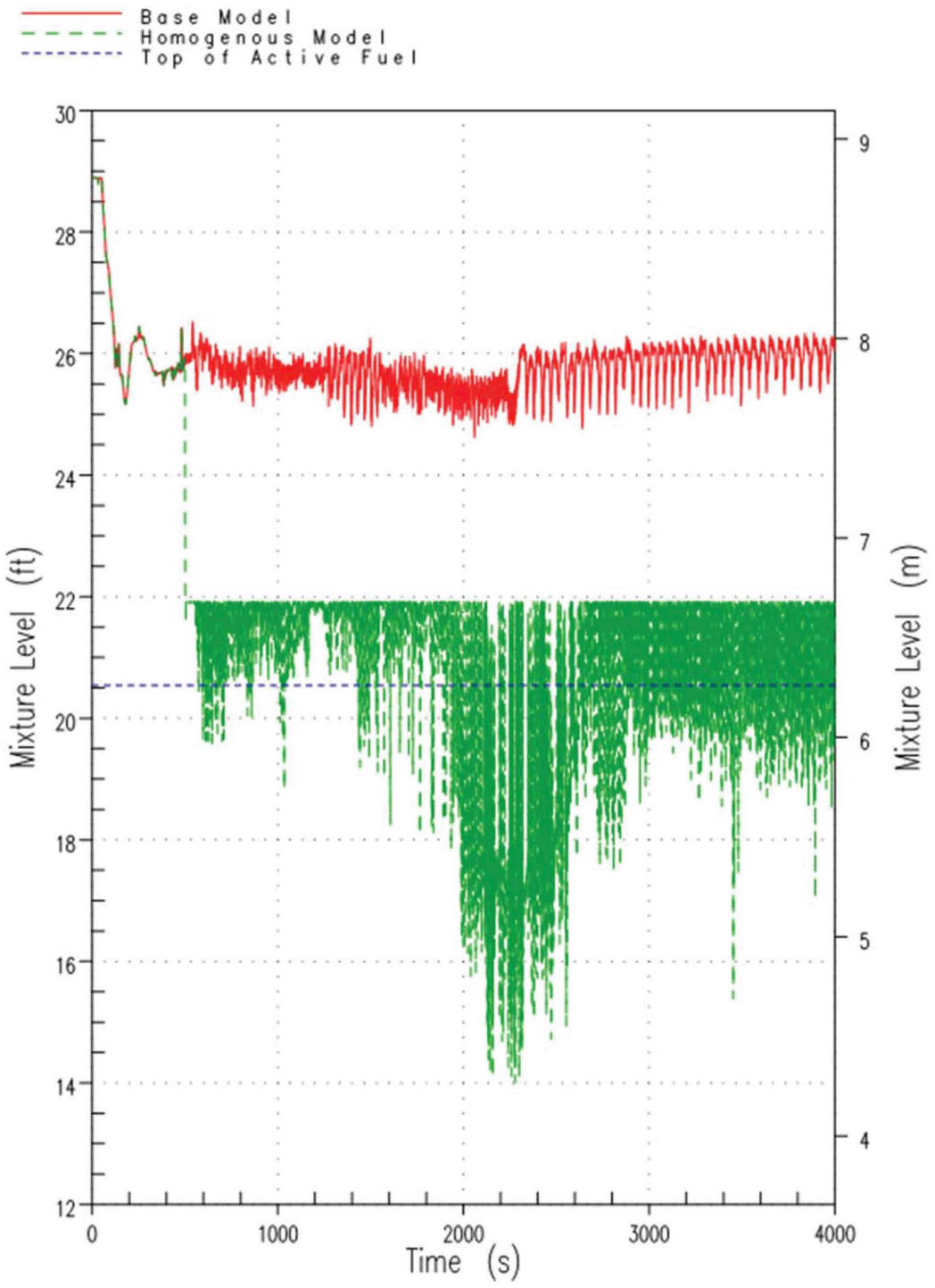

Figure 15.6.5.4B-88(a)
DEDVI Entrainment – Core/Upper Plenum Mixture Level Comparison

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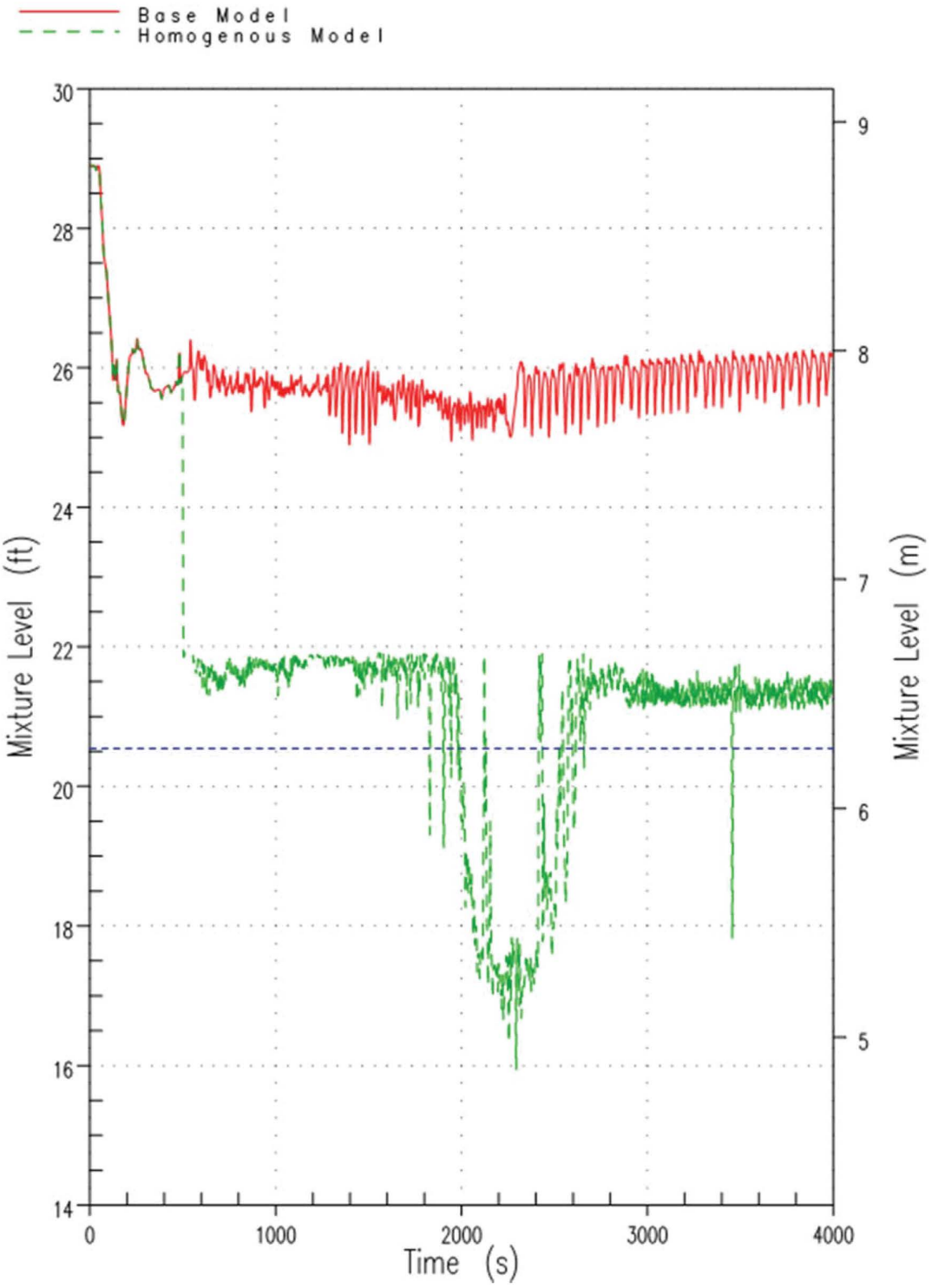

Figure 15.6.5.4B-88(b)
DEDVI Entrainment – Core/Upper Plenum Mixture Level Comparison (Smoothed)

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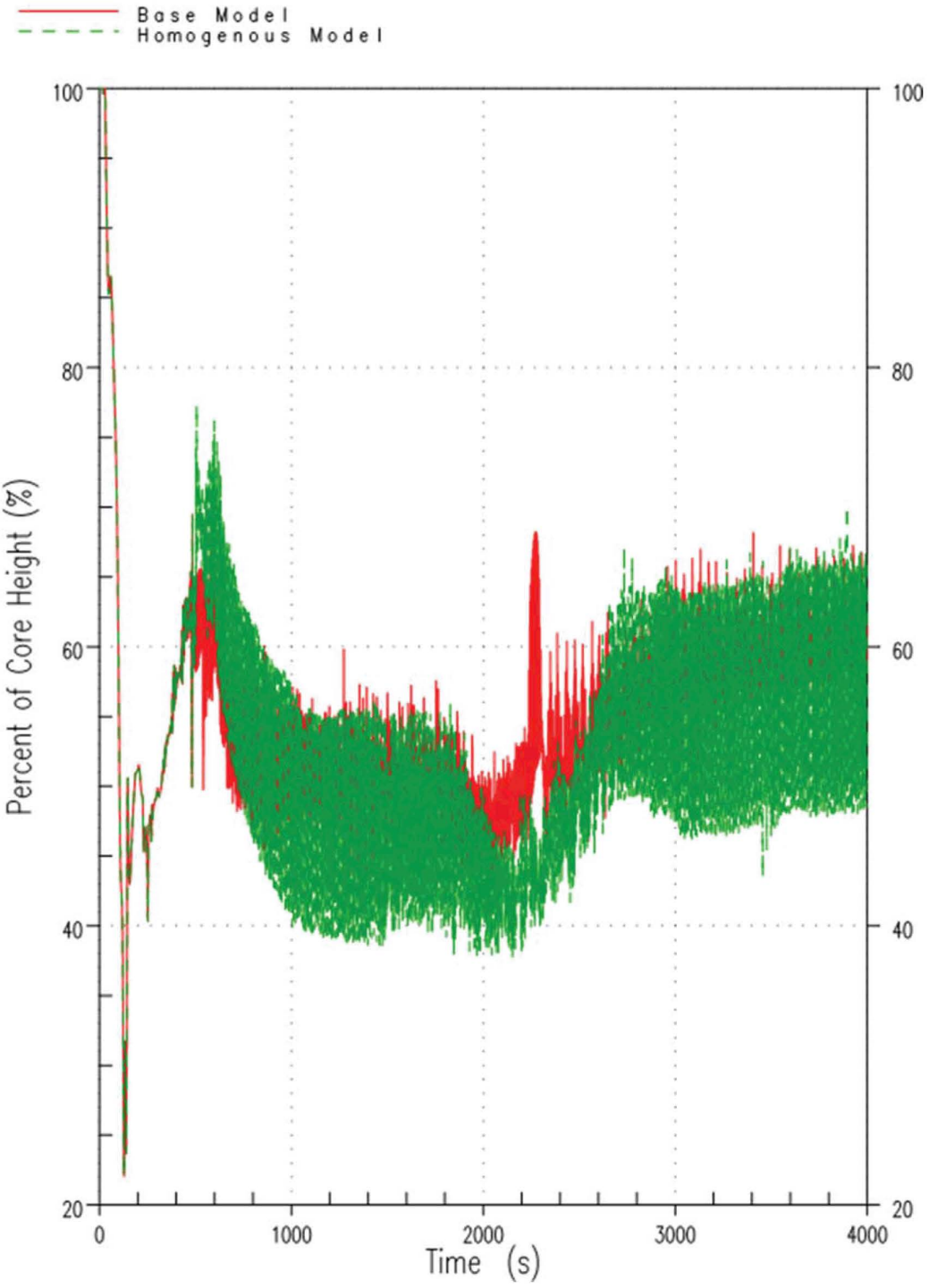

Figure 15.6.5.4B-89(a)
DEDVI Entrainment – Core Collapsed Liquid Level Comparison

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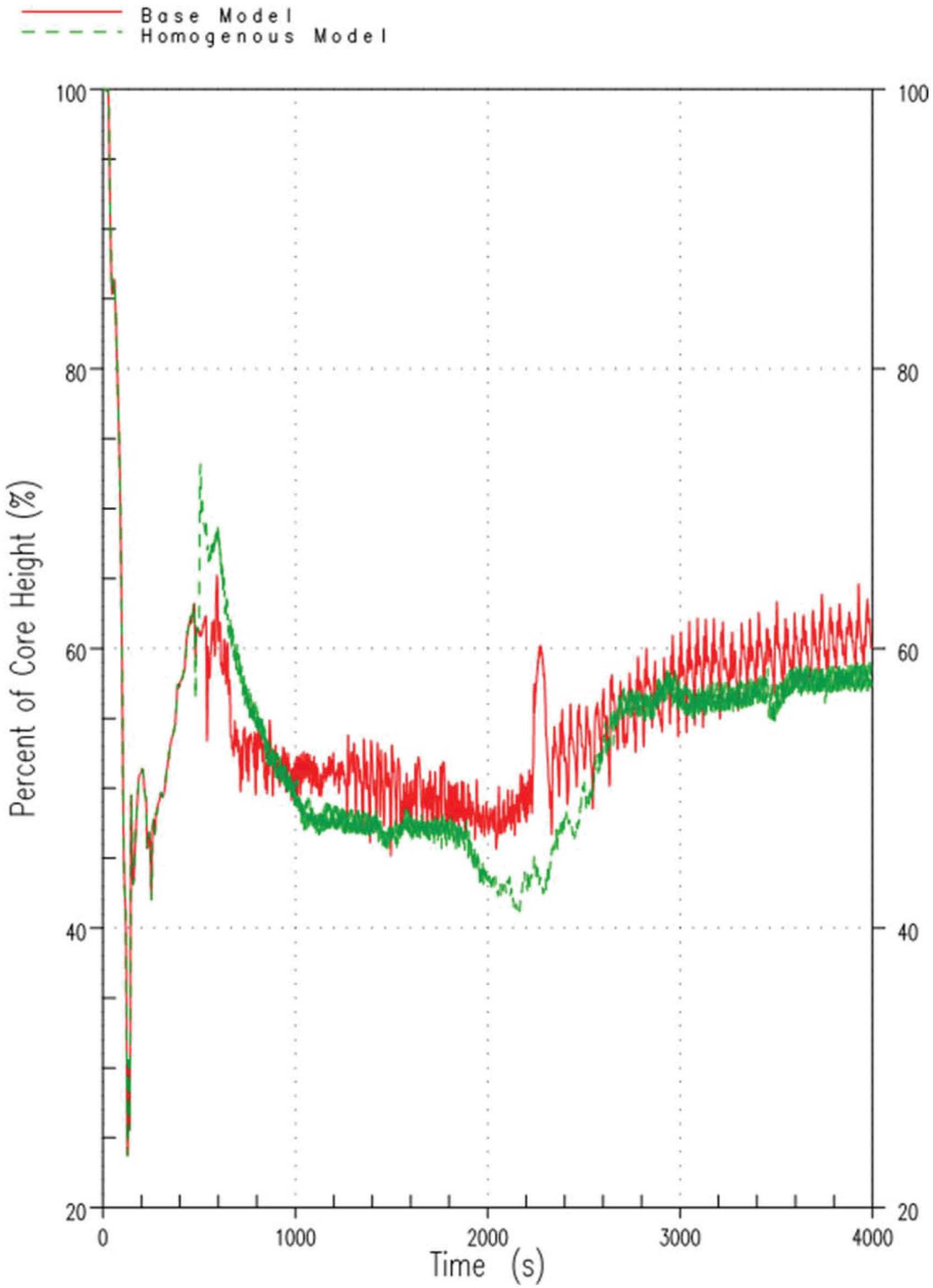

Figure 15.6.5.4B-89(b)
DEDVI Entrainment – Core Collapsed Liquid Level Comparison (Smoothed)

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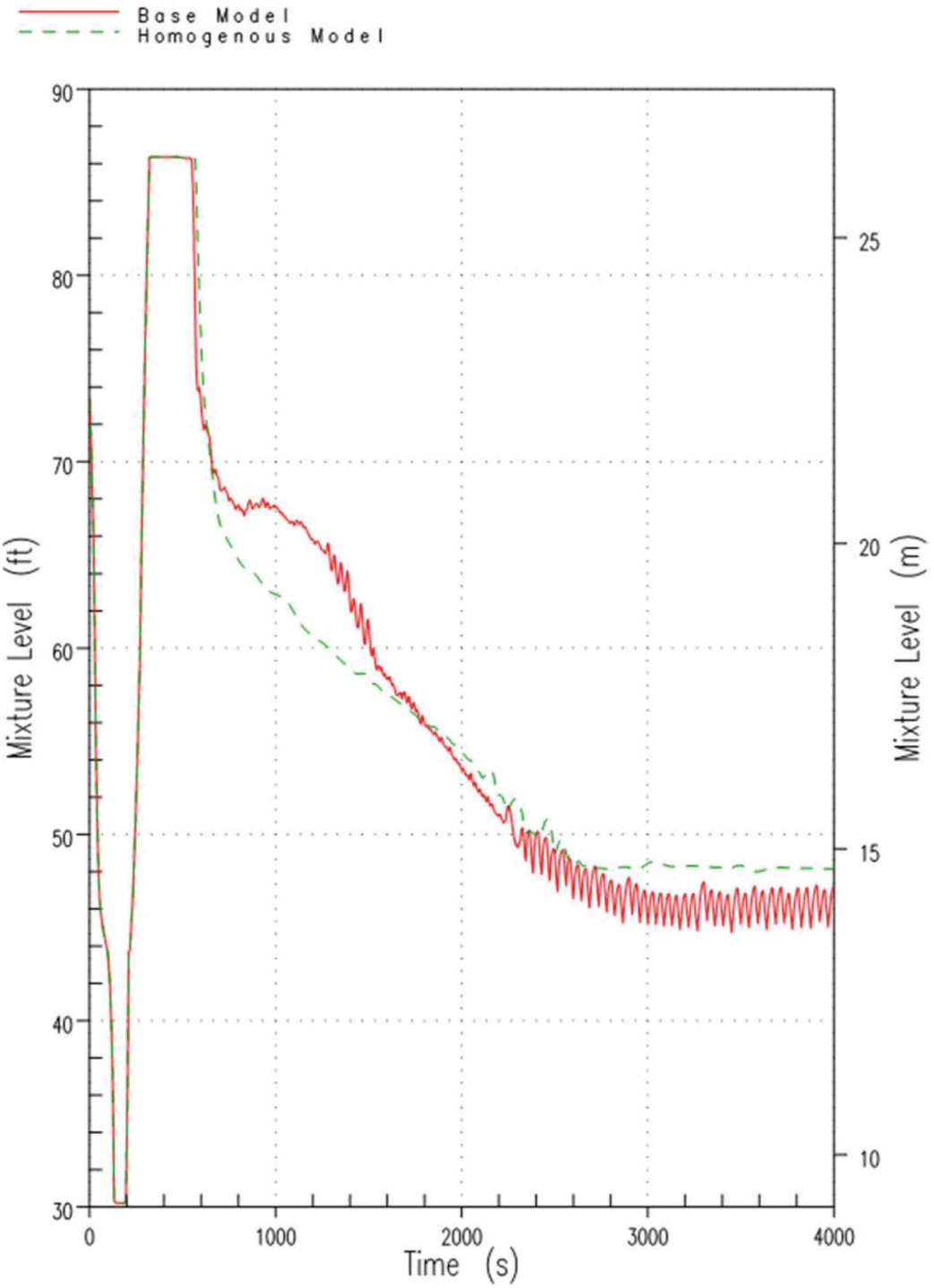

Figure 15.6.5.4B-90
DEDVI Entrainment – Pressurizer Mixture Level Comparison

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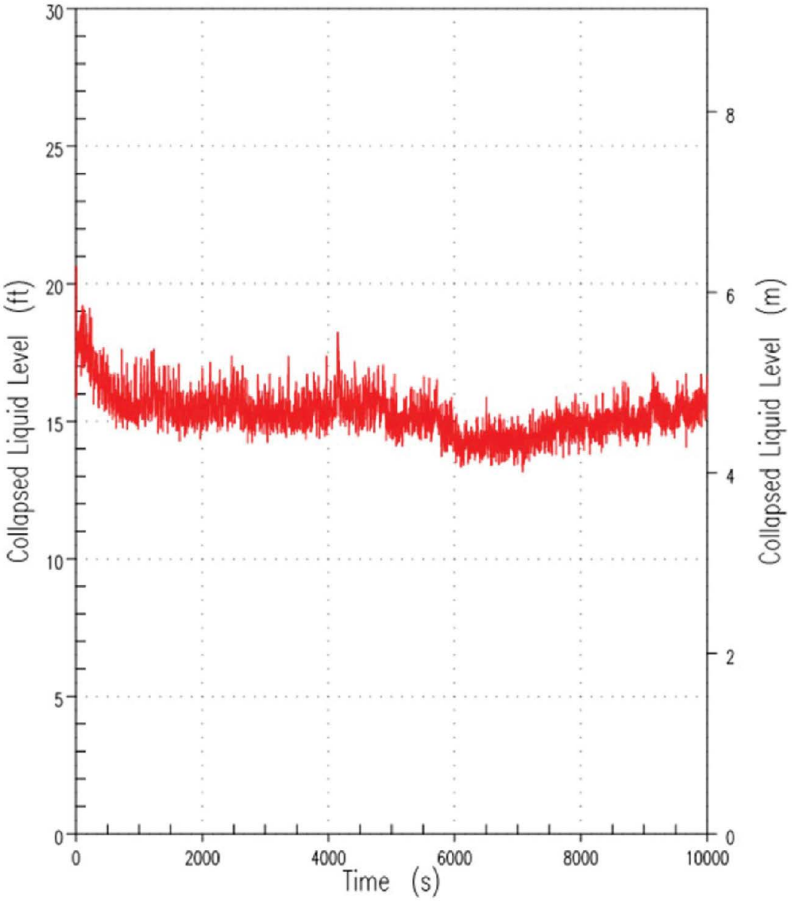

Figure 15.6.5.4C-1
Collapsed Level of Liquid in the Downcomer
(DEDVI Case)

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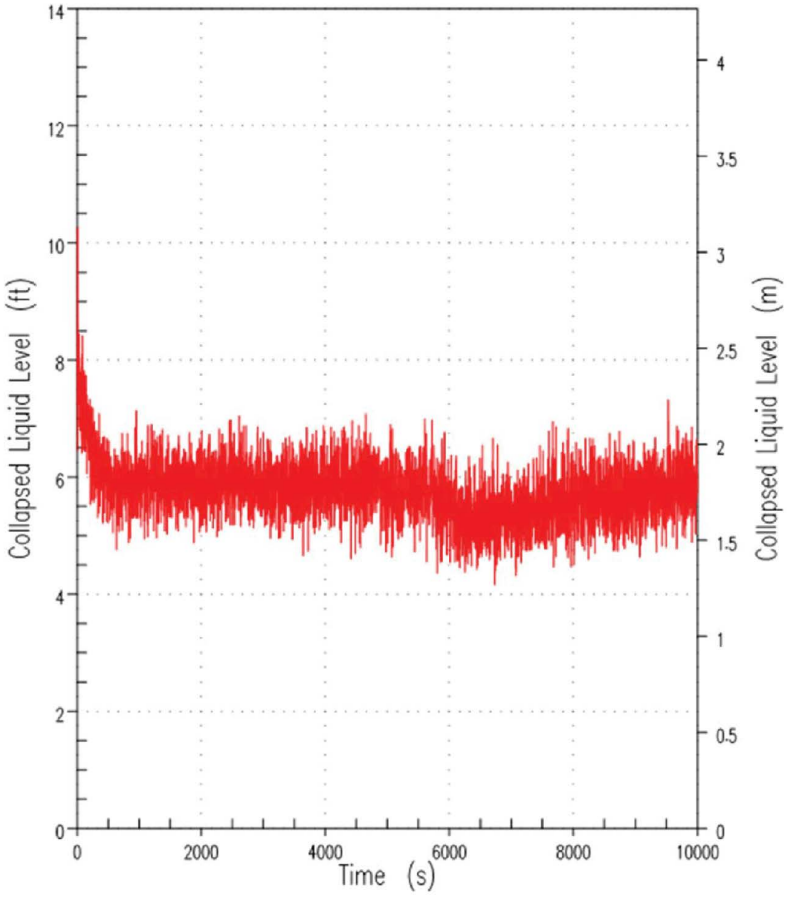

Figure 15.6.5.4C-2
Collapsed Level of Liquid over the Heated Length of the Fuel (DEDVI Case)

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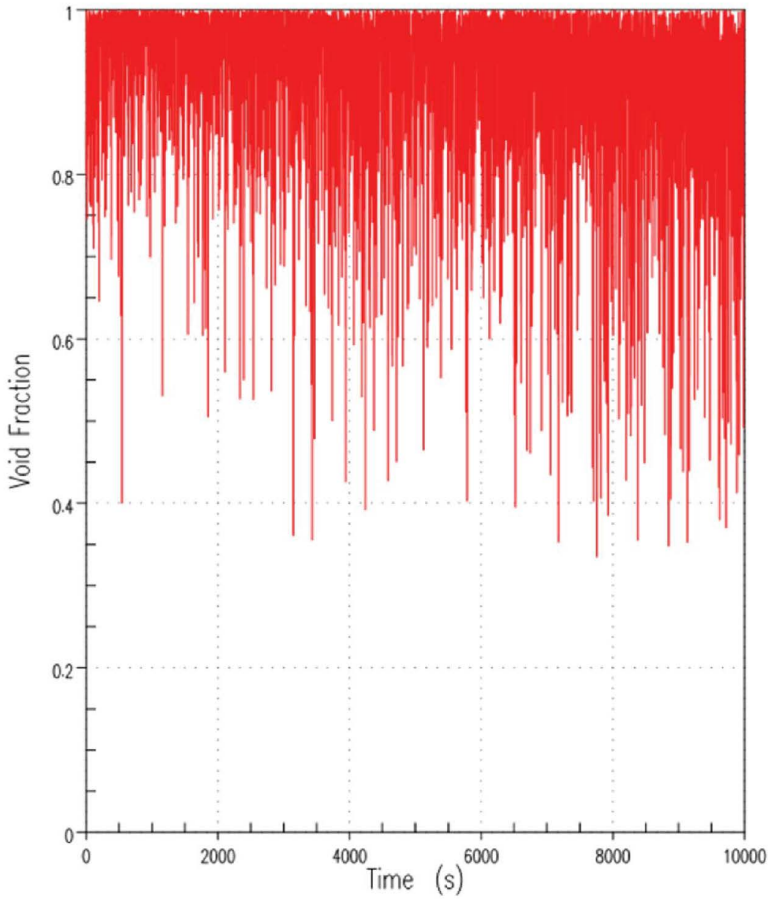

Figure 15.6.5.4C-3
Void Fraction in Core Hot Assembly Top Cell
(DEDVI Case)

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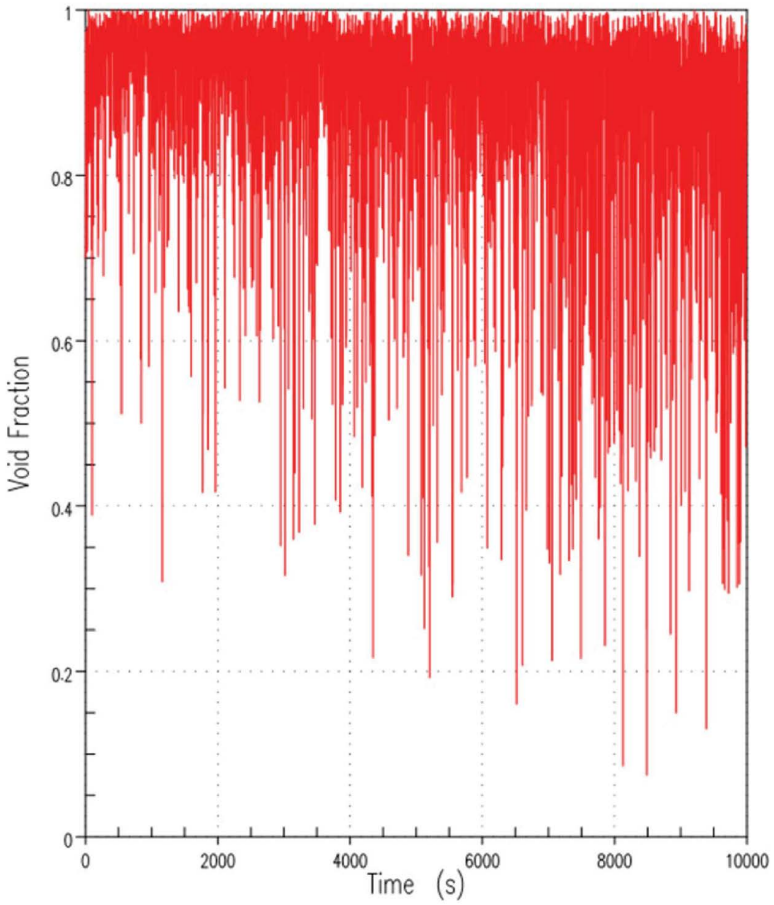

Figure 15.6.5.4C-4
Void Fraction in Core Hot Assembly Second from Top Cell
(DEDVI Case)

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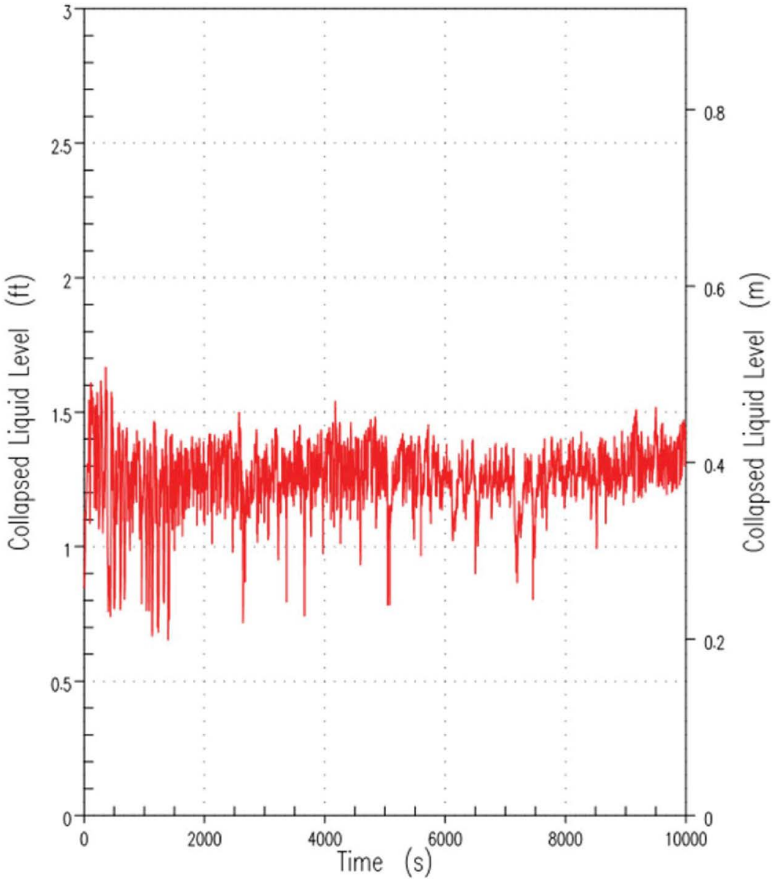

Figure 15.6.5.4C-5
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (DEDVI Case)

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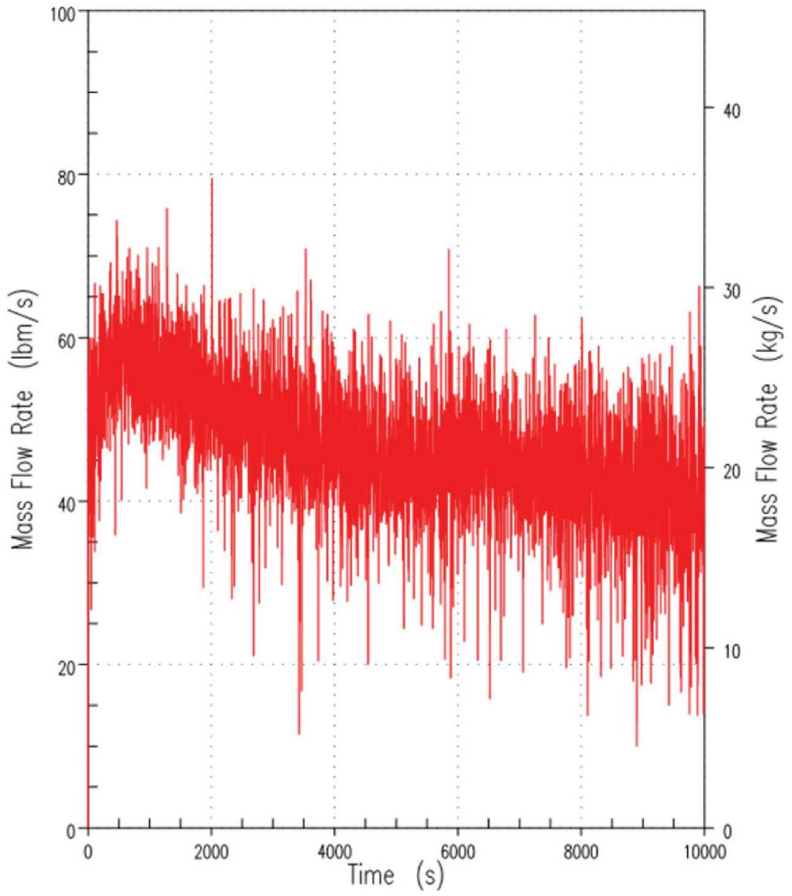

Figure 15.6.5.4C-6
Vapor Rate out of the Core
(DEDVI Case)

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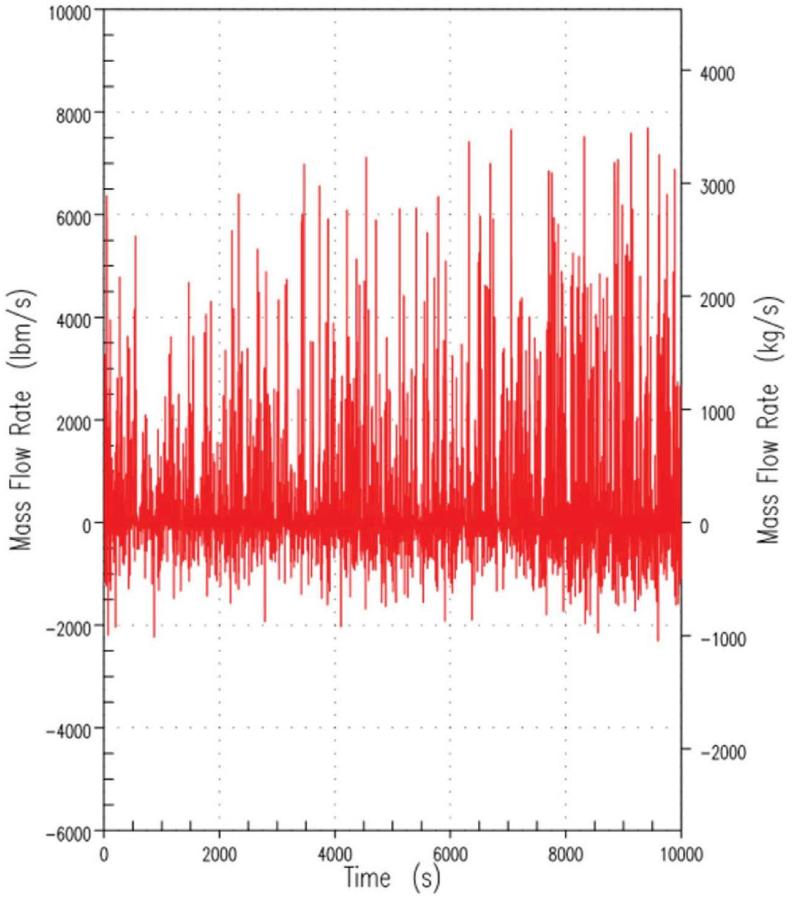

Figure 15.6.5.4C-7
Liquid Flow Rate out of the Core
(DEDVI Case)

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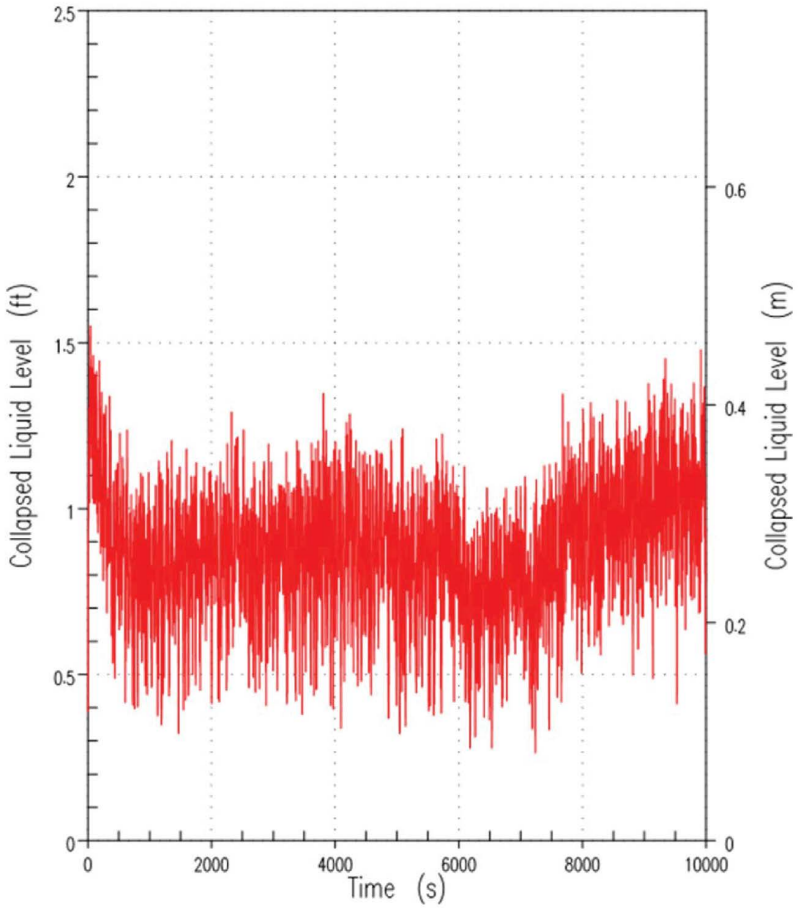

Figure 15.6.5.4C-8
Collapsed Liquid Level in the Upper Plenum
(DEDVI Case)

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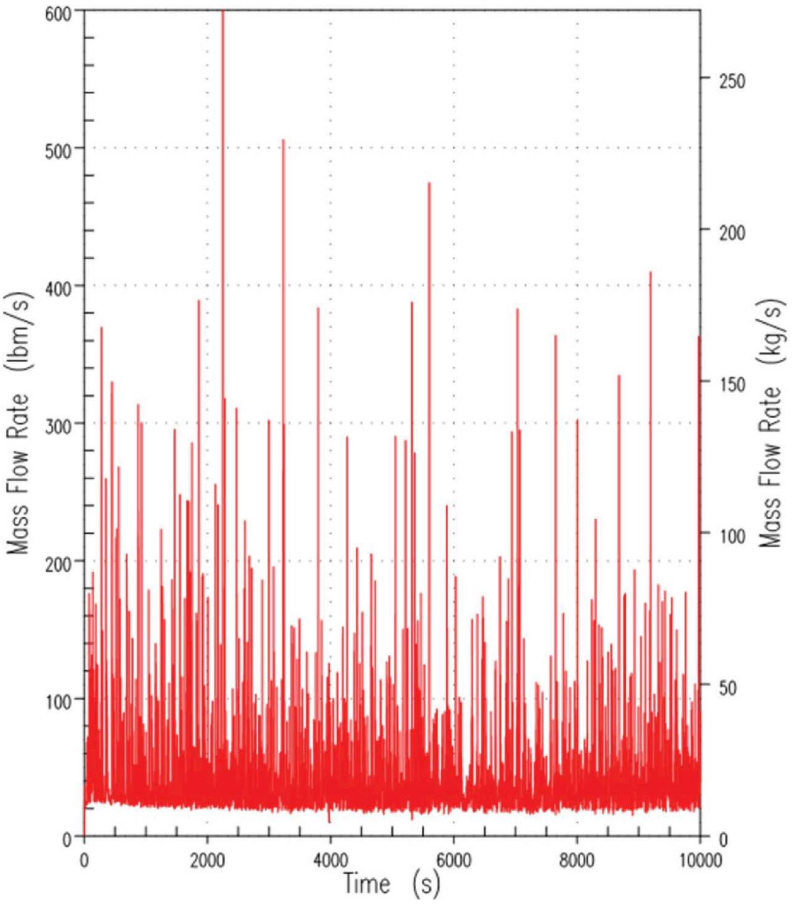

Figure 15.6.5.4C-9
Mixture Flow Rate Through ADS Stage 4A Valves
(DEDVI Case)

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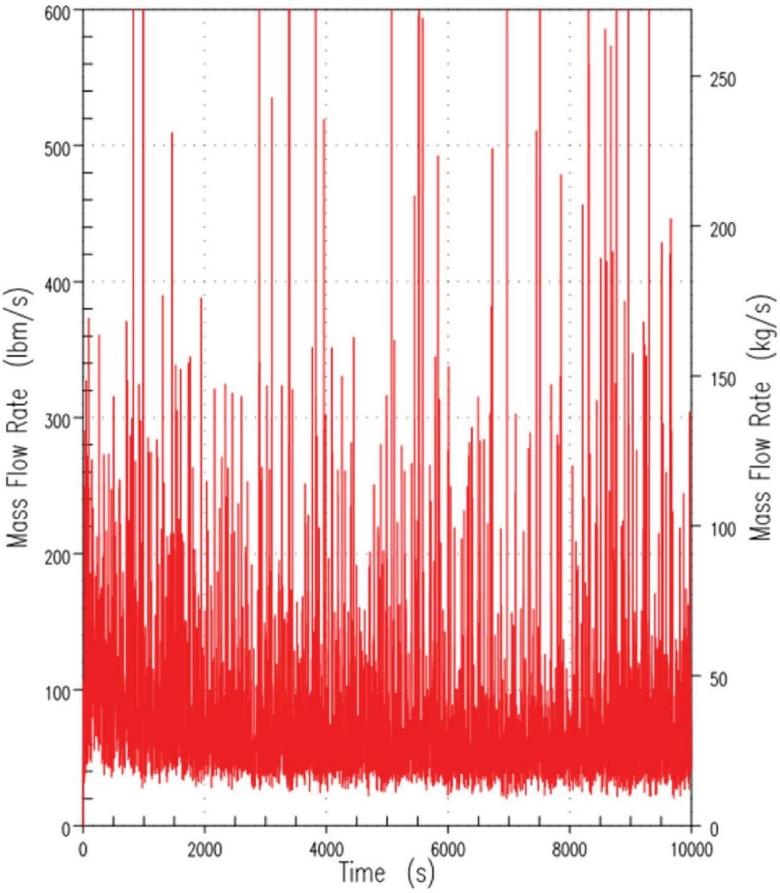

Figure 15.6.5.4C-10
Mixture Flow Rate Through ADS Stage 4B Valves
(DEDVI Case)

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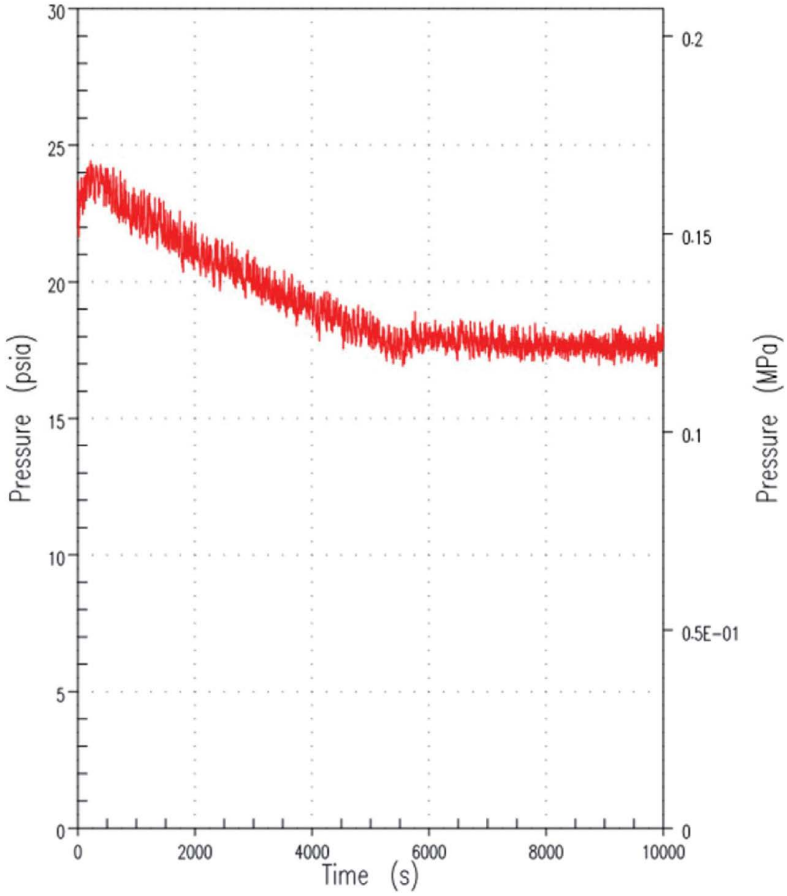

Figure 15.6.5.4C-11
Upper Plenum Pressure
(DEDVI Case)

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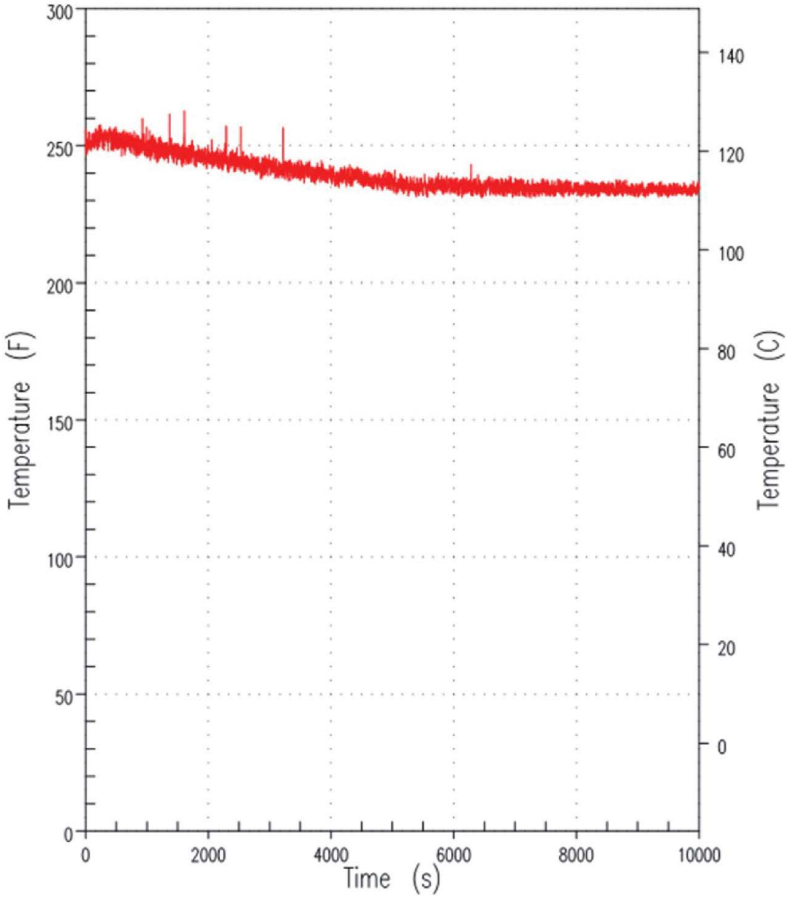

Figure 15.6.5.4C-12
Peak Cladding Temperature
(DEDVI Case)

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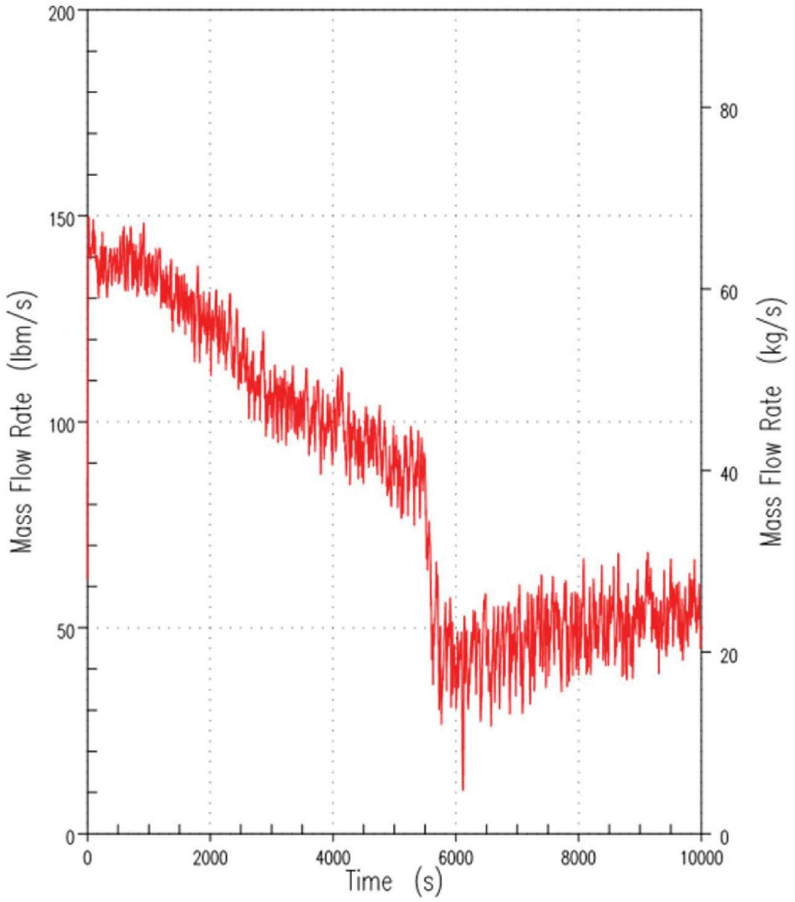

Figure 15.6.5.4C-13
DVI–A Mixture Flow Rate
(DEDVI Case)

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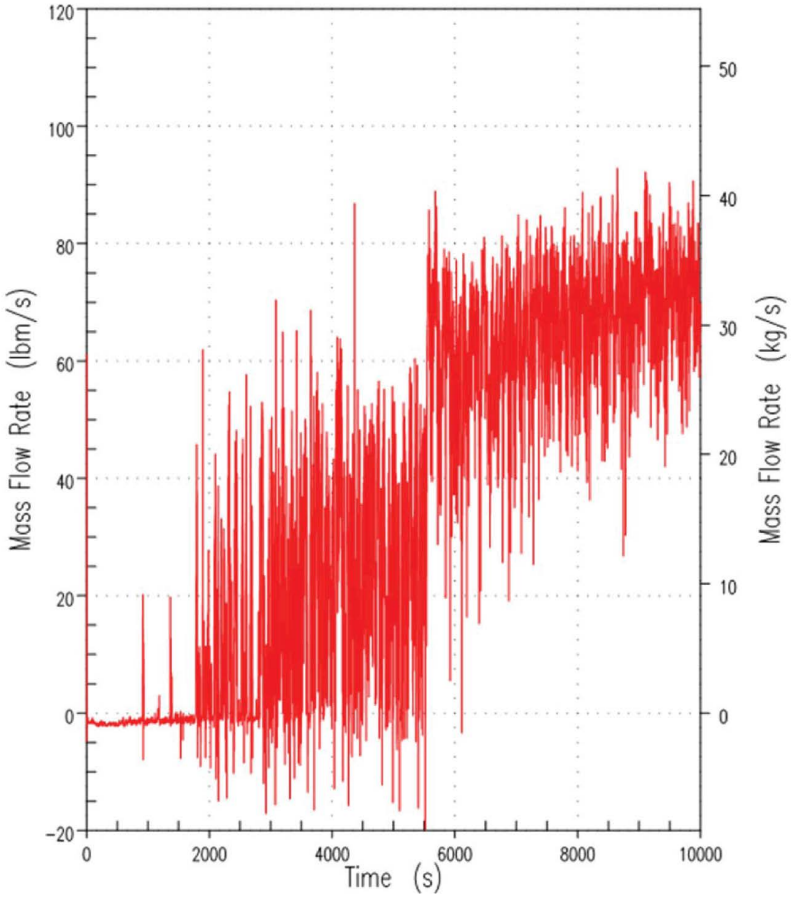

Figure 15.6.5.4C-14
DVI–B Mixture Flow Rate
(DEDVI Case)

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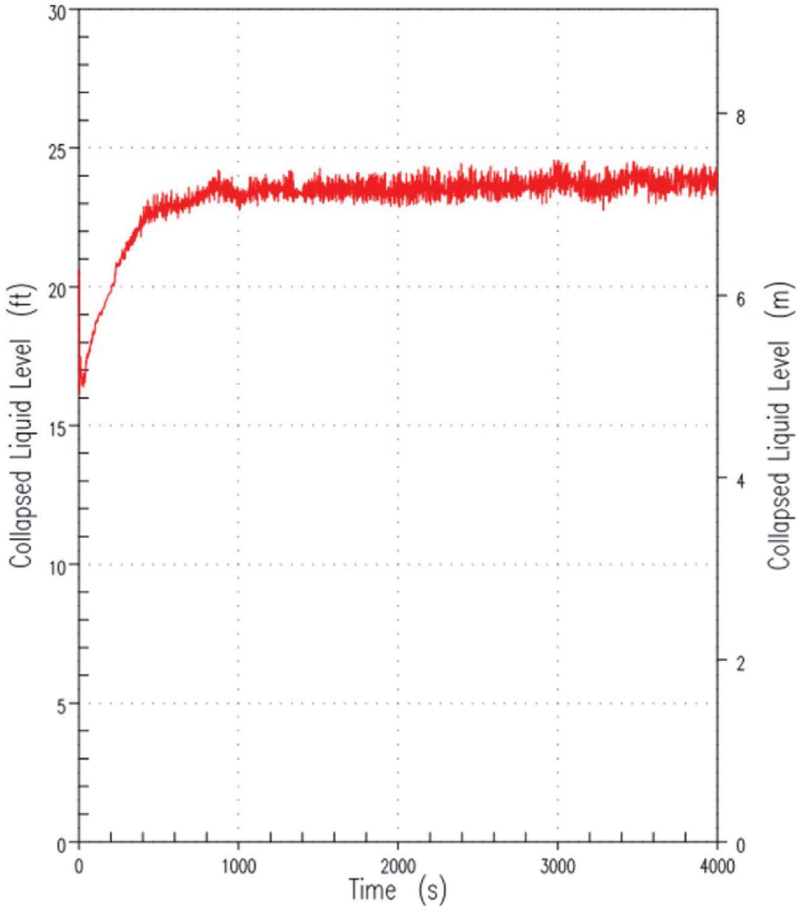

Figure 15.6.5.4C-15
Collapsed Level of Liquid in the Downcomer
(Wall-to-Wall Floodup Case) – 14.7 psia

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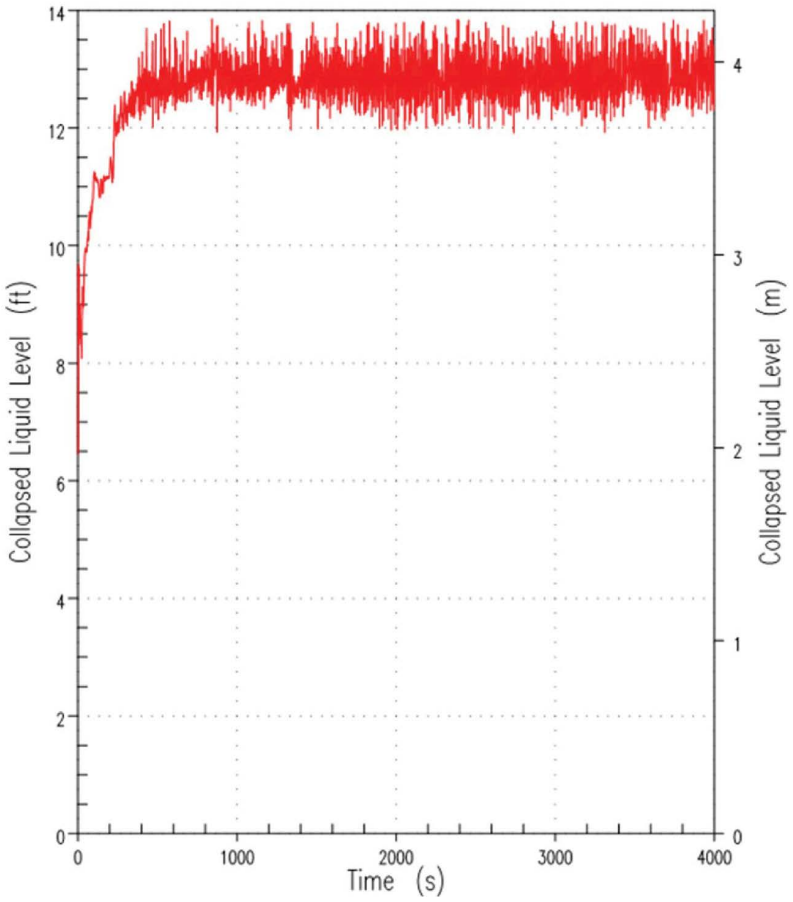

Figure 15.6.5.4C-16
Collapsed Level of Liquid Over the Heated Length of the Fuel (Wall-to-Wall Floodup Case) – 14.7 psia

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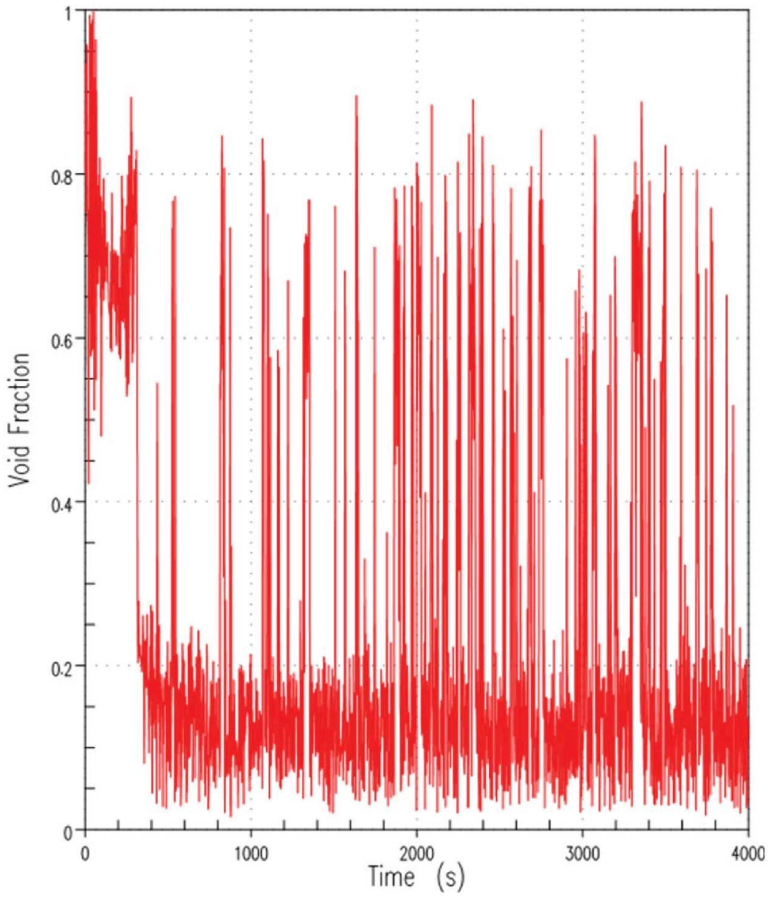

Figure 15.6.5.4C-17
Void Fraction in Core Hot Assembly Top Cell
(Wall-to-Wall Floodup Case) – 14.7 psia

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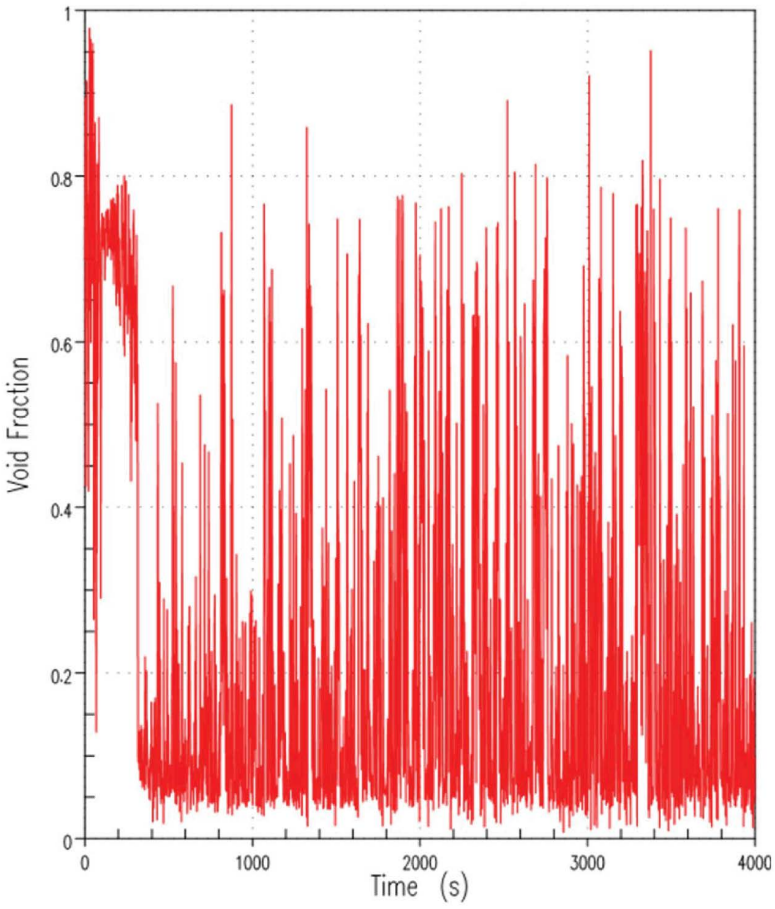

Figure 15.6.5.4C-18
Void Fraction in Core Hot Assembly Second from Top Cell (Wall-to-Wall Floodup Case) – 14.7 psia

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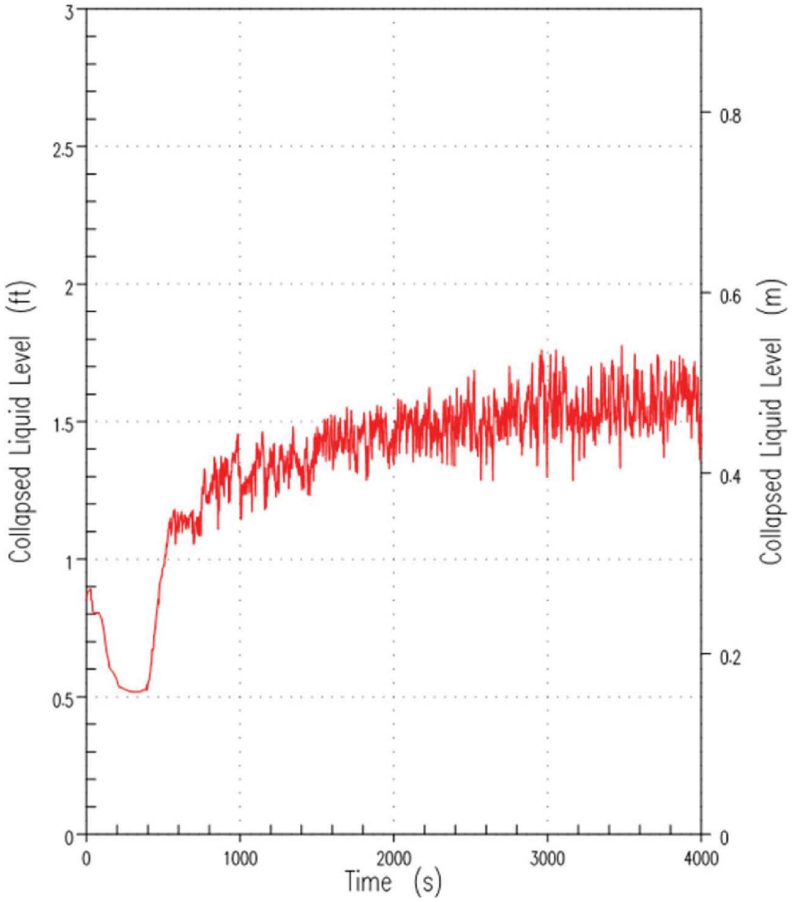

Figure 15.6.5.4C-19
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop
(Wall-to-Wall Floodup Case) – 14.7 psia

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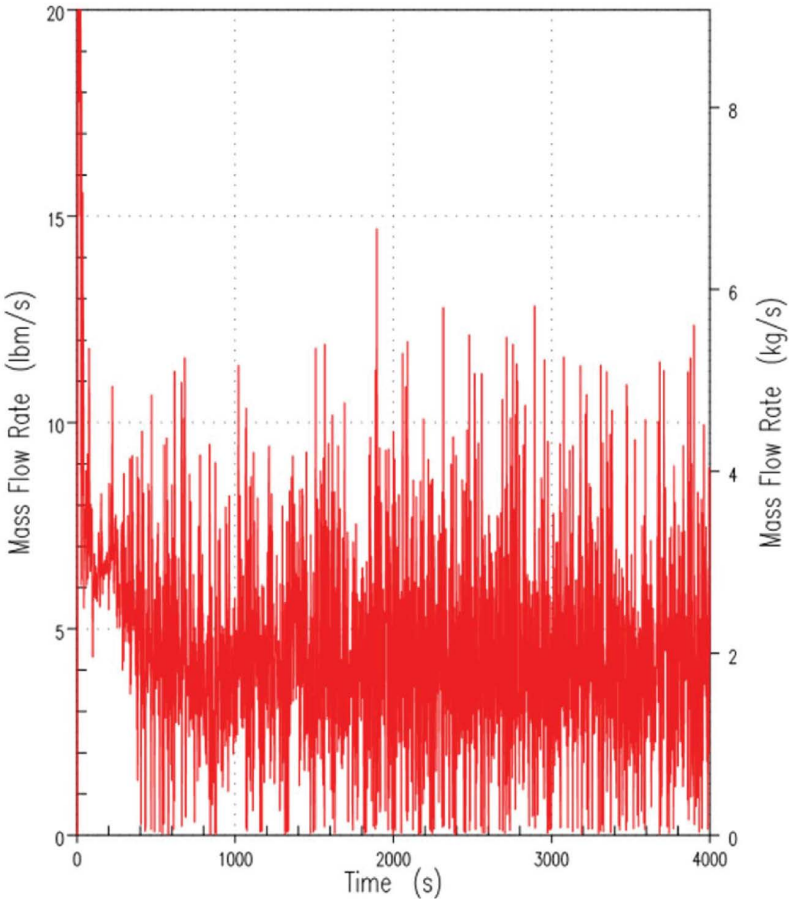

Figure 15.6.5.4C-20
Vapor Rate out of the Core
(Wall-to-Wall Floodup Case) – 14.7 psia

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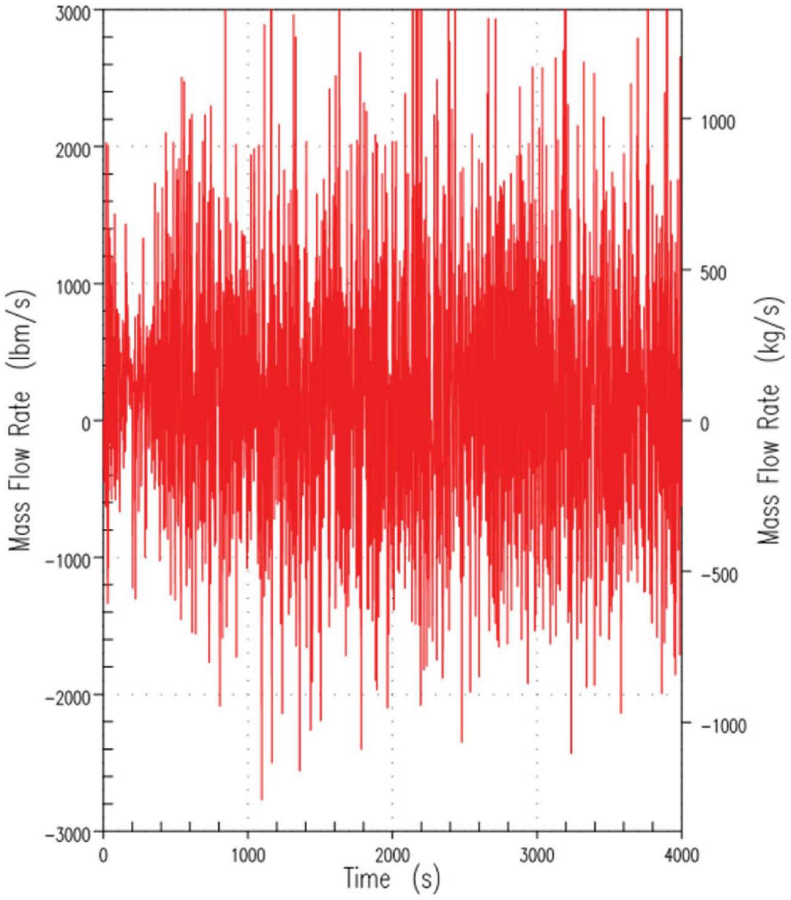

Figure 15.6.5.4C-21
Liquid Flow Rate out of the Core
(Wall-to-Wall Floodup Case) – 14.7 psia

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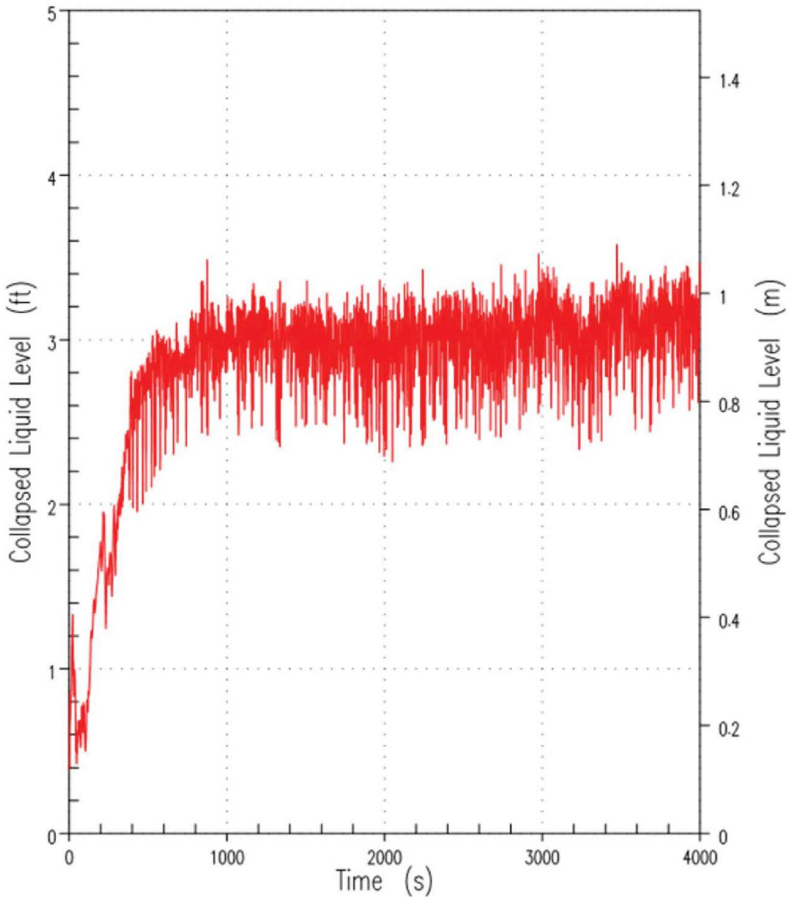

Figure 15.6.5.4C-22
Collapsed Liquid Level in the Upper Plenum
(Wall-to-Wall Floodup Case) – 14.7 psia

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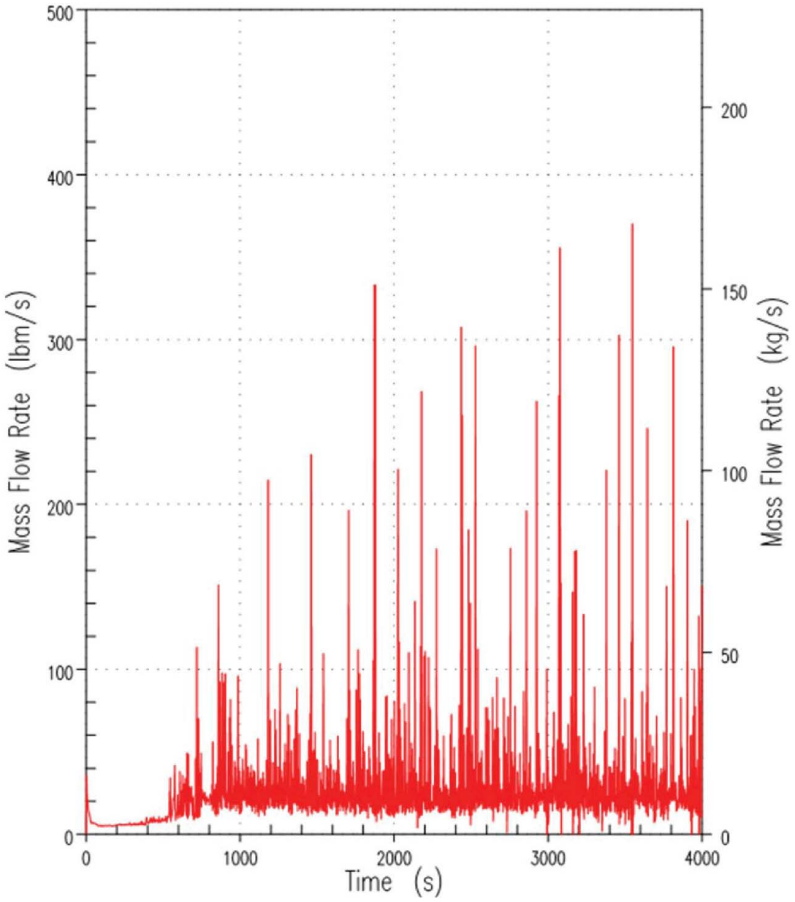

Figure 15.6.5.4C-23
Mixture Flow Rate Through ADS Stage 4A Valves
(Wall-to-Wall Floodup Case) – 14.7 psia

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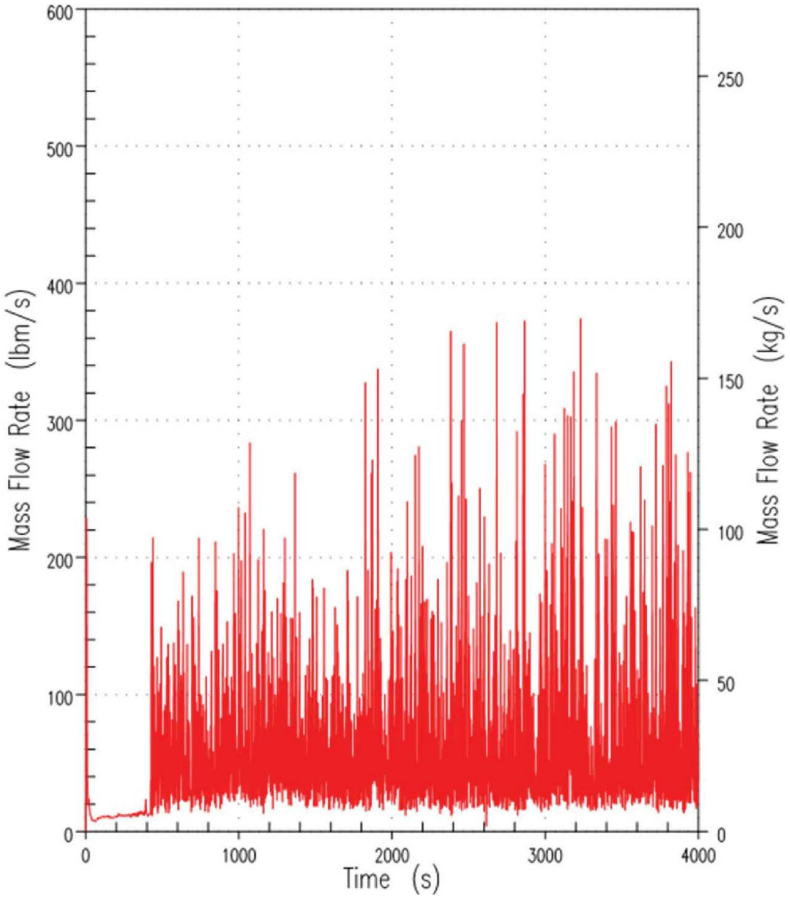

Figure 15.6.5.4C-24
Mixture Flow Rate Through ADS Stage 4B Valves
(Wall-to-Wall Floodup Case) – 14.7 psia

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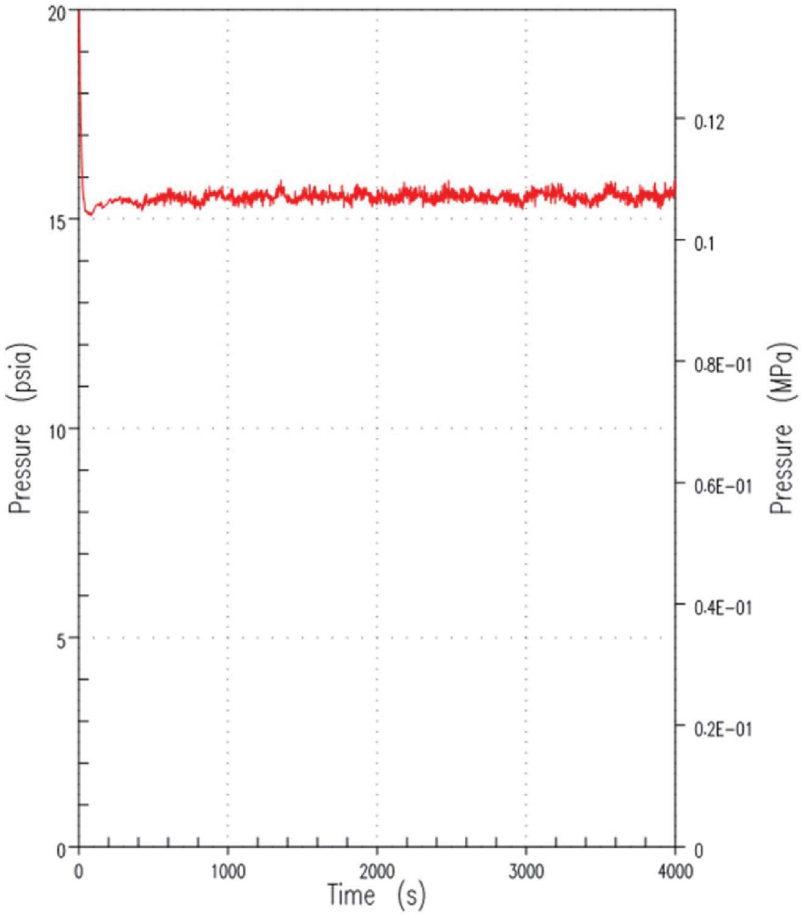

Figure 15.6.5.4C-25
Upper Plenum Pressure
(Wall-to-Wall Floodup Case) – 14.7 psia

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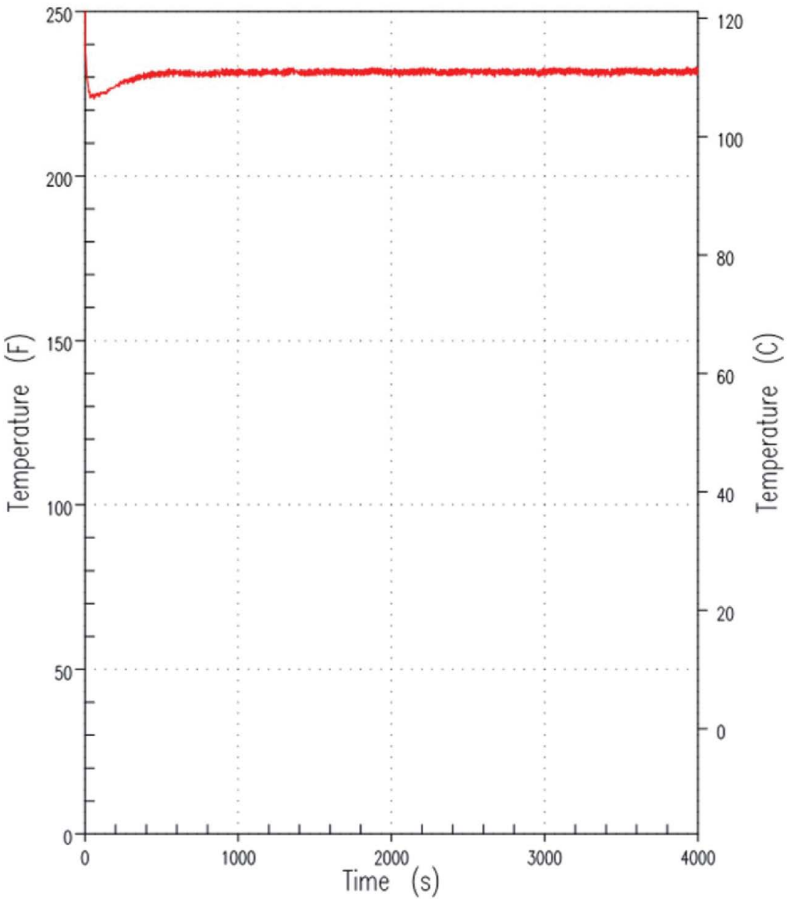

Figure 15.6.5.4C-26
Hot Rod Cladding Temperature Near Top of Core
(Wall-to-Wall Floodup Case) – 14.7 psia

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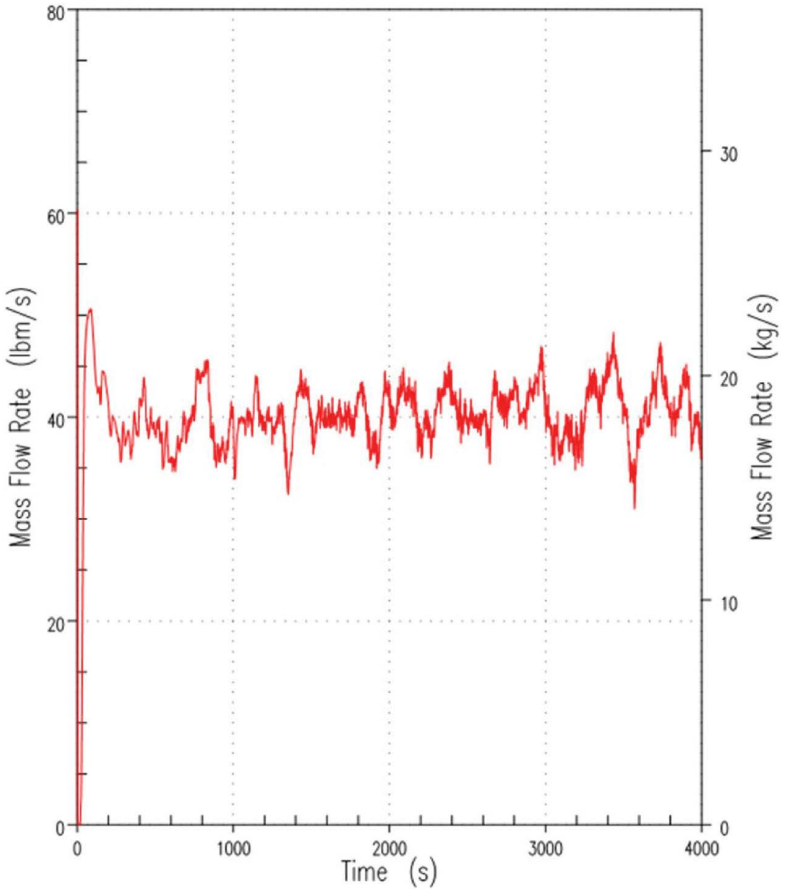

Figure 15.6.5.4C-27
DVI-A Mixture Flow Rate
(Wall-to-Wall Floodup Case) – 14.7 psia

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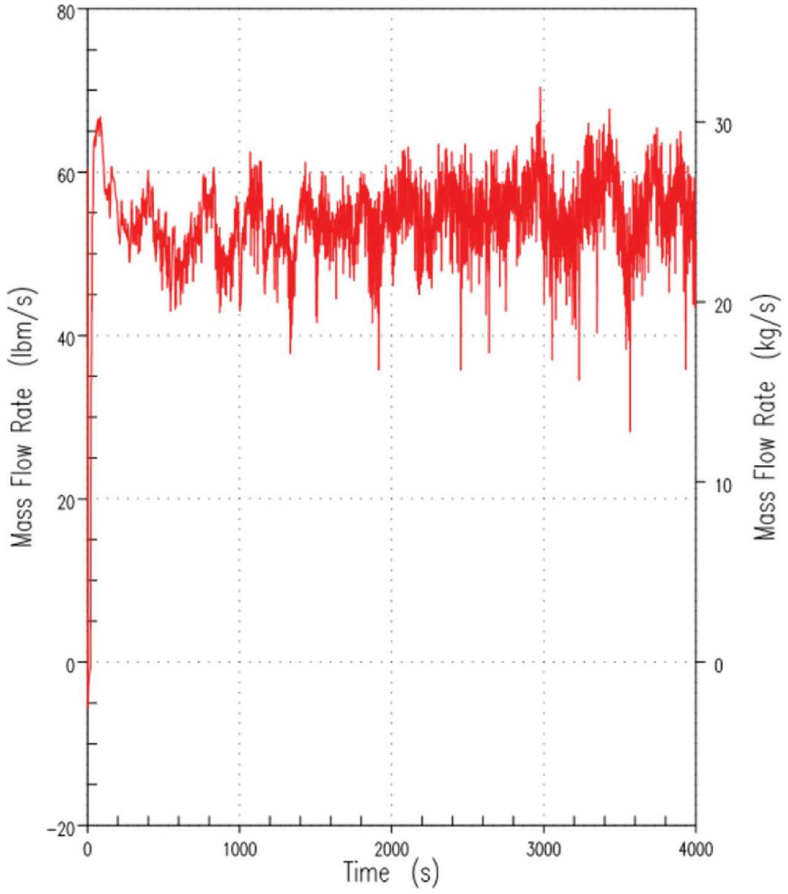

Figure 15.6.5.4C-28
DVI-B Mixture Flow Rate
(Wall-to-Wall Floodup Case) – 14.7 psia

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15.7 Radioactive Release from a Subsystem or Component

This group of events includes the following:

- Gas waste management system leak or failure
- Liquid waste management system leak or failure (atmospheric release)
- Release of radioactivity to the environment via liquid pathways
- Fuel handling accident
- Spent fuel cask drop accident

15.7.1 Gas Waste Management System Leak or Failure

The AP1000 gaseous radwaste system is a low-pressure, low-flow charcoal delay process. Failure of the gaseous radwaste system results in a minor release of activity that is not significant. The Standard Review Plan no longer includes this event as part of the review. Therefore, no analysis is provided.

15.7.2 Liquid Waste Management System Leak or Failure (Atmospheric Release)

The AP1000 liquid radwaste system tanks do not contain significant levels of gaseous activity because liquids expected to contain gaseous radioactivity are processed by a gas stripper before being directed to storage. The tanks are open to the atmosphere so that any evolution of gaseous activity is continually released through the monitored plant vent. The Standard Review Plan no longer includes this event as part of the review. Therefore, no analysis is provided.

15.7.3 Release of Radioactivity to the Environment Due to a Liquid Tank Failure

Tanks containing radioactive fluids are located inside plant structures.

In the event of a tank failure, the liquid would be drained by the floor drains to the auxiliary building sump. From the sump, the water would be directed to the waste holdup tank. The basemat of the auxiliary building is 6-feet thick, the exterior walls are 3-feet thick, and the building is seismic Category I. The exterior walls are sealed to prevent leakage. Thus, it is assumed that there is no release of the spilled liquid waste to the environment. However, the Standard Review Plan states that credit cannot be taken for liquid retention by unlined building foundations. Analysis of the impact of this event will be performed as discussed in subsection 15.7.6. This analysis should include consideration of tank liquid level, processing and decay of tank contents, potential paths of spilled waste to the environment, as well as other pertinent factors.

15.7.4 Fuel Handling Accident

A fuel handling accident can be postulated to occur either inside the containment or in the fuel handling area inside the auxiliary building. The fuel handling accident is defined as the dropping of a spent fuel assembly such that every rod in the dropped assembly has its cladding breached so that the activity in the fuel/cladding gap is released.

The possibility of a fuel handling accident is remote because of the many administrative controls and the equipment operating limits that are incorporated in the fuel handling operations (see Subsection 9.1.4). Only one spent fuel assembly is lifted at a time, and the fuel is moved at low speeds, exercising caution that the fuel assembly not strike anything during movement. The containment, auxiliary building, refueling pool, and spent fuel pool are designed to seismic Category I requirements to thus provide their integrity in the event of a safe shutdown earthquake. The spent fuel storage racks are located to prevent a credible external missile from reaching the stored fuel assemblies. The fuel handling equipment is designed to prevent the handling equipment from falling

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onto the fuel in the reactor vessel or that stored in the spent fuel pool. The facility is designed so that heavy objects, such as the spent fuel shipping cask, cannot be carried over or tipped into the spent fuel pool.

15.7.4.1 Source Term

The inventory of fission products available for release at the time of the accident is dependent on a number of factors, such as the power history of the fuel assembly, the time delay between reactor shutdown and the beginning of fuel handling operations, and the volatility of the nuclides.

The fuel handling accident source term is derived from the core source term detailed in Appendix 15A by taking into account the factors below. The assumptions used to define the fuel handling accident initial airborne release source term are provided in Table 15.7-1 along with the derived source term.

15.7.4.1.1 Fission Product Gap Fraction

During power operation, a portion of the fission products generated in the fuel pellet matrix diffuses into the fuel/cladding gap. The fraction of the assembly fission products found in the gap depends on the rate of diffusion for the nuclide in question as well as the rate of radioactive decay. In the event of a fuel handling accident, the gaseous and volatile radionuclides contained in the fuel/cladding gap are free to escape from the fuel assembly. The radionuclides of concern are the noble gases (kryptons and xenons) and iodines. Based on NUREG-1465 (Reference 1), the fission product gap fraction is 3-percent of fuel inventory. For this analysis, the gap fractions are increased to be consistent with the guidance of Regulatory Guide 1.183 (Reference 2). The gap fractions are listed in Table 15.7-1.

15.7.4.1.2 Iodine Chemical Form

Consistent with NUREG-1465 guidance, the iodine released from the damaged fuel rods is assumed to be 95-percent cesium iodide, 4.85-percent elemental iodine, and 0.15-percent organic iodine.

Cesium iodide is nonvolatile, and the iodine in this form dissolves in water but does not readily become airborne. However, consistent with the guidance in Regulatory Guide 1.183, it is assumed that the cesium iodide is instantaneously converted to the elemental form when released from the fuel into the low pH water pool.

15.7.4.1.3 Assembly Power Level

All fuel assemblies are assumed to be handled inside the containment during the core shuffle so a peak power assembly is considered for the accident. Any fuel assembly can be transferred to the spent fuel pool; during a core off-load, all fuel assemblies are discharged to the spent fuel pool. To obtain a bounding condition for the fuel handling accident analysis, it is assumed that the accident involves a fuel assembly that operated at the maximum rated fuel rod peaking factor. This is conservative because the entire fuel assembly does not operate at this level.

15.7.4.1.4 Radiological Decay

The fission product decay time experienced prior to the fuel handling accident is at least 48 hours.

15.7.4.2 Release Pathways

The spent fuel handling operations take place underwater. Thus, activity releases are first scrubbed by the column of water 23 feet in depth. This has no effect on the releases of noble gases or organic

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iodine but there is a significant removal of elemental iodine. Consistent with the guidance in Regulatory Guide 1.183, the overall pool scrubbing decontamination factor for iodine is assumed to be 200.

In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, there may be less than 23 feet of water above the top of the fuel bundle and the surface of the water, indicated by the width of the bundle. The fuel handling accident analysis bounds the case of a single bundle lying horizontally on top of the spent fuel racks by demonstrating that the overall decontamination factor of 200 is valid for pool depths of 21.5 feet.

After the activity escapes from the water pool, it is assumed that it is released directly to the environment within a 2-hour period without credit for any additional iodine removal process.

If the fuel handling accident occurs in the containment, the release of activity can be terminated by closure of the containment purge lines on detection of high radioactivity. No credit is taken for this in the analysis. Additionally, no credit is taken for removal of airborne iodine by the filters in the containment purge lines.

For the fuel handling accident postulated to occur in the spent fuel pool, there is assumed to be no filtration in the release pathway. Activity released from the pool is assumed to pass directly to the environment with no credit for holdup or delay of release in the building.

15.7.4.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

Table 15.7-1 lists the assumptions used in the analysis. The guidance of Regulatory Guide 1.183 is reflected in the analysis assumptions.

15.7.4.4 Identification of Conservatisms

The fuel handling accident dose analysis assumptions contain a number of conservatisms. Some of these conservatisms are described in the following subsections.

15.7.4.4.1 Fuel Assembly Power Level

The source term is based on the assumption that all of the fuel rods in the damaged assembly have been operating at the maximum fuel rod radial peaking factor. In actuality, this is true for only a small fraction of the fuel rods in any assembly. The overall assembly power level is less than the maximum radial peaking factor.

15.7.4.4.2 Fission Product Gap Fraction

The assumption of Regulatory Guide 1.183 gap fractions for the short-lived nuclides is conservative by a factor of 2 or more, depending on the nuclide.

15.7.4.4.3 Amount of Fuel Damage

It is assumed that all fuel rods in a fuel assembly are damaged so as to release the fission product inventory in the fuel/cladding gap. In an actual fuel handling accident, it is expected that there would be few rods damaged to this extent.

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15.7.4.4.4 Iodine Plateout on Fuel Cladding

Although it is expected that virtually all elemental iodine plates out on the fuel cladding and is unavailable for atmospheric release, no credit is taken for plateout.

15.7.4.4.5 Presence of Organic Iodine

Although 0.15% of the iodine is assumed to be in the organic form (and thus not subject to scrubbing removal in the water pool), there would be no organic iodine in the fuel rods. Any formation of organic iodine would occur gradually and would not contribute to early releases of activity.

15.7.4.4.6 Conversion of Cesium Iodide to Form Elemental Iodine

The analysis assumes that all of the cesium iodide converts immediately to the elemental iodine form after release to the water pool and is treated in the same manner as the iodine initially in the elemental form. While the low pH solution does support conversion to the elemental form, the conversion would not occur unless the cesium iodide was dissolved in the water. The elemental iodine that is formed would thus be in the water solution and not in the bubbles of gas released from the damaged fuel. Additionally, conversion of cesium iodide would occur slowly and the elemental iodine formed would not be immediately available for release.

15.7.4.4.7 Meteorology

It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

15.7.4.4.8 Time Available for Radioactive Decay

The dose analysis assumes that the fuel handling accident involves one of the first fuel assemblies handled. If it were one of the later fuel handling operations, there is additional decay and a reduction in the source term.

The dose evaluation was performed assuming 48 hours decay.

15.7.4.5 Offsite Doses

Using the assumptions from Table 15.7-1, the calculated doses from the initial releases are determined to be 2.8 rem TEDE at the site boundary and 1.2 rem TEDE at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "well within" is taken as meaning 25 percent or less.

15.7.5 Spent Fuel Cask Drop Accident

The spent fuel cask handling crane is prevented from travelling over the spent fuel. No radiological consequences analysis is necessary for the dropped cask event.

15.7.6 Combined License Information

Combined License applicant referencing the AP1000 certified design will perform an analysis of the consequences of potential release of radioactivity to the environment due to a liquid tank failure as outlined in subsection 15.7.3.

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15.7.7 References

1. Sofer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

2. U. S. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, " July 2000.

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Table 15.7-1
Assumptions Used to Determine
Fuel Handling Accident Radiological Consequences

Source term assumptions – Core power (MWt) – Decay time (hr)	 3434(1) 48
Core source term after 48 hours decay (Ci) I-130 I-131 I-132 I-133 I-135 Kr-85m Kr-85 Kr-88 Xe-131m Xe-133m Xe-133 Xe-135m Xe-135	 1.28 E+05 8.18 E+07 9.10 E+07 4.06 E+07 1.17 E+06 1.52 E+04 1.07 E+06 5.45 E+02 1.02 E+06 4.47 E+06 1.70 E+08 1.91 E+05 1.04 E+07
Number of fuel assemblies in core	157
Amount of fuel damage	One assembly
Maximum rod radial peaking factor	1.75
Percentage of fission products in gap I-131 Other iodines Kr-85 Other noble gases	 8 5 10 5
Pool decontamination factor for iodine	200
Activity release period (hr)	2
Atmospheric dispersion factors	See Table 15A-5 in Appendix 15A
Breathing rates (m^3/sec)	3.5 E-4
Nuclide data	See Appendix 15A

Note:
1. The main feedwater flow measurement supports a 1-percent power uncertainty.

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15.8 Anticipated Transients Without Scram

15.8.1 General Background

An anticipated transient without scram (ATWS) is an anticipated operational occurrence during which an automatic reactor scram is required but fails to occur due to a common mode fault in the reactor protection system. Under certain circumstances, failure to execute a required scram during an anticipated operational occurrence could transform a relatively minor transient into a more severe accident. ATWS events are not considered to be in the design basis for Westinghouse plants.

15.8.2 Anticipated Transients Without Scram in the AP1000

For Westinghouse plants, the ATWS rule (10 CFR 50.62) requires the installation of equipment that is diverse from the reactor protection system to automatically trip the turbine and initiate decay heat removal. This equipment must be designed to perform its function in a reliable manner and be independent from sensor output to final actuation device from the existing reactor protection system.

The basis for the ATWS rule requirements, as outlined in SECY-83-293 (Reference 1), is to reduce the risk of core damage because of ATWS to less than 10^{-5} per reactor year.

The AP1000 includes a diverse actuation system, which provides the AMSAC protection features mandated for Westinghouse plants by 10 CFR 50.62, plus a diverse reactor scram (see Section 7.7). Thus, the ATWS rule is met.

15.8.3 Conclusion

The AP1000 is equipped with a diverse actuation system, which provides the functions required by the ATWS rule (10 CFR 50.62). The ATWS core damage frequency for the AP1000 is well below the SECY-83-293 goal of 10^{-5} per reactor year. The AP1000 ATWS core damage frequency is discussed in Chapter 33 of the Probabilistic Risk Assessment (PRA). The AP1000 design meets the ATWS rule (10 CFR 50.62) and its ATWS core damage frequency safety goal basis.

15.8.4 Combined License Information

This section contained no requirement for additional information.

15.8.5 References

1. Dircks, W. J., "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," SECY-83-293, U.S. NRC, July 19, 1983.

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Appendix 15A Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents

This appendix contains the parameters and models that form the basis of the radiological consequences analyses for the various postulated accidents.

15A.1 Offsite Dose Calculation Models

Radiological consequences analyses are performed to determine the total effective dose equivalent (TEDE) doses associated with the postulated accident. The determination of TEDE doses takes into account the committed effective dose equivalent (CEDE) dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the effective dose equivalent (EDE) dose resulting from immersion in the cloud of activity.

15A.1.1 Immersion Dose (Effective Dose Equivalent)

Assuming a semi-infinite cloud, the immersion doses are calculated using the equation:

$$D_{im} = \sum_i DCF_i \sum_j R_{ij} (\chi/Q)_j$$

where:

D_{im} = Immersion (EDE) dose (rem)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-s)

R_{ij} = Amount of isotope i released during time period j (Ci)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

15A.1.2 Inhalation Dose (Committed Effective Dose Equivalent)

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j R_{ij} (BR)_j (\chi/Q)_j$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

R_{ij} = Amount of isotope i released during time period j (Ci)

$(BR)_j$ = Breathing rate during time period j (m^3/s)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

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15A.1.3 Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.2 Main Control Room Dose Models

Radiological consequences analyses are performed to determine the TEDE doses associated with the postulated accident. The determination of TEDE doses takes into account the CEDE dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the EDE dose resulting from immersion in the cloud of activity.

15A.2.1 Immersion Dose Models

Due to the finite volume of air contained in the main control room, the immersion dose for an operator occupying the main control room is substantially less than it is for the case in which a semi-infinite cloud is assumed. The finite cloud doses are calculated using the geometry correction factor from Murphy and Campe (Reference 1).

The equation is:

$$D_{im} = \frac{1}{GF} \sum_{i} DCF_i \sum_{j} (IAR)_{ij} O_j$$

where:

D_{im} = Immersion (EDE) dose (rem)

GF = Main control room geometry factor $=1173/V^{0.338}$

V = Volume of the main control room (ft^3)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-s)

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j (Ci-s/m^3)

O_j = Fraction of time period j that the operator is assumed to be present

15A.2.2 Inhalation Dose

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_{i} DCF_i \sum_{j} (IAR)_{ij} (BR)_j O_j$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j (Ci-s/m^3)

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$(BR)_j$ = Breathing rate during time period j (m^3/s)

O_j = Fraction of time period j that the operator is assumed to be present

15A.2.3 Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.3 General Analysis Parameters

15A.3.1 Source Terms

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the core if the accident involves fuel failures. The radiological consequences analyses use conservative design basis source terms.

15A.3.1.1 Primary Coolant Source Term

The design basis primary coolant source terms are listed in Table 11.1-2. Primary coolant alkali metal source terms are contained in Table 15A-8. These source terms are based on continuous plant operation with 0.25-percent fuel defects. The remaining assumptions used in determining the Table 11.1-2 primary coolant source terms are listed in Table 11.1-1. The concentrations in Table 15A-8 were calculated using the same methods as those used to develop the source term in Table 11.1-2, only with slightly different inputs (i.e. using a power of 3430 MWt, an initial boron concentration of 1490 ppm, and a shorter operating time).

The accident dose analyses take into account increases in the primary coolant source terms for iodines, alkali metals, and noble gases above those listed in Table 15A-8, consistent with the Tech Spec limits of 1.0 μCi/g dose equivalent I-131 for the iodines and 280 μCi/g dose equivalent Xe-133 for the noble gases. The accident dose analyses concentrations reflecting these increases are contained in Table 15A-9.

The radiological consequences analyses for certain accidents also take into account the phenomenon of iodine spiking, which causes the concentration of radioactive iodines in the primary coolant to increase significantly. Table 15A-1 lists the concentrations of iodine isotopes associated with a pre-existing iodine spike. This is an iodine spike that occurs prior to the accident and for which the peak primary coolant activity is reached at the time the accident is assumed to occur. These isotopic concentrations are also defined as 60 μCi/g dose equivalent I-131. The probability of this adverse timing of the iodine spike and accident is small.

Although it is unlikely for an accident to occur at the same time that an iodine spike is at its maximum reactor coolant concentration, for many accidents it is expected that an iodine spike would be initiated by the accident or by the reactor trip associated with the accident. Table 15A-2 lists the iodine appearance rates (rates at which the various iodine isotopes are transferred from the core to the primary coolant by way of the assumed cladding defects) for normal operation. The iodine spike appearance rates are assumed to be as much as 500 times the normal appearance rates.

15A.3.1.2 Secondary Coolant Source Term

The secondary coolant source term used in the radiological consequences analyses is conservatively assumed to be 1 percent of the primary coolant equilibrium source term. This is more conservative than using the design basis secondary coolant source terms listed in Table 11.1-5.

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Because the iodine spiking phenomenon is short-lived and there is a high level of conservatism for the assumed secondary coolant iodine concentrations, the effect of iodine spiking on the secondary coolant iodine source terms is not modeled.

There is assumed to be no secondary coolant noble gas source term because the noble gases entering the secondary side due to primary-to-secondary leakage enter the steam phase and are discharged via the condenser air removal system.

15A.3.1.3 Core Source Term

Table 15A-3 lists the core source terms at shutdown for an assumed three-region equilibrium cycle at end of life after continuous operation at 1 percent above full core thermal power. The main feedwater flow measurement supports a 1-percent power uncertainty. In addition to iodines and noble gases, the source terms listed include nuclides that are identified as potentially significant dose contributors in the event of a degraded core accident. The design basis loss-of-coolant accident analysis is not expected to result in significant core damage, but the radiological consequences analysis assumes severe core degradation.

15A.3.2 Nuclide Parameters

The radiological consequence analyses consider radioactive decay of the subject nuclides prior to their release, but no additional decay is assumed after the activity is released to the environment. Table 15A-4 lists the decay constants for the nuclides of concern.

Table 15A-4 also lists the dose conversion factors for calculation of the CEDE doses due to inhalation of iodines and other nuclides and EDE dose conversion factors for calculation of the dose due to immersion in a cloud of activity. The CEDE dose conversion factors are from EPA Federal Guidance Report No. 11 (Reference 2) and the EDE dose conversion factors are from EPA Federal Guidance Report No. 12 (Reference 3).

15A.3.3 Atmospheric Dispersion Factors

Subsection 2.3.4 lists the off-site short-term atmospheric dispersion factors (χ/Q) for the reference site. Table 15A-5 (Sheet 1 of 2) reiterates these χ/Q values.

The atmospheric dispersion factors (χ/Q) to be applied to air entering the main control room following a design basis accident are specified at the HVAC intake and at the annex building entrance (which would be the air pathway to the main control room due to ingress/egress). A set of reference **AP1000** χ/Q values is identified for each potential activity release location that has been identified and the two control room receptor locations. These reference **AP1000** χ/Q values are listed in Table 15A-6 and are provided in Table 2-1.

Table 15A-7 identifies the AP1000 source and receptor data to be used when determining the site-specific control room χ/Q values using the ARCON96 code (References 4 and 5).

The main control room reference **AP1000** χ/Q values do not incorporate occupancy factors.

The locations of the potential release points and their relationship to the main control room air intake and the annex building access door are shown in Figure 15A-1.

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15A.4 References

1. Murphy, K. G., Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," paper presented at the 13th AEC Air Cleaning Conference.

2. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.

3. EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA 402-R-93-081, September 1993.

4. NUREG/CR-6331, Ramsdell, J. V. and Simonen, C. A., "Atmospheric Relative Concentrations in Building Wakes," Revision 1, May 1997.

5. Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003.

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Table 15A-1
Reactor Coolant Iodine Concentrations for
Maximum Iodine Spike of 60 µCi/g Dose Equivalent I-131

Nuclide	µCi/g
I-130	0.38
I-131	42.8
I-132	61.8
I-133	81.6
I-134	15.7
I-135	51.9

Table 15A-2
Iodine Appearance Rates in the Reactor Coolant

Nuclide	Equilibrium Appearance Rate (Ci/min)
I-130	4.05E-03
I-131	3.34E-01
I-132	1.50E+00
I-133	7.70E-01
I-134	8.07E-01
I-135	6.96E-01

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Table 15A-3 (Sheet 1 of 2)
Reactor Core Source Term(1)

	Nuclide	Inventory (Ci)		Nuclide	Inventory (Ci)
Iodines	I-130	1.87E+06	Noble Gases	Kr-85m	2.52E+07
	I-131	9.48E+07		Kr-85	1.07E+06
	I-132	1.38E+08		Kr-87	4.96E+07
	I-133	1.95E+08		Kr-88	6.66E+07
	I-134	2.19E+08		Xe-131m	1.03E+06
	I-135	1.85E+08		Xe-133m	6.07E+06
Cs Group	Cs-134	1.65E+07		Xe-133	1.92E+08
	Cs-136	3.87E+06		Xe-135m	4.08E+07
	Cs-137	1.09E+07		Xe-135	3.87E+07
	Cs-138	1.82E+08		Xe-138	1.66E+08
	Rb-86	1.80E+05	Sr & Ba	Sr-89	9.48E+07
Te Group	Te-127m	1.37E+06		Sr-90	8.34E+06
	Te-127	8.44E+06		Sr-91	1.17E+08
	Te-129m	4.74E+06		Sr-92	1.25E+08
	Te-129	2.49E+07		Ba-139	1.73E+08
	Te-131m	1.88E+07		Ba-140	1.68E+08
	Te-132	1.35E+08	Ce Group	Ce-141	1.59E+08
	Sb-127	8.71E+06		Ce-143	1.48E+08
	Sb-129	2.67E+07		Ce-144	1.22E+08
Ru Group	Ru-103	1.44E+08		Pu-238	2.22E+05
	Ru-105	9.92E+07		Pu-239	2.53E+04
	Ru-106	4.54E+07		Pu-240	3.96E+04
	Rh-105	9.21E+07		Pu-241	1.01E+07
	Mo-99	1.77E+08		Np-239	1.99E+09
	Tc-99m	1.56E+08			

Note:
1. The following assumptions apply:
   - Core thermal power of 3434 MWt (1 percent above the design core power of 3400 MWt). The main feedwater flow measurement supports a 1-percent power uncertainty.
   - Three-region equilibrium cycle core at end of life

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Table 15A-3 (Sheet 2 of 2)
Reactor Core Source Term(1)

	Nuclide	Inventory (Ci)
La Group	Y-90	8.73E+06
	Y-91	1.23E+08
	Y-92	1.26E+08
	Y-93	1.42E+08
	Nb-95	1.65E+08
	Zr-95	1.63E+08
	Zr-97	1.62E+08
	La-140	1.77E+08
	La-142	1.52E+08
	Pr-143	1.44E+08
	Nd-147	6.12E+07
	Am-241	1.04E+04
	Cm-242	2.81E+06
	Cm-244	2.42E+05

Note:

1. The following assumptions apply:
   - Core thermal power of 3434 MWt (1 percent above the design core power of 3400 MWt). The main feedwater flow measurement supports a 1-percent power uncertainty.
   - Three-region equilibrium cycle core at end of life

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Table 15A-4 (Sheet 1 of 4)
Nuclide Parameters

A. HALOGENS			
Isotope	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
I-130	5.61×10^{-2}	1.04×10^{-13}	7.14×10^{-10}
I-131	3.59×10^{-3}	1.82×10^{-14}	8.89×10^{-9}
I-132	3.01×10^{-1}	1.12×10^{-13}	1.03×10^{-10}
I-133	3.33×10^{-2}	2.94×10^{-14}	1.58×10^{-9}
I-134	7.91×10^{-1}	1.30×10^{-13}	3.55×10^{-11}
I-135	1.05×10^{-1}	7.98×10^{-14}	3.32×10^{-10}
B. NOBLE GASES			
Isotope	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	
Kr-85m	1.55×10^{-1}	7.48×10^{-15}	
Kr-85	7.38×10^{-6}	1.19×10^{-16}	
Kr-87	5.45×10^{-1}	4.12×10^{-14}	
Kr-88	2.44×10^{-1}	1.02×10^{-13}	
Xe-131m	2.43×10^{-3}	3.89×10^{-16}	
Xe-133m	1.32×10^{-2}	1.37×10^{-15}	
Xe-133	5.51×10^{-3}	1.56×10^{-15}	
Xe-135m	2.72	2.04×10^{-14}	
Xe-135	7.63×10^{-2}	1.19×10^{-14}	
Xe-138	2.93	5.77×10^{-14}	

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Table 15A-4 (Sheet 2 of 4)
Nuclide Parameters

C. ALKALI METALS			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Cs-134	$3.84x10^{-5}$	$7.57x10^{-14}$	$1.25x10^{-8}$
Cs-136	$2.2x10^{-3}$	$1.06x10^{-13}$	$1.98x10^{-9}$
Cs-137(1)	$2.64x10^{-6}$	$2.88x10^{-14}$	$8.63x10^{-9}$
Cs-138	1.29	$1.21x10^{-13}$	$2.74x10^{-11}$
Rb-86	$1.55x10^{-3}$	$4.81x10^{-15}$	$1.79x10^{-9}$
D. TELLURIUM GROUP			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Te-127m	$2.65x10^{-4}$	$1.47x10^{-16}$	$5.81x10^{-9}$
Te-127	$7.41x10^{-2}$	$2.42x10^{-16}$	$8.60x10^{-11}$
Te-129m	$8.6x10^{-4}$	$1.55x10^{-15}$	$6.47x10^{-9}$
Te-129	$5.98x10^{-1}$	$2.75x10^{-15}$	$2.42x10^{-11}$
Te-131m	$2.31x10^{-2}$	$7.01x10^{-14}$	$1.73x10^{-9}$
Te-132	$8.86x10^{-3}$	$1.03x10^{-14}$	$2.55x10^{-9}$
Sb-127	$7.5x10^{-3}$	$3.33x10^{-14}$	$1.63x10^{-9}$
Sb-129	$1.6x10^{-1}$	$7.14x10^{-14}$	$1.74x10^{-10}$
E. STRONTIUM AND BARIUM			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Sr-89	$5.72x10^{-4}$	$7.73x10^{-17}$	$1.12x10^{-8}$
Sr-90	$2.72x10^{-6}$	$7.53x10^{-18}$	$3.51x10^{-7}$
Sr-91	$7.3x10^{-2}$	$3.45x10^{-14}$	$4.49x10^{-10}$
Sr-92	$2.56x10^{-1}$	$6.79x10^{-14}$	$2.18x10^{-10}$
Ba-139	$5.02x10^{-1}$	$2.17x10^{-15}$	$4.64x10^{-11}$
Ba-140	$2.27x10^{-3}$	$8.58x10^{-15}$	$1.01x10^{-9}$

Note:
1. The listed average gamma disintegration energy for Cs-137 is due to the production and decay of Ba-137m.

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Table 15A-4 (Sheet 3 of 4)
Nuclide Parameters

F. NOBLE METALS			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Ru-103	$7.35x10^{-4}$	$2.25x10^{-14}$	$2.42x10^{-9}$
Ru-105	$1.56x10^{-1}$	$3.81x10^{-14}$	$1.23x10^{-10}$
Ru-106	$7.84x10^{-5}$	0.0	$1.29x10^{-7}$
Rh-105	$1.96x10^{-2}$	$3.72x10^{-15}$	$2.58x10^{-10}$
Mo-99	$1.05x10^{-2}$	$7.28x10^{-15}$	$1.07x10^{-9}$
Tc-99m	$1.15x10^{-1}$	$5.89x10^{-15}$	$8.80x10^{-12}$
G. CERIUM GROUP			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Ce-141	$8.89x10^{-4}$	$3.43x10^{-15}$	$2.42x10^{-9}$
Ce-143	$2.1x10^{-2}$	$1.29x10^{-14}$	$9.16x10^{-10}$
Ce-144	$1.02x10^{-4}$	$8.53x10^{-16}$	$1.01x10^{-7}$
Pu-238	$9.02x10^{-7}$	$4.88x10^{-18}$	$1.06x10^{-4}$
Pu-239	$3.29x10^{-9}$	$4.24x10^{-18}$	$1.16x10^{-4}$
Pu-240	$1.21x10^{-8}$	$4.75x10^{-18}$	$1.16x10^{-4}$
Pu-241	$5.5x10^{-6}$	$7.25x10^{-20}$	$2.23x10^{-6}$
Np-239	$1.23x10^{-2}$	$7.69x10^{-15}$	$6.78x10^{-10}$

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Table 15A-4 (Sheet 4 of 4)
Nuclide Parameters

H. LANTHANIDE GROUP			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Y-90	$1.08x10^{-2}$	$1.90x10^{-16}$	$2.28x10^{-9}$
Y-91	$4.94x10^{-4}$	$2.60x10^{-16}$	$1.32x10^{-8}$
Y-92	$1.96x10^{-1}$	$1.30x10^{-14}$	$2.11x10^{-10}$
Y-93	$6.86x10^{-2}$	$4.80x10^{-15}$	$5.82x10^{-10}$
Nb-95	$8.22x10^{-4}$	$3.74x10^{-14}$	$1.57x10^{-9}$
Zr-95	$4.51x10^{-4}$	$3.60x10^{-14}$	$6.39x10^{-9}$
Zr-97	$4.1x10^{-2}$	$9.02x10^{-15}$	$1.17x10^{-9}$
La-140	$1.72x10^{-2}$	$1.17x10^{-13}$	$1.31x10^{-9}$
La-141	$1.76x10^{-1}$	$2.39x10^{-15}$	$1.57x10^{-10}$
La-142	$4.5x10^{-1}$	$1.44x10^{-13}$	$6.84x10^{-11}$
Nd-147	$2.63x10^{-3}$	$6.19x10^{-15}$	$1.85x10^{-9}$
Pr-143	$2.13x10^{-3}$	$2.10x10^{-17}$	$2.19x10^{-9}$
Am-241	$1.83x10^{-7}$	$8.18x10^{-16}$	$1.20x10^{-4}$
Cm-242	$1.77x10^{-4}$	$5.69x10^{-18}$	$4.67x10^{-6}$
Cm-244	$4.37x10^{-6}$	$4.91x10^{-18}$	$6.70x10^{-5}$

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Table 15A-5
Offsite Atmospheric Dispersion Factors (χ/Q)
for Accident Dose Analysis

Site boundary χ/Q (s/m^3)	
0–2 hours(1)	5.1×10^{-4}
Low population zone χ/Q (s/m^3)	
0–8 hours	2.2×10^{-4}
8–24 hours	1.6×10^{-4}
24–96 hours	1.0×10^{-4}
96–720 hours	8.0×10^{-5}

Note:
1. Nominally defined as the 0- to 2-hour interval but is applied to the 2-hour interval having the highest activity releases in order to address 10 CFR Part 50.34 requirements

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Table 15A-6
Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[1]						
	Plant Vent or PCS Air Diffuser[3]	**Ground Level Containment Release Points[4]**	**PORV and Safety Valve Releases[5]**	**Steam Line Break Releases**	**Fuel Handling Area[6]**	**Condenser Air Removal Stack[7]**
0 – 2 hours	2.53E-03	4.00E-03	1.92E-02	2.13E-02	6.0E-3	6.0E-3
2 – 8 hours	1.98E-03	2.28E-03	1.60E-02	1.76E-02	4.0E-3	4.0E-3
8 – 24 hours	7.96E-04	1.03E-03	6.90E-03	7.50E-03	2.0E-3	2.0E-3
1 – 4 days	6.40E-04	9.03E-04	4.96E-03	5.43E-03	1.5E-3	1.5E-3
4 – 30 days	4.78E-04	7.13E-04	4.16E-03	4.55E-03	1.0E-3	1.0E-3
χ/Q (s/m^3) at Annex Building Door for the Identified Release Points[2]						
	Plant Vent or PCS Air Diffuser[3]	**Ground Level Containment Release Points[4]**	**PORV and Safety Valve Releases[5]**	**Steam Line Break Releases**	**Fuel Handling Area[6]**	**Condenser Air Removal Stack[7]**
0 – 2 hours	1.0E-3	1.0E-3	4.0E-3	4.0E-3	6.0E-3	2.0E-2
2 – 8 hours	7.5E-4	7.5E-4	3.2E-3	3.2E-3	4.0E-3	1.8E-2
8 – 24 hours	3.5E-4	3.5E-4	1.2E-3	1.2E-3	2.0E-3	7.0E-3
1 – 4 days	2.8E-4	2.8E-4	1.0E-3	1.0E-3	1.5E-3	5.0E-3
4 – 30 days	2.5E-4	2.5E-4	8.0E-4	8.0E-4	1.0E-3	4.5E-3

Notes:

1. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the non-safety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.
2. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.
3. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.
4. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.
5. The listed values bound the dispersion factors for releases from the steam line safety & power-operated relief valves. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and for the secondary side release from a rod ejection accident.
6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.
7. This release point is included for information only as a potential activity release point. None of the design basis accident radiological consequences analyses model release from this point.

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Table 15A-7
Control Room Source/Receptor Data for Determination Of Atmospheric Dispersion Factors

Source Description	Release Elevation Note 1 (m)	Horizontal Straight-Line Distance To Receptor		
		Control Room HVAC Intake (Elevation 19.7 m) (Δ1)	Annex Building Access (Elevation 1.5 m) (Δ2)	Comment
Plant Vent (⬡1)	55.7	128 ft (39.0 m)	350 ft (106.6 m)	
PCS Air Diffuser (⬡2)	69.8	114 ft (34.7 m)	332 ft (101.1 m)	
Auxiliary Building Fuel Handling Area Blowout Panel (⬡3)	17.4	201 ft (61.2 m)	416 ft (126.8 m)	Note 3
Radwaste Building Truck Staging Area Door (⬡4)	1.5	204 ft (62.1 m)	411 ft (125.2 m)	Note 3
Steam Vent (⬡5)	17.1	55 ft (16.7 m)	250 ft (76.2 m)	
PORV/Safety Valves (⬡6)	19.2	58 ft (17.6 m)	235 ft (71.6 m)	
Condenser Air Removal Stack (⬡7)	49.5	307 ft (93.5 m)	112 ft (34.1 m)	Note 3
Containment Shell (Diffuse Area Source) (⬡8)	Same as Receptor Elevation (19.7 m or 1.5 m)	48 ft (14.6 m)	268 ft (81.6 m)	Note 2

Notes:
1. All elevations relative to grade at 0.0 m.
2. For calculating distance, the source is defined as the point on the containment shell closest to receptor.
3. Vertical distance traveled is conservatively neglected.
4. ⬡ – Refer to Symbols on Figure 15A-1.
5. Δ – Refer to Symbols on Figure 15A-1.

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Table 15A-8
Reactor Coolant Concentrations at Design Fuel Defect Level of 0.25%

Nuclide	Activity (μCi/g)
Kr-85m	7.10E-01
Kr-85	2.50E+00
Kr-87	4.10E-01
Kr-88	1.24E+00
Xe-131m	1.06E+00
Xe-133m	1.23E+00
Xe-133	1.01E+02
Xe-135m	1.41E-01
Xe-135	2.53E+00
Xe-138	2.14E-01
I-130	4.57E-03
I-131	5.17E-01
I-132	7.46E-01
I-133	9.84E-01
I-134	1.90E-01
I-135	6.27E-01
Cs-134	4.53E-01
Cs-136	6.66E-01
Cs-137	3.70E-01
Cs-138	3.19E-01
Rb-86	5.25E-03

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Table 15A-9
Reactor Coolant Concentrations for Radiological Consequence Analyses

Nuclide	Activity (μCi/g)
Kr-85m	3.40E+00
Kr-85	1.91E-01
Kr-87	1.08E+01
Kr-88	8.11E+01
Xe-131m	2.64E-01
Xe-133	1.01E+02
Xe-133m	1.08E+00
Xe-135m	1.84E+00
Xe-135	1.93E+01
Xe-138	7.92E+00
I-130	3.67E-04
I-131	5.17E-01
I-132	8.65E-03
I-133	1.75E-01
I-134	7.59E-04
I-135	2.34E-02
Cs-134	6.25E-01
Cs-136	9.19E-01
Cs-137	5.10E-01
Cs-138	4.40E-01
Rb-86	7.24E-03

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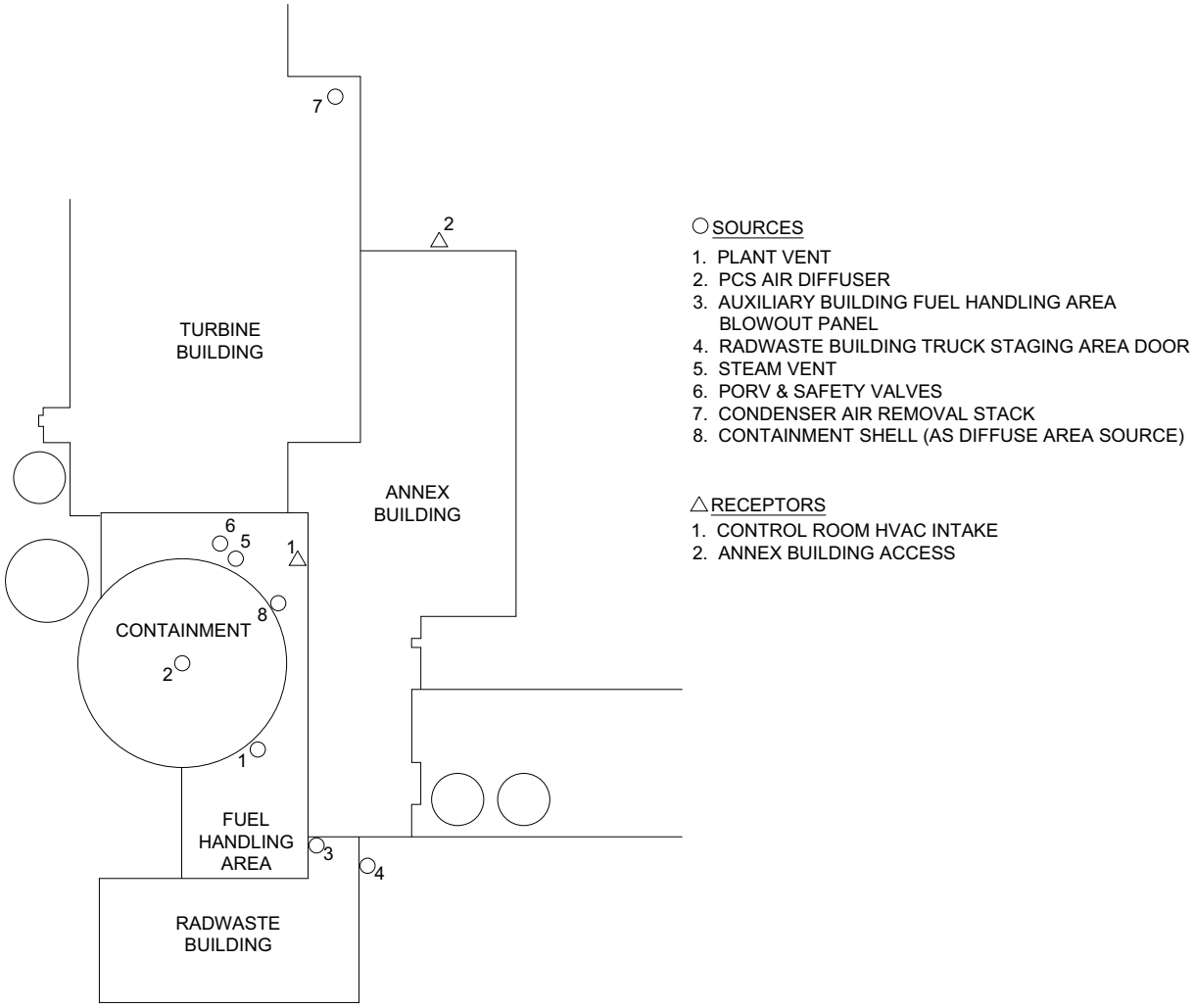

Figure 15A-1
Site Plan with Release and Intake Locations

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Appendix 15B Removal of Airborne Activity from the Containment Atmosphere Following a LOCA

The AP1000 design does not depend on active systems to remove airborne particulates or elemental iodine from the containment atmosphere following a postulated loss-of-coolant accident (LOCA) with core melt. Naturally occurring passive removal processes provide significant removal capability such that airborne elemental iodine is reduced to very low levels within a few hours and the airborne particulates are reduced to extremely low levels within 12 hours.

15B.1 Elemental Iodine Removal

Elemental iodine is removed by deposition onto the structural surfaces inside the containment. The removal of elemental iodine is modeled using the equation from the Standard Review Plan (Reference 1):

$$\lambda_d = \frac{K_w A}{V}$$

where:

λ_d = first order removal coefficient by surface deposition

K_w = mass transfer coefficient (specified in Reference 1 as 4.9 m/hr)

A = surface area available for deposition

V = containment building volume

The available deposition surface is 251,000 ft^2, and the containment building net free volume is 2.06×10^6 ft^3. From these inputs, the elemental iodine removal coefficient is 1.9 hr^{-1}.

Consistent with the guidance of Reference 1, credit for elemental iodine removal is assumed to continue until a decontamination factor (DF) of 200 is reached in the containment atmosphere. Because the source term for the LOCA (defined in Subsection 15.6.5.3) is modeled as a gradual release of activity into the containment, the determination of the time at which the DF of 200 is reached needs to be based on the amount of elemental iodine that enters the containment atmosphere over the duration of core activity release.

15B.2 Aerosol Removal

The deposition removal of aerosols from the containment atmosphere is accomplished by a number of processes including sedimentation, diffusiophoresis, and thermophoresis. All three of the deposition processes are significant contributors to the overall removal process in the AP1000. The large contributions from diffusiophoresis and thermophoresis to the total removal are a direct consequence of the high heat transfer rates from the containment atmosphere to the containment wall that characterize the passive containment cooling system.

Because of the AP1000 passive containment cooling system design, there are high sensible heat transfer rates (resulting in higher thermophoretic removal of aerosols) when condensational heat transfer is low (and the aerosol removal by diffusiophoresis is also low). The reverse is also true. Thus, there is an appreciable deposition removal throughout the accident from either diffusiophoresis or thermophoresis, in addition to the removal by sedimentation.

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15B.2.1 Mathematical Models

The models used for the three aerosol removal processes are discussed as follows.

15B.2.1.1 Sedimentation

Gravitational sedimentation is a major mechanism of aerosol removal in a containment. A standard model (Stokes equation with the Cunningham slip correction factor) for this process is used. The Stokes equation (Reference 2) is:

$$\upsilon_s = \frac{2\rho_p g r^2 Cn}{9\mu}$$

where:

υ_s = settling velocity of an aerosol particle

ρ_p = material density of the particle

g = gravitational acceleration

r = particle radius

μ = gas viscosity

Cn = Cunningham slip correction factor, a function of the Knudsen number (Kn) which is the gas molecular mean free path divided by the particle radius

However, the Stokes equation makes the simplifying assumption that the particles are spherical. The particles are expected to be nonspherical, and it is conventional to address this by introducing a "dynamic shape factor" (Reference 2) in the denominator of the Stokes equation, such that the settling velocity for the nonspherical particle is the same as for a spherical particle of equal volume. The value of the dynamic shape factor (ϕ) thus depends on the shape of the particle and, in general, must be experimentally determined.

The concept of dynamic shape factor can also be applied to a spherical particle consisting of two components, one of which has the density of the particle material, while the other component has a different density (Reference 9). In this manner, the impact of the void fraction in the particle can be modeled. Thus, the revised Stokes equation is:

$$\upsilon_s = \frac{2\rho_p g r^2 Cn}{9\mu\phi}$$

The derivation of ϕ follows.

The two-component particle is considered to have a density ρ_{av} and an effective radius of r_e. Assuming that the second component of the particle is the void volume and letting the void fraction be ε, then the average density of the particle is:

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ρ_{av} = the average density of the particle = $\rho_p (1-\varepsilon) + \rho_v\varepsilon$

where:

ρ_v = density of the void material (0.0 for gas filled, 1.0 for water filled)

ε = void fraction

ρ_p = material density (solid particle with no voids)

The definition of ϕ is obtained from the Stokes equation and the equation for mass of a sphere:

$$\frac{2\rho_p g r^2 Cn}{9\mu\phi} = \frac{2\rho_{av} g r_e^2 Cn}{9\mu}$$

which reduces to:

$$\rho_p r^2 = \phi\rho_{av} r_e^2$$

and

$$\frac{4\rho_p \pi r_0^3}{3} = \frac{4\rho_{av} \pi r_e^3}{3}$$

which reduces to:

$$\rho_p r_0^3 = \rho_{av} r_e^3$$

Then:

$$\phi = \frac{\rho_p r^2}{\rho_{av} r_e^2}$$

and

$$r_e = r\left(\frac{\rho_{av}}{\rho_p}\right)^{-1/3}$$

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From these two relationships, the dynamic shape factor is given by:

$$\phi = \left(\frac{\rho_{av}}{\rho_p} \right)^{-1/3}$$

15B.2.1.2 Diffusiophoresis

Diffusiophoresis is the process whereby particles are swept to a surface (for example, containment wall) by the flow set up by a condensing vapor (Stefan flow). The deposition rate is independent of the particle size and is proportional to the steam condensation rate on the surface. The standard equation for this phenomenon is due to Waldmann and Schmitt (Reference 3):

$$\upsilon_d = \frac{\sqrt{M_v}}{\sqrt{M_v} + \chi_{a/v}\sqrt{M_a}} \frac{W}{\rho_v}$$

where:

υ_d = diffusiophoretic deposition velocity

$\chi_{a/v}$ = ratio of mole fraction of air to mole fraction of steam in the containment atmosphere

M_v = molecular weight of steam

M_a = molecular weight of air

W = steam condensation rate on the wall

ρ_v = mass density of steam in the containment atmosphere

Because of the design of the passive containment cooling system, steam condensation rates are high at certain times in the design basis LOCA; thus at these times, diffusiophoretic deposition rates are significant.

15B.2.1.3 Thermophoresis

Thermophoresis is the process whereby particles drift toward a surface (for example, the containment wall) under the influence of a temperature gradient in the containment atmosphere at the surface. The effect arises because the gas molecules on the hot side of the particles undergo more collisions with the particle than do those on the cold side. Therefore, there is a net momentum transfer to the particle in the hot-to-cold direction. There are several models in the literature for this effect; the one used is the Brock equation in a form due to Talbot et al. (Reference 4). As indicated below, this model is in agreement with experimental data. The thermophoretic deposition rate is somewhat dependent on particle size and is proportional to the temperature gradient at the wall, or equivalently, the sensible heat transfer rate to the wall. The Talbot equation is:

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$$\upsilon_{th} = \frac{2\,C_s\,Cn\,(\mu_g/\rho_g)[\alpha + C_T\,Kn]\,dT}{[1+2(\alpha + C_T\,Kn)][1+3\,C_M\,Kn]}\left(\frac{1}{T}\right)\frac{dT}{dy}$$

where:

υ_{th} = thermophoretic deposition velocity

α = k_g/k_p which is the ratio of the thermal conductivities of the gas (evaluated at the gas temperature at each time step) and the aerosol particle (k_p is set equal to the thermal conductivity of water – the results are not sensitive to k_p or α.)

Kn = Knudsen number (equal to the gas molecular mean free path divided by the particle radius)

Cn = Cunningham slip correction factor, a function of the Knudsen number

μ_g = gas viscosity

ρ_g = gas density

C_S = slip accommodation coefficient (Reference 4 gives the best value as 1.17.)

C_T = thermal accommodation coefficient (Reference 4 gives the best value as 2.18.)

C_M = momentum accommodation coefficient (Reference 4 gives the best value as 1.14.)

The temperature gradient at the wall, dT/dy, can be evaluated as

$$\frac{dT}{dy} = \frac{\phi_s}{k_g}$$

where ϕ_s is the sensible heat flux to the wall, and k_g is the thermal conductivity of the gas. The sensible heat flux used in the analysis is the convective heat transfer calculated as discussed in Subsection 15B.2.4.7.

15B.2.2 Other Removal Mechanisms

In addition to the above mechanisms, there are others that were not considered, including turbulent diffusion and turbulent agglomeration. The neglect of these mechanisms adds further conservatism to the calculation.

15B.2.3 Validation of Removal Mechanisms

The aerosol processes are well established and have been confirmed in many separate effects experiments, which are discussed in standard references (References 2 through 4). The Stokes formula for sedimentation velocity has been well confirmed for particles whose diameters are less than about 50 μm. In the present calculations, these make up basically all of the aerosol.

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There are some separate effects validations of the diffusiophoretic effect, but the best confirmation comes from integral experiments such as the LACE tests (Reference 5). Calculations of these and other integral tests accurately predict the integrated mass of plated aerosol material only if diffusiophoresis is taken into account. If it is neglected, the predicted plated mass is about two orders of magnitude too small, compared to the observed plated mass.

The Talbot equation for the thermophoretic effect has been experimentally confirmed to within about 20 to 50 percent over a wide range of particle sizes (Reference 4). The temperature gradient at the wall, which drives this phenomenon, can be approximated by the temperature difference between the bulk gas and the wall divided by an appropriate length scale obtained from heat transfer correlations. Alternatively, because sensible heat transfer rates to the wall are available, it is easier and more accurate to use these rates directly to infer the temperature gradient.

15B.2.4 Parameters and Assumptions for Calculating Aerosol Removal Coefficients

The parameters and assumptions were selected to conservatively model the environment that would be expected to exist as a result of a LOCA with concurrent core melt.

15B.2.4.1 Containment Geometry

The containment is assumed to be a cylinder with a volume of 55,481 m^3 (1.959 x 10^6 ft^3). This volume includes those portions of the containment volume that would be participating in the aerosol transport and mixing; this excludes dead-ended volumes and flooded compartments. The horizontal surface area available for aerosol deposition by sedimentation is 2900 m^2 (31,200 ft^2). This includes projecting areas such as decks in addition to the floor area and excludes areas in dead-ended volumes and areas that would be flooded post-LOCA. The surface area for Brownian diffusive plateout of aerosols is 8008 m^2 (86,166 ft^2).

15B.2.4.2 Source Size Distribution

The aerosol source size distribution is assumed to be lognormal, with a geometric mean radius of 0.22 μm and a geometric standard deviation equal to 1.81. These values are derived from an evaluation of a large number of aerosol distributions measured in a variety of degraded-fuel tests and experiments. The sensitivity of aerosol removal coefficient calculations to these values is small.

15B.2.4.3 Aerosol Void Fraction

Review of scanning electron microscope photographs of deposited aerosol particles from actual core melt and fission product vaporization and aerosolization experiments (the Argonne STEP-4 test [Reference 10] and the INEL Power Burst Facility SFD 1-4 test [Reference 6]) indicates that the deposited particles are relatively dense, supporting a void fraction of 0.2.

The above-mentioned test results indicate that a void fraction of 0.2 is appropriate for modeling the aerosols resulting from a core melt. As part of the sensitivity study that was performed for the AP600 project, a case was run with a void fraction of 0.9. That analysis showed that the high void fraction resulted in an integrated release of aerosols over a 24-hour period that was less than 14 percent greater than that calculated when using the void fraction of 0.2. Thus, it is clear that the removal of aerosols from the containment atmosphere is not highly sensitive to the value selected for the void fraction. This is largely due to the fact that, while the selected value for void fraction has a significant impact on the calculated sedimentation removal, the impact on thermophoresis and diffusiophoresis removal is slight or none. The impact for AP1000 of using the higher value for void fraction would be less than was determined for the AP600 since sedimentation removal comprises a smaller fraction of the total removal calculated for the AP1000.

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For additional conservatism, the AP1000 aerosol removal analysis uses a void fraction of 0.4 and assumes the voids are filled with air.

15B.2.4.4 Fission Product Release Fractions

Core inventories of fission products are from ORIGEN calculations for the AP1000 at end of the fuel cycle. Fractional releases to the containment of the fission products are those specified in Subsection 15.6.5.3.

15B.2.4.5 Inert Aerosol Species

The inert species include SnO_2, UO_2, Cd, Ag, and Zr. These act as surrogates for all inert materials forming aerosols. The ratio of the total mass of inert species to fission product species was assumed to be 1.5:1. This value and the partitioning of the total inert mass among its constituents are consistent with results from degraded fuel experiments (Reference 6).

15B.2.4.6 Aerosol Release Timing and Rates

Aerosol release timing is in accordance with the source term defined in Subsection 15.6.5.3. Aerosol release takes place in two main phases: a gap release lasting for 0.5 hour, followed by an early in-vessel release of 1.3 hours duration. During each phase, the aerosols are assumed to be released at a constant rate. These rates were obtained for each species by combining its core inventory, release fraction, and times of release.

Only cesium and iodine are released during the gap release phase. During the in-vessel release phase, the other fission product and inert species are released as well.

15B.2.4.7 Containment Thermal-Hydraulic Data

The thermal-hydraulic parameters used in the aerosol removal calculation are the containment gas temperature, the containment pressure, the steam condensation rate on the wall, the steam mole fraction, and the convective heat transfer rate, all as functions of time. The AP1000-specific parameters were obtained using MAAP4 (Reference 7) for the 3BE-1 severe accident sequence (medium LOCA with failure to inject water from the refueling water storage tank into the reactor vessel). The thermal-hydraulic data are thus consistent with a core melt sequence.

15B.2.5 Aerosol Removal Coefficients

The aerosol removal coefficients are provided in Table 15B-1 starting at the onset of core damage through 24 hours. The removal coefficients for times beyond 24 hours are not of concern because there would be so little aerosol remaining airborne at that time. The values range between 0.29 hr^{-1} and 1.1 hr^{-1} during the time between the onset of core damage (0.167 hour) and 24 hours.

These removal coefficients conservatively neglect steam condensation on the airborne particles, turbulent diffusion, and turbulent agglomeration. Additionally, the assumed source aerosol size is conservatively small being at the low end of the mass mean aerosol size range of 1.5 to 5.5 μm used in NUREG/CR-5966 (Reference 8). Selection of smaller aerosol size would underestimate sedimentation.

Unlike the case for the elemental iodine removal, there is no limit assumed on the removal of aerosols from the containment atmosphere.

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Table 15B-1 (Sheet 1 of 3)
Aerosol Removal Coefficients in the AP1000 Containment Following a Design Basis LOCA With Core Melt

Time Interval (hours)			Removal Coefficient (hr^{-1})
0.167	–	0.179	1.141
0.179	–	0.200	1.013
0.200	–	0.251	0.944
0.251	–	0.292	0.882
0.292	–	0.433	0.842
0.433	–	0.631	0.901
0.631	–	0.684	0.821
0.684	–	0.801	0.781
0.801	–	0.893	0.735
0.893	–	1.033	0.699
1.033	–	1.171	0.662
1.171	–	1.233	0.627
1.233	–	1.331	0.594
1.331	–	1.395	0.562
1.395	–	1.429	0.551
1.429	–	1.475	0.576
1.475	–	1.519	0.537
1.519	–	1.579	0.510
1.579	–	1.653	0.483
1.653	–	1.776	0.458
1.776	–	1.903	0.430
1.903	–	1.991	0.462
1.991	–	2.067	0.429
2.067	–	2.176	0.396
2.176	–	2.371	0.380
2.371	–	2.621	0.337

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Table 15B-1 (Sheet 2 of 3)
Aerosol Removal Coefficients in the AP1000 Containment Following a Design Basis LOCA With Core Melt

Time Interval (hours)			Removal Coefficient (hr^{-1})
2.621	–	2.822	0.320
2.822	–	2.872	0.357
2.872	–	2.973	0.327
2.973	–	3.176	0.302
3.176	–	3.684	0.287
3.684	–	3.737	0.328
3.737	–	3.839	0.304
3.839	–	3.990	0.298
3.990	–	4.090	0.317
4.090	–	4.438	0.346
4.438	–	4.684	0.369
4.684	–	4.880	0.396
4.880	–	4.928	0.449
4.928	–	5.362	0.435
5.362	–	5.460	0.459
5.460	–	5.511	0.518
5.511	–	5.608	0.487
5.608	–	6.040	0.479
6.040	–	6.090	0.537
6.090	–	6.615	0.506
6.615	–	6.753	0.567
6.753	–	7.194	0.513
7.194	–	7.285	0.594
7.285	–	7.814	0.518
7.814	–	7.904	0.581

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Table 15B-1 (Sheet 3 of 3)
Aerosol Removal Coefficients in the AP1000 Containment Following a Design Basis LOCA With Core Melt

Time Interval (hours)			Removal Coefficient (hr^{-1})
7.904	–	8.431	0.528
8.431	–	8.521	0.589
8.521	–	9.387	0.529
9.387	–	9.553	0.568
9.553	–	11.189	0.530
11.189	–	14.937	0.516
14.937	–	17.610	0.506
17.610	–	24	0.492

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16.1 Technical Specifications

16.1.1 Introduction to Technical Specifications

LCO Selection Criteria

The screening criteria of 10CFR50.36, c(2)(ii) stated below has been used to identify the structures, systems, and parameters for which Limiting Conditions for Operation (LCOs) have been included in the AP1000 Technical Specifications.

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient Analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Technical Specification Content

The content of the AP1000 Technical Specifications meets the 10CFR50.36 requirements and is consistent with the Technical Specification Improvement Program, NUREG 1431, Rev. 2, to the maximum extent possible. The content differs from NUREG 1431 only as necessary to reflect technical differences between the “typical” Westinghouse design and the AP1000 design.

Completion Times and Surveillance Frequencies

The Completion Times and Surveillance Frequencies specified in NUREG 1431 have been applied to similar Actions and Surveillances Requirements in AP1000. Refer to Westinghouse letter DCP/NRC0891 for a discussion regarding selection of Completion Times and Surveillance Frequencies for those AP1000 Tech Specs for which no comparable NUREG 1431 system/function exists and for those AP1000 system design differences which lead to deviations from NUREG 1431 Completion Times and Surveillance Frequencies.

Shutdown Completion Times/Mode Definitions

The AP1000 plant design is different from current Westinghouse designs in that the systems normally used for MODE reduction are non-safety systems; and therefore, are not covered by LCO requirements in Technical Specifications. The passive safety systems, which shut down the plant require a longer period of time to accomplish mode changes and can not reduce the RCS temperature to below 200°F.

Combined License Information

This set of generic technical specifications is intended to be used as a guide in the development of the plant-specific technical specifications. Combined License applicants referencing the AP1000 design certification will be required to provide the final information for the remaining brackets [ ] with final plant-specific information.

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1.0 USE AND APPLICATION

1.1 Definitions

- NOTE -

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST may be performed by means of any series of sequential, overlapping, or total steps.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two-section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD), thermocouple, or reactor coolant pump speed sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

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1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same committed effective dose equivalent as the quantity and isotopic mixture of I-130, I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Table 2.1 of EPA Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” EPA-520/ 1-88-020, September 1988.
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same effective dose equivalent as the quantity and isotopic mixture of noble gases (Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138) actually present. The dose conversion factors used for this calculation shall be those listed in Table III.1 of EPA Federal Guidance Report No. 12, “External Exposure to Radionuclides in Air, Water, and Soil,” EPA 402-R-93-081, September 1993.

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1.1 Definitions

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from seals or valve packing, that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of leakage detection systems; 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System (primary to secondary LEAKAGE); or 4. RCS LEAKAGE through the passive residual heat removal heat exchanger (PRHR HX) to the In-containment Refueling Water Storage Tank (IRWST). b. <u>Unidentified LEAKAGE</u> All LEAKAGE that is not identified LEAKAGE. c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except primary to secondary LEAKAGE and PRHR HX tube LEAKAGE) through a fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

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1.1 Definitions

MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE-OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3400 MWt.

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1.1 Definitions

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of gripper coils voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single assembly of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCAs not capable of being fully inserted, the reactivity worth of these assemblies must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level. In MODE 2 with $k_{eff} < 1.0$, and MODES 3, 4, and 5, the worth of verified fully inserted Gray Rod Cluster Assemblies (GRCAs) which have passed the acceptance criteria for GRCA bank worth measurements performed during startup physics testing may be included in the SDM calculation.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

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1.1 Definitions

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.
VENTED	VENTED shall be the condition when all required flow paths in ADS stage 1, 2, and 3, or alternative flow path with equivalent area, required by LCO 3.4.13, "Automatic Depressurization System (ADS) - Shutdown, RCS Open," are open.

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MODES

MODES	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	> 420
4	Safe Shutdown[(b)]	< 0.99	NA	420 ≥ T_{avg} > 200
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

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1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in Technical Specifications are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meaning.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

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1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example, the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

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1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip ... OR A.2.1 Verify ... AND A.2.2.1 Reduce ... OR A.2.2.2 Perform ... OR A.3 Align ...	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

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1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition. Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition. However, when a subsequent train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within

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1.3 Completion Times

DESCRIPTION (continued)

limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified “time zero.” This modified “time zero” may be expressed as a repetitive time (i.e., “once per 8 hours,” where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase “from discovery ....”

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1.3 Completion Times

EXAMPLES The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 in 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

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1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One valve inoperable.	A.1 Restore valve to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a valve is declared inoperable, Condition A is entered. If the valve is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion time clocks for Required Actions B.1 and B.2 start. If the inoperable valve is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second valve is declared inoperable while the first valve is still inoperable, Condition A is not re-entered for the second valve. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable valve. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable valves is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable valves is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

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1.3 Completion Times

EXAMPLES (continued)

On restoring one of the valves to OPERABLE status the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. This Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second valve being inoperable for > 7 days.

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

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1.3 Completion Times

EXAMPLES (continued)

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A.

It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

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1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours. If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

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1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

- NOTE -

Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was only applicable to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve which caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve. Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

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1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a “once per” Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hours interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

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1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each “Once per 8 hours thereafter” interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour, or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

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1.3 Completion Times

IMMEDIATE COMPLETION TIME	When “Immediately” is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

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1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The “specified Frequency” is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The “specified Frequency” consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are “otherwise stated” conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillances, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only “required” when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of “met” or “performed” in these instances conveys specific meanings. A Surveillance is “met” only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not “met.” “Performance” refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be

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1.4 Frequency

DESCRIPTION (continued)

performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discusses these special situations.

EXAMPLES The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

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1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated surveillance must be performed at least one time. Performance of the surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside the specified limits, or the Unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

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1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the surveillance must be performed within 12 hours.

The use of "Once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

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1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------- **- NOTE -** Not required to be performed until 12 hours after ≥ 25% RTP. ---------- Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

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1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------- **- NOTE -** Only required to be met in MODE 1. ----------	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an “otherwise stated” exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

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1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------- **- NOTE -** Only required to be performed in MODE 1. ---------- Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

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1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
--- **- NOTE -** Not required to be met in MODE 3. ---	
Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an “otherwise stated” exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

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2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop cold leg temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the WRB-2M DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5 the RCS pressure shall be maintained ≤ 2733.5 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

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3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and 3.0.6.

If the LCO is met, or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours; and

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified

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3.0 LCO Applicability

LCO 3.0.4 (continued)

condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the test required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.7, “Safety Function Determination Program (SFDP).” If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system’s Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7

Test Exception LCOs 3.1.8 and 3.1.10 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

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3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability of individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the Surveillance, shall be a failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as “once”, the above interval extension does not apply.

If a Completion Time requires periodic performance on a “once per...” basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, which ever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period, and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

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3.0 SR Applicability

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 2 with k_{eff} < 1.0.
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM to be within limits.	24 hours

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ±1% Δk/k of the predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Measured core reactivity not within limit.	A.1	Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	7 days
	AND		
	A.2	Establish appropriate operating restrictions and SRs.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	---------- **- NOTE -** The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ---------- Verify measured core reactivity is within ±1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling AND ---------- **- NOTE -** Only required to be performed after 60 EFPD ---------- 31 EFPD thereafter

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 for the upper MTC limit.
MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit.
MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within upper limit.	A.1 Restore MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours
C. MTC not within lower limit.	C.1 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC within upper limit.	Once prior to entering MODE 1 after each refueling

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	---------- **- NOTES -** 1. Not required to be performed provided applicable criteria in the COLR are satisfied. 2. Not required to be performed if the MTC measured at the equivalent of equilibrium RTP all rods out (ARO) boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. ---------- Verify MTC is within lower limit.	Once within 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP ARO boron concentration of 300 ppm AND 14 EFPD thereafter when MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 Each rod cluster control assembly (RCCA) shall be OPERABLE.

AND

Individual indicated rod positions of each RCCA and Gray Rod Cluster Assembly shall be within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more RCCA(s) inoperable.	A.1.1	Verify SDM to be within the limits specified in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM within limit.	1 hour
	AND		
	A.2	Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1	Restore rod to within alignment limits.	1 hour with the On-Line Power Distribution Monitoring System not monitoring parameters AND 8 hours
	OR		

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1.1 Verify SDM to be within the limits specified in the COLR. OR	1 hour
	B.2.1.2 Initiate boration to restore SDM within limit. AND	1 hour
	B.2.2 Reduce THERMAL POWER to ≤ 75% RTP. AND	2 hours
	B.2.3.1 Perform SR 3.2.5.1. OR	Once per 12 hours
	B.2.3.2.1 Verify SDM is within the limits specified in the COLR. AND	Once per 12 hours
	B.2.3.2.2 Perform SR 3.2.1.1 and SR 3.2.1.2. AND	72 hours
	B.2.3.2.3 Perform SR 3.2.2.1. AND	72 hours
	B.2.4 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1	Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	D.1.2	Initiate boration to restore required SDM to within limit.	1 hour
	AND		
	D.2	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	---------- **- NOTE -** Not required to be performed for rods associated with inoperable digital rod position indication or demand position indication. ---------- Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each RCCA not fully inserted in the core ≥ 10 steps in either direction.	92 days

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify rod drop time of each RCCA, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 500°F, and b. All reactor coolant pumps operating.	Once prior to reactor criticality after each removal of the reactor head, and after each earthquake requiring plant shutdown

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each Shutdown Bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

- NOTE -

This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	A.2	Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the insertion limits specified in the COLR.	12 hours

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1.
MODE 2 with $k_{eff} \geq 1.0$.

- NOTES -

1. This LCO is not applicable while performing SR 3.1.4.2.

2. This LCO is not applicable to Gray Rod Cluster Assembly (GRCA) banks for up to one hour during GRCA bank sequence exchange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Control Bank insertion limits not met.	A.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM to within limits.	1 hour
	AND		
	A.2	Restore control bank(s) to within insertion limits.	2 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM is within the limits specified in the COLR. OR B.1.2 Initiate boration to restore SDM to within limits. AND B.2 Restore control bank sequence and overlap to within limits.	1 hour 1 hour 2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify the estimated critical control bank position is within limits specified in the COLR.	Within 4 hours prior to achieving criticality
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	12 hours
SR 3.1.6.3	Verify sequence and overlap limits, specified in the COLR, are met for control banks not fully withdrawn from the core.	12 hours

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) for each control rod and the Bank Demand Position Indication for each group shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each inoperable DRPI and each demand position indication.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DRPI per group inoperable in one or more groups.	A.1	Verify the position of the rods with inoperable DRPI indirectly by using the incore detectors.	Once per 8 hours
	OR		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. More than one DRPI per group inoperable in one or more groups.	B.1	Place the control rods under manual control.	Immediately
	AND		
	B.2	Restore inoperable DRPI(s) to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more rods with inoperable DRPI have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1 Verify the position of the rods with inoperable DRPI indirectly by using the incore detectors. OR C.2 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours 8 hours
D. One or more demand position indication per bank inoperable in one or more banks.	D.1.1 Verify by administrative means all DRPIs for the affected banks are OPERABLE. AND D.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart. OR D.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours Once per 8 hours 8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	---------- **- NOTE -** Not required to be met for DRPI associated with a rod that does not meet LCO 3.1.4, Rod Group Alignment Limits. ---------- Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Once prior to criticality after each removal of the reactor head

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3 "Moderator Temperature Coefficient (MTC),"
LCO 3.1.4 "Rod Group Alignment Limits,"
LCO 3.1.5 "Shutdown Bank Insertion Limits,"
LCO 3.1.6 "Control Bank Insertion Limits," and
LCO 3.4.2 "RCS Minimum Temperature for Criticality"

may be suspended, and the number of required channels for LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," Functions 1, 2, 3, and 4 may be reduced to 3 provided:

a. Reactor Coolant System (RCS) lowest loop average temperature is ≥ 541°F,

b. SDM is within the limits specified in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and Associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify the RCS lowest loop average temperature is ≥ 541°F.	30 minutes
SR 3.1.8.2	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes
SR 3.1.8.3	Verify SDM is within the limits specified in the COLR.	24 hours

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves

LCO 3.1.9 Two CVS Demineralized Water Isolation Valves and two CVS Makeup Line Isolation Valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

- NOTES -

1. Flow path(s) may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CVS demineralized water isolation valve inoperable. OR One CVS makeup line isolation valve inoperable.	A.1 Restore isolation valve to OPERABLE status.	72 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Two CVS demineralized water isolation valves inoperable. OR Two CVS makeup line isolation valves inoperable.	B.1 Isolate the affected flow path(s) to the Reactor Coolant System by use of at least one closed manual or one closed and de-activated automatic valve.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify two CVS demineralized water isolation valves and two CVS makeup line isolation valves stroke closed.	In accordance with the Inservice Testing Program
SR 3.1.9.2	Verify closure time of each CVS makeup isolation valve is within limits on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.1.9.3	Verify each CVS demineralized water isolation valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 Rod Withdrawal Test Exception – MODE 5

LCO 3.1.10 During the performance of rod movement and rod drop time testing, the requirements of LCO 3.4.4, “RCS Loops,” may be suspended provided boron concentration of the reactor coolant system is greater than the all rods out (ARO) boron concentration that provides k_{eff} < 0.99.

APPLICABILITY: MODE 5 with LCO 3.4.4 not met.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Initiate action to fully insert all rods.	Immediately
	AND		
	A.2	Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.10.1	Verify boron concentration of the reactor coolant system is greater than the ARO boron concentration providing k_{eff} < 0.99.	12 hours

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3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 25% RTP and with On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------- **- NOTES -** 1. Required Action A.3 shall be completed whenever this Condition is entered prior to increasing THERMAL POWER above the limit of Required Action A.1. 2. SR 3.2.1.2 is not required to be performed if this Condition is entered prior to THERMAL POWER exceeding 75% RTP after refueling. ---------- $F_Q^C(Z)$ not within limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	AND		
	A.2	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that THERMAL POWER is limited below RTP by Required Action A.1.	72 hours after each $F_Q^C(Z)$ determination
	AND		
	A.3	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F_Q^W(Z)$ not within limits.	B.1.1 ---------- **- NOTE -** Required Action B.1.2 shall be completed if control rod motion is required to comply with the new operating space implemented by Required Action B.1.1. ---------- Restore $F_Q^W(Z)$ to within limits specified in the COLR.	4 hours
	AND	
	B.1.2 Perform SR 3.2.1.1 and SR 3.2.1.2.	72 hours
	OR	
		(continued)

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 ---------- **- NOTE -** Required Action B.2.3 shall be completed whenever Required Action B.2.1 is performed prior to increasing THERMAL POWER above the limit of Required Action B.2.1. ---------- Limit THERMAL POWER to less than RTP by amount specified in the COLR.	4 hours after each $F_Q^W(Z)$ determination
	AND	
	B.2.2 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that THERMAL POWER is limited below RTP by Required Action B.2.1.	72 hours after each $F_Q^W(Z)$ determination
	AND	
	B.2.3 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.2.1
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER < 25% RTP.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	---------- **- NOTE -** Not required to be performed if OPDMS was monitoring parameters upon exceeding 75% RTP. ---------- Verify $F_Q^C(Z)$ within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP
SR 3.2.1.2	---------- **- NOTE -** Not required to be performed if OPDMS was monitoring parameters upon exceeding 75% RTP. ---------- Verify $F_Q^W(Z)$ within limits.	Once after each refueling within 24 hours after achieving equilibrium conditions after THERMAL POWER exceeds 75% RTP

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	---------- **- NOTE -** Not required to be performed until 24 hours after OPDMS not monitoring parameters. ---------- Verify $F_Q^C(Z)$ within limit.	Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified AND 31 effective full power days (EFPD) thereafter

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.4	---------- **- NOTE -** Not required to be performed until 24 hours after OPDMS not monitoring parameters. ---------- Verify $F_Q^W(Z)$ within limits.	Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

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3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 25% RTP and with On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------- **- NOTE –** Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limits.	A.1.1 Restore $F^N_{\Delta H}$ to within limit. OR A.1.2.1 Reduce THERMAL POWER to < 50% RTP. AND A.1.2.2 Reduce Overpower ΔT trip setpoints to ≤ 55% RTP. AND A.2 Perform SR 3.2.2.1. AND	4 hours 4 hours 72 hours 24 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.3 ---------- **- NOTE -** THERMAL POWER does not have to be reduced to comply with this Required Action. ---------- Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	---------- **- NOTE -** Not required to be performed if OPDMS was monitoring parameters upon exceeding 75% RTP. ---------- Verify $F^N_{\Delta H}$ within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP
SR 3.2.2.2	---------- **- NOTE -** Not required to be performed until 24 hours after OPDMS not monitoring parameters. ---------- Verify $F^N_{\Delta H}$ within limits specified in the COLR.	31 effective full power days (EFPD)

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3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band specified in the COLR about the target flux difference.

b. May deviate outside the target band with THERMAL POWER < 90% RTP, but ≥ 50% RTP, provided AFD is within the acceptable operation limits specified in the COLR and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours.

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

- NOTES -

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

2. With THERMAL POWER ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

3. With THERMAL POWER < 50% RTP and > 15% RTP, penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of Power Range Neutron Flux channels in accordance with SR 3.3.1.4, provided AFD is maintained within acceptable operation limits.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP and with the On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP. AND AFD not within the target band.	A.1 Restore AFD to within the target band.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP	15 minutes
C. ----------NOTE---------- Required Action C.1 must be completed whenever Condition C is entered. ---------- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. OR THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes
D. Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to ≤ 15% RTP.	9 hours

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SURVEILLANCE REQUIREMENTS

- NOTE -

Not required to be performed until 7 days after the last verification of OPDMS parameters.

	SURVEILLANCE	FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days
SR 3.2.3.2	Update the target flux difference.	Once within 31 EFPD after each refueling AND 31 EFPD thereafter
SR 3.2.3.3	--- **- NOTE -** The initial target flux difference after each refueling may be determined from design predictions. --- Determine, by measurement, the target flux difference.	Once within 31 EFPD after each refueling AND 92 EFPD thereafter

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3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP and with the On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	AND		
	A.2	Perform SR 3.2.4.1.	Once per 12 hours
	AND		
	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 AND Once per 7 days thereafter
	AND		
	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 ---------- **- NOTES -** 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed. ---------- Normalize excore detectors to restore QPTR to within limit.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND	
	A.6 ---------- **- NOTE -** Perform Required Action A.6 only after Required Action A.5 is completed. ---------- Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

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SURVEILLANCE REQUIREMENTS

- NOTE -

Not required to be performed until 12 hours after the last verification of OPDMS parameters.

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	--- **- NOTES -** 1. With one Power Range Neutron Flux channel inoperable and THERMAL POWER < 75% RTP, the remaining three Power Range Neutron Flux channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --- Verify QPTR within limit by calculation.	7 days
SR 3.2.4.2	--- **- NOTE -** Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. --- Verify QPTR is within limit using a minimum of 4 symmetric pairs of fixed incore detectors.	12 hours

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3.2 POWER DISTRIBUTION LIMITS

3.2.5 On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters

LCO 3.2.5 The following parameters shall not exceed their operating limits as specified in the COLR:

a. Peak Linear Heat Rate;
b. $F_{\Delta H}^{N}$;
c. DNBR; and
d. SDM.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 25% RTP and with OPDMS monitoring parameters a, b, and c.
MODE 1 with OPDMS monitoring parameter d.
MODE 2 with k_{eff} ≥ 1.0 and OPDMS monitoring parameter d.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the parameters a. through c. above not within limits.	A.1 Restore all parameters to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours
C. Parameter d above not within limits.	C.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.5.1	Verify the parameters a. through d. to be within their limits.	24 hours

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3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 Place inoperable channel in bypass or trip.	6 hours
B. One or more Functions with two channels inoperable.	B.1 Place one inoperable channel in bypass. AND B.2 Place one inoperable channel in trip.	6 hours 6 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR One or more Functions with three or more channels inoperable.	C.1 Enter the Condition Referenced in Table 3.3.1-1 for the channel(s).	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1 Be in MODE 3.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1 Reduce THERMAL POWER to < P-10.	6 hours

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	**- NOTES -** 1. Required to be met within 12 hours after reaching 15% RTP. 2. If the calorimetric heat balance is ≥ 15% RTP, and if the nuclear instrumentation channel indicated power is: a. lower than the calorimetric measurement by > 5% RTP, then adjust the nuclear instrumentation channel upward to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required. Compare results of calorimetric heat balance to nuclear instrument channel output.	24 hours

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.1.2 ---------- **- NOTES -** 1. Adjust the conversion factor, ΔT°, in the ΔT power calculation ($q_{\Delta T}$) if absolute difference between $q_{\Delta T}$ and the calorimetric measurement is > 3% RTP. 2. Required to be met within 12 hours after reaching 50% RTP. 3. If the calorimetric heat balance is < 70% RTP, and if $q_{\Delta T}$ is: a. lower than the calorimetric measurement by > 5%, then adjust ΔT° to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required. ---------- Compare results of calorimetric heat balance to the ΔT power calculation ($q_{\Delta T}$) output.	24 hours
SR 3.3.1.3 ---------- **- NOTES -** 1. Adjust nuclear instrument channel in PMS if absolute difference is ≥ 1.5% AFD. 2. Required to be met within 24 hours after reaching 20% RTP. ---------- Compare results of the incore detector measurements to nuclear instrument channel AXIAL FLUX DIFFERENCE.	31 effective full power days (EFPD)
SR 3.3.1.4 ---------- **- NOTE -** Required to be met within 24 hours after reaching 50% RTP. ---------- Calibrate excore channels to agree with incore detector measurements.	92 EFPD

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	---------- **- NOTE -** This Surveillance shall include verification that the time constants are adjusted to within limits. ---------- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.1.6	---------- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.1.7	---------- **- NOTE -** Verification of setpoint is not required. ---------- Perform TADOT.	24 months
SR 3.3.1.8	---------- **- NOTE -** Neutron detectors are excluded from response time testing. ---------- Verify RTS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

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Table 3.3.1-1 (page 1 of 2)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS
1. Power Range Neutron Flux				
a. High Setpoint	1,2	4	D	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8
b. Low Setpoint	1[(a)],2	4	D	SR 3.3.1.6 SR 3.3.1.8
2. Power Range Neutron Flux High Positive Rate	1,2	4	D	SR 3.3.1.6 SR 3.3.1.8
3. Overtemperature ΔT	1,2	4 (2/loop)	D	SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.8
4. Overpower ΔT	1,2	4 (2/loop)	D	SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.8
5. Pressurizer Pressure				
a. Low 2 Setpoint	1[(b)]	4	E	SR 3.3.1.5 SR 3.3.1.8
b. High 2 Setpoint	1,2	4	D	SR 3.3.1.5 SR 3.3.1.8
6. Pressurizer Water Level – High 3	1[(b)]	4	E	SR 3.3.1.5 SR 3.3.1.8

(a) Below the P-10 (Power Range Neutron Flux) interlocks.

(b) Above the P-10 (Power Range Neutron Flux) interlock.

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Table 3.3.1-1 (page 2 of 2)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS
7. Reactor Coolant Flow – Low 2	1[(b)]	4 per hot leg	E	SR 3.3.1.2 SR 3.3.1.5 SR 3.3.1.8
8. Reactor Coolant Pump (RCP) Bearing Water Temperature – High 2	1,2	4 per RCP	D	SR 3.3.1.5 SR 3.3.1.8
9. RCP Speed – Low 2	1[(b)]	4 (1/pump)	E	SR 3.3.1.5 SR 3.3.1.8
10. Steam Generator (SG) Narrow Range Water Level – Low 2	1,2	4 per SG	D	SR 3.3.1.5 SR 3.3.1.8
11. Steam Generator (SG) Narrow Range Water Level – High 3	1,2[(c)]	4 per SG	D	SR 3.3.1.5 SR 3.3.1.8
12. Passive Residual Heat Removal Actuation	1,2	4 per valve	D	SR 3.3.1.7 SR 3.3.1.8

(b) Above the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-11 (Pressurizer Pressure) interlock.

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3.3 INSTRUMENTATION

3.3.2 Reactor Trip System (RTS) Source Range Instrumentation

LCO 3.3.2 Four channels of RTS Source Range Neutron Flux - High Setpoint instrumentation shall be OPERABLE.

APPLICABILITY: MODE 2 with intermediate range neutron flux below the P-6 interlock,
MODES 3, 4, and 5 with Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One channel inoperable in MODE 2.	A.1	Place inoperable channel in bypass or trip.	2 hours
B. Two channels inoperable in MODE 2.	B.1	Place one inoperable channel in bypass.	2 hours
	AND		
	B.2	Place one inoperable channel in trip.	2 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Suspend operations involving positive reactivity additions.	Immediately
D. One or two channels inoperable in MODE 3, 4, or 5.	D.1	Restore three of four channels to OPERABLE status.	48 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition D not met.	E.1 Intiate action to fully insert all rods. AND E.2 Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour 1 hour
F. Three or more channels inoperable.	F.1 Open reactor trip breakers (RTBs).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.2	---------- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.2.3	---------- **- NOTE -** Neutron detectors are excluded from response time testing. ---------- Verify RTS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

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3.3 INSTRUMENTATION

3.3.3 Reactor Trip System (RTS) Intermediate Range Instrumentation

LCO 3.3.3 Four channels of RTS Intermediate Range Neutron Flux – High instrumentation shall be OPERABLE.

APPLICABILITY: MODE 1 with Power Range Neutron Flux below the P-10 interlock, MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable with THERMAL POWER ≥ P-6.	A.1 Place one inoperable channel in bypass or trip. OR A.2 Reduce THERMAL POWER to < P-6. OR A.3 Increase THERMAL POWER to > P-10.	2 hours 2 hours 2 hours
B. Two channels inoperable with THERMAL POWER ≥ P-6.	B.1.1 Place one inoperable channel in bypass. AND B.1.2 Place one inoperable channel in trip. OR B.2 Reduce THERMAL POWER to < P-6. OR B.3 Increase THERMAL POWER to > P-10.	2 hours 2 hours 2 hours 2 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or two channels inoperable with THERMAL POWER < P-6.	C.1 Restore three of four channels to OPERABLE status.	Prior to increasing THERMAL POWER to > P-6
D. Three or more channels inoperable.	D.1 Suspend operations involving positive reactivity additions. AND D.2 Reduce THERMAL POWER to < P-6. AND D.3 Be in MODE 3.	Immediately 2 hours 7 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	---------- **- NOTE -** Not required to be met in MODE 1. ---------- Perform CHANNEL CHECK.	12 hours
SR 3.3.3.2	---------- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.3.3	---------- **- NOTE -** Neutron detectors are excluded from response time testing. ---------- Verify RTS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

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3.3 INSTRUMENTATION

3.3.4 Reactor Trip System (RTS) Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.4 The RTS ESFAS instrumentation for each Function in Table 3.3.4-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.4-1.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or two channels inoperable in MODE 1 or 2.	A.1 Restore three of four channels to OPERABLE status.	6 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more Functions with three or more channels inoperable in MODE 1 or 2.	B.1 Be in MODE 3.	6 hours
C. One or more Functions with one or two channels inoperable in MODE 3, 4, or 5.	C.1 Restore three of four channels to OPERABLE status.	48 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met. OR One or more Functions with three or more channels inoperable in MODE 3, 4, or 5.	D.1 Initiate action to fully insert all rods. AND D.2 Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour 1 hour

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.4-1 to determine to which RTS ESFAS Function the SR applies.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Verify RTS RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

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Table 3.3.4-1 (page 1 of 1)
Reactor Trip System Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Safeguards Actuation Input from Engineered Safety Feature Actuation System – Automatic	1,2	4	SR 3.3.4.1
2. ADS Stages 1, 2, and 3 Actuation Input from Engineered Safety Feature Actuation System – Automatic	1,2,3[(a)],4[(a)],5[(a)]	4	None
3. Core Makeup Tank Actuation Input from Engineered Safety Feature Actuation System – Automatic	1,2,3[(a)],4[(a)],5[(a)]	4	None

(a) With Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

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3.3 INSTRUMENTATION

3.3.5 Reactor Trip System (RTS) Manual Actuation

LCO 3.3.5 The RTS manual actuation channels for each Function in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one manual actuation channel inoperable.	A.1 Restore manual actuation channel to OPERABLE status.	48 hours
B. Required Action and associated Completion Time of Condition A not met in MODE 1 or 2. OR One or more Functions with two manual actuation channels inoperable in MODE 1 or 2.	B.1 Be in MODE 3.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 3, 4, or 5. OR One or more Functions with two manual actuation channels inoperable in MODE 3, 4, or 5.	C.1 Initiate action to fully insert all rods. AND C.2 Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	---------- **- NOTE -** Verification of setpoint is not required. ---------- Perform TADOT.	 24 months

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Table 3.3.5-1 (page 1 of 1)
Reactor Trip System Manual Actuation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS
1. Manual Reactor Trip	1,2,3[(a)],4[(a)],5[(a)]	2
2. Safeguards Actuation Input from Engineered Safety Feature Actuation System – Manual	1,2	2
3. ADS Stages 1, 2, and 3 Actuation Input from Engineered Safety Feature Actuation System – Manual	1,2,3[(a)],4[(a)],5[(a)]	2 switch sets
4. Core Makeup Tank Actuation Input from Engineered Safety Feature Actuation System – Manual	1,2,3[(a)],4[(a)],5[(a)]	2

(a) With Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

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3.3 INSTRUMENTATION

3.3.6 Reactor Trip System (RTS) Automatic Trip Logic

LCO 3.3.6 Four divisions of RTS Automatic Trip Logic shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or two divisions inoperable in MODE 1 or 2.	A.1	Restore three of four divisions to OPERABLE status.	6 hours
B.	Required Action and associated Completion Time of Condition A not met. OR Three or more divisions inoperable in MODE 1 or 2.	B.1	Be in MODE 3.	6 hours
C.	One or two divisions inoperable in MODE 3, 4, or 5.	C.1	Restore three of four divisions to OPERABLE status.	48 hours
D.	Required Action and associated Completion Time of Condition C not met. OR Three or more divisions inoperable in MODE 3, 4, or 5.	D.1 AND D.2	Initiate action to fully insert all rods. Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour 1 hour

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SURVEILLANCE REQUIREMENTS

None

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3.3 INSTRUMENTATION

3.3.7 Reactor Trip System (RTS) Trip Actuation Devices

LCO 3.3.7 Four divisions of RTS trip actuation devices for the following Functions shall be OPERABLE:

a. Reactor Trip Breakers (RTBs); and

b. Undervoltage and Shunt Trip Mechanisms on in-service RTBs.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both Functions within one division inoperable.	A.1 Open affected RTB(s) in inoperable division.	8 hours
B. One or both Functions within two divisions inoperable.	B.1 Restore one division to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1 or 2. OR One or both Functions within three or more divisions inoperable in MODE 1 or 2.	C.1 Be in MODE 3.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met in MODE 3, 4, or 5. OR One or both Functions within three or more divisions inoperable in MODE 3, 4, or 5.	D.1 Initiate action to fully insert all rods. AND D.2 Place the Plant Control System in a condition incapable of rod withdrawal.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform TADOT on both reactor trip breakers in one division.	92 days on a STAGGERED TEST BASIS

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3.3 INSTRUMENTATION

3.3.8 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.8 The ESFAS instrumentation channels for each Function in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.8-1.

ACTIONS

- NOTE -
Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 Place inoperable channel in bypass or trip.	6 hours
B. One or more Functions with two channels inoperable.	B.1 Place one inoperable channel in bypass. AND B.2 Place one inoperable channel in trip.	6 hours 6 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR One or more Functions with three or more channels inoperable.	C.1 Enter the Condition referenced in Table 3.3.8-1 for the channel(s).	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.8-1.	D.1 Be in MODE 3.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.8-1.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Reduce Reactor Coolant System (RCS) pressure below P-11 (Pressurizer Pressure) interlock.	12 hours
	AND		
	E.3	Establish RCS boron concentration greater than or equal to that necessary to meet the SDM requirements at an RCS temperature of 200°F.	12 hours
F. As required by Required Action C.1 and referenced in Table 3.3.8-1.	F.1	Be in MODE 3.	6 hours
	AND		
	F.2	Be in MODE 4 with the RCS cooling provided by the Normal Residual Heat Removal System (RNS).	24 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. As required by Required Action C.1 and referenced in Table 3.3.8-1.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2	Establish RCS boron concentration greater than or equal to that necessary to meet the SDM requirements at an RCS temperature of 200°F.	12 hours
	AND		
	G.3	Block Steamline/ Feedwater isolation and Safeguards actuations.	1 hour from discovery of RCS boron concentration greater than or equal to that necessary to meet SDM at an RCS temperature of 200°F while in MODE 3
H. As required by Required Action C.1 and referenced in Table 3.3.8-1.	H.1	Be in MODE 3.	6 hours
	AND		
	H.2	Be in MODE 5	36 hours
I. As required by Required Action C.1 and referenced in Table 3.3.8-1.	I.1	Declare affected isolation valve(s) inoperable.	Immediately

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
J. As required by Required Action C.1 and referenced in Table 3.3.8-1.	J.1	Be in MODE 5.	37 hours with three or more inoperable channels AND 180 hours
	AND		
	J.2	Initiate action to restore channel(s) to OPERABLE status.	37 hours with three or more inoperable channels AND 180 hours
K. As required by Required Action C.1 and referenced in Table 3.3.8-1.	K.1	Suspend positive reactivity additions.	Immediately
	AND		
	K.2	Initiate action to restore channel(s) to OPERABLE status.	Immediately
L. Not Used.			
M. As required by Required Action C.1 and referenced in Table 3.3.8-1.	M.1	Suspend positive reactivity additions.	Immediately
	AND		
	M.2	Be in MODE 5.	12 hours
	AND		
	M.3	Initiate action to restore channel(s) to OPERABLE status.	12 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
N. Not Used.			
O. As required by Required Action C.1 and referenced in Table 3.3.8-1.	O.1	Declare affected isolation valve(s) inoperable.	Immediately
	AND		
	O.2	Be in MODE 3.	6 hours
P. As required by Required Action C.1 and referenced in Table 3.3.8-1.	P.1	Be in MODE 3.	6 hours
	AND		
	P.2	Be in MODE 5.	36 hours
	AND		
	P.3	Initiate action to restore channel(s) to OPERABLE status.	36 hours
Q. As required by Required Action C.1 and referenced in Table 3.3.8-1.	Q.1	Be in MODE 3.	6 hours
	AND		
	Q.2	Be in MODE 4 with at least one cold leg temperature ≤ 275°F.	24 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	---------- **- NOTE -** Only required to be met for Source Range Neutron Flux Doubling. ---------- Perform CHANNEL CHECK.	12 hours
SR 3.3.8.2	---------- **- NOTES -** 1. This surveillance shall include verification that the time constants are adjusted to within limits. 2. Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.8.3	---------- **- NOTES -** 1. Not applicable to Function 1.a. 2. Neutron detectors are excluded from response time testing. ---------- Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

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Table 3.3.8-1 (page 1 of 2)
Engineered Safeguards Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
1.	Containment Pressure			
	a. – Low	1,2,3,4,5[a],6[a]	4	P
	b. – Low 2	1,2,3,4,5[a],6[a]	4	P
2.	Containment Pressure – High 2	1,2,3,4	4	H
3.	Containment Radioactivity – High	1,2,3,4[b]	4	I
4.	Containment Radioactivity – High 2	1,2,3	4	I
5.	Pressurizer Pressure – Low 3	1,2,3[c][l]	4	E
6.	Pressurizer Water Level – Low	1,2	4	D
7.	Pressurizer Water Level – Low 2	1,2,3,4[b]	4	F
		4[d],5[e]	4	J
8.	Pressurizer Water Level – High	1,2,3	4	I
9.	Pressurizer Water Level – High 2	1,2,3,4[f]	4	I
10.	Pressurizer Water Level – High 3	1,2,3,4[f]	4	Q
11.	RCS Cold Leg Temperature (T_{cold})			
	a. – Low 2	1,2,3[c][l]	4 per loop	E
	b. – High	1,2,3,4[b]	4 per loop	F
12.	Reactor Coolant Average Temperature (T_{avg}) – Low	1,2	4	D
13.	Reactor Coolant Average Temperature (T_{avg}) – Low 2	1,2	4	D
14.	RCS Wide Range Pressure – Low	1,2,3,4	4	H
		5	4	K
		6[g]	4	K

(a) Without an open containment air flow path ≥ 6 inches in diameter.

(b) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(c) Above the P-11 (Pressurizer Pressure) interlock.

(d) With the RCS being cooled by the RNS.

(e) With RCS not VENTED and CMT actuation on Pressurizer Water Level - Low 2 not blocked.

(f) With all four cold leg temperatures > 275°F.

(g) With upper internals in place.

(l) Below the P-11 (Pressurizer Pressure) interlock and RCS boron concentration is less than that necessary to meet the SDM requirements at an RCS temperature of 200°F.

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Table 3.3.8-1 (page 2 of 2)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
15. Core Makeup Tank (CMT) Level – Low 3	1,2,3,4[b]	4 per tank	F
	4[d],5[h]	4 per OPERABLE tank	J
16. CMT Level – Low 6	1,2,3,4[b]	4 per tank	F
	4[d],5[h]	4 per OPERABLE tank	J
17. Source Range Neutron Flux Doubling	3[i],4[j]	4	I
	5[j]	4	I
18. IRWST Lower Narrow Range Level – Low 3	1,2,3,4[b]	4	F
	4[d],5	4	M
	6[g]	4	K
19. Reactor Coolant Pump Bearing Water Temperature – High 2	1,2,3,4	4 per RCP	O
20. SG Narrow Range Water Level – Low 2	1,2,3,4[b]	4 per SG	F
21. SG Wide Range Water Level – Low 2	1,2,3,4[b]	4 per SG	F
22. SG Narrow Range Water Level – High	1,2,3,4	4 per SG	I
23. SG Narrow Range Water Level – High 3	1,2	4 per SG	D
	3,4	4 per SG	I
24. Steam Line Pressure – Low 2	1,2,3[c][l][m]	4 per steam line	G
25. Steam Line Pressure – Negative Rate – High	3[k]	4 per steam line	I

(b) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(c) Above the P-11 (Pressurizer Pressure) interlock.

(d) With the RCS being cooled by the RNS.

(g) With upper internals in place.

(h) With RCS not VENTED.

(i) With unborated water source flow paths not isolated except during intentional approach to criticality.

(j) With unborated water source flow paths not isolated.

(k) Below the P-11 (Pressurizer Pressure) interlock when Steam Line Pressure – Low 2 is blocked.

(l) Below the P-11 (Pressurizer Pressure) interlock and RCS boron concentration is less than that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(m) Below the P-11 (Pressurizer Pressure) interlock when Steam Line Pressure – Low 2 is not blocked.

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3.3 INSTRUMENTATION

3.3.9 Engineered Safety Feature Actuation System (ESFAS) Manual Initiation

LCO 3.3.9 The ESFAS manual initiation channels for each Function in Table 3.3.9-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.9-1.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --- **- NOTE -** Not applicable to Functions 1, 6, 7, 8, 12, and 13 in MODE 5 or 6. --- One or more Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. --- **- NOTE -** Only applicable to Functions 1, 6, 7, 8, 12, and 13 in MODE 5 or 6. --- One or more Functions with one channel inoperable.	B.1 Restore channel to OPERABLE status.	72 hours

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ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
C.	Required Action and associated Completion Time of Condition A or B not met. OR One or more Functions with two channels inoperable.	C.1	Enter the Condition referenced in Table 3.3.9-1 for the channel(s).	Immediately
D.	As required by Required Action C.1 and referenced in Table 3.3.9-1.	D.1 AND D.2	Be in MODE 3. Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS).	6 hours 24 hours
E.	As required by Required Action C.1 and referenced in Table 3.3.9-1.	E.1 AND E.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours
F.	As required by Required Action C.1 and referenced in Table 3.3.9-1.	F.1	Declare affected isolation valve(s) inoperable.	Immediately
G.	As required by Required Action C.1 and referenced in Table 3.3.9-1.	G.1 AND G.2	Be in MODE 5. Initiate action to restore channel(s) to OPERABLE status.	12 hours 12 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. As required by Required Action C.1 and referenced in Table 3.3.9-1.	H.1	Suspend positive reactivity additions.	Immediately
	AND		
	H.2	Initiate action to restore channel(s) to OPERABLE status.	Immediately
I. Not Used.			
J. As required by Required Action C.1 and referenced in Table 3.3.9-1.	J.1	Suspend positive reactivity additions.	Immediately
	AND		
	J.2	Be in MODE 5.	12 hours
	AND		
	J.3	Initiate action to restore channel(s) to OPERABLE status.	12 hours
K. Not Used.			
L. As required by Required Action C.1 and referenced in Table 3.3.9-1.	L.1	Be in MODE 3.	6 hours
	AND		
	L.2	Be in MODE 5.	36 hours
	AND		
	L.3	Initiate action to restore channel(s) to OPERABLE status.	36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	---------- **- NOTE -** Verification of setpoint not required . ---------- Perform TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT).	24 months

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Table 3.3.9-1 (page 1 of 2)
Engineered Safeguards Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
1.	Safeguards Actuation - Manual Initiation	1,2,3,4	2 switches	E
		5	2 switches	J
2.	Core Makeup Tank (CMT) Actuation - Manual Initiation	1,2,3,4[(a)]	2 switches	D
		4[(b)], 5[(d)]	2 switches	G
3.	Containment Isolation - Manual Initiation	1,2,3,4	2 switches	E
4.	Steam Line Isolation - Manual Initiation	1,2,3,4	2 switches	F
5.	Feedwater Isolation - Manual Initiation	1,2,3,4	2 switches	F
6.	ADS Stages 1, 2 & 3 Actuation - Manual Initiation	1,2,3,4	2 switch sets	E
		5[(d)]	2 switch sets	H
7.	ADS Stage 4 Actuation - Manual Initiation	1,2,3,4	2 switch sets	E
		5	2 switch sets	H
		6[(e)]	2 switch sets	H
8.	Passive Containment Cooling Actuation - Manual Initiation	1,2,3,4	2 switches	E
		5[(f)]	2 switches	J
		6[(f)]	2 switches	H
9.	Passive Residual Heat Removal Heat Exchanger Actuation - Manual Initiation	1,2,3,4	2 Switches	E
		5[(c)]	2 switches	G
10.	Chemical and Volume Control System Makeup Isolation - Manual Initiation	1,2,3,4[(h)]	2 switches	F
11.	Normal Residual Heat Removal System Isolation - Manual Initiation	1,2,3	2 switch sets	F

(a) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(b) With the RCS being cooled by the RNS.

(c) With the RCS pressure boundary intact.

(d) With RCS not VENTED.

(e) With upper internals in place.

(f) With decay heat > 7.0 MWt.

(h) With all four cold leg temperatures > 275°F.

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Table 3.3.9-1 (page 2 of 2)
Engineered Safeguards Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
12.	In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valve Actuation – Manual Initiation	1,2,3,4[(a)]	2 switch sets	D
		4[(b)],5	2 switch sets	J
		6	2 switch sets	H
13.	IRWST Containment Recirculation Valve Actuation – Manual Initiation	1,2,3,4[(a)]	2 switch sets	D
		4[(b)],5	2 switch sets	J
		6	2 switch sets	H
14.	SG Power Operated Relief Valve and Block Valve Isolation – Manual Initiation	1,2,3,4[(a)]	2 switches	D
15.	Containment Vacuum Relief Valve Actuation – Manual Initiation	1,2,3,4,5[(g)],6[(g)]	2 switches	L

(a) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(b) With the RCS being cooled by the RNS.

(g) Without an open containment air flow path ≥ 6 inches in diameter.

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3.3 INSTRUMENTATION

3.3.10 Engineered Safety Feature Actuation System (ESFAS) Reactor Coolant System (RCS) Hot Leg Level Instrumentation

LCO 3.3.10 The ESFAS RCS Hot Leg Level instrumentation channels for each function in Table 3.3.10-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.10-1.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place inoperable channel in bypass. AND A.2 ---------- **- NOTE -** Only applicable to Function 1. ---------- Continuously monitor hot leg level.	6 hours 6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Enter the Condition referenced in Table 3.3.10-1 for the channel.	Immediately

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action B.1 and referenced in Table 3.3.10-1.	C.1 Suspend positive reactivity additions. AND C.2 Initiate action to restore channel(s) to OPERABLE status.	Immediately Immediately
D. As required by Required Action B.1 and referenced in Table 3.3.10-1.	---------- **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. ---------- D.1.1 Isolate the affected flow path(s). AND D.1.2.1 Isolate the affected flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. OR D.1.2.2 Verify the affected flow path is isolated OR D.2 Initiate action to restore channel(s) to OPERABLE status.	 24 hours 7 days Once per 7 days 12 hours
E. As required by Required Action B.1 and referenced in Table 3.3.10-1.	E.1 Initiate action to restore channel(s) to OPERABLE status.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	---------- **- NOTE -** This surveillance shall include verification that the time constants are adjusted to within limits. ---------- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.10.2	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

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Table 3.3.10-1 (page 1 of 1)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
1. Hot Leg Level – Low 4	5[(a)],6[(b)]	1 per loop	C
2. Hot Leg Level – Low 2	5[(c)]	1 per loop	D
	6[(d)]	1 per loop	E

(a) With CMT actuation on Pressurizer Water Level - Low 2 blocked.

(b) With upper internals in place and with CMT actuation on Pressurizer Water Level - Low 2 blocked.

(c) Below the P-12 (Pressurizer Level) interlock.

(d) With the water level < 23 feet above the top of the reactor vessel flange.

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3.3 INSTRUMENTATION

3.3.11 Engineered Safety Feature Actuation System (ESFAS) Startup Feedwater Flow Instrumentation

LCO 3.3.11 Two channels of ESFAS Startup Feedwater Flow - Low 2 instrumentation for each startup feedwater line shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with the Reactor Coolant System (RCS) not being cooled by the Normal Residual Heat Removal Sysem (RNS).

ACTIONS

- NOTE -

Separate condition entry is allowed for each startup feedwater line.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more startup feedwater lines with one channel inoperable.	A.1 Place channel in trip.	6 hours
B. Required Action and associated Completion Time of Condition A not met. OR One or more startup feedwater lines with two channels inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with the RCS cooling provided by the RNS.	6 hours 24 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.11.1	---------- **- NOTE -** This surveillance shall include verification that the time constants are adjusted to within limits. ---------- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.11.2	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

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3.3 INSTRUMENTATION

3.3.12 Engineered Safety Feature Actuation System (ESFAS) Reactor Trip Initiation

LCO 3.3.12 Three ESFAS Reactor Trip (P-4) divisions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required division inoperable.	A.1 Restore required division to OPERABLE status.	6 hours
B. Required Action and associated Completion Time of Condition A not met. OR Two or three required divisions inoperable.	B.1 Declare affected isolation valve(s) inoperable. AND B.2 Be in MODE 3. AND B.3 Be in MODE 4.	Immediately 6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.12.1	Perform TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT).	24 months

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3.3 INSTRUMENTATION

3.3.13 Engineered Safety Feature Actuation System (ESFAS) Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization

LCO 3.3.13 The ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization instrumentation channels for each Function in Table 3.3.13-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.13-1.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.13-1 for the channel(s).	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.13-1.	B.1 Verify one channel OPERABLE. AND B.2 Verify main control room isolation, air supply initiation, and electrical load de-energization manual controls are functional.	Immediately 72 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.13-1.	C.1 Verify one channel OPERABLE. AND C.2 Restore channel to OPERABLE status.	Immediately 72 hours
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Declare VES inoperable.	Immediately
E. ---------- **- NOTE -** Separate Condition entry is allowed for each channel. ---------- As required by Required Action A.1 and referenced in Table 3.3.13-1.	E.1 Place inoperable channel in trip. OR E.2.1 Verify actuation capability is maintained. AND E.2.2 Restore channel to OPERABLE status.	6 hours 6 hours 7 days
F. Required Action and accociated Completion time of Condition E not met.	F.1 De-energize both MCR air supply radiation monitor sample pumps.	6 hours

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SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.13-1 to determine which SRs apply for each ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization Function.

SURVEILLANCE		FREQUENCY
SR 3.3.13.1	--- **- NOTE -** This surveillance shall include verification that the time constants are adjusted to within limits. --- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.13.2	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

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Table 3.3.13-1 (page 1 of 1)
ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-Energization Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS
1. Main Control Room Air Supply Iodine or Particulate Radiation – High 2	1[(a)],2[(a)],3[(a)],4[(a)]	2	B	SR 3.3.13.1 SR 3.3.13.2
	(b)[(a)]	2	C	SR 3.3.13.1 SR 3.3.13.2
2. Main Control Room Differential Pressure – Low	1,2,3,4	2	B	SR 3.3.13.1 SR 3.3.13.2
	(b)	2	C	SR 3.3.13.1 SR 3.3.13.2
3. Class 1E 24-Hour Battery Charger Undervoltage	1,2,3,4,5,6,(b)	2/24-hour battery charger	E	SR 3.3.13.1

(a) Not applicable for the Main Control Room Air Supply Iodine or Particulate Radiation - High 2 function when the Main Control Room Envelope is isolated and the Main Control Room Emergency Habitability System is in operation.

(b) During movement of irradiated fuel assemblies.

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3.3 INSTRUMENTATION

3.3.14 Engineered Safety Feature Actuation System (ESFAS) In-containment Refueling Water Storage Tank (IRWST) and Spent Fuel Pool Level Instrumentation

LCO 3.3.14 The ESFAS IRWST and Spent Fuel Pool Level instrumentation channels for each Function in Table 3.3.14-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.14-1.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Spent Fuel Pool Level – Low 2 channel inoperable.	A.1 Place channel in trip.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more IRWST Wide Range Level – Low channels inoperable. OR Two or more Spent Fuel Pool Level – Low 2 channels inoperable.	---------- **- NOTE -** Penetration flow path(s) may be unisolated intermittently under administrative controls. ---------- B.1 Isolate affected penetration flow paths. AND B.2.1 Isolate affected penetration flow paths by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. OR B.2.2 Verify affected penetration flow paths are isolated.	24 hours 7 days Once per 7 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Declare the IRWST inoperable.	Immediately

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.14.1	---------- **- NOTE -** This surveillance shall include verification that the time constants are adjusted to within limits. ---------- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.14.2	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

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Table 3.3.14-1 (page 1 of 1)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS
1. Spent Fuel Pool Level - Low 2	6[(a)]	3
2. IRWST Wide Range Level - Low	1,2,3,4	2

(a) With refueling cavity and spent fuel pool volumes in communication.

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3.3 INSTRUMENTATION

3.3.15 Engineered Safety Feature Actuation System (ESFAS) Actuation Logic – Operating

LCO 3.3.15 Four divisions with one subsystem for each of the following Functions shall be OPERABLE:

a. ESF Coincidence Logic; and

b. ESF Actuation.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions within one division inoperable.	A.1 Restore division to OPERABLE status.	6 hours
B. Required Action and associated Completion Time of Condition A not met. OR One or more Functions within two or more divisions inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

None

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3.3 INSTRUMENTATION

3.3.16 Engineered Safety Feature Actuation System (ESFAS) Actuation Logic – Shutdown

LCO 3.3.16 Four divisions with one subsystem for each of the following Functions shall be OPERABLE:

a. ESF Coincidence Logic; and

b. ESF Actuation.

- NOTE -

Only the divisions necessary to support Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization are required to be OPERABLE during movement of irradiated fuel assemblies when not in MODE 1, 2, 3, 4, 5, or 6.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions within one required division inoperable.	A.1 Restore required division to OPERABLE status.	72 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met in MODE 5. OR One or more Functions within two or more divisions inoperable in MODE 5.	B.1 Suspend positive reactivity additions. AND B.2 Initiate action to restore Function(s) to OPERABLE status. AND B.3 Initiate action to isolate the flow path from the demineralized water storage tank to the RCS by use of at least one closed and de-activated automatic valve or closed manual valve.	Immediately Immediately Immediately
C. Required Action and associated Completion Time of Condition A not met in MODE 6. OR One or more Functions within two or more divisions inoperable in MODE 6.	C.1 Suspend positive reactivity additions. AND C.2 Initiate action to restore Function(s) to OPERABLE status.	Immediately Immediately

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies. OR One or more Functions within two or more required divisions inoperable during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

None

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3.3 INSTRUMENTATION

3.3.17 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.17 The PAM instrumentation for each Function in Table 3.3.17-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.5.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.17-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.17-1.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

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SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.17-1 to determine which SRs apply for each PAM Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.17.1	--- **- NOTE -** Not required to be met for Neutron Flux (Intermdiate Range) in MODE 1. --- Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.17.2	--- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	24 months

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Table 3.3.17-1 (page 1 of 1)
Post-Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITION REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS
1.	Neutron Flux (Intermediate Range)	2	E	SR 3.3.17.1 SR 3.3.17.2
2.	Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	2	E	SR 3.3.17.2
3.	RCS Cold Leg Temperature (Wide Range)	2	E	SR 3.3.17.2
4.	RCS Pressure (Wide Range)	2	E	SR 3.3.17.2
5.	RCS Subcooling	2	E	SR 3.3.17.2
6.	Containment Water Level	2	E	SR 3.3.17.2
7.	Containment Pressure	2	E	SR 3.3.17.2
8.	Containment Pressure (Extended Range)	2	E	SR 3.3.17.2
9.	Containment Area Radiation (High Range)	2	E	SR 3.3.17.2
10.	Pressurizer Level and Associated Reference Leg Temperature	2	E	SR 3.3.17.2
11.	In-Containment Refueling Water Storage Tank (IRWST) Wide Range Water Level	2	E	SR 3.3.17.2
12.	Passive Residual Heat Removal (PRHR) Heat Removal	2	E	SR 3.3.17.1 SR 3.3.17.2
13.	Core Exit Temperature -- Quadrant 1	2[(a)]	E	SR 3.3.17.2
14.	Core Exit Temperature -- Quadrant 2	2[(a)]	E	SR 3.3.17.2
15.	Core Exit Temperature -- Quadrant 3	2[(a)]	E	SR 3.3.17.2
16.	Core Exit Temperature -- Quadrant 4	2[(a)]	E	SR 3.3.17.2
17.	Passive Containment Cooling System (PCS) Heat Removal	2	E	SR 3.3.17.1 SR 3.3.17.2
18.	Penetration Flow Path Remotely Operated Containment Isolation Valve Position	2 per penetration flow path[(b)(c)(d)]	E	SR 3.3.17.1 SR 3.3.17.2
19.	IRWST to Normal Residual Heat Removal System (RNS) Suction Valve Status	2	E	SR 3.3.17.1 SR 3.3.17.2
20.	Pressurizer Pressure	2	E	SR 3.3.17.2

(a) A channel consists of two thermocouples within a single division. Each quadrant contains two divisions. The minimum requirement is two OPERABLE thermocouples in each of the two divisions.

(b) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(c) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(d) Penetration Flow Path Remotely Operated Containment Isolation Valve Position applies to components that receive the ESF containment isolation signal (T signal).

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3.3 INSTRUMENTATION

3.3.18 Remote Shutdown Workstation (RSW)

LCO 3.3.18 The RSW shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with Reactor Coolant System (RCS) average temperature (T_{avg}) ≥ 350°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RSW inoperable.	A.1 Restore to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with T_{avg} < 350°F.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.18.1	Verify each required transfer switch is capable of performing the required function.	24 months
SR 3.3.18.2	Verify the RSW communicates indication and controls with Division A, B, C and D of the Protection and Safety Monitoring System (PMS).	24 months
SR 3.3.18.3	Verify the OPERABILITY of the RSW hardware and software.	24 months

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.18.4	Perform TADOT of the reactor trip breaker open/closed indication.	24 months

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3.3 INSTRUMENTATION

3.3.19 Diverse Actuation System (DAS) Manual Controls

LCO 3.3.19 The DAS manual controls for each function in Table 3.3.19-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.19-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more manual DAS controls inoperable.	A.1 Restore DAS manual controls to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met for inoperable DAS manual reactor trip control.	B.1 Perform SR 3.3.7.1. AND B.2 Restore all controls to OPERABLE status.	Once per 31 days on a STAGGERED TEST BASIS Prior to entering MODE 2 following next MODE 5 entry
C. Required Action and associated Completion Time of Condition A not met for inoperable DAS manual actuation control other than reactor trip.	C.1 Restore all controls to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B not met. OR Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.19.1	---------- **- NOTE -** Verification of setpoint not required. ---------- Perform TRIP ACTUATION DEVICE OPERATIONAL TEST (TADOT).	 24 months

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Table 3.3.19-1 (page 1 of 1)
DAS Manual Controls

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CONTROLS
1.	Reactor trip manual controls	1,2	2 switches
2.	Passive Residual Heat Removal Heat Exchanger (PRHR HX) control and In-Containment Refueling Water Storage Tank (IRWST) gutter control valves	1,2,3,4,5[(a)]	2 switches
3.	Core Makeup Tank (CMT) isolation valves	1,2,3,4,5[(a)]	2 switches
4.	Automatic Depressurization System (ADS) stage 1 valves	1,2,3,4,5[(a)]	2 switches
5.	ADS stage 2 valves	1,2,3,4,5[(a)]	2 switches
6.	ADS stage 3 valves	1,2,3,4,5[(a)]	2 switches
7.	ADS stage 4 valves	1,2,3,4,5,6[(c)]	2 switches
8.	IRWST injection squib valves	1,2,3,4,5,6	2 switches
9.	Containment recirculation valves	1,2,3,4,5,6	2 switches
10.	Passive containment cooling drain valves	1,2,3,4,5[(b)],6[(b)]	2 switches
11.	Selected containment isolation valves	1,2,3,4,5,6	2 switches

(a) With Reactor Coolant System (RCS) pressure boundary intact.

(b) With the reactor decay heat > 7.0 MWt.

(c) With upper internals in place.

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3.3 INSTRUMENTATION

3.3.20 Automatic Depressurization System (ADS) and In-containment Refueling Water Storage Tank (IRWST) Injection Blocking Device

LCO 3.3.20 Four divisions of ADS and IRWST Injection Blocking Device channels for each Function in Table 3.3.20-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.20-1.

ACTIONS

- NOTE -
Separate condition entry is allowed for each Division.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more divisions inoperable.	A.1 Unblock component interface module (CIM) in the affected division.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Declare affected ADS and IRWST valves inoperable.	Immediately

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SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.20-1 to determine which SRs apply for each ADS and IRWST Injection Blocking Device Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.20.1	Verify each ADS and IRWST Injection Block switch is in the “unblock” position.	7 days
SR 3.3.20.2	Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.20.3	Perform ACTUATION LOGIC TEST of ADS and IRWST Injection Blocking Devices.	24 months
SR 3.3.20.4	**- NOTE -** Verification of setpoint not required. Perform TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) of ADS and IRWST Injection Block manual switches.	24 months
SR 3.3.20.5	The following SRs of Specification 3.5.2, “Core Makeup Tanks (CMTs) – Operating” are applicable for each CMT: SR 3.5.2.2 SR 3.5.2.5 SR 3.5.2.6	In accordance with applicable SRs

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Table 3.3.20-1 (page 1 of 1)
ADS and IRWST Injection Blocking Device

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER DIVISION	SURVEILLANCE REQUIREMENTS
1.	Core Makeup Tank Level for Automatic Unblocking[a]	1,2,3,4[b]	2	SR 3.3.20.2 SR 3.3.20.3 SR 3.3.20.5
2.	ADS and IRWST Injection Block Switches for Manual Unblocking	1,2,3,4[b]	1	SR 3.3.20.3 SR 3.3.20.4
		4[c],5,6	1	SR 3.3.20.1 SR 3.3.20.3 SR 3.3.20.4

(a) Not required to be OPERABLE with associated divisional ADS and IRWST Injection Block switch in the “unblock” position.

(b) With the Reactor Coolant System (RCS) not being cooled by the Normal Residual Heat Removal System (RNS).

(c) With the RCS being cooled by the RNS.

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer Pressure is greater than or equal to the limit specified in the COLR

b. RCS Average Temperature is less than or equal to the limit specified in the COLR, and

c. RCS total flow rate ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

- NOTE -

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute, or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or more RCS DNB parameters not within limits.	A.1	Restore RCS DNB parameter(s) to within limit.	2 hours
B.	Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	Perform a CHANNEL CALIBRATION of differential pressure RCS total flow rate indication channels.	24 months
SR 3.4.1.5	---------- **- NOTE -** Not required to be performed until 24 hours after ≥ 90% RTP. ---------- Verify RCS total flow rate is ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR as determined by precision heat balance or differential pressure RCS total flow rate indication measurements.	24 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 551°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ 551°F.	12 hours

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

- NOTE -

No reactor coolant pump (RCP) shall be started with any RCS cold leg temperature ≤ 350°F unless the secondary side water temperature of each steam generator is ≤ 50°F above each of the RCS cold leg temperatures and the RCP is started at ≤ 25% of rated RCP speed.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. --- **- NOTE -** Required Action A.2 shall be completed whenever this Condition is entered. --- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1	Restore parameters to within limits.	30 minutes
	<u>AND</u>		
	A.2	Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------- **- NOTE -** Required Action C.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	---------- **- NOTE -** Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates within limits specified in the PTLR.	30 minutes

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops

LCO 3.4.4 Two RCS loops shall be OPERABLE with four Reactor Coolant Pumps (RCPs) in operation with variable speed control bypassed.

- NOTE -

All RCPs may be removed from operation in MODE 3, 4, or 5 for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------- **- NOTE -** Required Actions must be completed whenever Condition A is entered. ---------- Requirements of LCO not met in MODE 1 or 2.	A.1	Suspend start of any RCP.	Immediately
	AND		
	A.2	Be in MODE 3.	6 hours
	AND		
	A.3	Initiate action to fully insert all rods.	6 hours
	AND		
	A.4	Place the Plant Control System in a condition incapable of rod withdrawal.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ---------- **- NOTE -** Required Actions must be completed whenever Condition B is entered. ---------- Requirements of LCO not met in MODE 3, 4, or 5.	B.1	Suspend start of any RCP.	Immediately
	AND		
	B.2	Initiate action to fully insert all rods.	1 hour
	AND		
	B.3	Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop is in operation with variable speed control bypassed.	12 hours

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 Pressurizer

LCO 3.4.5 The pressurizer water level shall be ≤ 92% of span.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all four cold leg temperatures > 275°F.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Pressurizer water level not within limit.	A.1	Be in MODE 3.	6 hours
	AND		
	A.2	Initiate action to fully insert all rods.	6 hours
	AND		
	A.3	Place the Plant Control System in a condition incapable of rod withdrawal.	6 hours
	AND		
	A.4	Be in MODE 4 with at least one cold leg temperature ≤ 275°F.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify pressurizer water level ≤ 92% of span.	12 hours

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 Pressurizer Safety Valves

LCO 3.4.6 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all four cold leg temperatures > 275°F.

- NOTE -

The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. One pressurizer safety valve at a time may be inoperable for hot lift setting adjustment.

This exception is allowed for 36 hours following entry into MODE 3, provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time of Condition A not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with at least one cold leg temperature ≤ 275°F.	6 hours 24 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify each pressurizer safety valve OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ±1%.	In accordance with the Inservice Testing Program

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Operational LEAKAGE

LCO 3.4.7 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 0.5 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE from the RCS,

d. 150 gallons per day primary to secondary LEAKAGE through any one Steam Generator (SG), and

e. 500 gallons per day primary to In-Containment Refueling Water Storage Tank (IRWST) LEAKAGE through the passive residual heat removal heat exchanger (PRHR HX).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressure boundary LEAKAGE exists.	A.1 Isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve.	4 hours
B. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	B.1 Reduce LEAKAGE to within limits.	4 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met. OR Primary to secondary LEAKAGE not within limit.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	---------- **- NOTES -** 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ---------- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.7.2	---------- **- NOTE -** Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	72 hours

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Minimum RCS Flow

LCO 3.4.8 At least one Reactor Coolant Pump (RCP) shall be in operation with a total flow through the core of ≥ 3,000 gpm.

- NOTE -

a. All RCPs may be removed from operation for ≤ 1 hour per 8 hour period for the purpose of testing; or

b. With no RCPs in operation, an unborated water source through the chemical mixing tank may be unisolated under administrative controls for ≤ 1 hour for the purpose of chemical addition to the pressurizer;

provided:

i. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

ii. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODES 3, 4, and 5 with unborated water sources not isolated from the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --- **- NOTE -** Required Action A.2 shall be completed prior to starting any RCP whenever this Condition is entered. --- No RCP in operation.	A.1 Isolate all sources of unborated water. AND A.2 Perform SR 3.1.1.1.	1 hour 1 hour

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify at least one RCP is in operation with total flow through the core ≥ 3,000 gpm.	12 hours

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Leakage Detection Instrumentation

LCO 3.4.9 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Two containment sump level channels; and

b. One containment atmosphere F18 particulate monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

- NOTES -

1. The following RCS leakage detection instrumentation is not required to be OPERABLE provided SR 3.4.7.1 is performed once per 24 hours after 12 hours of steady state operation:

   a. The required containment sump level channels when In-containment Refueling Water Storage Tank (IRWST) gutter drain isolation valves are closed and for 2 hours after reopening IRWST gutter drain isolation valves; and

   b. The containment atmosphere F18 particulate monitor and required containment sump level channels when containment purge flow path is open and for 2 hours after containment purge flow path is closed.

2. The containment atmosphere F18 particulate monitor is only required to be OPERABLE in MODE 1 with THERMAL POWER > 20% RTP.

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required containment sump channel inoperable.	A.1 Restore two containment sump channels to OPERABLE status.	14 days
B. Two required containment sump channels inoperable.	B.1 ---------- **- NOTE -** Not required until 12 hours after establishment of steady state operation. ---------- Perform SR 3.4.7.1.	Once per 24 hours
	AND	
	B.2 Restore one containment sump channel to OPERABLE status.	72 hours
C. Containment atmosphere F18 particulate monitor inoperable.	C.1.1 Analyze grab samples of containment atmosphere.	Once per 24 hours
	OR	
	C.1.2 ---------- **- NOTE -** Not required until 12 hours after establishment of steady state operation. ---------- Perform SR 3.4.7.1.	Once per 24 hours
	AND	
	C.2 Restore containment atmosphere F18 particulate monitor to OPERABLE status.	30 days

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. All required monitors inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Perform a CHANNEL CHECK.	12 hours
SR 3.4.9.2	Perform a COT of containment atmosphere F18 particulate monitor.	92 days
SR 3.4.9.3	Perform a CHANNEL CALIBRATION.	24 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 RCS Specific Activity

LCO 3.4.10 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2.
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 µCi/gm.	---------- **- NOTE -** LCO 3.0.4.c is applicable. ---------- A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm. <u>AND</u> A.2 Restore DOSE EQUIVALENT I-131 to within limit.	Once per 4 hours 48 hours
B. DOSE EQUIVALENT XE-133 > 280 µCi/gm.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 > 60 µCi/gm.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 280 µCi/gm.	7 days
SR 3.4.10.2	---------- **- NOTE -** Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm.	14 days AND Between 2 to 6 hours after a THERMAL POWER change of ≥ 15% of RTP within a 1 hour period

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Automatic Depressurization System (ADS) – Operating

LCO 3.4.11 Ten ADS flow paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One flow path in ADS stage 1, 2, or 3 inoperable.	A.1 Restore flow path to OPERABLE status.	7 days
B. One flow path in ADS stage 4 inoperable.	B.1 Restore flow path to OPERABLE status.	72 hours
C. One flow path in ADS stage 1 inoperable and one flow path in ADS stage 2 or 3 inoperable. OR Two flow paths in ADS stage 1 inoperable.	C.1 Restore one flow path to OPERABLE status.	72 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met. OR Condition A and Condition B entered concurrently. OR Three or more flow paths in ADS stage 1, 2, and 3 inoperable. OR LCO not met for reasons other than Condition A, B, or C.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify the motor operated valve in series with each 4th stage ADS valve is open.	12 hours
SR 3.4.11.2	Verify each stage 1, 2, and 3 ADS valve strokes open.	In accordance with the Inservice Testing Program
SR 3.4.11.3	Verify each stage 4 ADS valve is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.4	Verify each stage 1, 2, and 3 ADS valve actuates to the open position on an actual or simulated actuation signal.	24 months
SR 3.4.11.5	---------- **- NOTE -** Squib actuation may be excluded. ---------- Verify continuity of the circuit from the Protection Logic Cabinets to each stage 4 ADS valve.	24 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Automatic Depressurization System (ADS) – Shutdown, RCS Intact

LCO 3.4.12

A. With reactor subcritical for < 28 hrs:

1. Five flow paths in ADS stage 1, 2, and 3 shall be OPERABLE; and

2. Four flow paths in ADS stage 4 shall be OPERABLE

B. With reactor subcritical for ≥ 28 hrs:

1. Three flow paths in ADS stage 1, 2, and 3, with a minimum of two flow paths in ADS stage 2 or 3, shall be OPERABLE; and

2. Three flow paths in ADS stage 4 shall be OPERABLE.

APPLICABILITY: MODE 5 with RCS pressure boundary intact and pressurizer level ≥ 20%.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One required flow path in ADS stage 1, 2, or 3 inoperable.	A.1	Restore required flow path to OPERABLE status.	7 days
B.	One required flow path in ADS stage 4 inoperable.	B.1	Restore required flow path to OPERABLE status.	72 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One required flow path in ADS stage 1 inoperable and one required flow path in ADS stage 2 or 3 inoperable. OR Two required flow paths in ADS stage 1 inoperable.	C.1 Restore one required flow path to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met. OR Condition A and Condition B entered concurrently. OR Three or more required flow paths in ADS stage 1, 2, and 3 inoperable. OR LCO not met for reasons other than Condition A, B, or C.	D.1 Initiate action to restore compliance with the LCO.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	For flow paths required to be OPERABLE, the SRs of LCO 3.4.11, “Automatic Depressurization System (ADS) – Operating” are applicable.	In accordance with applicable SRs

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 Automatic Depressurization System (ADS) – Shutdown, RCS Open

LCO 3.4.13

A. With reactor subcritical for < 28 hrs:

1. Five flow paths in ADS stage 1, 2, and 3 shall be open; and

2. Four flow paths in ADS stage 4 shall be OPERABLE.

B. With reactor subcritical for ≥ 28 hrs:

1. Three flow paths in ADS stage 1, 2, and 3, with a minimum of two flow paths in ADS stage 2 or 3, shall be open; and

2. Three flow paths in ADS stage 4 shall be OPERABLE.

- NOTE -

In MODE 5, required flow paths in ADS stage 1, 2, and 3 may be closed provided they meet OPERABILITY requirements of LCO 3.4.12, ADS – Shutdown, RCS Intact, for the following:

a. To facilitate RCS vacuum fill operations until a pressurizer level of ≥ 20% is established; or

b. To facilitate LCO compliance during transitions between LCO 3.4.12 and LCO 3.4.13.

APPLICABILITY: MODE 5 with pressurizer level < 20%,
MODE 5 with RCS pressure boundary open,
MODE 6 with upper internals in place.

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required flow path in ADS stage 1, 2, and 3 not open.	A.1 Restore required flow path in ADS stage 1, 2, and 3 to open status. OR A.2 Open alternative flow path(s) with an equivalent area.	72 hours 72 hours
B. One required flow path in ADS stage 4 inoperable.	B.1 Restore required flow path in ADS stage 4 to OPERABLE status. OR B.2 Open an alternative flow path with an equivalent area.	36 hours 36 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR Condition A and Condition B entered concurrently. OR LCO not met for reasons other than Condition A or B.	C.1 Initiate action to restore compliance with the LCO. AND C.2 Suspend positive reactivity additions.	Immediately Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	Verify each required ADS stage 1, 2, and 3 valve is in the open position.	12 hours
SR 3.4.13.2	For each required flow path in ADS stage 4, the following SRs are applicable: SR 3.4.11.1 SR 3.4.11.3 SR 3.4.11.5	In accordance with applicable SRs

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 Low Temperature Overpressure Protection (LTOP)

LCO 3.4.14 At least one of the following overpressure protection methods shall be OPERABLE, with the accumulators isolated:

a. Two Normal Residual Heat Removal System (RNS) suction relief valves and Chemical and Volume Control System (CVS) makeup line containment isolation valve, CVS-PL-V091, closed; or

b. The RCS depressurized and an RCS vent of ≥ 4.15 square inches.

- NOTE -

Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

APPLICABILITY: MODE 4 when any cold leg temperature is ≤ 275°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

- NOTE -

LCO 3.0.4b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	A.1 Isolate affected accumulator.	1 hour

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ---------- **- NOTE -** Not applicable when an RCS vent of ≥ 4.15 square inches is established. ---------- CVS-PL-V091 not closed.	B.1 Close CVS-PL-V091.	1 hour
C. Required Action and associated Completion Time of Condition A not met.	C.1 Increase RCS cold leg temperature to a level acceptable for the existing accumulator pressure allowed in the PTLR. <u>OR</u> C.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours 12 hours
D. Required LTOP method inoperable for reasons other than Condition A, B, or C.	D.1 Restore two RNS suction relief valves to OPERABLE status. <u>OR</u> D.2 Depressurize RCS and establish RCS vent of ≥ 4.15 square inches.	12 hours 12 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	---------- **- NOTE -** Only required to be met when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. ---------- Verify each accumulator is isolated.	12 hours
SR 3.4.14.2	---------- **- NOTE -** Only required to be met when complying with LCO 3.4.14.a. ---------- Verify both RNS suction isolation valves in one RNS suction flow path are open.	12 hours
SR 3.4.14.3	---------- **- NOTE -** Only required to be met when complying with LCO 3.4.14.a. ---------- Verify CVS makeup line containment isolation valve, CVS-PL-V091, is closed.	12 hours
SR 3.4.14.4	---------- **- NOTE -** Only required to be met when complying with LCO 3.4.14.b. ---------- Verify RCS vent ≥ 4.15 square inches is open.	12 hours for unlocked-open vent AND 31 days for locked-open vent

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.5	---------- **- NOTE -** Only required to be met when complying with LCO 3.4.14.a. ---------- Verify the lift setting of each RNS suction relief valve in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Pressure Isolation Valve (PIV) Integrity

LCO 3.4.15 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4, with the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

- NOTES -

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Leakage from one or more RCS PIVs not within limit.	--- **- NOTE -** Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.15.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system. --- A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. <u>AND</u>	8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Verify a second OPERABLE PIV can meet the leakage limits. This valve is required to be a check valve, or a closed valve, if it isolates a line that penetrates containment.	72 hours
B. Required Action and associated Completion Time not met.	B.1 <u>AND</u> B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Verify leakage of each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 and ≤ 2255 psig.	24 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Reactor Vessel Head Vent (RVHV)

LCO 3.4.16 The Reactor Vessel Head Vent shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all four cold leg temperatures > 275°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One flow path inoperable.	A.1 Restore flow path to OPERABLE status.	72 hours
B. Two flow paths inoperable.	B.1 Restore at least one flow path to OPERABLE status.	6 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4 with at least one cold leg temperature ≤ 275°F.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify each RVHV valve strokes open.	In accordance with the Inservice Testing Program

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Once prior to entering MODE 4 following a SG tube inspection

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Pressurizer Heater Circuit Breakers

LCO 3.4.18 Each pressurizer heater circuit breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all four cold leg temperatures > 275°F.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each pressurizer heater.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more pressurizer heaters with one circuit breaker inoperable.	A.1 Restore pressurizer heater circuit breaker to OPERABLE status.	6 hours
B. Required Action and associated Completion Time of Condition A not met. OR One or more pressurizer heaters with two circuit breakers inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with at least one cold leg temperature ≤ 275°F.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify each pressurizer heater circuit breaker trips open on an actual or simulated actuation signal, except for circuit breaker maintained in the tripped condition.	24 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Auxiliary Spray and Purification Line Isolation Valves

LCO 3.4.19 Each auxiliary spray and purification line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

- NOTE -

1. Flow path(s) may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each flow path.

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	Auxiliary spray flow path isolation valve inoperable. OR One or two purification line flow path isolation valves inoperable.	A.1	Restore isolation valve(s) to OPERABLE status.	6 hours
B.	Required Action and associated Completion Time of Condition A not met. OR Three purification line flow path isolation valves inoperable.	B.1 OR B.2	Isolate affected flow path by use of at least one closed and deactivated automatic valve. Be in MODE 3	6 hours 6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.19.1	Verify each auxiliary spray and purification line isolation valve actuate to the isolation position on an actual or simulated actuation signal.	24 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.20 Chemical and Volume Control System (CVS) Letdown Isolation Valves

LCO 3.4.20 Each CVS letdown isolation valve shall be OPERABLE.

APPLICABILITY: MODE 5 below P-12 (Pressurizer Level) interlock,
MODE 6 with water level < 23 feet above the top of the reactor vessel flange.

ACTIONS

- NOTE -

CVS letdown flow path may be unisolated intermittently under administrative controls.

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One CVS letdown isolation valve inoperable.	A.1	Restore CVS letdown isolation valve to OPERABLE status.	72 hours
B.	Required Action and associated Completion Time of Condition A is not met. OR Two CVS letdown isolation valves inoperable.	B.1	Initiate action to isolate CVS letdown flow path by use of at least one closed and deactivated automatic valve.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.20.1	Verify CVS letdown isolation valves actuate to the isolation position on an actual or simulated actuation signal.	24 months

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.1 Accumulators

LCO 3.5.1 Both accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.
MODES 3 and 4 with Reactor Coolant System (RCS) pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration outside limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	1 hour from discovery of LCO 3.5.1 Condition B entry concurrent with LCO 3.5.2 Condition C or E entry AND 8 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce RCS pressure to ≤ 1000 psig.	6 hours 12 hours
D. Two accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify the borated water volume in each accumulator is ≥ 1667 cu. ft., and ≤ 1732 cu. ft.	12 hours
SR 3.5.1.3	Verify the nitrogen cover gas pressure in each accumulator is ≥ 637 psig and ≤ 769 psig.	12 hours
SR 3.5.1.4	Verify the boron concentration in each accumulator is ≥ 2600 ppm and ≤ 2900 ppm.	31 days AND ---------- **- NOTE -** Only required for affected accumulators. ---------- Once within 6 hours after each solution volume increase of ≥ 51 cu. ft. that is not the result of addition from the in-containment refueling water storage tank
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when pressurizer pressure is ≥ 2000 psig.	31 days
SR 3.5.1.6	Verify system flow performance of each accumulator in accordance with the System Level OPERABILITY Testing Program.	10 years

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.2 Core Makeup Tanks (CMTs) – Operating

LCO 3.5.2 Both CMTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with the Reactor Coolant System (RCS) not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CMT inoperable due to one CMT outlet isolation valve inoperable.	A.1 Restore outlet isolation valve to OPERABLE status.	72 hours
B. One CMT inoperable due to water temperature or boron concentration not within limits.	B.1 Restore water temperature and boron concentration to within limits.	72 hours
C. Two CMTs inoperable due to water temperature or boron concentration not within limits.	C.1 Restore water temperature and boron concentration to within limits for one CMT.	1 hour from discovery of LCO 3.5.2 Condition C entry concurrent with LCO 3.5.1 Condition B entry AND 8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One CMT inlet line with noncondensible gas volume not within limit.	D.1 Restore CMT inlet line noncondensible gas volume to within limit.	24 hours
E. One CMT inoperable for reasons other than Condition A, B, or D.	E.1 Restore CMT to OPERABLE status.	1 hour from discovery of LCO 3.5.2 Condition E entry concurrent with LCO 3.5.1 Condition B entry AND 8 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR Two CMTs inoperable for reasons other than Condition C.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the temperature of the borated water in each CMT is < 120°F.	24 hours
SR 3.5.2.2	Verify each CMT inlet isolation valve is fully open.	12 hours
SR 3.5.2.3	Verify the volume of noncondensible gases in each CMT inlet line has not caused the high-point water level to drop below the sensor.	24 hours

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.4	Verify the boron concentration in each CMT is ≥ 3400 ppm and ≤ 4500 ppm.	31 days
SR 3.5.2.5	Verify each CMT outlet isolation valve strokes open.	In accordance with the Inservice Testing Program
SR 3.5.2.6	Verify each CMT outlet isolation valve actuates to the open position on an actual or simulated actuation signal.	24 months
SR 3.5.2.7	Verify system flow performance of each CMT in accordance with the System Level OPERABILITY Testing Program.	10 years

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.3 Core Makeup Tanks (CMTs) – Shutdown, Reactor Coolant System (RCS) Intact

LCO 3.5.3 One CMT shall be OPERABLE.

APPLICABILITY: MODE 4 with the RCS cooling provided by the Normal Residual Heat Removal System (RNS).
MODE 5 with the RCS not VENTED.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required CMT inoperable due to one outlet isolation valve inoperable.	A.1 Restore required isolation valve to OPERABLE status.	72 hours
B. Required CMT inoperable due to water temperature or boron concentration not within limits.	B.1 Restore required CMT water temperature and boron concentration to within limits.	72 hours
C. Required CMT inlet line noncondensible gas volume not within limit in MODE 4.	C.1 Restore required CMT inlet line noncondensible gas volume to within limit. OR C.2 Be in MODE 5.	24 hours 24 hours
D. Required CMT inoperable for reasons other than Condition A, B, or C.	D.1 Restore required CMT to OPERABLE status.	8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, or D not met.	E.1 Initiate action to be in MODE 5. AND	Immediately
	E.2 Initiate action to restore required CMT to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	The following SRs are applicable: SR 3.5.2.1 SR 3.5.2.2 SR 3.5.2.4 SR 3.5.2.5 SR 3.5.2.6 SR 3.5.2.7	In accordance with applicable SRs
SR 3.5.3.2	---------- **- NOTE -** Only required to be met in MODE 4 with RCS cooling provided by the RNS. ---------- The following SR is applicable: SR 3.5.2.3	In accordance with applicable SR

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.3	---------- **- NOTE -** Only required to be met in MODE 5 with RCS not VENTED. ---------- Verify the borated water volume is ≥ 2450 cu. ft.	7 days

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.4 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating

LCO 3.5.4 The PRHR HX shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with the Reactor Coolant System (RCS) not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One air operated PRHR HX outlet isolation valve inoperable.	A.1	Restore air operated PRHR HX outlet isolation valve to OPERABLE status.	72 hours
B. One air operated In-Containment Refueling Water Storage Tank (IRWST) gutter isolation valve inoperable.	B.1	Restore air operated IRWST gutter isolation valve to OPERABLE status.	72 hours
C. PRHR HX inlet line noncondensible gas volume not within limit.	C.1	Restore PRHR HX inlet line noncondensible gas volume to within limit.	24 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4 with the RCS cooling provided by the RNS.	24 hours
E. LCO not met for reasons other than Condition A, B, or C.	E.1	Restore PRHR HX to OPERABLE status.	8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition E not met.	F.1 ---------- **- NOTE -** If redundant means of providing steam generator (SG) feedwater are not available, suspend LCO 3.0.3 and all other LCO Required Actions requiring MODE changes until redundant means are available. ---------- Be in MODE 3.	6 hours from discovery of redundant means of providing SG feedwater
	AND	
	F.2 ---------- **- NOTE -** If redundant means of cooling the RCS to MODE 5 are not available, suspend LCO 3.0.3 and all other LCO Required Actions requiring MODE changes until redundant means are available. ---------- Be in MODE 5.	36 hours from discovery of redundant means of cooling the RCS to MODE 5

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify the PRHR HX outlet manual isolation valve is fully open.	12 hours
SR 3.5.4.2	Verify the PRHR HX inlet motor operated isolation valve is open.	12 hours
SR 3.5.4.3	Verify the volume of noncondensible gases in the PRHR HX inlet line has not caused the high-point water level to drop below the sensor.	24 hours
SR 3.5.4.4	---------- **- NOTE -** Only required to be met when one or more reactor coolant pumps (RCPs) are in operation. ---------- Verify one Loop 1 RCP is in operation.	12 hours
SR 3.5.4.5	Verify power is removed from the PRHR HX inlet motor operated isolation valve.	31 days
SR 3.5.4.6	Verify both PRHR HX air operated outlet isolation valves stroke open and both IRWST gutter isolation valves stroke closed.	In accordance with the Inservice Testing Program
SR 3.5.4.7	Verify by visual inspection that the IRWST gutter and downspout screens are not restricted by debris.	24 months
SR 3.5.4.8	Verify both PRHR HX air operated outlet isolation valves actuate to the open position and both IRWST gutter isolation valves actuate to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.5.4.9	Verify PRHR HX heat transfer performance in accordance with the System Level OPERABILITY Testing Program.	10 years

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.5 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, Reactor Coolant System (RCS) Intact

LCO 3.5.5 The PRHR HX shall be OPERABLE.

APPLICABILITY: MODE 4 with the RCS cooling provided by the Normal Residual Heat Removal System (RNS).
MODE 5 with the RCS pressure boundary intact and pressurizer level ≥ 20%.

- NOTE -

PRHR HX is not required to be OPERABLE in MODE 5 during RCS vacuum fill operations.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One air operated PRHR HX outlet isolation valve inoperable.	A.1 Restore air operated PRHR HX outlet valve to OPERABLE status.	72 hours
B. One air operated In-Containment Refueling Water Storage Tank (IRWST) gutter isolation valve inoperable.	B.1 Restore air operated IRWST gutter isolation valve to OPERABLE status.	72 hours
C. PRHR HX inlet line noncondensible gas volume not within limit.	C.1 Restore PRHR HX inlet line noncondensible gas volume to within limit.	24 hours
D. PRHR HX inoperable for reasons other than Condition A, B, or C.	D.1 Restore PRHR HX to OPERABLE status.	8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Initiate action to be in MODE 5. AND	Immediately
	E.2 Initiate action to restore PRHR HX to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	The SRs of Specification 3.5.4, “Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating” are applicable.	In accordance with applicable SRs

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.6 In-containment Refueling Water Storage Tank (IRWST) – Operating

LCO 3.5.6 The IRWST, with two injection flow paths and two containment recirculation flow paths, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One IRWST injection line actuation valve flow path inoperable. OR One containment recirculation line actuation valve flow path inoperable.	A.1 Restore the inoperable actuation valve flow path to OPERABLE status.	72 hours
B. One IRWST injection flow path with noncondensible gas volume in one squib valve outlet line pipe stub not within limit.	B.1 Restore noncondensible gas volume in squib valve outlet line pipe stub to within limit.	72 hours
C. One IRWST injection flow path with noncondensible gas volume in both squib valve outlet line pipe stubs not within limit.	C.1 Restore noncondensible gas volume in one squib valve outlet line pipe stub to within limit.	8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. IRWST boron concentration not within limits. OR IRWST borated water temperature not within limits. OR IRWST borated water volume < 73,100 cu. ft and ≥ 70,907 cu. ft.	D.1 Restore IRWST to OPERABLE status.	8 hours
E. One motor operated IRWST isolation valve not fully open. OR Power is not removed from one or more motor operated IRWST isolation valves.	E.1 Restore motor operated IRWST isolation valve to fully open condition with power removed from both valves.	1 hour
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR LCO not met for reasons other than Condition A, B, C, D, or E.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.6.1	Verify the IRWST water temperature is < 120°F.	24 hours
SR 3.5.6.2	Verify the IRWST borated water volume is ≥ 73,100 cu. ft.	24 hours
SR 3.5.6.3	Verify the volume of noncondensible gases in each of the four IRWST injection squib valve outlet line pipe stubs has not caused the high-point water level to drop below the sensor.	24 hours
SR 3.5.6.4	Verify the IRWST boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days AND Once within 6 hours after each solution volume increase of ≥ 15,000 gal
SR 3.5.6.5	Verify each motor operated IRWST isolation valve is fully open.	12 hours
SR 3.5.6.6	Verify power is removed from each motor operated IRWST isolation valve.	31 days
SR 3.5.6.7	Verify each motor operated containment recirculation isolation valve is fully open.	31 days
SR 3.5.6.8	Verify each IRWST injection and containment recirculation squib valve is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.5.6.9	---------- - NOTE - Squib actuation may be excluded. ---------- Verify continuity of the circuit from the Protection Logic Cabinets to each IRWST injection and containment recirculation squib valve on an actual or simulated actuation signal.	24 months

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.6.10	Verify by visual inspection that the IRWST screens and the containment recirculation screens are not restricted by debris.	24 months
SR 3.5.6.11	Verify IRWST injection and recirculation system flow performance in accordance with the System Level OPERABILITY Testing Program.	10 years

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.7 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5

LCO 3.5.7 The IRWST, with one injection flow path and one containment recirculation flow path, shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required motor operated containment recirculation isolation valve not fully open.	A.1 Open required motor operated containment recirculation isolation valve.	72 hours
B. Required IRWST injection flow path with noncondensible gas volume in one squib valve outlet line pipe stub not within limit.	B.1 Restore noncondensible gas volume in squib valve outlet line pipe stub to within limit.	72 hours
C. Required IRWST injection flow path with noncondensible gas volume in both squib valve outlet line pipe stubs not within limit.	C.1 Restore noncondensible gas volume in one squib valve outlet line pipe stub to within limit.	8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. IRWST boron concentration not within limits. OR IRWST borated water temperature not within limits. OR IRWST borated water volume < 73,100 cu. ft. and ≥ 70,907 cu ft.	D.1 Restore IRWST to OPERABLE status.	8 hours
E. Required motor operated IRWST isolation valve not fully open. OR Power is not removed from required motor operated IRWST isolation valve.	E.1 Restore required motor operated IRWST isolation valve to fully open condition with power removed.	1 hour
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR LCO not met for reasons other than Condition A, B, C, D, or E.	F.1 Initiate action to restore IRWST to OPERABLE status. AND F.2 Suspend positive reactivity additions.	Immediately Immediately

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.7.1	---------- **- NOTE -** Penetration flow path(s) may be unisolated intermittently under administrative controls. ---------- Verify Spent Fuel Pool Cooling System containment isolation valves are closed.	31 days
SR 3.5.7.2	For the IRWST and flow paths required to be OPERABLE, the following SRs are applicable: SR 3.5.6.1 SR 3.5.6.2 SR 3.5.6.4 SR 3.5.6.5 SR 3.5.6.6 SR 3.5.6.7 SR 3.5.6.8 SR 3.5.6.9 SR 3.5.6.10 SR 3.5.6.11	In accordance with applicable SRs
SR 3.5.7.3	---------- **- NOTE -** Not required to be met during RCS vacuum fill operations. ---------- For the IRWST and flow paths required to be OPERABLE, the following SR is applicable: SR 3.5.6.3	In accordance with applicable SR

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.8 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6

LCO 3.5.8 The IRWST, with one injection flow path and one containment recirculation flow path, shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required motor operated containment recirculation isolation valve not fully open.	A.1 Open required motor operated containment recirculation isolation valve.	72 hours
B. Required IRWST injection flow path with noncondensible gas volume in one squib valve outlet line pipe stub not within limit.	B.1 Restore noncondensible gas volume in squib valve outlet line pipe stub to within limit.	72 hours
C. Required IRWST injection flow path with noncondensible gas volume in both squib valve outlet line pipe stubs not within limit.	C.1 Restore noncondensible gas volume in one squib valve outlet line pipe stub to within limit.	8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. IRWST and refueling cavity boron concentration not within limits. OR IRWST and refueling cavity borated water temperature not within limits. OR IRWST and refueling cavity borated water volume < 73,100 cu. ft and ≥ 70,907 cu. ft.	D.1 Restore IRWST to OPERABLE status.	8 hours
E. Required motor operated IRWST isolation valve not fully open. OR Power is not removed from required motor operated IRWST isolation valve.	E.1 Restore required motor operated IRWST isolation valve to fully open condition with power removed.	1 hour
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR LCO not met for reasons other than Condition A, B, C, D, or E.	F.1 Initiate action to restore IRWST to OPERABLE status. AND F.2 Suspend positive reactivity additions.	Immediately Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.8.1	Verify the IRWST and refueling cavity water temperature is < 120°F.	24 hours
SR 3.5.8.2	Verify the IRWST and refueling cavity water total borated water volume is ≥ 73,100 cu. ft.	24 hours
SR 3.5.8.3	---------- **- NOTES -** 1. Penetration flow path(s) may be unisolated intermittently under administrative controls. 2. Only required to be met with refueling cavity and spent fuel pool volumes not in communication. ---------- Verify Spent Fuel Pool Cooling System containment isolation valves are closed.	31 days
SR 3.5.8.4	Verify the IRWST and refueling cavity boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days AND Once within 6 hours after each solution volume increase of ≥ 15,000 gal
SR 3.5.8.5	For the IRWST and flow paths required to be OPERABLE, the following SRs are applicable: SR 3.5.6.3 SR 3.5.6.6 SR 3.5.6.8 SR 3.5.6.10 SR 3.5.6.5 SR 3.5.6.7 SR 3.5.6.9 SR 3.5.6.11	In accordance with applicable SRs

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3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.9 Reactor Coolant Pump (RCP) Breakers

LCO 3.5.9 Two RCP breakers for each RCP shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
MODE 5 with the RCS not VENTED.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each RCP.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCPs with one breaker inoperable in MODE 1, 2, 3, or 4.	A.1 Restore RCP breaker to OPERABLE status.	6 hours
B. Required Action and associated Completion Time of Condition A is not met. OR One or more RCPs with two breakers inoperable in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. One or more RCPs with one breaker inoperable in MODE 5.	C.1 Restore RCP breaker to OPERABLE status.	72 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C is not met. OR One or more RCPs with two breakers inoperable in MODE 5.	D.1 Initiate action to trip affected RCP breaker(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.9.1	Verify each RCP breaker trips open on an actual or simulated actuation signal, except for breaker maintained in the tripped condition.	24 months

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3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage-rate testing except for containment air-lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program

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3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTES -

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, “Containment,” when air lock leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	--- **- NOTES -** 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. --- A.1 Verify the OPERABLE door is closed in the affected air lock. <u>AND</u>	1 hour

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2 Lock the OPERABLE door closed in the affected air lock. AND	24 hours
	A.3 ---------- **- NOTE -** Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	---------- **- NOTES -** 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. ---------- B.1 Verify an OPERABLE door is closed in the affected air lock. AND	1 hour

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Lock an OPERABLE door closed in the affected air lock. AND	24 hours
	B.3	---------- **- NOTE -** Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. AND	Immediately
	C.2	Verify a door is closed in the affected air lock. AND	1 hour
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3. AND	6 hours
	D.2	Be in MODE 5.	36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	---------- **- NOTES -** 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	24 months

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3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE, except for containment isolation valves associated with closed systems and for vacuum relief valves.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTES -

1. Penetration flow path(s) may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, “Containment,” when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one containment isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	---------- **- NOTES -** 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. One or more penetration flow paths with two containment isolation valves inoperable.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. Required Action and associated Completion Time not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.1	Verify each 16 inch containment purge valve is closed, except when the 16 inch containment purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances which require the valves to be open.	31 days
SR 3.6.3.2	--- **- NOTE -** Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	--- **- NOTE -** Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months

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3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.2 psig and ≤ +1.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.
MODES 5 and 6 without an open containment air flow path ≥ 6 inches in diameter.

- NOTE -

The high pressure LCO limit is not applicable in MODES 5 or 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condtion A not met in MODE 5 or 6.	C.1 Initiate action to restore containment pressure to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

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3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4,
MODES 5 and 6 with both containment equipment hatches and both containment airlocks closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6.	C.1 Initiate action to restore containment average air temperature to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	12 hours

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3.6 CONTAINMENT SYSTEMS

3.6.6 Passive Containment Cooling System (PCS)

LCO 3.6.6 The passive containment cooling system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
MODES 5 and 6 with the reactor decay heat > 7.0 MWt.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One passive containment cooling water flow path inoperable.	A.1 Restore flow path to OPERABLE status.	7 days
B. Two passive containment cooling water flow paths inoperable.	B.1 Restore one flow path to OPERABLE status.	72 hours
C. One or more water storage tank parameters not within limits.	C.1 Restore water storage tank to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4. OR LCO not met for reasons other than Condition A, B, or C in MODE 1, 2, 3, or 4.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 84 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion time of Condition A, B, or C not met in MODE 5 or 6. OR LCO not met for reasons other than Condition A, B, or C in MODE 5 or 6.	E.1 Initiate action to restore passive containment cooling to OPERABLE status. AND E.2 Suspend positive reactivity additions.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify the water storage tank temperature ≥ 40°F and ≤ 120°F.	24 hours
SR 3.6.6.2	Verify the water storage tank volume ≥ 756,700 gallons.	7 days
SR 3.6.6.3	Verify each passive containment cooling system manual, power operated, and automatic valve in each flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.6.6.4	Verify each passive containment cooling system automatic valve in each flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.5	Verify the air flow path from the shield building annulus inlet to the exit is unobstructed and, that all air baffle sections are in place.	24 months
SR 3.6.6.6	Verify passive containment cooling system flow and water coverage performance in accordance with the System Level OPERABILITY Testing Program.	At first refueling AND 10 years

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3.6 CONTAINMENT SYSTEMS

3.6.7 Containment Penetrations

LCO 3.6.7 The containment penetrations shall be in the following status:

a. The equipment hatches closed and held in place by four bolts or, if open, can be closed prior to steaming into the containment.

b. One door in each air lock closed or, if open, can be closed prior to steaming into the containment.

c. The containment spare penetrations, if open, can be closed prior to steaming into the containment.

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere, if open, can be closed by a manual or automatic isolation valve, blind flange, or equivalent prior to steaming into the containment.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Restore containment penetrations to required status.	1 hour

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR LCO not met for reasons other than Condition A.	B.1 Initiate action to restore compliance with the LCO. AND B.2 Suspend positive reactivity additions.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify each required containment penetration is in the required status.	7 days
SR 3.6.7.2	---------- **- NOTE -** Only required to be met for an open equipment hatch. ---------- Verify the hardware, tools, equipment and power source necessary to close the equipment hatch are available.	 Prior to hatch removal AND 7 days

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3.6 CONTAINMENT SYSTEMS

3.6.8 pH Adjustment

LCO 3.6.8 The pH adjustment baskets shall contain ≥ 26,460 lbs of trisodium phosphate (TSP).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The weight of TSP in the pH adjustment baskets not within limit.	A.1 Restore weight of TSP in the pH adjustment baskets to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify the pH adjustment baskets contain ≥ 26,460 lbs of TSP.	24 months
SR 3.6.8.2	Verify a sample from the pH adjustment baskets provides adequate pH adjustment of the post-accident water.	24 months

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3.6 CONTAINMENT SYSTEMS

3.6.9 Vacuum Relief Valves

LCO 3.6.9 Two vacuum relief check valves and two vacuum relief isolation valves shall be OPERABLE.

AND

Containment inside to outside differential air temperature shall be ≤ 90°F.

APPLICABILITY: MODES 1, 2, 3, and 4.
MODES 5 and 6 without an open containment air flow path ≥ 6 inches in diameter.

- NOTE -

Vacuum relief valve OPERABILITY for closing is only required in MODES 1, 2, 3, and 4.

ACTIONS

- NOTE -

Enter applicable Conditions and Required Actions of LCO 3.6.1, “Containment,” when vacuum relief valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One vacuum relief check valve inoperable for opening.	A.1	Restore vacuum relief check valve to OPERABLE for opening status.	72 hours
B.	One vacuum relief isolation valve inoperable for opening.	B.1	Restore vacuum relief isolation valve to OPERABLE for opening status.	72 hours

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ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
C.	One or more vacuum relief check valves inoperable for closing. OR One or more vacuum relief isolation valves inoperable for closing.	C.1	Restore affected valve(s) to OPERABLE for closing status.	7 days
D.	One or more vacuum relief check valves inoperable for closing. AND One or more vacuum relief isolation valves inoperable for closing.	D.1 OR D.2	Restore both vacuum relief check valves to OPERABLE for closing status. Restore both vacuum relief isolation valves to OPERABLE for closing status.	1 hour 1 hour
E.	Containment inside to outside differential air temperature > 90°F.	E.1 OR E.2	Restore containment inside to outside differential air temperature to within limit. Reduce containment average temperature ≤ 80°F.	8 hours 8 hours

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ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
F.	Required Action and associated Completion Time of Condition A, B, C, D, or E not met in MODE 1, 2, 3, or 4. OR Two vacuum relief check valves inoperable for opening in MODE 1, 2, 3, or 4. OR Two vacuum relief isolation valves inoperable for opening in MODE 1, 2, 3, or 4.	F.1 AND F.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours
G.	Required Action and associated Completion Time of Condition A, B, or E not met in MODE 5 or 6. OR Two vacuum relief check valves inoperable for opening in MODE 5 or 6. OR Two vacuum relief isolation valves inoperable for opening in MODE 5 or 6.	G.1	Initiate action to restore compliance with the LCO.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.9.1	Verify containment inside to outside differential air temperature is ≤ 90°F.	12 hours
SR 3.6.9.2	---------- **- NOTES -** 1. Not required to be met for vacuum relief valves open during Surveillances. 2. Not required to be met for vacuum relief valves open when performing their vacuum relief function. ---------- Verify each vacuum relief isolation valve is closed.	31 days
SR 3.6.9.3	Verify each vacuum relief check valve and each vacuum relief isolation valve is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.6.9.4	Verify each vacuum relief isolation valve actuates on actual or simulated signals.	24 months

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3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Six MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

- NOTE -

The MSSVs are not required to be OPERABLE for opening in MODE 4 when the Reactor Coolant System (RCS) is being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

- NOTE -

Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both steam generators with one or more MSSVs inoperable for opening.	A.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. AND	4 hours

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	---------- **- NOTE -** Only required in MODE 1. ---------- Reduce the Overpower ΔT reactor trip setpoints to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours
B. One or both steam generators with one or more MSSVs inoperable for closing.	B.1	Restore MSSV to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or both steam generators with ≥ 5 MSSVs inoperable for opening.	C.1 <u>AND</u> C.2	Be in MODE 3. Be in MODE 4 with the RCS cooling provided by the RNS.	6 hours 24 hours
D. Required Action and associated Completion Time of Condition B not met.	D.1 <u>AND</u> D.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	---------- **- NOTE -** Only required to be performed in MODES 1 and 2. ---------- Verify each MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

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Table 3.7.1-1 (page 1 of 1)
OPERABLE MSSVs versus Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
5	60
4	46
3	32
2	18

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Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING (psig ± 1%)
STEAM GENERATOR		
#1	#2	
V030A	V030B	1185
V031A	V031B	1197
V032A	V032B	1209
V033A	V033B	1221
V034A	V034B	1232
V035A	V035B	1232

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3.7 PLANT SYSTEMS

3.7.2 Main Steam Line Flow Path Isolation Valves

LCO 3.7.2 Each of the following main steam line flow path isolation valves shall be OPERABLE:

a. Main steam isolation valves (MSIVs);

b. MSIV bypass valves;

c. Main steam line drain valves;

d. Turbine stop valves or turbine control valves;

e. Turbine bypass valves; and

f. Moisture separator reheater 2nd stage steam isolation valves.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore valve to OPERABLE status.	8 hours
B. One or more of the turbine stop valves and associated turbine control valves, turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 1.	B.1 Restore valve(s) to OPERABLE status.	72 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two MSIVs inoperable in MODE 1. OR One MSIV inoperable and one or more of the turbine stop valves and associated turbine control valves, turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 1. OR Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ---------- **- NOTE -** Separate Condition entry is allowed for each main steam line flow path. ---------- One or two MSIVs inoperable in MODE 2, 3, or 4. OR One or more of the turbine stop valves and associated turbine control valves, turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 2, 3, or 4.	D.1 Isolate affected main steam line flow path. AND D.2 Verify affected main steam line flow path is isolated.	8 hours Once per 7 days
E. ---------- **- NOTE -** Separate Condition entry is allowed for each penetration flow path. ---------- One or more MSIV bypass or main steam line drain valves inoperable.	---------- **- NOTE –** Penetration flow path(s) may be unisolated intermittently under administrative controls. ---------- E.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND	72 hours

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.2 ---------- **- NOTES -** 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days
F. Required Action and associated Completion Time of Condition D or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS). <u>AND</u> F.3 Be in MODE 5.	6 hours 24 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	---------- **- NOTE -** Only required to be performed prior to entry into MODE 2. ---------- Verify MSIV closure time is within limits on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.2.2	---------- **- NOTE -** Only required to be performed prior to entry into MODE 2. ---------- Verify required turbine stop, turbine control, turbine bypass, and moisture separator reheater 2nd stage steam isolation valves' closure time is within limits on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.2.3	Verify the isolation time of each MSIV bypass valve and main steam line drain isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.7.2.4	Verify each MSIV bypass valve and main steam line drain isolation valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

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3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Control Valves (MFCVs), and Main Feedwater (MFW) Pump Breakers

LCO 3.7.3 Each MFIV, MFCV, and MFW pump breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each feedwater flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both feedwater flow paths with MFIV or MFCV inoperable.	A.1 Isolate the affected flow path. AND A.2 Verify affected flow path is isolated.	72 hours Once per 7 days
B. One or both feedwater flow paths with associated MFIV and MFCV inoperable.	B.1 Isolate affected flow path.	8 hours
C. One or more feedwater flow paths with MFW pump breaker inoperable.	C.1 Restore the affected MFW pump breaker to OPERABLE status.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS).	24 hours
	AND		
	D.3	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	--- **- NOTE -** Only required to be performed prior to entry into MODE 2. --- Verify the closure time of each MFIV and MFCV is within limits on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify each MFW pump breaker trips open on an actual or simulated actuation signal, except for pump breaker maintained in the tripped condition.	24 months

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3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be ≤ 0.01 µCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify the specific activity of the secondary coolant ≤ 0.01 µCi/gm DOSE EQUIVALENT I-131.	31 days

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3.7 PLANT SYSTEMS

3.7.5 Spent Fuel Pool Water Level

LCO 3.7.5 The spent fuel pool water level shall be ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: When irradiated fuel assemblies are stored in the spent fuel pool.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level < 23 ft.	A.1 Suspend movement of irradiated fuel assemblies in the spent fuel pool. AND	Immediately
	A.2 Initiate action to restore water level to ≥ 23 ft.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify the spent fuel pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

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3.7 PLANT SYSTEMS

3.7.6 Main Control Room Emergency Habitability System (VES)

LCO 3.7.6 The VES shall be OPERABLE.

- NOTE -

The main control room envelope (MCRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --- **- NOTE -** Separate Condition entry is allowed for each valve and each damper. --- One or more valves inoperable with VES safety function maintained. <u>OR</u> One or more dampers inoperable with VES safety function maintained.	A.1 Restore valve or damper to OPERABLE status.	7 days
B. One PMS Division inoperable in one or more MCR load shed panel(s).	B.1 Restore PMS division in both MCR load shed panels to OPERABLE status.	7 days

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Thermal mass of one or more required heat sink(s) not within limit(s).	C.1	Restore required heat sink air temperatures to within limit(s).	24 hours
	AND		
	C.2	Restore thermal mass of required heat sink(s) to within limit(s).	5 days
D. VES inoperable due to inoperable MCRE boundary in MODE 1, 2, 3, or 4.	D.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	D.2	Verify mitigating actions ensure MCRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	AND		
	D.3	Restore MCRE boundary to OPERABLE status.	90 days
E. VES compressed air storage tanks compressed air volume not within limit.	E.1	Verify compressed air storage tanks contain > 245,680 scf of compressed air.	2 hours AND Once per 12 hours thereafter
	AND		
	E.2	Verify VBS MCRE ancillary fans and supporting equipment are available.	24 hours
	AND		
	E.3	Restore VES compressed air storage tanks compressed air volume to within limit.	7 days

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, or E not met. OR VES inoperable for reasons other than Condition A, B, C, D, or E.	F.1 Verify VBS available for mitigating VES inoperability. AND F.2 Restore VES to OPERABLE status.	2 hours 7 days
G. Required Action and associated Completion Time of Condition D or F not met in MODE 1, 2, 3, or 4.	G.1 Be in MODE 3. AND G.2 Be in MODE 5.	6 hours 36 hours
H. Required Action and associated Completion Time of Condition F not met during movement of irradiated fuel. OR VES inoperable due to inoperable MCRE boundary during movement of irradiated fuel.	H.1 Suspend movement of irradiated fuel assemblies.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the compressed air storage tanks contain > 327,574 scf of compressed air.	24 hours
SR 3.7.6.2	Verify thermal mass for the following heat sink locations is within limit: a. MCRE; b. Each required individual room adjacent to and below MCRE; c. Each required room-pair adjacent to and below MCRE; and d. Room above MCRE.	24 hours
SR 3.7.6.3	Operate VES for ≥ 15 minutes.	31 days on a STAGGERED TEST BASIS
SR 3.7.6.4	Verify each VES air header manual isolation valve is in an open position.	31 days
SR 3.7.6.5	Verify the air quality of the compressed air storage tanks meets the requirements of CGA G-7.1, Commodity Specification for Air, Grade E, with a pressure dew point of ≤ 40°F at ≥ 3400 psig.	92 days
SR 3.7.6.6	Verify each MCRE isolation valve is OPERABLE and will close upon receipt of an actual or simulated actuation signal, except for valves that are locked, sealed, or otherwise secured in the closed position.	24 months
SR 3.7.6.7	Verify each VES pressure relief isolation valve within the MCRE pressure boundary is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.8	Verify each VES pressure relief damper is OPERABLE.	24 months

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.6.9	Perform required MCRE unfiltered air inleakage testing in accordance with the Main Control Room Envelope Habitability Program.	In accordance with the Main Control Room Envelope Habitability Program
SR 3.7.6.10	Perform required VES Passive Filtration system filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.6.11	Verify the MCR load shed function actuates upon receipt of an actual or simulated actuation signal.	24 months
SR 3.7.6.12	Verify each VES main air delivery isolation valve actuates to the correct position upon receipt of an actual or simulated actuation signal.	24 months

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3.7 PLANT SYSTEMS

3.7.7 Startup Feedwater (SFW) Isolation Valves, Control Valves, and Pump Breakers

LCO 3.7.7 Each SFW isolation valve, control valve, and pump breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTES -

1. Flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with one inoperable valve.	A.1 Isolate the affected flow path. AND A.2 Verify affected flow path is isolated.	72 hours Once per 7 days
B. One flow path with two inoperable valves.	B.1 Isolate the affected flow path.	8 hours
C. One or more flow paths with pump breaker inoperable.	C.1 Restore the affected pump breaker to OPERABLE status.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS).	24 hours
	AND		
	D.3	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify each SFW isolation and control valve is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.7.2	Verify each SFW isolation and control valve actuates to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.7.7.3	Verify each SFW pump breaker trips open on an actual or simulated actuation signal, except for pump breaker maintained in the tripped condition.	24 months

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3.7 PLANT SYSTEMS

3.7.8 Main Steam Line Leakage

LCO 3.7.8 Main Steam Line leakage through the pipe walls inside containment shall be ≤ 0.5 gpm.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Main Steam Line leakage > 0.5 gpm.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify main steam line leakage into the containment sump ≤ 0.5 gpm.	Per SR 3.4.7.1

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3.7 PLANT SYSTEMS

3.7.9 Not used

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3.7 PLANT SYSTEMS

3.7.10 Steam Generator (SG) Isolation Valves

LCO 3.7.10 Each SG power operated relief valve (PORV), PORV block valve, and SG blowdown isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

- NOTE -

PORV OPERABILITY is not required in MODE 4 with Reactor Coolant System (RCS) being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

- NOTES -

1. SG blowdown flow path(s) may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG PORV flow paths with one isolation valve inoperable.	A.1 Isolate the flow path by use of at least one closed and deactivated automatic valve. AND	72 hours

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	---------- **- NOTES –** 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ---------- A.2 Verify the affected flow path is isolated.	Once per 31 days
B. One or more SG blowdown flow paths with one isolation valve inoperable.	B.1 Isolate the flow path by one closed valve. AND B.2 Verify the affected flow path is isolated.	72 hours Once per 7 days
C. One or more SG PORV flow paths with two isolation valves inoperable.	C.1 Isolate the affected flow path by use of at least one closed and deactivated automatic valve.	8 hours
D. One or more SG blowdown flow paths with two isolation valves inoperable.	D.1 Isolate the flow path by one closed valve.	8 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4 with the RCS cooling provided by the RNS. AND ---------- **- NOTE –** Not applicable for inoperable PORV(s). ---------- E.3 Be in MODE 5.	6 hours 24 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify each SG PORV, PORV block valve, and SG blowdown isolation valve strokes closed.	In accordance with the Inservice Testing Program
SR 3.7.10.2	Verify the isolation time of each PORV block valve and SG blowdown isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.7.10.3	Verify each SG PORV, PORV block valve, and SG blowdown isolation valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

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3.7 PLANT SYSTEMS

3.7.11 Spent Fuel Pool Boron Concentration

LCO 3.7.11 The spent fuel pool boron concentration shall be ≥ 2300 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pool. AND	Immediately
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify the spent fuel pool boron concentration is within limit.	7 days

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3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool Storage

LCO 3.7.12 The combination of initial nominal enrichment and burnup of each fuel assembly stored in Region 2 shall be within the limits specified in Figure 3.7.12-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify by administrative means the initial nominal enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.12-1.	Prior to storing the fuel assembly in Region 2

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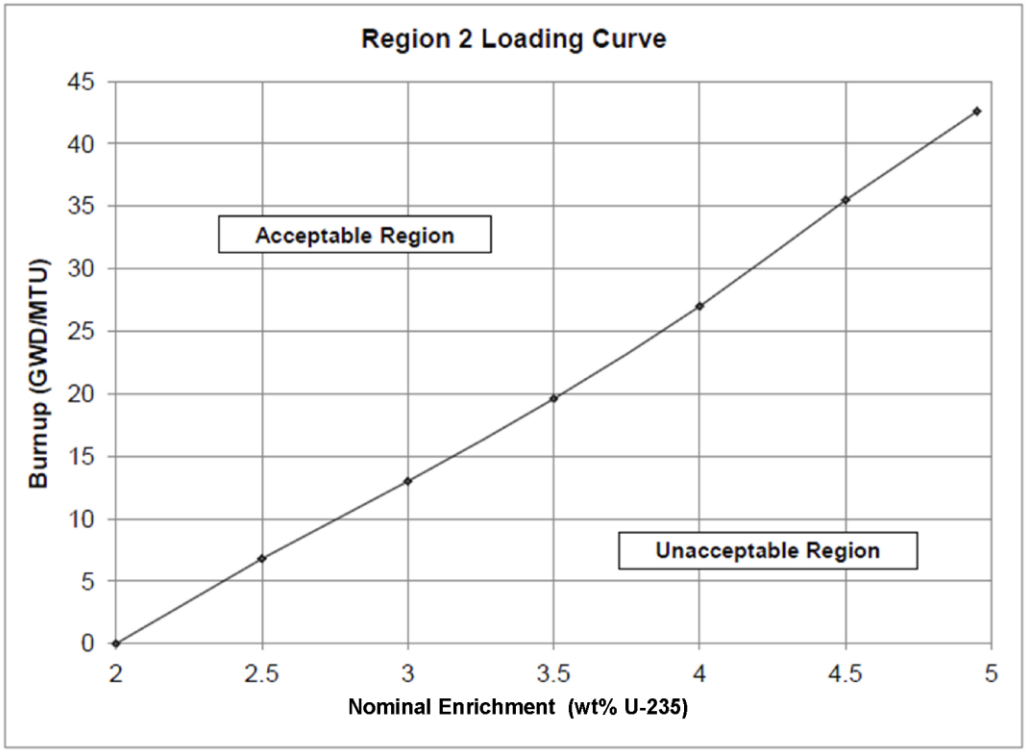

Figure 3.7.12-1

Minimum Fuel Assembly Burnup Versus
Initial Nominal Enrichment for Region 2 Spent Fuel Cells

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3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Cooling System (SFS) Containment Isolation Valves

LCO 3.7.13 The SFS containment isolation valves shall be OPERABLE.

APPLICABILITY: MODE 6 with refueling cavity and spent fuel pool volumes in communication.

ACTIONS

- NOTES -

1. Penetration flow path(s) may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each SFS penetration flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one or more SFS containment isolation valves inoperable.	A.1 Isolate affected penetration flow path. AND	24 hours
	A.2.1 Isolate affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. OR	7 days
	A.2.2 Verify affected penetration flow path is isolated.	Once per 7 days
B. Required Action and associated Completion Time not met.	B.1 Declare the In-containment Refueling Water Storage Tank inoperable.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	For SFS containment isolation valves required to be OPERABLE, the following SRs are applicable: SR 3.6.3.4 SR 3.6.3.5	In accordance with applicable SRs

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3.8 ELECTRICAL POWER SYSTEMS

3.8.1 DC Sources – Operating

LCO 3.8.1 The Division A, B, C, and D Class 1E DC power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more battery chargers in one division inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage. AND A.2 Verify battery float current ≤ 2 amps. AND A.3 Restore battery charger(s) to OPERABLE status.	6 hours Once per 24 hours 7 days
B. One or more battery chargers in two divisions inoperable.	B.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage. AND B.2 Verify battery float current ≤ 2 amps. AND B.3 Restore battery charger(s) to OPERABLE status.	2 hours Once per 24 hours 7 days

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more batteries in one division inoperable.	C.1 Restore batteries to OPERABLE status.	6 hours
D. One or more batteries in two divisions inoperable.	D.1 Restore batteries to OPERABLE status.	2 hours
E. One DC electrical power subsystem inoperable for reasons other than Condition A or C.	E.1 Restore DC electrical power subsystem to OPERABLE status.	6 hours
F. Two DC electrical power subsystems inoperable for reasons other than B or D.	F.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours
G. Required Action and associated Completion Time not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	7 days

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	Verify each battery charger supplies ≥ 150 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. OR Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	24 months
SR 3.8.1.3	---------- **- NOTE -** The modified performance discharge test in SR 3.8.7.6 may be performed in lieu of SR 3.8.1.3. ---------- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months
SR 3.8.1.4	---------- **- NOTE -** Only required to be met when the main control room air supply radiation monitor sample pumps are energized. ---------- Verify main control room air supply radiation monitor sample pump deenergizes on an actual or simulated actuation signal.	24 months

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3.8 ELECTRICAL POWER SYSTEMS

3.8.2 DC Sources – Shutdown

LCO 3.8.2 DC electrical power subsystems shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.6, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required battery chargers in one division inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	6 hours
	AND	
	A.2 Verify battery float current ≤ 2 amps	Once per 24 hours
	AND	
	A.3 Restore battery charger(s) to OPERABLE status.	72 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more required DC electrical power subsystems inoperable for reasons other than Condition A. OR Required Action and associated Completion Time of Condition A not met.	B.1 Declare affected required features inoperable. OR B.2.1 Suspend movement of irradiated fuel assemblies. AND B.2.2 Suspend operations with a potential for draining the reactor vessel. AND B.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND B.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately Immediately Immediately Immediately Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	---------- **- NOTE -** The following SRs are not required to be performed: SR 3.8.1.2 and SR 3.8.1.3. ---------- The following SRs are applicable: SR 3.8.1.1 SR 3.8.1.2 SR 3.8.1.3 SR 3.8.1.4	In accordance with applicable SRs

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3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Inverters – Operating

LCO 3.8.3 The Division A, B, C, and D inverters shall be OPERABLE.

- NOTES -

One inverter may be disconnected from its associated DC bus for ≤ 72 hours to perform an equalizing charge on its associated battery, providing:

1. The associated instrument and control bus is energized from its Class 1E voltage regulating transformer; and

2. All other AC instrument and control buses are energized from their associated OPERABLE inverters.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two inverter(s) within one division inoperable.	A.1	--- **- NOTE -** Enter applicable Conditions and Required Actions of LCO 3.8.5 “Distribution Systems – Operating” with any instrument and control bus de-energized. --- Restore inverter(s) to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1 AND B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify correct inverter voltage, frequency, and alignment to required AC instrument and control buses.	7 days

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3.8 ELECTRICAL POWER SYSTEMS

3.8.4 Inverters – Shutdown

LCO 3.8.4 Inverters shall be OPERABLE to support the onsite Class 1E power distribution subsystems required by LCO 3.8.6, “Distribution Systems – Shutdown.”

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required features inoperable.	Immediately
	OR	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.2 Suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify correct inverter voltage, frequency, and alignments to required AC instrument and control buses.	7 days

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3.8 ELECTRICAL POWER SYSTEMS

3.8.5 Distribution Systems – Operating

LCO 3.8.5 The following Division A, B, C, and D electrical power distribution subsystems shall be OPERABLE:

a. DC; and

b. AC instrument and control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC instrument and control division inoperable.	A.1 Restore AC instrument and control division to OPERABLE status.	6 hours
B. One DC division inoperable.	B.1 Restore DC division to OPERABLE status.	6 hours
C. Two AC instrument and control divisions inoperable.	C.1 Restore one AC instrument and control division to OPERABLE status.	2 hours
D. Two DC divisions inoperable.	D.1 Restore one DC division to OPERABLE status.	2 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours
F. Two inoperable divisions that result in a loss of safety function.	F.1 Enter LCO 3.0.3.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	Verify correct breaker and switch alignments and voltage to required DC and AC instrument and control electrical power distribution subsystems.	7 days

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3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Distribution Systems – Shutdown

LCO 3.8.6 The necessary portions of DC and AC instrument and control electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC or AC instrument and control electrical power distribution subsystems inoperable.	A.1 Declare affected required features inoperable. OR	Immediately
	A.2.1 Suspend movement of irradiated fuel assemblies. AND	Immediately
	A.2.2 Initiate action to suspend operations with a potential for draining the reactor vessel. AND	Immediately
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND	Immediately

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.4	Initiate actions to restore required DC and AC instrument and control electrical power distribution subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify correct breaker and switch alignments and voltage to required DC and AC instrument and control electrical power distribution subsystems.	7 days

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3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Battery Parameters

LCO 3.8.7 Battery Parameters for Division A, B, C, and D batteries shall be within limits.

APPLICABILITY: When associated DC electrical power sources are required to be OPERABLE.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries in one division with one or more battery cells float voltage < 2.07 V.	A.1 Perform SR 3.8.1.1. AND A.2 Perform SR 3.8.7.1. AND A.3 Restore affected cell voltage ≥ 2.07 V.	2 hours 2 hours 24 hours
B. One or more batteries in one division with float current > 2 amps.	B.1 Perform SR 3.8.1.1. AND B.2 Restore battery float current to ≤ 2 amps.	2 hours 24 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
---------- **- NOTE -** Required Action C.2 shall be completed if electrolyte level was below the top of plates. ----------	---------- **- NOTE -** Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of plates. ----------	
C. One or more batteries in one division with one or more cells electrolyte level less than minimum established design limits.	C.1 Restore electrolyte level to above top of plates. AND	8 hours
	C.2 Verify no evidence of leakage. AND	12 hours
	C.3 Restore electrolyte level to greater than or equal to minimum established design limits.	31 days
D. One or more batteries in one division with pilot cell electrolyte temperature less than minimum established design limits.	D.1 Restore battery pilot cell temperature to greater than or equal to minimum established design limits.	12 hours
E. One or more batteries in two or more divisions with battery parameters not within limits.	E.1 Restore battery parameters for batteries in three divisions to within limits.	2 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met. OR One or more batteries in one division with one or more battery cells float voltage < 2.07 V and float current > 2 amps.	F.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	---------- **- NOTE -** Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.1.1. ---------- Verify each battery float current is ≤ 2 amps.	7 days
SR 3.8.7.2	Verify each battery pilot cell float voltage is ≥ 2.07 V.	31 days
SR 3.8.7.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	31 days
SR 3.8.7.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	31 days
SR 3.8.7.5	Verify each battery connected cell float voltage is ≥ 2.07 V.	92 days

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.7.6	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months or 25% of expected life, whichever is less AND 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

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3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentration of the Reactor Coolant System (RCS), the fuel transfer canal, and the refueling cavity shall be maintained within the limit specified in COLR.

APPLICABILITY: MODE 6.

- NOTE -

Applicable to the fuel transfer canal and the refueling cavity only when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions. AND A.2 Initiate actions to restore boron concentration to within limits.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

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3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Flow Paths

LCO 3.9.2 One valve in each unborated water source flow path shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

- NOTE -

Separate condition entry is allowed for each unborated water source flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------- **- NOTE -** Required Action A.2 must be completed whenever Condition A is entered. ---------- One or more unborated water source flow paths with no valve secured in the closed position.	A.1 Initiate actions to secure one valve in the flow path in the closed position. AND A.2 Perform SR 3.9.1.1.	Immediately 4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify one valve in each unborated water source flow path is secured in the closed position.	31 days

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3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1 Suspend positive reactivity additions. AND	Immediately
	A.2 Suspend operations that would cause introduction into the Reactor Coolant System (RCS), coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status. AND	Immediately
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform a CHANNEL CHECK.	12 hours
SR 3.9.3.2	---------- **- NOTE-** Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	24 months

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3.9 REFUELING OPERATIONS

3.9.4 Refueling Cavity Water Level

LCO 3.9.4 Refueling Cavity Water Level shall be maintained ≥ 23 ft above the top of the reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

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3.9 REFUELING OPERATIONS

3.9.5 Decay Time

LCO 3.9.5 The reactor shall be subcritical for ≥ 48 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical < 48 hours.	A.1 Suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify the reactor has been subcritical for ≥ 48 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor vessel

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4.0 DESIGN FEATURES

4.1 Site

[Not applicable to AP1000 Design Certification. Site specific information to be provided by the COL Applicant.]

4.1.1 Site and Exclusion Boundaries

[This information will be provided by the combined license applicant.]

4.1.2 Low Population Zone (LPZ)

[This information will be provided by the combined license applicant.]

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with a zirconium based alloy and containing an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium based alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod and Gray Rod Assemblies

The reactor core shall contain 53 Rod Cluster Control Assemblies (RCCAs), each with 24 rodlets/RCCA. The RCCA absorber material shall be silver indium cadmium as approved by the NRC.

Additionally, there are 16 low worth Gray Rod Cluster Assemblies (GRCAs), with 24 rodlets/GRCA, which, in conjunction with the RCCAs, are used to augment mechanical shim (MSHIM) operation.

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4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 4.95 weight percent;

b. $k_{eff} \leq 0.95$ if flooded with unborated water which includes an allowance for uncertainties (Region 1 racks);

c. A nominal 10.93 inch center-to-center distance between fuel assemblies placed in Region 1 of the spent fuel storage racks;

d. A nominal 9.04 inch center-to-center distance between fuel assemblies placed in Region 2 of the spent fuel storage racks;

e. A nominal 11.65 inch center-to-center distance between fuel assemblies placed in the Defective Fuel Cells;

f. New or partially spent fuel assemblies with any discharge burnup may be allowed unrestricted storage in Region 1 and the Defective Fuel Cells of Figure 4.3-1;

g. Partially spent fuel assemblies meeting the initial nominal enrichment and burnup requirements of LCO 3.7.12, “Spent Fuel Pool Storage,” may be stored in Region 2 of Figure 4.3-1; and

h. $k_{eff} < 1.0$ if flooded with unborated water and $k_{eff} \leq 0.95$ if flooded with borated water at a minimum soluble boron concentration described in the Bases for LCO 3.7.12 for normal and design basis criticality-related accident conditions, which includes an allowance for uncertainties (Region 2 racks).

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. The maximum k_{eff} value, including all biases and uncertainties, shall be less than or equal to 0.95 with full density unborated water;

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4.0 DESIGN FEATURES

c. The maximum k_{eff} value, including all biases and uncertainties, shall be less than or equal to 0.98 with optimum moderation and full reflection conditions; and

d. A nominal 10.90 inch center-to-center distance between fuel assemblies placed in the new fuel storage racks.

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below a minimum water depth of ≥ 23 ft above the surface of the fuel storage racks.

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 889 fuel assemblies.

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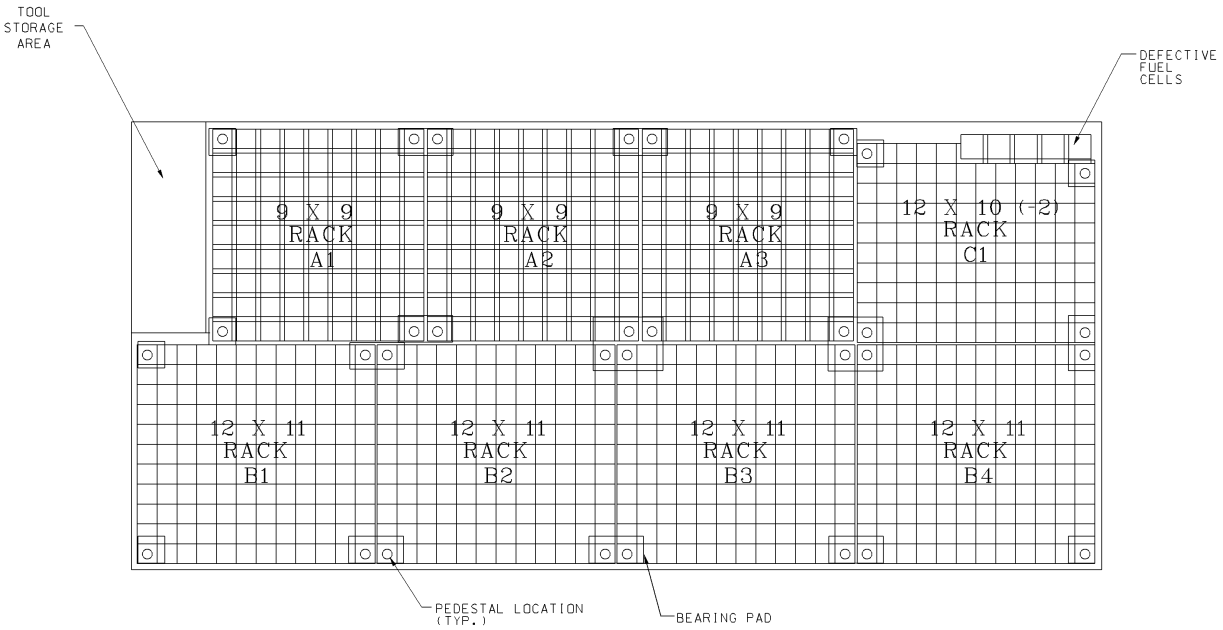

Region 1 (A1, A2, A3) – 243 locations

Region 2 (B1, B2, B3, B4, C1) – 641 locations

Defective Fuel Cells (DFCs) – 5 locations

Total Storage Locations – 889

Figure 4.3-1

Discrete Two Region Spent Fuel Pool Rack Layout

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5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The [Plant Manager] shall be responsible for overall unit operations and shall delegate in writing the succession to this responsibility during his absence.

The [Plant Manager] or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The [Shift Supervisor (SS)] shall be responsible for the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

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5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the [FSAR/QA Plan];

b. The [Plant Manager] shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operation pressures.

5.2.2 Unit Staff

- REVIEWER'S NOTE -

[Determination of the unit staff positions, numbers, and qualifications are the responsibility of the COL applicant. Input provided in WCAP-14694, Revision 0, for the MCR staff and WCAP-14655, Revision 1, for other than the MCR staff will be used in the determination. Each of the following paragraphs may need to be corrected to specify the plant staffing requirements.]

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5.2 Organization

5.2.2 Unit Staff (continued)

The unit staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODE 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), health physicists, auxiliary operators, and key maintenance personnel).

   The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

   Any deviation from the above guidelines shall be authorized in advance by the plant manager or the plant manager’s designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

   Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not be assigned.

e. The operations manager or assistant operations manager shall hold an SRO license.

f. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

- REVIEWER'S NOTE -

[Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.]

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standards acceptable to the NRC staff]. [The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptale to NRC staff].

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

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5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

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5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.1 and Specification 5.6.2.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

i. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

ii. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20. 1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the changed portion of the ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

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5.5 Programs and Manuals

5.5.2 Radioactive Effluent Control Program

a. This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoints determination in accordance with the methodology in the ODCM;

2. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20;

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

4. Limitations on the annual and quarterly doses or dose commitment to a member of the public for radioactive materials in liquid effluents released form each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

5. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

6. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be in accordance with the following:

i. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin and

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5.5 Programs and Manuals

5.5.2 Radioactive Effluent Control Program (continued)

ii. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

b. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.3 Inservice Testing Program

The Inservice Testing Program is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

5.5.4 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

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5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gpd per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.7, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

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5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following installation.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

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5.5 Programs and Manuals

5.5.5 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.6 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

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5.5 Programs and Manuals

5.5.7 Safety Function Determination Program (SFDP)

a. This program ensures loss of safety function is detected and appropriate action taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirement of LCO 3.0.6. The SFDP shall contain the following:

   1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

   2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

   3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

   4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

   1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

   2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

   3. A required system redundant to the support system(s) for the supported systems b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

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5.5 Programs and Manuals

5.5.8 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program, dated September 1995,” as modified by approved exceptions.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 58.1 psig. The containment design pressure is 59 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.10% of primary containment air weight per day.

d. Leakage Rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and Type C tests and ≤ 0.75 L_a for Type A tests;

   2. Air lock testing acceptance criteria are:

      i. Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a,

      ii. For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

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5.5 Programs and Manuals

5.5.9 System Level OPERABILITY Testing Program

The System Level OPERABILITY Testing Program provides requirements for performance tests of passive systems. The System Level OPERABILITY Test Requirements specified in FSAR Table 3.9-17 apply when specified by individual Surveillance Requirements.

a. The provisions of SR 3.0.2 are applicable to the test frequencies specified in FSAR Table 3.9-17 for performing system level OPERABILITY testing activities; and

b. The provisions of SR 3.0.3 are applicable to system level OPERABILITY testing activities.

5.5.10 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Table 3.9-1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.11 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,” or of the battery manufacturer including the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

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5.5 Programs and Manuals

5.5.12 Main Control Room Envelope Habitability Program

A Main Control Room Envelope (MCRE) Habitability Program shall be established and implemented to ensure that MCRE habitability is maintained such that, with an OPERABLE Main Control Room Emergency Habitability System (VES), MCRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the MCRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the MCRE and the MCRE boundary.

b. Requirements for maintaining the MCRE boundary in its design condition, including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the MCRE boundary into the MCRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors,” Revision 0, May 2003, and (ii) assessing MCRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the MCRE pressure relative to all external areas adjacent to the MCRE boundary during the pressurization mode of operation of one VES air delivery flow path, operating at the required flow rate of 65 ± 5 scfm, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the MCRE boundary.

e. The quantitative limits on unfiltered air inleakage into the MCRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of MCRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing MCRE habitability, determining MCRE unfiltered inleakage, and measuring MCRE pressure and assessing the MCRE boundary as required by paragraphs c and d, respectively.

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5.5 Programs and Manuals

5.5.13 Ventilation Filter Testing Program (VFTP)

a. A program shall be established to implement the following required testing of the VES.

Tests described in Specification 5.5.13.a.1 and 5.5.13.a.2 shall be performed: i) initially, ii) once each 24 months, iii) after partial or complete replacement of a HEPA filter or charcoal adsorber, iv) following detection of, or evidence of, penetration or intrusion of water or other material into any portion of the VES that may have an adverse effect on the functional capability of the filters, and v) following painting, fire, or chemical release in any ventilation zone communicating with the VES that may have an adverse effect on the functional capability of the system.

Tests described in Specification 5.5.13.a.3 shall be performed: i) after each 720 hours of system operation or at least once each 24 months, whichever comes first, ii) following painting, fire, or chemical release in any ventilation zone communicating with the VES that may have an adverse effect on the functional capability of the carbon media, and iii) following detection of, or evidence of, penetration or intrusion of water or other material into any portion of the VES that may have an adverse effect on the functional capability of the carbon media.

Tests described in 5.5.13.a.4 shall be performed once per 24 months.

1. Demonstrate for the VES that an inplace test of the high efficiency particulate air (HEPA) filter shows a penetration and system bypass ≤ 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at a flow rate at least 600 cfm greater than the VES makeup flow rate.

Ventilation System	Flow Rate
VES	≥ 600 + VES makeup flow rate (cfm)

2. Demonstrate for the VES that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at a flow rate at least 600 cfm greater than the VES makeup flow rate.

Ventilation System	Flow Rate
VES	≥ 600 + VES makeup flow rate (cfm)

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5.5 Programs and Manuals

5.5.13 Ventilation Filter Testing Program (continued)

3. Demonstrate for the VES that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below.

Ventilation System	Penetration	RH
VES	5%	95%

4. Demonstrate for the VES that the pressure drop across the combined HEPA filter, the charcoal adsorber, and the post filter is less than the value specified below when tested at the system flow rate specified below +/- 10%.

ESF Ventilation System	Delta P	Flow Rate
VES	5 in. water gauge	660 cfm

b. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.14 Setpoint Program (SP)

a. The Setpoint Program (SP) implements the regulatory requirement of 10 CFR 50.36(c)(1)(ii)(A) that technical specifications will include items in the category of limiting safety system settings (LSSS), which are settings for automatic protective devices related to those variables having significant safety functions.

b. The Nominal Trip Setpoint (NTS), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) for each Technical Specification required automatic protection instrumentation function shall be calculated in conformance with WCAP-16361-P, “Westinghouse Setpoint Methodology for Protection Systems – AP1000,” February 2011.

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5.5 Programs and Manuals

5.5.14 Setpoint Program (SP) (continued)

c. For each Technical Specification required automatic protection instrumentation function, performance of a CHANNEL CALIBRATION surveillance “in accordance with the Setpoint Program” shall include the following:

1. The as-found value of the instrument channel trip setting shall be compared with the previously recorded as-left value.

i. If the as-found value of the instrument channel trip setting differs from the previously recorded as-left value by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated to verify that it is functioning in accordance with its design basis before declaring the surveillance requirement met and returning the instrument channel to service. An Instrument Channel is determined to be functioning in accordance with its design basis if it can be set to within the ALT. This as-found condition shall be entered into the plant’s corrective action program.

ii. If the as-found value of the instrument channel trip setting is less conservative than the specified AFT, the surveillance requirement is not met and the instrument channel shall be immediately declared inoperable.

2. The instrument channel trip setting shall be set to a value within the specified ALT around the specified NTS at the completion of the surveillance; otherwise, the surveillance requirement is not met and the instrument channel shall be immediately declared inoperable.

d. The difference between the instrument channel trip setting as-found value and the previously recorded as-left value for each Technical Specification required automatic protection instrumentation function shall be trended and evaluated to verify that the instrument channel is functioning in accordance with its design basis.

e. The SP shall establish a document containing the current value of the specified NTS, AFT, and ALT for each Technical Specification required automatic protection instrumentation function and references to the calculation documentation. Changes to this document shall be governed by the regulatory requirement of 10 CFR 50.59. In addition, changes to the specified NTS, AFT, and ALT values shall be governed by the approved setpoint methodology. This document, including any revisions or supplements, shall be provided upon issuance to the NRC.

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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Annual Radiological Environmental Operating Report

- NOTE -
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 Radioactive Effluent Release Report

- NOTE -
A single submittal may be made for a multiple unit station.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

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5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1, “Reactor Core SLs”;
3.1.1, “SHUTDOWN MARGIN (SDM)”;
3.1.3, “Moderator Temperature Coefficient (MTC)”;
3.1.5, “Shutdown Bank Insertion Limits”;
3.1.6, “Control Bank Insertion Limits”;
3.2.1, “Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology)”;
3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)”;
3.2.3, “AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)”;
3.2.5, “On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters”;
3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits”; and
3.9.1, “Boron Concentration.”

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985 (Westinghouse Proprietary) and WCAP-9273-NP-A (Non-Proprietary).

   (Methodology for Specifications 3.1.1 - Shutdown Margin (SDM), 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limits, 3.1.6 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - AXIAL FLUX DIFFERENCE, 3.4.1 - RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, and 3.9.1 - Boron Concentration.)

2a. WCAP-8385, “Power Distribution Control and Load Following Procedures - Topical Report,” September 1974 (Westinghouse Proprietary) and WCAP-8403 (Non-Proprietary).

   (Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

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5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

2b. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

3. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification,” February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

4a. WCAP-16009-P-A, “Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),” Revision 0, January 2005 (Westinghouse Proprietary) and WCAP-16009-NP-A (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor and Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

4b. APP-GW-GLE-026, “Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000,” Revision 1, February 2009 (Westinghouse Proprietary) and APP-GW-GLE-026-NP (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor and Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

5. WCAP-12472-P-A, (Westinghouse Proprietary) and WCAP-12473-A (Non-Proprietary), “BEACON Core Monitoring and Operations Support System,” August 1994, Addendum 1, May 1996, and Addendum 2, March 2001 and WCAP-12472-P-A (Westinghouse Proprietary) and WCAP-12472-NP-A (Non-Proprietary) Addendum 4, September 2012.

(Methodology for Specification 3.2.5 - OPDMS - Monitored Parameters.)

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5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. APP-GW-GLR-137, Revision 1, “Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OTΔT) Reactor Trips,” Westinghouse Electric Company LLC.

   (Methodology for Specification 2.1.1 – Reactor Core Safety Limits.)

7a. WCAP-16045-P-A, “Qualification of the Two-Dimensional Transport Code PARAGON,” August 2004 (Westinghouse Proprietary) and WCAP-16045-NP-A, (Non-Proprietary).

   (Methodology for Specification 3.1.3 - Moderator Temperature Coefficient (MTC).)

7b. WCAP-16045-P-A, Addendum 1-A, “Qualification of the NEXUS Nuclear Data Methodology,” August 2007 (Westinghouse Proprietary) and WCAP-16045-NP-A, Addendum 1-A, (Non-Proprietary).

   (Methodology for Specification 3.1.3 - Moderator Temperature Coefficient (MTC).)

7c. WCAP-13749-P-A, “Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement,” March 1997 (Westinghouse Proprietary).

   (Methodology for Specification 3.1.3 - Moderator Temperature Coefficient (MTC).)

8. WCAP-17661-P-A, Revision 1, "Improved RAOC and CAOC F_Q Surveillance Technical Specifications," February 2019 (Westinghouse Proprietary).

   (Methodology for Specifications 3.1.6 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology), and 3.2.3 - AXIAL FLUX DIFFERENCE.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Passive Core Cooling Systems limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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5.6 Reporting Requirements

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

3.4.3, “RCS Pressure and Temperature (P/T) Limits”; and
3.4.14, “Low Temperature Overpressure Protection (LTOP).”

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.” (Limits for LCO 3.4.3 and LCO 3.4.14).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B of LCO 3.3.17, “Post Accident Monitoring (PAM) Instrumentation,” a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

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5.6 Reporting Requirements

5.6.6 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.4, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

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5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

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5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designees, and

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

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5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

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5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the “As Low As is Reasonably Achievable” principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core Safety Limits (SLs)

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not to be exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur, and by requiring that the fuel centerline temperature stays below the melting temperature.

The restriction of this SL prevents overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR or power peaking in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (Zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Protection and Safety Monitoring System (PMS) and steam generator safety valves prevents violation of the reactor core SLs.

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BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System (RTS) setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the PMS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RTS setpoints. LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,” or the assumed initial conditions of the safety analyses (as indicated in FSAR Section 7.2, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

These limits are developed using the methodologies identified in Technical Specification 5.6.3.

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and cold leg temperature for which the minimum DNBR is not less than the safety analysis limit, that fuel centerline temperature remains below melting, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience DNB and

b. There must be at least a 95% probability at a 95% confidence level that the hottest fuel pellet in the core does not experience centerline fuel melting.

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BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various PMS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the PMS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the core exit quality is within the limits defined by the DNBR correlation and that the Overtemperature and Overpower ΔT reactor trip protection functions continue to provide protection if local hot leg streams approach saturation temperature. Appropriate functioning of the PMS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS cold leg temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.” In MODES 3, 4, 5, and 6, applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 7.2, “Reactor Trip.”

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, “Reactor Coolant Pressure Boundary,” and GDC 15, “Reactor Coolant System Design” (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, “Reactivity Limits” (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia (2485 psig). During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases.

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor High 2 pressurizer pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load with loss of feedwater flow, without a direct reactor trip. During the transient, no control actions are assumed except that the

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BASES

APPLICABLE SAFETY ANALYSES (continued)

safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor High 2 pressurizer pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the High 2 pressurizer pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. RCS depressurization valves (Automatic Depressurization System [ADS] valves);

b. Steam line relief valves (SG PORVs);

c. Turbine Bypass System (Steam Dump System);

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel, piping, valves, and fittings under the ASME Code, Section III, is 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2733.5 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 since the reactor vessel head is removed, or overpressure protection is provided by meeting the requirements of LCO 3.4.14, “Low Temperature Overpressure Protection (LTOP)”.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

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BASES

SAFETY LIMIT VIOLATIONS (continued)

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for abnormal radioactive releases.

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 50.34.
5. FSAR Section 7.2, “Reactor Trip.”

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B 3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirements for when the LCO is required to be met (i.e. when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification.)

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that the ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case compliance with the Required Actions provides an acceptable level of safety for continued operation.

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BASES

LCO 3.0.2 (continued)

Completing the Required Actions is not required when an LCO is met, or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The ACTIONS for not meeting a single LCO adequately manage any increase in plant risk, provided any unusual external conditions (e.g., severe weather, offsite power instability) are considered. In addition, the increased risk associated with simultaneous removal of multiple structures, systems, trains or components from service is assessed and managed in accordance with 10 CFR 50.65(a)(4). Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met; and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

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LCO 3.0.3 (continued)

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. Planned entry into LCO 3.0.3 should be avoided. If it is not practicable to avoid planned entry into LCO 3.0.3, plant risk should be assessed and managed in accordance with 10 CFR 50.65(a)(4), and the planned entry into LCO 3.0.3 should have less effect on plant safety than other practicable alternatives.

Upon entering into LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, “Completion Times.”

A unit shutdown required in accordance with LCO 3.0.3 may be terminated, and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition was initially entered and not from the time LCO 3.0.3 is exited.

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BASES

LCO 3.0.3 (continued)

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

Compliance with the time limits of Specification 3.0.3 rely on the use of nonsafety-related systems, which are not governed by Technical Specification LCOs.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive condition required by LCO 3.0.3.

Exceptions to 3.0.3 are provided in instances where requiring a unit shutdown in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.5, Spent Fuel Pool Water Level. This Specification has an Applicability of “At all times.” Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.5 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.5 of “Suspend movement of irradiated fuel assemblies in the spent fuel pool” is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with either LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

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BASES

LCO 3.0.4 (continued)

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered following entry into the MODE or other specified condition in the Applicability will permit continued operation within the MODE or other specified condition for an unlimited period of time. Compliance with ACTIONS that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made and the Required Actions followed after entry into the Applicability.

For example, LCO 3.0.4.a may be used when the Required Action to be entered states that an inoperable instrument channel must be placed in the trip condition within the Completion Time. Transition into a MODE or other specified condition in the Applicability may be made in accordance with LCO 3.0.4 and the channel is subsequently placed in the tripped condition within the Completion Time, which begins when the Applicability is entered. If the instrument channel cannot be placed in the tripped condition and the subsequent default ACTION ("Required Action and associated Completion Time not met") allows the OPERABLE train to be placed in operation, use of LCO 3.0.4.a is acceptable because the subsequent ACTIONS to be entered following entry into the MODE include ACTIONS (place the OPERABLE train in operation) that permit safe plant operation for an unlimited period of time in the MODE or other specified condition to be entered.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.160, "Monitoring the Effectiveness of

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BASES

LCO 3.0.4 (continued)

Maintenance at Nuclear Power Plants." Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a

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BASES

LCO 3.0.4 (continued)

specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity), and may be applied to other Specifications based on NRC plant specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance of restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of Surveillance Requirements to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

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BASES

LCO 3.0.5 (continued)

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

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BASES

LCO 3.0.6 (continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.7, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety functions exists are required to be entered.

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. There are no support system LCO requirements for offsite power based on the safety-related passive design.

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BASES

LCO 3.0.6 (continued)

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.1.10 allow specified Technical Specification (TS) requirements to be changed to permit performance of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

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B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification ensures that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

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BASES

SR 3.0.1 (continued)

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met in accordance with SR 3.0.2 prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Actions with a Completion Time that requires the periodic performance of the Required Action on a “once per…” interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.2 are applicable, a 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted.

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The

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BASES

SR 3.0.2 (continued)

requirements of regulations take precedence over the TS. Examples of where SR 3.0.2 does not apply are the Containment Leakage Rate Testing Program required by 10 CFR 50, Appendix J, and the inservice testing of pumps and valves in accordance with applicable American Society of Mechanical Engineers Operation and Maintenance Code, as required by 10 CFR 50.55a. These programs establish testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations directly or by reference.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per …" basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed, in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before compliance with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform

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SR 3.0.3 (continued)

the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit Conditions or operational situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity. SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

SR 3.0.3 is only applicable if there is a reasonable expectation the associated equipment is OPERABLE or that variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or activities that might support the expectation that the Surveillance will be met when performed. An example of the use of SR 3.0.3 would be a component actuation that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the component actuation; the subject component actuation has been tested by another SR; or historical actuation of the subject component has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment. The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed. For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.

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SR 3.0.3 (continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used repeatedly to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.160, 'Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this specification, or within the Completion Time of the ACTIONS restores compliance with SR 3.0.1.

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BASES

SR 3.0.3 (continued)

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval (including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification).

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

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BASES

SR 3.0.4 (continued)

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1) the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to assure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all Rod Cluster Control Assemblies (RCCAs), assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn. When confirming SDM requirements in MODES 1 and 2 with $k_{eff} \geq 1.0$, the calculation conservatively ignores the insertion of the Gray Rod Cluster Assemblies (GRCAs) upon reactor trip. When SDM is confirmed in MODE 2 with $k_{eff} < 1.0$, and in MODES 3, 4, and 5, the calculation may include the negative reactivity from the GRCAs if the GRCAs are confirmed inserted and have passed the acceptance criteria for GRCA bank worth measurements performed during startup physics testing (Ref. 5).

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Plant Control System (PLS) can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the PLS, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM is calculated and monitored by the Online Power Distribution Monitoring System (OPDMS) and controlled by operating with RCCAs sufficiently withdrawn to meet the SDM requirement. When the OPDMS is not monitoring parameters, SDM control is ensured by operating within the limits of LCO 3.1.5 “Shutdown

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BASES

BACKGROUND (continued)

Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analyses (Ref. 2) establish an SDM that ensures that specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departures from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and must not exceed applicable limits (Ref. 4) for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accidents for the SDM requirements are typically from the categories of an increase in Heat Removal from the Primary System or Reactivity and Power Distribution Anomalies, e.g., Steam System Piping Failure or Decrease in Boron Concentration in the Reactor Coolant, as described in the accident analyses (Ref. 2). The increased steam flow in the main steam system causes an increased energy removal from the affected SG, and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient (MTC), this cooldown causes an increase in core reactivity. The positive reactivity addition from the moderator temperature decrease will terminate when the affected

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APPLICABLE SAFETY ANALYSES (continued)

SG boils dry, thus terminating RCS heat removal and cooldown. Following an increase in heat removal from the primary system, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and the THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the increase in heat removal from the primary system, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition;

c. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting when critical boron concentrations are highest.

The uncontrolled rod withdrawal transient is terminated by a high neutron flux trip. Power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time-dependent redistribution of core power.

These limits are developed using the methodologies identified in Technical Specification 5.6.3.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). SDM is considered an initial condition process variable because it is periodically monitored to provide assurance that the unit is operating within the bounds of accident analysis assumptions.

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BASES

LCO

SDM is a core design condition that can be ensured during operation by compliance with LCO 3.2.5, “OPDMS Monitored Parameters,” item d. When the OPDMS is not monitoring parameters, SDM is ensured during operation by complying with LCO 3.1.5, “Shutdown Bank Insertion Limits,” and LCO 3.1.6, “Control Bank Insertion Limits.” When the reactor is in shutdown and refueling MODES, SDM is ensured by maintaining a minimum soluble RCS boron concentration. This minimum boron concentration is based on the present reactor core design, condition and operating history.

The most limiting analyses that establish the SDM value of the LCO are typically from the categories of an Increase in Heat Removal from the Primary System or Reactivity and Power Distribution Anomalies (Ref. 2). For SLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 50.34 limits (Ref. 3). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$, and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, “Boron Concentration.” In MODE 1 and in MODE 2 with $k_{eff} \geq 1.0$, SDM is ensured by one of two means, depending on the status of the OPDMS. When the OPDMS is continuously monitoring parameters, SDM is ensured by compliance with LCO 3.2.5, “OPDMS Monitored Parameters,” item d. When the OPDMS is not monitoring parameters, SDM is ensured by complying with LCO 3.1.5, “Shutdown Bank Insertion Limits,” and LCO 3.1.6, “Control Bank Insertion Limits.”

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

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BASES

ACTIONS (continued)

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a concentrated solution. The operator should begin boration with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at hot shutdown conditions when boron concentration is highest. For example, assuming that a value of 1.0% Δk/k must be recovered and the boration flow rate is 100 gpm, it is possible to increase the boron concentration of the RCS by approximately 111 ppm in 21 minutes utilizing boric acid solution having a concentration of 4375 ppm. If a boron worth of 9 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1.0% Δk/k. The boron concentration and system parameters used above represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2 with $k_{eff} \geq 1.0$, SDM is verified by either compliance with the requirements of LCO 3.2.5, “OPDMS Monitored Parameters,” item d, or when the OPDMS is not monitoring parameters, by observing that the requirements of LCO 3.1.5 “Shutdown Bank Insertion Limits,” and LCO 3.1.6, “Control Bank Insertion Limits” are met. In the event that an RCCA is known to be untrippable, however, SDM verification must account for the worth of both the untrippable RCCA as well as another RCCA of maximum worth. This is not applicable to GRCAs as they are not credited in the verification of SDM in MODES 1 and 2 with $k_{eff} \geq 1.0$.

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering at least the listed reactivity effects:

a. RCS boron concentration;

b. RCCA and GRCA position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

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BASES

SURVEILLANCE REQUIREMENTS (continued)

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR Chapter 15, “Accident Analysis.”
3. 10 CFR 50.34.
4. NUREG-0800, Standard Review Plan, Section 4.2, Revision 3, “Fuel System Design” (Appendix B).
5. WCAP-16943-P-A, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design."

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, “SHUTDOWN MARGIN (SDM)”) in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance since parameters are being maintained relatively stable under steady-state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as control rod position, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculation models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and a negative moderator temperature coefficient, the

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BASES

BACKGROUND (continued)

excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel and burnable absorbers are being depleted and excess reactivity is changing. As the fuel and burnable absorbers deplete, the RCS boron concentration is periodically adjusted to compensate for the net core reactivity change and to reposition the Control Banks within the limits defined in LCO 3.1.6, “Control Bank Insertion Limits.” A predicted RCS boron concentration can be determined for any core condition using a calculational model consistent with the design analysis. Therefore, deviations from the predicted RCS boron concentration with otherwise consistent core conditions may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Certain accident evaluations (Ref. 2) are, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are sensitive to accurate predictions of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analysis are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at BOC do not agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron

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APPLICABLE SAFETY ANALYSES (continued)

concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the Conditions of the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% $\Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is

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BASES

APPLICABILITY (continued)

operating as designed. This specification does not apply in MODE 3, 4, and 5 because the reactor is shutdown and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, “Boron Concentration”) ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

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BASES

ACTIONS (continued)

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by LCO 3.1.1 Required Action A.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC.

The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPDs) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPDs following the initial 60 EFPDs after entering MODE 1 is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, OPDMS, AFD, etc.) for prompt indication of an anomaly.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. FSAR, Chapter 15, “Accident Analysis.”

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a non-positive MTC over the range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by surveillances. Both initial and reload cores are designed so that the MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons (burnable absorbers) to yield an MTC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles designed to achieve high burnups that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR Chapter 15 accident and transient analyses (Ref. 2).

If the LCO limits are not met, the plant response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

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BASES

BACKGROUND (continued)

The SRs for verification of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the RCS boron concentration changes associated with fuel burnup and burnable absorbers depletion.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

FSAR Chapter 15 (Ref. 2) contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the least negative value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core heat-up must be evaluated when the MTC is least negative. Such accidents include the rod withdrawal transient from either zero (Ref. 2) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is BOC or EOC. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at the limiting time in cycle life. An EOC verification is conducted for conditions of an RCS boron concentration of approximately 300 ppm. The value determined during the verification may be extrapolated to project the EOC value, in order to confirm reload design predictions.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

These limits are developed using the methodologies identified in Technical Specification 5.6.3.

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the accident analysis during operation.

Assumptions made in safety analyses require that the MTC be more negative than a given upper limit and less negative than a given lower limit. The MTC is least negative near BOC; this upper bound must not be exceeded. This maximum upper limit occurs at all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, the LCO is ensured through surveillances. The surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

During the core safety evaluation, MTC values are predicted at selected burnup accumulation during operations. Based on these predictions, administrative limits may be required on maximum RCS boron concentration and control bank position to maintain the MTC at less than or equal to the upper MTC limit provided in the COLR. These administrative limits consist of tables of the maximum RCS boron concentration versus control rod position, power level, and cycle burnup. A means for adjusting the maximum RCS boron concentrations is also provided to account for differences between the MTC measured during startup physics testing and the corresponding predicted value. The RCS boron concentration needed to maintain a critical core state may increase with cycle burnup accumulation due to burnable absorber depletion. However, as the cycle burnup continues to accumulate, the effect of fuel depletion on core reactivity will exceed the effect of burnable absorber depletion and the RCS boron concentration needed to maintain criticality will decrease. The reduction in critical boron concentration causes the MTC to become more negative. The time in cycle life at which the

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BASES

LCO (continued)

calculated MTC will meet the upper MTC limit can be determined. At this point in the cycle and beyond, the administrative limits on maximum RCS boron concentration and control bank position are no longer required to meet the upper MTC limit.

The BOC limit and the EOC limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to assure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod or control rod group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

If the upper MTC limit is violated, administrative limits are placed on the combination of maximum RCS boron concentration and control bank position to maintain the MTC within its upper limit. A Completion Time of 24 hours provides enough time for evaluating the MTC surveillance and computing the required administrative limits.

B.1

If the required administrative limits are not established within 24 hours, the unit must be placed in MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC which is less negative than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

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BASES

ACTIONS (continued)

C.1

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must be placed in a MODE or Condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 4 within 12 hours.

The allowed Completion Time is a reasonable time based on operating experience to reach the required MODE from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires verification of the MTC once at BOC prior to entering MODE 1 in order to demonstrate compliance with the most limiting MTC LCO. Meeting the limit prior to entering MODE 1 assures that the limit will also be met at higher power levels.

The BOL MTC can be used to determine if administrative limits on the combination of maximum RCS boron concentration and control bank position are required at any time in the cycle.

SR 3.1.3.2

In similar fashion, the LCO requires that the MTC be less negative than the specified value for EOC full power conditions. This surveillance may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to provide assurance that the LCO limit will be met at EOC when the 300 ppm Surveillance criterion is met.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The SR is required to be performed once within 7 effective full power days (EFPD) after reaching an RCS boron concentration that is equivalent to an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the second Frequency of 14 EFPD thereafter is sufficient to avoid exceeding the EOC limit.

SR 3.1.3.2 is modified by two Notes. The first Note states that an exemption from the measurement may be allowed provided applicable criteria provided in the COLR are met. This is based on the requirements of WCAP-13749-P-A (Ref. 4). Reference 4 allows that if a specified revised prediction of the MTC and limits for several core parameters measured during the cycle are within specified bounds, the end of life (EOL) MTC measurement need not be made. The second Note states that the Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup, and the continued performance of SR 3.1.3.2 is no longer required.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR Chapter 15, “Accident Analysis.”
3. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.
4. WCAP-13749-P-A, “Safety Evaluation Supporting The Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement,” March 1997.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the Rod Cluster Control Assemblies (RCCAs) is an initial assumption in all safety analyses which assume rod insertion upon reactor trip. Although Gray Rod Cluster Assemblies (GRCAs) are released on reactor trip, GRCAs are conservatively excluded from this assumption, in that their reactivity worth is not credited in advance when calculating the total negative reactivity worth potentially available from control rod insertion on a reactor trip. As a result, GRCAs are excluded from the OPERABILITY (i.e., trippability) requirements of this LCO. However, the basis for LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" defines conditions under which negative reactivity from GRCA banks may be credited for SDM after a shutdown has already occurred. Rod misalignment to the maximum extent allowed by this LCO is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

The control rods in the AP1000 design are divided among seven control banks and four shutdown banks. Shutdown banks are designated S1, S2, S3 and S4. Control banks are designated MA, MB, MC, MD, M1, M2, and AO. The control banks provide reactivity and power distribution control during unit operation while the shutdown banks provide rapid shutdown capability for the reactor.

Each bank may be further subdivided into groups to provide for more precise reactivity control. A group consists of four control rods in symmetric core locations that are electrically paralleled to step simultaneously. The only exception to this is the AO bank, which contains a five-rod group that steps simultaneously. A bank consists of one or more groups that are moved in a staggered fashion, but always within one step of each other.

Control rod assemblies are further characterized by their neutron absorption characteristics. More absorbent RCCAs are used in all shutdown banks as well as in control banks M1, M2 and AO. Less absorbent GRCAs are used in control banks MA, MB, MC and MD.

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BASES

BACKGROUND (continued)

Mechanical or electrical failures may cause an RCCA to become inoperable or cause an RCCA or GRCA to become misaligned from its group. RCCA inoperability or misalignment of a control rod of either type, may cause increased power peaking due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, RCCA and GRCA alignment and RCCA OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

RCCAs and GRCAs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA or GRCA one step (approximately 5/8 inch) at a time but at varying rates (steps per minute) depending on the signal output from the Plant Control System (PLS).

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are part of the MSHIM (Mechanical Shim) Control System which utilizes two independent groups of control banks for control of reactivity and axial power distribution.

Reactivity control is provided primarily by control banks MA, MB, MC, MD, M1, and M2, which are referred to as the M banks. These banks operate with a fixed overlap. The bank worth and overlap are defined so as to minimize the impact on axial offset with control bank maneuvering and still retain the reactivity required to meet the desired load changes.

Axial power distribution control is provided by the AO control bank, a relatively high worth bank which moves independently of the M banks.

The control rods are arranged in a radially symmetric pattern so that control bank motion does not introduce radial asymmetries in the core power distributions.

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BASES

BACKGROUND (continued)

The axial position of shutdown rods and control rods is indicated by two separate and independent means, which are the Bank Demand Position Indication (commonly called group step counters) and the Digital Rod Position Indication (DRPI). DRPI and Bank Demand Position Indication are described in more detail in the Bases for LCO 3.1.7, “Rod Position Indication.”

Certain control banks will be pre-selected for rapid insertion by the Rapid Power Reduction (RPR) system during load rejection transients or if the main turbine or a main feedwater pump trip occurs above a pre-selected plant power level. When the RPR system actuates, any control rods that are dropped into the core as a result of the RPR actuation, and that are confirmed to reach the fully inserted position using the DRPI system, remain aligned with their group step counter demand position. The group step counter demand position for control rod groups dropped by the RPR system is automatically set to zero steps withdrawn upon actuation of the RPR system. The prescribed sequence and overlap limits are restored upon the first withdrawal demand after actuation of the RPR system.

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment is that:

a. There be no violations of:

   1. Specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement with the maximum worth rod stuck fully withdrawn.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at or above their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 3).

The Required Actions in this LCO assure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and available SHUTDOWN MARGIN (SDM) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. Verification of $F_Q(Z)$, $F^N_{\Delta H}$, and SDM can be performed with the OPDMS provided the OPDMS rod position input is corrected to reflect the misaligned rod condition. If the OPDMS is not monitoring parameters, the verifications can be performed by off-line processing of in-core detector measurements and by performing a reactivity balance calculation for SDM confirmation. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by offline power distribution measurement if the OPDMS is not monitoring parameters. With the OPDMS monitoring parameters, and the rod position input corrected to the misaligned condition, the design peaking factors are confirmed by continuous monitoring such that off-line power distribution measurement is not required. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

RCCA OPERABILITY and the alignment of all control rods (RCCAs and GRCAs) are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore, they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

The limits on shutdown or control rod alignments assure that the assumptions in the safety analysis will remain valid. The requirements on RCCA OPERABILITY assure that upon reactor trip, the assumed reactivity will be available and will be inserted. The RCCA OPERABILITY requirements (i.e., trippability) apply to all RCCAs in the shutdown and control banks (not GRCAs) and are separate from the alignment requirements. The alignment requirements ensure that all control rods (RCCAs and GCRAs), and their respective banks maintain power distribution and rod alignment within analyzed limits. The RCCA OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move an RCCA (e.g., rod lift coil failures), but that do not impact trippability, do not result in RCCA inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and linear heating rates (LHR), or insufficient available SDM, which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment of all control rods are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. See LCO 3.1.1, “SHUTDOWN MARGIN (SDM),” for SDM in MODES 3, 4, and 5 and LCO 3.9.1, “Boron Concentration,” for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

In this situation, SDM verification must include the worth of the untrippable rod as well as an additional rod of maximum worth.

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BASES

ACTIONS (continued)

When one or more RCCA(s) are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate to identify the inoperable RCCA(s) to the OPDMS and establish continuous monitoring of SDM including the effects of the inoperable rod(s) or to otherwise determine SDM and, if necessary, to initiate boration to restore SDM.

A.2

If the inoperable RCCA(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. With the OPDMS monitoring parameters adverse peaking factors resulting from the misalignment can be detected. If the rod can be realigned within the Completion Time of 8 hours adverse burnup shadowing in the location of the misaligned rod can be avoided. With the OPDMS not monitoring parameters xenon redistribution can potentially cause adverse peaking factors which may not be detected. However, if the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant and operation may proceed without further restriction.

An alternative to realigning a single misaligned control rod to the group average position is to align the remainder of the group to the position of the misaligned control rod. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, “Shutdown Bank Insertion Limits,” and LCO 3.1.6, “Control Bank Insertion Limits.” The Completion Time of 1 hour with OPDMS not monitoring parameters gives the operator sufficient time to adjust the rod positions in an orderly manner. The system design allows alignment of a single rod to the bank or the bank to the rod to allow minimal impact on core flux.

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BASES

ACTIONS (continued)

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified within limit or boration must be initiated to restore SDM within limit.

Power operation may continue with one control rod misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary to update the OPDMS with the misaligned rod position indication information or use other methods to determine the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3.1, B.2.3.2.1, B.2.3.2.2, B.2.3.2.3, and B.2.4.

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% of RTP ensures that local LHR increases due to a misaligned control rod will not cause the core design criteria to be exceeded (Ref. 3). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Protection and Safety Monitoring System.

When a rod is known to be misaligned, there is a potential to impact the SDM and core margin to thermal limits. Since the core conditions can change with time, periodic verification of these parameters is required. The OPDMS provides continuous monitoring of these parameters following update with the misaligned rod position data. Performing SR 3.2.5.1 (Verify OPDMS monitored parameters within limits) on a frequency of 12 hours is sufficient to ensure that these requirements continue to be met while the reactor is operating with a misaligned control rod at THERMAL POWER less than or equal to 75% RTP.

When the OPDMS is not monitoring parameters, an offline verification of SDM and the peaking factors $F_Q(Z)$ and $F^N_{\Delta H}$ must be performed. The offline verification of SDM is performed with a frequency of 12 hours, but the verification of the peaking factors is performed on a frequency of 72 hours. The allowed Completion Time of 72 hours is reasonable

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BASES

ACTIONS (continued)

considering that the THERMAL POWER is reduced to ≤ 75% RTP. This provides significant margin to the fuel LHR and DNBR and subsequently to the peaking factor $F_Q(Z)$ and $F^N_{\Delta H}$ limits.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Accident (DBA) for the duration of operation under these conditions. The accident analyses presented in FSAR Chapter 15 (Ref. 3) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions for a single misaligned control rod cannot be completed within their Completion Times, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power condition in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM either through update of the OPDMS rod position data to reflect the misaligned control rods or a reactivity balance calculation, and to initiate boration if necessary to restore SDM to within its limit.

Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the CVS makeup pumps. Boration will continue until the required SDM is restored.

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BASES

ACTIONS (continued)

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing could continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect that a rod is beginning to deviate from its expected position. The specified Frequency takes into account other rod position information that is continuously available to the operator in the main control room so that during actual rod motion, deviations can immediately be detected.

The SR is modified by a Note that permits it to not be performed for rods associated with an inoperable demand position indication or an inoperable DRPI. The alignment limit is based on the position indication, and would not be available if the indication is inoperable. LCO 3.1.7, “Rod Position Indication,” provides Actions to verify the rods are in alignment when one or more rod position indications are inoperable.

Verification that individual rod positions are within alignment limits may utilize any position indication data, e.g., DRPI and/or incore detector data.

SR 3.1.4.2

Verifying each RCCA is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each RCCA would result in radial or axial power tilts, or oscillations. Exercising each individual RCCA every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each RCCA by 10 steps will not cause radial or axial power tilts, or oscillations, to occur.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the RCCAs. Between required performances of SR 3.1.4.2 (determination of RCCA OPERABILITY by movement), if a RCCA(s) is discovered to be immovable, but remains trippable and aligned, the RCCA(s) is considered to be OPERABLE. At any time, if a RCCA(s) is immovable, a determination of the trippability (OPERABILITY) of the RCCA(s) must be made, and appropriate action taken.

GRCA are excluded from this Surveillance because they are not considered in the calculation of SDM in MODES 1 and 2 with $k_{eff} \geq 1.0$. In Mode 2 with $k_{eff} < 1.0$, SR 3.1.4.2 is not applicable because GRCAs must be verified to be fully inserted to be credited in the calculation of SDM.

SR 3.1.4.3

Verification of RCCA drop times allows the operator to determine that the maximum RCCA drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring RCCA drop times prior to reactor criticality, after each reactor vessel head removal and each earthquake requiring plant shutdown, ensures that the reactor internals and rod drive mechanism will not interfere with RCCA motion or RCCA drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under conservative conditions. GRCA are excluded from this Surveillance because they are not considered in the calculation of SDM in MODES 1 and 2 with $k_{eff} \geq 1.0$. In MODE 2 with $k_{eff} < 1.0$, SR 3.1.4.3 is not applicable because GRCAs must be verified to be fully inserted to be credited in the calculation of SDM.

This Surveillance is performed during a plant outage due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR Chapter 15, “Accident Analysis.”

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown banks are initial assumptions in the safety analyses which assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, “Reactor Design,” GDC 26, “Reactivity Control System Redundancy and Protection,” GDC 28, “Reactivity Limits” (Ref. 1), and 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors” (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The control rods are divided among seven control banks and four shutdown banks. Shutdown banks are designated S1, S2, S3, and S4. Control banks are designated MA, MB, MC, MD, M1, M2, and AO. The control banks provide reactivity and power distribution control during unit operation while the shutdown banks provide rapid shutdown capability for the reactor. Each shutdown bank consists of eight Rod Cluster Control Assemblies (RCCAs). Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of four RCCAs that are electrically paralleled to step simultaneously. A bank consists of two groups that are moved in a staggered fashion, but always within one step of each other. See LCO 3.1.4, “Rod Group Alignment Limits,” for RCCA OPERABILITY and control rod alignment requirements, and LCO 3.1.7, “Rod Position Indication,” for position indication requirements.

The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control operator. During normal unit operation,

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BASES

BACKGROUND (continued)

the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs, except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core as allowed by LCO 3.1.6, “Control Bank Insertion Limits.” The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, “SHUTDOWN MARGIN (SDM)”) following a reactor trip from full power. The insertion of all RCCAs (except for the most reactive RCCA which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at the rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown bank rod.

The acceptance criteria for addressing shutdown bank insertion limits are that:

a. There be no violations of:

   1. specified acceptable fuel design limits, or,

   2. RCS pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

These limits are developed using the methodologies identified in Technical Specification 5.6.3.

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This in conjunction with LCO 3.1.6, “Control Bank Insertion Limits,” and LCO 3.2.5, “On-line Power Distribution Monitoring System (OPDMS) Monitored Parameters,” Item d, “SDM,” ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits with the reactor in MODE 1 and MODE 2. The applicability in MODE 2 begins at initial AO bank withdrawal, during an approach to criticality, and continues throughout MODE 2 until all control rods are again fully inserted by reactor trip or by shutdown. In MODE 3, 4, 5, or 6 the shutdown banks are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 2 with $k_{eff} < 1.0$, 3, 4, and 5. LCO 3.9.1, “Boron Concentration” ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating that the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and may require the shutdown banks to move below the LCO limits temporarily, which would normally violate the LCO. The rod movements in the plant procedure for performing SR 3.1.4.2 are small and result in no more than one shutdown bank potentially moving below the LCO at a time. If a reactor trip occurs during performance of SR 3.1.4.2, the potential loss in available negative reactivity insertion will be negligible.

ACTIONS

A.1.1, A.1.2, and A.2

When one or more shutdown banks is not within insertion limits, 2 hours are allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by the continuous monitoring of SDM by the OPDMS (see LCO 3.2.5) and adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by the OPDMS or by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

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BASES

ACTIONS (continued)

The allowed Completion Time of 1 hour for verification of SHUTDOWN MARGIN is an acceptable time for an operator to interface with the OPDMS to update the rod position of the control rods in violation of the Shutdown Bank Insertion Limits, or for the performance of the reactivity balance calculation.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the main control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hours Frequency takes into account other information available in the main control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. FSAR, Chapter 15, “Accident Analysis.”

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the control banks are initial assumptions in the safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, “Reactor Design,” GDC 26, “Reactivity Control System Redundancy and Protection,” GDC 28, “Reactivity Limits” (Ref. 1) and 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors” (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The control rods are divided among seven control banks and four shutdown banks. Shutdown banks are designated S1, S2, S3, and S4. Control banks are designated MA, MB, MC, MD, M1, M2, and AO. The control banks provide reactivity and power distribution control during unit operation while the shutdown banks provide rapid shutdown capability for the reactor. Each control bank consists of either Rod Cluster Control Assemblies (RCCAs) or Gray Rod Cluster Assemblies (GRCAs). RCCAs are used in control banks M1, M2, and AO, while GRCAs are used in control banks MA, MB, MC, and MD. Each bank may be further subdivided into two groups to provide for precise reactivity control. A control bank group consists of four RCCAs or GRCAs that are electrically paralleled to step simultaneously. The only exception to this is the AO bank, which contains a five-rod group that steps simultaneously. The five-rod group is powered by two separate sides of the same power selecting cabinet. A bank consists of one or more groups that are moved in a staggered fashion, but always within 1 step of each other. See LCO 3.1.4, “Rod Group Alignment Limits,” for RCCA OPERABILITY and control rod alignment requirements, and LCO 3.1.7, “Rod Position Indication,” for position indication requirements.

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BASES

BACKGROUND (continued)

The control bank insertion sequence and overlap limits are specified in the COLR. The control banks are required to be at or above the applicable insertion limit lines. There are two sets of insertion limits applicable to control banks MA, MB, MC, MD, M1 and M2. Which set is applicable depends on the status of the On-Line Power Distribution Monitoring System (OPDMS).

When the OPDMS is monitoring parameters, it monitors SHUTDOWN MARGIN as well as other key power distribution parameters defined in LCO 3.2.5, “OPDMS Monitored Parameters.” Because of this, the consideration of these parameters can be removed from the basis of the control bank insertion limits that are applicable when the OPDMS is monitoring parameters. When the OPDMS is not monitoring parameters, the applicable rod insertion limits must include SHUTDOWN MARGIN and the key power distribution limits normally monitored by the OPDMS as part of their basis. As a result, the control bank insertion limits applicable when OPDMS is not monitoring parameters may be more restrictive. If the insertion limits applicable when the OPDMS is monitoring parameters are conservatively defined to include consideration of SHUTDOWN MARGIN and the other key power distribution limits normally monitored by OPDMS, then both sets of insertion limits may be identical. The COLR identifies which control bank insertion limits apply to bank MA, MB, MC, MD, M1, and M2 when the OPDMS is monitoring and not monitoring parameters.

There is also a single insertion limit applicable to the AO bank in the COLR, which does not depend on the status of the OPDMS. Since the AO bank is moved independently of the other control banks, there is no sequence and overlap setting applicable for the AO bank.

The control banks are used for precise reactivity control of the reactor (banks MA, MB, MC, MD, M1, and M2) and axial power distribution control (AO bank). Banks M1, M2, and AO, which contain only RCCAs, are also credited as being available, as required by LCO 3.1.4, to provide negative trip reactivity for SHUTDOWN MARGIN calculations in MODE 1 and in MODE 2 with $k_{eff} \geq 1.0$. The positions of the control banks are normally controlled automatically by the Plant Control System (PLS), but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting), while the AO bank is capable of effectively controlling the axial power distribution to near the AO target value.

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BASES

BACKGROUND (continued)

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.4, “Rod Group Alignment Limits,” LCO 3.1.5, “Shutdown Bank Insertion Limits,” LCO 3.1.6, “Control Bank Insertion Limits,” and LCO 3.2.5, “On-Line Power Distribution Monitoring System (OPDMS) – Monitored Parameters,” when the OPDMS is monitoring parameters, or LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” and LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” when the OPDMS is not monitoring parameters, provide limits on control component operation and on monitored process variables which ensure that the core operates within the fuel design criteria.

The control bank insertion and alignment limits and power distribution limits are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the control bank insertion limits assure the required SDM is maintained when the OPDMS is not monitoring parameters.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

Certain control banks are pre-selected for rapid insertion by the Rapid Power Reduction (RPR) system during load rejection transients or if the main turbine or a main feedwater pump trip occurs above a pre-selected plant power level. The prescribed sequence and overlap limits are restored upon the first withdrawal demand after actuation of the RPR system.

APPLICABLE SAFETY ANALYSES

The applicable control bank insertion limits, in conjunction with the applicable LCOs in Section 3.2, “Power Distribution Limits,” are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function. In addition, when the OPDMS is not monitoring parameters, the applicable control bank insertion limits are required to maintain SHUTDOWN MARGIN.

The acceptance criteria for addressing control bank insertion limits are that:

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BASES

APPLICABLE SAFETY ANALYSES (continued)

a. There be no violations of:

   1. Specified fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by the continuous monitoring of the OPDMS and by limiting the control and shutdown bank insertion limits when the OPDMS is not monitoring parameters. When the OPDMS is not monitoring parameters, the rod insertion limits assure that there is sufficient negative reactivity insertion capability available from the RCCAs to shut down the reactor to hot zero power with a reactivity margin which assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worth.

The control bank insertion limits ensure that safety analyses assumptions for SDM (with OPDMS not monitoring parameters), ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

These Limits are developed using the methodologies identified in Technical Specification 5.6.3.

The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and physical insertion as defined in the COLR, must be maintained because they serve the function of preserving power distribution within applicable limits, ensuring that the SDM is maintained (when OPDMS is not monitoring parameters), ensuring that ejected rod worth is maintained, consistent with safety analysis assumptions and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more

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BASES

LCO (continued)

uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. There are two sets of insertion limits applicable to control banks MA, MB, MC, MD, M1 and M2 depending on OPDMS status. With OPDMS not monitoring parameters, insertion limits must be maintained which preserve the assumed power distribution, ejected rod worth, SDM, and reactivity insertion rate accident analysis assumptions. With OPDMS continuously monitoring power distribution and SDM, insertion limits must be maintained which preserve the accident analysis assumptions. The insertion limits with OPDMS monitoring parameters may be less restrictive than the insertion limits with OPDMS not monitoring parameters. Sequence and overlap limits must be maintained since they are necessary to preserve the assumed power distribution in the accident analysis.

Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor accident analysis precondition assumptions would be exceeded in these MODES.

The applicability requirements are modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the RCCA to move, and may require the control bank to move below the LCO limits, which could violate the LCO.

The second Note suspends LCO applicability to GRCA banks during GRCA bank sequence exchange operations for up to one hour. If a GRCA bank sequence exchange is initiated with the GRCA banks at different starting positions, the exchanging banks will move out of sequence and overlap limits for several minutes during the sequence exchange. This operation, which occurs frequently throughout the fuel cycle, would normally violate the LCO. One hour provides sufficient time to complete the exchange while limiting any impact on local power distribution from the GRCA banks being out of sequence and overlap. A prerequisite to performing the GRCA sequence exchange is that a sufficient peaking factor margin be available. This will provide assurance that core peaking factor limits will not be challenged while performing the exchange. Considering that this operation is a planned maneuver and involves exchanging the position of low worth GRCA banks over a short time period, it is acceptable to suspend applicability of this LCO.

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BASES

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits (except during a GRCA sequence exchange), they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2, may have been reduced below the applicable limit. One hour allows enough time for the operator to ensure the OPDMS is updated with correct control rod position data if necessary or by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1 if OPDMS is not monitoring parameters.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration (except during a GRCA sequence exchange), they must be restored to meet the limits.

If Condition A and/or B is entered due to RPR system actuation, it is possible that the selected control banks may be dropped into a configuration that exceeds the allowed control bank insertion limit, and it is likely that the inserted control banks will be out of their normal overlap condition. The Required Actions are acceptable because the immediate power reduction associated with the actuation of the RPR system prevents exceeding fuel design basis limits after the actuation. In addition, the immediate power reduction and subsequent xenon transient results in maintaining the available SDM at or above the SDM that was present just before the RPR actuation occurred. Required Action A.2 and/or Required Action B.2 will continue to require restoration of the control banks to an acceptable configuration within 2 hours. This is sufficient to preclude reaching an unacceptable power distribution or SDM condition in the longer term after an RPR actuation.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain outside the limits for an extended period of time.

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BASES

ACTIONS (continued)

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with $k_{eff} < 1.0$, or to a lower MODE, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable based on operating experience for reaching the required MODE from full power condition in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was verified long before criticality, xenon concentration could change sufficiently to make the ECP substantially in error. Conversely, verifying the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP verification within 4 hours prior to criticality allows the operator some flexibility to schedule the ECP verification with other startup activities.

SR 3.1.6.2

Verification of the control banks insertion limits at a Frequency of 12 hours is sufficient to detect control banks that may be approaching the insertion limits since the insertion limits are monitored and alarms will occur on approach to and/or the exceeding of the limit.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. The sequence and overlap requirements must be met while in the MODES of APPLICABILITY except during rod trippability testing per SR 3.1.4.2, or GRCA bank sequence exchange when the LCO does not apply for up to one hour. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

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BASES

REFERENCES

1. 10CFR50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10CFR50.46.
3. FSAR Chapter 15, “Accident Analysis.”

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences (AOOs), and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indication to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control bank Rod Cluster Control Assemblies (RCCAs) is an initial assumption in the safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the misalignment safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

RCCAs and Gray Rod Cluster Assemblies (GRCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The rods are divided among control banks and shutdown banks. Each bank may be further subdivided into one or more groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent means: the Bank Demand Position Indication (commonly called group step counters) and the Digital Rod Position Indication (DRPI).

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BASES

BACKGROUND (continued)

Bank Demand Position Indication counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group receive the same signal to move and should, therefore, be at the same position indicated by the group step counter for that group. Bank Demand Position Indication is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

DRPI provides a highly accurate indication of actual control rod position, at a lower precision than the step counters. This indication is based on inductive analog signals from detectors consisting of a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the indication, the inductive coils are divided into A and B coil stacks. The adjacent coils in a given detector will alternate between the A and B stacks so that if there is any single coil failure in a DRPI detector (or coil failures limited to one stack) the DRPI will function at half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches. Coil stack A data is transmitted to Data Cabinet A, and coil stack B data is transmitted to Data Cabinet B. Any failure of a single data cabinet would result in DRPI detectors functioning at half accuracy.

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, “Shutdown Bank Insertion Limits,” LCO 3.1.6, “Control Bank Insertion Limits,” LCO 3.2.5, “OPDMS Monitored Parameters”). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, “Rod Group Alignment Limits”). Control rod positions are continuously monitored to provide operators with information that assures the plant is operating within the bounds of the accident analysis assumptions.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The control rod position indication channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The control rod position indication monitors control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that one DRPI for each control rod and one Bank Demand Position Indication for each group shall be OPERABLE. For the control rod position indication to be OPERABLE requires meeting the SR of the LCO and the following:

a. The DRPI for each control rod functioning with full or half accuracy, and

b. The Bank Demand Indication for each group has been calibrated either in the fully inserted position or to the DRPI.

OPERABILITY of the DRPI is not affected by the coils as long as all A or all B coils in a given detector are functioning, because position indication for the rod would be provided with half accuracy or better. Half accuracy indication would also be provided if a single data cabinet fails, as long as there are no additional failures for the coils associated with the functioning data cabinet.

The 12 step agreement limit between the Bank Demand Position Indication and the DRPI indicates that the Bank Demand Position Indication is adequately calibrated and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit given in LCO 3.1.4 in position indication for a single control rod ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements provide adequate assurance that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indication channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

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BASES

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCOs 3.1.4, 3.1.5, and 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods has the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System (RCS).

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable DRPI and each demand position indication. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indication.

A.1

When one DRPI per group fails, the position of the rod can still be determined indirectly by use of the incore detectors. The Required Action may also be satisfied by verifying at least once per 8 hours that applicable power distribution limits (e.g., $F_Q(Z)$ and $F^N_{\Delta H}$) and SDM (or insertion limits) are within the limits provided in the COLR, provided the nonindicating rods have not been moved. Steady state equilibrium normal power operation does not require excessive movement of banks. If a bank has been significantly moved (more than 24 steps in one direction), the Actions of C.1 or C.2 below are required. Therefore, verification of rod position within the Completion Time of 8 hours is adequate to allow continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 2).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

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BASES

ACTIONS (continued)

B.1 and B.2

When more than one DRPI per group is inoperable, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual (both the MSHIM and AO banks) assures automatic unplanned rod motion will not occur. This action, together with the indirect position determination available via incore detectors will minimize the potential for rod misalignment. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned or automatic rod motion must be prevented while in this Condition. Manual rod motion, RCS boron concentration changes, or changes in reactor load may occur as necessary to respond to changes in RCS average temperature and AFD.

Verification of control rod position once per 8 hours (per Required Action A.1) is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the DRPI to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

If one or more rods has been significantly moved, the Required Action of C.1 and C.2 below is required.

C.1 and C.2

When one or more rods with inoperable DRPI have been moved in excess of 24 steps in one direction since the position was last determined, the Required Action C.1 is to be performed within 4 hours by analysis of the incore dectector data to determine the rod position to verify that these rods are still properly positioned relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

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BASES

ACTIONS (continued)

D.1.1 and D.1.2

With one or more demand position indication per bank inoperable in one or more banks, the rod positions can be determined by the DRPI. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indication is OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

D.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 2). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power to ≤ 50% RTP.

E.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps provides assurance that the DRPI is operating correctly.

This surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

The Surveillance is modified by a Note which states it is not required to be met for DRPI associated with a rod that does not meet LCO 3.1.4. If a rod is known to not be within 12 steps of the group demand position, the ACTIONS of LCO 3.1.4 provide the appropriate Actions.

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BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
2. FSAR, Chapter 15, “Accident Analysis.”

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions – MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B, (Ref. 1) requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for prior approval by the NRC, for the purpose of conducting tests and experiments not described in the FSAR, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power and after each refueling. The PHYSICS TEST requirements for reload fuel cycles assure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TEST procedures are written and approved in accordance with established formats. The procedures include information necessary to permit a detailed execution of the testing required, to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long-term power operation.

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BASES

BACKGROUND (continued)

The typical PHYSICS TESTS performed for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration – Control Rods Withdrawn;

b. Control Rod Worth;

c. Isothermal Temperature Coefficient (ITC).

These tests are performed in MODE 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration – Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has four alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is calculated based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and while varying the reactor coolant boron concentration to maintain HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated

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BASES

BACKGROUND (continued)

for the remaining control banks. The fourth method, Dynamic Rod Worth Measurement (DRWM), moves each bank, individually, into the core to determine its worth. The bank is dynamically inserted into the core while data is acquired from the excore channel. While the bank is being withdrawn, the data is analyzed to determine the worth of the bank. This is repeated for each control and shutdown bank. Performance of this test will violate LCO 3.1.4, “Rod Group Alignment Limits,” LCO 3.1.5, “Shutdown Bank Insertion Limits,” or LCO 3.1.6, “Control Bank Insertion Limits.”

c. The ITC Test measures the ITC of the reactor. This test is performed at HZP. The method is to vary the RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The final measured ITC may be the average of more than one measurement. Performance of this test could violate LCO 3.4.2, “RCS Minimum Temperature for Criticality.”

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

FSAR Chapter 14 defines requirements for initial testing of the facility, including low power PHYSICS TESTS. FSAR Sections 14.2.10.2 and 14.2.10.3 (Ref. 6) summarize the initial criticality and low power tests.

Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-2005 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for the LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in:

LCO 3.1.3 “Moderator Temperature Coefficient (MTC),”
LCO 3.1.4 “Rod Group Alignment Limits,”
LCO 3.1.5 “Shutdown Bank Insertion Limits,”

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BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.6 “Control Bank Insertion Limits,” and
LCO 3.4.2 “RCS Minimum Temperature for Criticality,”

are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 541°F, and SDM is within the limits provided in the COLR.

PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," Functions 1, 2, 3, and 4 may be reduced to 3 required channels during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 541°F,

b. SDM is within the limits provided in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

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BASES

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as “During PHYSICS TESTS initiated in MODE 2” to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER EXCEED 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limits.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest T_{avg} is < 541°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 541°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Action of Condition C cannot be completed within the associated Completion Time, the plant must be placed in a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, to reach MODE 3 from MODE 2 HZP conditions in an orderly manner and without challenging plant systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the RCS lowest loop T_{avg} is ≥ 541°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will provide assurance that the initial conditions of the safety analyses are not violated.

SR 3.1.8.2

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

SDM may also be verified using the OPDMS in place of a reactivity balance calculation. The OPDMS continuously monitors the available SDM during PHYSICS TESTS and considers all of the above effects.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

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BASES

REFERENCES

1. 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”
2. 10 CFR 50.59, “Changes, Tests and Experiments.”
3. Regulatory Guide 1.68, Revision 2, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” August 1978.
4. ANSI/ANS-19.6.1-2005, “Reload Startup Physics Tests for Pressurized Water Reactors,” American National Standards Institute, November 29, 2005.
5. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.
6. FSAR Chapter 14, “Initial Testing Program.”

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves

BASES

BACKGROUND

One of the principal functions of the CVS system is to maintain the reactor coolant chemistry conditions by controlling the concentration of boron in the coolant for plant startups, normal dilution to compensate for fuel depletion, and shutdown boration. In the dilute mode of operation, unborated demineralized water may be supplied directly to the reactor coolant system. Another of the principal functions of the CVS system is to maintain the reactor coolant inventory by providing water makeup for reactor coolant system (RCS) LEAKAGE, shrinkage of the reactor coolant during cooldowns, and RCS boron concentration changes. In the automatic makeup mode of operation, the pressurizer water level starts and stops CVS makeup to the RCS.

Although the CVS is not considered a safety related system, certain functions of the system are considered safety related functions. The appropriate components have been classified and designed as safety related. The safety related functions provided by the CVS include containment isolation of CVS lines penetrating containment, termination of inadvertent boron dilution, and preservation of the Reactor Coolant System (RCS) pressure boundary, including isolation of CVS letdown from the RCS. Another of the safety related functions provided by the CVS is the termination of RCS makeup to prevent overfilling of the pressurizer during non-LOCA transients or to prevent steam generator overfilling during a steam generator tube rupture. The CVS makeup line isolation valves provide this RCS makeup isolation function.

APPLICABLE SAFETY ANALYSES

One of the initial assumptions in the analysis of an inadvertent boron dilution event (Ref. 1) is the assumption that the increase in core reactivity, created by the dilution event, can be detected by the source range instrumentation. The source range instrumentation will then supply a signal to the CVS demineralized water isolation valves and the CVS makeup line isolation valves in the CVS causing these valves to close and terminate the boron dilution event. Thus the makeup line isolation valves and the demineralized water isolation valves are components which function to mitigate or prevent an anticipated operational occurrence (AOO).

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BASES

APPLICABLE SAFETY ANALYSES (continued)

One of the initial assumptions in the analysis of several non-LOCA events and during a steam generator tube rupture accident is that excessive CVS makeup to the RCS may aggravate the consequences of the accident. The need to isolate the CVS makeup to the RCS is detected by the pressurizer level instruments or the steam generator narrow range level instruments. These instruments will supply a signal to the CVS makeup line isolation valves causing these valves to close and terminate RCS makeup. Thus the CVS makeup isolation valves are components which function to mitigate an accident.

CVS isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that at least two CVS demineralized water isolation valves (CVS-PL-V136A and V136B) and two CVS makeup line isolation valves (CVS-PL-V090 and V091) be OPERABLE assures that there will be redundant means available to terminate or prevent an inadvertent boron dilution event. The requirement that at least two CVS makeup isolation valves (CVS-PL-V090 and V091) be OPERABLE also assures that there will be redundant means available to terminate CVS makeup to the RCS during a non-LOCA event or a steam generator tube rupture accident should that become necessary to provide protection from overfilling the pressurizer or steam generator. In addition, LCO 3.6.3, “Containment Isolation Valves,” provides additional requirements for the CVS makeup line isolation valves.

APPLICABILITY

The requirement that at least two CVS demineralized water isolation valves and two CVS makeup line isolation valves be OPERABLE is applicable in MODES 1, 2, 3, 4, and 5 because a boron dilution event is considered possible in these MODES, and the automatic closure of these valves is assumed in the safety analysis.

In MODES 1 and 2, the detection and mitigation of a boron dilution event does not assume the detection of the event by the source range instrumentation. In these MODES, the event would be signaled by an intermediate range trip, a trip on the Power Range Neutron Flux - High (low setpoint nominally at 25% RTP), or Overtemperature delta T. The two demineralized water isolation valves close automatically upon reactor trip in MODES 1 and 2. In MODE 6, a dilution event is precluded by the requirement in LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path.

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BASES

APPLICABILITY (continued)

The pressurizer overfill event is possible in MODES 1, 2, and 3, and MODE 4 with all four cold leg temperatures > 275°F. In these applicable MODES, the need to isolate the CVS makeup to the RCS is detected by the pressurizer level instruments (High setpoint coincident with safeguards actuation or High 2 setpoint).

The steam generator tube rupture (SGTR) event is possible in MODES 1, 2, and 3, and in MODE 4 at an RCS temperature ≥ 350°F. In MODE 4 with an RCS temperature < 350°F and MODES 5 and 6, the RCS pressure and temperature are reduced and an SGTR is not credible. However, the Applicability conservatively requires OPERABILITY of this Function in MODE 4 with RCS temperature < 350°F (i.e., all of MODE 4). In these applicable MODES, the need to isolate the CVS makeup to the RCS is detected by the steam generator narrow range level instruments (High setpoint coincident with reactor trip (P-4) or High 3 setpoint).

ACTIONS

The ACTIONS are modified by two Notes. Note 1 allows the affected flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the main control room. In this way, the flow path(s) can be rapidly isolated when a need for isolation is indicated.

A second Note provides clarification that, for this LCO, separate Condition entry is allowed for each valve. This is acceptable, since the Required Actions provide appropriate compensatory actions for each inoperable isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If only one demineralized water isolation valve and/or only one makeup line isolation valve are/is OPERABLE, the redundant valve(s) must be restored to OPERABLE status in 72 hours. The allowed Completion Time assures expeditious action will be taken, and is acceptable because the safety function of automatically isolating the clean water source and/or RCS makeup can be accomplished by the redundant isolation valve(s).

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BASES

ACTIONS (continued)

B.1

If the Required Actions and associated Completion Time of Condition A are not met, or if both CVS demineralized water isolation valves or both CVS makeup line isolation valves are not OPERABLE (i.e., not able to be closed automatically), then the affected flow path(s) must be isolated.

For inoperabilities of CVS makeup isolation valves, isolating the "affected flow path" must prevent flow from the CVS makeup pumps to the RCS in order to provide protection from overfilling. Isolation can be accomplished by manually closing the CVS makeup isolation MOVs or alternatively, manual valve(s) in the makeup line between the makeup pumps and the RCS.

For inoperabilities of CVS demineralized water isolation valves isolation from the demineralized water supply flow path to the RCS can be accomplished by manually isolating the CVS demineralized water isolation valve(s) or by positioning the 3-way blend valve to only take suction from the boric acid tank. Alternatively, the dilution path may be isolated by closing appropriate isolation valve(s) in the flow path(s) from the demineralized water storage tank to the reactor coolant system.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the CVS demineralized water isolation valves and makeup line isolation valves stroke closed demonstrates that the valves can perform their safety related function. The Frequency is in accordance with the Inservice Testing Program.

SR 3.1.9.2

Verification that the closure time of each RCS makeup isolation valve is less than that assumed in the safety analysis (i.e., < 30 seconds), is performed by measuring the time required for each valve to close on an actual or simulated actuation signal. The actual or simulated actuation signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. The Frequency is in accordance with the Inservice Testing Program.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.3

This SR verifies that each CVS demineralized water isolation valve actuates to the correct position on an actual or simulated actuation signal. The actual or simulated actuation signal is processed through the component interface module to verify the continuity between the output of component interface module and the valve. The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 Rod Withdrawal Test Exception – MODE 5

BASES

BACKGROUND

In MODE 5, there are exceptions required to allow performance of tests that require rod withdrawal and movement. The typical tests that require rod withdrawal in MODE 5 are:

a. Cold Rod Drop Time Testing. Successful performance of cold rod drop time testing in MODE 5 following reactor vessel assembly provides an early opportunity to discover any significant anomalies in rod drop performance that might indicate that the rod drop time test of SR 3.1.4.3 in LCO 3.1.4, “Rod Group Alignment Limits,” would not be met when performed;

b. DRPI System and Bank Demand Position Indication System Surveillance Testing. SR 3.1.7.1 specifies verification that each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel, prior to criticality after each removal of the reactor head. These indication systems are needed to support Cold Rod Drop time testing;

c. Rod Control System Testing. Successful testing of the Rod Control System in MODE 5 is necessary to support the performance of Cold Rod Drop time testing following reactor vessel (RV) assembly; and

d. Control Rod Drive Mechanism (CRDM) Testing. Successful testing of the CRDMs in MODE 5 is necessary to support the performance of Cold Rod Drop time testing following RV assembly.

The MODE 5 Rod Withdrawal Test Exception permits relaxation of LCO3.4.4 to allow performance of tests that require rod withdrawal and movement. Compensating requirements are applied to provide adequate assurance of subcriticality during the testing.

Preliminary rod drop time testing, Digital Rod Position Indication (DRPI) System and Bank Demand Position Indication System surveillance testing, Rod Control System testing, and Control Rod Drive Mechanism (CRDM) testing may be performed prior to plant heatup following refueling.

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BASES

APPLICABLE SAFETY ANALYSES

The tests described above require operating the plant with less than the RCS flow required by LCO 3.4.4, “RCS Loops.” However, this exception is acceptable under the subcritical boron concentration limitations imposed by the LCO. Protection from potential reactivity events during shutdown conditions is provided by adequate boration to ensure the reactor core remains subcritical at all times. As such, prevention of criticality is the acceptance criterion for exercising this test exception. The required boration will assure that the core remains subcritical at all times during any planned rod withdrawals or any inadvertent rod withdrawal errors. As such, there is no challenge to DNB or other fuel acceptance criteria.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

This LCO allows a MODE 5 exception to the LCO 3.4.4 requirement that 100% reactor coolant system flow is required whenever the Plant Control System is capable of rod withdrawal or one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s). An exception to this requirement is needed to permit tests involving rod withdrawal.

Subcritical boron concentrations must be met during the use of this test exception LCO, assuming that all shutdown, control, and gray rods are fully withdrawn. The boration limitation specified precludes reactor criticality.

APPLICABILITY

This LCO is applicable in MODE 5 with LCO 3.4.4, “RCS Loops,” not met. This allows testing that requires rod withdrawal. During testing, rod withdrawal is enabled with the Reactor Trip Breakers (RTBs) closed and the Plant Control System capable of rod withdrawal. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

Rod withdrawal under this LCO may only be enabled in MODE 5 to perform testing.

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BASES

ACTIONS

A.1 and A.2

If the LCO is not met, the reactor must be brought to a MODE in which the LCO does not apply. To achieve this status, action is initiated immediately to fully insert all rods and, within 1 hour, the PLS is placed in a condition incapable of rod withdrawal. This precludes rod withdrawal events which are unanalyzed. The Completion Times for Required Actions A.1 and A.2 are reasonable, based on operating experience, to reach the specified condition in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

This SR requires verification that boron concentration of the reactor coolant system is greater than the all-rods-out boron concentration providing a keff < 0.99. Assuming all rods are fully withdrawn ensures that the reactor will remain subcritical for all planned tests.

The Frequency of 12 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required boron concentration. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES None.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation with the On-line Power Distribution Monitoring System (OPDMS) not monitoring parameters, the global power distribution is limited by LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” and LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” which are directly and continuously measured process variables. These LCOs along with LCO 3.1.6, “Control Bank Insertion Limits,” maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

With the OPDMS monitoring parameters, peak linear heat rate (which is proportional to $F_Q(Z)$) is measured continuously. With the OPDMS not monitoring parameters, $F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

With the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$ with the OPDMS not monitoring parameters. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ which are present during a nonequilibrium situation such as load following.

To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor to account for the calculated worst case transient conditions according to the analytical method in Reference 5.

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BASES

BACKGROUND (continued)

Core monitoring and control under non-equilibrium conditions and the OPDMS not monitoring parameters are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

In the unlikely event that measurements indicate that the limit for $F_Q^W(Z)$ could be exceeded during future non-equilibrium operation, a more restrictive CAOC operating space specified in the Core Operating Limits Report (COLR) may be implemented to restore margin to the $F_Q^W(Z)$ limit. A CAOC operating space is a unique combination of an allowable AFD band and Control Bank Insertion Limits. A more restrictive CAOC operating space would employ a narrower AFD band, shallower Control Bank Insertion Limits, or a combination of the two. W(z) functions for each CAOC operating space are specified in the COLR. If none of the CAOC operating spaces provides adequate margin to the $F_Q^W(Z)$ limit, then THERMAL POWER must be limited to less than RTP.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed a limit of 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed applicable limits (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

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APPLICABLE SAFETY ANALYSES (continued)

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting (i.e., lower) relative to the $F_Q(Z)$ assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents. These limits are developed using the methodologies identified in Technical Specification 5.6.3.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} \qquad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The actual values of CFQ are given in the COLR; however, CFQ is normally a number on the order of 2.60. The normalized $F_Q(Z)$ as a function of core height is independent.

For Constant Axial Offset Control (CAOC) operation (Ref. 6), $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

An $F_Q^C(Z)$ evaluation requires obtaining an incore power distribution measurement in MODE 1. From the incore power distribution measurement results the measured value of $F_Q(Z)$, called $F_Q^M(Z)$ is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z) * F_Q^{MU}(Z)$$

where $F_Q^{MU}(Z)$ is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty (Ref. 4). $F_Q^{MU}(Z)$ is provided in the COLR.

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BASES

LCO (continued)

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z) * \frac{[W(Z)]^{COLR}}{P} * A_Q(Z)\ R_j, \text{ for } P > 0.5$$

$$F_Q^W(Z) = F_Q^C(Z) * \frac{[W(Z)]^{COLR}}{0.5} * A_Q(Z)\ R_j, \text{ for } P \leq 0.5$$

Where $[W(Z)]^{COLR}$ is a cycle and burnup dependent function, specified in the COLR, that accounts for power distribution transients encountered during non-equilibrium normal operation. $[W(Z)]^{COLR}$ functions are specified for each analyzed CAOC operating space (i.e., each unique combination of AFD band and Control Bank Insertion Limits). The $[W(Z)]^{COLR}$ functions account for the limiting non-equilibrium axial power shapes postulated to occur during normal operation for each CAOC operating space. Limiting power shapes at both full and reduced power operation are considered in determining the maximum values of $[W(Z)]^{COLR}$. The $[W(Z)]^{COLR}$ functions also account for the following effects: (1) the increase in radial peaking in rodded core planes due to the presence of control rods during non-equilibrium normal operation, (2) the increase in radial peaking that occurs during part-power operation due to reduced fuel and moderator temperatures, and (3) the increase in radial peaking due to non-equilibrium xenon effects. The $[W(Z)]^{COLR}$ functions are normally calculated assuming that the Surveillance is performed at the Target Axial Offset core conditions. Surveillance specific $[W(Z)]^{COLR}$ values may be generated for a given surveillance core condition.

P is the THERMAL POWER / RTP.

$A_Q(Z)$ is a function that adjusts the $F_Q^W(Z)$ surveillance for differences between the reference core conditions assumed in generating the $[W(Z)]^{COLR}$ function and the actual core condition that exists when the Surveillance is performed. Normally, this reference core condition is defined as the same core condition that is used to establish the target AO (i.e., the target axial flux difference at RTP) in LCO 3.2.3. For simplicity, $A_Q(Z)$ can be assumed to be 1.0 when the $F_Q^W(Z)$ surveillance is performed at a core condition consistent with updating the target AO and when no control bank is inserted below the axial elevation where the limiting F_Q margin is predicted to occur (as specified in the COLR). $A_Q(Z)$ can also be assumed to be 1.0 if the $F_Q^W(Z)$ surveillance is not performed

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BASES

LCO (continued)

at conditions consistent with updating the target AO, and a surveillance specific $[W(Z)]^{COLR}$ function associated with performing the surveillance at the applicable core conditions is specified in the COLR. If the conditions specified in the COLR for assuming $A_Q(Z) = 1.0$ are not met and a surveillance specific $[W(Z)]^{COLR}$ function has not been specified, then the $A_Q(Z)$ function is calculated using the NRC approved methodology in Reference 7.

R_j is a cycle and burnup dependent analytical factor specified in the COLR that accounts for potential increases in $F_Q^W(Z)$ between Surveillances. R_j values are provided for each CAOC operating space.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Violating the $F_Q(Z)$ LCO limits could result in unacceptable consequences if a design basis event were to occur while $F_Q(Z)$ exceeds its specified limits. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits.

APPLICABILITY

When the OPDMS is not monitoring parameters and core power distribution parameters cannot be continuously monitored, it is necessary to determine $F_Q(Z)$ on a periodic basis. Furthermore, the $F_Q(Z)$ limits must be maintained in MODE 1 with THERMAL POWER ≥ 25% RTP to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in MODE 1 with THERMAL POWER < 25% RTP is not required because local power peaking at very low power levels is not limiting with respect to design basis accident analyses and because the incore instrumentation system may not be able to provide an accurate measurement of the core flux distribution at very low THERMAL POWER levels. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

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BASES

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% of RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for fuel manufacturing tolerances and flux map measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

If an F_Q surveillance is performed at 100% RTP conditions, and both $F_Q^C(Z)$ and $F_Q^W(Z)$ exceed their limits, the option to reduce the THERMAL POWER limit in accordance with Required Action B.2.1 instead of implementing a new operating space in accordance with Require Action B.1.1, will result in a further power reduction after Required Action A.1 has been completed. However, this further power reduction would be permitted to occur over the next 4 hours. In the event the evaluated THERMAL POWER reduction in the COLR for Required Action B.2.1 did not result in a further power reduction (for example, if both Condition A and Condition B were entered at less than 100% RTP conditions), then the THERMAL POWER level established as a result of completing Required Action A.1, which is more restrictive limit than the THERMAL POWER level established by Required Action B.2.1, will establish the effective operating power level limit for the unit until both Conditions A and B are exited.

A.2

Reduction in the Overpower ΔT Trip setpoints (value of K_4) by ≥ 1% for each 1% that THERMAL POWER is limited below RTP by Required Action A.1 is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined

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BASES

ACTIONS (continued)

$F_Q^C(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints.

A.3

Verification that $F_Q^C(Z)$ has been restored to within its limit by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, assures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition A is modified by two Notes. Note 1 requires Required Action A.3 to be performed whenever the Condition is entered. Note 2 states that SR 3.2.1.2 is not required to be performed if this Condition is entered prior to THERMAL POWER exceeding 75% RTP after a refueling. This ensures that SR 3.2.1.1 and SR 3.2.1.2 (if required) will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.3. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

B.1.1

If it is found that the maximum calculated value of $F_Q(Z)$ which can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Implementing a more restrictive CAOC operating space, specified in the COLR, within the allowed Completion Time of 4 hours will restrict the AFD such that core peaking factor limits will not be exceeded during non-equilibrium normal operation. Several CAOC operating spaces, representing successively smaller AFD bands and, optionally, shallower Control Bank Insertion Limits, may be specified in the COLR. The corresponding $[W(Z)]^{COLR}$ functions for these operating spaces can be used to determine which CAOC operating space would result in acceptable non-equilibrium operation within the $F_Q^W(Z)$ limit, and thus, restore $F_Q^W(Z)$ to within limits.

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BASES

ACTIONS (continued)

Required Action B.1.1 is modified by a Note that states Required Action B.1.2 shall be completed whenever Required Action B.1.1 is performed and control rod motion is required to comply with the new operating space implemented by Required Action B.1.1. Required Action B.1.2 requires the performance of SR 3.2.1.1 and SR 3.2.1.2 within 72 hours. The Note ensures that the SRs will be performed even though Required Action B.1.1 restores $F_Q^W(Z)$ to within limits and Condition B is exited.

B.1.2

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. As discussed above, Required Action B.1.1 requires that a new CAOC operating space be implemented to restore $F_Q^W(Z)$ to within its limits. The performance of SR 3.2.1.1 and SR 3.2.1.2 is necessary to assure $F_Q(Z)$ is properly evaluated after any rod motion resulting from the implementation of a new CAOC operating space in accordance with Required Action B.1.1. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt implementation of the required CAOC operating space in accordance with Required Action B.1.1.

B.2.1

When $F_Q^W(Z)$ exceeds its limits, Required Action B.2.1 may be implemented instead of Required Action B.1.1. Required Action B.2.1 limits THERMAL POWER to less than RTP by the amount specified in the COLR. This maintains an acceptable absolute power density relative to the maximum power density value assumed in the safety analyses.

If the required $F_Q^W(Z)$ margin improvement exceeds the margin improvement available from the pre-analyzed THERMAL POWER reductions provided in the COLR, then THERMAL POWER must be further reduced to less than or equal to 50% RTP. In this case, reducing THERMAL POWER to less than or equal to 50% RTP will provide additional margin in the transient F_Q by the required change in THERMAL POWER and the increase in the F_Q limit. This will ensure that the F_Q limit is met during transient operation that may occur at or below 50% RTP.

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BASES

ACTIONS (continued)

The Completion Time of 4 hours provides an acceptable time to reduce power in an orderly manner to preclude entering an unacceptable condition during future nonequilibrium operation. The limit on THERMAL POWER initially determined by Required Action B.2.1 may be affected by subsequent determinations of $F_Q^W(Z)$ and would require power reductions within 4 hours of the $F_Q^W(Z)$ determination, if necessary to comply with the decreased THERMAL POWER limit. Decreases in $F_Q^W(Z)$ would allow increasing the THERMAL POWER limit and increasing THERMAL POWER up to this revised limit.

Required Action B.2.1 is modified by a Note that states Required Action B.2.3 shall be completed whenever Required Action B.2.1 is performed prior to increasing THERMAL POWER above the limit of Required Action B.2.1. Required Action B.2.3 requires the performance of SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit established by Required Action B.2.1. The Note ensures that the SRs will be performed even if Condition B may be exited prior to performing Required Action B.2.3. The performance of SR 3.2.1.1 and SR 3.2.1.2 is necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

If an F_Q, surveillance is performed at 100% RTP conditions, and both $F_Q^C(Z)$ and $F_Q^W(Z)$ exceed their limits, the option to reduce the THERMAL POWER limit in accordance with Required Action B.2.1 instead of implementing a new operating space in accordance with Required Action B.1.1, will result in a further power reduction after Required Action A.1 has been completed. However, this further power reduction would be permitted to occur over the next 4 hours. In the event the evaluated THERMAL POWER reduction in the COLR for proposed Required Action B.2.1 did not result in a further power reduction (for example, if both Condition A and Condition B were entered at less than 100% RTP conditions), then the THERMAL POWER level established as a result of completing Required Action A.1, which is more restrictive limit than the THERMAL POWER level established by Required Action B.2.1, will establish the effective operating power level limit for the unit until both Conditions A and B are exited.

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BASES

ACTIONS (continued)

B.2.2

Reduction in the Overpower ΔT trip setpoints value of K_4 by ≥ 1% for each 1% that THERMAL POWER is limited below RTP by Required Action B.2.1, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in the THERMAL POWER limit in accordance with Required Action B.2.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action B.2.2 may be affected by subsequent determinations of $F_Q^W(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^W(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^W(Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints.

B.2.3

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by Required Action B.2.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

C.1

If Required Actions A.1 through A.3 or B.1.1 through B.2.3 are not met within their associated Completion Times, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by reducing THERMAL POWER to < 25% RTP within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach < 25% RTP from full power operation in an orderly manner without challenging plant systems.

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SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note, which applies to the situation where the OPDMS is not monitoring parameters at the beginning of cycle startup, i.e., the Note applies during the first power ascension after a refueling. It states that performance of these SRs is not required if OPDMS was monitoring parameters upon exceeding 75% RTP. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, the SR 3.2.1.1 and SR 3.2.1.2 Frequency is applicable only for reload cores, and requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, SR 3.2.1.1 and SR 3.2.1.2 Frequency, together with the SR 3.2.1.3 and SR 3.2.1.4 first Frequency requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The SR 3.2.1.3 and SR 3.2.1.4 first Frequency is not intended to require verification of these parameters after every 10% increase in power level above the last verification. They only require verification after an equilibrium power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

The SR 3.2.1.3 Note and SR 3.2.1.4 Note 1 apply to the situation where the OPDMS is no longer monitoring parameters while the plant is in MODE 1. Without the continuous monitoring capability of the OPDMS, F_Q limits must be monitored on a periodic basis. The first measurement must be made within 24 hours of the OPDMS being declared not monitoring parameters. This is expected to be performed using the most recent contingency surveillance report created automatically during OPDMS monitoring. Subsequent surveillances are required consistent with the 31 EFPD Frequency.

<u>SR 3.2.1.1 and SR 3.2.1.3</u>

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing the measured values of $F_Q^C(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z) * F_Q^{MU}(Z)$. $F_Q^C(Z)$ is then compared to its specified limits.

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SURVEILLANCE REQUIREMENTS (continued)

The limit to which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP following a refueling outage assures that some determination of $F_Q^C(Z)$ is made prior to achieving a significant power level where the peak linear heat rate could approach the limits assumed in the safety analyses.

If THERMAL POWER has been increased by ≥ 10% RTP since the most recent determination of $F_Q^C(Z)$, another evaluation of this factor is required 24 hours after achieving equilibrium conditions at this higher power level (to assure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits). Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions required to perform the Surveillance.

The allowance of up to 24 hours after achieving equilibrium conditions at the increased THERMAL POWER level to complete the next $F_Q^C(Z)$ surveillance applies to situations where the $F_Q^C(Z)$ has already been measured at least once at a reduced THERMAL POWER level. The observed margin in the previous surveillance will provide assurance that increasing power up to the next plateau will not exceed the F_Q limit, and that the core is behaving as designed.

This Frequency condition is not intended to require verification of these parameters after every 10% increase in RTP above the THERMAL POWER at which the last verification was performed. It only requires verification after a power level is achieved for extended operation that is 10% higher than the THERMAL POWER at which $F_Q^C(Z)$ was last measured.

In the absence of these Frequency conditions (discussed above) it is possible to operate for 31 EFPD without verification of $F_Q^C(Z)$.

The Frequency of 31 effective full power days (EFPDs) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with Technical Specifications.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2 and SR 3.2.1.4

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called $[W(Z)]^{COLR}$. Multiplying the measured total peaking factor, $F_Q^C(Z)$, by $[W(Z)]^{COLR}$ gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit to which $F_Q^W(Z)$ is compared varies inversely with power.

The $[W(Z)]^{COLR}$ factors are provided in the COLR for discrete core elevations. $F_Q^W(Z)$ evaluations are not applicable for axial core regions near the top and bottom of the core. The excluded regions, usually the top and bottom 8%, are specified in the COLR and are defined to surveil the minimum margin location. A slightly smaller exclusion zone may be specified, if necessary, to include the limiting margin location in the surveilled region of the core.

These regions of the core are excluded from the evaluation because of the difficulty of making a precise measurement in these regions and because of the low probability that they would be more limiting than the safety analyses.

SR 3.2.1.2 requires a Surveillance of $F_Q^W(Z)$ during the initial startup following each refueling within 24 hours after exceeding 75% RTP and achieving equilibrium conditions. THERMAL POWER levels below 75% are typically non-limiting with respect to the limit for $F_Q^W(Z)$. Furthermore, startup physics testing and flux symmetry measurements, also performed at low power, provide confirmation that the core is operating as expected. This Frequency ensures that verification of $F_Q^W(Z)$ is performed prior to extended operation at power levels where peak LHR could be challenged and that the first required performance of SR 3.2.1.2 after a refueling is performed at a power level high enough to provide a high level of confidence in the accuracy of the Surveillance result.

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SURVEILLANCE REQUIREMENTS (continued)

Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions required to perform the Surveillance.

If a previous Surveillance of $F_Q^W(Z)$ was performed at part power conditions, SR 3.2.1.4 also requires that $F_Q^W(Z)$ be verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification within 24 hours after achieving equilibrium conditions. This ensures that $F_Q^W(Z)$ is within its limit using radial peaking factors measured at the higher power level.

The allowance of up to 24 hours after achieving equilibrium conditions will provide a more accurate measurement of $F_Q^W(Z)$ by allowing sufficient time to achieve equilibrium conditions and obtain the power distribution measurement.

In the absence of these Frequency conditions (discussed above) it is possible to operate for 31 EFPD without verification of $F_Q^W(Z)$. The Surveillance Frequency of 31 EFPDs is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with Technical Specifications, to preclude the occurrence of adverse peaking factors between 31 EFPD Surveillances. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

REFERENCES

1. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors,” 1974.
2. NUREG-0800, Standard Review Plan, Section 4.2, Revision 3, “Fuel System Design” (Appendix B).
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-7308-L-P-A, “Evaluation of Nuclear Hot Channel Factor Uncertainties,” June 1988 (Westinghouse Proprietary) and WCAP-7308-L-A (Non-Proprietary).
5. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification,” February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

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REFERENCES (continued)

6. WCAP-8385, “Power Distribution Control and Load Following Procedures - Topical Report,” September 1974.

7. WCAP-17661-P-A, “Improved RAOC and CAOC F_Q Surveillance Technical Specifications,” February 2019.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors assures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

With the On-line Power Distribution Monitoring System (OPDMS) monitoring parameters, $F^N_{\Delta H}$ is determined continuously by the OPDMS. When the OPDMS is not monitoring parameters, $F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 effective full power days (EFPDs). Also, during power operation with the OPDMS not monitoring parameters, the global power distribution is monitored by LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” and LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” which are directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio. Transient

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BASES

BACKGROUND (continued)

events that may be DNB limited are assumed to begin with a $F^N_{\Delta H}$ that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ prevent core power distributions from occurring which would exceed the following fuel design limits:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed applicable limits (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when the control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System (RCS) flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNB ratio (DNBR) to the 95/95 DNB criterion. This value provides a high degree of assurance that the hottest fuel rod in the core will not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this

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BASES

APPLICABLE SAFETY ANALYSES (continued)

variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which provide assurance that the initial conditions assumed in the safety and accident analyses remain valid. With the OPDMS monitoring parameters, peak linear heat rate and $F^N_{\Delta H}$ are directly monitored. Should the OPDMS cease monitoring parameters, the following LCOs assure that the conditions assumed for the safety analysis remain valid: LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” LCO 3.1.6, “Control Bank Insertion Limits,” LCO 3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$),” and LCO 3.2.1, “Heat Flux Hot Channel Factor ($F_Q(Z)$).”

When the OPDMS is not available to measure power distribution parameters continuously, $F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the incore detector system. Measurements are generally taken with the core at, or near, steady-state conditions. Without the OPDMS, core monitoring and control under transient conditions (Condition I events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

These limits are developed using the methodologies identified in Technical Specification 5.6.3.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

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BASES

LCO (continued)

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

When the OPDMS is not monitoring parameters and core power distribution parameters cannot be continuously monitored, it is necessary to monitor $F^N_{\Delta H}(Z)$ on a periodic basis. Furthermore, $F^N_{\Delta H}$ limits must be maintained in MODE 1 with THERMAL POWER ≥ 25% RTP to preclude core power distributions from exceeding the fuel design limits for DNBR and peak cladding temperature (PCT). Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power-dependent limit.

When the $F^N_{\Delta H}$ limit is exceeded, it is not likely that the DNBR limit would be violated in steady state operation, since events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain outside $F^N_{\Delta H}$ limits for an extended period of time.

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BASES

ACTIONS (continued)

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 would nevertheless require another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power were reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 would be performed if power ascension were delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Overpower ΔT trip setpoints to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those specified in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Time of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may cause an inadvertent reactor trip.

A.2

Once Condition A is entered, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or

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BASES

ACTIONS (continued)

Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence assures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation will proceed within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% of RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note, that states that THERMAL POWER does not have to be reduced prior to performing this action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by reducing THERMAL POWER to < 25% RTP within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach < 25% RTP from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 and SR 3.2.2.2

When the OPDMS is monitoring parameters, the value of $F^N_{\Delta H}$ is directly and continuously monitored. With the OPDMS not monitoring parameters, the value of $F^N_{\Delta H}$ is determined by using the incore detector system data. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the data. The measured value of $F^N_{\Delta H}$ must be multiplied by a measurement uncertainty factor before making comparisons to the $F^N_{\Delta H}$ limit.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

After each refueling, with the OPDMS not monitoring parameters, $F^N_{\Delta H}$ must be determined prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

SR 3.2.2.1 is modified by a Note, which applies to the situation where the OPDMS is not monitoring parameters at the beginning of cycle startup, i.e., applies during the first power ascension after a refueling. It states that performance is not required if OPDMS was monitoring parameters upon exceeding 75% RTP.

With the OPDMS not monitoring parameters, the 31 EFPDs Frequency is acceptable because the power distribution will change relatively slowly over this amount of fuel burnup. This Frequency is short enough so that the $F^N_{\Delta H}$ limit will not be exceeded for any significant period of operation.

SR 3.2.2.2 is modified by a Note, which applies to the situation where the OPDMS is no longer monitoring parameters. Without the continuous monitoring capability of the OPDMS, $F^N_{\Delta H}$ limits must be monitored on a periodic basis. The first measurement must be made within 24 hours of the OPDMS being declared not monitoring parameters. This is expected to be performed using the most recent contingency surveillance report created automatically during OPDMS monitoring. Subsequent surveillances are required consistent with the 31 EFPD Frequency.

REFERENCES

1. NUREG-0800, Standard Review Plan, Section 4.2, Revision 3, “Fuel System Design” (Appendix B).
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.”

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core when the On-Line Power Distribution Monitoring System (OPDMS) is not monitoring parameters. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing which is a significant factor in axial power distribution control.

The operating strategy used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band about a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined with THERMAL POWER at or near the RTP value, equilibrium xenon conditions, and the control banks positioned at or near a nominally assumed position for steady-state operation at high power levels. The value of the target flux difference obtained at these conditions divided by the fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

The AFD is monitored on an automatic basis using the computer which has an AFD monitor alarm. The frequency of monitoring the AFD by the computer is frequent enough (typically once per minute) to provide an essentially continuous accumulation of penalty deviation time that allows the operator to assess the status of the penalty deviation time. The computer determines a time average (typically over 1 minute) of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside the target band and the THERMAL POWER is > 90% RTP. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

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BASES

BACKGROUND (continued)

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup. The Nuclear Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$ and QPTR LCOs limit the radial component of the peaking factors.

It is expected that AFD will normally be automatically controlled by the Plant Control System (PLS), which will demand movement of the AO bank to maintain AFD within a control dead-band around the target flux difference. The control dead-band is defined to be more restrictive than the CAOC limits associated with this LCO such that automatic control can typically be expected to maintain adherence to this LCO.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology (Refs. 1, 2, and 3) entails:

a. Establishing an envelope of allowed power shapes and power densities,

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes,

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics, and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

With the OPDMS not monitoring parameters, the limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition II, III, or IV events. This ensures that the fuel cladding integrity is

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BASES

APPLICABLE SAFETY ANALYSES (continued)

maintained for these postulated accidents. The most important Condition IV event is the LOCA. The most important Condition III event is the loss of flow accident. The most important Condition II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition II accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

These limits are developed using the methodologies identified in Technical Specification 5.6.3.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through either the manual or automatic motion of control banks, particularly motion of the AO bank. Using the MSHIM operation and control strategy, automatic motion of the AO bank is in response to deviations of AFD from the target flux difference resulting from changes in power level, xenon distribution, or positioning of the MSHIM control banks. Automatic motion of the MSHIM control banks is in response to temperature deviations resulting from changes in power level, fuel and burnable absorber depletion, xenon concentration, or RCS boron concentration. The impact of MSHIM control bank motion on AFD is compensated by movements of the AO bank.

Signals are available to the operator from the Protection and Safety Monitoring System (PMS) excore neutron detectors (Ref. 4). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup. With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.

Violating this LCO on the AFD, with the OPDMS not monitoring parameters, could produce unacceptable consequences if a Condition II, III or IV event occurs while the AFD is outside its specified limits.

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BASES

LCO (continued)

The LCO is modified by four Notes. Note 1 states the conditions necessary for declaring the AFD outside of the target band. Notes 2 and 3 describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part b of this LCO), a 1 hour cumulative penalty deviation time, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside the target band, but within the acceptable operation limits provided in the COLR (Note 2). This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part b of this LCO (i.e., THERMAL POWER ≥ 50% RTP). The cumulative penalty time is the sum of penalty times from Parts b and c of this LCO.

For THERMAL POWER levels >15% RTP and < 50% RTP (i.e., Part c of this LCO), deviations of the AFD outside of the target are less significant. Note 3 allows the accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band and reflects this reduced significance. With THERMAL POWER ≤ 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and therefore requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulation penalty deviation time.

For surveillance of the Power Range Neutron Flux channels performed according to SR 3.3.1.4, Note 4 allows deviation outside the target band for 16 hours and no penalty deviation time accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 effective full power days (EFPD).

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BASES

APPLICABILITY

The AFD requirements are applicable in MODE 1 above 15% RTP and OPDMS not monitoring parameters. Above 50% RTP the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis (Ref. 1).

Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

Low signal levels in the excore channels may preclude obtaining valid AFD signals at or below 15% RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

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BASES

ACTIONS (continued)

C.1

With THERMAL POWER < 90% RTP, but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour during the preceding 24 hours if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

Condition C is modified by a Note that requires that Required Action C.1 must be completed whenever this Condition is entered.

D.1

If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.

Reducing the power level to ≤ 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

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BASES

SURVEILLANCE REQUIREMENTS

The surveillances are modified by a Note allowing 7 days without the continuous monitoring capability of the OPDMS before AFD must be initially verified. The first performance must be made within 7 days of the most recent date where the OPDMS data has verified parameters. This is consistent with the 7 day Surveillance Frequency for AFD.

SR 3.2.3.1

This surveillance verifies that the AFD, as indicated by the PMS excore channel, is within the target band. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.2

This Surveillance requires that the target flux difference is updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup by performing SR 3.2.3.3.

SR 3.2.3.3

Measurement of the target flux difference is determined from an incore power distribution measurement at, or very near equilibrium xenon conditions, preferably at high power levels with the control banks at a reference position. This power distribution measurement provides the reference axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady-state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference values that account for changes in incore excore calibrations that may have occurred in the interim.

A Note modifies this SR to allow the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

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BASES

REFERENCES

1. WCAP-8385, “Power Distribution Control and Load Following Procedures,” Westinghouse Electric Corporation, September 1974 (Westinghouse Proprietary) and WCAP-8403 (Non-Proprietary).

2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), Attachment: “Operation and Safety Analysis Aspects of an Improved Load Follow Package,” January 31, 1980.

3 C. Eicheldinger to D. B. Vassallo (Chief Light Water Reactors Branch, NRC), Letter NS-CE-687, July 16, 1975.

4. FSAR Chapter 15, “Accident Analysis.”

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

With the On-Line Power and Distribution Monitoring System (OPDMS) not monitoring parameters, the QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. With OPDMS monitoring parameters, the key power distribution parameters associated with fuel design criteria and bounds of the safety analyses are continuously and directly monitored. With the OPDMS not monitoring parameters, LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” and LCO 3.1.6, “Control Bank Insertion Limits,” provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions from occurring which would violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed applicable limits (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions from occurring which would exceed the safety analyses limits.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Should the OPDMS cease monitoring parameters, the QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, with the OPDMS not monitoring parameters, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, where corrective action is required, provides a margin of protection for both the DNB ratio (DNBR) and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP and OPDMS not monitoring parameters to preclude core power distributions from exceeding the design limits. Applicability with OPDMS monitoring parameters is not required since OPDMS directly and continuously monitors key power distribution parameters.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, and the OPDMS not monitoring parameters, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

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BASES

ACTIONS (continued)

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level and increasing power up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR alarm may be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are of primary importance in assuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limits, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^N$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such

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ACTIONS (continued)

reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors which best characterize the core power distribution. This re-evaluation is required to assure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly,

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BASES

ACTIONS (continued)

then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limit (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve the status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note allowing 12 hours without the continuous monitoring capability of the OPDMS before QPTR must be initially verified. The first verification must be made within 12 hours of the most recent date where the OPDMS data has verified parameters. This is consistent with the 12 hour Surveillance Frequency for QPTR in SR 3.2.4.2.

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three Power Range Neutron Flux channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR as indicated by the Protection and Safety Monitoring System (PMS) excore channels is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

For those causes of QPTR that occur quickly (a dropped rod), there are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With a PMS Power Range Neutron Flux channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts would likely be detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for assuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one Power Range Neutron Flux channel is inoperable, the incore detectors are used to confirm that the normalized symmetric power distribution is acceptable.

With the OPDMS not monitoring parameters and one PMS channel inoperable, the surveillance of the incore power distribution on a 12 hour basis is sufficient to maintain peaking factors within their normal limits, especially, considering the other LCOs and ACTIONS required when the OPDMS is out of service.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.”
2. NUREG-0800, Standard Review Plan, Section 4.2, Revision 3, “Fuel System Design” (Appendix B).
3. Title 10, Code of Federal Regulations, Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” GDC 26, “Reactivity Control System Redundancy and Capability.”

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters

BASES

BACKGROUND

The function of the On-line Power Distribution Monitoring System (OPDMS) is the display of core monitoring information to the control room. Its purpose is to support the continuous surveillance of the parameters of LCO 3.2.5 and provide information about core power distribution to the reactor operator. It consists of hardware and software for data transfer, core monitoring instrumentation hardware and software, continuous control room displays (in MODES 1, 2, and 3), operator accessible detailed displays through control room workstations and alarms associated with the monitoring of the parameter limits and the system functionality.

The purpose of the limits on the OPDMS-monitored power distribution parameters is to provide assurance of fuel integrity during Conditions I (Normal Operation) and II (incidents of Moderate Frequency) events by: (1) not exceeding the minimum departure from boiling ratio (DNBR) in the core, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear heat rate during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the peak cladding temperature (PCT) limit of 2200°F is not exceeded.

The purpose of the limits on SHUTDOWN MARGIN (SDM) are discussed in the Bases of LCO 3.1.1, “SDM.”

The definition of certain quantities used in these specifications are as follows:

Peak linear heat rate	Peak linear heat rate as measured in kw/ft.
$F^N_{\Delta H}$	Ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.
Minimum DNBR	Minimum ratio of the critical heat flux to actual heat flux at any point in the reactor that is allowed in order to assure that certain performance and safety criteria requirements are met over the range of plant conditions.

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BASES

BACKGROUND (continued)

By continuously monitoring the core and following its actual operation, it is possible to significantly limit the adverse nature of power distribution initial conditions for transients which may occur at any time.

For simplicity in the continuous monitoring of the compliance to limits of LCO 3.2.5 parameters a, b and c, the margin of the parameters to the limits are converted to the parameter power margin. Power margin is the percent THERMAL POWER increase that would result in the parameter limit being exceeded. Minimum power margin is the smallest margin to limit of the LCO 3.2.5 a, b and c parameters. Thus, compliance with limits of all three parameters is demonstrated by the minimum power margin (MPM) being greater than zero. The units of percent power facilitate management of the margins to thermal limits. Operators presented with a challenge to thermal limits would know quantitatively what power reduction is necessary to restore desired margins.

For simplicity in the monitoring of the compliance to the limit of LCO 3.2.5 parameter d, the margin to the limit is expressed as Excess Shutdown Margin (ESDM) and displayed in units of RCS boron concentration. ESDM is the excess amount over the SHUTDOWN MARGIN limit that the reactor would be shutdown if a trip were to occur, expressed in units of boron concentration. Thus compliance with the requirement of LCO 3.2.5 for SHUTDOWN MARGIN is demonstrated by ESDM greater than zero. The units of RCS boron concentration facilitate management of the margin to SDM limit. Operators presented with a challenge to SDM limits would know quantitatively what change in RCS boron concentration would restore desired margin.

The core monitoring system is the component of the OPDMS that calculates from present plant data the core power distribution and reactivity and determines margin to core thermal limits and required SHUTDOWN MARGIN. The core monitoring system is the Westinghouse BEACON Core Monitoring System. The technology utilized is licensed by the NRC for direct monitoring of core parameters to limits (Ref. 3). The core monitoring system requires present plant data of reactor THERMAL POWER, control rod position and inlet temperature to provide monitoring of SHUTDOWN MARGIN (SDM) in MODES 1 and 2 with $k_{eff} \geq 1.0$. Valid data from the Incore Instrumentation System neutron detectors is required in addition to that required for SDM to monitor reactor core margins to thermal limits in MODE 1 with THERMAL POWER ≥ 25%.

The ability of the OPDMS to monitor the required parameters is dependent on the functionality of the BEACON system, Plant Control System (PLS) and Data Display System (DDS). The requirements of OPDMS capabilities fall into the following categories:

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BASES

BACKGROUND (continued)

- A cycle specific nuclear model is installed in the core monitoring system and is updated to reflect the present characteristics of the core
- The core monitoring software is providing indication that the processes are running appropriately
- Present data of good quality and acceptable accuracy from an adequate number of critical sensors is available to the core monitoring system
- The PLS and DDS are functioning to transmit data to and from the core monitoring system and to correctly display the data and actuate alarms in the main control room

The minimum requirements for data input to the core monitoring system are in Table B 3.2.5-1. The core monitoring system continuously checks for the presence and quality of the minimum required parameters. If the input data falls below the minimum required, an alarm will actuate for loss of functionality.

APPLICABLE SAFETY ANALYSES

The limits on LCO 3.2.5 parameters a, b, and c preclude core power distributions from occurring which would violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the PCT must not exceed a limit of 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed applicable limits (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

Limits on linear power density or peak linear heat rate assure that the peak linear heat rate assumed as a base condition in the LOCA analyses is not exceeded during normal operation.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Limits on $F_{\Delta H}$ ensure that the LOCA analysis assumptions and assumptions made with respect to the Overtemperature ΔT Setpoint are maintained.

The limit on DNBR ensures that if transients analyzed in the safety analyses initiate from the conditions within the limit allowed by the OPDMS, the DNB criteria will be met.

The limit on SDM ensures that the safety analysis assumptions related to reactor trip effectiveness and the consequences of design basis actions initiating from shutdown reactor conditions remain valid.

These limits are developed using the methodologies identified in Technical Specification 5.6.3.

The OPDMS-monitored parameters of this LCO satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within these limits. If the LCO limits cannot be maintained within limits, reduction of the core power is required.

Violating the OPDMS-monitored power distribution parameter limits could result in unanalyzed conditions should a design basis event occur while the parameters are outside their specified limits.

Peak linear heat rate limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. The highest calculated linear power densities in the core at specific core elevations are displayed for operator visual verification relative to the COLR values.

The determination of $F^N_{\Delta H}$ identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB. Should $F^N_{\Delta H}$ exceed the limit given in the COLR, the possibility exists for DNBR to exceed the value used as a base condition for the safety analysis.

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BASES

LCO (continued)

Two levels of alarms on power distribution parameters and shutdown margin are provided to the operator. One level of alarm serves as a warning before any of the the three parameters (peak linear heat rate, $F^N_{\Delta H}$, DNBR) or SDM exceed their limits in the COLR. The other alarm indicates when the parameters have reached their limits.

APPLICABILITY

The OPDMS-monitored power distribution limits with OPDMS monitoring parameters a, b, and c must be maintained in MODE 1 with THERMAL POWER ≥ 25% RTP to preclude core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES, and MODE 1 with THERMAL POWER < 25% RTP, is not required because there is either insufficient stored energy in the fuel or insufficient energy transferred to the reactor coolant to require a limit on the distribution of core power. The OPDMS monitoring of SDM is applicable in MODES 1 and 2 with $k_{eff} \geq 1.0$.

Specifically for $F^N_{\Delta H}$ and DNBR, the design bases accidents (DBAs) that are sensitive to $F^N_{\Delta H}$ and DNBR in other MODES (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ and DNBR in these modes.

In addition to the alarms discussed in the LCO section above (alarms on OPDMS-monitored power distribution parameters), there is an alarm indicating the potential for the OPDMS to be not monitoring parameters.

Should the OPDMS be determined to be not monitoring parameters for other than reasons of alarms inoperable, this LCO is no longer applicable and LCOs 3.2.1 through 3.2.4 become applicable.

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BASES

ACTIONS

A.1

With any of the OPDMS-monitored power distribution parameters outside of their limits, the assumptions used as most limiting base conditions for the DBA analyses may no longer be valid. The 1 hour operator ACTION requirement to restore the parameter to within limits is consistent with the basis for the anticipated operational occurrences and provides time to assess if there are instrumentation problems. It also allows the possibility to restore the parameter to within limits by control rod motion if this is possible. The OPDMS will continuously monitor these parameters and provide an indication when they are approaching their limits.

B.1

If the OPDMS-monitored power distribution parameters cannot be restored to within their limits within the Completion Time of ACTION A.1, it is likely that the problem is not due to a failure of instrumentation. Most of these parameters can be brought within their respective limits by reducing THERMAL POWER because this will reduce the absolute power density at any location in the core thus providing margin to the limit.

If the parameters cannot be returned to within limits as power is being reduced, THERMAL POWER must be reduced to < 25% RTP where the LCOs are no longer applicable.

The Completion Time of 4 hours provides an acceptable time to reduce power in an orderly manner and without allowing the reactor to exceed the limits for an extended period of time.

C.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a concentrated solution. The operator should begin boration with the best source available for the plant conditions.

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BASES

SURVEILLANCE REQUIREMENTS

With OPDMS monitoring parameters, the power distribution parameters and SDM are continuously computed and compared against their limits, and margins to limits displayed. Two levels of alarms are provided to the operator. The first alarm provides a warning before these parameters (peak linear heat rate, $F^N_{\Delta H}$, DNBR), or SDM exceed their limits. The second alarm indicates when they actually reach their limits. A third alarm indicates trouble with the OPDMS system.

SR 3.2.5.1

This Surveillance requires the operator to verify that the power distribution parameters are within their limits. This confirmation is a verification in addition to the automated checking performed by the OPDMS system. A 24 hour Surveillance interval provides assurance that the system is functioning properly and that the core limits are met.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors,” 1974.

2. NUREG-0800, Standard Review Plan, Section 4.2, Revision 3, “Fuel System Design” (Appendix B).

3. WCAP-12472-P-A, (Westinghouse Proprietary) and WCAP-12473-A (Non-Proprietary), “BEACON Core Monitoring and Operations Support System,” August 1994, Addendum 1, May 1996, and Addendum 2, March 2001 and WCAP-12472-P-A (Westinghouse Proprietary) and WCAP-12472-NP-A (Non-Proprietary) Addendum 4, September 2012.

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Table B 3.2.5-1 (page 1 of 1)
OPDMS Minimum Required Input Parameters

Data or Sensor	Minimum Data Required
IIS Self Powered Neutron Detector segments[a]	75% at cycle initial power distribution measurement[b]
	40% following initial cycle power distribution measurement
	15 Detector segments/core quadrant[c]
	6 Detector segments/quadrant above core mid-plane
	6 Detector segments/quadrant below core mid-plane
Reactor Power	One indication [d]
RCS Cold Leg Temperature	Two (2) Channels
Control Rod Bank Positions	One (1) indication per bank[e]

(a) Required at THERMAL POWER ≥ 25%

(b) The requirement of 75% detector segments for initial power distribution measurement is not continuously monitored and is administratively implemented

(c) A detector segment is the difference in signal from two OPERABLE detectors in the same assembly or a single detector signal for the shortest OPERABLE detector in an assembly

(d) Either calorimetric or average of available DT channels

(e) Either a bank group demand or Digital Rod Position Indication

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B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based upon the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Feature Actuation System (ESFAS) in mitigating accidents.

The Protection and Safety Monitoring System (PMS) has been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as “Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded.” The Safety Analysis Limit (SAL) is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to assure that a SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Safety Analysis Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur. The LSSS values are identified and maintained in the Setpoint Program (SP) and are controlled by 10 CFR 50.59.

The Nominal Trip Setpoint (NTS) specified in the SP is a predetermined field setting for a protection channel chosen to initiate automatic actuation prior to the process variable reaching the Safety Analysis Limit and, thus, assures that the SL is not exceeded. As such, the NTS accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTS assures that the SLs are not exceeded. Therefore, the NTS meets the 10 CFR 50.36 definition of an LSSS.

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as “...being capable of performing its

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BASES

BACKGROUND (continued)

safety functions(s)." Relying solely on the NTS to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTS due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTS, and thus, the automatic protective action would still have assured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function. If the as-found condition of the channel is near the as-found tolerance, recalibration is considered appropriate to allow for drift during the next surveillance interval.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite doses are within the acceptance criteria during AOOs.

Design Basis Accidents (DBA) are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of the limits. Different accident categories are allowed a different fraction of these limits, based on the probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS maintains surveillance on key process variables which are directly related to equipment mechanical limitations, such as pressure, and on variables which directly affect the heat transfer capability of the reactor, such as flow and temperature. Some limits, such as

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BASES

BACKGROUND (continued)

Overtemperature ΔT, are calculated in the Protection and Safety Monitoring System cabinets from other parameters when direct measurement of the variable is not possible.

Protection and Safety Monitoring System (PMS)

The Protection and Safety Monitoring System cabinets contain the necessary equipment to (Ref. 9):

- Permit acquisition and analysis of the sensor inputs, including plant process sensors and nuclear instrumentation, required for reactor trip and Engineered Safety Features (ESF) calculations;
- Perform computation or logic operations on variables based on these inputs;
- Provide trip signals to the reactor trip switchgear and ESF actuation data to the ESF coincidence logic as required;
- Permit manual trip or bypass of each individual reactor trip Function and permit manual actuation or bypass of each individual automatic ESF Function;
- Provide data to other systems in the Instrumentation and Control (I&C) architecture;
- Provide additional redundancy (within a single division) for the reactor trips and ESF actuations; and
- Provide isolation circuitry for control Functions that require input from sensors that are also required for protection Functions.

PMS Overview

The four divisions of PMS each consist of the following:

- Field Transmitters and Sensors (non-redundant)
- Nuclear Instrumentation System (NIS) - 1 per division (non-redundant)
- Bistable Processor Logic (BPL) System- 2 BPL subsystems per division (redundant)

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BASES

BACKGROUND (continued)

- Local Coincidence Logic (LCL) System - 2 LCL subsystems per division (redundant), each subsystem with two reactor trip processors (non-redundant)

- Reactor Trip Initiation Logic - 2 Reactor Trip Matrices (RTMs) per division (undervoltage (UV) and shunt trip (ST) - redundant)

- Reactor Trip Breakers (RTBs) - 2 per division (non-redundant). See LCO 3.3.7, “RTS Trip Actuation Devices” for further discussion.

The vital buses that power each PMS division are backed up by the battery bank in the same division. See LCO 3.8.1, “DC Sources - Operating” for further discussion.

PMS equipment duplication within a division (redundancy) is generally provided for active equipment such as processors and communication hardware. Where provided, redundant sets of equipment within a division are not independent and can be subject to common cause failures. This redundancy is provided to increase plant availability and facilitate surveillance testing. A division or channel is OPERABLE if it is capable of performing its specified safety function(s) and all the required supporting functions or systems are also capable of performing their related support functions.

Within each division, the sensor signals and setpoint bistables are processed in the BPL. The results of the bistable logic (partial reactor trip signals) are communicated to the LCL subsystems in all divisions.

The LCL subsystems perform the divisional voting logic (e.g., two-out-of-four (2oo4) logic) and other associated logic.

Within each division, the LCL reactor trip contact outputs are inputs to two RTMs: one for the RTB undervoltage (UV) mechanism (coil), and one for the shunt trip (ST) mechanism (coil). The RTMs trip the RTBs by deenergizing the UV coils and energizing the ST coils. Each of the four PMS divisions consists of two reactor trip circuit breakers. The reactor is tripped when the RTBs in two or more divisions are opened.

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BACKGROUND (continued)

Reactor Trip (RT) Channel

An RT channel extends from the sensor to the output of the associated BPL subsystem and shall include the sensor (or sensors), the signal conditioning, the BPL subsystem, and associated datalinks to the LCL subsystems. For RT channels containing nuclear instrumentation, the RT channel also includes the signal processing and power supplies for the detectors provided by the NIS. In some cases, sensors provide inputs to more than one RT function. Thus an inoperability of a shared sensor can affect the OPERABILITY of more than one RT function.

Automatic Trip Logic

The Automatic Trip Logic includes the LCL subsystems and Reactor Trip Initiation logic matrices, including the outputs of the various RT Channels (BPL outputs). It does not include the RTBs.

PMS Division

The PMS consists of four redundant divisions, designated A, B, C, and D. All safety system divisions are physically, functionally, and electrically separated from each other and from non-safety systems. Redundant divisions are provided to satisfy single failure criteria and improve plant availability. Each PMS division actuates two RTBs. Actuation of the RTBs in two divisions will trip the reactor.

Field Transmitters and Sensors

Normally, four redundant measurements using four separate sensors are made for each variable used for reactor trip. One measurement is processed by each division. The use of four channels for protection functions is based on a minimum of two channels being required for a reactor trip (e.g., two-out-of-four (2oo4) voting logic), with one channel in test or bypass, and a single failure on the remaining channel. Minimum requirements for protection are achieved with only three channels OPERABLE. The fourth channel is provided to increase plant availability, and permits the plant to run for an indefinite time with a single channel out of service. The circuit design is able to withstand both an input failure to the control system, which may then require the protection Function actuation, and a single failure in the other channels providing the protection Function actuation.

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BACKGROUND (continued)

Analog measurements are converted to digital form by analog to digital (A/D) converters within the BPL subsystems. Signal conditioning may be applied to selected inputs following the conversion to digital form. Except for selected dual element resistance temperature detectors (RTDs), the sensors within each division are non-redundant.

Nuclear Instrument System (NIS)

In each division the NIS cabinet provides signal processing and power supplies for the source range, intermediate range, and power range detectors. The analog and digital pulse signals generated are sent to the BPL for processing. Within each division, each transmitter, sensor, and NIS signal is input in parallel to both BPLs.

Bistable Processor Logic (BPL) System

Each PMS division contains two identical and redundant BPL subsystems receiving the same input signals and performing the same calculations and logic. The BPL subsystems receive data from field sensors and manual inputs (such as system-level blocks and resets) from the MCR to perform the protective function calculations.

Within each division, the input from each sensor is provided in parallel to each of the two BPL subsystems' analog or digital input modules. Analog transmitter and sensor measurements are converted from analog to digital form by analog to digital input modules within the BPL subsystems. Signal conditioning may be applied to selected inputs following the conversion to digital form. The subsystems provide signal outputs for MCR indications and, via the Plant Control System (PLS), for non-safety Remote Shutdown Workstation indications.

Following necessary calculations, processing, and logic, the measurements are compared to the applicable setpoints. The basis of the setpoints is described in References 2 and 3. The results of the required calculations and setpoint comparisons determine the partial trip status for each reactor trip function.

The partial reactor trip signals are transmitted to all eight LCLs in the PMS for use in coincidence logic voting. The PMS uses datalinks to communicate the partial trips and related status information calculated in the BPL subsystems to the LCL subsystems. These datalinks are used both locally within a division and externally across divisions. A partial trip signal from either BPL subsystem in a division is interpreted by the LCL coincidence logic as a partial trip signal.

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BACKGROUND (continued)

During periodic tests and maintenance performed using the Maintenance and Test Panel and Interface and Test Processor, each BPL subsystem is tested individually. During testing, the LCL processors use the redundant BPL subsystem output for trip status. This permits the division being tested to remain OPERABLE while the testing is being conducted. In combination with manual tests required by Surveillance Requirements, the BPLs are tested via continuous system self-checking features.

Local Coincidence Logic (LCL) System

Each PMS division contains two identical and redundant LCL subsystems receiving the same input signals and performing the same logic. The LCL subsystems perform the logic to combine the partial trip signals from the BPL subsystems and generate a trip output signal to the Reactor Trip Initiation logic.

Eight LCL subsystems are provided in the PMS architecture. Each of the LCL subsystems receives partial reactor trip signals and status signals from each of the eight BPL subsystems. In normal operation, the LCL logic is programmed to assume that a partial trip signal in either BPL subsystem in a given division is equivalent to a partial trip signal.

In the event of a single BPL processor or associated datalink failure in a division is detected by diagnostics, the LCL logic will reject the input from the failed component and use the input from the other BPL subsystem from the affected division as the source of the trip information. This allows the function logic to remain in a 2oo4 logic configuration.

LCL processors will only permit bypass of both BPL inputs of a given function from one division (such as due to a failed sensor). In the event of a division bypass, the LCL reverts to bypass logic in the affected function (e.g., 2oo3 for 2oo4 logic).

The LCL subsystems act to initiate a reactor trip when the required number of divisions reaches a partial trip state (e.g., 2oo4, 1oo2).

Each LCL subsystem provides four contact outputs to the Reactor Trip Initiation logic, two for the Undervoltage (UV) RTM and two for the Shunt Trip (ST) RTM.

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BACKGROUND (continued)

The LCL also provides for the bypass of trip functions to accommodate periodic tests and maintenance. During LCL subsystem testing, each LCL subsystem is tested individually. During testing, the redundant LCL subsystem is available to provide the UV and ST output signals to the RTMs. This permits the division being tested to remain OPERABLE while the testing is being conducted. In combination with manual tests required by Surveillance Requirements, the LCLs are tested via continuous system self-checking features.

Reactor Trip Initiation Logic

The Reactor Trip Matrix (RTM) acts as an interface between the LCL subsystems and the RTBs. The RTM receives contact inputs from the LCL subsystems and performs the logic to determine if a division will issue a reactor trip command.

Each PMS division contains two redundant RTMs; one is configured as a ST matrix and the other a UV matrix. The combination of the two forms the complete RTM for a given division. If the ST logic is satisfied, the RTB ST coils are energized, opening both RTBs in the division. If the UV logic is satisfied, the RTB UV coils are de-energized, opening both RTBs in the division.

The PMS boundary ends at the interposing relay contacts of the RTMs.

Manual RT

A manual reactor trip is initiated from the MCR by redundant momentary switches. The switches directly control the power from the RTM logic, actuating the UV and ST attachments in all four divisions.

Nominal Trip Setpoint (NTS)

The NTS is the nominal value at which the trip output is set. Any trip output is considered to be properly adjusted when the “as-left” value is within the band for CHANNEL CALIBRATION (i.e., ± rack calibration accuracy).

The trip setpoints used in the trip output are based on the Safety Analysis Limits stated in Reference 2. The determination of these NTSs is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49

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(Ref. 5), the NTSs specified in the SP are conservative with respect to the Safety Analysis Limits. A detailed description of the methodology used to calculate the NTSs, including their explicit uncertainties, is provided in the "Westinghouse Setpoint Methodology for Protection Systems" (Ref. 3). The as-left tolerance and as-found tolerance band methodology is provided in the SP.

The NTSs listed in the SP are based on the methodology described in Reference 3, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each NTS. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes. Transmitter and signal processing equipment calibration tolerances and drift allowances must be specified in plant calibration procedures, and must be consistent with the values used in the setpoint methodology.

The OPERABILITY of each transmitter or sensor can be evaluated when its "as-found" calibration data are compared against the "as-left" data and are shown to be within the setpoint methodology assumptions. The basis of the setpoints is described in References 2 and 3. Trending of calibration results is required by the program description in Technical Specifications 5.5.14.d.

Note that the as-left and as-found tolerances listed in the SP define the OPERABILITY limits for a channel during a periodic CHANNEL CALIBRATION. Trip setpoints are continuously and automatically verified by PMS self-checking features between performances of CHANNEL CALIBRATIONS. Before unit startup, the CHANNEL CALIBRATION verifies that the trip setpoint values in the Maintenance and Test Panel (MTP) match the SP specified values.

The Protection and Safety Monitoring System testing features are designed to allow for complete testing by using a combination of system self-checking and manual tests. Successful testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded. For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing. To the extent possible, Protection and Safety Monitoring System testing will be accomplished with continuous system self-checking features in lieu of manual surveillance tests. As a result, some functions do not have manual surveillance requirements.

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BACKGROUND (continued)

The Protection and Safety Monitoring System incorporates continuous system self-checking features wherever practical. Self-checking features include on-line diagnostics for the computer system and the hardware and communications tests. Faults detected by the self-checking features are alarmed in the main control room. These self-checking tests do not interfere with normal system operation.

Manual tests are included for those parts of the system which are not tested with self-checking features. This includes manual functional checks, calibration verification, response time testing, and component testing. The test features again include a combination of continuous testing features and manual testing features.

All of the tests are designed so that the duration of the testing is as short as possible. The manual tests are designed so that the actual logic is not modified. To prevent unwanted actuation, the tests are designed with either the capability to bypass a Function during testing and/or limit the number of signals allowed to be placed in test at one time.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain compliance with the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

Each of the analyzed accidents and transients which require reactor trip can be detected by one of more RTS Functions. The accident analysis described in Reference 2 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These RTS trip Functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate function performance. These RTS trip Functions may also serve as backups to RTS trip Functions that were credited in the accident analysis.

Permissive and interlock functions are based upon the associated protection function instrumentation. Because they do not have to operate in adverse environmental conditions, the trip settings of the permissive and interlock functions use the normal environment, steady-state instrument uncertainties of the associated protection function instrumentation. This results in OPERABILITY criteria (i.e., as-found tolerance and as-left tolerance) that are the same as the associated protection function sensor and process rack modules. The NTSs for permissives and interlocks are based on the associated protection

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

function OPERABILITY requirements; i.e., permissives and interlocks performing enabling functions must be set to occur prior to the specified trip setting of the associated protection function.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. The as-left and as-found tolerances specified in the SP define the OPERABILITY limits for a channel during a CHANNEL CALIBRATION or COT. As such, the as-left and as-found tolerances differ from the NTS by plus or minus the PMS rack calibration accuracy and envelope the expected calibration accuracy and drift. In this manner, the actual setting of the channel (NTS) prevents exceeding an SL at any given point in time as long as the channel has not drifted beyond the expected tolerances during the surveillance interval. Note that the as-left and as-found recorded values must be confirmed to be within the assumptions of the statistical uncertainty calculations.

If the actual setting of the channel is found outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

A trip setpoint may be set more conservative than the NTS as necessary in response to plant conditions. However, in this case, the OPERABILITY of this instrument must be verified based on the actual field setting and not the NTS. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current plant status. They back up operator actions to ensure protection system Functions are not blocked during plant conditions under which the safety analysis assumes the Functions are OPERABLE. Therefore, the interlocks do not need to be OPERABLE when the associated reactor trip Functions are outside the applicable MODES. Proper operation of these interlocks supports

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

OPERABILITY of the associated TS Functions and/or the requirement for actuation logic OPERABILITY. Interlocks must be in the required state, as appropriate, to support OPERABILITY of the associated Functions. The interlocks are:

Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when the respective Intermediate Range Neutron Flux channel increases to approximately one decade above the channel lower range limit. The P-6 interlock ensures that the following are performed:

(1) On increasing power, the P-6 interlock allows the manual block of the respective Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed.

(2) On decreasing power, the P-6 interlock automatically energizes the source range detectors and enables the Source Range Neutron Flux reactor trip.

(3) On decreasing power, the P-6 interlock automatically resets the flux doubling block control ensuring the source range neutron flux doubling circuit is enabled. Normally, the source range neutron flux doubling circuit is manually blocked by the main control room operator during the reactor startup.

Reactor Coolant Average Temperature (T_{avg}), P-9

With T_{avg} channels less than the P-9 setpoint, the operator can manually block the Steam Generator Narrow Range Water Level – Low 2 Reactor Trip. This allows rod testing during routine maintenance. With T_{avg} channels greater than P-9 setpoint, the Steam Generator Narrow Range Water Level - Low 2 Reactor Trip is automatically enabled.

Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power as determined by the respective power-range detector. The P-10 interlock ensures that the following are performed:

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

(1) On increasing power, the P-10 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure – Low 2 Setpoint,
- Pressurizer Water Level – High 3,
- Reactor Coolant Flow – Low 2, and
- RCP Speed – Low 2.

These reactor trips are only required when operating above the P-10 setpoint (approximately 10% power). These reactor trips provide protection against violating the DNBR limit. Below the P-10 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) On increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip.

(3) On increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux – Low Setpoint reactor trip.

(4) On increasing power, the P-10 interlock automatically provides a backup block signal to the Source Range Neutron Flux reactor trip and also to de-energize the source range detectors.

(5) On decreasing power, the P-10 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure – Low 2 Setpoint,
- Pressurizer Water Level – High 3,
- Reactor Coolant Flow – Low 2, and
- RCP Speed – Low 2.

(6) On decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux – Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Pressurizer Pressure, P-11

With pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Steam Generator Narrow Range Water Level – High 3 reactor Trip. This allows rod testing with the steam generators in cold wet layup. With pressurizer pressure channels greater than P-11 setpoint, the Steam Generator Narrow Range Water Level – High 3 reactor Trip is automatically enabled. The operator can also enable these actuations by use of the respective manual reset.

The LCO generally requires OPERABILITY of four channels in each instrumentation Function.

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Power Range Neutron Flux

   The power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The power range detectors provide input to the PLS. Minimum requirements for protection and control is achieved with three channels OPERABLE. The fourth channel is provided to increase plant availability, and permits the plant to run for an indefinite time with a single channel in trip or bypass. This Function also satisfies the requirements of IEEE 603 (Ref. 4) with 2/4 logic. This Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor. Each flux signal is compensated for changes in cold leg density, using cold leg temperature and pressurizer pressure.

   a. Power Range Neutron Flux – High

      The Power Range Neutron Flux – High trip Function provides backup protection, at high power levels, against a positive reactivity excursion during power operations. Positive reactivity excursions can be caused by rod withdrawal or reductions in RCS temperature.

      The LCO requires four Power Range Neutron Flux – High channels to be OPERABLE in MODES 1 and 2.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux – High trip must be OPERABLE. This Function provides a backup trip to Overtemperature ΔT, Overpower ΔT, and Power Range Neutron Flux – High Positive Rate to terminate the reactivity excursion and shutdown the reactor. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux – High trip does not have to be OPERABLE because the reactor is shutdown and a reactivity excursion in the power range cannot occur. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6. In addition, the power range detectors cannot detect neutron levels in this range.

b. Power Range Neutron Flux – Low

The LCO requirement for the Power Range Neutron Flux – Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions. The Trip Setpoint reflects only steady state instrument uncertainties as this Function does not provide primary protection for any event that results in a harsh environment.

The LCO requires four of the Power Range Neutron Flux – Low channels to be OPERABLE in MODE 1 below the Power Range Neutron Flux P-10 Setpoint and MODE 2.

In MODE 1, below the Power Range Neutron Flux P-10 setpoint and in MODE 2, the Power Range Neutron Flux – Low trip must be OPERABLE. This Function may be manually blocked by the operator when the respective power range channel is greater than approximately 10% of RTP (P-10 setpoint). This Function is automatically unblocked when the respective power range channel is below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux – High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux – Low trip Function does not have to be OPERABLE because the reactor is shutdown and the power range detectors cannot detect neutron levels generated in MODES 3, 4, 5, and 6. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2. Power Range Neutron Flux – High Positive Rate

The Power Range Neutron Flux – High Positive Rate trip Function ensures that protection is provided against moderate to rapid increases in neutron flux which are characteristic of an uncontrolled rod cluster control assembly (RCCA) withdrawal or a drive rod housing rupture and the accompanying ejection of the RCCA. This Function works with the Overtemperature ΔT and Overpower ΔT trip functions, through overlap, to prevent overpower and low DNBR for moderate to rapid power excursions. The Power Range Neutron Flux – High Positive Rate trip uses the same channels as discussed for Function 1 above. Each flux signal is compensated for changes in cold leg density, using cold leg temperature and pressurizer pressure.

The LCO requires four Power Range Neutron Flux – High Positive Rate channels to be OPERABLE. In MODE 1 or 2, when there is a potential to add positive reactivity from an uncontrolled RCCA withdrawal or a rod ejection accident (REA), the Power Range Neutron Flux – High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux – High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or overpressure protection is provided by meeting the requirements of LCO 3.4.14, “Low Temperature Overpressure Protection (LTOP).” In addition, the power range detectors cannot detect neutron levels present in this MODE.

3. Overtemperature ΔT

The Overtemperature ΔT trip Function ensures that protection is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. This Function works with the Power Range Neutron Flux – High Positive Rate trip to provide DNB protection for low to moderate positive reactivity excursions due to uncontrolled RCCA withdrawal events. The inputs to the Overtemperature ΔT trip include all combinations of pressure, power, coolant temperature, and axial power distribution, assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

the core to the measurement system. The Overtemperature ΔT trip Function uses the measured T_{HOT} and T_{COLD} in each loop, together with the measured pressurizer pressure, to compute the reactor core thermal power. Equations to fit the properties of density and enthalpy are programmed in the software, such that the ΔT power signal is presented as a percent of RTP for direct comparison with measured calorimetric power. The overtemperature ΔT setpoint is automatically varied for changes in the parameters that affect DNB as follows:

- reactor core inlet temperature – the Trip Setpoint is varied to correct for changes in core inlet temperature based on measured changes in cold leg temperature with dynamic compensation to account for cold leg-to-core transit time;

- pressurizer pressure – the Trip Setpoint is varied to correct for changes in system pressure; and

- axial power distribution – the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the PMS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower power range detectors, the Trip Setpoint is reduced in accordance with algorithms documented in the SP.

Dynamic compensation of the ΔT power signal is included for system piping delays from the core to the temperature measurement system. The Overtemperature ΔT trip Function is calculated for each loop as described in the SP. A detailed description of this trip is provided in Reference 7. This Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip. No credit is taken in the safety analyses for the turbine runback.

The LCO requires four channels (two per loop) of the Overtemperature ΔT trip Function to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

4. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip function. This Function works with the Power Range Neutron Flux – High Positive Rate trip to limit overpower for low to moderate positive reactivity excursions due to uncontrolled RCCA withdrawal events. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the same ΔT power signal generated for the Overtemperature ΔT. The setpoint is automatically varied with the following parameter:

- Axial power distribution – the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the PMS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower power range detectors, the Trip Setpoint is reduced in accordance with algorithms documented in the SP.

The Overpower ΔT trip Function is calculated for each loop as described in the SP. A detailed description of this trip is provided in Reference 7. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide protection for a steam line break and may be in a harsh environment. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback reduces turbine power and reactor power. A reduction in power normally alleviates the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels (two per loop) of the Overpower ΔT trip Function to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The Overpower ΔT Function receives input from channels shared

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

5. Pressurizer Pressure

a. Pressurizer Pressure – Low 2

The Pressurizer Pressure – Low 2 trip Function ensures that protection is provided against violating the DNBR limit due to low pressure. The Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide primary protection for an event that results in a harsh environment.

The LCO requires four channels of Pressurizer Pressure – Low 2 to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure – Low 2 trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-10 interlock. On decreasing power, this trip Function is automatically blocked below P-10. Below the P-10 setpoint, no conceivable power distributions can occur that would cause DNB concerns.

b. Pressurizer Pressure – High 2

The Pressurizer Pressure – High 2 trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the safety valves to prevent RCS overpressure conditions. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires four channels of the Pressurizer Pressure – High 2 to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 or 2, the Pressurizer Pressure – High 2 trip must be OPERABLE to help prevent RCS overpressurization and LCOs, and minimizes challenges to the safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure – High 2 trip Function does not have to be OPERABLE because transients which could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate plant conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

6. Pressurizer Water Level – High 3

The Pressurizer Water Level – High 3 trip Function provides a backup signal for the Pressurizer Pressure – High 2 trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. Each water level signal is compensated using pressurizer reference leg temperature, pressurizer pressure and RCS wide range pressure. A reactor trip is actuated prior to the pressurizer becoming water solid. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment. The level channels do not actuate the safety valves.

The LCO requires four channels of Pressurizer Water Level – High 3 to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 when there is a potential for overfilling the pressurizer, the Pressurizer Water Level – High 3 trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-10 interlock. On decreasing power, this trip Function is automatically blocked below P-10. Below the P-10 setpoint, transients which could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate plant conditions and take corrective actions.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

7. Reactor Coolant Flow – Low 2

The Reactor Coolant Flow – Low 2 trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS hot legs. Above the P-10 setpoint, a loss of flow in any RCS hot leg will actuate a Reactor trip.

Each RCS hot leg has four flow detectors to monitor flow. Each flow signal is compensated for changes in reactor coolant density using the ΔT power signal. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment. Two first-order lag filters in series are used to reduce noise on the input signals.

The LCO requires four Reactor Coolant Flow – Low 2 channels per hot leg to be OPERABLE in MODE 1 above P-10. Four OPERABLE channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 above the P-10 setpoint, when a loss of flow in one RCS hot leg could result in DNB conditions in the core, the Reactor Coolant Flow – Low 2 trip must be OPERABLE.

8. Reactor Coolant Pump (RCP) Bearing Water Temperature – High 2

The RCP Bearing Water Temperature – High 2 reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS cold leg. This Function is lead-lag compensated. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires four RCP Bearing Water Temperature – High 2 channels per RCP to be OPERABLE in MODE 1 or 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 or 2, when a loss of flow in any RCS cold leg could result in DNB conditions in the core, the RCP Bearing Water Temperature – High 2 trip must be OPERABLE.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

9. Reactor Coolant Pump Speed – Low 2

The RCP Speed – Low 2 trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS cold legs. The speed of each RCP is monitored. Above the P-10 setpoint a low speed detected on two or more RCPs will initiate a reactor trip. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires four RCP Speed – Low 2 channels (one per pump) to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 above the P-10 setpoint, the RCP Speed – Low 2 trip must be OPERABLE. Below the P-10 setpoint, all reactor trips on loss of flow are automatically blocked since no power distributions are expected to occur that would cause a DNB concern at this low power level. Above the P-10 setpoint, the reactor trip on loss of flow in two or more RCS cold legs is automatically enabled.

10. Steam Generator Narrow Range Water Level – Low 2

The Steam Generator (SG) Narrow Range Water Level – Low 2 trip Function ensures that protection is provided against a loss of heat sink. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range Low 2 water level in any steam generator is indicative of a loss of heat sink for the reactor. Each water level signal is compensated using steam line pressure. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide primary protection for an event that results in a harsh environment. This Function also contributes to the coincidence logic for the ESFAS Function of opening the Passive Residual Heat Removal (PRHR) discharge valves.

The LCO requires four channels of SG Narrow Range Water Level – Low 2 per SG to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

In MODE 1 or 2, when the reactor requires a heat sink, the SG Narrow Range Water Level – Low 2 trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater System (non-safety related). The Main Feedwater System is normally in operation in MODES 1 and 2. PRHR is the safety related backup heat sink for the reactor. During normal startups and shutdowns, the Main and Startup Feedwater Systems (non-safety related) can provide feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Narrow Range Water Level – Low 2 Function does not have to be OPERABLE because the reactor is not operating or even critical. The P-9 interlock is provided on this Function to permit bypass of the trip Function when T_{avg} is below P-9.

11. Steam Generator Narrow Range Water Level – High 3

The SG Narrow Range Water Level – High 3 trip Function ensures that protection is provided against excessive feedwater flow by closing the main feedwater control valves, tripping the turbine, and tripping the reactor. Each water level signal is compensated using steam line pressure. While the transmitters (d/p cells) are located inside containment, the events which this function protects against cannot cause severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

The LCO requires four channels of SG Narrow Range Water Level – High 3 per SG to be OPERABLE. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODES 1 and 2 above the P-11 interlock, the SG Narrow Range Water Level – High 3 trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater System (non-safety related). The Main Feedwater System is only in operation in MODES 1 and 2. In MODE 3, 4, 5, or 6, the SG Narrow Range Water Level – High 3 Function does not have to be OPERABLE because the reactor is not operating or even critical. The P-11 interlock is provided on this Function to permit bypass of the trip Function when the pressure is below P-11. This bypass is necessary to permit rod testing when the steam generators are in wet layup.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

12. Passive Residual Heat Removal Actuation

The Passive Residual Heat Removal (PRHR) Actuation reactor trip ensures that a reactivity excursion due to cold water injection will be minimized upon an inadvertent operation of the PRHR discharge valves. The reactor trip is needed to ensure the reactor is tripped in the event PRHR is not initiated by ESFAS actuation signals. The two discharge valves for the PRHR HX are monitored by PMS using valve position indicators as inputs into PMS.

The LCO requires four channels of PRHR discharge valve position indication per valve to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODES 1 and 2, the Passive Heat Removal Actuation reactor trip must be OPERABLE. In MODES 3, 4, 5, and 6, the Passive Heat Removal Initiation reactor trip Function does not have to be OPERATIONAL because the reactor is not operating or critical.

The RTS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.1-1.

In the event a channel's as-found condition is outside the as-found tolerance described in the SP, or the channel is not functioning as required, or the transmitter, instrument loop, signal processing electronics, or trip output is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

A.1

Condition A addresses the situation where one required channel for one or more Functions is inoperable. With one channel inoperable, the inoperable channel must be placed in a bypass or trip condition within 6 hours. If one inoperable channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one inoperable channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three

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ACTIONS (continued)

remaining channels will not prevent the protective function.) The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 6.

B.1 and B.2

Condition B addresses the situation where one or more Functions have two required channels inoperable. With two channels for a Function inoperable, one inoperable channel must be placed in a bypass condition within 6 hours and one inoperable channel must be placed in a trip condition within 6 hours. If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place the inoperable channels in the bypassed or tripped condition is justified in Reference 6.

C.1

Condition C addresses the situation where any Required Action and associated Completion Time of Condition A or B is not met, or three or more channels are inoperable for one or more Functions. Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1.

D.1

Condition D is entered from Required Action C.1 when any Required Action and associated Completion Time of Condition A or B is not met, or three or more channels are inoperable for one or more Functions, and it is identified as the appropriate Condition referenced in Table 3.3.1-1. If the channel(s) is not restored to OPERABLE status or placed in trip or bypass, as applicable, within the allowed Completion Time, or three or more channels are inoperable for a Function, the plant must be placed in MODE 3. Six hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

E.1

Condition E is entered from Required Action C.1 when any Required Action and associated Completion Time of Condition A or B is not met, or three or more channels for one or more Functions are inoperable, and it is identified as the appropriate Condition referenced in Table 3.3.1-1. If the channel(s) is not restored to OPERABLE status or placed in trip or bypass, as applicable, within the allowed Completion Time, or three or

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BASES

ACTIONS (continued)

more channels are inoperable for a Function, thermal power must be reduced to below the P-10 interlock; a condition in which the LCO does not apply. The allowed Completion Time is reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified in the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

SR 3.3.1.1

This SR compares the calorimetric heat balance to the nuclear instrumentation channel output every 24 hours. If the calorimetric measurement at ≥ 15% RTP, differs from the nuclear instrument channel output by > 5% RTP, the nuclear instrument channel is not declared inoperable, but must be adjusted. If the nuclear instrument channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify this SR. The first Note clarifies that this Surveillance is required only if THERMAL POWER is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels the calorimetric data from feedwater flow venturi measurements are less accurate. The second Note is required because, at power levels ≥ 15% RTP, calorimetric uncertainty and control rod insertion create the potential for miscalibration of the nuclear instrumentation channel. Therefore, if the calorimetric heat measurement is ≥ 15% RTP, and if the nuclear instrumentation channel indicated power is lower than the calorimetric measurement by > 5% RTP, then the nuclear instrumentation channel shall be adjusted upward to match the calorimetric measurement. No nuclear instrumentation channel adjustment is required if the nuclear instrumentation channel is higher than the calorimetric measurement.

The Frequency of every 24 hours is adequate based on plant operating experience, considering instrument reliability and operating history data for instrument drift.

Together, these factors demonstrate the change in the absolute difference between nuclear instrumentation and heat balance calculated powers rarely exceeds 5% RTP in any 24 hours period.

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SURVEILLANCE REQUIREMENTS (continued)

In addition, main control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.2

This SR compares the calorimetric heat balance to the calculated ΔT power ($q_{\Delta T}$) in each Division every 24 hours. If the calorimetric measurement between 70% and 100% RTP, differs from the calculated ΔT power by > 3% RTP, the Function is not declared inoperable, but the conversion factor, ΔT°, must be adjusted. If ΔT° cannot be properly adjusted, the Function is declared inoperable in the affected Division(s).

Three Notes modify this SR. The first Note indicates that ΔT° shall be adjusted consistent with the calorimetric results if the absolute difference between the calculated ΔT power and the calorimetric measurement between 70% and 100% RTP is > 3% RTP.

The second Note clarifies that this Surveillance is required only if THERMAL POWER is ≥ 50% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 50% RTP. At lower power levels, the calorimetric data from feedwater flow venturi measurements are less accurate. The calculated ΔT power is normally stable (less likely to need adjustment or to be grossly affected by changes in the core loading pattern than the nuclear instrumentation), and its calibration should not be unnecessarily altered by a possibly inaccurate calorimetric measurement at low power.

The third Note is required because at power levels below 70%, calorimetric uncertainty creates the potential for non-conservative adjustment of the ΔT° conversion factor, in cases where the calculated ΔT power would be reduced to match the calorimetric power. Therefore, if the calorimetric heat measurement is less than 70% RTP, and if the calculated ΔT power is lower than the calorimetric measurement by > 5%, then the ΔT° conversion factor shall be adjusted so that the calculated ΔT power matches the calorimetric measurement. No ΔT° conversion factor adjustment is required if the calculated ΔT power is higher than the calorimetric measurement.

The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and the limited effects of fuel burnup and rod position changes on the accuracy of the calculated ΔT power.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

This SR compares the AXIAL FLUX DIFFERENCE determined using the incore system to the nuclear instrument channel AXIAL FLUX DIFFERENCE every 31 effective full power days (EFPD) and adjusts the excore nuclear instrument channel if the absolute difference between the incore and excore AFD is ≥ 1.5% AFD.

Each nuclear instrument channel is calibrated to an average weighted peripheral AFD, which accounts for the fact that neutron leakage from the peripheral fuel assemblies nearest each excore detector will have the largest effect on the channel response. This calibration method reduces the effect of changes in the radial power distribution, caused by either burnup or control rod motion, on the channel AFD calibration. The calibration method is consistent with the development of the f(ΔI) penalty functions for the overpower ΔT and overtemperature ΔT functions, which are made a function of the same average weighted peripheral AFD (i.e., the AFD used in determining the f(ΔI) penalty is calculated using the same radial weighting factors as are used to calibrate the excore detector nuclear instrument channels). The incore AFD used as the basis for comparison when performing this SR is also calculated in the same weighted peripheral manner.

If the absolute difference is ≥ 1.5% AFD the nuclear instrument channel is still OPERABLE, but must be readjusted. If the nuclear instrument channel cannot be properly readjusted, the channel is declared inoperable. This surveillance is performed to verify the f(ΔI) input to the overpower ΔT and overtemperature ΔT functions.

Two Notes modify this SR. The first Note indicates that the excore nuclear instrument channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 1.5% AFD. Note 2 clarifies that the Surveillance is required only if THERMAL POWER is ≥ 20% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP. Below 20% RTP, the design of the incore detector system, low core power density, and detector accuracy make use of the incore detectors inadequate for use as a reference standard for comparison to the excore channels.

The Frequency of every 31 EFPD is adequate based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4

This SR is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be adjusted to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overpower ΔT and the overtemperature ΔT Function.

A Note modifies this SR. The Note states that this Surveillance is required only if THERMAL POWER is > 50% RTP and that 24 hours is allowed for performing the first surveillance after reaching 50% RTP.

The Frequency of 92 EFPD is adequate based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.5

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION is performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation. Transmitter calibration must be performed consistent with the assumptions of the setpoint methodology. The differences between the current as-found values and the previous as-left values must be consistent with the transmitter drift allowance used in the setpoint methodology.

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SURVEILLANCE REQUIREMENTS (continued)

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months).

Interlocks implicitly required to support the Function's OPERABILITY are also addressed by this CHANNEL CALIBRATION. This portion of the CHANNEL CALIBRATION ensures the associated Function is not bypassed when required to be enabled. This can be accomplished by ensuring the interlocks are calibrated properly in accordance with the SP. If the interlock is not automatically functioning as designed, the condition is entered into the Corrective Action Program and appropriate OPERABILITY evaluations performed for the affected Function. The affected Function's OPERABILITY can be met if the interlock is manually enforced to properly enable the affected Function. When an interlock is not supporting the associated Function's OPERABILITY at the existing plant conditions, the affected Function's channels must be declared inoperable and appropriate ACTIONS taken.

The RCP speed sensor (Function 9) is calibrated using a qualitative assessment of the behavior that is obtained through insulation resistance and sensor resistance testing to complement the normal calibration of the remaining adjustable devices in Function 9.

This SR is modified by a Note stating that this test shall include verification that the time constants are adjusted to within limits where applicable.

SR 3.3.1.6

This SR is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION is performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

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SURVEILLANCE REQUIREMENTS (continued)

The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 20% RTP. Below 20% RTP, the design of the incore detector system, low core power density, and detector accuracy make use of the incore detectors inadequate for use as a reference standard for comparison to the excore channels.

Interlocks implicitly required to support the Function's OPERABILITY are also addressed by this CHANNEL CALIBRATION. This portion of the CHANNEL CALIBRATION ensures the associated Function is not bypassed when required to be enabled. This can be accomplished by ensuring the interlocks are calibrated properly in accordance with the SP. If the interlock is not automatically functioning as designed, the condition is entered into the Corrective Action Program and appropriate OPERABILITY evaluations performed for the affected Function. The affected Function’s OPERABILITY can be met if the interlock is manually enforced to properly enable the affected Function. When an interlock is not supporting the associated Function’s OPERABILITY at the existing plant conditions, the affected Function's channels must be declared inoperable and appropriate ACTIONS taken.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 24 month Frequency.

SR 3.3.1.7

This SR is the performance of a TADOT of the Passive Residual Heat Removal Actuation valve position indicator contact inputs. This TADOT is performed every 24 months.

The Frequency is based on the known reliability of the Function and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.8

This SR verifies that the individual channel/division actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response Time testing criteria are included in Reference 1.

In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by a series of overlapping tests and allocated values such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 1) response time.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the allocated sensor, signal processing and actuation logic response times, where approved, with actual response time tests on the remainder of the channel. Allocations for signal processing and actuation logic response times may be obtained from the Protection and Safety Monitoring System Functional Requirements document. The response time of the RCP speed sensor (Function 9) and its associated preamplifier is allocated in lieu of measurement, based on the specific design of the component models being used.

The Passive Residual Heat Removal (PRHR) Actuation Function RTS RESPONSE TIME is the time interval between input of a PRHR discharge valve not-fully-closed position feedback signal and the loss of gripper coil voltage. The RTS RESPONSE TIME for the PRHR actuation does not include testing actuation of the discharge valves by ESFAS instrumentation signals because it cannot be tested if an ESFAS function (e.g., CMT Actuation) has already caused a reactor trip.

Each division response must be verified every 24 months on a STAGGERED TEST BASIS (i.e., all four Protection Channel Sets would be tested after 96 months). Response times cannot be determined during plant operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed on a refueling frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a note indicating that neutron detectors may be excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. FSAR Chapter 7, “Instrumentation and Controls.”
2. FSAR Chapter 15, “Accident Analysis.”
3. WCAP-16361-P, “Westinghouse Setpoint Methodology for Protection Systems – AP1000,” February 2011 (proprietary).
4. Institute of Electrical and Electronic Engineers, IEEE-603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” June 27, 1991.
5. 10 CFR 50.49, “Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants.”
6. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
7. APP-GW-GLR-137, Revision 1, “Bases of Digital Overpower and Overtemperature Delta-T (OPΔT/OPΔT) Reactor Trips,” Westinghouse Electric Company LLC.
8. Not used.
9. WCAP-16675-P, “AP1000 Protection and Safety Monitoring System Architecture Technical Support.”
10. Not used.

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B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Trip System (RTS) Source Range Instrumentation

BASES

BACKGROUND

A description of the RTS Instrumentation is provided in the Bases for LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.”

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain compliance with the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed. A description of RTS interlocks is provided in the Bases for LCO 3.3.1.

The RTS Source Range Neutron Flux – High Setpoint trip Function provides protection against an uncontrolled bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux – Low Setpoint and Intermediate Range Neutron Flux trip Functions. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The safety analyses do not take credit for the Source Range Neutron Flux trip Function. Even though the safety analyses take no credit for the Source Range Neutron Flux trip, the functional capability at the specified Trip Setpoint is assumed to be available and the trip is implicitly assumed in the safety analyses.

The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. This trip can be manually blocked by the main control room operator when above the P-6 setpoint (Intermediate Range Neutron Flux interlock) and is automatically unblocked when below the P-6 setpoint. The manual block of the trip function also de-energizes the source range detectors. The source range detectors are automatically re-energized when below the P-6 setpoint. The trip is automatically blocked when above the P-10 setpoint (Power Range Neutron Flux interlock). The source range trip is the only RTS automatic protective Function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires four channels of Source Range Neutron Flux to be OPERABLE in MODE 2 below the P-6 setpoint, and in MODE 3, 4, or 5 with the Plant Control System (PLS) capable of rod withdrawal or one or

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

more rods not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s). Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODE 3, 4, or 5 with the Plant Control System (PLS) incapable of rod withdrawal and all rods fully inserted, the LCO does not require the Source Range Neutron Flux channels for reactor trip Functions to be OPERABLE.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux – Low Setpoint trip will provide core protection for reactivity accidents.

In MODE 3, 4, or 5 with the reactor shutdown, the Source Range Neutron Flux trip Function must also be OPERABLE. If the PLS is capable of rod withdrawal or one or more rods are not fully inserted, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the PLS is not capable of rod withdrawal, the source range detectors are required to be OPERABLE to provide monitoring of neutron levels and provide protection for events like an inadvertent boron dilution. These Functions are addressed in LCO 3.3.8, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” LCO 3.3.15, “Engineered Safety Feature Actuation System (ESFAS) Actuation Logic – Operating,” and LCO 3.3.16, “Engineered Safety Feature Actuation System (ESFAS) Actuation Logic – Shutdown.” The requirements for the source range detectors in MODE 6 are addressed in LCO 3.9.3, “Nuclear Instrumentation.”

The RTS Source Range instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

In the event a channel’s as-found condition is outside the as-found tolerance described in the SP, or the channel is not functioning as required, or the transmitter, instrument loop, signal processing electronics, or trip output is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

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BASES

ACTIONS (continued)

A.1

Condition A addresses the situation where one source range instrumentation channel is inoperable in MODE 2. With one channel inoperable, the inoperable channel must be placed in a bypass or trip condition within two hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function). The 2 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is consistent with the Intermediate Range Neutron Flux instrumentation Completion Times provided in LCO 3.3.3.

B.1 and B.2

Condition B addresses the situation where two source range instrumentation channels are inoperable in MODE 2. With two channels inoperable, one affected channel must be placed in a bypass condition within 2 hours and one affected channel must be placed in a trip condition within 2 hours. If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 2 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is consistent with the Intermediate Range Neutron Flux instrumentation Completion Times provided in LCO 3.3.3.

C.1

Condition C is entered when any Required Action and associated Completion Time of Conditions A or B is not met. If the channel(s) is not restored to OPERABLE status or placed in trip or bypass, as applicable, within the allowed Completion Time, Required Action C.1 requires immediate suspension of operations involving positive reactivity additions.

D.1

Condition D addresses the situation where one or two source range instrumentation channels are inoperable in MODE 3, 4, or 5. With one or two source range instrumentation channels inoperable, three of the four required channels must be restored to OPERABLE status within 48 hours. The 48 hours to restore three of four source range instrumentation channels is justified in Reference 2.

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BASES

ACTIONS (continued)

E.1 and E.2

Condition E is entered when the Required Action and associated Completion Time of Condition D is not met. If three of the four required source range instrumentation channels are not restored to OPERABLE status within the allowed Completion Time, Required Action E.1 requires that action be initiated to fully insert all rods within 1 hour, and Required Action E.2 requires that the PLS be placed in a condition incapable of rod withdrawal within 1 hour. The allowed Completion Time is reasonable, based on operating experience, to reach the specified condition in an orderly manner and without challenging plant systems.

F.1

Condition F addresses the situation where three or more source range instrumentation channels are inoperable. With three or more channels inoperable, single failure criterion cannot be met and the reactor trip breakers must be opened immediately.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment have drifted outside their corresponding limits.

The Frequency is based on operating experience that demonstrates that channel failure is rare. Automated operator aids may be used to facilitate the performance of the CHANNEL CHECK.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION is performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

The CHANNEL CALIBRATION for the source range neutron detectors consists of obtaining the preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data.

Interlocks implicitly required to support the Function's OPERABILITY are also addressed by this CHANNEL CALIBRATION. This portion of the CHANNEL CALIBRATION ensures the associated Function is not bypassed when required to be enabled. This can be accomplished by ensuring the interlocks are calibrated properly in accordance with the SP. If the interlock is not automatically functioning as designed, the condition is entered into the Corrective Action Program and appropriate OPERABILITY evaluations performed for the affected Function. The Function's OPERABILITY can be met if the interlock is manually enforced to properly enable the affected Function. When an interlock is not supporting the associated Function's OPERABILITY at the existing plant conditions, the affected Function's channels must be declared inoperable and appropriate ACTIONS taken.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 24 month Frequency.

SR 3.3.2.3

This SR verifies that the individual channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response Time testing criteria are included in Reference 1.

In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by a series of overlapping tests and allocated values such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 1) response time.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the allocated sensor, signal processing and actuation logic response times, where approved, with actual response time tests on the remainder of the channel.

Each channel response must be verified every 24 months on a STAGGERED TEST BASIS (i.e., all four Protection Channel Sets would be tested after 96 months). Response times cannot be determined during plant operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed on a refueling frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a note exempting neutron detectors from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

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BASES

REFERENCES

1. FSAR Chapter 7, “Instrumentation and Controls.”
2. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
3. Not used.

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B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Trip System (RTS) Intermediate Range Instrumentation

BASES

BACKGROUND

A description of the RTS Instrumentation is provided in the Bases for LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.”

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain compliance with the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

The Intermediate Range Neutron Flux – High trip Function ensures that protection is provided against an uncontrolled RCCA bank withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux – Low Setpoint trip Function. The intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The safety analyses do not take credit for the Intermediate Range Neutron Flux trip Function. Even though the safety analyses take no credit for the Intermediate Range Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of the RTS. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. This trip can be manually blocked by the main control room operator when above the P-10 setpoint, which is the respective power range channel greater than 10% power, and is automatically unblocked when below the P-10 setpoint, which is the respective power range channel less than 10% power.

This Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires four channels of Intermediate Range Neutron Flux to be OPERABLE. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 below the P-10 setpoint, and in MODE 2, when there is a potential for an uncontrolled rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux – High Setpoint

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

trip and the Power Range Neutron Flux – High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the intermediate range detectors cannot detect neutron levels present in this MODE.

The RTS Intermediate Range instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

In the event a channel's as-found condition is outside the as-found tolerance described in the SP, or the channel is not functioning as required, or the transmitter, instrument loop, signal processing electronics, or trip output is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

A.1, A.2, and A.3

Condition A addresses the situation where one intermediate range instrumentation channel is inoperable with THERMAL POWER greater than or equal to the P-6 interlock setpoint. With one channel inoperable, the inoperable channel must be placed in a bypass or trip condition within two hours, or THERMAL POWER must be either reduced below the P-6 interlock setpoint or increased above the P-10 interlock setpoint within 2 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function). If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function). The 2 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 2.

As an alternative to placing the inoperable channel in bypass or trip if THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 2 hours are allowed to reduce THERMAL POWER below

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BASES

ACTIONS (continued)

the P-6 setpoint or to increase the THERMAL POWER above the P-10 setpoint. The Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the power range detectors perform the monitoring and protective functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment below P-6, and takes into account the redundant capability afforded by the remaining OPERABLE channels and the low probability of their failure during this period.

B.1.1, B.1.2, B.2, and B.3

Condition B addresses the situation where two intermediate range instrumentation channels are inoperable with THERMAL POWER greater than or equal to the P-6 interlock setpoint. With two intermediate range channels inoperable, one inoperable channel must be placed in a bypass condition and one inoperable channel must be placed in a trip condition within 2 hours, or THERMAL POWER must be either reduced below the P-6 interlock setpoint or increased above the P-10 interlock setpoint within 2 hours. If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 2 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 2.

As an alternative to placing the channels in bypass or trip if THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 2 hours are allowed to reduce THERMAL POWER below the P-6 setpoint or to increase the THERMAL POWER above the P-10 setpoint. The Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the power range detectors perform the monitoring and protective functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment below P-6, and takes into account the redundant capability afforded by the two remaining OPERABLE channels and the low probability of their failure during this period.

C.1

Condition C addresses the situation of one or two intermediate range instrumentation channels are inoperable with THERMAL POWER below the P-6 interlock setpoint. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. With one or two

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BASES

ACTIONS (continued)

intermediate range instrumentation channels inoperable, three of the four required channels must be restored to OPERABLE status prior to increasing THERMAL POWER above the P-6 setpoint. With the unit in this condition, below P-6, the Source Range Neutron Flux channels perform the monitoring and protection functions.

D.1, D.2, and D.3

Condition D addresses the situation where three or more intermediate range instrumentation channels are inoperable. With three or more channels inoperable, operations involving positive reactivity addition must be suspended immediately. This will preclude any power level increase since there are insufficient OPERABLE Intermediate Range channels to adequately monitor power escalation. In addition, THERMAL POWER must be reduced below the P-6 interlock setpoint within 2 hours, and the plant must be placed in MODE 3 within 7 hours. The allowed Completion Times for Required Actions D.2 and D.3 are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment have drifted outside their corresponding limits.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note. The Note states that the SR is not applicable in MODE 1. In MODE 1, the OPERABILITY of the Intermediate Range Neutron Instrumentation is verified by self-checking features in lieu of performing a CHANNEL CHECK.

The Frequency is based on operating experience that demonstrates that channel failure is rare. Automated operator aids may be used to facilitate the performance of the CHANNEL CHECK.

SR 3.3.3.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION is performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

The CHANNEL CALIBRATION for the intermediate range neutron detectors consists of obtaining the detector plateau curves, evaluating those curves, and comparing the curves to the manufacturer's data.

Interlocks implicitly required to support the Function's OPERABILITY are also addressed by this CHANNEL CALIBRATION. This portion of the CHANNEL CALIBRATION ensures the associated Function is not bypassed when required to be enabled. This can be accomplished by ensuring the interlocks are calibrated properly in accordance with the SP. If the interlock is not automatically functioning as designed, the condition is entered into the Corrective Action Program and appropriate OPERABILITY evaluations performed for the affected Function. The

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BASES

SURVEILLANCE REQUIREMENTS (continued)

affected Function's OPERABILITY can be met if the interlock is manually enforced to properly enable the affected Function. When an interlock is not supporting the associated Function's OPERABILITY at the existing plant conditions, the affected Function's channels must be declared inoperable and appropriate ACTIONS taken.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 24 month Frequency.

SR 3.3.3.3

This SR verifies that the individual channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response Time testing criteria are included in Reference 1.

In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by a series of overlapping tests and allocated values such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 1) response time.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the allocated sensor, signal processing and actuation logic response times, where approved, with actual response time tests on the remainder of the channel.

Each channel response must be verified every 24 months on a STAGGERED TEST BASIS (i.e., all four Protection Channel Sets would be tested after 96 months). Response times cannot be determined during plant operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed on a refueling frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a note exempting neutron detectors from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. FSAR Chapter 7, “Instrumentation and Controls.”
2. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
3. Not used.

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B 3.3 INSTRUMENTATION

B 3.3.4 Reactor Trip System (RTS) Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

A description of the RTS Instrumentation is provided in the Bases for LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.”

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain compliance with the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

This LCO provides requirements for the automatic inputs from the Engineered Safety Feature Actuation System (ESFAS) to the RTS. The safety analyses and OPERABILITY requirements applicable to the RTS ESFAS Instrumentation Functions are discussed below:

1. Safeguards Actuation Input from Engineered Safety Feature Actuation System – Automatic

   The Safeguards Actuation Input from ESFAS – Automatic ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal which initiates the Safeguards Actuation signal. This is a condition of acceptability for the Loss of Coolant Accident (LOCA). However, other transients and accidents take credit for varying levels of ESFAS performance and rely upon rod insertion, except for the most reactive rod which is assumed to be fully withdrawn, to ensure reactor shutdown.

   The LCO requires four automatic channels of Safeguards Actuation Signal Input from ESFAS to be OPERABLE in MODES 1 and 2. Four automatic channels are provided to permit one channel to be in bypass indefinitely and still ensure no single random failure will disable this trip Function.

   A reactor trip is initiated every time a Safeguards Actuation signal is present. Therefore, this trip Function must be OPERABLE in MODES 1 and 2, when the reactor is critical, and must be shutdown in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2. ADS Stages 1, 2 and 3 Actuation Input from Engineered Safety Feature Actuation System – Automatic

   The LCO requirement for this Function provides a reactor trip for any event that may initiate depressurization of the reactor.

   The LCO requires four automatic channels of the ADS Stages 1, 2, and 3 Actuation Input from ESFAS Function to be OPERABLE. Four OPERABLE channels are provided to ensure that a random failure of a single logic channel will not prevent reactor trip.

   This trip Function must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, this RTS trip Function must be OPERABLE when the Plant Control System (PLS) is capable of rod withdrawal, or one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

3. Core Makeup Tank (CMT) Actuation Input from Engineered Safety Feature Actuation System – Automatic

   The LCO requirement for this Function provides a reactor trip for any event that may initiate CMT injection.

   The LCO requires four channels of the CMT Actuation Input from ESFAS Function to be OPERABLE. Four OPERABLE channels are provided to ensure that random failure of a single logic channel will not prevent reactor trip.

   This trip Function must be OPERABLE in MODES 1 and 2 when the reactor is critical. In MODE 3, 4, and 5 this RTS trip Function must be OPERABLE when the PLS is capable of rod withdrawal, or one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

The RTS ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.4-1.

A.1

Condition A addresses the situation where one or more Functions have one or two channels inoperable in MODE 1 or MODE 2. With one or two channels inoperable, three of four channels must be restored to OPERABLE status within 6 hours. Restoring all channels but one to OPERABLE status ensures that a single failure will not prevent the protective function, nor will it cause the protective function (with the exception of a limited number of PMS component failures). The Completion Time takes into consideration the redundant capability provided by the remaining redundant OPERABLE channels and the low probability of occurrence of an event during this period that may require the protection afforded by this Function. The 6 hour Completion Time is considered reasonable since the protective function will still function.

B.1

Condition B addresses the situation where the Required Action and associated Completion Time of Condition A is not met, or there are one or more Functions with three or more channels inoperable in MODE 1 or MODE 2. Required Action B.1 directs that the plant must be placed in MODE 3 within 6 hours. The allowed Completion Time for Required Action B.1 is reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

C.1

Condition C addresses the situation where one or more Functions have one or two channels inoperable in MODE 3, 4 or 5. With one or two channels inoperable, three of the four channels must be restored to OPERABLE status in 48 hours. Restoring all channels but one to OPERABLE status ensures that a single failure will not prevent the protective function, nor will it cause the protective function (with the exception of a limited number of PMS component failures). The 48 hour Completion Time is considered reasonable since the protective function will still function.

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BASES

ACTIONS (continued)

D.1 and D.2

Condition D addresses the situation where the Required Action and associated Completion Time of Condition C is not met, or there are one or more Functions with three or more channels inoperable in MODE 3, 4, or 5. Required Action D.1 requires initiating action to fully insert all rods within 1 hour, and Required Action D.2 requires that the Plant Control System be placed in a condition incapable of rod withdrawal within 1 hour. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS ESFAS Function are identified in the SRs column of Table 3.3.4-1 for that Function. A Note has been added to the SR table stating that Table 3.3.4-1 determines which RTS ESFAS Function the SR applies to. Function 2 and Function 3 do not require surveillance requirements because self-checking features continuously monitor logic OPERABILITY and alert the operator to any failures.

SR 3.3.4.1

This SR verifies that the individual channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by any series of sequential, overlapping, or total channel measurements such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 1) response time.

This SR measures the response time for the generation of a reactor trip signal from the Safeguards Actuation Input from ESFAS Automatic channels. SR 3.3.8.2 measures the ESF RESPONSE TIME for the generation of the safeguards signal itself.

Response time may be verified by actual response time tests or by the allocated response times, where approved, with actual response time tests on the remainder of the channel.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

Each channel response must be verified every 24 months on a STAGGERED TEST BASIS (i.e., all four Protection Channel Sets would be tested after 96 months). Response times cannot be determined during plant operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed on a refueling frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCE

1. FSAR Chapter 7, “Instrumentation and Controls.”
2. Not used.

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B 3.3 INSTRUMENTATION

B 3.3.5 Reactor Trip System (RTS) Manual Actuation

BASES

BACKGROUND

A description of the RTS Instrumentation is provided in the Bases for LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.”

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain compliance with the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

The LCO requires all RTS Manual Actuation channels performing an RTS Function, listed in Table 3.3.5-1, to be OPERABLE.

The safety analyses and OPERABILITY requirements applicable to the RTS Manual Actuation Functions are discussed below:

1. Manual Reactor Trip

   The Manual Reactor Trip Function ensures that the main control room operator can initiate a reactor trip at any time by using either of two reactor trip actuation devices in the main control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It can be used by the reactor operator to shutdown the reactor whenever any parameter is rapidly trending toward its Trip Setpoint. The safety analyses do not take credit for the Manual Reactor Trip.

   The LCO requires two Manual Reactor Trip actuation channels to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when the Plant Control System (PLS) is capable of rod withdrawal, or one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s). Two independent actuation channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

   In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the shutdown or control rods are withdrawn or the

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

PLS is capable of withdrawing the shutdown or control rods. In MODES 3, 4, and 5, manual initiation of a reactor trip does not have to be OPERABLE if the PLS is not capable of withdrawing the shutdown or control rods or all rods are fully inserted. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function does not have to be OPERABLE.

2. Safeguards Actuation Input from Engineered Safety Feature Actuation System – Manual

The Safeguards Actuation Signal from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal which initiates the Safeguards Actuation signal. This is a condition of acceptability for the Loss of Coolant Accident (LOCA). However, other transients and accidents take credit for varying levels of ESFAS performance and rely upon rod insertion, except for the most reactive rod which is assumed to be fully withdrawn, to ensure reactor shutdown.

The LCO requires two manual channels of Safeguards Actuation Input from ESFAS to be OPERABLE in MODES 1 and 2.

A reactor trip is initiated every time a Safeguards Actuation signal is present. Therefore, this trip Function must be OPERABLE in MODES 1 and 2, when the reactor is critical, and must be shutdown in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical.

3. ADS Stages 1, 2 and 3 Actuation Input from Engineered Safety Feature Actuation System – Manual

The LCO requirement for this Function provides a reactor trip for any event that may initiate depressurization of the reactor.

The LCO requires two manual actuation switch sets for the ADS Stages 1, 2 and 3 Actuation Input from Engineered Safety Feature Actuation System to be OPERABLE. Two OPERABLE manual actuation switch sets are provided to ensure that a random failure of a single logic channel will not prevent reactor trip.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

This trip Function must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, this RTS trip Function must be OPERABLE when the PLS is capable of rod withdrawal or one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

4. Core Makeup Tank (CMT) Actuation Input from Engineered Safety Feature Actuation System – Manual

The LCO requirement for this Function provides a reactor trip for any event that may initiate CMT injection.

The LCO requires two manual actuation channels for the CMT Actuation Input from Engineered Safety Feature Actuation System to be OPERABLE. Two OPERABLE manual actuation channels are provided to ensure that random failure of a single logic channel will not prevent reactor trip.

This trip Function must be OPERABLE in MODES 1 and 2 when the reactor is critical. In MODE 3, 4, and 5 this RTS trip Function must be OPERABLE when the PLS is capable of rod withdrawal or one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

The RTS Manual Actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.5-1.

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BASES

ACTIONS (continued)

A.1

Condition A applies to the RTS Manual Actuation Features identified in Table 3.3.5-1. Condition A addresses the situation where one or more Functions have one manual actuation channel inoperable. One manual actuation channel consists of an actuation switch and the associated hardware (such as contacts and wiring) up to but not including the eight Reactor Trip Breakers. With one manual actuation channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE device is adequate to perform the safety function.

B.1

Condition B addresses the situation where the Required Action and associated Completion Time of Condition A are not met in MODE 1 or 2, or there are one or more Functions with two manual actuation channels inoperable in MODE 1 or 2. Required Action B.1 directs that the plant must be placed in MODE 3 within 6 hours. The allowed Completion Time for Required Action B.1 is reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Condition C addresses the situation where the Required Action and associated Completion Time of Condition A is not met in MODE 3, 4, or 5, or there are one or more Functions with two manual actuation channels inoperable in MODE 3, 4, or 5. Required Action C.1 requires that action be initiated to fully insert all control rods within 1 hour, and Required Action C.2 requires that the Plant Control System be placed in a condition incapable of rod withdrawal within 1 hour. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition in an orderly manner and without challenging plant systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 is the performance of a TADOT of the RTS inputs for Manual Reactor Trip, and from the ESFAS logic for Safeguards Actuation, ADS Stage 1, 2, and 3 Actuation, and CMT Actuation. This TADOT is performed every 24 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

REFERENCES None.

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B 3.3 INSTRUMENTATION

B 3.3.6 Reactor Trip System (RTS) Automatic Trip Logic

BASES

BACKGROUND

A description of the RTS Instrumentation and logic is provided in the Bases for LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.”

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain compliance with the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

The RTS Automatic Trip Logic is required to ensure RTS Automatic Functions can provide the necessary protection.

The Automatic Trip Logic ensures that means are provided to interrupt the power to the CRDMs and allow the rods to fall into the reactor core.

The automatic trip logic includes the Engineered Safety Features (ESF) coincidence logic and the voting logic.

The LCO requires four divisions of RTS Automatic Trip Logic to be OPERABLE. Four OPERABLE divisions are provided to ensure that a random failure of a single logic channel will not prevent reactor trip.

The trip Function must be OPERABLE in MODE 1 or 2, and in MODE 3, 4, or 5, when the Plant Control System (PLS) is capable of rod withdrawal or one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

The RTS Automatic Trip Logic satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A.1

Condition A addresses the situation where one or two RTS Automatic Trip Logic divisions are inoperable in MODE 1 or 2. With one or two divisions inoperable, the Required Action is to restore three of four divisions to OPERABLE status within 6 hours. Restoring all divisions but one to OPERABLE status ensures that a single failure will not prevent the protective function, nor will it cause the protective function (with the exception of a limited number of PMS component failures). The 6 hour Completion Time is considered reasonable since the protective function will still function.

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BASES

ACTIONS (continued)

B.1

Condition B addresses the situation where the Required Action and associated Completion Time of Condition A is not met, or there are three or more divisions inoperable in MODE 1 or 2. Required Action B.1 directs that the plant must be placed in MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

C.1

Condition C addresses the situation where one or two RTS Automatic Trip Logic divisions are inoperable in MODE 3, 4, or 5. With one or two divisions inoperable, the Required Action is to restore three of four divisions to OPERABLE status within 48 hours. Restoring all channels but one to OPERABLE status ensures that a single failure will not prevent the protective function, nor will it cause the protective function (with the exception of a limited number of PMS component failures). The 48 hour Completion Time is considered reasonable since the protective function will still function.

D.1 and D.2

Condition D addresses the situation where the Required Action and associated Completion Time of Condition C is not met, or three or more RTS Automatic Trip Logic divisions are inoperable in MODE 3, 4, or 5. Required Action D.1 requires that action be initiated to fully insert all control rods within 1 hour, and Required Action D.2 requires that the Plant Control System be placed in a condition incapable of rod withdrawal within 1 hour. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

None are required due to self-checking features that continuously monitor logic OPERABILITY and alert the operator to any failures. The OPERABILITY of the Reactor Trip Digital Outputs and the Reactor Trip Matrix Termination Units are verified by the TADOT performed in SR 3.3.7.1.

REFERENCES

None.

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B 3.3 INSTRUMENTATION

B 3.3.7 Reactor Trip System (RTS) Trip Actuation Devices

BASES

BACKGROUND

A description of the RTS Instrumentation is provided in the Bases for LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation.”

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain compliance with the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

The RTS Trip Actuation devices are required to ensure RTS Automatic and Manual Functions can provide the necessary protection.

The LCO requires OPERABILITY of four RTS divisions with two reactor trip breakers (RTB) per division, and one undervoltage and shunt trip mechanism per in-service RTB.

The safety analyses and OPERABILITY requirements applicable to the RTS Trip Actuation Devices are discussed below:

1. Reactor Trip Breakers

   This trip Function applies to the RTBs exclusive of individual trip mechanisms. There are eight reactor trip breakers with two breakers in each division. The reactor trip circuit breakers are arranged in a two-out-of-four logic configuration, such that the tripping of the two circuit breakers associated with one division does not cause a reactor trip. This circuit breaker arrangement is illustrated in FSAR Figure 7.2-1. The LCO requires four divisions of the Reactor Trip Switchgear to be OPERABLE with two trip breakers associated with each required division. This logic is required to meet the safety function assuming a single failure.

   This trip Function must be OPERABLE in MODE 1 or 2. In MODE 3, 4, or 5, this RTS trip Function must be OPERABLE when the Plant Control System (PLS) is capable of rod withdrawal or when one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the PLS, or declared inoperable under Function 1 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening the breakers on a valid signal.

This trip Function must be OPERABLE in MODES 1 and 2. In MODES 3, 4, and 5, this RTS trip Function must be OPERABLE when the Plant Control System (PLS) is capable of rod withdrawal or when one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

The RTS Trip Actuation Devices satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in LCO 3.3.7.

A.1

Condition A addresses the situation where one or both RTS Trip Actuation functions within one division are inoperable. With one division inoperable, the Required Action is to open the affected RTB(s) in the inoperable division within 8 hours. The 8 hour Completion Time is considered reasonable since the protective function will still function.

B.1

Condition B addresses the situation where one or both RTS Trip Actuation functions within two divisions are inoperable. With two divisions inoperable, the Required Action is to restore one division to OPERABLE status within 1 hour. The 1 hour Completion Time is considered reasonable since the protective function will still function.

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BASES

ACTIONS (continued)

C.1

Condition C addresses the situation where the Required Action and associated Completion Time of Condition A or B are not met in MODE 1 or 2, or there are one or both RTS Trip Actuation functions within three or more divisions inoperable in MODE 1 or MODE 2. Required Action C.1 directs that the plant must be placed in MODE 3 within 6 hours. The allowed Completion Times for Required Action C.1 is reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

D.1 and D.2

Condition D addresses the situation where the Required Action and associated Completion Time of Condition A or B are not met in MODE 3, 4, or 5, or there are one or both RTS Trip Actuation functions within three or more divisions inoperable in MODE 3, 4, or 5. Required Action D.1 requires initiating action to fully insert all control rods within 6 hours, and Required Action D.2 requires that the Plant Control System be placed in a condition incapable of rod withdrawal within 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a TADOT on the Reactor Trip Digital Outputs, the Reactor Trip Matrix Termination Units, and on both reactor trip breakers associated with a single division every 92 days on a STAGGERED TEST BASIS for four divisions. This test shall verify OPERABILITY by actuation of the end devices.

The Reactor Trip Breaker (RTB) test shall include separate verification of the undervoltage and shunt trip mechanisms. Each RTB in a division shall be tested separately in order to minimize the possibility of an inadvertent trip. Both breakers in a single division are tested during each STAGGERED TEST.

The Frequency of every 92 days on a STAGGERED TEST BASIS is adequate based on industry operating experience, considering instrument reliability and operating history data. In addition, the design provides additional breakers to enhance reliability.

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”

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B 3.3 INSTRUMENTATION

B 3.3.8 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the ESFAS, as well as specifying LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as “Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded.” The Safety Analysis Limit (SAL) is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that an SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Safety Analysis Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur. The LSSS values are identified and maintained in the Setpoint Program (SP) and are controlled by 10.CFR.50.59.

The Nominal Trip Setpoint (NTS) specified in the SP is a predetermined field setting for a protection channel chosen to initiate automatic actuation prior to the process variable reaching the Safety Analysis Limit and, thus, ensuring that the SL is not exceeded. As such, the NTS accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTS ensures that the SL is not exceeded. Therefore, the NTS meets the 10 CFR 50.36 definition of an LSSS.

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as “...being capable of performing its safety functions(s).” Relying solely on the NTS to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the “as-found” value of a

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BACKGROUND (continued)

protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule that are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTS due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTS, and thus, the automatic protective action would still have ensured that the SL would not be exceeded with the “as-found” setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function. If the as-found condition of the channel is near the as-found tolerance, recalibration is considered appropriate to allow for drift during the next surveillance interval.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB),

2. Fuel centerline melt shall not occur, and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, “Safety Limits (SLs),” also maintains the above values and ensures that offsite dose will be within the acceptance criteria during AOOs.

Design Basis Accidents (DBAs) are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of the limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

Protection and Safety Monitoring System (PMS)

The Protection and Safety Monitoring System contains the necessary equipment to (Ref. 8):

- Permit acquisition and analysis of the sensor inputs, including plant process sensors and nuclear instrumentation, required for reactor trip and ESF calculations;

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BASES

BACKGROUND (continued)

- Perform computation or logic operations on variables based on these inputs;
- Provide trip signals to the reactor trip switchgear and ESF actuation data to the ESF coincidence logic as required;
- Permit manual trip or bypass of each individual reactor trip Function and permit manual actuation or bypass of each individual voted ESF Function;
- Provide data to other systems in the Instrumentation and Control (I&C) architecture;
- Provide additional redundancy (within a single division) for the reactor trips and ESF actuations; and
- Provide isolation circuitry for control Functions that require input from sensors that are also required for protection Functions.

PMS Overview

The four divisions of PMS each consist of the following:

- Field Transmitters and Sensors (non-redundant)
- Nuclear Instrumentation System (NIS) - 1 per division (non-redundant)
- Bistable Processor Logic (BPL) System - 2 BPL subsystems per division (redundant)
- ESF Coincidence Logic (i.e., Local Coincidence Logic (LCL)) - 2 LCL subsystems per division (redundant)
- ESF Actuation Logic – 2 subsystems with Integrated Logic Processors (ILP) and Safety Remote Node Controller (SRNC) (redundant)
    - Division A - 6 ILPs
    - Division B - 8 ILPs
    - Division C - 4 ILPs
    - Division D - 8 ILPs
- Component Interface Modules (CIM) (non-redundant)

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BACKGROUND (continued)

The vital buses that power each PMS division are backed up by the battery bank in the same division. See LCO 3.8.1, “DC Sources – Operating” for further discussion.

PMS equipment duplication within a division (redundancy) is generally provided for active equipment such as processors and communication hardware. Where provided, redundant sets of equipment within a division are not independent and can be subject to common cause failures. This redundancy is provided to increase plant availability and facilitate surveillance testing. A division or channel is OPERABLE if it is capable of performing its specified safety function(s) and all the required supporting functions or systems are also capable of performing their related support functions.

Within each division, the sensor signals and setpoint bistables are processed in the BPL. The results of the bistable logic (partial ESF actuation signals) are communicated to the LCL subsystems in all divisions. The LCL subsystems perform the divisional ESF Coincidence Logic voting (e.g., two-out-of-four (2oo4) logic) and other associated logic. In some cases interlock logic is applied to each partial ESF actuation prior to performing the ESF Coincidence Logic voting (“pre-vote” interlock); in other cases an interlock is applied after the two-out-of-four logic (“post-vote” interlock).

ESF system-level actuation outputs from the LCL are inputs to the ESF Actuation Logic performed by redundant ILPs and SRNCs (which comprise the ESF Actuation Logic subsystems), and non-redundant CIMs. The ILPs decode the system commands, transmit the commands via the SRNCs, and actuate the final equipment through the CIM interlocking logic specific to each component (Ref. 8).

The CIMs provide actuation signals to each of the actuated devices. While the PMS boundary ends at the output terminals of the CIMs, individual CIM inoperability only impacts operability of the actuated device. Therefore, the CIMs and the actuated devices are outside the scope of this specification.

Engineered Safety Features (ESF) Channel

An ESF channel extends from the sensor to the output of the associated BPL subsystem and shall include the sensor (or sensors), the signal conditioning, the BPL subsystem, and associated datalinks to the LCL subsystems. For ESF channels containing nuclear instrumentation (e.g., boron dilution block, CVS isolation), the ESF channel also includes the signal processing and power supplies for the detectors provided by

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BACKGROUND (continued)

the NIS. In some cases, sensors provide inputs to more than one ESF function (e.g., sensors can provide a primary actuation signal as well as an interlock signal utilized in conjunction with other actuation signals). Thus, an inoperability of a shared sensor can affect the OPERABILITY of more than one ESF function.

PMS Division

The PMS consists of four redundant divisions, designated A, B, C, and D (with the exception of the Component Interface Modules). All safety system divisions are physically, functionally, and electrically separated from each other and from non-safety systems. Redundant divisions are provided to satisfy single failure criteria and improve plant availability. PMS divisions A and C actuate one train of ESF equipment and divisions B and D actuate the other ESF train.

Field Transmitters and Sensors

Normally, four redundant measurements using four separate sensors are made for each variable used for actuation of ESF. One measurement is processed by each division. The use of four channels for protection Functions is based on a minimum of two channels being required for an ESF actuation (two-out-of-four (2oo4) voting logic), with one channel in test or bypass, and a single failure on the remaining channel. Minimum requirements for protection are achieved only with three channels OPERABLE. The fourth channel is provided to increase plant availability, and permits the plant to run for an indefinite time with a single channel out of service. The circuit design is able to withstand both an input failure to the control system, which may then require the protection Function actuation, and a single failure in the other channels providing the protection Function actuation.

Analog measurements are converted to digital form by analog to digital (A/D) converters within the BPL subsystems. Signal conditioning may be applied to selected inputs following the conversion to digital form.

Except for selected dual element resistance temperature detectors (RTDs), the sensors within each division are non-redundant.

Nuclear Instrumentation System (NIS)

In each division the NIS cabinet provides signal processing and power supplies for the source range, intermediate range, and power range detectors. The analog and digital pulse signals generated are sent to the BPL for processing. Within each division, each transmitter, sensor, and NIS signal is input in parallel to both BPLs.

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BACKGROUND (continued)

Bistable Processor Logic (BPL) System

Each PMS division contains two identical and redundant BPL subsystems receiving the same input signals and performing the same calculations and logic. The BPL subsystems receive data from field sensors and manual inputs (such as system-level blocks and resets) from the MCR to perform the protective function calculations.

Within each division, the input from each sensor is provided in parallel to each of the two BPL subsystems' analog or digital input modules. Analog transmitter and sensor measurements are converted from analog to digital form by analog to digital input modules within the BPL subsystems. Signal conditioning may be applied to selected inputs following the conversion to digital form. The subsystems provide signal outputs for MCR indications and via the Plant Control System (PLS) for the non-safety related Remote Shutdown Workstation indications.

Following necessary calculations, processing, and logic, the measurements are compared against the applicable setpoints. The basis of the setpoints is described in References 2 and 6. The results of the required calculations and setpoint comparisons determine the partial actuation status for each ESF actuation function.

The partial ESF actuation signals are transmitted to all eight LCLs in the PMS for use in coincidence logic voting. The PMS uses datalinks to communicate the partial actuations and related status information calculated in the BPL subsystems to the LCL subsystems. These datalinks are used both locally within a division and externally across divisions. A partial actuation signal from either BPL subsystem in a division is interpreted by the LCL coincidence logic as a partial actuation signal.

During periodic tests and maintenance performed using the Maintenance and Test Panel and Interface and Test Processor, each BPL subsystem is tested individually. During testing, the LCL processors use the redundant BPL subsystem for actuation status. This permits the division being tested to remain OPERABLE while the testing is being conducted. In combination with manual tests required by Surveillance Requirements, the BPLs are tested via continuous system self-checking features.

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BASES

BACKGROUND (continued)

ESF Coincidence Logic Subsystem (i.e., Local Coincidence Logic (LCL))

Each PMS division contains two identical and redundant LCL subsystems receiving the same input signals and performing the same logic. The LCL subsystems perform the logic to combine the partial actuation signals from the BPL subsystems, along with certain permissives, blocks, and resets, and generate an actuation output signal to the ESF Actuation Logic subsystems.

Eight LCL subsystems are provided in the PMS architecture. Each of the LCL subsystems receives partial ESF actuation signals and status signals from each of the eight BPL subsystems. In normal operation, the LCL logic is programmed to assume that a partial actuation signal in either BPL subsystem in a given division is equivalent to a partial actuation signal.

In the event of a single BPL processor or associated datalink failure in a division is detected by diagnostics, the LCL logic will reject the input from the failed component and use the input from the other BPL subsystem from the affected division as the source of actuation information. This allows the function logic to remain in a 2oo4 logic configuration.

LCL processors will only permit bypass of both BPL inputs of a given function from one division (such as due to a failed sensor). In the event of a division bypass, the LCL reverts to 2oo3 logic in the affected function.

The LCL subsystems act to initiate an ESF actuation when the required number of divisions reaches a partial actuation state (e.g., 2oo4, 1oo2).

The LCL also provides for the bypass of actuation functions to accommodate periodic tests and maintenance. During LCL subsystem testing, each LCL subsystem is tested individually. During testing, the redundant LCL subsystem is available to provide output signals to the ESF Actuation Logic subsystems. For the CMT Actuation function only, both LCL processors in each division must be capable of functioning to maintain a single failure proof design, because the same two divisions of PMS are required to actuate both outlet valves for a given CMT. Therefore, if one of the two redundant PMS LCLs in a division is taken out of service or has no output or a bad quality signal, the CMT outlet valve component interface modules (CIMs) in the effected division must be locally placed in an actuate condition. This ensures CMT actuation will occur on a valid actuation signal in the event of a subsequent single failure of an LCL. This permits the division being tested to remain OPERABLE while the testing is being conducted. In combination with manual tests required by Surveillance Requirements, the LCLs are tested via continuous system self-checking features.

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BASES

BACKGROUND (continued)

Within each division, datalinks transfer the ESF system-level actuations and related status information calculated in the LCL controllers to the ESF Actuation Logic subsystems. Actuation signals from both LCL subsystems are transmitted to the redundant ILPs in each ESF Actuation Logic subsystem.

MCR ESF Manual Controls

Each of the manual system-level actuations is implemented using switches wired to digital input modules included in the LCL. The switch inputs to the LCL produce the ESF system-level actuation signals that are communicated to the ESF Actuation Logic subsystems.

ESF Actuation Logic Subsystem

Within each division, ESF system-level actuation outputs from the ESF Coincident Logic are inputs to the ESF Actuation Logic for generation of logic commands to the actuated components. The ESF Actuation Logic consists of internally redundant ILPs and SRNCs. The ILPs decode the system commands, transmit the commands via the SRNCs, and actuate the final equipment through the CIM logic specific to each component.

Each of the two ILP processors receives inputs from both ESF Coincidence Logic processors (i.e., LCLs) in a division. The ILPs perform the component fan-out for each ESF system-level actuation command. As long as the outputs from both ESF Coincidence Logic processors (i.e., LCLs) agree that an actuation should occur (i.e., 2oo2 logic), the ILPs will generate actuation signals to the CIMs. For each Function, if one of the LCLs has no output signal, provides a bad quality signal, or is in test, a good quality signal input to the ILP processors from the other LCL is sufficient to maintain OPERABILITY. With a good quality signal from only one LCL, the ILP logic becomes 1oo1.

Component Interface Module (CIM)

Each CIM provides an actuation signal to its associated actuated device. Each CIM receives inputs from both processors in an ILP and produces component actuation signals if the input signals agree (i.e., 2oo2 logic). If one of the ILP processors or SRNCs has no output signal or provides a bad quality signal, a good quality signal input to the CIM from the other ILP processor and SRNC is sufficient to maintain OPERABILITY. With a good quality signal from only one ILP processor, the logic to the CIM becomes 1oo1.

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BACKGROUND (continued)

While the PMS boundary ends at the output terminals of the CIMs, individual CIM inoperability only impacts inoperability of the actuated device. Therefore, the CIMs and the actuated devices are outside the scope of this specification. In lieu of manual tests required by a surveillance requirement, the ILPs, SRNCs, and CIMs are tested via continuous system self-checking features. The output of the CIMs up to, but not including the actuated component, are tested via a combination of manual surveillance tests and continuous self-checking features.

ADS and IRWST Injection Blocking Device

The ADS and IRWST injection blocking device is a Class 1E module physically located within each of the PMS divisions. The blocking device is diverse from the PMS hardware and software that is used to create the automatic ADS and IRWST injection ESF actuation signals. There are no inter-divisional connections between the blocking devices nor is there any coincidence voting among the blocking devices. The ADS and IRWST injection blocking device uses core makeup tank (CMT) level. The ADS and IRWST injection block in each division receives an input from a level sensor on each CMT that removes the block if the water level in either CMT is below a predetermined setpoint. Additionally, one switch for each division is provided in the Main Control Room (MCR) to allow the operators to manually clear the ADS and IRWST blocks.

The ADS and IRWST injection blocking device design uses conventional analog components that do not rely on software. The ADS and IRWST injection blocking device outputs provide CIM inputs for ADS stage 1, 2, and 3 MOVs, and the ADS Stage 4 and IRWST injection squib valves. The ADS and IRWST injection blocking device outputs block any attempt to open the ADS and IRWST injection valves from the PMS Integrated Logic Processors.

Nominal Trip Setpoints (NTSs)

The NTS is the nominal value at which the trip output is set. Any trip output is considered to be properly adjusted when the “as-left” value is within the band for CHANNEL CALIBRATION, i.e., ± rack calibration accuracy.

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BASES

BACKGROUND (continued)

The trip setpoints used in the trip output are based on the Safety Analysis Limits stated in Reference 2. The determination of these NTSs is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 4), the NTSs specified in the SP are conservative with respect to the Safety Analysis Limits. A detailed description of the methodology used to calculate the NTSs, including their explicit uncertainties, is provided in the "Westinghouse Setpoint Methodology for Protection Systems" (Ref. 6). The as-left tolerance and as-found tolerance band methodology is provided in the SP.

The NTSs listed in the SP are based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each NTS. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes. Transmitter and signal processing equipment calibration tolerances and drift allowances must be specified in plant calibration procedures, and must be consistent with the values used in the setpoint methodology.

The OPERABILITY of each transmitter or sensor can be evaluated when its "as-found" calibration data are compared against the "as-left" data and are shown to be within the setpoint methodology assumptions. The basis of the setpoints is described in References 2 and 6. Trending of calibration results is required by the program description in Technical Specification 5.5.14.d.

Note that the as-left and as-found tolerances listed in the SP define the OPERABILITY limits for a channel during a periodic CHANNEL CALIBRATION or a TRIP ACTUATING DEVICE OPERATIONAL TEST. Trip setpoints are continuously and automatically verified by PMS self-checking features between performances of CHANNEL CALIBRATIONS. Before unit startup, the CHANNEL CALIBRATION verifies that the trip setpoint values in the Maintenance and Test Panel (MTP) match the SP specified values.

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BASES

BACKGROUND (continued)

The protection and safety monitoring system testing features are designed to allow for complete testing by using a combination of system self-checking and manual tests. Successful testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded. For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Since software does not degrade, software testing involves verifying that the software code has not changed and that the software code is executing. To the extent possible, protection and safety monitoring system testing will be accomplished with continuous system self-checking features in lieu of manual surveillance tests. As a result, some functions do not have manual surveillance requirements.

The protection and safety monitoring system incorporates continuous system self-checking features wherever practical. Self-checking features include on-line diagnostics for the computer system and the hardware and communications tests. Faults detected by the self-checking features are alarmed in the main control room. These self-checking tests do not interfere with normal system operation.

Manual tests are included for those parts of the system which are not tested with self-checking features. This includes manual functional checks, calibration verification, response time testing, and component testing.

All of the tests are designed so that the duration of the testing is as short as possible. The manual tests are designed so that the actual logic is not modified. To prevent unwanted actuation, the tests are designed with either the capability to bypass a Function during testing and/or limit the number of signals allowed to be placed in test at one time.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure – Low 3 is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation not specifically credited in the accident safety analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These Functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate Function performance. These

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 2).

Permissive and interlock functions are based upon the associated protection function instrumentation. Because they do not have to operate in adverse environmental conditions, the trip settings of the permissive and interlock functions use the normal environment, steady-state instrument uncertainties of the associated protection function instrumentation. This results in OPERABILITY criteria (i.e., as-found tolerance and as-left tolerance) that are the same as the associated protection function sensor and process rack modules. The NTSs for permissives and interlocks are based on the associated protection function OPERABILITY requirements; i.e., permissives and interlocks performing enabling functions must be set to occur prior to the specified trip setting of the associated protection function. Interlocks can be applied to individual channels within each division “pre-vote” (i.e., affecting OPERABILITY of the ESF instrument channel) or in some cases applied to the “post-vote” actuation signal (i.e., affecting OPERABILITY of the ESF Actuation Logic Function).

The LCO requires all instrumentation performing an ESFAS Function, listed in Table 3.3.8-1 in the accompanying LCO, to be OPERABLE. The as-left and as-found tolerances specified in the SP define the OPERABILITY limits for a channel during the CHANNEL CALIBRATION. As such, the as-left and as-found tolerances differ from the NTS by plus or minus the PMS rack calibration accuracy and envelope the expected calibration accuracy and drift. In this manner, the actual setting of the channel (NTS) prevents exceeding an SL at any given point in time as long as the channel has not drifted beyond the expected tolerances during the surveillance interval. Note that the as-left and as-found recorded values must be confirmed to be within the assumptions of the statistical uncertainty calculations.

If the actual setting of the channel is found outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance) and evaluating the channel’s response. If the channel is functioning as required and expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

A trip setpoint may be set more conservative than the NTS as necessary in response to plant conditions. However, in this case, the OPERABILITY of this instrument must be verified based on the actual field setting and not the NTS. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

ESFAS Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlocks backup manual actions to ensure bypassable Functions are in operation under the conditions assumed in the safety analyses. Proper operation of these interlocks supports OPERABILITY of the associated ESF Instrumentation channel OPERABILITY (i.e., of the “pre-vote” interlocks) and/or the requirement for ESF Actuation Logic OPERABILITY (i.e., of the “post-vote” interlocks). Interlocks must be in the required state, as appropriate, to support OPERABILITY of ESFAS.

Reactor Trip Breaker Open, P-3

The P-3 interlock is provided to permit the block of automatic Safeguards Actuation after a predetermined time interval following automatic Safeguards Actuation (this interlock is applied “post-vote”).

The reactor trip breaker position switches that provide input to the P-3 interlock only function to energize or de-energize (open or close) contacts. Therefore, this interlock does not have an adjustable trip setpoint.

Reactor Trip, P-4

There are eight reactor trip breakers with two breakers in each division. The P-4 interlock is enabled when the breakers in two-out-of-four divisions are open. Additionally, the P-4 interlock is enabled by all Automatic Reactor Trip Actuations (this interlock is applied “post-vote”). The Functions of the P-4 interlock are:

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

- Trip the main turbine;
- Block boron dilution;
- Isolate main feedwater control valves coincident with Low reactor coolant average temperature (This function is not assumed in safety analysis);
- Isolate main feedwater isolation valves coincident with Low 2 reactor coolant average temperature (This function is not assumed in the safety analysis); and
- Isolate startup feedwater and CVS makeup coincident with high narrow range water level in either steam generator.

The reactor trip breaker position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this interlock has no adjustable trip setpoint.

Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when the respective NIS intermediate range channel increases to approximately one decade above the channel lower range limit (this interlock is applied “pre-vote”). Above the setpoint, the P-6 interlock allows manual block of the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions, and deenergization of the source range neutron flux detectors. As intermediate range flux decreases from above the setpoint to below the setpoint, the P-6 interlock automatically resets the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions, and automatically energizes the source range neutron flux detectors. Normally, the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions is blocked by the main control room operator during reactor startup.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Reactor Coolant Average Temperature (T_{avg}), P-8

The P-8 interlock is used to permit manual blocking, and for the resetting of a manual block, of the automatic Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation (this interlock is applied “pre-vote”). The automatic Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions may be manually blocked (disabled) to permit plant startup and normal power operation when above the P-8 reactor coolant average temperature setpoint. The manual block to disable the automatic Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions is automatically reset upon decreasing reactor coolant average temperature to below the P-8 setpoint.

Once reactor coolant average temperature is below the P-8 setpoint and after the operator isolates unborated water source flow paths, the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions may also be manually blocked to prevent inadvertent actuation during refueling operations and post-refueling control rod testing.

When the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions is manually blocked below the P-8 setpoint, an automatic close signal is also sent to the CVS demineralized water system isolation valves to prevent inadvertent boron dilution.

Reactor Coolant Average Temperature (T_{avg}), P-9

The P-9 interlock permits routine maintenance without core makeup tank actuation, reactor coolant pump trip, PRHR heat exchanger actuation, and SG blowdown isolation. The P-9 interlock also serves to permit the steam generators to be flooded for wet-layup in MODES 5 and 6 without automatic startup feedwater isolation, CVS isolation, main feedwater

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

isolation, or actuation of turbine trip signal (this interlock is applied "pre-vote"). With T_{avg} channels less than the P-9 setpoint, the operator can manually block the Steam Generator Wide Range Water Level – Low 2 coincident with Hot Leg Temperature High, Steam Generator Wide Range Water Level – Low 2 and Steam Generator Narrow Range Water Level – Low 2 signals. Utilizing a separate block control with RCS T_{avg} below the P-9 setpoint, the operator can also manually block startup feedwater isolation and CVS isolation on High narrow range steam generator water level coincident with reactor trip (P-4), as well as startup feedwater isolation, main feedwater isolation, CVS isolation, and turbine trip on High-3 narrow range steam generator water level. These functions are automatically enabled above P-9. With T_{avg} channels greater than P-9 setpoint, the Steam Generator Wide Range Water Level – Low 2 coincident with Hot Leg Temperature High, Steam Generator Wide Range Water Level – Low 2 and Steam Generator Narrow Range Water Level – Low 2 signals are automatically enabled.

The P-9 interlock also permits the draining of the IRWST to support refueling operations (this interlock is applied "post-vote" in Divisions A and B). With T_{avg} channels less than the P-9 setpoint, the operator can manually block the IRWST Wide Range Level - Low isolation signal. With T_{avg} channels greater than P-9 setpoint, the IRWST Wide Range Level - Low isolation signal is automatically enabled. This block is necessary to permit IRWST volume to be transferred to and from the refueling cavity during MODE 6 with the refueling cavity and spent fuel pool volumes not in communication.

Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power as determined by the respective power-range detector (this interlock is applied "pre-vote"). Below the setpoint, the P-10 interlock allows manual energization of the source range neutron flux detectors and reset of the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions. Above the setpoint, the P-10 interlock automatically blocks the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation, and de-energizes the source range neutron flux detectors. Normally, the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block and CVS isolation Functions is blocked by the main control room operator during reactor startup.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without Safeguards Actuation or main steam line and feedwater isolation. With pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Pressurizer pressure – Low 3, Steam Line Pressure – Low 2, and T_{cold} – Low 2 Safeguards Actuation signals and the Steam Line Pressure – Low 2 and T_{cold} – Low 2 steam line isolation signals (this interlock is applied "pre-vote"). When the Steam Line Pressure – Low 2 is manually blocked, a main steam isolation signal on Steam Line Pressure-Negative Rate – High is enabled. This provides protection for an SLB by closure of the main steam isolation valves. Manual block of feedwater isolation on T_{avg} – Low, Low 2, and T_{cold} – Low 2 is also permitted below P-11. With pressurizer pressure channels ≥ P-11 setpoint, the Pressurizer Pressure – Low 3, Steam Line Pressure – Low 2, and T_{cold} – Low 2 Safeguards Actuation signals and the Steam Line Pressure Low 2 and T_{cold} – Low 2 steam line isolation signals are automatically enabled. The feedwater isolation signals on T_{cold} – Low 2, T_{avg} – Low and Low 2 are also automatically enabled above P-11. The operator can also enable these signals by use of the respective manual reset buttons. When the Steam Line Pressure – Low 2 and T_{cold} – Low 2 steam line isolation signals are enabled, the main steam isolation on Steam Line Pressure-Negative Rate – High is disabled.

RNS isolation provided by Containment Radioactivity – High 2 channels (this interlock is applied "post-vote" in Divisions A, B, and D) and Safeguards Actuation (this interlock is applied "pre-vote") are automatically unblocked above the P-11 interlock, with manual block permitted below P-11. The Setpoint reflects only steady state instrument uncertainties.

Pressurizer Level, P-12

The P-12 interlock is provided to permit midloop operation without core makeup tank actuation, reactor coolant pump trip, CVS letdown isolation, or purification line isolation. With pressurizer level channels less than the P-12 setpoint, the operator can manually block Pressurizer Water Level – Low and Pressurizer Water Level – Low 2 signals used for these actuations (this interlock is applied "pre-vote"). Concurrent with blocking CMT actuation on Pressurizer Water Level – Low 2, ADS 4^{th} Stage actuation on Low 4 RCS hot leg level is enabled (this interlock is applied "pre-vote"). Also CVS letdown isolation on Low 2 RCS hot leg level is enabled (this interlock is applied "post-vote" in Divisions A and D). When the pressurizer level is above the P-12 setpoint, the Pressurizer Water Level – Low 2 signal is automatically enabled (this interlock is applied "pre-vote").

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

RCS Pressure, P-19

The P-19 interlock is provided to permit water solid conditions (i.e., when the pressurizer water level is > 92%) in lower MODES without automatic isolation of the CVS makeup pumps (this interlock is applied “pre-vote”). With RCS pressure below the P-19 setpoint, the PMS allows the operator to manually block CVS isolation on High 2 pressurizer water level, and block Passive RHR actuation and Pressurizer Heater Trip on High 3 pressurizer water level. The operator can manually block these functions once they are no longer required to be OPERABLE when at least one RCS cold leg temperature is ≤ 275°F. When RCS pressure is above the P-19 setpoint, these Functions are automatically unblocked.

The LCO generally requires OPERABILITY of four channels in each instrumentation/logic Function and two devices for each manual initiation Function. The two-out-of-four configurations allow one channel to be bypassed during maintenance or testing without causing an ESFAS initiation. Two manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS protective functions are as follows:

Safeguards Actuation

The Safeguards Actuation signal actuates the alignment of the Core Makeup Tank (CMT) valves for passive injection to the RCS. The Safeguards Actuation signal provides two primary Functions:

- Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal and clad integrity, peak clad temperature < 2200°F); and

- Boration to ensure recovery and maintenance of SHUTDOWN MARGIN (k_{eff} < 1.0).

These Functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The Safeguards Actuation signal is also used to initiate other Functions such as:

- Containment Isolation;

- Reactor Trip;

- Close Main Feedwater Control Valves;

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

- Trip Main Feedwater Pumps and Closure of Isolation and Crossover Valves; and
- Reactor Coolant Pump Trip.

These other Functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the turbine and reactor to limit power generation;
- Isolation of main feedwater to limit secondary side mass losses;
- Trip of the reactor coolant pumps to ensure proper CMT actuation; and
- Enabling automatic depressurization of the RCS on CMT Level – Low 3 to ensure continued safeguards actuated injection.

Safeguards Actuation is initiated by the following signals:

- Containment Pressure – High 2;
- Pressurizer Pressure – Low 3;
- RCS Cold Leg Temperature (T_{cold}) – Low 2;
- Steam Line Pressure – Low 2; and
- Safeguards Actuation – Manual Initiation.

Core Makeup Tank (CMT) Actuation

CMT Actuation provides the passive injection of borated water into the RCS. Injection provides RCS makeup water and boration during transients or accidents when the normal makeup supply from the Chemical and Volume Control System (CVS) is lost or insufficient. Two tanks are available to provide passive injection of borated water. CMT injection mitigates the effects of high energy line breaks by adding primary side water to ensure maintenance or recovery of reactor vessel water level following a LOCA, and by borating to ensure recovery or maintenance of SHUTDOWN MARGIN following a steam line break.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

CMT Actuation is initiated by the following signals:

- Safeguards Actuation;
- Pressurizer Water Level – Low 2;
- ADS Stages 1, 2, and 3 Actuation; and
- CMT Injection Actuation – Manual Initiation.

Containment Vacuum Relief Valve Actuation

The purpose of the vacuum relief lines is to protect the containment vessel against damage due to a negative pressure (i.e., a lower pressure inside than outside). Containment Vacuum Relief Valve Actuation is actuated by the following signals:

- Containment Pressure – Low 2; and
- Containment Vacuum Relief Valve Actuation – Manual Initiation coincident with Containment Pressure – Low.

Containment Isolation (T signal)

Containment Isolation provides isolation of the containment atmosphere and selected process systems which penetrate containment from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

Containment Isolation is actuated by the following signals:

- Safeguards Actuation;
- Passive Containment Cooling Actuation – Manual Initiation; and
- Containment Isolation – Manual Initiation.

Containment Air Filtration System Isolation

Some DBAs such as a LOCA may release radioactivity into the containment where the potential would exist for the radioactivity to be released to the atmosphere and exceed the acceptable site dose limits. Isolation of the Containment Air Filtration System provides protection to prevent radioactivity inside containment from being released to the atmosphere.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Containment Air Filtration System Isolation is actuated on the following signals:

- Containment Radioactivity – High; and
- Containment Isolation Actuation.

Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one steam generator (SG) at most. For an SLB upstream of the isolation valves, inside or outside of containment, closure of the isolation valves limits the accident to the blowdown from only the affected SG. For a SLB downstream of the isolation valves, closure of the isolation valves terminates the accident as soon as the steam lines depressurize.

Closure of the main steam branch isolation valves is initiated by this Function. Closure of these valves limits the accidental depressurization of the main steam system associated with an inadvertent opening of a single steam dump, relief, safety valve, or a rupture of a main steam line. Closure of these valves also supports a steam generator tube rupture event by isolating the faulted steam generator.

Steam Line Isolation is actuated by the following signals:

- Containment Pressure – High 2;
- RCS Cold Leg Temperature (T_{cold}) – Low 2;
- Steam Line Pressure – Low 2;
- Steam Line Pressure – Negative Rate – High; and
- Steam Line Isolation – Manual Initiation.

SG Power Operated Relief Valve and Block Valve Isolation

The Function of the SG Power Operated Relief Valve and Block Valve Isolation is to ensure that the SG PORV flow paths can be isolated during a SG tube rupture (SGTR) event. The PORV flow paths must be isolated following a SGTR to minimize radiological releases from the ruptured steam generator into the atmosphere. The PORV flow path is assumed

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

to open due to high secondary side pressure, during the SGTR. Dose analyses take credit for subsequent isolation of the PORV flow path by the PORV and/or the block valve which receive a close signal on Low 2 steam line pressure. Additionally, the PORV flow path can be isolated manually.

SG Power Operated Relief Valve and Block Valve Isolation is actuated by the following signals:

- Steam Line Pressure – Low 2; and
- SG Power Operated Relief Valve and Block Valve Isolation – Manual Initiation.

Steam Generator Blowdown Isolation

The primary Function of the steam generator blowdown isolation is to preserve water inventory in the steam generators to support removing the excess heat being generated until the decay heat has decreased to within the PRHR HX capability.

Steam Generator Blowdown Isolation is actuated on the following signals:

- PRHR HX Actuation;
- SG Narrow Range Water Level – Low 2; and
- Containment Isolation Actuation.

Turbine Trip

The primary Function of the Turbine Trip is to prevent damage to the turbine due to water in the steam lines. Closure of the turbine stop and control valves is initiated by this Function. This Function is necessary in MODES 1 and 2, and 3 above P-11 to mitigate the effects of a large SLB or a large feedwater line break (FLB). Failure to trip the turbine following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Turbine Trip is actuated by the following signals:

- SG Narrow Range Water Level – High 3;
- Reactor Trip Signal (P-4); and
- Feedwater Isolation – Manual Initiation.

Main Feedwater Control Valve Isolation

The primary Function of Main Feedwater Control Valve Isolation is to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs.

Main Feedwater Control Valve Isolation is actuated by the following signals:

- SG Narrow Range Water Level – High 3;
- Safeguards Actuation;
- Reactor Coolant Average Temperature (T_{avg}) – Low coincident with Reactor Trip Signal (P-4); and
- Main Feedwater Control Valve Isolation – Manual Initiation.

Main Feedwater Pump Trip and Valve Isolation

The primary function of the Main Feedwater Pump Trip and Isolation is to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs. Valve isolation includes closing the main feedwater isolation and crossover valves. Isolation of main feedwater is necessary to prevent an increase in heat removal from the reactor coolant system in the event of a feedwater system malfunction. Addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature.

Main Feedwater Pump Trip and Valve Isolation is actuated by the following signals:

- SG Narrow Range Water Level – High 3;
- Safeguards Actuation;

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- Reactor Coolant Average Temperature (T_{avg}) – Low 2 coincident with Reactor Trip Signal (P-4); and
- Main Feedwater Control Valve Isolation – Manual Initiation.

Startup Feedwater Isolation

The primary Function of the Startup Feedwater Isolation is to stop the excessive flow of feedwater into the SGs. This Function is necessary in MODES 1, 2, 3, and 4 to mitigate the effects of a large SLB or a large FLB. Failure to isolate the startup feedwater system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment.

Startup Feedwater Isolation is actuated by the following signals:

- SG Narrow Range Water Level – High 3;
- RCS Cold Leg Temperature (T_{cold}) – Low 2;
- SG Narrow Range Water Level – High coincident with Reactor Trip Signal (P-4); and
- Main Feedwater Control Valve Isolation – Manual Initiation.

ADS Stages 1, 2, & 3 Actuation

The Automatic Depressurization System (ADS) provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the in-containment refueling water storage tank (IRWST) to mitigate the effects of a LOCA. The depressurization is accomplished in four stages, with the first three stages discharging into the IRWST and the last stage discharging into containment. Each of the first three stages consists of two parallel paths with each path containing an isolation valve and a depressurization valve.

The first stage isolation valves open on any ADS Stages 1, 2, and 3 actuation. The first stage depressurization valves are opened following a preset time delay after the actuation of the isolation valves. The second stage isolation valves are opened following a preset time delay after actuation of the first stage depressurization valves open. The second stage depressurization valves are opened following a preset time delay after the second stage isolation valves are actuated, similar to stage one. Similar to the second stage, the third stage isolation valves are opened

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following a preset time delay after the actuation of the second stage depressurization valves. The third stage depressurization valves are opened following a preset time delay after the third stage isolation valves are actuated.

ADS Stages 1, 2, & 3 is actuated on the following signals:

- CMT Level – Low 3 coincident with CMT Actuation; and
- ADS Stages 1, 2 & 3 Actuation – Manual Initiation.

ADS Stage 4 Actuation

The ADS provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the IRWST to mitigate the effects of a LOCA. The depressurization is accomplished in four stages, with the first three stages discharging into the IRWST and the fourth stage discharging into containment.

The fourth stage of the ADS consists of four parallel paths. Each of these paths consists of a normally open isolation valve and a depressurization valve. The four paths are divided into two groups with two paths in each group. Within each group, one path is designated to be substage A and the second path is designated to be substage B.

The substage A depressurization valves are opened upon receipt of a system level ADS Stage 4 actuation signal. The substage B depressurization valves are opened following a preset time delay after the substage A depressurization valves receive their open signal.

ADS Stage 4 is actuated on the following signals:

- CMT Level – Low 6 coincident with both ADS Stage 1, 2, & 3 Actuation and RCS Wide Range Pressure – Low;
- Hot Leg Loop 1 Level Low 4 coincident with Hot Leg Loop 2 Level Low 4;
- ADS Stage 4 Actuation – Manual Initiation coincident with ADS Stages 1, 2, & 3 Actuation; and
- ADS Stage 4 Actuation – Manual Initiation coincident with RCS Wide Range Pressure – Low.

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Reactor Coolant Pump Trip

Reactor Coolant Pump (RCP) Trip allows the passive injection of borated water into the RCS. Injection provides RCS makeup water and boration during transients or accidents when the normal makeup supply from the CVS is lost or insufficient. Two tanks provide passive injection of borated water by gravity when the reactor coolant pumps are tripped. CMT injection mitigates the effects of high energy line breaks by adding primary side water to ensure maintenance or recovery of reactor vessel water level following a LOCA, and by borating to ensure recovery or maintenance of SHUTDOWN MARGIN following a steam line break. RCP trip on High 2 bearing water temperature protects the RCP coast down.

RCP trip is actuated on the following signals:

- Safeguards Actuation;
- ADS Stages 1, 2, & 3 Actuation;
- Reactor Coolant Pump Bearing Water Temperature – High 2;
- Pressurizer Water Level – Low 2; and
- CMT Injection Actuation – Manual Initiation.

Component Cooling Water System Containment Isolation Valve Closure

The function of the Component Cooling Water System (CCS) containment isolation valve closure is to ensure that the CCS flow paths can be isolated during an RCP heat exchanger tube rupture event. The CCS flow paths must be isolated following an RCP heat exchanger tube rupture event to minimize radiological releases from the ruptured tube into the turbine building. CCS Isolation Valve Closure is actuated by Reactor Coolant Pump Bearing Water Temperature – High 2 and Safeguards Actuation.

Passive Containment Cooling Actuation

The Passive Containment Cooling System (PCS) transfers heat from the reactor containment to the environment. This Function is necessary to prevent the containment design pressure and temperature from being exceeded following any postulated DBA (such as LOCA or SLB).

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A Passive Containment Cooling Actuation signal initiates water flow by gravity by opening the isolation valves. The water flows onto the containment dome, wetting the outer surface. The path for natural circulation of air along the outside walls of the containment structure is always open.

Passive Containment Cooling is actuated on the following signals:

- Containment Pressure – High 2; and
- Passive Containment Cooling Actuation – Manual Initiation.

Passive Residual Heat Removal (PRHR) Heat Exchanger Actuation

The PRHR Heat Exchanger (HX) provides emergency core decay heat removal when the Startup Feedwater System is not available to provide a heat sink.

PRHR is actuated on the following signals:

- SG Narrow Range Water Level – Low 2 coincident with Startup Feedwater Flow – Low 2 coincident with T_{cold} - High;
- SG Wide Range Water Level – Low 2;
- ADS Stages 1, 2, & 3 Actuation;
- CMT Actuation;
- Pressurizer Water Level – High 3; and
- PRHR Heat Exchanger Actuation – Manual Initiation.

Boron Dilution Block

The block of boron dilution is accomplished by closing the CVS makeup line isolation valves or closing the demineralized water system isolation valves to the CVS.

Boron Dilution Block is actuated on the following signals:

- Source Range Neutron Flux Doubling; and
- Reactor Trip Signal (P-4).

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Chemical and Volume Control System Makeup Isolation

The CVS makeup line is isolated following certain events to prevent overfilling of the RCS. In addition, this line is isolated on High 2 containment radioactivity to provide containment isolation following an accident. This line is not isolated on a containment isolation signal, to allow the CVS makeup pumps to perform their defense-in-depth functions. However, if very high containment radioactivity exists (above the High 2 setpoint) this line is isolated.

Chemical and Volume Control System Makeup Line Isolation is actuated on the following signals:

- Containment Radioactivity – High 2;
- Pressurizer Water Level – High 2;
- Pressurizer Water Level – High coincident with unlatched Safeguards Actuation;
- Source Range Neutron Flux Doubling;
- SG Narrow Range Water Level – High 3;
- SG Narrow Range Water Level – High coincident with Reactor Trip Signal (P-4); and
- Chemical and Volume Control System Makeup Isolation – Manual Initiation.

Chemical and Volume Control System Letdown Isolation

The CVS provides letdown to the liquid radwaste system to maintain the pressurizer level. To help maintain RCS inventory in the event of a LOCA, the CVS Letdown Isolation is actuated on Hot Leg Level – Low 2.

Auxiliary Spray and Purification Line and Zinc/Hydrogen Addition Isolation

The CVS maintains the RCS fluid purity and activity level within acceptable limits. The CVS purification line receives flow from the discharge of the RCPs. The CVS also provides auxiliary spray to the pressurizer. To preserve the reactor coolant inventory in the event of a break in the CVS loop piping, the auxiliary spray line, purification line, zinc addition line, and hydrogen addition line are isolated to help maintain reactor coolant system inventory.

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Auxiliary Spray and Purification Line and Zinc/Hydrogen Addition Isolation is actuated on the following signals:

- Pressurizer Water Level – Low; and
- Chemical and Volume Control System Makeup Isolation – Manual Initiation.

Additionally, Zinc/Hydrogen Addition Isolation is actuated by the Containment Isolation actuation signal.

Pressurizer Heater Trip

Pressurizer heaters are automatically tripped to reduce the potential for steam generator overfill and automatic ADS Stages 1, 2, and 3 actuation for a steam generator tube rupture event. Automatically tripping the pressurizer heaters reduces the pressurizer level swell for certain non-LOCA events such as loss of normal feedwater, inadvertent CMT operation, and CVS malfunction resulting in an increase in RCS inventory. For small break LOCA analysis, tripping the pressurizer heaters supports depressurization of the RCS following actuation of the CMTs.

Pressurizer Heater Trip is actuated on the following signals:

- CMT Actuation; and
- Pressurizer Water Level – High 3.

Normal Residual Heat Removal System (RNS) Isolation

The RNS suction line is isolated by closing the containment isolation valves on High 2 containment radioactivity to provide containment isolation following an accident. This line is isolated on a safeguards actuation signal. However, the valves may be reset to permit the RNS pumps to perform their defense-in-depth functions post-accident. Should a high containment radiation signal (above the High 2 setpoint) develop following the containment isolation signal, the RNS valves would re-close. A high containment radiation signal is indicative of a high RCS source term and the valves would re-close to assure offsite doses do not exceed regulatory limits.

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RNS Isolation is actuated on the following signals:

- Containment Radioactivity – High 2;
- Safeguards Actuation; and
- Normal Residual Heat Removal System Isolation – Manual Initiation.

IRWST Injection Line Valve Actuation

The PXS provides core cooling by gravity injection and recirculation for decay heat removal following an accident. The IRWST has two injection flow paths. Each injection path includes a normally open motor operated isolation valve and two parallel lines, each isolated by one check valve and one squib valve in series.

IRWST Injection Line Valve Actuation is actuated on the following signals:

- ADS Stage 4 Actuation; and
- IRWST Injection Line Valve Actuation – Manual Initiation.

IRWST Containment Recirculation Valve Actuation

The PXS provides core cooling by gravity injection and recirculation for decay heat removal following an accident. The PXS has two containment recirculation flow paths. Each path contains two parallel flow paths, one path is isolated by a motor operated valve in series with a squib valve and one path is isolated by a check valve in series with a squib valve.

IRWST Containment Recirculation Valve Actuation is actuated on the following signals:

- ADS Stage 4 Actuation coincident with IRWST Lower Narrow Range Level – Low 3 level; and
- IRWST Containment Recirculation Valve Actuation – Manual Initiation.

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Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization

Isolation of the main control room and initiation of the VES air supply provides a breathable air supply for the operators following an uncontrolled release of radiation. De-energizing non-essential main control room electrical loads maintains the room temperature within habitable limits. Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization is actuated on any one of the following signals:

- Main Control Room Air Supply Iodine or Particulate Radiation – High 2;
- Main Control Room Differential Pressure – Low; and
- Class IE 24-hour Battery Charger Undervoltage.

The main control room air supply radiation monitor sample pumps are de-energized by PMS on Class 1E 24-hour battery charger undervoltage signal as shown on FSAR Figure 7.2-1, Sheet 13. The main control room air supply radiation monitor sample pumps are de-energized to conserve battery capacity, maintain room temperature below the equipment qualification limitation, and maintain the assumed heat load for Main Control Room habitability.

Refueling Cavity and Spent Fuel Pool Cooling System (SFS) Isolation

The SFS can be connected to the spent fuel pool, the fuel transfer canal, the refueling cavity, and the IRWST to clarify and purify the water. It can also connect the IRWST and refueling cavity to transfer water in preparation for refueling activities, and to return to normal operations from refueling activities. In the event of a leak in the nonsafety-related SFS, Refueling Cavity and SFS Isolation is actuated on the following signals:

- Spent Fuel Pool Level - Low 2; and
- IRWST Wide Range Level - Low.

ESF Coincidence Logic

LCO 3.3.15 and LCO 3.3.16 require four sets of ESF Coincidence Logic (“pre-vote” interlock logic is addressed by ESF instrument channel OPERABILITY), each set with one battery backed subsystem OPERABLE to support automatic actuation. The ESF coincidence logic provides the system level logic interfaces for the divisions.

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ESF Actuation Logic

LCO 3.3.15 and LCO 3.3.16 require that for each division of ESF Actuation Logic (including consideration of the “post-vote” interlock logic), one battery backed subsystem be OPERABLE to support both automatic and manual actuation. The ESF Actuation Logic subsystems provide the logic and power interfaces to the CIM for each actuated component.

ESF Instrument Functions

The following are descriptions of the individual instrument Functions required by this LCO as presented in Table 3.3.8-1.

1.a. Containment Pressure - Low

The Containment Pressure - Low provides an interlock to block opening of the Containment Vacuum Relief Valves above the Low setpoint to prevent inadvertent opening of these valves. The Containment Pressure - Low also provides an automatic reset of the Containment Vacuum Relief manual initiation function to ensure the Containment Vacuum Relief Valves are allowed to close when Containment Vacuum Relief function is no longer needed.

The OPERABILITY of the Containment Pressure - Low function is necessary for the manual actuation function for the Containment Vacuum Relief Valves. The Containment Pressure - Low actuation must be OPERABLE in MODES 1, 2, 3, and 4 and in MODES 5 and 6 when there is no open containment air flow path ≥ 6 inches in diameter. With a 6 inch diameter or equivalent containment air flow path, the vacuum relief function that the Containment Pressure – Low signal supports is not needed to prevent inadvertent opening of these valves.

1.b Containment Pressure – Low 2

This signal provides protection against a negative pressure in containment due to loss of ac power or inadvertent actuation of containment cooling and a low outside ambient air temperature in combination with limited containment heating that reduces the atmospheric temperature (and hence pressure) inside containment.

Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

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The Containment Vacuum Relief Valve Actuation ESFAS protective function is actuated by Containment Pressure – Low 2.

Automatic Containment Vacuum Relief Valve actuation must be OPERABLE in MODES 1 through 4 and in MODES 5 and 6 without an open containment air flow path ≥ 6 inches in diameter. With a 6 inch diameter or equivalent containment air flow path, the vacuum relief function is not needed to mitigate a low pressure event.

2. Containment Pressure – High 2

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

The ESFAS protective functions actuated by Containment Pressure – High 2 are:

- Safeguards Actuation;
- Steam Line Isolation; and
- Passive Containment Cooling Actuation.

The transmitters (d/p cells) and electronics are located outside of containment. Since the transmitters and electronics are located outside of containment, they will not experience adverse environmental conditions. The Containment Pressure – High 2 setpoint has been specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with the TMI action item (NUREG-0933, Item II.E.4.2) guidance.

The LCO requires four channels of Containment Pressure – High 2 to be OPERABLE in MODES 1, 2, 3, and 4. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODES 5 and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure – High 2 setpoint.

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3. Containment Radioactivity – High

This signal to isolate Containment Air Filtration System results from the coincidence of containment radioactivity above the High setpoint in any two of the four divisions.

The Containment Air Filtration System Isolation ESFAS protective function is actuated by Containment Radioactivity – High.

Four channels of Containment Radioactivity – High are required to be OPERABLE in MODES 1, 2, and 3, and MODE 4 with the RCS not being cooled by the RNS, when the potential exists for a LOCA, to protect against radioactivity inside containment being released to the atmosphere. This Function is not required to be OPERABLE in MODE 4 with the RCS being cooled by the RNS, because, if required, the Containment Air Filtration System is isolated by the Containment Isolation Actuation. This Function is not required to be OPERABLE in MODES 5 and 6, because any DBA release of radioactivity into the containment in these conditions would not require this containment isolation function.

4. Containment Radioactivity – High 2

This signal to isolate CVS makeup and to isolate the normal residual heat removal system results from the coincidence of containment radioactivity above the High 2 setpoint in any two of the four divisions.

The ESFAS protective functions actuated by Containment Radioactivity – High 2 are:

- Chemical and Volume Control System Makeup Isolation; and
- Normal Residual Heat Removal System Isolation.

Four channels of Containment Radioactivity – High 2 are required to be OPERABLE in MODES 1, 2, and 3 when the potential exists for a LOCA, to ensure that the radioactivity inside containment is not released to the atmosphere. This signal for RNS isolation may be manually blocked by the operator below the P-11 setpoint. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is no credible release of radioactivity into the containment in these MODES that would result in a High 2 actuation.

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5. Pressurizer Pressure – Low 3

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer safety valve;
- LOCAs; and
- Steam Generator Tube Rupture (SGTR).

The Safeguards Actuation ESFAS protective function is actuated by Pressurizer Pressure – Low 3. The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment). Therefore, the NTS reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

The LCO requires four channels of Pressurizer Pressure – Low 3 to be OPERABLE in MODES 1 and 2. Additionally, they are required to be OPERABLE in MODE 3 above P-11 or in a condition below P-11 where the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F, to mitigate the consequences of a high energy line rupture inside containment. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic actuation below this pressure is then performed by the Containment Pressure – High 2 signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint (as long as the boron concentration is greater than that necessary to meet the SDM requirements at an RCS temperature of 200°F). Other ESF Functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

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6. Pressurizer Water Level – Low

A signal to isolate the purification line, the auxiliary spray line, the zinc addition line, and the hydrogen addition line is generated upon the coincidence of pressurizer level below the Low setpoint in any two-out-of-four divisions. Each water level signal is compensated using pressurizer reference leg temperature, pressurizer pressure and RCS wide range pressure.

The Auxiliary Spray and Purification Line and Zinc/Hydrogen Addition Isolation ESFAS protective function is actuated by Pressurizer Water Level – Low.

This function can be manually blocked when the pressurizer water level is below the P-12 setpoint to permit operations during shutdown conditions that require the operation of the lines isolated by this Function. This Function is automatically unblocked when the pressurizer water level is above the P-12 setpoint.

Four channels of Pressurizer Water Level - Low are required to be OPERABLE in MODES 1 and 2 to help maintain RCS inventory. In MODES 3, 4, 5, and 6, this Function is not needed for accident detection and mitigation.

7. Pressurizer Water Level – Low 2

This Function initiates CMT Actuation and tripping of the RCPs from the coincidence of pressurizer level below the Low 2 Setpoint in any two of the four divisions. Each water level signal is compensated using pressurizer reference leg temperature, pressurizer pressure and RCS wide range pressure.

The ESFAS protective functions actuated by Pressurizer Water Level – Low 2 are:

- CMT Actuation; and
- Reactor Coolant Pump Trip.

This function can be manually blocked when the pressurizer water level is below the P-12 setpoint to prevent spurious CMT actuations on Pressurizer Water Level - Low 2. This Function is automatically unblocked when the pressurizer water level is above the P-12 Setpoint. The setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment.

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This Function is required to be OPERABLE in MODES 1, 2, 3, and 4. This Function is also required to be OPERABLE in MODE 5 with RCS not VENTED and CMT actuation on Pressurizer Water Level - Low 2 not blocked.

8. Pressurizer Water Level – High

Four channels of pressurizer level are provided on the pressurizer. Two-out-of-four channels on indicating level greater than the High setpoint coincident with a Safeguards Actuation signal will close the containment isolation valves for the CVS. This Function prevents the pressurizer level from reaching a level that could lead to water relief through the pressurizer safety valves during some DBAs. Each water level signal is compensated using pressurizer reference leg temperature, pressurizer pressure and RCS wide range pressure.

The Chemical and Volume Control System Makeup Isolation ESFAS protective function is actuated by Pressurizer Water Level – High.

This Function is required to be OPERABLE in MODES 1, 2, and 3. This function is not required to be OPERABLE in MODES 4, 5, and 6, because it is not required to mitigate a DBA in these MODES.

9. Pressurizer Water Level – High 2

A signal to close the CVS isolation valves is generated on Pressurizer Water Level – High 2. This Function results from the coincidence of pressurizer level above the High 2 setpoint in any two of the four divisions. The PMS allows this Function to be manually blocked when the pressurizer pressure is below the P-19 permissive setpoint. The operator can manually block this function once it is no longer required to be OPERABLE when at least one RCS cold leg temperature is ≤ 275°F to permit pressurizer water solid conditions with the plant cold and to permit level makeup during plant cooldowns. This Function is automatically unblocked when RCS pressure is above the P-19 (RCS Pressure) setpoint. Each water level signal is compensated using pressurizer reference leg temperature, pressurizer pressure and RCS wide range pressure.

The Chemical and Volume Control System Makeup Isolation ESFAS protective function is actuated by Pressurizer Water Level – High 2.

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This Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when all four cold leg temperatures are > 275°F. This Function is not required to be OPERABLE in MODE 4 with at least one cold leg temperature ≤ 275°F or MODES 5 and 6 because it is not required to mitigate a DBA in these MODES.

10. Pressurizer Water Level – High 3

PRHR is actuated and the pressurizer heaters are tripped when the pressurizer water level reaches its High 3 setpoint. This signal provides protection against a pressurizer overfill following an inadvertent core makeup tank actuation with consequential loss of offsite power. This Function is automatically unblocked when RCS pressure is above the P-19 (RCS pressure) setpoint. Each water level signal is compensated using pressurizer reference leg temperature, pressurizer pressure and RCS wide range pressure.

The ESFAS protective functions actuated by Pressurizer Water Level – High 3 are:

- PRHR Heat Exchanger Actuation; and
- Pressurizer Heater Trip.

This Function is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when all four cold leg temperatures are > 275°F. This Function is not required to be OPERABLE in MODE 4 with at least one cold leg temperature ≤ 275°F or in MODES 5 and 6 because it is not required to mitigate DBA in these MODES.

11a. RCS Cold Leg Temperature (T_{cold}) – Low 2

This signal provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

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The ESFAS protective functions actuated by RCS Cold Leg Temperature (T_{cold}) – Low 2 are:

- Safeguards Actuation;
- Steam Line Isolation; and
- Startup Feedwater Isolation.

This Function provides closure of the MSIVs during a SLB or inadvertent opening of a SG relief or a safety valve to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. This Function also closes the startup feedwater control and isolation valves and trips the startup feedwater pumps if reactor coolant system cold leg temperature is below the T_{cold} setpoint in any loop.

The LCO requires four channels of T_{cold} – Low 2 to be OPERABLE in MODES 1 and 2. Additionally, they are required to be OPERABLE in MODE 3 above P-11 or in a condition below P-11 where the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F. At these conditions, a secondary side break or stuck open valve could result in the rapid cooldown of the primary side. Two of the four channels in each loop average a fast-response RTD with a cold leg dual range temperature element while the remaining two channels use only the fast-response RTD. Four channels are provided in each loop to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation because the cold leg temperature is reduced below the actuation setpoint.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

11.b. RCS Cold Leg Temperature (T_{cold}) – High

PRHR is actuated when the T_{cold} reaches its High setpoint in either loop coincident with SG Narrow Range Water Levels indicating Low 2 coincident with Startup Feedwater Flows indicating Low 2 in both SGs. The T_{cold} – High Function requires four channels per loop to be OPERABLE to satisfy the requirements with a two-out-of-four logic in either loop. Two of the four channels in each loop average a fast-response RTD with a cold leg dual range temperature element while the remaining two channels use only the fast-response RTD. Four channels are provided to permit on channel to be in trip of bypass indefinitely and still ensure no single random failure will disable this Function. The Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment. The ESFAS protective functions actuated by T_{cold} – High are:

- PRHR Heat Exchanger Actuation.

The T_{cold} – High Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the RNS. This ensures that the PRHR can be actuated in the event of a loss of the normal heat removal systems. In MODE 4 when the RCS is being cooled by the RNS, and in MODES 5 and 6, the PRHR is not required to provide the normal RCS heat sink.

12. T_{avg} Low

This signal provides protection against excessive feedwater flow by closing the main feedwater control valves. This signal results from a coincidence of two of the four divisions of reactor loop average temperature below the Low setpoint coincident with the P-4 permissive. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function.

The Main Feedwater Control Valve Isolation ESFAS protective function is actuated by T_{avg} Low.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Closing the Main Feedwater Control Valves on T_{avg} Low coincident with Reactor Trip (P-4) is required to be OPERABLE in MODES 1 and 2. Failure to close the main feedwater control valves following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment.

13. T_{avg} Low-2

This signal provides protection against excessive feedwater flow by closing the main feedwater isolation and crossover leg valves, and tripping of the main feedwater pumps. This signal results from a coincidence of two out of four divisions of reactor loop average temperature below the Low 2 setpoint coincident with the P-4 permissive.

Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function. This Function may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint.

The Main Feedwater Pump Trip and Valve Isolation ESFAS protective function is actuated by T_{avg} Low-2.

This Function is required to be OPERABLE in MODES 1 and 2 to mitigate the effects of a large SLB or a large FLB. Failure to trip the turbine or isolate the main feedwater system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment.

14. RCS Wide Range Pressure – Low

The fourth stage depressurization valves open on manual actuation, but is interlocked to actuate with either the low RCS pressure signal or with the ADS Stages 1, 2, & 3 actuation. These interlocks minimize the potential for inadvertent actuation of this Function. This consideration is important in PRA modeling to improve the reliability of reducing the RCS pressure following a small LOCA or transient event.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The ADS Stage 4 Actuation ESFAS protective function is actuated by RCS Wide Range Pressure – Low.

This Function must be OPERABLE in MODES 1, 2, 3, 4, and 5. This Function must also be OPERABLE in MODE 6 with the upper internals in place.

15. CMT Level – Low 3

This Function ensures continued passive injection or borated water to the RCS following a small break LOCA. ADS Stages 1, 2 and 3 actuation is initiated when the CMT Level reaches its Low 3 setpoint coincident with any CMT Actuation signal. Four channels are provided in each CMT to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The ADS Stages 1, 2, & 3 Actuation ESFAS protective function is actuated by CMT Level – Low 3.

This Function must be OPERABLE in MODES 1, 2, 3, and 4. This Function must also be OPERABLE in MODE 5 with the RCS not VENTED. In MODE 4 with the RCS being cooled by the RNS and in MODE 5 with the RCS not VENTED only one CMT is required to be OPERABLE in accordance with LCO 3.5.3, CMTs - Shutdown, RCS Intact; therefore, CMT level channels are only required on an OPERABLE CMT.

16. CMT Level – Low 6

The fourth stage isolation valves receive a confirmatory open signal (nonsafety-related function) following a preset time delay after the first stage depressurization valves are sent a signal to open. The fourth stage depressurization valves open on CMT Level – Low 6 in two-out-of-four channels in either CMT. Actuation of the fourth stage depressurization valves is interlocked with the third stage depressurization signal such that the fourth stage is not actuated unless the third stage has been previously actuated following a preset time delay. Actuation of the fourth stage ADS valves are further interlocked with a low RCS pressure signal such that the ADS Stage 4 actuation is not actuated unless the RCS pressure is below a predetermined setpoint.

Four channels of CMT level instrumentation are provided per tank to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The ADS Stage 4 Actuation ESFAS protective function is actuated by CMT Level – Low 6.

This Function must be OPERABLE in MODES 1, 2, 3, 4, and in MODE 5 with the RCS not VENTED. In MODE 4 with the RCS being cooled by the RNS and in MODE 5 with the RCS not VENTED, only one CMT is required to be OPERABLE in accordance with LCO 3.5.3, CMTs - Shutdown, RCS Intact; therefore, CMT level channels are only required on an OPERABLE CMT.

17. Source Range Neutron Flux Doubling

The source range neutron detectors are used for this Function. A signal to block boron dilution is derived from source range neutron flux increasing at an excessive rate (source range neutron flux doubling). The LCO requires four divisions to be OPERABLE. There are four divisions and two-out-of-four logic is used. On a coincidence of excessively increasing source range neutron flux in two of the four divisions, demineralized water is isolated (CVS demineralized water system isolation valves closed) from the makeup pumps and reactor coolant makeup is isolated (CVS makeup line isolation valves closed) from the reactor coolant system to preclude a boron dilution event.

The Boron Dilution Block ESFAS protective function is actuated by Source Range Neutron Flux Doubling.

The signal to block boron dilution on source range neutron flux increasing at an excessive rate (source range neutron flux doubling) must be OPERABLE in MODE 3 if unborated water source flow paths are not isolated except when an intentional approach to criticality is in progress. It must also be OPERABLE in MODES 4 and 5 if unborated water flow paths are not isolated. In MODE 6, a dilution event is precluded by the requirement in LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path.

Above the P-6 setpoint the source range detectors can be manually deenergized and above the P-10 setpoint they are automatically deenergized. OPERABILITY of the Source Range Neutron Flux Doubling Function requires the source range neutron flux detectors to be energized.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

When the Source Range Neutron Flux Doubling feature is blocked below P-8, an automatic close signal is sent to both valves in the CVS demineralized water flow path to ensure a single failure does not reestablish this potential dilution flow path.

18. IRWST Lower Narrow Range Level – Low 3

A Low 3 IRWST level coincident with a ADS Stage 4 Actuation signal will open the containment recirculation valves. Four channels of IRWST Lower Narrow Range Level – Low 3 instrumentation are provided to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this trip Function.

The IRWST Containment Recirculation Valve Actuation ESFAS protective function is actuated by IRWST Lower Narrow Range Level – Low 3.

Four channels of IRWST Lower Narrow Range Level - Low 3 are required to be OPERABLE in MODES 1, 2, 3, 4, and 5, and MODE 6 with the upper internals in place.

19. Reactor Coolant Pump Bearing Water Temperature – High 2

The CCS containment isolation valves are closed and the RCPs are tripped if two-out-of-four sensors on any RCP indicate High 2 bearing water temperature. This Function is lead-lag compensated.

The ESFAS protective functions actuated by Reactor Coolant Pump Bearing Water Temperature – High 2 are:

- Reactor Coolant Pump Trip; and
- Component Cooling Water System Containment Isolation Valve Closure.

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4. Four channels are provided for each RCP to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

20. SG Narrow Range Water Level – Low 2

PRHR is actuated when the SG Narrow Range Water Levels reaches its Low 2 setpoint coincident with an indication of Low 2 Startup Feedwater Flows in both SGs, coincident with an indication of T_{cold} – High in either loop. The LCO requires four channels per steam generator to be OPERABLE to satisfy the requirements with a two-out-of-four logic in each steam generator. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Each water level signal is compensated using steam line pressure. The Setpoint reflects both steady state and adverse environmental instrument uncertainties as the detectors provide protection for an event that results in a harsh environment.

The ESFAS protective functions actuated by SG Narrow Range Water Level – Low 2 are:

- PRHR Heat Exchanger Actuation; and
- SG Blowdown Isolation

The SG Narrow Range Water Level – Low 2 Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the Normal Residual Heat Removal System (RNS). This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems. In MODE 4 when the RCS is being cooled by the RNS, and in MODES 5 and 6, the SGs are not required to provide the normal RCS heat sink.

Therefore, startup feedwater flow is not required, and PRHR actuation on Low 2 steam generator narrow range water level is not required. This ESFAS function can be manually blocked by the main control room operator when below the P-9 setpoint (T_{avg} interlock) and is automatically unblocked when above the P-9 setpoint. This block is necessary to permit routine maintenance without PRHR heat exchanger actuation or SG blowdown isolation.

21. SG Wide Range Water Level – Low 2

PRHR is also actuated when the SG Wide Range Water Level reaches its Low 2 Setpoint. There are four wide range level channels for each steam generator and a two-out-of-four logic is used. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

disable this trip Function. Each water level signal is compensated using steam line pressure.

The PRHR Heat Exchanger Actuation ESFAS protective function is actuated by SG Narrow Range Water Level – Low 2.

This Function is required to be OPERABLE in MODES 1, 2, and 3 and in MODE 4 when the RCS is not being cooled by the RNS. This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems. In MODE 4 when the RCS is being cooled by the RNS, and in MODES 5 and 6, the SGs are not required to provide the normal RCS heat sink. Therefore, SG Wide Range Level is not required, and PRHR actuation on SG Wide Range Water Level – Low 2 is not required. This ESFAS function can be manually blocked by the main control room operator when below the P-9 setpoint (T_{avg} interlock) and is automatically unblocked when above the P-9 setpoint. This block is necessary to permit routine maintenance without PRHR heat exchanger actuation.

22. SG Narrow Range Water Level – High

If steam generator narrow range level reaches the High setpoint in either steam generator coincident with a Reactor Trip (P-4), then all startup feedwater control and isolation valves are closed, the startup feedwater pumps are tripped, and the isolation valves for the CVS are closed. This Function prevents adding makeup water to the RCS during an SGTR. Four channels are provided in each steam generator to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this function. Each water level signal is compensated using steam line pressure.

The ESFAS protective functions actuated by SG Narrow Range Level – High are:

- Startup Feedwater Isolation; and
- Chemical and Volume Control System Makeup Isolation.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4. In MODE 4 with an RCS temperature < 350°F and MODES 5 and 6, the RCS pressure and temperature are reduced and an SGTR is not credible. However, the Applicability conservatively requires OPERABILITY of this Function in MODE 4 with RCS temperature < 350°F (i.e., all of MODE 4). Therefore, startup feedwater isolation and CVS makeup isolation actuations on High narrow range steam generator water level coincident with reactor trip (P-4) are not required in MODES 5 and 6. These ESFAS functions can be manually blocked by the main control room operator when below the P-9 setpoint (T_{avg} interlock) and are automatically unblocked when above the P-9 setpoint. This block is necessary to permit the steam generators to be flooded for wet-layup without startup feedwater isolation and CVS makeup isolation.

23. SG Narrow Range Water Level – High 3

This signal provides protection against excessive feedwater flow by closing the main feedwater control, isolation and crossover valves, tripping of the main feedwater pumps, and tripping the turbine. The signal also prevents adding makeup water to the RCS during a SGTR by closing the isolation valves for the CVS. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Each water level signal is compensated using steam line pressure.

The ESFAS protective functions actuated by SG Narrow Range Water Level – High 3 are:

- Turbine Trip;
- Main Feedwater Control Valve Isolation;
- Main Feedwater Pump Trip and Valve Isolation;
- Startup Feedwater Isolation, and
- Chemical and Volume Control System Makeup Isolation.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The transmitters (d/p cells) are located inside containment. However, the events which this Function protect against cannot cause severe environment in containment. Therefore, the Setpoint reflects only steady state instrument uncertainties. The LCO requires four channels of SG Narrow Range Water Level – High 3 instrumentation per steam generator to be OPERABLE in MODES 1, 2, 3, and 4 when there is significant mass and energy in the RCS and the steam generators. In MODES 5 and 6, the energy in the RCS and the steam generators is low; therefore, startup feedwater isolation, main feedwater isolation, CVS makeup isolation, and turbine trip on High-3 narrow range steam generator water level are not required to be OPERABLE.

These ESFAS functions can be manually blocked by the main control room operator when below the P-9 setpoint (T_{avg} interlock) and are automatically unblocked when above the P-9 setpoint. This block is necessary to permit the steam generators to be flooded for wet-layup without startup feedwater isolation, main feedwater isolation, CVS makeup isolation, or actuation of turbine trip.

24. Steam Line Pressure – Low 2

Steam Line Pressure – Low 2 provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure – Low 2 provides closure of the PORV flow paths in the event of SGTR in which the PORV(s) open, to limit the radiological releases from the ruptured steam generator into the atmosphere. Steam Line Pressure – Low 2 also provides closure of the MSIVs in the event of an SLB to limit the mass and energy release to containment and limit blowdown to a single SG.

Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure that no single random failure will disable this Function.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

This Function is anticipatory in nature and has a typical leading/lag ratio of 50/5. It is possible for the transmitters to experience adverse environmental conditions during a secondary side break. Therefore, the NTS reflects both steady state and adverse environmental instrument uncertainties.

The ESFAS protective functions actuated by Steam Line Pressure – Low 2 are:

- Safeguards Actuation;
- Steam Line Isolation; and
- SG Power Operated Relief Valve and Block Valve Isolation.

The LCO requires four channels per steam line of Steam Line Pressure – Low 2 Function to be OPERABLE in MODES 1 and 2. Additionally, they are required to be OPERABLE in MODE 3 above P-11 or in a condition below P-11 where the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F, or in a condition below P-11 where Steam Line Pressure – Low 2 is not blocked. This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint as long as the boron concentration is greater than that necessary to meet the SDM requirements at an RCS temperature of 200°F and the Function is blocked.

25. Steam Line Pressure-Negative Rate – High

Steam Line Pressure-Negative Rate – High provides closure of the MSIVs for an SLB, when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure – Low 2 when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate – High signal is automatically enabled.

The Steam Line Isolation ESFAS protective function is actuated by Steam Line Pressure-Negative Rate – High.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires four channels of Steam Line Pressure-Negative Rate – High instrumentation per steam line to be OPERABLE in MODE 3 when less than the P-11 setpoint and Steam Line Pressure – Low 2 is blocked, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). Four channels are provided in each steam line to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. In MODES 1 and 2, and in MODE 3 when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure – Low 2 signal is automatically enabled.

In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

While the transmitters may experience elevated ambient temperatures due to a steam line break, the Trip Function is on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the NTS reflects only steady state instrument uncertainties.

ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this specification may be entered independently for each Function listed on Table 3.3.8-1. The Completion Time(s) of the inoperable equipment of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

In the event a channel's as-found condition is outside the as-found tolerance described in the SP, or the channel is not functioning as required, or the transmitter, or the Protection and Safety Monitoring System Division, associated with a specific Function is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the particular protection Function(s) affected. When the Required Channels are specified only on a per steam line, per loop, per SG, basis, then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

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BASES

ACTIONS (continued)

A.1

Condition A is applicable to the ESFAS protection Functions listed in Table 3.3.8-1. Condition A addresses the situation where one channel for one or more functions is inoperable. With one channel inoperable, the affected channel must be placed in a bypass or trip condition within 6 hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) The 6 hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference 5.

B.1 and B.2

With one or more functions with two channels inoperable, one affected channel must be placed in bypass and one affected channel must be placed in trip within 6 hours. If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The 6 hours allowed to place one inoperable channel(s) in bypass and one inoperable channel(s) in trip is justified in Reference 5.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.8-1. The applicable Condition referenced in the table is Function dependent. If the Required Action and the associated Completion Time of Condition A or B are not met or if three or more channels for one or more Functions are inoperable Condition C is entered to provide for transfer to the appropriate subsequent Condition.

D.1

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels for one or more Functions are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours. The allowed time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

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BASES

ACTIONS (continued)

E.1, E.2 and E.3

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and by reducing reactor pressure below the P-11 (Pressurizer Pressure) interlock and establishing RCS boron concentration greater than or equal to that necessary to meet the SDM requirements at an RCS temperature of 200°F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

F.1 and F.2

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 with the RCS being cooled by the RNS within 24 hours. The allowed time is reasonable, based on operating experience, to reach the required plant conditions in an orderly manner without challenging plant systems.

G.1, G.2, and G.3

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours, and establishing RCS boron concentration greater than or equal to that necessary to meet the SDM requirements at an RCS temperature of 200°F within 12 hours. The Required Action G.3 requirement to block Steamline/Feedwater isolation and Safeguards actuations is not required until 1 hour from discovery of RCS boron concentration greater than or equal to that necessary to meet SDM at an RCS temperature of 200°F while in MODE 3. Establishing SDM prior to reducing pressurizer pressure and cooling down the RCS maintains the necessary margins assumed for the SLB. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

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ACTIONS (continued)

H.1 and H.2

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

I.1

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the affected isolation valve(s) must be declared inoperable immediately. Declaring the affected isolation valve inoperable allows the supported system Actions (i.e., for inoperable valves) to dictate the required measures. The respective isolation valve LCO provides appropriate actions for the inoperable components. This actions is in accordance with LCO 3.0.6, which requires that the applicable Conditions and Required Actions for the isolation valves declared inoperable shall be entered in accordance with LCO 3.0.2.

J.1 and J.2

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. Required Action J.1 requires that the plant shall be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 5.

It is further required to immediately initiate action to restore the instrument channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function. For Function 7, the Applicability (and ACTIONS Condition) can also be exited by manually blocking P-12 when the pressurizer water level is below the P-12 setpoint.

The Completion Time to be in MODE 5 and initiate action to restore compliance (Required Actions J.1 and J.2) is 37 hours with three or more channels for the affected Function inoperable. This time is based on the time provided in LCO 3.0.3 to reach MODE 5. The 180 hour Completion Time is based on the ability of the two remaining OPERABLE channels to provide the protective Function.

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ACTIONS (continued)

K.1 and K.2

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a condition in which the likelihood and consequences of a criticality event are minimized. This is accomplished by immediately suspending positive reactivity additions. Additionally, action is required to immediately initiate action to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function.

L.1 and L.2 – Not used.

M.1, M.2, and M.3

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a condition in which the potential for a criticality event is minimized. This is accomplished by immediately suspending positive reactivity additions (Required Action M.1).

The plant must also be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 5 within 12 hours per Required Action M.2. The 12 hours is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

Required Action M.3 requires initiation of action within 12 hours to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function. The 12 hour Completion Time is consistent with the transition to MODE 5 in accordance with Required Action M.2.

N.1 and N.2 – Not used.

O.1 and O.2

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the affected isolation valve(s) must be declared inoperable immediately. Declaring the affected isolation valve inoperable allows the supported system Actions (i.e., for inoperable valves) to dictate the required measures. The respective isolation valve LCOs provide appropriate actions for the inoperable components. This actions is in accordance with LCO 3.0.6,

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ACTIONS (continued)

which requires that the applicable Conditions and Required Actions for the isolation valves declared inoperable shall be entered in accordance with LCO 3.0.2. Additionally, Required Action O.2 requires that the plant must be brought to at least MODE 3 within 6 hours. The allowed time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

P.1, P.2, and P.3

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Required Action P.3 requires initiation of action within 36 hours to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function. The 36 hour Completion Time is consistent with the transition to MODE 5 in accordance with Required Action P.2.

Q.1 and Q.2

If the Required Action and associated Completion Time of Condition A or B is not met or if three or more channels are inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 with at least one cold leg temperature ≤ 275°F within 24 hours. The allowed time is reasonable, based on operating experience, to reach the required plant conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each ESFAS Instrumentation Function in Table 3.3.8-1, except where noted below.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside their corresponding limits.

The Surveillance Requirement is modified by a Note. The Note states that this SR is only required for Source Range Neutron Flux Doubling. The OPERABILITY for the other Functions is verified by self-checking features in lieu of performing a CHANNEL CHECK.

The Surveillance Frequency is based on operating experience that demonstrates that channel failure is rare. Automated operator aids may be used to facilitate performance of the CHANNEL CHECK.

SR 3.3.8.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor and the integrated protection cabinets (IPC). The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found

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SURVEILLANCE REQUIREMENTS (continued)

condition will be entered into the Corrective Action Program for further evaluation. Transmitter calibration must be performed consistent with the assumptions of the setpoint methodology. The difference between the current as-found values and the previous as-left values must be consistent with the transmitter drift allowance used in the setpoint methodology.

Interlocks implicitly required to support the Function's OPERABILITY are also addressed by this CHANNEL CALIBRATION. This portion of the CHANNEL CALIBRATION ensures the associated Function is not bypassed when required to be enabled. This can be accomplished by ensuring the interlocks are calibrated properly in accordance with the SP. If the interlock is not automatically functioning as designed, the condition is entered into the Corrective Action Program and appropriate OPERABILITY evaluations performed for the affected Function. The affected Function’s OPERABILITY can be met if the interlock is manually enforced to properly enable the affected Function. When an interlock is not supporting the associated Function’s OPERABILITY at the existing plant conditions, the affected Function's channels must be declared inoperable and appropriate ACTIONS taken. In the case of an interlock that is applied “post-vote,” the ESF Actuation Logic Function of LCO 3.3.15 or LCO 3.3.16 (as applicable to the plant condition) is declared inoperable.

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months).

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by two Notes. Note 1 states that this test should include verification that the time constants are adjusted to within limits where applicable. Note 2 states that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consisting of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer’s data.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.3

This SR ensures the individual channel ESF RESPONSE TIME is less than or equal to the maximum value assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the NTS value at the sensor, to the point at which the equipment reaches the required functional state (e.g., valves in full open or closed position).

In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by a series of overlapping tests and allocated values such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 1) response time.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the allocated sensor, signal processing and actuation logic response times, where approved, with actual response time tests on the remainder of the channel.

ESF RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Testing of the devices is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

The Surveillance Requirement is modified by two Notes: Note 1 states that it is not applicable to Function 1.a for Containment Pressure – Low. The exception is appropriate because the Containment Pressure – Low signal provides an interlock function for the containment vacuum relief valves manual initiation function and does not directly actuate any ESF. Note 2 exempts neutron detectors from ESF RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

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BASES

REFERENCES

1. FSAR Chapter 7, “Instrumentation and Controls.”
2. FSAR Chapter 15, “Accident Analysis.”
3. Institute of Electrical and Electronic Engineers, IEEE-603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” June 27, 1991.
4. 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”
5. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
6. WCAP-16361-P, “Westinghouse Setpoint Methodology for Protection Systems – AP1000,” February 2011 (proprietary).
7. Not used.
8. WCAP-16675-P, “AP1000 Protection and Safety Monitoring System Architecture Technical Report.”

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B 3.3 INSTRUMENTATION

B 3.3.9 Engineered Safety Feature Actuation System (ESFAS) Manual Initiation

BASES

BACKGROUND

A description of the ESFAS Instrumentation is provided in the Bases for LCO 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

A description of the following ESFAS protective functions is provided in the Bases for LCO 3.3.8:

- Safeguards Actuation
- Core Makeup Tank (CMT) Actuation
- Containment Isolation (T signal)
- Steam Line Isolation
- Main Feedwater Control Valve Isolation
- Automatic Depressurization System (ADS) Stages 1, 2, & 3 Actuation
- ADS Stage 4 Actuation
- Passive Containment Cooling Actuation
- Passive Residual Heat Removal Heat Exchanger Actuation
- Chemical and Volume Control System Makeup Isolation
- Normal Residual Heat Removal System Isolation
- In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valve Actuation
- IRWST Containment Recirculation Valve Actuation
- Steam Generator (SG) Power Operated Relief Valve and Block Valve Isolation
- Containment Vacuum Relief Valve Actuation

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires OPERABILITY of two devices for each manual initiation Function listed in Table 3.3.9-1. Two manual initiation channels are required to ensure no single random failure disables the ESFAS. The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS manual initiation functions are as follows:

1. Safeguards Actuation - Manual Initiation

   The LCO requires that two manual initiation devices are OPERABLE. The operator can initiate the Safeguards Actuation signal at any time by using either of two switches in the main control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

   The LCO on Safeguards Actuation - Manual Initiation ensures the proper amount of redundancy is maintained in the manual Safeguards actuation circuitry to ensure the operator has manual Safeguards Actuation capability.

   Each device consists of one switch and the interconnecting wiring to all four divisions. Each manual initiation device actuates all four divisions. This configuration does not allow testing at power.

   Manual Safeguards Actuation must be OPERABLE in MODES 1, 2, 3, and 4.

   Manual initiation is required in MODE 5 to support system level initiation.

   This Safeguards Actuation Function is not required to be OPERABLE in MODE 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many engineered safety features (ESF) components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

2. Core Makeup Tank (CMT) Actuation - Manual Initiation

   CMT Actuation provides the passive injection of borated water into the RCS. Injection provides RCS makeup water and boration during transients or accidents when the normal makeup supply from the Chemical and Volume Control System (CVS) is lost or insufficient.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Manual CMT Actuation is accomplished by either of two switches in the main control room. Either switch activates all four divisions.

Manual CMT Actuation must be OPERABLE in MODES 1 through 3, and MODE 4 with the Reactor Coolant System (RCS) not cooled by the Normal Residual Heat Removal System (RNS). Manual actuation of the CMT is additionally required in MODE 4 when the RCS is being cooled by the RNS, and MODE 5 with the RCS not VENTED. Actuation of this Function is not required in MODE 5 with the RCS VENTED, or MODE 6 because the CMTs are not required to be OPERABLE in these MODES.

3. Containment Isolation - Manual Initiation

Containment isolation is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

Manual Containment Isolation is accomplished by either of two switches in the main control room. Either switch actuates all four ESFAS divisions. Manual initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3, and 4, when containment integrity is required.

4. Steam Line Isolation - Manual Initiation

Isolation of the main steam lines provides protection in the event of a Steam Line Break (SLB) inside or outside containment.

Manual initiation of Steam Line Isolation can be accomplished from the main control room. There are two switches in the main control room and either switch can initiate action to immediately close all main steam isolation valves (MSIVs).

The LCO requires two OPERABLE channels in MODES 1, 2, 3, and 4. In MODES 5 and 6, this Function is not required to be OPERABLE because there is insufficient energy in the secondary side of the unit to cause an accident.

5. Feedwater Isolation - Manual Initiation

The primary Function of Feedwater Isolation is to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Manual Feedwater Isolation can be accomplished from the main control room. There are two switches in the main control room and either switch can initiate action in both divisions to close all main and startup feedwater control, isolation and crossover valves, trip all main and startup feedwater pumps, and trip the turbine.

Feedwater isolation is necessary in MODES 1, 2, 3, and 4 to mitigate the effects of a large SLB or feedwater line break (FLB). In MODES 5 and 6, the energy in the RCS and the steam generators is low and this function is not required to be OPERABLE.

6. ADS Stages 1, 2, & 3 Actuation - Manual Initiation

The Automatic Depressurization System (ADS) provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the in-containment refueling water storage tank (IRWST) to mitigate the effects of a LOCA.

The operator can initiate an ADS Stages 1, 2, and 3 actuation from the main control room by simultaneously actuating two ADS actuation devices in the same set. There are two sets of two switches each in the main control room. Simultaneously actuating the two devices in either set will actuate ADS Stages 1, 2, and 3.

This Function must be OPERABLE in MODES 1 through 4, and MODE 5 with the RCS not VENTED. In MODE 5 with the RCS VENTED and in MODE 6 with upper internals in place, LCO 3.4.13, ADS - Shutdown, RCS Open, required the ADS Stages 1, 2, and 3 valves to be open. Thus, Manual actuation is not required.

7. ADS Stage 4 Actuation - Manual Initiation

The ADS provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the IRWST to mitigate the effects of a LOCA.

The fourth stage depressurization valves open on manual actuation. The operator can initiate Stage 4 of ADS from the main control room. There are two sets of two switches each in the main control room. Actuating the two switches in either set will actuate all 4th stage ADS valves. This manual actuation is interlocked to actuate with either the low RCS pressure signal or with the ADS Stages 1, 2, & 3 actuation. These interlocks minimize the potential for inadvertent

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

actuation of this Function. This interlock with the ADS Stages 1, 2, & 3 actuation Function allows manual actuation of this Function if automatic or manual actuation of the ADS Stages 1, 2, & 3 valves fails to depressurize the RCS due to common-mode failure. This consideration is important in PRA modeling to improve the reliability of reducing the RCS pressure following a small LOCA or transient event. This Function must be OPERABLE in MODES 1 through 5, and MODE 6 with the upper internals in place. In MODE 6 with the upper internals not in place, the Stage 4 ADS valves are not required to be OPERABLE by LCO 3.4.13, thus Manual Initiation of the valves is not required.

8. Passive Containment Cooling Actuation - Manual Initiation

The Passive Containment Cooling System (PCS) transfers heat from the reactor containment to the environment. This Function is necessary to prevent the containment design pressure and temperature from being exceeded following any postulated DBA (such as LOCA or SLB).

The operator can initiate Containment Cooling at any time from the main control room by actuating either of the two containment cooling actuation switches. There are two switches in the main control room, either of which will actuate containment cooling in all divisions. Manual Initiation of containment cooling also actuates containment isolation.

The LCO requires this Function to be OPERABLE in MODES 1, 2, 3, and 4 when the potential exists for a DBA that could require the operation of the Passive Containment Cooling System. In MODES 5 and 6, with decay heat greater than 7.0 MWt, manual initiation of the PCS provides containment heat removal. Section B 3.6.6, Applicability, provides the basis for the decay heat limit.

9. Passive Residual Heat Removal Heat Exchanger Actuation - Manual Initiation

The PRHR Heat Exchanger (HX) provides emergency core decay heat removal when the Startup Feedwater System is not available to provide a heat sink.

Manual PRHR actuation is accomplished by either of two switches in the main control room. Either switch actuates all four ESFAS Divisions.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4, and MODE 5 with the RCS pressure boundary intact. This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems.

10. Chemical and Volume Control System Makeup Isolation - Manual Initiation

The CVS makeup line, auxiliary spray line, zinc addition line, hydrogen addition line, and letdown purification line are isolated following certain events to prevent overfilling of the RCS.

Manual Chemical and Volume Control System Makeup Isolation is actuated by either of two switches in the main control room. Either switch closes Chemical and Volume Control System makeup line, auxiliary spray line, zinc addition line, hydrogen addition line, and letdown purification line isolation valves. The LCO requires two switches to be OPERABLE.

This Function is required to be OPERABLE in MODES 1 through 3, and MODE 4 with all four cold leg temperatures > 275°F. This Function is not required to be OPERABLE in MODE 4 with at least one cold leg temperature ≤ 275°F or in MODES 5 and 6 because it is not required to mitigate a DBA in these MODES.

11. Normal Residual Heat Removal System Isolation - Manual Initiation

The RNS suction line is isolated by closing the containment isolation valves to provide containment isolation following an accident

The operator can initiate RNS isolation at any time from the control room by simultaneously actuating two switches in the same actuation set. Because an inadvertent actuation of RNS isolation could have serious consequences, two switches must be actuated simultaneously to initiate isolation. There are two sets of two switches in the control room. Simultaneously actuating the two switches in either set will isolate the RNS in the same manner as the automatic actuation signal. Two Manual Initiation switches in each set are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function.

This Function is required to be OPERABLE in MODES 1, 2, and 3.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

12. In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valve Actuation - Manual Initiation

    The PXS provides core cooling by gravity injection and recirculation for decay heat removal following an accident. Manual initiation will generate a signal to open the IRWST injection line and actuate IRWST injection.

    The operator can open IRWST injection line valves at any time from the main control room by actuating two IRWST injection actuation switches in the same actuation set. There are two sets of two switches each in the main control room.

    This Function is required to be OPERABLE in MODES 1 through 3, and MODE 4 with the Reactor Coolant System (RCS) not cooled by the Normal Residual Heat Removal System (RNS). Manual actuation of the IRWST injection line valves is additionally required in MODE 4 when the RCS is being cooled by the RNS, MODE 5, and MODE 6.

13. IRWST Containment Recirculation Valve Actuation - Manual Initiation

    The PXS provides core cooling by gravity injection and recirculation for decay heat removal following an accident. Manual initiation will open these valves.

    The operator can open the containment recirculation valves at any time from the main control room by actuating two containment recirculation actuation switches in the same actuation set. There are two sets of two switches each in the main control room.

    This Function is required to be OPERABLE in MODES 1 through 3, and MODE 4 with the Reactor Coolant System (RCS) not cooled by the Normal Residual Heat Removal System (RNS). Manual actuation of the IRWST containment recirculation valves is additionally required in MODE 4 when the RCS is being cooled by the RNS, MODE 5, and MODE 6.

14. SG Power Operated Relief Valve and Block Valve Isolation - Manual Initiation

    The Function of the SG Power Operated Relief Valve and Block Valve Isolation is to ensure that the SG PORV flow paths can be isolated during a SG tube rupture (SGTR) event.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Manual initiation of SG Power Operated Relief Valve and Block Valve Isolation can be accomplished from the control room. There are two switches in the control room and either switch can close the SG PORVs and PORV block valves. The LCO requires two switches to be OPERABLE.

This Function is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 with the RCS cooling not being provided by the RNS. In MODE 4 with an RCS temperature < 350°F and MODES 5 and 6, the RCS pressure and temperature are reduced and an SGTR is not credible. However, the Applicability conservatively requires OPERABILITY of this Function in MODE 4 with RCS temperature < 350°F until the RCS is being cooled by RNS.

15. Containment Vacuum Relief Valve Actuation - Manual Initiation

The purpose of the vacuum relief lines is to protect the containment vessel against damage due to a negative pressure (i.e., a lower pressure inside than outside).

The operator can open the vacuum relief valves from the main control room by actuating either of the two vacuum relief actuation switches. There are two switches in the main control room, either of which will actuate vacuum relief in all divisions. Manual Containment Vacuum Relief Actuation is interlocked to prevent actuation until the Containment Pressure – Low setpoint is reached. After the operator initiates Manual Containment Vacuum Relief Actuation, the actuation signal is automatically reset when containment pressure rises above the Low setpoint due to the interlock condition no longer being satisfied. Following actuation signal reset, the valves will remain open until closed by a different manual or automatic signal.

Manual Containment Vacuum Relief Valve actuation must be OPERABLE in MODES 1 through 4 and in MODES 5 and 6 without an open containment air flow path ≥ 6 inches in diameter. With a 6 inch diameter or equivalent containment air flow path, the vacuum relief function is not needed to mitigate a low pressure event.

ESFAS Manual Initiation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this specification may be entered independently for each Function listed on Table 3.3.9-1. The Completion Time(s) of the inoperable equipment of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A addresses the inoperability of the system level manual initiation capability for the ESF Functions listed in Table 3.3.9-1. With one channel inoperable for one or more Functions, the system level manual initiation capability is reduced below that required to meet single failure criterion. Required Action A.1 requires the manual initiation channel to be restored to OPERABLE status within 48 hours. The specified Completion Time is reasonable considering that the remaining switch or switch set is capable of performing the safety function.

Condition A is modified by a Note stating that this Condition is not applicable to Functions 1, 6, 7, 8, 12, and 13 in MODE 5 or 6.

B.1

As noted, Condition B addresses the inoperability of one channel of Function 1, 6, 7, 8, 12, and 13 when in MODE 5 or 6. With one channel inoperable for one or more Functions, the system level initiation capability is reduced below that required to meet single failure criterion. Therefore, the required channel must be returned to OPERABLE status within 72 hours. The specified Completion Time is reasonable considering the remaining switch or switch set is capable of performing manual initiation.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.9-1. If the Required Action and the associated Completion Time of Condition A or B are not met or if two channels for one or more Functions are inoperable Condition C is entered to provide for transfer to the appropriate subsequent Condition.

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BASES

ACTIONS (continued)

D.1 and D.2

If the Required Action and associated Completion Time of Condition A are not met or if two channels for one or more Functions are inoperable, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 with the RCS being cooled by the RNS within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

E.1 and E.2

If the Required Action and associated Completion Time of Condition A are not met or if two channels for one or more Functions are inoperable, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

F.1

If the Required Action and associated Completion Time of Condition A are not met or if two channels for one or more Functions are inoperable, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. Under these circumstances, the affected isolation valves(s) must be declared inoperable immediately. Declaring the affected isolation valve inoperable allows the supported system Actions (i.e., for inoperable valves) to dictate the required measures. The respective isolation valve LCO provides appropriate actions for the inoperable components. This actions is in accordance with LCO 3.0.6, which requires that the applicable Conditions and Required Actions for the isolation valves declared inoperable shall be entered in accordance with LCO 3.0.2.

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ACTIONS (continued)

G.1 and G.2

If the Required Action and associated Completion Time of Condition A are not met or if two channels for one or more Functions are inoperable, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by placing the plant in MODE 5 within 12 hours. It is further required to initiate action within 12 hours to restore the required instrument channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function. The 12 hour Completion Time is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems and initiate action to restore compliance.

H.1 and H.2

If the Required Action and associated Completion Time of Condition B are not met or if two channels for one or more Functions are inoperable, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by immediately suspending positive reactivity additions. In addition, action must be immediately initiated to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function.

I.1 and I.2 – Not used.

J.1, J.2, and J.3

If the Required Action and associated Completion Time of Condition B are not met or if two channels for one or more Functions are inoperable, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by immediately suspending positive reactivity additions and placing the plant in MODE 5 within 12 hours (Required Action J.2). The 12 hours is a reasonable time to reach MODE 5 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

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BASES

ACTIONS (continued)

Required Action J.3 requires initiation of action within 12 hours to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function. The 12 hour Completion Time is consistent with the transition to MODE 5 in accordance with Required Action J.2.

K.1 and K.2 – Not used.

L.1, L.2, and L.3

If the Required Action and associated Completion Time of Condition A are not met or if more than two channels are inoperable, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Required Action L.3 requires initiation of action within 36 hours to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function. The 36 hour Completion Time is consistent with the transition to MODE 5 in accordance with Required Action L.2.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

SR 3.3.9.1 is the performance of a TADOT of the manual initiation for the various ESF Functions. This TADOT is performed every 24 months.

The Frequency is based on the known reliability of the ESF Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that states verification of setpoint is not required, since these functions have no setpoint associated with them.

REFERENCES 1. FSAR Chapter 15, “Accident Analysis.”

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B 3.3 INSTRUMENTATION

B 3.3.10 Engineered Safety Feature Actuation System (ESFAS) Reactor Coolant System (RCS) Hot Leg Level Instrumentation

BASES

BACKGROUND

A description of the ESFAS Instrumentation is provided in the Bases for LCO 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. A description of ESFAS P-12 (Pressurizer Level) interlocks is provided in the Bases for LCO 3.3.8. ESFAS protective functions include:

ADS Stage 4 Actuation

A description of the ADS Stage 4 Actuation is provided in the Bases for LCO 3.3.8.

Chemical and Volume Control System (CVS) Letdown Isolation

A description of the Chemical and Volume Control System (CVS) Letdown Isolation is provided in the Bases for LCO 3.3.8.

The following are descriptions of the individual instrument Functions required by this LCO as presented in Table 3.3.10-1. Each Function also provides the ESFAS protective functions actuated by the instrumentation.

1. Hot Leg Level – Low 4

   A signal to initiate the opening sequence of the fourth stage is also generated upon coincident loop 1 and loop 2 hot leg levels below an established setpoint. First, the fourth stage isolation valves receive a confirmatory open signal. Then, the depressurization valves receive an open signal if the hot leg levels remain below the setpoint for a duration exceeding a preset time delay. The ADS provides a sequenced depressurization of the reactor coolant system to allow passive injection from the Core Makeup Tanks (CMTs), accumulators, and the IRWST to mitigate the effects of a LOCA. This Function is required to be OPERABLE in MODE 5 with CMT actuation on Pressurizer Water Level - Low 2 blocked, and in MODE 6 with the upper internals in place and with CMT actuation on Pressurizer Water Level - Low 2 blocked.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2. Hot Leg Level – Low 2

A signal to isolate the Chemical and Volume Control System (CVS) letdown valves is generated upon the occurrence of a Low 2 hot leg level in either of the two RCS hot leg loops. This helps to maintain reactor system inventory in the event of a LOCA. This Function can be blocked in MODES 1, 2, and 3 and is automatically reset when P-12 is first activated. It may be manually reset as well. This Function is required to be OPERABLE in MODE 5 below the P-12 (Pressurizer Level) interlock, and in MODE 6 with the water level < 23 feet above the top of the reactor vessel flange.

ESFAS instrumentation Hot Leg Level satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this specification may be entered independently for each Function listed on Table 3.3.10-1. The Completion Time(s) of the inoperable equipment of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function. Where the required channels are specified on a per loop basis, separate Condition entry is allowed for each loop.

In the event a channel's as-found condition is outside the as-found tolerance described in the Setpoint Program, or the channel is not functioning as required, or the transmitter, or the Protection and Safety Monitoring System Division, associated with a specific Function is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the particular protection Function(s) affected.

A.1 and A.2

With one channel inoperable, the affected channel must be placed in a bypass condition within 6 hours. For Function 1, if one channel is placed in bypass, automatic actuation will not occur. For Function 2, the 6 hours allowed to place the inoperable channel in the bypass condition is justified in Reference 3. If one CVS isolation channel is bypassed, the logic becomes one-out-of-one. A single failure in the remaining channel could cause a spurious CVS isolation. Spurious CVS

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ACTIONS (continued)

isolation, while undesirable, would not cause an upset plant condition. Therefore, Required Action A.2 requires continuous monitoring of the hot leg level. This provides sufficient information to permit timely operator action to ensure that ADS Stage 4 actuation can occur, if needed to mitigate events requiring RCS makeup, boration, or core cooling. Operator action to manually initiate ADS Stage 4 actuation is assumed in the analysis of shutdown events (Ref. 4). It is also credited in the shutdown PRA (Ref. 5) when automatic actuation is not available.

Required Action A.2 is modified by a Note stating that the action is only applicable to Function 1.

B.1

Condition B addresses the situation where the Required Action and associated Completion Time of Condition A is not met. The Required Action is to refer to Table 3.3.10-1 and to take the Required Actions for the protection Functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A are not met for Function 1 in Table 3.3.10-1, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by immediately initiating action to suspend positive reactivity additions (Required Action C.1).

Required Action C.2 requires immediate initiation of action to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function.

D.1.1, D.1.2.1, D.1.2.2, and D.2

If the Required Action and associated Completion Time of Condition A are not met for Function 2 in Table 3.3.10-1, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path within 24 hours. By isolating the CVS letdown flow path from the RCS, the need for automatic isolation is eliminated.

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BASES

ACTIONS (continued)

To assure that the flow path remains closed, the flow path shall be isolated by the use of one of the specified means (Required Action D.1.2.1) or the flow path shall be verified to be isolated (Required Action D.1.2.2). A means of isolating the affected flow path includes at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured, within 7 days. If one of the Required Action D.1.2.1 specified isolation means is not used, the affected flow path shall be verified to be isolated once per 7 days.

This action is modified by a Note allowing the flow path to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

Optionally, if the flow path is not isolated in accordance with Required Actions D.1.1, D.1.2.1 and D.1.2.2, action must be initiated within 12 hours to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function. The 12 hour Completion Time allows the opportunity to isolate the affected flow path in an orderly manner without challenging plant systems (in accordance with Required Action D.1.1).

E.1

If the Required Action and associated Completion Time of Condition A are not met for Function 2 in Table 3.3.10-1, action must be immediately initiated to restore the channel(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

This SR is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor and the integrated protection cabinets (IPC). The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation. Transmitter calibration must be performed consistent with the assumptions of the setpoint methodology. The difference between the current as-found values and the previous as-left values must be consistent with the transmitter drift allowance used in the setpoint methodology.

Interlocks implicitly required to support the Function's OPERABILITY are also addressed by this CHANNEL CALIBRATION. This portion of the CHANNEL CALIBRATION ensures the associated Function is not bypassed when required to be enabled. This can be accomplished by ensuring the interlocks are calibrated properly in accordance with the SP. If the interlock is not automatically functioning as designed, the condition is entered into the Corrective Action Program and appropriate OPERABILITY evaluations performed for the affected Function. The affected Function's OPERABILITY can be met if the interlock is manually enforced to properly enable the affected Function. When an interlock is not supporting the associated Function's OPERABILITY at the existing plant conditions, the affected Function's channels must be declared inoperable and appropriate ACTIONS taken.

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months).

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by a Note. The Note states that this test should include verification that the time constants are adjusted to within limits.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.10.2

This SR ensures the individual channel ESF RESPONSE TIME is less than or equal to the maximum value assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the NTS value at the sensor, to the point at which the equipment reaches the required functional state (e.g., valves in full open or closed position).

In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by a series of overlapping tests and allocated values such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 2) response time.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the allocated sensor, signal processing and actuation logic response times, where approved, with actual response time tests on the remainder of the channel.

ESF RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Testing of the devices is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

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BASES

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”
2. FSAR Chapter 7, “Instrumentation and Controls.”
3. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
4. APP-GW-GLR-004, Rev. 0, “AP1000 Shutdown Evaluation Report,” July 2002.
5. FSAR Chapter 19, “Probabilistic Risk Assessment,” Appendix 19E, “Shutdown Evaluation.”
6. Not used.

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3.3 INSTRUMENTATION

B 3.3.11 Engineered Safety Feature Actuation System (ESFAS) Startup Feedwater Flow Instrumentation

BASES

BACKGROUND

A description of the ESFAS Instrumentation is provided in the Bases for LCO 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents ESFAS protective functions include PRHR Heat Exchanger Actuation.

PRHR is actuated when the Steam Generator Narrow Range Water Level reaches its Low 2 setpoint coincident with indications of Low 2 Startup Feedwater Flows for both SGs, coincident with T_{cold} – High in either loop.

Startup Feedwater Flow – Low 2 uses a one-out-of-two logic on each of the two startup feedwater lines. These two startup feedwater line Functions are required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the Reactor Coolant System (RCS) is not being cooled by the Normal Residual Heat Removal System (RNS). This ensures that PRHR can be actuated in the event of a loss of the normal heat removal systems. In MODE 4 when the RCS is being cooled by the RNS, and in MODES 5 and 6, the steam generators (SGs) are not required to provide the normal RCS heat sink. Therefore, startup feedwater flow is not required, and PRHR actuation on Low 2 startup feedwater flow is not required.

ESFAS Instrumentation Startup Feedwater Flow satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this specification may be entered independently for each startup feedwater line Function. The Completion Time(s) of the inoperable equipment of each startup feedwater line Function will be tracked separately starting from the time the Condition was entered for that Function. Because the Required Channels are specified on a per startup feedwater line basis, separate Condition entry is allowed for each startup feedwater line.

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BASES

ACTIONS (continued)

In the event a channel's as-found condition is outside the as-found tolerance described in the Setpoint Program, or the channel is not functioning as required, or the transmitter, or the Protection and Safety Monitoring System Division associated with a specific Function is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the particular protection Function(s) affected.

A.1

With one or more startup feedwater lines with one startup feedwater channel inoperable, the inoperable channel must be placed in a trip condition within 6 hours. If one channel is tripped, the interlock condition is satisfied. The specified Completion Time is reasonable considering the time required to complete this action.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met or if one or more startup feedwater lines has two channels inoperable, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 with the RCS being cooled by the RNS within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1

This SR is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor and the integrated protection cabinets (IPC). The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the

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BASES

SURVEILLANCE REQUIREMENTS (continued)

Corrective Action Program for further evaluation. Transmitter calibration must be performed consistent with the assumptions of the setpoint methodology. The difference between the current as-found values and the previous as-left values must be consistent with the transmitter drift allowance used in the setpoint methodology.

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months).

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by a Note. The Note states that this test should include verification that the time constants are adjusted to within limits where applicable.

SR 3.3.11.2

This SR ensures the individual channel ESF RESPONSE TIME is less than or equal to the maximum values assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the NTS value at the sensor, to the point at which the equipment reaches the required functional state (e.g., valves in full open or closed position).

In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by a series of overlapping tests and allocated values such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 2) response time.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the allocated sensor, signal processing and actuation logic response times, where approved, with actual response time tests on the remainder of the channel.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

ESF RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Testing of the devices is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”
2. FSAR Chapter 7, “Instrumentation and Controls.”
3. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
4. Not used.

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B 3.3 INSTRUMENTATION

B 3.3.12 Engineered Safety Feature Actuation System (ESFAS) Reactor Trip Initiation

BASES

BACKGROUND

A description of the ESFAS Instrumentation is provided in the Bases for LCO 3.3.8, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation.”

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS protective functions include the ESFAS Reactor Trip Initiation (P-4) function.

There are eight reactor trip breakers with two breakers in each division (a total of four divisions). The P-4 interlock is enabled when the breakers in two-out-of-four divisions are open. Additionally, the P-4 interlock is enabled by all Automatic Reactor Trip Actuations. Additional discussion of P-4 interlock support for ESFAS systems is in LCO 3.3.8. In summary the Functions of P-4 are:

- Trip the main turbine
- Block boron dilution
- Isolate CVS (closes the isolation valves) coincident with a high steam generator (SG) narrow range signal (Table 3.3.8-1, Function 22) to prevent adding makeup water to the SG following a SGTR event
- Isolate startup feedwater control and isolation valves coincident with a high SG narrow range level signal (Table 3.3.8-1, Function 22)
- Isolate main feedwater control valves coincident with Low reactor coolant average temperature (This function is not assumed in safety analysis)
- Isolate main feedwater isolation valves coincident with Low 2 reactor coolant average temperature. (This function is not assumed in safety analysis)

The reactor trip breaker position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable trip setpoint.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Three divisions of this Function must be OPERABLE in MODES 1, 2, 3, and 4 when the reactor may be critical or approaching criticality. This ensures that a single failure will not prevent the protective function, nor will it cause the protective function (with the exception of a limited number of PMS component failures). These MODES (MODES 1, 2, 3, and 4) are also consistent with the Applicability of the various ESFAS Instrument Functions to which the P-4 interlock provides input.

The P-4 Function to trip the main turbine supports the required alternate downstream isolation of each main steam line flow path into the high pressure turbine as required by LCO 3.7.2, “Main Steam Line Flow Path Isolation Valves,” in MODES 1, 2, 3, and 4.

The P-4 Function does not have to be OPERABLE in MODE 4 or 5 to block boron dilution, because Table 3.3.8-1, Function 17, Source Range Neutron Flux Doubling, provides the required block. In MODE 6, the P-4 interlock with the Boron Dilution Block Function is not required, since the unborated water source flow path isolation valves are locked closed in accordance with LCO 3.9.2.

ESFAS Reactor Trip Initiation satisfies Criterion 3 of
10 CFR 50.36(c)(2)(ii).

ACTIONS A.1

With one required division inoperable, the affected division must be restored to OPERABLE status within 6 hours.

Condition A applies to one inoperable required division of the P-4 Interlock. With one required division inoperable, the two remaining OPERABLE divisions are capable of providing the required interlock function, but without a single failure. The P-4 Interlock is enabled when RTBs in two divisions are detected as open. The status of the other inoperable, non-required P-4 division is not significant, since P-4 divisions cannot be tripped or bypassed. In order to provide single failure tolerance, three required divisions must be OPERABLE.

The 6 hours allowed to restore the inoperable division is reasonable based on the capability of the remaining OPERABLE divisions to mitigate all DBAs and the low probability of an event occurring during this interval.

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BASES

ACTIONS (continued)

B.1, B.2, and B.3

If the Required Action and associated Completion Time Condition A is not met, or if two or three required divisions are inoperable, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized.

Under these circumstances, the affected isolation valves(s) must be declared inoperable immediately, per Required Action B.1. Declaring the affected isolation valve inoperable allows the supported system Actions (i.e., for inoperable valves) to dictate the required measures. The respective isolation valve LCO provides appropriate actions for the inoperable components. This actions is in accordance with LCO 3.0.6, which requires that the applicable Conditions and Required Actions for the isolation valves declared inoperable shall be entered in accordance with LCO 3.0.2. Additionally, Required Action B.2 requires that the plant must be brought to MODE 3 within 6 hours and Required Action B.3 requires that the plant be brought to MODE 4 in 12 hours. The allowed time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

SR 3.3.12.1 is the performance of a TADOT of the blocks for the reactor trip (P-4) input from the integrated protection cabinets (IPCs). This TADOT is performed every 24 months.

The Frequency is based on the known reliability of the engineered safety features (ESF) function and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”

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B 3.3 INSTRUMENTATION

B 3.3.13 Engineered Safety Feature Actuation System (ESFAS) Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization

BASES

BACKGROUND

A description of the ESFAS Instrumentation is provided in the Bases for LCO 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

A description of the Main Control Room (MCR) Isolation, Air Supply Initiation, and Electrical Load De-energization is also provided in the Bases for LCO 3.3.8. The ESFAS protective functions required by this LCO, as presented in Table 3.3.13-1, include:

1. Main Control Room Air Supply Iodine or Particulate Radiation – High 2

Two radiation monitoring channels are provided on the main control room (MCR) air intake. Each MCR Air Supply Iodine or Particulate Radiation monitoring channel requires both the iodine radiation monitor and the particulate radiation monitor to be OPERABLE for the MCR Air Supply Iodine or Particulate Radiation channel to be OPERABLE. If either MCR Air Supply Iodine or Particulate Radiation channel exceeds the High 2 setpoint, main control room isolation, air supply initiation, and electrical load de-energization are actuated. Two channels of the Main Control Room Air Supply Iodine or Particulate Radiation - High 2 Function (each with both the iodine and particulate monitors) are required to be OPERABLE in MODES 1, 2, 3, and 4, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident, or other DBA.

The MCR Air Supply Iodine or Particulate Radiation - High 2 channels are not required to be OPERABLE if the Main Control Room Envelope (MCRE) is isolated and the Main Control Room Emergency Habitability System (VES) is in operation. In the event of a Class 1E 24-hour battery charger undervoltage signal, the MCRE is isolated, the VES is initiated, and the MCR air supply radiation monitor sample pumps are de-energized to conserve battery capacity. With the MCR air supply radiation monitor sample pumps de-energized, the MCR Air Supply Iodine or Particulate Radiation - High 2 function would be inoperable. Whenever both the MCRE has been isolated (VBS cannot supply air flow to the MCR) and the VES is providing pressurization and breathing air to the MCRE, the safety function of the MCR Air Supply Iodine or Particulate Radiation - High 2 channels is not required.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2. Main Control Room Differential Pressure – Low

Two differential pressure sensor monitors are provided for the MCR pressure boundary. If either sensor exceeds the Low setpoint for greater than 10 minutes, main control room isolation, air supply initiation, and electrical load deenergization are actuated. Two channels of the Main Control Room Differential Pressure – Low Function are required to be OPERABLE in MODES 1, 2, 3, and 4, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident, or other DBA.

3. Class 1E 24-hour Battery Charger Undervoltage

Two undervoltage sensors are provided on the input to each of the four 24-hour battery charger inputs. The main control room isolation, air supply initiation, and electrical load de-energization is actuated by an undervoltage condition on the Class 1E 24-hour battery chargers for greater than 10 minutes. The logic is based on a two-out-of-two undervoltage to the 24-hour battery charger for divisions A or C coincident with an undervoltage to the 24-hour battery chargers for divisions B or D. When the undervoltage to Class 1E 24-hour battery chargers signal is present, the main control room air supply radiation monitor sample pumps are de-energized.

Two channels per Class 1E 24-hour battery charger are required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel because in the result of a loss of ac power event, the main control room air supply radiation monitor sample pumps are required to be de-energized to conserve battery capacity, maintain room temperature below the equipment qualification limitation, and maintain the assumed heat load for Main Control Room habitability.

ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

ACTIONS

In the event a channel's as-found condition is outside the as-found tolerance described in the SP, or the channel is not functioning as required, or the transmitter, or the Protection and Safety Monitoring System Division, associated with a specific Function is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the particular protection Function(s) affected.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this specification may be entered independently for each Function (i.e., Main Control Room Air Supply Iodine or Particulate Radiation – High 2, Main Control Room Differential Pressure – Low, and Class 1E 24-Hour Battery Charger Undervoltage). The Completion Time(s) of the inoperable Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A addresses one or more Functions with one or more channels inoperable. In this case, the Required Action is to enter the Condition referenced in Table 3.3.13-1 immediately.

B.1 and B.2

Required Action B.1 verifies that one channel in the affected Function is OPERABLE. The Completion Time for Required Action B.1 is immediately because if two channels are inoperable then Condition D is also applicable. With one channel inoperable in either or both Functions in MODE 1, 2, 3, or 4, the logic becomes one-out-of-one in the affected Function and is unable to meet single failure criterion.

Required Action B.2 requires that the main control room isolation, air supply initiation, and electrical load de-energization manual controls must be verified to be functional within 72 hours. These provisions for operator action provide compensation for inoperability of one channel of the system actuation function. The 72 hour Completion Time is reasonable considering that there is one remaining channel OPERABLE and the low probability of an event occurring during this interval.

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BASES

ACTIONS (continued)

C.1 and C.2

Required Action C.1 verifies that one channel in the affected Function is OPERABLE. The Completion Time for Required Action C.1 is immediately because if two channels are inoperable then Condition D is also applicable. With one channel inoperable during movement of irradiated fuel assemblies, the system level initiation capability is reduced below that required to meet single failure criterion. The required channel must be returned to OPERABLE status within 72 hours. The specified Completion Time is reasonable considering the remaining channel is capable of performing the initiation and the low probability of an event occurring during this interval.

D.1 and D.2

If the Required Action and associated Completion Times of Condition B or C are not met, VES must be declared inoperable immediately. Declaring VES inoperable allows the supported system Actions (i.e., TS 3.7.6 Actions for VES inoperable) to dictate the required measures. TS 3.7.6 provides appropriate actions for the inoperability. This action is in accordance with LCO 3.0.6, which requires that the applicable Conditions and Required Actions for VES declared inoperable shall be entered in accordance with LCO 3.0.2.

E.1, E.2.1, and E.2.2

Required Action E.1 allows the inoperable channel to be placed in a trip condition within 6 hours. The Completion Time of 6 hours to place the inoperable channel in trip is reasonable considering the time required to complete this action. Once an inoperable channel is placed in trip, single failure capability of the action is restored and no additional action is required for the inoperable channel. If more than one channel is inoperable at a time, the configuration of inoperable channels in the trip condition may generate an undesirable Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization and de-energize the MCR air supply radiation monitor sample pumps. Therefore, one or more inoperable channels may be left untripped by optionally complying with Required Actions E.2.1 and E.2.2.

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BASES

ACTIONS (continued)

In place of Required Action E.1, the Required Actions E.2.1 and E.2.2 may be performed. Required Action E.2.1 is to verify that actuation capability is maintained. In order to verify actuation capability is maintained, the combination of channels either in trip or OPERABLE must be able to maintain the actuation capability with an undervoltage to the 24-hour battery chargers for divisions A or C coincident with an undervoltage to the 24-hour battery chargers for divisions B or D. Performing the Required Action E.2.1 within the 6 hour Completion Time is a reasonable time to verify that actuation capability is maintained. Required Action E.2.2 also requires restoring the channel to OPERABLE status within 7 days. This reflects a reasonable time to effect restoration of the single failure capability of the undervoltage actuation function.

Condition E is modified by a Note that states separate Condition entry is allowed for each channel. The Required Actions provide appropriate compensatory actions for each inoperable channel. Complying with the Required Actions may allow for continued operation, and subsequent inoperable channels are governed by subsequent Condition entry and application of associated Required Actions. The Completion Time(s) of the inoperable channel will be tracked separately for each channel starting from the time the Condition was entered for that channel.

F.1

If the Required Action and associated Completion Time of Condition E is not met, the plant must be placed in a Condition in which the likelihood and consequences of an event are minimized. This is accomplished by de-energizing both main control room air supply radiation monitor sample pumps within 6 hours. This allowed Completion Time is reasonable considering the time required to complete this action. Note that de-energizing main control room air supply radiation monitor sample pumps results in inoperability of the Main Control Room Air Supply Iodine or Particulate Radiation - High 2 channels and the need to enter other applicable Conditions.

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BASES

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization Function are identified in the SRs column of Table 3.3.13-1 for that Function. A Note has been added to the SR table stating that Table 3.3.13-1 determines which SRs apply to which ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization Function.

SR 3.3.13.1

This SR is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor and the integrated protection cabinets (IPC). The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation. Transmitter calibration must be performed consistent with the assumptions of the setpoint methodology. The difference between the current as-found values and the previous as-left values must be consistent with the transmitter drift allowance used in the setpoint methodology.

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months).

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by a Note. The Note states that this test should include verification that the time constants are adjusted to within limits where applicable.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.13.2

This SR ensures the individual channel ESF RESPONSE TIME is less than or equal to the maximum value assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the NTS value at the sensor, to the point at which the equipment reaches the required functional state (e.g., valves in full open or closed position).

In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by a series of overlapping tests and allocated values such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 1) response time.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the allocated sensor, signal processing and actuation logic response times, where approved, with actual response time tests on the remainder of the channel.

ESF RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Testing of the devices is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

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BASES

REFERENCES

1. FSAR Chapter 7, “Instrumentation and Controls.”
2. FSAR Chapter 15, “Accident Analysis.”
3. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
4. Not used.

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B 3.3 INSTRUMENTATION

B 3.3.14 Engineered Safety Feature Actuation System (ESFAS) In-containment Refueling Water Storage Tank (IRWST) and Spent Fuel Pool Level Instrumentation

BASES

BACKGROUND

A description of the ESFAS Instrumentation is provided in the Bases for LCO 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The required channels of ESFAS IRWST and Spent Fuel Pool Level Instrumentation provide plant protection in the event of a leak in the nonsafety-related Spent Fuel Pool Cooling System (SFS). A description of the ESFAS P-9 (T_{avg}) interlock is provided in the Bases for TS 3.3.8. A description of the two SFS penetration flow paths is provided in the Bases for TS 3.7.13.

The following are descriptions of the individual instrument Functions required by this LCO as presented in Table 3.3.14-1.

1. Spent Fuel Pool Level - Low 2

   Three spent fuel pool level channels are provided. If any two spent fuel pool level channels decrease to below the Low 2 setpoint, Refueling Cavity and SFS Isolation is actuated closing the SFS containment isolation valves to terminate draining of the refueling cavity before reducing the combined IRWST and refueling cavity water inventory below the minimum volume required to provide adequate decay heat removal in MODE 6. With refueling cavity and spent fuel pool volumes in communication in MODE 6, the spent fuel pool level is the same as the refueling cavity, so Refueling Cavity and SFS Isolation on Spent Fuel Pool Level - Low 2 also terminates draining of the spent fuel pool. In MODE 6 with refueling cavity and spent fuel pool volumes not in communication, SR 3.5.8.3 requires the SFS containment isolation valves to be maintained closed, except under administrative controls. With the SFS containment isolation valves closed, the nonsafety-related SFS is isolated from the IRWST and refueling cavity, and a leak in the SFS does not cause draining of the IRWST and refueling cavity. Three channels of ESFAS Spent Fuel Pool Level – Low 2 Function are required to be OPERABLE in MODE 6 with refueling cavity and spent fuel pool volumes in communication.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

2. IRWST Wide Range Level - Low

Two IRWST wide range level channels are provided. If either IRWST wide range level channel decreases to below the Low setpoint, Refueling Cavity and SFS Isolation is actuated closing the SFS containment isolation valves to terminate draining of the IRWST before reducing the IRWST water inventory below the minimum volume required to mitigate the consequences of a design basis event in MODES 1, 2, 3, and 4. The IRWST Wide Range Level - Low isolation signal can be manually blocked by the main control room operator when below the P-9 setpoint, and is automatically unblocked when above the P-9 setpoint. This block is necessary to permit IRWST volume to be transferred under administrative control to and from the refueling cavity during MODE 6 with refueling cavity and spent fuel pool volumes not in communication. In MODE 5 and in MODE 6 with refueling cavity and spent fuel pool volumes not in communication, the isolation valves are maintained closed, except under administrative controls as required by SR 3.5.7.1 and SR 3.5.8.3. In MODE 6 with refueling cavity and spent fuel pool volumes in communication, the Spent Fuel Pool Level - Low 2 isolation signal provides protection. Two channels of IRWST Wide Range Level - Low are required to be OPERABLE in MODES 1, 2, 3, and 4.

ESFAS IRWST and Spent Fuel Pool Level instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.14-1.

In the event a channel's as-found condition is outside the as-found tolerance described in the Setpoint Program, or the channel is not functioning as required, or the transmitter, or the Protection and Safety Monitoring System Division, associated with a specific Function is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the particular protection Function(s) affected.

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BASES

ACTIONS (continued)

A.1

Condition A addresses the situation where one spent fuel pool level channel is inoperable. With one spent fuel pool level channel inoperable, the inoperable channel must be placed in a trip condition within 6 hours. If one of the three spent fuel pool level channels is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The specified Completion Time is reasonable considering the time required to complete this action.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, or one or more IRWST level channels are inoperable, or two or more spent fuel pool level channels are inoperable, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer applicable. To achieve this, the affected penetration flow paths must be isolated within 24 hours.

Additionally, to assure that the penetration flow paths remain closed, the penetration flow paths shall be isolated by the use of one of the specified means (Required Action B.2.1) or the penetration flow paths shall be verified to be isolated (Required Action B.2.2). A means of isolating the penetration flow paths includes at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured within 7 days. If one of the Required Action B.2.1 specified isolation means is not used, the penetration flow paths shall be verified to be isolated once per 7 days.

This action is modified by a Note allowing the penetration flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the penetration flow path(s) can be rapidly isolated when a need for penetration flow path isolation is indicated.

C.1

If the Required Action and associated Completion Time of Condition B is not met, the IRWST must be declared inoperable immediately. Declaring the IRWST inoperable allows the supported system Required Actions (i.e., depending on the MODE, LCO 3.5.6, “In-containment Refueling Water Storage Tank (IRWST) – Operating,” or LCO 3.5.8, “In-containment

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BASES

ACTIONS (continued)

Refueling Water Storage Tank (IRWST) - Shutdown, MODE 6") to dictate the required measures. The IRWST LCO(s) provide appropriate Required Actions for the inoperability of the IRWST and Spent Fuel Pool Level Instrumentation. This action is in accordance with LCO 3.0.6, which requires that the applicable Conditions and Required Actions for the IRWST declared inoperable shall be entered in accordance with LCO 3.0.2.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

This SR is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor and the integrated protection cabinets (IPC). The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation. Transmitter calibration must be performed consistent with the assumptions of the setpoint methodology. The difference between the current as-found values and the previous as-left values must be consistent with the transmitter drift allowance used in the setpoint methodology.

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months).

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by a Note. The Note states that this test shall include verification that the time constants are adjusted to within limits where applicable.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.14.2

This SR ensures the individual channel ESF RESPONSE TIME is less than or equal to the maximum value assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the NTS value at the sensor, to the point at which the equipment reaches the required functional state (e.g., valves in full open or closed position).

In lieu of measurement, the response time for the protection and safety monitoring system equipment is based on allocated values. The overall response time may be determined by a series of overlapping tests and allocated values such that the entire response time is verified, with the resulting verified response time compared to the appropriate FSAR Chapter 7 (Ref. 2) response time.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the allocated sensor, signal processing and actuation logic response times, where approved, with actual response time tests on the remainder of the channel.

ESF RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Testing of the devices is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

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BASES

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”
2. FSAR Chapter 7, “Instrumentation and Controls.”
3. APP-GW-GSC-020, “Technical Specification Completion Time and Surveillance Frequency Justification.”
4. Not used.

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B 3.3 INSTRUMENTATION

B 3.3.15 Engineered Safety Feature Actuation System (ESFAS) Actuation Logic - Operating

BASES

BACKGROUND

A description of the ESFAS Instrumentation and ESFAS Actuation Logic Functions is provided in the Bases for LCO 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The required divisions of ESFAS Actuation Logic Functions provide plant protection in the event of any of the analyzed accidents.

A description of the Engineered Safety Features (ESF) Coincidence Logic and ESF Actuation Logic is provided in the Bases for LCO 3.3.8.

The following are descriptions of the ESFAS Actuation Logic Functions required by this LCO:

a. ESF Coincidence Logic

This LCO requires four divisions of ESF Coincidence Logic (note that consideration of any “pre-vote” interlock logic is addressed by LCO 3.3.8 instrument channel OPERABILITY requirements), each division with one battery backed logic group (i.e., subsystem) be OPERABLE to support automatic actuation. If one battery backed ESF Coincidence Logic subsystem is OPERABLE in all four divisions, a single failure will not prevent ESF actuations because the ESF Coincidence Logic subsystems in the other three divisions will still be available to provide redundant actuation for all ESF functions. For the CMT Actuation function only, both PMS Local Coincidence Logic (LCL) processors in each division must be capable of functioning to maintain a single failure proof design, because the same two divisions of PMS are required to actuate both outlet valves for a given CMT. Therefore, if one of the two redundant PMS LCLs in a division is taken out of service or has no output or a bad quality signal, the CMT outlet valve component interface modules (CIMs) in the affected division must be locally placed in an actuate condition. Otherwise Condition A will be entered for CMT actuation function until the CIMs are placed in a local actuate condition or the division is restored.

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4. The ESF Coincidence Logic Function requirements for MODES 5 and 6 are discussed in LCO 3.3.16, "ESFAS Actuation Logic Function - Shutdown."

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

b. ESF Actuation

This LCO requires four divisions of ESF Actuation (including consideration of any “post-vote” interlock logic), each division with one battery backed logic group (i.e., subsystem) be OPERABLE to support ESF actuations. If one battery backed ESF Actuation logic subsystem is OPERABLE in all four divisions, a single failure will not prevent ESF actuations because ESF Actuation subsystems in the other three divisions are still available to provide redundant actuation for ESF functions.

This Function is required to be OPERABLE in MODES 1, 2, 3, and 4. The ESF Actuation Function requirements for MODES 5 and 6 are discussed in LCO 3.3.16.

ESFAS Actuation Logic - Operating satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this specification may be entered independently for each Function (i.e., ESF Coincidence Logic and ESF Actuation). The Completion Time(s) of the inoperable equipment of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A addresses the situation where one or more ESFAS actuation logic Functions within one division are inoperable. The ESF Coincidence Logic and ESF Actuation subsystem divisions are inoperable when both their associated battery backed subsystems are inoperable. The ESF coincidence logic is also inoperable when one of the two redundant PMS LCLs in a division is inoperable, until the CMT outlet valve CIMs in the affected division are locally placed in an actuate condition or the division is restored.

With one ESFAS actuation logic division inoperable, the inoperable division must be restored to OPERABLE status within 6 hours. With one division inoperable, the three remaining OPERABLE divisions are capable of mitigating all DBAs, but without a single failure.

The 6 hours allowed to restore the inoperable division is reasonable based on the capability of the remaining OPERABLE divisions to mitigate all DBAs and the low probability of an event occurring during this interval.

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BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, or one or more ESFAS actuation logic Functions within two or more divisions are inoperable, the plant must be placed in a condition where the Function is no longer applicable. This is accomplished by placing the plant in MODE 3 within 6 hours, and in MODE 5 within 36 hours . The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Due to self-checking features that continuously monitor logic OPERABILITY and alert the operator to any failures, there are no Surveillances required for ESF Coincidence Logic. Similarly, there are no Surveillances required for ESF Actuation.

REFERENCES None.

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B 3.3 INSTRUMENTATION

B 3.3.16 Engineered Safety Feature Actuation System (ESFAS) Actuation Logic - Shutdown

BASES

BACKGROUND

A description of the ESFAS Instrumentation and ESFAS Actuation Logic is provided in the Bases for LCO 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The required divisions of ESFAS Actuation Logic Functions provide plant protection in the event of any of the analyzed accidents.

A description of the Engineered Safety Features (ESF) Coincidence Logic and ESF Actuation Logic is provided in the Bases for LCO 3.3.8.

The following are descriptions of the ESFAS Actuation Logic Functions required by this LCO:

a. ESF Coincidence Logic

This LCO requires four divisions of ESF Coincidence Logic (note that consideration of any “pre-vote” interlock logic is addressed by LCO 3.3.8 instrument channel OPERABILITY requirements), each division with one battery backed logic group (i.e., subsystem) be OPERABLE to support automatic actuation. If one battery backed ESF Coincidence Logic subsystem is OPERABLE in all four divisions, an additional single failure will not prevent ESF actuations because the ESF Coincidence Logic subsystems in the other three divisions will still be available to provide redundant actuation for all ESF functions. For the CMT Actuation function only, both PMS Local Coincidence Logic (LCL) processors in each division must be capable of functioning to maintain a single failure proof design, because the same two divisions of PMS are required to actuate both outlets valves for a given CMT. Therefore, if one of the two redundant PMS LCLs in a division is taken out of service or has no output or a bad quality signal, the CMT outlet valve component interface modules (CIMs) in the effected division must be locally placed in an actuate condition. Otherwise Condition A will be entered for CMT actuation function until the CIMs are placed in a local actuate condition or the division is restored.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

This Function is required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident, or other DBA.

The ESF Coincidence Logic requirements for MODES 1, 2, 3, and 4 are discussed in LCO 3.3.15, "ESFAS Actuation Logic - Operating."

b. ESF Actuation

This LCO requires four divisions of ESF Actuation (including consideration of any “post-vote” interlock logic), each division of ESF actuation, with one battery backed logic group be OPERABLE to support ESF actuations. If one battery backed ESF Actuation logic subsystem is OPERABLE in all four divisions, a single failure will not prevent ESF actuations because ESF Actuation subsystems in the other three divisions are still available to provide redundant actuation for ESF functions.

This Function is required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident, or other DBA. The LCO is modified by a Note stating that only the ESFAS Actuation divisions necessary to support Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization are required to be OPERABLE during movement of irradiated fuel assemblies when not in MODE 1, 2, 3, 4, 5, or 6. This supports TS 3.3.13, “Engineered Safety Feature Actuation System (ESFAS) Main Control Room Isolation, Air Supply Initiation, and Electrical Load De energization Instrumentation,” actuation of Main Control Room Emergency Habitability System (VES).

The ESF Actuation Function requirements for MODES 1, 2, 3, and 4 are discussed in LCO 3.3.15.

The LCO is modified by a Note stating that only the divisions necessary to support Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization are required to be OPERABLE during movement of irradiated fuel assemblies when not in MODE 1, 2, 3, 4, 5, or 6. This supports TS 3.3.13, "Engineered Safety Feature Actuation System (ESFAS) Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization Instrumentation," actuation of Main Control Room Emergency Habitability System (VES).

ESFAS Actuation Logic - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this specification may be entered independently for each Function (i.e., ESF Coincidence Logic and ESF Actuation). The Completion Time(s) of the inoperable equipment of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A addresses the situation where one or more ESFAS actuation logic Functions within one division are inoperable. The ESF Coincidence Logic and ESF Actuation subsystem divisions are inoperable when both their associated battery backed subsystems are inoperable. The ESF Coincidence Logic is also inoperable when one of the two redundant PMS LCLs in a division is inoperable, until the CMT outlet valve CIMs in the effected division are locally placed in an actuate condition or the division is restored.

With one ESFAS actuation logic division inoperable, the initiation capability is reduced below that required to meet the single failure criterion. Therefore, the required division must be returned to OPERABLE status within 72 hours. The specified Completion Time is reasonable considering the remaining divisions are capable of performing the associated safety function.

B.1, B.2, and B.3

If the Required Action and associated Completion Time of Condition A is not met in MODE 5, or one or more ESFAS actuation logic Functions within two divisions are inoperable, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by immediately suspending positive reactivity additions (Required Action B.1). In addition, action must be immediately initiated to restore the Function(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function.

Action must also be immediately initiated to isolate the flow path from the demineralized water storage tank to the RCS by use of at least one closed and de-activated automatic valve or closed manual valve (Required Action B.3). This requirement minimizes the potential for a criticality event by isolation of the demineralized water storage tank.

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BASES

C.1 and C.2

If the Required Action and associated Completion Time of Condition A is not met in MODE 6, or one or more ESFAS actuation logic Functions within two or more divisions are inoperable, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. This is accomplished by immediately suspending positive reactivity additions. In addition, action must be immediately initiated to restore the Function(s) to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary channels for the Function.

D.1

If the Required Action and associated Completion Time of Condition A is not met during movement of irradiated fuel assemblies, or one or more ESFAS actuation logic Functions within two or more divisions are inoperable, the plant must be placed in a condition in which the likelihood and consequences of an event are minimized. Required Action D.1 requires immediately suspending movement of irradiated fuel assemblies. This required action suspends activities with potential for releasing radioactivity that might enter the Main Control Room. This action does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

Due to self-checking features that continuously monitor logic OPERABILITY and alert the operator to any failures, there are no Surveillances required for ESF Coincidence Logic. Similarly, there are no Surveillances required for ESF Actuation.

REFERENCES None.

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B 3.3 INSTRUMENTATION

B 3.3.17 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM Instrumentation is to display unit variables that provide information required by the main control room operators during accident situations. These plant variables provide the necessary information to assess the process of accomplishing or maintaining critical safety functions. The instruments which monitor these variables are designated in accordance with Reference 1.

The OPERABILITY of the PAM Instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Reference 1.

A PAM CHANNEL shall extend from the sensor up to the display device, and shall include the sensor (or sensors), the signal conditioning, any associated datalinks, the display device, any signal gathering or processing subsystems, and any data processing subsystems. Note that for digital PAM CHANNELs, the information may be displayed on multiple display devices. For this case, the PAM CHANNEL shall extend to any available qualified display device. The two OPERABLE channels must be from Protection and Safety Monitoring System Divisions B and C, except for valves and electrical breakers powered from the 24 hour battery supplies of the Class 1E dc and uninterruptible power supply system. Once the 24 hour battery supply associated with the actuation signal for these valves and electrical breakers has been exhausted, remote capability for changing the position of these components is lost, and there is no need for further position monitoring.

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables. The unit specific implementation of Regulatory Guide 1.97 has not identified any Type A variables, therefore, only Category 1 variables are specified.

Category 1 variables are the key variables deemed risk significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions; and
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release.

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BASES

APPLICABLE SAFETY ANALYSES

The PAM Instrumentation ensures that the main control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM Instrumentation that is required in accordance with Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for those monitors which provide information required by the control room operators to assess the process of accomplishing or maintaining critical safety functions. This LCO addresses those Regulatory Guide 1.97 instruments which are listed in Table 3.3.17-1.

The OPERABILITY of the PAM Instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status following an accident. This capability is consistent with the recommendations of Reference 1.

Category 1 non-type A variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 1) design and qualification requirements for seismic and environmental qualification, single-failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument functions listed in Table 3.3.17-1. Each of these is a Category 1 variable.

1. Intermediate Range Neutron Flux

   Neutron Flux indication is provided to verify reactor shutdown. The neutron flux intermediate range is sufficient to cover the full range of flux that may occur post accident.

   Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

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BASES

LCO (continued)

2, 3. Reactor Coolant System (RCS) Wide Range Hot and Cold Leg Temperature

RCS Hot and Cold Leg Temperatures are provided for verification of core cooling and long-term surveillance. The channels provide indication over a range of 50°F to 700°F.

In addition to this, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the plant conditions necessary to establish natural circulation in the RCS.

4. RCS Pressure

RCS wide range pressure is provided for verification of core cooling and RCS integrity long term surveillance.

5. RCS Subcooling

RCS Subcooling is calculated from core exit temperature and RCS wide range pressure. RCS Subcooling is provided for verification of core cooling. Subcooling margin is available when the RCS pressure is greater than the saturation pressure corresponding to the core exit temperature.

6. Containment Water Level

Containment Water Level is used to monitor the containment environment during accident conditions. The containment water level can also provide information to the operators that the various stages of safety injection along with system depressurization are progressing.

7. Containment Pressure

The containment pressure transmitters monitor the containment pressure over the range of -5 to 10 psig. This provides information on post accident containment pressure and containment integrity.

8. Containment Pressure (Extended Range)

The extended range containment pressure transmitters are instruments that operators use for monitoring the potential for breach of containment, a fission product barrier. The extended range sensors monitor containment pressure over the range of 0 to 240 psig.

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BASES

LCO (continued)

9. Containment Area Radiation (High Range)

   Containment Area Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

10. Pressurizer Level and Associated Reference Leg Temperature

    Pressurizer level is provided to monitor the RCS coolant inventory. During an accident, operation of the safeguards systems can be verified based on coolant inventory indicators.

    The reference leg temperature is included in the Technical Specification since it is used to compensate the level signal. The pressurizer instruments for RCS wide range pressure and pressurizer pressure also provide input to compensate the level signal.

11. In-Containment Refueling Water Storage Tank (IRWST) Wide Range Water Level

    The IRWST provides a long term heat sink for non-LOCA events and is a source of injection flow for LOCA events. When the IRWST is a heat sink, the level will change due to increased volume associated with the temperature increase. When saturation temperature is reached, the IRWST will begin steaming and initially lose mass to the containment atmosphere until condensation occurs on the steel containment shell which is cooled by the passive containment cooling system. The condensate is returned to the IRWST via a gutter and downspouts.

    During a LOCA, the IRWST is available for injection. Depending on the severity of the event, when a fully depressurized RCS has been achieved, the IRWST will inject by gravity flow.

12. Passive Residual Heat Removal (PRHR) Heat Removal

    Two channels of PRHR Flow are provided to monitor primary system heat removal during accident conditions when the steam generators are not available. PRHR Heat Removal provides protection for non LOCA events when the normal heat sink is lost.

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BASES

LCO (continued)

Alternatively, two channels of not closed position indication for the PRHR HX control valves are also provided to monitor primary system heat removal during accident conditions when the steam generators are not available. These two channels, along with the confirmation that the PRHR HX inlet isolation valve is open per SR 3.5.4.2, are sufficient to verify that PRHR heat removal is fulfilled.

The PRHR HX control valves are redundant to each other; they are located in parallel flow paths and only one valve is required to open to satisfy the heat sink function.

Note also that the flow channels and valve position channels cannot be mixed.

13, 14, 15, 16. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

An evaluation was made of the minimum number of valid core exit thermocouples necessary for In-Core Cooling (ICC) detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate ICC detection is assured with two valid core exit thermocouples per quadrant. Core Exit Temperature is also used for plant stabilization and cooldown monitoring.

Two OPERABLE channels of Core Exit Temperature are required in each quadrant (defined below) to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two thermocouples in each of the two divisions ensure a single failure will not disable the ability to determine the temperature at two locations within a quadrant.

Core Exit Temperature quadrants are defined as shown in Figures B 3.3.17-1 and B 3.3.17-2. Either the normal or alternate quadrant definitions and associated core exit thermocouple assignments can be used to determine OPERABILITY.

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BASES

LCO (continued)

17. Passive Containment Cooling System (PCS) Heat Removal

The PCS Heat Removal must be capable of removing the heat from the containment following a postulated LOCA or steam line break (SLB). Two tank level instruments provide indication that sufficient water is available to meet this requirement. The PCS flow instrument on the lowest standpipe, which is also an input to the summation representing total PCS flow, provides a diverse indication of the PCS heat removal capability. The lowest standpipe indication can be used to satisfy one of the two required channels when the PCS level channel in the same electrical division is inoperable. When available, the flow from the other three standpipes is combined with the lowest standpipe to provide a total flow measurement. However, these other standpipe instruments are not required.

18. Penetration Flow Path Remotely Operated Containment Isolation Valve Position

The Penetration Flow Path Remotely Operated Containment Isolation Valve Position is provided to support verification of containment integrity by confirming flowpaths without accident mitigation function to be open (non-essential) are closed. Table 3.3.17-1 Note (d) states that the Function is limited to components that receive an ESF containment isolation signal. Only nonessential systems are isolated automatically by the containment isolation signal. SGS-PL-V075A/B are not containment isolation valves, but are conservatively included because they receive the T signal. SGS-PL-V075A/B are in series with containment isolation valves SGS-PL-V074A/B that also receive the T signal, and are functionally equivalent. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each valve in a containment penetration flow path actuated on a containment isolation signal, i.e., two total channels of valve position indication for a penetration flow path with two active valves. For containment penetrations with only one active valve having post-accident monitoring control room indication, Note (c) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each nonessential, isolable penetration either via indicated status of the active valve, as applicable, and prior knowledge of a passive valve, or via system boundary status. If a normally active valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this

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BASES

LCO (continued)

state is not required to be OPERABLE. Note (b) to the Required Channels states that the Function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. Each penetration is treated separately and each penetration flow path is considered a separate function. Therefore, separate Condition entry is allowed for each penetration flow path with one or more inoperable position indicators.

19. IRWST to RNS Suction Valve Status

The position of the two motor-operated valves in the line from the IRWST to the RNS pump suction header is monitored to verify that the flow path is isolated following postulated events. The flow path must be isolated to prevent loss of IRWST inventory into the RNS. Valve position indication is provided from the external stem-mounted limit switch on the motor-operated valve in the line from the IRWST to the RNS pump suction header located inside reactor containment. Valve position indication is provided from the internal limit switch on the motor-operated valve in the line from the IRWST to the RNS pump suction header located outside reactor containment.

20. Pressurizer Pressure

Pressurizer Pressure is used to provide the control room operating staff with information to assess reactor coolant system inventory control.

APPLICABILITY The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables provide the information necessary to assess the process of accomplishing or maintaining critical safety functions following Design Basis Accidents (DBAs). The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

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BASES

ACTIONS

The ACTIONS Table has been modified by a Note that clarifies the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.17-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that function.

A.1

When one or more Functions have one required channel which is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

The Required Action directs actions to be taken in accordance with Specification 5.6.5 immediately. Each time an inoperable channel has not met Required Action A.1, and the associated Completion Time has expired, Condition B is entered.

C.1

When one or more Functions have two required channels which are inoperable, (two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information.

Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM function will be in a degraded condition should an accident occur.

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BASES

ACTIONS (continued)

D.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.17-1. The applicable Condition referenced in the Table is Function dependent.

Each time an inoperable channel has not met any Required Action of Condition C, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C are not met for the Functions in Table 3.3.17-1, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for each PAM Function are identified in the SRs column of Table 3.3.17-1 for that Function. A Note has been added to the SR table stating that Table 3.3.17-1 determines which SRs apply to each PAM Function.

SR 3.3.17.1

Performance of the CHANNEL CHECK once every 31 days verifies that a gross instrumentation failure has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal-processing equipment has drifted outside its limit. If the channels are within the match criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Surveillance Requirement is modified by a Note. The Note states that this SR is not required for Neutron Flux (Intermediate Range) in MODE 1. In MODE 1, the OPERABILITY of the Intermediate Range Neutron Instrumentation is verified by self-checking features in lieu of performing a CHANNEL CHECK.

The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the required channels of this LCO.

SR 3.3.17.2

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. This SR is modified by a Note that excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.3, "Reactor Trip System (RTS) Intermediate Range Instrumentation." RTD and Thermocouple channels are to be calibrated in place using cross-calibration techniques. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. Regulatory Guide 1.97, Rev. 3, “Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” U.S. Nuclear Regulatory Commission.

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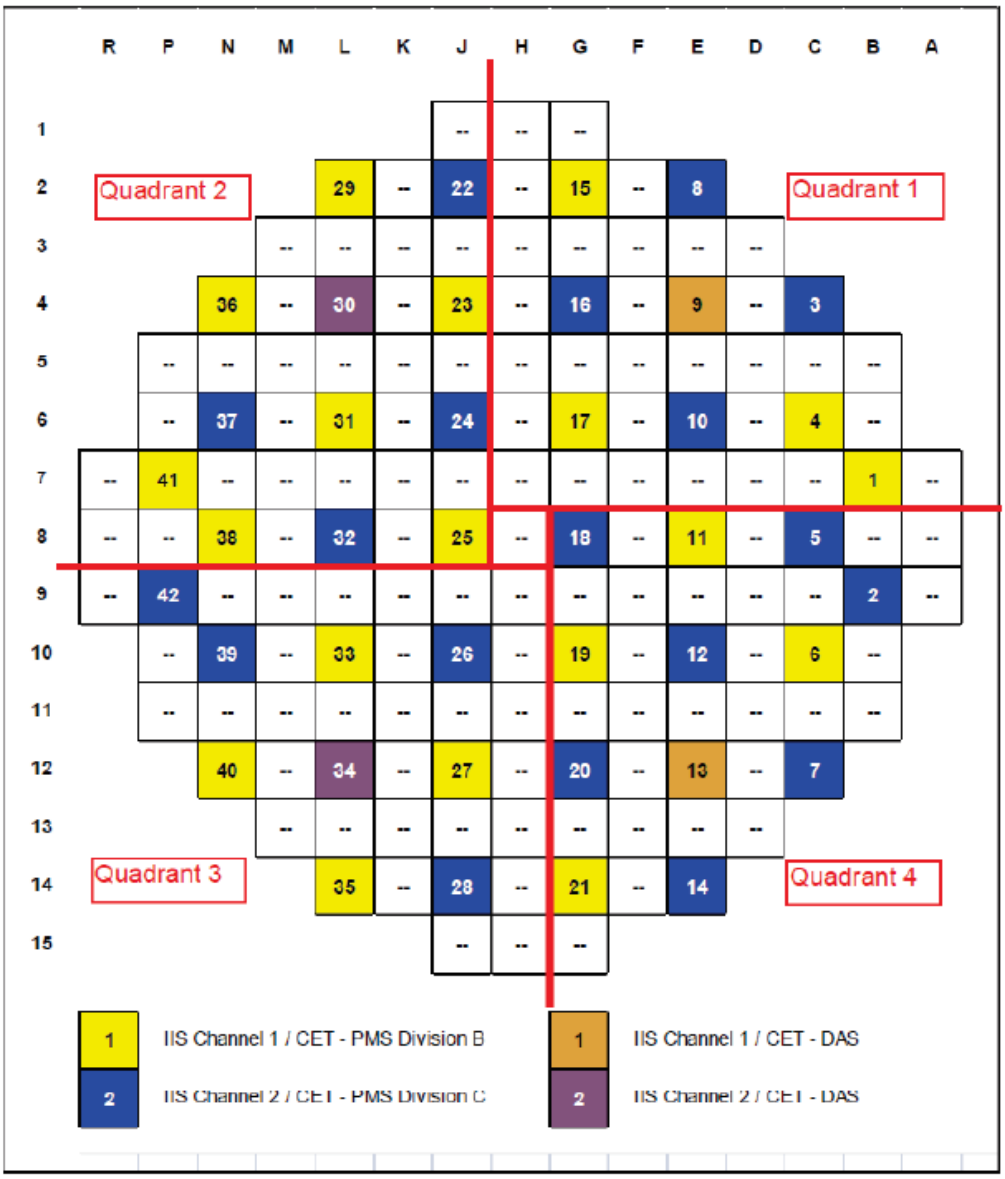

Figure B 3.3.17-1

Normal Core Exit Temperature Quadrant Definition

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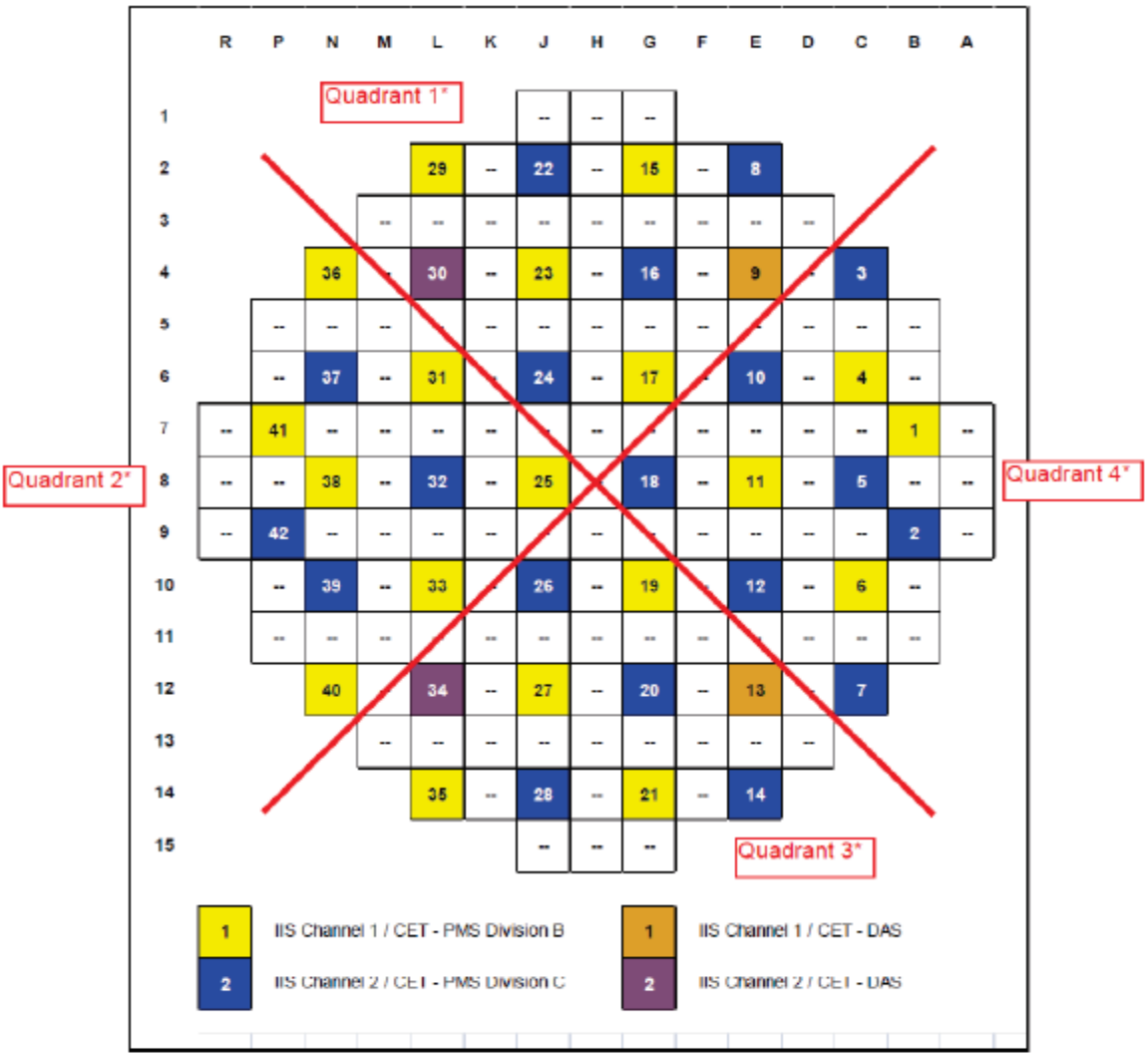

Figure B 3.3.17-2

Alternate Core Exit Temperature Quadrant Definition

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B 3.3 INSTRUMENTATION

B 3.3.18 Remote Shutdown Workstation (RSW)

BASES

BACKGROUND

The RSW provides the control room operator with sufficient displays and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. Passive residual heat removal (PRHR), the core makeup tanks (CMTs), and the in-containment refueling water storage tank (IRWST) can be used to remove core decay heat. The use of passive safety systems allows extended operation in MODE 4.

If the control room becomes inaccessible, the operators can establish control at the RSW and place and maintain the unit in MODE 4 with T_{avg} < 350°F. The unit can be maintained safely in MODE 4 with T_{avg} < 350°F for an extended period of time.

The OPERABILITY of the remote shutdown control and display functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 4 with T_{avg} < 350°F should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The RSW is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 4 with T_{avg} < 350°F.

The criteria governing the design and the specific system requirements of the RSW are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

Since the passive safety systems alone can establish and maintain safe shutdown conditions for the unit, nonsafety systems are not required for safe shutdown of the unit. Therefore, no credit is taken in the safety analysis for nonsafety systems.

The RSW satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The RSW LCO provides the OPERABILITY requirements of the displays and controls necessary to place and maintain the unit in MODE 4 from a location other than the control room.

The RSW is OPERABLE if the display instrument and control functions needed to support the RSW are OPERABLE.

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BASES

LCO (continued)

The RSW covered by this LCO does not need to be energized to be considered OPERABLE. This LCO is intended to ensure the RSW will be OPERABLE if unit conditions require that the RSW be placed in operation.

APPLICABILITY

The RSW LCO is applicable in MODES 1, 2, and 3 and in MODE 4 with $T_{avg} \geq 350°F$. This is required so that the facility can be placed and maintained in MODE 4 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4 with $T_{avg} < 350°F$ or in MODE 5 or 6. In these MODES, the unit is already subcritical and in a condition of reduced Reactor Coolant System (RCS) energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

A.1

Condition A addresses the situation where the RSW is inoperable. The Required Action is to restore the RSW to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 with $T_{avg} < 350°F$ within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.18.1

SR 3.3.18.1 verifies that each required RSW transfer switch performs the required functions. This ensures that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 4 with T_{avg} < 350°F from the RSW. This Surveillance also demonstrates that the ADS and IRWST injection block is unblocked by the RSW transfer switch.

The 24 month Frequency was developed considering it is prudent that these types of surveillances be performed during a unit outage. However, this surveillance is not required to be performed only during a unit outage. This is due to the unit conditions needed to perform the surveillance and the potential for unplanned transients if the surveillance is performed with the reactor at power. Operating experience demonstrates that RSW transfer switches usually pass the surveillance test when performed on the 24 month Frequency.

SR 3.3.18.2

This Surveillance verifies that the RSW communicates controls and indications with Divisions A, B, C, and D of the PMS. The operator can select the controls and indications available through each PMS division.

The Frequency is based on the known reliability of the Functions and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.18.3

SR 3.3.18.3 verifies the OPERABILITY of the RSW hardware and software by performing diagnostics to show that operator displays are capable of being called up and displayed to an operator at the RSW. The RSW has several video display units which can be used by the operator. The video display units are identical to that provided in the control room and the operator can display information on the video display units in a manner which is identical to the way the information is displayed in the control room. The operator normally selects an appropriate set of displays based on the particular operational goals being controlled by the operator at the time. Each display consists of static graphical and legend information which is contained within the display processor associated with each video display unit and dynamic data which is updated by the data display system.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of 24 months is based on the use of the data display capability in the control room as part of the normal unit operation and the availability of multiple video display units at the RSW. The Frequency of 24 months is based upon operating experience and consistency with control room hardware and software.

SR 3.3.18.4

SR 3.3.18.4 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) every 24 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the RSW by actuating the RTBs. The Frequency of 24 months was chosen because the RTBs may not be exercised while the facility is at power and is based on operating experience and consistency with the refueling outage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.
2. FSAR Section 7.4.1, “Safe Shutdown.”

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B 3.3 INSTRUMENTATION

B 3.3.19 Diverse Actuation System (DAS) Manual Controls

BASES

BACKGROUND

The Diverse Actuation System (DAS) manual controls provide non-Class 1E backup controls in case of common-mode failure of the Protection and Safety Monitoring System (PMS) automatic and manual actuations evaluated in the PRA. These DAS manual controls are not credited for mitigating accidents in the FSAR Chapter 15 analyses.

The specific DAS controls were selected based on PRA risk importance as discussed in Reference 1. As noted in Reference 1, electrical power for these controls and instrument indications need not be covered by Technical Specifications. The rational is that these controls use the same nonsafety-related power supply used by the plant control system. This power is required to be available to support normal operation of the plant. With offsite power available, there are several sources to provide this power including AC power to non-Class 1E battery chargers, AC power to rectifiers, and non-Class 1E batteries. As a result, with offsite power available it is very likely that power will be available for these DAS controls. If offsite power is not available, then there is still the likelihood that the non-1E batteries or the non-1E diesel generators will be available. Even if these sources are unavailable, the desired actions will occur without operator action for the more probable events. The rods will insert automatically on loss of offsite power. The passive residual heat removal heat exchanger (PRHR HX), core makeup tanks (CMT), passive containment cooling system (PCS), and containment isolation features are initiated by operation of fail-safe, air-operated valves. If all offsite and onsite AC power is lost, the instrument air system will depressurize by the time these functions are needed in the 1-hour time frame.

Instrument readouts are expected to be available even in case of complete failure of the PMS due to common cause failure. These instruments include both DAS and PLS instruments. They are powered by DC sources for 24 or 72 hours following a loss of AC power, as described in FSAR Section 8.3.2. As discussed above, it is expected that AC power will be available to power the instruments. Even if the operators have no instrument indications, they are expected to actuate the controls most likely to be needed (PRHR HX, CMT, PCS, and containment isolation). If all AC power fails, then the rods will drop and the air-operated valves will go to their fail-safe positions.

The DAS uses equipment from sensor output to the final actuated device that is diverse from the PMS to automatically initiate a reactor trip, or to manually actuate the identified safety-related equipment. FSAR Section 7.7.1.11 (Ref. 2) provides a description of the DAS.

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BASES

APPLICABLE SAFETY ANALYSES

The DAS manual controls are required to provide a diverse capability to manually trip the reactor and actuate the specified safety-related equipment, based on risk importance in the PRA.

The DAS manual controls are not credited for mitigating accidents in the FSAR Chapter 15 safety analyses.

The PRA, Appendix A, provides additional information, including the thermal and hydraulic analyses of success sequences used in the PRA.

The DAS manual controls satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The DAS LCO provides the requirements for the OPERABILITY of the DAS manual trip and actuation controls necessary to place the reactor in a shutdown condition and to remove decay heat in the event that the PMS automatic actuation and manual controls are inoperable.

Two switches are required for each of the DAS Manual Control Functions. These switches correspond to the available switches on the Main Control Room DAS Panel. For the function of containment recirculation valves, the required switches can be met by either the two switches for Containment Recirculation or the two switches for Containment Recirculation / IRWST Drain.

APPLICABILITY

The DAS manual controls are required to be OPERABLE in the MODES specified in Table 3.3.19-1.

The manual DAS reactor trip control is required to be OPERABLE in MODES 1 and 2 to mitigate the effects of an ATWS event occurring during power operation.

The other manual DAS actuation controls are required to be available in the plant MODES specified, based on the need for operator action to actuate the specified components during events that may occur in these various plant conditions, as identified in the PRA.

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BASES

ACTIONS

A.1

Condition A applies when one or more DAS manual controls are inoperable.

The Required Action A.1 to restore the inoperable DAS manual control(s) to OPERABLE status within 30 days is reasonable because the DAS is a separate and diverse non-safety backup system for the manual reactor trip and manual safety-related equipment actuation controls. The 30-day Completion Time allows sufficient time to repair an inoperable manual DAS control but ensures the control is repaired to provide backup protection.

B.1 and B.2

Condition B applies when Required Action A cannot be completed for the DAS manual reactor trip control within the required completion time of 30 days.

Required Action B.1 requires SR 3.3.7.1, “Perform TADOT” for the reactor trip breakers, to be performed once per 31 days, instead of once every 92 days. Condition A of Example 1.3-6 illustrates the use of the Completion Time for Required Action B.1. The initial performance of SR 3.3.7.1 on the first division (since it is performed on a STAGGERED TEST BASIS) must be completed within 31 days of entering Condition B. The normal surveillance test frequency requirements for SR 3.3.7.1 must still be satisfied while performing SR 3.3.7.1 for Required Action B.1. The predominant failure requiring the DAS manual reactor trip control is common-mode failure of the reactor trip breakers. This change in surveillance frequency for testing the reactor trip breakers increases the likelihood that a common-mode failure of the reactor trip breakers would be detected while the DAS manual reactor trip control is inoperable. This reduces the likelihood that a diverse manual reactor trip is required. It is not required to perform a TADOT for the manual actuation control. The manual reactor trip control is very simple, highly reliable, and does not use software in the circuitry. Although the DAS manual controls are non-Class 1E, they have been shown to be PRA risk important as discussed in Reference 1. The impact of an inoperable DAS manual control is compensated for by increasing the reactor trip breaker surveillance frequency from once every 92 days to once every 31 days.

Action B.2 requires that the inoperable DAS manual reactor trip control be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

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BASES

ACTIONS (continued)

C.1

Condition C applies when Required Action A cannot be completed for any DAS manual actuation control (other than reactor trip) within the required completion time of 30 days.

Action C.1 requires that the inoperable DAS manual actuation control(s) be restored to OPERABLE status prior to entering MODE 2 following any plant shutdown to MODE 5 while the control is inoperable. This ACTION is provided to ensure that all DAS manual controls are restored to OPERABLE status following the next plant shutdown.

D.1 and D.2

Condition D is entered if the Required Action associated with Condition B or C is not met within the required Completion Time.

Required Actions D.1 and D.2 ensure that the plant is placed in a condition where the probability and consequences of an event are minimized. The allowed Completion Times are reasonable based on plant operating experience, for reaching the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.19.1

SR 3.3.19.1 is the performance of a TADOT of the DAS manual trip and actuation controls for the specified safety-related equipment. This TADOT is performed every 24 months.

The Frequency is based on the known reliability of the DAS functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of the setpoints from the TADOT. The functions have no setpoints associated with them.

REFERENCES

1. WCAP-15985, “AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process,” Revision 2, dated August 2003.

2. FSAR Section 7.7.1.11.

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B 3.3 INSTRUMENTATION

B 3.3.20 Automatic Depressurization System (ADS) and In-containment Refueling Water Storage Tank (IRWST) Injection Blocking Device

BASES

BACKGROUND

The ADS and IRWST Injection Blocking Device is provided to minimize the likelihood of spurious ADS and IRWST injection valve actuation. The ADS and IRWST injection block function logic is performed by devices, one in each PMS division, that provide an input to the ADS and IRWST injection valve component interface module (CIM) in the same division. The devices provide an independent block of the ADS and IRWST injection valves actuations, such that the valves can be opened only when both the software-based PMS logic and the analog ADS and IRWST Injection Blocking Device both agree that an actuation is needed. Each ADS and IRWST Injection Blocking Device division is supplied with one Core Makeup Tank (CMT) level instrumentation channel from each CMT to automatically unblock ADS and IRWST injection actuation during event scenarios requiring actuation of ADS or the IRWST injection valves, based on low level in one or both Core Makeup Tanks (CMTs). The operator can also manually unblock each division by placing the Main Control Room ADS and IRWST Injection Block switch in the respective division in the “unblock” position.

While not required to be OPERABLE, two additional signals result in ADS and IRWST injection unblock via the blocking device: (1) low battery charger input voltage; and (2) transfer of control from the MCR to the remote shutdown workstation.

The ADS and IRWST injection blocking device logic is implemented in hardware using conventional analog components and does not use or rely upon software.

Failure of a single ADS and IRWST Injection Blocking Device will, at most, prevent the opening of one ADS Stage 1 path and one ADS Stage 3 path, or an ADS Stage 2 path. Since each of these stages has two paths, failure of one division (path) does not defeat the ADS function. Each ADS Stage 4 path (valve) is operated by two divisions; therefore, failure of a single ADS and IRWST Injection Blocking Device will not defeat any Stage 4 path. Each IRWST injection path (valve) is also operated by two divisions, and therefore, the failure of a single blocking device will not defeat any injection path. Loss of power to the ADS blocking device removes the block from the CIM.

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

Four Divisions of ADS and IRWST Injection Blocking Devices for automatic unblocking with two level instrumentation channels (one from each CMT) per division provide plant protection in the event of any of analyzed accident.

Four Divisions of ADS and IRWST Injection Blocking Devices (with one manual switch per division) for manual unblocking, with Main Control Room ADS and IRWST Injection Block manual switches in the “unblock” position, provide plant protection in the event of any of analyzed accident. ADS and IRWST Injection Block manual switches in the “unblock” position is required for manual ADS and IRWST injection actuation.

The ADS provides a sequenced depressurization of the reactor coolant system to allow passive injection from the CMTs, accumulators, and the IRWST to mitigate the effects of a LOCA. The depressurization is accomplished in four stages, with the first three stages discharging into the IRWST and the fourth stage discharging into containment.

OPERABILITY of the ADS and IRWST Injection Blocking Device is required to assure ADS and IRWST injection actuation, as assumed in the safety analysis. When the blocking device is not unblocked, a low level in either one or both CMTs will unblock the ADS and IRWST injection actuation in the respective division. Each OPERABLE ADS and IRWST Injection Blocking Device division requires one upper narrow range level transmitter on each of the two CMTs. Additionally, both CMTs must be capable of injecting to assure appropriate redundancy for the unblocking function. The CMT low level setpoint for the ADS and IRWST Injection Blocking Device is set above the ADS Stages 1, 2, and 3 CMT Level – Low 3 actuation setpoint. Thus on decreasing level, the CMT low level unblocks ADS actuation prior to reaching the ADS Stages 1, 2, and 3 CMT Level – Low 3 setpoint. OPERABILITY of the CMT low level unblock is required from each CMT.

The ADS and IRWST Injection Blocking Device is required to be OPERABLE for automatic unblocking in MODES 1, 2, and 3, and in MODE 4 with the Reactor Coolant System (RCS) not being cooled by the Normal Residual Heat Removal System (RNS). This aligns with the Applicability for the CMTs in LCO 3.5.2, Core Makeup Tanks (CMTs) – Operating.

The ADS and IRWST Injection Blocking Device is required to be OPERABLE for manual unblocking in MODES 1, 2, 3, 4, 5, and 6. This aligns with the Applicability for the manual actuation functions for ADS and IRWST injection required by LCO 3.3.9, “Engineered Safety Feature Actuation System (ESFAS) Manual Initiation.”

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

In MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS), and in MODES 5 and 6, less than two Core Makeup Tanks (CMT) are required to be OPERABLE, and therefore, may not be available to support the required automatic unblocking to assure ADS and IRWST injection actuation. Therefore, ADS and IRWST Injection Block manual switches are required to be in the “unblock” position to support automatic ADS and IRWST injection actuation OPERABILITY.

For postulated shutdown events in MODE 6, the IRWST provides the reactor coolant system (RCS) heat removal by IRWST injection and containment sump recirculation. In MODE 6 with the reactor vessel internals removed, IRWST Manual actuation continues to be required by LCO 3.3.9, Engineered Safety Feature Actuation System (ESFAS) Manual Initiation. As such, the ADS and IRWST Injection Block manual switches in the “unblock” position supports IRWST OPERABILITY.

The CMT level OPERABILITY requirement is provided with a Note stating that ADS and IRWST Injection Blocking Device OPERABILITY for automatic unblocking from CMT level is not required with ADS and IRWST Injection Block switch in the “unblock” position. This supports unblocking prior to the Applicability transition of establishing RNS cooling, by establishing unblocked state that is credited in the safety analysis.

ADS and IRWST Injection Blocking Device satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

The ACTIONS are modified by a Note to clarify the application of Completion Time rules. The Condition A may be entered independently for each division with an inoperable ADS and IRWST Injection Blocking Device. The Completion Time(s) of the inoperable division will be tracked separately for each division starting from the time the Condition was entered for that division.

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BASES

ACTIONS (continued)

A.1

Condition A addresses the situation where one or more divisions of ADS and IRWST Injection Blocking Device(s) is inoperable (e.g., one or both CMT level channels in one or more divisions inoperable when required, or ADS and IRWST Injection Block in one or more divisions not unblocked when required). In this condition, the component interface module (CIM) in the affected division is required to be unblocked in the affected division within 8 hours. The ADS and IRWST Injection Block manual switches may be utilized to implement the unblock. The 8 hours is reasonable based on the low probability of an event occurring during this interval.

B.1

If the Required Action and associated Completion Time of Condition A is not met the affected ADS and IRWST injection valves must be declared inoperable immediately. Declaring the affected valves inoperable allows the supported system Actions (i.e., for ADS and IRWST inoperable valves) to dictate the required measures. The ADS and/or IRWST LCO(s) provide appropriate actions for the inoperable components. This action is in accordance with LCO 3.0.6, which requires that the applicable Conditions and Required Actions for valves declared inoperable shall be entered in accordance with LCO 3.0.2.

SURVEILLANCE REQUIREMENTS

The SRs for each ADS and IRWST Injection Blocking Device Function are identified in the SRs column of Table 3.3.20-1 for that Function.

A Note has been added to the SR table stating that Table 3.3.20-1 determines which SRs apply to which ADS and IRWST Injection Blocking Device Function.

SR 3.3.20.1

Verification that the position of each ADS and IRWST Injection Block switch is in the “unblock” position is required when less than two CMTs are required to be OPERABLE. This assures the actuation of ADS and IRWST injection is not blocked when there may be reduced or no capability for automatic unblocking from CMT level. The 7 day Frequency is adequate considering the availability of main control room status monitoring of the block signal.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.20.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop. The test is performed in accordance with the SP. If the actual setting of the channel is found to be outside the as-found tolerance, the channel is considered inoperable. This condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTS (within the allowed tolerance), and evaluating the channel's response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation. Transmitter calibration must be performed consistent with the assumptions of the setpoint methodology. The difference between the current as-found values and the previous as-left values must be consistent with the transmitter drift allowance used in the setpoint methodology.

The setpoint methodology requires that 30 months drift be used (1.25 times the surveillance calibration interval, 24 months).

The Frequency is based on operating experience and consistency with the refueling cycle.

SR 3.3.20.3

This SR is the performance of an ACTUATION LOGIC TEST for unblocking. This test overlaps the ADS and IRWST injection functional tests (i.e., SR 3.4.11.4, SR 3.4.11.5, and SR 3.5.6.9) that verify actuation on an actual or simulated actuation signal.

The Frequency of 24 months is based on the need to perform this SR during periods in which the plant is shut down for refueling to prevent any additional risks associated with inadvertent operation of the ADS and IRWST injection valves.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.20.4

This SR is the performance of a TADOT of the of required ADS and IRWST Injection Block manual switch. This TADOT is performed every 24 months.

The Frequency is based on the known reliability of manual switch Functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that states verification of setpoint is not required, since these functions have no setpoint associated with them.

SR 3.3.20.5

This SR requires performance of LCO 3.5.2 Surveillances associated with ensuring CMTs are capable of injecting to the RCS. CMT injection supports OPERABILITY of the ADS and IRWST Injection Blocking Devices for automatic unblocking. If one or both CMTs are inoperable for injection, all four divisions of ADS and IRWST Injection Blocking Devices are inoperable. Therefore, SRs 3.5.2.2, 3.5.2.5, and 3.5.2.6 are required to be met. See the corresponding Bases for LCO 3.5.2 for a discussion of each Surveillance and its Frequency.

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”
2. FSAR Chapter 7, “Instrumentation and Controls.”
3. WCAP-10271, “Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System, June 1996” Supplement 2.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within the limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope of operating conditions. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. At the beginning of the first fuel cycle, precision (calorimetric) flow measurements, augmented by hydraulic measurements in the reactor coolant loop and pump performance, provide a value for comparison to the limit, and to determine the calibration coefficients for future use with differential pressure measurements. The reactor coolant flow rate channels are normalized to these test measurements for 100% indication using these calibration coefficients and are frequently monitored to determine flow degradation. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

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BASES

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown transients initiated within the requirements of this LCO will result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit which could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed include loss of coolant flow events and dropped or stuck rod events. An assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, “Control Bank Insertion Limits,” as well as within the limits of either LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” and LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” or within the limits of LCO 3.2.5, “On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters.”

The pressurizer pressure limit and the RCS average temperature limit specified in the COLR correspond to analytical limits, with an allowance for steady state fluctuations and measurement errors. The RCS average temperature limit corresponds to the analytical limit with allowance for controller deadband and measurement uncertainty.

These limits are developed using the methodologies identified in Technical Specificaiton 5.6.3.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables, pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on maximum analyzed steam generator tube plugging, is retained in the LCO. Operating within these limits will result in meeting DNBR criterion in the event of a DNB limited transient.

The COLR RCS total flow rate limit is equal to or more restrictive than the ≥ 301,670 gpm limit specified in the LCO. The COLR limit reflects the cycle-specific core design and plant conditions as well as added margin.

Separate minimum RCS total flow rate limits are specified in the COLR for measurement by precision heat balance or by differential pressure instrumentation. Different flow limits may apply for each measurement method since the two methods have unique measurement errors and instrument allowances that are included in the COLR RCS flow rate limits.

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BASES

LCO (continued)

The calibration coefficients for the differential pressure (hot-leg elbow and cold-leg bend) RCS total flow rate indication channels are established based on the comprehensive RCS flow measurements taken at the beginning of the first fuel cycle. These measurements include precision (calorimetric) flow, differential temperature, reactor coolant loop hydraulic tests, and pump performance. The differential pressure calibration coefficients are not expected to change during plant life. Measurement errors associated with the method used to determine the calibration coefficients are included in the differential pressure COLR RCS flow rate limit.

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location but have been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state plant operation in order to ensure DNBR criterion will be met in the event of an unplanned loss of forced coolant flow or other DNB-limiting transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counter productive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, “Reactor Core SLs.” The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether an SL may have been exceeded.

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BASES

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This surveillance demonstrates that the pressurizer pressure remains greater than or equal to the limit specified in the COLR. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency of pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

This surveillance demonstrates that the average RCS temperature remains less than or equal to the limit specified in the COLR. Indications of temperature are averaged to determine a value for comparison to the limit. Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

This surveillance demonstrates that the RCS total flow rate remains ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR. The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed differential pressure flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

A CHANNEL CALIBRATION of the differential pressure RCS total flow rate indication channels is performed every 24 months, at the beginning of each fuel cycle.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter with the necessary range and accuracy.

The Frequency is based on consistency with the refueling cycle.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.5

Measurement of RCS total flow rate by performance of precision test measurements once every 24 months, at the beginning of each fuel cycle, allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow is greater than or equal to the minimum required RCS flow rate. These test measurements may be based on a precision heat balance, or by differential pressure measurements of static elements in the RCS piping (such as elbows) that have been calibrated by previous precision tests, or by a combination of those two methods. In all cases, the measured flow, less allowance for error, must exceed the corresponding value used in the safety analysis and specified in the COLR.

The Frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until after 24 hours after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES 1. FSAR Chapter 15, “Accident Analyses.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, “Moderator Temperature Coefficient (MTC).” In the transient and accident analyses, the MTC is assumed to be in a range from zero to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil-ductility reference temperature when the reactor is critical.

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assume the failure of, or presents a challenge to, the integrity of a fission product barrier.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures greater than or equal to the HZP temperature of 557°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality parameter satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.8, “MODE 2 PHYSICS TEST Exceptions,” permits PHYSICS TESTS to be performed at ≤ 5.0% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

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BASES

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with k_{eff} < 1.0 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with k_{eff} < 1.0 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 551°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

REFERENCES

1. FSAR Chapter 15, “Accident Analyses.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, RCS inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The limit requirements are for ferritic materials of pressure-retaining components of the RCPB. The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer or individual auxiliary lines, which have different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 1) requires the establishment of P/T limits for ferritic material fracture toughness requirements of the RCPB materials. An adequate margin to brittle failure must be provided during normal operation, anticipated operational occurrences, and system hydrostatic tests. Reference 1 mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the

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BASES

BACKGROUND (continued)

recommendations of Regulatory Guide 1.99 (Ref. 5).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the P/T span of the limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the requirement to be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for RCS ISLH testing per Reference 1. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, “RCS Minimum Temperature for Criticality.”

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the ferritic materials of the RCPB, possibly leading to a leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the ferritic materials of pressure-retaining components of the RCPB, an unanalyzed condition. Reference 7 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and RCS ISLH testing; and

b. Limits on the rate of change of temperature

The LCO limits apply to components of the RCS based on the bulk RCS conditions, except the pressurizer and individual auxiliary lines. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and RCS ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

The LCO is modified by a Note that requires the secondary side water temperature of each SG to be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with at least one RCS cold leg temperature ≤ 350°F, and the RCP must be started at ≤ 25% of rated RCP speed. This limitation helps to ensure that the RNS system pressure remains below both the piping design pressure and the acceptable RNS relief valve inlet pressure. This restraint also assures the thermal transient when an RCP is started is within the bounds of the low temperature overpressure event analysis.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile (brittle) failure in accordance with 10 CFR 50, Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or RCS ISLH testing, they are applicable at all times in keeping with the

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BASES

APPLICABILITY (continued)

concern for nonductile failure. The limits do not apply to the pressurizer or individual auxiliary lines.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits”; LCO 3.4.2, “RCS Minimum Temperature for Criticality”; and Safety Limit 2.1, “Safety Limits,” also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS A.1 and A.2

Operation outside the limits of the LCO must be restored to within the limits. The RCS must be returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with preanalyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the

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ACTIONS (continued)

allowable limits. Restoration per Required Action A.1 alone is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished in 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 within 36 hours.

The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power condition in an orderly manner without challenging plant systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the limits of the LCO at times other than when in MODE 1, 2, 3, or 4, so that the RCS is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

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BASES

ACTIONS (continued)

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within PTLR limits is required every 30 minutes when RCS P/T conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or RCS ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this surveillance to be performed during system heatup, cooldown, and RCS ISLH testing. No SR is given for criticality operations because LCO 3.4.2, “RCS Minimum Temperature for Criticality,” contains a more restrictive requirement.

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BASES

REFERENCES

1. 10 CFR 50, Appendix G, “Fracture Toughness Requirements.”
2. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, “Protection Against Non-Ductile Failure.”
3. ASTM E 185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” July 1982.
4. 10 CFR 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.”
5. Regulatory Guide 1.99, “Radiation Embrittlement of Reactor Vessel Materials,” May 1988.
6. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, “Evaluation of Unanticipated Operating Events.”
7. WCAP-14040-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops

BASES

BACKGROUND The functions of the RCS include:

a. Removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs) to the secondary plant;

b. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

c. Improving the neutron economy by acting as a reflector;

d. Carrying the soluble neutron poison, boric acid;

e. Providing a second barrier against fission-product release to the environment; and

f. Removal of the heat generated in the fuel due to fission-product decay following a unit shutdown.

The reactor coolant is circulated through two loops connected in parallel to the reactor vessel, each containing a SG, two reactor coolant pumps (RCPs), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the fuel. In MODES 1 and 2 the SGs provide the heat sink to the secondary coolant. In order to act as a heat sink, the SGs must have a narrow range water level above Low 2 in both SGs, otherwise a reactor trip is initiated and Passive Residual Heat Removal (PRHR) actuation occurs. The RCPs circulate the primary coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control. The steam generator heat transfer function and associated secondary water and steam systems are not required to provide a safety-related safe shutdown of the plant (Ref. 1).

The RCPs must be started using the variable speed controller with the Plant Control System (PLS) incapable of rod withdrawal and all rods fully inserted. The controller shall be bypassed prior to making the PLS capable of rod withdrawal or withdrawing one or more rods.

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BASES

APPLICABLE SAFETY ANALYSES

MODES 1 and 2

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops and RCPs in service.

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming two RCS loops are initially in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses, where RCP operation is most important are the four pump coastdown, single pump locked rotor, single pump broken shaft or coastdown, and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the DNB ratio (DNBR) limit) assumes a maximum power level of 100% RTP. This is the design overpower condition for two RCS loop operation. The DNBR limit defines a locus of pressure and temperature points which result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with both RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

MODES 3, 4, and 5

Whenever the PLS is capable of rod withdrawal or one or more rods are not fully inserted, there is the possibility of an inadvertent rod withdrawal from subcritical, resulting in a power excursion in the area of the withdrawn rod. Such a transient could be caused by a malfunction of the PLS. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible. Such a transient could be caused by the mechanical failure of a CRDM. The initial power rise is terminated by doppler broadening in the fuel pins, followed by rod insertion. During this event, if there is not adequate coolant flow along the clad surface of the fuel, there is a potential to exceed the DNBR limit. Therefore, the required coolant flow is an initial condition of a design basis event that presents a challenge to the integrity of a fission product barrier.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, in MODE 3, 4 or 5 with the PLS capable of rod withdrawal or one or more rods not fully inserted, accidental control rod withdrawal from subcritical is postulated and requires the RCPs to be OPERABLE and in operation with variable speed control bypassed to ensure that the accident analysis limits are met.

RCS Loops satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required in MODES 1 and 2. The requirement that at least four RCPs must be operating in MODES 3, 4 and 5 when the PLS is capable of rod withdrawal or one or more rods are not fully inserted provides assurance that, in the event of a rod withdrawal accident, there will be adequate flow in the core to avoid exceeding the DNBR limit. Bypass of the RCP variable speed control ensures that the pumps are operating at full flow.

The LCO is modified by a Note that permits all RCPs to be removed from operation in MODE 3, 4, or 5 for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analysis values. One of these tests is for the validation of the pump coastdown curve, used as input to a number of accident analyses including a loss of flow accident.

This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve may need to be revalidated by conducting the test again.

No-flow testing may be performed in MODE 3, 4, or 5, and requires that the pumps be stopped for a short period of time. Note 3 permits removing all RCPs from operation in order to perform testing and validate the assumed analysis values. The 1 hour time period specified is adequate to perform the desired tests and experience has shown that boron stratification is not a problem during this short period with no forced flow.

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BASES

LCO (continued)

Utilization of the Note is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause natural circulation flow obstruction.

An OPERABLE RCS loop is composed of two OPERABLE RCPs in operation with variable speed control bypassed providing forced flow for heat transport.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, both RCS loops are required to be OPERABLE and in operation with variable speed control bypassed in these MODES to prevent DNB and core damage.

In MODES 3, 4 and 5, this LCO ensures forced circulation of the reactor coolant to protect from exceeding the DNBR limit and to provide proper boron mixing. For these purposes and because the PLS is capable of rod withdrawal or one or more rods are not fully inserted, there is the possibility of an inadvertent rod withdrawal event. Four RCPs are required to be operating in MODES 3, 4, and 5, with variable speed control bypassed, whenever the PLS is capable of rod withdrawal or one or more rods are not fully inserted. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

ACTIONS

A.1, A.2, A.3, and A.4

If the requirements of the LCO are not met while in MODE 1 or 2, the Required Action is to suspend the start of any RCP, reduce power and bring the plant to MODE 3, initiate action to fully insert all rods, and place the PLS in a condition incapable of rod withdrawal. The Condition A Note

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BASES

ACTIONS (continued)

requires these Required Actions to be completed (i.e., to exit the Applicability prior to attempting to restart a RCP) when Condition A is entered. This prevents startup of a RCP and the resulting circulation of cold and/or unborated water from an inactive loop into the core, precluding reactivity excursion events which are unanalyzed and lowers power level; thus reducing the core heat removal needs and minimizing the possibility of violating DNB limits.

When all four reactor coolant pumps are operating, a loss of a single reactor coolant pump above power level P-10 will result in an automatic reactor trip.

The Completion Time of 6 hours is reasonable to allow for an orderly transition to MODE 3. The applicable safety analyses described above bound Design Basis Accidents (DBA) initiated with three reactor coolant pumps operating at power levels below P-10.

B.1, B.2, and B.3

If the requirements of the LCO are not met while in MODE 3, 4 or 5, the Required Action is to suspend the start of any RCP, initiate action to fully insert all rods, and place the PLS in a condition incapable of rod withdrawal. The Condition B Note requires these Required Actions to be completed (i.e., to exit the Applicability prior to attempting to restart a RCP) when Condition B is entered. The actions prevent startup of a RCP and the resulting circulation of cold and/or unborated water from an inactive loop into the core, precluding reactivity excursion events which are unanalyzed and eliminate the possibility of a rod withdrawal event with one or more pumps not operating and thus minimizing the possibility of violating DNB limits.

The Completion Time of 1 hour is reasonable to allow for making PLS incapable of rod withdrawal and fully inserting all control rods, since plant cool-down is not required.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation with the pump variable speed control bypassed. Verification helps ensure that forced flow is maintaining the margin to DNB and proper boron mixing. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the main control room to monitor RCS loop performance.

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BASES

REFERENCES

1. FSAR Section 5.4.2.1, “Design Bases.”
2. FSAR Chapter 15, “Accident Analysis.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The normal level and pressure control components addressed by this LCO include the pressurizer water level, the heaters, their controls, and power supplies. Pressurizer safety valves and automatic depressurization valves are addressed by LCO 3.4.6, “Pressurizer Safety Valves,” and LCO 3.4.11, “Automatic Depressurization System (ADS) – Operating,” respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, and MODE 4 with all four cold leg temperatures > 275°F, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. A steam bubble is credited in the safety analyses whenever the PSVs provide overpressure protection of the RCS. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in FSAR Chapter 15 (Ref. 1) do not take credit for pressurizer heater operation, however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirement for the pressurizer water volume ≤ 92% of span, ensures that an adequate steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3 and for MODE 4 with all four cold leg temperatures > 275°F. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup. With at least one cold leg temperature ≤ 275°F, overpressure protection is provided by LCO 3.4.14, Low Temperature Overpressure Protection (LTOP), and a steam bubble is not required.

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions.

If the pressurizer water level is above the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. This is done by placing the unit in MODE 3 with the Plant Control System in a condition incapable of rod withdrawal and action initiated to fully insert all control rods within 6 hours, and placing the unit in MODE 4 with at least one cold leg temperature ≤ 275°F within 24 hours. This takes the unit out of the applicable MODES and restores the unit to operation within the bounds of the safety analyses. Control rods are capable of withdrawal when one or more control rod drive mechanisms (CRDMs) is energized such that an automatic or manual signal could power the coils to withdraw the rod(s).

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BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 Pressurizer Safety Valves

BASES

BACKGROUND

The two pressurizer safety valves provide, in conjunction with the Protection and Safety Monitoring System (PMS), overpressure protection for the RCS. The pressurizer safety valves are totally enclosed, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2733.5 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The minimum relief capacity for each valve, 750,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The pressurizer safety valves discharge into the containment atmosphere. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 when the reactor vessel head is on; however, in MODE 4 with at least one RCS cold leg temperature ≤ 275°F, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.14, “Low Temperature Overpressure Protection (LTOP).”

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the ASME Code, Section III pressure limit (Ref. 1) could include damage to RCS components, increased LEAKAGE, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

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BASES

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in FSAR Chapter 15 (Ref. 3) that require safety valve actuation assume operation of two pressurizer safety valves to limit increases in the RCS pressure. The overpressure protection analysis (Ref. 2) is also based on operation of the two safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Locked rotor; and

e. Loss of AC power/loss of normal feedwater

Detailed analyses of the above transients are contained in Reference 3. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer Safety Valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig.

The limit protected by this specification is the Reactor Coolant Pressure Boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and MODE 4 with all four cold leg temperatures > 275°F, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included although the listed accidents may not require the safety valves for protection.

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BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 4 with any cold leg temperature ≤ 275°F and in MODE 5, because overpressure protection is provided by LCO 3.4.14, Low Temperature Overpressure Protection (LTOP). Overpressure protection is not required in MODE 6 with reactor vessel head removed, or overpressure protection is provided by meeting the requirements of LCO 3.4.14, “Low Temperature Overpressure Protection (LTOP)”.

The Note allows entry into MODES 3 and 4 with the lift setpoints outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time may be made inoperable for hot lift setting adjustment. The 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two pressurizer safety valves are inoperable, the plant must be placed in a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with at least one cold leg temperature ≤ 275°F within 24 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With at least one cold leg temperature ≤ 275°F, overpressure protection is provided by the LTOP System per LCO 3.4.14. The change from MODE 1, 2, or 3 to MODE 4 with at least one cold leg temperature ≤ 275°F reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested one at a time and in accordance with the requirements of ASME OM Code (Ref. 4), which provides the activities and Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 1% for OPERABILITY, and the values are reset to remain within ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7500.
2. WCAP-16779, “AP1000 Overpressure Protection Report, April 2007.”
3. FSAR Chapter 15, “Accident Analyses.”
4. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core comprise the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of LEAKAGE inside containment is expected from auxiliary systems that cannot be made 100% leaktight. LEAKAGE from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

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BASES

BACKGROUND (continued)

LCO 3.4.15, "RCS Pressure Isolation Valve (PIV) Integrity," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA. The amount of LEAKAGE can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 300 gpd primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leak contaminates the secondary fluid.

The FSAR Chapter 15 (Ref. 3) analyses for the accidents involving secondary side releases assume 150 gpd primary to secondary LEAKAGE in each generator as an initial condition. The design basis radiological consequences resulting from a postulated SLB accident and SGTR are provided in Sections 15.1.5 and 15.6.3 of FSAR Chapter 15, respectively.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

Pressure boundary LEAKAGE is prohibited as the leak itself could cause further RCPB deterioration, resulting in higher LEAKAGE.

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BASES

LCO (continued)

b. Unidentified LEAKAGE

0.5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air F18 particulate radioactivity monitoring and containment sump level monitoring equipment, can detect within a reasonable time period. This leak rate supports leak before break (LBB) criteria. Separating the sources of LEAKAGE (i.e., LEAKAGE from an identified source versus LEAKAGE from an unidentified source) is necessary for prompt identification of potentially adverse conditions, assessment of the safety significance, and corrective action.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources.

d. Primary to Secondary LEAKAGE through One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

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BASES

LCO (continued)

e. Primary to In-Containment Refueling Water Storage Tank (IRWST) LEAKAGE through the Passive Residual Heat Removal Heat Exchanger (PRHR HX)

The 500 gpd limit from the PRHR HX is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress condition of an RCS pressure increase event. If leakage is through many cracks, and the cracks are very small, then the above assumption is conservative. This is conservative because the thickness of the PRHR HX tubes is approximately 60% greater than the thickness of the SG tubes. Furthermore, a PRHR HX tube rupture would result in an isolable leak and would not lead to a direct release of radioactivity to the atmosphere.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

ACTIONS

A.1

If pressure boundary LEAKAGE exists, the affected component, pipe, orvessel must be isolated from the RCS by a closed manual valve, closedand de-activated automatic valve, blind flange, or check valve within4 hours. While in this condition, structural integrity of the system shouldbe considered because the structural integrity of the part of the system within the isolation boundary must be maintained under all licensing basis conditions, including consideration of the potential for further degradation of the isolated location. Normal LEAKAGE past the isolation device is acceptable as it will limit RCS LEAKAGE and is included in identified or unidentified LEAKAGE. This action is necessary to prevent further deterioration of RCPB.

B.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

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BASES

ACTIONS (continued)

C.1 and C.2

If primary to secondary LEAKAGE is not within limits, or if any of theRequired Actions and associated Completion Times are not met, the reactor must be brought to lower pressure conditions to reduce theseverity of the LEAKAGE and its potential consequences. The reactormust be brought to MODE 3 within 6 hours and to MODE 5 within36 hours. This action reduces the LEAKAGE and also reduces the factors which tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without ACTIONS challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

Verifying RCS LEAKAGE within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection.

Unidentified LEAKAGE and identified LEAKAGE are determined by performance of a RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions. The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer level, and reactor coolant drain tank and in-containment refueling water storage tank levels.

RCS inventory monitoring via the pressurizer level changes is valid in MODES 1, 2, 3, and 4 only when RCS conditions are stable, i.e., temperature is constant, pressure is constant, no makeup and no letdown.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere F18 particulate radioactivity and the containment sump level. These LEAKAGE detection systems are specified in LCO 3.4.9, “RCS LEAKAGE Detection Instrumentation.”

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72-hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.7.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and reactor coolant drain tank and in-containment refueling water storage tank levels, and makeup and letdown flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

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BASES

REFERENCES

1. 10 CFR 50, Appendix A GDC 30.
2. Regulatory Guide 1.45, Revision 0, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” May 1973.
3. FSAR Chapter 15, “Accident Analysis.”
4. NEI-97-06 – “Steam Generator Program Guidelines.”
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 Minimum RCS Flow

BASES

BACKGROUND

The RCS consists of the reactor vessel and two heat transfer loops, each containing a steam generator (SG), two reactor coolant pumps (RCPs), a single hot leg and two cold legs for circulating reactor coolant. Loop 1 also contains connections to the pressurizer and passive residual heat removal (PRHR).

The primary function of the reactor coolant is removal of decay heat and the transfer of this heat, via the SGs to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

APPLICABLE SAFETY ANALYSES

An initial condition in the Design Basis Accident (DBA) analysis of a possible Boron Dilution Event (BDE) in MODE 3, 4, or 5 is the assumption of a minimum mixing flow in the RCS. In this scenario, dilute water is inadvertently introduced into the RCS, is uniformly mixed with the primary coolant, and flows to the core. The increase in reactivity is detected by the source range instrumentation which provides a signal to terminate the inadvertent dilution before the available SDM is lost. If there is inadequate mixing in the RCS, the dilute water may stratify in the primary system, and there will be no indication by the source range instrumentation that a dilution event is in progress. When primary flow is finally increased, the dilution event may have progressed to the point that mitigation by the source range instrumentation is too late to prevent the loss of SDM.

Thus, a minimum mixing flow in the RCS is a process variable which is an initial condition in a DBA analysis.

Minimum RCS Flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that at least one RCP be in operation with a minimum core flow ≥ 3,000 gpm provides assurance that in the event of an inadvertent BDE, the diluted water will be properly mixed with the primary system coolant, and the increase in core reactivity will be detected by the source range instrumentation. A core flow < 3,000 gpm is considered equivalent to no RCP in operation.

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BASES

LCO (continued)

The LCO is modified by a Note that permits all RCPs to be removed from operation for ≤ 1 hour per 8 hour period for the purpose of testing. This Note also permits an unborated water source through the chemical mixing tank to be temporarily unisolated under administrative controls with no RCPs in operation for the purpose of chemical addition to the pressurizer. The purpose of the testing allowance is to permit tests that are designed to validate various accident analysis values. One of these tests is for the validation of the pump coastdown curve, used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve may need to be revalidated by conducting the test again.

No-flow testing may be performed in MODE 3, 4, or 5, and requires that the pumps be stopped for a short period of time. The Note permits removing all RCPs from operation in order to perform this testing and validate the assumed analysis values. The 1 hour time period specified is adequate to perform the desired tests and experience has shown that boron stratification is not a problem during this short period with no forced flow.

Additionally, the Note allows chemical additions to be performed and mixed with CVS makeup operations using a blend of borated and unborated water sources for the purpose of achieving proper RCS water quality. In this case, RCPs are not initially running; therefore, the allowance is made to permit chemical addition to the pressurizer under administrative controls so that the Applicability can be entered without violating the RCP in operation requirement. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with a dedicated operator in the control room. In this way, the flow path from the chemical mixing tank to the RCS can be rapidly isolated locally or from the control room after completion of the limited duration chemical addition, or if flow rates indicate the potential for inadequately borated blended flow.

Utilization of the Note is permitted provided the following conditions are met along with any other conditions imposed by procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron

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BASES

LCO (continued)

concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause natural circulation flow obstruction.

APPLICABILITY

Minimum RCS flow is required in MODES 3, 4, and 5 with unborated water sources not isolated from the RCS because an inadvertent BDE is considered possible in these MODES.

In MODES 1 and 2, and in MODES 3, 4, and 5 with the Plant Control System capable of rod withdrawal or one or more rods not fully inserted, LCO 3.4.4 requires all four RCPs to be in operation. Thus, in the event of an inadvertent boron dilution, adequate mixing will occur.

A minimum mixing flow is not required in MODE 6 because LCO 3.9.2 requires that all valves used to isolate unborated water sources shall be secured in the closed position. In this situation, an inadvertent BDE is not considered credible.

ACTIONS

A.1

If no RCP is in operation, all sources of unborated water must be isolated within 1 hour. This action assures that no unborated water will be introduced into the RCS when proper mixing cannot be assured. The allowed Completion Time requires that prompt action be taken, and is based on the low probability of a DBA occurring during this time.

A.2

The Requirement to perform SR 3.1.1.1 (SDM verification) within 1 hour assures that if the boron concentration in the RCS has been reduced and not detected by the source range instrumentation, prompt action may be taken to restore the required SDM. The allowed Completion Time is consistent with that required by Action A.1 because the conditions and consequences are the same.

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BASES

ACTIONS (continued)

Condition A is modified by a Note that requires Required Action A.2 to be performed prior to starting any RCP whenever the Condition is entered. This ensures that SR 3.1.1.1 will be performed prior to starting an RCP, even when Condition A is exited prior to performing Required Action A.2. Performance of SR 3.1.1.1 is necessary to assure SDM is properly evaluated prior to starting an RCP.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that a minimum mixing flow is present in the RCS. A Frequency of 12 hours is adequate considering the low probability of an inadvertent BDE during this time, and the ease of verifying the required RCS flow.

REFERENCES None.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting, and, to the extent practical, identifying the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting LEAKAGE detection systems.

LEAKAGE detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

The containment sump used to collect unidentified LEAKAGE, is instrumented to alarm for increases of 0.5 gpm in the normal flow rates. Note that the containment sump level instruments are also used to identify leakage from the main steam lines inside containment. Since there is not another method to identify steam line leakage in a short time frame, two sump level sensors are required to be OPERABLE. The containment water level sensors (LCO 3.3.17) provide a diverse backup method for steam line leakage detection inside containment. However, the containment water level sensors are not required to be OPERABLE by this LCO.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity used for leak detection is the decay of F18. The production of F18 is proportional to the reactor power level; and is not dependent on the presence of fuel defects or coolant impurities such as corrosion products. Due to its short half-life, F18 is not present in the reactor coolant after refueling. Its concentration in the reactor coolant only increases with increasing neutron flux. By the time THERMAL POWER has reached 20% RTP, the concentration of F18 in the reactor coolant is great enough that its presence in the containment atmosphere due to LEAKAGE would be detectable. F18 becomes a particulate after leaving the RCS; therefore, its particulate form is used for leak detection. The Radiation Monitoring System includes monitoring F18 particulate activity to provide leak detection.

The containment sump level and F18 particulate leakage monitoring methods differ in sensitivity and response time.

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BASES

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The LEAKAGE detection systems are described in FSAR Chapter 5 (Ref. 3).

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur.

RCS LEAKAGE detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requires three instruments to be OPERABLE, subject to the exceptions and timing of plant evolutions in the Applicability Notes, as discussed below.

The containment sump is used to collect unidentified LEAKAGE. The monitors on the containment sump detect level and are instrumented to detect when there is an increase above the normal value by 0.5 gpm. The identification of an increase in unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the containment sump; and it may take longer than one hour to detect a 0.5 gpm increase in unidentified LEAKAGE due to this transit time. This sensitivity is acceptable for containment sump monitor OPERABILITY.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by the containment atmosphere F18 particulate monitor. The LCO requires the containment atmosphere F18 particulate monitor because of the sensitivity of the particulate monitor and its rapid response to RCS LEAKAGE. A gaseous radioactivity monitor is also available; but has recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup, and for a few weeks thereafter, until activated corrosion products form, or in the event that fission products appear from fuel element cladding contamination or

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BASES

LCO (continued)

cladding defects. If there are few fuel element cladding defects and low levels of activation products, it might not be possible for the containment atmosphere gaseous radiation monitor to detect a 0.5 gpm increase within 1 hour during normal operation.

The containment atmosphere F18 particulate monitor has adequate detection sensitivity to detect LEAKAGE without the presence of fuel defects or activated cladding contamination or corrosion products. The containment atmosphere F18 particulate monitor is OPERABLE when it is capable of detecting 0.5 gpm increase in unidentified LEAKAGE within 1 hour given nominal plant operating conditions (e.g., no containment purge operations) at full power and an RCS F18 activity consistent with industry experience in the absence of fuel defects or significant activated cladding contamination or corrosion products.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, two containment sump level monitors, in combination with a containment atmosphere F18 particulate monitor, provide an acceptable minimum. Containment sump level monitoring is performed by two of the three redundant, seismically qualified level instruments.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS LEAKAGE detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are lower than those for MODES 1, 2, 3, and 4, the likelihood of LEAKAGE and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

Containment sump level monitoring is required for detecting LEAKAGE in MODES 1, 2, 3, and 4.

The Applicability is modified by Note 1 in recognition of reductions in LEAKAGE detection capability caused by plant evolutions that divert potential LEAKAGE away from the containment sump (i.e., redirection from the containment sump to the in-containment refueling water storage tank (IRWST) by the IRWST gutter drain isolation valves or containment purge operation, which may add or remove moisture from the containment atmosphere) or the containment atmosphere F18 particulate monitor (i.e., containment purge operation, which removes airborne radiation from the containment atmosphere). Those evolutions include

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BASES

APPLICABILITY (continued)

IRWST gutter drain isolation for testing or maintenance (during which closure of valves PXS-PL-130A/B redirects potential LEAKAGE to the IRWST) and containment purge operations during which none of the RCS LEAKAGE detection instrumentation is required to be OPERABLE.

As provided by Note 1, when plant evolutions that affect the RCS LEAKAGE detection instrumentation occur, the instrumentation is not required to be OPERABLE as long as the RCS water inventory balance prescribed in SR 3.4.7.1 of LCO 3.4.7, “RCS Operational LEAKAGE,” is performed once per 24 hours after 12 hours of steady state operation (stable RCS pressure, temperature, power level, pressurizer level, and reactor coolant drain tank (RCDT) and IRWST levels). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 24 hour Frequency required by Applicability Note 1 also applies during a 2 hour normalization period following the reopening of the IRWST gutter drain isolation valves and after closure of the containment purge flow path.

Applicability Note 2 states that the containment atmosphere F18 particulate radioactivity LEAKAGE measurement during MODE 1 is only required when THERMAL POWER is > 20% RTP.

ACTIONS A.1

With one of the two required containment sump level channels inoperable, the one remaining required OPERABLE channel is sufficient for RCS leakage monitoring since the containment radiation provides a method to monitor RCS leakage. The remaining required OPERABLE containment sump level monitor is adequate for steam line leakage monitoring. The containment sump level instruments are capable of detecting a volume change of less than 2 gallons.

Restoration of two containment sump level channels to OPERABLE status is required to regain the function in a Completion Time of 14 days after the monitor’s failure. This time is acceptable, considering the frequency of SR 3.4.9.1 and the available required channel.

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BASES

ACTIONS (continued)

B.1 and B.2

With two of the two required containment sump level channels inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere F18 particulate monitor will provide indications of changes in LEAKAGE. The containment atmosphere F18 particulate monitor together with performance of an RCS inventory balance (SR 3.4.7.1) on an increased frequency of every 24 hours provides information that is adequate to detect RCS LEAKAGE. The sensors monitoring containment water level provide information that is adequate for steam line leakage. A Note is added allowing that SR 3.4.7.1 is not required to be performed until 12 hours after establishing steady state operation (stable RCS pressure, temperature, power level, pressurizer level, and RCDT and IRWST levels). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of one containment sump level channel to OPERABLE status is required to regain the function in a Completion Time of 72 hours. This time is acceptable, considering the leak detection diversity and the frequency of performing SR 3.4.7.1.

C.1.1, C.1.2, and C.2

With the containment atmosphere F18 particulate monitoring instrumentation channel inoperable, action is required. Either grab samples of the containment atmosphere must be taken and analyzed or an RCS inventory balance (SR 3.4.7.1) must be performed every 24 hours, to provide alternate periodic information.

With a sample obtained and analyzed or an RCS inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the containment atmosphere F18 particulate monitor.

The 24 hour interval for grab samples or RCS inventory balance provides periodic information that is adequate to detect LEAKAGE. A Note is added allowing that SR 3.4.7.1 is not required to be performed until 12 hours after establishing steady state operation (stable RCS pressure, temperature, power level, pressurizer level, and RCDT and IRWST levels). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leak detection is available.

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BASES

ACTIONS (continued)

D.1 and D.2

If a Required Action of Condition A, B or C cannot be met within the required Completion Time, the reactor must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

E.1

With all required monitors inoperable, no required means of monitoring leakage is available and plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

SR 3.4.9.1 requires the performance of a CHANNEL CHECK of the containment atmosphere F18 particulate monitor and the required containment sump level channels. The check gives reasonable confidence that the required instrumentation and associated channels are operating properly. The Frequency of 12 hours is based on instrument reliability and risk and is reasonable for detecting off normal conditions.

SR 3.4.9.2

SR 3.4.9.2 requires the performance of a CHANNEL OPERATIONAL TEST (COT) on the containment atmosphere F18 particulate monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers risks and instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.9.3

This SR requires the performance of a CHANNEL CALIBRATION for each of the required RCS Leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument strings, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has proven that this Frequency is acceptable.

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BASES

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45, Revision 0, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” May 1973.
3. FSAR Section 5.2.5, “Detection of Leakage Through Reactor Coolant Pressure Boundary

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 RCS Specific Activity

BASES

BACKGROUND

The limits on RCS specific activity ensure that the doses due to postulated accidents are within the doses reported in FSAR Chapter 15.

The RCS specific activity LCO limits the allowable concentration of iodines and noble gases in the reactor coolant. The LCO limits are established to be consistent with a fuel defect level of 0.25% and to ensure that plant operation remains within the conditions assumed for shielding and Design Basis Accident (DBA) release analyses.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to limit the doses due to postulated accidents to within the values calculated in the radiological consequences analyses (as reported in FSAR Chapter 15).

APPLICABLE SAFETY ANALYSES

The LCO limits on the reactor coolant specific activity are a factor in accident analyses that assume a release of primary coolant to the environment.

The events which incorporate the LCO values for primary coolant specific activity in the radiological consequence analysis include the following:

Steam generator tube rupture (SGTR)
Steam line break (SLB)
Locked RCP rotor
Rod ejection
Small line break outside containment
Loss of coolant accident (LOCA) (early stages)

The limiting event for release of primary coolant activity is the SLB. The SLB dose analysis considers the possibility of a pre-existing iodine spike (in which case the maximum LCO of 60 µCi/gm DOSE EQUIVALENT I-131 is assumed) as well as the more likely initiation of an iodine spike due to the reactor trip and depressurization. In the latter case, the LCO of 1.0 µCi/gm DOSE EQUIVALENT I-131 is assumed at the initiation of the accident, but the primary coolant specific activity is assumed to increase with time due to the elevated iodine appearance rate in the coolant. The reactor coolant noble gas specific activity for both cases is

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BASES

APPLICABLE SAFETY ANALYSES (continued)

assumed to be the LCO of 280 µCi/gm DOSE EQUIVALENT XE-133. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.01 µCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.4, "Secondary Specific Activity."

The LCO limits ensure that, in either case, the doses reported in FSAR Chapter 15 remain bounding.

The RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the specific noble gas activity is limited to 280 µCi/gm DOSE EQUIVALENT XE-133. These limits ensure that the doses resulting from a DBA will be within the values reported in FSAR Chapter 15. Secondary coolant activities are addressed by LCO 3.7.4, "Secondary Specific Activity."

The SLB and SGTR accident analyses (Refs. 1 and 2) show that the offsite doses are within acceptance limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SLB or SGTR accident, lead to doses that exceed those reported FSAR Chapter 15.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity are necessary to contain the potential consequences of a SGTR to within the calculated site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODE 4, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves. In MODES 5 and 6, the RCS pressure and temperature are reduced and an SGTR event is not credible.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to verify that DOSE EQUIVALENT I-131 is ≤ 60 µCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is to continue to provide a trend.

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BASES

ACTIONS (continued)

The DOSE EQUIVALENT I-131 must be restored to normal within 48 hours. If the concentration cannot be restored to within the LCO limit in 48 hours, it is assumed that the LCO violation is not the result of normal iodine spiking.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With DOSE EQUIVALENT XE-133 in excess of the allowed limit, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 3 with RCS average temperature < 500˚F within 6 hours.

The change to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the set points of the main steam safety valves, and prevents venting the SG to the environment in a SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience to reach MODE 3 from full power conditions in an orderly manner, without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is > 60 µCi/gm, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operation experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SR 3.4.10.1 requires performing a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This is a quantitative measure of radionuclides with half lives longer than

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BASES

SURVEILLANCE REQUIREMENTS (continued)

15 minutes. This Surveillance provides an indication of any increase in the release of noble gas activity from fuel rods containing cladding defects.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the unlikelihood of a significant increase in fuel defect level during the time.

SR 3.4.10.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when increased releases of iodine from the fuel (iodine spiking) is apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level. The Frequency, between 2 and 6 hours after a power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failures; samples at other times would provide inaccurate results.

REFERENCES

1. FSAR Section 15.1.5, “Steam System Piping Failure.”
2. FSAR Section 15.6.3, “Steam Generator Tube Rupture.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Automatic Depressurization System (ADS) – Operating

BASES

BACKGROUND

The ADS is designed to assure that core cooling and injection can be achieved for Design Basis Accidents (DBA). The four stages of ADS valves are sequenced in coordination with the passive core cooling system injection performance characteristics. Actuation of the ADS stage 1, 2, and 3 valves depressurizes the Reactor Coolant System (RCS) to a pressure below which the ADS stage 4 valves may be actuated without exceeding allowed stage 4 piping loads. Actuation of the stage 4 valves opens flow paths from the hot legs into the associated steam generator compartment that facilitate gravity injection (draining) of the in containment refueling water storage tank (IRWST) into the RCS.

The ADS consists of 10 flow paths arranged in four different stages that open sequentially (Ref. 1). Stages 1, 2, and 3 each include 2 flow paths. Each of the stage 1, 2, 3 flow paths has a common inlet header connected to the top of the pressurizer. The outlets of the stage 1, 2, 3 flow paths combine into one of the two common discharge lines to the spargers located in the IRWST. The first stage valves are 4 inch valves with DC motor operators. The second and third stage valves are 8 inch valves with DC motor operators. An OPERABLE stage 1, 2, or 3 automatic depressurization flow path consists of two OPERABLE normally closed motor operated valves, in series. A single failure of electrical division A or B affects one 4-inch stage 1 flow path and one 8-inch stage 3 flow path. A single failure of electrical division C or D affects one 8-inch stage 2 flow path.

Stage 4 includes 4 flow paths. The fourth stage ADS valves are 14 inch squib valves. The four fourth stage flow paths connect directly to the top of the reactor coolant hot legs and vent directly into the associated steam generator compartment. An OPERABLE stage 4 flow path consists of an open motor operated valve and an OPERABLE closed squib valve. These motor operated valves are not required to be OPERABLE because they are open. Each stage 4 squib valve is actuated by two divisions; A and C, or B and D. Therefore, loss of a single electrical division will not prevent any opening of any 14-inch flow path. ADS stage 4 flow paths are also actuated by the Diverse Actuation System and a non-Class 1E battery.

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BASES

BACKGROUND (continued)

The automatic depressurization valves are designed to open automatically when actuated, and to remain open for the duration of any automatic depressurization event. The valves are actuated sequentially. The stage 1 valves are actuated on a Low 3 core makeup tank (CMT) level. Stages 2 and 3 are actuated on the stage 1 signal plus time delays. Stage 4 is actuated on a Low 6 CMT level signal with a minimum time delay after stage 3. Stage 4 is blocked from actuating at normal RCS pressure.

ADS blocking devices are provided to minimize the likelihood of spurious ADS actuation. The ADS injection blocking device in each PMS division provides an output to the ADS valve component interface module (CIM) to prevent the ADS actuation unless other plant parameters indicate that an actuation is necessary. A low level in either one or both CMTs will unblock the ADS actuation in each division.

In order to perform a controlled, manual depressurization of the RCS, the valves are opened starting with the first stage. The first stage valves can also be modulated to perform a partial RCS depressurization if required. ADS stage 1, 2, 3 valves may be manually operated under controlled conditions for testing purposes.

ADS stages 1, 2 and 3 valves are designed to open relatively slowly, from approximately 48 seconds for the first stage valves, to approximately 120 seconds for the second and third stage valves.

The ADS valves are powered by batteries. In the unlikely event that offsite and onsite AC power is lost for an extended period of time, a timer will actuate ADS within 24 hours of the time at which AC power is lost, before battery power has been degraded to the point where the valves cannot be opened.

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of up to seven (all ADS stage 1 to 3 and one ADS stage 4) flow paths. The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1, 2, 3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.

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BASES

APPLICABLE SAFETY ANALYSES

For non-LOCA events, use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the CMTs may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

For events which involve a loss of primary coolant inventory, such as a LOCA, the ADS will be actuated, allowing for injection from the accumulators, the IRWST, and the containment recirculation (Ref. 2).

The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that ten ADS flow paths be OPERABLE ensures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses.

For the ten ADS flow paths to be considered OPERABLE, the 16 ADS valves must be capable of opening on an actuation signal. In addition, the stage 4 motor operated isolation valves must be open. These stage 4 motor operated isolation valves are not required to be OPERABLE because they are maintained open per SR 3.4.11.1. Each stage 4 squib valve is OPERABLE when capable of actuation by two igniters supplied by independent Class 1 E electrical divisions.

APPLICABILITY

In MODES 1, 2, 3 and 4 the ADS must be OPERABLE to mitigate the potential consequences of any event which causes a reduction in the RCS inventory, such as a LOCA.

The requirements for the ADS in MODES 5 and 6 are specified in LCO 3.4.12, “Automatic Depressurization System (ADS) – Shutdown, RCS Intact,” and LCO 3.4.13, “Automatic Depressurization System – Shutdown, RCS Open.”

ACTIONS

A.1

If any one flow path in ADS stage 1, 2, or 3 is determined to be inoperable, the remaining OPERABLE ADS flow paths are more than adequate to perform the required safety function as long as a single failure involving a flow path in ADS stage 4 or two additional flow paths in ADS stage 1, 2, or 3 does not also occur. A Completion Time of 7 days is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a DBA during this time period.

If more than one flow path in ADS stage 1, 2, or 3 is inoperable, Condition C or D is also applicable.

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BASES

ACTION (continued)

B.1

If any one flow path in ADS stage 4 is determined to be inoperable, the remaining OPERABLE flow paths in ADS stage 4 are adequate to perform the required safety function as long as a single failure does not also occur. A Completion Time of 72 hours is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a DBA during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.

C.1

If one flow path in ADS stage 1 and one flow path in ADS stage 2 or 3, or two flow paths in ADS stage 1 are determined to be inoperable, the remaining OPERABLE ADS flow paths are adequate to perform the required safety function as long as a single failure does not also occur. This Condition is equivalent to the worst case single failure of electrical division A or B, which would render one 4-inch flow path in ADS stage 1 and one 8-inch flow path in ADS stage 3 inoperable.

A Completion Time of 72 hours to restore one flow path to OPERABLE status is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a DBA during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.

If three or more flow paths in ADS stages 1, 2, or 3 are inoperable, Condition D is applicable.

D.1 and D.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met, Condition A and Condition B are entered concurrently, three or more flow paths in ADS stage 1, 2, or 3 are inoperable, or LCO 3.4.11 is not met for reasons other than Condition A, B, or C, the plant must be brought to MODE 5 where the probability and consequences on an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Each ADS stage 4 isolation motor operated valve must be verified to be open every 12 hours. Note that these valves receive confirmatory open signals. The Surveillance Frequency is acceptable considering valve position is manually monitored in the control room.

SR 3.4.11.2

This Surveillance requires verification that each ADS stage 1, 2, 3 valve strokes to its fully open position. Note that this surveillance is performed during shutdown conditions.

The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

SR 3.4.11.3

This Surveillance requires verification that each ADS stage 4 squib valve is OPERABLE in accordance with the Inservice Testing Program. The Surveillance Frequency for verifying valve OPERABILITY references the Inservice Testing Program.

The squib valves will be tested in accordance with the ASME OM Code (Ref. 5). The applicable ASME OM Code squib valve requirements are specified in paragraph ISTC-5260.

SR 3.4.11.4

This SR verifies that each Stage 1, 2, and 3 ADS valve actuate to the correct position on an actual or simulated actuation signal. The ADS and IRWST injection blocking device ACTUATION LOGIC TEST provides overlap with this Surveillance.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.5

This SR verifies that each Stage 4 ADS valve can actuate to the correct position on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. The ADS and IRWST injection blocking device ACTUATION LOGIC TEST provides overlap with this Surveillance. The OPERABILITY of the squib valves is checked by performing a continuity check of the circuit from the Protection Logic Cabinets to the squib valve.

This Surveillance is modified by a Note that excludes squib valve actuation as a requirement for this Surveillance to be met. This is acceptable because the design of the squib actuated valve was selected for this application because of its very high reliability. The OPERABILITY of squib actuated valves is verified by the Inservice Test Program for squib actuated valves.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

REFERENCES

1. FSAR Section 6.3, “Passive Core Cooling System.”
2. FSAR Section 15.6, “Decrease in Reactor Coolant Inventory.”
3. AP1000 Probabilistic Risk Assessment, Appendix A.
4. FSAR Section 3.9.6, “Inservice Testing of Pumps and Valves.”
5. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Automatic Depressurization System (ADS) – Shutdown, RCS Intact

BASES

BACKGROUND

A description of the ADS is provided in the Bases for LCO 3.4.11, "Automatic Depressurization System (ADS) – Operating."

APPLICABLE SAFETY ANALYSES

For postulated loss of normal decay heat removal events in MODE 5 with the RCS pressure boundary intact, the primary protection is the Passive Residual Heat Removal Heat Exchanger (PRHR HX). For loss of RCS inventory, injection of borated water into the core from the core makeup tanks (CMTs) may be required for makeup or boration. However, the amount of water necessary may not reduce the level in the CMTs to the point of ADS actuation.

In MODE 5, the ADS, in conjunction with the in-containment refueling water storage tank (IRWST) injection, provides the primary protection for mitigation of events requiring RCS makeup, boration, or core cooling. To ensure sufficient steam venting area is provided to adequately depressurize the RCS, thus enabling IRWST injection and containment recirculation, the number of ADS flow paths required to be OPERABLE depends on the level of reactor decay heat. Single failures are assumed to render any one flow path in ADS stage 1, 2, 3, or 4 inoperable. A single failure of electrical division A or B renders one 4-inch ADS stage 1 flow path and one 8-inch ADS stage 3 flow path inoperable. A single failure of electrical division C or D would render one 8-inch ADS stage 2 flow path inoperable.

The minimum ADS flow path requirements applicable 28 hours after shutdown (i.e., subcriticality) are based on the conditions during containment recirculation, which exhibits the minimum driving head for injection flow. Additional driving head is available when the IRWST still contains sufficient inventory early in events.

No LOCAs are postulated during plant operation in MODE 5, however loss of primary coolant through LEAKAGE or inadvertent draining may occur. For such shutdown events occurring in MODE 5 it is anticipated that the ADS will be actuated, allowing injection from the IRWST and the containment recirculation if containment flooding occurs (Ref. 1).

The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

The number of ADS flow paths required to be OPERABLE assures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses.

An ADS stage 1, 2, or 3 flow path is considered OPERABLE if both valves in the line are closed and OPERABLE (capable of opening on an actuation signal). In addition, an ADS stage 4 flow path is OPERABLE if the motor operated isolation valve is open and the squib valve is closed and OPERABLE. Each stage 4 squib valve is OPERABLE when capable of actuation by two igniters supplied by independent Class 1E electrical divisions.

If the reactor has been subcritical < 28 hours, five flow paths in ADS stage 1, 2, and 3 and four flow paths in ADS stage 4 are required OPERABLE. If the reactor has been subcritical for ≥ 28 hours, three flow paths in ADS stage 1, 2, and 3, with a minimum of two flow paths in ADS stage 2 or 3, and three flow paths in ADS stage 4 are required OPERABLE. A minimum of two flow paths in ADS stage 2 or 3 are required to ensure that sufficient steam venting area will be provided following the assumed single failure of an electrical division.

APPLICABILITY

In MODE 5 with the reactor coolant pressure boundary (RCPB) intact and pressurizer level ≥ 20%, all required flow paths of the ADS must be OPERABLE to mitigate the potential consequences of any event which causes a loss of decay heat removal or reduction in the RCS inventory.

The requirements for the ADS in MODES 1 through 4 are specified in LCO 3.4.11, “Automatic Depressurization System (ADS) – Operating;” and in MODE 5 with the RCS pressure boundary open, MODE 5 with the pressurizer level < 20%, and MODE 6 with upper internals in place, in LCO 3.4.13, “Automatic Depressurization System (ADS) – Shutdown, RCS Open.”

ACTIONS

A.1

If any one required flow path in ADS stage 1, 2, or 3 is determined to be inoperable, the remaining OPERABLE required flow paths in ADS stage 1, 2, and 3 are more than adequate to perform the required safety function as long as a single failure involving a flow path in ADS stage 4 or two additional flow paths in ADS stage 1, 2, or 3 does not also occur. A Completion Time of 7 days is based on the capability of the remaining ADS flow paths to perform the required safety functions assumed in the safety analyses and the low probability of a DBA occurring during this time period.

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BASES

ACTIONS (continued)

If more than one required flow paths in ADS stage 1, 2, or 3 are inoperable, Condition C or D is also applicable.

B.1

If any one required flow path in ADS stage 4 is determined to be inoperable, the remaining OPERABLE required flow paths in ADS stage 4 are adequate to perform the required safety function as long as a single failure does not also occur. A Completion Time of 72 hours is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a shutdown event during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.

C.1

If one required flow path in ADS stage 1 and one required flow path in ADS stage 2 or 3, or two flow paths in ADS stage 1 are determined to be inoperable, the remaining OPERABLE required ADS flow paths are adequate to perform the required safety function as long as a single failure does not also occur. This Condition is equivalent to the worst case single failure of electrical division A or B, which would render one 4-inch flow path in ADS stage 1 and one 8-inch flow path in ADS stage 3 inoperable.

A Completion Time of 72 hours to restore one required flow path to OPERABLE status is reasonable based on the capability of the remaining ADS valves to perform the required safety functions assumed in the safety analyses and the low probability of a shutdown event during this time period. This Completion Time is the same as is used for two train ECCS systems which are capable of performing their safety function without a single failure.

If three or more required flow paths in ADS stage 1, 2, or 3 are inoperable, Condition D is applicable.

D.1

If the Required Actions and associated Completion Times of Condition A, B, or C are not met, Condition A and Condition B are entered concurrently, three or more required flow paths in ADS stage 1, 2, or 3 are inoperable, or LCO 3.4.12 is not met for reasons other than Condition A, B, or C, action must be initiated, immediately, to restore compliance with the LCO and to continue this action until restoration is accomplished in order to provide the necessary ADS flow paths.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

The LCO 3.4.11 Surveillance Requirements are applicable to the ADS flow paths required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.4.11 for a discussion of each SR.

REFERENCES

1. FSAR Section 19E.4, “Safety Analyses and Evaluations.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 Automatic Depressurization System (ADS) – Shutdown, RCS Open

BASES

BACKGROUND

A description of the ADS is provided in the Bases for LCO 3.4.11, "Automatic Depressurization System (ADS) – Operating."

When all required flow paths in ADS stage 1, 2, and 3 are open, the RCS is considered VENTED. In this condition, the open flow paths in ADS stage 1, 2, and 3 provide sufficient vent area that, when combined with the required ADS stage 4 actuation, lower the RCS pressure to enable injection from the IRWST and containment recirculation.

During a normal Reactor Coolant System (RCS) cooldown from full power, MODE 5 is entered approximately 8 hours after reactor shutdown. In MODE 5, the Normal Residual Heat Removal System (RNS) is used to cool the RCS with Reactor Coolant Pumps (RCPs) in operation. Once the RCS temperature reaches 160°F, the RCPs may be stopped, the RCS pressure boundary may be opened and the pressurizer may be drained. During drain down to reduced RCS inventory conditions, automatic CMT actuation can be manually blocked when the pressurizer water level is below P-12 (approximately 16%) in order to prevent inadvertent CMT actuation on Pressurizer Water Level - Low 2 (approximately 10%).

APPLICABLE SAFETY ANALYSES

Because the CMTs may be isolated to prevent inadvertent injection, or steam flow through open flow paths in ADS stage 1, 2, or 3 may preclude automatic CMT actuation on low pressurizer level due to pressurizer inventory hold up, manual operator actuation of ADS stage 4 and IRWST injection may be required. This provides the primary protection for mitigation of events requiring RCS makeup, boration, or core cooling (Ref. 1).

The minimum ADS flow path requirements applicable 28 hours after shutdown (subcriticality) are based on the conditions during containment recirculation, which exhibits the minimum driving head for injection flow. Additional driving head is available when the IRWST still contains sufficient inventory early in events.

In MODE 6, with the upper internals removed, IRWST injection and containment recirculation can occur without actuation of ADS Stage 4.

The ADS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

When required flow paths in ADS stage 1, 2, and 3 are open, from the pressurizer through the spargers into the IRWST, and required flow paths in ADS stage 4 are OPERABLE sufficient vent area is available to support IRWST injection. Each stage 4 squib valve is OPERABLE when capable of actuation by two igniters supplied by independent Class 1E electrical divisions.

With the reactor subcritical < 28 hours, five ADS stage 1, 2, and 3 flow paths are required to be open and four ADS stage 4 flow paths are required to be OPERABLE. With the reactor subcritical for ≥ 28 hours, three ADS stage 1, 2, and 3 (maximum of one ADS stage 1) are required to be open and three ADS stage 4 flow paths are required OPERABLE. A maximum of one ADS stage 1 flow path is allowed to satisfy the LCO requirement to ensure that sufficient steam venting area will be provided following the assumed failure of an electrical division. After 28 hours the reduced number of ADS stage 1, 2, and 3 flow paths support vacuum fill operations when the discharge header of one train of flow paths in ADS stage 1, 2, and 3 is blocked, rendering those flow paths unavailable to meet the LCO requirement.

An ADS stage 4 flow path is OPERABLE if the motor operated isolation valve is open and the squib valve is OPERABLE. Each stage 4 squib valve is OPERABLE when capable of actuation by two igniters supplied by independent Class 1E electrical divisions.

The LCO is modified by a Note that allows required flow paths in ADS stage 1, 2, and 3 to be closed for two operational evolutions, provided their OPERABILITY meets LCO 3.4.12, ADS - Shutdown, RCS Intact. The first is to facilitate vacuum fill operations until a pressurizer water level ≥ 20% is established, and the second is to facilitate compliance during transitions between LCO 3.4.12 and LCO 3.4.13.

APPLICABILITY

In MODE 5 with pressurizer level < 20%, in MODE 5 with the reactor coolant system pressure boundary (RCPB) open, and in MODE 6 with the upper internals in place, the required flow paths in ADS stage 1, 2, and 3 must be open and required flow paths in ADS stage 4 must be OPERABLE to support IRWST injection and containment recirculation, if needed, in order to mitigate events requiring RCS makeup, boration, or core cooling (Ref 1).

The requirements for the ADS in MODES 1 through 4 are specified in LCO 3.4.11, “Automatic Depressurization System (ADS) – Operating;” and in MODE 5 with the RCPB intact and pressurizer water level ≥ 20% in LCO 3.4.12, “Automatic Depressurization System (ADS) – Shutdown, RCS Intact.”

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BASES

ACTIONS

A.1 and A.2

If one required flow path in ADS stage 1, 2, or 3 is not open, action must be taken to open the required flow path or open alternative flow path(s) with an equivalent area within 72 hours. The removal of one pressurizer safety valve is an acceptable alternate flow path for a flow path in ADS stage 1, 2, or 3. The removal of a pressurizer safety valve or the pressurizer manway is also an acceptable alternative.

A Completion Time of 72 hours is reasonable based on capability of the remaining open flow paths in ADS stage 1, 2, or 3 to provide the required vent area assumed in safety analyses, and the low probability of a DBA occurring during this time period. This completion time is the same as is used for two train ECCS systems which are capable of performing their safety function without an additional single failure.

B.1 and B.2

If one required flow path in ADS stage 4 is inoperable, action must be taken to restore the required flow path in ADS stage 4 to OPERABLE status or open an alternative flow path with an equivalent area within 36 hours. In this Condition the remaining open flow paths in ADS stage 1, 2, and 3 and the OPERABLE flow paths in ADS stage 4 are adequate to perform the required safety function without an additional single failure. Required ADS stage 4 flow paths may be restored to OPERABLE status, or the required vent area may be restored by opening alternate vent path with an equivalent area.

An acceptable alternate ADS stage 4 vent path is a manway cover on the hot leg side of the steam generator channel head. Note that the hot leg nozzle dam must not be installed to credit removal of the steam generator manway as an alternate flow path. The pressurizer manway is not an acceptable alternate for ADS stage 4 vent path as pressurizer surge line flooding phenomena can negate the IRWST elevation head necessary for successful gravity injection. Therefore, a Completion Time of 36 hours is considered acceptable.

C.1 and C.2

While in MODE 5 or 6 if the Required Actions and associated Completion Times of Condition A or B are not met, Condition A and Condition B are entered concurrently, or LCO 3.4.13 is not met for reasons other than Condition A or B, action must be immediately initiated to restore compliance with the LCO and to continue this action until restoration is accomplished in order to provide the necessary ADS flow paths. In addition, the plant must be placed in a condition which minimizes the likelihood and consequences of an event. This is accomplished by

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BASES

ACTIONS (continued)

immediately suspending positive reactivity additions is required to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of control rods, and excessive cooling of the RCS.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Each flow path in ADS stage 1, 2, and 3 required to be open is verified to be open by verifying that each required ADS stage 1, 2, and 3 valve is in its open position every 12 hours, as indicated in the control room. This Surveillance Frequency is acceptable based on administrative controls which preclude repositioning the valves.

SR 3.4.13.2

SR 3.4.11.1, SR 3.4.11.3, and SR 3.4.11.5 are applicable to the ADS stage 4 valves required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.4.11 for a discussion of each SR.

REFERENCES

1. FSAR Section 19E.4.8.3, “Loss of Normal Residual Heat Removal System Cooling in Mode 5 with Reactor Coolant System Open.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 Low Temperature Overpressure Protection (LTOP)

BASES

BACKGROUND

LTOP limits RCS pressure at low temperatures so that the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The Pressure and Temperature Limits Report (PTLR) provides the limits which set the maximum allowable setpoints for the Normal Residual Heat Removal System (RNS) suction relief valves, RNS-PL-V020 (1 inch) and RNS-PL-V021 (3 inch). LCO 3.4.3 provides the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, “RCS Pressure and Temperature (P/T) Limits,” requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a maximum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires isolating the accumulators and closing CVS makeup line containment isolation valve, CVS-PL-V091. The pressure relief capacity requires the RNS suction relief valves or a depressurized RCS and an RCS vent of sufficient size. The RNS suction relief valves or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

RNS Suction Relief Valve Requirements

During the LTOP MODES, the RNS system is operated for decay heat removal. Therefore, the RNS suction isolation valves are open in the

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BASES

BACKGROUND (continued)

piping from the RCS hot legs to the inlet of the RNS system. While these valves are open, the RNS suction relief valves are exposed to the RCS and able to relieve pressure transients in the RCS.

The RNS suction relief valves are spring loaded, water relief valves with a pressure tolerance and an accumulation limit established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

The RNS suction isolation valves must be open to make the RNS suction relief valves OPERABLE for RCS overpressure mitigation.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it may require removing one or more pressurizer safety valves or manually opening one or more Automatic Depressurization System (ADS) valves. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 all four cold leg temperatures > 275°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. When the RNS is aligned and open to the RCS, overpressure protection is provided by the RNS suction relief valves, or a depressurized RCS and a sufficiently sized open RCS vent.

The actual temperature at which the pressure in the P/T limit curve falls below the suction relief setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, LTOP must be re-evaluated to ensure its functional requirements can still be met using the RNS suction relief valves, or the depressurized and vented RCS condition.

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APPLICABLE SAFETY ANALYSES (continued)

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients. The events listed below were used in the analysis to size the RNS suction relief valves. Therefore, any events with a mass or heat input greater than the listed events cannot be accommodated and must be prevented.

Mass Input

a. Makeup water flow rate to the RCS assuming one or both CVS makeup pumps are in operation and letdown is isolated.

Heat Input

a. Restart of one reactor coolant pump (RCP) with water in the steam generator secondary side 50°F hotter than the primary side water, and the RCS water solid.

RNS Suction Relief Valve Performance

Since each of the RNS suction relief valves do not have a variable P/T lift setpoint, the analysis must show that with chosen setpoint, the relief valves will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the lowest of either the P/T limit curve, 110% of the design pressure of the normal residual heat removal system, or the acceptable RNS relief valve inlet pressure.

When the RCS is aligned to the RNS with RNS suction relief valves OPERABLE, Chemical and Volume Control System (CVS) valve, CVS-PL-V091, is closed to limit the flow rate to the capacity of the smaller RNS suction relief valve by directing flow to CVS flow-restricting orifice, CVS-PY-R10. During a mass input transient, the smaller RNS suction relief valve will lift at its lift setting and the larger RNS suction relief valve will remain closed. If the pressure exceeds the capacity of the smaller RNS suction relief valve, the larger RNS suction relief valve will open.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

During a heat input transient, the larger RNS suction relief valve is relied upon for overpressure protection and the smaller RNS suction relief valve opens to provide reliability and reduce potential chattering in the larger valve. The heat input transient is limiting when the RNS suction relief valves are relied upon for overpressure protection.

To prevent the possibility of a heat input transient, and thereby limit the required flow rate of the RNS suction relief valves, administrative requirements in the Note for LCO 3.4.3, “RCS Pressure and Temperature (P/T) Limits,” have been imposed for starting an RCP.

RCS Vent Performance

With the RCS depressurized, a vent size of 4.15 square inches is capable of mitigating a limiting overpressure transient. The area of the vent is equivalent to the area of the inlet pipe to the larger RNS suction relief valve so the capacity of the vent is greater than the flow possible with either the mass or heat input transient, while maintaining the RCS pressure less than the lower of either the maximum pressure on the P/T limit curve or 110% percent of the design pressure of the normal residual heat removal system. When an RCS vent of ≥ 4.15 square inches is established, CVS-PL-V091 may be opened.

The required vent area may be obtained by opening one ADS Stage 1, 2, or 3 flow path.

The RCS vent size will be reevaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

LTOP satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that LTOP is OPERABLE. LTOP is OPERABLE when the maximum coolant input and minimum pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires all accumulator discharge isolation valves closed and immobilized, when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed in the PTLR.

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BASES

LCO (continued)

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are one of the following:

a. Two OPERABLE RNS suction relief valves.

An RNS suction relief valve is OPERABLE for LTOP when both RNS suction isolation valves in one flow path are open, its setpoint is set within the Inservice Testing Program limits, and testing has proven its ability to open at this setpoint. Additionally, CVS-PL-V091 is closed to limit the flow rate to within the relieving capacity of the smaller RNS suction relief valve when the RNS suction relief valves are OPERABLE; or

b. A depressurized RCS and an RCS vent path is open with an area of ≥ 4.15 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

The LCO is modified by a Note that provides that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

APPLICABILITY

This LCO is applicable in MODE 4 when any cold leg temperature is ≤ 275°F, MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits with all four cold leg temperatures above 275°F. In MODE 6 with the reactor vessel head off an overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.6, “Pressurizer Safety Valves,” requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 with all four cold leg temperature > 275°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure with little or no time for operator action to mitigate the event.

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BASES

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP System. Due to the increased risk associated with an inoperable LTOP system, the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

The Completion Time is based on operating experience that this activity can be accomplished in this time period and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed time.

B.1

CVS makeup line containment isolation valve, CVS-PL-V091, requires closure within 1 hour. Condition B is modified by a Note stating that the Action is only applicable when an RCS vent of ≥ 4.15 square inches is not established.

The Completion Time is based on operating experience that this activity can be accomplished in this time period and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed time.

C.1 and C.2

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 275°F, the accumulator pressure cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

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BASES

ACTIONS (continued)

D.1 and D.2

If the RNS suction relief valves are inoperable and the RCS is not depressurized, there is a potential to overpressurize the RCS and exceed the limits allowed in LCO 3.4.3. The suction relief valves are considered inoperable if the RNS isolation valves have isolated the RNS from the RCS in such a way that the suction relief valves cannot perform their intended safety function, or if the valves themselves will not operate to perform their intended safety function. If the RCS is depressurized but the RCS vent path does not provide a flow area sufficient to mitigate any of the design low temperature overpressure events and the RNS suction relief valves are inoperable, there is a potential to overpressurize the RCS and exceed the limits allowed in LCO 3.4.3. The RCS vent path is considered inoperable if the area of the vent is not equivalent to the area of the inlet pipe to the larger RNS suction relief valve.

Under these conditions, Required Action D.1 or D.2 provide two options, either of which must be accomplished in 12 hours. If the RNS suction relief valves cannot be restored to OPERABLE status, the RCS must be depressurized and vented with a RCS vent which provides a flow area sufficient to mitigate any of the design low temperature overpressure events.

The 12 hour Completion Time represents a reasonable time to repair the relief valves, open the RNS isolation valves or otherwise restore the LTOP to OPERABLE status, or depressurize and vent the RCS, without imposing a lengthy period when the LTOP is not able to mitigate a low temperature overpressure event.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, the accumulator discharge isolation valves are verified closed and locked out. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the main control room to verify the required status of the equipment.

SR 3.4.14.1 is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note requires the accumulator discharge isolation valve Surveillance to be met only under these pressure and temperature conditions.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2

The RNS suction relief valves shall be demonstrated OPERABLE by verifying two RNS suction isolation valves in one flow path are open. This Surveillance is only performed if the RNS suction relief valves are being used to satisfy this LCO.

The RNS suction isolation valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RNS suction isolation valves remain open. This Surveillance is required to be met if the RNS suction relief valves are being used to satisfy the pressure relief requirements of LCO 3.4.14.a.

SR 3.4.14.3

To minimize the potential for a low temperature overpressure event when the RNS suction relief valves are OPERABLE, the flow from the CVS is limited to the capacity of the smaller RNS suction relief valve. The CVS makeup line containment isolation valve, CVS-PL-V091, is closed to limit CVS mass and heat injection flow to the CVS flow-restricting orifice, CVS-PY-R10. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the main control room to verify the required status of the equipment. This Surveillance is required to be met if the RNS suction relief valves are being used to satisfy the pressure relief requirements of LCO 3.4.14.a.

SR 3.4.14.4

The RCS vent of ≥ 4.15 square inches is verified open either:

a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered “locked” in this context) or

b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position or a removed pressurizer safety valve or open manway also fits this category).

This Surveillance is modified by a Note that states it is only required to be met if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.14.b.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.5

The RNS suction relief valves shall be demonstrated OPERABLE by verifying that two RNS suction isolation valves in one flow path are open and by testing them in accordance with the Inservice Testing Program. (Refer to SR 3.4.14.2 for the RNS suction isolation valve Surveillance.) The ASME OM Code (Ref. 5) test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint. This Surveillance is modified by a Note specifying that it is only required to be met if the RNS suction relief valves are being used to satisfy the pressure relief requirements of LCO 3.4.14.a.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, “Fracture Toughness Requirements.”
2. Generic Letter 88-11, “NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation.”
3. ASME Boiler and Pressure Vessel Code, Section III.
4. FSAR Section 5.2.2, “Overpressure Protection.”
5. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Pressure Isolation Valve (PIV) Integrity

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define the RCS pressure boundary as all those pressure containing components such as pressure vessels, piping, pumps, and valves which are connected to the reactor coolant system, up to and including the outermost containment isolation valve in system piping which penetrates primary reactor containment, the second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment, and the reactor coolant system safety and relief valves. This includes any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can experience varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The PIVs are listed in FSAR Chapter 3, Table 3.9-18. The RCS PIV Leakage LCO allows RCS high pressure operation when PIV leakage has been verified to be within limits.

The purpose of this specification is to prevent overpressure failure or degradation of low pressure portions of connecting systems. The following criteria was used in identifying PIVs for inclusion in the specification. A valve was included in this specification if its failure may result in:

1. Failure of low pressure portions of connected systems, such as a Loss of Coolant Accident (LOCA) outside of containment, which could place the plant in an unanalyzed condition.

2. Degradation of low pressure portions of connected systems, such as damage to a core cooling system, which could degrade a safety related function that mitigates a DBA.

Valves considered for inclusion in this specification are used to isolate the RCS from the following connected systems:

a. Passive Core Cooling System (PXS) Accumulators;

b. Normal Residual Heat Removal System (RNS); and

c. Chemical and Volume Control System (CVS).

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BASES

BACKGROUND (continued)

The RNS pressure boundary isolation valves meet the first criterion for inclusion in this specification. The PXS accumulator check valves were determined to meet the second PIV criteria for inclusion in this specification. It is determined that the CVS PIVs do not meet either criterion for inclusion in this specification.

The PIVs that are addressed by this specification are listed in FSAR Chapter 3, Table 3.9-18.

The CVS pressure isolation valves were not included in this specification based on the defined criteria. The justification for excluding the CVS PIVs is discussed in the following paragraph.

The CVS contains five high pressure/low pressure connections with the RCS. Since the portion of the CVS which is located inside reactor containment is designed to full RCS pressure, the high pressure/low pressure interfaces with the RCS are the lines that penetrate the reactor containment. The CVS lines that penetrate containment include the makeup line, the letdown line to the Liquid Radwaste System, the hydrogen supply line, the zinc supply line, and the demineralizer resin sluice line used to transfer spent resins from the demineralizers to the Solid Radwaste System. These lines each contain two safety related containment isolation valves which are addressed by the Containment Isolation Specification (LCO 3.6.3). In addition to the containment isolation valves in each of the CVS lines that interface with the RCS, there are additional valves in each line that provide diverse isolation capability. Since more restrictive requirements are imposed by LCO 3.6.3, the CVS isolation valves are not included in this LCO.

Since the purpose of this LCO is to verify that the PIVs have not suffered gross failures, the valve leakage test in conjunction with tests specified in the IST program provide an acceptable method of determining valve integrity. The ability of the valves to transition from open to closed provides assurance that the valve can perform its pressure isolation function as required. A small amount leakage through these valves is allowed, provided that the integrity of the valve was demonstrated.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system or the failure of a safety related function to mitigate a DBA.

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BASES

APPLICABLE SAFETY ANALYSES

Pressure isolation valve integrity is not considered in any design basis accident analyses. This specification provides for monitoring the condition of the reactor coolant pressure boundary to detect degradation which could lead to accidents or which could impair a connected system's ability to mitigate DBAs.

RCS PIV integrity satisfies, Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually small. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per inch nominal valve size up to a maximum of 5 gpm per valve. This limit is well within the makeup capability of the CVS makeup pumps. This leak rate will not result in the overpressure of a connected low pressure system. Reference 5 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage of the valve. In such cases, the observed leakage rate at lower differential pressures shall be adjusted to the maximum pressure differential assuming leakage is directly proportional to the pressure differential to the one-half power.

APPLICABILITY

In MODES 1, 2, and 3 and MODE 4, with RCS not being cooled by the RNS, this LCO applies when the RCS is pressurized.

In MODE 4, with RNS in operation, and MODES 5 and 6, the RCS pressure is reduced and is not sufficient to overpressurize the connected low pressure systems.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The pressurization may have affected system OPERABILITY, or isolation of an affected flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

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BASES

ACTIONS (continued)

A.1

With one or more PIVs inoperable, the affected flow path(s) must be isolated. Required Action A.1 is modified by a Note that the valves used for isolation must meet the same integrity requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required Action A.1 requires that the isolation with one valve must be performed within 8 hours. Eight hours provides time to verify IST compliance for the alternate isolation valve and isolate the flow path. The 8 hour Completion Time allows the actions and restricts the operation with inoperable isolation valves.

A.2

Required Action A.2 specifies verification that a second OPERABLE PIV can meet the leakage limits. This valve is required to be a check valve, or a closed valve, if it isolates a line that penetrates containment. For the accumulator valves, the normally open accumulator isolation valve is a suitable replacement PIV, but can remain open because leakage into the accumulator is continuously monitored. If leakage into the accumulators increased to the allowable operational leakage limit, then the valve could be used to isolate the accumulators from the RCS.

The 72 hour Completion Time allows the actions and restricts the operation with inoperable isolation valves.

B.1 and B.2

If PIV integrity cannot be restored, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and reduces the potential for a LOCA outside containment.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking

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BASES

SURVEILLANCE REQUIREMENTS (continued)

valve. The leakage limit of 0.5 gpm per inch nominal valve size up to a maximum of 5 gpm applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing shall be performed every 24 months, a typical refueling cycle. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 4) as contained in the Inservice Testing Program and is within frequency allowed by the American Society of Mechanical Engineers (ASME) OM Code (Ref. 5).

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. 10 CFR 50.55a(g).
5. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 Reactor Vessel Head Vent (RVHV)

BASES

BACKGROUND

The reactor vessel head vent (RVHV) is designed to assure that long-term operation of the Core Makeup Tanks (CMTs) does not result in overfilling of the pressurizer during Condition II Design Basis Accidents (DBAs). The RVHV can be manually actuated by the operators in the main control room to reduce the pressurizer water level during long-term operation of the CMTs.

The RVHV consists of two parallel flow paths each containing two RVHV isolation valves in series. The RVHV valves are connected to the reactor vessel head via a common line. The outlets of the RVHV flow paths combine into one common discharge line which connects to a single ADS discharge header that discharges to spargers located in the incontainment refueling water storage tank (IRWST). The RVHV valves are 1 inch valves with DC solenoid operators.

The RVHV valves are designed to open when actuated by the operator, and to reclose when actuated by the operator from the main control room.

The number and capacity of the RVHV flow paths are selected so that letdown flow from the RCS is sufficient to prevent pressurizer overfill for events where extended operation of the CMTs causes the pressurizer water level to increase. Although realistic evaluations of the Condition II non-LOCA events does not result in pressurizer overfill, conservative analyses of some of these events can result in pressurizer overfill if no operator actions are assumed.

APPLICABLE SAFETY ANALYSES

For Condition II non-LOCA events, such as inadvertent passive core cooling system operation and chemical and volume control system malfunction, the use of the RVHV may be required to prevent long-term pressurizer overfill (Ref. 1).

For LOCA events, the RVHV is not required.

The RVHV satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that all four RVHV valves be OPERABLE ensures that upon actuation, the RVHV can reduce the pressurizer water level as assumed in the DBA safety analyses.

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BASES

LCO (continued)

For the RVHV to be considered OPERABLE, all four valves must be closed and OPERABLE (capable of opening from the main control room).

APPLICABILITY

In MODES 1, 2, and 3, and MODE 4 with all four cold leg temperatures > 275°F, the RVHV must be OPERABLE to mitigate the potential consequences of any event which causes an increase in the pressurizer water level that could otherwise result in overfilling of the pressurizer.

In MODE 4, any cold leg temperature ≤ 275°F, and in MODES 5 and 6, pressurizer overfill is not a concern since overpressure protection in these MODES is provided by LCO 3.4.14, Low Temperature Overpressure Protection (LTOP), and the credited valves are designed for water relief.

ACTIONS

A.1

If one or two RVHV valves in a single flow path are determined to be inoperable, the flow path is inoperable. The remaining OPERABLE RVHV flow path is adequate to perform the required safety function. A Completion Time of 72 hours is acceptable since the OPERABLE RVHV paths can mitigate DBAs without a single failure.

B.1

If both flow paths are determined to be inoperable, the RVHV is degraded such that the system is not available for some DBA non-LOCA analyses for which it may be required. A Completion Time of 6 hours is permitted to restore at least one flow path. This Completion Time is acceptable considering that the realistic analysis of these non-LOCA events do not result in pressurizer overfill.

C.1 and C.2

If the Required Actions and associated Completion Times are not met the plant must be brought to MODE 4 with at least one cold leg temperature ≤ 275°F where the probability and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with at least one cold leg temperature ≤ 275°F within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

The dedicated component level remote manual valve switches in the main control room shall be used to stroke each RVHV valve to demonstrate OPERABILITY of the controls.

This Surveillance requires verification that each RVHV valve strokes to its fully open position. The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

REFERENCES

1. FSAR Section 15.5, “Increase in Reactor Coolant System Inventory.”

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.4, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.4, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.4. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

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BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.7, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere includes primary to secondary SG tube LEAKAGE equivalent to the operational leakage limit of 150 gpd per SG. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.10, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.34 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.4, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 150 gpd per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

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BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.7, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination

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ACTIONS (continued)

is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

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SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.4 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.4 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of once prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

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BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.34.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 Pressurizer Heater Circuit Breakers

BASES

BACKGROUND

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. Power to the pressurizer heaters is blocked when the core makeup tanks are actuated. This action reduces the potential for steam generator overfill for a steam generator tube rupture accident.

Pressurizer heaters are automatically tripped to reduce the potential for steam generator overfill and automatic ADS Stages 1, 2, and 3 actuation for a steam generator tube rupture event. Automatically tripping the pressurizer heaters reduces the pressurizer level swell for certain non-LOCA events such as loss of normal feedwater, inadvertent CMT operation, and CVS malfunction resulting in an increase in RCS inventory. For small break LOCA analysis, tripping the pressurizer heaters supports depressurization of the RCS following actuation of the CMTs.

As described in FSAR Subsection 7.3.1.2.23, Division A of the protection and safety monitoring system provides actuation signals to five load center circuit breakers which provide the power feed to five pressurizer heater electrical control centers. When these five power feed breakers are opened, the electrical power is removed from the pressurizer heaters. In addition, each division of the protection and safety monitoring system provides a separate signal to the plant control system. Within the plant control system, these signals are voted two-out-of four and the result is used to open the pressurizer heater circuits at the motor control center. This arrangement provides for complete disabling of the pressurizer heaters, even if a single component failure occurs. Pressurizer heater trip on High-3 pressurizer water level may be manually blocked when wide range RCS pressure is below the P-19 interlock.

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BASES

APPLICABLE SAFETY ANALYSES

For Condition II non-LOCA events, such as inadvertent operation of the core makeup tanks during power operation (Ref. 1) and chemical and volume control system malfunction that increases reactor coolant inventory (Ref. 2), the blocking of pressurizer heater operation may be required to prevent long-term pressurizer overfill.

For the steam generator tube rupture events, power to the pressurizer heaters is shut off so that they will not provide additional heat to the primary should the pressurizer level return (Ref. 3).

The pressurizer heater circuit breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement that each pressurizer heater circuit breaker be OPERABLE ensures that upon signal to block pressurizer heaters, the breakers will trip open as assumed in the DBA safety analyses.

For a pressurizer heater circuit breaker to be considered OPERABLE, it must be capable of tripping on an actuation signal or maintained in the tripped condition.

APPLICABILITY

In MODES 1, 2, and 3, and MODE 4 with all four cold leg temperatures > 275°F, the pressurizer heater circuit breakers must be OPERABLE to mitigate the potential consequences of any event which causes an increase in the pressurizer water level that could otherwise result in overfilling of the pressurizer.

In MODE 4, any cold leg temperature ≤ 275°F, and in MODES 5 and 6, pressurizer overfill is not a concern since overpressure protection in these MODES is provided by LCO 3.4.14. Low Temperature Overpressure Protection (LTOP), and the credited valves are designed for water relief.

ACTIONS

The actions are modified by a Note indicating that separate Condition entry is allowed for each pressurizer heater. This is acceptable, since the Required Actions provide appropriate compensatory actions for each pressurizer heater with inoperable breaker(s). Complying with the Required Actions allows subsequent inoperable circuit breakers to be governed by subsequent Condition entry and application of associated Required Actions.

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BASES

ACTIONS (continued)

A.1

If one or more pressurizer heaters with one circuit breaker is inoperable, action must be taken to restore the breaker(s) to OPERABLE status within 6 hours. The 6 hours allowed to restore the inoperable breaker(s) is reasonable based on the low probability of an event occurring during this interval.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met or if one or more pressurizer heaters with two circuit breakers are determined to be inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with at least one cold leg temperature ≤ 275°F within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.18.1

This SR demonstrates that the pressurizer heater circuit breakers trip open in response to an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the circuit breakers. The OPERABILITY of the motor control center breakers is checked by opening these breakers using the Plant Control System.

The SR excludes circuit breakers that are maintained in the tripped position since a tripped open circuit breaker is already performing the safety function. Restoration of a circuit breaker to the non-tripped position requires verification that the SR has been met within its required Frequency.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation. This Frequency is adequate based on the use of multiple circuit breakers to prevent the failure of any single circuit breaker from disabling the function and that all circuit breakers are tested.

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BASES

REFERENCES

1. FSAR Section 15.5.1, “Inadvertent Operation of the Core Makeup Tanks During Power Operation.”
2. FSAR Section 15.5.2, “Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory.”
3. FSAR Section 15.6.3, “Steam Generator Tube Rupture."

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.19 Auxiliary Spray and Purification Line Isolation Valves

BASES

BACKGROUND

The Chemical and Volume Control System (CVS) maintains the reactor coolant system (RCS) fluid purity and activity level within acceptable limits. The CVS purification line receives flow from the discharge of the reactor coolant pumps (RCPs). The CVS also provides auxiliary spray to the pressurizer. To preserve the reactor coolant inventory in the event of a break in the CVS loop piping, the auxiliary spray line, purification line, zinc addition line, and hydrogen addition line are isolated on Pressurizer Water Level - Low to help maintain reactor coolant system inventory.

In the case of CVS-V092 & -V094 (Zinc Injection Isolation) and CVS-V219 (Hydrogen Injection Isolation) these end devices are also Containment Isolation Valves and have OPERABILITY covered by LCO 3.6.3, "Containment Isolation Valves." As such, only CVS-V084 (Aux Spray Isolation) and CVS-V001, -V002, -V003 (Purification Line Isolations) are addressed by this LCO.

Auxiliary Spray and Purification Line Isolation is actuated on the following signals:

- Pressurizer Water Level - Low; and
- Chemical and Volume Control System Makeup Isolation – Manual Initiation.

APPLICABLE SAFETY ANALYSES

To help preserve the reactor coolant inventory in the event of a break in the CVS loop piping, the auxiliary spray and purification line isolation valves can be closed (Ref. 1).

The auxiliary spray and purification line Isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The requirement for OPERABILITY of the CVS auxiliary spray and purification line isolation valves ensures that upon signal to isolate, the valves will be isolated. For each auxiliary spray and purification line isolation valve to be considered OPERABLE, it must be capable of tripping on an actuation signal.

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BASES

APPLICABILITY

This Function is required to be OPERABLE in MODES 1 and 2. In MODES 3, 4, 5, and 6, this Function is not needed for accident detection and mitigation

ACTIONS

The ACTIONS are modified by two Notes. Note 1 allows the auxiliary spray and purification line flow path to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for auxiliary spray and purification line isolation is indicated.

Note 2 indicates that separate Condition entry is allowed for each flow path. This is acceptable, since the Required Actions provide appropriate compensatory actions for each inoperable isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

A.1

In the event the auxiliary spray flow path isolation valve and/or one or two of the purification line flow path isolation valves is inoperable, the valve(s) must be restored to OPERABLE status in 6 hours. The 6 hour Completion Time is reasonable considering the time required to isolate the penetration and the relative importance of supporting automatic RCS isolation.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met or if three purification line flow path isolation valves are inoperable then the affected flow path must be isolated by use of at least one closed and deactivated automatic valve within 6 hours. Alternatively, the plant must be placed in MODE 3 in which the likelihood and consequences of an event are minimized. The 6 hour Completion Time is a reasonable time based on operating experience, to isolate the flow path or reach MODE 3 from full power in an orderly manner and without challenging plant systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1

This SR demonstrates that the auxiliary spray and purification line isolation valves actuate to the isolation position in response to an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

REFERENCES

1. FSAR Section 7.3.1.2.18 “Auxiliary Spray and Purification Line and Zinc/Hydrogen Addition Isolation."

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.20 CVS Letdown Isolation Valves

BASES

BACKGROUND	The CVS provides letdown to the liquid radwaste system to maintain the pressurizer level. To help maintain RCS inventory in the event of a LOCA, the CVS letdown isolation valves (CVS-V045 and CVS-V047) are actuated on Hot Leg Level - Low 2.
APPLICABLE SAFETY ANALYSES	Letdown isolation assists the operators when draining the RCS to a mid-loop level. If the operators fail to isolate letdown, these channels send a signal to close the letdown valves and stop the draining process (Ref. 1). The CVS letdown isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The requirement for OPERABILITY of the CVS letdown isolation valves ensures that upon signal to isolate, the penetration will be isolated. For each CVS letdown isolation valve to be considered OPERABLE, it must be capable of isolating on an actuation signal.
APPLICABILITY	This Function is required to be OPERABLE in MODE 5 below the P-12 (Pressurizer Level) interlock, and in MODE 6 with the water level < 23 feet above the top of the reactor vessel flange to assist the operators when draining the RCS to a mid-loop level. In other conditions, the RCS is not being drained to mid-loop and operator assistance is not needed.
ACTIONS	The Actions are modified by a Note allowing the CVS letdown flow path to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for CVS letdown isolation is indicated.

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BASES

ACTIONS (continued)

A.1

If one CVS letdown isolation valve is inoperable, action must be taken to restore the valve within 72 hours. The 72 hours allowed to restore the inoperable valve is reasonable based on the low probability of an event occurring during this interval and OPERABILITY of the redundant letdown isolation valve.

B.1

If the Required Action or associated Completion Time of Condition A is not met or two CVS letdown isolation valves are inoperable, action must be initiated immediately to isolate the CVS letdown flow. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve.

SURVEILLANCE REQUIREMENTS

SR 3.4.20.1

This SR demonstrates that the CVS letdown isolation valves actuate to the isolation position in response to an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown to prevent any upsets of plant operation.

REFERENCES

1. FSAR Section 19E.2.1.2.4, "Improved RCS Draindown Method."

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.1 Accumulators

BASES

BACKGROUND

Two redundant PXS accumulators provide sufficient water to the reactor vessel during the blowdown phase of a large-break loss-of-coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, to provide Reactor Coolant System (RCS) makeup for a small-break LOCA, and to provide RCS boration for steam line breaks (Ref. 2).

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The accumulator inventory is available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core.

The accumulators are pressure vessels, partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required for them to perform their function. Internal accumulator pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the static accumulator pressure.

Each accumulator is piped into the reactor vessel via an accumulator line and is isolated from the RCS by two check valves in series.

A normally open motor operated valve is arranged in series with the check valves. Upon initiation of a safeguards actuation signal, the normally open valves receive a confirmatory open signal.

Power lockout and position alarms ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 603-1991 (Ref. 1) for “operating bypasses” and that the accumulators will be available for injection without being subject to a single failure.

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BASES

BACKGROUND (continued)

The accumulator size, water volume, and nitrogen cover pressure are selected so that both of the accumulators are sufficient to recover the core cooling before significant clad melting or zirconium water reaction can occur following a large break LOCA. One accumulator is adequate during a small break LOCA where the entire contents of one accumulator can possibly be lost via the pipe break. This accumulator performance is based on design basis accident (DBA) assumptions and models (Ref. 3). The Probabilistic Risk Assessment (PRA) (Ref. 4) shows that one of the two accumulators is sufficient for a large break LOCA caused by spurious ADS actuation and that none of the accumulators are required for small break LOCAs, assuming that at least one core makeup tank (CMT) is available. In addition, both accumulators are required for a large break LOCA caused by the break of a cold leg pipe; the probability of this break has been significantly reduced by incorporation of leak-before-break.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed to be OPERABLE in both the large and small break LOCA analyses at full power (Ref. 3) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

For a small break LOCA, a large range of break sizes and locations were analyzed to verify the adequacy of the design. The cases analyzed include the rupture of one 8 inch direct vessel injection line and several smaller break sizes. Acceptable PXS performance was demonstrated.

For a larger LOCA, including a double ended RCS piping rupture, the PXS can provide a sufficiently large flow rate, assuming both accumulators are OPERABLE, to quickly fill the reactor vessel lower plenum and downcomer. Both accumulators, in conjunction with the CMTs, ensure rapid reflooding of the core. For a large LOCA, both lines are available since an 8 inch line break would be a small LOCA.

Following a non-LOCA event such as a steam line break, the RCS experiences a decrease in temperature and pressure due to an increase in energy removal by the secondary system. The cooldown results in a reduction of the core SHUTDOWN MARGIN with a potential for return to power. During such an event the accumulators provide injection of borated water to assist the CMT’s boration to mitigate the reactivity transient and ensure the core remains shut down.

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

This LCO establishes the limits necessary to ensure that accumulator parameters meet the initial conditions assumed in the safety analyses to satisfy the necessary acceptance criteria established for core cooling by 10 CFR 50.46 (Ref. 5). These conditions are:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. The core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For an accumulator to be OPERABLE, the isolation valve must be fully open (and power removed when pressurizer pressure is ≥ 2000 psig), and the limits established in the Surveillance Requirements for contained water, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODES 3 and 4 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that adequate injection flow from other sources exists to retain peak clad temperatures below the 10 CFR 50.46 limit of 2200°F.

In MODES 3 and 4 with RCS pressure ≤ 1000 psig, and in MODES 5 and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows the RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

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BASES

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, action must be taken to restore the parameter.

Deviations in boron concentration are expected to be slight, considering that the pressure and volume are verified once per 12 hours. For one accumulator, boron concentration not within limits will have an insignificant effect on the ability of the accumulators to perform their safety function. Therefore, a Completion Time of 72 hours is considered to be acceptable.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 8 hours. With one accumulator inoperable, the remaining accumulator is capable of providing the required safety function, except for one low probability event (large cold leg LOCA) discussed in the background section. The effectiveness of one accumulator is demonstrated in analysis performed to justify PRA success criteria (Ref. 4). The analysis contained in this reference shows that for a range of other events including small break LOCAs and large hot leg break LOCAs that with one accumulator unavailable the core is adequately cooled.

The 8 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time is reasonable since the CMTs are required to be available to provide small break LOCA mitigation. The effectiveness of backup CMT injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one CMT without any accumulator injection supports adequate core cooling. This analysis provides a high confidence that with the unavailability of one accumulator, the core can be cooled following design bases accidents.

If LCO 3.5.2 Condition C or E is also entered concurrent with this Condition, then a 1 hour Completion Time from discovery of LCO 3.5.1 Condition B entry concurrent with LCO 3.5.2 Condition C or E entry also applies. This 1 hour Completion Time requires very prompt actions to

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BASES

ACTIONS (continued)

restore either the accumulator or the CMT (per LCO 3.5.2 Condition C or E) to OPERABLE status. This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

C.1 and C.2

If the Required Action and associated Completion Time of Conditions A or B are not met, the plant must be placed in a MODE or condition in which the LCO does not apply. This is done by placing the plant in MODE 3 within 6 hours and with pressurizer pressure to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures each accumulator isolation valve is fully open, as indicated in the control room, and timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a partially closed valve could result in not meeting DBA analyses assumptions (Ref. 3). A 12 hour Frequency is considered reasonable in view of the other administrative controls which ensure that a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and 3.5.1.3

Verification every 12 hours of the borated water volume and nitrogen cover pressure in each accumulator is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Considering that control room alarms are provided for both parameters these limits are effectively subject to

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SURVEILLANCE REQUIREMENTS (continued)

continuous monitoring. The 12 hour Frequency is considered reasonable considering the availability of the control room alarms and the likelihood that, with any deviation which may occur, the accumulators will perform their safety function with slight deviations in these parameters.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days, since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as in-leakage. Sampling the affected accumulator within 6 hours after a 51 cu. ft. (i.e., 3% of nominal required borated water volume of 1700 cu. ft.) volume increase will promptly identify whether the volume change has caused a reduction of boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the in-containment refueling water storage tank (IRWST), because the water contained in the IRWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 6).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, reduced accumulator capacity might be available for injection following a DBA that required operation of the accumulators. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startup or shutdowns.

Should closure of a valve occur, the safeguard actuation signal provided to the valve would open a closed valve, if required.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.6

This SR requires performance of a system performance test of each accumulator to verify flow capabilities. The system performance test demonstrates that the accumulator injection line resistance assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. IEEE Standard 603-1991, “Criteria for Safety Systems for Nuclear Power Generating Stations.”
2. FSAR Section 6.3 “Passive Core Cooling System.”
3. FSAR Section 15.6 “Decrease in Reactor Coolant Inventory.”
4. AP1000 PRA.
5. 10 CFR 50.46.
6. NUREG-1366, December 1992, “Improvements to Technical Specifications Surveillance Requirements.”

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.2 Core Makeup Tanks (CMTs) – Operating

BASES

BACKGROUND

Two redundant CMTs provide sufficient borated water to assure Reactor Coolant System (RCS) reactivity and inventory control for all design basis accidents (DBAs), including both loss of coolant accident (LOCA) events and non-LOCA events (Ref. 1). Flow from the core makeup tank increases if the reactor coolant pumps (RCPs) have coasted down, therefore, the RCP breaker OPERABILITY is addressed in LCO 3.5.9, Reactor Coolant Pump Breakers.

The CMTs are cylindrical tanks with hemispherical upper and lower heads. They are made of carbon steel and clad on the internal surfaces with stainless steel. They are located in containment at an elevation slightly above the reactor coolant loops. Each tank is full of borated water at > 3400 ppm. During normal operation, the CMTs are maintained at RCS pressure through a normally open pressure balance line from the cold leg.

The outlet line from each CMT is connected to one of two direct vessel injection lines, which provides an injection path for the water supplied by the CMT. The outlet line from each CMT is isolated by parallel, normally closed, fail open valves. Upon receipt of a safeguards actuation signal, these four valves open to align the CMTs to the RCS. The same two PMS divisions are needed to actuate both CMT outlet valves for a given CMT (divisions B and D for PXS-PL-V014A and PXS-PL-V015A and divisions A and C for PXS-PL-V014B and PXS-PL-V015B). Redundant Local Coincidence Logics (LCLs) within each PMS division are required to ensure CMT injection will occur when necessary with the single failure of an LCL.

The CMTs will inject to the RCS as inventory is lost and steam or reactor coolant is supplied to the CMT to displace the water that is injected. Steam or reactor coolant is provided to the CMT through the cold leg balance line, depending upon the specific event that has occurred. The inlet line from the cold leg is sized for LOCA events, where the cold legs become voided and higher CMT injection flows are required.

The injection line from each CMT contains a flow tuning orifice that is used to provide a mechanism for the field adjustment of the injection line resistance. The orifice is used to establish the required flow rates for the associated plant conditions assumed in the CMT design. The CMT flow is based on providing injection for approximately 20 minutes after CMT actuation.

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BASES

BACKGROUND (continued)

The CMT size and injection capability are selected to provide adequate RCS boration and safety injection for the limiting DBA. One CMT is adequate for this function during a small break LOCA where one CMT completely spills via the pipe break (Ref. 2). The Probabilistic Risk Assessment (PRA) (Ref. 3) shows that none of the CMTs are required for small LOCAs, assuming that at least one accumulator is available.

APPLICABLE SAFETY ANALYSES

The CMTs are assumed to be OPERABLE to provide emergency boration and core makeup when the Chemical and Volume Control System (CVS) is unavailable or insufficient, and to mitigate the consequences of any DBA which requires the safety injection of borated water (Ref. 2).

Following a non-LOCA event such as a steam line break, the RCS experiences a decrease in temperature and pressure due to an increase in energy removal by the secondary system. The cooldown results in a reduction of the core SHUTDOWN MARGIN due to the negative moderator temperature coefficient, with a potential for return to power. The actuation of the CMTs following this event provides injection of borated water to mitigate the reactivity transient and ensure the core remains shut down.

In the case of a steam generator tube rupture (SGTR), CMT injection provides borated water to compensate for RCS LEAKAGE.

In the case of an RCS leak of 10 gallons per minute, the CMTs can delay depressurization for at least 10 hours, providing makeup to the RCS and remain able to provide the borated water to compensate for RCS shrinkage and to assure the RCS boration for a safe shutdown.

In the case of a LOCA, the CMTs provide a relatively large makeup flow rate for approximately 20 minutes, in conjunction with the accumulators to provide the initial core cooling.

In the case of a small break LOCA, the borated water volume of one CMT is adequate for RCS safety injection, where one CMT completely spills via the pipe break.

CMTs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO establishes the minimum conditions necessary to ensure that CMT parameters meet the initial conditions assumed in the safety analyses. OPERABILITY is not expected to be challenged due to small gas accumulations in the high point, and rapid gas accumulations are not

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BASES

LCO (continued)

expected during plant operation. However, a relatively small gas volume was incorporated into the design for alerting operators to provide sufficient time to initiate venting operations before the gas volume would be expected to increase to a sufficient volume that might potentially challenge the OPERABILITY of natural circulation flow. Therefore, noncondensible gas accumulation in the inlet line high point that causes the water level to drop below the sensor will require operator action to investigate the cause of the gas accumulation and to vent the associated high point(s).

Each CMT represents 100% of the total injected borated water assumed in LOCA analysis. If the injection line from a single CMT to the vessel breaks, no single active failure on the other CMT will prevent the injection of borated water into the vessel. Thus the assumptions of the LOCA analysis will be satisfied.

For non-LOCA analysis, two CMTs are assumed. Note that for non-LOCA analysis, the accident cannot disable a CMT.

APPLICABILITY

In MODES 1, 2, and 3, and MODE 4 when the RCS is not being cooled by the Normal Residual Heat Removal System (RNS) the CMTs are required to be OPERABLE to provide borated water for RCS inventory makeup and reactivity control following a design basis event and subsequent cooldown.

The CMT requirements in MODE 5 with the RCS pressure boundary intact are specified in LCO 3.5.3, “Core Makeup Tanks (CMTs) – Shutdown, RCS Intact.”

The CMTs are not required to be OPERABLE while in MODE 5 with the RCS pressure boundary open or in MODE 6 because the RCS is depressurized and borated water can be supplied from the In-containment Refueling Water Storage Tank (IRWST), if needed.

In the unlikely event of a total loss of AC power sources, coupled with an inoperable Passive Residual Heat Removal Heat Exchanger (PRHR HX) (beyond DBA), the CMTs may be used in a feed and bleed sequence to remove heat from the RCS.

ACTIONS

A.1

With one outlet isolation valve inoperable on one CMT, action must be taken to restore the valve. In this Condition, the CMT is capable of performing its safety function, provided a single failure of the remaining parallel isolation valve does not occur. A Completion Time of 72 hours is

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ACTIONS (continued)

acceptable for a two train emergency core cooling system (ECCS) which is capable of performing its safety function without a single failure.

B.1

If the water temperature or boron concentration of one CMT is not within limits, it must be returned to within limits within 72 hours. The deviations in these parameters are expected to be slight, considering the frequent surveillances and control room monitors. With the temperature above the limit, the full core cooling capability assumed in the safety analysis may not be available. With the boron concentration not within limits, the ability to maintain subcriticality following a DBA may be degraded. However, because only one of two CMTs is inoperable, and the deviations of these parameters are expected to be slight, it is probable that more than a required amount of boron and cooling capability will be available to meet the conditions assumed in the safety analysis.

Since the CMTs are redundant, safety class components, the 72 hour Completion Time is consistent with the times normally allowed for this type of component.

C.1

With two CMTs inoperable due to water temperature or boron concentration, at least one CMT must be restored to within limits in 8 hours. The deviations in these parameters are expected to be slight, considering the frequent surveillances and control room monitors. A Completion Time of 8 hours is considered reasonable since the CMTs are expected to be capable of performing their safety function with slight deviations in these parameters and the accumulators are required to be available for LOCA mitigation (i.e., concurrent entry into Condition B of LCO 3.5.1 has not occurred). The effectiveness of accumulator injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one accumulator without any CMT injection supports adequate core cooling. This analysis provides a high confidence that with the unavailability of two CMTs due to water temperature or boron concentration deviations, the core can be cooled following design bases accidents.

If LCO 3.5.1 Condition B is entered concurrent with this Condition, then a 1 hour Completion Time from discovery of LCO 3.5.2 Condition C entry concurrent with LCO 3.5.1 Condition B entry also applies. This 1 hour Completion Time requires very prompt actions to restore either the CMT or the accumulator (per LCO 3.5.1 Condition B) to OPERABLE status.

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ACTIONS (continued)

This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

D.1

Excessive amounts of noncondensible gases in a CMT inlet line may interfere with the natural circulation flow (hot water from the RCS through the balance line into the CMT and cold water from the CMT through the direct vessel injection line into the vessel) assumed in the safety analyses for some transients. For CMT injection following a LOCA (steam will enter the CMT through the balance line, displacing the CMT water), gases in the CMT inlet line are not detrimental to the CMT function. The presence of some noncondensible gases does not mean that the CMT natural circulation capability is immediately inoperable, but that gases are collecting and should be vented.

The level sensor location has been selected to permit additional gas accumulation prior to significantly affecting the natural circulation flow so that adequate time may be provided to permit containment entry for venting the gas. Anticipated noncondensible gas accumulation in this piping segment is expected to be relatively slow.

The venting of these gases requires containment entry to manually operate the vent valves. A Completion Time of 24 hours is permitted for venting noncondensible gases and is acceptable, since, for the transients, the natural circulation capability of one CMT is adequate to ensure mitigation assuming less conservative analysis assumptions regarding stuck rods and core characteristics.

E.1

With one CMT inoperable for reasons other than Condition A, B, or D, operation of the CMT may not be available. Action must be taken to restore the inoperable CMT to OPERABLE status within 8 hours. The remaining CMT is sufficient for DBAs except for a LOCA in the OPERABLE CMTs direct vessel injection (DVI) line. The 8 hour Completion Time is based on the required availability of injection from the accumulators (provided that concurrent entry into Condition B of LCO 3.5.1 has not occurred) to provide SI injection. The effectiveness of accumulator injection is demonstrated in analysis performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a small LOCA, the injection from one accumulator without any CMT supports adequate core cooling. This analysis provides a high confidence that with the unavailability of one CMT, the core can be cooled following design bases accidents.

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BASES

ACTIONS (continued)

If LCO 3.5.1 Condition B is entered concurrent with this Condition, then a 1 hour Completion Time from discovery of LCO 3.5.2 Condition C entry concurrent with LCO 3.5.1 Condition B entry also applies. This 1 hour Completion Time requires very prompt actions to restore either the CMT or the accumulator (per LCO 3.5.1 Condition B) to OPERABLE status. This Completion Time is considered reasonable because of the low probability of simultaneously entering these multiple PXS Conditions and the very small likelihood of a LOCA occurring at the same time.

F.1 and F.2

If the Required Action or associated Completion Time of Condition A, B, C, D, or E are not met or two CMTs are inoperable for reasons other than Condition C, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification every 24 hours that the temperature of the borated water in each CMT is within limits ensures that when a CMT is needed to inject water into the RCS, the injected water temperature will be within the limits assumed in the accident analysis. The 24 hour Frequency is adequate, based on the fact that no mechanism exists to rapidly change the temperature of a large tank of water such as a CMT. This parameter is normally monitored in the control room by indication and alarms. Also, there are provisions for monitoring the temperature of the inlet and outlet lines to detect in-leakage which may affect the CMT water temperature.

SR 3.5.2.2

Each CMT inlet isolation valve must be verified to be fully open each 12 hours. Frequent verification is considered to be important, since a CMT cannot perform its safety function, if the valve is closed. Control room instrumentation is normally available for this verification.

SR 3.5.2.3

Verification that excessive amounts of noncondensible gases have not caused the water level to drop below the sensor in the inlet line is required every 24 hours. The inlet line of each CMT has a vertical section

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SURVEILLANCE REQUIREMENTS (continued)

of pipe which serves as a high point collection point for noncondensible gases. Control room indication of the water level in the high point collection point is available to verify that noncondensible gases have collected to the extent that the water level is depressed below the allowable level. The thermal dispersion sensor locations on the vertical pipe sections have been selected to permit additional gas accumulation before injection flow is significantly affected so that adequate time may be provided to permit containment entry for venting the gas.

This level is well above the CMT level corresponding to that assumed in the safety analysis for CMT injected volume. Verifying the noncondensible gas volume is within the noncondensible gas limit confirms ≥ 2487 cu. ft. of borated water volume in the CMT.

The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication.

SR 3.5.2.4

The minimum boron concentration is 3400 ppm to meet the CMT safety analysis minimum reactivity control requirements. The maximum boron concentration allowed within each CMT at any time is 4500 ppm to prevent overboration.

Verification every 31 days that the boron concentration in each CMT is within the required limits ensures that the reactivity control from each CMT, assumed in the safety analysis, will be available as required. The 31 day Frequency is adequate to identify changes which could occur from mechanisms such as in-leakage considering the provisions for monitoring temperature of the inlet line and top of the CMT. Additionally, the top of the CMT has control room indication and an alarm on increased temperature, which would be indicative of in-leakage.

SR 3.5.2.5

Verification that the redundant outlet isolation valves are OPERABLE by stroking the valves open ensures that each CMT will function as designed when these valves are actuated. Prior to opening the outlet isolation valves, the inlet isolation valve should be closed temporarily. Closing the inlet isolation valve ensures that the CMT contents will not be diluted or heated by flow from the RCS. Upon completion of the test, the inlet isolation valves must be opened. The Surveillance Frequency references the inservice testing requirements.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.6

This SR verifies that CMT outlet isolation valve actuates to the correct position on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valve. The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

SR 3.5.2.7

This SR requires performance of a system performance test of each CMT to verify flow capabilities. The system performance test demonstrates that the CMT injection line resistance assumed in DBA analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. FSAR Section 6.3, “Passive Core Cooling System.”
2. FSAR Chapter 15, “Accident Analysis.”
3. AP1000 PRA.

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.3 Core Makeup Tanks (CMTs) – Shutdown, RCS Intact

BASES

BACKGROUND

A description of the CMTs is provided in the Bases for LCO 3.5.2, “Core Makeup Tanks – Operating.”

APPLICABLE SAFETY ANALYSES

When the plant is shutdown with the Reactor Coolant System (RCS) pressure boundary intact, the CMT and Passive Residual Heat Removal (PRHR) are the preferred methods for mitigation of postulated events such as loss of normal decay heat removal capability (either loss of Startup Feedwater or loss of normal residual heat removal system). The CMT and PRHR are preferred because the RCS pressure boundary can remain intact, thus preserving one of the barriers to fission product release. For these events, the PRHR provides the safety related heat removal path. And the CMT maintains RCS inventory control. These events can also be mitigated by In-containment Refueling Water Storage Tank (IRWST) injection; however, the RCS must be depressurized through actuation of the Automatic Depressurization System (ADS) in order to facilitate IRWST injection.

Since no loss of coolant accidents (LOCAs) are postulated within the applicability of the LCO, the possibility of a break in the direct vessel injection line is not considered. As a result, only one CMT is required to be available to provide core cooling in response to postulated events. The two parallel CMT outlet isolation valves ensure that injection from one CMT occurs in the event of a single active failure. Note that a volume of noncondensible gases in the CMT inlet line does not affect the mechanism for draindown injection capability or required core cooling, and therefore does not affect the CMT OPERABILITY in MODE 5 with the RCS not VENTED.

CMTs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO establishes the minimum conditions necessary to ensure that one CMT will be available for RCS inventory control in the event of the loss of normal decay heat removal capability. The two CMT outlet isolation valves must be OPERABLE to ensure that at least one valve will operate, assuming that the other valve is disabled by a single active failure.

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BASES

APPLICABILITY

In MODE 4 without steam generator heat removal and in MODE 5 with the RCS not VENTED, one CMT is required to provide borated water to the RCS in the event the nonsafety related chemical and volume control system makeup pumps are not available to provide RCS inventory control.

The RCS is considered VENTED when the ADS stage 1, 2, and 3 flow paths (or alternative flow path with equivalent area) required by LCO 3.4.13, "Automatic Depressurization System (ADS) - Shutdown, RCS Open," are open, such that when combined with required ADS stage 4 actuation, sufficient steam venting area is provided to lower RCS pressure sufficiently to allow IRWST injection.

The CMT requirements in MODES 1, 2, 3, and 4 are specified in LCO 3.5.2, “Core Makeup Tanks (CMTs) – Operating.”

The CMTs are not required to be OPERABLE while in MODE 5 with the RCS VENTED or in MODE 6 because the RCS is depressurized and borated water can be supplied from the IRWST, if needed.

ACTIONS

A.1

With one outlet isolation valve inoperable action must be taken to restore the valve. In this Condition the CMT is capable of performing its safety function, provided a single failure of the remaining parallel isolation valve does not occur. A Completion Time of 72 hours is consistent with times normally applied to an emergency core cooling system (ECCS) which is capable of performing its safety function without a single failure.

B.1

If the water temperature or boron concentration in the required CMT is not within limits, it must be returned to within limits within 72 hours. With the temperature above the limit the makeup capability assumed in the safety analysis may not be available. With the boron concentration not within limits, the ability to maintain subcriticality may be degraded.

Because the mechanisms for significantly altering these parameters in the CMT are limited, it is probable that more than the required amount of boron and cooling capacity will be available to meet the conditions assumed in the safety analysis. Therefore, the 72 hour Completion Time is acceptable.

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BASES

ACTIONS (continued)

C.1 and C.2

While in MODE 4, with the RCS being cooled by the RNS, excessive amounts of noncondensible gases in a required CMT inlet line might interfere with the natural circulation flow (hot water from the RCS through the direct vessel injection line into the vessel) assumed in safety analyses for certain transients. The presence of some noncondensible gases does not mean that the CMT natural circulation capability is immediately inoperable, but that gases are collecting and should be vented.

A Completion Time of 24 hours is required to either vent the required CMT inlet line or bring the plant to MODE 5, where SR 3.5.3.2 is not applicable. This Completion Time is permitted to allow containment entry and manual operation of vent valves. It is reasonable because the air void does not impact the ability of the CMT to maintain RCS inventory.

D.1

With the required CMT inoperable for reasons other than Condition A, B, or C operation of the CMT may not be available. Action must be taken to restore the inoperable CMT to OPERABLE status within 8 hours. The 8 hour Completion Time is based on the availability of injection from the IRWST to provide RCS makeup. The ability of the IRWST to provide RCS injection is demonstrated by analysis performed to show that IRWST injection together with ADS venting provides adequate core cooling. Such analysis was performed for the loss of RNS cooling during midloop operations. The analysis was performed in support of the AP1000 PRA (Ref. 2).

E.1 and E.2

If the Required Action or associated Completion Time of Conditions A, B, or D are not met action must be initiated, immediately, to place the plant in MODE 5 which minimizes the likelihood and consequences of an event. In addition, action must be immediately initiated to restore the required CMT to OPERABLE status and to continue this action until restoration is accomplished in order to provide the required CMT for the mitigation of postulated events.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The LCO 3.5.2 Surveillance Requirements (SRs 3.5.2.1 through 3.5.2.7, except SR 3.5.2.3) are applicable to the CMT required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.5.2 for a discussion of each SR.

SR 3.5.3.2

The LCO 3.5.2 SR 3.5.2.3 is applicable to the CMT required to be OPERABLE in MODE 4, with RCS cooling provided by the RNS. It is not required to be met in MODE 5, with the RCS not VENTED, because a noncondensible gas volume in the CMT inlet line does not affect its draindown capability. This level is well above the CMT level corresponding to that assumed in the analysis for loss of RNS CMT volume. Verifying the noncondensible gas volume is within the noncondensible gas limit also confirms the required borated water volume in the CMT. The Frequency associated with the specified SR is applicable. Refer to the corresponding Bases for LCO 3.5.2 for a discussion of this SR.

SR 3.5.3.3

Verification every 7 days that the volume of the borated water is ≥ 2450 cu. ft. for the CMT required to be OPERABLE in MODE 5 with the RCS not VENTED verifies that when the OPERABLE CMT is needed to inject water into the RCS, the injected water volume will be within the limits assumed in the analyzed event. This parameter is normally monitored in the control room by indication and alarm. The 7 day Frequency is adequate, based on the fact that no mechanism exists which will drain water below the top nozzle of the CMT.

REFERENCES

1. FSAR Section 6.3, “Passive Core Cooling System.”
2. AP1000 PRA.

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.4 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating

BASES

BACKGROUND

The normal heat removal mechanism is the steam generators, which are supplied by the startup feedwater system. However, this path utilizes non-safety related components and systems, so its failure must be considered. In the event the steam generators are not available to remove decay heat for any reason, including loss of startup feedwater, the heat removal path is the PRHR HX (Ref. 1).

The principal component of the PRHR HX is a 100% capacity heat exchanger mounted in the In-containment Refueling Water Storage Tank (IRWST). The heat exchanger is connected to the Reactor Coolant System (RCS) by an inlet line from one RCS hot leg, and an outlet line to the associated steam generator cold leg channel head. The inlet line to the passive heat exchanger contains a normally open, motor operated isolation valve. The outlet line is isolated by two parallel, normally closed air operated valves, which fail open on loss of air pressure or PMS control signal. There is a vertical collection point at the top of the common inlet piping high point which serves as a gas collector. It is provided with level detectors that indicate when noncondensible gases have collected in this area. There are provisions to manually vent these gases to the IRWST.

In order to preserve the IRWST water for long-term PRHR HX operation, downspouts and a gutter are provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation, any water collected by the downspouts or gutter is directed to the normal containment sump. During PRHR HX operation, redundant series air operated valves are actuated to block the draining of condensate to the normal sump and to force the condensate into the IRWST. These valves fail closed on loss of air pressure or control signal.

The PRHR HX size and heat removal capability is selected to provide adequate core cooling for the limiting non-LOCA heatup Design Basis Accidents (DBAs) (Ref. 2). The Probability Risk Assessment (PRA) (Ref. 3) shows that PRHR HX is not required assuming that passive feed and bleed is available. Passive feed and bleed uses the Automatic Depressurization System (ADS) for bleed and the CMTs/accumulators/ IRWST for feed.

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BASES

APPLICABLE SAFETY ANALYSES

In the event of a non-LOCA DBA during normal operation, the PRHR HX is automatically actuated to provide decay heat removal path in the event the normal path through the steam generators is not available (Ref. 2).

The non-LOCA events which establish the PRHR HX parameters are those involving a decrease in heat removal by the secondary system, such as loss of main feedwater or other failure in the feedwater system. Since the PRHR HX is passive, it will mitigate the consequences of these events with a complete loss of all AC power sources. The PRHR HX actuates when the CMTs are actuated during LOCA events.

The PRHR HX satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the PRHR HX be OPERABLE so that it can respond appropriately to the DBAs which may require its operation. Since this is a passive component, it does not require the actuation of active components such as pumps for its OPERABILITY and will be OPERABLE if the inlet valves are in their normally open position, and the normally closed, fail open outlet valves open on receipt of an actuation signal.

In addition to the appropriate valve configuration, OPERABILITY may be impaired by noncondensible gases collecting in the system. OPERABILITY is not expected to be challenged due to small gas accumulations in the high point, and rapid gas accumulations are not expected during plant operation. However, a relatively small gas volume was incorporated into the design for alerting operators to provide sufficient time to initiate venting operations before the gas volume would be expected to increase to a sufficient volume that might potentially challenge the OPERABILITY of natural circulation flow. Therefore, noncondensible gas accumulation in the inlet line high point that causes the water level to drop below the sensor will require operator action to investigate the cause of the gas accumulation and to vent the associated high point(s).

A reactor coolant pump (RCP) is required to be operating in the loop with the PRHR HX, Loop 1, if any RCPs are operating. If RCPs are only operating in Loop 2 and no RCPs are operating in Loop 1, there is a possibility there may be reverse flow in the PRHR HX.

APPLICABILITY

The PRHR HX must be OPERABLE in MODES 1, 2, and 3, and in MODE 4 with the RCS not cooled by the Normal Residual Heat Removal System (RNS). Under these conditions, the PRHR HX may be actuated to provide core cooling and to mitigate the consequences of a DBA.

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BASES

APPLICABILITY (continued)

The PRHR HX requirements in MODE 4 with RCS cooling provided by the RNS and in MODE 5 with the RCS pressure boundary intact are specified in LCO 3.5.5, “Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, RCS Intact.”

The PRHR HX is not capable of natural circulation cooling of the RCS in MODE 5 with the RCS pressure boundary open or in MODE 6.

ACTIONS

A.1

The outlet line from the PRHR HX is controlled by a pair of normally closed, fail open, air operated valves, arranged in parallel. Thus they are redundant and, if either valve is OPERABLE, the system can function at 100% capacity, assuming other OPERABILITY conditions are met.

If one valve is inoperable, a Completion Time of 72 hours has been allowed to restore the inoperable valve(s) to OPERABLE status. This Completion Time is consistent with the Completion Times specified for other parallel redundant safety related systems.

B.1

With one air operated IRWST gutter isolation valve inoperable, the remaining isolation valve can function to drain the gutter to the IRWST. Action must be taken to restore the inoperable gutter isolation valve to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable based on the capability of the remaining valve to perform 100% of the required safety function assumed in the safety analyses.

C.1

Excessive amounts of noncondensible gases in the PRHR HX inlet line may interfere with the natural circulation flow of reactor coolant through the PRHR HX. The presence of some noncondensible gases does not mean that the PRHR HX is immediately inoperable, but that gases are collecting and should be vented.

The level sensor location has been selected to permit additional gas accumulation before natural circulation flow is significantly affected so that sufficient time may be provided to permit containment entry for venting the gas. Anticipated noncondensible gas accumulation in this piping segment is expected to be relatively slow.

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BASES

ACTIONS (continued)

The venting of these gases requires containment entry to manually operate the appropriate vent valves. A Completion Time of 24 hours is acceptable considering that passive feed and bleed cooling is available to remove heat from the RCS.

D.1 and D.2

If any of the above Required Actions have not been accomplished in the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4, with the RCS cooled by the RNS, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

With the LCO not met for reasons other than Condition A, B, or C, the PRHR HX must be restored within 8 hours. The 8 hour Completion Time is based on the availability of passive feed and bleed cooling to provide RCS heat removal. The effectiveness of feed and bleed cooling has been demonstrated in analysis and evaluations performed to justify PRA success criteria (Ref. 3). The analysis contained in this reference shows that for a range of events including loss of main feedwater, SGTR, and small LOCA (as small as 1/2") that feed and bleed cooling provides adequate core cooling.

These analyses and evaluations provide a high confidence that with the unavailability of the PRHR HX the core can be cooled following design bases accidents.

F.1 and F.2

If the PRHR HX is not restored in accordance with Action E.1 within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 5 within 36 hours.

Action F.1 is modified by a Note which requires that prior to initiating cooldown of the plant to MODE 3, redundant means of providing SG feedwater be verified as available. Possible means include two main feedwater pumps (with two condensate pumps), or two startup feedwater pumps, or a combination of one main feedwater pump (with one condensate pump) and one startup feedwater pump. With the PRHR HX

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BASES

ACTIONS (continued)

inoperable and redundant means of feeding the SGs not available, the unit is in a seriously degraded condition with no means for conducting a controlled cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. If redundant means of feeding the SGs are not available, the plant should be maintained in the current MODE until redundant means are restored. LCO 3.0.3 and all other Required Actions shall be suspended until the redundant means are restored, because they could force the unit into a less safe condition.

The Completion Time for Required Action F.1 is intended to allow the operator time to evaluate availability of redundant means. In this Required Action, the Completion Time only begins on discovery that redundant means of feeding the SGs is available.

Action F.2 is modified by a Note which requires that prior to stopping SG feedwater, redundant means of cooling the RCS to cold shutdown conditions must be verified as available. One redundant means of cooling the RCS to cold shutdown includes both subsystems of the normal residual heat removal system (RNS). Without availability of these redundant cooling means, the unit is in a seriously degraded condition with no means for continuing the controlled cooldown. Until the redundant cooling means are restored, heat removal using SG feedwater should be maintained. LCO 3.0.3 and all other Required Actions shall be suspended until the systems and equipment required for further cooldown are restored, because they could force the unit into a less safe condition.

The Completion Time for Required Action F.2 is intended to allow the operator time to evaluate availability of redundant means. In this Required Action, the Completion Time only begins on discovery that redundant means of cooling the RCS to cold shutdown conditions is available

Support equipment should be evaluated when assessing whether redundant means are available.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification, using remote indication, that the common PRHR HX outlet manual isolation valve is fully open ensures that the flow path from the heat exchangers to the RCS is available. Misalignment of this valve could render the heat exchanger inoperable. A 12 hour Frequency is reasonable considering that the valve is manually positioned and has control room position indication and alarm.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

Verification that the motor operated PRHR HX inlet valve is fully open, as indicated in the main control room, ensures timely discovery if the valve is not fully open. The 12 hour Frequency is consistent with the ease of verification, confirmatory open signals, and redundant series valve controls that prevent spurious closure.

SR 3.5.4.3

Verification that excessive amounts of noncondensible gases have not caused the water level to drop below the sensor in the inlet line is required every 24 hours. The inlet line of the PRHR HX has a vertical section of pipe which serves as a high point collection point for noncondensible gases. The thermal dispersion sensor location on the vertical pipe section has have been selected to permit additional gas accumulation before natural circulation flow is significantly affected so that sufficient time may be provided to permit containment entry for venting the gas.

Control room indication of the water level in this high point collection point is available to verify that noncondensible gases have not collected to the extent that the water level is depressed below the allowable level. The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication.

SR 3.5.4.4

Verification is required to confirm that one Loop 1 RCP is in operation, if any RCPs are operating. If RCPs are only operating in Loop 2 and no RCPs are operating in Loop 1, there is a possibility there may be reverse flow in the PRHR HX. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the main control room to monitor RCS loop performance.

The SR is modified by a Note which only requires that the Surveillance be met if one or more RCPs are in operation. If no RCPs are in operation, there is no need to verify one Loop 1 RCP is in operation.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.5

Verification is required to confirm that power is removed from the motor operated PRHR HX inlet isolation valve every 31 days. Removal of power from this valve reduces the likelihood that the valve will be inadvertently closed as a result of a fire. The 31 day Frequency is acceptable considering the frequent surveillance of valve position and that the valve has a confirmatory open signal.

SR 3.5.4.6

Verification that both air operated PRHR HX outlet valves stroke open and both IRWST gutter isolation valves stroke closed ensures that the PRHR HX will actuate on command, with return flow from the gutter to the IRWST. Since these valves are redundant, if one valve is inoperable, the system can function at 100% capacity. Verification requires the actual operation of each valve to move it to its safe position. The Surveillance Frequency is provided in the Inservice Testing Program.

SR 3.5.4.7

This surveillance requires visual inspection of the IRWST gutter and downspout screens to verify that the return flow to the IRWST will not be restricted by debris. A Frequency of 24 months is adequate since there are no known sources of debris with which the gutter or downspout screens could become restricted.

SR 3.5.4.8

This SR verifies that both PRHR HX air operated outlet isolation valves and both IRWST gutter isolation valves actuate to the correct position on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.9

This SR requires performance of a system performance test of the PRHR HX to verify system heat transfer capabilities. The system performance test demonstrates that the PRHR HX heat transfer assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

REFERENCES

1. FSAR Section 6.3, “Passive Core Cooling System.”
2. FSAR Chapter 15, “Safety Analysis.”
3. AP1000 PRA.

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.5 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, RCS Intact

BASES

BACKGROUND

A description of the PRHR HX is provided in the Bases for LCO 3.5.4, "Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating."

APPLICABLE SAFETY ANALYSES

In the event of a loss of normal decay heat removal capability during shutdown with the Reactor Coolant System (RCS) pressure boundary intact, the PRHR HX provides the preferred safety related heat removal path. When required, the PRHR HX is manually actuated and can maintain the RCS < 420°F. Alternatively, the heat removal function can be provided by depressurizing the RCS with the Automatic Depressurization System (ADS) and injection of the In-containment Refueling Water Storage Tank (IRWST) with containment closure capability provided. The PRHR HX is preferred because the RCS pressure boundary remains intact, thus preserving a barrier to fission product release.

The PRHR HX satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires the PRHR HX to be OPERABLE so that it can be placed in service in the event normal decay heat removal capability is lost. Since this a passive component, it does not require the actuation of active components such as pumps for its OPERABILITY and will be OPERABLE if the inlet valves are in their normally open position, and the normally closed, fail open outlet valves open on receipt of an actuation signal.

OPERABILITY is not expected to be challenged due to small gas accumulations in the high point, and rapid gas accumulations are not expected during plant operation. However, a relatively small gas volume was incorporated into the design for alerting operators to provide sufficient time to initiate venting operations before the gas volume would be expected to increase to a sufficient volume that might potentially challenge the OPERABILITY of passive safety injection flow. Therefore, noncondensible gas accumulation in the injection line high point that causes the water level to drop below the sensor will require operator action to investigate the cause of the gas accumulation and to vent the associated high point(s).

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BASES

LCO (continued)

A reactor coolant pump (RCP) is required to be operating in the loop with the PRHR HX, Loop 1, if any RCPs are operating. If RCPs are only operating in loop 2 and no RCPs are operating in loop 1, there is a possibility there may be reverse flow in the PRHR HX.

APPLICABILITY

The PRHR HX must be OPERABLE in MODE 4 with RCS cooling provided by the Normal Residual Heat Removal System (RNS) and in MODE 5 with the RCS pressure boundary intact and pressurizer level ≥ 20% to provide decay heat removal in the event the normal residual heat removal system is not available.

The Applicability is modified by a Note that states that the PRHR is not required to be OPERABLE in MODE 5 during RCS vacuum fill operations. In this scenario, the heat removal function is provided by the vent path through the ADS valves and the injection flow path from the IRWST with containment closure capability provided.

The PRHR HX requirements in MODES 1, 2, and 3, and in MODE 4 with RCS cooling not provided by the RNS are specified in LCO 3.5.4, “Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating.”

The PRHR HX is not credited for natural circulation cooling of the RCS in MODE 5 with the RCS pressure boundary open or when pressurizer level < 20%, or in MODE 6. When in MODE 5 with either the RCS pressure boundary open, or RCS intact with pressurizer level < 20%, or in MODE 6 with the upper internals in place, OPERABILITY of the PRHR HX is not required because decay heat removal by IRWST injection and ADS steam venting is credited. Decay heat removal by RCS heat-up and refueling cavity steam venting is credited in MODE 6 with upper internals removed.

ACTIONS

A.1

The outlet line from the PRHR HX is isolated by a pair of normally closed, fail open, air operated valves, arranged in parallel. They are redundant, and if either valve is OPERABLE the system can function at 100% capacity, assuming other OPERABILITY conditions are met.

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BASES

ACTIONS (continued)

Since these valves are redundant, if one valve is inoperable, a Completion Time of 72 hours has been allowed to restore the inoperable valve to OPERABLE status. This Completion Time is consistent with the Completion Times specified for other parallel redundant safety related systems.

B.1

With one air operated IRWST gutter isolation valve inoperable, the remaining isolation valve can function to drain the gutter to the IRWST. Action must be taken to restore the inoperable gutter isolation valve to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable based on the capability of the remaining valve to perform 100% of the required safety function assumed in the safety analyses.

C.1

At the inlet piping high point there is a vertical chamber which serves as a collection point for noncondensible gases. This collection point is provided with detectors which alarm to indicate when gases have collected in this area. The presence of an alarm does not mean that PRHR HX is immediately inoperable, but that gases are collecting and should be vented. A Completion Time of 24 hours is acceptable, considering that passive feed and bleed cooling is available to remove heat from the RCS.

D.1

With the LCO not met for reasons other than Condition A, B, or C, the PRHR HX must be restored within 8 hours. The 8 hour Completion Time is acceptable based on the availability of passive feed and bleed cooling to provide RCS heat removal. The effectiveness of feed and bleed cooling is discussed in the bases for LCO 3.5.4, Action E.1.

E.1 and E.2

If any of the above Required Actions have not been accomplished in the required Completion Time, action must be initiated, immediately, to be in MODE 5 which minimizes the likelihood and consequences of an event. In addition, action must be immediately initiated to restore the required PRHR HX to OPERABLE status and to continue this action until restoration is accomplished in order to provide the PRHR HX for the mitigation of postulated events.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

The LCO 3.5.4 Surveillance Requirements are applicable to the PRHR HX required to be OPERABLE. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.5.4 for a discussion of each SR.

REFERENCES None.

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.6 In-containment Refueling Water Storage Tank (IRWST) – Operating

BASES

BACKGROUND

The IRWST is a large stainless steel lined tank filled with borated water (Ref. 1). It is located below the operating deck in containment. The tank is designed to meet seismic Category 1 requirements. The floor of the IRWST is elevated above the reactor coolant loop so that borated water can drain by gravity into the Reactor Coolant System (RCS). The IRWST is maintained at ambient containment pressure.

The IRWST has two injection flow paths. The injection paths are connected to the reactor vessel through two direct vessel injection lines which are also used by the accumulators and the core makeup tanks. Each path includes an injection flow path and a containment recirculation flow path. Each injection path includes a normally open motor operated isolation valve and two parallel actuation lines each isolated by one check valve and one squib valve in series.

ADS and IRWST injection blocking devices are provided to minimize the likelihood of spurious IRWST injection. The ADS and IRWST injection blocking device in each PMS division provides an output to the IRWST injection valve component interface module (CIM) to prevent the IRWST injection actuation unless other plant parameters indicate that an actuation is necessary. A low level in either one or both CMTs will unblock the IRWST injection actuation in each division.

The IRWST has two containment recirculation flow paths. Each containment recirculation path contains two parallel actuation flow paths, one path is isolated by a normally open motor operated valve in series with a squib valve and one path is isolated by a check valve in series with a squib valve.

During refueling operations, the IRWST is used to flood the refueling cavity. During abnormal events, the IRWST serves as a heat sink for the passive residual heat removal heat exchangers, as a heat sink for the depressurization spargers, and as a source of low head (ambient containment pressure) safety injection during loss of coolant accidents (LOCAs) and loss of decay heat removal in MODE 5 (loops not filled). The IRWST can be cooled by the Normal Residual Heat Removal System (RNS) system.

The IRWST size and injection capability is selected to provide adequate core cooling for the limiting Design Basis Accidents (DBAs) (Ref. 2).

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BASES

APPLICABLE SAFETY ANALYSES

During non-LOCA events, the IRWST serves as the initial heat sink for the Passive Residual Heat Removal (PRHR) Heat Exchanger (PRHR HX) if used during reactor cooldown to MODE 4. If RNS is available, it will be actuated in MODE 4 and used to continue the plant cooldown to MODE 5. If RNS is not available, cooldown can continue on PRHR. Continued PRHR HX operation will result in the water in the IRWST heating up to saturation conditions and boiling. The steam generated in the IRWST enters the containment through the IRWST vents. Most of the steam generated in the IRWST condenses on the inside of the containment vessel and drains back to the IRWST.

For events which involve a loss of primary coolant inventory, such as a large break LOCA, or other events involving automatic depressurization, the IRWST provides low pressure safety injection (Ref. 2). The IRWST drain down time is dependent on several factors, including break size, location, and the return of steam condensate from the passive containment cooling system. During drain down, when the water in the IRWST reaches the Low 3 level, the containment sump will be sufficiently flooded, to initiate containment sump recirculation. This permits continued cooling of the core by recirculation of the spilled water in the containment sumps via the sump recirculation flow paths. In this situation, core cooling can continue indefinitely.

When the plant is in midloop operation, the pressurizer Automatic Depressurization System (ADS) valves are open, and the RNS is used to cool the RCS. The RNS is not a safety related system, so its failure must be considered. In this situation, with the RCS drained and the pressure boundary open, the PRHR HX cannot be used. In such a case, core cooling is provided by gravity injection from the IRWST, venting the RCS through the ADS. Injection from the IRWST provides core cooling until the tank empties and the containment is flooded up to a level sufficient to provide recirculation flow through the gravity injection lines back into the RCS. With the containment closed, the recirculation can continue indefinitely, with the decay heat generated steam condensing on the containment vessel and draining back into the IRWST.

The IRWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The IRWST requirements ensure that an adequate supply of borated water is available to act as a heat sink for PRHR and to supply the required volume of borated water as safety injection for core cooling and reactivity control.

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BASES

LCO (continued)

To be considered OPERABLE, the IRWST must meet the water volume, boron concentration, and temperature limits defined in the surveillance requirements. The motor operated injection isolation valves must be open with power removed, and the motor operated sump recirculation isolation valves must be open. OPERABILITY is not expected to be challenged due to small gas accumulations in the high point, and rapid gas accumulations are not expected during plant operation. However, a relatively small gas volume was incorporated into the design for alerting operators to provide sufficient time to initiate venting operations before the gas volume would be expected to increase to a sufficient volume that might potentially challenge the OPERABILITY of passive safety injection flow. Therefore, noncondensible gas accumulation in the injection line high point that causes the water level to drop below the sensor will require operator action to investigate the cause of the gas accumulation and to vent the associated high point(s).

APPLICABILITY

In MODES 1, 2, 3, and 4, a safety related function of the IRWST is to provide a heat sink for PRHR. In MODES 1, 2, 3, and 4, a second safety related function is the low pressure safety injection of borated water following a LOCA for core cooling and reactivity control. Both of these functions must be available to meet the initial assumptions of the safety analyses. These assumptions require the specified boron concentration, the minimum water volume, and the maximum water temperature.

The requirements for the IRWST in MODES 5 and 6 are specified in LCO 3.5.7, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5 and LCO 3.5.8, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6.

ACTIONS

A.1

If an IRWST injection line actuation valve flow path or a containment recirculation line actuation valve flow path is inoperable, then the valve actuation flow path must be restored to OPERABLE status within 72 hours. In this condition, three other IRWST injection or containment sump recirculation flow paths are available and can provide 100% of the required flow assuming a break in the direct vessel injection line associated with the other injection train, but with no single failure of the actuation valve flow path in the same injection or sump recirculation flow path. The 72 hour Completion Time is consistent with times normally applied to a degraded two train emergency core cooling system (ECCS) which can provide 100% of the required flow without a single failure.

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ACTIONS (continued)

B.1

Excessive amounts of noncondensible gases in one of the injection flow path squib valve outlet line pipe stubs in one IRWST injection line may interfere with the passive injection of IRWST water into the reactor vessel from the associated parallel flow path in the affected injection line. Analyses have shown that with enough noncondensible gas accumulation, IRWST injection through the affected flow path could be delayed. However, the presence of some noncondensible gases does not mean that the IRWST injection capability is immediately inoperable, but that gases are collecting and should be vented. The venting of these gases requires containment entry to manually operate the vent valves. In this Condition, the parallel flow path in the affected injection line is capable of providing 100% of the required injection flow and the other IRWST injection line remains fully OPERABLE. These IRWST flow paths can provide the credited flow in the event of a direct vessel injection (DVI) line break downstream of the fully OPERABLE injection line, provided a single failure of the remaining parallel isolation valve does not occur. A Completion Time of 72 hours is acceptable for two train ECCS systems, which are capable of performing their safety function without a single failure.

C.1

Excessive amounts of noncondensible gases in both of the injection flow path squib valve outlet line pipe stubs in one IRWST injection line may affect the passive injection of IRWST water into the reactor vessel from the affected injection line. Sufficient gas accumulation could potentially challenge IRWST injection capability. However, the presence of some noncondensible gases does not immediately render the IRWST injection capability inoperable, but that gases are collecting and should be vented.

The level sensor location has been selected to permit additional gas accumulation before injection flow is significantly affected so that adequate time may be provided to permit containment entry for venting the gas. Anticipated noncondensible gas accumulation in this piping segment is expected to be relatively slow.

In this Condition, the remaining OPERABLE IRWST injection line is capable of performing the safety function for all plant events except for one, DVI line break. For this one event, the line with gas accumulation in both injection flow path squib valve outlet line pipe stubs will be capable of performing the safety function with a small amount of voiding that is not expected to significantly challenge the required injection flow.

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BASES

ACTIONS (continued)

The venting of these gases requires containment entry to manually operate the vent valves. Considering the relatively slow rate of gas accumulation, venting within 8 hours should normally prevent accumulation of amounts of noncondensible gases that could significantly challenge IRWST injection capability. A Completion Time of 8 hours is permitted for venting noncondensible gases and is acceptable since the injection capability of the other IRWST injection line is sufficient to ensure event mitigation, or in the event of a break in the DVI line connected to the OPERABLE injection line, the injection line with gas accumulation will be capable of providing the required injection flow with some voiding. If only one of the affected injection flow path squib valve outlet line pipe stubs is vented, then Condition B will apply to the remaining injection flow path squib valve outlet line pipe stub with noncondensible gas accumulation.

D.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The allowable deviation of the water volume is limited to 3%. This limit prevents a significant change in boron concentration and is consistent with the long-term cooling analysis performed to justify probabilistic risk assessment (PRA) success criteria (Ref. 3), which assumed multiple failures with as many as 3 CMTs/Accum not injecting. This analysis shows that there is significant margin with respect to the water supplies that support containment recirculation operation. The 8-hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to DBAs with slight deviations in these parameters.

E.1

If the motor operated IRWST isolation valves are not fully open or valve power is not removed, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valves must be restored to fully open with valve power removed in 1 hour. This Completion Time is acceptable based on risk considerations.

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BASES

ACTIONS (continued)

F.1 and F.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times of Condition A, B, C, D, or E, or the LCO is not met for reasons other than Conditions A, B, C, D, or E, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.6.1

The IRWST borated water temperature must be verified every 24 hours to ensure that the temperature is within the limit assumed in the accident analysis. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

SR 3.5.6.2

Verification every 24 hours that the IRWST borated water volume is above the required minimum level will ensure that a sufficient initial supply is available for safety injection and floodup volume for recirculation and as the heat sink for PRHR. During shutdown with the refueling cavity flooded with water from the IRWST, this Surveillance requires that the combined volume of borated water in the IRWST and refueling cavity meet the specified limit. Since the IRWST volume is normally stable, and is monitored by redundant main control indication and alarm, a 24 hour Frequency is appropriate.

SR 3.5.6.3

Verification that excessive amounts of noncondensible gases have not caused the water level to drop below the sensor in the four IRWST injection line squib valve lines is required every 24 hours. The 8x8x8 inch tee after the outlet of the IRWST injection line squib valve lines has a vertical section of pipe which serves as a high point collection point for noncondensible gases. The thermal dispersion sensor locations on the vertical pipe sections have been selected to permit additional gas accumulation prior to significantly affecting the injection flow so that adequate time may be provided to permit containment entry for venting the gas.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

Control room indication of the water level in this high point collection point is available to verify that noncondensible gases have not collected to the extent that the water level is depressed below the allowable level. The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication.

SR 3.5.6.4

Verification every 31 days that the boron concentration of the IRWST is greater than the required limit, ensures that the reactor will remain subcritical following a LOCA. Since the IRWST volume is large and normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each solution volume increase of ≥ 15,000 gal. In addition, the relatively frequent surveillance of the IRWST water volume provides assurance that the IRWST boron concentration is not changed.

SR 3.5.6.5

This surveillance requires verification that each motor operated injection and sump recirculation isolation valve is fully open. This surveillance may be performed with available remote position indication instrumentation. The 12 hour Frequency is acceptable, considering the redundant remote indication and alarms and that power is removed from the valve operator.

SR 3.5.6.6

Verification is required to confirm that power is removed from each motor operated IRWST isolation valve each 31 days. Removal of power from these valves reduces the likelihood that the valves will be inadvertently closed. The 31 day Frequency is acceptable considering frequent surveillance of valve position and that the valve has a confirmatory open signal.

SR 3.5.6.7

Each motor operated containment recirculation isolation valve must be verified to be fully open. This valve is required to be open to improve containment recirculation reliability. The 31 day Frequency is acceptable considering the valve has a confirmatory open signal. This surveillance may be performed with available remote position indication instrumentation.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.6.8

This Surveillance requires verification that each IRWST injection and each containment recirculation squib valve is OPERABLE in accordance with the Inservice Testing Program. The Surveillance Frequency for verifying valve OPERABILITY references the Inservice Testing Program.

The squib valves will be tested in accordance with the ASME OM Code (Ref. 4). The applicable ASME OM Code squib valve requirements are specified in paragraph ISTC-5260.

SR 3.5.6.9

This SR ensures that each IRWST injection and containment recirculation squib valve can actuate to the correct position on an actual or simulated actuation signal. The ADS and IRWST injection blocking device ACTUATION LOGIC TEST, provides overlap with this Surveillance. The OPERABILITY of the squib valves is checked by performing a continuity check of the circuit from the Protection Logic Cabinets to the squib valve. The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

SR 3.5.6.10

Visual inspection is required each 24 months to verify that the IRWST screens and the containment recirculation screens are not restricted by debris. A Frequency of 24 months is adequate, since there are no known sources of debris with which these screens could become restricted.

SR 3.5.6.11

This SR requires performance of a system inspection and performance test of the IRWST injection and recirculation flow paths to verify system flow capabilities. The system inspection and performance test demonstrates that the IRWST injection and recirculation capabilities assumed in accident analyses is maintained. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The System Level Operability Testing Program provides specific test requirements and acceptance criteria.

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BASES

REFERENCES

1. FSAR Section 6.3, “Passive Core Cooling.”
2. FSAR Section 15.6, “Decrease in Reactor Coolant Inventory.”
3. AP1000 PRA.
4. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.7 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5

BASES

BACKGROUND

A description of the IRWST is provided in LCO 3.5.6, “In-containment Refueling Water Storage Tank – Operating.”

APPLICABLE SAFETY ANALYSES

For postulated shutdown events in MODE 5 with the Reactor Coolant System (RCS) pressure boundary intact, the primary protection is Passive Residual Heat Removal (PRHR), where the IRWST serves as the initial heat sink for the PRHR heat exchanger (PRHR HX). For events in MODE 5 with the RCS pressure boundary open, PRHR is not available and RCS heat removal is provided by IRWST injection and containment sump recirculation.

IRWST injection could be required to mitigate some events by providing RCS inventory makeup.

No loss of coolant accidents (LOCAs) are postulated during plant operation in MODE 5; therefore, the rupture of the direct vessel injection line (DVI) is not assumed. Since the DVI rupture is not assumed, only one train of IRWST injection and recirculation flow paths is required to mitigation postulated events, assuming a single failure.

The IRWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The IRWST requirements ensure that an adequate supply of borated water is available to act as a heat sink for PRHR and to supply the required volume of borated water as safety injection for core cooling and reactivity control.

To be considered OPERABLE, the IRWST must meet the water volume, boron concentration, and temperature limits defined in the Surveillance Requirements, and one path of injection and recirculation must be OPERABLE (the motor operated injection isolation valve must be open with power removed, and the motor operated sump recirculation isolation valves must be open). OPERABILITY is not expected to be challenged due to small gas accumulations in the high point, and rapid gas accumulations are not expected during plant operation. However, a relatively small gas volume was incorporated into the design for alerting operators to provide sufficient time to initiate venting operations before

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BASES

LCO (continued)

the gas volume would be expected to increase to a sufficient volume that might potentially challenge the OPERABILITY of passive safety injection flow. Noncondensible gas accumulation in the injection line high point that causes the water level to drop below the sensor will require operator action to investigate the cause of the gas accumulation and to vent the associated high point(s).

Note that during vacuum fill operations, a vapor void may form in the high point vent lines, causing the water level to drop below the high point water level sensor. Noncondensible gas accumulation will not increase to a volume that could potentially challenge the OPERABILITY of the passive safety injection flow.

APPLICABILITY

In MODE 5 with the RCS pressure boundary intact or with the RCS open with pressurizer level ≥ 20%, the IRWST is an RCS injection source of borated water for core cooling and reactivity control. Additionally, in MODE 5 with the RCS pressure boundary intact, the IRWST provides the heat sink for PRHR.

The requirements for the IRWST in MODES 1, 2, 3, and 4 are specified in LCO 3.5.6, In-containment Refueling Water Storage Tank (IRWST) – Operating. The requirements for the IRWST in MODE 6 are specified in LCO 3.5.8, In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6.

ACTIONS

A.1

If a motor operated containment sump isolation valve in the required sump recirculation flow path is not fully open, the valve must be fully opened within 72 hours. The 72 hour Completion Time is consistent with times normally applied to a degraded two train emergency core cooling system (ECCS) which can provide 100% of the required flow without a single failure.

B.1

Excessive amounts of noncondensible gases in one of the injection flow path squib valve outlet line pipe stubs in the required IRWST injection line may interfere with the passive injection of IRWST water into the reactor vessel from the associated parallel flow path in the affected injection line. Analyses have shown that with enough noncondensible gas accumulation, IRWST injection through the affected flow path could be

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BASES

ACTIONS (continued)

delayed. However, the presence of some noncondensible gases does not mean that the IRWST injection capability is immediately inoperable, but that gases are collecting and should be vented. Venting of these gases requires containment entry to manually operate the vent valves. In this Condition the parallel flow path in the affected injection line is capable of providing 100% of the required injection. A direct vessel injection (DVI) line break is not postulated in MODE 5. A Completion Time of 72 hours is acceptable since the IRWST is capable of performing the safety function without a single failure of the remaining parallel isolation valve. In addition, the 72-hour Completion Time is consistent with the time normally applicable to one inoperable train in a two train ECCS system.

C.1

Excessive amounts of noncondensible gases in both of the injection flow path squib valve outlet line pipe stubs in the required IRWST injection line may interfere with the passive injection of IRWST water into the reactor vessel from the affected injection line. Analyses have shown that with enough noncondensible gas accumulation, IRWST injection could be delayed long enough to cause core uncover. However, the presence of some noncondensible gases does not mean that the IRWST injection capability is immediately inoperable, but that gases are collecting and should be vented. Venting of these gases requires containment entry to manually operate the vent valves. Considering the slow rate of gas accumulation, venting within 8 hours should normally prevent accumulation of amounts of noncondensible gases that could interfere with IRWST injection. A Completion Time of 8 hours is permitted for venting noncondensible gases and is acceptable since the injection capability is not significantly affected. If only one of the affected injection flow path squib valve outlet line pipe stub is vented, then Condition B will apply to the remaining high point vent with noncondensible gas accumulation.

D.1

If the IRWST water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR heat transfer and the reactivity benefit of injection assumed in the MODE 1 safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The allowable deviation of the water volume is limited to 3%. This limit

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BASES

ACTIONS (continued)

prevents a significant change in boron concentration. During shutdown conditions, the control rods are inserted, the decay heat is lower compared to operation in higher modes due to the time to get to shutdown, the RCS is borated for cold shutdown conditions, and the RCS temperatures are significantly reduced below normal operating conditions. Therefore, anticipated minor deviations in these IRWST parameters are acceptable, since they provide substantial core cooling and boration margins during shutdown events. The 8-hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to design basis accidents (DBAs) with slight deviations in these parameters.

E.1

If the required motor operated IRWST isolation valve is not fully open or valve power is not removed, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valve must be restored to fully open with valve power removed in 1 hour. This Completion Time is acceptable based on risk considerations.

F.1 and F.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times of Condition A, B, C, D, or E, or the LCO is not met for reasons other than Conditions A, B, C, D, or E, action must be immediately initiated to restore the IRWST to OPERABLE status and to continue this action until restoration is accomplished in order to provide the IRWST for the mitigation of postulated events. In addition, the plant must be placed in a condition in which the probability and consequences of an event are minimized. This is done by immediately suspending positive reactivity additions to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of control rods, and excessive cooling of the RCS.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.7.1

This SR ensures that the Spent Fuel Pool Cooling System (SFS) containment isolation valves are closed. With the nonsafety-related SFS isolated from the IRWST, a leak in the SFS does not cause draining of the IRWST.

The Surveillance Requirement is modified by a Note allowing the affected penetration flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration flow path(s) can be rapidly isolated when a need for isolation is indicated.

SR 3.5.7.2

The LCO 3.5.6 Surveillance Requirements and Frequencies (SR 3.5.6.1 through SR 3.5.6.11, except SR 3.5.6.3) are applicable to the IRWST and the flow paths required to be OPERABLE. Refer to the corresponding Bases for LCO 3.5.6 for a discussion of each SR.

The LCO 3.5.6 Surveillance Requirement 3.5.6.3 is not applicable to the IRWST and the flow paths required to be OPERABLE. During RCS vacuum fill operations, a vapor void may form in the high point vent line, causing the water level to drop below the high point water level sensor. Gas accumulation will not increase to a volume that could potentially challenge the OPERABILITY of the passive safety injection flow.

SR 3.5.7.3

The LCO 3.5.6 Surveillance Requirement 3.5.6.3 is applicable to the IRWST and the flow paths required to be OPERABLE. SR 3.5.7.2 is not required to be met during RCS vacuum fill operations, as a vapor void may form in the high point vent line, causing the water level to drop below the high point water level sensor. Gas accumulation will not increase to a volume that could potentially challenge the OPERABILITY of the passive safety injection flow.

REFERENCES None.

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.8 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6

BASES

BACKGROUND

A description of the IRWST is provided in LCO 3.5.6, “In-containment Refueling Water Storage Tank (IRWST) – Operating.”

APPLICABLE SAFETY ANALYSES

For MODE 6, heat removal is provided by IRWST injection and containment sump recirculation.

IRWST injection could be required to mitigate some events by providing RCS inventory makeup.

One line with redundant, parallel valves is required to accommodate a single failure (to open) of an isolation valve.

The IRWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The IRWST requirements ensure that an adequate supply of borated water is available to supply the required volume of borated water as safety injection for core cooling and reactivity control.

To be considered OPERABLE, the IRWST in combination with the refueling cavity must meet the water volume, boron concentration, and temperature limits defined in the Surveillance Requirements, and one path of injection and recirculation must be OPERABLE. The motor operated injection isolation valve must be open and power removed, and the motor operated sump recirculation isolation valves must be open and OPERABLE. Any cavity leakage should be estimated and made up with borated water such that the volume in the IRWST plus the refueling cavity will meet the IRWST volume requirement. OPERABILITY is not expected to be challenged due to small gas accumulations in the high point, and rapid gas accumulations are not expected during plant operation. However, a relatively small gas volume was incorporated into the design for alerting operators to provide sufficient time to initiate venting operations before the gas volume would be expected to increase to a sufficient volume that might potentially challenge the OPERABILITY of passive safety injection flow. Noncondensible gas accumulation in the injection line high point that causes the water level to drop below the sensor will require operator action to investigate the cause of the gas accumulation and to vent the associated high point(s).

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BASES

APPLICABILITY

In MODE 6, the IRWST is an Reactor Coolant System (RCS) injection source of borated water for core cooling and reactivity control.

ACTIONS

A.1

With the required motor operated containment sump isolation valve not fully open, the valve must be fully opened within 72 hours. The 72 hour Completion Time is consistent with times normally applied to a degraded two train emergency core cooling system (ECCS) which can provide 100% of the required flow without a single failure.

B.1

Excessive amounts of noncondensible gases in one of the high point vents in the required IRWST injection line may interfere with the passive injection of IRWST water into the reactor vessel from the associated parallel flow path in the affected injection line. Analyses have shown that with enough noncondensible gas accumulation, IRWST injection through the affected flow path could be delayed. However, the presence of some noncondensible gases does not mean that the IRWST injection capability is immediately inoperable, but that gases are collecting and should be vented. Venting of these gases requires containment entry to manually operate the vent valves. In this Condition, the parallel flow path in the affected injection line is capable of providing 100% of the required injection. A direct vessel injection (DVI) line break is not postulated in MODE 6. A Completion Time of 72 hours is acceptable since the IRWST is capable of performing the safety function without a single failure of the remaining parallel isolation valve. In addition, the 72-hour Completion Time is consistent with the time normally applicable to one inoperable train in a two train ECCS system.

C.1

Excessive amounts of noncondensible gases in both of the high point vents in the required IRWST injection line may interfere with the passive injection of IRWST water into the reactor vessel from the affected injection line. Analyses have shown that with enough noncondensible gas accumulation, IRWST injection could be delayed long enough to cause core uncover. However, the presence of some noncondensible gases does not mean that the IRWST injection capability is immediately inoperable, but that gases are collecting and should be vented. Venting of these gases requires containment entry to manually operate the vent valves. Considering the slow rate of gas accumulation, venting within 8 hours should normally prevent accumulation of amounts of noncondensible gases that could interfere with IRWST injection. A Completion Time of 8 hours is permitted for venting noncondensible

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ACTIONS (continued)

gases and is acceptable since the injection capability is not significantly affected. If only one of the affected high point vents is vented, then Condition B will apply to the remaining high point vent with noncondensible gas accumulation.

D.1

If the IRWST and refueling cavity water volume, boron concentration, or temperature are not within limits, the core cooling capability from injection or PRHR HX heat transfer and the reactivity benefit of injection assumed in the MODE 1 safety analyses may not be available. Due to the large volume of the IRWST, online monitoring of volume and temperature, and frequent surveillances, the deviation of these parameters is expected to be minor. The allowable deviation of the water volume is limited to 3%. This limit prevents a significant change in boron concentration. During shutdown conditions, the control rods are inserted, the decay heat is lower compared to operation in higher modes due to the time to get to shutdown, the RCS is borated for cold shutdown conditions, and the RCS temperatures are significantly reduced below normal operating conditions. Therefore, anticipated minor deviations in these IRWST parameters are acceptable, since they provide substantial core cooling and boration margins during shutdown events. The 8-hour Completion Time is acceptable, considering that the IRWST will be fully capable of performing its assumed safety function in response to design basis accidents (DBAs) with slight deviations in these parameters.

E.1

If the required motor operated IRWST isolation valve is not fully open or valve power is not removed, injection flow from the IRWST may be less than assumed in the safety analysis. In this situation, the valve must be restored to fully open with valve power removed in 1 hour. This Completion Time is acceptable based on risk considerations.

F.1 and F.2

If the IRWST cannot be returned to OPERABLE status within the associated Completion Times or the LCO is not met for reasons other than Conditions A, B, C, D, or E, action must be immediately initiated to restore the IRWST to OPERABLE status and to continue this action until restoration is accomplished in order to provide the IRWST for the mitigation of postulated events. In addition, the plant must be placed in a condition in which the probability and consequences of an event are minimized.

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BASES

ACTIONS (continued)

This is done by immediately suspending positive reactivity additions to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of control rods, and excessive cooling of the RCS.

SURVEILLANCE REQUIREMENTS

SR 3.5.8.1

The IRWST and refueling cavity borated water temperature must be verified every 24 hours to ensure that the temperature is within the limit assumed in accident analysis. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

SR 3.5.8.2

Verification every 24 hours that the IRWST and refueling cavity borated water volume is above the required minimum level will ensure that a sufficient intitial supply is available for safety injection and floodup volume for recirculation and as the heat sink for PRHR. During shutdown with the refueling cavity flooded with water from the IRWST, the Surveillance requires that the combined volume of borated water in the IRWST and refueling cavity meets the specified limit. Since the IRWST volume is normally stable, and is monitored by redundant main control indication and alarm, a 24 hour Frequency is appropriate.

SR 3.5.8.3

This SR ensures that the Spent Fuel Pool Cooling System (SFS) containment isolation valves are closed. With the nonsafety-related SFS isolated from the IRWST and refueling cavity, a leak in the SFS does not cause draining of the IRWST and the refueling cavity.

The Surveillance Requirement is modified by two Notes. Note 1 allows the affected penetration flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration flow path(s) can be rapidly isolated when a need for isolation is indicated.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Requirement is modified by Note 2 stating that it is only required to be met with refueling cavity and spent fuel pool volumes not in communication. In MODE 6 with refueling cavity and spent fuel pool volumes in communication, Refueling Cavity and SFS Isolation is actuated on a Spent Fuel Pool Level - Low 2 isolation signal as required by LCO 3.3.14, “ESFAS IRWST and Spent Fuel Pool Level Instrumentation.”

SR 3.5.8.4

Verification every 31 days that the boron concentration of the IRWST and refueling cavity is greater than the required limit ensures that the reactor will remain subcritical following shutdown events. Since the IRWST volume is large and normally stable, the 31 day Frequency is acceptable, considering additional verifications are required within 6 hours after each combined volume increase of ≥ 15,000 gal.

SR 3.5.8.5

LCO 3.5.6 Surveillance Requirements and Frequencies SR 3.5.6.3 and 3.5.6.5 through 3.5.6.11 are applicable to the IRWST and the flow paths required to be OPERABLE. Refer to the corresponding Bases for LCO 3.5.6 for a discussion of each SR.

REFERENCES None.

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B 3.5 PASSIVE CORE COOLING SYSTEM (PXS)

B 3.5.9 Reactor Coolant Pump Breakers

BASES

BACKGROUND

Core Makeup Tank (CMTs) provide sufficient borated water to assure Reactor Coolant System (RCS) reactivity and inventory control for all design basis accidents (DBAs), including both loss of coolant accident (LOCA) events and non-LOCA events. CMT performance assumes reactor coolant pumps (RCPs) have tripped (Ref. 1). Flow from the core makeup tank increases if the RCPs have coasted down, therefore, the RCP breakers are required to trip open after time delay to support the assumptions for CMT injection.

RCP trip is actuated on the following signals (refer to the Bases for LCO 3.3.8, Engineered Safety Feature Actuation System (ESFAS) Instrumentation):

- Safeguards Actuation;
- ADS Stages 1, 2, & 3 Actuation;
- Reactor Coolant Pump Bearing Water Temperature - High 2;
- Pressurizer Water Level - Low 2; and
- CMT Injection Actuation - Manual Initiation.

Each RCP is powered through two Class 1E circuit breakers connected in series, with one breaker receiving a trip signal from either Protection and Safety Monitoring System (PMS) Division B and the second breaker from Division C. The two breakers in series scheme is used to ensure that the RCPs are shutdown when the PMS tripping signal is initiated.

APPLICABLE SAFETY ANALYSES

When the CMTs are actuated, the RCPs are tripped (after a time delay) to allow steam and water to separate in the RCS and to allow steam displacement draindown. Tripping of the RCPs is done to enhance safety injection and avoid potential adverse interactions.

RCP Breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

This LCO establishes the OPERABILITY for two RCP breakers for each RCP. For RCP breakers to be considered OPERABLE they must trip open on an actuation signal or be maintained in the tripped condition.

APPLICABILITY

The Applicability is established consistent with the CMTs, which are required to be OPERABLE to provide borated water for RCS inventory makeup and reactivity control following a design basis event and subsequent cooldown. In MODES 1, 2, 3, and 4, CMT OPERABILITY is required to provide borated water for RCS inventory makeup and reactivity control following a design basis event and subsequent cooldown.

In MODE 5 with the RCS not VENTED, CMT OPERABILITY is required to provide borated water to the RCS in the event the nonsafety related chemical and volume control system makeup pumps are not available to provide RCS inventory control.

The RCP breakers are not required to be OPERABLE while in MODE 5 with the RCS VENTED, or in MODE 6, because the RCS is depressurized and borated water can be supplied from the In-containment Refueling Water Storage Tank (IRWST), if needed. The RCS is considered VENTED when the ADS stage 1, 2, and 3 flow paths (or alternative flow path with equivalent area) required by LCO 3.4.13, "Automatic Oepressurization System (ADS) – Shutdown, RCS Open," are open, such that when combined with required ADS stage 4 actuation, sufficient steam venting area is provided to lower RCS pressure sufficiently to allow IRWST injection.

ACTIONS

The Actions are modified by a note indicating that separate condition entry is allowed for each RCP. This is acceptable, since the Required Actions provide appropriate compensatory actions for each RCP. Complying with the Required Actions allows subsequent RCPs with inoperable breakers to be governed by subsequent Condition entry and application of associated Required Actions.

A.1

With one or more RCPs with one breaker inoperable in MODE 1, 2, 3, or 4, action must be taken to restore the breaker(s) within 6 hours. The 6 hours allowed to restore the inoperable breaker is reasonable based on the capability of the remaining OPERABLE breaker to mitigate all DBAs and the low probability of an event occurring during this interval and OPERABILITY of the redundant RCP breaker.

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BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action or associated Completion Time of Condition A is not met or one or more RCPs with two breakers are inoperable in MODE 1, 2, 3, or 4, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

With one or more RCPs with one breaker inoperable in MODE 5, action must be taken to restore the breaker(s) within 72 hours. The 72 hours is reasonable considering the remaining OPERABLE breaker is capable of performing the associated safety function and the low probability of an event occurring during this interval.

D.1

If the Required Action or associated Completion Time of Condition C is not met or one or more RCPs with two breakers are inoperable in MODE 5, action must be initiated immediately to trip the inoperable RCP breaker(s). Tripping the affected RCP(s) accomplishes the necessary safety function to support CMT OPERABILITY.

SURVEILLANCE REQUIREMENTS

SR 3.5.9.1

This SR verifies that RCP breakers trip open on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valve.

The SR excludes circuit breakers that are maintained in the tripped position since a tripped open circuit breaker is already performing the safety function. Restoration of a circuit breaker to the non-tripped position requires verification that the SR has been met within its required Frequency.

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

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BASES

REFERENCES 1. FSAR Section 6.3, “Passive Core Cooling System.”

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low-leakage steel vessel designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) such that offsite radiation exposures are maintained within limits. The containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with elliptical upper and lower heads, completely enclosed by a seismic Category I reinforced concrete shield building. A 4.5 foot wide annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to permit inservice inspection and air flow over the steel dome for containment cooling. The containment utilizes the outer concrete building for shielding and a missile barrier, and the inner steel containment for leak tightness and passive containment cooling.

Containment piping penetration assemblies provide for the passage of process, service and sampling pipelines into the containment vessel while maintaining containment integrity. The shield building provides biological shielding and environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate Surveillance Requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

    1. Capable of being closed by an OPERABLE automatic containment isolation system, or

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BASES

BACKGROUND (continued)

2. Closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, “Containment Isolation Valves”;

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, “Containment Air Locks”; and

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. The DBA analyses assume that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment is designed with an allowable leakage rate of 0.10% of containment air weight of the original content of containment air after a DBA per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on containment leakage rate testing. L_a is assumed to be 0.10% per day in the safety analysis.

Satisfactory leakage rate test results is a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program Leakage Test. At this time, the applicable leakage limits must be met.

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BASES

LCO (continued)

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. The MODES 5 and 6 requirements are specified in LCO 3.6.7, “Containment Penetrations.”

ACTIONS A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and < 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. 10 CFR 50, Appendix J, Option B, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Performance-Based Requirements.”
2. FSAR Chapter 15, “Accident Analysis.”
3. FSAR Section 6.2, “Containment Systems.”

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, 10 feet in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBA that results in the largest release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 3). In the analyses of DBAs, it is assumed that containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of containment leakage. The containment is designed with an allowable leakage rate of 0.10% of containment air weight of the original content of containment air per day after a DBA (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of offsite radiation exposures resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is necessary to support containment OPERABILITY following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4 a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES and large inventory of coolant. Therefore, containment air locks are not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment. However, containment closure capability is required within MODES 5 and 6 as specified in LCO 3.6.7, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for repair without interrupting containment integrity. If containment entry is required, it is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

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BASES

ACTIONS (continued)

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event that air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, “Containment.”

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, “Containment,” which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is reasonable based on engineering judgement and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified to be locked closed by administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

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BASES

ACTIONS (continued)

The Required Actions are modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the airlock are inoperable. With both doors in the same airlock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of an airlock for entry and exit for 7 days, under administrative controls if both airlocks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specification (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (non-TS-related activities) if the containment is entered, using the inoperable airlock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock door interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same airlock are inoperable. With both doors in the same airlock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the airlock to ensure that only one door is opened at a time (the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to airlock doors located in high radiation areas that allows these doors to be verified locked closed by administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position is small.

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BASES

ACTIONS (continued)

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with respect to air lock leakage (Type B leakage tests). The acceptance

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SURVEILLANCE REQUIREMENTS (continued)

criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is as required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock door interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at 24 month Frequency. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.

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BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Performance-Based Requirements.”
2. FSAR Section 6.2, “Containment Systems.”
3. FSAR Chapter 15, “Accident Analysis.”

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. FSAR Section 6.2 (Ref. 1) identifies parameters which initiate isolation signal generation for containment isolation valves. The containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that containment function assumed in the safety analysis will be maintained.

<u>Containment Air Filtration System 16-inch purge valves</u>

The Containment Air Filtration System operates to:

a. Supply outside air into the containment for ventilation and cooling or heating,

b. Reduce the concentration of noble gases within containment prior to and during personnel access, and

c. Equalize internal and external pressures.

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BASES

BACKGROUND (continued)

Since the valves used in the Containment Air Filtration System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3 and 4.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 2). In the analyses for each of the accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized.

The DBA dose analysis assumes that, following containment isolation signal generation, the containment purge isolation valves are closed within 10 seconds. The remainder of the automatic isolation valves are assumed closed and the containment leakage is terminated except for the design leakage rate, L_a. Since the containment isolation valves are powered from the 1E division batteries no diesel generator startup time is applied.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the design of the containment purge isolation valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are pneumatically operated, spring closed valves that fail in the closed position and are provided with power via independent sources.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

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BASES

LCO (continued)

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in the FSAR Section 6.2 (Ref. 1).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, or blind flanges are in place. These passive isolation valves/devices are those listed in Reference 1.

This LCO provides assurance that the containment isolation valves, except for the closed system valves and vacuum relief valves, and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents. The containment isolation valves associated with closed systems are not included in this LCO since they are covered in LCO 3.7.1, "Main Steam Safety Valves (MSSVs)," LCO 3.7.2, “Main Steam Line Flow Path Isolation Valves,” LCO 3.7.3, “Main Feedwater Isolation Valves (MFIVs), Main Feedwater Control Valves (MFCVs), and Main Feedwater (MFW) Pump Breakers,” LCO 3.7.7, “Startup Feedwater (SFW) Isolation Valves, Control Valves, and Pump Breakers,” and LCO 3.7.10, “Steam Generator (SG) Isolation Valves.”

The containment system vacuum relief valves provide containment isolation but are also required to open to mitigate a negative pressure event within containment. Therefore, the vacuum relief valves are not included in this LCO since they are covered in LCO 3.6.9, “Vacuum Relief Valves.”

APPLICABILITY In MODES 1, 2, 3, and 4 a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment isolation valves are not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment. However, containment closure capability is required in MODES 5 and 6. The requirements for containment isolation valves during MODES 5 and 6 are addressed in LCO 3.6.7, “Containment Penetrations.”

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BASES

ACTIONS

The ACTIONS are modified by a Note allowing containment penetration flow paths to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event that the containment isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, or a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable considering the time required to isolate the penetration, the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4, and the availability of a second barrier.

For affected penetrations that cannot be restored to OPERABLE status within the 4 hour Completion Time and have been isolated in accordance with Required Action A.1, the affected penetrations must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations that are required to be isolated following an accident and that are no longer capable of being automatically isolated, will be in the

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BASES

ACTIONS (continued)

isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high-radiation areas, and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2 which remains in effect. This periodic verification is necessary to ensure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

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BASES

ACTIONS (continued)

C.1 and C.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

This SR ensures that the 16 inch purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the 16 inch purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The 16 inch purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency specified as “prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days” is appropriate since these containment isolation valves are operated under administrative control and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation times are specified in FSAR Section 6.2.3 (Ref. 1) and Frequency of this SR is in accordance with the Inservice Testing Program.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Automatic containment isolation valves close on their respective ESF signal to prevent leakage of radioactive material from containment following a DBA. The actual or simulated actuation signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valve. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on its respective ESF signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 6.2, “Containment Systems.”
2. FSAR Chapter 15, “Accident Analysis.”
3. NUREG-1449, “Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States.”

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of transients which result in a negative pressure.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the operating band of conditions used in the containment pressure analyses for the Design Basis Events which result in internal or external pressure loads on the containment vessel. Should operation occur outside these limits, the initial containment pressure would be outside the range used for containment pressure analyses.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst cast SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). This resulted in a maximum peak pressure from a LOCA, P_a, of 58.1 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, P_a, does not exceed the containment design pressure, 59 psig.

The containment was also designed for an external pressure load equivalent to 1.7 psid. The limiting negative pressure transient is a loss of all AC power sources coincident with extreme cold weather conditions which cool the external surface of the containment vessel. The initial pressure condition used in this analysis was -0.2 psig. This resulted in a minimum pressure inside containment, as illustrated in Reference 1,

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BASES

APPLICABLE SAFETY ANALYSES (continued)

which is less than the design load. Other external pressure load events evaluated include:

Failed fan cooler control

Malfunction of containment purge system

Inadvertent Passive Containment Cooling System (PCS) actuation

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure.

Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following negative pressure transients. If the containment pressure does not meet the low pressure limit, the containment vacuum relief capacity of one flow path may not be adequate to ensure the containment pressure meets the negative pressure design limit.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within the high pressure limit is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the high pressure limit of the LCO is not required in MODE 5 or 6.

In MODES 1 through 6, the potential exists for excessive containment cooling events to produce a negative containment pressure below the design limit. However, in MODES 5 and 6, a containment air flow path may be opened (LCO 3.6.7, Containment Penetrations), providing a vacuum relief path that is sufficient to preclude a negative containment pressure below the design limit.

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BASES

APPLICABILITY (continued)

Therefore, maintaining containment pressure within the low pressure limit is essential to ensure initial conditions assumed in the containment cooling events in MODES 1 through 4 and in MODES 5 and 6 without an open containment air flow path ≥ 6 inches in diameter. With a 6 inch diameter or equivalent containment air flow path, the vacuum relief function is not needed to mitigate a low pressure event.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, “Containment,” which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If the containment pressure is not restored to within its limits within the required Completion Time in MODE 1, 2, 3, or 4, the plant must be placed in a condition in which the probability and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If the containment pressure is not restored to within its limits within the required Completion Time in MODE 5 or 6, Required Action C.1 requires that action be immediately initiated to restore containment pressure to within kimits and to continue this action until restoration is accomplished.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the main control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. FSAR, Section 6.2, “Containment Systems.”

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the passive containment cooling system during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment Engineered Safety Feature (ESF) systems, assuming the loss of one Class 1E Engineered Safety Features Actuation Cabinet (ESFAC) Division, which is the worst case single active failure, resulting in one Passive Containment Cooling System flow path being rendered inoperable.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting DBA for the maximum peak containment air temperature is a LOCA or SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F.

The DBA temperature transients are used to establish the environmental qualification operating envelope for containment. The basis of the containment environmental qualification temperature envelope is to ensure the performance of safety related equipment inside containment (Ref. 2). The containment vessel design temperature is 300°F. The containment vessel temperature remains below 300°F for DBAs. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBAs.

The temperature limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Passive Containment Cooling System (Ref. 1).

The containment is designed for an external pressure load equivalent to 1.7 psid. The limiting negative pressure transient is a loss of all ac power sources coincident with extreme cold weather conditions, which cool the external surface of the containment vessel. The initial containment average air temperature condition used in this analysis is 120°F. This resulted in a minimum pressure inside containment, as illustrated in Reference 1, which is less than the design load.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is an SLB or LOCA. The temperature limit is used in the DBA analyses to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is computed to remain within acceptable limits. As a result, the ability of containment to perform its design function is ensured.

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BASES

LCO (continued)

The LCO establishes the maximum containment average air temperature initial condition required for the excessive cooling analysis. If the containment average air temperature exceeds the limit, the containment vacuum relief capacity of one flow path may not be adequate to ensure the containment pressure meets the negative pressure design limit.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6 for a DBA LOCA or SLB.

In MODES 1 through 6, the potential exists for excessive containment cooling events to produce a negative containment pressure below the design limit. However, in MODES 5 and 6, a containment equipment hatch or airlock may be opened (LCO 3.6.7, “Containment Penetrations”), providing a vacuum relief path that is sufficient to preclude a negative containment pressure below the design limit.

Therefore, maintaining containment average air temperature within the limit is essential to ensure initial conditions assumed in the cooling events in MODES 1 through 4 and in MODES 5 and 6 with both containment equipment hatches and both containment airlocks closed.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within its limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the conservative analysis to variations in this parameter, and provides sufficient time to correct minor problems.

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BASES

ACTIONS (continued)

B.1 and B.2

If the containment average air temperature is not restored to within its limit within the required Completion Time, the plant must be placed in a condition in which probability and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If the containment average air temperature is not restored to within its limits within the required Completion Time in MODE 5 or 6, Required Action C.1 requires that action be immediately initiated to restore containment average air temperature to within limits and to continue this action until restoration is accomplished.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that the containment average air temperature is within the LCO limit ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, a weighted average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the associated containment atmosphere. The 12 hour Frequency of this Surveillance Requirement is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the main control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. FSAR Section 6.2, “Containment Systems.”
2. 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Passive Containment Cooling System (PCS)

BASES

BACKGROUND

The PCS provides containment cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Passive Containment Cooling System is designed to meet the requirements of 10 CFR 50 Appendix A GDC 38 “Containment Heat Removal” and GDC 40 “Testing of Containment Heat Removal Systems” (Ref. 1).

The PCS consists of an 800,000 gal (nominal) Passive Containment Cooling Water Storage Tank (PCCWST), four headered PCCWST discharge lines with flow restricting orifices, and two separate full capacity discharge headers to the containment vessel with 3 sets of isolation valves (i.e., 3 flow paths), each flow path capable of meeting the design bases. Algae growth is not expected within the PCCWST; however, to assure water clarity is maintained, a prevailing concentration of hydrogen peroxide is maintained at 50 ppm. The recirculation pumps and heater provide freeze protection for the PCCWST. However, OPERABILITY of the PCCWST is assured by compliance with the temperature limits specified in SR 3.6.6.1 and not by the recirculation pumps and heater. In addition to the recirculation pumps and heater, the PCCWST temperature can be maintained within limits by the ambient temperature, the large thermal inertia of the PCCWST, or heat from other sources. The PCS valve room temperature must not be below freezing for an extended period to assure the water flow path to the containment shell is available. The isolation valves on each flow path are powered from a separate Division.

Upon actuation of the isolation valves, gravity flow of water from the PCCWST (contained in the shield building structure above the containment) onto the upper portion of the containment shell reduces the containment pressure and temperature following a DBA. The flow of water to the containment shell surface is initially established to assure that the short term containment cooling requirements following the postulated worst case LOCA are achieved. As the decay heat from the core becomes less with time, the water flow to the containment shell is reduced in three steps. The change in flow rate is attained without active components in the system and is dependent only on the decreasing water level in the elevated PCCWST. In order to ensure the containment

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BASES

BACKGROUND (continued)

surface is adequately and effectively wetted, the water is introduced at the center of the containment dome and flows outward. Weirs are placed on the dome surface to distribute the water and ensure effective wetting of the dome and vertical sides of the containment shell.

The path for the natural circulation of air is from the air intakes in the shield building, down the outside of the baffle, up along the containment shell to the top center exit in the shield building and is always open. The drains in the upper annulus region must be clear to prevent water from blocking the air flow path. Heat is removed from within the containment utilizing the steel containment shell as the heat transfer surface combining conductive heat transfer to the water film, convective heat transfer from the water film to the air, radiative heat transfer from the film to the air baffle, and mass transfer (evaporation) of the water film into the air. As the air heats up and water evaporates, it becomes less dense than the cooler air in the air inlet annulus. This differential causes an increase in the natural circulation of the air upward along the containment surface, with heated air/water vapor exiting the top center of the shield building. Additional system design details are provided in Reference 3.

The PCS is actuated either automatically, by a containment High-2 pressure signal, or manually. Automatic actuation opens the PCCWST discharge valves, allowing gravity flow of the cooling water onto the containment shell. The manual containment cooling actuation consists of two momentary controls. Operating either control will actuate all applicable divisions. The discharge continues for at least three days.

The PCS is designed to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The PCS is an engineered safety features (ESF) system and is designed to ensure that the heat removal capability required during the post accident period can be attained.

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BASES

APPLICABLE SAFETY ANALYSES

The Passive Containment Cooling System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF system, assuming the loss of one Class 1E Engineered Safety Features Actuation Cabinet (ESFAC) Division, which is the worst case single active failure and results in one PCS flow path being inoperable.

The analyses and evaluations assume 100% RTP, one passive containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Passive Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment cooling system performance for post accident conditions is given in Reference 3. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition.

The modeled Passive Containment Cooling System actuation response time from the containment analysis is based upon a response time associated with exceeding the containment High-2 pressure setpoint to opening of isolation valves.

The PCS limits the temperature and pressure that could be experienced during shutdown following a loss of decay heat removal. For shutdown events, the Reactor Coolant System (RCS) sensible and decay heat removal requirements are reduced as compared to heat removal requirements for MODE 1, 2, 3, or 4 events. Therefore, the shutdown containment heat removal requirements are bounded by analyses of MODES 1, 2, 3, and 4 events.

The Passive Containment Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

During a DBA, one passive containment cooling water flow path is required to maintain the containment peak pressure and temperature below the design limits (Ref. 3). To ensure that this requirement is met, two passive containment cooling water flow paths are provided.

Therefore, in the event of an accident, at least one flow path operates, assuming the worst case single active failure occurs. A third PCS flow path is provided for protection against multiple failure scenarios modeled in the PRA. To ensure that these requirements are met, three PCS water flow paths must be OPERABLE.

The PCS includes the PCCWST, valves, piping, instruments and controls to ensure an OPERABLE flow path capable of delivering water from the PCCWST upon an actuation signal. An OPERABLE flow path consists of a normally closed valve capable of automatically opening in series with a normally open valve. For the two flow paths containing air-operated valves, it is preferred because of PRA insights that these valves be normally closed.

The PCCWST ensures that an adequate supply of water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA). To be considered OPERABLE, the PCCWST must meet the water volume and temperature limits established in the SRs. To be considered OPERABLE, the air flow path from the shield building annulus inlet to the exit must be unobstructed, with unobstructed upper annulus safety-related drains providing a path for containment cooling water runoff to preclude blockage of the air flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the PCS.

OPERABILITY of the PCS is required in either MODE 5 or 6 with the reactor decay heat (normally determined by calculation) greater than 7.0 MWt for heat removal in the event of a loss of nonsafety decay heat removal capabilities. With the decay heat at or below 7.0 MWt, the decay heat can be removed from containment with air cooling alone for 72 hours. Confirmation of decay heat levels may be determined consistent with the assumptions and analysis basis of ANS 1979 plus 2 sigma or via an energy balance of the reactor coolant system.

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BASES

ACTIONS

A.1

With one passive containment cooling water flow path inoperable, the affected flow path must be restored within 7 days. In this degraded condition, the remaining flow paths are capable of providing greater than 100% of the heat removal needs after an accident, even considering the worst single failure. The 7 day Completion Time was chosen in light of the remaining heat removal capability and the low probability of a DBA occurring during this period.

B.1

With two passive containment cooling water flow paths inoperable, at least one affected flow path must be restored to OPERABLE status within 72 hours. In this degraded condition, the remaining flow path is capable of providing greater than 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen in light of the remaining heat removal capability and the low probability of DBA occurring during this period.

C.1

If the PCCWST water volume or temperature is not within limits, it must be restored to within limits within 8 hours. The 8 hour Completion Time is reasonable based on the remaining heat removal capability of the system and the availability of cooling water from alternate sources.

D.1 and D.2

If any of the Required Actions and associated Completion Times are not met in MODE 1, 2, 3, or 4, or if the LCO is not met for reasons other than Condition A, B, or C when in MODE 1, 2, 3, or 4, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

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BASES

ACTIONS (continued)

E.1 and E.2

If any of the Required Actions and associated Completion Times of Condition A, B, or C are not met in MODE 5 or 6, or if the LCO is not met for reasons other than Condition A, B, or C when in MODE 5 or 6, action must be initiated, immediately, to restore compliance with the LCO and to continue this action until restoration is accomplished in order to provide the PCS for required heat removal needs. In addition, action be immediately initiated to suspend positive reactivity additions to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

This surveillance requires verification that the PCCWST water temperature is within the limits assumed in the accident analyses. The 24 hour Frequency is adequate to identify a temperature change that would approach the temperature limits since the PCCWST is large and temperature variations are slow.

SR 3.6.6.2

Verification that the cooling water volume is above the required minimum ensures that a sufficient supply is available for containment cooling. Since the cooling water volume is normally stable and low level is indicated by a main control room alarm, a 7 day Frequency is appropriate and has been shown to be acceptable in similar applications.

SR 3.6.6.3

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the PCS flow path provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through control room instrumentation or a system walkdown, that valves capable of potentially being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single flow path. This Frequency has been shown to be acceptable through operating experience.

SR 3.6.6.4

This SR requires verification that each automatic isolation valve actuates to its correct position upon receipt of an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.5

Periodic inspections of the PCS air flow path from the shield building annulus inlet to the exit ensure that it is unobstructed, the baffle plates are properly installed, and the upper annulus safety-related drains are unobstructed. Although there are no anticipated mechanisms which would cause air flow path or annulus drain obstruction and the effect of a missing air baffle section is small, it is considered prudent to verify this capability every 24 months. Additionally, the 24 month Frequency is based on the desire to perform this Surveillance under conditions that apply during a plant outage, the need to have access to the locations, and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation in similar situations.

SR 3.6.6.6

This SR requires performance of a Passive Containment Cooling System test to verify system flow and water coverage capabilities. The system performance test demonstrates that the containment cooling capability assumed in accident analyses is maintained by verifying the flow rates via each standpipe and measurement of containment wetting coverage. The System Level OPERABILITY Testing Program provides specific test requirements and acceptance criteria. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The first refueling and 10 year Frequency is based on the ability of the more frequent surveillances to verify the OPERABILITY of the active components and features which could degrade with time.

REFERENCES

1. 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants.”
2. 10 CFR 50, Appendix K, “ECCS Evaluation Models.”
3. FSAR Chapter 6.2, “Containment Systems.”

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Containment Penetrations

BASES

BACKGROUND

Containment closure capability is required during shutdown operations when there is fuel inside containment. Containment closure is required to maintain the cooling water inventory within containment. Due to the large volume of the In-Containment Refueling Water Storage Tank (IRWST) and the reduced sensible heat during shutdown, the loss of some of the water inventory can be accepted. Since there are no unique MODES 5 and 6 containment pressurization response analyses, containment pressurization is assumed to be bounded by the design basis accident containment response. Therefore, containment closure must be capable of maintaining containment isolation at the containment design pressure of 59 psig to prevent gross failure of containment opening barriers that would challenge maintaining long-term core cooling water inventory (Ref. 1).

Further, accident analyses have shown that containment closure capability is not required to meet offsite dose requirements. Therefore, containment does not need to be leak tight as required for MODES 1 through 4. In MODES 5 and 6, the LCO requirements are referred to as “containment closure” rather than “containment OPERABILITY.” Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no requirement for containment leak tightness, compliance with the Appendix J leakage criteria and tests are not required.

In MODES 5 and 6, there is no potential for steam release into the containment immediately following an accident. Pressurization of the containment could only occur after heatup of the IRWST due to Passive Residual Heat Removal Heat Exchanger (PRHR HX) operation (MODE 5 with RCS intact) or after heatup of the RCS with direct venting to the containment (MODE 5 with reduced RCS inventory or MODE 6 with the refueling cavity not fully flooded) or after heatup of the RCS and refueling cavity (MODE 6 with refueling cavity fully flooded). The time from loss of normal cooling until steam release to the containment for four representative sets of plant conditions is shown in Figure B 3.6.7-1 as a function of time after shutdown. Because local manual action may be required to achieve containment closure it is assumed that the containment hatches, air locks and penetrations must be closed prior to steaming into containment.

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BASES

BACKGROUND (continued)

Figure B 3.6.7-1 provides allowable closure times for four representative sets of plant conditions. The time to steaming is dependent on various plant parameters (RCS temperature, IRWST temperature, etc.) and plant configuration (RCS Pressure Boundary Intact, RCS Open, etc.). Therefore, the actual representation of the time to steaming may be different than that provided in Figure B 3.6.7-1. In determining the minimum time to steaming, conservative assumptions regarding core decay heat, RCS configuration, and initial RCS inventory are used to minimize the calculated time to steaming. The curves are based on the core decay heat prior to refueling so that closure times are longer following the core reload.

As presented in Tables 54-1 and 54-4 of Reference 2, the most risk significant events during shutdown are events that lead to a loss of Normal Residual Heat Removal System (RNS) cooling. Of these, the limiting events that lead to steaming to containment are the loss of shutdown cooling events, specifically:

- Loss of decay heat removal during drained conditions due to a failure of component cooling water or service water system;
- Loss-of-offsite power during drained conditions; and
- Loss of decay heat removal during drained conditions due to failure of the normal residual heat removal system.

These events are further discussed in FSAR Section 19.19. Time to steaming is dependent on the postulated RCS configuration (intact versus open), and is based on the response of the plant considering features such as the operation of the 4th stage Automatic Depressurization System (ADS) valves if necessary, status of the upper internals, status of refueling cavity, etc. Conservative assumptions regarding these features are made in the determination of the minimum time to steaming. The time assumed in the PRA to close the penetrations before steaming to containment included 15 minutes for the diagnosis and decision-making time, in addition to the time required to physically complete the closure action.

The risk of overdraining the RCS has been significantly reduced due to the automatic protection features associated with the hot leg level instruments which isolate letdown on Low 2 hot leg water level. Overdraining the RCS is no longer a significant contributor to core damage, as shown in Table 54-4 of Reference 2.

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BASES

BACKGROUND (continued)

The assumptions used in determining the required closure time for the various containment openings should be conservative, and should be consistent with the plant operating procedures, staffing levels, and status of the containment openings. The evaluation should consider the ability to close the containment for the limiting loss of shutdown cooling event, and considering the possibility of a station blackout. In determining if containment can be closed within the time permitted to containment closure specified in Figure B 3.6.7-1, the time to close containment penetrations must include both the diagnosis and decision-making time and the time required to physically complete the closure action.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that bolts required by this LCO be approximately equally spaced. Alternatively, if open, each equipment hatch can be installed using a dedicated set of hardware, tools and equipment. A self-contained power source is provided to drive each hoist while lowering the hatch into position. Large equipment and components may be moved through the hatches as long as they can be removed and the hatch closed prior to steaming into the containment.

The design of the equipment hatch is such that the four bolts would only be needed to support the hatch in place and provide adequate strength to support the hatch dead weight and associated loads. The hatch is installed on the inside containment and is held in place against a matching flange surface with mating bolt pattern by the bolts. Once the dead weight is supported, any pressure (greater than atmospheric) within containment will serve to exert closure force on the hatch toward the mating flange surface serving to reduce stresses on bolts. Therefore the determination of the number of bolts is limited to the quantity required to support the hatch itself and not related to any potential containment pressure.

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BASES

BACKGROUND (continued)

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, “Containment Air Locks.” Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Temporary equipment connections (e.g., power or communications cables) are permitted as long as they can be removed to allow containment closure prior to steaming into the containment.

Containment spare penetrations which also provide a part of the containment boundary provide for temporary support services (electrical, I&C, air, and water supplies) during MODES 5 and 6. Each penetration is flanged and normally closed. During periods of plant shutdown, temporary support systems may be routed through the penetrations; temporary equipment connections (e.g., power or communications cables) are permitted as long as they can be removed to allow containment closure prior to steaming into the containment. The spare penetrations must be closed or, if open, capable of closure prior to steaming to containment.

Containment penetrations, including purge system flow paths, that provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being isolated prior to steaming into the containment on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary barrier for the containment penetrations. The equivalent isolation barrier must be capable of maintaining containment isolation at the containment design pressure of 59 psig (Ref. 1).

APPLICABLE SAFETY ANALYSES

For postulated shutdown events in MODES 5 and 6, RCS heat removal is provided by either passive residual heat removal (PRHR) or IRWST injection and containment sump recirculation. To support RCS heat removal, containment closure is required to limit the loss of the cooling water inventory from containment (Ref. 1).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

This LCO limits the loss of cooling water inventory in containment to assure continued coolant inventory by limiting the potential escape paths for water released within containment. Penetrations closed in accordance with these requirements are not required to be leak tight.

The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed or capable of being closed prior to steaming into the containment. The equipment hatches may be open; however, the hatches shall be clear of obstructions such that capability to close the hatch within the indicated time period is maintained. The hardware, tools, equipment and power sources necessary to install the hatches shall be available when the hatch is open. Both doors in each containment air lock may be open; however, the air locks shall be clear of obstructions such that the capability to close at least one door within the indicated time period is maintained. Alternatively, one door in an air lock may be closed. Containment spare penetrations may be open; however, the penetrations shall be clear of obstructions such that the penetrations are capable of being closed within the indicated time period. Direct access penetrations shall be closed by at least one manual or automatic isolation valve, blind flange or equivalent, or capable of being closed by at least one valve actuated by their respective ESF signal. If direct access penetrations are open, then they must be capable of being closed prior to steaming into the containment. Figure B 3.6.7-1 provides the acceptable required closure times for various representative MODES and conditions.

APPLICABILITY

The containment penetration requirements are applicable during conditions for which the primary safety related core cooling and boration capabilities are provided by IRWST or injection or PRHR – MODES 5 and 6. The capability to close containment is required to ensure that the cooling water inventory is not lost in the event of an accident.

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1.

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BASES

ACTIONS

A.1

If the containment equipment hatches, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the containment isolation capability when automatic isolation valves are open, the penetration(s) must be restored to the required status within 1 hour.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met or the LCO is not met for reasons other than Condition A, action must be initiated, immediately, to restore compliance with the LCO and to continue this action until restoration is accomplished. In addition, action be immediately initiated to suspend positive reactivity additions to ensure that the SDM is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of control rods, and excessive cooling of the RCS.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. For any open purge and exhaust valves, this SR will demonstrate that the valves are not blocked from closing. Also if the purge and/or exhaust valves are open and relying on automatic closure, this Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an automatic containment purge and exhaust isolation signal prior to steaming into containment. Alternately, purge and exhaust penetrations may be verified closed or capable of being closed prior to steaming into the containment, which will limit loss of the cooling water inventory from containment. Open containment spare penetrations shall be verified capable of being closed prior to steaming into the containment by removal of obstructions and installation of the flange or by other closure means which will limit loss of the cooling water inventory from containment.

The Surveillance is performed every 7 days. The Surveillance interval is selected to ensure that the required penetration status is maintained during shutdown inspections, testing, and maintenance.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

Each of the two equipment hatches is provided with a set of hardware, tools, equipment, and self-contained power source for moving the hatch from its storage location and installing it in the opening. The required set of hardware and tools shall be visually inspected to ensure that they can perform the required functions. The equipment and power source shall be inspected and/or operated as necessary to verify that the hatch can be closed properly. The power source shall be verified as containing sufficient energy to install the hatch from the storage location.

The 7 day Frequency is adequate considering that the hardware, tools, equipment, and power sources are dedicated to the associated equipment hatch and not used for any other functions.

The SR is modified by a Note which only requires that the surveillance be met for an open equipment hatch. If the equipment hatch is installed in position, then the availability of the means to install the hatch is not required.

REFERENCES

1. FSAR subsection 19E.2.6.2.

2. AP1000 Probabilistic Risk Assessment.

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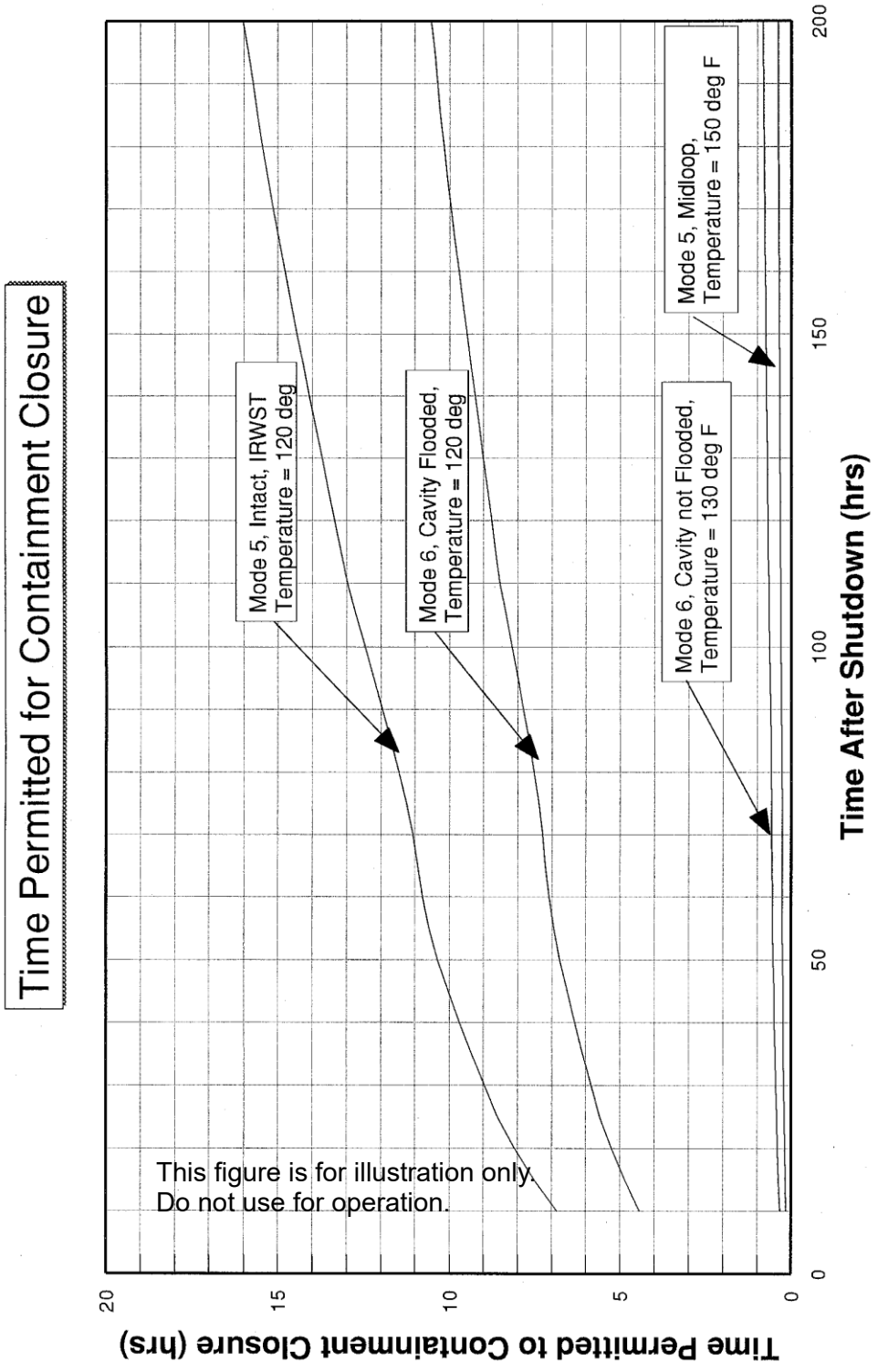

Figure B 3.6.7-1 (page 1 of 1)
Time Prior to Coolant Inventory Boiling

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 pH Adjustment

BASES

BACKGROUND

The Passive Core Cooling System (PXS) includes four pH adjustment baskets which provide adjustment of the pH of the water in the containment following an accident where the containment floods.

Following an accident with a large release of radioactivity, the containment pH is automatically adjusted to greater than or equal to 7.0, to enhance iodine retention in the containment water. Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-LOCA environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and pyrolysis of electric cable insulation). The desired pH values significantly reduce formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking of safety related containment components during long-term cooling.

Dodecahydrate trisodium phosphate (TSP) contained in baskets provides a passive means of pH control for such accidents. The baskets are made of stainless steel with a mesh front that readily permits contact with water. These baskets are located inside containment at an elevation that is below the minimum floodup level. The baskets are placed at least a foot above the floor to reduce the chance that water spills will dissolve the TSP. Natural recirculation of water inside the containment, following a LOCA, is driven by the core decay heat and provides mixing to achieve a uniform pH. The dodecahydrate form of TSP ($Na_3PO_4 \cdot 12H_2O$) is initially loaded into the baskets because it is hydrated and will undergo less physical and chemical change than would anhydrous TSP as a result of the humidity inside containment. (Refs. 1 and 2)

APPLICABLE SAFETY ANALYSES

In the event of a Design Basis Accident (DBA), iodine may be released from the fuel to containment. To limit this iodine release from containment, the pH of the water in the containment sump is adjusted by the addition of TSP. Adjusting the sump water to neutral or alkaline pH (pH ≥ 7.0) will augment the retention of the iodine, and thus reduce the iodine available to leak to the environment.

pH adjustment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

The requirement to maintain the pH adjustment baskets with ≥ 26,460 lbs of TSP assures that for DBA releases of iodine into containment, the pH of the containment sump will be adjusted to enhance the retention of the iodine.

APPLICABILITY

In MODES 1, 2, 3, and 4 a DBA could cause release of radioactive iodine to containment requiring pH adjustment. The pH adjustment baskets assist in reducing the airborne iodine fission product inventory available for release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, pH adjustment is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

If the TSP weight in the pH adjustment baskets is not within limits, the iodine retention may be less than that assumed in the accident analysis for the limiting DBA. Due to the very low probability that the weight of TSP may change, the variations are expected to be minor such that the required capability is substantially available. The 72 hour Completion Time for restoration to within limits is consistent with times applied to minor degradations of ECCS parameters.

B.1 and B.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

The minimum amount of TSP is 26,460 lbs. This weight is based on providing sufficient TSP to buffer the post accident containment water to a minimum pH of 7.0. Additionally, the TSP weight is based on treating the maximum volume of post accident water (867,830 gallons) containing the maximum amount of boron (3050 ppm) as well as other sources of acid. The minimum required mass of TSP is 26,460 lbs at an assumed assay of 100%.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

While a weight is specified, the normal manner to confirm the weight limit is met is by measuring the volume of the TSP contained in the pH adjustment baskets. The minimum measured volume of TSP is 540 ft^3 to meet the minimum required mass of TSP (26,460 lbs) and assumes the minimum density of TSP. The minimum TSP density is based on the manufactured density (49 lbm/ft^3), since the density may increase and the volume decrease, during plant operation, due to agglomeration from humidity inside the containment. A TSP volume of 560 ft^3 at the initial loading (i.e., prior to compaction and agglomeration which may occur over time) provides about 4% margin above the minimum required TSP mass of 26,460 lbs.

The periodic verification is required every 24 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 24 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

SR 3.6.8.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of TSP from one of the baskets in containment is submerged in 1.0 ± 0.01 liter of water at a boron concentration of 3050 ppm that has been heated to a temperature of 71 ± 5°C (160 ± 9°F). The solution is allowed to stand at this temperature for 4 hours without agitation. The solution is then cooled to 25 ± 5°C (77 ± 9°F) and the pH is measured. The solution pH should rise to ≥ 7.0. A TSP sample size of 2.19 grams is used if the TSP volume is verified to be at least 540 ft^3 (minimum required). However, a larger TSP sample size can be used if the measured volume is verified to be greater than the minimum. For example, if the TSP volume is verified to be 560 ft^3, then a representative sample of 2.27 grams can be used.

Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The time of 4 hours without agitation is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA sump area, rapid mixing would occur due to liquid flow, significantly decreasing the actual amount of time before the required pH is achieved.

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BASES

REFERENCES

1. FSAR Section 6.3.2.1.4, “Containment pH Control.”
2. FSAR Section 6.3.2.2.4, “pH Adjustment Baskets.”

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Vacuum Relief Valves

BASES

BACKGROUND

The purpose of the vacuum relief lines is to protect the containment vessel from damage due to a negative pressure (that is, a lower pressure inside than outside). Excessive negative pressure inside containment can occur, if there is a loss of ac power (containment recirculation cooling system (VCS) containment heating not available, reactor trip decay heating only) with a differential (inside to outside) ambient temperature > 90°F. In this case, the relative low outside ambient temperature may cool containment faster than the available heat sources (primarily, reactor decay heat) can heat containment, resulting in a reduction of the containment temperature and pressure below the negative pressure design limit since normal non-safety-related pressure control means are not available due to loss of ac power. In addition, excessive negative pressure inside containment can occur, in the event of malfunction of the Containment Fan Coolers (containment air filtration system (VFS)) control, in combination with low outside ambient temperature, which reduces containment temperature. The vacuum relief valves have an additional function to provide containment isolation to establish a containment boundary during positive pressure accident conditions.

The containment pressure vessel is protected from excessive external pressure loading by redundant 100% capacity vacuum relief capability within a shared containment penetration. There are two outside containment normally closed vacuum relief isolation valves, which are motor-operated valves (MOVs). The vacuum relief isolation valves receive an engineered safety features (ESF) “open” signal on Containment Pressure–Low 2. The vacuum relief isolation valves close on an ESF containment isolation signal, as well as on High containment radioactivity. There are two normally closed, self-actuated check valves inside containment that open on a negative differential pressure of 0.2 psi. To provide 100% capacity vacuum relief capability, either one of the two vacuum relief isolation valves and either one of the two check valves need to open.

The parallel vacuum relief isolation valves are interlocked with the 16 inch containment purge discharge isolation valve inside containment, VFS-PL-V009, which shares the containment penetration. The vacuum relief isolation valves are blocked from opening if VFS-PL-V009 is not closed. If VFS-PL-V009 is not closed, then the vacuum relief isolation valves will automatically close to direct VFS purge exhaust through the normal VFS discharge flow path. However, if vacuum relief actuation is required, the vacuum relief isolation valves actuation signal overrides the

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BASES

BACKGROUND (continued)

closing interlock with VFS-PL-V009 to allow the vacuum relief isolation valves to open ensuring that the vacuum relief protection actuates (Ref. 3).

APPLICABLE SAFETY ANALYSES

Design of the vacuum relief system involves calculating the effect of loss of ac power and a low outside ambient air temperature in combination with limited containment heating that reduces the atmospheric temperature (and hence pressure) inside containment (Ref. 1). Conservative assumptions are used for relevant parameters in the calculation; for example, maximum inside containment temperature, minimum outside air temperature, maximum humidity, and maximum heat transfer coefficients (Ref. 1). The resulting containment pressure versus time is calculated, including the effect of the opening of the vacuum relief valves when their negative pressure setpoint is reached. It is also assumed that one valve fails to open.

The containment was designed for an external pressure load equivalent to 1.7 psid. The excessive containment cooling events were analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was -0.2 psig. This resulted in a minimum pressure inside the containment less than the design load.

The applicable safety analyses results for the loss of ac power event bounds the analyses for the other external pressure load events described in the Bases for LCO 3.6.4, “Containment Pressure.”

The vacuum relief valves must also perform the containment isolation function during a containment high pressure event. For this reason, the system is designed to take the full containment positive design pressure and the environmental conditions (temperature, pressure, humidity, radiation, chemical attack, and the like) associated with the containment DBA.

The vacuum relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the maximum containment temperature initial condition and the minimum equipment required to accomplish the vacuum relief function following excessive containment cooling events (Ref. 1). This LCO also addresses the minimum equipment required to perform the containment isolation function of the vacuum relief valves.

The vacuum relief check valves are located in independent parallel paths inside containment and the vacuum relief isolation valves are located in independent parallel paths outside containment; they are connected by a

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BASES

LCO (continued)

common containment penetration. Therefore, two vacuum relief check valves and two vacuum relief isolation valves are required to be OPERABLE to ensure that at least one vacuum relief path is available with the failure to open of one check valve and/or one isolation valve and to ensure that there are two barriers for containment isolation. To be considered OPERABLE for opening, a vacuum relief check valve must be capable of opening on a differential pressure of less than or equal to 0.2 psi outside atmosphere-to-containment pressure. To be considered OPERABLE for closing, a vacuum relief check valve must close with a differential pressure less than or equal to 0 psi outside atmosphere-to-containment pressure. To be considered OPERABLE for opening, a vacuum relief isolation valve must be capable of opening on an ESF open signal and to be considered OPERABLE for closing, the isolation valve must be capable of closing on an ESF closure signal.

The containment inside to outside differential air temperature limit of ≤ 90°F ensures that the initial condition for the excessive cooling analysis is met. If the differential air temperature exceeds the limit, the containment vacuum relief capacity of one flow path may not be adequate to ensure the containment pressure meets the negative pressure design limit.

APPLICABILITY

In MODES 1 through 6, the potential exists for excessive containment cooling events to produce a negative containment pressure below the design limit. However, in MODE 5 or 6, a containment air flow path may be opened (LCO 3.6.7, “Containment Penetrations”), providing a vacuum relief path that is sufficient to preclude a negative containment pressure below the design limit.

Therefore, the vacuum relief valves are required to be OPERABLE for opening in MODES 1 through 4 and in MODES 5 and 6 without an open containment air flow path ≥ 6 inches in diameter. With a 6 inch diameter or equivalent containment air flow path, the vacuum relief function is not needed to mitigate a low pressure event.

The Applicability is modified by a Note that the vacuum relief valve OPERABILITY for closing is only required in MODES 1, 2, 3, and 4. In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the vacuum relief valve OPERABILITY for closing is only required in MODES 1, 2, 3, and 4 to prevent leakage of radioactive material from containment during positive pressure accident conditions. However, containment closure capability is required in MODES 5 and 6. The requirements for

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BASES

APPLICABILITY (continued)

containment isolation valves, including the vacuum relief valves, during MODES 5 and 6 are addressed in LCO 3.6.7, “Containment Penetrations.”

ACTIONS

The Actions are modified by a Note that directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, “Containment,” in the event that vacuum relief valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

A.1

With one vacuum relief check valve inoperable for opening, vacuum relief capability is reduced below that required to meet single failure criterion. The inoperable valve must be restored to OPERABLE for opening status within 72 hours. It is acceptable to enter Condition A and B concurrently. The vacuum relief function is maintained with at least one check valve and one isolation valve OPERABLE, which is assured provided Condition F or G is not entered for two vacuum relief check valves or two vacuum relief isolation valves inoperable for opening. The specified Completion Time is consistent with other LCOs for the loss of one train of a system required to mitigate the consequences of a LOCA or other DBA.

B.1

With one vacuum relief isolation valve inoperable for opening, vacuum relief capability is reduced below that required to meet single failure criterion. The inoperable valve must be restored to OPERABLE for opening status within 72 hours. Provided Condition F or G is not entered for two vacuum relief check valves or two vacuum relief isolation valves inoperable for opening, then the vacuum relief function is maintained with at least one check valve and one isolation valve (i.e., it is acceptable to enter Conditions A and B concurrently). The Completion Time period is consistent with other LCOs for the loss of one train of a system required to mitigate the consequences of a LOCA or other DBA.

C.1

With one or more vacuum relief check valves inoperable for closing or with one or more vacuum relief isolation valves inoperable for closing (including either two vacuum relief check valves inoperable for closing or two vacuum relief isolation valves inoperable for closing), provided Condition D is not entered, the penetration has not lost single failure isolation capability.

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BASES

ACTIONS (continued)

The Completion Time to restore the valve(s) to OPERABLE for closing status in 7 days has been shown to be acceptable based on probability risk assessment considering the importance of supporting containment OPERABILITY during MODES 1, 2, 3, 4, with the availability of one barrier for containment isolation, while also maintaining the capability for vacuum relief.

D.1

With one or more vacuum relief check valves and one or more vacuum relief isolation valves inoperable for closing, either both vacuum relief check valves or both vacuum relief isolation valves are to be restored to OPERABLE for closing status within 1 hour, to restore the capability to isolate the containment penetration. With one check valve and one isolation valve inoperable for closing, the containment isolation capability is lost and a release path remains open. To address this, either both vacuum relief check valves or both vacuum relief isolation valves need to be restored to OPERABLE for closing status to provide a containment isolation barrier. The 1 hour Completion Time is consistent with Action B of LCO 3.6.3, “Containment Isolation Valves.”

E.1 and E.2

If the containment inside to outside differential air temperature is > 90°F, then the differential air temperature shall be restored to within the limit within 8 hours. The 8-hour Completion Time is reasonable, considering that limit is based on a worst case condition and the time needed to reduce the containment temperature while controlling pressure within limits of LCO 3.6.4, Containment Pressure.

If the differential temperature cannot be restored, Required Action E.2 provides an alternate requirement. Reduction of the containment average temperature to ≤ 80°F provides an initial condition for excessive cooling events that ensures the vacuum relief system capacity is sufficient (Ref. 1).

F.1 and F.2

If the Required Action and associated Completion Time of Conditions A, B, C, D, or E are not met in MODE 1, 2, 3, or 4, or two vacuum relief check valves are inoperable for opening in MODE 1, 2, 3, or 4, or two vacuum relief isolation valves are inoperable for opening in MODE 1, 2, 3, or 4, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable,

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BASES

ACTIONS (continued)

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

G.1

If the Required Action and associated Completion Time of Condition A, B, or E is not met, or if two vacuum relief check valves are inoperable for opening, or if two vacuum relief isolation valves are inoperable for opening in MODE 5 or 6, action must be immediately initiated to restore compliance with the LCO and to continue this action until restoration is accomplished.

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Verification that the containment inside to outside differential air temperature is ≤ 90°F is required every 12 hours. The containment inside to outside differential air temperature is the difference between the outside ambient air temperature (measured by the site meteorological instrumentation or equivalent) and the inside containment average air temperature (measured using the same instrumentation as used for SR 3.6.5.1).

The Frequency is based on the normally stable containment average air temperature and the relatively small outside ambient air temperature changes within this time.

SR 3.6.9.2

Each vacuum relief isolation valve must be verified to be closed every 31 days. This SR assures that the vacuum relief isolation valves are closed as required, or if open, the valves are open for an allowable reason. The vacuum relief isolation valves are normally closed. The Frequency of 31 days is appropriate considering the valves should only be opened to relieve vacuum or manually opened to perform Surveillances and, therefore, the probability of the valve misalignment is low.

SR 3.6.9.2 is modified by two Notes. Note 1 states that SR 3.6.9.2 is not required to be met for vacuum relief valves that are open during Surveillances. Surveillances which open the vacuum relief isolation valves are performed infrequently and the valves will only be open for a limited period of time. Note 2 states that SR 3.6.9.2 is not required to be met for vacuum relief valves performing their vacuum relief function. If vacuum relief is required, the valves open to perform their safety function

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BASES

SURVEILLANCE REQUIREMENTS (continued)

to maintain the integrity of the containment vessel. Air will flow into containment from the atmosphere due to the differential pressure, which will prevent radiological release from containment during vacuum relief.

SR 3.6.9.3

Each vacuum relief check valve and each vacuum relief isolation valve is tested to be OPERABLE for opening and OPERABLE for closing. This SR cites the Inservice Testing Program, which establishes the requirement that inservice testing of the ASME Code Class 1, 2, and 3 valves shall be performed in accordance with the ASME OM Code (Ref. 2). Therefore, SR Frequency is governed by the Inservice Testing Program.

SR 3.6.9.4

This SR ensures that each vacuum relief isolation valve will actuate to the open position on an actual or simulated actuation signal and will actuate to the closed position on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operations.

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BASES

REFERENCES

1. FSAR Subsection 6.2.1.1.4, “External Pressure Analysis.”
2. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”
3. FSAR Subsection 9.4.7, “Containment Air Filtration System.”

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B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available. The MSSVs also are containment isolation valves.

Six MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in Reference 1. The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, as shown in Table 3.7.1-2 of the specification, so that only the needed valves actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open the valves following a turbine-reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operating occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, are those characterized as decreased heat removal events, which are presented in FSAR Section 15.2 (Ref. 3). Of these, the full power turbine trip without turbine bypass is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on High 2 pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure.

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APPLICABLE SAFETY ANALYSES (continued)

All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux – High Positive Rate setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The FSAR Section 15.4.2 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The FSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Overpower ΔT reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Overpower ΔT setpoints to an appropriate value.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition, the MSSVs are containment isolation valves and support the assumptions related to minimizing the loss of inventory and establishing the containment boundary during major accidents. Therefore, the safety analysis of any event requiring isolation of containment is applicable to the MSSVs.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analysis requires six MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% of RTP. A MSSV will be considered inoperable if it fails to open in the event of a pressure excursion in excess of the setpoint. The LCO requires that six MSSVs be OPERABLE in compliance with Reference 2. Operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 of the specification and Required Action A.1.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings specified in Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODE 1, 2, 3, or 4, six MSSVs per steam generator are required to be OPERABLE. However, as stated in the Note to the Applicability, the MSSVs are not required to be OPERABLE for opening in MODE 4 when the Reactor Coolant System (RCS) is being cooled by the Normal Residual Heat Removal System (RNS).

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BASES

APPLICABILITY (continued)

In MODE 4 with the RNS in service, there are no credible transients requiring the MSSVs to be opened. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized. Thus, there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

With one or both steam generators with one or more MSSVs inoperable for opening, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all six MSSVs OPERABLE for opening for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described below and in the attachment to Reference 6, with an appropriate allowance for calorimetric power uncertainty.

To determine the maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs, the governing heat transfer relationship is the equation $q = \dot{m}\Delta h$, where q is the heat input from the primary side, $\dot{m}$ is the mass flow rate of the steam, and Δh is the increase in enthalpy that occurs in converting the secondary side water to steam. If it is conservatively assumed that the secondary side water is all saturated liquid (i.e., no subcooled feedwater), then the Δh is the heat of vaporization (h_{fg}) at the steam relief pressure. The following equation is used to determine the maximum allowable power level for continued operation with inoperable MSSVs.

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BASES

ACTIONS (continued)

Maximum NSSS Power $\leq$ (100/Q) (W_s h_{fg} N) / K

where:

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), MWt

K = Conversion factor, 947.82 (Btu/sec)/MWt

w_s = Minimum total steam flow rate capability of the OPERABLE MSSVs on any one steam generator at the highest OPERABLE MSSV opening pressure, including tolerance and accumulation as appropriate, lbm/sec

h_{fg} = Heat of vaporization at the highest MSSV opening pressure, including tolerance and accumulation as appropriate, Btu/lbm

N = Number of steam generators in the plant

To determine the Table 3.7.1-1 Maximum Allowable Power, the Maximum NSSS Power calculated using the equation above is reduced by 9% RTP to account for Overpower ΔT trip channel uncertainties.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

B.1

With one or both steam generators with one or more MSSVs inoperable for closing, the inoperable MSSV must be restored to OPERABLE status within 72 hours. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4.

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BASES

ACTIONS (continued)

C.1 and C.2

If any Required Action and associated Completion Time of Condition A is not met, or if one or both steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, with RCS cooling provided by the RNS, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1 and D.2

If the Required Action and associated Completion Time of Condition B is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The safety and relief valve tests are required to be performed in accordance with ASME OM Code (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Set pressure determination (lift setting);

d. Compliance with owner’s seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The ANSI/ASME standard requires that all valves be tested every 5 years and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ±1% setpoint tolerance for OPERABILITY, and the valves are reset to remain within ±1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR Chapter 10, “Steam and Power Conversion Systems Description.”
2. ASME Boiler and Pressure Vessel Code, Section III, Article NC-7000, “Overpressure Protection,” Class 2 Components.
3. FSAR Section 15.2, “Decreased Heat Removal by Secondary System.”
4. ASME OM Code, Appendix I, “Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants.”
5. ASME OM Code, “Code for Operation and Maintenance of Nuclear Power Plants.”
6. NRC Information Notice 94-60, “Potential Overpressurization of the Main Steam System,” August 22, 1994.

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B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Line Flow Path Isolation Valves

BASES

BACKGROUND

Each main steam line has one MSIV (which is safety related) to isolate steam flow from the secondary side of the steam generators, which may be required following a high energy line break.

One MSIV is located in each main steam line outside containment. The MSIVs are downstream from the main steam safety valves (MSSVs). Downstream of the MSIVs and the pressure equalization header, steam in each of the two main steam line flow paths enters the high pressure turbine through four stop valves and four governing control valves – two parallel stop valves and two associated control valves in each main steam line flow path. Closing the MSIVs isolates each steam generator from the other and isolates the turbine bypass system, and other steam supplies from the steam generator.

The MSIVs, turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves close on a main steam isolation signal generated by either Low 2 steam line pressure, High 2 containment pressure, Low 2 T_{cold}, or high negative steam pressure rate. The turbine stop and control valves close on a turbine trip. The MSIVs close on an actuation signal from either of two 1E power divisions. These valves also fail to a closed position upon loss of actuating hydraulic power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in FSAR Section 10.3 (Ref. 1). Descriptions for the turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves are found in FSAR Section 10.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in FSAR Section 6.2 (Ref. 3). It is also affected by the accident analysis of the SLB events presented in FSAR Section 15.1 (Ref. 4). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Design basis events of concern for containment analysis are SLB inside containment with the failure of the associated MSIV to close, or a main feedline break with the associated failure of a feedline isolation or control valve to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers, downstream from the other MSIV, contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Core Makeup Tanks (CMTs).

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes consideration of scenarios with offsite power available, and with a loss of offsite power. With offsite power available, the reactor coolant pumps continue to circulate coolant for a longer period through the steam generators, maximizing the Reactor Coolant System cooldown. The reactor protection system includes a safety related signal that initiates the coastdown of the reactor coolant pumps early in the large SLB transient. Therefore, there is very little difference in the predicted departure from nucleate boiling ratio between cases with and without offsite power. Significant single failures considered include failure of an MSIV to close.

The non-safety related turbine stop or control valves, in combination with the turbine bypass, and moisture separator reheater 2nd stage steam isolation valves, are assumed as a backup to isolate the steam flow path given a single failure of an MSIV. The safety analyses do not differentiate between the availability of the turbine stop valves or the turbine control valves in each main steam line flow path. These valves, along with the turbine bypass, and moisture separator reheater 2nd stage steam isolation valves are considered as alternate downstream valves.

The MSIVs serve a safety related function and remain open during power operation. These valves operate under the following situations:

a. High energy line break inside containment. In order to maximize the mass and energy release into containment, the analysis assumes

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BASES

APPLICABLE SAFETY ANALYSES (continued)

that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from both steam generators until the unaffected loop MSIV closes. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIV in the unaffected loop. Closure of the MSIV isolates the break from the unaffected steam generator.

b. A break outside of containment, and upstream or downstream from the MSIVs. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled Reactor Coolant System (RCS) cooldown and positive reactivity addition. Closure of the MSIVs or alternate downstream valves isolates the break, and limits the blowdown to a single steam generator.

c. Following a steam generator tube rupture. Closure of the MSIVs isolates the ruptured steam generator to minimize radiological releases.

d. Other events such as a feedwater line break; however, these events are less limiting so far as MSIV OPERABILITY is concerned.

In addition, the MSIV bypass valves and main steam line drain valves are containment isolation valves and support the assumptions related to minimizing the loss of inventory and establishing the containment boundary during major accidents. Therefore, the safety analysis of any event requiring isolation of containment is applicable to the MSIV bypass valves and main steam line drain isolation valves.

The MSIVs, the alternate downstream valves, the MSIV bypass valves, and the main steam line drain valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Following an SLB and main steam isolation signal, the analyses assume continued steam loss through the main steam line drain lines, turbine gland seal system, and the main steam to auxiliary steam header which supplies the auxiliary steam line to the deaerator. Since these valves are not assumed for steam isolation to mitigate an SLB, they do not satisfy the 10 CFR 50.36(c)(2)(ii) criteria; however, main steam line drain isolation valves satisfy the 10 CFR 50.36(c)(2)(ii) criteria for containment isolation as discussed above.

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BASES

LCO

This LCO requires the minimum combination of valves for isolation of each main steam line flow path to be OPERABLE. This requires that one MSIV in each of the two main steam line flow paths be OPERABLE and requires MSIV bypass valves and main steam line drain valves to be OPERABLE. The MSIVs are considered OPERABLE when their isolation times are within limits, and they close on an MSIV actuation signal.

This LCO requires the minimum combination of turbine stop and control valves required for alternate downstream isolation of each main steam line flow path into the high pressure turbine to be OPERABLE. This LCO requires two turbine stop valves or the two associated turbine control valves in main steam line flow path “A” and two turbine stop valves or the two associated turbine control valves in main steam line flow path “B” be OPERABLE. A valve is considered OPERABLE when its closure time is within the safety analysis isolation time limit and the valve closes on an actuation signal. Example combinations of turbine stop and associated turbine control valves required by this LCO are as follows:

- If V001A and/or V003A (i.e., one or both turbine stop valve(s) in main steam line flow path “A”) are/is not OPERABLE, then associated valves V002A and V004A (i.e., both turbine control valves in main steam line flow path “A”) are required to be OPERABLE.

- If V002B and/or V004B (i.e., one or both turbine control valves in main steam line flow path “B”) are/is not OPERABLE, then associated valves V001B and V003B (i.e., both turbine stop valves in main steam line flow path “B”) are required to be OPERABLE.

This LCO also requires six turbine bypass valves and two moisture separator reheater 2nd stage steam isolation valves be OPERABLE. A valve is considered OPERABLE when its isolation time is within the safety analysis isolation time limit and it closes on an MSIV actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 50.34 limits or the NRC staff approved licensing basis.

This LCO provides assurance that the design and performance of the alternate downstream valves are compatible with the accident conditions for which they are called upon to function (Ref. 5).

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BASES

APPLICABILITY

The MSIVs, MSIV bypass valves, main steam line drain valves, turbine stop or associated turbine control valves, turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves must be OPERABLE in MODES 1, 2, 3, and 4, when there is significant mass and energy in the RCS and steam generators and where a DBA could cause a release of radioactive material to containment.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs and alternate downstream valves are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the valves can be made with the plant hot. The 8 hour Completion Time is reasonable considering the low probability of an accident occurring during this time period that would require a closure of these valves. With a single MSIV inoperable, the safety function, isolation of the main steam line flow path, is provided by the OPERABLE alternate downstream valves, but cannot accommodate a single failure. The assumptions and criteria of the accident analyses are preserved by the ability to automatically isolate the steam flow path.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides a positive means for containment isolation.

B.1

With any number of the turbine stop valves and the associated turbine control valves, turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 1, action must be taken to restore the inoperable valve(s) to OPERABLE status within 72 hours. Some repairs to the valves can be made with the plant hot. The 72 hour Completion Time is reasonable considering the low probability of an accident occurring during this time period that would require a closure of these valves. With the backup isolation valves inoperable, the safety function, isolation of the main steam line flow path, is provided by the remaining OPERABLE valves, but cannot accommodate a single failure. The assumptions and criteria of the accident analyses are preserved by the ability to automatically isolate the steam flow path.

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BASES

ACTIONS (continued)

C.1

With two MSIVs inoperable in MODE 1 or one MSIV and an alternate downstream valve inoperable or if the valves cannot be restored to OPERABLE status in accordance with Required Action A.1 or B.1, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition D would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner and without challenging unit systems.

D.1 and D.2

Condition D is modified by a Note indicating that a separate Condition entry is allowed for each main steam line flow path.

With one or two MSIVs inoperable or any number of turbine stop valves and the associated turbine control valves, turbine bypass, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 2, 3, or 4, the inoperable valve(s) may either be restored to OPERABLE status or the affected main steam line flow path isolated. When isolated, the main steam line flow path is in the condition required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A, and conservative considering the reduced energy in the steam generators in MODES 2, 3, and 4.

For inoperable main steam line flow path isolation valves that cannot be restored to OPERABLE status within the specified Completion Time but whose affected flow paths are isolated, these isolated flow paths must be verified to be continually isolated on a periodic basis. This is necessary to ensure that the assumptions in the safety analyses remain valid. The 7 day Completion Time is based on engineering judgment, and is considered reasonable in view of MSIV status indications available in the control room and other administrative controls which ensure that these valves will continue to be closed.

E.1 and E.2

With one or more MSIV bypass or main steam line drain valves inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be

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BASES

ACTIONS (continued)

adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action E.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event that the affected penetration is isolated in accordance with Required Action E.1, the affected penetration must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition E is modified by a Note indicating that separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The Required Actions are modified by a Note allowing containment penetration flow paths to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated.

Required Action E.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas, and allows these devices to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components

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BASES

ACTIONS (continued)

is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

F.1, F.2, and F.3

If the main steam line flow path isolation valves cannot be restored to OPERABLE status or affected flow paths isolated within the associated Completion Times of Condition D or E, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, in MODE 4 with normal residual heat removal system in service within 24 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5.0 seconds, on an actual or simulated actuation signal. The MSIV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power.

The Frequency is in accordance with the Inservice Testing Program.

The SR is modified by a note which states that the SR is only required to be performed prior to entry into MODE 2. This allows the option to perform testing in MODES 3 or 4.

SR 3.7.2.2

This SR verifies that the turbine stop, turbine control, turbine bypass, and moisture separator reheater 2nd stage steam isolation valves' closure time is ≤ 5.0 seconds, on an actual or simulated actuation signal. These alternate downstream isolation valves must meet the MSIV isolation time

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BASES

SURVEILLANCE REQUIREMENTS (continued)

assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The alternate downstream valves should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power.

The Frequency is in accordance with the Inservice Testing Program.

The SR is modified by a note which states that the SR is only required to be performed prior to entry into MODE 2. This allows the option to perform testing in MODES 3 or 4.

SR 3.7.2.3

Verifying that the isolation time of each MSIV bypass and steam line drain valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation times are specified in FSAR Section 6.2.3 (Ref. 6) and Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.7.2.4

This SR ensures that each MSIV bypass and steam line drain valve will actuate to its isolation position on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. The 24 month Frequency is based on the need to perform this Surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

REFERENCES

1. FSAR Section 10.3, “Main Steam System.”
2. FSAR Section 10.4, “Other Features of Steam and Power Conversion Systems.”
3. FSAR Section 6.2.1, “Containment Functional Design.”

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BASES

REFERENCES (continued)

4. FSAR Section 15.1, “Increase in Heat Removal by Secondary System.”

5. NUREG-138, Issue 1, “Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director NRR to NRR Staff.”

6. FSAR Section 6.2, “Containment Systems.”

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B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Control Valves (MFCVs), and Main Feedwater (MFW) Pump Breakers

BASES

BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break. The safety related function of the MFCVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following a high energy line break. Closure of the MFIVs or MFCVs terminates flow to the steam generators, terminating the event for feedwater line breaks occurring upstream of the MFIVs or MFCVs. The MFW pumps also trip to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the steam generators. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure and the MFW pump trip. Closure of the MFIVs or MFCVs, and trip of the MFW pumps, effectively terminates the addition of main feedwater to an affected steam generator, limiting the mass and energy release for steam or feedwater line breaks inside containment, and reducing the cooldown effects for steam line breaks (SLBs).

The MFIVs or MFCVs isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves isolate and the pumps trip to limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of startup feedwater (SFW) to the intact loops of the steam generator.

One MFIV and one MFCV are located on each MFW line, outside but close to containment. The MFIVs and MFCVs are located in the MFW line and are independent of the delivery of the MFW or SFW via the SFW line which is separately connected and isolated from the steam generator. This configuration permits MFW or SFW to be supplied to the steam generators following MFIV or MFCV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases following either an SLB or FWLB.

The three MFW pumps operate in parallel each constituting a MFW pump flow path. The MFW pumps discharge to a common header, which then feeds two MFW flow paths each to one steam generator.

The MFIVs and MFCVs close, and the MFW pump breaker trips, on receipt of engineered safeguards feedwater isolation signal generated from any of the following conditions:

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BASES

BACKGROUND (continued)

- Automatic or manual safeguards actuation “S” signal;
- High 3 steam generator level;
- Low-2 T_{avg} signal coincident with reactor trip (P-4) (for the MFIVs and MFW pumps);
- Low T_{avg} signal coincident with reactor trip (P-4) (for the MFCVs); and
- Manual actuation

Each valve and each pump may be actuated manually. In addition to the MFIVs and the MFCVs, a check valve is available outside containment to isolate the feedwater line penetrating containment. In the event of feedwater line depressurization due to pump trip on line break, the check valve provides rapid backup isolation of the steam generators limiting the inventory loss. A description of the MFIVs and MFCVs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs, MFCVs, and MFW pump trip is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large Feedwater Line Break (FWLB). Closure of the MFIVs (or MFCVs), and trip of the MFW pump, may also be relied on to mitigate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level – High 3 signal.

Failure of an MFIV (or MFCV) to close, or the MFW pumps to trip, following an SLB or FWLB, can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

In addition, the MFIVs are containment isolation valves and support the assumptions related to minimizing the loss of inventory and establishing the containment boundary during major accidents. Therefore, the safety analysis of any event requiring isolation of containment is applicable to the MFIVs.

The MFIVs, MFCVs, and MFW pump breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

This LCO ensures that the MFIVs and the MFCVs will isolate the main feedwater system to the secondary side of the steam generators and ensures that the MFW pumps will trip.

This LCO requires that the one isolation valve and one control valve on each feedwater line and each MFW pump breaker be OPERABLE. These valves are considered OPERABLE when their isolation times are within limits and they close on an isolation actuation signal. For a MFW pump breaker to be considered OPERABLE, it must trip on an actuation signal or be in the tripped condition.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A main feedwater isolation signal on High 3 steam generator level is relied on to terminate an excess feedwater flow event, and therefore failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MFIVs, MFCVs, and MFW pump breakers must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and the steam generators. This ensures that, in the event of a high energy line break, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, 3, and 4, these valves and pump trips are required to be OPERABLE to limit the amount of available fluid that could be added to the containment in the case of a secondary system pipe break inside containment and where a DBA could cause a release of radioactive material to containment.

In MODES 5 and 6 steam generator energy is low. Therefore, the MFIVs and the MFCVs are normally closed, and MFW pumps are not running, since MFW is not required.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate condition entry is allowed for each feedwater flow path, where there are two MFW flow paths with an MFIV and MFCV in series and three MFW pump flow paths with one MFW pump in each flow path.

A.1 and A.2

With one or both feedwater flow paths with an MFIV or MFCV inoperable (one or two MFIVs, or one or two MFCVs inoperable), the affected flow path must be isolated in 72 hours. When these flow paths are isolated, they are performing their required safety function.

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ACTIONS (continued)

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event that would require isolation of the main feedwater flow paths occurring during this period.

For inoperable MFIVs and MFCVs that cannot be restored to OPERABLE status within the specified Completion Time but whose affected flow path is isolated, the flow paths must be verified on a periodic basis to be isolated. This is necessary to ensure that the assumptions in the safety analyses remain valid. The 7 day Completion Time is reasonable based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these flow paths are isolated.

B.1

With one or both feedwater flow paths with associated MFIV and MFCV inoperable (two inoperable valves in the same flow path), there may be no redundant system to operate automatically to perform the required safety function. Under these conditions, one valve in the affected flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the situation in which at least one valve in the affected flow path is performing the required safety function. The 8 hour Completion Time is a reasonable amount of time to complete the actions required to close the MFIV, or MFCV, which includes performing a controlled plant shutdown. The Completion Time is reasonable based on operating experience to reach MODE 2 with the MFIV or MFCV closed, from full-power conditions in an orderly manner and without challenging plant systems.

C.1

With one or more feedwater flow paths with a MFW pump breaker inoperable, the affected MFW pump breaker must be restored to OPERABLE status within 6 hours. The Completion Time takes into account the redundancy afforded by the MFIV and MFCV and the low probability of an event that would require tripping the MFW pump during this period.

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BASES

ACTIONS (continued)

D.1, D.2, and D.3

If the affected flow path cannot be isolated or the MFIVs, MFCVs, or MFW pump breaker cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, in MODE 4 with the normal residual heat removal system in service within 24 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV and MFCV is ≤ 5.0 seconds, on an actual or simulated actuation signal. The MFIV and MFCV isolation times are assumed in the accident and containment analyses. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power.

The Frequency is in accordance with the Inservice Testing Program.

The SR is modified by a note which states that the SR is only required to be performed prior to entry into MODE 2. This allows the option to perform testing in MODES 3 or 4.

SR 3.7.3.2

This SR verifies that each main feedwater pump breaker trip open in response to an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the pump breaker.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The SR excludes main feedwater pump breakers that are maintained in the tripped position since a tripped open MFW pump breaker is already performing the safety function. Restoration of a breaker to the non-actuated position requires verification that the SR has been met within its required Frequency.

The 24 month Frequency is based on the need to perform this Surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

REFERENCES

1. FSAR Section 10.4.7, “Condensate and Feedwater System.”

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B 3.7 PLANT SYSTEMS

B 3.7.4 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube LEAKAGE from the Reactor Coolant System (RCS). Therefore, fission product isotopes, as well as activated corrosion products in lesser amounts, may be found in the secondary coolant. While fission products present in the primary coolant, as well as activated corrosion products, enter the secondary coolant system due to the primary to secondary LEAKAGE, only the iodines are of a significant concern relative to airborne release of activity in the event of an accident or abnormal occurrence (radioactive noble gases that enter the secondary side are not retained in the coolant but are released to the environment via the condenser air removal system throughout normal operation).

The limit on secondary coolant radioactive iodines minimizes releases to the environment due to anticipated operational occurrences or postulated accidents.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (SLB) as discussed in FSAR Chapter 15 (Ref. 1) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.01 µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of a postulated SLB are within the acceptance criteria in SRP Section 15.0.1, and within the exposure guideline values of 10 CFR Part 50.34.

Secondary specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit of the secondary coolant is required to be ≤ 0.01 µCi/gm DOSE EQUIVALENT I-131 to maintain the validity of the analyses reported in FSAR Chapter 15 (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

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BASES

APPLICABILITY

In MODES 1, 2, 3, and 4 the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or leakage. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. FSAR, Chapter 15, “Accident Analyses.”

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B 3.7 PLANT SYSTEMS

B 3.7.5 Spent Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are at their maximum capacity. The water also provides shielding during the movement of spent fuel, and a large capacity heat sink in the event the spent fuel pool cooling system is unavailable.

A general description of the spent fuel pool design is given in FSAR Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling System is given in FSAR Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in FSAR Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The design basis radiological consequences resulting from a postulated fuel handling accident are within the dose values provided in FSAR Section 15.7.4 (Ref. 3).

According to Reference 3 there is 23 ft of water between the damaged fuel bundle and the fuel pool surface during a fuel handling accident. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. This slight reduction in water depth does not adversely affect the margin of conservatism associated with the assumed pool scrubbing factor of 200 for iodine.

In addition to mitigation of the effects of a fuel handling accident, the required minimum water level in the spent fuel pool and connected fuel transfer canal provides a large capacity heat sink for spent fuel pool cooling in the event the spent fuel pool cooling system is unavailable.

The Spent Fuel Pool Water Level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The spent fuel pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident

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BASES

LCO (continued)

analysis (Ref. 3) and loss of spent fuel pool cooling. As such, it is the minimum required for fuel storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies when irradiated fuel assemblies are stored in the spent fuel pool. Irradiated fuel assemblies generate decay heat and cooling of the irradiated fuel assemblies would be negatively impacted by the loss of spent fuel pool cooling.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool cooling requirements apply when irradiated fuel assemblies are stored in the spent fuel pool, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. Spent fuel pool cooling requirements are independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1

When the initial conditions of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assemblies shall be suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 4, 5, or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2 and 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

A.2

If the water level in the spent fuel pool is < 23 ft, the heat capacity of the spent fuel pool will be less than that assumed in the event of a loss of spent fuel pool cooling. In this case, action must be initiated within 1 hour to restore the water level in the spent fuel pool to ≥ 23 ft above the top of the irradiated fuel assemblies. Initiation of this action requires that the action be continued until a water level of ≥ 23 ft is attained.

The Completion Time of 1 hour assures prompt action to compensate for a degraded condition.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR verifies sufficient spent fuel pool water is available in the event of a fuel handling accident or loss of spent fuel pool cooling. The water level in the spent fuel pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.4.1.

REFERENCES

1. FSAR Section 9.1.2, “Spent Fuel Storage.”
2. FSAR Section 9.1.3, “Spent Fuel Pool Cooling System.”
3. FSAR Section 15.7.4, “Fuel Handling Accident.”
4. Regulatory Guide 1.183 Rev. 0, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

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B 3.7 PLANT SYSTEMS

B 3.7.6 Main Control Room Emergency Habitability System (VES)

BASES

BACKGROUND

The Main Control Room Emergency Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available and provide protection from the release of radioactivity. If AC power is lost for greater than 10 minutes, or Low main control room differential pressure is sensed for greater than 10 minutes, or a High-2 iodine or particulate Main Control Room Envelope (MCRE) radiation signal is received, the VES is actuated. The MCRE radioactivity is measured by detectors in the Main Control Room (MCR) supply air duct, downstream of the filtration units. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air (Ref. 5) for the MCRE occupants; 2) to provide forced ventilation to maintain the MCRE at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; 3) provide passive filtration to filter contaminated air in the MCRE; and 4) to limit the temperature increase of the MCRE equipment and facilities that must remain functional during an accident, via de-energizing (load shedding) nonessential, non-safety main control room (MCR) electrical equipment (e.g., wall panel information system displays, office equipment, water heater, kitchen appliances, and non-emergency lighting) and the heat absorption of passive heat sinks.

The VES consists of compressed air storage tanks, main (with two parallel automatically actuated flow paths) and an alternate (a manually actuated flow path) air delivery flow paths, an eductor, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), associated valves or dampers, piping, and instrumentation. The tanks contain enough breathable air to supply the required air flow to the MCRE for at least 72 hours. The VES system is designed to maintain CO_2 concentration less than 0.5% for up to 11 MCRE occupants.

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BASES

BACKGROUND (continued)

There are 32 compressed air storage tanks arranged in 4 banks. Each bank nominally provides at least 81,893.5 scf of compressed air to support the minimum required 327,574 scf. With the compressed air equivalent of one bank of tanks unavailable, verifying > 245,680 scf of compressed air ensures the equivalent of three banks remains available to supply air to the MCRE for 54 hours (75% of the required 72 hours). The VBS supplemental filtration mode or MCRE ancillary fans are both capable of maintaining the habitability of the MCRE after 54 hours.

The MCRE is the area within the confines of the MCRE boundary that contains the spaces that control room operators inhabit to control the unit during normal and accident conditions. This area encompasses the main control area, operations work area, operational break room, shift supervisor's office, kitchen, and toilet facilities (Ref. 1). The MCRE is protected during normal operation, natural events, and accident conditions. The MCRE boundary is the combination of walls, floor, roof, electrical and mechanical penetrations, and access doors. The OPERABILITY of the MCRE boundary must be maintained to ensure that the inleakage of unfiltered air into the MCRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to MCRE occupants. The MCRE and its boundary are defined in the Main Control Room Envelope Habitability Program.

Heat sources inside the MCRE include operator workstations, emergency lighting and occupants. During normal operation, temperatures in the main control room, instrumentation and control rooms, dc equipment rooms, Class 1E electrical penetration rooms, and some adjacent rooms are maintained within a specified range by the VBS. As described in FSAR Section 9.4.1.2, the VBS consists of independent subsystems, including the main control room / control support area HVAC subsystem and the Class 1E Electrical room HVAC subsystem. The Class 1E Electrical room HVAC subsystem is further divided into two independent subsystems, with one serving the Division A & C Class 1E electrical division rooms and the other serving the Division B & D Class 1E electrical division rooms. Each independent subsystem serves its associated rooms with two redundant, 100 percent capacity equipment trains, maintaining temperatures within the specified range.

To support OPERABILITY of the VES, passive heat sink air temperatures are maintained by VBS in required dc Equipment rooms and required I&C rooms. Certain required room-pairs (i.e., 12201/12301, 12203/12302, 12205/12305, and 12207/12304) require the average temperature of the combined room-pair to be ≤ 85°F, as monitored by temperature elements located in the shared return air ducting. Other required individual rooms

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BASES

BACKGROUND (continued)

(i.e., 12202, 12204, 12300, 12313, 12412, and 12501) are each required to be ≤ 85°F. Additionally, a maximum air temperature limit of ≤ 75°F is also placed on the MCRE. The passive heat sinks limit the temperature rise inside each room and the MCRE during the 72 hour period following VES actuation.

Access corridors, stairwells, rooms separated by an air gap, and other rooms without significant heat loads are not monitored because these areas do not contain significant heat sources and their temperatures are assumed to match the connected spaces. These unmonitored rooms are identified as: 12211, 12311, 12400, 12405, 12411, 21480, 40400, and Stairwells.

Initial temperatures assumed for remaining rooms are conservatively selected to match the initial 115°F outdoor ambient (12212, 12213, 12306, 12312, 12404, and 12406) or do not have an appreciable impact on the analyses. These unmonitored rooms are identified as: 12212, 12213, 12306, 12312, 12404, 12406, 12504, 12505, 12506, and Level 1 rooms.

Nonessential, non-safety MCR heat loads are de-energized by the Protection and Safety Monitoring System (PMS) VES actuation signal, which is generated by the “Main Control Room Isolation, Air Supply Initiation and Electrical Load De-energization” ESF actuation signal, to maintain the MCRE within habitable limits for 72 hours.

Upon receipt of a “Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization” ESF actuation signal, PMS Divisions A and C energize associated redundant relays in each of the two safety-related electrical panels (VES-EP-01 and VES-EP-02). Energizing one set of relays in each panel disconnects non-safety related electrical power to the non-safety electrical loads in the MCRE. Energizing just one set of relays in one panel de-energizes the non-safety loads associated only with that panel.

De-energized non-safety loads are separated into stage 1 and stage 2 to maximize the availability of the non-safety related wall panel information system which is de-energized with stage 2 loads. Timers and associated relays, which actuate to de-energize the stage 1 and stage 2 non-safety loads, are internal to each safety-related load shed panel. Stage 1 loads are de-energized by both panels immediately after the timers in each panel receive the PMS “Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization” ESF actuation signal. Stage 2 loads are de-energized by both panels within 180 minutes after the timers in each panel receive the PMS “Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization” ESF actuation signal.

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BASES

BACKGROUND (continued)

OPERABILITY of two redundant divisions of MCR Class 1E load shed relays and timers located in two safety-related panels is required to meet the single failure criterion. Each panel contains redundant load shed relays and timers actuated by the two PMS divisions such that actuation of either division deenergizes the specified loads associated with both panels.

In the unlikely event that power to the VBS is unavailable for more than 72 hours, MCRE habitability is maintained by operating one of the two MCRE ancillary fans to supply outside air to the MCRE.

The compressed air storage tanks are initially filled to contain greater than 327,574 scf of compressed air. The compressed air storage tanks, the tank pressure, and the room temperature are monitored to confirm that the required volume of breathable air is stored. During operation of the VES, a self-contained pressure regulating valve maintains a constant downstream pressure regardless of the upstream pressure. The eductor which operates with an integrated inlet flow metering orifice, or the orifice in the VES bypass line downstream of the regulating valve, is used to control the air flow rate into the MCRE. The MCRE is maintained at a 1/8 inch water gauge positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas. The VES operation in maintaining the MCRE habitable is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The compressed air storage tanks are sized such that the set of tanks has a combined capacity that provides at least 72 hours of VES operation.

Operation of the VES is automatically initiated by any of the following safety related signals:

- Main Control Room Air Supply Iodine or Particulate Radiation - High-2;
- Loss of AC power for more than 10 minutes; or
- Main Control Room Differential Pressure - Low (for greater than 10 minutes).

In the event that a High 1 radioactivity setpoint value is reached, the non-safety VBS re-aligns to supplemental filtration mode, providing MCRE pressurization, cooling, and filtration. Upon High-2 particulate or iodine radioactivity setpoint, a safety related signal is generated to isolate the MCRE and to initiate air flow from the VES storage tanks. Isolation of

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BASES

APPLICABLE SAFETY ANALYSES (continued)

the MCRE consists of closing safety related valves in the lines that penetrate the MCRE pressure boundary. Valves in the VBS supply and exhaust ducts, and the Sanitary Drainage System (SDS) vent lines are automatically isolated. VES air flow is initiated by a safety related signal which opens the isolation valves in the VES supply lines.

The VES provides protection from smoke and hazardous chemicals to the MCRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the MCRE following a hazardous chemical release (Ref. 1). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the MCRE occupants to control the reactor either from the control room or from the remote shutdown room (Ref. 2).

The VES functions to mitigate a DBA or transient that either assumes the failure of or challenges the integrity of the fission product barrier.

The VES satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The VES limits the MCRE temperature rise and maintains the MCRE at a positive pressure relative to the surrounding environment.

The main air delivery flow path, with two parallel automatically actuated flow paths, and the alternate manually actuated air delivery flow paths are required to be OPERABLE for opening to ensure that at least one is available, assuming a single failure. The three air delivery flow paths (VES-PL-V002A with VES-PL-V005A, VES-PL-V002A with VES-PL-V005B, and VES-PL-V002B with VES-PL-V001) comprise the air delivery function.

The VES pressure relief isolation valves and pressure relief dampers are required to be OPERABLE for opening to ensure that adequate vent area is available to mitigate MCRE overpressurization. The two pressure relief flow paths (VES-PL-V022A with VES-MD-D001A and VES-PL-V022B with VES-MD-D001B) comprise the pressure relief function. Each flow path is also required to be closed when not relieving pressure to maintain the MCRE boundary and is provided with redundant isolation devices to ensure that at least one is available assuming a single failure.

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BASES

LCO (continued)

The VES eductor bypass isolation valve (VES-PL-V046) is required to be OPERABLE for opening to ensure continued delivery of breathable air in the event a passive failure (e.g., a blockage) disables the passive filtration flow path after 24 hours as assumed in the safety analysis assumptions (Ref. 1) for modeling the doses to the main control room personnel. This flow path comprises the eductor bypass function.

The VES MCR air filtration line balancing dampers (VES-MD-D002 and -D003) are required to be OPERABLE for opening for flow distribution in the ducting for delivery of recirculated air throughout the MCR boundary. The dampers split flow between the Main Control Area and the Shift Managers Office; however, single failure analysis (Ref. 1) demonstrates acceptable VES performance in the event one of these dampers fails. The air delivery ducting and dampers comprise the distribution function.

The VBS supply flow path isolation valves (VBS-PL-V186 and -V187) and return flow path isolation valves (VBS-PL-V188 and -V189), as well as toilet exhaust flow path isolation valves (VBS-PL-V190 and -V191) are required to be OPERABLE to isolate the MCRE. Each flow path is provided with redundant isolation valves to ensure that at least one is available assuming a single failure.

The MCR SDS vent line isolation valves (SDS-PL-V001 and -V002) are required to be OPERABLE to isolate the MCR envelope from the surrounding areas and outside environment. The vent line flow path is provided with redundant isolation valves to ensure that at least one is available assuming a single failure.

The VES OPERABILITY requires that the individual components necessary to deliver a supply of breathable air to the MCRE are OPERABLE. Each tank must be aligned and each bank must be aligned to be considered OPERABLE and the combined volume of compressed air of the aligned storage tanks must be > 327,574 scf, which is equivalent to each of four banks providing 81,893.5 scf.

In addition, the MCRE pressure boundary must be maintained, including the integrity of the walls, floors, ceilings, electrical and mechanical penetrations, and access doors. The MCRE pressure boundary includes the Potable Water System (PWS) and SDS running (piping drain) traps, which retain a fluid level sufficient to maintain a seal preventing gas flow through the piping. The MCRE pressure boundary also includes the Waste Water System (WWS) drain line, which is isolated by a normally closed isolation valve.

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BASES

LCO (continued)

In order for the VES to be considered OPERABLE, the MCRE boundary must be maintained such that the MCRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analysis for DBAs (Ref. 3), and that MCRE occupants are protected from hazardous chemicals and smoke.

The initial MCRE heat sink thermal mass, required individual room heat sink thermal mass, and required room-pair heat sink thermal mass are initial conditions required to limit the maximum MCRE temperature. Thermal mass is the ability of a material to absorb and store heat energy. In the context of the MCRE heatup analysis, the thermal mass of the heat sinks provides inertia against temperature changes. Establishing the passive heat sink nominal conditions is related to the time of exposure and magnitude of relevant heat sources, and is dependent upon material properties such as specific heat capacity and density of concrete. The thermal mass of the required MCRE heat sinks (the MCRE, individual required rooms adjacent to and below the MCRE, required room-pairs adjacent to and below the MCRE, and the room above the MCRE) must be within limits to support VES OPERABILITY and limit the maximum MCRE temperature for 72 hours after VES actuation.

The LCO is modified by a Note allowing the MCRE boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the MCRE. This individual will have a method to rapidly close the opening and to restore the MCRE boundary to a condition equivalent to the design condition when a need for MCRE isolation is indicated.

Both PMS Divisions A and C in the two safety-related electrical panels are required to be OPERABLE, so that non-safety stage 1 and stage 2 MCR heat loads can be de-energized by the VES system actuation signal within the required time, assuming a single failure. This maintains the MCR temperature within habitable limits.

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BASES

APPLICABILITY

In MODES 1, 2, 3, and 4 and during movement of irradiated fuel assemblies, the VES must be OPERABLE to ensure that the MCRE will remain habitable during and following a DBA.

The VES is not required to be OPERABLE in MODES 5 and 6 when irradiated fuel is not being moved because accidents resulting in fission product release are not postulated.

ACTIONS

A.1

When one or more of the following VES functions have only one inoperable flow path (e.g., one or more inoperable for opening VES valve or VES damper in a flow path of one of the below listed types of functions), the VES safety function is maintained for that type of function:

- Air Delivery Function,
- Pressure Relief Function,
- Eductor Bypass Function, and
- Distribution Function.

When one or more of the following VES, SDS vent line, or a main control room boundary isolation flow paths have only one valve inoperable for closing, the VES safety function is maintained:

- Pressure relief flow path,
- VBS Isolation flow path, and
- SDS vent line flow path.

In this condition, action is required to restore the affected VES valve or damper to OPERABLE status. A Completion Time of 7 days is permitted to restore the valve or damper to OPERABLE status before action must be taken to reduce power. The Completion Time of 7 days is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release from the fuel, the low probability of not containing the radiation, and that the remaining components can provide the required VES safety function. In the event the VES safety function is not maintained, entry into Condition F and/or Condition G is required.

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ACTIONS (continued)

Condition A is modified by a Note that states separate Condition entry is allowed for each valve and each damper. This is acceptable, since each function and flow path is provided with redundancy such that sufficient remaining OPERABLE valves and dampers can provide appropriate VES safety function during the 7 day Completion Time. Complying with the Required Actions may allow for continued operation during the 7 day Completion Time, and a subsequent inoperable valve or damper is governed by subsequent Condition entry and application of associated Required Action.

B.1

If one division of one or more MCR load shed panel(s) is inoperable, all divisions of both MCR load shed panels must be restored to OPERABLE status within 7 days. In this condition, the OPERABLE unaffected division of the panel is capable of providing 100% of the load shed function.

A Completion Time of 7 days is permitted to restore the inoperable division of MCR load shed panel(s) to OPERABLE status before action must be taken to reduce power. The Completion Time of 7 days is based on engineering judgment, considering the low probability of an accident that would require VES actuation, and that the remaining panel division can provide the required load shed function.

As described in FSAR Subsection 6.4.2.3 of Ref. 1, any component failure in a PMS division of the load shed panel(s) renders that division inoperable. If this failure affects only one PMS division, leaving the remaining division of PMS unaffected, including the associated power and control circuit, it renders the panel(s) inoperable, while still maintaining the full load shed function. An event or action that impacts both PMS divisions in either panel does not maintain the full load shed function, and Condition F or G of LCO 3.7.6 would apply.

C.1 and C.2

When the thermal mass of one or more of the required MCRE heat sinks (the MCRE, individual required rooms adjacent to and below the MCRE, required room-pairs adjacent to and below the MCRE, and the room above the MCRE) is not within the required limit(s), the heat sink air temperature must be restored to within limit in 24 hours and the thermal mass of the required heat sink(s) must be restored to within limit(s) in 5 days.

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ACTIONS (continued)

The Required Action C.1 Completion Time of 24 hours to initially restore the heat sink air temperature to within limit is based on engineering judgment, considering the low probability of an accident that would require VES actuation under the worst case temperature conditions, and is permitted based upon the availability of temperature indication in the MCRE and individual required rooms, and in the air return ducts to the adjacent required room-pairs. It is judged to be a sufficient amount of time allotted to correct the deficiency in the non-safety VBS ventilation system.

The MCRE heatup analysis demonstrates that the heat sink thermal mass returns to baseline assumptions after a variable time period depending upon the extent of VBS HVAC system degradation and outage time (i.e., extent and duration of the loss of VBS cooling) and upon heat sink wall thickness. For a total loss of VBS cooling that lasts for 24 hours, maintaining ambient air temperature below the limit for the MCRE (i.e., ≤ 75ºF), the individual required rooms (i.e., ≤ 85°F), and adjacent required room-pairs (i.e., ≤ 85°F) for 4 days is one method of re-establishing the heat sink thermal mass assumed in the safety analysis. Alternatively, analyses or local measurements can evaluate ambient air temperature excursions for impact on meeting the thermal mass assumed in the main control room heatup calculations and Condition C can be exited once the thermal mass of the required heat sinks is determined to be within limits. The Completion Time for Required Action C.2 is 5 days.

The selection of air temperature is an indication of heat sink temperature and heat sink thermal mass. It is recognized that the thermal mass of the passive heat sinks will not be restored to baseline assumptions after air temperature is restored within limits because the heat sinks take longer to be restored to the initial conditions assumed in the MCRE heatup analysis.

D.1, D.2, and D.3

If the unfiltered inleakage of potentially contaminated air past the MCRE boundary and into the MCRE can result in MCRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of MCRE occupants from hazardous chemicals or smoke, the MCRE boundary is inoperable. Actions must be taken to restore an OPERABLE MCRE boundary within 90 days.

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ACTIONS (continued)

During the period that the MCRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on MCRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that MCRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that MCRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable MCRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions.

Required Action D.3 allows time to restore the MCRE boundary to OPERABLE status provided mitigating actions can ensure that the MCRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3 (Ref. 4) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating actions as required by Required Action D.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 7).

Options for restoring the MCRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the MCRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the MCRE boundary has been restored to OPERABLE status. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of MCRE occupants within analyzed limits while limiting the probability that MCRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the MCRE boundary.

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ACTIONS (continued)

E.1, E.2, and E.3

With VES compressed air storage tanks compressed air volume not within limit (i.e., containing ≤ 327,574 scf of compressed air), action must be taken to restore VES compressed air storage tanks compressed air volume to within limit within 7 days. In this Condition, Required Action E.1 directs that the stored amount of compressed air in the VES compressed air storage tanks must be verified within 2 hours and every 12 hours thereafter to contain > 245,680 scf. The 245,680 scf value is 75% of the minimum amount of stored compressed air that must be available in the compressed air storage tanks. The standard volume is determined using the compressed air storage tank room temperature (VAS-TE-080A/B) and compressed air storage tanks pressure (VES-PT-001A/B) via the plant control system and chart in the system procedures. Verification that the minimum volume of compressed air is contained in the compressed air storage tanks ensures a 54 hour air supply will be available if needed. Additionally, within 24 hours, the VBS ancillary fans are verified to be available so that, if needed, can be put into use once the compressed air storage tanks have been exhausted.

The Completion Times associated with these actions and the 7 day Completion Time to restore VES compressed air storage tanks compressed air volume to within limit is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release from the reactor core, the low probability of radioactivity release, and that the remaining components and compensatory systems can provide the required capability. The 54 hours of air remaining in the compressed air storage tanks, along with compensatory operator actions, are adequate to protect the main control room envelope habitability. Dose calculations verify that the MCRE dose limits will remain within the requirements of GDC 19 with the compensatory actions taken at 54 hours.

F.1 and F.2

If the Required Actions and Completion Times of Condition A, B, C, or E are not met, or the VES is inoperable for reasons other than Condition A, B, C, D, or E (e.g., VES declared inoperable in accordance with TS 3.3.18 Required Action D.1), VBS must be verified available to mitigate the VES inoperability within 2 hours (Required Action F.1). Verifying VBS availability for mitigating VES inoperability includes verifying at least one train of VBS and its standby diesel generator and supporting subsystems is available to isolate the MCR from the normal outdoor air intake and provide filtered outdoor air to pressurize the MCR to a positive pressure

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BASES

ACTIONS (continued)

of at least 1/8 inch wg when a “High-1” radioactivity concentration (gaseous, particulate, or iodine) is detected in the main control room supply air duct, as well as the availability of the supply air handling unit to provide cooling with recirculation air to maintain the MCR and CSA areas within their design temperatures.

The 7 day Completion Time is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release, the low probability of radioactivity release, and that the available VBS can provide the required capability. Dose calculations verify that the MCRE dose limits will remain within the requirements of GDC 19 with VBS mitigation in the event of VES inoperability.

G.1 and G.2

In MODE 1, 2, 3, or 4, if the Required Actions and associated Completion Times of Condition D or F are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

H.1

During movement of irradiated fuel assemblies, if the Required Actions and associated Completion Times of Condition F are not met, or the VES is inoperable due to an inoperable MCRE boundary, action must be taken immediately to suspend the movement of fuel. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Verification is required every 24 hours that compressed air storage tanks contain > 327,574 scf of breathable air.

The standard volume is determined using the compressed air storage tank room temperature (VAS-TE-080A/B) and compressed air storage tanks pressure (VES-PT-001A/B) via the plant control system and chart in the system procedures. Verification that the minimum volume of compressed air is contained in the compressed air storage tanks ensures that there will be an adequate supply of breathable air to maintain MCRE habitability for a period of 72 hours. The Frequency of 24 hours is based on the availability of pressure indication in the MCRE.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.2

SR 3.7.6.2 verifies that the thermal mass of the required heat sinks is within limit(s) every 24 hours. One method of satisfying SR 3.7.6.2 is maintaining ambient air temperature below the limit for the MCRE (i.e., ≤ 75°F), the individual required rooms (i.e., ≤ 85°F), and adjacent required room-pairs (i.e., ≤ 85°F) for 4 days. Alternatively, analyses or local measurements can satisfy the verification of the heat sink thermal mass assumed in the main control room heatup calculation.

Satisfying the required heat sink room temperature limits (i.e., ≤ 75°F for the MCRE and ≤ 85°F for the individual required rooms and the adjacent required room-pairs) for sufficient duration (i.e., 4 days) establishes the equilibrium initial heat sink thermal mass assumed in the main control room heatup calculation.

Passive heat sink air temperatures in required dc Equipment Room and required I&C room-pairs (12201/12301, 12203/12302, 12205/12305, and 12207/12304) are verified by temperature elements located in the shared return air ducting (alternatively, local measurement of each room may be utilized). Other required individual rooms (12202, 12204, 12300, 12303, 12313, 12412, and 12501) are verified using indication from the temperature elements in each room.

This is done to verify that the VBS is performing as required to maintain the initial conditions assumed in the safety analyses, and to verify the VES heat sinks provide adequate thermal mass to limit the temperature increase in the MCRE, dc Equipment Rooms, and I&C Rooms from exceeding the allowable limits after VES actuation.

The 24 hour Frequency is acceptable based on the availability of automatic VBS temperature controls, alarms, and indication in the MCRE. Air temperatures may also be verified using local measurement.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.3

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing VES once every month provides an adequate check of the system. This monthly VES operation requires operation of one of the two pressure regulating valves. Additionally, the main air delivery line has two parallel air delivery flow paths via automatically actuated isolation valves (VES-V005A/B) that are also required to be exercised quarterly. Monthly testing of the three air delivery flow paths on a STAGGERED TEST BASIS will result in at least quarterly exercising of each air delivery valve. The 31 day on a STAGGERED TEST BASIS Frequency is based on the reliability of the equipment and the availability of system redundancy.

SR 3.7.6.4

VES air header isolation valves are required to be verified open at 31 day intervals. This SR is designed to ensure that the pathways for supplying breathable air to the MCRE are available should loss of VBS occur. These valves should be closed only during required testing or maintenance of downstream components, or to preclude complete depressurization of the system should the VES isolation valves in the air delivery line open inadvertently or begin to leak.

SR 3.7.6.5

Verification that the air quality of the compressed air storage tanks meets the requirements of CGA G-7.1, Commodity Specification for Air, Grade E, (Ref. 5) with a pressure dew point of ≤ 40°F at ≥ 3400 psig is required every 92 days. If air has not been added to the air storage tanks since the previous verification, verification may be accomplished by confirmation of the acceptability of the previous surveillance results along with examination of the documented record of air makeup. Verification of the initial air quality (in combination with the other surveillances) ensures that breathable air is available for 11 MCRE occupants for at least 72 hours. Confirmation of the pressure dew point verifies that water has not formed in the line, eliminating the potential for freezing at the pressure regulating valve during VES operation. In addition, the dry air allows the MCRE to remain below the maximum relative humidity to support the 90°F WBGT required for human factors performance.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.6

Verification that the VBS isolation valves and the SDS isolation valves are OPERABLE and will actuate upon demand is required every 24 months to ensure that the MCRE can be isolated upon loss of VBS operation. The SR excludes automatic valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to valves that are locked, sealed, or otherwise secured in the actuated position since the affected valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve to the non-actuated position requires verification that the SR has been met within its required Frequency. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves.

SR 3.7.6.7

Verification that each VES pressure relief isolation valve within the MCRE pressure boundary is OPERABLE is required in accordance with the Inservice Testing Program. The valves should actuate on an actual or simulated signal to the correct position. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. The SR is used in combination with SR 3.7.6.8 to ensure that adequate vent area is available to mitigate MCRE overpressurization.

SR 3.7.6.8

Verification that the VES pressure relief damper is OPERABLE is required at 24 month intervals. The SR is used in combination with SR 3.7.6.7 to ensure that adequate vent area is available to mitigate MCRE overpressurization.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.9

This SR verifies the OPERABILITY of the MCRE boundary by testing for unfiltered air inleakage past the MCRE boundary and into the MCRE. The details of the testing are specified in the Main Control Room Envelope Habitability Program.

The MCRE is considered habitable when the radiological dose to MCRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the MCRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the MCRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition D must be entered.

SR 3.7.6.10

This SR verifies that the required VES testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VES filter tests are in accordance with the FSAR Appendix 1A position for Regulatory Guide 1.52 (Ref. 7). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.6.11

Verification that the MCR load shed function actuates on an actual or simulated signal from each PMS Division is required every 24 months to confirm that the non-safety stage 1 and stage 2 MCR heat loads can be de-energized by the VES actuation signal within the required time. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage to minimize the potential for adversely affecting MCR operations.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.12

Verification that the main VES air delivery isolation valves actuate on an actual or simulated signal to the correct position is required every 24 months to confirm that the VES operates as assumed in the safety analysis. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage to minimize adversely affecting MCR operations.

REFERENCES

1. FSAR Section 6.4, “Main Control Room Habitability Systems.”
2. FSAR Section 9.5.1, “Fire Protection System.”
3. FSAR Subsection 15.6.5.3, “Radiological Consequences.”
4. Regulatory Guide 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors.”
5. CGA G-7.1-1997, “Commodity Specification for Air.”
6. NEI 99-03, “Control Room Habitability Assessment,” June 2001.
7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, “NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability.” (ADAMS Accession No. ML040300694).
8. Regulatory Guide 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Revision 3.

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B 3.7 PLANT SYSTEMS

B 3.7.7 Startup Feedwater (SFW) Isolation Valves, Control Valves, and Pump Breakers

BASES

BACKGROUND

The SFW system supplies feedwater to the steam generators during plant startup, hot standby and cooldown, and in the event of main feedwater unavailability.

The SFW system serves no safety related function and has no safetyrelated design basis, except to isolate feedwater in the event of a Feedline Break (FLB), Steam Line Break (SLB), a steam generator tube rupture (SGTR), or other secondary side event.

The SFW system consists of two SFW pumps, which are headered at the pump discharge, and a separate flow path to each of the steam generators. Each flow path consists of two series SFW valves to provide feedwater control for low feedwater demand conditions. Feedwater can be supplied to the SFW line via either the main or SFW. The feedwater is delivered directly to the steam generator (SG) independent of the main feedwater line. Each SFW line contains one control valve and one isolation valve (Ref. 1).

APPLICABLE SAFETY ANALYSES

The basis for the requirement to isolate the SFW system is established by the analysis for large SLB inside containment. It is also based on the analysis for a large FLB and a steam generator tube rupture.

Failure to isolate the SFW system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. Failure to isolate the SFW following a steam generator tube rupture may result in overfilling the steam generator.

Low 2 T_{cold} or high steam generator level signals close the SFW control and isolation valves and trips the SFW pumps.

In addition, the SFW isolation valves are containment isolation valves and support the assumptions related to minimizing the loss of inventory and establishing the containment boundary during major accidents. Therefore, the safety analysis of any event requiring isolation of containment is applicable to these valves.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The SFW isolation valves, control valves, and pump breakers are components which actuate to mitigate a Design Basis Accident, and as such meet Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the SFW isolation and control valves will isolate, and SFW pump breakers will trip, on command, following a SLB, FLB or SGTR, to isolate SFW flow to the steam generators.

The SFW isolation and control valves are considered OPERABLE when they automatically close on an isolation actuation signal and their isolation times are within the required limits. For a SFW pump breaker to be considered OPERABLE, it must trip on an actuation signal or be in the tripped condition.

APPLICABILITY

The SFW isolation and control valves, and SFW pump breakers, must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and the steam generators. In MODES 1, 2, 3 and 4, the SFW isolation and control valves, and SFW pump breakers, are required to be OPERABLE in order to limit the amount of mass and energy that could be added to containment in the event of a SLB or FLB and prevent steam generator overfill in the event of an SGTR and where a DBA could cause a release of radioactive material to containment.

In MODES 5 and 6, the energy in the steam generators is low, and isolation of the SFW system and trip of the feedwater pump breakers are not required.

ACTIONS

The Actions are modified by two Notes. Note 1 allows flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the flow paths can be rapidly isolated.

The second Note allows separate Condition entry for each flow path, where there are two flow paths each with an isolation and control valve in series and two SFW pump flow paths. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable flow path.

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BASES

ACTIONS (continued)

A.1 and A.2

With only one isolation or control valve OPERABLE in one or more flow paths, there is no redundant capability to isolate the flow paths. In this case, both an isolation and a control valve in each flow path must be restored to OPERABLE status with 72 hours, or the flow path must be isolated. A Completion Time of 72 hours is acceptable since, with one valve in a flow path inoperable, there is a second valve available in the flow path to isolate the line.

If the inoperable valve in the flow path cannot be restored to OPERABLE status, then the flow path must be isolated within a Completion Time of 72 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure.

For flow paths isolated in accordance with Required Action A.1, the affected flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that flow paths required to be isolated following an accident will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that the isolation devices are in the correct position. The Completion Time of “once per 7 days” is appropriate considering the fact that the devices are operated under administrative controls, valve status indications in the main control room and the probability of their misalignment is low.

B.1

With both the isolation and control valves inoperable in one flow path, the affected flow path must be restored to OPERABLE status or isolated within a Completion Time of 8 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure.

C.1

With one or more flow paths with the pump breaker inoperable the affected pump breaker must be restored to OPERABLE status within 6 hours. The Completion Time takes into account the redundancy afforded by the isolation and control valves and the low probability of an event that would require tripping the SFW pump during this period.

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BASES

ACTIONS (continued)

D.1, D.2, and D.3

If the affected flow paths cannot be isolated or the isolation valves, control valves, or pump breakers cannot be restored to OPERABLE status, closed, or isolated within the associated Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in least MODE 3 within 6 hours, in MODE 4 with RCS cooling provided by the normal residual heat removal system within 24 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This surveillance requires verification in accordance with the Inservice Testing Program to assure that each SFW isolation and control valve is OPERABLE. The Surveillance Frequency is provided in the Inservice Testing Program.

SR 3.7.7.2

This SR ensures that each SFW isolation valve and SFW control valve will actuate to its isolation position on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves.

The 24 month Frequency is based on the need to perform this Surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

SR 3.7.7.3

This SR verifies that each SFW pump breakers trip open on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the pump breaker.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The SR excludes SFW pump breakers that are maintained in the tripped position since a tripped open SFW pump breaker is already performing the safety function. Restoration of a breaker to the non-actuated position requires verification that the SR has been met within its required Frequency.

The 24 month Frequency is based on the need to perform this Surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

REFERENCES

1. FSAR Section 10.4.9, “Startup Feedwater System.”

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B 3.7 PLANT SYSTEMS

B 3.7.8 Main Steam Line Leakage

BASES

BACKGROUND

A limit on leakage from the main steam line inside containment is required to limit system operation in the presence of excessive leakage. Leakage is limited to an amount which would not compromise safety consistent with the Leak-Before-Break (LBB) analysis discussed in FSAR Chapter 3 (Ref. 1). This leakage limit ensures appropriate action can be taken before the integrity of the lines is impaired.

LBB is an argument which allows elimination of design for dynamic load effects of postulated pipe breaks. The fundamental premise of LBB is that the materials used in nuclear plant piping are strong enough that even a large throughwall crack leaking well in excess of rates detectable by present leak detection systems would remain stable, and would not result in a double-ended guillotine break under maximum loading conditions. The benefit of LBB is the elimination of pipe whip restraints, jet impingement effects, subcompartment pressurization, and internal system blowdown loads.

As described in FSAR Section 3.6 (Ref. 1), LBB has been applied to the main steam line pipe runs inside containment. Hence, the potential safety significance of secondary side leaks inside containment requires detection and monitoring of leakage inside containment (FSAR Section 3.6.3.3 of Ref. 1). This LCO protects the main steam lines inside containment against degradation, and helps assure that serious leaks will not develop. The consequences of violating this LCO include the possibility of further degradation of the main steam lines, which may lead to pipe break.

APPLICABLE SAFETY ANALYSES

The safety significance of plant leakage inside containment varies depending on its source, rate, and duration. Therefore, detection and monitoring of plant leakage inside containment are necessary. This is accomplished via the instrumentation required by LCO 3.4.9, “RCS Leakage Detection Instrumentation,” and the Reactor Coolant System (RCS) water inventory balance (SR 3.4.7.1). Subtracting RCS leakage as well as any other identified non-RCS leakage into the containment area from the total plant leakage inside containment provides qualitative information to the operators regarding possible main steam line leakage. This allows the operators to take corrective action should leakage occur which is detrimental to the safety of the facility and/or the public.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Although the main steam line leakage limit is not required by the 10 CFR 50.36(c)(2)(ii) criteria, this specification has been included in Technical Specifications in accordance with NRC direction (Ref. 2).

LCO

Main steam line leakage is defined as leakage inside containment in any portion of the two (2) main steam line pipe walls. Up to 0.5 gpm of leakage is allowable because it is below the leak rate for LBB analyzed cases of a main steam line crack twice as long as a crack leaking at ten (10) times the detectable leak rate under normal operating load conditions. Violation of this LCO could result in continued degradation of the main steam line.

APPLICABILITY

Because of elevated main steam system temperatures and pressures, the potential for main steam line leakage is greatest in MODES 1, 2, 3, and 4.

In MODES 5 and 6, a main steam line leakage limit is not provided because the main steam system pressure is far lower, resulting in lower stresses and a reduced potential for leakage. In addition, the steam generators are not the primary method of RCS heat removal in MODES 5 and 6.

ACTIONS

A.1 and A.2

With main steam line leakage in excess of the LCO limit, the unit must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The reactor must be placed in MODE 3 with 6 hours and MODE 5 within 36 hours. This action reduces the main steam line pressure and leakage, and also reduces the factors which tend to degrade the main steam lines. The Completion Time of 6 hours to reach MODE 3 from full power without challenging plant systems is reasonable based on operating experience. Similarly, the Completion Time of 36 hours to reach MODE 5 without challenging plant systems is also reasonable based on operating experience. In MODE 5, the pressure stresses acting on the main steam line are much lower, and further deterioration of the main steam line is less likely.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying that main steam line leakage is within the LCO limit assures the integrity of those lines inside containment is maintained. An early warning of main steam line leakage is provided by the automatic system which monitors the containment sump level. Main steam line leakage would appear as unidentified leakage inside containment via this system, and can only be positively identified by inspection. However, by performance of an RCS water inventory balance (SR 3.4.7.1) and evaluation of the cooling and chilled water systems inside containment, determination of whether the main steam line is a potential source of unidentified leakage inside containment is possible.

The Frequency is “Per SR 3.4.7.1,” which includes SR 3.4.7.1 Note 1 stating that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and reactor coolant drain tank and in-containment refueling water storage tank levels, and makeup and letdown flows.

REFERENCES

1. FSAR, Section 3.6, “Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping.”

2. NRC letter, Diane T. Jackson to Westinghouse (Nicholas J. Liparulo), dated September 5, 1996, “Staff Update to Draft Safety Evaluation Report (DSER) Open Items (OIs) Regarding the Westinghouse AP600 Advanced Reactor Design,” Open Item #365.

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B 3.7 PLANT SYSTEMS

B 3.7.9 Not Used

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B 3.7 PLANT SYSTEMS

B 3.7.10 Steam Generator (SG) Isolation Valves

BASES

BACKGROUND

The steam generator isolation valves consist of the power operated relief valve (PORV) block valves (SGS-PL-V027A & B), PORVs (SGS-PL-V233A & B), and blowdown isolation valves (SGS-PL-V074A & B and SGS-PL-V075A & B). The PORV flow paths must be isolated following a Steam Generator Tube Rupture (SGTR) to minimize radiological releases. The blowdown flow path must be isolated following Loss of Feedwater and Feedwater Line Break events to retain the steam generator water inventory for Reactor Coolant System (RCS) heat removal.

A PORV is installed in a branch line off of the main steam line piping from each steam generator, to provide for controlled removal of reactor decay heat during normal reactor cooldown when the main steam isolation valves are closed or the turbine bypass system is not available. A normally-open block valve is provided in each PORV line to provide backup isolation capability. Both the PORV and the block valve receive a Protection and Safety Monitoring System (PMS) isolation signal on Low 2 steam line pressure. The block valve is also a containment isolation valve.

The blowdown line from each steam generator is provided with two series isolation valves, both located outside, but close to, containment. The blowdown valves receive a PMS isolation signal on SG Narrow Range Water Level – Low 2 and on Passive Residual Heat Removal (PRHR) actuation. The first blowdown isolation valve outside of containment is also a containment isolation valve.

The steam generator PORVs and the blowdown isolation valves fail closed on loss of control or actuation power. The steam generator PORV block valves fail as-is on loss of control or actuation power. The steam generator isolation valves may also be actuated manually.

Descriptions of the PORVs and SG blowdown isolation are found in FSAR Section 10.3.2.2.3 and FSAR Section 10.4.8 (Refs. 1 & 2).

APPLICABLE SAFETY ANALYSES

The PORV flow paths must be isolated following an SGTR to minimize radiological releases from the ruptured steam generator into the atmosphere. The PORV flow path is assumed to open due to high secondary side pressure, during the SGTR. Dose analyses take credit for subsequent isolation of the PORV flow path by the PORV and/or the block valve which receive a close signal on Low 2 steam line pressure.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The blowdown flow path on each SG must be isolated following Loss of Feedwater and Feedwater Line Break events to retain the steam generator water inventory for use in Reactor Coolant System (RCS) heat removal via the SGs. RCS heat removal for these events is, primarily, provided by the Passive Residual Heat Removal Heat Exchanger (PRHR HX); however, the SG heat removal is assumed. The SG blowdown isolation valves receive an isolation signal on SG Narrow Range Water Level – Low 2 or PRHR actuation. These events take credit for steam generator heat removal using the water inventory retained after blowdown isolation. If the blowdown line were not isolated, much of the inventory would drain from the SG rather than cool the RCS.

In addition, the PORV block valves and steam generator blowdown valves are containment isolation valves and support the assumptions related to minimizing the loss of inventory and establishing the containment boundary during major accidents. Therefore, the safety analysis of any event requiring isolation of containment is applicable to the PORV block valves and steam generator blowdown valves.

The steam generator isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the steam generator PORV, PORV block valve, and steam generator blowdown isolation valves on each steam generator to be OPERABLE. These isolation valves are considered OPERABLE when the valves are capable of closing on a PMS isolation signal.

This LCO provides assurance that the steam generator PORV and PORV block valve will perform their design safety function to mitigate the consequences of an SGTR that could result in offsite exposures.

Additionally, this LCO provides assurance that the steam generator blowdown isolation valves will perform their design safety function to mitigate the consequences of Loss of Feedwater and Feedwater Line Break events by retaining the steam generator water inventory for Reactor Coolant System (RCS) heat removal.

APPLICABILITY

The steam generator PORV, PORV block valve, and steam generator blowdown isolation valves must be OPERABLE in MODES 1, 2, 3, and 4, where a DBA could cause a release of radioactive material to containment. The Applicability is modified by a Note indicating that PORV OPERABILITY is not required in MODE 4 with the RCS cooling being provided by the Normal Residual Heat Removal System (RNS).

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BASES

APPLICABILITY (continued)

In MODE 4 with the RCS cooling being provided by the RNS and in MODES 5 and 6, the steam generators are not needed for RCS cooling and the potential for Loss of Feedwater and Feedwater Line Break events is minimized due to the reduced mass and energy in the RCS and steam generators.

In MODE 4 with an RCS temperature < 350°F and MODES 5 and 6, the RCS pressure and temperature are reduced and an SGTR is not credible. However, the Applicability conservatively requires OPERABILITY in MODE 4 with RCS temperature < 350°F for PORVs until the RCS is being cooled by RNS.

ACTIONS

The ACTIONS are modified by a Note allowing the steam generator blowdown flow paths to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the flow path can be rapidly isolated when a need for blowdown isolation is indicated.

The second Note allows separate Condition entry for each steam generator PORV and blowdown flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable PORV and blowdown flow path.

A.1 and A.2

With one valve in one or more steam generator PORV flow paths (i.e., a PORV or PORV block valve) inoperable, action must be taken to isolate the flow path with a closed and deactivated valve. The valve must be deactivated to assure that the flow path will not be opened by a high pressure signal during the course of an SGTR event. This action places the flow path in a condition which assures the safety function is performed. A Completion Time of 72 hours is based on the availability of one OPERABLE PORV flow path isolation valve which is fully capable of performing the required isolation function.

To ensure the PORV flow path remains isolated, periodic verification is required. This is necessary to ensure that the assumptions in the safety analysis remain valid. This Requried Action does not require any testing or valve maniplation. Rather, it involves verification that the valve is in the correct position and deactivated. The 31 day Completion Time is reasonable, considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

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ACTIONS (continued)

Required Action A.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas, and allows these devices to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

B.1 and B.2

With one valve in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. A Completion Time of 72 hours to isolate the flow path is based on the availability of one OPERABLE blowdown flow path isolation valve which is fully capable of performing the required isolation function.

Since the blowdown isolation valve is not deactivated, periodic verification is required to assure that the flow path remains isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of status indications available in the control room, and other administrative controls, to ensure that the valve remains in the closed position.

C.1

With both valves in one or more steam generator PORV flow paths inoperable, action must be taken to isolate the flow path with a closed and deactivated valve. The valve must be deactivated to assure that the flow path will not be opened by a high pressure signal during the course of an SGTR event. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG PORV flow path isolation valves.

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BASES

ACTIONS (continued)

In the event the affected flow path is isolated in accordance with Required Action C.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect.

D.1

With two valves in one or more steam generator blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG blowdown flow path isolation valves.

In the event the affected flow path is isolated in accordance with Required Action D.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action B.2, which remains in effect.

E.1, E.2, and E.3

If the SG PORV flow path or blowdown flow path isolation valves cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, in MODE 4 with the RCS cooling provided by the RNS within 24 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems. Required Action E.3 to be in MODE 5 is modified by a Note stating that it is not applicable to inoperable PORV(s). PORV Applicability is exited on completion of Required Action E.2.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

The function of the SG PORV isolation valves (PORV block valves (SGS-PL-V027A & B) and PORVs (SGS-PL-V233A & B)) and blowdown isolation valves (SGS-PL-V074A & B and SGS-PL-V075A & B) is to isolate the steam generators in the event of SGTR, Loss of Feedwater or Feedwater Line Break. Stroking the valves closed demonstrates their capability to perform the isolation function. The Frequency for this SR is in accordance with the Inservice Testing Program.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

Verifying that the isolation time of each PORV block valve and SG blowdown isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation times are specified in Section 6.2.3 (Ref. 3) and Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.7.10.3

This Surveillance verifies that each SG PORV, SG PORV block valve, and SG blowdown isolation valve actuates to the isolation position on an actual or simulated actuation signal. The actual or simulated signal is processed through the component interface module to verify the continuity between the output of the component interface module and the valves.

The Frequency of 24 months is based on the need to perform this Surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.

REFERENCES

1. FSAR Section 10.3.2.2.3, “Power-Operated Atmospheric Relief Valves.”
2. FSAR Section 10.4.8, “Steam Generator Blowdown System.”
3. FSAR Section 6.2.3, "Containment Isolation System."

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B 3.7 PLANT SYSTEMS

B 3.7.11 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The water in the spent fuel pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel racks, the design basis for preventing criticality outside the reactor is that there is a 95% probability at a 95% confidence level, without soluble boron, that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.997, including uncertainties and tolerances. The NRC guidelines specify a k_{eff} limit of < 1.0 for normal storage in the absence of soluble boron. Therefore, the design is based on the use of unborated water, which maintains a subcritical condition (Ref. 1). The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 2) allows credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has a potential for more than negligible positive reactivity effect is an inadvertent mislocating of a new fuel assembly. This accident has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the location of each assembly in accordance with LCO 3.7.12, “Spent Fuel Pool Storage.” Prior to movement of an assembly, it is necessary to perform SR 3.7.12.1.

APPLICABLE SAFETY ANALYSES

Although credit for the soluble boron normally present in the spent fuel pool water is permitted under abnormal or accident conditions, most abnormal or accident conditions will not result in exceeding the limiting reactivity even in the absence of soluble boron. The effects on reactivity of credible abnormal and accident conditions due to temperature increase, dropped assembly, and mislocating/misloading of a fuel assembly have been analyzed. The reactivity effects of bulk spent fuel pool temperature increase (> 140°F) and steaming from the pool water surface or dropped assembly are assessed as none or negligible. The spent fuel pool k_{eff} storage limit of 0.95 is maintained during mislocating/misloading of a fuel assembly events by a minimum boron concentration of greater than or equal to 800 ppm

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BASES

APPLICABLE SAFETY ANALYSES (continued)

established by criticality analysis (Ref. 3). Compliance with the LCO minimum boron concentration limit of 2300 ppm provides sufficient margin to ensure that the credited concentration is available.

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The spent fuel pool boron concentration is required to be ≥ 2300 ppm. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 1 and 3. This concentration of dissolved boron is the minimum required concentration for spent fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool cooling requirements apply in all MODES when fuel is stored in the spent fuel pool, the ACTIONS have been modified by the Note stating that LCO 3.0.3 is not applicable. Spent fuel pool boron concentration requirements are independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, and A.2

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. FSAR Sections 9.1.2, “Spent Fuel Storage” and 15.7.4, “Fuel Handling Accident.”
2. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
3. APP-GW-GLR-029, “AP1000 Spent Fuel Storage Racks Criticality Analysis,” Westinghouse Electric Company LLC (Westinghouse Proprietary).

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B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool Storage

BASES

BACKGROUND

The high density spent fuel storage racks are divided into two separate and distinct regions and include locations for storage of defective fuel as shown in Figure 4.3-1. Region 1, with a maximum of 243 storage locations and the Defective Fuel Cells, with 5 storage locations, are designed and analyzed (Ref. 2) to accommodate new fuel assemblies with a maximum nominal enrichment of 4.95 weight percent U-235, or spent fuel assemblies regardless of the combination of initial nominal enrichment and burnup. Region 2, with a maximum of 641 storage locations is designed and analyzed (Ref. 2) to accommodate spent fuel assemblies in all locations which comply with the combination of initial nominal enrichment and burnup specified in LCO Figure 3.7.12-1, Minimum Fuel Assembly Burnup Versus Initial Nominal Enrichment for Region 2 Spent Fuel Cells. To allow for burnup uncertainty, which the Region 2 loading curve of LCO Figure 3.7.12-1 does not include, the fuel assembly burnup from the reactor records shall be conservatively reduced.

The water in the spent fuel pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel racks, the design basis for preventing criticality outside the reactor is that there is a 95% probability at a 95% confidence level, without soluble boron, that the effective multiplication faction (k_{eff}) of the fuel assembly array will be less than 0.997, including uncertainties and tolerances. The NRC guidelines specify a k_{eff} limit of < 1.0 for normal storage in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains a subcritical condition for the allowed loading patterns.

The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal and accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has the potential for more than negligible positive reactivity effect is an inadvertent mislocating of a new fuel assembly. This accident has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the combination of initial enrichment and burnup in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.12.1.

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BASES

APPLICABLE SAFETY ANALYSES

The fuel handling accident can only take place during or as a result of the movement of an assembly (Refs. 2 and 3). For these accident occurrences, the presence of soluble boron in the spent fuel pool (controlled by LCO 3.7.11, “Spent Fuel Pool Boron Concentration”) prevents criticality. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within Region 2 of the spent fuel pool in the accompanying LCO, ensure the k_{eff} of the spent fuel pool will always remain < 0.997, assuming the pool to be flooded with unborated water and ≤ 0.95, with a boron concentration of greater than or equal to 800 ppm.

Region 2 permits storage of spent fuel assemblies in any cell location provided the assembly meets the combination of initial nominal enrichment and burnup shown in LCO Figure 3.7.12-1, Minimum Fuel Assembly Burnup Versus Initial Nominal Enrichment for Region 2 Spent Fuel Cells. The Acceptable Region of the Figure is to the left and above the curve.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of this spent fuel pool.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool storage requirements apply in all MODES when fuel is stored in Region 2, the ACTIONS have been modified by a Note stating the LCO 3.0.3 is not applicable. Spent fuel pool storage requirements are independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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BASES

ACTIONS (continued)

A.1

The LCO is not met if spent fuel assemblies stored in Region 2 spent fuel assembly storage locations do not meet the applicable initial nominal enrichment and burnup limits in accordance with Figure 3.7.12 1.

When the LCO is not met, action must be initiated immediately to make the necessary fuel assembly movement(s) in Region 2 to bring the storage configuration into compliance with Figure 3.7.12-1 by moving the affected fuel assemblies to Region 1 or the Defective Fuel Cells.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies by administrative means that the initial nominal enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.12-1. Fuel assemblies stored in Region 2 that do not meet the Figure 3.7.12-1 enrichment and burnup limits shall be stored in Region 1– or Defective Fuel Cells.

REFERENCES

1. Double contingency principle ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. APP-GW-GLR-029, “AP1000 Spent Fuel Storage Racks Criticality Analysis,” Westinghouse Electric Company LLC (Westinghouse Proprietary).
3. FSAR Sections 9.1.2, “Spent Fuel Storage” and 15.7.4, “Fuel Handling Accident.”

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B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool Cooling System (SFS) Containment Isolation Valves

BASES

BACKGROUND

The safety-related SFS containment isolation valves are used to isolate the SFS from the refueling cavity in the event of a leak in the nonsafety-related SFS.

The SFS containment isolation valves are on the SFS supply and return lines that penetrate the containment barrier, which can be aligned to and from the refueling cavity and In-containment Refueling Water Storage Tank (IRWST). The SFS supply line is isolated by one motor-operated valve located outside containment (SFS-PL-V038) and one check valve located inside containment (SFS-PL-V037). The SFS return line is isolated by two motor-operated isolation valves, one located inside containment (SFS-PL-V034) and one located outside containment (SFS-PL-V035). The two SFS containment isolation valves on each of the two penetration flow paths are required to meet the single failure criterion. Each motor-operated isolation valve may be actuated manually from the main control room.

The SFS containment isolation valves close automatically as required by LCO 3.3.14, “ESFAS IRWST and Spent Fuel Pool Level Instrumentation.”

APPLICABLE SAFETY ANALYSES

In the event of a leak in the nonsafety-related SFS in MODE 6 with refueling cavity and spent fuel pool volumes in communication, a loss of refueling cavity water inventory may occur. Automatic actuation to close the SFS containment isolation valves on a Spent Fuel Pool Level - Low 2 signal with refueling cavity and spent fuel pool volumes in communication terminates draining of the refueling cavity before reducing the combined IRWST and refueling cavity water inventory below the minimum volume required to provide adequate decay heat removal in MODE 6.

In MODE 6 with refueling cavity and spent fuel pool volumes in communication, the spent fuel pool level is the same as the refueling cavity, and Refueling Cavity and SFS Isolation on Spent Fuel Pool Level – Low 2 also terminates draining of the spent fuel pool.

Therefore, each SFS containment isolation valve is required to be OPERABLE in MODE 6 with refueling cavity and spent fuel pool volumes in communication.

The SFS containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

The requirement that the SFS containment isolation valves be OPERABLE assures that there will be redundant means available to terminate draining of the refueling cavity and spent fuel pool in the event that a leak in the nonsafety-related SFS occurs in MODE 6 with refueling cavity and spent fuel pool volumes in communication.

APPLICABILITY

The SFS containment isolation valves must be OPERABLE in MODE 6 with refueling cavity and spent fuel pool volumes in communication.

LCO 3.6.3, “Containment Isolation Valves” requires OPERABILITY of the SFS containment isolation valves in MODES 1, 2, 3, and 4. Surveillance Requirements of LCO 3.5.7, “In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5,” and LCO 3.5.8, “IRWST – Shutdown, MODE 6,” require the SFS containment isolation valves to be closed, except under administrative controls, in MODE 5 and in MODE 6 with refueling cavity and spent fuel pool volumes not in communication.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 allows the penetration flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration flow path can be rapidly isolated when a need for penetration flow path isolation is indicated. Note 2 allows separate Condition entry for each SFS penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SFS penetration flow path.

A.1 and A.2

With one or more penetration flow paths with one or more SFS containment isolation valves inoperable, then the affected penetration flow path must be isolated within 24 hours. When the affected penetration flow path is isolated, it is performing its required safety function.

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BASES

ACTIONS (continued)

Additionally, to assure that the penetration flow path remains closed, the penetration flow path shall be isolated by the use of one of the specified means (Required Action A.2.1) or the penetration flow path shall be verified to be isolated (Required Action A.2.2). A means of isolating the affected penetration flow path includes at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured within 7 days. If one of the Required Action A.2.1 specified isolation means is not used, the affected penetration flow path shall be verified to be isolated once per 7 days.

B.1

If the Required Action and associated Completion Time of Condition A is not met, the IRWST must be declared inoperable immediately. Declaring the IRWST inoperable allows the supported system Required Actions (i.e., LCO 3.5.8) to dictate the required measures. The IRWST LCO provides appropriate Required Actions for the inoperability of the SFS containment isolation valves. This action is in accordance with LCO 3.0.6, which requires that the applicable Conditions and Required Actions for the IRWST declared inoperable shall be entered in accordance with LCO 3.0.2.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

SR 3.7.13.1 requires performance of Surveillances required by SR 3.6.3.4 and SR 3.6.3.5 for the SFS containment isolation valves required OPERABLE. The Frequencies associated with each specified SR are applicable. Therefore, see the corresponding Bases for LCO 3.6.3 for a discussion of each SR.

REFERENCES None.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 DC Sources – Operating

BASES

BACKGROUND

The Class 1E DC and UPS System (IDS) provides electrical power for safety related and vital control instrumentation loads, including monitoring equipment and main control room emergency lighting. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost and cannot be recovered for up to 72 hours. As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the Class 1E DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The Class 1E DC electrical power system also conforms to the requirements of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 250 VDC electrical power system consists of four independent safety related Class 1E DC electrical power subsystems (Division A, B, C, and D). Divisions A and D each consist of one battery bank, one battery charger, and the associated control equipment and interconnecting cable. Divisions B and C each consist of two battery banks, two battery chargers, and the associated control equipment and interconnecting cabling. The first battery bank in each of the four divisions, designated as the "24 hour" battery bank, provides power to the loads required for the first 24 hours following an event. The second battery bank in Divisions B and C, designated as the "72 hour" battery bank, is used for those loads requiring power for 72 hours following an event. The loads on the battery banks (including those on the associated inverters) are grouped according to their role in response to a Design Basis Accident (DBA). Loads which are a one time or limited duration load (engineered safety features (ESF) actuation cabinets and reactor trip function) required within the first 24 hours following an accident are connected to the “24 hour” battery bank. Loads which are continuous or required beyond the first 24 hours following an accident (emergency lighting, post accident monitoring, and Qualified Data Processing System) are connected to the “72 hour” battery bank. There are a total of six battery banks. A battery bank (also referred to as a battery) consists of two battery strings connected in series. Each battery string consists of 60 cells connected in series. Divisions A and D each have one 2400 ampere hour battery bank and Divisions B and C each have two 2400 ampere hour battery banks.

Additionally, there is one installed spare battery bank and one installed spare battery charger, which provide backup service in the event that one of the battery banks and/or one of the preferred battery chargers is out of service. The spare battery bank and charger are Class 1E and have the same rating as the primary components. If the spare battery bank with

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BASES

BACKGROUND (continued)

the charger is substituted for one of the preferred battery banks or chargers, then the requirements of independence and redundancy between subsystems are maintained and the division is OPERABLE.

During normal operation, the 250 VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

Each battery bank provides power to an inverter, which in turn powers an AC instrumentation and control bus. The AC instrumentation and control bus loads are connected to inverters according to the battery bank type, 24 hour or 72 hour.

The Class 1E DC power distribution system is described in more detail in Bases for LCO 3.8.5, “Distribution Systems – Operating,” and LCO 3.8.6, “Distribution Systems – Shutdown.”

Each battery has adequate storage capacity to carry the required load for the required duration as discussed in Reference 4.

Each 250 VDC battery bank, including the spare battery bank, is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystems to ensure that a single failure in one subsystem does not cause a failure in a separate subsystem. There is no sharing between separate Class 1E subsystems such as batteries, battery chargers, or distribution panels.

The batteries for each Class 1E electrical power subsystem are based on 125% of required capacity. The battery terminal minimum established float voltage limit is 2.20 V per cell, which corresponds to a total minimum voltage output of 264 V per battery discussed in Reference 4. The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger has sufficient capacity to restore the battery bank from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads (Ref. 4).

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BASES

BACKGROUND (continued)

The main control room air supply radiation monitor sample pumps are de-energized by the Protection and Safety Monitoring System (PMS) on a "Class 1E 24-hour Battery Charger Undervoltage" signal (refer to LCO 3.3.13, "ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load Deenergization," Bases) to remove non-essential loads from the batteries.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR Chapter 6 (Ref. 6) and FSAR Chapter 15, (Ref. 7), assume that engineered safety features are OPERABLE. The Class 1E DC electrical power system provides 250 volts power for safety related and vital control instrumentation loads including monitoring and main control room emergency lighting during all MODES of operation. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost.

The OPERABILITY of the Class 1E DC sources is consistent with the initial assumptions of the accident analyses. This includes maintaining at least three of the four divisions of DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite and onsite AC power sources; and

b. A worst case single failure.

The DC Sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Class 1E DC electrical power subsystems are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of Class 1E DC electrical power from one division does not prevent the minimum safety function from being performed (Ref. 4).

De-energization of each main control room air supply radiation monitor sample pump on a valid PMS signal is also required to be OPERABLE to support the Class 1E 24-hour battery OPERABILITY based on the assumed battery load profile, to maintain room temperature below the equipment qualification limitation, and maintain the assumed heat load for Main Control Room habitability.

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BASES

LCO (continued)

An OPERABLE Class 1E DC electrical power subsystem requires all required batteries and respective chargers to be operating and connected to the associated DC bus(es). The spare battery and/or charger may be used by one subsystem for OPERABILITY.

APPLICABILITY

The Class 1E DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

Class 1E DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.2, “DC Sources – Shutdown.”

ACTIONS

A.1, A.2, and A.3

Condition A represents one division with one or two battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 6 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion

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BASES

ACTIONS (continued)

Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 6 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 6 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 24 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 24 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable.

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BASES

ACTIONS (continued)

Required Action A.3 limits the restoration time for the inoperable battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used. The 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

B.1, B.2, and B.3

Condition B represents two divisions with one or more battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action B.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to the charger inoperability.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 24 hours (Required Action B.2).

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BASES

ACTIONS (continued)

Required Action B.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 24 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action B.3 limits the restoration time for the inoperable battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used. The 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

C.1

Condition C represents one division with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the OPERABLE battery chargers. Any event that results in a loss of the AC bus supporting the battery chargers will also result in loss of DC to that train.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

D.1

Condition D represents two divisions with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the OPERABLE battery charger. Any event that results in a loss of the AC bus supporting the battery charger will also result in loss of

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BASES

ACTIONS (continued)

DC to that train. The 2 hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than 2.07 V, etc.) are identified in Specifications 3.8.1, 3.8.2, and 3.8.7 together with additional specific completion times.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

E.1

If one of the Class 1E DC electrical power subsystems is inoperable, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all design basis accidents, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System (PMS) actuation divisions (LCO 3.3.15, Engineered Safety Feature Actuation System (ESFAS) Actuation Logic - Operating). Depending on the nature of the DC electrical power subsystem inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS Division is similar to loss of one DC electrical power subsystem. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

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BASES

ACTIONS (continued)

F.1

Condition F represents two subsystems with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected distribution subsystems. The 2 hour limit is consistent with the allowed time for two inoperable DC distribution subsystems.

If two of the required DC electrical power subsystems are inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the two remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all but the very worst case events. Since a subsequent worst case single failure would, however, result in the loss of the third subsystem, leaving only one subsystem with limited capacity to mitigate events, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 11) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

G.1 and G.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers which support ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery

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SURVEILLANCE REQUIREMENTS (continued)

cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. This voltage maintains the battery plates in a condition that supports maintaining the grid life (expected to be approximately 20 years). The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8).

SR 3.8.1.2

This SR verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32 (Ref. 9), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying 150 amps at the minimum established float voltage for 8 hours. The ampere requirements are based on the design minimum charger size of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least 2 hours.

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest combined demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, an the exponential decay in charging current. The battery is recharged when the measured charging current is ≤ 2 amps.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 24 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

A battery service test, as defined in IEEE 450-1995 (Ref. 8), is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the Class 1E DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 24 months is consistent with accepted industry practice. This Surveillance may be performed during any plant condition with the spare battery and charger providing power to the bus.

This SR is modified by a Note. The Note allows the performance of a modified performance discharge test in lieu of a service test.

The modified performance discharge test is a test, in the “as found” condition, of a battery’s capacity with the discharge rate(s) modified to encompass every portion of the battery duty cycle. This allows the performance test to be accomplished in the minimun amount of time while still demonstrating the high rate capability of the battery to meet duty cycle requirements. Thus, a modified performance test satisfies both the requirement to perform periodic performance tests as well as the requirement to perform periodic service tests.

The battery terminal voltage for the modified performance discharge test is required to remain above the minimum battery terminal voltage specified in the battery service test for the the duration of the service test portion of the test. Initial conditions for the modified performance discharge test are identical to those specified for a service test.

SR 3.8.1.4

Verification that the main control room air supply radiation monitor sample pump de-energize on an actual or simulated signal from the PMS is required every 24 months to assure that the non-essential Class 1E 24-hour battery loads are shed to maintain the assumed battery load profile.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and other administrative controls existing to ensure adequate charger performance during these 24 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is modified by a Note stating that it is only required to be met when the main control room air supply radiation monitor sample pump is energized.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, “Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems,” U.S. Nuclear Regulatory Commission, March 10, 1971.
3. IEEE-308 1991, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers.
4. FSAR Section 8.3.2, “Class 1E DC Power Systems.”
5. IEEE-485 1997, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications,” Institute of Electrical and Electronic Engineers.
6. FSAR Chapter 6, “Engineered Safety Features.”
7. FSAR Chapter 15, “Accident Analyses.”
8. IEEE-450 1995, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,” Institute of Electrical and Electronic Engineers.
9. Regulatory Guide 1.32, “Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, February 1977.
10. Not used.
11. Regulatory Guide 1.93, “Availability of Electric Power Sources,” U.S. Nuclear Regulatory Commission, December 1974.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 DC Sources – Shutdown

BASES

BACKGROUND

A description of the Class 1E DC power sources is provided in the Bases for LCO 3.8.1, “DC Sources – Operating.”

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the emergency auxiliaries and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystem is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems’ OPERABILITY.

The OPERABILITY of the minimum Class 1E DC power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate Class 1E DC power sources are provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. To demonstrate robust design, a single failure is assumed, consistent with the single failure assumptions in TS 3.8.1 and the shutdown evaluation in FSAR Appendix 19E (Ref. 3).

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BASES

APPLICABLE SAFETY ANALYSES (continued)

These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case Design Basis Accidents which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The Class 1E DC Sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Class 1E DC electrical power subsystems are required to be OPERABLE to support required trains of Class 1E Distribution System divisions required to be OPERABLE by LCO 3.8.6. This ensures the availability of sufficient Class 1E DC power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents, inadvertent reactor vessel draindown).

De-energization of the main control room air supply radiation monitor sample pumps on a valid PMS signal is also required to be OPERABLE to support the Class 1E 24-hour battery OPERABILITY based on the assumed battery load profile, to maintain room temperature below the equipment qualification limitation, and maintain the assumed heat load for Main Control Room habitability.

As described in the previous section, “Applicable Safety Analyses,” in the event of an accident during shutdown, the Technical Specifications are designed to maintain the plant in such a condition that, even with a single failure, the plant will not be in immediate difficulty.

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BASES

APPLICABILITY

The Class 1E DC power sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Required features needed to mitigate a fuel-handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The Class 1E DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1, “DC Sources – Operating.”

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2, and A.3

Condition A represents one division with one or two battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 6 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

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BASES

ACTIONS (continued)

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus, there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If the charger is operating in the current limit mode after 6 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 24 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 24 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action A.3 limits the restoration time for the inoperable battery charger to 72 hours. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used. The 72 hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

B.1, B.2.1, B.2.2, B.2.3, and B.2.4

With one or more of the required (per LCO 3.8.6, “Distribution Systems – Shutdown”) Class 1E DC power subsystems inoperable, the remaining subsystems may be capable of supporting sufficient systems to allow continuation of fuel movement and/or operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO

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BASES

ACTIONS (continued)

ACTIONS. In many instances this option would likely involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit) to assure continued safe operation. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary Class 1E DC electrical power to the unit safety systems.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC power subsystem; however, all applicable surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required Class 1E DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires performance of all Surveillances required by SR 3.8.1.1 through SR 3.8.1.4. Therefore, see the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

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BASES

REFERENCES

1. FSAR Chapter 6, “Engineered Safety Features.”
2. FSAR Chapter 15, “Accident Analysis.”
3. FSAR Appendix 19E, “Shutdown Evaluation.”

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Inverters – Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the Class 1E AC instrument and control buses because of the stability and reliability they achieve. Divisions A and D, each consist of one Class 1E inverter. Divisions B and C, each consist of two inverters. The function of the inverter is to convert Class 1E DC electrical power to AC electrical power, thus providing an uninterruptible power source for the instrumentation and controls for the Protection and Safety Monitoring System (PMS). The inverters are powered from the Class 1E 250 V battery sources (Ref. 1).

Under normal operation, a Class 1E inverter supplies power to the Class 1E AC instrument and control bus. If the inverter is inoperable or the Class 1E 250 VDC input to the inverter is unavailable, the Class 1E AC instrument and control bus is powered from the backup source associated with the same division via a static transfer switch featuring a make-before-break contact arrangement. In addition, a manual mechanical bypass switch is used to provide a backup power source to the Class 1E AC instrument and control bus when the inverter is removed from service. The backup source is a Class 1E 480-208/ 120 volt voltage regulating transformer providing a regulated output to the Class 1E AC instrument and control bus through a static transfer switch and a manual bypass switch.

In addition to powering safety loads, the Class 1E AC power sources are used for emergency lighting in the main control room and remote shutdown workstation. When a normal AC power source for emergency lighting is lost, the loads are automatically transferred to a Class 1E AC power source. Specific details on inverters and their operating characteristics are found in FSAR Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) transient analyses in FSAR Chapter 6 (Ref. 2) and FSAR Chapter 15 (Ref. 3), assume engineered safety features are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the PMS instrumentation and controls so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining at least three of the four Divisions of AC instrument and control buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite and onsite AC power source; and

b. A worst case single failure.

Inverters are a part of distribution systems, and as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the PMS instrumentation and controls is maintained. The six inverters (Divisions A and D, one each and Divisions B and C two each; six total) ensure an uninterruptible supply of AC electrical power to the six Class 1E AC instrument and control buses even if all AC power sources are de-energized.

OPERABLE inverters require that the Class 1E AC instrument and control bus be powered by the inverter with output voltage and frequency within tolerances, and the power input to the inverter from a 250 VDC station battery.

This LCO is modified by a Note that allows one inverter to be disconnected from its associated Class 1E DC bus for ≤ 72 hours, if the associated Class 1E AC instrument and control bus is powered from its Class 1E voltage regulating transformer during the period and all other inverters are OPERABLE. This allows an equalizing charge to be placed on one battery bank. If the inverter was not disconnected, the resulting voltage condition might damage the inverter. These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 72 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected Class 1E AC instrument and control bus while taking into consideration the time required to perform an equalizing charge on the battery bank.

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BASES

LCO (continued)

The intent of this Note is to limit the number of inverters that may be disconnected. Only the inverter associated with the single battery bank undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.4, “Inverters – Shutdown.”

ACTIONS

A.1

With one or two inverters inoperable in one division, the associated Class 1E AC instrument and control bus is automatically energized from its voltage regulating transformer. A manual switch is also provided which can be used if the static transfer switch does not properly function.

For this reason a Note has been included in Condition A requiring the entry into the applicable Conditions and Required Actions of LCO 3.8.5, “Distribution Systems – Operating.” This ensures that each affected vital bus is re-energized within 6 hours.

Required Action A.1 allows 14 days to fix each inoperable inverter and return it to service. The 14 day time limit is a risk-informed Completion Time based on site-specific risk analysis, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC instrument and control bus is powered from its voltage regulating transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC instrument and control buses is the preferred source for powering instrumentation trip setpoint devices.

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BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to MODE 5 where the probability and consequences on an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This Surveillance verifies that the inverters are functioning properly with all required switches and circuit breakers closed and Class 1E AC instrument and control buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the PMS instrumentation connected to the Class 1E AC instrument and control buses. The 7 day Frequency takes into account the effectiveness of the voltage and frequency instruments, the redundant capability of the inverters, and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. FSAR Section 8.3.2.1.1.2, “Class 1E Uninterruptible Power Supplies.”
2. FSAR Chapter 6, “Engineered Safety Features.”
3. FSAR Chapter 15, “Accident Analyses.”

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 Inverters – Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for Specification 3.8.3, “Inverters – Operating.”

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Protection and Monitoring System Engineered Safety Feature Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems’ OPERABILITY.

The OPERABILITY of the minimum inverters to each Class 1E AC instrument and control bus during MODES 5 and 6, and during movement of irradiated fuel assemblies, ensures that (Refs. 1 and 2):

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal

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BASES

APPLICABLE SAFETY ANALYSES (continued)

consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

To demonstrate robust design, a single failure is assumed, consistent with the single failure assumptions in TS 3.8.3 and the shutdown evaluation in FSAR Appendix 19E (Ref. 3).

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case Design Basis Accidents which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The Class 1E UPS inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or postulated DBA. The battery powered inverters provide an uninterruptible supply of AC electrical power to the Class 1E AC instrument and control buses, even if the normal power supply from the 480 VAC is deenergized. OPERABILITY of the inverters requires that the Class 1E instrument and control buses be powered by the inverter with output voltage and frequency within tolerances, and the power input to the inverter from a 250 VDC station battery. This ensures the availability of sufficient inverter power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents, inadvertent reactor vessel draindown).

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BASES

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Class 1E UPS inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.3, “Inverters – Operating.”

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2.1. A.2.2, A.2.3, and A.2.4

If one or more required (per LCO 3.8.6, Distribution Systems – Shutdown) inverters are inoperable, the remaining OPERABLE inverters may be capable of supporting required features to allow continuation of fuel movement and operations with a potential for draining the reactor vessel. By allowance of the option to declare required features inoperable with associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs’ Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., suspend movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)). Suspending positive reactivity additions that

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BASES

ACTIONS (continued)

could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a voltage regulating transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and Class 1E AC instrument and control buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the Class 1E AC instrument and control buses. The 7 day Frequency takes into account the redundant capability of the inverters, and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. FSAR Chapter 6, “Engineered Safety Features.”
2. FSAR Chapter 15, “Accident Analysis.”
3. FSAR Appendix 19E, “Shutdown Evaluation.”

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 Distribution Systems – Operating

BASES

BACKGROUND

The onsite Class 1E and DC and UPS electrical power distribution system is divided by division into four independent divisions of AC and DC electrical power distribution subsystems (Divisions A, B, C, and D).

The Class 1E AC distribution Divisions A and D each consists of one 208/120 V bus. The Class 1E AC distribution Divisions B and C each consists of two 208/120 V buses. The buses are normally powered from separate inverters which are connected to the respective division Class 1E battery banks. The backup source provided for each division for the Class 1E AC instrument and control buses is a Class 1E voltage regulating transformer providing regulated output to the Class 1E AC instrument and control buses through a static transfer switch and a manual bypass switch. Power to the transformer is provided by the nonsafety related Main AC Power System. Additional description of this system may be found in the Bases for Specification 3.8.3, “Inverters – Operating.”

The Class 1E DC distribution Divisions A and D each consists of one 250 VDC bus. The Class 1E DC distribution Divisions B and C each consists of two 250 VDC buses. The buses for the four Divisions are normally powered from their associated Division battery chargers. The backup source for each Class 1E DC bus is its associated Class 1E battery bank. Additionally, there is one installed spare Class 1E battery bank and one installed spare Class 1E battery charger, which can provide backup power to a Class 1E DC bus in the event that one of the battery banks or one of the chargers is out of service. Additional description of this system may be found in the Bases for Specification 3.8.1, “DC Sources Operating.”

The list of all required distribution buses and/or panels is presented in Table B 3.8.5-1 and shown in FSAR Section 8.3.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in FSAR Chapter 6 (Ref. 2) and FSAR Chapter 15 (Ref. 3), assume engineered safety features (ESFs) are OPERABLE. The Class 1E AC instrument and control and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the ESFs so that the fuel, Reactor Coolant System (RCS) and containment design limits are not exceeded.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

These limits are discussed in more detail in the Bases for Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

The OPERABILITY of the Class 1E AC instrument and control and DC electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least three of the four Divisions of Class 1E AC instrument and control and DC power distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite and onsite AC power sources; and

b. A worst case single failure.

The Class 1E AC and DC electrical power distribution system satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The required power distribution subsystems listed in Table B 3.8.5-1 ensure the availability of Class 1E AC instrument and control and DC electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The Division A, B, C, and D Class 1E AC instrument and control and DC electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Division A, B, C, and D AC instrument and control and DC electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of the ESFs is not defeated. Three of the four Class 1E AC instrument and control and DC power distribution subsystems are capable of providing the necessary electrical power to the associated ESF components. Therefore, a single failure within any subsystem or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

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BASES

LCO (continued)

OPERABLE Class 1E DC electric power distribution subsystems require the associated buses, motor control centers, and electrical circuits to be energized to their proper voltage from either the associated battery bank or charger. The spare battery bank and/or chargers may be used by one subsystem for OPERABILITY. OPERABLE Class 1E AC instrument and control electrical power distribution subsystems require the associated buses to be energized to their proper voltages and frequencies from the associated inverter or voltage regulating transformer.

APPLICABILITY

The Class 1E AC instrument and control and DC electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The Class 1E AC instrument and control and DC electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for Specification 3.8.6, “Class 1E Distribution Systems – Shutdown.”

ACTIONS

A.1

With one Class 1E AC instrument and control division inoperable (for Division B or C, either one or both required distribution panels inoperable can cause that division to be inoperable) the remaining Class 1E AC instrument and control divisions have the capacity to support a safe shutdown and to mitigate all DBAs, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E AC instrument and control divisions have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement

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BASES

ACTIONS (continued)

balancing the risks of operation without one AC instrument and control division against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

This 6 hour limit is shorter than Completion Times allowed for most supported systems which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 6 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 6 hour Completion Time takes into account the importance to safety of restoring the Class 1E AC instrument and control division to OPERABLE status, the passive design of the ESF systems, the redundant capability afforded by the other OPERABLE Class 1E AC instrument and control divisions, and the low probability of a DBA occurring during this period which requires more than two OPERABLE AC instrument and control divisions.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System actuation division (LCO 3.3.15, Engineered Safety Feature Actuation System (ESFAS) Actuation Logic - Operating). Depending on the nature of the AC instrument and control inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS division is similar to loss of one AC instrument and control division. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

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BASES

ACTIONS (continued)

B.1

With one Class 1E DC electrical power distribution subsystem division inoperable (for Division B or C, either one or more required buses or distribution panels inoperable can cause that division to be inoperable), the remaining divisions have the capacity to support a safe shutdown and to mitigate all DBAs, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining divisions have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one division against the risks of a forced shutdown. Additionally, the completion time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System division (LCO 3.3.15, Engineered Safety Feature Actuation System (ESFAS) Actuation Logic – Operating). Depending on the nature of the DC electrical power distribution subsystem inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS division is similar to loss of one DC electrical power distribution subsystem division. In both cases, actuation of the safety functions associated with one of the four divisions may no longer be available.

This 6 hour limit is shorter than Completion Times allowed for most supported systems which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 6 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected division; and

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BASES

ACTIONS (continued)

c. The potential for an event in conjunction with a single failure of a redundant component.

C.1

With two AC instrument and control divisions inoperable, the remaining OPERABLE divisions are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, one required division of AC instrument and control must be restored to OPERABLE status within 2 hours by powering the division from the associated inverter via inverted DC, inverter using internal AC source, or Class 1E voltage regulating transformer.

Condition C represents two AC instrument and control divisions without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptable power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining divisions and restoring power to one affected division.

This 2 hour time limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate AC instrument and control power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, which would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue);

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

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BASES

ACTIONS (continued)

The 2 hour Completion Time takes into account the importance to safety of restoring the AC instrument and control division to OPERABLE status, the redundant capability afforded by the other OPERABLE divisions, and the low probability of a DBA occurring during this period.

D.1

With two DC electrical power distribution subsystem divisions inoperable, the remaining DC electrical power distribution subsystem divisions are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem division could result in the minimum required ESF functions not being supported. Therefore, one required DC division must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition D represents two divisions without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining divisions and restoring power to one affected division.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected divisions; and

c. The potential for an event in conjunction with a single failure of a redundant component.

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BASES

ACTIONS (continued)

The 2 hour Completion Time for one DC electrical power distribution subsystem division is consistent with Regulatory Guide 1.93 (Ref. 4).

E.1 and E.2

If the inoperable division(s) cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to MODE 5 where the probability and consequences on an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With two inoperable divisions that result in a loss of safety function, adequate core cooling, containment OPERABILITY, and other vital functions for DBA mitigation would be compromised, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

This Surveillance verifies that the Class 1E AC instrument and control and DC electrical power distribution subsystems are functioning properly, with the required circuit breakers and switches properly aligned. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the Class 1E AC instrument and control and DC electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

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BASES

REFERENCES

1. FSAR Section 8.3.2, “DC Power Systems.”
2. FSAR Chapter 6, “Engineering Safety Features.”
3. FSAR Chapter 15, “Accident Analyses.”
4. Regulatory Guide 1.93, “Availability of Electric Power Sources,” U.S. Nuclear Regulatory Commission, December 1974.

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Table B 3.8.5-1 (page 1 of 1)
Class 1E AC and DC Electrical Power Distribution System

TYPE	VOLTAGE	DIVISION A	DIVISION B	DIVISION C	DIVISION D
DC Buses	250 Vdc	IDSA-DS-1	IDSB-DS-1 IDSB-DS-2	IDSC-DS-1 IDSC-DS-2	IDSD-DS-1
DC Distribution Panels	250 Vdc	IDSA-DD-1 IDSA-DK-1	IDSB-DD-1 IDSB-DK-1	IDSC-DD-1 IDSC-DK-1	IDSD-DD-1 IDSD-DK-1
AC Instrumentation and Control Panels	120 Vac	IDSA-EA-1	IDSB-EA-1 IDSB-EA-3	IDSC-EA-1 IDSC-EA-3	IDSD-EA-1

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Distribution Systems – Shutdown

BASES

BACKGROUND

A description of the Class 1E AC instrument and control and Class 1E DC electrical power distribution system is provided in the Bases for Specification 3.8.5, “Distribution Systems – Operating.”

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in FSAR Chapter 6 (Ref. 1) and FSAR Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The Class 1E AC and DC electrical power sources and associated power distribution systems are designed to provide sufficient capacity, redundancy, and reliability to ensure the availability of necessary power to the engineered safety features (ESF) systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the minimum Class 1E AC and DC electrical power sources and associated power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

The Class 1E AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components—all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

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BASES

LCO (continued)

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents), including single failure. To demonstrate robust design, a single failure is assumed, consistent with the single failure assumptions in TS 3.8.5 and the shutdown evaluation in FSAR Appendix 19E (Ref. 4).

APPLICABILITY

The Class 1E AC instrument and control and DC electrical power distribution subsystems are required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The Class 1E AC instrument and control and DC electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.5, “Distribution Systems – Operating.”

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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BASES

ACTIONS (continued)

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If one or more required Class 1E DC or Class 1E AC instrument and control bus electrical power distribution subsystems are inoperable, the remaining OPERABLE divisions may be capable of supporting required features to allow continuation of fuel movement and/or operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions will be implemented in accordance with the affected equipment LCO Required Actions. In many instances this would likely involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions will minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC instrument and control and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This Surveillance verifies that the Class 1E AC instrument and control and DC electrical power distribution subsystems are functioning properly, with the required circuit breakers and switches properly aligned. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the electrical power distribution subsystems and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. FSAR Chapter 6, “Engineered Safety Features.”
2. FSAR Chapter 15, “Accident Analysis.”
3. FSAR Section 8.3.2, “DC Power Systems.”
4. FSAR Appendix 19E, “Shutdown Evaluation.”

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Battery Parameters

BASES

BACKGROUND

LCO 3.8.7, Battery Parameters, delineates the limits on electrolyte temperature, electrolyte level, float voltage, and float current for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.1, "DC Sources – Operating," and LCO 3.8.2, "DC Sources – Shutdown." In addition to the limitations of this Specification, the licensee controlled program also implements a program specified in Specification 5.5.11 for monitoring various battery parameters that is based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice For Maintenance, Testing, And Replacement Of Vented Lead-Acid Batteries For Stationary Applications" (Ref. 3).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in FSAR Chapter 6 (Ref. 1), and FSAR Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for safety related and vital control instrumentation loads including monitoring and main control room emergency lighting during all MODES of operation. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least three of the four Divisions of DC sources OPERABLE during accident conditions, in the event of:

a. An assumed loss of all offsite and onsite AC power sources; and

b. A worst case single failure.

Battery parameters satisfy the Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with

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BASES

LCO (continued)

limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the licensee controlled program is conducted as specified in Specification 5.5.11.

APPLICABILITY

The battery parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery parameter limits are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.1, and LCO 3.8.2.

ACTIONS

A.1, A.2, and A.3

With one or more cells in one or more batteries in one Division < 2.07 V, the battery cell is degraded. Within 2 hours verification of the required battery charger, OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.1.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.7.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one or more batteries < 2.07 V, and continued operation is permitted for a limited period up to 24 hours.

Since the Required Actions only specify “perform,” a failure of SR 3.8.1.1 or SR 3.8.7.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.7.1 is failed then there is not assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

B.1 and B.2

One or more batteries in one Division with float > 2 amps indicate that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is operating in the current limit mode. Condition A addresses charger inoperability. If the

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ACTIONS (continued)

charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 24 hours (Required Action B.2). The battery must therefore be declared inoperable.

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than 2.07 V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than 2.07 V there is good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to a temporary loss of the battery charger.

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If the condition is due to one or more cells in a low voltage condition but still greater than 2.07 V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and 24 hours is a reasonable time prior to declaring the battery inoperable.

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.1.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

C.1, C.2, and C.3

With one or more batteries in one Division with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to

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ACTIONS (continued)

perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.11, Battery Monitoring and Maintenance Program). They are modified by a note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.11.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from Annex D of IEEE Standard 450-1995. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing the batteries may have to be declared inoperable and the affected cells replaced.

D.1

With one or more batteries in one Division with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met.

E.1

With one or more batteries in two or more Divisions with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. With redundant batteries involved this potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters on non-redundant batteries not within limits are therefore not appropriate, and the parameters must be restored to within limits in three Divisions within 2 hours.

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BASES

ACTIONS (continued)

F.1

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or more batteries in one Division with one or more battery cells float voltage less than 2.07 V and float current greater than 2 amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 3). The 7 day Frequency is consistent with IEEE-450 (Ref. 3).

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.1.1. When this float voltage is not maintained the Required Actions of LCO 3.8.1 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of 2 amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

SR 3.8.7.2 and SR 3.8.7.5

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 264.0 V at the battery terminals, or 2.20 Volts per cell. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 Volts per cell, are addressed in Specification 5.5.11. SRs 3.8.7.2 and 3.8.7.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of 2.07 V. The Frequency for cell voltage verification

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BASES

SURVEILLANCE REQUIREMENTS (continued)

every 31 days for pilot cell and 92 days for each connected cell is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. The Frequency is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 60°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provided the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. The Frequency is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.6

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.7.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.1.3. This Surveillance may be performed during any plant condition with the spare battery and charger providing power to the bus.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

The modified performance discharge test is a test, in the "as found" condition, of a battery's capacity with the discharge rate(s) modified to encompass every portion of the battery duty cycle. This allows the performance test to be accomplished in the minimum amount of time while still demonstrating the high rate capability of the battery to meet

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BASES

SURVEILLANCE REQUIREMENTS (continued)

duty cycle requirements. Thus, a modified performance test satisfies both the requirement to perform periodic performance tests as well as the requirement to perform periodic service tests.

The battery terminal voltage for the modified performance discharge test is required to remain above the minimum battery terminal voltage specified in the battery service test for the duration of the service test portion of the test. Initial conditions for the modified performance discharge test are identical to those specified for a service test.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 3) and IEEE-485 (Ref. 4). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this 80% limit.

The Surveillance Frequency for this test is normally 60 months, or 25% of expected life, whichever is less. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 3), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is below 90% of the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 3) and Information Notice 2013-05 (Ref. 5).

REFERENCES

1. FSAR Chapter 6, "Engineered Safety Features."

2. FSAR Chapter 15, "Accident Analyses."

3. IEEE-450 1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

4. IEEE-485 1983, "IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations."

5 NRC Information Notice 2013-05, March 19, 2013.

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B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentration of the Reactor Coolant System (RCS), the refueling cavity, and the fuel transfer canal during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by procedures.

GDC 26 of 10 CFR 50, Appendix A requires that two independent reactivity control systems of different design principles be provided (Ref. 3). One of these systems, the Passive Core Cooling System (PXS), is capable of holding the core subcritical under safe shutdown conditions as described in FSAR Section 7.4 (Ref. 4).

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled down and depressurized, the vessel head is unbolted and slowly removed. The refueling cavity and the fuel transfer canal are then flooded with borated water from the In-containment Refueling Water Storage Tank (IRWST) by the use of the Spent Fuel Pool Cooling System (SFS).

During refueling, the water volumes in the RCS, the fuel transfer canal and the refueling cavity are contiguous. However, the soluble boron concentration is not necessarily the same in each volume. If additions of boron are required during refueling, the Chemical and Volume Control System (CVS) provides the borated makeup.

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BASES

BACKGROUND (continued)

The pumping action of the Normal Residual Heat Removal System (RNS) in the RCS, the SFS pumps in the spent fuel pool and refueling cavity, and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the fuel transfer canal. The RNS is in operation during refueling to provide forced circulation in the RCS, while the SFS is in operation to cool and purify the spent fuel pool and refueling cavity. Their operation assists in maintaining the boron concentration in the RCS, the refueling cavity, and fuel transfer canal above the COLR limit.

APPLICABLE SAFETY ANALYSES

The boron concentration limit, specified in the COLR, is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% Δk/k margin of safety is established during refueling.

These limits are developed using the methodologies identified in Technical Specification 5.6.3.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that boron concentration be maintained within limit in the RCS, the refueling cavity and the fuel transfer canal while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} of ≤ 0.95. Above MODE 6, LCO 3.1.1, “SHUTDOWN MARGIN (SDM)” ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are applicable to the fuel transfer canal and the refueling cavity only when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists from those volumes.

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BASES

ACTIONS

A.1

Continuation of positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the plant in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling cavity, or the fuel transfer canal is less than its limit, all operations involving positive reactivity additions must be suspended immediately.

Suspension of positive reactivity additions shall not preclude completion of actions to establish a safe condition, including moving a component to a safe position.

A. 2

In addition to immediately suspending positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique design basis accident (DBA) must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator shall begin boration with the best source available for plant operations.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR verifies that the coolant boron concentration in the RCS, the refueling cavity and the fuel transfer canal is within the COLR limit. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time between verifications of the boron concentration. The surveillance interval is based on operating experience, isolation of unborated water sources in accordance with LCO 3.9.2, and the availability of the source range neutron flux monitors required by LCO 3.9.3.

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BASES

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”
2. NS-57.2, ANSI/ANS-57.2-1983, Section 6.4.2.2.3, American Nuclear Society, American National Standard, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” 1983.
3. 10 CFR 50, Appendix A, GDC 26.
4. FSAR Section 7.4, “Systems Required for Safe Shutdown.”

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B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Flow Paths

BASES

BACKGROUND

During MODE 6 operation, all flow paths for reactor makeup water sources containing unborated water which are connected to the Reactor Coolant System (RCS) must be closed to prevent an unplanned dilution of the reactor coolant. At least one isolation valve in each flow path must be secured in the closed position.

The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition, made by reducing the boron concentration, is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution event.

APPLICABLE SAFETY ANALYSES

The possibility of an unplanned boron dilution event (Ref. 1) in MODE 6 is precluded by adherence to this LCO which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portions of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required in MODE 6.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that at least one valve in each flow path to the RCS from unborated water sources be secured in the closed position to prevent unplanned boron dilution during MODE 6 and, thus, avoid a reduction in SHUTDOWN MARGIN.

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an unplanned boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

In MODES 1 through 5, the requirements of LCO 3.1.9, “Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves,” apply.

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BASES

ACTIONS

The ACTIONS Table has been modified by a Note which allows separate Condition entry for each unborated water source flow path.

A.1

Preventing unplanned dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position verifies that the valves cannot be inadvertently opened. The Completion Time of "Immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

A.2

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to verify that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

These valves are to be secured closed to isolate possible dilution flow paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water source flow paths are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This surveillance demonstrates that the valves are closed. The 31 day Frequency is based on engineering judgement and is considered reasonable in view of other administrative controls that will verify that the valve opening is an unlikely possibility.

REFERENCES

1. FSAR Chapter 15, "Accident Analyses."
2. NUREG-0800, Standard Review Plan, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the RCS."

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B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used to monitor the core reactivity during refueling operations. The source range neutron flux monitors are part of the Protection and Safety Monitoring System (PMS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (0.1 to $1x10^{+6}$ cps) with a 5% instrument accuracy. The detectors also provide continuous visual and audible indication in the main control room and an audible alarm in the main control room and containment building.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as those associated with an improperly loaded fuel assembly. During initial fuel loading, or when otherwise required, temporary neutron detectors may be used to provide additional reactivity monitoring (Ref. 2). The potential for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2 (Ref. 1).

The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires two source range neutron flux monitors to be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. The source range neutron flux monitor OPERABILITY includes only a visual monitoring function (audible alarm is not required for OPERABILITY).

APPLICABILITY

In MODE 6, the source range neutron flux monitors are required to be OPERABLE to determine possible changes in core reactivity. There are no other direct means available to monitor the core reactivity conditions. In MODES 2, 3, 4, and 5, the source range detectors and associated

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BASES

APPLICABILITY (continued)

circuitry are also required to be OPERABLE by LCO 3.3.2, "Reactor Trip System (RTS) Source Range Instrumentation," and LCO 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Function 17.

ACTIONS

A.1 and A.2

Redundancy has been lost if only one source range neutron flux monitor is OPERABLE. Since these instruments are the only direct means of monitoring core reactivity conditions, positive reactivity additions, and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Performance of Required Action A.1 shall not preclude completion of actions to establish a safe condition.

B.1

If no source range neutron flux monitors are OPERABLE, actions to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

If no source range neutron flux monitors are OPERABLE, there is no direct means of detecting changes in core reactivity. However, since positive reactivity additions are discontinued, the core reactivity condition is stabilized and no changes are permitted until the source range neutron flux monitors are restored to OPERABLE status. This stable condition is confirmed by performing SR 3.9.1.1 to verify that the required boron concentration exists.

The Completion Time of 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable considering the low probability of a change in core reactivity during this time period.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is the comparison of the indicated parameter values monitored by each of these instruments. It is based on the assumption that the two indication channels should be consistent for the existing core conditions. Changes in core geometry due to fuel loading can result in significant differences between the source range channels, however each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for these same instruments in LCO 3.3.2, “Reactor Trip System (RTS) Source Range Instrumentation” and LCO 3.3.8, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” Function 17.

SR 3.9.3.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consisting of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer’s data. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at a 24 month Frequency.

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”
2. FSAR Section 14.2.7.1, “Initial Fuel Loading.”

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B 3.9 REFUELING OPERATIONS

B 3.9.4 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in containment, refueling cavity, fuel transfer canal, and spent fuel pool to retain iodine fission product activity in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within the values reported in FSAR Chapter 15 (Ref. 1).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling cavity and the refueling canal is an initial condition design parameter in the analysis of a fuel-handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 2).

The fuel handling accident analysis inside containment is described in Reference 3. This analysis assumes a minimum water level of 23 feet.

In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, there may be less than 23-feet of water above the top of the fuel bundle and the surface of the water, indicated by the width of the bundle. This slight reduction in water depth does not adversely affect the margin of conservatism associated with the assumed pool scrubbing factor of 200 for iodine.

Refueling Cavity Water Level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within the values calculated in Reference 3.

APPLICABILITY

Refueling Cavity Water Level is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of radioactive release due to a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.5, “Spent Fuel Pool Water Level.”

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BASES

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement to safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4 1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 3).

The Frequency of 24 hours is based on engineering judgement and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions which make significant unplanned level changes unlikely.

REFERENCES

1. FSAR Chapter 15, “Accident Analysis.”
2. Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”
3. FSAR Section 15.7.4, “Fuel Handling Accident.”

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B 3.9 REFUELING OPERATIONS

B 3.9.5 Decay Time

BASES

BACKGROUND

The movement of irradiated fuel assemblies in the reactor pressure vessel requires allowing at least 48 hours for radioactive decay time before fuel assembly handling can be initiated. During fuel handling, this ensures that sufficient radioactive decay has occurred in the event of a fuel handling accident (Refs. 1 and 2). Sufficient radioactive decay of short-lived fission products would have occurred to limit offsite doses from the accident to within the values reported in FSAR Chapter 15.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the radioactivity decay time is an initial condition design parameter in the analysis of a fuel-handling accident inside containment or in the fuel handling area inside the auxiliary building, as postulated by Regulatory Guide 1.183 (Ref. 1).

The fuel handling accident analysis inside containment or in the fuel handling area inside the auxiliary building is described in Reference 2. This analysis assumes a minimum radioactive decay time of 48 hours.

Radioactive decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum radioactive decay time of 48 hours is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment or in the fuel handling area inside the auxiliary building are within the values calculated in Reference 2.

APPLICABILITY

Radioactive decay time is applicable when moving irradiated fuel assemblies in the reactor pressure vessel. The LCO minimizes the possibility of radioactive release due to a fuel handling accident that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved, there can be no significant radioactivity release as a result of a postulated fuel handling accident. If irradiated fuel assemblies are being moved outside of the reactor pressure vessel, then they were previously assured of having been subcritical for more than 48 hours before being moved from the reactor pressure vessel. Requirements for fuel handling accidents in the spent fuel pool are also covered by LCO 3.7.5, “Spent Fuel Pool Water Level.”

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BASES

ACTIONS

A.1

With a decay time of less than 48 hours, all operations involving movement of irradiated fuel assemblies within the reactor pressure vessel shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement to safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

Verification that the reactor has been subcritical for $\geq$ 48 hours prior to movement of irradiated fuel in the reactor pressure vessel ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Specifying radioactive decay time limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident (Ref. 2).

REFERENCES

1. Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

2. FSAR Section 15.7.4, “Fuel Handling Accident.”

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16.2 Design Reliability Assurance Program

See Section 17.4 for information on the AP1000 Design Reliability Assurance Program (D-RAP).

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16.3 Investment Protection

16.3.1 Investment Protection Short-Term Availability Controls

The importance of nonsafety-related systems, structures and components in the AP1000 has been evaluated. The evaluation uses PRA insights to identify systems, structures and components that are important in protecting the utilities investment and for preventing and mitigating severe accidents. To provide reasonable assurance that these systems, structures and components are functional during anticipated events short-term availability controls are provided. These investment protection systems, structures and components are also included in the D-RAP/OPRAAs (refer to Section 17.4), which provides confidence that availability and reliability are designed into the plant and that availability and reliability are maintained throughout plant life through the use of reliability assurance activities as listed in Subsection 17.4.4. Technical Specifications are not required for these systems, structures and components because they do not meet the selection criteria applied to the AP1000 (refer to Subsection 16.1.1).

Table 16.3-1 lists nonsafety-related systems, structures and components that have investment protection short-term availability controls which are contained in the Technical Requirements Manual (TRM). The portions of the TRM containing these short-term availability controls are incorporated into this subsection as Tier 2 information. These are identified in the TRM Table of Contents with Scope as “DCD Tier 2.” These short-term availability controls define:

– Equipment that shall be FUNCTIONAL
– Operational MODES when the equipment shall be FUNCTIONAL
– Testing and inspections that are used to demonstrate the equipment’s FUNCTIONALITY
– Operational MODES that are used for planned maintenance operations
– Remedial actions that are taken if the equipment is not FUNCTIONAL

Station procedures govern and control the FUNCTIONALITY of investment protection systems, structures, and components, in accordance with the Tier 2 information within the TRM, and provide the operating staff with instruction for implementing required actions when FUNCTIONALITY requirements are not met. Procedure development is addressed in Section 13.5. TRM Use and Application Sections 1.2, 1.3, and 1.4, as well as TRM Applicability Sections TR 3.0.1, TR 3.0.2, TRS 3.0.1, TRS 3.0.2, and TRS 3.0.3 constitute a portion of these procedural controls, but are not considered Tier 2 information. These, and other procedural controls, are identified in the TRM Table of Contents with Scope as “Administrative.”

TRM Section 1.1, Definitions, contains defined terms that appear in capitalized type. TRM Section 1.1, Definitions, is also incorporated into this subsection as Tier 2 information.

TRM material that is incorporated as Standard SAR Content, is noted in the TRM Table of Contents with Scope as "Standard SAR Content."

16.3.2 Combined License Information

The procedure to control the functionality of investment protection systems, structures and components is addressed in Subsection 16.3.1.

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Table 16.3-1 List of Investment Protection Short-term Availability Controls

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TRM 3.8.4	Non Class 1E DC and UPS System (EDS)

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TR 3.7.3	Component Cooling Water System (CCS) – Reactor Coolant System (RCS) Open	DCD Tier 2
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TRM 1.0 USE AND APPLICATION

TRM 1.1 Definitions

- NOTE -

1. Technical Requirements (TR) Definitions follow those terms defined in Technical Specifications (TS) Section 1.1. Only definitions specific to the TRM will be defined in this section.

2. The defined terms appear in capitalized type and are applicable throughout the TRM.

3. When a term is defined in both the TS and the TRM, the TRM definition takes precedence within the TRM.

4. When a term defined in TS Section 1.1 refers to “OPERABLE” or “OPERABILITY” its use within the TRM is understood to apply the TRM requirement-specific defined terms (i.e., FUNCTIONAL or FUNCTIONALITY) as required in each Technical Requirement.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Technical Requirement that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
FUNCTIONAL - FUNCTIONALITY	A system, subsystem, train, component, or device shall be FUNCTIONAL or have FUNCTIONALITY when it is capable of performing its specified function(s), as set forth in the current licensing basis and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified function(s) are also capable of performing their related support function(s).

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TRM 1.0 USE AND APPLICATION

TRM 1.2 Logical Connectors

Logical Connectors are discussed in Technical Specification Section 1.2 and are applicable throughout the Technical Requirements Manual (TRM). When TS Section 1.2 refers to “OPERABLE” or “OPERABILITY” its use as applied to the TRM is understood to apply the TRM requirement-specific defined terms (i.e., FUNCTIONAL or FUNCTIONALITY) as required in each Technical Requirement.

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TRM 1.0 USE AND APPLICATION

TRM 1.3 Completion Times

Completion Times are discussed in Technical Specification Section 1.3 and are applicable throughout the Technical Requirements Manual (TRM). When TS Section 1.3 refers to "OPERABLE" or "OPERABILITY" its use as applied to the TRM is understood to apply the TRM requirement-specific defined terms (i.e., FUNCTIONAL or FUNCTIONALITY) as required in each Technical Requirement. Similarly, when TS Section 1.3 refers to an "LCO" its use as applied to the TRM is understood to apply to Technical Requirements (TRs).

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TRM 1.0 USE AND APPLICATION

TRM 1.4 Frequency

Frequency is discussed in Technical Specification Section 1.4 and is applicable throughout the Technical Requirements Manual (TRM). When TS Section 1.4 refers to “OPERABLE” or “OPERABILITY” its use as applied to the TRM is understood to apply the TRM requirement-specific defined terms (i.e., FUNCTIONAL or FUNCTIONALITY) as required in each Technical Requirement. Similarly, when TS Section 1.4 refers to an “LCO” its use as applied to the TRM is understood to apply to Technical Requirements (TRs). TS Section 1.4 references to LCO 3.0.4 and SR 3.0.4 are not applicable to the TRM.

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TRM 3.0 TECHNICAL REQUIREMENT (TR) APPLICABILITY

TR 3.0.1 Each TR shall be met during the MODES or other specified conditions in the Applicability, except as provided in TR 3.0.2.

TR 3.0.2 Upon discovery of a failure to meet a TR, the Required Actions of the associated Conditions shall be met.

If the TR is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

TR 3.0.3 When a TR is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS:

a. Initiate action to restore compliance with the TR or associated ACTIONS immediately;

b. Initiate action to assess and manage the risk of the resulting unit configuration immediately, and

- NOTE -
TR 3.0.3.c shall be completed if TR 3.0.3 is entered.

c. Enter the circumstances of entry into TR 3.0.3 into the Corrective Action Program within 24 hours.

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TRM 3.0 TECHNICAL REQUIREMENT SURVEILLANCE (TRS) APPLICABILITY

TRS 3.0.1	TRSs shall be met during the MODES or other specified conditions in the Applicability for the individual TR, unless otherwise stated in the TRS. Failure to meet a TRS, whether such failure is experienced during the performance of the TRS or between performances of the TRS, shall be failure to meet the TR. Failure to perform a TRS within the specified Frequency shall be failure to meet the TR. TRSs do not have to be performed on nonfunctional equipment or variables outside specified limits.
TRS 3.0.2	The specified Frequency for each TRS is met if the TRS is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as “once,” the above interval extension does not apply. If a Completion Time requires periodic performance on a “once per . . .” basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this TRS are stated in the individual TRSs.
TRS 3.0.3	If it is discovered that a TRS was not performed within its specified Frequency, then compliance with the requirement to declare theTR not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, which ever is greater. This delay period is permitted to allow performance of the TRS. A risk evaluation shall be performed for any TRS delayed greater than 24 hours and the risk impact shall be managed. If the TRS is not performed within the delay period, the TR must immediately be declared not met, and the applicable Condition(s) must be entered. When the TRS is performed within the delay period, and the TRS is not met, the TR must immediately be declared not met, and the applicable Condition(s) must be entered.

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TRM 3.2 POWER DISTRIBUTION LIMITS

TRM 3.2.1 On-Line Power Distribution Monitoring System (OPDMS) – Input Parameters

TR 3.2.1 OPDMS Input Parameters listed in Table TR 3.2.1-1 shall be FUNCTIONAL.

APPLICABILITY: MODE 1
MODE 2 with keff ≥ 1.0

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required input parameters not FUNCTIONAL.	A.1 Restore required parameter(s) to FUNCTIONAL status.	7 days

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.2.1.1	Perform CHANNEL CHECK on each required channel.	7 days

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Table TR 3.2.1-1 (page 1 of 1)
OPDMS Input Parameters

	PARAMETER	MINIMUM DATA REQUIRED
1.	Individual Control Rod Position	Digital Indicated Rod Position (DRPI) signal for each control rod
2.	Control Rod OPERABILITY Status	Indication of OPERABILITY status of each control rod
3.	Source of Control Rod Position Data	Indication of source of control rod position data for each rod
4.	Inoperable / misaligned Control Rod Position	Indication of position of inoperable / misaligned control rods

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TRM 3.2.1 On-Line Power Distribution Monitoring System (OPDMS)-Input Parameters

Bases

The core monitoring system is the component of the OPDMS that calculates from present plant data the core power distribution and reactivity and determines margin to core thermal limits and required SHUTDOWN MARGIN. The core monitoring system is the Westinghouse BEACON™ Core Monitoring System. The technology utilized is licensed by the NRC for direct monitoring of core parameters to limits. The core monitoring system requires present plant data of reactor THERMAL POWER, control rod position and inlet temperature to provide monitoring of SHUTDOWN MARGIN (SDM) in MODES 1 and 2 with $k_{eff} \geq 1.0$ per TS 3.2.5. Valid data from the Incore Instrumentation System neutron detectors is required in addition to that required for SDM to monitor reactor core margins to thermal limits in MODE 1 with THERMAL POWER ≥ 25%.

In addition to the minimum required inputs for the core monitoring system as governed by TS 3.2.5, additional inputs are also used. The core monitoring system primarily uses control rod position from the group demand indicators to perform the necessary calculations. In order to calculate margins to thermal and SDM limits correctly in the event control rods are misaligned or inoperable, it is necessary that operations personnel update the data used by the core monitoring system so that the correct position and operability status is used. An interface is provided through control room work stations to facilitate this action. Options to enter operability status, a fixed position or the instruction for the core monitoring system to use the individual rod position from the DRPI signal are available.

These additional input signals are not continuously monitored by the core monitoring system and loss or degradation of this data will not result in an alarm. Therefore, a CHANNEL CHECK of this additional data is provided to ensure the data transfer is functioning correctly and the data is of adequate quality. Operating experience with similar core monitoring system inputs indicates a frequency of 7 days for the data check is adequate.

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TRM 3.3 INSTRUMENTATION

TRM 3.3.1 Diverse Actuation System (DAS) Automatic Actuation

TR 3.3.1 DAS automatic actuation functions and DAS automatic actuation initiation signals listed in Table TR 3.3.1-1 shall be FUNCTIONAL.

APPLICABILITY: According to Table TR 3.3.1-1.

ACTIONS

- NOTE -

Separate condition entry is allowed for each actuation function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more automatic actuation functions with one or more required channels not FUNCTIONAL.	A.1 Restore required channel(s) to FUNCTIONAL status.	14 days

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TECHNICAL REQUIREMENT SURVEILLANCE

- NOTE -
Refer to Table TR 3.3.1-1 to determine which TRSs apply for each actuation function.

SURVEILLANCE		FREQUENCY
TRS 3.3.1.1	Perform CHANNEL CHECK on each required channel.	30 hours
TRS 3.3.1.2	Perform CHANNEL OPERATIONAL TEST (COT) on each required channel.	24 months
TRS 3.3.1.3	Perform CHANNEL CALIBRATION on each required channel.	24 months
TRS 3.3.1.4	--- **– NOTE –** Only required to be met in MODE 1. --- Verify rod drive MG set field control relays open on demand.	24 months

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Table TR 3.3.1-1 (page 1 of 1)
DAS Automatic Actuation Functions

	ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	TECHNICAL REQUIREMENT SURVEILLANCE
1.	SG Wide Range Level – Low[a]	1,2,3,4[b]	1 per SG	TRS 3.3.1.1 TRS 3.3.1.2 TRS 3.3.1.3 TRS 3.3.1.4
2.	Hot Leg Temperature – High[a]	1,2,3,4,5[c]	1 per Hot Leg	TRS 3.3.1.1 TRS 3.3.1.2 TRS 3.3.1.3 TRS 3.3.1.4
3.	Pressurizer Level - Low	1,2,3,4,5[d]	2	TRS 3.3.1.1 TRS 3.3.1.2 TRS 3.3.1.3
4.	Containment Temperature - High	1,2,3,4	2	TRS 3.3.1.1 TRS 3.3.1.2 TRS 3.3.1.3

(a) DAS ATWS mitigation functions of reactor trip and turbine trip actuation are only required in MODE 1.

(b) With the RCS not being cooled by the normal residual heat removal system (RNS).

(c) With the RCS pressure boundary intact and pressurizer level ≥ 20%. Not required to be FUNCTIONAL in MODE 5 during RCS vacuum fill operations.

(d) With the RCS not VENTED and CMT actuation on Pressurizer Level - Low not blocked.

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TRM 3.3.1 Diverse Actuation System (DAS) Automatic Actuation

BASES

The DAS automatic actuation functions include: (1) the DAS ATWS mitigation functions of reactor trip (i.e., MG set field control relay open), turbine trip, and passive residual heat removal heat exchanger (PRHR HX) actuation; and (2) the DAS Engineered Safety Features Actuation (ESFA) functions of PRHR HX actuation, CMT actuation, RCP trip, Passive Containment Cooling actuation, and selected containment isolation actuation.

The DAS ATWS mitigation function actuation provides ATWS mitigation capability on:

- SG Wide Range Level – Low; and
- Hot Leg Temperature – High.

This function is important based on 10 CFR 50.62 (ATWS Rule) and because it provides margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to actuate.

The DAS ESFA functions provide accident mitigation capability on:

- SG Wide Range Level – Low;
- Hot Leg Temperature – High;
- Pressurizer Level – Low; and
- Containment Temperature - High.

This function is important because it provides margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to actuate.

The DAS uses a 2-out-of-2 logic to actuate automatic functions. When a required channel is not FUNCTIONAL the automatic DAS function is not FUNCTIONAL. FSAR subsection 7.7.1.11 (Ref. 1) provides additional information. A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Technical Requirement may be entered independently for each non-FUNCTIONAL actuation function listed in Table TR 3.3.1-1.

Table TR 3.3.1-1 DAS function nominal trip setpoints are determined using the setpoint methodology referenced in FSAR subsection 7.7.1.11 (Ref. 1). The nominal trip setpoints in the setpoints program results report (Ref. 2) are used to support the necessary surveillance testing. The nominal trip setpoint is a digital value and, therefore, does not drift.

Automated operator aids may be used to facilitate performance of the CHANNEL CHECK. An automated tester may be used to facilitate performance of the CHANNEL OPERATIONAL TEST.

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BASES (continued)

The DAS ATWS mitigation function is required during MODE 1 when ATWS is a limiting event. Planned maintenance affecting the DAS ATWS mitigation function should be performed in MODES 3, 4, 5 or 6. These MODES are selected because the reactor is tripped in these MODES and ATWS cannot occur. The DAS ESFA mitigation functions are required during MODES in which the DAS provides backup to the automatic protection system functions as described in the following paragraphs, which is when automatic DAS ESFA is beneficial to the PRA results. Planned maintenance affecting overall FUNCTIONALITY of the ESFA mitigation functions should not be performed during a required MODE of operation.

The SG Wide Range Level – Low actuation function is required during MODES 1, 2, and 3 and during MODE 4 with the RCS not being cooled by the RNS. Planned maintenance affecting the SG Wide Range Level – Low actuation function should be performed during MODE 4 with the RCS being cooled by the RNS or during MODES 5 or 6, which is when the steam generators are no longer being relied upon as a heat sink.

The Hot Leg Temperature – High actuation function is required during MODES 1, 2, 3, and 4, and during MODE 5 when the RCS pressure boundary intact and pressurizer level ≥ 20%, which is when the PRHR HX can provide core cooling. This actuation function is not required during vacuum fill, which hinders natural circulation through the PRHR HX. Planned maintenance affecting the Hot Leg Temperature – High actuation function should be performed while the RCS pressure boundary is not intact, when natural circulation through the PRHR HX is inhibited and automatic PRHR HX actuation is not required.

The Pressurizer Level – Low actuation function is required during MODES 1, 2, 3, and 4, and during MODE 5 when the RCS is not VENTED and CMT actuation on Pressurizer Level – Low is not blocked, which is when automatic DAS actuation of the CMTs is required to provide borated water for RCS inventory makeup and reactivity control. Planned maintenance affecting the Pressurizer Level – Low actuation function should be performed during MODE 5 after a vent has been established or in MODE 6, when injection from the IRWST is available to provide RCS makeup and boration.

The Containment Temperature – High actuation function is required during MODES 1, 2, 3, and 4, which is when containment integrity is required. Planned maintenance affecting the Containment Temperature – High actuation function should be performed in MODES 5 or 6 when containment isolation is administratively controlled.

REFERENCES

1. FSAR Section 7.7.1.11, “Diverse Actuation System.”

2. APP-PMS-GS-001, “Protection and Safety Monitoring System and Diverse Actuation System Setpoints for the AP1000 Plants,” Revision 4, December 2024.

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TRM 3.3 INSTRUMENTATION

TRM 3.3.2 Feedwater Flow

TR 3.3.2 Caldon/Cameron Leading Edge Flow Meter (LEFM) CheckPlus system feedwater flow calorimetric input shall be FUNCTIONAL.

APPLICABILITY: MODE 1 with THERMAL POWER > 3366 MWt.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. LEFM CheckPlus system not FUNCTIONAL.	A.1	Initiate action to verify calorimetric power is utilizing feedwater flow input from venturi elements.	Immediately
	AND		
	A.2	Restore LEFM CheckPlus system to FUNCTIONAL status.	48 hours
B. Required action and associated Completion Time not met.	B.1	Initiate action to reduce THERMAL POWER to ≤ 3366 MWt.	Immediately

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TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.3.2.1	Perform CHANNEL CHECK on LEFM CheckPlus system feedwater flow calorimetric input.	24 hours
TRS 3.3.2.2	Perform CHANNEL CALIBRATION and preventative maintenance on the required plane in accordance with LEFM CheckPlus system technical manuals and recommendations.	Per technical manuals and recommendations

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TRM 3.3.2 Feedwater Flow

Bases

The plant operating instrumentation selected for feedwater flow measurement is a Caldon [Cameron] LEFM CheckPlus System. This selected plant operating instrumentation has documented instrumentation uncertainties to calculate a power calorimetric uncertainty that confirms the 1% uncertainty assumed for the initial reactor power in the safety analysis bounds the calculated calorimetric power uncertainty values. While the main feedwater flow measurement supports a 1% RTP uncertainty, the safety analysis uses a conservative 2% RTP uncertainty (i.e., initial power of 102% [3468 MWt]).

The Caldon/Cameron LEFM CheckPlus System consists of two ultrasonic flow meters (UFMs), one in each steam generator loop, to measure volumetric flow rates. Volumetric flow rate measurements are made by the UFM transmitters. Each UFM has two transmitters. Each transmitter is connected to one measurement plane. Each measurement plane is comprised of 4 acoustic measurement chords or paths. A measurement path consists of two transducers aiming at each other, through the pressure boundary transducer housing. An acoustic pulse is transmitted along each path and the receiver records the arrival of the pulse. Another pulse is transmitted in the opposite direction and the time for that pulse is also recorded. Since the speed of an acoustic pulse will increase in the direction of flow and will decrease when transmitted against the flow, the difference in the upstream and downstream transit times for the acoustic pulse provides information on flow velocity. Once the difference in travel times is determined, the average velocity of the fluid along the acoustic path can be determined. Therefore, the difference in transit time is proportional to the average velocity of the fluid along the acoustic path.

Both LEFM CheckPlus System Status registers return maintained integers to the Data Display and Processing System (DDS) representative of the UFM Status condition for Stem Generator Loops 1 and 2. The maintained Status integers are 0, 1, 2, and 3, representative of the individual UFM health. A UFM Status register returns a 0 when the system is operating without error. This indicates that the UFM is functioning with two measurement planes. The guaranteed instrument flow uncertainty associated with Status 0 is less than the 1% RTP uncertainty allowance assumed for the initial reactor power in the safety analysis. A UFM Status register returns a 1 when a minor alert is present. This indicates that minor maintenance is required, and that the system is functioning with two measurement planes. The guaranteed instrument flow uncertainty associated with Status 1 is less than the 1% RTP uncertainty allowance assumed for the initial reactor power in the safety analysis. A UFM Status register returns a 2 when a major alert is present. This indicates that a critical maintenance condition exists, and the system is functioning with only one measurement plane. The guaranteed instrument flow uncertainty increases in Status 2 but remains less than the 1% RTP uncertainty allowance assumed for the initial reactor power in the safety analysis. A UFM Status register returns a 3 when the system is not FUNCTIONAL. This indicates that a critical maintenance condition exists, and measurement uncertainty cannot be guaranteed.

The LEFM CheckPlus System may only be used as input to the feedwater flow power calorimetric when the guaranteed uncertainty of the UFMs is less than or equal to the bounding uncertainty assumed in the Power Calorimetric Measurement Uncertainty calculation.

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Bases (continued)

This provides a calorimetric input uncertainty that is less than the calorimetric power uncertainty allowance of 1% assumed for the initial reactor power in the safety analysis. As such, the UFM must be operating with a Status of 0, 1, or 2 to meet the uncertainty requirements. The LEFM CheckPlus System is considered FUNCTIONAL when both Loop 1 and Loop 2 UFMs are operating in Status 0, 1, or 2.

Operation at indicated core power levels at 3400 MWt (i.e., RATED THERMAL POWER) requires a calorimetric power uncertainty determination of less than 2.0%. This is only possible if the LEFM CheckPlus System feedwater flow instrumentation is available as the input to the calorimetric is FUNCTIONAL. The LEFM system measures and transmits data with lower uncertainty than the instrumentation from the feedwater flow venturi. The feedwater flow venturi input to the calorimetric will support operation above 3366 MWt for up to 48 hours while the LEFM system is not FUNCTIONAL. Upon the expiration of the 48-hour Completion Time, action is immediately initiated to reduce reactor core power to ≤ 3366 MWt.

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TRM 3.6 CONTAINMENT SYSTEMS

TRM 3.6.1 Passive Containment Cooling Water Storage Tank (PCCWST) and Spent Fuel Pool Makeup - Long Term Shutdown

TR 3.6.1 Long term makeup to the PCCWST and the spent fuel pool shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,

When irradiated fuel assemblies are stored in the spent fuel pool.

- NOTES -

1. Makeup flow path to the PCCWST is only required to be FUNCTIONAL when the core decay heat is > 4.0 MWt.
2. Makeup flow path to the spent fuel pool is only required to be FUNCTIONAL when irradiated fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required long term makeup to the PCCWST not FUNCTIONAL.	A.1 Restore required long term makeup to the PCCWST to FUNCTIONAL status.	14 days
B. Required long term makeup to the spent fuel pool not FUNCTIONAL.	B.1 Restore required long term makeup to the spent fuel pool to FUNCTIONAL status.	14 days

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TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.6.1.1	Verify water volume in the PCCAWST is > 885,000 gal.	31 days
TRS 3.6.1.2	Verify required PCS recirculation pump provides recirculation of the PCCWST at > 100 gpm.	92 days
TRS 3.6.1.3	Verify required PCS recirculation pump transfers > 100 gpm from the PCCAWST to the PCCWST. During this test, required PCS recirculation pump shall be powered from an ancillary diesel.	10 years

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TRM 3.6.1 Passive Containment Cooling Water Storage Tank (PCCWST) and Spent Fuel Pool Makeup - Long Term Shutdown

BASES

The PCS recirculation pumps provide long-term shutdown support by transferring water from the PCCAWST to the PCCWST and the spent fuel pool. The specified PCCAWST volume is sufficient to maintain PCS and spent fuel pool cooling during the 3 to 7 day time period following an accident. The specified PCCAWST volume includes the base amount of water needed to provide 135 gpm for four days (780,000 gal.) plus an additional volume intended to provide margin for potential instrument inaccuracies, mis-positioning of flow control valve, and makeup inventory for the fire protection system. After 7 days, water brought in from offsite allows the PCCWST to continue to provide PCS cooling and makeup to the spent fuel pool. This PCCWST makeup function is important because it supports long-term shutdown operation. A minimum availability of 90% is assumed for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The PCCWST and spent fuel pool makeup functions involve the use of one PCS recirculation pump, the PCCAWST and the line connecting the PCCAWST with the PCCWST and/or spent fuel pool. The flow path between the PCCAWST and PCS-V023 (PCS Recirculation Return Isolation Valve) is common to both makeup functions. Downstream of PCS-V023, the flow path to the spent fuel pool branches off from the flow path to the PCCWST. One PCS recirculation pump normally operates to recirculate the PCCWST. FSAR subsections 6.2.2 and 9.1.3 contain additional information on the long-term PCCWST and spent fuel pool makeup function.

The PCCWST and spent fuel pool makeup functions are required during MODES of operation when PCS and spent fuel pool cooling is required.

Planned maintenance should be performed on the redundant pump (i.e., the pump not required to be FUNCTIONAL). Planned maintenance affecting the PCCAWST that requires less than 72 hours to perform can be performed in any MODE of operation. Planned maintenance requiring more than 72 hours should be performed in MODE 5 or 6 when the core decay heat is ≤ 4.0 MWt and an alternate means of makeup to the spent fuel pool is available for the duration of the maintenance based on stored decay heat load. The basis for this recommendation is that the long-term PCS makeup to the PCCWST is not required in this condition, the PCCWST, along with other makeup sources can provide the required makeup to the spent fuel pool.

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TRM 3.6 CONTAINMENT SYSTEMS

TRM 3.6.2 Hydrogen Igniters

TR 3.6.2 Hydrogen igniters shall be FUNCTIONAL in accordance with Table TR 3.6.2-1.

APPLICABILITY: MODES 1 and 2,
MODE 5 with RCS pressure boundary open,
MODE 6 with upper internals in place,
MODE 6 with cavity level < 23 feet above the top of the reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required hydrogen igniters not FUNCTIONAL.	A.1 Restore required igniters to FUNCTIONAL status.	14 days

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.6.2.1	Energize each required hydrogen igniter and verify the surface temperature is ≥ 1700°F.	Each refueling outage

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Table TR 3.6.2-1 (page 1 of 1)
Hydrogen Igniters

	LOCATION	REQUIRED IGNITERS
1.	Loop Compartment 01	3
2.	Loop Compartment 02	3
3.	Pressurizer Compartment	3
4.	Tunnel connecting Loop Compartments	5
5.	Southeast Valve Room & Southeast Accumulator Room	2
6.	East Valve Room, Northeast Accumulator Room, & Northeast Valve Room	2(1)
7.	North CVS Equipment Room	2
8.	Lower Compartment Area (CMT and Valve Area)	10
9.	IRWST Sparger Side	2
10.	IRWST Vacuum Breaker Vents	2
11.	IRWST Roof Vents	2
12.	IRWST Hooded Vents	3
13.	Refueling Cavity	3
14.	Upper Compartment - Lower Region	9
15.	Upper Compartment - Mid Region	3
16.	Upper Compartment - Upper Region	3

NOTE:
(1) Igniter 18 and either Igniter 17 or Igniter 19.

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TRM 3.6.2 Hydrogen Igniters

Bases

The hydrogen igniters are required to provide the capability of burning hydrogen generated during severe accidents in order to prevent failure of the containment due to hydrogen detonation. These hydrogen igniters are required by 10 CFR 50.34 to limit the buildup of hydrogen to less than 10% assuming that 100% of the active zircaloy fuel cladding is oxidized.

This function is also important because it provides margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The igniters are distributed in the containment to limit the buildup of hydrogen in local areas. Two groups of igniters are provided in each area; one of which is sufficient to limit the buildup of hydrogen. When an igniter is energized, the igniter surface heats up to ≥ 1700°F. This temperature is sufficient to ignite hydrogen in the vicinity of the igniter when the lower flammability limit is reached. FSAR subsection 6.2.4 provides additional information.

The hydrogen igniter function is required during MODES 1 and 2 when core decay heat is high and during MODE 5 when the RCS pressure boundary is open and in MODE 6 with cavity level < 23 feet above the top of the reactor vessel flange. MODES 1, 2, 5, and 6 were selected as MODES at risk for a severe accident event occurrence combined with the production and burning of hydrogen, with MODES 3 and 4 excluded due to low risk for an event at these initial conditions. This selection excludes MODES with a low probability of event occurence and low reliability on the hydrogen igniters where maintenance can be performed. Table TR 3.6.2-1 indicates the minimum number of hydrogen igniters that are required.

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TRM 3.7 PLANT SYSTEMS

TRM 3.7.1 Normal Residual Heat Removal System (RNS) - Reactor Coolant System (RCS) Makeup

TR 3.7.1 One train of RNS - RCS makeup shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required RNS - RCS makeup train not FUNCTIONAL.	A.1 Restore required RNS - RCS makeup train to FUNCTIONAL status.	14 days

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.7.1.1	Verify required RNS pump develops a differential head of 330 feet on recirculation flow.	92 days
TRS 3.7.1.2	Verify the following valves stroke open: RNS V011, RNS Discharge Containment Isolation RNS V022, RNS Suction Header Containment Isolation RNS V023, RNS Suction from IRWST Isolation RNS V055, RNS Suction from Cask Loading Pit	92 days

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TRM 3.7.1 Normal Residual Heat Removal System (RNS) - Reactor Coolant System (RCS) Makeup

Bases

The RNS – RCS makeup function provides a nonsafety-related means of injecting water from the cask loading pit (CLP) into the RCS and water from the IRWST into the RCS following ADS actuations. The RNS – RCS makeup function is important because it provides margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to operate.

One train of RNS – RCS makeup includes one RNS pump, the lines from the CLP and the IRWST to the RCS, and RNS valves V011, V022, V023 and V055. This equipment does not normally operate during MODES 1, 2, and 3. FSAR subsection 5.4.7 contains additional information on the RNS.

The RNS – RCS makeup function is required during MODES 1, 2, and 3 because decay heat is higher and the need for ADS is greater.

Planned maintenance on redundant RNS SSCs should be performed during MODE 1, 2, or 3. Such maintenance should be performed on an RNS SSC not required to be FUNCTIONAL. The basis for this recommendation is that the RNS is more risk important during shutdown MODES when it is normally operating than during other MODES when it only provides a backup to PXS injection.

Planned maintenance on non-redundant RNS valves (such as V011, V022, V023, V055) should be performed to minimize the impact on their RNS – RCS makeup and their containment isolation capability. Non-pressure boundary maintenance should be performed during MODE 5 with a visible pressurizer level or MODE 6 with cavity level ≥ 23 feet above the top of the reactor vessel flange. In these MODES, these valves need to be open but they do not need to be able to close. Containment closure which is required in these MODES can be satisfied by one normally open operable valve. Pressure boundary maintenance can not be performed during MODES when the RNS is used to cool the core, therefore such maintenance should be performed during MODE 1, 2, or 3. Since these valves are also containment isolation valves, maintenance that renders the valves inoperable requires that the containment isolation valve located in series with the inoperable valve has to be closed and de-activated. The basis for this recommendation is that the RNS is more risk important during shutdown MODES when it is normally operating than during other MODES when it only provides a backup to PXS injection. In addition, it is not possible to perform pressure boundary maintenance of these valves during RNS operation.

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TRM 3.7 PLANT SYSTEMS

TRM 3.7.2 Normal Residual Heat Removal System (RNS) – Reactor Coolant System (RCS) Open

TR 3.7.2 Two trains of RNS RCS shutdown heat removal shall be FUNCTIONAL and one RNS pump in operation.

APPLICABILITY: MODE 5 with RCS pressure boundary open,
MODE 6 with upper internals in place,
MODE 6 with cavity level < 23 feet above the top of the reactor vessel flange.

- NOTE -

Not applicable with no irradiated fuel assemblies in the reactor vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RNS RCS shutdown heat removal train not FUNCTIONAL. <u>OR</u> Required RNS pump not in operation.	A.1 Initiate actions to increase the water inventory above the core. <u>AND</u> A.2 Restore two trains of RNS RCS shutdown heat removal to FUNCTIONAL status and one RNS pump in operation.	12 hours 72 hours

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TECHNICAL REQUIREMENT SURVEILLANCE

	SURVEILLANCE	FREQUENCY
TRS 3.7.2.1	Verify one RNS pump is in operation and that the required RNS pump circulates reactor coolant at a flow > 1580 gpm. OR Verify both RNS pumps are in operation and circulating reactor coolant at a flow > 2000 gpm.	Within 24 hours prior to entering the Applicability

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TRM 3.7.2 Normal Residual Heat Removal System (RNS) – Reactor Coolant System (RCS) Open

Bases

The RNS cooling function provides a nonsafety-related means to normally cool the RCS during shutdown operations (MODES 4, 5, and 6). This RNS cooling function is important during conditions when the RCS pressure boundary is open and the refueling cavity is not flooded because it reduces the probability of an initiating event due to loss of RNS cooling and because it provides margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The RCS is considered open when its pressure boundary is not intact. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The RNS cooling of the RCS involves the RNS suction line from the RCS HL, the two RNS pumps and the RNS discharge line returning to the RCS through the DVI lines. Prior to entering the Applicability, one of the RNS pumps has to be operating. The other pump may be operating or may be in standby. If one RNS pump and HX are in standby, the system shall be arranged such that the standby pump and HX can be placed into service from the main control room. FSAR subsection 5.4.7 contains additional information on the RNS.

Both RNS pumps are required during the Applicability when the loss of RNS cooling is risk important. The applicability is modified by a note that allows the TR to be not applicable when there is no irradiated fuel in the reactor vessel, which is based on there being no decay heat in this condition. If both RNS pumps are not FUNCTIONAL, the plant should not enter these conditions. If the plant has entered reduced inventory conditions, then the plant should take action to restore full system operation or leave the Applicability. If the plant has not restored full system operation or left the Applicability within 12 hours, then actions need to be initiated to increase the RCS water level to either 20% pressurizer level or to a refueling cavity ≥ 23 ft.

Planned maintenance affecting this RNS cooling function should be performed in MODE 1, 2, or 3, or in MODE 6 with a refueling cavity level $\geq$ 23 ft above the top of the reactor vessel flange, or with no irradiated fuel in the reactor vessel, when the RNS is not considered risk significant. The basis for this recommendation is that the RNS is more risk important during the Applicability conditions than during other MODES when it only provides a backup to PXS injection.

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TRM 3.7 PLANT SYSTEMS

TRM 3.7.3 Component Cooling Water System (CCS) – Reactor Coolant System (RCS) Open

TR 3.7.3 Two CCS trains shall be FUNCTIONAL and one CCS pump in operation to support the operating Normal Residual Heat Removal System (RNS) RCS shutdown heat removal train.

APPLICABILITY: MODE 5 with RCS pressure boundary open,
MODE 6 with upper internals in place,
MODE 6 with cavity level < 23 feet above the top of the reactor vessel flange.

- NOTE -

Not applicable with no irradiated fuel assemblies in the reactor vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCS train not FUNCTIONAL. OR Required CCS pump not in operation to support operating RNS RCS shutdown heat removal train.	A.1 Initiate actions to increase the water inventory above the core. AND A.2 Restore two CCS trains to FUNCTIONAL status and one CCS pump in operation to support operating RNS RCS shutdown heat removal train.	12 hours 72 hours

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TECHNICAL REQUIREMENT SURVEILLANCE

	SURVEILLANCE	FREQUENCY
TRS 3.7.3.1	Verify one CCS pump is in operation and each CCS pump operating individually can provide a CCS flow through one RNS heat exchanger > 2685 gpm. OR Verify both CCS pumps are in operation with CCS flow through each RNS heat exchanger > 2685 gpm.	Within 24 hours prior to entering the Applicability

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TRM 3.7.3 Component Cooling Water System (CCS) – Reactor Coolant System (RCS) Open

Bases

The CCS cooling of the RNS HXs provides a nonsafety-related means to normally cool the RCS during shutdown operations (MODES 4, 5, and 6). This RNS cooling function is important because it reduces the probability of an initiating event due to loss of RNS cooling and because it provides margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The RCS is considered open when its pressure boundary is not intact. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The CCS cooling of the RNS involves two CCS pumps and HXs and the CCS line to the RNS HXs. Prior to entering the Applicability, one of the CCS pumps and its HX has to be operating. One of the lines to a RNS HX also has to be open. The other CCS pump and HX may be operating or may be in standby. If one CCS pump and HX are in standby, the system shall be arranged such that the standby pump can be placed into service from the main control room. FSAR subsection 9.2.2 contains additional information on the CCS.

Both CCS pumps are required during the Applicability when the loss of RNS cooling is risk important. The applicability is modified by a note that allows the TR to be not applicable when there is no irradiated fuel in the reactor vessel, which is based on there being no decay heat in this condition. If both CCS pumps are not FUNCTIONAL, the plant should not enter these conditions. If the plant has entered these conditions, then the plant should take action to restore both CCS pumps or to leave these conditions. If the plant has not restored full system operation or left the Applicability within 12 hours, then actions need to be initiated to increase the RCS water level to either 20% pressurizer level or to a refueling cavity ≥ 23 ft.

Planned maintenance affecting this CCS cooling function should be performed in MODE 1, 2, or 3 when the CCS is not supporting RNS operation. The basis for this recommendation is that the CCS is more risk important during shutdown MODES, especially during the Applicability conditions than during other MODES.

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TRM 3.7 PLANT SYSTEMS

TRM 3.7.4 Service Water System (SWS) – Reactor Coolant System (RCS) Open

TR 3.7.4 Two SWS trains shall be FUNCTIONAL and one SWS pump in operation to support Component Cooling Water System (CCS) cooling.

APPLICABILITY: MODE 5 with RCS pressure boundary open,
MODE 6 with upper internals in place,
MODE 6 with cavity level < 23 feet above the top of the reactor vessel flange.

- NOTE -

Not applicable with no irradiated fuel assemblies in the reactor vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS train not FUNCTIONAL. OR Required SWS pump not in operation to support CCS cooling.	A.1 Initiate actions to increase the water inventory above the core. AND A.2 Restore two SWS trains to FUNCTIONAL status and one SWS pump in operation to support CCS cooling.	12 hours 72 hours

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TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.7.4.1	Verify one SWS pump is in operation and each SWS pump operating individually can provide a SWS flow > 10,000 gpm.	Within 24 hours prior to entering the Applicability
TRS 3.7.4.2	Operate each cooling tower fan for > 15 min.	Within 24 hours prior to entering the Applicability

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TRM 3.7.4 Service Water System (SWS) – Reactor Coolant System (RCS) Open

Bases

The SWS cooling of the CCS HXs provides a nonsafety-related means to normally cool the RNS HX which cools the RCS during shutdown operations (MODES 4, 5, and 6). This RNS cooling function is important because it reduces the probability of an initiating event due to loss of RNS cooling and because it provides margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The RCS is considered open when its pressure boundary is not intact. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The SWS cooling of the CCS HXs involves two SWS pumps and cooling tower fans and the SWS line to the CCS HXs. Prior to entering the Applicability, one of the SWS pumps and its cooling tower fan has to be operating. The other SWS pump and cooling tower fan may be operating or may be in standby. If one SWS pump and cooling tower fan are in standby, the system shall be arranged such that the standby pump and cooling tower fan can be placed into service from the main control room. FSAR subsection 9.2.1 contains additional information on the SWS.

Both SWS pumps and cooling tower fans are required during the Applicability when the loss of RNS cooling is risk important. The applicability is modified by a note that allows the TR to be not applicable when there is no irradiated fuel in the reactor vessel, which is based on there being no decay heat in this condition. If both SWS pumps and cooling tower fans are not FUNCTIONAL, the plant should not enter these conditions. If the plant has entered these conditions, then the plant should take action to restore both SWS pumps / fans or to leave these conditions. If the plant has not restored full system operation or left the Applicability within 12 hours, then actions need to be initiated to increase the RCS water level to either 20% pressurizer level or to a refueling cavity ≥ 23 ft.

Planned maintenance affecting this SWS cooling function should be performed in MODES when the SWS is not supporting RNS operation, i.e., during MODE 1, 2, or 3. The basis for this recommendation is that the SWS is more risk important during shutdown MODES, especially during the Applicability conditions than during other MODES.

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TRM 3.7 PLANT SYSTEMS

TRM 3.7.5 Main Control Room (MCR) Cooling - Long Term Shutdown

TR 3.7.5 Long term cooling of the MCR shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required MCR ancillary fan not FUNCTIONAL.	A.1 Restore required MCR ancillary fan to FUNCTIONAL status.	14 days

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.7.5.1	Operate required MCR ancillary fan for > 15 min.	92 days
TRS 3.7.5.2	Verify required MCR ancillary fan can provide a flow of air into the MCR for > 15 min. During this test, the required MCR ancillary fan will be powered from an ancillary diesel.	10 years

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TRM 3.7.5 Main Control Room (MCR) Cooling - Long Term Shutdown

Bases

The MCR ancillary fans provide long term shutdown support by cooling the main control room. For the first three days after an accident the emergency HVAC system (VES) together with the passive heat sinks in the MCR provide cooling of the MCR. After 3 days, the MCR ancillary fans can be used to circulate ambient air through the MCR to provide cooling. The long term MCR cooling function is required during all MODES of operation. This long term MCR cooling function is important because it supports long-term shutdown operation. A minimum availability of 90% is assumed for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The long term MCR cooling function involves the use of a MCR ancillary fan. During TRS 3.7.5.1 the required fan will be run to verify that it operates without providing flow to the MCR. During TRS 3.7.5.2 the required fan will be connected to the MCR and operated such that it provides flow to the MCR. FSAR subsection 9.4.1 contains additional information on the long term MCR cooling function.

One MCR ancillary fan is required during all MODES of plant operation. Planned maintenance should not be performed on the required MCR ancillary fan during a required MODE of operation; planned maintenance should be performed on the redundant MCR ancillary fan (i.e., the fan not required to be FUNCTIONAL) during MODE 3 or 4, MODE 5 with a visible pressurizer level, or MODE 6 with the refueling cavity ≥ 23 ft; these MODES are selected because the reactor is tripped in these MODES and the risk of core damage is low.

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TRM 3.7 PLANT SYSTEMS

TRM 3.7.6 I&C Room Cooling - Long Term Shutdown

TR 3.7.6 Long term cooling of I&C room B or I&C room C shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required I&C room ancillary fan not FUNCTIONAL.	A.1 Restore required I&C room ancillary fan to FUNCTIONAL status.	14 days

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.7.6.1	Operate required I&C room ancillary fan for > 15 min.	92 days
TRS 3.7.6.2	Verify required I&C room ancillary fan can provide a flow of air into an I&C room for > 15 min. During this test, the required I&C room ancillary fan will be powered from its associated ancillary diesel.	10 years

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TRM 3.7.6 I&C Room Cooling - Long Term Shutdown

Bases

The I&C room ancillary fans provide long term shutdown support by cooling I&C rooms B & C, which contain post accident instrument processing equipment. For the first three days after an accident the passive heat sinks in the I&C rooms provide cooling. After 3 days, the I&C room ancillary fans can be used to circulate ambient air through the I&C rooms to provide cooling. The long term I&C room cooling function is required during all MODES of operation. This long term I&C room cooling function is important because it supports long-term shutdown operation. A minimum availability of 90% is assumed for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The long term I&C room cooling function involves the use of two I&C room ancillary fans. Each I&C room ancillary fan is associated with one I&C room (B or C) and one ancillary diesel generator. Each of the I&C room ancillary fans is permanently installed in its associated I&C room ductwork. During TRS 3.7.6.1, the required I&C room ancillary fan will be run to verify that it operates. During TRS 3.7.6.2, the required I&C room ancillary fan will be powered from the required ancillary diesel generator; and will be operated such that the fan provides the cooling required for the I&C room. FSAR subsection 9.4.1 contains additional information on the long term I&C room cooling function.

One I&C room ancillary fan is required during all MODES of plant operation. Planned maintenance should not be performed on the required I&C room ancillary fan during a required MODE of operation. Planned maintenance should be performed on the redundant I&C room ancillary fan.

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TRM 3.7 PLANT SYSTEMS

TRM 3.7.7 Spent Fuel Pool Makeup Water Sources

TR 3.7.7 Spent fuel pool makeup water sources shall be FUNCTIONAL.

- NOTES -

1. FUNCTIONALITY of the cask washdown pit is required when the spent fuel pool decay heat > 4.0 MWt and ≤ 7.0 MWt.

2. FUNCTIONALITY of the cask loading pit is required when the spent fuel pool decay heat > 5.0 MWt and ≤ 7.0 MWt.

3. FUNCTIONALITY of the Passive Containment Cooling Water Storage Tank (PCCWST) is required as a spent fuel pool makeup water source when the spent fuel pool decay heat > 7.0 MWt. If the reactor decay heat is > 7.0 MWt, the PCCWST must be exclusively available for containment cooling in accordance with Technical Specifications LCO 3.6.6.

4. FUNCTIONALITY of the Fuel Transfer Canal is required.

APPLICABILITY: When irradiated fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required spent fuel pool makeup water sources not FUNCTIONAL.	A.1 Initiate action to restore the required makeup water source(s) to FUNCTIONAL status.	Immediately

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TECHNICAL REQUIREMENT SURVEILLANCE

	SURVEILLANCE	FREQUENCY
TRS 3.7.7.1	---------- **- NOTE -** Only required to be performed when spent fuel pool decay heat is > 7.0 MWt. ---------- Verify one passive containment cooling system, motor-operated valve in each flow path is closed and locked, sealed, or otherwise secured in position.	7 days
TRS 3.7.7.2	---------- **- NOTE -** Only required to be performed when spent fuel pool decay heat is > 7.0 MWt. ---------- Verify the PCCWST volume is ≥ 756,700 gallons.	7 days
TRS 3.7.7.3	---------- **- NOTE -** Only required to be performed when spent fuel pool decay heat is > 4.0 MWt and ≤ 7.0 MWt. ---------- Verify the water level in the cask washdown pit is ≥ 13.75 ft.	31 days
TRS 3.7.7.4	---------- **- NOTE -** Only required to be performed when spent fuel pool decay heat is > 5.0 MWt and ≤ 7.0 MWt. ---------- Verify the water level in the cask loading pit is ≥ 43.9 ft. and in communication with the spent fuel pool.	31 days
TRS 3.7.7.5	Verify the fuel transfer canal is in communication with the spent fuel pool.	31 days

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TECHNICAL REQUIREMENT SURVEILLANCE (continued)

SURVEILLANCE		FREQUENCY
TRS 3.7.7.6	Verify the spent fuel pool makeup isolation valves PCS-PL-V009, PCS-PL-V045, PCS-PL-V051, SFS-PL-V066, and SFS-PL-V068 are FUNCTIONAL in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

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TRM 3.7.7 Spent Fuel Pool Makeup Water Sources

BASES

BACKGROUND

The spent fuel pool is normally cooled by the nonsafety spent fuel pool cooling system. In the event the normal cooling system is unavailable, the spent fuel pool can be cooled by the normal residual heat removal system. Alternatively, the spent fuel pool and fuel transfer canal contain sufficient water inventory for decay heat removal by boiling. To support extended periods of loss of normal pool cooling, makeup water is required to provide additional cooling by boiling. Both safety and non-safety makeup water sources are available on-site.

Safety-related, gravity fed sources of makeup water are provided to the spent fuel pool. These makeup water sources contain sufficient water to maintain spent fuel pool cooling for 72 hours. When irradiated fuel is in the spent fuel pool, the fuel transfer canal, with the gate open, must be available to provide makeup water to the spent fuel pool. When the spent fuel pool decay heat is > 4.0 MWt and ≤ 7.0 MWt, the fuel transfer canal and cask washdown pit must be available to provide makeup to the spent fuel pool. When the spent fuel pool decay heat is > 5.0 MWt and ≤ 7.0 MWt the fuel transfer canal, the cask washdown pit, and the cask loading pit must be available to provide makeup to the spent fuel pool. When the spent fuel pool decay heat is > 7.0 MWt and the reactor decay heat is ≤ 7.0 MWt, the Passive Containment Cooling Water Storage Tank (PCCWST) must be available to provide makeup water to the spent fuel pool (when the tank is no longer required for containment cooling purposes). Additional on-site makeup water sources are available to provide spent fuel pool cooling between 3 and 7 days.

The PCCWST is isolated by two normally closed valves. The normally closed valves will be opened only to provide emergency makeup to the spent fuel pool. A third downstream valve permits the operator to regulate addition of water to the spent fuel pool as required to maintain the cooling water inventory.

Once decay heat in the spent fuel pool is reduced to ≤ 4.0 MWt, the spent fuel pool water inventory, as required by Technical Specifcation (TS) LCO 3.7.5, Spent Fuel Pool Water Level, and fuel transfer canal inventory, is sufficient, without makeup, to maintain the spent fuel pool for 72 hours.

A general description of the spent fuel pool design is given in FSAR Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in FSAR Section 9.1.3 (Ref. 2).

APPLICABLE SAFETY ANALYSES

In the event the normal spent fuel pool cooling system is unavailable, the spent fuel cooling is provided by the heat capacity of the spent fuel pool and spent fuel pool makeup water sources available to the pool. The worst case decay heat load (decay heat > 7.0 MWt) is produced by a full core off-load following a refueling plus fifteen years of spent fuel. The worst case event is a seismic event concurrent with a station blackout where the spent fuel pool cooling suction line to

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BASES (continued)

the spent fuel pool is assumed to fail, and the water in the pool and makeup sources are assumed to drain to the bottom elevation of the spent fuel pool cooling suction piping attachment to the spent fuel pool. For this case the spent fuel pool and fuel transfer canal inventory provided by the water over the stored fuel and below the pump suction connection is capable of cooling the spent fuel pool without boiling for at least 2.0 hours, following a loss of normal spent fuel pool cooling. After boiling starts, makeup water may be required to replace water lost by boiling and is available, without offsite support, via the PCCWST.

The requirements of Technical Specifications (TS) LCO 3.6.6, “Passive Containment Cooling System,” are applicable in MODES 1, 2, 3, and 4, and MODES 5 and 6 with reactor decay heat > 7.0 MWt. TS LCO 3.6.6 requires availability of the containment cooling water tank for containment heat removal. At or below 7.0 MWt reactor decay heat, containment air cooling is adequate for 72 hours without support from the PCCWST, which can then provide a makeup water source for the spent fuel pool.

None of the FSAR Design Basis Accident analyses assume availability of the PCCWST, the cask washdown pit, or the cask loading pit for the spent fuel pool makeup water sources. This TR is included as defense-in-depth.

TECHNICAL REQUIREMENT

The spent fuel pool makeup water sources are required to contain the following amounts of water to be considered FUNCTIONAL:

- Fuel transfer canal with the gate open in communication with the spent fuel pool (which will establish water level as required by TS LCO 3.7.5, Spent Fuel Pool Water Level).
- Cask washdown pit water level must be ≥ 13.75 ft and capable of being manually aligned.
- Cask loading pit water level must be ≥ 43.9 ft with the gate open in communication with the spent fuel pool.
- PCCWST is required to contain ≥ 756,700 gallons of water.

A FUNCTIONAL flow path from the required makeup source assures spent fuel cooling for at least 72 hours. Several additional makeup sources are available, including the ground level PCCAWST. These makeup sources assure spent fuel cooling for at least 7 days.

Note 1 specifies that the cask washdown pit is required to be FUNCTIONAL when the spent fuel pool decay heat is > 4.0 MWt and ≤ 7.0 MWt.

Note 2 specifies that the cask loading pit is required to be FUNCTIONAL when the spent fuel pool decay heat is > 5.0 MWt and ≤ 7.0 MWt.

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BASES (continued)

Note 3 specifies that the PCCWST is required to be FUNCTIONAL when the spent fuel pool decay heat is > 7.0 MWt, which is normal following a full core off load. The larger makeup source is necessary for the higher decay heat load. In MODES 5 and 6, with the calculated reactor decay heat > 7.0 MWt, the PCCWST is reserved for containment cooling in accordance with TS LCO 3.6.6, Passive Containment Cooling System (PCS). Thus, fuel movement from the reactor to the spent fuel pool must be suspended until reactor decay heat is ≤ 7.0 MWt if the fuel movement will increase the spent fuel pool decay heat to > 7.0 MWt.

Note 4 specifies that the fuel transfer canal is required to be FUNCTIONAL when irradiated fuel is in the spent fuel pool.

The decay heat specified in the three Notes is normally determined by calculation.

When a portion of the fuel is returned to the reactor vessel in preparation for startup, the pool decay heat is reduced to ≤ 5.0 MWt and makeup from the fuel transfer canal and cask washdown pit is sufficient to maintain spent fuel pool water cooling for 72 hours.

APPLICABILITY

This TR applies during storage of irradiated fuel in the spent fuel pool. Irradiated fuel assemblies generate decay heat and the required makeup sources provide for 3 days of cooling in the event of a station blackout with seismic event, which results in the spent fuel pool water inventory at the bottom of the pump suction connection to the pool.

ACTIONS

If the fuel transfer canal, cask washdown pit (with spent fuel pool decay heat > 4.0 and ≤ 7.0 MWt), the cask loading pit (with spent fuel pool decay heat > 5.0 MWt and ≤ 7.0 MWt), or the PCCWST (with spent fuel pool decay heat > 7.0 MWt) is not FUNCTIONAL, Action must be initiated immediately to restore the makeup source or its associated flow path to FUNCTIONAL status.

Additionally, in order to provide the maximum cooling capability, the spent fuel pool should be filled to its maximum level. Nonsafety related makeup sources can be used to fill the pool. This action is not specified in the specification, since the benefit of adding approximately 6 inches of water to the pool is less than a 5% improvement in cooling capability.

TECHNICAL REQUIREMENT SURVEILLANCES

TRS 3.7.7.1 verifies that the three flow paths from the PCCWST to the containment vessel are isolated and secured to prevent inadvertent opening and loss of required tank volume. The verification is required to be performed prior to declaring the PCCWST FUNCTIONAL for spent fuel pool usage. The 7 day Frequency is appropriate because the valves in the passive containment cooling system are controlled by plant procedures.

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BASES (continued)

TRS 3.7.7.2 verifies sufficient PCCWST volume is available in the event of a loss of spent fuel cooling prior to declaring the tank FUNCTIONAL for spent fuel pool usage. The 7 day Frequency is appropriate because the volume in the PCCWST is normally stable and water level changes are controlled by plant procedures.

TRS 3.7.7.3 verifies sufficient cask washdown pit water volume is available in the event of a loss of spent fuel cooling. The 13.75 ft level specified provides makeup water for stored fuel with decay heat (normally determined by calculation) > 4.0 and ≤ 7.0 MWt. The cask washdown pit is no longer required when the decay heat level requires that the PCCWST is FUNCTIONAL for spent fuel pool usage. The 31 day Frequency is appropriate because the cask washdown pit has only one drain line which is isolated by series manual valves which are only operated in accordance with plant procedures, thus providing assurance that inadvertent level reduction is not likely.

TRS 3.7.7.4 verifies sufficient cask loading pit water volume is available and connected to the spent fuel pool such that no action is required in the fuel handling area, in the event of a loss of spent fuel cooling. The 43.9 ft level specified provides makeup water for stored fuel with decay heat (normally determined by calculation) > 5.0 and ≤ 7.0 MWt. The cask loading pit is no longer required when the decay heat level requires that the PCCWST is FUNCTIONAL for spent fuel pool usage. The 31 day Frequency is appropriate because the cask loading pit has only one drain line, which is isolated by series manual valves, which are operated only in accordance with plant procedures. This provides assurance that inadvertent level reduction is not likely.

TRS 3.7.7.5 verifies sufficient fuel transfer canal water volume is available and connected to the spent fuel pool such that no action is required in the fuel handling area, in the event of a loss of spent fuel cooling. The 31 day Frequency is appropriate because the fuel transfer canal has only one drain line isolated by a locked-closed manual valve and is isolated from the refueling cavity by a gate valve and blind flange. Connections to the fuel transfer canal are operated only in accordance with plant procedures. This provides assurance that inadvertent level reduction is not likely.

TRS 3.7.7.6 requires verification of the FUNCTIONALITY of the manual makeup water source isolation valves in accordance with the requirements and Frequency specified in the Inservice Testing Program. Manual valves PCS-PL-V009, PCS-PL-V045, PCS-PL-V051, isolate the makeup flow path from the PCCWST. Manual valves SFS-PL-V066 and SFS-PL-V068 isolate the makeup flow path from the cask washdown pit.

REFERENCES

1. FSAR Section 9.1.2, “Spent Fuel Storage.”

2. FSAR Section 9.1.3, “Spent Fuel Pool Cooling System.”

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TRM 3.8 ELECTRICAL POWER SYSTEMS

TRM 3.8.1 AC Power Supplies - Operating

TR 3.8.1 One standby diesel generator shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel volume in required standby diesel fuel tank ≤ 55,000 gal.	A.1 Restore fuel volume in required standby diesel fuel tank to > 55,000 gal.	14 days
B. Required fuel transfer pump not FUNCTIONAL. <u>OR</u> Required standby diesel generator not FUNCTIONAL for reasons other than Condition A.	B.1 Restore required fuel transfer pump and required standby diesel generator to FUNCTIONAL status.	14 days

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TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.8.1.1	Verify fuel oil volume in required standby diesel generator fuel tank is > 55,000 gal.	31 days
TRS 3.8.1.2	Verify required fuel oil transfer pump provides a recirculation flow of > 8 gpm.	92 days
TRS 3.8.1.3	Verify required standby diesel generator starts and operates at > 4000 kw for > 1 hour. This test may utilize diesel engine prelube prior to starting and a warmup period prior to loading.	92 days
TRS 3.8.1.4	Verify the required standby diesel generator starts and operates at > 4000 kw for > 24 hours. This test may utilize diesel engine prelube prior to starting and a warmup period prior to loading.	10 years
TRS 3.8.1.5	Verify the required standby diesel generator starts and operates (unloaded). This test may utilize diesel engine prelube prior to starting. Both diesel generators shall be started at the same time for this test.	10 years

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TRM 3.8.1 AC Power Supplies - Operating

Bases

AC power is required to power the RNS and to provide a nonsafety-related means of supplying power to the safety-related PMS for actuation and post accident monitoring. The RNS provides a nonsafety-related means to inject water into the RCS following ADS actuations in MODES 1, 2, 3, and 4 (when steam generators cool the RCS). This AC power supply function is important because it adds margin to the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to operate.

Two standby diesel generators are provided. Each standby diesel generator has its own fuel oil transfer pump and fuel oil tank. The volume of fuel oil required is that volume that is above the connection to the fuel oil transfer pump. FSAR subsection 8.3.1 contains additional information.

This AC power supply function is required during MODES 1, 2, 3, 4, and 5 when RNS injection and PMS actuation are more risk important. Planned maintenance should not be performed on required AC power supply SSCs during a required MODE of operation; planned maintenance should be performed on redundant AC power supply SSCs during MODE 1, 2, or 3 when the RNS is not normally in operation. The basis for this recommendation is that the AC power is more risk important during shutdown MODES, especially when the RCS is open as defined in TRM 3.7.2 Bases, than during other MODES.

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TRM 3.8 ELECTRICAL POWER SYSTEMS

TRM 3.8.2 AC Power Supplies – Reactor Coolant System (RCS) Open

TR 3.8.2 The following AC power supplies shall be FUNCTIONAL to support Normal Residual Heat Removal System (RNS) operation:

a. One offsite circuit; and
b. One standby diesel generator.

APPLICABILITY: MODE 5 with RCS pressure boundary open,
MODE 6 with upper internals in place,
MODE 6 with cavity level < 23 feet above the top of the reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required AC power supply not FUNCTIONAL.	A.1 Initiate actions to increase the water inventory above the core. AND A.2 Restore required AC power supply to FUNCTIONAL status.	12 hours 72 hours

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.8.2.1	Verify required AC power supplies are FUNCTIONAL.	Within 24 hours prior to entering the Applicability

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TRM 3.8.2 AC Power Supplies – Reactor Coolant System (RCS) Open

Bases

AC power is required to power the RNS and its required support systems (CCS & SWS); the RNS provides a nonsafety-related means to normally cool the RCS during shutdown operations. This RNS cooling function is important when the RCS pressure boundary is open and the refueling cavity is not flooded because it reduces the probability of an initiating event due to loss of RNS cooling during these conditions and because it provides margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The RCS is considered open when its pressure boundary is not intact. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to operate.

Two AC power supplies, one offsite and one onsite supply, are required as follows:

a) Offsite power through the transmission switchyard and either the main step-up transformer / unit auxiliary transformer or the reserve auxiliary transformer supply from the transmission switchyard, and

b) Onsite power from one of the two standby diesel generators.

FSAR subsection 8.3.1 contains additional information on the standby diesel generators. FSAR Section 8.2 contains information on the offsite AC power supply.

One offsite and one onsite AC power supply are required during the Applicability when the loss of RNS cooling is important. If both of these AC power supplies are not FUNCTIONAL, the plant should not enter these conditions. If the plant has already entered these conditions, then the plant should take action to restore this AC power supply function or to leave these conditions. If the plant has not restored full system operation or left the Applicability within 12 hours, then actions need to be initiated to increase the RCS water level to either 20% pressurizer level or to a refueling cavity ≥ 23 ft.

Planned maintenance should not be performed on required AC power supply SSCs. Planned maintenance affecting the standby diesel generators should be performed in MODE 1, 2, or 3 when the RNS is not normally in operation. Planned maintenance of the other AC power supply should be performed in MODE 2, 3, or 6 with the refueling cavity ≥ 23 ft. The basis for this recommendation is that the AC power is more risk important during shutdown MODES, especially during the Applicability conditions than during other MODES.

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TRM 3.8 ELECTRICAL POWER SYSTEMS

TRM 3.8.3 AC Power Supplies – Long Term Shutdown

TR 3.8.3 One ancillary diesel generator shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
When irradiated fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel volume in ancillary diesel generator fuel tank ≤ 600 gal.	A.1 Restore fuel volume in ancillary diesel generator fuel tank > 600 gal.	14 days
B. Required ancillary diesel generator not FUNCTIONAL for reasons other than Condition A.	B.1 Restore required ancillary diesel generator to FUNCTIONAL status.	14 days

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TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.8.3.1	Verify fuel volume in ancillary diesel generator fuel tank is > 600 gal.	31 days
TRS 3.8.3.2	Verify required ancillary diesel generator starts and operates for > 1 hour connected to a test load > 35 kw. This test may utilize diesel engine warmup period prior to loading	92 days
TRS 3.8.3.3	Verify required ancillary diesel generator starts and operates for 4 hours while providing power to the regulating transformer, an ancillary main control room fan, the required ancillary I&C room fan and a passive containment cooling water storage tank recirculation pump that it will power in a long term post accident condition. Test loads will be applied to the output of the regulating transformers that represent the loads required for post-accident monitoring and control room lighting. This test may utilize diesel engine warmup prior to loading. Both diesel generators will be operated at the same time during this test.	10 years

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TRM 3.8.3 AC Power Supplies – Long Term Shutdown

Bases

The ancillary diesel generators provide long term power supplies for post accident monitoring, MCR and I&C room cooling, MCR lighting, and PCS and spent fuel pool water makeup. For the first 3 days after an accident, the Class 1E batteries provide power for post accident monitoring. Passive heat sinks provide cooling of the MCR and the I&C rooms. The initial water supply in the PCCWST provides for at least 3 days of PCS cooling. The initial water volume in the spent fuel pool, along with required makeup sources, normally provides for at least 3 days of spent fuel cooling; in some shutdown events the PCCWST is used to supplement the spent fuel pool.

After 3 days, ancillary diesel generators can be used to power the MCR and I&C room ancillary fans, two divisions of post accident monitoring, the PCS recirculation pumps, and MCR lighting. In this time frame, the PCCWST provides water makeup to both the PCS and the spent fuel pool. An ancillary diesel generator is required during all MODES of operation, and when irradiated fuel assemblies are stored in the spent fuel pool. This long-term AC power supply function is important because it supports long-term shutdown operation. A minimum availability of 90% is assumed for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The long-term AC power supply function involves the use of two ancillary diesel generators and an ancillary diesel generator fuel tank. Each ancillary diesel generator provides power to one MCR ancillary fan, one I&C room ancillary fan and the associated division of post accident monitoring, one PCS recirculation pump, and a subset of MCR emergency lighting. The specified ancillary diesel generator fuel tank volume is based on operation of both ancillary diesel-generators for 4 days. FSAR subsection 8.3.1 contains additional information on the long-term AC power supply function.

One ancillary diesel generator and the ancillary diesel generator fuel tank are required during all MODES of plant operation, and when irradiated fuel assemblies are stored in the spent fuel pool. Planned maintenance should not be performed on the required ancillary diesel generator during a required MODE of operation; planned maintenance should be performed on the redundant ancillary diesel generator. Planned maintenance affecting the ancillary diesel generator fuel tank that requires less than 72 hours to perform can be performed in any MODE of operation. Planned maintenance requiring more than 72 hours should be performed in MODE 6 with the refueling cavity water level ≥ 23 ft, or when the core has been completely offloaded to the spent fuel pool. Alternate means of makeup to the spent fuel pool should be available for the duration of the maintenance, based on stored decay heat load.

The basis for this recommendation is that in MODE 6, with the refueling cavity water level ≥ 23 ft, core decay heat is low, the risk of core damage is low, and the inventory of the refueling cavity results in slow response of the plant to accidents. When the core has been completely offloaded to the spent fuel pool, the PCCWST and other makeup sources can provide required long-term makeup to the spent fuel pool.

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TRM 3.8 ELECTRICAL POWER SYSTEMS

TRM 3.8.4 Non Class 1E DC and UPS System (EDS)

TR 3.8.4 Power for DAS automatic actuation functions required by TR 3.3.1 shall be FUNCTIONAL.

APPLICABILITY: According to Table TR 3.3.1-1 to support required DAS actuation functions.

- NOTE -

EDS support for DAS ATWS mitigation functions of reactor trip and turbine trip actuation are only required in MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Power to required DAS Actuation Function not FUNCTIONAL.	A.1 Restore required power supply to FUNCTIONAL status.	14 days

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.8.4.1	Verify power supply voltage at each required DAS cabinet is 120 volts ± 5%.	92 days

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TRM 3.8.4 Non Class 1E DC and UPS System (EDS)

Bases

The EDS function of providing power to DAS to support ATWS mitigation is important based on 10 CFR 50.62 (ATWS Rule). The EDS function of providing power to DAS to support ESFA is important based on providing margin in the PRA sensitivity performed assuming no credit for nonsafety-related SSCs to mitigate at-power and shutdown events. The margin provided in the PRA study assumes a minimum availability of 90% for this function during the Applicability, considering both maintenance unavailability and failures to operate.

The DAS uses a 2-out-of-2 logic to actuate automatic functions. EDS power is required for the DAS sensors, DAS actuation, and the devices which control the actuated components. Power may be provided by EDS to DAS by non-1E batteries through non-1E inverters. Other means of providing power to DAS include the spare battery through a non-1E inverter or non-1E regulating transformers.

The EDS support of the DAS mitigation functions is required during MODE 1 when ATWS is a limiting event. The EDS support of the DAS ESFA mitigation functions is required during the MODES and other conditions for which DAS ESFA mitigation functions are required as specified in Table TR 3.3.1-1. Planned maintenance should not be performed on a required EDS SSC during a required MODE of operation; planned maintenance should be performed on redundant supplies of EDS power.

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TRM 3.9 REFUELING OPERATIONS

TRM 3.9.1 Containment Penetrations

TR 3.9.1 The containment penetrations shall be in the following status:

a. The equipment hatches closed and held in place by four bolts or, if open, the Containment Air Filtration System (VFS) shall be FUNCTIONAL and operating;

b. One door in each air lock closed or, if open, the VFS shall be FUNCTIONAL and operating;

c. The containment spare penetrations closed or, if open, the VFS shall be FUNCTIONAL and operating;

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either:

1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. VFS being FUNCTIONAL and operating.

- NOTE -

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TR not met.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

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TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.9.1.1	Verify each required containment penetration is in the required status.	7 days
TRS 3.9.1.2	---------- – NOTE – Only required to be met if any TR items a through d has an open penetration. ---------- Verify the required VFS can maintain a pressure ≤ -0.03 inches water gauge relative to outside atmospheric pressure in the area enclosed by the containment and alternate barrier.	24 months
TRS 3.9.1.3	Operate each required VFS train for ≥ 15 continuous minutes with the heaters operating.	Within 31 days prior to fuel movement

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TRM 3.9.1 Containment Penetrations

BASES

BACKGROUND

During movement of irradiated fuel assemblies within containment, potential releases of fission product radioactivity within containment are monitored and filtered or are restricted from escaping to the environment when the TR requirements are met. Monitoring of potential releases of radiation is performed in accordance with Technical Specifications (TS) Section 5.5.2, “Radioactive Effluent Control Program.” In MODES 1, 2, 3, and 4, containment OPERABILITY is addressed in TS LCO 3.6.1, “Containment.” In MODES 5 and 6, closure capability of containment penetrations is addressed in TS LCO 3.6.7, “Containment Penetrations.” Since there is no potential for containment pressurization due to a fuel handling accident, the Appendix J leakage criteria and tests are not required in MODES 5 and 6.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 50.34. For a fuel handling accident, the Reference 1 analysis does not rely on containment closure to meet the offsite radiation exposure limits. This TR is provided as an additional level of defense against the possibility of a fission product release from a fuel handling accident.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. During movement of irradiated fuel assemblies within containment, an equipment hatch is considered closed if the hatch cover is held in place by at least four bolts. Good engineering practice dictates that the bolts required by this TR be approximately equally spaced.

If an equipment hatch is open, an alternate barrier between the containment atmosphere and the outside atmosphere shall be in place. Each containment equipment hatch opens into a staging area in the auxiliary building. These staging areas contain doors that open to the radiologically controlled areas of the annex building. The annex building contains a door that opens to the outside atmosphere. The alternate barrier may consist of the staging area in the auxiliary building, or may consist of the staging areas in the auxiliary building and the radiologically controlled areas in the annex building provided the doors from the annex building to the outside atmosphere are closed. The alternate barrier may also consist of a temporary equipment hatch cover that provides equivalent isolation capability. The alternate barrier prevents the airborne fission products from being readily released to the atmosphere if the equipment hatches were open during a fuel handling accident.

If an equipment hatch is open during movement of irradiated fuel assemblies within containment, the Containment Air Filtration System (VFS) shall be FUNCTIONAL, and at least one exhaust fan shall be operating to provide for monitoring of air-borne radioactivity. This system services the containment, and upon detection of high radiation, also services the fuel handling area, the auxiliary building (including the staging areas), and the annex building. If high airborne radioactivity is detected in the area enclosed by the alternate barrier, the

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BASES (continued)

Radiologically Controlled Area Ventilation System (VAS) supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment, and the VAS exhaust is automatically aligned to the VFS exhaust subsystem. The operation of the VFS exhaust fans provides the system with the ability for monitoring of radioactivity releases from containment following a fuel handling accident and, if operating, will provide filtration of the containment atmosphere.

If a personnel air lock, or containment spare penetration is open during movement of irradiated fuel assemblies within containment, then the VFS shall be FUNCTIONAL and operating to monitor for the release of radioactivity and to provide filtration of the air inside containment. These penetrations open into the auxiliary building. Upon detection of high radiation in the exhaust air from the auxiliary building, VFS will provide filtered exhaust of these areas. Considering that these penetrations open into the auxiliary building and not directly to the atmosphere, and that the VFS is in operation, an alternate barrier to the release of radioactivity directly to the environment is provided.

APPLICABLE SAFETY ANALYSES

There are no safety analyses that require containment closure during movement of irradiated fuel assemblies within containment, other than those discussed in TS LCO 3.6.7. Fuel handling accidents, analyzed in Reference 1, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of TS LCO 3.9.4, “Refueling Cavity Water Level,” ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 50.34. Standard Review Plan, Section 15.0.1 (Reference 2), defines the dose acceptance limit to be 25% of the limiting dose guideline values.

This TR is included as defense-in-depth.

TECHNICAL REQUIREMENT

This TR provides defense-in-depth against the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. This TR requires that if an equipment hatch, personnel air lock, containment spare penetration, or penetration providing direct access from the containment atmosphere to the outside atmosphere is open during movement of irradiated fuel assemblies within containment, then the VFS shall be FUNCTIONAL and operating to monitor for the release of radioactivity and to provide filtration of the air inside containment.

The VFS is FUNCTIONAL when:

a. One VFS exhaust fan is capable of operating; the associated HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and air circulation can be maintained, including maintaining ≤ -0.03 inches water gauge relative to outside atmospheric pressure;

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BASES (continued)

b. An alternate barrier between the containment atmosphere and the outside atmosphere is in place. The alternate barrier may consist of the staging area in the auxiliary building, or may consist of the staging areas in the auxiliary building and the radiologically controlled areas in the annex building provided the doors from the annex building to the outside atmosphere are closed.

Doors in the alternate barrier which are normally closed may be opened for short periods of time for ingress and egress. The alternate barrier may also consist of a temporary equipment hatch cover that provides equivalent isolation capability.

APPLICABILITY

The containment penetration requirements are applicable during movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by TS LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Containment closure capability in MODES 5 and 6 are addressed by TS LCO 3.6.7.

ACTIONS

The required status for the containment equipment hatch, air locks or spare penetration is either closed, or open with the VFS FUNCTIONAL and operating. The required status for the containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere is either closed by a manual or automatic isolation valve, blind flange or equivalent, or open with the VFS FUNCTIONAL and operating. If the containment equipment hatch or air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

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TECHNICAL REQUIREMENT SURVEILLANCE

TRS 3.9.1.1 verifies that each of the containment penetrations required to be in its closed position is in that position or the VFS is FUNCTIONAL and operating. For the VFS to be considered FUNCTIONAL, this surveillance also requires that an alternate barrier is in place.

TRS 3.9.1.2 verifies the ability of the VFS to maintain a pressure ≤ -0.03 inches water gauge relative to outside atmospheric pressure in the containment and the portions of the auxiliary and/or annex building that comprise the envelope defined as the alternate barrier. This surveillance is performed with the VFS in containment operating. Doors in the alternate barrier which are normally closed may be opened for ingress and egress. The portion of the VAS which services the area enclosed by the alternate barrier is aligned to the VFS exhaust subsystem. The portion of the VAS supplying the alternate barrier is secured. The Frequency of 24 months is consistent with the guidance provided in NUREG 0800, Standard Review Plan, Section 6.5.1 (Ref. 3).

Changes to the alternate barrier are permitted while the TR is applicable provided the configuration has been verified to maintain a pressure ≤ -0.03 inches water gauge relative to outside atmospheric pressure within the 24-month frequency. If implementation of a new, untested alternate barrier configuration is required, movement of irradiated fuel must be suspended while the new configuration is established and verified according to this SR.

TRS 3.9.1.3 addresses that the VFS should be checked periodically to ensure that it functions properly. As the operating conditions on this system are not severe, testing each train within 31 days prior to fuel movement provides an adequate check on this system. Operation with the heaters on for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Operation ensures that heater failure, blockage, fan or motor failure, or excessive vibration can be detected for corrective action.

REFERENCES

1. FSAR Section 15.7.4, “Fuel Handling Accident.”

2. NUREG-0800, Standard Review Plan, Section 15.0.1, Rev. 0.

3. NUREG-0800, Standard Review Plan, Section 6.5.1, Rev. 2, July 1981.

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TRM 3.9 REFUELING OPERATIONS

TRM 3.9.2 Containment Air Filtration System (VFS)

TR 3.9.2 One VFS exhaust subsystem shall be FUNCTIONAL.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required VFS exhaust subsystem not FUNCTIONAL.	A.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.9.2.1	Operate required VFS exhaust subsystem for ≥ 15 continuous minutes with the heaters operating.	Within 31 days prior to fuel movement
TRS 3.9.2.2	Verify the VAS fuel handling area subsystem aligns to required VFS exhaust subsystem on an actual or simulated actuation signal.	24 months
TRS 3.9.2.3	Verify required VFS exhaust subsystem can maintain a pressure ≤ -0.03 inches water gauge relative to outside atmospheric pressure in the fuel handling area.	24 months

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TRM 3.9.2 Containment Air Filtration System (VFS)

BASES

BACKGROUND

The radiologically controlled area ventilation system (VAS) serves the fuel handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas which are provided with a separate ventilation system (VHS). If high airborne radioactivity is detected in the exhaust air from the fuel handling area, the auxiliary building, or the annex buildings, the VAS supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment and the containment air filtration exhaust subsystem starts. The VFS exhaust subsystem prevents exfiltration of unfiltered airborne radioactivity by maintaining the isolated zone at ≤ -0.03 inches water gauge pressure relative to the outside atmosphere. Monitoring of potential releases of radiation is performed in accordance with Technical Specification (TS) 5.5.2, “Radioactive Effluent Control Program.”

For a fuel handling accident, the Reference 3 analysis does not rely on the FUNCTIONALITY of the VAS or VFS exhaust subsystem to meet the offsite radiation exposure limits. This TR is provided as an additional level of defense-in-depth against the possibility of a fission product release from a fuel handling accident in the fuel building. The plant vent radiation detectors monitor effluents discharged from the plant vent to the environment.

Each VFS exhaust subsystem includes one 100% capacity exhaust air filtration unit, and the associated exhaust fan, heater and ductwork.

The filtration units are connected to a ducted system with isolation dampers to provide HEPA filtration and charcoal adsorption of exhaust air from the containment, fuel handling area, radiologically controlled areas of the auxiliary and annex buildings. A gaseous radiation monitor is located downstream of the exhaust air filtration units to provide an alarm if abnormal gaseous releases are detected. The plant vent exhaust flow is monitored for gaseous, particulate and iodine releases to the environment. During conditions of abnormal airborne radioactivity in the fuel handling area, auxiliary and/or annex buildings, the VFS exhaust subsystem provides filtered exhaust to minimize unfiltered offsite releases.

The VAS is described in Reference 1 and the VFS is described in Reference 2.

APPLICABLE SAFETY ANALYSES

The VFS is not required to mitigate the consequences of the limiting design basis accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident does not assume that the VFS provides a filtered exhaust, and its operation would reduce the consequences of the fuel handling accident.

This specification is included for defense-in-depth.

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BASES (continued)

TECHNICAL REQUIREMENT

One VFS exhaust subsystem is required to be FUNCTIONAL to reduce the consequences of a fuel handling accident by filtering the fuel building atmosphere.

A VFS exhaust subsystem is considered FUNCTIONAL when its associated:

a. Exhaust fan is capable of operating, including maintaining ≤ -0.03 inches water gauge relative to outside atmospheric pressure;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function;

c. The associated heater and ductwork are capable of operating.

APPLICABILITY

During movement of irradiated fuel in the fuel handling area, one VFS exhaust subsystem is FUNCTIONAL to alleviate the potential consequences of a fuel handling accident.

ACTIONS

When the required VFS exhaust subsystem is not FUNCTIONAL during movement of irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the TR does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

TECHNICAL REQUIREMENT SURVEILLANCE

TRS 3.9.2.1 verifies the required VFS exhaust subsystem be checked 31 days prior to fuel movement in the fuel handling area to ensure that it functions properly. As the operating conditions on this subsystem are not severe, testing each subsystem within one month prior to fuel movement provides an adequate check on this system. Operation with the heaters on for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Operation ensures that heater failure, blockage, fan or motor failure, or excessive vibration can be detected for corrective action.

TRS 3.9.2.2 verifies that the VAS fuel handling area subsystem aligns to the VFS and that the required VFS exhaust subsystem starts and operates on an actual or simulated actuation signal. During the post-accident mode of operation, the VAS fuel handling area subsystem aligns to the VFS filtered exhaust subsystem. The 24 month Frequency is consistent with Reference 4.

TRS 3.9.2.3 verifies the integrity of the fuel handling area of the auxiliary building enclosure. The ability of the VAS and VFS to maintain pressure ≤ -0.03 inches water gauge relative to outside atmospheric pressure in the fuel handling area of the auxiliary building is periodically tested to verify proper function of the VAS and VFS exhaust subsystem. During this

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BASES (continued)

surveillance, the VAS fuel handling area subsystem is aligned to the operating VFS exhaust subsystem. The fan for the VAS fuel handling area subsystem is off. In this configuration, the required VFS exhaust subsystem is designed to maintain a pressure in the fuel handling area of the auxiliary building ≤ -0.03 inches water gauge relative to outside atmospheric pressure, to prevent unfiltered and unmonitored leakage. Doors may be opened for short periods of time to allow ingress and egress. During this surveillance, the VAS may be servicing the remaining portions of the auxiliary and annex buildings. The Frequency of 24 months is consistent with the guidance provided in NUREG 0800, Standard Review Plan, Section 6.5.1 (Ref. 5).

REFERENCES

1. FSAR Section 9.4.3, “Radiologically Controlled Area Ventilation System.”

2. FSAR Section 9.4.7, “Containment Air Filtration System.”

3. FSAR Section 15.7.4, “Fuel Handling Accident.”

4. Regulatory Guide 1.140 (Rev. 2).

5. NUREG 0800, Standard Review Plan, Section 6.5.1, Rev. 2, July 1981.

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Chapter 17 Quality Assurance

17.1 Quality Assurance During the Design and Construction Phases

See Section 17.7.

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17.2 Quality Assurance During the Operations Phase

See Section 17.5.

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17.3 Quality Assurance During Design, Procurement, Fabrication, Inspection, and/or Testing of Nuclear Power Plant Items

This section outlines the quality assurance program applicable to the design, procurement, fabrication, inspection, and/or testing of items and services for the AP1000 Project. The design for AP1000 is based upon employing the design of AP600 to the maximum extent possible. As a result, a continuous quality program spanning AP600 design as well as AP1000 design has been used. Westinghouse has and will continue to maintain a quality assurance program meeting the requirements of 10 CFR 50 Appendix B for the AP1000 program that will be applicable to the design, procurement, fabrication, inspection, and/or testing activities.

Effective March 31, 1996, activities affecting the quality of items and services for the AP600 Project during design, procurement, fabrication, inspection, and/or testing were being performed in accordance with the quality plan described in "Westinghouse Electric Corporation – Energy Systems Business Unit, Quality Management System," (Reference 1). The Quality Management System (QMS) has been maintained as the Quality Plan for the AP1000 program and subsequent revisions have been submitted to and accepted by the NRC as meeting the requirements of 10 CFR 50 Appendix B.

Prior to introduction of the QMS as the quality plan applicable to the AP1000 project, activities on the AP600/AP1000 program were performed in accordance with topical report WCAP 8370 (References 2 and 3), Westinghouse Energy Systems Business Unit/Power Generation Business Unit Quality Assurance Plan. WCAP 8370 was subsequently superseded by the Westinghouse QMS to describe the quality assurance plan and Westinghouse commitments to meet the requirements of 10 CFR 50 Appendix B.

The current Westinghouse quality plan for work being performed on the AP1000 is the Westinghouse Electric Company Quality Management System (QMS) (Reference 9). The referenced revision of the QMS was accepted by the NRC as meeting the requirements of 10 CFR 50, Appendix B, on December 29, 2014.

A project-specific quality plan was issued to supplement the quality management system document and the topical reports for design activities affecting the quality of structures, systems, and components for the AP600 project (Reference 4). This plan referenced the NQA-1-1989 edition through NQA-1b-1991 addenda and was applicable to work performed for the AP1000 design prior to March 16, 2007.

Effective March 16, 2007, NQA-1 is applicable for work performed for the AP1000 project. As such, a project-specific quality plan is no longer required, and the Westinghouse Electric Company Quality Management System (QMS) (Reference 9) is the quality program for work performed for the AP1000 project. Work by Westinghouse for the AP1000 plant is performed in accordance with the Westinghouse QMS in effect at the time of the work.

Quality Assurance requirements for systems, structures, and components will be graded based on the safety classification as indicated in Section 3.2. Safety-related systems are classified as Equipment Classes A, B and C, and will meet the requirements of 10 CFR 50, Appendix B. For systems, structures, and components included in the regulatory treatment of nonsafety systems (RTNSS), the quality requirements are identified in Table 17-1. See Section 16.3 for systems that should be considered for designation of systems and components included in the regulatory treatment of nonsafety systems.

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While Westinghouse retains the overall responsibility for the AP1000 design, portions of the design are developed by external organizations. Each organization maintains a quality assurance program that meets the NQA-1 criteria that apply to its work scope. In accordance with the QMS, Westinghouse performs an initial evaluation of these programs and monitors their continued effective implementation through audits, surveillance, and evaluation of the performance of external organizations.

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Table 17-1 (Sheet 1 of 3)
Quality Assurance Program Requirements for Systems, Structures, And Components Important to Investment Protection

The following outlines the quality assurance program requirements for suppliers of systems, structures, or components to which the requirements for investment protection short-term availability controls apply.
1. Organization The normal line organization may verify compliance with the requirements of this table. A separate or dedicated quality assurance organization is not required.
2. Quality Assurance Program It is expected that the existing body of supplier's procedures or practices will describe the quality controls applied to the subject equipment. A new or separate QA program is not required.
3. Design Control Measures shall be established to ensure that contractually established design requirements are included in the design. Applicable design inputs shall be included or correctly translated into design documents, and deviations therefrom shall be controlled. Normal supervisory review of the designer's work is an adequate control measure.
4. Procurement Document Control Applicable design bases and other requirements necessary to assure component performance, including design requirements, shall be included or referenced in documents for procurement of items and services, and deviations therefrom shall be controlled.
5. Instructions, Procedures, and Drawings Activities affecting quality shall be performed in accordance with documented instructions, procedures, or drawings of a type appropriate to the circumstances. This may include such things as written instructions, plant procedures, cautionary notes on drawings, and special instructions on work orders. Any methodology which provides the appropriate degree of guidance to personnel performing activities important to the component functional performance will satisfy this requirement.
6. Document Control The issuance and change of documents that specify quality requirements or prescribe activities affecting quality shall be controlled to assure that correct documents are employed.
7. Control of Purchased Items and Services Measures are to be established to ensure that all purchased items and services conform to appropriate procurement documents.

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Table 17-1 (Sheet 2 of 3)
Quality Assurance Program Requirements for Systems, Structures, And Components Important to Investment Protection

8. Identification and Control of Purchased Items Measures shall be established where necessary, to identify purchased items and preserve their investment protection important functional performance capability. Examples of circumstances requiring such control include the storage of environmentally sensitive equipment or material, and the storage of equipment or material that has a limited shelf-life.
9. Control of Special Processes Measures shall be established to control special processes, including welding, heat treating, and non-destructive testing. Applicable codes, standards, specifications, criteria, and other special requirements may serve as the basis of these controls.
10. Inspection Inspections shall be performed where necessary to verify conformance of an item or activity to specified requirements, or to verify that activities are being satisfactory accomplished. Inspections need not be performed by personnel who are independent of the line organization. However, inspections, where necessary, shall be performed by knowledgeable personnel.
11. Test Control Measures shall be established, as appropriate, to test equipment prior to installation to demonstrate conformance with design requirements. Tests shall be performed in accordance with test procedures. Test results shall be recorded and evaluated to ensure that test requirements have been met.
12. Control of Measuring and Test Equipment Measures shall be established to control, calibrate, and adjust measuring and test equipment at specific intervals.
13. Handling, Storage, and Shipping Handling, storage, cleaning, packaging, shipping, and preservation of items shall be controlled to prevent damage or loss and to minimize deterioration.
14. Inspection, Test, and Operating Status Measures shall be established to identify items that have satisfactory passed required tests and inspections, and to indicate status of inspection, test, and operability as appropriate.
15. Control of Nonconforming Items Items that do not conform to specified requirements shall be identified and controlled to prevent inadvertent installation or use.

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Table 17-1 (Sheet 3 of 3)
Quality Assurance Program Requirements for Systems, Structures, And Components Important to Investment Protection

16. Corrective Action Measures shall be established to ensure that failures, malfunctions, deficiencies, deviations, defective components, and nonconformances are properly identified, reported, and corrected.
17. Records Records shall be prepared and maintained to furnish evidence that the above requirements for design, procurement, document control, inspection, and test activities have been met.
18. Audits Audits which are independent of line management are not required, if line management periodically reviews and documents the adequacy of the suppliers process and takes any necessary corrective action. Line management is responsible for determining whether reviews conducted by line management or audits conducted by and organization independent of line management are appropriate. If performed, audits shall be conducted and documents to verify compliance with design and procurement documents, instructions, procedures, drawings, and inspection and test activities.

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17.4 Design Reliability Assurance Program

This subsection presents the AP1000 Design Reliability Assurance Program (D-RAP).

17.4.1 Introduction

The AP1000 D-RAP is implemented as an integral part of the AP1000 design process to provide confidence that reliability is designed into the plant and that the important reliability assumptions made as part of the AP1000 probabilistic risk assessment (PRA) (Reference 5) will remain valid throughout plant life. The PRA quantifies plant response to a spectrum of initiating events to demonstrate the low probability of core damage and resultant risk to the public. PRA input includes specific values for the reliability of the various structures, systems, and components (SSCs) in the plant that are used to respond to postulated initiating events.

The D-RAP, shown in Figure 17.4-1, is implemented during Design Certification. The D-RAP identifies risk-significant SSCs for inclusion into the site Operational Phase Reliability Assurance Activities (OPRAAs) using probabilistic, deterministic, and other methods.

The OPRAAs provides confidence that the operations and maintenance activities performed by the operating plant support should maintain the reliability assumptions made in the plant PRA.

17.4.2 Scope

The D-RAP includes a design evaluation of the AP1000 and identifies the aspects of plant operation, maintenance, and performance monitoring pertinent to risk-significant SSCs. In addition to the PRA, deterministic tools, industry sources, and expert opinion are used to identify and prioritize those risk-significant SSCs.

The quality assurance requirements for non-safety related SSCs within the scope of D-RAP is in accordance with the Quality Assurance Program Description (QAPD), Part III.

17.4.3 Design Considerations

As part of the design process, risk-significant components are evaluated to determine their dominant failure modes and the effects associated with those failure modes. For most components, a substantial operating history is available which defines the significant failure modes and their likely causes.

The identification and prioritization of the various possible failure modes for each component lead to suggestions for failure prevention or mitigation. This information is provided as input to the OPRAAs.

The design reflects the reliability values assumed in the design and PRA as part of procurement specifications. When an alternative design is proposed to improve performance in either area, the revised design is first reviewed to provide confidence that the current assumptions in the other areas are not violated. When a potential conflict exists between safety goals and other goals, safety goals take precedence.

17.4.4 Relationship to Other Administrative Programs

The D-RAP manifests itself in other administrative and operational programs. The technical specifications provide surveillance and testing frequencies for certain risk-significant SSCs, providing confidence that the reliability values assumed for them in the PRA will be maintained during plant operations. Risk-significant systems that provide defense-in-depth or result in significant improvement in the PRA evaluations are included in the scope of the D-RAP.

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The OPRAAs are comprised of site administrative, maintenance, operational, and testing programs to enhance operational phase reliability throughout the designed plant life. As documented in Reference 10 and Reference 12, the following reliability assurance programs are credited as OPRAAs:

- Maintenance Rule Program (Reference 10)
- Quality Assurance Program (Section 17.2)
- Inservice Testing Program (Section 3.9)
- Inservice Inspection Program (Section 5.2 and Section 6.6)
- Technical Specifications Surveillance Test Program (Section 16.1)
- AP1000 Investment Protection Short Term Availability Controls Program (Section 16.3)
- Site Maintenance Program

17.4.5 The AP1000 Design Organization

The AP1000 organization of Section 1.4 formulates and implements the AP1000 D-RAP.

The AP1000 management staff is responsible for the AP1000 design and licensing.

The AP1000 staff coordinates the program activities, including those performed within Westinghouse as well as work completed by the architect-engineers and other supporting organizations listed in Section 1.4.

The AP1000 staff is responsible for development of the D-RAP and the design, analyses, and risk and reliability engineering required to support development of the program. Westinghouse is responsible for the safety analyses, the reliability analyses, and the PRA.

The reliability analyses are performed using common databases from Westinghouse and from industry sources such as INPO and EPRI.

The Risk and Reliability organization is responsible for developing the D-RAP and has direct access to the AP1000 staff. Risk and Reliability is responsible for keeping the AP1000 staff cognizant of the D-RAP risk-significant items, program needs, and status. Risk and Reliability participates in the design change control process for the purpose of providing D-RAP-related inputs to the design process. Additionally, a cognizant representative of Risk and Reliability is present at design reviews. Through these interfaces, Risk and Reliability can identify interfaces between the performance of risk-significant SSCs and the reliability assumptions in the PRA. Meetings between Risk and Reliability and the designer are then held to manage interface issues.

17.4.6 Objective

The objective of the D-RAP is to design reliability into the plant and to maintain the AP1000 reliability consistent with the NRC-established PRA safety goals.

The following goals have been established for the D-RAP:

- Provide reasonable assurance that
  - The AP1000 is designed, procured, constructed, maintained and operated in a manner consistent with the assumptions and risk insights in the AP1000 PRA for these risk-significant SSCs
  - The risk-significant SSCs do not degrade to an unacceptable level during plant operations

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  - The frequency of transients that challenge the AP1000 risk-significant SSCs are minimized
  - The risk-significant SSCs function reliably when they are challenged

- Provide a mechanism for establishing baseline reliability values for risk-significant SSCs identified by the risk determination methods used to implement the Maintenance Rule (10 CFR 50.65) and consistent with PRA reliability and availability design basis assumptions used for the AP1000 design
- Provide a mechanism for establishing baseline reliability values for SSCs consistent with the defense-in-depth functions to minimize challenges to the safety-related systems
- Generate design and operational information to be used for ongoing plant reliability assurance activities

Development of maintenance assessments and recommendations and the site-specific portion of the program is the responsibility of the Combined License applicant.

17.4.7 D-RAP

The definition portion of the D-RAP includes the initial identification of SSCs to be included in the program, implementation of the aspects applicable to design efforts, and definition of the scope, requirements, and implementation options to be included in the later phases.

17.4.7.1 SSCs Identification and Prioritization

The initial task of the D-RAP is identification of risk-significant SSCs to be included within the scope of the program. As shown in Figure 17.4-1, the AP1000 PRA is used to identify those SSCs, consistent with the criteria of Reference 7 for risk achievement worth (RAW), risk reduction worth (RRW), and Fussel-Vesely worth (FVW). Note that, although Reference 7 was developed for AP600, it is directly applicable to AP1000. The review of light water reactor industry experience and industry notices (such as licensee event reports) supports the process. An expert panel is also employed in the selection process.

PRA-based measurements provide information that contributes to the identification and prioritization of SSCs. A component's RAW is the factor by which the plant's core damage frequency increases if the component reliability is assigned the value 0.0. Components with risk achievement worth values of 2 or greater are considered for inclusion in the D-RAP.

RRW is used in the selection process. A component's risk reduction worth is the amount by which the plant's core damage frequency decreases if the component's reliability is assigned the value 1.0. A threshold measure of 1.005 or greater is used as the cutoff. Components with RRW of 1.005 or greater are considered for inclusion in the D-RAP.

FVW is also used in the screening process. This is a measure of an event's contribution to the overall plant core damage frequency. Components with Fussel-Vesely worth of 0.5 percent or greater are considered for inclusion in the D-RAP.

Deterministic considerations are also instrumental in identifying risk-significant SSCs. The deterministic identification of risk-significant SSCs encompasses the following guidelines and considerations:

- ATWS rule (10 CFR 50.62)

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- Loss of all ac power (10 CFR 50.63)
- Post-72-hour actions
- Containment performance
- Adverse interactions with the AP1000 safety-related systems
- Seismic considerations

Nonsafety-related systems identified as risk-significant are considered in the scope of the D-RAP:

- Diverse actuation system
- Non-Class 1E dc and uninterruptible power supply system
- Offsite power, main ac power, and onsite standby power systems
- Normal residual heat removal system
- Component cooling water system
- Service water system

Finally, risk-significant SSCs are selected using industry experience, regulations, and engineering judgment.

17.4.7.1.1 Level 1 PRA and Shutdown Analysis

The Level 1 PRA evaluates accident sequences from initiating events and failures of safety functions to core damage events. The probability of core damage and the identification of dominant contributors to that state are also determined in this analysis.

A low-power and shutdown assessment is conducted to address concerns about risk of operations during shutdown conditions. It encompasses operation when the reactor is in a subcritical state or is in a transition between subcriticality and power operation up to 5 percent of rated power. It consists of a Level 1 PRA and an evaluation of release frequencies and magnitudes.

Included in the D-RAP are events that meet the threshold risk achievement worth, risk reduction worth, or Fussel-Vesely worth values defined in Subsection 17.4.7.1.

17.4.7.1.2 Level 2 Analysis

The Level 2 analysis predicts the plant response to severe accidents and offsite fission product releases. Specifically, the analysis includes the following sections:

- Evaluating severe accident phenomena and fission product source terms
- Modeling the containment event tree
- Analyzing hydrogen burn, mixing, and igniter placement
- Modeling the AP1000 utilizing the MAAP4 code

Equipment used in the prevention of severe accidents and severe post-accident boundary conditions is credited in the Level 1 and Level 2 PRA analyses. An example of this preventive equipment is the reactor coolant system automatic depressurization system (ADS). Successful depressurization leads to core cooling, and in the event that injection fails, results in a low pressure core damage sequence that has fewer uncertainties and can be more easily mitigated than high pressure core damage.

The containment event tree used in the AP1000 Level 2 PRA examines the operation of equipment which mitigates the threat to the containment from severe accident phenomena. The systems credited for the mitigation of large fission product releases are containment isolation, passive containment cooling water (PCS), and operator action to flood the cavity by opening the recirculation valves and energizing the hydrogen igniters.

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17.4.7.1.3 External Event Analyses

These analyses consider the events whose cause is external to all the systems associated with normal and emergency operations situations. They include the following:

- Internal flood
- Seismic margins analysis
- External events evaluations (such as high winds and tornados, external floods, and transportation accidents)
- Fire

The internal flood analysis identifies, analyzes, and quantifies the core damage risk contribution as a result of internal flooding during at-power and shutdown conditions. The analysis models potential flood vulnerabilities in conjunction with random failures modeled as part of the internal events PRA.

The seismic margins analysis identifies potential vulnerabilities and demonstrates seismic margin beyond the safe shutdown earthquake. The capacity of those components required to bring the plant to a safe, stable shutdown is evaluated.

17.4.7.1.4 Expert Panel

Meetings were held among Systems Engineering, PRA, and Reliability Engineering to perform the final selection of SSCs that should be included in the D-RAP. As shown in Figure 17.4-1, industry-wide information sources and engineering judgment were employed in considering the addition of SSCs to the D-RAP.

17.4.7.1.5 SSCs to be Included in D-RAP

Table 17.4-1 lists the non-site-specific SSCs included in the D-RAP. In Figure 17.4-1, this list is denoted as "Risk-significant items (non-site-specific)." For each item listed in the "SSC" column, there is a corresponding "Rationale" given. Items whose values exceed the thresholds for RAW or RRW are included and noted as such. Other SSCs are included based upon their significance to Level 2 analysis, external event analyses, or seismic margin analysis. Additional items are included based upon an expert panel review. The "Insights and Assumptions" column provides additional insights into the selection process.

The use of Fussel-Vesely worth resulted in no SSC selections.

17.4.7.2 Not Used

17.4.7.2.1 Not Used

17.4.7.3 Not Used

17.4.7.4 D-RAP Implementation

The following is an example of a system that was reviewed and modified under the D-RAP. The design and analytical results presented here are intended as an example.

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The automatic depressurization system, which is part of the reactor coolant system, acts in conjunction with the passive core cooling system to mitigate design basis accidents. The automatic depressurization system valves are discussed in Subsection 5.4.6.

An earlier AP600 automatic depressurization system design contained four depressurization stages, with motor-operated valves in all stages. Preliminary PRA analysis established that fourth stage failure, in certain combination with failures of other stages, was a major contributor to core damage frequency. Thus, it was concluded that the fourth stage valves should be diverse in design from the valves in other stages to reduce common cause failure.

As a result of joint meetings among the AP600 PRA, Design, and staff organizations to discuss core melt frequency improvements, the fourth stage automatic depressurization system was changed from a motor-operated valve to a squib (explosively actuated) valve. The new configuration of the system is shown in the reactor coolant system P&ID (Figure 5.1-5). An example of the analytical results that reflect this change is provided in Table 17.4-2. This design feature is included in the AP1000 design to maintain the core melt frequency improvements included in the AP600 design.

As part of the evaluation of the squib valves, a failure modes and effects analysis (FMEA) was prepared to identify subcomponent failures and critical items that could lead to hazardous or abnormal conditions of the automatic depressurization system and the plant. The identification of failure modes facilitated the development of recommended maintenance and in-service testing activities to maximize valve reliability.

The squib valve is a completely static electromechanical assembly. Prior to activation, there are no moving parts. No powered components are needed to hold a stem seat or globe in place by torque, solenoid coils, or friction. The explosive actuator is a simple, passive device that is triggered by an applied voltage.

Because the automatic depressurization system fourth stage valves perform safety-related functions, they will be subject to in-service testing to verify that they are ready to function in an accident.

Example FMEA results for the fourth stage squib valves and the second and third stage motor-operated valves are included in Table 6.3-3.

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17.4.8 Glossary of Terms

D-RAP	Design Reliability Assurance Program – performed as part of the AP1000 design effort to assure that the reliability assumptions of the PRA remain valid throughout the plant operating lifetime.
FVW	Fussel-Vesely Worth
MR	Maintenance Rule
OPRAAs	Operational Phase Reliability Assurance Activities
PRA	Probabilistic Risk Assessment
RAW	Risk Achievement Worth
Risk-significant	Any SSC determined in the PRA or by risk-significance analysis (e.g., Level 2 PRA and shutdown risk analysis) to be a major contributor to overall plant risk
RRW	Risk Reduction Worth
RTNSS	Regulatory Treatment of Nonsafety Systems
SSC	Structures, Systems, and Components

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Table 17.4-1 (Sheet 1 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
System: Component Cooling Water (CCS)		
Component Cooling Water Pumps (CCS-MP-01A/B)	EP	These pumps provide cooling of the normal residual heat removal system (RNS) and the spent fuel pool heat exchanger. Cooling the RNS heat exchanger is important to investment protection during shutdown reduced-inventory conditions. CCS valve realignment is not required for reduced-inventory conditions.
System: Containment System (CNS)		
Containment Vessel (CNS-MV-01)	EP, L2	The containment vessel provides a barrier to steam and radioactivity released to the atmosphere following accidents.
Hydrogen Igniters (VLS-EH-1 through -66)	RAW/CCF, L2, Regulations	The hydrogen igniters provide a means to control H_2 concentration in the containment atmosphere, consistent with the hydrogen control requirements of 10 CFR 50.34f.
System: Chemical and Volume Control System (CVS)		
Makeup Pumps (CVS-MP-01A/B)	EP	These pumps provide makeup to the RCS to accommodate leaks and to provide negative reactivity for shutdowns, steam line breaks, and ATWS.
Makeup Pump Suction and Discharge Check Valves (CVS-PL-V113, -V160A/B)	EP	These CVS check valves are normally closed and have to open to allow makeup pump operation.
Letdown Isolation Valves (CVS-PL-V045, -V047)	RAW	The CVS letdown isolation valves automatically close to prevent excessive reactor coolant letdown and provide containment isolation. These containment isolation valves are important in limiting offsite releases following core melt accidents.
System: Diverse Actuation System (DAS)		
DAS Processor Cabinets and Control Panel (used to provide automatic and manual actuation) (DAS-JD-001, -002, -003, OCS-JC-20)	RAW	The DAS is diverse from the PMS and provides automatic and manual actuation of selected plant features including control rod insertion, turbine trip, passive residual heat removal (PRHR) heat exchanger actuation, core makeup tank actuation, isolation of critical containment lines, and passive containment cooling system (PCS) actuation.
Auxiliary Building UPS Distribution Panels (EDS2-EA-12, EDS3-EA-14A)	RAW	These panels distribute power to the DAS equipment.

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Table 17.4-1 (Sheet 2 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
Control Cabinets for the Rod Drive MG Sets (Generator Field Control Relays) (PLS-JD-RDM001, PLS-JD-RDM002)	RAW	These relays open on a DAS reactor trip signal demand to de-energize the control rod MG sets and allow the rods to drop.
System: Main ac Power System (ECS)		
Reactor Coolant Pump Switchgear (ECS-ES-31, -32, -41, -42, -51, -52, -61, -62)	RAW/CCF	These breakers open automatically to allow core makeup tank operation.
Ancillary Diesel Generators (ECS-MS-01, -02)	EP	For post-72 hour actions, these generators are available to provide power for Class 1E monitoring, MCR lighting and for refilling the PCS water storage tank and spent fuel pool.
6900 Vac Buses (ECS-ES-1, -2)	RAW	These are ac power buses fed by the onsite DGs and offsite power.
System: Main and Startup Feedwater System (FWS)		
Startup Feedwater Pumps (FWS-MP-03A/B)	EP	The startup feedwater system pumps provide feedwater to the steam generator. This capability provides an alternate core cooling mechanism to the PRHR heat exchangers for non-loss-of-coolant-accidents or steam generator tube ruptures.
System: General I&C[4]		
Low Pressure/DP Sensors – IRWST Wide Range Level (PXS-046, -047, -048) – IRWST Lower Narrow Range Level (PXS-066, -067, -068, -069)	RAW/CCF	These in-containment refueling water storage tank (IRWST) level sensors support PMS functions. They are used in automatic actuation of refueling cavity and SFS isolation, and IRWST recirculation actions, and they provide indications to the operator.
High Pressure/DP Sensors – RCS Hot Leg Level (RCS-160A/B) – Pressurizer Pressure (RCS-191A/B/C/D) – Pressurizer Level (RCS-195A/B/C/D) – SG Narrow-Range Level (SGS-001, -002, -003, -004, -005, -006, -007, -008) – SG Wide-Range Level (SGS-011, -012, -013, -014, -015, -016, -017, -018)	RAW/CCF/EP	The following sensors are included in this group. These sensors support PMS and PLS functions. They are used in reactor trip and ESF functions, and provide indications to the operator. Main feedwater and nonsafety-related startup feedwater flow sensors support startup feedwater actuation and other safety-related startup feedwater flow sensors support PRHR actuation. The hot leg level sensors automatically actuate the IRWST injection and automatic depressurization system (ADS) valves during shutdown conditions.

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Table 17.4-1 (Sheet 3 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
– Main Steam Line Pressure (SGS-030, -031, -032, -033, -034, -035, -036, -037) – Main Feedwater Low-Range Flow (FWS-050A/C/E, -051A/C/E) – Startup Feedwater Flow (SGS-055A/B/C/D/E, -056A/B/C/D/E)		
CMT Level Sensors (PXS-011A/B/C/D, -012A/B/C/D, -013A/B/C/D, -014A/B/C/D)	RAW/CCF	These level sensors provide input for automatic actuation of the ADS. They also provide indications to the operator.
System: Class 1E DC Power and Uninterruptible Power System (IDS)		
250 Vdc 24-hour Buses, Batteries, Inverters, and Chargers (IDSA-DB-1A/B, IDSB-DB-1A/B, IDSC-DB-1A/B, IDSD-DB-1A/B, IDSA-DU-1, IDSB-DU-1, IDSC-DU-1, IDSD-DU-1, IDSA-DC-1, IDSB-DC-1, IDSC-DC-1, IDSD-DC-1, IDSA-DS-1, IDSB-DS-1, IDSC-DS-1, IDSD-DS-1)	RAW/CCF	The batteries provide power for the PMS and safety-related valves. The chargers are the preferred source of power for Class 1E dc loads and are the source of charging for the batteries. The inverters provide uninterruptible ac power to the I&C system. The buses distribute power to the Class 1E dc loads.
250 Vdc and 120 Vac Distribution Panels (IDSA-DD-1, -EA-1/2, IDSB-DD-1, -EA-1/2/3, IDSC-DD-1, -EA-1/2/3, IDSD-DD-1, -EA-1/2)	RAW	These panels distribute power to components in the plant that require 1E power support and for the PMS.
Fused Transfer Switch Boxes (IDSA-DF-1, IDSB-DF-1/-2, IDSC-DF-1/-2, IDSD-DF-1)	RAW	The fused disconnect switches connect the different levels of Class 1E distribution panels.

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Table 17.4-1 (Sheet 4 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
250 Vdc Motor Control Centers (IDSA-DK-1, IDSB-DK-1, IDSC-DK-1, IDSD-DK-1)	EP	These buses provide power for the PMS and safety-related valve operation.
System: Passive Containment Cooling System (PCS)		
Recirculation Pumps (PCS-MP-01A/B)	EP	These pumps provide the motive force to refill the PCS water storage tank during post-72 hour support actions.
PCCWST Drain Isolation Valves (PCS-PL-V001A/B/C)	EP, L2	These valves (two AOVs and one MOV) open automatically to drain water from a water storage tank onto the outside surface of the containment shell. This water provides evaporative cooling of the containment shell following accidents.
System: Plant Control System (PLS)		
PLS Actuation Hardware (Control functions listed in Note 5)	RAW/CCF	This common cause failure event is assumed to disable all logic outputs from the PLS associated with CVS reactor makeup, RNS reactor injection, spent fuel cooling, component cooling of RNS SFS heat exchangers, service water cooling of CCS heat exchangers, standby diesel generators, and hydrogen igniters.
PLS Actuation Software (Control functions listed in Note 5)	RAW/CCF	This common cause failure event is assumed to disable the software in the PLS associated with CVS reactor makeup, RNS reactor injection, spent fuel cooling, component cooling of RNS SFS heat exchangers, service water cooling of CCS heat exchangers, standby diesel generators, and hydrogen igniters.
System: Protection and Safety Monitoring System (PMS)		
PMS Actuation Software	RAW/CCF	The PMS software provides the automatic reactor trip and ESF actuation functions listed in Tables 7.2-2 and 7.3-1.
PMS Actuation Hardware	RAW/CCF	The PMS hardware provides the automatic reactor trip and ESF actuation functions listed in Tables 7.2-2 and 7.3-1.
Main Control Room (MCR) 1E Displays and System Level Controls (OCS-JC-10, -11)	RAW/CCF	This includes the Class 1E PMS displays and controls. These displays and system level controls provide important plant indications to allow the operator to monitor and control the plant during accidents.
Reactor Trip Switchgear (PMS-JD-RTS A01/02, B01/02, C01/02, D01/02)	RAW/CCF	These breakers open automatically to allow insertion of the control rods.

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Table 17.4-1 (Sheet 5 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
System: Passive Core Cooling System (PXS)		
IRWST Hood Vent Covers (PXS-MY-Y21, PXS-MY-Y22, PXS-MY-Y23, PXS-MY-Y24, PXS-MY-Y25, PXS-MY-Y26, PXS-MY-Y27, PXS-MY-Y28, PXS-MY-Y29, PXS-MY-Y30, PXS-MY-Y31, PXS-MY-Y32, PXS-MY-Y33, PXS-MY-Y41, PXS-MY-Y47, PXS-MY-Y48)	RAW/CCF	IRWST hood vent covers provide the functionality to vent steam into containment. The IRWST hood vent covers prevent debris from entering the IRWST and minimize evaporation. Safety functions support hydrogen venting strategy and prevent overpressurization of the IRWST.
IRWST SG Wall Vent Covers (PXS-MY-Y61, PXS-MY-Y62, PXS-MY-Y63, PXS-MY-Y64)	RAW/CCF	IRWST SG wall vent covers provide the functionality to vent steam into containment. The IRWST SG wall vent covers prevent debris from entering the IRWST and minimize evaporation. Safety functions support hydrogen venting strategy and prevent overpressurization of the IRWST.
IRWST Overflow Weir Covers (PXS-MY-Y71, PXS-MY-Y72, PXS-MY-Y73, PXS-MY-Y74, PXS-MY-Y75, PXS-MY-Y76)	RAW/CCF	IRWST overflow weir covers allow water overflow into the refueling cavity during ADS blowdown and PRHR Hx operation. The IRWST overflow weir covers prevent debris from entering the IRWST and minimize evaporation. Safety functions accommodate volume and mass increases and support hydrogen venting strategy.
IRWST Screens (PXS-MY-Y01A/B/C)	RAW/CCF	The IRWST injection lines provide long-term core cooling following a LOCA. These screens are located inside the IRWST and prevent large particles from being injected into the RCS. They are designed so that they will not become obstructed.
Containment Recirculation Screens (PXS-MY-Y02A/B)	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. The screens are located in the containment and prevent large particles from being injected into the RCS. They are designed so that they will not become obstructed.
CMT Discharge Isolation Valves (PXS-PL-V014A/B, PXS-PL-V015A/B)	RAW/CCF	These air-operated valves automatically open to allow core makeup tank injection.
CMT Discharge Check Valves (PXS-PL-V016A/B, PXS-PL-V017A/B)	RAW/CCF	These check valves are normally open. They close during rapid accumulator injection.
Accumulator Discharge Check Valves (PXS-PL-V028A/B, -V029A/B)	RAW/CCF	These check valves open when the RCS pressure drops below the accumulator pressure to allow accumulator injection.
PRHR Heat Exchanger Control Valves (PXS-PL-V108A/B)	RAW/CCF	The PRHR heat exchangers provide core cooling following non-LOCAs, steam generator tube ruptures, and anticipated transients without scram. The air-operated valves automatically open to initiate PRHR heat exchanger operation.

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Table 17.4-1 (Sheet 6 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
Containment Recirculation Squib Valves (PXS-PL-V118A/B, PXS-PL-V120A/B)	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. These squib valves open automatically to allow containment recirculation when the IRWST level is reduced to about the same level as the containment level. These squib valves can also allow long-term core cooling to be provided by the RNS pumps. These squib valves can provide a rapid flooding of the containment to support in-vessel retention during a severe accident.
IRWST Injection Check Valves (PXS-PL-V122A/B, -V124A/B)	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. These check valves open when the IRWST level is reduced to approximately the same level as the containment level.
IRWST Injection Squib Valves (PXS-PL-V123A/B, -V125A/B)	RAW/CCF	The IRWST injection lines provide long-term core cooling following a LOCA. These squib valves open automatically to allow injection when the RCS pressure is reduced to below the IRWST injection head.
IRWST Gutter Bypass Isolation Valves (PXS-PL-V130A/B)	RAW/CCF	These valves direct water collected in the IRWST gutter to the IRWST. This capability extends PRHR heat exchanger operation.
System: Reactor Coolant System (RCS)		
ADS Stage 1/2/3 Valves (MOV) (RCS-PL-V001A/B, -V002A/B, -V003A/B, -V011A/B, -V012A/B, -V013A/B)	RAW/CCF	The ADS provides a controlled depressurization of the RCS following LOCAs to allow core cooling from the accumulator, IRWST injection, and containment recirculation. The ADS provides "bleed" capability for feed/bleed cooling of the core. The ADS also provides depressurization of the RCS to prevent a high-pressure core melt sequence.
ADS Stage 4 Valves (Squib) (RCS-PL-V004A/B/C/D)	RAW/CCF	The ADS provides a controlled depressurization of the RCS following LOCAs to allow core cooling from the accumulator, IRWST injection, and containment recirculation. The ADS provides "bleed" capability for feed/bleed cooling of the core. The ADS also provides depressurization of the RCS to prevent a high-pressure core melt sequence.
Pressurizer Safety Valves (RCS-PL-V005A/B)	RRW	These valves provide overpressure protection of the RCS.

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Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
Reactor Vessel Insulation Water Inlet and Steam Vent Devices (RXS-MN-01)	EP	These devices provide an engineered flow path to promote in-vessel retention of the core in a severe accident.
Reactor Cavity Doorway Damper	EP	This device provides a flow path to promote in-vessel retention of the core in a severe accident.
Fuel Assemblies (157 assemblies with tag numbers beginning with RXS-FA)	SMA	The nuclear fuel assembly includes the fuel pellets, fuel cladding, and associated support structures. This equipment, which provides a first barrier for release of radioactivity and allows for effective core cooling, had the least margin in the seismic margin analysis.
System: Normal Residual Heat Removal System (RNS)		
Residual Heat Removal Pumps (RNS-MP-01A/B)	RAW/CCF	These pumps provide shutdown cooling of the RCS. They also provide an alternate RCS lower pressure injection capability following actuation of the ADS. The operation of these pumps is important to investment protection during shutdown reduced-inventory conditions. RNS valve realignment is not required for reduced-inventory conditions.
RNS Motor-Operated Valves (RNS-PL-V011, -V022, -V023, -V055)	RRW	These MOVs align a flow path for nonsafety-related makeup to the RCS following ADS operation, initially from the cask loading pit and later from the containment.
RNS Stop Check Valves (RNS-PL-V015A/B), RNS Check Valves (RNS-PL-V017 A/B)	CCF/EP	These stop check valves and check valves are in the discharge of the RNS pumps. They prevent backflow from the RCS.
RNS Check Valves (RNS-PL-V007 A/B, -V013, -V056)	L2 RAW/EP	Check valves V007 A/B and V013 provide a flow path from the RNS pumps to the RCS. Failure of these valves to open will result in the loss of long-term cooling from the RNS. Check valve V056 provides a flow path from the cask loading pit to the RNS pump inlet.
System: Spent Fuel Cooling System (SFS)		
Spent Fuel Cooling Pumps (SFS-MP-01A/B)	EP	These pumps provide flow to the heat exchangers for removal of the design basis heat load.
System: Steam Generator System (SGS)		
Main Steam Safety Valves (SGS-PL-V030A/B, -V031A/B, -V032A/B, -V033A/B, -V034A/B, -V035A/B)	RRW	The steam generator main steam safety valves provide overpressure protection of the steam generator. They also provide core cooling by venting steam from the steam generator.

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Table 17.4-1 (Sheet 8 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
Main Steam and Feedwater Isolation Valves (SGS-PL-V040A/B, -V057A/B)	RAW/EP	The steam generator main steam and feedwater isolation valves provide isolation of the steam generator following secondary line breaks and steam generator tube rupture.
System: Service Water System (SWS)		
Service Water Pumps and Cooling Tower Fans (SWS-MP-01A/B, SWS-MA-01A/B)	EP	These pumps and fans provide cooling of the CCS heat exchanger which is important to investment protection during shutdown reduced-inventory conditions. Service water system valve realignment is not required for reduced-inventory conditions.
System: Nuclear Island Nonradioactive Ventilation System (VBS)		
VBS MCR and I&C Rooms B/C Ancillary Fans (VBS-MA-10A/B, -11, -12)	EP	For post-72 hour actions, these fans are available to provide cooling of the MCR and the two I&C rooms (B/C) that provide post-accident monitoring.
System: Containment Air Filtration System (VFS)		
VFS Containment Purge Isolation Valves (VFS-PL-V003, -V004, -V009, -V010)	RAW	The VFS containment purge isolation valves provide isolation of containment following an accident. These containment isolation valves are important in limiting offsite releases following core melt accidents.
System: Chilled Water System (VWS)		
Air Cooled Chillers and Pumps (VWS-MS-02, -03, VWS-MP-02, -03)	EP	This VWS subsystem provides chilled cooling water to the CVS makeup pump room. The pumps and chillers are important components of the VWS.
System: Liquid Radwaste System (WLS)		
Sump Containment Isolation Valves (WLS-PL-V055, -V057)	RAW	The sump containment isolation valves provide isolation of containment following an accident. These containment isolation valves are important in limiting offsite releases following core melt accidents.
System: Onsite Standby Power System (ZOS)		
Onsite Diesel Generators (ZOS-MS-05A/B)	RAW/CCF	These diesel generators provide ac power to support operation of nonsafety-related equipment such as the startup feedwater pumps, CVS pumps, RNS pumps, CCS pumps, SWS pumps, and the PLS. Providing ac power to the RNS and the equipment necessary to support its operation is important to investment protection during reduced inventory conditions.
Engine Room Exhaust Fans (VZS-MY-V01A/B, -V02A/B)	RAW/CCF	These fans provide ventilation of the rooms containing the onsite diesel generators.

Notes:

1. Only includes equipment at the **component** level. Other parts of the SSC or support systems are not included unless specifically listed.
2. Definition of Rationale Terms:
   CCF = Common Cause Failure (for the SSCs whose inclusion rationale is RAW/CCF, the RAW is based on common cause failure of two or more of the specified SSCs.
   EP = Expert Panel
   RAW = Risk Achievement Worth
   RRW = Risk Reduction Worth
   SMA = Seismic Margin Analysis
3. Maintenance/surveillance recommendations for equipment are documented in each appropriate section.
4. This category captures instrumentation and control equipment common cause failures across systems.
5. The PLS provides control of the following functions:
   CVS Reactor Makeup
   RNS Reactor Injection from Cask Loading Pit
   Startup Feedwater from CST
   Spent Fuel Cooling
   Component Cooling of RNS and SFS Heat Exchangers
   Service Water Cooling of the CCS Heat Exchangers
   Onsite Diesel Generators
   Hydrogen Igniters

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Table 17.4-2
Example of Risk-Significant Ranking of SSCs for the Automatic Depressurization System

Rank(1)	Event Code	Description
1	ED3MOD07	EDS3 EA1 distribution panel failure or unavailable due to testing and maintenance
2	AD4MOD07, AD4MOD08, AD4MOD09, AD4MOD10	Hardware failure of 2 of 4 automatic depressurization system Stage 4 squib valves
3	EC1BS001TM, ECBS012TM, EC1BS121TM, EC2BS002TM, EC2BS022TM, EC2BS221TM	Unavailability of bus ECS ES due to unscheduled maintenance
4	AD2MOD01, AD2MOD02, AD2MOD03, AD2MOD04	Hardware failure of 2 of 4 automatic depressurization system Stages 2 and 3 of lines 1 and 2 (includes motor-operated valves)
5	EC0MOD01	Main generator breaker ES01 fails to open
6	ED3MOD01	Fixed component fails: circuit breaker, inverter or static transfer switch
7	Z01MOD01, Z02MOD01	Diesel generator fails to start and run or breaker 102 fails to close
8	Z02DG001TM, Z02DG001TM	Standby diesel generator unavailable due to testing and maintenance

Note:
1. The ranking is in the order of decreasing risk achievement component importance.

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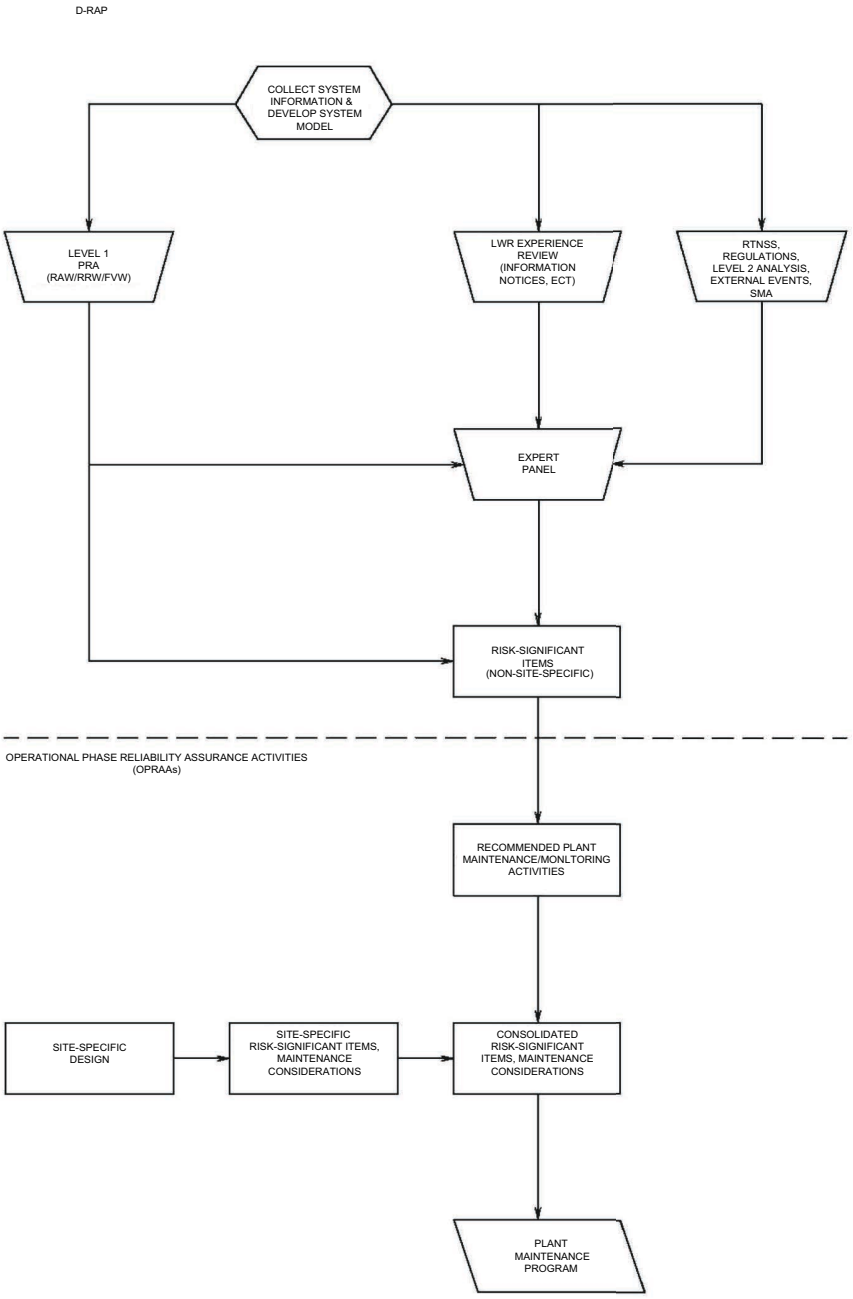

Figure 17.4-1
Design Reliability Assurance Program and Operational Phase Reliability Assurance Activities

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17.5 Quality Assurance Program Description

The Quality Assurance Program in place during the design, construction, and operations phases is described in the QAPD, which is maintained as a separate document. This QAPD is incorporated by reference (see Table 1.6-201). This QAPD is based on NEI 06-14A, “Quality Assurance Program Description” (Reference 201).

Conformance statements for QA-related Regulatory Guides (including Regulatory Guides 1.28, 1.30, 1.33, 1.38, 1.39, 1.94, and 1.116) are provided in Appendix 1A. While many Regulatory Guide positions can be identified as applicable to the scope of work identified and addressed by the DCD and others can be identified as applicable to the scope of work identified and addressed by the COLA, some QA guidance related positions could be accomplished by either scope of work and thus be addressed in either the DCD or the COLA. These positions are primarily dependent on who performs the work. The DCD conformance statement indicates an exception to apply NQA-1. The COLA identifies an exception to apply NQA-1. Per Section 17.3, WEC work performed up to March 15, 2007 applied a 1991 version of the standard. A 1994 version of the standard is applied for work performed after that date by WEC. If the work is performed under the applicant’s COL program, the 1994 version of NQA-1 identified in the COLA QAPD is applied. Thus, DCD scope (identified in Appendix 1A) and “remaining scope” differentiate the application of the guidance identified in these Regulatory Guides.

Table 13.4-201 provides milestones for operational quality assurance program implementation.

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17.6 Maintenance Rule Program

This section incorporates NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52" Reference 202.

Table 13.4-201 provides milestones for maintenance rule program implementation.

The Maintenance Rule (MR) Program provides assurance that structures, systems, and components (SSCs) within the scope of the program remain reliable and capable of fulfilling their intended functions and provides processes for assessing and managing potential increases in risk that might result from proposed maintenance activities. The MR Program meets the requirements of 10 CFR 50.65 Reference 203.

17.6.1 Maintenance Rule Program Description

The MR program follows the guidance in NUMARC 93-01 Reference 204, as endorsed and modified by Regulatory Guide 1.160, Reference 205, without any exceptions that could materially and negatively affect the effectiveness of the program. The principal functions of the program are described in the following subsections.

The MR program includes appropriate control of procedures, documents, computer software and data, as applicable.

17.6.1.1 Maintenance Rule Scoping per 10 CFR 50.65(b)

17.6.1.1.a The SSCs within the scope of the MR program include safety-related SSCs and certain non-safety-related SSCs, as determined using a MR scoping procedure. The scoping procedure addresses:

- Safety-related SSCs.
- Non-safety-related SSCs that mitigate accidents or transients.
- Non-safety-related SSCs that are used in Emergency Operating Procedures, where 'used' means directly used to mitigate the accident or transient via explicit reference in the EOP or used in steps of procedures referenced by the EOP. Additionally, SSCs explicitly referenced in back-up or lower-tier methods in the EOPs and provide reasonable assurance of mitigation success, or whose use is implied in an EOP and essential to the completion of an EOP step, are considered within scope of the Maintenance Rule.
- Non-safety-related SSCs whose failure prevents safety-related SSCs from fulfilling their safety-related functions.
- Non-safety-related SSCs whose failure causes scrams or actuates safety systems.

The SSCs within the scope of the MR program are evaluated against performance criteria to determine which SSCs will have goals established and monitoring activities performed in accordance with 10 CFR 50.65(a)(1).

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17.6.1.1.b Safety significance classifications and bases of in-scope SSCs, e.g., high safety significance (HSS) or low safety significance (LSS), are determined using processes consistent with Subsection 9.3.1 of NUMARC 93-01. They include determination of risk significance criteria and appropriate consideration of operating experience, generic failure data, component reliability information, probabilistic risk assessment (PRA) insights, and the recommendations of an expert panel. All SSCs identified as risk-significant via the Reliability Assurance Program for the design phase (DRAP – see Section 17.4) are included within the initial MR scope as HSS SSCs. This includes risk-significant SSCs identified as part of the design certification phase or follow-on COL applicant/holder phases of DRAP.

17.6.1.1.c The expert panel is established in accordance with NUMARC 93-01 prior to fuel load authorization and utilizes operating, maintenance and systems expertise, PRA insights, and other applicable information to update and maintain the MR scope and SSC classifications.

17.6.1.2 Monitoring and Corrective Action per 10 CFR 50.65(a)(1)

SSCs within the scope of the MR are initially classified as (a)(2) (ref. Section 17.6.1.3), except where it is determined that an SSC should be initially classified as (a)(1), e.g., an SSC that fails during start-up testing.

SSCs that do not meet performance criteria established for (a)(2) monitoring (ref. Section 17.6.1.3) are evaluated for (a)(1) classification in accordance with MR program procedures, with recommended corrective actions identified as appropriate. Necessary corrective actions are implemented in accordance with the site Corrective Action Program. The MR expert panel reviews whether SSCs are to be classified as (a)(1). Monitoring goals are established for SSCs classified as (a)(1), as appropriate, commensurate with the SSCs' safety significance, and considering applicable industry operating experience, with the objective of providing reasonable assurance that the SSC is proceeding to acceptable performance levels and that the corrective actions taken were effective.

For SSCs that do not meet established (a)(1) monitoring goals following corrective actions initially identified and implemented, appropriate additional corrective actions are taken.

17.6.1.3 Preventive Maintenance per 10 CFR 50.65(a)(2)

Monitoring as specified in 10 CFR 50.65(a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance (PM), such that the SSC remains capable of performing its intended function.

The MR program includes procedures for managing SSC performance in accordance with 10 CFR 50.65(a)(2) requirements during plant operation consistent with NUMARC 93-01. To monitor the effectiveness of the maintenance performed on the various SSCs, performance criteria are established at the plant, system, train, or component level commensurate with safety, risk significance and SSC function. SSC performance criteria (e.g., failure rate, unavailability or condition-based) are chosen that are reasonable, measurable, and technically appropriate for the purpose of timely identification of degraded SSC performance or condition. For risk-significant SSCs identified via DRAP, performance criteria are consistent with the reliability and availability assumptions used in the PRA.

When a performance criterion is not met, the SSC is evaluated for (a)(1) classification in accordance with MR program procedures, including review by the Expert Panel. Should the Expert Panel conclude that the SSC should not be classified as (a)(1), or that no (a)(1) monitoring goals need be

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established, a technical justification establishing the appropriateness of continued management of SSC performance under (a)(2) is documented and maintained.

SSCs that provide little or no contribution to system safety function or can be allowed to run to failure due to an acceptable risk may be categorized in a “run-to-failure” status (i.e., perform corrective maintenance rather than preventive maintenance) consistent with NUMARC 93-01.

Preventive maintenance is subject to risk assessment and management per 10 CFR 50.65(a)(4) (ref. Section 17.6.1.5).

17.6.1.4 Periodic Evaluation of Monitoring and Preventive Maintenance per 10 CFR 50.65(a)(3)

The MR program includes procedures for the periodic evaluation of the performance and condition monitoring activities and associated goals and preventive maintenance activities in accordance with 50.65(a)(3). The following considerations are included:

- how procedures govern the scheduling and timely performance of (a)(3) evaluations.
- documenting, reviewing and approving evaluations, providing and implementing results.
- review of 50.65(a)(1) goals and 50.65(a)(2) performance criteria, condition monitoring criteria, SSC performance and condition history and effectiveness of corrective action
- making adjustments to achieve or restore balance between reliability and availability.
- industry operating experience.

17.6.1.5 Risk Assessment and Risk Management per 10 CFR 50.65(a)(4)

The MR program includes procedures for maintenance risk assessment and management in accordance with 10 CFR 50.65(a)(4), employing the methods described in NUMARC 93-01, Section 11. The risk from maintenance activities is both assessed (i.e., using a risk-informed process to evaluate the overall contribution to risk of the planned maintenance activities) and managed (i.e., providing plant personnel with proper awareness of the risk, and taking actions as appropriate to control the risk).

The MR program and procedures reflect, as appropriate, consideration of issues associated with grid/offsite power reliability as identified in NRC Generic Letter 2006-02, items 5 and 6.

17.6.2 Maintenance Rule Training and Qualification

The MR program is supported by appropriate training and qualification for designated personnel. Training is commensurate with MR responsibilities, including MR program administration, the expert panel process, operations, engineering, maintenance, licensing, and plant management, as appropriate. Maintenance Rule Program training and qualification materials are based on regulatory requirements and guidance, and training records are maintained in accordance with plant procedures.

17.6.3 Maintenance Rule Program Relationship With Reliability Assurance Activities

Reliability during the operations phase is assured through the implementation of operational programs, i.e., the MR program, the Quality Assurance Program, inservice inspection and testing programs, the Technical Specifications surveillance test program, and maintenance programs.

Descriptions of the programs are provided in the following chapters/sections:

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The maintenance rule program (Section 17.6)

The quality assurance program (Section 17.5)

Inservice inspection program (Sections 5.2 and 6.6)

Inservice testing program (Section 3.9)

The technical specifications surveillance test program (COL Appendix A)

17.6.4 Maintenance Rule Program Relationship With Industry Operating Experience Activities

Industry Operating Experience (IOE) comprises information from a variety of sources that is applicable and available to the nuclear industry with the intent of minimizing, through shared experiences, adverse plant conditions or situations. Sources of IOE include information programs organized by the reactor vendor, safety-related equipment suppliers, the NRC, the Institute of Nuclear Power Operations (INPO) and the Electric Power Research Institute (EPRI).

IOE is reviewed for plant-specific applicability and, where appropriate, is applied in various elements of the MR program and procedures, including scoping, performance/condition criteria development, monitoring, goal-setting, corrective action, training, program assessment, and maintenance and procurement activities. The specific steps for employing IOE in the various MR program areas are contained in program procedures.

Condition monitoring of underground or inaccessible cables is incorporated into the maintenance rule program. The cable condition monitoring program incorporates lessons learned from industry operating experience, addresses regulatory guidance, and utilizes information from detailed design and procurement documents to determine the appropriate inspections, tests and monitoring criteria for underground and inaccessible cables within the scope of the maintenance rule (i.e., 10 CFR 50.65). The program takes into consideration Generic Letter 2007-01.

17.6.5 Maintenance Rule Program Implementation

MR Program documents will be developed and maintained, and the MR program will be implemented by the time that initial fuel loading has been authorized.

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17.7 Combined License Information Items

17.7.1 The design phase Quality Assurance program is addressed in Sections 17.1 and 17.5.

17.7.2 The Quality Assurance program for procurement, fabrication, installation, construction and testing of structures, systems and components in the facility, including provisions for seismic Category II structures, systems, and components, is addressed in Section 17.5.

17.7.3 The PRA importance measures, the expert panel process, and other deterministic methods to determine the site-specific list of SSCs under the scope of RAP are addressed in APP-GW-GLR-117 (Reference 11).

17.7.4 The Quality Assurance program for operations is addressed in Section 17.5.

17.7.5 The activities represented in Figure 17.4-1 as "Plant Maintenance Program" include the tasks necessary to maintain the reliability of risk-significant SSCs as addressed in APP-GW-GLR-117 (Reference 11). Reference 8 contains examples of cost-effective maintenance enhancements, such as condition monitoring and shifting time-directed maintenance to condition-direction maintenance.

17.7.6 The Maintenance Rule (10 CFR 50.65) activities that prescribe SSC performance-related goals during plant operation are addressed in APP-GW-GLR-117 (Reference 11).

17.7.7 The D-RAP activities are addressed in APP-GW-GLR-117 (Reference 11), and include:

- Reliability data base — Historical data available on equipment performance. The compilation and reduction of this data provides the plant with source of component reliability information.
- Surveillance and testing — In addition to maintaining the performance of the components necessary for plant operation, surveillance and testing provides a high degree of reliability for the safety-related SSCs.
- Maintenance plan — This plan describes the nature and frequency of maintenance activities to be performed on plant equipment. The plan includes the selected SSCs identified in the D-RAP.

17.7.8 The integration of the objectives of the OPRAAs into the Quality Assurance Program developed to implement 10 CFR 50, Appendix B, including failures of non-safety-related, risk-significant SSCs that result from design and operational errors in accordance with SECY-95-132, Item E, is addressed in Section 17.5.

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2. WCAP-8370, Revision 12a, “Energy Systems Business Unit - Power Generation Business Unit Quality Assurance Plan,” April 1992.

3. WCAP-8370/7800, Revision 11A/7A, “Energy Systems Business Unit - Nuclear Fuel Business Unit Quality Assurance Plan,” December 1988.

4. WCAP-12600, Revision 4, "AP600 Advanced Light Water Reactor Design Quality Assurance Program Plan," January 1998.

5. APP-GW-GL-022, AP1000 Probabilistic Risk Assessment Rev. 8, August 2004 [ML042230497].

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8. Lofgren, E. V., Cooper, et al., "A Process for Risk-Focused Maintenance," NUREG/CR-5695, March 1991.

9. Westinghouse Electric Company Quality Management System (QMS), Revision 7, dated October 1, 2013.

10. NEI 07-02, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52.”

11. APP-GW-GLR-117, “Incorporation of the Maintenance Rule,” Westinghouse Electric Company LLC.

12. SECY 95-132, “Policy and Technical Issue With the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY 94-084).”

201. Nuclear Energy Institute, Technical Report NEI 06-14A, “Quality Assurance Program Description,” Revision 7, July 2009.

202. Nuclear Energy Institute, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52,” NEI 07-02A, Revision 0, March 2008.

203. 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants.”

204. Nuclear Management and Resources Council, Inc., "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01, Rev. 4F, April 2018 [ML18120A069].

205. Regulatory Guide 1.160, Rev. 4, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

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Chapter 18 Human Factors Engineering

18.1 Overview

Human factors engineering deals with designing and implementing resources and environments that help people perform tasks more reliably. Traditionally, human factors engineering includes the consideration of:

- Anthropometric or physical fit of humans to either their task-assisting machines or to their surroundings (for example, height, reach, and visual limitations)
- Biomechanical fit of the physical capabilities and limitations of humans relative to the requirements of their tasks (for example, lifting limits and push-pull limits)
- Biophysical fit of the physiological capabilities and limitation of humans to their environment (for example, tolerance to heat or cold, harmful chemicals, and noise)

More recently, the human factors engineering discipline also models human error. Human errors include:

- Errors of execution or “slips”
- Errors of intention or “mistakes” (Reference 1)

Slips are errors in which a person’s intentions are correct, but an incorrect method for executing the action is chosen. Mistakes are errors in which the person forms an incorrect intention but then correctly executes it. Slips tend to be the result of poorly designed physical interfaces (for example, switches on a control board that look or feel alike) or of a poorly designed work environment (for example, temperatures that cause worker exhaustion). Mistakes are cognitive or mental errors. Human factors engineering includes cognitive systems engineering. This discipline focuses on the design of interfaces between humans and machines that support the operator decision-making activities that are required by the task. Cognitive systems engineering is particularly important when designing an interface for operators that control a real-time process, such as a nuclear power plant.

The rapid changes in digital computer and color graphics display technology offer the AP1000 design team an opportunity to improve the real-time decision support for the AP1000 operating staff. The AP1000 has a plant-wide network that provides pre-processed plant data to those members of the plant’s staff who have need of it. The real-time process control interface between the plant’s staff and the plant’s process equipment is the instrumentation and control (I&C) equipment driving graphical display devices in an integrated Human System Interface. Cognitive systems engineering is applied in the design of the human system interface.

The layout and environmental design of the main control room and the remote shutdown room, and the supplementary support areas, such as the technical support center, are sites of application of the traditional disciplines of human factors engineering.

Design input including decisions made in the design of the AP1000 that affect interfaces is provided. This includes input on the operating staff training program and on the development of the plant operating procedures.

Because of the rapid changes that are taking place in the digital computer and graphic display technology employed in a modern human system interface, design certification of the AP1000 focuses upon the process used to design and implement human system interfaces for the AP1000, rather than on the details of the implementation. As a result, this chapter describes the processes used to provide human factors engineering in the design of the AP1000.

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This chapter describes the application of the human factors engineering disciplines to the design of the AP1000. [*The basis for the human factors engineering program is the human factors engineering process specified in Reference 2.*]* Figure 18.1-1 illustrates the elements of the human factors engineering program. These elements correspond to the elements specified in References 2, 10, and 11. The organization of this chapter parallels these elements. In addition to the elements of the program review model, this chapter includes a description of the minimum inventory of controls, displays, and alarms present in the main control room and at the remote shutdown workstation.

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6. Not used.

7. Not used.

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11. NUREG-0711, Revision 2, “Human Factors Engineering Program Review Model,” U.S. NRC, February 2004.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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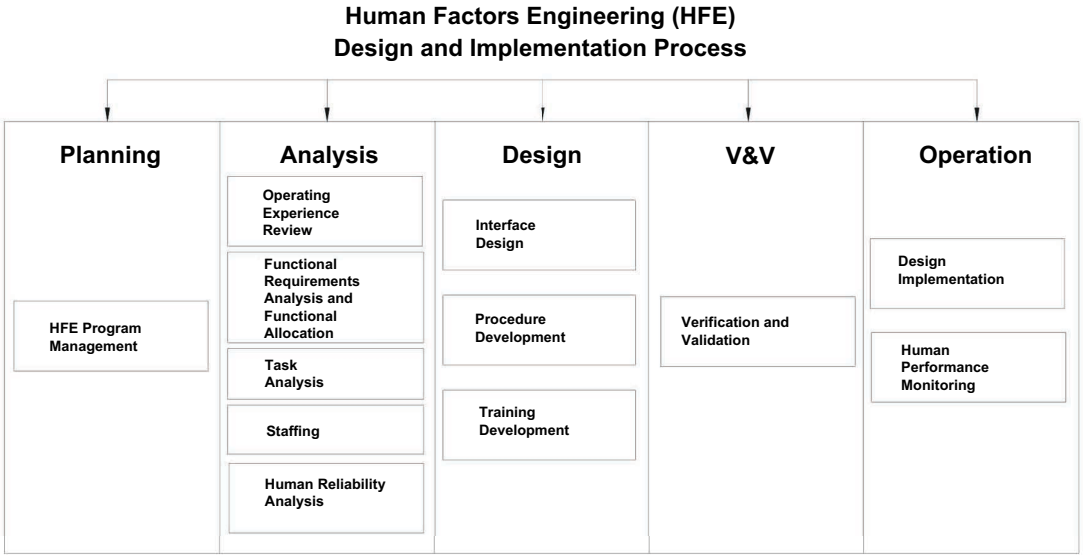

Figure 18.1-1
Human Factors Engineering (HFE)
Design and Implementation Process

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18.2 Human Factors Engineering Program Management

The purpose of this section is to describe the goals of the AP1000 human factors engineering program, the technical program to accomplish these goals, the human system interface design team, and management and organizational structure that support the implementation of the technical program.

Human factors engineering is the system engineering of human system interfaces. The program management tools and procedures that govern the design of AP1000 systems apply to the human factors engineering activity. This approach integrates the design of human system interfaces with other plant systems.

18.2.1 Human Factors Engineering Program Goals, Scope, Assumptions, and Constraints

18.2.1.1 Human Factors Engineering Program Goals

The goal of the human factors engineering program is to provide the users of the plant operation and control centers effective means for acquiring and understanding plant data and executing actions to control the plant's processes and equipment.

The objective is to enable personnel tasks to be accomplished within time and performance criteria.

18.2.1.2 Assumptions and Constraints

There are a number of inputs to the human factors engineering design process that specify assumptions or constraints on the human factors engineering program and the human system interfaces design.

Major design inputs include regulatory guidelines, guidance from utilities and utility representative groups, utility requirements documents, and AP1000 plant systems design specifications. The requirements resulting from these design inputs are captured in human system interface specification documents and functional requirements documents.

While assumptions and constraints specified by design inputs are provisionally treated as design requirements, the appropriateness of these requirements is evaluated as part of the human factors engineering design process. Results of human factors engineering activities such as operating experience review, task analyses, mockup activities and verification and validation activities are used to provide feedback on the adequacy of initial human system interface design assumptions and constraints. If results of human factors engineering analyses or evaluations indicate that initial human system interface design assumptions or constraints are inadequate, then the human system interface design requirements are revised utilizing the standard AP1000 design configuration change control process.

Listed below are some of the major inputs to the AP1000 human system interface design and the assumptions and constraints they impose on the AP1000 human system interface design process and human system interfaces design.

Regulatory Requirements

One of the requirements for the AP1000 human factors engineering program is that it complies with applicable regulatory requirements. [*The human factors engineering process is designed to meet the human factors engineering design process requirements specified in NUREG-0711 (Reference 1).*]* This is also in accordance with References 7 and 10.

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Utility Requirements

Another source of design input is utility customer requirements. Utility input can take the form of utility requirements documents, and/or input from utility representative groups serving in an advisory capacity.

Examples of utility requirements that impact the human system interface design are:

- **Operating staff assumptions.** A single reactor operator (RO) should be able to control major plant functions performed from the main control room during normal power operations.

- **Assumptions with respect to human system interface resources.** The human system interface design shall include an integrating overview display and mimic in the main control room.

The AP1000 design goals with respect to control room staffing are addressed in Section 18.6 and WCAP-14694 (Reference 3). As noted in WCAP-14694, a number of elements of the AP1000 human factors engineering design process are used to help achieve, verify and validate the control room staffing design goal. These include operating experience review, function analysis and allocation, task analysis, human reliability assessment, human system interface design, procedures, training, and human factors engineering verification and validation.

As described in Section 18.8, one of the human system interface resources is a wall panel information system. The wall panel information system is intended to meet the utility requirement for an integrating overview display and mimic in the main control room. A number of design activities establish the basis and functional requirements for the wall panel information system. Design activities include conducting operating experience reviews in nuclear power plants and related industries to examine the requirements for individual and group situation awareness and how these can best be supported.

Plant System Design Information

The design of the plant systems is an essential input to the human system interface design process. The physical implementation specifications as well as the systems designer's intent with regard to expected systems operation and performance are used as input to the design of the AP1000 human system interfaces. System design data are documented in the individual system specification documents. The input representing the plant's physical structure is represented by the piping and instrumentation drawings, general arrangement drawings, and equipment drawings.

System design specifications include specifications with respect to function allocation between automated systems and human operators. The system design specifications indicate which functions are to be automated, which are to be manual, and which require joint input of person and machine. In addition, the system design specifications indicate the set of instruments and controls that are implemented in the AP1000.

The AP600 function requirements analysis and function allocation document (Reference 4) provides information on the approach to initial function allocation and presents the results for AP600 safety functions. The results include a specification of level of automation and personnel responsibility for AP1000 safety functions, processes, and systems. The results also document the rationale for function allocation decisions for AP1000 safety functions.

The report also describes human factors activities that are conducted as part of the AP1000 human system interface design process to verify the adequacy of function allocation decisions, and establish the ability of operators to perform the role assigned to them. Function-based task analyses are used to verify that the sensors and controls that are provided are sufficient to enable operators to perform

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the role assigned to them in system performance. Workload analyses are used to evaluate the adequacy of the integrated roles assigned to operators across systems. Integrated system validation is used to establish the adequacy of the function allocation using man-in-the-loop tests in dynamic simulated plant conditions.

Technology Limits

Recent advances in the technology of digital computing have made it possible and practical to change the performance and role of the human system interface in a process control application such as a nuclear power plant. For the AP1000, a position regarding the limits of the implementation technology to be assumed for the human system interfaces is derived from assessment of existing technology and anticipated advancements. An emphasis is placed on utilization of proven, reliable technology. The decision on the specific technology to be employed is made on a case-by-case basis after available technology alternatives are evaluated.

18.2.1.3 Applicable Facilities

[*Facilities included in the scope of the AP1000 human factors engineering program are the main control room (MCR), the technical support center (TSC), the remote shutdown room, the emergency operations facility (EOF), and local control stations.*]*

The EOF is designed as discussed in Subsection 18.2.6, including specification of a location, in accordance with the AP1000 human factors engineering program. Communication with the emergency operations facility is also as discussed in Subsection 18.2.6. Section 13.3 discusses the responsibility for emergency planning.

18.2.1.4 Applicable Human System Interfaces

[*The scope of the human system interfaces encompasses the instrumentation and control systems which perform the monitoring, control, and protection functions associated with all modes of plant normal operation as well as off-normal, emergency, and accident conditions. Both the physical and the cognitive characteristics of those humans involved in the use, control, maintenance, test, inspection, and surveillance of plant systems are accommodated.*]*

18.2.1.5 Applicable Plant Personnel

[*The AP1000 human factors engineering program and the design of the human system interfaces includes the selection, synthesis, and distribution of process data to plant operations personnel as well as other plant personnel. These additional users include management, engineering, maintenance, health physics and chemistry personnel.*]*

18.2.1.6 Technical Basis

[*The human factors engineering program is performed in accordance with accepted industry standards, guidelines, and practices.*]* The references listed at the end of each Chapter 18 section and within any supporting documentation and reports are used to guide the human factors engineering program. [*The human factors engineering process specified in Reference 1 is used.*]* This is also in accordance with References 7 and 10.

18.2.2 Human System Interface Design Team and Organization

The human system interface design team is part of the AP1000 systems engineering function and has similar responsibility, authority, and accountability as the rest of the design disciplines. Figure 18.2-1 depicts the process used by the human system interface design team members.

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Figure 18.2-2 shows the organization of the human system interface design team and its relationship to the AP1000 design organization.

18.2.2.1 Responsibility

[*The mission of the human system interface design team is to develop the main control room and ancillary control facilities (such as remote shutdown workstation) that support plant personnel in the safe operation and maintenance of the plant. The human system interface design team is responsible for coordinating the human factors aspects associated with designing the structures, systems, and components that make up the main control room and ancillary control facilities.*

The human system interface design team is responsible for:

- *Development of human system interface plans and guidelines*
- *Oversight and review of human system interface design, development, test, and evaluation activities*
- *Initiation, recommendation, and provision of solutions for problems identified in the implementation of the human system interface activities*
- *Assurance that human system interface activities comply with the human system interface plans and guidelines*]*

18.2.2.2 Organizational Placement and Authority

The organization of the human system interface design team and its relation to the AP1000 design organization is depicted in Figure 18.2-2. The structure of the organization may change, but the functional nature of the human system interface design team is retained through the change. The human system interface design team consists of an instrumentation and control system manager, advisors/reviewers team, core human system interface design team, and human system interface technical lead. The technical disciplines described in Subsections 18.2.2.3 and 18.2.2.4 are organized by function within the core human system interface design team. The core human system interface design team and the advisors/reviewers team report to the instrumentation and control system manager. The human system interface technical lead works within the human system interface design function and reports to the instrumentation and control system manager through the manager of the human system interface design function. The instrumentation and control system manager is responsible for the design of the AP1000 instrumentation and control systems which include the human system interfaces. The instrumentation and control system manager reports to the AP1000 project manager.

The manager of the human system interface design function, who performs the function of technical project management for the human factors engineering design process, is responsible for the overall human system interface design and for integration of the human system interface design with the overall plant design. The advisors/reviewers team is responsible for overseeing the general progress of the human system interface design, providing guidance within the core human system interface design team, reviewing and providing comments on documents, specifications, and drawings pertaining to the human system interface design, and providing supplemental expertise in particular areas of design. The responsibility of the core human system interface design team is to produce the detailed design of the human system interfaces. The human system interface design function is responsible for the functional design of the human system interfaces, main control room and workstation layout (ergonomics), controls, the plant information system displays, the wall panel information system, the PMS safety displays, the alarm system, and computerized procedures system design and specification. The responsibilities of the human system interface technical lead

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include coordinating the technical work of the functional engineering groups, providing the administrative and technical interface between the functional engineering groups and the advisors/ reviewers team, and tracking the identification and resolution of human factors engineering design issues through operating experience review.

18.2.2.3 Composition

[*The human system interface design team consists of a multi-disciplinary technical staff. The team is under the leadership of an individual experienced in the management of the design and operation of process control facilities for complex technologies. The technical disciplines of the design team include:*

- *Technical project management*
- *Systems engineering*
- *Nuclear engineering*
- *Instrumentation and control (I&C) engineering*
- *Architect engineering*
- *Human factors engineering*
- *Plant operations*
- *Computer system engineering*
- *Plant procedure development*
- *Personnel training*
- *Systems safety engineering*
- *Maintainability/inspectability engineering*
- *Reliability/availability engineering*]*

The responsibilities of the individual technical disciplines include:

- Technical Project Management
  - Provide central point of contact for management of the human factors engineering design and implementation process
  - Develop and maintain schedule for human factors engineering design process
- Systems Engineering
  - Provide knowledge of the purpose, technical specifications, and operating characteristics of plant systems
  - Provide input to human factors engineering task analyses
  - Participate in development of procedures and scenarios for task analyses, and integrated system validation
- Nuclear Engineering
  - Provide knowledge of the processes involved in reactivity control and power generation
  - Provide input to human factors engineering task analyses
  - Participate in development of scenarios for task analyses, and integrated system validation

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- Instrumentation and Control (I&C) Engineering
  - Provide knowledge of control and display hardware design, selection, functionality, and installation
  - Provide input to software quality assurance programs
  - Participate in the design, development, test, and evaluation of the human system interfaces
- Architect Engineering
  - Provide knowledge of plant component layout and the overall structure of the plant including design characteristics and performance requirements for the containment building, control room, remote shutdown room, and local control stations
  - Provide input to human factors engineering task analyses
  - Participate in development of scenarios for task analyses, and integrated system validation
- Human Factors Engineering
  - Provide knowledge of human performance capabilities and limitations, human factors design and evaluation practices, and human factors principles, guidelines, and standards
  - Develop and perform human factors analyses and participate in resolution of human factors problems
- Plant Operations
  - Provide knowledge of operational activities relevant to characterizing tasks and environment and development of human system interface components, procedures, and training programs
  - Participate in development of scenarios for task analyses, and integrated system validation
- Computer System Engineering
  - Provide knowledge of data processing required for human system interface displays and controls
  - Participate in design and selection of computer-based equipment
  - Participate in development of scenarios for task analyses, and integrated system validation, particularly those involving failures of the human system interface data processing systems
- Plant Procedure Development
  - Provide knowledge of operational tasks and procedure formats

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  - Provide input for development of emergency operating procedures, computer-based procedures, and training systems
  - Participate in development of scenarios for task analyses, and integrated system validation

- Personnel Training
  - Develop content and format of personnel training programs
  - Participate in development of scenarios for task analyses, and integrated system validation

- Systems Safety Engineering
  - Identify safety concerns
  - Perform system safety hazard analysis such as thermal atmospheric analysis, toxicology analysis, and radiological analysis

- Maintainability/Inspectability Engineering
  - Provide knowledge of maintenance, inspection, and surveillance activities
  - Provide input in the areas of maintainability and inspectability
  - Support design, development, and evaluation of control room and other human system interface components
  - Participate in development of scenarios for task analyses, and integrated system validation

- Reliability/Availability Engineering
  - Provide knowledge of plant system and component reliability and availability and assessment methodologies
  - Provide input to design of human system interface equipment
  - Participate in development of scenarios for task analyses, and integrated system validation

18.2.2.4 Team Staffing Qualifications

In choosing the human system interface design team members, greater emphasis is placed on the individual's relevant experience to the specific discipline than on formal education. Alternative personal credentials may be selectively substituted for the education and experience requirements specified below. The professional experience of the human system interface design team as a collective whole satisfies the experience qualifications. The human system interface design team members have the following backgrounds:

- Technical Project Management
  - Bachelor's degree

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  - Five years experience in nuclear power plant design or operations and three years of management experience

- Systems Engineering
  - Bachelor of Science degree
  - Four years of cumulative experience of the following areas of systems engineering: design, development, integration, operation, and test and evaluation

- Nuclear Engineering
  - Bachelor of Science degree
  - Four years of experience in the following areas of nuclear engineering: design, development, test, or operations

- Instrumentation and Control (I&C) Engineering
  - Bachelor of Science degree
  - Four years of experience in hardware and software design aspects of process control systems; familiarity with software quality assurance and control
  - Experience in at least one of the following areas of instrumentation and control engineering: development, power plant operations, test evaluations

- Architect Engineering
  - Bachelor of Science degree
  - Four years experience in design of power plant structures and building services

- Human Factors Engineering
  - Bachelor’s degree in Human Factors Engineering, Engineering Psychology, or related science
  - Four years experience in the following areas of human factors engineering: human factors aspects of human system interfaces (design, development, and test and evaluation of human system interfaces for process control applications) and four years experience in human factors aspects of workplace design (design, development, and test and evaluation of workplaces)

- Plant Operations
  - Current or prior senior reactor operator (SRO) license/senior reactor operator instructor certification
  - Two years experience in PWR nuclear power plant operations

- Computer System Engineering

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  - Bachelor's degree in Electrical Engineering or Computer Science or graduate degree in other engineering discipline
  - Four years experience in design of computer systems and real-time system applications; familiarity with software quality assurance and control
- Plant Procedure Development
  - Bachelor's degree
  - Four years experience in developing nuclear power plant operating procedures
- Personnel Training
  - Bachelor's degree
  - Four years experience in the development of personnel training programs for power plants and experience in the application of systematic training development methods
- Systems Safety Engineering
  - Bachelor of Science degree or Bachelor's degree in Science
  - Experience in system safety engineering, such as thermal atmospheric analysis, toxicology, radiological analysis and applicable OSHA limits
- Maintainability/Inspectability Engineering
  - Bachelor of Science degree or Bachelor's degree in Science
  - Four years of cumulative experience in at least two of the following areas of power plant maintainability and inspectability engineering activity: design, development, integration, test and evaluation, and analysis/resolution of maintenance problems.
- Reliability/Availability Engineering
  - Bachelor's degree
  - Four years of cumulative experience in at least two of the following areas of power plant reliability engineering activity: design, development, integration, and test and evaluation. Knowledge of computer-based, human system interfaces.

18.2.3 Human Factors Engineering Processes and Procedures

Activities performed relating to human factors engineering are performed in accordance with documented procedures under the quality assurance program for the AP1000. These procedures provide for control of processes as described below.

18.2.3.1 General Process and Procedures

The instrumentation and control system function is responsible for development of the AP1000 instrumentation and control (I&C), including human system interfaces, and coordinating and integrating AP1000 instrumentation and control and human system interfaces with other AP1000 plant design activities. The overall operation of the project instrumentation and control systems

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function is defined. The function includes human system interface design of control rooms and control boards, instrumentation and control design, and control room/equipment design. The function includes definition of an engineering plan, review of inputs, production of system documentation, verification of work, procurement and manufacturing follow-up, and acceptance testing. An iterative feature is built into the process.

Documents produced as part of the instrumentation and control and human system interface design process include:

- Operating experience review documents
- Task analysis documents
- Functional requirements documents
- Human system interface design guidelines documents
- Design specification documents
- Instrumentation and control architecture diagrams
- Block diagrams
- Room layout diagrams
- Instrumentation lists
- System specification documents

The procedures governing instrumentation and control engineering work specify methods for verification of work. The types of verification include:

- Design verification by design reviews
- Design verification by independent review/alternative calculations
- Design verification by testing

System Specification Documents

System specification documents identify specific system design requirements and show how the design satisfies the requirements. They provide a vehicle for documenting the design and they address information interfaces among the various design groups.

System specification documents follow established format and content requirements. The content of a system specification document includes:

- Purpose of the system
- Functional requirements and design criteria for the system
- System design description including system arrangement and performance parameters
- Layout
- Instrumentation and control requirements
- Interfacing system requirements

The section on interfacing system requirements describes the support needed from and provided to other systems.

System specification documents document human factors and human system interaction requirements. This includes specification of task requirements, information requirements, and equipment requirements for operations, surveillance, test, and maintenance activities.

System specification documents provide specification of instrument and control requirements including:

- System input to the I&C channel list
- Reference to control logic diagrams

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- Alarm requirements and characteristics
- Requirements and characteristics of plant status indications

A system specification document for the operation and control centers system provides a mechanism for documenting and tracking human system interface requirements and design specifications. The operation and control centers system specification document is the umbrella document for capturing generic human factors requirements. It provides a uniform operational philosophy and a design consistency among human system interface resources, including alarm system, plant information system, wall panel information system, and computerized procedures.

Functional requirements and design specifications for the AP1000 operation and control centers system, including the main control room, the technical support center, the remote shutdown room, and local control stations are provided in the operation and control centers system specification document. Functional requirements documents and design specification documents are generated for each of the individual human system interface resources (including alarm system, plant information system, wall panel information system, computerized procedures, controls). Functional requirements documents specify the applicable codes, standards, and design requirements and constraints to be met by the design. These documents are referenced by the operation and control centers system specification document.

Design specification documents provide the design specifications for individual human system interface resources and their integration. Included in these specifications are layout and arrangement drawings, algorithms, and display system descriptions, including display task descriptions, display layouts, and navigation mechanisms.

The operation and control centers system specification document, the functional requirement documents, and design specification documents provide input to the generation of I & C system specification documents such as the system specification document for the data display and processing system.

Design Configuration Change Control Process

Design changes are controlled to assure that proposed changes to design documents under configuration control are appropriately evaluated for impacts and that approved changes are communicated to the responsible design organizations.

The design configuration change control process is used to control and implement changes to the design. It is used when the design to be changed has been previously released in a document for project use and placed under configuration control. A design change proposal is the vehicle used to initiate and document review of proposed design changes. Design change proposals include identification of impacts of the proposed design change from affected functional groups. In some instances, human factors engineering issues are addressed by the initiation of design change proposals. In other instances, they are addressed as a consequence of human factor engineering review of design change proposals originating from other disciplines. Design change proposals are maintained in a database that is used to track the status of each design change proposal from initiation through implementation and closure.

Design Review of Human Factors Engineering Products

Design reviews by a multi-disciplined review team are established as a verification method. Requirements for the design review process, including selection of the review team, preparation of information for review, identification and follow of action items, and documentation of the proceedings, are defined.

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Design reviews provide a method of design verification consisting of a systematic overall evaluation of a design that is conducted by an independent design review team. Design reviews are conducted at appropriate stages of design development to provide an objective, independent review of design adequacy, safety, performance, and cost. Design reviews are performed by persons not directly associated with the specific design development, but who, as a group, are knowledgeable in the appropriate technical disciplines.

Original designs, as well as major design changes, are subject to the design review process. For each design review, a design review data package is prepared. It includes checklists, including one specifically addressing human factor engineering questions, which are used by design review committee members to aid their review. For each design issue identified through the use of checklists or otherwise, an action item is initiated.

[*Action items are tracked through the design issues tracking system database as described in subsection 18.2.4. The responsibility of entering design review action items into the design issues tracking system database is assigned to the manager responsible for the system reviewed. The responsible design manager is responsible for tracking and addressing open action items.*]*

18.2.3.2 Process Management Tools

Tools are provided to facilitate communication across design disciplines and organizations to enhance consistency. An AP1000 design database enables parties involved in the engineering design of the plant to access up-to-date plant design data. Procedures define requirements and responsibilities for moving data into the database.

Tools are provided to guide the design review process. These include design review checklists that support evaluation of design adequacy, and a database for tracking action items generated as a result of the design review process. Further details on the process of tracking action items generated by design reviews are provided in Subsections 18.2.3.1 and 18.2.4.

A design configuration change control process is used to control and implement proposed design changes. Design change proposals are maintained in a database that is used to track the status of each design change proposal from initiation through implementation and closure.

A design issues tracking system database is used to document and track design issues that are identified during the plant design process. Further details on the design issues tracking system are provided in Subsection 18.2.4.

18.2.3.3 Integration of Human Factors Engineering and Other Plant Design Activities

The AP1000 design process provides for the integration of human factors engineering activities among the design groups.

The instrumentation and control systems design function is responsible for the development of the AP1000 instrumentation and control systems, including the human system interface. Coordination and integration of the instrumentation and control and human system interface design with other plant design activities is performed by the instrumentation and control systems design function. An iterative design process that includes review and feedback from other engineering and design groups at the design interface is specified. Subsection 18.2.3.1 describes the responsibilities and design process of the instrumentation and control system design function.

System specification documents provide the primary vehicle for transmitting system design data and interface requirements, including human factors engineering and human system interface requirements, to the affected AP1000 design and analysis groups. The system specification

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documents include a section on interfacing system requirements that describes the support needed from and provided to other systems in the plant. Interface control is performed at the design interfaces and design changes affecting the interfaces are coordinated. Subsection 18.2.3.1 provides details on system specification documents.

A design configuration change control process provides the process and actions to implement design changes. Subsection 18.2.3.1 provides further details on this process.

Engineering design databases serve as a repository of AP1000 design data for parties involved in engineering design activities of the plant. A technical document control system is used to track the status of AP1000 documents. By using the engineering design databases and the technical document control system, parties have access to up-to-date design data to perform their respective design activities.

Section 18.8 presents the implementation plan for the design of the human system interface. Figure 18.2-3 provides an overview of the AP1000 human factors engineering process, including the design stages of the human system interface. The relationship of other human factors engineering process elements to the human system interface design is shown.

18.2.3.4 Human Factors Engineering Documentation

Procedures address documentation for AP1000, including document preparation, review, retention, access, and configuration control. These procedures apply to all AP1000 activities, including human factors engineering.

Documents refer to any self contained portrayal of the AP1000 design or its basis. These include design criteria, descriptions, specifications, drawings, analysis reports, safety reports and calculations.

A procedure establishes requirements and responsibilities for the preparation, review, and approval of AP1000 design documents. The procedure specifies that documents are to be reviewed by appropriate reviewers, and comments are to be resolved prior to issuance of the document. Appropriate reviewers include responsible engineers or managers impacted by the information in the document.

Changes to released documents are reviewed and approved in accordance with the design configuration change control procedure for the AP1000 program.

Procedures establish content and format requirements for system specification documents. Other procedures addressing documentation requirements include those for design configuration change control, design reviews, design criteria, and control of subcontractor submittals.

Sections 18.3 through 18.12 provide information on the types of documents that are generated as part of the AP1000 human factors engineering program.

18.2.3.5 Human Factors Engineering in Subcontractor Efforts

Human factors engineering and human system interface requirements are passed on to subcontractors through engineering documents including design criteria and system specification documents.

Activities within subcontractor design organizations are performed in accordance with the written procedures of those organizations. [*Effective implementation of each organization's quality assurance program is monitored by their respective internal audit programs, and by supplier audits.*]*

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See Section 17.3 for quality assurance requirements associated with subcontractor human factors engineering design efforts.

18.2.4 Human Factors Engineering Issues Tracking

A tracking system is used to address human factors issues that are known to the industry and/or identified throughout the life cycle of the human factors engineering/human system interface design, development, and evaluation. The tracking system enables the documentation and tracking of issues that need to be addressed at some later date.

Tracking of human factors engineering issues is accomplished within the framework of the overall plant design process. In this manner, human factors engineering issues are addressed in the same way as those for other disciplines.

[*The design issues tracking system database is used to track AP1000 design issues to resolution, including human factors engineering issues. This database receives input from the following three sources:*

- *Operating experience review*
- *Design reviews*
- *Design issues associated with the design of the human system interface and the operation and control centers system*]*

For each design issue entered into the database, the actions taken to address the issue and the final resolution of the issue are documented.

The human factors issues in the operating experience review report (Reference 2) that are identified as requiring further consideration by the AP1000 design are entered into the design issues tracking system database.

[*The design review process also provides input to the design issues tracking system database. For each design issue identified through the design review process, an action item is initiated. Action items are entered into the design issues tracking system database. Human factors action items from design reviews are included in the database. For preliminary and intermediate design reviews, some action items may be deferred to a more appropriate, subsequent design review. The responsibility of entering design review action items into the design issues tracking system database is assigned to the manager responsible for the system reviewed.*]*

Human factors engineering design issues directly associated with the AP1000 human system interfaces and the operation and control centers system (such as the main control room, remote shutdown room, and technical support center) are entered into the design issues tracking system database. These are design issues that are identified by the human system interface and operation and control centers system designers as issues that need to be addressed by the human system interface design.

The AP1000 project manager, as shown on Figure 18.2-2, is responsible for the maintenance and documentation of the design issues tracking system. For each issue entered into the design issues tracking system database, a “responsible engineer” field is used to assign an engineer the responsibility for resolution of the issue.

18.2.5 Human Factors Engineering Technical Program and Milestones

[*The human factors engineering program is performed in accordance with the human factors engineering process specified in NUREG-0711 (Reference 1).*]* Figure 18.1-1 shows the elements of

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the AP1000 human factors engineering program. [*These elements conform to the elements of the Program Review Model specified in Reference 1, as augmented by Reference 7.*]* This is also in accordance with Reference 10.

Human factors engineering Program Management is addressed in Section 18.2. The remaining elements are addressed in Sections 18.3 through 18.11, 18.13, and 18.14.

These sections address the activities conducted as part of the corresponding human factors engineering element, including the accepted industry standards, guidelines, and practices used as technical guidance, the inputs to the element, and the products, including documents that are generated as output. The facilities, equipment, and tools employed are also addressed in the section corresponding to each element.

Figure 18.2-3 provides an overview of the Westinghouse human factors engineering process. The figure summarizes the major activities of the human factors engineering program, their relative order, and the inputs and outputs for the major activities. The boxes in the diagram indicate major human factors engineering activities. The activities are presented in approximate chronological order, with the outputs of each activity serving as inputs to subsequent activities. The items listed below the activity boxes are the document outputs from that human factors engineering activity. The human factors engineering process includes iterations considering the outcomes of subsequent analysis and design activities, design reviews, and testing. In this approach, design issues are addressed and resolved through the iterative stages of the human factors engineering process. Potential points of iteration are indicated in Figure 18.2-3. Further details on the activities, inputs, and output documents associated with the various elements of the human factors engineering program are provided in the sections corresponding to each human factors engineering element.

Figure 18.2-3 provides a program milestone schedule of human factors engineering tasks showing relationships between human factors engineering elements and activities, products, and reviews. Internal design reviews are performed at various points throughout the design process.

18.2.6 Combined License Information

18.2.6.1 Human Factors Engineering Program

The execution of the NRC approved human factors engineering program as presented by Section 18.2 is addressed in APP-OCS-GBH-001 (Reference 8), and the applicable changes are incorporated.

The AP1000 Human Factors Engineering Program Plan (Reference 8) fully captures the information certified in Section 18.2. Reference 8 provides execution guidance for the NRC-approved HFE program.

18.2.6.2 Emergency Operations Facility

The design of the emergency operations facility in accordance with the AP1000 human factors engineering program is addressed in Reference 9 (APP-GW-GLR-136).

Reference 9 captures the method by which the AP1000 Human Factors Engineering Program Plan (Reference 8) will be applied to TSCs and EOFs that support an AP1000 plant.

The following activities are to be addressed by the Combined License applicant:

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Specific information regarding EOF and TSC communications, and EOF and TSC human factors attributes will be provided by the Combined Operating License applicant to address the Combined License information requested in this subsection.

18.2.7 References

[*1. NUREG-0711, "Human Factors Engineering Program Review Model," U.S. NRC, July 1994.*]*

2. WCAP-14645, "Human Factors Engineering Operating Experience Review Report For The AP1000 Nuclear Power Plant," Revision 3, May 2006.

3. WCAP-14694, "Designers Input to Determination of the AP600 Main Control Room Staffing Level," Revision 0, July 1996.

4. WCAP-14644, "AP600/AP1000 Functional Requirements Analysis and Allocation," Revision 1, October 2008.

5. Not used.

6. Not used.

[*7. NUREG-0711, Rev. 1, "Human Factors Engineering Program Review Model," U.S. NRC, May 2002.*]*

8. APP-OCS-GBH-001, "AP1000 Human Factors Engineering Program Plan," Westinghouse Electric Company LLC.

9. APP-GW-GLR-136, "AP1000 Human Factors Program Implementation for the Emergency Operations Facility and Technical Support Center," Westinghouse Electric Company LLC.

10. NUREG-0711, Revision 2, "Human Factors Engineering Program Review Model," U.S. NRC, February 2004.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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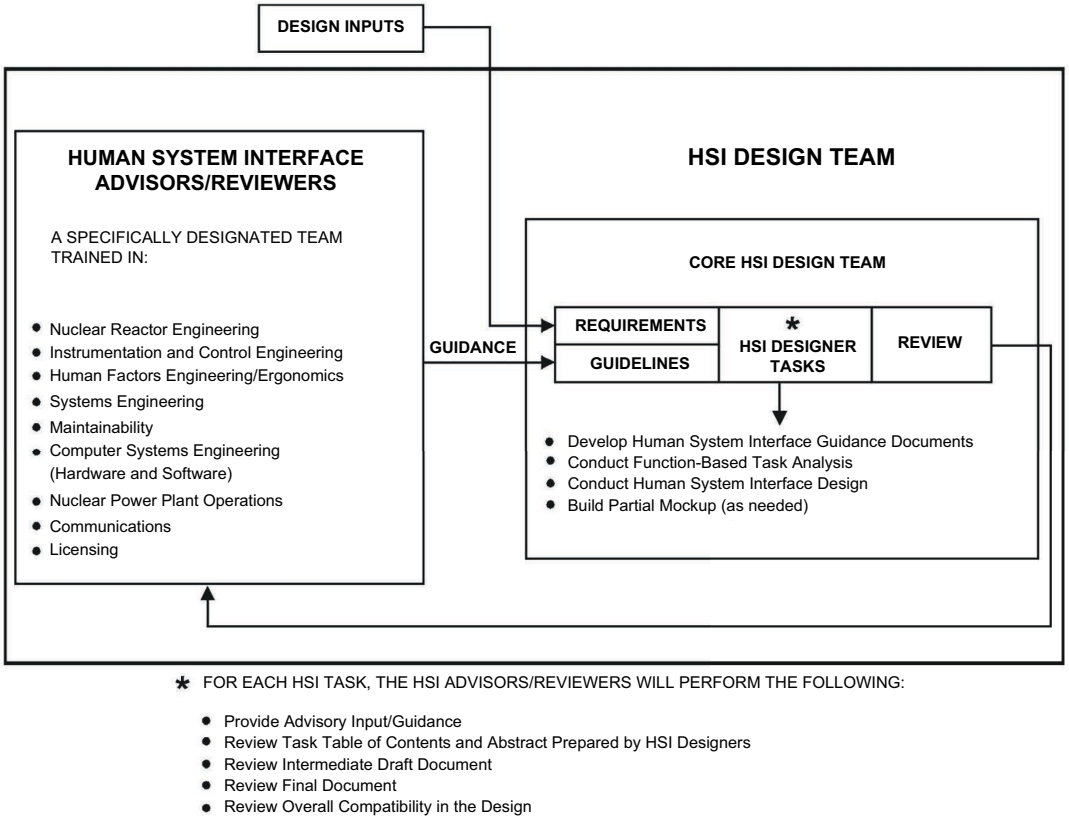

Figure 18.2-1
[*Human System Interface (HSI) Design Team Process*]*

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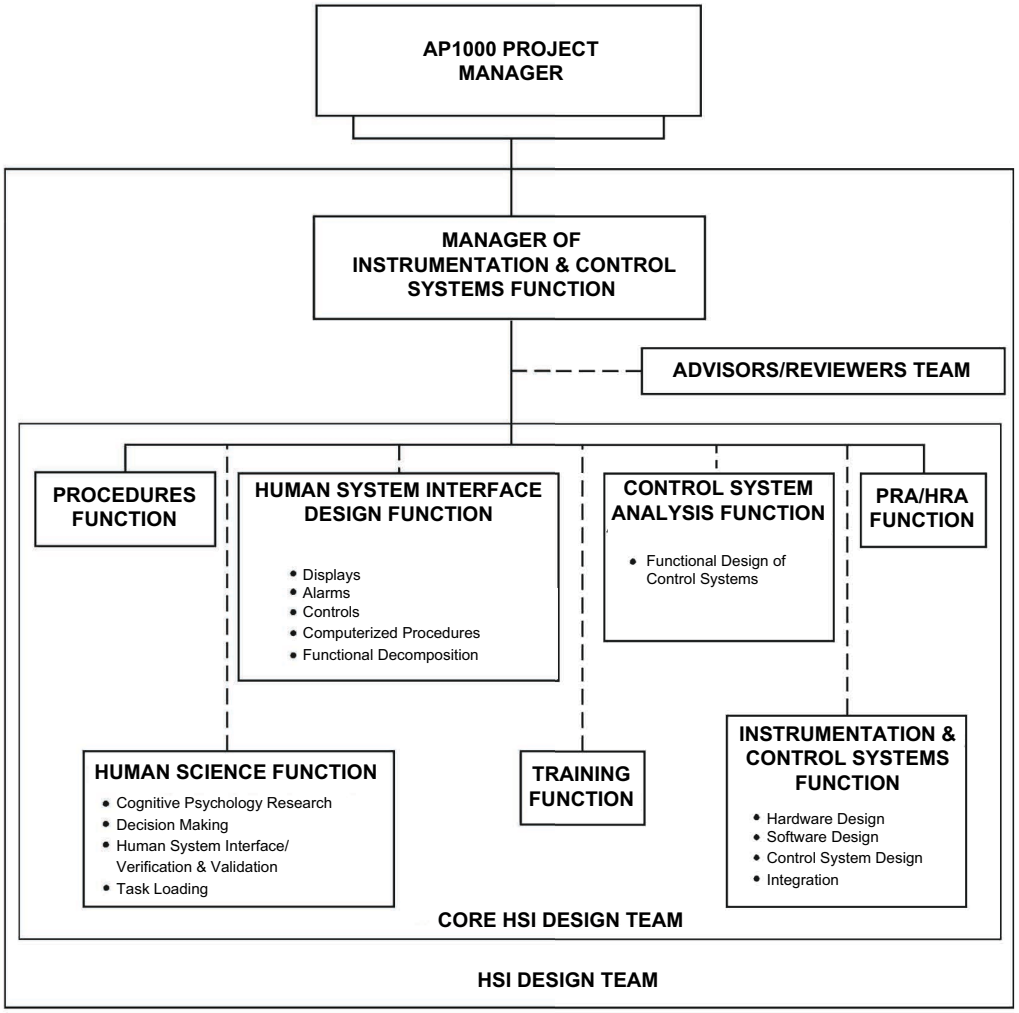

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Figure 18.2-2
Human System Interface (HSI) Design Team
Organization and Relationship to AP1000 Organization

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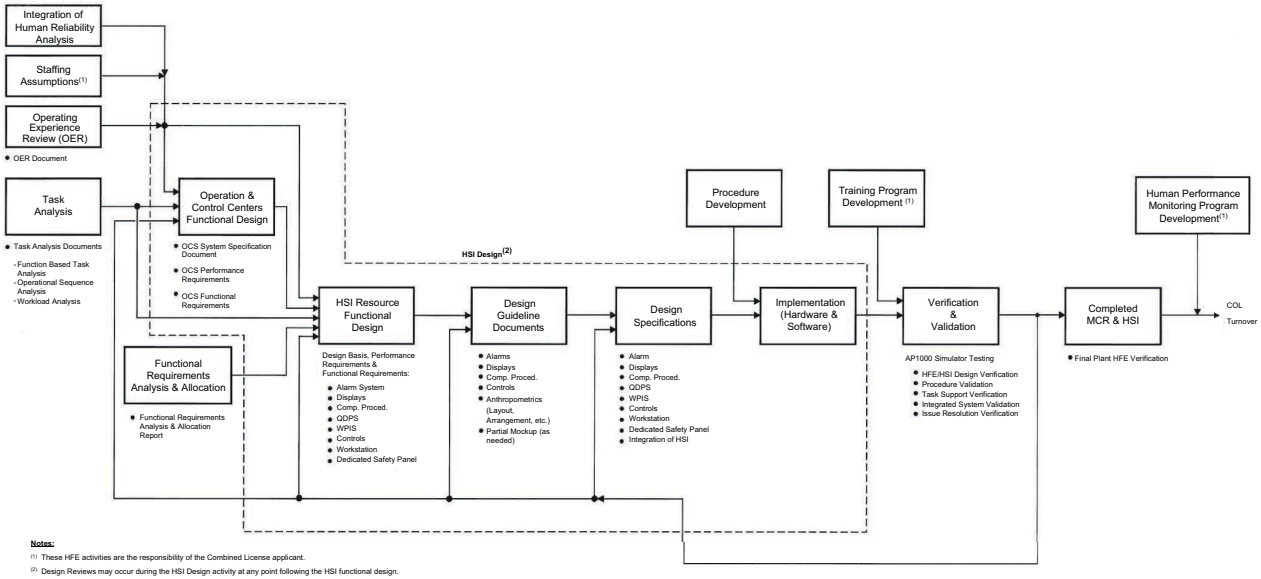

Figure 18.2-3
Overview of the AP1000 Human Factors Engineering Process

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18.3 Operating Experience Review

The objective of the operating experience review is to identify and analyze human factors engineering-related problems and issues encountered in previous designs that are similar to the AP1000. Reference 1 documents the results of this review, including descriptions of how the AP1000 design addresses each identified issue.

18.3.1 Combined License Information

Combined License applicant responsibilities identified in Reference 1 are presented in Sections 10.4.12, 16.2, 18.2.6, 18.6.1, and 18.10.1.

18.3.2 References

1. WCAP-14645, “Human Factors Engineering Operating Experience Review Report for the AP1000 Nuclear Power Plant,” Revision 3, May 2006.

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18.4 Functional Requirements Analysis and Allocation

Functional requirements and function allocation analyses are performed to establish and document design decisions with respect to level of plant automation.

Functional requirements analysis is defined as the "identification of those functions that must be performed to satisfy plant safety objectives, that is, to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public" (Reference 1).

Function allocation is defined as the "analysis of the requirements for plant control and the assignment of control functions to (1) personnel (e.g., manual control), (2) system elements (e.g., automatic control and passive, self-controlling phenomena), and (3) combinations of personnel and system elements (e.g., shared control and automatic systems with manual backup)" (Reference 1).

Reference 2 documents the methods and results of the functional requirements analysis and function allocation conducted for AP600.

The report provides a description of the AP600 approach to functional requirements analysis and presents the results for AP600 safety functions. The results include a description of AP600 processes, systems, and components involved in maintaining AP600 safety functions. The report also includes a similar analysis for current Westinghouse pressurized water reactor designs to serve as a reference in identifying areas where the AP600 plant differs from previous designs for which operating experience exists. An explicit comparison of the AP600 design with the reference plant design is provided that identifies plant functions, processes, and systems that are new or modified relative to the reference plant design. This includes changes in level of automation.

The report also describes the AP600 approach to initial function allocation and presents the results for AP600 safety functions. A methodology adapted from Reference 3 is used to document the rationale for initial allocation decisions and verify the acceptability of the initial allocation from a human factors perspective. The results include a specification of level of automation and personnel responsibility for AP600 safety functions, processes, and systems. The rationale for the function allocation decisions for AP600 safety functions is documented.

Since AP1000 is like AP600 in its operation and approach to safety functions, Reference 2 is directly applicable to AP1000. It is used as is for functional requirements and function allocation analyses for AP1000.

The report includes a description of human factors activities that are conducted as part of the AP600 HSI design process to verify the adequacy of function allocation decisions and establish the capability of operators to perform the role assigned to them. This is applied to AP1000 and includes:

- How human factors input is provided early in the design process
- How the integrated role of the operator across the systems is confirmed for acceptability
- Mechanisms available for reconsidering, and if necessary, changing AP1000 function allocations in response to operating experience, and the outcomes of ongoing analyses and trade studies

18.4.1 Combined License Information

This section contained no requirement for additional information.

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18.4.2 References

1. NUREG-0711, “Human Factors Engineering Program Review Model,” U.S. NRC, July 1994.

2. WCAP-14644, “AP600/AP1000 Functional Requirements Analysis and Function Allocation,” Revision 1, October 2008.

3. NUREG/CR-3331, “A Methodology for Allocation of Nuclear Power Plant Control Functions to Human and Automated Control,” 1983.

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18.5 AP1000 Task Analysis Implementation Plan

[*Task analysis, according to the Human Factors Engineering Program Review Model (Reference 1), has the following objectives:*

- *Provide one of the bases for the human system interface design decisions*
- *Match human performance requirements with human capabilities*
- *Provide input to procedure development*
- *Provide input to staffing, training, and communications requirements of the plant*]*

This section describes the scope of the AP1000 task analysis activities and the task analysis implementation plan. In addition to References 1 and 16, References 2 through 13 are inputs to this plan.

18.5.1 Task Analysis Scope

[*The scope of the AP1000 task analysis is divided into two complementary activities: function-based task analysis (FBTA) and traditional task analysis, or operational sequence analysis (OSA). The scope of the function-based task analysis is the Level 4 functions*]* identified in Figure 18.5-1. This figure is the functional decomposition (goal-means analysis) for normal power operations in a standard pressurized water reactor. Examples of functions at Level 4 are "Control RCS Coolant Pressure" and "Control Containment Pressure." This set of functions defines the breadth of functions to be analyzed. The function-based task analysis will be expanded in scope to include any additional Level 4 functions identified.

[*The traditional task analysis, or operational sequence analysis, is developed for a representative set of operational and maintenance tasks. The following guidelines are applied to select tasks:*

- *Tasks are selected to represent the full range of operating modes, including startup, normal operations, abnormal and emergency operations, transient conditions, and low-power and shutdown conditions.*

- *Tasks are selected that involve operator actions that are identified as either critical human actions or risk-important tasks, based on the criteria in Reference 13.*

- *Tasks are selected to represent the full range of activities in the AP1000 emergency response guidelines.*

- *Tasks are selected that involve maintenance, test, inspection, and surveillance (MTIS) actions. A representative set of maintenance, test, inspection, and surveillance tasks are analyzed for a subset of the "risk-significant" systems/structures/components (SSCs).*

The set of tasks to be analyzed are not identified as a part of design certification. The OSAs listed below are included in the set of tasks to be analyzed: (Each of these satisfies one or more of the selection criteria described above.)

  - *Plant heatup and startup from post-refueling to 100% power*
  - *Reactor trip, turbine trip, and safety injection*
  - *Natural circulation cooldown (startup feedwater with steam generator)*
  - *Loss of reactor or secondary coolant*
  - *Post loss-of-coolant accident cooldown and depressurization*
  - *Loss of RCS inventory during shutdown*
  - *Loss of the normal residual heat removal system (RNS) during shutdown*
  - *Manual automatic depressurization system (ADS) actuation*

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– *Manual reactor trip via PMS, via diverse actuation system (DAS)*
– *ADS valve testing during Mode 5*

The human factors engineering program review model (Reference 1) indicates that task analysis should include tasks that are considered to be high-risk and tasks that require critical human actions. Reference 13 defines criteria for critical human actions and risk-important tasks and has identified a list of examples of AP600 tasks that meet these criteria. Reference 13 is applicable to AP1000.]*

Section 16.2 identifies the systems/structures/components included in the Reliability Assurance Program. A subset of these systems/structures/components and a representative set of associated maintenance, tests, inspection and surveillance tasks will be selected by an expert panel. This panel will be comprised of representatives with expertise from relevant groups in the design process, such as systems engineering, reliability engineering, probabilistic risk analysis, human factors engineering, and human system interface design. The set of maintenance, test, inspection and surveillance tasks identified through the expert panel process will be considered to be "risk important" tasks, and will be included in task analysis activities.

18.5.2 Task Analysis Implementation Plan

Figure 18.5-2 shows the proposed sequence of task analyses. Figure 18.5-2 provides information concerning the task analysis and human system interface design elements. [*Task analysis includes both a function-based task analysis and an operational sequence analysis.*]* In Figure 18.5-2, the operational sequence analysis in the task analysis box is designated as OSA-1 since two operational sequence analyses will be implemented.

18.5.2.1 Function-Based Task Analyses

Function-based task analysis is applied to each of the Level 4 functions. There are four components to a function-based task analysis. First, analysis is performed to identify the set of goals relevant to the function. Second, a functional decomposition is performed. This decomposition identifies the processes that, either individually or in combination, have a significant effect on the function. Third, a process analysis is performed by applying a set of questions derived from Rasmussen's model (References 6–9) analysis approach. [*The set of questions used and basis for the methodology is provided in Reference 12.*]* An example of a question from the process analysis is "Are the process data valid?" The results of the process analysis identify the indications, parameters, and controls that the operator uses to make decisions about the respective function. Finally, there is a verification that the indications and controls, identified in the process analysis are included in the AP1000 design.

From the function-based task analyses, the following types of information are obtained:

- A completeness check on the availability of needed indications, parameters, and controls. This includes indications and controls needed for supervisory control of automated systems and manual over-ride.

- Input to the specification and layout of functional displays.

18.5.2.2 OSA-1

The operational sequence analysis completed as part of the task analysis process focuses on specifying the operational requirements for the complete set of tasks selected. For each task, an analysis of the task is conducted that includes the following:

- Plant state data required at each step
- Source of the data (alarm, display, oral communication)

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- Action to be taken or decision to be made from the data
- Relevant criterion or reference values
- Information that provides feedback on the action's adequacy
- Other temporal constraints (ordering, tasks that need to be done in parallel)
- Task support requirements needed (required tools)
- Considerations of work environment

The task analyses are developed from the emergency response guidelines and the Probabilistic Risk Assessment event sequences associated with critical or risk-important actions. The following potential limitations on task performance are considered:

- Limits on human performance
- Limits on crew communications

This first operational sequence analysis provides the following types of information:

- Frequency and co-occurrence of plant state parameters and controls
- Display design and organization constraints
- Inventory of alarms, controls, and parameters needed to perform the sequences

As shown in Figure 18.5-2, the function-based task analysis and OSA-1 feed into the human system interface design by providing task performance guidance and constraints. The display and operator workstation design is based on this information.

18.5.2.3 OSA-2

The critical issues for the second operational sequence analysis are:

- Task requirements - This analysis identifies the requirements necessary for the operator to perform the task activities.

- Time to perform tasks - A set of performance time assumptions will be established and used to determine the time required for actions to be completed. These assumptions will provide estimates of task performance times that can be compared to performance time requirements.

- Operator workload analysis - An evaluation of the effect of the human system interface design and the task demands on operator workload will be conducted.

- Operational crew staffing - The workload analysis provides an indication of the adequacy of staffing assumptions. In cases where the operational sequence analysis indicates high operator workload values, or insufficient time available for performance, alternative staffing assumptions or changes to the human system interface design or task allocation to reduce operator workload is evaluated.

This second operational sequence analysis is performed for a representative subset of tasks that include the critical human actions and risk-important tasks and tasks that have human performance concerns (for example, potential for high workload or high error rates).

18.5.2.4 Task Analysis of Maintenance, Test, Inspection and Surveillance Tasks

The maintenance, test, inspection, and surveillance tasks that are identified to be "risk-important" are analyzed using operational sequence task analyses. OSA-1 analyses are conducted on the set of maintenance, test, inspection, and surveillance tasks identified to be "risk-important."

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18.5.2.5 Technical Support Center and Emergency Operations Facility

OSA-1 analysis is conducted for the technical support center (TSC) and emergency operations facility (EOF) for the tasks where the data and displays available in the main control room may be utilized to support the TSC and/or EOF functions.

18.5.3 Job Design Factors

Section 18.6 addresses the control room staffing that applies to the AP1000. The staffing level of the main control room, job design considerations, and crew skills are discussed in Subsection 18.6.1.

18.5.4 Combined License Information Item

18.5.4.1 Task Analysis Implementation

The execution and documentation of the task analysis implementation plan presented in Section 18.5 are addressed in APP-GW-GLR-081 (Reference 14), and the applicable changes are incorporated.

18.5.4.2 Main Control Room Position Scope and Responsibilities

The scope and responsibilities of each main control room position, considering the assumptions and results of the task analysis are addressed in APP-OCS-GJR-003 (Reference 15), and the applicable changes are incorporated.

18.5.5 References

[*1. NUREG-0711, "Human Factors Engineering Program Review Model," U.S. NRC, July 1994.*]*

2. U.S. NRC Guidance, NUREG/CR-3371, "Task Analysis of Nuclear Power Plant Control Room Crews."

3. IEC-964, "Design for Control Rooms of Nuclear Power Plants."

4. Department of Defense Documents: DI-H-7055, "Critical Task Analysis Report," and MIL-STD-1478, "Task Performance Analysis."

5. NATO Document, "Applications of Human Performance Models to System Design," edited by McMillan, Beevis, Salas, Strub, Sutton, & van Breda, New York: Plenum Press, 1989.

6. Rasmussen, J., "Information Processing and Human-Machine Interaction, An Approach to Cognitive Engineering," New York: North-Holland, 1986.

7. Hollnagel, E. and Woods, D. D., "Cognitive Systems Engineering: New Wine in New Bottles," International Journal of Man-Machine Studies, Volume 18, 1983, pages 583-600.

8. Roth, E. and Mumaw, R., "Using Cognitive Task Analysis to Define Human Interface Requirements for First-of-a-Kind Systems," Proceedings of the Human Factors and Ergonomics Society 39th Annual Meeting, San Diego, Ca., 1995, pp. 520-524.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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9. Vicente, K. J., "Task Analysis, Cognitive Task Analysis, Cognitive Work Analysis: What=s the Difference?" Proceedings of the Human Factors and Ergonomics Society 39th Annual Meeting, San Diego, Ca., 1995, pp. 534-537.

10. Drury, C. G., Paramour, B., Van Cott, H. P., Grey, S. N., and Corlett, E. N., "Task Analysis," Handbook of Human Factors, Salvendy, G. (ed.), New York: John Wiley & Sons, 1987.

11. Woods, D. D., "Application of Safety Parameter Display Evaluation Project to Design of Westinghouse SPDS," Appendix E to "Emergency Response Facilities Design and V & V Process," WCAP-10170, submitted to the U.S. Nuclear Regulatory Commission in support of their review of the Westinghouse Generic Safety Parameter Display System (Non-Proprietary) (Pittsburgh, PA, Westinghouse Electric Corp.), April 1982.

[*12. WCAP-14695, "Description of the Westinghouse Operator Decision Making Model and Function Based Task Analysis Methodology," Revision 0, July 1996.*]*

[*13. WCAP-14651, "Integration of Human Reliability Analysis and Human Factors Engineering Design Implementation Plan," Revision 2, May 1997.*]*

14. APP-GW-GLR-081, "Closure of COL Information Item 18.5-1, Task Analysis," Westinghouse Electric Company LLC.

15. APP-OCS-GJR-003, "AP1000 Main Control Room Staff Roles and Responsibilities," Westinghouse Electric Company LLC.

16. NUREG-0711, “Human Factors Engineering Program Review Model,” Revision 2, U.S. NRC, February 2004.

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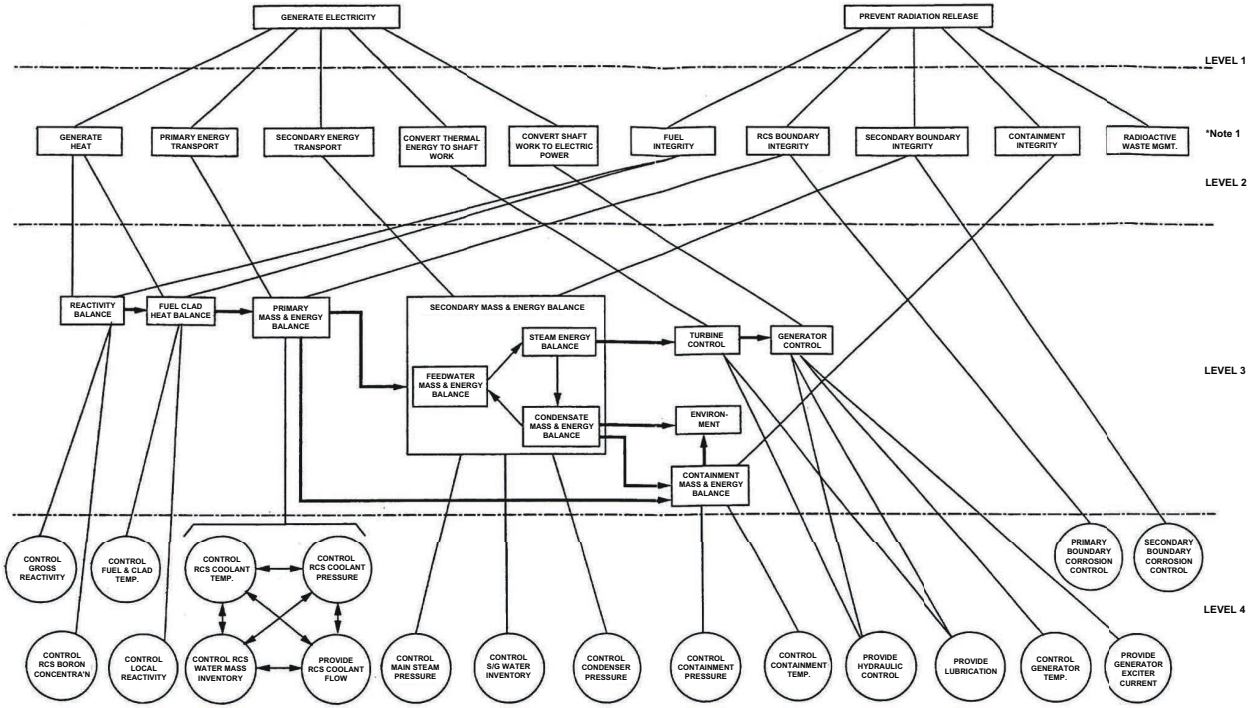

*Note 1: Decomposition and subsequent task analysis of this activity is performed as part of a similar process applied to the radioactive waste control center.

Figure 18.5-1
Top Four Levels of the Normal
Power Operation for a Westinghouse PWR

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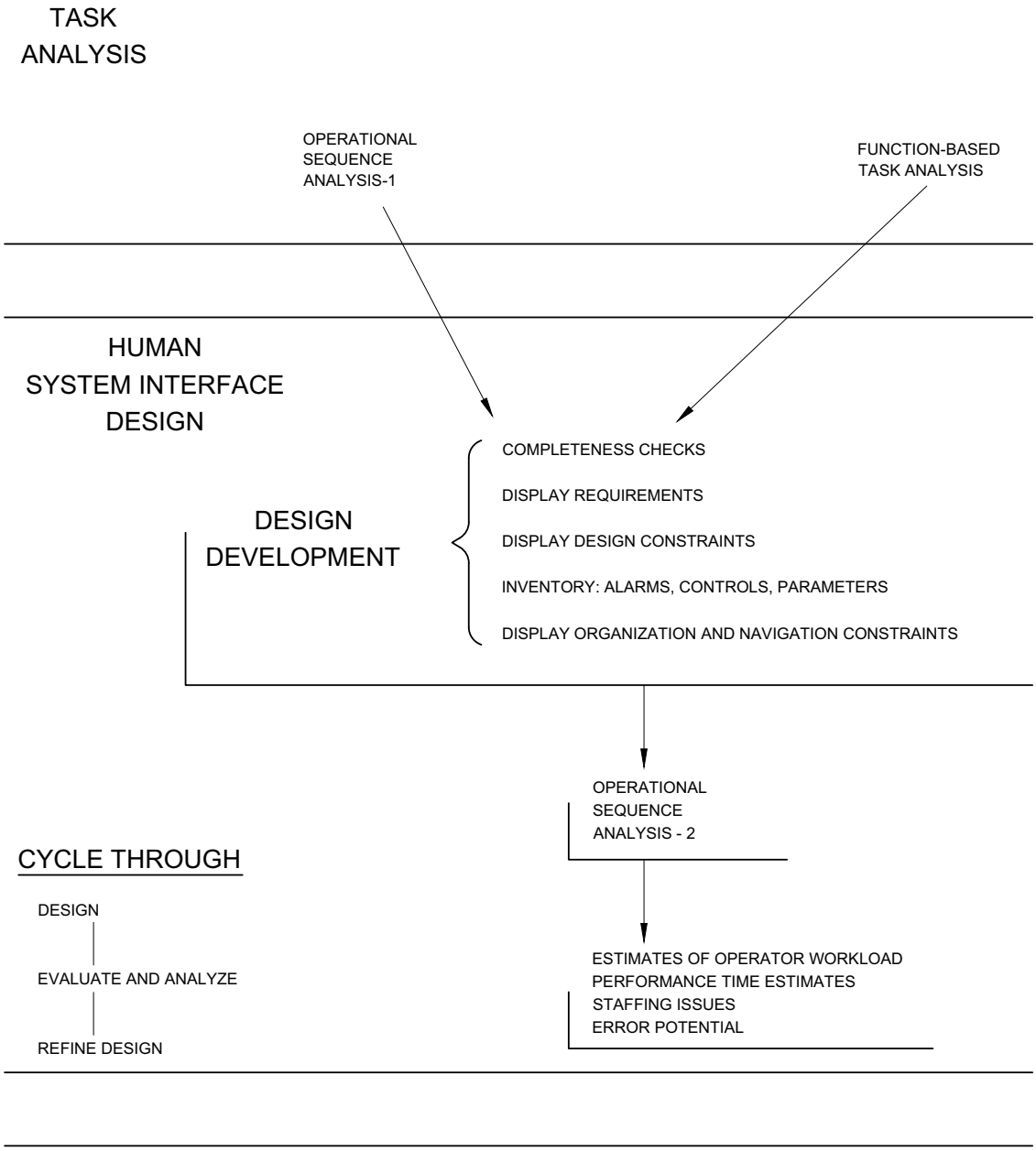

Figure 18.5-2
Task Analysis Utilized as Design Input

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18.6 Staffing

Categories of personnel that are addressed by the Human Factors Engineering program per NUREG-0711, “Human Factors Engineering Program Review Model” (Reference 201), include:

- Licensed operators
- Shift Supervisors
- Non-licensed operators
- Shift technical advisors
- Instrumentation and control technicians
- Mechanical maintenance technicians
- Electrical maintenance technicians
- Radiation protection technicians
- Chemistry technicians
- Engineering support

The minimum level of staffing for control room personnel who directly monitor and control the plant meets the requirements of 10 CFR 50.54(m). Information about the staffing levels of security personnel is contained in the separately submitted physical security plan.

Qualification requirements of security personnel are discussed in the physical security plan.

The baseline level of staffing for the categories of personnel discussed above is derived from experience in current operating nuclear power plants. The number of personnel in operating plants has evolved over many years to a level that is safe and efficient and provides adequate personnel to operate the plant under all conditions, including abnormal and emergency, meets regulatory requirements, and supports individual training and personal needs.

Iterative adjustments are implemented to the level of staffing, as necessary, based on findings and input from periodic reviews and staffing analysis. Input to this analysis includes information derived from the other elements of the human factors engineering program, particularly operating experience review, functional requirements analysis and function allocation, task analysis, human reliability analysis, human-system interface design, procedure development, and training program development.

In addition to the regulatory requirements referenced, input to the analyses and the level of staffing is provided by WCAP-14694, “Designer’s Input to Determination of the AP600 Main Control Room Staffing Level” (Reference 1), AP1000 Combined License Technical Report APP-GW-GLR-010, “AP1000 Main Control Room Staff Roles and Responsibilities” (Reference 202), and EPRI Technical Report 1011717, “Program on Technology Innovation: Staff Optimization Scoping Study for New Nuclear Power Plants” (Reference 203).

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18.6.1 Combined License Information Item

Combined License applicants referencing the AP1000 design will address the staffing levels and qualifications of plant personnel including operations, maintenance, engineering, instrumentation and control technicians, radiological protection technicians, security, and chemists. The number of operators needed to directly monitor and control the plant from the main control room, including the staffing requirements of 10CFR50.54(m), will be addressed.

18.6.2 References

1. WCAP-14694, "Designer's Input To Determination of the AP600 Main Control Room Staffing Level," Revision 0, July 1996.

201. United States Nuclear Regulatory Commission, “Human Factors Engineering Program Review Model,” NUREG-0711, Revision 2, February 2004.

202. Westinghouse, “AP1000 Main Control Room Staff Roles and Responsibilities,” APP-GW-GLR-010, Rev. 2, June 2007.

203. EPRI, “Program on Technology Innovation: Staff Optimization Scoping Study for New Nuclear Power Plants,” Technical Report 1011717, Final Report, August 2005.

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18.7 Integration of Human Reliability Analysis with Human Factors Engineering

Human reliability analysis (HRA) evaluates the potential for human error that may affect plant safety. There are important interfaces between the human factors engineering program and human reliability analysis. Human reliability analysis makes use of outputs of human factors engineering/HSI design activities including analyses of operator functions and tasks and specifications of HSI characteristics. Human reliability analysis is a source of input to human factors engineering/HSI design in identifying plant scenarios, human actions, and HSI components that are important to plant safety and reliability.

[*The objective of integration of human reliability analysis with human factors engineering is to specify the interfaces between human reliability analysis and human factors engineering activities. Reference 1 documents the implementation plan for the integration of human reliability analysis with human factors engineering design*.]* Reference 2 documents the execution and documentation of this implementation plan.

[*The objective of the human reliability analysis/human factors engineering integration implementation plan is to enable:*

- *Human reliability analysis activity to integrate the results of the human factors engineering design activities*

- *Human factors engineering design activities to address critical human actions, risk important tasks, and human error mechanisms, in order to minimize the likelihood of personnel error and to provide for error detection and recovery capability*]*

Human reliability analysis methodology and results are described in Chapter 30 of the AP1000 PRA.

18.7.1 Combined License Information

The execution and documentation of the human reliability analysis/human factors engineering integration implementation plan that is presented in Section 18.7 is addressed in Reference 2 (APP-GW-GL-011, WCAP-16555).

18.7.2 References

[*1. WCAP-14651, "Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan," Revision 2, May 1997.*]*

2. WCAP-16555, "AP1000 Identification of Critical Human Actions and Risk Important Tasks," Revision 0, March 2006.

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18.8 Human System Interface Design

This section provides an implementation plan for the design of the human system interface (HSI) and information on the human factors design for the non-HSI portion of the plant. The human system interface includes the design of the operation and control centers system (OCS) and each of the human system interface resources.

The operation and control centers system includes the main control room, the technical support center, the remote shutdown room, emergency operations facility, local control stations and associated workstations for each of these centers. The AP1000 human system interface resources include:

- Wall panel information system
- Alarm system (also identified as alarm presentation system)
- Plant information system
- Computerized procedure system
- Soft controls/dedicated controls
- PMS safety displays

The wall panel information system presents information about the plant for use by the operators. No control capabilities are included. The wall panel information system provides dynamic display of plant parameters and alarm information so that a high level understanding of current plant status can be readily ascertained. It is located at one end of the main control area at a height such that persons seated at the reactor operator and senior reactor operator workstations can view it while sitting at their respective workstations. It provides information important to maintaining the situation awareness of the crew and for supporting crew coordination. The wall panel information station provides a dynamic display of the plant. It also serves as the alarm system overview panel display. The display of plant disturbances (alarms) and plant process data is integrated on this wall panel information system display. The wall panel information system is a nonsafety-related system. It is designed to have a high level of reliability.

The mission of the AP1000 alarm system, together with the other human system interface resources, is to provide the operation and control centers operating staff with the means for acquiring and understanding the plant's behavior. The alarm system improves the performance of the operating crew members, when acting both as individuals and as a team, by improving the presentation of the plant's process alarms. [*The alarm system supports the control room crew members in the following steps or activities of Rasmussen's operator decision-making model (Reference 25):*]*

- The "alert" activity, which alerts the operator to off-normal conditions

- The "observe what is abnormal" activity, which aids the user in focusing on the important issue(s)

- The process "state identification" activity, which aids the user in understanding the abnormal conditions and provides corrective action guidance. It guides the operating crew into the information display system.

The plant information system is a subset of the data display and processing system (non-Class 1E system), presenting plant process information for use by the operators. The plant information system provides dynamic indications of plant parameters and visual alerts so that an understanding of current plant conditions and status is readily ascertained. The plant information system uses color-graphic visual display units located on the operation and control centers workstations to display plant process data. These displays provide information important to monitoring, planning, and controlling the operation of plant systems and obtaining feedback on control actions. The displays provided by

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the plant information system are nonsafety-related displays, but provide information on both safety-related and nonsafety-related systems.

The computerized procedure system has a mission to assist plant operators in monitoring and controlling the execution of plant procedures. The computerized procedures system is a software system. It runs on the hardware selected for the operation and control centers. The computerized procedure system is accessible from the workstations in the main control room. A procedure writer's guide is developed as part of the human system interface design implementation plan for the computerized procedure system. The writer's guide is the design guidelines document for the computerized procedure system. Information on the writer's guide and on the computerized procedure system is found in Reference 31. Application of the computerized procedure system for emergency operating procedures is licensed outside the United States and is being used in an operating nuclear power plant. Additionally, the application of the computerized procedure system for turbine-generator startup and shutdown is being used in another operating nuclear power plant located outside the United States. Human factors engineering review guidance for computer-based procedures is presented by Reference 9. The design of a backup to the computerized procedure system, to handle the unlikely event of a loss of the computerized procedure system, is developed as part of the human system interface design process. Design options include the use of a paper backup. [*The acceptability of the computerized procedure system and its backup will be confirmed as integral elements of the AP1000 design by the implementation of the AP1000 verification and validation program (Reference 24).*]* Procedure development is addressed in Sections 13.5 and 18.9.

The mission of the controls in the main control room is to allow the operator to operate the plant safely under normal conditions, and to maintain it in a safe condition under accident conditions. The main control room includes both safety-related and nonsafety-related controls. The types of controls in the main control room include both discrete (dedicated) control switches and soft controls. The discrete control switches are controls dedicated to a single function. Figure 18.8-1 shows a representative model of the soft controls used in AP1000. The soft control units are control devices whose resulting actions are selectable by the operator. The instrumentation and control architecture uses both discrete control switches and soft control units. The soft control units are used to provide a compact alternative to the traditional control board switches by substituting virtual switches in the place of the discrete switches.

The final configuration of these elements is dependent upon the results of the human system interface design process described in Subsection 18.8.1 below.

The mission of the PMS safety displays is to provide a Class 1E system to present to the main control room operators the plant parameters which demonstrate the safety of the plant. The qualified data processing system provides for the display of the variables as described in Section 7.5. The informational content of qualified data processing system is provided to the remote shutdown workstation through the plant information system.

18.8.1 Implementation Plan for the Human System Interface Design

Figure 18.2-3 provides an overview of the AP1000 human factors engineering process, including the design stages of the human system interface. The relationship of other human factors engineering process elements to the human system interface design is shown.

The functional design of the operation and control centers system and the human system interface is the activity where the functional requirements for the human system interface resources of the main control room and related operation and control centers system are developed. The output of the functional design is a set of documents that specify the mission, design bases, performance requirements, and functional requirements for each human system interface resource. These

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functional requirement documents and the human system interface design guidelines are used to develop the design specifications. The design specifications are provided as input to the hardware and software system designers for design implementation.

The following subsections describe the activities conducted as part of the human system interface design and the documents that are produced.

18.8.1.1 Functional Design

A system specification document for the operation and control centers system documents and tracks human system interface requirements and design specifications. The operation and control centers system specification document is the umbrella document for capturing human factors requirements and providing a uniform operational philosophy, and design consistency among the individual human system interface resources.

Included in the operation and control centers system specification document are functional requirements and specifications for the AP1000 operation and control centers system, including the main control room, the technical support center, the remote shutdown room, and local control stations. In addition, functional requirement documents are generated for each of the individual human system interface resources. These documents are referenced by the operation and control centers system specification document.

The operation and control centers system specification document and the individual human system interface functional requirement documents include mission statements and performance requirements. The mission statements establish the high level goals and main tasks to be supported by the control center or human system interface resource. Performance requirements represent high level design goals and help to clarify the functional designer's intent. They are high level requirements that may not be readily verifiable by testing or other quantitative means, but are important considerations for meeting the goals defined in the mission statements. The design bases establish the foundation for the design and the rationale behind engineering decisions made and criteria established for the design. Functional requirements include requirements needed to meet the criteria defined in the applicable codes, standards, and customer requirements.

The operations and control centers functional requirements document includes requirements to meet failure, diversity, electrical separation, and other applicable criteria. This document establishes requirements related to access control, redundancy, independence, identification and test capability, and defines requirements on system inputs and outputs. It specifies the system safety classification and defines applicable quality assurance, reliability goals, and environmental qualification requirements. The specification of the cognitive activities in the operator decision-making model that each human system interface resource is intended to support is provided in the operation and control centers functional requirements document.

Reference 25 describes the operator decision-making model and associated operator cognitive activities. As shown in Figure 18.8-2, the HSI interface resources are mapped to four major classes of operator cognitive activities in the model (detection and monitoring, interpretation, control, and feedback).

The contents of this map are then considered in terms of sources of operational complexity that add operator performance demands. The two general sources of complexity considered are 1) use of multiple as opposed to single HSI resources, and 2) increasing situational or scenario-based complexity. Considering the impact of complexity on the mapping leads to "issues"; that is, general cases where adequate human performance should be confirmed.

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Table 18.8-1 presents the resulting set of human performance issues. Note that “feedback” issues have been addressed under “control,” rather than as a separate activity, because feedback activities follow directly from control activities. These human performance issues serve as input to the development of the performance requirements for the operation and control centers system specification document and to the individual human system interface functional requirement documents. The human performance issues and requirements will be addressed by the verification and validation activities described by Reference 24.

18.8.1.2 Design Guidelines

Guidelines for the human system interface design have been developed for the human system interface resources to facilitate the standard and consistent application of human factors engineering (HFE) principles to the design (see Reference 1). Reference 1 contains standards and conventions guidelines and tailors generic human factors engineering guidance to the AP1000 human system interface design and defines how those human factors engineering principles are applied.

These guidelines enable groups of people to simultaneously develop the human system interface in a consistent manner in accordance with the human factors engineering principles established for the design. [*The guidelines are used to perform the human factors engineering design verification activity of the human factors verification and validation plan (Reference 24).*]*

Human system interface design guideline documents include:

- Anthropometric guidelines
- Alarm guidelines
- Display guidelines
- Controls guidelines
- Computerized procedures guidelines

The AP1000 human system interface design guidelines document provides:

- Statements of their intended scope, references to source materials, and instructions for their proper use.

- Specification of accepted human factors engineering guidelines, standards, and principles to which the AP1000 human system interface conforms.

- Specification of design conventions (for example, coding conventions) to which the AP1000 human system interface conforms.

- Documentation of deviations from human factors engineering guidelines, standards and principles, and justification based on documented rationale such as trade study results, literature-based evaluations, demonstrated operational experience, and tests and experiments.

The accepted human factors engineering guidelines documents that were used in compiling the AP1000 human system interface design guidelines document are found in References 2 through 8.

18.8.1.3 Design Specifications

Design specifications are written for the operation and control centers system and the human system interface resources. The design specification documents are the result of applying the guidelines to the functional design. They provide the design for each human system interface resource, including the integration of the hardware and software modules, to satisfy the human system interface

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functional design requirements. Included in these specifications are layout and arrangement drawings, algorithms, display layouts, display task descriptions, navigation mechanisms and resource lists.

The functional requirement documents are used to define the bases for the system design specifications.

The operation and control centers system specification document and human system interface functional requirements and design specification documents provide input to the generation of instrumentation and control system specification documents, such as the system specification document for the data display and processing system. These specification documents are used as inputs to the hardware and software system designers to generate implementation documents such as hardware and software specifications.

18.8.1.4 Man-in-the-Loop Testing

An integral part of the human system interface design process is the conduct of man-in-the-loop engineering tests to obtain feedback from prototype design products early in the design process.

The use of engineering tests is a good engineering practice, which reflects an iterative design process. By providing feedback early, before the detailed design is complete, engineering tests can help to improve the design and to avoid problems in the final product. Engineering tests also may offer concrete insight on questions that cannot be resolved logically (for example, by guidance or analysis). Finally, results from engineering tests provide evidence of design adequacy. Engineering tests thus serve to increase confidence and reduce project risk in the design process.

Engineering tests are performed to obtain empirical results that can be applied directly to understanding and improving the design product. More specifically, engineering tests are designed to produce the following types of results for the prototype design:

- Design-specific operating experience
- Confirmation of necessary performance and integration
- Identification of specific problems
- Subjective feedback from expert users and observers

[*The man-in-the-loop test plan to obtain feedback from prototype design products early in the design process is defined and documented in Reference 46.*]* The results of the engineering testing are used to refine the design of the operation and control centers system and the human system interface.

18.8.1.5 Mockup Activities

A mockup of portions of the main control room working area is constructed as part of the human system interface design process. The partial mockup consists mainly of non-operational representations of the desks, displays, and panels. The mockups are constructed to the anthropometric profiles and arranged in the floor layout intended for the main control room.

The partial mockup is used to examine and verify, as needed, physical layout aspects such as availability of workspace, physical access, visibility, and related anthropometric and human factors engineering issues. It will also be used for walk-through exercises to examine issues such as staffing levels, task allocation, and procedure usage.

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18.8.1.6 Human System Interface Design Documentation

The human system interface design is documented through a system specification document for the operation and control centers system, functional requirement documents, design criteria documents, design review documents, and documentation of design configuration change control.

18.8.1.7 Task-Related Human System Interface Requirements

As shown in Figure 18.2-3, the results of other human factors engineering program elements are used as input and bases for developing the operation and control center system and human system interface resources functional design (mission statements, performance requirements, design bases, functional requirements), guideline documents and the design specification documents. Staffing assumptions, operating experience reviews, functional requirements analysis and allocations, task analysis, and integration of human reliability analysis provide the bases for identifying the human system interface requirements needed to support human functions and tasks. The resulting human system interface requirements are documented in the human system interface resource functional design documents (operation and control centers system specification document and the individual human system interface resource functional requirements document), guidelines document and design specification documents. Subsections 18.8.1.1 through 18.8.1.3 provide descriptions of these documents.

The AP1000 task analysis, described in Section 18.5, includes two complementary activities: function-based task analysis (FBTA) and traditional task analysis, or operational sequence analysis (OSA). The function-based task analysis identifies the indications, parameters, and controls that the operator needs to make decisions about the respective function. There is also a verification that the indications and controls identified in the process analysis are included in the design. The operational sequence analysis, completed as part of the task analysis process, focuses on specifying the operational requirements for the complete set of tasks selected. One of the guidelines used in selecting tasks for analysis are those tasks that represent the full range of activities in the AP1000 emergency response guidelines. One type of information provided by the operational sequence analysis is an inventory of alarms, controls, and parameters needed to perform the task sequences. The operational task analysis results include the identification of controls, alarms, and parameters needed by the operator to execute task sequences found within the emergency response guidelines. These results serve as a cross-check with the function-based task analysis results. Design reviews held during the human system interface design serve as another means of verifying completeness and identifying and correcting omissions. [*The task support verification activity of the human factors verification and validation (Reference 24) verifies that the human system interface design provides the necessary alarms, displays, and controls to support personnel tasks.*]*

The collective results of the task analysis activities identify the tasks and operational information needed by the operator to execute these tasks. For each display, a display task description is written. The display task description includes the identification of the informational needs to be supported by the display. The features, dynamic characteristics, calculated values, and supporting algorithms for the display are part of the display task description. The design specification of a display includes the range, precision, and measurement units of the parameters provided in the display. These parametric characteristics are chosen to support the task and the operator informational needs. The parametric characteristics, identified in the design specification, are provided using the guidelines presented in the design guidelines document for displays. The basis for the parametric characteristics chosen for the displays is found in the design guidelines document.

18.8.1.8 General Human System Interface Design Feature Selection

The AP1000 human system interface resources include the wall panel information system, alarm system, plant information system, computerized procedure system, controls, (soft and dedicated)

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and the PMS safety displays. These human system interface resources are used as a starting point to define how the human system interface supports operator performance. [*Reference 25 describes the operator decision-making model that is used by the task analysis activities to identify the operator's information and control requirements.*]* The human system interface resources are mapped to the major classes of operator activities identified from this model. Figure 18.8-2 illustrates this mapping. The human performance requirements that each human system interface resource supports are identified as part of the design process.

The human system interface resources are chosen based upon utility requirements and review of operating experience. The goal of the human system interface design is to provide the operators with effective means for acquiring and understanding plant data and executing actions to control the plant's processes and equipment. Through implementation of the human system interface design process, the identified AP1000 human system interface resources are developed.

Design alternatives for a feature within a human system interface resource (such as the use of a mouse, trackball, or touchscreen for soft controls) are evaluated. A decision is made based upon evaluation methods including human factors/trade-off studies, reviews of nuclear industry operating experience or reviews of other industry experience, experience gained from past projects, and utility input. The basis and rationale for the decisions are provided in the functional design documentation.

18.8.1.9 Human System Interface Characteristics: Identification of High Workload Situations

Identification of high operator workload situations and their consequent changes in operator response times or likelihood of operator error is a usability issue. Potential impact on operator workload is a criterion in selecting the human performance issues identified in Table 18.8-1.

Identification of high-workload situations through analytic techniques and part-task simulations is part of the human factors engineering program (Section 18.5 on Task Analysis).

Use of Workload Measurement Techniques

As part of task analysis activities (Section 18.5), analytic approaches are used to estimate workload. Analytic methods include the use of task analysis.

Usability Guidance

Usability guidance is included in the human system interface design guidelines, as discussed in Subsection 18.8.1.2.

Workstation Usage Scenarios

The physical layout of the AP1000 control room and related control centers follows established ergonomic guidelines including consideration of fatigue and alertness of operators sitting at workstations.

Environmental Conditions

Determination of environmental conditions (lighting, noise, ambient working temperatures, radiation, air quality, and humidity) in the control room, the remote shutdown room, and at local control stations employ well-accepted standards from the fields of industrial and human engineering such as References 14, 15 and 16. Relevant guidance from prior studies in the nuclear power area (References 17 through 20) is also used.

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The worst credible conditions that can be encountered by operators in the main control room are identified as outcomes of design basis scenarios. Effects on operator performance and the effects of extremes of habitability during degraded conditions are considered in the design specification.

Local Control Actions

Critical human actions and risk important tasks are identified by the probabilistic risk assessment/ human reliability analysis process. [*Reference 23 presents the process of identifying the critical human actions and risk important tasks and the implementation plan for integrating human reliability analysis into the human factors engineering program.*]* Critical human actions or risk important tasks are examined by task analysis, human system interface design, and procedure development, to identify changes to the operator task or the control and display environment to reduce or eliminate sources of error.

18.8.1.10 Human System Interface Software Design and Implementation Process

This subsection describes the software design, implementation, and verification process established to verify that human system interface functional requirements are implemented by the software. The software design, implementation, and verification process uses a top-down approach to incorporate the system design requirements and the functional requirements into software module design.

Software refers to the computer instructions and information provided to implement a subset of the human system interface functional requirements. The software design and implementation process is a subset of the overall human system interface design process. It consists of system software design specifications, software design, software implementation, and software verification.

The system software design specification activity takes as its input the system functional requirement and specification documents and produces software design requirements documents and the software verification test procedures. Software design requirements documents list the functions, performance, design constraints, and attributes of the system software.

The software design activity takes software design requirements and produces software design specification documents. Software design specification documents provide the details for the software design at the module level and assembly level. These documents define the software language, logical structure, variable names, information flow, logical processing steps, and data structure of the system software programs. They also describe the functions performed, support software, storage and execution limitations, interface constraints, error conditions, error detection, error response actions, and details of the software operation in the hardware environment.

The software implementation activity implements the software design specifications in the form of documented source programs and object code. The source program and associated documentation contain the comments, functional diagrams, external references, and internal module descriptions.

The object code is generated from the source program and installed in processor memory to perform the functions specified by the software design specifications.

In the software verification testing activity, the software is tested to verify that it complies to the system software design requirements. The software is tested according to the software verification test procedures.

Nonconformances of the software to the software verification test procedures are documented by trouble reports, and changes are made. In the case where the error is a result of an error in the system software design requirements or the software design specifications, these documents are revised. The software test results report presents a summary of the software verification testing results.

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18.8.2 Safety Parameter Display System (SPDS)

[*The Safety Parameter Display System is designed following the human system interface design implementation plan*]* described in Subsection 18.8.1. [*The Safety Parameter Display System is integrated into the design of the AP1000 human system interface resources.*]*

As noted in Section 4.1.a of Reference 27 "...the principle purpose and function of the Safety Parameter Display System is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident." Since the main intended use is during relatively rare occurrences, human-factors engineering suggests that the operators will find that the use of data acquisition habits acquired and repeated during the normal operation of the plant will be the most successful. A system in the control room that only varies its output during abnormalities may require a shift in mental focus and in data acquisition habits and subsequent analysis. An effective means for conveying the safety state of the plant is to provide data and displays for normal operation that employs the Safety Parameter Display System required principles for data synthesis, concentration and display. This operator interface is operational over the range of plant conditions specified by the Safety Parameter Display System requirements, as well as during normal operations.

The operator-interface to the plant is improved by integrating Safety Parameter Display System requirements into the overall human system interface design to avoid the need for another system that is infrequently used.

The following subsections describe [*the approach to meeting the regulatory requirements for a Safety Parameter Display System by addressing the Safety Parameter Display System requirements of References 26 and 27.*]*

18.8.2.1 General Safety Parameter Display System Requirements

The AP1000 human system interface resources used to address the Safety Parameter Display System requirements are the alarm system, plant information system (workstation visual display unit displays), and the computerized procedure system. The AP1000 human system interface data display (alarms and visual display unit displays) is organized around the Safety Parameter Display System requirement of plant process functions. Expressing plant state in terms of process functions is incorporated in the AP1000 control room design. This is expected to improve the human interface by making the data presentation interface seamless as the plant moves from one operational state to another.

An alarm system which organizes the presentation of alarms by process function and adapts a "dark board" approach (for all plant modes) continually indicates the state of each of the functions. By remaining dark when the process is performing as expected, the process functions are interpreted as being satisfied. An alarm indication displayed in any function indicates that the function is in jeopardy. In this way, the set of alarms that is active is the minimum set. The alarm system is capable of displaying a full range of alarms based on important plant parameters and data trends. The alarms indicate when process limits are being approached and exceeded.

Section 18.7 and [*Reference 23 present an implementation plan for integrating the human reliability analysis with human factors engineering.*]* The critical human actions and the risk important tasks identified through the execution of this plan are used as an input to the task analysis activities and subsequently to the design of the human system interface. They are also used to evaluate the Safety Parameter Display System functions and parameters selected to monitor these functions. The human system interface, which includes the integration of Safety Parameter Display System

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requirements, is designed to reduce the likelihood of operator error and provide for error detection and recovery capability for the identified critical human actions and risk important tasks.

18.8.2.2 Display of Safety Parameters

The functionally organized plant information system displays, including the Safety Parameter Display System-related displays, are accessed on the workstation visual display units (VDU) using a cursor. The AP1000 operator workstations employ a windowing system which allows a single cursor to cover the visual display unit screens. The design allows the operator to recover a specific parameter within one or two actuations of the pointing device.

The design goal for the AP1000 human system interface is to update the displays every 1 to 2 seconds. The process data sampling rate is 1 second or less. Sequence of events (SOE) points can be sampled at a rate of once every milli-second and are available within the AP1000 human system interface. The Safety Parameter Display System responds to user commands in less than 10 seconds. The design goal for graphical display response time, from user command to developed graphical display, in the AP1000 human system interface is 2 seconds.

The AP1000 alarm system includes plant overview alarms that are organized around the concept of plant process functions. These process functions address the five SPDS functions. The alarm system overviews, including the functional organization, are integrated into the wall panel information system displays.

During the execution of emergency operating procedures, the computerized procedure system provides a continuous display of the status of each critical safety function.

The Safety Parameter Display System data and data display organization are available to the control room staff.

[*The AP1000 human system interface process display set (from the plant information system) is organized into two hierarchies that are linked together. One is focused upon providing the process data from a functional perspective and the other from a physical perspective. Both follow the concept of abstraction/aggregation suggested by Rasmussen as described in Reference 25. Top levels in the hierarchy are plant wide summaries, lower levels are component details. The hierarchy is structured so as to reflect the plant process functional decomposition performed during the function based task analysis described in Reference 25.*]*

Process display presentation for the control room users is organized by functions. The function based task analysis integrates the functional organization design principles dictated by the Safety Parameter Display System requirements into the AP1000 human system interface.

Plant process displays and plant controls necessary to operate the plant are located on the reactor operator console. There are a total of six redundant workstations on the reactor operator console.

Because the Safety Parameter Display System requirements are an integral part of the AP1000 human system interface design, the Safety Parameter Display System workstation is the AP1000 human system interface control room workstation, the Safety Parameter Display System displays are the workstation displays; and the display accessing “controls” used to access Safety Parameter Display System displays are the same as those used to access any workstation display.

Safety Parameter Display System-related information is physically displayed such that the information can be read from the Safety Parameter Display System user’s position. Each reactor operator’s workstation contains the human system interface operator process displays. The senior

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reactor operator has separate workstations that have the operator process displays. The wall panel information system is available to the main control room staff.

The AP1000 human system interface provides the status of the Safety Parameter Display System functions. The Safety Parameter Display System functions include:

- Reactivity control
- Reactor core cooling and heat removal from the primary system
- Reactor coolant system integrity
- Radioactivity control
- Containment conditions

The AP1000 alarm system provides overview alarms addressing the five Safety Parameter Display System functions. These overview alarms, integrated into the wall panel information system displays, are continuously displayed. Most of the safety parameters used to monitor the status of each Safety Parameter Display System function are continuously displayed on the wall panel information system displays. Those that are not continuously displayed on the wall panel are accessible at the operator's workstation. During the execution of emergency operating procedures, the AP1000 computerized procedure system provides a continuous display of the status of the critical safety functions.

Safety Parameter Display System-related information is physically displayed such that the information is readable from the reactor operator workstation. Each reactor operator's workstation contains the plant information system process displays. The control room supervisor (shift foreman) has an independent workstation that also has the process displays. The wall panel information system is available to the main control room staff.

18.8.2.3 Reliability

The AP1000 instrumentation and control (I&C) systems, including the human system interface, have reliability/availability design criteria. A description of the instrumentation and control system design features is found within Section 7.1.

The human system interface design includes the capability to build password or key-lock accessibility on the human system interface database. In addition, the system carries and displays data quality on the data in the system.

The alarm overviews integrated into the wall panel information system include indication of the operability of the alarm system itself.

18.8.2.4 Isolation

The Safety Parameter Display System as integrated into the overall human system interface is isolated from safety systems. Electrical isolation devices are discussed in Subsection 7.1.2.

18.8.2.5 Human Factors Engineering

Section 5 of Reference 28 presents the need for human-factors engineering in the design of the Safety Parameter Display System. The Safety Parameter Display System is designed using the implementation plan described in Subsection 18.8.1. [*This implementation plan includes the application of human factors engineering principles that address the criteria of the Human Factors Engineering Program Review Model (Reference 29).*]* This is also in accordance with Reference 48.

The AP1000 main control room and human system interface design reduces the number of individual computerized operator support systems by incorporating the requirements of the Safety Parameter

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Display System into the design requirements for the AP1000 human system interface. This is accomplished primarily by those human system interface resources that produce and display the process abnormality alarms and the process graphical visual display units.

Parameter units of measure, labels, and abbreviations displayed by the human system interface resources are consistent with the units of measure, labels, and abbreviations included in the emergency operating procedures.

The human system interface displays information is in a form that does not require transformation or calculation. High- and low-level setpoints are consistent with the reactor protection system setpoints. The high- and low-level setpoints are visible in both the messages created by the AP1000 alarm system and on the indications, trends and graphs that appear as part of the process displays of the AP1000 plant information system.

Consistency of calculated values, such as subcooling margin, is maintained. The AP1000 instrumentation and control and human system interface architecture shares process data through a database.

The technical basis for software specifications are verified with plant data (for example, heat-up and cool-down limits, steam generator setpoints and high- and low-level alarm setpoints). The AP1000 human system interface is designed so that the plant data is a separate data file independent of the software specifications.

18.8.2.6 Minimum Information

The AP1000 human system interface resources used to address the Safety Parameter Display System requirements are the alarm system, plant information system, and the computerized procedure system. The AP1000 human system interface displays sufficient information to determine plant safety status with respect to the Safety Parameter Display System safety functions. [*The safety functions and respective parameters presented in Table 2 of Reference 32 are used as a starting point for the AP1000.*]* The human system interface design implementation plan is described in Subsection 18.8.1 and includes the integration of Safety Parameter Display System requirements into the human system interface. [*The Safety Parameter Display System design issue of "minimum information" is tracked by the human factors engineering issues tracking system.*]*

18.8.2.7 Procedures and Training

Sections 13.2 and 13.5 describe the development of training programs and plant procedures respectively. Reference 30 describes how training insights are passed from the designer to operations personnel who participate as subjects in the HFE V&V activities. Reference 31 provides input to the development of plant operating procedures.

18.8.3 Operation and Control Centers System

The human system interface includes the design of the operation and control centers system. The design of each of these control centers is conducted using the human system interface implementation plan presented in Subsection 18.8.1. The mission for each of the operation and control centers in the AP1000 is provided in the following subsections. Coupled with each mission statement is a brief description of the major tasks and design features that are supported by that center.

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18.8.3.1 Main Control Room Mission and Major Tasks

The mission of the main control room is to provide a seismically qualified habitable and comfortable location for housing the resources for a limited number of humans to monitor and control the plant processes.

The major tasks performed in the main control room include monitoring, supervising, managing, and controlling those aspects of the plant processes related to the thermodynamic and energy conversion processes under normal, abnormal, and emergency conditions. Operating staff can monitor, supervise, manage, and control processes that have a real-time requirement for protecting the health and safety of operating personnel. The main control room supports the operator's decision-making process, and promotes the interaction with other plant personnel, while preventing distractions by non-operating personnel. The main control room provides the interfacing resources between the operation of the plant and the maintenance of the plant. Its areas include the main control area, the operations work area, the shift supervisor's office, and the operations break room (see Figure 1.2-8). Habitability systems are described in Sections 6.4 and 9.4.

18.8.3.2 Main Control Area Mission and Major Tasks

[*The mission of the main control area is to provide the support facilities necessary for the operators to monitor and control the AP1000 efficiently and reliably. Figure 6.4-1 provides a view of the main control area. The main control area includes the reactor operator workstations, the supervisor's workstation, the dedicated safety panel and the wall panel information system. The layout, size and ergonomics of the operator workstations and the wall panel information system depicted in this figure does not reflect the results of the human system interface design implementation plan*]* described in Subsection 18.8.1. The actual size, shape, ergonomics and layout of the operator workstations and the wall panel information system is an output of the implementation plan.

[*The major task of the main control area is to provide the human system interface resources that determine the plant state and implement the desired changes to the plant state during both normal and emergency plant operations. The main control area provides alarms to alert the operator to the need for further investigation. Plant process data displays permit the operator to observe abnormal conditions and identify the plant state. The controls enable the operator to execute actions. The process data displays and the alarms provide feedback to enable the operator to observe the effects of the control actions.*

Each reactor operator workstation contains the displays and controls to start up the plant, maneuver the plant, and shut down the plant.]* Reference 44 presents input for the determination of the staffing level of the operating crew in the main control room. [*Each workstation is designed to be manned by one operator. There is sufficient space and operator interface devices for two operators. The physical makeup of the reactor operator workstations is identical. The human system interface resources available at each workstation are:*

- *Plant information system displays*
- *Control displays (soft controls)*
- *Alarm system displays*
- *Computerized procedure displays*
- *Screen and component selector controls*

The supervisor workstation is identical to the reactor operator workstations, except that its controls are locked-out. The supervisor workstation contains both internal plant and external plant communications systems.

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Upon failure of a reactor operator workstation, the failed workstation is locked out, and the supervisor workstation controls are unlocked. This modified workstation configuration maintains independent, redundant workstations.

A dedicated safety panel is located in the main control area. The PMS safety displays and the dedicated safety system controls are provided in this panel. These visual display units are the only monitoring display devices in the main control room that are seismically qualified and provide the post-accident monitoring capabilities in accordance with Regulatory Guide 1.97. Dedicated system-level safety system control switches are located on the dedicated safety panel to provide the operators with safety system actuation capabilities.]* A minimum inventory of these dedicated displays and controls are presented in Section 18.12.

[*There is storage space for supplies, protective clothing and some spare parts. Cabinets are provided for necessary documents, and a drawing laydown area is provided for the operators' use. Restroom and kitchen facilities are provided for the main control room operations crew.*]*

18.8.3.3 Operations Work Area Mission and Major Tasks

The operations work area provides an area for personnel who support plant operations to work in close proximity to the main control area, but not in the main control area, in order to minimize distractions to the plant operators. Personnel in the operations work area can access plant data via one or more workstations to enable personnel to monitor the current state of systems, major components, and equipment. Additional support equipment may be provided as needed.

18.8.3.4 Remote Shutdown Workstation Mission and Major Tasks

[*The mission of the remote shutdown workstation is to provide the resources to bring the plant to a safe shutdown condition after an evacuation of the main control room. The remote shutdown workstation resources are based on an assumed evacuation of the main control room without an opportunity to accomplish tasks involved in the shutdown except reactor trip.*]* Subsection 7.4.3 discusses safe shutdown using the remote shutdown workstation, including design basis information.

18.8.3.5 Technical Support Center Mission and Major Tasks

The mission of the technical support center (TSC) is to provide an area and resources for use by personnel providing plant management and technical support to the plant operating staff during emergency evolutions. The TSC relieves the reactor operators of peripheral duties and communications not directly related to reactor system manipulations and prevents congestion in the control room. The Technical Support Center (TSC) location is described in the Emergency Plan.

Communications needs are established for the staff within the TSC, and between the TSC and the plant (including the main control room and operational support center), the emergency operations facility, the Combined License holder management, the outside authorities (including the NRC), and the public.

The design includes adequate shielding as discussed in Chapter 12. Adequate space, resources and access is provided for maintenance, emergency equipment and storage.

Consistent with NUREG 0737, the technical support center is nonsafety-related and is not required to be available after a safe shutdown earthquake.

The size of the TSC complies with the size requirements of Reference 28. [*The TSC complies with the habitability requirements of Reference 27 when electrical power is available.*]*

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Should habitability be challenged within the TSC due to lack of cooling or a high radiation level resulting from a beyond-design-basis accident, the plant management function of the TSC is transferred to the main control room.

The EOF design is discussed in Chapters 13 and 18, including the specification of its location (Subsection 18.2.6) and emergency planning, and associated communication interfaces among the main control room, the TSC, and the EOF (Section 13.3).

Subsection 18.2.1.2 provides a description of assumptions and constraints, including utility requirements, that are used as inputs to the human factors engineering program and the human system interface design. As stated earlier under Section 18.8, the human system interface design includes the design of the operation and control centers system (main control room, TSC, remote shutdown room, emergency operations facility, local control stations and associated workstations) and each of the human system interface resources. The main control room design (environment, layout, number and design of workstations) supports emergency operations with a maximum crew compliment consisting of eleven individuals. These eleven include two individuals with senior reactor operator licenses, three with reactor operator licenses, one observer from the NRC, one from the plant owner's management and one communicator.

[*The design of the TSC's interfaces is included with the design of the human system interface.*]* Subsection 18.8.1 provides an implementation plan for the design of the human system interface. As shown in Figure 18.2-3, the results of the human factors engineering program elements are used as input and bases for developing the operation and control center system and human system interface resources functional design. This includes task analysis. Section 18.5 provides the implementation plan for the task analysis activities.

An uninterruptible power supply system provides approximately two hours of backup power supply to the TSC displays should ac power become unavailable.

18.8.3.6 Operations Support Center Mission and Major Tasks

The operations support center (OSC) is not within the scope of the human factors engineering program, but it is an emergency response facility. The mission of the operations support center is to provide a habitable area for operations support personnel and the resources to coordinate the assignment of duties and tasks to personnel outside of the main control room and the technical support center in support of plant emergency operation. The operations support center and the TSC are in different locations. The Operations Support Center (OSC) location is described in the Emergency Plan.

The major task of the operations support center is to provide a centralized area and the necessary supporting resources for the assembly of predesignated operations support personnel during emergency conditions. The operations support center provides the resources for communicating with the main control room and the technical support center. This permits personnel reporting to the operations support center to be assigned to duties in support of emergency operations.

18.8.3.7 Radwaste Control Area Mission and Major Tasks

The mission of the radwaste control area is to provide a habitable area and the appropriate resources for the operation of the radwaste processing systems. These resources include alarms, displays, controls, and procedures. These resources are located in a control area outside of the main control room.

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18.8.3.8 Local Control Stations Mission and Major Tasks

The mission of local control stations is to provide the resources, outside of the main control room, the remote shutdown room, and the radwaste control area, for operations personnel to perform monitoring and control activities. The capability to access displays and controls (controls as assigned by the main control room operators) for local control and monitoring, from selected locations throughout the plant, is provided. Activities that are implemented through local control stations are reviewed to verify that their removal from the main control room is consistent with the operator staffing and performance considerations. Human system interface locations are provided for single task operations such as the operation of a manual valve.

18.8.3.9 Emergency Operations Facility

The design of the emergency operations facility, including specification of the location, in accordance with the AP1000 human factors engineering program, is discussed in Subsection 18.2.6.

18.8.4 Human Factors Design for the Non-Human-System Interface Portion of the Plant

18.8.4.1 General Plant Layout and Design

The AP1000 design process incorporates a human engineering approach to operations and maintenance. Maintainability design guidelines and human factors and as-low-as-reasonably-achievable (ALARA) checklists are used to meet the requirements of a human engineered environment. The design objectives include reducing worker exposure and eliminating unnecessary inspection and maintenance tasks.

18.8.4.1.1 Maintainability

Design features such as component selection, layout and standardization increase the probability that targeted repair times are achieved. These features coupled with a preventative maintenance program help the AP1000 meet its objectives for operation and maintenance. Design requirements from the utility industry and industry design practices establish criteria for layout, changeout, and replacement for parts and components; access for major pieces of equipment; and vehicle passage.

Critical path outage models are prepared for the AP1000. A typical refueling and maintenance outage schedule is used by design engineers. The model indicates maintenance windows for major outage events. Maintenance and testing of equipment and necessary plant operations (for example, refueling, heatup, and cooldown) are scheduled within the outage window.

18.8.4.1.2 Accessibility and Equipment Laydown Provisions

AP1000 maintainability design guidelines assist designers in identifying top-level layout requirements for equipment accessibility. Component engineers specify space requirements for routine maintenance, inservice inspection, testing and component replacement.

Frequency of inspection and maintenance dictates whether permanent platforms, ladders, and scaffolding are provided.

Overhead access is considered when equipment or tooling must be lifted into place or supported by a crane. Removable floor gratings and plugs are examples of features that provide overhead accessibility.

Permanent lifting devices are provided to enhance maintainability.

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The use of robotics and automated devices are considered in the AP1000 design.

Robotic devices, such as refueling cavity decontamination units, are considered in the layout of the refueling cavity so that such interferences as light fixtures, tool hangers and personnel ladders are removable or do not affect the use of the robotic units.

Valve space enveloping drawings indicate the minimum space requirements. Equipment and module designers locate and arrange the valves to maintain the required space envelope.

The turbine-generator contains built-in features to increase accessibility for in-place inspection and maintenance. Access ports in the turbine housings allow routine inspections to be performed without dismantling the turbine casing. Laydown area is provided in the turbine building to access components and to allow for concurrent work.

18.8.4.1.3 Lighting

The AP1000 normal and emergency lighting system is designed to provide illumination levels required for the safe performance of plant operation under normal and emergency conditions.

18.8.4.1.4 Radiation Protection and Safety

The AP1000 design process incorporates radiation exposure reduction principles to keep worker dose ALARA. ALARA checklists are used in design evaluations. Exposure length, distance, shielding, and source reduction are the fundamental criteria incorporated into the design process.

Design features such as readily detachable insulation, as-built smooth surfaces for non-destructive examination, and “modular type” replacement components reduce worker time in radiation areas.

The large AP1000 containment vessel provides laydown space to transfer subcomponents to storage areas until needed. The reactor head is remotely located on the operating deck to reduce background radiation to refueling personnel.

Provisions for remotely operated tooling are considered during the design process. Space is provided to clean and inspect the reactor vessel O-ring grooves using a remotely operated device. Remotely controlled radiation and surveillance equipment is considered for high radiation areas.

Special provisions for radiation shielding are included in the AP1000 design. Permanent shielding built into the integrated head package reduces worker exposure resulting from the incore instrumentation operation.

Material selection and surface conditioning are important elements in radiation exposure reduction. Electropolishing of surfaces exposed to reactor coolant primary water is considered to reduce crud deposits and aid in decontamination.

The AP1000 radioactive waste processing facilities are designed to concentrate radioactive waste processing and drumming activities in remote areas to reduce contact with the majority of plant personnel.

18.8.4.1.5 Communication

The AP1000 communication system provides voice communication during normal operations, plant outages, and emergency operations. The system includes broadcast of alarm signals in plant-wide emergency situations. The wireless telephone system enables plant personnel to remain in direct communication via wireless, hand-carried telephones throughout the plant. Headset-style telephones

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are available for individuals requiring hands-free operation. Some communication devices have built-in compatibility with protective clothing including respirators.

A paging system is used as a backup to the wireless telephone system. In the event of a failure of the wireless system, personnel communicate via a plant-wide broadcast and five party lines. Emergency broadcasts are announced through this system.

Communication during AP1000 refueling and maintenance outages is enhanced by a sound-powered communication system. Refueling, maintenance, and cold shutdown loops are provided. Jacks are placed in locations where plant personnel are located during these activities.

A private branch exchange system is capable of duplex voice communication between stations. These telephones are placed in acoustic booths in those areas having high ambient noise levels to improve user interface. See Subsection 9.5.2 for information on the communication system.

A site wide radio system provides primary wireless communications for Security and fire brigade activities. Radio system is also used by other departments as needed.

18.8.4.1.6 Temperature, Humidity, Ventilation

Radioactive and nonradioactive ventilation systems are provided in required areas. The ventilation systems are designed to control the environment within the plant and to protect the environment outside the plant. Requirements for temperature, humidity, and ventilation vary, depending on work location, frequency of use, and work description.

18.8.4.1.7 Emergency Equipment

Emergency equipment for treatment of injured personnel is placed in appropriate locations. Provisions for emergency equipment are considered during plant layout.

18.8.4.1.8 Storage

Storage facilities are identified in the AP1000. Radioactively clean and contaminated storage areas are designated.

18.8.4.1.9 Coding and Labeling

Equipment located in the AP1000 has a unique identifier and plant descriptive name. The configuration management system includes the identification of the equipment in the plant. Each component is assigned an identifier during the design process. The identifier is maintained through manufacturing, construction, and operation. The components are labeled according to the assigned identifier. These labels help avoid errors in operating or working on the wrong equipment and in reporting problems or conditions observed in the plant. The labels help reduce the training burden for operating and maintenance personnel.

Color, syntax, abbreviations and symbols are consistently applied. The labels are located in an easily visible location on the component and are not hidden by insulation, equipment covers, or surrounding equipment. Labels are fastened to the component to prevent easy detachment of the label.

18.8.5 Combined License Information

The execution and documentation of the human system interface design implementation plan that is presented by Section 18.8 is addressed in APP-GW-GLR-082 (Reference 47), and the applicable changes are incorporated.

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18.8.6 References

1. APP-OCS-J1-002, “AP1000 Human System Interface Design Guidelines,” (Westinghouse Proprietary).

2. CEI/IEC 964, “Design for Control Rooms of Nuclear Power Plants,” International Electrotechnical Commission, Geneva, Switzerland, 1989.

3. IEEE Std 1023-2004, “IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities.”

4. IEEE Std 1289-1998, “IEEE Guide for the Application of Human Factors Engineering in the Design of Computer-Based Monitoring and Control Displays for Nuclear Power Generating Stations.”

5. NUREG-0700, “Human-System Interface Design Review Guideline,” Rev. 2, U.S. Nuclear Regulatory Commission, Washington, D.C., May 2002.

6. Not used.

7. NUREG/CR-6105, “Human Factors Engineering Guidelines for the Review of Advanced Alarm Systems,” U.S. Nuclear Regulatory Commission, Washington, D.C., September 1994.

8. MIL-STD-1472, Department of Defense Design Criteria Standard: Human Engineering, Revision F, August 1999.

9. NUREG/CR-6634, “Computer-Based Procedure Systems: Technical Basis and Human Factors Review Guidance,” U.S. Nuclear Regulatory Commission, Washington, D.C., March 2000.

10. Not used.

11. Not used.

12. Not used.

13. Not used.

14. Beranek, L. L., “Revised Criteria for Noise in Buildings,” Noise Control, Vol. 3, Nr.1, p. 19ff.

15. Grandjean, E., “Fitting the Task to the Man: An Ergonomic Approach,” London: Taylor and Francis Ltd., 1981.

16. Van Cott and Kinkade, “Human Engineering Guide to Equipment Design,” Washington D.C.: U.S. Government Printing Office, 1972.

17. Electric Power Research Institute, “Human Factors Guide for NPP Control Room Development,” Final Report on Project 1637-1. EPRI NP-3659, 1984.

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18. Electric Power Research Institute, “Advanced Light Water Reactor Utility Requirements Document, Vol. III. ALWR Passive Plant, Chapter 10: Man-Machine Interface Systems,” Revision 6, December 1993.

19. International Electrotechnical Commission, “Design for Control Rooms of Nuclear Power Plants,” IEC Standard 964, 1989.

20. International Electrotechnical Commission, “Operating Conditions for Industrial-Process Measurement and Control Equipment,” IEC Standard 654-1, 1979.

21. Not used.

22. Not used.

[*23. WCAP-14651, “Integration of Human Reliability Analysis With Human Factors Engineering Design Implementation Plan,” Revision 2, May 1997.*]*

[*24. WCAP-15860, “Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan,” Revision 2, October 2003.*]*

[*25. WCAP-14695, “Description of the Westinghouse Operator Decision Making Model and Function Based Task Analysis Methodology,” Revision 0, July 1996.*]*

[*26. 10 CFR 50.34 (f) (2) (iv).*]*

[*27. NUREG-0737, Supplement 1; “Requirements for Emergency Response Capability.”*]*

28. NUREG-0696, “Functional Criteria For Emergency Response Facilities.”

[*29. NUREG-0711, “Human Factors Engineering Program Review Model,” U.S. NRC, July 1994.*]*

30. WCAP-14655, “Designer’s Input for the Training of the Human Factors Engineering Verification and Validation Personnel,” Revision 1, August 1996.

31. WCAP-14690, “Designer’s Input to Procedure Development for the AP600,” Revision 1, June 1997.

[*32. NUREG-1342, “A Status Report Regarding Industry Implementation of Safety Parameter Display Systems.”*]*

33. Not used.

34. Not used.

35. Not used.

36. Not used.

37. Not used.

38. Not used.

39. Not used.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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40. Not used.

41. Not used.

42. Not used.

43. Not used.

44. WCAP-14694, “Designer’s Input to Determination of the AP600 Main Control Room Staffing Level,” Revision 0, July 1996.

45. Not used.

[*46. WCAP-14396, “Man-in-the-Loop Test Plan Description,” Revision 3, November 2002.*]*

47. APP-GW-GLR-082, “Execution and Documentation of the Human System Interface Design Implementation Plan,” Westinghouse Electric Company LLC.

48. NUREG-0711, Revision 2, “Human Factors Engineering Program Review Model,” U.S. NRC, February 2004.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 18.8-1 (Sheet 1 of 2)
[*HUMAN PERFORMANCE ISSUES TO BE ADDRESSED BY THE HSI DESIGN*]*

Operator Activity: Detection and Monitoring	
Issue 1:	*Do the wall panel information system and the workstation summary and overview displays support the operator in maintaining an awareness of plant status and system availability without needing to search actively through the workstation displays?*
Issue 2:	*Does the wall panel information system support the operator in getting more detail about plant status and system availability by directed search of the workstation functional and physical displays?*
Issue 3:	*Do the HSI features support efficient navigation to locate specific information?*
Issue 4:	*Do the HSI features effectively support crew awareness of plant condition?*
Operator Activity: Interpretation and Planning	
Issue 5:	*Does the alarm system convey information in a way that enhances operator awareness and understanding of plant condition?*
Issue 6:	*Does the physical and functional organization of plant information on the workstation displays enhance diagnosis of plant condition and the planning/selection of recovery paths?*
Issue 7:	*Does the integration of alarms, wall panel information system, workstation, and procedures support the operator in responding to single-fault events?*
Issue 8:	*Does the integration of alarms, wall panel information system, workstation and procedures support the operator in interpretation and planning during multiple-fault events?*
Issue 9:	*Does the integration of alarms, wall panel information system, workstation and procedures support the crew in interpretation and planning during multiple-fault events?*
Issue 10:	*Does the integration of alarms, wall panel information system, workstation, and procedures support the crew in interpretation and planning during severe accidents?*

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Table 18.8-1 (Sheet 2 of 2)
[*HUMAN PERFORMANCE ISSUES TO BE ADDRESSED BY THE HSI DESIGN*]*

Operator Activity: Controlling Plant State
Issue 11: Do the HSI features support the operator in performing simple, operator-paced control tasks?
Issue 12: Do the HSI features support the operator in performing control tasks that require assessment of preconditions, side effects and post-conditions?
Issue 13: Do the HSI features support the operator in performing control tasks that require multiple procedures?
Issue 14: Do the HSI features support the operator in performing event paced control tasks?
Issue 15: Do the HSI members features support the operator in performing control tasks that require coordination among crew?

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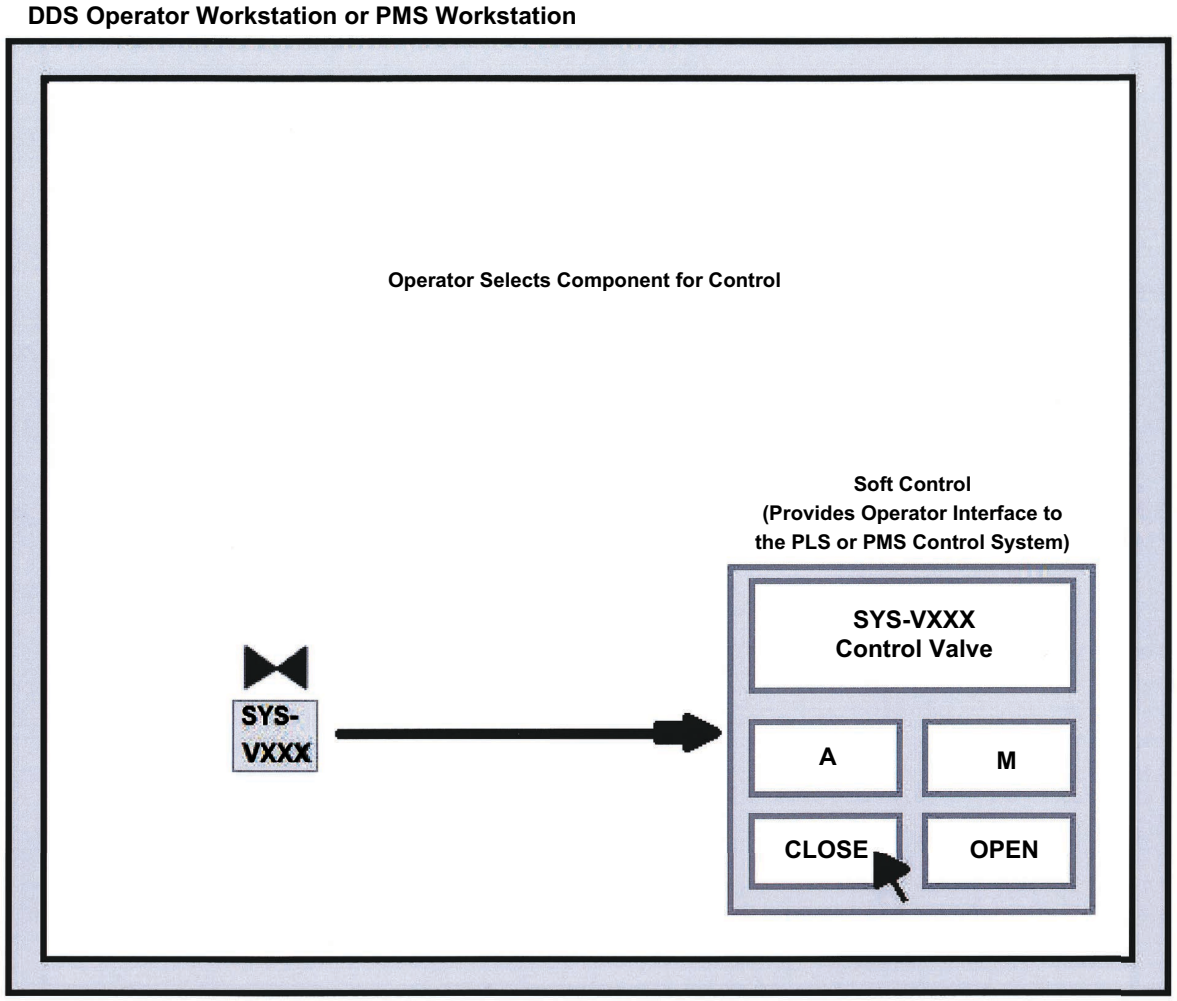

Figure 18.8-1
Soft Control Interactions

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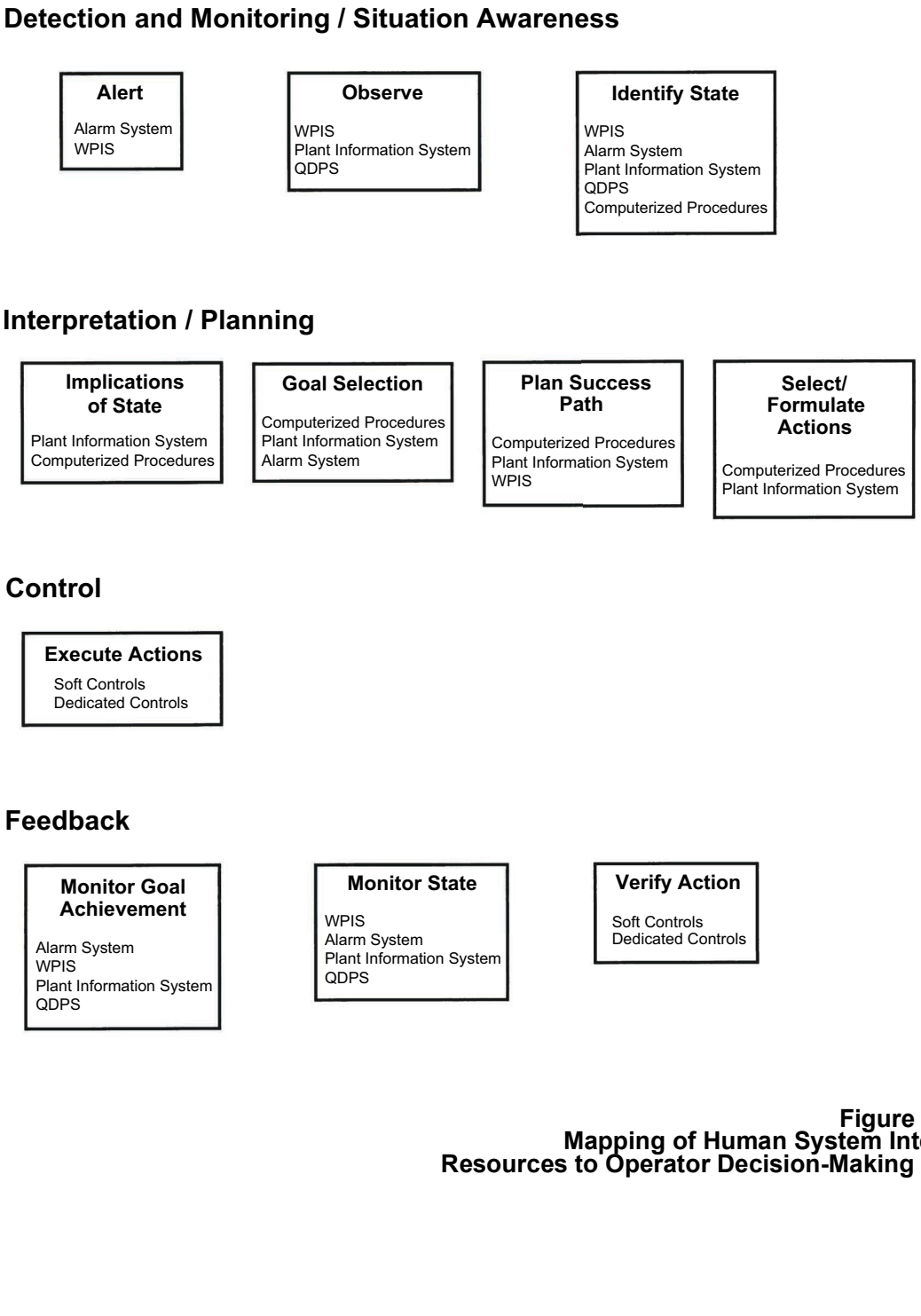

Figure 18.8-2
Mapping of Human System Interface Resources to Operator Decision-Making Model

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18.9 Procedure Development

WCAP-14690, "Designer's Input to Procedure Development for the AP600" (Reference 1), provides input for the development of plant operating procedures, including information on the development and design of the AP600 emergency response guidelines and emergency operating procedures. It applies directly to AP1000 since AP1000 is operated in the same manner as AP600. The WCAP also includes information on the computerized procedure system, which is the human system interface through which operators execute the plant procedures.

18.9.1 Combined License Information

The responsibility for procedure development is addressed in APP-GW-GLR-040 (Reference 2), and the applicable changes are incorporated.

18.9.2 References

1. WCAP-14690, "Designer's Input to Procedure Development for the AP600," Revision 1, June 1997.

2. APP-GW-GLR-040, "Plant Operations, Surveillance, and Maintenance Procedures," Westinghouse Electric Company LLC.

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18.10 Training Program Development

WCAP-14655, "Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel" (Reference 1), describes the design and implementation of the training program for the training of the operations personnel who participate as subjects in the Human Factors Engineering (HFE) Verification and Validation. The WCAP also describes the process used to develop the specification of the role of the operator for AP1000 and how this role and training insights can be passed from the designer to the developer of the training program.

Information regarding training program development is located in Section 13.2, Training.

18.10.1 Combined License Information

See Section 13.2 for a discussion of the responsibility for training program development.

18.10.2 References

1. WCAP-14655, "Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel," Revision 1, August 1996.

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18.11 Human Factors Engineering Verification and Validation

A programmatic level description of the AP1000 human factors engineering verification and validation program is provided by Reference 1. Using the programmatic level description, the development of an implementation plan for the AP1000 human factors engineering verification and validation is executed and documented as discussed in Reference 2. The implementation of the verification and validation activities is detailed in the five documents References 3 to 7.

The verification and validation activities are in accordance with Reference 1. There are a number of exceptions in respect to human factors engineering verification and validation. The details of these exceptions and the corresponding justifications are provided in References 3 to 5.

18.11.1 Combined License Information

The development, execution and documentation of an implementation plan for the verification and validation of the AP1000 human factors engineering program is addressed in Reference 2 (APP-GW-GLR-084).

18.11.2 References

[1. *WCAP-15860 , "Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan," Revision 2, October 2003.*]*

2. APP-GW-GLR-084, "AP1000 Human Factors Engineering Verification and Validation," Westinghouse Electric Company LLC.

[3. *APP-OCS-GEH-120, "AP1000 Human Factors Engineering Design Verification Plan,"*]* Revision 3, December 2014, Westinghouse Electric Company LLC.(1)

[4. *APP-OCS-GEH-220, "AP1000 Human Factors Engineering Task Support Verification Plan,"*]* Revision 6, February 2018, Westinghouse Electric Company LLC.(2)

[5. *APP-OCS-GEH-320, "AP1000 Human Factors Engineering Integrated System Validation Plan,"*]* Revision 7, February 2018, Westinghouse Electric Company LLC(3) [*(as modified by changes provided in Appendix 18A)*]*

[6. *APP-OCS-GEH-420, "AP1000 Human Factors Engineering Discrepancy Resolution Process,"*]* Revision 2, December 2014, Westinghouse Electric Company LLC.(4)

[7. *APP-OCS-GEH-520, "AP1000 Plant Startup Human Factors Engineering Design Verification Plan,"*]* Revision 4, December 2014, Westinghouse Electric Company LLC.(5)

Notes:

1. [*Section 1, Section 2, and Section 3 of APP-OCS-GEH-120 are Tier 2*. Changes to these portions of the document require prior NRC approval.*]* The remainder of the document, including its revision number, is Tier 2.

2. [*Section 1, Section 2, Section 3, Section 4, and Section 5 of APP-OCS-GEH-220 are Tier 2*. Changes to these portions of the document require prior NRC approval.*]* The remainder of the document, including its revision number, is Tier 2.

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3. [*Section 1, Section 2, Section 3, Section 4, Section 5, Section 6, and Section 7 of APP-OCS-GEH-320 are Tier 2*. Changes to these portions of the document require prior NRC approval.*]* The remainder of the document, including its revision number, is Tier 2.

4. [*Section 1 and Section 2 of APP-OCS-GEH-420 are Tier 2*. Changes to these portions of the document require prior NRC approval.*]* The remainder of the document, including its revision number, is Tier 2.

5. [*Section 1, Section 2, and Section 3 of APP-OCS-GEH-520 are Tier 2*. Changes to these portions of the document require prior NRC approval.*]* The remainder of the document, including its revision number, is Tier 2.

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Figure 18.11-1 Not Used
(See Reference 5, Figure 1.1-1, "AP1000 Verification and Validation Activities.")

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18.12 Inventory

18.12.1 Inventory of Displays, Alarms, and Controls

[*An inventory of instruments, alarms, and controls for the AP1000 systems is provided in the respective system piping and instrumentation diagrams and/or the respective system specification documents.*

The AP1000 system design engineers determine the specific sensors, instrumentation, controls, and alarms that are needed to operate the various plant systems. The instruments, alarms, and controls for each system are documented in the piping and instrumentation diagram and/or the respective system specification documents. An instrument, alarm, and control is specified by the system design engineer if it is needed to control, verify, or monitor the operation of the system and its components. System functions and their respective functional requirements are considered by the system designer when determining the need for a specific instrument, alarm, or control.

The role of the Human System Interface design team in the determination of the total inventory list is one of verification. As described in Section 18.5, human system interface design team has functionally decomposed the plant. The top four levels of this model for the AP1000 are shown in Figure 18.5-1. Each Level 4 function has a function-based task analysis (FBTA) performed as described in the Task Analysis Implementation Plan. Considering the plant operating modes and emergency operations, the function-based task analysis:

- *Identifies the functions goals*
- *Identifies the processes used to achieve each goal*
- *Documents the performance of a cognitive task analysis of each process*

The cognitive task analysis of each process answers the monitoring/feedback, planning, and controlling questions. The answers to these questions identify the data for each functional process (instrumentation, indications, alarms, and controls) needed by the operator to make decisions. The results of the cognitive task analysis phase of each function-based task analysis are used to verify the inventory list of instruments, controls, and alarms developed by the AP1000 system designers and documented in the respective design documents.]*

18.12.2 Minimum Inventory of Main Control Room Fixed Displays, Alarms, and Controls

Background

[The human system interface design includes the appropriate plant displays, alarms, and controls needed to support a broad range of expected power generation, shutdown, and accident mitigation operations. Soft control displays and plant information displays are generated by a computer and can be changed to perform different functions, allow control of different devices, or display different information. These displays appear on display devices such as cathode ray tubes, flat panel screens, or visual display units. Alarms are used to direct operator attention. Soft controls are provided through devices such as a keyboard, touch screen, mouse, or other equivalent input devices. The majority of the operations for both the AP1000 main control room and the remote shutdown workstation are expected to employ soft control displays and plant information displays.

The AP1000 human system interface design also includes a minimum inventory of dedicated or fixed-position displays and controls. The minimum inventory of AP1000 fixed-position instrumentation includes those displays, controls, and alarms that are used to monitor the status of critical safety functions and to manually actuate the safety-related systems that achieve these critical safety functions.

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Fixed-position alarms and displays are available at a fixed location and are continuously available, though not necessarily displayed, to the operator. Fixed-position displays can be accessed by the operator to monitor the plant status, based on indications from critical plant variables or parameters. Fixed-position alarms are designed to direct operator attention to the need to perform safety-related functions for which there is no automatic actuation function. Although not continuously displayed, the fixed-position displays and alarms are quickly and easily retrievable.

Fixed-position controls provide a means for manual reactor and turbine trip, and safety-related system/component actuation. Fixed-position controls are available to the operator to perform tasks in the operation of safety-related systems and components used to mitigate the consequences of an accident and to establish and maintain safe shutdown conditions following an accident. The fixed-position controls are a manual backup to the automatic protection signals provided by the protection and safety monitoring system.]*

Design Basis and Minimum Inventory

[*A systematic process was implemented to identify the minimum inventory of AP1000 fixed-position controls, displays, and alarms, using established selection criteria directly related to the specific AP1000 accident mitigation operator actions and the critical safety functions identified in the emergency response guidelines.*

The AP1000 design basis for accident mitigation protects the following three fission product barriers:

- *Fuel matrix/fuel rod cladding*
- *Reactor coolant system pressure boundary*
- *Containment*

Therefore, the minimum inventory of fixed instrumentation includes those displays, controls, and alarms used to monitor the status of these fission product barriers and manually actuate the safety-related systems that achieve the critical safety functions protecting these barriers.

Six critical safety functions are identified in the Emergency Response Guidelines (ERGs). These critical safety functions are physical processes, conditions, or actions designed to maintain the plant conditions within the acceptable design basis.

The AP1000 critical safety functions are:

- *Reactivity control*
- *Reactor core cooling*
- *Heat sink maintenance*
- *Reactor coolant system integrity*
- *Containment environment*
- *Reactor coolant system inventory control*

The minimum inventory of AP1000 fixed instrumentation includes those displays, controls, and alarms that are used to monitor the status of these critical safety functions and to manually actuate the safety-related systems that achieve these critical safety functions.]*

Minimum Inventory Selection Criteria

[*The following selection criteria are used to develop the minimum inventory of instrumentation controls, displays, and alarms:*

- *Regulatory Guide 1.97 Types A, B, and C, Category 1 instrumentation*

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- *Dedicated controls for manual safety-related system actuation (reactor trip, turbine trip, engineered safety feature actuation)*
- *Controls, displays, and alarms required to perform critical manual actions as identified from the PRA analysis*
- *Alarms provided for operator use in performing safety functions to respond to design basis events for which there is no automatically-actuated safety function*
- *Controls, displays, and alarms necessary to maintain the critical safety functions and safe shutdown conditions*

For the main control room, the minimum inventory of displays is provided by the PMS safety displays. For the remote shutdown workstation, the minimum inventory of displays is provided by the nonsafety-related displays of the plant information system.

An alarm is a device that provides warning by means of a signal or sound. The parameters and associated alarms, listed in DCD Table 18.12.2-1, identify challenges to the critical safety functions. This minimum inventory of alarms is embedded in displays as visual signals. For example, the visual signal may involve a change of color, brightness, flashing, or a combination of these. For clarity, these alarms are called visual alerts to distinguish them from other alarms which may include sound. For the main control, the visual alerts are embedded in the safety-related displays. For the remote shutdown workstation, the visual alerts are embedded in the nonsafety-related displays.

The minimum inventory resulting from the implementation of these selection criteria is provided in Table 18.12.2-1.]*

Regulatory Guide 1.97

[*The guidelines in Regulatory Guide 1.97 provide an effective basis for selection criteria to identify the minimum inventory of fixed displays, controls, and alarms, since these guidelines are consistent with monitoring the status of the fission product barriers and the associated critical safety functions in the AP1000 Emergency Response Guidelines.*

Regulatory Guide 1.97 provides a method to identify the post-accident monitoring (PAMS) instrumentation to monitor plant variables and systems during and following an accident. Selected post-accident monitoring instrumentation is required to remain functional over the range of the accident conditions and must be able to survive the accident environment for the length of time its function is required. The instrumentation helps the operator to identify the accident, to implement proper corrective actions, and to observe plant response to these actions in order to determine the need for additional actions. Five types of accident monitoring instrumentation and associated performance criteria are provided in the regulatory guide.

Within each type of post-accident monitoring instrumentation, there are three categories (Categories 1, 2, and 3) that are related to the qualification (seismic and environmental conditions) and reliability (safety-related power supply and single failures) of the specific instrumentation.

The Category 1 variables are considered as primary variables and meet appropriate qualification, design, and interface requirements discussed in subsection 7.5.2.2.1 and listed in Tables 7.5-2 and 7.5-3. These variables provide the appropriate capabilities and reliability that are required for the parameters. Only the Category 1 (primary) variables are included in the minimum inventory selection criteria. Category 2 and Category 3 instrumentation are not included in the selection criteria for the minimum inventory.

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Type A, Type B, and Type C are considered in developing the selection criteria for identification of the minimum inventory, since these three types are related to monitoring the three fission product barriers. The details of instrumentation designed to meet the guidelines in Regulatory Guide 1.97 are presented in Section 7.5.

Type A variables are defined in subsection 7.5.2.1. As discussed in subsection 7.5.3.1, Type A variables provide primary information to permit the main control room operating staff to:

- *Perform the diagnosis in the AP1000 emergency operating procedures*
- *Take specified preplanned, manually-controlled actions, for which automatic controls are not provided, and that are required for safety-related systems to accomplish their safety-related function to recover from a design basis accident*

There are no specific, preplanned, manually-controlled actions for safety-related systems to recover from design basis accidents in the AP1000 design. Therefore, as reflected in Table 7.5-4, there are no Type A variables.

Type B variables are defined in subsection 7.5.2.1. As discussed in subsection 7.5.3.2, Type B variables provide information to the main control room operating staff to assess the process of accomplishing critical safety functions in the emergency response guidelines. The Type B variables are identified in Table 7.5-5.

Type C variables are defined in subsection 7.5.2.1. As discussed in subsection 7.5.3.3, Type C variables provide the control room operating staff with information to monitor the potential for breach or the actual gross breach of:

- *Incore fuel cladding*
- *Reactor coolant system boundary*
- *Containment boundary*

The Type C variables are identified in Table 7.5-6.]*

Dedicated Controls

[*The selection criteria of AP1000 minimum inventory include dedicated, fixed-position controls that provide the capability to manually initiate system-level actuation signals for the safety-related systems and components that are used to achieve the critical safety functions. These dedicated controls provide the capability to initiate manual reactor and turbine trip, safeguards actuation, individual actuation of various safety-related, passive components and containment isolation.*]*

Probabilistic Risk Assessment Critical Human Actions

[*As described in Section 18.7 and Reference 1, the human factors engineering design process includes integration of PRA and the associated human reliability analysis insights into the AP1000 design. The human reliability analysis integration includes the identification of critical human actions through the consideration of specific deterministic and PRA criteria. These selection criteria for minimum inventory identify dedicated, fixed-position displays, alarms, and controls required to support critical human actions identified from the integration of human reliability analysis into the human factors engineering design process.*]*

Dedicated Alarms

[*As specified by Criterion 1, the minimum inventory of instrumentation requires dedicated instrumentation displays of the Regulatory Guide 1.97 Type A variables so that the operator can identify the need to take preplanned manually-controlled actions to mitigate the consequences of a*

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design basis accident, where a safety-related system needed to support a critical safety function is not automatically actuated.

The fourth criterion for minimum inventory is included to identify alarms needed to automatically alert the operator to the need to take these preplanned manually controlled actions.

One of the design goals of the AP1000 is to minimize the need for operator actions to mitigate the consequences of design basis events. As part of the implementation of this design goal, the safety-related systems required to mitigate the consequences of design basis accidents and most anticipated operational occurrences are automatically actuated. There are no specific preplanned, manually-controlled actions required for the safety-related systems to mitigate design basis accidents in the AP1000 design.

Another design goal for the AP1000 is to enhance defense in depth, which includes the use of automatically actuated safety-related systems as a backup to other automatically actuated safety-related systems. For example, if beyond-design-basis failures occurred such that the safety-related passive residual heat removal heat exchanger failed to actuate, other safety-related systems would automatically actuate to provide core cooling, without the need for operator action for either group of safety-related components. This design approach enhances overall plant safety.

The AP1000 minimum inventory includes a criterion for evaluating the need for dedicated alarms for preplanned operator actions. However, as a result of these two design approaches, the level of protection available to mitigate the consequences of an accident and to achieve the critical safety functions is provided without the need for preplanned operator actions for either the primary safety-related systems or the backup safety-related systems. Since there are no specific preplanned, manually-controlled actions for safety-related systems required to respond to design basis accidents in the AP1000 design, there are also no dedicated, fixed-position alarms identified in the minimum inventory list.]*

Critical Safety Functions and Safe Shutdown

[*The design basis for the AP1000 requires protecting the three fission product barriers in the plant (the fuel matrix and cladding, the reactor coolant system pressure boundary, and containment) following design basis events. The AP600 system/event matrix (Reference 2) identifies four safety-related, post-accident mitigation functions that are required as part of the design basis for the AP600 to protect the integrity of these fission product barriers. This document is directly applicable to AP1000. The design basis of the AP1000 requires safety-related systems that can perform these four safety-related functions for design basis events.*

The AP1000 Emergency Response Guidelines were developed by using the system/event matrix document as the plant response design basis and following the standardized process for Emergency Response Guideline development for Westinghouse PWRs. The design approach described in the system/event matrix document organizes the identified safety-related and nonsafety-related systems, structures and components into the appropriate groups that perform the four safety-related design basis functions. In developing the AP1000 Emergency Response Guidelines, the same groups of safety-related and nonsafety-related systems in the system/event matrix are used to perform their basic design functions, but they are organized somewhat differently from the system/event matrix to support development of symptom-based functional guidelines that can be more effectively used by the operators. These four design basis safety functions identified in (Reference 2) are expanded into the six critical safety functions in writing the symptom-based AP1000 Emergency Response Guidelines.

The six Emergency Response Guidelines critical safety functions (and the four design basis safety functions that the critical safety functions must satisfy) are physical processes, conditions, or actions taken using the safety-related and nonsafety-related systems to maintain the plant conditions within

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the acceptable design basis. These systems provide the physical equipment used to initiate and control the processes that achieve the critical safety functions.

By accomplishing the emergency response guideline critical safety functions following a design basis event, the plant is able to mitigate the consequences of the event and to establish and maintain safe shutdown conditions. The minimum inventory list identifies sufficient controls, displays, and alarms to monitor and control operation of the safety-related systems to achieve the six critical safety functions identified in the Emergency Response Guidelines and to establish and maintain safe shutdown conditions following an accident.

Tables 7.5-4, 7.5-5, and 7.5-6 identify the instrumentation and the associated Emergency Response Guidelines critical safety functions that each instrument supports for each of the Type A, B, and C post-accident instrumentation, respectively.]*

Minimum Inventory Selection Criteria Implementation Process

[*Section 7.5 provides a discussion of the development of the requirements of Regulatory Guide 1.97 and the implementation process for the AP1000 (Criteria 1, 2, and 4).*

Section 18.7 and Reference 1 provide a discussion of the implementation process for identification of critical PRA operator actions (Criteria 3). Chapter 30 of the AP1000 PRA describes the process for the human reliability analysis.]*

18.12.3 Remote Shutdown Workstation Displays, Alarms, and Controls

[*Subsection 7.4.3 discusses safe shutdown using the remote shutdown workstation following an evacuation of the main control room.*

The main control room provides the capability to perform accident mitigation and safe shutdown tasks for design basis events. The only types of events that would require evacuation of the main control room and control from the remote shutdown workstation are localized emergencies where the main control room environment is unsuitable for the operators or where the main control room workstations and equipment become damaged.

Evacuation of the main control room is not expected to occur coincident with any other design basis events. Subsection 9.5.1 of the Standard Review Plan (NUREG-0800) specifically excludes consideration of other design basis events coincident with a fire.

The design capability for the remote shutdown workstation is to provide the capability to establish and maintain safe shutdown conditions following a main control room evacuation, as described in subsection 7.4.3.1.1. The controls, displays, and alarms listed in Table 18.12.2-1 are retrievable from the remote shutdown workstation.]*

18.12.4 Combined License Information

This section contained no requirement for additional information.

18.12.5 References

[*1. WCAP-14651, "Integration of Human Reliability Analysis With Human Factors Engineering Design Implementation Plan," Revision 2, May 1997.*]*

2. WCAP-13793, "The AP600 System/Event Matrix," June 1994.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.6.

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Table 18.12.2-1 (Sheet 1 of 2)
Minimum Inventory of
Fixed Position Controls, Displays, and Alerts

Description	Control	Display	Alert[2]
Neutron flux		x	x
Neutron flux doubling[3]			x
Startup rate		x	x
RCS pressure		x	x
Wide range T_{hot}		x	
Wide range T_{cold}		x	x
RCS cooldown rate compared to the limit based on RCS pressure		x	x
Wide range T_{cold} compared to the limit based on RCS pressure		x	x
Change of RCS temperature by more than 5°F in the last 10 minutes			x
Containment water level		x	x
Containment pressure		x	x
Pressurizer water level		x	x
Pressurizer water level trend		x	
Pressurizer reference leg temperature		x	
Reactor vessel - Hot leg water level		x	x
Pressurizer pressure		x	
Core exit temperature		x	x
RCS subcooling		x	x
RCS cold overpressure limit		x	x
IRWST water level		x	x
PRHR flow		x	x
PRHR HX outlet temperature		x	x
PRHR HX inlet isolation and control valve status		x	x
PCS storage tank water level		x	
PCS cooling flow		x	
IRWST to RNS suction valve status[3]		x	x
Remotely operated containment isolation valve status[3]		x	
Containment area high range radiation level		x	x
Containment pressure (extended range)		x	
CMT level[1]		x	

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Table 18.12.2-1 (Sheet 2 of 2)
Minimum Inventory of
Fixed Position Controls, Displays, and Alerts

Description	Control	Display	Alert(2)
Manual reactor trip (Also initiates turbine trip Figure 7.2-1, sheet 14.)	x		
Manual safeguards actuation	x		
Manual CMT actuation	x		
Manual main control room emergency habitability system actuation(4)	x		
Manual ADS actuation (1-3 and 4)	x		
Manual PRHR actuation	x		
Manual containment cooling actuation	x		
Manual IRWST injection actuation	x		
Manual containment recirculation actuation	x		
Manual containment isolation	x		
Manual main steam line isolation	x		
Manual feedwater isolation	x		
Manual containment hydrogen igniter (nonsafety-related)(5)	x		
Manual ADS and IRWST injection unblock(6)	x		
Manual containment vacuum relief	x		

Notes:
1. Although this parameter does not satisfy any of the selection criteria of Subsection 18.12.2, its importance to manual actuation of ADS justifies its placement on this list.
2. These parameters are used to generate visual alerts that identify challenges to the critical safety functions. For the main control room, the visual alerts are embedded in the safety-related displays as visual signals. For the remote shutdown workstation, the visual alerts are embedded in the nonsafety-related displays as visual signals.
3. These instruments are not required after 24 hours. (Subsection 7.5.4 includes more information on the class 1E valve position indication signals, specified as part of the post-accident monitoring instrumentation.)
4. This manual actuation capability is not needed at the remote shutdown workstation.
5. At the remote shutdown workstation, containment hydrogen igniter control is provided as a “soft” control.
6. These controls are not required at the remote shutdown workstation.

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18.13 Design Implementation

This process element is added by Reference 2 to the Program Review Model specified in Reference 1. However, it mostly applies to plant modernization. The portions of the added element that apply to new plants were formerly addressed under the Verification and Validation element in Reference 1. Since these aspects of the Program Review Model are unchanged, AP1000 will continue to address them under Section 18.11 as “Issue Resolution Verification” and “Final Plant HFE Verification.”

18.13.1 References

1. NUREG-0711, “Human Factors Engineering Program Review Model,” U.S. NRC, July 1994.

2. NUREG-0711, Rev. 1, “Human Factors Engineering Program Review Model,” U.S. NRC, May 2002.

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18.14 Human Performance Monitoring

Human performance monitoring applies after the plant is placed in operation. The human performance monitoring process implements the guidance and methods as described in Reference 1.

The human performance monitoring process provides reasonable assurance that:

- The design can be effectively used by personnel, including within the control room and between the control room and local control stations and support centers.
- Changes made to the human system interface(s), procedures, and training do not have adverse effects on personnel performance, (e.g., a change does not interfere with previously trained skills).
- Human actions can be accomplished within time and performance criteria.
- The acceptable level of performance established during the design integrated system validation is maintained.

The human performance monitoring process is structured such that:

- Human actions are monitored commensurate with their safety importance.
- Feedback of information and corrective actions are accomplished in a timely manner.
- Degradation in performance can be detected and corrected before plant safety is compromised (e.g., by use of the plant simulator during training exercises).

The human performance monitoring process for risk-informed changes is integrated into the corrective action program, training program and other programs as appropriate. Identified human performance conditions/issues are evaluated for human factors engineering applicability.

Human factors engineering conditions are assigned specific human factors cause determination codes, trended for indications of degraded performance or potential human performance failures and have specific corrective actions identified.

The cause investigation:

- Identifies the cause of the failure or degraded performance to the extent that corrective action can be taken consistent with the corrective action program requirements.
- Addresses failure significance which includes the circumstances surrounding the failure or degraded performance, the characteristics of the failure, and whether the failure is isolated or has generic or common cause implications.
- Identifies and establishes corrective actions necessary to preclude the recurrence of unacceptable failures or degraded performance in the case of a significant condition adverse to quality.

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When appropriate, design changes are integrated into training exercises to monitor for degradation in performance and allow for early detection and corrective actions before plant safety is challenged (e.g., by use of the plant simulator during training exercises).

Plant or personnel performance under actual design conditions may not be readily measurable. When actual conditions cannot be simulated, monitored, or measured, the available information that most closely approximates performance data in actual conditions should be used.

Monitoring strategies for human performance trending after the implementation of design changes is capable of demonstrating that performance is consistent with that assumed in the various analyses conducted to justify the change.

Risk-informed changes are screened commensurate with their safety importance to determine if the change requires monitoring of actions. For changes which require monitoring, the appropriate monitoring requirements are determined and implemented in the training program or other program as appropriate.

18.14.1 References

1. NUREG-0711, Rev. 1, “Human Factors Engineering Program Review Model,” U.S. NRC, May 2002.

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Appendix 18A Not Used

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Chapter 19 Probabilistic Risk Assessment

19.1 Introduction

Chapter 19 presents the Problematic Risk Assessment (PRA) completed for the Reference Plant.

This section presents an overview of the Probabilistic Risk Assessment (PRA) performed for the Reference Plant. The PRA models developed for the Reference Plant are built upon the generic AP1000® plant PRA model and have been upgraded to the ASME/ANS NRC-endorsed standard (Reference 1) including site-specific information and expanded to evaluate the site seismic hazard. The Reference Plant PRA models are peer reviewed in accordance with current industry practices, as endorsed by the NRC. Results of the PRA models summarized in this chapter show that the Reference Plant peer reviewed PRA models and results meet the NRC PRA safety goals and the associated regulatory guidance, i.e., as addressed for Regulatory Guide 1.200 (Reference 2) in Appendix 1A. The site-specific detailed risk analyses present the associated core damage frequency (CDF) and large early release frequency (LERF).

The purpose of the AP1000 PRA was to provide inputs to the optimization of the AP1000 design and to verify that the US NRC PRA safety goals were satisfied as part of the design certification process. The purpose of the PRA performed for the Reference Plant is to assess site-specific risk.

The process of upgrading the generic AP1000 plant PRA to a site-specific PRA included reviewing and evaluating changes to the design, procedures and programs of the Reference Plant for impact to the PRA. This process captures the differences between the generic AP1000 plant and the Reference Plant.

The upgraded Chapter 19 Section 19.1 through Section 19.10 reflect the Reference Plant site-specific PRA. The licensing basis cross-references throughout the Current Licensing Basis excluding these sections continue to refer to the generic AP1000 plant PRA and remain acceptable as the site-specific PRA does not invalidate or change the information contained in these sections. The site-specific PRA is consistent with the described information.

The purpose of this chapter is to focus on the scope and results for the reference plant PRA models for:

- Internal Events (Section 19.2)
- Internal Flood (Section 19.3)
- Internal Fire (Section 19.4)
- Seismic PRA (Section 19.5)
- Other External Hazards (Screening) (Section 19.6)

Sections 19.7 and 19.8 are not used. Section 19.9 discusses the PRA insights, the revised results for internal events, internal flood, internal fire, seismic and other external hazards, and compares the revised results to the Commission's goals for CDF and LERF.

Section 19.10 lists overall references used as part of the standards and guidance used for the Reference Plant site-specific PRA. The standard review plan (Reference 17) was used as the overall justification for References 1 and Reference 2. Section 19.10 also lists a set of industry accepted guidance documents from the NRC, EPRI and ANS which provide the standard of practice for RG 1.200 (Reference 2) recommended peer reviews.

Section 19.11 contains combined license information that has been retained from the generic AP1000 plant PRA. Sections 19.12, 19.13, 19.14, 19.15, 19.16, 19.17, and 19.18 contain severe accident information that has been retained from the generic AP1000 plant PRA. Section 19.19

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contains shutdown evaluation information that has been retained from the generic AP1000 plant PRA. These sections are periodically updated pursuant to maintenance regulations. Section 19.20 is retained in Chapter 19 as historical seismic margins analysis information and is not maintained.

19.1.1 Scope

The PRA analysis for the Reference Plant consists of full-scope Level 1 and Level 2 “at-power” PRA. Detailed PRA models were developed for internal events, internal floods, internal fire, and seismic hazards. Other external hazards were screened from detailed evaluation in accordance with the qualitative and quantitative screening requirements of Part 6 of Reference 1. An expanded discussion of the external hazard screening is presented in Section 19.6.

The Level 1 analysis includes:

- Internal initiating events (IEs) evaluation
- Event tree and success criteria analyses
- Plant systems analysis using fault tree models
- Common cause failure and human reliability analyses
- Data analysis
- Quantification to calculate CDF

Level 1 analyses were applied to plant internal events and the following hazards:

- Internal fire
- Internal flooding
- Seismic hazard

Level 1 analyses have been performed in accordance with the ASME/ANS PRA Standard (Reference 1). Reference 1 provides the specific requirements for performing PRAs used to support risk-informed decisions for commercial light water reactor nuclear power plants.

The Level 2 analysis includes:

- An evaluation of severe accident phenomena and fission product source terms
- Modelling of the containment event tree and associated success criteria
- Analysis of hydrogen burning and mixing
- Quantification to calculate LERF

A containment event tree (CET) supported by plant system and phenomenological fault trees are used to calculate the plant large early release frequency and large release frequency.

Level 2 PRA analyses were performed in accordance with the ASME/ANS RA-S-1.2-2014 Trial Use Level 2 Standard (Reference 8). As with Reference 1, Reference 8 provides the specific guidance for performing Level 2 PRAs.

CDF and LERF analyses performed using the above standards have been peer reviewed consistent with the available guidance at the time. (References 3, 4, 5, and 6)

The generic AP1000 plant PRA included specific supplemental analyses addressing select risk-related shutdown issues. Shutdown risk assessments were limited to developing and integrating risk insights into deterministic plant operational strategies and were evaluated based on standard engineering practice.

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The generic AP1000 plant Low-Power Shutdown Evaluation has not been upgraded as part of the development of the Reference Plant Level 1 and Level 2 PRA. A defense-in-depth shutdown model was developed to qualitatively assess risk and is managed in accordance with 10 CFR 50.65(a)(4). That effort uses qualitative methods and is adequate to manage risk during shutdown conditions.

The generic AP1000 plant PRA included an off site dose evaluation. This off site dose quantification is bounding for the Reference Plant site (Reference 18). Accident off site doses at the site boundary that could exist in the first 24 hours after a severe accident are either less than 25 rem, or for those releases that are greater than 25 rem, have a frequency of much less than 1E-06.

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19.2 Internal Events

The internal events plant model includes: 1) Level 1 quantification of CDF that also supports quantification of Level 2 release categories, 2) use of appropriate models and computer codes accounting for methodology limitations, 3) identification and evaluation of dependencies related to multiple human failure events in the same cutsets, 4) detailed review of the quantification results for reasonableness and identification of significant contributors to CDF and LERF, 5) uncertainty and sensitivity analysis, and 6) detailed documentation of the quantification process and results.

The plant model integrates the accident sequences, system models, data and human reliability analysis (HRA) for each IE group to support evaluation of CDF for each accident sequence. The total mean CDF and LERF was determined using methodology that accounts for plant response in addition to failures in the evaluation of accident sequences, accounts for the state-of-knowledge correlation, and provides detailed importance measures.

Table 19.2-1 provides a list of the front line and support systems addressed within the internal events PRA.

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Table 19.2-1 Summary of Systems in the PRA

System in the PRA	Acronym (System - Function)
Passive Core Cooling System - Passive Residual Heat Removal	PXS- PRHR
Passive Core Cooling System - Core Makeup Tanks	PXS - CMT
Passive Core Cooling System - Accumulator	PXS - ACC
Passive Core Cooling System - Automatic Depressurization System	RCS - ADS
Passive Core Cooling System - In-containment Refueling Water Storage Tank	PXS - IRWST
Passive Containment Cooling	PCS
Main and Startup Feedwater System	FWS
Main Steam System	MSS
Condensate System	CDS
Condenser Air Removal System	CMS
Circulating Water System	CWS
Steam Generator System	SGS
Chemical and Volume Control System	CVS
Demineralized Water Treatment System	DTS
Demineralized Water Transfer and Storage System	DWS
Containment Hydrogen Control System	VLS
Normal Residual Heat Removal System	RHR
Component Cooling Water System	CCS
Service Water System	SWS
Central Chilled Water System	VWS
Air-Conditioning, Heating, Cooling, and Ventilation System	VAS, VBS, VES
Non-Class 1E AC Power System	ECS
Class 1E dc & UPS System	IDS
Non-Class 1E dc & UPS System	EDS
Standby Diesel Fuel Oil System	DOS
Containment Isolation	CNS
Compressed and Instrument Air System	CAS
Turbine Building Closed Cooling Water System	TCS
Protection and Safety Monitoring System	PMS
Diverse Actuation System	DAS
Plant Control System	PLS

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19.2.1 Internal Events Results

A quantitative internal events PRA of the Reference Plant design performed to evaluate plant CDF and LERF for at-power events provided the following results:

Table 19.2-2 Reference Plant At-Power Internal Events PRA Results

	Plant CDF	Plant LERF
Internal Events At-Power	3.94E-07/reactor-year	3.83E-08/reactor-year

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19.3 Internal Flood

An internal flooding PRA was performed based on Reference Plant site-specific information.

Fluid systems that are potential flood sources were identified for the internal flooding PRA. For these systems, the information on pipe length, pipe diameter and associated flood area was collected. The piping information obtained was used in addition to industry pipe rupture frequency data to quantify the initiating event frequencies for a specific flood source system in a specific room.

Flood source capacity along with door and penetration information were used to identify propagation pathways for each flood scenario to determine impacted PRA-related structures, systems, and components (SSCs).

Flood scenarios were screened and identified for further evaluation. Each of the flood scenarios were characterized by the flood area, flood source, failure mechanism, potential failure modes of the flood source, potential impact on components in the flood area and areas along the propagation path and plant impact. Four distinct failure modes (spray, flood, major flood, high-energy line break (HELB)) were considered in characterizing flood scenarios.

19.3.1 Internal Flood Results

A quantitative internal flooding PRA of the Reference Plant design performed to evaluate plant CDF and LERF for at-power events provided the following results:

Table 19.3-1 Reference Plant At-Power Internal Flooding PRA Results

	Plant CDF	Plant LERF
Internal Flooding During At-Power Events	2.17E-07/reactor-year	8.40E-08/reactor-year

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19.4 Internal Fire

The at-power internal fire PRA was developed following NUREG/CR-6850 and supplemental methodology as endorsed by the NRC. The internal fire PRA analysis was based on the Level 1 and Level 2 release categories from the internal events modified to account for fire specific failures and multiple spurious actuations. The plant fire PRA model includes: 1) Plant Boundary Definition and Partitioning, 2) Component Selection, 3) Cable selection, 4) Qualitative Screening, 5) Fire Risk Model, 6) Ignition Frequency Development, 7) Quantitative Screening, 8) Scoping Fire Modeling, 9) Detailed Circuit Failure Analysis, 10) Circuit Failure Likelihood Analysis, 11) Detailed Fire Modeling, 12) Fire Human Reliability Analysis, 13) Fire Risk Quantification, 14) Uncertainty and Sensitivity Analysis, and 15) detailed documentation of the quantification process and results.

19.4.1 Internal Fire Results

A quantitative internal fire PRA of the Reference Plant design performed to evaluate plant CDF and LERF for at-power events provided the following results:

Table 19.4-1 Reference Plant At-Power Internal Fire PRA Results

	Plant CDF	Plant LERF
Internal Fire During At-Power Events	8.54E-07/reactor-year	3.42E-07/reactor-year

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19.5 Seismic PRA

The Reference Plant Units Seismic PRA (SPRA) follows industry recognized guidance such as References 7, 9, 10, and 11 to estimate the frequencies of core damage and Level 2 release categories. The SPRA assembles the results of the seismic hazard analysis, fragility analysis, and the seismic accident sequence model into a large fault tree including seismic initiating events (ground motion intervals) with separate top events for CDF and Level 2 release categories. The quantification is a convolution of the seismic hazard fragility for each screened in SSCs including contributions of random hardware and human failures to contribute to the CDF and Level 2 results. The convolution is approximated by FRANX software (Reference 12) and the SPRA in general as follows:

- **Hazard Treatment**: Each of the 16 identified intervals is assigned a representative peak ground acceleration (PGA) (geometric mean between the interval lower and upper bounds) and a corresponding occurrence frequency (calculated as the interval lower bound exceedance frequency minus the upper bound exceedance frequency).

- **Fragility Treatment**: The fragility for each SSC is used to calculate the failure probability for each interval (i.e., failure probability at the representative PGA assigned to each interval).

- **Model Logic**: For each interval, the model logic ANDs the seismic initiating event (i.e., the hazard) with the fragility. Solution of the fault tree generates minimal cut sets each containing a seismic initiating event AND fragility event(s). The result is an approximate solution to the hazard convolved with the fragilities.

19.5.1 Seismic Fragility

The seismic fragility evaluation estimates the conditional probability of failure of each screened in SSC. Fragilities for SPRA are expressed in terms of PGA using a double lognormal model with three parameters: median acceleration capacity (A_m), logarithmic standard deviation of the aleatory (randomness) uncertainty (β_R), and logarithmic standard deviation of the epistemic (modeling and data) uncertainty (β_U). β_R and β_U can be combined into a composite variability (β_C) referred to as mean fragility. A high confidence, low probability of failure (HCLPF) capacity is the ground motion level at which there is 95% confidence of at most 5% probability of failure. The fragility parameters are used to develop a complete family of fragility curves for each screened in SSC.

This analysis has been developed in accordance with industry guidance provided in Electric Power Research Institute (EPRI) 3002012994 (Reference 7). The Conservative Deterministic Failure Margins (CDFM) methodology was used as a first step. For components that were risk significant, the fragilities were revised using a Separation of Variables (SOV) approach.

Most fragilities were developed based on the Hybrid approach which uses the CDFM method to generate HCLPF values. The HCLPF capacity is computed, and conservatively biased generic random variability β_R and uncertainty β_U logarithmic standard deviation values are used to estimate the median acceleration capacity, A_m, value. Significant top contributors and SSCs that require a more detailed/refined fragility evaluation are identified through several sets of quantifications and sensitivity studies. The SOV method, which estimates a realistic median acceleration capacity, is employed and corresponding logarithmic standard deviations β_R and β_U for each variable affecting the response and capacity is evaluated.

Fragility estimates for risk significant contributors are calculated realistically, while conservatism is acceptable for fragility estimates associated with components that are not important risk contributors.

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The degree of realism (and associated effort) is commensurate with the importance of the fragility estimates in the risk profile. The PRA logic is used through several sets of quantifications and sensitivity studies to identify fragilities that should be further refined. After each quantification and completion of the sensitivity study, important risk contributors were identified, and the fragilities were refined to understand seismic risk insights and identify those components with the highest risk contribution.

19.5.2 Seismic PRA Results

A quantitative Seismic PRA of the Reference Plant design performed to estimate plant CDF and LERF for at-power events provided the following results:

Table 19.5-1 Reference Plant At-Power Seismic PRA Results

	Plant CDF	Plant LERF
Seismic Events At-Power	6.89E-08/reactor- year	2.83E-08/reactor- year

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19.6 Other External Hazards

A qualitative External Hazards Analysis was performed based on the Reference Plant site-specific information.

Potential external hazards were identified and evaluated for the Reference Plant site to determine if any hazards should be further evaluated in a detailed PRA. The potential hazards evaluation concluded that solar flares and solar storms and electromagnetic interference should be added to the Reference Plant site list of potential hazards. In addition to the list of external hazards requiring consideration in the PRA Standard (Reference 1), several other site-specific sources (including industry guidance documents and Reference Plant site records) were consulted to ensure the list of hazards considered is complete. The requirements of the ASME/ANS RA-Sa Addendum to PRA Standard (Reference 1) were applied throughout this assessment.

Qualitative screening of external hazards was performed in two (2) parts based on the criteria outlined in SR EXT-B1 and SR EXT-C1 of the PRA Standard (Reference 1). SR EXT-B1 provides a preliminary screen that resulted in eliminating most hazards. Hazards that could not be eliminated based on the criteria of SR EXT-B1 were evaluated against the criteria of SR EXT-C1. SR EXT-C1 screening criteria requires an acceptable basis for bounding analysis or demonstrably conservative analysis for the hazard exists.

The screening results for each hazard considered in the External Hazards analysis are provided in Table 19.6-1.

Table 19.6-1 Summary of External Hazards Screening

ID	External Hazard	Screening Conclusion	
		SR EXT-B1(1) Criteria Applied	SR EXT-C1(1) Criteria Applied
1	Aircraft Impacts	N/A	C
2	Avalanche	3	N/A
3	Biological Events	1, 5	N/A
4	Coastal Erosion	1, 3	N/A
5	Drought	1, 5	N/A
6	External Flooding	1	A
7	Electromagnetic Interference	1	N/A
8	Extreme Winds and Tornadoes	N/A	B, C
9	Fog	1, 4	N/A
10	Forest Fire	1, 3	N/A
11	Frost	1, 4	N/A
12	Hail	1, 4	N/A

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Table 19.6-1 Summary of External Hazards Screening (Continued)

ID	External Hazard	Screening Conclusion	
		SR EXT-B1[(1)] Criteria Applied	SR EXT-C1[(1)] Criteria Applied
13	High Summer Temperature	1, 5	N/A
14	High Tide	3	N/A
15	Hurricane	4	N/A
16	Ice Cover	1	N/A
17	Industrial or Military Facility Accident	1	A
18	Internal Flooding	Not Screenable - Full Scope at-power Internal Flooding PRA Performed to Part 3 of the ASME PRA Standard. See Section 19.3	
19	Landslide	3	N/A
20	Lightning	1, 4	N/A
21	Low Lake or River Water Level	1, 5	N/A
22	Low Winter Temperature	1, 5	N/A
23	Meteorite/Satellite Strikes	N/A	C
24	Pipeline Accident	3	N/A
25	Precipitation, Intense	4	N/A
26	Release of Chemicals from On-Site Storage	1	A
27	River Diversion	1	N/A
28	Sandstorm	3	N/A
29	Seiche	3	N/A
30	Seismic Activity	Not Screenable - Full Scope at-power Seismic PRA Performed to Part 5 of the ASME PRA Standard. See Section 19.4	
31	Snow	N/A	C
32	Soil Shrink-Swell	1	N/A
33	Solar Flares or Solar Storms	N/A	C
34	Storm Surge	4	N/A

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Table 19.6-1 Summary of External Hazards Screening (Continued)

ID	External Hazard	Screening Conclusion	
		SR EXT-B1(1) Criteria Applied	SR EXT-C1(1) Criteria Applied
35	Toxic Gas	1	N/A
36	Transportation Accidents	3	A
37	Tsunami	3	N/A
38	Turbine-Generated Missiles	N/A	B, C
39	Volcanic Activity	3	N/A
40	Waves	4	N/A

Notes:
1. Screening criteria SR EXT-B1 and SR EXT-C1 are presented in ASME/ANS-RA-Sa 2009 (Reference 1)

19.6.1 Other External Hazards PRA Results

A quantitative other external hazards PRA of the Reference Plant design was not performed to evaluate plant CDF and LERF for at-power events. Application of the screening criteria SR EXT-B1 and SR EXT-C1 demonstrate that the events are either not significant enough to impact the plant or have an acceptable basis for bounding analysis or demonstrably conservative analysis.

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19.7 Not Used

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19.8 Not Used

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19.9 Conclusions

19.9.1 PRA Insights

PRA insights refer to the results and findings of the PRA. A synopsis of the insights gained from the PRA about the Reference Plant design includes:

- The Reference Plant design benefits from the high level of redundancy and diversity of the passive safety-related systems. The passive systems have been shown to be reliable; their designs are simple so that a limited number of components are required to function.

- The Reference Plant are less dependent on non-safety-related systems than current plants licensed under 10 CFR Part 50 regulations or advanced light water reactor evolutionary plants.

- The nonsafety-related support systems (AC power, component cooling water, service water, and instrument air) have a limited role in the plant risk profile because the passive safety-related systems do not require cooling water or AC power.

- The Reference Plant are less dependent on human actions than current plants licensed under 10 CFR Part 50 regulations or advanced light water reactor evolutionary plants.

- Single system or component failures are not overly important due to the redundancy and diversity of safety-related systems in the design. For example, the following lines of defense are available for reactor coolant system (RCS) makeup:

  - CVS

  - CMTs with power locked-out from the open CMT balance line inlet isolation MOVs which prevents inadvertent closure of the inlet isolation valves and loss of CMT due to fire-induced hot-short

  - Partial automatic depressurization system in combination with normal residual heat removal

  - Full automatic depressurization system with accumulators and in-containment refuelling water storage tank

  - Full automatic depressurization system with core makeup tanks and in-containment refuelling water storage tank

- Typical current PRA dominant IEs are significantly less important for the Reference Plant. For example, the RCP seal LOCA event has been eliminated as a core damage initiator since the Reference Plant uses sealless reactor coolant pumps. Another example is the LOOP event. The station blackout and loss of offsite power event is a minor contributor compared to legacy plants since the Reference Plant passive safety-related systems do not require the support of AC power for 72 hours following loss of offsite power.

- Passive safety-related systems are available in shutdown modes. Planned maintenance of passive features is only performed during shutdown modes when that feature is not risk important. In addition, planned maintenance of non-safety-related defense-in-depth features used during shutdown is performed at power when these systems are less risk important.

- The Reference Plant design includes features to contribute to the low likelihood of interfacing system LOCAs between the RNS and the RCS.

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- A relief valve located in the common RNS discharge line outside containment provides protections against excess pressure.

- Two remotely operated MOVs connecting the suction and discharge headers to the IRWST are interlocked with the isolation valves connecting the RNS pumps to the hot leg. This prevents inadvertent opening of the two MOVs when the RNS is aligned for shutdown cooling and potential diversion and draining of reactor coolant system.

- Power to the four isolations MOVs connecting the RNS pumps to the RCS hot leg is administratively blocked at their motor control centers during normal power operation.

- The potential for containment isolation and containment bypass is lessened by having fewer penetrations to allow fission product release. In addition, normally open and risk important penetrations are fail-closed, thus eliminating the dependence on instrumentation and controls and batteries for these penetrations.

- The Reference Plant Design includes features to enhance the likelihood of retaining the core within the reactor vessel for severe accident sequences. These features include:

  - The depressurization of the reactor coolant system below 150 psi facilitates in-vessel retention of molten core debris

  - Passive containment can withstand severe accidents without PCS water cooling the containment shell. Air cooling alone is sufficient to maintain containment pressure below failure pressure with high probability.

  - Reflooding the reactor pressure vessel through the break can have a significant effect on a severe accident by quenching core debris, achieving a controlled stable state, and producing hydrogen

  - The reactor vessel lower head has no vessel penetrations, thus eliminating penetration failure as a potential vessel failure mode. Preventing the relocation of molten core debris to the containment eliminates the occurrence of several severe accident phenomena, such as ex-vessel fuel-coolant interactions and core-concrete interaction, which may threaten the containment integrity. Therefore, the Reference Plant, through the prevention of core debris relocation to the containment, significantly reduces the likelihood of containment failure.

- The potential for the spreading of fires and floods to safety-related equipment is reduced by the Reference Plant layout.

- The design philosophy of minimizing the number of potential flooding sources in safety related areas, along with the physical separation of redundant safety-related components and systems from each other and from nonsafety-related components, minimizes the consequences of internal flooding.

19.9.2 Conclusions

The Reference Plant design has evolved over a period of years, including the work done for the AP600 design. PRA techniques have been used since the beginning in an iterative process to optimize the AP600/AP1000 Nuclear Power Plant (NPP) with respect to public safety. Each of these iterations has included:

- Development/refinement of the PRA model
- Use of the model to identify weaknesses

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- Quantification of PRA benefits of alternate designs and operational strategies
- Adoption of selected design and operational improvements.

The scope and detail of the PRA model has increased from the early studies as the plant design has matured. This iterative design process has resulted in a number of design and operational improvements. This Reference Plant PRA revision includes the revised results for the RG 1.200 models for internal events, internal flood, internal fire, seismic and other external hazards. The risk associated with the Reference Plant design compares favorably against the Commission's goals of less than 10E-4 per year (/yr) for CDF and less than 10E-5/yr for LERF. The total CDF and LERF for Reference Plant during at-power events provide the following results:

Table 19.9-1 Summary of Reference Plant At-Power PRA Results

	Plant CDF	Plant LERF
Internal Events	3.94E-07/reactor-year	3.83E-08/reactor-year
Internal Flood	2.17E-07/reactor-year	8.40E-08/reactor-year
Internal Fire	8.54E-07/reactor-year	3.42E-07/reactor-year
Seismic Events	6.89E-08/reactor-year	2.83E-08/reactor- year
Total During At-Power Events	1.53E-06/reactor-year	4.93E-07/reactor-year

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19.10 References

1. ASME/ANS-RA-Sa 2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” American Society of Mechanical Engineers, February 2009.

2. NRC Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities,” Revision 2, March 2009.

3. NEI 00-02, “Probabilistic Risk Assessment (PRA) Peer Review Process Guidance,” Revision A3, Nuclear Energy Institute, March 2000.

4. NEI 05-04, “Process for Performing Internal Events PRA Peer Reviews Using the ASME/ ANS PRA Standard,” Revision 2, Nuclear Energy Institute, November 2008.

5. NEI 07-12, “Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines,” Revision 0, Nuclear Energy Institute, November 2008.

6. NEI 12-13, “External Hazards PRA Peer Review Process Guidelines,” Revision 0, Nuclear Energy Institute, August 2012.

7. EPRI 3002012994, “Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessment,” Electric Power Research Institute, Palo Alto, CA: September 2018.

8. ASME/ANS RA-S-1.2-2014, “Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWR), Trial Use and Pilot Application”, American Society of Mechanical Engineers, 2014.

9. EPRI 3002000709, “Seismic Probabilistic Risk Assessment Implementation Guide,” Electric Power Research Institute, Palo Alto, CA: December 2013.

10. EPRI 3002008093, “An Approach to Human Reliability Analysis for External Events with a Focus on Seismic,” Electric Power Research Institute, Palo Alto, CA: December 2016.

11. EPRI 3002012980, “Methodology for Seismically Induced Internal Fire and Flood Probabilistic Risk Assessment,” Electric Power Research Institute, Palo Alto, CA: June 2018.

12. FRANX Software, EPRI, www.EPRI.com.

13. Not Used.

14. Not Used.

15. Not Used.

16. Not Used.

17. NRC Standard Review Plan 19.0 Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors, Revision 3, December 2015.

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18. APP-GW-GL-022, AP1000 Probabilistic Risk Assessment, Westinghouse Electric, LLC, Rev. 8, August 2004 [ML042230497].

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19.11 Combined License Information

A review of the differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis will be completed prior to fuel load. A verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any differences will be evaluated and the seismic margins analysis modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design. A comparison of the as-built SSC high confidence, low probability of failures (HCLPFs) to those assumed in the AP1000 seismic margin evaluation will be performed prior to fuel load. Deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis will be evaluated to determine if vulnerabilities have been introduced. The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include:

1. Specific minimum seismic requirements consistent with those used to define the Table 19.20-1 HCLPF values.

   This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The test response spectra are chosen so as to demonstrate that no more than one percent rate of failure is expected when the equipment is subjected to the applicable seismic margin ground motion for the equipment identified to be applicable in the seismic margin insights of the site-specific PRA.The range of frequency response that is required for the equipment with its structural support is defined.

2. Hardware enhancements that were determined in previous test programs and/or analysis programs will be implemented.

A review of the differences between the as-built plant and the design used as the basis for the AP1000 PRA will be completed prior to fuel load. The plant-specific PRA-based insight differences will be evaluated and the plant-specific PRA model modified as necessary to account for plant-specific design and any design changes or departures from the generic AP1000 plant PRA.

A review of the differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analyses will be completed prior to fuel load. Plant specific internal fire and internal flood analyses will be evaluated and the analyses modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design.

The AP1000 Severe Accident Management Guidance (SAMG) from APP-GW-GLR-070, Reference 19.11-1, is implemented on a site-specific basis. Key elements of the implementation include:

- SAMG based on APP-GW-GLR-070 is provided to Emergency Response Organization (ERO) personnel in assessing plant damage, planning and prioritizing response actions and implementing strategies that delineate actions inside and outside the control room.

- Severe accident management strategies and guidance are interfaced with the Emergency Operating Procedures (EOP's) and Emergency Plan.

- Responsibilities for authorizing and implementing accident management strategies are delineated as part of the Emergency Plan.

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- SAMG training is provided for ERO personnel commensurate with their responsibilities defined in the Emergency Plan.

A thermal lag assessment of the equipment required to mitigate severe accidents (hydrogen igniters and containment penetrations) will be performed to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. This assessment will be performed prior to fuel load and is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The ability of the equipment to perform during severe accident hydrogen burns will be assessed using the Environment Enveloping method or the Test Based Thermal Analysis method discussed in EPRI NP-4354 (Reference 19.11-2).

The Combined License applicant referencing the AP1000 certified design will confirm that the Seismic Margin Assessment analysis documented in Section 19.20 is applicable to the COL site. This will include a confirmation that the COL site seismic demand based on the site GMRS is enveloped by the Certified Seismic Design Response Spectra (CSDRS) seismic demand as defined by Tier 1 criteria for SSE as well as an assessment that no site specific effects such as seismically induced liquefaction settlements, slope stability, foundation failure, and relative displacements have the potential to lower the HCLPF values calculated for the certified design. Further evaluation will be required if the COL site is shown to be outside of the bounds of the SMA analysis documented in Section 19.20.

The Combined License holder will develop a probabilistic risk assessment in accordance with 10 CFR 50.71(h) prior to initial fuel load.

19.11.1 References

19.11-1. APP-GW-GLR-070, “Development of Severe Accident Management Guidance,” Westinghouse Electric Company LLC.

19.11-2. “Large Scale Hydrogen Burn Equipment Experiments,” EPRI-NP-4354, December 1985.

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19.12 Severe Accident Phenomena Treatment

19.12.1 Introduction

This section describes how the AP1000 containment addresses challenges from severe accident phenomena, and how the challenges are evaluated in the AP1000 probabilistic risk assessment (PRA). In the PRA, the Modular Accident Analysis Program (MAAP) version 4.04 code (Reference 19.12-8) is used to evaluate severe accident scenarios. Severe accident phenomenological uncertainties are treated with Risk-Oriented Accident Analysis Methodology (ROAAM) (Reference 19.12-2) phenomenological evaluations, with AP1000-specific decomposition event tree phenomenological evaluations, or with assumptions that certain low-frequency severe accident phenomena fail the containment. The objective of these studies is to show, with a high degree of confidence, that the AP1000 containment will accommodate the effects of severe accidents in a range of scenarios for at least the first 24 hours after the onset of core damage. Such evaluations demonstrate the robustness of the containment design.

19.12.2 Treatment of Physical Processes

The following eight issues are identified in Reference 19.12-1 as being representative of the phenomenological issues pertaining to severe accident conditions:

1. Loss-of-coolant accident (LOCA)
2. Fuel-coolant interaction (steam explosion)
3. Hydrogen combustion and detonation
4. Melt attack on concrete structure or containment pressure boundary
5. High-pressure melt ejection
6. Core-concrete interaction (CCI)
7. Containment pressurization from decay heat
8. Elevated temperature (equipment survivability)

The challenge to the containment integrity from a LOCA blowdown is covered in the containment design basis and is not specifically addressed here. Treatment of physical processes affecting the remaining challenges is discussed in this chapter.

19.12.2.1 In-Vessel Retention of Molten Core Debris

In-vessel retention (IVR) of core debris by external reactor vessel cooling is a severe accident mitigation attribute of the AP1000 design; it is discussed in detail in Section 19.16. With the reactor vessel intact and debris retained in the lower head, phenomena such as molten core-concrete interaction and ex-vessel steam explosion, which occur as a result of core debris relocation to the reactor cavity, are prevented.

The AP1000 reactor vessel insulation and containment geometry promote in-vessel retention. Engineered design features of the AP1000 containment flood the containment reactor cavity region during accidents, and thereby, submerge the reactor vessel in water.

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Chapter 39 of the AP1000 PRA presents an AP1000-specific evaluation to determine the likelihood that sufficient heat can be removed from the outside surface of the submerged reactor pressure vessel lower head to prevent reactor vessel failure and relocation of debris to containment. The methodology used to quantify the margin to vessel failure in Reference 19.12-2 for the AP600 was adapted to the AP1000. For the AP1000 the methodology assumes that:

- The RCS is depressurized.
- The reactor vessel is submerged above the 98-ft elevation in the containment.
- The reactor vessel insulation promotes the two-phase natural circulation in the reactor vessel cooling annulus.
- The reactor vessel external surface is bare metal.

The containment event tree includes a node to ascertain that the reactor coolant system (RCS) is depressurized and a node to determine if adequate water is available in the cavity to achieve two-phase natural circulation. Success at both of these nodes is required to demonstrate that the conditions and assumptions of the IVR analysis are met. The AP1000 design specifies that the reactor vessel insulation is designed appropriately and that the outer surface of the reactor vessel promotes wetability.

Accounting for the uncertainties in thermal-hydraulic parameters, the heat fluxes to the vessel wall and reactor vessel internals from the debris pool are calculated. The results show large margin to failure for the reactor vessel if it is externally cooled by water.

19.12.2.2 Fuel-Coolant Interaction (Steam Explosions)

A steam explosion may occur as a result of molten metal or oxide core debris mixing with water and interacting thermally. Steam explosions are postulated to occur inside the reactor vessel when debris relocates from the core region into the lower plenum and in the reactor cavity if the vessel fails and debris is ejected from it into water in the reactor cavity.

19.12.2.2.1 In-Vessel Fuel-Coolant Interaction

In-vessel steam explosions were studied extensively in the AP600 analyses. A ROAAM analysis of the AP600 reactor vessel lower head integrity under in-vessel steam explosion loading is presented in Reference 19.12-3. Typically, in-vessel steam explosion analyses focus on the α-mode containment failure, which is induced by the reactor vessel upper head failure. The ROAAM analysis focused on failure of the lower head since that steam explosion vessel failure mode would impair the in-vessel retention capability of the reactor vessel. The ROAAM analysis concludes that lower-head vessel failure from in-vessel steam explosion is physically unreasonable with very large margin to failure.

Based on the in-vessel core relocation scenario for the AP1000, the in-vessel steam explosion ROAAM analysis presented for the AP600 can be extended to the AP1000. The mass flow rate, superheat and composition of debris in the relocation from the upper core region to the lower head is expected to be essentially the same as the AP600. The geometry of the lower head of the AP1000 is the same as the AP600. Therefore, it is reasonable to extend the results of the AP600 in-vessel steam explosion ROAAM analysis to the AP1000.

The results of the in-vessel steam explosion ROAAM can also be extended to containment failure induced by in-vessel steam explosions (α-mode containment failure). The likelihood for vessel failure and subsequent containment failure due to in-vessel steam explosion is so small as to be negligible.

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This conclusion is in agreement with the conclusions of the U.S. Nuclear Regulatory Commission (NRC)-sponsored Steam Explosion Review Group (Reference 19.12-4).

19.12.2.2.2 Ex-Vessel Fuel-Coolant Interaction

The first level of defense for ex-vessel steam explosion is the in-vessel retention of the molten core debris. If molten debris does not relocate from the vessel to the containment, there are no conditions for ex-vessel steam explosion. In the event that the reactor cavity is not flooded and the vessel fails, the PRA containment event tree assumes that the containment fails in the early time frame.

An analysis of the structural response of the reactor cavity was performed for the AP600 (Reference 19.12-5, Appendix B). As in the in-vessel steam explosion analysis, the results of this AP600 ex-vessel steam explosion analysis are extended to the AP1000. The vessel failure modes for AP600 and AP1000 are the same. The initial debris mass, superheat and composition are assumed to be the same as the AP600. The reactor cavity geometry and water depth prior to vessel failure are the same as AP600. Therefore, the results of the AP600 ex-vessel steam explosion analysis are considered to be appropriate for the AP1000.

19.12.2.3 Hydrogen Combustion and Detonation

A decomposition event tree analysis discussed in Section 19.18 evaluates the potential for hydrogen combustion threatening the containment integrity during a severe accident sequence in the AP1000. The analysis examines diffusion flame burning and local detonation occurring during in-vessel hydrogen generation prior to hydrogen mixing in the containment and global deflagration and detonation, which may occur later when the hydrogen is mixed throughout the containment. Only in-vessel hydrogen generation is considered, since vessel failure and ex-vessel debris relocation is assumed to fail containment.

The AP1000 provides defense-in-depth to address hydrogen diffusion flames that may challenge containment integrity. The first level of defense is the stage four automatic depressurization system (ADS Stage 4) lines from the RCS, which prevent significant hydrogen releases to the in-containment refueling water storage tank (IRWST) and Passive Core Cooling System (PXS) compartments. ADS Stage 4 vents from the RCS hot legs to the loop compartments, which are shielded from the containment shell and have a constant source of oxygen from the natural circulation in the containment. Hydrogen can burn as a diffusion flame in the loop compartments without threatening the containment integrity. If ADS Stage 4 fails, the AP1000 has provided design considerations in IRWST vents to mitigate diffusion flames near the containment walls. Vents from the passive injection system compartments and chemical and volume control system compartment are located away from the containment shell and penetrations in order to mitigate the threat from hydrogen diffusion flames.

Containment failure from a directly initiated detonation wave is not considered to be a credible event for the AP1000 containment. There are no ignition sources of sufficient energy to directly initiate a detonation in the AP1000 containment. Deflagration to detonation transition (DDT) is considered to be the only likely mechanism to produce a detonation in the AP1000 containment.

The likelihood of DDT in the AP1000 containment is evaluated locally in confined compartments during in-vessel hydrogen generation and globally after in-vessel generation is concluded and hydrogen is mixed in the containment. For a DDT to occur, the combination of the gas mixture sensitivity to detonation and the geometric configuration potential for flame acceleration must be conducive to DDT. Since the hydrogen concentration necessary to form a detonable mixture depends on the size of the enclosure, concentration requirements for DDT in different regions of the AP1000 containment are extrapolated from the FLAME facility data (Reference 19.12-6) using scaling arguments based on the detonation cell width. The geometric requirement is evaluated considering aspects such as the degree of confinement and the extent and type of obstacles present in the

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postulated flame propagation path. In all cases, DDT is assumed to result in containment failure in the containment event tree analysis.

Global hydrogen deflagration and the potential for containment failure are modeled on the containment event tree. Adiabatic, isochoric, complete combustion (AICC) is assumed, and peak pressure probability distributions are developed for the accident scenarios. The probability of containment failure due to hydrogen deflagration is evaluated from the containment failure probability distribution combined with the peak pressure probability distribution.

19.12.2.4 High-Pressure Melt Ejection

The AP1000 incorporates design features that prevent high-pressure core melt. These features include the passive residual heat removal (PRHR) system and the ADS. These design features provide primary system heat removal and depressurization to prevent high pressure core damage conditions. The consequences from postulated high pressure melt ejection (HPME) are mitigated by the containment layout which provides a torturous pathway to the upper compartment, and no direct pathway for the impingement of debris on the containment shell.

In high-pressure core damage sequences the potential exists for creep-rupture-induced failures of the RCS piping at the hot-leg nozzles, the surge line, the steam generator tubes and, given debris relocation to the lower plenum, in the reactor vessel lower head. Failure of the hot-leg nozzle or surge line prior to failures of other components results in the rapid depressurization of the RCS. Failure of the steam generator tubes results in a containment bypass and a large release of fission products to the environment. Failure of the lower head of the reactor vessel results in the potential for HPME.

Hot-leg nozzle failure is expected prior to steam generator tube failure, but because of large uncertainties, hot-leg nozzle creep rupture failure is not credited with preventing steam generator tube failure. In the PRA, steam generator tube failure is assumed for high-pressure sequences in the containment event tree analysis unless operator action to depressurize the RCS with the ADS is successful.

19.12.2.5 Core Debris Coolability

In accident sequences where the reactor pressure vessel failure is not prevented, core debris may be discharged into the reactor cavity. The likely vessel failure modes produce a low pressure melt ejection (LPME) to the containment. The AP1000 cavity design provides area for the core debris to spread. Passive containment cooling system (PCS) heat removal ensures steam condensation on the inside containment shell. This condensate is recirculated back to the reactor cavity to provide a long-term supply of water to cool the core debris.

At vessel failure it is very likely that the cavity will be filled with water from the RCS, core makeup tanks (CMTs), and accumulators to at least the 83-ft elevation. There are significant uncertainties associated with debris spreading into a water-filled cavity. Debris-spreading is mainly a function of the highly uncertain vessel failure mode. A large-scale lower-head failure releasing debris at a high rate would enhance spreading, while a localized failure mode would release debris at a slow rate, which would most likely cause the debris to pile up under the reactor vessel and minimize spreading.

Given the uncertainties in the debris-spreading and in non-condensable gas generation and combustion, the containment event tree analysis does not credit containment integrity in the event of failure of the lower head of the vessel and relocation of the core.

A limited set of deterministic analyses of debris spreading and core-concrete interaction in the AP1000 cavity is presented in Appendix 19B. The analyses show that basemat melt-through is not predicted to occur within 24 hours of the accident initiation. Basemat melt through is predicted to

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occur before pressurization of the containment by non-condensable gases challenges the containment integrity.

19.12.2.6 Containment Pressurization from Decay Heat

The AP1000 containment is cooled via the PCS (see Section 19.17). Evaporative water cooling of the containment shell provides long term containment cooling and limits the containment pressure to less than the design pressure for all severe accident events except hydrogen combustion (which is addressed separately). Containment water is provided to the top of the containment via redundant, diverse system of valves and lines, including a line that can be connected to an outside water source, such as a fire truck.

In the unlikely event that water cannot be supplied to the top of the containment shell for an extended period of time, air-only cooling by air flowing through the PCS annulus provides significant cooling to the containment. Under the right environmental conditions, the containment is expected to reach an equilibrium pressure that will not challenge containment integrity. However, under nominal-to-conservative environmental conditions, containment integrity by air-cooling alone cannot be assured. In this case, containment failure is predicted to occur more than 24 hours after accident initiation.

A significant amount of time is available for operator action to vent the containment under the severe accident management guidance (SAMG). Containment venting mitigates uncontrolled releases of fission products from a failed containment. The AP1000 can be vented on an ad-hoc basis under the SAMG from a number of containment penetrations. Containment venting also reduces the partial pressure of non-condensable gases in the containment, and thus creates a new containment underpressure failure mode that may occur if containment is cooled after venting.

19.12.2.7 Elevated Temperatures (Equipment Survivability)

Reference 19.12-7 states that equipment identified as being useful to mitigate the consequences of severe accidents must be designed to provide reasonable assurance that it will continue to operate in a severe accident environment for the length of time needed to accomplish its function. Also, 10 CFR 50.44 requires safety equipment to continue performing its function after being exposed to a containment environment created as a consequence of generating a quantity of hydrogen equivalent to that from 100-percent cladding oxidation. As the AP1000 design uses thermal igniters to burn hydrogen in a controlled manner, it is necessary to demonstrate that the safety equipment can continue to perform its function in the high-temperature environment created by the hydrogen burning.

The functions of the equipment in containment for which credit is taken in the AP1000 PRA were reviewed to determine if the equipment is required to operate in a severe accident environment and beyond design basis limits. The equipment and the basis for operation are the same as the AP600 except for the igniter equipment, which is similar to the AP600. Therefore, the results of the AP600 are extended to the AP1000 for equipment survivability (see Appendix 19D).

19.12.2.8 Summary

The potential for and the consequences of severe accident phenomena are evaluated. The preventive and mitigative features of the AP1000 addressing the severe accident phenomena are discussed. This information is applied to the containment event trees and used in the quantification of the large release frequency.

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19.12.3 Analysis Method

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

19.12.4 Severe Accident Analyses

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

19.12.5 Insights and Conclusions

The analyses of the severe accident phenomena for the AP1000 PRA highlight the following insights and conclusions:

- The design of the AP1000 reactor vessel, vessel insulation, and reactor cavity; and the ability to flood the cavity after a severe accident reduce the potential challenges to the containment integrity by maintaining the vessel integrity.
- Should a failure of the reactor vessel occur, the design of the reactor cavity enhances the ability to cool any core debris that exits the vessel.
- Lower head vessel failure due to in-vessel steam explosions is physically unreasonable.
- The ADS and PRHR system are design features that can be used to prevent high-pressure core melt in a severe accident.
- A directly-initiated hydrogen detonation in the AP1000 containment is not a credible event.
- The equipment needed to mitigate the consequences of a severe accident is designed to provide reasonable assurance that it will continue to operate during an accident.

19.12.6 References

19.12-1. Letter from D. A. Ward, Advisory Committee on Reactor Safeguards, to K. A. Carr, Chairman, Nuclear Regulatory Commission, “Proposed Criteria to Accommodate Severe Accidents in Containment Design,” dated May 17, 1991.

19.12-2. Theofanous, T. G., et al., “In-Vessel Coolability and Retention of a Core Melt,” DOE/ID-10460, July 1995.

19.12-3. Theofanous, T. G., et al., “Lower Head Integrity Under In-Vessel Steam Explosion Loads,” DOE/ID-10541, July 1996.

19.12-4. NUREG-1116, *A Review of the Current Understanding of the Potential for Containment Failure From In-Vessel Steam Explosions*, 1985.

19.12-5. GW-GL-022, AP600 Probabilistic Risk Assessment, August 1998.

19.12-6. Sherman, M. P., Tieszen, S. R., and Benedick, W. B., *FLAME Facility - The Effects of Obstacles and Transverse Venting on Flame Acceleration and Transition to Detonation for Hydrogen-Air Mixtures at Large Scale*, SAND85-1264, Sandia National Laboratories, NUREG/CR-5275, April 1989.

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19.12-7. Attachment to letter from D. M. Crutchfield, Office of Nuclear Reactor Regulation, to E. E. Kintner, Advanced Light Water Reactor Steering Committee, “Major Technical and Policy Issues Concerning the Evolutionary and Passive Plant Designs,” dated February 27, 1992.

19.12-8. “EPRI MAAP 4.0 Users Manual.”

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19.13 Reactor Coolant System Depressurization

19.13.1 Introduction

Depressurization of the reactor coolant system is required for the external water cooling of the reactor vessel that will prevent vessel failure and core debris relocation to the containment (Reference 19.13-1). If the reactor coolant system (RCS) is at high pressure during core damage, containment failure may be postulated by several severe accident phenomena including induced failure of the steam generator tubes, high-pressure melt ejection, and direct containment heating.

19.13.2 Definition of High Pressure

High pressure is defined to support the assumptions of the PRA model. Induced steam generator tube rupture, high-pressure melt ejection, and reactor vessel failure into the flooded cavity will not occur if there is successful reactor coolant system depressurization.

Vessel failure can occur at elevated pressures with melted core debris in the vessel. This could cause the ejection of core debris from the vessel, followed by entrainment of debris in the high-velocity steam and water blowdown that would follow. Direct containment heating (DCH) and shifting of the reactor vessel are postulated containment failure mechanisms related to high pressure ejection of molten core debris.

Vessel failure and ex-vessel severe accident phenomena are prevented in the AP1000 by external cooling of the reactor vessel when the cavity is flooded. Flooding of the cavity occurs when the in-containment refueling water storage tank (IRWST) water fills the cavity. This can happen because of depressurization and subsequent in-containment refueling water storage tank water injection or when the cavity flood lines on the in-containment refueling water storage tank are opened. This cooling confirms the core debris will remain in the vessel. Heat transfer from the molten debris in the vessel through the lower head will thin the vessel wall and reduce the capability of the vessel to withstand higher pressures. It is conservatively assumed in this analysis that molten core debris in the vessel could cause failure at pressures greater than 150 psig.

If there is no molten core debris in the vessel, the vessel and the rest of the reactor coolant system, including the stream generator tubes, are expected to remain intact at pressures up to 3200 psig. This is consistent with the design of the reactor coolant system primary side.

19.13.3 References

19.13-1. Theofanous, T. G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

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19.14 Containment Isolation

Containment isolation is required before significant fission-product release after core uncovery. If the containment is not isolated, then the core damage results in a fission-product release to the environment. Containment isolation following an accident is achieved automatically by the protection and safety monitoring system or by the operator as instructed by an Emergency Response Guideline.

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19.15 Reactor Vessel Reflooding

Reflooding of the in-vessel core debris will occur following an accident if the reactor coolant system is sufficiently depressurized and if the in-containment refueling water storage tank water can enter the reactor vessel, either through one or both in-containment refueling water storage tank gravity injection lines or through a break in the reactor coolant system.

Successful reflooding of the reactor vessel following an accident that resulted in core damage provides additional cooling to core debris and the vessel wall. It may also have the undesirable effect of leading to the production of hydrogen if water reacts with unoxidized zirconium and molten core debris.

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19.16 In-Vessel Retention of Molten Core Debris

19.16.1 Introduction

In-vessel retention of molten core debris through water cooling of the external surface of the reactor vessel is a severe accident management feature of the AP1000. During postulated severe accidents, the accident management strategy to flood the reactor cavity with in-containment refueling water storage tank water and submerge the reactor vessel is credited with preventing vessel failure in the AP1000 probabilistic risk assessment. The water cools the external surface of the vessel and prevents molten debris in the lower head from failing the vessel wall and relocating into containment. Retaining the debris in the reactor vessel protects containment integrity by eliminating the occurrence of ex-vessel severe accident phenomena, such as ex-vessel steam explosion and core-concrete interaction, which have large uncertainties with respect to containment integrity.

The AP1000 provides for in-vessel retention with features that promote external cooling of the reactor vessel:

- The reliable multi-stage reactor coolant system depressurization system results in low stresses on the vessel wall after the pressure is reduced.
- The vessel lower head has no vessel penetrations to provide a failure mode for the vessel other than creep failure of the wall itself.
- The floodable reactor cavity can submerge the vessel above the coolant loop elevation with water intentionally drained from the in-containment refueling water storage tank.
- The reactor vessel insulation provides an engineered pathway for water-cooling the vessel and for venting steam from the reactor cavity.

19.16.2 Background on the Application of In-Vessel Retention to the Passive Plant

The Risk-Oriented Accident Analysis Methodology (ROAAM) analysis of the in-vessel retention phenomena (References 19.16-1 and 19.16-2) provided the basis for the application of the in-vessel retention accident management strategy to the AP600 passive plant and quantification of vessel failure in the AP600 PRA (Reference 19.16-3). The ROAAM included an analysis of the in-vessel melt progression and evaluation of the structural and thermal challenges to the vessel during the relocation to the lower head, including in-vessel steam explosion. Testing and evaluation of the uncertainties associated with the thermal loads produced by the in-vessel circulating molten debris pool, and heat removal limitations due to boiling crisis on the exterior vessel surface were performed in the ACOPO (Reference 19.16-4) and ULPU programs (References 19.16-1 and 19.16-5). The ROAAM concluded that the limiting challenge to the vessel integrity is the thermal loading produced during the steady-state heat transfer to the lower head wall after complete debris relocation to the lower plenum. The in-vessel retention ROAAM analyses and testing showed that the water in the AP600 cavity will remove the heat produced by the molten debris bed in the lower head with significant margin while the structural integrity of the lower head was maintained.

Based on the ROAAM results, vessel failure in the AP600 was considered to be physically unreasonable, and a probability of zero was applied to vessel failure in the AP600 PRA (Reference 19.16-3) if the following conditions of the ROAAM analysis were met:

- The reactor coolant system was depressurized.
- The reactor vessel was submerged sufficiently to wet the heated surface.

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- Reactor vessel insulation and containment water recirculation flow paths allowed sufficient ingress of water and venting of steam from the cavity.

- The treatment of the lower head outside surface (painting, coatings, etc.) did not interfere with water cooling of the vessel.

19.16.3 Application of In-Vessel Retention to the AP1000 Passive Plant

To establish a strong basis for crediting in-vessel retention in the AP1000, the following steps are taken:

- Establish design measures to increase the capability of the water to remove heat from the external surface of the reactor vessel (increase critical heat flux).

- Demonstrate that the thermal failure remains the limiting failure over the structural failure for the AP1000.

- Demonstrate that the AP1000 in-vessel melt progression does not change from the AP600 melt progression in such as way as to challenge the vessel integrity during relocation.

- Demonstrate that the heat load correlation, as applied from the ACOPO program (Reference 19.16-4), scales appropriately to the AP1000.

- Quantify the thermal loads using probability distributions developed specifically for the AP1000.

These items are discussed in the following sections.

19.16.4 Reactor Vessel Failure Criteria

The conclusions of the structural analyses performed for the AP600 in Reference 19.16-1 can be extrapolated to the AP1000. Thus, for the AP1000, success of in-vessel retention can be based solely on the thermal success criterion.

19.16.5 In-Vessel Melt Progression and Relocation

The AP1000 core and lower internals geometry has been changed from the AP600 geometry as a result of the higher power output. The core is made up of 157 fuel assemblies with a 14-foot active fuel length. To accommodate the larger reactor core, the thick stainless steel reflector has been replaced by a nominal 1″ thick core stainless steel shroud. The thick bottom plate of the shroud is mounted flush on the support plate. There are no former plates in the annulus between the shroud and the core barrel. The core barrel is 2″ thick and hangs from the upper head flange. Cooling holes through the core shroud provide cooling flow to the shroud from the core flow.

The phenomena associated with melting the core and the relocation of the molten debris to the lower plenum play an important role in the composition and configuration of the debris pool (Reference 19.16-2). In turn, the characteristics of the debris pool significantly impact the heat loading to the lower head wall and the challenge to lower head integrity (Reference 19.16-1). Therefore, understanding the melting and relocation scenarios plays an important role in the assessment of in-vessel retention of molten core debris in the lower plenum.

The important conclusions from the analysis of the lower plenum debris pool formation are:

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- The lower plenum debris bed is cooled with water during the entire relocation process prior to contact with the support plate. Transient debris configurations are not predicted to threaten vessel integrity.

- The lower plenum oxide debris subsumes the lower core support plate before dry out in the lower plenum occurs. If the relocated debris is assumed to be instantaneously quenched in the lower plenum water, the oxide debris contacts the lower support plate before the debris can return to a superheated condition. Therefore, the lower core support plate, core shroud and a sizeable fraction of the core barrel are subsumed in the debris bed. The focusing effect is mitigated.

- The lower plenum debris bed is predicted to form a metal layer over oxide pool configuration.

- The potential for debris interaction creating a bottom metal pool of uranium dissolved in zirconium is expected to be small.

- The earliest time to achieve the fully molten, circulating debris bed in the lower plenum is 2.7 hours after event initiation.

19.16.6 Application of Heat Transfer Correlations to the AP1000

19.16.6.1 Debris Pool to Vessel Wall Heat Transfer

The heat transfer from the oxide pool containing the decay heat producing fission products to the lower head of the reactor vessel is described using correlations that were developed in the Department of Energy (DOE) program for the AP600 in-vessel retention assessment.

The correlations developed in the ACOPO experiments and used in the AP600 in-vessel retention ROAAM for heat transfer from the debris pool to the lower head wall are applicable for use in the AP1000 in-vessel retention analysis.

19.16.6.2 Vessel Wall to External Cooling Water Heat Transfer

The heat transfer from the vessel wall to the cooling water is limited by the transition to film boiling at the external surface of the vessel wall. The maximum heat flux that can be removed prior to the transition to film boiling is the critical heat flux. If the heat flux from the debris pool to the wall is less than the critical heat flux, the vessel maintains sufficient strength to carry the load on the vessel. At heat fluxes above the critical heat flux, the external wall temperature increases significantly, the strength of the wall is lost, and the vessel fails.

Testing has been performed with ULPU-2000 Configuration IV (Reference 19.16-4) which demonstrates the feasibility of increasing the critical heat flux for AP1000. The heat removal capability is enhanced by constructing a hemispherical baffle outside the lower head to channel the cooling water flow and by assuring the flooding level in the containment outside the reactor vessel is sufficient for two phase natural circulation (Reference 19.16-4).

The AP1000 employs a reactor vessel insulation design that provides water inlet, steam venting and a baffle around the lower head to enhance the heat removal and increase the critical heat flux on the reactor vessel external surface. The insulation is vented from the annulus between the insulation and vessel to the vessel nozzle gallery at the 98 ft elevation.

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19.16.7 Quantification of Heat Load on the Reactor Vessel Wall

With the baffle installed in the AP1000 and the cavity adequately flooded, significant margin-to-failure for in-vessel retention via external reactor vessel cooling is achieved.

Based on the results of the ROAAM testing and analysis and the UPLU-2000 Configuration IV testing, vessel failure is concluded to be physically unreasonable in the AP1000 PRA provided the following conditions are met:

- The reactor coolant system is depressurized.
- The vessel is submerged adequately to promote natural circulation of water through the baffle surrounding the lower head.
- Reactor vessel insulation remains structurally sound under the pressure loads produced by the boiling external to the reactor vessel, allows water inlet at the bottom and venting of steam at the top, and provides the proper baffling to increase the critical heat flux on the external surface of the vessel lower head.
- The reactor vessel external surface conditions do not preclude the wetting phenomena identified as the cooling mechanism in the ULPU testing.

19.16.8 Reactor Coolant System Depressurization

Reactor coolant system depressurization is discussed in Section 19.13.

19.16.9 Reactor Cavity Flooding

Reactor cavity flooding is accomplished through either operator action or through the progression of the accident. The operator floods the cavity by opening a motor-operated valve and a squib valve in the recirculation lines between the in-containment refueling water storage tank and the containment recirculation sump, as shown in Figure 6.3-3. The operator action is prescribed by entering the AFR-C.1 Function Restoration Guideline (Reference 19.16-6) when the core-exit thermocouples reach 1200°F. The water floods the containment by flowing out of the recirculation screens, filling the containment floodable region of the containment, shown in Figure 19.16-1, to at least the 107′ 2″ elevation, shown in Figure 19.16-2.

To achieve the high critical heat flux for the AP1000 lower head, water level in containment must be sufficient for two phase natural circulation flow. The vents from the AP1000 reactor vessel insulation exit to the vessel nozzle gallery at the 98 ft elevation. It is conservatively assumed that the water level in the containment has to reach the 98 ft elevation within seventy minutes after the core exit temperature exceeds 1200°F, for successful vessel cooling.

The AP600 procedures instructed the operator to flood the reactor cavity at the end of AFR-C.1 Function Restoration Guideline before entering the severe accident management guidelines. The AP1000 requires the cavity to be flooded to a higher level and more quickly than the AP600. For the AP1000, the operator action to initiate cavity flooding has been moved to the entry of the AFR-C.1 Function Restoration Guideline to meet the time requirement for cavity flooding success.

19.16.10 Reactor Vessel Insulation Design Concept

With respect to in-vessel retention severe accident management, the goal of the reactor vessel insulation is to ensure that there will always be an adequate water layer next to the reactor vessel to promote heat transfer from the reactor vessel. The insulation will define an optimized flow path next

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to the lower head to enhance the critical heat flux. The cooling of the vessel in a severe accident is accomplished by providing:

- A means of allowing water free access to the region between the reactor vessel and insulation.
- A frame that maintains the structural integrity of the insulation surrounding the lower head which provides the baffle for the water flow next to the vessel.
- A means to vent steam generated by the water cooling the vessel wall from the insulation surrounding the reactor vessel.
- A support frame to prevent the insulation panels above the vessel lower head from breaking free and blocking water from cooling the reactor vessel exterior surface.

19.16.10.1 Description of Reactor Vessel Insulation and Venting

Subsection 5.3.5 provides a description of the reactor vessel insulation and the functional requirements for the insulation.

19.16.10.2 Design Analysis of the Insulation and Support Frame

The insulation forms an engineered pathway to enhance the cooling of the external surface of the reactor vessel during in-vessel retention. Structural support to maintain this pathway is provided.

19.16.10.3 Reactor Vessel External Surface Treatment

Based on the reactor vessel system design specification, the surface is not coated and remains as bare metal.

19.16.11 Reactor Vessel Failure

Based on the analysis of in-vessel retention, an intact reactor vessel remains intact if the reactor coolant system is depressurized and the reactor vessel is adequately submerged.

19.16.12 Summary

In-vessel retention of molten core debris via external reactor vessel cooling can be accomplished in the AP1000.

- The reactor vessel insulation must provide a structurally sound baffle around the lower head and lower cylinder of the vessel to channel the flow between the vessel and insulation. An insulation design that provides the proper water inlet, steam venting and flow baffling is specified for the AP1000.
- The reactor cavity must be flooded to an elevation of at least 98 ft prior to the onset of the steady–state heat flux to the vessel wall from the debris to produce the driving head required to enhance the critical heat flux on the vessel surface. The operator action to flood the cavity has been moved to the first step of the emergency operating procedures to provide adequate flooding.

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19.16.13 References

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19.16-2. Theofanous, T.G., et al., “Lower Head Integrity Under In-Vessel Steam Explosion Loads,” DOE/ID-10541, June 1996.

19.16-3. AP600 PRA Report, GW-GL-022, August 1998.

19.16-4. Theofanous, T.G., and S. Angeli, “Natural Convection for In-Vessel Retention at Prototypic Rayleigh Numbers,” Nuclear Engineering and Design, 200, 1-9 (2000).

19.16-5. Angelini, S., et al., “The Mechanism and Prediction of Critical Heat Flux in Inverted Geometries,” Nuclear Engineering and Design, 200, 83-94 (2000).

19.16-6. AP600 Emergency Response Guidelines.

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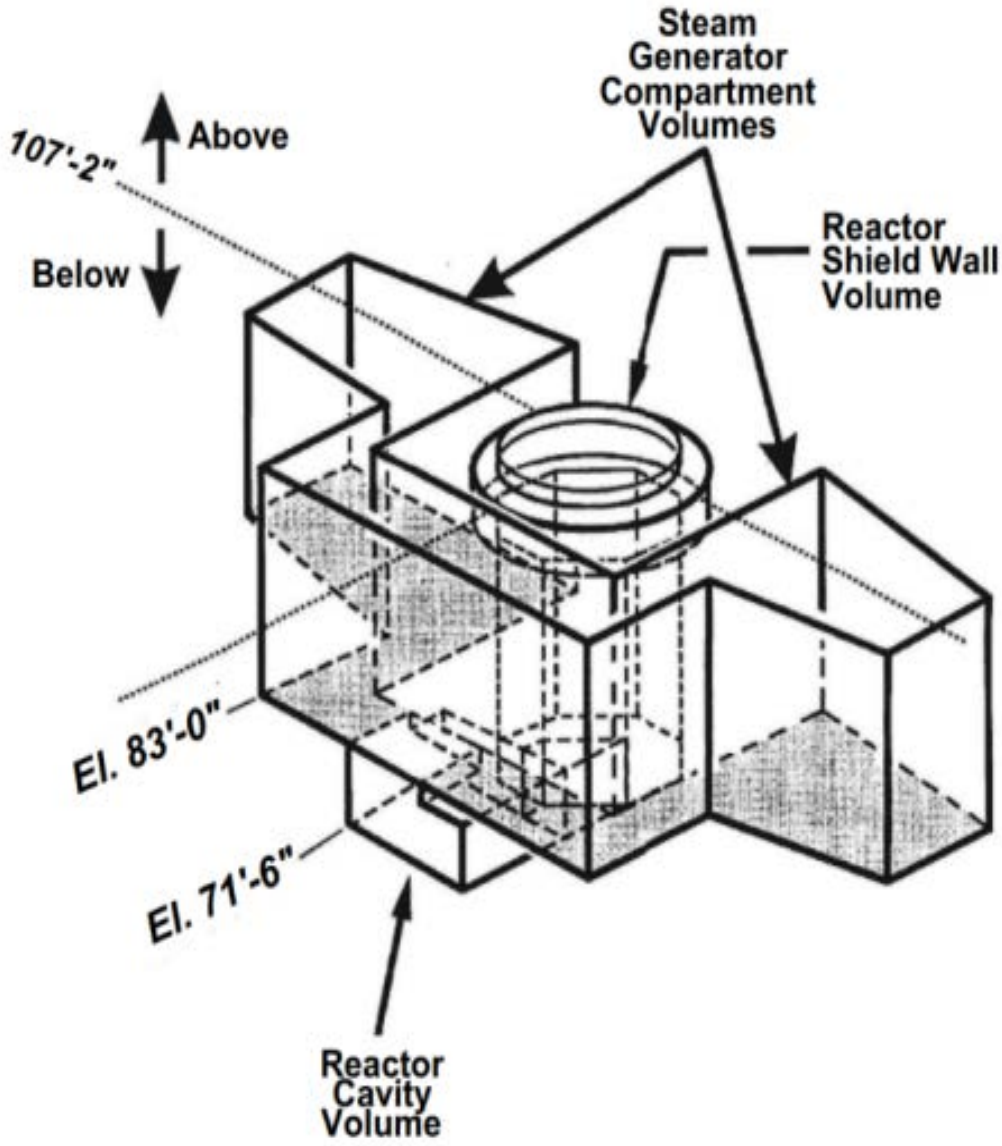

Figure 19.16-1
Containment Floodable Region

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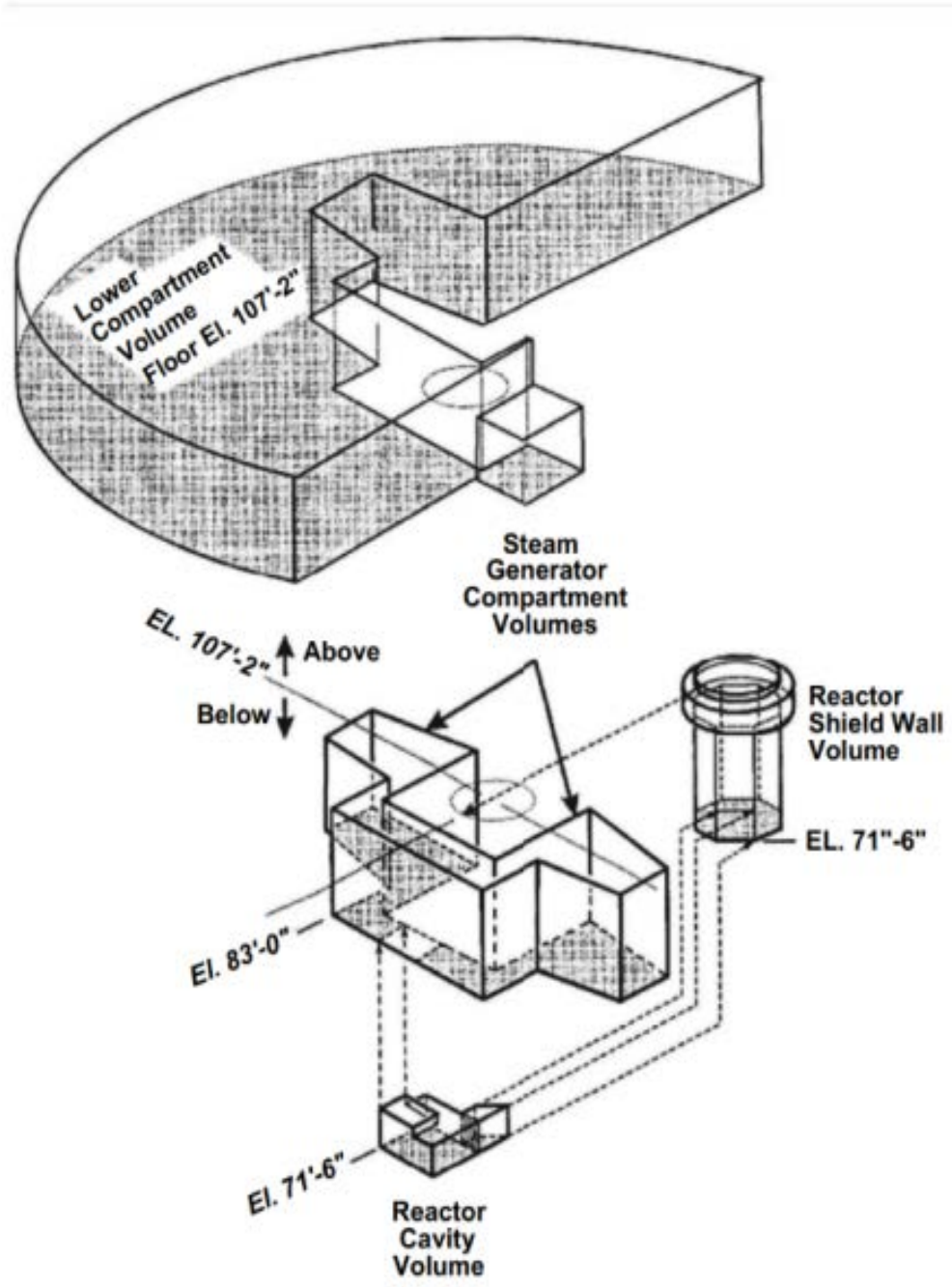

Figure 19.16-2
Containment Floodable Region – Expanded View

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19.17 Passive Containment Cooling

The Nuclear Regulatory Commission (NRC) containment performance goal for advanced containment systems is to provide a leak-tight barrier to fission product release for 24 hours following an accident, and to remain as a barrier against uncontrolled releases following that time. The AP1000 containment is cooled via the passive containment cooling system (PCS). Barring hydrogen combustion and ex-vessel phenomena that are addressed elsewhere in the AP1000 containment event tree analysis, the AP1000 containment is not expected to exceed the design-basis pressure during a severe accident. No threat to the containment integrity from long-term overpressurization is predicted.

In the event that design-basis cooling fails, the containment pressure will exceed the design basis, although containment failure within 24 hours is predicted to be highly unlikely. After 24 hours, the operator may vent the containment to prevent uncontrolled failure of the containment per the severe accident management guidelines. Once vented, the steam concentration in the containment will increase and improve the heat removal capacity of the passive cooling such that no further venting would be required.

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19.18 Hydrogen Mixing and Combustion Analysis

19.18.1 Introduction

In the course of a severe accident, a substantial amount of combustible gases can be generated in-vessel from the oxidation of the zirconium and other metals. The AP1000 containment is provided with nonsafety-related hydrogen igniters to control the concentration of combustible gases. If the igniters operate, combustion of hydrogen plumes may present a thermal load to the containment. Combustible gas can accumulate in the containment at flammable concentrations if the igniter system fails to function. The AP1000 hydrogen analysis quantifies the threat to containment integrity with and without hydrogen igniters.

If vessel failure does not occur, the amount of hydrogen in the containment is limited to the mass generated during the in-vessel core heatup and relocation. If vessel failure occurs with water in the cavity, an additional amount of hydrogen may be generated from ex-vessel fuel-coolant interactions. Furthermore, if the debris layer in the cavity is not coolable or if insufficient water is available in the containment to cool the debris, and subsequent thermal attack of concrete occurs, additional hydrogen and other combustible gas, such as carbon monoxide, will be generated. The AP1000 PRA assumes containment failure if vessel failure is predicted, so the evaluation of containment integrity from hydrogen combustion only considers in-vessel hydrogen generation.

Hydrogen combustion is evaluated during two time frames: early (during the in-vessel relocation and hydrogen generation) and intermediate (prior to 24 hours after the onset of core damage). In the early time frame, containment challenge is considered from hydrogen burning as an unmixed plume (diffusion flame) and from local detonation at high concentrations in confined compartments below the operating deck. In the intermediate time frame when the hydrogen is mixed, containment challenge from global deflagration and potential detonation due to stratification of gases is considered. The hydrogen is assumed to burn within 24 hours of core damage.

19.18.2 Controlling Phenomena

The conditions required for combustion in the containment are flammable gas mixtures and the presence of an ignition source. Typically, a spark is sufficient to cause ignition. If the mixture temperature is above ~1000 K, auto-ignition can occur without the presence of an ignition source. The flammability limits are determined by the concentrations and temperature of the combustible gas-air-diluent mixture. Hydrogen and the oxygen in the air are the reactants in the combustion reaction. Steam, carbon dioxide, and excess nitrogen in the mixture act as inertants that may inhibit the reaction.

Hydrogen-air-steam mixtures can burn in several modes: diffusion flames, slow and accelerated deflagrations, and detonations (Reference 19.18-1). Burning of an unmixed hydrogen plume near the source results in a diffusion flame. Diffusion flames are stationary and result primarily in thermal loads on nearby structures or equipment. Deflagrations or detonations are burning of premixed gases. In practical terms, a slow deflagration is a flame that travels at a speed much slower than the speed of sound such that the pressure inside the containment equilibrates during the combustion. No dynamic loads are generated. Accelerated deflagrations travel fast enough to generate shock waves and dynamic loads. Detonations travel at supersonic velocities and also generate dynamic loads. The static loads that result from deflagrations can be predicted and bounded. The maximum dynamic loads from accelerated flames and detonations are difficult to calculate.

Standing diffusion flames on the in-containment refueling water storage tank pool or at the in-containment refueling water storage tank vents can be postulated early into an accident following core uncovery for sequences in which the automatic depressurization system stages 1 through 3 provide a primary depressurization mechanism. A standing diffusion flame at the vent could present a

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thermal load to the containment steel shell, which is close to some of the vents. If the primary system break is in one of the PXS valve/accumulator rooms which flood with water and submerge the break, diffusion flames can also be postulated at the room exit in the maintenance floor. This location has a direct line of sight with the personnel and equipment hatches, electrical penetrations, and the containment shell, and may present a thermal loading challenge.

The static loads associated with deflagrations are limited by thermodynamics. If all of the chemical energy available in the mixture is converted to temperature and pressure, then the maximum pressure is limited by the adiabatic, isochoric (constant volume), complete combustion (AICC) pressure. The actual pressure would drop over time from this peak because of heat losses to water, structures, and equipment in containment. Dynamic pressure loads are not limited by the adiabatic, isochoric, complete combustion value because the local pressure is due to very rapid, nonequilibrium combustion.

The mode of combustion depends on the mixture concentrations, initial conditions, and boundary conditions (Reference 19.18-1). Near the hydrogen source, hydrogen may not be mixed significantly with the air in the containment. If ignition occurs there, then a diffusion flame may be formed. Further downstream from the hydrogen source, mixing will have occurred and a deflagration or detonation may result, depending on the hydrogen concentration and geometric boundary factors. In some cases, accelerated flames may also develop to detonations, which are called deflagration-to-detonation transition (DDT). The occurrence of flame acceleration and deflagration-to-detonation transition is complex and not completely understood. It is dependent on a number of parameters. These include hydrogen and oxygen concentrations; nature and concentration of inertants; gas temperature and pressure before ignition; ignition source; the size and shape of the compartment in which the combustion occurs; and the number, size, and shape of any obstacles in the compartment.

In AP1000, direct initiation of detonation by sufficiently high-energy sources from equipment in containment is unlikely (Reference 19.18-2: Since AP1000 is very similar to AP600, the phenomenological evaluations are valid for AP1000.), but mechanisms to accelerate a flame to a detonation may occur. Deflagration-to-detonation transition is considered the most likely mechanism. Transition to detonation is considered in several sections of the containment for accident sequences that result in hydrogen concentrations greater than 10 volume percent, including the passage connecting the two steam generator compartments, the core makeup tank and equipment bay, in-containment refueling water storage tank gas space, steam generator compartments, and steam generator operating deck.

19.18.3 Major Assumptions and Phenomenological Uncertainties

Because of phenomenological uncertainties, a number of assumptions are necessary in the hydrogen analysis.

19.18.3.1 Hydrogen Generation

The degree to which the cladding is oxidized during the in-vessel phase of the accident sequence and the availability of water to the core determines the rate and the mass of hydrogen released to the containment during the early time phase. The rate and mass of hydrogen produced are important parameters in determining the hydrogen concentration and the flammability limits of the gas mixtures in the containment compartments.

19.18.3.2 Containment Pressure

The containment pressure is an important parameter in the determination of the pre-burn boundary conditions. A higher initial pressure can result in a higher peak pressure, but the increased steam

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mass can inert the mixture and prevent combustion. If the passive containment cooling system water is not operational, containment pressures are elevated and combustion is steam inerted.

19.18.3.3 Flammability Limits

A flammable condition is determined by flammability limits. Flammability limits of a combustible gas mixture are defined as the limiting gas compositions at a given temperature and pressure in which a deflagration will propagate once ignited. There is information on flammability limits of hydrogen-air-steam mixtures at temperatures less than 149°C. For hydrogen, there are two lean propagation limits considered, upward and downward. At lean upward propagation limits, flames will propagate upward because of buoyancy. At lean downward propagation limits, flames will propagate upward and downward throughout the volume by their own reaction kinetics. Hence, the extent of flame propagation (or combustion completeness) for combustion at lean flammability limits is determined by the hydrogen concentration. This relation is a result from the Nevada Test Site (Reference 19.18-3). The addition of steam or other inert gas has a strong effect on the hydrogen concentration and flammability (Reference 19.18-4).

Combustion initiated by igniters occurs at lean upward flammability limits with a small pressure rise. However, with the failure of igniters, combustion at a hydrogen mixture at a concentration above the lean downward propagation limits may result in much larger pressure and temperature consequences. The global burn considered in the analysis is defined as combustion at or above the lean downward propagation limits. This definition includes the possibility that a global burn becomes a detonation, since the occurrence of a detonation requires a hydrogen concentration much above the lean downward propagation limits. Combustion regimes and associated adiabatic, isochoric, complete combustion pressure are approximately demonstrated for hydrogen-air mixtures in Reference 19.18-5.

19.18.3.4 Detonation Limits and Loads

A detonation is a supersonic combustion front that produces a dynamic load in excess of the adiabatic, isochoric, complete combustion value. The energy release from the combustion of the hydrogen-air-steam mixture sustains the shock structure that ignites and burns the mixture. The detonation limits cannot currently be predicted by any first-principles theory. Engineering correlations used to predict the limits have been developed based on a measurable quantity called the detonation cell width. For simplified discussion, the detonation cell width can be considered a characteristic length that describes the sensitivity of the mixture to detonation. The smaller the detonation width, the easier it is to get the mixture to detonate and sustain propagation. Deflagration-to-detonation transition is considered, and the method of NUREG/CR-4803 (Reference 19.18-6) is used to evaluate the potential for flame acceleration.

Since the lowest hydrogen concentration for which deflagration-to-detonation transition has been observed in the intermediate-scale FLAME facility at Sandia is 15 percent (Reference 19.18-7), and 10 CFR 50.44 limits hydrogen concentration to less than 10 percent, the likelihood of deflagration-to-detonation transition is assumed to be zero if the hydrogen concentration is less than 10 percent.

19.18.3.5 Igniter System

The AP1000 nonsafety-related hydrogen igniter system, if operational during a severe accident, will burn hydrogen as soon as the lean upward flammability limits are met. Thus, the concentration of hydrogen is maintained, on average, at the lean upward flammability limits. However, depending on the hydrogen release rate, location and oxygen availability, locally high concentrations may exist in the in-containment refueling water storage tank or in the subcompartment where the pipe break occurs.

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The hydrogen igniters are actuated by manual action when core-exit temperature exceeds a predetermined temperature as directed by the emergency response guidelines (ERG). The indication and actuation are done with containment conditions within the equipment qualification limits of the systems used, within the design basis of the plant and systems, and before fission-product releases to the containment, so equipment survivability of the monitoring and actuation systems during the time frame that they are required to perform is supported (see Appendix 19D).

19.18.3.6 Other Ignition Sources

A flammable mixture will not burn without an ignition source unless the temperature of the mixture is sufficiently high (~1000 K) that auto-ignition becomes possible. Hot surfaces or random sparks from equipment or static electricity may be postulated ignition sources. High-temperature gas jets exiting from the reactor coolant system may become an ignition source. However, the gas stream may not have enough momentum to entrain the surrounding flammable mixture, especially in the depressurized cases.

19.18.3.7 Severe Accident Management Actions

Severe accident management guidance that is considered in the AP1000 PRA is the operator action to flood the reactor cavity in the event of core damage. This action often results in the late reflooding of a damaged core due to the time required for the operator to diagnose the problem and take the action. Some events will lead to core reflooding through the natural progression of the accident.

19.18.4 Hydrogen Generation and Mixing

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

19.18.5 Hydrogen Burning at Igniters

Analyses of AP600 demonstrated the effectiveness of the hydrogen igniter system as placed in the passive containment geometry. The cases in the burning analysis were chosen for variation in hydrogen generation rate, release locations into containment, in-containment refueling water storage tank water level, and PXS compartment flooding. The cases considered 100 percent cladding reaction. The behavior of the AP1000 is essentially the same as the AP600 with respect to hydrogen release rates and locations.

Generally, the reactor coolant system is depressurized prior to hydrogen generation. Hydrogen is released to the containment through ADS stage 4 as it is generated in the core. Natural circulation in the containment provides oxygen for burning the hydrogen at the igniters in the loop compartments, close to the source. The loop compartments are shielded from the containment shell and most equipment and instrumentation that would be used to mitigate and monitor the accident.

Igniters located in the IRWST, PXS and CVS compartments, CMT room and at various elevations in the upper compartment provide coverage for hydrogen that may be released through the IRWST, PXS/CVS or in the CMT room.

The igniter system maintains the global uniform hydrogen concentration in the containment at or below lower flammability limits. In the most likely severe accidents, the hydrogen is burned primarily in a favorable location that protects the integrity of the containment and mitigative and monitoring equipment.

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19.18.6 Early Hydrogen Combustion

Early hydrogen combustion is defined as burning that occurs during the period the hydrogen is released from the primary system to the containment. During this time, the hydrogen may not be well mixed in the containment and, depending on release locations, may be concentrated in the in-containment refueling water storage tank, PXS valve/accumulator rooms or chemical and volume control system room, steam generator compartments or maintenance floor. If sufficient oxygen is available, the compartments may become locally detonable. If oxygen is not available in the compartment, the plume may travel to a location where oxygen is available and it can burn as a diffusion flame.

19.18.6.1 Hydrogen Generation Rates

Qualitative hydrogen generation characteristics can be inferred from the availability of steam and the availability of overheated, unreacted zirconium in the reactor vessel. Based on the insights from hydrogen generation and mixing analyses, the hydrogen generation can be classified into one of three categories: boiloff generation rate, early-reflood generation rate, and late-reflood generation rate. This section briefly defines each type of hydrogen release in the AP1000 hydrogen analysis and the conditions under which they occur.

19.18.6.1.1 Boiloff Hydrogen Generation

Boiloff hydrogen generation occurs as the water inventory in the reactor vessel is depleted by decay heat. The steam generation is limited to the decay heat boiloff in the covered fraction of the core and overheated, unreacted zirconium surface area is limited to the upper regions of the core, which have not relocated below the water line. Core relocation to the lower head may produce a rapid steam generation that produces a brief period of rapid oxidation, but by this time, the core geometry is lost and very little unoxidized zirconium surface area is available for sustained hydrogen production.

19.18.6.1.2 Early-Reflood Hydrogen Generation

Early-reflood hydrogen generation occurs in the event of the reflooding of an overheated, relatively intact core. Quenching of the core provides a large quantity of steam and a large, overheated, unreacted zirconium surface area for oxidation. Shattering of the cladding due to thermal stresses can enhance the oxidation rate. In the early-reflood case, the oxidation of the zirconium is limited only by the degree of core uncovery prior to the reflood. The rate and degree of zirconium oxidation is expected to be greater than the no-reflood case.

19.18.6.1.3 Late-Reflood Hydrogen Generation

Late-reflood hydrogen generation occurs in the event of a reflood after the core has degraded significantly and possibly after relocation to the lower head. Much of the core geometry is lost and little surface area is available for oxidation, even when steaming from quenching debris is available.

19.18.6.2 Hydrogen Release Locations

The hydrogen release locations in the containment determine the hydrogen mixing in the containment and regions of high hydrogen concentration in the event that the igniters fail. The flow paths from release points in confined compartments to the volumes where oxygen is available determine possible locations where diffusion flames may occur.

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19.18.6.2.1 Automatic Depressurization System Stages 1, 2, and 3

Stages 1, 2 and 3 of the automatic depressurization system relieve the reactor coolant system pressure from the top of the pressurizer to the in-containment refueling water storage tank. The water level in the in-containment refueling water storage tank at the time of the release determines the steam concentration in the tank. If the spargers are covered, the steam is quenched out of the gas flow and the hydrogen is released to the gas space of the tank. If the spargers are not covered, the steam concentration is high and will drive the air out of the tank. If the igniters are available, diffusion flames may be postulated at the in-containment refueling water storage tank vent exits for large sustained hydrogen releases. If igniters are not available, the possibility of hydrogen detonation is evaluated.

19.18.6.2.2 Automatic Depressurization System Stage 4

Stage 4 of the automatic depressurization system relieves steam and hydrogen from the hot leg of the reactor coolant system to the steam generator compartments in the containment. The steam generator compartments, along with the maintenance floor and the upper compartment, form the major natural-circulation path in the containment. Oxygen starvation of any potential diffusion flames in the steam generator compartment is not expected for low-pressure hydrogen releases from automatic depressurization system stage 4. The containment shell is sheltered from flames in the steam generator compartments by the concrete walls, so diffusion flames at the igniters in the steam generator compartments are not considered to be a threat to the containment integrity. If igniters are not available, good mixing in the compartment mitigates the threat of detonation for the low-pressure releases.

19.18.6.2.3 Break Location

The reactor coolant system break provides a pathway from the reactor coolant system to one of several compartments in the containment. A failure of a component in the reactor coolant system loop (hot leg or cold leg) will relieve hydrogen to the loop compartment. Hydrogen released from the break to the loop compartment will behave similarly to the hydrogen released from stage 4 automatic depressurization system.

A failure of the direct vessel injection line or a break in the chemical and volume control system piping will relieve hydrogen to one of the small compartments under the maintenance floor, the chemical and volume control system room or one of the two PXS valve/accumulator rooms. These compartments are dead-ended and communicate with the maintenance floor through stairway or room vents. The initial blowdown through the break fills the compartment with steam and drives the air out of the compartment. After the blowdown and reactor coolant system depressurization, countercurrent flow between the compartment and the maintenance floor slowly replenishes the air.

Each of the dead-ended compartments has a one-way drain to the containment sump in the cavity. The break flow into a dead-ended compartment will not fill the compartment with water, as the draining and flashing of the break flow removes the water to the containment sump. However, a broken direct vessel injection line in a PXS valve/accumulator room may allow the in-containment refueling water storage tank to drain into the PXS valve/accumulator room if the injection valves open in the broken line. The draining of the in-containment refueling water storage tank water into the PXS valve/accumulator room will fill the PXS valve/accumulator room and spill water over the curb into the maintenance floor.

If the igniters are available, hydrogen released to the dead-ended compartments during the core degradation may burn initially, but may become oxygen starved. The plume then rises through the stairway to the maintenance floor, which is amply supplied with oxygen by the containment natural circulation. A diffusion flame can be postulated at the exit of the dead ended compartments in the

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maintenance floor. The exterior wall of the maintenance floor is the steel containment shell below the passive containment cooling system annulus, the lower-level equipment hatch, and the personnel hatch. Many electrical penetrations pass through the maintenance floor wall to the auxiliary building.

19.18.6.3 Early Hydrogen Combustion Ignition Sources

For a burn to be initiated, an ignition source is required. Igniters mitigate the threat to the containment integrity from global deflagration and detonation. If a hydrogen plume can produce a diffusion flame, the igniters provide the ignition source.

19.18.7 Diffusion Flame Analysis

Diffusion flames can be postulated to occur at vents or exits from compartments with a hydrogen source that are dead-ended or not well-mixed. Incombustible gas mixtures that include a high concentration of hydrogen may develop in the compartment. When the plume of hydrogen exits the compartment into a room containing oxygen and an ignition source, burning of the plume as a standing flame at the vent may produce locally high temperatures. If the release of hydrogen is sustained, the heat load from the burning may threaten equipment, including the containment shell integrity.

The overall geometry of the AP1000 containment is relatively open. Ninety-seven percent of the containment free volume participates in containment natural circulation and is well-mixed. However, the IRWST, PXS and CVS compartments are small, confined rooms that may have a hydrogen source, and thus may be postulated to produce a diffusion flame at vents. This section discusses the conditions that may produce a standing diffusion flame in these locations, and presents the quantification of the containment failure probability given the presence of a sustained diffusion flame at a dead-ended compartment vent.

AP1000 Diffusion Flame Mitigation Strategy

Hydrogen is a byproduct of a severe accident, and hydrogen pathways to the IRWST, PXS and CVS subcompartments cannot be completely ruled out, particularly in the IRWST, to which the effluent of the first stages of the reactor coolant system automatic depressurization system are directed. The other compartments can only have hydrogen releases in the event that a break occurs there, but some of the highest frequency severe accident sequences have breaks in a DVI line, which traverses a PXS compartment. Therefore, the potential for diffusion flames from these subcompartment locations cannot be excluded from the probabilistic risk assessment.

The AP1000 addresses diffusion flames by adopting a defense-in-depth philosophy in the design. In the highest frequency severe accidents, sustained hydrogen release is prevented from occurring in the dead-ended compartments. In sequences where diffusion flames at IRWST or PXS/CVS compartment vents may be postulated, design strategies are initiated to mitigate the threat to the containment integrity by locating hydrogen plumes where they do not challenge containment integrity.

The first level of defense against the threat to containment integrity from diffusion flames is the prevention of sustained hydrogen releases to dead-ended compartments. The highest frequency severe accident sequences have full reactor coolant system depressurization prior to core damage. Hydrogen is released at low pressure to the containment as it is produced in the core. Stage four of the automatic depressurization system provides a pathway of substantially lower resistance (by approximately one order of magnitude) compared to the maximum break size in the DVI line that relieves to the PXS compartment and to the other three ADS stages that relieve to the IRWST. Additionally, the ADS spargers in the IRWST generally have a 10-ft static head of water above them, which further increases the resistance to flow of hydrogen to the IRWST.

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Hydrogen released from ADS stage 4 is relieved to the loop compartments, which are supplied with oxygen by the containment natural circulation and shielded from the containment shell by high concrete walls. Hydrogen is able to burn in the loop compartments without threatening the containment integrity. Therefore, ADS stage 4 provides the first level of defense against diffusion flames.

In the event that ADS stage 4 fails to adequately direct hydrogen away from confined compartments, the compartment vents are designed to preferentially release the hydrogen at locations where it burns, but does not challenge containment integrity.

Vents from the PXS and CVS compartments to the CMT room are located well away from the containment shell and containment penetrations. Access hatches to the subcompartments that are near the containment shell are covered and secured closed such that they will not open as a result of a pipe break inside the compartment. Therefore, hydrogen releases to the CMT room from the subcompartments have been shown to not challenge the containment integrity.

19.18.8 Early Hydrogen Detonation

Hydrogen detonation can be initiated from a high-energy ignition source or by deflagration-to-detonation transition during flame acceleration. A review of potential ignition sources in containment concludes that the maximum source is too small to directly initiate a detonation (Reference 19.18-2: Since AP1000 is very similar to AP600, the phenomenological evaluations are valid for AP1000.). Therefore, the occurrence of detonation is related to the potential for deflagration-to-detonation transition in the AP1000 containment analysis.

The methodology of Sherman and Berman (Reference 19.18-6) is used to evaluate the likelihood of deflagration-to-detonation transition. The analysis considers the hydrogen release rates to the containment, core reflooding, the containment release locations, and in-containment refueling water storage tank and PXS valve/accumulator room water levels to determine the probabilities.

19.18.9 Deflagration in Time Frame 3

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

19.18.10 Detonation in Intermediate Time Frame

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

19.18.11 Safety Margin Basis Containment Performance Requirement

The AP1000 containment meets the criteria of the safety margin basis containment performance requirement.

19.18.12 Summary

The major insights of the hydrogen mixing and combustion analysis are as follows:

- No containment failure from hydrogen is predicted if the hydrogen igniters are operational.

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- Operation of the stage 4 automatic depressurization system valves releases much of the hydrogen generated in the reactor coolant system to the steam generator rooms where it can be well mixed in the containment to mitigate the threat of diffusion flames from sustained hydrogen released through the in-containment refueling water storage tank.

- The threat of detonation is predominantly due to hydrogen releases to the PXS valve/ accumulator rooms below the 107′ 2″ containment elevation (direct vessel injection line breaks). The compartment is a confined region with little ventilation. Equipment and grating are present to promote turbulence. A break in the compartment induces a high-temperature environment creating good conditions for potential deflagration-to-detonation transition.

- The probability of containment failure due to diffusion flame is very small.

- No containment failure is predicted from deflagration.

Analyses are performed to meet the requirements of 10 CFR 50.44. Igniter burning analyses with rapid hydrogen generation and 100-percent cladding reaction conclude that the igniter system maintains the global uniform hydrogen concentration in the containment at or below lower flammability limits. If the stage 4 automatic depressurization system is available, the hydrogen is well mixed in the containment and no excessive concentrations are predicted in the in-containment refueling water storage tank or PXS valve/accumulator rooms. If the stage 4 automatic depressurization system is failed, hydrogen in the in-containment refueling water storage tank and PXS valve/accumulator rooms can reach high concentrations. However, the mixtures are oxygen starved and are not flammable or detonable. The safety margin basis containment performance requirement is met as the loss-of-coolant accident plus 100-percent active cladding reaction hydrogen burn peak pressure provides margin to the ASME Service Level C stress limits.

19.18.13 References

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19.19 Core Damage and Severe Release Frequency from Events at Shutdown

19.19.1 Summary of Shutdown Level 1 Results

As shown by the dominant cutsets of the AP600 and AP1000 shutdown models, the risk profiles of these plants for events during shutdown conditions are almost identical. The results indicate that the three events dominating the CDF are loss of component cooling/service water during drained condition, loss of RNS during drained condition, and loss of offsite power during drained condition. The AP1000 and AP600 initiating event core damage contributions are similar for the two plants.

The dominant sequences are described in the subsections that follow. The dominant accident sequences comprise 95.3 percent of the level 1 shutdown PRA core damage frequency. These dominant sequences consist of:

- Loss of component cooling or service water system initiating event during drained condition with a contribution of 76.7 percent of the CDF
- Loss of RNS initiating event during drained condition with a contribution of 10.4 percent of the CDF
- Loss of offsite power initiating event during drained condition with a contribution of 8.2 percent of the CDF

Loss of Component Cooling or Service Water System Initiating Event During Drained Condition

These sequences are described as the loss of decay heat removal initiated by failure of the component cooling water or service water system during drained condition. The loss of decay heat removal occurs following loss of component cooling water system (CCS) or service water system (SWS) during mid-loop/vessel flange operation, which has an estimated duration of 120 hours per 18 months refueling cycle.

The major contributors to risk due to loss of CCS or SWS during drained condition are the following:

- Hardware failures of both service water pumps or common cause failure of digital input/output modules from the protection and monitoring system (PMS)
- Common cause failure of the ADS 4^{th} stage squib valves
- Common cause failure of the recirculation line squib valves
- Common cause failure of the IRWST injection squib valves
- Common cause failure of the strainers in the IRWST tank
- Common cause failure of the recirculation sump strainers

Loss of RNS Initiating Event During Drained Condition

This sequence is described as the loss of decay heat removal initiated by failure of the RNS during drained condition. The loss of decay heat removal occurs following loss of RNS during mid-loop/ vessel flange operation, which has an estimated duration of 120 hours per 18 months refueling cycle.

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The major contributors to risk due to loss of RNS during drained condition are the following:

- Common cause failure of the RNS pumps to run
- Common cause failure of the recirculation line squib valves
- Common cause failure of the ADS 4th stage squib valves
- Common cause failure of the IRWST injection squib valves
- Common cause failure of the strainers in the IRWST tank
- Common cause failure of the recirculation sump strainers

Loss of Offsite Power Initiating Event During Drained Condition (with failure of grid recovery within 1 hour)

This sequence is initiated by loss of offsite power during mid-loop/vessel flange operation, which has an estimated duration of 120 hours per 18 months refueling cycle. Following this initiating event, the RNS does not restart automatically, and the grid is not recovered within 1 hour.

The major contributors to risk given loss of offsite power (without grid recovery) are the following:

- Failure of the RNS pump to run or restart
- Failure of the diesel generator to start or run
- Failure of the main breaker to open
- Failure to recover ac power within 1 hour
- Failure of Ovation digital output modules for RNS-V055
- Common cause failure of the ADS 4th stage squib valves
- Common cause failure of batteries IDSA-DB-1A/1B
- Common cause failure to start engine-driven fuel pumps
- Common cause failure of the IRWST injection squib valves
- Common cause failure of the strainers in the IRWST tank
- Common cause failure of the recirculation sump strainers

Loss of Offsite Power Initiating Event During Drained Condition (with success of grid recovery within 1 hour)

This sequence is initiated by loss of offsite power during mid-loop/vessel flange operation which has an estimated duration of 120 hours per 18 months refueling cycle. Following this initiating event, the RNS does not restart automatically, the grid is recovered within 1 hour but manual RNS restart after grid recovery fails.

The major contributors to risk, given loss of offsite power (with grid recovery), are the following:

- Failure of the RNS pump to run or restart
- Common cause failure of the ADS 4th stage squib valves
- Failure of Ovation digital output modules for RNS-V055
- Common cause failure of the recirculation line squib valves
- Common cause failure of the IRWST injection squib valves
- Common cause failure of the strainers in the IRWST tank
- Common cause failure of the recirculation sump strainers

Conclusions

The conclusions drawn from the shutdown Level 1 study are as follows:

- The overall shutdown core damage frequency is very small (1.03E-07/year).

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- Initiating events during reactor coolant system drained conditions contribute approximately 95 percent of the total shutdown core damage frequency. Loss of decay heat removal capability (during drained condition) due to failure of the component cooling water system or service water system is the initiating event with the greatest contribution (approximately 77 percent of the shutdown core damage frequency).

- Common cause failures of in-containment refueling water storage tank components contribute approximately 56 percent of the total shutdown core damage frequency. Common cause failure of the in-containment refueling water storage tank valves contributes approximately 45 percent of the total shutdown core damage frequency.

- Common cause failures of the automatic depressurization system stage 4 squib valves contribute approximately 26 percent to the total shutdown core damage frequency. The function of the automatic depressurization system is important to preclude the effects of surge line flooding. This indicates that maintaining the reliability of the automatic depressurization system is important.

- Common cause failures of the containment sump recirculation squib valves contribute approximately 22 percent to the total shutdown core damage frequency. This function is important during drained conditions. This indicates that maintaining the reliability of the recirculation line squib valves is important.

- Human errors are not overly important to shutdown core damage frequency. There is no particular dominant contributor. Sensitivity results show that the shutdown core damage frequency would remain very low even with little credit for operator actions.

- One action, operator failure to recognize the need for reactor coolant system depressurization during safe/cold shutdown conditions, is identified as having a significant risk increase value. This indicates it is important that the procedures include this action and the operators understand and are appropriately trained for it.

- Individual component failures are not significant contributors to shutdown core damage frequency, and there is no particular dominant contributor. This confirms the at-power conclusion that single independent component failures do not have a large impact on core damage frequency for AP1000 and reflects the redundancy and diversity of protection at shutdown as well.

- The in-containment refueling water storage tank provides a significant benefit during shutdown because it serves as a passive backup to the normal residual heat removal system.

19.19.2 Large Release Frequency for Shutdown and Low-Power Events

The baseline PRA shutdown large release frequency (LRF) for AP600 was calculated to be 1.5E-08 per reactor-year, associated with a shutdown CDF of 9.0E-08 per year. The AP1000 LRF is estimated to be 1.72E-08 per year, with the same risk profile as that of AP600 (see Table 19.19-1). This LRF compares well with the AP1000 at-power LRF of 1.95E-08 per year.

19.19.3 Shutdown Results Summary

The results of the low-power and shutdown assessment show that the AP1000 design includes redundancy and diversity at shutdown not found in current plants. In particular, the in-containment refueling water storage tank provides a unique safety backup to the normal residual heat removal system. Maintenance at shutdown has less impact on the defense-in-depth features for AP1000 than for current plants. In accordance with plant technical specifications, safety-related system planned

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maintenance is performed only during those shutdown modes when the protection provided by the safety-related system is not required. Further, maintenance of nonsafety systems, such as the normal residual heat removal system, component cooling water system, and service water system, is performed at power to avoid adversely affecting shutdown risk. These contribute to the extremely low shutdown core damage frequency and the low large release frequency.

A quantitative internal events, internal flood, and internal fire PRA of the AP1000 design performed to estimate plant CDF and LRF during shutdown events provided the following results:

Table 19.19-1 Summary of AP1000 PRA during Shutdown Events Results

Events	Core Damage Frequency (per year)	Large Release Frequency (per year)
Internal Events	1.03E-07	1.72E-08
Internal Flood	3.22E-09	5.37E-10
Internal Fire	8.5E-08(1)	1.43E-08
Sum =	1.91E-07	3.20E-08

Note:
1. Internal fire during shutdown is evaluated quantitatively as a response to an NRC question and isnot reported elsewhere in this document.

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19.20 Seismic Margin Analysis

19.20.1 Introduction

In accordance with Section II.N, Site-Specific Probabilistic Risk Assessments and Analysis of External Events, of SECY-93-087 (Reference 19.20-1), the U.S. Nuclear Regulatory Commission (NRC) approved the following staff recommendations:

> "PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margin analysis will consider sequence-level High Confidence, Low Probability of Failures (HCLPFs) and fragilities for all sequences leading to core damage or containment failures up to approximately one and two-thirds the ground motion acceleration of the Design Basis SSE."

The PRA based seismic margin analysis (SMA) and the methodology described in this section is consistent with the recommendation of SECY-93-087.

Seismic margins methodology is employed to identify potential vulnerabilities and demonstrate seismic margin beyond the design-level safe shutdown earthquake (SSE). The capacity of those components required to bring the plant to a safe, stable condition is assessed. The structures, systems, and components identified as important to seismic risk are addressed. For this PRA-based seismic margin analysis, HCLPFs are calculated and reported at the sequence level. In addition, insights related to random and/or human failures are reported, as deemed appropriate, for each sequence.

19.20.2 Calculation of HCLPF Values

19.20.2.1 Seismic Margin HCLPF Methodology

The seismic margin analysis is based on established criteria, design specifications, existing qualification test reports, established basic design characteristics and configurations, and public domain generic data.

The seismic margin assessment is used to demonstrate margin over the SSE of 0.3g. Consistent with SECY-93-087 (Reference 19.20-1), the goal of the SMA is therefore to demonstrate that the plant HCLPF is at least 0.5g peak ground acceleration (pga). This is also called the review level earthquake (RLE). The AP1000 seismic response spectra are included in Tier 1, Chapter 5 (see Tier 1, Figures 5.0-1 through 5.0-4). It will be necessary for a COL (combined operating license) applicant to demonstrate that the seismic response for the applicant's plant is equal to or less than that used in the calculation of the HCLPF values, and to evaluate the potential for soil liquefaction using the applicant's site specific conditions. This will ensure a reserve margin that exceeds a 0.5g seismic level.

19.20.2.2 Calculation of HCLPF Values

A seismic margin analysis is made up of two major tasks:

1. A PRA-based model to determine the plant HCLPF
2. Determination of the plant structure and component HCLPFs

The second task, determination of HCLPF seismic acceleration values for plant structures and components, is discussed in this section; the PRA-based model is herein discussed as far as the seismic event trees and major assumptions associated with seismic fault trees development are

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concerned. The HCLPF values used in the analysis, which now include HCLPF values for hard-rock, high-frequency sites and soil sites, are summarized in Table 19.20-1.

19.20.2.2.1 Review of Plant Information

The assessment uses the following plant information:

- Structural and seismic design criteria and procedures
- Structural design calculations
- Layout and design drawings
- Test reports
- Piping and instrumentation diagrams
- Equipment design specifications
- Generic fragility data
- AP1000 plant response spectra.

19.20.2.2.2 System Analysis

Section 7.4 provides a discussion of the systems required for safe shutdown. The structures and components associated with these systems are considered in the seismic margin assessment. It is noted that the same success criteria as in the AP1000 PRA sensitivity case where no credit is taken for non-safety related systems, is used as the starting point for the AP1000 PRA-based seismic margins analysis. This success criterion is not necessarily defined in terms of reaching specific plant modes, but rather on reaching a sustainable safe plant state. The bases for these success criteria are given in the AP1000 PRA report (Reference).

19.20.2.2.3 Analysis of Structure Response

The purpose of a seismic fragility analysis is to define the maximum limit, seismic capacity, of functional capability or operability with the associated uncertainty for plant components and structures that could have an effect on safe shutdown of the plant following a seismic event. Capacity in the seismic margin assessment, expressed in terms of the free field peak ground level acceleration, is the level of the seismic event that results in failure of a given component or structure to perform its safety-related function. Failures leading to loss of safety function could result from such things as: loss of a pressure boundary; significant inelastic deformation; partial collapse; loss of support functions; or a combination of failure modes. In the calculation of the HCLPF value for a system, structure, or component, the governing failure mode is established by examining the different potential failure modes possible. Each failure mode has different reserve margin. As an example, ductility may be very large for tension failure, whereas, for buckling, ductility generally does not contribute to reserve margin.

A fragility evaluation is made for the key structures and components. The HCLPF for the equipment and structures is established using one of the following:

- Probabilistic fragility analysis
- Conservative deterministic failure margin (CDFM) method
- Test results
- Deterministic approach
- Generic fragility data

These methods are briefly discussed below.

Probabilistic Fragility Analysis

This method is used to define HCLPF values for structures such as:

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- Steam generator supports
- Reactor pressure vessel supports
- Pressurizer supports
- Containment vessel

There are many sources of conservatism and variability in the estimation of seismic peak ground acceleration capacity for seismic margin assessment. HCLPF values reflective of the seismic capacity are derived from median capacity using formulas based on the log-normal distribution. The HCLPF values reflect a 95-percent confidence (probability) of not exceeding a 5-percent probability of failure (Reference).

The HCLPF is defined by a lognormal probability distribution that is a function of median seismic capacity and composite standard deviation, βc:

$$\text{HCLPF} = \text{Median Capacity} \times e^{[-2.3 \times \beta c]}$$

The median seismic capacity is related to the mean seismic capacity by the expression:

$$\text{Median Capacity} = \text{Mean Capacity} \times e^{[-(\beta c \text{\^{}} 2)/2]}$$

The mean peak seismic ground capacity, A_m, is related to the stress and strength design margin factors by the following expression:

$$A_m = (\Pi_i [X_i])A_o$$

where:

A_m = Mean peak seismic ground capacity

X_i = i^{th} design mean margin factor

Π_i = Product notation

A_o = Nominal seismic peak ground capacity

It is noted that the composite standard deviation is equal to the root mean square of the composite standard deviation associated with each of the margin factors. That is:

$$\beta_c = \sqrt{[\Sigma_i (\beta_c)_i^2]}$$

The conservatisms and variability identified and considered in this assessment are associated with stress and strength margin factors. The basic grouping of margin factors are: deterministic strength factor; variable strength factors; material; damping; inelastic energy absorption, ductility; and analysis or modeling error.

Conservative Deterministic Failure Margin Method

The HCLPF values for the shield building and the exterior walls of the Auxiliary Building were calculated using the conservative deterministic failure margin approach. A finite element analysis was performed of the structures that considered cracking of the concrete and redistribution of the loads. Deterministic margin factors were defined for three items: strength; inelastic energy absorption; and damping.

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The polar crane HCLPF is calculated using the Westinghouse's design specification of Polar Crane and the vendor structural qualification calculation. The CDFM approach is used allowing the stress to reach yield and using a ductility factor of 1.25.

In addition, the HCLPF values for the Reactor Coolant Pump external heat exchanger and for the Passive Containment Cooling System are calculated with the CDFM approach.

Test Results

For the electrical equipment where documented test results are available, the HCLPF value is defined from comparison of required response spectra (RRS) and test response spectra (TRS). The method employed follows a deterministic approach using existing test data for similar types of equipment.

The existing test data was reviewed to determine a lower bound seismic capacity.

When the natural frequency of the equipment is not known, it was assumed that the natural frequency coincided with the required response spectra peak acceleration so that the lowest HCLPF value was calculated. It is noted that where equipment frequencies are known, and are used for comparing the RRS and TRS, these frequencies will be included in the design specification for the equipment to assure that the dynamic characteristics are the same as those expected.

Relay Chatter

Solid-state switching devices and electro-mechanical relays will be used in the AP1000 protection and control systems. Solid-state switching devices are inherently immune to mechanical switching discontinuities such as contact chatter. Robust electro-mechanical relays are selected for AP1000 applications such that inherent mechanical contact chatter is within the required system performance criteria. Therefore, contact chatter has no effect on system operation and was, therefore, not included in the seismic margin analysis. The COL must confirm the use of seismically robust electro-mechanical relays in the engineered safety features actuation and control systems.

Moreover, the loss of offsite power event has a very low HCLPF value (0.09g). The control rod motor generator sets are powered by AC load centers that are de-energized on loss of offsite power sources. When the control rod motor generator sets are de-energized, current to the magnetic jack mechanisms stops and the gripper coils open, allowing the rods to drop into the core. Therefore, relay chatter is not an issue for reactor trip.

Finally, passive residual heat removal (PRHR) and core makeup tank (CMT) system valves automatically fail open upon loss of instrument air due to loss of seismically induced loss of offsite power. Thus, relay chatter is not an issue for PRHR and CMT system functions.

Deterministic Approach

A lower bound estimate of the HCLPF is obtained for selected structures or equipment based on margin to design limit for the appropriate load combination defined by the fault tree logic. Where applicable, the increased capacity due to inelastic energy absorption is defined using the recognized and recommended ductility factor of 1.25.

This approach was used for the primary components to verify that their supports would control the HCLPF value. It was also used for a few cases to define the HCLPF when it was apparent that its seismic capacity would not control the plant HCLPF value. This approach was used for: containment baffle plate supports; Interior Containment Structure and IRWST; PRHR heat exchanger; core makeup tank; and valves.

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Generic Fragility Data

Generic fragility data was used when insufficient information was available to define the HCLPF value using one of the methods described above. Those cases where this approach was used were:

- Reactor internals and core assembly that includes fuel
- Control rod drive mechanism (CRDM)
- Reactor coolant pump
- Accumulator tank
- Piping
- Cable trays
- Valves
- Ceramic insulators

The Utility Requirements Document for Advanced Light Water Reactor, Reference , was used for all of the components listed above except ceramic insulators, which used recognized industry low seismic capacity data.

19.20.2.2.4 Evaluation of Seismic Capacities of Components and Plant

Table 19.20-1 provides the HCLPF values for the equipment, structures, and systems considered in the seismic margin evaluation. Also shown in this table is the approach used to define the HCLPF value, as described in Subsection 19.20.2.2.3. The evaluation considers the effect of uplift and sliding of the nuclear island basemat foundation. The nuclear island seismic response has been evaluated at 1.1 times the Review Level Earthquake (RLE) and was found to retain its stability against sliding and overturning.

In the design of the AP1000, careful consideration is given to those areas that are recognized as important to plant seismic risk. In addition to paying special attention to those critical components that have HCLPF values close to the review level earthquake, the design process considers potential interaction with both safety-related and nonsafety-related systems or structures, as well as adequate anchorage load transfer and structural ductility. The seismic margin evaluation provides a means of identifying specific equipment and/or structures that are vulnerable to beyond design basis seismic events.

Equipment qualification is the generation and maintenance of evidence to ensure that safety systems and equipment will operate on demand to meet system performance requirements during normal/ abnormal and accident environmental conditions. The methodology for qualification of safety-related electrical and mechanical equipment is defined in Appendix 3D and further expanded for seismic high frequency considerations in Appendix 3I. The intent of the qualification process defined in these Appendixes is to ensure a high reliability for equipment and system safety. Qualification by test, analysis or a combination of test and analysis is performed to verify the safety-related electrical and mechanical equipment will operate as intended under normal/abnormal and accident environmental conditions over the installed life. Details on the qualification process are provided to the equipment vendors in specifications and qualification methodology documents during procurement under a 10CFR50 Appendix B quality assurance program.

19.20.2.2.5 Verification of Equipment Fragility Data

The AP1000 safety-related equipment is designed to meet the safe shutdown earthquake requirements defined in Chapter 3. This seismic margin evaluation has focused on demonstrating that the design of the nuclear island structures, safety-related equipment, and equipment supports can carry the loads induced by the review level earthquake discussed here. This evaluation incorporates as-specified equipment data. After the plant has been built, it will be necessary to perform a verification of the seismic margin assessment for the installed conditions.

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19.20.2.2.6 Turbine Building Seismic Interaction

As part of the seismic margin assessment, the seismic interaction between the turbine building and the nuclear island was evaluated according to guidance provided in Reference . It was determined that:

- To protect the adjacent nuclear island auxiliary building the first bay of the turbine building has been classified as seismic category II.
- It is not likely that the size and energy of debris from the turbine building will be large enough to result in penetration through the auxiliary building roof structure.

Even though it is not likely that penetration of turbine building debris could be large enough or have sufficient energy for penetration through the auxiliary building roof structure, this event was evaluated. The consequences of damage to the safety-related equipment in the auxiliary building were investigated. It was determined from this investigation that should an event occur that causes the failure of equipment in the upper elevations of the auxiliary building, the results of the seismic margin assessment, the plant HCLPF value, and the insights derived from the seismic margin assessment are not affected. Moreover, the steam line break events, which would result from the damage of equipment in the upper elevations, are not dominant contributors to the core damage frequency. Further, the loss of equipment in the upper elevations will not affect the passive safety systems that would be used to put the plant in a safe shutdown condition should an event occur.

19.20.3 Seismic Margin Model

In this section, the AP1000 Risk-Based Seismic Margins Model is summarized and the plant HCLPF for AP1000 is determined.

HCLPFs are calculated for the seismic Category I safety-related systems that are called upon via the seismic event trees to mitigate an accident caused by the initiating seismic event.

19.20.3.1 Major SMA Model Assumptions

In this section, the general characteristics and major assumptions of the AP1000 SMA model are discussed.

1. The seismic event is assumed to occur while the plant is operating at full power.
2. A review level earthquake equal to 0.5g is used for the seismic margin analysis.
3. It is assumed that the seismic event would result in loss of offsite power since the AC power equipment is not seismic Category I. (The offsite insulators on the feed lines from the offsite power grid fail such that a loss of offsite power occurs.) No credit is taken for onsite emergency AC power (diesel generators).
4. No credit is taken for non-safety related systems. They are assumed to have failed or be non-functional due to the seismic event. This includes all equipment in the turbine building and the turbine building itself; as discussed in Subsection 19.20.3.3, structural failure of the turbine building is assumed not to impact the structural integrity of the adjacent auxiliary building.
5. The seismically induced SMA initiating event categories and their event trees are taken from the AP600 PRA model. For each initiating event, the PRA logical modeling (i.e., seismic event and fault trees) developed for AP600 structures, systems, and components have been

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used as the starting point and their applicability to the AP1000 design has been assessed and confirmed. The applicability of the base AP600 to the AP1000 has been addressed in a supporting calculation. Cutsets associated with each sequence are generated and then the min-max method is used to calculate the plant HCLPF value.

19.20.3.2 Seismic Initiating Events

The first step in Seismic Margins Model is to evaluate which initiating events could occur as a result of a seismic event. For this purpose, a Seismic Initiating Event Hierarchy Tree is constructed. This event tree is given in Figure 19.20-1 and discussed below. Based on this hierarchy event tree, seismic initiating event categories are defined and their event tree models are constructed (as discussed in Subsection 19.20.3.3).

Given that a seismic event occurs, the hierarchy event tree is constructed such that the seismically-induced initiating event with the most challenge to the plant safety systems is considered first: gross structure collapse. This category is labeled as EQ-STRUC and is the first initiating event category to be modeled and quantified.

If gross structure collapse does not occur, next the reactor coolant system (RCS) loss-of-coolant-accident (LOCA) category in excess of emergency core cooling system (ECCS) capacity (also termed as “Vessel Failure”) is considered. This category is labeled as EQ-RVFA.

If vessel failure does not occur, then large RCS LOCAs are considered. This category is labeled as EQ-LLOCA.

If EQ-LLOCA does not occur, then small RCS LOCAs are considered. This category is labeled as EQ-SLOCA. Steam generator tube rupture (SGTR) and large secondary line break (SLB) events are folded into the small LOCA category, as discussed in Subsection 19.20.3.3.

Next considered is the seismically induced anticipated transient without scram (ATWS) event. This event is labeled as EQ-ATWS.

Finally, all other transients are considered in the category labeled EQ-LOSP. The seismically induced LOSP event occurs at low HCLPF values (e.g., lower than the SSE at 0.3g) and does not affect the plant HCLPF, as discussed in Subsection 19.20.4.2. The cutsets for this event are all “mixed cutsets,” containing seismically induced initiating event coupled with random failures leading to core damage. This event is included in the model for additional insights and completeness.

Thus, the hierarchy tree defines six initiating event categories. Each of these is discussed and an event tree for each is constructed in Subsection 19.20.3.3.

The PRA-based seismic margins analysis does not consider seismic hazard curves. Therefore, initiating event frequencies are not calculated for each seismically generated initiating event category. Although seismically generated initiating event frequencies are not calculated, it is important to evaluate the seismic vulnerability of the components and systems that contribute to the initiating event categories. This is done by estimating a HCLPF for each seismic initiating event category, as discussed in Subsection 19.20.3.3.

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19.20.3.3 Seismic Event Trees

The six seismically induced initiating event categories defined by the hierarchy event tree model of Subsection 19.20.3.2 are further discussed to model seismically induced failures that will determine the HCLPF for each of these initiating events. The six categories considered are:

1. EQ-STRUC	Gross structural collapse
2. EQ-RVFA	LOCA in excess of emergency core cooling system capacity
3. EQ-LLOCA	Large LOCA
4. EQ-SLOCA	Small LOCA
5. EQ-ATWS	ATWS
6. EQ-LOSP	Loss of offsite power

The small LOCA category also covers SGTR and SLB events. As discussed later in the success paths, the SLOCA success path used for SMA is also applicable (conservatively) to the SGTR and unisolated SLB events given that only safety-related systems are credited and considered in the PRA-based SMA.

The last event, LOSP, is postulated at 0.09g. This event may also be viewed to represent a larger family of transients associated with loss of main feedwater, loss of compressed air, turbine trip, reactor trip, loss of service water/component cooling water, etc, following a seismic event and LOSP since no credit is taken for these non-safety systems in the SMA models. Moreover, a seismically induced transient containing LOSP, becomes a station blackout (SBO) event since no credit is taken for diesel generators that are not seismically qualified.

Each of the SMA events are further discussed below.

1. EQ-STRUC (Gross Structural Collapse)

   This event includes seismically induced failures of AP1000 structures that may result in core damage and large fission product release.

   The AP1000 structures are classified in 5 groups:

   1. Nuclear Island

      This consists of the containment, shield building, and auxiliary building.

      Nuclear island is structurally designed to meet seismic Category I.

   2. Turbine Building

      The first bay (the portion of the turbine building adjacent to the nuclear island outlined in Table 3.2-2 and including the portion of the basemat under this area) of the turbine building is classified as Seismic Category II, and the main area of the turbine building structure is designed to meet the International Building Code. For the SMA model, it is assumed to have failed. Thus no credit is taken for systems in this building.

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3. Annex Building

   The high rise portion of the annex building is designed to meet seismic Category II. For the SMA model, it is assumed to have failed. Thus, no credit is taken for systems in this building.

4. Diesel Generator Building

   The diesel generator building is designed to meet the UBC. For the SMA model, it is assumed to have failed. Thus, no credit is taken for systems in this building.

5. Radwaste Building

   The radwaste building is designed to meet the UBC. For the SMA model, it is assumed to have failed. Thus, no credit is taken for systems in this building.

Thus, only the nuclear island is considered for the SMA model; the interaction between the other buildings and the nuclear island is assumed to have no detrimental effect on the nuclear island structures. This assumption needs to be verified by a plant walkdown when an AP1000 plant is built.

The failures of the nuclear island structures are modeled in terms of the driving structures of the steel containment vessel, the shield building, and the auxiliary building.

The EQ-STRUC event tree is shown in Figure 19.20-2; HCLPF value for EQ-STRUC is calculated in Subsection 19.20.4.

2. EQ-RVFA (LOCA in Excess of ECCS Capacity)

   This event represents the “vessel failures” where the event leads to excessive loss of RCS inventory that can not be made up by the ECCS capacity. In this case, core damage is postulated. A complete dependency between seismic induced failures of SSCs that share basic characteristics (i.e., component type, location/elevation, etc.), the “vessel failure” event comprises the following types of structural and component failures:

   1. Seismically induced failures of the reactor vessel
   2. Seismically induced failures of the steam generators
   3. Seismically induced failures of the other RCS components
   4. Seismically induced failures of two direct vessel injection (DVI) lines
   5. Seismically induced failures of fuel.

   The EQ-RVFA event tree is shown in Figure 19.20-3; HCLPF value for EQ-RVFA is calculated in Subsection 19.20.4.

3. EQ-LLOCA (Large LOCA)

   Seismically induced large LOCA initiating event category, EQ-LLOCA, contains RCS breaks with break sizes greater than 9 inches. Since the seismic event failures assume that if one pipe breaks by a seismic event, all redundant similar pipes will break at the same time, all major RCS pipe breaks are conservatively included in this category; thus, no medium LOCA is defined in the initiating event hierarchy tree. Also included in this category are the failures of the PRHR heat exchanger by a seismic event.

   The EQ-LLOCA event tree is shown in Figure 19.20-4; HCLPF value for EQ-LLOCA is calculated in Subsection 19.20.4.

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4. EQ-SLOCA (Small LOCA)

   Seismically induced small LOCA initiating event category, EQ-SLOCA, contains RCS breaks with break sizes less than 2 inches of equivalent diameter. Since the seismic event failures assume that if one pipe breaks by a seismic event, all redundant similar pipes will break at the same time, all major RCS pipe breaks are conservatively included in the large LOCA category. For the small LOCA category, RCS leaks from instrument lines are used as the representative event. The small LOCA category also includes and bounds events such as

   - Steam Generator Tube Rupture (SGTR)
   - Large Steam Line Breaks (SLB) (due to generation of S signal and RCS inventory shrinkage)

   For SGTR events, breaks of one or more (up to 5) tubes have been considered for the AP1000 design. An event with 5 steam generator tubes rupturing has an equivalent LOCA break flow area of a 1.46 inch diameter hole. The rupture of more than 5 tubes by a seismic event is conservatively bounded by the structural failure of a steam generator, which is included in the EQ-RVFA initiating event.

   Due to the modification of the Reactor Coolant Pump (RCP) Heat Exchanger (HX) from the AP600 design to the AP1000 design, an additional entry is added to the seismic induced Small LOCA. This reflects the possibility that in the event of a RCP HX pipe break, a small LOCA will be induced. Flow from the RCS inventory will be restricted by the labyrinth seal surrounding the RCP motor shaft; tolerances on the labyrinth seal allow for a maximum flow area of $1.389 in^2$. This corresponds to approximately a 1.3 inch pipe break. A postulated seismic induced break of all eight tubes does not change the equivalent break flow rate for each pump and when considering the break in all pumps, a total of approximately 2.7 inch pipe break equivalent LOCA needs to be considered. This is judged to be consistent with the definition of seismically induced small LOCA given above.

   The EQ-SLOCA event tree is shown in Figure 19.20-5; HCLPF value for EQ-SLOCA is calculated in Subsection 19.20.4.

5. EQ-ATWS (Anticipated Transients without Scram)

   The EQ-ATWS event addresses the seismically induced ATWS initiating event related to the failure of the core assembly or guide tubes or the control rod drive systems to remain functional so that the rods can not fall into the core. The fuel is still intact and can be cooled. The failure mode associated with seismically induced fuel failure has been already addressed in EQ-RVFA event.

   Because offsite power is postulated to have been lost, the control rod motor generator sets would be de-energized even if the reactor trip function failed. If the core assembly or the control rod system failed, the rods are postulated to fail to insert into the core.

   The EQ-ATWS event tree is shown in Figure 19.20-6; the HCLPF value for EQ-ATWS is calculated in Subsection 19.20.4.

6. EQ-LOSP (Loss of Offsite Power)

   The EQ-LOSP event addresses the seismically induced loss of offsite power. This event occurs at relatively low intensity earthquakes. The driving failure for loss of offsite power is represented by failure of ceramic insulators in the switchyard. The HCLPF value for these insulators is 0.09g,

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which is lower than the review level earthquake of 0.5g, and the plant SSE of 0.3g. Such an earthquake does not challenge any of the safety-related systems that are built to withstand the SSE and have margin for higher g levels. Thus, this event does not lead to purely seismically driven failure combinations for a core damage sequence. This event model contains only “mixed cutsets” for core damage; these are failure combinations of seismically induced initiating event coupled with random failures of safety-related systems.

The EQ-LOSP event tree is shown in Figure 19.20-7; this event does not contribute to plant HCLPF.

19.20.3.4 Seismic Fault Trees

System fault trees for mitigation functions have been modified to account for seismically-induced failures. The AP600 system seismic fault trees have been reviewed for applicability to the AP1000 and only limited and minor changes have been deemed necessary.

19.20.4 Calculation of Plant HCLPF

This section presents the SMA calculations based on the model developed in Subsection 19.20.3.

The initiating event HCLPFs are calculated in Subsection 19.20.4.2. The plant HCLPF is calculated in Subsection 19.20.4.3.

The analysis demonstrates that all structures and components required to maintain the plant in a safe stable state are expected to function following a seismic event of 0.5g acceleration.

19.20.4.1 HCLPFs for Basic Events

The HCLPF values for various AP1000 structures and components were determined in a supporting calculation and are given in Table 19.20-1. The basic events defined in the SMA model for seismic failures are assigned their own HCLPF values, as shown in Table 19.20-2. These HCLPF values are taken from Table 19.20-1. When not self-evident, the “Source” column in Table 19.20-2 explains how the information Table 19.20-1 has been used.

For reasons beyond the development of the PRA-based AP1000 SMA, Table 19.20-1 groups all the electrical equipment into two major categories: “Non-Sensitive to High Frequency Excitation” and “Sensitive to High Frequency Excitation”. For the purposes of the PRA-based SMA, all electrical equipment has been assumed to be from the limiting categories among the two, which has an HCLPF value of 0.5; this assumption is for the purposes of this analysis only and is conservative for this purpose.

19.20.4.2 Calculation of Initiating Event HCLPFs

Initiating event HCLPFs are calculated by assigning the HCLPF values from Table 19.20-2 to the seismically induced failures modeled in Subsection 19.20.3.3 for initiating events. The HCLPF associated to the initiating events will be the minimum among those for each of the potential initiator; the results of these calculations are given in Tables 19.20-3 through 19.20-7; results are presented for the AP1000 before and after this modification for DCD Revision 17. EQ-IEV-LOSP is already assigned a HCLPF 0.09g, representing the failure of ceramic insulators but it does not contribute to plant HCLPF since it has only mixed cutsets (seismic and random failures combined in cutsets).

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The initiating event HCLPFs are summarized below:

Initiating Event	HCLPF	Dominated by
EQ-IEV-STRUC	0.55g	Polar crane
EQ-IEV-RVFA	0.50g	Fuel and pressurizer failure
EQ-IEV-LLOCA	0.81g	RCS piping
EQ-IEV-SLOCA	0.54g	Steam generator tube failure
EQ-IEV-ATWS	0.50g	Core assembly failures
EQ-IEV-LOSP	0.09g	Ceramic insulator failure

When the min-max method is used, the HCLPF of seismic sequences resulting from an initiating event can not be less than the initiating event HCLPF since it appears in every cutset. If the initiating event is postulated to lead directly to core damage, the IE HCLPF is used in the determination of the plant HCLPF.

Since both EQ-STRUC and EQ-RVFA events are postulated to lead to core damage, and EQ-STRUC is postulated to go to large early release as well, plant HCLPF can be determined at this point to be at least 0.50g for core damage and at least 0.55g for large, early release consequences.

19.20.4.3 Calculation of AP1000 Plant HCLPF

The final AP1000 plant HCLPF calculation also considers the mitigation portion of the PRA logic. Even though this is not going to change the values identified in Subsection 19.20.4.2, the complete calculation provides further insights on the seismic margin of the AP1000 design.

All basic events in the AP1000 SMA model (listed in Table 19.20-2) are assigned a dummy probability value of 0.5; the model is then quantified and cutsets are generated. The min-max approach is then applied to the obtained cutsets at each failure sequence level to evaluate the sequence HCLPF value, the event tree HCLPF value and the overall plant HCLPF value.

The cutset generated from the SMA model are listed and analyzed through the min-max approach discussed above in a supporting calculation. Sequence level results are presented in Table 19.20-8 where also the plant level HCLPF value is presented.

19.20.5 Sensitivity Analyses

A 99% confidence associated with the test response spectra is expected for all the HCLPF extracted from tests (method [6] in Table 19.20-1). To address this expectation a sensitivity case was run to the AP1000 PRA-based SMA.

Since electrical equipment is tested and qualified to the SSE (i.e., 0.30g), the HCLPF values in Table 19.20-1 for all tested equipment are set to 0.3g. While the selected values are extremely conservative due to the engineering margins normally adopted for the qualification tests, such values would not change either the overall AP1000 plant HCLPF value or any sequence or event tree level HCLPF value.

The Polar Crane HCLPF value dominates the plant level HCLPF for the Gross Structural Collapse initiating event. Therefore, the fragility analysis of the polar crane was performed using both CDFM and PRA-based fragility analysis. It was demonstrated that the calculated HCLPF values from these two methods are above 0.5g and have a difference of less than 5%.

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19.20.6 Results and Insights

19.20.6.1 AP1000 SMA Results

The AP1000 PRA-based SMA has demonstrated that for structures, systems, and components required for safe shutdown, the HCLPF magnitudes are equal to or greater than 0.50g. This HCLPF is determined by various structures, systems, and components with an HCLPF value of 0.5g.

Thus, the AP1000 plant can meet or exceed the requirement to withstand a review level earthquake of 0.5g. It is observed that electrical equipment qualification consistent with the Certified Seismic Design Response Spectra (CSDRS) at 0.3g (with a 99% confidence associated to the Test Response Spectra – TRS) supports the overall plant HCPLF value of 0.5g.

The success paths used for the SMA are taken conservatively in many cases, and credit for operator actions for events at 0.5g review level earthquake has been avoided. Thus, the results are valid without operator intervention, which indicates a strong point of the AP1000 design to mitigate seismically induced core damage and large release sequences.

All SMA sequences are evaluated with loss of offsite power and loss of onsite AC power leading to a station blackout event. The plant design is shown to be robust against seismic event sequences each of which contain station blackout coupled with other seismic or random failures.

19.20.6.2 AP1000 SMA Insights

The SMA results also point out the following insights:

1. Design Features

   The AP1000 design provides some aspects that make the plant more robust against the review level earthquakes. Namely:

   - Reactor trip is ensured without the actuation signal due to the loss of offsite power occurring and rods inserting by gravity.
   - PRHR system valves fail open without actuation signal following loss of power/loss of instrument air. Thus, PRHR cooling is immediately available.
   - CMT system valves fail open without actuation signal following loss of power/loss of instrument air. Thus, CMT injection is immediately available.

   Thus, three key mitigating systems, reactor trip, PRHR cooling, and CMT injection are available with high confidence and low probability of failure, without dependence on actuation signals immediately after a review level seismic event.

   Moreover, the passive containment cooling system air operated valves also fail open in a review level earthquake, due to loss of offsite power/instrument air. As a result, the passive containment cooling system is automatically actuated and has enough water inventory to last for 72 hours.

2. DC System Fragility

   Control rods, PRHR, CMT, and passive core cooling systems would be operational after potential loss of protection and safety monitoring system (PMS) or DC control power. Thus, the plant can successfully mitigate a transient event even with a failure of PMS or DC control power. However, the DC control power system HCLPF is the same as the plant HCLPF (0.50g). This HCLPF has

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the potential to become a driving failure, if it were to be coupled with a LOCA event with low HCLPF. However, no such low HCLPF LOCA events are identified in the current model.

3. Importance of Valve Room Fragilities

   Fragility of certain valve rooms, where the passive core cooling system valves are concentrated, becomes an important factor; the SMA model depends on the successful functioning of these valves to mitigate LOCA accidents. These rooms are labeled as 11206/11207 and contain CMT, accumulator, IRWST injection, and cavity recirculation valves. Since the HCLPF of these rooms is relatively high, compared to the plant HCLPF value, the seismic failure of many passive core cooling system valves does not become a contributor to plant HCLPF.

4. Operator Actions

   Operator actions are not credited in the SMA model for the 0.50g review level events. Inclusion of operator actions in the models would provide additional success paths, such as manual actuation of the automatic depressurization system (ADS) after failure of CMTs to inject. However, this inclusion would not affect the plant HCLPF or the major conclusions of the SMA. Thus, the AP1000 design is already robust with respect to its response to seismic events, even without taking credit for operator actions.

5. IRWST Failure

   This failure is modeled to render PRHR, gravity injection, and recirculation systems inoperable. Thus, it becomes a single point failure that affects both the transient (e.g. LOSP events) and LOCA success paths. Failure of IRWST is modeled as a part of gross structural failure, as well as in PRHR and gravity injection system fault trees. The IRWST HCLPF is 0.71g and therefore significantly above the plant level HCLPF.

   Additionally, an argument can be made that when the IRWST fails, its inventory would end up in the containment cavity and can be used to recirculate cavity water back into the RCS, leading to successful core cooling. Although this scenario is plausible and credible, such success sequences (e.g. sequences where gravity injection is skipped, directly going into cavity recirculation) are not analyzed in the AP1000 PRA. For this purpose, no credit for such a success path is taken in the present model.

6. Large Fission Product Release

   The large fission product release is driven by the same seismic sequences that dominate the plant core damage. This is due to either the nature of the initiating event (such as gross structural failure initiating event, EQ-STRUC), or postulated containment failure following a reactor vessel failure (RVFA) (such as EQ-RVFA initiating event or some ATWS sequences leading the RVFA). Failure of containment isolation or containment cooling system due to their system components or system actuation failures does not dominate the plant large release HCLPF.

19.20.6.3 Not Used

19.20.7 References

19.20-1. “SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” USNRC Memorandum, July 21, 1993, Chilk to Taylor.

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19.20-2. Budnitz, R. J., et al., “An Approach to the Quantification of Seismic Margins in Nuclear Power Plants,” NUREG/CR-4334, UCID-20444, August 1985.

19.20-3. Advanced Light Water Reactor Utility Requirements Document, Volume III, ALWR Passive Plant, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules, Revisions 5 & 6, Issued December 1993.

19.20-4. “A Methodology for Assessment of Nuclear Power Plant Seismic Margin,” Electric Power Research Institute, EPRI NP-6041, October 1988.

19.20-5. APP-GW-GL-022, Revision 8, AP1000 Probabilistic Risk Assessment, Westinghouse Electric, LLC, August 2004 [ML042230487].

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Table 19.20-1 (Sheet 1 of 2)
Seismic Margin Parameters and HCLPF Values

Description	Median pga[1]	βc	HCLPF Value[1]	Basis
Buildings/Structures				
Shield Building – Tension Ring	-	-	0.73	[2]
Shield Building – Air Inlet	-	-	0.71	[2]
Shield Building – Conical Roof	-	-	0.71	[2]
Shield Building – PCS Tank	-	-	0.81	[2]
Shield Building – SC/RC Connection	-	-	>0.67	[2]
Shield Building – RC Cylindrical Wall	-	-	0.67	[2]
Steel Containment Vessel – Buckling	1.94	0.42	0.73	[3]
Steel Containment Vessel – Overturning	5.74	0.62	1.38	[3]
Containment Baffle – Support Failure	-	-	0.91	[4]
Interior Containment Structure & IRWST Tank	-	-	0.71	[4]
Exterior Walls of Auxiliary Building – Wall 1	-	-	0.97	[2]
Exterior Walls of Auxiliary Building – Wall 11	-	-	0.88	[2]
Primary Components				
Reactor Pressure Vessel	-	-	0.56	[4]
Reactor Pressure Vessel Supports	1.58	0.35	0.71	[3]
Reactor Internals and Core Assembly (includes fuel)	1.5	0.51	0.5	[5]
Control Rod Drive Mechanism (CRDM)	2.2	0.51	0.7	[5]
Steam Generator	-	-	0.54	[4]
Steam Generator Support Column Buckling	1.14	0.33	0.54	[3]
Steam Generator Lower Lateral Support	1.23	0.34	0.57	[3]
Steam Generator Intermediate Supports	1.17	0.30	0.59	[3]
Pressurizer	-	-	0.58	[4]
Pressurizer Upper Support Weld[10]	1.02	0.31	0.50	[3]
Pressurizer Upper Support Strut	1.11	0.29	0.56	[3]
Pressurizer Lower Support Strut	1.41	0.29	0.72	[3]
Reactor Coolant Pump[9]	2.2	0.51	0.68	[5]
Reactor Coolant Pump Heat Exchanger[9]	-	-	0.55	[2]
Mechanical Equipment				
Polar Crane	-	-	0.55	[2]

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Table 19.20-1 (Sheet 2 of 2)
Seismic Margin Parameters and HCLPF Values

Description	Median pga[1]	βc	HCLPF Value[1]	Basis
Piping – Support Controlled	3.3	0.61	0.81	[5]
Cable trays – Support Controlled	2.2	0.61	0.54	[5]
Accumulator Tank	2.2	0.46	0.76	[5]
Core Make Up Tank	-	-	0.87	[4]
Heat Exchanger (PRHR)	-	-	1.11	[4]
Valves				
Higher than El. 100′	3.3	0.61	0.81	[5]
Equal to or Lower than El. 100′	-	-	1.02	[4]
Passive Containment Cooling System	-	-	0.67	[2]
Electrical Equipment				
Non-Sensitive to High Frequency Excitation	-	-	0.5	[6]
Sensitive to High Frequency Excitation	-	-	0.52	[6]
Ceramic Insulators[7]	0.2	0.35	0.09	[8]

Notes of Table 19.20-1:

[1] pga is the free field peak ground acceleration level for the seismic event.
[2] HCLPF based on conservative deterministic fragility margin approach.
[3] HCLPF based probabilistic fragility analysis.
[4] HCLPF based on deterministic approach.
[5] HCLPF based on URD recommended generic fragility data.
[6] HCLPF based on design margin, code requirements and test margins inherent to the seismic qualification testing. Qualification testing with 99% confidence on the TRS will be limited to 0.3g.
[7] The capacity of the ceramic insulators is less than the review level earthquake of 0.5g. The failure of the ceramic insulators is considered in the PRA analysis.
[8] HCLPF based on recognized generic fragility data
[9] Both the Reactor Coolant Pump Support and Reactor Coolant Pump External Heat Exchanger HCLPF values are controlled by Steam Generator Support.
[10] The HCLPF value of the Pressurizer Upper Support Weld is calculated as 0.6 g using conservative deterministic failure margin method. The value of 0.5 g in the table is used in the PRA/SMA and is more conservative.

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Table 19.20-2 (Sheet 1 of 5)
Basic Events HCLPF Values

BE ID	BE Description	HCLPF (g)	Source
EQ-AB-EXTWALL	Failure of Auxiliary Building Exterior Wall	0.88	Exterior walls of auxiliary building, limiting values between *wall 1* and *wall 11*
EQ-AB-FLOOR	Failure of Auxiliary Building Floor	0.88	Same as auxiliary building exterior wall
EQ-AB-INTWALL	Failure of Auxiliary Building Interior Wall	0.88	Same as auxiliary building exterior wall
EQ-ACC-CV28	Accumulator Check Valves 28A and 28B Fail	1.02	In rooms 11206/11207, below elevation 100′
EQ-ACC-CV29	Accumulator Check Valves 29A and 29B Fail	1.02	In rooms 11206/11207, below elevation 100′
EQ-ACC-TANKS	Accumulator Tanks Fail	0.76	
EQ-ACDISPANEL	120 Volt AC Distribution Panels Fail	0.5	Limiting value among those provided for electrical equipment
EQ-ADS-S1MOVS	ADS Stage 1 MOVs RCS-PL-V001A/B and RCS-PL-V011A/B Fail	0.81	In rooms 11603/11703, above elevation 100′
EQ-ADS-S2MOVS	ADS Stage 2 MOVs RCS-PL-V002A/B and RCS-PL-V012A/B Fail	0.81	In rooms 11603/11703, above elevation 100′
EQ-ADS-S3MOVS	ADS Stage 3 MOVs RCS-PL-V003A/B and RCS-PL-V013A/B Fail	0.81	In rooms 11603/11703, above elevation 100′
EQ-ADS-S4VALVES	ADS Stage 4 Squib Valves 4A/B/C/D Fail	0.81	In rooms 11301/11302, above elevation 100′
EQ-BAF-SUPP	Failure of Containment Baffle Support	0.91	
EQ-BAT-RACK	Battery Racks Fail	0.5	Limiting value among those provided for electrical equipment.
EQ-BATTERY	250 Vdc Batteries Fail	0.5	Limiting value among those provided for electrical equipment.

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Table 19.20-2 (Sheet 2 of 5)
Basic Events HCLPF Values

BE ID	BE Description	HCLPF (g)	Source
EQ-CABINETS	PMS Cabinet Fail	0.5	Limiting value among those provided for electrical equipment.
EQ-CABLETRAY	Cable Trays Fail	0.54	
EQ-CAS-AOV-1415	Containment CAS Isolation Valves AOV 14 and 15 Fail	0.81	In rooms 12405/11400, above elevation 100′
EQ-CER-INSULATOR	Seismically induced failure of ceramic insulators	0.09	
EQ-CMT-AOV	CMT AOV 14A/B and 15A/B Fail by Seismic Event	1.02	In rooms 11206/11207, below elevation 100′
EQ-CMT-CV	CMT CV 16A/B or 17A/B Fail by Seismic Event	1.02	In rooms 11206/11207, below elevation 100′
EQ-CMT-LEVELSWT	CMT Level Switch Fails	0.5	Limiting value among those provided for electrical equipment.
EQ-CMT-TANKS	CMT Tanks Fail by Seismic Event	0.87	
EQ-CONTPR-SENSOR	Containment Pressure Sensor or Transmitter Fails	0.5	Limiting value among those provided for electrical equipment.
EQ-CORE-ASSEMBLY	Failure of Core Assembly	0.5	
EQ-CRDM	Failure of Control Rod Drive Mechanism	0.7	
EQ-CV-BUCKLE	Containment Vessel Buckling	0.73	
EQ-CV-INTER	Failure of the Interior (concrete) Structure of Containment	0.71	
EQ-CV-OVERT	Containment Vessel Overturning	1.38	
EQ-DCDISPANEL	250 Vdc Distribution Panel Fails	0.5	Limiting value among those provided for electrical equipment.
EQ-DCMCC	DC Motor Control Centers Fail	0.5	Limiting value among those provided for electrical equipment.

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Table 19.20-2 (Sheet 3 of 5)
Basic Events HCLPF Values

BE ID	BE Description	HCLPF (g)	Source
EQ-DC-SWBRD	250 Vdc Switchboard Fails	0.5	Limiting value among those provided for electrical equipment.
EQ-DVI-PIPES	Seismically Induced Failure of Both DVI Lines	0.81	
EQ-ELECTRONICS	PMS Electronic Fail	0.5	Limiting value among those provided for electrical equipment.
EQ-INSTR-PIPES	Failure of RCS Instruments Lines	0.81	
EQ-INVERTER	250 Vdc Inverters Fail	0.5	Limiting value among those provided for electrical equipment.
EQ-IRW-INJCV	IRWST Injection CV 122A/B and 124A/B Fail by Seismic Event	1.02	In rooms 11206/11207, below elevation 100′
EQ-IRW-INJSQ	IRWST Injection Squib Valves 123A/B and 125A/B Fail by Seismic Event	1.02	In rooms 11206/11207, below elevation 100′
EQ-IRW-RECCV	Sump Recirculation Check valves 119A/B Fail by Seismic Event	1.02	In rooms 11206/11207, below elevation 100′
EQ-IRW-RECMOV	Sump Recirculation MOVs 117A/B Fail by Seismic Event	1.02	In rooms 11206/11207, below elevation 100′
EQ-IRW-RECSQ	Failure of Recirculation Squib Valves 118A/B and 120A/B by Seismic Event	1.02	In rooms 11206/11207, below elevation 100′
EQ-IRWST-TANK	Failure of IRWST	0.71	
EQ-MSL-SENSOR	Main Steam Line Pressure Sensor or Transmitter Fails	0.5	Limiting value among those provided for electrical equipment.
EQ-PCC-TANK	Passive Containment Core Cooling Tank Fails	0.81	
EQ-POL-CRANE	Failure of the Polar Crane	0.55	
EQ-PRHR-AOV	Passive RHR AOVs PXS-PL-V108A and B Fail by Seismic Event	0.81	In room 11300, above elevation 100′
EQ-PRHR-HX	Failure of Passive RHR Heat Exchanger	1.11	

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Table 19.20-2 (Sheet 4 of 5)
Basic Events HCLPF Values

BE ID	BE Description	HCLPF (g)	Source
EQ-PRZR-FAILS	Seismically Induced Failures of the Pressurizer	0.5	Pressurizer upper support weld (limiting HCLPF among pressurizer components)
EQ-PRZR-LVTRANS	Seismically Induced Failure of Pressurizer Level Transmitter	0.5	Limiting value among those provided for electrical equipment.
EQ-PRZR-SENSOR	Pressurizer Sensor Or Transmitter Fails	0.5	Limiting value among those provided for electrical equipment.
EQ-PRZR-SV	Pressurizer Safety Valves RCS-PL-V005A/B Fail Seismically	0.81	In rooms 11603/11703, above elevation 100′
EQ-RCP-FAILS	Reactor Coolant Pumps Fail	0.54	Same as SG due to connection between RCP & SG.
EQ-RCP-HX	Seismically Induced RCP HX Failure Inducing a LOCA	0.55	
EQ-RCS-PIPES	Failure of RCS Piping	0.81	
EQ-RV-FAILS	Reactor Pressure Vessel Fails	0.56	
EQ-RV-FUEL	Fuel in Reactor Vessel Fails	0.5	
EQ-RV-HDPK	Reactor Vessel Integrated Head Package Fails	0.7	Same as CRDM due to physical location
EQ-SG-FAILS	Seismically Induced Failures of the Steam Generators	0.54	
EQ-SGTR	Seismically Induced SGTR	0.54	Same as SG failure
EQ-SHBLD-ROOF	Shield Building Roof Fails	0.71	
EQ-SHBLD-WALL	Shield Building Wall Fails	0.71	Same as roof
EQ-SLB	Failure of Feed and Steam Pipes on Secondary Side	0.81	
EQ-TRSFSWITCH	Transfer Switches Fail	0.5	Limiting value among those provided for electrical equipment.

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Table 19.20-2 (Sheet 5 of 5)
Basic Events HCLPF Values

BE ID	BE Description	HCLPF (g)	Source
EQ-VFS-AOV-0304	Containment Air Filtration System Containment Air Supply Isolation Valves AOV 03 and 04 Fail	0.81	In rooms 12452/11400, above elevation 100′
EQ-VFS-AOV-0910	Containment Air Filtration System Containment Air Exhaust Isolation Valves Fail (009, 010, 800A/B, and 803A/B)	0.81	In rooms 12452/11400, above elevation 100′
EQ-WLS-AOV-5557	WLS Cont. Sump Isolation Valves AOV 55 and 57 Fail	0.81	In rooms 11300/12244, above elevation 100′

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Table 19.20-3
EQ-IEV-STRUC (EQSTR-02) HCLPF

		Original AP1000	Updated AP1000
1	EQ-AB-FLOOR	0.51g	0.88g
2	EQ-AB-EXTWALL	0.51g	0.88g
3	EQ-AB-INTWALL	0.51g	0.88g
4	EQ-BAF-SUPP	1.30g	0.91g
5	EQ-PCC-TANK	0.51g	0.81g
6	EQ-SHBLD-ROOF	0.51g	0.71g
7	EQ-SHBLD-WALL	0.51g	0.71g
8	EQ-CV-INTER	0.50g	0.71g
9	EQ-CV-BUCKLE	0.66g	0.73g
10	EQ-CV-OVERT	1.11g	1.38g
11	EQ-IRWST-TANK	0.50g	0.71g
12	EQ-POL-CRANE	0.77g	0.55g
	IE HCLPF=	0.50g	0.55g

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Table 19.20-4
EQ-IEV-RVFA (EQRVF-02) HCLPF

		Original AP1000	Updated AP1000
1	EQ-DVI-PIPES	0.81g	0.81g
2	EQ-SG-FAILS	0.54g	0.54g
3	EQ-RCP-FAILS	0.68g	0.54g
4	EQ-PRZR-FAILS	0.55g	0.50g
5	EQ-RV-FUEL	0.50g	0.50g
6	EQ-RV-HDPK	0.70g	0.70g
7	EQ-RV-FAILS	0.64g	0.56g
	IE HCLPF =	0.50g	0.50g

Table 19.20-5
EQ-IEV-LLOCA HCLPF

		Original AP1000	Updated AP1000
1	EQ-PRHR-HX	0.76g	1.11g
2	EQ-RCS-PIPES	0.81g	0.81g
	IE HCLPF =	0.76g	0.81g

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Table 19.20-6
EQ-IEV-SLOCA HCLPF

		Original AP1000	Updated AP1000
RCS Instrumentation Pipe Breaks	EQ-INSTR-PIPES	0.81g	0.81g
Secondary Line Breaks	EQ-SLB	0.81g	0.81g
SGTR	EQ-SGTR	0.54g	0.54g
RCP HX	EQ-RCP-HX	-	0.55g
	HCLPF =	0.54g	0.54g

Table 19.20-7
EQ-IEV-ATWS HCLPF

		Original AP1000	Updated AP1000
1	EQ-CORE-ASSEMBLY	0.50g	0.50g
2	EQ-CRDM	0.70g	0.70g
	HCLPF =	0.50g	0.50g

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Table 19.20-8
Sequence and Plant HCLPF

ET	Original AP1000	Updated AP1000
EQ-STRUC	EQSTR-02	0.55
	EQ-STRUC HCLPF	*0.55*
EQ-RVFA	EQRVF-02	0.50
	EQ-RVFA HCLPF	*0.50*
EQ-LLOCA	EQLLO-02	0.81
	EQLLO-03	0.81
	EQLLO-05	0.81
	EQLLO-06	0.81
	EQLLO-08	0.81
	EQLLO-09	0.81
	EQLLO-10	0.81
	EQLLO-11	0.81
	EQ-LLOCA HCLPF	*0.81*
EQ-SLOCA	EQSLO-02	0.54
	EQSLO-03	0.54
	EQSLO-04	0.54
	EQSLO-05	0.87
	EQ-SLOCA HCLPF	*0.54*
EQ-ATWS	EQATW-02	0.50
	EQATW-03	0.50
	EQATW-04	0.50
	EQATW-05	0.87
	EQATW-06	0.81
	EQATW-07	0.71
	EQ-ATWS HCLPF	*0.50*
EQ-LOSP	*All mixed cut sets (IE HCLP =0.09)*	*N/A*
	Plant HCLPF	***0.50***

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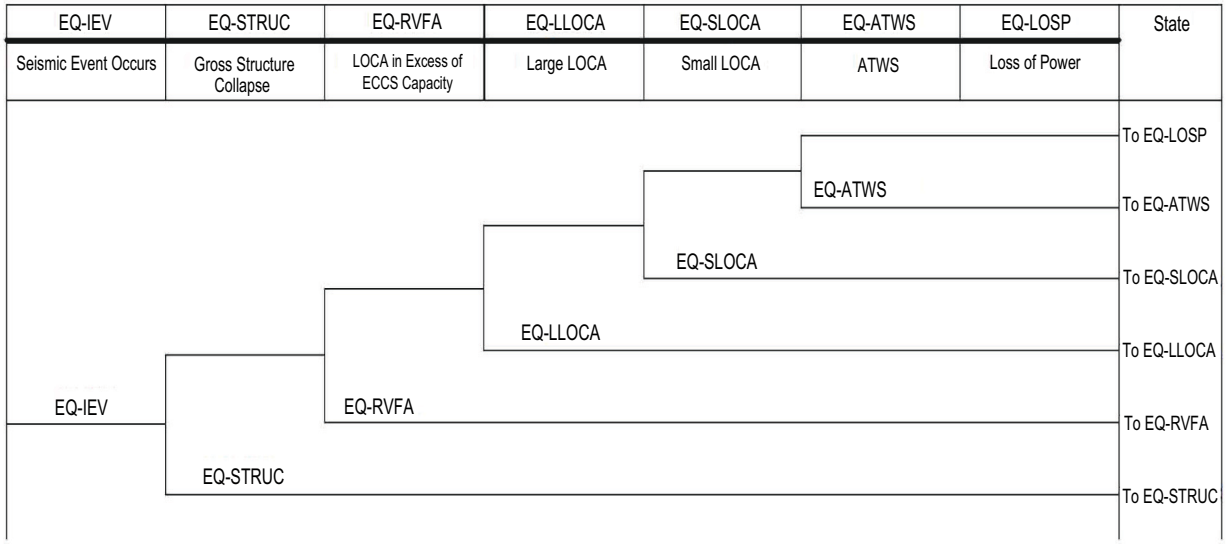

Figure 19.20-1
Seismic Initiating Event Hierarchy Tree

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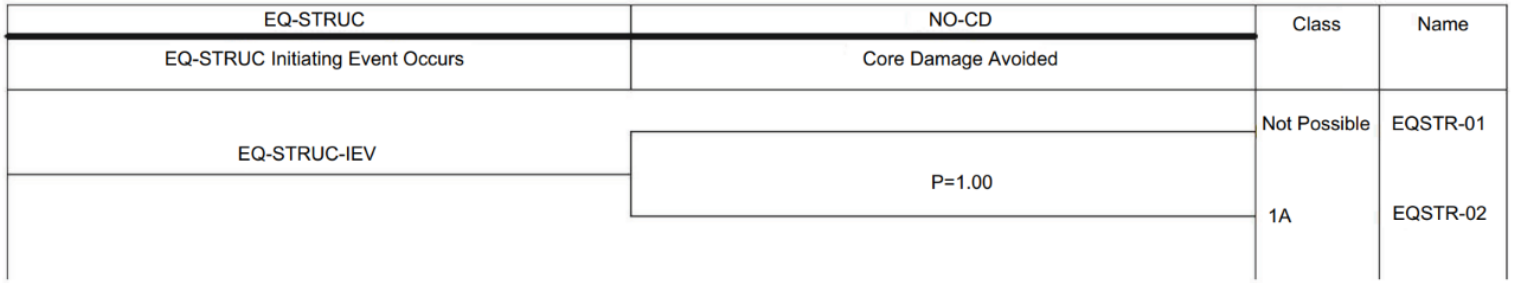

Figure 19.20-2
Seismic Induced Gross Structural Collapse Event Tree

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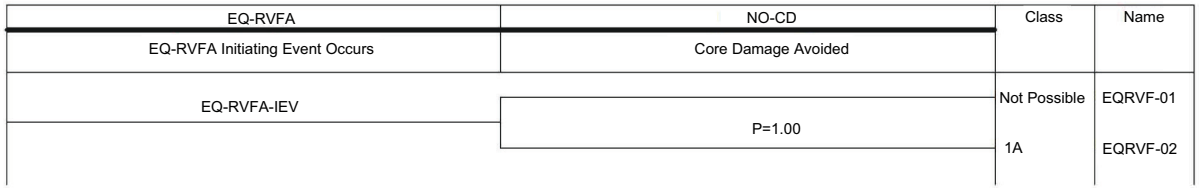

Figure 19.20-3
Seismic Induced Excessive LOCA Event Tree

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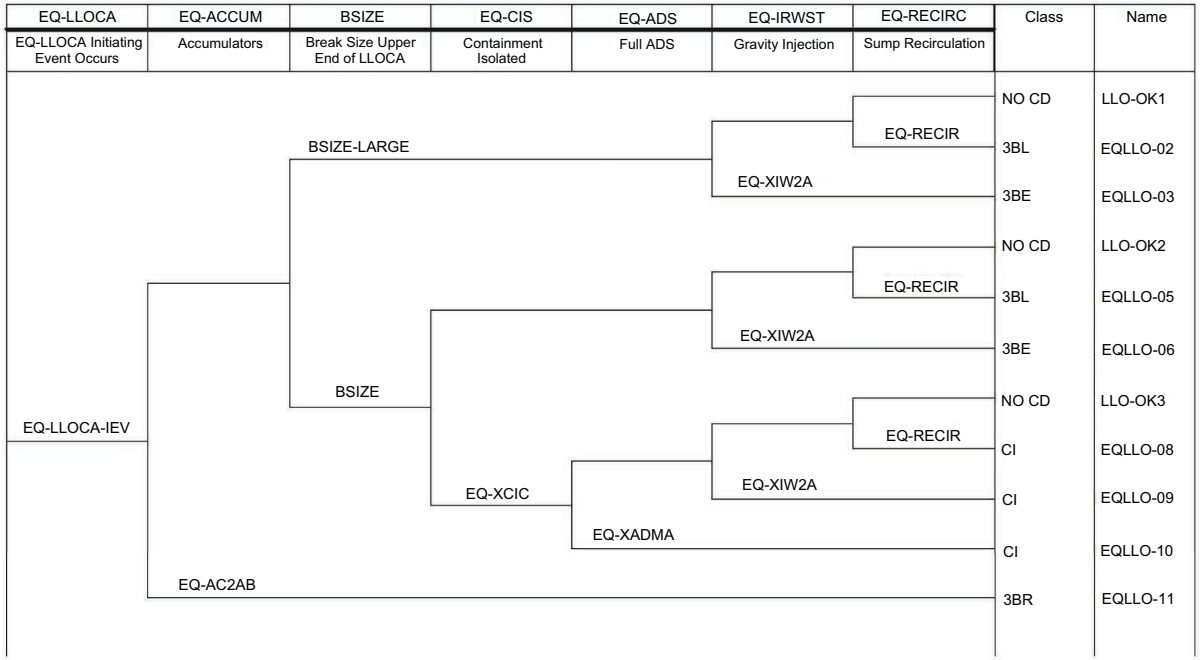

Figure 19.20-4
Seismic Induced Large LOCA Event Tree

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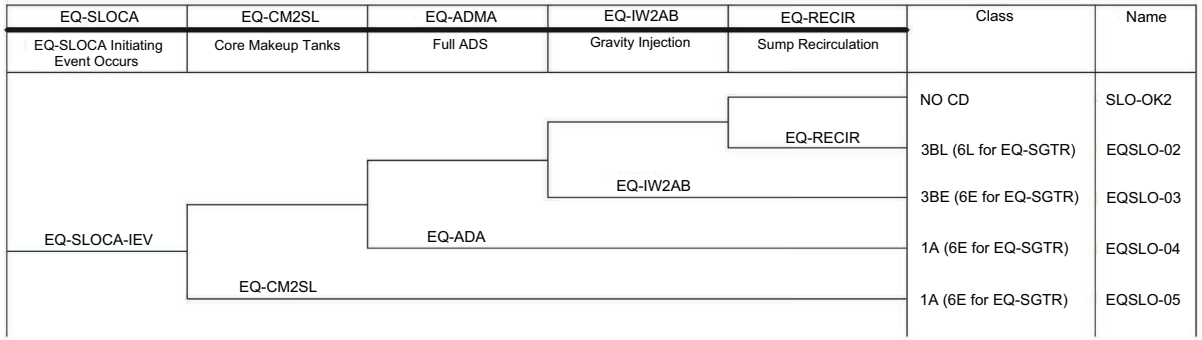

Figure 19.20-5
Seismic Induced Small LOCA Event Tree

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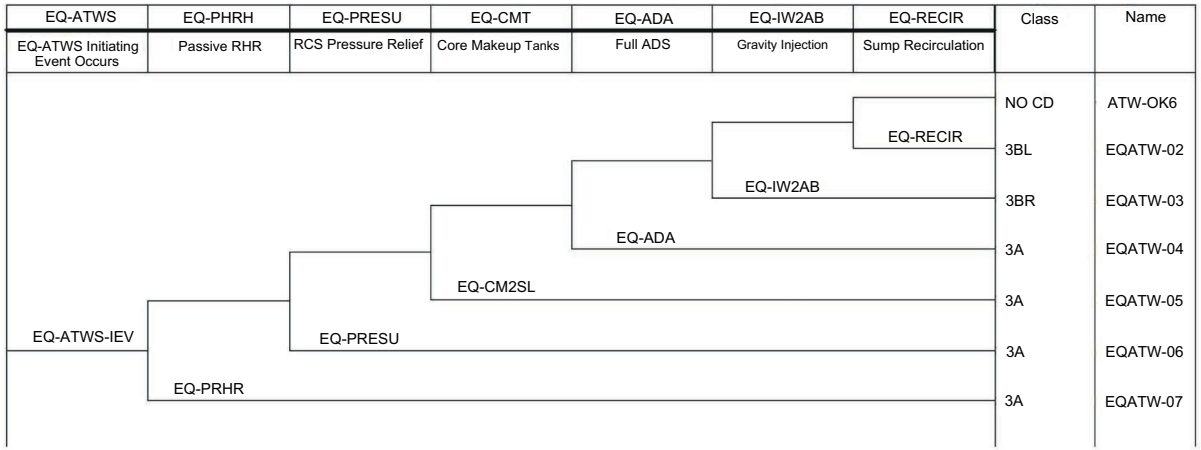

Figure 19.20-6
Seismic Induced ATWS Event Tree

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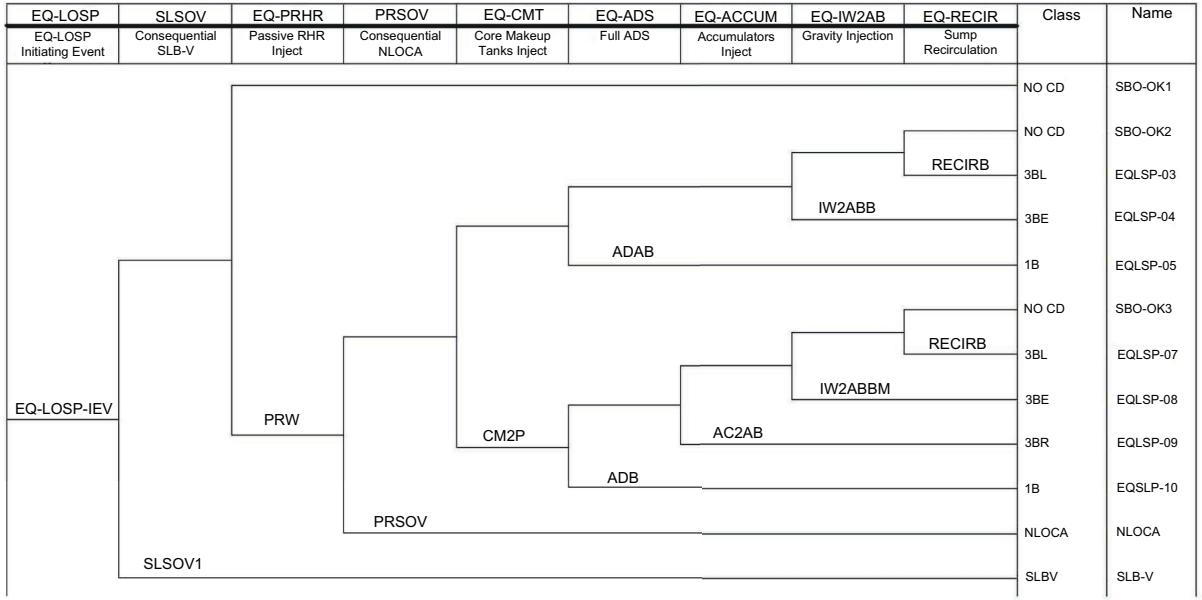

Figure 19.20-7
Seismic Induced LOSP Event Tree

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Appendix 19A THERMAL HYDRAULIC ANALYSIS TO SUPPORT SUCCESS CRITERIA

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

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Appendix 19B Ex-Vessel Severe Accident Phenomena

One of the key AP1000 severe accident design features is the capability to retain the core debris within the reactor vessel for a large number of severe accident sequences by flooding the reactor cavity and submerging the outer surface of the reactor vessel. The heat removal capability of the water on the external surface of the reactor vessel prevents the reactor vessel wall from reaching temperatures where failure of the reactor vessel could occur. This has been termed in-vessel retention (IVR) and is described in detail in Chapter 39 of the AP1000 Level 2 PRA. The primary benefit of in-vessel retention of the core is that ex-vessel severe accident phenomena associated with relocation of core debris to the containment, which can be a dominant containment failure mechanism, are physically prevented. Thus, retention of the core within the reactor vessel results in a significant reduction in the potential for large fission product releases to the environment for core damage accidents.

The probability of various levels of fission product releases (release categories) has been determined in the AP1000 Level 2 PRA, using a containment event tree which describes the various severe accident phenomena that can impact the fission product release quantities and probability of release. In the quantification of the AP1000 Level 2 PRA it was conservatively assumed that the containment would fail at the time of reactor vessel failure for all core damage sequences in which the core debris could not be retained within the reactor vessel. The two principle ways identified in the AP1000 Level 2 PRA of retaining the core within the reactor vessel are reflooding the core with water before the core begins to relocate within the reactor vessel and submerging the outer surface of the reactor vessel to the reactor coolant loop nozzles. Using this approach, the regulatory and industry severe accident performance targets for the AP1000 design criteria were met. Therefore, it was considered unnecessary to investigate the consequences of reactor vessel failure on a realistic basis, including quantification of uncertainties.

The AP1000 design includes features to enhance the likelihood of retaining the core within the reactor vessel for severe accident sequences. These features include:

- Depressurization of the reactor coolant system (RCS) in the event of an accident by either automatic or manual actuation of the highly reliable automatic depressurization system (ADS)
- A containment layout wherein the water relieved from the reactor coolant system (either from the ADS discharge or a break in the RCS) accumulates in the reactor cavity region
- The capability to manually initiate flooding of the reactor cavity by gravity draining the in-containment refueling water storage tank (IRWST) into the reactor cavity
- The absence of in-core penetrations in the reactor vessel bottom head eliminates a possible reactor vessel failure mode
- The reactor cavity layout provides for rapid flooding of the reactor vessel to the reactor coolant loop nozzle elevation
- The reactor vessel insulation design promotes the two-phase natural circulation in the vessel cooling annulus
- The external reactor vessel surface treatment is bare metal

Some of the AP1000 design features to reduce the probability of a core damage accident and to enhance the likelihood of in-vessel retention of core debris in the event of a core damage accident are counter to the design philosophy that would be used to mitigate the consequences of ex-vessel severe accident phenomena. In particular, two of the design features are mutually exclusive between

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preventing ex-vessel phenomena and mitigating the consequences of ex-vessel phenomena. On balance, the AP1000 severe accident risk profile is substantially reduced by the features that prevent ex-vessel severe accident phenomena. Two of the more noteworthy features are:

- The large mass of the AP1000 core provides for a slower accident progression, which enhances the capability to prevent a core damage accident (i.e., a reduced core damage frequency). The larger mass of core materials may result in more severe consequences from some of the potential ex-vessel phenomena such as core debris coolability and core concrete interactions.

- The small reactor cavity floor area reduces the amount of water required to completely submerge the reactor vessel. The small cavity floor area also provides for a more rapid flooding of the cavity if manual initiation of IRWST draining to the reactor cavity is required to submerge the reactor vessel. The small reactor cavity floor area may result in more severe consequences from some of the severe accident ex-vessel phenomena such as core debris coolability and core concrete interactions.

The purpose of this section is to provide the results of a limited number of deterministic investigations of the consequences of ex-vessel severe accident phenomena for the AP1000 design. The results of these deterministic investigations show that the challenges to the integrity of the containment posed by ex-vessel severe accident phenomena are generally within the structural capability of the containment. From these investigations, the conclusion is the capability to prevent large fission product releases to the environment does not depend on the ability to retain the core within the reactor vessel for core damage accident sequences.

The limited deterministic investigations of ex-vessel severe accident phenomena described in this section includes: ex-vessel steam explosions, direct containment heating and core concrete interactions. These ex-vessel phenomena are strongly dependent on the assumptions made concerning the mode of reactor vessel failure for the AP1000 design. Therefore, the reactor vessel failure mode is described first, followed by a description of the ex-vessel phenomena investigations.

19B.1 Reactor Vessel Failure

The AP1000 reactor vessel has a main cylindrical section approximately 4 meters in diameter and a hemispherical bottom head. The bottom head is approximately 15 cm (6 inches) thick and is made of carbon steel with an inner cladding of stainless steel to prevent contact between reactor coolant and carbon steel during normal plant operations. The bottom head of the reactor vessel does not contain any discontinuities or penetrations that could impact the mode of reactor vessel failure as the molten core material relocates to the bottom head.

Based on the similar vessel configurations of AP600 and AP1000, the possible failure modes for the AP600 reactor vessel, as documented in Reference 19B-1, are extended to the AP1000. The most likely failure mode is creep failure of the vessel wall due to heating of the vessel wall by the core debris that has relocated to the reactor vessel bottom head. Since creep failure is a strongly temperature-dependent phenomenon, the location of the failure is predicted to be at the upper surface of the core debris pool that has relocated to the reactor vessel bottom head. For most severe accident sequences, this location is near the junction of the hemispherical bottom head and the cylindrical portion of the vessel.

As described in Reference 19B-2, the presence of water on the external surface of the reactor vessel, as in the case of a flooded reactor cavity, does not alter the conclusion that the highest heat fluxes to the reactor vessel walls will be at a point near the top of the in-vessel molten core pool. This would correspond to the region of the reactor vessel most susceptible to creep failure. However,

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reactor vessel failure will not occur for the case in which the reactor coolant system is depressurized and the reactor cavity is filled with water to the reactor coolant loop elevation.

For the case in which the outside of the reactor vessel is initially submerged but a sufficient in-flow of water to the reactor cavity cannot be maintained, the reactor vessel wall location experiencing the highest heat fluxes would uncover and lose its external cooling before other locations on the reactor vessel lower head. Thus, creep failure of the vessel would be expected to occur at the same location as the case with no water in the reactor cavity.

Two reactor vessel failure cases, as described below, are carried through the deterministic analyses of ex-vessel steam explosions and core concrete interactions. For the consideration of ex-vessel steam explosions and core concrete interactions, it is assumed that the reactor vessel is initially submerged in water but that gravity draining of water from the IRWST does not occur. As the water in the reactor cavity boils down, the outside of the reactor vessel at the elevation at the top of the in-vessel core pool will dry out and begin to heat up. As the vessel wall heats up, it undergoes thinning due to dissolution and melting until failure occurs. The manner in which the reactor vessel fails is treated in two separate scenarios described below.

In the first scenario, the formation of a localized opening occurs due to asymmetric heating around the circumference followed by the vessel tearing around nearly all of its circumference. This would result in the bottom part of the reactor vessel and the bottom head hinging such that the lower head swings downward and comes to rest on the cavity floor. This behavior is illustrated in Figure 19B-1. A hinging type of failure would result in an immediate pouring of core debris onto the cavity floor with metal flowing ahead of oxide. The relationship between the height of the reactor vessel above the floor is such that all but a minor part of the oxide melt would be free to flow immediately out of the head.

In the second scenario, the head and bottom part of the vessel do not hinge downward. In this scenario, the formation of a localized opening permits molten core debris to drain into the cavity lowering the in-vessel core debris depth and thereby decreasing the thermal load on the vessel wall formerly adjacent to the melt. This type of failure is illustrated in Figure 19B-2. In this case, the continued boildown of water level is followed by the release of the core debris located above the water level after a delay interval during which heatup, thinning, and localized failure of the wall will occur. Over time, the elevation of the failure location moves downward over the vessel wall and lower head. This type of failure gives rise to a very slow release rate with the core debris first relocating downward through the water before collecting and spreading on the cavity floor.

19B.2 Direct Containment Heating

Direct containment heating (DCH) is defined as the rapid energy addition to the containment atmosphere as a result of several physical and chemical processes that can occur if the core debris is forcibly ejected from the reactor vessel. The prerequisites for direct containment heating are vessel failure occurs at a location where a substantial portion of the core debris that has relocated to the lower head is ejected into the reactor cavity before the RCS gases are discharged from the RCS and the RCS is at a high pressure (sometimes called high pressure melt ejection or HPME).

To preclude the potential for high-pressure core melt ejection leading to containment failure via DCH, SECY-93-087 (Reference 19B-4) directs passive light water reactor (LWR) designs to:

- Provide a reliable depressurization system
- Provide cavity design features to decrease the amount of ejected core debris that reaches the upper compartment

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The AP1000 design incorporated design features that prevent high-pressure core melt. These features include the passive residual heat removal (PRHR) system and the ADS, both subsystems of the passive core cooling system (PXS). Depressurization of the AP1000 RCS in the event of an accident is provided by automatic or manual actuation of the ADS. Redundancy and diversity are included within the ADS design to ensure a highly reliable depressurization system. The ADS consists of four different valve stages that open sequentially to reduce reactor coolant system pressure in a controlled fashion. All four-valve stages are arranged into two identical groups. Different valve types/sizes are utilized within the ADS stages to provide diversity. Based on these ADS design features, a highly reliable depressurization system is provided which precludes the potential for high-pressure core melt ejection in the AP1000 design. The AP1000 PRHR and ADS subsystems are described in additional detail in Chapters 8 and 11 of the AP1000 PRA and in Section 6.3.

Even though high-pressure core melt ejection is not a likely scenario for the AP1000, SECY-93-087 directs passive LWR designs to include cavity design features to decrease the amount of ejected core debris from reaching the upper compartment. The AP1000 design includes design features to retain and quench the core debris within the reactor cavity in the unlikely event of core debris relocation outside the reactor vessel. These features include:

- A containment layout wherein the water accumulates in the reactor cavity region
- The capability to manually initiate flooding of the reactor cavity by gravity draining the IRWST into the reactor cavity
- The reactor cavity geometry is arranged to provide a torturous pathway from the reactor cavity to the loop compartment and no direct pathway for the impingement of debris on the containment shell

19B.3 Ex-Vessel Steam Explosions

The first level of defense for ex-vessel steam explosion is the in-vessel retention of the molten core debris. If molten debris does not relocate from the vessel to the containment, there are no conditions for ex-vessel steam explosion. In the event that the reactor cavity is not flooded and the vessel fails, the PRA containment event tree assumes that the containment fails in the early time frame.

An analysis of the structural response of the reactor cavity was performed for the AP600 (Reference 19B-3). As in the in-vessel steam explosion analysis, the results of this AP600 ex-vessel steam explosion analysis are extended to the AP1000. The vessel failure modes for AP600 and AP1000 are the same. The initial debris mass participating in the interaction, superheat and composition are assumed to be the same as for AP600. The mass assumption is conservative since the AP1000 reactor vessel lower head is closer to the cavity floor resulting in less debris mass participating in the interaction. The reactor cavity geometry and water depth prior to vessel failure are the same as AP600. Therefore, the results of the AP600 ex-vessel steam explosion analysis are considered to be appropriate for the AP1000.

19B.4 Core Concrete Interactions

If the reactor vessel fails when the RCS is at a low pressure, the molten core debris will pour from the reactor vessel onto the reactor cavity floor. If a steam explosion does not occur, the pour will spread over the cavity floor and begin to transfer heat to the concrete floor of the reactor cavity. Due to the predicted mode of reactor vessel failure and the shape of the AP600 reactor cavity, analyses of the possible spreading of the core debris over the cavity floor were conducted. The results were used as input to the MAAP4 code for analysis of core concrete interactions for AP1000.

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An investigation of the spreading of core debris that pours into the reactor cavity was conducted for reactor vessel failure that occurs at low RCS pressure. The investigation considered the vessel failure mode and location, as well as the recognition that the oxide and metal components of the in-vessel core debris are predicted to be separated. Since the oxide and metal components of the core debris have very different physical characteristics (e.g., viscosity or heat capacity), the separated in-vessel layers influence the spreading of the core debris in the reactor cavity. The melt spreading analysis was conducted for two reactor vessel failure modes, hinged and localized failures.

For the hinged vessel failure case, the analysis results show that the core debris is spread relatively uniformly over the reactor cavity floor. However, the distribution of the metal and oxide components of the core debris is not uniformly distributed over the reactor cavity floor. In the region directly under the reactor vessel, the core debris consists primarily of the oxide component. At the opposite end of the reactor cavity, the core debris consists mainly of the metal component of the core debris released from the reactor vessel. The core debris is still almost totally molten at the end of the spreading analysis.

A different behavior is predicted for the localized reactor vessel failure case. The analysis predicts that the core debris will accumulate at the reactor vessel end of the reactor cavity. The distribution of the metal and oxide components of the core debris is not uniformly distributed over the reactor cavity floor. In the region directly under the reactor vessel, the core debris consists primarily of the oxide component. At the opposite end of the reactor cavity, the core debris consists mainly of the metal component of the core debris released from the reactor vessel. The core debris is almost totally frozen at the end of the spreading analysis.

The core concrete interactions for the AP1000 design were analyzed for two concrete types: basaltic concrete and common limestone-sand concrete. The common limestone-sand concrete has a significantly higher noncondensable gas generation rate, compared to basaltic concrete and should therefore present a more severe containment pressurization transient. On the other hand, the basaltic concrete suffers higher ablation rate, due to its physical properties (mainly, its lower decomposition energy), and should therefore present a more severe basemat penetration failure mode, compared to common limestone-sand concrete. In all cases, a 3.5 m deep water pool is initially present in the cavity while debris is being released into it.

Based on analyses, it can be concluded that: a) the goal of protecting the containment fission product boundary during the first 24 hours of a core melt accident is met, b) it is not necessary to specify a concrete type for the containment basemat since credible containment basemat failure that could lead to fission product releases to the atmosphere are likely to occur at times well beyond 24 hours, and c) the reactor cavity sump is adequately protected such that it is not a weakness in containment basemat integrity during postulated accidents that lead to core concrete interactions.

19B.4.1 Containment Pressurization due to Core Concrete Interactions

The containment pressurization due to steam and noncondensable gas generation during the episodes of core concrete interactions described above was assessed to determine the effect of core concrete interactions on the containment integrity.

The indicator of a challenge to containment integrity for the containment pressurization due to the noncondensable gases produced from core concrete interactions is the Service Level “C” pressure, which is 91 psig (0.73 MPa). This is well below the 50 percent containment failure probability value of 135 psig (1.03 MPa).

The results also show that, in all cases the containment does not pressurize to Service Level “C” containment challenge indicator value prior to the time that the core debris completely penetrates the containment basemat. Thus, for these cases there is no potential challenge to containment integrity

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due to overpressurization since: a) there is no longer a source of mass and energy input to the containment after the core debris penetrates the entire basemat, and b) basemat penetration assures that the containment will be depressurized through the basemat failure.

Based on these analyses, it can be concluded that it is not necessary to specify a concrete type for the containment basemat since containment overpressure failure due to non-condensable gas generation from core concrete interactions is not likely for any credible severe accident scenarios.

19B.5 Conclusions

The results of the limited deterministic analyses of ex-vessel severe accident phenomena presented in this section show that early containment failure is not a certainty if the reactor vessel fails. Based on the deterministic analyses, direct containment heating that might ensue from a high pressure melt ejection would not challenge the integrity of the containment. Ex-vessel steam explosions, assessed on a very conservative basis would not produce impulse loads that would challenge the integrity of the containment due to localized failures of the reactor cavity floor and walls. In addition, these analyses indicate that the ex-vessel steam explosion loads are not strong enough to displace the reactor vessel from its location inside the biological shield. Thus, there is no challenge to any containment penetrations connected to the reactor vessel or to the reactor coolant loops. In the case of a vessel failure at a low RCS pressure, the core concrete interactions analyses indicate that the containment integrity would not be challenged in the first 24 hours of the event and thus no significant releases of fission products are predicted in that time frame.

Thus, it is concluded that prevention of large fission product releases to the environment is not dependent on the integrity of the reactor vessel. If reactor vessel failure occurs, there may be challenges to the containment integrity, but these challenges are highly uncertain and the most likely challenge (containment failure by core penetration of the cavity basemat) would not occur in the first 24 hours of the accident. Thus, the AP1000 assumption that reactor vessel failure always leads to containment failure is a conservatism in the AP1000 risk profile.

19B.6 References

19B-1. “AP600 Phenomenological Evaluation Summaries,” WCAP-13388 (Proprietary) Rev. 0, June 1992 and WCAP-13389 (Nonproprietary), Rev. 1, 1994.

19B-2. Theofanous, T. G., et al., “In-Vessel Coolability and Retention of a Core Melt,” DOE/ID-10460, July 1995.

19B-3. “AP600 Probabilistic Risk Assessment,” GW-GL-022, August 1998.

19B-4. “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Design,” SECY-93-087, dated April 2, 1993.

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Table 19B-1 Not Used

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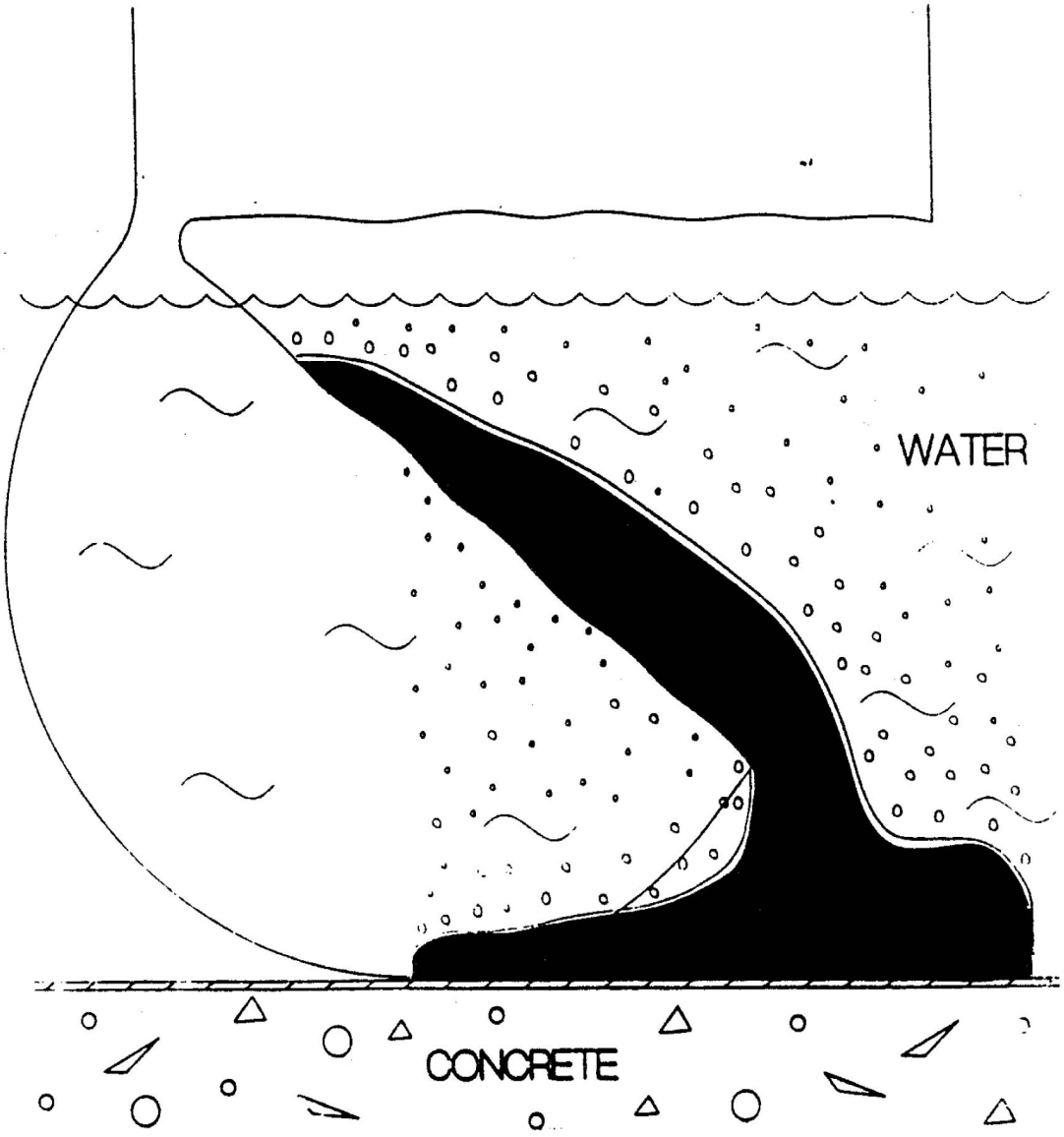

Figure 19B-1
Illustration of Hinging Type of Failure Resulting in Rapid Melt Release

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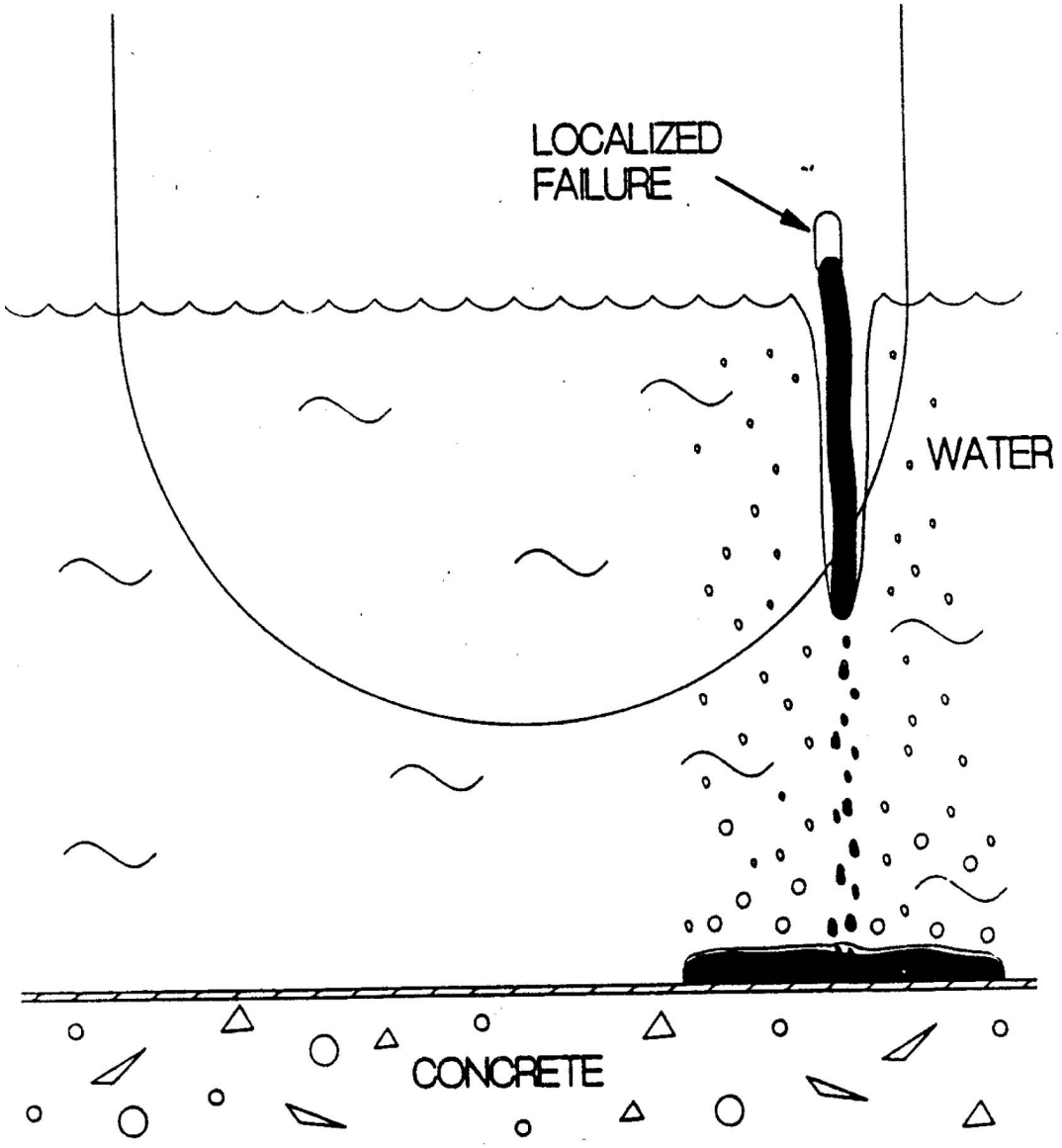

Figure 19B-2
Illustration of Localized Type of Failure Resulting in Slow Melt Release

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Figures 19B-3–19B-8b Not Used

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Appendix 19C Not Used

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Appendix 19D Equipment Survivability Assessment

19D.1 Introduction

The purpose of the equipment survivability assessment is to evaluate the availability of equipment and instrumentation used during a severe accident to achieve a controlled, stable state after core damage under the unique containment environments. Severe accident phenomena may create harsh, high temperature and pressure containment environments with a significant concentration of combustible gases. Local or global burning of the gases may occur, presenting additional challenges to the equipment. Analyses demonstrate that there is reasonable assurance that equipment used to mitigate and monitor severe accident progression is available at the time it is called upon to perform.

The methodology used to demonstrate equipment survivability is:

- Identify the high level actions used to achieve a controlled, stable state
- Define the accident time frames for each high level action
- Determine the equipment and instruments used to diagnose, perform and verify high level actions in each time frame
- Determine the bounding environment within each time frame
- Demonstrate reasonable assurance that the equipment will survive to perform its function within the severe environment.

19D.2 Applicable Regulations and Criteria

Equipment that is classified as safety-related must perform its function within the environmental conditions associated with design-bases accidents. The level of assurance provided by equipment required for design-bases events is “equipment qualification.”

The environmental conditions resulting from beyond design basis events may be more limiting than conditions from design-bases events. The NRC has established criteria to provide a reasonable level of assurance that necessary equipment will function in the severe accident environment within the time span it is required. This criterion is referred to as “equipment survivability.”

The applicable criteria for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents are provided in 10 CFR 50.34(f):

- Part 50.34(f)(2)(ix)(c) states that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.
- Part 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluent.
- Part 50.34(f)(2)(xix) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

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- Part 50.44(c)(2) states that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100-percent fuel-clad metal-water reaction.

Part 50.44(c)(4) states that equipment must be provided for monitoring hydrogen in the containment that is functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning. The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents are discussed in Section III.F, “Equipment Survivability” of SECY-90-016. The NRC recommends in SECY-93-087 that equipment provided only for severe accident protection need not be subject to 10 CFR 50.49 equipment qualification requirements, the 10 CFR 50 Appendix B quality assurance requirements, or 10 CFR 50 Appendix A redundancy/diversity requirements. However, mitigation features must be designed to provide reasonable assurance they will operate in the severe accident environment for which they are intended and over the time span for which they are needed.

19D.3 Definition of Controlled, Stable State

The goal of accident management is to achieve a controlled, stable state following a beyond design basis accident. Establishment of a controlled, stable state protects the integrity of the containment pressure boundary. The conditions for a controlled, stable state are defined by APP-GW-GL-027, the “Framework for AP1000 Severe Accident Management Guidance” (SAMG) (Reference 19D-1).

For a controlled, stable core state:

- A process must be in place for transferring the energy being generated in the core to a long-term heat sink.
- The bulk core temperature must be well below the point where chemical or physical changes might occur.

For a controlled, stable containment state:

- A process must be in place for transferring the energy that is released to the containment to a long-term heat sink.
- The containment boundary must be protected.
- The containment and reactor coolant system conditions must be well below the point where chemical or physical processes (severe accident phenomena) might result in a dynamic change in containment conditions or a failure of the containment boundary.

19D.4 Definition of Equipment Survivability Time Frames

The purpose of the equipment survivability time frames is to identify the time span in the severe accident in which specific equipment is required to perform its function. The phenomena and environment associated with that phase of the accident defines the environment which challenges the equipment survivability. The equipment survivability time frame definitions are summarized in Table 19D-1.

19D.4.1 Time Frame 0 - Pre-Core Uncovery

Time Frame 0 is defined as the period of time in the accident sequence after the accident initiation and prior to core uncovery. The fuel rods are cooled by the water/steam mixture in the reactor vessel.

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The accident has not yet progressed beyond the design basis of the plant, and hydrogen generation and the release of fission products from the core is negligible. Emergency Operating Procedures (EOPs) are designed to maintain or recover the borated water inventory and heat removal in the reactor coolant system to prevent core uncovery and establish a safe, stable state. Recovery within Time Frame 0 prevents the accident from becoming a severe accident. Equipment survivability in Time Frame 0 is covered under the design basis equipment qualification program for the primary accident management strategies.

19D.4.2 Time Frame 1 - Core Heatup

Time Frame 1 is defined as the period of time after core uncovery and prior to the onset of significant core damage as evidenced by the rapid zirconium-water reactions in the core. This is the transition period from design basis to severe accident environment. The overall core geometry is intact and the uncovered portion of the core is overheating due to the lack of decay heat removal. Hydrogen releases are limited to relatively minor cladding oxidation and some noble gas and volatile fission products may be released from the fuel-clad gap due to rupture of fuel rod cladding at these higher temperatures. As the core-exit gas temperature increases above 1200 degrees F, the EOPs transition to a red path indicating inadequate core cooling (FR-C.1). Upon entry into FR-C.1, the control room staff initiates actions to mitigate a severe accident by turning on the hydrogen igniters for hydrogen control and flooding the reactor cavity to prevent reactor pressure vessel failure. The operators attempt to reduce the core temperature by depressurizing the RCS and re-establish the borated water inventory in the reactor coolant system. Recovery in Time Frame 1 prevents the accident from becoming a core melt. In general, the containment conditions are expected to be within the design basis conditions while the reactor vessel and RCS conditions will be slightly above the design basis. Equipment survivability in Time Frame 1 is evaluated to demonstrate it is within the equipment qualification envelope except for components inside the RCS pressure boundary.

19D.4.3 Time Frame 2 - In-Vessel Severe Accident Phase

Time Frame 2 is the period of time in the severe accident after the accident progresses beyond the onset of rapid zirconium-water reactions and prior to the establishment of a controlled, stable state (end of in-vessel core relocation), or prior to reactor vessel failure. The onset of rapid zirconium-water reactions of the fuel rod cladding and hydrogen generation defines the beginning of Time Frame 2. The heat of the exothermic reaction accelerates the degradation, melting and relocation of the core. Fission products are released from the fuel-clad gap as the cladding bursts and from the fuel matrix as the UO_2 pellets melt. Over the period of Time Frame 2, the initial, intact geometry of the core is lost as it melts and relocates downward. Severe accident management strategies exercised during Time Frame 2 are designed to recover reactor coolant system inventory and heat removal, to maintain reactor vessel integrity and to maintain containment integrity. Recovery actions in Time Frame 2 may create containment environmental challenges by increasing the rate of hydrogen and steam generation.

19D.4.4 Time Frame 3 - Ex-Vessel Severe Accident Phase

Time Frame 3 is defined as the period of time after the reactor vessel fails until the establishment of a controlled, stable state. The AP1000 design and the AP1000 EOPs provide the capability to flood the reactor vessel and depressurize the RCS to prevent vessel failure in a severe accident. This severe accident Time Frame 3 is predicted to be a very low probability event. However, it is included in the SAMG to provide guidance in the event that reactor vessel failure occurs. Molten core debris is relocated from the reactor vessel onto the containment cavity floor which creates the potential for rapid steam generation, core-concrete interaction and non-condensable gas generation. Severe accident management strategies implemented in Time Frame 3 are designed to monitor the accident progression, attempt to re-establish a coolable core configuration on the containment floor, maintain containment integrity and mitigate fission product releases to the environment.

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19D.5 Definition of Active Operation Time

Equipment only needs to survive long enough to perform its function to protect the containment fission product boundary. In the case of some items, such as valves or motor-operators, once the equipment performs its function, and changes state (e.g., opens), the function is completed. An exception to this is solenoid-operated valves that must maintain a position other than their design basis failure position (e.g., a fail closed AOV that must remain open for a strategy to remain effective). For other items, such as pumps, the equipment must operate continuously to perform its function. The time of active operation is the time during which the equipment must perform its function.

19D.6 Equipment and Instrumentation for Severe Accident Management

The AP1000 EOPs (Reference 19D-2) and severe accident management guidance (SAMG) framework (Reference 19D-1) define actions that accomplish the goals for achieving a controlled, stable state and terminating fission product releases in an accident. The high level actions from the accident management framework are summarized in Table 19D-2 and provide the basis for identifying equipment. This section discusses the EOP and SAMG actions within each of the time frames of the accident to determine the equipment and instrumentation and the active operation time in which they are needed to provide reasonable assurance of achieving a controlled, stable state.

The AP1000 SAMG (Reference 19D-3) provides the primary input to the selection of the instrumentation used for monitoring the actions. The instrument used to diagnose the need for the action and monitor the response are listed. Instruments to evaluate potential negative impacts are covered under other high level actions in the framework and therefore are also considered for survivability.

The equipment and instrumentation used in each time frame are summarized in Tables 19D-3 through 19D-5. Although the SAMG considers all possible paths for each high level action, only the primary method is listed in Tables 19D-4 and 19D-5 for the equipment survivability assessment.

19D.6.1 Time Frames 0 and 1 - Accident Initiation, Core Uncovery and Heatup

Time Frame 0 represents the accident time prior to core uncovery. Time Frame 1 represents the time following core uncovery, but prior to the rapid oxidation of the core. Aside from potential ballooning of the cladding, the core has not lost its initial intact geometry and coolability is assured by recovering the core with borated water.

During Time Frames 0 and 1, most of the equipment that is automatically actuated will receive a signal to start. However, given that the accident has progressed to core uncovery and heatup, some critical equipment has not actuated. From accident initiation until the time of core uncovery (Time Frame 0) the conditions are bounded by the design basis and covered under equipment qualification. During Time Frame 1, the containment environment is still within the design basis of the plant and the control room is operating within the Emergency Operating Procedures, but the conditions have degraded. Accident management to achieve a controlled, stable state, via the EOPs, is geared toward recovering the core cooling before the coolable geometry is lost.

19D.6.1.1 Injection into the RCS

Failure of RCS injection is likely to be the reason the accident has proceeded to core uncovery. Successful injection into the RCS removes the sensible and decay heat from the core. Prior to the onset of rapid oxidation of the cladding, successful RCS injection recovers the accident before it progresses to substantial damage and establishes a controlled, stable state. Failure to inject into the RCS at a sufficient rate allows the accident to proceed into Time Frame 2 and the SAMG.

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The equipment and systems used to inject into the RCS during Time Frame 0 and 1 are the core makeup tanks, accumulators and IRWST (which are part of the passive core cooling system (PXS)), the chemical and volume control system (CVS) makeup pumps, and the normal residual heat removal (RNS) pumps. For non-LOCA and small LOCA sequences, depressurization of the RCS using the automatic depressurization system (ADS) is required for successful injection.

The plant response is monitored using the system flow rates, IRWST water level indication, RCS pressure, core-exit temperature, and RCS temperature.

19D.6.1.2 Injection into Containment

The operator is instructed via the EOPs to inject water into the containment to submerge the reactor vessel and cool the external surface if core overheating begins to occur. This action is performed later in Time Frame 1, but prior to entry into the SAMG. Successful cavity flooding, in conjunction with RCS depressurization, prevents vessel failure in the event of molten core relocation to the vessel lower head. Failure of cavity flooding allows the accident to proceed to vessel failure and molten core relocation into the containment (Time Frame 3) if timely injection into the reactor vessel cannot be established to cool the core and prevent substantial core relocation to the lower head.

The PXS motor-operated and squib recirculation valves are opened manually to drain the IRWST water into the containment in Time Frame 1.

The plant response is monitored by core-exit temperature, containment water level indication, and IRWST water level indication.

19D.6.1.3 Decay Heat Removal and Injection into the Steam Generators

In the event of non-LOCA or small LOCA sequences, the RCS pressure is elevated above the secondary pressure. In Time Frame 0, the SGs and PRHR are used for decay heat removal. Note that PRHR is effective only in Time Frame 0. Failure of the PRHR may be the reason that the event proceeds to core overheating. Recovery of the PRHR will provide decay heat removal. Failure of feedwater to the steam generators with the PRHR failed may also be a cause for core overheating and recovery of injection to the steam generators may be required. If the steam generators remain dry without PRHR recovery and the core is uncovered, the tube integrity or hot leg nozzle integrity may be threatened by creep rupture failure at the onset of rapid oxidation (entry into Time Frame 2) if the RCS is at a high pressure. Injecting to the steam generators provides a heat sink to the RCS by boiling water on the secondary side, and protects the tubes by cooling them. Successful steam generator injection can establish a controlled, stable state if the losses from the RCS can be recovered and mitigated. Failure to inject to the steam generator requires depressurization of the RCS to prevent creep rupture failure of the tubes and loss of the containment integrity at the onset of rapid oxidation in Time Frame 2.

For accident sequences initiated by steam generator tube rupture, the procedures instruct the control room to isolate feedwater to the faulted steam generator, and to use feedwater to the intact steam generator in conjunction with steam generator depressurization and PRHR initiation to cooldown the reactor coolant system and isolate the break. In Time Frame 1, PRHR initiation or feed to the intact steam generators may be used to re-establish a primary heat sink to cooldown the RCS and a controlled, stable state if the losses from the RCS can be recovered and mitigated. Failure to recover the PRHR or to feed the intact steam generator may lead to a continued loss of coolant to the faulted steam generator and progression to Time Frame 2.

The main feedwater and startup feedwater pumps are used to inject into a pressurized secondary system. The AP1000 plant design does not allow for use of low pressure systems (e.g., condensate, fire water, or service water) to feed the steam generators.

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The plant response is monitored with the steam generator water level indication, steam line pressure, core-exit temperature, RCS temperature, IRWST temperature, and IRWST water level indication.

19D.6.1.4 Depressurize Reactor Coolant System

19D.6.1.4.1 Non-LOCA and Small LOCA Sequences

In Time Frame 0, RCS depressurization is not used for most accidents because the steam generators and PRHR are used to establish a controlled stable state.

In the event of non-LOCA or a small LOCA sequences, the RCS pressure will remain above the secondary pressure. If the steam generators are dry and the core is uncovered, the hot leg nozzle or tube integrity is threatened by creep rupture failure at the onset of rapid cladding oxidation (beginning of Time Frame 2). Timely depressurization (prior to significant cladding oxidation) of the RCS mitigates the threat to the tubes, allows injection of the accumulators and IRWST water, and provides a long-term heat sink to establish a controlled, stable state. Failure to depressurize can result in the failure of the tubes and a loss of containment integrity when oxidation begins.

For steam generator tube rupture (SGTR) initiated sequences, depressurization of the RCS can be used to isolate the faulted steam generator, and re-establish core cooling via injection.

The automatic depressurization system (ADS) is required to depressurize the RCS to allow the PXS systems to inject. However, the recovery of passive residual heat removal (PRHR) or feedwater to the steam generators will provide a substantial heat sink to depressurize the RCS and mitigate the threat to the tubes. The auxiliary pressurizer sprays are not evaluated for survivability since the inclusion of several other safety-related systems which perform the same function provides reasonable assurance of RCS depressurization in the event of a non-LOCA or small LOCA severe accident.

The RCS pressure, steam generator pressure, IRWST water level, and IRWST temperature can be used to monitor the plant response to the RCS depressurization.

19D.6.1.4.2 LOCA Sequences

In Time Frame 0, steam generators and PRHR are not effective due to low RCS pressure.

LOCA sequences (other than small LOCA sequences) by definition are depressurized below the secondary system pressure by the initiating event and therefore, are not a threat to steam generator tube integrity upon the onset of rapid oxidation. Depressurization may be required for injection to establish a long-term heat sink. Medium LOCAs require additional depressurization to allow the injection of RNS or PXS. Large LOCAs are fully depressurized by the initiating event.

In LOCA sequences, the ADS is effective in providing depressurization capability to allow injection to the RCS. While RCS cooldown and depressurization using the steam generators could be effective, it is not evaluated here for survivability for LOCA sequences. RCS cooldown using pressurizer sprays was determined to not be effective for the larger LOCA sequences because of the loss of communication between the RCS and the pressurizer for these sequences.

The RCS pressure can be used to monitor the plant response to the RCS depressurization.

19D.6.1.4.3 Prevent Reactor Vessel Failure

Depressurization of the RCS, along with injecting into the containment is an accident management strategy to prevent vessel failure. The depressurization of the RCS reduces the stresses on the

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damaged vessel wall facilitating the in-vessel retention of core debris. To prevent reactor vessel failure, the RCS must be depressurized to nearly containment conditions.

The ADS is used to depressurize the RCS to prevent reactor vessel failure. The use of the steam generators to depressurize the RCS to prevent vessel failure was determined to not be effective because it cannot bring the RCS pressure down far enough in the time frame of interest for accidents that progress to Time Frame 1 (i.e., no water on primary side of steam generators).

The RCS pressure can be used to monitor the plant response to the RCS depressurization.

19D.6.1.5 Depressurize Steam Generators

The steam generators may be depressurized to depressurize the RCS in non-LOCA and small LOCA sequences. Injection to the steam generator must be available to depressurize the secondary system to prevent creep rupture failure of the tubes.

The steam generator PORV and main steam bypass valves are used for depressurizing the steam generators. The MSIV must be opened in order to use the main steam bypass valves.

Depressurization of the steam generators is used in the EOPs as a means to cool down and depressurize the RCS. Depressurization of the steam generators is called for in the EOPs and is appropriate only in Time Frame 1 as the RCS is depressurized in order to minimize the pressure differential across the steam generator tubes.

The steam line pressure, steam generator water level, and RCS pressure can be used to monitor the plant response.

19D.6.1.6 Containment Heat Removal

Containment heat removal is not explicitly listed as a high level action in the AP1000 SAMG Framework, but it is implicit in the high level action “Depressurize Containment.” Containment heat removal is provided by the passive containment cooling system (PCS). Water cooling of the shell is needed to establish a controlled, stable state with the containment depressurized. The actuation of PCS water is typically automatic in Time Frame 0.

PCS water is supplied to the external surface of the containment shell from the PCS water storage tank or the post-72 hour passive containment cooling ancillary water storage tank. Alternative water sources can be provided via separate connections outside containment.

The containment heat removal can be monitored with the containment pressure and the PCS water flowrate or PCS water and passive containment cooling ancillary water storage tank levels.

19D.6.1.7 Containment Isolation

Containment isolation is not explicitly listed as a high level action in the AP1000 SAMG Framework, but it is implicit as a requirement to protect the fission product barrier.

Containment isolation is provided by an intact containment shell and the containment isolation system which closes the isolation valve in lines penetrating the containment shell that may be open to either the RCS or containment atmosphere following an accident.

The containment isolation can be monitored by the containment pressure and the containment isolation system valve positions.

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19D.6.1.8 Hydrogen Control

Maintaining the containment hydrogen concentration below a globally flammable limit is a requirement for a controlled, stable state. The containment can withstand the pressurization from a global deflagration. While hydrogen is not generated in a significant quantity until Time Frame 2, provisions are provided in the EOPs within Time Frame 1 to turn on the hydrogen igniters before hydrogen generation begins so that hydrogen can be burned as it is produced.

Severe accident hydrogen control in the AP1000 is provided by hydrogen igniters. The containment has passive auto-catalytic recombiners (PARs) as well, which are credited for reducing long-term hydrogen releases in cases with igniter failure. The PARs are passive equipment that cannot be controlled by the operating staff from the control room.

The igniters are manually actuated from the control room in the EOPs on high core-exit temperature. The intention is to actuate the igniters prior to the onset of significant cladding oxidation (Time Frame 1). The containment hydrogen concentration is monitored prior to igniter actuation so that a globally flammable mixture is not unintentionally ignited by the hydrogen igniters.

The plant response to the igniter actuation can be monitored by containment hydrogen concentration using the hydrogen monitors or containment atmosphere sampling, which is part of the primary sampling system. The containment pressure response can also be used to indicate hydrogen burning, which creates a distinctive pressure global peak, but not continual hydrogen burning by the igniters because the energy release to containment is at a low rate and the containment pressure response cannot be distinguished from other heat generation processes.

19D.6.1.9 Accident Monitoring

Accident monitoring is a post-TMI requirement as outlined in 10 CFR 50.34(f). Aside from the accident management purposes outlined above, monitoring the progression of the accident and radioactive releases provides input to emergency response and emergency action levels.

Accident monitoring is provided by the in-containment monitors for pressure, hydrogen concentration, water levels, temperature and radiation, core-exit temperature, IRWST water level, RCS pressure, and steam generator radiation monitors.

19D.6.2 Time Frame 2 - In-Vessel Core Melting and Relocation

Time Frame 2 represents the period of core melting and relocation and the entry into the SAMG. The intact and coolable in-vessel core geometry is lost, and relocation of core debris into the lower head is likely. The in-vessel hydrogen generation and fission product releases from the fuel matrix occur during this time frame.

19D.6.2.1 Injection into the RCS

In Time Frame 2, the in-vessel core configuration loses its coolable geometry and it is likely that at least some of the core debris will migrate to the reactor vessel lower head. If the RCS is depressurized and the reactor vessel is submerged, the core debris will be retained in the reactor vessel. However, injection into the RCS to cover and cool the core debris is required to achieve a controlled, stable state. RCS injection is not required to protect the containment fission product boundary. Injection is successful if it is sufficient to quench the sensible heat from the core debris and refill the reactor vessel. Decay heat removal will then be accomplished by a combination of heat transfer to the water in the reactor vessel and heat transfer to the water on the exterior surface of the reactor vessel.

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Severe accident studies for the AP1000 indicate that even with the reactor vessel refilled and the exterior surface of the reactor vessel submerged, the entire core debris may not return to low temperatures (e.g., less than 1200°F for a substantial period of time (e.g., months) if most of the core debris has relocated to the reactor vessel bottom head. This is due to the heat transfer rate through the outer shell of frozen core debris in relation to the heat generation in the central unfrozen core debris. However, this is an extreme case (i.e., no recovery of injection to the RCS until the entire core debris is in the reactor vessel bottom head).

Guidance for establishing RCS injection would be from the AP1000 SAMG (Reference 19D-3). Water can be injected into the RCS using the CVS or the RNS systems. The PXS (CMT, accumulator, IRWST) is not credited in Time Frame 2 in survivability assessments because automatic and manual activation of the system is attempted several times in Time Frame 0 and 1.

Post-core damage, the actions may be monitored with RCS pressure or temperature, containment pressure or CVS or RNS flow rates.

19D.6.2.2 Injection into Containment

The objective of injection to the containment prior to reactor vessel failure (Time Frame 3) is to cool the external surface of the reactor vessel to maintain the core debris in the vessel. Due to the lead time required to submerge the bottom head of the reactor vessel prior to core relocation of the bottom head, injecting to the containment for in-vessel retention is achieved by instructing the operator to drain the IRWST in the EOPs within Time Frame 1.

Since a long lead time is required to submerge the exterior surface of the RPV, the objective of injecting into containment in Time Frame 2 is to provide water in the containment if the accident progresses to RPV failure and Time Frame 3. Two methods are used to inject into containment during Time Frame 2; the containment spray and the addition of water to the IRWST to overflow into containment. There are three methods used to add makeup to the IRWST to overflow; RNS pumps, makeup pumps, and spent fuel system pumps. Draining the IRWST to containment is not credited in Time Frame 2 in survivability assessments because activation of the system is attempted several times in Time Frame 1, and diverse systems are credited to provide reasonable assurance of containment injection survivability in this time frame. If the vessel fails, the accident progresses to Time Frame 3.

Post-core damage, the actions may be monitored with containment water level indication or IRWST water level indication if IRWST overfill is used.

19D.6.2.3 Decay Heat Removal and Injection into the Steam Generators

In transients and small LOCAs, initiation of PRHR or injection into the steam generators is required to be recovered in Time Frame 1 to be successful. If the secondary side is dry and the RCS is not depressurized, the steam generator tubes can experience creep rupture due to circulation of hot gases when the cladding oxidation begins at the onset of Time Frame 2. Steam generator injection is not required for LOCAs which depressurize the RCS below the secondary system pressure.

Within Time Frame 2, steam generator injection can be utilized in unisolated SGTR sequences to maintain the water level on the secondary side for mitigation of fission product releases. Injecting into the steam generators, along with depressurization of the RCS, is an accident management action to isolate containment or scrub fission products. Failure to inject to the ruptured steam generator in Time Frame 2 can lead to continued breech of the containment fission product boundary and large offsite doses.

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Steam generator feed for non-ruptured SGs is not credited in Time Frame 2 because it is attempted several times in Time Frame 0 and Time Frame 1. However, re-initiation of feedwater to the ruptured steam generator is not attempted until the SAMG, which is not used until Time Frame 2. Thus, re-initiation of feedwater is a Time Frame 2 activity.

The main feedwater and startup feedwater pumps are used to inject into a pressurized secondary system.

The plant response is monitored with the core-exit temperature, RCS temperature, steam generator water level and steam line pressure.

19D.6.2.4 Depressurize RCS

RCS depressurization is required within Time Frame 1 for facilitating in-vessel retention of core debris and for successfully preventing steam generator tube failure in high pressure severe accident sequences. The steam generator tubes or hot leg nozzles may fail due to creep rupture after the onset of rapid oxidation at the beginning of Time Frame 2. RCS depressurization facilitates in-vessel retention of core debris in conjunction with injection into the containment to give time to recover pumped injection sources to the RCS to establish a controlled, stable state. RCS depressurization is provided by instructing the operator to depressurize the system in the EOPs in Time Frame 1.

Three methods are used to depressurize the RCS during Time Frame 2: ADS, auxiliary pressurizer spray, and reactor vessel head vent. ADS and auxiliary pressurizer spray are not credited in Time Frame 2 in survivability assessments because activation of the system is attempted several times in Time Frame 1. Survivability of the reactor vessel head vent is assessed only in Time Frame 2.

19D.6.2.5 Depressurize Steam Generators

Active operation to depressurize a steam generator can be used to cooldown the RCS prior to Time Frame 2. After the onset of core melting and relocation, depressurizing steam generators could threaten steam generator tube integrity. Depressurizing the steam generator in Time Frame 2 does not facilitate the establishment of a controlled, stable state. Depressurization of the steam generators is called for in the EOPs and is appropriate only in Time Frame 1 if the RCS is depressurized in order to minimize the pressure differential across the steam generator tubes.

19D.6.2.6 Containment Heat Removal

Automatic actuation of PCS water occurs in Time Frame 0 or 1. In Time Frame 2, PCS flowrate and level are monitored to determine if additional water is needed to permit continuation of PCS flow. Alternate water sources can be provided by connections to the external passive containment cooling water storage tank (PCCWST) which is outside the containment pressure boundary and not subjected to the harsh environment.

In addition to PCS water, a nonsafety-related containment spray system can provide heat removal from containment. The design basis purpose of containment spray is scrubbing fission products and containment spray is actuated on high containment radiation levels. This would most likely occur in Time Frame 2 when the fuel rods are overheated and melting. Manually actuating the containment spray system involves opening an air-operated valve inside the containment and actuating valves and a pump outside the containment. Once open, the active operation of the valve inside the containment is completed.

Post-core damage, the actions may be monitored with PCS flow rate and PCCWST water level, containment water level, and containment pressure.

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19D.6.2.7 Containment Isolation

Active operation of containment isolation valves is required in Time Frame 0 or 1 to establish the containment fission product barrier. Therefore, only the survivability of the containment pressure boundary, including penetrations, is required to maintain containment isolation after Time Frame 1.

19D.6.2.8 Hydrogen Control

The operator action to actuate the igniters occurs prior to the hydrogen generation at the onset of Time Frame 2. The igniters need to survive and receive power throughout the hydrogen release to maintain the hydrogen concentration below the lower flammability limit during the hydrogen generation in Time Frame 2.

If containment becomes steam inert in Time Frame 2, the igniters will become ineffective and hydrogen will accumulate in containment. The passive auto-catalytic recombiners (PARs) are also available to control hydrogen in containment and can be effective in a steam inert environment. The PARs are passive equipment that are not controlled by the operating staff from the control room.

The plant response to the igniter actuation can be monitored by containment hydrogen concentration using the hydrogen monitors or containment atmosphere sampling, which is part of the primary sampling system. The containment pressure response can also be used to indicate hydrogen burning which creates a distinctive pressure global peak, but not continual hydrogen burning by the igniters because the energy release to containment is at a low rate and the containment pressure response cannot be distinguished from other heat generation processes.

19D.6.2.9 Control Fission Product Releases

A nonsafety-related containment spray system is provided in AP1000 to wash aerosol fission products from the containment atmosphere. The spray system is manually actuated from the SAMG which is entered at the onset of Time Frame 2. Operating the spray involves opening an air-operated valve inside the containment and actuating valves and a pump outside the containment. Once open, the active operation of the valve inside the containment is completed.

Post-core damage, this action may be monitored with containment water level.

19D.6.2.10 Accident Monitoring

During the initial core melting and relocation, containment hydrogen and radiation monitors are used for core damage assessment and verification of the hydrogen igniter operation. Steam generator radiation monitoring is used to determine steam generator tube integrity. In the longer term, containment atmosphere and containment liquid sampling can be used to monitor hydrogen and radiation. Containment pressure, temperature, and water level indication and RCS pressure need to be monitored throughout Time Frame 2.

During a severe accident, the instrumentation may be subjected to conditions well beyond its design basis. The SAMG does not automatically eliminate instrumentation based on its design basis in comparison to severe accident conditions. Instead, the AP1000 SAMG relies on all available instrumentation indications and instructs the user to constantly compare instrumentation readings to diverse sources to validate the instrumentation reading. It is also noteworthy that while target values are established for various plant parameters to indicate a controlled stable state, the trends of the parameters being monitored are equally as important in managing the accident. The parameter trends indicate whether strategies are effective and determine if additional strategies need to be considered.

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19D.6.3 Time Frame 3 - Ex-Vessel Core Relocation

Time Frame 3 represents the phase of the accident after vessel failure. The core debris is in the reactor cavity, and the IRWST water is not injected into the containment.

19D.6.3.1 Injection into the RCS

The RCS is failed. Injection to the RCS is no longer needed in Time Frame 3. Note that the AP1000 SAMG considers RCS injection as a means to inject water into the reactor cavity in Time Frame 3.

19D.6.3.2 Injection into Containment

Water coverage to the ex-vessel debris bed is passively provided by the containment design to drain water from the RCS via the IRWST. Water condensing on the PCS shell is returned to the reactor cavity after filling the IRWST to the overflow. The addition of water to the IRWST from other sources to overflow into containment is also a method of injecting water into containment. Containment spray can also be used to inject water into containment in Time Frame 3. Draining the IRWST to containment is not credited in Time Frame 3 in survivability assessments because activation of the system is attempted several times in Time Frame 1, and diverse systems are credited to provide reasonable assurance of containment injection survivability in this time frame. Containment spray and overflowing the IRWST are also not credited in Time Frame 3 survivability assessments because these methods are already credited in Time Frame 2.

19D.6.3.3 Decay Heat Removal and Injection into the Steam Generators

The RCS is failed. PRHR activation or injection into the steam generators is no longer needed in Time Frame 3. Injection to the steam generator for SGTR fission product scrubbing is not required to maintain the water level.

19D.6.3.4 Depressurize RCS

The RCS is depressurized by the vessel failure in Time Frame 3.

19D.6.3.5 Depressurize Steam Generators

The RCS is failed. Steam generator depressurization is not needed in Time Frame 3.

19D.6.3.6 Containment Heat Removal

Active initiation of PCS water is completed prior to Time Frame 3. PCS flowrate and level are monitored for post-72 hour activities. Alternate water sources can be provided by connections to the external PCCWST which is outside the containment pressure boundary and not subjected to the harsh environment.

In addition to PCS water, a nonsafety-related containment spray system can provide heat removal from containment. The design basis purpose of containment spray is scrubbing fission products and containment spray is actuated on high containment radiation levels. This would most likely occur in Time Frame 2 when the fuel rods are overheated and melting. Manually actuating the containment spray system involves opening an air-operated valve inside the containment and actuating valves and a pump outside the containment. Once open, the active operation of the valve inside the containment is completed.

Post-core damage, the actions may be monitored with PCS flowrate and PCCWST water level, containment water level and containment pressure.

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19D.6.3.7 Containment Isolation and Venting

Continued operation of the containment shell as a pressure boundary is needed to maintain containment isolation in Time Frame 3. Containment temperature needs to be monitored because prolonged exposure of organic materials (e.g., equipment and personnel hatch seals) to high temperatures (> 400°F) can degrade the material.

In the event of containment pressurization above design pressure due to core concrete interaction non-condensable gas generation, the containment can be vented. Venting protects containment isolation by preventing an uncontrolled containment failure airborne release pathway. The vent can be opened and closed as required to maintain pressure in the containment below its failure pressure. Containment venting does not prevent or mitigate containment basemat failure due to core concrete interaction. Containment venting to the spent fuel pool is available through RNS hot leg suction line MOVs.

19D.6.3.8 Combustible Gas Control

The hydrogen igniters are used to control combustible gases. Active operation of igniters continues to control the release of combustible gases (e.g., hydrogen and carbon monoxide) from the degradation of concrete in the reactor cavity.

If containment becomes steam inert in Time Frame 3, the igniters will become ineffective and hydrogen will accumulate in containment. The passive auto-catalytic recombiners (PARs) are also available to control hydrogen in containment and can be effective in a steam inert environment. The PARs are passive equipment that are not controlled by the operating staff from the control room.

The plant response to the igniter actuation can be monitored by containment hydrogen concentration using the containment atmosphere sampling, which is part of the primary sampling system. The containment pressure response can also be used to indicate hydrogen burning which creates a distinctive pressure global peak, but not continual hydrogen burning by the igniters because the energy release to containment is at a low rate and the containment pressure response cannot be distinguished from other heat generation processes.

19D.6.3.9 Control Fission Product Releases

The nonsafety-related sprays are actuated in Time Frame 2. The operation of the nonsafety-related containment spray continues, possibly into Time Frame 3, until the water from the source tank is depleted.

Post-core damage, this action may be monitored with containment water level.

19D.6.3.10 Accident Monitoring

Containment pressure, temperature, water level and radiation, steam generator radiation and the containment hydrogen concentration are sufficient to monitor the accident in the long-term. Hydrogen concentration and radiation can be monitored with containment atmosphere and liquid sampling functions. In both Time Frame 2 and Time Frame 3, auxiliary building radiation monitors, if properly correlated, could be used for containment radiation monitoring.

19D.6.4 Summary of Equipment and Instrumentation

The equipment and instrumentation used in achieving a controlled, stable state following a severe accident, and the time it operates are summarized in Tables 19D-3 through 19D-5.

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19D.7 Severe Accident Environments

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

19D.8 Assessment of Equipment Survivability

Since severe accidents are very low probability events, the NRC recommends in SECY-93-087, that equipment desired to be available following a severe accident need not be subject to the qualification requirements of 10CFR50.49, the quality assurance requirements of 10CFR50 Appendix B, or the redundancy/diversity requirements of 10CFR50 Appendix A. It is satisfactory to provide reasonable assurance that the designated equipment will operate following a severe accident by comparing the AP1000 severe accident environments to design basis event/severe accident testing or by design practices.

19D.8.1 Approach to Equipment Survivability

The approach to survivability is by equipment type, equipment location, survival time required, and the use of design basis event qualification requirements and severe environment experimental data.

19D.8.1.1 Equipment Type

The various types of equipment needed to perform the activities discussed above are transmitters, thermocouples, resistance temperature detectors (RTDs), hydrogen and radiation monitors, valves, pumps, valve limit switches, containment penetration assemblies, igniters, and cables.

19D.8.1.2 Equipment Location

Some of the in-containment equipment, such as transmitters, has been deliberately located to avoid the most severe calculated environments. Other equipment is located outside containment. The performance of the equipment was judged based on the most severe postulated event for that location.

19D.8.1.3 Time Duration Required

Requirements are defined for each time frame, so the equipment evaluation only discusses performance during these periods. A limited amount of equipment has been designated for the long term (Time Frame 3) and these parameters can be monitored outside containment.

19D.8.1.4 Severe Environment Experiments

The primary source for performance expectations of similar equipment in severe accident environments is EPRI NP-4354, “Large Scale Hydrogen Burn Equipment Experiments.” This information is supplemented by NUREG/CR-5334, “Severe Accident Testing of Electrical Penetration Assemblies.” These programs tested equipment types that had previously been qualified for design basis event environmental conditions. The temperature in the chamber for the first program was in the 700°F - 800°F range for ten to twenty minutes during the continuous hydrogen injection tests. Although the conditions at the equipment would be somewhat less severe, the chamber conditions envelop all of the longer duration profiles indicated for the AP1000 events. The equipment in this program was also exposed to significant hydrogen burn spikes that are also postulated for the AP1000 plant. The same equipment was exposed to and survived several events, both pre-mixed and continuous hydrogen injection which provides confidence in its ability to survive a postulated severe accident. The second program tested containment penetrations to high temperatures for long durations. A penetration was tested under severe accident conditions simulated with steam up to

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400°F and 75 psia for ten days. The results indicated that the electrical performance of the penetration would not lead to degraded equipment performance for the first four days. The mechanical performance did not degrade (no leaks) during the entire test.

19D.8.2 Equipment Located in Containment

The exposure to elevated temperatures as a direct result of the postulated severe accident or as a result of hydrogen burning is the primary parameter of interest. Pressure environments do not exceed the design basis event conditions for which the equipment has been qualified if PCS is operating as designed. Radiation environments also do not exceed the design basis event conditions throughout Time Frames 1 and 2.

19D.8.2.1 Differential Pressure and Pressure Transmitters

The functions defined for accident management that utilize in-containment transmitters are IRWST water level, reactor coolant system pressure, steam generator wide range water level, and containment pressure. Most of these transmitters that provide this information are located in rooms where the environment is limited to short duration temperature transients. These transients exceed ambient design basis temperature conditions but should not impact the transmitter performance since the internal transmitter temperature do not increase significantly above that experienced during design basis testing. EPRI NP-4354 documents transmitter performance during several temperature transients with acceptable results. The IRWST water level transmitters are located in the maintenance floor and are only required during Time Frames 1 and 2. The environment during Time Frames 1 and 2 does not exceed the design basis qualification parameters of the transmitters if PCS is operating as designed. Reactor system pressure and steam generator wide range water level are required through the second time frame. The only long term application is the containment pressure transmitter which may eventually be impacted by the severe accident radiation dose.

19D.8.2.2 Thermocouples

The function defined for severe accident management that uses thermocouples is core-exit temperature. The core-exit temperature is only required during Time Frame 1. The temperatures to which the thermocouples are exposed during the defined time frames do not exceed the thermocouple design.

19D.8.2.3 Resistance Temperature Detectors (RTDs)

Both hot and cold leg temperatures are defined as parameters for severe accident management in Time Frame 1. RTDs are utilized for these measurements and will perform until their temperature range is exceeded. The hot leg RTDs could fail as the temperature increases well above the design conditions of the RTDs but the cold leg RTDs should perform throughout Time Frame 1. RTDs are also utilized through Time Frame 3 for the containment temperature measurement and are exposed to temperature transients that exceed design basis qualification conditions. EPRI NP-4354 documents RTD performance during several temperature transients with acceptable results.

19D.8.2.4 Hydrogen Monitors

Containment hydrogen is defined as a parameter to be monitored throughout the severe accident scenarios. Early in the accident, the hydrogen may be monitored by a galvanic device with an electrochemical sensor. The hydrogen monitors are located in the main containment area. The design limits of this device may be exceeded after the first few hours of some of the postulated accidents and performance may be uncertain. If the device fails, post-accident sampling of containment atmosphere using analysis of grab samples may be used to determine containment hydrogen concentrations.

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19D.8.2.5 Radiation Monitors

Containment radiation is defined as a parameter to be monitored throughout the severe accident scenarios. The containment radiation monitors are located in the main containment area. Early in the accident, the design basis event qualified containment radiation monitor provides the necessary information until the environment exceeds the design limits of the monitor. If the device fails, containment radiation is determined through the containment atmosphere sampling function or by portable monitors located against the outside of the containment shell.

19D.8.2.6 Solenoid-Operated Valves (SOVs) and Air-Operated Valves (AOVs)

Qualified solenoid valves are used to vent air-operated valves (AOVs) to perform the function required. In Time Frame 1, the core makeup tank AOVs located in the accumulator room provide a path for RCS injection, the PRHR AOVs located in the maintenance floor provide a path for RCS heat removal and the containment is isolated by AOVs located in the maintenance floor and the PXS valve/accumulator room. The environment to which these solenoid valves may be exposed in Time Frame 1 is not significantly different than the design basis events to which the devices are qualified. In Time Frame 2, the RCS boundary AOV located in the maintenance floor is used for CVS injection into the RCS and the containment spray AOV located in the maintenance floor is used for control of fission product release. Also in Time Frame 2, the reactor vessel head vent solenoid valves provide a path for RCS depressurization. In addition, throughout Time Frame 3, access to the containment environment from the containment atmosphere sampling function is through solenoid valves located in the maintenance floor. During Time Frame 2 and Time Frame 3, these valves may be exposed to transient conditions due to hydrogen burns that exceed design basis event qualification. Solenoid valves in an energized condition were included in the hydrogen burn experiments (EPRI NP-4354) and survived many transients. Shielding provided by the location of the valves limits the severe accident radiation dose to the typical design basis qualification dose for these valves.

19D.8.2.7 Motor-Operated Valves

Motor-operated valves (MOVs) are utilized in several applications during the severe accident scenarios. MOVs in the accumulator and core makeup tank path are normally open and remain open. In Time Frame 1, the PXS recirculation MOVs located in the PXS valve/accumulator room are required for injection of water into the containment, MOVs for the first three stages of ADS located in a compartment above the pressurizer are required for RCS depressurization and the containment is isolated by MOVs located in the maintenance floor and the PXS valve/accumulator room. The environment to which these MOVs may be exposed in Time Frame 1 is not significantly different than the design basis events to which they are qualified. In Time Frame 2, the charging and injection MOV located in the maintenance floor provides a path from the CVS for RCS injection and an RNS MOV located in the PXS valve/accumulator room provides a path from the IRWST for RCS injection. In addition, throughout Time Frame 3, containment venting to the spent fuel pool is available through RNS hot leg suction line MOVs located in the RNS valve room. During Time Frames 2 and 3, these valves may be exposed to transient conditions due to hydrogen burns that exceed design basis event qualification. MOVs were included in the hydrogen burn experiments (EPRI NP-4354) and survived many transients. Shielding provided by the location of the valve limits the severe accident radiation dose to the typical design basis qualification dose for these valves.

19D.8.2.8 Squib Valves

Squib valves are only required in Time Frame 1 when the severe accident environment is not significantly different than the design basis environment for which these valves are qualified. IRWST and PXS recirculation squib valves located in the accumulator room are used for injection into the RCS and containment, respectively. For RCS depressurization, the fourth stage ADS squib valves are located in steam generator compartments 1 and 2.

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19D.8.2.9 Position Sensors

Position sensors are required to monitor the position of containment isolation valves that could lead directly to an atmospheric release. These isolation valves actuate early in the transient, so verification is only required during Time Frame 1. The position sensors are located in the maintenance floor and the environment in this time frame does not exceed the design basis event qualification environment of the position sensors.

19D.8.2.10 Hydrogen Igniters

The hydrogen igniters are distributed throughout the containment and are designed to perform in environments postulated for severe accidents. The successful results of igniter testing through several hydrogen burns is documented in EPRI NP-4354 and provides confidence in the performance of these devices.

19D.8.2.11 Electrical Containment Penetration Assemblies

The electrical containment penetrations are located in the lower compartment and are required to perform both electrically and mechanically throughout the severe accident. The hydrogen burn equipment experiments documented by EPRI NP-4354 included penetrations qualified for nuclear plants. Electrical testing on the penetration cables after all the pre-mixed and continuous injection tests concluded that most of the cables passed the electrical tests while submerged in water. These tests consisted of ac (at rated voltage) and dc (at three times rated voltage) withstand tests and insulation resistance tests at 500 volts. The penetrations were also tested under simulated severe accident conditions at 400°F and 75 psia for about 10 days (NUREG/CR-5334). The results indicated that some degradation in instrumentation connected to the penetration may occur in four days under these severe conditions. The maintenance floor may experience short temperature transients above 400°F but stable temperatures are significantly less, so it is expected that the electrical performance would be maintained throughout the event. The only long term measurement utilizing these penetrations is containment pressure and this can be measured outside containment if necessary. There was no degradation of mechanical performance of the electrical penetrations (maintaining the seal) in either test program.

19D.8.2.12 Cables

The hydrogen burn equipment experiments documented by EPRI NP-4354 included twenty-four different cable types qualified for nuclear plants. Electrical testing on these cables after all the pre-mixed and continuous injection tests concluded that all (fifty two samples) of the cables passed the electrical tests while submerged. These tests consisted of ac (at rated voltage) and dc (at three times rated voltage) withstand tests and insulation resistance tests at 500 volts. Due to the exposure to many events, some cable samples had extensive damage in the form of charring, cracking and bulging of the outer jackets and still performed satisfactorily. The cables tested are representative of cables specified for the AP1000 and are only exposed to short single temperature transients in their respective locations. Proper performance can be expected. The only long term measurement utilizing cables is containment pressure, which can be measured outside containment if necessary.

19D.8.2.13 Float Level Sensors

The function defined for severe accident management that uses float level sensors is containment water level. Containment water level is required through Time Frame 2. The temperature to which the float level sensors are exposed during the time frame does not exceed the float level sensor design.

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19D.8.2.14 Containment Hatches (Gasket Material Only)

The equipment hatch and a personnel hatch are located at the operating deck elevation. The maintenance hatch and a second personnel hatch are located at the maintenance floor elevation. The gasket material of the hatches is required to perform throughout the severe accident to maintain the containment pressure boundary. In the severe accident environment, during hydrogen combustion, the equipment experiences ambient temperature conditions beyond their design basis. However, the hatches are enclosed in the thick metal of the hatch barrels. The internal temperature does not rise beyond that experienced during design basis equipment qualification testing. Therefore, there is reasonable assurance of survivability of this equipment.

19D.8.2.15 Containment Shell

The containment shell is used to maintain containment integrity. The shell is exposed to the severe accident environment in the upper compartment and Room 11300. The upper compartment shell is externally cooled by the PCS. In the analysis, the lower shell is modeled as a heat sink that is adiabatic on the external side and therefore provides the bounding temperature transient for the whole of the containment steel. The active operation time of the containment extends through Time Frame 3 (ex-vessel severe accident environment). The containment shell remains within the code limits during the severe accident. Therefore, the containment shell is expected to survive the severe accident pressure and temperatures predicted for the AP1000 plant.

19D.8.2.16 Assessment of Equipment for Sustained Burning

The equipment necessary for equipment survivability in sustained burning environments is defined in Tables 19D-3 through 19D-5. The equipment in Table 19D-3 includes equipment and instrumentation operation during Time Frame 1 - core uncovery and heatup, and is prior to the release of significant quantities of hydrogen. Therefore, it does not have to be qualified for sustained hydrogen burning. Table 19D-7 specifies the equipment and instrumentation used in Time Frames 2 and 3 to provide reasonable assurance of achieving a controlled stable state. The specific equipment located inside containment which must survive a severe accident for Time Frames 0 and 1 is listed in Table 19D-8 and in Table 19D-9 for severe accident Time Frames 2 and 3.

19D.8.3 Equipment Located Outside Containment

Other functions defined for severe accident management are performed outside containment and the equipment is not subjected to the harsh environment of the event. This equipment includes, but is not limited to:

- Steam line radiation monitor,
- Transmitters for monitoring steam line pressure,
- Passive containment cooling system flow and tank level,
- Containment atmosphere sampling function,
- Makeup pumps and flow measurement,
- RNS pumps and flow measurement,
- SFS pumps and flow measurement,
- RNS MOVs
- MFW pumps and valves,
- SFW pumps and valves,
- Steam generator PORVs and main steam bypass valves for depressurization,
- Recirculation pumps, PCS valves and fire water pumps and valves for containment heat removal,
- Containment isolation valves (outside containment),
- Auxiliary building radiation monitor,

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- MOV and manual valve from RNS hot leg suction lines to the spent fuel pool and
- Fire water, fire pumps, valves and flow measurement used to provide containment spray and backup containment cooling.

19D.9 Conclusions of Equipment Survivability Assessment

The equipment defined for severe accident management was reviewed for performance during the environments postulated for these events. Survivability of the equipment was evaluated based on design basis event qualification testing, severe accident testing, and the survival time required following the initiation of the severe accident. The equipment that is qualified for design basis events has a high probability of surviving postulated severe accident events and performing satisfactorily for the time required.

This assessment provides reasonable assurance that equipment, both electrical and mechanical, used to mitigate the consequences of severe accidents and achieve a controlled, stable state can perform over the time span for which they are needed.

19D.10 References

19D-1. APP-GW-GL-027, “Framework for AP1000 Severe Accident Management Guidance,” Westinghouse Electric Company LLC.

19D-2. AP1000 Emergency Operating Procedures.

19D-3. APP-GW-GJR-400, “AP1000 Severe Accident Management Guidelines,” Westinghouse Electric Company LLC.

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Table 19D-1
Definition of Equipment Survivability Time Frames

Time Frame	Beginning Time	Ending Time	Comments
0	Accident initiation	safe, stable state or core uncovery	• Bounded by design basis equipment qualification environment
1	Core uncovery	controlled, stable state or rapid cladding oxidation	• Core uncovery and heatup • Bounded by design basis equipment qualification environment
2	Rapid cladding oxidation	controlled, stable state or vessel failure	• In-vessel core melting and relocation • Entry into SAMG
3	Vessel failure	controlled, stable state or containment failure	• Ex-vessel core relocation

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Table 19D-2
AP1000 High Level Actions Relative to Accident Management Goals

Goal	Element	High Level Action*
Controlled, stable core	water inventory in RCS	• inject into RCS • depressurize RCS
	water inventory in containment	• inject into containment
	heat transfer to IRWST	• initiate PRHR
	heat transfer to SGs	• inject into RCS • inject into SGs
	heat transfer to containment	• inject into RCS • inject into containment • depressurize RCS • initiate PRHR
Controlled, stable containment	heat transfer from containment	• depressurize containment • vent containment • water on outside containment
	isolation of containment	• inject into SGs • depressurize RCS
	hydrogen prevention/control	• burn hydrogen • pressurize containment • depressurize RCS • inject into containment • vent containment • water on outside containment
	core concrete interaction prevention	• inject into containment
	high pressure melt ejection prevention	• inject into containment • depressurize RCS
	creep rupture prevention	• depressurize RCS • inject into SGs
	containment vacuum prevention	• pressurize containment
Terminate fission product release	isolation of containment	• inject into SGs • depressurize RCS
	reduce fission product inventory	• inject into containment • depressurize RCS
	reduce fission product driving force	• depressurize containment • water on outside containment

Note:
* See Tables 19D-3, 19D-4 and 19D-5

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Table 19D-3 (Sheet 1 of 3)
Equipment and Instrumentation Operation Prior to End of Time Frame 1 - Core Uncovery and Heatup

Action	Equipment	Instrumentation	Purpose	Comment
Inject into RCS	• CMT • accumulator • IRWST • CVS • RNS	• core-exit t/c's • RCS pressure • RCS RTDs • CVS flow • RNS flow • IRWST water level	• restore core cooling	• injection must often be recovered to be successful in severe accident
Inject to SGs	• MFW • SFW	• SG WR water level • steam line pressure	• decay heat removal • make SGs available to depressurize RCS • prevent SG tube creep rupture	• injection source must often be recovered to be successful in severe accident
Decay Heat Removal	• PRHR Hx • via SGs	• IRWST water level • IRWST temperature • core-exit t/c's • RCS RTDs	• decay heat removal	• only works if RCS is reflooded and IRWST water level covers PRHR Hx
Depressurize RCS	• ADS • aux pressurizer spray • via SGs • PRHR Hx	• RCS pressure • IRWST water level • IRWST temperature • steam line pressure	• facilitate injection to RCS • long term heat transfer path	• ADS often automatic
			• prevent SG tube creep rupture • containment integrity	• RCS depressurization required prior to significant cladding oxidation to prevent creep rupture
			• isolate break in SGTR	• uses intact SG or PRHR
			• prevent vessel failure	• requires injection to containment to be successful

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Table 19D-3 (Sheet 2 of 3)
Equipment and Instrumentation Operation Prior to End of Time Frame 1 - Core Uncovery and Heatup

Action	Equipment	Instrumentation	Purpose	Comment
Depressurize SGs	• SG PORV • main steam bypass	• steam line pressure • RCS pressure • SG WR water level	• depressurize RCS • minimize pressure differential across SG tubes	• requires injection into SGs to prevent creep rupture
Inject Into Containment	• IRWST drains	• core-exit t/c's • containment water level • IRWST water level	• prevent vessel failure	• manual cavity flooding action in EOP
Containment Isolation	• containment isolation system • containment shell • penetrations • containment hatches (gasket material only)	• containment isolation system valve position • containment pressure	• containment integrity	• containment isolation system often automatic • manual action in EOP
Control Hydrogen	• igniter • PARs	• containment hydrogen monitors • containment atmosphere sampling functions • containment pressure	• containment integrity	• manual igniter action in EOP
Containment Heat Removal	• PCS water • external water	• containment pressure • PCS flowrate • PCS tank level	• containment integrity • alleviate environmental challenge to equipment • long term heat transfer path	• PCS water automatic

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Table 19D-3 (Sheet 3 of 3)
Equipment and Instrumentation Operation Prior to End of Time Frame 1 - Core Uncovery and Heatup

Action	Equipment	Instrumentation	Purpose	Comment
Accident Monitoring		• SG radiation • containment pressure • containment temperature • containment hydrogen monitors • containment water level • containment radiation • containment atmosphere sampling functions • containment liquid sampling function • auxiliary building radiation • core-exit t/c's • RCS pressure • IRWST water level	• accident management • emergency response(1) • emergency action levels(1)	• required by 10 CFR 50.34(f)

Note:
1. Note that the instrumentation required for emergency response and emergency action levels is an open item because the EALs are not yet developed.

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Table 19D-4 (Sheet 1 of 3)
Equipment and Instrumentation Operation During Time Frame 2 -
In-Vessel Core Melting and Relocation

Action	Equipment	Instrumentation	Purpose	Comment
Inject into RCS	• CMT • accumulator • IRWST • CVS • RNS	• RCS pressure • containment pressure • CVS flow • RNS flow • RCS temperature	• cool core debris in-vessel	• RCS injection needed to cool in-vessel debris for reasonable assurance of controlled, stable state
Decay Heat Removal	• via SGs	• SG WR water level • steam line pressure • core-exit t/c's • RCS RTDs	• decay heat removal	
Inject Into Containment	• containment spray • overflow IRWST • RNS • IRWST drains	• containment water level	• prevent vessel failure	• containment spray only actuated on high containment radiation in SAMG which occurs in Time Frame 2
Inject to SGs	• MFW • SFW	• SG WR water level • steam line pressure	• isolate containment in SGTR • scrub fission products	• also requires RCS depressurization for containment isolation
Depressurize RCS	• ADS • aux pressurizer spray • reactor vessel head vent	• RCS Pressure	• prevent vessel failure • containment integrity	• needed for in-vessel retention of core debris • needed for prevention of TI-SGTR • try to recover in Time Frame 2, if not successful in Time Frame 1

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Table 19D-4 (Sheet 2 of 3)
Equipment and Instrumentation Operation During Time Frame 2 -
In-Vessel Core Melting and Relocation

Action	Equipment	Instrumentation	Purpose	Comment
Depressurize SGs				• not needed in Time Frame 2
Containment Heat Removal	• PCS water • external water • containment spray	• PCS flowrate • PCS tank level • containment water level • containment pressure	• containment integrity	• active operation completed in Time Frame 1; needs to be continued in Time Frame 2
Containment Isolation	• containment shell • penetrations • containment hatches (gasket material only)	• containment pressure	• containment integrity	• containment isolation system active operation completed in Time Frame 1
Control Hydrogen	• igniters • PARs	• containment hydrogen monitors • containment atmosphere sampling function • containment pressure	• containment integrity	• active operation continues in Time Frame 2 • monitors only required initially to verify hydrogen igniter operation
Control Fission Product Releases	• containment spray	• containment water level	• scrub fission products	• containment spray only actuated on high containment radiation in SAMG which occurs in Time Frame 2

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Table 19D-4 (Sheet 3 of 3)
Equipment and Instrumentation Operation During Time Frame 2 -
In-Vessel Core Melting and Relocation

Action	Equipment	Instrumentation	Purpose	Comment
Accident Monitoring		• SG radiation • containment pressure • containment temperature • containment hydrogen monitors • containment water level • containment radiation • containment atmosphere sampling functions • containment liquid sampling function • auxiliary building radiation • RCS pressure	• accident management • emergency response(1) • emergency action levels(1)	• active operation continues in Time Frame 2

Note:
1. Note that the instrumentation required for emergency response and emergency action levels is an open item because the EALs are not yet developed.

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Table 19D-5 (Sheet 1 of 2)
Equipment and Instrumentation Operation During Time Frame 3 - Ex-Vessel Core Relocation

Action	Equipment	Instrumentation	Purpose	Comment
Inject into RCS				• not needed in Time Frame 3
Decay heat removal				• not needed in Time Frame 3
Inject into SGs				• not needed in Time Frame 3
Depressurize RCS				• not needed in Time Frame 3
Depressurize SGs				• not needed in Time Frame 3
Inject Into Containment	• Containment spray • Overflow IRWST - CVS - RNS - SFS • IRWST drains	• containment water level	• cool ex-vessel core debris to prevent or mitigate consequences of CCI • scrub fission products released from ex-vessel core debris	• only get to Time Frame 3 if there is no water in containment or if RCS depressurization fails
Containment Heat Removal	• PCS water • external water • containment spray	• PCS flowrate • PCS tank level • containment water level • containment pressure	• containment integrity	• active operation completed in Time Frame 1; needs to be continued in Time Frame 3
Containment Isolation	• containment shell • penetrations • containment hatches (gasket material only)	• containment pressure • containment temperature	• containment integrity	• active operation of containment isolation system completed in Time Frame 1
	• RNS hot leg suction MOVs	• containment pressures • SFP water level	• containment vent	• manual action within SAMG
Control Hydrogen	• igniters	• containment atmosphere sampling function • containment pressure	• containment integrity	• active operation continues in Time Frame 3 • PARS may be effective in Time Frame 3 if igniters are not effective

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Table 19D-5 (Sheet 2 of 2)
Equipment and Instrumentation Operation During Time Frame 3 - Ex-Vessel Core Relocation

Action	Equipment	Instrumentation	Purpose	Comment
Control Fission Product Release	• containment spray	• containment water level	• scrub fission products	• containment spay only actuated on high containment radiation in SAMG which occurs in Time Frame 3
Accident Monitoring		• SG radiation • containment pressure • containment temperature • containment hydrogen monitors • containment water level • containment radiation • containment atmosphere sampling function • containment liquid sampling function • auxiliary building radiation monitors	• accident management • emergency response(1) • emergency action levels(1)	• active operation continues in Time Frame 3

Note:
1. The instrumentation required for emergency response and emergency action levels is an open item because the EALs are not yet developed.

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Table 19D-6 Not Used

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Table 19D-7 (Sheet 1 of 3)
Sustained Hydrogen Combustion Survivability Assessment

EQUIPMENT AND INSTRUMENTATION	SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT
Equipment	
PXS equipment (injection)	The PXS equipment utilized for introduction of cooling water includes component redundancy and is separated into two delivery flow paths. The two flow paths are physically separated into two trains such that if one train is disabled due to a sustained burn from DVI or other line break within that subsystem, the other subsystem will function.
CVS equipment (injection)	The equipment providing for CVS injection is located within the CVS compartment with the exception of the CVS makeup isolation valve. In accordance with the above, a sustained burn will not occur within the CVS compartment and, therefore, the equipment within this compartment utilized for CVS makeup will be operable. The CVS makeup isolation valve is normally in the correct position for severe accident scenario and is considered operable.
RNS equipment (injection)	Injection via the RNS is dependent only upon check valves within containment and, therefore, is not susceptible to sustained burning effects.
Main Feedwater	The operability of main feedwater system to inject feedwater to steam generators is not dependent upon equipment located within containment and, therefore, is not susceptible to sustained burning effects.
Startup Feedwater	The operability of startup feedwater system to inject feedwater to steam generators is not dependent upon equipment located within containment and, therefore, is not susceptible to sustained burning effects.
Fire Water, containment spray, and external containment vessel cooling	The operability of the fire water system to provide makeup for containment spray and for external containment vessel cooling is not dependent upon equipment located within containment and, therefore, is not susceptible to sustained burning effects.

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Table 19D-7 (Sheet 2 of 3)
Sustained Hydrogen Combustion Survivability Assessment

EQUIPMENT AND INSTRUMENTATION	SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT
Equipment	
Containment Shell	As discussed in Subsection 19.18.7 of this document, hydrogen plumes are located away from the containment shell to mitigate the threat to the containment integrity.
Containment Lower Equipment Hatch and Seals	The lower equipment hatch and seals on the containment vessel may be exposed to heat transfer from a sustained flame at the vents from the PXS valve/accumulator room to the maintenance floor. The equipment hatch and seals have been shown by analysis to be unlikely to fail or leak.
Igniters	Igniters are specified and designed to withstand the effects of sustained burning and, therefore, are considered operable for these events.
Instrumentation	
RCS Pressure	There are four RCS pressurizer pressure transmitters. Two transmitters are located at a distance greater than 75 feet from the vent from the PXS valve/accumulator room and are therefore beyond the distance that potentially causes operability concerns from a sustained flame. The other two transmitters are located in a different room from the fourth stage ADS valves. This precludes radiative heating, which could potentially cause operability concerns.
Containment Pressure	There are three extended range containment pressure transmitters. The three transmitters are located such that they cannot all be exposed to a sustained flame from either of the vents from the PXS valve/accumulator room into the maintenance floor at the base of the CMTs. Therefore, continued operability of the containment pressure function is provided.
SG 1 Wide Range Level	There are four steam generator wide range levels for SG 1. Two of the transmitters are located at a distance of greater than 20 feet from a CMT and are, therefore, beyond the distance that could potentially cause operability concerns from a sustained flame from the vent from the PXS valve/accumulator room into the maintenance floor at the base of the CMT. The other two transmitters are located over 20 feet below the fourth stage ADS valves. This precludes radiative heating, which could potentially cause operability concerns.
SG 2 Wide Range Level	Based on the layout of the four steam generator wide range levels for SG 2, at least two of the transmitters will not be exposed to a sustained flame from either of the vents from the PXS valve/accumulator room into the maintenance floor at the base of the CMTs. Therefore, continued operability of the SG 2 wide range level indication function is provided.

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Table 19D-7 (Sheet 3 of 3)
Sustained Hydrogen Combustion Survivability Assessment

EQUIPMENT AND INSTRUMENTATION	SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT
Instrumentation	
Containment Hydrogen Monitors	There are 3 containment hydrogen monitors located in the containment dome. There are no sustained burns that could potentially affect the sensors.
Containment Atmosphere Sampling Function	The capabilities to perform containment atmosphere sampling are discussed in Subsection 9.3.3.1.2.2 – Post-Accident Sampling. Successful containment atmosphere sampling is dependent on the availability of the containment atmosphere sampling isolation valves. The sample return function is provided by the containment air sample return containment isolation valves. The containment atmosphere sampling function provides a backup to the VLS hydrogen monitoring. The containment isolation valves are located less than 20 feet from the PXS-A compartment vent. The survivability of the valves has been considered in their placement in the containment. The sampling valve is located in Room 11300 which does not normally have a direct hydrogen source term. In the event of a specific primary pipe break (DVI line break), the potential hydrogen releases from the break would be mitigated by the ADS-4 valves and the hydrogen would be released to the loop compartment instead of Room 11300. The combination of failures that may be postulated to produce hydrogen releases and sustained burning in the vicinity has a low probability. There is reasonable assurance for the survivability of the valves by demonstrating that the valves survive the more likely hydrogen release scenarios.
Containment Liquid Sampling Function	The capabilities to perform containment liquid sampling are discussed in Subsection 9.3.3.1.2.2 – Post-Accident Sampling. Successful sampling is dependent on the availability of either of the hot leg sample source isolation valves and the containment isolation valves in series with the isolation valve. The sample isolation valve from reactor coolant hot leg number 1 is located in a different room from the fourth stage ADS valves. This precludes radiative heating, which could potentially cause operability concerns. The sample isolation valve from reactor coolant hot leg number 2 is located in a different room from the fourth stage ADS valves. This precludes radiative heating, which could potentially cause operability concerns. The containment isolation valves are located less than 20 feet from the PXS-A compartment vent. The survivability of the valves has been considered in their placement in the containment. The sampling valve is located in Room 11300 which does not normally have a direct hydrogen source term. In the event of a specific primary pipe break (DVI line break), potential hydrogen releases from the break would be mitigated by the ADS-4 valves and the hydrogen would be released to the loop compartment instead of Room 11300. The combination of failures that may be postulated to produce hydrogen releases from the break (DVI line break and common cause failures of multiple ADS-4 valves) and sustained burning in the vicinity of these valves has a low probability. Defense-in-depth precluding sustained combustion in this room provides reasonable assurance for the survivability of the valves, which are demonstrated to survive the more likely hydrogen release scenarios.

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Table 19D-8 (Sheet 1 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
19D.8.2.1	**Differential Pressure and Pressure Transmitters**				
	PXS – IRWST Water Level	**T0,T1-A1,A2,A3,A4,A10**		**2**	**Sense level**
PXS-JE-LT046	IRWST Level Transmitter		11300		
PXS-JE-LT047	IRWST Level Transmitter		11300		
PXS-JE-LT048	IRWST Level Transmitter		11300		
	RCS Pressure	**T0,T1-A1,A4,A5,A10**		**2**	**Sense pressure**
RCS-JE-PT140A	RCS Wide Range Pressure Trans. A		11300		
RCS-JE-PT140C	RCS Wide Range Pressure Trans. C		11300		
RCS-JE-PT140B	RCS Wide Range Pressure Trans. B		11306		
RCS-JE-PT140D	RCS Wide Range Pressure Trans. D		11306		
	SG Wide Range Water Level	**T0,T1-A3,A5**		**2 per SG**	**Sense level**
SGS-JE-LT012	SG 1 Wide Range Level Transmitter		11201		
SGS-JE-LT016	SG 1 Wide Range Level Transmitter		11201		
SGS-JE-LT011	SG 1 Wide Range Level Transmitter		11300		
SGS-JE-LT015	SG 1 Wide Range Level Transmitter		11300		
SGS-JE-LT013	SG 2 Wide Range Level Transmitter		11300		
SGS-JE-LT014	SG 2 Wide Range Level Transmitter		11300		
SGS-JE-LT017	SG 2 Wide Range Level Transmitter		11300		
SGS-JE-LT018	SG 2 Wide Range Level Transmitter		11300		
	Containment Pressure	**T0,T1-A6,A7,A8,A10**		**2**	**Sense pressure**
PCS-JE-PT012	Extended Range Cont. Pressure 1		11400		
PCS-JE-PT013	Extended Range Cont. Pressure 2		11400		
PCS-JE-PT014	Extended Range Cont. Pressure 3		11400		
19D.8.2.2	**Thermocouples**				
	Core-exit Temperature	**T0,T1-A1,A2,A3,A10**		**2 (Note 2)**	**Sense temperature**
IIS-JE-TE002	Core-exit Thermocouple, B09		11105		
IIS-JE-TE003	Core-exit Thermocouple, C04		11105		
IIS-JE-TE005	Core-exit Thermocouple, C08		11105		
IIS-JE-TE008	Core-exit Thermocouple, E02		11105		
IIS-JE-TE010	Core-exit Thermocouple, E06		11105		
IIS-JE-TE012	Core-exit Thermocouple, E10		11105		

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Table 19D-8 (Sheet 2 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
IIS-JE-TE015	Core-exit Thermocouple, G02		11105		
IIS-JE-TE016	Core-exit Thermocouple, G04		11105		
IIS-JE-TE017	Core-exit Thermocouple, G06		11105		
IIS-JE-TE020	Core-exit Thermocouple, G12		11105		
IIS-JE-TE021	Core-exit Thermocouple, G14		11105		
IIS-JE-TE025	Core-exit Thermocouple, J08		11105		
IIS-JE-TE026	Core-exit Thermocouple, J10		11105		
IIS-JE-TE030	Core-exit Thermocouple, L04		11105		
IIS-JE-TE033	Core-exit Thermocouple, L10		11105		
IIS-JE-TE035	Core-exit Thermocouple, L14		11105		
IIS-JE-TE038	Core-exit Thermocouple, N08		11105		
IIS-JE-TE039	Core-exit Thermocouple, N10		11105		
IIS-JE-TE040	Core-exit Thermocouple, N12		11105		
IIS-JE-TE041	Core-exit Thermocouple, P07		11105		
IIS-JE-TE001	Core-exit Thermocouple, B07		11105		
IIS-JE-TE004	Core-exit Thermocouple, C06		11105		
IIS-JE-TE006	Core-exit Thermocouple, C10		11105		
IIS-JE-TE009	Core-exit Thermocouple, E04		11105		
IIS-JE-TE013	Core-exit Thermocouple, E12		11105		
IIS-JE-TE014	Core-exit Thermocouple, E14		11105		
IIS-JE-TE018	Core-exit Thermocouple, G08		11105		
IIS-JE-TE019	Core-exit Thermocouple, G10		11105		
IIS-JE-TE022	Core-exit Thermocouple, J02		11105		
IIS-JE-TE023	Core-exit Thermocouple, J04		11105		
IIS-JE-TE024	Core-exit Thermocouple, J06		11105		
IIS-JE-TE027	Core-exit Thermocouple, J12		11105		
IIS-JE-TE029	Core-exit Thermocouple, L02		11105		
IIS-JE-TE031	Core-exit Thermocouple, L06		11105		
IIS-JE-TE034	Core-exit Thermocouple, L12		11105		

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Table 19D-8 (Sheet 3 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
IIS-JE-TE036	Core-exit Thermocouple, N04		11105		
IIS-JE-TE037	Core-exit Thermocouple, N06		11105		
IIS-JE-TE007	Core-exit Thermocouple, C12		11105		
IIS-JE-TE011	Core-exit Thermocouple, E08		11105		
IIS-JE-TE028	Core-exit Thermocouple, K14		11105		
IIS-JE-TE032	Core-exit Thermocouple, L08		11105		
IIS-JE-TE042	Core-exit Thermocouple, P09		11105		
19D.8.2.3	**Resistance Temperature Detectors (RTDs)**				
	Hot Leg RTDs	**T0,T1-A1,A3**		**2**	**Sense temperature**
RCS-JE-TE135A	RCS Hot Leg 1 Wide Range RTD		11201		
RCS-JE-TE135B	RCS Hot Leg 2 Wide Range RTD		11202		
	Cold Leg RTDs	**T0,T1-A1,A3**		**2**	**Sense temperature**
RCS-JE-TE125A	RCS Cold Leg 1A Wide Range RTD		11201		
RCS-JE-TE125C	RCS Cold Leg 1B Wide Range RTD		11201		
RCS-JE-TE125B	RCS Cold Leg 2A Wide Range RTD		11202		
RCS-JE-TE125D	RCS Cold Leg 2B Wide Range RTD		11202		
	Containment Temperature	**T0,T1-A10**		**2**	**Sense temperature**
VCS-JE-TE053A	Operating Floor Area Temperature A		11500		
VCS-JE-TE053B	Operating Floor Area Temperature B		11500		
	IRWST Water Temperature	**T0,T1-A3,A4**		**2**	**Sense temperature**
PXS-JE-TE041	IRWST Water Temperature		11300, 11305		
PXS-JE-TE042	IRWST Water Temperature		11300, 11305		
PXS-JE-TE043	IRWST Water Temperature		11300, 11305		
PXS-JE-TE044	IRWST Water Temperature		11300, 11305		
19D.8.2.4	**Hydrogen Monitors**				
	Hydrogen Monitors	**T0,T1-A8,A10**		**2**	**Sense concentration**
VLS-JE-AE001	Hydrogen Monitor		11500		
VLS-JE-AE002	Hydrogen Monitor		11500		
VLS-JE-AE003	Hydrogen Monitor		11500		

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Table 19D-8 (Sheet 4 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
19D.8.2.5	**Radiation Monitors**				
	PXS Radiation Monitors	**T0,T1-A10**		**2**	**Sense radiation**
PXS-JE-RE160	Containment Radiation Detector A		11500		
PXS-JE-RE161	Containment Radiation Detector B		11500		
PXS-JE-RE162	Containment Radiation Detector C		11500		
PXS-JE-RE163	Containment Radiation Detector D		11500		
19D.8.2.6	**Solenoid-Operated Valves (SOVs) and Air-Operated Valves (AOVs)**				
	PXS Core Makeup Tank AOVs	**T0,T1-A1**		**1 per CMT**	**Transfer open**
PXS-PL-V014A	CMT A Discharge Isolation		11206		
PXS-PL-V015A	CMT A Discharge Isolation		11206		
PXS-PL-V014B	CMT B Discharge Isolation		11207		
PXS-PL-V015B	CMT B Discharge Isolation		11207		
	PXS PRHR AOVs	**T0,T1-A3,A4**		**1**	**Transfer open**
PXS-PL-V108A	PRHR HX Control A		11300		
PXS-PL-V108B	PRHR HX Control B		11300		
	Containment Isolation AOVs and SOVs	**T1-A7**		**All**	**Varies, see below**
CVS-PL-V045	Letdown Containment Isolation		11300		Transfer close
PSS-PL-V008	Cont. Air Sample Cont. Isolation		11400		Transfer open and close
PSS-PL-V010A	Liquid Sample Line Cont. Isolation A		11400		Transfer open and close
PSS-PL-V010B	Liquid Sample Line Cont. Isolation B		11400		Transfer open and close
PSS-PL-V024	Containment Atmosphere Sample Return Isolation		11400		Transfer open and close
RNS-PL-V061	RNS Return from CVS Cont. Isolation		11208		Transfer close
VFS-PL-V004	Cont. Purge Inlet Cont. Isolation		11400		Transfer open and close
VFS-PL-V009	Cont. Purge Discharge Cont. Isolation		11400		Transfer open and close
VWS-PL-V082	Fan Coolers Return Cont. Isolation		11500		Transfer close
WLS-PL-V055	Sump Discharge Cont. Isolation		11300		Transfer close
WLS-PL-V067	RCDT Gas Outlet Cont. Isolation		11300		Transfer close

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Table 19D-8 (Sheet 5 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
	CVS RCS Boundary AOVs	**T0,T1-A1**		**All**	**Transfer open and close**
CVS-PL-V081	RCS Purification Return Line Stop Valve		11304		
CVS-PL-V084	Auxiliary Pressurizer Spray Line Isolation		11304		
	Containment Sampling Function	**T0,T1-A8,A10**		**All**	**Transfer open and close**
PSS-PL-V001A/B	Hot Leg Sample Isol Valves	A10	11300		
PSS-PL-V010A/B	Liquid Sample Cont. Isol Valves	A10	11400		
PSS-PL-V008	Containment Air Sample Cont. Isolation	A8, A10	11400		
PSS-PL-V024	Containment Atmosphere Sample Return Isolation	A8, A10	11400		
PSS-PL-V002	Containment Sump Sample Isolation Valve	A10	11400		
	Reactor Vessel Head Vent SOVs	**T1-A4**		**2 valves in the same flowpath**	**Transfer open and close**
RCS-PL-V150A	Reactor Vessel Head Vent		11601		
RCS-PL-V150B	Reactor Vessel Head Vent		11601		
RCS-PL-V150C	Reactor Vessel Head Vent		11601		
RCS-PL-V150D	Reactor Vessel Head Vent		11601		
19D.8.2.7	**Motor-Operated Valves (MOVs)**				
	PXS Accumulator MOVs	**T1-A1**		**1 per Accum**	**Transfer open**
PXS-PL-V027A	Accumulator A Discharge Isolation		11206		
PXS-PL-V027B	Accumulator B Discharge Isolation		11207		
	PXS Core Makeup Tank MOVs	**T0,T1-A1**		**1 per CMT**	**Transfer open**
PXS-PL-V002A	CMT A Inlet Isolation		11400		
PXS-PL-V002B	CMT B Inlet Isolation		11400		
	PXS Recirculation MOVs	**T1-A2**		**1**	**Transfer open**
PXS-PL-V117A	Cont. Recirculation. A Isolation		11206		
PXS-PL-V117B	Cont. Recirculation. B Isolation		11207		
	ADS Stages 1, 2, 3, & 4 MOVs	**T1-A4**		**1 flowpath per stage**	**Transfer open**
RCS-PL-V001A	First Stage ADS A		11703		
RCS-PL-V011A	First Stage ADS Isolation A		11703		
RCS-PL-V002A	Second Stage ADS A		11703		
RCS-PL-V012A	Second Stage ADS Isolation A		11703		

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Table 19D-8 (Sheet 6 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
RCS-PL-V003A	Third Stage ADS A		11703		
RCS-PL-V013A	Third Stage ADS Isolation A		11703		
RCS-PL-V001B	First Stage ADS B		11603		
RCS-PL-V011B	First Stage ADS Isolation B		11603		
RCS-PL-V002B	Second Stage ADS B		11603		
RCS-PL-V012B	Second Stage ADS Isolation B		11603		
RCS-PL-V003B	Third Stage ADS B		11603		
RCS-PL-V013B	Third Stage ADS Isolation B		11603		
RCS-PL-V014A	Fourth Stage ADS MOV		11301		
RCS-PL-V014B	Fourth Stage ADS MOV		11302		
RCS-PL-V014C	Fourth Stage ADS MOV		11301		
RCS-PL-V014D	Fourth Stage ADS MOV		11302		
	Containment Isolation MOVs	**T1-A7**		**All**	**Varies, see below**
CCS-PL-V207	CCS Containment Isolation Valve		11400		Transfer close
CVS-PL-V091	Makeup Line Cont. Isolation		11300		Transfer open and close
RNS-PL-V002A	RNS HL Suction & Cont. Isolation – A		11208		Transfer open and close
RNS-PL-V002B	RNS HL Suction & Cont. Isolation – B		11208		Transfer open and close
RNS-PL-V023	RNS Suction from IRWST – Cont. Isolation		11208		Transfer open and close
SFS-PL-V034	SFS Suction Line Cont. Isolation		11206		Transfer close
	CVS Charging and Injection MOVs	**T0, T1-A1**		**1**	**Transfer open and close**
CVS-PL-V091	Makeup Line Cont. Isolation		11300		
	RNS MOVs	**T1-A1**		**2**	**Transfer open and close**
RNS-PL-V001A/B	RNS Suction Isol from Hot Leg		11208		
RNS-PL-V002A/B	RNS Suction Isol from Hot Leg		11208		
RNS-PL-V023	RNS Suction from IRWST – Cont. Isolation		11208		
RNS-PL-V024	IRWST Discharge Isolation Valve		11206		

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Table 19D-8 (Sheet 7 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
	PXS PRHR MOV	**T0, T1-A3, A4**		**1**	**Transfer open**
PXS-PL-V101	PRHR HX Inlet Isolation		11500		
19D.8.2.8	**Squib Valves**				
	IRWST Injection	**T1-A2**		**2**	**Transfer open**
PXS-PL-V123A	IRWST Injection Line A Squib Valve		11206		
PXS-PL-V125A	IRWST Injection Line A Squib Valve		11206		
PXS-PL-V123B	IRWST Injection Line B Squib Valve		11207		
PXS-PL-V125B	IRWST Injection Line B Squib Valve		11207		
	PXS Containment Recirculation	**T1-A2**		**2**	**Transfer open**
PXS-PL-V118A	Cont. Recirculation A Squib Valve		11206		
PXS-PL-V120A	Cont. Recirculation A Squib Valve		11206		
PXS-PL-V118B	Cont. Recirculation B Squib Valve		11207		
PXS-PL-V120B	Cont. Recirculation B Squib Valve		11207		
	Fourth Stage ADS	**T1-A4**		**3**	**Transfer open**
RCS-PL-V004A	Fourth Stage ADS A		11301		
RCS-PL-V004C	Fourth Stage ADS C		11301		
RCS-PL-V004B	Fourth Stage ADS B		11302		
RCS-PL-V004D	Fourth Stage ADS D		11302		
19D.8.2.9	**Position Sensors**				
	Valve Position Sensors	**T1-A7**		**All**	**Sense valve position**
CCS-PL-V207-L	CCS Containment Isolation Valve		11400		
CVS-PL-V045-L	Letdown Containment Isolation		11300		
CVS-PL-V091-L	Makeup Line Cont. Isolation		11300		
PSS-PL-V008-L	Containment Air Sample Cont. Isolation		11400		
PSS-PL-V010A-L	Liquid Sample Cont. Isolation Valve A		11400		
PSS-PL-V010B-L	Liquid Sample Cont. Isolation Valve B		11400		
PSS-PL-V024-L	Containment Atmosphere Sample Return Isolation		11400		
RNS-PL-V002A-L	RNS HL Suction & Cont. Isolation – A		11208		
RNS-PL-V002B-L	RNS HL Suction & Cont. Isolation – B		11208		
RNS-PL-V023-L	RNS Suction from IRWST – Cont. Isolation		11208		
RNS-PL-V061-L	RNS Return from CVS – Cont. Isolation		11208		

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Table 19D-8 (Sheet 8 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
SFS-PL-V034-L	SFS Suction Line Cont. Isolation.		11206		
VFS-PL-V004-L	Containment Purge Inlet Cont. Isolation		11400		
VFS-PL-V009-L	Containment Discharge Cont. Isolation		11400		
VWS-PL-V082-L	Fan Coolers Return Cont. Isolation Valve		11500		
WLS-PL-V055-L	Sump Discharge Cont. Isolation		11300		
WLS-PL-V067-L	RCDT Gas Outlet Cont. Isolation		11300		
19D.8.2.10	**Hydrogen Igniters**				
	Hydrogen Igniters	**T1-A8**		**Note 3**	**Ignite hydrogen**
VLS-EH-01	Hydrogen Igniter 01, 1		11204		
VLS-EH-02	Hydrogen Igniter 02, 2		11204		
VLS-EH-03	Hydrogen Igniter 03, 1		11204		
VLS-EH-04	Hydrogen Igniter 04, 2		11204		
VLS-EH-05	Hydrogen Igniter 05, 1		11402		
VLS-EH-06	Hydrogen Igniter 06, 2		11502		
VLS-EH-07	Hydrogen Igniter 07, 2		11402		
VLS-EH-08	Hydrogen Igniter 08, 1		11502		
VLS-EH-09	Hydrogen Igniter 09, 1		11305		
VLS-EH-10	Hydrogen Igniter 10, 2		11305		
VLS-EH-11	Hydrogen Igniter 11, 2		11401		
VLS-EH-12	Hydrogen Igniter 12, 1		11501		
VLS-EH-13	Hydrogen Igniter 13, 1		11401		
VLS-EH-14	Hydrogen Igniter 14, 2		11501		
VLS-EH-15	Hydrogen Igniter 15, 2		11305		
VLS-EH-16	Hydrogen Igniter 16, 1		11305		
VLS-EH-17	Hydrogen Igniter 17, 2		11207		
VLS-EH-18	Hydrogen Igniter 18, 1		11207		
VLS-EH-19	Hydrogen Igniter 19, 2		11208		
VLS-EH-20	Hydrogen Igniter 20, 2		11206		
VLS-EH-21	Hydrogen Igniter 21, 1		11206		
VLS-EH-22	Hydrogen Igniter 22, 1		11400		
VLS-EH-23	Hydrogen Igniter 23, 2		11400		
VLS-EH-24	Hydrogen Igniter 24, 2		11400		

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Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
VLS-EH-25	Hydrogen Igniter 25, 2		11400		
VLS-EH-26	Hydrogen Igniter 26, 2		11400		
VLS-EH-27	Hydrogen Igniter 27, 1		11300		
VLS-EH-28	Hydrogen Igniter 28, 1		11400		
VLS-EH-29	Hydrogen Igniter 29, 1		11400		
VLS-EH-30	Hydrogen Igniter 30, 2		11403		
VLS-EH-31	Hydrogen Igniter 31, 1		11400		
VLS-EH-32	Hydrogen Igniter 32, 1		11400		
VLS-EH-33	Hydrogen Igniter 33, 2		11209		
VLS-EH-34	Hydrogen Igniter 34, 1		11209		
VLS-EH-35	Hydrogen Igniter 35, 1		11500		
VLS-EH-36	Hydrogen Igniter 36, 2		11500		
VLS-EH-37	Hydrogen Igniter 37, 1		11500		
VLS-EH-38	Hydrogen Igniter 38, 2		11500		
VLS-EH-39	Hydrogen Igniter 39, 1		11500		
VLS-EH-40	Hydrogen Igniter 40, 2		11500		
VLS-EH-41	Hydrogen Igniter 41, 2		11500		
VLS-EH-42	Hydrogen Igniter 42, 1		11500		
VLS-EH-43	Hydrogen Igniter 43, 1		11500		
VLS-EH-44	Hydrogen Igniter 44, 1		11500		
VLS-EH-45	Hydrogen Igniter 45, 2		11500		
VLS-EH-46	Hydrogen Igniter 46, 2		11500		
VLS-EH-47	Hydrogen Igniter 47, 1		11500		
VLS-EH-48	Hydrogen Igniter 48, 2		11500		
VLS-EH-49	Hydrogen Igniter 49, 1		11503		
VLS-EH-50	Hydrogen Igniter 50, 2		11503		
VLS-EH-51	Hydrogen Igniter 51, 1		11500		
VLS-EH-52	Hydrogen Igniter 52, 2		11500		
VLS-EH-53	Hydrogen Igniter 53, 2		11500		
VLS-EH-54	Hydrogen Igniter 54, 1		11500		
VLS-EH-55	Hydrogen Igniter 55, 1		11504		
VLS-EH-56	Hydrogen Igniter 56, 2		11504		

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Table 19D-8 (Sheet 10 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
VLS-EH-57	Hydrogen Igniter 57, 2		11504		
VLS-EH-58	Hydrogen Igniter 58, 1		11504		
VLS-EH-59	Hydrogen Igniter 59, 2		11503		
VLS-EH-60	Hydrogen Igniter 60, 1		11503		
VLS-EH-61	Hydrogen Igniter 61, 1		11500		
VLS-EH-62	Hydrogen Igniter 62, 2		11500		
VLS-EH-63	Hydrogen Igniter 63, 1		11500		
VLS-EH-64	Hydrogen Igniter 64, 2		11500		
VLS-EH-65	Hydrogen Igniter 65, 1		11500		
VLS-EH-66	Hydrogen Igniter 66, 2		11500		
19D.8.2.11	**Electrical Containment Penetration Assemblies**				
	Electrical Containment Penetration Assemblies	**T1-A7**		**All**	**Maintain containment boundary; Conduct electricity for sensors and igniters**
ECS-EY-P01X	Electrical Penetration P01		11400		
ECS-EY-P02X	Electrical Penetration P02		11400		
DAS-EY-P03Z	Electrical Penetration P03		11400		
ECS-EY-P06Y	Electrical Penetration P06		11400		
ECS-EY-P07X	Electrical Penetration P07		11400		
ECS-EY-P09W	Electrical Penetration P09		11400		
ECS-EY-P10W	Electrical Penetration P10		11400		
IDSA-EY-P11Z	1E Electrical Penetration P11		11400		
IDSA-EY-P12Y	1E Electrical Penetration P12		11400		
IDSA-EY-P13Y	1E Electrical Penetration P13		11400		
IDSD-EY-P14Z	1E Electrical Penetration P14		11306		
IDSD-EY-P15Y	1E Electrical Penetration P15		11306		
IDSD-EY-P16Y	1E Electrical Penetration P16		11306		
ECS-EY-P17X	Electrical Penetration P17		11300		
ECS-EY-P18X	Electrical Penetration P18		11300		

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Table 19D-8 (Sheet 11 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
ECS-EY-P19Z	Electrical Penetration P19		11300		
ECS-EY-P20Z	Electrical Penetration P20		11300		
EDS-EY-P21Z	Electrical Penetration P21		11300		
ECS-EY-P22X	Electrical Penetration P22		11300		
ECS-EY-P23X	Electrical Penetration P23		11300		
ECS-EY-P24	Spare Electrical Penetration		11300		
ECS-EY-P25W	Electrical Penetration P25		11300		
ECS-EY-P26W	Electrical Penetration P26		11300		
IDSC-EY-P27Z	1E Electrical Penetration P27		11300		
IDSC-EY-P28Y	1E Electrical Penetration P28		11300		
IDSC-EY-P29Y	1E Electrical Penetration P29		11300		
IDSB-EY-P30Z	1E Electrical Penetration P30		11306		
IDSB-EY-P31Y	1E Electrical Penetration P31		11306		
IDSB-EY-P32Y	1E Electrical Penetration P32		11306		
19D.8.2.12	**Cables**				
	Cables	**T0,T1-A10**	**Note 1**	**All**	**Conduct electricity for valves, sensors and igniters**
19D.8.2.13	**Float Level Sensors**				
	PXS Containment Water Level	**T0,T1-A2,A10**		**2**	**Sense level**
PXS-JE-LE050	Containment Floodup Level		11104		
PXS-JE-LE052	Containment Floodup Level		11104		
PXS-JE-LE051	Containment Floodup Level		11104		
19D.8.2.14	**Containment Hatches (Gasket Material Only)**				
	Containment Hatches (Gasket Material Only)	**T0,T1-A7**		**All**	**Maintain containment boundary**
CNS-MY-Y01	Equipment Hatch		11500		
CNS-MY-Y02	Maintenance Hatch		11300		
CNS-MY-Y03	Personnel Hatch – Upper		11500		
CNS-MY-Y04	Personnel Hatch – Lower		11300		

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Table 19D-8 (Sheet 12 of 12)
Equipment Located Inside Containment (Time Frames T0 and T1)

Section and Equip Tag No.	Description	Time Frames T0,T1 and Actions (Notes 4, 5)	Room Number	Required Channels/ Divisions	Function
19D.8.2.15	**Containment Vessel**				
	Containment Vessel	**T0,T1-A7**			**Maintain containment boundary**
CNS-MV-01	Containment Shell				

Notes:
1. Cables are located in most rooms inside containment.
2. A channel consists of two thermocouples within a single division. Each quadrant contains two divisions. The minimum requirement is two OPERABLE thermocouples in each of the two divisions for each quadrant. All Core Exit Thermocouples are located inside the Reactor Pressure Vessel which is in Room 11105 (reactor cavity).
3. There are two groups of igniters. Each containment compartment and hydrogen pathway is covered by at least one igniter in each group. The minimum requirement is one igniter is OPERABLE in each compartment and hydrogen pathway.
4. See Appendix 19D, Section 19D.4 for definitions of Equipment Survivability Time Frames T0 and T1.
5. See Appendix 19D, Section 19D.6 for Actions associated with Time Frames T0 and T1.
   - A1 = Subsection 19D.6.1.1
   - A2 = Subsection 19D.6.1.2
   - A3 = Subsection 19D.6.1.3
   - A4 = Subsection 19D.6.1.4
   - A5 = Subsection 19D.6.1.5
   - A6 = Subsection 19D.6.1.6
   - A7 = Subsection 19D.6.1.7
   - A8 = Subsection 19D.6.1.8
   - A10 = Subsection 19D.6.1.9

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Table 19D-9 (Sheet 1 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
19D.8.2.1	**Differential Pressure and Pressure Transmitters**				
	RCS Pressure	**T2-A1, A4, A10**		**2**	**Sense pressure**
RCS-JE-PT140A	RCS Wide Range Pressure Trans. A		11300		
RCS-JE-PT140C	RCS Wide Range Pressure Trans. C		11300		
RCS-JE-PT140B	RCS Wide Range Pressure Trans. B		11306		
RCS-JE-PT140D	RCS Wide Range Pressure Trans. D		11306		
	SG Wide Range Water Level	**T2-A3**		**2 per SG**	**Sense level**
SGS-JE-LT012	SG 1 Wide Range Level Transmitter		11201		
SGS-JE-LT016	SG 1 Wide Range Level Transmitter		11201		
SGS-JE-LT011	SG 1 Wide Range Level Transmitter		11300		
SGS-JE-LT015	SG 1 Wide Range Level Transmitter		11300		
SGS-JE-LT013	SG 2 Wide Range Level Transmitter		11300		
SGS-JE-LT014	SG 2 Wide Range Level Transmitter		11300		
SGS-JE-LT017	SG 2 Wide Range Level Transmitter		11300		
SGS-JE-LT018	SG 2 Wide Range Level Transmitter		11300		
	Containment Pressure	**T2-A1, T2, T3-A6, A7, A8, A10**		**2**	**Sense pressure**
PCS-JE-PT012	Extended Range Cont. Pressure 1		11400		
PCS-JE-PT013	Extended Range Cont. Pressure 2		11400		
PCS-JE-PT014	Extended Range Cont. Pressure 3		11400		
19D.8.2.2	**Thermocouples**				
	Core-exit Temperature	**T2-A3, A10**		**2 (Note 2)**	**Sense temperature**
IIS-JE-TE002	Core-exit Thermocouple, B09		11105		
IIS-JE-TE003	Core-exit Thermocouple, C04		11105		
IIS-JE-TE005	Core-exit Thermocouple, C08		11105		
IIS-JE-TE008	Core-exit Thermocouple, E02		11105		
IIS-JE-TE010	Core-exit Thermocouple, E06		11105		
IIS-JE-TE012	Core-exit Thermocouple, E10		11105		
IIS-JE-TE015	Core-exit Thermocouple, G02		11105		
IIS-JE-TE016	Core-exit Thermocouple, G04		11105		

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Table 19D-9 (Sheet 2 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
IIS-JE-TE017	Core-exit Thermocouple, G06		11105		
IIS-JE-TE020	Core-exit Thermocouple, G12		11105		
IIS-JE-TE021	Core-exit Thermocouple, G14		11105		
IIS-JE-TE025	Core-exit Thermocouple, J08		11105		
IIS-JE-TE026	Core-exit Thermocouple, J10		11105		
IIS-JE-TE030	Core-exit Thermocouple, L04		11105		
IIS-JE-TE033	Core-exit Thermocouple, L10		11105		
IIS-JE-TE035	Core-exit Thermocouple, L14		11105		
IIS-JE-TE038	Core-exit Thermocouple, N08		11105		
IIS-JE-TE039	Core-exit Thermocouple, N10		11105		
IIS-JE-TE040	Core-exit Thermocouple, N12		11105		
IIS-JE-TE041	Core-exit Thermocouple, P07		11105		
IIS-JE-TE001	Core-exit Thermocouple, B07		11105		
IIS-JE-TE004	Core-exit Thermocouple, C06		11105		
IIS-JE-TE006	Core-exit Thermocouple, C10		11105		
IIS-JE-TE009	Core-exit Thermocouple, E04		11105		
IIS-JE-TE013	Core-exit Thermocouple, E12		11105		
IIS-JE-TE014	Core-exit Thermocouple, E14		11105		
IIS-JE-TE018	Core-exit Thermocouple, G08		11105		
IIS-JE-TE019	Core-exit Thermocouple, G10		11105		
IIS-JE-TE022	Core-exit Thermocouple, J02		11105		
IIS-JE-TE023	Core-exit Thermocouple, J04		11105		
IIS-JE-TE024	Core-exit Thermocouple, J06		11105		
IIS-JE-TE027	Core-exit Thermocouple, J12		11105		
IIS-JE-TE029	Core-exit Thermocouple, L02		11105		
IIS-JE-TE031	Core-exit Thermocouple, L06		11105		
IIS-JE-TE034	Core-exit Thermocouple, L12		11105		

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Table 19D-9 (Sheet 3 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
IIS-JE-TE036	Core-exit Thermocouple, N04		11105		
IIS-JE-TE037	Core-exit Thermocouple, N06		11105		
IIS-JE-TE007	Core-exit Thermocouple, C12		11105		
IIS-JE-TE011	Core-exit Thermocouple, E08		11105		
IIS-JE-TE028	Core-exit Thermocouple, K14		11105		
IIS-JE-TE032	Core-exit Thermocouple, L08		11105		
IIS-JE-TE042	Core-exit Thermocouple, P09		11105		
19D.8.2.3	**Resistance Temperature Detectors (RTDs)**				
	Hot Leg RTDs	**T2-A1,A3**		**2**	**Sense temperature**
RCS-JE-TE135A	RCS Hot Leg 1 Wide Range RTD		11201		
RCS-JE-TE135B	RCS Hot Leg 2 Wide Range RTD		11202		
	Cold Leg RTDs	**T2-A1,A3**		**2**	**Sense temperature**
RCS-JE-TE125A	RCS Cold Leg 1A Wide Range RTD		11201		
RCS-JE-TE125C	RCS Cold Leg 1B Wide Range RTD		11201		
RCS-JE-TE125B	RCS Cold Leg 2A Wide Range RTD		11202		
RCS-JE-TE125D	RCS Cold Leg 2B Wide Range RTD		11202		
	Containment Temperature	**T3-A7,T2,T3-A10**		**2**	**Sense temperature**
VCS-JE-TE053A	Operating Floor Area Temperature A		11500		
VCS-JE-TE053B	Operating Floor Area Temperature B		11500		
19D.8.2.4	**Hydrogen Monitors**				
	Hydrogen Monitors	**T2-A8,T2,T3-A10**		**2**	**Sense concentration**
VLS-JE-AE001	Hydrogen Monitor		11500		
VLS-JE-AE002	Hydrogen Monitor		11500		
VLS-JE-AE003	Hydrogen Monitor		11500		
19D.8.2.5	**Radiation Monitors**				
	PXS Radiation Monitors	**T2,T3-A10**		**2**	**Sense radiation**
PXS-JE-RE160	Containment Radiation Detector A		11500		
PXS-JE-RE161	Containment Radiation Detector B		11500		
PXS-JE-RE162	Containment Radiation Detector C		11500		
PXS-JE-RE163	Containment Radiation Detector D		11500		

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Table 19D-9 (Sheet 4 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
19D.8.2.6	**Solenoid-Operated Valves (SOVs) and Air-Operated Valves (AOVs)**				
	CVS RCS Boundary AOVs	**T2-A1,A4**		**All**	**Transfer open and close**
CVS-PL-V081	RCS Purification Return Line Stop Valve		11304		
CVS-PL-V084	Auxiliary Pressurizer Spray Line Isolation		11304		
	Containment Spray AOVs	**T2,T3-A2,A6,A9**	**Note 4**	**All**	**Transfer open**
FPS-PL-V701	Containment Spray Header AOV		11500		
	Containment Sampling Function	**T2,T3-A8,A10**		**All**	**Transfer open and close**
PSS-PL-V001A/B	Hot Leg Sample Isol Valves	**A10**	11300		
PSS-PL-V010A/B	Liquid Sample Cont. Isol Valves	**A10**	11400		
PSS-PL-V008	Containment Air Sample Cont. Isolation	**A8, A10**	11400		
PSS-PL-V024	Containment Atmosphere Sample Return Isolation	**A8, A10**	11400		
PSS-PL-V002	Containment Sump Sample Isolation Valve	**A10**	11400		
	Reactor Vessel Head Vent SOVs	**T2-A4**		**2 valves in the same flowpath**	**Transfer open and close**
RCS-PL-V150A	Reactor Vessel Head Vent		11601		
RCS-PL-V150B	Reactor Vessel Head Vent		11601		
RCS-PL-V150C	Reactor Vessel Head Vent		11601		
RCS-PL-V150D	Reactor Vessel Head Vent		11601		
	PXS AOV	**T2, T3-A2**		**1 valve**	**Transfer open**
CVS-PL-V089	PXS Makeup AOV		11300		
PXS-PL-V234	IRWST makeup AOV		11400		
19D.8.2.7	**Motor-Operated Valves (MOVs)**				
	CVS Charging and Injection MOVs	**T2-A1**		**All**	**Transfer open**
CVS-PL-V091	Makeup Line Cont. Isolation		11300		
	RNS MOVs	**T2,T3-A2, A7**		**All**	**Transfer open and close**
RNS-PL-V024	IRWST Discharge Isolation Valve	A2	11206		
RNS-PL-V001A	RNS HL Suction Isolation – Inner A	A2, A7	11208	**2 valves in the same flowpath**	
RNS-PL-V001B	RNS HL Suction Isolation – Inner B	A2, A7	11208		
RNS-PL-V002A	RNS HL Suction & Cont. Isolation – A	A2, A7	11208		
RNS-PL-V002B	RNS HL Suction & Cont. Isolation – B	A2, A7	11208		

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Table 19D-9 (Sheet 5 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
19D.8.2.10	**Hydrogen Igniters**				
	Hydrogen Igniters	**T2,T3-A8**		**Note 3**	**Ignite hydrogen**
VLS-EH-01	Hydrogen Igniter 01, 1		11204		
VLS-EH-02	Hydrogen Igniter 02, 2		11204		
VLS-EH-03	Hydrogen Igniter 03, 1		11204		
VLS-EH-04	Hydrogen Igniter 04, 2		11204		
VLS-EH-05	Hydrogen Igniter 05, 1		11402		
VLS-EH-06	Hydrogen Igniter 06, 2		11502		
VLS-EH-07	Hydrogen Igniter 07, 2		11402		
VLS-EH-08	Hydrogen Igniter 08, 1		11502		
VLS-EH-09	Hydrogen Igniter 09, 1		11305		
VLS-EH-10	Hydrogen Igniter 10, 2		11305		
VLS-EH-11	Hydrogen Igniter 11, 2		11401		
VLS-EH-12	Hydrogen Igniter 12, 1		11501		
VLS-EH-13	Hydrogen Igniter 13, 1		11401		
VLS-EH-14	Hydrogen Igniter 14, 2		11501		
VLS-EH-15	Hydrogen Igniter 15, 2		11305		
VLS-EH-16	Hydrogen Igniter 16, 1		11305		
VLS-EH-17	Hydrogen Igniter 17, 2		11207		
VLS-EH-18	Hydrogen Igniter 18, 1		11207		
VLS-EH-19	Hydrogen Igniter 19, 2		11208		
VLS-EH-20	Hydrogen Igniter 20, 2		11206		
VLS-EH-21	Hydrogen Igniter 21, 1		11206		
VLS-EH-22	Hydrogen Igniter 22, 1		11400		
VLS-EH-23	Hydrogen Igniter 23, 2		11400		
VLS-EH-24	Hydrogen Igniter 24, 2		11400		
VLS-EH-25	Hydrogen Igniter 25, 2		11400		
VLS-EH-26	Hydrogen Igniter 26, 2		11400		

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Table 19D-9 (Sheet 6 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
VLS-EH-27	Hydrogen Igniter 27, 1		11300		
VLS-EH-28	Hydrogen Igniter 28, 1		11400		
VLS-EH-29	Hydrogen Igniter 29, 1		11400		
VLS-EH-30	Hydrogen Igniter 30, 2		11403		
VLS-EH-31	Hydrogen Igniter 31, 1		11400		
VLS-EH-32	Hydrogen Igniter 32, 1		11400		
VLS-EH-33	Hydrogen Igniter 33, 2		11209		
VLS-EH-34	Hydrogen Igniter 34, 1		11209		
VLS-EH-35	Hydrogen Igniter 35, 1		11500		
VLS-EH-36	Hydrogen Igniter 36, 2		11500		
VLS-EH-37	Hydrogen Igniter 37, 1		11500		
VLS-EH-38	Hydrogen Igniter 38, 2		11500		
VLS-EH-39	Hydrogen Igniter 39, 1		11500		
VLS-EH-40	Hydrogen Igniter 40, 2		11500		
VLS-EH-41	Hydrogen Igniter 41, 2		11500		
VLS-EH-42	Hydrogen Igniter 42, 1		11500		
VLS-EH-43	Hydrogen Igniter 43, 1		11500		
VLS-EH-44	Hydrogen Igniter 44, 1		11500		
VLS-EH-45	Hydrogen Igniter 45, 2		11500		
VLS-EH-46	Hydrogen Igniter 46, 2		11500		
VLS-EH-47	Hydrogen Igniter 47, 1		11500		
VLS-EH-48	Hydrogen Igniter 48, 2		11500		
VLS-EH-49	Hydrogen Igniter 49, 1		11503		
VLS-EH-50	Hydrogen Igniter 50, 2		11503		
VLS-EH-51	Hydrogen Igniter 51, 1		11500		
VLS-EH-52	Hydrogen Igniter 52, 2		11500		
VLS-EH-53	Hydrogen Igniter 53, 2		11500		
VLS-EH-54	Hydrogen Igniter 54, 1		11500		

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Table 19D-9 (Sheet 7 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
VLS-EH-55	Hydrogen Igniter 55, 1		11504		
VLS-EH-56	Hydrogen Igniter 56, 2		11504		
VLS-EH-57	Hydrogen Igniter 57, 2		11504		
VLS-EH-58	Hydrogen Igniter 58, 1		11504		
VLS-EH-59	Hydrogen Igniter 59, 2		11503		
VLS-EH-60	Hydrogen Igniter 60, 1		11503		
VLS-EH-61	Hydrogen Igniter 61, 1		11500		
VLS-EH-62	Hydrogen Igniter 62, 2		11500		
VLS-EH-63	Hydrogen Igniter 63, 1		11500		
VLS-EH-64	Hydrogen Igniter 64, 2		11500		
VLS-EH-65	Hydrogen Igniter 65, 1		11500		
VLS-EH-66	Hydrogen Igniter 66, 2		11500		
19D.8.2.11	**Electrical Containment Penetration Assemblies**				
	Electrical Containment Penetration Assemblies	**T2,T3-A7**		**All**	**Maintain containment boundary; Conduct electricity for sensors and igniters**
ECS-EY-P01X	Electrical Penetration P01		11400		
ECS-EY-P02X	Electrical Penetration P02		11400		
DAS-EY-P03Z	Electrical Penetration P03		11400		
ECS-EY-P06Y	Electrical Penetration P06		11400		
ECS-EY-P07X	Electrical Penetration P07		11400		
ECS-EY-P09W	Electrical Penetration P09		11400		
ECS-EY-P10W	Electrical Penetration P10		11400		
IDSA-EY-P11Z	1E Electrical Penetration P11		11400		
IDSA-EY-P12Y	1E Electrical Penetration P12		11400		
IDSA-EY-P13Y	1E Electrical Penetration P13		11400		
IDSD-EY-P14Z	1E Electrical Penetration P14		11306		
IDSD-EY-P15Y	1E Electrical Penetration P15		11306		

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Table 19D-9 (Sheet 8 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
IDSD-EY-P16Y	1E Electrical Penetration P16		11306		
ECS-EY-P17X	Electrical Penetration P17		11300		
ECS-EY-P18X	Electrical Penetration P18		11300		
ECS-EY-P19Z	Electrical Penetration P19		11300		
ECS-EY-P20Z	Electrical Penetration P20		11300		
EDS-EY-P21Z	Electrical Penetration P21		11300		
ECS-EY-P22X	Electrical Penetration P22		11300		
ECS-EY-P23X	Electrical Penetration P23		11300		
ECS-EY-P24	Spare Electrical Penetration		11300		
ECS-EY-P25W	Electrical Penetration P25		11300		
ECS-EY-P26W	Electrical Penetration P26		11300		
IDSC-EY-P27Z	1E Electrical Penetration P27		11300		
IDSC-EY-P28Y	1E Electrical Penetration P28		11300		
IDSC-EY-P29Y	1E Electrical Penetration P29		11300		
IDSB-EY-P30Z	1E Electrical Penetration P30		11306		
IDSB-EY-P31Y	1E Electrical Penetration P31		11306		
IDSB-EY-P32Y	1E Electrical Penetration P32		11306		
19D.8.2.12	**Cables**				
	Cables	**T2,T3-A10**	**Note 1**	**All**	**Conduct electricity for valves, sensors and igniters**
19D.8.2.13	**Float Level Sensors**				
	PXS Containment Water Level	**T2,T3-A2,A6,A9,A10**		**2**	**Sense level**
PXS-JE-LE050	Containment Floodup Level		11104		
PXS-JE-LE052	Containment Floodup Level		11104		
PXS-JE-LE051	Containment Floodup Level		11104		
19D.8.2.14	**Containment Hatches (Gasket Material Only)**				
	Containment Hatches (Gasket Material Only)	**T2,T3-A7**		**All**	**Maintain containment boundary**
CNS-MY-Y01	Equipment Hatch		11500		
CNS-MY-Y02	Maintenance Hatch		11300		
CNS-MY-Y03	Personnel Hatch – Upper		11500		
CNS-MY-Y04	Personnel Hatch – Lower		11300		

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Table 19D-9 (Sheet 9 of 9)
Equipment Located Inside Containment (Time Frames T2 and T3)

Section and Equip Tag No.	Description	Time Frames T2,T3 and Actions (Notes 6, 7)	Room Number	Desired Channels/ Divisions (Note 5)	Function
19D.8.2.15	**Containment Vessel**				
	Containment Vessel	**T2,T3-A7**		**All**	**Maintain containment boundary**
CNS-MV-01	Containment Shell				

Notes:
1. Cables are located in most rooms inside containment.
2. A channel consists of two thermocouples within a single division. Each quadrant contains two divisions. The minimum requirement is two OPERABLE thermocouples in each of the two divisions for each quadrant. All Core Exit Thermocouples (CETs) are located inside the Reactor Pressure Vessel which is in Room 11105 (reactor cavity).
3. There are two groups of igniters. Each containment compartment and hydrogen pathway is covered by at least one igniter in each group. The minimum requirement is one igniter is OPERABLE in each compartment and hydrogen pathway.
4. Equipment is part of the fire protection system, on vertical pipe inside containment, feeding water to top rings, Room 11500.
5. Time Frames T2 and T3 are beyond design basis, and therefore, only one channel of each instrumentation parameter is required to be OPERABLE. However, it is preferred to have more than one channel of each parameter available to resolve information ambiguity.
6. See Appendix 19D, Section 19D.4 for definitions of Equipment Survivability Time Frames T2 and T3.
7. See Appendix 19D, Section 19D.6 for Actions associated with Time Frames T2 and T3.

Time Frame T2

- A1 = Subsection 19D.6.2.1
- A2 = Subsection 19D.6.2.2
- A3 = Subsection 19D.6.2.3
- A4 = Subsection 19D.6.2.4
- A5 = Subsection 19D.6.2.5
- A6 = Subsection 19D.6.2.6
- A7 = Subsection 19D.6.2.7
- A8 = Subsection 19D.6.2.8
- A9 = Subsection 19D.6.2.9
- A10 = Subsection 19D.6.2.10

Time Frame T3

- A1 = Subsection 19D.6.3.1
- A2 = Subsection 19D.6.3.2
- A3 = Subsection 19D.6.3.3
- A4 = Subsection 19D.6.3.4
- A5 = Subsection 19D.6.3.5
- A6 = Subsection 19D.6.3.6
- A7 = Subsection 19D.6.3.7
- A8 = Subsection 19D.6.3.8
- A9 = Subsection 19D.6.3.9
- A10 = Subsection 19D.6.3.10

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Appendix 19E Shutdown Evaluation

19E.1 Introduction

Westinghouse has considered shutdown operations in the design of the A1000 nuclear power plant. The AP1000 defense-in-depth design philosophy to provide normally operating active systems and passive safety-related systems gives the AP1000 a greater degree of safety during shutdown operations as well as normal power operation when compared to currently operating plants. This appendix presents and evaluates the AP1000 design features in the context of the specific shutdown issues identified by the Nuclear Regulatory Commission.

19E.1.1 Purpose

This appendix presents AP1000 design features that address the issues of shutdown risk and shutdown safety. This appendix further evaluates these design features with respect to their ability to reduce and or mitigate the consequences of events that can occur during shutdown.

19E.1.2 Scope

The scope of this appendix includes discussions of the following:

- Systems designed to operate during shutdown
- Shutdown operations – including maintenance insights, risk management, and Emergency Response Guidelines (ERGs) (Reference 1)
- Safety analyses and evaluations for shutdown operations
- Technical Specifications
- Shutdown risk evaluations – including shutdown PRA results and insights and fire/flood risk
- Compliance with the guidance in NUREG-1449 (Reference 2)

19E.1.3 Background

The Diablo Canyon event of April 10, 1987, and the loss of ac power at the Vogtle plant on March 20, 1990, led the NRC staff to issue NUREG-1449, which provides an evaluation of the shutdown risk issue. During the AP600 Design Certification review, the NRC requested that Westinghouse perform a systematic assessment of the shutdown risk issue to address areas identified in NUREG-1449 as applicable to the AP600 design. The AP1000 design is based extensively on the AP600, and the systems, structures and components that are important in maintaining a low shutdown risk for AP600 are generally the same design and/or have the same design basis with respect to their role in reducing shutdown risk. Therefore, the conclusions from the assessment of the shutdown risk for the AP600 are applicable to the AP1000. This appendix summarizes the assessment of the shutdown risk issue for AP1000.

19E.2 Major Systems Designed to Operate During Shutdown

Westinghouse has considered shutdown modes, shutdown alignments, and industry issues related to shutdown in the design of the AP1000 safety-related and nonsafety-related systems designed to operate or be available during shutdown. This section provides descriptions of the important systems designed to operate during shutdown and includes specific design features that have been

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incorporated for shutdown operations with a discussion of their operating modes or alignment during shutdown.

In this appendix, references are made to the various AP1000 operating modes. The AP1000 operating modes have been defined in Technical Specifications, Table 1.1-1. The mode definitions for the AP1000 are similar to that of current Westinghouse pressurized water reactors (PWRs), with the difference being the definition of Mode 4, safe shutdown.

In the AP1000, Mode 4 has been redefined as safe shutdown and corresponds to the range of RCS temperature between 420°F and 200°F. The upper temperature limit corresponds to the RCS temperature that can be achieved by the passive safety-related systems 36 hours after shutdown. The ability of the passive safety-related systems to achieve Mode 4 within 36 hours is shown in Subsection 4.10.2 of this appendix.

19E.2.1 Reactor Coolant System

19E.2.1.1 System Description

The reactor coolant system (RCS) is described in Chapter 5.

19E.2.1.2 Design Features to Address Shutdown Safety

The AP1000 has incorporated design features that address issues related to shutdown operations. This subsection provides a discussion of the RCS design features that are incorporated to address shutdown operations or that are important to minimizing the risk to plant safety during shutdown.

19E.2.1.2.1 Loop Piping Offset

The RCS hot legs and cold legs are vertically offset. This permits draining of the steam generators for nozzle dam insertion with the hot leg level much higher than traditional designs. The RCS must be drained to a level sufficient to provide a vent path from the pressurizer to the steam generators. This loop piping offset also allows an RCP to be replaced without removing the full core.

19E.2.1.2.2 RCS Instrumentation

Instrumentation is provided to monitor the RCS process parameters as required by the PLS and PMS as discussed in Chapter 7. This subsection describes RCS instrumentation designed to accommodate shutdown operations.

RCS Hot Leg Level

There are two safety-related RCS hot leg level channels, one located in each hot leg. These level indicators are provided primarily to monitor the RCS water level during mid-loop operation following shutdown operations. These are totally independent of each other. One level tap is at the bottom of the hot leg, and the other tap is on the top of the hot leg close to the steam generator. The steam generator tap is located at the high point of the tubing run. The level tap for the instrument in the hot leg with the normal residual heat removal system (RNS) step-nozzle suction line connection is between the reactor vessel and the step-nozzle. Figure 19E.2-1 shows a simplified sketch of the RCS level instruments.

These channels provide signals for the following protection functions:

- Isolation of letdown on Low-4 level on a one-out-of-two basis.

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- Actuation of fourth-stage ADS valves on Low-4 (empty) hot leg level on a two-out-of-two basis. Actuation of fourth-stage ADS causes actuation of IRWST injection.

These functions protect the plant during shutdown operations. Letdown isolation assists the operators when draining the RCS to a mid-loop level. If the operators fail to isolate letdown, these channels send a signal to close the letdown valves and stop the draining process.

In the event of a loss of the RNS during shutdown, coolant inventory could be boiled away. When the hot leg water level indicates that the loops are empty, IRWST injection and fourth-stage ADS are actuated 25 minutes after receipt of the empty hot leg level signal.

These channels are also used to generate the alarms on low hot leg water level. The alarm setpoints are selected to give the operator sufficient time to take the manual actions necessary to prevent the automatic actuation described previously. Indication of these channels is retrievable in the main control room. This variable is used by the operator to monitor the status of RCS inventory following an accident and is, therefore, classified as a post-accident monitoring system (PAMS) variable as discussed in Section 7.5.

The accuracy and response time of the hot leg level instruments are consistent with the standard engineered safety features (ESF) actuation discussed in Section 7.3. Concerns related to potential problems of noncondensible gases in the hot leg level instrument lines that have been raised in NRC Information Notice 92-54, Level Instrumentation Inaccuracies Caused by Rapid Depressurization (Reference 3), have been addressed in the layout of the instrument lines. In addition, as the hot leg level instruments are provided primarily for shutdown operations, off-gassing due to sudden depressurization of the RCS in shutdown modes is not a concern.

In the AP1000, draining of the RCS to mid-loop conditions is achieved in a controlled manner as discussed in Subsection 19E.2.1.2.4. Due to the low RCS drain rate, and the RCS step-nozzle as discussed in Subsection 19E.2.1.2.3, the amount of air-entrainment, and therefore RCS level perturbation during mid-loop, is negligible. Draining of the RCS is conducted in a quasi-steady-state, and the reliability of an accurate level reading is high.

Pressurizer Level

A fifth nonsafety-related independent pressurizer level transmitter, calibrated for low temperature conditions, provides water level indication during startup, shutdown, and refueling operations in the main control room and in the remote shutdown workstation. The upper level tap is connected to an ADS valve inlet header above the top of the pressurizer. The lower level tap is connected to the bottom of the hot leg. This provides level indication for the entire pressurizer and a continuous reading as the level in the pressurizer decreases to mid-loop levels during shutdown operations.

RCS Hot Leg Wide-Range Temperatures

The RCS contains two safety-related thermowell-mounted hot leg wide-range temperature detectors, one in each hot leg. The orientation of the resistance temperature detectors enables measurement of the reactor coolant fluid in the hot leg when in normal inventory conditions. Their range is selected to accommodate the low RCS temperatures that can be attained during shutdown. Two nonsafety-related RNS temperature instruments upstream of the RNS heat exchangers indicate RCS hot leg temperature when in reduced inventory conditions. In addition, at least two incore thermocouple channels are available to measure the core exit temperature during mid-loop RNS operation. These two thermocouple channels are associated with separate electrical divisions.

Pressurizer Surge Line Temperatures

There are three nonsafety-related temperature detectors located on the RCS pressurizer surge line. These instruments monitor the pressurizer surge line fluid temperature during plant normal

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operations to detect thermal stratification in the surge line. Two of the temperature detectors are on a moderately sloped run approximately midway between the RCS hot leg and the pressurizer. One detector is on the bottom of the pipe and the other detector on the top. The third detector is located on the pressurizer surge line close to the pressurizer nozzle. This detector is used to monitor cold water insurges to the pressurizer during transient operations.

The temperature is monitored at the three locations using strap-on resistance temperature detectors. Temperature indication is provided in the main control room. One low-temperature alarm is provided to alert the operator of thermal stratification in the surge line. This alarm is associated with the detector on the bottom of the pipe.

During shutdown operations, this temperature instrumentation will be monitored to detect possible surge line stratification. If stratification is detected, the operators can increase spray flow to increase the outsurge from the pressurizer and reduce stratification in the surge line.

19E.2.1.2.3 Step-nozzle Connection

The AP1000 RNS uses a step-nozzle connection to the RCS hot leg. The step-nozzle connection has two effects on mid-loop operation. One effect is to lower the RCS hot leg level at which a vortex occurs in the residual heat removal pump suction line due to the lower fluid velocity in the hot leg nozzle. This increases the margin from the nominal mid-loop level to the level where air entrainment into the pump suction begins.

Another effect of the step-nozzle is that, if a vortex should occur, the maximum air entrainment into the pump suction as shown experimentally will be no greater than 5 percent (Reference 4). The RNS pumps can operate with 5% air-entrainment. As discussed in NUREG-0897 (Reference 5), low levels of air ingestion can be tolerated, and a pump inlet void fraction of 5% has been shown experimentally to reduce the pump head less than 15%. At this level of degradation, the RNS pumps would maintain decay heat removal. The step-nozzle thereby precludes air binding of the pump and will allow for RNS pump operation with low water levels in the hot leg.

19E.2.1.2.4 Improved RCS Draindown Method

During the cooldown operations, the RCS water level is drained to a mid-loop level to permit steam generator draining and maintenance activities. The AP1000 has improved the reliability of draindown operations by incorporating a dedicated drain path to be used to reduce the water level in the RCS controlled in the main control room. In current plants, various drain paths can be used either locally or remotely from the control room. These drain paths include the safety-related residual heat removal system, loop drain valves, and letdown. The result is that draining of the RCS can be difficult to control, and perturbations in water level can occur due to inadvertent system manipulations of which the operators are not always aware.

The AP1000 RCS drain path is via the CVS letdown line from the RNS cross-connect provided to maintain full RCS purification flow during shutdown. The letdown line flow control valve controls the letdown rate, which controls the RCS draindown rate. At the appropriate time during the cooldown, the operator initiates the draindown by manually positioning the CVS letdown control valve. At this time, the makeup pumps are turned off and the letdown flow control valve controls the drain rate to the liquid radwaste system at the initial maximum rate of approximately 100 gpm. The rate is reduced as the level in the RCS approaches the top of the hot leg. The letdown rate is manually controlled based upon the difference in flow instruments readings in the CVS letdown line and injection line. The letdown flow control valve as well as the letdown line containment isolation valve receives a signal to automatically close once the appropriate level is attained. Alarms actuate in the control room if the RCS level falls below the automatic letdown valve closure setpoint so that the operator is alerted to manually isolate the letdown line. Furthermore, an automatic isolation of the letdown line is actuated

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on Low-2 hot leg level. This draindown method provides a reliable means of attaining mid-loop conditions.

19E.2.1.2.5 ADS Valves

The ADS first-, second-, and third-stage valves, connected to the top of the pressurizer, are open whenever the core makeup tanks (CMTs) are blocked during shutdown conditions while the reactor vessel upper internals are in place. This provides a vent path that, when combined with the required ADS stage 4 actuation, provides adequate steam venting area to reduce RCS pressure during shutdown conditions if decay heat removal is lost. Actuation of ADS stage 4, either automatically or manually (as described in Subsection 19E.4.8.3) allows the in-containment water storage tank to provide injection flow following a loss of decay heat removal. In addition, ADS fourth-stage valves are required to be available during reduced inventory operations to preclude surge line flooding following a loss of the RNS. Because of the potential for counter-current flow limitation in the surge line, it is essential to establish and maintain venting capability with ADS Stage 4 for gravity injection and containment recirculation following an extended loss of RNS when the RCS is open during shutdown operations.

19E.2.1.2.6 Steam Generator Channel Head

The AP1000 steam generator is a vertical-shell U-tube evaporator with integral moisture separating equipment. The generator is discussed in Subsection 5.4.2.

On the primary side, the reactor coolant flow enters the primary chamber via the hot leg nozzle. The lower portion of the primary chamber is hemispherical and merges into a cylindrical portion, which mates to the tubesheet. This arrangement provides enhanced access to all tubes, including those at the periphery of the bundle, with robotics equipment. This feature enhances the ability to inspect, replace, and repair portions of the AP1000 unit compared to the more hemispherical primary chamber of earlier designs. The channel head is divided into inlet and outlet chambers by a vertical divider plate extending from the apex of the head to the tubesheet.

The reactor coolant enters the inverted U-tubes, transferring heat to the secondary side during its traverse, and returns to the cold leg side of the primary chamber. The flow exits the steam generator via two cold leg nozzles to which the reactor coolant pumps are directly attached.

The AP1000 steam generator channel head has provisions to drain the head. For minimizing deposits of radioactive corrosion products on the channel head surfaces and for enhancing the decontamination of these surfaces, the channel head cladding is machined or electropolished for a smooth surface.

The steam generator is equipped with permanently mounted nozzle dam brackets, which are designed to support nozzle dams during refueling operations. The design pressure of the nozzle dam bracket and nozzle dam is selected to withstand the RCS pressures that can occur during a loss of shutdown cooling. The nozzle dam design pressure is at least 50 psia.

The AP1000 nozzle dams can be installed with the hot leg water level at the nominal water level for mid-loop operations. The nozzle dams can be inserted via the steam generator manway. The ADS valves connected to the pressurizer are open during all reduced inventory operations including nozzle dam installation, and when combined with the required ADS stage 4 actuation, provide adequate steam venting area to reduce RCS pressure following a loss of decay heat removal when the nozzle dams are installed.

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19E.2.2 Steam Generator and Feedwater Systems

19E.2.2.1 System Description

This section discusses the AP1000 steam generator system (SGS) and the main and startup feedwater system (FWS) designs as they relate to shutdown operations. These systems are discussed in Chapter 10.

19E.2.2.2 Design Features to Address Shutdown Safety

19E.2.2.2.1 Feedwater Control

The AP1000 provides improvements in feedwater control that minimizes the probability of loss of feedwater transients during low power and shutdown modes. The main feedwater pumps are capable of providing feedwater during all modes of operation, including plant startup and standby conditions. In addition, the startup feedwater pumps are automatically started in the event that the main feedwater pumps are unable to continue to operate. The startup feedwater pumps are also automatically loaded on the diesels for operation following a loss of normal ac power and offsite power, during operating modes when the steam generators can be used for decay heat removal.

19E.2.2.2.2 Safety-Related Actuation in Shutdown Modes

The AP1000 has safety-related actuations associated with the SGS that are operable during shutdown modes. These include the PRHR HX actuation on Low-2 steam generator wide range water level during shutdown modes, and this is discussed in Subsection 19E.2.3 of this appendix. Also included is the isolation of the main steam line on a high (large) negative rate of change in steam pressure. This safety-related signal is provided to address a steam line break that could occur in Mode 3. If actuated, this signal causes the MSIVs to close to terminate the blowdown of the SGS following a steam line break. This signal is placed into service below the setpoint that disables the Low-2 steam line pressure signal (P11) that actuates steam line isolation as discussed in Section 7.3. When the operator manually blocks the Low-2 steam line pressure signal, the steam line high pressure-negative rate signal is automatically enabled.

This signal is operable during Mode 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In Modes 4, 5, and 6, this function is not needed for accident detection and mitigation. Subsection 19E.4.2.3 discusses steam line break events that could occur in shutdown modes. Operability of this actuation logic is discussed in the AP1000 Technical Specifications.

19E.2.2.2.3 Steam Generator Cooling in Shutdown Modes

The secondary side of the steam generators can be recirculated during shutdown by circulating their contents through the blowdown system to promote heat transfer from the steam generators to the primary system. This feature reduces the challenges to low-temperature overpressure events. During RCS water-solid operation, heat input from the steam generators is capable of challenging the low-temperature relief valves. The Technical Specifications prevent the operators from starting an RCP with the steam generator secondary side temperature more than 50°F higher than the primary side, with the pressurizer water-solid. With the RCS water-solid, the heat input that could occur would cause the system to be pressurized to the setpoints of the low-temperature overpressure relief valves in the RNS.

When the RCPs are operating, the secondary side of the steam generator is cooled by steaming to the MSS. Once the RNS is aligned, and steaming to the MSS is decreased, the secondary side of the steam generators is cooled by operation of the RNS. However, once the RCPs are tripped, water

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does not circulate through the primary side of the tubes and the secondary side of the steam generators remains at elevated temperature. With the ability to circulate the secondary side via the blowdown system, the AP1000 reduces the probability that an RCP would be started with the secondary side of the generator at elevated temperature.

The AP1000 has also incorporated steam generator fluid thermocouples to monitor the temperature of the fluid in the secondary side of the steam generator. This improves the ability of the operators to monitor this temperature to prevent them from inadvertently starting an RCP with the secondary side at elevated temperatures.

19E.2.3 Passive Core Cooling System

19E.2.3.1 System Description

The passive core cooling system (PXS) is described in Section 6.3.

19E.2.3.2 Design Features to Address Shutdown Safety

A significant improvement in shutdown safety for the AP1000 is the availability of a dedicated safety-related system that can be automatically or manually actuated in response to an accident that can occur during shutdown. In current plants, the safety-related systems that mitigate the consequences of an accident are also the operating systems that are used for decay heat removal. In the AP1000, nonsafety-related active systems provide the first level of defense, while the passive safety-related systems are available during shutdown modes to mitigate the consequences of an accident. This design approach results in a significant improvement in the AP1000 shutdown risk.

19E.2.3.2.1 Core Makeup Tanks

The CMTs provide RCS makeup. During shutdown, the CMTs are available in Modes 3, 4, and 5, until the RCS pressure boundary is open and the pressurizer water level is reduced. During power operation, the CMTs are automatically actuated on various signals including a safeguards actuation signal (low RCS pressure, low RCS temperature, Low-2 steam line pressure, and High-2 containment pressure) and on Low-2 pressurizer water level. See Chapter 7 for a description of the AP1000 PMS actuation logic. In shutdown modes, portions of the safeguards actuation signal are disabled to allow the RCS to be cooled and depressurized for shutdown. For instance, the low RCS pressure and temperature, and Low-2 steam line pressure signals are blocked in Mode 3 prior to cooling and depressurizing the RCS. Therefore, during shutdown Modes 3, 4, and 5, the primary signal that actuates the CMTs due to a loss of inventory is the pressurizer level signal. In Mode 5, with the RCS open (in preparation for reduced inventory operations), the Low-2 pressurizer level signal is blocked prior to draining the pressurizer. Therefore, in Mode 5 with the RCS open, the CMTs are not required to be available and the RCS makeup function is provided by the IRWST.

The CMTs also provide an emergency boration function for accidents such as steam line breaks. However, the signals that provide the primary protection for this function (Low-2 steam line pressure, low RCS pressure, and low RCS temperature) are blocked in Mode 3 as discussed above. Prior to blocking these signals in Mode 3, the Technical Specifications require that the RCS be sufficiently borated. For these events, the pressurizer level signal provides automatic actuation of the CMTs for a steam line break that might occur due to the RCS shrinkage that would occur.

19E.2.3.2.2 Accumulators

The PXS accumulators provide safety injection following a LOCA. In Mode 3, the accumulators must be isolated to prevent their operation when the RCS pressure is reduced to below their set pressure.

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The accumulator isolation valves are closed when the RCS pressure is reduced to 1000 psig to block their injection when the RCS pressure is reduced to below the normal accumulator pressure.

19E.2.3.2.3 In-containment Refueling Water Storage Tank

The IRWST provides long-term RCS makeup. During shutdown, the IRWST is available until Mode 6, when the reactor vessel upper internals are removed and the refueling cavity flooded. At that time, the IRWST is not required, due to the large heat capacity of the water in the refueling cavity.

The IRWST injection paths are actuated on a Low-6 CMT water level. This signal is available in shutdown Modes 3, 4, and 5, with the RCS intact. When the RCS is open to transition to reduced inventory operations, the CMT actuation logic on Low-2 pressurizer level is removed, and the CMTs can be taken out of service. For these modes, automatic actuation of the IRWST can be initiated (on a two-out-of-two basis) on Low-4 hot leg level.

19E.2.3.2.4 Passive Residual Heat Removal Heat Exchanger

The PRHR HX provides decay heat removal during power operation and is required to be available in shutdown Modes 3 and 4, and in Mode 5 until the RCS is open except during vacuum fill. In these modes, the PRHR HX provides a passive decay heat removal path. It is automatically actuated on a CMT actuation signal, which would eventually be generated on a loss of shutdown decay heat removal, as shown in the analysis provided in Section 19E.4 of this appendix. In modes with the RCS open (portions of Mode 5 and Mode 6), decay heat removal is provided by “feeding” water from the IRWST and “bleeding” steam from the ADS.

19E.2.3.2.5 Reduced Challenges to Low-Temperature Overpressure Events

Another design feature of the PXS that reduces challenges to shutdown safety is the elimination of high-head safety injection pumps in causing low temperature overpressure events. In current plants, during water solid operations that may be necessary to perform shutdown maintenance, the high-head safety injection pumps are a major source of cold overpressure events. To address this, plants are required to lock out safety injection pumps to prevent them from inadvertently causing a cold overpressure event. This eliminates a potential source of safety injection for a loss of inventory event that could occur at shutdown. With the AP1000 PXS, the CMTs are not pressurized above RCS pressure and are, therefore, not capable of causing a cold overpressure event. Therefore, they are not isolated until the pressurizer is drained for mid-loop. Low-temperature overpressure events are discussed in Subsection 19E.4.10.1.

19E.2.3.2.6 Discussion of Safe Shutdown for AP1000

The functional requirements for the PXS specify that the plant be brought to a safe, stable condition using the PRHR HX for events not involving a loss of coolant. As stated in Subsection 6.3.1.1.1, the PRHR HX, in conjunction with the passive containment cooling system (PCS), provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the PXS, in conjunction with the PCS, and the ADS, has the capability to establish long-term safe shutdown conditions in the RCS as identified in Subsection 7.4.1.1.

The CMTs automatically provide injection to the RCS after they are actuated on low reactor coolant temperature or low pressurizer pressure or level. The PXS can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak, a longer time is available.

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In scenarios when ac power sources are unavailable for approximately 22 hours, the ADS automatically actuates. However, after the initial plant cooldown following a non-LOCA event, operators assess plant conditions and have the option to perform recovery actions to further cool and depressurize the RCS in a closed-loop mode of operation, i.e., without actuation of the ADS. After verifying the RCS is in an acceptable, stable condition, such that automatic depressurization is not needed, the operators may take action to extend PRHR HX operation by de-energizing the loads on the Class 1E dc batteries powering the protection and safety monitoring system actuation cabinets. After operators have taken action to extend its operation, the PRHR HX, in conjunction with the PCS, has the capability to maintain safe, stable conditions. The ADS remains available to maintain safe shutdown conditions at a later time.

In most sequences, the operators would return the plant to normal system operations and terminate passive system operation within several hours in accordance with the plant emergency operating procedures. For LOCAs and other postulated events, when the core makeup tank level reaches the automatic depressurization actuation setpoint, and other postulated events where the PRHR HX operation is not extended or exhausted, the ADS may be initiated. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the RCS is nearly depressurized. For these conditions, the RCS depressurizes to saturated conditions at about 250°F within 24 hours. The PXS can maintain this safe shutdown condition as identified in Subsection 7.4.1.1.

The primary function of the PXS during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Analysis is provided in Subsection 19E.4.10.2 of this appendix that verifies the ability of the AP1000 passive safety systems to meet the safe shutdown requirements.

19E.2.3.2.7 Containment Recirculation Screens

The PXS containment recirculation screens may have to function in the longer-term during a shutdown accident that results in ADS operation. Effective screen design, plant layout, and other factors prevent clogging of these screens by debris during such accident operations.

- Two very large interconnected screens are provided.
- A significant delay is provided between the accident/ADS stage opening and the initiation of recirculation (at least 2 hours).
- Deep flood up levels are provided post ADS operation (31 ft of water above the lowest level in containment and 25.5 ft above floors around screens).
- Bottom of screens are located well above the lowest containment level (13.5 feet) as well as the floors around them (2 feet).
- Top of screens are located well below the containment floodup level (~10 ft from top screens to minimum flood level).
- Screens have a protective plate located no more than 1 foot, 3 inches above the top of the face of the screens, which extends at least 8 feet, 3 inches perpendicular to the front and at least 7 feet to the side of the face of the screens.

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- Screens have conservative flow areas to account for plugging. Operation of the nonsafety-related normal residual heat removal pumps with suction from the IRWST and the containment recirculation lines is considered in sizing screens. Note that adequate PXS performance can be supported by one screen with more than 90 percent of its surface area completely blocked.

- During recirculation operation, the velocity approaching the screens is very low, which limits the transport of debris.

- Each screen has a fine screen.

- Technical Specifications require the screens to be inspected during each refueling outage.

- As discussed in Subsection 6.3.8.1, a cleanliness program to limit the amount of foreign materials that might be left in the containment following refueling and maintenance outages and become debris during an accident.

19E.2.3.3 Shutdown Operations

Operation of the PXS during operating modes and during accident events including shutdown events is discussed in Subsection 6.3.3. The following is a discussion of a loss of shutdown cooling during reduced inventory operations which can be a limiting shutdown event.

19E.2.3.3.1 Operation During Loss of Normal Residual Heat Removal Cooling During Mid-loop Events

During RCS maintenance, the most limiting shutdown condition anticipated is with the reactor coolant level reduced to the hot leg (mid-loop) level and the RCS pressure boundary opened. It is normal practice to open the steam generator channel head manway covers to install the hot leg and cold leg nozzle dams during a refueling outage. In this situation, the RNS is used to cool the RCS. The AP1000 incorporates features to reduce the probability of losing RNS. However, because the RNS is nonsafety-related, its failure has been considered.

In this situation, core cooling is provided by the safety-related PXS, using gravity injection from the IRWST, while venting through the ADS valves (and possibly through other openings in the RCS). Note that with the RCS depressurized and the pressure boundary opened, the PRHR HX is unable to remove the decay heat because the RCS cannot heat sufficiently above the IRWST temperature.

During plant shutdown, at 1000 psig, the accumulators are isolated to prevent inadvertent injection. Prior to draining the RCS inventory below the no-load pressurizer level, the CMTs are isolated by closing the inlet MOVs to preclude inadvertent draining into the RCS while preparing for mid-loop operation. Although these tanks are isolated from the RCS, the valves can be remotely opened by the operators to provide additional makeup water injection.

Prior to initiating the draindown of RCS to mid-loop level, the automatic depressurization first-, second-, and third-stage valves are opened. This alignment, when combined with the required ADS stage 4 actuation, provides adequate steam venting area to reduce RCS pressure in the event of a loss of nonsafety-related decay heat removal during mid-loop operation. The ADS first- to third-stage valves are required to be opened before blocking the CMTs. They are required to remain open until either the RCS level is increased and the RCS is closed, or until the upper core internals are removed and the refueling cavity flooded. Note that the upper internals can restrict the vent flow path and prevent water in the refueling cavity from draining into the RCS unless ADS valves are open.

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The IRWST injection squib valves and fourth stage ADS valves are automatically opened if the RCS hot leg level indication decreases below a Low-4 setpoint or manually as described in Subsection 19E.4.8.3. A time delay is provided to provide time for the operators to restore nonsafety-related decay heat removal prior to actuating the PXS. The time delay with an alarm in the containment serves to protect maintenance personnel. Once the IRWST injection valves and fourth stage ADS valves open, the IRWST provides gravity-driven injection to cool the core. Containment recirculation flow would be automatically initiated when the IRWST level dropped to the Low-3 level to provide long-term core cooling.

Subsection 19E.4.8.3 provides the assessment of the loss of the RNS during mid-loop operations. Table 19E.2-1 provides the results of calculations performed to demonstrate the amount of time between a loss of RNS that could occur at mid-loop until core uncovery. This calculation is performed with the RCS water level at the nominal mid-loop water level and is performed with conservative, design basis assumptions for decay heat. As described previously and shown in Table 19E.2-1, the operators have a significant amount of time to actuate gravity injection before core uncovery. In addition, the PMS, on a two-out-of-two basis, provides a signal to actuate the IRWST when the hot legs empty.

This arrangement provides automatic core cooling protection, in mid-loop operation, while also providing protection (an evacuation alarm and sufficient time to evacuate) for maintenance personnel in containment during mid-loop operation.

Containment closure capability is required to be maintained during mid-loop operation, as discussed in Subsection 19E.2.6.2 of this appendix. With the containment closed, containment recirculation can continue indefinitely with decay heat generating steam condensed on the containment vessel and drained back into the IRWST and/or the containment recirculation.

19E.2.4 Normal Residual Heat Removal System

19E.2.4.1 System Description

The normal residual heat removal system (RNS) is discussed in Subsection 5.4.7. Operability of the RNS is tested, via connections to the IRWST, before its alignment to the RCS hot leg for shutdown cooling.

19E.2.4.2 Design Features to Address Shutdown Safety

The AP1000 has incorporated various design features to improve shutdown safety. The RNS features that have been incorporated to address shutdown safety are described in this subsection.

19E.2.4.2.1 RNS Pump Elevation and NPSH Characteristics

The AP1000 RNS pumps are located at the lowest elevation in the auxiliary building. This location provides the RNS pumps with a large available NPSH during all modes of operation including RCS mid-loop and reduced inventory operations. The large NPSH provides the pumps with the capability to operate during most mid-loop conditions without throttling the RNS flow. If the RCS is at mid-loop level and saturated conditions, some throttling of a flow control valve is necessary to maintain adequate net positive suction head for the RNS pumps. The RNS pumps can be restarted and operated with RCS conditions that might occur following a temporary loss of RNS cooling.

The plant piping configuration, piping elevations and routing, and the pump characteristics allow the RNS pumps to be started and operated at their full design flow rates in most conditions without the need to reduce RNS pump flow to meet pump NPSH requirements. This reduces the potential failure

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mechanism that exists in current PWRs, where failure of an air-operated control valve can result in pump runout and cavitation during mid-loop operations.

19E.2.4.2.2 Self-Venting Suction Line

Most of the RNS pump suction line is sloped upward from the RNS pump to the reactor coolant system hot leg. In the level portions of piping, there are no local high points. This eliminates potential problems with refilling the pump suction line if an RNS pump is stopped due to pump cavitation and/or excessive air entrainment. With the self-venting suction line, the line will refill and the pumps can be immediately (no additional operator action required) restarted once an adequate level in the hot leg is re-established.

19E.2.4.2.3 IRWST Injection via the RNS Suction Line

During shutdown modes, initiating events such as the loss of the nonsafety-related RNS are postulated. Such events would require IRWST injection as discussed in Subsection 19E.2.3 of this appendix, and as shown in the accident analyses provided in Section 19E.4. For initiating IRWST injection, the operation of PXS squib valves in the IRWST injection line is required. However, the operators can use the RNS pump suction line that connects to the IRWST to provide controlled IRWST injection. This flow path, shown in Figure 19E.2-2, connects the IRWST directly to the RCS via the RNS hot leg suction isolation valves and provides a diverse method for IRWST injection. In addition, it would be the preferred method of providing IRWST injection because the flow would be controllable by the operation of the IRWST suction line isolation valve. The RNS isolation valve is equipped with a throttle capability to provide the operators with the capability to control the injection flow via this path. The operator would monitor the RCS hot leg level while controlling flow through this valve. This path provides IRWST injection regardless of whether the RNS pumps are operating.

19E.2.4.2.4 Codes and Standards/Seismic Protection

The portions of the RNS located outside containment (that serve no active safety functions) are classified as AP1000 equipment Class C so that the design, manufacture, installation, and inspection of this pressure boundary is in accordance with the following industry codes and standards and regulatory requirements: 10 CFR 50, Appendix B (Reference 6); Regulatory Guide 1.26, quality group C (Reference 7); and ASME Boiler and Pressure Vessel Code, Section III, Class 3 (Reference 8), with restriction as described in Subsection 5.2.1.1. The pressure boundary is classified as seismic Category I.

19E.2.4.2.5 Increased Design Pressure

The portions of the RNS from the RCS to the containment isolation valves outside containment are designed to the operating pressure of the RCS. The portions of the system downstream of the suction line containment isolation valve and upstream of the discharge line containment isolation valve are designed so that its ultimate rupture strength is not less than the operating pressure of the RCS. The design pressure of the RNS is 900 psig, which is 40 percent of operating RCS pressure.

19E.2.4.2.6 Reactor Coolant System Isolation Valve

The RNS contains an isolation valve in the pump suction line from the RCS. This motor-operated containment isolation valve is designed to the RCS pressure. It provides an additional barrier between the RCS and lower pressure portions of the RNS.

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19E.2.4.2.7 Normal Residual Heat Removal System Relief Valve

The inside containment RNS relief valves are connected to the residual heat removal pump suction line. These valves are designed to provide low-temperature, overpressure protection of the RCS as described in Subsection 5.2.2. The valves, connected to the high-pressure portion of the pump suction line, reduce the risk of overpressurizing the low-pressure portions of the system.

19E.2.4.2.8 Features Preventing Inadvertent Opening of Isolation Valves

The RCS isolation valves are interlocked to prevent their opening at RCS pressures above 450 psig. Section 7.6 discusses this interlock. The power to these valves is administratively blocked during normal power operation.

In addition, these valves are interlocked with the RNS/IRWST isolation valves to prevent their opening with the RNS open to the IRWST. This precludes the blowdown of the RCS to the IRWST through the RNS upon system initiation.

19E.2.4.2.9 RCS Pressure Indication and High Alarm

The AP1000 RNS contains an instrumentation channel that indicates pressure in each normal residual heat removal pump suction line. A high-pressure alarm is provided in the main control room to alert the operator to a condition of rising RCS pressure that could eventually exceed the design pressure of the RNS.

19E.2.4.2.10 RNS Break Outside Containment

For a postulated RNS break outside the containment while in Modes 4, 5, and 6, the RCS loses inventory through the postulated break. Manual operator actions are taken to isolate RNS and isolate containment within ten minutes of the break. If the RCS inventory level continues to decrease, a backup automatic isolation signal is provided on an ADS Stage 4 actuation to isolate RNS isolation and containment isolation valves to prevent further loss of inventory.

19E.2.5 Component Cooling and Service Water Systems

Two different means are provided to protect the lower-pressure CCS from overpressure if RNS heat exchanger tube leakage occurs during plant cooldown or shutdown operations.

A relief valve is located on the CCS cooling water line, inside the upstream and downstream manual isolation valves for each RNS heat exchanger. The valve satisfies requirements in Section VIII of the ASME code for overpressure protection of heat transfer equipment. This relief valve provides both thermal overpressure and tube leakage protection in the event that the section of piping containing the RNS heat exchanger is isolated from the remainder of the CCS. The valve discharges to the auxiliary building sump through the WRS.

If RNS heat exchanger tube leakage occurs with the affected heat exchanger not isolated from the CCS, the excess volume added to the CCS by the leak will begin to fill the CCS surge tank. If the CCS surge tank fills before the leak is isolated, fluid is discharged through the tank vent into the turbine building sump to prevent over-pressurization of any portion of the CCS. Leakage into the system will produce a CCS liquid radiation monitor alarm, and an increase in CCS surge tank level that results in a tank high level alarm.

Doses from the RNS heat exchanger tube rupture event would be below those produced by the primary sample line break outside containment with the plant at power.

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19E.2.6 Containment Systems

19E.2.6.1 System Description

The containment systems are described in Section 6.2.

19E.2.6.2 Design Features to Address Shutdown Safety

The AP1000 has addressed the issue of containment closure at shutdown and incorporated the following requirements in the Technical Specifications. In shutdown Modes 3 and 4, containment status is the same as at-power. Specifically, containment integrity is required, the major equipment hatches are closed and sealed, and containment air locks and isolation valves are operable.

In Modes 5 and 6, containment closure capability is required during shutdown operations when there is fuel inside containment. Containment closure is required to maintain, within containment, the cooling water inventory. Due to the large volume of the IRWST and the reduced sensible heat during shutdown, the loss of some of the water inventory can be accepted. Further, accident analyses provided in Section 19E.4 of this appendix show that containment closure capability is not required to meet offsite dose requirements. Therefore, containment does not need to be leak-tight as required for Modes 1 through 4.

In Modes 5 and 6, there is no potential for steam release into the containment immediately following an accident. Pressurization of the containment could occur only after heatup of the IRWST due to PRHR HX operation (Mode 5 with RCS intact), after heatup of the RCS with direct venting to the containment (Mode 5 with reduced RCS inventory or Mode 6 with the refueling cavity not fully flooded), or after heatup of the RCS and refueling cavity (Mode 6 with refueling cavity fully flooded). To limit the magnitude of cooling water inventory losses and because local manual action may be required to achieve containment closure, the containment hatches, air locks, and penetrations must be closed prior to steaming into containment.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. If closed, the equipment hatch is held in place by at least four bolts. If open, each equipment hatch can be closed using a dedicated set of hardware, tools, and equipment. A self-contained power source is provided to drive each hoist while lowering the hatch into position. Large equipment and components may be moved through the hatches as long as they can be removed and the hatch closed prior to steaming into the containment.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during Modes 1, 2, 3, and 4 unit operation. Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment operability is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Temporary equipment connections (for example, power or communications cables) are permitted as long as they can be removed to allow containment closure prior to steaming into the containment.

Containment spare penetrations, which also provide a part of the containment boundary, provide for temporary support services (electrical, I&C, air, and water supplies) during Modes 5 and 6. Each penetration is flanged and normally closed. During periods of plant shutdown, temporary support systems may be routed through the penetrations; temporary equipment connections (for example, power or communications cables) are permitted as long as they can be removed to allow containment closure prior to steaming into the containment.

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The spare penetrations must be closed or, if open, capable of closure prior to reaching boiling conditions within reactor coolant inventory. Temporary containment penetrations that may be employed during shutdown modes must have a design pressure equal to the containment design pressure of 59 psig.

Containment penetrations, including purge system flow paths, that provide direct access from containment atmosphere to outside atmosphere must be isolated or isolatable on at least one side. Isolation may be achieved by an operable automatic isolation valve or by a manual isolation valve, blind flange, or equivalent.

The fuel transfer canal may be opened to provide for the transfer of new and spent fuel into and out of containment during Modes 5 and 6. At times when the canal is opened, it must be isolatable on at least one side by closure of the flange within containment or the gate valve outside containment.

19E.2.7 Chemical and Volume Control System

19E.2.7.1 System Description

The chemical and volume control system (CVS) is described in Subsection 9.3.6.

19E.2.7.2 Design Features to Address Shutdown Safety

The AP1000 CVS is a nonsafety-related system. However, portions of the system are safety-related and perform safety-related functions, such as containment isolation, termination of inadvertent RCS boron dilution, RCS pressure boundary preservation, and isolation of excessive makeup.

Boron dilution events during low power modes can occur for a number of reasons, including malfunctions of the reactor makeup control system. Regardless of the cause, the protection is the same. The CVS is designed to avoid and/or terminate boron dilution events by automatically closing either one of two series, safety-related valves in the demineralized water supply line to the makeup pump suction to isolate the dilution source. Additionally, the suction line for the CVS makeup pump is automatically realigned to draw borated water from the boric acid tank. The automatic boron dilution protection signal is safety-related and is generated upon any reactor trip signal, source-range flux multiplication signal, or low input voltage to the Class 1E dc and uninterruptible power supply system battery chargers.

The safety analysis of boron dilution accidents is provided in Chapter 15 and is discussed in Subsection 19E.4.5 of this appendix. For dilution events that occur during shutdown, the source-range flux-doubling signal closes the safety-related remotely operated CVS makeup line isolation valves to terminate the event. In addition, the signal is used to isolate the line from the demineralized water system to the makeup pump suction by closing the two safety-related remotely operated valves. The three-way pump suction control valve aligns the makeup pumps to take suction from the boric acid tank and, therefore, stops the dilution.

For refueling operations, administrative controls are used to prevent boron dilutions by verifying that the valves in the line from the demineralized water system are closed and locked. These valves block the flow paths that can allow unborated makeup water to reach the RCS. Makeup required during refueling uses borated water supplied from the boric acid tank by the CVS makeup pumps.

During refueling operations (Mode 6), two source-range neutron flux monitors are operable to monitor core reactivity. This is required by the plant Technical Specifications. The two operable source-range neutron flux monitors provide a signal to alert the operator to unexpected changes in core reactivity. The potential for an uncontrolled boron dilution accident is precluded by isolating the unborated water sources. This is also required by the plant Technical Specifications.

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The source range flux doubling function can be manually blocked during shutdown conditions when below the P-8 setpoint after the operator isolates unborated water source flow paths. When blocked during shutdown conditions, an automatic close signal is also sent to the CVS demineralized water system isolation valves to prevent inadvertent boron dilution.

19E.2.8 Spent Fuel Pool Cooling System

19E.2.8.1 System Description

The spent fuel pool cooling system (SFS) is discussed in Subsection 9.1.3.

19E.2.8.2 Design Features to Address Shutdown Safety

The AP1000 has incorporated various design features to improve shutdown safety. The SFS features that have been incorporated to address shutdown safety are described in this subsection.

19E.2.8.2.1 Seismic Design

The spent fuel pool, fuel transfer canal (FTC), cask loading pit (CLP), cask washdown pit (CWP), and gates from the spent fuel pool-CLP and FTC-spent fuel pool are all integral with the auxiliary building structure. The auxiliary building is seismic Category I design and is designed to retain its integrity when exposed to a safe shutdown earthquake (SSE). The suction and discharge connections between the spent fuel pool and RNS are safety Class C, which is also seismic Category I. The emergency makeup water line from the PCS water storage tank to the spent fuel pool actually connects with the RNS pump suction line. This emergency makeup line is also safety Class C and seismic Category I. The spent fuel pool level instruments connections to the spent fuel pool are safety Class C, seismic Category I, and have 3/8-inch flow restricting orifices at the pool wall to limit the amount of a leak from the pool if the instrument or its piping develops a leak.

The refueling cavity is integral with the containment internal structure, and as such, is seismic Category I, and is designed to retain its integrity when exposed to an SSE. In addition, the AP1000 has incorporated a permanently welded seal ring to provide the seal between the vessel flange and the refueling cavity floor. This permanent cavity seal ring is part of the refueling cavity and is seismic Category I. Figure 19E.2-3 is a simplified drawing of the AP1000 permanent cavity seal ring. The permanent cavity seal ring is designed to accommodate the thermal transients associated with the reactor vessel flange.

19E.2.9 Control and Protection Systems

The AP1000 control and protection systems support the operations necessary for the AP1000 to achieve shutdown. These systems consist of a nonsafety-related plant control system (PLS), a safety-related protection and safety monitoring system (PMS), and a nonsafety-related diverse actuation system (DAS). These systems are discussed in Chapter 7.

19E.3 Shutdown Maintenance Guidelines and Procedures

This section presents an overview discussion of AP1000 shutdown maintenance guidelines and procedures captured as part of the AP1000 design and design certification program. Shutdown maintenance requirements and guidelines have been identified in various licensing submittals, such as the AP1000 Technical Specifications and the design reliability assurance program, (Section 17.4).

Shutdown procedures were addressed in the AP600 design certification program by the submittal of the AP600 Emergency Response Guidelines (ERGs) (Reference 1), which include shutdown emergency procedures. These shutdown emergency procedures are applicable to the AP1000.

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This section summarizes the major shutdown maintenance guidelines and procedures that have been identified.

19E.3.1 Maintenance Guidelines and Insights Important to Reducing Shutdown Risk

This section presents an overview of AP1000 shutdown maintenance guidelines and insights, which are either required for plant safety or are effective at reducing shutdown risk.

19E.3.1.1 Availability Requirements for Safety-Related Systems

Availability controls of the AP1000 safety-related systems are provided by the Technical Specifications. These availability requirements cover all modes of operation including shutdown.

19E.3.1.2 Availability Guidelines for Systems Important for Investment Protection

Availability guidelines for systems important for investment protection are discussed in the AP1000 Design Reliability Assurance Program, Section 17.4.

19E.3.1.3 Reactor Coolant System Precautions and Limitations at Shutdown

Precautions and limitations for RCS operation at shutdown are considered to minimize the risk to plant safety at shutdown. The most important of these are captured in the AP1000 Technical Specifications. However, other precautions and limitations associated with maintenance and operation at shutdown have been identified during the design of the AP1000. These are based on both the past operating experience of PWRs, as well as the designer's knowledge of the unique AP1000 design features. A summary of these precautions and limitations that apply to shutdown maintenance and operation is provided in this section.

19E.3.1.3.1 General Shutdown

Precautions and limitations for general shutdown are as follows:

- To provide mixing, at least one reactor coolant pump (RCP) or a normal residual heat removal pump should be in service while chemicals are being added to the system or the boron concentration is being changed.

- Reactor coolant samples must be taken at the regular intervals to check coolant chemistry, activity level, and boron concentration as specified in the appropriate Technical Specifications including 3.1.1, 3.4.11, and 3.1-1. In addition, during shutdown modes, more frequent checks on RCS boron concentration should be made when changes in RCS boron concentration are being made.

- When the RNS is in operation, the reactor coolant temperature should not exceed 350°F. The reactor coolant pressure should be limited to avoid approaching the RNS relief valve setpoints.

- The maximum allowable heatup and cooldown rates for the RCS are provided in the Technical Specifications.

- During cooldown, the RCPs located in the loop containing the pressurizer spray line should be operated to provide adequate pressurizer spray.

- The accumulators must be isolated prior to reducing the RCS pressure to the accumulator pressure (637 to 769 psig).

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19E.3.1.3.2 Water-Solid Operation

Precautions and limitations for water-solid operation are as follows:

- The RNS inlet line should not be isolated from the reactor coolant loop unless there is a steam bubble in the pressurizer or the makeup pumps are stopped. This precaution provides relief valve protection of the RCS when it is at low pressure and water-solid.

- Whenever the plant is water-solid and the reactor coolant pressure is being maintained by the letdown containment isolation outside-containment valve, the RNS should remain open to the reactor coolant loops to maintain sufficient letdown flow through the bypass line from the RNS to the letdown heat exchanger, until a steam bubble is formed in the pressurizer. During this mode of operation, the isolation valve in the bypass line from the RNS to the letdown heat exchanger should be in the full-open position and the letdown orifice bypass valve should also be open.

- If all RCPs are stopped and the reactor coolant temperature is greater than 200°F, the first pump should not be restarted until a steam bubble has formed in the pressurizer. This precaution will minimize the pressure transient when the first pump is started. The steam bubble will accommodate the resultant expansion.

- When the reactor coolant pressure is being maintained by the letdown containment isolation outside containment valve, changes to the flow rate through the RNS loop by throttling of valves or starting and stopping the RNS pumps will result in changes to the reactor coolant pressure.

- Whenever the reactor coolant temperature is above 160°F, at least one RCP should be in operation.

19E.3.1.3.3 Steam Generators

Precautions and limitations for steam generators are as follows:

- During cooldown, all steam generators should be connected to the steam header to provide uniform cooldown of the reactor coolant loops.

- During steam plant warmup and at hot standby, draw steam slowly and regulate feedwater additions carefully to avoid rapid cooling of the reactor coolant.

- During cooldown, once RNS is in operation, and after the RCPs have been tripped, actions should be taken to cool the contents of the steam generator secondary side, either by recirculation of this water to promote reverse heat transfer to the primary side or by draining the contents via the blowdown lines.

19E.3.1.3.4 Surge Line

During heatup and cooldowns, the temperature difference between the pressurizer and the hot legs should be less than 320°F. This prevents unacceptable stress levels in the surge line.

19E.3.1.3.5 Reduced-Inventory Operations

Precautions and limitations for reduced-inventory operations are as follows:

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- The timing of the initiation of draindown is highly dependent of the scenario that requires the drained condition. However, in order to drain down to mid-loop conditions, the reactor coolant pumps must be tripped, and the RCS temperature must be less than saturation. Typically, the reactor coolant pumps operate until the RCS temperature is reduced to less than 160°F. For a refueling outage, the transition to reduced inventory conditions should typically begin about 3-4 days after shutdown. For a forced outage condition, reduced inventory operations should not begin until the RCS temperature is less than 160°F.

  The time after shutdown directly affects the time that the RCS would boil and the rate at which inventory would be depleted following a loss of cooling event. Table 19E.2-1 presents the time to reach saturation, and the time to core uncovery for a loss of heat sink event initiated from mid-loop conditions at 28 hours after shutdown. For loss of heat sink events initiated earlier, the time to reach saturation and the time to uncover the core would be slightly decreased. The performance of the IRWST injection, in conjunction with ADS, is sufficient to mitigate the consequences of the event.

  The time after shutdown impacts the requirements for containment closure during shutdown as discussed in Subsection 19E.2.6.2 of this appendix (and captured in the Technical Specifications). For reduced inventory conditions, if the time to steaming (inside containment) following a loss of heat sink event is less than the time required to close the containment equipment hatches, then these hatches should be closed. If the time after shutdown is sufficiently long, such that steaming to containment would not occur prior to the containment hatches being able to be closed, then the equipment hatches could be open, with the ability to close them.

- As the RCS is drained, the rate of change in water level will vary non-linearly for a given drain rate due to the geometry of the RCS and the offset hot leg and cold leg piping. It is important to drain the RCS at a low rate to minimize the possibility of overdraining the system. Evaluations have been performed that indicate that a drain rate of 20 gpm is sufficient once the water level has been reduced to the top of the hot leg.

- After maintenance operations that result in draining the RCS, the system should be refilled with borated makeup water at the prevailing RCS boron concentration via the chemical and volume control system (CVS) makeup pumps. If the RCPs are drained, the pumps should be refilled with borated water via the pump drain line so that the pump is completely filled with borated water.

- After maintenance operations on the CVS purification loop (demineralizer, filters, and heat exchangers), the system should be purged, draining potential unborated water to the liquid radwaste system, and refilling it with borated water from the RCS. These operations should not be conducted at mid-loop or reduced inventory operations to avoid an inadvertent drop in RCS water level during mid-loop.

- The RCS hot leg level instruments should be operable and available prior to reduced inventory operations. Their automatic actuation functions are required to be operable in shutdown modes as described in Technical Specification 3.3.10.

19E.3.2 Shutdown Risk Management

This appendix contains insights of which Westinghouse is currently aware and which are related to AP1000 design certification.

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19E.3.3 Shutdown Emergency Response Guidelines Overview

The AP600 ERGs (Reference 1) provide functional guidance for responding to accidents and transients that affect plant safety during shutdown modes of operation (operational Modes 5 and 6). The shutdown ERGs consist of a shutdown safety status tree for monitoring the critical safety functions and six shutdown guidelines for responding to the respective challenges to plant safety. The AP600 ERGs are applicable to AP1000 for the purpose of developing Emergency Operating Procedures.

The shutdown safety status tree provides a systematic method of determining the safety status of the plant. This status tree represents the critical safety functions that are of concern during plant shutdown conditions. Prior to this shutdown condition, the plant can be in any state ranging from heatup and pressurization (from 200°F to no-load temperature) to full power operation. Under these conditions (plant Modes 4 through 1), plant monitoring and response to a reactor trip or requirement for safety injection are covered by the optimal recovery guidelines, status trees, and function restoration guidelines of the at-power ERGs.

By using the shutdown status tree, plant conditions are monitored during plant shutdown after entering Mode 5 while normal operating procedures are in use for plant shutdown operations. The shutdown safety status tree is arranged so that the functions are checked in order of importance. Core cooling during shutdown conditions is addressed first. During plant shutdown conditions, the RNS provides core cooling, which requires adequate RCS inventory to operate properly. RCS inventory checks are made first to show core cooling will not be interrupted because of inadequate RCS inventory and as an early symptom to a loss of shutdown core cooling. After adequate RCS inventory is checked, RNS operation is checked to verify shutdown core cooling is being provided by the RNS. After RNS operation is verified, containment radiation is checked so that an unexpected uncontrolled release will not occur because containment integrity may be breached during plant shutdown maintenance activities. Core reactivity is then checked by monitoring source range flux doubling as an early symptom of an unintended RCS boron dilution, which should occur at a slow enough rate to allow appropriate action to be taken to reestablish shutdown margin. RCS cold overpressure symptoms of RCS pressure and temperature are monitored for maintaining the RCS pressure boundary integrity safety function.

Lastly, RCS temperature change, aside from any normal expected RCS temperature change, is used as an early symptom for potential degradation of the core cooling safety function and the RCS pressure boundary integrity safety function. The shutdown safety status tree is considered to be satisfied when all status tree blocks have been satisfied. If a challenge is identified during the monitoring of the tree, the tree directs plant operators to one of the appropriate six shutdown guidelines for mitigating actions.

The format and arrangement of the shutdown ERG documentation is similar to the at-power ERGs consisting of guidelines and background documents. Implementation of the shutdown ERGs into plant procedures will also be similar to the at-power ERGs with the task allocation between the man and the computer for doing this to be decided when designing features of the man-machine interface system.

19E.4 Safety Analyses and Evaluations

19E.4.1 Introduction

This section reviews each of the design basis accidents (DBAs) and transients presented in Chapter 15, with respect to lower power and shutdown modes. In Subsections 19E.4.2 through 19E.4.9, evaluations or analyses are performed for each case of the transient and LOCA analyses for events occurring at low power and shutdown operations, including the reduced reactor coolant system

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(RCS) inventory and refueling operations. The evaluations discuss the effects of key plant parameters (for example, plant control parameters, neutronic and thermal hydraulic parameters, and engineering safety features [ESFs]) on plant transient response (such as departure from nucleate boiling ratio [DNBR], peak pressure, and peak cladding temperature). The limiting case for each event category is identified. For those limiting cases bounded by the cases analyzed at power conditions, supporting rationales are provided.

For those events where analyses are presented in the shutdown modes, a discussion of the adequacy of the codes used is presented in Subsection 19E.4.1.2.

In Subsection 19E.4.10, additional analyses and evaluations demonstrate that the passive systems can bring the plant to a stable, safe condition and maintain this condition.

19E.4.1.1 Matrix of Chapter 15 Events

Table 19E.4.1-1 provides a list of Chapter 15 events. This table categorizes the events as “E” (requiring evaluation), “A” (requiring analysis), or “n/a” (not applicable). The “n/a” events are bounded by at-power analyses or current analyses.

The events denoted by an “n/a” in Table 19E.4.1-1 are as follows:

- Boron dilution design basis transient explained in Subsection 15.4.6 because it explicitly considers all modes such that no analysis or evaluation is required for this appendix
- Rod cluster control assembly (RCCA) withdrawal at-power explained in Subsection 15.4.2 because this event occurs only at-power

19E.4.2 Increase in Heat Removal from the Primary System

19E.4.2.1 Feedwater System Malfunctions Which Increase Heat Removal from the Primary System

Faults that decrease feedwater temperature or increase feedwater flow can be postulated in the feedwater system. These faults could increase heat removal from the primary system, which reduces RCS temperature. The reduction in RCS temperature could lead to an increase in core power generation (due to a negative moderator temperature coefficient) and result in a reduction in margin-to-core design limits. Unchecked, excessive feedwater flow could also result in overfilling the steam generators.

Discussions and analyses, initiated from Modes 1 and 2, of RCS cooldowns caused by feedwater system malfunctions are presented in Subsections 15.1.1 and 15.1.2. Subsection 15.1.1 covers reductions in feedwater temperature, and Subsection 15.1.2 covers increases in feedwater flow. Modes 1 and 2 are the limiting initial conditions for feedwater system induced RCS cooldown transients.

Protection against feedwater system induced cooldown transients is provided by the protection and safety monitoring system (PMS) through automatic functions that trip the reactor and isolate the feedwater system. The protection functions are available in all modes during which the feedwater system is in operation. Reactor trip includes overpower Δt, high power-range nuclear flux, high intermediate-range nuclear flux, or high source-range nuclear flux. The PMS closes the main feedwater control valves on low-1 RCS average temperature signal. The PMS also closes the main feedwater isolation valves and trips the booster/main feedwater pumps when RCS average temperature decreases below the low-2 RCS T_{avg} setpoint. These protection functions are arranged to detect symmetrical plant transients with a channel out of service and a single channel failure.

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Additional PMS functions are provided to detect and protect against asymmetrical feedwater system malfunctions. Automatic reactor trip, closure of the main feedwater control and isolation valves, closure of the startup feedwater control and isolation valves, tripping of the booster/main feedwater pumps, and tripping of the startup feedwater pumps occur if the level in a single steam generator is above the High-3 water level setpoint. Similar actions occur if cold leg temperature in a single RCS loop decreases below the Low-2 T_{cold} setpoint. The High-3 steam generator level setpoint is active in Modes 1 through 4 unless the various feedwater valves are closed. This ensures that the steam generators cannot inadvertently be overfilled. The Low-2 T_{cold} signal is available in Modes 1 through 3. In Mode 3 prior to blocking the Low-2 T_{cold} signal, the RCS must be borated to cold shutdown conditions. With the RCS borated, no feedwater malfunction can be postulated to cool the RCS such that a core power excursion would occur.

The feedwater malfunction associated with a drop in feedwater temperature is less severe as power level is decreased. Normal operating feedwater temperature decreases as plant power level decreases. Therefore, if a fault suddenly reduces the feedwater temperature, the maximum change in feedwater temperature will occur if the plant is operating at full power.

As discussed in Subsection 19E.2.2 of this appendix, in Modes 2 and below, feedwater entering the steam generators is routed through the startup feedwater control valves. The maximum achievable flow rate through the startup feedwater path is much less than when flow is being controlled by the main feedwater control valves. Therefore, failure of a main feedwater control valve in Mode 2 and below is not likely. The assumption of a failed open startup feedwater control valve, in Mode 2 and below, will result in a relatively slow transient due to low feedwater flow rate.

The most severe RCS cooldowns caused by feed system malfunctions will occur in Modes 1 or 2. In Modes 3 or 4, RCS cooldowns due to feedwater malfunctions would be precluded, inconsequential, or less severe than in Modes 1 or 2. The analyses presented in Chapter 15 bound the consequences of this class of events initiated in the shutdown modes.

19E.4.2.2 Excessive Increase in Secondary Steam Flow

An excessive increase in secondary steam flow (excessive load increase) is caused by a rapid increase in steam flow that results in a power mismatch between the reactor core power and the steam generator load demand. The plant control system (PLS) is designed to accommodate a 10-percent step load increase in steam flow in the range of 25 to 100 percent of full power. Analyses results for a 10-percent step increase in steam flow are presented in Subsection 15.1.3. The analyses are performed for Mode 1 from full-power initial conditions. Depending upon the plant and PMS characteristics (setpoint uncertainties), a reactor trip signal may or may not be generated for an excessive load increase from full power.

An excessive load increase in Mode 1 is considered limiting because an excessive load increase at full power will put the plant at the highest achievable power level. Load increases at less than full power, or during startup (Mode 2), will not reach as high a power level. The excessive load increase, in Mode 2, will not be as severe as the Mode 1 excessive load increase.

In Mode 3, the excessive load increase may be considered to be a simple steam release because there can be no load, per se, when the turbine is off-line and the core is subcritical. The Mode 3 load increase will be less limiting than the Mode 1 or Mode 2 case because the core is already subcritical. Automatic safeguards actuation signals may not be available if blocked by the operator (blocking is necessary to depressurize and cool down the RCS). However, the RCS must be borated to meet shutdown margin requirements at cold shutdown (200°F) prior to blocking automatic safeguards actuation signals to prevent a return to criticality in the event of a cooldown.

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The Mode 4 situation is bounded by Mode 3 because pressure and temperature conditions in the primary and secondary systems are reduced. At some point in Mode 4, the RNS will be placed in service. In Modes 5 and 6, the RNS should be in operation. Any steam release will have little or no effect upon the core.

19E.4.2.3 Credible and Hypothetical Steam Line Breaks

The spurious opening of a steam generator safety or relief valve is a Condition II event and referred to as a credible steam line break. This event affects the core like a load increase but the analysis assumptions that are applied are different. The credible steam line break is usually assumed to be an unisolatable, uncontrolled steam release, which causes a non-uniform core cooldown (typical of an open safety valve) during the period immediately following a reactor trip which inserts all but the most reactive rod cluster control assembly (RCCA). The resulting reactivity excursion may be large enough to overcome the shutdown margin and return the core to critical, especially when there is little or no decay heat (with power peaking in the region of the stuck RCCA). The credible steam line break is analyzed in Mode 2, and the results are presented in Subsection 15.1.4. The assumptions used in the analysis lead to a more severe, post-trip transient than will result from a load increase initiated in Mode 1.

In Mode 1, prior to reactor trip, the transient characteristics of an inadvertent opening of a steam generator safety or relief valve are similar to the excessive load increase. A reactor trip signal, if needed, may result from overpower ΔT logic. After the reactor trip, the concern becomes a possible return to criticality with the most reactive RCCA stuck in the fully withdrawn position, leading to high local power levels. However, a post-trip return to criticality is less likely when this event occurs in Mode 1 than in Mode 2 because there will be more decay heat present, which tends to retard the cooldown.

In Mode 3, results are expected to be better than the Mode 2 case because pressure, temperature, and flow conditions will be less limiting. An occurrence in Mode 4 will be less severe than in Modes 2 or 3 due to the lower initial RCS temperature, and an effective decoupling of the secondary system from the primary system as the reactor coolant pumps (RCPs) are removed from service and the RNS is started. Automatic safeguards actuation signals are available through Mode 3, until the RCS is borated and the automatic safeguards signals are blocked (see excessive load increase discussion). Both CMTs continue to be available for automatic actuation on low-2 pressurizer level or manual actuation through Mode 4 with the RCS not being cooled by the RNS (see Technical Specification LCO 3.5.2). In Mode 4 with the RNS in operation and in Mode 5 with the RCS pressure boundary intact, one CMT is available for activation if needed.

Any cooldown in Modes 5 and 6 caused by depressurization of the secondary system is meaningless because the RCS is already cold, and the RNS system effectively decouples the steam generators from the core.

The steam line rupture is a Condition IV event, producing a greater uncontrolled steam release than the spurious opening of a steam generator safety valve (described above), but the relative effects in the various modes and requirements for protection equipment are the same. This is the most severe cooldown event.

19E.4.2.4 Inadvertent PRHR HX Operation

Inadvertent actuation of the PRHR HX causes an injection of relatively cold water into the RCS. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. Because the PRHR HX is connected to only one RCS loop, the cooldown resulting from its actuation is asymmetric with respect to the core. Inadvertent actuation of the PRHR HX could lead to an asymmetric power increase and a reduction in margin-to-core design limits.

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A limiting analysis of an inadvertent actuation of the PRHR HX heat exchanger is presented in Subsection 15.1.6. The analysis is initiated in Mode 1 from hot full-power conditions. This is the most limiting case.

The PRHR HX heat transfer rate is a function of the inlet temperature to the heat exchanger and the flow rate through the heat exchanger. PRHR HX heat transfer rate is higher with high flow rates and high inlet temperatures. Therefore, the maximum heat removal rate will occur when the plant is at full-power condition with forced RCS flow and a high hot leg temperature. At plant full-power conditions, the PRHR HX heat removal rate is approximately 10 percent of full power. At hot zero power (HZP) conditions with natural circulation, heat removal by the PRHR HX is approximately 1.5 percent to 2 percent of full power.

The heat sink for the PRHR HX is the in-containment refueling water storage tank (IRWST), in which the heat exchanger is submerged. Prior to actuation of the PRHR HX, the fluid within the heat exchanger is in thermal equilibrium with the fluid in the IRWST. Thus, the PRHR HX is initially filled with relatively cold fluid which is at containment ambient temperature. When the PRHR HX is actuated, the initial fluid outsurge is fluid at containment ambient temperature. Once the original fluid in the PRHR HX is purged, the out-flow temperature trend of the heat exchanger is set by the temperature entering the heat exchanger from the RCS hot leg minus the temperature drop through the heat exchanger. Thus, the outlet fluid temperature is limited by the cooling capacity of the PRHR HX.

If the reactor is at power (Mode 1 or 2) when the PRHR HX is inadvertently actuated, a cooldown induced increase in core power will occur. The transient response will have two parts. As the cold fluid from the PRHR HX, which is initially at the ambient IRWST temperature, enters the RCS, a large core power increase will occur. The magnitude of the power increase is proportional to the volume of the cold fluid in the PRHR HX. Once the original fluid is purged from the PRHR HX, the fluid temperature exiting the PRHR HX increases to a value limited by the cooling capacity of the PRHR HX. Core power will then decrease to a value higher than the initial core power, but in equilibrium with the heat removal capability of the steam generators plus the PRHR HX.

With the assumptions of a protection system channel out of service as allowed by the Technical Specifications, a failure of an additional protection system channel, and maximum instrument uncertainties, the asymmetric core power transient may not result in actuating any overpower reactor trips, such as high nuclear flux or overpower Δt. In this case, the core power transient is controlled only by the initial volume of cold water in the PRHR HX and the heat removal capability of the heat exchanger.

Higher initial core power will result in the largest achievable core power and in more severe consequences. Therefore, if the reactor is at-power, the full-power case produces the worst results.

In Mode 3, because the reactor is subcritical, inadvertent actuation of the PRHR HX produces a less severe power excursion than if the reactor is at power or at HZP with the reactor just critical. If in Mode 3 below no-load temperature, the cooldown caused by the actuation of the PRHR HX results in the cold leg temperature dropping below the Low-2 T_{cold} safeguards signal setpoint. This function actuates a reactor trip, initiates boration by the CMTs, and most importantly, trips all the RCPs. When the RCPs trip, natural circulation flow begins in the RCS and the PRHR HX loop. When natural circulation flow is initiated, the heat removal capability of the PRHR HX decreases to approximately 1.5 percent of full power and the severity of the transient is minimized. With the RCS in natural circulation, the cooldown rate of the RCS is also slowed. If criticality is obtained, boration by the CMTs will bring the core subcritical again.

The Low-2 T_{cold} safeguards signal may be blocked by the operator in Mode 3 to allow plant depressurization and cooldown to lower modes. However, prior to blocking the Low-2 T_{cold}

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safeguards signal, the RCS is borated to the shutdown margin requirements at cold shutdown (200°F). Therefore, in Mode 3 with safeguards signals blocked or in Mode 4, cooldown of the RCS by inadvertent actuation of the PRHR HX will not result in a reactivity excursion, which produces a power increase.

In Modes 5 and 6, the RCS will be borated such that a cooldown-induced power excursion could not be postulated. The RCS will be at 200°F or less, and with initial RCS temperatures this low, no significant cooling of the RCS by inadvertent actuation of the PRHR HX could be postulated.

19E.4.3 Decrease in Heat Removal by the Secondary System

19E.4.3.1 Loss of Load and Turbine Trip

Discussions and analyses of the consequences of loss of load, turbine trip, inadvertent closure of main steam isolation valves (MSIVs), or loss of condenser vacuum are presented in Subsections 15.2.2 through 15.2.5. These events are characterized by a rapid reduction in steam flow from the steam generators. This results in an increase in steam pressure and a heatup of the primary side if the reactor power is not reduced. The effects of the primary to secondary power mismatch during these events are mitigated by tripping the reactor and opening secondary and primary side safety valves. The severity of these events is increased if the primary to secondary power mismatch is increased. Therefore, the most severe results occur if the plant is initially operating in Mode 1 at maximum-rated plant power conditions rather than lower power conditions. The turbine is off-line below Mode 1 and transients related to turbine-related faults cannot occur.

In Modes 2, 3, or 4, the plant may be removing decay heat by dumping steam to the condenser. In Mode 4 when the RCS is below 350°F, decay heat is removed using the RNS. In Modes 2, 3, or 4, the transient response to a loss of condenser vacuum or inadvertent MSIV closure is bounded by the turbine trip analysis from full power because the power mismatch is low. Decay heat removal can still be accomplished by the steam generators through atmospheric steam relief through power-operated relief valves (PORVs) if available or through steam generator safety valves, which are available through Mode 4 (see Technical Specification LCO 3.7.1). Additionally, decay heat can be removed with the PRHR HX, which is available through Mode 5 with the RCS intact (see Technical Specifications LCO 3.5.4 and 3.5.5).

19E.4.3.2 Loss of ac Power

A discussion and an analysis of a loss of ac power event are provided in Subsection 15.2.6. The loss of ac power results in the loss of forced primary coolant flow and the loss of main feedwater flow. This results in a heatup and pressurization of the RCS. If the reactor is at power, the event is mitigated by tripping the reactor. The reactor may be automatically tripped on Low-2 RCP speed, Low-2 RCS flow, Low-2 steam generator narrow range water level, or several other primary side heatup signals. Also reactor trip may occur due to the loss of power to the control rod drive mechanisms.

Following reactor trip, the PRHR HX is activated for decay heat removal. Automatic PRHR HX actuation on Low-2 steam generator wide range water level is available in Modes 1 through 3 and in Mode 4 when the RCS is not being cooled by the RNS. The most limiting case for loss of ac power would be if the plant were at full rated power. This will result in the highest decay heat levels and stored energy in the RCS and the heat removal capability of the PRHR HX will be maximized. In Modes 4 or 5 with the RNS in operation, the plant response to a loss of ac power is the same at the loss of RNS cooling as discussed in Subsection 19E.4.8 of this appendix.

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19E.4.3.3 Loss of Normal Feedwater

The main feedwater system is in operation during Modes 1 and 2. The startup feedwater system is used in Mode 2 below approximately 2 percent power, in Mode 3, and in Mode 4 before the RNS is aligned. In Mode 4 with the RNS aligned and in Modes 5 and 6, the feedwater system is not used, and therefore, loss of feedwater events is irrelevant.

A discussion and an analysis of a loss of normal feedwater event from rated full-power conditions are provided in Subsection 15.2.7. The loss of normal feedwater flow results in a heatup and pressurization of the RCS. If the reactor is at-power, the event is mitigated by tripping the reactor on Low-2 steam generator narrow range water level.

Following reactor trip, the PRHR HX is activated for decay heat removal. Automatic PRHR HX actuation on Low-2 steam generator wide range water level is available in Modes 1 through 3 and in Mode 4 when the RCS is not being cooled by the RNS. The most limiting case for a loss of normal feedwater is with the plant initially at full rated power. This case will have the highest decay heat levels and stored energy in the RCS and the heat removal capability of the PRHR HX will be maximized. The analysis initiated from full power bounds cases initiated from the shutdown modes.

19E.4.3.4 Feedwater System Pipe Break

Depending upon the size of the break and plant operating conditions, the break could cause either an RCS heatup or an RCS cooldown. The cooldown aspects are less severe than a steam line break, which is discussed in Subsection 19E.4.2.3 of this appendix and is not considered in the following discussion.

The main feedwater system is in operation during Modes 1 and 2. The startup feedwater system is used in Mode 2 below approximately 2 percent power, in Mode 3, and in Mode 4 before the RNS is aligned. In Mode 4 with the RNS aligned and in Modes 5 and 6, the feedwater system is not used, and therefore, a loss of feedwater caused by a feedwater system pipe break will not cause a heatup of the RCS.

A discussion and an analysis of feedwater system pipe break from rated full-power conditions are provided in Subsection 15.2.8. A rupture of a feedwater system pipe results in a loss of feedwater flow causing a heatup and pressurization of the RCS. If the reactor is at-power, the event is mitigated by tripping the reactor on Low-2 steam generator narrow range water level.

Following reactor trip, the PRHR HX is activated for decay heat removal. Automatic PRHR HX actuation on Low-2 steam generator wide range water level is available in Modes 1 through 3 and in Mode 4 when the RCS is not being cooled by the RNS. The most limiting case for a feedline break occurs with the plant at full rated power. This case will have the highest decay heat levels and the highest stored energy in the RCS and the heat removal capability of the PRHR HX will be maximized.

19E.4.4 Decrease in Reactor Coolant Flow Rate

19E.4.4.1 Partial and Complete Loss of Forced RCS Flow

A partial loss of forced RCS flow may be caused by a mechanical or an electrical failure in an RCP or from a fault in the power supply to the pumps. An RCP failure will result in only the loss of a single RCP. A fault in the power supplies for the RCPs could result only in the loss of one, two, or all four RCPs.

The loss of one or more RCPs reduces the heat removal rate from the primary to the secondary coolant system and thereby causes a heatup in the RCS. The heatup of the RCS results in an

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increase in RCS pressure and a decrease in margin-to-core design limits (that is, departure from nucleate boiling [DNB]). An occurrence at full power will produce a greater and more rapid heatup than at part-power conditions or low-power conditions in Mode 2. Therefore, for evaluating the maximum RCS pressure or the minimum DNB ratio, analyses are performed at full-power conditions. Analyses for partial loss of forced RCS flow transients are presented in Subsection 15.3.1. Analyses for a complete loss of flow are presented in Subsection 15.3.2. These analyses bound loss of flow events initiated in other modes.

Protection for loss of forced RCS flow events is provided by tripping the reactor. This reduces reactor power and preserves margin-to-DNB limits. The AP1000 PMS includes a reactor trip on Low-2 RCS flow in any cold leg and a reactor trip on Low-2 RCP speed in any two of four RCPs. These two reactor trips are used to detect all possible partial and complete loss of RCS flow transients. Opening of the pressurizer safety valves in conjunction with the reactor trip prevents overpressurization of the RCS.

Below Mode 2, when the core is subcritical, forced RCS flow is not needed because margin-to-DNB is not an issue. It is common to have one or more RCPs out of service below Mode 2 because full RCS flow is no longer needed. In Modes 3 through 5, LCO 3.4.4 of the Technical Specifications requires that all four RCPs need to be operating if the Plant Control System is capable of rod withdrawal or one or more rods is not fully inserted, to ensure that DNB limits are not exceeded, in the event RCCAs are inadvertently withdrawn. If the Plant Control System is incapable of rod withdrawal, all rods are fully inserted, and unborated water sources are isolated from the RCS, no RCPs are required to be operating in Modes 3 through 5.

Following reactor trip in loss of forced RCS flow events, decay heat removal is required. The PRHR HX or the steam generators can be used for decay heat removal. In the event of a complete loss of forced RCS flow, RCS natural circulation is adequate to remove core decay heat. This is demonstrated by the loss of ac power analysis presented in Subsection 15.2.6.

19E.4.4.2 Reactor Coolant Pump Shaft Seizure or Break

An RCP shaft seizure or break results in a partial loss of forced RCS flow. The results are similar to partial loss of flow events discussed in Subsection 19E.4.4.1 of this appendix except that the rate of flow reduction is much more rapid if an RCP shaft breaks or seizes. Like the partial loss of flow, a locked or broken RCP shaft reduces the heat removal rate from the primary to secondary coolant system and thereby causes a heatup of the RCS. An occurrence at full power produces the most severe heatup transient. The discussion for the partial loss of flow with respect to limiting modes and protection is applicable to the RCP shaft seizures or breaks.

Analyses and evaluation of RCP shaft seizures and breaks for Mode 1, from full-power conditions, are provided in Subsections 15.3.3 and 15.3.4. The analyses bound events initiated from the shutdown modes.

19E.4.5 Reactivity and Power Distribution Anomalies

19E.4.5.1 Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

An uncontrolled RCCA bank withdrawal from a subcritical condition could cause a reactivity excursion, which if not terminated by a reactor trip, could result in DNB. Subsection 15.4.2 presents an analysis for the uncontrolled RCCA bank withdrawal from a subcritical condition in Mode 2. Assumptions are used that make the analysis bound an occurrence in Modes 2, 3, 4, or 5. Specific conservative assumptions are made for the number of RCPs operating, the reactor trip functions credited, initial RCS temperature, and the magnitude of the reactivity excursion.

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A single failure in the rod control system could cause the withdrawal of only one bank, and its withdrawal rate would be expected to be slower than the maximum rod speed possible when in automatic rod control. The analysis assumes the simultaneous withdrawal of the combination of two sequential RCCA banks having the greatest combined worth at the maximum possible speed.

LCO 3.3.1–3.3.7 of the AP1000 Technical Specifications gives the operational requirements for reactor trips. The source-range high neutron flux trip must be in operation in Modes 3, 4, and 5 if the reactor trip breakers are closed. If the reactor trip breakers are open, then an RCCA withdrawal is precluded from occurring. The source-range high neutron flux trip is available in Mode 2 if power is below the P-6 interlock. In these instances, the source-range high neutron flux trip would be available to terminate the event, by tripping any withdrawn and withdrawing rods, before any significant power level could be attained. Therefore, DNB would be precluded. The intermediate-range high neutron flux reactor trip is also available in Mode 2. The analysis assumes that reactor trip does not occur until the power-range (low setting) high neutron flux setpoint is reached. No credit is assumed in the analysis for the source-range high neutron flux reactor trip or the intermediate-range high neutron flux reactor trip.

LCOs 3.4.4 and 3.4.5 of the AP1000 Technical Specifications give the operation requirements for RCPs. LCO 3.4.4 specifies that all four RCPs must be operating whenever the Plant Control System is capable of rod withdrawal or one or more rods is not fully inserted in Modes 1 through 5.

The RCS temperature is assumed to be at the HZP value in the analysis. This is more limiting than that of a lower initial system temperature for DNB and core kinetics feedback calculations.

These conservative assumptions result in the core returning to critical and generating power before reactor trip occurs. The analysis presented in Chapter 15 bounds the inadvertent RCCA bank withdrawal from a subcritical condition transient in Modes 2 through 5.

19E.4.5.2 Uncontrolled RCCA Bank Withdrawal at Power

This transient is defined only in Mode 1.

19E.4.5.3 RCCA Misalignment

RCCA misalignment events are analyzed in Subsection 15.4.3. RCCA misalignment events include the following:

- One or more dropped RCCAs
- Statically misaligned RCCA
- Withdrawal of a single RCCA

This group of events may result in core radial power distribution perturbations, which may cause allowable design power peaking factors and DNB design limits to be exceeded. Therefore, these events are a concern only in the at-power modes, and the severity will be increased at high power. If the reactor is subcritical, DNB will not be a concern.

Following the dropping of one or more RCCAs while at-power, core power will immediately be reduced. The reduced core power and the continued steam demand to the turbine causes a reactor coolant temperature decrease. If the reactor is in manual control, the core power rises due to moderator feedback to the initial power level at a reduced core inlet temperature. If the reactor is in automatic control, the control system detects the drop in power and initiates withdrawal of a control bank. Power overshoot above the initial power level may occur as the control system withdraws a bank. Following dropping of one or more RCCAs, the most severe results occur when the control system overshoots the initial power level in conjunction with a perturbation in the radial power

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distribution. This is the most limiting case for this event, and the results are presented in Chapter 15. If the reactor is in any of the subcritical modes, dropping RCCAs will not result in any power transient.

As in the case of dropped RCCAs, statically misaligned RCCAs have no effect in the absence of a critical neutron flux and are not a concern below Mode 2. The most limiting case, and analysis, is for Mode 1 which also bounds Mode 2 operation.

The most limiting case for the withdrawal of a single RCCA is an occurrence while in Mode 1. An occurrence in any of the subcritical modes will have no effect. The shutdown margin requirements are specified in LCO 3.1.1 of the AP1000 Technical Specifications. The shutdown margin requirements are determined assuming the most reactive RCCA is fully withdrawn from the core. Therefore, no single RCCA withdrawal initiated from the subcritical modes will insert enough reactivity to attain criticality.

19E.4.5.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

This event is precluded from occurring during at-power modes by Technical Specifications. Startup of an inactive RCP while in any of the subcritical modes will have relatively little effect upon core temperature because there will be little or no temperature difference between the loops. Subsection 15.4.4 discusses the consequences of this event for the AP1000.

19E.4.5.5 Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant

Boron dilution analyses and evaluations for Modes 1 through 5 are provided in Subsection 15.4.6. In Mode 6, administrative controls isolate the RCS from potential sources of unborated water by locking closed specified valves in the chemical and volume control system (CVS), and Demineralized Water Transfer and Storage System, and thereby preclude an uncontrolled boron dilution transient. Makeup needed during refueling is supplied from the boric acid tank which contains borated water.

19E.4.5.6 Inadvertent Loading of a Fuel Assembly in an Improper Position

Fuel loading errors – such as inadvertent loading of one or more fuel assemblies into improper positions, having a fuel rod with one or more pellets of the wrong enrichment, or having a fuel assembly with pellets of the wrong enrichment – may result in power shapes in excess of design values. Subsection 15.4.7 presents Mode 1 results for this event which bound the results for operation in Mode 2. This event is meaningful only if the reactor is at-power and, therefore, not applicable in the subcritical Modes of 3 through 6.

19E.4.5.7 RCCA Ejection

Analyses for RCCA ejections in Mode 1 and Mode 2 are presented in Tier 2 Information Subsection 15.4.8. The cases analyzed in Chapter 15 are the most limiting cases. The shutdown margin requirements are specified in LCO 3.1.1 of the AP1000 Technical Specifications. The shutdown margin requirements are determined assuming the most reactive RCCA is fully withdrawn from the core. Therefore, the ejection of a single RCCA initiated from the subcritical modes would not insert enough reactivity to attain criticality.

19E.4.6 Increase in Reactor Coolant Inventory

An increase in RCS inventory could be caused by inadvertent actuation of the CMTs or by malfunctions in the CVS system. Analyses of events that increase the RCS inventory are provided in Section 15.5. Subsection 15.5.1 presents the analysis results for inadvertent actuation of the CMT. Subsection 15.5.2 contains results from the analysis of a CVS malfunction which increases RCS

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inventory. These events do not present a challenge to core design limits. If unchecked, these events could lead to an overfill of the pressurizer and possible loss of reactor coolant from the system. The increase in pressurizer water volume is slow during these events and is controlled by the injection rate, core decay heat produced, and heat removal rate from the RCS. While the pressurizer safety valves may open, the steam relief from the pressurizer safety valves is low and no serious challenge to the RCS pressure boundary occurs (if the pressurizer does not fill).

The Chapter 15 analyses for these events are performed with the plant initially in Mode 1 at full-power conditions. This results in the maximum amount of stored energy in the plant and in the maximum core decay heat. If the plant was assumed to be at part power, or in the subcritical modes, the amount of stored energy and decay heat will be significantly reduced.

If a spurious “S” signal occurs causing the CMTs to be actuated, the reactor is also tripped and the PRHR HX is also actuated. The CMTs will begin injecting cold, borated fluid into the RCS. The injected fluid expands as it is heated in the RCS by decay heat. The expansion is counteracted by decay heat removal through the PRHR HX. The severity of the expansion is increased with higher decay heat levels.

Malfunctions in the CVS, which add excess inventory to the RCS, are protected against by the inclusion of automatic CVS isolation functions in the PMS. If a safeguards signal has occurred (which also would activate the CMTs), the CVS is automatically isolated if the pressurizer level exceeds the high-1 pressurizer level setpoint. Above the high-1 pressurizer level setpoint, there is a high-2 pressurizer level setpoint, which also isolates the CVS. The high-2 pressurizer level function is not interlocked with the safeguards signal. The high-2 function protects in situations where the reactor is at-power or a safeguards signal has not occurred. The high-2 pressurizer level function is available in Modes 1 through Mode 3 and in Mode 4 when the RNS is not operating. These functions effectively prevent overfilling of the pressurizer when the CVS acts alone or where CVS interacts to also cause the CMTs to be actuated.

Isolation of CVS on high-2 pressurizer level is available in Modes 1 through 4 until the plant is operating on RNS. There are applications where the RCS may be filled water-solid when the RNS is in operation. In Modes 4, 5, and 6 when the RNS is in operation, low-temperature overpressure protection (LTOP) of the RCS pressure boundary is provided by the RNS relief valves. A discussion of this is provided in Subsection 19E.4.10.1 of this appendix.

19E.4.7 Decrease in Reactor Coolant Inventory

19E.4.7.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the Automatic Depressurization System

Subsection 15.6.1 includes analyses and evaluations of the inadvertent opening of a pressurizer safety valve or the inadvertent operation of the automatic depressurization system (ADS).

When analyzed as depressurization events, inadvertent opening of primary side relief valves, if the reactor is at-power, could result in exceeding core design limits, specifically DNB criteria. Violation of DNB criteria is not a realistic concern if the reactor is in any of the subcritical modes. Therefore, these events are analyzed in Mode 1 at the maximum rated power and the analysis performed bounds cases initiated from Mode 2. These events bound events that can occur at shutdown.

The inadvertent ADS is analyzed as a loss-of-coolant accident in Mode 1 to demonstrate acceptance to the limits specified in 10 CFR 50.46. As described in Subsection 15.6.5, this analysis is a “no-break” small-break LOCA calculation. The inadvertent opening of the 4-inch nominal size ADS Stage 1 valves is a situation that minimizes the venting capability of the RCS. Only the ADS valve vent area is available; no additional vent area exists due to a break. This case examines whether sufficient vent

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area is available to completely depressurize the RCS and achieve injection from the IRWST without core uncovery. The case analyzed at-power bounds the inadvertent ADS during shutdown because the lower decay heat levels at shutdown reduce the challenge to the ADS vent capacity. More limiting loss-of-coolant accidents at shutdown are analyzed as described in Subsection 19E.4.8.

19E.4.7.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

This event is reported in Subsection 15.6.2 as the rupture of a primary coolant sample line; the radiological consequences of this event are analyzed during Mode 1 because the coolant temperature and iodine concentrations bound those that would exist in the other modes. Concerning shutdown risk, the consequences of a sample line break during Modes 2, 3, 4, or 5 are no more severe than if the accident occurs during Mode 1 operation.

19E.4.7.3 Steam Generator Tube Rupture in Lower Modes

The steam generator tube rupture (SGTR) analysis presented in Chapter 15 is the limiting case with respect to offsite doses. The analysis was performed at full power because this results in the maximum offsite dose. The key inputs from the thermal-hydraulic SGTR analysis performed with the LOFTTR2 computer code to the offsite dose analysis are the amount of flashed primary to secondary break flow and the steam released from the faulted steam generator. Both of these will be significantly reduced at lower power levels and in lower modes of operation.

Margin to overfill analyses are not presented in Chapter 15, however an analysis is performed to demonstrate margin to steam generator overfill with no operator actions modeled. This is necessary because the dose analysis does not include consideration of water relief from the ruptured steam generator PORV/MSSV. This margin to steam generator overfill analysis was supported by the assertion that an analysis with operator actions modeled will also demonstrate margin to overfill. The overfill analysis with no operator actions discussed in Chapter 15 was initiated at full power. WCAP-10698-P-A (Reference 9) indicates that margin to overfill is reduced when the SGTR is initiated at zero power because of the higher initial steam generator secondary liquid inventory. WCAP-10698-P-A concludes that zero power and lower mode SGTR overfill analyses are not limiting, based primarily on more rapid operator responses expected in those conditions. This is discussed further in the Appendices to WCAP-10698-P-A. When operator actions are credited for AP1000 SGTR mitigation, the plant behaves in a manner comparable to a standard Westinghouse PWR and the conclusions of WCAP-10698-P-A apply.

When operator actions are not relied upon and only the AP1000 automatic RCS cooling and depressurization are credited, margin to overfill would still be maintained for SGTR events initiated at lower power levels and lower modes despite the increased initial steam generator secondary side inventory corresponding to the lower initial power assumption. This is because the automatic protection system actions that prevent overfill are independent of the operator actions. For operating plants, there is a set period of time from the start of the event until the operator can reverse the trend toward filling the steam generator. Therefore, the initial margin to overfill directly impacts the final margin. For the AP1000, the primary cooldown and depressurization occur automatically when the PRHR HX is actuated on a Low-3 pressurizer pressure “S” signal or Low-2 pressurizer level CMT actuation signal. The primary pressure may still be held up by the CVS, until it is isolated on a high steam generator level signal. For the AP1000, a higher initial steam generator water level results in the CVS flow being terminated earlier.

In Mode 4, the PRHR HX actuation is provided by the Low-2 pressurizer level signal. Although this results in delayed cooling and depressurization, margin to steam generator overfill is still maintained. The increase in mass in the secondary side of the ruptured steam generator is directly related to the reduction in pressurizer water level, because (once the CVS is isolated on high steam generator water level) there is no source of makeup to the RCS. The steam generator secondary side can

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accommodate the amount of fluid initially contained in the pressurizer and still retain a significant amount of margin to steam generator overfill. The PRHR HX will, therefore, be actuated on Low-2 pressurizer water level in sufficient time for the PRHR HX to cool and depressurize the primary and terminate break flow before steam generator overfill will occur.

In Mode 4 with an RCS temperature less than 350° F and in Modes 5 and 6, the RCS pressure and temperature are reduced; thus, an SGTR event is not considered credible.

19E.4.8 Loss-of-Coolant Accident Events in Shutdown Modes

The AP1000 DCD presents a spectrum of break sizes of the postulated LOCAs at the full-power operating condition. Other things being equal, the reduction in power to decay heat levels associated with shutdown mode operations will make all LOCA events less limiting than those analyzed at full power and reported in Subsection 15.6.5. However, as the plant proceeds through shutdown modes of operation, various PXS equipment are removed from service at identified points in time. One particularly significant action in the course of taking the AP1000 to cold shutdown, in the elimination of PXS equipment, is the isolation of the accumulators at 1000 psig. This procedural action reduces the capability of the PXS to mitigate LOCAs. For assessing the adequacy of the remaining PXS components to mitigate postulated LOCA events, the limiting double-ended cold-leg guillotine (DECLG) break, analyzed in Chapter 15, is analyzed assuming it occurs immediately after the isolation of the accumulators. The analysis is performed using the AP1000 Large-Break LOCA WCOBRA-TRAC model used for the at-power Design Basis Accident analysis. Only safety-related systems are modeled in the analysis of this event.

Depressurization of the AP1000 primary system during shutdown operations will be performed with the same care taken to avoid the flashing of liquid in the core and upper head that is taken by current operating plants. Prudent plant operation dictates that subcooling margin be retained as pressure is reduced. Therefore, since the AP1000 shutdown operations will be conducted in a prudent, controlled manner, it is anticipated that the RCS temperature will be near the 420°F lower limit of Mode 3 when the accumulators are isolated.

For these analyses, the plant was assumed to be shut down in Mode 3 at steady-state conditions of 1000 psig and 425°F with the accumulators isolated. An initial pressure of 1000 psig is assumed because this is the highest pressure with the accumulators isolated and a hot-leg temperature of 425°F is the highest expected temperature when the pressure is 1000 psig. The decay heat level accounts for a ~2-hour power descension from full power to reactor trip plus a ~1-hour post-trip period until the accumulators are isolated. When accounting for the power descension, this leads to a conservative analyzed shutdown time equivalent to 1.71 hours. The Low-3 pressurizer pressure safeguards signal is also assumed to be disabled because the initial pressure is below the setpoint.

19E.4.8.1 Double-Ended Cold-Leg Guillotine

The DECLG break is analyzed using the WCOBRA/TRAC computer code and the AP1000-specific noding, which is based on the AP600 noding, presented in WCAP-14171, Revision 1 (Reference 10). Table 19E.4.8-1 summarizes the results.

This case models the double-ended rupture of one of the two cold legs in the RCS loop without the PRHR HX at a pressure of 1000 psig just after the accumulators are isolated. Only the core makeup tanks (CMTs) and IRWST are available to deliver PXS flow. This break evaluates the ability of the plant to withstand a large LOCA during shutdown with its conditions and equipment availability. The nominal discharge coefficient (1.0) is modeled. The analysis is performed with ANSI-5.1-1979+2σ decay heat, and Technical Specification/Core Operating Limits Report maximum peaking factors.

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The break is assumed to open instantaneously at 0.0 seconds. The subcooled discharge from the broken cold leg (Figure 19E.4.8-1) causes a rapid RCS depressurization (Figure 19E.4.8-2). In Figure 19E.4.8-1, the positive flow direction is the normal operation direction. The reversal of flow entering the vessel to flow out of the break is shown. Due to High-2 containment pressure, an “S” signal is generated at 2.2 seconds. Following a 2.0-second delay, the isolation valves on the CMT and PRHR HX outlet lines begin to open. The reactor coolant pumps trip at 9.5 seconds. The nominal discharge coefficient of 1.0, identified in full-power LOCA analyses, is assumed.

Within a few seconds, the collapsed liquid level drops within the upper plenum due to voiding (Figure 19E.4.8-3). The downcomer collapsed liquid level (Figure 19E.4.8-4) quickly falls below the elevation of the cold legs; the elevation of the top of the core is 20.47 feet.

CMT injection from both tanks replenishes the RCS mass inventory. Injection from the CMTs as the RCS pressure declines terminates the peak cladding temperature (PCT) transient because the stable injection of water from the CMTs exceeds the break flow. The core collapsed level refills as shown in Figure 19E.4.8-5. The draining of the CMTs is delayed by condensation occurring at the top of the tank (Figure 19E.4.8-6(b)). After brief periods of draining early in the transient, sustained CMT injection is established at ~150 seconds for the intact cold leg and ~200 seconds for the broken cold leg. The maximum PCT value is approximately 1840°F for this bounding break size as shown in Figure 19E.4.8-6(a), and all the 10 CFR 50.46 (Reference 15) acceptance criteria are met.

19E.4.8.2 Loss of Normal Residual Heat Removal System Cooling in Mode 4 and Mode 5 with Reactor Coolant System Intact

This analysis is applicable when the RCS pressure boundary is intact less than 28 hours after shutdown. For this analysis, it is assumed that the RNS has just been placed in operation at 2.5 hours after reactor shutdown with the RCS at 350°F and 400 psig (414.7 psia). It is assumed that a loss of offsite power occurs, resulting in a loss of flow through the RNS, and thus, in a loss of RNS cooling. The MSS is assumed to be unavailable for heat removal during the event, except for the main steam safety valves. The steam generator secondary side is assumed to be at saturated conditions for 350°F with the normal water level. Because the Mode 4 plant conditions assumed for the analysis are more limiting than Mode 5 conditions, this analysis is also applicable for a loss of RNS cooling in Mode 5 when the RCS is intact.

The Technical Specifications indicate one CMT is operable during Mode 4 with RCS cooling provided by RNS and Mode 5 with RNS not vented, and all of the fourth-stage ADS flow paths are operable when the RCS is intact less than 28 hours after shutdown. The Technical Specifications permit one of six first-, second-, and third-stage ADS flow paths to be unavailable, and this analysis assumes the second-stage ADS flow path is not operable. One of the four available fourth-stage ADS valves is assumed to fail to open on demand as the single failure, consistent with the single failure assumption used for the full power small-break LOCA analyses.

Two cases were analyzed. The first allowed for automatic safety system actuation on a Low-2 pressurizer level signal late in the event. During this time, the only mechanism for removing decay heat is boiling off the RCS inventory and venting through the RNS relief valve. The second calculation assumes operator action 1800 seconds (30 minutes) after the loss of RNS cooling.

For the scenario with the RCS pressure boundary intact for at least 28 hours after shutdown, the Technical Specifications permit one fourth-stage ADS flow path to be unavailable and indicate only three ADS Stage 1, 2, and 3 flow paths are operable, including a minimum of two paths in the second- and third-stage. This scenario shows increased margin to core uncovery relative to the cases for Mode 4 and Mode 5 RCS intact less than 28 hours after reactor shutdown. Therefore, the cases described herein are bounding with respect to core uncovery margin, and the Mode 5 scenario with RCS boundary intact for at least 28 hours after reactor shutdown is not discussed any further.

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Automatic Safety System Actuation

The accident analyzed is a loss of RNS cooling, which is assumed to result in a complete loss of heat removal for the RCS. The sequence of events for this analysis is presented in Table 19E.4.8-2.

Following the loss of RNS cooling, there is no active mechanism for heat removal from the RCS. The core decay heat generation causes the reactor coolant temperature and pressure to increase. Although the MSS is assumed to be unavailable for heat removal (with the exception of the main steam safety valves), the steam generators represent a heat sink that slows the rate of heatup of the reactor coolant. The fluid temperature at the core outlet for the transient is shown in Figure 19E.4.8-7. The reactor coolant heatup causes the system pressure to increase, as shown in Figure 19E.4.8-8, until the pressure reaches the RNS relief valve setpoint of 500 psig (514.7 psia) at approximately 742 seconds (0.21 hours). The normal relieving capacity of the RNS relief valve is conservatively approximated at 991 gpm at 550 psig (564.7 psia), and the pressure is maintained by the capacities of the relief valves based on the NOTRUMP Henry-Fauske Homogeneous Equilibrium Model of the choke flow area and back pressures at different water pressures and temperatures as conditions change in the RCS. The analysis model considers the negative impact of increases in RCS temperature on relief capacity. After the additional installation of relief valve RNS-V020, in parallel with RNS-V021, the total relieving capacity is within 0.3% (988 gpm) of the flow used in Subsection 19E.4.8, but considering that the 0.3% reduction is based on a higher inlet water temperature for the relief valves (400°F), the actual relieving capacity compared to the analysis of record rated RNS relief flowing temperature of 350°F would show a small benefit to the fluid conditions in the reactor vessel. Flow out the relief valve is shown in Figure 19E.4.8-9. The expansion of the water due to the coolant temperature increase also causes the pressurizer level to increase slightly as shown in Figure 19E.4.8-10.

The loss of reactor coolant through the relief valve is not sufficient to remove the core decay heat, and the reactor coolant temperature continues to increase until the core outlet temperature reaches saturation near the relief valve setpoint at approximately 2785 seconds (0.77 hours). The generation of steam in the core causes the system pressure to increase above the RNS relief valve setpoint and the pressurizer level to continue to increase. A mixture level begins to form in the upper plenum at approximately 2850 seconds (0.79 hours) and drops to the top of the hot-leg elevation as shown in Figure 19E.4.8-11. At about 3319 seconds (0.92 hours), enough mass has been discharged such that a mixture level also forms in the downcomer (Figure 19E.4.8-12) and the downcomer two-phase level begins to decrease. As the boiling front moves lower and lower into the core, more steam generation occurs and the pressure continues to increase. When the entire core length is boiling, the upper plenum mixture level is within the hot-leg perimeter. There is a brief period of two-phase discharge out the RNS relief valve when the upper plenum mixture level drops near the bottom of the hot leg at approximately 7225 seconds (2.00 hours). Following this period there is sufficient steam generation due to the loss of heat sink, that vapor is able to flow into the pressurizer and promote pressurizer draining.

As the pressurizer level decreases, a safety systems actuation signal is generated automatically on Low-2 pressurizer level. Following a 2.0-second delay, the isolation valves on the available CMT tank delivery lines begin to open and CMT injection flow is initiated at approximately 8620 seconds (2.39 hours) as shown in Figure 19E.4.8-13. The opening of the PRHR HX isolation valve on a safety systems actuation signal starts the flow through the heat exchanger. The CMT injection and PRHR actuation causes the reactor coolant pressure to decrease below the RNS relief valve setpoint, and the loss of reactor coolant via the RNS relief valve is terminated at approximately 9145 seconds (2.54 hours). As the CMT level decreases (Figure 19E.4.8-14), the first-stage ADS setpoint at 67.5 percent volume is reached and flow initiates at 9078 seconds (2.52 hours). The second-stage and third-stage ADS valves also open following the timer delays for the actuation of the second-stage and third-stage ADS valves. The vapor and liquid flow through the ADS valves (Figures 19E.4.8-15 and 19E.4.8-16) results in a rapid depressurization of the reactor coolant system. The CMT reaches the fourth-stage ADS setpoint of 20 percent volume, and the first of the fourth-stage paths begins to

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open at 9688 seconds (2.69 hours). As noted previously, one of the four available fourth-stage paths is assumed to fail as the single active failure. The vapor and liquid flow through the fourth-stage ADS paths (Figures 19E.4.8-17 and 19E.4.8-18) further reduces the pressure to the point where IRWST injection begins at approximately 10,018 seconds (2.78 hours) (Figure 19E.4.8-19).

The CMT and IRWST injection reverses the decrease in the core stack and downcomer mixture levels as shown in Figures 19E.4.8-11 and 19E.4.8-12, respectively. As shown in Figure 19E.4.8-11, the core stack mixture level is maintained above the elevation of the top of the core active fuel (20.54 feet) throughout the transient. At the end of the transient, the core stack mixture level has been restored to within the hot-leg perimeter. The fluid temperature at the core outlet has also been reduced and is being maintained at less than 250ºF. As shown in Figure 19E.4.8-20, the reactor coolant mass inventory reaches a minimum of approximately 111,000 lbm; IRWST injection causes the subsequent increase in the inventory. The reactor coolant mass inventory has stabilized near 130,500 lbm at the end of the transient. Thus, it is concluded that the consequences of a loss of RNS in Modes 4 and 5 with the RCS intact are acceptable.

Manual Safety System Actuation

If operator action occurs 1800 seconds (0.50 hours) after the loss of RNS event, the CMT and PRHR isolation valves would open earlier when compared to the automatic case, allowing for heat removal via CMT and PRHR circulation. However, the decay heat is greater than the PRHR capacity and therefore the RCS pressure increases to the RNS relief valve setpoint, causing RCS inventory to be vented (Figure 19E.4.8-21 and Figure 19E.4.8-22). With heat removal being achieved through CMT and PRHR circulation, and via RCS mass discharge out of the RNS relief valve, CMT draining and ADS actuation occurs approximately 1.3 hours later compared to the automatic actuation case. Even with a conservative modeling approach allowing for bulk saturation of the IRWST, the core continues to be cooled by a two phase mixture (Figure 19E.4.8-23). The sequence of events for this case is also shown in Table 19E.4.8-2.

19E.4.8.3 Loss of Normal Residual Heat Removal System Cooling in Mode 5 with Reactor Coolant System Open

This analysis is applicable when the RCS pressure boundary is open at any time after shutdown in Mode 5. For this analysis, it is assumed that the RNS is in operation in Mode 5 at 24 hours after reactor shutdown with one tier of ADS Stage 1, 2, and 3 valves open and the RCS vented to the IRWST. The reactor coolant temperature is assumed to be at 160°F, and the pressurizer pressure is near equilibrium with the IRWST. The steam generator secondary side is assumed to be drained, and thus, there is no secondary heat sink for this case. It is assumed that the CMTs and the PRHR are not available because the Technical Specifications permit them to be taken out of service when the RCS is open in Mode 5. It is also assumed that three of the fourth-stage ADS valves are available for potential use by the operators. In addition, one of the three available fourth-stage ADS valves is assumed to fail to open on demand as the single failure. The Technical Specifications also permit one of the two IRWST injection paths to be out of service in Mode 5 with the RCS open, and thus, only one of the IRWST injection paths is assumed to be available.

It is assumed that a loss of offsite power occurs, resulting in a loss of RNS flow, and thus a loss of RNS cooling. The sequence of events for this analysis is presented in Table 19E.4.8-3.

Following the loss of RNS cooling, there is no mechanism for heat removal from the RCS and the core decay heat generation results in an increase in the reactor coolant temperature. The fluid temperature at the core outlet for the transient is shown in Figure 19E.4.8-24. The core outlet fluid temperature increases steadily until approximately 2160 seconds (0.60 hours) when saturation temperature is reached and voiding is initiated in the core. Because the RCS is vented to the IRWST via ADS Stages 1, 2, and 3, the pressure initially remains relatively constant until approximately 2400 seconds (0.67 hours) as shown in Figure 19E.4.8-25. As the void generation in the system

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continues, the vapor flow through ADS Stages 1, 2, and 3 is not sufficient to maintain the pressure, and it begins to increase. The pressure increases to approximately 93.0 psia, and then begins to decrease. As shown in Figure 19E.4.8-26, the pressurizer level also increases as the reactor coolant temperature increases. The level subsequently reaches the top of the pressurizer as a result of the steam generation in the system. As shown in Figures 19E.4.8-27 and 19E.4.8-28, a mixture of steam and water is discharged via ADS Stages 1, 2, and 3 after the pressurizer fills.

The continued loss of reactor coolant through ADS Stages 1, 2, and 3 causes the pressure to begin to decrease after approximately 4100 seconds (1.14 hours). The core outlet temperature is at saturation and also begins to decrease as the pressure decreases. A mixture level begins to form in the upper plenum at approximately 2380 seconds (0.66 hours), and the level begins to decrease, as shown in Figure 19E.4.8-29, as the voiding continues in the system. At about 3080 seconds (0.86 hours), enough mass has been discharged that a mixture level forms in the downcomer (Figure 19E.4.8-30) and the downcomer level also begins to decrease. The pressurizer level does not decrease significantly due an increasing void fraction in the pressurizer.

As the voiding in the core continues, the core stack mixture level continues to decrease as shown in Figure 19E.4.8-29. The void fraction in the hot legs also increases, and the mixture level in the hot leg begins to decrease after 2375 seconds (0.66 hours). The hot leg is empty at approximately 4554 seconds (1.27 hours) as shown in Figure 19E.4.8-31. This is the normal signal for opening the fourth-stage ADS valves and to initiate IRWST injection when the systems are aligned for automatic actuation. Thus, it is assumed that the operator will initiate manual action at 4554 seconds (1.27 hours) to open the fourth-stage ADS valves and to open the IRWST flow path to permit IRWST injection when the downcomer pressure is sufficiently low. Discharge through one of the fourth-stage ADS valves is initiated at 4674 seconds (1.30 hours) as shown in Figures 19E.4.8-32 and 19E.4.8-33. As noted previously, one of the three available fourth-stage ADS paths is assumed to fail to open as the single active failure. The flow through the fourth-stage ADS path results in a further reduction in the pressurizer pressure and a rapid decrease in the pressurizer level. The downcomer pressure is also reduced to the point where IRWST injection is initiated at approximately 5029 seconds (1.40 hours) (Figure 19E.4.8-34).

The IRWST injection reverses the decrease in the core stack and downcomer mixture levels as shown in Figures 19E.4.8-29 and 19E.4.8-30, respectively. As shown in Figure 19E.4.8-29, the core stack mixture level is maintained well above the elevation of the top of the core active fuel (20.54 feet) throughout the transient. At the end of the transient, the core stack mixture level has been restored to approximately the middle of the hot-leg elevation. The fluid temperature at the core outlet has also been reduced to approximately 230°F. As shown in Figure 19E.4.8-35, the reactor coolant mass inventory reaches a minimum of approximately 122,500 pounds and then begins to increase as a result of the IRWST injection. Thus, it is concluded that when the appropriate operator action is performed, two ADS Stage 4 valves are effective in reducing system pressure so that the consequences of a loss of RNS in Mode 5 with the RCS vented are acceptable.

The analysis presented here is a conservative analysis of a loss of RNS cooling during reduced inventory conditions. During Mode 5, prior to draining to mid-loop conditions from a water solid initial condition, the operator manually opens the ADS Stages 1 through 3 paths to the IRWST. With the RCS open, the operator could proceed to slowly drain the RCS to mid-loop conditions for performing steam generator maintenance or other maintenance that requires a reduced RCS water level. Before the operator starts draining to mid-loop conditions, it is postulated that a loss of decay heat removal via the nonsafety-related RNS occurs. A loss of RNS cooling at this time is selected because it is the earliest time the RCS could be placed into a reduced inventory (that is, RCS open) condition. In addition, the backpressure on the reactor vessel, due to the presence of water in the pressurizer, is higher at this time. This presents the most challenging condition for the ADS to depressurize the RCS to IRWST cut-in pressure. This transient represents the most limiting “surge line flooding” scenario, a term commonly used for operating plants to refer to the phenomenon associated with water in the

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pressurizer and surge line causing a high backpressure in the RCS. This potentially challenges the ability of the low head safety injection systems to inject properly. In addition, this scenario can potentially challenge the design pressure of temporary nozzle dams placed in the steam generators to facilitate maintenance of the RCS during refueling.

Mid-Loop Operations

For a loss of the RNS during mid-loop operations, calculations have been performed to determine the time until core uncovery would occur. The results of these calculations are presented in Table 19E.2-1. The progression of events following a loss of RNS cooling during mid-loop results in a heatup of the core to saturation, followed by a boiling off of the coolant to the IRWST via the ADS Stages 1, 2, and 3 valves. Eventually, the operator actuates the IRWST based upon indications of a loss of RCS subcooling and/or a loss of RCS inventory. The conditions in the RCS following IRWST and fourth-stage ADS actuation are similar to the other cases in this evaluation. As shown in Table 19E.2-1, the operator has at least 42 minutes from the loss of RNS cooling until the onset of core uncovery to manually actuate the IRWST and ADS Stage 4. In general, when considering appropriate operator action to mitigate the event, the results of a loss of RNS during mid-loop conditions are similar, but slightly less severe than the other cases presented in this evaluation due to the lower levels of decay heat and to the absence of the initial water inventory in the pressurizer. This serves to reduce the surge line flooding phenomenon that degrades the depressurization capability of the ADS Stages 1 through 3 vent paths.

19E.4.9 Radiological Consequences

This section presents evaluations that confirm that the radioactive material releases from the AP1000 events postulated to be initiated in a shutdown mode have acceptable consequences.

- The Standard Review Plan (Reference 12) no longer includes the atmospheric releases from radioactive gas waste system failure and radioactive liquid waste system leak or failure events as part of the review. As discussed in Subsections 15.7.1 and 15.7.2, no analysis for these events is provided.

- Release of radioactivity to the environment due to a liquid tank failure is addressed in Subsection 15.7.3 and is not mode dependent.

- The fuel handling accident described in Subsection 15.7.4, while not mode dependent, is analyzed in the applicable and bounding mode and accounts for spent fuel pool boiling. This accident analysis bounds radioactivity releases from other Chapter 15 events during low power and shutdown operations. The LOCA analysis results show PCT remains below 2200°F, and there are no fuel cladding failures.

- The spent fuel cask drop accident described in Subsection 15.7.5 is not mode dependent.

- Appendix 15A contains the evaluation models and parameters that form the basis of the radiological consequences analyses for the various postulated accidents. This methodology applies in all modes of operation.

In summary, there are no shutdown risks associated with the radiological consequences methodology or parameters, or the postulated or applicable events, which need to be considered outside the scope of what is already analyzed for Section 15.7.

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19E.4.10 Other Evaluations and Analyses

19E.4.10.1 Low Temperature Overpressure Protection

For the AP1000, the normal residual heat removal system (RNS) suction relief valves are located immediately downstream of the RCS suction isolation valves. These relief valves protect the RNS from overpressurization and provide low temperature overpressure protection (LTOP) for the RCS components when the RNS is aligned to the RCS to provide decay heat removal during plant shutdown and startup operations. The RNS relief valves are sized to provide LTOP by limiting the RCS and RNS pressure to less than the 10 CFR 50 Appendix G (Reference 13) steady-state pressure limit. Subsection 5.2.2 provides a discussion of the AP1000 low temperature overpressure protection design bases.

19E.4.10.2 Shutdown Temperature Evaluation

As discussed in Subsection 6.3.1.1.4, the PRHR HX is required to be able to cool the RCS to a safe, stable condition after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the PRHR HX can meet this criterion and cool the RCS to the specified, safe shutdown condition of 420°F within 36 hours. This analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (Reference 15) criteria.

As discussed in Subsections 6.3.3 and 7.4.1.1, the PRHR HX operates to reduce the RCS core average temperature to the safe shutdown condition following a non-LOCA event. An analysis of the loss of main feedwater with a loss of ac power event demonstrates that the passive systems can bring the plant to a safe, stable condition following postulated transients. A non-bounding, conservative analysis is represented in Figures 19E.4.10-1 through 19E.4.10-4 The progression of this event is outlined in Table 19E.4.10-1. Though some of the assumptions of this evaluation are based on nominal conditions, many of the analysis assumptions are bounding.

The performance of the PRHR HX is affected by the containment pressure. Containment pressure determines the PRHR HX heat sink (the IRWST water) temperature. The WGOTHIC containment response model described in Subsection 6.2.1.1.3 was used to determine the containment pressure response to this transient, which was used as an input to the plant cooldown analysis performed with LOFTRAN. Some changes were made to the WGOTHIC model to provide conservative results for the long-term safe shutdown analysis.

The PRHR HX performance is also affected by the IRWST water level when the level drops below the top of the PRHR HX tubes. The IRWST water level is affected by the heat input from the PRHR HX and by the amount of steam that leaves the IRWST and does not return to the IRWST through the IRWST gutter arrangement. The principal steam condensate losses include steam that stays in the containment atmosphere, steam that condenses on heat sinks inside containment other than the containment vessel, and dripping or splashing losses due to obstructions on the inner containment vessel wall. The WGOTHIC containment response model also provided the mass balance with respect to the steam lost to the containment atmosphere and to condensation on passive heat sinks other than the containment vessel. The WGOTHIC analysis inputs (including the mass of the heat sinks and heat transfer rates) were biased to increase steam condensate losses. The WGOTHIC model provides the time-dependent condensate return rate, which was incorporated into the LOFTRAN computer code described in Subsection 15.0.11.2 to demonstrate that the RCS core average temperature could be cooled to 420°F within 36 hours.

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Summarizing this transient, the loss of normal feedwater occurs, followed by the reactor trip and loss of ac power. The PRHR HX is actuated on the Low-2 steam generator wide range water level signal. Eventually a safeguards actuation signal is actuated on Low-2 cold leg temperature and the CMTs are actuated.

Once actuated, at about 3,000 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold water, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about 6,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about 49,000 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from Figure 19E.4.10-1 the cold leg temperature in the loop with the PRHR is reduced to 420°F at about 56,530 seconds, while the core average temperature reaches 420°F at about 128,260 seconds (approximately 35.6 hours).

As discussed in Subsection 7.4.1.1, a timer is used to automatically actuate the ADS if offsite and onsite power are lost for about 24 hours. This timer automates putting the open loop cooling features into service prior to draining the Class 1E dc 24-hour batteries that operate the ADS valves. Before 22 hours, if the plant conditions indicate that the ADS would not be needed until well after 24 hours, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the ADS and preserves the ability to align open loop cooling at a later time. Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

As discussed in Subsection 6.3.3.2.1.1, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this section shows the PRHR HX is expected to maintain safe shutdown conditions for greater than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooling can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

19E.5 Technical Specifications

While the Technical Specification guidance provided in NUREG-1449 (Reference 2) relates to existing plant shutdown operation concerns, the underlying concerns relating to causes of events and recovery from those events during shutdown operations are applicable to the AP1000. Subsection 19E.5.1 summarizes the shutdown Technical Specifications. Subsection 19E.5.2 summarizes the AP1000's compliance with SECY-93-190 (Reference 16).

19E.5.1 Summary of Shutdown Technical Specifications

The content of the AP1000 Technical Specifications meets the requirements of 10 CFR 50.36 (Reference 17) and is consistent with the guidance provided in NUREG-1431 (Reference 18). For the AP1000, passive systems are used to safely shut down the plant. Because this design feature is different from existing plants, and because NUREG-1449 provides a reasonable basis for creating shutdown Technical Specifications, the AP1000 Technical Specifications are improved to include specifications for these systems in the shutdown modes.

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19E.6 Shutdown Risk Evaluation

The “AP1000 Probabilistic Risk Assessment (PRA)” (Reference 19) provides an assessment of the plant risk associated with events at shutdown.

19E.7 Compliance with NUREG-1449

The Diablo Canyon event of April 10, 1987, and the loss of ac power event at the Vogtle plant on March 20, 1990, led the Nuclear Regulatory Commission (NRC) staff to issue NUREG-1449, “Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States” (Reference 2), to provide an evaluation of the shutdown risk issue. The scope of NUREG-1449 includes subjects such as operating experiences as documented in generic letters, operator training, technical specifications, residual heat removal capacity, temporary reactor coolant boundaries, rapid boron dilution, containment capacity, fire protection, outage planning and control, and instrumentation.

The NRC requested Westinghouse to assess the compliance of AP600 with NUREG-1449. It was recognized that some of the issues discussed in NUREG-1449 are the responsibility of the plant owners because they relate to operation, maintenance, and refueling plans, procedures, and risk management. However, the NRC believed that the level of defense-in-depth against shutdown events would be improved if clear guidance is provided to the areas discussed above by the plant designer. The NRC requested that Westinghouse perform a systematic assessment of the shutdown risk issue to address areas identified in NUREG-1449 as they are applicable to the AP1000 design and document the results.

This Appendix provides the systematic assessment of the shutdown risk issue to address areas identified in NUREG-1449. This assessment includes design basis evaluations of events that can occur during shutdown and a probabilistic assessment of plant risk at shutdown. The design of the AP1000 builds on the lessons-learned from the industry with regard to shutdown safety, including the guidance provided in NUREG-1449.

19E.8 Conclusion

This AP1000 Shutdown Evaluation provides a systematic evaluation of the AP1000 during shutdown operations. As demonstrated in this appendix, the AP1000 is designed to mitigate events that can occur during shutdown modes. In addition, the risk of core damage as a result of an accident that may occur during shutdown has been demonstrated to be acceptably low.

19E.9 References

1. Letter, Westinghouse to NRC, DCP/NRC1385, AP600 Emergency Response Guidelines.

2. NUREG-1449, “Shutdown and Low Power Operations at Commercial Nuclear Power Plants in the United States,” September 1993.

3. NRC Information Notice 92-54, “Level Instrumentation Inaccuracies Caused by Rapid Depressurization,” July 24, 1992.

4. Letter, Westinghouse to NRC, DCP/NRC0124, APWR-0452, “AP600 Vortex Mitigator Development Test for RCS Mid-loop Operation,” July 6, 1994.

5. NUREG-0897, Rev. 1, “Containment Emergency Sump Performance,” October 1985.

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6. Title 10, Code of Federal Regulations, Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants.”

7. NRC Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Revision 3, February 1976.

8. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, 1998 with 2000 Addenda.

9. Lewis, R. N., Huang, P., Behnke, D. H., Fittante, R. L., and Gelman, A., WCAP-10698-P-A (Proprietary) and WCAP-10750-A (Non-Proprietary), “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,” August 1987.

10. WCAP-14171, Revision 2 (Proprietary) and WCAP-14172, Revision 2 (Non-Proprietary), “WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident,” March 1998.

11. Not used.

12. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Revision 1, July 1981.

13. Title 10, Code of Federal Regulations, Part 50, Appendix G, “Fracture Toughness Requirements.”

14. Not used.

15. Title 10, Code of Federal Regulations, Part 50, (10 CFR 50.46).

16. NRC letter, SECY-93-190, “Regulatory Approach to Shutdown and Low-Power Operations,” July 12, 1993.

17. Title 10, Code of Federal Regulations, Part 50.36, “Technical Specifications.”

18. NUREG-1431, “Standard Technical Specifications – Westinghouse Plants,” April 1995.

19. APP-GW-GL-022, “AP1000 Probabilistic Risk Assessment,” Westinghouse Electric, LLC, Rev. 8, August 2004 [ML042230497].

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Table 19E.2-1
Evaluation of a Loss of RNS at Mid-Loop With No IRWST Injection

Time After Shutdown	Time After Loss of RNS to Boiling	Time After Loss of RNS to Empty Hot Leg	Time After Loss of RNS to Core Uncovery
28 hours	3 minutes	20 minutes	42 minutes

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Table 19E.4.1-1 (Sheet 1 of 2)
AP1000 Accidents Requiring Shutdown Evaluation or Analysis

Tier 2 Section	Titles	Evaluation or Analysis Required
15.1	Increase in Heat Removal from the Primary System	
15.1.1	Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	E
15.1.2	Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	E
15.1.3	Excessive Increase in Secondary Steam Flow	E
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	E
15.1.5	Steam System Piping Failure	E
15.1.6	Inadvertent Operation of the Passive Residual Heat Removal Heat Exchanger	E
15.2	Decrease in Heat Removal by the Secondary System	
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	E
15.2.2	Loss of External Electrical Load	E
15.2.3	Turbine Trip	E
15.2.4	Inadvertent Closure of Main Steam Isolation Valves	E
15.2.5	Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	E
15.2.6	Loss of ac Power to the Plant Auxiliaries	E
15.2.7	Loss of Normal Feedwater Flow	E
15.2.8	Feedwater System Pipe Break	E
15.3	Decrease in Reactor Coolant System Flow Rate	
15.3.1	Partial Loss of Forced Reactor Coolant Flow	E
15.3.2	Complete Loss of Forced Reactor Coolant Flow	E
15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	E
15.3.4	Reactor Coolant Pump Shaft Break	E
15.4	Reactivity and Power Distribution Anomalies	
15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition	E
15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	n/a
15.4.3	Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)	E
15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	E

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Table 19E.4.1-1 (Sheet 2 of 2)
AP1000 Accidents Requiring Shutdown Evaluation or Analysis

Tier 2 Section	Titles	Evaluation or Analysis Required
15.4.6	Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant	n/a
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	E
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	
15.5	Increase in Reactor Coolant Inventory	
15.5.1	Inadvertent Operation of the Core Makeup Tanks (CMT) During Power Operation	E
15.5.2	Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	E
15.6	Decrease in Reactor Coolant Inventory	
15.6.1	Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	E
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	E
15.6.3	Steam Generator Tube Rupture	E
15.6.5	Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	E/A
15.7	Radioactive Release From a Subsystem or Component	E

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Table 19E.4.8-1
Double-Ended Cold-Leg Guillotine Break – Sequence of Events

Event	Time (seconds)
Break open	0.0
“S” signal receipt (including delay)	4.2
RCPs start to coast down	9.5
Sustained CMT draindown begins (CMT connected to intact cold leg)	~150
Sustained CMT draindown begins (CMT connected to broken cold leg)	~200
Lower plenum refilled	~250

Table 19E.4.8-2
Loss of Normal Residual Heat Removal System Cooling in Mode 4 and Mode 5 With Reactor Coolant System Intact – Sequence of Events

Event	Automatic Actuation Time After Loss of RNS		Manual Actuation Time After Loss of RNS	
	(seconds)	(hours)	(seconds)	(hours)
Loss of RNS cooling	0	0	0	0
RNS relief valve flow starts	742	0.21	742	0.21
CMT and PRHR actuated	8619	2.39	1800	0.50
RNS relief valve flow terminated	9141	2.54	14,024	3.90
ADS Stage 1 flow starts	9078	2.52	13,830	3.84
ADS Stage 2 flow starts	9126	2.54	13,878	3.86
ADS Stage 3 flow starts	9246	2.57	13,998	3.89
ADS Stage 4 flow starts	9688	2.69	14,353	3.99
IRWST injection starts	10,018	2.78	14,666	4.07

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Table 19E.4.8-3
Loss of Normal Residual Heat Removal System Cooling in Mode 5 With Reactor Coolant System Open – Sequence of Events

Event	Time After Loss of RNS	
	(seconds)	(hours)
Loss of RNS cooling	0	0
Hot leg empty	4554	1.27
ADS Stage 4 flow initiated	4674	1.30
IRWST injection starts	5029	1.40

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Tables 19E.4.8-4–19E.4.8-5 Not Used

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Table 19E.4.10-1
Sequence of Events Following a Loss of ac Power
Flow with Condensate from the Containment Shell
Being Returned to the IRWST

Event	Time (seconds)
Feedwater is Lost	10.0
Low-2 Steam Generator Narrow Range Water Level Reactor Trip Setpoint Reached	61.3
Rods Begin to Drop	63.3
Turbine Trip	68.3
Loss of Offsite ac Power, Reactor Coolant Pumps Begin to Coast Down	71.3
Low-2 Steam Generator Wide Range Water Level Reached	206.6
PRHR HX Actuation on Low-2 Steam Generator Wide Range Water Level	218.6
Low-2 T_{cold} Setpoint Reached	2,952
Steam Line Isolation on Low-2 T_{cold} Signal	2,964
CMTs Actuated on Low-2 T_{cold} Signal	2,964
IRWST Reaches Saturation Temperature	15,520
Heat Extracted by PRHR HX Matches Core Decay Heat	49,000
Cold Leg Temperature Reaches 420°F (loop with PRHR)	56,530
Core Average Temperature Reaches 420°F (loop with PRHR)	128,260

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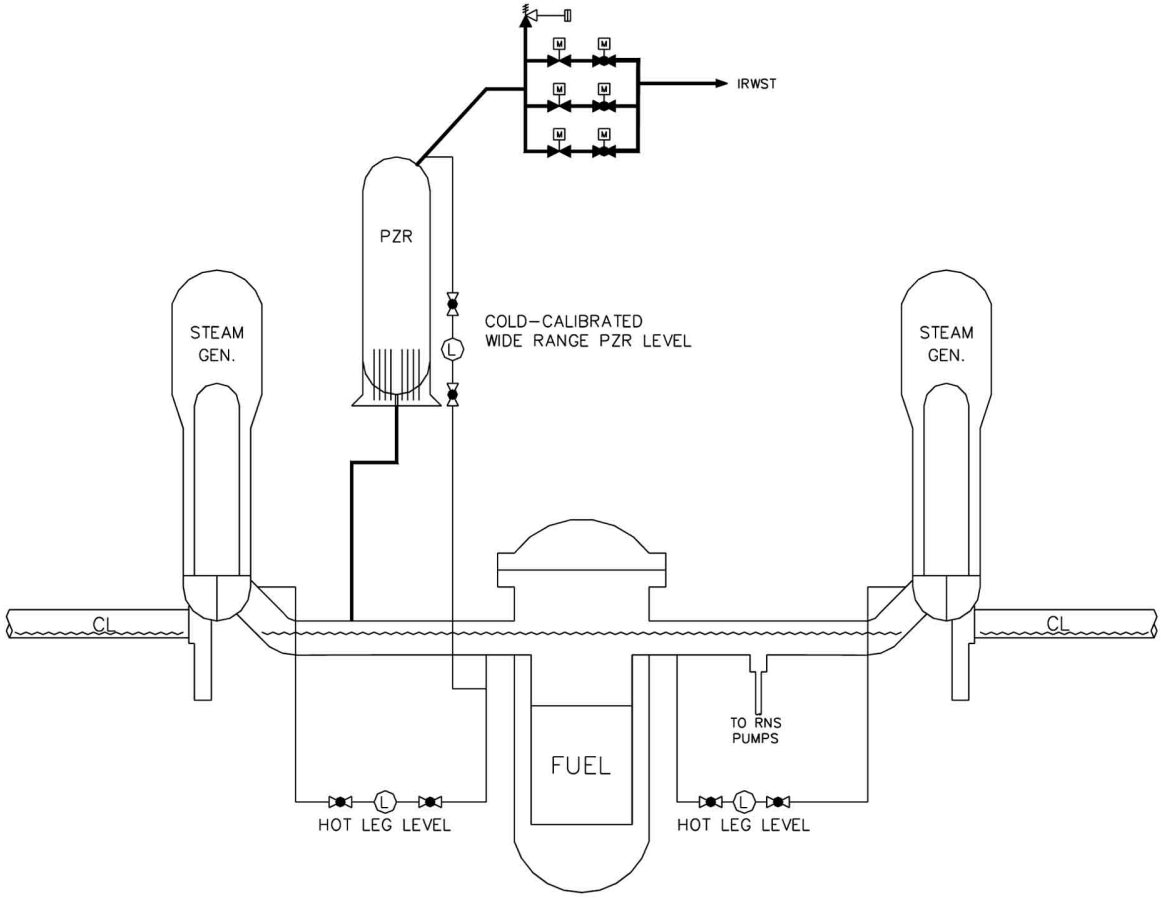

Figure 19E.2-1
Reactor Coolant System Level Instruments Used During Shutdown

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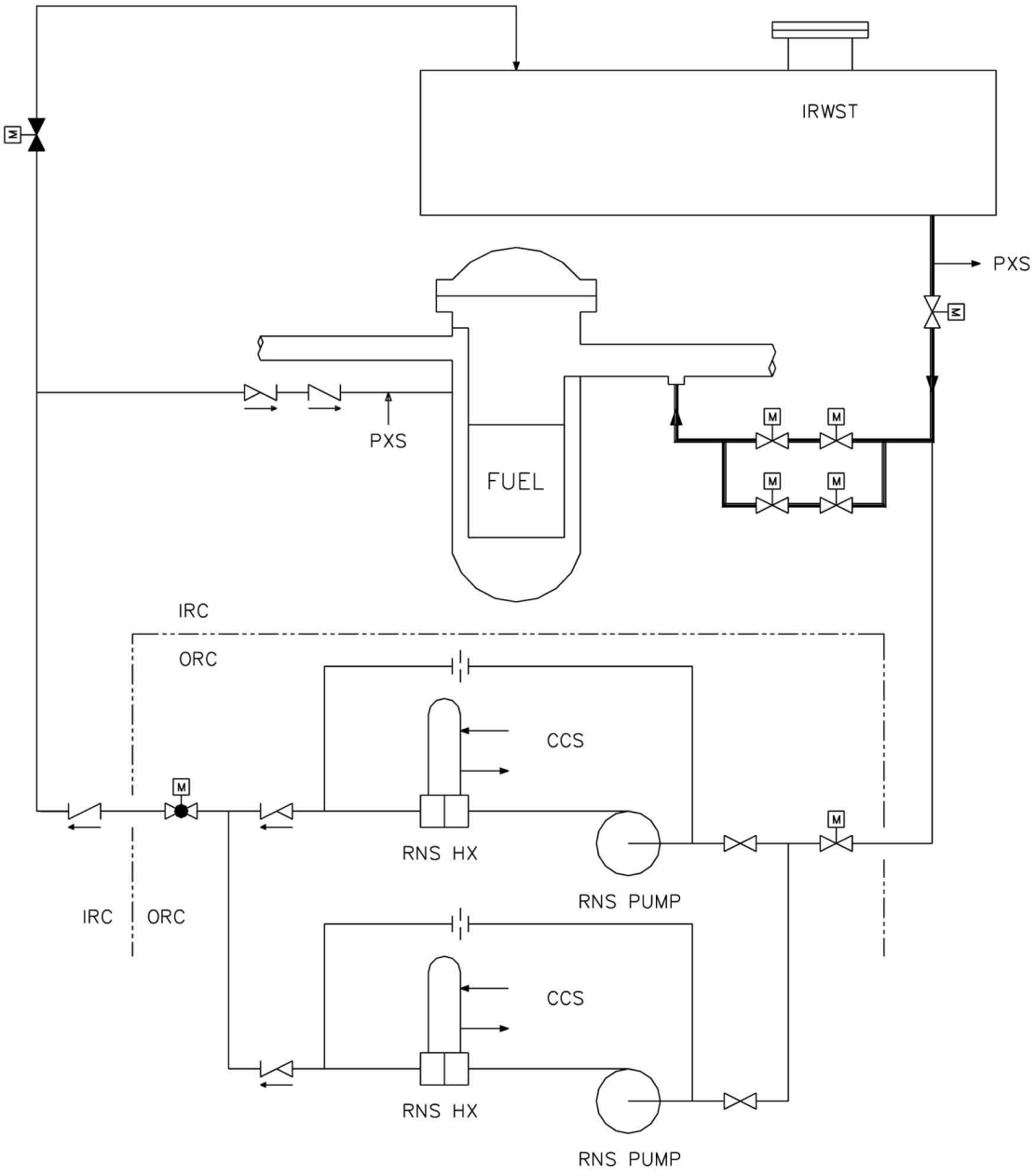

Figure 19E.2-2
IRWST Injection Flow Path

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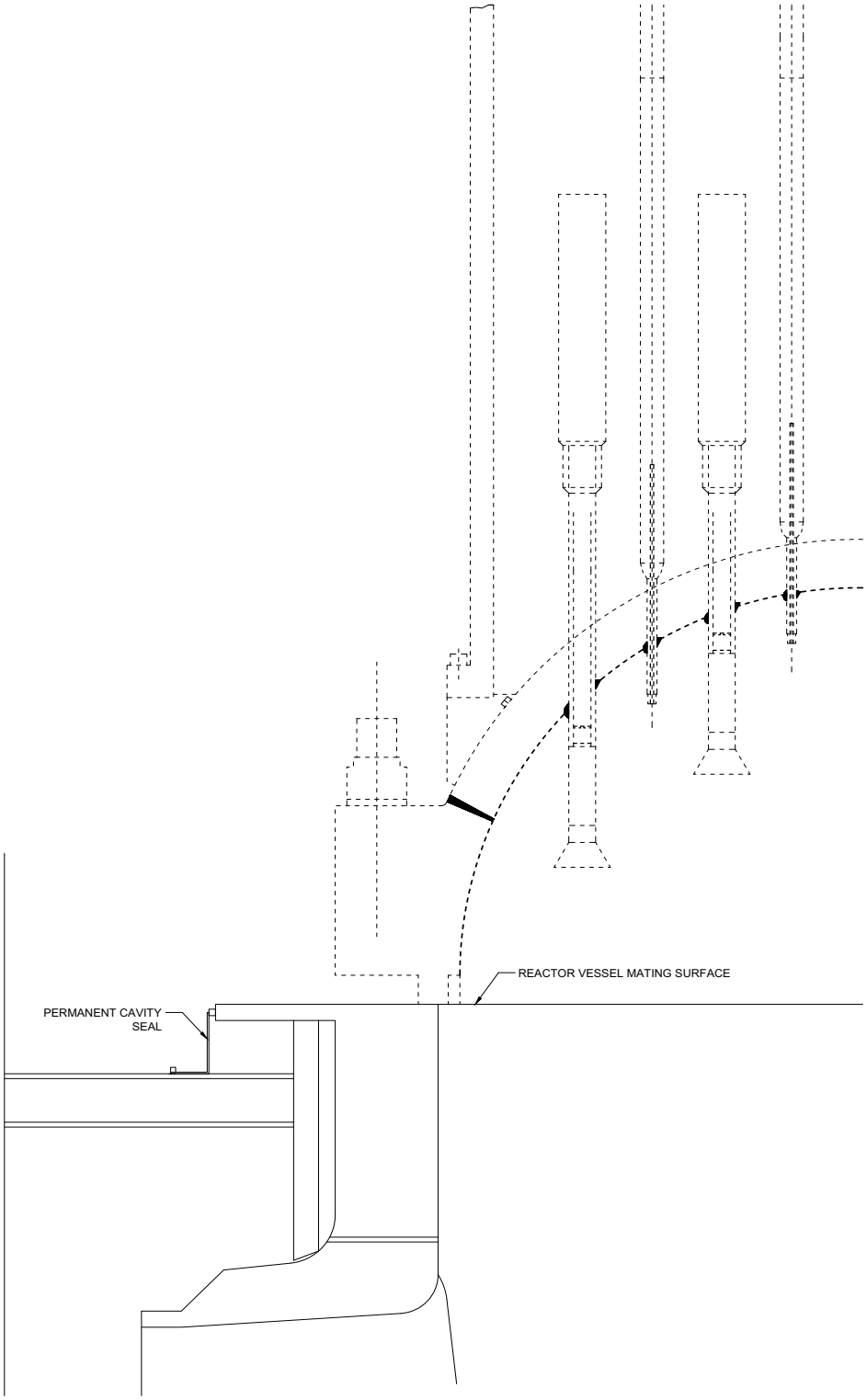

Figure 19E.2-3
AP1000 Permanent Cavity Seal Ring

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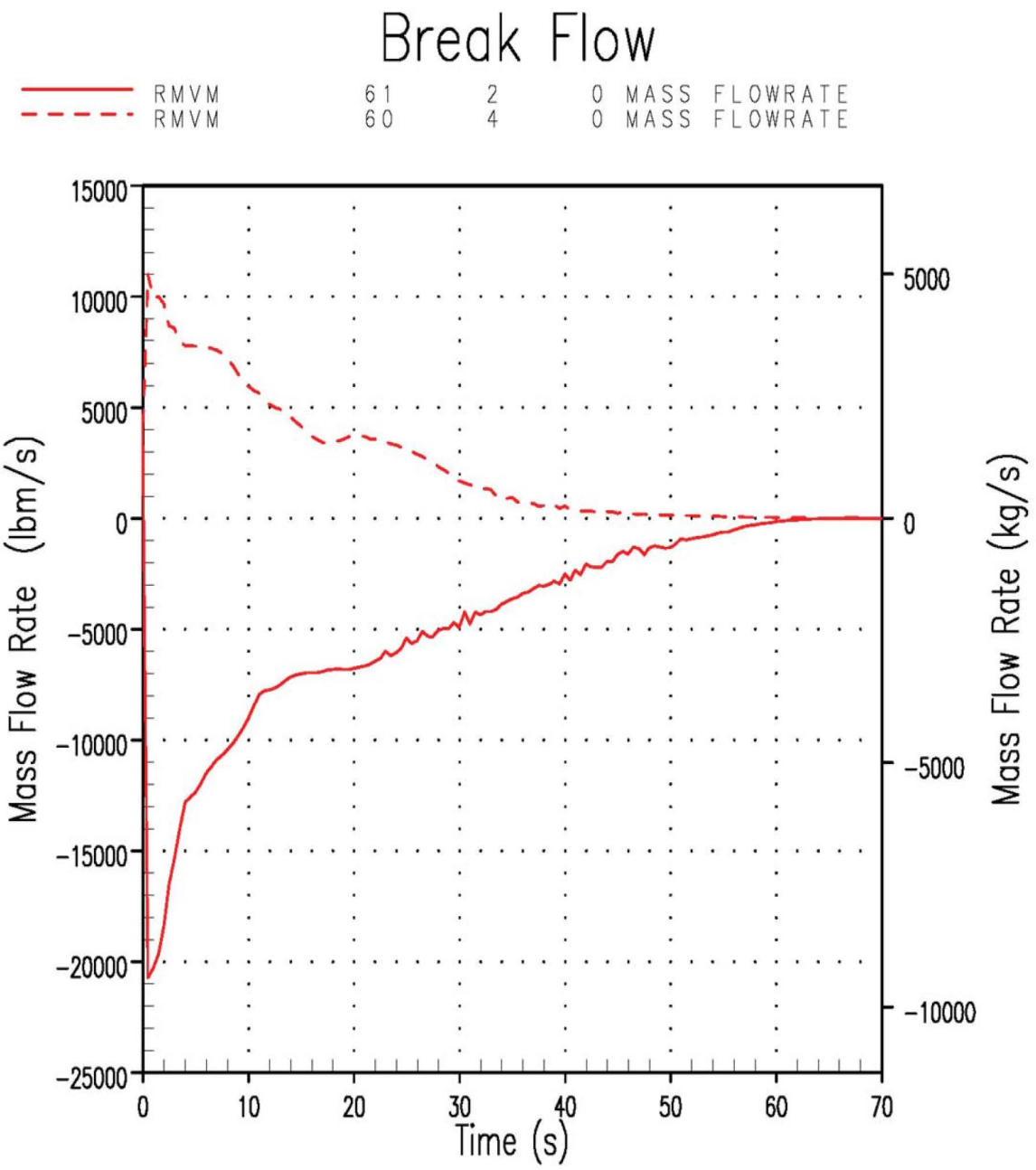

Figure 19E.4.8-1
Mode 3 DECLG Break, Break Flow Rates, Vessel and RCP Sides

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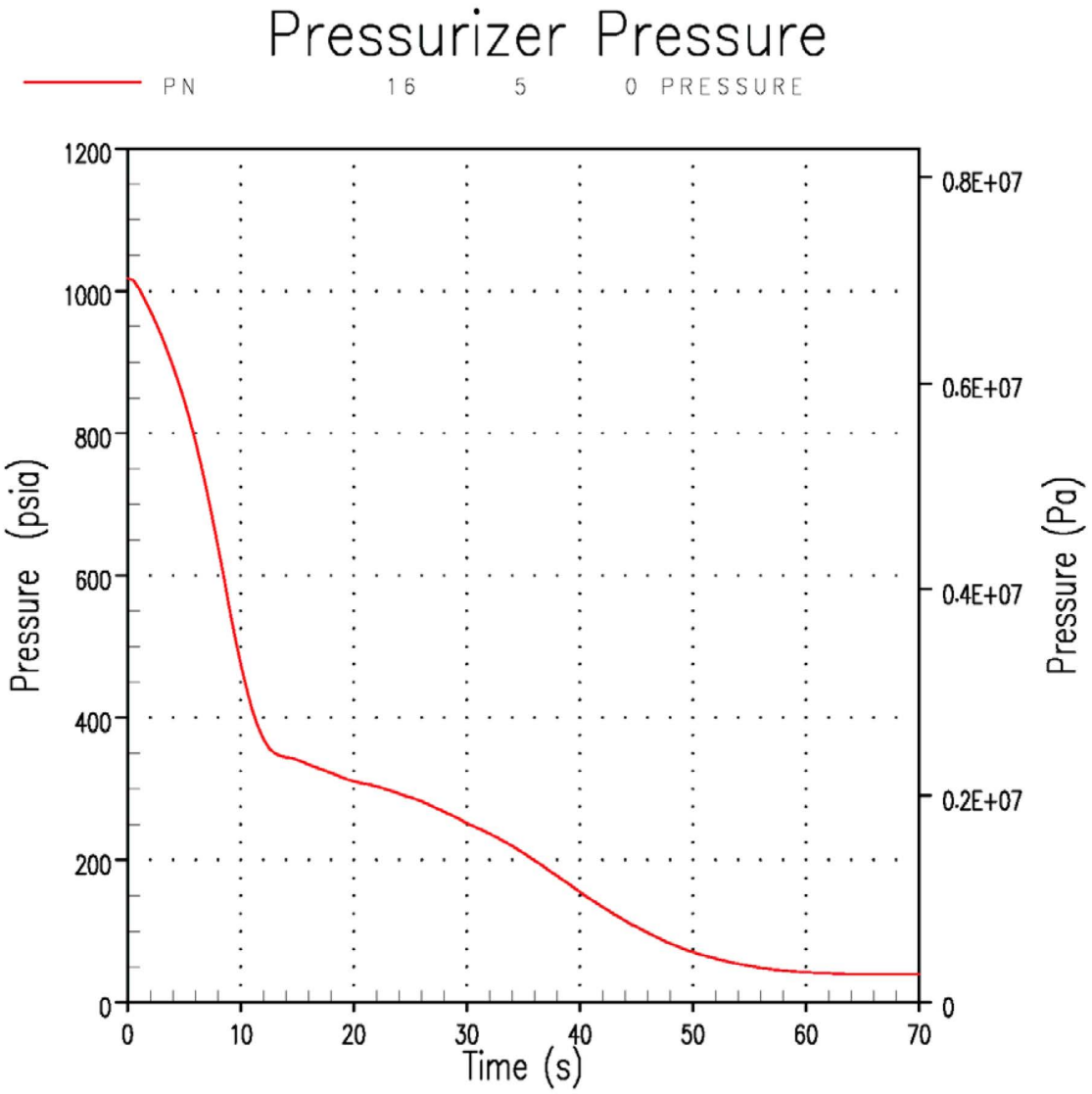

Figure 19E.4.8-2
Mode 3 DECLG Break, Pressurizer Pressure

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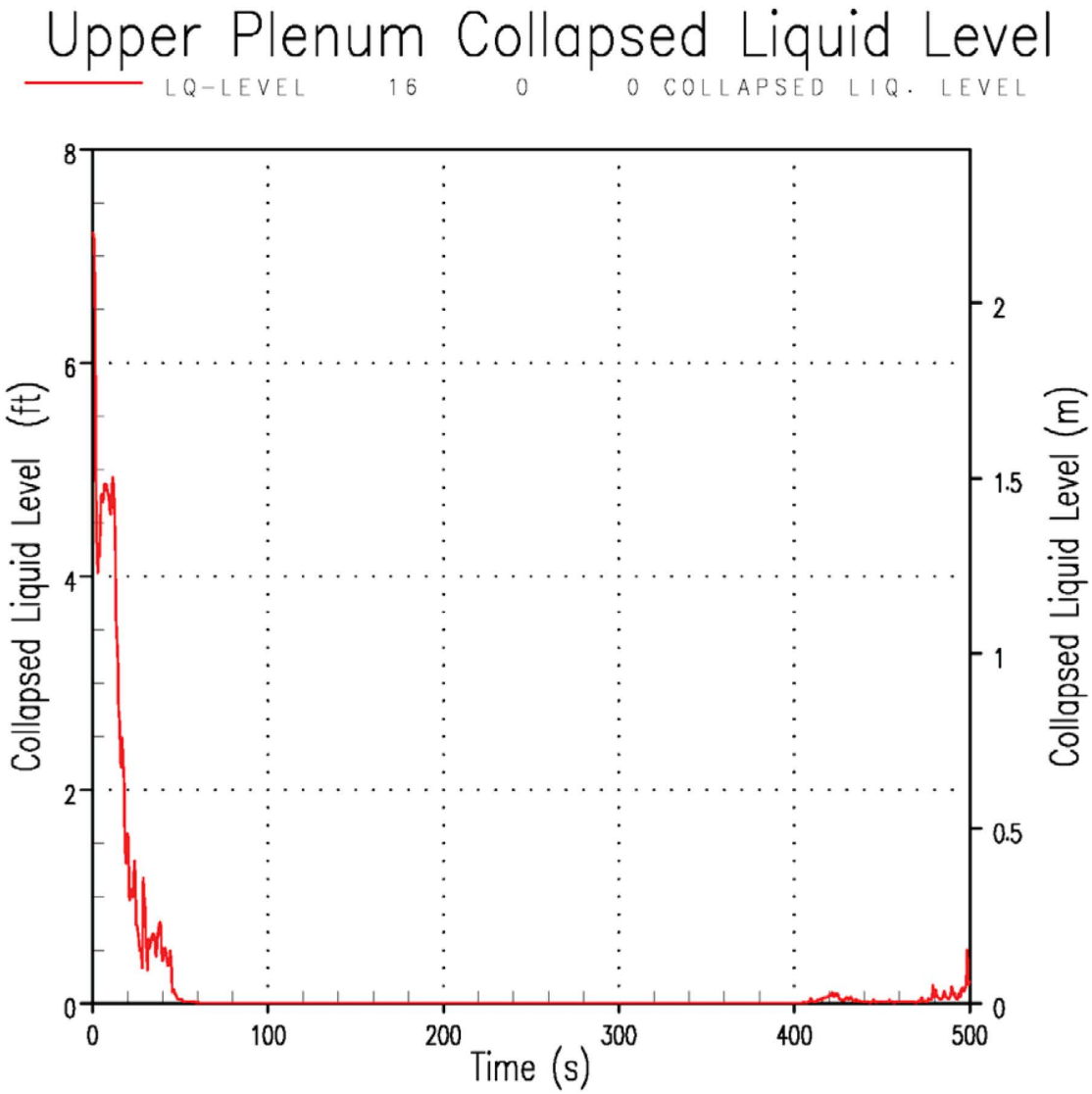

Figure 19E.4.8-3
Mode 3 DECLG Break, Upper Plenum Collapsed Liquid Level

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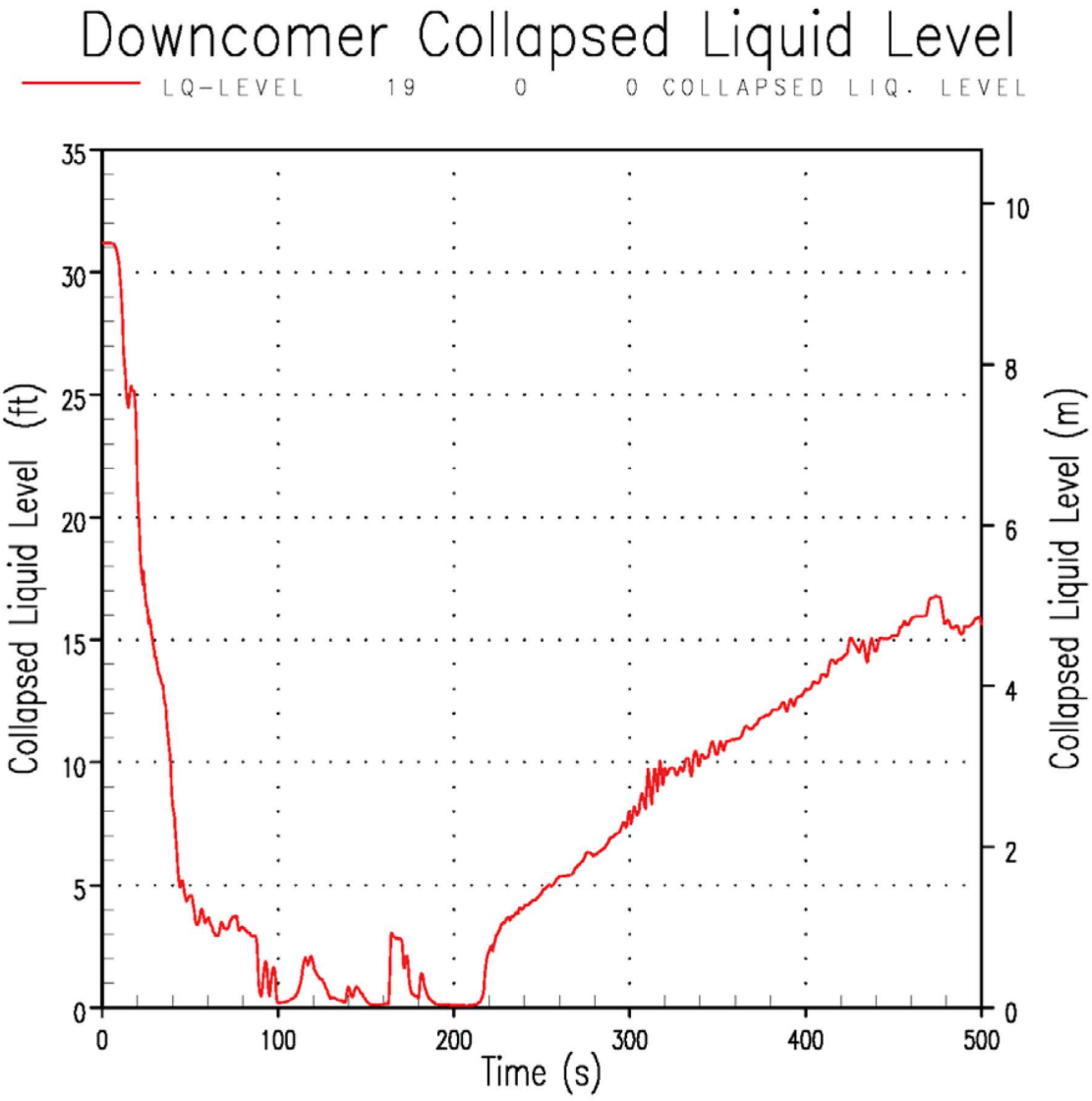

Figure 19E.4.8-4
Mode 3 DECLG Break, Downcomer Collapsed Liquid Level

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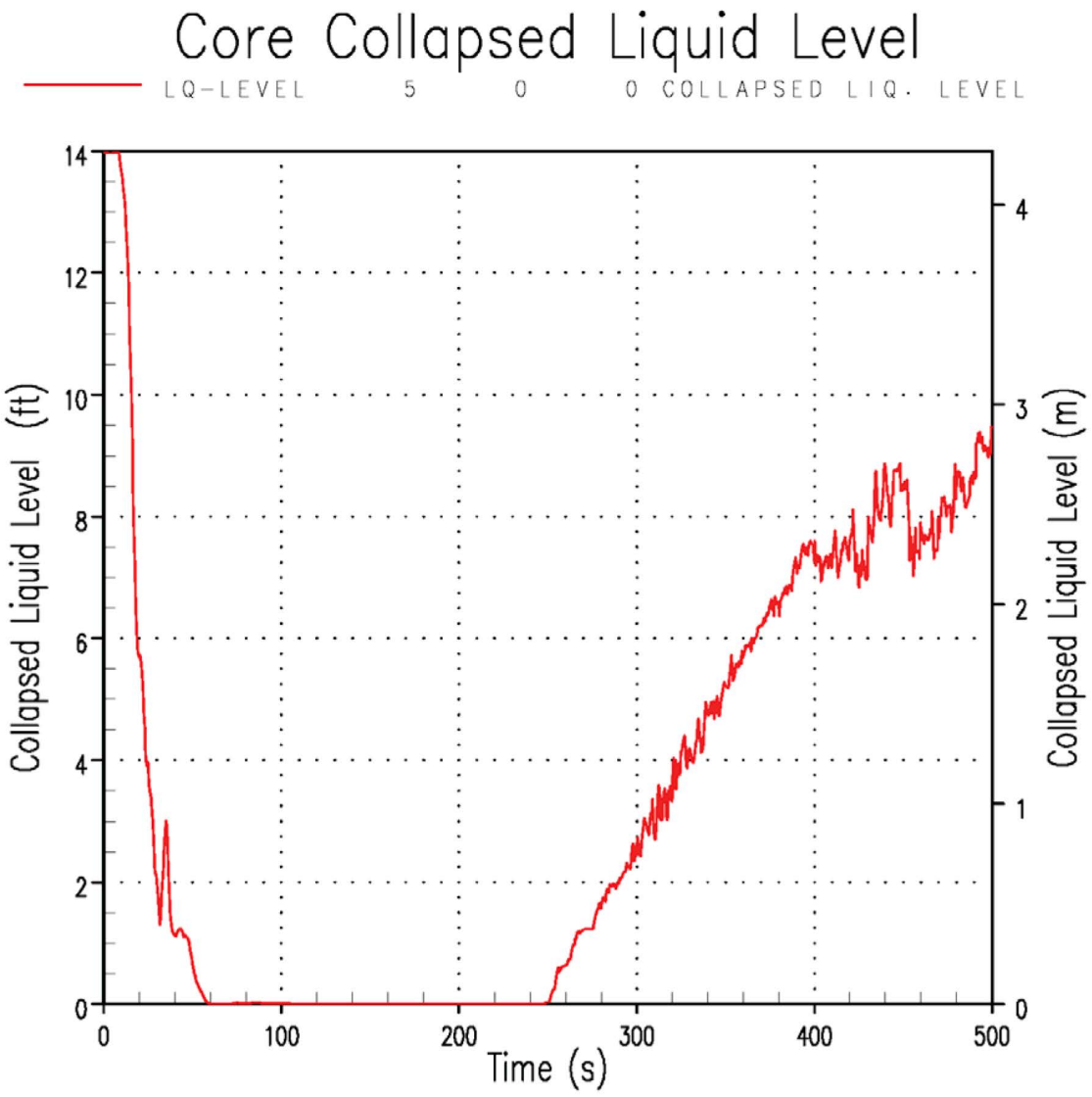

Figure 19E.4.8-5
Mode 3 DECLG Break, Core Collapsed Liquid Level

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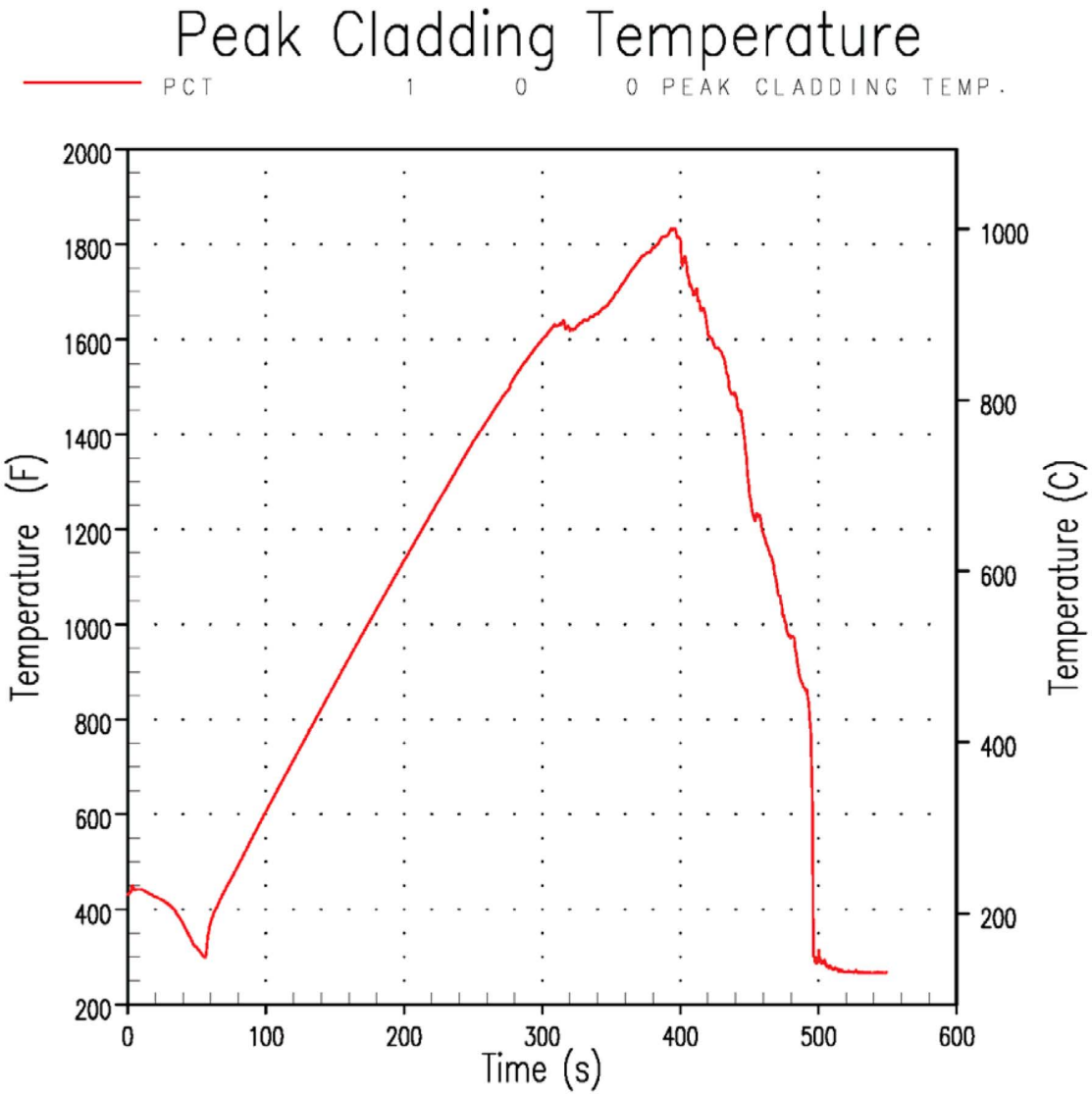

Figure 19E.4.8-6(a)
Mode 3 DECLG Break, Peak Cladding Temperature

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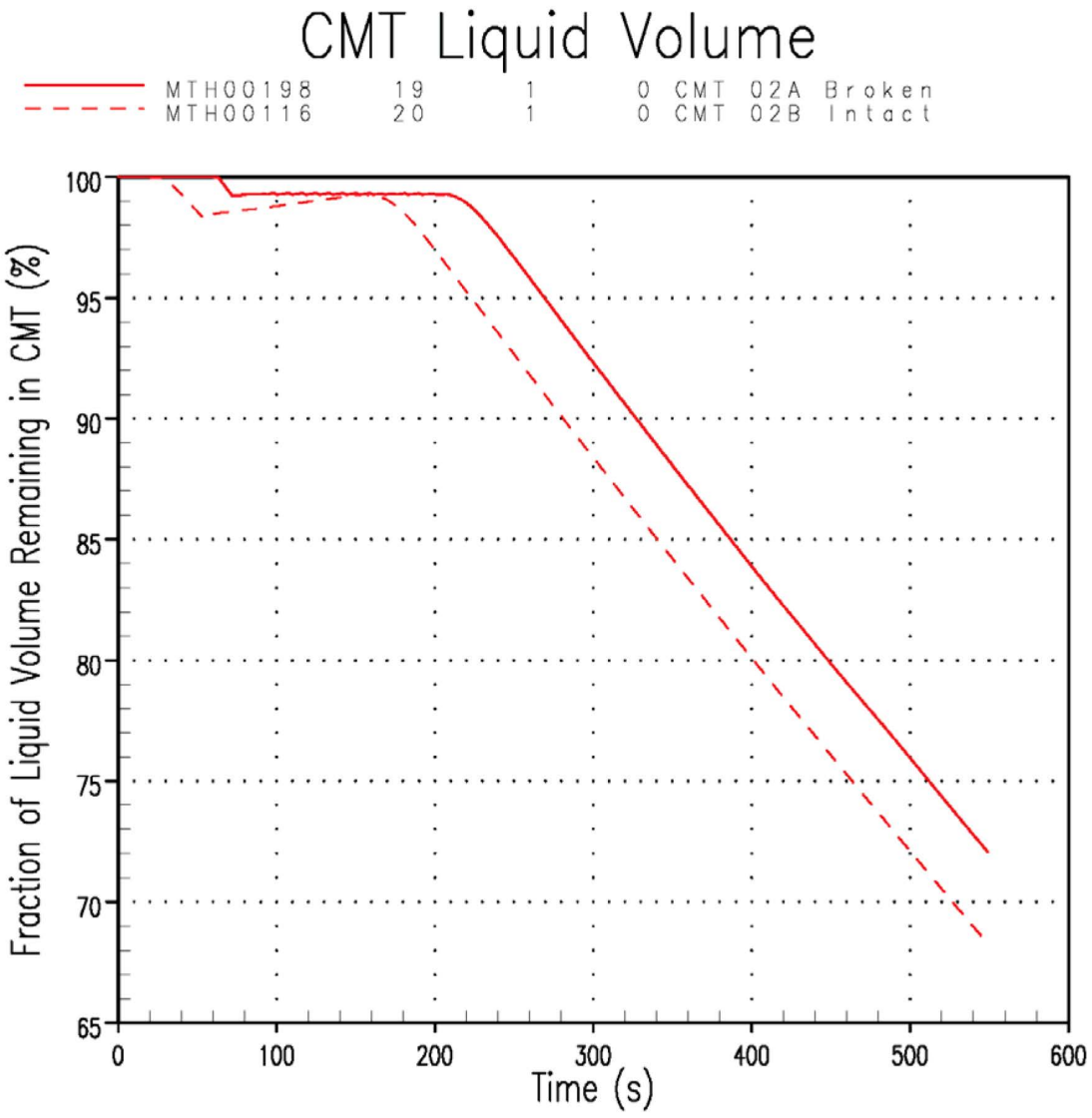

Figure 19E.4.8-6(b)
Mode 3 DECLG Break, CMT Liquid Volume

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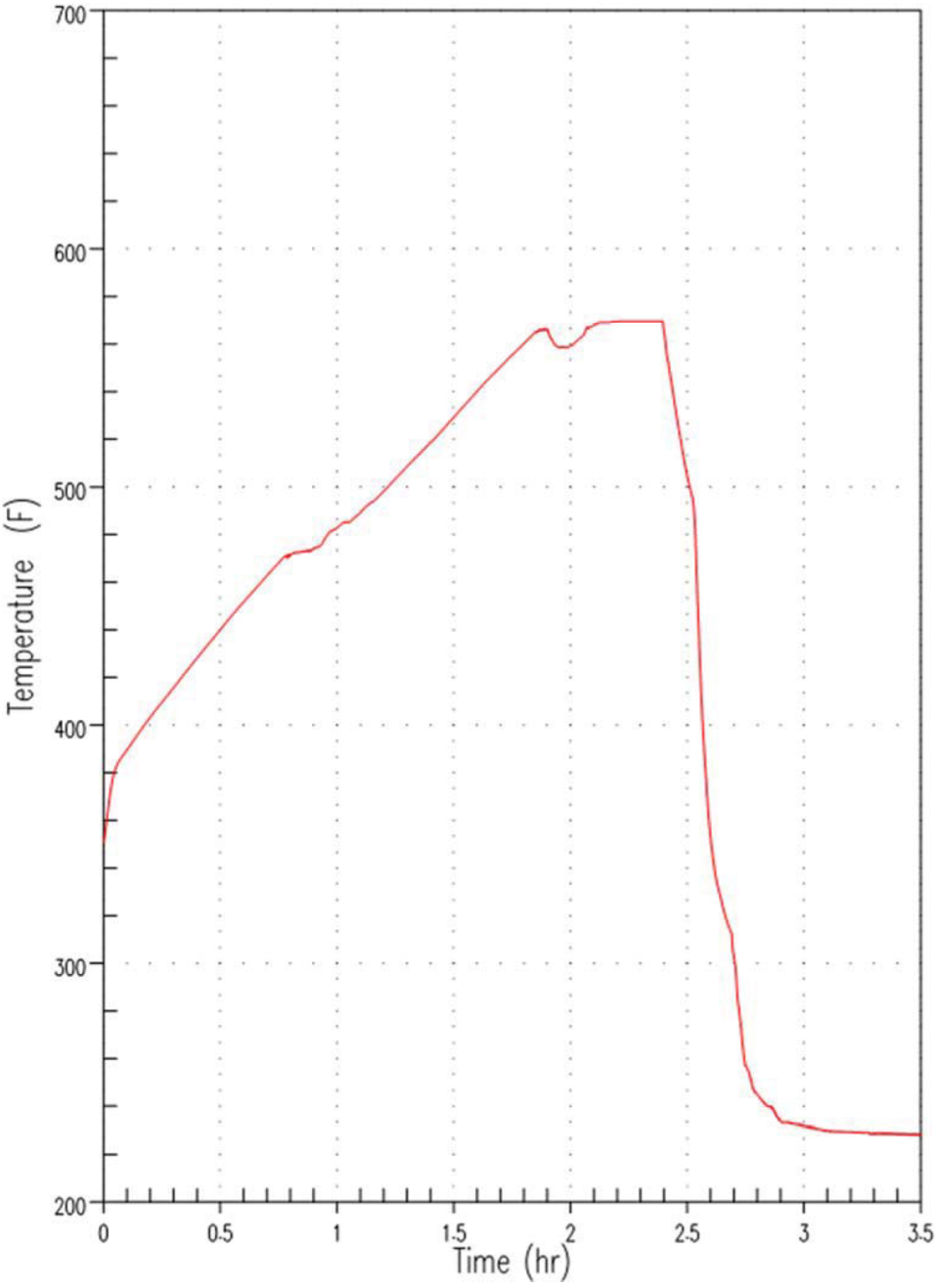

Figure 19E.4.8-7
Core Outlet Temperature, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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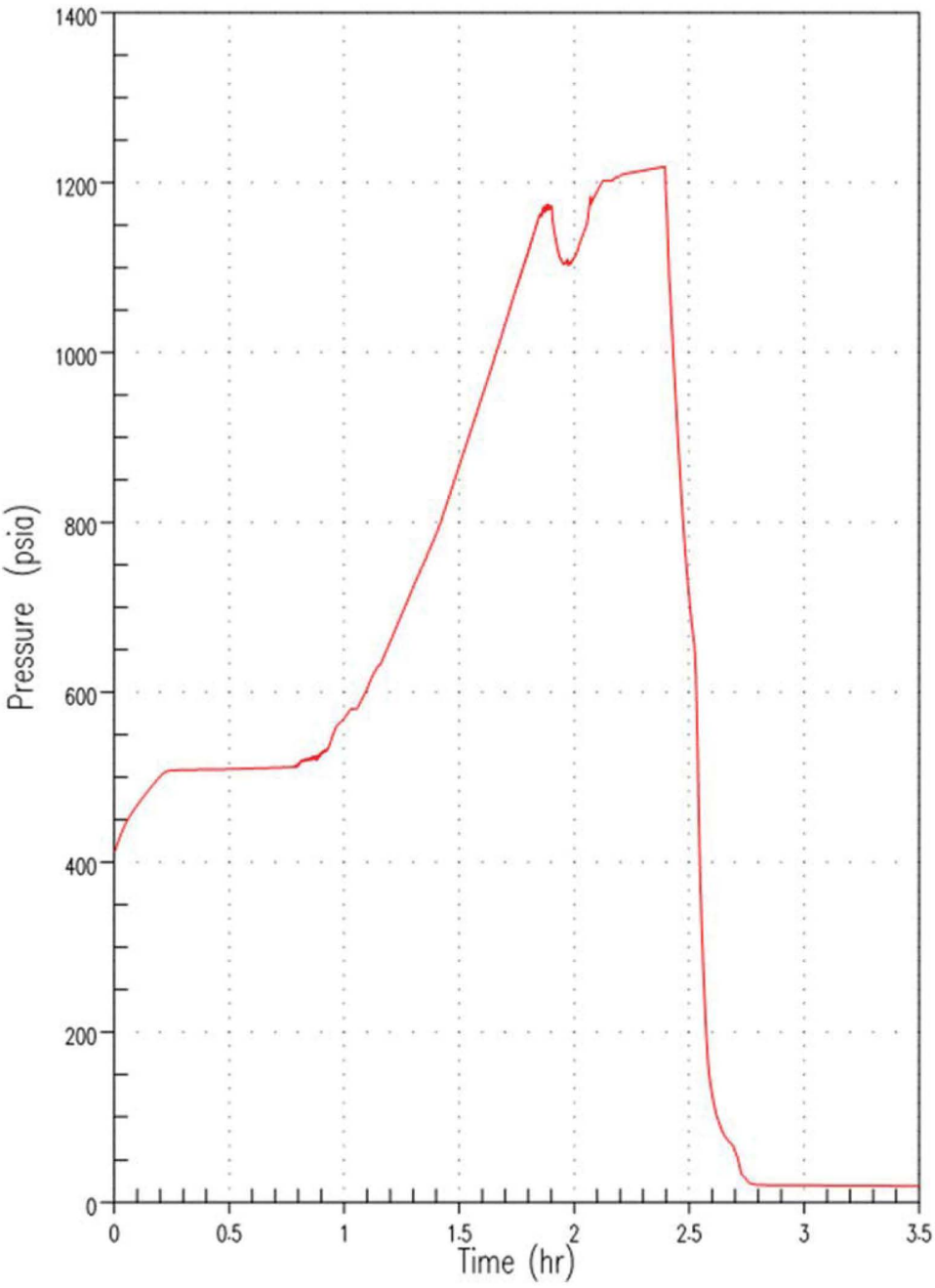

Figure 19E.4.8-8
Pressurizer Pressure, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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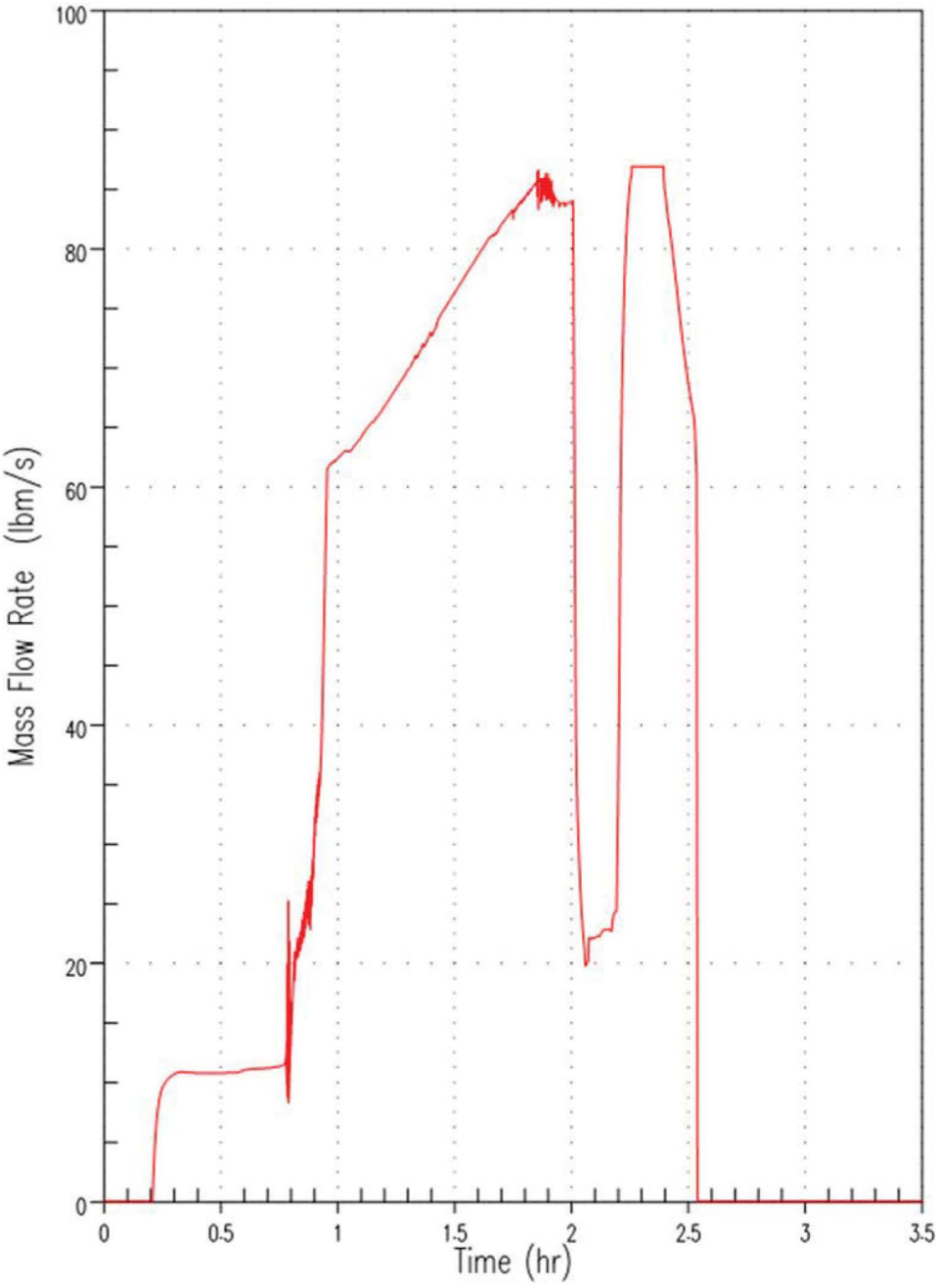

Figure 19E.4.8-9
RNS Relief Valve Flow, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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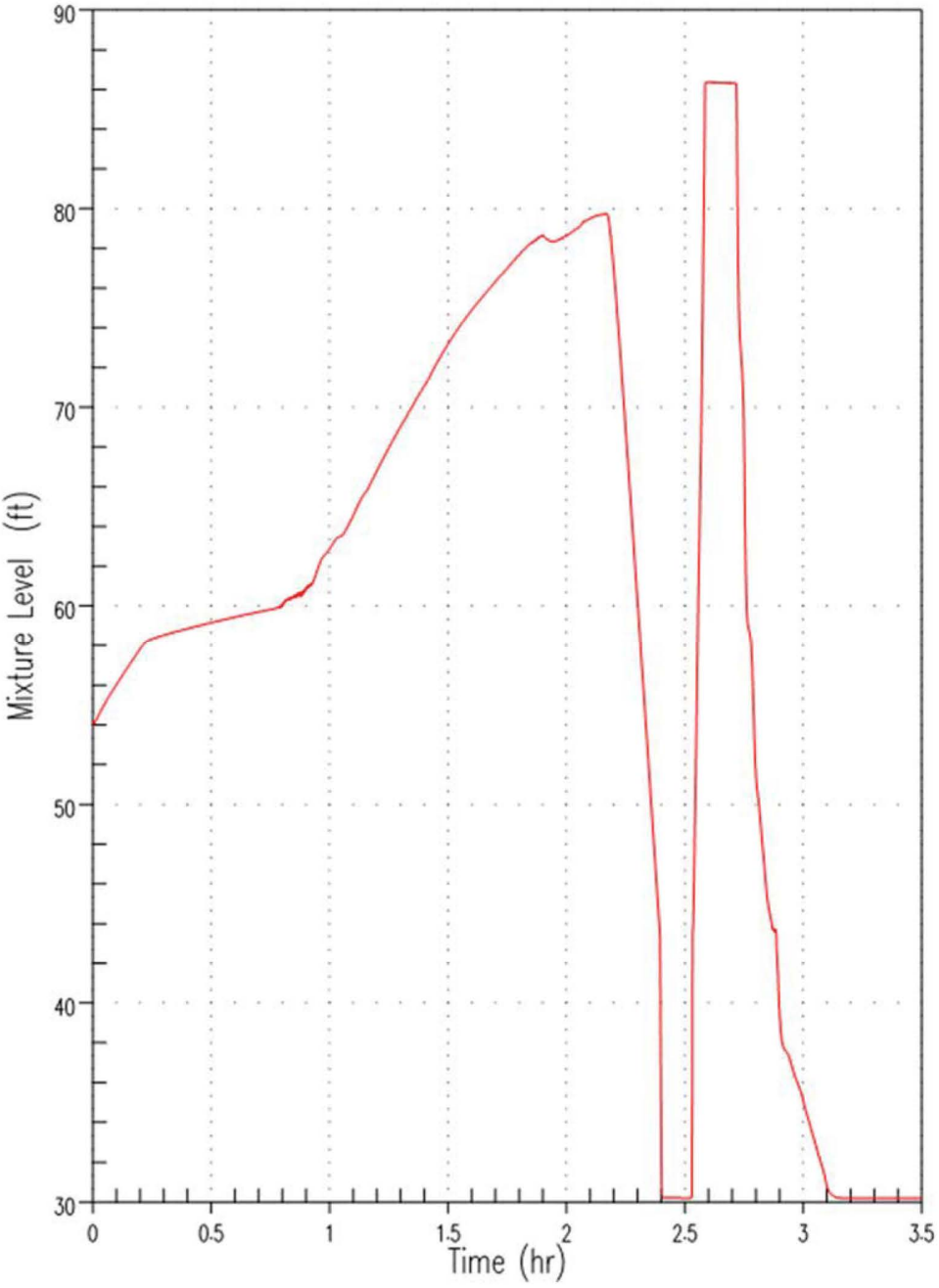

Figure 19E.4.8-10
Pressurizer Mixture Level, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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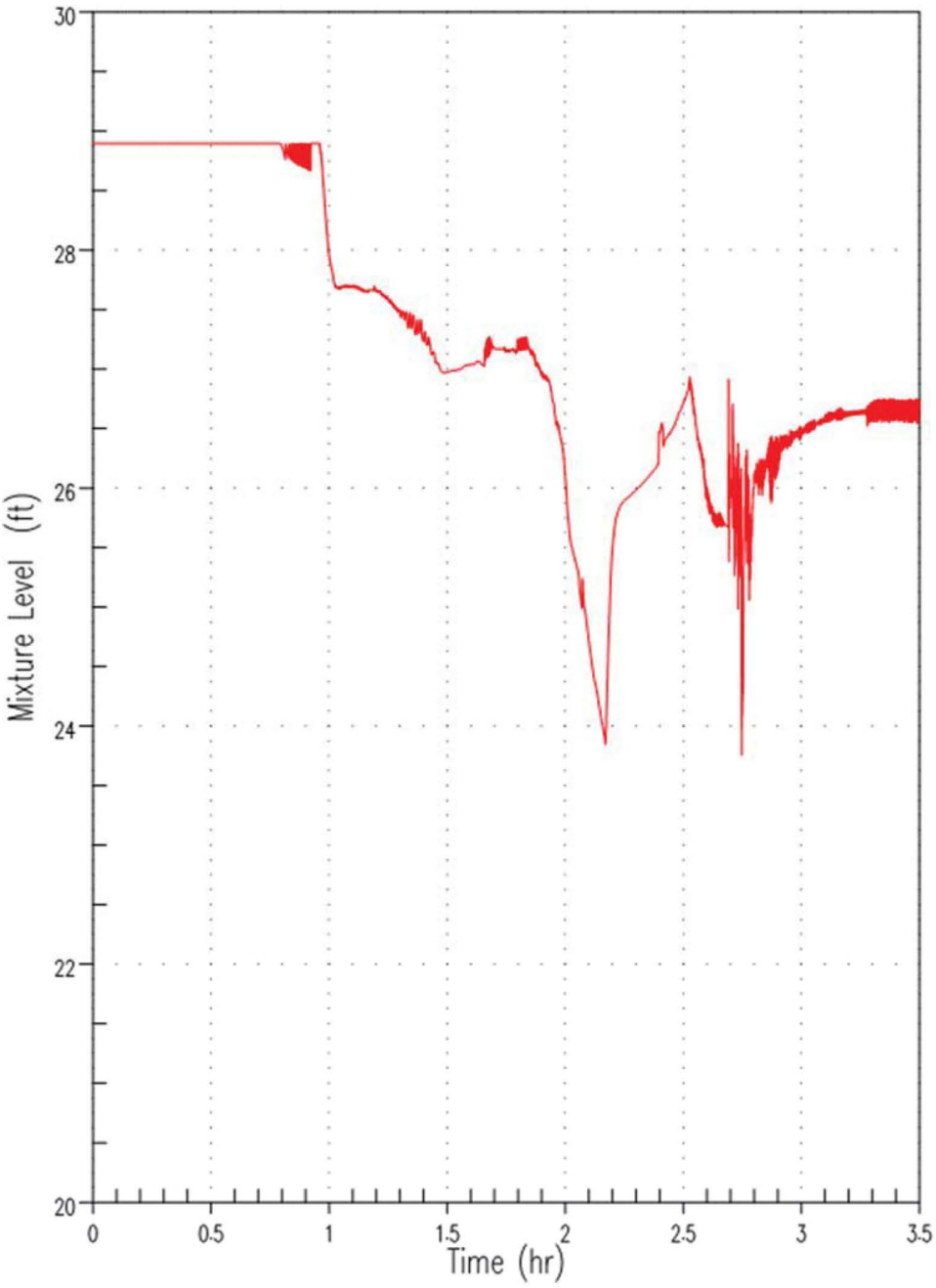

Figure 19E.4.8-11
Core Stack Mixture Level, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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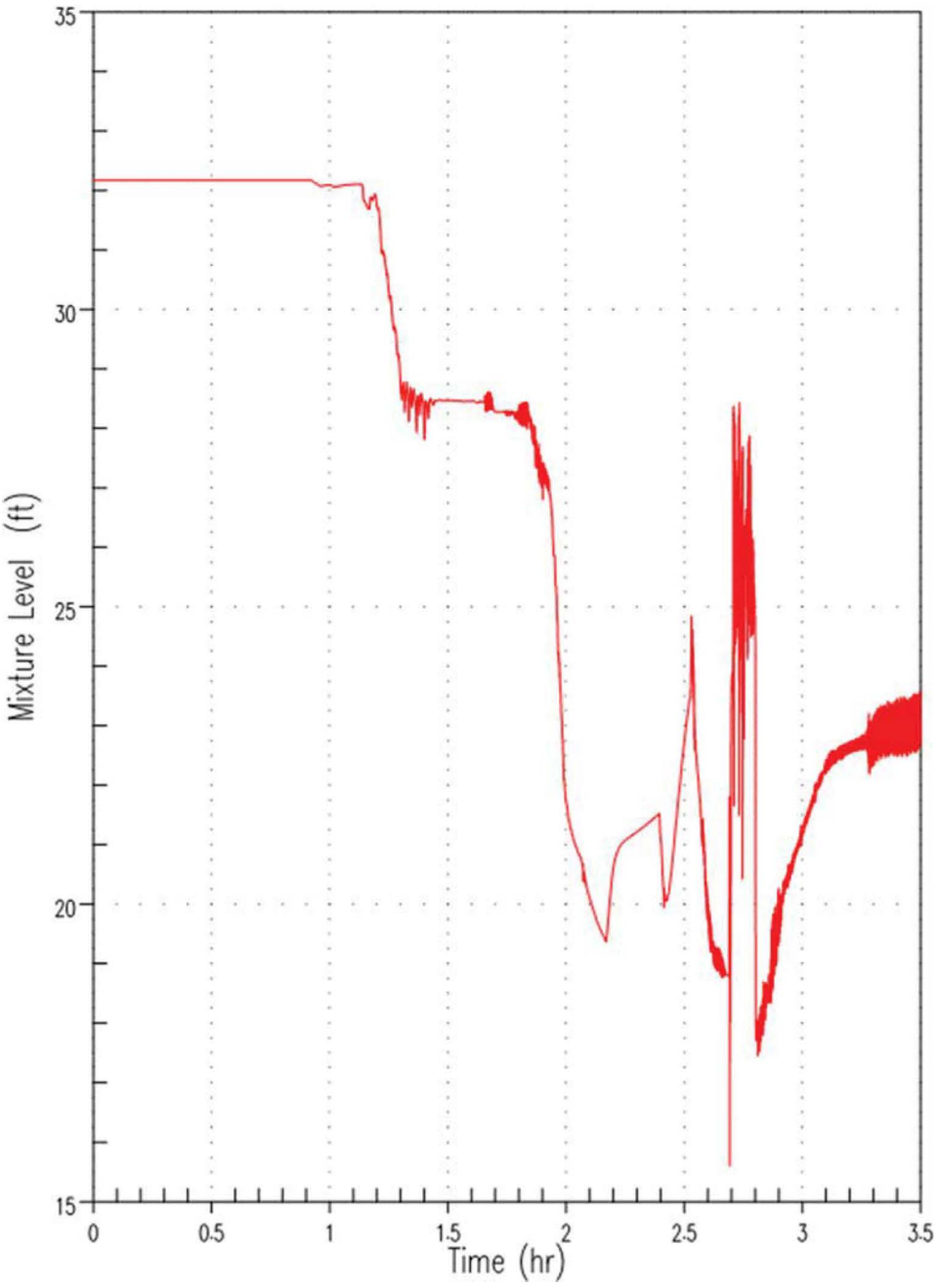

Figure 19E.4.8-12
Downcomer Mixture Level, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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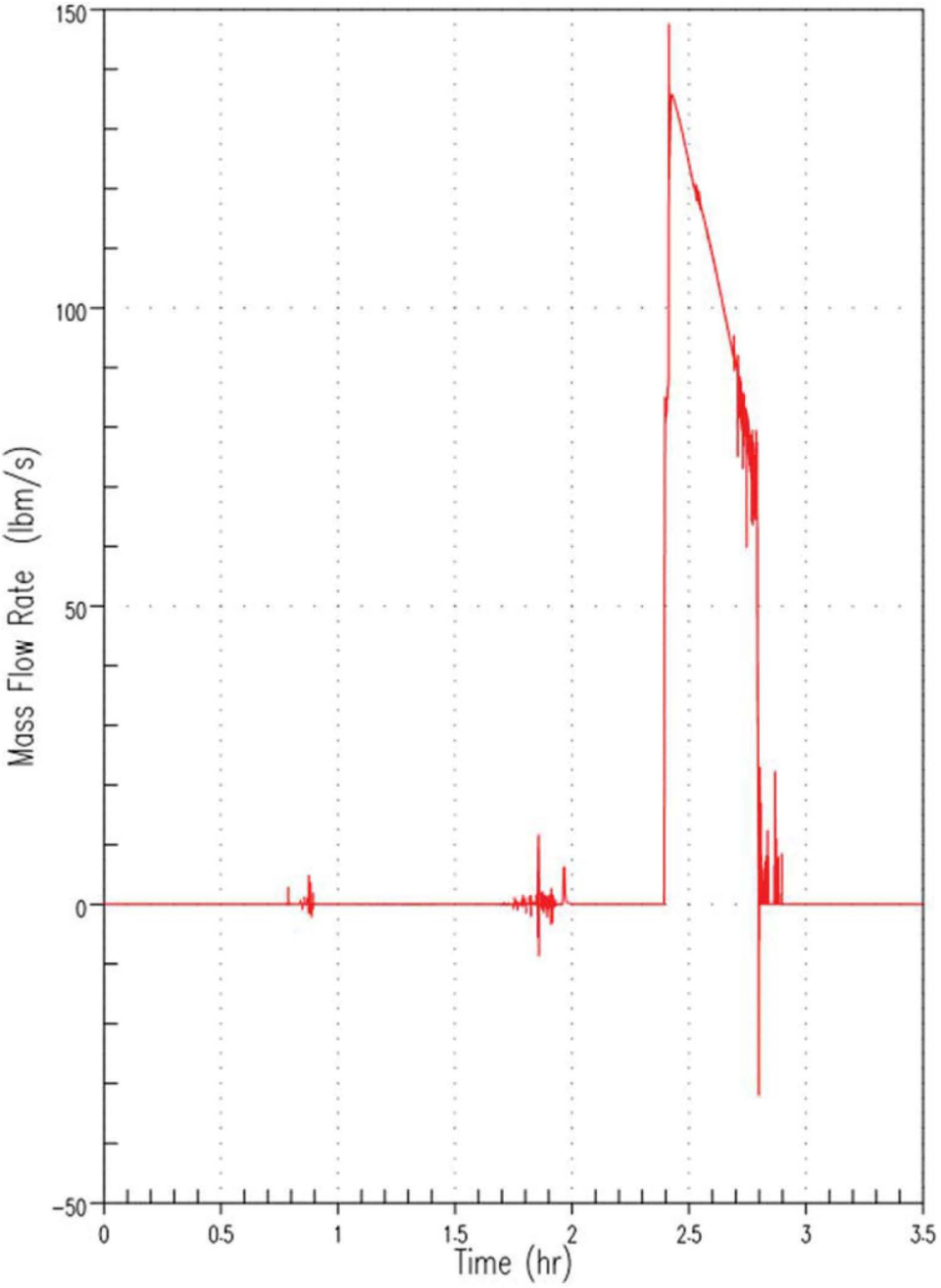

Figure 19E.4.8-13
CMT to DVI Flow, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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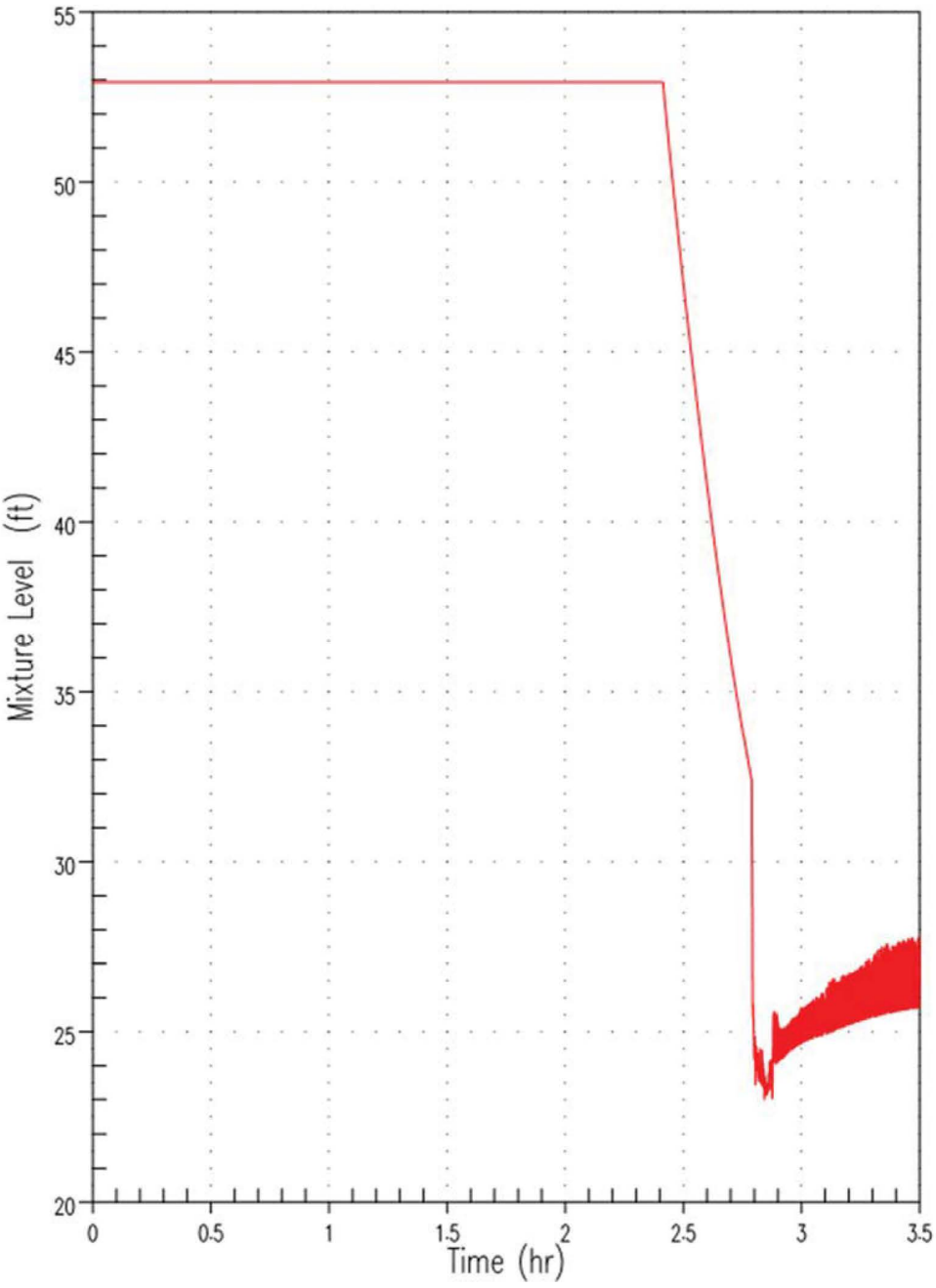

Figure 19E.4.8-14
CMT Mixture Level, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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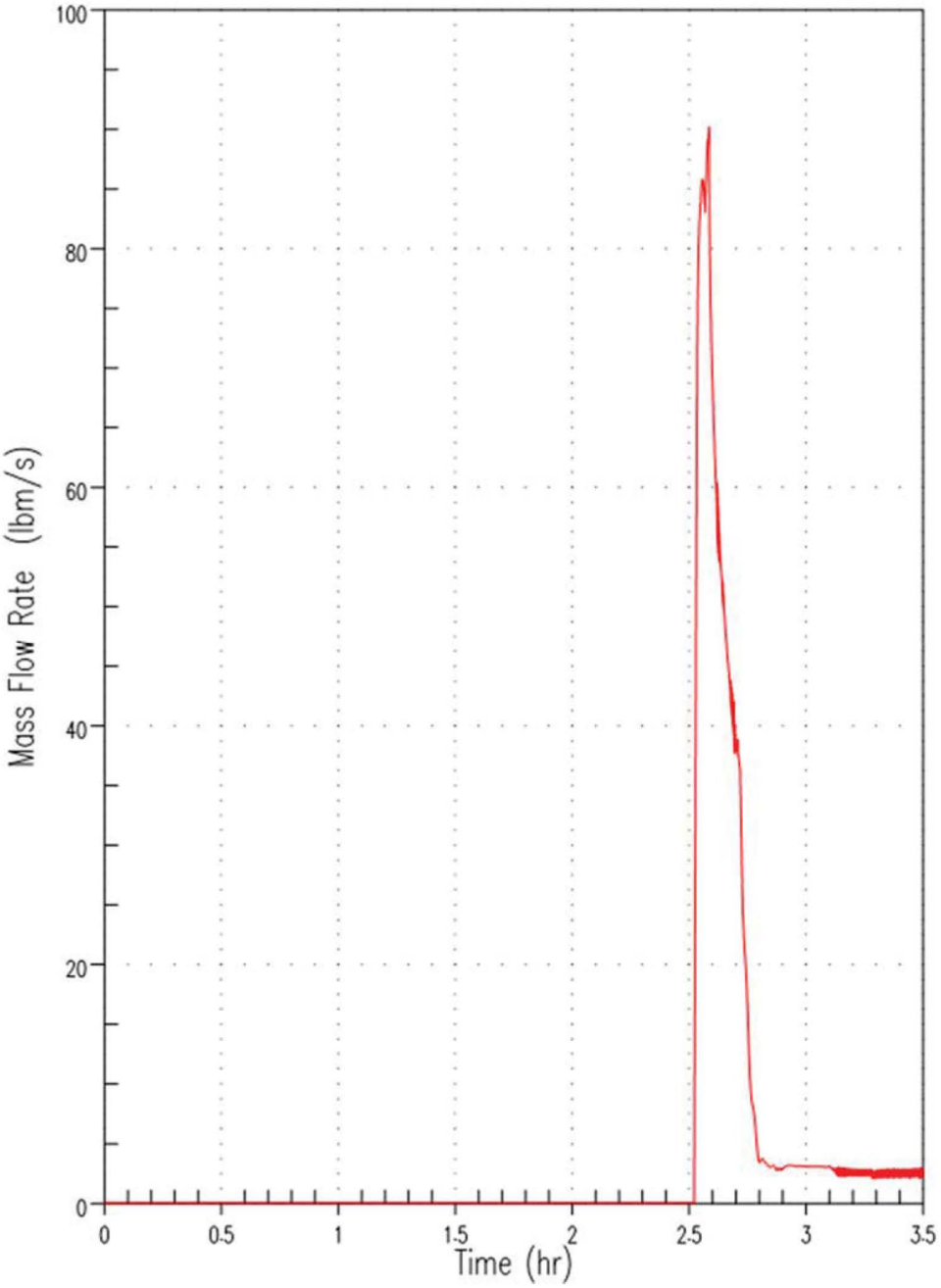

Figure 19E.4.8-15
ADS Stages 1-3 Vapor Flow, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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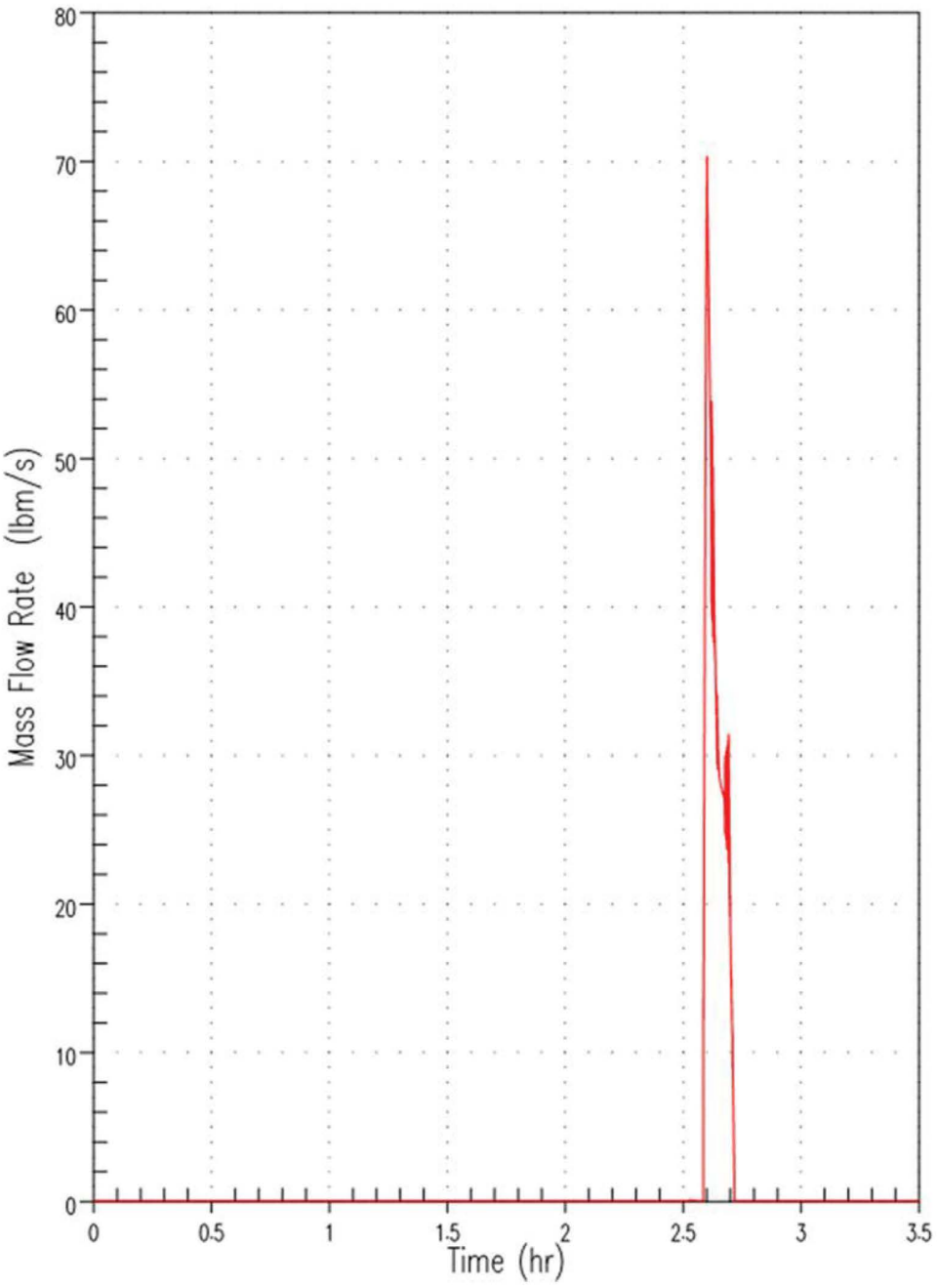

Figure 19E.4.8-16
ADS Stages 1-3 Liquid Flow, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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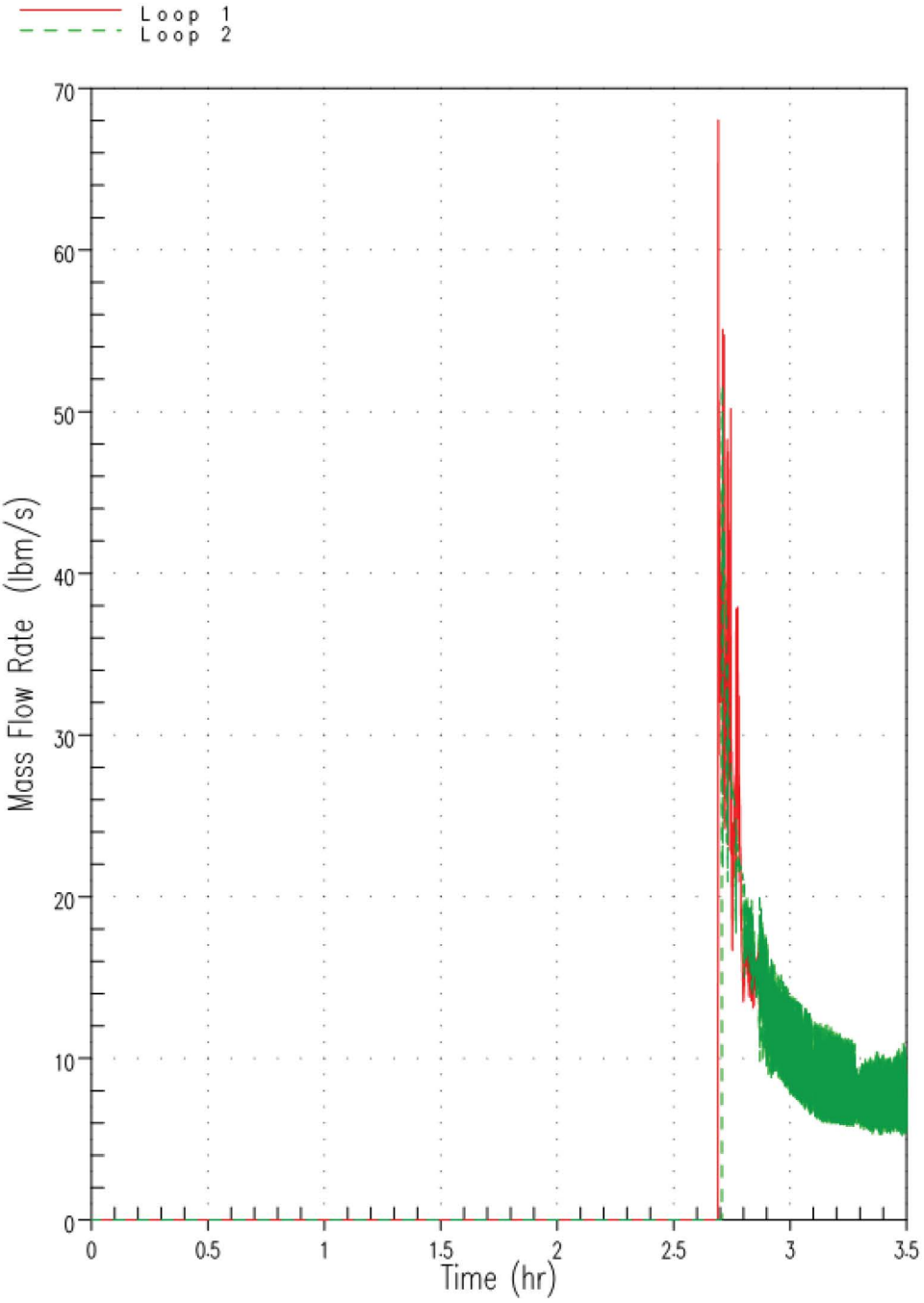

Figure 19E.4.8-17
ADS Stage 4 Vapor Flow, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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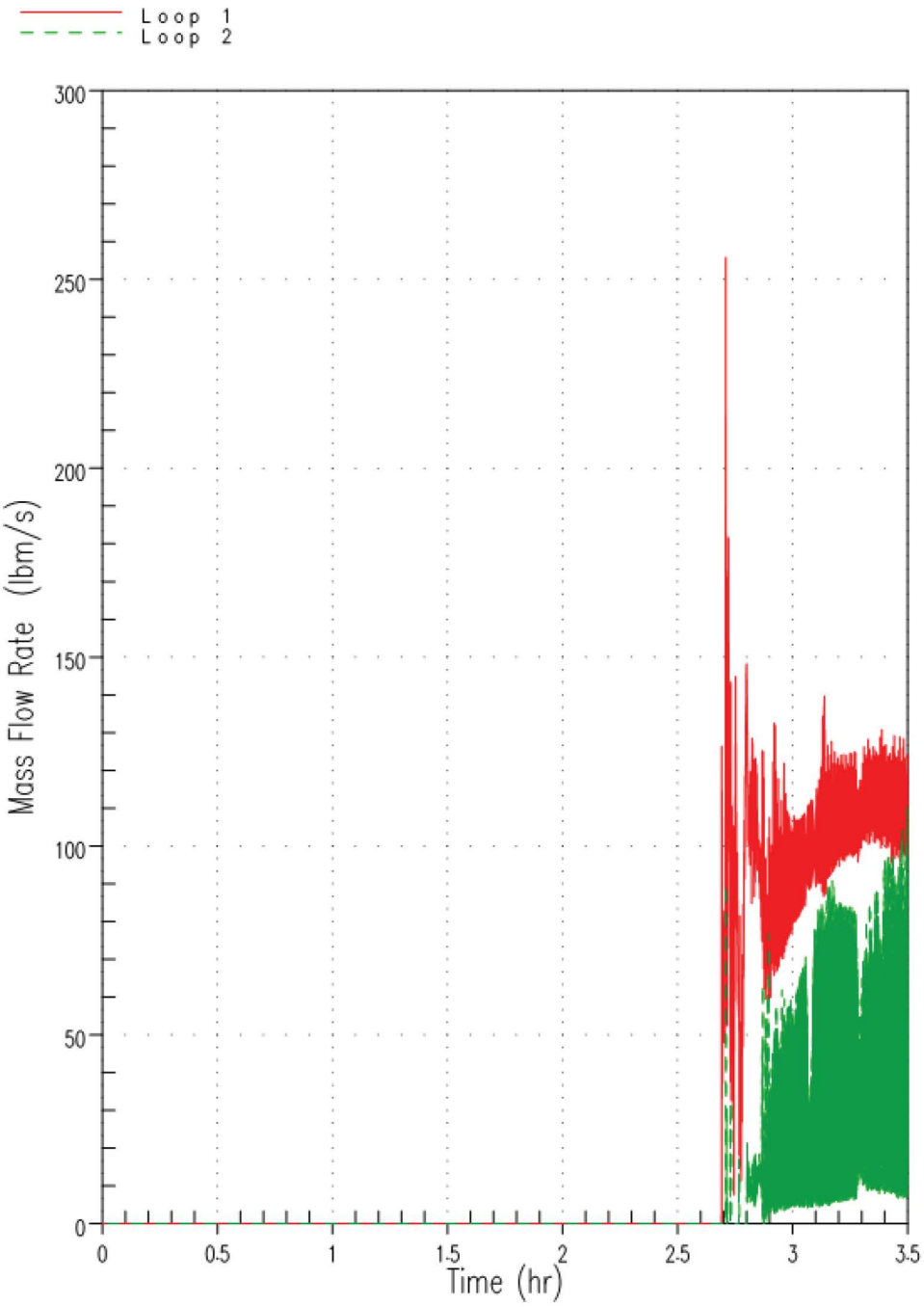

Figure 19E.4.8-18
ADS Stage 4 Liquid Flow, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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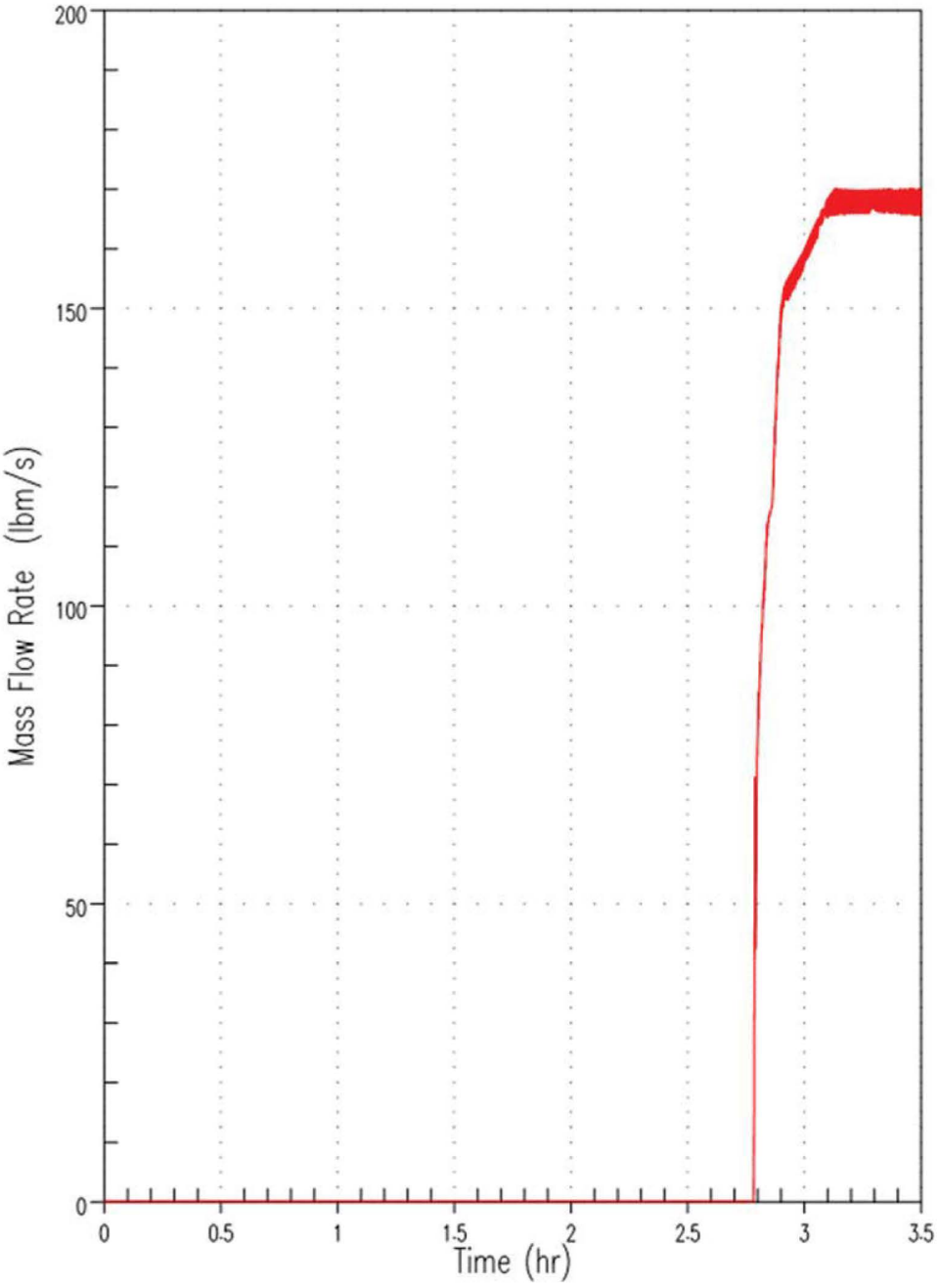

Figure 19E.4.8-19
Loop 1 IRWST Injection Flow, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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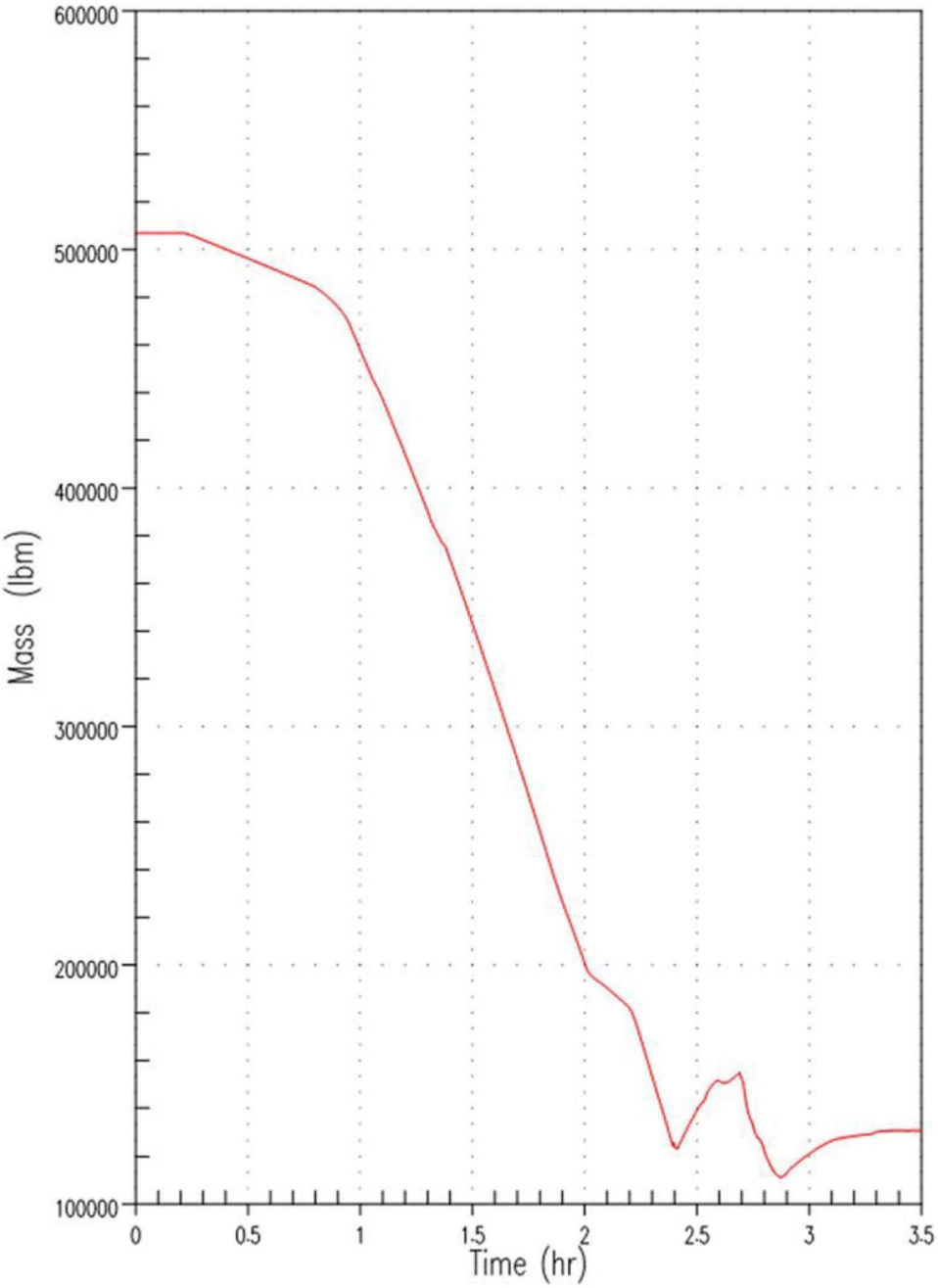

Figure 19E.4.8-20
Primary Mass Inventory, Loss of RNS in Mode 4 and Mode 5 with RCS Intact

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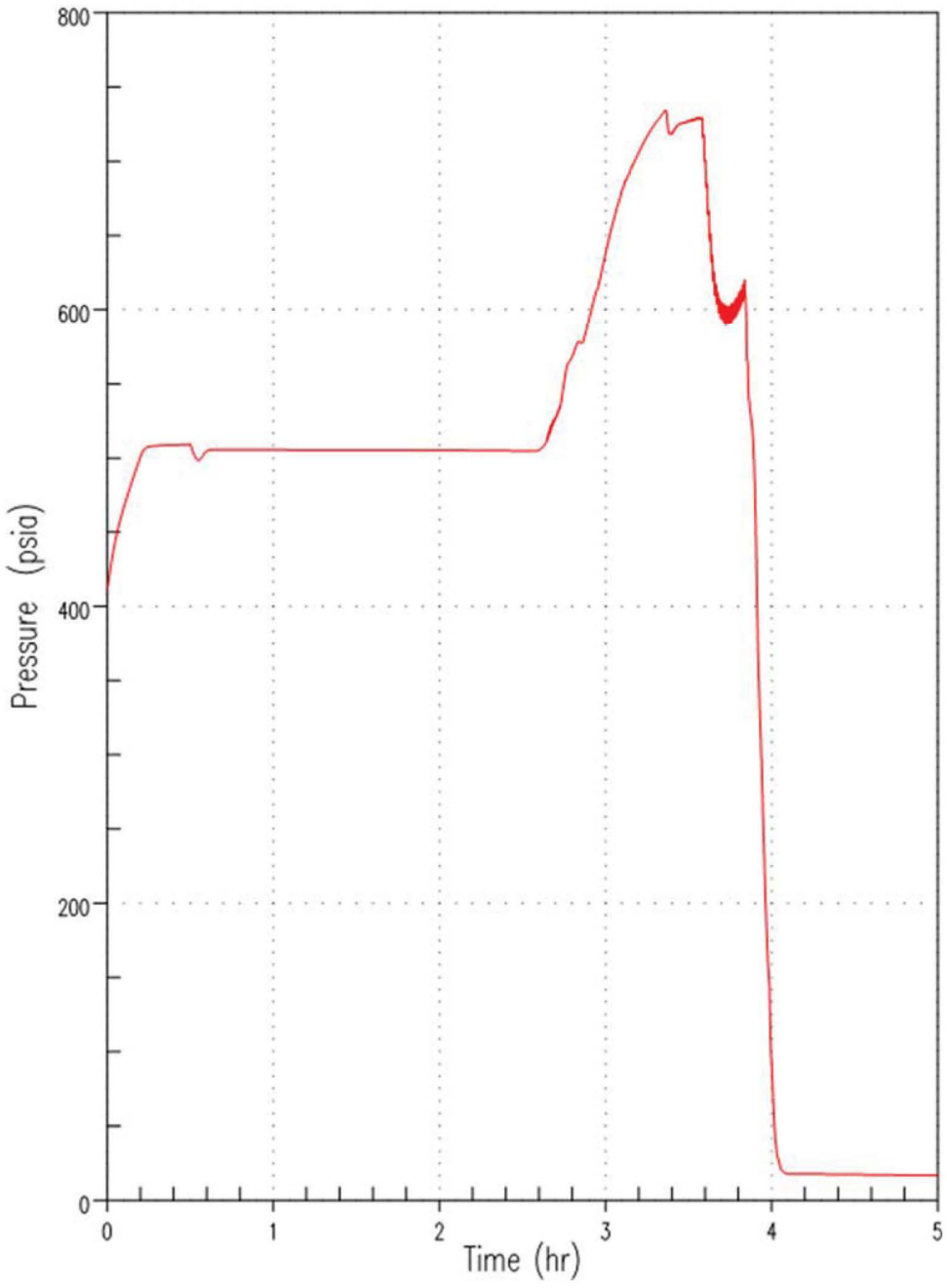

Figure 19E.4.8-21
Pressurizer Pressure, Loss of RNS in Mode 4 and Mode 5 with RCS Intact, Manual Safety System Actuation at 1800 Seconds

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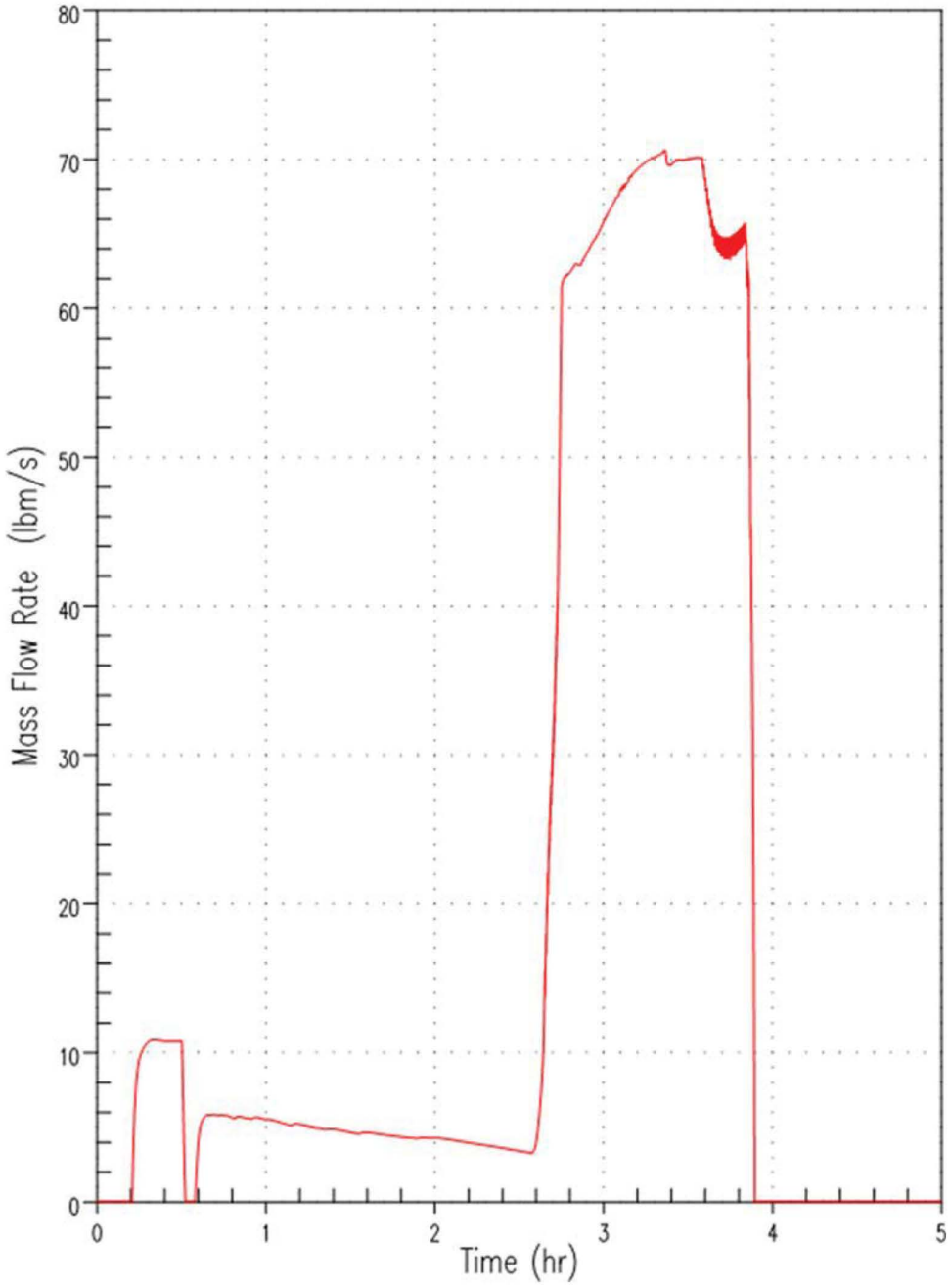

Figure 19E.4.8-22
RNS Relief Valve Flow, Loss of RNS in Mode 4 and Mode 5 with RCS Intact, Manual Safety System Actuation at 1800 Seconds

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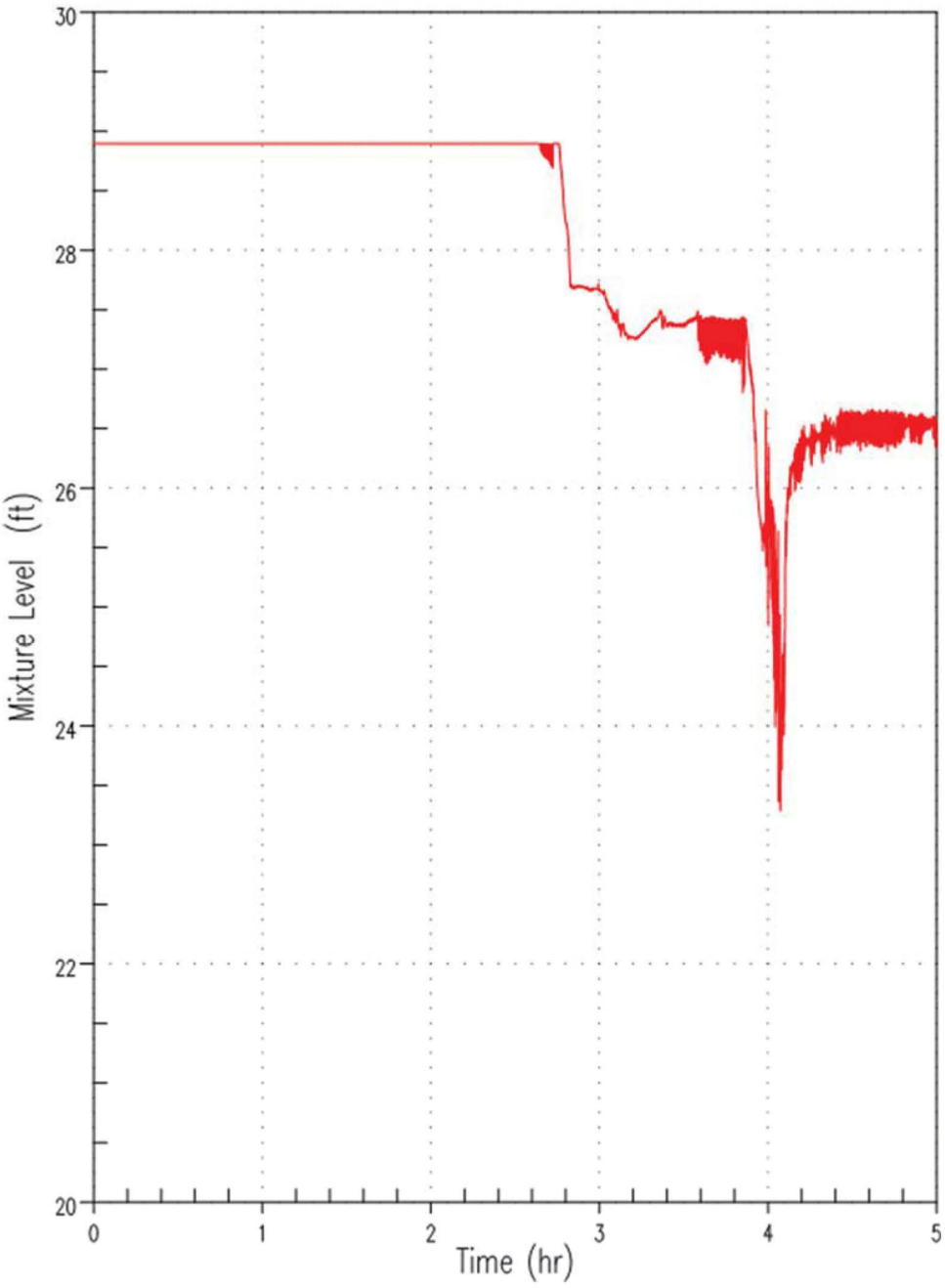

Figure 19E.4.8-23
Core Stack Mixture Level, Loss of RNS in Mode 4 and Mode 5 with RCS Intact, Manual Safety System Actuation at 1800 Seconds

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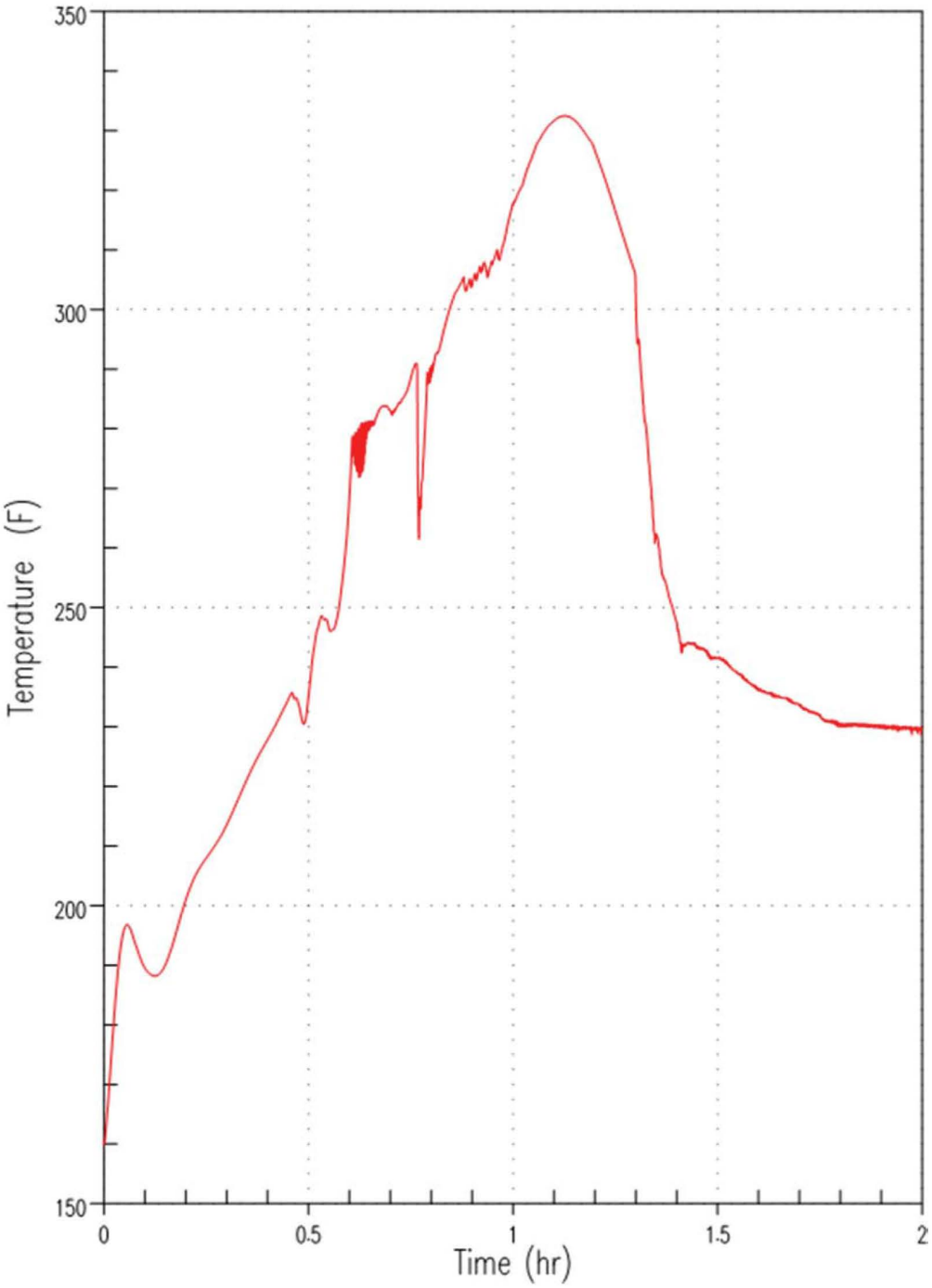

Figure 19E.4.8-24
Core Outlet Fluid Temperature, Loss of RNS in Mode 5 with RCS Open

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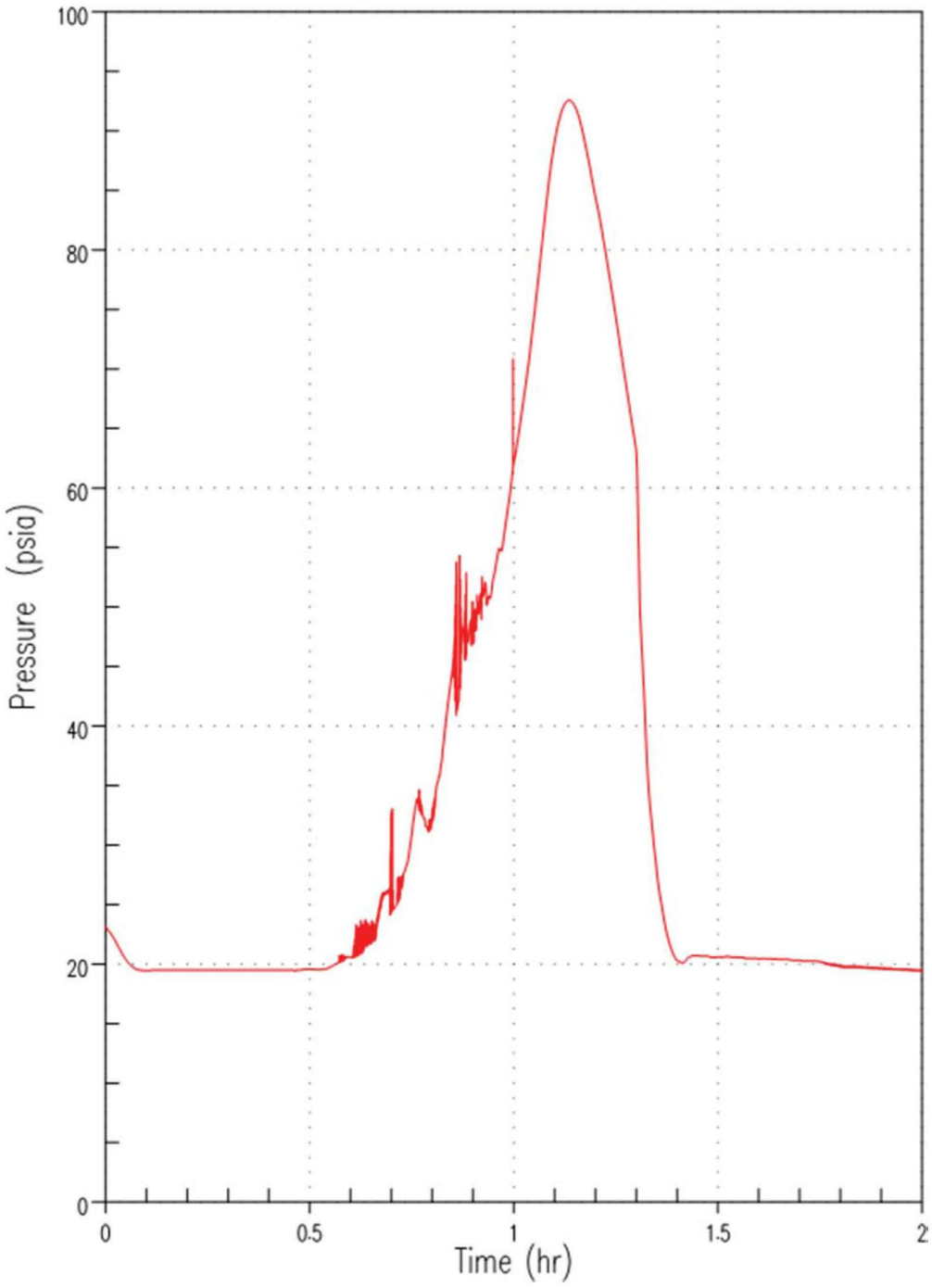

Figure 19E.4.8-25
Pressurizer Pressure, Loss of RNS in Mode 5 with RCS Open

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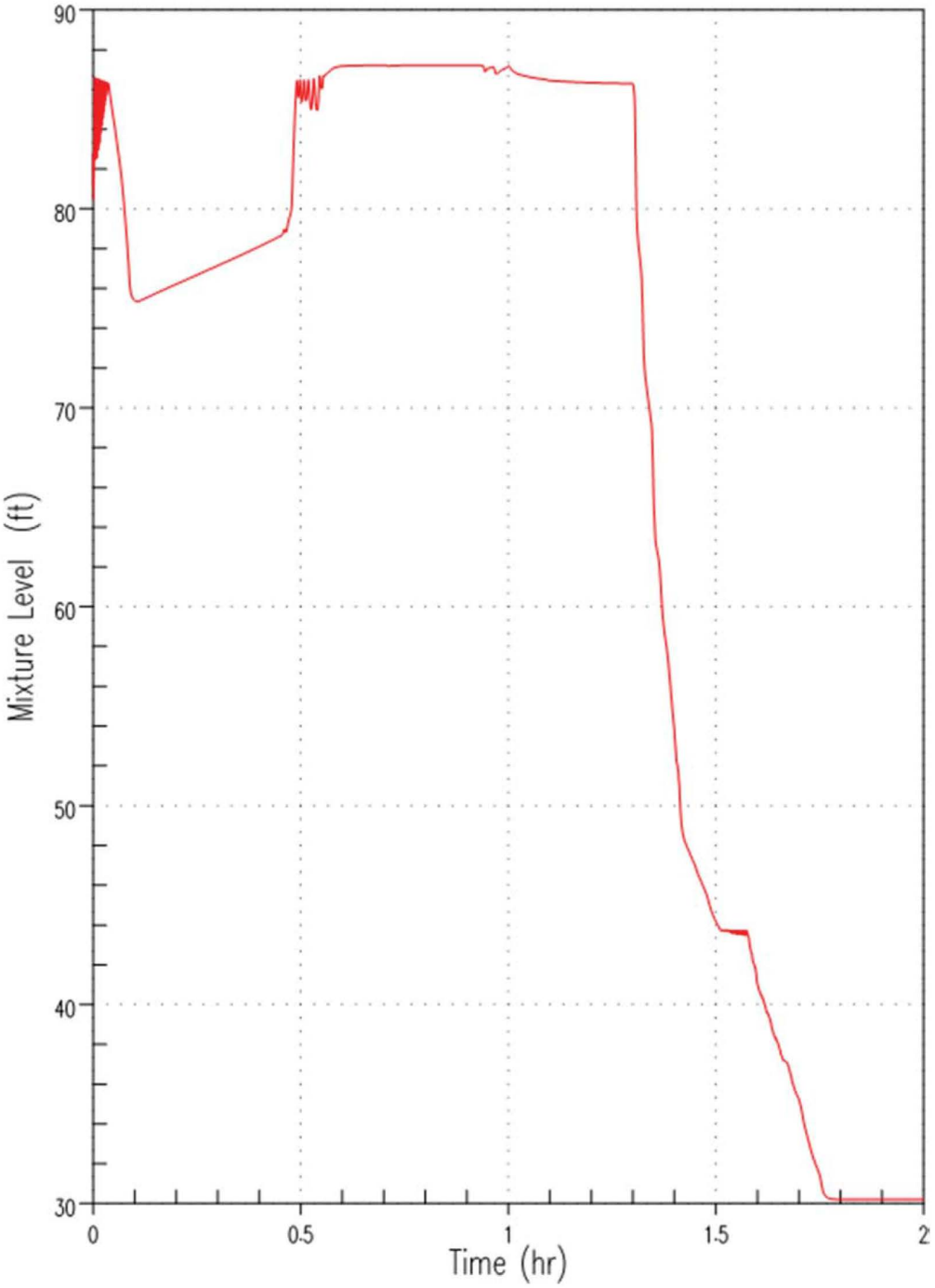

Figure 19E.4.8-26
Pressurizer Mixture Level, Loss of RNS in Mode 5 with RCS Open

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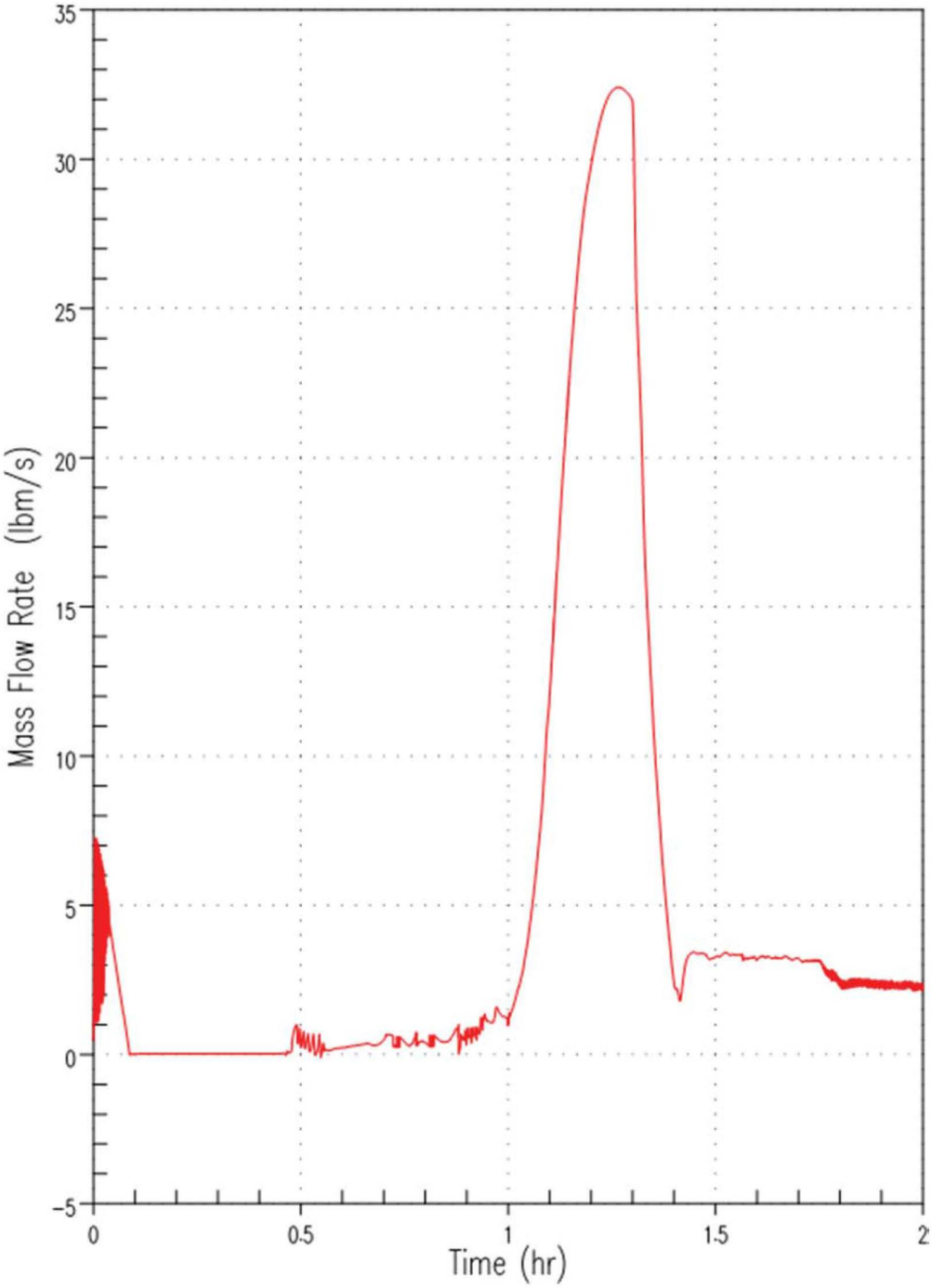

Figure 19E.4.8-27
ADS Stages 1-3 Vapor Flow, Loss of RNS in Mode 5 with RCS Open

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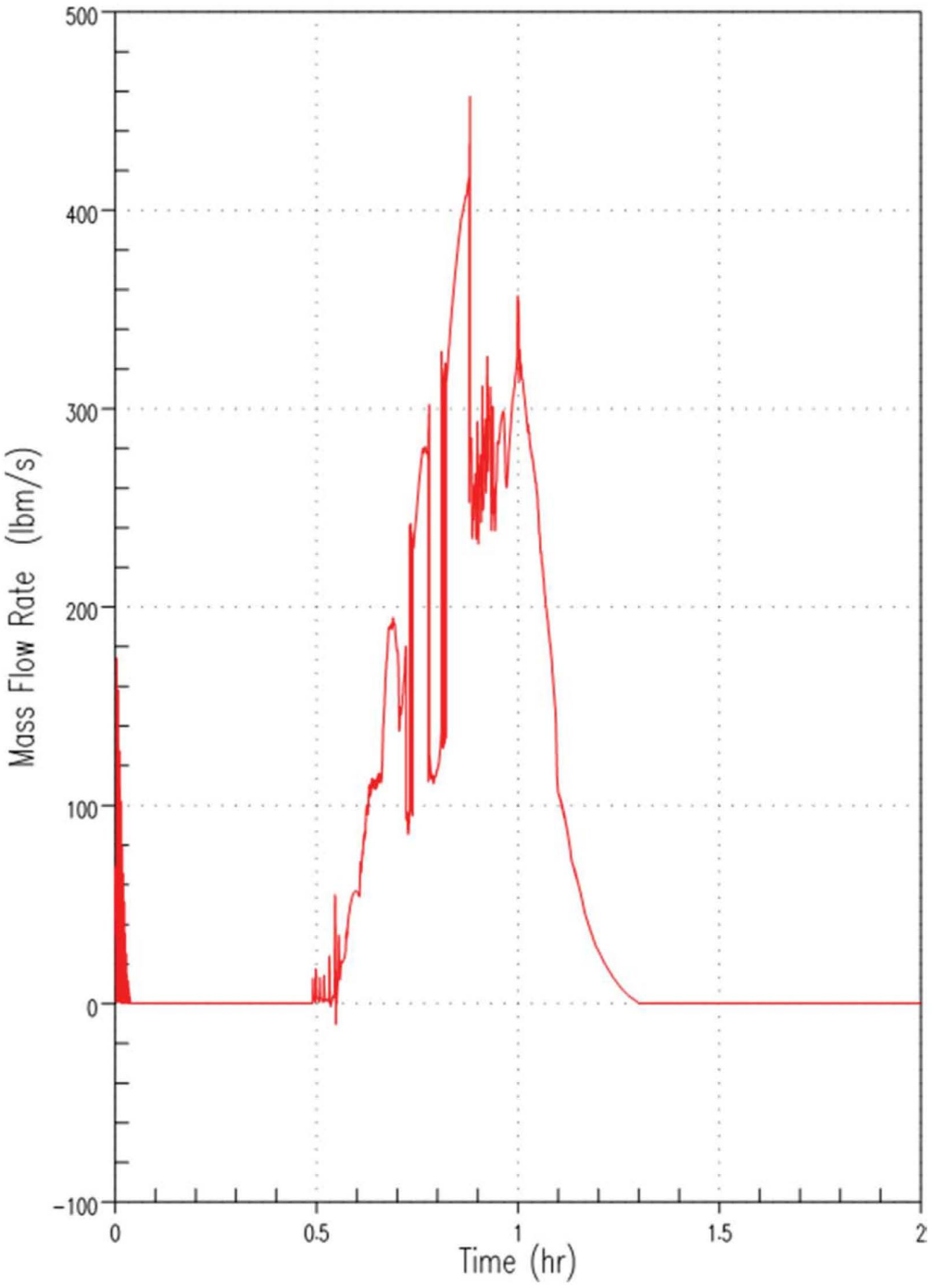

Figure 19E.4.8-28
ADS Stages 1-3 Liquid Flow, Loss of RNS in Mode 5 with RCS Open

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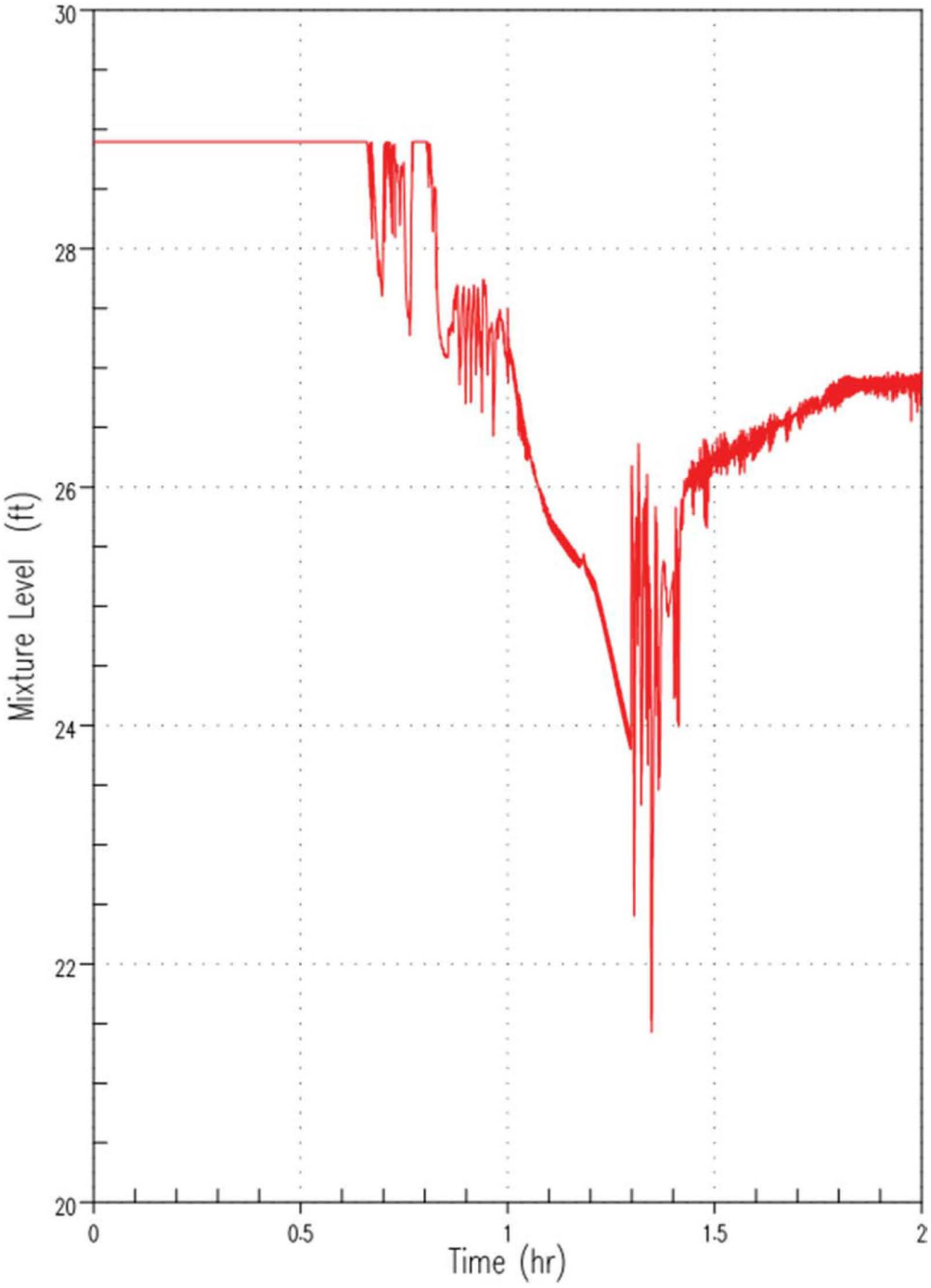

Figure 19E.4.8-29
Core Stack Mixture Level, Loss of RNS in Mode 5 with RCS Open

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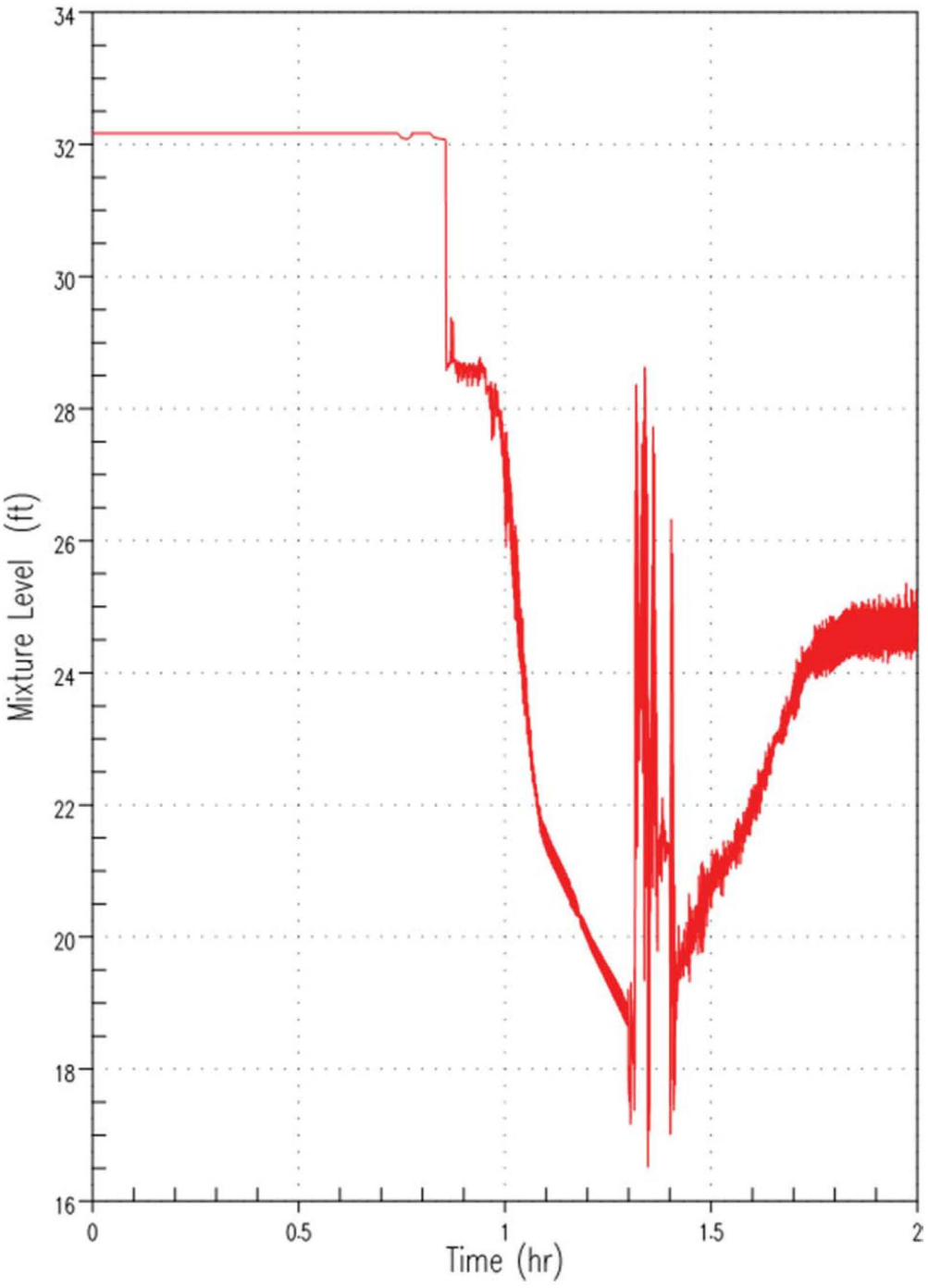

Figure 19E.4.8-30
Downcomer Mixture Level, Loss of RNS in Mode 5 with RCS Open

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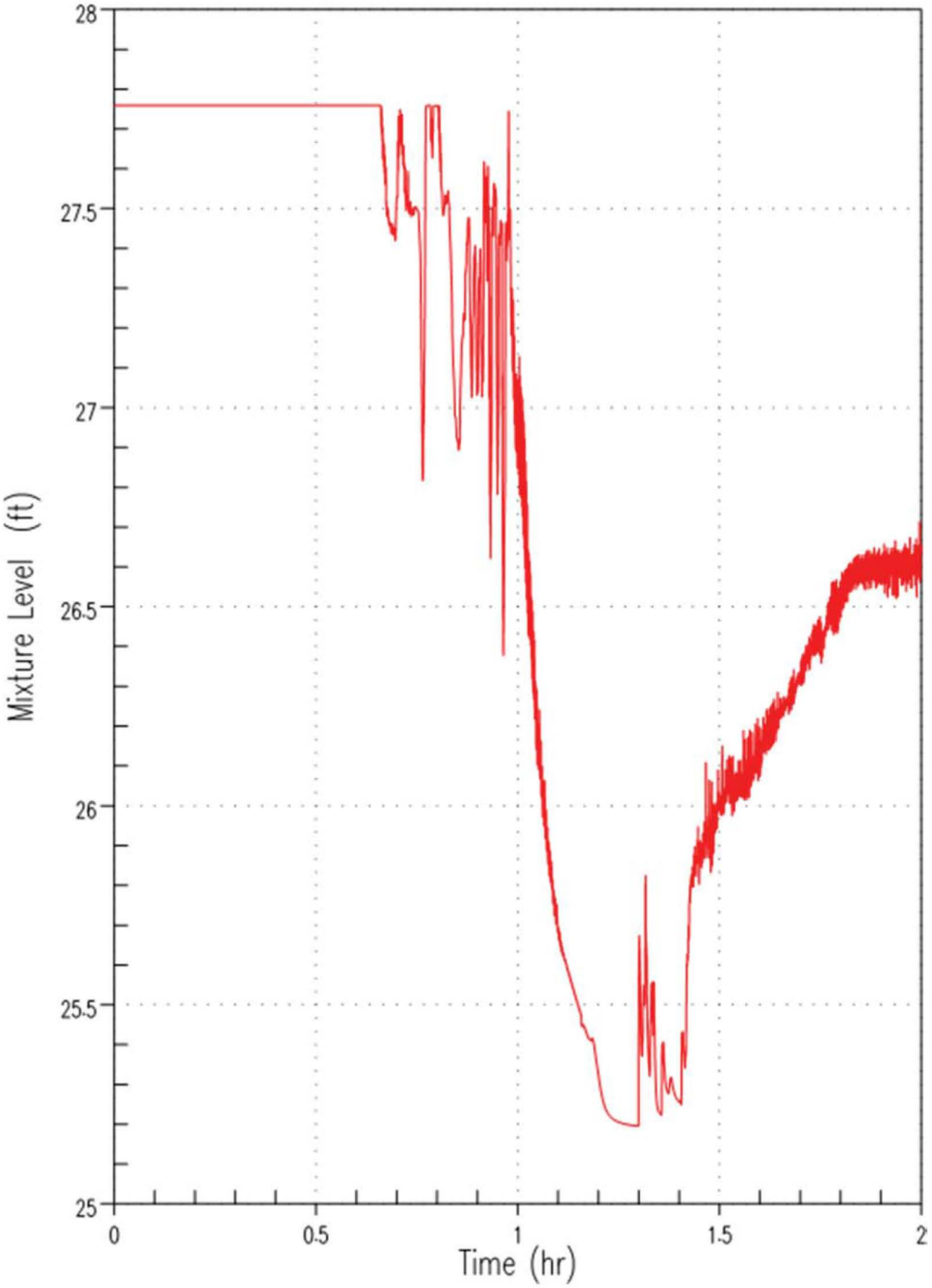

Figure 19E.4.8-31
Loop 2 Hot-Leg Mixture Level, Loss of RNS in Mode 5 with RCS Open

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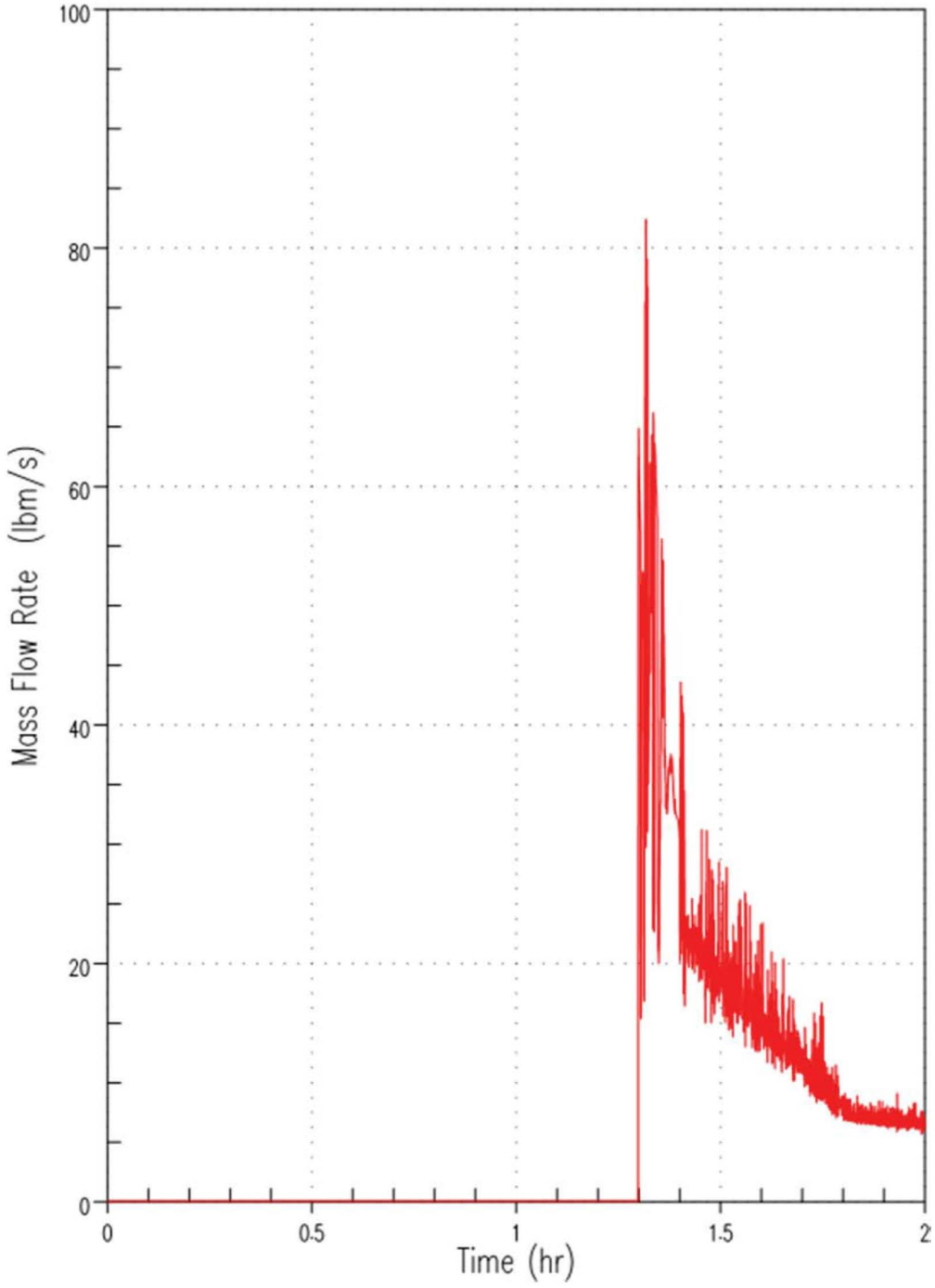

Figure 19E.4.8-32
ADS Stage 4 Vapor Flow, Loss of RNS in Mode 5 with RCS Open

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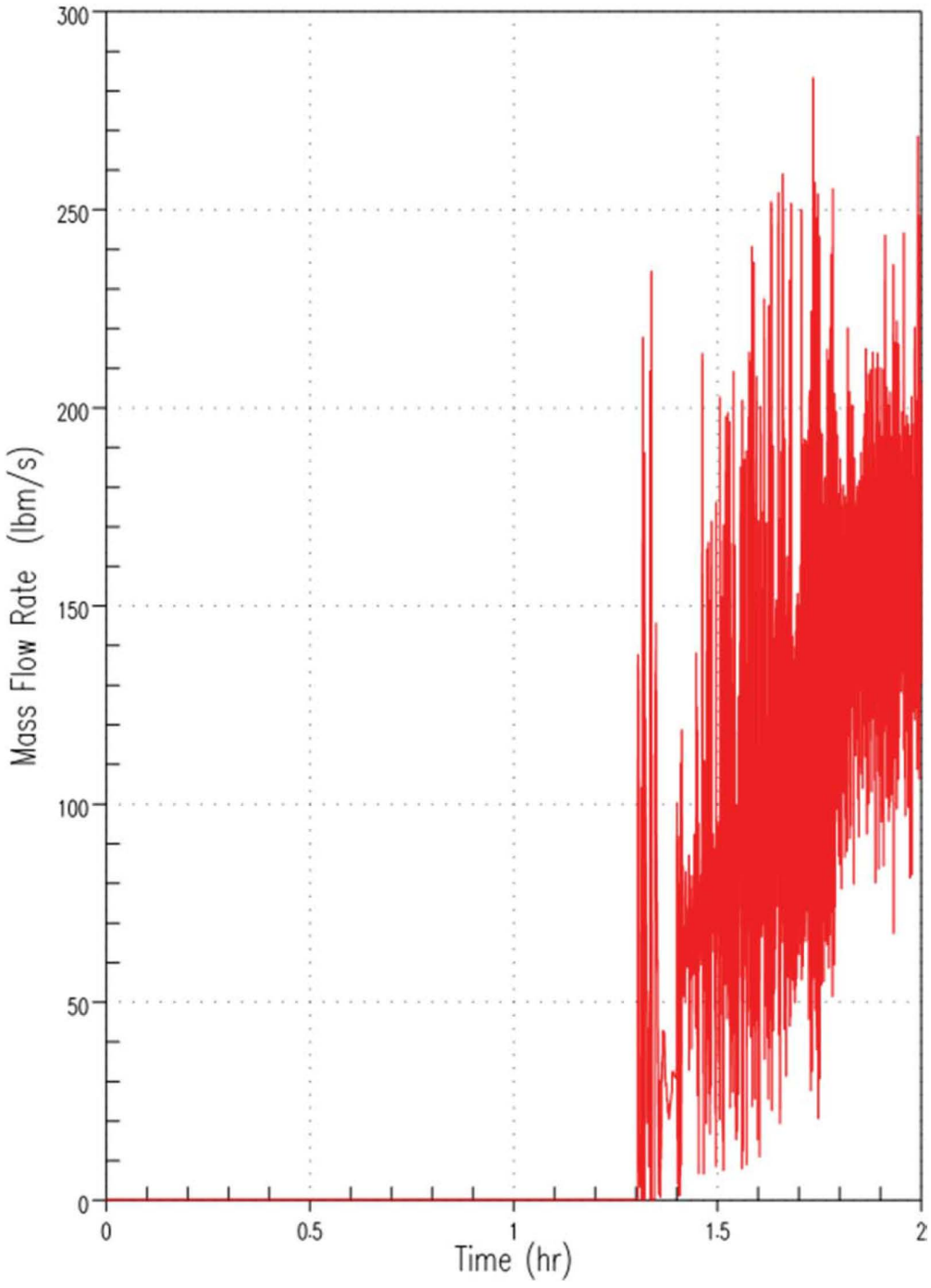

Figure 19E.4.8-33
ADS Stage 4 Liquid Flow, Loss of RNS in Mode 5 with RCS Open

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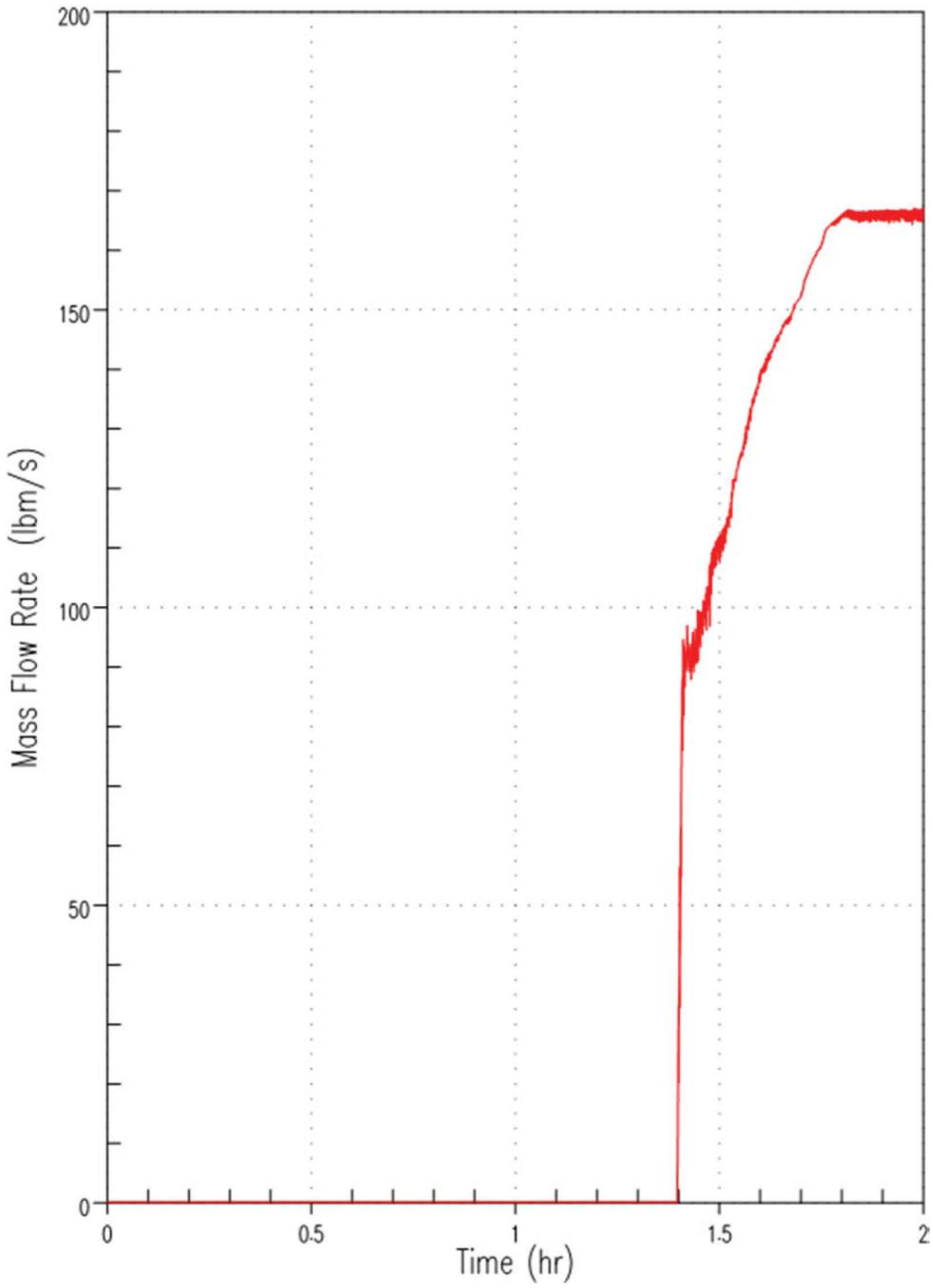

Figure 19E.4.8-34
IRWST Injection Flow, Loss of RNS in Mode 5 with RCS Open

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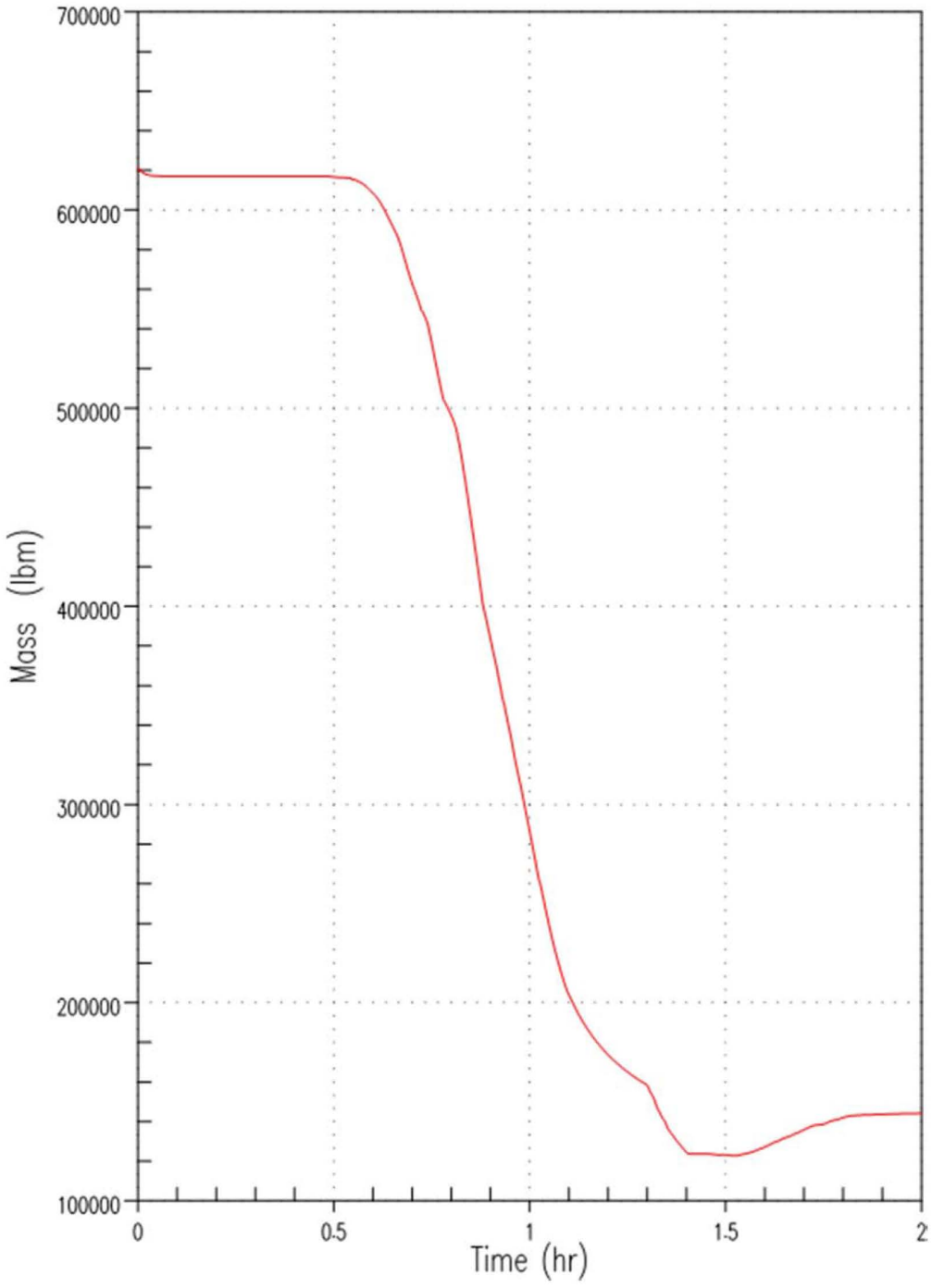

Figure 19E.4.8-35
Primary Mass Inventory, Loss of RNS in Mode 5 with RCS Open

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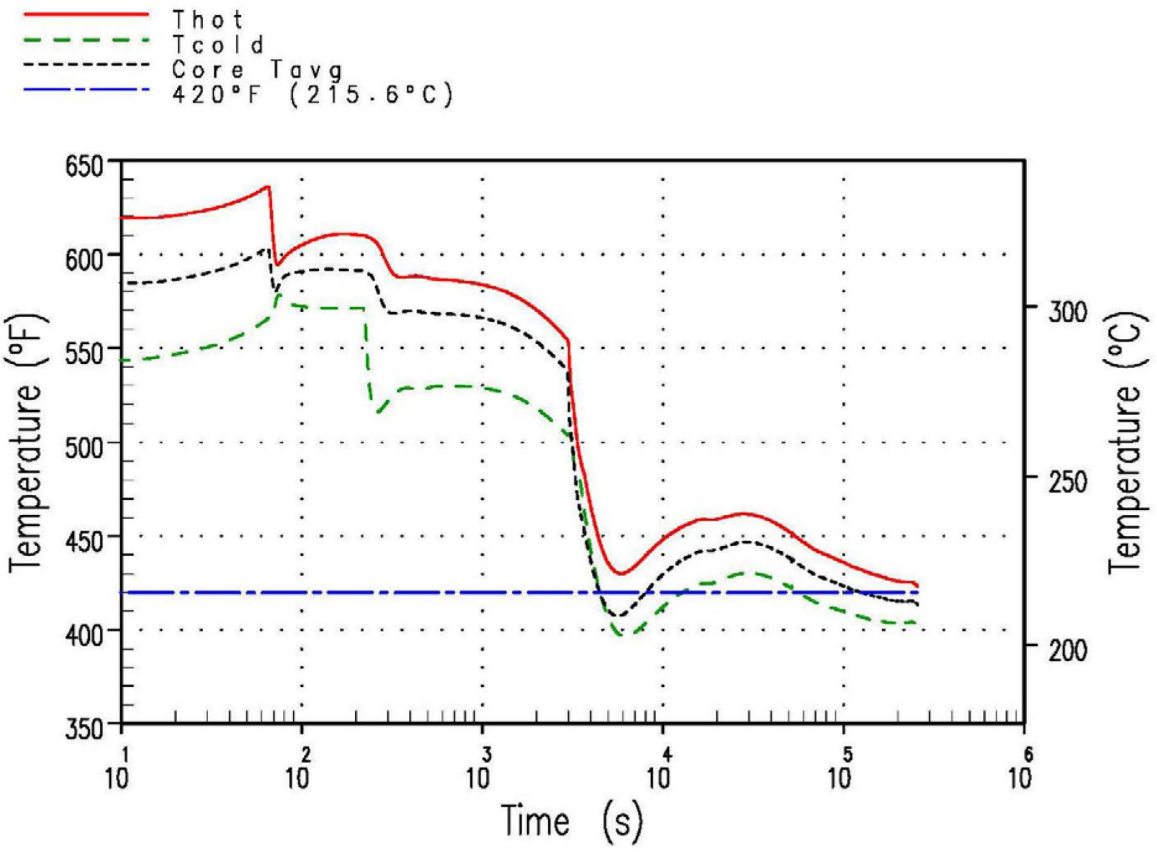

Figure 19E.4.10-1
Shutdown Temperature Evaluation, RCS Temperature

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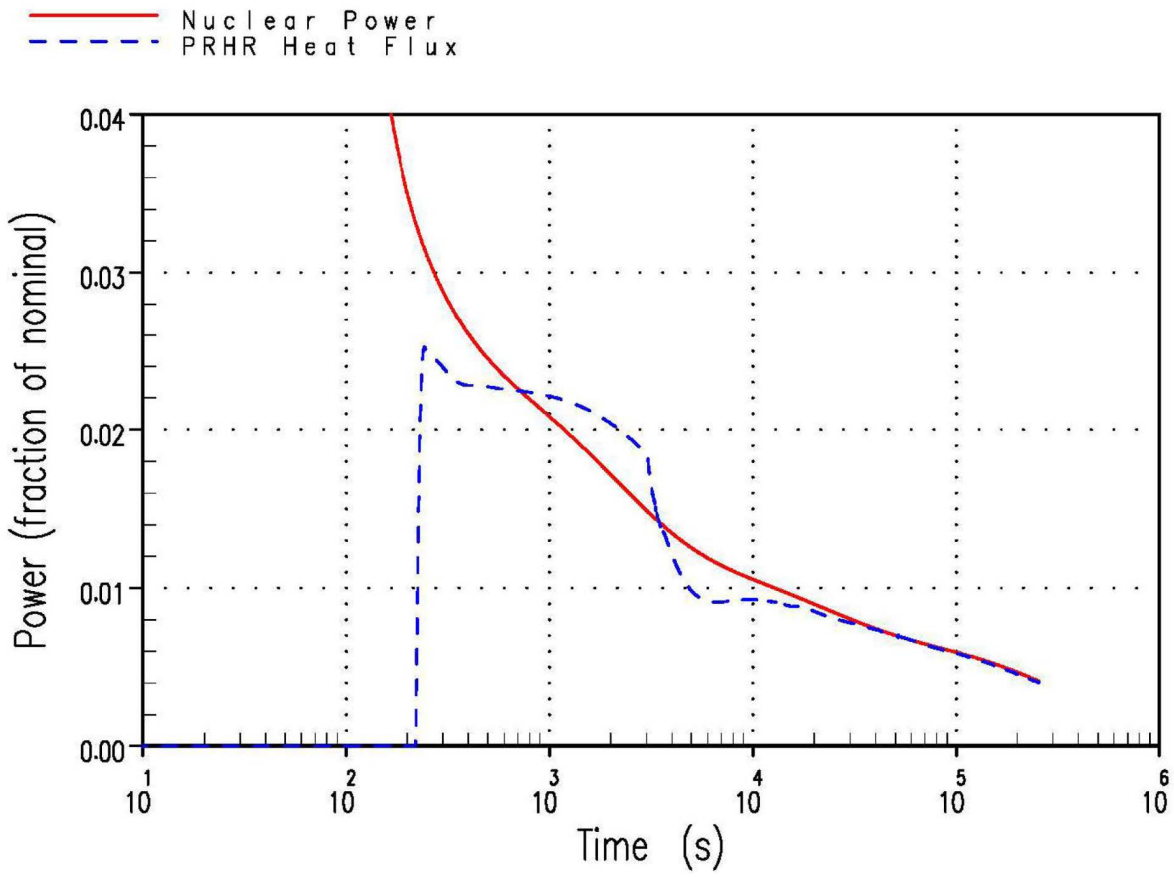

Figure 19E.4.10-2
Shutdown Temperature Evaluation, PRHR Heat Transfer

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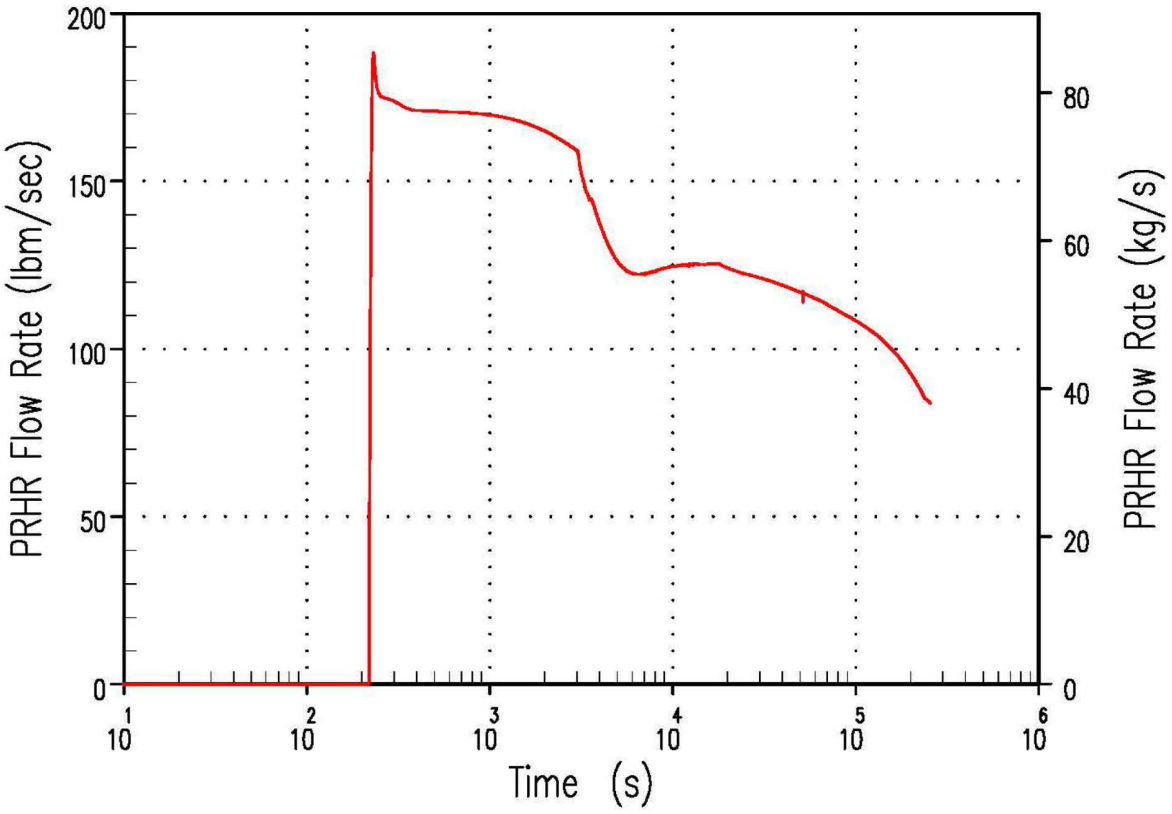

Figure 19E.4.10-3
Shutdown Temperature Evaluation, PRHR Flow Rate

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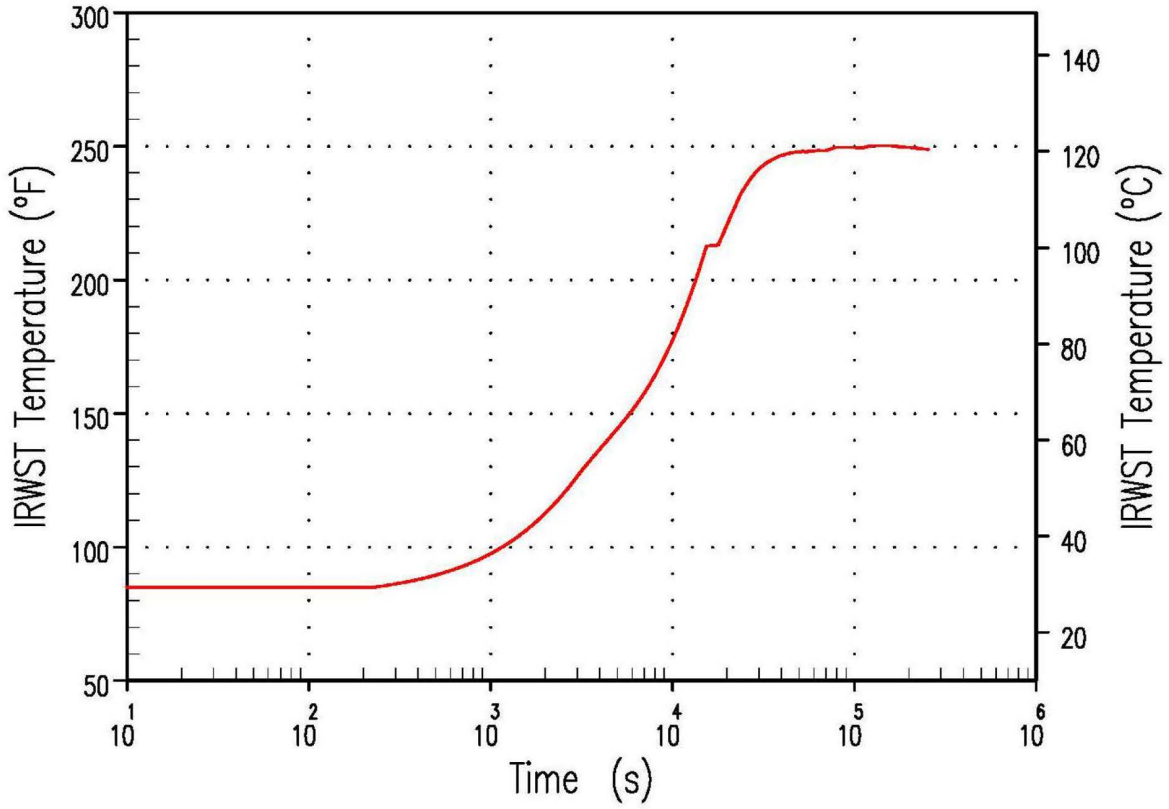

Figure 19E.4.10-4
Shutdown Temperature Evaluation, IRWST Heatup

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Appendix 19F Malevolent Aircraft Impact

19F.1 Introduction and Background

A design-specific assessment of the effects on the AP1000 of the beyond design basis impact of a large commercial aircraft has been performed in accordance with 10 CFR 50.150(a) to identify design features and functional capabilities that demonstrate with reduced use of operator actions: (i) the reactor core remains cooled, the containment remains intact, and (ii) spent fuel pool integrity is maintained. The specific assumptions regarding the aircraft impact were based on guidance provided by the NRC and the Nuclear Energy Institute, including the loading function derived from the aircraft impact characteristics for use in assessments of aircraft impact effects.

This appendix describes those design features and functional capabilities identified in the assessment, and discusses how the identified design features and functional capabilities show that with reduced use of operator actions, the reactor core remains cooled and the containment remains intact, and spent fuel pool integrity is maintained. In the following discussion, the identified design features are designated as “key design features.”

19F.2 Scope

The evaluation of plant damage caused by the impact of a commercial aircraft is a complex analysis problem involving phenomena associated with structural impact, shock-induced vibration, and fire effects. The assessment of the aircraft impact also considers structural damage, such as that caused by the impact/penetration of hardened components (e.g., engine rotors, landing gear).

An assessment of the effects of aircraft fuselage and wing structure is also performed.

An assessment of the effects of shock-induced vibration on systems, structures, and components is performed.

An assessment of the impact/penetration of hardened aircraft components, such as engine rotors and landing gear is performed.

Perforation of analyzed structural components, including the containment vessel and the spent fuel pool liner, is not predicted; therefore, realistic assessments of the damage to internal systems, structures, and components caused by 1) burning aviation fuel and 2) secondary impacts are not required.

19F.3 Assessment Methodology

Methods described in NEI 07-13, Revision 7 (Reference 1) were followed to assess the effects on the structural integrity of the primary containment and spent fuel pool, and to assess the physical, fire, and vibration effects of the aircraft impact on the core cooling capability of the existing and enhanced design. In accordance with the recommendation set forth in subsection 2.4.1(3) of NEI 07-13, Revision 7, an analytical evaluation and experimental verification has been performed for the first of a kind steel-concrete modular design feature subjected to the aircraft impact loading.

19F.4 Results/Conclusions

A detailed aircraft impact assessment was performed for AP1000 in accordance with the guidance in NEI 07-13 (Reference 1). The assessment concludes that an aircraft impact would not inhibit AP1000’s core cooling capability, would not impact containment integrity, and would not impact spent fuel pool integrity based on best-estimate calculations.

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The assessment resulted in the identification of the following design features and functional capabilities; changes to which are evaluated and reported in accordance with 10 CFR 50.150(d).

19F.4.1 Shield Building and Auxiliary Building

The shield building, as described in Chapter 3, is a key design feature for the protection of the safety systems located inside containment from the impact of a large commercial aircraft. The design of access openings is also a key design feature included as part of the shield building design. The assessment concludes that a strike upon the shield building would not result in perforation of the shield building so damage to the containment vessel would not occur. Therefore, the systems and equipment within the containment vessel are not damaged from the impact or from exposure to jet fuel.

The assessment finds that safety-related components inside containment, including the reactor pressure vessel and passive core cooling system, remain intact and maintain their intended capabilities following the shock-induced vibrations resulting from the impact of a large commercial aircraft based on applying the methodology in Reference 1.

This assessment also concludes that because of the structural integrity of the auxiliary building, a strike upon the auxiliary building would not result in loss of spent fuel pool liner integrity nor would it prevent the ability to manually actuate the system and equipment potentially required for core cooling. Both the structural design of the shield building and the auxiliary building, as described in Chapter 3, are considered key design features.

19F.4.2 Site Arrangement

The assessment credits the design and arrangement of certain building features, depicted in Figures 3.7.2-12 and 3.7.2-19, to limit the effects of a potential aircraft impact on the auxiliary building. These key features are as follows:

- The design of the wall along the south end of the turbine building at column line 11.2, as described in Subsection 3.7.2.8.3, is a key design feature for the protection of the auxiliary building from the impact of a large commercial aircraft.

- The design of the wall along the east side of the annex building at column line E, as depicted in Figure 3.7.2-19, is a key design feature for the protection of the auxiliary building from the impact of a large commercial aircraft.

- The design and location of the spent fuel pool in the southern portion of the auxiliary building, as depicted in Figure 3.7.2-12 and described in Subsection 9.1.2.2, is a key design feature for the protection of the spent fuel from the effects of an impact of a large commercial aircraft. The spent fuel pool is located in area 6 of the auxiliary building. The spent fuel pool liner is protected from the east, south, and west by a minimum of 7 feet, 3 inches of concrete and from the north by the location of the shield building. Therefore, the liner is not impacted and the spent fuel pool integrity is maintained.

- The locations of the main control room (MCR), remote shutdown station, and secondary diverse actuation system (DAS) panels are a key design feature for the protection against the physical and fire damage resulting from the impact of a large commercial aircraft. The detailed aircraft impact assessment shows that an aircraft impact cannot destroy all three of these locations due to the number of barriers associated with these locations. The main control room is located in room 12401, the remote shutdown station is located in room 12303, and the secondary DAS panels are located in room 12554. The assessment determined that any impact scenario would not destroy all three of these locations, and from any one of these locations, passive safety injection and recirculation for long-term core cooling can be initiated.

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- The design of the five oversized doors located on the east wall of room [ ]^{SRI}, the east wall of room [ ]^{SRI}, the shield building wall on the west side of room [ ]^{SRI}, east wall of room [ ]^{SRI} and the shield building wall on the west side of room [ ]^{SRI} are key design features for the protection against the physical and fire damage resulting from the impact of a large commercial aircraft. These doors and their connections to the walls are considered key design features because they are designed with a thickness that provides impact resistance equivalent to that of the wall. The doors at the east wall of room [ ]^{SRI}, the shield building wall on the west side of room [ ]^{SRI}, east wall of room [ ]^{SRI}, and the shield building wall on the west side of room [ ]^{SRI} are normally thicker than what is required for impact resistance due to radiation shielding. The door on the east wall of room [ ]^{SRI} is calculated to have a thickness of at least 5" steel to meet the impact resistance requirement. The walls are considered key design features for protecting containment integrity and core cooling.

19F.4.3 Core Cooling and Containment Integrity

If necessary, core cooling can be maintained by actuating the passive safety injection portion of the Passive Core Cooling System (PXS) and Reactor Coolant System (RCS) as described in Section 6.3. The portions of the PXS and RCS required for safety injection are located inside containment and are key design features. Their location protects them from damage due to an aircraft impact because the containment vessel remains intact and has no structural damage. The following valves are key design features and need to actuate for passive safety injection and recirculation for long-term core cooling:

- ADS Stage 4 squib valves, RCS-V004A/B/C/D (3 of 4)
- In-containment refueling water storage tank (IRWST) injection line squib valves, PXS-V123A/B and PXS-V125A/B (1 of 4)
- Recirculation line squib valves, PXS-V118A/B and PXS-V120A/B (1 of 4)

The steel containment vessel is protected by the shield building and is a key design feature. Based on beyond design basis calculations, the steel containment vessel is not impacted as a result of an aircraft impact on the shield building. If necessary, containment integrity is maintained by portions of the Passive Containment Cooling System (PCS). Containment integrity is maintained via air-only cooling by the passive containment cooling system. As discussed in Section 19.17, with air-only cooling (without design basis cooling), containment failure within 24 hours is predicted to be highly unlikely.

For design basis accidents, containment cooling is provided by water cooling of containment. Water cooling is distributed onto the containment vessel by the PCS water distribution bucket located above the containment vessel. Although the water distribution bucket is predicted to be unnecessary following an aircraft impact, an assessment has been performed on the water distribution bucket and predicts the support structure to be intact.

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19F.4.4 Reactor Trip

The reactor trip equipment is a key design feature. This equipment includes the sensors and manual inputs, protection and safety monitoring system cabinets, and reactor trip switchgear as described in Subsection 7.2.1. In the event of an aircraft impact, it is likely that ac power will be lost. On a loss of ac power, the control rods are de-energized and fall by gravity into the reactor core. If ac power is not lost, plant shutdown will be controlled by the intact protection and safety monitoring system or initiated manually from the main control room, remote shutdown room, or reactor trip switchgear. Additionally, if PMS is not intact as a result of the impact, the reactor trip breakers will open due to undervoltage. This results in the control rods being de-energized and falling into the reactor core. If the reactor trip switchgear or rod drive motor-generator sets are not intact, the rods also are de-energized and fall by gravity into the reactor core.

19F.4.5 Supporting Power, Instrumentation, and Control Equipment

The supporting equipment for the main control room, remote shutdown station, and secondary DAS panels are key design features. These include the class 1E batteries, the supporting PMS control and instrumentation cabinets and cabling for the equipment identified in Subsection 19F.4.3, the transfer switch to isolate the MCR and transfer controls to the remote shutdown room, and the DAS cabling for the DAS Processor Cabinet 1, Processor Cabinet 2, and Squib Valve Control Cabinet, which are referred to as the DAS secondary panels. These key design features enable the actuation of safety injection through operation of the squib valves. The functional capabilities of the secondary DAS panels are described in Subsection 7.7.1.11. These key design features are protected by their spatial separation as described in Subsection 19F.4.2.

19F.4.6 Fire Barriers

The design and location of 3-hour fire barriers within the auxiliary building are key design features for the protection of equipment needed to manually actuate the systems and equipment potentially required for core cooling following the impact of a large commercial aircraft. The assessment credited the design and location of fire barriers (including doors), as described in Appendix 9A, to limit the effects of fire damage created by the impact of a large commercial aircraft. Penetrations through

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specific barriers in the auxiliary building are rated to withstand a differential pressure of 5 psid based on the methodology in Reference 1. These barriers are identified in Subsection 9.5.1.2.1.1.

19F.5 References

1. NEI 07-13, Revision 7, “Methodology for Performing Aircraft Impact Assessments for New Plant Designs.”